MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
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-11332

Subject: MHI's Response to US-APWR DCD RAI No. 809-5957 Revision 3 (SRP 15.0)

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. 809-5957 Revision 3 (SRP 15.0)". The enclosed material provides MHI's response to the NRC's "Request for Additional Information (RAI) 809-5957 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. 809-5957 Revision 3 (SRP 15.0) (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 809-5957 Revision 3 (SRP 15.0) (non-proprietary)

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

Contact Information

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-11332

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. 809-5957 Revision 3 (SRP 15.0)", dated September 30, 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 30th day of September, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11332
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 809-5957 Revision 3

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. 809-5957 REVISION 3**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0**
DATE OF RAI ISSUE:	**8/22/2011**

QUESTION NO.: 15-33

The application states it is not necessary to include the loss of offsite power (LOOP) case for all events because the three second delay between the reactor/turbine trip and LOOP assures that the minimum DNBR is captured, which is the principal concern with LOOP. The response to RAI 2287, Question 15.0.0-2 demonstrates that this approach is bounding with respect to minimum DNBR, peak RCS pressure, peak fuel centerline temperature and peak cladding temperature, but not peak secondary side pressure. Provide justification that this approach bounds peak secondary side pressure. Because the staff could not find a case designed to maximize secondary side pressure, include a discussion of how the peak secondary side pressure event was identified.

ANSWER:

The input parameters and initial conditions are determined in order to maximize the RCS and main steam pressures for the Loss of External Load event as described in DCD Subsection 15.2.1.4.1. This is the limiting event for main steam pressure with a peak secondary pressure of( ), as shown in Table 15.0.0-16.1 in the response to RAI 297-2287 Question 15.0.0-16. In addition, MHI performed sensitivity analyses for the main steam pressure in some RAI responses as follows.

- As shown in Figure 15.0.0-3.31 in the response to RAI 297-2287 Question 15.0.0-3, the impact of LOOP on the secondary side pressure is negligible for the Loss of External Load event. The results in that response showed that the peak main steam pressure of the LOOP case is slightly less than the value of the without LOOP case (DCD 15.2.1 case).

- MHI performed a sensitivity analysis to determine the impact of the SG tube plugging assumption in the response to RAI 789-5920 Question 15.02.01-15.02.05-10. The results showed that the impact of SG tube plugging on the peak main steam pressure is negligible. SG tube plugging is not a key parameter.

- MHI performed a sensitivity analysis to determine the impact of the MSSV modeling assumptions in the response to Question 15-34 of this RAI. The results show that the detailed MSSV model has a negligible effect on RCS pressure, but causes a small increase in peak main steam pressure. Although the sensitivity case shows higher SG pressures than the DCD case, the results still have margin to the limit of 1320 psia. It is also important to note that even if using a simplified MSSV model, if the MSSV total steam release rate is less than the maximum capacity, the secondary maximum pressure is assured to remain less than the limit of 1320 psia.

In addition to the assumptions above, MHI performed a sensitivity analysis assuming the following initial condition (with the detailed MSSV model).

- The initial reactor coolant temperature is 4°F above the nominal value for the main steam pressure case while it is assumed to be 4°F below the nominal value for the RCS pressure case.

The first two assumptions (LOOP and SG tube plugging) have a negligible impact on peak main steam pressure. Therefore, the current DCD assumptions are satisfactory. The detailed MSSV model and initial RCS temperature considering +4°F uncertainty have been shown to increase the numerical value of the peak main steam pressure, although it remains below the safety limit. In order to maximize the peak main steam pressure, MHI will revise DCD Section 15.2.1 Loss of External Load and 15.2.2 Turbine Trip events to include new cases to incorporate the detailed MSSV model and change in the assumption of the initial reactor coolant temperature uncertainty. These are applicable only to the cases of the maximum secondary pressure evaluation in DCD Section 15.2.1 Loss of External Load and 15.2.2 Turbine Trip events. The maximum RCS pressure and minimum DNBR evaluations are not changed. Note that the comparison between the current DCD Section 15.2.1 Loss of External Load analysis, the sensitivity case which only includes the detailed MSSV model (the case presented in the response to Question 15-34), and the case including both the detailed MSSV model and change in the initial RCS temperature is shown in Figure 15-33.1.

