REQUEST FOR ADDITIONAL INFORMATION 872-6144 REVISION 0

11/21/2011

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)

19-561

Follow-up to RAI 752-5614, Question 19-523. The APET for the TI-SGTR currently assumes that RCP seal LOCAs greater than 170 gpm per pump could depressurize the reactor, thus precluding TI-SGTR. However, even with seal leakage of 300 gpm per pump, NRC-sponsored MELCOR calculations for the US-APWR have shown that there is a strong likelihood of induced SGTR in the same time frame as the earliest failure of any other RCS components. To account for this, please add new branches to this event tree, or correct the branch probability for large RCP seal LOCAs to be the complement of 0.808 for no large RCP seal LOCAs, instead of the assigned value of 0.027. Alternatively, please provide an analysis to justify the value chosen for seal LOCAs greater than 200 gpm per pump. In addition, please revise the LRF and CCFP based on the results obtained.

19-562

In the responses to RAI No. 752-5614 Questions 521 through 523, the applicant used sensitivity analyses to determine the impacts of various assumptions on LRF. Based on these results, even though the individual assumptions may or may not have a large impact; the combined effect could lead to a result that exceeds the Commission's LRF guideline of 1.0×10^{-6} per reactor year.
The staff was able to perform an independent verification of the sensitivity results for the internal at-power events. However, due to the lack of sufficient details on PDS assignments of fire and flood events provided in the MHI Level-2 DCD and PRA Revision 3, the staff was not able to perform an independent verification of the sensitivity analyses and the changes in LRF values that are cited for the fire and flood initiators. In the revised sections on Level-2 PRA related to fire and flood initiators that are in response to RAI 627-4926 Question 19-447, the applicant provides additional information in the DCD and in Sections 23S.6 and 22D.7 of the PRA, and it indicates that the approach for quantification of release categories is basically the same as that for internal events at-power. However, the supplied information does not contain any results that can be used for independent confirmation. For example, the sensitivity analysis for scenario 1 in the Question 521 response shows that the changes in fire and flood LRF due to an increase in ex-vessel steam explosion probability are about 16% and 42%, respectively. According to the discussion in Chapter 17 of the PRA, the ex-vessel steam explosion is only possible for PDS 3* (*=<A-H>). Therefore, all the relevant PDS frequencies, the conditional containment failure probabilities, and finally the release frequencies are already available to the applicant and can be readily provided.

In order to independently verify the effects of changes in phenomenological events conditional probabilities on release categories in the sensitivity analyses, please provide additional information that summarizes the various accident class and PDS assignment probabilities for fire and flood events, in a manner similar to those provided in Chapter 17 for internal events at-power. Specifically, please provide the individual fire and floods PDS frequencies to be used as input to the containment phenomenological event tree analyses and include this information in the DCD.

In addition, please justify why the CDF contributions from fire and flood events during LPSD are not included in the tabular summary of CDF or LRF values for LPSD sensitivity studies. If the contribution from LPSD LRF results in a total LRF value that is above the Commission goal of 1.0×10^{-6} per reactor year for all initiators including the LPSD modes of operation, please justify the acceptability of this result.