MITSUBISHI HEAVY INDUSTRIES, LTD.

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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-09301

Subject: MHI's Response to US-APWR DCD RAI No. 306-2333 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. 306-2333 Revision 1". The material in Enclosure 1 provides MHI's response to the NRC's "Request for Additional Information (RAI) 306-2333 Revision 1," dated April 2, 2009.

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.

Enclosures:

1. MHI's Response to US-APWR DCD RAI No. 306-2333 Revision 1 (non-proprietary)

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

Contact Information

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ENCLOSURE 1

UAP-HF-09301
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 306-2333 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. 306-2333 REVISION 1**
SRP SECTION:	**15.03.01 – 15.03.02 – LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP MOTOR AND FLOW CONTROLLER MALFUNCTIONS**
APPLICATION SECTION:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-1

SRP Sections 15.3.1-2 and 15.3.3-4 state, "For new applications, LOOP should not be considered a single failure; [all AOOs] should be analyzed with and without LOOP in combination with a single active failure." In compliance with this requirement, provide the results of calculations that include the occurrence of a LOOP unless otherwise exempted by the NRC staff. If exempted by the NRC, provide documentation of the exemption.

ANSWER:

For the event in DCD Subsection 15.3.1.1 (partial loss of forced reactor coolant flow), the basis for the limiting DNBR occurring prior to the RCP coastdown (i.e. the LOOP analysis is not necessary) will be provided in the response to RAI 297-2287 Question 15.0.0-3. However, the transient curve for DNBR for the 15.3.1.1 analysis considering a LOOP occurs 3 seconds after the turbine trip (turbine trip is assumed to occur at the same time as reactor trip) which causes the other two RCPs to coast down is shown below in Figure 15.3.1-1.1. For comparison, the same event without LOOP is shown on the same curve. For the event in DCD Subsection 15.3.1.2 (complete loss of forced reactor coolant flow), all of the RCPs trip as the initiating event of this transient, so it is not necessary to consider a LOOP. For the RCP rotor seizure event presented in DCD Subsection 15.3.3, the basis for not considering LOOP will also be covered by the response for RAI 297-2287 Question 15.0.0-3. However, the transient curve for cladding inside temperature for the 15.3.3 analysis considering a LOOP occurs 3 seconds after turbine trip (turbine trip is assumed to occur at the same time as reactor trip) which causes the other three RCPs to coast down is shown below in Figure 15.3.1-1.2. Again, the event without LOOP is shown on the same figure for comparison.

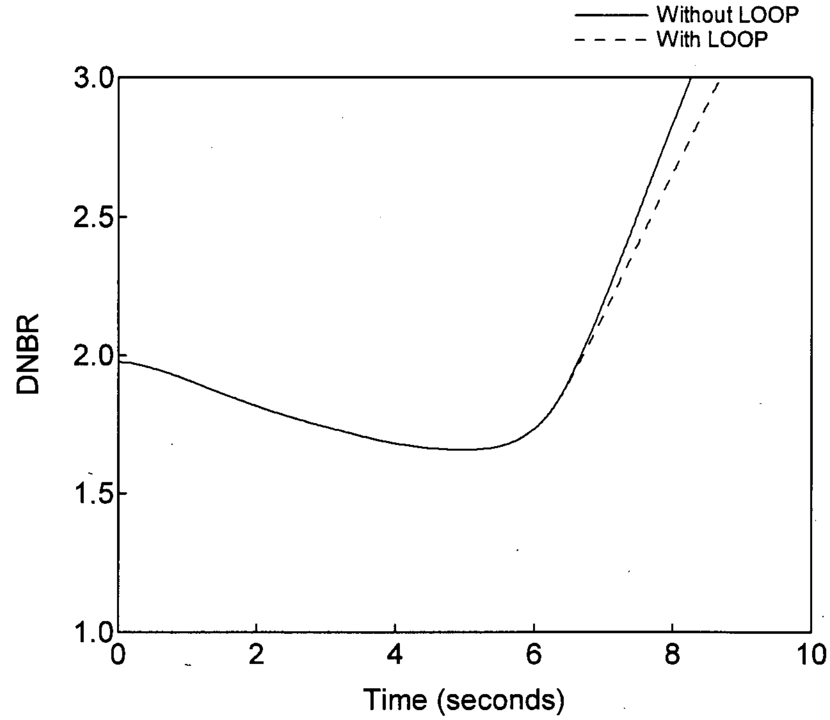

Figure 15.3.1-1.1 **DNBR versus Time with and without LOOP**
Partial Loss of Forced Reactor Coolant Flow

