Attachment (1) to Question #19-300

*KNFCS-MHI-2007-5*

# Probabilistic Risk Assessment Peer Review Report for Level 2 PRA

## Results of a Peer Review for the PRA for MHI US APWR Design Certification

**Prepared for:**

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## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**3/11/2009**

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** **NO. 197-1800 REVISION 0**

**SRP SECTION:** **19 – Probabilistic Risk Assessment and Severe Accident Evaluation**

**APPLICATION SECTION:** **19**

**DATE OF RAI ISSUE:** **2/9/2009**

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**QUESTION NO.: 19-301**

Please provide the discussion of the quantification of the CPET, particularly the levels of uncertainty associated with the qualitative evaluations and the assignment of quantitative probabilities to qualitative failure attributes.

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**ANSWER:**

One of the previously performed discussions on the uncertainty can be found in the NUREG/CR-4700 (Reference 1), in which the sensitivity analysis for the levels of uncertainty associated with the qualitative evaluations and the assignment of quantitative probabilities to qualitative failure attributes is presented.

In the NUREG/CR-4700, three kinds of evaluations, including optimistic evaluation, central evaluation and pessimistic evaluation are performed in order to evaluate the difference in the qualitative judgment among experts. Optimistic evaluation and pessimistic evaluation directly reflect the variation in the probability determined in accordance with various opinions by experts. In each evaluation, the probability values of each branch point of the containment event tree are different. In the case of pessimistic evaluation, the values are assigned to be higher probability for the pathways that lead to the higher source terms, and lower probability for the pathways that lead to lower source terms. In the case of optimistic evaluation, the values are determined in the opposite direction to the pessimistic evaluation. Meanwhile the central evaluation represents the median of the reactor safety community on any particular issue of the containment event tree. The central evaluation is accordingly defined to be that half of experts would consider the actual situation becomes better than that indicated by the central evaluation, and another half would consider the situation becomes worse. Therefore, optimistic evaluation and pessimistic evaluation reflect the variation in the probability determined in accordance with various opinions by experts.

The standard shown in the Table 19.301-1 is established to quantify the qualitative judgment by experts. ALT. 1(Base Case) is used for normal evaluations and other alternatives (ALT. 2~4) are used for the sensitivity analysis. The sensitivity analysis result is shown in Table 19.301-2. The effect of the assignment of quantitative probabilities to qualitative failure attributes on the conditional containment failure probability is evaluated as much as approximately 10%. As the result, in the NUREG/CR-4700, it is concluded that there is influence on the sensitivity results due to the difference in the qualitative judgment such as optimistic evaluation or pessimistic evaluation although the magnitude of influence on the results by the assignment of quantitative probabilities to qualitative failure attributes is very limited.

It is considered that these conclusions are also applicable to quantification of the CPET of US-APWR.

The assignment of failure fractions in the CPET is shown in Table 19.301-3. It is considered that the uncertainty caused by the qualitative judgment whether it is "likely" or "unlikely" resulting in containment failure from a certain phenomenon may influence the evaluation, as concluded in NUREG/CR-4700. However, in the evaluation of the US-APWR CPET, the magnitude of influence is considered very limited. It is because the qualitative judgments in the CPET are based on the severe accident progression analysis (shown in Chapter 14 of the PRA technical Report), separate effect analysis (shown in Chapter 15 of the PRA technical Report) and the previously performed PRA result and experimental result which are considered that application to US-APWR, accordingly the anticipated uncertainty involved in the qualitative judgment is to be small.

On the other hand, it is considered that the influence caused by uncertainty associated with the assignment of quantitative probabilities to qualitative failure attributes is also very small, same with the conclusion of NUREG/CR-4700. This can be justified in the sensitivity analysis for the debris coolability performed in Subsection 18.6.2 of the PRA technical report. The failure fraction of the debris coolability is increased by 50 times from 0.002 to 0.1, then the LRF is evaluated as 2.1E-07/ry, which is as little as approximately twice of the base case LRF.

Reference:

1 Containment Event Analysis for Postulated Severe Accidents: Surry Power Station, Unit 1, NUREG/CR-4700, Vol.1, February 1987

**Table 19.301-1 Alternative Assignment of Values to Verbal Descriptors in the NUREG/CR-4700**

| Verbal Descriptor | Likelihood | | | |
|---|---|---|---|---|
| | ALT. 1 (BASE CASE) | ALT. 2 | ALT. 3 | ALT. 4 |
| Certain or Almost Certain | 1.0 | 1.0 | 1.0 | 1.0 |
| Likely | 0.9 | 0.9 | 0.9 | 0.9 |
| indeterminate | 0.5 | 0.5 | 0.5 | 0.5 |
| Unlikely | 0.1 | 0.01 | 0.01 | 0.1 |
| Remotely Possible | 0.001 | 0.001 | 0.0001 | 0.01 |
| Impossible | 0 | 0 | 0 | 0 |

