

July 25, 2011

Dr. Raymond J. Juzaitis  
Department Head, Nuclear Engineering  
Texas A&M University  
129 Zachry Engineering Center  
College Station, TX 77843-3133

SUBJECT: TEXAS A & M UNIVERSITY - REQUEST FOR ADDITIONAL INFORMATION  
REGARDING THE TEXAS A & M UNIVERSITY AGN-201M REACTOR  
LICENSE RENEWAL APPLICATION (TAC NO. ME1588)

Dear Dr. Juzaitis:

The U.S. Nuclear Regulatory Commission (NRC) is continuing our review of your application for renewal of Facility Operating License No. R-23, Docket No. 50-59 for the Texas A & M University AGN-201M Reactor dated July 22, 1997, as supplemented by letters dated June 30, and July 11, 2011. During our review of the documentation for your renewal request, questions have arisen for which we require additional information and clarification. Enclosed are requests for additional information (RAI). We are requesting a response to this request within 60 days of the date of this letter.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.30(b), you must execute your response in a signed original document under oath or affirmation. Your response must be submitted in accordance with 10 CFR 50.4, "Written Communications." Information included in your response that is considered security, sensitive, or proprietary, that you seek to have withheld from the public, must be marked in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Following receipt of the additional information, we will continue our evaluation of your renewal request.

If you have any questions regarding this review, please contact Walter Meyer at (301) 415-0897 or by electronic mail at [walter.meyer@nrc.gov](mailto:walter.meyer@nrc.gov).

Sincerely,

**/RAI**

Duane A. Hardesty, Project Manager  
Research and Test Reactors Licensing Branch  
Division of Policy and Rulemaking  
Office of Nuclear Reactor Regulation

Docket No. 50-59

Enclosure:  
As stated

cc w/encl: See next page

Texas A&M University

Docket No. 50-59

cc:

Mayor, City of College Station  
P.O. Box Drawer 9960  
College Station, TX 77840-3575

Governor's Budget and  
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Austin, TX 78711

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Radiation Program Officer  
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Susan M. Jablonski  
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Office of Permitting, Remediation & Registration  
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P.O. Box 13087, MS 122  
Austin, TX 78711-3087

Test, Research, and Training  
Reactor Newsletter  
University of Florida  
202 Nuclear Sciences Center  
Gainesville, FL 32611

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Sincerely,  
/RA/

Duane A. Hardesty, Project Manager  
Research and Test Reactors Licensing Branch  
Division of Policy and Rulemaking  
Office of Nuclear Reactor Regulation

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TEMPLATE # NRR-

OFFICE	PRLB:PM	PRLB:PM	PRLB:LA	PRLB:BC	PRLB:PM
NAME	WMeyer	DHardesty	GLappert	JQuichocho	DHardesty
DATE	7/19/11	7/19/2011	7/22/2011	7/25/11	7/25/11

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**OFFICE OF NUCLEAR REACTOR REGULATION**  
**REQUEST FOR ADDITIONAL INFORMATION**  
**FOR THE RENEWED FACILITY OPERATING LICENSE**  
**TEXAS A & M UNIVERSITY AGN-201M RESEARCH REACTOR**  
**LICENSE NO. R-23**  
**DOCKET NO. 50-59**

The U.S. Nuclear Regulatory Commission (NRC) is continuing our review of your application for renewal of Facility Operating License No. R-23 for the Texas A & M AGN-201M Reactor which you submitted on July 22, 1997, as supplemented by letters dated June 30, and July 11, 2011. During our review of your renewal request, questions have arisen for which we require additional information and clarification. Please provide responses to each of the following requests for additional information no later than 60 days from the date of this letter.

