APPLICATION

FOR

AMENDMENT NO. 1

TO

MEMPHIS STATE UNIVERSITY FACILITY OPERATING LICENSE

NO. R-127, DOCKET 50-538

Submitted by Billy M. Jones

Billy M. Jones
President
Memphis State University

STATE OF TENNESSEE
COUNTY OF SHELBY

Billy M. Jones being duly sworn, states that he is President of Memphis State University; that he executed this document for the purpose set forth; that the statements made herein are true to the best of his knowledge, information and belief; and he is authorized to execute this document on behalf of said University.

Sworn and attested this day before me July 31st, 1979.

John William Cotton
Notary Public

My commission expires March 5, 1983

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ILLUSTRATIONS

I. GENERAL INFORMATION

A. Introductio... Pursuant to the Code of Federal Regulations, Title 10, Chapter 1, Part 50.90, Memphis State University (MSU) hereby applies for amendment to its Class 104 (c) Facility Operating License No. R-127, Docket 50-538, to:

1. Operate the Model AGN-201, serial 108 Nuclear Reactor at continuous power levels up to and including 20 Watts (thermal) and intermittently at power levels up to and including 1000 Watts (thermal) and,

2. transport up to 700 grams of contained U-235 between Oak Ridge National Laboratories, Oak Ridge, TN and the Memphis State University, South Campus, Memphis, TN as supplemental fuel loading for the reactor and,

3. receive and possess up to 1400 grams of contained U-235 in connection with operation of the facility.

Approval of reactor operation at the requested power levels is essential to the performance of scheduled research which requires radiation doses in excess of those obtainable under the current operating license and, in general, will significantly increase the overall educational and research capabilities of the AGN-201 Reactor. Approval of transport, receipt, and possession of an additional 700 grams of contained U-235 is necessary to ensure the continued availability of replacement fuel which was removed from an AGN-201 reactor that has been scrapped, and which is in temporary storage at Oak Ridge National Laboratories.

If approved, the existing facility will be modified as described in part II of this application.

B. The Applicant.

General information required by 10CFR50.33 concerning the applicant, except as shown below, is contained in the Application for Construction Permit and License to Operate the Model AGN-201, Serial 108 Nuclear Reactor at Memphis State University dated April 11, 1975, as amended (see Docket 50-538).

1. Earliest date for completion of alteration: 9/1/79

2. Latest date for completion of alteration: 11/30/79

II. PROPOSED FACILITY MODIFICATIONS

A. Introduction. The core of the AGN-201, serial 108 reactor is composed of a homogeneous mixture of approximately 20% enriched UO_2 in polyethylene surrounded by a graphite reflector and lead and water shielding. The assembly is contained in a reactor tank, 6½ feet diameter by 9½ feet high, and is located in the Reactor Room of Building 113, MSU South Campus. A control console is connected to the reactor assembly by instrumentation and control cabling and is located approximately 14 feet from the reactor in a separate Reactor Control Room. The existing facility arrangement is shown in Figure II-1. A more complete description of the Reactor Assembly and control console is given in references a and b.

The MSU Reactor has been operated for approximately eighteen years (Argonne National Laboratory, 1957 - 1972; MSU, 1976 to present) with no apparent indication of fuel deterioration, and has not been operated at power levels greater than 0.1 Watt. By the addition of concrete shielding and by extending the instrumentation, other AGN-201 reactors have been operated at continuous power levels up to 5 Watts with no significant increases in operational hazards over the past twenty years. Moreover, by modifications similar to those proposed in this application, the AGN-201, serial 100 Reactor was successfully operated at continuous power levels up to 20 Watts, and intermittently at power levels up to 1000 Watts by the U.S. Naval Post graduate School (USNPGS) at Monterey, California for a period of approximately eight years (references c, d, e). This reactor was subsequently transferred to the California Polytechnic Institute where it has been satisfactorily operated at power levels up to 0.1 Watt since 1973.

Experience with the serial 100, AGN-201 Reactor at USNPGS has shown that a maximum apparent core temperature of 34.2°C was achieved for a single operation commencing at room temperature and remaining at 1000 Watts for nine minutes. Core temperatures versus time at various power levels are summarized in reference e. Operations at power levels greater than 20 Watts also resulted in gas diffusion from the AGN-201 fuel for periods as long as two-to-three days following such operations. The evolved gas was predominantly hydrogen, containing detectable concentrations of fission product gases, and was estimated to result in a reactivity loss of (.0135 ± .001) % per KW-hr of operation. The increased neutron flux in void areas such as the AGN-201 Glory Hole (when empty) and the reactor skirt area produced measureable concentrations of Argon-41 (References d, e).

Proposed modifications described in subsequent sections of this application consider the increased operational concerns described above, as well as necessary additional shielding and radiological controls, and an extension of existing instrumentation necessary to safely operate the AGN-201, serial 108, reactor at the power levels requested.

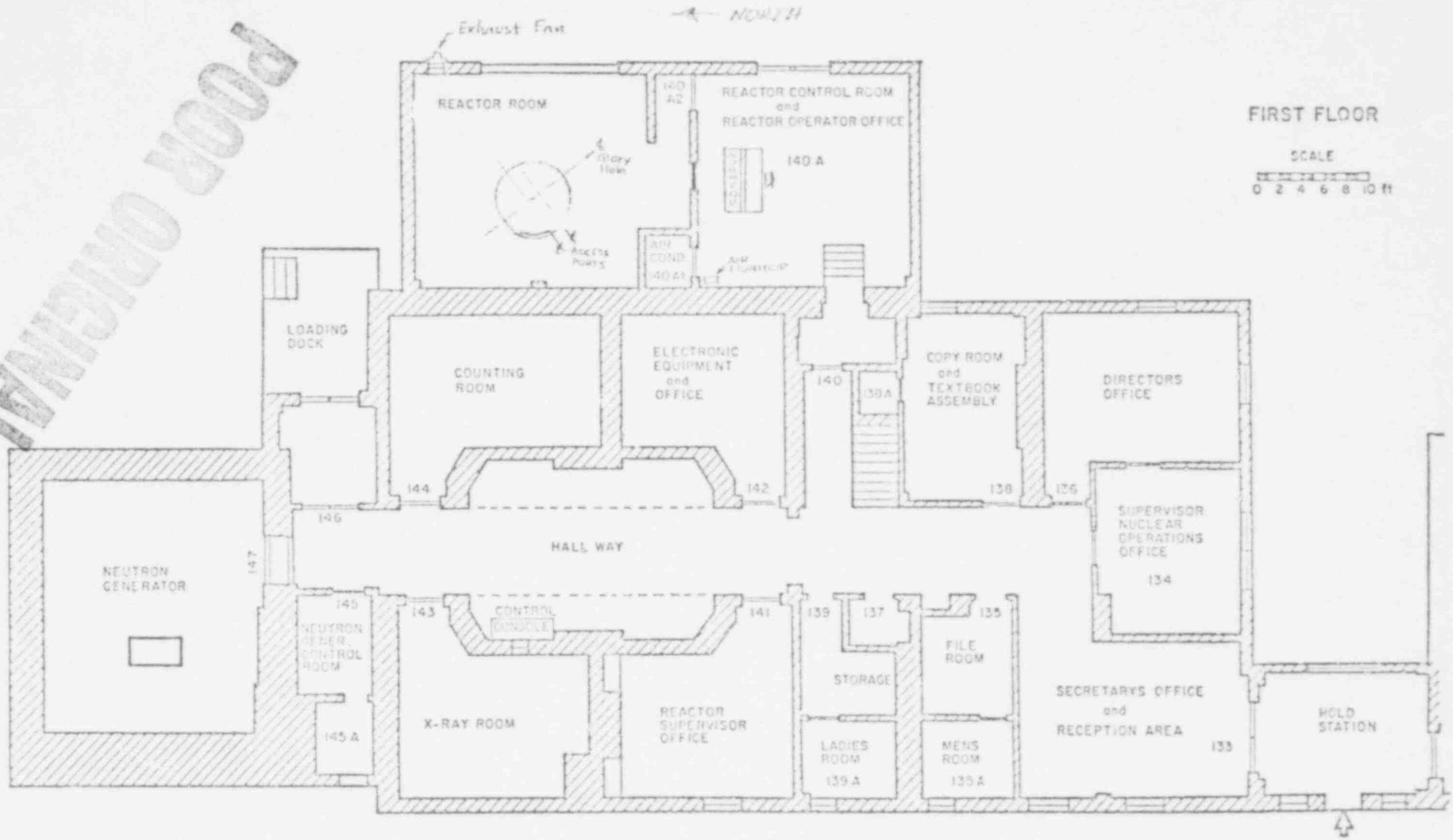

FIGURE II-1. AGN-201 REACTOR FACILITY (existing)

B. AGN-201, Serial 108 Reactor Modifications.

1. Core Tank Assembly (see Figure II-2).

The lower core-half support assembly (aluminum rod) will be replaced by a longer support assembly which will extend through the lower core tank cover and which will accommodate installation of a temperature sensor. The penetration through the core tank cover will be sealed to maintain gas tight integrity of the core tank. The new support assembly will be sufficiently long to extend through the bottom of the AGN-201 Reactor Tank and into the rod drive cavity.

An adapter will be fabricated which penetrates the bottom core tank cover, and which will interface with the Gas Handling System discussed in part II.B.4 of this application. The penetration through the core tank cover will be sealed to maintain gas tight integrity of the core tank. The adapter will be sufficiently long to extend through the bottom of the AGN-201 Reactor Tank and into the rod drive cavity.

The core tank assembly will be pressure tested to at least 6 psig to verify gas-tight integrity following installation of the proposed alterations.

2. Shield Water Tank Assembly.

a. The Reactor Tank and bottom lead shield only have penetrations to accommodate the four Safety and Control Rod thimbles. Two additional holes will be bored through the lead shield and Reactor Tank to accept the extended lower core-half support assembly and the Gas Handling System Adapter discussed in part II.B.1. These additional penetrations will not violate the secondary gas seal established by original AGN-201 design.

b. The existing Shield Water Tank has only one instrument tube (4¼" I.D. X 17"). The Channel 1 and 2 neutron detectors are located in Access Ports 4 and 1 respectively. To accommodate compensated ion chambers for Channels 2 and 3 (estimated overall length 24 inches), new instrument tubes (aluminum) will be fabricated. The Channel 1 detector (BF_3 proportional counter) will remain in Access Port No. 4.

c. The Red Warning Light mounted on top of the Shield Water Tank will be removed to eliminate interference with the top shield proposed in part II.C. This light serves no useful purpose under the current facility configuration.

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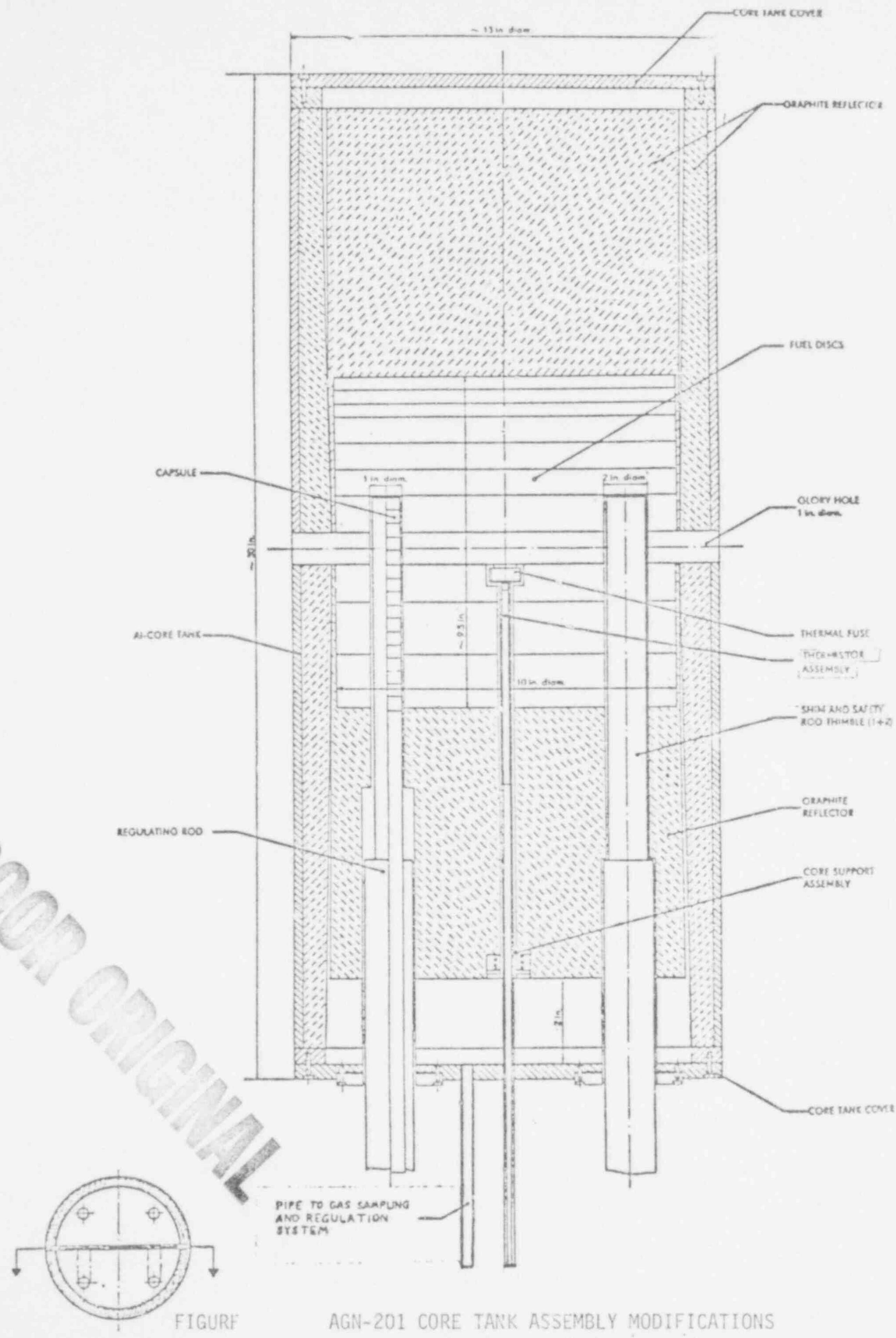