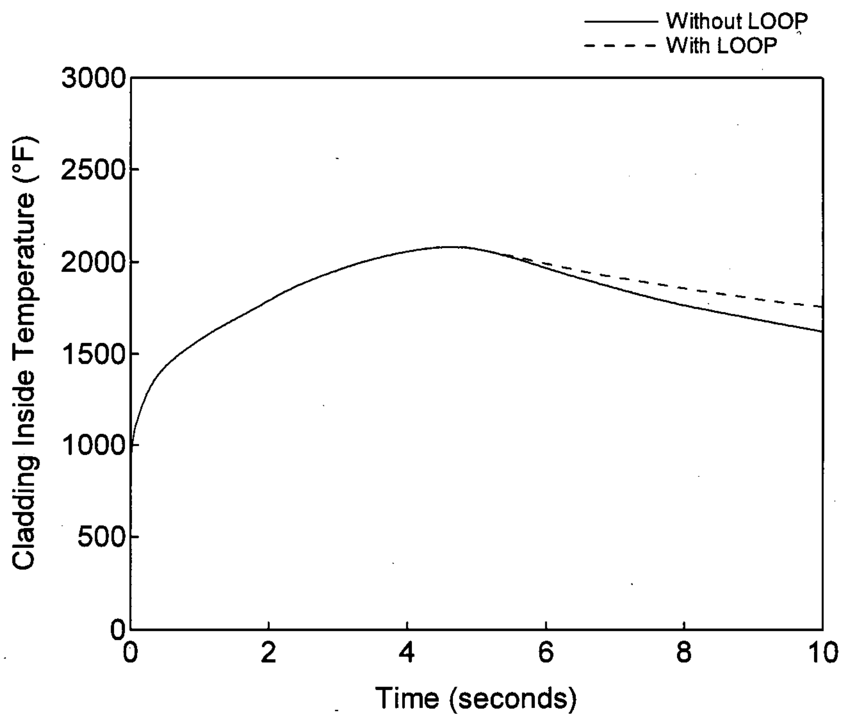

Figure 15.3.1-1.2 **Cladding Inside Temperature versus Time with and without LOOP**
RCP Rotor Seizure
- Cladding Temperature Analysis

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:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-2

In DCD Section 15.3.1.1, Partial Loss of Forced Reactor Coolant Flow, it is assumed in the analysis that two of the four RCPs trip at the same time to initiate the RCS coolant flow transient. Which two pumps are assumed to trip, pumps in adjacent loops or pumps in opposite loops? Are the calculated results for minimum DNBR sensitive to this parameter? Is the minimum DNBR sensitive to the mixing factors FMXI and FMXO assumed for this transient? What values for FMXI and FMXO were used in the calculations? When does the LOOP occur and when do the other two RCPs trip?

ANSWER:

The partial loss of forced reactor coolant flow in DCD Subsection 15.3.1.1 assumes that two of the four reactor coolant pumps begin to coast down at the same time. In the analysis, which two pumps are tripped has no effect on the minimum DNBR because a constant RCS pressure and inlet temperature are used in the DNBR calculation for conservatism. For this same reason, the inlet and outlet mixing factors also have no effect on the minimum DNBR. The values of FMXI and FMXO were provided in the response to RAI 2.1-13-1 on the Non-LOCA Methodology Topical Report (MUAP-07010), which was submitted to the NRC by MHI letter UAP-HF-09040-P (R0), dated February 12, 2009. As described in Subsection 15.3.1.1.2, the analysis did not originally consider a LOOP that would cause the coastdown of the other two RCPs. The basis of the assumptions concerning LOOP will be provided in the response to RAI 297-2287 Question 15.0.0-3. The transient curve for DNBR, assuming that the other two RCPs coast down 3 seconds after the turbine trip (turbine trip is assumed to occur at the same time as reactor trip), is provided in the response to Question 15.3.1-1 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.

