APPENDIX A

FACILITY LICENSE NO. R-79

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE

UNIVERSITY OF MISSOURI-ROLLA REACTOR

DOCKET NO. 50-123

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1. INTRODUCTION

1.1 Scope

This document constitutes the Technical Specifications for Facility License No. R-79 and supersedes all prior Technical Specifications. Included are the "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.

This document was written to be in conformance with ANSI/ANS-15.1-1982[1] and NRC Regulatory Guide 1.16[2]. The content of the Technical Specifications includes: Definitions, Safety Limits, Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, and Administrative Controls.

1.2 Application

1.2.1 Purpose

These Technical Specifications have been written specifically for the University of Missouri-Rolla Reactor.

The Technical Specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission on administrative controls, equipment availability, and operational parameters.

Specific limitations and equipment requirements for safe reactor operation and for dealing with abnormal situations, typically derived from the Safety Analysis Report (SAR), are called specifications. These specifications represent a comprehensive envelope for safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed.

1.2.2 Format

The format of this document is in general accordance with ANSI/ANS-15.1-1982[1].

1.3 Definitions

The following definitions are listed in alphabetical order:

administrative controls - those organizational and procedural requirements which are established by the Commission and/or the facility management.

ALARA - as low as is reasonably achievable.

channel - the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

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

channel check - a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

channel test - the introduction of a signal into the channel for verification that it is operable.

Commission - the U.S. Nuclear Regulatory Commission (or NRC).

configuration - the specific arrangement of specific fuel elements and control rods in the grid plate to constitute a reactor core.

confinement - a closure on the overall facility which controls the movement of air into it and out through a controlled path.

control rod - a device fabricated from neutron absorbing material which is used to establish neutron flux changes.

core - the general arrangement of fuel elements and control rods.

critical - when the effective multiplication factor (k_{eff}) of the reactor is equal to unity.

direct supervision - in visual and audible contact.

excess reactivity - that amount of reactivity that would exist if all control rods were removed from the core.

experiment - any apparatus, device, or material installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility.

experimental facility - 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 - any solid or liquid that is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in <u>Dangerous Properties of Industrial Materials</u>(3) by N.I. Sax, or is given an Identification of Reactivity (Stability) index 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, <u>Identification System for Fire Hazards of</u>

Materials,[4] or enumerated in the Handbook for Laboratory Safety[5] published by the Chemical Rubber Co.

fueled experiment - any experiment that contains U-235 or U-233 or Pu-239, not including the normal reactor fuel elements.

grid plate - the structural member which supports the fuel elements.

licensee - the Board of Curators of the University of Missouri.

limiting conditions for operation (LCO) - the lowest functional capability or performance levels of equipment required for safe operation of the facility.[6]

limiting safety system settings (LSSS) - settings 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 so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.[6]

measured value - the value of a parameter as it appears on the output of a channel.

mode - when the reactor is positioned as close as possible to the thermal column it is in the T mode and when it is moved away from the thermal column and reflected by water it is in the W mode.

movable experiment - an entire experiment which is intended to be moved in or near the core or into and out of the reactor while the reactor is operating.

operable - a component or system which is capable of performing its intended functions in a normal manner.

operating - a component or system which is performing its intended function.

protective action - the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

reactivity limits - those limits imposed on reactor core excess reactivity based upon a reference core condition.

reactivity worth of an experiment - the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter an experiment's position or configuration.

reactor facility - that portion of the Reactor Building that constitutes the confinement but which does not include the front office area.

reactor operating - whenever the reactor is not secured or not shutdown.

reactor operator - an individual who is licensed to manipulate the controls of the reactor.

reactor safety systems - those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

reactor secured - whenever (1) all shim/safety rods are fully inserted, (2) the console key is in the OFF position and is removed from the lock, and (3) no in-core work is in progress involving fuel or experiments or maintenance of the core structure, control rods, or control rod drive mechanisms.

reactor shutdown - when the reactor is subcritical by at least

1% delta k/k in the reference core condition and the reactivity worth of all experiments is accounted for.

reference core condition - when the core is at ambient temperature and the reactivity worth of xenon is negligible (<0.21% delta k/k).

regulating rod - a low reactivity-worth control rod used primarily to maintain an intended power level. Its position may be varied either by manual control or by the automatic servo-controller.

reportable occurrence - any of the conditions described in section 6.5.2 of these specifications.

safety channel - a measuring or protective channel in the reactor safety system.

safety limits (SL) - limits on important process variables which are found to be necessary to reasonable protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.[6] (The principal physical barrier is the fuel cladding.)

scram - the elapsed time between reaching a limiting safety system set point and the time when a control rod is fully inserted.

secured experiment - 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.

senior operator - an individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

shall, should and may - the word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, which is neither a requirement nor a recommendation.

shim/safety rods - high reactivity-worth rods used primarily to provide coarse reactor control. They are connected electromagnetically to their drive mechanisms and have scram capabilities.

shutdown margin - the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition although the most reactive rod is in its most reactive position (fully withdrawn) and that the reactor will remain subcritical without further operator action.

startup source - a spontaneous source of neutrons which is used to provide a channel check of the startup (fission chamber) channel.

surveillance time intervals -

- two-year (interval not to exceed 30 months).
- annually (interval not to exceed 15 months).
- semiannually (interval not to exceed 7 1/2 months).
- quarterly (interval not to exceed 4 months).
- monthly (interval not to exceed 6 weeks).
- weekly (interval not to exceed 10 days).
- daily (must be done during the working day).

true value - the actual value of a parameter.

unscheduled shutdown - 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 which could adversely affect safe operation, not including shutdowns which occur during testing or check-out operations.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability: This specification applies to the melting temperature of the fuel cladding.

Objective: The objective is to ensure that the integrity of the fuel cladding is maintained in order to guard against an uncontrolled release of radioactivity.

Specification: The safety limit shall be on the temperature of fuel element cladding, which shall be less than 580°C (1076°F).

