May 2, 2006

Dr. William G. Vernetson
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SUBJECT: UNIVERSITY OF FLORIDA—REQUEST FOR ADDITIONAL INFORMATION RE: HIGH ENRICHED TO LOW ENRICHED URANIUM CONVERSION FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR (TAC NO. MC9037)

Dear Dr. Vernetson:

We are continuing our review of your request for high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel conversion for the University of Florida Training Reactor which you submitted on December 2, 2005. During our review of your request, 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. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. Following receipt of the additional information, we will continue our evaluation of your amendment request.

If you have any questions regarding this review, please contact me at (301) 415-1127.

Sincerely,

/RA/
Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Branch
Division of Policy and Rulemaking
Associate Director for Risk Assessment & New Projects
Office of Nuclear Reactor Regulation

Docket No. 50-83

Enclosure:
As stated

cc:
See next page

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REQUEST FOR ADDITIONAL INFORMATION
UNIVERSITY OF FLORIDA TRAINING REACTOR
DOCKET NO. 50-83

1. License conditions and Technical Specification 5.3. Please propose and justify license possession limits for the new low-enriched uranium (LEU) fuel if they are different than your current approved possession limits. Your answer may impact the possession limits given in Technical Specification (TS) 5.3. Will any additional changes be needed to the facility license to allow for conversion from high-enriched uranium (HEU) to LEU fuel?

2. Section 1.5, Comparison with Similar Facilities Already Converted. It is stated that a surface treatment will be applied to the fuel. This is a design feature that was not evaluated in NUREG-1313, "Safety Evaluation Report related to the evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors." Will this surface treatment have any impact on the thermal-hydraulic analysis (e.g., impact fuel thermal-conductivity, hydraulic diameter, create hot spots if spalling occurs, etc.)? The reference cited in the conversion Safety Analysis Report (SAR) discusses the importance of maintaining primary coolant pH in a range of 5.4 to 6.0. Please propose and justify a TS to maintain pH within a proposed range or discuss why limits on primary coolant pH are not needed. The reference also discusses not letting primary coolant stagnate in the core and not draining the core if water or air quality is poor. Discuss any steps you take in these areas.

3. Section 4.5, Dynamic Design. Are any changes in fuel element location planned from the LEU reference core during the life of the core and if so, how does that affect the results of the safety analysis?

4. Section 4.5, Dynamic Design. The LEU core will age differently than the HEU core because of the addition of uranium-238 and the production of plutonium. Are the various calculated results given for the LEU core (e.g., kinetics parameters, reactivity coefficients, etc.) and the accident analysis results based on the most conservative values over the 87 MWD life of the fuel? If not, please provide these results or explain why changes in parameters over core life is not significant.

5. Section 4.5.1, Calculation Model. In order to understand the engineering uncertainty factor assumed for the calculated power density (Table B-1) it is important to understand the validity of the MCNP5 model. Can the data in Table 4-9 and estimates of the uncertainies in measurements be used to determine an uncertainty factor with greater confidence than the one given in Table B-1?

6. Section 4.7.1 Fuel Assembly and Fuel Box Geometry. A form-loss coefficient K is used to represent local losses at the inlet, outlet and the grid plate. There are at least two types of coolant flow paths in the fuel box, gaps between the fuel plates, and the region between the fuel bundles (vertical bypass) plus the region between the fuel bundles and the fuel box (channel against fuel box). Explain the selection of the value(s) of K for the flow paths between the plates and outside the plates and justify that conservative value(s) were used in the analyses. Since the coolant channels between the fuel plates are not sealed by a channel box there will be mixing of coolant between the two types of coolant flow paths. Explain the impact of ignoring the diversion of heated coolant to the unheated flow channel (e.g. the vertical bypass region between fuel bundles).

7. 4.7.2 PLTEMP/ANL v2.14 Code Description. The reference (Reference 11) cited for the code PLTEMP/ANL v2.14 is for an earlier version that does not model laminar forced convection. Provide an updated reference that documents the implementation of the new models and the validation of the new models for application to plate type geometries.

8. 4.7.2 PLTEMP/ANL v2.14 Code Description. The placement of Table 4-20 appears to be out of place. There is no mention of the table in the text. Also it is not clear if the table is based on Table B-1 in Appendix B because values of hot channel factors from the two tables do not match. This table misplacement is probably also the reason that tables after Table 22 are misnumbered.

9. 4.7.2 PLTEMP/ANL v2.14 Code Description. The thermal-hydraulic analyses did not include systematic uncertainties, such as measurements of reactor power and coolant flow rate, in the calculations. According to the SAR, “These systematic uncertainties will be included in the interpretation of the results.” This does not seem to be the case so how was this done?

10. 4.7.2 PLTEMP/ANL v2.14 Code Description. The thermal-hydraulic analysis assumed that power generation is uniform along the width of a fuel plate. Explain how the analysis accounts for the peaking of power density along the width of a fuel plate.

11. 4.7.2 PLTEMP/ANL v2.14 Code Description. PLTEMP/ANL v2.14 requires the specification of where the hot channel factors are to be applied. Explain the process and the calculations that were used to determine the limiting safety case with hot channel factors included.

12. Appendix B. One of the random uncertainties not explicitly represented in Appendix B, Table B-1 but is noted in Reference 25 is the fuel density from end-to-end along the axial length of the plate. Explain the significance/insignificance of this uncertainty for the LEU plate to be used in UFTR.

13. Section 4.7.3 Thermal-Hydraulic Analysis Results. The thermal-hydraulic parameters shown in Table 4-1 in the SAR do not agree with a similar tabulation in Table 4-22. Explain the differences between the two tables and why the maximum fuel temperature is higher for the LEU core in one table and lower in the other.

