May 17, 1989

Docket No. 50-284

Dr. John M. Hutchinson
Vice President of Academic Affairs
Idaho State University, Box 8063
Pocatello, Idaho 83209

Dear Dr. Hutchinson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE NO. R-110 - IDAHO STATE UNIVERSITY

The Commission has issued the enclosed Amendment No. 4 to Facility Operating License No. R-110 for the Idaho State University AGN-201 M Reactor. The amendment consists of changes to the Technical Specifications (TS) in response to your application (letter from A. E. Wilson to C. Miller) dated October 28, 1988 and supplemented by letters dated February 15 and April 14, 1989.

The amendment consists of changes to the TS to include administrative changes and to reflect the operational status of the facility.

A copy of the related Safety Evaluation supporting Amendment No. 4 is enclosed.

Sincerely,

/s/
Theodore S. Michaels, Project Manager
Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 4
2. Safety Evaluation

cc w/enclosures:
See next page

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

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Idaho State University

Docket No. 50-284

cc: State Planning and Community
Affairs Agency
State of Idaho
Boise, Idaho 83720

Mr. Alert E. Wilson
Reactor Supervisor
Idaho State University
Pocatello, Idaho 83209

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

IDAHO STATE UNIVERSITY

DOCKET NO. 50-284

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. R-110

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

   A. The application for amendment to Facility Operating License No. R-110 filed by the Idaho State University (the licensee), dated October 28, 1988 and supplemented by letters dated February 15, and April 14, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the 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 set forth in 10 CFR Chapter I;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of License No. R-110 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 4, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore S. Michaels for
Charles L. Miller, Director
Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance: May 17, 1989

ENCLOSURE TO LICENSE AMENDMENT NO. 4

FACILITY OPERATING LICENSE NO. R-110

DOCKET NO. 50-284

The Appendix A Technical Specifications have been replaced in its entirety. Apart from page number revisions, caused by retyping, the only changes that have been made appear in the following pages, which are identified by Amendment Number and contain a vertical line indicating the area of change.

Revised Pages

6, 10, 12, 19, 20, 21, 25, and 26

<u>APPENDIX A</u>

<u>TO FACILITY OPERATING</u>

<u>LICENSE NO. R-110</u>

<u>TECHNICAL SPECIFICATIONS</u>

<u>FOR</u>

<u>IDAHO STATE UNIVERSITY AGN-201 M REACTOR (SERIAL #103)</u>

<u>DOCKET NO. 50-284</u>

AMENDMENT NO. 4

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; or

(3) 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, buoyant, 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.

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 shutdown.

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

1.14 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 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.

2.0 SAFETY LIMITS AND LIMITED SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

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

Objective

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

Specification

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

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

Basis

The polyethylene core material does not 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 0.44C/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 reactivity loss greater than ~~5%Δ k~~ 5% Δ k/k.

Basis

Based on instrumentation response times and scram tests, the AGN Hazards Report concluded that reactor periods in excess of 30-50 milliseconds 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 milliseconds. 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 characteristic 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% Δk/k.

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

Basis

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

3.2 Control and Safety Systems

Applicability

These specifications apply to the reactor control and safety systems.

Objective

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

TABLE 3.1

SAFETY CHANNEL	SET POINT	FUNCTION
Nuclear Safety #1 Low Power	5% Full Scale	Scram at source levels < 5% of full scale
Nuclear Safety #2 High Power	10 Watt	Scram at power > 10 watt
Low Power	3.0×10^{-13} amps	Scram at source levels < 3×10^{-13} amps
Reactor Period	5 sec	Scram at periods < 5 sec
Nuclear Safety #3 (Linear Power)		
High Power	10 Watt	Scram at power > 10 watt
Low Power	5% full scale	Scram at source levels < 5% of full scale
Manual Scram	-----	Scram at operator option
Radiation Monitor	-----	Alarm at or below level set to meet requirements of 10 CFR Part 20

Specification

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

b. The average reactivity addition rate for each control or safety rod shall not exceed 0.065% $\Delta k/k$ 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 with the exception that Safety Channels 1 or 3 may be bypassed whenever the reactor control or safety rods are not in their fully withdrawn position.