Figure 15-33.1 Steam Generator Pressure versus Time
Loss of External Load – Main Steam Pressure Analysis

Impact on DCD

The detailed three setpoint MSSV model and initial RCS temperature assumption are incorporated into the peak main steam pressure analysis of the DCD Section 15.2.1 Loss of External Load and 15.2.2 Turbine Trip events. The mark-up of the DCD for Section 15.2.1 is indicated in Attachment 1. Note that the impact on DCD Section 15.2.2 has already been incorporated into the mark-up for the response to RAI 789-5920 Question 15.02.01-15.02.05-9 such that the required change is not included in the mark-up for this RAI response.

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.

The DNBR calculations use 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.

15.2.1.3.2 Input Parameters and Initial Conditions

The following assumptions are utilized in order to calculate conservative DNBR transient results for the loss of external load 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 high pressurizer pressure signal. Table 15.0-4 summarizes the reactor trip setpoint and signal delay time 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. The automatic reactor trip following turbine trip is conservatively ignored, delaying the reactor trip until the plant trips on other RTS signals. Since the transient is modeled in this manner, the results bound both the loss of load and the turbine trip without loss of offsite power~~(Section 15.2.2)~~ events. DCD_15-33

- 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.

- 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. ~~A s~~Separate cases to evaluate RCS and main steam system peak pressures ~~is~~are described in Section 15.2.1.4, Barrier Performance. DCD_15-33

Figure 15.2.1-1 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 loss of external load~~ and turbine trip~~ is to evaluate peak RCS and main steam system pressures, and is presented as part of the barrier performance analysis in Section 15.2.1.4. Therefore, only the DNBR versus time parameter plot is provided for the core response analysis. The responses of the other parameters shown in Figures 15.2.1-2 through 15.2.1-7 are approximately the same for this case. DCD_15-33

15.2.1.4 Barrier Performance

15.2.1.4.1 Evaluation Model

The barrier performance evaluation for this transient employs the same basic model as is used for the core and system performance evaluation (described in Section 15.2.1.3.1), except that certain input parameters and initial conditions are different so the calculations will produce either the maximum RCS ~~and~~or maximum main steam system pressure~~s~~ instead of minimum DNBR. DCD_15-33

15.2.1.4.2 Input Parameters and Initial Conditions

For the peak RCS pressure case, ~~T~~the same input parameters are used as in Section 15.2.1.3.2 with the exception of the initial conditions and pressurizer spray, which are discussed below. DCD_15-33

- The initial power level is taken as 102 percent of the licensed core thermal power level with initial reactor coolant temperature 4°F below the nominal value and the pressurizer pressure 30 psi below the nominal value. This combination of initial condition uncertainties 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 loss of external load/turbine trip 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 are assumed to be operable. The pressurizer safety valves begin to open at 2525 psia, corresponding to 101% of RCS design pressure.

For the peak main steam pressure case, the same input parameters are used as in the RCS pressure analysis above with the following exceptions. DCD_15-33

- 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.

- The detailed MSSV model is used instead of the simplified MSSV model used in the RCS pressure analysis. DCD_15-33

15.2.1.4.3 Results

Table 15.2.1-2 and Table 15.2.1-3 list~~s~~ the key events and times at which they occur, relative to the initiation of the transient. DCD_15-33

Figures 15.2.1-2 through 15.2.1-7 are plots of system parameters versus time for the Barrier Performance Evaluation cases. DCD_15-33

The loss of external load~~/turbine trip~~ event does not result in exceeding any RCS pressure boundary or containment volume fission product barrier design limits. The RCP outlet pressure (Figure 15.2.1-3) is the highest pressure in the RCS and is presented in place of RCS pressure for the purpose of confirming the reactor coolant 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 15.2.1-7) does not exceed 110% of the main steam system design pressure. Therefore, the integrity of the reactor coolant pressure boundary and main steam system pressure boundary are maintained.

