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

June 16, 2009

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09299

Subject: MHI's Response to US-APWR DCD RAI No. 301-2324 Revision 1

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. 301-2324 Revision 1". The enclosed materials provide MHI's response to the NRC's "Request for Additional Information (RAI) 301-2324 Revision 1," dated April 2, 2009.

As indicated in the enclosed materials, this document contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted 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. 301-2324 Revision 1 (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 301-2324 Revision 1 (non-proprietary)

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

<u>Contact Information</u>

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-09299

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. 301-2324 Revision 1" dated June 16, 2009, 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 all 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 16th day of June, 2009.

Y. Ogata

Yoshiki Ogata

Enclosure 3

UAP-HF-09299
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 301-2324 Revision 1

June 2009

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/16/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 301-2324 REVISION 1**
SRP SECTION:	**15.01.01 – 15.01.04 – DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE**
APPLICATION SECTION:	**15.1.1 – 15.1.4**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.1-1

In DCD Sections 15.1.2 and 15.1.5, the applicant presents the following argument concerning the timing of a LOOP with respect to the timing of reactor or turbine trip from at-power conditions and the subsequent RCP coast down. It is used as justification for neglecting LOOP for these events with respect to the minimum DNBR and is based upon the discussion in DCD Section 15.0.0.7: "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 DCD 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, the case where offsite power is unavailable is not presented." The applicant should present additional technical justification as to why minimum DNBR always occurs before the LOOP, why cases where offsite power is unavailable are not presented, as well as technical justification as to why the remaining duration of the affected transients are similarly unaffected by the loss of offsite power.

ANSWER:

A sensitivity analysis concerning the US-APWR LOOP assumptions and their supporting bases will be described in detail in the response that will be submitted for RAI 297-2287 Question 15.0.0-3. Figure 15.1-1.1 below provides the transient DNBR curve for the DCD Subsection 15.1.2 event (increase in feedwater flow) considering a LOOP, in which the reactor coolant pump coastdown is delayed 3 seconds after the turbine trip (turbine trip is assumed to occur at the same time as reactor trip). For comparison, the curve without LOOP is provided on the same figure. For the main steam line break event described in Subsection 15.1.5, cases are presented for HZP double-ended breaks both with and without LOOP (Cases A & B) and for a spectrum of breaks at HFP (Case C). The transient DNBR curve for the limiting Case C break with LOOP, in which the reactor coolant pump coastdown is delayed 3 seconds after the turbine trip (turbine trip is assumed to occur at the same time as reactor trip), is shown below in Figure 15.1-1.2. Again, the case without LOOP is also provided on the figure for comparison purposes. For the two DNBR figures shown in this response, the results are generated using the

MARVEL-M/VIPRE-01M methodology rather than the MARVEL-M lookup table methodology. Both these methodologies are described in detail in the Non-LOCA Methodology Topical Report (MUAP-07010). Since it was necessary to use the MARVEL-M/VIPRE-01M methodology for the LOOP case due to the flow coastdown, the same methodology was used for the without LOOP case for consistency. As previously described, a more detailed discussion of the bases for these two figures will be provided in the response to RAI 297-2287 Question 15.0.0-3.

