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TOKYO, JAPAN

September 30, 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-11331

Subject: MHI's Response to US-APWR DCD RAI No. 789-5920 Revision 3 (SRP 15.2.1 – 15.2.8) and Revised Response to US-APWR DCD RAI No. 297-2287 Revision 2 (SRP 15.0.0)

References: 1) "MHI's Response to US-APWR DCD RAI No. 297-2287 Revision 2", MHI Letter UAP-HF-09340, dated July 3, 2009 (ML 091890966).

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the documents entitled "MHI's Response to US-APWR DCD RAI No. 789-5920 Revision 3 (SRP 15.2.1 – 15.2.8)" and "MHI's Revised Response to US-APWR DCD RAI No. 297-2287 Revision 2 (SRP 15.0.0)". The enclosed materials provide MHI's response to the NRC's "Request for Additional Information (RAI) 789-5920 Revision 3," dated July 26, 2011. The MHI response to RAI No. 789-5920 Revision 3 resulted in changes to the RAI response previously submitted in Reference 1. Therefore, the enclosed material also provides a revised version of the response to Questions 15.0.0-10 and 15.0.0-16 in Reference 1. Only Questions 15.0.0-10 and 15.0.0-16 of Reference 1 have been revised; the responses to all of the other questions are not changed and are not included in the enclosed material.

As indicated in the enclosed materials, Enclosures 2 and 4 contain 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. Non-proprietary versions of the documents are also being submitted in this package (Enclosures 3 and 5). In the non-proprietary versions, the proprietary information, bracketed in the proprietary versions, is replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2) and non-proprietary version (Enclosure 3) of the RAI 789-5920 response, a copy of the proprietary version (Enclosure 4) and non-proprietary version (Enclosure 5) of the revised RAI 297-2287 response, and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all material designated as "Proprietary" in Enclosures 2 and 4 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.

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NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Response to US-APWR DCD RAI No. 789-5920 Revision 3 (SRP 15.2.1 – 15.2.8) (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 789-5920 Revision 3 (SRP 15.2.1 – 15.2.8) (non-proprietary)
4. MHI's Revised Response to US-APWR DCD RAI No. 297-2287 Revision 2 (SRP 15.0.0) (proprietary)
5. MHI's Revised Response to US-APWR DCD RAI No. 297-2287 Revision 2 (SRP 15.0.0) (non-proprietary)

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

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ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11331

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 documents entitled "MHI's Response to US-APWR DCD RAI No. 789-5920 Revision 3 (SRP 15.2.1 – 15.2.8)" and "MHI's Revised Response to US-APWR DCD RAI No. 297-2287 Revision 2 (SRP 15.0.0), both dated September 30, 2011, and have determined that the documents 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 documents indicate 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 30th day of September, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11331
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 789-5920 Revision 3
(SRP 15.2.1 – 15.2.8)

September 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/30/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 789-5920 REVISION 3**
SRP SECTION:	**15.02.01 - 15.02.05 - LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR); AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)**
APPLICATION SECTION:	**15.2.1 - 15.2.8**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.02.01-15.02.05-9

DCD Section 15.2.2 presents analysis of a turbine trip (TT) event. The staff issued RAI 2329, question 15.2-2 asking why the loss of offsite power (LOOP) case was not included. In its response, MHI stated the TT event with LOOP was bound by either the 15.2.1 Loss of External Load or the 15.3.1.2 Complete Loss of Forced Coolant Flow. The staff was unable to reach the same conclusion because of the limited information included in the response. In order to find that the application contains adequate information to demonstrate that SRP criteria regarding LOOP are met, the staff requests that the offsite power unavailable case be added to the DCD and that this evaluation include time related variations of all key parameter.

ANSWER:

Although not credited in the safety analyses, the US-APWR design provides for a reactor trip initiation function on a turbine trip, which is described in DCD Subsection 7.2.1.4.8 as follows:

> RT on turbine trip (TT) is an anticipatory trip that is not credited in the safety analysis. Therefore, this is not a safety function but it is designed to be highly reliable. The high reliable design meets the guidance of BTP 7-9 (Reference 7.2-12).

Not crediting the direct reactor trip on turbine trip is conservative since it results in a delay of the reactor trip and produces the most severe result.

MHI performed a sensitivity study of the turbine trip event and the timing of LOOP which results in the lowest DNBR. The study showed that a LOOP occurring 7.4 seconds after the turbine trip where the high pressurizer pressure and low RCP speed reactor trips occur at the same time (after the LOOP occurs) gives the most severe result. The resulting time sequence of events for the most severe case is shown in Table 15.02.01-15.02.05-9.1. Figures 15.02.01-15.02.05-9.1 through 15.02.01-15.02.05-9.3 provide transient plots of the key parameters for the most severe case. The DNB result shown in Figure 15.02.01-15.02.05-9.3 demonstrates that the minimum DNBR for the sensitivity study case is higher than the complete loss of flow analysis result provided in DCD Subsection 15.3.1.2. This result confirms the original MHI position that the turbine trip analysis is bounded by the complete loss of flow analysis for minimum DNBR.

MHI also performed a sensitivity study of the turbine trip event regarding the timing of LOOP

occurrence which results in the maximum RCS pressure and main steam pressure. The study showed that a LOOP occurring 7.6 seconds after the turbine trip where the high pressurizer pressure and low RCP speed reactor trips occur at the same time (after the LOOP occurs) gives the most severe RCS pressure. The resulting time sequence of events for the most severe case is shown in Table 15.02.01-15.02.05-9.2. Figures 15.02.01-15.02.05-9.4 through 15.02.01-15.02.05-9.9 provide transient plots of the key parameters for the most severe RCS pressure case. The RCP outlet pressure result shown in Figure 15.02.01-15.02.05-9.5 demonstrates that the RCP outlet pressure for the sensitivity study case is slightly higher than the loss of external load result provided in DCD Subsection 15.2.1. The sensitivity analysis also includes an additional case designed to maximize main steam pressure by using a detailed MSSV model and modifying other initial conditions (from the RCS pressure case) as necessary. Note that in the main steam pressure case, the LOOP timing (7.4 seconds after the turbine trip) is also different from the RCS pressure case. The resulting time sequence of events and main steam pressure transient for this case are shown in Table 15.02.01-15.02.05-9.3 and Figure 15.02.01-15.02.05-9.10, respectively.

MHI will revise DCD Section 15.2.2 to include the DNBR, RCS, and main steam pressure cases of a turbine trip event assuming a LOOP. Note that the response to RAI 809-5957 Questions 15-33 showed the impact of the initial RCS temperature assumption and detailed MSSV model, respectively. Although the peak main steam system pressure for this event is slightly lower than the Loss of External Load analysis in Section 15.2.1, these changes are also applied to the main steam system pressure analysis for the turbine trip event.

Table 15.02.01-15.02.05-9.1
Time Sequence of Events for Turbine Trip (Assuming LOOP) – DNBR Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached*	6.7
RCP coastdown begins due to loss of offsite power	7.4
Low reactor coolant pump speed analytical limit reached*	7.9
Reactor trip initiated (rod motion begins)	8.5
Pressurizer safety valves open	8.6
Main steam safety valves open	9.8
Minimum DNBR occurs	10.7

* Response time of high pressurizer pressure reactor trip is 1.8 seconds and low reactor coolant pump speed reactor trip is 0.6 seconds as described in Table 15.0-4 in DCD Chapter 15.

Table 15.02.01-15.02.05-9.2
Time Sequence of Events for Turbine Trip (Assuming LOOP)
– RCS Pressure Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached*	6.9
RCP coastdown begins due to loss of offsite power	7.6
Low reactor coolant pump speed analytical limit reached*	8.1
Pressurizer safety valves open	8.6
Reactor trip initiated (rod motion begins)	8.7
Peak RCP outlet pressure occurs	11.2
Main steam safety valves open	11.8

* Response time of high pressurizer pressure reactor trip is 1.8 seconds and low reactor coolant pump speed reactor trip is 0.6 seconds as described in Table 15.0-4 in DCD Chapter 15.