Bases: The melting temperature of the aluminum alloy used for cladding in the fuel element fabrication is 580°C (1076°F). Therefore, in order to maintain the fuel element integrity its cladding temperature must not exceed 580°C (1076°F). Although the temperature distribution within a fuel element is not measured during the reactor operation, it can be calculated from the known reactor power and its distribution within the reactor core (see section 3.4.5 of the Safety Analysis Report).

2.2 Limiting Safety System Settings

Applicability: This specification applies to the set points for the safety channels monitoring reactor thermal power, P.

Objective: To ensure that automatic protective action is initiated to prevent the maximum fuel cladding temperature from exceeding the safety limit.

Specifications: The limiting safety system setting shall be on reactor thermal power, P, which shall be no greater than 300 kWt, or 150% of full power.

Bases: The reactor cooling is provided by natural convection in the reactor pool. Therefore, the only parameter which can be used to limit the fuel cladding temperature is the reactor power. The reactor power is, therefore, specified as the limiting safety parameter. The analysis in section 3.4.5 of the Safety Analysis Report (SAR) shows that at the reactor power of 300 kWt, the maximum fuel plate centerline temperature is about 105^{o}C (221^{o}F). This temperature is much lower than the temperature at which fuel element damage could occur. Therefore, a large safety margin exists between the limiting safety system set point and the fuel safety limit.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods and experiments.

Objectives: To ensure that the reactor can be operated safely and to ensure that it can be shut down at all times.

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

(1) The fuel loading shall be such that the excess reactivity above the reference core condition will be no more than 1.5% delta k/k, except that the excess reactivity may be increased up to a maximum of 3.5% delta k/k for purposes of control rod calibration only. This increase in excess reactivity above 1.5% delta k/k will be permitted no more than twice a year and for no more than five consecutive working days each time. The reactor shall be operated only by a licensed Senior Operator when the excess reactivity is greater than 1.5%.

(2) The reactor shall be operated only when all lattice positions internal to the active fuel boundary are occupied by either a fuel element or control rod fuel element or by an experimental facility.

(3) The minimum shutdown margin under any condition of operation with the highest worth control rod fully withdrawn shall be no less that 1.0% delta k/k.

(4) The regulating rod shall be worth no more than 0.7% delta k/k in reactivity.

<u>Bases</u>:

(1) A sufficient excess reactivity is needed to provide for temperature effect override, xenon override, and operational and experimental flexibility. The limit of 1.5% delta k/k on excess reactivity is to assure that the operational characteristics of a reactor core are such as analyzed in the Safety Analysis Report. It has been shown in section 9.6 of the SAR that a stepwise reactivity insertion of 1.5% delta k/k does not adversely affect the health and safety of the public and the reactor staff. The limit of 3.5% delta k/k allows for the complete, direct calibration of the most worth shim/safety rod. This excess reactivity is not allowed for any other operational purposes. In the accident analysis performed in section 9.6 of the SAR it was concluded that no credible physical mechanism exists which could possibly lead to a sudden release of this amount of reactivity. Past experience has shown that it takes about five working days to perform control rod calibrations.

(2) This specification precludes the possibility of having an internal vacancy into which a fuel element could be inadvertently inserted.

(3) The shutdown margin is necessary so that the reactor can be shut down from any operating condition and remain shut down after cool down and xenon decay, even if one control rod should become stuck in the fully withdrawn position.

(4) Since the regulating rod is used for automatic control, it is prudent to limit its reactivity worth to less than the delayed neutron fraction, so that a prompt neutron criticality cannot inadvertently be caused by its total withdrawal.

3.2 Reactor Control and Safety Systems

Applicability: This specification applies to the instrumentation that must be operable for safe operation of the reactor.

Objective: To require that sufficient control information and automatic protective signals are available to the operator to ensure safe operation of the reactor.

Specification: The reactor shall not be operated unless the channels described in Table 3.1 are operable.

Table 3.1 Control Channels

Channel	Set Point	Function
Linear Power Demand	120%*	Rundown
Low Compensating Ion Chamber Voltage	80%*	Rundown
Log Power	120%*	Rundown
Reactor Period	15 s*	Rundown
Reg. Rod on Insert Limit in Auto	Not applicable.	Rundown
Radiation Area Monitors+	20 mrem/hr*	Rundown
Pool Water Temperature	135^{0}F*	Rod Withdrawal Prohibit
Startup Count Rate+	2 cps*	Rod Withdrawal Prohibit
Reactor Period+	30 s*	Rod Withdrawal Prohibit
Recorder Off	Not applicable.	Rod Withdrawal Prohibit

* Values listed are the limiting set points. For operational convenience the actual set points may be on more restrictive values.

+ These channels may be key bypassed at the reactor console by the Senior Operator on Duty as provided for in the Standard Operating Procedures.

Bases: The power channels provide assurance that measurement of the reactor power is adequately covered at both low and high power ranges.

The radiation area monitors provide information to operating personnel of a decrease in pool water level and of any impending or existing danger from radiation contamination or streaming, allowing ample time to take necessary precautions to initiate safety action.

The startup interlock, which requires a neutron count rate of at least 2 counts per second (cps) before the reactor is operated, ensures that sufficient neutrons are available for proper operation of the startup channel, and for a controlled approach to criticality.

The pool water temperature prohibit provides protection to keep the demineralizer resins below their suggested temperature limit, which is 140°F (60°C).

3.2.2 Reactor Safety Systems

Applicability: This specification applies to the reactor safety system channels.

Objective: To stipulate the minimum number of reactor safety system channels that must be operable to ensure that the limiting safety system settings are not exceeded during normal operation.

Specification: The reactor shall not be operated unless the safety system channels described in the Table 3.2 are operable.

Table 3.2 Safety System Channels

Channel	Set Point	Function
Manual Button	Not applicable.	Scram
Reactor Power	300 kWt*	Scram
Reactor Period	5 s*	Scram
Bridge Motion	Not applicable.	Scram
Log N & Period Not Operative	Not applicable.	Scram

* Values listed are the limiting set points. For operational convenience the actual set points may be on more restrictive values.

