TEXAS A&M AGN-201M
RESEARCH REACTOR
LICENSE NO. R-23
DOCKET NO. 50-59

REVISED SAFETY ANALYSIS REPORT, TECHNICAL SPECIFICATIONS, OPERATOR REQUALIFICATION PROGRAM, AND FINANCIAL QUALIFICATIONS

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

DWIGHT LOOK COLLEGE OF ENGINEERING

Department of Nuclear Engineering

TEXAS A&M UNIVERSITY

June 30, 2011

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Docket No. 50-059

SUBJECT: Response to "Texas A&M University – Follow –up on request for additional information regarding the Texas A&M University AGN-201M reactor (TAC. NO. ME1588), October 9, 2009"

In response to the RAI dated October 9, 2009, the following items are enclosed:

- Attachment A - Updated Appendix for A, License No. R-23
- Attachment B – Operator Licensing Requalification Plan
- Attachment C – Safety Analysis Report
- Attachment D – Response to RAI Item C. Financial Qualifications

If you have any questions, please do not hesitate to contact me at: (979) 845-4161, or e-mail at rjuzaitis@tamu.edu.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 30, 2011.

Sincerely,

Raymond J. Juzaitis, Ph.D.
Department Head

cc: Dr. Emile Schweikert
Mr. Chris Crouch

Attachments

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A020
NRR

Attachment A

Updated Appendix for A
License No. R-23

Appendix for A
License No. R-23

Technical Specifications

Texas A&M University AGN-201M (Serial Number 106)
Docket Number 50-59

Date: June 1, 2011

Amendment No. 12 to NRC License R-23

TECHNICAL SPECIFICATIONS
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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 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 **Coarse Control Rod** – The control rod with a scram function that can be mechanically withdrawn/inserted at two possible speeds.

1.5 **Excess Reactivity** – The amount of reactivity above a $k_{eff} = 1$. This is the amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$)

1.6 **Experiment** – An experiment is any of the following

a. An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;

b. An evaluation or test of a reactor system; operational, surveillance, or maintenance technique;

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

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

1.8 **Explosive Material** – Explosive material is any solid or liquid which is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its 704 Diamond, Hazard Rating System.

1.9 **Fine Control Rod** – A low worth control rod (about 25% of the worth of the other control rods) used primarily to maintain an intended power level. Its position may be varied manually. The fine control rod does not drop on a scram signal.

1.10 **Measuring Channel** – A measuring channel is the combination of sensor, lines, cables, amplifiers, and output devices which are connected for the purpose of measuring or responding to the value of a process variable.

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

1.12 **Operable** – Operable means a component or system is capable of performing its intended function in its normal manner.

1.13 **Operating** – Operating means a component or system is performing its intended function in its normal manner.

1.14 **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.

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

1.16 **Reactor Operation** – Reactor operation is any condition wherein the reactor is not shutdown or secured.

1.17 **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.

1.18 **Reactor Secured** – The reactor shall be considered secured when the following conditions exist:

(a) All control rods removed from the core;

(b) The console key switch is in the off position, and the key is removed from the lock;

(c) No work in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;

(d) No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding 0.65% $\Delta k/k$ referenced to 20°C.

1.19 **Reactor Shutdown** – The reactor shall be considered shutdown if it is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments included and the following conditions exist:

(a) No work in progress involving core fuel , core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;
(b) No experiments being moved or serviced that have, on movement, a reactivity worth exceeding 0.65% $\Delta k/k$ referenced to 20°C.

1.20 **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.

1.21 **Safety Channel** – A safety channel is a measuring channel in the reactor safety system.

1.22 **Safety Control Rod** – One of two scrammable control rods that can be mechanically withdrawn/inserted at only one speed.

1.23 **Scram Time** – The time for the control rods acting under gravity to change the reactor from a critical to a subcritical condition. In most cases, this is less than or equal to the time it takes for the rod to fall from full-in to full-out position.

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

1.25 **Shall, Should and May** – The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission – neither a requirement nor a recommendation.

1.26 **Shutdown Margin** – Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition with the most reactive rod in its most reactive condition, and the fine control rod full out, and that the reactor will remain subcritical without further operator action.

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

1.28 **Unsecured Experiment** – Any experiment, or component of an experiment is deemed to be unsecured whenever it is nor secured as defined in 1.24 above. Moving parts of experiments are deemed to be unsecured when they are in motion.

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.

Specifications

a. 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 would be below 200° C thus assuring integrity of the core and retention of fission products.

2.2 Limiting Safety

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	≤ 10 watts
Nuclear Safety #3	High Power	≤ 10 watts

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

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 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 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 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 two safety rods, course control rod and fine control rod shall all be operable when not in their fully withdrawn position.

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

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

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

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

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

e. Nuclear safety channel instrumentation shall be operable in accordance with the Table 3.1 whenever the reactor control or safety rods are not at their fully withdrawn position.

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

g. 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.

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

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

j. A manual scram shall be provided on the reactor console.

Table 3.1

Safety Channel	Set Point	Function
Nuclear Safety #1		
Low count rate	≥ 10 cps	scram below 10 cps
Nuclear Safety #2		
High Power	≤ 10 watt	scram at power > 10 watts
Low Power	$\geq 1.0 * 10^{-12}$ amps	scram at source levels $< 1.0 * 10^{-12}$ 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	$\geq 1.0 * 10^{-12}$ amps	scram at source levels $< 1.0 * 10^{-12}$
Manual scram	-----------------	scram at operator option

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 levels low enough to assure safe shutdown without exceeding any safety limits. The period scram conservatively limits the rate of rise of reactor power to periods which are manually controllable and will automatically scram the reactor in the event of unexpected large reactivity additions.

The AGN-201M'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 6cm 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 shutdown 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 experiment 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 liquid or gaseous, fissionable materials shall be doubly encapsulated.

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

c. The radioactive material content, including fission products of any experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components from the experiment will not result in: (1) a total effective dose equivalent to any person occupying an unrestricted area in excess of 0.1 rem or (2) a total effective dose equivalent to any person occupying a restricted area during the length of time required to evacuate the restricted area in excess of 5 rem.

Bases

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

3.4 Radiation Monitoring, Control, and Shielding

Applicability

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

Objective

The objective is 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 (when the shield plug is removed) 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 with the shield plug removed.

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 and the reactor 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.

<u>Bases</u>

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.

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.

- biennial (interval not to exceed 2.5 years);
- annual (interval not to exceed 15 months);
- semiannual (interval not to exceed 7.5 months);
- quarterly (interval not to exceed 4 months);
- monthly (interval not to exceed 6 weeks);
- weekly (interval not to exceed 10 days);
- daily (must be done during the calendar day)

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 15 months.

b. Total excess reactivity and shutdown margin shall be determined annually, but at intervals not to exceed 15 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 15-month period.

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 15 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 systems.

g. The period, count rate, and power level measuring channels shall be calibrated and set points verified annually, but at intervals not to exceed 15 months.

h. The shield water level interlock and shield water temperature interlock shall be calibrated by perturbing the sensing element to the appropriate set point. These calibrations shall be performed annually, but at intervals not to exceed 15 months.

Bases

The channel tests and checks required daily or before each startup will assure that the safety channels 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

The objective is to assure integrity of the reactor structures.

Specification

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

b. Visual inspection for water leakage from the shield tank shall be performed 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 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 Radiological Safety annually, but at intervals not to exceed 15 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 (Ref. 3.4e) shall be verified to be operable.

c. A radiation survey of the reactor room, reactor control room, and accelerator room shall be performed under the supervision of Radiological Safety annually, but at intervals not to exceed 15 months, to determine the location of radiation and high radiation areas corresponding to reactor operating power levels.

Bases

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

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 Room and Accelerator Room

Applicability

This specification applies to the physical orientation of the facility.

Objective

To identify the location of the facility and other associated equipment.

Specification

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

b. The accelerator room is directly above the reactor room and a hole in the accelerator room floor provides access to the thermal column.

c. The reactor room and accelerator room are separate rooms in the Zachary Engineering Center, 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.

d. The access doors to the reactor room and accelerator room shall contain locks.

Bases

The reactor room and accelerator room provide a secure, controlled access area with appropriate shielding for personnel radiation protection.

5.2 Reactor

Applicability

This specification applies to the physical characteristics of the reactor.

Objective

To identify the physical characteristics of the reactor.

Specification

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.

NOTE: All dimensions, masses, and densities given in the above description are nominal values.

Bases

These basic design criteria are relevant to the safe operation of the reactor and should not be changed or modified without NRC approval.

5.3 Fuel Storage

Applicability
This specification applies to fuel storage.

Objective

To assure fuel is stored in a subcritical array.

Specification

Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in locked rooms in the Department of Nuclear Engineering laboratories. The storage array shall be such that K_{eff} if is not greater than 0.8 for all conditions of moderation and reflection.

Bases

The limits imposed are conservative and assure safe storage (NUREG-1537).

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. 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 President

The President (Level 1) 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 (Level 1) 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 (Level 2) 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 Committee 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 ex officio member of the Reactor Safety Board.

6.1.4 Reactor Supervisor

The Reactor Supervisor (Level 3) shall be a licensed SRO and shall be responsible for the preparation, promulgation, and enforcement of administration controls including all rules, regulation, 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. He shall also be an ex officio member of the Reactor Safety Board.

6.1.5 Reactor Operators

Reactor Operators (Level 4) 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 50 paragraph 55.13 shall only operate the reactor under the direct 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 affects its safe operation; determining whether proposed experiments, procedures, or modifications meet the requirements of 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 Head of the Department of Nuclear Engineering.

6.1.7 Radiological Safety Committee

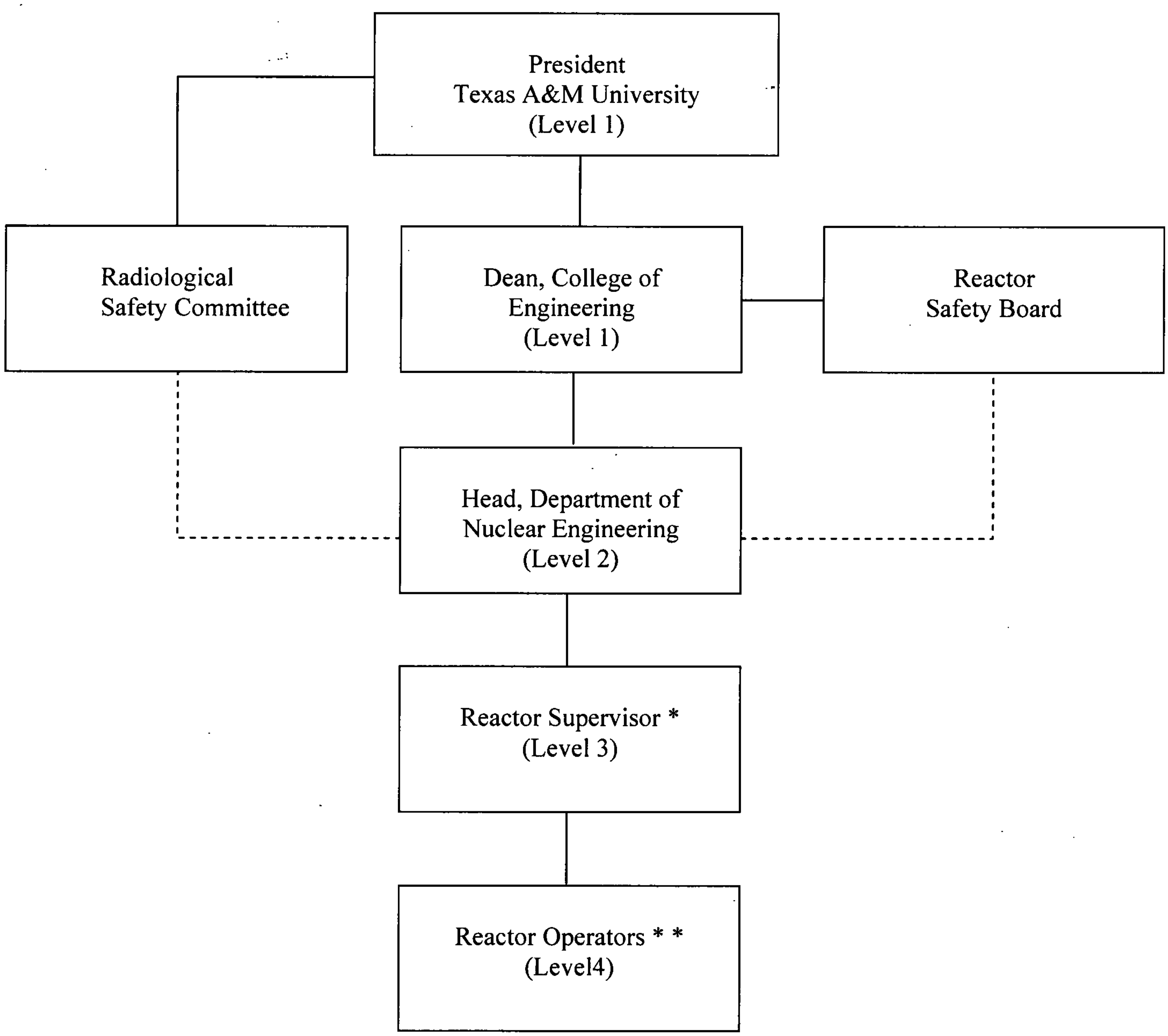
The Radiological Safety Committee shall advise the University administration and the Radiological Safety Officer on all matters concerning radiological safety at university facilities.

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 of the Reactor Safety Board.

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 55.13

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 at the reactor control console.
   2. One other person in the reactor room certified by the Reactor Supervisor as qualified to activate a 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.

c. A listing of reactor facility personnel by name and phone number shall be conspicuously posted in the reactor control console area.

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 ANSI/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 biology 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 ANSI/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 paragraph 50.59 to verify that such actions do not constitute a license amendment.

b. Proposed changes to procedure, equipment or systems that change the original intent or use, and are non-conservative, or those that do not meet the criteria set forth in 10 CFR 50 paragraph 50.59 (c).

c. Proposed tests or experiments which are significantly different from previously approved tests or experiments, or those that do not meet the criteria set forth in 10 CFR 50 paragraph 50.59 (c).

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statures, 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. Deficiencies uncovered that affect reactor safety shall immediately be reported to Level 1 management. A written report of the findings of the audit shall be submitted to Level 1 management and the review and audit group members within 3 months after the audit has been completed. 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.

c. The Facility Security Plan and implementing procedure, at least every two years.

d. The Requalification Program and records, at least every two years.

e. Results of actions taken to correct deficiencies, at least annually.

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 Nuclear Reactor Safety Board.

c. The Chairman of the Reactor Safety Board shall submit the proposal to the Reactor Safety Board members for the 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 irradiations and experiments that could affect the operation of 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.

g. Radiation Safety Program

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 the criteria set forth in 10 CFR 50 paragraph 50.59 (c)may be employed on approval by the Reactor Supervisor.

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 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 NRC Operations Center, the Director of NRR, the Reactor Safety Board, the Head of the Department of Nuclear Engineering, and the Reactor Supervisornot 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 actions 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 the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Document Control Desk, USNRC, Washington D.C., 20555.

6.9.1 Annual Operating Report

Routine annual operating reports shall be submitted no later than ninety (90) days following May 31. 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 for the shutdowns and corrective actions 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, paragraph 50.59 which clearly shows the reason leading to the conclusion that the criteria set forth in 10 CFR 50 paragraph 50.59 (c) was followed 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. Liquid Waste (summarized for each release)

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

   b. Solid Waste (summarized for each release)

      (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 received by facility personnel and visitors where such exposures are >25% of 10CFR20 limits..

6.9.2 Reportable Occurrences

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

Prompt Notification With Written Follow-up

The types of events listed below shall be reported as expeditiously as possible by telephone and confirmed in writing by facsimile or similar conveyance to the NRC Operations Center no later than the first work day following the event, with a written follow-up 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 has reached the set point 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% $\Delta k/k$.

    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% $\Delta k/k$ or any unplanned criticality.

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

6. Personnel error or procedural inadequacy which prevents, or could prevent, by itself, the fulfillment of the functional requirements of system 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 I 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. Release of radiation or radioactive materials from the facility above allowed limits.

10. Performance of structure, 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, Office of Nuclear Reactor Regulation, USNRC within the time period specified for each report. This includes personnel changes in Level 1 (University President, Dean of the College of Engineering) or 2 (Head of the Nuclear Engineering Department) administration, as shown in Figure 1, which shall be reported within 30 days of such a change.

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.

Attachment B

Operator Licensing Requalification Plan

REQUALIFICATION PROGRAM FOR LICENSED REACTOR OPERATORS

AND SENIOR REACTOR OPERATORS

TEXAS A&M UNIVERSITY

AGN-201M REACTOR LICENSE R-23

Approved: 26 May 2011

AGN-201 BIENNIAL REQUALIFICATION PROGRAM

A. Each licensed operator and senior operator will be required to operate the reactor or otherwise perform the duties as a licensed operator or senior operator for a minimum of four hours per calendar quarter. Each licensed operator must perform the following control manipulations or demonstrate knowledge of actions to be taken, as appropriate, annually: reactor startup, reactor shutdown, reactor power change of 10 percent or more, and a loss of electrical power. Each licensed operator must demonstrate knowledge of actions to be taken during the following transients biennially: inability to drive control rods, reactor trip, and nuclear instrumentation failure. Senior operators may be credited with these activities if they direct control manipulations as they are performed.

B. Each licensed operator and senior operator will be required to take an annual operating examination to establish the operators abilities in the following categories: manipulations of console control rods; identification of alarms and any remedial action required; identify instrumentation systems and their significance; observe and control the operating characteristics of the facility; perform controlled manipulations during normal, abnormal and emergency situations; describe the use of the facility's radiation monitoring systems; demonstrate a knowledge of radiation hazards; demonstrate knowledge of the facility emergency plan; demonstrate ability to perform the function of an operator in the emergency plan; and demonstrate ability to perform the operator's function as a member of the control room team.

C. Each licensed operator and senior operator will be required to take a written examination over the elements listed in 10 CFR 55.41 and 55.43 respectively. Any licensed scoring less than 70 percent on the annual written examination will be removed from their licensed duties until they have corrected the deficiency. They will not be restored to duty until they have retaken the exam and score a grade of at least 80 percent in all categories. Any licensed person scoring less than 80 percent in any category on the annual written exam must be tutored to bring their performance up to a higher level.

D. Each licensed operator and senior operator shall participate in an accelerated requalification program where performance evaluations pursuant to A, B or C above clearly indicate the need. A licensed individual preparing and grading any portion of the written examination is exempt from taking the written examination in that category.

E. All operators and senior operators must have a physical examination performed once every two years. The results of the physical examinations will be maintained in the operator's training file.

F. Individuals who maintain operator or senior operator licenses for the purpose of providing backup capability to the operating staff participate in this requalification program except to the extent of their normal duties preclude the need for retraining in particular areas. All licensed operators who do not participate in the Requalification Program, because of other duties, will have their names removed from the list of authorized operators. Operators will then be required to satisfactorily complete an accelerated Requalification Program in order to be listed on the authorized operator's list. The accelerated program will include at least six hours of experience with the reactor. Prior to placing an operator back on the authorized list, the NRC will be notified by letter indicating an operator has satisfactorily completed a retaining program in accordance with 10 CFR 55.