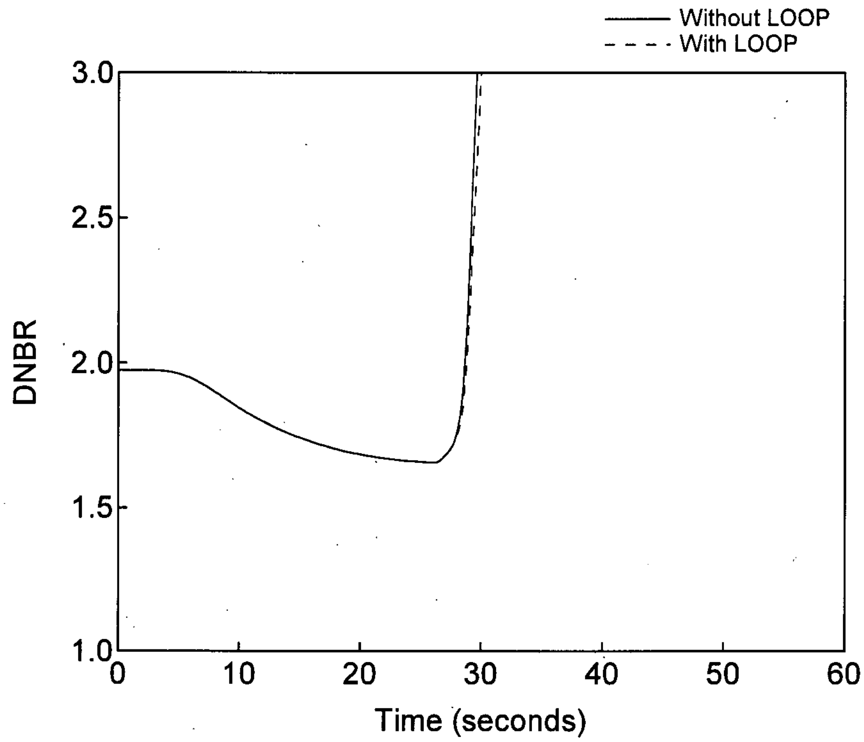

Figure 15.1-1.1 **DNBR versus Time with and without LOOP**
Increase in Feedwater Flow

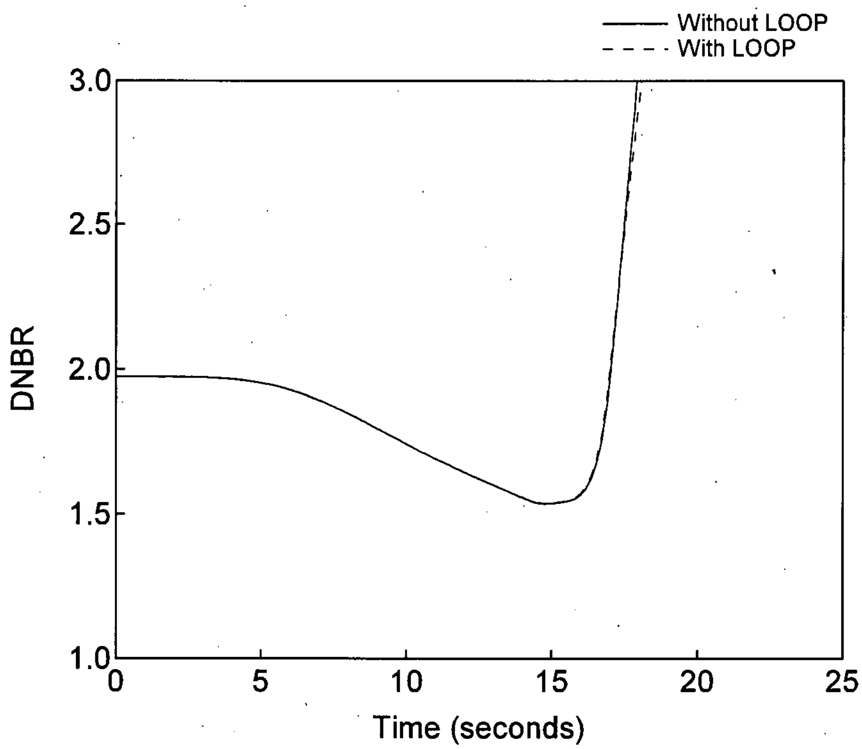

Figure 15.1-1.2 **DNBR versus Time with and without LOOP**
Steam System Piping Failure
- Case C: Limiting case for spectrum of breaks at 100% power

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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APPLICATION SECTION:	**15.1.1 – 15.1.4**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.1-2

In DCD Section 15.1.2, Increase in Feedwater Flow, verify that the valves that are closed by the feedwater isolation system (DCD pg.15.1-14, paragraph 7) are safety-related. Discuss the effect of LOOP on the timing of the main feedwater isolation valve closures.