Bases: Power channels are provided to ensure that the power level is limited to protect against abnormally high fuel temperatures. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The period scram is provided to ensure that the power level does not increase on a period less than 5 seconds.

3.2.3 Shim/Safety Rod Drop Times

Applicability: This specification appplies to the time from the receipt of a safety signal to the time it takes for a shim/safety rod to drop from the fully withdrawn to the fully inserted position (free-drop time).

Objective: To ensure that the reactor can be shut down within a specified period of time.

Specification: The reactor will not be operated unless the free-drop time for each of the three shim/safety rods is less than 600 msec.

Bases: Shim/safety rod drop times as specified will ensure that the safety limit will not be exceeded in a worst-case delayed critical transient which has been analyzed in section 9.4 of the SAR. Establishing a limit on the rod drop time also provides means to detect any gradual degradation in the rod insertion performance.

3.3 Coolant System

Applicability: This specification applies to the height of water above the reactor core and to the quality of the coolant water.

Objective: To ensure that adequate cooling is provided at all times for the reactor core and to ensure that there is sufficient biological shielding available. The objective of the water quality requirement is to reduce corrosion of the fuel element cladding and to reduce neutron activation of dissolved materials.

Specification:

(1) The reactor shall not be operated unless there is at least 16 feet (4.88 m) of water above the core.

(2) The resistivity of the pool water shall be greater than 0.5 megaohm-cm as long as there are fuel elements in the pool.

Bases:

(1) Cooling of the reactor core is provided by the natural convection in the reactor pool. In order to maintain the convection flow path intact the reactor core must be fully submerged. Radiation levels at licensed power require a sufficient depth of water for shielding.

(2) 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 purification system is required. Experience with water quality control at this and many other reactor facilities has shown that maintenance within the specified limit provides acceptable control of the corrosion rate. (See section 5.2 of the SAR for further information.)

3.4 Confinement

Applicability: This specification applies to the capability of isolating the reactor facility from the unrestricted environment when necessary.

Objective: To prevent the exposure to the public resulting from airborne activity released into the reactor facility from exceeding the limits of 10 CFR 20.105 for unrestricted areas, and to be consistent with the ALARA concept.

Specification: The reactor shall not be operated unless the Reactor Building bay door, ventilation intake and exhaust duct louvers, and the personnel security door are operable.

Basis: The basis for the availability of these items is to ensure that the reactor facility can be isolated in the case of a release of airborne radioactivity from the reactor or associated experimental facilities.

3.5 Ventilation System

Applicability: This specification applies to all ventilation fans and the associated intakes and exhausts.

Objective: To provide for normal building ventilation and the reduction of airborne radioactivity within the reactor bay during reactor operation.

Specification: A ventilation fan with a capacity of at least 4,500 cubic feet per minute (cfm) (127.4 m^3/min) shall be turned on when the reactor is at full power.

Bases: Experience has shown that during normal operation this specification is sufficient to maintain radioactive gaseous effluents below 10 CFR 20 (Appendix B) limits. Section 7.6.1 of the Safety Analysis Report shows that releasing the air does not unduly expose the public.

3.6 Radiation Monitoring Systems and Radioactive Effluents

3.6.1 Radiation Monitoring Systems

Applicability: This specification applies to the radiation monitoring instrumentation.

Objective: The objective is to ensure that radiation exposure to the personnel in the reactor building stays below the limits specified in 10 CFR 20.101 for restricted areas and is consistent with the ALARA principle.

Specification: The reactor shall not be operated unless the Radiation Area Monitors (RAMs) located at the reactor bridge, at the demineralizer, and in the experiment room are operable. Table 3.3 specifies their location, their set points and functions.

Table 3.3 Radiation Area Monitors

Location	Set Point*	Function
Reactor Bridge	20 mrem/hr 30 mrem/hr	Rundown Building Evacuation
Demineralizer	20 mrem/hr	Rundown
Experiment Room	20 mrem/hr	Rundown

* Values listed are the limiting set points. For operational convenience the actual set points may be on more restrictive values.

Bases: The RAMs provide information to operating personnel about the radiation level above the reactor pool, at the demineralizer, and in the experiment room. It ensures that in the case of a failure of an experiment or a significant drop in the pool water level the appropriate action can be automatically initiated.

3.6.2 Radioactive Effluents

Applicability: This specification applies to the monitoring of radioactive effluents from the reactor facility. Airborne and liquid effluents are discussed separately in the following sub-sections.

3.6.2(1) Airborne Effluents

Objective: To ensure that exposure to the public resulting from the routine release of radioactive airborne effluents will not endanger the health and safety of the public.

Specification: The activity of Ar-41 released beyond the site boundary shall not exceed the limits of 10 CFR 20, Appendix B, Table II, Column 1 for unrestricted areas.

Basis: The basis for this specification is given in section 7.6 of the SAR.

3.6.2(2) Liquid Effluents

Objective: To ensure that exposure to the public resulting from the release of radioactive liquid effluents will be well below the limits of 10 CFR 20, Appendix B, Table I, Column 2.

Specification: The activity of liquids released beyond the site boundary shall not exceed 10 CFR 20.303 specified limits at the point of release.

Basis: The basis for this specification is given in section 7.6 of the SAR.

3.7 Experiments

Applicability: These specifications apply to the reactivity condition of experiments, to materials limitations, to failures or malfunctions of experiments, and to fueled experiments.

Objectives: To ensure the reactor can be shut down at all times, that the reactor fuel will not be damaged, that the limiting conditions for operation will not be exceeded, and that a malfunction of an experiment will not result in undue radioactivity release to the environment.

3.7.1 Reactivity Limits

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

(1) If any experiment worth more than 0.4% delta k/k is inserted in the reactor, a procedure approved by the Radiation Safety Committee shall be followed.

(2) Any experiment with a reactivity worth greater than 0.4% delta k/k shall be a secured experiment.

(3) Experiments worth more than 0.4% delta k/k shall be inserted or removed with the reactor shut down.

(4) The total reactivity worth of all experiments shall be no greater than 1.2% delta k/k.

(5) Experiments having moving parts shall not have an insertion rate greater that 0.05% delta k/k per second.

