Center for Nuclear Studies
Memphis State University

1982
Annual Report
Nuclear Reactor Operations

License R-127, Docket 50-538
AGN-201 Nuclear Reactor, Serial 108

FEBRUARY , 1983

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Center for Nuclear Studies

Memphis State University

1982 ANNUAL REPORT

of

NUCLEAR REACTOR OPERATIONS

AGN-201 Nuclear Reactor, Serial 108
Facility Operating License R-127, Docket 50-538

February 25, 1983

R. L. Dietz
Supervisor
Nuclear Operations

Dr. D. W. Jones
Director

ABSTRACT

The 1982 Annual Report of Nuclear Reactor Operations is prepared in compliance with Technical Specification 6.9 of Appendix A to Memphis State University Facility Operating License R-127, Docket 50-538. The report includes facility operation from January 1 to December 31, 1982.

Reactor operations during 1982 were primarily for the purpose of operator training and no new or untried experiments were performed. The maximum steady-state power level at which the reactor operated was 54 milliwatts. Five unscheduled shutdowns are described in section B of the report and an additional 22 training scrams were experienced throughout the operating year. Safety-related corrective maintenance was required due to failure of the Shield Water Level Float Assembly and is described in section C. A copy of the written report of this event, Reportable Occurrence 82-1, is contained herein as Appendix A. Results of major surveillance tests were satisfactory and included Safety and Control Rod Fuel Capsule disassembly and inspection. Annual reactivity measurements of core parameters were made in November, 1982, and do not significantly differ from data contained in previous reports of AGN-201 operation.

MSU requested and the USNRC issued Amendment No. 4 to the Facility Operating License. This change is described in section D and a copy of the amendment is contained herein as Appendix B. On October 26, 1982, an upgraded facility Emergency Plan was submitted to the NRC pursuant to 10 CFR 50.54(r) and is currently in the review process.

Section K of this report provides a statistical summary of personnel radiation exposures in accordance with 10 CFR 20.407. A description of audits and inspections performed by the MSU Reactor Safety Committee is contained in section L.

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A. REACTOR OPERATING EXPERIENCE

1. Student Training Programs

Five power plant employees participated in training exercises designed by Memphis State University (MSU) to provide research reactor startup experience for cold license candidates. In addition, 69 students from MSU's specialized on-campus Nuclear Skills Training Program performed reactor startups and related training exercises as part of that program's normal curriculum. A total of 199 reactor startups were conducted for purposes of student training.

2. Staff Operator Training

Seventy-seven reactor startups were conducted for purposes of preparing new staff operators for license examinations and maintaining and evaluating licensed operator proficiency. As of December 31, 1982, the MSU Center for Nuclear Studies Staff held two Senior Operator and two Operator Licenses for the AGN-201 Reactor.

3. Additional Operations and Operating Experience Summary

Additional reactor operations were conducted for purposes of satisfying surveillance requirements, routine tests, and calibrations.

Operations Summary

Month	Hours Critical	Max. Power (Milliwatts)	Month	Hours Critical	Max. Power (Milliwatts)
Jan	9.62	32	Jul	0	0
Feb	0.57	48	Aug	31.07	42
Mar	26.37	51	Sep	8.73	41
Apr	4.72	40	Oct	0	0
May	0	0	Nov	18.30	54
Jun	17.28	52	Dec	7.07	52

TOTAL NUMBER OF REACTOR STARTUPS DURING 1982: 295

TOTAL HOURS OF CRITICAL OPERATION DURING 1982: 123.73

B. UNSCHEDULED REACTOR SHUTDOWNS.

Five unscheduled reactor scrams were experienced during 1982. No operating limitations were exceeded nor were any conditions achieved that would have required reactor shutdown, operation of safety systems, or other protective measures required by Technical Specifications. The shutdowns were not considered to be reportable occurrences as defined in the facility operating license.

CHANNEL 1 HIGH LEVEL TRIP (3). On January 14, April 8, and April 9, 1982, high level trips occurred on the Channel 1 Neutron Flux Monitor due to operators not upranging the instrument properly while increasing reactor power. This channel utilizes an eight-step range switch and has a High Level Trip set at 90-95% of full-scale indication to prevent over-ranging the instrument. The highest power level corresponding to the maximum setting of this trip point is approximately 62 milliwatts.

SEISMIC DISPLACEMENT INSTRUMENT TRIP. On April 13, 1982, a Seismic Displacement Instrument Trip occurred during the performance of a radiation survey around the reactor. The survey technician inadvertantly struck the reactor skirt with a neutron survey instrument Bonner-ball at a point immediately adjacent to the seismic displacement sensor. The minor disturbance to this sensitive instrument resulted in a reactor trip from a power level of approximately 48 milliwatts.

CHANNEL 3 HIGH LEVEL TRIP. On April 15, 1982, a High Level Trip occurred on the Channel 3 Neutron Flux Monitor at approximately 10 milliwatts due to the operator improperly positioning the range switch while changing reactor power. This channel utilizes a 20-step range switch and has a High Level Trip set at 90-95% of full-scale indication.

Twenty-two additional scrams were experienced by trainees during the conduct of student training programs and are classified as training scrams.

C. PREVENTIVE AND CORRECTIVE MAINTENANCE

1. Safety Related Corrective Maintenance

On August 11, 1982, the Shield Water Level Float Assembly failed to initiate a protective trip signal during performance of the reactor Prestartup Checkoff procedure. A gradual buildup of corrosion products in the internal operating mechanism of the microswitch due to its location in the humid atmosphere of the Reactor Shield Tank was the cause of failure. The faulty switch was replaced with a new switch of the same type (Microswitch V3-101) and operation immediately returned to normal. This event was reported as MSU Reportable Occurrence No. 82-1 and a copy of this report is included herein as Appendix A.

On October 15, 1982, the Shield Water Level Float Assembly microswitch was replaced with a hermetically sealed model (Microswitch 1-HS-3) as a permanent corrective measure to prevent recurrence of this type of failure.

