STATE OF RHODE ISLAND AND PROVIDENCE PLANTATIONS

RHODE ISLAND ATOMIC ENERGY COMMISSION
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Mr. William B. Kennedy, Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D.C. 20555-0001

December 7, 2010

Re: Renewal of License No. R-95
Docket No. 50-193

Dear Mr. Kennedy:

We are responding to your requests for additional information (RAIs) regarding aging issues raised in those RAIs. We have expressed our concerns over those aging issues in an earlier letter.[1] We will first repeat the RAI and follow that with our response. The RAIs addressed in this letter are RAI NRC Numbers 4.5, 4.8 and 4.9 from your letter dated April 13, 2010.

RAI 4.5 Section 4.2.3 states that the graphite reflectors are designed for expansion "from an integrated flux of $2X10^{21}$ nvt (expansion based on a more than two-year, full-power operation factor)." Given that the TS do not explicitly limit the duration of full-power operation, provide a discussion of the methods used to ensure that the graphite reflectors will not be exposed to an integrated neutron flux greater than the expansion design basis (e.g., calculation of integrated flux, surveillance programs, etc.). The discussion should include consideration of current integrated flux and integrated flux during the period of the renewed license.

Response: The graphite reflector element is a block contained in a 3-inch square aluminum can with handles to allow remote handling. It should be noted that graphite has been used for many years as a reflector and crumbling or other catastrophic failures of graphite pieces have never occurred. AGOT reactor-grade graphite is used to avoid

[1] Letter dated December 2, 2010 from H. J. Bicehouse, Assistant Director to W. B. Kennedy, Project Manger, USNRC

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trace boron contamination. Graphite[2] is known to undergo several changes when exposed to neutron irradiation:

- Dimensional change due to neutron induced swelling
- Elastic modulus change as measured by the impulse excitation technique
- Coefficient of thermal expansion change
- Thermal conductivity change
- Electrical resistivity change
- Irradiation-induced creep

SAR Section 4.2.3 discusses the expansion of graphite due to irradiation and gas evolution. The design allows for a maximum increase in graphite dimensions of 1.1% due to irradiation growth and gas evolution. The SAR suggests that the graphite will be fine up to an integrated flux of 2×10^{21} neutrons /cm^2. Our maximum flux at the center of the core is estimated to be 10^{13} n/cm^2-s. Therefore to reach 2×10^{21} nvt:
[2×10^{21} n/cm^2] / [10^{13} n / cm^2-s] = 2×10^8 seconds or about 5.6×10^4 hours of operation. At 7 hours/day operation, this amounts to 8,000 days of full-power operation or about 22 years of full-power operation. It should be noted that the graphite reflectors are now on the periphery rather than at the center of the core (LEU core configuration). The neutron flux at that location is at least an order of magnitude less than that at the center of the core. Thus, a conservative estimate of the time needed to reach the integrated flux would be 80,000 days or more of full-power operation. That amounts to about 219 years or well beyond the requested period of the renewed license.

RAI 4.8 Section 4.2.5. Provide justification for the design of the core support structure as to its ability to support the weight of the core and its ability to withstand radiation damage, mechanical stress, and chemical degradation over the period of the renewed license.

Response: Relative to commercial power reactors, the RINSC reactor operates at very low power, temperature, and pressure. Consequently, damage to the core support structure due to radiation exposure and thermal aging will be significantly less than the damage typical of power reactors. No reports of significant radiation damage to core components of small research reactors have been published. Since the power industry does damage studies to show that their facilities can continue to operate safely with extended lifetimes, it is reasonable to assume that research reactors can safely operate within similar lifetimes.

The reactor core support consists of a suspension frame which is bolted to a moveable bridge, operated by a hand crank, which can relocate the entire core plus core support structure to various positions in the reactor pool. The four corners of the structure are occupied by the suspension posts. These corner posts connect the grid plate to the reactor bridge. The core suspension system includes a locating plate, made of heavy steel that spans the upper end of the suspension frame to provide support and location for the control blade drive mechanisms. The control blade drive guide tubes are flanged to the bottom of this locating plate. Core elements are contained in a grid box that is enclosed

[2] Marsden, B.J., Preston, S. D., Wickham, A. J., "Evaluation of graphite safety issues for the British production piles at Windscale," IAEA-TECDOC-1043, September 1997.

on four sides to confine the flow of cooling water between elements (See Fig. 4-2 of the SAR). The grid box assembly, including the drive mechanisms, is supported by the suspension frame. The elements that make up the core sit on a 7 x 9 grid plate with the four corner positions occupied by the suspension frame corner posts.

