MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

September 22, 2011

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-11323

Subject: MHI's Response to US-APWR DCD RAI No. 808-5921 Revision 3 (SRP 15.6.3)

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Response to US-APWR DCD RAI No. 808-5921 Revision 3 (SRP 15.6.3)". The enclosed material provides MHI's response to the NRC's "Request for Additional Information (RAI) 808-5921 Revision 3," dated August 22, 2011.

As indicated in the enclosed materials, Enclosure 2 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 in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]".

This letter includes a copy of the proprietary version of the RAI response (Enclosure 2), a copy of the non-proprietary version of the RAI response (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all material designated as "Proprietary" in Enclosure 2 be withheld from 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 this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

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

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Response to US-APWR DCD RAI No. 808-5921 Revision 3 (SRP 15.6.3) (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 808-5921 Revision 3 (SRP 15.6.3) (non-proprietary)

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

C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11323

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and 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 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 "MHI's Response to US-APWR DCD RAI No. 808-5921 Revision 3 (SRP 15.6.3)", dated September 22, 2011, and have determined that the document contains 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 information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The basis for holding the referenced information confidential is that it describes the unique design of the safety analysis, developed by MHI (the "MHI Information").

4. The MHI Information is not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years. Therefore public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

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 22nd day of September, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11323
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 808-5921 Revision 3
(SRP 15.6.3)

September, 2011
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 808-5921 REVISION 3**
SRP SECTION:	**15.06.03 - RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE (PWR) 07/1981**
APPLICATION SECTION:	**15.6.3**
DATE OF RAI ISSUE:	**8/22/2011**

QUESTION NO.: 15.06.03-3

The design basis analysis of the steam generator tube rupture (SGTR) event, described in DCD Section 15.6.3, assumed a series of operator actions as a part of mitigating and recovery procedures to stabilize the plant in a timely manner to terminate the primary-to-secondary break flow, and minimize contamination of the secondary system and the release of radioactivity to the atmosphere. It is also stated that the operator actions for SGTR recovery are proceduralized in the emergency operating procedures (EOP). However, the plant-specific EOPs are not yet available as they will be developed based on the emergency response guidelines (ERG). The staff will review the ERGs to assure that the design basis operator action assumptions in the safety analysis are consistent with the ERG-specified steps and bound the operator response times.

(a) Provide the US-APWR ERG for the SGTR event.

(b) Provide an evaluation of the operator actions and completion times credited in the SGTR analysis. The evaluation should consider each operator action for consistency with the corresponding steps in the ERG, the alarms and indications of event symptoms, quality and adequacy of diagnostic instrumentation, and training which would enable the operators to properly interpret the symptoms to take proper action within the assumed operator response time. The operator actions assumed in the SGTR analysis for the radiological dose evaluation case include:

- Manual reactor trip and main feedwater (MFW) isolation at 15 minutes
- Identification and isolation (main steam isolation valve closure) of the ruptured SG at 20 minutes
- Opening of intact SG main steam depressurization valves (MSDV) at 25 minutes
- Opening of pressurizer safety depressurization valve (SDV) at ~45.3 minutes
- Closure of SDV at ~47.5 minutes
- Manual termination of ECCS at 48 minutes

ANSWER:

As described in the response to RAI 297-2287 Question 15.0.0-12 (MHI letter UAP-HF-09340, dated July 3, 2009), MHI is currently developing Emergency Response Guidelines (ERGs) for the US-APWR for the purpose of supporting plant-specific Emergency Operating Procedures (EOPs). The development of ERGs is expected to result in the US-APWR ERG Revision 0 at the end of December 2011.

As discussed in a conference call with the NRC on August 10, 2011, MHI proposes to present the

Revision 0 ERG documents of interest in a closed audit sometime after January 2012 to confirm that the design basis operator action assumptions in the safety analysis are consistent with the ERG-specified steps and bound the operator response times.

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.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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APPLICATION SECTION:	**15.6.3**
DATE OF RAI ISSUE:	**8/22/2011**

QUESTION NO.: 15.06.03-4

In the analyses of the SGTR for both the radiological dose evaluation and margin to overfill cases, the limiting single failure is assumed to be the loss of one emergency feedwater system (EFWS) train.

