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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO FACILITY OPERATING LICENSE NO. R-127

MEMPHIS STATE UNIVERSITY

DOCKET NO. 50-538

INTRODUCTION

By letter dated March 23, 1979, as supplemented August 3 and 28, 1979, Memphis State University (MSU or the licensee) requested that Facility Operating License No. R-127 be amended to permit:

1. Operation of 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),

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

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

Operation at these power levels would require modification of the reactor instrumentation and control system, installation of additional shielding, installation of a gas handling system, and additional physical and administrative controls.

Upon completion of the modifications for steady state 20 Watt operation and intermittent 1000 Watt operation, the reactor would be designated Model AGN-201H, Serial No. 108.

DISCUSSION

The AGN-201 reactor is a portable, self-contained reactor using a homogenous fuel mixture of uranium oxide and polyethylene enriched with U-235. The reactor is presently authorized to operate at a maximum power level of 100 milliwatts (thermal).

The increased power level is needed for the MSU scheduled research programs that require higher neutron flux levels than current power levels can attain. This will, in general, significantly enhance the overall educational and research capabilities of the AGN-201 reactor.

Oak Ridge National Laboratories (ORNL) is currently storing 700 grams of contained U-235 which was removed from a previously scrapped AGN-201. This fuel can serve as replacement fuel for the MSU reactor. MSU will supervise the shipment of the fuel from ORNL which is proposed to be accomplished in two separate shipments. This fuel will be stored on site in the shipping containers which will be locked in a secure area protected by locked doors, an electronic **burglar** alarm, and a patrol security force.

Modifications

A. Reactor Instrument[illegible] [illegible]trol Systems

Instrumentation and [illegible] will be modified to provide required operational control and to meet [illegible] considerations. The three nuclear safety systems will be modified with new instruments and ion chambers; a high core tank pressure scram circuit and instruments will be provided; appropriate core temperature instruments will be installed.

B. Shielding

Additional shielding to that provided by the water-filled reactor tank is required. It will consist of two concentric cylinders. one cylinder of borated concrete and the other of ordinary concrete block. They will be arranged so that the seams will overlap to prevent radiation streaming and to provide a total wall thickness of 44 inches. The cylindrical shield will have a top support and shield assembly consisting of 18 inches of borated paraffin.

C. Monitoring-Gas Handling System

A gas handling connection will be installed to interface with a new gas handling system that will penetrate the core tank to permit monitoring of gas and gaseous products in the core void areas.

Administrative Controls

Areas exposed to higher levels of radioactivity will be posted as exclusion or restricted areas during reactor operation. Areas inside the building will be controlled by facility procedures; areas outside the building that may experience radiation during operation will be restricted by use of a security fence and gate.

I. SAFETY EVALUATION

The modified MSU AGN reactor is similar in design and operation to the U. S. Naval Post Graduate School (USNPGS) reactor which operated at a steady state power of 20 watts and intermittent power level of 1000 watts. The USNPGS reactor was an AGN-201 Serial 100 reactor (Docket No. 50-43). Modifications similar to those proposed by MSU were evaluated for the USNPGS reactor and were approved by the Commission on June 15, 1961. The reactor was operated at 20 watts steady state and 1000 watts intermittent for approximately eight (8) years with no apparent fuel deterioration and no danger to the public health and safety. The reactor was then transferred to California State Polytechnic University at San Luis Obispo, California. Following receipt of its license in 1973 (Docket No. 50-394), the (USNPGS) reactor has operated satisfactorily.

In addition, AGN-201, Serial 108 reactor design feature characteristics and operating conditions had been previously evaluated in support of Facility Operating License No. R-127 issued December 10, 1976.

Core Modifications

The modifications to the core tank and shield water tank assembly and the operating procedures assure that the reactor will be operated with the maximum degree of core tank integrity and safety similar to that previously evaluated for the AGN-201 reactor design (Docket No. 50-538, December 10, 1976).

Instrumentation modifications will increase the monitoring and associated reactor safety component response capabilities over those included in the previously approved design (Docket No. 50-538).

Radiation Calculations

The licensee provided the following evaluation with regard to the proposed shielding calculations. We have reviewed these calculations and concur with them.

a. Continuous Operation at 20 Watts

Calculations for 44" concrete shielding around the AGN-201 reactor 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 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 (10 CFR 20) on the roof directly above the reactor. Existing procedures prohibit access to this area during reactor operation.

b. Intermittent Operation at Power Levels Greater than 20 Watts

Calculations for the proposed shielding at 1000 watt operation indicate that dose rates at the surface of the concrete shield would be ≤38 mR/hr gamma and ≤.06 mrem/hr 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/hr neutrons (water filled thermal column) or ≤3.1 R/hr gamma and ≤27 mrem/hr neutrons (graphite filled thermal column) at a height of 10 feet above the reactor (height of roof).

Assuming the reactor could be operated at 1000 watts for 15 minutes (approximately 50% longer than expected), the highest dose available would be ≤8 rem on top of the polyshield, 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 ≤780 mR gamma and ≤7 mrem neutrons. The dose at the surface of the concrete shield would be ≤10 mR gamma and ≤.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, the administrative controls governing access to posted restricted areas are considered sufficient to assure personnel protection from radiation exposure.

