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

July 3, 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-09340

Subject: MHI's Response to US-APWR DCD RAI No. 297-2287 Revision 2

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. 297-2287 Revision 2". The enclosed materials provide MHI's response to the NRC's "Request for Additional Information (RAI) 297-2287 Revision 2," dated May 4, 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.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Response to US-APWR DCD RAI No. 297-2287 Revision 2 (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 297-2287 Revision 2 (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-09340

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. 297-2287 Revision 2" dated July 3, 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 3rd day of July, 2009.

Y. Ogata

Yoshiki Ogata

ENCLOSURE 3

UAP-HF-09340
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 297-2287 Revision 2

July 2009

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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.: 15.0.0-1

SRP 15.0, Rev.3, March 2007 states that for new applications, the categorizations and acceptance criteria of this SRP section shall apply, i.e., only two classes of events shall be considered in Chapter 15 analyses, AOO and PA. In DCD Section 15.0.0.1, it appears as if a category of event is defined that is neither an AOO nor a PA. The applicant states, "AOOs with an assumed coincident single failure or operator error...are no longer considered AOOs." This is confusing because single failures are accounted for in the analysis of AOOs. It also states "such events are either evaluated as if they were AOOs or less restrictive acceptance criteria are applied." This is also confusing. It is not clear what acceptance criteria are applied, those for an AOO or those for a PA. Please clarify this paragraph in this section in light of the review guidance provided in SRP 15.0, rev.3, March 2007.

ANSWER:

The second paragraph of DCD Subsection 15.0.0.1 clearly states that the current SRP classifies all Chapter 15 events into only two categories - AOOs or PAs - and that the US-APWR "DCD utilizes the categorization and classification schemes adopted by the current SRP."

Subsection 15.0.0.1 of the DCD will be revised in order to remove the statements indicated as confusing by this question.

Impact on DCD

DCD Subsection 15.0.0.1 will be revised as follows:

15.0.0.1 Classification of Plant Conditions

Initiating events are categorized by event type and by frequency of occurrence. Categorization by event type provides for logical comparison between events with similar effects on the plant, which allows for the identification of limiting events. Classification by frequency of occurrence provides a basis for the selection of applicable acceptance criteria.

Initiating events are first categorized by their effect on the plant (i.e., event type), such as an increase in heat removal by the secondary system, and then further classified according to their expected frequency of occurrence. Historically, the frequency of each event was categorized as an incident of moderate frequency (ANSI N18.2 Category II), an infrequent event (ANSI N18.2 Category III), or a limiting fault (ANSI N18.2 Category IV) (Ref. 15.0-1). However, the current SRP does not use the historical ANSI N18.2 frequency classification but rather classifies an event as either an anticipated operational occurrence (AOO) or a postulated accident (PA). This DCD utilizes the categorization and classification schemes adopted by the current SRP.

~~It is important to note that AOOs and PAs apply to certain initiating events, but that there are transients and accidents that are more severe and infrequent than AOOs, but not as severe and infrequent as PAs. Examples of these events include AOOs with an assumed coincident single failure or operator error, as well as infrequent events that can only result from coincident component active failures or passive failures. AOOs that occur with such a coincident failure are no longer considered AOOs. Such events are either evaluated as if they were AOOs or less restrictive acceptance criteria are applied.~~

Due to the similarities between the MHI US-APWR and the current generation of PWRs operating in the United States, MHI has determined that no new event types are required to bound the possible initiating events.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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.: 15.0.0-2

In DCD Section 15.0.0.7, the applicant states, "The reactor trip is assumed to cause a disturbance in the utility grid, which causes the loss of offsite power (LOOP)." In other places in the DCD, the applicant states that, "A turbine trip could cause a disturbance to the grid, which could, in turn, cause a loss of offsite power, which could, in turn, cause a reactor coolant pump coast down." Please clarify this apparent contradiction in the sequence of events as described. Is LOOP caused by the turbine trip, the reactor trip, or by either one independent of the other? Or is the turbine trip the causal event for both the LOOP and the reactor trip? Please clarify the sequence of events that lead to LOOP.