**Table 19.301-2 Sensitivity of Results of Conditional Probabilities to Alternative Assignments of Numerical Values in the NUREG/CR-4700**

| | Optimistic | | | | Central | | | | Pessimistic | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALT 1 | ALT 2 | ALT 3 | ALT 4 | ALT 1 | ALT 2 | ALT 3 | ALT 4 | ALT 1 | ALT 2 | ALT 3 | ALT 4 |
| Unlikely | .1 | .01 | .01 | .1 | .1 | .01 | .01 | .1 | .1 | .01 | .01 | .1 |
| Remote Poss. | .001 | .001 | .0001 | .01 | .001 | .001 | .0001 | .01 | .001 | .001 | .0001 | .01 |
| No failure | .95 | .99 | .99 | .93 | .46 | .54 | .54 | .45 | .006 | .007 | .007 | .006 |
| Meltthrough | .05 | .006 | .005 | .06 | .45 | .45 | .45 | .45 | .06 | .06 | .06 | .06 |
| Late Overpressure | .001 | .001 | | .01 | .09 | .01 | .01 | .09 | .03 | .02 | .02 | .03 |
| Late Leak | | | | | | | | | .01 | .01 | .01 | .01 |
| Late H2 burn | | | | | .001 | | | .001 | .06 | .05 | .05 | .06 |
| Early Leak | | | | | | | | | .03 | .03 | .03 | .03 |
| Steam Spike | | | | | | | | | | | | |
| Early H2 burn | | | | | | | | | .04 | .04 | .04 | .04 |
| Steam Spike + H2 burn | | | | | | | | | .77 | .77 | .77 | .77 |
| Isolation Failure | | | | | .002 | .002 | .002 | .002 | .002 | .002 | .002 | .002 |

**Table 19.301-3 Assign of Failure Fractions in the CPET**

| Verbal attribute | Probability value |
|---|---|
| Practically impossible | 0.0001 |
| Extremely unlikely | 0.001 |
| Very Unlikely | 0.01 |
| Unlikely | 0.1 |
| Not very likely | 0.25 |
| Undecided | 0.5 |
| Not very unlikely | 0.75 |
| Likely | 0.9 |
| Very likely | 0.99 |
| Extremely likely | 0.999 |
| Practically certain | 0.9999 |

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO.: 19-302**

Please provide a summary of the effects of the severe accident containment environment conditions on the CPET and the CSET, including a discussion of the effects of including emergency and SAMG operator actions in both parts of the CET.

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**ANSWER:**

The CSET deals with the containment systems to mitigate the severe accident containment environment. The survivability of equipment used in these mitigation features is discussed in Subsection 15.7 of the PRA technical report. It is concluded from the study that the containment systems significant to mitigate severe accident progression keep the functions available to maintain containment integrity. Therefore there is no effect from the severe accident containment environment to the system function considered in the CSET. On the other hand, operation of the systems considered in the CSET significantly influences the containment environment. The plant damage states (PDS) are defined in accordance with the containment environmental conditions and the CPET is developed to model all PDSs. The CPET deals with the containment phenomenological event during the postulated severe accident progression. The environmental change of containment is therefore addressed in the CPET portion. Meanwhile, influence of the change in the containment environmental condition caused by other phenomenological event considered in the CPET is eliminated in this study through the selection of the CPET top event.

The effects of containment environmental change by operator actions influencing the CPET are summarized in Table 19.302-1.

**Table 19.302-1 Summary of Effect of the CPET**

| No | CPET Top Events | | Effect of severe accident containment environment change by operator actions | Description in the PRA technical report |
|---|---|---|---|---|
| 1 | IHL | Temperature-induced hot leg rupture before temperature-induced SGTR and vessel melt through | No effect; since this event scarcely depends on the containment condition but highly does on the RCS and SG conditions. | Subsection 17.3.3.1 |
| 2 | BP | No temperature-induced SGTR before temperature-induced hot leg rupture and vessel melt through | No effect; since this event scarcely depends on the containment condition but highly does on the RCS and SG conditions. | Subsection 17.3.3.2 |
| 3 | ISX | No containment failure from in-vessel steam explosion | No effect; since this event scarcely depends on the containment condition but highly does on the RCS condition. | Subsection 17.3.3.3 |
| 4 | HB1 | No containment failure from hydrogen-burn before vessel melt through | Change in the steam and hydrogen concentrations significantly influence the possibility of this event. | Subsection 17.3.3.4 |
| 5 | ESX | No containment failure from ex-vessel steam explosion | Change in the status of reactor cavity flooding significantly influences the possibility of this event. | Subsection 17.3.3.5 |
| 6 | DH | No containment failure from direct containment heating and rocket-mode reactor vessel failure | Change in the status of reactor cavity flooding significantly influences the possibility of this event. | Subsection 17.3.3.6 |
| 7 | HB2 | No containment failure from hydrogen-burn after vessel melt through | Change in the steam and hydrogen concentrations significantly influence the possibility of this event. | Subsection 17.3.3.7 |
| 8 | EVC | Debris quenched, cooled long-term and containment cooled | Change in the containment pressure and the status of reactor cavity flooding significantly influence the possibility of this event. | Subsection 17.3.3.8 |

Impact on DCD

There is no impact on DCD from this RAI.

