IV UNIVERSITY OF LOWELL

TECHNICAL SPECIFICATIONS

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Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included 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.

1.0 DEFINITIONS

1.1 ABNORMAL OCCURRENCES - An abnormal occurrence is any of the following:

a. Any actual safety system setting less conservative than specified in Paragraph 12.2.2 of these Technical Specifications;

b. Operation in violation of a limiting condition for operation;

c. Safety system component malfunction or other component or system malfunction which could, or threaten to, render the system incapable of performing its intended function;

d. Release of fission products from a fuel element in a quantity that would indicate a fuel element cladding failure;

e. An uncontrolled or unanticipated change in reactivity greater than 0.5% delta k/k;

f. 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; and

g. Conditions arising from natural or offsite manmade events that affect or threaten to affect the safe operation of the facility.

1.2 CHANNEL CALIBRATION - A channel calibration is an adjustment of the channel such that its output responds, with acceptable range and accuracy, to known values of the parameter which the channel measures.

1.3 CHANNEL CHECK - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with other independent channels or methods measuring the same variable.

1.4 CHANNEL TEST - A channel test is the introduction of an input signal into the channel to verify that it is operable.

1.5 CONTAINMENT BUILDING INTEGRITY - Integrity of the containment building is said to be maintained when all isolation system equipment is operable or secured in an isolating position.

1.6 CONTROL ROD OR SHIM SAFETY ROD - A control rod (or shim safety rod) is one having a magnet, scram capability, and a drive mechanism.

1.7 INDEPENDENT EXPERIMENTS - Experiments not connected by a mechanical, chemical, or electrical link.

1.8 MEASURED VALUE - The measured value of a parameter is the value as it appears at the output of a measuring channel.

1.9 MEASURING CHANNEL - A measuring channel is the combination of sensors, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable.

1.10 MOVABLE EXPERIMENT - A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into or out of the reactor while the reactor is operating.

1.11 <u>NON-SECURED EXPERIMENTS</u> - Experiments where it is intended that the experiment should not move while the reactor is operating, but is held in place with less restraint than secured experiment.

1.12 <u>OPERABLE</u> - A system or component is operable when it is capable of performing its intended function in a normal manner.

1.13 <u>OPERATING</u> - A system or component is operating when it is performing its intended function in a normal manner.

1.14 <u>REACTOR OPERATING MODE</u> - Reactor operating mode refers to the method by which the core is cooled, either natural convection mode of operation or forced convection mode of operation.

1.15 <u>PROTECTIVE CHANNEL</u> - A protective channel is a safety channel in the reactor safety system which is not a measuring channel.

1.16 <u>REACTOR OPERATION</u> - The reactor is in operation when it is not secured.

1.17 <u>REACTOR SAFETY SYSTEM</u> - The reactor safety system is that combination of safety channels and associated circuitry which forms the automatic protective system of the reactor, or provides information which requires manual protective action to be initiated.

1.18 <u>REACTOR SECURED</u> - That overall condition where all of the following conditions are satisfied:

a. the reactor is shutdown;

b. electrical power to the control rod circuits is switched off and switch key is in proper custody; and

c. no work is in progress involving fuel or incore experiments, or maintenance of the core structure, control rods, or control rod drives.

1.19 REACTOR SHUTDOWN - That condition where all control rods are fully inserted or reactivity condition equivalent to one where all control rods are fully inserted.

1.20 REGULATING ROD - The regulating rod is one with a drive mechanism, without scram capability, and of specified maximum reactivity worth.

1.21 SAFETY CHANNEL - A safety channel is a measuring or protective channel in the reactor safety system.

1.22 SECURED EXPERIMENT - A secured experiment is an experiment or experimental facility held firmly in place by a mechanical device or by gravity, such that the restraining forces are substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment or by forces which can arise as a result of credible malfunctions.

1.23 SHUTDOWN MARGIN - Shutdown margin is the amount of negative reactivity by which the reactor is subcritical.

1.24 TRUE VALUE - The true value of a parameter is its exact value at any instant.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 Safety limits in the forced convection mode of operation.

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance with forced convection flow. These variables are:

P = Reactor Thermal Power

W = Reactor Coolant Flow Rate

T_i = Reactor Coolant Inlet Temperature

L = Height of water above the center line of the core

Objective

To assure that the integrity of the fuel cladding is maintained.

Specification

Under the conditions of forced convection flow:

1. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 1 under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time above the curve shown in Figure 1.

2. The true value of pool water level (L) shall not be less that 24 feet above the center line of the

core.

3. The true value of reactor coolant inlet temperature (T_i) shall not be greater than 110°F.

Above 34% of the full core flow of 1400 gpm in the region of full power operation, the criterion used to establish the safety limit was the onset of nucleate boiling at the hot spot in the hot channel. The analysis is given in Paragraph 9.1.1 of the FSAR.

In the region below 34% of full core flow, where under a loss of flow transient at power, the flow coasts down, reverses, and then establishes natural convection, the criterion for selecting a safety limit was taken as the fuel cladding temperature (FSAR Paragraph 9.1.1.7). The analysis of a loss of flow transient is presented in Paragraphs 9.1.2.3.1 and 9.1.3 of the FSAR.

For initial conditions of full flow and operating power of 1 MW under the conservative assumptions of the analysis, no Safety Limit is violated during the transient. The Safety Limit shown in Figure 1 for flow less than 34% of full flow is the steady state power corresponding to the maximum fuel clad temperature of 250°F with natural convection flow, namely, 0.66 MWt.

