September 3, 2015

Dr. Cameron Goodwin, Director
Rhode Island Nuclear Science Center
Rhode Island Atomic Energy Commission
16 Reactor Road
Narragansett, RI 02882-1165

SUBJECT: RHODE ISLAND ATOMIC ENERGY COMMISSION - REQUEST FOR ADDITIONAL INFORMATION FOR THE RHODE ISLAND NUCLEAR SCIENCE CENTER REACTOR LICENSE RENEWAL (TAC NO. ME1598)

Dear Dr. Goodwin:

The U.S. Nuclear Regulatory Commission is continuing the review of the Rhode Island Atomic Energy Commission application for the renewal of Facility Operating License No. R-95 by letters dated May 3, 2004, as supplemented, for the Rhode Island Nuclear Science Center reactor. During our review, questions have arisen, which we require additional information and clarification. Please provide responses to the enclosed request for additional information no later than 90 days from the receipt of this letter.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.30(b), your response must be executed in a signed original under oath or affirmation. You must submit your response 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.”

If you have any questions regarding this review, please contact me at 301-415-3936 or by electronic mail at Patrick.Boyle@nrc.gov.

Sincerely,

/RA/

Patrick G. Boyle, Nuclear Engineer
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-193

Enclosure:
Request for Additional Information

cc: See next page

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Test, Research, and Training Reactor Newsletter
University of Florida
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Gainesville, FL 32611

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>FOR THE LICENSE RENEWAL FOR</u>

<u>THE RHODE ISLAND ATOMIC ENERGY COMMISSION</u>

<u>RHODE ISLAND NUCLEAR SCIENCE CENTER REACTOR</u>

<u>LICENSE NO. R-95</u>

<u>DOCKET NO. 50-193</u>

The U.S. Nuclear Regulatory Commission (NRC) is continuing the review of your application for renewal of Facility Operating License No. R-95 for the Rhode Island Nuclear Science Center (RINSC) reactor submitted by letters dated May 3, 2004 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML041270519 and ML14038A386), as supplemented.

The following requests for additional information (RAIs) are related to the Safety Analysis Report (SAR) submitted May 3, 2004, as supplemented with additional correspondence. The number of each RAI begins with the corresponding chapter of the SAR. The numbering sequence continues from the previously-identified RAIs. Responses to the RAIs should be in the form of discussion or analysis, or both. The responses must provide sufficient information for the NRC staff to independently verify all safety-related conclusions. Response to the RAIs may be in the form of replacement pages for the SAR. NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Application for the Licensing of Non-Power Reactors: Format and Content," dated February 1996, contains guidance for providing sufficient information to satisfy the related regulatory requirements in Title 10 of the *Code of Federal Regulations* (10 CFR).

RAI 4.34

The guidance in NUREG-1537, Section 4.5.1, "Normal Operating Conditions," discusses both the limiting core configuration (LCC) and operational core configuration (OCC), and requests the licensee to characterize the neutronic conditions that are applicable. The conditions include power distributions, control rod worth, and core reactivity. The reviewers are considering the material submitted in SAR Chapter 4 to determine whether the LCC models used to support the safety analysis are acceptable. Acceptability of the model is determined by comparing the calculated and measured behavior of the OCC. In SAR Table 4-2, the excess reactivity values for 10 different core configurations are presented. In SAR Table 4-3, several reactivity coefficient calculations are presented and it is stated in the following paragraph that measurements are taken for confirmation.

a. Provide measured excess reactivity values for comparison to the calculated estimates in Table 4-2.
b. Provide a tabulation of control rod worth calculations and measurements for the core conditions described in Table 4-2.
c. Provide confirming measurements for the coefficient calculations in Table 4-3.

Enclosure

d. Describe the core condition that is considered to represent LCC conditions (i.e., the core with the highest power density at full power or at the limited safety system Setting (LSSS) set point (please specify)). For that core, provide the peak fuel assembly power as well as the core and fuel plate peaking factors that are required to support safety analysis.

RAI 4.35

The guidance in NUREG-1537, Section 4.6, “Thermal-Hydraulic Design,” states that the licensee should take into account uncertainties in thermal-hydraulic and nuclear parameters. SAR Section 4.6.1 discusses a report titled “Report on the Determination of Hot Spot Factors for the Rhode Island Nuclear Science Research Reactor Using LEU [low-enriched uranium] Fuel (Reference 4-Y).” If this information is used in support of the application, describe where and how it is used, and provide a copy for staff review.

