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TOKYO, JAPAN

April 28, 2009

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09201

Subject: MHI's Third Responses to US-APWR DCD RAI No.197-1800 Revision 0

References:
1) "Request for Additional Information No. 197-1800 Revision 0, SRP Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: 19," dated February 9, 2009.
2) Letter MHI Ref: UAP-HF-09085 from Y. Ogata (MHI) to the U.S. NRC, "MHI's Responses to US-APWR DCD RAI No. 197-1800," dated March 11, 2009.
3) Letter MHI Ref: UAP-HF-09163 from Y. Ogata (MHI) to the U.S. NRC, "MHI's Second Responses to US-APWR DCD RAI No. 197-1800," dated April 10, 2009

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document as listed in Enclosures.

Enclosed are the third responses to the RAI of Reference 1, especially for the Question #19-303 contained within Reference 1. In the first responses submitted with Reference 2, MHI committed to submit responses to 19-304 within 60 days after RAI issue date (answered within Reference 3), and to submit responses to 19-303 by 28 April 2009.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

D081
NRO

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Third Responses to Request for Additional Information No. 197-1800 Revision 0 (proprietary version)

3. Third Responses to Request for Additional Information No. 197-1800 Revision 0 (non-proprietary version)

CC: J. A. Ciocco
C. K. Paulson

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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-09201

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Third Responses to Request for Additional Information No. 197-1800 Revision 0" dated April 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design and methodology developed by MHI for performing the design of the US-APWR reactor.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

    A. Loss of competitive advantage due to the costs associated with development of methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of modeling information.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 28th day of April 2009.

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

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Enclosure 3

UAP-HF-09201
Docket Number 52-021

Third Responses to Request for Additional Information No. 197-1800 Revision 0

April 2009
(Proprietary Information Excluded)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

4/28/2009

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**

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.

ANSWER:

Please explain the methodology and results of the evaluation of induced steam generator tube ruptures, given failure to depressurize.

The NRC and MHI had a face-to-face meeting to discuss issues on temperature induced steam generator tube rupture and related problems from 17 to 19 February 2009. As the conclusion, the NRC staff requested MHI to further consider in the following two aspects:

(1) The overall SGTR issue needs to be addressed in more detail. The probability values used for SGTR need more explanation, specifically a discussion on the basis for the use of 1/3 probability for competing path determination assumption.

(2) Severe accident management would be more benefit from more cases being analyzed. For example, hot leg creep rupture case should be evaluated for more than 24 hours in order to provide adequate understanding event progression to support SAMG.

In order to address the aspect (1), MHI reviewed the technical report which evaluated the risks for steam generator tube integrity [1]. The study in the report involves detailed computational calculations employing specialized codes to evaluate several properties, such as STEIN code for calculation of burst probability versus tube differential pressure, HOTLEG code for overcoming the limitation of MAAP code in modeling heat transfer in the hot leg pipes, PROBFAIL code for calculation of contribution to SG tube rupture probability, etc. MHI does not own these specialized computational codes, thus it is practically impossible to perform such quantitative evaluation in the same level achieved in the report. Instead, MHI qualitatively analyzed the applicability of the assessment conclusion of the report to the US-APWR design.

Summary of reviewing EPRI technical report [1]

A flowchart of the methodology for calculating the probability of rupture in U-tube plants is shown in Figure 1, which is Figure 6-1 extracted from the report.