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

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.

g. The seismic displacement interlock sensor 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.

Basis

The specifications on scram withdrawal time in conjunction with the safety system instrumentation and set points assure safe reactor shutdown during the most severe 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~~ levels initiate redundant automatic protective action at power level scrams low enough to assure safe shutdown without exceeding any safety limits. The period scram conservatively limits the rate of rise of reactor power to periods which are manually controllable and will automatically scram the reactor in the event of unexpected large reactivity additions.

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

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

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

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

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

3.3 Limitations on Experiments

Applicability

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

Objective

To prevent damage to the reactor or excessive release of radioactive materials in the event of an 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 of (2) restricted areas during the length of time required to evacuate the restricted area.

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

Basis

These specifications are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from an 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 and installed 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 shall be considered a restricted area whenever the reactor is not shutdown.

c. 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 measurement of reactivity worth of thermal column water or graphite, or when the neutron radiography collimator is being used.

   3. The movable shield doors above the thermal column shall be maintained in a closed position whenever the reactor is operated at a power greater than 0.5 watts.

Basis

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 outside the designated high radiation areas is less than 25 mrem/hr at reactor power levels less than 5.0 watts. ~~and that the total gamma, thermal neutron, and fast 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 designated restricted radiation areas is designed to limit radiation doses to facility personnel and to the public to a level below 10 CFR 20 limits under operating conditions, and to a level below Criterion 19, Appendix A, 10 CFR 50 recommendations under accident conditions.

4.0 SURVEILLANCE REQUIREMENTS

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

4.1 Reactivity Limits

Applicability

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

Basis

The control and safety 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 safely shutdown with one rod not functioning and that the maximum possible reactivity insertion will not result in reactor periods shorter than those than can be adequately terminated by either operator or automatic action. Based on experience with AGN reactors, significant changes in reactivity or rod worth are not expected within a 16-month period.

4.2 Control and Safety System

Applicability

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

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 drive shall be inspected for deterioration at intervals not to exceed 2 years.

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

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

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

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

   Nuclear Safety #1, #2, and #3

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

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

h. The shield tank water level interlock, shield water temperature interlock, and seismic displacement safety channel 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.

Basis

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.

Basis

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 areas within the reactor facility are identified and controlled as required by Specification 3.4.

Specification

a. All portable and installed radiation survey instruments assigned to the reactor facility shall be calibrated under the supervision of the Radiation Safety Officer 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 alarm shall be verified to be operable.

c. A radiation survey of the reactor room and reactor control room shall be performed under the supervision of the Radiation Safety Officer 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.

Basis

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.

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.

5.0 DESIGN FEATURES

5.1 Reactor

a. The reactor core, including control and safety rods, contains approximately 670 grams of U^{235} in the form of 20% enriched UO_2 dispersed in approximately 11 kilograms of polyethylene. The lower section of the core is supported by an aluminum rod hanging from a fuse link. The fuse melts at temperatures below 120°C causing the lower core section to fall away from the super 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/cm3) 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 of "thermal column tank" may serve as a shield tank when filled with water or a thermal column when filled with graphite.

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

e. Shielding is provided by a concrete wall constructed of 4" X 8" X 16" concrete blocks and 4" X 8" X 12" barytes concrete blocks for 5 watt operation. The blocks are held to close dimensional tolerance in manufacture and stacked in such a manner that voids in the completed wall are at a minimum. Near the beam ports and glory hole, high density blocks are used between 40 inches and 112 inches above the base. The use of these blocks further reduces radiation level in these areas. Overhead shielding is provided by 8 inch thick barytes blocks (minimum density 3.7 gcc).

As detailed in the amendment for 5 watt operation for Aerojet-General Nucleonics, dated 11 February 1957, and on file with the Commission in Docket 50-32, an 18 inch additional concrete shield wall was sufficient to maintain sub-tolerance radiation levels external to the shield when operating at 5 watts. Subsequent analysis by Aerojet-General Nucleonics indicated that 16 inches of ordinary concrete shielding was sufficient. Twelve inches of barytes concrete is more effective than 18 inches of ordinary concrete.

