September 29, 1998

Stephen M. Sohinki, Director
Office of Commercial Light
Water Reactor Production
Defense Programs
U.S. Department of Energy
Washington, DC 20585

SUBJECT: NRC STAFF'S REQUESTS FOR ADDITIONAL INFORMATION REGARDING DOE TOPICAL REPORT ON TRITIUM PRODUCTION CORE (TAC NO. MA0118)

Dear Mr. Sohinki:

The NRC staff is reviewing your topical report entitled "Tritium Production Core (TPC) Topical Report" submitted by letter dated July 30, 1998, and concludes that additional information is needed before it can complete its review. You are requested to provide a response to the enclosed requests for additional information (RAIs) in the form of a letter or as revisions to your TPC topical report by December 15, 1998.

You should note that these RAIs are the result of the staff's review to date in all technical areas, except for those under review by the Plant Systems Branch. As the review proceeds, additional questions, if needed, will be transmitted to you promptly.

If you have any questions regarding this request, or if you wish to schedule a meeting with the staff to discuss the RAIs and your proposed response, please contact the project manager, J. H. Wilson, at (301) 415-1108.

Sincerely,

Ralph Architzel for/

Thomas H. Essig, Acting Chief
Generic Issues and Environmental
Projects Branch
Division of Reactor Program Management

Enclosure: As stated

Project No. 697

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Thomas H. Essig, Acting Chief
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OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ADDITIONAL INFORMATION REGARDING
DOE'S "TRITIUM PRODUCTION CORE (TPC) TOPICAL REPORT
PROJECT NO. 691

A. Containment Systems and Severe Accident Branch

Section 2.6.2 of the TPC topical report states that the tritium-producing burnable absorber rods (TPBARs) will have a negligible impact on the core stored energy and that there is sufficient conservatism in the current methodology to cover the relatively small amount of heat generated by the TPBARs. It is also noted elsewhere that the TPC decay heat (see for example Section 2.5.4) is lower than the reference core and that the primary coolant inventory is less than the reference core. These two factors would reduce the mass and energy releases to containment from a LOCA when evaluating the containment response. However, their effects have not been quantified. These two factors have the opposite impact when evaluating the containment minimum pressure, as discussed in Section 2.6.4 (although only the mass inventory is addressed).

Under current regulations, operating reactors may have a calculated peak containment pressure at ("less than") or near the design pressure. The staff's acceptance of the licensee's analysis includes the use of an acceptable computer program and the factors which have been included in the analysis "to maximize the calculated peak containment pressure and temperature" (SRP 6.2.1.1.A, Section III. REVIEW PROCEDURES).

1. Quantify the "relatively small" heat generation of the TPBARs and the changes in the integrated mass and the integrated energy into containment for a LOCA and their estimated impact on the calculated containment pressure. This information is needed to make the determination that the assumptions result in a conservative containment response that maximizes the calculated peak containment pressure and temperature. It will also provide a means to assess the need to remove some of the added conservatism resulting from the use of the TPC to offset non-conservative design basis accident (DBA) attributes.

2. Section 2.6.3.2 refers to a Westinghouse topical report for steamline and feedwater line break mass and energy and specifically the Westinghouse LOFTRAN computer program. This section should also discuss SRP 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," as the generic guidance for secondary side mass and energy releases. It is stated that the mass and energy releases are sensitive to the core reactivity coefficients and that, in general, these values are typically bounded in the analyses. Further, the potential effect of the TPC on the calculated mass and energy releases is based on the Westinghouse LOFTRAN computer program, and therefore likely only for Westinghouse-type steam generators. Discuss these issues for non-Westinghouse plants. Can it be confirmed that the Westinghouse methods and methodologies are consistent with other NSSS PWR vendor methods and methodologies? Would other designs require reanalyses for their current licensing basis based on different sensitivity to these TPC parameters?