FIGURF AGN-201 CORE TANK ASSEMBLY MODIFICATIONS

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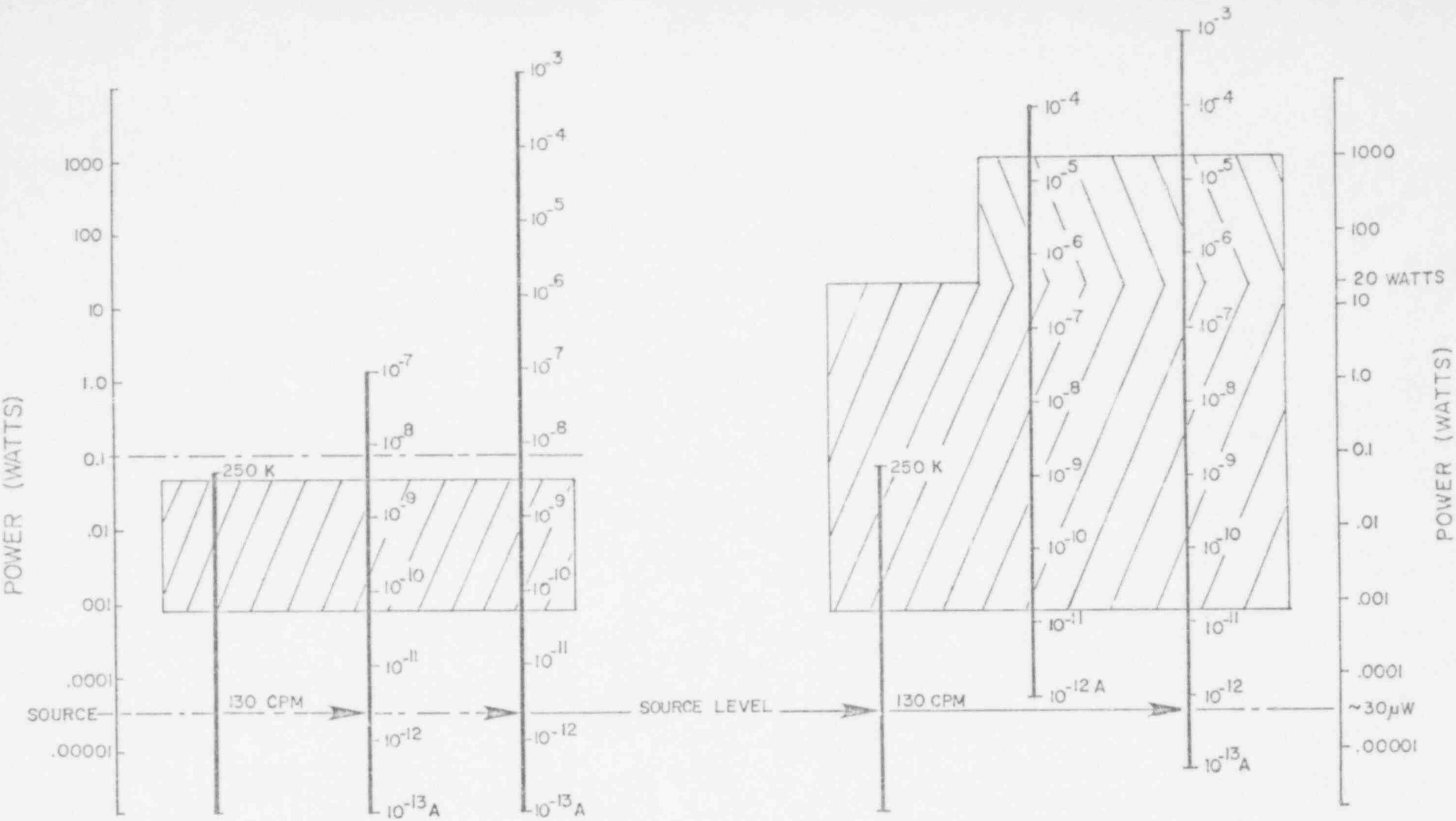
3. Instrumentation and Safety Systems (Figure II-3).

a. Nuclear Safety No. 1. The existing system utilizes a Tracerlab SC-34A (0-250K CPM) Countrate Instrument with BF_3 proportional counter and preamplifier. The system provides indication from below source level up to approximately 55 milliwatts of reactor power and provides a Low Level Scram for countrates below 120 CPM. A redundant High Level Scram function is also provided at 90-95% of full scale meter deflection.

The redundant High Level Scram will be eliminated. The existing high voltage switch for the Channel 1 neutron detector will be modified such that it will simultaneously disable the Low Level Scram function by insertion of a dummy electrical signal and remove high voltage from the detector. This operator action will be controlled by operating procedures which will permit high voltage removal above 90% full scale meter deflection and require reenergization below 40 milliwatts. An automatic time delay will be incorporated into the Low Level Scram disabling circuit to ensure on-scale indication prior to removing the dummy signal and re-arming the Low Level Scram function. The time delay will be necessary to prevent a scram from occurring simultaneously with reenergizing the proportional counter high voltage.

b. Nuclear Safety No. 2. The existing system utilizes a Keithley - 412 Log Picoammeter (10^{-13} - 10^{-7} AMPS) with an uncompensated ion chamber. The system provides Log-N indication from below source level up to approximately 1.6 Watts of reactor power and provides an analog signal for remote Reactor Period measurement. Nuclear Safety No. 2 provides a reactor Low Level Scram at source levels less than 0.5 X 10^{-13} AMPS, a High Level Scram at power greater than 0.2 Watts, and a Period Scram at periods less than 5 seconds.

The picoammeter will be replaced with a new Keithley-25012 Log-N Period Amplifier (10^{-12} - 10^{-4} AMPS), or equivalent, and the uncompensated ion chamber will be replaced with an equivalent compensated ion chamber. This system should provide Log-N and period indication from approximately 45 micro Watts (slightly above source level) to 4.5 Kilowatts of reactor power. The existing period measurement circuit will be removed and the period scram circuitry modified to interface with the new Log-N period amplifier. The existing Channel 2 Low Level Scram will be eliminated since on-scale indication may not be achieved until after reactor startup has commenced.



(A) EXISTING SYSTEM

Channel 1: Linear countrate (0-250K CPM) W/BF_3 P.C.

Channel 2: Log-N Picoammeter (10^{-13}-10^{-7} AMPS) W/U.I.C.

Channel 3: Linear Picoammeter (10^{-13}-10^{-3} AMPS) W/U.I.C.

Present range of critical operation

(B) PROPOSED SYSTEM

Channel 1: Linear countrate (0-250K CPM) W/BF_3 P.C.

Channel 2: Log-N Picoammeter (10^{-12}-10^{-4} AMPS) W/C.I.C

Channel 3: Linear Picoammeter (10^{-13}-10^{-3} AMPS) W/C.I.

Normal range of critical operation

Intermittent operation

FIGURE II-3. INSTRUMENTATION RANGES

c. Nuclear Safety No. 3. The existing system utilizes a Keithley-410 Linear Picoammeter (10^{-13} -10^{-3} AMPS) with an uncompensated ion chamber. The system provides linear indication from below source level to approximately 16 kilowatts of reactor power. Nuclear Safety No. 3 provides a Low Level Scram at source levels less than 5% of full scale and a High Level Scram at power greater than 0.2 watts.

The uncompensated ion chamber will be replaced with an equivalent compensated ion chamber. A mechanical interlock will be incorporated with the existing range selector switch to prevent exceeding the 10 x 10^{-7} AMPS scale (approximately 30 watts) except for approved high power operation and instrument calibration. This operator action will be controlled by operating procedures which will only permit removal of the interlock for approved operations above 20 watts, or for instrument calibration when the reactor is shutdown. No changes in the scram circuitry are anticipated for Channel 3.

d. Interlock Line. A High Core Tank Pressure trip function will be added to the existing Interlock Line continuity circuit. This function will be provided by a pressure-sensitive transducer which will be located in the Gas Handling System discussed in part II.B.4 of this application. The sensor will be installed such that the Interlock Line will be interrupted and an automatic reactor scram initiated at core tank pressure greater than 5 psig.

e. Temperature Monitoring. The present system only provides local indication of Shield Water Tank temperature and utilizes a well-type bimetallic thermometer. An additional temperature monitoring system will be added to monitor in-core temperatures and shield water temperature of the reactor assembly. The system will utilize thermistor or thermocouple sensor elements with a common remote readout instrument located at the control console. The sensor element(s) used for the in-core monitor will be selected from materials that would result in a negligible reactivity effect on the reactor core.

The core temperature sensor will be installed into the lower core-half support assembly previously discussed in part II.B.1 of this application. Since the sensor will be in contact with the aluminum tube which supports the core fuse rather than in direct contact with the fuse, it is anticipated that the indicated temperature will be slightly less than actual fuse temperature while temperature is rising (Reference e).

Interconnecting wiring from the core temperature sensor to the console will be routed through the existing conduit between the Rod Drive Assembly cavity and the reactor skirt area. Thus, the original AGN-201 design secondary gas seals will not be violated.

4. Gas Handling System (Figure II-4).

Due to radiation damage in polyethylene, hydrogen evolution with detectable concentrations of fission product gases are expected to result from reactor operations greater than 20 watts. Thus, a system capable of monitoring, sampling, and handling these gasses will be necessary to ensure that compliance with 10CFR20 can be accomplished as well as to prevent formation of hazardous concentrations of hydrogen. A means must also be provided to permit air samples to be drawn from within the reactor shielding for the purpose of determining Argon 41 concentrations which may accumulate in the AGN-201 Reactor Skirt area and other void spaces during high power operations.

A Gas Handling System will be fabricated to perform the following functions:

a. Maintain a dry nitrogen blanket in void spaces of the AGN-201 Core Tank Assembly.

b. Monitor core tank pressure.

c. Interrupt the Interlock Line Safety System and thereby initiate a reactor scram for core tank pressure greater than 5 psig.

d. Prevent core tank overpre urization.

e. Provide sample capability for reactor skirt area and other void spaces within the reactor shield.

f. Provide radiation detection capability for gases within the handling system.

g. Provide gas discharge capability.

It is anticipated that MSU will fabricate and use a system similar to that shown in Figure II-4 and which is discussed in detail in reference f. The system will interface with the core tank bottom cover adapter discussed in part II.B.1 of this application. Gases that may be released from this system will be routed via a permanent extension of the system shown in Figure II-4 to an existing exhaust fan in the east wall of the Reactor Room. This fan is approximately 17 feet above ground level and the discharge flow path is directly from the east wall of the room, approximately 3 feet below the roof and 3 feet from the northeast corner.