Table 15.02.01-15.02.05-9.3
Time Sequence of Events for Turbine Trip (Assuming LOOP)
– Main Steam Pressure Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached*	6.7
RCP coastdown begins due to loss of offsite power	7.4
Main steam safety valves open (1st setpoint)	7.7
Low reactor coolant pump speed analytical limit reached*	7.9
Pressurizer safety valves open	8.4
Reactor trip initiated (rod motion begins)	8.5
Main steam safety valves open (2nd setpoint)	9.2
Main steam safety valves open (3rd setpoint)	11.7
Peak main steam system pressure occurs	15.1

* Response time of high pressurizer pressure reactor trip is 1.8 seconds and low reactor coolant pump speed reactor trip is 0.6 seconds as described in Table 15.0-4 in DCD Chapter 15.

Figure 15.02.01-15.02.05-9.1 Reactor Power versus Time –DNBR Analysis

Figure 15.02.01-15.02.05-9.2 RCS Total Flow versus Time –DNBR Analysis

Figure 15.02.01-15.02.05-9.3 DNBR versus Time –DNBR Analysis

Figure 15.02.01-15.02.05-9.4 Reactor Power versus Time –RCS Pressure Analysis

Figure 15.02.01-15.02.05-9.5 RCP Outlet Pressure versus Time –RCS Pressure Analysis

Figure 15.02.01-15.02.05-9.6 Pressurizer Water Volume versus Time –RCS Pressure Analysis

Figure 15.02.01-15.02.05-9.7 Pressurizer Safety Valve Flow Rate versus Time –RCS Pressure Analysis

Figure 15.02.01-15.02.05-9.8 RCS Average Temperature versus Time –RCS Pressure Analysis

Figure 15.02.01-15.02.05-9.9 RCS Total Flow versus Time –RCS Pressure Analysis

Figure 15.02.01-15.02.05-9.10 Steam Generator Pressure versus Time –Main Steam Pressure Analysis

Impact on DCD

DCD Section 15.2.2 will be revised as indicated in Attachment 1. To maintain consistency with DCD Section 15.2.2, DCD Table 15.0-1 will also be revised 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.

Attachment 1

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 1 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)	
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler*7		
15.1.1	Decrease in feedwater temperature	AOO	MARVEL-M	max	--	min feedback Figure 15.0-2	4466	
15.1.2	Increase in feedwater flow	AOO	MARVEL-M	max	--	min feedback Figure 15.0-2	4466	
15.1.3	Increase in steam flow	AOO	MARVEL-M	min and max	--	min feedback Figure 15.0-2	4466	
15.1.4	Inadvertent opening of a steam generator relief or safety valve	AOO	MARVEL-M, ANC, VIPRE-01M	See Figure 15.1.4-1	--	See Figure 15.1.4-2*8	0	DCD_15.0.0-30
15.1.5	Steam system piping failures - Minor/Major	AOO/PA	MARVEL-M, ANC, VIPRE-01M*1	Hot standby: Figure 15.1.4-1 HFP: max	--	Hot standby: Figure 15.1.4-2*8 HFP: min feedback Figure 15.0-2	0%, 75%, & 100% of 4466	DCD_15.0.0-30
15.2.1	Loss of external load	AOO	MARVEL-M	min	--	min feedback Figure 15.0-2	4466 for DNBR 4555*2 for RCS and main steam pressure	DCD_15.02.01-15.02.05-9
15.2.2	Turbine trip	AOO	~~Bounded by loss of load~~MARVEL-M, VIPRE-01M	~~-~~min	--	~~-~~min feedback Figure 15.0-2	~~-~~4466 for DNBR 4555*2 for RCS and main steam pressure	DCD_15.02.01-15.02.05-9
15.2.3	Loss of condenser vacuum	AOO	Bounded by loss of load	--	--	--	--	

15.2.2 Turbine Trip

15.2.2.1 Identification of Causes and Frequency Classification

In a turbine trip event, the main turbine stop valves rapidly close on loss of fluid pressure actuated by one the following turbine trip initiation signals:

- Low bearing oil pressure
- Low emergency trip header pressure
- Low condenser vacuum
- Turbine overspeed
- Thrust bearing wear
- High exhaust hood temperature
- High shaft vibration
- Low shaft-driven lube oil pump discharge pressure
- Generator trip
- Manual trip
- High-high steam generator water level
- Reactor trip

The turbine trip event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.2-1). Event frequency conditions are described in Section 15.0.0.1.

15.2.2.2 Sequence of Events and Systems Operation

The sequence of events for the turbine trip transient is similar to the loss of external load transient (Section 15.2.1) except that the steam flow following a turbine trip transient is isolated by closure of the main turbine stop valves rather than the main turbine control valves. The sudden reduction in steam flow leads to an increase in pressure and temperature in the shell side of the steam generators. As a result, the reactor coolant system (RCS) temperature and pressure increases, the coolant density decreases, and the pressurizer water volume increases.

~~However, t~~The loss of external load scenario evaluated in Section 15.2.1 was developed by assuming a step load decrease in both steam flow and feedwater flow from their full valve (100%) to zero at the beginning of the transient. This scenario thus bounds both DCD_15.02.01-15.02.05-9

the loss of external load and turbine trip events where offsite power is available. ~~Therefore, a separate detailed transient analysis is not presented for the turbine trip event.~~ DCD_15.02.01-15.02.05-9

~~In addition to turbine trip signals generated by the turbine and condenser signals listed above, the automatic reactor trip signals assumed to be available to provide protection from the loss of external load transient are also applicable to the turbine trip event. See Section 15.2.1.2 for additional details.~~

A turbine trip could cause a disturbance to the utility grid, which could in turn, cause a loss of offsite power, which could, in turn cause a reactor coolant pump (RCP) coastdown. ~~As discussed in Section 15.0.0.7, the resulting RCP coastdown would not start until after the time of minimum DNBR so that the minimum DNBR for the entire transient is the same whether offsite power is available or unavailable. Since the two cases have equally limiting minimum DNBRs,~~Only the case where offsite power is unavailable as a consequence of the turbine trip is ~~not~~ presented here since the case where offsite power is available is bounded by the loss of external load scenario evaluated in Section 15.2.1. DCD_15.02.01-15.02.05-9

The limiting single failure for this event is the failure of one train of the reactor trip system (RTS). Any one of the remaining trains is adequate to provide the protection functions credited in this assessment. Additional details about the RTS are provided in Section 7.2.

The RTS, main steam safety valves, and pressurizer safety valves may be required to mitigate the transient. No single active failure will prevent this equipment from performing their required functions. For this event, no normal reactor control systems or engineered safety systems are required to function and are thus not credited in the evaluation. However, the emergency feedwater (EFW) may be automatically actuated on loss of main feedwater; thus providing further mitigation of the transient.

In addition to the turbine trip signals generated by the turbine and condenser signals listed above, the following automatic reactor trip signals are assumed to be available to provide protection from this transient:

- High pressurizer pressure
- High pressurizer water level
- Low steam generator water level
- Low RCP speed
- Over temperature ΔT

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3. Non safety-related systems are not assumed to mitigate the consequences of this event (as discussed in Section 15.0.0.5). Long-term cooling using the residual heat removal system following this event is discussed in Section 15.0.0.8.

15.2.2.3 Core and System Performance

15.2.2.3.1 Evaluation Model

DCD_15.02.01-15.02.05-9

The MARVEL-M plant transient analysis code is used to calculate the transient response of various parameters following a turbine trip event with a loss of offsite power. This evaluation model is described in Section 15.0.2.2.1. Additional details regarding the MARVEL-M code are provided in Reference 15.2-4. The MARVEL-M code generates an interface file that includes the time-dependent histories of the reactor power, core pressure, core inlet temperature, and core inlet flow rate for use in the VIPRE-01M code.

The VIPRE-01M code (Ref.15.2-2) calculates the minimum DNBR during the transient using this interface as a boundary condition assuming a constant design power distribution. The DNBR calculation uses the RTDP and the WRB-2 DNB correlation. See Section 4.4.1.1.2 for additional details regarding the RTDP method of addressing uncertainties and Section 4.4.2.2.1 for details regarding the WRB-2 DNB correlation.

