APPENDIX A

LICENSE NO. R-110

TECHNICAL SPECIFICATIONS

FOR

IDAHO STATE UNIVERSITY AGN-201 REACTOR (SERIAL #103)

DOCKET NO. 50-284

DATE: November 15, 1978

AS MODIFIED TO INCLUDE ANSI N378-1974 AND

REGULATORY GUIDE 2.2 GUIDANCE

1.0 DEFINITIONS

The terms Safety Limit (SL), Limiting Safety System Setting (LSSS), and Limiting Conditions for Operation (LCO) are as defined in 50.36 of 10 CFR part 50.

1.1 Reactor Shutdown - The reactor shall be considered shutdown whenever

1. either: A. All safety and control rods are fully withdrawn from the core, or

   B. The core fuse melts resulting in separation of the core,

   and:

2. The reactor console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator.

1.2 Reactor Operation - Reactor operation is any condition wherein the reactor is not shutdown.

1.3 Measuring Channel - Ameasuring channel is the combination of sensor,lines, amplifiers, and output devices which are connected for the purpose of measuring or responding to the value of a process variable.

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

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

1.6 Reactor Component - A reactor component is any apparatus, device, or material that is a normal part of the reactor assembly.

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

1.8 Operating - Operating means a component or system is performing its intended function in its normal manner.

1.9 Channel Check - A channel check is a qualitative verification of acceptable performance by observaion of channel behavior. This verification may include comparison of the channel with other independent channels or methods measuring the same variable.

1.10 Channel Test - A channel test is the introduction of a signal into the channel to verify that it is operable.

1.11 Channel Calibration - A channel calibration is an adjustment of the cahnnel such that its output responds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment, actuation, alarm, or trip.

1.12 Experiment - An experiment is any of the following:

a. An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;
b. An evaluation or test of a reactor system operational, surveillance, or maintenance technique;

c. The material content of any of the foregoing, including structural components, encapsulation or confining boundaries, and contained fluids or solids.

1.13 Secured Experiment - Any experiment, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraint shall exert sufficient force on the experiment to overcome the expected effects of hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment or which might arise as a result of credible malfunctions.

1.14 Unsecured Experiment - Any experiment, or component of an experiment is deemed to be unsecured whenever it is not secured as defined in 1.13 above. Moving parts of experiments are deemed to be unsecured when they are in motion.

1.15 Movable Experiment - A movable experiment is one which may be inserted, removed or manipulated while the reactor is critical.

1.16 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 reasonable be anticipated to be moved one or more times during the life of the reactor.

1.17 Experimental Facilities - Experimental facilities are those portions of the reactor assembly that are used for the introduction of experiments into or adjacent to the reactor core region or allow beams of radiation to exist from the reactor shielding. Experimental facilities shall include the thermal column, glory hole, and access ports.

1.18 Potential Reactivity Worth - 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 experiment position or configuration.

The evaluation must consider possible trajectories of the experiment in motion relative to the reactor, its orientation along each trajectory, and circumstances which can cause internal changes such as creating or filling of void spaces or motion of mechanical components. For removable experiments, the potential reactivity worth is equal to or greater than the static reactivity worth.

1.19 Static Reactivity Worth - The static reactivity worth of an experiment is the 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.

1.20 Explosive Material - Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion 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.

2.0 SAFETY LIMITS AND LIMITED SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the maximum steady state power level and maximum core temperature during steady state or transient operation.

Objective

To assure that the integrity of the fuel material is maintained and all fission products are retained in the core matrix.

Specification

a. The reactor power level shall not exceed 100 watts.

b. The maximum core temperature shall not exceed 200°C during either steady state or transient operation.

Basis

The polyethylene core material does not m[illegible]ow 200°C and is expected to maintain its integrity and [illegible]ain essentially all of the fission products temperatures below 200°C. The Hazards Summary Report dated August 1956 submitted on Docket F-15 by Aerojet-General Nucleonics (AGN) calculated a steady state core average temperature rise of 0.5°C/watt. Therefore, a steady state power level of 100 watts would result in an average core temperature rise of 50°C. The corresponding maximum core temperature would be below 200°C thus assuring integrity of the core and retention of fission products.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the parts of the reactor safety system which will limit maximum power and core temperature.

