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

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TEXAS A&M UNIVERSITY

DOCKET NO. 50-59

AMENDED FACILITY OPERATING LICENSE

Amendment No. 12
License No. R-23

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

   A. The application for amendment by Texas A&M University (the licensee) dated May 31, 1977, as supplemented by filings dated September 29, 1978, December 11, 1978 and December 18, 1978, and March 23, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

   D. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the regulations of the Commission;

   E. The licensee is a nonprofit educational institution, and will use the facility for the conduct of educational activities, and has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Idemnity Agreements," of the Commission's regulations;

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

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

   H. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including 10 CFR Sections 30.33, 70.23 and 70.31.

2. Facility License No. R-23 is hereby amended in its entirety to read:

   A. This license applies to the homogeneous nuclear reactor model AGN-201M, Serial No. 106 (the reactor), owned by the Texas A&M University (the licensee), located on its campus at College Station, Texas and described in the application for license dated June 13, 1957, and subsequent amendments and supplements thereto, including the application for license renewal dated May 31, 1977, and supplements thereto dated September 29, December 11 and December 18, 1978, and March 23, 1979.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Texas A&M University:

      (1) Pursuant to Section 104c of the Act and 10 CFR, Chapter, 1 Part 50, “Licensing of Production and Utilization Facilities” to possess, but not use or operate the reactor as a utilization facility at the designated locations in College Station, Texas, in accordance with the procedures and limitations set forth in this license.

      (2) Pursuant to the Act and 10 CFR Part 30 “Rules of general applicability to domestic licensing of byproduct material” to possess, but not separate, such byproduct material present in the AGN-201M reactor non-fuel components.

      (3) Pursuant to the Act and 10 CFR Part 70 “Domestic licensing of special nuclear material” to possess, but not separate, such special nuclear material in the AGN-201M reactor non-fuel components.

   C. This license shall be deemed to contain, and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

      (1) <u>Maximum Power Level</u>

         The licensee is authorized to operate the reactor at a steady-state power levels up to a maximum of 5 watts (thermal)

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment 16, are hereby incorporated in their entirety in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) [DELETED]

D. [DELETED]

E. This license is effective as of the date of issuance and shall expire at midnight, August 26, 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Brian K. Grimes, Assistant Director
For Engineering & Projects
Division of Operating Reactors

Attachment:
Appendix A, Technical
Specifications dated

Date of Issuance: April 25, 1979

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APPENDIX A

LICENSE NO. R-23

TECHNICAL SPECIFICATIONS

FOR

TEXAS A&M UNIVERSITY AGN-201M REACTOR (SERIAL #106)

DOCKET NO. 50-59

Amendment No. 12

Dated: April 25, 1979

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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 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, actuation, alarm, or trip.

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.

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

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 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 1.4.b. above. Moving parts of experiments are deemed to be unsecured when they are in motion.

Correction - 5-24-79

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.

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 exit from the reactor shielding. Experimental facilities shall include the thermal column, glory hole, and access ports.

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 Company.

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.

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

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

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.

Evaluations of potential reactivity worth of experiments also shall include effects of 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.

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

1.12 Reactor Operation - Reactor operation is any condition wherein the reactor is not shut down.

Correction - 5-24-79

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.

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.

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

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.

Correction - 5-24-79

2.0 SAFETY LIMITS AND LIMITING 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.

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 products 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 0.44 C/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 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 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 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.

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.

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% Δ 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% Δ k/k.

c. The reactivity worth of the control and safety rods shall ensure subcriticality 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.

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

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

b. The average reactivity addition rate for each control or safety rod shall not exceed 0.065% Δ k/k per second.

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 3.1 whenever the reactor control or safety rods are not in their fully withdrawn position.

e. The shield water level interlock shall be set to prevent reactor start-up and scram the reactor if the shield water level falls 10.5 inches below the highest point on the reactor shield tank manhole opening.

f. 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.

Table 3.1

Safety Channel	Set Point	Function
Nuclear Safety #1		
Low count rate	$\geq$ 10 cps	scram below 10 cps
Nuclear Safety #2		
High Power	$\leq$ 10 watt	scram at power > 10 watts
Low Power	$\geq 1.0 \times 10^{-12}$ amps	scram at source levels $< 1.0 \times 10^{-12}$ amps
Reactor period	$\geq$ 5 sec	scram at periods <5 sec
Nuclear Safety #3 (Linear Power)		
High Power	$\leq$ 10 watt	scram at power >10 watt
Low Power	$\geq 1.0 \times 10^{-12}$ amps	scram at source levels $< 1.0 \times 10^{-12}$ amps
Manual scram	------------------	scram at operator option

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

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

Bases

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

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 thus assuring safe and immediate shutdown in case of a power outage.