RAI 4.36

The guidance in NUREG-1537, Section 4.5.3, “Operating Limits,” requests an analysis of a rod withdrawal accident. The response to RAI 4.33 (correspondence dated March 15, 2013) provides the requested analysis. The methodology appears to be consistent with current guidance and the appropriate safety limit (SL) (i.e., fuel temperature) is cited as a basis for acceptability; the temperatures attained (~80°C) is suitably below the safety limit (530°C).

a. Confirm that the core configuration used for this analysis is the LCC and the cited reactivity insertion rate (0.02%Δk/k per second) establishes the maximum control rod reactivity insertion rate.
b. Propose a technical specification (TS) with the reactivity rate converted into a linear withdrawal rate or provide an explanation why a TS is not required to protect this safety analysis assumption.

RAI 4.37

The guidance in NUREG-1537, Section 4.6, “Thermal-Hydraulic Design,” requests a thermal-hydraulic (T&H) analysis for the LCC (see RAI 4.34, above). The T&H analysis results are submitted under forced and natural convection conditions. However, the acceptability of the results are quoted in terms of: (1) the power at which the onset of nucleate boiling is predicted to occur; (2) the onset of flow instability; and (3) the onset of critical heat flux. There are no conclusions based upon the SL for RINSC (530°C) or the guidance in NUREG-1537 (departure from nucleate boiling ratio [DNBR] > 2.0). For the LCC operating at the LSSS, document the results of T&H calculations showing compliance with the SL and the cited DNBR guidance. Ensure that the peaking factors determined from the LCC are used. If such analysis is provided using a computer code, provide the supporting documentation demonstrating that the code(s) in question are validated for such purposes and fully describe the model used.

RAI 13.26

The guidance in NUREG-1537, Section 13.2, “Accident Analysis and Determination of Consequences,” states that analysis should consider functions and actions assumed to occur

that change the course of the accident or mitigate the consequences, such as reactor scram. The rising power transient analysis, in response to RAI 13.7 (correspondence dated September 8, 2010), uses a previously-approved method, but concludes acceptability based upon a trip value of 2.3 megawatt (MW) (the LSSS is 2.1), and cites a 2.4 MW SL, which is not the value in the TS. It is not clear which flow rates and feedback coefficients are used or whether the analysis is suitably conservative. Provide a revised analysis for the LCC using limiting peaking factors, conservative conditions, approved methods, and show that the TS SL (530°C) is not exceeded.

RAI 5.2

The guidance in NUREG-1537, Section 5.2, "Primary Coolant System," requests that the licensee describe fully the reactor coolant system. The Safeguards Report for the Rhode Island Open Pool Reactor (1962) identifies a single primary coolant loop. SAR Section 5.2.1.4 identifies two primary coolant loops. The presence of two connected heat exchangers impacts the review for potential loss-of-coolant accidents.

a. Identify the means used for making this change to the facility.
b. Describe all piping and electrical connections to this system.
c. Explain whether the second system constructed has undergone any functional tests and if so, describe how those tests were accomplished.
d. Explain whether the second system has previously been operated, is currently operable, and if so, describe the operational characteristics, performance attributes, the conditions under which this system is used or usable, and modify your submittal for Chapter 5 to incorporate the required documentation.
e. Explain whether both primary loops are assumed to be operating in any of the safety analyses submitted for this review.

RAI 6.2

The guidance in NUREG-1537, Chapter 6, "Engineered Safety Features," requests that the licensee describe fully any engineered safety feature in use. Section 6.2.3, "Emergency Core Cooling System [ECCS]," identifies specific information that must be included for a system relied upon "... to remove decay heat from the fuel to prevent failure or degradation of the cladding if cooling is lost." The Safeguards Report for the Rhode Island Open Pool Reactor (1962) does not identify an ECCS. SAR Section 1.7.4.1 states that such a system was added as part of the power upgrade program. However, SAR Chapter 6 does not include a description of this system. SAR Section 1.7.4.1 does state that the system was installed. In addition it states, in part, that "the primary pump diaphragm isolation valves were replaced with low d/p [differential/pressure] butterfly valves. As part of the primary flow upgrade, new components with a larger range were added to the reactor flow sensing circuits to monitor the increased flow capability."

a. Identify if the installed system is intended to be an ECCS.
b. Identify the means for making this change to the facility.
c. Describe all piping and electrical connections of this system and how such connections interact with the primary coolant system.

d. Explain whether this system has undergone any functional tests and if so, describe how those tests were accomplished.
e. Describe how use of the system will not degrade any systems relied upon for safety (e.g., what prevents important to safety electrical connections from being wetted?).
f. Explain whether this system is currently operable or has been operated, and if so, describe the operational characteristics, performance attributes, the conditions under which this system is used or usable, and revise Chapter 6 of the SAR to incorporate the required documentation.
g. Explain whether this system has been modeled and/or included for the safety analyses submitted for this review.

