Proposed Amendment to

CHAPTER 16.0 TECHNICAL SPECIFICATIONS

Facility License R-31

(Docket Number 50-77)

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16.1.0 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.

16.1.1 Channel Calibration - A channel calibration is an adjustment of the channel 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, acutation, alarm, or trip.

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

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

16.1.4 Experiment -

a. An experiment is any of the following:

(1) An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;

(2) An evaluation or test of a reactor system operational, surveillance, or maintenance technique;

(3) An experimental or testing activity which is conducted within the confinement or containment system of the reactor; or

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

b. 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, bouyant, or other forces which are normal to the operating environment of the experiment or which might arise as a result of credible malfunctions.

c. Unsecured Experiment - Any experiment, or component of an experiment is deemed to be unsecured whenever it is not secured as defined in 16.1.4.b. above. Moving parts of experiments are deemed to be unsecured when they are in motion.

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

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

16.1.5 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.

16.1.6 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.

16.1.7 Measuring Channel - A measuring 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.

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

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

16.1.10 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.

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

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

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

16.1.14 Reactor Shutdown - The reactor shall be considered shutdown whenever

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

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

and:

b. 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.

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

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

16.2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

16.2.1 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 fragments 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.

Bases

The polyethylene core material does not melt below 200°C and is expected to maintain its integrity and retain essentially all of the fission fragments at temperatures below 200°C. The Hazards Summary Report dated February 1962 submitted on Docket F-15 by Aerojet-General Nucleonics (AGN) calculated a steady state core average temperature rise of 0.044C/watt. Therefore, a steady state power level of 100 watts would result in an average core temperature rise of 44°C. The corresponding maximum core temperature would be below 200°C thus assuring integrity of the core and retention of fission fragments.

Specification

c. The reactor shield tank water temperature shall be maintained above 10°C, and the water level in the tank shall not be more than 10 inches below the highest point on the manhole opening.

Bases

Low reactor shield tank water temperature may result in freezing of the water. The resultant expansion due to freezing of the water may damage the shield tank and other reactor components. This condition would degrade core containment and shielding capability. A safety limit of 10°C provides a margin for confidence that the reactor will not be operated with frozen shielding water.

The shield tank water level of 10 inches below the highest point on the manhole opening provides adequate biological shielding during reactor operation.

16.2.2 Limiting Safety Systems 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 #1	High Power	$\leq$ 0.2 watts
Nuclear Safety #2	High Power	$\leq$ 0.2 watts
Nuclear Safety #3	High Power	$\leq$ 0.2 watts

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

Bases

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 0.2 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 at about 120°C will assure safe shutdown without exceeding a core temperature of 200°C.

16.3 LIMITING CONDITIONS FOR OPERATION

16.3.1 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.

Specification

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 2% $\Delta k/k$.

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.

Bases

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.

16.3.2 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.

Specification

The reactor shall not be made critical unless the following specifications are met:

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

b. The maximum reactivity addition rate for each rod shall not exceed 0.04% $\Delta k/k$/sec.

c. 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.

d. All reactor safety system instrumentation shall be operable in accordance with Table 16-1 with the following allowable exceptions:

   1. Nuclear Safety Channel No.1 may be bypassed for a period not to exceed twelve consecutive hours provided Nuclear Safety Channel Nos. 2 and 3 are verified to be operable.

   2. Nuclear Safety Channel No. 3 may be bypassed for a period not to exceed 12 consecutive hours provided Nuclear Safety Channel Nos. 1 and 2 are verified to be operable.

   3. The seismic displacement scram may be out of service during reactor operation for no more than 24 hours in any 3-month period.

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

Bases

The specification on scram reactivity in conjunction with the safety system instrumentation and set points assure safe reactor shutdown during the most severe foreseeable transients. The limitations on reactivity addition rates allow only relatively slow increases of reactivity so that ample time will be available for manual or automatic scram during any operating conditions.