Provide an evaluation, which shows the loss of one EFWS train being the limiting single failure. The evaluation should be done for both cases, and determine whether other single failures can give worse results, e.g., failure of an intact SG MSRV to open, failure of EFW control valve to the ruptured SG to close, failure of MFW control valve to the ruptured SG to close, failure of the MSIV to the ruptured SG to close, failure of the pressurizer SDV to open on manual demand, or failure of high pressure safety injection (HPSI) pump switches when SI is to be terminated.

ANSWER:

The MHI response to RAI 297-2287 Question 15.0.0-11 (MHI letter UAP-HF-09340, dated July 3, 2009) provided a detailed discussion of the systematic event-specific review performed to determine the limiting single failure for each Chapter 15 event. Table 15.0.0-11.1 included in that RAI response provides an event-specific tabular summary of the mitigative systems assumed in the Chapter 15 Safety Analysis. This table indicates that the following systems are used to mitigate the effects of an SGTR: RTS, EFWS, EFWIV, MSRV, MSDV, SDV, and GTG. Note that ECCS is assumed to operate in order to conservatively increase primary-to-secondary leakage, but is not considered as a mitigative system.

The following system and component failures were considered by MHI as part of the evaluation that determined the limiting single failure for an SGTR for both the radiological dose evaluation and the margin to overfill cases.

RTS:	A single failure of the RTS has no impact due to the redundant RTS trains.
EFWS:	A single failure of this system is assumed in the DCD 15.6.3 analysis.
EFWIV:	EFWIV valve closure is assumed in the analysis. The EFWIV design is a redundant valve configuration, which includes two EFWIVs in series. The closure of either valve is sufficient to ensure EFW isolation. Therefore, there is no impact of a single failure.
MSRV:	There is one air-operated MSRV installed on the main steam line from each SG. No credit is taken for these valves to perform a mitigative function. Therefore, it is not necessary to assume a single failure of an MSRV. However, for the

radiological dose analysis, an additional failure of a stuck open MSRV in the ruptured SG is assumed to ensure the resulting doses are conservative. Note that the stuck open MSRV is eventually isolated by automatic closure of the MSRV block valve on low main steam line pressure.

MSDV: There is one motor-operated MSDV installed on the main steam line from each SG. The MSDVs of the intact SGs are assumed to be opened by the operators in the analysis. Although not described in the DCD, the MSDV on the intact SG with no EFW (due to assumed single failure in EFWS described above) is assumed to be inoperable as an additional failure. If an inoperable MSDV on an intact SG with EFW is assumed as a single failure instead, then there is no need to postulate the previous EFW pump failure. The results of this scenario are less limiting than the current DCD analysis result. Also, the US-APWR design includes MSRVs that can be manually opened as backup equipment for the MSDV. It is also necessary for the operators to close the MSDVs after using them. If a MSDV can not be closed, it can be isolated by the block valve which is in series with the MSDV. Therefore, there is no impact of a single failure of the MSDV to close.

SDV: One pressurizer SDV is assumed to be opened and closed by manual action. The pressurizer has two redundant SDVs and each SDV has a block valve. Therefore, there is no impact of the single failure of one SDV to either open or close.

GTG: Assuming one GTG fails results in the associated EFW and ECCS pump not starting. One EFW pump not starting is already covered by the assumed EFWS single failure described above. One SI pump not starting actually reduces the primary-to-secondary leakage and thus it is more conservative to assume the SI pump remains running. Therefore, a single failure of a GTG is bounded by the single failure of the EFWS assumed in the DCD.

MSIV: The MSIV is a normally open, system medium actuated gate type valve using system internal pressure for valve closure. The valve will be closed when the main steam pressure is provided to push the valve body downward, therefore, the valve can be considered as passive in nature. The system pressure is normally isolated by solenoid valves attached to the pressure supply path in order to maintain the valve open position. No single failure is assumed because the solenoid valves are redundant with different power supplies and control trains.

MFIV: The MFIV is a normally open, system medium actuated gate type valve using system internal pressure for valve closure. The valve will be closed when the main feedwater pressure is provided to push the valve body downward, therefore, the valve can be considered as passive in nature. The system pressure is normally isolated by solenoid valves attached to the pressure supply path in order to maintain the valve open position. No single failure is assumed because the solenoid valves are redundant with different power supplies and control trains.