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 experiment 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 or stored within the confines of the reactor facility.

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.

Bases

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

3.4 RADIATION MONITORING, CONTROL, AND SHIELDING

Applicability

This specification applies to radiation monitoring, control, and reactor shielding required during reactor operation.

Objective

To protect facility personnel and the public from radiation exposure.

Specification

a. An operable portable radiation survey instrument capable of detecting gamma radiation shall be immediately available to reactor operating personnel whenever the reactor is not shutdown.

b. The reactor room and accelerator room shall be considered restricted areas whenever the reactor is not shutdown.

c. The accelerator room shall be considered a radiation area whenever the reactor is not shutdown.

d. The reactor room shall be considered a radiation area whenever the reactor is operated at a power level less than 0.9 watts.

e. Whenever the reactor is operated at a power level equal to or greater than 0.9 watts the reactor room shall be considered a high radiation area and the reactor room entrance shall be equipped with a control device which shall energize a conspicuous visible and audible alarm signal in such manner that the individual entering the reactor room and the reactor operator are made aware of the entry, or the reactor room entrance shall be maintained locked except during periods when access to the area is required, with positive control over each entry.

f. The following shielding requirements shall be fulfilled during reactor operation:

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

   2. The thermal column shall be filled with water or graphite except during a critical experiment (core loading) or during other approved experiments requiring the thermal column to be empty.

Bases

Radiation surveys performed under the supervision of a qualified health physicist have shown that the total gamma, thermal neutron, and fast neutron radiation dose rate in the reactor room, at the closest approach to the reactor, is less than 100 mrem/hr at reactor power levels less than 1.0

watt, and that the total gamma, thermal neutron, and fast neutron dose rate in the accelerator room is less than 15 mrem/hr at reactor power levels less than or equal to 5.0 watts and the thermal column filled with water.

The facility shielding in conjunction with radiation monitoring, control, and restricted 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.

3.5 AGN-201M Reactor Components Stored at the Nuclear Science Center (NSC) Facility

Applicability

This specification applies to all AGN-201M reactor components stored at the NSC facility.

Objective

To verify the AGN-201M reactor components, are in two specified secured locations with no evidence of tampering, while stored at the NSC facility.

Specifications

1. Accelerator Building

   AGN-201M reactor components shall be stored in a secured fenced area in the Accelerator Building. The AGN-201M Reactor Supervisor or designee shall control access to the secured fenced area.

   The following AGN-201M reactor components shall be stored in the Accelerator Building:

   a. AGN-201M reactor control panel and associated electronic equipment

   b. AGN-201M Shield Tank, Reactor Tank, Core Tank, and associated internal components

2. Cargo Container

   AGN-201M reactor components not stored in the Accelerator Building shall be stored in a secured cargo container with a tamper proof seal affixed in such a way that opening the cargo container will break the seal. Access to the cargo container shall be restricted to personnel authorized by the AGN-201M Reactor Supervisor or designee.

Bases

These Technical Specifications ensure that the AGN-201M reactor components are secured and prevent tampering while stored at the NSC facility.

4.2 Control and Safety Systems

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 3.2.

Specification

a. Safety and control rod scram times and average reactivity 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 the first reactor startup of the day or prior to each reactor operation extending more than one day:

   Nuclear Safety #1, #2, and #3

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 reactor operation or prior to each reactor operation extending more than one day, safety rod #1 shall be inserted and scrammed to verify operability of the manual scram system.

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 interlock, shield water temperature interlock and seismic displacement safety channel shall be tested by perturbing the sensing element to the appropriate set point. These tests shall be performed anually, but at intervals not to exceed 16 months.

Correction - 5-24-79

4.0 SURVEILLANCE REQUIREMENTS

Actions specified in this section, with the exception of Section 4.5, 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 unless reactor operation is required for performance of the surveillance. Any surveillance that can only be performed during reactor operation may be postponed until the reactor is operable.

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 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 rod reactivity worths are 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.

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, 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.

4.3 Reactor Structure

Applicability

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

Objective

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 annually. 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.

4.4 Radiation Monitoring and Control

Applicability

This specification applies to the surveillance requirements of the radiation monitoring and control systems.

Objective

To assure that the radiation monitoring and control systems are operable and that all radiation and high radiation areas within the reactor facility are identified and controlled as required by Specification 3.4.