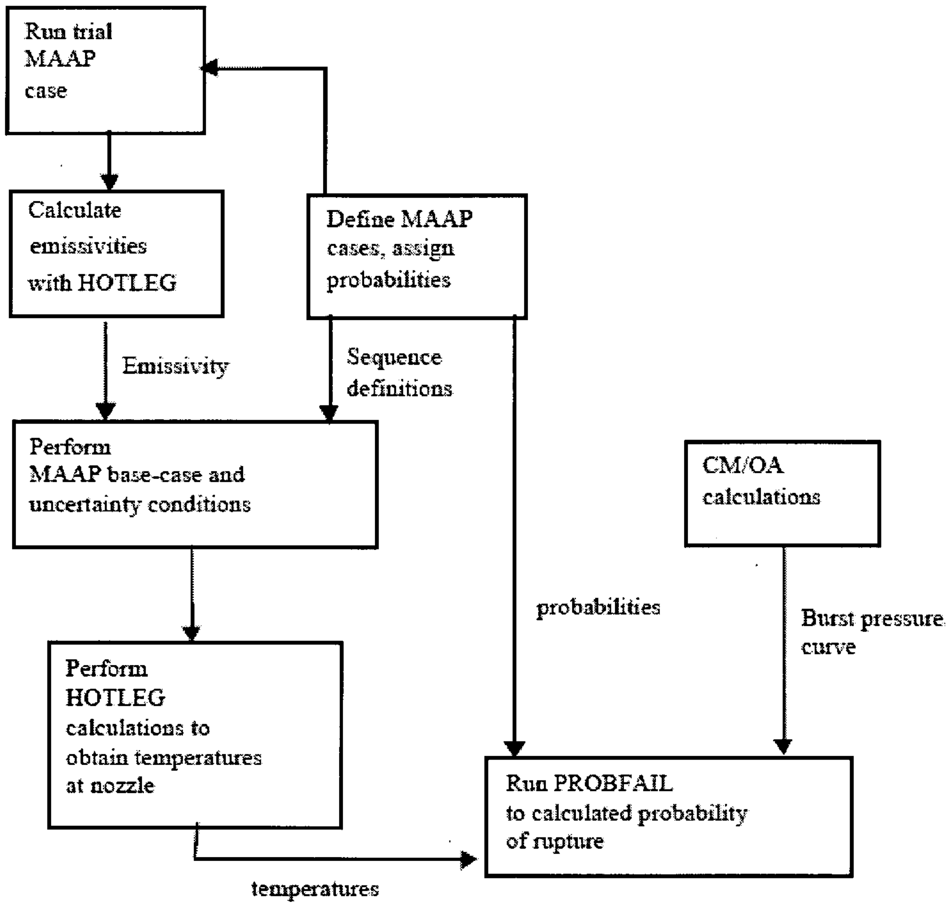

Figure 6-1
Methodology for Calculating Probability of Temperature-induced Rupture

Figure 1 Methodology for Calculating Probability of TISGTR

To illustrate the application of the methodology, several typical sequences were analyzed for a four-loop Westinghouse plant, including "high/dry/high" case, "high/dry/low" case with and without loop seal clearing.

A series of STEIN calculation were performed to estimate the probability of burst at 650F as a function of pressure, as the first step to perform the tube rupture calculations. The calculation result is shown in Figure 2 and summarized in Table 1, which are Figure A-4 and Table 6-3 extracted from the report, respectively. It can be observed that the probability of burst at 2560 psid is very small, and 2560 psi is approximately the RCS pressure during operations at power. This implies that pressure induced rupture is practically negligible except for ATWS events.

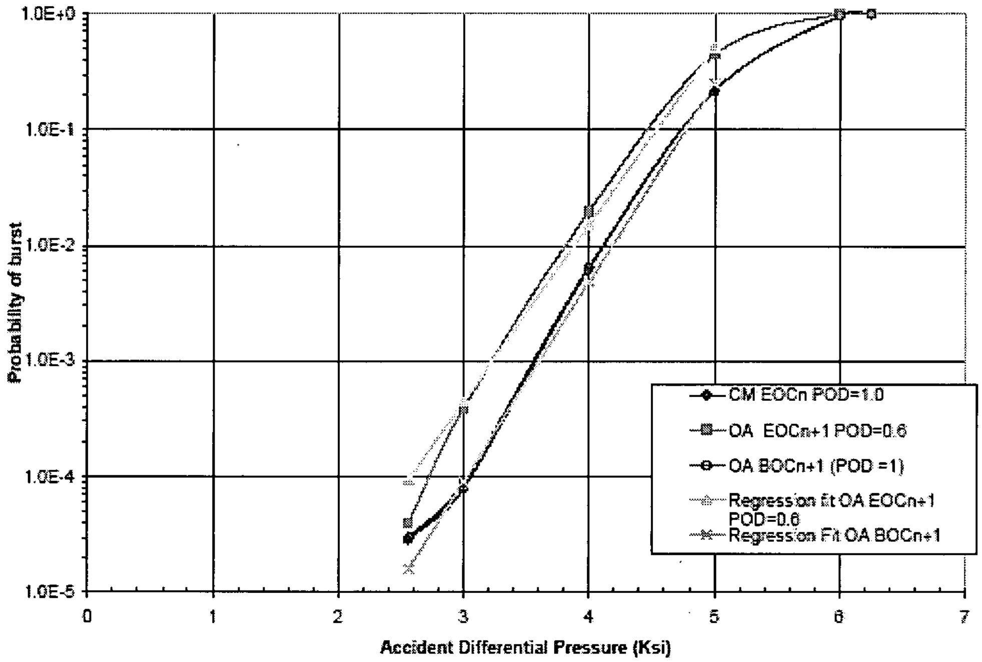

Figure A-4
Burst Probability Versus Accident Induced Differential Pressure Across SG Tubes

