UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

STATE UNIVERSITY OF NEW YORK AT BUFFALO

DOCKET NO. 50-57

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. R-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to Facility Operating License No. R-77, filed by the State University of New York at Buffalo (the licensee), dated March 8, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's Regulations as set forth in 10 CFR Chapter I:

   B. The facility will operate in conformity with the amended license, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

   F. Publication of notice of this amendment is not required since it does not involve a significant hazards consideration nor amendment of a license of the type described in 10 CFR Section 2.106(a)(2).

2. Accordingly, the license is amended by reissuing the technical specifications to include the necessary changes throughout. Paragraph 2.C.2 of License No. R-77 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 20, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Attachment:
Appendix A Technical
Specifications Changes

Date of Issuance: August 19, 1985

APPENDIX A

FACILITY LICENSE NO. R-77

TECHNICAL SPECIFICATIONS

FOR THE

BUFFALO MATERIALS RESEARCH CENTER

STATE UNIVERSITY OF NEW YORK AT BUFFALO

DOCKET NO. 50-57

DATE: AUGUST 1985

Amendment No. 20

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1.0 DEFINITIONS

Channel Calibration: A channel calibration is an adjustment of the channel so that its output corresponds, with acceptable accuracy, to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall include a channel test.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification should include comparison of the channel with other independent channels or systems measuring the same variable, where this capability exists.

Channel Test: A channel test is the introduction of a signal into the channel to verify that it is operating.

Control Blade: A neutron absorbing blade used to control core reactivity that is not magnetically coupled to its drive unit.

Control-Safety Blade: A neutron absorbing blade used to control the reactivity of the core. A control-safety blade is magnetically coupled to its drive unit allowing it to perform the function of a safety device when the magnet is energized.

Experiment: An experiment is any of the following:

(1) an activity using the reactor system or its components or the neutrons or radiation generated therein

(2) an evaluation or test of a reactor system operation, surveillance, or maintenance technique

(3) an experimental or testing activity that is conducted within the confinement or containment system of the reactor

(4) the material content of any of the foregoing, including structural components, encapsulation or confining boundaries, and contained fluids or solids

Experimental Facility: An experimental facility is any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

Explosive Material: Explosive material is any solid or liquid that is categorized as a Severe, Dangerous, or Very Dangerous Explosive Hazard in "Dangerous Properties of Industrial Materials" by N. I. Sax, Third Ed. (1968), or is given an Identification of Reactivity (Stability) index of 2, 3 or 4 by the National Fire Protection Association in its publication 704-M, 1966, "Identification System for Fire Hazards of Materials," also enumerated in the "Handbook for Laboratory Safety" 2nd Ed. (1971) published by the Chemical Rubber Co.

Fast Scram: Fast scram is a rapid reduction of the magnet holding current of the control-safety blades until the blades fall by gravity into the reactor core.

Fuel Assembly: A grouping of fuel elements that is not taken apart during the charging and discharging of a reactor core.

Fuel Element: The smallest structurally discrete part of a reactor that has fuel as its principal constituent (same as fuel pin).

Limiting Condition for Operation: Limiting conditions for operation (LCO) are those administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility (10 CFR 50.36).

Limiting Safety System Settings: Limiting safety system settings (LSSS or LS^3) are for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be chosen so that the automatic protective action will correct the abnormal situation before a safety limit is exceeded (10 CFR 50.36).

Measured Value: The measured value of a process variable is the value of the variable as indicated by a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, amplifiers, and output devices that are used for the purpose of measuring the value of a process variable.

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

Operable: Operable means that a component or system is capable of performing its intended function in its normal manner.

Operating: Operating means that a component or system is performing its intended function in its normal manner.

Permanent Experimental Facility: Those experimental facilities that would require considerable effort and planning to remove or alter such as beam tubes, thermal column, etc.

Potential Reactivity Worth of an Experiment: The potential reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter equipment position or configuration.

Reactivity Limits: The reactivity limits are those limits imposed on reactor core excess reactivity. Quantities are referenced specifically to a cold core (80 - 100°F) with the effect of xenon poisoning on core activity accounted for if greater than or equal to 0.05% $\Delta k/k$. The reactivity worth of samarium in the core will not be included in excess reactivity limits. The reference core condition will be known as the cold, xenon-free critical condition.

Reactor Operations: Reactor operation means (1) that the control blades installed in the core are not fully inserted, (2) that the console key is in the keyswitch, or (3) that manipulations are being conducted in the pool that could affect core reactivity.

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

Reactor Secured: The reactor is secured when a shutdown checklist has been completed.

Reactor Shutdown: The reactor is considered shut down if all control-safety blades are fully inserted, the console key is removed, and no manipulations are being conducted in the pool that could affect core reactivity. When the reactor is shut down, an operator must be in the facility but not necessarily in the control room.

Readily Available/On Call: Readily available/on call shall mean that the licensed senior operator shall ensure that he is within a reasonable driving time (1 hour) from the reactor building. The licensed senior operator shall always keep the licensed operator informed of where he may be contacted.

Removable Experiment: A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

Reportable Occurrence: A reportable occurrence is any of the conditions described in Section 6.4 of these specifications.

Rundown: Rundown is the automatic insertion of the control-safety blades.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Safety Limits: Safety limits (SLs) are limits upon important process variables, which are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity (10 CFR 50.36).

Secured Experiment: A secured experiment is any experiment, experimental facility, 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.

Slow Scram: Slow scram is the shutoff of electrical power to the units providing the magnet holding current with subsequent decay of the magnet holding current until the blades fall by gravity into the reactor core.

Static Reactivity Worth: The static reactivity worth of an experiment is the absolute value of the reactivity change, which is measurable by calibrated control or regulating rod comparison methods between two defined terminal positions or configurations of the experiment. For removable experiments, the terminal positions are fully removed from the reactor and fully inserted or installed in the normal functioning or intended position.

True Value: The true value of a process variable is its actual value at any instant.

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 to include shutdowns that occur during testing or checkout operations.

Unsecured Experiment: An unsecured experiment is any experiment, experimental facility, or component of an experiment that is not a secured experiment as defined above.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

2.1.1 Safety Limits in Forced Convection Mode

Applicability: These specifications apply to pertinent process variables, the variation of which may potentially compromise fuel or cladding integrity, during forced convection operation.

Objective: The objective is to protect the reactor core from the loss of fuel or cladding integrity during operation in the forced convection mode.

Specifications:

(1) The true value of reactor thermal power (P) shall not exceed 3.3 MW.

(2) The true value of pool water level (H) from the surface to the top of the core shall not be less than 5.2 meters (17 ft).

(3) The true value of primary coolant flow rate (W) shall not be less than 63 LPS (1,000 gpm).

(4) The true value of the bulk pool temperature (T_{bulk}) shall not exceed 60° C (140°F).

Basis: It has been demonstrated that boiling in the coolant outlet channel does not necessarily jeopardize the reactor core. Fuel centerline melting or departure from nucleate boiling (DNB) are, therefore, selected as the criteria defining compromise of fuel or cladding integrity.

An analysis has been performed using the Bernath correlation*, to determine the critical heat flux. In this analysis, W, H, and T_{bulk} were fixed at their limiting values. Manufacturer tolerances were further assumed to be at their worst case values and coincident with the flux hot spot. The number of fuel assemblies (and hence channel velocity) were varied, and the corresponding critical heat fluxes calculated. A limiting value of 1.09×10^6 BTU/hr-ft^2 was identified for a 16 element core. This corresponds to a core thermal power of 3.3 MW. A correlational error of 30% was assumed, and a DNB ratio of 2.0 was applied. Maximum fuel and cladding temperatures were calculated to be 3,794° F and 285° F, respectively.

2.1.2 Safety Limits in Natural Convection Mode

Applicability: This specification applies to core power in natural convection mode.

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*Bernath, L., "A theory of local boiling burnout," Heat Transfer Symposium, AICHE 1955.

Objective: The objective is to protect fuel and cladding integrity during operation in the natural convection mode.

Specification: The true value of reactor thermal power shall not exceed 500 kW in the natural convection mode.

Basis: Performance tests conducted in 1966 demonstrated that Pulstar fuel may be operated in the natural convection mode at powers in excess of 1 MW without the temperatures of fuel or cladding reaching their respective melting points. These tests are documented in WYNYNRC technical note J-435, dated December 1966.

2.2 Limiting Safety System Settings

2.2.1 Limiting Safety System Settings in Forced Convection Mode

Applicability: These specifications shall provide worst upper bounds on set points of safety systems, when the reactor is operating in forced convection mode.