Bases:

(1) Thorough Radiation Safety Committee review, which requires that a detailed experimental procedure be provided and a reactor staff review be conducted, provides assurance that such experiments will take reactor and personnel safety, and the environment into proper account.

(2) This limit is provided in order to prevent a moveable

experiment from inserting a large reactivity insertion into the operating reactor. An analysis of this reactivity limit is given in section 9.4 of the SAR.

(3) In order not to accidentally insert too much reactivity when the reactor is operating, such experiments need to be assembled or disassembled only when the reactor is shut down.

(4) The total reactivity of 1.2% delta k/k places an acceptable upper limit on the worth of all experiments. This limit is lower than the reactivity for which an accident analysis was performed in section 9.6 of the Safety Analysis Report. It was shown in this analysis that the maximum fuel cladding temperature would not exceed the safety limit.

(5) This specification allows for certain reactor kinetics experiments to be performed, while maintaining constraint upon the rate of change of reactivity insertions. It is well within the envelope of the reactivity insertion rate which was analyzed in section 9.5 of the Safety Analysis Report. Results have shown that the health and safety of the public and the reactor staff would not be endangered in such an accident.

3.7.2 Materials

Specifications:

(1) All materials to be irradiated in the reactor shall be either corrosion resistant in reactor pool water or encapsulated within corrosion resistant containers.

(2) Explosive material shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Experiments reviewed by the Radiation Safety Committee in which the material is potentially

explosive, either while contained or if it leaked from the container, shall be designed to prevent damage to the reactor core or to the control rods or instrumentation, and to prevent any changes in reactivity. Known explosives in the amount greater than 25 milligrams shall not be irradiated in or near the reactor core. In addition the pressure shall be calculated or experimenatlly determined such that it will not cause the sample container to fail.

(3) Fueled experiments shall not be allowed in or near the reactor unless specifically approved by the Radiation Safety Committee. Fueled experiments in the amount which would generate a power greater than 100 W shall not be irradiated at the UMRR facility. Fueled experiments which generate more than 1 W power shall be irradiated in the reactor pool at least 4.88 m (16 ft) deep under the pool water surface. Fueled experiments which generate less than 1 W power may be irradiated in the beam port or the thermal column.

(4) Cooling shall be provided to prevent the surface temperature of an experiment being irradiated from exceeding the boiling point of the reactor pool water.

Bases:

(1) The requirement concerning either corrosion resistant materials or corrosion resistant encapsulation provides assurance that irradiation samples will not contaminate the pool water and thus cause fuel damage.

(2) Special case-by-case precautions would be taken before the unlikely irradiation of explosive materials would be allowed. The quantities would be restricted to very small masses and most likely such irradiations would be done at the far end of the beam tube or of the thermal column. In which case, the potential for core damage or reactivity

changes would be very small.

(3) Special case-by-case precautions would be taken before irradiation of fueled experiments. The Radiation Safety Committee must determine whether there are any unreviewed safety questions. Section 9.7 of the Safety Analysis Report addresses the impact of the failure of a fueled experiment.

(4) Samples or containers irradiated in the pool are in contact with a large heat sink. However, in order to assure that departure from nucleate boiling does not occur, adequate heat removal must be provided.

3.7.3 Failure and Malfunction

Specifications:

Experimental apparatus, material, or equipment to be inserted in the reactor shall not be positioned so as to cause continuous shadowing of the nuclear instrumentation, interference with the control rods, or other perturbations that may interfere with the safe operation of the reactor.

Bases:

An experiment which shadows any of the nuclear instrumentation could possibly cause it to give erroneous information and thus degrade its performance. Experiments which could adversely affect proper operation of the control rods must be avoided for obvious reasons.

4. SURVEILLANCE REQUIREMENTS

This section prescribes the frequency and scope of tests which are required to demonstrate performance of the systems and their limiting conditions for operation. Allowable surveillance time intervals shall not exceed the times shown in the definition section 1.3.

The maximum intervals on surveillance frequencies indicated are to provide operational flexibility. The established frequencies of surveillance shall be maintained over the long term.

Surveillance tests (except those specifically required for safety when the reactor is shutdown) may be deferred during reactor shutdown; however, they must be completed prior to or at the time of the next reactor startup. Surveillance tests scheduled to occur during an operating cycle which cannot be performed with the reactor operating may be deferred to the end of the cycle.

4.1 <u>Reactor Core Parameters</u>

4.1.1 Excess Reactivity

<u>Applicability</u>: This specification applies to the surveillance requirements for determining the excess reactivity of the reactor core.

<u>Objectives</u>: To make certain that the excess reactivity limits of specification 3.1 are not exceeded.

<u>Specification</u>:

(1) Following any change in core configuration involving reactivity changes greater than 0.2% delta k/k, including any control rod grid position changes, the excess reactivity of the core shall be determined for both the W and T modes.

(2) Before a new core configuration is taken critical it shall be visually confirmed by a licensed operator on duty that there are no unoccupied internal lattice positions.

Bases:

(1) An experimental determination of the excess reactivity at reference core conditions is necessary in order to preclude operating the reactor without adequate shutdown capability.

(2) Visual confirmation of the reactor core is the most reliable way to assure that all internal positions are occupied and that no space exists for rapid insertion of a fuel element.

4.2 Reactor Control and Safety Systems

4.2.1 Shim/Safety Rods and Regulating Rod

Applicability: This specification applies to the surveillance requirements for the shim/safety rods and the regulating rod.

Objectives: To ensure that the control rods are capable of performing their function and to establish that no significant physical degradation in the rods has occurred.

Specifications:

(1) Shim/safety rod drop times shall be measured semiannually. Shim/safety rod drop times shall also be measured if the control assembly is moved to a new position in the core or if the magnet assembly has been removed and reinstalled, and after rod inspections.

(2) The shim/safety rod reactivity worths shall be measured whenever one or more of the rods are installed in a new core configuration.

(3) The regulating rod worth shall be measured whenever the rod is installed in a new core configuration.