Specification

a. All portable radiation survey instruments assigned to the reactor facility shall be calibrated under the supervision of the Radiological Safety Office annually, but at intervals not to exceed 16 months.

b. Prior to each day's reactor operation or prior to each reactor operation extending more than one day, the reactor room high radiation area alarm shall be verified to be operable. (See Article 3.4.e)

c. A radiation survey of the reactor room, reactor control room, and accelerator room shall be performed under the supervision of the Radiological Safety Office annually, but at intervals not to exceed 16 months, to determine the location of radiation and high radiation areas corresponding to reactor operating power levels.

Bases

The periodic calibration of radiation monitoring equipment and the surveillance of the reactor room high radiation area alarm will assure that the radiation monitoring and control systems are operable during reactor operation. (See Article 3.4.e).

The periodic radiation surveys will verify the location of radiation and high radiation areas and will assist reactor facility personnel in properly labeling and controlling each location in accordance with 10 CFR 20.

4.5 Reactor Components Stored at the Nuclear Science Center (NSC) Facility

Applicability

This applies to the surveillance requirements of the AGN-201M reactor components stored at the NSC Facility.

Objective

To verify the AGN-201M reactor components remain stored in specified locations and protected from tampering while at the NSC facility.

Specifications

a. NSC Accelerator Building

1. Once a quarter the secured fenced area in the Accelerator Building shall be inspected to verify all reactor components are present and no indications of tampering exist. If indications of tampering are discovered, the Director of Nuclear Engineering or designee shall be notified. In addition, a special report in accordance with Technical Specification Section 6.9.3 shall be transmitted to the U.S. NRC.

2. Once a quarter a radiation and contamination survey shall be conducted around the exterior of the stored AGN-201M reactor components to verify that contamination is not migrating from the contained reactor components. If detectable loose surface contamination exceeds levels acceptable for an unrestricted area, the reactor components shall be decontaminated and repackaged as necessary.

b. Cargo Container

1. Once a quarter a survey of the cargo container is required to verify that the tamper proof seal has not been broken. In the event the seal is found broken, the Director of Nuclear Engineering or designee shall be notified and an inventory of the cargo container shall be performed. In addition, a special report in accordance with Technical Specification Section 6.9.3 shall be transmitted to the U.S. NRC.

2. Once a quarter a radiation and contamination survey shall be conducted of the exterior of the cargo container to verify that contamination is not migrating from the contained components. If detectable loose surface contamination exceeds levels acceptable for an unrestricted area, the cargo container exterior shall be decontaminated and the source of contamination identified and secured.

Bases

These surveillances shall verify the components necessary for reassembly of the AGN-201M reactor remain secure, no indications of tampering exist, and the radiological conditions of storage remain unchanged.

5.0 DESIGN FEATURES

5.1 Reactor

a. The reactor core, including control and safety rods, contains approximately 660 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 a fuse temperature of about 120°C causing the lower core section to fall away from the upper section reducing reactivity by at least 5% Δ 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 6 ½ 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 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 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 department laboratories. The storage array shall be such that Keff is no greater than 0.8 for all conditions of moderation and reflection.

5.3 AGN-201M Reactor and Associated Components Storage Locations

The AGN-201M reactor and associated components shall be stored in the following locations:

- Texas A&M Engineering Experiment Station Nuclear Science Center facility
  - Accelerator Building
  - Cargo Container