Objective: The objective is to provide appropriate safety system actions so that safety limits shall not be exceeded.

Specifications:

(1) Control blade reverse shall occur at 2.2 MW.

(2) Control blade scram shall occur at 2.4 MW.

(3) Reactor scram shall occur if the depth of water above the core falls below 6.13 m (20 ft).

(4) Reactor scram shall occur if coolant flow falls below 68 lps (1,080 gpm).

(5) Reactor scram shall occur if the bulk pool temperature reaches 52°C (126°F).

Basis: The limiting safety system settings (LS^3s) specified have been chosen to ensure that automatic protective actions will correct the most severe abnormal situation before a safety limit (SL) is exceeded. The margin that is provided between the SLs and LS^3s, allows for the most adverse combination of instrument uncertainties associated with the protective systems. These uncertainties include a power level variance of 5%, a pool water level variance of 3 in., a flow variance of 5%, and a bulk pool temperature sensor variance of 5%.

The actual values associated with the worst case instrument variances are

Power	2.2 MW x 1.05 = 2.31 MW
Flow	1,080 gpm ÷ 1.05 = 1,030 gpm
Temperature	126°F x 1.05 = 132°F
Height	20 ft 3 in. = 19 ft 9 in.

The margin of safety between the true values at maximum variance and the safety limits would be

Power	3.3 MW - 2.3 MW = 1 MW (30% of SL)
Flow	1,030 gpm - 1,000 gpm = 30 gpm (3% of SL)
Temperature	140°F - 132°F = 8°F (6% of SL)
Height	19 ft 9 in. - 17 ft = 2 ft 9 in. (14% of SL)

Because the analysis to determine the safety limits assumed that W, T_{bulk}, and H were fixed at these extremes, the margins on W, T_{bulk}, and H are adequate. The 30% margin on reactor power, coupled with the conservative DNB ratio of 2.0 will ensure that unforeseen factors do not compromise fuel integrity.

2.2.2 Limiting Safety System Settings in Natural Convection Mode

Applicability: This specification shall establish the maximum set point for overpower scram in natural convection mode.

Objective: The objective is to limit reactor power in natural convection mode.

Specification: Reactor scram shall occur if reactor power exceeds 2[illegible] kW, while in the natural convection mode.

Basis: Without accurate calorimetric power indication, a maximum 20% variance in power sensing instrumentation is assumed. Thus, at an indicated power of 250 kW, the true power could be 300 kW or 60% of the safety limit. A 40% margin will reasonably ensure that protective actions will curtail any abnormal situation before the safety limit is exceeded.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability: These specifications apply to the reactivity of the reactor core, the control rods, and experiments.

Objectives: The objectives are to ensure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded.

Specifications:

(1) The shutdown margin relative to the cold, xenon-free critical condition shall be not less than the total worth of the most reactive safety-control blade + 0.5% $\Delta k/k$.

(2) The sum of the absolute values of all experiments shall not exceed 3% $\Delta k/k$.

(3) The worth of individual experiments shall be limited as follows:

Experiment	Maximum Worth
movable	± 0.3% $\Delta k/k$
unsecured	± 0.6% $\Delta k/k$
secured	± 1.5% $\Delta k/k$

(4) The reactor shall be subcritical by at least 3% $\Delta k/k$ during any fuel manipulations or experiment manipulations involving reactivities > 0.3% $\Delta k/k$.

(5) Control-safety blades shall not be removed from the core unless all four adjacent fuel assemblies have first been removed from the grid plate.

(6) The maximum available excess reactivity shall not exceed 5.2% $\Delta k/k$ during reactor operation, including the worth of all experiments.

Bases: The shutdown margin required by specification 3.1(1) ensures that the reactor can be shut down from any operating condition and will remain subcritical after cooldown and xenon decay even if the control rod of the highest reactivity worth should be in the fully withdrawn position.

Specification 3.1.(2) limits the total worth of experiments to a value that can be compensated for by the control blades while maintaining the required shutdown margin.

Specification 3.1(3) limits the worth of individual experiments. The 0.3% $\Delta k/k$ for movable experiments was chosen because it covers the needs of isotope production, and experience has shown that a combination of temperature coefficient and operator action provide adequate control of the change. The value of 0.6% for unsecured experiments was chosen to be slightly less than the value of β_{eff}. The value of 1.5% for secured experiments was chosen because this value of a step increase in reactivity was demonstrated to lead to no loss of fuel-clad integrity during the pulse-test program.

Specification 3.1(4) ensures that the reactor will remain subcritical during any manipulation of fuel or experiments.

Specification 3.1(5) ensures that the reactor will remain subcritical even if any or all control blades are removed entirely from the reactor.

Specification 3.1(6). While this specification, in conjunction with specification 3.1(1) tends to overconstrain the excess reactivity, it helps ensure that the operable core is reasonably similar to the core analyzed in the SAR.

3.2 Reactor Safety System

Applicability: These specifications apply to the reactor safety system and other safety-related instrumentation.

Objective: The objective of these specifications is to specify the lowest acceptable level of performance or the minimum number of operable components for the reactor safety system and other safety-related instrumentation.

Specifications: The reactor shall not be operated unless

(1) The reactor safety systems and safety-related instrumentation are operating in accordance with Table 3.1, including the minimum number of channels and the indicated maximum or minimum set points.

(2) All control-safety blades are operable.

(3) The time from the initiation of a scram signal until each of the control-safety blades is fully inserted shall not exceed 650 msec.

Bases: Specification 3.2(1) ensures that the reactor operator will have sufficient information at his disposal to operate the reactor within the safety limits and to ensure safe shutdown.

The Log Count Rate interlocks ensure that the channel is operational and on scale, with the detector properly positioned during startup.

Specifications on the pool water level (1) prevent operation of the reactor if the LS^3 for pool level is exceeded, (2) scrams the reactor if a major leak in the primary coolant boundary occurs, and (3) warns the reactor operator of an unusually high pool level.

The power level reverse and scram ensure that the safety limit on reactor power is not exceeded, thus preventing DNB and/or fuel melting. The manual scrams allow the operator to shut down the reactor if an unsafe or otherwise abnormal condition arises. The Dry Chamber Door Open scram prevents operation of the reactor while the dry chamber is occupied, or the shielding door is otherwise not fully closed.

The Flow scram prevents overheating of the reactor core.

The Flapper Open scram prevents core flow bypass when operating in the forced convection mode at powers in excess of the natural convection LS^3.

Table 3.1 Required instrumentation

Instrument channel	Minimum number operating	Function	Set point	Modes in which required
Log count rate (a, b)	1	Indication/ inhibit	<2 cps; 9800 cps	Startup
Linear power (a, b)*	1	Indication	-	All
Log power (a, b)	1	Indication	-	All
Period (a, b)	1	Indication		All
Power safety (a, b)	2	Indication/scram	120%	All
Power safety (a, b)	1	Reverse	110%	All
Manual scram (a, b)	5	Scram	-	All
Dry chamber door open (a)	1	Scram	Door < full closed	All
Flow (a, b)	1	Indication/scram	68 lps	Forced convection
Flapper open (a, b)	1	Scram	> 250 kW	Forced convection
Water level low (b)	1	Scram	6.13 m over fuel	All
Water level low (b)	1	Annunciation	6.43 m over fuel	All
Water level high (b)	1	Annunciation	6.74 m over fuel	All
Pool temperature (a)	1	Scram	52°C	Forced convection
Core outlet temperature (a)	1	Annunciation	52°C +ΔT	Forced convection
Recorders inoperative (b)	3	Inhibit	-	Startup
Conductivity (a)	0	Annunciation	200 K ohms	None
EPF valve open (a)**	0	Annunciation	Valve open	None
Suction valve closed (a)	1	Disables primary pump	Valve < full open	Forced convection
Servo deviation (a)	1	Annunciation/ transfer to manual	±10%	Servo control
Blade position - analog (a)	1 of 2	Indication	-	All
Blade position - digital (b)	1 of 2	Indication	-	All
Nitrogen-16 (a)	2 of 3	Indication	-	Forced convection
Primary temperature (a)	2 of 3	Indication	-	All
Core Δ T (a)	2 of 3	Indication	-	Forced convection

(a) - Test and/or calibration required four times/year
(b) - Operability check required prior to operation
* - Linear power channel and any recorder may be inoperable for short periods while operating
** - Emergency pool filling valve

The Pool Temperature scram ensures that the reactor will shut down if the LS^3 for bulk pool temperature is exceeded, as this parameter enters into the thermal-hydraulics safety analyses.