Objective

To assure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specification

a. The safety channels shall initiate a reactor scram at the following limiting safety system settings:

Channel	Condition	LSSS
Nuclear Safety #2	High Power	$\leq$10 watts
Nuclear Safety #3	High Power	$\leq$10 watts

b. The core thermal fuse shall melt when heated to a temperature of 120°C or less resulting in core separation and a reactivity loss greater than 5%Δk.

Basis

Based on instrumentation response times and scram tests, the AGN Hazards Report concluded that reactor periods in excess of 30-50 milli-seconds would be adequately arrested by the scram system. Since the maximum available excess reactivity in the reactor is less than one dollar the reactor cannot become prompt critical and the corresponding shortest possible period is greater than 200 milli-seconds. The high power LSSS of 10 watts in conjunction with automatic safety systems and/or manual scram capabilities will assure that the safety limits will not be exceeded during steady state or as a result of the most severe credible transient.

In the event of failure of the reactor to scram, the self-limiting characteristics due to the high negative temperature coefficient, and the melting of the thermal fuse at a temperature below 120°C will assure safe shutdown without exceeding a core temperature of 200°C.

Channel	Condition	LSSS
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3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments.

Objective

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

Specifications

a. The available excess reactivity with all control and safety rods fully inserted and including the potential reactivity worth of all experiments shall not exceed 0.65% $\Delta k/k$ referenced to 20°C.

b. The shutdown margin with the most reactive safety or control rod fully inserted shall be at least 1% $\Delta k/k$.

c. The reactivity worth of the control and safety rods shall ensure sub-criticality on the withdrawal of the coarse control rod or any one safety rod.

Basis

The limitations on total core excess reactivity assure reactor periods of sufficient length so that the reactor protection system and/or operator action will be able to shut the reactor down without exceeding any safety limits. The shutdown margin and control and safety rod reactivity limitations assure that the reactor can be brought and maintained subcritical if the highest reactivity rod fails to scram and remains in its most reactive position.

3.2 Control and Safety Systems

Applicability

These specifications apply to the reactor control and safety systems.

Objective

To specify lowest acceptable level of performance, instrument set points,and the minimum number of operable components for the reactor control and safety systems.

TABLE 3.1

Safety Channel	Set Point	Function
Nuclear Safety #1		
Low power	5% Full scale	scram at source levels < 5% of full scale
Nuclear Safety #2		
High power	10 watt	scram at power < 10 watt
Low power	3.0×10^{-13} amps	scram at source levels < 3×10^{-13} amps
Reactor period	5 sec	scram at periods < 5 sec
Nuclear Safety #3 (Linear Power)		
High power	10 watt	scram at power > 10 watt
Low power	5% full scale	scram at source levels < 5% of full scale
Manual scram	-------	scram at operator option
Radiation Monitor	-------	alarm at or below level set to meet requirements of 10 CFR part 20

Specification

a. The total scram withdrawal time of the safety rods and coarse control rod shall be less than 200 milliseconds.

b. The safety rods and coarse control rod shall be interlocked such that:

   1. Reactor startup cannot commence unless both safety rods and coarse control rod are fully withdrawn from the core.

   2. Only one safety rod can be inserted at a time.

   3. The coarse control rod cannot be inserted unless both safety rods are fully inserted.

c. Nuclear safety channel instrumentation shall be operable in accordance with Table 3.1 with the exception that Safety Channels 1 or 3 may be bypassed whenever the reactor control or safety rods are not in their fully withdrawn position.

d. The shield water level interlock shall be set to prevent startup and scram the reactor if the shield water level falls 10 inches below the highest point on the reactor shield tank manhole opening.

e. The shield water temperature interlock shall be set to prevent reactor startup and scram the reactor if the shield water temperature falls below 15°C.

f. The seismic displacement interlock sensor shall be installed in such a manner to prevent reactor startup and scram the reactor during a seismic displacement.

g. A loss of electric power shall cause the reactor to scram.