ANSWER:

The US-APWR is designed such that a reactor trip will cause a turbine trip. The delay time between reactor trip and turbine trip is conservatively ignored. The turbine trip will cause a generator trip. The time delay between turbine trip and generator trip is also conservatively ignored, as described in DCD Subsection 15.0.0.7. The generator trip could then cause a disturbance in the utility grid, which could, in turn, cause a loss of offsite power (LOOP). As a result, a LOOP may be indirectly initiated by a reactor trip (via a turbine-generator trip) or may be initiated indirectly by the turbine-generator trip. Therefore, the event-specific descriptions in Chapter 15 may refer to a LOOP as a result of a reactor trip or as a result of a turbine trip. The intent of these descriptions was to accurately describe the causal event of the LOOP. However, for clarity, the description in DCD Subsection 15.0.0.7 will be modified to better describe the sequence of events.

Impact on DCD

Subsection 15.0.0.7 of the DCD will be revised as follows. Note that the changes shown below also include changes made based on the response to Question 15.0.0-3 of this RAI.

15.0.0.7 Loss of Offsite AC Power

The analyses for AOOs and other accidents consider transients both with and without offsite power available for cases where the ~~transient~~ event may be accompanied by a

turbine-generator trip. Since all reactor trips are accompanied by a turbine trip, this extends to all events resulting in ~~includes~~ a reactor trip. This analysis approach is consistent with 10 CFR 50, Appendix A, General Design Criteria (GDC) 17 (Ref. 15.0-12).

The unavailability of offsite power is not considered in characterizing the frequency of the event sequence (i.e., for transients that are AOOs, the AOO acceptance criteria are applied even when the offsite power is considered to be unavailable).

The loss of offsite power is considered in addition to the limiting single failure assumed for the event sequence where offsite power is available.

The US-APWR is designed such that the normal source of electrical power for the RCPs is the plant generator. The plant design incorporates a time delay between turbine and generator trips, ~~assuring that~~ allowing power to the RCPs ~~is~~ to be maintained for a period of time following a turbine trip. This design feature is conservatively ignored in the accident analyses. The reactor trip causes a turbine trip, which then causes a generator trip. The generator trip is assumed to cause a disturbance in the utility grid, which is conservatively assumed to cause~~s the~~ a loss of offsite power (LOOP). The accident analyses assume a loss of offsite power occurs a minimum of 3 seconds after the reactor trip. This 3-second delay accounts for the time it would take for a grid instability caused by the turbine-generator trip due to the reactor trip to propagate through the grid to the plant offsite power source. A turbine-generator trip without a prior reactor trip is also assumed to ultimately cause a LOOP with the same 3-second delay time.

The principal concern with a LOOP occurring at the time of reactor trip is that a complete loss of flow transient would be superimposed on the initiating event. With the beginning of the reactor coolant pump coastdown delayed more than 3 seconds after reactor trip, the rods are inserted to the dashpot by the time the LOOP (and corresponding loss of flow) is initiated. (Refer to Figure 15.0-3) This time delay between the reactor trip and pump coastdown assures that the portion of the transient following a postulated LOOP occurs after the limiting DNBR. Therefore, the minimum DNBR at any time during the transient is the same with offsite power available or unavailable. For this reason, the LOOP cases following reactor trip are not presented in each of the event-specific analyses.

For peak pressure analyses, the time assumed for the loss of offsite power with respect to the ~~at the time of~~ reactor or turbine-generator trip is not a key parameter ~~considered only if this assumption is conservative~~. The assumed time for the loss of offsite power is described in the applicable DCD Chapter 15 subsection.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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.: 15.0.0-3

In DCD Section 15.0.0.7, the applicant presents arguments concerning the timing of a LOOP with respect to the timing of reactor trip and RCP coast down. By assuming a time delay in the onset of a LOOP, the onset of DNBR is decoupled from the availability of offsite power because the rods are inserted to the dashpot by the time the LOOP is initiated; therefore, the RCPs continue to run until the rods have been inserted in the core. This argument repeats itself throughout many Chapter 15 scenarios, allowing for the neglect of any LOOP consequences. Please provide more substantial arguments in defense of this assumed time delay, supported by sensitivity calculations that directly address this issue. (see also Question 15.1.2-1).