~~The response of this event is bounded by the plant response for the loss of external load event described in Section 15.2.1.3.~~

15.2.2.3.2 Input Parameters and Initial Conditions

The following assumptions were utilized in order to calculate conservative DNB transient results for the turbine trip with offsite power unavailable event:

- Consistent with the use of RTDP, the assumed initial values of reactor power, reactor coolant average temperature, and RCS pressure are assumed to be the nominal values as defined in Table 15.0-3.

- The moderator density coefficient is assumed to have the minimum value as defined in Section 15.0.0.2.4. The Doppler power coefficient is assumed to be the minimum feedback limit shown in Figure 15.0-2. Core reactivity coefficients used in the analysis are summarized in Table 15.0-1.

- Conservative assumptions for the trip simulation (trip reactivity curve, rod drop time, RTS signal processing delays) are used in the analysis. Rod cluster control assembly insertion characteristics assumed in the analysis are described in Section 15.0.0.2.5.

- The reactor is assumed to be automatically tripped by the concurrent actuation of the high pressurizer pressure signal and the low RCP speed signal. Table 15.0-4 summarizes the reactor trip setpoints and signal delay times used in the analysis.

- In this analysis, a bounding scenario that assumes an instantaneous step load decrease in both steam flow and feedwater flow from their full value (100%) to 0 initiates the event. Then a loss of offsite power (LOOP) is assumed to occur 7.4 seconds after the turbine trip. This time results in the high pressurizer pressure and low reactor coolant pump speed reactor trip signals causing concurrent rod motion. The automatic reactor trip following turbine trip is conservatively ignored, delaying the reactor trip until the plant trips on these other RTS signals.

- In the automatic rod control mode, the control rod banks would be inserted to decrease power before the reactor trip occurs. Therefore, it is conservatively assumed that the reactor is in manual rod control. DCD_15.02.01-15.02.05-9

- Additionally, the event is analyzed with the pressurizer spray actuating at 2275 psia and the safety valves credited to operate at 2525 psia. The availability of this equipment minimizes the RCS pressure, which is conservative in calculating the minimum DNBR. Separate cases to evaluate RCS and main steam system peak pressures are described in Section 15.2.2.4, Barrier Performance.

15.2.2.3.3 Results

Table 15.2.2-1 lists the key events and times at which they occur, relative to the initiation of the transient.

Figure 15.2.2-1 demonstrates that the minimum DNBR remains above the 95/95 limit and no fuel failures are predicted.

It should be noted that the base analysis for the turbine trip is to evaluate peak RCS and main steam system pressures, and is presented as part of the barrier performance analysis in Section 15.2.2.4. Therefore, only the DNBR versus time parameter plot is provided for the core response analysis. The responses of the other parameters shown in Figures 15.2.2-2 through 15.2.2-8 are approximately the same for this case.

15.2.2.4 Barrier Performance

15.2.2.4.1 Evaluation Model

The barrier performance evaluation for this transient employs the same basic model as is used for the core and systems performance evaluation (described in Section 15.2.2.3.1), except that certain input parameters and initial conditions are different so the calculations will produce either the maximum RCS or maximum main steam system pressure instead of minimum DNBR.

15.2.2.4.2 Input Parameters and Initial Conditions

For the peak RCS pressure case, the same input parameters as used as in Section 15.2.2.3.2 with the exception of the initial conditions and pressurizer spray, which are discussed below:

- The initial power level is taken as 102 percent of the licensed core thermal power level with the initial reactor coolant temperature 4°F below the nominal value and the pressurizer pressure 30 psi below the nominal value. This combination of initial conditions maximizes RCS pressure. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- For the peak RCS pressure evaluation, the turbine trip assuming LOOP event is analyzed with the pressurizer spray assumed to be unavailable. The unavailability of this equipment maximizes the RCS pressure for the barrier performance

evaluation. However, the safety valves begin to open at 2525 psia, corresponding to 101% of RCS design pressure.

- In this analysis, a LOOP is assumed to occur 7.6 seconds after a turbine trip. This time results in the high pressurizer pressure and low reactor coolant pump speed reactor trip signals causing concurrent rod motion. The automatic reactor trip following turbine trip is conservatively ignored, delaying the reactor trip until the plant trips on these other RTS signals. DCD_15.02.01-15.02.05-9

For the peak main steam pressure case, the same input parameters are used as in the RCS pressure analysis above with the following exceptions.

- The initial power level is taken as 102 percent of the licensed core thermal power level with initial reactor coolant temperature 4°F above the nominal value and the pressurizer pressure 30 psi below the nominal value. This combination of initial condition uncertainties maximizes main steam pressure. The nominal value of core power, reactor coolant temperature, and RCS pressure conditions are described in Table 15.0-3.

- In this analysis, a LOOP is assumed to occur 7.4 seconds after a turbine trip. This time results in the high pressurizer pressure and low reactor coolant pump speed reactor trip signals causing concurrent rod motion. The automatic reactor trip following turbine trip is conservatively ignored, delaying the reactor trip until the plant trips on these other RTS signals.

- The detailed MSSV model is used instead of the simplified MSSV model used in the RCS pressure analysis.

15.2.2.4.3 Results

Table 15.2.2-2 and Table 15.2.2-3 list the key events and times at which they occur, relative to the initiation of the transient.

Figures 15.2.2-2 through 15.2.2-8 are plots of system parameters versus time for the Barrier Performance Evaluation cases.

The turbine trip with LOOP event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The RCP outlet pressure (Figure 15.2.2-3) is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the reactor pressure boundary limits are not exceeded. The maximum RCS pressure remains well below 110% of the design pressure. In addition, the steam generator pressure (Figure 15.2.2-8) does not exceed 110% of the main steam design pressure. Therefore, the integrity of the reactor coolant pressure boundary and main steam system pressure boundary are maintained.

~~The response of this event is bounded by the plant response for the loss of external load event described in Section 15.2.1.4.~~

15.2.2.5 Radiological Consequences

The radiological consequences of this event are bounded by the radiological consequences of the feedwater system piping failure event evaluated in Section 15.2.8.

15.2.2.6 Conclusions

The sudden reduction in steam flow due to a turbine trip leads to an increase in pressure and temperature on the secondary side. As result the RCS temperature and pressure increases. DCD_15.02.01-15.02.05-9

The turbine trip event with offsite power available is bounded by the loss of external load event described in Section 15.2.1. Based on the results presented in Section 15.2.1.3.3, the resulting transient with offsite power available does not cause the minimum DNBR to decrease below the 95/95 limit. Additionally, the resulting transient for a turbine trip event assuming LOOP described in Section 15.2.2.3.3 does not cause the minimum DNBR to decrease below the 95/95 limit. Therefore, no fuel failures are predicted for a turbine trip event. DCD_15.02.01-15.02.05-9

The results of Section 15.2.1.4.3 for a turbine trip with offsite power available and Section 15.2.2.4.3 for a turbine trip assuming LOOP demonstrate that RCS pressure and the main steam system pressure remain well below 110% of their system design pressures. Thus, the integrity of the reactor coolant pressure boundary and the main steam system pressure boundary are maintained for the turbine trip event.