14. Section 12.4, Emergency Plan. For any emergency plan changes you want to make under the reactor conversion please submit the specific changes and their justification along with replacement Emergency Plan pages.

15. Section 13.0 Accident Analysis. The explanation of Table 13-1 says that coefficients with the smallest magnitude were used in the analysis in order to be conservative. Explain why the fuel temperature coefficient used is not the smallest value for the LEU core according to Table 4-18.

16. Section 13.1.3 Sudden Insertion of the Maximum Allowed Excess Reactivity. In addition to peak clad temperature, the safety of the LEU core in a sudden insertion of the maximum allowed excess reactivity accident hinges on the total energy released in the transient that

is scaled from the SPERT-I tests. What is the calculated value for the accident with a step insertion of reactivity?

17. Section 13.1.3 Sudden Insertion of the Maximum Allowed Excess Reactivity. In the analysis of energy deposition the reactivity feedback coefficient is based on an empirical correlation from the SPERT-I test results. How does this value differ from what would be calculated based on the UFTR core design and using MCNP5?

18. Section 13.3, Fuel Handling Accident. It is stated that in all cases, the reactor would be shutdown from power operation for at least three days before fuel handling operations would occur. Propose a technical specification that captures this condition or discuss why this limit is not needed.

19. Section 13.3, Fuel Handling Accident and Section 13.4, Maximum Hypothetical Accident. The locations chosen for evaluations of doses from these accidents is not consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," Section 13.2. The NRC staff does not support the use of site and urban boundaries as discussed in ANSI/ANS -15.7 (note that this is not an active standard). Please provide dose results and bases for the following locations: (1) occupational doses for staff inside the reactor cell from the initiation of the event until they evacuate the reactor cell, (2) non-occupational dose to the person with maximum exposure (this could be someone at the outside of the operations boundary but may be at a different location depending on the release point) and (3) non-occupational dose at the nearest residence (if it is closer than East Hall Housing).

20. Section 13.3, Fuel Handling Accident. An assumption of a reactor cell leak rate of 0.2% volume/hour is used in this analysis. What is the basis for this leakage assumption? This type of assumption normally becomes a technical specification limiting condition of operation with a surveillance requirement to periodically (annually has been accepted by NRC) confirm the leak rate. Please discuss.

21. Section 13.4, Maximum Hypothetical Accident. It is not clear if there is an assumed 3-day period of reactor shutdown before shielding block handling would occur (similar to the fuel handling restriction in the fuel handling accident analysis). If so, please propose a technical specification that captures this condition or discuss why this limit is not needed.

22. Section 13.4.1 Radionuclide Inventories. In Table 13-11, the third and fifth columns are meant to be 2.7% of the previous column. However, the results for the LEU core are not correct, except for the I-135m case, even though the results for the dose calculation are correct. It appears that the HEU 2.7% column was copied into the LEU 2.7% column and never changed to reflect the 2.7% of the LEU inventory.

23. Appendix A.1, Determination of Material Composition, last paragraph. There appears to be an equation missing from the text where an error message appears. Please correct.

24. Table A.3-1. There appear to be formatting errors in the first footnote on this table. Please correct.

25. TS 2.1, Safety Limits. The regulations in 10 CFR 50.36 define safety limits (SLs) as limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. Exceeding a SL means that there is a possibility that the physical barrier has failed. That is the basis for requiring Commission approval for reactor restart if a SL is exceeded. As pointed out in the SAR, fuel cladding is the physical barrier of concern. It is stated that the integrity of the barrier is protected if the cladding temperature is kept below the incipient melting temperature of 582EC for the 6061-aluminum cladding of the LEU silicide fuel (however, NRC has accepted an upper temperature for aluminum-clad fuel of 530EC based on blister formation temperature - see NUREG-1313, "Safety Evaluation Report related to the evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors," Section 3.3.3). You have based your SLs on the prevention of localized boiling. This results in SL process variable values significantly below those needed to protect against clad blister.

    As a result, in some cases, the accident analysis does not appear to support the technical basis for the SLs. For example, for the analysis of a step addition of reactivity of 0.6% $\Delta k/k$ into the core, the peak power for the LEU core with a SCRAM occurring is 320 kW. This would violate the SL in the proposed Technical Specification 2.1, namely, that "the power level shall not exceed 190 kW." Also, this result would indicate that the limiting safety system setting (LSSS) of 125 kW would not be in compliance with the regulations in 10 CFR 50.36(c)(1)(ii)(A) which states when a LSSS is specified for a variable on which a SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded.

    Please propose revised SLs and LSSSs, as needed, that support the conclusions in your accident analyses for reactivity insertions and loss-of-coolant or explain why the proposed values in your SAR are acceptable.

26. TS 5.3. In your conversion SAR only aspects of using U_3Si_2-Al dispersion fuel were discussed. In the proposed change to TS 5.3 it is stated: "The fuel matrix may be fabricated by alloying high purity aluminum-uranium alloy or the fuel matrix may be fabricated from uranium silicide-aluminum (U_3Si_2-Al) using the powder metallurgy process." Please explain why high purity aluminum-uranium alloy fuel remains in the technical specifications. If the high purity aluminum-uranium alloy were to be used, what effect would it have on all the safety analysis covered in the submittal? Is it possible that there could also be cores with a mix of the two types of fuel?

27. TS 5.5.1. This TS has integral worth values for the control rods. Does this information need to be updated based on calculated worths for control rods in the LEU core?

28. TS 6.6.3(2). Should significant changes in the transient or accident analysis for the HEU to LEU conversion also be reported to NRC within 30 days of occurrence?