ANSWER:

As described in DCD Subsection 15.0.0.7 and the response to Question 15.0.0-2 of this RAI, the US-APWR DCD Chapter 15 analyses currently assume a loss of offsite power (LOOP) occurs a minimum of 3 seconds after the reactor trip. This 3-second delay accounts for the time it would take for a grid instability caused by the turbine-generator trip due to the reactor trip to propagate through the grid to the plant offsite power source. A turbine-generator trip without a prior reactor trip is also assumed to ultimately cause a LOOP with the same 3-second delay time.

MHI performed a series of analyses to determine the sensitivity of the relevant key parameters (such as power, hot spot heat flux, RCS pressure, T_{avg}, etc.) to the assumed delay time. Delay periods of 0, 1, 2, and 3 seconds were used for the sensitivity analyses. Representative events were chosen for evaluation based on the events ability to challenge one or more of the following acceptance criteria:

- Minimum DNBR
- Peak RCS Pressure
- Peak Fuel Centerline Temperature
- Peak Fuel Enthalpy
- Peak Cladding Temperature

Minimum DNBR Analyses

Four representative events were evaluated using MARVEL-M and VIPRE-01M to assess the impact of the assumed LOOP delay time on the minimum DNBR: rod withdrawal at power (Subsection 15.4.2) for the 75 and 5.0 pcm/sec withdrawal rates, the limiting main steam line break at 100% power (Subsection 15.1.5 Case C), and loss of load (Subsection 15.2.1). For the DNBR figures shown in this RAI, the results are generated using the MARVEL-M/VIPRE-01M methodology rather than the MARVEL-M lookup table methodology utilized in the DCD; therefore, the legend for these figures indicates "without LOOP" instead of "DCD case". Both of 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 (i.e., DCD) case for consistency. Transient results for the key analysis parameters for the rod withdrawal at power (RWP) event assuming a 75 pcm/sec withdrawal rate and LOOP are provided in Figures 15.0.0-3.1 through 15.0.0-3.7. For DNBR, Figure 15.0.0-3.7 shows that the cases with 2 and 3 second delays have minimum DNBRs that are bounded by the case without LOOP. The case with a 1 second delay has a minimum DNBR that is slightly more limiting than the case without LOOP, but is still above the DNBR limit. The case with no delay has a minimum DNBR that is less than the DNBR limit. Similar results for the RWP case assuming a 5.0 pcm/sec withdrawal rate and LOOP are provided in Figures 15.0.0-3.8 through 15.0.0-3.14. Transient results for the key analysis parameters for the limiting main steam line break at 100% power with LOOP are shown in Figures 15.0.0-3.15 through 15.0.0-3.20. For DNBR, Figure 15.0.0-3.20 shows that the case without LOOP bounds the minimum DNBR for the 2 and 3 second delay cases, while the 0 and 1 second delay cases have minimum DNBRs slightly less than the case without LOOP. However, the minimum DNBR remains above the DNBR limit in all cases. Finally, the transient results for the key analysis parameters for the loss of load with LOOP are provided in Figures 15.0.0-3.21 through 15.0.0-3.25. For DNBR, Figure 15.0.0-3.25 shows that the cases with 2 and 3 second delays have minimum DNBRs that are bounded by the DCD. The case with a 1 second delay has a minimum DNBR that is slightly less than the case without LOOP while the case with a 0 second delay has a minimum DNBR that is much less than the case without LOOP. However, the minimum DNBR remains significantly above the DNBR limit in all cases.

The results of the DNBR sensitivity analysis of these four events validate the assumption described in DCD Subsection 15.0.0.7 (and reiterated in event-specific subsections) that, with a 3 second delay between the turbine-generator trip and LOOP, the minimum DNBR is the same for the case with and without LOOP. The results of this sensitivity also show that there is some available margin in the delay time assumed since the minimum DNBR is the same for the case with LOOP assuming a 2 second delay and without LOOP. In addition, while the minimum DNBR for the 1 second delay cases were slightly worse than the cases without LOOP, the DNBR remained above the safety analysis analytical limit. Since the four events selected for this sensitivity analysis were representative DNBR events in Chapter 15, the remaining DNBR analysis in Chapter 15 would also be adequate even assuming a 1 second delay (excluding those events that already violated the DNBR limit, such as the RCP rotor seizure). Although the results showed that the two 0 second delay RWP cases result in a minimum DNBR below the acceptance limit, the US-APWR DCD contains COL Item 8.2(11) which requires that the COL applicant perform a grid stability analysis to confirm that the LOOP occurs at 3 seconds or greater. This is also described in the response to Question 15.0.0-4 of this RAI.

