REQUEST FOR ADDITIONAL INFORMATION 719-5352 REVISION 0

3/17/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
Application Section: 15.6.5

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

15.06.05-87

If fluid mixing between coolant in the US-APWR lower plenum and in adjacent reactor core regions can take place, assess the effects of possible localized coolant temperature variations in the lower plenum and core inlet areas in the US-APWR boric acid precipitation analysis. Due to the strong dependence of the boric acid solubility limit on the solution temperature, precipitation can first be triggered by such local coolant temperature distributions in areas where colder coolant can reside.
Provide a calculation for the boric acid solubility limit at a solution temperature that conservatively bounds expected coolant temperature variations in the reactor vessel lower plenum during post-LOCA long term cooling. Provide a plot showing the determined precipitation limit as a function of time after the LOCA initiation. Provide relevant data and/or equations used to compute the result as well as those used to compute any other boric acid precipitation limits applied in the US-APWR precipitation analysis. List all assumptions made in calculating the precipitation limits and discuss the impact of each individual assumption on the limiting concentrations obtained. If a parameter that changes in time is represented by a single value, explain how this value was computed and the point in time or time period for which it is representative of. Also, if a volume average quantity is used to represent the conditions in a certain region modeled by a control volume, explain how the spatial distribution effects associated with this parameter have been accounted for in obtaining the volume average value. In considering possible effects related to time and space variations, show that the results applied led to conservative predictions.

15.06.05-88

In the US-APWR design, a switchover from direct vessel ECCS injection mode to a simultaneous injection mode involving direct vessel and hot leg ECCS injection is used to prevent boric acid precipitation and to ensure core cooling following a LOCA. During the simultaneous injection mode, the steam flow through the reactor hot legs can cause liquid entrainment and thus impede delivery of ECCS flow into the upper plenum. In addition, liquid holdup in the hot leg horizontal and inclined sections as well as in the connected steam generator regions can increase the loop resistance. In turn, this will cause a corresponding increase of the upper plenum pressure thus limiting the growth of the control mixing volume. US-APWR FSAR Section 15.6.5.3.3.3 "Post-LOCA Long Term Cooling Evaluation Results" only refers to entrainment threshold calculations as an evaluation basis for concluding that sufficient reactor core cooling is provided following the switchover to simultaneous ECCS injection after a LOCA.

Describe the entrainment model and provide the results from entrainment calculations performed for the US-APWR to demonstrate that hot leg injection is capable of preventing effectively boric acid precipitation for this reactor design. Discuss the applicability of the selected correlations under US-APWR specific conditions. List all assumptions made in the calculations including assumptions related to the decay heat model and core decay rate calculations as well as ECCS performance. Provide an assessment for the earliest point in time after which the liquid delivery into the upper plenum is sufficient enough to compensate for the core boil-off rate and flush the core. Address possible impacts of assumptions and uncertainties associated with key parameters on the critical time point obtained. Present plots showing the time variation of quantities such as pressure, temperature, injected ECCS flow rate, steam flow rate, liquid flow rate, and entrainment rate as used and obtained in the analysis.

15.06.05-89

Fibrous debris, in combination with other types of debris, can bypass the US-APWR sump strainer and reach the reactor core region where fuel blockage can take place. Debris can cause fuel blockage near the reactor core inlet region in a direct vessel ECCS injection mode and, in a simultaneous ECCS injection mode, fuel blockage in the top core regions becomes possible.

Discuss effects from fuel blockage by debris in the reactor coolant on the US-APWR boric acid precipitation evaluation. If fluid mixing between the reactor lower plenum and adjacent core regions has been credited in the precipitation analysis, demonstrate that fuel blockage at the core inlet will not preclude or adversely impact coolant mixing between the lower plenum and the core. In addition, show that fuel blockage by debris in the top core area will not interfere with downwards coolant penetration into the core region during the core flushing process.

15.06.05-90

In the response to RAI Question 15.6.5-56 provided in UAP-HF-09384 “MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1” (July 2009), the issue of inherent boron dilution during small break LOCAs in the US-APWR is discussed.

Referring to an evaluation by the applicant, the RAI response cites a minimum core boron concentration required to maintain the reactor subcritical. This value is used as a criterion for assessing the available margin to recriticality following the restart of natural circulation and associated transport of diluted condensate towards the core inlet. It is explained that this value is based on the assumptions stated in the above referenced RAI response. It is also stated that the uncertainty associated with the core criticality evaluation is taken into account.

Provide a full list of reactor core conditions that have been assumed in the criticality calculation for determining the minimum core boron concentration required to maintain the reactor subcritical. In particular, specify the reactor core temperature, reactor coolant pressure, and core life cycle point in time. In addition, quantify any conservative margins included in the calculated minimum core boron concentration such as available shutdown margin and additions to the criticality result for conservatism.

15.06.05-91

According to a core recriticality evaluation for small break LOCAs, as described in the response to RAI Question 15.6.5-56 provided in UAP-HF-09384 "MHI's Response to US-APWR DCD RAI No. 352-2369 Revision 1" (July 2009), the minimum core entry boron concentration during the process of dilute slug propagation towards the core is used to determine if the reactor will remain subcritical. It is stated in this response that the minimum core entry boron concentration provides a safety margin of 307 ppm when compared to the minimum core boron concentration required to maintain the reactor subcritical under certain assumed core conditions. In addition, it is explained that Assumption Number 5 in the above referenced RAI response is considered when determining the minimum core entry boron concentration.

Provide a detailed description of the analytical mixing model used to calculate the minimum core entry boron concentration during the dilute slug propagation process. As appropriate, include the modeling equations as well as any computer programs used to perform the calculations. List all assumptions used to develop the model and to perform the calculations. In particular, describe the initial conditions and provide the input values for the model calculations. Discuss the conservatisms of the obtained results and substantiate the appropriateness of the applied approach.