1. NUREG-1537, Part 1, Section 1.4, "Shared Facilities and Equipment," requests complete descriptions and any safety implications that result from shared facilities. The safety analysis report (SAR) Section 1.4 describes a subcritical assembly within the reactor room. Please provide a description and purpose for this facility and any potential for this assembly to affect reactor operations. Please address any other potential public health and safety or environmental issues this shared facility may present.
2. NUREG-1537, Section 4.5.3, "Operating Limits," requests principle parameters of the reactor that support the conclusion the reactor is designed for safe operation and can be shutdown under all credible conditions. SAR Table 4.1.D. provides a value for the measured excess reactivity. Please provide the date of the most recent determination of excess reactivity and shutdown margin and describe the method used to make these determinations.
3. NUREG-1537, Section 4.2, "Reactor Core," requests a description of major core components including in-core experimental facilities. SAR Table 4.1.D. provides the reactivity associated with a cadmium shutdown rod but does not describe its purpose. Please describe this rod and how and when it is used.
4. NUREG-1537, Part 1, Section 4.3, "Reactor Tank or Pool," requests information that will ensure the integrity of the reactor tanks during their projected lifetime. SAR Section 4.3 describes the tanks but does not discuss planned methods to assess possible deterioration during their projected lifetime. Please discuss any inspections to date that have been performed on the core tank, reactor tank and shield water tank and what monitoring and assessments are planned as part of a continuing program to ensure there is no deterioration of these tanks during their projected lifetime.
5. NUREG-1537, Part 1, Section 4.5, "Nuclear Design," requests descriptions and analyses of all safety issues in the design and operation of the reactor. SAR Section 4.5.3 states

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that the core thermal fuse shall melt when heated to a temperature of 120 degrees C or less resulting in core separation and a reactivity loss of greater than 5% delta K. Please provide an evaluation of a safety analysis for the core thermal fuse that supports the conclusion that the core thermal fuse will melt when heated to 120 degrees C or less resulting in core separation and creating a greater than 5% reactivity loss.

6. NUREG-1537, Part I, Section 7.2, "Design of Instrumentation and Control (I&C) Systems," requests a discussion of the design of I&C Systems to show that it is designed to perform its protective function after experiencing a single random failure within the system. SAR Section 7.2.1 is limited to a brief description of the functions of the I&C systems. Please describe the criteria (standards, codes and guidance) that form the design basis for the AGN Reactor Console Instrumentation and Electronics Upgrade. Please describe how the digital software incorporated into this modification was designed such that a failure of the computer or software will not prevent the safety functions of the hardwired scram circuits from performing their intended functions.
7. NUREG-1537, Part 1, Section 7.2.2, "Design-Basis Requirements," requests the function or purpose of systems or instruments that monitor reactor parameters. SAR Sections 7.1 and 7.2.3 state that the skirt monitor is intended to scram the reactor and sound an evacuation alarm. Please include a limited condition for operation (LCO) and surveillance requirement for the skirt monitor scram and alarm or justify why this scram and evacuation alarm are not required to be included in Technical Specifications (TS). Please describe how often the skirt monitor detector is calibrated and how the setpoint is derived and implemented.
8. NUREG-1537, Part 1, Section 7.2.4, "System Performance Analysis," requests description of the operation of I&C systems and presentation of the analysis of how the system design meets the design criteria and design bases. SAR Section 7.2.3 states that the Channel #2 detector and circuitry, which was changed during the Reactor Console Instrumentation and Electronics Upgrade, sends a signal to the AGN computer which calculates reactor period and drives the period meter display. Please provide an evaluation of the safety analysis regarding the design change of the Channel #2 period function that shows the impact on response time changes and how the new design will not impact the safety margin or accident analyses in SAR Chapter 13.
9. NUREG-1537, Part 1, Section 7.2.4, "System Performance Analysis," requests description of the operation of I&C systems and presentation of the analysis of how the system design meets the design criteria and design bases. SAR Section 7.2.3 states that the Channel #3 detector and circuitry, which was changed during the Reactor Console Instrumentation and Electronics Upgrade, sends a signal to the AGN computer which generates a linear power reading in watts. Please provide an evaluation of a safety analysis regarding the design change of the Channel #3 linear power function that shows the impact on response time changes and how the new design will not impact the safety margin or accident analyses in SAR Chapter 13.
10. NUREG-1537, Part 1, Section 7.2.4, "System Performance Analysis," requests description of the operation of I&C systems and presentation of the analysis of how the system design meets the design criteria and design bases. SAR Figure 7.3 shows how a computer was integrated into the reactor control safety system as part of a console and

instrumentation upgrade. Please describe the circuits in the reactor control safety system where solid state circuits have replaced relay circuits and provide an evaluation of a safety analysis that shows these changes have not impacted circuit response time or created an impact on the function of other systems that were not upgraded.