The Figure 15.2.1-1 parameter plot for DNBR for the loss of external load/turbine trip event is presented as part of the core response analysis in Section 15.2.1.3. The response of reactor power, average core heat flux, and maximum core heat flux are almost indistinguishable for this transient. Therefore, only a plot of reactor power is provided. Because there is subcooling margin and DNB does not occur, plots for average and hot channel exit temperatures and steam fractions are not provided. These parameters are bounded by the more severe feedwater system pipe break event evaluated in Section 15.2.8, which provides plots for hot leg, cold leg, and saturation temperatures. An RCS average temperature plot is provided to characterize the temperature response for this event. Steam generator pressure (Figure 15.2.1-7) is provided in place of steam line pressure for the purpose of demonstrating that the main steam system pressure meets the acceptance limit. All pressurizer safety valve flow is steam since the pressurizer does not fill. Containment parameters are not reported for this event because there are no releases directly from the RCS or steam generators inside containment.

15.2.1.5 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.

15.2.1.6 Conclusions

The sudden reduction in steam flow due to a loss of external load/turbine trip leads to an increase in pressure and temperature in the secondary side of the steam generators. As a result, the RCS temperature and pressure increases. However, the resulting transient does not cause the minimum DNBR to decrease below the 95/95 limit, and no fuel failures are predicted.

Table 15.2.1-1
Time Sequence of Events for Loss of External Load~~/Turbine Trip Transient~~ - DNBR Analysis

DCD_15-33

Event	Time (sec)
Loss of main steam flow, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.7
Reactor trip initiated (rod motion begins)	8.5
Pressurizer safety valves open	8.6
Minimum DNBR occurs	9.5
Main steam safety valves open	9.7
~~Peak RCP outlet pressure occurs~~	~~10.3~~
~~Peak main steam system pressure occurs~~	~~14.3~~

DCD_15-33

Table 15.2.1-2
Time Sequence of Events for Loss of External Load ~~/Turbine Trip Transient~~ - RCS ~~& Main Steam~~ Pressure Analysis

DCD_15-33

Event	Time (sec)
Loss of main steam flow, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.9
Pressurizer safety valves open	8.6
Reactor trip initiated (rod motion begins)	8.7
Peak RCP outlet pressure occurs	10.9
Main steam safety valves open	11.5
~~Peak main steam system pressure occurs~~	~~14.9~~

DCD_15-33

Table 15.2.1-3
Time Sequence of Events for Loss of External Load - Main Steam Pressure Analysis

DCD_15-33

Event	Time (sec)
Loss of main steam flow, loss of main feedwater flow	0.0
High pressurizer pressure analytical limit reached	6.7
Main steam safety valves open (1st setpoint)	7.7
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)	11.2
Peak main steam system pressure occurs	14.5