ANSWER:

For the event described in DCD Subsection 15.1.2, the high-high steam generator water level signal trips the reactor and actuates a feedwater isolation signal. The feedwater isolation signal closes all the main feedwater bypass regulation valves, trips all main feedwater pumps, closes all main feedwater isolation valves, closes all steam generator water filling valves (already closed during this event), and closes all main feedwater regulation valves. The main feedwater isolation valves are safety-related, seismic category I, ASME Code, Section III, Class 2. The other valves are seismic category I, ASME Code, Section III, Class 3. These safety-related valves are described in more detail in DCD Subsection 10.4.7.2.2. All of these valves use pressurized fluid to maintain the valves in an open position. When either of the redundant solenoids, supplied from a class 1E DC battery source, are energized, the pressurized fluid is vented, and the valves close rapidly. All of these valves will close without relying on AC power. Therefore, the timing of the LOOP does not affect the feedwater isolation function.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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APPLICATION SECTION:	**15.1.1 – 15.1.4**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.1-3

Provide transient curves for steam generator pressure verses time in Sections 15.1.1 through 15.1.3.

ANSWER:

The transient curve for steam generator pressure versus time for the decrease in feedwater temperature analysis in DCD Subsection 15.1.1 is shown below in Figure 15.1-3.1. The transient curve for steam generator pressure versus time for the increase in feedwater flow analysis in DCD Subsection 15.1.2 is shown below in Figure 15.1-3.2. The transient curves for steam generator pressure versus time for the increase in steam flow analysis in DCD Section 15.1.3 for Cases A through D are shown below in Figures 15.1-3.3 through 15.1-3.6, respectively. These steam generator pressure figures confirm the statement in Subsections 15.1.1.3.3, 15.1.2.3.3, and 15.1.3.3.3 that steam line pressure is not a key parameter for heat removal events where steam generator pressures are stable or decreasing.

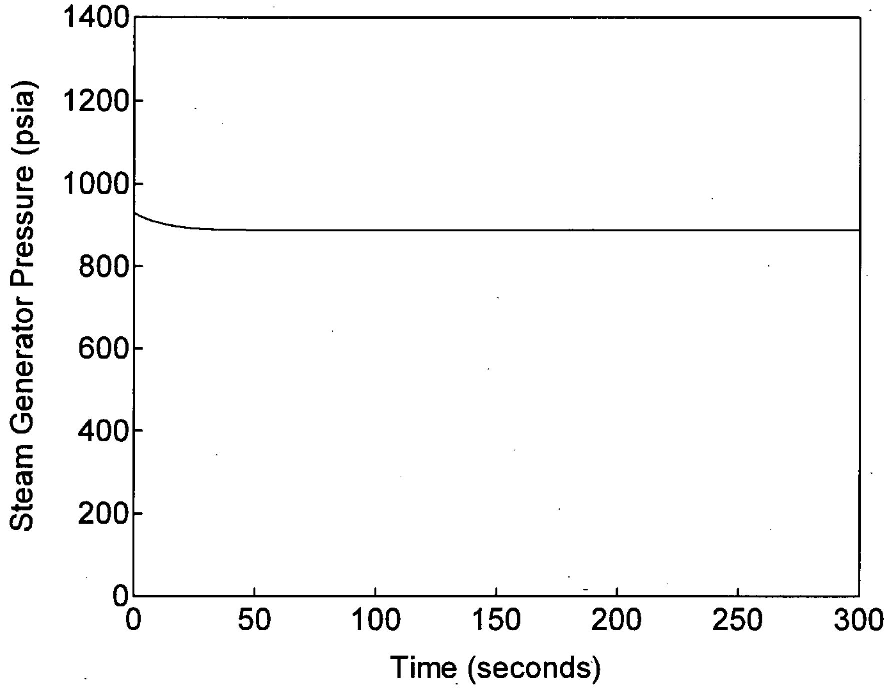

Figure 15.1-3.1 **Steam Generator Pressure versus Time Decrease in Feedwater Temperature**