~~In addition, t~~This event does not lead to a more serious fault condition. DCD_15.02.01-15.02.05-9

DCD_15.02.01-15.02.05-9

Table 15.2.2-1
Time Sequence of Events for Turbine Trip with LOOP - DNBR Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.7
RCP coastdown begins due to loss of offsite power	7.4
Low reactor coolant pump speed analytical limit reached	7.9
Reactor trip initiated (rod motion begins)	8.5
Pressurizer safety valves open	8.6
Main steam safety valves open	9.8
Minimum DNBR occurs	10.7

Table 15.2.2-2
Time Sequence of Events for Turbine Trip with LOOP - RCS Pressure Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.9
RCP coastdown begins due to loss of offsite power	7.6
Low reactor coolant pump speed analytical limit reached	8.1
Pressurizer safety valves open	8.6
Reactor trip initiated (rod motion begins)	8.7
Peak RCP outlet pressure occurs	11.2
Main steam safety valves open	11.8

Table 15.2.2-3
Time Sequence of Events for Turbine Trip with LOOP - Main Steam Pressure Analysis

Event	Time (sec)
Turbine trip, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.7
RCP coastdown begins due to loss of offsite power	7.4
Main steam safety valves open (1st setpoint)	7.7
Low reactor coolant pump speed analytical limit reached	7.9
Pressurizer safety valves open	8.4
Reactor trip initiated (rod motion begins)	8.5
Main steam safety valves open (2nd setpoint)	9.2
Main steam safety valves open (3rd setpoint)	11.7
Peak main steam system pressure occurs	15.1