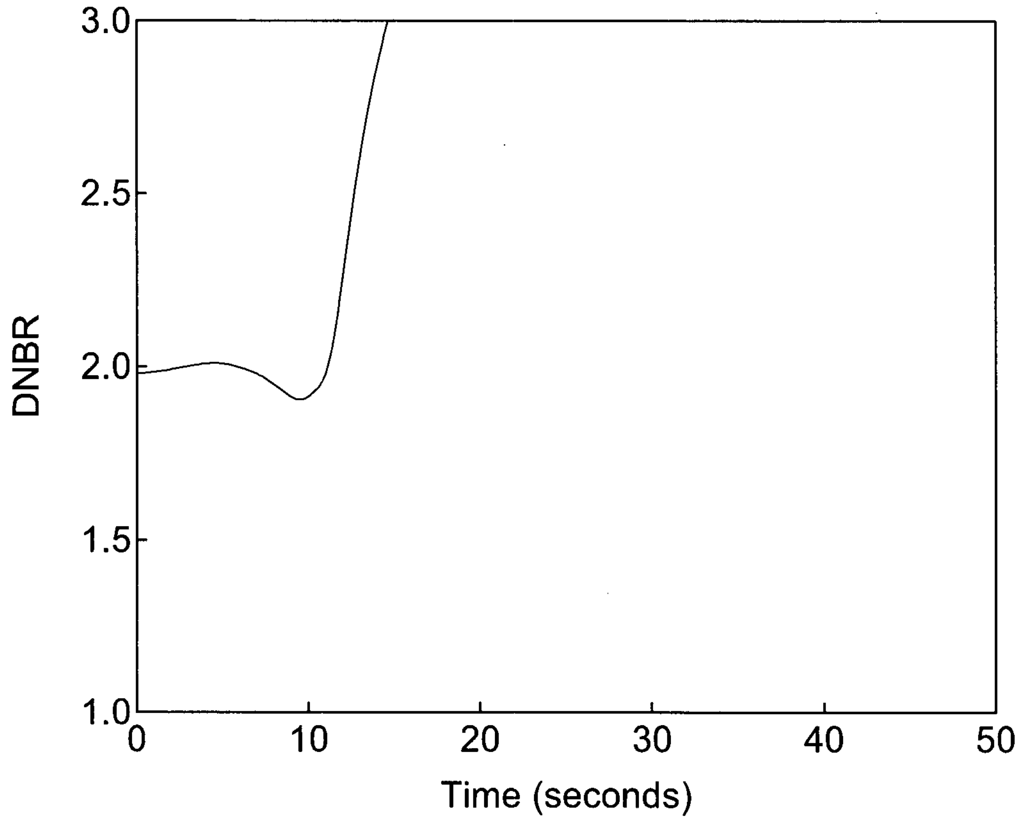

Figure 15.2.1-1 **DNBR versus Time**

Loss of External Load~~/Turbine Trip Transient~~ - DNBR Analysis DCD_15-33

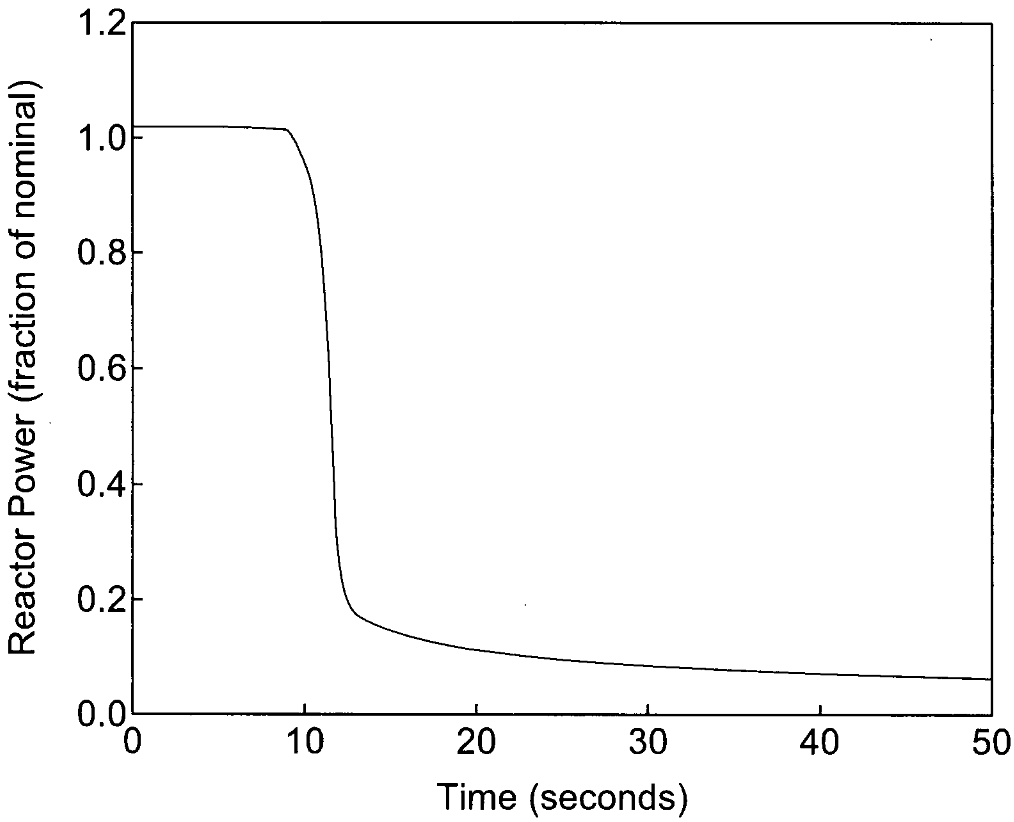

Figure 15.2.1-2 Reactor Power versus Time

Loss of External Load/~~Turbine Trip Transient~~ - RCS ~~& Main Steam~~ Pressure Analysis DCD_15-33