The Recorders Inoperative interlock ensures that the recorders indicating reactor power or startup count rate are operative during startup and operation of the reactor.

The neutron detectors provide assurance that measurements of reactor power or source multiplication are adequately covered at both low and high ranges.

The blade positions indicators ensure that information is available to the operator of the control element positions and, hence, the amount of available negative reactivity.

The N-16, core ΔT and primary temperature systems provide a measure of reactor power that is not sensitive to neutron flux distribution. The suction-valve-closed interlock prevents implosion of the N-16 delay tank by primary pump suction.

Specifications 3.2(2) and 3.2(3) ensure that primary control devices are operable and that upon exceeding any LS^3, automatic protective action will promptly avoid exceeding a safety limit. These specifications help to ensure that the safety margins between LS^3s and SLs are adequate.

3.3 Radiation Monitoring Systems

3.3.1 Fixed Area Monitors

Applicability: These specifications apply to the permanently mounted radiation monitors in the containment building that have readouts and alarms in the control room.

Objective: The objective of these specifications is to set a minimum level of performance for the facility area monitor system when the reactor is operating.

Specifications:

(1) The normal contingent of operating monitors when the reactor is operating shall be (a) neutron deck #1, 2, and 3; (b) reactor bridge; and (c) hot cell.

(2) The alarm set points for each monitor shall be clearly stated in a facility operating procedure.

(3) Monitors may be removed from service for repair, replacement, or calibration in accordance with the following restrictions:

   (a) No more than one of the three monitors on the neutron deck may be out of service at a given time.

   (b) The hot cell monitor may be out of service for extended periods provided the key to the cell is administratively controlled.

(c) The bridge monitor may be out of service for periods up to 4 hours, provided reactor power level is not increased during that outage.

(d) Any monitor may be temporarily replaced by a portable unit that provides equivalent functions.

Bases: Specification 3.3.1(1) provides that the neutron deck area monitors alert the operator and experimentors of unusually high radiation levels on the neutron deck; the bridge monitor alerts operators and others of high radiation over the pool, which could conceivably result from low pool water level or experiment transfers; the hot cell monitor warns personnel who may potentially enter the cell that high radiation exists within.

Specification 3.3.1(2) ensures that each monitor will have a clearly defined alarm point that cannot be altered without formal review. Specification 3.3.1(3) provides for normal repair and calibration of the area monitor system without shutting down the reactor.

3.3.2 Effluent Monitors and Primary Coolant Monitor

Applicability: These specifications apply to the permanently installed systems that monitor the airborne radioactivity leaving the facility and also monitor the activity of the primary coolant.

Objective: The objective of these specifications is to establish a minimum operability level for the effluent and primary coolant monitor system.

Specifications:

(1) The normal contingent of operating monitors when the reactor is operating shall be the building air continuous monitor, stack air continuous monitor, stack particulate continuous monitor, and primary water monitor.

(2) The alarm points for the two gaseous effluent monitors shall be clearly stated in a facility operating procedure. The alarm points for the primary water monitor and stack particulate monitor shall be posted on the radiation monitor panel in the control room.

(3) The outputs of the building air, stack air, and stack particulate monitors shall be recorded on a strip chart.

(4) Both air monitor systems shall provide fixed filters for evaluating particulate releases.

(5) One of the two gaseous effluent monitors may be out of service for up to 4 hours while the reactor is operating, provided that no unusual experiments are being conducted and no radioactive chemical processing is being done in the hoods. The primary water monitor may be inoperative for up to 8 hours while the reactor is operating, provided that both air effluent monitors are operating. The stack particulate monitor may be out of service for periods not exceeding one week, provided that the fixed filter is evaluated daily. The recorder may be out of service for periods of time not to exceed 48 hours as long as the effluent monitor values are logged

at nominal 15-min intervals while the reactor is operating. The primary water monitor need not be operative in the natural convection mode.

Bases: Specification 3.3.2(1) provides assurance that adequate operating instrumentation exists to monitor airborne effluents and primary coolant activity when the reactor is operating. The building air and stack air monitors measure the concentrations of gaseous radioactive materials being released to the environs. The stack particulate monitor measures particulate airborne radioactive materials in the stack exhaust. These monitors enable the operator to ensure that applicable release limits are not being exceeded and maintain released materials to a level that is as low as reasonably achievable. The primary water monitor provides indication of unusual levels of radioactivity in the primary coolant that might arise as a result of fuel cladding or experiment failure.

Specification 3.3.2(2) provides for the unambiguous establishment of alarm settings for the effluent and coolant monitors.

Specification 3.3.2(3) ensures that a permanent record of effluent releases shall exist.

Specification 3.3.2(4) ensures that a means shall be available to evaluate particulate releases from the facility.

Specification 3.3.2(5) provides for limited inoperability of the monitor equipment in order to permit maintenance and calibration of the systems without shutting the reactor down.

3.4 Engineered Safety Features

3.4.1 Reactor Containment Building

Applicability: These specifications apply to the facility containment vessel and its associated airlocks and ventilation system.

Objective: The objective of these specifications is to control the release of airborne radioactivity from the facility under both normal and off normal conditions.

Specifications: The reactor will not be operated unless the following conditions are satisfied:

(1) The truck door shall be closed and sealed, one door in each of two airlocks shall be closed and sealed, and all other penetrations other than ventilation ducts shall be sealed.

(2) The stack fan in the university steam plant shall be operating and the fan in the building air duct shall be operating.

(3) The pressure in the containment vessel shall be negative with respect to the outside atmosphere.

(4) The dampers in the containment ventilation ducts must be operable and be capable of closing in 5 sec or less. They must close automatically in

response to coincident alarms from the building air monitor and the bridge monitor or in response to a manual signal.

(5) A charcoal filter shall be maintained in the emergency exhaust duct and the modulating damper in the exhaust duct and its controller shall be operable.

(6) The leak rate of the containment vessel shall not exceed 3.3 lps (7 cfm) standard air at a negative differential of 1.27 cm of water (½ in. water).

Bases: Specification 3.4.1(1) ensures that all openings in the containment vessel other than air ducts will be closed during operation, so that uncontrolled escape of confinement air is prevented.

Specification 3.4.1(2) ensures that fans capable of maintaining a negative pressure in the building under both normal and emergency conditions are operating.

Specification 3.4.1(3) ensures that any air leakage between the containment vessel and the environment will be inward.

Specification 3.4.1(4) ensures operability of the emergency dampers in the event of an accidental release of radioactivity within containment.

Specification 3.4.1(5) ensures that, under accident conditions, the containment will be vented in a controlled manner through an activated charcoal filter to the stack.

Specification 3.4.1(6) ensures that the leak rate of the containment building will not be greater than the 5% free air volume per day as used in the analysis of the design-basis accident.

3.5 Primary Coolant Conditions

Applicability: These specifications apply to the primary coolant purity, radioactivity, and flow distribution.

Objectives: The objectives of these specifications are (1) to ensure that coolant conditions are such that corrosion to the fuel and pool components is minimized, (2) to minimize the concentration of dissolved materials subject to neutron activation, and (3) to ensure that coolant does not bypass the fuel.

Specifications:

(1) The primary coolant pH shall be maintained between 5.0 and 7.5.

(2) The resistivity of the primary coolant when the reactor is operating shall be no less than 200,000 ohms per centimeter, averaged over a period of 2 weeks.

(3) During forced convection operation, grid plate holes not occupied by fuel must be blocked by plugs, reflector elements, or experiments.

Bases: A small rate of corrosion continuously occurs in a water-metal system. To limit this rate and, thereby, extend the longevity and integrity of the fuel

cladding, a water cleanup system is required. Experience with water quality control at many research reactor facilities has shown that maintenance within the limits of specifications (1) and (2) provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations.

Specification 3.5(3) ensures that all primary coolant flow is through the fuel assemblies as was assumed in the analysis performed to establish the safety limits. Diversion of primary coolant flow would decrease channel velocity and, thus, lower the critical heat flux.

3.6 Airborne Effluents

Applicability: These specifications apply to levels of radioactivity discharged to the environment through the building air duct and the power plant stack.

Objective: The objective is to ensure that persons in the facility or any location outside the facility will not be exposed to concentrations of airborne radioactivity in excess of the maximum permissible concentrations (MPC) stated in 10 CFR 20, Appendix B, Tables I or II.