G. A training file shall be maintained on each licensed operator and senior operator to document their participation in the Requalification Program. Each file shall include copies of written examinations administered, the answer given by the licensees, results of evaluations, documentation of any additional or accelerated training administered in areas in which an operator or senior operator has exhibited deficiencies, and a copy of the reactor operating log documenting the required reactivity manipulations. The Nuclear Engineering Department Head shall sign documentation of any accelerated training.

Attachment C

Safety Analysis Report

Safety Analysis Report

Facility License R-23
Docket Number 50-59

AGN-201M (Serial Number 106)

Department of Nuclear Engineering
Texas A&M University

SAFETY ANALYSIS REPORT
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Chapter 1: THE FACILITY

1.1 Introduction

This report is intended to be a part of the application to the United States Nuclear Regulatory Commission (USNRC) by Texas A&M University (TAMU) at College Station, Texas for the renewal of the class 104, Facility License R-23, Docket # 50-59 for the University's AGN-201M research reactor. The reactor is primarily used for the teaching of classes in the Department of Nuclear Engineering and for the training of reactor operators. The University has operated this reactor for a period exceeding fifty years and in its present location, the Zachry Engineering Center room 61B for nearly thirty-five years.

This application contains information on the local communities of Bryan and College Station as well as the reactor facility located inside the Zachry Engineering Center. The local area information includes: meteorology, geology, seismology, and other pertinent data required by NUREG-1537 Part 1.

This reactor has been rated for a maximum power level of 5 watts thermal since 1972. The AGN-201M reactor design has several inherent safety features. The most significant of these are the negative temperature coefficient and the design limitation of a maximum of 0.65% excess reactivity above delayed critical. These designed safety features along with the core thermal fuse and the trips and interlocks associated with the nuclear instrumentation safety systems provide for protection of the reactor fuel even under the worst case accident conditions. If a sudden insertion of positive reactivity were to occur, the reactor power rise would be attenuated by the negative temperature coefficient and then terminated by the separation of the reactor core due to the melting of the core thermal fuse. Other important reactor design considerations include: the type of fuel used, the core can, , and sealed fueled control rods. These features provide a complete barrier between the reactor fuel and the environment. While the core tank allows for thermal dispensation the heat generated due to the fissioning of the fuel; because of this, no outside secondary cooling system is necessary. A feature of the TAMU reactor room is the thickness of the walls and ceiling. These components which act as a boundary to the reactor facility are constructed of concrete. The walls are 3.5 feet thick and the ceiling is 4 feet thick. The AGN reactor design has proven to be a safe and effective training device for thousands of future engineers at this as well as other universities. The system has an established data and collaborated data base, so no research and development activities were required to evaluate the system.

1.2 Summary and Conclusions on Principal Safety Considerations

The safety criterion associated with the AGN-201M reactor at Texas A&M University is to limit as much as possible any adverse effect that reactor operations may have on the general public and students within the reactor facility.

It is believed that no adverse consequences have resulted from many years of reactor operations in this current facility. Radiation surveys conducted at 5 watts indicate very low dose rates outside the reactor room and none outside the areas controlled by the Department of Nuclear Engineering.

A complete description of the AGN-201M reactor is presented in Chapter 4 of this report. All the design bases for the AGN-201M reactor system and components are presented in Chapter 14 of this report.

The reactor facility in the Zachry Engineering Center is constructed in such a way that the reactor room and accelerator laboratory are in a cube. This cube has 3.5 foot thick reinforced concrete walls and 4 foot thick ceilings. This area of the building is isolated from the general public, as well as most personnel other than nuclear engineering students that have laboratories or offices in the area. Because of this design and the design of the reactor and its components, the reactor will remain in a safe, isolated condition even under the most severe hypothetical accident.

This reactor facility has been in its present location since the early 1970's and a renewal license was issued by the USNRC in August 1977, for this reactor in its present location. Some small changes have been made to the reactor facility in this time period.

1.3 General Description of the Facility

The AGN-201M reactor facility is located on the main campus of Texas A&M University in the Zachry Engineering Center. The location of the Center on campus is presented in Figure 1.3-1. The facility is located in areas under the direct control of the Department of Nuclear Engineering on the 1st and ground floors of the center with the reactor room below unique the ground floor in a specially constructed reactor room. The floor plans for the facility are presented in Figures 1.3-2 and 1.3-3.

The location of the facility in the Center allows for complete isolation of the reactor room, Rm. 61B from the rest of the Engineering Center. The AGN-201M reactor is designed to be operated in populated areas without undue risk to the general public. This facility produces no effluents to the environment and is located in a room with thick reinforced concrete walls and below ground. The AGN-201M reactor system has been in service for decades and the University has possessed an operating license since 1957.

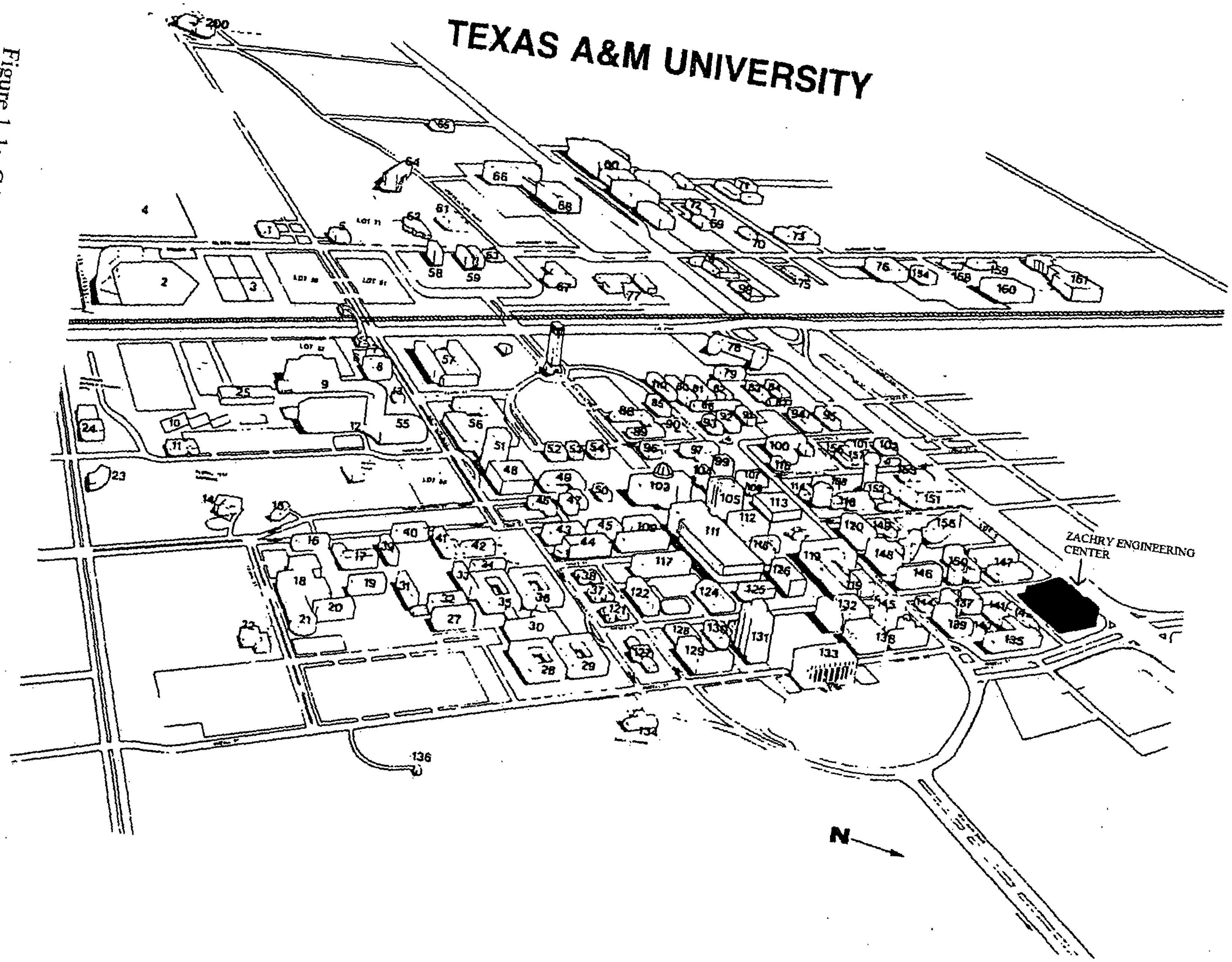

Figure 1-1: CAMPUS MAP OF TEXAS A&M UNIVERSITY

[redacted]

Figure 1-2: REACTOR FACILITY FIRST FLOOR

[redacted]

Figure 1-3: REACTOR FACILITY GROUND FLOOR

Specific site characteristics of the facility are addressed in Chapter 2 of this report. The site is in an area with a very low threat of earthquakes or flooding, and the facility is designed to be fail safe. The reactor system has many safety features associated with it. Some are passive systems while others are designed into the reactor protective systems. The reactor systems and safety features are described in greater detail in other chapters of this report.

A unique feature of this reactor facility is the design of the reactor room itself. The control console is shielded by an L-shaped wall to lower the dose to the operators and students. The steel reinforced concrete walls and ceiling also provide additional shielding to the general public. This greatly enhances the ability of the operating staff to operate reactor with confidence that no member of the general public is being negatively affected.

1.4 Shared Facilities and Equipment

The only facilities that this reactor facility shares with the surrounding building are the electrical power and air supply systems. The reactor room has a ventilation unit inside, and the discharge from the reactor room passes through a grate in the ceiling and into an exit filter located inside the accelerator room. A separate fan on the roof provides suction on this filter unit and air is drawn out of the reactor room and discharged on the roof. Power to the AGN-201M reactor console is supplied from the building power supply, but power can be remotely isolated by operation of two large breakers outside of room 60C. Inside the reactor room is a large graphite pile and a subcritical assembly.

1.5 Comparisons to Similar Facilities

Two other AGN- 201M research reactors are currently licensed by the USNRC. These are both university reactors and have a similar training and teaching role as does the AGN-201M at Texas A&M University. Although, the roles are somewhat different since TAMU has a much larger TRIGA reactor facility on the campus. This reactor is used for irradiation of materials for medical and commercial use while the AGN-201M is used as a teaching and training device only.

1.6 Summary of Operations

The AGN-201M operations have been nonexistent over the last 10 years due to upgrading the reactor control console. This shifted teaching responsibilities to the TRIGA reactor facility. Now that the control console upgrade has been completed and the subsequent testing and restart have been completed, the AGN-201M will resume previous teaching uses. The AGN-201M is being used for student reactor startups and limited experiments in two different nuclear engineering classes. The reactor has again become an important learning device for the students. With the presence of two reactors at one university, the students get the unique opportunity to work with and operate both reactors. The reactor should have critical run hours of 60-80 hours per year at an average power level of 0.75 - 1.0 watts. As part of the ALARA program, the reactor operated at power levels which are as low as possible to achieve the desired results in order to keep doses to the student as low as possible.

In the future the AGN-201M will become more integrated into the teaching curriculum and see greater use over the semester. The reactor is currently used throughout the fall and spring semesters for both graduate and undergraduate reactor laboratory classes.

1.7 Compliance With the Nuclear Waste Policy Act of 1982

Texas A&M University is in compliance with the Waste Policy Act. The University has entered into an agreement with Idaho National Engineering Laboratory for the final, no-cost disposal of the AGN-201M reactor fuel.

1.8 Facility Modifications and History

This AGN-201M reactor has been licensed by Texas A&M University since 1957. The initial maximum rated power was 100 mW but this was changed in 1973 to the present 5 watt thermal power limit. The current reactor room was constructed in 1970 under construction permit CPRR-112 and the reactor moved to its current location in 1972.

In September of 1984, the reactor control console was moved from room 60C to its present location in the reactor room, 61B, behind the installed shielding. This was done so that the operator could be in the same room as the reactor during experiments. At the same time another modification was made to add a series of concrete blocks around the base of the reactor to lower general area radiation levels. Current radiation surveys indicate that the reactor produces very low levels of radiation outside the reactor room and in the accelerator room.

During the period from August 1999 until June 2010 the reactor console underwent an instrumentation and electronics upgrade. This upgrade installed a new digital control console. The reactor control console was designed, tested, and installed by Thomas H. Fisher. This upgrade improves the human-machine interface, as well as, better equipping students for commercial nuclear power plants.

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Chapter 2: SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Specification and Location

The AGN-201M reactor facility is physically located a[redacted]. The facility is on the main campus of Texas A&M University, in College Station, Texas in the county of Brazos. The facility is located on the first and ground floors of the Zachry Engineering Center and the reactor room is on the ground floor in room 61 B.A map, Figure 2.1.1.1-2 is provided of the main campus of the University with the Zachry Engineering Building clearly marked.

2.1.1.2 Boundary and Zone Area Maps

The reactor room and accelerator room areas are easily seen on Figures 2.1.1.2-1 and 2.1.1.2-2, the dark outer line indicates the boundary of the facility. The campus of Texas A&M University is bordered by Texas Highways Business 6, Texas Farm to Market Road 60, Texas Farm to Market Road 2818, and George Bush Dr. No emergency planning zones are needed for this facility due to its size and design.

2.1.2 Population Distribution

It can be seen from the population density on Figure 2.1.1.1-1A&B that the population density remains constant until you leave the towns of Bryan or College Station and it falls to near zero. The 2000 census recorded the population of the Bryan and College

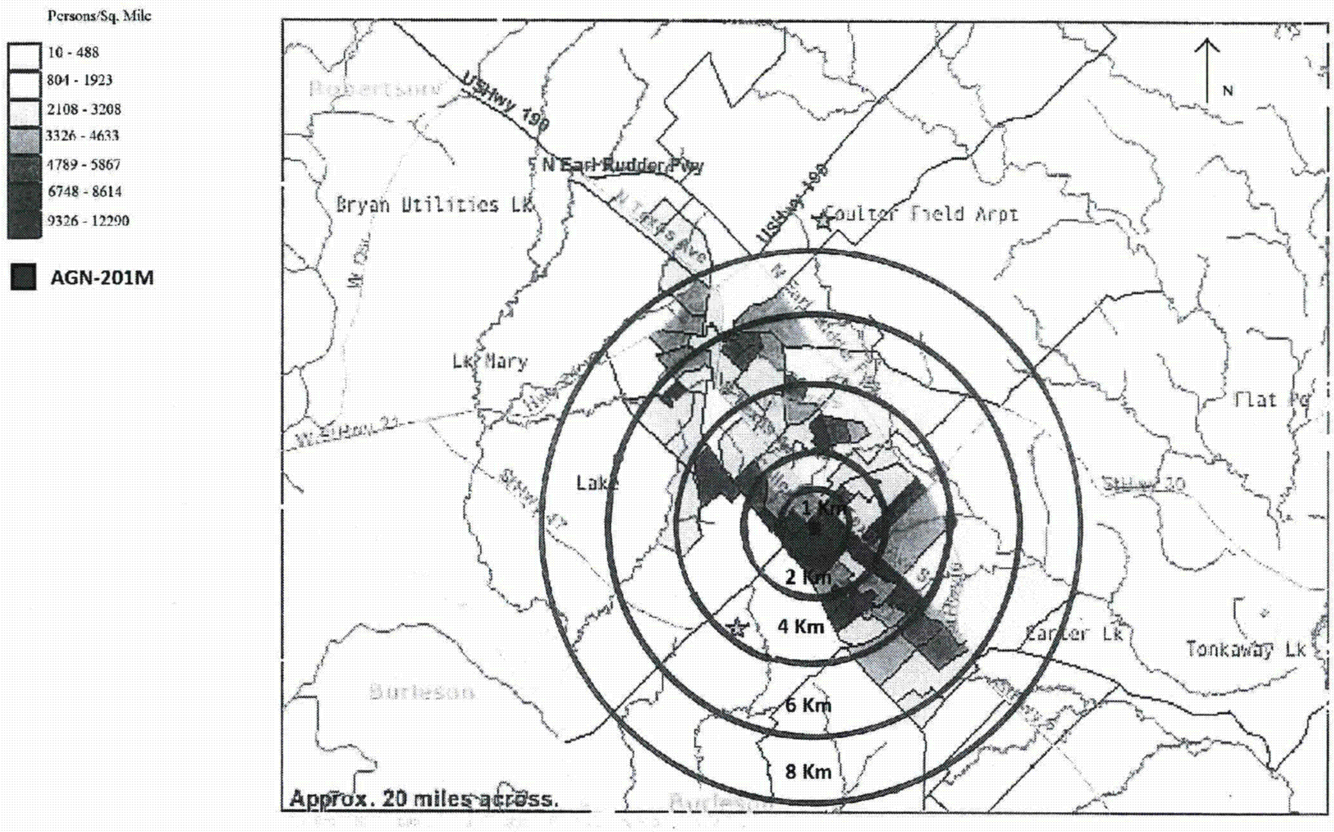

Figure 2-1: POPULATION DENSITY

Station area at 184,885 residents, many of which are associated with the University. The 2015 population projection for the Bryan/College Station area is 224,325. Due to the reactor being located on the main campus, up to 55,000 students and employees could be on campus during reactor operations. Since the reactor has no effluents and has been proven safe under the most serve conditions this does not pose a safety hazard.

2.2 Nearby Industrial, Transportation and Military Facilities

Any potential accident associated with the AGN-201M reactor will have no effect upon the environment nor the surrounding area. No military installations are close the Bryan/College Station area, and industrial development is very limited within the 8 kilometer limit, since most of that area is still part of Texas A&M University or a residential area.

2.2.1 Locations and Routes

The only major route of concern is the railroad that cuts across campus, 1.8 kilometers southwest from the facility. Texas Highway Business 6 is located one kilometer northeast of the facility. But in both cases the facility is located on the far side of the most direct point from the routes.

2.2.2 Air Traffic

The reactor facility is located 6.5 kilometers from a small commercial airport. The flight path for this airport is some 1.0 kilometers to the northwest of the facility.
Since the facility is located on the first floor and the reactor room is in a steel reinforced concrete room, the danger posed by passing planes is very small.

2.2.3 Analysis of Potential Accidents at Facilities

No heavy industry or chemical plants are located within 20 kilometers of the reactor facility. The worst possible accident would involve trains that use the railroad tracks through the middle of campus. If an accident occurred that would require an evacuation of the facility and the surrounding building, it is believed that the reactor facility would be in no real danger. If a large explosion occurred the shockwave would tear apart many other structures before getting to the Zachry Engineering Center. The subsequent danger to the reactor room and accelerator room is limited because of its location inside of the steel reinforced concrete cube. Even in the event of a highway accident the reactor is well protected from potential harm.

2.3 Meteorology

The Bryan-College Station area is located approximately 100 miles inland from the Texas Gulf Coast. The local weather is determined to a great extent by the high pressure areas which are predominant over the Gulf of Mexico. As a direct result of this condition, warm southeasterly winds are present a large majority of the time. It should be noted that in the winter months the prevailing winds come out of the north. The average wind speed is 7.34 knots. Hurricanes are very infrequent in this area do to the distance from the Texas Gulf Coast. In the last 75 years only two hurricanes have caused any damage in this area and that was due to the thunderstorms and gusty winds associated with the remnants of the system.