2.1.2 Safety Limits in the natural convection mode of operation.

Applicability

This specification applies to the interrelated variables associated with core thermal and hydraulic performance with natural convection flow. These variables are:

P = Reactor Thermal Power

T_p = Reactor Pool Temperature

L = Height of water above the center line of the core

Objective

To assure that the integrity of the fuel cladding is maintained.

Specification

Under conditions of natural convection flow:

1. The true value of the reactor thermal power (P) shall not exceed 0.66 MW.
2. The true value of the reactor thermal power (P) shall not exceed 1.33 kW when the true value of the pool water level (L) is less that 2 feet above the center line of the core.
3. The reactor shall not be taken critical when the true value of the pool water level (L) is less than 2 feet above the center line of the core.
4. The true value of the reactor coolant inlet temperature (pool temperature, T_p) shall not be greater than 110°F.

Bases

The criterion for establishing a safety limit with natural convection flow is established as the fuel clad temperature. This is consistent with Figure 1 for forced convection flow during a transient. The analysis of natural convection flow given in paragraph 9.1.1.7 of the FSAR shows that at 0.66 MW the maximum fuel clad temperature is 250°F which is well below the temperature at which fuel clad damage could occur. Operation of the reactor with less than full water height above the core is limited to a power nearly 500 times lower than the limit with full water height; there is no possibility of fuel clad damage under water immersion at 1.33 kW.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 Limiting Safety System Settings in the forced convection mode of operation.

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), reactor coolant inlet temperature (T_i), and the height of water above the center line of the core (L).

Objective

To assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specification

Under conditions of forced convection flow the values of the Limiting Safety System Settings shall be as follows:

P = 1.25 MWt (max)

W = 1170 gpm (min)

T_i = 108°F (max)

$L = 24.25$ ft (min)

Bases

The Limiting Safety System Settings that are given in Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in an automatic protective actions that will prevent Safety Limits from being exceeded during the course of the most adverse anticipated transient. To determine the LSSS given above, an analysis of the uncertainties in the instruments and measurements was taken into account. These safety settings are adjusted so that the true value of the measured parameter will not exceed the specified Safety Limits. The results of these adjustments included a flow variation of 4%, a temperature variation of $2^{o}F$, a power level variation of 6%, and a pool water level variation of three inches. (See FSAR Paragraph 9.1.2.)

2.2.2 Limiting Safety System Settings in the natural convection flow mode of operation.

Applicability

This specification applies to the setpoints for the safety channel monitoring reactor thermal power (P), reactor pool temperature (T_p), and the height of water above the center line of the core (L).

Objective

To assure that automatic protective action is initiated in order to prevent an undesirable level of ^{16}N.

Specification

Under conditions of natural convection flow the measured values of the Limiting Safety System Settings shall be as follows:

P = 125 kW (max) P = 1.25 kW (max)

T_p = 108°F (max) or T_p = 108°F (max)

L = 24.25 ft (min) L = 2.25 ft (min)

Bases

The Limiting Safety System Settings that are given in Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent an undesirable level of production and escape of ^{16}N. The specifications given above assure that an adequate safety margin exists between the LSS s and the SL s for natural convection, because the values of the power LSSS would be much higher (620 kW, Paragraph 9.1.2.3.2 of the FSAR) if the specifications were based on Safety Limits rather than on ^{16}N production. The ^{16}N criterion is not related to fuel clad damage which was the criterion used in establishing the Safety Limits (see Specification 12.2.1.2). It is desirable to eliminate to the greatest extent possible ^{16}N release during upflow in natural convection.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTIVITY

Applicability

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

Objective

To assure that the reactor can be shut down at all times and that the Safety Limits will not be exceeded.

Specification

The reactor shall not be operated unless the following conditions exist:

1. The minimum shutdown margin relative to the cold, clean (xenon-free) critical condition, with the most reactive shim safety rod in the fully withdrawn position, is greater than 3% delta k/k.
2. The reactor core is loaded so that the excess reactivity in the cold clean (xenon-free) critical condition does not exceed 4.7% delta k/k.
3. All core grid positions are filled with fuel elements, irradiation baskets, source holders, servo regulating rod, graphite reflector elements or grid plugs. For low power, <10 kW, physics tests without forced flow, this specification will not apply.
4. The drop time of each shim safety rod from a fully withdrawn position is less than 1.0 second.
5. The isothermal temperature coefficient of reactivity

is negative at temperatures >70°F.

6. The reactivity insertion rates of the shim safety rods are less than 0.025% delta k/k per second.

7. The total reactivity worth of the regulating rod is less than the effective delayed neutron fraction.

8. The reactivity worth of experiments shall not exceed the values indicated in the following table:

Kind	Single Experiment Worth	Total Worth
Movable (includes the pneumatic rabbit)	0.1% delta k/k	0.5% delta k/k together
Non-secured	0.5% delta k/k	
Secured	0.5% delta k/k	2.5% delta k/k

9. The total reactivity worth of all experiments shall not be greater than 2.5% delta k/k.

Bases

1. The shutdown margin required by Specification 1 assures that 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.

2. The maximum allowed excess reactivity of 4.7% delta k/k 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 rod fully withdrawn.