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APPLICATION SECTION:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-3

In DCD Section 15.3.1.2, Complete Loss of Forced Reactor Coolant Flow, can the minimum DNBR be influenced by the mixing factors FMXI and FMXO assumed for this transient, or is the calculation insensitive to these parameters due to the symmetry of the flow transient in the core? What values for FMXI and FMXO were used in the calculations?

ANSWER:

The complete loss of forced reactor coolant flow in DCD Subsection 15.3.1.2 assumes that all 4 reactor coolant pumps begin to coast down at the same time. Therefore, the flow is symmetric during the transient and the inlet and outlet mixing factors have no effect on the minimum DNBR. The values of FMXI and FMXO were provided in the response to RAI 2.1-13-1 on the Non-LOCA Methodology Topical Report (MUAP-07010), which was submitted to the NRC by MHI letter UAP-HF-09040-P (R0), dated February 12, 2009.

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:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-4

Provide the transient curve for Steam Generator Pressure verses time in the analysis for Loss of Flow event in Section 15.3.1.

ANSWER:

The transient curves for the steam generator pressure versus time for the analyses in DCD Subsection 15.3.1, for partial loss of flow, complete loss of flow, and frequency decay resulting in a complete loss of flow, are shown below in Figures 15.3.1-4.1 through 15.3.1-4.3, respectively. The SG safety valve is modeled assuming 103% of the SG design pressure (1236 psia). These steam generator pressure figures confirm the statement in Subsections 15.3.1.1.3.3 and 15.3.1.2.3.3 that the steam line pressure for these events are bounded by that of the RCP rotor seizure event which is less than 110% of the design pressure.

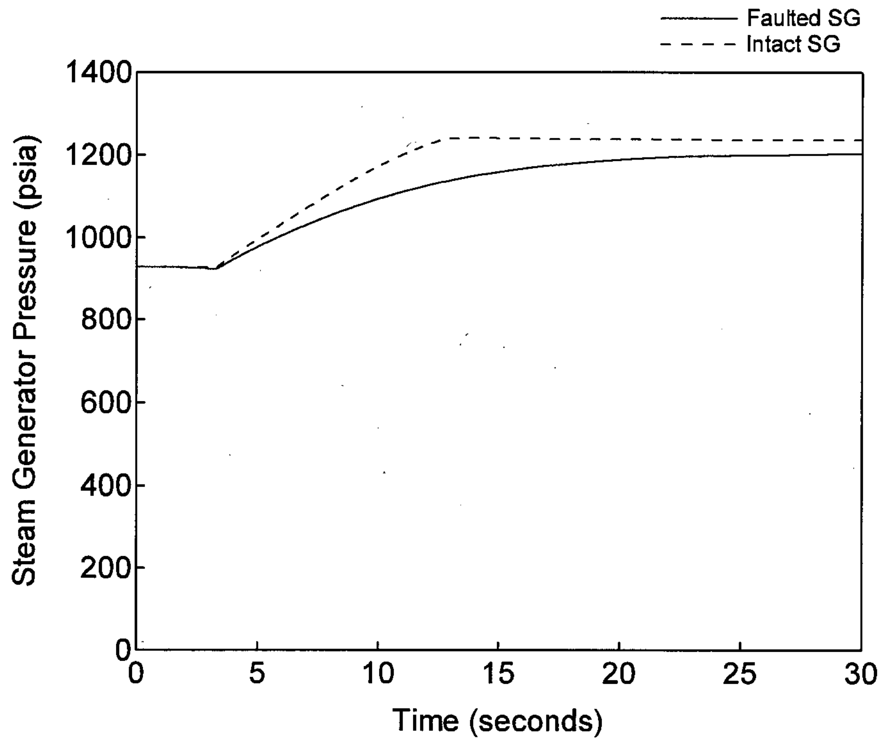

Figure 15.3.1-4.1 **Steam Generator Pressure versus Time Partial Loss of Forced Reactor Coolant Flow**