Figure 2 STEIN Calculation Result

Table 1 Summary of STEIN Calculation Result Shown in Figure 2

Table 6-3
Burst Pressure Values Entered in File st2_oa.dat for Performing the Illustrative Calculations

Differential pressure (psid)	Probability of burst
2560	4.E-5
3000	3.9E-4
4000	0.02
5000	0.45
6000	1.0

- *High/Dry/High Sequence*

Regarding the evaluation for high/dry/high sequence, the result is shown in Table 2, which is Table 6-4 of the report. It can be concluded that high/dry/high sequence in this plant apparently pose little risk of induced tube rupture and attention should instead be focused on other sequence variations that pose a greater risk of induced tube rupture.

Table 2 SG Tube Rupture Evaluation Result for High/dry/high Sequences

Table 6-4
Contents of PROBFAIL output file case1a.prf

Hot leg properties used: C-Mo Steel				
Sequence	**Seq.Prob.**	**Cprob(BSGrup)**	**Cprob(USGrup)**	**Rupture Prob.**
case1awrs	1.000	1.125E-12	5.520E-12	1.769E-11
Total probability of SG rupture is: 1.768574E-11				

- *High/Dry/low Sequence*

Study result for high/dry/low with one steam generator depressurized is shown in Table 3, which is Table 6-6 of the report. Contribution to overall SG tube rupture probability for all sequences is evaluated as 1.1%.

Table 3 SG Tube Rupture Evaluation Result for High/dry/low with One SG Depressurized

Table 6-6
Results of PROBFAIL Calculations for Base Case High/Dry/Low Sequence (B-loop Steam Generator)

Sequence	**Probability assigned to Sequence in Table 6-2 (percent)**	**Conditional probability of B-loop steam generator rupture in this sequence (percent)**	**Contribution to overall SG Rupture Probability (percent)**
Base case	40.8	0.55	0.18
High mixing	11.7	0.31	0.036
Low mixing	5.83	1.5	0.09
High radiation	13.6	0.1	0.01
Low radiation	13.6	4.1	0.56
Low SG/HL ratio	10.2	1.2	0.12
Best combination	3.88	0.07	0.003
Worst comb.	0.485	13.7	0.07
Total:	100		1.1

This evaluation is expanded to a sequence that all four steam generators are depressurized, and the PROBFAIL calculation resulted in a probability of tube rupture of 5.8%. This is roughly four times the value given in the calculation for a sequence with single SG depressurized, as expected.

- *Pump Seal LOCA Leading to Loop Seal Clearing*

Three cases are considered, including all steam generators pressurized, cleared loop steam generator depressurized, and uncleared loops steam generators depressurized.

<All SG Pressurized>

Evaluation result showed that tube rupture was not predicted to occur in any loop by either MAAP or PROBFAIL. Compared to the high/dry/low case, the higher relative tube temperatures that could occur due to loop seal clearing were not enough to make up for the lower tube differential pressure and the slower hot leg heat-up rates.

<SG in Cleared Loop Depressurized>

Evaluation result is shown in Table 4, which is Table 6-8 of the report.

Table 4 SG Tube Rupture Evaluation Result for SG in Cleared Loop Depressurized

Table 6-8
Results of Calculations with Forced Loop Seal Clearing and Depressurized Steam Generators in the Cleared Loop

Case	MAAP hot leg rupture time (secs)	MAAP steam generator rupture time (secs)	Conditional probability of tube rupture (percent)
Base	16400	14300	85
high radiation	16280	14160	79
low radiation	16540	14220	92
total			85

The probability of rupture in the affected loop SG is very high as expected, to lead to unfavorable results since the high temperature that develop in the cleared loop are applied to SG with a maximized tube differential pressure.

<SG in Uncleared Loop Depressurized>

Probability of SG tube failure is calculated assuming all the SGs in loops with filled (uncleared) loop seals are depressurized, and the cleared loop SG remains at pressure. Evaluation results are shown in Table 5 and 6, which are Table 6-9 and 6-10 of the report, respectively.

The aggregated probability of rupture is approximately 0.9%, which has contributions from both cleared broken-loop (high tube temperatures but low differential pressure and stress) and the uncleared unbroken-loops (lower tube temperatures but high differential pressure and stress).