(4) The shutdown margin shall be determined after the excess reactivity of the core and the total worth of each control rod have been experimentally determined for a new core configuration.

(5) The shim/safety rods shall be visually inspected annually for pitting and cracking and whenever rod drop times exceed the limiting conditions for operation (section 3.2.3 of these specifications).

Bases: The reactivity worth of the shim/safety rods is measured to assure that the required shutdown margin is available and to provide means for determining the reactivity worth of experiments inserted in the core. The visual inspection of the shim/safety rods and measurement of their drop times are made to determine whether they are capable of performing properly. The determination of the regulating rod worth is to make certain that its value does not exceed the delayed neutron fraction.

4.2.2 Safety Channels

Applicability: This specification applies to the surveillance requirements for the reactor safety system channels for the reactor.

Objective: To ensure that the reactor safety system channels are operable as required by Specification 3.2.2.

Specifications:

(1) A channel test of each of the reactor safety system channels shall be performed before each day's operation or before each operation expected to extend more than one day, except for the bridge motion monitor which shall be done

weekly.

(2) A channel calibration of the reactor power range safety channel and period channel shall be performed semiannually.

(3) The thermal power shall be experimentally verified semiannually.

Bases:

(1) The daily channel tests will ensure that the safety channels are operable.

(2) The semi-annual calibration will permit any long-term drift of the channels to be corrected.

(3) The semi-annual verification of the power measuring channels will correct for drift and ensure operation within the requirements of the license.

4.2.3 Maintenance

Applicability: This specification applies to the surveillance requirements following maintenance of control or safety systems.

Objective: To ensure that a system is operable before being used after maintenance has been performed.

Specification: Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable either before it is returned to service or during its initial operation.

Bases: The intent of the specification is to ensure that work on the system or component has been properly performed and that the system or component has been properly reinstalled or reconnected. Correct operation of some systems, such as power range monitors, cannot be verified unless the reactor is operating. Operation of these systems will be verified during their initial operation following maintenance or modification.

4.3 Coolant System

Applicability: This specification applies to the surveillance of coolant water quality.

Objective: To ensure that water quality does not deteriorate over extended periods of time even if the reactor is not operated.

Specification: The resistivity of the coolant water shall be measured at least once every 2 weeks.

Bases: Section 3.3 of these specifications ensures that the water quality is adequate during reactor operation, and this section ensures that the water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate. The demineralizer resins should be regenerated in order to improve the water quality. If that is not sufficient, then the resins should be replaced.

4.4 Confinement

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

Objective: To ensure that the closure equipment to the reactor bay is operable.

Specifications: At least once each month, a test shall be made to ensure that the following equipment is operable: bay door, ventilation inlet and exhaust duct louvers, and the personnel security door.

Bases: Monthly surveillance of this equipment will verify that the confinement of the reactor bay can be maintained, if confinement is needed.

4.5 Ventilation Systems

Applicability: This specification applies to the ventilation fans and associated closure devices.

Objective: The objective is to ensure that the ventilation fans and closure devices perform their function satisfactorily.

Specification: Ventilation fans and all closure devices shall be visually checked at least monthly for proper operation.

Bases: Monthly surveillance is to ensure proper exchange of air through the reactor facility to reduce the buildup of radioactive gases or particles within the reactor bay.

4.6 Radiation Monitoring Systems and Radioactive Effluents

4.6.1 Radiation Monitoring Systems

Applicability: This specification applies to the area radiation monitoring equipment required by section 3.6.1 of these specifications.

Objectives: To ensure that the radiation monitoring equipment is operating and to verify appropriate alarm settings.

Specification: The operation of the radiation monitoring equipment and the position of their associated alarm set points shall be verified daily during periods when the reactor is in operation. Calibration of the radiation monitoring equipment shall be performed annually.

Bases: Adequate radiation control requires operable monitors, and experience has shown that an annual calibration of the monitoring systems is adequate to ensure their proper functioning within the specified limits.

4.6.2 Radioactive Effluents

4.6.2(1) Airborne Effluents

Applicability: This specification applies to the surveillance of the air in the reactor building while the reactor is operated.

Objective: To verify the method used to calculate the airborne effluents.

Specifications: An experimental verification of calculated release values shall be performed annually.

Basis: This is to ensure that the airborne radioactive effluents will be properly accounted. The basis for this specification is given in section 7.6.1 of the SAR.

4.6.2(2) Liquid Effluents

Applicability: This specification applies to the surveillance of liquid radioactive effluents.

Specifications: Before any release of potentially radioactive liquid effluent, samples shall be drawn and analyzed.

Basis: This is to ensure that radioactive liquid effluents will be properly analyzed before being released to the unrestricted environment. The basis for this specification is given in section 6.2 of the SAR.

4.7 Experiments

Applicability: These specifications apply to the specific surveillance activities related to experiments.

Objectives: To make certain that all of the restrictions on experiments in Specification 3.7 are met.

Specification: Preparation of samples to be irradiated and of experiments to be performed at the reactor shall be done in accordance with proper prescribed procedures.

Bases: The preparation of samples and experiments by proper technique means laboratory safety is better assured.

5. DESIGN FEATURES

Only those design features of the facility describing materials of construction and geometric arrangements, which if altered or modified would significantly affect safety (and which are not included in sections 2, 3 or 4 of the Technical Specifications), are included in this section.

The Safety Analysis Report contains the details necessary for establishing criteria for the following Technical Specifications.

5.1 Site and Facility Description

5.1.1 The Nuclear Reactor Building is located on the east side of the University of Missouri-Rolla campus in Rolla, Missouri, near 14th Street and Pine Street.

5.1.2 The reactor is housed in a steel-framed, double-walled aluminum building designed to restrict leakage. All air and other gases will be exhausted through vents in the reactor bay ceiling 30 feet (or 11 meters) above grade. The Reactor Building free volume is approximately 17000 cubic meters.

5.2 Reactor Coolant System

5.2.1 The minimum temperature of the reactor pool should be no less than 15.5^{0}C (60^{0}F) when the reactor is operated.

5.3 Reactor Core and Fuel

5.3.1 Core Configurations

Various core configurations may be used to accommodate experiments.