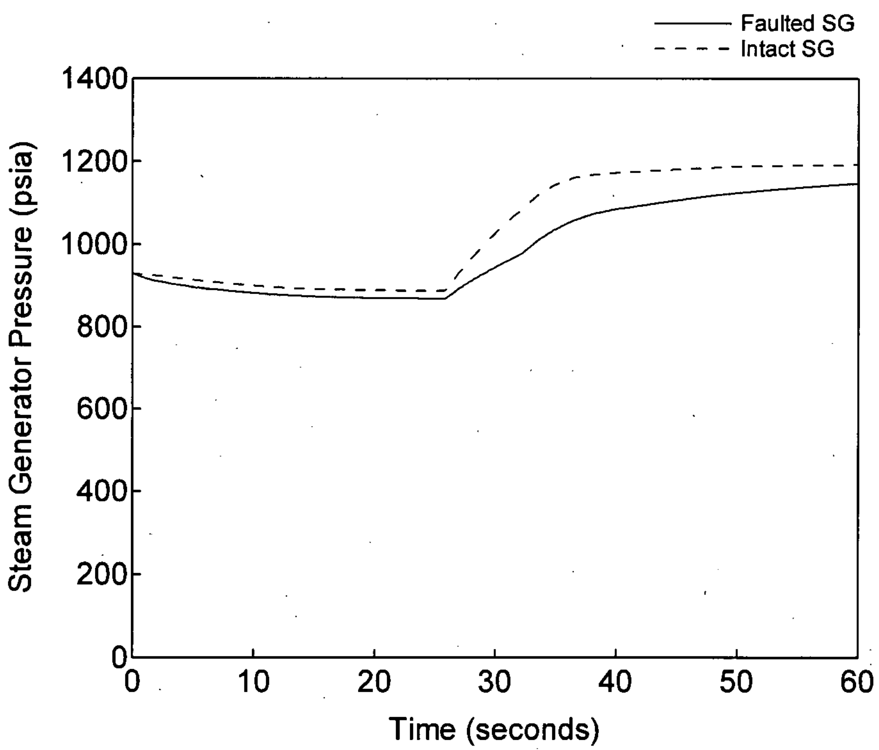

Figure 15.1-3.2 **Steam Generator Pressure versus Time Increase in Feedwater Flow**

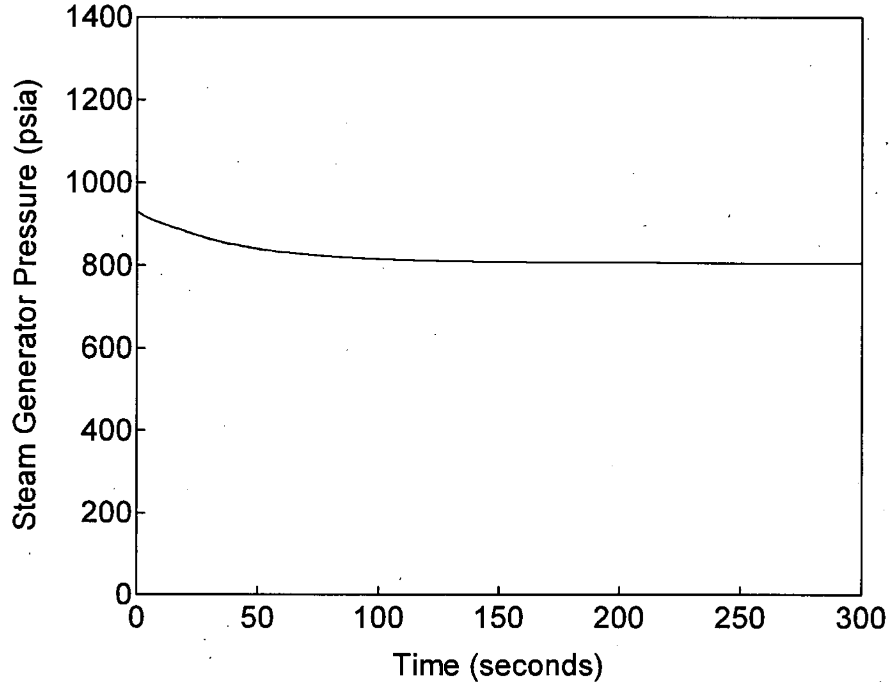

Figure 15.1-3.3 **Steam Generator Pressure versus Time**
Increase in Steam Flow
- Case A: Manual Rod Control, Minimum Moderator Feedback