Peak RCS Pressure Analyses

Two representative events were evaluated using MARVEL-M to assess the impact of the assumed LOOP delay time on the peak pressure: loss of load (Subsection 15.2.1) and feedwater line break (Subsection 15.2.8). Transient results for the key analysis parameters for the loss of load event with LOOP are provided in Figures 15.0.0-3.26 through 15.0.0-3.32. For the RCP outlet pressure shown in Figure 15.0.0-3.27, the cases with 2 and 3 second delay times have peak pressures that

are bounded by the DCD case, while the cases with 0 and 1 second delay times have slightly higher peak pressures than the DCD case. However, the difference in peak pressure is very small and in all cases, the peak pressures meet the relevant acceptance criteria. Transient results for the key analysis parameters of the feedwater line break event with LOOP are provided in Figures 15.0.0-3.33 through 15.0.0-3.44. It is important to note that the DCD feedwater line break case already assumes LOOP occurs with a 0 second (as opposed to a 3 second) delay time. The figures show that all parameters are mostly insensitive to the LOOP assumption. The relatively small differences in LOOP delay times are insignificant compared to the relatively long duration of the feedwater line break event. The DCD case shown in Figure 15.0.0-3.34 (which assumes a LOOP with 0 seconds delay) does have a slightly higher peak RCP outlet pressure than the other cases, but in all cases, the peak pressures meet the relevant acceptance criteria.

The results of the peak pressure analysis of these two events show that peak pressure is not as sensitive as DNBR to the LOOP assumptions. With a 3, 2, or 1 second delay between the turbine-generator trip and LOOP, the peak pressure is the same for the case with and without LOOP. The peak pressure is slightly higher for the 0 second delay case, but still within the acceptance criteria. Since the two events selected for this sensitivity analysis were the most limiting peak pressure events in Chapter 15, the remaining peak pressure analyses in Chapter 15 would also be adequate regardless of the assumptions about the LOOP delay time. While the results of this sensitivity analysis show that the DCD analyses meet the acceptance criteria, the description of the LOOP assumptions for peak pressure analyses in DCD Section 15.0.0.7 will be revised to more accurately describe the results of this sensitivity study.

Fuel Centerline Temperature and Enthalpy Analyses

Three representative cases of the rod ejection event (Subsection 15.4.8) were evaluated to assess the impact of the assumed LOOP delay time on fuel parameters: hot full power (HFP) at end-of-cycle (EOC), hot zero power (HZP) at beginning-of-cycle (BOC), and HZP at EOC. The HFP-EOC case was evaluated to assess the impact of the LOOP assumptions on the calculated fuel centerline temperature. Figures 15.0.0-3.45 through 15.0.0-3.47 provide the transient results for the key analysis parameters for the rod ejection event at HFP-EOC conditions with LOOP. The calculated fuel centerline temperatures of all the LOOP cases are bounded by the fuel centerline temperature of the DCD case (no LOOP), as shown in Figure 15.0.0-3.46. The HZP-BOC and HZP-EOC cases were evaluated to assess the impact of the LOOP assumptions on the calculated fuel enthalpy. Figures 15.0.0-3.48 through 15.0.0-3.50 and Figures 15.0.0-3.51 through 15.0.0-3.53 provide the transient results for the key analysis parameters for the rod ejection event at HZP for BOC and EOC conditions, respectively, with LOOP. The calculated peak fuel enthalpies of all of the LOOP cases are bounded by peak fuel enthalpy of the DCD case (no LOOP) as shown in Figures 15.0.0.3-50, and 15.0.0.3-53.