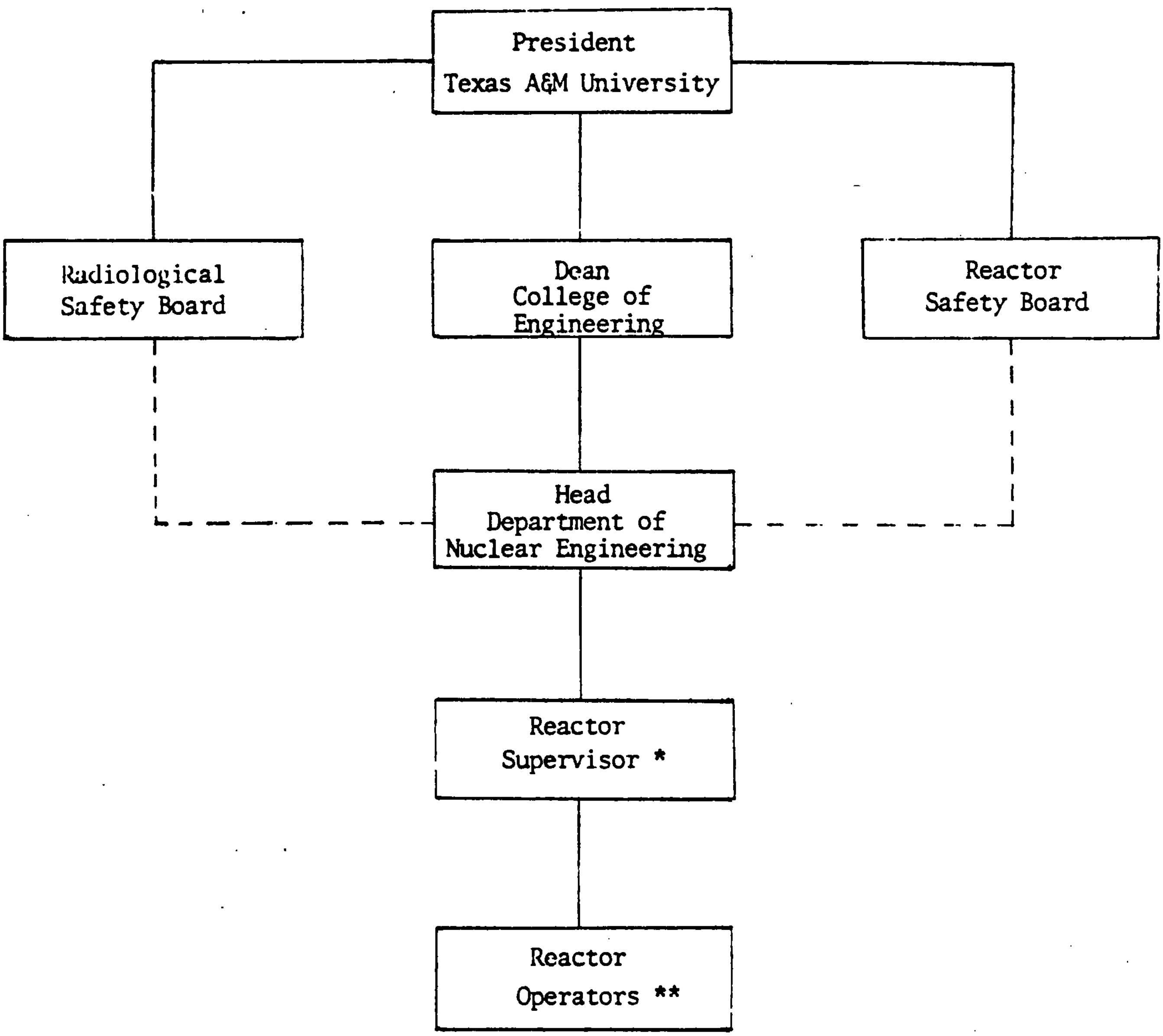
6.0 ADMINISTRATIVE CONTROLS

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.

FIGURE 1

Administrative Organization of the Texas A&M University AGN-201M Reactor Facility NRC License R-23



* Requires NRC Senior Operators License

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

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.

6.1.2 Dean, College of Engineering

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

6.1.3 Head, Department of Nuclear Engineering

The Head of the Department of Nuclear Engineering is the administrative officer responsible for the operation of the Department of Nuclear Engineering, including the AGN-201M 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 Radiological Safety Board and/or the Reactor Safety Board in all matters concerning unresolved safety questions, new experiments and new procedures, and facility modifications which might affect safety. He shall be an <u>ex officio</u> member of the Reactor Safety Board.

6.1.4 Reactor Supervisor

The Reactor Supervisor shall be a licensed SRO and 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 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 Board and/or the Radiological Safety Committee and the Head of the Department of Nuclear Engineering; 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 Board

The Reactor Safety Board 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 recurrence; reporting all their findings and recommendations concerning the reactor facility to the Head of the Department of Nuclear Engineering.

6.1.7 Radiological Safety Board

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

6.1.8 Radiological Safety Officer

The Radiological Safety Officer shall review and approve all procedures and experiments involving radiological safety. 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 Head of the Department of Nuclear Engineering. He shall provide personnel dosimetry and keep records of personnel radiation exposure. He shall advise the Head of the Department of Nuclear Engineering on all matters concerning radiological safety at the reactor facility. The Radiological Safety Officer shall be an *ex officio* member of the Reactor Safety Board.

6.1.9 Operating Staff

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

   1. One licensed Reactor Operator in the reactor control room.

   2. One other person in the reactor room or reactor control room certified by the Reactor Supervisor as qualified to activate manual scram and initiate emergency procedures.

