REQUEST FOR ADDITIONAL INFORMATION 92-1237 REVISION 0

11/5/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

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

QUESTIONS for PRA Licensing, Operations Support and Maintenance Branch 2 (ESBWR/ABWR Projects) (SPLB)

19-151

On page 11-3 of MUAP-07030 (R0), it is stated that hydrogen control is to be achieved through twenty glow-plug type hydrogen igniters. Please identify the power source for these igniters.

19-152

Please justify the statements on the last paragraph of Section 11.2.2.2 of the PRA that dismiss the potential for recriticality following firewater injection. Please provide the results of your calculations or other technical evidence that support these statements.

19-153

The results of the Sandia tests (Table 15-17 of the PRA) geared to the U.S. designed PWR cavities may not be directly applicable to the assessment of high pressure melt ejection for U.S. APWR (i.e., it is not clear why the APWR cavity is considered to be similar to that of Zion). Please demonstrate either by presenting scaled test data or analyses that are supported by data prototypical of APWR reactor cavity configuration that support the discussions of Section 11.2.2.3, and the analysis approach of Section 11.3.4.3, and Chapter 15 (15.6.3), specifically:

(a) Please justify the range of the RV breach size that is considered in Table 15-19. Why are larger breach areas excluded?

(b) Please discuss the applicability of TCE to APWR geometry and conditions? Please list all the TCE dispersal parameters that have been used in the analysis, and the basis for their selection.

In addition, on page 19.1-11 (under the heading Core debris trap), it is stated that "...the effect of this design feature is not explicitly addressed in the Level 2 PRA…" Please describe:

(c) The degree of trapping expected for the reactor cavity (and subsequently the degree of dispersal to adjacent compartments).

(d) The existence of any flow paths around the reactor vessel that could directly connect the reactor cavity to the upper containment compartments.

19-154

In Section 15.3.3.1 of the PRA, it is not clear how the hydrogen is being partitioned between the in-vessel and ex-vessel phases (last paragraph before Section 15.3.3.1 talks about 1/3 released in one minute after RV failure as hydrogen generation by “breaking up” and 2/3 in 5 minutes after that as hydrogen due to MCCI). Please explain why this is considered as bounding and/or representative of severe accident conditions for APWR.

19-155

Please provide the technical basis for the concentration criteria of various hydrogen combustion modes (e.g., deflagration, DDT, etc.) in Section 15.3.3.4.2 of the PRA.

19-156

Please provide justification for the scenarios considered for the evaluation of hydrogen generation and distribution in Section 15.3.3 of the PRA are bounding for APWR severe accidents. Please explain the impact of uncertainties in hydrogen release and transport on the overall conclusions.

19-157

Page 19.2-8 of the FSAR lists a maximum pressure in the containment vessel under adiabatic isochoric complete combustion of 137 psia, whereas the results on page 15-15 of the PRA chapter (MUAP-07030 [R0]), lists a pressure ranging from 127 to 152 psia, depending on the extent of Zr oxidation. Please provide the source for the pressure of 137 psia that is listed in the FSAR. If this value is based on a revised analysis, please provide the details of the calculations and the results.

19-158

The hydrogen combustion evaluation in PRA Section 15.3.3.5 notes that deflagration to detonation transition is not expected when the hydrogen concentration does not exceed 10%.

(a) Please provide specific citations of experimental or analytical results in support of this statement.

(b) If some of the glow plug igniters were to become inoperable, please explain whether situations would arise where pressure waves may arise locally, propagate, and/or cause structural failures. Also, please explain how many igniters would be needed to fail for this to occur. Please provide the results of any sensitivity studies that have been performed to assess such situations.

19-159

The PRA Chapter 17, quantification of the probability of induced steam generator tube ruptures refers to the 1993 report NUREG/CR-4551. A remark is also made referring to unspecified "recent studies". Please identify these studies, and present an updated analysis of this phenomenon for the US ABWR using current technology and analyses (for example, NUREG-1570, and EPRI Technical Reports 100693 and TR-107623) in support of the numerical quantifications presented. This analysis should also include the impacts of depressurizing of the secondary side of the steam generators on SG tube integrity.