The radiation levels associated with 5 watt operation (peak thermal flux of 2.5 x 10^8 n/cm^2 -sec) have been calculated by Aerojet-General Nucleonics. The table below gives Aerojet-General results for gamma dose for an 18" concrete shield:

Energy (MeV)	u	ux	B	e^{-ux}	Dose (mrem/hr)
3.0	0.081	3.64	1.9	0.027	1.5
2.2	0.105	4.73	2.2	0.0088	1.0
				Total =	2.5 mrem/hr (18" concrete)

Changing the e^{-ux} for a 16" concrete wall and using the same buildup factors yields the following table:

Energy (MeV)	u	ux	B	e^{-ux}	Dose (mrem/hr)
3.0	0.081	3.30	1.9	0.037	2.1
2.2	0.105	4.28	2.2	0.014	1.2
				Total =	3.3 mrem/hr (16" concrete)

The National Naval Medical Center reported neutron dose rates were less than 0.2 mrem/hr for 18" shield.

Although calculations and operating surveys of similar facilities show that at 5 watts power there will be no area on the reactor floor outside the concrete shield where the total radiation exceeds tolerance levels; nevertheless, the reactor floor is a control area with restricted access.

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

5.2 Fuel Storage

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

5.3 Reactor Room

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

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

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

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

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

6.1.1 University Officer

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

6.1.2 Dean, ~~School~~ College of Engineering

The Dean of the ~~School~~ College of Engineering is the administrative officer responsible for the operation of the ~~School~~ College of Engineering. ~~(also Reactor Administrator.)~~

6.1.3 Reactor Administrator

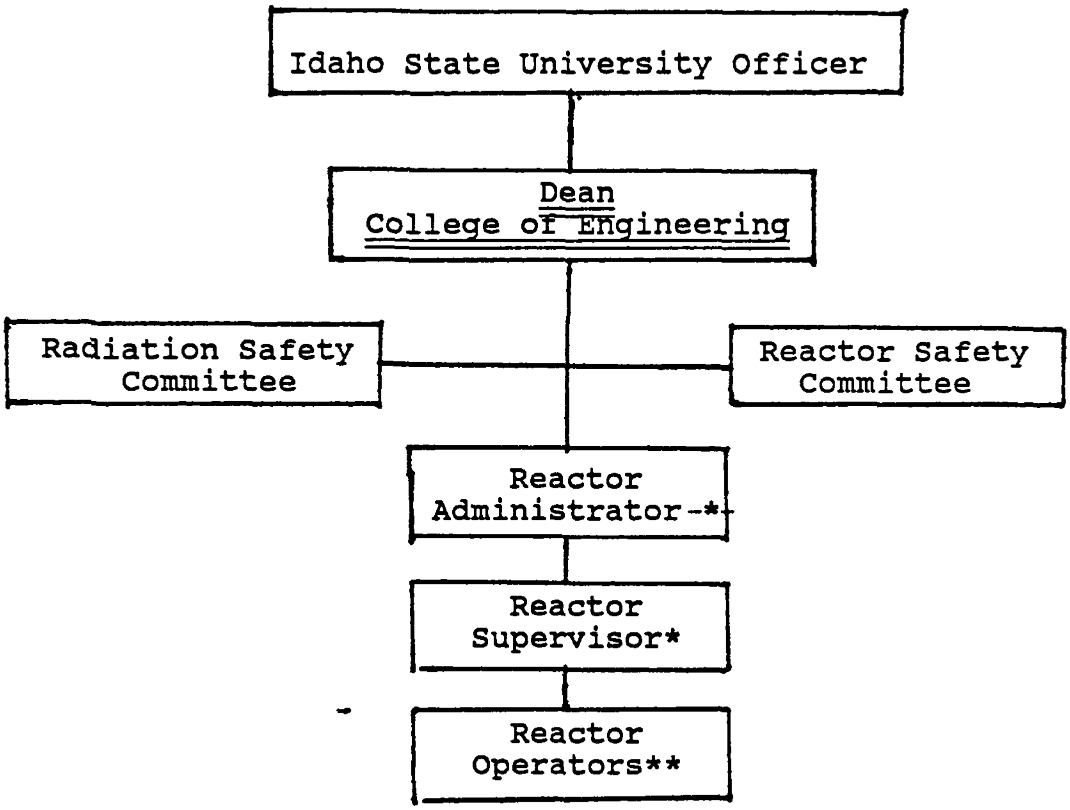
The Reactor Administrator is the administrative officer responsible for the operation of the AGN-201M Reactor Facility. In this capacity he shall have final authority and ultimate responsibility for the operation, maintenance, and safety of the reactor facility within the limitations set forth in the 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 Radiation Safety Committee and/or the Reactor Safety Committee in all matters concerning unresolved safety questions, new experiments and new procedures, and facility modifications which might affect safety. He shall be an ex officio member of the Reactor Safety Committee.