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 levels 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. In order to provide some time to correct channel defects, a maximum of 12 hours is allowed for operation with either Nuclear Safety Channels Nos. 1 or 3 bypassed if the remaining two channels are verified to be operable. Although some redundancy in the reactor protection system is lost during the limited time interval, all scram functions and monitoring capabilities are still available.

The AGN-201's negative temperature coefficient of reactivity causes a reactivity increase with decreasing core temperature. The shield water temperature safety channel 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 safety channel 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 safety channel 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. Due to the low probability of earthquake damage, the seismic instrument can be out of service during reactor operation for 24 hours in any 3-month period.

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 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.

16.3.3 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 results 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 results 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.

Bases

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.

16.3.4 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 fullfilled prior to reactor startup and 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 graphi...

Bases

The inherent reactor 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 10CFR20 limits under operating conditions, and to a level below Criterion 19, Appendix A, 10CFR50 recommendations under accident conditions.

16.4 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 surveillance requirements must be fulfilled prior to subsequent startup of the reactor.

16.4.1 Reactivity Limits

Applicability

This specification applies to the surveillance requirements for reactivity limits.

Objective

To assure that reactivity limits for Specification 16.3.1 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 at 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.

Bases

The control and safety rods are inspected and their 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 shutdown safely with one rod not functioning and that the maximum possible reactivity insertion will not result in periods shorter than those that can be terminated adequately 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.

16.4.2 Control and Safety System

Applicability

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

Objective

To assure that the reactor control and safety systems are operable as required by Specification 16.3.2

Specification

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

b. Safety and control rods and drives 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 each day's operation or prior to each operation extending more than one day:

Nuclear Safety #1, #2, and #3
Manual scram
Area radiation monitor

d. A channel test of the following safety channels shall be performed quarterly:

Shield water temperature
Shield water level
Seismic displacement

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

Nuclear Safety #1, #2, and #3
Area radiation monitor

f. Daily, prior to startup, each of the two safety rods 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 tank water level and temperature and seismic displacement safety channels shall be calibrated by perturbing the sensing element to the appropriate set point. These calibrations shall be performed 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.

Bases

The channel tests and checks required daily or before each startup will assure that the safety channels and scram functions are operable. Based on operating experience with reactors of this type, the annual scram measurements, channel calibrations and set point verifications are of sufficient frequency to assure, with a high degree of confidence, that the safety system setting will be within acceptable drift tolerance for operation. The periodic surveillance and calibration of the radiation monitoring instrumentation will assure that the radiation monitoring equipment is operable during reactor operation.

16.4.3 Reactor Structure

Applicability

This specification applies to surveillance requirements for reactor components other than control and safety rods.

Objective

The objective is to assure integrity of the reactor structures.

Specification

a. The shield tank shall be visually inspected every two years. If apparent excessive corrosion or other damage is observed, corrective measures shall be taken prior to subsequent reactor operation.

b. Visual inspection for water leakage from the shield tank shall be performed every year. Leakage shall be corrected prior to subsequent reactor operation.

Bases

Based on experience with reactors of this type, the frequency of inspection and leak test requirements of the shield tank will assure capability for radiation protection during reactor operation.

16.5.0 DESIGN FEATURES

16.5.1 Reactor

a. The reactor core, including control and safety rods, contains approximately 675 grams of U-235 in the form of 19.9% enriched UO_2 dispersed in approximately 11 kilograms of polethylene.

   The lower section of the core is supported by an aluminum rod hanging from a fuse link. The fuse melts at fuse temperatures of about 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 20 cm 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 6½ 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 1000 gallons of water. The complete reactor shield shall limit doses to operating personnel in restricted and unrestricted areas to levels less than permitted in 10 CF 20 under operating conditions.

e. Two safety rods and one control rod (identical in size) contain less than 15 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 contains unfueled polyethylene.

16.5.2 Fuel Storage

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

16.5.3 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 Pangborn Engineering building, 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. The reactor room doors are self-closing and-locking.