Impact on COLA

There is no impact on COLA from this RAI.

Impact on PRA

There is no impact on PRA from this RAI.

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**QUESTION NO.: 19-303**

Please explain the methodology and results of the evaluation of induced steam generator tube ruptures, given failure to depressurize. Also, provide the discussion of the results of scenarios where the steam generators are also depressurized. In addition, please justify the assumption of zero break areas for hot leg creep rupture and temperature-induced steam generator tube rupture.

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**ANSWER:**

The NRC and MHI had a face-to-face meeting to discuss this issue from 17 to 19 February, and MHI understood more about the intention of the NRC staff for this question. MHI would therefore like to request the NRC staff to allow two more months to prepare the answer, in which additional analysis results and discussions will be included.

MHI will answer to this question by 28 April 2009.

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**QUESTION NO. : 19-304**

Please describe and justify the criteria that would be used to manually depressurize the reactor coolant system during a high-pressure severe accident.

For sequences AM001 and AM002 on the effectiveness of RCS depressurization for a small-break LOCA and a main steam line break outside containment, respectively, leading to a severe accident, RCS depressurization was enabled 10 minutes after core melt. What are your definitions of core damage and core melt? At the times of core damage and core melt, what are the core outlet temperatures, the amounts of hydrogen generated, and the damage fractions for the hot leg and steam generator tubes for each sequence? For each sequence, please provide plots, from the start of the sequence until the time of vessel failure, of the core-to-upper plenum natural circulation, the natural circulation between the upper plenum and the steam generators, and the countercurrent natural circulation flow rates in the hot legs and in the steam generators.

Please verify that the study of the effectiveness of RCS depressurization features in Section 15.6.2 of the PRA is based on these two sequences. Note that, in Section 15.6.2, it is stated that the depressurization valves are assumed to be manually opened 10 minutes after core damage.

Since the large release frequency (LRF) in existing reactors is dominated by high pressure sequences where the secondary side has been depressurized, please provide analyses of variations of the two cases described above, in which one or more steam generators have been depressurized prior to the onset of zircaloy oxidation. Provide the same results requested above for comparison, and report when the hot leg and/or the steam generator tubes would fail from creep rupture.

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**ANSWER:**

This question will be answered by 10 April 2009.

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**QUESTION NO. : 19-305**

It is important to assure containment integrity for at least 24 hours after accident initiation. Chapter 14 of the PRA presents results of accident sequences AM003, AM004, AM005, AM006, and AM007 to evaluate the ability of alternative containment cooling and/or cavity flooding to avert overpressurization, basemat meltthrough, or failure of RPV support. Please run a similar scenario, in which there are no containment sprays, no fan coolers, and no fire water injection. Please provide results in the format of Table 14.5 for comparison.

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**ANSWER:**

Analysis results of accident sequences in which there are no accident management are also provided in Chapter 14 of the PRA technical report. IDs of these accident sequences are represented as AP***. Relationship between the sequences with accident management relating to containment heat removal and those without it is summarized below.

- The accident sequence AM003 corresponds to AP005 in which alternate containment cooling using containment fan cooler unit is started 30 minutes after the containment pressure exceeds the containment design pressure.

- The accident sequence AM004 corresponds to AP103 in which alternate containment cooling using containment fan cooler unit is started 30 minutes after the containment pressure exceeds the containment design pressure.

- The accident sequence AM005 corresponds to AP001 in which firewater injection to the reactor cavity is started 10 minutes after onset of core melt and then alternate containment cooling using containment fan cooler unit is started 30 minutes after the containment pressure exceeds the containment design pressure.

- The accident sequence AM006 corresponds to AP101 in which firewater injection to the reactor cavity is started 10 minutes after onset of core melt and then alternate containment cooling using containment fan cooler unit is started 30 minutes after the containment pressure exceeds the

containment design pressure.

- The accident sequence AM007 corresponds to AP101 in which firewater injection to the reactor cavity is started 10 minutes after onset of core melt and then firewater injection to the spray header is started 30 minutes after the containment pressure exceeds the containment design pressure.

Analysis results of these accident sequences are summarized in Table 14-3 through Table 14-11. The results show that containment integrity is maintained more than 24 hours after initiating event for any accidental scenarios including no containment heat removal function is available.

Impact on DCD
There is no impact on DCD.

Impact on COLA
There is no impact on COLA.

Impact on PRA
There is no impact on PRA.