DCD_15.02.01-15.02.05-9

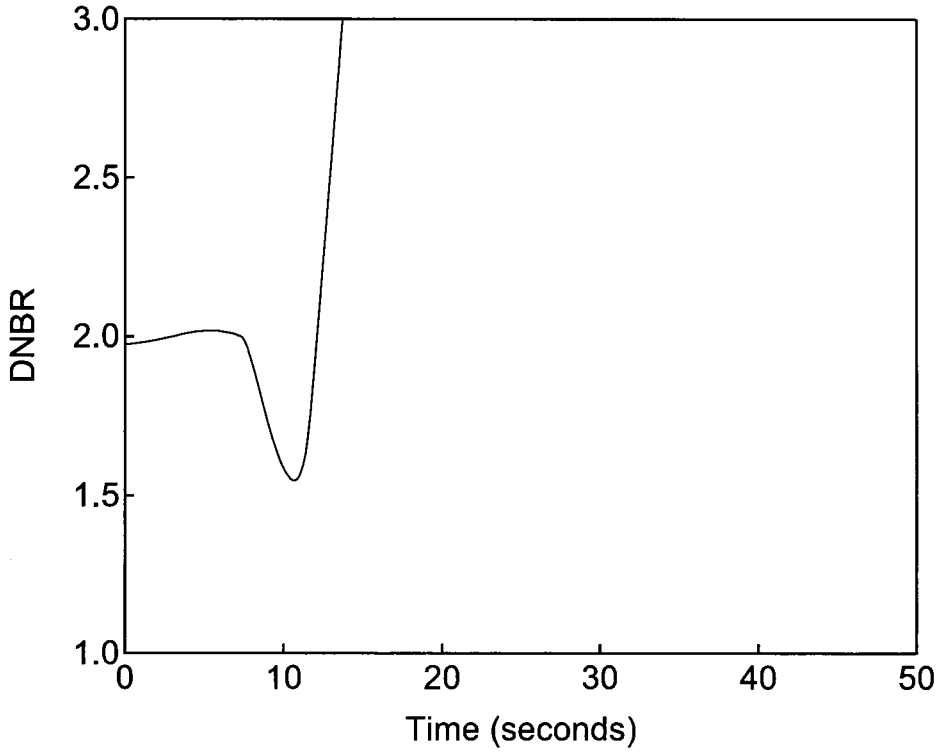

Figure 15.2.2-1 DNBR versus Time

Turbine Trip (Assuming LOOP) - DNBR Analysis

DCD_15.02.01-15.02.05-9

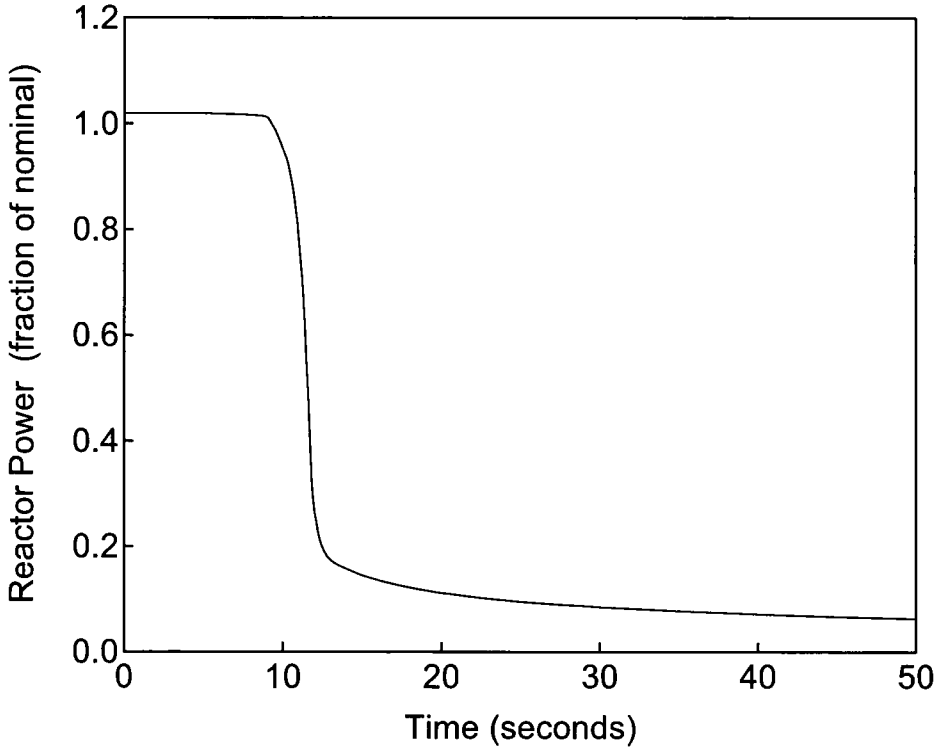

Figure 15.2.2-2 **Reactor Power versus Time**

Turbine Trip with LOOP - RCS Pressure Analysis

DCD_15.02.01-15.02.05-9

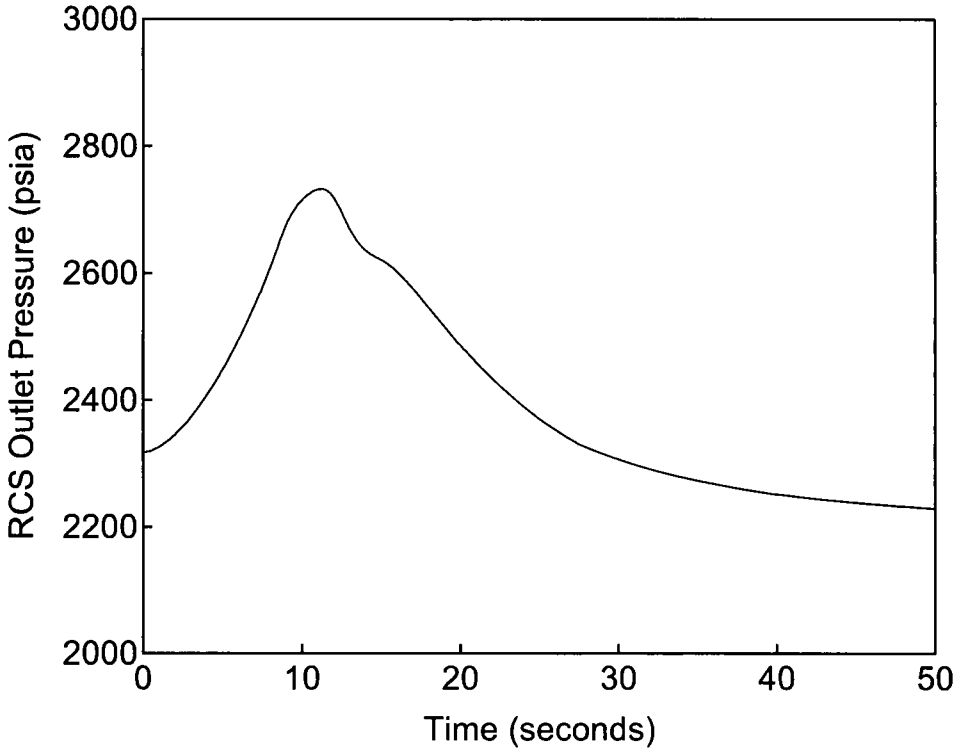

Figure 15.2.2-3 **RCP Outlet Pressure versus Time**

Turbine Trip with LOOP
- RCS Pressure Analysis

DCD_15.02.01-15.02.05-9

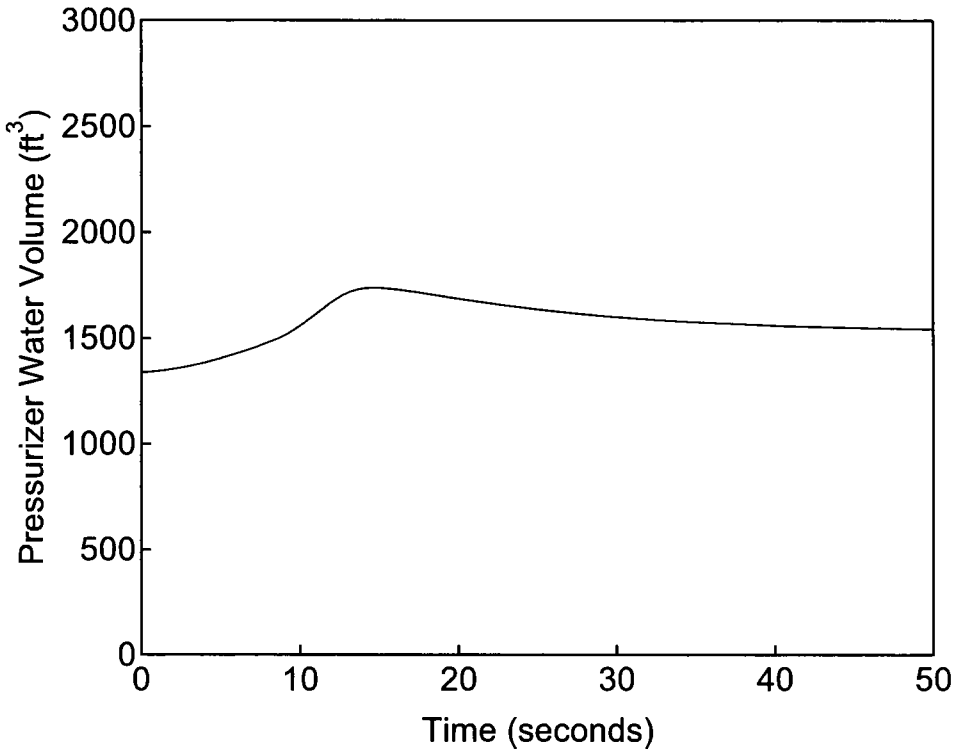

Figure 15.2.2-4 Pressurizer Water Volume versus Time

Turbine Trip with LOOP
- RCS Pressure Analysis

DCD_15.02.01-15.02.05-9

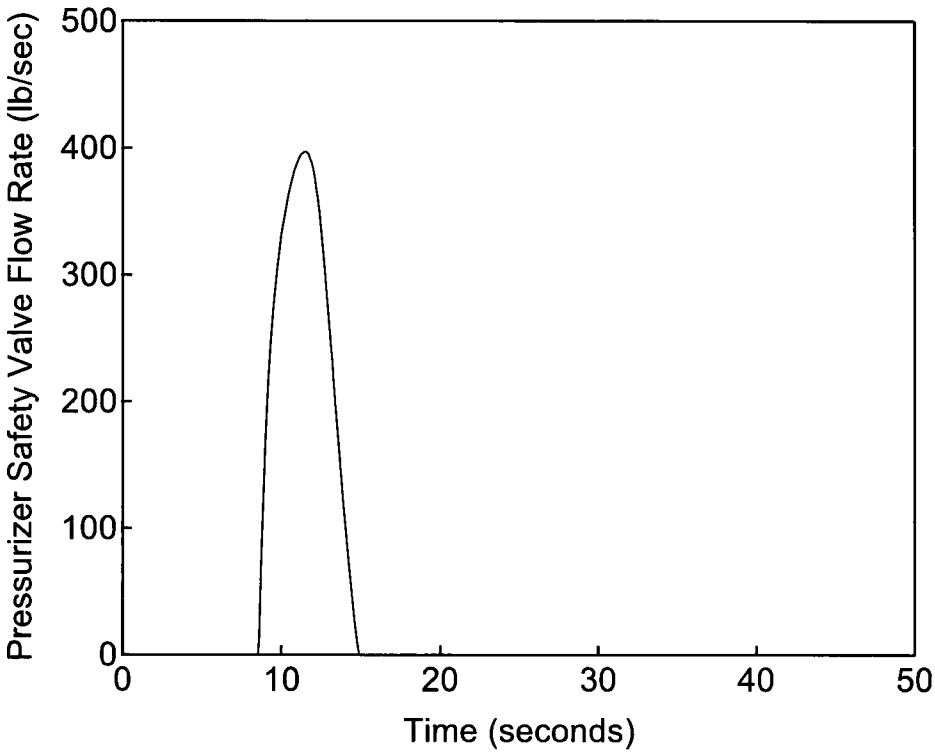

Figure 15.2.2-5 **Pressurizer Safety Valve Flow Rate versus Time**

Turbine Trip with LOOP
- RCS Pressure Analysis

DCD_15.02.01-15.02.05-9

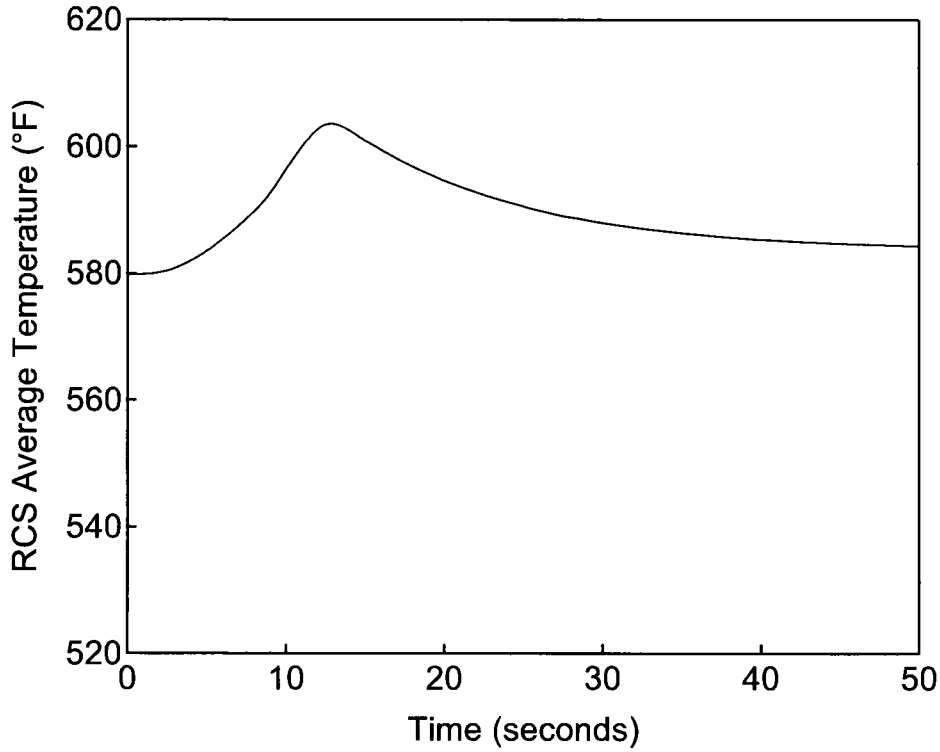

Figure 15.2.2-6 **RCS Average Temperature versus Time**

Turbine Trip with LOOP
- RCS Pressure Analysis

DCD_15.02.01-15.02.05-9

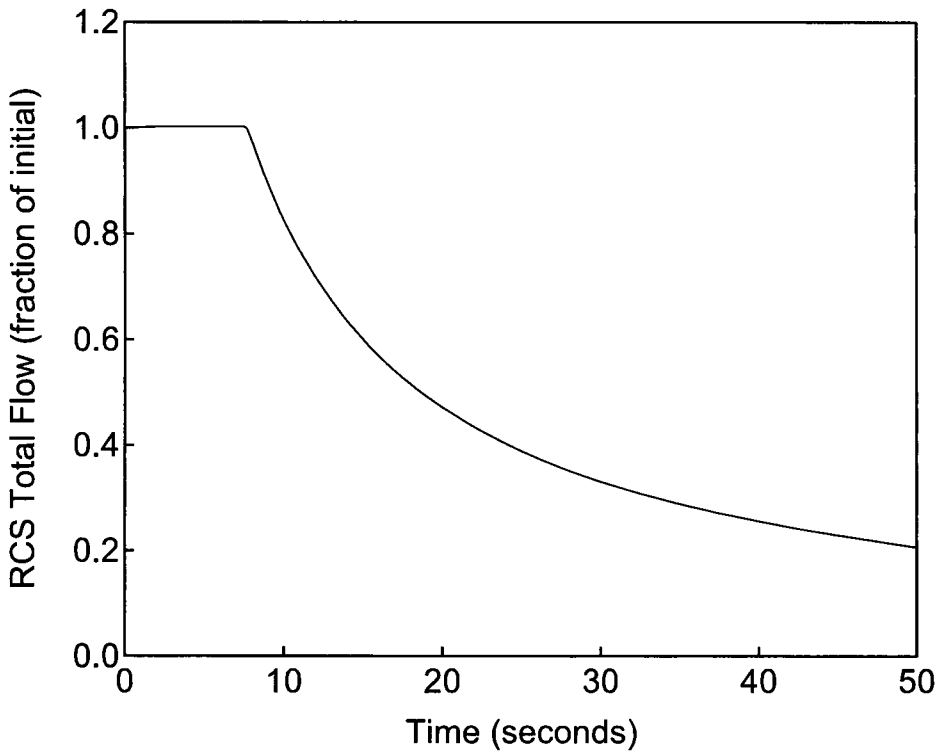

Figure 15.2.2-7 **RCS Total Flow versus Time**

Turbine Trip with LOOP - RCS Pressure Analysis

DCD_15.02.01-15.02.05-9

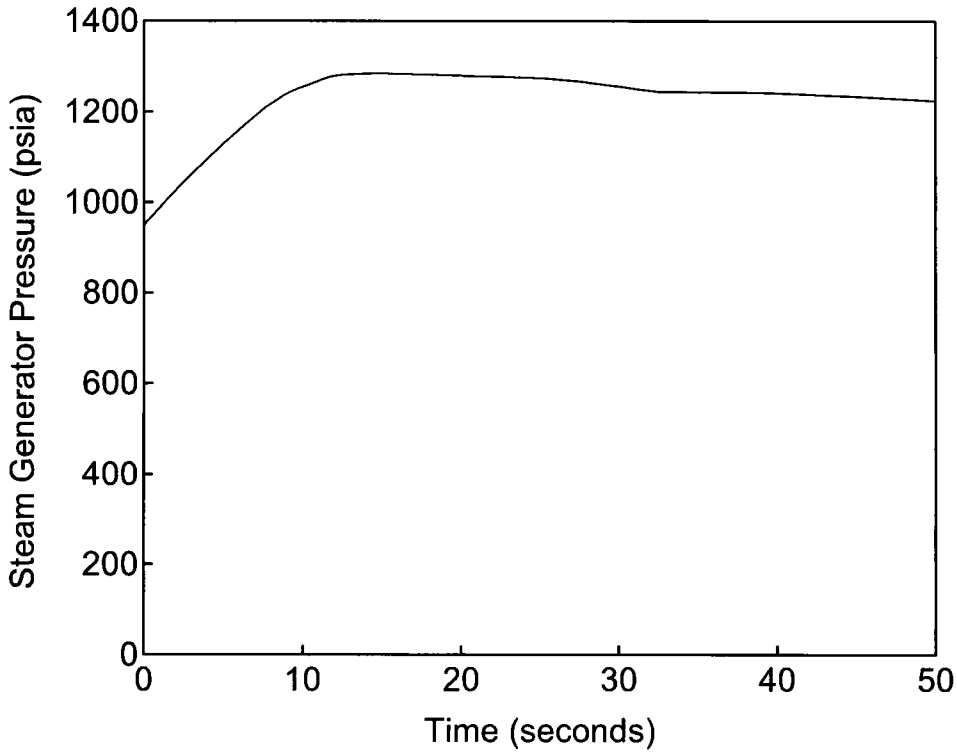

Figure 15.2.2-8 Steam Generator Pressure versus Time

Turbine Trip with LOOP
- Main Steam Pressure Analysis

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/30/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 789-5920 REVISION 3**
SRP SECTION:	**15.02.01 - 15.02.05 - LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR); AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)**
APPLICATION SECTION:	**15.2.1 - 15.2.8**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.02.01-15.02.05-10

Per Table 15.0.0-10.2 (included in MHI's response to RAI 2287, question 15.0.0-10), the decrease in heat removal transients analyzed in Sections 15.2.1 through 15.2.8 assume 10% of the steam generator tubes are plugged. This appears to be non-conservative for the secondary side overpressure analyses because it reduces the heat transfer from the primary to secondary system, reducing the heat up in the secondary side. Please justify this assumption for the 15.2.1-15.2.8 events.

ANSWER:

Generally, the SG tube plugging has minor effects in Non-LOCA accident analysis; therefore it is not a key parameter. The events in Sections 15.2.1 through 15.2.8 are heatup events. The analyses for these events assume 10% SG tube plugging. MHI uses the steam generator design UA at 10% SG tube plugging. In fact, the actual UA is greater than the design UA for both the 0% and 10% plugging cases. The 10% plugging assumption may slightly underestimate the primary-to-secondary system heat transfer capabilities and hence overestimate the heat up and pressurization of the RCS.