16.6 ADMINSTRATIVE CONTROLS

16.6.1 ORGANIZATION The administrative organization for control of the reator facility and its operation shall be as set forth in Figure 16-1 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.

16.6.1.1 PRESIDENT The President is the chief Administrative Officer responsible for the University and in whose name the application for licensing is made.

16.6.1.2 EXECUTIVE VICE PRESIDENT AND PROVOST. The executive Vice President and Provost reports to the President and is responsible for the internal administration of the University, both academic and non-academic. In this capacity he represents the President in all matters pertaining to the reactor except in those cases of health and safety for which the Radiation Safety Committee has authority.

16.6.1.3 DEAN OF ENGINEERING AND ARCHITECTURE. The Dean of Engineering and Architecture is responsible to the Executive Vice President and Provost for all departments in the School of Engineering and Architecture. In this capacity he shall have final authority and ultimate responsibility for the reactor facility and, within the limitations set forth by the facility license, make final policy decisions with respect to reactor operation; appoint the Reactor Administrator, with the advice of the Chairman of the Department of Mechanical Engineering; be advised on all matters concerning reactor safety by the Reactor Safety Committee.

16.6.1.4 CHAIRMAN, DEPARTMENT OF MECHANICAL ENGINEERING. The Chairman of the Department of Mechanical Engineering is responsible to the Dean of Engineering and Architecture for the routine functioning of the reactor facility. He shall advise the Dean of Engineering and Architecture concerning the appointment of the Reactor Administrator; and appoint the members of the Reactor Safety Committee.

16.6.1.5 REACTOR ADMINISTRATOR The Reactor Administrator is responsible to the Chairman of Mechanical Engineering Department for the daily administration of the reactor facility. In this capacity, he shall, within the policies set forth by the Chairman and the facility license, prepare all regulations for the facility, review and approve all procedures, seek approval of all procedures and proposals for changes and experiments from the Radiation Safety Committee, and be responsible for the health and safety of all personnel in the reactor facility. He shall be responsible for the official files of the facility, including storage of such prescribed logs and records of the facility which are no longer required for current operations. Prior to periods of scheduled absence, he shall designate an alternate and notify the Chairman.

16.6.1.6 REACTOR SUPERVISOR The RS shall be a licensed SRO. He shall be responsible for the preparation, promulgation, and enforcement of administrative controls including all rules, regulations, instructions and operating procedures to ensure that the 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 of the reactor; schedule reactor operations and maintenance; be responsible for the preparation and authentication of all prescribed logs and operating records of the facility; authorize all experiments, procedures, and changes thereto which have first received approval of the Reactor Safety Committee, the Radiation Safety Committee, and the Reactor Administrator, and be responsible for the preparation of all instructional manuals and experimental procedures involving use of the reactor. The Reactor Supervisor shall advise the Reactor Administrator of any scheduled periods of absence.

16.6.1.7 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. The Reactor Operator shall be in direct charge of the reactor console at all times during reactor operation and when the reactor is not secured and conform to the rules, instructions, and procedures established by the Reactor Administrator and Reactor Supervisor for operation of the reactor and the performance of experiments.

16.6.1.8 REACTOR SAFETY COMMITTEE The Reactor Safety Committee (RSC) shall be responsible for independent reviews and audits of facility operations to insure that the reactor is operated in a safe and competent manner and advise the Reactor Administrator in all matters related to reactor safety and personnel safety.

16.6.1.9 RADIATION SAFETY COMMITTEE. The Chairman of the Radiation Safety Committee shall be appointed by the Executive Vice President and Provost (EVPP). Additional members of the Committee shall be appointed by the EVPP with the advice of the Chairman. The Committee shall advise the EVPP in all matters concerning radiological aspects of the health and safety of personnel who might be exposed to radiation produced by University owned and/or operated sources or equipment. The committee shall review, approve, and promulgate a Radiation Safety Manual for the University. The Committee shall be informed of all occurrences related to radiation health and safety and reactor safety which are reportable to any authorities outside the University and advise the EVPP of such occurrences and make recommendations to the EVPP with regard to such matters.