Fuel Integrity

The fuel consists of polyethylene material with uranium dioxide (enriched to 19.9% in U-235) uniformly dispersed throughout the polyethylene. Polyethylene is an organic material that can sustain radiation damage when exposed to fission product bombardment. Test data was provided by Aerojet-General Nucleonics of samples of core material exposed in the Argonne National Laboratory CP-5 reactor. The CP-5 reactor is a 5 megawatt (flux-10^{12} n/cm^2 sec) reactor. Tests included exposures at full power for periods up to one week continuous operation. Analyses of these tests revealed that radiation damage was evident in a reduced density and there was some loss of hydrogen from the polyethylene. An extrapolation of these results, assuming that the integrated flux-time (nvt) is responsible for the damage, for continuous operation at 100 watts equates to a core life of six years prior to any damage occurring. At 20 watts continuous operation the core life would be approximately 30 years. As the normal operating cycle is less than 40 hours per week, or less than 24% of the total time, the projected life approaches 125 years. Intermittent operation at 1000 watts (thermal) at flux levels of 4.2×10^{10} n/cm^2 sec for short periods of time (less than 20 minutes) will not greatly affect core life. From this analysis it is reasonable to conclude that the AGN-201 core operating 40 hours per week at 20 watts (flux - 9×10^8 n/cm^2 sec) and intermittent operation at 1000 watts would sustain no radiation damage over the remaining 16 years of reactor operation authorized by MSU's license.

Fuel Storage

Three hundred fifty grams of the fuel will be stored in each of two Department of Transportation (DOT) specified 6J type containers. Each container is filled with vermiculite and sealed. Calculations for AGN-201 fuel indicate that the critical mass is greater than 650 grams of U-235. Accordingly, it is not possible for critical mass to be achieved with the fuel containerization devised.

Design Basis Accident (DBA)

We have postulated a DBA for an AGN-201 operating at 20 watts to be the failure of the polyethylene moderator cladding of a single fuel disc in air. If all elements ruptured and all fission products were released, the activity would be approximately 200 curies, principally Iodine-131, when operating at 1000 watts and 600 millicuries at 20 watts. This equates to 6.0×10^{-4} µCi/ml if uniformly dispersed throughout the reactor room (10^6 liters). The release from the postulated DBA when uniformly dispersed throughout the reactor room is 1.27×10^{-7} µCi/ml which is less than the Part 20 limiting dose for an unrestricted area. Any release to the outside atmosphere would be only a small fraction of this dose and therefore presents no hazard to the public. Restricting 1000 watt operation to less than 20 minutes will ensure that any DBA occurring when at this power level will not significantly affect the foregoing assumptions which are conservative by a factor of 100.

Although the DBA is only remotely possible, MSU is designating the reactor room a controlled area; and should it occur, the additional precautions of evacuating the reactor room will be accomplished and emergency procedures implemented.

Gas Handling System (GHS)

The GHS modifications are described in detail in Section II B4 (p.9) Figure II-4 and Section IV A4 (p.29) of the application for amendment dated March 23, 1979. The proposed system and facility operating procedures will have a degree of performance and reliability equal to or greater than previously evaluated systems used in AGN-201 Serial 100 reactors.

As stated above, the core material will sustain no damage over the 16 years of operation remaining in the authorized MSU license. The GHS, however, is designed to identify radiation damage by being able to measure fission products that escape from radiation damage to the fuel discs. MSU concludes and we agree that the proposed GHS, the existing air particulate activity monitor and alarm system and associated procedures will ensure that personnel protection requirements of 10 CFR 20 can be adequately met.

CONCLUSION ON SAFETY

Moreover, due to the fact that: (1) no unusual problems have arisen with this reactor during over 20 years of authorized operation at 0.1 watts at both Argonne National Laboratory and MSU, (2) the revised TS require surveillance and periodic testing of safety related equipment to assure continued safe operation of the reactor at 20 watts and to assure that any significant component degradation will be detected in a timely manner, and (3) the USNPGS operated a similar AGN-201 reactor at 20 and 1000 watts without evidence of any unusual problems over a period of eight (8) years, we have concluded that the MSU AGN-201 reactor can be operated in a safe manner at 20 watts steady state and intermittently at 1000 watts. Furthermore, based on the foregoing considerations, we have concluded that the estimated life of the facility will extend far beyond the end of the current license period. Therefore, from a reactor safety standpoint, the amendment is acceptable.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation at 20 watts steady state and intermittently at 1000 watts 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.

II. ENVIRONMENTAL IMPACT APPRAISAL

Inasmuch as the power level of this research reactor facility remains below 2 MWt, the environmental impact remains as stated in the license granted to MSU in 1976 (Docket No. 50-538). The 1976 Environmental Impact Appraisal was based upon a generic evaluation for research reactors of 2 MWt or less which was approved January 28, 1974.

Based on the foregoing analysis, we have concluded that there will be no significant environmental impact attributed to this proposed license power increase. Having made this conclusion, we have further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Dated: March 28, 1980