In addition, please provide and justify the maximum number of broken tubes that were assumed in the case of a temperature-induced SGTR, considering the fact that it is not clear that a single broken tube would necessarily depressurize the primary side of the system to an extent that would mitigate further tube failures.

19-160

In Section 15.4.2.1, it is stated, “This condition is a conservative estimation in terms of the molten core spreading behavior.” Please explain why the sub-cooled cavity water and high heat removal capability are considered to be conservative when evaluating melt spreading.

19-161

The FLOW-3D code is used to assess core melt spreading in the cavity. Please present a brief discussion of the special capabilities of this code. Describe the method of qualifying the code for this application in terms of validation and verification. Describe special modeling, if any, added to the code for the US-APWR analyses. Please provide the experimental validation basis of this code for application to debris pool spreading. In addition:

(a) Provide references supporting the statements (Section 15.4.2.3 of the PRA) that it is " widely recognized" for its ability to evaluate the solidification behavior of molten materials, that the solidification model is considered having a good ability in its prediction, and that this model is expected to predict precisely the solidification form and spreading size of molten metal in the casting processes.

(b) Please describe what is meant by the volume of fluid (VOF) method in FLOW-3D code.

(c) In Section 15.4.2.3 please provide the references for “Previously performed studies show that the heat transfer coefficient of molten core and a reactor cavity….decreases to a value ranging from 1.76 x10^2 to 4.40 x10^2…as a result of formation of a crust layer.”

19-162

The conclusions based on the OECD MCCI experiments in Section 15.4.3.1 of the PRA on debris coolability need to be clearly tied to the specific experimental data. Please cite the exact data and experiments that have shown that debris was quenched and coolable, or identify and justify any other method that was used.

19-163

In Section 15.4.2.3 of the PRA, the film boiling heat transfer coefficient is listed as α = 8.81 x 10 Btu/hr-ft^2-ºF, please provide the missing power (exponent) of 10.

In addition, reference is made to the use of FLOW-3D; however, the submittal does not contain any information on the modeling features, solution methods, and the experimental validation and technical basis for the code. Please provide the technical details of the FLOW-3D computer code, including its experimental validation basis for application to debris pool spreading.

19-164

Please justify why the violation of acceptance criterion for basaltic concrete (case M1-2) in PRA Section 15.4.4 L/CS is not considered to be significant.

19-165

Please provide a list of various analysis cases that have been considered for ex-vessel steam explosions. Please include the following information for each case:

- Debris pour composition
- Lower head hole size
- Pour temperature
- Pour velocity
- Cavity water temperature
- Cavity water depth
- Location of RV failure (middle or at the side)

For each case, please provide the peak pressure and the impulse load on the cavity wall.

19-166

The assessment of ex-vessel steam explosions considers two potential containment failure modes, namely, (1) displacement of primary system loops and connecting steam generators challenging containment penetrations; and

(2) dynamic loads on the reactor cavity wall structure. Dynamic structural responses for both modes are assessed using the LS-DYNA code (a non-linear finite element analysis program developed by Livermore Software Technology Corp). Describe the method of qualifying the LS-DYNA code for this application in terms of validation and verification. Describe special modeling, if any, added to the code for the US-APWR analyses.

19-167

The probability of an ISLOCA is described to be negligible due to design provisions to mitigate this accident. Please provide a more quantitative basis in regard to the design and the probabilities in support of the omission of the ISLOCA accident from the PRA.

19-168

The Accident Classes (ACLs) and plant damage states (PDSs) do not have specific identification of accident progression timing such as early or late. Please explain if it is the intent of MHI to include the Level 3 PRA reported in FSAR Chapter 19.2 as part of the PRA in Chapter 19.1 in a future revision.

19-169

MAAP version 4.06 is used to support the U.S. APWR Level 2 PRA. Describe the method of qualifying the code for this application in terms of validation and verification. Also, please describe special modeling, if any, added to the code for the US-APWR analyses.

