TECHNICAL SPECIFICATIONS
FOR THE
UNIVERSITY OF MASSACHUSETTS LOWELL
RESEARCH REACTOR

FACILITY OPERATING LICENSE NO. R-125

Contents

1. INTRODUCTION

1.1 Scope

This document constitutes the technical specifications for The University of Massachusetts Lowell Research Reactor under facility license No. R-125. The technical specifications include definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls in accordance with 10CFR 50.36. Also included are the bases for the technical specifications. The bases, which provide the technical support for the individual technical specifications, are for information purposes only. They are not part of the technical specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.2 Application

1.2.1 Purpose

The technical specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission (NRC) on administrative controls, operational parameters, and equipment requirements, for safe reactor operation and for dealing with abnormal situations. They are typically derived from the safety analysis report (SAR). These specifications represent a comprehensive envelope for safe operation. The operational parameters and equipment requirements directly related to preserving this safe envelope are included.

1.2.2 Format

The format of this document is in general accordance with ANSI/ANS-15.1-2007.

1.3 Definitions

ADMINISTRATIVE CONTROLS – Those organizational and procedural requirements established by the NRC and/or the facility management.

CHANNEL – A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

CHANNEL CALIBRATION – A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

CHANNEL CHECK – A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.

CHANNEL TEST – A channel test is the introduction of a signal into the channel for verification that it is operable.

CONFINEMENT – Confinement is an enclosure of the reactor building that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways (see also Reactor Building).

CONTROL BLADE – See Rod, Control.

CORE CONFIGURATION – The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and control rods and regulating rod/ occupying the core grid.

EXCESS REACTIVITY – Excess reactivity is that amount of reactivity that would exist if all control rods and the regulating devices were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff} = 1) at reference core conditions and with all installed experiments in their most reactive condition.

EXPERIMENT – Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of their design to carry out experiments is not normally considered an experiment.

LICENSE – The written authorization, by the NRC, for an individual or the organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

LICENSEE – An individual or organization holding a license.

MEASURED VALUE – The measured value is the value of a parameter as it appears on the output for a channel.

MOVABLE EXPERIMENT – A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

OPERABLE – Operable means a component or system is capable of performing its intended function.

OPERATING – Operating means a component or system is performing its intended function.

OPERATIONS MODE – Operations mode refers to the method by which the reactor core is cooled, either natural convection mode or forced convection mode of operation.

PROTECTIVE ACTION – Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specific limit.

REACTIVITY WORTH OF AN EXPERIMENT – The reactivity worth of an experiment is the value of the reactivity change that results from the experiment, being inserted into or removed from its intended position.

REACTOR BUILDING – The reactor building is the enclosure housing the research reactor (see also Confinement).

REACTOR OPERATING – The reactor is operating whenever it is not secured or shut down.

REACTOR OPERATOR – An individual who is licensed by the NRC to manipulate the controls of the reactor.

REACTOR SAFETY SYSTEM – Reactor safety systems are those systems, including their associated input channels, that are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. The reactor safety system is also referred to as the reactor protection system.

REACTOR SECURED – The reactor is secured when:

(1) *Either* there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

(2) *Or* the following conditions exist:

   (a) The minimum number of neutron absorbing control devices are fully inserted or other safety devices are in shutdown position, as required by technical specifications;

   (b) The console key switch is in the off position and the key is removed from the lock;

   (c) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;

   (d) No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment ($<0.5\%\ \Delta k/k$).

REACTOR SHUTDOWN – The reactor is shut down if it is subcritical by at least one dollar (0.78% $\Delta k/k$) in the reference core condition with the reactivity worth of all installed experiments included.

REFERENCE CORE CONDITION – The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.2% $\Delta k/k$).

RESEARCH REACTOR –A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, developmental, educational, training, and experimental purposes and that may have provisions for the production of radioisotopes.

RESEARCH REACTOR FACILITY – Includes those areas described in TS 5.1.2 within which the licensee directs authorized activities associated with the reactor. The terms research reactor facility and facility may be used interchangeably.

ROD, CONTROL – A control rod is a device fabricated from neutron-absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. The terms control rod and control blade may be used interchangeably.

ROD, REGULATING – The regulating rod is a low worth control device, used primarily to maintain an intended power level and does not have scram capability. Its position may be varied manually or by a servo-controller.

SCRAM TIME – Scram time is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device.

SECURED EXPERIMENT – A secured experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

SENIOR REACTOR OPERATOR – An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

SHALL, SHOULD, AND MAY – The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

SHUTDOWN MARGIN – Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive control blade and regulating rod in the most reactive positions and that the reactor will remain subcritical without further operator action.

SITE – The UMLRR site includes the reactor confinement building and the attached academic building (Pinanski Hall).

SURVEILLANCE TIME INTERVALS – The maximum allowable intervals listed as follows are to provide operational flexibility only. Established frequencies shall be maintained over the long term. Any extension of these intervals shall be occasional and for a valid reason.

- 5 Year (interval not to exceed 6 years)
- Biennial (interval not to exceed 30 months)
- Annual (interval not to exceed 15 months)
- Semiannual (interval not to exceed 7-1/2 months)
- Quarterly (interval not to exceed 4 months)
- Monthly (interval not to exceed 6 weeks)
- Weekly (interval not to exceed 10 days)
- Daily (shall be done during the same working day)
- Prior to the first reactor start-up of the day

TRUE VALUE – The true value is the actual value of a parameter.

UNSCHEDULED SHUTDOWN – An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

End Definitions

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT

Applicability:
This specification applies to the reactor fuel.

Objective:
The objective is to ensure that the integrity of the fuel cladding is maintained.

Specification:
The reactor fuel clad temperature shall be less than 530°C (986°F).

Bases:
The melting temperature of aluminum is 660°C (1220°F). Fuel damage occurs with blister formation. The blister threshold temperature for both uranium silicide and uranium aluminide fuel is above 530°C (986°F) (NUREG-1313).

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 Forced Convection Mode

Applicability:
This specification applies to the set points for the safety channels monitoring reactor thermal power, coolant flow rate, reactor coolant inlet temperature, and the height of water above the center line of the core under the condition of the forced convection mode of operation.

Objective:
To ensure that automatic protective action is initiated in order to prevent the Safety Limit from being exceeded.

Specifications:

(1) The Limiting Safety System Setting for the reactor power level shall initiate automatic protective action at or below a measured value of 1.15 MW_t.

(2) The Limiting Safety System Setting for the primary coolant flow shall initiate automatic protective action at or above a measured value of 1400 GPM.

(3) The Limiting Safety System Setting for the primary coolant inlet temperature shall initiate automatic protective action at or below a measured temperature of 108^{o}F.

(4) The Limiting Safety System Setting for pool height above the core centerline shall initiate automatic protective action at or above a measured value of 24.25 ft.

Bases:
The Limiting Safety System Settings (LSSS) for forced convection mode are set points which if reached, will cause an automatic protective action to prevent the Safety Limit (SL) from being exceeded during the course of the most adverse anticipated transient. The LSSS values for this specification are more conservative than the values used in both the analyses for steady-state (SAR 4.6) and various transient conditions (SAR 13.2.2). Of the transient conditions analyzed, the step-reactivity addition is the most limiting condition. Using values for the variables more conservative than those in this specification, and using a step-reactivity value greater than the maximum reactivity value for a single secured experiment given in Specification 3.7.1, the analysis found the step-reactivity transient will not lead to an onset of nucleate boiling (ONB) before the reactor protection system begins to shut down the transient. The ONB limit provides an adequate margin to ensure the SL is not reached.

