

  
**MITSUBISHI HEAVY INDUSTRIES, LTD.**  
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TOKYO, JAPAN

June 3, 2010

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-11168

**Subject: MHI's Responses to US-APWR DCD RAI No. 752-5614 Revision 0 (SRP 19)**

**Reference:** 1) "Request for Additional Information No. 752-6514 Revision 0, SRP Section: 19 – Probabilistic Risk Assessment and Severe Accident Evaluation, Application Section: SRP Chapter 19," dated May 3, 2011.

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

Enclosed is the response to one RAI contained within Reference 1. Of these RAIs, three questions #19-520, #19-521 and #19-522 will not be answered within this package. These questions require additional time for internal discussions and computations, and will be answered by 2<sup>nd</sup> July 2011.

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,



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

Enclosures:

1. Responses to Request for Additional Information No. 752-5614 Revision 0

D081  
NRQ

CC: J. A. Ciocco  
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Docket No. 52-021  
MHI Ref: UAP-HF-11168

Enclosure 1

UAP-HF-11168  
Docket Number 52-021

Responses to Request for Additional Information No. 752-5614  
Revision 0

June 2011

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**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION**

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6/3/2011

**US-APWR Design Certification**

**Mitsubishi Heavy Industries**

**Docket No.52-021**

**RAI NO.:** NO. 752-5614 REVISION 0  
**SRP SECTION:** 19 – Probabilistic Risk Assessment and Severe Accident Evaluation  
**APPLICATION SECTION:** SRP Chapter 19  
**DATE OF RAI ISSUE:** 5/3/2011

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**QUESTION NO.: 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 ICIS 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 ICIS 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.

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**ANSWER:**

Sensitivity for the APET is evaluated considering the parameter combinations in accordance with the suggestion by the NRC staff, and the results are shown in Table 19-523-1 and 19-523-2 for the APET base case and for the APET sensitivity case, respectively.

Table 19-523-1 Sensitivity for the APET Base Case

	RCS depressurized due to ICIS tube release	No concurrent loop seal and core barrel clearing		CCFP	Delta from base case
		No T/D EFW	T/D EFW		
base	0.5	0.9	0.99	1.36E-2	-
(1)	0.5	0.5	0.99	1.36E-2	0.0%
(2)	0.5	0.5	0.5	1.52E-2	11.9%
(3)	0.5	0.9	0.5	1.52E-2	11.8%
(4)	0.0	0.9	0.99	2.71E-2	100.0%
(5)	0.0	0.5	0.99	2.71E-2	100.1%
(6)	0.0	0.5	0.5	3.04E-2	123.8%
(7)	0.0	0.9	0.5	3.03E-2	123.7%

Table 19-523-2 Sensitivity for the APET Sensitivity Case

	RCS depressurized due to ICIS tube release	No concurrent loop seal and core barrel clearing		CCFP	Delta from base case
		No T/D EFW	T/D EFW		
base	0.5	0.9	0.99	1.65E-2	-
(1)	0.5	0.5	0.99	1.65E-2	0.0%
(2)	0.5	0.5	0.5	1.74E-2	5.1%
(3)	0.5	0.9	0.5	1.74E-2	5.1%
(4)	0.0	0.9	0.99	3.30E-2	100.0%
(5)	0.0	0.5	0.99	3.30E-2	100.0%
(6)	0.0	0.5	0.5	3.47E-2	110.3%
(7)	0.0	0.9	0.5	3.47E-2	110.2%

It can be observed from this study that the sensitivity of the RCS depressurization through ICIS tube failure shows significant influence to the APET, and the CCFP with no RCS depressurization is double that of the base case. This is reasonable because the high RCS pressure scenario has significant contribution to the possibility of TISGTR, and the challenge of the low RCS pressure scenario to TISGTR is considered negligible. Sensitivity of the concurrent loop seal and core barrel clearing with no T/D EFW is negligibly small. It is because the contribution of no T/D EFW scenario path is evaluated as small as 0.5% of the total SBO sequence, as answered to the Question 19-450. On the other hand, the sensitivity of the concurrent loop seal and core barrel clearing with T/D EFW shows substantial influence.

The results shown in the answer to Question 19-450 were calculated based on the DCD Rev. 2 PRA evaluation results; therefore, they do not reflect the latest design activities. Table 19-523-3 shows the difference in PRA evaluation results between the DCD Rev. 2 with the APET base case, and DCD Rev. 3. DCD Rev. 3 PRA results considering the above APET sensitivity case are shown in Table 19-523-4.