16.6.1.10 RADIATION SAFETY OFFICER. The Radiation Safety Officer (RSO) shall be appointed by the Executive Vice President and Provost, with the advice of the Chairman of the Radiation Safety Committee. He is responsible to the Chairman for the day-to-day administration of the radiation safety program and shall serve as the Secretary of the Radiation Safety Committee and as an ex officio member of the Reactor Safety Committee. He shall prepare the University's Radiation Safety Manual and have the authority to enforce the regulations, rules and procedures set forth in the Radiation Safety Manual, suspend the operation and use of radiation producing devices when their use is in violation of these rules, and secure such sources of radiation until corrective action is taken. He shall also have the authority to disapprove the acquisition of radiation producing sources until satisfactory evidence is presented to ensure the safe storage and use of these facilities. The Radiation Safety Officer is also responsible for preparing, for the signature of the Chairman of the Radiation Safety Committee, reports of all reportable occurrences to the appropriate regulatory agency and for ensuring that appropriate follow-up action is taken.

16.6.1.11 OPERATING STAFF

a. The minimum staff during any time in which the reactor is not shutdown (1) shall consist of:

1. A licensed reactor operator in the control room (RO).

2. A second person present at the facility complex capable of carrying out any prescribed written instructions and instructions of operators and to summon help in the event the licensed operator becomes incapacitated. Unexpected absence for a period of 2 hours or less for this person is acceptable providing action is taken to obtain a replacement.(2)

(1) Reactor Shutdown is defined in Section 16.1

(2) See page 16-15

3. A licensed Senior Reactor Operator (SRO), if there are more than one licensed operators at the facility, shall be readily available on call. If there is only one licensed operator, he shall be an SRO and shall also fulfill sub paragraph 1 above.

   a. "Readily Available on Call: is defined to mean an individual who (1) has been specifically designated and whose designation is known to the RO, (2) keeps the RO informed of where he may be rapidly contacted by phone, and (3) is capable of getting to the reactor facility within a reasonable time period under normal conditions (e.g., 30 minutes or within a 20 mile radius).

16.6.2 STAFF QUALIFICATIONS The Reactor Administrator, Reactor Supervisor, Reactor Operator, and any Technicians performing work on the reactor shall meet the minimum qualifications set forth in ANS 15.4 "Standards for Selection and Training of Personnel for Research Reactors".

The qualifications of the Reactor Safety Committee members shall be five (5) years of professional experience in the field represented by the member or a baccalaureate degree plus at least two years 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 Chairman of Mechanical Engineering Department. In this case, a baccalaureate degree plus five (5) years experience will be required.

16.6.3 TRAINING The Reactor Administrator shall be responsible for the facility retraining program.

16.6.4 REACTOR SAFETY COMMITTEE REVIEWS, AUDITS, AND AUTHORITY

16.6.4.1 MEETINGS AND QUORUM The Reactor Safety Committee shall meet as necessary but at least twice each calendar year. A quorum for review shall consist of the chairman, or his designated alternate, and two other members, or alternate members as long as a majority of those present shall be regular members, and shall include representation in reactor operations and radiation protection. However, the operating staff shall not be a voting majority.

16.6.4.2 ALTERNATES Alternate members may be appointed by the Chairman of Mechanical Engineering Department to serve on a temporary basis; each appointment shall be in writing. No more than two alternates shall participate on a voting basis in Reactor Safety Committee activities at any one time.

16.6.4.3 REVIEWS The Reactor Safety Committee shall review:

a. Safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments, conducted without Nuclear Regulatory Commission approval under the provision of Section 50.59, 10 CFR, to verify that such actions do not constitute an unreviewed safety question.