Tornadoes are fairly common in Texas, however; they are not very common in this region of the state. Data collected on tornado frequency between 1950 and 1976 indicates that 17 tornadoes were reported within 25 nautical miles radius of College Station.

2.4 Hydrology

The nearest large independent free flowing water is the Brazos river located about 10 kilometers from the facility. Several small lakes and streams are located within 3 kilometers but pose no hazard to the facility. The facility is located 314 feet above sea level and not in any flood plains. The highest recorded crest of the Brazos River was recorded at 54 feet above flood stage or 246 feet above sea level. The nearest aquifer is the Bryan Sandstone and it is located under other geological at a depth beyond concern for his study.

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

The facility is located in a geological region known as the Gulf Coastal Plain. This region of Texas has only one fault zone associated with it, and the fault lies directly on the western boundary of the coastal plain.

2.5.2 Site Geology

The local geology is dominant by a formation known as the Easterwood Shale. The thickness of this formation varies from 10-300 feet. This formation is located throughout the area.

2.5.3 Seismology

The State of Texas lies in a geological area that has minor seismic activity. Most of the activity that exists is limited to extreme West Texas. This region lies over 600 miles west from the local area, but is the nearest active belt along the west coast of Mexico and the United States. No recorded earthquakes have ever occurred in the local area. The nearest fault zone is known as the Balcones Escarpment and is located on the western boundary of the Gulf Coast Plain, 100 miles to the west of this facility.

2.5.4 Maximum Earthquake Potential

The largest earthquake ever recorded in the State of Texas occurred near El Paso and measured 6.4 on the Richter scale. No seismic activity has ever been recorded in the local area and the possibility of any activity occurring is very low.

2.5.5 Vibratory Ground Motion

The area surrounding the facility has no geological activity associated with it. The area has been stable for as long as people have recorded such things and no threat of earthquakes or other seismic actives exists for this local region of Texas.

2.5.6 Surface Faulting

The local area is considered to have a very low potential for surface faulting. No earthquakes have ever been reported in the local area.

2.5.7 Liquefaction

The local foundation material has no potential for liquefaction. The facility is constructed in such a manner that this is not considered to be a potential problem.

Chapter 3: DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Design Criteria

The AGN-201M reactor control and instrumentation system has two basic design criteria. First, the ability to safely control the reactor under all normal and foreseeable accident possibilities, and second is the "fail-safe" design mode.

The fail-safe design mode is a concept that the reactor will be placed in a safe shutdown condition if any equipment or power failure were to occur. The nuclear instrumentation system is constructed with a series of trips and interlock systems that will accomplish this design criterion. A complete description of these trips and interlocks is presented in Chapter 7 of this document. The most critical passive design feature in this fail-safe mode is in the form of large springs on each of the safety rods and the coarse control rod. These springs ensure that the rods will be forcibly removed from the core if power is interrupted to the scram magnets, no matter the cause of the power interruption. The Technical Specification limit is less than 200 msecs for the safety system rods to be removed from the core, this limit ensures that the reactor will be safety shutdown even under serve transient conditions.

The reactor control and instrumentation system is designed to monitor reactor power all ranges for the AGN-201M reactor system.

The construction of the reactor room and adjoining room provides for a safe isolated environment in which to conduct reactor operations. The facility is designed to have very limited interaction with the building systems. The reactor control console power and power to the local ventilation unit and exhaust fan come from the in building power system. No additional systems are associated with the reactor facility.

3.2 Meteorological Damage

The location of Texas A&M University provides some threat to severe weather conditions. Rare tornadoes have occurred in the Brazos County and have been within 10 miles of the facility, however; these storms have dealt little damage to the University. Only the remnants of one hurricane have been detected in the county over the last 50 years and this led to some damage in the local community.

The reactor facility in the Zachry Engineering Center is designed to be a large cube inside the southwest corner of the building. This cube extends from the first floor down to the reactor room and laboratory adjacent to it. The walls of this cube are 3.5 feet of steel reinforced concrete, the ceilings are 4 feet of steel reinforced concrete, and the outer wall is a direct component of the building foundation. This system will stand up to any storm that may pass through the Bryan/College Station area.

In the event a tornado is sighted within a 5 mile radius of Texas A&M University, the radio operator at the University's Communications Center will notify the first available person listed on the AGN-201M Emergency Response phone numbers list. The communications center receives notification of tornadoes from both the Texas A&M weather radar and the Brazos County, Bryan-College Station Disaster Emergency Planning Organization. Tornadoes are detected by the use of the Texas A&M radar and the Doppler radar at a local television station.

3.3 Water Damage

The design of the reactor facility and the Zachry Engineering Center makes the likelihood of water damage very remote. The only route that floodwater could enter the area is through two sets of double doors. In the event of weather severe enough to warrant concern, these doors will be sandbagged to eliminate any possibility of water entering the facility. A large drain is located in the parking lot adjacent to these double doors to drain off the water.

If water entered the facility it would have to rise up to the level of the double doors leading to room 60C. Then flood the floor in this room in order to reach the double doors leading to the laboratory outside the reactor room. If water were to enter the reactor room a drain system is set up so that the water would drain into a 1000 gallon holding tank located in the basement floor of the facility. This sump would have to be completely filled in order to raise the water level in the reactor room.

3.4 Seismic Damage

The possibility of seismic activity is extremely low in this region of Texas. This combined with the fact that the walls of this facility are constructed on 3.5 feet of steel reinforced concrete in the form of a cube give the facility the strength to resist any seismic activity in the area.

3.5 Systems and Components

The fail-safe AGN-201M control and instrumentation system and the earthquake switch are designed to place the reactor in a shutdown condition in the event of a natural disaster. The loss of electrical power would also scram the reactor, since this is a likely byproduct of such an event. This also ensures the safe condition of the reactor. An Emergency Procedure EA-4, Severe Natural Phenomena, has been written for this facility in the case of such events and ensures the reactor is in a safe condition if the likelihood of such events is present or has already occurred.

The scram system for the AGN-201M is of a fail-safe design. Loss of power to the electromagnets will allow for the reactor to scram and be placed in a safe condition. The design basis for the scram system is to remove enough positive reactivity from the core to lessen the severity of any accident. The design basis for the control and safety systems for the AGN-201M reactor is presented in section 3.2 of the Technical Specifications.

Chapter 4: REACTOR DESCRIPTION

4.1 Summary Description

The AGN-201M research reactor in operation at Texas A&M University is essentially identical to all the other AGN-201M reactors that have been operated since 1957. Therefore, the reactor description given in the original Aerojet-General Nucleonics AGN-201M reactor report (Hazard Report and Preliminary Design Report by Aerojet- General Nucleonics, Docket F-40) is applicable to this reactor as are the Hazard and Safety Analysis Reports for the other AGN-201M reactors on file with the USNRC.

The characteristics and operating parameters of this reactor have been calculated and experimentally determined from data from this and other AGN-201M reactors.

The rated thermal power level for this AGN-201M reactor is 5 watts. This power level was approved by the USNRC in Amendment No. 10 to License R-23 issued on January 18, 1973.

The critical mass of the reactor is approximately [redacted] grams of ^{235}U. The fuel itself is in a plate type configuration and is homogeneously mixed powder of polyethylene and UO_2 [redacted] mg $^{235}U/cm^3$) in the form of 20 micron diameter particles. The core discs have been formed by pressing, under high pressure, the homogeneously mixed powder.

The reactor is a free standing unit. The reactor core is surrounded by a graphite reflector, a lead shield and a large water shield tank for fast neutrons.

The reactor has a natural convection system through the graphite reflector and the lead shield to the 1000 gallon tank of water that surrounds the reactor core. This removes any thermal heat due to the fission of the fuel.

The reactor has a hermetically sealed core tank surrounding it. This tank is designed to contain fission products gases if any core damage were to occur.

The experimental facilities associated with this AGN-201M are the small [redacted] the 4 -inch diameter access ports that pass through the water shield, graphite moderator, and lead gamma shield. A side view of the AGN-201M reactor is presented in Figure 4-1.

4.2 Reactor Core

4.2.1 Reactor Fuel

The reactor core of the AGN-201M reactor consists of nine separate fuel discs which are stacked into a right circular cylinder as shown in figure 4.2. This figure clearly depicts the physical relationship between the control rods, the core, and the core tank. The final core configuration is 10.15 inches (25.8 cm) in diameter and 9.34 inches (23.75 cm) in height. The fuel in the core is made up of approximatel[redacted] grams of ^{235}U in [redacted]grams of U for an enrichment of just less than 20%. Figure 4-2 shows the relationship between the core and the rods. The uranium is homogeneously mixed with polyethylene to form a series of fuel plates. A physical description of these fuel plates is included in the reactor characteristics table 4-1

The AGN-201M reactor core also contains four fueled control and safety rods. Three of these rods, Safety Rods 1&2 and the Coarse Control Rod, are constructed in a similar manner while the Fine Control Rod is much smaller in size. The fuel contained in these rods is exactly of the same type as the rest of the core. The reactor fuel has some limitations associated with it, in particular the breakdown of the fuel at high temperature. Several design features of the AGN-201M will prevent this temperature (200 °C) from being reached and if some core damage were to occur the gases and other materials would be trapped inside the core tank.

4.2.2 Control Rods

The AGN-201M design contains four fueled control and safety rods. Three of these rods, the two safety rods and the coarse control rod, are identical in design but differ in their function. Each of these rods contains about [redacted] grams of ^{235}U and operate in such a manner that as the rod is driven into the core the reactivity in the core increases. The exact amount of fuel contained in each of the rods is listed on table 4-1. Figure 4-3 shows the design of the rod which fits into a 2 inch (5 cm) hole in the lower disc in the reactor core.

The coarse control rod and the two safety rods are designed to contain 1.25% of reactivity each, while the smaller fine control rod has a reactivity worth of 0.31%. The fine control rod is designed to allow the reactor operator to make small adjustments in reactor power. The active length of these rods is 5.9 inches (15 cm) and consists of UO_2 embedded in radiation-stabilized polyethylene identical to that of the fuel discs in the reactor core. The active fuel material is enclosed in two aluminum containers. The outer containers provide fluid seals between the rods and their drive mechanisms and the shield tank. The innermost aluminum containers are used to seal the active fuel material contained in the fuel slugs. These containers provide a double seal between the fuel and the outside environment. These rods are calibrated as part of the AGN maintenance program The reactivity worth's of these control rods must be known to safety operate the reactor.

78"

[redacted]

Figure 4-1: SIDE VIEW OF AGN-201M REACTOR

[redacted]

Core Tank Cover

Graphite Reflector

Fuel Discs

Glory Hole

Thermal Fuse

Core Tank

Control & Safety
Rod Thimbles

Core Support
Assembly

Core Tank Cover

Figure 4-2: CORE TANK AND CORE

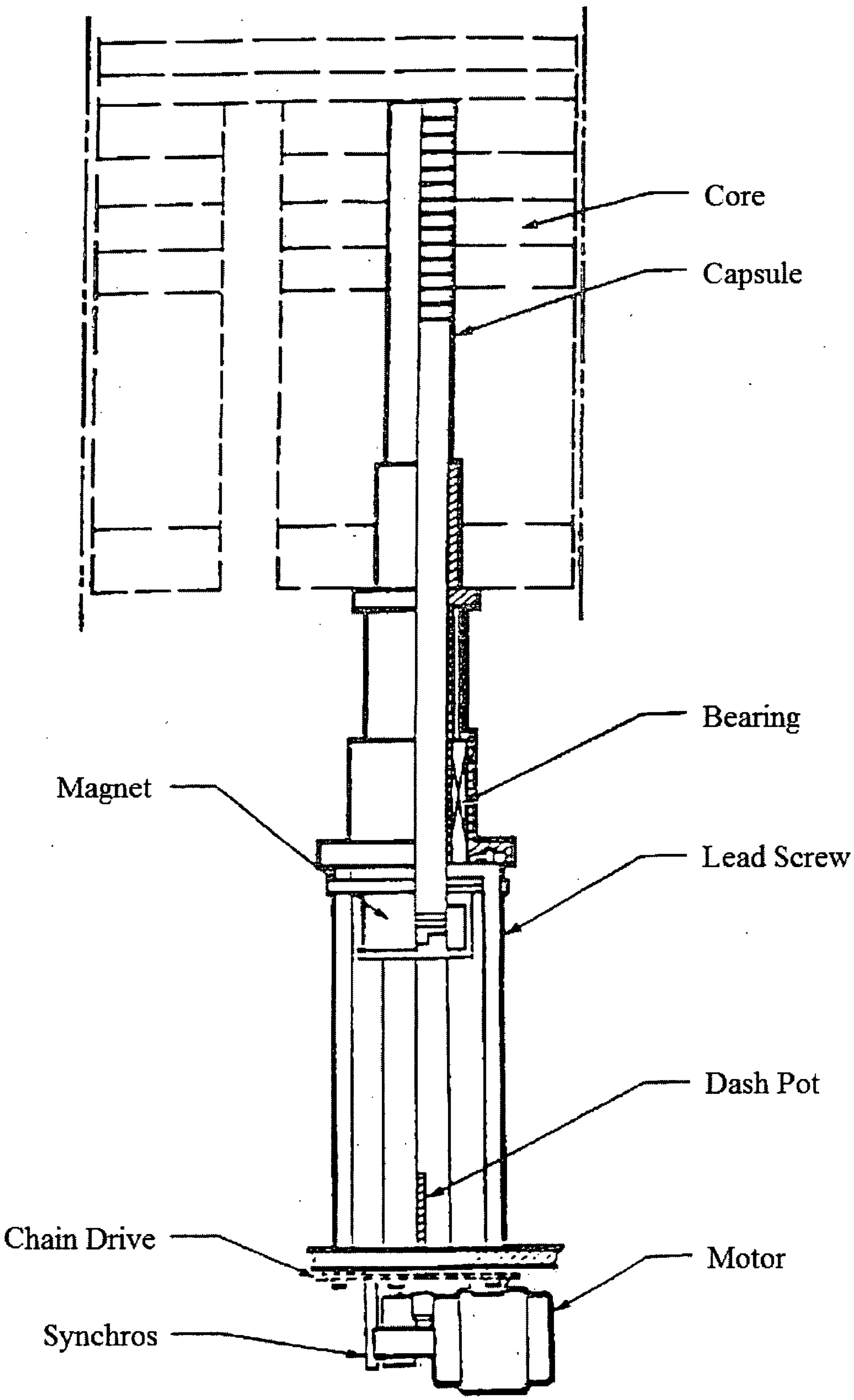

Figure 4-3: CONTROL ROD

Table 4-1

Reactor Characteristics

A. **General Description**	
Type	Homogeneous thermal reactor
Principal Uses	Education and Training
Maximum Rated Power	5 watts (thermal)
Core Design	UO_2 in the form of < 20% enriched ^{235}U particles homogeneously distributed in a solid polyethylene moderator, in nine fuel discs, in a cylindrical geometry
Reflector	Graphite
Shielding	Lead and Water
Rods	Two safety and two control, all fueled
B. **Fuel**	
Fuel Material	< 20% enriched UO_2
Fuel Disc	Right circular cylinder fabricated from a compressed mixture of fuel material and polyethylene powder Specifications: UO_2 ^{235}U enrichment < 20% Particle size – 15 ± 10 μm Polyethylene Powder Particle size – 15 ± 10 μm Purity – commercial grade
Approximate Core Fuel Disc Size and ^{235}U content	9 separate fuel disc Four discs 1.57" (4 cm) high [redacted] g ^{235}U each) Three discs 0.79" (2 cm) high [redacted] g ^{235}U each) Two discs 0.39" (1 cm) high [redacted] g ^{235}U each)

Approximate Safety and Coarse Control Rods fuel disc sizes: 4 per rod

1.85" (4.7 cm) diameter and 1.57" (4 cm) in height

Approximate Safety and Coarse Control Rods fuel disc loading: 4 per rod

[redacted] g ^{235}U each

Approximate Fine Control fuel disc sizes: 4 per rod

0.9" (2.3 cm) diameter and 1.57" (4 cm) in height

Total Fuel Loading	≈ [redacted] g ^{235}U
^{235}U Density	[redacted] g $^{235}U/cm^3$
Core Thermal Fuse	Small, right circular cylinder, 0.87" (2.2 cm) in diameter and 0.35" (0.9 cm) in height, enriched with [redacted] g of ^{235}U. Fabricated from a mixture of fuel materials and polystyrene powder.

C. **Reactor**

Core-containing Tank	Gas-tight, [redacted]) aluminum cylindrical tank, 12.7" (32.2 cm) in diameter and 30" (76 cm) high. [redacted]
Core	9 fuel disc, separated at core midpoint by glory hole and thin aluminum baffle. Lower half of the core has appropriate access holes for control and safety rod thimbles and core thermal fuse.
Reflector	Heavy density (1.75 g/cm^3) graphite, approximately 7.87" (20 cm) on all sides of the core. Top and bottom reflectors are contained in the core tank, side reflector surrounds the core tank and contains four 4" (10 cm) access ports running tangentially to the core tank
Gamma Shield	3.94" (10 cm) of lead shielding completely surrounding the graphite reflector
Reactor Tank	A [redacted]) thick steel tank, 37.4" (95 cm) diameter and 58.3" (148 cm) in height tank. This tank contains core tank, reflector and lead shield and the appropriate access holes for the control and safety rods, the glory hole and access ports. This is a gas-tight vessel when all the seals are made. The upper portion of this tank contains the removable thermal column.