Specifications:

(1) For the power plant stack, the release rates of the quantity of radioactive material shall be limited so that

Yearly average:
$$\sum_{i=1}^{N} Qi/MPC_i \leq 8.1 \times 10^3 \frac{Ci \cdot Sec^{-1}}{Ci \cdot m^{-3}}$$

Instantaneously:
$$\sum_{i=1}^{N} Qi/MPC_i \leq 2.4 \times 10^4 \frac{Ci \cdot Sec^{-1}}{Ci \cdot m^{-3}}$$

where Qi = release rate (Ci/s) of radioisotope (i).
MPC_i = unrestricted area MPC (Ci/m^3) of radioisotope (i) (for air) (10 CFR 20, Appendix B, Table II).
N = number of radioisotopes being released.

(2) For the building air duct, effluent concentrations shall be limited so that:

Yearly average:
$$\sum_{i=1}^{N} (C_i \div MPC_{R_i}) \leq 1$$

Instantaneously:
$$\sum_{i=1}^{N} (C_i \div MPC_{R_i}) \leq 2$$

where C_i = concentration (μCi/cc) of the ith isotope

MPC_{R_i} = restricted area MPC (μCi/cc) of radioisotpe (i) (10 CFR 20, Appendix B, Table I)

N = number of radioistopes present

Basis: The specified limits are based on an analysis in Appendix B of the Safety Analysis Update, transmitted by letter dated 4/3/81, which demonstrates that below these limits maximum potential exposures to workers and to the public to airborne radioactive materials will be well below 10 CFR 20 Appendix B limits. A conservative factor of 3.0 was used in the computations for yearly maximum potential annual exposures to the public. Based on a review of meteorological data from 1968-1978, population exposures are estimated to be less than 3% of the 10 CFR 20 guidelines even if airborne releases were at the maximum value continuously.

3.7 Liquid Effluents

Applicability: These specifications shall apply to liquid radioactive effluents released to the unrestricted sanitary sewer system.

Objectives: The objectives of these specifications are (1) to prevent unmonitored releases to the sanitary sewer and (2) to ensure that radioactive releases in excess of limits specified in 10 CFR 20, Appendix B, are not made intentionally or inadvertently.

Specifications:

(1) All potentially radioactive liquid effluent first shall be retained by a hold tank system.

(2) Before releasing effluent from a hold tank to the unrestricted sanitary sewer, the solution in the tank shall be mixed and a representative sample shall be analyzed for radioactive content.

(3) The effluent shall not be released to the sanitary sewer unless the sample analysis has demonstrated that

$$\sum_{i=1}^{N} [(C_i \div MPC_i) + (C_u \div MPC_u)] \leq 1$$

where C_i = concentration (μCi/cc) of radioisotope (i)

MPC_i = unrestricted area MPC (μCi/cc) of radioisotope (i) (10 CFR 20, Appendix B)

N = number of radioisotopes in the solution to be released

C_u = concentration (μCi/cc) of all unidentified βγ emitters

MPC_u = MPC (μCi/cc of water) for unidentified βγ emitters (10 CFR 20, Appendix B).

For the purposes of establishing the above concentrations, the most recently established nominal annual release rate of water for the Winspear Avenue sewer trunk shall be explicitly included when calculating C_i and C_u.

Bases: Specification 3.7(1) will ensure that no liquid radioactive materials will be released to the unrestricted environment in an uncontrolled, unmonitored manner. Specifications 3.7(2) and 3.7(3) will ensure that liquid effluents will be monitored accurately and that releases to the unrestricted environment are not made unless compliance with the appropriate limits of 10 CFR 20, Appendix B is ensured.

3.8 Limitations on Experiments

Applicability: These specifications apply to all experiments installed in the BMRC reactor.

Objectives: The objectives of these specifications are (1) to prevent damage to the reactor fuel and (2) to prevent excessive releases of radioactivity in the event of an experiment failure.

Specifications:

(1) Experiments shall be designed and operated to prevent local boiling of the moderator coolant or melting of any structural component of the experiment.

(2) All samples or experiments shall be doubly encapsulated if release of the contained material could cause (a) excessive corrosive attack of the reactor or experimental facility, (b) excessive contamination or production of airborne radioactivity, or (c) a violent chemical reaction. Each such capsule shall be capable of containing the reaction and resulting pressure.

(3) The container for experiments shall be designed to withstand the maximum pressure that could be produced by the sum of gamma heating, fission heating, external heating, and radiolytic decomposition. Double encapsulation shall be provided where appropriate, with the outer encapsulation capable of containing the stored energy of the inner container. Pressure relief devices shall be provided where appropriate.

(4) During normal functioning, experiments shall not interfere with the normal operation of the control safety blades, nor shall they adversely affect any other safety system or required instrument channel.

(5) Experiment variables that could affect the safety of the reactor facility shall be monitored and limits established. Alarms and reactor trips shall be provided as appropriate.

(6) No single experiment shall occupy more than two adjacent fuel spaces on the grid plate.

(7) Explosive materials shall not be irradiated in the reactor tank nor inside a beam tube. Irradiations of explosive material in the thermal column, dry chamber, or outside a beam tube must be analyzed in detail in an experiment plan and approved by the Nuclear Safety Committee. Any explosives that are irradiated shall be separated from the reactor core or safety systems by an adequate blast shield, or shall be confined to a container adequate to withstand detonation. The aggregate mass of explosive material that may be irradiated at any one time shall not exceed 175 g equivalent

TNT, and the maximum stored inside the reactor containment building shall not exceed 2 lbs equivalent TNT.

(8) A fission plate containing up to 1 kg of highly enriched U-235 may be used at the outside face of the thermal column only.

(9) Except for the fission plate, no special nuclear material or source material other than trace quantities shall be irradiated unless a detailed experiment plan is proposed and approved by the Nuclear Safety Committee.

Bases: Specification 3.8(1) will prevent reactivity oscillations resulting from voiding caused by failure of an experiment, and will ensure that reactor thermal-hydraulic characteristics remain within the envelope assumed in the safety analysis report and update.

Specifications 3.8(2) and 3.8(3) are necessary to protect personnel and the reactor core from corrosive or mechanical attack, and thus to help ensure integrity of fuel and safety-related components.

Specification 3.8(4) will ensure that experiments do not invalidate the assumptions made in establishing LS^3 or in the analysis of normal operations or accidents as presented in the SAR.

Specification 3.8(5) is necessary to ensure that possible decrease in a margin of safety caused by an experiment would lead to automatic corrective or preventive protective action.

Specification 3.8(6) ensures that no incore experiment with a cross section area greater than 16 in.^{2} (103.2 cm^2) will be considered. This limitation precludes the BMRC reactor being operated as a testing facility, as defined in 10 CFR 50.2(r). A double hole has a cross section of 17.26 in.^{2} (111.4 cm^2); thus, any experiment designed for such a hole must be slightly smaller.

Specification 3.8(7) applies to Class C explosive devices (as defined by the U.S. Department of Transportation) or other potentially explosive materials. Damage to a reactor core or to reactor safety systems is to be prevented by these limitations on the use of explosives in experiments.

Specification 3.8(8) ensures that the facility-owned fission plate will be used only in a very low neutron flux, where the power produced is estimated not to be greater than 1 watt (thermal). At such a power level, temperature increases in the fission plate itself will occur slowly so that melting and loss of confinement of radioactivity is very unlikely.

Specification 3.8(9) is a general limitation that any new experiment involving special nuclear material be reviewed and approved by the Nuclear Safety Committee.

3.9 Fuel Limitations

Applicability: These specifications pertain to the fuel used in the BMRC reactor. They do not pertain to experiments that contain fissile material.

Objective: The objective of these specifications is to ensure that only PULSTAR-type fuel is used in the BMRC reactor.

Specifications:

(1) Only PULSTAR-type fuel using nominal 6% enriched sintered UO_2 pellets in Zircaloy-2 cladding (as specified in Section 5.5 of these Technical Specifications) shall be used in the BMRC reactor.

(2) The reactor will not be operated with fuel in the core that is known to be leaking fission products from inside the cladding except for such operation that may be necessary to find the leaking fuel pin.

(3) Fuel will be limited to a maximum burnup of 20,000 MW days per metric ton. The first fuel assembly to reach 15,000 MW days per ton will be removed from the core. After 6 months' decay, this fuel assembly will be disassembled and a representative sampling of pins will be inspected and measured. The results will be reported to the NRC. If significant degradation of the cladding is found, all fuel in the core with >15,000 MW days per ton will be removed and no additional fuel will be allowed to exceed 15,000 MW days per ton without NRC approval.