On the other hand, the RAI states that the assumption may not be conservative with respect to SG pressure since increased primary-to-secondary heat transfer associated with less SG tube plugging could result in higher SG pressures. Therefore, MHI performed a sensitivity analysis assuming 0% SG plugging for the loss of external load event. In this case, the SG UA is increased about 10% from the DCD case.

Figure 15.02.01-15.02.05-10.1 through 15.02.01-15.02.05-10.4 shows the sensitivity analysis result. These figures show that SG tube plugging has negligible impact on the primary and secondary system pressures. Therefore, SG tube plugging is not a key parameter when determining peak primary or secondary pressure and MHI will keep the assumption of 10% SG tube plugging with the steam generator design UA at 10% SG tube plugging for all Non-LOCA AOO and PA events. Note that this sensitivity analysis was performed based on the analysis provided in DCD Revision 3 Section 15.2.1 Loss of External Load event. The DCD Section 15.2.1 analysis will be revised to incorporate the detailed MSSV model as a separate main steam pressure case as discussed in the response to RAI 809-5957 Questions 15-33 and 15-34.

Figure 15.02.01-15.02.05-10.1 Reactor Power versus Time –RCS & Main Steam Pressure Analysis SG Tube Plugging Sensitivity Analysis

Figure 15.02.01-15.02.05-10.2 RCP Outlet Pressure versus Time –RCS & Main Steam Pressure Analysis SG Tube Plugging Sensitivity Analysis

Figure 15.02.01-15.02.05-10.3 RCS Average Temperature versus Time –RCS & Main Steam Pressure Analysis SG Tube Plugging Sensitivity Analysis

Figure 15.02.01-15.02.05-10.4 RCS Average Temperature versus Time –RCS & Main Steam Pressure Analysis SG Tube Plugging Sensitivity Analysis

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.

ENCLOSURE 5

UAP-HF-11331
Docket No. 52-021

MHI's Revised Response to US-APWR DCD RAI No. 297-2287
Revision 2 (SRP 15.0.0)

September 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/30/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 297-2287 REVISION 2**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0.0**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.0.0-10

Please extend the tables shown in Chapter 15.0 to show for each transient and accident the limiting power, temperatures, flows, levels, scram reactivity, reactivity coefficients, heat transfer coefficients, and degree of SG tube plugging.

ANSWER:

Tables 15.0.0-10.1 and 15.0.0-10.2 below provide a summary of key input parameters for each of the events analyzed in Chapter 15.