2. Results of Major Surveillance Tests and Inspections

a. Safety and Control Rod Assemblies Fuel Inspection: This bi-annual surveillance procedure was completed on October 14, 1982. A small quantity of loose fuel particles (< 1 milligrams U-235 content, collectively) was observed in the fuel capsules for the Coarse Control Rod and both Safety Rods. The quantity and circumstances were similar to those reported from the inspections of 1978 and 1980 and are not considered to be the result of abnormal deterioration of the AGN-201 design fission product barriers.

b. Reactor Shield Tank Visual Inspection: This bi-annual surveillance procedure was completed on October 22, 1982. The shield tank water was sampled (gross activity less than minimum detectable, where MDA = 4.7×10^{-8} μCi/ml referenced to Cobalt-60) and drained to permit entry. The Shield Water Level Float Assembly microswitch was replaced with a hermetically sealed model (Microswitch 1-HS-3) and minor areas of spot-rust were cleaned and repainted. The shield tank was re-filled with reactor grade pure water and the float assembly was re-calibrated.

c. Control Rod Drive Assembly Inspection and Lubrication: This annual surveillance procedure was completed on October 28, 1982. The drive assembies were found in satisfactory condition with no evidence of abnormal wear or deterioration.

d. Measurement of Safety and Control Rod Scram and Insertion Times: This annual surveillance procedure was completed on October 28, 1982, with the following results:

	Insertion (cm/sec)	Scram (millisec)
Safety Rod No. 1	0.411	125
Safety Rod No. 2	0.408	110
Coarse Control Rod	0.401	121
Fine Control Rod	0.420	N/A

e. Reactivity Measurements: This annual surveillance procedure was completed on November 12, 1982 with the following results:

Parameter	% Reactivity
Control Rod Integral Worth:	
Fine	0.319
Coarse	1.22
Reactivity Insertion Rate:	
Safety Rod No. 1	0.030/sec
Safety Rod No. 2	0.030/sec
Coarse Control Rod	0.030/sec
Fine Control Rod	0.019/sec
Excess Reactivity (Glory Hole empty, 20°C, all rods IN)	0.202
Shutdown Margin (Most reactive rod IN)	2.65

D. CHANGES IN FACILITY DESIGN, PERFORMANCE CHARACTERISTICS, OR PROCEDURES RELATED TO REACTOR SAFETY

Amendment No. 4 to MSU Facility Operating License R-127 was issued by the USNRC on August 24, 1982, in accordance with MSU's request to cancel changes in the licensed operating conditions of the reactor which had been previously authorized by Amendment No. 1. In order to minimize confusion about the applicable technical specifications, Appendix A of the amended license, NRC reissued the original 1976 Technical Specifications with the Amendment 4 date. A copy of this amendment and supporting safety evaluation is included herein as Appendix B.

There were no changes in facility design or performance characteristics during 1982 that were related to reactor safety.

E. CHANGES WHICH WOULD AFFECT THE FACILITY DESCRIPTION

None

F. CHANGES TO ADMINISTRATIVE PROCEDURES

An upgraded facility Emergency Plan was submitted to the Director, Division of Licensing, USNRC, on October 26, 1982, pursuant to 10 CFR 50.54(r) and USNRC Generic Letter to All Research and Test Reactor Licensees dated June 16, 1982. The proposed plan is in the review process as of the date of this report.

G. NEW OR UNTRIED EXPERIMENTS

None

H. RADIOACTIVE EFFLUENTS

1. Liquid: None
2. Airborne: None
3. Solid: None

I. ENVIRONMENTAL RADIOLOGICAL SURVEYS PERFORMED OUTSIDE THE FACILITY

Areas of unrestricted access begin at the outside walls of the Reactor Room. A general area radiation survey conducted December 15, 1982, revealed the maximum level of gamma radiation to be 0.15 mR/hr measured upon contact with the outside east wall. The maximum level of neutron radiation measured 0.2 mrem/hr at the same location. The reactor was operating at a steady power level of 51 milliwatts for the duration of the survey.

Random wipes/smears of surfaces both inside and outside the reactor facility did not reveal any loose surface contamination above natural background levels.

The reactor was not operated at steady-state power levels above 54 milliwatts during 1982. Therefore, full power radiation survey results are not available.

J. RADIATION EXPOSURES GREATER THAN 100 MILLIREM (50 MILLIREM FOR PERSONS UNDER 18 YEARS OF AGE)

None

K. PERSONNEL EXPOSURE AND MONITORING: 10 CFR 20, Part 407 (a)(2) and Part 407(b)

1. Personnel monitoring was provided for a total of 131 persons during 1982. The highest cumulative exposure for an individual was .092 Rem. The average exposure for the 131 individuals monitored was .0087 Rem.

2. Statistical Summary:

Estimated Whole Body Exposure (Rems)	Number of Individuals In Each Range
No measureable exposure	82
Measureable exposure less than 0.1	49
0.1 to 0.25	0
0.25 to 0.5	0
0.5 to 0.75	0
0.75 to 1.0	0
1 to 2	0
2 to 3	0
3 to 4	0
4 to 5	0
5 to 6	0
6 to 7	0
7 to 8	0
8 to 9	0
9 to 10	0
10 to 11	0
11 to 12	0
12 +	0

L. AUDITS AND INSPECTIONS

Audits and inspections of the AGN-201 Reactor Facility were conducted by the following agencies during 1982.

1. MSU Reactor Safety Committee (RSC):

   a. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety (1-18-82).

   b. The performance, training, and qualifications of the entire facility staff (1-29-82).

   c. The conformance of facility operation to the Technical Specifications and applicable license conditions (2-4-82).

2. Others:

   None

APPENDIX A

FOLLOW-UP REPORT

TO

REPORTABLE OCCURRENCE NO. 82-1

APPENDIX A
(6 pps)

MEMPHIS STATE UNIVERSITY

AGN-201 NUCLEAR REACTOR FACILITY

LICENSE R-127, DOCKET NO. 50-538

FOLLOW-UP REPORT TO REPORTABLE OCCURRENCE NO. 82-1

Date of Report:	August 12, 1982
Date of Occurrence:	August 11, 1982
Initial NRC Notification:	August 11, 1982

1. Reactor.

   AGN-201, Serial 108. Located at the Center for Nuclear Studies, Memphis State University, Memphis, Tennessee. Facility Operating License No. R-127; Docket No. 50-538.