Bases: Specification 3.9(1) ensures that only fuel that is of a type that has been thoroughly tested and characterized will be used. The safety analysis and the specifications in this text were based on PULSTAR fuel, having the same procurement and quality assurance specifications as used in the procurement of the new fuel loading in 1977-1978.

Specification 3.9(2) ensures that only nonleaking fuel will be used for normal operation. Operation with a leaking fuel pin would lead to levels of gaseous effluents and pool contamination that have not been fully analyzed.

Specification 3.9(3) ensures that the burnup will be limited to the operating experience for zircaloy clad power reactor fuel (reference license amendment #10 dated May 29, 1975).

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactivity Limit Measurements

Applicability: These specifications apply to the surveillance requirements for reactivity limits.

Objective. The objective of these specifications is to ensure that the specifications of Section 3.1 are satisfied.

Specifications:

(1) The shutdown margin as specified in Section 3.1(1) shall be verified each time the core fuel configuration is changed and at any time that an experiment worth $>$ + 0.3% $\Delta k/k$ is loaded.

(2) The worth of individual samples or experiments representing a new type shall be measured when they are initially loaded into the reactor.

(3) At least one control-safety blade shall be removed from the core annually, not to exceed 15 months, for visual inspection. Inspection of all control element blades shall be on a rotating basis.

(4) The worth of the control blades shall be measured at least annually, not to exceed 15 months.

Bases: Specification 4.1(1) ensures that the minimum shutdown margin is maintained throughout fuel or experiment manipulations.

Specification 4.1(2) ensures that the worth of each experiment/sample or class of experiments/samples is known, so that reactivity conditions will be confined within the authorized limits.

Specification 4.1(3) ensures that the control-safety blades will be examined on a fixed schedule so that mechanical defects can be spotted before gross failure occurs.

Specification 4.1(4) ensures that valid blade worth data are available for measuring experiment worth, excess reactivity, shutdown margin, etc.

4.2 Reactor Safety System Tests

Applicability: These specifications apply to the surveillance of the reactor safety systems. They apply at any time that a critical mass is assembled on the grid plate.

Objective: The objective of these specifications is to ensure operability of the reactor safety system as described in Section 3.2.

Specifications:

(1) A channel test for each channel labeled (b) in Table 3.1 shall be performed before each startup following a period when the reactor was secured.

(2) A channel calibration for each channel labeled (a) in Table 3.1 shall be performed in each calendar quarter.

(3) Any safety-related instrument requiring maintenance or repair shall be operability checked and calibrated before being returned to service.

(4) Ion chamber channels shall be calibrated as frequently as required to maintain true power indication as determined by either heat balance measurements or the N-16 channel. The linear and log channel may deviate by +10 and -30% from nominal true power and the safety channels may vary -5 + 10%. These limits may be exceeded for short periods of time immediately following a startup, a sample loading, or other event that alter rod shadowing of the ion chambers.

(5) Control blade drop times shall be measured quarterly, not to exceed 4 months, or whenever maintenance or repairs are made that could affect their performance.

Bases: Specification 4.2(1) ensures that all safety-related systems are on and functioning each time the reactor is to be operated.

Specification 4.2(2) ures that all safety-related systems are tested on a periodic basis to en e proper operability, trip-point settings, and accurate relationship between measured variable and output signals.

Specification 4.2.(3) ensures that all instruments will perform as intended when installed in the reactor systems.

Specification 4.2.(4) ensures that the ion chamber power indicating channels will be calibrated frequently when operating near full power. The neutron flux incident to the ion chambers is significantly affected by control rod positions and hence xenon concentration, sample loadings, etc. Two temperature channels and an N-16-based power channel are available for power determination and are not affected by control blade positions. The ion chamber positions are easily adjustable to permit frequent cross calibration of the power channels. The limits on deviation from true power are somewhat arbitrary and are based on past practice.

Specification 4.2(5) ensures that control blade operation will be verified on a routine basis or as warranted by maintenance.

4.3 Radiation Monitoring Systems Tests

Applicability: These specifications pertain to the surveillance requirements for the permanently installed area monitors, effluent monitors, and primary water monitor. They apply at all times that radioactive material is being used, stored, or generated within the containment building.

Objective: The objective of these specifications is to ensure that all radiation monitoring channels are tested frequently for operability and satisfactory performance.

Specifications:

(1) The area monitors, effluent monitors, and primary water monitor shall be tested for operability monthly, at intervals not to exceed 6 weeks.

(2) A complete calibration of the area monitors, effluent monitors, and primary coolant monitor shall be performed quarterly, at intervals not to exceed 4 months.

(3) The hot cell monitor may be excluded from specifications (1) and (2) above, if conditions in the cell would lead to unnecessary excess exposure to radiation.

(4) Effluent monitor sensitivities shall be determined experimentally annually, not to exceed 15 months.

Bases: Specifications 4.3(1) and 4.3(2) ensure that the various radiation monitors are tested and calibrated on a routine basis.

Specification 4.3(3) allows the hot cell monitor be exempted from normal test and calibration schedules when activity levels in the cell prohibit entry. During such periods, the monitor will be maintained to the extent reasonably possible.

Specification 4.3(4) ensures that accurate sensitivity data are available for calculating effluent releases.

4.4 Engineered Safety Features Tests (Containment)

Applicability: These specifications apply to the surveillance requirements for the reactor containment vessel.

Objective: The objective of these specifications is to ensure that the containment and ventilation systems are tested on a routine basis.

Specifications:

(1) Before each startup that follows a period when the reactor was secured, it will be confirmed that the airlocks and truck door are sealed, that the containment vessel pressure is negative (with respect to the outside atmosphere), and that hydraulic pressure is available to the ventilation dampers.

(2) The following items will be tested quarterly, at intervals not to exceed 4 months.

(a) All dampers close in less than 5 sec in response to both manual trip and radiation signals.

(b) The control systems maintain negative pressure in the building under both normal and emergency conditions.

(c) The condition of the emergency duct charcoal filter will be visually inspected.

(3) The volumetric leak rate of the containment vessel will be measured annually, at intervals not to exceed 15 months.

Bases: Specification 4.4(1) ensures that basic containment and ventilation system components are operable when the reactor is in use.

Specification 4.4(2) ensures that ventilation components required for emergency conditions will function properly if called upon to do so.

Specification 4.4.(3) ensures that all penetration seals are intact and that there has been no general deterioration of containment integrity.

4.5 Primary Coolant Conditions Measurements

Applicability: These specifications apply to the surveillance of pool water conditions. They are in effect as long as fuel is present in the reactor tank.

Objective: The objective of these specifications is to ensure that water quality is measured often enough to prevent corrosion of system components and to detect higher than normal activity levels in the coolant.

Specifications:

(1) Pool water pH, conductivity, and gross beta activity will be measured weekly, at intervals not to exceed 10 days.

(2) A detailed radioisotopic analysis will be made of pool water semiannually at intervals not to exceed seven and one half months.

(3) A record will be maintained of all pool water additions.

(4) The pH of the pool water shall be between 5 and 7.5

(5) The resistivity of the pool water shall average no less than 0.2×10^6 ohm-cms.

Bases: Specification 4.5(1) ensures that poor pool water quality could not exist for long without being detected.

Specification 4.5(2) provides for detailed water analysis that can be useful in detecting small fuel leaks, experiment failure, etc.

Specification 4.5(3) provides a means for detecting primary system leaks.

Specifications 4.5(4) and 4.5(5) provide limits on the quality of the water to support the other parts of this specification.

4.6 Liquid Effluent Monitoring

Applicability: This specification applies to the surveillance of liquid wastes discharged to the sanitary sewer system.

Objective: The objective of this specification is to ensure that no waste is discharged to the sewer without first being assayed.

Specification: No potentially radioactive liquid wastes shall be discharged to the sanitary sewer without first being assayed. Records of the assay data and volume of liquids discharged must be maintained. All such discharges must be within applicable state and federal limits.

Basis: The basis for this specification is the federal and state requirements that all radioactive material released to the environment be monitored and that certain specified limits will not be exceeded.

5.0 DESIGN FEATURES

5.1 Site Description

The site of the BMRC reactor is the south center edge of the Main Street Campus of the State University of New York at Buffalo.