MSSV: No single failure is assumed due to the passive nature (spring-loaded code safety valve) of this mitigative component.

ECCS As described above, SI is not a mitigative system for this event, but actually increases primary-to-secondary leakage. So assuming the failure of one ECCS train is less limiting than assuming all ECCS trains are running. Therefore, a single failure of one ECCS train is not assumed. However, termination of SI is a mitigative operator action credited in the analysis. This action is performed using the SI pump switches to stop the pumps. There are redundant HSI means to stop the pumps in that the pumps can be stopped from either the operational or safety VDUs. In addition, if an SI pump can not be stopped, SI flow can be terminated by closing the DVI isolation valves or hot leg injection isolation valves. Finally, an SI pump could be stopped by opening the incoming breaker for the Class 1E 6.9 kV bus to terminate power to the SI pump (this would also result in the loss of one EFWS train, but that single failure is already assumed in the DCD analysis). Therefore, the single failure of an SI pump switch to stop an SI pump has no impact.

As a result of the evaluation briefly summarized above, MHI concluded that the limiting single failure for an SGTR is the loss of one emergency feedwater system (EFWS) train.

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.

This completes MHI's response to the NRC's question.

9/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 808-5921 REVISION 3

SRP SECTION: 15.06.03 - RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE (PWR) 07/1981

APPLICATION SECTION: 15.6.3

DATE OF RAI ISSUE: 8/22/2011

QUESTION NO.: 15.06.03-5

Regarding the input assumptions in the analysis of the SGTR event,

(a) Explain why the initial reactor coolant temperature is assumed to be 4°F above the nominal value for the radiological dose evaluation case, and 4°F below the nominal value for the overfill margin case.

(b) Explain why the MFW control system is not credited in the SG overfill margin case, but is assumed in the radiological dose evaluation case.

ANSWER:

(a)

The uncertainty of the initial RCS temperature has been selected independently for each case in order to provide the most severe results based on sensitivity analyses. For the radiological dose evaluation, the reactor coolant temperature uncertainty is applied in the positive direction in order to increase the RCS temperature resulting in higher temperature of the primary-to-secondary leakage. This leakage is then easier to vaporize, ultimately resulting in an increase in the amount of vapor released from the secondary side. This is conservative from a radiological dose standpoint. For the overfill evaluation on the other hand, the uncertainty is applied in the negative direction in order to reduce the amount of vaporized leakage, which results in an increase in water level inside the steam generator (SG). This is more conservative for the margin to overfill of the ruptured SG.

(b)

Since the ruptured SG water level is increasing, the MFW control system would automatically reduce the feedwater flow. Hence, automatic MFW control results in a lower steam generator water level, which is not conservative for the ruptured SG overfill evaluation. Therefore, the MFW control system is not assumed for the ruptured SG overfill case. On the other hand for the radiological dose evaluation case, assuming the feedwater control system keeps the SG water level lower. This prevents the level from reaching the high-high SG water level reactor trip setpoint and delays the time of reactor trip until the operators manually trip the reactor. The delay time results in additional primary-to-secondary leakage, which is conservative for the radiological evaluation. Therefore the MFW control system is assumed for the radiological dose evaluation case.

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.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

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APPLICATION SECTION:	**15.6.3**
DATE OF RAI ISSUE:	**8/22/2011**

QUESTION NO.: 15.06.03-6

Clarify the discussion in Section 15.6.3.4.3 regarding the results of the radiological dose evaluation case.

(a) It indicates that the EFW flow is not provided for the ruptured SG since the MSIV closed before the EFW initiation. However, though the EFWS is designed with the capability to automatically terminate EFW flow to prevent overfilling of the SGs, the EFW isolation is not based on an MSIV closure. According to the US-APWR technical specifications, the EFW isolation can be initiated by high SG water level, coincident with reactor trip and no low main steam line pressure.

Clarify whether the MSIV closure or other engineered safety actuation function prevents the EFW flow to the ruptured SG.

(b) Figure 15.6.3-5 indicates the SG pressure increasing and exceeding the MSSV setpoint at about 900 seconds.

Explain why the sequence of event in Table 15.6.3-1 does not indicate the MSSV opening.