3. The requirement that all grid plate positions be filled during reactor operation assures that the quantity of primary coolant which bypasses the heat producing elements will be kept within the limits used in establishing Safety Limits in Paragraph 9.1 of the FSAR. Under natural circulation conditions at low power, this requirement does not apply.
4. The rod drop time required by Specification 4 assures that the Safety Limit will not be exceeded during the flow coast down which occurs upon loss of forced convection coolant flow. The analysis of this situation, which is given in Paragraphs 9.1.2.3.1 and 9.1.3 of the FSAR, assumes a 1-second rod drop time.
5. The requirement for a negative temperature coefficient of reactivity assures that any temperature rise caused by a reactor transient will not cause a further increase in reactivity.
6. The maximum rate of reactivity insertion by the shim safety rods which is allowed in Specification 6 assures that the Safety Limit will not be exceeded during a startup accident due to a continuous linear reactivity insertion. Analysis in Paragraph 9.1.11 of the FSAR shows that a maximum power of less than 1.4 MW would be reached assuming a continuous linear reactivity insertion rate of 0.035% delta k/k per second, which is greater than the maximum allowed.

7. Limiting the reactivity worth of the regulating rod to a value less than the effective delayed neutron fraction assures that a failure of the automatic servo control system could not result in a prompt critical condition.
8. Specification 8 assures that the failure of a single experiment will not result in the exceeding of a Safety Limit; the analysis of the step insertion of 0.5% delta k/k is given in Paragraph 9.1.10 of the FSAR. Limiting a movable experiment such as the pneumatic rabbit to 0.1% delta k/k assures that the prompt jump, which is about 17%, will result in a power below the power level scram setting, i.e., below 125% of power.
9. The total reactivity of 2.5% in Specification 9 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.

3.2 REACTOR INSTRUMENTATION

Applicability

This specification applies to the instrumentation which must be available and operable for safe operation of the reactor.

Objective

The objective is to require that sufficient information be available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the measuring channels listed in the following table are operable:

Measuring Channel	Minimum Required	Operating Mode in Which Required
Pool Water Level	1	All modes
Startup Count Rate	1	All modes (during reactor startup)
Log N	1	All modes
Power Level (Linear N)	2	All modes
Reactor Coolant Inlet Temperature	1	Forced convection
Coolant Flow Rate	1	Forced convection
Reactor Pool Temperature	1	Natural convection

Bases

The neutron detectors assure that measurements of the reactor power level are adequately displayed during reactor startup and low and high power operation.

The temperature and flow detectors give information to the operator to prevent the exceeding of a Safety Limit.

3.3 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the reactor safety system channels.

Objective

To require the minimum number of reactor safety system channels that must be operable in order to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the reactor safety system channels described in the following table are operable.

Reactor Safety System Component/Channel	Minimum Required	Function	Operating Mode in Which Required
Startup Count Rate	1	Prevent blade withdrawal when N count rate $\leq$ 2 cps.	Reactor startup in all modes
Reactor Period	1	Automatic reactor scram with $\leq$ 3 sec period	All modes
Reactor Power Level	2	Automatic scram when $\geq$ 125% of range scale	All modes
Coolant Flow Rate	2	Automatic scram at 1170 gpm	Forced convection above 0.1 MW
Coolant Inlet Temperature	1	Automatic scram $\geq$ 108°F	Forced convection above 0.1 MW
Pool Temperature (Amendment #1)	1	Automatic scram $\geq$ 108°F	All modes
Pool Water Level	1	Automatic scram at: (1) 24.25 ft above core center line (2) 2.25 ft above core center line	All modes above 1.25 kW (measured value) Operation below 1.25 kW (measured value)
Seismic Disturbance	1	Automatic scram at Modified Mercalli Scale IV	All modes
Primary Piping Alignment	1	Automatic scram	Forced convection above 0.1 MW

Reactor Safety System Component/Channel	Minimum Required	Function	Operating Mode in Which Required
Bridge Movement	1	Automatic scram if moved $\geq$1 inch	All modes
Coolant Gates Open (Amendment #1)	1	Automatic scram if either the coolant riser or coolant downcomer gates open	Forced convection above 0.1 MW; down comer flow pattern
Coolant Gate Opens (Amendment #1)	1	Automatic scram if the coolant riser gate opens	Forced convection above 0.1 Mw; cross pool flow pattern
High Voltage Failure in Control Console	1	Automatic scram if Voltage $\leq$500V.	All modes
Thermal Column Door Open	1	Automatic scram	All modes
Truck Door and/or Air Lock Integrity	3	Automatic scram	All modes
Manual Scram Button	1	Manual scram	All modes
"Reactor On" Key-Switch	1	Manual scram	All modes

Bases

The inhibit function on the startup channel assures the required startup neutron source is sufficient and in a proper location for the reactor startup, such that a minimum source multiplication count rate level is being detected to ensure proper operation of the startup channel.

The automatic protective action initiated by the reactor period channel, high flux channels, flow rate channels, coolant inlet temperature channel, pool temperature channel, and pool water level channel provides the redundant protection to assure that a Safety Limit is not exceeded.

Automatic protective action initiated by the seismic detector, bridge misalignment, opening of coolant gates, high voltage failure, and opening of thermal column door assures shutdown of the reactor under conditions that could lead to a safety problem.

The automatic protective action covering the condition of the air lock doors assures that containment capability is maintained.

The manual scram button and the "Reactor On" Key-Switch provide two manual scram methods to the operator if any abnormal condition should occur.

3.4 RADIATION MONITORING EQUIPMENT

Applicability

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

Objective

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

Specification

1. When the reactor is operating, gaseous and

particulate sampling of the stack effluent will be effected by the stack monitor which has a readout in the control room.