Table 5 SG Tube Rupture Evaluation Result for Pressurized SG in Cleared Loop

Table 6-9
Results of Calculations with Forced Loop Seal Clearing and Depressurized Steam Generators in the 3 Uncleared Loops: Results for Cleared Loop with Pressurized Steam Generator

Case	MAAP cleared-loop hot leg rupture time (secs)	MAAP cleared-loop steam generator rupture time (secs)	Conditional probability of tube rupture in cleared loop (percent)
base	15160	18260	0.14
High radiation	14920	19250	0.08
Low radiation	15300	18200	0.16

Table 6 SG Tube Rupture Evaluation Result for Depressurized SGs in uncleared Loops

Table 6-10
Results of Calculations with Forced Loop Seal Clearing and Depressurized Steam Generators in the 3 Uncleared Loops: Results for Loops with Filled Loop Seals and Depressurized Steam Generator

Case	MAAP uncleared-loop hot leg rupture time (secs)	MAAP uncleared-loop steam generator rupture time (secs)	Conditional probability of tube rupture in uncleared loop (percent)
base	15900	17400	0.25
High radiation	15560	17740	0.11
Low radiation	16350	17020	0.41

- *Medium-to Large Pump Seal LOCA with No Loop Seal Clearing*

Assuming that pump seal LOCA does not result in loop seal clearing, for example seal LOCA equivalent to a leak rate of 180 gpm in all four loops, the calculation resulted in probability of tube failure of 0.3%. This is not very different from 1.1% result obtained in High/dry/low with one SG depressurized.

This sample evaluation demonstrated in the report was performed in an application to a Westinghouse four-loop plant, which equipped U-tube steam generators.

--- End of summary of EPRI technical report ---

The evaluation performed in the EPRI technical report are considered generally applicable for U-tube SG type PWR plants because the relationship between primary and secondary system examined in the report does not change for a unique plant. It should be therefore applicable to the US-APWR design as well.

The most significant risk contributor for TISGTR is evaluated as the cases of RCP seal LOCA with loop seal cleared and depressurized steam generator in the cleared loop; with having the total SG tube failure probability of 85%. This calculated value may vary depending on each specific plant, although the conclusion of risk significance should be maintained regardless of any plants. Therefore, this analysis results should be carefully addressed in the evaluation of Level 2 PRA for the US-APWR.

By reviewing the design of the US-APWR, it is concluded that the most risk significant scenario of RCP seal LOCA with loop seal cleared and SG depressurized in the cleared loop can be screened out and negligible because of the following discussions.

- RCP seal LOCA is an event basically resulting from station blackout (SBO) or loss of CCW. For this concerned scenario, loss of CCW is not important because emergency feedwater system is maintained functional as long as power is available, and it is possible to cool down primary system through SG. Therefore, only SBO is a contributor on this risk.

- For the US-APWR PRA, it is assumed that RCP seal LOCA occurs if SBO is not recovered within one hour. Emergency feedwater system is temporarily available through turbine driven pumps even if there is no ac power; however this works only several hours while battery is capable to supply power.

- After emergency feedwater system stops, primary and secondary system pressure gradually increases. When secondary system pressure reaches the set pressure of MSSV, spring type valve, it spontaneously opens and maintains the SG pressure below the set pressure. It is practically negligible to manually operate MSRV or MSDV under SBO condition. MSSV is therefore only the measures to depressurize SG.

- If MSSV is stuck open under SBO condition, then it is possible to result in a situation of the concerned scenario, RCP seal LOCA with loop seal cleared and depressurized SG in the cleared loop. The probability of this situation can be very roughly evaluated as:

  - CDF due to LOOP (f_1): 5.8×10^{-7}(/ry) (Reference [2])
  - Probability of stuck open MSSV (p_1): 6.8×10^{-5} (Reference [3])
  - Number of MSSV possible to open under this situation: 4
  - Probability that stuck open MSSV is at the loop where loop seal is cleared: 1/4
  - Frequency of concerned scenario (f_2): $f_1 \times p_1 \times 4 \times 1/4 = \underline{3.9\times10^{-11}\text{(/ry)}}$

If CCFP of this scenario is assumed to be one, the contribution to LRF is evaluated as f_2 above. It can be acceptable to ignore values with this magnitude of order, and accordingly the possibility of TISGTR from RCP seal LOCA is assumed to be negligible in the current Level 2 PRA. As evaluated in the EPRI report, high RCS pressure scenario, specifically high/dry/low case still remains as the risk contributor for occurrence of TISGTR.

Reviewing the evaluation result of current Level 2 PRA, the ACL "TEI" is evaluated to have the largest CDF among the high RCS pressure events. In this ACL, the sequences for secondary system depressurization (SLBO, SLBI, FWBL) occupies approximately 30% of total CDF. The PDS "9A", evaluated having the largest CDF among the high RCS pressure events, also involves approximately 30% of secondary system depressurization sequences. It is a reverse direction for verification, however this PRA result can be one of a basis for the use of 1/3 probability. CCFP of high/dry/low scenarios should be much less than one as evaluated in the EPRI technical report,

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thus this assumption is perhaps excessively conservative however acceptable as long as the safety goal for LRF is satisfied.