2. Reportable Occurrence.

   Shield Water Level Float Switch Assembly failed to initiate protective trip signal during Prestartup Checkoff. Item 6.9.2.a(5) of the Facility Technical Specifications applies.

3. Conditions at Time of Occurrence.

   a. The reactor was shutdown.

   b. Shield water level was 8 inches below the highest point on the tank manhole opening which is the normal operating level for the system.

   c. Prestartup Checks of the Shield Water Level Safety Channel were in progress per Facility Operating Procedure OP-2.

4. Narrative.

At 9:45 a.m., CST, on August 11, 1982 Prestartup Checks of the AGN-201, Serial 108 Reactor were being performed to verify operability of the Shield Water Level Safety Channel. The check is a Channel Test which consists of manually depressing the water level float to operate an associated microswitch and thereby interrupt the Interlock Line continuity circuit. Interruption of the Interlock Line initiates a protective trip signal by deenergizing the system's main scram relay which removes power from—and/or prevents application of power to—the control rod drive/latch magnets. The Channel Test is performed while the reactor is shutdown and with normal water level in the Shield Tank.

Upon depressing the float to its lowest level, simulating a water level of approximately 9.5 inches below the tank top, the operator did not hear an audible "click" of the associated microswitch which is physically located about 2½ inches above the water line nor was the Interlock line continuity circuit interrupted as observed by the Senior Operator stationed at the control console.

Operation of the reactor was prohibited. The event was reported to the Region II, U. S. Nuclear Regulatory Commission Office of Inspection and Enforcement via telephone conversation between the AGN-201 Reactor Supervisor and Mr. Austin Hardin, and confirmed by mailgram on August 11, 1982 in accordance with Section 6.9.2a of the Facility Technical Specifications.

Maintenance records indicate that the microswitch had been installed by MSU personnel on October 30, 1982 1981 as part of a routine surveillance procedure. The Shield Water Level Safety Channel had been calibrated to initiate a reactor scram at a water level of 9-1/8 inches below the highest point on the manhole opening which conforms to the Limiting Conditions for Operation specified in Table 3.1 of the Facility Technical Specifications. Subsequent Channel Tests required by Item 4.2.d of the Technical Specifications had been satisfactorily performed at least once each month, the trip set-point verified annually, and Channel Tests satisfactorily conducted during approximately 68 Prestartup Checkoffs since the time of switch installation. The most recent satisfactory Channel Test had been performed within 24 hours of the time of failure.

The microswitch was replaced with a new switch of the same type, was tested satisfactorily, and normal operations were resumed at approximately 10:15 a.m., CST, on August 11, 1982.

5. Safety Significance of the Occurrence.

In the event of a shield water leak, failure of the Shield Water Level Safety Channel to initiate a reactor scram at water levels > 10.5 inches below the highest point on the manhole opening during critical operation would have violated the specified Limiting Conditions for Operation. At levels greater than 12 inches below the top of the tank, adequate biological shielding would not be provided during reactor operation as specified by the Safety Limit of Technical Specification 2.1.

At normal operating power, an undetected loss of shield water could increase the gamma dose-rate at the reactor exterior by a factor of 7-8 and increase the neutron background by a factor of several hundred. Compounding an undetected loss of shield water with a nuclear runaway and additional scram circuit failure could result in an exposure of 200-300 Rem of fast neutrons to a person standing next to the reactor. (Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor: Aerojet-General Nucleonics Report No. 23, Revised April 1, 1959.)

6. Redundancy.

There is no design redundancy in the Shield Water Level Safety Channel. However, a loss of shield water would be indicated by the following additional means:

a. Prior to Startup

(1) Shield tank level is verified at the proper level by visual observation and documented as part of each Pre-Startup Checkout.

(2) Visual inspection of areas which would receive shield water leakage is made and documented as part of each Pre-Startup Checkout.

(3) An increase in radiation levels may be observed during Pre-Startup radiation survey.

b. During Operation

(1) An increase in reactor room radiation levels sufficient to activate the facility evacuation alarm would be detected by the area gamma monitor located approximately 6 feet from the reactor.

(2) Shield water levels below $\sim$ 20 inches from the tank top are visible from the control room via a viewing window directly in front of the AGN console and viewing window in the reactor shield tank (Operator is approximately 17 feet from shield tank $32\frac{1}{2} \times 27\frac{1}{2}$ inch viewing window).

7. Cause of Failure.

Gradual buildup of corrosion products in the internal operating mechanism of the microswitch due to its location in the humid atmosphere of the shield tank resulted in the internal switch operating lever being stuck to the ceramic switch housing (Figure 1). Spring pressure, which normally opens the active switch contacts when the float operating arm permits movement of the microswitch plunger, was insufficient to overcome the corrosion effects; thus, the switch contacts remained closed and prevented interruption of the Interlock Line.

8. Corrective Action.

The microswitch was replaced with a new switch of the same type on August 11, 1982.

Manufacturer: Micro-Switch Company; Freeport, Illinois
Type: V3-101, SPDT, 0.5 A @ 125 volts D.C.
Mechanical Life: 10,000,000 cycles
Minimum Release Force: 2 oz.

9. Measures to Prevent Recurrence.

The microswitch casing is of riveted construction which precludes periodic disassembly and cleaning of switch internals. The availability of a model more suitable to a humid environment or a model constructed such that periodic cleaning of switch internals is possible will be investigated.

10. Applicability to Other Equipment in the Reactor System.

None. Microswitches of similar internal construction but with different terminal connectors and actuating mechanisms are utilized for interlocks and position indication devices in the Rod Control System. However, due to their location outside the humid atmosphere which is characteristic of the Shield Tank and due to the greater frequency with which they are exercised during normal reactor operation, it is anticipated that corrosion will not be a problem throughout the mechanical life of the switches.

11. Similar Reportable Occurrences.

MSU Followup Report No. 79-3 dated November 26, 1979, and No. 77-1 dated June 27, 1977.