RAI 13.27

The guidance in NUREG-1537 requests that a spectrum of potential accidents be evaluated to demonstrate acceptable levels of safety for the facility. The accidents evaluated generally fall into two categories: (1) accidents for which dose consequences need to be evaluated; and (2) accidents for which SL compliance needs to be evaluated. In the SAR, the accidents analyzed for SL compliance, the SL of 2.4 MW is consistently used. This is not the SL in the latest version of the RINSC TSs. Provide a comprehensive restatement of your safety analysis for all applicable accidents ensuring that the analysis uses conservative peaking factors, feedback coefficients, and flow conditions. In the case of accidents or transients for which power is an important component, ensure that the LSSS power is used in the evaluation. Demonstrate that the TS SL - 530°C - is conservatively bounded by these analyses.

RAI 13.28

NUREG-1537 requests that the licensee identify and document the analysis of the maximum hypothetical accident. In the responses to RAIs 13.2 through 13.5 the licensee has provided this analysis. However, the following issues are noted and need to be addressed:

a. The 200-MeV per fission the fission rate is estimated to be ~3.1×10^{10} fissions per watt-sec. The analysis uses 3.1×10^{11} fissions per watt-sec. Revise the analysis with the corrected value and provide the result or explain why your values are correct.
b. Per 10 CFR Part 20, Appendix B, a breathing rate volume under light work condition is 20 liters per minute. The analysis uses 2 liters per minute. Revise the analysis with the corrected value and provide the result or justify the use of your value.
c. It appears that the derived air concentration value of I-130 is incorrectly used for I-131. Revise the analysis with the corrected value and provide the result or justify the use of your value.
d. The analysis does not appear to consider the fuel plate power peaking factors in determining the radionuclide inventory in the damaged fuel plate. Revise the analysis to use the hot plate power density and provide the results or justify your methodology.
e. The atmospheric dispersion calculation refers to an effective x/Q, which includes a factor of 1/3 for the building dispersal effects. This assumption differs from the guidance in Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” which recommends

a horizontal plume meander in a building wake. Revise the analysis with the corrected values or provide additional justification for the factors that were used.

f. The noble gas and halogen release fractions are based on the RINSC interpretation of University of Virginia fuel failure results without justification. In general, the inventory of fission gas release depends on the extent of damage, fuel temperature, cladding integrity, and location (under water or in air). Release assumptions should be consistent with the expected conditions of the fuel plate at the time of the accident. Provide a technical justification for the use of this data.

g. The build up of Kr-85 is mainly through the isomeric transition of Kr-85m. Given the yield for the 85 atomic mass (1.317 percent), the Kr-85 yield is about 0.277 percent (1.317% x 0.211); the analysis uses a yield of 0.026 percent. Revise the analysis with the corrected value and provide the result or explain why your value is correct.

h. The reactor building concentration is based on a volume of 6.15×10^9 cm^3, whereas TS 5.1 provides a volume of 5.15×10^9 cm^3. Revise TS 5.1 to define the free volume in the reactor building; if a volume other than the TS 5.1 value is used, provide the supporting justifications and the related calculations. A larger free volume is considered to be less conservative.

i. The analysis uses a charcoal filter efficiency of 0.99 for iodine removal. The basis for this efficiency, given the conditions that can adversely affect adsorption, is not provided. Although TS 3.5.2 does provide a limiting condition for operation requiring the filter for operation, there is no acknowledgment of the potential for filter efficiency degradation over time and the resulting required replacement frequency that should be used to maintain the cited efficiency. Describe the methodology in place at RINSC that will ensure the cited efficiency is maintained.

j. There appears to be transposition errors in the calculations. The dose conversion factor for I-132 from FGR 11 should be 0.381 mrem/μCi (it is cited as 0.011). In addition, the release of Kr-85 to the reactor building is increased by a factor of 3.7 without any explanation. Justify the conversion factor utilized and revise the analysis with the corrected values and provide the results for any updated correction factors.