3. Address the impact of the TPC design differences from the reference core and their effect on the minimum containment pressure calculation for DBA analyses. These differences would appear to result in a non-conservative evaluation of the containment response, based on SRP 6.2.1.5 guidance, and on the ECCS performance (the TPC core releases

would be lower resulting in a lower calculated pressure). The current discussion only includes a reference to the primary system inventory. Include in the discussion the core stored energy and the decay heat release, and their impact on the calculation.

4. Address the issues identified in question 3 for the subcompartment loading analysis, based on SRP 6.2.1.2 guidance. This information is needed to make the determination that the distribution of the mass and energy released into the break compartment results in a conservative assessment of the subcompartment loadings. It will also provide a means to assess the need to remove some of the added conservatism resulting from the use of the TPC to offset non-conservative design basis accident (DBA) attributes.

Section 6.2.5 of the TPC topical report provides an assessment of the TPC on the combustible gas control system. The discussions center on the additional combustible gas generated by the TPC as being "small" but does not address the expected margins to allowable concentrations, for example Regulatory Guide 1.7.

5. SRP 6.2.5 (Appendix A) guidance indicates that one pound of reacted zirconium generates 0.021978 pound-moles of free hydrogen, or 8.4866 standard cubic feet (scf) of hydrogen. Reconcile the difference between this value and the 7.9 scf value referenced in NDP-98-181.

6. Address the margin to the allowable hydrogen (four volume percent) concentration following a LOCA. While the expected amount of additional hydrogen is small in comparison to the total allowable, under what circumstances (for example containment free volume or core-clad generated hydrogen) would the margin become unacceptable? Provide numerical values for each source of combustible gas and a range of containment free volumes to demonstrate that margin is available and that the hydrogen recombiners remain adequate.

7. The TPC topical report does not discuss the 75% metal-water reaction aspect of the hydrogen control system (hydrogen ignitors). Provide a discussion of the impact, if any, of the TPC on the hydrogen ignitor system.

B. Emergency Preparedness and Radiation Protection Branch

1. Section 2.11.2 describes the source terms used in the analysis of the radioactive waste management systems. Detail how the reactor coolant activity (Table 2.11.2-2) was calculated. Why is attention limited to the radioiodines and the noble gases in the primary coolant, when the primary issue here is tritium and other nuclides (e.g. the cesiums) likely to affect the whole body doses via liquid effluents? The original PWR GALE code is cited on Page 2-231. Why was the updated version, NUREG-0017 Rev. 1, 1985 not used and what difference would it have made if the newer version had been used?

   The key element here is the assumed failure of two rods; how is this small number justified?

2. Section 2.11.3 addresses the liquid waste management systems. The distribution of tritium in liquid and in gaseous effluents is noted (page 2-237), citing NUREG-0017, Rev. 1. However, practices at nuclear power plants have changed so that waste evaporators are used much less that they were earlier. How is this change reflected in the analysis?

The discussion of discharge concentration limits (Page 2-240) does not seem to reflect the fact that this is an instantaneous limit on concentration. The average concentration in the discharge must be much lower. This should be clarified.

Considerable detail about the reference plant will be needed to complete the Appendix I dose assessment. For example, do the two units share a single discharge canal? Is extensive dilution in the receiving stream assured before any individual can drink the discharged water? Presumably this is the case or the concentrations in Table 2.11.3-2 would be too high.

3. Section 2.11.6 of the topical report discusses the process and effluent radiological monitoring and sampling systems evaluation. Because of the potential importance and variability of the tritium concentrations in effluents, the assumptions underlying the current radioactive liquid waste sampling and analysis program seem no longer valid. A suitable sampling and analysis program should be developed and justified.

4. Section 2.12 of the topical report addresses radiation protection issues. The substantial increase in tritium concentrations and the potential for substantial variation in tritium levels suggest that added tritium monitoring provisions are in order. Describe the tritium monitoring capabilities for both airborne and surface contamination, and show why they are adequate. What provisions are, or will be, in place to ensure that workers do not leave the plant with significant levels of tritium contamination?