2. When the reactor is operating, at least one constant air monitoring unit located in the containment building on the reactor pool level and having a readout in the control room shall be operating.
3. The reactor shall not be continuously* operated without a minimum of one radiation monitor on the experimental level of the reactor building and one monitor over the reactor pool operating and capable of warning personnel of high radiation levels.

Bases

A continuing evaluation of the radiation levels within the reactor building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Chapter 10 of the FSAR.

A continuing evaluation of the stack effluent will be made using the information recorded from the particulate and gas monitors.

*In order to continue operation of the reactor, replacement of an inoperative monitor must be made within 15 minutes of recognition of failure, except that the reactor may be operated in a steady-state power mode if the installed systems are replaced with portable gamma-sensitive instruments having their own alarm.

3.5 CONTAINMENT AND EMERGENCY EXHAUST SYSTEM

Applicability

This specification applies to the operation of the reactor containment and emergency exhaust system.

Objective

To assure that the containment and emergency exhaust system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated unless the following equipment is operable, and conditions met:

Equipment/Condition	Function
1. At least one door in each of the personnel air locks is closed and the truck door is closed.	To maintain containment system integrity
2. All isolation valves, except that reactor operation can proceed if a failed isolation valve is in the closed (isolated) position.	To maintain containment system integrity
3. Initiation system for containment isolation.	To maintain containment system integrity
4. Emergency exhaust system	To maintain the ability to tend toward a negative building pressure without unloading any large fraction of possible air-borne activity.

Equipment/Condition	Function
5. Vacuum relief device	To ensure that building underpressure will not exceed 0.2 psi.
6. Reactor alarm system*	To assure that proper emergency action is taken.

Bases

In the unlikely event of a release of fission products, or other airborne radioactivity, the containment isolation initiation system will secure the normal ventilation exhaust fan, will bypass the normal ventilation supply up the stack, and will close the normal inlet and exhaust valves. In containment, the emergency exhaust system will tend to maintain a negative building pressure with a combination of controls intended to prevent unloading any large fraction of airborne activity if the internal building pressure is high. The emergency exhaust purges the building air through charcoal and absolute filters and controls the discharge, which is diluted by supply air, through a 100-foot stack on site. Chapter 3 of the FSAR describes the system's sequence of operation; Chapter 9 provides the analysis.

*The public address system can serve as a temporary substitute for reactor evacuation and formation of the Emergency Team during short periods of maintenance.

3.6 LIMITATIONS OF EXPERIMENTS

Applicability

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

Objectives

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

Specification

The reactor shall not be operated unless the following conditions governing experiments exist:

1. All materials to be irradiated shall be either corrosion resistant or encapsulated within corrosion resistant containers to prevent interaction with reactor components or pool water. Corrosive materials shall be doubly encapsulated.
2. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2.
3. Explosive material such as (but not limited to) gunpowder, dynamite, TNT, nitroglycerine, or PETN in quantities $<$ 25 mg may be irradiated in the reactor or experimental facilities provided out-of-core

tests indicate that, with the containment provided, no damage to the explosive containers, the reactor or the reactor components shall occur upon detonation of the explosive.

4. Explosive materials, in any quantity, shall not be allowed in the reactor pool or experimental facilities without special authorization from the USNRC. (Amendment #5.)
5. All experiments shall be designed against failure from internal and external heating at the true values associated with the LSSS for reactor power level and other process variables.
6. The outside surface temperature of a submerged experiment or capsule shall not exceed the saturation temperature of the reactor coolant during operation of the reactor.
7. Experimental apparatus, material or equipment to be irradiated shall be positioned so as not to cause shadowing of the nuclear instrumentation, interference with control rods, or other perturbations which may interfere with safe operation of the reactor.
8. Cyrogenic liquids shall not be used in any experiment within the reactor pool.
9. 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. (Amendment

#5.)

Bases

Specifications 1 through 6 are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure, including those involving the Co-60 Source and, along with the reactivity restriction of pertinent specification in 12.3.1, serve as a guide for the review and approval of new and untried experiments by the operations staff as well as the Reactor Safety Subcommittee. (Amendment #5.)

Specification 7 assures that no physical or nuclear interferences compromise the safe operation of the reactor by, for example, tilting the flux in a way that could affect the peaking factor used in the Safety Limit calculations. Review of the experiments using the appropriate LCO's and the Administrative Controls of Section 6 assures that the insertion of experiments will not negate the consideration implicit in the Safety Limits.

Specification 8 insures NRC review of experiments containing or using cyrogenic materials. Cyrogenic liquids present structural and explosive problems which enhance the potential of an experiment failure. (Amendment #2.)

Specification 9 assures that there will be no interference, either instrumental or procedural between the reactor and the cobalt source during reactor operation. (Amendment #5.)

3.7 GASEOUS EFFLUENTS

Applicability

This specification applies to the routine release of gaseous radioactive effluents from the facility.

Objective

The objective is to minimize the release of gaseous radioactive effluents, particularly argon-41, the effluent most likely to be generated in routine operation.

Specification

The release rate of gaseous radioactive material from the reactor stack shall be limited to 8 microcuries per second averaged over a year.

Bases

Calculations based on a very conservative model, allowing for no atmospheric dilution of the gaseous effluent, have predicted an annual dose of 12 mr to an individual exposed to the effluent on a continual basis for an argon-41 release rate of 8 microcuries per second. Allowance for even minimal atmospheric turbulence would reduce this dose number by about a factor of three.

