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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Mendonca:

This is in response to your letter of May 17, 1994 concerning the conversion of the GTRR from high-to-low-enriched fuel. The answers to your questions were provided for the most part by Dr. William Woodruff and Dr. James Matos of Argonne National Laboratory. With your consent, this response was delayed in order to have in hand Argonne's complete response.

Question 1.

Specification 2.1.1, Safety Limits in the Forced Convection Mode

a. Specification 2.1.1.a refers to Figure II-1, for which you have provided a revised version II-1 (new) to replace the existing II-1 (old).

   1. Because the line for HEU will no longer be applicable after the reactor is converted to LEU, the HEU line from Fig. II-1 (new) should be eliminated to avoid confusion.

   2. The remaining line for LEU (flow instability) should represent the acceptable safety limit envelope of the converted Georgia Tech Research Reactor, so it seems appropriate to ink that plot in solid, instead of dashed lines.

   Provide these changes or rationale as to why they are not needed.

   Response: A revised Figure II-1 is attached

b. Specification 2.1.1, Basis, discusses departure from nucleate boiling (DNB) initially and then later discusses DNB and flow instability criteria. While mention of departure from nucleate boiling is acceptable, emphasis should be clearly placed on initiation of flow

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instability as the limiting criteria, since it is limiting. Provide changes that clarify this issue.

Response: The first sentence - "Gross fuel element failure and concomitant fission produce release will not occur unless there is departure from nucleate boiling." Should be replaced with the following - "Gross fuel element failure and concomitant fission product release will not occur until after there is onset of flow instability."

2. Specification 2.2.1, Limiting Safety System Settings in the Forced Convection Mode, Basis, uses the phrase "with no incipient boiling." For internal consistency in your T.S., the same phrases should be used wherever appropriate. If "incipient boiling" is inferred from either "departure from nucleate boiling" or "initiation of flow instability" calculations, use only the term that applies. If they are not interchangeable, please provide a reference as to your analyses that support this different usage or an explanation of the difference in the basis.

Response: Replace the use of "incipient boiling" with "onset of nucleate boiling".

3. Specification 5.2, Fuel Elements; provide changes to this specification to accommodate the conversion to LEU fuel.

The LEU fuel elements shall be of the MTR type consisting of 18 fuel plates of uranium silicide with an enrichment less than 20%. Each fuel plate will have a nominal loading of 12.5 grams of U-235. The HEU fuel elements shall also be of the MTR type consisting of 16 fuel plates of uranium aluminide with an enrichment of 93%. Each fuel plate will have nominal loading of 11.75 grams of U-235.

4. Provide a description of how the PARET code uses boundary conditions for inlet and outlet of the flow channels (e.g., pressure or flow, or can either be specified). Explain what boundary conditions are used for the transient analyses.

Response: PARET can be provided with either an inlet flow rate per unit area or a pressure drop as boundary conditions. Since the flow for the reactor is specified, this value was used for these transient analyses.

5. Describe the PARET heat transfer modeling from channel to channel (e.g., heat transfer from the hot to average channels).

   Response: The PARET code does not model channel to channel heat transfer. Since plate type fuel has closed channels there is no cross flow, and conduction through the side plates would be negligible for most transients. This is a conservative model.

6. Describe how the PARET subcooled boiling model has been benchmarked against any separate effects test, such as the Christensen, Marchaterre, or the Shoukri data. Describe the results of the benchmarking.

   Response: The PARET code has been benchmarked against the "subcooled" SPERT experimental transient data for plate type fuel: See - William L. Woodruff, "A Kinetics and Thermal-hydraulics Capability for the Analysis of Research Reactors," Nucl. Technol., 64, pp. 199-202, 1984 and W. L. Woodruff, "Additional Capabilities and Benchmarking with the SPERT Transients for Heavy Water Applications of the PARET Code," Proc. XIIth International Meeting on Reduced Enrichment for Research and Test Reactors, Berlin, 10-14 Sept 1989, pp. 357-365, Konferenzen des Forschungs-zentrums Julich (1991).

7. Describe how the PARET code flow instability model has been benchmarked against any experiments. Were the comparisons for upflow and downflow? Describe the results of the benchmarking. Describe any nodalization studies that were performed to verify the effects of modeling on flow instability.

   Response: A flow instability model has not been incorporated into the PARET code. Instead, the code estimates the eta parameter as a function of time for use with the Whittle and Forgan correlation for flow instability. See the steady-state analyses in IAEA-TECDOC-233, pp. 99-106 (1980). The estimate of eta was found to be insensitive to the node selection.

8. Describe how PARET models the "dynamic" pressure. Is it based on the average channel flow from the point where the boundary pressure is known using the momentum equation? Have any calculations been performed to assess the local pressure to a thermodynamic "state" pressure for two-phase flow?

Response: The PARET code has a single fluid, incompressible thermal-hydraulics model based on a modified momentum integrated model (channel averaged mass flow and coolant properties based on a reference pressure). Only the coolant density is evaluated as a function of local pressure. Given an inlet pressure, the local pressure is determined based on friction, elevation, and spatial and transient acceleration. See - C. F. Obenchain, "PARET - A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Engineering Laboratory (1969).

