REQUEST FOR ADDITIONAL INFORMATION 377-2629 REVISION 1

5/29/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 04.04 - Thermal and Hydraulic Design
Application Section: 4.4.2.5

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

04.04-1

In MUAP-07009-P, the Thermal Design Topical Report, Section 4.6, there is a statement that the inlet mass velocity of the hot assembly could be approximately 5 to 10% lower than the core inlet mass velocity average. In MUAP-07022-P, the 1/7 scale flow testing model, in Section 1.3.1 item b., there is a statement that DNBR(design margin) will not be evaluated for flow rates 0 to 20% below the core average value. Explain the apparent difference in these two reports. If an allowed 20% lower flow rate is MHI's criteria for evaluation, please provide quantitative results, for a range of power distributions, that a 20% decrease in assembly inlet flow rate has a negligible effect on minimum DBNR. If not, what inlet flow rate maldistribution is assumed in the DNBR analysis?

04.04-2

In Design Control Document, Section 4.4.2.2.4, 'Effects of Rod Bow on DNBR', it is stated that "the maximum DNBR rod bow penalty for the US-APWR core is less than 1 percent."

Provide the basis for the 1 percent value. Include reference to relevant test data for 14-foot US-APWR fuel.

04.04-3

Design Control Document, Section 4.4.2.6, 'Core Pressure Drops and Hydraulic Loads', does not contain sufficient description of the hydraulic loads calculation method or results for staff evaluation. Provide a discussion of the calculation method, assumptions, and results. A reference to the structural evaluation of the fuel and internal vessel components should be included.

04.04-4

Provide a tabulation of all numerical uncertainty values considered in the statistical evaluation of DNBR for the US-APWR fuel. These uncertainties are discussed qualitatively in Design Control Document, Section 4.4.2.9.1 for DNB analyses.

04.04-5

Provide a tabulation of all numerical uncertainty values considered in the statistical evaluation of fuel and cladding temperatures for the US-APWR fuel. These uncertainties are discussed qualitatively in Design Control Document, Section 4.4.2.9.2 for fuel and cladding temperatures.

04.04-6

Provide a tabulation of all numerical uncertainty values considered in the statistical evaluation of core hydraulics for the US-APWR core design. These uncertainties are discussed qualitatively in Design Control Document, Section 4.4.2.9.3 for core hydraulics.