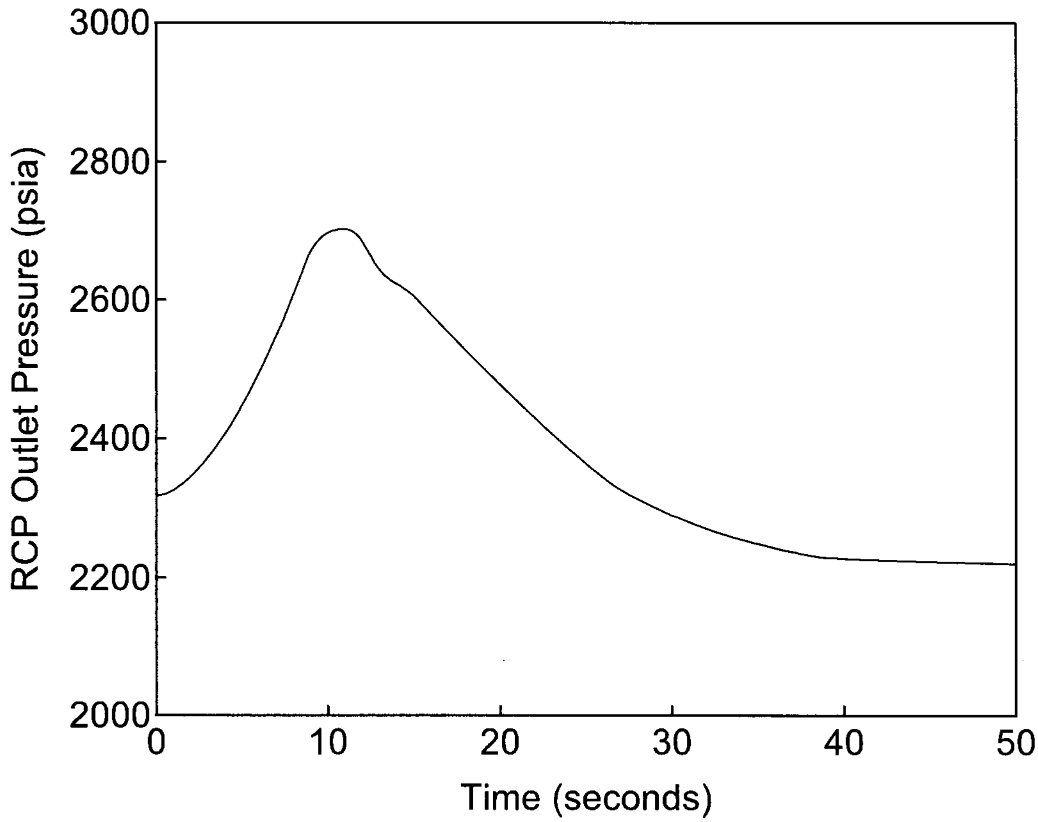

Figure 15.2.1-3 RCP Outlet Pressure versus Time

Loss of External Load~~/Turbine Trip Transient~~
- RCS ~~& Main Steam~~ Pressure Analysis