2.2.2 Natural Convection Mode

Applicability:
This specification applies to the set points for the safety channels monitoring reactor thermal power, reactor pool temperature, and the height of water above the center line of the core under the condition of the natural convection mode of operation.

Objective:
To ensure that automatic protective action is initiated in order to prevent undesirable radiation levels on the surface of the pool.

Specifications:

(1) The Limited Safety System Setting for the reactor power level shall initiate automatic protective action at or below a measured value of 115 kW_t.

(2) The Limited Safety System Setting for the pool temperature shall initiate automatic protective action at or below a measured temperature of 108 ^oF.

(3) The Limited Safety System Setting for pool height above the core centerline shall initiate automatic protective action at or above a measured value of 24.25 ft.

Bases:
The Limiting Safety System Settings (LSSS) for natural convection mode are set points which if reached, will cause an automatic protective action to prevent undesirable radiation levels on the surface of the pool due to the production and escape of ^{16}N during the natural convection mode of operation. The specifications also ensure an adequate safety margin exists between the LSSS and the SL for natural convection. The value for the power LSSS would be much higher (>200 kW, SAR 4.5) if the specifications were based on ONB rather than on ^{16}N production (see Bases for Forced Convection).

End Section 2

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR CORE PARAMETERS

3.1.1 Reactivity

Applicability:
These specifications apply to the reactivity condition of the reactor and the reactivity worths of the control blades, the regulating rod, and experiments.

Objective:
To ensure that the reactor can be safely operated and shutdown and maintained in a safe shutdown condition at all times such that the Safety Limit will not be exceeded.

Specifications:
When the reactor is operating, the following conditions shall exist:

(1) The excess reactivity in the reference core condition shall be <4.7% $\Delta k/k$.

(2) The shutdown margin shall be >1% $\Delta k/k$ with the most reactive control blade and regulating blade in their fully withdrawn position; all installed experiments in their most reactive state; and the reactor in the reference core condition.

(3) All core grid positions shall be filled with fuel elements, irradiation baskets, source holders, regulating rod, graphite reflector elements, lead void boxes, or grid plugs.

(4) No more than five (5) of the radiation baskets shall be without flow restricting devices. This specification shall not apply for low power operation <100 kW without forced flow.

(5) The reactor shall not be knowingly operated with damaged fuel except as may be necessary to identify the location of the damaged fuel.

(6) The reactor shall not be operated whenever the reactor core is in the same end of the reactor pool as any portion of the cobalt-60 source.

Bases:
The maximum allowed excess reactivity of provides sufficient reactivity to accommodate fuel burnup, xenon and samarium poisoning buildup, experiments, and control requirements, but gives a sufficient shutdown margin even with the highest worth control blade and the regulating rod fully withdrawn. (SAR 4.5.3) The shutdown margin provides adequate negative reactivity to ensure the reactor

can be shut down from any operating condition and will remain shutdown after cooldown and xenon decay, even if the highest worth control rod should be in the fully withdrawn position. The requirement that all grid plate positions be filled and the restriction on radiation baskets during reactor operation ensures that the quantity of primary coolant which bypasses the heat producing elements will be kept within the limits used for the transient analyses (SAR 4.5.7 and 13.2.2). This requirement does not apply under natural circulation conditions given the analyses for natural convection show that ONB does not occur for power levels <248kW (SAR 4.6.1). Specification 5 assures that fuel elements found to be defective are no longer used. Fresh fuel elements are initially inspected in accordance with written procedures to assure the fuel elements are not damaged. In-core fuel elements are periodically re-inspected. Specification 6 prevents the Co-60 from causing signal interference with the power measuring detectors, particularly at low power levels.

3.1.2 Maximum Power Level

Applicability:
This specification applies to the reactor thermal power level.

Objective:
To ensure the safety limit is not exceeded.

Specification:
The reactor shall not be continuously operated at a power level exceeding $1MW_t$.

Basis:
Thermal hydraulic calculations presented in Chapter 13 of the SAR demonstrate that the fuel may be safely operated at power levels up to 1.25 MW. The LSSS specification in 2.2.1(1) takes into account the reactor power measurement uncertainty. Automatic protective action would be initiated at or below that value. Momentary drifts of power level beyond $1MW_t$ would be corrected by the reactor operator.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

3.2.1 Control Blades

Applicability:
This specification applies to the reactor control blades.

Objective:
To specify the minimum number of operable control blades and their maximum scram time to ensure the reactor can be shutdown and the Safety Limit is not exceeded.

Specifications:
(1) All four control blades shall be operable when the reactor is operating.

(2) The time from initiation of a scram signal and movement of each control blade from the fully withdrawn position to 80% of the fully inserted position shall be less than one second.

Bases:
The UMLRR is equipped with four control blades and one regulating rod as described in SAR 4.2.2. The control blades are connected to their drives by electromagnets and hence drop by gravity into the core upon initiation of a scram signal. The last few inches of travel are dampened to prevent damage to the control blade due to its momentum. (SAR 4.2.2.1) Analyses in Chapter 13 of the SAR show that for the most limiting transient, the peak clad temperature is well below the ONB point during the 1.0 second scram time interval. The analyses also assume only 3 of the 4 control blades are scrammed. For added conservativeness, the specification requires all four control blades to be operable.

3.2.2 Maximum Reactivity Insertion Rate and Regulating Rod Worth

Applicability:
This specification applies to the maximum positive reactivity insertion rate by the most reactive control rod and the regulating rod simultaneously.

Objective:
To ensure that the reactor is operated safely and the safety limit is not exceeded during any credible ramp reactivity insertion.

Specifications:

(1) The maximum reactivity insertion rate by the most reactive control blade and the regulating rod simultaneously shall not exceed 0.05% $\Delta k/k$ per second.

(2) The total reactivity worth of the regulating rod shall be < 0.5% $\Delta k/k$.

(3) Only one control blade shall be withdrawn at a time.

Basis:
The maximum reactivity insertion rate limit and requirement for withdrawal of only one control at a time ensures that the safety limit will not be exceeded as a result of a continuous linear reactivity insertion. The analyses show that the peak clad temperature would be well below the ONB point even under the conservative assumption that the reactor is operating at the LSSS values for power and temperature when the ramp begins and using a reactivity addition rate greater than that allowed by the specification (SAR 13.2.2.2). An analysis of a step insertion >0.5% $\Delta k/k$ shows the step-reactivity transient will not lead to ONB before the reactor protective system begins to shut down the transient (SAR 13.2.2.1). Limiting the reactivity worth of the regulating rod to this value ensures that any failure of the automatic servo control system could not result in the Safety Limit being exceeded.

3.2.3 Reactor Protection System Scrams

Applicability:
This specification applies to the reactor protection system..

Objective:
To stipulate the minimum number of reactor protection system scrams that shall be operable to ensure that the safety limit is not exceeded.

Specification:
The reactor shall not be operated unless the reactor protection system scrams described in Table 3.2.3-1 are operable.