Prepared by: ____________________
Reactor Supervisor

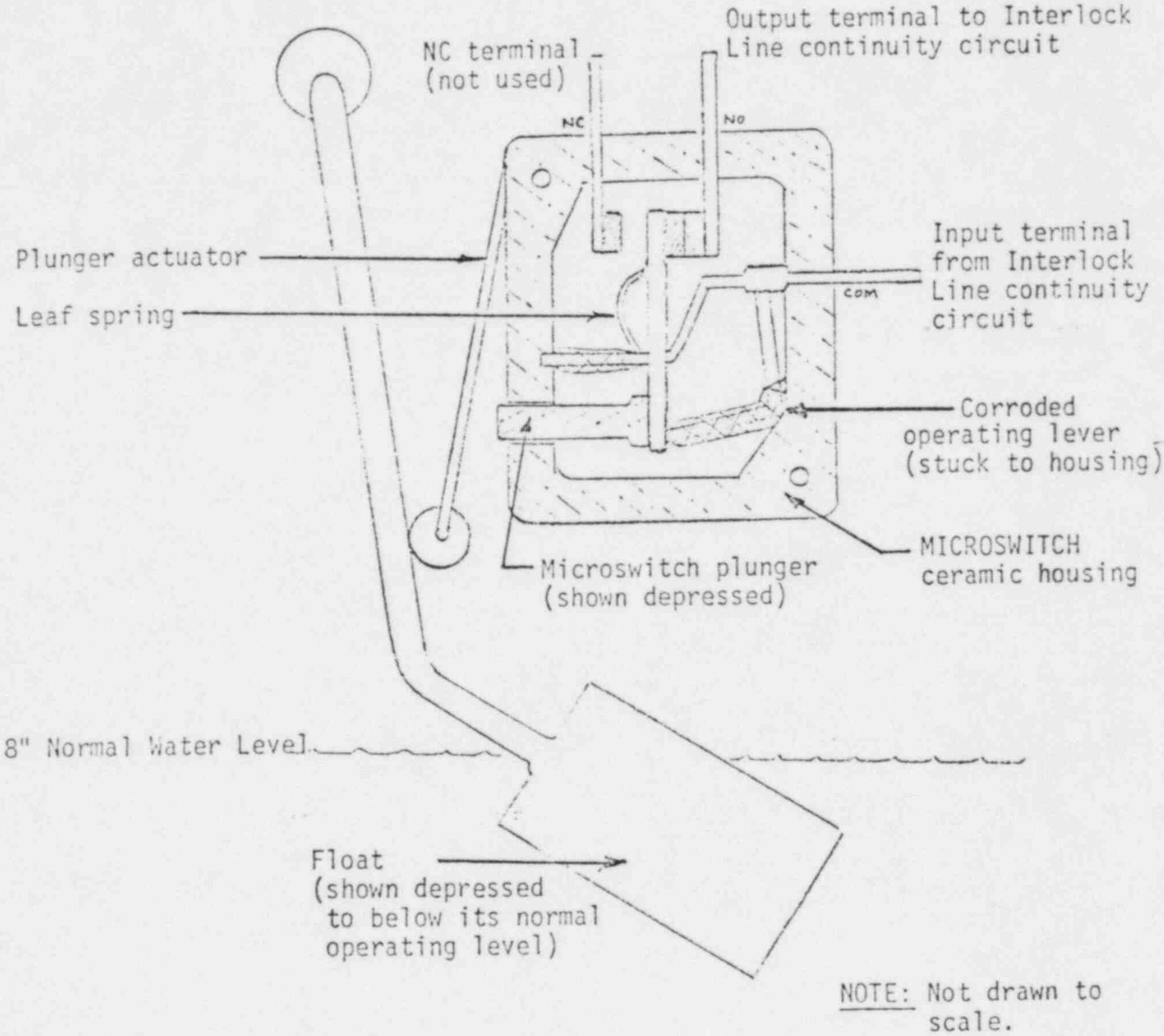

FIGURE 1. Shield Tank Float Switch Assembly (Microswitch shown in failed position)

AMENDMENT NO. 4

TO

MEMPHIS STATE UNIVERSITY

AGN-201 REACTOR OPERATING LICENSE

APPENDIX B
(37 pps.)

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

MEMPHIS STATE UNIVERSITY

DOCKET NO. 50-538

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. R-127

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

   A. The application for amendment by Memphis State University (the licensee) dated May 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.A, 2.C.(1) and 2.C.(2) of Facility Operating License No. R-127 are hereby revised as follows:

   2.A This license applies to the Model AGN-201, Serial No. 108, nuclear research reactor and associated equipment (the facility) owned by Memphis State University. The facility is located on its campus in Memphis, Tennessee, and described in the licensee's application for construction permit and operating license dated April 11, 1975 and amendments thereto.

2.C.(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the reactor at steady state power levels not in excess of 100 milliwatts (thermal).

2.C.(2) <u>Technical Specifications</u>

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

H Bernard

Harold Bernard, Acting Branch Chief
Standardization and Special
Projects Branch
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: AUG 24 1982

APPENDIX A

LICENSE NO. R-127

TECHNICAL SPECIFICATIONS

FOR

MEMPHIS STATE UNIVERSITY AGN-201 (SERIAL 108)

DOCKET NO. 50-538

DATE: AUGUST 24, 1982

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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 Reactor Shutdown - The reactor shall be considered shutdown whenever

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

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

and:

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

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

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

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

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

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

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

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

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

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

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

1.12 Experiment - An experiment is (1) an apparatus, device, or material other than a reactor component, placed in an experimental facility or in line with a beam of radiation emanating from the reactor, or (2) any operation designed to measure reactor characteristics.

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

b. Unsecured Experiment - Any experiment, or component of an experiment is deemed to be unsecured whenever it is not secured as defined in 1.12a above.

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

1.13 The potential reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

The evaluation must consider possible trajectories of the experiment in motion relative to the reactor, its orientation along each trajectory, and circumstances which can cause internal changes such as creating or filling of void spaces or motion of mechanical components.

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

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 fragments are retained in the core matrix.