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(2) Persons capable of performing emergency procedures shall be authorized by the Reactor Supervisor; be familiar with the Emergency Plan and capable of initiating the Evacuation Alarm; know the locations and be capable of using emergency and radiation survey equipment.

b. Proposed changes to procedures, equipment or systems that change the original intent or use, and are non-conservative, or those that involve an unreviewed safety question as defined in Section 50.59, 10CFR.

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.

d. Proposed changes in Technical Specifications or licenses.

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

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

g. Events which have been reported in writing within 24 hours to the Nuclear Regulatory Commission.

h. Audit reports.

16.6.4.4 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. Individual audits may be performed by one individual who need not be an identified Reactor Safety Committee member. These audits shall examine the operating records and encompass:

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.

f. Any other area of facility operation considered appropriate by the Reactor Safety Committee or the Chairman of Mechanical Engineering Department.

16.6.4.5 AUTHORITY The Reactor Safety Committee shall report to the Chairman of Mechanical Engineering Department and advise the Reactor Administrator on those areas of responsibility specified in sections 16.6.4.3 and 16.6.4.4.

16.6.4.6 RECORDS AND REPORTS OF THE REACTOR SAFETY COMMITTEE The chairman of the Reactor Safety Committee shall prepare, maintain, and distribute records of its activities as indicated below:

a. Minutes of each Reactor Safety Committee meeting shall be prepared, and forwarded to the Chairman of Mechanical Engineering and the Committee members within 30 days following each meeting.

b. Reports of all reviews and audits shall be prepared and forwarded to the Chairman of Mechanical Engineering within 30 days following completion of the review or immediately upon completion if corrective action is required.

c. Reviews of approvals requested by the Reactor Administrator for proposed changes shall be forwarded to the Chairman of Mechanical Engineering upon completion.

16.6.5 APPROVALS The procedure for obtaining approval for any change, modification, or other item which requires approval of the Reactor Safety Committee shall be as follows:

1. The Reactor Supervisor shall prepare a proposal for review of the Reactor Administrator who shall submit it for approval to the Reactor Safety Committee. The Reactor Safety Committee shall be responsible for review and audit as prescribed in Section 16.6.4. A copy of the findings of this committee shall be submitted to the Radiation Safety Officer for action as required by the University Radiation Safety Committee.

2. The Reactor Administrator shall submit copies of proposals reviewed by the Reactor Safety Committee to the Chairman of Mechanical Engineering.

3. The Reactor Administrator shall upon receipt of the required approvals from the Reactor Safety Committee and the Radiation Safety Committee authorize the Reactor Supervisor to proceed with the proposed change or modification.

16.6.6 PROCEDURES There shall be written operating procedures that cover the following activities. They shall be approved by the Reactor Administrator.

a. Conduct of irradiations and experiments that could affect the operation or safety of the reactor.

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

c. Fuel movement and changes to the core and experiments that can effect the reactivity.

d. Preventive or corrective maintenance which could have an effect on the safety of the reactor.

e. Surveillance, testing and calibration of instruments, components and systems involving nuclear safety.

f. Review and approval of changes to procedures.

g. Personnel radiation protection consistent with 10 CFR Part 20.

h. Implementation of the Security Plan and Emergency Plan.

i. Administrative control of operation and maintenance.

Though substantive changes to the above procedures shall be made only with approval by the Reactor Administrator, temporary changes to the procedures that do not change their original intent may be made by the Reactor Supervisor. All such temporary changes shall be documented, and subsequently approved by the Reactor Administrator within 14 days.

16.6.7 EXPERIMENTS

a. Prior to initiating any new reactor experiment, e.g., class of experiments that could affect reactivity of the reactor or result in release of radioactive materials, an experiment plan shall be prepared, reviewed by the Reactor Safety Committee, and the Radiation Safety Committee and approved by the Reactor Administrator.

b. Each experiment plan shall (1) identify the type of experiment (previously approved or recently reviewed per 16.6.4), (2) identify the experimenters and (3) have been approved by the licensed senior operator in charge of reactor operation.

