September 14, 1992

Docket No. 50-193

Mr. Terry Tehan, Director
Nuclear Science Center
Rhode Island Atomic Energy Commission
South Ferry Road
Narragansett, Rhode Island 02882-1197

Dear Mr. Tehan:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION

We are continuing our review of the Rhode Island Atomic Energy Commission's proposal of November 18, 1991, as supplemented on July 2[illegible], 1992, for conversion to Low Enriched Uranium (LEU) fuel for Facility License No. R-95. During our review of your application, questions have arisen for which we require additional information and clarification. Please provide responses to the enclosed Request for Additional Information within 30 days of the date of this letter. Following receipt of the additional information, we will continue our evaluation of your application. If you have any questions regarding this review, please contact me at (301) 504-1128.

This request affects nine or fewer respondents and, therefore, is not subject to Office of Management and Budget review under P. L. 96-511.

Sincerely,
- original signed by -
Marvin M. Mendonca, Senior Project Manager
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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Rhode Island Atomic Energy Commission		Docket No. 50-193

cc:

President, Town Council
Town of Narragansett
Town Hall
Narragansett, Rhode Island 02882

Governor of Rhode Island
Providence, Rhode Island 02903

REQUEST FOR ADDITIONAL INFORMATION

RHODE ISLAND ATOMIC ENERGY COMMISSION

DOCKET NO. 50-193

1. Since the safety analysis has changed and the response to this request for additional information appears to be relatively extensive, a revised version that incorporates and integrates the submittals and the responses to the questions that follow should be provided.

2. For Part A, page 1 of the November 1991, Safety Analysis Report (SAR), verify that the date of the amendment to permit operation at two megawatts (Mw) was September 12, 1988 versus September 10, 1968.

3. For Part A, pages 2 through 4 on the "DESCRIPTION OF REACTOR SYSTEMS" in the November 1991, SAR, describe the function of the regulating rod on reactor trip.

4. For Part A, pages 6 through 14 of the November 1991, SAR, provide a thermal flux profile for the LEU and HEU cores at full power for the all rods out and the rod in configurations. Provide the peak-to-average power ratio that was assumed for the HEU core.

5. For Part A, page 14, Table 3 of the November 1991, SAR, provide rod worth values for each of the safety blades, and which safety blades contributed to the calculated shutdown margin.

6. With regard to temperature reactivity coefficient:

   a. Provide clarification of the discrepancy between the temperature reactivity coefficient in Part A, page 15 of the November 1991, SAR which indicates values as low a -0.82×10^{-4} percent delta k/k/degree Celsius, and the water temperature reactivity coefficient (calculated density only) in the proposed Technical Specification K.3.e.(3)(c)1. which indicates a value of 0.82×10^{-4}/degree Celsius. That is, is the value negative or positive?

   b. Verify that the previous Technical Specification temperature coefficient was for density only. If not provide the corresponding value for the LEU core or justification for the use of the density only value.

   c. Provide a description of the techniques used to verify that this parameter (or the total water temperature reactivity coefficient) is correct for each core change.

7. With regard to the void reactivity coefficient

   a. Provide clarification of the discrepancy between the void reactivity coefficient (calculated) in Part A, page 15 of the November 1991, SAR which indicates values as low a 2.7×10^{-3} delta k/k/percent void and the void reactivity coefficient in the proposed Technical Specification K.3.e.(3)(c)2 which indicates a value of -2.7×10^{-3} delta k/k/percent void. That is, is the value negative or positive?

   b. Provide a description of the techniques used to assure or verify that this parameter and the reactor power reactivity coefficient are correct for each core change

8. For Part A, page 15 of the November 1991, SAR, provide design details for the replacement regulating rod or provide reference to descriptions and drawing of the replacement regulating rod.

9. For Part A, pages 6 through 15 of the November 1991, SAR, provide design details for the beryllium reflectors and the beryllium central core element, or provide reference to descriptions and drawing for these components.

10. For Part B, page 10 of the November 1991, SAR, verify that the analysis for high power and low flow conditions was a steady state analysis. If this is the case, provide an estimate of the transient effect. That is, for an increasing power level or decreasing flow rate at the time of the scram, provide an estimate of the effect on the fuel surface temperature.

11. Provide a calculation of the fission product inventory for the Low Enriched Uranium (LEU) fuel and core. Compare that calculations results to the fission product inventory for the High Enriched Uranium (HEU) fuel and core. Determine if the difference changes the assumptions and results of the radiological dose assessments in the previous SAR.

12. Provide the bases for the change to Technical Specification (TS) Table F.1 on pool temperature alarm.

13. Provide the bases for the change to TS G.2.d.2 on Area and Exhaust Gas Monitor Design Features.

14. Provide the bases for the change to TS K.3.b.(2)(a)4 and 5 on Methods for Retest of Confinement.

15. Provide the bases for the change to TS K.3.e.(4)(a) on Maximum Pool Temperature Limitations. Is this change to match the changes to TS Table F.1?

16. With regard to neutron fluence:

    a. Provide the bases for deleting TS K.3.e.(4)(f). Does the TS apply to the change of fuel materials, and is there data and analyses that show that the LEU silicide fuel is not subject to problems due to irradiation?

    b. Provide a TS to limit the fluence (fission density or neutron exposure) of the proposed beryllium components in the Low Enriched Uranium (LEU) core. The limit should be consistent with the November 1991, SAR analyses. If a TS is not judged necessary, provide justification referencing data and analyses.