DCD_15-33

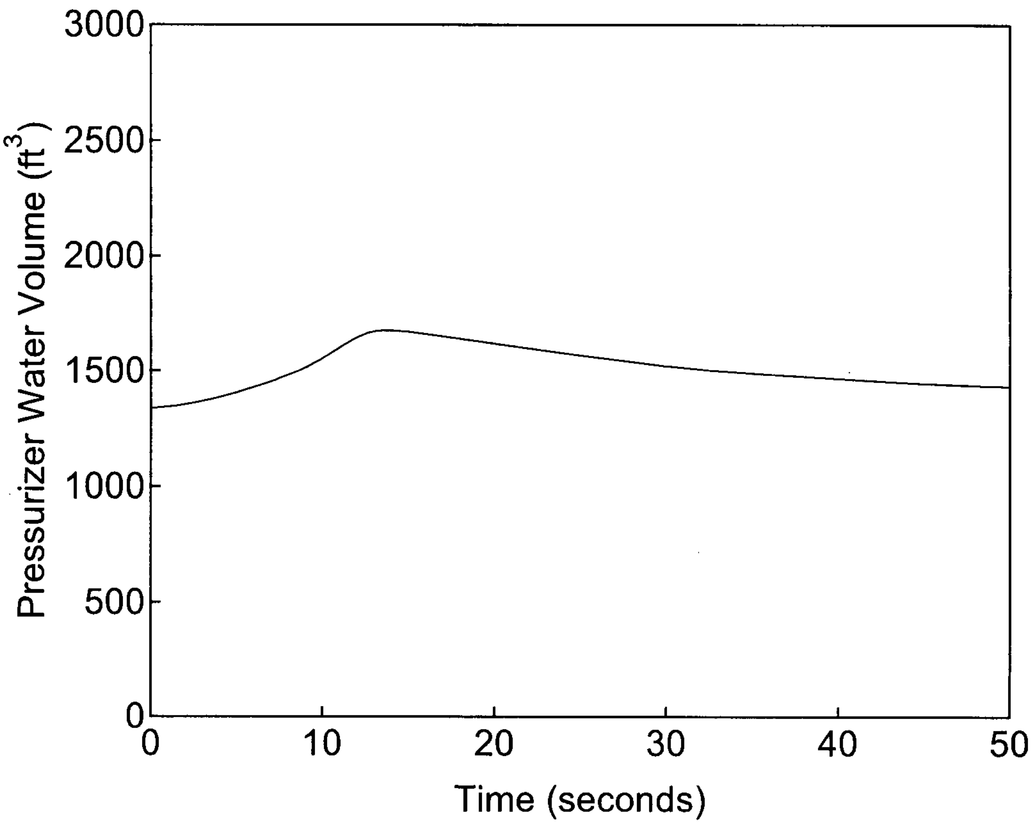

Figure 15.2.1-4 **Pressurizer Water Volume versus Time**

Loss of External Load~~/Turbine Trip Transient~~ - RCS ~~& Main Steam~~ Pressure Analysis DCD_15-33

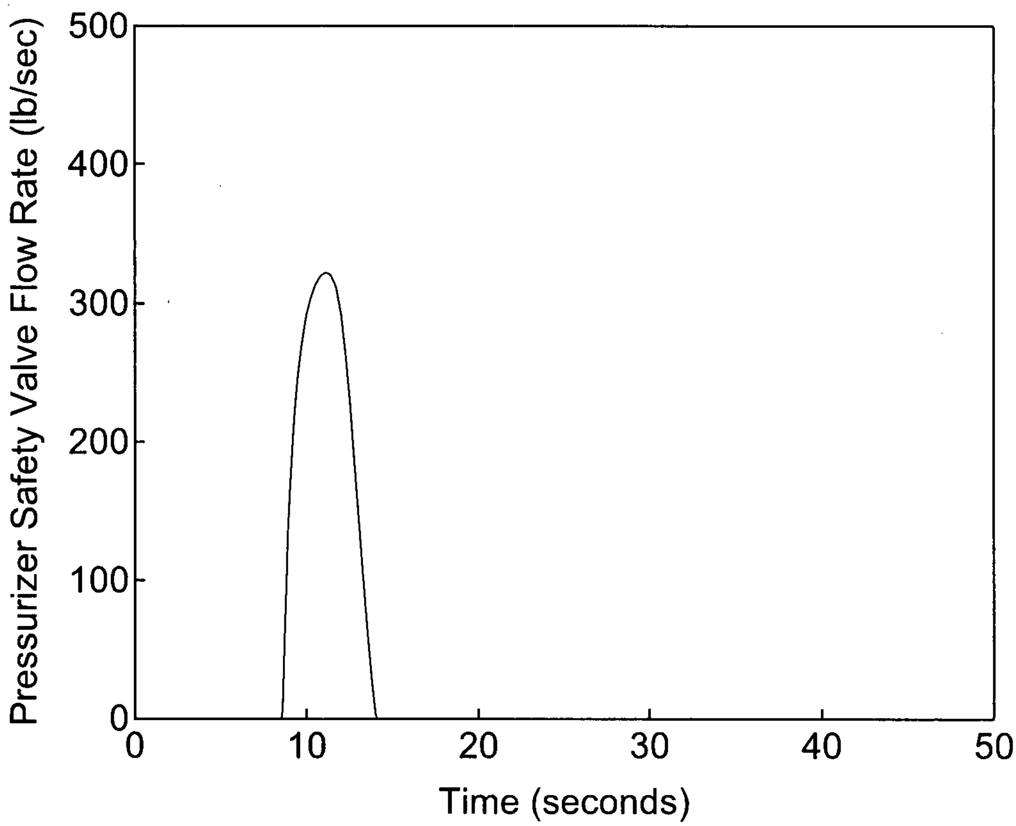

Figure 15.2.1-5 **Pressurizer Safety Valve Flow Rate versus Time**

Loss of External Load~~/Turbine Trip Transient~~ - RCS ~~& Main Steam~~ Pressure Analysis