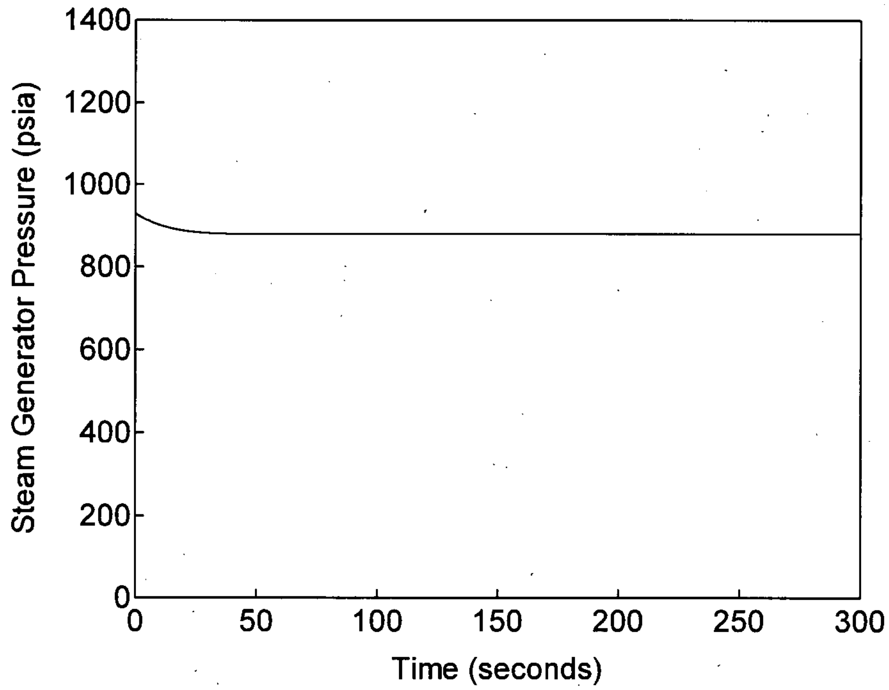

Figure 15.1-3.4 **Steam Generator Pressure versus Time**
Increase in Steam Flow
- Case B: Manual Rod Control, Maximum Moderator Feedback

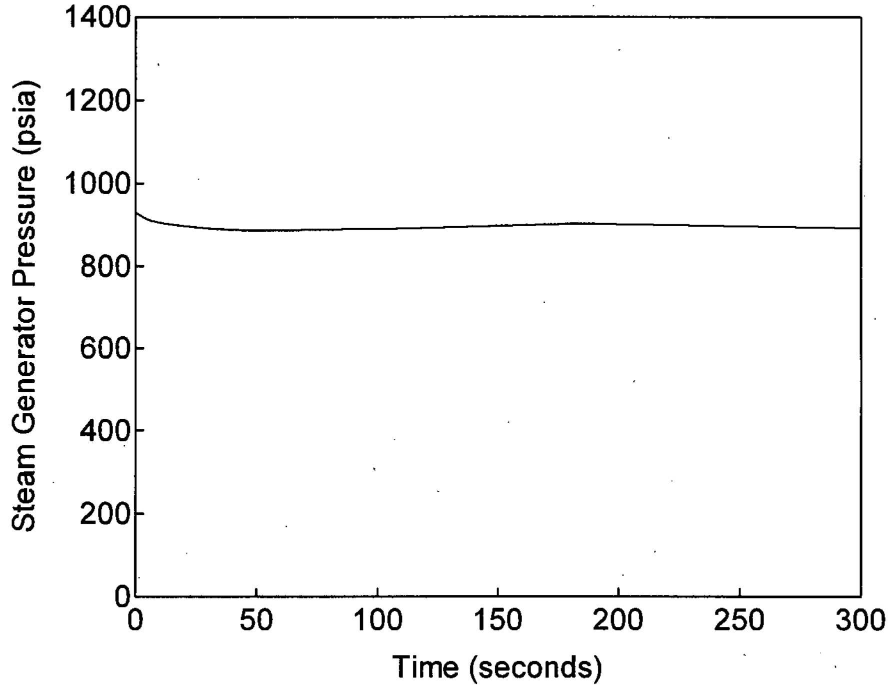

Figure 15.1-3.5 **Steam Generator Pressure versus Time**
Increase in Steam Flow
- Case C: Automatic Rod Control, Minimum Moderator Feedback