   3. One licensed Senior Reactor Operator readily available on call. This requirement can be satisfied by having a licensed Senior Reactor Operator perform the duties stated in paragraph 1 or 2 above or by designating a licensed Senior Reactor Operator who can be readily contacted by telephone and who can arrive at the reactor facility within 30 minutes.

b. A licensed Senior Reactor Operator shall supervise all reactor maintenance or modification which could affect the reactivity of the reactor.

6.2 Staff Qualifications

The Head of the Department of Nuclear Engineering, 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 Board members shall have a minimum of five (5) years experience in their profession or a baccalaureate degree and two (2) years of professional experience. Reactor Safety Board members will generally be University faculty members with considerable experience in their area of expertise. The Radiological Safety Officer shall have a baccalaureate degree in biological or physical science and have at least two (2) years experience in health physics.

6.3 Training

The Head of the Department of Nuclear Engineering 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 Board

6.4.1 Meetings and Quorum

The Reactor Safety Board shall meet as often as deemed necessary by the Reactor Safety Board 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 Board meeting.

6.4.2 Reviews

The Reactor Safety Board 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 do not constitute an unreviewed safety question.

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 10 CFR 50 paragraph 50.59.

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 10 CFR 50 Section 50.59.

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. Reportable occurrences.

h. Audit reports.

6.4.3 Audits

Audits of facility activities shall be performed at least quarterly under the cognizance of the Reactor Safety Board 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 the facility operation to the Technical Specifications and applicable license conditions, at least annually.

b. The Facility Emergency Plan and implementing procedures, at least every two years.

6.4.4 Authority

The Reactor Safety Board shall report to the President and shall advise the Head of the Department of Nuclear Engineering on those areas of responsibility outlined in section 6.1.6 of these Technical Specifications.

6.4.5 Minutes of the Reactor Safety Board

The Chairman of the Reactor Safety Board 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 Approvals

The procedure for obtaining approval for any change, modification, or procedure which requires approval of the Reactor Safety Board shall be as follows:

a. The Reactor Supervisor shall prepare the proposal for review and approval by the Head of the Department of Nuclear Engineering.

b. The Head of the Department of Nuclear Engineering shall submit the proposal to the Chairman of the Reactor Safety Board.

c. The Chairman of the Reactor Safety Board shall submit the proposal to the Reactor Safety Board members for review and comment.

d. The Reactor Safety Board can approve the proposal by majority vote.

6.6 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 irradiation and experiments that could affect the operation or safety of the reactor.

d. Preventative 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 Emergency Plan.

The above listed procedures shall be approved by the Head of the Department of Nuclear Engineering and the Reactor Safety Board. Temporary procedures which do not change the intent of previously approved procedures and which do not involve any unreviewed safety question may be employed on approval by the Reactor Supervisor.

6.7 Experiments

a. Prior to initiating any new reactor experiment and experiment procedures shall be prepared by the Reactor Supervisor and reviewed and approved by the Head of the Department of Nuclear Engineering and the Reactor Safety Board.

b. Approved experiments shall only be performed under the cognizance of the Head of the Department of Nuclear Engineering and the Reactor Supervisor.

6.8 Safety Limit Violation

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

a. The reactor will be shutdown 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 Board not later than the next work day.

c. A Safety Limit Violation Report shall be prepared for review by the Reactor Safety Board. 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 Board within 14 days of the violation.

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.

6.9.1 Annual Operating Report

Routine annual operating reports shall be submitted no later than ninety (90) days following the end of the operating 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 occurred during the reporting period.

   (b) Results of major surveillance tests and inspections.

2. A tabulation showing the hours the reactor was operated and the energy produced by the reactor in watt-hours.

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, Section 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 licensee as determined at or prior to the point of such release or discharge.

   (a) <u>Liquid Waste (summarized for each release)</u>

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

   (b) <u>Solid Waste (summarized for each release)</u>

      (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 visitors.

6.9.2 Reportable Occurrences

Reportable occurrences, 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.

4. Reactivity balance anomalies involving:

   (a) Disagreement between expected and actual critical rod positions of approximately 0.3% Δ k/k.

   (b) Exceeding excess reactivity limit.

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

   (d) If sub-critical, an unplanned reactivity insertion of more than approximately 0.5% Δ 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.

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.

6.10 Record Retention

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 Board for the performance of new or special experiments.

g. Changes to operating procedures.

6.10.2 <u>Records to be Retained for the Life of the Facility</u>:

a. Records of liquid and solid 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 10 CFR 50.59.

i. Records of meetings of the Reactor Safety Board.