DCD_15-33

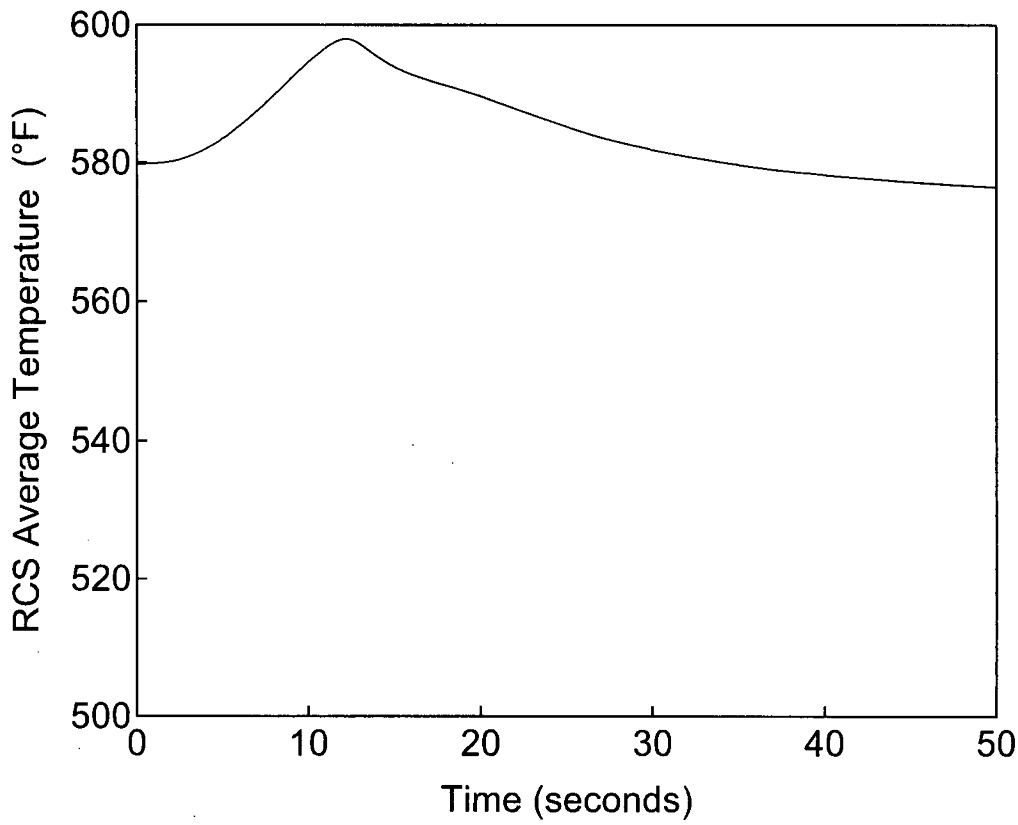

Figure 15.2.1-6 RCS Average Temperature versus Time

Loss of External Load~~/Turbine Trip Transient~~ - RCS ~~& Main Steam~~ Pressure Analysis