Table 3.2.3-1
Minimum Reactor Protection System Scrams

		Forced Convection Mode		**Natural Convection Mode**	
	Scrams	Function	Minimum Required	Function	Minimum Required
1.	**Reactor Period**	Scram at $\leq$ 3 second period	1	Scram at $\leq$ 3 second period	1
2.	**Reactor Power Level**	Scram at $\geq$ 1.15 MW	2	Scram at $\geq$115kW	2
3.	**Primary Coolant Flow Rate**	Scram at $\leq$1400 GPM	1	n/a	n/a
4.	**Pool Water Level**	Scram at $\leq$ 24.25 ft above core centerline	1	Scram at $\leq$ 24.25 ft above core centerline	1
5.	**Pool Inlet Temperature**	Scram $\geq$ 108^{o}F	1	n/a	n/a
6.	**Pool Temperature**	Scram $\geq$ 108^{o}F	1	Scram $\geq$ 108^{o}F	1
7.	**Control Room Manual Scram Button**	Scram if pressed	1	Scram if pressed	1
8.	**Detector High Voltage (each period and power channel)**	Scram $\leq$ 500 V	1	Scram $\leq$ 500 V	1
9.	**Process Controls Display Watch Dog Timer**	Scram for communication loss >10 second	1	Scram for communication loss >10 second	1
10.	**Drives Controls Display Watch Dog Timer**	Scram for communication loss >10 second	1	Scram for communication loss >10 second	1
11.	**Seismic Disturbance**	Scram on seismic motion	1	Scram on seismic motion	1

Table 3.2.3-1 (continued)
Minimum Reactor Protection System Scrams

		Forced Convection Mode			**Natural Convection Mode**	
	Scrams	Function	Minimum Required		Function	Minimum Required
12.	**Bridge Movement**	Scram if moved > 1 inch	1		Scram if moved > 1 inch	1
13.	**Primary Piping Alignment**	Scram when alignment limit switches not met	1		n/a	n/a
14.	**Riser Coolant Gate Open**	Scram when gate opens in cross-pool mode	1		n/a	n/a
15.	**Coolant Gate Open**	Scram when either gate opens in downcomer mode	2		n/a	n/a

Bases

The reactor protection system is described in SAR section 7.4. The automatic protective action initiated by the reactor period channel, the reactor power level channels, the flow rate channel, the pool water level channel, and the coolant temperature channels all provide redundant protection to ensure that the Safety Limit is not exceeded. The manual scram button provides a manual method to shutdown the reactor if the operator determines an unsafe condition has occurred or could occur. Automatic protection action initiated by a detector high voltage failure or displays watchdog timers ensures a reactor shutdown occurs for potential instrumentation problems. Automatic protection action initiated by a seismic event ensures the reactor will be shutdown should structural or system damage occur due to seismic activity. The bridge movement, primary piping alignment, and coolant gate trips ensure adequate coolant flow is maintained in the reactor core during forced convection operations.

3.2.4 Radiological Protection Scrams

Applicability:
This specification applies to reactor scrams associated with radiological protection.

Objective:
Radiological protection scrams are incorporated in the scram circuit to protect personnel, the public, and the environment from possible radiation exposure.

Specification:
The reactor shall not be operated unless the following radiological protection scrams described in Table 3.2.4-1 are operable.

Table 3.2.4-1
Radiological Protection System Scrams

	Scram	Function	Minimum Required
1.	Thermal Column Door Open	Scram if door limit switch open	1
2.	Beam port Chamber Door Open	Scram if door limit switch open	1
3.	First Floor Airlock Integrity	Scram if both doors unsealed	1
4.	Third Floor Airlock Integrity	Scram if both doors unsealed	1
5.	Truck Door Seal	Scram if door unsealed	1

Bases:
The radiological protection scrams minimize the possibility of exceeding 10CFR Part 20 limits for radiation exposure.

3.2.5 Minimum Channels Needed for Reactor Operation

Applicability:
This specification applies to channels in the reactor protection and control systems.

Objective:
To stipulate the minimum number of channels that shall be operable to ensure that the reactor operator has sufficient information for safe operation of the reactor.

Specification:
The reactor shall not be operated unless the channels in the Table 3.2.5-1 are operable.

Table 3.2.5-1
Minimum Reactor Protection Channels

	Channel	Operations Mode	Minimum Required
1.	Start-up Count Rate	Both	1
2.	Reactor Period	Both	1
3.	Reactor Linear Power Level	Both	1
4.	Reactor Log Power Level	Both	1
4.	Primary Coolant Flow Rate	Forced	1
5.	Pool Water Level	Both	1
6.	Pool Inlet Temperature	Forced	1
7.	Pool Temperature	Both	1

Bases:
The channels associated with the reactor protection system are described in SAR section 7.4. The channels listed in the above table ensure that measurements of the reactor power level and the process variables are adequately displayed during reactor startup and during low-power natural convection and high-power forced convection modes of operation.

3.2.6 Reactor Control System Interlocks

Applicability:
This specification applies to the reactor control system.

Objective:
To stipulate the minimum number of interlocks available to inhibit control blade withdrawal.

Specifications:
The following interlocks to prevent control blade withdrawal shall be operable when the reactor is operating:

(1) Scram circuit not reset.

(2) Start-up neutron count rate is $\leq$ 2 counts per second.

(3) The reactor period $\leq$15 seconds.

Bases:
Interlocks associated with the reactor control system are described in SAR sections 7.2.2.1 and 7.3.3. The requirement for the scram circuit to be reset ensures that reactor conditions are normal and radiological hazards are minimized. The inhibit function for startup neutron count rate ensures the required startup neutron source is sufficient and in a proper location for reactor startup, such that a minimum source multiplication count rate level is being detected. The inhibit function for the reactor period channel limits the rate of power increase when withdrawing a control rod and K_{eff} >1.

3.3 REACTOR COOLANT SYSTEMS

Applicability:
This specification applies to the reactor primary coolant system water quality requirements and pool configuration.

Objective:
The objectives are to minimize corrosion and radioactive contaminants, and to ensure the full volume of pool water is available in the event of a loss of coolant accident.

Specifications:
The reactor shall not be operated under any of the following conditions:

(1) The conductivity of the pool water, averaged over the previous two weeks, is >5 µmho/cm.

(2) The pH of the pool water is <5.0 or >7.5.

(3) The concentrations of radionuclides in the bulk pool water meet or exceed the values presented for water in 10 CFR Appendix B to Part 20 Table 2. If such a condition occurs, the source of the radionuclide(s) shall be identified and corrected.

(4) The pool divider gate is in position to separate the bulk pool and the stall pool.

Bases:
Pool water of high purity minimizes the rate of corrosion and minimizes neutron activation of impurities. The purpose of a pH limit is to minimize corrosion of the fuel, core components, and the primary coolant loop structure. The fuel cladding, core structure, pool liner, and primary piping are all made of aluminum alloy. A portion of the primary coolant loop is constructed of stainless steel. Lower pH will reduce aluminum alloy corrosion and oxide formation. Higher pH is favored to control stainless steel corrosion. Thus, a pH range between 5 and 7.5 is selected for the primary coolant. Electrical conductivity is also monitored to control purity of the primary coolant. With a limit of $\leq$5 µmho/cm no corrosion issues have been identified with either the fuel or the core structural materials since operations began in 1974. Conductivity may occasionally and briefly exceed the higher limit immediately following regeneration of the water purification system. Radionuclide analysis of the pool water allows for early determination of any significant buildup of radioactivity from operation of the reactor or the cobalt-60 source. Specifying the pool gate not be in position to isolate the bulk and stall pools during reactor operations assures the entire pool volume and surface area is available for cooling in normal and off-normal conditions.

3.4 CONFINEMENT

3.4.1 Operations Requiring Confinement

Applicability:
This specification applies to the reactor building.

Objective:
To restrict the release of airborne radioactive material into the environment in the event of an accident.

Specifications:
Confinement shall be maintained for any of the following conditions:

(1) The reactor is not secured.

(2) Movement of irradiated fuel is being performed, except when the fuel is in a properly sealed and approved shipping container.