ANSWER:

(a)

The isolation of the EFW in the safety analysis is credited as an operator action and not an automatic isolation function. The actuation of EFW for the dose evaluation case is assumed to occur 140 seconds after the SI signal, which is initiated after the isolation of the ruptured SG. Therefore, the EFW flow to the ruptured SG is already manually isolated when the EFW pump is actuated. MHI will modify the description of the SG isolation operator action to clarify that the EFW isolation is an operator action rather than an automatic function in the safety analysis.

Note: the automatic closure of the EFWIV is actuated by two different signals.

a) High SG water level, coincident with reactor trip and no low main steam line pressure signal (described in "b" below).
b) Low main steam line pressure (not lead/lag compensated) and no other loop EFWIV is closed.

(b)

Since the SGTR is not a limiting event for maximum RCS or secondary pressure, the opening of the MSSVs was not originally included in the sequence of events table. However, DCD Table 15.6.3-1 will be revised to indicate that the MSSVs open.

Impact on DCD

DCD Subsection 15.6.3.4.3 Results, 1. Radiological Dose Evaluation Case, Item c is revised as indicated in Attachment 1.

DCD Table 15.6.3-1 is revised to add the time the MSSVs open as indicated in Attachment 1.

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.

This completes MHI's response to the NRC's question.

c. Isolating the ruptured steam generator

The main steam isolation valve (MSIV) is closed 1200 seconds after SGTR initiation. ~~(Therefore, EFW flow is not provided for the ruptured steam generator since the MSIV closed before the EFW initiated).~~ Emergency feedwater flow to the ruptured steam generator is also isolated by operator action at this time. DCD_15.06.03-6

d. Reducing the RCS temperature

To reduce the RCS temperature, the MSDVs of the intact steam generators are opened by operators 1500 seconds after SGTR initiation to reduce the RCS temperature.

At the same time, the MSRV on the ruptured steam generator is assumed to fail open. This failure causes the ruptured steam generator to rapidly depressurize, resulting in an increase in primary-to-secondary leakage and energy transfer. The MSRV on the ruptured steam generator is automatically isolated at 1826 seconds when the associated block valve is closed by the low main steam line pressure signal.

e. Depressurizing the RCS

Operators reduce the RCS pressure by opening a SDV at 2717 seconds until the primary-to-secondary pressure balance is attained.

f. Terminating the ECCS

After the successful establishment of a secondary heat sink, adequate subcooling margin, and RCS depressurization, SI is no longer needed and should be stopped to prevent repressurization of the primary system.

All the requirements for the termination of SI described in Section 15.6.3.4.2 are met and SI termination occurs 2880 seconds into the transient. After SI termination, the RCS pressure decreases as shown in Figure 15.6.3-1. Figure 15.6.3-7 shows that the primary-to-secondary leak flow continues after the SI is stopped until the pressures of the RCS and the ruptured steam generator equalize, which occurs at 4183 seconds.

The water volume in the ruptured steam generator as a function of time is shown in Figure 15.6.3-6. It can be seen that the water volume in the ruptured steam generator is 3030 ft^3 when the break flow stops, which is significantly less than the total steam generator volume of 7220 ft^3. Thus, the steam generator does not overfill. Radiological calculations were performed based on parameters shown in Figures 15.6.3-8 through 15.6.3-10.

2. Steam Generator Overfill Case

In the steam generator overfill case, no credit is taken for the feedwater control system, which causes the increase of the water level in the ruptured steam generator. Therefore, a reactor trip is automatically initiated by the high-high steam generator water level signal. The main feedwater system is also isolated by the high-high steam generator water level

Table 15.6.3-1
Time Sequence of Events for Steam Generator Tube Rupture - Radiological Dose Evaluation Input Analysis

Event	Time (sec)
SG tube rupture	0
Manual reactor trip (rod motion begins) and loss of offsite power	900
Reactor coolant pumps trip	900
Main feedwater isolation	900
Turbine trip	900
Main steam safety valves open	912
Ruptured steam generator isolated (MSIV closed)	1200
Intact SGs MSDV open (initiation of RCS cooling)	1500
Ruptured SG MSRV fails open	1500
ECCS initiated	1634
EFW pumps actuated	1774
Ruptured SG MSRV block valve closed	1826
SDV open	2717
SDV closed	2848
ECCS terminated (by operator)	2880
Primary leakage terminated	4183

DCD_15.06.03-6

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/22/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

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APPLICATION SECTION:	**15.6.3**
DATE OF RAI ISSUE:	**8/22/2011**

QUESTION NO.: 15.06.03-7

One of key parameters affecting the results of the SG margin to overfill (MTO) analysis during an SGTR event is the initial SG water level. Section 15.6.3.4.2 states that the initial water level is assumed at its expected nominal programmed level with positive uncertainty applied.

