DWIGHT LOOK COLLEGE OF ENGINEERING

Department of Nuclear Engineering

TEXAS A&M UNIVERSITY

December 22, 2011

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

Docket No. 50-059

SUBJECT: Response to "Texas A&M University – Request for additional information regarding the Texas A&M University AGN-201M reactor license renewal application (TAC. NO. ME1588), July 25, 2011"

In response to the RAI dated July 25, 2011 Texas A&M University is submitting answers to the following questions: 6, 11, 32 and 37. In order to fully analyze and answer the remaining questions, the University is coordinating with Idaho State University's AGN facility to obtain a baseline measurement for response times of their analog console. Once these times are obtained, they will be used to compare the response times addressed in the RAIs above. The University is also in the process of implementing changes to the Technical Specifications as a result of the last RAI submittal. Once these changes have been approved by the Reactor Safety Board, the approved final draft of the Technical Specifications will be submitted. This process is estimated to be complete prior to February 15, 2012 at which time all documents will be submitted together.

If you have any questions, please do not hesitate to contact me at: (979) 862-1956, or e-mail at rjuzaitis@tamu.edu.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 22, 2011.

Sincerely,

Raymond J. Juzaitis

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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.

The digital software was designed such that a failure of the software will not interfere with the hardwired scram circuits. This is accomplished by having an isolated scram circuit. There is no interface between these hardwired scrams and the computer or software. The software also has protection against software failure, through the use of a watch-dog circuit.

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.

The Interlock Relay scram function was not incorporated into the Reactor Console Instrumentations and Electronics Upgrade. There was no Technical Specification requirement for this interlock feature. The interlock circuit was previously interrupted when any of the following took place: a rod drive system plug is removed, reactor tank temperature ≤ 20°C, earthquake switch opening, shield water level low, the relay chassis is removed, channel #3 low level, channel #2 low level, and channel #1 low level. With the exception of the rod drive system plug being removed and the relay chassis interlock, all other components of the interlock relay are provided for through scram functions. The two remaining interlocks are provided for in the console design and/or reinforced through operator action.

In the case of the rod drive system plug interlock, if any amphenol is removed from its associated rod that rod will scram and a loss of rod indication occurs for that rod. If this situation were to occur, the operator would initiate a manual scram of the reactor due to the loss of rod indication. As far as the relay chassis interlock, protection is provided through a loss of signal scram for each channel. The low power scram setpoint for each channel provides protection against a loss of electrical signal.

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.

The reactor nominally contains 9 stacked circular fuel disks with four fueled control rods that are inserted into the bottom of the core for operation. The control rods are removed and placed on an inspection stand when outside of the reactor. The stand places these 4 rods in a vertical orientation 30 cm apart. This was modeled using MCNP5 and produced an estimated K_{eff} of 0.04414 with a standard deviation of 0.00010. This is not considered a criticality concern.

The K_{eff} of the fuel is lowered when removed from the reactor tank due to the needed moderation and reflection provided by the internal reactor material during operation. A model was created using MCNP5 to model the fuel outside of the reactor. The fuel disks were modeled as a complete, stacked reactor core as it would appear if the whole core was simply removed from the reactor at one time. The K_{eff} of this was found to be 0.73949 with a standard deviation of 0.00037. This is below the allowable limit of K_{eff} for fuel.

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.

The change to Channel #3 low power scram from "5% full scale" to "1.0 E-12 amps" should require a license amendment to implement due to changing a technical specification requirement. Modification Authorization 2008-1 should have resulted in a request to amend the license of Docket No. 050-059. All reactor operations have ceased until a license amendment is approved. License amendment no. 14 was submitted to the USNRC on September 5, 2011.