(3) The handling of radioactive material with the potential for significant airborne release.

Bases:
Confinement provides means to isolate and release effluents through a controlled pathway, thereby mitigating possible radiological exposures to the public or workers. Reactor operation requires building confinement due to a remote possibility for the release of radioactive gasses or airborne particulates. It is not required when the reactor is shutdown. The movement of irradiated fuel introduces a remote possibility of fuel cladding damage. The handling of radioactive materials consisting of volatile, gaseous, or particulate components has the potential for creating an airborne release beyond 10CFR Part 20 Appendix B values for airborne radioactivity.

3.4.2 Conditions Needed to Achieve Confinement

Applicability:
This specification applies to the reactor building equipment required to achieve a confinement configuration.

Objective:
To prevent the release of reactor building exhaust air through other than defined pathways.

Specifications:
For any of the conditions in specification 3.4.1, the following equipment requirements shall be met:

(1) At least one door in each of the personnel air locks is sealed and the truck door is sealed.

(2) All ventilation isolation valves are either operable or in the closed (isolated) position.

Bases:
Effective confinement is achieved by maintaining a negative building pressure or by completely sealing the building. Chapter 6 of the SAR describes the building isolation equipment operation. The reactor building personnel airlocks and truck door are not provided with automatic closure devices. Confinement cannot be maintained if any of these portals are open to the outside atmosphere. The confinement building normal ventilation valves are designed to automatically seal upon an initiating signal from the Radiation Monitoring System or manual signal by the control room operator. If one or more ventilation valves are closed in the sealed position, then automatic isolation of that valve (or valves) is unnecessary.

3.5 VENTILATION SYSTEM

Applicability:
This specification applies to the normal and emergency exhaust ventilation equipment.

Objective:
To maintain a controlled pathway for reactor building exhaust air and minimize exposures from a release of airborne radioactive materials.

Specifications:
For any of the operations specified in 3.4.1:

(1) The main intake fan shall be operating.

(2) Any single or combination of exhaust fans capable of maintaining a minimum negative building pressure at or more negative than 0.1 inch water column shall be operable.

(3) The emergency exhaust system shall be operable.

(4) The emergency exhaust system charcoal filter shall have an efficiency of 95% or greater.

Bases:
Chapter 6 of the SAR describes the ventilation system operation. The main intake fan provides fresh air to the confinement building, or under the condition where building isolation occurs (see Chapter 6), dilution air up the stack. The main exhaust fan is designed to operate at a flow rate greater than the main intake fan in order to produce negative pressure in the building. In the event the main exhaust fan is not operating, the isolation valves are interlocked to divert the main intake up the stack. Negative building pressure can be maintained by one or a combination of smaller exhaust fans. In the unlikely event a release of fission products or other airborne radioactivity, an isolation signal from the Radiation Monitoring System or the control room operator will shutdown the main exhaust fan and close the building ventilation valves. The emergency exhaust system will start and purge the building air through charcoal and absolute filters, which is then diluted by the diverted intake air through the 100 foot exhaust stack.

3.6 RADIATION MONITORING SYSTEMS AND EFFLUENTS

3.6.1 Radiation Monitoring

Applicability:
This specification applies to the availability of radiation monitoring equipment which must be operable during reactor operation.

Objective:
To ensure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specifications:

(1) For any of the conditions in specification 3.4.1, the following minimum radiation monitors shall be operating with readouts and alarm indicators in the control room:

a. Stack gaseous and stack particulates radiation monitors.

b. A constant air monitor, located on the reactor pool level (third floor).

c. An area radiation monitor on the reactor experimental level (first floor).

d. An area radiation monitor over the reactor pool.

(2) Each gamma irradiation facility shall have an operating area radiation monitor having a readout and alarm indicator in the control room when irradiations are performed.

(3) If a required radiation monitor becomes inoperable, operations may continue only if the monitor can be repaired or replaced with a monitor of similar function within 1 hour of discovery.

(4) There shall be an environmental monitoring program that shall include the placement of dosimeters, or other devices at points outside the reactor building.

Bases:
The radiation monitoring system is described in Section 7.7 of the SAR. Specifications 1 and 2 provide the minimum equipment for evaluating the radiation levels within the stack effluent and within the reactor building. Specification 3 provides a reasonable time period to take corrective action after a failure of the minimum equipment is recognized. Specification 4 provides a means to assure exposures outside the restricted area are within regulatory limits.

3.6.2 Effluents

Applicability:
This specification applies to the monitoring and control of radioactive effluents from the reactor building.

Objectives:
To ensure that releases of liquid and airborne effluents are within 10 CFR Part 20 limits.

Specifications:
(1) The discharge of licensed material into sanitary sewage shall meet the requirements of 10CFR 20.2003(a), “Disposal by Release into Sanitary Sewerage.”

(2) The concentration of argon-41 into the unrestricted area shall not exceed the unrestricted area effluent concentration limit in 10CFR Part 20 Appendix B, Table 2, Column 1 for argon-41 when averaged over 1 year.

Bases:
Chapter 11 of the SAR evaluates liquid releases into the sanitary sewer system and evaluates the release of argon-41 to unrestricted areas. Both analyses show exposures are well within 10CFR Part 20 limits.

3.7 EXPERIMENTS

Applicability:
This specification applies to experiments to be installed in the reactor and associated experimental facilities.

Objective:
To prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

3.7.1 Reactivity Limits

Specifications:

(1) The absolute value of reactivity worth of any single movable experiment shall not exceed 0.25%Δk/k.

(2) The sum total absolute value of reactivity worth of all movable experiments shall not exceed 0.5%Δk/k.

(3) The absolute value reactivity worth of any single secured experiment shall not exceed 0.5%Δk/k.

(4) The sum total absolute reactivity worth of all secured experiments shall not exceed 2.5%Δk/k.

(5) The sum absolute value of reactivity worth of all experiments shall not be greater than 2.5%Δk/k.

Bases:
Specifications (1), (2) and (3) ensure that the failure of a single or multiple moveable experiments, or a single secured experiment, will not result in exceeding the Safety Limit. The analysis of a step insertion >0.5%Δ k/k is given in Chapter 13 of the SAR. The analysis shows the step-reactivity transient will not lead to ONB before the reactor protective system begins to shut down the transient. The total reactivity of 2.5% in Specifications (4) and (5) places a reasonable upper limit on the worth of all experiments which is compatible with the allowable excess reactivity and the shutdown margin and is consistent with the functional mission of the reactor.

3.7.2 Design and Materials

Specifications:
Experiments irradiated with either neutrons from the reactor or gamma rays from the Co-60 sources shall conform to the following:

(1) Experiments shall be designed such that a credible failure of the experiment shall not result in releases or exposures in excess of 10 CFR Part 20 limits.

(2) Experiments shall be designed such that a failure of an experiment shall not contribute to the failure of another experiment, core components, or principal physical barriers to uncontrolled release of radioactivity.

(3) All materials to be irradiated shall be either corrosion resistant or encapsulated within corrosion resistant containers to prevent interaction with reactor components, pool water, or Co-60 sources. Corrosive materials shall be doubly encapsulated. Should a failure of the encapsulation occur that could damage the reactor or Co-60 sources, the potentially damaged components shall be inspected.

(4) Explosive materials shall not be irradiated nor shall they be allowed to generate in any experiment in quantities over 25 milligrams of TNT-equivalent explosives. In addition, the irradiation container for this material shall be designed and tested for a pressure exceeding two times the maximum expected pressure.

(5) Each fueled experiment shall be limited such that the total inventory of iodine-131 through iodine-135 in the experiment is not greater than 100 mCi.