(a) What is the value of the expected nominal programmed SG water level? Provide the basis to determine that bounding initial water level has been used in the MTO analysis.

(b) What is the value of uncertainty applied to the MTO analysis? Describe the analysis which determines the uncertainty value. The information should also include a discussion of the effect of potential turbine runback on the SG level assumed in the MTO analysis.

ANSWER:

(a)

During normal operations, the feedwater control system maintains the steam generator water level at its nominal programmed level of approximately [ ] The initial water level in the ruptured steam generator is assumed to be [ ] In this case, the uncertainty is applied to the water level in the positive direction since it reduces the margin to overfill. The initial water level in the intact steam generators is assumed to be [ ] In this case, the uncertainty is applied in the negative direction to conservatively reduce the heat removal capability of the intact steam generators.

(b)

The values described in Item (a) above are calculated based on the [ ] Although a turbine runback has the potential to decrease the reactor power and increase the water level in the steam generator, the impact of the turbine runback is small and adequately bounded by the other conservatisms already included in the analysis assumptions and input parameters. MHI performed a sensitivity study to demonstrate the limited impact of the turbine runback on SG water level. The ΔT margin between the turbine runback setpoint and measured ΔT for each loop are shown in Figure 15.06.03-7.1 through 15.06.03-7.4. The turbine runback setpoint is 3% above the ΔT trip set point as indicated in the figures. These figures show that a turbine runback does not occur during the SGTR transient since the measured ΔT does not reach the turbine runback setpoint as shown in the figures.

In addition, MHI performed a sensitivity analysis to the initial SG water level. The sensitivity case shown in Figures 15.06.03-7.5 and 15.06.03-7.6 assumes that the initial SG water level (in all SGs) is at the nominal value without uncertainty. As shown in the figures, the reactor is tripped on the high-high SG water level reactor trip due to the ruptured SG which causes the water inventories of the faulted SG for both cases to be almost the same at the time the reactor trip occurs. Since the increase in the SG water level after the reactor trip is mainly determined by the amount of the leakage from the RCS, the eventual maximum SG water level is almost the same as the value for the DCD case. Therefore, the initial SG water level is not a key parameter for the MTO analysis.

Figure 15.06.03-7.1 ΔT margin (LOOP-A) versus Time
Steam Generator Tube Rupture Event
-SG Overfill Analysis Turbine Runback Sensitivity

Figure 15.06.03-7.2 ΔT margin (LOOP-B) versus Time
Steam Generator Tube Rupture Event
-SG Overfill Analysis Turbine Runback Sensitivity

Figure 15.06.03-7.3 Δ T margin (LOOP-C) versus Time
Steam Generator Tube Rupture Event
-SG Overfill Analysis Turbine Runback Sensitivity

Figure 15.06.03-7.4 Δ T margin (LOOP-D) versus Time
Steam Generator Tube Rupture Event
-SG Overfill Analysis Turbine Runback Sensitivity

Figure 15.06.03-7.5 Pressure versus Time
Steam Generator Tube Rupture Event
-SG Overfill Analysis Initial SG Water Level Sensitivity

Figure 15.06.03-7.6 Steam Generator Water Volume versus Time
Steam Generator Tube Rupture Event
-SG Overfill Analysis Initial SG Water Level Sensitivity

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.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/22/2011

US-APWR Design Certification

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APPLICATION SECTION:	**15.6.3**
DATE OF RAI ISSUE:	**8/22/2011**

QUESTION NO.: 15.06.03-8

List systems, components and instruments that are credited in the SGTR analysis. Discuss whether each system and component is safety grade. For non-safety grade systems and components, discuss whether safety grade backups are available which can be expected to function, or justify that non-safety grade systems and components can be used for the SGTR analysis.