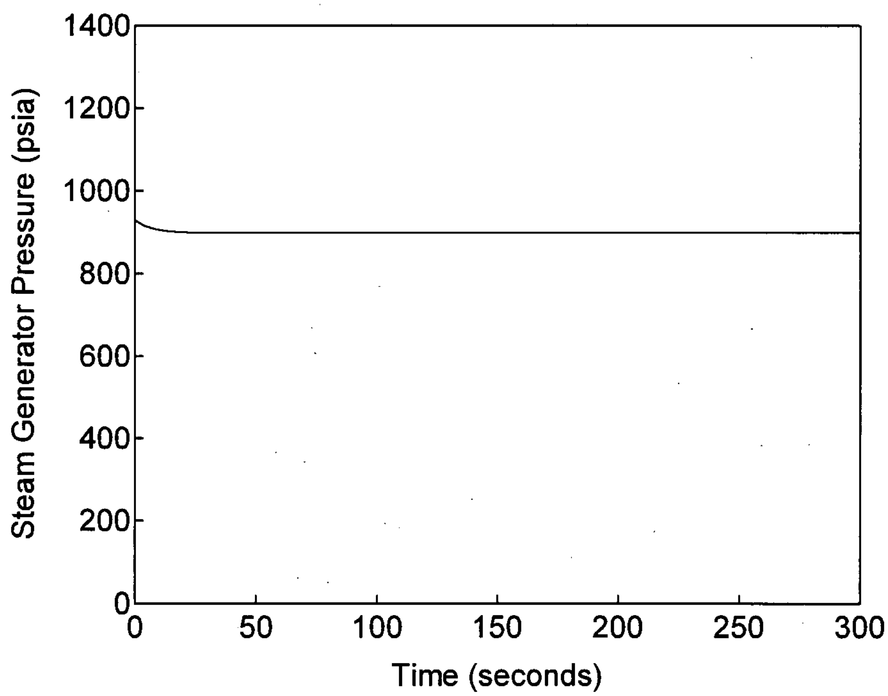

Figure 15.1-3.6 **Steam Generator Pressure versus Time**
Increase in Steam Flow
- Case D: Automatic Rod Control, Maximum Moderator Feedback

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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APPLICATION SECTION:	**15.1.1 – 15.1.4**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.1-4

Provide transient curve for DNBR verses time in Section 15.1.4.

ANSWER:

The transient curve for DNBR versus time for the analysis in Subsection 15.1.4 is shown below in Figure 15.1-4.1. These results are based on the steady state evaluation described in the Non-LOCA Methodology Topical Report (MUAP-07010), Section 5.4, "Method of Analysis", (b) DNBR calculation.

Figure 15.1-4.1 **DNBR versus Time**
Inadvertent Opening of a Steam Generator Relief or Safety Valve

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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APPLICATION SECTION:	**15.1.1 – 15.1.4**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.1-5

In Case B, Double-ended Steam Line Break from hot standby without offsite power, the RCPs are assumed to begin coast down at the time of ECCS actuation. Provide the basis for this assumption.

ANSWER:

For the steam line break at hot zero power, the turbine-generator is already disconnected before the accident. Therefore, there will not be any disturbance to the electrical grid, so there is no need to account for a LOOP from this point of view.

However, a LOOP is assumed in Case B, coincident with the ECCS actuation signal, considering that the US-APWR has logic such that an ECCS actuation signal will cause an automatic trip of the reactor coolant pumps (RCPs). This sequence of events is described in DCD Subsection 15.1.5.2. The RCP trip logic is shown in detail in sheet 11 of DCD Figure 7.2-2.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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APPLICATION SECTION:	**15.1.1 – 15.1.4**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.1-6

In Case C, Spectrum of Steam Break from Power Operation, provide an analysis that considers LOOP.

ANSWER:

The basis for the effect of LOOP for Case C presented in DCD Subsection 15.1.5 (steam line break at power) will be covered by the response to RAI 297-2287 Question 15.0.0-3. The transient DNBR curve for the limiting Case C break with LOOP, in which the reactor coolant pump coastdown is delayed 3 seconds after the turbine trip (turbine trip is assumed to occur at the same time as reactor trip), has been provided in the response to Question 15.1-1 of this RAI.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.