Bases
Specification (1) requires an evaluation to assure the experiment materials and apparatuses do not lead to airborne and/or area radiation exposures that could exceed 10 CFR Part 20 limits under credible failure conditions. Specification (2) requires an evaluation to assure materials and apparatuses used do not cause a failure of other experiments or structures, systems, or components (SSC) resulting in a radiological consequence. Specification (3) provides assurance that no unintended chemical reaction will take place that could adversely affect SSC resulting in a radiological consequence. Specification (4) provides assurance that the detonation of explosive materials will not lead to the failure of encapsulation and possible damage to the reactor or SSC resulting in a radiological consequence. Specification (5) limits the inventory of iodine radioisotopes to approximately one-half that used in the MHA analysis (SAR Chapter 13) in which the occupational and public dose consequences were determined to be well below 10CFR Part 20 regulatory limits.

3.8 Beam Port Operations

Applicability:
This specification applies to restrictions associated with operation of the beam ports.

Objective:
To prevent a loss of coolant accident that may cause the safety limit to be exceeded.

Specifications:

(1) The reactor shall not be operated with both the beam port lead shutter in the up (open) position and the beam-port shield plug removed.

(2) The shield plug may be substituted or modified so long as the overall open diameter does not exceed an area equivalent to 4 inches in diameter.

(3) In order to access the beam ports with both the lead shutter in the up position and shield plug removed, the reactor shall be positioned in the bulk pool.

Bases
A conservative loss of coolant analysis (SAR chapter 13) involving a beam port rupture demonstrates the beam ports can be safely utilized under the provided specifications.

End Section 3.

4.0 SURVEILLANCE REQUIREMENTS

Applicability:
This specification applies to the surveillance requirements of systems related to reactor safety.

Objective:
To verify the proper operation of systems related to reactor safety.

Specification:

A. Surveillance requirements may be deferred during reactor shutdown (except TS 4.1(7); 4.2.1(2); 4.3; and 4.6(2)); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

B. The appropriate surveillance testing on any Limiting Condition of Operation required equipment shall be conducted after replacement, repair, or modification before the equipment is considerable operable and returned to service.

Basis:
Specification 4A allows for the deferral of surveillances when the reactor is shut down provided they are performed prior to reactor operation or if operation is required to perform the surveillance, they are performed as soon as practical after reactor start up. This ensures that the requirements for limiting conditions of operation in accordance with section 3.0 are met. Specification 4B ensures that the affected LCO required equipment will operate as intended and as described in the SAR.

4.1 REACTOR CORE PARAMETERS

Applicability:
This specification applies to surveillance requirements for the various reactor core parameters.

Objective:
To ensure the reactor core parameters meet the specified limiting conditions for operation.

Specifications:

(1) The reactor core excess reactivity above reference core condition shall be verified annually or following any significant core configuration and/or control blade change. A significant core configuration change is defined as a change in reactivity greater than 0.2 % $\Delta k/k$.

(2) The shutdown margin shall be verified annually or following any significant core configuration and/or control blade change. A significant core configuration change is defined as a change in reactivity greater than 0.2 % $\Delta k/k$.

(3) Prior to the first reactor start-up of the day, visual verification shall be made that each core grid positions is filled with either a fuel element, a radiation basket, a source holder, the regulating rod, a graphite reflector element, a lead void box, or a grid plug.

(4) Prior to the first reactor start-up of the day, visual verification shall be made that all but 5 of the radiation baskets contain flow restricting devices. This specification shall be optional for low power operation <100 kW without forced flow.

(5) Prior to the first reactor start-up of the day, visual verification shall be made that the reactor is not in the same end of the reactor pool as any portion of the cobalt-60 source.

(6) Prior to the first reactor start-up of the day, a visual verification shall be made confirming the beam ports meet the criteria of TS 3.8.

(7) The linear and logarithmic power channels signals shall be checked against a heat balance annually.

(8) Visual inspection of one fifth of the in-core reactor fuel elements shall be performed every two years, such that all fuel elements in the core are inspected over a 10 year period.

Bases:
Specifications (1) through (4) provide verification the reactor is being operated within the nuclear and hydraulics design parameters used in the steady state and transient analyses. Specification (5) and (6) provide verification that the measured reactor power level is correct. Specification (7) provides verification of acceptable fuel condition.

4.2 REACTOR CONTROL AND SAFETY SYSTEMS

4.2.1 Control Blades

Applicability:
This specification applies to the surveillance requirements for operability of the reactor control blades.

Objective:
To ensure the control blades meet the specified limiting conditions for operation.

Specifications:
(1) Prior to the first reactor start-up of the day, all the control blades shall be verified as operable.

(2) The control blades shall be visually inspected annually.

(3) Control blade scram times and drive times, and regulating rod drive time shall be determined annually, or if maintenance or modification is performed on the mechanism.

Bases:
The operability checks, visual inspection of the control blades, and the measurements of scram times ensure that the blades are capable of operating properly and within the considerations used in transient analyses in Chapter 13 of the SAR. Drive times are used for calculating reactivity addition rates. Verification of operability after maintenance or modification of a drive mechanism will ensure proper operation after reinstallation or reconnection.

4.2.2 Rod Reactivity Insertion Rate

Applicability:
This specification applies to the surveillance requirements for the reactivity insertion rates.

Objective:
To ensure the reactivity insertion rates do not exceed the specified limiting conditions for operation.

Specifications:

(1) The reactivity worth and maximum reactivity insertion rate of the regulating rod and each control blade shall be determined annually or following any significant core configuration and/or change in a control blade or the regulating rod. A significant core configuration change is defined as a change in reactivity greater than 0.2 % $\Delta k/k$.

(2) Prior to the first reactor start-up of the day, the control blade drive system shall be tested to verify only control blade can be withdrawn at a time.

Bases:
The reactivity worth of the control blades and regulating rods is measured to ensure that the required shutdown margin is available, and to provide a means for determining the reactivity worths of experiments inserted in the core. Annual measurement of reactivity worths provides a correction for the slight variations expected because of burnup, and the required measurement after a core configuration change ensures that possibly altered rod worths will be known before routine operation.

4.2.3 Reactor Protection System Scrams

Applicability:
This specification applies to the surveillance requirements for the Reactor Protection System.

Objective:
To ensure Reactor Protection System limiting conditions for operation are met.

Specifications:

(1) A channel check of each channel listed in Specification 3.2.5, specific to the operating mode, shall be performed daily when the reactor is in operation.

(2) A channel test, including scram function where applicable, of each channel listed in Specification 3.2.5, specific to the operating mode, shall be performed prior to each day's operation, or prior to each operation extending more than one day.

(3) A channel calibration of the reactor power level channels (Linear and Log-N), and the period channel shall be made annually.

(4) Thermal power level shall be verified annually.

(5) A channel calibration of the following channels shall be made annually:
   a. Pool water temperature
   b. Primary coolant flow rate
   c. Pool water level
   d. Primary coolant inlet temperature

(6) The manual scram in the control room shall be verified to be operable prior to each day’s operation, or prior to each operation extending more than one day.

(7) All scrams listed in Specifications 3.2.3 items 8 – 15 and 3.2.4 shall be verified operable annually.

(8) The interlocks listed in Specification 3.2.6 shall be verified operable annually.

Bases:
The daily channel tests and checks and periodic verifications will ensure that channels used to measure the process variables are operable. Annual calibrations will ensure that any long term drift of the process measuring channels is corrected. Appropriate annual tests of other scrams in the scram chain and control system interlocks will ensure that those functions not tested before daily operation remain operable.