4.0 SURVEILLANCE REQUIREMENTS

4.1 CONTROL AND REGULATING RODS

Applicability

This specification applies to the surveillance requirements for the control and regulating rods.

Objective

To assure the operability of the control and regulating rods.

Specifications

1. The reactivity worth of the regulating rod and each control rod shall be determined at intervals not to exceed 13 months. The reactivity worth of all rods shall also be determined prior to routine operation of any new fuel configuration in the reactor core.
2. Control rod drop and drive times and regulating rod drive time shall be determined at intervals not to exceed 13 months, or if maintenance or modification is performed on the mechanism.
3. The control and regulating rods shall be visually inspected at intervals not to exceed 13 months.

Bases

The reactivity worth of the control and regulating rods is measured to assure that the required shutdown margin is available, and to provide a means for determining the reactivity worths of experiments inserted in the core. The measurement of reactivity worths on a 13-month basis provides a correction for the slight variations expected because of burnup, and the required measurement after any new arrangement of fuel in the core assures that possibly altered rod worths will be known before routine operation.

The visual inspection of the regulating and control rods and the measurements of drive and drop times are made to assure that the rods are capable of performing properly and within the considerations used in transient analyses in Chapter 9 of the FSAR. Appropriate inspection data will be recorded and analyzed for trends. Verification of operability after maintenance or modification of the control system will ensure proper reinstallation or reconnection.

4.2 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the surveillance requirements for the Reactor Safety System.

Objective

To assure that the Reactor Safety System (RSS) will remain operable and will prevent the Safety Limits from being exceeded.

Specifications

1. A channel check of each measuring channel in the RSS shall be performed daily when the reactor is in operation.
2. A channel test of each measuring channel in the RSS shall be performed prior to each day's operation, or prior to each operation extending more than one day.
3. A channel calibration (reactor power level) of the Log N and linear safety power level measuring

channels shall be made at intervals not to exceed 13 months.

4. A channel calibration of the following channels shall be made at intervals not to exceed 13 months:
   a. Pool water temperature
   b. Primary coolant flow rate
   c. Pool water level
   d. Primary coolant inlet and outlet temperature
5. The manual scram shall be verified to be operable prior to each reactor startup.
6. Any RSS instrument channel replacement must have undergone a channel calibration prior to installation.
7. Any RSS instrument repaired or replaced while the reactor is shutdown must have a channel test prior to reactor operation.
8. Each protective channel in the RSS shall be verified to be operable at intervals not to exceed 7 months.

Bases

The daily channel tests and checks and periodic verifications will assure that the safety channels are operable. The calibrations at 13-month intervals will assure that long-term drift of the channels is corrected. The calibration of the reactor power level will provide continual reference for the adjustment of the Log N and safety channel detectors positions and current alignments.

4.3 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 assure that the equipment used for monitoring radioactivity is operable and to verify the appropriate alarm settings.

Specification

1. The operation of the area radiation monitoring equipment and systems for monitoring airborne radioactivity, and their associated alarm set points, shall be verified prior to reactor startup.
2. All radiation monitoring systems shall be calibrated semiannually, but at intervals not to exceed eight months.

Bases

The area radiation monitoring system, described in the Emergency Plan, includes the stack air monitor, two building constant air monitors, a fission product monitor, 12 GM detectors and two ion chamber detectors at selected sites throughout the building. The detectors used have been chosen for stability and operational reliability. The large number of detectors in the area monitoring system ensures that if a particular monitor should malfunction or drift out of calibration, sufficient backup monitors are available for reliable information. Calibration of the area monitors at eight-month intervals is sufficient to insure the required reliability. Daily checks (during operating days) of the area monitors ensure that any obvious malfunctions will be corrected.

4.4 CONTAINMENT BUILDING

Applicability

This specification applies to the surveillance requirements for the containment building.

Objective

To assure that the containment system is operable.

Specification

1. Building pressure will be verified at least every eight hours during reactor operation to ensure that it is less than ambient atmospheric pressure.
2. The containment building isolation system including the initiating system shall be tested at intervals not exceeding six months. The test shall verify that valve closure is achieved in $<$ 2.5 seconds after the initial signal.
3. An integrated leakage rate of the containment building as-is* shall be performed at a pressure of at least 0.5 psig at intervals of 24 months (+ 4

*Non-routine maintenance or repair for the purpose of reducing containment leakage shall not be performed prior to the leak test.

mo.) to verify leakage rate of less than 10% of the building air volume/day at 2 psig. (Amendment #3.)

4. All additions, modifications, or maintenance of the containment building or its penetrations that could effect building containment capability shall be tested to verify containment requirements.

5. At intervals not to exceed 12 months (+ 2 mo.) the emergency exhaust system including the initiating system shall be verified to be operable.

6. At intervals not to exceed 24 months, and subsequent to replacement of the facility filters and prior to reactor operation thereafter, the filter trains shall be tested to verify that they are operable.

7. At intervals not to exceed 24 months, the air flow rate in the stack exhaust duct shall be measured.

Bases

Maintaining a negative pressure ensures that any leakage in the containment is inward.

Valve closure time was chosen to be 1/2 the time required for a given sample of air to travel from the first to the second valve in series in the exhaust line under regular flow conditions. Semi-annually is considered a reasonable frequency of testing.

The containment building was designed to withstand a 2.0 psig internal pressure. An overpressure of less than 0.5 psig would result from an excursion of 538 MWs, which is nearly four times the energy release achieved in the Borax tests. A 0.5 psig test pressure is therefore adequate.