16.6.8 SAFETY LIMIT VIOLATION The following actions shall be taken in the event a Safety Limit is violated:

a. The reactor will be 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 Director of the appropriate NRC Regional Office of Inspection and Enforcement (or his designee), the Chairman of Mechanical Engineering and to the Reactor Safety Committee not later than the next work day.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Reactor Safety Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

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

16.6.9 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 unless otherwise noted.

16.6.9.1 ROUTINE REPORTS

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the Hazards Summary Report (hereinafter Safety Analysis Report) and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

   Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of power operation, (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e. initial criticality, completion of startup test program and resumption or commencement of power operation), supplementary reports shall be submitted at least every three months untill all three events have been completed.

b. Annual Operating Report Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 31 of each year.

   The annual operating reports made by licensees shall provide 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 related to reactor safety that occured 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 question existed and that no technical specification change 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 license as determined at or prior to the point of such release or discharge.

   (a) <u>Liquid Waste</u> (summarized on a 3 month basis)

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

   (b) <u>Airborne Waste</u> (Summarized on a 3 month basis)

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

   (c) <u>Solid Waste</u> (summarized on an annual basis)

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

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

      (3) 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 vistors.

16.6.9.2 REPORTABLE OCCURENCES Reportable occurences, including causes probable consequences, corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, an amended licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate NRC Regional Office, or his designated representative no later than the first work 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 or other systems 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 or failure to complete the required protective function.

(2) Operation of the reactor or affected systems when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the technical specifications without taking permitted remedial action.

(3) Abnormal degradation discovered in a fission product barrier, i.e., cracked fuel disc, primary gas-tight seals.

(4) Reactivity balance anomalies involving:

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

(b) exceeding excess reactivity limit;

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

(d) unexpected short-term reactivity changes that resulted in a period of 10 seconds or less;

(e) if sub-critical, an unplanned reactivity insertion of more than approximately 0.5% $\Delta k/k$ or any unplanned criticality.

(5) Failure or malfunction of one (or more) component(s) which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the 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) Unscheduled Conditions arising from natural or man-made events that, as a direct result of the event require reactor shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

(8) 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 bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) 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 analysis in the Safety Analysis Report or technical specifications bases; 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.

16.6.9.3 SPECIAL REPORTS Special reports which may be required by the Nuclear Regulatory Commission shall be submitted to the Director of the appropriate NRC Regional Office within the time period specified for each report.

16.6.10 RECORD RETENTION

16.6.10.1 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.

16.6.10.2 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.

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 10CFR 50.59.

i. Records of meetings of the Reactor Safety Committee.

TABLE 16-1 LIMITING CONDITIONS FOR OPERATION, INSTRUMENTATION

Safety Channel	Set Point	Function
Nuclear Safety #1		
Low count rate	$\geq$ 120 cpm	scram below 120 cpm
Nuclear Safety #2 (log)		
High power	$\leq$ 0.2 watts	scram at power > 0.2 watts
Low power	$\geq 0.5 \times 10^{-13}$ amps	scram at source levels $< 0.5 \times 10^{-13}$ amps
Reactor period	$\geq$ 8 sec	scram at period < 8 sec
Nuclear safety #3 (linear)		
High power	$\leq$ 0.2 watt	scram at power > 0.2 watt
Low power	$\geq$ 5% full scale	scram at source levels < 5% of full scale
Shield water temperature	$\geq$ 15°C	scram at temperature < 15°C
Shield water level	$\leq$ 10.5 inches	scram at water levels > 10.5 inches below highest point on manhold opening
Seismic displacement	$\leq$ 1/16"	scram at displacements > 1/16"
Manual scram	--	scram at operator option
Radiation monitor	--	alarm at or below level set to meet requirements of 10 CFR Part 20