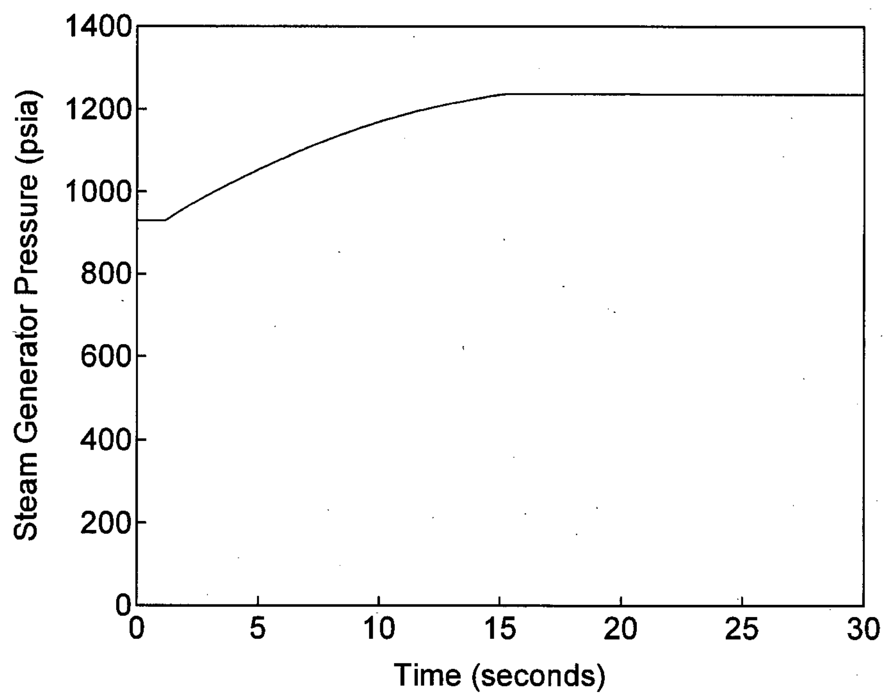

Figure 15.3.1-4.2 **Steam Generator Pressure versus Time Complete Loss of Forced Reactor Coolant Flow**

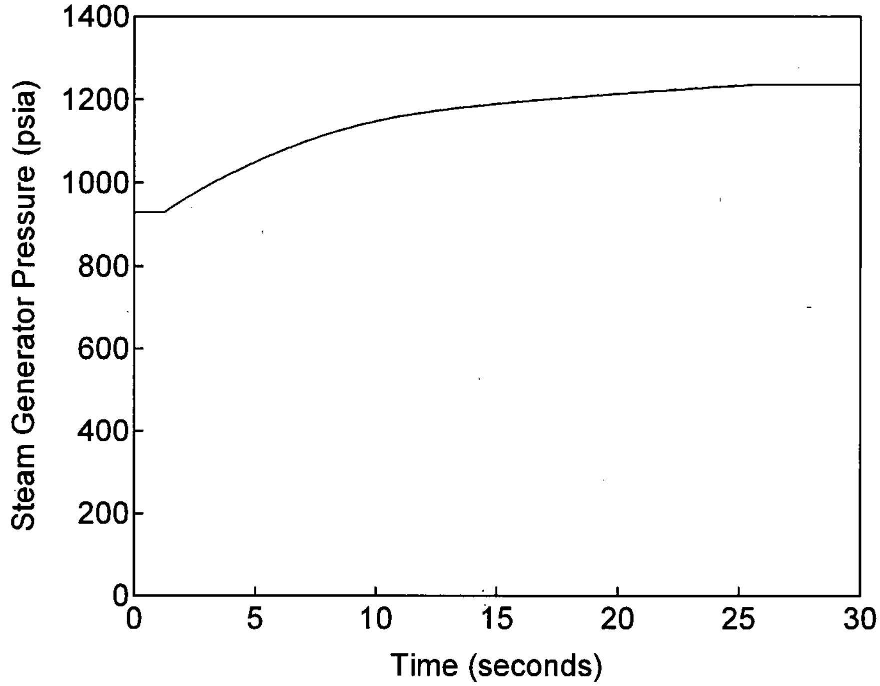

Figure 15.3.1-4.3 **Steam Generator Pressure versus Time Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow**

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:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-5

Confirm that the transient curve for RCS total flow verses time In FSAR Section 15.3.1 (Figure 15.3.1.2-1) is consistent with the RCP coast down characteristics.

ANSWER:

The RCS total flow versus time shown in DCD Figure 15.3.1.2-1 is consistent with the RCP coastdown characteristics. The RCP coastdown characteristics were verified by comparison to actual data from a typical 4-loop plant. This comparison was described in the response to RAI 2.1-16 on the Non-LOCA Methodology Topical Report, which was submitted to the NRC by MHI letter UAP-HF-08170-P (R0), dated September 12, 2008. The RCP coastdown in DCD Figure 15.3.1.2-1 is slightly more severe than that shown in the comparison in that RAI response due to the conservative assumption about the RCP moment of inertia described in DCD Subsection 15.3.1.1.3.2.