5.3.2 Fuel Elements

(1) Plate fuel elements of the MTR type are used. The overall dimensions of each element are approximately 3 inches by 3 inches by 36 inches. The active length of fuel is approximately 24 inches and the fuel is clad in aluminum. The fuel plates are joined to two side plates in each element. The whole assembly is joined at the bottom to a cylindrical nose piece that fits into the core grid plate.

(2) There are control rod fuel elements which are similar to the elements described in (1) with the exception that the center plates have been removed and have been replaced with guide plates such that the control rod cannot come in contact with fuel plates.

5.3.3 Control Rods

(1) Poison sections of the three shim/safety rods are stainless steel and contain approximately 1.5% natural boron. The rods have nominal dimensions of 2-1/4 inches by 7/8 inches in cross section and are 29 inches long.

(2) The poison section of the regulating rod is a stainless steel oval-shaped tube, 25 inches long, having a wall thickness of 65 mils, and is mechanically coupled to the rod drive.

5.3.4 Control Rod Drive Mechanisms

(1) The shim/safety rod drives have a maximum vertical travel of 24 inches and a withdrawal rate of approximately 6-inches per minute.

(2) The regulating rod drive has a maximum vertical travel of 24 inches and a withdrawal rate of approximately 24-inches per minute.

(3) Lights are provided on the operator's console to indicate

upper limit, lower limit, shim range, and magnet contact for each shim/safety rod.

5.3.5 Start-up Source

A neutron source is available of such a strength as to satisfy the requirements that the count rate is greater than 2 counts per second during a cold reactor start-up.

5.4 Fissionable Material Storage

5.4.1 The fuel storage pit, which is located below the floor of the reactor pool and at the end opposite from the core, will be capable of storing the complete fuel inventory. The neutron multiplication factor of the fully loaded storage pit shall not exceed 0.9 under any conditions.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The Nuclear Reactor Facility is a part of the School of Mines and Metallurgy of the University of Missouri-Rolla. The organizational structure is shown in Figure 6.1.

6.1.2 Responsibility

The Dean of the School of Mines and Metallurgy is the individual responsible for the reactor facility's licenses (Level 1).

The Director of the Nuclear Reactor Facility is the contact person for the NRC and will have overall responsibility for management of the facility (Level 2).

The Reactor Manager shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Radiation Safety Committee (Level 3). During periods when the Reactor Manager is absent, his responsibilities are delegated to a Senior Operator (Level 4).

The Reactor Manager shall have a Bachelor's degree in engineering or science or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired.

The Reactor Maintenance Engineer shall have a Bachelor's degree in engineering or science or an equivalent combination of education and experience from which comparable knowledge and abilities can be acquired.

As soon as reasonably possible after being assigned to the

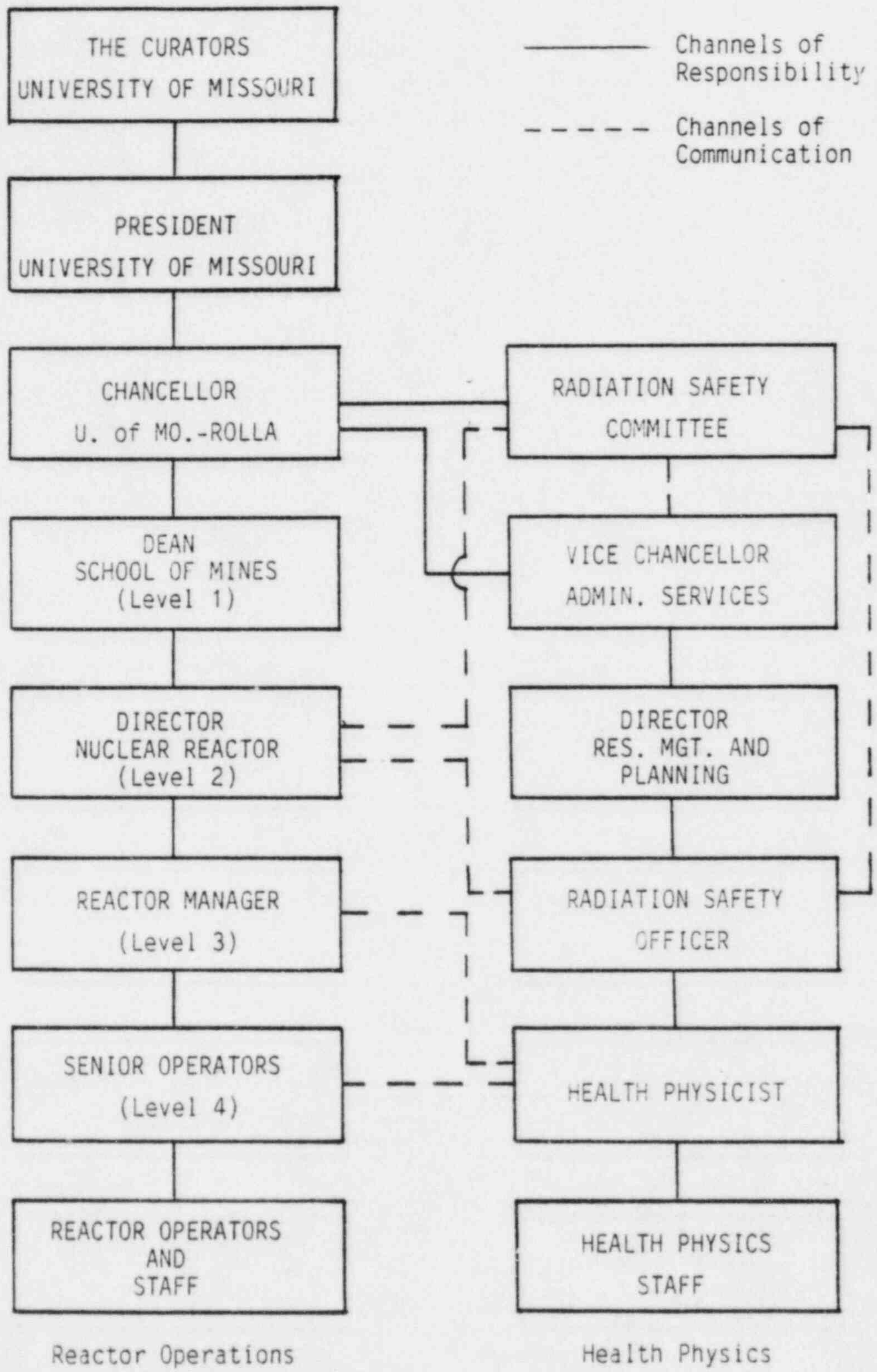

Figure 6.1 Organizational structure of the University of Missouri related to the UMR Nuclear Reactor Facility.