Basis

The specifications on scram withdrawal time in conjunction with the safety system instrumentation and set points assure safe reactor shutdown during the most severe forseeable transients. Interlocks on control and safety rods assure an orderly approach to criticality and an adequate shutdown capability.

The neutron detector channels (nuclear safety channels 1 through 3) assure that reactor power levels are adequately monitored during reactor startup and operation. Requirements

on minimum neutron levels will prevent reactor startup unless channels are operable and responding, and will cause a scram in the event of instrumentation failure. The power level scrams initiate redundant automatic protective action at power level scrams low enough to assure safe shutdown without exceeding any safety limits. The period scram conservatively limits the rate of rise of reactor power to periods which are manually controllable and will automatically scram the reactor in the event of unexpected large reactivity additions.

The AGN-201's negative temperature coefficient of reactivity causes a reactivity increase with decreasing core temperature. The shield water temperature interlock will prevent reactor operation at temperatures below 15°C thereby limiting potential reactivity additions associated with temperature decreases.

Water in the shield tank is an important component of the reactor shield and operation without the water may produce excessive radiation levels. The shield tank water level interlock will prevent reactor operation without adequate water levels in the shield tank.

The reactor is designed to withstand 0.6g accelerations and 6 cm displacements. A seismic instrument causes a reactor scram whenever the instrument receives a horizontal acceleration that causes a horizontal displacement of 1/16 inch or greater. The seismic displacement interlock assures that the reactor will be scrammed and brought to a subcritical configuration during any seismic disturbance that may cause damage to the reactor or its components.

The manual scram allows the operator to manually shut down the reactor if an unsafe or otherwise abnormal condition occurs that does not otherwise scram the reactor. A loss of electrical power de-energizes the safety and coarse control rod holding magnets causing a reactor scram and thus assuring safe and immediate shutdown in case of a power outage.

A radiation monitor must always be available to operating personnel to provide an indication of any abnormally high radiation levels so that appropriate action can be taken to shut the reactor down and assess the hazards to personnel.

3.3 Limitations on Experiments

Applicability

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

Objective

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

Specification

a. Experiments containing materials corrosive to reactor components or which contain liquid or gaseous, fissionable materials shall be doubly encapsulated.

b. Explosive materials shall not be inserted into experimental facilities of the reactor.

c. The radioactive material content, including fission products of any experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components from the experiment will not result in doses in excess of 10% of the equivalent annual doses stated in 10 CFR part 20 for persons occupying (1) unrestricted areas continuously for two hours starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.

d. The radioactive material content, including fission products of any doubly encapsulated experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components of the experiment shall not result in exposures in excess of 0.5 Rem whole body or 1.5 Rem thyroid to persons occupying an unrestricted area continuously for a period of two hours starting at the time of release or exposure in excess of 5 Rem whole body or 30 Rem thyroid to persons occupying a restricted area during the length of time required to evacuate the restricted area.

Basis

These specifications are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from an experimental failure and to protect operating personnel and the public from excessive radiation doses in the event of an experimental failure.

3.4 Shielding

Applicability

This specification applies to reactor shielding required during reactor operation.

Objective

The objective is to protect facility personnel and the public from radiation exposure.

Specification

The following shielding requirements shall be fulfilled during reactor operation:

a. The reactor shield tank shall be filled with water to a height within 10 inches of the highest point on the manhole opening.

b. The thermal column shall be filled with water or graphite except during a critical experiment (core loading) or during measurement of reactivity worth of thermal column water or graphite.

Basis

The facility shielding in conjunction with designated restricted radiation areas is designed to limit radiation doses to facility personnel and to the public to a level below 10 CFR 20 limits under operating conditions, and to a level below criterion 19, Appendix A, 10 CFR 50 recommendations under accident conditions.

4.0 SURVEILLANCE REQUIREMENTS

Actions specified in this section are not required to be performed if during the specified surveillance period the reactor has not been brought critical or is maintained in a shutdown condition extending beyond the specified surveillance period. However, the surveillnace requirements must be fulfilled prior to subsequent startup of the reactor.