Water Shield Tank	Steel tank serves as the main structural tank as well as a fast neutron shield. The tank is 6.5' (198 cm) in diameter and 7' (213 cm) in height.
Reactor Dimensions	6.5' (198 cm) in diameter and 9.17' (280 cm) in height
Reactor Weight	20,000 lbs. 11,400 lbs. without shield water
Reactor Control	Two Safety Rods Safety Rod #1 [redacted] grams of ^{235}U Safety Rod #2 [redacted] grams of ^{235}U Coarse Control Rod [redacted] grams ^{235}U Fine Control Ro[redacted] grams of ^{235}U All rods, with the exception of the FCR are mechanically coupled, and are magnetically coupled to the carriage which is driven into the reactor on a lead screw by a reversible DC motor. Total travel distance is about 25 cm

D. **Nuclear Data**

(1) Fuel Loading

(a)	Approximate Critical Mass	[redacted] grams ^{235}U
(b)	Last measured Excess Reactivity corrected to 20 °C with glory hole empty	0.137% Δk/k

(2) Neutron Flux

(a)	Average Thermal Flux	1.5×10^8 n/cm^2-sec at 5 watts
(b)	Peak Thermal Flux	2.4×10^8 n/cm^2-sec at 5 watts

(3) Reactivity Worth of Reactor Components

(a)	Safety and Coarse Rods	≈ 1.25 % Δk/k (each)
(b)	Fine Control Rod	≈ 0.31 % Δk/k
(c)	Cadmium Shutdown Rod	2.35 % Δk/k
(d)	Polyethylene Rod in Glory Hole	
	New Poly Rod	0.29 % Δk/k
	Old Poly Rod	0.26 % Δk/k
(e)	Temperature Coefficient of Reactivity	≈ - 0.024 % Δk/k per degree centigrade

(4) Important Figures

(a) Fine Control Rod Calibration Curve Figure 4-4

(b) Inhour Equation Figure 4-5

(c) Flux Profile Figure 4-6

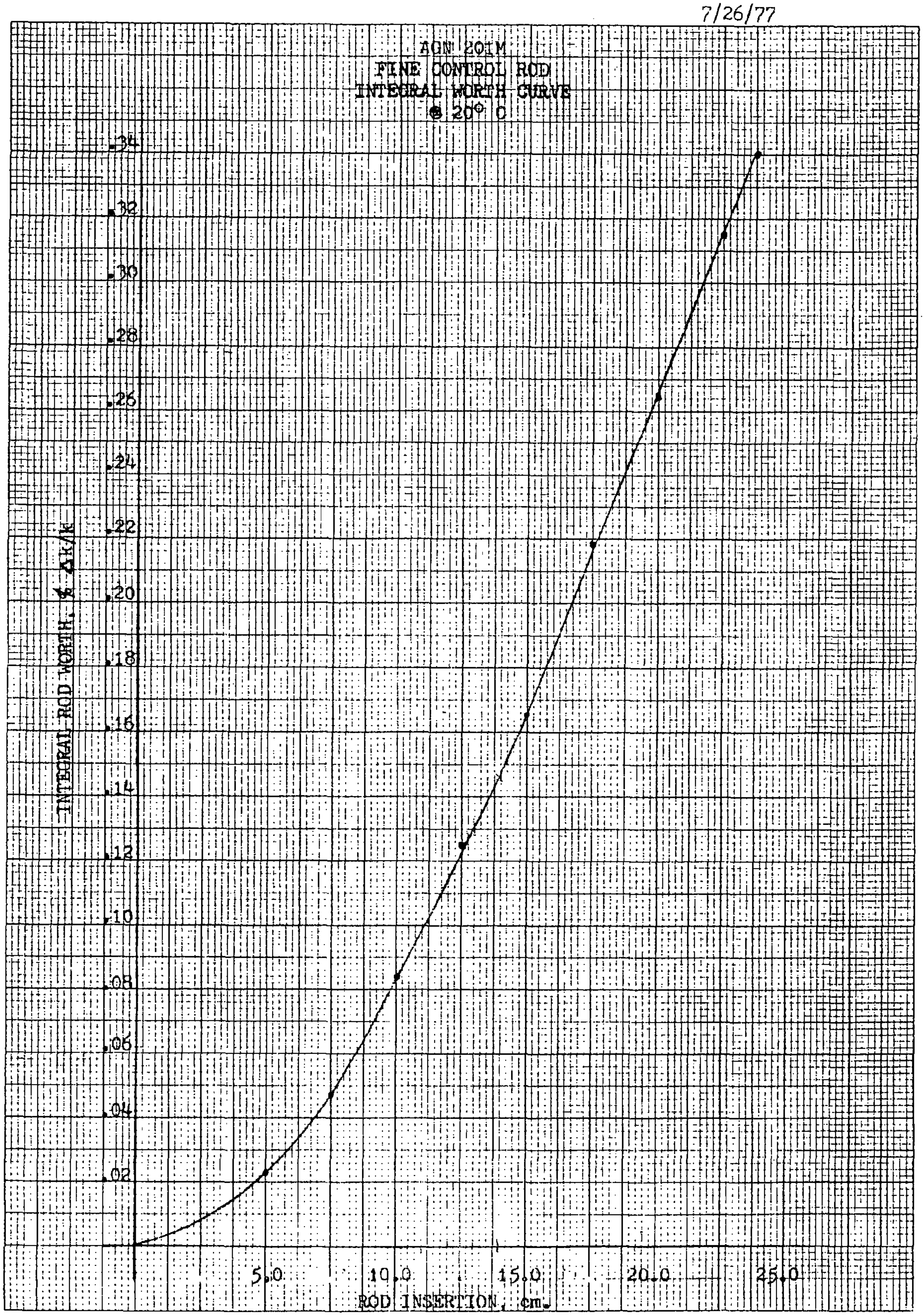

Figure 4-4: Fine Control Rod Calibration Curve

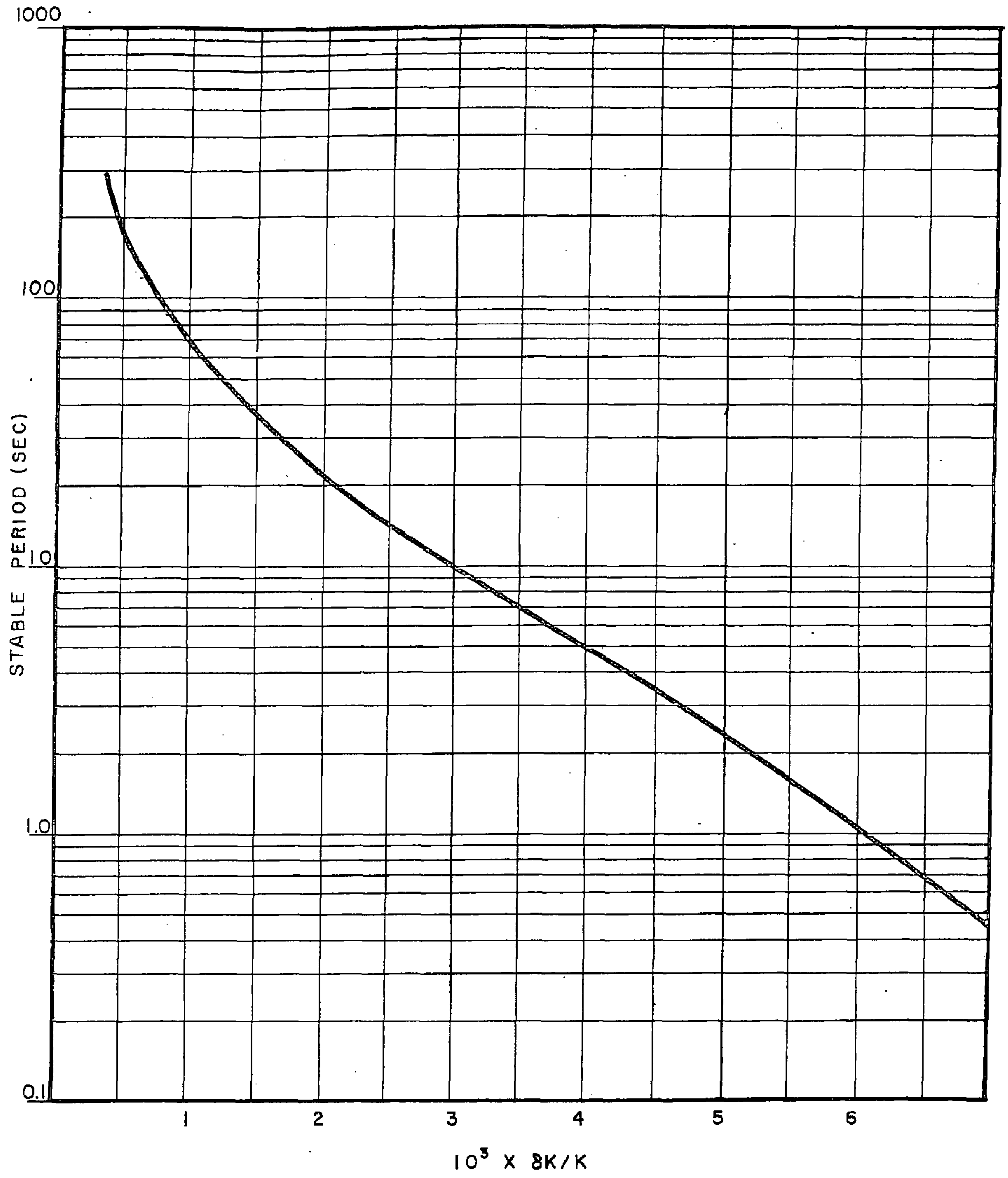

Figure 4-5: AGN-201 INHOUR EQUATION

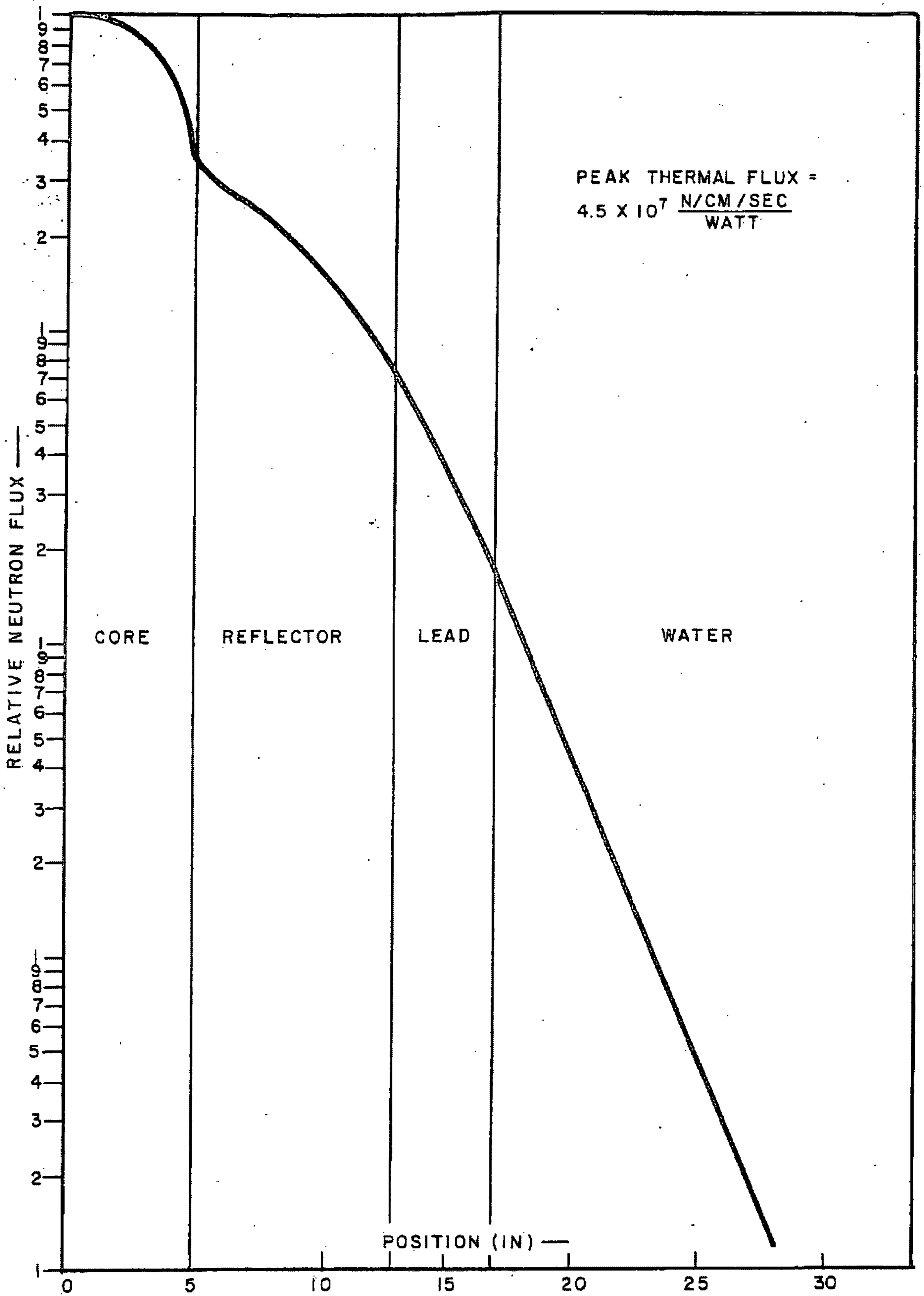

Figure 4-6: 100mW THERMAL NEUTRON FLUX PROFILE

When the reactor is operated the rods are inserted into the reactor in a specific sequence to ensure that sufficient shutdown margin is maintained at all times. This sequence is controlled by a series of limit switches associated with each of the safety rods and the coarse control rod. In order for rod drive motor power to be applied to Safety Rod #2, Safety Rod #1 must be fully inserted in to the core. This feature is tested during the pre-startup checkout in the operating procedures. A similar system is associated with Safety Rod #2. The Control Rod will not have rod drive motor power applied until Safety Rod #2 is in its fully inserted position in the reactor core.

All three of these rods are part of the reactor safety system and will be ejected from the core anytime a scram signal is received, even during rod insertion. The safety system is designed to be a "fail safe" system. This means that in the event of a scram signal the electromagnets holding the rods have their power interrupted. This will allow the rods to be accelerated out of the core downward by gravity and by the force of spring loading. The spring constant is such that these three rods are ejected from the core with a force of 5 g. With this amount of downward acceleration, these rods are out of the core in $\approx$ 120 milliseconds. The safety rods insert a negative 0.7 % $\Delta k/k$ during the first 50 milliseconds following the scram signal. The rods are decelerated by dashpots during the last 10 cm of travel to avoid damage to reactor components. The fine control rod due to its limited reactivity is not associated with the reactor safety system. This rod is driven out of the core at a rate of 0.5 cm/sec.

Both the fine and coarse control rods are driven by reversible DC motors through lead screw assemblies that are controlled by switches located on the control console. The maximum speed associated with rod movement is 1.0 cm/sec, this speed will result is a maximum possible reactivity change of 2×10^{-4} % Δ k/k per second for the coarse control rod. Both the coarse and the fine control rods can move at two speeds, 0.5 cm/sec and 1.0 cm/sec, a switch for each rod is located on the control console to allow for the different control rod speeds.

The positions of all rods are indicated on the reactor control console. Each rod has an indication of distance traveled into the core on a percentage basis. The trips and interlocks associated with the scram and interlock systems for the control rods are summarized in the following table.

Table 4.2.2-1

Trips and Interlocks Leading to a Reactor Scram

Scram Systems	Interlock Relay, High Level Channel #3, High Level, Channel #2, Manual Scram, Channel #2 Min. Period, Skirt Monitor High Level
Interlock System	Shield Water Temperature Switch, Earthquake Switch, Shield Water Level Switch, Channel #1 Low Count Rate, Channel #3 Low Level, Channel #2 Low Level, Relay Chassis Interlock, Rod Drive System Plug

4.2.3 Neutron Moderator and Reflector

The neutron moderator associated with the AGN-201M is the polyethylene that is part of the reactor fuel. The fuel is homogeneously mixed with 10,900 grams of polyethylene, to form the fuel disc in the reactor core. Since the fuel and polyethylene are part of the same matrix, the neutrons are able to be slowed down to a certain extent right after fission.

The reflector associated with the AGN-201M is made-up of high density graphite (1.75 g/cm^3) and is 20 cm thick. This reflector surrounds the reactor on all sides and is used to move neutrons back into the core after they scatter out of it.

4.2.4 Neutron Startup Source

A [redacted] gram plutonium-beryllium startup source is currently installed in the AGN-201M. This source was installed in access port #3 on May 1, 1972 after being approved as Amendment No.11 to License R-23 by the USNRC. The relative position of the source as it relates to the nuclear instrumentation is shown in Figure 4.2.4-1. The source is in a position in the access port in the access port 11.25 in. from centerline based upon a study conducted by facility staff in 1972. It was determined at this location the source provide sufficient neutron flux to ensure the detectors are in their normal operating range (10^{-12} to 10^{-6} A).

4.3 Reactor Tank or Pool

A large 1000 gallon tank of water surrounds the AGN-201M reactor, while the thermal column at the top of the tank itself is a separate tank. The bottom of the AGN -201M is not shielded by this tank in order to provide access to the control drive mechanisms and the rods. The tank is filled with water and provides a fast neutron shield. Scram features associated with this tank include a low-level trip and a low-temperature trip. This tank while providing fast neutron shielding is also the only cooling system associated with the AGN-201M reactor system. The tank is a solid unit and rest on the base of the AGN-201M system. The reactor core, lead shielding, and graphite reflector are in the center of the tank but not directly associated with it.

4.4 Biological Shield

The biological shield is designed to limit total gamma and fast and thermal neutron dose. At closest approach to the reactor, the total dose is less than 100 mrem/hr at reactor power less than 1.0 watts. This dose rate should be as low as possible to allow access to the experimental

[redacted]

Figure 4-7: TOP VIEW OF REACTOR

areas of the reactor room, while observing the ALARA principle. The other requirement is that the dose rate due to gamma, thermal and fast neutron dose is less than 15 mren/hr in the accelerator room. The construction of the accelerator room floor prevents exceeding this limit during operation. A complete copy of a radiation survey has been included in the back of this section of the Safety Analysis Report. The biological shielding must also perform under emergency conditions.

4.5 Nuclear Design

The construction and description reactor core and fueled control rods for the AGN-201M have been described earlier in this chapter. The behavior of the AGN-201M reactor core has been well defined over many years of operations and its previous Safety Analysis Reports and Hazard Safety Report from Aerojet-General Nucleonics. During normal and emergency operations no damage to the reactor core or fuel is expected due to the compact and conservative design of the reactor core as well as the presence of many passive safety features associated with this reactor system.

4.5.1 Normal Operating Conditions

During normal operating conditions the AGN-201M reactor will exhibit little fuel burnup or usage. The reactor core has only a single configuration for its control rods and operates in a very low and limited power range.

4.5.2 Reactor Core Physic Parameters

Table 4-1 Reactor Characteristics provides a listing of some core physics parameters. The reactivity worth's for the control and safety rods are determined annually using maintenance procedure for rod worth. The values for the polyethylene rods can be determined in the same manner, as well as the cadmium shutdown rod. This procedure has been used for many years and has proven an effective method for the determination of rod worth's.

4.5.3 Operating Limits

The limiting conditions are designed so that the limitations on total core excess reactivity will help to 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 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. These reactivity limits must apply to the reactivity condition of the reactor and the reactivity worth's of control rods and experiments.

If these limits are followed it will ensure that the reactor can be shut down at all times and that the safety limits will not be exceeded.

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

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

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

The core excess reactivity limits are designed to ensure that the reactor will have reactor periods of sufficient length so as not to exceed safety limits if the reactor protective system and/or operator action will be able to shutdown the reactor. These operating limits are verified during annual preventive maintenance procedure ROEX-3, Calculation of Shutdown Margin and Total Excess Reactivity.

When setting operating limits the maximum steady state power level and maximum core temperature during steady state and transient operating conditions one must always consider the integrity of the fuel matrix so that no fission products are allowed to escape the core matrix.

The polyethylene core material does not melt below a temperature of 200°C and is expected to maintain its integrity and retain essentially all of the fission products at temperatures below that point. The Hazards Summary Report dated February 1962 and 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.

The safety limits shall be set to ensure the fuel matrix maintains it integrity at these levels

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

Now that the limits exist, reactor safety system set points must be determined which will limit maximum power and core temperature. These shall be the Limiting Safety System Settings. At these setting an automatic protective action will be initiated to scram the reactor and prevent a safety limit from being exceeded.

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 120°C, or below, will assure safe shutdown without exceeding a core temperature of 200°C.

So the settings have to be ≤ 10 watts

Channel	Condition	LSSS
Nuclear Safety #2	High Power	≤ 10 watts
Nuclear Safety #3	High Power	≤ 10 watts

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

4.6 Thermal-Hydraulic Design

The thermal-hydraulic design of the AGN-201M reactor system is limited to the transfer of heat to the shield tank. The shield tank is so large, 1000 gallons and the heat load so low from the AGN-201M reactor that the reactors effect on the tank is not noticed. But the reverse is not true; the tank must be filled to within 10 inches of the top on order to meet shielding requirements. The shield tank is the primary fast neutron shield in the AGN-201M reactor design.