19-170

GOTHIC7.2a-p5(QA) code is used for certain containment analyses in support of the Level 2 PRA. Please present a brief discussion on the special capabilities of this code, emphasizing the benefits and limitations of its application to complement and supplement MAAP. Please describe the method of qualifying the code for this application in terms of validation and verification, and describe special modeling, if any, added to the code for the US-APWR analyses.

When describing the capabilities of the code related to hydrogen burn in containment, please explain why MAAP 4.0.6 was not used in the accident progression analyses (see the assumptions in Section 14.2.1 of the PRA), and why the igniter model was not included in the developed MAAP model.

19-171

A "Summary of Level 2 PRA Results from Internal Events at Power" is presented in PRA Table 17.3-9. It is not clear if internal floods and fires are included in these internal events. This question also applies to the results and tables in FSAR Section 19.1. Please clarify this in the related sections of the PRA report.

19-172

In PRA Section 15.4.2, the FLOW-3D calculation of the corium spreading behavior on the US-APWR cavity floor is discussed. Previously performed studies are noted as showing that the heat transfer coefficient between the molten corium and the floor decreased as a result of crust formation. The PRA analysis however assumed no crust formation as conservatism.

(a) Please reference and summarize these previous studies.

(b) Discuss the conservatism of this assumption.

(c) Explain how this assumption is evaluated or included in the sensitivity studies performed for corium cooling and concrete erosion.

19-173

In PRA Section 15.4 "Core Debris Coolability and Molten Core Concrete Interaction", the capability of the US-APWR design to withstand the effects of molten corium release from the vessel is analyzed through calculations based on MAAP and FLOW-3D. A summary of early relevant studies is presented in support in Table 15-3. Since then, more experiments, analyses, and models have been developed. Please present an updated analysis of this phenomenon for the US-APWR using current technology and analyses in support of the numerical quantifications presented, or justify why no further analysis is needed.

19-174

(a) The "No" column of PRA Table 17.3-10 starts at Number two, and there is no Number one. Please explain or correct this error.

(b) The note to FSAR Table 19.1.50 says that the uncertainty sources are categorized into three types. Only two are listed, and "completeness" seems to have been omitted or deleted. Please explain or correct this error.

19-175

Please confirm whether the overall LPDS analyses were only for the Level 1 PRA model and did not include the CSETs or the CPETs.

19-176

Some CET results are missing from PRA (MUAP-07030 R0) Table 17.3-7. Sheet 5 appears to be a duplicate of Sheet 4. Please provide corrected Sheet 5.

19-177

The information on the Level-3 PRA is provided outside of the FSAR (MUAP-08004-P). It is, however, pertinent to the evaluation of severe accident mitigation design alternatives. This analysis provides information on release categories, source terms and consequence analysis. For consequence analysis, the study primarily uses the data extracted from the MACCS2 manual for the majority of the site-specific input data, including the regional economic data.

(a) The analysis indicates consequence analysis was limited to releases occurred within 24 hours after the onset of core damage. Please provide technical bases for not considering releases after 24 hours, given that the release durations have been broken into four plumes, with the final plume release exceeding the 24 hours.

(b) For release category 5, late containment failure, since releases occur after 24 hours, no consequence analysis was performed. Please provide clarification on mitigation measures that eliminate the need for consequence analysis for this release category.

(c) It is not clear how the limiting 24 hour release duration affects various durations used in the MACCS2 sample problem. For example, the duration of emergency phase in the sample problem is one week. It is not clear if the analysis in this report used similar duration as that of the sample problem. In addition, release duration in MACCS2 should be less than 10 hours. The plume release duration data in the report indicates values in excess of 10 hours. Please provide MCCAS2 input files used for the consequence analysis.

(d) Please provide equilibrium core inventory source term given potential for high burnup.

(e) The property data used in the economic loss is essentially those of 1970 values. Please explain how the current property values would affect the results presented for offsite property damage.

(f) For release category 6, intact containment, the release would be continuous and in the order of technical specification limit. Please provide the details of how this release is modeled using MACCS.