4.1 Reactivity Limits

Applicability

This specification applies to the surveillance requirements for reactivity limits.

Objective

To assure that reactivity limits for Specification 3.1 are not exceeded.

Specification

a. Safety and control rod reactivity worths shall be measured annually, but at intervals not to exceed 16 months.

b. Total excess reactivity and shutdown margin shall be determined annually, but intervals not to exceed 16 months.

c. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before or during the first startup subsequent to the experiment's insertion.

Basis

The control and safety rod reactivity worths measured annually to assure that no degradation or unexpected changes have occurred which could adversely affect reactor shutdown margin or total excess reactivity. The shutdown margin and total excess reactivity are determined to assure that the reactor can always be safely shutdown with one rod not functioning and that the maximum possible reactivity insertion will not result in reactor periods shorter than those that can be adequately terminated by either operator or automatic action. Based on experience with AGN reactors, significant changes in reactivity or rod worth are not expected within a 16-month period.

4.2 Control and Safety System

Applicability

This specification applies to the surveillance requirements of the reactor control and safety system.

Specifications

a. Safety and control rod scram times shall be measured annually, but at intervals not to exceed 16 months.

b. Safety and control rods and drive shall be inspected for deterioration at intervals not to exceed 2 years.

c. A channel test of the following safety channels shall be performed prior to the first reactor startup of the day or prior to each operation extending more than one day:

   Nuclear Safety #1, #2, and #3
   Manual scram

d. A channel test of the seismic displacement interlock shall be performed semiannually.

e. A channel check of the following safety channels shall be performed daily whenever the reactor is in operation:

   Nuclear Safety #1, #2, and #3

f. Prior to each day's operation or prior to each operation extending more than one day, safety rods #1, and #2 shall be inserted and scrammed to verify operability.

g. The period, count rate, and power level measuring channels shall be calibrated and set points verified annually, but at intervals not to exceed 16 months.

h. The shield water level interlock and shield water temperature interlock shall be calibrated by perturbing the sensing element to the appropriate set point . These calibrations shall be perform d annually, but at intervals not to exceed 16 months.

i. The radiation monitoring instrumentation shall be calibrated annually, but at intervals not to exceed 16 months.

Basis

The channel tests and checks required daily or before each startup will assure that the safety channels and scram funcitons are operable. Based on operating experience with reactors of this type, the annual scram measurements, channel calibrations, set point verifications, and inspections are of sufficient frequency to assure, with a high degree of confidence, that the safety system settings will be within acceptable drift tolerance for operation.

5.0 DESIGN FEATURES

5.1 Reactor

a. The reactor core, including control and safety rods, contains approximately 670 grams of U-235 in the form of 20% enriched UO_2 dispersed in approximately 11 kilograms of polyethylene. The lower section of the core is supported by an aluminum rod hanging from a fuse link. The fuse melts at temperatures below 120°C causing the lower core section to fall away from the upper section reducing reactivity by at least 5% $\Delta k/k$. Sufficient clearance between core and reflector is provided to insure free fall of the bottom half of the core during the most severe transient.

b. The core is surrounded by a 20cm thick high density (1.75 gm/cm^3) graphite reflector followed by a 10 cm thick lead gamma shield. The core and part of the graphite reflector are sealed in a fluid-tight aluminum core tank designed to contain any fission gases that might leak from the core.

c. The core, reflector, and lead shielding are enclosed in and supported by a fluid-tight steel reactor tank. An upper or "thermal column tank" may serve as a shield tank when filled with water or a thermal column when filled with graphite.

d. The 6-1/2 foot diameter, fluid-tight shield tank is filled with water constituting a 55 cm thick fast neutron shield. The fast neutron shield is formed by filling the tank with approximately 1000 gallons of water. The complete reactor shield shall limit doses to personnel in unrestricted areas to levels less than permitted by 10 CFR 20 under operating conditions.

e. Two safety rods and one control rod (identical in size) contain up to 20 grams of U-235 each in the same form as the core material. These rods are lifted into the core by electromagnets, driven by reversible DC motors through lead screw assemblies. Deenergizing the magnets causes a spring-driven, gravity-assisted scram. The fourth rod or fine control rod (approximately one-half the diameter of the other rods) is driven directly by a lead screw. This rod may contain fueled or unfueled polyethylene.