11. NUREG-1537, Part 1, Section 7.1, "Summary Description," requests a description of the I&C systems showing major components and subsystems and connections among them. SAR Figure 7.1, Reactor Control Safety System, does not show how the Interlock Relay scram function is incorporated as a result of the Reactor Console Instrumentation and Electronics Upgrade. Please provide additional detail to indicate how this scram function is implemented following the Reactor Console Instrumentation and Electronics Upgrade.
12. NUREG-1537, Part 1, Section 7.2.2, "Design-Basis Requirements," requests a description of the function or purpose of systems or instruments. SAR Section 7.5 describes a criticality monitor for the source locker. Please provide a description of this monitor and its purpose and indicate if this monitor is required by the reactor license.
13. NUREG-1537, Part 1, Section 7.2.2, "Design-Basis Requirements," requests a description of the function and purpose of safety or control functions. Modification Authorization 2008-1, submitted as part of the application, describes the logic in the new console as the same as in the old console except a computer used as digital recorder and monitor. In the same paragraph it states the scram circuit is designed to trip the reactor immediately when the computer fails. This scram function is not described in the SAR or included in TS. Please provide an evaluation of a safety analysis of this design change and describe any impact on the safety analyses in SAR Chapter 13. Please indicate how this "watch-dog" scram is initiated and propose changes to TS to include this scram as an LCO with a corresponding surveillance or provide an explanation describing your reasons for not incorporating the changes.
14. NUREG-1537, Part 1, Section 11.1.4, "Radiation Monitoring and Surveying," requests the applicant describe how routine monitoring provided at the facility provides reasonable assurance that all radiation at, and released, from the site will be appropriately monitored. Section 20.1501(a)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires licensees to perform surveys to evaluate the magnitude and extent of radiation levels and the potential radiological hazards. Section 20.1101 of 10 CFR requires licensees to the extent practical achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA). SAR Section 11.1 states the Radiation Safety personnel will conduct an annual radiation survey of the reactor. Please describe the annual radiation survey and provide an explanation why more frequent surveys are not required to meet ALARA considerations and to evaluate potential radiological hazards as required in 20.1501.
15. NUREG-1537, Part 1, Section 11.1.1.1, "Airborne Radiation Sources," requests discussion and calculations that show that facility design ensures doses to the facility staff and the public will not exceed 10 CFR Part 20 limits for effluents. SAR Section 11.2.1 states only that Ar-41 production is minimal. Please provide an evaluation of a safety analysis, with calculations, that indicate the maximum argon-41 (Ar-41) concentration in the reactor room and released from the facility based on current operations are within the limits of 10 CFR 20 Appendix B.

16. NUREG-1537, Part 1, Section 11.1.3, "ALARA Program," requests a description of the ALARA program for the facility as required by 10 CFR 20.1101. The description should include the basis for the program and the management level and authority by which the facility ALARA policy is established. SAR Section 11.4 provides little of the description requested. Please provide a discussion that shows how the University "Radiological Safety Program Manual" applies to ALARA policy at your facility.
17. NUREG-1537, Part 1, Section 11.1.5, "Radiation Exposure Control and Dosimetry," requests description of the procedures utilized for controlling exposures to personnel and releases of radioactive materials from the facility. SAR Section 11.6 and 11.7 discuss how irradiations are conducted and handled but does not discuss radiological monitoring during experiment removal from the reactor. Please describe how personnel are monitored for exposure when an experiment is being removed from the reactor and show how it applies to the ALARA policy for your facility.
18. NUREG-1537, Part 1, Section 12.1.2, "Responsibility," requests discussion of responsibilities for the safe operation of the reactor and the reactor facility for individuals that appear in the organization structure. SAR Section 12.1.1 description of the Reactor Supervisor responsibilities does not include the SAR Section 10.3 responsibility that experiments will be conducted under the direct supervision of the Reactor Supervisor. Please verify that this responsibility is one of the Reactor Supervisor responsibilities, if this responsibility is permitted to be delegated and if delegated by what mechanism.
19. NUREG-1537, Part 1, Section 12.2.2, "Charter and Rules," requests discussion of requirements for a quorum when voting. SAR Section 12.2.2 states the operating organization will not comprise a voting majority of the members at any Reactor Safety Board. Please discuss how this voting control is met, specify by title which persons are considered members of the operating organization and propose changes to TS 6.4.1 that incorporate these details or provide an explanation describing your reasons for not incorporating the changes.
20. NUREG-1537, Part 1, Section 13, "Accident Analyses," states that the NRC staff has generally found acceptable that doses to facility staff for accident analysis results are less than 5 rem whole-body. SAR Section 13.1.1 provides a Maximum Hypothetical Accident (MHA) with calculated results that exceed the acceptable limit for the occupationally exposed staff member.