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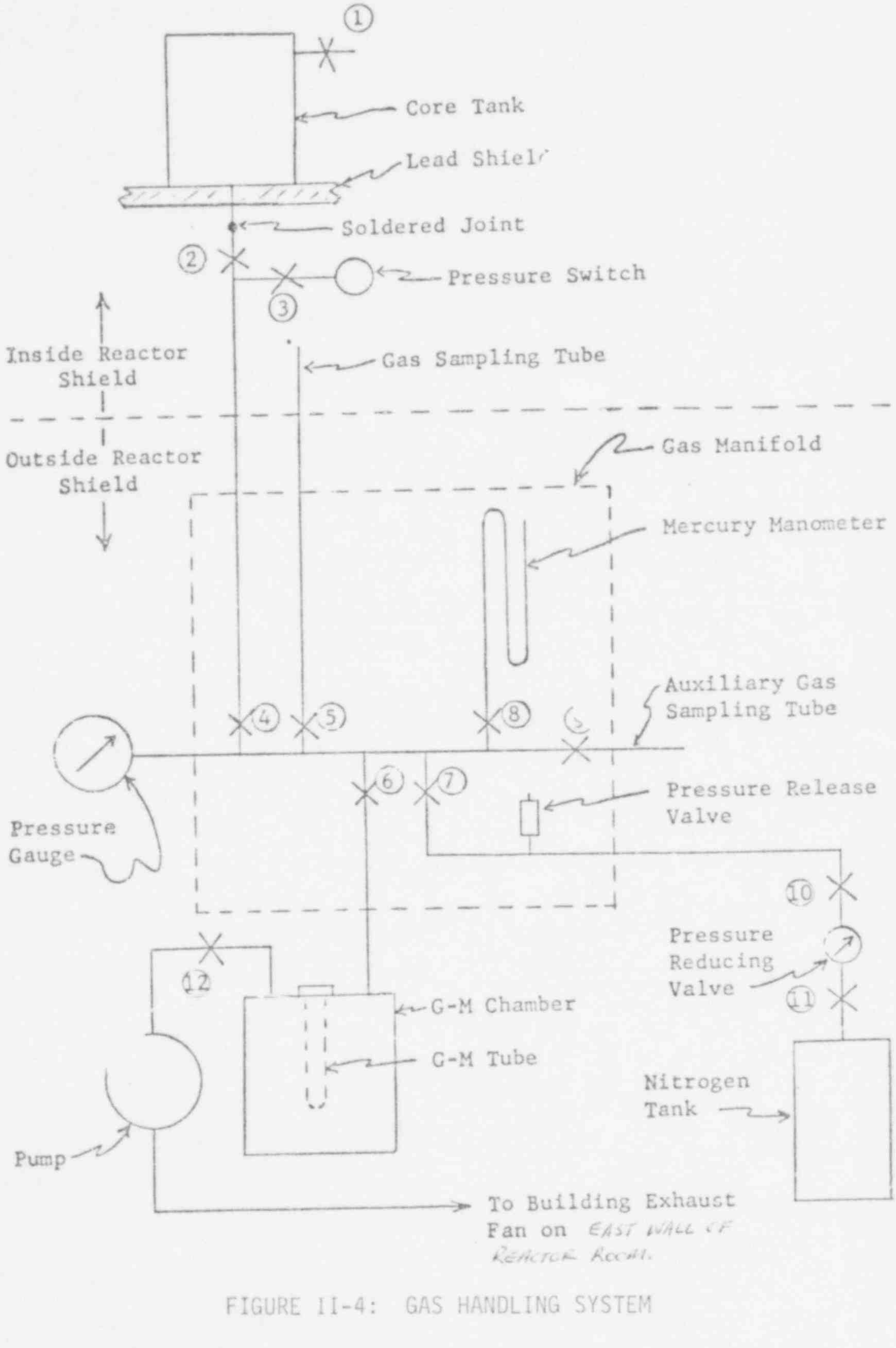

FIGURE II-4: GAS HANDLING SYSTEM

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The existing facility utilizes a fixed Air Particulate Sampling system which continuously monitors and returns air in the reactor room. This system provides audible and visual alarms. MSU intends to supplement the capabilities of this system by purchasing a portable high volume sampler, and by developing a portable gaseous activity sampling capability.

5. Safety and Control Rod Drive Assemblies (Figure II-5).

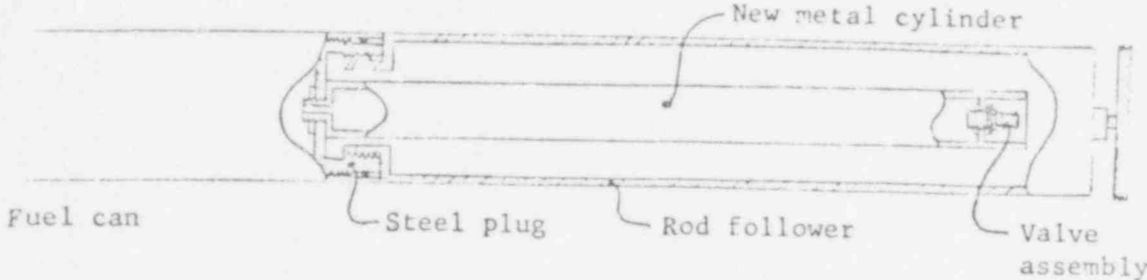
In order to provide additional free void volume for accumulation of gases from the Safety and Control Rod Fuel Capsules, and to provide a method for disposing of these gases, modifications to the fuel capsules and rod follower tubes for each of the rods are necessary. MSU proposes to modify the AGN-201, serial 108 reactor Rod Drive Assemblies similar to the manner shown in Figure II-5 (a), and to fabricate a Gas-Release Tool similar to that shown in Figure II-5 (b). The modifications should not interfere with nor in any way alter the existing Control and Safety Rod motion and response times. These alterations are discussed in more detail in references e and g.

C. Reactor Building and Shielding (Figures II-6 and II-7).

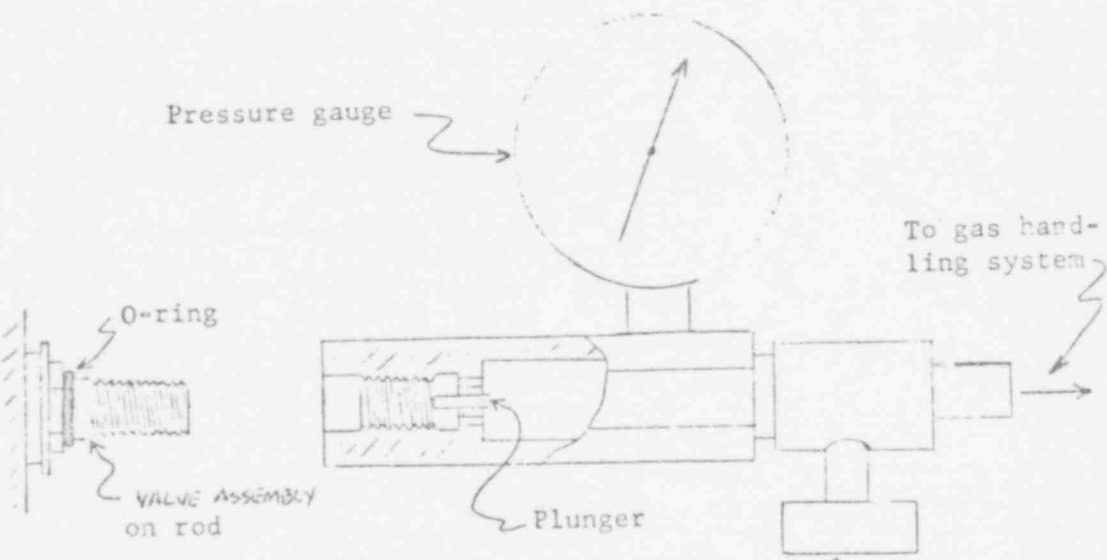
The existing concrete block partition shown at the southeast end of the reactor room in Figure II-1 will be removed in order to provide greater freedom of access to the AGN-201 experimental facilities.

In addition to the lead and water shielding provided by the existing AGN-201 design, a cylindrical shield composed of both borated and ordinary concrete block will be constructed around the reactor assembly (Figure II-6). The shield will consist of two concentric cylinders of blocks, one cylinder compressed by borated concrete, and arranged so that seams will overlap to minimize radiation streaming and so that a total wall thickness of 44 inches is provided. Penetrations through the shield will be made in order to extend the Glory Hole and Access Port liner tubes out to the shield face. These extensions will normally contain shield filler plugs to minimize streaming. A removable shield plug at the base of the shield will provide access to the reactor.

The cylindrical shield will have a top support and shield assembly consisting of aluminum angle and flatbar frame, approximately 18 inches of borated paraffin, and a wooden walking deck. This top shield will also contain a removable shield plug approximately 5 feet in diameter to provide access to the reactor. Seams will be arranged to minimize radiation streaming.



(a) Proposed Rod Assembly



(b) Rod-Gas-Release Device (Portable)

FIGURE II-5. SAFETY AND CONTROL ROD DRIVE ASSEMBLY MODIFICATIONS

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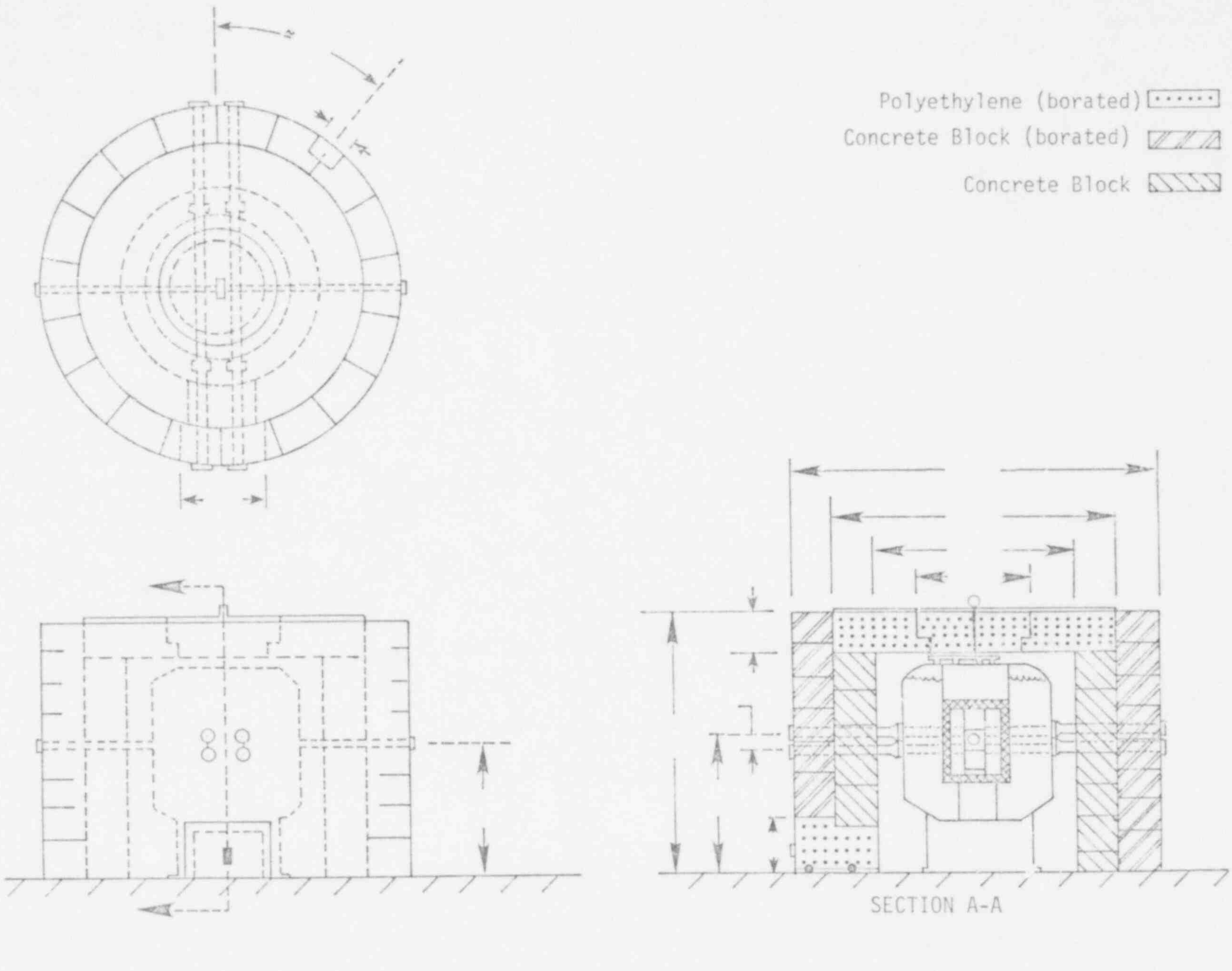

FIGURE II-6. PROPOSED AGN-201 SHIELD

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Instrumentation and control cables, and gas handling system piping/tubing, will be routed through an existing cable channel which is recessed into the reactor room floor. This channel is normally covered by steel plate and will pass beneath the proposed shield below floor level.