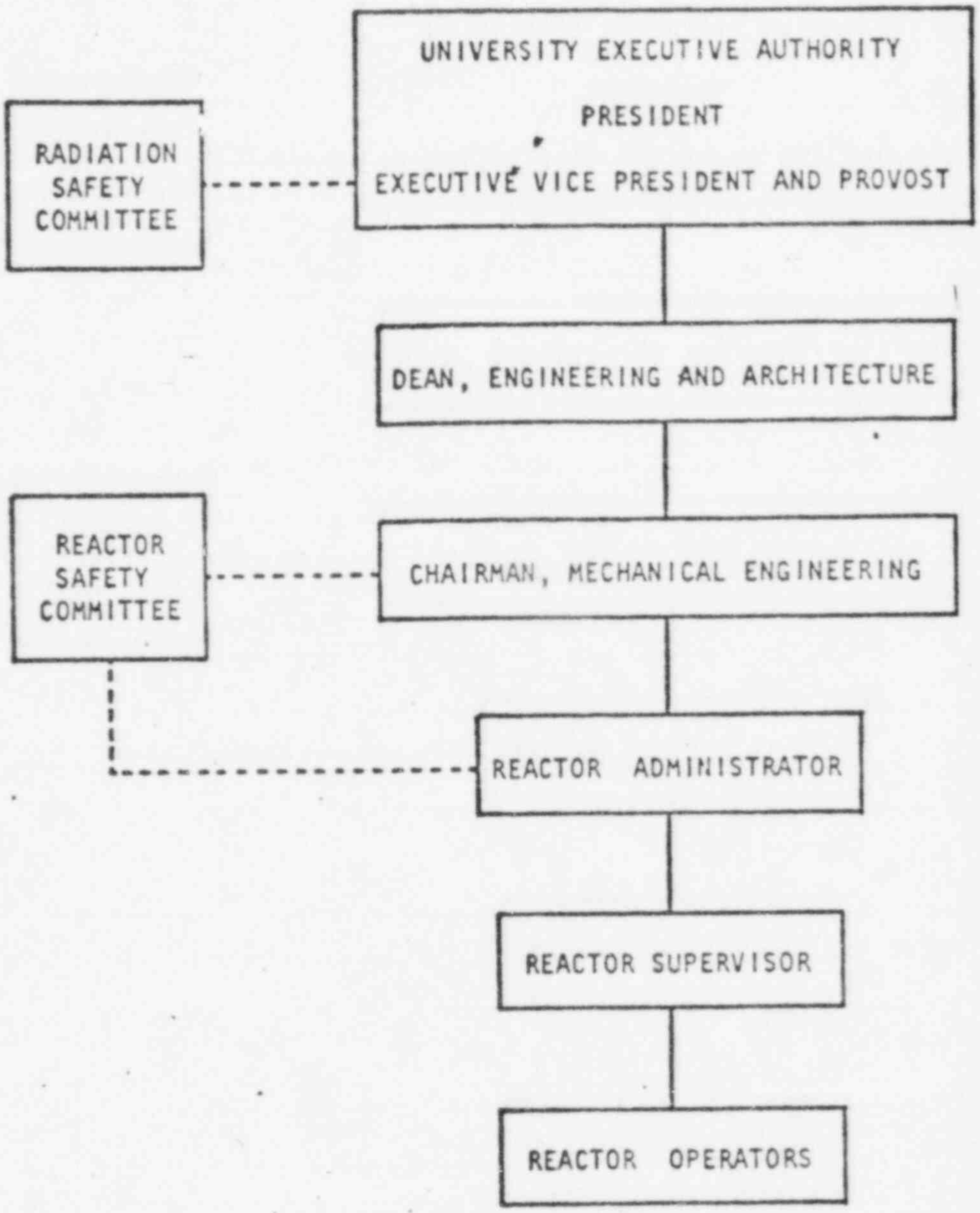

FIG. 16-1 ADMINISTRATIVE ORGANIZATION FOR REACTOR CONTROL AND SAFETY