**DWIGHT LOOK COLLEGE OF ENGINEERING**

**Department of Nuclear Engineering**

TEXAS A&M UNIVERSITY

August 25, 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)"

In response to the RAI dated July 25, 2011, the following questions are being submitted for review. Questions 1,2,3,4,5,14,15,16,17 and 26 are enclosed.

If you have any questions, please do not hesitate to contact me at: (979) 845-4161, or e-mail at rjuzaitis@tamu.edu.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 25, 2011.

Sincerely,

Raymond J. Juzaitis, Ph.D.
Department Head

Encl. (1)

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Texas A&M University
AGN-201M
License No. R-23
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Responses to Request for Additional Information
Questions 1,2,3,4,5,14,15,16,17 and 26

1. The subcritical assembly consists of 192 slugs of natural uranium, organized into 48 fuel pins each containing four slugs. The cladding for the slugs is 1.0 mm of aluminum, and the slugs are annular with an inner diameter of 1.3 cm, an outer diameter of 3.0 cm, and a length of 21.3 cm. The tank can be filled with approximately 700 gallons of water.

   The purpose of the subcritical assembly is undergraduate and graduate laboratory use. The subcritical assembly can be used in comparison to Monte Carlo calculations and in determining criticality ($k_{eff}$) changes by varying fuel geometry.

   The subcritical assembly is not used simultaneously with AGN-201M reactor operations, thus the assembly does not affect reactor operations. The facility poses no public safety or environmental issues. The combination of natural uranium fuel in a light water moderator cannot become critical, and the aluminum cladding contains all the natural uranium.

2. The most recent calculation of shutdown margin and excess reactivity was performed on March 29, 2011. Shutdown margin is calculated by subtracting the stuck rod configuration reactivity from the critical configuration reactivity corrected to 20°C. The stuck rod configuration reactivity assumes the most reactive rod remains in the core as well as the maximum experiment loading. Excess reactivity is determined by obtaining a critical rod configuration with Safety Rod #1, Safety Rod #2 and the Coarse Control Rod full in. The Fine Control Rod (FCR) position is recorded. The FCR position is used to obtain a reactivity value from the FCR calibration curve. This value is subtracted from the most recent rod calibration value for the FCR, giving the total excess reactivity.

3. The cadmium shutdown rod is an 83" long by 0.75" diameter circular aluminum rod with cadmium insert. This rod is inserted into the glory hole after the reactor has been shut down and removed prior to operation.

4. On November 7, 1973 the reactor tank was opened and the core tank removed for the purpose of replacing the lower o-ring seals. An inspection was performed at this time and showed that the 12 lower cap screws had some surface corrosion. The lower core half and reflector were in excellent condition, and no corrosion was observed on the inner surface of

the core tank. Due to an issue that arose while conducting a pressure drop test, the core and reflector were removed from the core tank on November 8, 1973. The cap screws used to attach the core tank lower plate were determined to be too short, and the cap screws were replaced with longer screws. The core and reflector were reinstalled in the core tank, and the core tank was reassembled. The core tank was replaced in the reactor tank and the thermal column replaced.

Inspections of the shield water tank are performed annually. These inspections occurred annually up to 1999 and started again in 2010 moving forward. There are no plans for regular monitoring or assessment of the core tank and reactor tank due to current facility configurations.

Annual inspections of the shield water tank take place during SWTI-12, Maintenance Procedure for Conducting a Detailed Shield Water Tank Inspection. The purpose of this preventative maintenance procedure is to assure mechanical integrity of the reactor shield water tank and to verify that shield water anticorrosion chemicals are within specifications. The pH of the water shall measure greater than 8.0, and the anticorrosion chemicals added as needed are Sodium Chromate, Sodium Hydroxide, and Sodium Silicate. A visual inspection of the inside and outside of the shield water tank is then performed.

5. The following is taken from AGN Preliminary Design Study (License F-32) and is applicable to the Texas A&M AGN-201M reactor.

<u>Fusing system</u>

The concept of a fuse for a reactor is directly analogous to the electrical fuse used in every household. The basic concept involves having a safety device that prevents large power overloads, and yet is almost foolproof (not depending upon electronic circuits) and is not easily sabotaged. The AGN 201 reactor core fuse is made of polystyrene containing 100 milligrams of U-235 cm-3 that acts as the support for the bottom half of the reactor core and a section of the reflector. The load on this fuse is some 15 Kg. Most of the stress in the fuse is in compression and sheer so as to circumvent any possible creep problem of polystyrene in tension.