19-178

The regulations in 10 CFR 52.47(b)(2) and 10 CFR 51.55(a) require the applicants to prepare an environmental report that includes a cost and benefits of severe accident mitigation design alternatives (SAMDA). The environmental report supporting the SAMDA analysis refers to a level 3 PRA that was performed to determine the overall risk perspective of the US APWR design (MAUP-DC201). The analysis uses a standard method (template) as provided in the NEI 05-01 for SAMDA analysis in support of license renewal.

Review of the methods and assumptions has identified the following:

(a) Table 2 of the ER provides a list of SAMDA candidates. The table does not provide any details on assumptions and basis for screening/modification evaluation. Please provide the assumptions and basis for the screening of candidate SAMDAs.

(b) One screening criterion is labeled “not a design alternative.” All except a few of these SAMDA items related to procedures and training. Please identify the screened out list that needs to be considered by COL holders of US APWR design. Also, please explain why the SAMDA items related to improving uninterruptable power supplies and enhancing control of combustible and ignition sources are screened under this criterion.

(c) NUREG/BR-0184 provides a range of doses and costs for occupational exposure, onsite clean up costs, and replacement power costs. The report used only the best estimate values with 3 and 7 per cent discount rates to estimate the range of potential averted costs. Given that these estimates are dated (1992 circa), please elaborate why the analysis did not consider potential uncertainties in these values, and evaluate the impacts on the cost benefit analysis.

(d) Table 12 of the ER provides a sensitivity analysis for the selected SAMDA items benefits. The report indicates that each SAMDA item benefit is calculated using ratio of each item contribution to decrease CDF or large release frequency (LRF). Please explain where these ratios are provided.

19-179

Section 19.5 of the PRA provides the results of the uncertainty analysis for the LRF. It does not discuss how this calculation was carried out. Please provide a detailed discussion of the analysis including methods and data, especially with respect to the CPET portion of the LRF calculation.

19-180

In Table 2.1-9, page 39 of the Level 3 PRA (MUAP-08004-P (R0)) and Figures 2.1-17 and 2.1-21, the release of CsI is significantly higher than that of CsOH. This larger release for CsI appears to be as result of late revaporization of previously deposited aerosols on the RCS structures (evident from Figure 2.1-17). Please explain the reasons for the absence of this revaporization contribution for CsOH resulting in about an order of magnitude lower release for CsOH as compared with CsI.

19-181

In the TMI-2 accident, it could be possible that the instrument tubes failed from oxidation of the Zircaloy cladding, causing steam, hydrogen, and fission products to be released to the containment building before the B-loop pump was restarted (see R. E. Henry’s presentation to the MAAP Users’ Group on May 7, 2008). Please explain what would be the consequences of such a development in the US-APWR for both high RCS pressure and low RCS pressure severe accident scenarios. In addition, please provide the location of hydrogen, steam, and fission product entry into the containment building, the hydrogen, steam, oxygen, and nitrogen mole fractions in the containment vs. time, the CsI, CsOH, and SrO fractions in the containment, and the containment pressure increase vs. time.

19-182

Please provide the following design information related to the in-core instrument tubes:

(a) A schematic of a typical in-core instrumentation tube and associated guide tube geometry.

(b) The outside guide tube diameter, material, and wall thickness.

(c) The material, the diameter of any casings where the thermocouples and/or neutron detectors are located, the wall thickness, and typical cross sectional drawing of the in-core detector and guide pipe.

(d) The material, its high-temperature yield strength as a function temperature, and its creep-rupture properties (e.g., Larson-Miller parameters) for the in-core instrument guide and other parts of the in-core detectors.

(e) A schematic of the in-core instrumentation guide tubes configuration, their location of entry into the reactor pressure vessel, and the location of entry into the instrumentation room inside the containment. This information should also show the actual location inside the containment (to be accompanied by a drawing of the containment marking the location of the instrumentation room).

(f) Description of instrumentation room compartment (room size), locations and opening areas for (flow) communication with adjacent compartments.

(g) Location of the instrument tube/guide pressure boundary between the reactor vessel and the containment building, and a discussion of failure of this boundary for both low pressure and high pressure scenarios.