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:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-6

Provide the RCP coast-down flow characteristics used for determining the mass flow rate in the primary reactor coolant recirculation loops for the reactor transients in Sections 15.3.1 through 15.3.4. Discuss how those characteristics were determined and explain any assumptions made.

ANSWER:

The characteristics of the RCP coastdown flow model were discussed in detail in Section 2.1.3.3 of the Non-LOCA Methodology Topical Report, MUAP-07010, dated July 2007. This information was further described in the responses to the following RAI Questions regarding the topical report:

- 2.1-7 (MHI letter UAP-HF-080141-P (R0), dated August 22, 2008)
- 2.1-16 (MHI letter UAP-HF-08170-P (R0), dated September 12, 2008)
- 2.1-16-1 and 2.1-16-2 (MHI letter UAP-HF-09040-P (R0), dated February 12, 2009)

In addition to the characteristics described in those locations, the partial and complete loss of flow analyses in DCD Subsection 15.3.1 conservatively assume that the moment of inertia of the RCP is 90% of the design value. This assumption is described in DCD Subsection 15.3.1.1.3.2.

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:	**15.3.1 – 15.3.2**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-7

Plot the peak fuel centerline temperature as a function of time for the reactor transients in Sections 15.3.1 through 15.3.4 and explain the associated safety limit.

ANSWER:

The plots of the peak fuel centerline temperature as a function of time for the reactor transients in DCD Subsection 15.3.1, partial loss of flow, complete loss of flow, and frequency decay resulting in a complete loss of flow, are shown below in Figures 15.3.1-7.1 through 15.3.1-7.3, respectively. As described in DCD Subsection 15.3.2, the flow controller malfunction event is not applicable to the US-APWR and is not analyzed. The plot of the peak fuel centerline temperature as a function of time for the reactor transient in DCD Subsection 15.3.3, the RCP rotor seizure, is shown below in Figure 15.3.1-7.4. This case conservatively assumes the film boiling coefficient from the beginning of the accident because DNB occurs in the RCP rotor seizure event as described in DCD Subsection 15.3.3.3.2. As described in DCD Subsection 15.3.4, the RCP shaft break event is bounded by the conservative analysis in DCD Subsection 15.3.3.

The US-APWR applies the following criterion as the safety limit in the fuel centerline temperature analyses for both AOOs and PAs:

- The maximum fuel centerline temperature should be less than the fuel melting point so that the fuel cladding will not be mechanically damaged.

The fuel pellet melting temperature is described in DCD Subsection 4.2.1.2.1 and Table 4.1-1 (sheet 2 of 3). The melting temperature is 5072°F (2800°C) for un-irradiated uranium dioxide (UO_2) fuel, and decreases with burnup by 58°F (32°C) per 10 GWD/MTU. Maximum fuel centerline temperature during AOOs is 4620°F which is decreased by a fuel temperature uncertainty of 452°F (about 250°C).

In the Chapter 15 analyses concerning fuel centerline temperature, 4620°F is used as the safety limit for AOOs. In case of the PAs, a fuel temperature uncertainty is considered in advance by including it in the initial fuel temperature. The safety limit is based on the melting temperature considering only fuel burnup effects.