Any additions, modifications or maintenance to the building or its penetrations shall be tested to verify that such work has not adversely affected the leaktightness of the building.

Surveillance of the emergency exhaust system and the various filters will verify that these are functioning. See Chapters 3 and 7 of the FSAR.

4.5 POOL WATER

Applicability

This specification applies to the surveillance requirement for monitoring the quality and the radioactivity in the pool water.

Objective

To assure high quality pool water and to monitor the radioactivity in the pool water in order to verify the integrity of the fuel cladding.

Specification

1. The conductivity of the pool water shall be measured weekly and shall be maintained at a value of 5 micromhos per centimeter or less averaged over a month. (Amendment #1.)
2. The radioactivity in the pool water shall be analyzed weekly (at intervals not to exceed 10 days). The pool water shall be analyzed for gross activity and for Cobalt-60. Analysis shall be

capable of detecting levels of $1x10^{-7}$ microcuries per milliliter. If a sample analysis reveals a significant increase of activity in the water, with respect to previous samples or a contamination level greater than $1x10^{-6}$ microcuries of Cobalt-60 per milliliter of water, prompt action shall be taken to prevent further contamination of pool water. If the gross activity of the sample is less than $1x10^{-7}$ microcuries per milliliter, specific analysis for Cobalt-60 need not be performed. If remedial action is required by this section, notification will be made to the USNRC within 10 days as required by Section 6.6.b(1) of these Technical Specifications. (Amendment #5.)

Bases

Surveillance of water conductivity assures that changes that could accelerate corrosion have not occurred. Radionuclide analysis of the pool water samples will allow early determination of any significant buildup of radioactivity from operation of the reactor or the Co-60 source. (Amendment #5.)

4.6 EMERGENCY POWER SYSTEM

Applicability

This specification applies to the emergency electrical power equipment.

Objective

To assure that the emergency power equipment is maintained in an operable condition.

Specification

The natural-gas generator shall be tested for proper operation at least monthly. All emergency power equipment shall be tested under a simulated complete loss of outside power at least annually.

Bases

These tests of the emergency power equipment will provide assurance that there is a source of power available for operating emergency equipment in the event of power failure at the site.

4.7 SCRAM BY PROCESS VARIABLE EFFECT

Applicability

This specification applies to the surveillance requirements applied to process variable scrams.

Objective

To assure that a Safety Limit is not exceeded.

Specification

Following a reactor scram caused by a process variable, the reactor shall not be operated until an evaluation has been made to determine if a safety limit was exceeded, the cause of the scram, the effects of operation to the scram point and the appropriate action to be taken.

Bases

This specification assures that if a safety limit should be exceeded as a result of a malfunction of process variable, the fact will be known.

5.0 DESIGN FEATURES

5.1 REACTOR FUEL

The reactor fuel shall be as follows:

1. Standard fuel element: the fuel elements shall be flat plate MTR-type elements. The plates shall be highly enriched (93%) uranium-aluminum alloy fuel clad with aluminum. There shall be 135 (+ 4) grams of uranium-235 per element. There shall be 18 plates per fuel element.
2. Half-element: same as a standard fuel element except each plate has one half the uranium loading.
3. Variable-load element: same as Specification 1 above but internal plates are removable.

5.2 REACTOR CORE

1. The reactor core consists of a 9 x 7 array of 3-inch square modules with the four corners occupied by posts. The reference core for these technical specifications consists of 26 standard fuel elements arranged symmetrically around four safety control blades as shown in Figure 4.23 of the FSAR.
2. Cores from 23 standard elements to 30 elements may be used, and cores from 24 elements to 30 elements may contain 2 half-loaded elements.
3. Cores with an internal fuel element replaced by a radiation basket may be operated under natural convection only after flux measurements made under natural convection establish that no alteration of

the LSSS's are required to preclude violation of a SL during the transients anticipated in the FSAR. (Amendment #1.)

5.3 REACTOR BUILDING

1. The reactor shall be housed in the reactor building, designed for containment. The minimum free volume in the reactor bulding shall be 8.5×10^9 cm^3 (300,000 ft^3).
2. The reactor building ventilation and containment systems shall be separate from the rest of the building systems and shall be designed to exhaust air or other gases from the building through a stack with a discharge at a minimum of 100 feet above ground level.
3. The openings into the reactor building are the truck entrance door, personnel entrance doors, and air supply and exhaust ducts.

5.4 FUEL STORAGE

All reactor fuel element storage facilities shall be designed in a geometrical configuration where k_{eff} is less than 0.8 under quiescent flooding with water.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION AND MANAGEMENT

1. The reactor facility shall be an integral part of the Radiation Laboratory of the University of Lowell. The reactor shall be related to the University structure as shown in Chart 6-1 and Chart

6-2. (Amendment #8.)

2. The reactor facility shall be under the direction of the Administrative Assistant for the Reactor and Accelerator, who shall be a member of the graduate faculty, and it shall be supervised by the Reactor Supervisor who shall be an NRC-licensed senior operator for the facility. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Reactor Safety Subcommittee. (Amendment #8.)
3. There shall be a Radiation Safety Officer responsible for the safety of operations from the standpoint of radiation protection. He does not report to the line organization responsible for reactor operations, but rather to the Vice-President for Academic Services and Technical Research (see Chart 6-2). (Amendment #8.)
4. An operator or senior operator licensed pursuant to 10 CFR 55 shall be present at the controls unless the reactor is secured as defined in these specifications.
5. A Licensed Senior Operator shall be readily available on call within the Pinanski Building whenever the reactor is in operation. (Amendment #8.)