6.1.4 Reactor Supervisor

The Reactor Supervisor shall be responsible for the preparation, promulgation, and enforcement of administrative controls including all rules, regulations, instructions, and operating procedures to ensure that the reactor facility is operated in a safe, competent, and authorized manner at all times. He shall direct the activities of operators and technicians in the daily operation and maintenance of the reactor; schedule reactor operations and maintenance; be responsible for the preparation, authentication, and storage of all prescribed logs and operating records; authorize all experiments, procedures, and changes thereto which have received the approval of the Reactor Safety Committee and/or the Radiation Safety Committee and the Reactor Administrator; and be responsible for the preparation of experimental procedures involving use of the reactor.

Figure 1

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



*   Requires NRC Senior Operators License

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

Persons holding positions on the Administrative Organization shall meet or exceed the qualification requirements of ANSI/ANS -15.4-1977 (N380), "Selection and Training of Personnel for Research Reactors."

6.1.5 Reactor Operators

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

6.1.6 Reactor Safety Committee

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

6.1.7 Radiation Safety Committee

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

6.1.8 Radiation Safety Officer

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

6.1.9 Operating Staff

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

6.2 Staff Qualifications

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

6.3 Training

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

6.4 Reactor Safety Committee

6.4.1 Meetings and Quorum

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

6.4.2 Reviews

The Reactor Safety Committee shall review:

a. Safety evaluations for changes to procedures, equipment or systems, and tests or experiments, conducted without Nuclear Regulatory Commission approval under the provision of 10 CFR 50 paragraph 50.59, to verify that such actions 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 previously approved tests or experiments, or those that involve an unreviewed safety question as defined in Section 50.59, 10 CFR 50 paragraph 50.59.

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

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

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

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

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

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

6.4.4 Authority

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

6.4.5 Minutes of the Reactor Safety Committee

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

6.5 Approvals

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

a. The Reactor Supervisor shall prepare the proposal for review and approval by the ~~Dean of the School of Engineering~~ Reactor Administrator.

b. The ~~Dean of the School of Engineering~~ Reactor Administrator shall submit the proposal ~~to the Chairman of the Reactor~~ Safety Committee.

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

d. The Reactor Safety Committee 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 irradiations and experiments that could affect the safety of the reactor.

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

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

f. Implementation of the Security Plan and Emergency Plan.

The above listed procedures shall be approved by the ~~Dean of the School of Engineering~~ Reactor Administrator and the Reactor Safety Committee. 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. ~~or Dean of the School of Engineering.~~

6.7 Experiments

a. Prior to initiating any new reactor experiment an experimental procedure shall be prepared by the Reactor Supervisor and reviewed and approved by the ~~Dean of the School of Engineering~~ Reactor Administrator and the Reactor Safety Committee.

b. Approved experiments shall only be performed under the cognizance of the ~~Dean of the School of 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 Committee not later than the next working day.

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

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

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

The annual operating reports made by licensees shall 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 relating to reactor safety that occurred during the reporting period.

b. Results of major surveillance tests and inspections.