The issue of the aspect (2) has probably raised by the NRC staff because the analysis description in Chapter 14 of the PRA technical report [2], specifically the evaluations of the effectiveness of RCS depressurization (i.e. AM001 and AM002) are forced stopped at 1 hour after reactor vessel failure. MHI therefore extended the calculation time until 1 hour after containment failure in order to evaluate the effectiveness of accident management measures as well as the influence to the subsequent physical phenomena.

In addition, in order to evaluate the influence of assumption of zero break area, sequence AM002 was re-evaluated without depressurization by valve, instead hot leg creep rupture and SG tube creep rupture was considered (sequence AM002-2). The differences in the accident progression between with accident management operation and without operation were evaluated.

Table 7 Summary of Additional Accident Progression Analysis Results

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Also, provide the discussion of the results of scenarios where the steam generators are also depressurized.

The sequence of AP201 described in Chapter 14 of the PRA technical report [2] is exactly the case to consider high/dry/low scenario. Very similar question was asked in Question #19-304 and MHI answered with additional discussions on possibility of hot leg rupture or SG tube rupture in the letter to the NRC [5].

In addition, please justify the assumption of zero break areas for hot leg creep rupture and temperature-induced steam generator tube rupture.

Regarding the assumption of zero break areas considered in the severe accident progression analysis in Chapter 14 of the PRA technical report [2], the assumption is applied in order to model specific accidental conditions, especially the conditions with high RCS pressure. For example, in order to evaluate the consequences of high pressure melt ejection from reactor vessel failure, or in another way, in order to consider the effectiveness of RCS depressurization valve provided as accident management measures, in these cases it is more favorable RCS is not depressurized due to hot leg creep rupture or TISGTR before RV failure or opening the depressurization valve. Hence, the RCS break model is manipulated not to depressurize before RV failure or the operation of opening RCS depressurization valve for convenience sake. It is still possible to evaluate the expected timing when hot leg creep rupture or TISGTR occurs from this model as the output of the code calculation, then this evaluated RCS break timing, due to either hot leg or SG tube, can be appropriately incorporated into the development of SAMG, for determination of necessary completion time, etc.

Realistic evaluation results considering hot leg rupture and TISGTR were provided to the NRC as the supplemental information within the MHI's letter [4]. In that evaluation, hydrogen release behavior is specifically discussed because hydrogen is one of the most significant risk contributors during severe accident. It is concluded from that evaluation that hydrogen can be properly controlled in any cases with or without the zero break area assumption, and accordingly this zero area assumption is considered acceptable. Additional discussions on severe accident progression for high RCS pressure scenarios with RCS depressurization through operating depressurization valve (AM002) and either hot leg rupture or SG tube rupture (AM002-2) are also provided before in this answer.

The occurrence possibility of hot leg rupture and TISGTR is also discussed in the answer for the RAI#19-304, provided to the NRC within the MHI's letter [5], and concluded that hot leg rupture is likely to occur before TISGTR according to the evaluation results utilizing MAAP. This may be a characteristic tendency of MAAP calculation on this aspect. Apart from the evaluation result by MAAP, MHI has comprehensively addressed various research and evaluations on TISGTR as discussed previously in this answer, and concluded the occurrence of these events should be treated in a probabilistic approach and maintaining the assumption taken in the DCD as 1/3 failure fraction for each of event in the high RCS pressure sequences.

References:

[1] Steam Generator Tube Integrity Risk Assessment, EPRI Technical Report TR-107623, March 2002

[2] US-APWR Probabilistic Risk Assessment, MUAP-07030 Rev.1, Mitsubishi Heavy Industries, September 2008

[3] Industry-Averaged Performance for Components and Initiating Event at U.S. commercial Nuclear Power Plants, NUREG/CR-6928, February 2007

[4] Letter from Mitsubishi Heavy Industries to the U.S. Nuclear Regulatory Commission, "Supplemental Information to US-APWR DCD RAI No. 92-1237 Revision 0" dated 24 April 2009 (MHI Ref: UAP-HF-09137)

[5] Letter from Mitsubishi Heavy Industries to the U.S. Nuclear Regulatory Commission, "MHI's Second Responses to US-APWR DCD RAI No. 92-1237 Revision 0" dated 10 April 2009 (MHI Ref: UAP-HF-09163)

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 current Level 2 PRA evaluation result from this RAI. PRA technical report [2] will be revised to address the discussion especially on TISGTR in the next revision.