A chain link fence topped with strands of barbed wire will be constructed around the existing Reactor Room and Control Room (Figure II-7). The fence will establish an area of controlled access extending approximately 20 feet from the east and north walls of the facility. The fence will contain a gate leading to the Control Room east entrance, and a gate to accommodate access to the north loading dock. For reactor operations above 20 watts, the fenced area will be posted as a Radiation Area and the gates will be locked. For operations at and below 20 watts, it is expected that radiation dose rates external to the north and east walls of the facility will not exceed limits for unrestricted access to the fenced area. Posting of signs and access requirements will be controlled by operating procedures for the facility.

The Reactor Room and the roof of the reactor building will be posted Radiation Areas for operations below 20 watts, and High Radiation Areas for operations above 20 watts. Access shall be controlled by facility procedures, and except for approved radiation surveys, access to the Reactor Room and the roof shall be prohibited for reactor operations above 20 watts.

Rooms 142, 144, and the Reactor Control Room will be posted as Radiation Areas for operations above 20 watts. Access will be controlled in accordance with existing facility procedures.

Rooms 248 and 252, which are located on the second floor directly above rooms 142 and 144 respectively, will be posted as Radiation Areas for operations above 20 watts unless operational surveys show that these rooms may be classified for unrestricted access.

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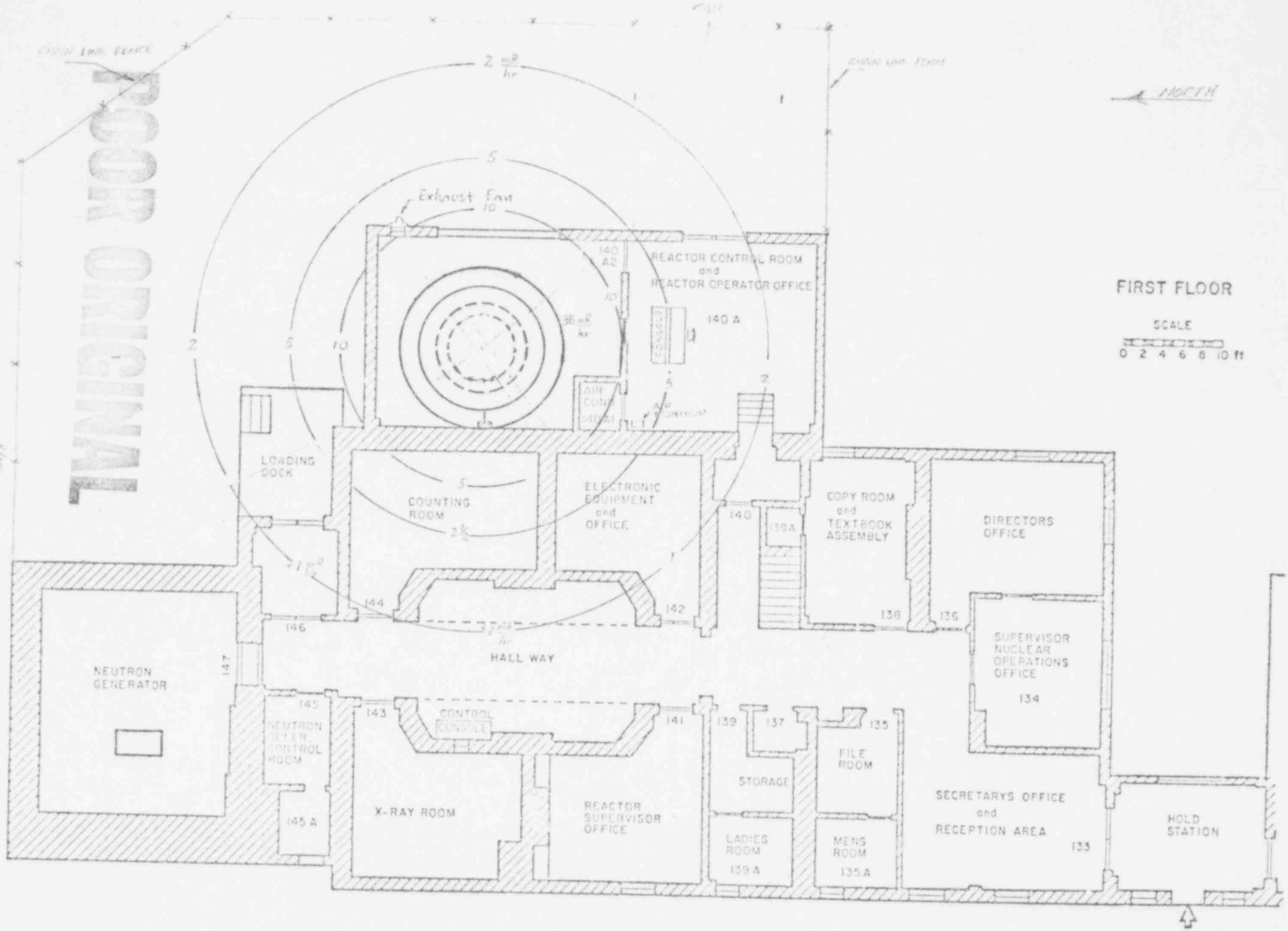

FIGURE II-7. PROPOSED AGN-201 REACTOR FACILITY (ESTIMATED DOSE-RATE RINGS AT 1000 WATTS)

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D. Reactor Characteristics.

Maximum continuous power	20 watts (thermal)
Peak thermal neutron flux	9×10^8 n/cm^2-sec.
Gamma field at vertical shield face	0.8 mR/hr.
Neutron field at vertical shield face	<.01 mrem/hr.
Maximum intermittent power	1000 watts (thermal)
Peak thermal neutron flux	4.5×10^{10} n/cm^2-sec.
Gamma field at vertical shield face	38 mR/hr.
Neutron field at vertical shield face	.06 mrem/hr.
Total Stored Fission Products (Immediately after shutdown)	200 Curies
Long-lived Fission Products (1 day after shutdown)	20 Curies
Total Thickness of Concrete Shield	44 inches
Total Shield Thickness above Reactor	18 inches borated paraffin
Maximum estimated operating time at 1000 watts (reactor core 20°C at startup)	10.3 minutes

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E. Administrative Controls.

Necessary changes to existing facility operating and emergency procedures will be made to ensure that reactor operation is in compliance with approved license and technical specification changes. In addition, new procedures will be developed which include the following:

1. Gas Handling System operation under normal and abnormal conditions (see reference f).

2. High Power Operation (operations greater than 20 watts) to include a Pre-operation Checklist, precautions, limitations, and exclusion area controls.

3. Maintenance Operations including defueling and refueling procedures to satisfy increased radiation control concerns resulting from operation in excess of 20 watts (see reference f).

4. Calibration and operation procedures for new instrumentation proposed in part II.B of this application.

5. Test procedures to include post-modification fuel loading, startup, physics tests, and radiation surveys.

6. Fuel Storage and Handling procedures to preclude simultaneous storage of more than 700 grams of contained U-235 in the existing fuel storage facility or the AGN-201 Reactor Facility.

New procedures and necessary changes to existing procedures will be processed in accordance with Section 6.0, Administrative Controls, of the existing facility technical specifications prior to implementation.

F. Training and Requalification.

The facility staff currently includes two licensed Senior Operators. One additional person is undergoing training and it is anticipated that he will complete Senior Operator license examinations during July, 1979.

A specific training and requalification program will be scheduled by the Reactor Supervisor to ensure that all staff Operators and Senior Operators are properly informed and knowledgeable of approved facility alterations and procedure changes. This program will consist of formal lectures and oral and written examinations. It is anticipated that operational training

will not be required since facility modifications should be completed within the time constraints normally specified to maintain operational proficiency and, except for intermittent high power operations, reactor operation up to the proposed steady-state limit will not be significantly different from existing procedures. The training program shall be approved by the Reactor Administrator and reviewed by the MSU Reactor Safety Committee.

In addition to scheduled training, the facility licensed Operators and Senior Operators will participate in preparation of procedures, procedure changes, and in implementation of approved alterations. Examinations will be administered and evaluated, and training records will be maintained in accordance with the MSU Operator Requalification Program approved by USNRC Operator Licensing Branch letter dated January 11, 1978 (Docket 50-538).

III. PROPOSED AMENDMENT TO OPERATING LICENSE R-127

A. Reactor Fuel (Part 2.B). Amend the existing authorization for reactor fuel to:

1. Transport up to 700 grams of contained U-235 as reserve and interchangeable fuel replacement loading for the reactor. Such shipment authorized between Oak Ridge National Laboratory, Oak Ridge, Tennessee and the Memphis State University South Campus, Memphis, Tennessee.

2. Receive and possess up to 1400 grams of contained U-235 in connection with operation of the facility.

B. Maximum Power Level (Part 2.C). Amend the existing authorization for operation to:

1. Operate the reactor at power levels not to exceed 1000 watts (thermal), and

2. Operate the reactor at power levels of 1000 watts (thermal) or less in such a manner that the integrated power level for any seven consecutive days shall not exceed 3.36 kilowatt-hours.

C. Technical Specifications, Appendix A of Operating License R-127.

1. Safety Limits and Limiting Safety System Settings

Replace existing section 2.0 with the following:

2.0. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1. Safety Limits.

Applicability. These specifications apply to the maximum core temperature and minimum shield water temperature and level during steady state or transient operation.

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

2.1.1. Specification. 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 therefore assures integrity of the core and retention of essentially all fission fragments at temperatures below 200°C.

2.1.2 Specification. The reactor shield tank water temperature shall be maintained above $10^{\circ}C$, 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 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 $10^{\circ}C$ provides a margin for confidence that the reactor will not be operated with frozen shield water.

The shield water level of 12 inches below the top of the tank provides an adequate medium for continuous neutron flux monitoring during reactor operation and ensures adequacy of the facility secondary radiation shield for operations greater than 100 milliwatts.

2.2 Limiting Safety System Settings.

Applicability. These specifications apply to the parts of the reactor safety system which will limit maximum core temperature.

Objective. To assure that automatic protective action is initiated to prevent a safety limit from being exceeded during steady state or transient operation.

2.2.1 Specification. The core thermal fuse shall melt when heated to a temperature of $120^{\circ}C$ or less resulting in core separation and a reactivity loss greater than 5% $\Delta k/k$.

Bases. 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 $120^{\circ}C$ will assure safe shutdown without exceeding a core temperature of $200^{\circ}C$.

2. Limiting Conditions for Operation: Control and Safety Systems

a. Revise the existing specification 3.1.a. to read:

"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% $\Delta k/k$.

b. Revise the existing specification 3.1.b. to read:

"The shutdown margin with the most reactive safety or control rod fully inserted shall be at least 2% $\Delta k/k$, referenced to $20^{\circ}C$.

c. Replace existing section 3.2 with the following:

3.2 Control and Safety Systems

Applicability. These specifications apply to the reactor control and safety systems.

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

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

e. The reactor source level measured by the Nuclear Safety Channel No. 1 countrate instrument is more than 120 CPM.

f. The reactor core tank is pressurized with dry nitrogen to at least 1 psig.

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

1. Nuclear Safety Channel No. 1 may be bypassed for reactor operations conducted above 40 milliwatts provided Nuclear Safety Channels 2 and 3 are verified to be operable.

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TABLE 3.1

SAFETY CHANNEL	SETPOINT	FUNCTION
Nuclear Safety #1 Low Countrate	$\geq$ 12% full scale	Scram at source levels < 12% of full scale
Nuclear Safety #2 (Log) High Power	$\leq$ 2000 watts	Scram at power >2000 watts
Reactor Period	$\geq$ 5 seconds	Scram at periods <5 sec.
Nuclear Safety #3 (linear)		
High Power	$\leq$ 95% full scale	Scram at power >95%
Low Power	$\geq$ 5% full scale	Scram at source levels < 5% of full scale
Shield Water Temperature	$\geq$ 15°C	Scram at temperature <15°C
Shield Water Level	$\leq$ 10.5 inches	Scram at levels >10.5 inches below top of shield water tank
Seismic Displacement	$\leq$ 1/16 inch	Scram at displacements >1/16"
Core Tank Pressure	$\leq$ 5 psig	Scram at core tank pressure >5 psig
Manual Scram	----------------	Scram at operator option
Radiation Monitor	----------------	Alarm at or below level set to meet requirements of 10CFR Part 20
Air Particulate Monitor	----------------	Alarm at or below level set to meet requirements of 10CFR Part 20

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

The minimum reactor source level assures that an adequate neutron countrate from which to conduct an orderly and controlled startup is registered on the startup channel before a reactor startup begins.

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Pressurizing the core tank with dry nitrogen to at least 1 psig assures that evolved hydrogen from high power operation cannot agglomerate into hazardous concentrations.

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 the startup channels (nuclear safety 1 and 3) are operable and responding, and will cause a scram in the event of instrumentation failure. In order to provide assurance that at least two nuclear safety channels are operative for all ranges of reactor operation and to prevent overranging the channel 1 startup instrument, Nuclear Safety Channel No. 1 is allowed to be bypassed for operations above 40 milliwatts only if the remaining two channels are verified to be operable.