Table 19-523-3 Comparison between DCD Rev. 2 and Rev. 3 Results

	PRA Evaluation Result DCD Rev. 2 with APET Base Case			PRA Evaluation Result DCD Rev. 3		
	CDF	LRF	CCFP	CDF	LRF	CCFP
Internal at power	1.03E-06	1.03E-07	0.100	1.03E-06	1.07E-07	0.103
Fire	1.77E-06	2.44E-07	0.138	8.60E-07	1.87E-07	0.217
Flood	1.36E-06	2.79E-07	0.205	8.91E-07	1.56E-07	0.175
Total	4.17E-06	6.27E-07	0.150	2.78E-06	4.49E-07	0.162

Table 19-523-4 Comparison between DCD Rev. 3 and DCD Rev. 3 with APET Sensitivity Case Results

	CDF	DCD Rev. 3		DCD Rev. 3 with APET Sensitivity Case	
		LRF	CCFP	LRF	CCFP
Internal at power	1.03E-06	1.07E-07	0.103	1.08E-07	0.104
Fire	8.60E-07	1.87E-07	0.217	1.88E-07	0.218
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175
Total	2.78E-06	4.49E-07	0.162	4.51E-07	0.162

By applying the sensitivity evaluation results shown in Table 19-523-1 and 19-523-2 to DCD Rev. 3 evaluation results, the CCFP, LRF and delta LRF results for both the base-case and the sensitivity case are calculated as shown in Tables 19-523-4 to 19-523-9. It has been confirmed that the sensitivity of these parameters to the total LRF is limited and not very significant.

Table 19-523-4 LRF and CCFP Sensitivity for Case (2) and (3) (APET Base Case)

	CDF	DCD Rev. 3		LRF and CCFP Sensitivity Case (2) and (3)		Delta in LRF
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.07E-07	0.103	1.07E-07	0.104	0.51%
Fire	8.60E-07	1.87E-07	0.217	1.87E-07	0.218	0.26%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	0.00%
Total	2.78E-06	4.49E-07	0.162	4.50E-07	0.162	0.23%

Table 19-523-5 LRF and CCFP Sensitivity for Case (4) and (5) (APET Base Case)

	CDF	DCD Rev. 3		LRF and CCFP Sensitivity Case (4) and (5)		Delta in LRF
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.07E-07	0.103	1.11E-07	0.108	4.27%
Fire	8.60E-07	1.87E-07	0.217	1.91E-07	0.222	2.20%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	0.01%
Total	2.78E-06	4.49E-07	0.162	4.58E-07	0.165	1.93%

Table 19-523-6 LRF and CCFP Sensitivity for Case (6) and (7) (APET Base Case)

	CDF	DCD Rev. 3		LRF and CCFP Sensitivity Case (6) and (7)		Delta in LRF
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.07E-07	0.103	1.12E-07	0.109	5.31%
Fire	8.60E-07	1.87E-07	0.217	1.92E-07	0.223	2.70%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	0.02%
Total	2.78E-06	4.49E-07	0.162	4.60E-07	0.165	2.39%



Table 19-523-7 LRF and CCFP Sensitivity for Case (2) and (3) (APET Sensitivity Case)

	CDF	DCD Rev. 3 with APET Sensitivity Case		LRF and CCFP Sensitivity Case (2) and (3)		Delta in LRF
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.08E-07	0.104	1.08E-07	0.105	0.28%
Fire	8.60E-07	1.88E-07	0.218	1.88E-07	0.219	0.15%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	0.00%
Total	2.78E-06	4.51E-07	0.162	4.52E-07	0.162	0.13%

Table 19-523-8 LRF and CCFP Sensitivity for Case (4) and (5) (APET Sensitivity Case)

	CDF	DCD Rev. 3 with APET Sensitivity Case		LRF and CCFP Sensitivity Case (4) and (5)		Delta in LRF
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.08E-07	0.104	1.13E-07	0.110	5.17%
Fire	8.60E-07	1.88E-07	0.218	1.93E-07	0.224	2.67%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	0.02%
Total	2.78E-06	4.51E-07	0.162	4.62E-07	0.166	2.35%

Table 19-523-9 LRF and CCFP Sensitivity for Case (6) and (7) (APET Sensitivity Case)

	CDF	DCD Rev. 3 with APET Sensitivity Case		LRF and CCFP Sensitivity Case (6) and (7)		Delta in LRF
		LRF	CCFP	LRF	CCFP	
Internal at power	1.03E-06	1.08E-07	0.104	1.14E-07	0.110	5.70%
Fire	8.60E-07	1.88E-07	0.218	1.93E-07	0.225	2.95%
Flood	8.91E-07	1.56E-07	0.175	1.56E-07	0.175	0.02%
Total	2.78E-06	4.51E-07	0.162	4.63E-07	0.166	2.59%

**Impact on DCD**

There is no impact on the DCD

**Impact on R-COLA**

There is no impact on the R-COLA

**Impact on S-COLA**

There is no impact on the S-COLA

**Impact on PRA**

There is no impact on the PRA