The higher loading density is used to generate heat at a higher rate in the fuse than in the core, such that the fuse rises in temperature about twice as fast as does the core proper. At about 100°C, the fuse melts and the core separates completely, thereby shutting down the reactor in the event of an accidental runaway. Polystyrene is used as the fuse material rather than polyethylene because of its resistance to changes in physical properties induced by radiation. Experiments indicate that the melting point of polystyrene is unaffected by radiation doses below 100 megarep. Thus, the properties of the fuse are not affected by several severe nuclear excursions, nor by normal operation for a score of years.

Care has been taken in the design of the reflector plug to insure that the plug actually drops after the fuse melts. Ample clearance and the tapered, hourglass design have been provided to insure a free fall. The separation of the core reduces the reactivity by at least 5%, and more likely 10%.

14. The annual radiation survey is performed at 4 different power levels. There are 11 survey points in the facility where surveys are taken. These surveys are taken at zero power, 1 watt, 3 watts, and 5 watts. The survey locations are depicted on page(s) 4-23 and 4-24 of the most recent SAR. This survey is performed by radiological safety staff members and is a part of the preventive maintenance program. Monthly radiation/contamination surveys are also taken of the facility. Additionally, prior to reactor operations a zero power survey is conducted. The facility includes three phantom monitors, used to provide a baseline for personnel dose received. These dosimeters are read quarterly along with facility personnel dosimetry.

    Due to the unchanging core configuration and low power levels at this facility, current survey practices are deemed adequate to comply with ALARA considerations.

15. Radioactive argon-41 is produced by neutron reactions with air in the vicinity of the reactor core. Air may be contained in experimental facilities (glory hole & access ports) and is in solution in the tank water. Experience with the AGN reactors operating at higher power has shown that no significant release of Ar-41 occurs from the glory hole during power operation at 5 W or less. The Naval Post Graduate School found some Ar-41 activity by irradiating a sample of air at atmospheric pressure in a closed tubular container just filling the AGN glory hole to the boundaries of the core. The irradiated air was transferred to a chamber counter with thin-walled glass G-M tube. Decay was followed over approximately one half-life and was consistent with the decay of Ar-41. The measured activities agreed with those estimated from a calculated efficiency of the counter. On the basis of Naval Post Graduate School operating experience, Ar-41 will not be formed in significant concentrations under the skirt at operation of 5 W. Since the resulting peak Ar-41 activity for the air volume in a sealed empty glory hole is only 45 times greater than the maximum permissible concentration (MPC) value of Ar-41, natural diffusion and mixing of this irradiated air volume will easily reduce the average air activity in the vicinity of the reactor to less than 1% of MPC values for uncontrolled areas. Also, the reactor area is presently and will continue to be a controlled area with limited access. Thus, no hazard from Ar-41 is anticipated, as shown below.

The maximum equilibrium concentration of Ar-41 produced can be easily calculated. The saturation reaction rate, $\Re$, for Ar-41 production is given by

$$\Re = \Sigma_\gamma \phi$$

$$= \sigma_\gamma \frac{\zeta m N_A}{A} \phi \qquad (1)$$

where $\sigma_\gamma$ = microscopic cross section for $^{40}Ar(n, \gamma)^{41}Ar$ [$cm^2$],

$\zeta$ = natural abundance of Ar-40 [dimensionless],

m = the mass of Ar-40 contained within the volume of the glory hole fully contained within the reactor core [grams],

$N_A$ = Avogadro's number [$mol^{-1}$],

$A$ = atomic mass of Ar-40 [grams $mol^{-1}$], and

$\phi$ = average thermal neutron flux [n/$cm^2$-s].