5.2 Fuel Storage

Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in locked rooms in the nuclear engineering department laboratories. The storage array shall be such that K_{eff} is no greater than 0.8 for all conditions of moderation and reflection.

5.3 Reactor Room

a. The reactor room houses the reactor assembly and accessories required for its operation and maintenance.

b. The reactor room is a separate room in the Lillibridge Engineering Laboratory, constructed with adequate shielding and other radiation protective features to limit doses in restricted and unrestricted areas to levels no greater than permitted by 10 CFR 20, under normal operating conditions, and to a level below criterion 19, Appendix A, 10 CFR 50 recommendations under accident conditions.

c. Access doors to the reactor room are self-locking.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

The administrative organization for control of the reactor facility and its operation shall be as set forth in Figure 1 attached hereto. The authorities and responsibilities set forth below are designed to comply with the intent and requirements for administrative controls of the reactor facility as set forth by the Nuclear Regulatory Commission.

6.1.1 University Officer

The University Officer is an administrative officer responsible for the University and in whose name the application for licensing is made.

6.1.2 Dean, School of Engineering

The Dean of the School of Engineering is the administrative officer responsible for the operation of the School of Engineering.

6.1.3 Reactor Administrator

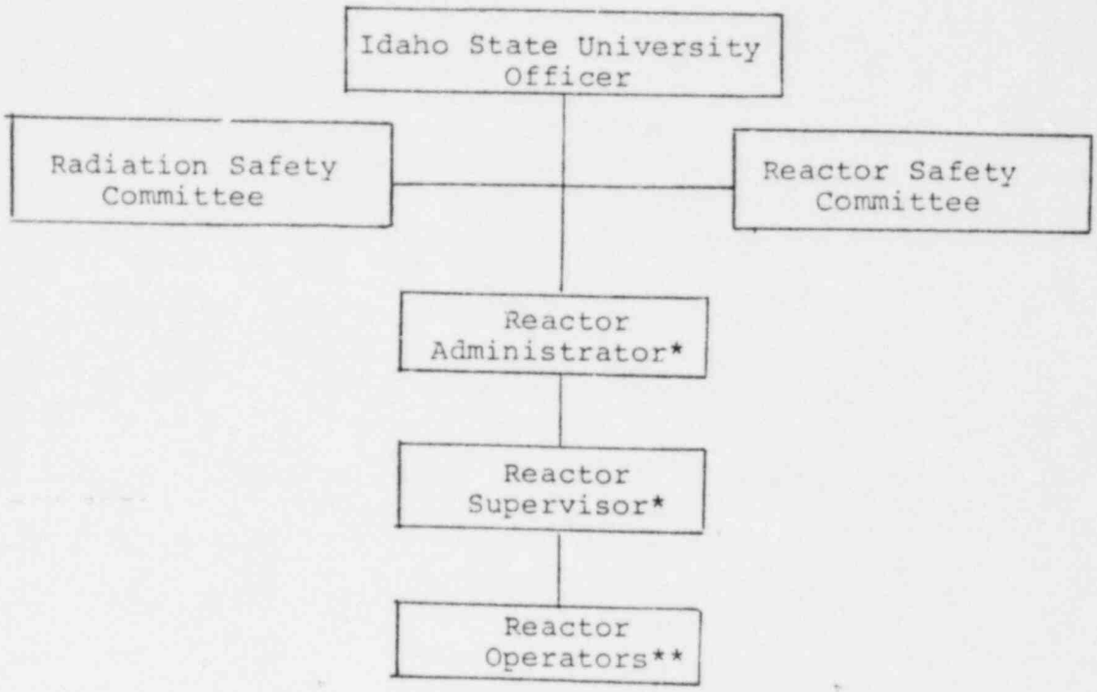
The Reactor Administrator is the administrative officer responsible for the operation of the AGN-201 Reactor Facility. In this capacity he shall have final authority and ultimate responsibility for the operation, maintenance, and safety of the reactor facility within the limitations set forth in the facility license. He shall be responsible for appointing personnel to all positions reporting to him as described in Section 6.1 of the Technical Specifications. He shall seek the advice and approval of the Radiation Safety Committee and/or the Reactor Safety Committee in all matters concerning unresolved safety questions, new experiments and new procedures, and facility modifications which might affect safety. He shall be an *ex officio* member of the Reactor Safety Committee.