Procedure for AGN Reactor Radiation Survey

1. Fill out cover page including instruments to be used in the course of the survey.

2. Perform a background radiation survey in the AGN Room. Record the results as being performed at "0" (zero) power level. Measurements made in the AGN Room are performed with the detector placed on top of a meter stick and the bottom of the stick on the "X" on the floor.

3. Perform surveys at various power levels. Although rated power is 5 watts, the reactor may not be able to achieve full power due to temperature constraints. At the highest power level achievable, survey the Accelerator Room. These measurements are made with the detector placed on the floor.

[redacted]

[redacted]

AGN Form 301
Rev. -
June 2010

Date: 23MAR11

Texas A&M University
AGN-201M Reactor Facility

Reactor Power Level 0 Watts

Survey Point #	Gamma Dose mrem/hr	Neutron Dose mrem/hr	Total Dose mrem/hr
AGN Room			
1	0.05	0	0.05
2	0.03	0	0.03
3	0.02	0	0.02
4	0.02	0	0.02
5	0.2	0	0.2
6	0.7	0.1	0.8
7	0.25	0	0.25
8	0.6	0	0.6
9	0.04	0	0.04
Accelerator Room			
Grating			
Shield Plug			

Gamma Instrument ID: B852M Calibration Due: 8/7/2011

Neutron Instrument ID: 00130 Calibration Due: 5/20/2011

Notes:

Signature: Caitlin Anderson 3-23-11

AGN Form 301
Rev. -
June 2010

Date: 23 MAR 11

Texas A&M University
AGN-201M Reactor Facility

Reactor Power Level 1 Watts

Survey Point #	Gamma Dose mrem/hr	Neutron Dose mrem/hr	Total Dose mrem/hr
AGN Room			
1	0.1	0	0.1
2	1.0	0	1.0
3	1.5	1.0	2.5
4	0.2	0	0.2
5	7.0	7.0	14.0
6	2.0	0.3	2.3
7	7.0	5.0	12.0
8	4.0	2.0	6.0
9	6.0	3.0	9.0
SKIRT DOOR	9.0	21.0	30.0
Accelerator Room			
Grating	—	—	—
Shield Plug	—	—	—

Gamma Instrument ID: B852M Calibration Due: 8/7/2011

Neutron Instrument ID: 00130 Calibration Due: 5/20/2011

Notes:

Signature: Caitlin Anderson 3-23-11

AGN Form 301
Rev. -
June 2010

Date: 23MAR11

Texas A&M University
AGN-201M Reactor Facility

Reactor Power Level 3 Watts

Survey Point #	Gamma Dose mrem/hr	Neutron Dose mrem/hr	Total Dose mrem/hr
AGN Room			
1	0.3	0	0.3
2	4.0	1.5	5.5
3	6.0	1.0	7.0
4	0.4	0	0.4
5	30.0	15.0	45.0
6	5.0	6.5	11.5
7	30.0	24.0	54.0
8	11.0	5.0	16.0
9	20.0	15.0	35.0
SKIRT DOOR	30.0	65.0	95.0
Accelerator Room			
Grating	—	—	—
Shield Plug	—	—	—

Gamma Instrument ID: B852M Calibration Due: 8/7/2011

Neutron Instrument ID: 00130 Calibration Due: 5/20/2011

Notes:

Signature: Caitlin Anderson 3-23-11

AGN Form 301
Rev. -
June 2010

Date: 23MAR11

Texas A&M University
AGN-201M Reactor Facility

Reactor Power Level 5 Watts

Survey Point #	Gamma Dose mrem/hr	Neutron Dose mrem/hr	Total Dose mrem/hr
AGN Room			
1	0.6	0	0.6
2	7.0	0.1	7.1
3	10.0	2.2	12.2
4	0.8	0	0.8
5	48.0	18.5	66.5
6	9.0	1.0	10.0
7	50.0	32.0	82.0
8	18.0	16.5	34.5
9	43.0	18.0	61.0
SKIRT DOOR	60.0	130.0	190.0
Accelerator Room			
Grating	0.4	0	0.4
Shield Plug	0.01	0	0.01

Gamma Instrument ID: B852M Calibration Due: 8/7/2011

Neutron Instrument ID: ~~15268~~ 00130 Calibration Due: 5/20/2011

Notes:

Signature: Caitlin Anderson 3-23-11

Chapter 5: REACTOR COOLANT SYSTEMS

The AGN-201M reactor is operated at very low power levels for short periods of time. The 1000-gallons of water in the shield tank, whose design function is that of a fast neutron shield, will by natural convection flow dissipate any heat that is rejected from the core. The amount of heat transferred to the shield tank has a little effect on the temperature of the tank.

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Chapter 6: ENGINEERED SAFETY FEATURES

Engineered safety features (ESFs) are systems provided to mitigate the radiological consequences of designed-basis accidents. Since the AGN-201M reactor system operates at such a low maximum power level, 5 watts thermal, the fission product inventory is very low. Evaluation of the maximum hypothetical accident has indicated no significant radiological releases. Therefore, no ESF systems are required or needed at an AGN-201M reactor facility

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Chapter 7: INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The instrumentation and control (I&C) system of an AGN-201M reactor consists mostly of three nuclear instrument drawers that allow for operation of the reactor and, if necessary, send a variety of scram signals to the reactor control safety systems. The design basis for these trips will be discussed in detail later in this chapter.

The reactor control system is designed, through a series of interlocks and relays, to control movement of the safety and control rods. This is done to ensure that a reactor startup cannot commence unless both safety rods are fully withdrawn from the core. The interlocks ensure that only one safety rod can be inserted at a time and that the coarse control rod cannot be inserted unless both safety rods are fully inserted.

The rods are controlled at the control console by a series of four switches, one for each rod, and position indication is provided on the console. Indicator lights will illuminate to give the operator indication of the status of each rod. These lights are illuminated when the rod and the carriage reach certain positions in their travel. A series of eight relays (K10 -17) actually provide the controlling features to this system.

The reactor control safety system receives input from the nuclear instruments, various other detectors and the sensing systems and voltage detectors on the interlock relay system itself.

The console and at its features are clearly presented in a series of four figures presented in section 7.6 of this chapter. The layout of the control console is presented in these diagrams and all detectors and instrumentation are clearly labeled.

The only area monitor device directly associated with the AGN-201M reactor is the skirt monitor under the reactor. This system is intended to scram the reactor if radiation levels greater than 2 times the last skirt monitor indication at 5 watts.

7.2 Design of Instrumentation and Control Systems

7.2.1 Design Criteria

All systems and components associated with this system are located in the reactor console, in the shield tank or under it. The reactor control system diagram is presented in Figure 7-1, this figure shows both the interlock system and the scram system for the AGN-201M. Figure 7-1 also includes a block diagram of the nuclear instrumentation systems for the AGN-201M is presented. A complete description of each of the nuclear instrumentation channels is presented in section 7.2.3.

The design and operations of these detection systems is such that the Limiting Safety Systems Settings will not be exceeded due to a reactor scram initiated by these systems. The control and safety setpoints are presented in Table 3.1 in the Technical Specifications of the AGN-201M reactor and will be presented later in this chapter.

7.2.2 Design – Basis Requirements

The design basis of this system is to monitor the entire range of reactor power and to provide protective features that will safely shutdown the reactor in the event of an equipment failure, operator error, or accident. The three nuclear instrumentation channels constantly monitor the reactor at all times. The scram and interlock reactor control safety systems are directly associated with the nuclear instruments since many of the signals to these systems come directly from this instrumentation.

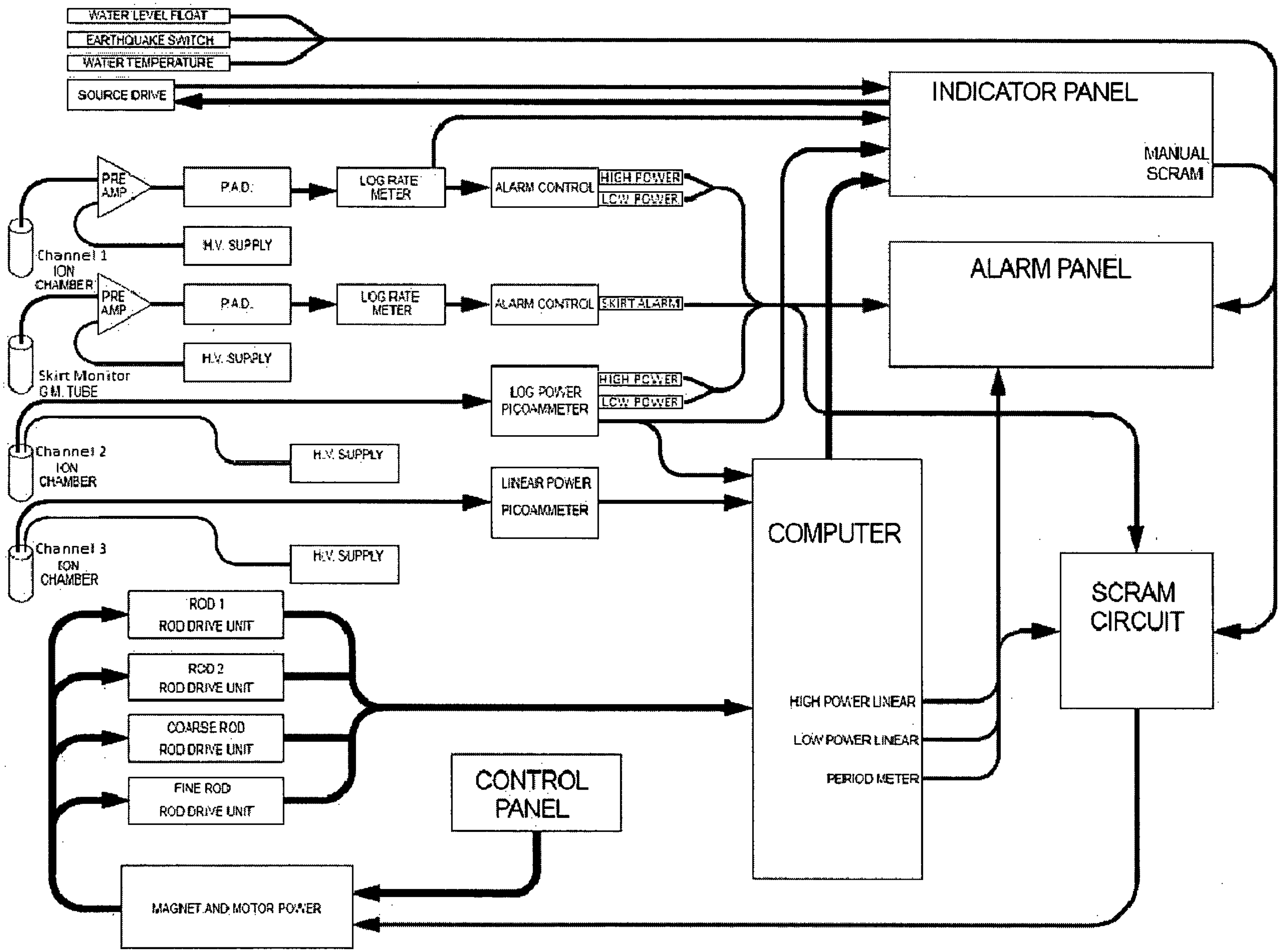

Figure 7-1: REACTOR CONTROL SAFETY SYSTEM

7.2.3 System Description

The nuclear instrumentation system consists of three channels and the skirt monitor.

Channel No. 1 The components that comprise nuclear instrument Channel No. 1 are presented in block format in Figure 7-1. The detector is a BF_3 filled proportional counter with a range of 10 -10^6 counts per second. The detector is mounted in the shield tank, the relative position of all detectors are presented in Figure 4-7. The channel drawer for this detector is located in the bottom right hand side of the instrument panel.

Channel No.1 is a low-range detector and its primary function is to ensure a neutron source is present in the reactor during startup. The detector system is made up of a high voltage power supply, a preamplifier and amplifier. The linear amplifier and power supply are located in the instrument panel of the control console; the preamplifier is located on the top of the reactor in the cabling runs.

The high voltage for this detection system is determined annually as part of the preventive maintenance program. The system is also checked prior to reactor startup during the pre-startup checklist. The function generator, on the instrument panel, is set to different values to ensure the meter responds to a known input value. Another check performed as part of the pre-startup checklist is the low count rate channel #1 interlock. This check is performed to ensure that if the count-rate drops below 10 counts per second the reactor will scram. An annual maintenance procedure also tests this interlock to ensure that it is functioning correctly.

Channel No. 2 The block diagram for Channel No.2 is presented in Figure 7-1. The function of this channel is to provide both high and low-level trips as well as an input signal to the period meter. This system present output in a logarithmic fashion on a display that reads from 10^{-6} – 10 watts allowing the power to be monitored over the complete range of reactor operations without switching scales.

The detector associated with Channel No. 2 is a BF_3-filled ionization chamber. The detector is physically located in the shield tank as shown in Figure 4-7 in a waterproof housing. The high voltage power supply for the detector is located in instrument panel portion of the control console.

Once the signal leaves the detector it goes to a Keithley Model 6487 Picoammeter for processing. This processed signal is fed to the alarm panel, scram circuit and the AGN computer. The alarm panel allows indication of the associated alarms. The scram circuit interrupts power to the rod magnets in a high or low power condition. The AGN computer sends an output signal to the Channel No. 2 panel display. The computer also calculates the reactor period and drives the period meter display. The reactor period also inputs to the alarm panel and scram circuit. The operation of Channel No. 2 is tested by using test current switch.

Prior to reactor startup, a pre-startup checkout is required to be performed. This procedure tests the operation of Channel No.2 instrumentation and the period meter as well. The first test ensures proper operation over the full range of reactor power. In order accomplish this, a series of test signals are fed into the system and verified against known values. The low level interlock is tested next. This interlock is tested by driving the startup source to the "out" position and verifying that rod magnets cannot be energized. The high-level trip is then checked. The test current knob is positioned to input a scram signal to the system. The operator depresses the test log channel button and observes that the rod magnets de-energize and the high power log channel alarm illuminates. These settings are verified annually during reactor maintenance and are used to ensure the reactor will automatically scram before power reaches 10 watts or if the signal input falls below 10^{-12} amps. Operation of the period meter is also checked during this procedure. A large test signal is fed into Channel No. 2, this rapid change in indicated power allows the operator to ensure a reactor period scram occurs at periods less than 5 seconds. An annual maintenance requirement also is in place to check the proper operation of the period meter.

The output signal from the Channel No.2 ammeter has a number of very important functions; it provides a signal to both the interlock system and scram system directly as well as to the period meter and also has a direct input to the scram system. The computer also displays the power level of Channel No. 2 over the last hour.

Channel No.3 The block diagram for Channel No.3 is presented in Figure 7-1. The primary function of this channel is to provide high power level scram protection and to ensure that the LSSS and LCO are not exceeded. This channel is the primary indication of reactor power over all ranges of operation.

The detector associated with Channel No. 3 is a BF_3-filled ionization chamber. The detector is physically located in the shield tank as shown in Figure 4-7 in a waterproof housing. The high voltage power supply for the detector is located in the instrument panel portion of the control console.

Once the signal leaves the detector it goes to a Keithley Model 6487 Picoammeter for processing. This processed signal is fed to the AGN computer. The computer uses this signal to calculate linear power readings in watts. Linear power is then displayed on an analog and digital display on the control panel. The computer also provides input to the alarm panel and scram circuit for any low/high linear power condition. The alarm panel allows indication of the associated alarms. The scram circuit interrupts power to the rod magnets during a high/low power condition. The operation of Channel No. 3 is tested by using test current switch.

Channel No. 3 instrumentation is also checked during the pre-startup checkout. The first test ensures proper operation over the full range of reactor power. In order to accomplish this, a series of test signals are fed into the system and verified against known values. The low level interlock is tested next. This interlock is tested by driving the startup source to the "out" position and verifying that rod magnets cannot be energized. The high-level trip is then checked. The test current knob is positioned to input a scram signal to the system. The operator depresses the test linear channel button and observes that the rod magnets de-energize and the high power linear channel alarm illuminates. These settings are verified annually during reactor maintenance and are used to ensure the reactor will automatically scram before power reaches 10 watts or if the input signal falls below 10^{-12} amps.

The output signal from the Channel No. 3 ammeter provides a signal directly to the AGN computer. The computer then inputs to the scram circuit and the alarm panel for any low/high linear power condition. The computer also displays the power level of Channel No. 3 over the last hour.

The automatic trips associated with Channel No. 3 are low and high linear power. These are set and verified by annual maintenance procedures.

<u>Skirt monitor</u> The skirt monitor is a G.M. tube placed directly under the skirt of the reactor and inside a series of lead blocks. This system diagram is presented in Figure 7-1. The function of this system is to provide a high area radiation level scram signal if the detector indicates a reading that is 2 times a previous 5 watt reading. The signal from this system goes directly to the scram relays and will also sound the evacuation alarm.

The Technical Specifications clearly state all the required testing for all channels of the nuclear instrumentation system. The LSSS and LCO are clearly described in this document.

7.3Reactor Control and Protective Systems

The AGN-201M reactor system only has a manual control system for control of the safety, fine, and coarse control rods. The first part of this section will discuss the system design to limit rod movement, and the second part will discuss the reactor protective system. The only interlocks associated with this system are directly related to rod movement and the maintenance of adequate shutdown margin.

The rods in the AGN-201M reactor system which are directly related to reactor safety, the two safety rods and coarse control rod, have limitations on their respective movement and cannot be moved unless certain conditions are met.

The first rod to be moved is Safety Rod #1, this rod must be driven into its fully inserted position prior to the K-10 relay activating, cutting power to Safety Rod #1 and allowing movement of Safety Rod #2. The K-relays are designed to activate only when the appropriate safety system rod is in its fully inserted position or in the fully withdrawn position depending upon their function. Once Safety Rod #2 is fully inserted and the K-12 relay operates, then the Coarse Control Rod can be moved. A set of these K relays also are activated when the carriages are returned to the down position so that rod movement may be started again after a reactor scram.

The reactor protective system, known as the Reactor Control Safety Systems consists of the Interlock System as well as the Scram System. The Interlock System was designed as an add-on to the original Scram System when it was found over the years of AGN operations that more automatic protective devices need to be added to ensure that none of the LSSS and LCO were exceeded as the reactor power levels were raised.

The Reactor Control Safety Systems rely upon signals from the nuclear instruments as well as other detection systems to perform their designated function, that is, to scram the reactor. The features directly associated with the nuclear instrumentation system were discussed in a previous section, so the other parts of the Interlock Relay and Scram System will be addressed in this section.

The manual scram switch is located in the middle of the channel display panel and may be activated in the event of any emergency condition that exists in or around the reactor facility. An integrated diagram of the Scram System is provided in Figure 7-1.

The Interlock Relay in the Scram System will activate anytime one of the Interlock System trips is activated, just as any trip circuit in the Scram System.

The Interlock System consists of some trips that are associated with the shield water tank, a low-temperature trip that activates at 20 $\pm$ 0.5 °C and a low level trip that activates when the level in the shield tank falls below 10 inches from the highest point.
These trips are important maintaining shielding requirements in the reactor room. Both of these trips are checked as part of the annual maintenance program and as directed by the Technical Specifications. The other trips will active if the scram bus voltage is sensed to be low a trip will occur.