Since the AGN-201 core negative temperature coefficient of reactivity in conjunction with the maximum potential excess reactivity specified in 3.1 prevents reactor operation at high power levels for time intervals necessary to approach established safety limits or limiting safety system settings, the high power level scrams are established to provide redundant automatic protective action at levels low enough to assure safe shutdown during rapid reactivity transients and to prevent exceeding requirements for design of the facility radiation shield. 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 large reactivity additions.

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

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

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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.

The core tank high pressure scram prevents reactor operation with internal tank pressures above that for which core tank integrity is assured.

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 and an air particulate monitor must be available to warn operating personnel of a degradation in core tank or gas monitoring system integrity so that appropriate action can be taken to shut the reactor down and assess the hazards to personnel.

3. Limiting Conditions for Operations: Shielding

Replace existing section 3.4 with the following:

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.

3.4.1 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. The facility secondary shield shall be in place with removable shield plugs installed.

3.4.2 Specification. Access to the reactor room shall be prohibited, except for radiation surveys, during operations above 20 watts.

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. Design Features.

Replace existing section 5.1 with the following:

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 and a thermal fuse. 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 reflectors is provided to ensure 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 (1175 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. A valved gas handling system is permanently connected to the core tank assembly to permit monitoring and disposal of gases which may accumulate in the tank from high power operations.

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. A 44 inch thick concrete block shield supporting a top cover that contains approximately 18 inch thick borated paraffin encloses the 6½ foot diameter shield tank to provide a secondary neutron and gamma shield for operations above 100 milliwatts. 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.

IV. SAFETY EVALUATION

AGN-201, serial 108 Reactor design features, characteristics, and operating conditions have been previously evaluated in the Safety Evaluation by the Office of Nuclear Reactor Regulation in support of Docket 50-538 (reference a). Modifications similar to those proposed in this application were evaluated for the AGN-201, serial 100 Reactor in support of Docket 50-43, and the safety and reliability of the modified serial 100 reactor was demonstrated for several years at the U.S. Naval Postgraduate Scnool (USNPGS) in Monterey, California (reference c).

A. AGN-201, Serial 108 Reactor. The following evaluation applies to alterations proposed in Part II.B. of this application.

1. Modifications to the core tank assembly would extend the tank containment boundary into a permanently installed gas handling piping system and provide access for core temperature instrumentation. Following installation, the core tank and piping system will be pressure-tested to 6 psig internal pressure which is approximately 200% of the expected pressure increase resulting from high power operation (reference d). Prior to each reactor startup, facility operating procedures will require verification that the core tank is pressurized with dry Nitrogen to at least 1 psig, but not more than 2 psig. A pressure sensor will initiate an automatic reactor scram for core tank pressures above 5 psig (maximum). MSU considers that the proposed test pressure, prestartup procedure requirement, and automatic protective action will assure that the reactor will not be operated without the core tank integrity provided by the previously evaluated AGN-201 design (reference a).

2. Modifications to the Shield Water Tank Assembly would not compromise the secondary fluid-tight seals nor alter the mechanical strength and shielding characteristics of the assembly previously evaluated in reference (a).

3. Modifications to Instrumentation and Safety Systems would assure a minimum of at least two neutron detection and safety channels for monitoring reactor source level, startup, steady state and transient operations. The use of compensated ion chambers in place of the existing uncompensated chambers would ensure accurate low level neutron detection for mixed radiation fields with thermal neutron to gamma ratios much less than those possible in the AGN-201 Reactor. The scope of existing instrumentation and safety system capabilities will be extended to provide an in-core temperature monitor and an interlock to assure gas-tight integrity of the core tank assembly.

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Nuclear Safety No. 1 would provide linear countrate indication from the minimum reactor source level to more than one decade beyond the power level at which criticality is normally achieved in the AGN-201 Reactor. This system will initiate an automatic reactor scram in the event of instrument or detector failure, and for countrates less than the minimum specified source level which assures that adequate neutron countrates are registered on this startup channel before startup can begin. This low level scram would be deactivated at power levels above which channel saturation would occur provided the Channel 2 log-N instrument has on-scale indication and has been verified to be operating properly. MSU considers that this mode of operation will not compromise reactor safety since the objective of the low-level scram feature will have been fulfilled prior to deactivation and the instrument will be restored to operation at power levels below which channel saturation will not occur, and since the Channel 3 low-level scram will be functional for all ranges of operation. Due to the self-limiting action of the large negative temperature coefficient in conjunction with operation of the core fuse, safe reactor shutdown without core damage is assured for instantaneous reactivity insertions as high as 2.0% $\Delta k/k$. This protective action has been evaluated independently from any instrumentation channel capabilities (reference a). MSU considers, therefore, that this assurance in addition to the redundancy provided by high power scram functions of Nuclear Safety Channels 2 and 3, provides an adequate margin for safety such that the Channel 1 high level scram function can be eliminated without degradation of existing safety system objectives.

Nuclear Safety No. 2 would provide log-N and reactor period indication from the subcritical range, slightly above the minimum source level, to levels greater than the maximum licensed power requested in this application. This safety system will initiate an automatic reactor scram at specified high power levels and at specified reactor periods which will ensure a controlled rate of power rise. MSU considers that raising the threshold of on-scale indication and eliminating the low-level scram function for Nuclear Safety No. 2 will not degrade safe operating performance since only subcritical operation is practical below the proposed threshold level and it is expected that Channel 2 will have on-scale indication following insertion of the two safety rods. In addition, a measure of redundancy for protection against low neutron levels is provided by Channels 1 and 3. In the event of Channel 2 instrument or detector failure, operating procedures require immediate reactor shutdown.

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Nuclear Safety No. 3 will provide linear indication of neutron levels from the minimum reactor source level to more than one decade beyond the maximum licensed power requested in this application. This safety system will initiate an automatic scram in the event of instrument or detector failure and will thus provide redundancy to assure that levels above the minimum specified for at least two channels are registered on the instruments before startup can begin. Nuclear Safety No. 3 will also initiate an automatic reactor scram at specified high power levels and will therefore assure high level scram redundancy for a minimum of two safety channels.

Specifications for the minimum reactor source level, maximum rate of reactivity insertion, maximum potential excess reactivity, and safety and control rod insertion sequence interlocks in conjunction with existing facility startup procedures will result in an additional degree of instrumentation redundancy since all three safety channels will have on-scale indication in the range for which criticality is normally achieved. Thus, MSU concludes that the proposed alterations to existing instrumentation and safety systems will provide adequate neutron monitoring capabilities and sufficient safety system redundancy to support controlled and safe reactor operation.

The modification to the Interlock Line continuity circuit would ensure that reactor operations could not be conducted with internal core tank pressures above those for which gas-tight integrity will have been tested. The addition of an in-core temperature monitor will provide a margin of confidence beyond that previously evaluated to ensure that Limiting Safety System Settings and Safety Limits will not be exceeded, and will not significantly alter core reactivity. Thus, MSU considers that the Interlock Line alteration and addition of in-core temperature monitor will provide assurances of reactor safety beyond those provided by existing AGN-201 design. Similar modifications have been evaluated in reference c.

4. The Gas Handling System proposed in Part II.B.4 of this application, in conjunction with facility operating procedures, will perform functions and have capabilities equal to or better than the system used to operate the AGN-201, serial 100 reactor (reference f) which was previously evaluated in reference c. The design, construction, fuel loading, void spaces, neutron flux levels, and general operating characteristics of the MSU serial 108, AGN-201 Reactor are similar to those documented in reference c. Due to the AGN-201 similarities, MSU considers that radiation damage to the serial 108 Reactor fuel discs, hydrogen and fission product gas diffusion rates from the polyethylene fuel discs, and Argon-41 production rates in reactor voids will not exceed those documented in references d through g for the serial 100 reactor.

Therefore, MSU concludes that the proposed Gas Handling System and associated operating procedures, in conjunction with the existing continuous air particulate activity monitor and alarm system, will ensure that the personnel protection requirements of 10CFR20 for airborne radioactivity can be adequately met.

5. The Safety and Control Rod Assembly modifications proposed in Part II.B.5 of this application are similar to those described in references e and g, which were previously evaluated in reference c. Due to similarities between MSU's serial 108, AGN-201 Reactor and the serial 100 reactor evaluated in reference c, MSU considers that the proposed modifications and corresponding gas handling procedures should pose no threat to safe operation beyond that previously evaluated in reference c.

B. Reactor Building and Shielding. The following evaluation applies to alterations proposed in Part II.C of this application.

1. Continuous Operation at 20 Watts. Calculations for 44" concrete shielding around the AGN-201 Reactor (Appendix A) indicate that dose rates from gamma radiation at the outer surface of the concrete shield will be ≤ 0.8 mR/hr Neutron radiation will not be transmitted through the shield. Calculations for the surface of the consisting top shield consisting of 18" of borated paraffin show that the maximum dose rates would be ≤ 68 mR/hr gamma and ≤ 0.2 mrem/hr fast neutrons for a water-filled thermal column, and ≤ 612 mR/hr gamma and ≤ 6 mrem/hr fast neutrons for a graphite-filled thermal column. At 10 feet above the shield (height of roof), these dose rates decrease to 7 mR/hr gamma and ≤ 0.1 mrem/hr fast neutrons (water filled), and 63 mR/hr gamma and < 0.6 mrem/hr fast neutrons (graphite filled). Thus, the radiation dose rates outside the reactor room will only exceed limits specified for unrestricted access (10CFR20) on the roof directly above the reactor. Existing procedures prohibit access to this area during reactor operation.

2. Intermittent Operation at Power Levels Greater than 20 Watts. Calculations for the proposed shielding at 1000 watt operation (Appendix A) indicate that dose rates at the surface of the concrete shield would be ≤ 38 mR/hr gamma and ≤ .06 mrem/hour neutrons. The borated paraffin top shield will limit dose rates to ≤ 3.4 R/hr gamma and ≤ 9 mrem/hr neutron for a water-filled thermal column, and ≤ 31 R/hr gamma and ≤ 270 mrem/hr neutrons for a graphite filled thermal column. These values would decrease to ≤ 340 mR/hr gamma and ≤ 1 mrem/hour neutrons (water filled thermal column) or ≤ 3.1 R/hr gamma and ≤ 27 mrem/hour neutrons (graphite filled thermal column) at a height of 10 feet above the reactor (height of roof).

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Assuming the reactor could be operated at 1000 watts for 15 minutes (approximately 50% longer than expected), the highest dose available would be $\leq$ 8 rem on top of the poly shield, but access to this area will be prohibited by physical barriers during high power operation and the area is within the viewing range of the console operator via a window in the reactor room to control room wall. The highest dose in an area not in visible range of the operator would be available on the roof while operating with a graphite-filled thermal column. This dose would be $\leq$ 780 mR gamma and $\leq$ 7 mrem neutrons. The dose at the surface of the concrete shield would be $\leq$ 10 mR gamma and $\leq$ .02 mrem neutrons. Since not more than one high power operation could be conducted within a one hour time interval, and since access to the roof is prohibited during all reactor operations, MSU considers the administrative controls governing access to posted restricted areas to be sufficient to assure personnel protection from radiation exposure.

3. Accident Conditions. The original design analysis (Docket F-15) calculated values for radiation doses available at the AGN-201 shield tank exterior wall for a hypothetical 2% step change of reactivity. Assuming the potential dose for the duration of the accident to be 6 rem at the shield tank wall, the addition of the proposed concrete shielding would decrease this dose by a factor of approximately 2600 at the surface of the concrete, and by a factor of approximately 5700 at 10 feet from the concrete surface (control room area). Thus, the dose available to personnel in the control room for the duration of such an accident would be approximately 1 millirem and would not exceed the limitations of 10CFR50, Appendix A, Criterion 19 for accident conditions.

Based upon the preceding information, MSU considers the additional shielding and extension of restricted and posted areas proposed by this application, in conjunction with written radiological controls procedures, to provide reasonable assurance that maximum personnel radiation exposures will be in compliance with federal regulations during normal and potential accident conditions.

C. Technical Specifications. The following evaluation applies to technical specification changes proposed in Part III.C of this application.

1. The revised Safety Limits would delete the existing specification which limits the maximum steady-state power level to less than 100 watts.

The negative temperature coefficient inherent in AGN-201 reactors in conjunction with the maximum potential excess reactivity, experiments included, provides assurance that the core cannot sustain significant power levels for periods of time necessary (steady-state) to approach melting temperatures of the polyethylene fuel matrix.