Specification

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

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

Bases

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

Specification

The reactor shield tank water temperature shall be maintained above 10°C, and the water level in the tank shall not be more than 12 inches below the top of the reactor shield tank.

Bases

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

The shield tank water level of 12 inches below the top of the tank provides adequate biological shielding during reactor operation.

2.2 Limiting Safety System Settings

Applicability

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

Objective

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

Specification

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

Channel	Condition	LSSS
Nuclear Safety #2	High Power	$\leq$ 0.2 watts
Nuclear Safety #3	High Power	$\leq$ 0.2 watts

b. The core thermal fuse shall melt when heated to a temperature of 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 the corresponding shortest possible period is greater than 200 milli-seconds. The high power LSSS of 0.2 watts in conjunction with automatic safety systems and/or manual scram capabilities will assure that the safety limits will not be exceeded during steady state or as a result of the most severe credible transient.

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

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability

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

Objective

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

Specification

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

b. The shutdown margin with the most reactive safety or control rod fully inserted shall be at least 2% $\Delta k/k$.

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

Bases

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

3.2 Control and Safety Systems

Applicability

These specifications apply to the reactor control and safety systems.

Objective

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

Specification

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

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

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

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

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

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

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

d. All reactor safety system instrumentation shall be operable in accordance with Table 3.1 with the following allowable exceptions:

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

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

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

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

Bases

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

TABLE 3.1

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

scram during any operating conditions. Interlocks on control and safety rods assure an orderly approach to criticality and an adequate shutdown capability.

The neutron detector channels (nuclear safety channels 1 through 3) assure that reactor power levels are adequately monitored during reactor startup and operation. Requirements on minimum neutron levels will prevent reactor startup unless channels are operable and responding, and will cause a scram in the event of instrumentation failure. The power level scrams initiate redundant automatic protective action at power levels low enough to assure safe shutdown without exceeding any safety limits. The period scram conservatively limits the rate of rise of reactor power to periods which are manually controllable and will automatically scram the reactor in the event of unexpected large reactivity additions. In order to provide some time to correct channel defects, a maximum of 12 hours is allowed for operation with either Nuclear Safety Channels Nos. 1 or 3 bypassed if the remaining two channels are verified to be operable. Although some redundancy in the reactor protection system is lost during the limited time interval, all scram functions and monitoring capabilities are still available.

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

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

The reactor is designed to withstand 0.6g accelerations and 6 cm displacements. A seismic instrument causes a reactor scram whenever the instrument receives a horizontal acceleration that causes a horizontal displacement of 1/16 inch or greater. The seismic displacement safety channel assures that the reactor will be scrammed and brought to a subcritical configuration during any seismic disturbance that may cause damage to the reactor or its components. Due to the low probability of earthquake damage, the seismic instrument can be out of service for 24 hours in any 3-month period of reactor operation.

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

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

3.3 Limitations on Experiments

Applicability

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

Objective

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

Specification

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

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

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

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

Bases

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

3.4 Shielding

Applicability

This specification applies to reactor shielding required during reactor operation.

Objective

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

Specification

The following shielding requirements shall be fulfilled prior to reactor startup and during reactor operation:

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

b. The thermal column shall be filled with water or graphite. Access to the reactor building roof area above the reactor shall be restricted during reactor operation.

c. Except for radiation surveys, entry to all areas in which dose rate is > 1mr/hr (measured at licensed reactor power) shall be prohibited during reactor operation.

Bases

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

4.0 SURVEILLANCE REQUIREMENTS

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

4.1 Reactivity Limits

Applicability

This specification applies to the surveillance requirements for reactivity limits.

Objective

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

Specification

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

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

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

Bases

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

4.2 Control and Safety System

Applicability

This specification applies to the surveillance requirements of the reactor control and safety 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 time and insertion rates shall be measured annually, but at intervals not to exceed 16 months.

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

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

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

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

   Shield water temperature
   Shield water level
   Seismic displacement

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

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

f. Daily, prior to startup, each of the two safety rods shall be inserted and scrammed to verify operability.

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

h. The shield tank water level and temperature and seismic displacement safety channels shall be calibrated by perturbing the sensing element to the appropriate set point. These calibrations shall be performed annually, but at intervals not to exceed 16 months.

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

Bases

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

4.3 Reactor Structure

Applicability

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

Objective

The objective is to assure integrity of the reactor structures.

Specification

a. At intervals of no more than 5 years, the following inspection and maintenance of core structures shall be performed:

   1. The core thermal fuse shall be inspected. If the inspection indicates the fuse may not be capable of performing its design function, corrective measures shall be taken or the defective fuse shall be replaced to assure the safety function of the fuse will, with a high degree of confidence, be performed. One fuse of the same batch as that of the replacement fuse shall be tested to demonstrate that it meets the requirements of Specification 2.2. The replacement fuse shall not be installed unless the tested fuse complies with this specification.

   2. The core components and structures, including graphite and lead shielding, shall be inspected to assure their functional capability. Major defects shall be corrected prior to reconstitution of the core assembly.

b. 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 startup.

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

Bases

Based on experience with reactors of this type, the 5-year intervals for core component and structure inspections, including the thermal fuse, will provide, with a high degree of confidence, that the core assembly will perform its design function. Similarly, the frequency of inspection and leak test requirements of the shield tank is based on experience with reactors of this type and will assure capability for radiation protection during reactor operation.

5.0 DESIGN FEATURES

5.1 Reactor

a. The reactor core, including control and safety rods, contains approximately 660 grams of U-235 in the form of 20% enriched UO_2 dispersed in approximately 11 kilograms of polyethylene. The lower section of the core is supported by an aluminum rod hanging from a fuse link. The fuse melts at temperatures below 120°C causing the lower core section to fall away from the upper section reducing reactivity by at least 5% $\Delta k/k$. Sufficient clearance between core and reflector is provided to insure free fall of the bottom half of the core during the most severe transient.

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

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

d. The 6½ foot diameter, fluid-tight shield tank is filled with water constituting a 55 cm thick fast neutron shield. The fast neutron shield is formed by filling the tank with 1000 gallons of water. The complete reactor shield shall limit doses to operating personnel in restricted and unrestricted areas to levels less than permitted in 10 CFR 20 under operating conditions.

