REQUEST FOR ADDITIONAL INFORMATION 752-5614 REVISION 0

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: SRP Chapter 19

QUESTIONS for PRA and Severe Accidents Branch (SPRA)

19-520

Follow-up to Question 19-442:
Although low power and shutdown (LPSD) events are considered in the Environmental Report, the contribution to the CDF and the fission product source terms used in the offsite consequence analysis may be too low. As discussed in the response to Question 19-442, the CDF could be as high as 2.1E-5 per reactor year if all requirements not required by technical specifications were removed. Moreover, the source terms calculated in the Level 3 PRA for accident scenarios that could occur during mid-loop operations (POS 4 and POS 8) may be underestimated. Specifically, none of the release categories represent severe accidents that may occur during shutdown, when the containment is likely to be open as would the reactor coolant system during mid-loop operations. An arbitrary assumption was made by the applicant to consider the containment to be isolated (but not completely) when the accident occurs, such that the leakage would be 100% of containment volume per day at the design pressure. No justification was given for this assumption, and no procedures to do this were discussed. Based on this assumption, the volatile fission product releases were limited to be a few percent of the initial core inventory.
Since LPSD scenarios are significant contributors to the CDF and LRF, and source terms are expected to be large when containment is open, please determine fission product release fractions and the offsite consequences of not isolating the containment during mid-loop accident scenarios. Specifically, the analyses should use more realistic pathways for releases representative of LPSD scenarios. In addition, please perform sensitivity studies to determine maximum averted costs and which SAMDAs, if any, would become cost-beneficial if the LPSD CDF is increased to 2.1E-5 per reactor year, using release fractions characteristic of partial containment isolation, as well as for no isolation.

19-521

Follow-up to Question 19-448:
In response to RAI 627-4926 Rev 2 (Question No: 19-448), the applicant performed sensitivity calculations using two different node sizes, 0.6 m (Model 1) and 0.2 m (Model 2), as compared to node size of 0.3 m used for the U.S. APWR design certification (DCD) calculations.
The applicant also performed several calculations to study the sensitivity to fragmentation model parameters. In the response to RAI, the applicant justified the ranges for the fragmentation model parameters based on the results of the benchmark

studies for the original TEXAS-V code against the KROTOS experiment data. However, the jet break-up model used in the MHI-version of the TEXAS-V code is considerably different from the original TEXAS-V model. Therefore, the ranges of the fragmentation model parameters that are found to be appropriate for the original TEXAS-V model may not be applicable to the MHI-version of the code. Furthermore, the applicant's analysis showed that the peak cavity pressure is almost insensitive to one of the fragmentation model parameters. However, the NRC calculations have shown a much larger sensitivity in the estimated peak pressure and impulse loads over the same range of the selected fragmentation model parameters using the original (unmodified) TEXAS-V code.

The RAI response also stated that for the U.S. APWR design certification analysis, the evaluated pressure is increased by 10% in order to take into account the modeling uncertainties.

Considering the above noted differences between the results of the original and the MHI-version of the TEXAS-V computer code calculations, the use of pressure/pressure impulse history predicted by the original TEXAS-V code in the U.S. APWR cavity structural analysis may lead to significantly lower margin between the calculated plastic strain and the maximum allowable strain. Therefore, please investigate the implications of using a larger range of uncertainties in the calculated peak pressure/pressure impulse (i.e., as much as 50% instead of the MHI-assumed 10%) associated with the steam explosions-induced dynamic loads on the cavity structural failure probability.

19-522

Follow-up to RAI 19-449:

The MHI analyses show results that are qualitatively consistent with the MELCOR analyses of ERI/NRC 10-205 regarding the build-up of potentially detonable mixtures in the Refueling Water Storage Pit (RWSP).

The MHI response states "It is assumed in this study that occurrence of a detonation will cause the containment structure to fail, and the potential of detonation is probabilistically treated."

In the MHI approach (as part of sensitivity calculation), a containment failure fraction for hydrogen combustion of 0.10 is <u>assumed</u> if hydrogen concentration is greater than 13% (Case 1). Furthermore, as part of a "bounding" analysis (Case 2), this failure fraction is set to 1.0.

MHI has not provided any basis for the <u>assumed</u> containment failure fraction of 0.10; therefore, the argument that this value takes into account the uncertainties is not strong. (The uncertainties that are associated with potentially higher hydrogen concentrations when containment cooling is not available and the RWSP water is not utilized for ECCS and/or CSS.).

MHI dismisses consideration of placement of hydrogen igniters inside the RWSP because of potential complications for maintenance. No other mitigative or design improvements have been considered.

The results of MHI sensitivity calculations show, irrespective of the assumed failure fraction, the Large Release Frequency (LRF) is approximately 10^{-6} per reactor year (ranges from 8.5×10^{-7} to 1.1×10^{-6}). Furthermore, the Conditional Containment Failure Probability is in the range of 0.102 to 0.165.