6.1.4 Reactor Supervisor

The Reactor Supervisor shall be responsible for the preparation, promulgation, and enforcement of administrative controls including all rules, regulations, instructions, and operating procedures to ensure that the reactor facility is operated in a safe, competent, and authorized manner at all times. He shall direct the activities of operators and technicians in the daily operation and maintenance of the reactor; schedule reactor operations and maintenance; be responsible for

FIGURE 1

Administrative Organization of the
Idaho State University AGN-201 Reactor Facility
NRC License R-110



* Requires NRC Senior Operators License

** Requires NRC Operators License except where exempt per 10 CFR 55 paragraph 55.9

6.1.4 contd the preparation, authentication, and storage of all prescribed logs and operating records; authorize all experiments, procedures, and changes thereto which have received the approval of the Reactor Safety Committee and/or the Radiation Safety Committee and the Reactor Administrator; and be responsible for the preparation of experimental procedures involving use of the reactor.

6.1.5 Reactor Operators

Reactor Operators shall be responsible for the manipulation of the reactor controls, monitoring of instrumentation, operation of reactor related equipment, and maintenance of complete and current records during operation of the facility. Reactor Operators who are exempt from holding an NRC license per 10 CFR 55 paragraph 55.9 shall only operate the reactor under the direct and immediate supervision of a licensed Reactor Operator.

6.1.6 Reactor Safety Committee

The Reactor Safety Committee shall be responsible for, but not limited to, reviewing and approving safety standards associated with the use of the reactor facility; reviewing and approving all proposed experiments and procedures and changes thereto; reviewing and approving all modifications to the reactor facility which might affect its safe operation; determining whether proposed experiments, procedures, or modifications involve unreviewed safety questions, as defined in 10 CFR 50 paragraph 50.59 (c), and are in accordance with these Technical Specifications; conducting periodic audits of procedures, reactor operations and maintenance, equipment performance, and records; review all reportable occurrences and violations of these Technical Specifications, evaluating the causes of such events and the corrective action taken and recommending measures to prevent reoccurence; reporting all their findings and recommendations concerning the reactor facility to the Reactor Administrator.

6.1.7 Radiation Safety Committee

The Radiation Safety Committee shall advise the University administration and the Radiation Safety Officer on all matters concerning radiological safety at university facilities.

6.1.8 Radiation Safety Officer

The Radiation Safety Officer shall review and approve all procedures and experiments involving radiological safety.

6.1.8 contd He shall enforce all federal, state, and university rules, regulations, and procedures relating to radiological safety. He shall perform routine radiation surveys of the reactor facility and report his findings to the Reactor Administrator. He shall provide personnel dosimetry and keep records of personnel radiation exposure. He shall advise the Reactor Administrator on all matters concerning radiological safety at the reactor facility. The Radiation Safety Officer shall be an ex officio member of the Reactor Safety Committee.

6.1.9 Operating Staff

The minimum operating staff during any time in which the reactor is not shutdown shall consist of one licensed Reactor Operator and one other person capable of carrying out any prescribed written instruction and oral instructions of operators and to summon help in the event the licensed operator becomes incapacitated.

6.2 Staff Qualifications

The Reactor Administrator, the Reactor Supervisor, licensed Reactor Operators, and technicians performing reactor maintenance shall meet the minimum qualifications set forth in ANS 15.4, "Standards for Selection and Training of Personnel for Research Reactors." Reactor Safety Committee members shall have a minimum of five (5) years experience in their profession or a baccalaureate degree and two (2) years of professional experience. Generally, these committee members will be made up of University faculty, but outside experience may be sought in areas where additional experience is considered necessary by the Reactor Administrator.

