March 5, 2008

Mr. Keith Paulson
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON US-APWR TOPICAL REPORT MUAP-07009-P, "THERMAL DESIGN METHODOLOGY"

Dear Mr. Paulson;

By letter dated July 13, 2007, Mitsubishi Nuclear Energy Systems, Inc. (MNES) submitted for NRC staff review Topical Report MUAP-07009-P, "Thermal Design Methodology." The NRC staff has reviewed this topical report and has determined that the following information is needed for the NRC staff to complete its review.

Mitsubishi Nuclear Energy Systems, Inc. is requested to provide a response to the NRC staff's request for additional information within 30 days of the date of this letter.

Sincerely,

/RA/

Stephen Raul Monarque
USAPWR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-021

Enclosure: As stated

cc: See next page

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OFFICIAL RECORD

TOPICAL REPORT MUAP-07009, "THERMAL DESIGN METHODOLOGY"

REQUEST FOR ADDITIONAL INFORMATION

General Comments

G-1. On page 3-2, MHI references the MARVEL code. Reference 17 appears to be a non-loss-of-coolant-accident (LOCA) methodology topical report. Provide a description of all changes made to the MARVEL code from the previously approved version.

G-2. The statement in section 6.4 on page 6-1, needs clarification on the term "conservative results," as compared to what, or how to select the value to be conservative.

G-3. The reference number for item (1) on page 8-1 should be 11 not 12.

G-4. In Appendix A, the number of axial nodes for cases 3 and 4 (Table A.1-2) does not agree with either the nodalization diagram (Figure A.1-4), or the result keys (Figures A1.5 thru 7). Please clarify this discrepancy.

G-5. In Appendix B, the database for the WRB-2 DNB correlation does not appear to support the lower grid spacing value (10 inches, database 22 inches). Please explain the effect of lower grid spacing on the departure from nucleate boiling (DNB) correlation.

G-6. Please confirm that the sensitivity studies directly relate to the MHI fuel to be used in the US-APWR.

Application of VIPRE-01M to Peak Cladding Temperature (PCT) Transients

MHI intends to replace FACTRAN with VIPRE-01M for the performance of Chapter 15 (condition III or IV) non-LOCA events transients, which are PCT limited instead of DNB limited. Options for forced film boiling at the hot spot and the Zr-H_2O reaction heat generation within the clad were added to the code.

Sensitivity studies reported only address DNB results.

S-1. Please provide sensitivity studies affecting PCT results, such as nodalization, time step, properties, etc.

S-2. Section 4.1(2) Axial Nodalization, states that the number of axial meshes does not have significant impact on DNB ratio results; however, an increase in elevational discrepancy between cell center and grid location may have an adverse effect. Please discuss the magnitude of the effect and the user guidance/control established to offset this effect.

Thermal properties in Section 6.2

TP-1. It is stated in this section that a degradation effect, due to burnup, is applied to the thermal conductivity. Please describe the burnup degradation model and provide results of analysis of the effect at various burnups up to the limiting fuel burnup.

The metal-water reaction model described in Section 6.6.

Limitations in VIPRE-01M's basic capabilities limit the potential for enhancements. For example, constant material density since the code does not account for changes, and the simplified Zr-H_2O layer treatment.

MW-1. Other than the comparison to FACTRAN, what assessment of the model was performed?

MW-2. Can fuel pin failure occur resulting in oxidation on the interior surface of the cladding?

MW-3. Provide a discussion of any differences between the FACTRAN models and the models implemented in VIPRE-O1M. Address the treatment of the oxide layer as a composite material.

<u>System Code (MARVEL) Capability for Asymmetrical Flows</u>

M-1. Can the VIPRE-01M core flow driver code (MARVEL) adequately treat asymmetrical core flow? For example, a locked pump rotor, with loss of flow in 1-of-4 reactor coolant loops resulting in some portion of the core receiving less than the total pumped flow. If the total flow is 400 units, then each core quadrant obtains 100 units. The loss of flow in one reactor coolant loop means a total of 300 units, but the remaining 3-of-4 reactor coolant loops would likely see more than the loop without the pump (not necessarily 75 units each, but, perhaps, 90 units in three and 30 units in the other). How does this impact the applicability of the 1/8-core model, Figure 4.1, page 4-7?