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

5.2 Fuel Storage

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

5.3 Reactor Building

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

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

c. Access doors to and from the reactor rooms will contain locks. Doors and windows will contain features to preclude entry by unauthorized personnel.

6.0 ADMINSTRATIVE CONTROLS

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

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

6.1.2 VICE PRESIDENT FOR PUBLIC SERVICE/CONTINUING EDUCATION The Vice President is the Administrative Officer responsible to the President for all divisions of Public Service and Continuing Education of the University. In this capacity he shall represent the President in all matters pertaining to the reactor facility except in those cases of health and safety for which the Radiation Control and Safety Committee has authority.

6.1.3 DIRECTOR The Director of the Center for Nuclear Studies is the Administrative Officer responsible to the University for all departments in the Center, including the Reactor Facility and its operation, maintenance, and safety. In this capacity he shall have final authority and ultimate responsibility for the reactor facility and, within the limitations set forth by the facility license, make final policy decisions on all phases of reactor operation; appoint personnel to all positions reporting to him as described in Section 6.1 of the Technical Specifications and as shown on Figure 1 of these specifications be advised in all matters concerning health and safety by the Radiation Control and Safety Committee; and be advised in all matters concerning reactor safety by the Reactor Safety Committee.

6.1.4 REACTOR ADMINISTRATOR The Reactor Administrator is responsible to the Director for the daily administration of the reactor facility. In this capacity, he shall, within the policies set forth by the Director and the facility license, prepare all regulations for the facility, review and approve all procedures, seek approval of all procedures and proposals for changes and experiments from the Radiation Control and Safety Committee, and be responsible for the health and safety of all personnel in the reactor facility. Prior to periods of scheduled absence, he shall designate an alternate and notify the Director.

6.1.5 SUPERVISOR OF NUCLEAR OPERATIONS The Supervisor of Nuclear Operations shall be responsible for licensing of all radiation sources and radiation producing facilities, for the procurement, calibration, and maintenance of all equipment, and the daily administration of all nuclear facilities; except that the authority of the Reactor Supervisor and Reactor Administrator shall supercede that of the Supervisor of Nuclear Operations in all matters related to reactor operation and safety.

6.1.6 REACTOR SUPERVISOR The Reactor Supervisor shall be responsible for the preparation, promulgation, and enforcement of administrative controls including all rules, regulations, instructions and operating procedures to ensure that the facility is operated in a safe, competent, and authorized manner at all times. He shall direct the activities of Operators and Technicians in the daily operation of the reactor; schedule reactor operations and maintenance; be responsible for the preparation, authentication, and storage of all prescribed logs and operating records of the facility; authorize all experiments, procedures, and changes thereto which have first received approval of the Reactor Safety Committee, the Radiation Safety and Control Committee, and the Reactor Administrator, and be responsible for the preparation of all instructional manuals and experimental procedures involving use of the reactor. The Reactor Supervisor shall advise the Reactor Administrator of any scheduled periods of absence.

6.1.7 REACTOR OPERATORS Reactor Operators shall be responsible for the manipulation of the reactor controls, monitoring of instrumentation, operation of reactor related equipment, and maintenance of complete and current records during operation of the facility. The Reactor Operator shall be in direct charge of the reactor console at all times during reactor operation and when the reactor is not secured and conform to the rules, instructions, and procedures established by the Reactor Administrator and Reactor Supervisor for operation of the reactor and the performance of experiments.

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

The Reactor Safety Committee shall hold meetings and have the authority and conduct reviews and audits of reactor operations in accordance with the provisions of Section 6.4 of Amendment No. 6 to the Memphis State University application for construction permit and license to operate the Model AGN-201, serial no. 108, Nuclear Research Reactor at Memphis State University.

6.1.9 RADIATION CONTROL AND SAFETY COMMITTEE The Radiation Control and Safety Committee (RCSC) shall advise the President in all matters concerning the health and safety of personnel who might be exposed to radiation produced by University owned and/or operated sources or equipment. This committee shall review, approve, and promulgate a Radiological Controls Program for the University. This committee shall be informed of all reportable occurrences related to radiation health and safety and reactor safety which are reportable to any authorities outside the University, and advise the President of such occurrences and make recommendations to the President with regard to any such matters.

6.1.10 RADIATION SAFETY OFFICER The Radiation Safety Officer (RSO) shall be the chief administrative officer of the Radiation Control and Safety Committee and represent the committee in matters concerning the radiation safety aspects of reactor operation. He shall prepare the University's Radiological Control manual and have the authority to enforce the regulations, rules, and procedures set forth in the University Radiological Controls manual, suspend the operation and use of radiation producing devices when their use is in violation of these rules, and secure such sources of radiation until corrective action is taken. He shall also have the authority to disapprove the acquisition of radiation producing sources until satisfactory evidence is presented to ensure the safe storage and use of these facilities. The Radiation Safety Officer is also responsible for the reporting of all reportable occurrencesto the appropriate regulatory agency and for ensuring that the appropriate follow up action is taken.

6.1.11 OPERATING STAFF

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

1. Two licensed operators, at least one of whom is licensed as a Senior Operator (SRO).

   (a) One of the two licensed operators shall be at the reactor controls; the other may be on call but shall be within the Center for Nuclear Studies so long as at least one person capable of performing emergency procedures(2), in addition to the operator at the controls, is present in the Control Room.

2. A Radiation Control Technician on call (this requirement may be fulfilled by one of the licensed operators specified above).

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(1) Reactor Shutdown is defined in Section 1.1
(2) Persons capable of performing emergency procedures shall be authorized by the Reactor Supervisor; be familiar with the Emergency Evacuation Plan and capable of initiating the Evacuation Alarm; know the locations and be capable of using emergency and radiation survey equipment; and be capable of operating the facility's Communication and Announcing system.

b. A Senior Operator (SRO) shall supervise all core alterations which can affect the reactivity of the reactor.

c. At least one person qualified to implement routine Radiation Protection procedures shall be present in the facility when any reactor experiment or facility is being serviced.