6.3 Training

The Reactor Administrator shall be responsible for directing training as set forth in ANS 15.4, "Standards for Selection and Training of Personnel for Research Reactors." All licensed reactor operators shall participate in requalification training as set forth in 10 CFR 55.

6.4 Reactor Safety Committee

6.4.1 Meetings and Quorum

The Reactor Safety Committee shall meet as often as deemed necessary by the Reactor Safety Committee Chairman but shall meet at least once each calendar year. A quorum for the conduct of official business shall be the chairman, or his designated alternate, and two (2) other regular members. At no time shall the operating organization comprise a voting majority of the members at any Reactor Safety Committee meeting.

6.4.2 Reviews

The Reactor Safety Committee shall review:

a. Safety evaluations for changes to procedures, equipment or systems, and tests or experiments, conducted without Nuclear Regulatory Commission approval under the provision of 10 CFR 50 paragraph 50.59, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems, and tests or experiments, conducted without Nuclear Regulatory Commission approval under the provision of 10 CFR 50 paragraph 50.59, to verify that such actions did not constitute an unreviewed safety question.

c. Proposed tests or experiments which are significantly different from previous approved tests or experiments, or those that involve an unreviewed safety question as defined in Section 50.59, 10 CFR 50 paragraph 50.59.

d. Proposed changes in Technical Specificaitons or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instruciton having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety.

g. Reportable occurrences.

h. Audit reports.

6.4.3 Audits

Audits of facility activities shall be performed under the cognizance of the Reactor Safety Committee but in no case by the personnel responsible for the item audited. These audits shall examine the operating records and encompass but shall not be limited to the following:

a. The conformance of facility operation to the Technical Specifications and applicable license conditions, at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff, at least once per 24 months.

c. The results of all actions taken to correct deficiencies occuring in facility equipment, structures, systems or method of operation that affect nuclear safety, at least once per calendar year.

d. The Facility Emergency Plan and implementing procedures, at least once per 24 months.

e. The Facility Security Plan and implementing procedures, at least once per 24 months.

6.4.4 Authority

The Reactor Safety Committee shall report to the President and shall advise the Reactor Administrator on those areas of responsibility outlined in section 6.1.6 of these Technical Specifications.

6.4.5 Minutes of the Reactor Safety Committee

The Chairman of the Reactor Safety Committee shall direct the preparation, maintenance, and distribution of minutes of its activities. These minutes shall include a summary of all meetings, actions taken, audits, and reviews.

6.5 Procedures

There shall be written procedures that cover the following activities:

a. Startup, operation, and shutdown of the reactor.

b. Fuel movement and changes to the core and experiments that could affect reactivity.

c. Conduct of irradiations and experiments that could affect the safety of the reactor.

d. Preventive or corrective maintenance which could affect the safety of the reactor.

e. Surveillance, testing, and calibration of instruments, components and systems as specified in section 4.0 of these Technical Specifications.

f. Implementation of the Security Plan and Emergency Plan.

6.6 Safety Limit Violation

The following actions shall be taken in the event a Safety Limit is violated:

a. The reactor will shut down immediately and reactor operation will not be resumed without authorization by the Nuclear Regulatory Commission (NRC).

b. The Safety Limit violation shall be reported to the appropriate NRC Regional Office of Inspection and Enforcement, the Director of the NRC, and the Reactor Safety Committee not later than the next working day.

c. A Safety Limit Violation Report shall be prepared for review by the Reactor Safety Committee. This report shall describe the applicable circumstances preceding the violation, the effects of the violation upon facility components, systems or structures, and corrective action to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the NRC, and Reactor Safety Committee within 14 days of the violation.

6.7 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the appropriate NRC Regional Office.

6.7.1 Annual Operating Report

Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to June 30 of each year.

The annual operating reports made by licensees shall provides a comprehensive summary of the operating experience having safety significance that was gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(1) A brief narrative summary of

(a) Changes in facility design, performance characteristics, and operating procedures

6.7.1 contd

related to reactor safety that occurred during the reporting period.