Please provide an evaluation of a safety analysis of the MHA for your facility that indicates the resulting occupational doses are within the Total Effective Dose Equivalent limits of 10 CFR 20.1201 for an occupationally exposed staff member. Please provide an evaluation of a safety analysis, with calculations, for:

- a. The maximally exposed staff member.
- b. The staff member at the reactor console, and;
- c. The potential direct dose to a person in the unrestricted area closest to the reactor room and show the dose is within the limits of 10 CFR 20.1302, compliance with dose limits for individual members of the public.

The following requests for additional information (RAIs) are related to the proposed TS submitted by letter dated June 30, 2011, for the Texas A & M AGN-201M reactor. In responding to the following RAIs, please provide a response to each individual RAI, including any revised wording for the proposed TS with justification for any changes. Also, please provide a revised TS page that incorporates any proposed changes made as a result of the responses to these RAIs.

Regulation 10 CFR 50.36, "Technical Specification," contains the requirements for proposed technical specifications (TS) submitted as part of a license application. American National Standards Institute/American Nuclear Society standard 15.1, 2007, "Development of Technical Specifications for Research Reactors," (ANSI/ANS-15.1) and NUREG-1537, Parts 1 and 2, "Guidance for Preparing and Reviewing Applications for the Licensing of Non-power Reactors," provide guidance for satisfying the requirements of 10 CFR 50.36.

21. ANSI/ANS-15., Section 1.3, "Definitions," provides definitions commonly used in Research and Test Reactors. TS 1.0 contains several definitions that do not conform or lacked recommended detail. Please propose changes to the definitions that conform to ANSI/ANS-15.1 guidance for: reactor secured, reactor shutdown and shutdown margin (SDM) or provide an explanation describing your reasons for not incorporating the changes.
22. NUREG-1537, Part 1, Appendix 14.1, Section 3.2(4), "Scram Channels," states that historically, there have been cases in which the NRC has accepted power level scrams higher than licensed power (1.2 times licensed power level is common) if supported by the safety analysis. TS 3.2.e, Table 3.1 lists the high power scram setpoint as <10 watts. Please consider revising the high power scram setpoints in TS Table 3.1 to 120% of licensed power (6 watts) or justify by safety analysis why this scram setpoint should remain as proposed (10 watts).
23. ANSI/ANS-15.1, Section 1.3, "Definitions" requests the (SDM) be made with the non-scrammable rods in the most reactive position. In TS 3.1.b, the (SDM) determination does not include the reactivity of the fine control rod, a non-scrammable rod, in the most reactive position (inserted). Please propose changes to TS 3.1.b to include the reactivity effect of the fine rod failing in its most reactive state to the SDM determination or provide an explanation describing your reasons for not incorporating the changes.
24. NUREG-1537, Part 1, Appendix 14.1, Section 3.2(4), "Scram Channels," requests a list of all required scram channels. TS 3.2 and TS Table 3.1 do not include the Interlock Relay Scram in TS that is described in the SAR Section 7.3. Please propose changes to TS 3.2 and Table 3.1 adding the Interlock Relay Scram as an LCO with corresponding surveillance requirement or provide an explanation describing your reasons for not incorporating the changes.