4.3 COOLANT SYSTEMS

Applicability:
This specification applies to verifying the quality of the primary coolant system water and the pool configuration.

Objective:
To ensure the primary coolant system limiting conditions for operation are met.

Specifications:

(1) The conductivity and pH of the pool water shall be measured weekly.

(2) The radioactivity in the pool water shall be analyzed monthly.

(3) The pool water shall be either monitored continuously for Co-60 or sampled once per week.

(4) Prior to the first reactor start-up of the day, the pool divider shall be verified as open.

Bases
The pH and conductivity reading are administratively recorded as part of the reactor checkout procedure. A minimum weekly measurement is consistent with the recommendations in ANSI/ANS 15.1. Monthly radionuclide analysis of the pool water samples will allow early determination of any significant buildup of radioactivity from operation of the reactor. Either continuous or weekly pool water sampling for determining if the Co-60 sources are leaking is consistent with the original technical specification for monitoring of source leakage.

4.4 CONFINEMENT

Applicability:
This specification applies to the surveillance requirements for the reactor building confinement.

Objective:
To ensure the confinement limiting conditions for operation are met.

Specifications:

(1) Prior to any of the operations specified in 3.4.1 the main intake fan shall be verified as operating.

(2) Prior to any of the operations specified in 3.4.1 and at no less than 8 hour intervals during, the building pressure compared to ambient shall be verified at or more negative than 0.1 inch water column.

(3) The confinement system shall be functionally tested semi-annually.

Bases
An initial verification of intake fan is operating assures that adequate dilution air is available during operations requiring confinement. Periodic measurement for negative pressure ensures that any confinement building leakage is inward. Semi-annual functional tests of the isolation system provide adequate assurance the isolation system performs as designed.

4.5 VENTILATION SYSTEMS

Applicability:
This specification applies to the surveillance requirements for the confinement building emergency exhaust system.

Objective:
To ensure the emergency exhaust system limiting conditions for operation are met.

Specifications:

(1) An operability check of the emergency exhaust system shall be performed quarterly or following any maintenance or modifications that could affect the operability of the system.

(2) The carbon filter in the emergency exhaust system shall be tested biennially.

Bases
Surveillance of the emergency exhaust system and the periodic testing of the carbon filter will verify the system is functioning as described in Chapter 6 of the SAR.

4.6 RADIATION MONITORING EQUIPMENT

Applicability:
This specification applies to the surveillance requirements for the area radiation monitoring equipment and systems for monitoring airborne radioactivity.

Objective:
To ensure radiation monitoring equipment limiting conditions for operation are met.

Specifications:

(1) A channel test of the radiation monitoring channels in Specification 3.6.1(1) shall be made prior to each day's operation.

(2) The radiation monitoring channels in the radiation monitoring system shall be calibrated and the trip set points verified when initially installed and annually thereafter.

Bases:
The channel tests verify the channel operability by the introduction of a test signal. The calibration provides a complete verification of the performance of the channel. An annual calibration is based upon manufacturer recommendations and is sufficient to ensure the required reliability.

End Section 4

5.0 DESIGN FEATURES

5.1 SITE AND FACILITY DESCRIPTION

Applicability:
These specifications apply to the physical location of the reactor and supporting structures.

Objective:
To specify the bounds of the facility.

Specifications:

(1) The reactor and associated equipment shall be located at 1 University Avenue, Lowell, Massachusetts.

(2) The facility is the area under the reactor license. It shall include the reactor building, designed for confinement, and the attached three story building. The reactor building shall be the restricted area as defined in 10 CFR Part 20. The reactor building shall have a minimum free volume of 335,000 ft^3 that is exhausted through a 100 ft. high stack. The three story building attached to the reactor building shall include spaces necessary for supporting licensed activities including radiation protection, emergency preparedness, physical security, and the reactor building ventilation.

Bases:
The site on which the reactor building is located is detailed in chapter 2 of the SAR. Chapters 3 and 6 provide details of the reactor building and its design features. The attached three story building includes spaces described in the radiation safety program (SER Chapter 11), in the Emergency Preparedness Plan, and in the Physical Security Plan.

5.2 REACTOR COOLANT SYSTEM

Applicability:
These specifications apply to the reactor pool and primary coolant system.

Objective:
To specify the major design features of the reactor coolant system.

Specifications:
The reactor coolant system shall consist of the following:

(1) An open pool containing approximately 75,000 gallons of demineralized water (H_2O).

(2) A single cooling loop containing a heat exchanger, a circulation pump, and various valves.

(3) All materials associated with the reactor coolant system shall be aluminum alloys, except for the heat exchanger which may include stainless steel components, and small non-corrosive components such as gaskets, filters, and valve diaphragms.

Bases:
Chapter 5 of the SAR provides detail on the reactor cooling system.

5.3 REACTOR CORE AND FUEL

Applicability:
These specifications apply to reactor core and fuel.

Objective:
To specify design features of the reactor core and fuel and allowable fuel configurations.

Specifications:

(1) The reactor core shall consist of a 9 x 7 array of 3-inch square modules with the four corners occupied by posts.

(2) Cores shall contain 21 elements to 26 elements, consisting of any combination of fuel elements as described in specifications 5.3.3, 5.3.4, and 5.3.5.

(3) A standard fuel element shall be either:

a. A flat plate MTR-type element having plates fueled with low enrichment (<20% U-235) U_3Si_2, clad with aluminum. There shall be 18 plates per element with 16 plates containing fuel and two outside plates of aluminum. There shall be 200 ± 2 grams of Uranium-235 per element when new, or

b. A flat plate MTR-type element having plates fueled with low enrichment (<20% U-235) UAl_x, clad with aluminum. There shall be 18 plates per element. There shall be 167 ± 2 grams of Uranium-235 per element when new.

(4) A partial fuel element shall be the same as Specification 5.3(3-a) except each plate shall have approximately half the uranium loading. No more than two (2) partial fuel elements shall be allowed in the core.

(5) A removable plate fuel element shall be the same as Specification 5.3(3-b), except the fuel plates are removable. No more than one (1) removable plate element shall be allowed in the core.

(6) Prior to operating the reactor with a removable plate element, a safety analysis shall be performed for each core configuration and configuration of the element to assure there are no changes to the safety margins

presented in the SAR. The analysis shall be reviewed and approved by the reactor safety subcommittee.

(7) The average fission density shall not exceed 2 x 10^{21}fissions/cm^3.

Bases:
Chapter 4 of the SAR provides details of the core design which are included in the safety analyses. The analyses show the reactor can be safely operated with any combination of U_3Si_2 and UAl_x fuel, including 2 partial U_3Si_2 elements, for core loadings from 21 to 26 elements. The removable plate element provides for numerous configurations of placement in the core and number of plates used, requiring each configuration to be separately analyzed before use. The negative reactivity coefficients are described in the SAR Chapter 4 and used in the transient analyses (SAR Chapter 13). NUREG-1313 provides data indicating fission densities up to 2.5 x 10^{21}fissions/cm^3 are acceptable.

5.4 FISSIONABLE MATERIAL STORAGE

Applicability:
These specifications apply to the storage of reactor fuel when not in the core and the storage of other fissionable material.

Objective:
To ensure that stored fuel or other fissionable material does not become critical and will not reach an unsafe temperature.

Specifications:

(1) Fuel, including fueled experiments and fueled devices not in the reactor shall be stored in the restricted area and in a configuration that ensures adequate cooling and is designed to maintain k_{eff} less than 0.9 under all conditions of moderation and reflection.