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position, the Reactor Manager and the Maintenance Engineer shall obtain and maintain NRC Senior Operator licenses.

A Health Physicist who is organizationally independent of the Reactor Facility operations group, as shown in Figure 6.1, shall be responsible for radiological safety at the facility.

6.1.3 Staffing

When the reactor is operating the following staffing conditions shall be met:

(1) At least two persons (one of whom is a licensed Senior Operator) shall be present in the Reactor Building.

(2) A licensed Reactor Operator or Senior Operator shall be present in the control room.

When fuel or control rods are being installed in or being unloaded from the core the following conditions shall be met:

(1) All rearrangements of the core or other nonroutine actions shall be under direct supervision of a licensed Senior Operator.

(2) The Health Physicist or his/her designated representative shall be present to monitor radiation levels.

When the reactor is being used for training purposes the following conditions shall be met:

(1) Students shall be under the direct supervision of a licensed Reactor Operator and shall not be permitted to operate the reactor when the excess reactivity is greater than 0.7% delta k/k.

(2) Trainees, who are preparing to become licensed at the facility or for a utility, shall be under the direct

supervision of a Senior Operator and shall not be permitted to operate the reactor when the excess reactivity is greater than 1.5% delta k/k.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall met or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4-6.[7]

6.2 Review and Audit

There shall be a committee that reviews and audits reactor operations to ensure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The Committee shall be referred to as the Radiation Safety Committee and shall report to the Chancellor of the campus and advise the Dean of the School of Mines and Metallurgy, and the Reactor Director on those areas of responsibility specified below.

6.2.1 Composition and Qualifications

The Committee shall be composed of at least five members, one of whom shall be the Radiation Safety Officer of the campus. No more than two members will be from the organization responsible for reactor operations. The membership of the Committee shall be such as to maintain a thorough knowledge in areas relating to reactor safety and research use of radioisotopes.

6.2.2 Charter and Rules

(1) A quorum of the Committee shall consist of at least one half of the voting members.

(2) The Committee shall meet at least quarterly. Minutes of all meetings shall be disseminated to Committee members and

to other responsible personnel as deisgnated by the Committee Chairman.

(3) The Committee shall have a written statement, or charter, defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee.

6.2.3 Review Function

As a minimum, the responsibilities of the Radiation Safety Committee include:

(1) review and approval in accordance with 10CFR50.59 of untried experiments and tests that are significantly different from those previously used or tested in the reactor, as determined by the Facility Director.

(2) review and approval in accordance with 10CFR50.59 of changes to the reactor core, reactor systems or design features that may affect the safety of the reactor.

(3) review and approval of all proposed amendments to the facility license and Technical Specifications.

(4) review reportable occurrences and the actions taken to identify and correct the cause of the occurrences.

(5) review significant operating abnormalities or deviations from normal performance of facility equipment that affect reactor safety.

This same Committee may have other responsibilities, for example review and approval of radioisotope use requests on the rest of the campus. The Committee may assign sub-committees to act on its behalf provided that said sub-committees report in writing all actions they take.

6.2.4 Audit Function

The Radiation Safety Committee will arrange for a knowledgeable and impartial individual (or individuals) to review reactor operation and audit the operational records for compliance with reactor procedures, Technical Specifications, and license provisions. An impartial individual is one who is not directly affected by the findings or recommendations of the audit and has no reason to be biased concerning the review. These audits shall be performed at least once each calendar year.

6.3 Operating Procedures

The reactor staff shall prepare and utilize written procedures for at least the items listed below. These procedures shall be adequate to ensure 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 foreseen potential malfunctions of systems or components, including responses to alarms, suspected 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) preventive and corrective maintenance operations that could have an effect on reactor safety.

(6) periodic surveillance (including testing and calibration)

of reactor instrumentation and safety systems.

(7) radiation control procedures which shall be maintained and made available to all operations personnel.

(8) implementation of emergency and physical security plans.

Substantive changes to the approved procedures shall be made only with the approval of the Radiation Safety Committee. Changes that do not change the original intent of the procedures may be made with the approval of the Facility Director.

6.4 Experiments Review and Approval

The reactor staff shall perform a thorough review of all proposed experiments in order to assure that they meet the requirements of sections 3.8 and 4.8 of these specifications.

Following the reactor staff review and approval, any proposed untried experiments will be forwarded to the Radiation Safety Committee for its review and approval or disapproval.

6.5 Required Actions

6.5.1 Action To Be Taken in the Event a Safety Limit is Exceeded

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

(2) The safety limit violation shall be promptly reported to the Director of the Reactor Facility.

(3) The safety limit violation shall be reported to the NRC.

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

(a) applicable circumstances leading to the violation including, when known, the cause and contributing factors.

(b) effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

(c) corrective action to be taken to prevent recurrence.

(5) The report shall be reviewed by the Radiation Safety Committee and any follow-up report shall be submitted to the NRC when authori ation is sought to resume operation of the reactor.

6.5.2 Action To Be Taken In the Event of a Reportable Occurrence

A reportable occurrence is any of the following conditions:

(1) any safety system setting less conservative than specified in section 2.2 of these specifications.

(2) operating in violation of a Limiting Condition Operation established in section 3 of these specifications, unless prompt remedial action is taken.

(3) safety system component malfunctions or other component or system malfunctions during reactor operation that could, or threaten to, render the safety system incapable of performing its intended safety function, unless immediate shutdown of the reactor is initiated.