Given the potential for build-up of large hydrogen concentration inside RWSP, please examine potential changes in the design, operation, and/or implementation of accident management strategies to circumvent the potential for build-up of high combustible gas

concentrations inside the RWSP compartment. Please use the Level-2 PRA to demonstrate compliance with NRC's deterministic and probabilistic guidelines as applicable to combustible gas control. Please discuss any potential accident management procedures that are considered for detection and mitigation of hydrogen build-up inside the RWSP.

19-523

Follow-up to RAI 626-4926, Question 19-450:
The response to Question 19-450 includes a new accident progression event tree (APET) for TI-SGTR, shown on Figure 19-450-1, that considers the research that has been done on relevant phenomena, as well as the potential failure locations. This APET is very similar to that in EPRI Technical Report 1006593, and also includes an event to address RCS depressurization as a result of neutron flux measurement tube (ICIS) failures in the core region. The other events include no turbine-driven emergency feed water, stuck-open MSSVs, no large RCP seal LOCAs, no concurrent loop seal and core barrel clearing, high SG pressure in loop where loop seal clears, and probability of temperature-induced tube ruptures. Quantification of the modified APET is as follows (the original APET appears in the response to RAI 480-3711, Question 19-xxx(3)).
The applicant utilized this newly-developed APET to estimate the conditional containment failure probabilities, for the base case and the sensitivity case, as 0.0136 and 0.0165, respectively. LRF evaluations addressing this APET calculation results are shown in Tables 19-450-1 and 19-450-2 for the APET base case and the sensitivity case, respectively. Only small increases are shown, relative to Revision 2 of the DCD.
The staff's confirmatory assessments on induced steam generator tube rupture evaluated the locations and timing of potential reactor coolant system (RCS) failures in the US-APWR during a high-pressure station blackout scenario with depressurized steam generators. It was found that, with hot leg counter-current natural circulation considered, the possibility of a creep-induced steam generator tube rupture cannot be ruled out under high-pressure accident conditions with depressurized steam generators. Moreover, if there are any pre-existing cracks in the tube walls of the steam generators, their resulting increased susceptibility to creep-rupture may make such a failure more likely than not. Specific conclusions drawn from this study are as follows:

· The most likely point of induced failure in the RCS would be at portions of the hot leg made up of carbon steel (i.e., the reactor vessel nozzle and welds) in the loop containing the pressurizer.
· Induced failure in the hot legs of non-pressurizer loops are predicted to occur almost simultaneously (within about a minute) with the pressurizer loop.
· Based on the hottest temperature of an average tube in the steam generator tube bundle, creep-induced rupture of a tube is predicted to occur either never or significantly past the time of hot leg failure, assuming no flaws in the tube wall.
· Average steam generator tubes (at the positions along their length of highest temperature) are predicted to become the earliest point of creep-induced failure if they are flawed by cracks of one inch length (e.g., due to presence of foreign objects as a result of maintenance) and at least 66% or more through-wall depth. A one-inch crack of 50% depth would cause the average tube to fail about 10 minutes after the hot leg. A crack of 40% depth would cause tube failure about 14 minutes after hot leg failure.

- The hottest tubes in the steam generator are predicted to fail at more or less the same time as the hot leg nozzle even if no flaws in the tube are assumed. For any significant flaws in the hottest tubes, induced tube rupture at this hottest tube bundle location would be the first point of failure in the RCS.
- If failure of the ICS neutron flux measurement thimble tubes occurs, it is likely that the resulting de-pressurization of the reactor cooling system is sufficient to allow accumulator injections. However, such depressurization may occur too late to significantly influence the timing of the earliest thermally induced ruptures of cooling system structures.
- If all pump seals are assumed to leak at an initial rate of 300 gpm (the MHI assumption), the failure of the average, unflawed tube in the pressurizer-loop steam generator follows the earliest failure of any other RCS component by only one minute. This enhanced propensity for early tube failure is due to vigorous whole-loop natural circulation in the pressurizer loop, enabled by complete clearing of the loop seal, in consequence of the pump seal leaks.

Comparing the results of the staff's confirmatory analyses to MHI's evaluation suggests that the conditional probabilities assigned in the APET, while reasonable, may not adequately cover the range of thermal-mechanical uncertainties, particularly for RCS depressurization following ICIS tube failure, and for no concurrent loop seal and core barrel clearing. Since there are significant uncertainties in the treatment of these phenomena, the staff requests the applicant to perform some sensitivity calculations, varying the success values of the two split fractions as follows:

- RCS depressurized due to ICS tube release: 0.5 or 0.0
- No concurrent loop seal and core barrel clearing: 0.9 and 0.5 for no turbine-driven EFW, and 0.99 or 0.5 for turbine-driven EFW values.

Please report the CCFP, LRF and delta LRF results for both the base-case and the sensitivity case for tube failure probability, in the same format as Tables 19-450-1 and 19-450-2, for all combinations of split fractions noted in the above two bullets. Please update the DCD, as necessary.