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 15.0.0-10.1 Summary of Key Input Parameters

DCD Subsection	Event Description	NSSS Thermal Output (MW_t)	RCS Flow*0 (gpm/loop)	RCS Avg Temp (°F)	RCS Pressure (psia)	Reactivity Coefficients		
						Moderator Density (Δk/k)/(g/cc)	Moderator Temperature (pcm/°F)	Doppler*1
15.1.1	Decrease in feedwater temperature	4466	115,000	583.8	2250	0.51	--	Min
15.1.2	Increase in feedwater flow	4466	115,000	583.8	2250	0.51	--	Min
15.1.3	Increase in steam flow	4466	115,000	583.8	2250	0.0 (Cases A & C) & 0.51 (Cases B & D)	--	Min
15.1.4	Inadvertent opening of a steam generator relief or safety valve	0	112,000	557	2250	See DCD Figure 15.1.4-1	--	See DCD Figure 15.1.4-2
15.1.5	Steam system piping failures – Cases A & B	0	112,000	557	2250	See DCD Figure 15.1.4-1	--	See DCD Figure 15.1.4-2
	Steam system piping failures – Case C	75% & 100% of 4466	115,000	577.1 (75%) & 583.8 (100%)	2250	0.51	--	Min
15.2.1	Loss of external load – DNBR Case	4466	115,000	583.8	2250	0.0	--	Min
	Loss of external load – RCS Pressure Case	4555*2	112,000	579.8	2220	0.0	--	Min
	Loss of external load – Main Steam System Pressure Case	4555*2	112,000	587.8	2220	0.0	--	Min
15.2.2	Turbine trip – DNBR Case	4466	115,000	583.8	2250	0.0	--	Min
	Turbine trip – RCS Pressure Case	4555*2	112,000	579.8	2220	0.0	--	Min
	Turbine trip – Main Steam System Pressure Case	4555*2	112,000	587.8	2220	0.0	--	Min
15.2.2	Turbine trip	--	--	--	--	--	--	--
15.2.3	Loss of condenser vacuum	--	--	--	--	--	--	--
15.2.4	Closure of main steam isolation valves	--	--	--	--	--	--	--
15.2.5	Steam pressure regulator failure	N/A	N/A	N/A	N/A	N/A	N/A	N/A
15.2.6	Loss of non-emergency AC power to the station auxiliaries	4555*2	112,000	579.8	2280	0.0	--	Max
15.2.7	Loss of normal feedwater flow – DNBR Case	4466	115,000	583.8	2250	0.0	--	Max
	Loss of normal feedwater flow – RCS Pressure Case	4555*2	112,000	587.8	2280	0.0	--	Max
	Loss of normal feedwater flow – Peak PRZR Water Volume Case	4555*2	112,000	579.8	2280	0.0	--	Max
15.2.8	Feedwater system pipe break – Peak RCS Pressure Case	4555*2	112,000	587.8	2280	0.0	--	Max

Table 15.0.0-10.1 Summary of Key Input Parameters

DCD Subsection	Event Description	NSSS Thermal Output (MW_t)	RCS Flow*0 (gpm/loop)	RCS Avg Temp (°F)	RCS Pressure (psia)	Reactivity Coefficients		
						Moderator Density (Δk/k)/(g/cc)	Moderator Temperature (pcm/°F)	Doppler*1
	Feedwater system pipe break – Hot Leg Boiling Case	4555*2	112,000	587.8	2220	0.0	--	Max
	Feedwater system pipe break – Peak PRZR Water Volume Case	4555*2	112,000	579.8	2280	0.0	--	Max
15.3.1.1	Partial loss of forced reactor coolant flow	4466	115,000	583.8	2250	0.0	--	Max
15.3.1.2	Complete loss of forced reactor coolant flow	4466	115,000	583.8	2250	0.0	--	Max
15.3.3	Reactor coolant pump rotor seizure – Peak Cladding Temperature Case	4555*2	112,000	587.8	2220	0.0	--	Max
	Reactor coolant pump rotor seizure – Peak RCS Pressure Case	4555*2	112,000	587.8	2280	0.0	--	Max
15.3.4	Reactor coolant pump shaft break	--	--	--	--	--	--	--
15.4.1	Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition	0	112,000	557.0	2250	--	+2	Temperature coefficient -20% from design
15.4.2	Uncontrolled control rod assembly withdrawal at power	10%, 75%, & 100% of 4466	115,000	559.7 (10%), 577.1 (75%), & 583.8 (100%)	2250	0.0 & 0.51	--	Min & Max
15.4.3	Control rod misoperation	4466	115,000	583.8	2250	0.0	--	Min
15.4.4	Startup of an inactive loop or recirculation loop at an incorrect temperature	N/A	N/A	N/A	N/A	N/A	N/A	N/A
15.4.5	Flow controller malfunction causing an increase in BWR recirculation loop	N/A	N/A	N/A	N/A	N/A	N/A	N/A
15.4.6	Inadvertent decrease in boron concentration in the RCS	0 and 4466	--	--	--	--	--	--
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper position	--	--	--	--	--	--	--
15.4.8	Spectrum of rod ejection accidents –Fuel Temperature Case (HFP)	4540*3	112,000	554.6*8	2220	--	Temperature coefficient -20% from design	Temperature coefficient -20% from design

Table 15.0.0-10.1 Summary of Key Input Parameters

DCD Subsection	Event Description	NSSS Thermal Output (MW_t)	RCS Flow*0 (gpm/loop)	RCS Avg Temp (°F)	RCS Pressure (psia)	Reactivity Coefficients		
						Moderator Density (Δk/k)/(g/cc)	Moderator Temperature (pcm/°F)	Doppler*1
	Spectrum of rod ejection accidents –Fuel Enthalpy Case (HZP)	0	112,000	557.0*8	2250	--	Temperature coefficient -20% from design	Temperature coefficient -20% from design
15.5.1	Inadvertent operation of ECCS that increases reactor coolant inventory	N/A	N/A	N/A	N/A	N/A	N/A	N/A
15.5.2	CVCS malfunction that increases reactor coolant inventory	4555*2	112,000	579.8	2280	0.0	--	Min
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve	4466	115,000	583.8	2250	0.0	--	Max
15.6.2	Radiological consequences of the failure of small lines carrying primary coolant outside containment	4540*3	--	--	--	--	--	--
15.6.3	Radiological consequences of SGTR – Dose Evaluation	4555*2	112,000	587.8	2280	0.0	--	Max
	Radiological consequences of SGTR – SG Overfill	4555*2	112,000	579.8	2280	0.0		Max
15.6.5	Loss-of-Coolant Accidents (LB LOCA)	4466	112,000	--*4	--*4	--*5	--	--*5
	Loss-of-Coolant Accidents (SB LOCA)	4555*2	112,000	587.8	2280	--*6	--	--*7

Notes:

***0** Per DCD Table 15.0-3, 112,000 gpm is used for events not analyzed using RTDP (RTDP events use 115,000 gpm)
***1** Unless otherwise stated, the reference figure for Doppler feedback is DCD Figure 15.0-2
***2** 102% of 4466 MW_t (NSSS thermal power)
***3** 102% of 4451 MW_t (core thermal power)
***4** Values are randomly sampled over their range in the calculations
***5** Applicability confirmed (DCD Ref. 15.0-18)
***6** Conservative moderator density coefficient changes with moderator density assumed (DCD Ref. 15.0-20)
***7** Conservative Doppler temperature coefficient changes with moderator density assumed (DCD Ref. 15.0-20)
***8** This value is indicated core inlet temperature

Table 15.0.0-10.2 Summary of Key Input Parameters

DCD Subsection	Event Description	PRZR Water Vol. (ft^3)	FW/EFW Temp (°F)	Scram Reactivity (%Δk/k)	PRZR SV Setpoint (psia)	MSSV Setpoint (psia)	Initial SG Mass (lbs/SG)	SG Plugging	h_{gap} (BTU/ (hr-ft^2-°F)$^{-2}$
15.1.1	Decrease in feedwater temperature			-4	2525	1236		10%	
15.1.2	Increase in feedwater flow			-4	2525	1236		10%	
15.1.3	Increase in steam flow			-4	2525	1236		10%	
15.1.4	Inadvertent opening of a steam generator relief or safety valve			1.6 (SDM)	2525	1236		10%	
15.1.5	Steam system piping failures – Cases A & B			1.6 (SDM)	2525	1236		10%	
	Steam system piping failures – Case C			1.6 (SDM, 75%) & -4 (100%)	2525	1236		10%	
15.2.1	Loss of external load – DNBR Case			-4	2525	1236		10%	
	Loss of external load – RCS Pressure Case			-4	2525	1236		10%	
	Loss of external load – Main Steam System Pressure Case			-4	2525	1212 1243 1272		10%	
15.2.2	Turbine trip – DNBR Case			-4	2525	1236		10%	
	Turbine trip – RCS Pressure Case			-4	2525	1236		10%	
	Turbine trip – Main Steam System Pressure Case			-4	2525	1212 1243 1272		10%	
15.2.3	Loss of condenser vacuum			--	--	--		--	
15.2.4	Closure of main steam isolation valves			--	--	--		--	
15.2.5	Steam pressure regulator failure			N/A	N/A	N/A		N/A	
15.2.6	Loss of non-emergency AC power to the station auxiliaries			-4	2525	1236		10%	