5. Section 2.15.6 of the topical report presents the radiological consequences of accidents. Normally, the design basis accident doses are calculated separately, so a change in the whole body dose does not result in a change in the thyroid dose. However, in this special case where the critical consideration is the dose from tritium, it seems that the internal dose from tritium should be considered a contributor to both the whole body dose and the thyroid dose. Show how this changes Table 2.15.6-2.

6. Justify the assumption that only 3 percent of the tritium will be available for leakage from containment in the event of a LOCA.

C. Electrical Engineering Branch

Both electrical systems and environmental qualification issues are within the Electrical Engineering Branch scope of review. The staff has no questions resulting from the review by the Electrical Engineering Branch at this time.

D. Instrumentation and Controls Branch

The staff has no questions resulting from the review by the Instrumentation and Controls Branch at this time.

E. Materials and Chemical Engineering Branch

1. Appendix G of 10 CFR Part 50 requires reactor vessel beltline materials to maintain Charpy upper-shelf energy (USE) throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture

equivalent to those of Section XI of the ASME Code. Section 2.5.3.2 of the topical report does not contain criteria and methodology for evaluating the integrity of reactor pressure vessels with projected end-of-license USE of less than 50 ft lb. Describe the criteria and methodology to be used if the USE is projected to be less than 50 ft lb.

F. Mechanical Engineering Branch

1. Section 2.3.3.1 of the topical report states that the incorporation of TPBARs affects LOCA forces because they slightly increase reactor vessel hydraulic resistance and thus reduce the best estimate primary loop flow rate. This results in a decrease in the cold leg temperature which in turn increases the limiting hydraulic forces associated with the cold leg break. Provide a more detailed discussion of the calculation of the hydraulic forces for the reference core associated with the postulated break which includes the type and size of break and assumed break opening time.

2. Section 2.4.2.1.4 of the topical report states that since the TPBAR assembly is a hanging structure supported by the top nozzle adapter plate of the fuel assembly and the rodlets are hanging in the guide thimble tubes, the added weight can be considered to be part of the fuel assembly nozzle support. It is further stated, that the added TPBAR assembly weight, together with the rodlet stiffness, has an insignificant effect on the fuel assembly's dynamic characteristics. Provide additional structural details and analysis or test data to validate this assumption.

G. Plant Systems Branch

The staff will issue any questions resulting from the review by the Plant Systems Branch in October 1998.

H. Quality Assurance and Maintenance Branch

1. Reg Guide 1.30 should be included in the series of regulatory guidance documents for SRP 17.1 (Page 2-375).

2. The topical report limits the scope of the review to the period when fuel assemblies (containing unirradiated TPBARs) arrive at the reactor site until the assemblies are placed in the fuel pool after one cycle.

   Confirm that manufacture and procurement processes are outside the scope of this review (with reference to deferral of NRC review/inspection of supplier/vendor QA programs).

3. The Pacific Northwest National Laboratory developed the design and fabrication techniques for the prototype TPBAR lead test assemblies.

   What processes, verification checks, and interface agreements will be implemented to ensure the accuracy and completeness of the data and technology transferred from PNNL to the production-scale supplier?

4. Describe the process to be used by suppliers/vendors for evaluating and reporting Part 21 defects and noncompliances. Describe the involvement of the DOE Part 21 program in the evaluation/reporting process.

5. Because DOE has not awarded the contract for manufacture of TPBARs and the scope of the topical report is limited to evaluation of TPBARs on the safety and operability of the reference reactor, the purpose of including a discussion of the Westinghouse QA program in Section 2.17.2 is unclear.

   Will these QA programmatic requirements be imposed if DOE awards the contract to a TPBAR supplier other than Westinghouse?

I. Reactor Systems Branch

1. The last sentence of the first paragraph of Section 2.4.4.2 (Page 2-119) states that the outside geometry of the TPBAR was selected so that it was within the envelope of existing burnable absorber component rods. What does "within the envelope" mean? Provide clarification substantiated by comparison to the reference core, if applicable.