6.2 REVIEW AND AUDIT

1. There shall be a Reactor Safety Subcommittee which shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The Subcommittee shall report to the Institute Radiation Safety Committee which has overall authority in the use of all radiation sources at the Institute.
2. The responsibilities of the Subcommittee include, but are not limited to, the following:
   a. Review and approval of normal, abnormal and emergency operating and maintenance procedures and records.
   b. Review and approval of proposed tests and experiments utilizing the reactor facilities in accordance with Paragraph 6.8 of these specifications.
   c. Review and approval of proposed changes to the facility systems or equipment, procedures, and operations.
   d. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question requiring a change to the Technical Specifications or facility license.
   e. Review of all violations of the Technical

Specifications and NRC Regulations, and significant violations of internal rules or procedures, with recommendations for corrective action to prevent recurrence. (Amendment #2.)

f. Review of the qualifications and competency of the operating organization to assure retention of staff quality.

3. The Reactor Safety Subcommittee 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. The Subcommittee shall be proficient in all areas of reactor operation and reactor safety. The membership of the Subcommittee shall include at least two senior scientific staff members, and the chairman will not have line responsibility for operation of the reactor.

4. The Subcommittee shall have a written charter defining such matters as the authority of the Subcommittee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Subcommittee. Minutes of all meetings of the Subcommittee shall be kept.

5. A quorum of the Subcommittee shall consist of not less than a majority of the full Subcommittee and shall include the Radiation Safety Officer or his designee, the Reactor Supervisor or his designee,

and the chairman or his designee.

6. The Subcommittee shall meet at least quarterly after reactor operation commences.

6.3 OPERATING PROCEDURES

Written procedures, reviewed and approved by the Reactor Safety Subcommittee and approved by the Professor in Charge shall be in effect and followed for the following items. The procedures shall be adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1. Startup, operation, and shutdown of the reactor.
2. Installation or removal of fuel elements, control rods, experiments and experimental facilities.
3. Actions to be taken to correct specific and potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, and abnormal reactivity changes.
4. Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.
5. Maintenance procedures which could have an effect on reactor safety.
6. Periodic surveillance of reactor instrumentation and safety systems, area monitors and continuous air monitors.

7. Civil disturbance on or near campus.

8. Radiation control procedures shall be maintained and available to all operations personnel.

9. Receipt, inspection, and storage of new fuel elements.

10. Handling and storage of irradiated fuel elements.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Subcommittee. Temporary changes to the procedures that do not change their original intent may be made by the Reactor Supervisor. Temporary changes to procedures shall be documented and subsequently reviewed by the Reactor Safety Subcommittee.

6.4

ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE

In the event of an abnormal occurrence:

The Reactor Supervisor shall be notified promptly and

1. corrective action shall be taken immediately to place the facility in a safe condition until the causes of the abnormal occurrence are determined and corrected.

2. The Reactor Supervisor shall report the occurrence to the Reactor Safety Subcommittee. The report shall include an analysis of the cause of the occurrence, corrective actions taken, and recommendations for appropriate action to prevent or reduce the probability of a repetition of the

occurrence.

3. The Reactor Safety Subcommittee shall review the report and the corrective actions taken. They shall inform the Professor in Charge with their findings for appropriate action.

4. Notification shall be made to the NRC in accordance with Paragraph 6.6 of these specifications. (Amendment #2.)

6.5

ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event a Safety Limit has been exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorization is obtained from the NRC. (Amendment #2.)

2. Immediate notification shall be made to the NRC in accordance with paragraph 6.6 of these specifications, to the Professor in Charge, and to the Director of the Nuclear Center. (Amendment #2.)

3. A prompt report shall be prepared by the Reactor Supervisor and approved by the Professor in Charge. The report shall include a complete analysis of the causes of the event and the extent of possible damage together with recommendations to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Subcommittee for review and appropriate action, and a suitable similar report shall be submitted to the

NRC in accordance with Paragraph 6.6 of these specifications and in support of a request for authorization for resumption of operations. (Amendment #2.)

6.6 REPORTING REQUIREMENTS

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made to the NRC as follows: (Amendment #2.)

1. Within 24 hours, a report by telephone and telegraph to NRC Region I Office of Inspection and Enforcement (OI&E) of:
   (Amendment #2.)
   a. Any accidental release of radioactivity to unrestricted areas above permissible limits, whether or not the release resulted in property damage, personal injury or exposure.
   b. Any significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor.
   c. Any abnormal occurrences as defined in Paragraph 1.1 of these specifications.
   d. Any violation of a Safety Limit.
2. A written report within 10 days to the Director of the Region I OI&E of:

(Amendment #2.)

a. Any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure; the written report (and, to the extent possible, the preliminary telephone and telegraph report) shall describe, analyze and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.

b. Any significant variation of measured values from a corresponding predicted or previously measured value of safety-related operating characteristics occurring during operation of the reactor.

c. Incidents or conditions relating to operation of the facility which prevented or could have prevented the performance of engineered safety features as described in these specifications.

d. Any abnormal occurrences as defined in Paragraph 1.1 of these specifications.

e. Any violation of a Safety Limit.

3. A written report within 30 days to the Director of the Regional I OI&E of:

(Amendment #2.)

a. Any substantial variance from performance specifications contained in these specifications or in the Final Safety Analysis Report.

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

c. Any observed inadequacies in the implementation of administrative or procedural controls.