(b) Results of major surveillance tests and inspections.

(2) A monthly tabulation showing the hours the reactor is operating.

(3) List of the unscheduled shutdowns, including the reasons therefore and corrective action taken, if any.

(4) Discussion of the major safety related corrective maintenance performed during the period, including the effects, if any, on the safe operation of the reactor, and the reasons for the 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 application for license and amendments thereto.

(b) Changes to the procedures as described in Facility Technical Specifications.

(c) Any new or untried experiments or tests performed during the reporting period.

(6) A summary of the safety evaluation made for each change, test, or experiment not submitted for NRC approval pursuant to 10 CFR 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety quesiton existed and that no technical specification was required.

(7) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the lecense as determined at or prior to the point of such release or discharge.

(a) Liquid Waste -

Total estimated quantity of radioactivity released (in curies) and total volume (in liters) of effluent water (including diluent) released.

6.7.1 contd

(b) Airborne Waste -

Total estimated quantity of radioactivity released (in curies) determined by an approved sampling and counting method.

(c) Solid Waste -

(i) Total amount of solid waste packaged (in cubic meters)

(ii) Total activity in solid waste (in curies)

(iii) The dates of shipments and disposition (if shipped off site)

(8) A description of the results of any environmental radiological surveys performed outside the facility.

(9) Radiation Exposure - A summary of radiation exposures greater than 100 mrem (50 mrem for persons under 18 years of age) received during the reporting period by facility personnel or visitors.

6.7.2 Reportable Occurrences

Reportable occurrences, including causes, probable consequences, corrective actions and measures to prevent recurrence, shall be reported to the NRC.

a. Prompt Notification with Written Followup -

The types of events listed shall be reported as expeditiously as possible by telephone and telegraph to the Director of the appropriate NRC Regional Office, or his designated representative no later than the first working day following the event, with a written followup report within two weeks. Information provided shall contain narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reached the setpoint specified as the limiting safety system setting in the technical specifications.

(2) Operation of the reactor when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the technical specifications.

6.7.2 contd

(3) Abnormal degradation discovered in a fission product barrier.

(4) Reactivity balance anomalies involving:

(a) disagreement between expected and actual critical positions of approximately 0.3% $\Delta k/k$;

(b) exceeding excess reactivity limits;

(c) shutdown margin less conservative than specified in technical specifications.

(5) Failure or malfunction of one (or more) component(s) which prevents, or could prevent, by itself, the fulfillment of the functional requirements of systems used to cope with accidents analyzed in Safety Analysis Report.

(6) Personnel error or procedural inadequacy which prevents, or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in Safety Analysis Report.

(7) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the basis for the Technical Specifications that have permitted reactor operation in a manner less conservative than assumed in the analyses.

(8) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications basis; or discovery during plant life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

6.8 Record Retention

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

a. Operating logs or data which shall identify:

1. Completion of pre-startup checkout, startup, power changes, and shutdown of the reactor.

2. Installation or removal of fuel elements, control rods or experiments that could affect core reactivity.

3. Installation or removal of jumpers, special tags or notices, or other temporary changes to reactor safety circuitry.

4. Rod worth measurements and other reactivity measurements.

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by technical specifications.

e. Facility radiation and contamination surveys.

f. Experiments performed with the reactor.

This requirement may be satisfied by the normal operations log book plus:

1. Records of radioactive material transferred from the facility as required by license.

2. Records required by the Reactor Safety Committee for the performance of new or special experiments.

g. Changes to operating procedures.

6.8.2 Records to be retained for the life of the facility:

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

b. Appropriate off site environmental monitoring surveys.

c. Fuel inventories and fuel transfers.

6.8.2 contd

d. Radiation exposures for all personnel.

e. Updated as-built drawings of the facility.

f. Records of transient or operational cycles for those components designed for a limited number of transients or cycles.

g. Records of training and qualification for members of the facility staff.

h. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

i. Records of meetings of the Reactor Safety Committee.