The campus is in the triangle bounded by Bailey Avenue, running almost due north and south, Winspear Avenue, running roughly east and west, and Main Street, running northeast and southwest. The reactor site is about 152 m (500 ft) due north of Winspear Avenue. The nearest buildings are Acheson Hall, Howe Building, the steam plant and Clark Gym. The reactor is about 30 m (100 ft) north of Acheson, 46 m (150 ft) west of the steam plant, and 91 m (300 ft) south of Clark Gym. A large Veterans Administration Hospital is situated about 610 m (2,000 ft) east of the reactor. The nearest residential area is on the south side of Winspear Avenue.

The reactor restricted access area consists of the containment building and the attached laboratory and office wing.

5.2 Containment Building

The containment building is a flat roofed, right cylinder, nominally 21.34 m (70 ft) in diameter and 15.84 m (52 ft) high. The vessel is constructed of normal density reinforced concrete. The walls are a nominal 61 cm (2 ft) thick and the roof is 10.2 cm (4 in.) supported by steel and concrete beams. The floor is 1.07 m (3½ ft) thick and the entire building rests on bedrock. The total free air volume of the building is 5,267 m^3 (186,000 ft^3).

The building contains two personnel airlocks and a single barrier truck door. All electrical and piping penetrations are sealed. Drain lines are provided with 61 cm (24 in.) dip legs to maintain a seal.

5.3 Ventilation

Under normal conditions, the containment building is ventilated by a single-pass-type system. Filtered, conditioned air is supplied to the vessel through two 75-cm (30-in.) diameter ducts. Inhabited areas and some intermediate level fume hoods are exhausted through the roof of the containment through a 91-cm (36-in.) duct. The blower in the containment exhaust duct has a variable damper on its suction side that is used to control the negative pressure of the building. Some reactor experimental facilities and several high level fume hoods are exhausted into a 46-cm (18-in.) duct that discharges at the top of the steam plant stack, which is 50.9 m (167 ft) above ground level. The stack duct is driven by two blowers, one in the facility basement and another in the steam plant. All air leaving the containment passes through absolute filters.

Under emergency conditions, all fans in the containment building are automatically turned off and dampers in the normal ventilation ducts are automatically closed. Only the fan in the steam plant remains operating. This fan draws air

from the containment through a 15¼-cm (6-in.) duct containing an absolute filter and an activated charcoal filter and exhausts it out the steam plant stack duct.

5.4 Primary Coolant System

The primary coolant system consists of the reactor tank, a N-16 decay tank, a pump, heat exchanger, and various support systems. The reactor tank is 8.8 m (29 ft) deep, aluminum lined and holds approximately 55,266 l (14,600 gal) of water. The remainder of the system contains approximately 21,803 l (5,760 gal) of water and is fabricated of either aluminum or stainless steel. Valves are located at points that permit isolation of the reactor tank or each of the other major components.

Two demineralizer systems are available; one provides pure water for makeup to the system, and the other is in a continuously circulating cleanup loop.

An emergency pool fill system is available for adding city water to the pool should this be desired. In the event of a gross leak from the reactor tank, a manual valve can be opened to supply water directly from the municipal water supply.

5.5 Fuel and Reflectors

Fuel assemblies are 8 cm (3.15 in.) x 6.96 cm (2.74 in.) in cross section and 96.5 cm (38 in.) long. Each assembly contains 25 pins in a 5 x 5 array. The pins are positioned by aluminum grids at each end. The end grids each contain 25 holes of 0.63-cm (¼-in.) diameter for coolant passage. The lower end of the assembly consists of an aluminum nosepiece that mates with the grid plate. The top of the assembly contains a bail for handling purposes. The center box portion is made of 0.152-cm (0.060-in.) zircaloy.

Each fuel pin is made up of a zircaloy tube 1.194 cm (0.47 in.) O.D. and 0.047 cm (0.0185 in.) minimum wall, containing a stack 60.96 cm (24 in.) long of sintered uranium dioxide pellets. Welded caps form the closure for the tube ends. Pellets are 1.067-cm (0.42-in.) diameter and have a minimum density of 10.2 g per cubic centimeter. Enrichment is to 6% in U-235.

Typical composition of a fuel assembly is as follows:

Uranium-235	= 0.768 Kg (1.69 lb)
Uranium	= 12.83 Kg (28.29 lb)
Uranium dioxide	= 14.56 Kg (32.1 lb)
Complete Fuel Assembly	= 20.37 Kg (45 lb)

The reactor may be reflected by normal water, graphite, lead, aluminum or voids.

5.6 Reactivity Control

Reactivity control is provided by six neutron absorbing blades. Each is composed of 80% silver, 15% indium, and 5% cadmium. The blades are 12.32 cm (4.85 in.) wide by 0.457 cm (0.180 in.) thick by 73.7 cm (29 in.) long. The blades are plated with 0.0076-cm (0.003-in.) nickel.

The blades are located in aluminum guides that are positioned between and supported by adjacent fuel assemblies. Five of the blades have extensions that extend above the surface of the pool where they mate with out-of-water magnets. The sixth blade (formerly the pulse blade) connects directly with its drive. The drives have a stroke of 66 cm (26 in.) and a nominal drive speed of 0.127 cm per sec (3 in. per minute).

The five blades supported by magnets provide scram capability for the functions listed in Table 3.1. The nonscramming blade is used for flux distribution control only. Its worth is not considered in computations of shutdown margin. It may be left at its fully withdrawn position and unused as was the case when it was intended for pulsing. The nonscramming blade may be converted to a scram-safety blade at any future time as long as its resulting performance characteristics are the same as the other five blades. The NRC will be notified before initiation of any such modification.

One of the control-safety blades may be used for automatic servo-control of reactor power. When in use, it maintains a constant power level as indicated by the linear power channel.

5.7 Fuel Storage

5.7.1 Cold Fuel Storage

Cold fuel pins and assemblies shall be stored within the containment building in secure vaults constructed of nonflammable materials. Fuel will be stored in rows of metal cylinders. The cylinders will have a minimum center to center spacing of 12.7 cm (5 in.) and the rows will have a minimum center line spacing of 40.6 cm (16 in.). Each cylinder will contain no more than one fuel assembly or 25 individual pins. Cold fuel is defined as any fuel that can be stored in the vault without creating a radiation field at the outer vault boundary in excess of 5 mrem per hour. The vaults are specifically exempted from the criticality alarm requirements of 10 CFR 70.24.

The fission plate may be stored in a fuel vault, a locked cask within the containment building or in the dry chamber or thermal column. A minimum distance of 40.6 cm (16 in.) will be maintained between the fission plate and any other fissile material.

5.7.2 Irradiated Fuel Storage

Normally all irradiated fuel will be stored in the reactor tank either on the grid plate or in storage racks. The pool storage racks will be linear with a minimum center to center spacing of 15.5 cm (6.1 in.).

In order to perform repairs in the lower tank, irradiated fuel may be stored out of the pool. Storage may be in the hot cell or in a specially fabricated shielded facility within the containment builidng. In either case, storage must be in accord with the following stipulations:

(1) The calculated k_{eff} if the storage array must be ≤ 0.85 for the flooded condition.

(2) The storage configuration must be approved by the Nuclear Safety Committee.

(3) Transfer of fuel to the facility must be done according to detailed written procedures approved by the Operating Committee.

(4) [illegible]lding must be adequate to reduce the radiation level to $\leq$ 100 mR per [illegible] the outer boundary of the storage facility.

(5) A [illegible]ity containing $\geq$ 15 fuel assemblies or an equivalent number of pins shall be equipped with a neutron sensing criticality alarm.

(6) The facility will be vented to the stack exhaust system.

6.0 ADMINISTRATION

6.1 Management

The Buffalo Materials Research Center (BMRC) is owned by the State University of New York at Buffalo. Operation of the reactor and its support facilities is managed by a private company, Buffalo Materials Research Inc. (BMR). The BMRC maintains a close working relationship with SUNY through the Office of the Vice President for Research. Independent overview and radiological safety monitoring is performed by University Radiation Protection Services (RPS) under the direction of the Radiation Safety Officer (RSO). The RSO reports to the Director of Environmental Health and Safety, who in turn reports to the Vice President for Finance and Management. The organization structure is presented in Figure 6.1.

The University will provide whatever resources are required to maintain the Center in a condition that poses no threat to the general public or to the environment. The University, as the licensee, bears overall responsibility for the operation of the facility.

6.1.1 Center Staff Requirements

(1) Figure 6.2 shows the minimum staffing requirements and their functional relationships.