ANSWER:

Table 15.06.03-8.1 provides a list of the systems, components, and instrumentation which are credited for mitigation in the safety analysis of an SGTR. All of the primary systems and components credited for the mitigation of the event are safety grade. In addition, most of the backup components are also safety grade. Instrumentation such as the radiation alarms and monitors that may be used to detect the event and identify the ruptured SG are non-safety grade. However, other safety grade instrumentation such as SG water level indication can be used to detect the event and identify the ruptured SG as backup to the radiation alarms and monitors.

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.

This completes MHI's response to the NRC's question.

Table 15.06.03-8.1 Systems, Components, and Instrumentation Credited for SGTR Mitigation (Sheet 1 of 3)

Item	Function in SGTR Analysis	Safety Grade Status	Available Backups
RTS	Trip the reactor	Safety Grade	N/A
EFWS	Supply EFW to SGs	Safety Grade	N/A
EFWIV	Close to isolate ruptured SG	Safety Grade	N/A
MSRV/associated block valve	Close to isolate ruptured SG (*1)	Safety Grade	N/A
MSDV	1) Open for RCS cooldown 2) Close when RCS cooldown completed	Safety Grade	N/A
Pressurizer SDV	1) Open for RCS depressurization 2) Close when RCS depressurization completed	Safety Grade	N/A
GTG	Provide power to SI and EFW pumps during LOOP	Safety Grade	N/A
MSIV	Close to isolate steam flow from SGs	Safety Grade	N/A
MFIV	Close to isolate MFW to ruptured SG	Safety Grade	N/A
MSSV	Automatically open/close to relieve SG pressure	N/A (code safety valve)	N/A

(*1) The analysis assumes a stuck open MSRV. The MSRV automatically opening and failing to automatically close increases the ruptured SG steam release and keeps its pressure low. A lower ruptured SG pressure results in increased primary-to-secondary leakage. Therefore, the MSRV of the ruptured SG automatically opening and then failing to close is a conservative assumption and not a mitigative function for SGTR. The analysis then later assumes the associated MSRV block valve automatically closes to isolate the ruptured SG. The closure function on low main steam line pressure is safety grade.

Table 15.06.03-8.1 Systems, Components, and Instrumentation Credited for SGTR Mitigation (Sheet 2 of 3)

Item	Function in SGTR Analysis	Safety Grade Status	Available Backups
SI Pump Switches	Stop SI pumps to terminate SI	Safety Grade	N/A
Pressurizer Water Level Indication	1) Detect event 2) Monitor RCS depressurization criteria 3) Verify pressurizer level for SI termination	Safety Grade	N/A
SG Water Level (narrow range) Indications	1) Detect event 2) Identify ruptured SG 3) Verify feedwater isolation to ruptured SG 4) Monitor RCS cooldown 5) Verify heat sink for SI termination	Safety Grade	N/A
High Sensitivity Main Steam Line Radiation (N-16) Alarms	1) Detect event 2) Identify ruptured SG	Non-Safety Grade	For all functions, two channels per loop are available. For Function 1, the pressurizer water level indication provides backup (see item above). For Functions 1 and 2, the SG water level (narrow range) indications and other radiation monitors provide backup (see items above and below).

Table 15.06.03-8.1 Systems, Components, and Instrumentation Credited for SGTR Mitigation (Sheet 3 of 3)

Item	Function in SGTR Analysis	Safety Grade Status	Available Backups
Radiation Monitors (any of the following: main steam line, condenser vacuum pump exhaust line, SG blowdown water)	1) Detect event 2) Identify ruptured SG	Non-Safety Grade	For Function 1, the pressurizer water level indication provides backup (see item above). For Functions 1 and 2, the SG water level (narrow range) indications and high sensitivity main steam line (N-16) radiation alarms provide backup (see items above).
RCS Hot Leg Temperature (wide range) Indications	Monitor RCS cooldown	Safety Grade	N/A
Main Steam Line Pressure Indications	Monitor RCS depressurization	Safety Grade	N/A
RCS Pressure Indication	1) Monitor RCS depressurization 2) Verify RCS pressure for SI termination	Safety Grade	N/A
RCS Subcooling Indication	1) Monitor RCS depressurization 2) Verify subcooling for SI termination	Safety Grade	N/A
EFW Flow Indication	1) Verify feedwater flow for RCS cooldown 2) Verify heat sink for SI termination	Safety Grade	N/A