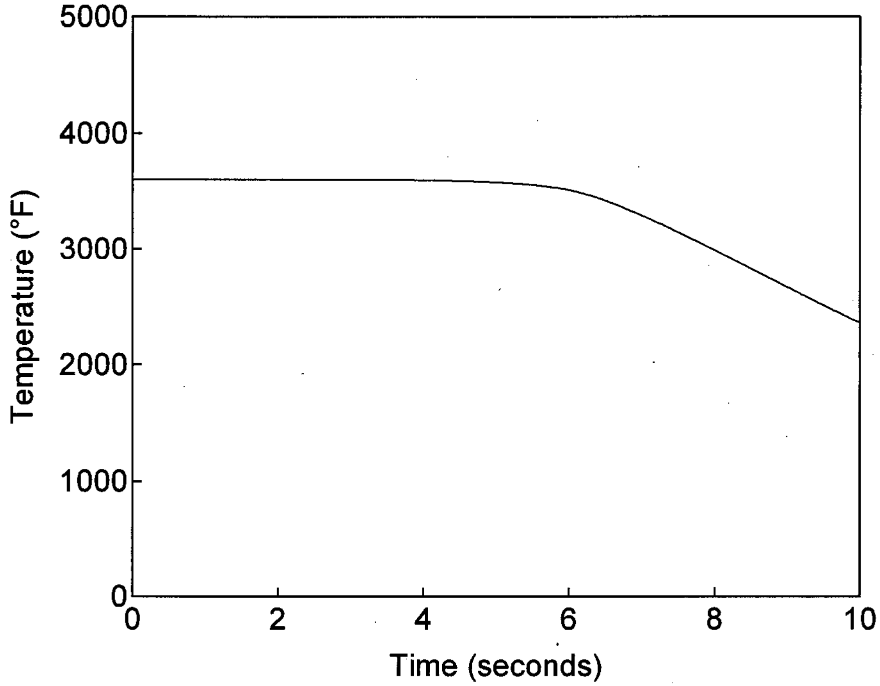

Figure 15.3.1-7.1 **Fuel Centerline Temperature versus Time Partial Loss of Forced Reactor Coolant Flow**

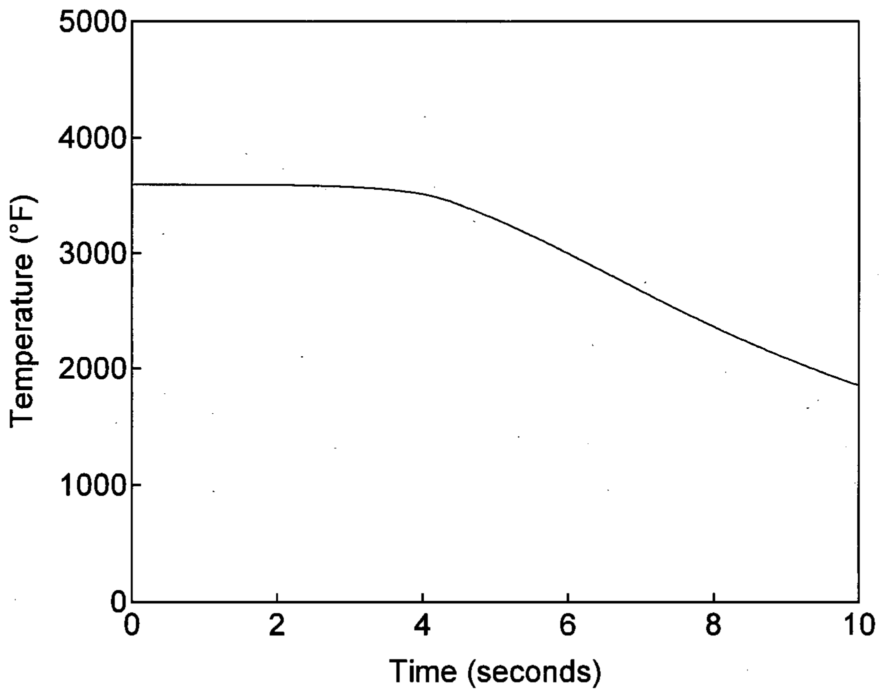

Figure 15.3.1-7.2 **Fuel Centerline Temperature versus Time Complete Loss of Forced Reactor Coolant Flow**