(2) The BMRC Director is assigned direct responsibility for the safe operation of the Center, on behalf of the licensee.

(3) A SUNY/Buffalo Radiation Safety Officer shall be responsible for monitoring, planning, and promoting radiological safety at the center. He has the responsibility and authority to stop, secure, or otherwise control as necessary any operation or activity that poses an unacceptable radiological hazard.

(4) The BMRC Health Physics Department shall be responsible for the routine and day-to-day radiological safety activities.

(5) A licensed reactor operator (RO) or a licensed senior reactor operator (SRO) shall be present in the control room whenever the reactor is operating.

(6) A minimum of two persons must be present in the facility when the reactor is operating; the operator in the control room and a second person who can be reached from the control room by telephone or intercom.

(7) The following operations must be supervised by a senior reactor operator:

   (a) fuel manipulations in the core when there are $\geq$ 15 fuel assemblies on the grid plate.

   (b) when experiments are being manipulated in the core that have an estimated reactivity worth > 0.5% $\Delta k/k$.

Figure 6.1

Buffalo Materials Research/SUNY/B Organization

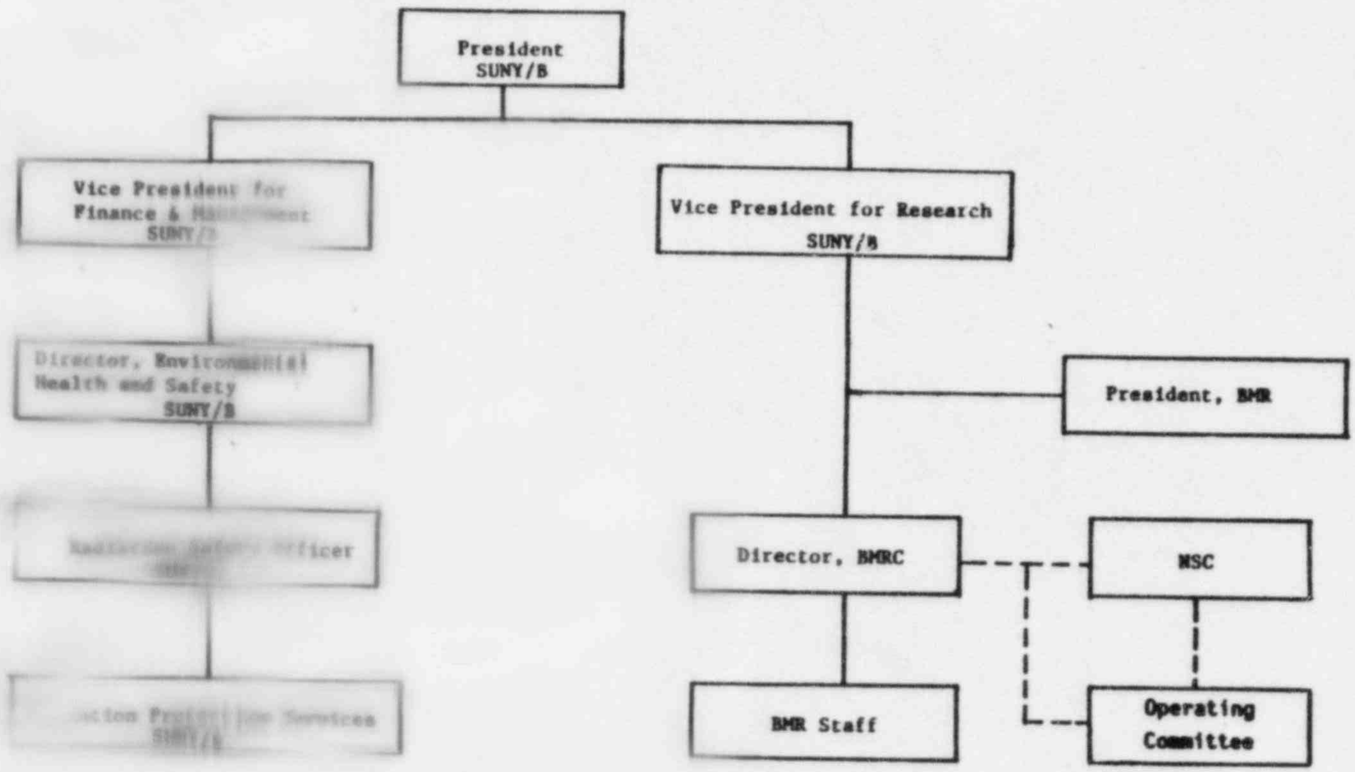
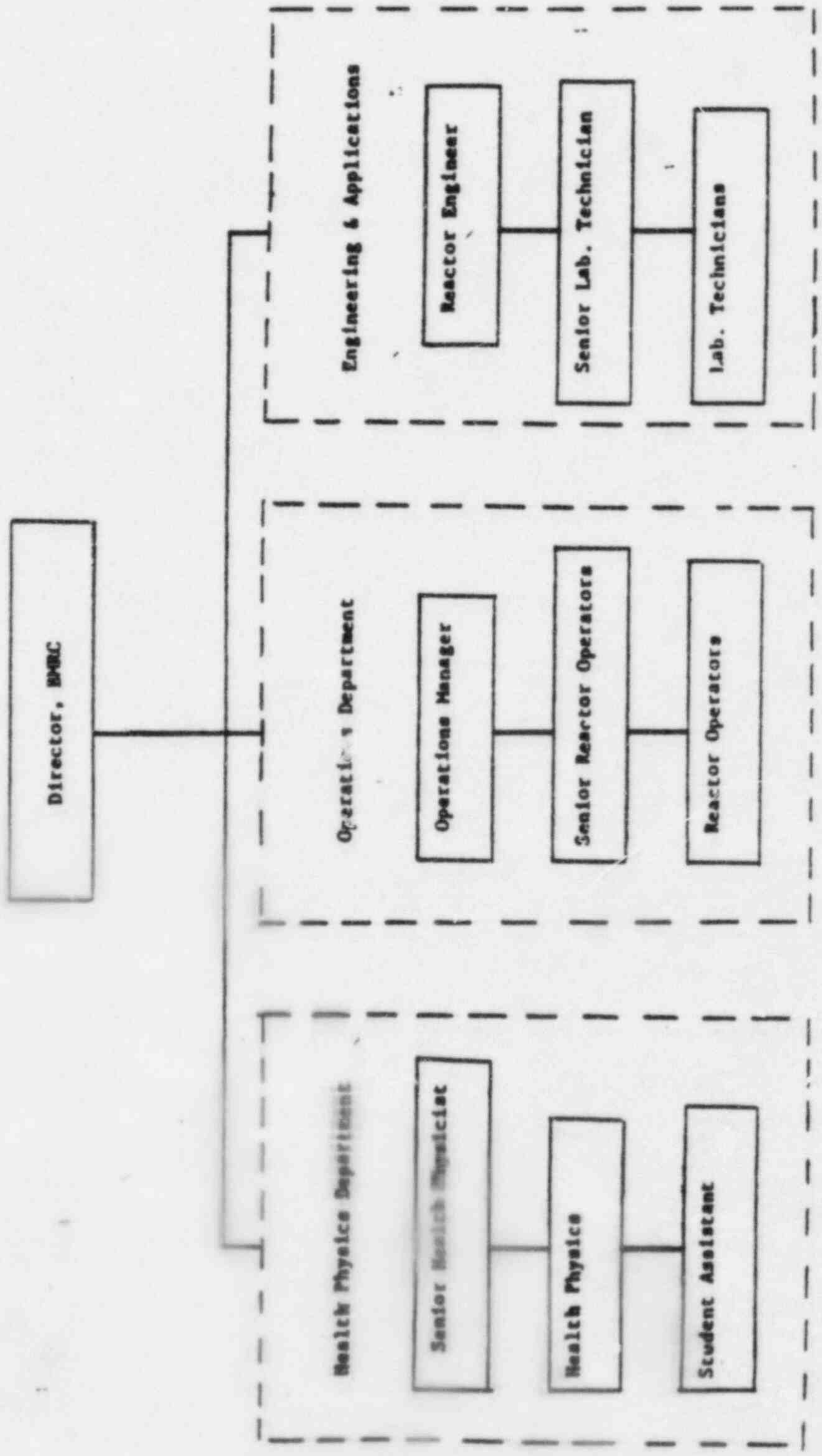


Figure 6.2

ORGANIZATIONAL CHART OF THE BMRC



(c) removal of one or more control blades if there are $\geq$ 10 fuel assemblies on the grid plate.