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

The qualifications of the Reactor Safety Committee members shall be five (5) years of professional experience in the field represented by the member or a baccalaureate degree plus at least two years experience. Generally, these committee members will be made up of University faculty; but outside experience may be sought in areas where additional experience is considered necessary by the Director of the Center. In this case, a baccalaureate degree plus five (5) years experience will be required.

6.3 TRAINING The Reactor Administrator shall be responsible for the facility retraining and replacement program.

6.4 REACTOR SAFETY COMMITTEE REVIEWS, AUDITS, AND AUTHORITY

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

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

6.4.3 REVIEWS The Reactor Safety Committee shall review:

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

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

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

d. Proposed changes in Technical Specifications or licenses.

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

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

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

h. Audit reports.

6.4.4 AUDITS Audits of facility activities shall be performed under the cognizance of the Reactor Safety Committee but in no case by the personnel responsible for the item audited. Individual audits may be performed by one individual who need not be an identified Reactor Safety Committee member. These audits shall examine the operating records and encompass:

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

b. The performance, training and qualifications of the entire facility staff, at least once per 12 months.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety, at least once per 6 to 12 months.

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

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

f. Any other area of facility operation considered appropriate by the Reactor Safety Committee or the Facility Director.

6.4.5 AUTHORITY The Reactor Safety Committee shall report to the Director Center for Nuclear Studies and advise the Reactor Administrator on those areas of responsibility specified in sections 6.4.3 and 6.4.4.

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

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

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

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

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

1. The Reactor Supervisor shall prepare a proposal for review of the Reactor Administrator who shall submit it for approval to the Reactor Safety Committee. The Reactor Safety Committee shall be responsible for review and audit as prescribed in the above Section 6.4. A copy of the findings of this committee shall be submitted to the Radiation Safety Officer for action as required by the University Radiation Control and Safety Committee.

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

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

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

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

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

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

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

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

f. Review and approval of changes to procedures.

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

h. Implementation of the Security Plan and Emergency Plan.

i. Administrative control of operation and maintenance.

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

6.7 EXPERIMENTS

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

b. Each experiment plan shall (1) identify the type of experiment (previously approved or recently reviewed per 6.4), (2) identify the experimenters and (3) have been approved by the licensed senior operator in charge of reactor operation.

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

a. The reactor will be shut down immediately and reactor operation will not be resumed without authorization by the Commission.

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

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

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

6.9 REPORTING REQUIREMENTS In addition to the applicable reporting requirements of TITLE 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the appropriate NRC Regional Office unless otherwise noted.

6.9.1 ROUTINE REPORTS

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

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

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

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

Each annual operating report shall include:

(1) A brief narrative summary of:

(a) Changes in facility design, performance characteristics, and operating procedures related to reactor safety that occurred during the reporting period.

(b) Results of major surveillance tests and inspections.

(2) A tabulation showing the energy generated by the reactor (in general, a monthly tabulation in megawatt-hours and/or hours the reactor is operating will be satisfactory).

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

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

(5) A brief description of:

(a) Each change to the facility to the extent that it changes a description of the facility in the application for license dated April 11, 1975, 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 Commission approval pursuant to 10 CFR 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety question existed and that no technical specification change was required.

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

(a) Liquid Waste (summarized on a 3 month basis)

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

(b) Airborne Waste (summarized on a 3 month basis)

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

(c) Solid Waste (summarized on an annual basis)

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

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

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

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

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

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

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

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

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

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

(4) Reactivity balance anomalies involving:

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

(b) exceeding excess reactivity limit;

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

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

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

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

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

(7) Unscheduled Conditions arising from natural or man-made events that, as a direct result of the event require reactor shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

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

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

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

6.10 RECORD RETENTION

6.10.1 Records to be Retained for a Period of at least five years:

a. Operating logs or data which shall identify:

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

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

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

4. Rod worth measurements and other reactivity measurements.

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by technical specifications.

e. Facility radiation and contamination surveys.

f. Experiments performed with the reactor.

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

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

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

g. Changes to operating procedures.

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

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

b. Appropriate off-site environmental monitoring surveys.

c. Fuel inventories and fuel transfers.

d. Radiation exposures for all personnel.

e. Updated as-built drawings of the facility.

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

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

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

i. Records of meetings of the Reactor Safety Committee.