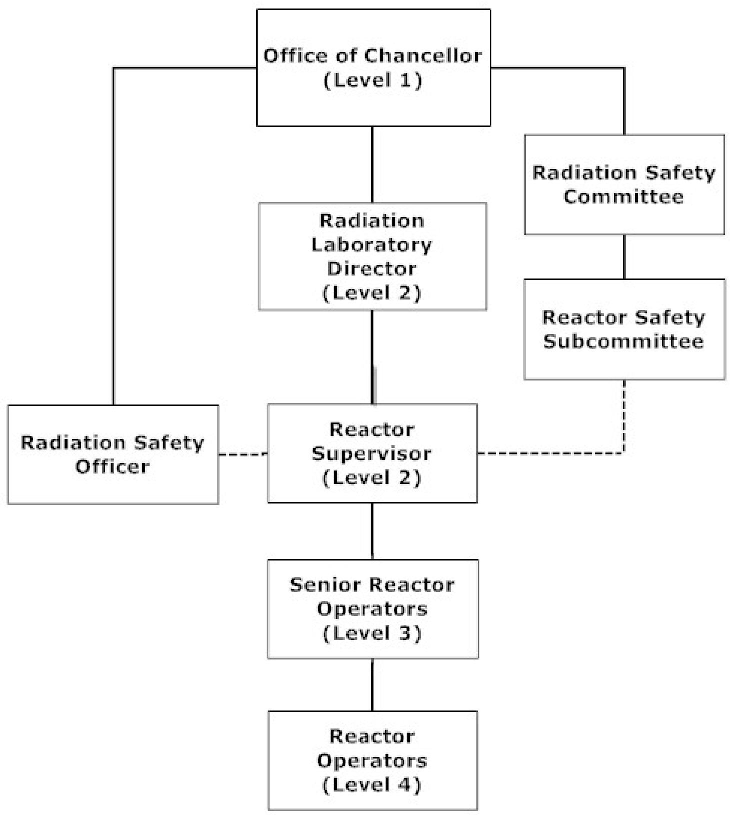
(2) Where a licensed shipping container is used, the k_{eff} and cooling design considerations of the container shall apply.

Bases:
Specification (1) assures criticality is not attained and temperatures do not reach a level where damage could occur. Specification (2) allows for shipments.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

6.1.1 Structure:
The organization for the management and operation of the research reactor facility in matters related to the license and these technical specifications shall be as shown in Figure 6-1.



Reporting Line
Communication Line

Figure 6-1

6.1.2 Responsibility:

(1) The Chancellor shall designate an individual (Level 1), at a position of associate vice chancellor or higher, to be responsible for ensuring that all research, education, and service activities at the research reactor facility are conducted in accordance with applicable federal, state and local regulations.

(2) The Reactor Supervisor (Level 2) shall be directly responsible for the safety of all operations at the research reactor facility, and in all matters pertaining to these Technical Specifications.

(3) In all matters pertaining to safe operation of the reactor and to these Technical Specifications, the Reactor Supervisor shall report to and be directly responsible to the Director of the Radiation Laboratory (Level 2).

(4) The UML Radiation Safety Officer shall be responsible for radiation protection at the UMLRR and shall advise the Reactor Supervisor on all matters pertaining to radiation protection.

(5) In matters pertaining to radiation safety, the UML Radiation Safety Officer shall report to and be directly responsible to the Level 1 individual in the Office of the Chancellor.

6.1.3 Staffing

(1) The following shall be the minimum staffing when the reactor is not secured:

a. A reactor operator or senior reactor operator shall be in the control room.

b. A second designated person shall be present at the facility. This individual shall be a senior reactor operator, reactor operator or an individual able to carry out prescribed written instructions.

c. If a senior reactor operator is not at the facility, a senior reactor operator shall be readily available on call. "Readily available on call" shall mean an individual who:

1. has been specifically designated and the designation known to the operator on duty,

2. keeps the operator on duty informed of where he/she may be rapidly contacted and the phone number, and

3. is capable of getting to the facility within a reasonable time

under normal conditions (e.g., 30 minutes or within a 15-mile radius).

(2) A list of reactor facility personnel by name and telephone number shall be readily available for use in the control room. The list shall include:

a. management personnel,

b. radiation safety personnel, and

c. other operations personnel

(3) The following events shall require the presence of a senior reactor operator at the facility:

a. initial startup and approach to power.

b. all fuel or control-rod relocations within the reactor core region.

c. recovery from an unplanned or unscheduled shutdown or power reduction of 200kW or greater.

6.1.4 <u>Selection and Training of Personnel</u>

(1) The Director of the Radiation Laboratory shall be a tenured faculty member in a science or engineering discipline.

(2) The selection, training, and requalification of operations personnel shall meet or exceed the requirements (most current revision) of American National Standard, ANSI/ANS-15.4 “Selection and Training of Personnel for Research Reactors.” (R2007 or later revision).

6.2 REVIEW AND AUDIT

There shall be a Reactor Safety Subcommittee (RSSC) which shall review reactor operations to ensure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The RSSC shall be a subcommittee of the University Radiation Safety Committee which has overall authority in the use of all radiation sources at the University.

6.2.1 Composition and Qualifications

The RSSC shall be composed of at least five members, one of whom shall be the Radiation Safety Officer and another of whom shall be the Reactor Supervisor. Members of the RSSC shall be knowledgeable in the areas of reactor operation and radiation safety. The membership of the RSSC shall include at least two faculty members from the engineering or science disciplines. Members shall be appointed by the Office of the Chancellor Level 1 designee. The RSSC chairman shall be elected from among the membership and shall not have line responsibility for operation of the reactor.

6.2.2 Charter and Rules

The RSSC shall follow the rules specific to it under the charter and rules of the Radiation Safety Committee. Notwithstanding that charter and rules, the RSSC functions shall be conducted as follows:

(1) Meetings shall be held at least once per calendar year and more frequently as circumstances warrant, consistent with effective monitoring of facility activities.

(2) A meeting quorum shall consist of at least one-half of the membership where the reactor staff does not constitute a majority.

(3) Meeting minutes shall be distributed to RSSC members within three months of the meeting.

6.2.3 Review Function

(1) The RSSC shall review the following:

a. Evaluations performed as required by 10 CFR 50.59.

b. All new procedures and major revisions thereto having safety significance and proposed changes in reactor facility equipment or systems having safety significance.

c. All new experiments or classes of experiments.

d. Proposed changes in technical specifications or license.

e. Violations of technical specifications or license and violations of internal procedures having safety significance.

f. Operating abnormalities having safety significance.

g. Reportable occurrences listed in specification 6.6.2.

h. Audit reports.

(2) A written report or minutes of the findings and recommendations of the RSSC shall be submitted to the Level 1 individual in the Office of the Chancellor and to the RSSC members within three months after a review has been completed.

6.2.4 Audit Function

(1) Audits of the following functions shall be performed by an individual or group without immediate responsibility for the area being audited.

(2) The scope of the audit shall include, as a minimum, the following:

a. Facility operations for conformance to the technical specifications and license conditions on an annual basis.

b. The requalification program for the operating staff on a biennial basis.

c. Corrective actions associated with deficiencies in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety on an annual basis.

d. The reactor facility emergency plan and implementing procedures on a biennial basis.

(3) Deficiencies uncovered that affect reactor safety shall immediately be reported to the Chancellor's Level 1 designee. . A written report of the findings of the audit shall be submitted to the Chancellor's Level 1 designee and to all RSSC members within three months after the audit has been completed.