This core support system was designed to support the weight of the core plus control and cooling elements. The design has satisfactorily supported the weight of the components for over forty years and there is no credible reason why it should not continue to function as designed. No appreciable deterioration of any components of the support structure has been seen during inspections.

The core support structural materials are predominantly made of 6061-T6 Al. In order to minimize corrosion of the aluminum, reactor pool water pH and conductivity (resistivity) levels are measured weekly to verify that the values are within the RINSC Technical Specification limits (pH between 5.5 and 7.5; resistivity greater than 500 kΩ/cm). As described in Section 5.5.1 of the SAR, make-up water for the pool passes through a five micron filter, an activated charcoal filter, two mixed bed demineralizers and a one micron filter before entering the pool. The pH and conductivity are measured weekly to verify that the water in the pool is within the specification limits.

The core fuel cladding material is also 6061-T6 Al. We perform an annual fuel element inspection that would provide another indication of whether or not aluminum core materials are beginning to suffer from corrosion, radiation damage or thermal stress.

RAI 4.9 Section 4.4. Discuss the ability of the biological shield and pool liner to continue to meet their design bases during the period of the renewed license. Include considerations of radiation, chemical, and thermal degradation. Describe any surveillance programs in place to detect degradation of the biological shield and pool liner.

Response: Relative to commercial power reactors, the RINSC reactor operates at very low power, temperature, and pressure. Consequently, damage to the biological shield and the pool liner due to radiation exposure and thermal aging will be significantly less than the damage to similar structures typical of power reactors. No reports of significant radiation damage to biological shields or pool liners of small research reactors have been published. Since the power industry does damage studies to show that their facilities can continue to operate safely with extended lifetimes, it is reasonable to assume that research reactors can safely operate within similar lifetimes.

The reactor pool is surrounded by thick (minimum 10 ft) reinforced concrete. The inner surface of the concrete is lined with ¼-in. thick aluminum. The liner is the primary pool water containment vessel. For the same reasons given in response to RAI 4.8, corrosion of the aluminum liner is expected to be minimal over the extended lifetime of the reactor. The same monitoring (pH and conductivity) used for the aluminum fuel clad would also alert operators to any corrosion of the liner. Required annual inspections of the pool supplement weekly water monitoring to confirm the integrity of the pool liner. Any significant degradation of the liner, whether from chemical or mechanical causes, that could lead to pool water leakage would be detected by routine monitoring of the makeup water system.

The combination of the water pool and the surrounding concrete provide a biological shield for facility workers that keeps the dose rate below 1 mrem/hr at all points above and outside the pool area (SAR, Section 4.4). Water level monitors and radiation monitors ensure that the water depth is sufficient (approximately 24 ft) to shield personnel near the top of the pool. During routine operations radiation surveys are performed to monitor dose rates (SAR Ch. 11).

The radiation attenuation properties of the pool water are based on the nuclear properties of the water and the attenuation level will not change over time as long as the water level is maintained. The aluminum liner is a minor contributor to the biological shielding relative to the water and concrete, but as explained above it is not expected to deteriorate over the lifetime of the facility. While concrete is susceptible to thermal and radiation damage, the low power and low temperature of the RINSC reactor will not lead to any degradation of the concrete over the lifetime of the reactor.

A survey of aging effects on concrete was performed at the Idaho National Laboratory[3]. According to this report, for conditions of radiation flux up to 2×10^{19} nvt (thermal) and temperatures to 120 °C, radiation damage to the type of concrete used in our facility was insignificant, while other types show considerable loss of strength (specifically high alumina cement concrete). All effects on concrete due to radiation, per se, were too slight to reliably measure because of the gross effects from the increased temperature during exposure. Generally speaking, the threshold of degradation in the concrete is approximately 95 °C.

The neutron flux at the core end of the beam ports (basically the inner surface of the concrete shielding) is approximately 1 to 4×10^{12} n/cm^2-s. Assuming 40 hours of operation per week, ten years of full-power operation would just approach the 10^{19} nvt threshold for the most susceptible type of concrete (not used at the RINSC). However, as noted in the referenced report, radiation damage is basically not measureable compared to temperature effects. The safety limit for the pool water temperature is 130 °F (54.4 °C). This is substantially below the 95 °C threshold value for thermal damage to the concrete. Based on these considerations it can be concluded that the biological shield will not deteriorate over the extended lifetime of the facility.

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[3] *Literature Review of the Effects of Radiation and Temperature on the Aging of Concrete, INEEL/EXT-04-02319, September 2004*

If you have questions regarding this letter, please address them to the undersigned.

Very truly yours,

Terrence Tehan Ph.D., Director
Rhode Island Atomic Energy Commission

I certify under penalty of perjury that the representations made above are true and correct.

Executed on: DEC 7 2010 By: ____________ Ph.D