7.4 Control Console and Display Instruments

The reactor control console is presented in Figures 7-3, 7-4, and 7-5. These figures provide a clear idea of the particular console design for the AGN-201M reactor. The console is designed so that the operator has within easy reach all control switches for the nuclear instrument channels as well as a clear view of all indications including the alarm panel. Figure 7-5 depicts the alarm panel. The indicator lights on the panel are different colors and give the operator a visual as well as audio indication of the reactor status.

The rod energized lights are green and the scram indicator lights are red, these are just two examples of how the alarm panel was designed with the operator in mind. The rod position indicator lights are different colors which makes the identification very simple as well as providing a visual indication of rod position. Figure 7-4 displays the computer panel. The logarithmic and linear power indications are presented in graph form on the computer monitor. The computer program also displays each rod position as well as the rod bottom lights, rod engaged light and the rod carriage position indicators. Below the computer monitor channel #2 high voltage reading is displayed. Figure 7-5 shows the channel display panel. This panel displays each channel's indicated power level, the reactor period, and magnet current display. The panel also includes the manual scram switch, start-up source control, reactor console keyhole, and rod magnet power control. These figures clearly show the ease with which the reactor operator can observe and monitor reactor operations. In Figure 7-5, the switches that control the rods are shown. This horizontal panel is located directly below the computer panel.

7.5 Radiation Monitoring Systems

The radiation monitoring system directly associated with the AGN-201M reactor is the skirt monitor. A G.M. tube, inside a lead shield, is placed under the reactor inside the skirt. The skirt monitor panel on the control console provides an indication of the radiation level under the reactor. As part of the annual maintenance program the radiation levels at a reactor power of 5 watts is recorded and a voltage plateau performed on the detector. The skirt monitor is then set to alarm at two times this 5-watt reading. It is assumed that, if the radiation levels reach these levels in the local area, some accident or loss of shielding has taken place. When this monitor reads its set point the reactor will scram and the evacuation horn will sound in the areas outside and above the reactor room. Should this occur Emergency Procedures will be initiated, and the cause of the problem will be investigated, and corrective actions will be taken. Another monitor is present as a criticality monitor for the source locker, this system is designed to measure radiation levels above the locker and ensures that conditions don't exist that a k_{eff} of 0.8 is possible.

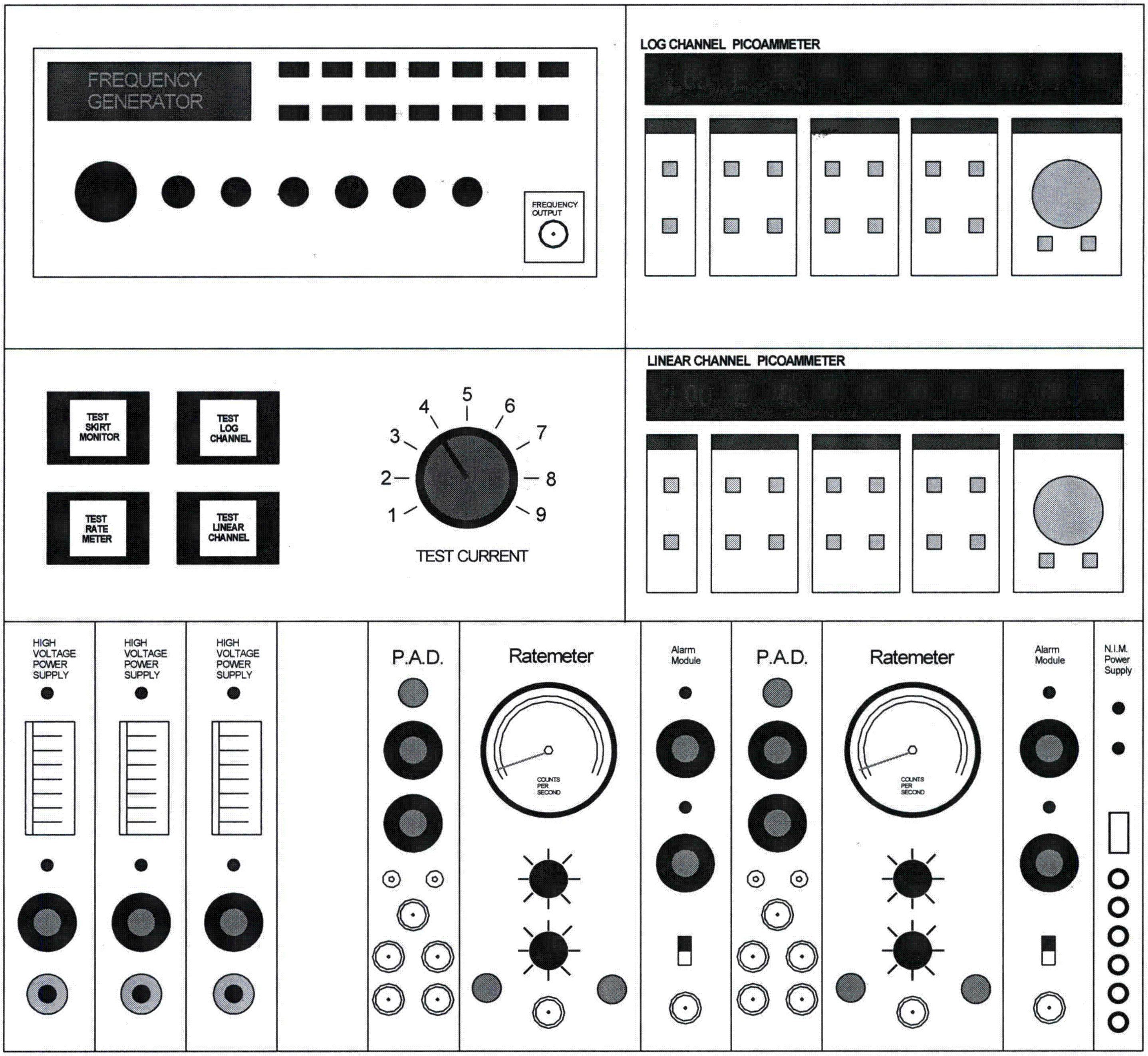

Figure 7-2: INSTRUMENT PANEL

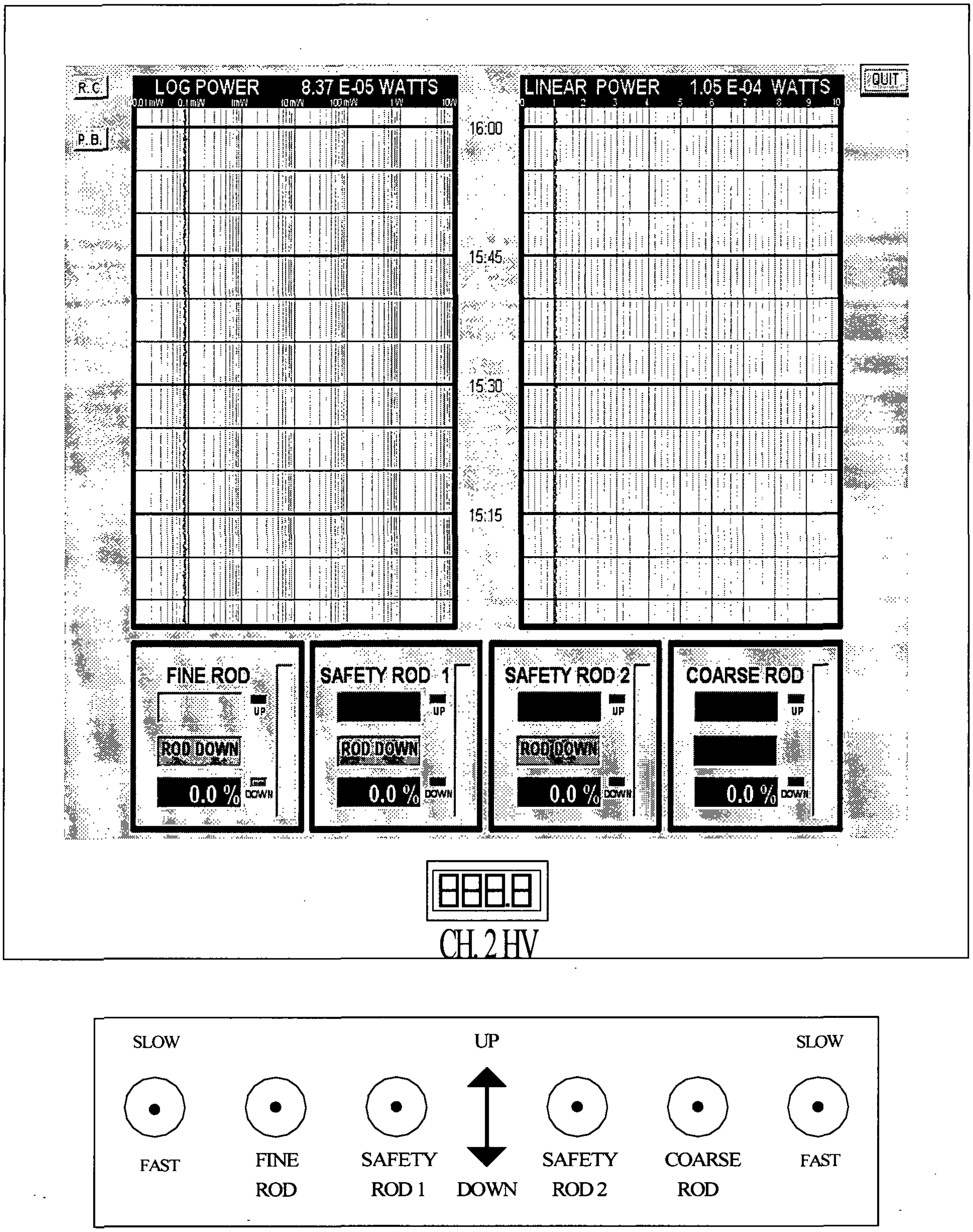

Figure 7-3: COMPUTER PANEL

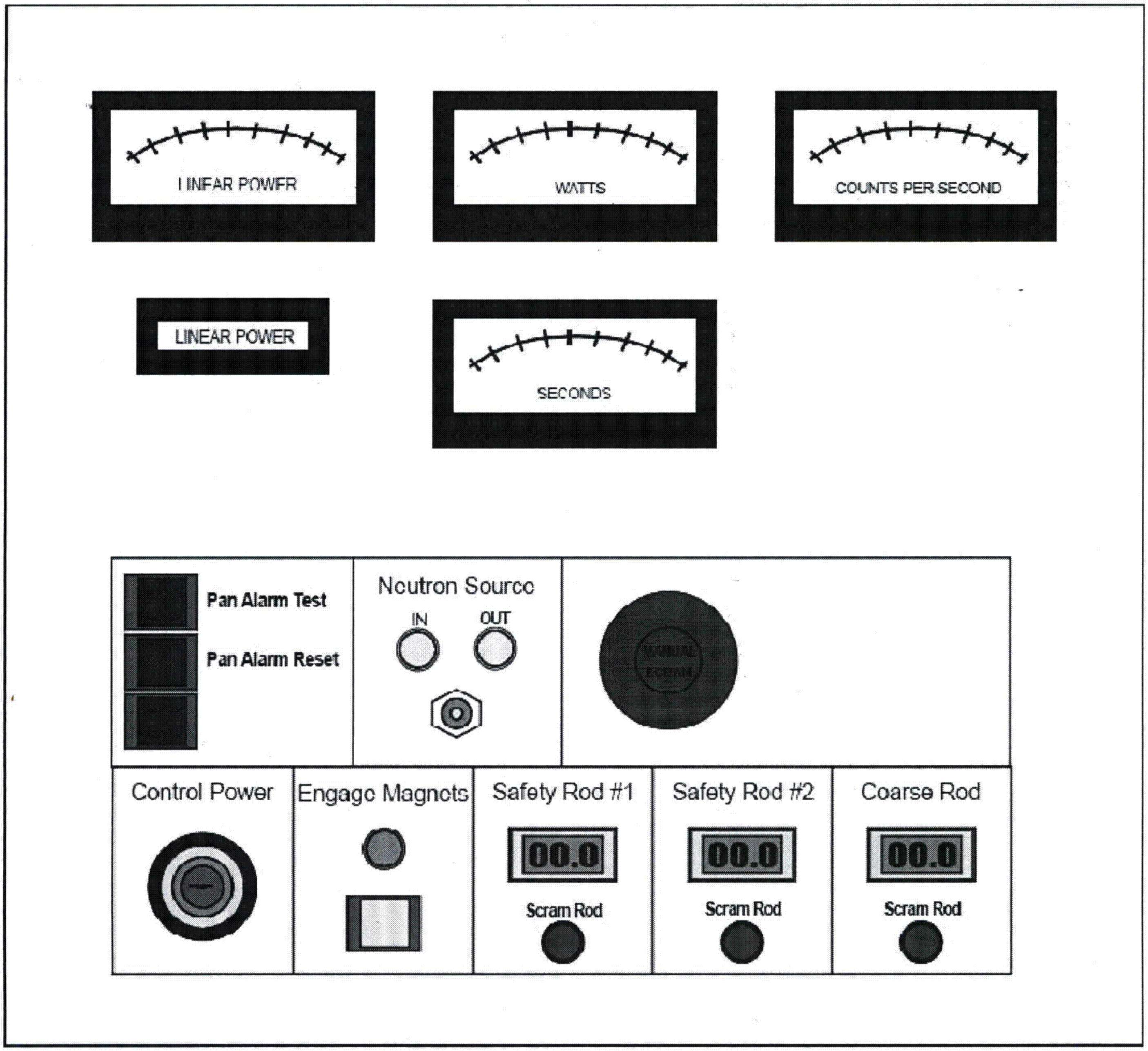

Figure 7-4: CHANNEL DISPLAY PANEL

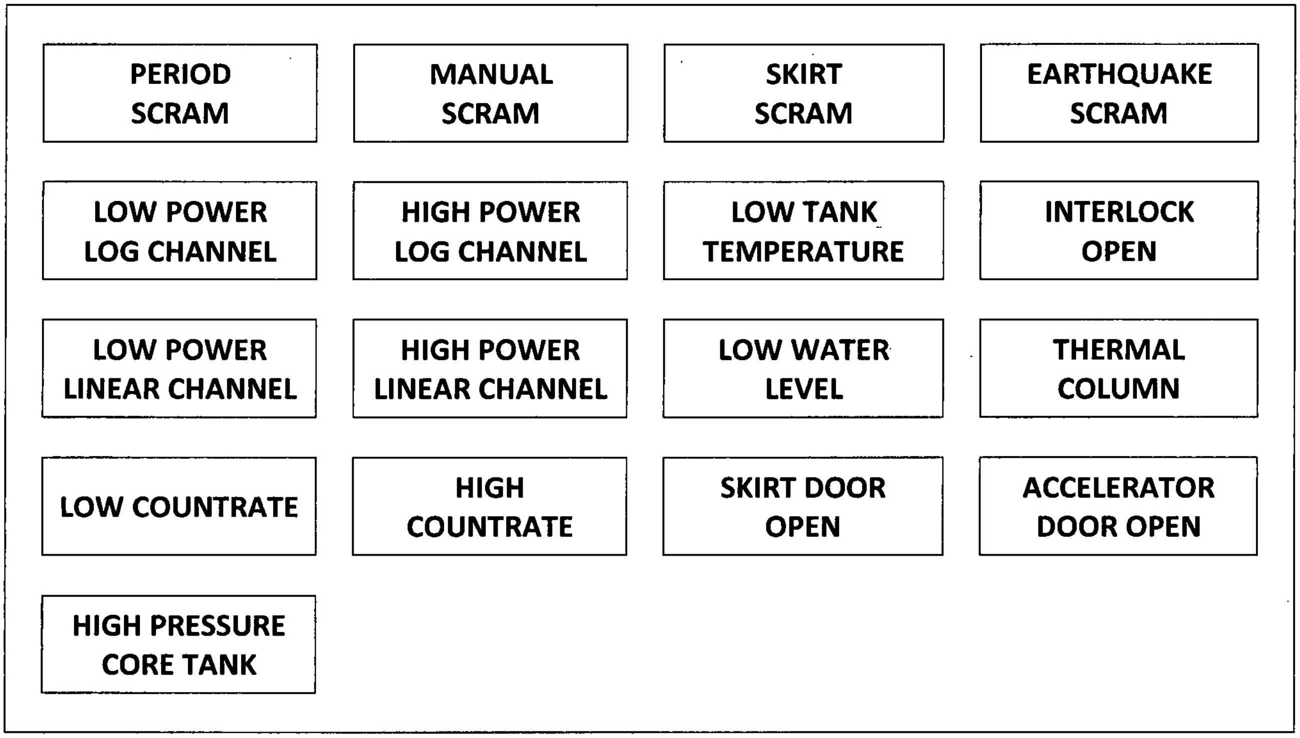

Figure 7-5: CONTROL CONSOLE ALARM PANEL

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Chapter 8: ELECTRICL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The AGN-201M reactor system has an electrical requirement of 2kw of 110 volt AC. This power is supplied directly to the reactor room from the Zachry Engineering Center power network. The actual power to the reactor room is both 220 and 110 volt AC. Both of these power systems can be remotely disconnected using two breakers located in the hallway outside of room 60C. The AGN-201M reactor system has no emergency power requirements. If a loss of power occurs during operation the rods are forced from the core and the reactor is considered shutdown. The cadmium rod is added to the core to ensure the reactor is shutdown since without instrumentation this is hard to verify. A radiation survey can also be performed to ensure the reactor is shutdown.

The AGN-201M reactor system uses a variety of voltages and currents in normal operations. It would be impossible to present all that information even in this context. The major power distribution electrical block diagram for the Texas A&M University AGN-201M reactor is Drawing Number 2-C-100631. This diagram breaks down the regulated 110 volt AC to all the detector high voltage power supplies and to each of the channel drawers. This regulated AC passed through two sola type voltage transformers. These transformers have been replaced by modern line conditioners to better handle the power from the Zachry Engineering Center.

8.2 Emergency Electrical Power Systems

The AGN-201M reactor does not require an emergency power system due to its design and lack of external systems. The reactor facility has an emergency lighting system that will energize when power is lost. This will ensure some lighting and allow for safe departure from the reactor room.

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Chapter 9: AUXILIARY SYSTEMS

AGN-201M has few auxiliary systems associated with it and none have any safety significance except for the ventilation system.

9.1 Heating, Ventilation, and Air Conditioning Systems

The ventilation system for the AGN-201M reactor room consists of one circulating unit by the reactor console, a supply area in the accelerator laboratory and an exit fan on the roof of the Zachry Engineering Center. The inlet plenum for the fan suction is located in room 135, the accelerator laboratory, and the air flow is designed to pass from the reactor room up through the grate in the ceiling and into this inlet plenum. Measurements have confirmed that air passes from outside both the accelerator laboratory and the reactor room in this manner. The supply header in the accelerator consist of three outlets supplying air at 500 cfm each and the exhaust fan drawing off air at 1500 cfm. The unit in the reactor room recirculates air at 1100 cfm.

9.2 Handling and Storage of Reactor Fuel

The AGN-201M reactor has such a low fuel burn-up rate that it will never have to be refueled in its useful lifetime. Some approved experiments can require the removal of fuel from the core but these experiments are no longer performed at Texas A&M University.