The mass of Ar-40 is calculated based on the assumption that the air entrapped within the glory hole is a dry, ideal gas at standard temperature and pressure with argon comprising 1.3% of air by mass. Thus, there are 6.2 mg or 6.2 x $10^{-3}$ g of Ar-40 contained within the 394 $cm^2$ of air entrapped within the portion of the glory hole fully contained within the reactor core. At 5 W, the average thermal neutron flux is 1.5 x $10^8$ n/$cm^2$-s. From the Chart of Nuclides the microscopic cross section for Ar-41 production is 0.65 barns (6.5 x $10^{-25}$ $cm^2$), the natural abundance of Ar-40 is 99.6003%, and the atomic mass of Ar-40 is 39.96238 grams/mol. The resulting Ar-41 production rate is 9,073 atoms/s. In order to obtain the Ar-41 production rate in terms of activity an activity equation must be used.

$$A = \lambda N$$

or in this case

$$= \frac{\ln(2)}{t_{1/2}} \Re \qquad (2)$$

where $A$ = activity production rate [Bq/s]

$t_{1/2}$ = half-life of Ar-41 [sec]

The Chart of Nuclides gives the half-life of Ar-41 to be 1.83 hours. Once the activity production rate is found, conversion to curies was done using the relationship, 1 Ci is equal to 3.7 x $10^{10}$ Bq. Thus, the resulting production rate in terms of activity is 2.58 x $10^{-11}$ Ci/s or 25.8 pCi/s.

The equilibrium concentration of Ar-41 in the reactor room at a power level of 5 W is given by

$$C_{41} = \frac{L}{V_R r} \tag{3}$$

where $C_{41}$ = concentration of Ar-41 [μCi/cm$^3$],

$L$ = leakage rate of Ar-41 from the core (assumed to be equal to the production rate) [μCi/s],

$V_R$ = the volume of the reactor room [cm$^3$], and

$r$ = fractional volumetric exchange rate of air in the reactor room equal to the ventilation flow rate, Q [cm$^3$/s], divided by the reactor room volume.

The dimensions of the reactor room are 29 ft x 25 ft x 12 ft which give a volume of 8700 ft$^3$ (2.46 x 10$^8$ cm$^3$). During normal operation, the ventilation flow rate is 250 cfm (1.18 x 10$^5$ cm$^3$/s) as measured August 1, 2011. Assuming the Ar-41 leakage rate from the core equal to the production rate of 2.58 x 10$^{-5}$ μCi/s, then substitution of these numerical values into Equation (3) yields an equilibrium concentration for 2.19 x 10$^{-10}$ μCi/cm$^3$.

From 10 CFR20 Appendix B, the limiting values for the Derived Air Concentration (DAC) and the effluent concentration for Ar-41 are 3 x 10$^{-6}$ μCi/ml and 1 x 10$^{-8}$ μCi/ml, respectively. DAC values establish the limiting concentrations of airborne radioactive materials to which occupational workers may be exposed while the effluent concentration provides limiting concentrations for the general public. Thus, the maximum equilibrium concentration at 5 W is 0.0073% of the DAC limit and 2.19% of the effluent concentration limit.

16. The ALARA program is designed to prevent excessive exposure to employees and the general public. The implementation of this policy at Texas A&M University comes directly from procedures created by the Environmental Health and Safety Department. These procedures as well as detailed instructions are presented in the "Radiological Safety Program Manual" which was last revised in July, 2004. ALARA is addressed in section 5 of the manual which sets forth radiation dose limits that comply with 10 CFR 20 limits. Best practices to comply with ALARA principles are outlined to control internal and external exposure.

The Radiological Safety Officer directs the radiation safety program at the University. The radiological safety staff is charged with maintaining the personnel dosimetry program for the facility. The facility is responsible for ensuring ALARA principles are implemented while the individual is responsible for maintaining these principles.

17. Per procedure, the activity of all experiment samples removed from the glory hole or access ports is monitored with a portable radiation survey instrument prior to removal. During operations and experiments facility personnel are equipped with a personal dosimetry badge and pocket ion chamber. Personnel minimize time near the sample and maximize shielding and distance from samples after removal from the reactor.

26. The provision "Reactor startup cannot commence unless both safety rods and coarse control rod are fully withdrawn from the core" was intentionally removed from the proposed Technical Specifications. This provision was removed to allow reactor recovery during rod calibrations. Previously all rods would have to be withdrawn from the reactor in order to continue rod calibrations. Removing this provision allows the rod being calibrated to be returned to the core in order to continue rod calibration of the other rods.