2. The first sentence of the second paragraph of Section 2.4.4.2 (Page 2-119) states that "performance differences were seen and accommodated". Provide tabulated performance differences of what is meant by "accommodated."

3. The last sentence of the third paragraph of Section 2.4.4.2 (Page 2-119) lists a number of parameters for the Tritium core. How do these parameters compare to the reference core?

4. In the last paragraph on Page 2-120, the subject of acceptance criteria is brought up. Provide clarification of what is meant by "acceptance criteria are generally met."

5. The last sentence In the second paragraph on Page 2-121 refers to "significant" power distribution changes. Please clarify the definition of "significant."

6. The fourth paragraph on Page 2-121 (Section 2.4.4.6) states that the "Bases" will continue to be met. Which bases are being referred to here?

7. The first paragraph on Page 3-50 refers to the Westinghouse Thermal-hydraulic Design Procedures. This presumes that the referenced procedure is an approved one. If so, please provide the approved topical report (WCAP-XXXX-A) number and verification that the conditions imposed on the approval of this WCAP have been adhered to.

8. The first paragraph on Page 3-51 refers to "generic analysis."

   (a) Please clarify what is meant by this term.

   (b) Are the criteria given in Table 3.6-1 the same as those applied to the reference core?

9. The third paragraph of Page 3-51 refers to a conservative, but more realistic, coolant temperature that is calculated with $F_{\Delta H} = 1.46$ to determine existing margin to bulk boiling. Why is this value conservative and more realistic?

10. The first paragraph of Page 3-53 refers to the thermal margin of the TPBAR in the context of the thermal design limits. Please provide table(s) or graphs showing comparisons between the Tritium loaded core to that of the Reference core regarding thermal margin.

11. The third paragraph of Page 2-369 refers to an ATWS transient calculation that is based upon "Unfavorable Exposure Time (UET)." The UET methodology was rejected by the staff after extensive review (see letter to Mr. Vance D. Vanderburg, Chairman, Westinghouse Owners Group analysis Subcommittee, "Review of WCAP-11992," July 1, 1997). Therefore, the Method of Analysis, as presented on Page 2-369 of Section 2.15.7 of the topical report, is unacceptable. Please provide an ATWS analysis based on the 1979 ATWS rule.

12. In addition to the LOCA, are any other accidents mitigated by the inclusion of TPBARs? If so, which ones?

13. Regarding Section 2.15.2.2, provide lists of bounding values assumed for the key safety analyses parameters for each of the transients evaluated in Sections 2.15.2 and 2.15.4. Discuss the changes of these bounding values from the TPBAR core. This information will be used by the staff in comparing the assessment of the effect from the TPBAR core to the current FSAR of the reference plant.

14. Regarding Section 2.15.2.5, provide discussions on why an increase of feedwater flow is the only transient among its category of events which requires actual analysis to assess the acceptability of the TPBAR core.

15. Section 2.15.2.6 states that the slightly increased maximum fuel temperatures considered for the TPBAR core do not significantly affect the results for the transients considered in Section 15.2 (Decrease in Heat Removal by the Secondary System) of the reference plant FSAR. Provide quantified information to support the conclusion that the reference plant FSAR will indeed bound the TPBAR core design.

16. Section 2.15.2.7.3 of the topical report states that the current technical specification allows the pressurizer power operated relief valves (PORVs) to be isolated during power operation. Therefore, the PORVs should not be considered available for accident mitigation and the analysis of the locked rotor event should be modified accordingly.

17. Section 2.15.4 states that the reactor core characteristics have only a minor impact on the SGTR analysis. Provide quantified information to support the conclusion that the SGTR analysis in the reference plant will indeed bound the TPBAR core design.

18. What is the effect of He-3 buildup after a 6-month shutdown on power peaking in the TPBAR and non-TPBAR assemblies?

Project No. 697
DOE Tritium Program

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