25. ANSI/ANS-15.1, Section 4.2(5), "Reactor Control and Safety Systems," requests appropriate surveillance testing (e.g., operability checks, calibrations, and inspections) for scram channels. TS 3.2.g states that a seismic displacement interlock switch shall be installed in such a manner to prevent reactor startup and scram the reactor during a seismic displacement. Please propose and justify a setpoint for this switch and propose changes to TS to provide a surveillance for this switch or provide an explanation describing your reasons for not incorporating the changes.
26. ANSI/ANS-15.1, Section 3.2, "Reactor Control and Safety Systems," requests appropriate LCOs for the reactor control and safety systems. Proposed TS 3.2.d does not include a requirement that appeared in previous TS. Please propose changes to TS 3.2.d to include "Reactor startup cannot commence unless both safety rods and coarse control rod are fully withdrawn from the core" or provide an explanation describing your reasons for not incorporating the changes.
27. ANSI/ANS-15.1, Section 3.8.3, "Failure and malfunctions," requests that failure of any experiment shall not result in releases or exposures in excess of 10 CFR 20 limits. TS 3.3.d. includes a two hour exposure for persons in the unrestricted area starting with the time of a release. Please provide an evaluation of a safety analysis showing persons can be removed from the exposure in the specified time and that the specified time is consistent with evacuation times in your Emergency Plan.
28. ANSI/ANS-15.1, Sections 4.2(1) and 4.2(4), "Reactor Control and Safety Systems," request measurement of scram time and rod reactivity worth be performed after any work is done on the rods or the rod drive systems. TS 4.2.a. does not include the requested measurement of scram time and rod reactivity worth after work is done on reactor rods and rod drives. Please propose changes to TS adding these requirements to TS 4.2.a or provide an explanation describing your reasons for not incorporating the changes.
29. ANSI/ANS-15.1, Section 3.1(4), "Core configurations," requests LCO specifications be included for core configuration components that assure the assumptions for accident analyses are maintained. The Maximum Hypothetical Accident scenario assumes that no fission products escape the core tank. Please propose changes to TS 3.4.f to include the specification that "the core tank shall be sealed during reactor operations" or provide an explanation describing your reasons for not incorporating the changes.
30. ANSI/ANS-15.1, Section 4.0, "Surveillance requirements," requests appropriate surveillance testing (e.g., inspections) for reactor component operability. TS 4.3.b. specifies that a visual inspection for water leakage from the shield water tank shall be performed annually and leakage shall be corrected prior to subsequent reactor operation. Please provide an evaluation of the proposed inspection frequency and propose changes to TS 4.3.b to perform a visual inspection or check more frequently or provide an explanation describing your reasons for not incorporating the change.
31. ANSI/ANS-15.1, Section 5.1, "Site and facility description," requests a description of the site and reactor facility. TS 5.1 describes the Reactor Room and Accelerator Room but does not specify the reactor licensed area. Please define in TS 5.1 the areas, including room numbers where appropriate, that are under the reactor license.

32. ANSI/ANS-15.1, Section 5.4, "Fissionable material storage," specifies that fuel shall be stored in a geometric array where  $K_{eff}$  is no greater than 0.9 for all conditions of moderation and reflection. TS 5.3 specifies that fueled experiments and fuel devices not in the reactor shall be stored in an array such that  $K_{eff}$  is not greater than 0.8 but the SAR provides no supporting analyses. Please provide an evaluation of a safety analysis that shows how this fuel storage specification is met.
33. ANSI/ANS-15.1, Section 6.1.3(3) "Staffing," describes events when the presence of an SRO is required. TS 6.1.9 does not include 1) initial startup and approach to power after the reactor has been secured and 2) recovery from unplanned or unscheduled shutdown or significant power reduction. Please include these events in TS 6.1.9 as these are required by 10 CFR 50.54(m)(1).
34. ANSI 15.1, Section 6.2.2. "Charter and rules," requests a timely dissemination, review and approval of minutes. TS 6.4.5, does not specify to whom the Reactor Safety Board minutes are distributed or a timeframe for the distribution. Please propose changes to TS 6.4.5. that provide the requested information or provide an explanation describing your reasons for not incorporating the changes.
35. ANSI/ANS-15.1, Section 6.6.2, "Actions to be taken in the event of an occurrence of the type identified in Sections 6.7.2(1)(b) and 6.7.2(1)(c)," describe actions to be taken for reportable occurrences. TS 6.9.2 has no specified actions required for reportable occurrences. Please propose changes to TS 6.9.2 to include the actions specified above for reportable occurrences or provide an explanation describing your reasons for not incorporating the changes.
36. ANSI-15.1, Section 6.8.3, "Records to be retained for the life of the facility," provides recommendations for record retention. TS 6.10.2 does not include copies of Reactor Safety Board Audits or records of review of violations of any safety limit, LSSS or LCO. Please propose changes to 6.10.2 to include these records or provide an explanation describing your reasons for not incorporating the changes.
37. ANSI-15.1, Section 6.2.3, "Review function," addresses the review function required by 10 CFR 50.59. TS 6.1.6 describes the review and approval responsibility for meeting regulation and license conditions. Modification Authorization 2008-1, submitted as part of the application, discusses a change in set point for the Channel #3 low power scram from "5% full scale" to "1.0 E-12 amps." This was required due to a change to an autoscaling display for Channel #3. The conclusion of the safety evaluation states it has been reviewed in accordance with 10 CFR 50.59. Section 50.59(c)(1)(i) of 10 CFR states that "a licensee may make changes in a facility as described in the final safety analysis report....without obtaining a license amendment pursuant to 50.90 only if a change to the TS incorporated in the license is not required..." Please explain why this change to TS setpoint should not be required to meet this 10 CFR 50.59 requirement.