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Data supporting this conclusion has been documented in reference e from actual operations up to and including 1000 watts (thermal). Maximum operating times experienced in the AGN-201, serial 100 Reactor at steady power levels ranging from 100-1000 watts with 0.5% excess reactivity were recorded when control rods reached full insertion indicating that temperature effects had reduced the reactivity to zero. Based upon this data, it was estimated that temperature effects would not impose a time limit on operation at powers below about 175 watts provided the reactor contained 0.7% excess reactivity, which is greater than the maximum potential reactivity presently permitted for AGN-201 reactors including any installed experiments.

The calculated steady-state temperature rise of the AGN-201 core is 0.5°C/watt (Docket F-16) and has traditionally been used for AGN-201 core evaluations over the past twenty years. This value would yield a steady-state temperature rise of 87.5°C above ambient for continuous operation at 175 watts. The temperature at the core center would be greater by a factor of approximately 1.54 (reference h), or 135°C above ambient. Assuming the core fuse would not function to shutdown the reactor, the resulting temperatures remain less than the melting temperature specified for the polyethylene fuel matrix. These steady-state values at 175 watts appear to be very conservative since the negative temperature coefficient of reactivity, both calculated and measured, is significantly greater than is implied by the postulated steady-state temperature rise vs. available excess reactivity, and, though it is recognized only as qualitative data, the apparent core temperature for 58 minutes operation at 200 watts documented in reference e is significantly lower than would be expected.

For these reasons, MSU considers that deleting a maximum power value from Section 2.1 of the technical specifications will not increase the probability that fission product retention and fuel matrix integrity would not be maintained during the maximum steady-state operation achievable, nor would establishing this parameter as a Safety Limit provide additional protection beyond that offered by the core fuse. Rather, it is considered that a limiting value for reactor power appears to be irrelevant to the stated objectives for Safety Limits and would be more appropriate as a Limiting Condition for Operation.

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2. The basis for the shield tank water level Safety Limit would be reworded to address the proposed modifications and represents no change to the existing specification for reactor operation.

3. The revised Limiting Safety System Settings would delete the existing specification which requires a reactor scram at power levels $\leq$ 0.2 watts. In view of the objectives for LSSS, evaluation of this action is the same as that for the proposed change to the Safety Limits. Reactor high power scram requirements would be specified as Limiting Conditions for Operation.

4. Proposed revisions to the wording of reactivity limits specified as Limiting Conditions for Operation would delete the reference to 20° C for available excess reactivity, and insert the reference to 20° C for minimum shutdown margin.

The available excess reactivity referenced to 20° C would permit an actual excess reactivity of more than 0.65% to exist for reactor operations conducted at temperatures below 20° C and is therefore not consistent with the basis for this specification. MSU considers that the available excess reactivity limit should be referenced to the actual conditions existing during reactor operation. Thus, deleting the reference to 20° C will provide additional clarification that the excess reactivity shall not exceed 0.65% for any condition of reactor operation.

The specification for minimum shutdown margin with the most reactive safety or control rod fully inserted has no common reference for measurement. Although the maximum reactivity worth of control and safety rods for AGN-201 reactors in conjunction with the maximum available excess reactivity, including potential reactivity worth of experiments, is such that a 2% $\Delta k/k$ specification for shutdown margin at any temperature typical of reactor operation will ensure the reactor can be brought and maintained subcritical, MSU considers that referencing this specification to 20° C will provide clarity and a more consistent basis for comparison during the conduct of annual surveillance requirements.

5. Proposed changes to the Limiting Conditions for Operation of Control and Safety Systems would require a minimum of two nuclear safety channels to be in operation, ensure that gas-tight integrity of the core tank is maintained, and ensure that hazardous concentrations of free hydrogen would not be formed during all conditions of reactor operation. A requirement for an air particulate monitor to be in service during reactor operation would also be added. Specifications for interlocks, minimum indicated neutron level for startup, shield water temperature and level, seismic displacement, manual scram operability, reactor period, and radiation monitors would remain as required by the existing license.

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a. Since the core tank and gas-handling system will be tested to at least 6 psig, specifications for at least 1 psig nitrogen pressure, and initiation of reactor scram at pressures greater than 5 psig (increasing) in the core tank would ensure that the reactor is not operated without core tank and gas handling system pressure-tight integrity. In addition, the minimum pressure specification will ensure that sufficient nitrogen is maintained in free-void spaces of the core tank to minimize the probability of an ignition or explosion of hydrogen which may be released from the UO_2 - polyethylene discs. This mode of operation has previously been evaluated as providing an adequate margin for safety (reference c), and operation of the AGN-201, serial 100 reactor has demonstrated the validity of this evaluation. Due to the similarities of design, construction, operating characteristics, and modes of operation, MSU considers the safety evaluation of reference c to be applicable to the AGN-201, serial 108 reactor and that the proposed Limiting Conditions for Operation for core tank pressures do not represent an unreviewed question of reactor safety.

b. The existing technical specifications require three nuclear safety channels to be operable with the exception that either Channel 1 or Channel 3 may be bypassed for up to 12 consecutive hours provided the remaining two channels are verified to be operable. The proposed change to this specification would permit only Channel 1 to be bypassed for operations above approximately 40 milliwatts, provided both Channels 2 and 3 are verified to be operable, to prevent overranging this instrument. Operation of the reactor would not be permitted with Channel 2 or Channel 3 out of service, and all three nuclear safety channels would be operable for reactor startup and the range in which criticality is normally achieved. MSU considers that the proposed change will provide adequate safety channel and neutron monitoring redundancy for all ranges of reactor operation and will not lessen the degree of redundancy required by the existing license.

c. The minimum indicated source level for reactor startup is presently specified as an instrument setpoint for Nuclear Safety Channel No. 1 which is established at 120 CPM. Since the Channel 1 instrument is a linear countrate instrument employing a range selector switch, a fixed neutron level at which the low level scram setpoint can be established is only appropriate for one corresponding position of this switch. As the operator changes ranges to monitor increasing or decreasing neutron levels, the percent of full scale meter deflection corresponding to a fixed countrate (neutron level) also changes.

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Therefore, the proposed alteration would clearly specify 120 CPM as the minimum indicated source level for commencing a reactor startup, and would establish the instrument low level scram setpoint at $\leq$ 12% of full-scale deflection which is sufficiently greater than the noise level for all positions of the range selection switch. MSU considers that the proposed change will increase the degree of confidence that the reactor will not be started up without an adequate source level and that the setpoint in terms of percent full scale meter deflection will ensure that the instrument will be responding sufficiently above electronic noise levels for all positions of the range selector switch.

d. Scram protection from high power levels would be provided by Nuclear Safety Channels 2 (log N) and 3 (linear). The Channel 3 setpoint would be established at $\leq$ 95% full scale meter deflection and Channel 2 would provide redundancy at $\leq$ 2000 watts. Channel 2 would also provide a scram for reactor periods less than 5 seconds.

Since Channel 3 contains a linear picoammeter which employs a range selector switch, a scram setpoint based upon a fixed neutron level would only be appropriate for one corresponding position of this switch. As the operator changes ranges to monitor increasing or decreasing neutron levels, the percent full scale deflection corresponding to a fixed neutron level also changes. Thus, the Channel 3 scram setpoint would be established at $\leq$ 95% full scale meter deflection. This instrument will also initiate a scram signal for indications < 5% full-scale deflection to ensure the range selected is providing indication greater than instrumentation noise levels. Since it would not be possible for an operator to operate the range switch rapidly enough to maintain on-scale indication for Channel 3 during a rapid reactivity insertion that resulted in a period less than that normally permitted, a scram signal would be initiated on the same scale for which the transient began and within a maximum possible interval of 0.9 decade meter indication above the initial power level. In the event Channel 3 failed to scram, the Channel 2 log-N instrument would provide redundant high power scram protection at a fixed power level $\leq$ 2000 watts.

The negative temperature coefficient of reactivity that has been demonstrated in AGN-201 Reactor cores, in conjunction with the maximum potential excess reactivity, provides assurance that safety limits would not be exceeded at the maximum steady-state power levels achievable (Part IV.C.1).

Moreover, the hazards summary contained in references h and i show that, without automatic action from neutron monitoring instruments, degradation in fission product containment should not occur from a positive step change in reactivity as large as 2%. More recently, a transient analysis was conducted for the AGN-201 Reactor which assumed a step change in reactivity corresponding to one-dollar and considered termination of the transient by scram rod withdrawal one second later, which is a time equivalent to a factor of more than 3 times the estimated instrumentation response characteristics (reference j). MSU considers this latter anaylsis to also be applicable to the AGN-201, serial 108 Reactor.

Utilizing the graphs shown in Figures 2 and 3 of reference j which are based upon a one-dollar step change in reactivity initiated from a power level of 0.1 watt, a scram actuated by Channel 3 at 0.9 decades above P(t=0) would correspond to rod withdrawal being initiated at t=0.34 seconds (Appendix B). This assumption considers a trip at 0.9 watts indicated plus 0.3 seconds response time, and shows a peak transient of < 30 watts. Considering a scram from Channel 2 at 2000 watts, rod withdrawal would occur at a point on the graph (extrapolated) located at t=1.3 seconds (Appendix B). The peak transient would be < 20 kilowatts. The total energy release would be < 3000 joules and resultant radiation dose to a person standing next to the reactor (without the additional shielding proposed in this application) would be < 83 mrem.

MSU considers that this analysis provides a conservative point of view since the transient begins at a power level below that for which temperature coefficient feedback effects are significant and thus represents a more rapid power rise during the interval between transient initiation and rod withdrawal than if calculated for P_0 > 0.1 watt. In addition, the Channel 2 period scram is not considered in the analysis, nor is the additional shielding that is proposed in this application. Therefore, MSU considers that the proposed high power scram setpoints would provide reasonable assurance that personnel exposure limits or reactor safety limits would not be exceeded during the maximum credible transient.

e. The proposed Limiting Conditions for Operation would require the existing Air Particulate Monitor to be in service during reactor operation. This monitor is located in the Control Room and continuously samples and returns air to the Reactor Room atmosphere. The monitor has a continuous readout capability and provides an audible and visual alarm if airborne particulate levels in the Reactor Room exceed an adjustable alarm setting.

MSU considers that the continuous Air Particulate Monitor, in conjunction with operating procedures, would provide redundant protection to ensure the reactor is not operated without pressure-tight integrity of the core tank and gas monitoring system. In addition, the monitor would provide sufficient warning so that an assessment of personnel hazards from airborne particulate activity can be made prior to entering the Reactor Room.

f. The proposed Limiting Conditions for Operation applicable to reactor shielding would delete the requirement to prohibit entry to all areas in which dose rate is > 1 mr/hr. (measured at licensed reactor power).

The additional shielding proposed in this application, in conjunction with reactor operating and facility radiological controls procedures, provide assurance that potential radiation doses will be in compliance with the requirements of 10CFR Part 20 and Criterion 19 of Appendix A to 10CFR Part 50 for accident conditions. For this reason, MSU considers that a basis no longer exists for the prohibition against entry into areas in which the dose rate > 1 mr/hr.

However, since operation at the maximum power level requested by this application would result in high radiation areas directly on top of the reactor and proposed additional top shield, a requirement has been added to prohibit access to the reactor room, except for radiation surveys, during operations greater than 20 watts. Since the area of concern is visible to the console operator via a viewing window, and since access to the reactor room is only available through a normally locked door from inside the control room, MSU considers that this requirement should increase the degree of confidence that 10CFR20 requirements will be met during high power operations.

D. Transportation and Storage of Fuel. Personnel from Memphis State University will supervise the transport of reactor fuel from Oak Ridge National Laboratory to Memphis, TN. Preparation and transfer will be in accordance with 10CFR Parts 70 and 71. The fuel will be packaged for transport as Fissile Class III shipments under the general license provisions for shipment of licensed material. The fuel will be divided and stored in 5 sealed DOT specification 6C containers. Each container will be packed with vermiculite and placed atop an empty 6C container inside a DOT specification 6J container, which will then be filled with vermiculite and sealed. The containers will be transported in two separate shipments in an enclosed and locked van with two drivers. Because calculations indicate that the critical mass for AGN-201 fuel is more than 650 grams of U-235, packaging and transporting the fuel in this manner precludes accidental criticality.

MSU considers that the fuel can be transported without hazard and will be adequately protected against theft by using the preceding method.

Memphis State University will store the fuel on site as replacement fuel for the installed AGN-201 Reactor core. The fuel will remain in the 6J containers which will be locked in an area which requires access through at least two locked doors and which is protected by an electronic burglar alarm and is patrolled by a security force (reference b). MSU considers that the storage precautions will provide adequate security since the licensee will not possess an amount of special nuclear material that is equal to or greater than the formula quantities specified in 10CFR Part 73.