(d) removal of shielding plugs from a beam tube.

(e) resumption of operation following an unscheduled shutdown, unless the shutdown was a result of an interruption of electrical power to the center.

(8) A licensed SRO must be present or readily available on call at any time the reactor is operating.

(9) The reactor must be secured at times when no RO or SRO is present in the facility.

(10) All licensed operators at the facility shall participate in an NRC approved requalification program as a condition of their continued assignment of operator duties.

6.2 Review Functions

6.2.1 Nuclear Safety Committee

(1) A Nuclear Safety Committee (NSC) shall exist for the purpose of reviewing matters relating to the health and safety of the public in accordance with the constitution and by-laws of that committee.

(2) The NSC shall consist of a minimum of six persons with expertise in the physical sciences and preferably some nuclear experience. Permanent members of the committee are the Center Director, the Campus Radiation Safety Officer, the Radiation Protection Services Manager, and the Operations Department Manager.

(3) The NSC shall meet at least twice per year, and more often as required.

(4) A quorum of the NSC must have at least five members present and must have a majority of non-BMRC members present.

(5) The Committee Chairperson shall notify the members of the Agenda not later than 48 hours before a regular meeting. For emergency meetings, the Agenda shall be included with the meeting notice.

(6) Questions before the committee shall be decided by vote of the members present, the concurrence of a majority of those present being required for approval, except that no question requiring specialized knowledge shall be decided unless a member or consultant who is qualified by training and experience in that field is present.

(7) Minutes of all meetings will be retained in a file and distributed to all NSC members.

(8) The NSC shall review the following:

(a) experiments referred to it by the Operating Committee because of the degree of hazard involved or the unusual nature of the experiment

(b) reportable occurrences (see Section 6.4)

(c) proposed changes to the facility license, changes to technical specifications, and experiments or changes made pursuant to 10 CFR 50.59.

6.2.2 Operating Committee

(1) An Operating Committee shall exist as a subgroup of the NSC.

(2) The Operating Committee shall consist of the Center Director, the Radiation Protection Services Manager, and the Operations Department Manager. The committee shall seek advice and comment from other qualified individuals as appropriate.

(3) The committee shall meet as frequently as needed. Minutes will be maintained for all formal meetings.

(4) The Operating Committee shall perform the following:

(a) review experiments which present no new significant safety problems

(b) approve additions to or revisions of any Operating Procedures

(c) review abnormal occurrences

(d) perform facility inspections

6.3 Facility Audit

(1) A consultant will be retained by the university to perform an annual audit of reactor operations. The consultant shall be nominated by the Director of the BMRC and shall be an individual with expertise in the nuclear field. The consultant shall submit a report on his findings to the Director of the BMRC.

(2) The consultant shall audit the following:

(a) reactor operators and operational records for compliance with internal rules, procedures, and regulations, and with license provisions

(b) existing operating procedures for adequacy and accuracy

(c) plant equipment performance and its surveillance requirements

(d) records of releases of radioactive effluents to the environment

6.4 Actions To Be Taken in the Event of a Reportable Occurrence

A reportable occurrence is any of the following:

(1) operation in excess of a safety limit as set forth in Section 2.1

(2) discovery of a safety system setting less conservative than the limiting setting established in the Technical Specifications

(3) operation in violation of any condition for operation established in the Technical Specifications

(4) a safety system component malfunction or other component or system malfunction that could, or threatens to, render the safety system incapable of performing its intended safety functions

(5) release of fission products from a leaking fuel element

(6) an uncontrolled or unplanned release of radioactive material from the restricted area of the facility in excess of applicable limits

(7) an uncontrolled or unplanned release of radioactive material that results in concentrations of radioactive materials within the restricted area in excess of the limits specified in Section 3.6 of these Technical Specifications

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

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

In the event of a reportable occurrence, as defined in these Technical Specifications, the following actions will be taken:

(1) Immediate action will be taken to correct the situation and to mitigate the consequences of the occurrence.

(2) The Operating Committee will investigate the causes of the occurrence. The Operating Committee will report its findings to the NRC, the Nuclear Safety Committee, and to the Vice President for Research. The report shall include an analysis of the causes of the occurrence, the effectiveness of corrective actions taken, and recommendations of measures to prevent or reduce the probability or consequences of recurrence.

6.5 Operating Procedures

Written procedures will exist that define how and when various aspects of facility operation will be performed. These procedures will be reviewed and updated as frequently as needed, but the review will be no less frequent than once per year. All new or revised procedures will be reviewed and approved by the Operating Committee.

Procedures will address the following areas:

(1) normal reactor operation

(2) use, surveillance, and maintenance of auxiliary systems

(3) use of experimental facilities

(4) abnormal and emergency situations - health and safety

(5) reactor electrical and mechanical surveillance and maintenance

Temporary changes to the procedures that do not change their original intent may be made with the approval of an SRO. All such changes shall be documented.

6.6 Operating Records

In addition to the requirements of applicable regulations, the following records and logs shall be maintained in a manner convenient for review and retained by the licensee for at least 5 years:

(1) normal facility operation and maintenance
(2) reportable occurrences
(3) tests, checks, and measurements documenting compliance with surveillance requirements
(4) records of experiments performed
(5) operator requalification program records
(6) facility radiation and contamination surveys
(7) minutes of the Operating Committee meetings
(8) principal maintenance activities

The following records shall be retained by the licensee for the life of the facility:

(1) gaseous and liquid radioactive waste released to the environs
(2) radiation exposure records for all facility personnel
(3) fuel inventories and transfers
(4) updated, corrected, and as-built facility drawings
(5) minutes of the Nuclear Safety Committee meetings
(6) offsite environmental monitoring surveys

6.7 Reporting Requirements

6.7.1 Annual Operating Report

A report summarizing facility operations will be prepared for each calendar year. A copy of this report shall be submitted to the Director, Office of Nuclear Reactor Regulation, Attn: Document Control Desk, with a copy to the Regional Administrator, Region I, USNRC, by March 31 of each year. The report shall include the following:

(1) a brief narrative summary of (a) changes in facility design or performance that relate to reactor safety and (b) results of surveillance tests and inspections

(2) a tabulation showing the energy generated for each month and for the year in MW hours

(3) a list of the unplanned shutdowns including the reasons therefor and corrective action taken, if any

(4) discussion of the major maintenance operations performed during the period, including the effects, if any, on safe operation of the reactor, and the reason for any corrective maintenance required

(5) a brief description of (a) each change to the facility to the extent that it changes a description of the facility in the Safety Analysis Report and (b) reviews of changes, tests, and experiments made pursuant to 10 CFR 50.59

(6) a summary of the nature and amount of radioactive effluents released or discharged to the environment

(7) a description of any environmental surveys performed outside the facility

(8) a summary of radiation exposures received by facility personnel and visitors, including details of unusual exposures, and a brief summary of the results of radiation and contamination surveys performed within the facility

(9) any changes in facility organization

6.7.2 Reportable Occurrence Reports

Notification shall be made within 24 hours by telephone or telegraph to the Regional Administrator, Region I, USNRC, followed by a written report within 14 days to the Director, Office of Nuclear Reactor Regulation, with a copy to the Regional Administrator, Region I, in the event of a reportable occurrence, as defined in Section 6.4. The written report and, to the extent possible, the preliminary telephone or telegraph notification shall:

(1) describe, analyze, and evaluate safety implications

(2) outline the measures taken to ensure that the cause of the condition is determined

(3) indicate the corrective action taken to prevent repetition of the occurrence including changes to procedures

(4) evaluate the safety implications of the incident in light of the cumulative experience obtained from the report of previous failure and malfunction of similar systems and components

6.7.3 Unusual Event Report

A written report shall be forwarded within 30 days to the Director, Office of Nuclear Reactor Regulation, with a copy to the Regional Administrator, Region I in the event of:

(1) discovery of any substantial errors in the transient or accident analysis or in the methods used for such analysis as described in the Safety Analysis Report or in the basis for the Technical Specifications

(2) discovery of any substantial variance from performance specifications contained in the Technical Specifications or Safety Analysis Report

(3) discovery of any condition involving a possible single failure which, for a system designed against assumed failure, could result in a loss of the capability of the system to perform its safety function

6.7.4 Special Nuclear Materials Reports

Material status reports and nuclear material transfer reports for special nuclear materials shall be made in accordance with Parts 70.53 and 70.54 of 10 CFR.