9.3 Fire Protection Systems and Programs

No permanent of fixed fire suppression system exists in the Zachry Engineering Center. Several CO_2 fire extinguishers are present in the reactor facility including one by the reactor control console and another outside the reactor room in room 61B. The design of the facility limits the effects of fire on the facility. The reactor console is in the corner of the room, a fire starting in the area it could be put out before spreading. The reactor room itself is large and without much equipment, so everything is fairly spread out. If a fire were to occur outside the facility, access to the facility would be limited and the fire would have to pass the thick concrete walls and ceiling of the accelerator laboratory and the reactor room and adjoining laboratory. The doors into both the reactor room and accelerator room are metal so the fire would have little chance of getting through.

Chemicals used by the department are stored inside explosion-proof lockers. During the biannual reactor emergency drill involving outside agencies, one of the local fire department companies responds at the scene. After the drill is over, firemen are given a tour of the facility and the different aspects of the facility are pointed out to them.

One of the Emergency Procedures directly addresses a fire in the facility and the Reactor Emergency Procedure (RE-1) addresses a fire that could be in a nuclear system.

With the core surrounded by 1000 gallons of water and the isolated nature of the reactor room in the Zachry Engineering Center the chances of any damage to the reactor or any reactor components is very small.

9.4 Communication Systems

A public address system is used by the reactor operator to make important announcements in the facility. The microphone is located on the reactor console and the speakers for the system are located in the hallway outside of room 60C and room 135.

9.5 Possession and Use of Byproduct, Source and Special Nuclear Material

The only special nuclear material directly associated with the AGN-201M reactor is the startup source located in access port #3. This is a [redacted] PuBe sealed source and is used to provide neutrons in the reactor. Several other sealed sources are located within the reactor room but these are just stored in the source locker and are under the control of Radiological Safety. Two sealed reactor fuel plates are also stored in the source locker, one from an AGN-211 and one from an

AGN-201. These plates will not be used and will be returned with the rest of the AGN-201M fuel when the reactor is decommissioned.

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Chapter 10: EXPERIMENTAL FACILITIES AND UTILIZATION

The experimental facilities associated with the AGN-201M reactor system consist of the [redacted] the access ports which pass outside the reactor in the graphite reflector, the lead, and water shields.

This particular facility has very little experimental use. The TRIGA reactor is primarily used by researchers at the University for experimental research. The AGN-201M reactor is used for experiments in nuclear engineering classes. The only experiments run using the reactor during classes are from the set of approved experiments for the AGN-201M. Section 3.3 of the Technical Specifications clearly listed all the limitations on experiments at the AGN-201M reactor facility.

If an experiment were proposed to be performed using the AGN-201M, the experimental procedure first would have to be approved by the reactor supervisor. Then, this approved proposed experiment would be presented to the Head of the Department of Nuclear Engineering. Once reviewed and approved by the Department Head, the proposed experiment must be presented to the Reactor Safety Board, if a majority of the Board approve the experiment only then can it be performed. It may only be performed under the direct supervision of the reactor supervisor and with the approval of the Department Head.

10.1 Summary Description

The AGN-201M reactor design has the in-core facility, the glory hole, and in-reflector facility with the access ports. A thermal column may be installed in the AGN-201M. However, this feature has not been used and this area is filled with water.

10.2 Experimental Facilities

The experimental facilities in an AGN-201M reactor design are limited to the glory hole and the access ports. While the center of the glory hole provides a relatively large neutron flux, the remaining area of the glory hole will be filled by either an experiment holder or a poly rod so the generation of radioactive gases is very limited.

10.3 Experiment Review

The experiment review process is a three-level system at the AGN-201M reactor facility. The proposed experiment must be presented to the reactor supervisor for review and consideration. His responsibilities include ensuring all requirements are met for limitations on experiments and reactivity as per the Technical Specifications. This review includes a verification of all reactivity values of the materials and experimental equipment that are exposed to the reactor neutron flux. Special care must be taken to ensure that if any material becomes activated during the experiment that proper radiological controls are taken and that personnel from Radiological Safety are directly involved in evaluating the experiment. The reactor supervisor must ensure that no part of the experiment can lead to damage to reactor systems or injury of personnel in the event of a failure. If the reactor supervisor is satisfied that the experiment can be conducted safely and poses no danger to the general public, he must present the proposed experiment to the Head of the Department of Nuclear Engineering.

The Department Head must then complete an evaluation of the proposed experiment and determine if it is in fact worth the resources to conduct and has no unresolved safety considerations. All the aspects of the experiment will be discussed with the reactor supervisor to ensure no questions remain. The Department Head must then present the proposed experiment to the Reactor Safety Board for consideration. If a majority of the Board approves then the experiment can be conducted.

The experiment can only be conducted with the direct approval of the Department Head and under the direct supervision of the reactor supervisor.

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11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

The radiation protection program at Texas A&M University is the responsibility of the Radiological Safety personnel which is part of the University's Environmental Health and Safety organization. The AGN-201M reactor room is part of the nuclear engineering area but all of these laboratories lie within the physical area of the reactor facility as defined by the Security Plan. The personnel directly involved in the radiation protection program for the AGN-201M are staff of the facility. The reactor operators control access to and control the actions of all personnel in the reactor room.

Radiological Safety personnel conduct an annual radiation survey in association with the AGN's preventive maintenance program. This survey verifies the extremely low levels of radiation present in uncontrolled areas outside the reactor room.

The main concern of the program is to limit exposure to both the students and the general public. The design of the reactor facility and its location in the Zachry Engineering Center has isolated the reactor room from unrestricted access by the general public. The Physical Security Plan does not allow unrestricted access to the reactor facility by the general public. The operators do not allow any personnel in the reactor during reactor operations unless they have some form of dosimetry. The reactor operating procedures do not allow for a reactor startup without verification of proper personal dosimetry.

An important design feature of the facility is the presence of the C-2 warning device on the door to the reactor room. This device will sound a horn if anyone opens the room door whenever the reactor console is energized. As a further limitation on access, the door to laboratory 61A is always locked closed and only personnel assigned to the facility or doing research in that laboratory have a key to the door.

11.2 Radiation Sources

The only radioactive source that is directly associated with license R-23 is [redacted] PuBe startup neutron source located inside access port #3. This sealed source is swiped quarterly and these records are maintained by Radiological Safety personnel. A radioactive source storage locker is located in the corner of the reactor room 61B. All sources contained in this storage locker are under the control of Radiological Safety personnel. Three [redacted] and two [redacted] sealed PuBe sources are stored in this locker, along with one [redacted] Co-60 source. These sealed sources are stored inside containers in the source locker. Inside the source locker are two fuel plates one from an AGN-211, and the other from an AGN-201, these plates are inside sealed aluminum storage holders and will remain inside these holders until a time when the AGN fuel is removed.

Another source locker is located inside the counting laboratory. This locker and laboratory are secured at all times unless in use. These sources are used as check sources in nuclear engineering laboratories and consist mostly of source sets. All of these sources remain inside the locked storage cabinet, inside a locker storage container unless they are required for a laboratory. The teaching assistant or professor must obtain a key to the source locker from the laboratory manager and be briefed on precautions regarding their use.

11.2.1 Airborne Radiation Sources

The only possible airborne radioactive source associated with the AGN-201M could be due to the generation of Ar-41 from a neutron interaction with Ar-40. Analysis of this interaction has shows a minimal production of Ar-41 at this particular type of reactor facility.

11.2.2 Liquid Radioactive Sources

The design of this reactor facility and the experiments conducted do not lead to the production of any liquid radioactive sources. The only possible source is the 1000 gallons of water that surround the reactor in a sealed tank. This water will be analyzed and the level of possible contamination determined as part of a decommissioning plan for the facility.

11.2.3 Solid Radioactive Sources

No solid radioactive sources exist at this facility other than those described in section 11.2 of this chapter. No solid radioactive waste is present at the facility other than small amounts of laboratory waste generated by students during experiments. This radioactive reactor waste is removed by Radiological Safety personnel, but by its very nature is a very small amount of waste. No waste other than laboratory trash-type waste is generated by experiments directly associated with AGN-201M reactor operations and maintenance.

11.3 Radiation Protection Program

The radiation protection program for the AGN-201M is cooperative program between Radiological Safety personnel and the facility staff. The Radiological Safety Officer directs the radiological safety personnel in the workings of the program. Under emergency conditions , the Head of the Department of Nuclear Engineering or his designated alternate is the emergency director for the emergency response personnel at the AGN but gets direct input from Radiological Safety personnel.

The main function of this program to protect the general public under normal and emergency conditions from radiation exposure and ensure that staff personnel and students use the ALARA principle to fullest extent possible to keep exposures low.

The dose to the students is directly controlled and limited by the actions of the facility staff. Any student that is working with the reactor will be issued a dosimeter. If they are classified as radiation workers in the department, Radiological Safety will issue them a suitable personnel monitoring device. If these individuals are nuclear engineering students, the reactor operator will issue pocket ion chambers, obtained from Radiological Safety personnel, prior to reactor operations.

When a nuclear engineering class is scheduled to use the AGN-201M for experiments in the laboratory, the class is lectured about reactor operations and construction by a qualified operator. During this lecture, a tour of the reactor room is performed and through this effort the student will become familiar with the reactor room layout. It is during this tour that the reactor operator points out the location of specific items (i.e. the source locker) of ALARA concern. This lecture will point out that avoidance of these areas will lower the radiation exposure of the students. When the student is to perform their reactor startup, one of the first steps of the pre-startup checklist is to conduct a radiation survey of the room. It is during this time that the student is trained on the location of all survey points and on the practice of ALARA during reactor operations.

Reactor operations during laboratory classes are limited to the lowest power level practical for the experiment. Student access to the reactor room is limited to only personnel necessary to successfully complete the experiment and to operate the reactor. All other students involved in the experiment, are allowed access only to room 61B, this way the students level of exposure is greatly limited.

11.4 ALARA Program

The ALARA program in place at Texas A&M University comes directly from procedures created by the Environmental Health and Safety Department. All of these procedures as well detailed instructions are presented in "Radiological Safety Program Manual" which was last revised in July, 2004. The ALARA Program is addressed in section 5 of the manual.

11.5 Radiation Monitoring and Surveying

At the University personnel from Radiological Safety perform the required radiation surveys and instrument calibrations. The basis for the radiation survey is detailed in the Technical Specifications section 3.4, and specifications and basis for radiation monitoring and control are contained in section 4.4 of the Technical Specifications. Section 4.4 also details the requirements for calibration of all portable radiation survey instruments assigned to the reactor facility by personnel for Radiological Safety.

11.6 Radiation Exposure Control and Dosimetry

Radiation exposure control is a very important concern at the AGN-201M facility. The reactor room is considered to be a radiation area whenever the reactor is operating at a power level of less than 0.9 watts and a high radiation area whenever the reactor is operated at power levels greater than 0.9 watts. These radiation levels raise many ALARA concerns but the reactor is only operated at power levels greater than 0.9 watts when required by maintenance procedures. When students operate the reactor for laboratory reactor startups, the power level rarely exceeds 50 milliwatts.

If irradiations are required for a laboratory only the reactor operator and one student will be in the reactor room and they are shielded by the shielding around the control console. If the activation foils are to be taken to the counting laboratory, they are moved in a lead container and handled with large tweezers and remain in the custody of the reactor operator. When the laboratory is completed, the foils are placed into the source locker inside the reactor room for decay.

All students performing reactor startups are issued pocket ion chambers and students performing experiments in the area around the reactor are also issued pocket dosimeters. These dosimeters are provided by Radiological Safety personnel and issued by the reactor operator just prior to reactor startup.

With the reactor operating for different hours per year, exposure levels in the reactor room over the course of a year due to reactor operations are relatively low. The reactor room has also been used for calibration of portable survey instruments by Radiological Safety personnel. These exposures have added to the dose measured by the local area badges. During years of consistent reactor usage and instrument calibration, the area badges read between 180 and 830 mrem per year of gamma and neutron radiation. Since the calibration of instruments has been moved to another location, the area badge inside the reactor room received 423 mrem in 2010 but the area badge outside the reactor room recieved only 16 mrem.

The only anti-contamination clothing worn at this facility is gloves, these are worn to perform swipe surveys on samples. No respiratory protective equipment is used at the AGN-201M reactor.

The area badges and personnel dosimetry issued to facility personnel are evaluated quarterly. If problems were to occur in this facility, personnel from Radiological Safety would be called in to assist facility personnel in accessing the problem and help in the restoration efforts.

Doses at this facility have been very low even to personnel assigned to the facility.

In calendar year 2010, the largest personnel dose received was 48 mrem. Great care is taken by the facility staff to ensure that all personnel associated with the AGN-201M reactor receive doses which are as low as possible.

11.7 Contamination Control

The reactor facility is swiped monthly by personnel from Radiological Safety. Radiation and contamination surveys are performed on all samples after they are removed from the reactor. With a sealed reactor system, no loose contamination is expected to be found in the AGN-201M reactor facility during normal operations. When the control rods are removed for reactor maintenance or repair, they are surveyed to ensure no loose contamination is present. All the sources stored in the reactor room are sealed.

11.8 Environmental Monitoring

No environmental monitoring, except for area badges, is conducted at this facility.

11.9 Radioactive Waste Management

The only waste generated directly from the reactor operations is simple laboratory trash. Efforts are made by facility staff to limit the amount of waste generated. The total amount is very small and Radiological Safety personnel dispose of this laboratory waste.

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Chapter 12: CONDUCT OF OPERATIONS

12.1 Organization

The AGN-201M reactor is owned by Texas A&M University and has been operated by the University since 1957 under USNRC license R-23, Docket 50-59. Since the reactor is at a university the organization surrounding it is quite unique to this particular university and will be discussed in the following chapter. A detailed description of all duties, responsibilities and requirements are presented in the Technical Specification, section 6.0.

12.1.1 Structure & Responsibility

The organizational structure at Texas A&M University is intended to fulfill the requirement set forth by the USNRC. The organizational structure is presented in flowchart format in Figure 12-1.

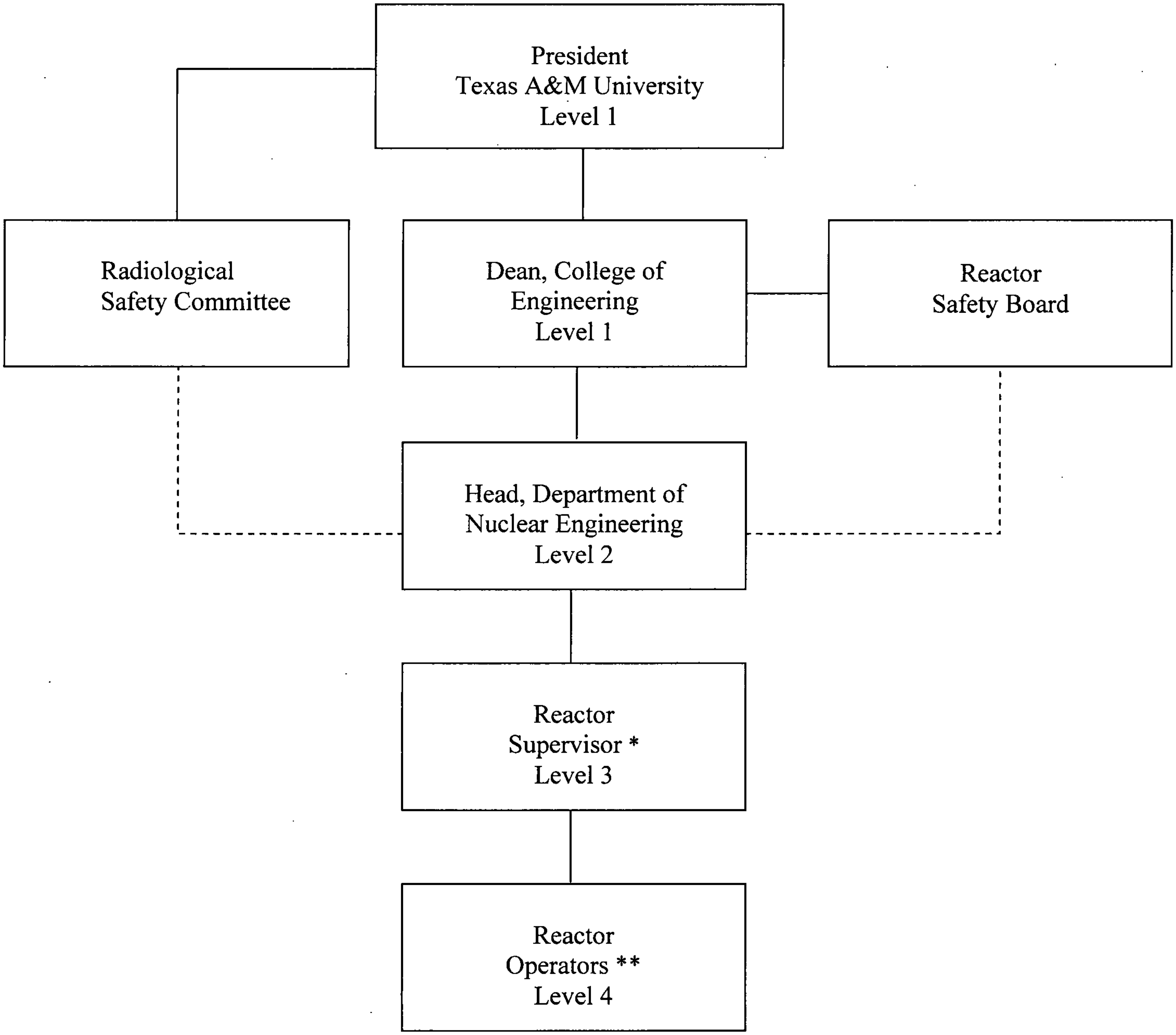
The President of the University is considered to be the chief administrative officer for the facility. The application for licensing of the reactor is made in his name. The Dean of the College of Engineering is the administrative officer responsible for the day to day operations of the College of Engineering and, since the Department of Nuclear Engineering is among those departments, he has an administrative role in the operations of the AGN-201M reactor.

The Head of the Department of Nuclear Engineering is the person administratively responsible for the department. This includes the AGN-201M reactor. The Department Head shall have final authority and ultimate responsibility for the operation, maintenance and safety of the reactor facility. The Department Head shall appoint personnel to the position of Reactor Supervisor and shall select qualified candidates as Reactor Operators. He shall work with members of the Reactor Safety Board and the Radiological Safety Committee to ensure that no unresolved safety questions are present. He shall present new experiments, procedures, and facility modifications to the Board or Committee for their review and approval.

The Reactor Supervisor shall be a licensed SRO and shall be responsible for the preparation, promulgation and enforcement of administrative controls including all the rules, regulations, instructions, and operating procedures to ensure that the reactor facility is operated in a safe competent and authorized manner. He shall direct the activities of operators and technicians in the daily operation and maintenance of the reactor; schedule reactor operations, maintenance; be responsible for the preparation, authentication, and storage of all prescribed logs and operating records: authorize all experiments, procedures, and changes thereto which receive the approval of the Reactor Safety Board and/or Radiological Safety Committee and the Head of the Department of Nuclear Engineering. He shall be responsible for the preparation of experimental procedures involving the use of the AGN-201M reactor.