Since facility procedures will limit the quantity of contained U-235 which will be handled, used, or stored in the proposed fuel storage area or the reactor room to $\leq$ 700 grams at any one time, MSU does not consider that the criticality monitors specified in 10CFR Part 70.24 will be necessary.

E. Environmental Considerations. Environmental considerations for licensing the Memphis State University Research Reactor to operate at 0.1 watt are contained in the Negative Declaration and Environmental Impact appraisal dated June 14, 1976, which was issued with Construction Permit No. CPRR-122 on June 15, 1976 (reference k). MSU considers that operation of the AGN-201 Reactor at the power levels requested by this application will not significantly alter the factors evaluated in that appraisal.

The fence to be constructed as described in Part II.C of this application would not enclose or alter areas occupied by wildlife, vegetation, nearby waters, or aquatic life. Other physical alterations to be performed are within the boundaries previously evaluated in reference k for site preparation and facility construction.

Additional thermal effluents from operation of the reactor at powers up to and including 1000 watts would be rejected to the surrounding water tank and eventually to the atmosphere by means of conduction and radiation. There would be no release of liquid effluents. However, the potential for gaseous effluents is increased due to diffusion of radiogases from the polyethylene fuel discs of the reactor core following operations at power levels significantly greater than 20 watts.

Operation of the AGN-201, serial 100 reactor demonstrated that a total activity of 40 μ Ci could be present in approximately 2700 cc of evolved gas corresponding to a 3 psi pressure increase in the core tank assembly.

The accumulation of the evolved gases, most of which was hydrogen, took place over a period of 2 to 3 days following a high power run of 10^4 watt-minutes. The total activity represented that activity present in two-day old gas. There was no evidence of gross fission products other than the inert radiogases with a long-lived component that was presumed to be Krypton-85. The amount of evolved gas that was released at any one time amounted to $\leq$ 14% of the total gas in the core tank, and was released in a controlled manner to ensure that effluents were below the maximum permissible concentrations for release to unrestricted areas. The procedure, assumptions, and calculations for such releases are contained in reference f.

Although the potential for formation of releasable gaseous activity to the environment is increased from high power operation of the AGN-201 Reactor, MSU considers that the additional precautions including the higher than design core tank and gas-handling system test pressure, high core-tank pressure scram interlock, and proposed technical specifications correspondingly increase the degree of confidence that gas-tight integrity of the core tank will be maintained. Therefore, the potential for an unplanned release of gaseous effluents should be no greater than previously evaluated in reference k.

Planned releases of evolved gases will be necessary. Since the proposed gas-handling system, method of operation, and method of discharge and dilution to be used with the MSU reactor are closely similar to that previously evaluated in reference f, MSU considers that evaluation to be applicable to the alterations proposed for the serial 108 AGN-201 Reactor. Because the planned releases are controlled to ensure that maximum permissible concentrations for release to unrestricted areas are not exceeded, MSU considers the environmental effects from gaseous effluents to be insignificant.

MSU considers that operation of the reactor at the power level requested by this application, in conjunction with the proposed alterations to radiation shielding, posted and controlled restricted areas, and technical specifications will not alter other environmental considerations appraised in reference k.

F. Emergency and Security Planning. MSU does not propose any changes to the Security Plan or the Emergency Plan previously evaluated and approved in reference a.

REFERENCES

a. Memphis State University Facility Operating License R-127, effective December 10, 1976 (Docket 50-538).

b. Application for Construction Permit and License to Operate the Model AGN-201, serial 108 Nuclear Reactor at Memphis State University, dated April 11, 1975, as amended (Docket 50-538).

c. U.S. Naval Postgraduate School Facility Operating License No. R-11, as amended (Docket 50-43).

d. Report of Operation of Reactor Facility License No. R-11 Amendment No. 1, transmitted by ltr. U.S. Naval Postgraduate School to U.S. Atomic Energy Commission, dated March 30, 1962 (Docket 50-43).

e. Report of Operation No. II of the U.S. Naval Postgraduate School's AGN-201 Reactor Facility, dated January 6, 1964 (Docket 50-43).

f. Supplement I to Application for Amendment to Facility License R-11 to Permit Disassembly and Reassembly of Reactor Core, dated June 10, 1963 (Docket 50-43).

g. Request for Amendment to Facility License R-11 to establish new procedures for monitoring for Argon-41 . . . and to modify the control and safety rods . . ., dated January 8, 1964 (Docket 50-43).

h. Aerojet-General Nucleonics, Elementary Reactor Experimentation, A. T. Biehl, et. al., Ed. (San Ramon, California: October, 1957).

i. "Hazards Summary Report for the AGN-201 Reactor," Aerojet-General Nucleonics, August, 1956 (Docket F-15).

j. "A Safety Analysis for the Georgia Tech AGN-201", J. Narl Davidson, July, 1976 (Docket 50-276).

k. "Environmental Considerations Regarding the Licensing of the Memphis State Univers.ty Research Reactor," Construction Permit CPRR-122, USNRC, June 15, 1976 (Docket 50-538).

APPENDIX A

DOSE RATE CALCULATIONS

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DOSE RATE CALCULATIONS FOR 44" CONCRETE SHIELD WITH 18" POLYETHYLENE (BORATED) TOP

A. Introduction. Radiation dose rates measured during low power operation of the AGN-201, serial 108 reactor are used as a basis for assuming the conservative values which are used in the dose-rate estimates for 20 watt and 1000 watt operation. The measurements are made with portable survey instruments and have been consistent over a period of more than two years. The gamma instruments are low range (0.50 mR/hr.) and are calibrated using a Cobalt-60 source whose strength is known to ± 4.6%. The neutron instrument utilizes a ten-inch polyethylene sphere with scintillation detector and has a spectral response closely approximating the dose curve for neutron energies from thermal to 7 MeV, and retains capabilities through 12 MeV. The neutron survey instrument is regularly calibrated using a California -252 source whose strength is known to ± 3%.

POWER	LOCATION	GAMMA	NEUTRON
50 m Watt	Shield Tank exterior (Glory Hole plane)	3.5 mR/hr.	< 0.1 mrem/hr.
50 m Watt	Thermal Column Top Cover (Water filled)	2.3 mR/hr.	0.2 mrem/hr.

The measured dose-rates are higher than those estimated in the AGN-201 Preliminary Design Analysis (docket F-15), presumably due to radiation scatter and due to operation without boron additions to the shield water.

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B. Assumptions.

1. Dose-rates at 100 milliwatt operation are conservatively assumed as:

LOCATION	GAMMA	NEUTRON
Shield Tank exterior	10 mR/hr.	0.3 mrem/hr.
Thermal column top (water filled)	6.6 mR/hr.	0.6 mrem/hr.
Thermal column top (graphite filled)*	60 mR/hr.	18 mrem/hr.

*Dose-rates at top of graphite filled thermal column are estimated to be greater than water filled by factors of 9 (gamma) and 30 (neutron).[1]

2. The assumed dose rates for 100 milliwatt operation are considered to be those caused by an equivalent point-source located 10 cm. within the core tank assembly. The AGN-201 shielding thus consists of 10 cm. core, 20 cm. graphite, 10 cm. lead, and 55 cm. water comprising a total distance of 95 cm. between shield tank exterior and source of radiation (see docket F-15).

3. An average attenuation coefficient for gamma photons is assumed for 44 inches of ordinary concrete, and is based upon the following factors:

γ ENERGY	μ/ρ[2]	ρ[2]	μ_i	x_i	$\mu_i x_i$	B[2]	$Be^{-\mu_i x_i}$
7.7 MeV	.024	2.35	.056	112	6.26	2.9	5.5×10^{-3}
6.0 MeV	.027	2.35	.064	112	7.15	3.5	2.7×10^{-3}
3.0 MeV	.036	2.35	.085	112	9.50	6.7	5.0×10^{-4}
2.2 MeV	.043	2.35	.101	112	11.30	11.4	1.4×10^{-4}
1.0 MeV	.064	2.35	.150	112	16.80	38	1.9×10^{-6}

$$\text{AVERAGE ATTENUATION COEFFICIENT} = \frac{\Sigma\, Be^{-\mu_i x_i}}{n} = 1.8 \times 10^{-3}$$

The average coefficient is considered conservative since it represents less attenuation (higher transmitted dose-rate) than for values corresponding to energies below approximately 6 MeV. Gamma photons with energy greater than 6 MeV constitute only a small fraction of the spectrum outside the design AGN-201 shield (docket F-15).

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[1] Biehl, et. al., Elementary Reactor Experimentation, Aerojet-General Nucleonics, October, 1 7.

[2] ANL-5800, Reactor Physics Constants, Second Edition, Argonne National Laboratory (USAEC: July, 1963)

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4. A representative relaxation length, λi, for gamma photons in polyethylene is assumed to be 21 cm. (docket F-15).

5. Representative relaxation lengths, λi, for fast neutrons are assumed to be 12.1 cm. in ordinary concrete and 8 cm. in polyethylene (Docket F-15, (2)).

6. It is assumed that the radiation dose-rates will be proportional to the rate of energy release in the reactor, and that the dose rates at 100 milliwatts can be extrapolated using the following relationship:

$\dot{D} = K \left(\frac{dE}{dt}\right) \left(\frac{Ro}{R1}\right)^2 A$, where

$\dot{D}$ = extrapolated dose-rate

K = dose/Joule, and is determined from the assumed dose-rate at 0.1 watt, i.e.,

$$K = \frac{\dot{D}_{0.1}}{(.1W)\ (3600\ J/w\text{-}hr)}$$

$\frac{dE}{dt}$ = rate of energy release in Joules/hour.

Ro = 95 cm.

R1 = distance in cm. from equivalent point source to the point of interest.

A = Attenuation coefficient for the proposed additional shielding.

7. No credit is taken for the free air distance between the reactor shield tank and the proposed 44" thick concrete shield, which will be approximately 16 inches. In addition, no credit is taken for the 8" concrete block walls that enclose the North, West, and South boundaries of the Reactor Room.

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C. Calculations.

1. Gamma Dose Rates at 20 Watts

a. Concrete shield:

$$\dot{D}_\gamma = \left(\frac{10\ \frac{mR}{hr.}}{360\ J/hr.}\right)(7.2 \times 10^4 \frac{J}{hr.})\left(\frac{95}{207}\right)^2(1.8 \times 10^{-3}) = \underline{0.76\ \frac{mR}{hr.}}$$

b. Polyethylene top cover (water filled thermal column):

(1) $\dot{D}_\gamma = \left(\frac{6.6\ mR/hr.}{360\ J/hr.}\right)(7.2 \times 10^4 \frac{J}{hr.})\left(\frac{95}{R1}\right)^2 \exp.\left\{-\frac{Xi}{\lambda i}\right\} = \underline{68\ mR/hr.}$

where R1 = 95 + 45.7 = 141 cm.
Xi = 45.7 cm.
λi = 21 cm.

(2) Ten feet above top cover (roof level)

$\dot{D}_\gamma = (68\ \frac{mR}{hr.})\left(\frac{141\ cm}{446\ cm}\right)^2 \qquad = \underline{7\ mR/hr.}$

c. Polyethylene top cover (graphite filled water column)

(1) $\dot{D}_\gamma = (68\ \frac{mR}{hr.})(9) \qquad = \underline{612\ mR/hr.}$

(2) Ten feet above top cover (roof level)

$\dot{D}_\gamma = (7\ \frac{mR}{hr.})(9) \qquad = \underline{63\ mR/hr.}$

2. Gamma Dose Rates at 1000 Watts

a. Concrete shield:

$\dot{D}_\gamma = \left(\frac{10\ mR/hr.}{360\ J/hr.}\right)(3.6 \times 10^6 \frac{J}{hr.})\left(\frac{95}{207}\right)^2(1.8 \times 10^{-3}) \qquad = \underline{38\ mR/hr.}$

b. Polyethylene top cover (water filled thermal column)

(1) $\dot{D}_\gamma = \left(\frac{6.6\ mR/hr.}{360\ J/hr.}\right)(3.6 \times 10^6\ J/hr.)\left(\frac{95}{141}\right)^2 \exp.\left\{-\frac{45.7}{21}\right\}$
$= \underline{3.4\ R/hr.}$

(2) Ten feet above top cover (roof level)

$\dot{D}_\gamma = (3.4\ R/hr.)\left(\frac{141}{446}\right)^2 \qquad = \underline{340\ mR/hr.}$

c. Polyethylene top cover (graphite filled thermal column)

(1) $\dot{D}_\gamma = (3.4\ R/hr.)(9) \qquad = \underline{30.6\ R/hr.}$

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(2) Ten feet above top cover (roof level)

$\dot{D}_\gamma = (340\ \frac{mR}{hr.})(9) \qquad = \underline{3.1\ R/hr.}$

3. Neutron Dose Rates at 20 Watts

a. Concrete shield:

$$\dot{D}_n = \left(\frac{0.3 \text{ mrem/hr.}}{360 \text{ J/hr.}}\right) (7.2 \times 10^4 \text{J/hr.}) \left(\frac{95}{207}\right)^2 \exp. \left\{-\frac{Xi}{\lambda i}\right\}$$

$$= \underline{.0012 \text{ mrem/hr.}}$$

where $Xi = 112$ cm.
$\lambda i = 12.1$ cm.

b. Polyethylene top cover (water filled thermal column)

$$\dot{D}_n = \left(\frac{0.6 \text{ mrem/hr.}}{360 \text{ J/hr.}}\right) (7.2 \times 10^4 \text{ J/hr.}) \left(\frac{95}{R1}\right)^2 \exp. \left\{-\frac{Xi}{\lambda i}\right\}$$

$$= \underline{0.18 \text{ mrem/hr.}}$$

where $R1 = 95 + 45.7 = 141$ cm.
$Xi = 45.7$ cm.
$\lambda i = 8$ cm.

c. Polyethylene top cover (graphite filled thermal column)

(1) $\dot{D}_n = (0.18 \text{ mrem/hr.}) (30)$ $= \underline{5.4 \text{ mrem/hr.}}$

(2) Ten feet above top cover (roof level)

$$\dot{D}_n = (5.4 \text{ mrem/hr.}) \left(\frac{141 \text{ cm}}{446 \text{ cm}}\right)^2 = \underline{0.54 \text{ mrem/hr.}}$$

NOTE: No credit taken for 5 $^w/o$ Boron in polyethylene or concrete.