4. A written report to the Director of the Region I OI&E within 60 days after completion of startup testing of the reactor upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level, describing the measured values of the operating conditions or characteristics of the reactor under the new conditions, including:

(Amendment #2.)

a. An evaluation of facility performance to date in comparison with design predictions and specifications; and

b. A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

5. An annual report shall be submitted in writing to the Director of the Region I OI&E within 60 days following the 30th of June of each year, providing the following information:

(Amendment #2.)

a. A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period, as well as results of surveillance tests and inspections.

b. 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.

c. The number of emergency shutdowns and inadvertent scrams, including the reasons therefore.

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required.

e. A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10 CFR 50 including a summary of the safety evaluation of each.

f. A description of any environmental surveys performed outside the facility.

g. A summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposures, and a summary of the results of radiation and contamination surveys performed within the facility.

h. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

<u>Liquid Waste</u> (Summarized on a monthly basis)

(1) Total gross beta radioactivity released (in curies) during the reporting period.

(2) Total radioactivity released (in curies) for specific nuclides, if the gross beta radioactivity exceeds 3×10^{-6} Ci/cm^3 at point of release, during the reporting period.

(3) Average concentration (Ci/cm^3) of release as diluted by sewage system flow of 2.7×10^8 cm^3/day.

<u>Gaseous Waste</u> (summarized on a monthly basis)

(1) Radioactivity discharged during the reporting period (in curies) for:

(a) Gases

(b) Particulates, with half lives greater than eight days.

(2) The MPC used and the estimated activity (in curies) discharged during the reporting period, by nuclide, based on representative isotopic analysis.

Solid Waste

(1) The total amount of solid waste packaged (in cubic feet).

(2) The total activity and type of activity involved (in curies).

(3) The dates of shipment and disposition (if shipped off-site).

6.7 PLANT OPERATING RECORDS

In addition to the requirements of applicable regulations and in no way substituting therefore, records and logs of the following items, as a minimum, shall be kept in a manner convenient for review and shall be retained as indicated:

1. Records to be retained for a period of at least five years:

   a. Reactor operations;

   b. Principal maintenance activities;

   c. Experiments performed including aspects of the experiments which could affect the safety of reactor operation or have radiological safety implications;

   d. Abnormal occurrences; and

e. Equipment and component surveillance activities.

2. Records to be retained for the life of the facility:

   a. Gaseous and liquid radioactive effluents released to the environs;

   b. Off-site environmental monitoring surveys;

   c. Facility radiation and monitoring surveys;

   d. Personnel radiation exposures;

   e. Fuel inventories and transfers;

   f. Changes to procedures, systems, components, and equipment; and

   g. Updated, "as-built" drawings of the facility.

6.8 APPROVAL OF EXPERIMENTS

1. All proposed experiments using the reactor shall be evaluated by the experimenter and a staff member who has been approved by the Reactor Safety Subcommittee. The evaluation shall be reviewed by the Reactor Supervisor and the Radiation Safety Officer to ensure compliance with the provisions of the facility license, these Technical Specifications, and 10 CFR 20. If the experiment complies with the above provisions, it shall be submitted by the Reactor Supervisor to the Reactor Safety Subcommittee for approval if it is a new experiment, as indicated in 4. below. The experimenter evaluation shall include:

   a. The reactivity worth of the experiment;

   b. The integrity of the experiment, including the effect of changes in temperature, pressure, chemical composition, or radiolytic decomposition;

c. Any physical or chemical interaction that could occur with the reactor components;

d. Any relation hazard that may result from the activation of materials or from external beams; and

e. An estimate of the amount of radioactive materials produced.

2. Prior to performing any new reactor experiment, an evaluation of the experiment shall be made by the Reactor Safety Subcommittee. The subcommittee evaluation shall consider:

a. The purpose of the experiment;

b. The effect of the experiment on reactor operation and the possibility and consequences of failure of some aspect of the experiment, including, where significant, chemical reactions, physical integrity, design life, proper cooling interaction with core components, and reactivity effects;

c. Whether or not the experiment, by virtue of its nature and/or design, includes an unreviewed safety question or constitutes a significant threat to the integrity of the core, the integrity of the reactor, or to the safety of personnel; and

d. A procedure for the performance of the experiment.

A favorable subcommittee evaluation shall conclude that failure of the experiment will not lead to direct failure of any reactor component or of other experments.

An experiment shall not be conducted until a favorable evaluation indicated in writing is rendered by the Reactor Safety Subcommittee.

3. In evaluating experiments, the following assumptions shall be used for the purpose of determining that failure of the experiment would not cause the appropriate limits of 10 CFR 20 to be exceeded:
   a. If the possibility exists that airborne concentration of radioactive gases or aerosols may be released within the containment building, 100% of the gases or aerosols will escape;
   b. If the effluent exhausts through a filter installation designed for greater than 90% efficiency for 0.3 micron particles, at least 10% of gases or aerosols will escape; and
   c. For a material whose boiling point is above 130°F and where vapors formed by boiling this material could escape only through a volume of water above the core, at least 10% of these vapors will escape.

4. An experiment that has had prior subcommittee approval and has been performed safely shall be a routine experiment and requires only the approval of the Reactor Supervisor and the Radiation Safety Officer to be repeated.

   An experiment that represents a minor variation from a routine experiment not involving safety considerations of a different kind nor of a magnitude greater than a routine experiment shall be considered the equivalent of a routine experiment and may be approved for the subcommittee by agreement of the Reactor Supervisor and the Radiation Safety Officer.