(4) an uncontrolled or unanticipated increase in reactivity in excess of 0.5% delta k/k.

(5) 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.

(6) abnormal and significant degradation in reactor fuel, and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks) where applicable that could result in exceeding prescribed radiation exposure limits of personnel and/or the environment.

In the event of a reportable occurrence, the following action shall be taken:

(1) The reactor conditions shall be returned to normal, or the reactor shall be shut down, if it is necessary in order to correct the occurrence.

(2) The Director of the Reactor Facility shall be notified as soon as possible and corrective action shall be taken before resuming the operation involved.

(3) A written report of the occurrence shall be made which shall include an analysis of the cause of the occurrence, the corrective action taken, and recommendations for measures to preclude or reduce the probability of recurrence. This report shall be submitted to the Director and the Radiation Safety Committee for review and approval.

(4) A report shall be submitted to the Nuclear Regulatory Commission in accordance with section 6.6.2 of these specifications.

6.6 Reports

Reports should be made to the Nuclear Regulatory Commission as follows:

6.6.1 Operating Reports

An annual progress report will be made by April 30 of each year to the Director, Office of Nuclear Reactor Regulation, U.S. NRC, with a copy to the Regional Administrator, Region III, U.S. NRC, providing the following information:

(1) A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics, and operating procedures related to the reactor safety occurring during the reporting period.

(2) A tabulation will be prepared showing the energy generated by the reactor (in kilowatt hours) and the number of hours the reactor was in use.

(3) The results of the safety-related maintenance and inspections. The reasons for corrective maintenance of safety-related items will be included.

(4) A table of unscheduled shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary of changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of section 50.59 of 10 CFR 50.[6]

(6) A summary of the nature and amount of radioactive gaseous, liquid, and solid effluents released or discharged to the environs beyond the effective control of the licensee as measured or calculated at or prior to the point of such

release or discharge.

(7) A summary of radiation exposures received by facility personnel and visitors, including the dates and times of significant exposures (greater than 500 mrem for adults and 50 mrem for persons under 18 years of age) and a summary of the results of radiation and contamination surveys performed within the facility.

6.6.2 Special Reports

(1) A telephone or telegraph report shall be submitted as soon as possible, but no later than the next working day, to the Regional Administrator, Region III, U.S. NRC of the following:

(a) any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure.

(b) any exceeding of the safety limit as defined in section 2.1 of these specifications.

(c) any reportable occurrences as defined in section 6.5.2 of these specifications.

(2) A written report shall be submitted within 14 days to the Director, Office of Nuclear Reactor Regulation, U.S. NRC, with a copy to the Regional Administrator, Region III, U.S. NRC of the following:

(a) any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure.

(b) any exceeding of the safety limit as defined in section 2.1.

(c) any reportable occurrence as defined in section 6.5.2 of these specifications.

(3) A written report shall be submitted within 30 days to the Director, Office of Nuclear Reactor Regulation, U.S. NRC, with a copy to the Regional Administrator, Region III, U.S. NRC of the following:

(a) any substantial variance from performance specifications contained in these specifications or in the SAR.

(b) any significant change in the transient or accident analyses as described in the SAR.

(c) changes in personnel serving as Dean of the School of Mines and Metallurgy, Reactor Facility Director, or Reactor Manager.

(4) A report shall be submitted within nine months after initial criticality of the reactor or within 90 days of completion of the startup test programs, whichever is earlier, to the Director, Office of Nuclear Reactor Regulation, U.S. NRC upon receipt of a new facility license, an amendment to the license authorizing an increase in power level or the installation of a new core of a different fuel element type or design than previously used.

The report will include the measured values of the operating conditions or characteristics of the reactor under the new conditions, including the following:

(a) total control rod reactivity worth.

(b) reactivity worth of the single control rod of highest reactivity worth.

(c) minimum shutdown margin both at ambient and operating temperatures.

6.7 <u>Records</u>

Records, or logs, of the items listed below shall be kept in a manner convenient for review and shall be retained for as long as indicated.

6.7.1 Records To Be Retained for a Period of at Least Five Years

(1) normal plant operation,
(2) principal maintenance activities,
(3) experiments performed with the reactor,
(4) reportable occurrences,
(5) equipment and component surveillance activity,
(6) facility radiation and contamination surveys,
(7) transfer of radioactive material,
(8) changes to operating procedures,
(9) minutes of Radiation Safety Committee meetings.

6.7.2 Records To Be Retained for at Least One Requalification Cycle

Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained at all times the individual is employed. The normal training cycle is one year, and the NRC requalification cycle is two years.

6.7.3 Records To Be Retained for the Life of the Facility

(1) gaseous and liquid radioactive effluents released to the environment,
(2) fuel inventories and transfers,
(3) radiation exposures for all personnel,
(4) changes to reactor systems, components, or equipment that may affect reactor safety,

(5) Updated, corrected, and as-built drawings of the facility.

7. REFERENCES

(1) "American National Standard for the Development of Technical Specifications for Research Reactors," ANSI/ANS-15.1-1982, American Nuclear Society, LaGrange Park, Illinois (1982).

(2) "Reporting of Operating Information - Appendix A. Technical Specifications," Regulatory Guide 1.16 (Rev. 4), U.S. Nuclear Regulatory Commission, Washington, D.C. (Rev. 4)(August, 1975).

(3) Dangercus Properties of Industrial Materials, N.I. Sax, Van Nostrand- Reinhold Co. New York, NY (1975).

(4) Identification System for Fire Hazards of Materials, Publication 704, National Fire Protection Association, Batterymarch Park, Quincy, MA (1980).

(5) Handbook for Laboratory Safety, Chemical Rubber Company, Cleveland, OH. (1970).

(6) "Domestic Licensing of Production and Utilization Facilities," 10 CFR 50, U.S. Government Printing Office, Washington, D.C. (Current).

(7) "Selection and Training of Personnel for Research Reactors," ANSI/ANS-15.4-1977, American Nuclear Society, LaGrange Park, IL (1977).