(2) A monthly tabulations 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 change to the Technical Specifications 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. Liquid waste-

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

b. Airborne waste-

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

c. Solid waste-

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

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

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

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

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

6.9.2 Reportable Occurrences

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

a. Prompt Notification with Written Followup-

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

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

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

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

(4) Reactivity balance anomalies involving:

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

b. exceeding excess reactivity limits;

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

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

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

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

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

6.10 <u>Record Retention</u>

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 requirements may be satisfied by the normal operations log book plus:

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

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

g. Changes to operating procedures.

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

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

b. Appropriate offsite 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 Committee.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 TO

FACILITY OPERATING LICENSE NO. R-110

IDAHO STATE UNIVERSITY

DOCKET NO. 50-284

1.0 INTRODUCTION

By letter dated October 28, 1988, and supplemented by letters dated February 15 and April 14, 1989, the Idaho State University (licensee) requested an amendment to their license which would revise certain Technical Specifications (TS). The TS changes requested are mostly administrative in nature, which reflect either typographical errors, the addition of an approved radiography experiment, the addition of shielding when operating at power levels above 0.5 watts, the removal of a reference to an accelerator room and changes with regard to organization and titles.

2.0 EVALUATION

All changes have been identified by vertical lines in the margin of the TS. The changes made to correct typographical errors are on pages 6 and 10. On page 12, Section 3.4.c.2 has been revised to not require that the thermal column be filled with water whenever the approved neutron radiography collimator is used, since the collimator displaces some of the water in the thermal column. The licensee, in the letters of February 15, 1989 and April 14, 1989, submitted data to show radiation levels in representative locations of the reactor laboratory with and without the collimator installed. These radiation levels were taken at 5 watts and with the movable shield door above the thermal column in a closed position. The results show that in some part of the laboratory there is no rise in the radiation level with the collimator in or out. For instance, there is no change at the operator location. At other locations there is a small rise in the radiation level with the collimator in place. However, the increased radiation levels will not preclude operation in these locations but instead will reduce the amount of time an experimenter can spend in these areas with the collimator installed.

The licensee has also proposed to change the basis for Section 3.4 of the TS in light of the data provided in the letters of February 15 and April 14, 1989. The previous basis said that "radiation surveys ....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." The data provided in the licensee's submittal of April 14, 1989 justify a revision to the basis to read "radiation surveys.... in the reactor room, at the closest approach to the reactor outside the designated high radiation area is less than 100 mrem/hr at power levels less than 5.0 watts." The data taken by the licensee justifies a value less than

100 mrem/hr at 5 watts as a basis for TS Section 3.4. Licensee dose measurements taken at various locations within the reactor room (see licensee letter of April 14, 1989) show that the highest measured radiation level outside the designated high radiation area is 14.5 mrem/hr (combined gamma and neutron) at the experimenter 4 position. Therefore, the licensee (A. E. Wilson), in a telecon of April 28, 1989 with T. S. Michaels, agreed to reduce the dose rate level in the basis to 25 mrem/hr. This is acceptable since this level is more in line with the maximum expected dose rates in the reactor room.

On page 12, Section 3.4.c.3, a requirement has been added that the movable shield doors above the thermal column should be maintained in a closed position whenever the reactor is operated at a power greater than 0.5 watts. The licensee has been following this procedure and is now proposing that it become a TS requirement. The licensee has provided data in the April 14, 1989 letter which shows that a considerable reduction is obtained in the high radiation area when the shield doors are closed. This reduction is significant at a power level of 5.0 watts and is part of the licensee's As Low as Reasonably Achievable Program to reduce exposure.

On page 12, the first paragraph of the basis has been changed to eliminate the reference to dose rates in the accelerator room since the reactor was moved in 1976 from the Physical Science Building, where the accelerator was located, to the Lillibridge Engineering Laboratory Building.

On pages 19, 20, 21, 25, and 26 changes to the Organization have been made to reflect current status. These changes are acceptable and where applicable conform to ANSI/ANS-15.4-1977.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements and in the category of recordkeeping, reporting, or administrative procedures or requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: Theodore S. Michaels

Dated: May 17, 1989