4. Neutron Dose Rates at 1000 Watts

a. Concrete shield:

$$\dot{D}_n = \left(\frac{0.3 \text{ mrem/hr.}}{360 \text{ J/hr.}}\right) (3.6 \times 10^6 \text{ J/hr.}) \left(\frac{95}{207}\right)^2 \exp. \left\{-\frac{112\text{cm}}{12.1 \text{ cm}}\right\}$$

$$= \underline{.06 \text{ mrem/hr.}}$$

b. Polyethylene top cover (water filled thermal column)

(1) $\dot{D}_n = \left(\frac{0.6 \text{ mrem/hr.}}{360 \text{ J/hr.}}\right) (3.6 \times 10^6 \text{ J/hr.}) \left(\frac{95}{141}\right)^2 \exp. \left\{-\frac{45.7}{8}\right\}$

$$= 9 \text{ mrem/hr.}$$

(2) Ten feet above top cover (roof level)

$$\dot{D}_n = (9 \text{ mrem/hr.}) \left(\frac{141}{446}\right)^2 = \underline{0.9 \text{ mrem/hr.}}$$

c. Polyethylene top cover (graphite filled thermal column)

(1) $\dot{D}_n$ = (9 mrem/hr.) (30) = 270 mrem/hr.

(2) Ten feet above top cover (roof level)

$\dot{D}_n$ = (0.9 mrem/hr.) (30) = 27 mrem/hr.

NOTE: No credit taken from 5 $^w/_o$ boron in polyethylene or concrete.

5. Thermal Neutrons

a. Assuming a typical Poly-Boron mixture with density 0.94 g/cc containing 5 $^w/_o$ Boron additive, the effective Boron density would be:

ρ_β = (.94) (.05)= .047 g/cc

The removal cross-section would be:

$$\Sigma_{thermal} = \frac{\rho \text{ No } \sigma a}{AW}, \text{ where}$$

ρ = density, g/cc

No = Avogadro's number, atoms/mole

σa = Microscopic absorption cross-section, cm^2

AW = Atomic weight

$$\Sigma_{thermal} = \frac{(.047)\ (.6023)\ (759)}{11} = 1.95\ cm^{-1}$$

b. The thermal neutron attenuation by the 18 inch top shield can be estimated by:

$$\frac{I}{I_o} = \exp. \{-\Sigma\ \chi i\}, \text{ where}$$

I = thermal neutrons from the shield

I_o = thermal neutrons in to the shield

Σ = removal cross-section

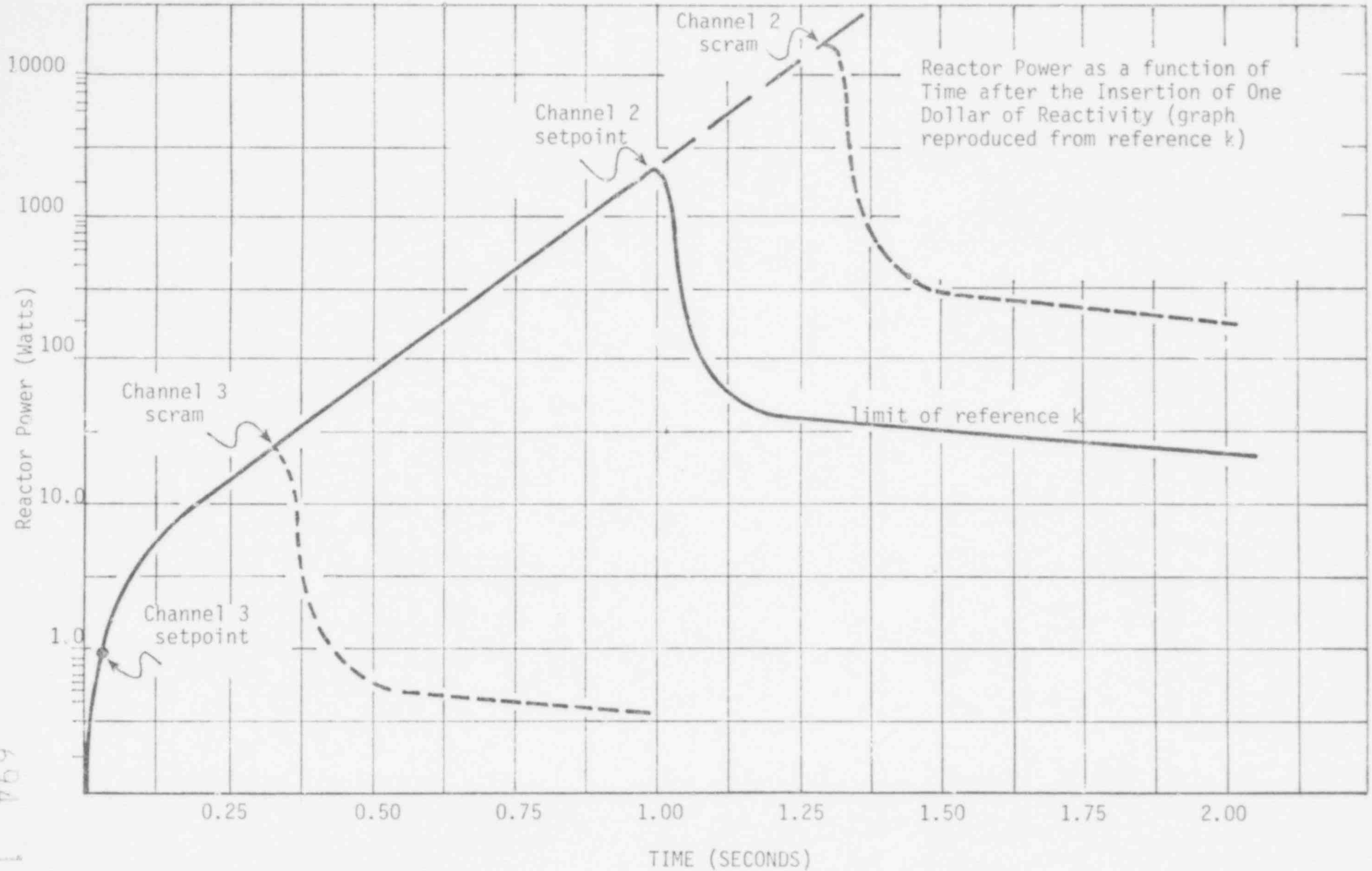
Xi = shield thickness

Thus, $\frac{I}{I_o}$ = exp. {-1.95 (45.7 cm.)} = 2 X 10^{-39}

c. Since it is intended that one cylinder of the blocks comprising the 44" thick concrete cylindrical shield also be borated to ≈ 5%, no thermal neutrons are expected to be transmitted through the proposed additional shielding.

APPENDIX B

TRANSIENT ANALYSIS


Reactor Power as a function of Time after the Insertion of One Dollar of Reactivity (graph reproduced from reference k)
Channel 2 scram
Channel 2 setpoint
Channel 3 scram
Channel 3 setpoint
limit of reference k
10000
1000
100
10.0
1.0
Reactor Power (Watts)
0.25
0.50
0.75
1.00
1.25
1.50
1.75
2.00
TIME (SECONDS)

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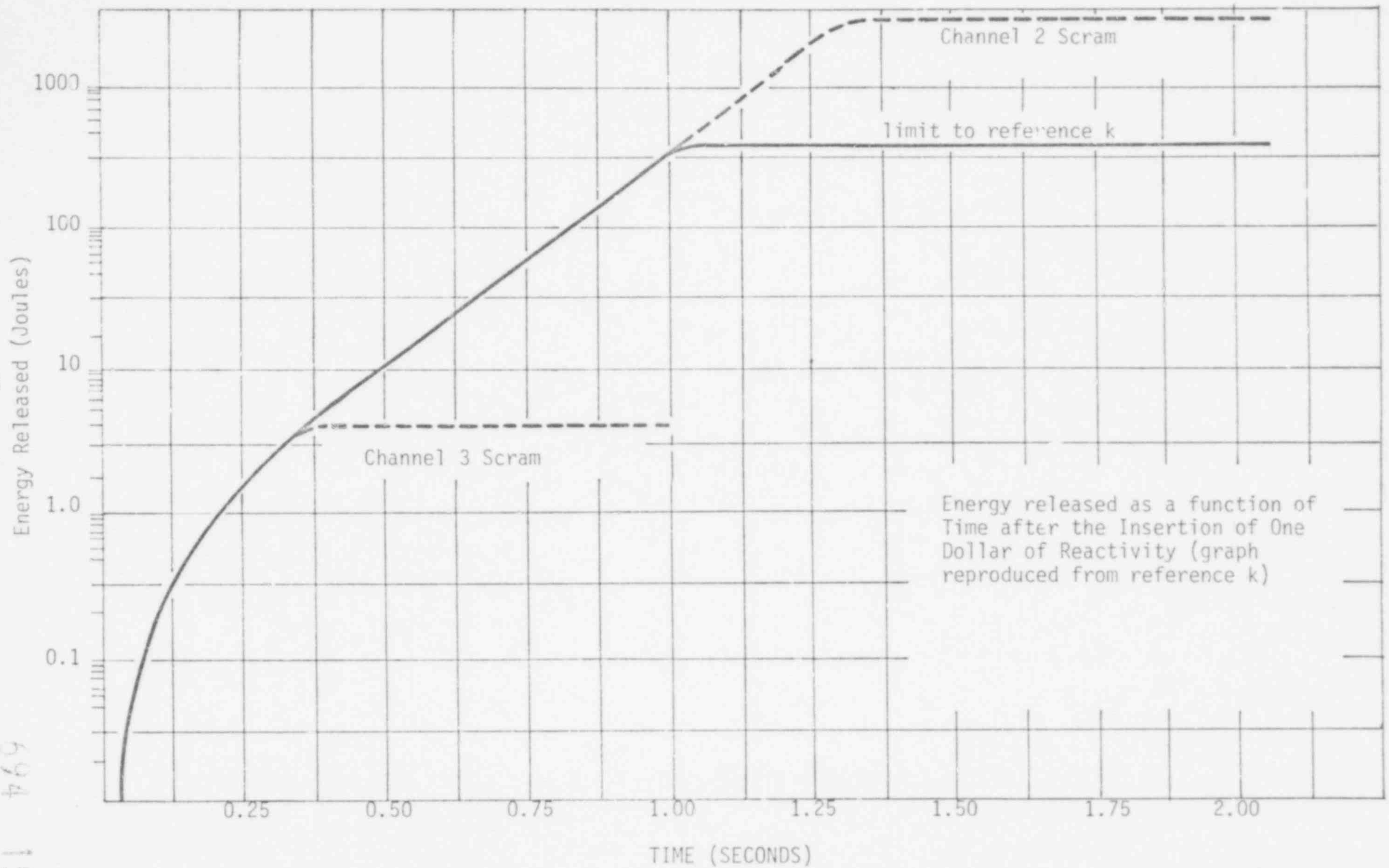
Channel 2 Scram
limit to reference k
Channel 3 Scram
Energy released as a function of Time after the Insertion of One Dollar of Reactivity (graph reproduced from reference k)
Energy Released (Joules)
1000
100
10
1.0
0.1
0.25
0.50
0.75
1.00
1.25
1.50
1.75
2.00
TIME (SECONDS)

B2

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