<u>DNB Correlations</u>

DNB-1. General**:** There are no 14 foot data referenced in the Topical Report MUAP-07009-P. All of the Columbia test data are for 12 foot Westinghouse/MHI fuel. MHI is assuming that the grid spacer studies conducted at Columbia for the 12 foot fuel are applicable to the MHI 14 foot fuel. Provide quantitative and qualitative technical justification for this assumption.

DNB-2. Based on figures in Appendix C, it appears that the grid spacing in the MHI fuel design is less than the minimum used in the test programs (database). In addition, a grid spacer is located 0.7 inches above the start of the heated length of the core. Two grid spacers occur before the 13 inch or 10 inch lower limit of the database.

It is not clear if the database referenced is complete or just a part of the database used to develop the actual DNB correlations. MHI omitted some data, see Appendix B. The database is used to establish a 95/95 DNB limit and then show the MHI fuel database supports the 1.17 minimum DNB value.

a. What is the expected impact of the grid spacers near the core entrance? Would they improve or degrade the grid spacer enhancements measured higher in the bundle? If the spacers streamline the flow then the turbulence enhancement seen higher up may actually be less - the DNB location needs to be considered. What does the “goodness” of the predicted-to-measured ratio mean?

b. Westinghouse, for example, typically does not credit improvements below the first grid spacer, and instead uses a different DNB correlation (no enhancements) for that region. It would seem MHI uses the enhanced DNB correlations over the full height, perhaps conceptually correct but not obvious from the database. Provide justification that this approach is appropriate.

DNB-3. On page 5-1 of Topical Report MUAP-07009-P, the first paragraph of section 5.1 alludes to the use of the W-3 correlation in regions outside the ranges of the WRB-1/2 correlations. No detail was provided as to where exactly the W-3 correlations apply. The applicable ranges of the W-3 correlation needs to be stated and demonstrated to be valid as per the NRC approved SER for this correlation.

DNB-4. On page 7-4 of Topical Report MUAP-07009-P, it is not clear from Tables 7-1 and 7-2 whether the stated thermal hydraulic parameters cover the expected ranges of operation (steady state and transient domains), for the US-APWR. Please relate the thermal hydraulic parameters to the expected operational ranges.

DNB-5. On page 7-5 of Topical Report MUAP-07009-P, Figures 7-1 thru 7-16 provide a variety of void model behavior as a function of chosen thermal hydraulic parameters. Of particular interest to the NRC staff is the void fraction behavior as a function of axial height. It appears that approximately 2/3 up the core, void effects come into play. Is this void behavior captured in the neutronic models as well as the thermal hydraulic models?

DNB-6. Regarding these same figures, it appears that the operational domain of the US-APWR could entail void fractions on the order of 80 to 85 percent. How do these high void fractions compare to current pressurized water reactors (PWRs), and how are these high voids accounted for in the thermal and neutronic models?

DNB-7. In Appendix B, page B-1 of Topical Report MUAP-07009-P, the first paragraph alludes to the use of the WRB-1 and WRB-2 DNB correlations in the sub-channel code VIPRE-01M to determine the DNB value for the US-APWR. It is not clear to the staff from this paragraph whether the data bases for the WRB-1 and 2 cover the intended operational range of the US-APWR.

DNB-8. Appendix C - page C-1, provides a description of the z1 and z2 spacer tests conducted at Columbia University. It appears that these tests were conducted on 12 foot fuel and only with a cosine power profile. No justification was provided in this appendix as to why 12 foot test data is applicable to 14 foot fuel. Please provide qualitative and quantitative technical justification in support of the intended application.

DNB-9. In neither Appendix B nor C are the operation ranges, pressure, mass flow, quality, temperatures, etc., stated. Please provide table(s) showing the full Operational ranges of the US-APWR compared to the current PWRs, (12 and 14 foot cores). Explain any differences.

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