6.3 RADIATION SAFETY

(1) The Radiation Safety Program shall be designed to achieve the requirements of 10 CFR 20 and should use the guidelines in American National Standard, ANSI/ANS-15.11 "Radiation Protection at Research Reactor Facilities." (R2009 or later revision).

(2) The Radiation Safety Program shall be the responsibility of the Radiation Safety Officer, having line authority as indicated in Figure 6-1.

(3) The Radiation Safety Program shall include management commitment to maintain exposures and releases as low as reasonably achievable.

6.4 OPERATING PROCEDURES

(1) Written procedures shall be reviewed by the RSSC and approved by the Reactor Supervisor or designee, and shall be in effect and followed for the following items. The procedures shall be adequate to ensure the safe operation of the reactor and gamma irradiation facilities, but should not preclude the use of independent judgment and action should the situation require such.

a. startup, operation, and shutdown of the reactor;

b. fuel loading, unloading, and movement within the reactor;

c. maintenance of major components of systems that could have an effect on reactor safety;

d. surveillance checks, calibrations, and inspections required by the technical specifications or those that may have an effect on reactor safety;

e. personnel radiation protection specific to the facility radiation monitoring system and the use of the reactor experimental facilities and the gamma irradiation facilities;

f. administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;

g. for the conduct of irradiations and experiments in the gamma irradiation facilities;

h. implementation of required plans such as emergency or security plans;

i. use, receipt, and transfer of byproduct material.

(2) Deviations from procedures may be made by a senior reactor operator (Level 3). Such deviations shall be documented, reviewed pursuant to 10 CFR 50.59, and reported within 24 hours or the next working day to the Reactor Supervisor or designee.

6.5 EXPERIMENTS REVIEW AND APPROVAL

(1) All new experiments or class of experiments shall be reviewed by the RSSC, subject to the requirements of 10CFR 50.59, and approved in writing by the Reactor Supervisor or designated alternate prior to initiation.

(2) Substantive changes to previously approved experiments shall be made only after review by the RSSC, subject to the requirements of 10CFR 50.59, and approved in writing by the Reactor Supervisor or designated alternate prior to initiation.

6.6 REQUIRED ACTIONS

6.6.1 Action To Be Taken In The Event The Safety Limit Is Exceeded

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorization is obtained from the NRC.

(2) The safety limit violation shall be promptly reported to the Reactor Supervisor or his designee, the Chancellor's Level 1 designee, and the Chairman of the RSSC.

(3) The safety limit violation shall be reported to the NRC in accordance with specification 6.7.2.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

a. The time and date of the violation, reactor status at the time of the violation, and a description of the violation.

b. The applicable circumstances leading to the violation including, when known, the cause and contributing factors.

c. The effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

d. Corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed and approved by the RSSC and shall be submitted to the NRC in accordance with specification 6.7.2 of this document.

6.6.2 Action To Be Taken in the Event of a Reportable Occurrence

(1) A reportable occurrence shall be any of the following conditions:

a. Release of radioactivity from the reactor confinement building above allowed limits.

b. Operating with any safety system setting less conservative than that stated in Section 2.2 these specifications.

c. Operating in violation of a limiting condition for operation established in Section 3.0 of these specifications unless prompt remedial action is taken as specified in Section 3.

d. A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report shall be required.

e. An uncontrolled or unanticipated change in reactivity in excess of 0.6%Δk/k. Reactor trips resulting from a known cause are excluded.

f. An abnormal and significant degradation in reactor fuel and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks).

g. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

(2) In the event of a reportable occurrence, the following actions shall be taken:

a. If involving the reactor, the reactor conditions shall be returned to normal, or the reactor shall be shutdown, to correct the occurrence. If shutdown, the reactor shall not be operated until authorized by the Reactor Supervisor.

b. If involving a gamma irradiation facility, the conditions shall be returned to normal, or gamma facilities operations shall cease, to correct the occurrence. If operations cease, the gamma irradiation facility shall not be operated until authorized by the Reactor Supervisor.

c. The Reactor Supervisor shall be notified as soon as possible.

d. The Nuclear Regulatory Commission shall be notified in accordance with specification 6.7.2.

e. A report shall be prepared that includes the time and date of the occurrence, facility status at the time of the occurrence, a description of the occurrence, an evaluation of the cause of the occurrence, a record of the corrective action taken, and recommendations for appropriate action to prevent or reduce the probability of recurrence. This report shall be reviewed by the RSSC no later than its next regularly scheduled meeting.

6.7 REPORTS

6.7.1 Operating Reports

An annual or operating report shall be submitted to the NRC Document Control Desk within ninety days following the 30th of June of each year. Its content shall include:

(1) A narrative summary of reactor operating experience including a tabulation showing the energy generated by the reactor (in megawatt days), the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

(2) The number of emergency shutdowns and inadvertent scrams, including the reasons therefore, and where applicable, corrective actions to preclude recurrence.

(3) Tabulation of major preventive and corrective maintenance operations having safety significance.

(4) A tabulation of changes in the facility and procedures and of tests and experiments carried out pursuant to 10 CFR 50.59.

(5) A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the licensee, as determined at, or before, the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is <25% of the 10 CFR 20 Appendix B concentration limits, a statement to this effect is sufficient.

(6) A summarized result of environmental surveys performed outside the facility.

(7) A summary of exposures received by facility personnel and visitors where such exposures are >25% of the regulatory limits in 10 CFR 20.

6.7.2 Special Reports

(1) A report shall be made not later than the following working day by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event and sent within 14 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, of any of the following:

a. Operation in violation of a safety limit.

b. Any release of radioactivity to unrestricted areas above permissible limits.

c. Any reportable occurrence as defined in Specification 6.6.2.

(2) A written report shall be provided as a follow-up to the verbal one within 14 days of the occurrence. This report shall provide the information required by Specification 6.6.2(2). The report shall be submitted to the NRC Document Control Desk.

(3) A written report shall be submitted within 30 days to the NRC Document Control Desk in the event of:

a. A permanent change in the personnel serving as Level 1or Level 2.

b. Any significant change in the transient or accident analyses as described in the SAR.

6.8 RECORDS

6.8.1 Five-Year Record Retention

The following records shall be retained for five years or for the life of the component involved if less than five years:

(1) Records of normal reactor operation including power levels and periods of operation at each power level. (Note: Excludes retention of supporting documents such as checklists, log sheets, etc., which shall be retained for a period of at least one year.)
(2) Records of principal maintenance activities including inspection, repair, substitution, or replacement of principal items of equipment pertaining to nuclear safety.
(3) Records of reportable occurrences.
(4) Records of surveillance activities that are required by these technical specifications.
(5) Records of reactor facility radiation and contamination surveys.
(6) Records of experiments performed with the reactor.
(7) Records of fuel inventories, receipt, and shipments.
(8) Records of changes made in the operating procedures.
(9) Minutes of the RSSC and audit reports including both internal audits and those performed for or by the RSSC.

6.8.2 Six-Year Record Retention

Records of individual licensed staff members indicating qualifications, experience, training, and requalification shall be retained at all times that the individual is employed or until the operator license is renewed.

6.8.3 Records To Be Retained for the Life of the Facility

Applicable annual reports, if they contain all of the required information, may be used as records in this section.

(1) Gaseous and liquid radioactive effluents released to the environs.
(2) Off-site environmental-monitoring surveys required by the technical specifications.
(3) Radiation exposure for all personnel monitored.
(4) Drawings of the reactor facility.
(5) Reviews and reports pertaining to a violation of a safety limit, limiting safety system setting, or limiting conditions for operations.

End Section 6