Table 15.0.0-10.2 Summary of Key Input Parameters

DCD Subsection	Event Description	PRZR Water Vol. (ft^3)	FW/EFW Temp (°F)	Scram Reactivity (%Δk/k)	PRZR SV Setpoint (psia)	MSSV Setpoint (psia)	Initial SG Mass (lbs/SG)	SG Plugging	h_{gap} (BTU/ (hr-ft^2-°F)$^{-2}$
15.2.7	Loss of normal feedwater flow – DNBR Case			-4	2525	1236		10%	
	Loss of normal feedwater flow – RCS Pressure Case			-4	2525	1236		10%	
	Loss of normal feedwater flow – Peak PRZR Water Volume Case			-4	2525	1236		10%	
15.2.8	Feedwater system pipe break – Peak RCS Pressure Case			-4	2525	1236		10%	
	Feedwater system pipe break – Hot Leg Boiling Case			-4	2525	1236		10%	
	Feedwater system pipe break – Peak PRZR Water Volume Case			-4	2525	1236		10%	
15.3.1.1	Partial loss of forced reactor coolant flow			-4	2525	1236		10%	
15.3.1.2	Complete loss of forced reactor coolant flow			-4	2525	1236		10%	
15.3.3	Reactor coolant pump rotor seizure – Peak Cladding Temperature Case			-4	2525	1236		10%	
	Reactor coolant pump rotor seizure – Peak RCS Pressure Case			-4	2525	1236		10%	
15.3.4	Reactor coolant pump shaft break			--	--	--		--	
15.4.1	Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition			-2	N/A	N/A		N/A	
15.4.2	Uncontrolled control rod assembly withdrawal at power			1.6 (10%, 75%), -4 (100%)	2525	1236		10%	
15.4.3	Control rod misoperation			-4	2525	1236		10%	

Table 15.0.0-10.2 Summary of Key Input Parameters

DCD Subsection	Event Description	PRZR Water Vol. (ft^3)	FW/EFW Temp (°F)	Scram Reactivity (%Δk/k)	PRZR SV Setpoint (psia)	MSSV Setpoint (psia)	Initial SG Mass (lbs/SG)	SG Plugging	h_{gap} (BTU/ (hr-ft^2-°F) ^{*2}
15.4.4	Startup of an inactive loop or recirculation loop at an incorrect temperature			N/A	N/A	N/A		N/A	
15.4.5	Flow controller malfunction causing an increase in BWR recirculation loop			N/A	N/A	N/A		N/A	
15.4.6	Inadvertent decrease in boron concentration in the RCS			--	--	--		10%	
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper position			--	--	--		--	
15.4.8	Spectrum of rod ejection accidents -Fuel Temperature Case (HFP)			-4	N/A	N/A		N/A	
	Spectrum of rod ejection accidents –Fuel Enthalpy Case (HZP)			-2	N/A	N/A		N/A	
15.5.1	Inadvertent operation of ECCS that increases reactor coolant inventory			N/A	N/A	N/A		N/A	
15.5.2	CVCS malfunction that increases reactor coolant inventory			-4	2525	1236		10%	
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve			-4	2525	1236		10%	
15.6.2	Radiological consequences of the failure of small lines carrying primary coolant outside containment			--	--	--	--	--	
15.6.3	Radiological consequences of SGTR – Dose Evaluation			-4	2525	1236		10%	
	Radiological consequences of SGTR – SG Overfill			-4	2525	1236		10%	

Table 15.0.0-10.2 Summary of Key Input Parameters

DCD Subsection	Event Description	PRZR Water Vol. (ft^3)	FW/EFW Temp (°F)	Scram Reactivity (%Δk/k)	PRZR SV Setpoint (psia)	MSSV Setpoint (psia)	Initial SG Mass (lbs/SG)	SG Plugging	h_{gap} (BTU/(hr-ft^2-°F)) *2
15.6.5	Loss-of-Coolant Accidents (LB LOCA)			N/A	N/A	N/A		10%	
	Loss-of-Coolant Accidents (SB LOCA)			-4	N/A	1296		10%	

*1 Values are randomly sampled over their range in the calculations.
*2 Hot spot fuel-to-cladding gap conductance

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/30/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 297-2287 REVISION 2**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0.0**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.0.0-16

In each of the transient and accident analyses, provide numeric values of MDNBR and peak primary and secondary pressure to compare with the allowable limits and demonstrate that all acceptance criteria are met.

ANSWER:

The numeric values of MDNBR, peak primary pressure, and peak secondary pressure, along with the allowable limits, for all events are shown in Table 15.0.0-16.1 below. As shown in the table, all acceptance criteria are met, except for the minimum DNBR for the RCP rotor seizure, single RCCA withdrawal, and RCCA ejection events. For some events (e.g. loss of external load), the DCD describes different cases to calculate minimum DNBR and peak pressure. As a result, the limiting values in Table 15.0.0-16.1 may not occur for the same case, but indicate the most limiting value from all of the different cases evaluated for that event.

Table 15.0.0-16.1 Results of Chapter 15 Accident Analyses Compared to Acceptance Criteria

DCD Subsection	Event Description	MDNBR	Peak Primary Pressure (psia) [Limit=2750]		Peak Secondary Pressure (psia) [Limit=1320]
			Maximum RCS#	Pressurizer Surge Line Connection	
15.1.1	Decrease in Feedwater Temperature				
15.1.2	Increase in Feedwater Flow				
15.1.3	Increase in Steam Flow				
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve				
15.1.5	Steam System Piping Failure (hot zero power)				
	Steam System Piping Failure (at power)				
15.2.1	Loss of External Load				
15.2.2	Turbine Trip				
15.2.3	Loss of Condenser Vacuum*1				
15.2.4	Closure of Main Steam Isolation Valve*1				
15.2.5	Steam Pressure Regulator Failure*0				
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries				
15.2.7	Loss of Normal Feedwater Flow				
15.2.8	Feedwater System Pipe Break				
15.3.1.1	Partial Loss of Forced Reactor Coolant Flow				
15.3.1.2	Complete Loss of Forced Reactor Coolant Flow				
	Frequency Decay Resulting in Complete Loss of Flow				
15.3.2	Flow Controller Malfunctions*0				
15.3.3	Reactor Coolant Pump Rotor Seizure				
15.3.4	Reactor Coolant Pump Shaft Break*2				
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition				
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power				
15.4.3	One or More Dropped RCCAs				
	One of More Misaligned RCCAs				
	Uncontrolled Withdrawal of a Single RCCA				
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature*0				
15.4.5	Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate*0				
15.4.6	Inadvertent Decrease in Boron Concentration in the RCS*4				
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position*4				

Table 15.0.0-16.1 Results of Chapter 15 Accident Analyses Compared to Acceptance Criteria

DCD Subsection	Event Description	MDNBR	Peak Primary Pressure (psia) [Limit=2750]		Peak Secondary Pressure (psia) [Limit=1320]
			Maximum RCS#	Pressurizer Surge Line Connection	
15.4.8	Spectrum of Rod Ejection Accidents				
15.4.9	Spectrum of Rod Drop Accidents in a BWR*0				
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory*0				
15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory				
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve				
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment*4				
15.6.3	Radiological Consequences of Steam Generator Tube Failure				
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)*0				
15.6.5	Loss-of-Coolant Accidents				
15.7.1	Gas Waste Management System Leak or Failure*4				
15.7.2	Liquid Waste Management System Leak or Failure*4				
15.7.3	Release of Radioactivity to the Environment Due to a Liquid Tank Failure*4				
15.7.4	Fuel Handling Accident*4				
15.7.5	Spent Fuel Cask Drop Accident*4				

Notes:

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.