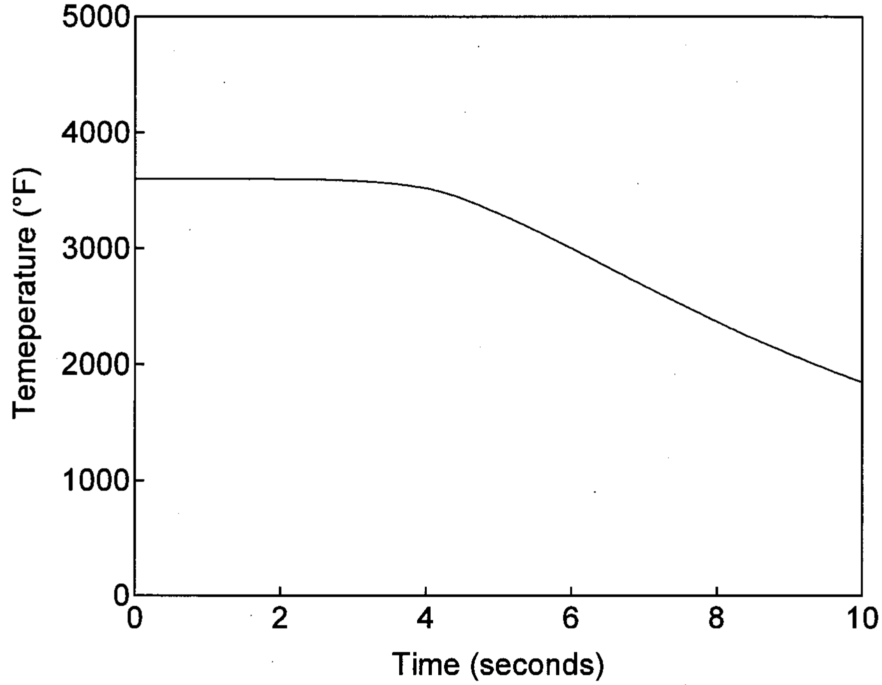

Figure 15.3.1-7.3 **Fuel Centerline Temperature versus Time Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow**

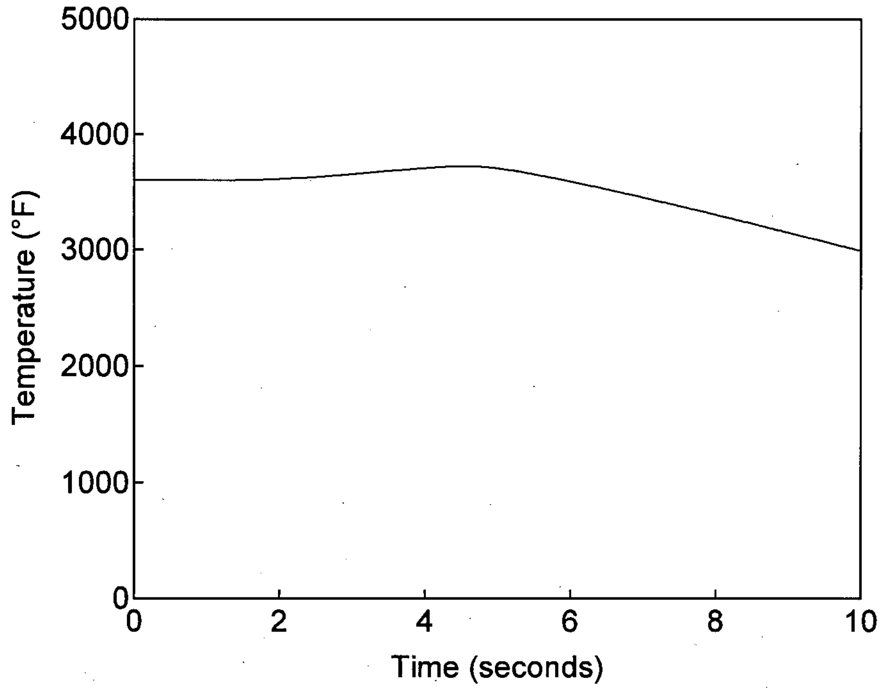

Figure 15.3.1-7.4 **Fuel Centerline Temperature versus Time RCP Rotor Seizure**

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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DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 15.3.1-8

For the three events analyzed in DCD Section 15.3, the "Time Sequence of Events" tables provide the timing of the event initiation, low flow/speed limit, reactor trip, and minimum DNBR or maximum clad temperature. It is not made clear if the reactor trip is accompanied by a turbine trip and a LOOP. Please clarify the event timings for these three events.

ANSWER:

As shown in the "Time Sequence of Events" table for each event in DCD Section 15.3, the reactor trip is initiated by the low reactor coolant flow or low reactor coolant pump speed reactor trip signal. The loss of offsite power (LOOP) is not assumed in the analyses in DCD Section 15.3. The basis for this assumption will be provided in the response to RAI 297-2287 Question 15.0.0-3. However, assuming LOOP occurrence has no effect on the minimum DNBR or maximum cladding temperature as shown in the response to Question 15.3.1-1 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.