DCD_15-33

DCD_15-33

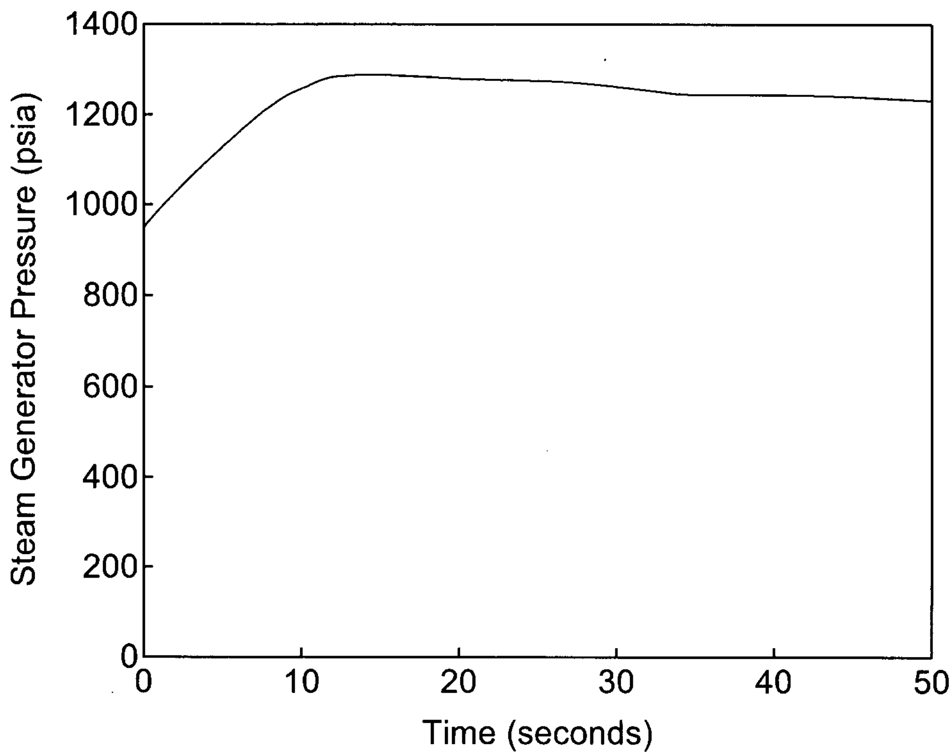

Figure 15.2.1-7 Steam Generator Pressure versus Time

Loss of External Load~~/Turbine Trip Transient~~ - ~~RCS &~~ 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. 809-5957 REVISION 3**

SRP SECTION: **15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**

APPLICATION SECTION: **15.0**

DATE OF RAI ISSUE: **8/22/2011**

QUESTION NO.: 15-34

As seen in Table 15.0.0-10.2 (from response to RAI 2287, Question 15.0.0-10), the Chapter 15 transient analyses models set the main steam safety valves (MSSV) to open at 1236 psia (1221.3 psig). In order to find that input to the Chapter 15 safety analyses is consistent with the range of values specified in the technical specifications (TS), justify the selection of a setpoint that does not bound all the MSSV lift settings defined in TS 3.7.1 (1185-1244 psig plus 1% uncertainty)

ANSWER:

As shown in Figure 15-34.1 below, a simplified, one setpoint MSSV model is used in the Chapter 15 analyses while the actual MSSVs have three different setpoints. The reason why the simplified one setpoint MSSV model is used is that even for the limiting secondary pressure events like the loss of load transient, the valves are sized for 100% steam flow capacity and therefore no over pressurization is expected. In other words, even if using a simplified MSSV model, if the MSSV total steam release rate is less than the maximum capacity, the secondary maximum pressure is assured to remain less than the limit of 1320 psia. If a more detailed MSSV model is used, the actual peak steam pressure may change, but will still be well under the safety analysis limit because the maximum MSSV flow rate is less than 100%. MHI performed a sensitivity analysis for the loss of external load (LOL) event using the detailed three setpoint MSSV model (plus uncertainty). The results of the sensitivity analysis are provided in Figures 15-34.2 through 15-34.4.

The results show that the impact on minimum DNBR and RCS pressure is negligible. The peak RCS pressure decreases slightly[ ]due to the earlier opening of the MSSV with the lowest setpoint. On the other hand, the peak secondary side pressure increases by [ ] Although the sensitivity case shows higher SG pressures than the DCD case, these results still have margin to the limit of 1320 psia.

Figure 15-34.1 MSSV Setpoint Model

Figure 15-34.2 DNBR versus Time
Loss of External Load - DNBR Analysis
MSSV Setpoint Sensitivity

Figure 15-34.3 RCP Outlet Pressure versus Time
Loss of External Load – RCS & Main Steam Pressure Analysis
MSSV Setpoint Sensitivity

Figure 15-34.4 Steam Generator Pressure versus Time
Loss of External Load – RCS & Main Steam Pressure Analysis
MSSV Setpoint Sensitivity