Reactor Operators shall be responsible for the safe manipulation of the reactor controls, monitoring of instrumentation, operation of reactor related equipment, and maintenance of complete and current records during the operation of the facility.

The Reactor Safety Board is responsible for, but not limited to, reviewing and approving safety standards associated with the use of the reactor facility. The Board shall review and approve of all proposed experiments, procedures, and changes to either. The Board must also approve all reactor facility modifications which might affect its safe operation. A complete listing of the Reactor Safety Boards duties and responsibilities are listed in Section 6.1.6 of the Technical Specifications.



* Requires NRC Senior Operators License

* * Requires NRC Operators License except where exempt per 10 CFR 55 paragraph 55.13

Figure 12-1: ADMINISTRATIVE ORGANIZATION OF THE TEXAS A&M UNIVERSITY AGN-201M REACTOR FACILITY NRC LICENSE R-23

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

12.1.2 Staffing

The staffing requirements for the AGN-201M are as follows

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

1. One licensed Reactor Operator at the reactor control console.
2. One other person in the reactor room certified by the Reactor Supervisor as qualified to activate the 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 performing 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 modifications which could affect the reactivity of the reactor.

12.1.3 Selection and Training of Personnel

The selection of personnel to operate the AGN-201M reactor typically has been restricted to former nuclear navy personnel or graduate students at Texas A&M University. These individuals require less basic training as operators than would other less-qualified personnel. All personnel that operate the AGN-201M reactor, even students, undergo some form of training.

The students are lectured on the construction and operations of the reactor prior to student startups.

12.1.4 Radiation Safety

The Radiological Safety Officer at the University is responsible for all aspects of radiological safety. In this facility, the staff will train the student operators on the importance of the ALARA program and what particular aspects of that program can be used at the facility. As stated above, all questions and reviews concerning radiological safety are presented directly to the Radiological Safety Committee. At Texas A&M University, Radiological Safety personnel have the authority to stop any experiment or activity they believe to be unsafe.

12.2 Review and Audit Activities

The Reactor Safety Board is responsible for reviews and audits at this facility. The review and audit activities of the Board are clearly outlined in Sections 6.4.2 and 6.4.3 of the AGN-201M Technical Specifications. The authority of the Reactor Safety Board comes directly from the office of the President of the University.

The Reactor Safety Board shall review:

A. Safety evaluations for changes to procedures, equipment, systems, and tests or experiments conducted without Nuclear Regulatory Commission approval under provision of paragraph 50.59 to verify that such actions do not constitute a license amendment.

B. Proposed changes to procedures, equipment or systems that change the original intent or use, and are non-conservative or those that do not meet the criteria set forth in 10 CFR 50 paragraph 50.59.

C. Proposed test or experiments which are significantly different from previously approved tests of experiments, or those that do not meet the criteria set forth in 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.

Audits of facility activities shall be performed at least quarterly under the cognizance of the Reactor Safety Board but in no case shall personnel responsible for the item conduct the audit. Deficiencies uncovered that affect reactor safety shall immediately be reported to Level 1 management. A written report of the findings of the audit shall be submitted to Level 1 management and the review and audit group members within 3 months after the audit has been completed. 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

C. The Facility Security Plan and implementing procedures, at least every two years.

D. The Requalification Program and records, at least every two years.

E. Results of actions taken to correct deficiencies, at least annually.

12.2.1 <u>Composition and Qualifications</u>

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. Members will generally be University faculty with considerable experience in their own field of expertise.

12.2.2 <u>Charter and Rules</u>

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

12.2.3 <u>Review Function</u>

Reviews are performed when either the facility staff has a proposed change to a standing item or when some sort of violation has occurred. The review of standing procedures, Technical Specifications, security and emergency plans, etc. is not required but encouraged since some minor changes and updates occur all the time.

12.2.4 <u>Audit Function</u>

The audit function at Texas A&M University is limited to members of the Reactor Safety Board, in special cases, audits may be conducted by others, e.g. qualified individuals from the University, consultants, etc.

12.3 <u>Procedures</u>

There shall be written procedures for the following;

A. Startup, operation and shutdown of the reactor.

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

C. Conduct of irradiations and experiments that could affect the operation or 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 the Technical Specifications.

F. Implementation of the Security and Emergency Plans.

G. Radiation Safety Protection.

This list is taken directly from section 6.6 of the Technical Specifications for the AGN-201M.

Any procedure must be submitted to the reactor supervisor for approval and then to the Head of the Department of Nuclear Engineering. Finally it must be submitted to the Reactor Safety Board for approval and signature of the Chairman.

12.4 Required Actions

Section 6.9.2 of the facility Technical Specifications clearly defines what are considered to be reportable occurrences. In all cases of reportable occurrences, the reactor shall be shutdown.

Reportable occurrences, including causes, probable consequences, corrective actions, and measures to prevent recurrences, shall be reported to the USNRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, an amended license event report shall be completed and reference shall be made to the original report date.

A. Prompt Notification With Written Follow-up.

The types of events listed below shall be reported as expeditiously as possible by telephone and confirmed by facsimile or similar conveyance to the NRC Operations Center no later than the first work day following the event, with a written follow-up report within two weeks. Information provided shall contain narrative material to provide a 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 has reached the set point 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. Error discovered in the transient or accident analyses, in the methods used for such analyses as described in the Safety Analysis Report, or in the bases of the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.
9. Release of radiation or radioactive materials from the facility above allowed limits.
10. Performance of structures, systems, or components that require 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.

12.5 Reports

Routine annual operating reports shall be submitted no later than ninety (90) days following May 31. 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 for the shutdowns and corrective actions 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, paragraph 50.59 which clearly shows the reason leading to the conclusion that the criteria set forth in 10 CFR 50 paragraph 50.59 (c) was followed and that no technical specification change was required.

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

   (a) Liquid Waste (summarized for each release)

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

   (b) Solid Waste (summarized for each release)

      (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 received by facility personnel and visitors where such exposures are >25% of 10CFR20 limits.

If a Safety Limit is violated then the violation shall be reported to the NRC Operations Center, the Director of the NRR, the Reactor Safety Board, the Head of the Department of Nuclear Engineering, and the Reactor Supervisor not later than the next work day. In addition a Safety Limit Violation Report shall be prepared for review by the Reactor Safety Board. This report shall be submitted to the USNRC and the Reactor Safety Board within 14 days of the violation.

12.6 Records

Generally record retention falls into two broad categories, records that need to be retained for a period of at least five years and records that are required to be retained for the life of the facility. The records that are retained in this facility are contained in a series of file cabinets located inside the reactor room.

The following is a list of records to be retained for a period of at least five years, taken from the Technical Specifications section 6.10

Records to be Retained or 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 the license.
2. Records required by the Reactor Safety Board for the performance of new or special experiments.

G. Changes to operating procedures.

Records to be Retained for the Life of the Facility:

A. Records of liquid and solid radioactive effluents released to the environment.

B. Appropriate off-sit 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.

12.7 Emergency Planning

The main function of the Emergency Plan is to provide a a guide by which the facility staff is able to contain and overcome emergency situations. The plan clearly states the role and response of area emergency services and with the procedures in place the severity of incidents associated with the AGN-201M will be limited.

12.8 Security Planning

The Security Plan associated with the AGN-201M is intended to take full advantage of the isolated nature of the facility inside the Zachry Engineering Center. The location of the facility in the building is over two floors but with restricted access to the outer doors, access to the reactor room itself is extremely limited.

12.9 <u>Operator Training and Requalification</u>

A copy of the Requalification Program for Licensed Reactor Operators and Senior Reactor Operators for the AGN-201M has been included in this application for license renewal as Appendix C.

12.10 <u>Environmental Reports</u>

An environmental report has been submitted with this license renewal application and is included as Appendix E.

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Chapter 13: Accident Analyses

13.1 Accident-Initiating Events and Scenarios

13.1.1 Maximum Hypothetical Accident

Studies have demonstrated the Maximum Hypothetical Accident (MHA) involving an AGN-201M reactor system is a large reactivity addition. This analysis is based upon the assumption of an instantaneous increase of 2% $\Delta k/k$ reactivity above delayed critical with the reactor at a power level $\leq$ 5 watts. This situation would not occur under normal situations but for the MHA it is being considered. This instantaneous 2% positive reactivity addition is considered to be the maximum creditable accident for the AGN-201M reactor system.

This scenario has been analyzed using one group theory with one group of delayed neutrons. This analysis was performed by Aerojet-General Nucleonics and was published in August 1957. This simple analysis assumed that: (1) at time zero a 2% step increase in reactivity was inserted with the reactor at a power level of 100 milliwattt and (2) the energy in the core at time zero was negligible in comparison with the energy liberated during the ensuing excursion. Further, this analysis assumed that there was no energy transfer from the core during the excursion. A complete description of the analysis can be found in the Hazard Summary Report, the results are summarized below.

Numerical solution of the governing equations by finite-difference methods yielded a total energy release of 1.7 MJ and raised the average core temperature by 71.3°C. This excursion produced a peak power excursion of 54.4 MW at 204 msec after the reactivity insertion. These results demonstrated that the core material would not sustain catastrophic damage nor release any fission products because the average core temperature remained well below the melting point of the polyethylene matrix.

13.1.2 Insertion of Excess Reactivity

Consider in 13.1.1.

13.1.3 Loss of Coolant

Not considered in the AGN-201M system.

13.1.4 Loss of Coolant Flow

Not considered in the AGN-201M system.

13.1.5 Mishandling or Malfunction of Fuel

Not considered in the AGN-201M system.

13.1.6 Experiment Malfunction

No explosives or corrosives are permitted in the AGN-201M.

13.1.7 Loss of Normal Electrical Power

In the event of a loss of normal electrical power the reactor will scram and be placed in a shutdown condition.

13.1.8 External Events

A large scale meteorological disturbance would certainly interrupt reactor operations. The reactor would be shutdown and placed in a safe condition, if an unforeseen event occurred the construction of the reactor facility would ensure the safety of personnel and the reactor would be shutdown in a controlled manner.

In the event of a large seismic event, the earthquake switch would activate and shutdown the reactor.

The walls of the facility at ground level are constructed of 3.5 feet of steel reinforced Type II concrete. The rest of the structures in the reactor facility around the reactor room and accelerator laboratory are not. If any large impact were to occur the weaker structure would bear the brunt of the damage.

If a large explosion were to occur the construction of the facility would mitigate most of the damage.

13.1.9 Mishandling or Malfunction of Equipment

Operator error is a very important consideration during reactor operation. The design of the AGN-201M reactor system, with all the scrams and associated interlocks, the likely result of operator error would be a scram. The goal of proper operations is not to let an engineered system scram the reactor because the operator has made a mistake.

The most dangerous error could occur if the operator exceeded 5 watts, on a low period. A scram would occur at ≤ 10 watts on Channel No. 3 so no core damage would result. Since the only 5 watt operations of the AGN-201M reactor occur during reactor maintenance procedures, which require controlled reactor operation, this event is unlikely to occur. If reactor period were to go to less than 5 seconds, due to a failure, the reactor interlock system would scram at≤10 watts on Channel No. 3.

If an equipment malfunction were to occur, such as failure of a nuclear instrument the reactor would scram on the interlock relay or the scram system.

13.2 Accident Analysis and Determination of Consequences

(1) In the case of the MHA, initial reactor conditions would be reactor critical, operating just below 5 watts, shield tank temperature at 21°C. Glory hole empty.

(2) The cause would be an instantaneous addition of 2% reactivity [redacted]

(3) Reactor would scram on high power, ≤10 watts, Channels No. 1 & 2 would be off scale high, all the scram lights would be on but with the release of so much energy in such a short amount of time it is possible the core thermal fuse would melt. If this happens the core will separate adding a negative 5% reactivity to the core which will control the reactivity addition.

(4) It has been shown that the reactor fuel would not be damaged but since the core separated some major repairs would be required.

(5) With the presence of the core tank to trap any gases released from the fuel and the separation of the core no releases to the environment would occur. The radiation levels in the reactor room would be very high, but will decrease rapidly.

13.3 Summary and Conclusions

Many studies have demonstrated with USNRC approval that the MHA or the AGN-201M reactor will not melt the polyethylene which surrounds the fuel. Consequently, an insignificant amount of fission products would be released and the direct radiation would only be small fraction of the 10CFR20 limits.

Chapter 14: TECHNICAL SPECIFICATIONS

The Texas A&M University AGN-201M Technical Specifications in their present format were approved by the NRC on April 25, 1979 as Amendment 12 to NRC License R-23 version submitted with this license renewal is the same set of Technical Specifications with a few minor revisions. At this time, these revisions are being submitted to the NRC for approval.

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Chapter 15: FINANCIAL QUALIFICATION

15.1 Financial Ability to Operate a Non-Power Reactor

Texas A&M University, for the last 50 years, has demonstrated the financial ability to own and operate an AGN-201M reactor. The overhead cost of operation of this facility is small. The reactor supervisor is only supported ½ time in this capacity and the other operating cost of the reactor are quite minimal. Based upon Department of Nuclear Engineering records, the operating cost for the AGN-201M in 2010 was about forty thousand dollars.

These costs include salary, student wages, repair parts, and some new equipment for the facility. In the next five years it is estimated that this figure will vary between forty and fifty thousand dollars. All operating capital associated with the AGN-201M reactor is provided by the Department of Nuclear Engineering.

The reactor is not used for any commercial work and no plans have been made to change the operating schedule of the reactor.

15.2 Financial Ability to Decommission the Facility

Texas A&M University is a state supported educational institution that receives the majority of its financial support directly from the state. Texas A&M University has pledged to apply to the State of Texas for whatever financial resources are necessary to decommission the AGN-201M. Texas A&M University has filed a letter of intent, dated July 19, 1990, with the USNRC concern with the decommission funding.

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Chapter 16: OTHER LICENSE CONSIDERATIONS

The AGN-201M reactor at Texas A&M University is not used for medical research and no future considerations are being made for this type of work. The components that presently make up all the systems associated with the AGN-201M contain no prior use components from any other reactors since the license was last renewed in August of 1977

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Chapter 17: DECOMMISSION AND POSSESSION ONLY LICENSE AMENDMENTS

At the current time Texas A&M University has no plans to decommission the AGN-201M reactor. When necessary, facility staff will prepare a preliminary decommissioning plan which will be followed by a complete decommissioning plan for the facility. The University responded to the NRC INFORMATION NOTICE NO. 90-16 with a letter to the Document Control Desk stating that since the University is a state institution that, at the time a decommissioning decision is made, the proper funds would be requested from the State of Texas.

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Attachment D

Response to RAI Item C.
Financial Qualifications

C. Financial Qualifications

1. Responses:

   (a) Texas A&M University is an institution of higher education located in the city of College Station. It is under the management and control of the board of regents of the Texas A&M University System (http://www.statutes.legis.state.tx.us/Docs/ED/htm/ED.86.htm Education Code Title 3. Subtitle D. Chapter 86. Texas A&M University).

   (b) Texas A&M University is not owned, controlled or dominated by an alien, a foreign corporation , or foreign government (http://www.statutes.legis.state.tx.us/docs/ed/htm/ed.61.htm Texas Education Code, Section 61.003 definition of General academic teaching institution and institution of higher education) .

   (c) None of the provisions of 10 CFR 50.33(d) apply to Texas A&M University.

2. Financial Statements for Texas A&M University for the Year ended August 31, 2010 with comparative totals for the year ended August 31, 2009.

   http://www.tamus.edu/apps/soba-afr/disclaimer.aspx?report=SOBAFY2010TAMU

3. Responses:

   (a) Estimated total annual operating costs for each of the first five years of operation of the facility, FY2012 thru FY2016 (the first five-year period after the projected license renewal date). Texas A&M University's source of funding to cover the operating costs for the first five fiscal years.

   Operating Costs $50,000 for 50% Lab Supervisor/Reactor Technician and 25% for Chief Reactor Supervisor salaries. These salaries are currently a part of Texas A&M's state-funded Instruction and General budget, and this will continue to serve as the revenue source for these salaries through FY2015, with no expected raise over the next fiscal years FY2012-FY2013. Assuming a maximum of 2% increase over the 2014-2015 period, the operating costs are projected to be:

FY2012	$50,000
FY2013	$50,000
FY2014	$51,000
FY2015	$52,020
FY2016	$53,060

   (b) Texas A&M University's primary source of funding to cover the operating costs for the above fiscal years will be funded directly by the Department of Nuclear Engineering as identified in the application.

4. Responses:

(a) Estimated Decommissioning Costs (2011 dollars) for each of the years FY2011through FY2016. Texas A&M's source of funding to cover the operating costs for the above fiscal years.

As the last AGN to be decommissioned was more than 15 years ago, there does not appear to be reliable records on the costs associated with such an effort. Therefore, the costs are estimated based on discussions with the University's Safety and Risk Services and Departmental personnel using estimates based on experiences in dealing with mixed, hazardous and radioactive wastes. Based on these conversations, the estimated decommissioning costs in FY2011 dollars are:

Waste Disposal $40,000 (based on estimates and conversations between Departmental Personnel and the waste consolidator and assuming mixed waste)

Fuel Shipment $10,000 (based on 10 Type A container shipments no more than 10 lbs per container air shipped to INL)

Reactor Vessel Disposal $10,000 (based on decontaminated disposal at area landfill)

Labor and Equipment $16,000 (based on skilled labor costs of $75/hour including fringe, overhead, etc. for 20 person days, rental of power washing equipment and torch needed to clean and dismantle tank)

This comes to $76,000 and with a 25% contingency is then $95,000.

These costs are then escalated using an average increase of 3% per year. Inflation has averaged 2.54% over the last 10 years, so a 3% escalation factor was used to conservatively estimate change (according to the CPI-U all urban consumers, Bureau of Labor Statistics, US Department of Labor).

Decommission (in FY2012$)

Waste Disposal	$41,200
Fuel Shipment	$10,300
Labor	$16,480
Reactor Vessel	$10,300
Total	$78,280
25% Contingency	$19,570
Total	$97,850

Costs were calculated in 2011$ and then extended to 2012$ based on an average increase of 3% per year.

(b) Statement of Decommissioning Method to be used.

To comply with Texas A&M University's commitment to the state of Texas and its citizens, the reactor laboratory will be decontaminated to meet the requirements of 10 CFR 20.1402 "Radiological Criteria for Unrestricted Use." The Nuclear Engineering Laboratory (Zachry Building) will remain in place, and the laboratory room will be analyzed and characterized to ensure that it meets the requirements. At the time of decommissioning, the College of Engineering will determine which computer codes and instrumentation are best suited for characterization of the area.

© Description of means of adjusting Decommissioning Costs.

The decommissioning cost estimate will be updated every 10 years beginning in FY2022. The adjusted costs will be based on the average CPI-U for the previous 10 years.

(d) Numerical example showing how the decommissioning costs will be updated periodically in the future.

Adjusted Cost Example for FY2022.

If previous 10 year average CPI was 2.75%, then the total cost will be adjusted to:
$97,850* (1.0275)^10 = $128,345.

5. Statement of Intent to ensure availability of sufficient funds

Department of Nuclear Engineering will provide a Statement of Intent according to Appendix A of NUREG-1757, Vol.3 along with documentation verifying that the signatory is authorized to represent the licensee in providing the statement of intent (signatory should be head of agency or designee), and the amount of the statement of intent equals or exceeds the required coverage level (attached).