ATTACHMENT 1

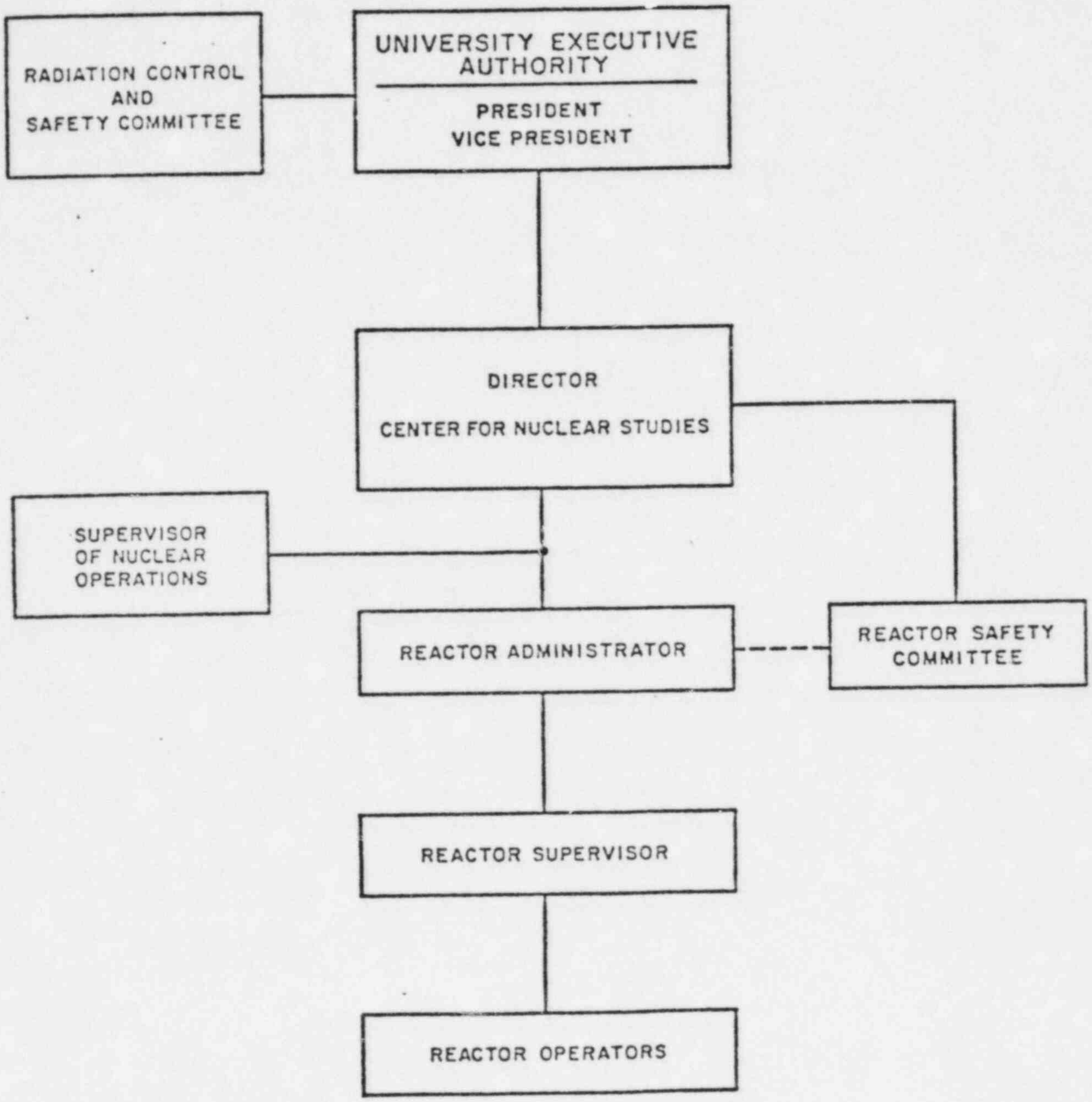

FIGURE I. ADMINISTRATIVE ORGANIZATION PERTINENT TO REACTOR CONTROL AND SAFETY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 4 TO FACILITY OPERATING LICENSE NO. R-127

MEMPHIS STATE UNIVERSITY

DOCKET NO. 50-538

Introduction

By letter dated May 14, 1982, Memphis State University (MSU or the licensee) requested that Facility Operating License No. R-127 be amended to authorize:

(1) Operation of the reactor at steady-state power levels not in excess of 100 milliwatts (thermal).

(2) The licensee to operate the reactor in accordance with the Technical Specifications contained in Appendix A of the original issue of this license dated December 10, 1976.

Discussion

By letter dated April 11, 1975, as supplemented, Memphis State University applied for a permit to construct an AGN-201 training and research reactor facility on the campus in Memphis, Tennessee. The requested Construction Permit was issued by NRC, dated June 15, 1976. Following satisfactory completion of construction, as authorized, and acquisition of a formerly used AGN 201 reactor from the Argonne National Laboratory, Memphis State was issued Facility Operating License No. R-127 on December 10, 1976. Among other license conditions, this license authorized Memphis State to operate the reactor at steady-state power levels up to 100 milliwatts. The license also included, as Appendix A, a set of Technical Specifications which provided for the safe operation through technical performance standards and management controls. This maximum authorized power level was typical of several other AGN-201 reactors already licensed by NRC and in operation at that time.

After approximately two years of uneventful routine operation of the reactor, as authorized, Memphis State University requested by letter dated March 23, 1979 an amendment to License No. R-127. The request proposed that the reactor be authorized to operate at steady-state power levels up to 20 Watts (thermal) with intermittent 1000 Watt operation. Appropriate changes in the Technical Specifications were also required and requested. In order to operate safely at these higher power levels, some modifications to instrumentation and shielding would be required, and the licensee promised to make the modifications before implementing the proposed license conditions. The applicant justified and supported the requested changes in operating parameters in two ways:

(1) A licensed AGN-201 reactor had been modified in a similar way and operated at the higher power levels for approximately 8 years at the U. S. Naval Post Graduate School.

(2) The applicant provided a Safety Analysis of the operation of the reactor in the upgraded modes.

The NRC found these justifications acceptable and issued Amendment No. 1 to license No. R-127, dated March 28, 1980. Furthermore, in order to acquire spare fuel for the reactor, the licensee had requested, and in the same amendment (No. 1) NRC approved, an increase in authorized Special Nuclear Material.

During the two years since Amendment No. 1 was issued, the licensee has continued to operate the reactor in accordance with the conditions of the initial license but has not found the funds, nor the timely opportunity to make the modifications required for operation at the higher authorized power levels. On the other hand, the Technical Specifications approved as part of Amendment No. 1 anticipated early modification, and contain some parameters, limitations, and instrumentation which do not apply to the unmodified AGN 201. When this status was brought to the licensee's attention by inspection personnel from USNRC Region III, the licensee applied to NRR to have the license and Technical Specifications changed back as they were in the initial license, before Amendment No. 1.

Evaluation

(1) The licensee's request involves no changes in instruments, equipment, operating conditions, surveillance, or management controls. The reactor would continue to be operated as it has been since the initial license was issued in 1976. Therefore, the staff considers that the amendment is purely administrative in nature, and no unreviewed technical, safety, or environmental issues are raised. Accordingly, the staff concludes, based on the consideration discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by continued operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

(2) The licensee has requested that the license, as amended, continue to authorize the possession and use of up to 1400 grams of U-235, enriched to less than 20%. Since the excess reactivity loaded into the core is governed by the Technical Specifications, and not by the amount of enriched uranium authorized or in inventory, approval of the request does not involve a safety consideration,

but is administrative in nature. Furthermore, Amendments No. 2 and 3 to License No. R-127 continue to provide for adequate safeguarding of the 1400 grams of U-235 when acquired. Therefore, the staff finds acceptable the request for continuation of the authorization to possess and use not more than 1400 grams of the low enrichment U-235.

Dated: AUG 2 4 1982