9. Describe how PARET models void propagation applied to downflow conditions. Include discussion of the modeling of void propagation when boiling will most likely take place at the exist of the channel and can result in flow reversal.

Response: The GTRR has upflow, and the reactor was modeled with upflow. PARET can model downflow conditions, and it can model flow reversal with loss-of-flow from a forced downflow condition to an upflow condition with natural convection (See R. S. Smith and W. L. Woodruff, "Thermal-hydraulic Aspects of Flow Inversion in a Research Reactor," Proc. 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, 3-6 Nov, 1986, ANL/RERTR/TM-9, CONF-861185, pp. 449-460 (May 1988).

10. Describe the rate of void production when using the Bergles-Rohsenow criteria for subcooled boiling in the PARET code. Is the void propagation model used in subcooled boiling?

Response: The Bergles-Rohsenow correlation is used in PARET as both a trigger for ONB and for part of a transition model to fully developed nucleate boiling. The void production model includes subcooled boiling and distinguishes between the boiling regimes of nucleate boiling, transition boiling and film boiling (independent of the correlation used for subcooled boiling). The voiding model is described in the PARET manual - C. F. Obenchain, "PARET - A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Engineering Laboratory (1969).

11. Described how the PARET code models the heated wall viscosity effects. Include discussion of the treatment of the viscosity decrease near the wall of a heated fuel plate. Describe how

the decrease in friction is modeled. Include the description of the treatment of single phase friction or two-phase conditions.

Response: The PARET code includes the Sieder-Tate correlation option, which has a surface temperature dependent viscosity. The single and two-phase friction treatment is as described in the original PARET manual - C. F. Obenchain, "PARET - A Program for the Analysis of Reactor Transients," IDO-17282, Idaho National Engineering Laboratory (1969).

12. Describe how PARET calculates the average channel flow. Is it equivalent to the imposition of an inlet pressure and an outlet pressure, and iteration for the friction loss and associated new time flow? Describe how the calculational approach precludes any local flow reversal within the channel if the average channel flow is calculated from the imposed pressures at the inlet and outlet.

Response: PARET uses either a fixed input flow or a fixed pressure drop that does not change with time (See responses #4 and #8). The GTRR is upflow, and flow reversal is not a consideration.

13. For other recent LEU conversion analyses (e.g., Rhode Island) the modeling may have been different than used in that of Georgia Tech. The following questions are to better understand the potential modeling differences and effects.

a. It is understood that the Whittle and Forgan flow instability model was recently instituted for use in the PARET code. When was that done? Is it an automatic option in the PARET code? For other recent LEU conversions, was this model used? Discuss the accuracy of the model and comparison to other flow instability models that have been used or are available in PARET.

Response: See response #7. The PLTEMP code (not the PARET code) was used for the steady-state data that was quoted for the Georgia tech reactor. The Whittle and Forgan correlation was included in the early 1980s. The modeling for the Georgia Tech reactor is consistent with the GTRR design. In other reactors the applicable design conditions were also modeled.

b. Describe the PARET modeling for heat transfer to the side plates. Was this function modeled in other recent LEU conversion analyses? Provide a comparison of this modeling and assumptions for the different PARET applications.

Response: PARET has a 1-D heat transfer model (See also response #5). By neglecting transverse heat transfer to the side plates, the model will give conservative estimates for all applications.

c. Describe how the channel tolerances were modeled in the PARET code. Was this function modeled in other recent LEU conversion analyses? Provide a comparison of this modeling and assumptions for the different PARET applications.

Response: Channel uncertainties are modeled only in the PLTEMP analyses for steady-state margins. A peaking factor is applied to the hot channel in PARET as predicted by the neutronics computations. The basis in PARET is to always provide an estimate for the transient behavior of each reactor under nominal conditions with conservative models and consistent with the SPERT experiments (see response #6).

d. Describe the modeling of the bypass flow in the PARET modeling and comparison to other recent LEU conversion analyses. What was the bypass percentage of total flow?

Response: PARET uses the flow that is provided to the active core in proportion to the channel modeled. No bypass flow was modeled in the GTRR case. The reference flow rates as described in the safety documentation were used.

e. Provide a comparison of radial and axial peaking factors used in the PARET code with other recent LEU conversions.

Response: The radial and axial nuclear power peaking factors that were used in the computations for the Georgia Tech heavy water reactor are provided in Figure 8 and 9 in Attachment 2, Table 2-1 of the ANL report "Analyses for Conversion of the Georgia Tech Research Reactor from HEU to LEU Fuel,' J. E. Matos, S. C. Mo, and W. L. Woodruff, September 1992. These factors for the Rhode Island light water reactor are provided in Table 3, p. 14, of the Rhode Island Atomic Energy Commission Report "Safety analysis Report for the Low Enrichment Fuel Conversion of the Rhode Island Nuclear Science Center Research Reactor," November 1991.

14. Describe how the fuel plate heat transfer area is calculated. Is the area based on the width of the plate or the active fuel?

    Response: The heat transfer area is based on the nominal height and width of the active fuel.

Should you have additional questions, please let me know.

Sincerely,

R.A. Karam

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Neely Nuclear Research Center

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Fig. II - 1. GTRR Safety Limit for Forced Convection