17. Provide the bases for the change to TS K.3.g.(1) on Waste Disposal and Reactor Monitoring Systems.

18. With regard to the postulated loss of coolant accident (LOCA):

    a. Provide a description of your program to ensure that the reactor pool tank has not degraded and that any potential leakage, including the through beam port penetrations, would be detected and properly dispositioned. Provide justification or Technical Specifications that require this surveillance and evaluation process.

    b. Describe the impact on the LOCA analyses of the vent illustrated on the beam port penetration drawing. That is, when is the vent open? Is there a potential loss of flow through this pathway?

    c. Clarify the potential reactor pool water makeup rates. That is, is normal makeup 20 gallons per minute (GPM)? What additional flow can the manual bypass flow connection provide? What flow can be assured with out power to the campus, five GPM?

    d. In the revised LOCA analysis, no credit is taken for the half-inch hole in the bottom of the grid box, however, plugging this hole could serve to mitigate the postulated LOCA in two ways:

        1. If the drain time is 35 hours, as assumed, then the grid box would serve to keep the core covered beyond this point, adding an extra margin of cooling capacity.

        2. If the drain time were to turn out to be actually much shorter than the 35 hours, then maintaining water in the grid box would serve to extend the time that the core is covered and possible accident consequence could be mitigated.

    Therefore, provide an analysis of the advantages and disadvantages of plugging the half-inch hole in the grid box. Based on

18. d. (continued) this analysis, propose appropriate modifications to the grid box to provide additional assurance that the LOCA consequences will remain acceptable.

e. Provide an estimate of the effect that closing the beam port shutter by operator action (10 minutes after identification of low water level would have on the LOCA analysis. Provide an estimate of the effect if this were an automatic actuation on the reactor scram signal. Provide an analysis of the advantages and disadvantages of such an operator action or modification. Based on this analysis, propose appropriate operator procedures, system modifications, and/or Technical Specifications to further mitigate the consequences of a potential LOCA.

f. For failure of the through tube, the flanges at the ends are assumed to contain the vessel water. Describe the design of these flanges with particular regard to the ability to install the flanges in the face of a leak (if necessary) and to contain the water pressure of the tank. Provide an analysis of the advantages and disadvantages of such an operator action or modification that start after approximately 10 minutes after identification of the LOCA. Based on this analysis, propose appropriate operator procedures, system modifications, and/or Technical Specifications to further mitigate the consequences of a potential LOCA.

g. Provide specific calculations of maximum clad and fuel temperatures assuming prolonged operation at two megawatts and no makeup water. Provide acceptance criteria to determine if fuel temperatures remains acceptable for the LOCA.

19 Provide a description of the start-up test program. Include measurements as specified in enclosure 1 that was provided to the University of Missouri, Rolla, with their LEU conversion order on March 5, 1991.

20. Provide proposed changes for appropriate license conditions for the LEU fuel. Enclosure 2 provides examples that was issued to the University of Missouri, Rolla, with their LEU conversion order on March 5, 1991.

Enclosure 1

OUTLINE OF REACTOR START-UP

REPORT AND COMPARISONS WITH CALCULATIONS

1. Critical Mass

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for both LEU and HEU.

2. Excess (operational) reactivity

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for both LEU and HEU.

3. Control and regulating rod calibrations

   Measurement of differential and total rod worths, and comparisons with calculations for both HEU and LEU.

4. Reactor power calibration

   Methods and measurements that assure operation within the license limit. Comparison between HEU and LEU nuclear instrumentation setpoints, detector positions, and detector output.

5. Shutdown margin

   Measurement with HEU
   Measurement with LEU
   Comparisons between these, and with computations for both.

6. Partial fuel element worths for LEU

   Measurements of the worth of the partial loaded fuel elements.

7. Thermal neutron flux distributions

   Measurements of the core and measured experimental facilities with HEU and LEU, and comparisons with each other calculations.

8. Results of determination of LEU effective delayed neutrons fraction, temperature coefficient, and void coefficient. Comparison with calculations and HEU core measurement.

9. Comparison of the various results, and discussion of the comparison, including an explanation of any significant differences that could affect both normal operation and possible accidents with the reactor.

10. Measurements made during initial loading of the LEU fuel, presenting subcritical multiplication measurements, predictions of multiplication for next fuel additions, and prediction and verification of final criticality conditions.

Enclosure 2

ATTACHMENT TO ORDER

MODIFYING FACILITY OPERATING LICENSE NO. R-79

A. License Conditions Revised and Added by this Order

2.B.2 Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess and use up to a maximum of 5.00 kilograms of contained uranium-235 at various enrichments, up to a maximum of 200 grams of plutonium-239 in the form of sealed plutonium-beryllium neutron sources in connection with operation of the reactor, and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility. Without exceeding the foregoing maximum possession limits, the maximum limits on specific enrichments of U-235 are as follows:

Maximum U-235 (kilograms)	% Enrichment	Form
4.95	< 20%	MTR-type fuel
0.05	Any	Fission chambers and flux foils used in connection with operation of the reactor

2.B.4 Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, a maximum of 4.90 kilograms of contained uranium-235 at greater than 20% enrichment in the form of MTR-type reactor fuel until the existing inventory of MTR-type reactor fuel is removed from the facility.

2.C.2 Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 9 are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications.

B. Technical Specifications Revised by this Order

1.3 Definitions

scram time - the elapsed time between reaching a limiting safety system set point and the time when a control rod is fully inserted.