The results of the fuel centerline temperature and enthalpy sensitivity analysis of the three cases of the rod ejection event validate the assumption described in DCD Subsection 15.0.0.7 (and reiterated in event-specific subsections) that, with a 3 second delay between the turbine generator trip and LOOP, the results are the same for the case with and without offsite power. In addition, the maximum fuel centerline temperatures and maximum fuel enthalpies for the 0, 1, and 2 second delay cases were also bounded by the DCD case (no LOOP).

Peak Cladding Temperature Analyses

The RCP rotor seizure event (Subsection 15.3.3) was evaluated as a representative event to determine the effect of the assumed LOOP time delay on the peak cladding temperature (PCT). Transient results for the key analysis parameters for the locked rotor event with LOOP are provided in Figures 15.0.0-3.54 through 15.0.0-3.55. The RCP coastdown due to the LOOP has a significant impact on the total and loop flow, including a reduction in reverse flow in the faulted loop, as shown in Figure 15.0.0-3.54. The results also indicate that while the maximum PCT for the DCD and 3 second delay case are the same, the maximum PCT increases with the shorter

LOOP time delays of 2, 1, and 0 seconds. However, the calculated PCT remains below the relevant acceptance criteria in all cases.

The results of the PCT sensitivity analysis of the rotor seizure event validate the assumption described in DCD Subsection 15.0.0.7 (and reiterated in event-specific subsections) that, with a 3 second delay between the turbine-generator trip and the LOOP, the results are the same for the case with and without LOOP. In addition, while the maximum PCT for the 0, 1, and 2 second delay cases were slightly worse than the DCD case (no LOOP), the PCT still met the acceptance criteria.

Conclusions

MHI performed a series of analyses to determine the sensitivity of the relevant key parameters (such as power, hot spot heat flux, RCS pressure, T_{avg}, etc.) to the assumed LOOP delay time. Delay periods of 0, 1, 2, and 3 seconds were used for the sensitivity analyses. The results generally indicate that a smaller delay time has minimal effect on the ability to meet the relevant acceptance criteria; although, in a few cases a zero second delay results in exceeding the relevant acceptance criteria. For this reason, COL Item 8.2(11) requires that the COL applicant perform a grid stability analysis to confirm that the LOOP occurs at 3 seconds or greater.

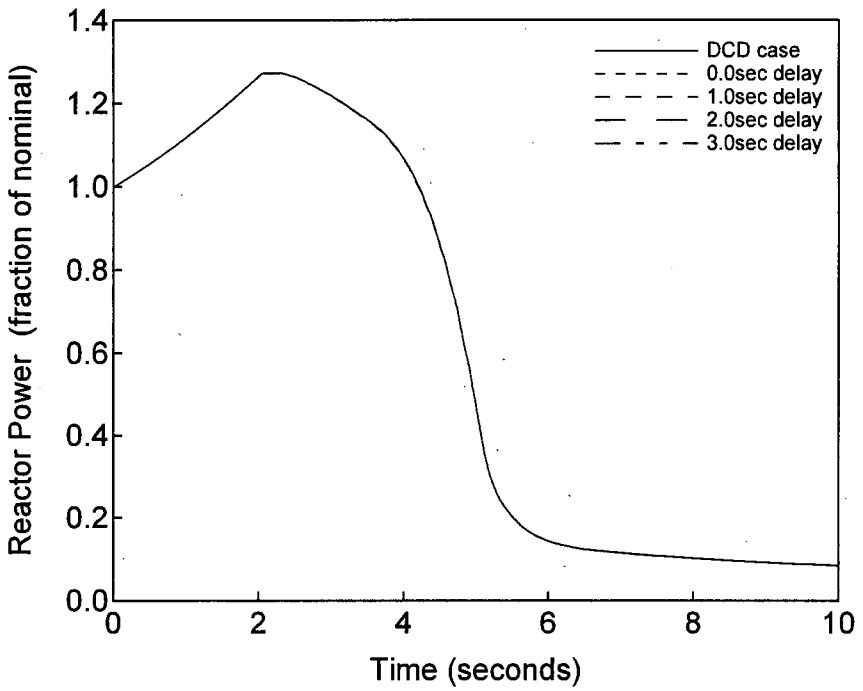

Figure 15.0.0-3.1 **Reactor Power versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

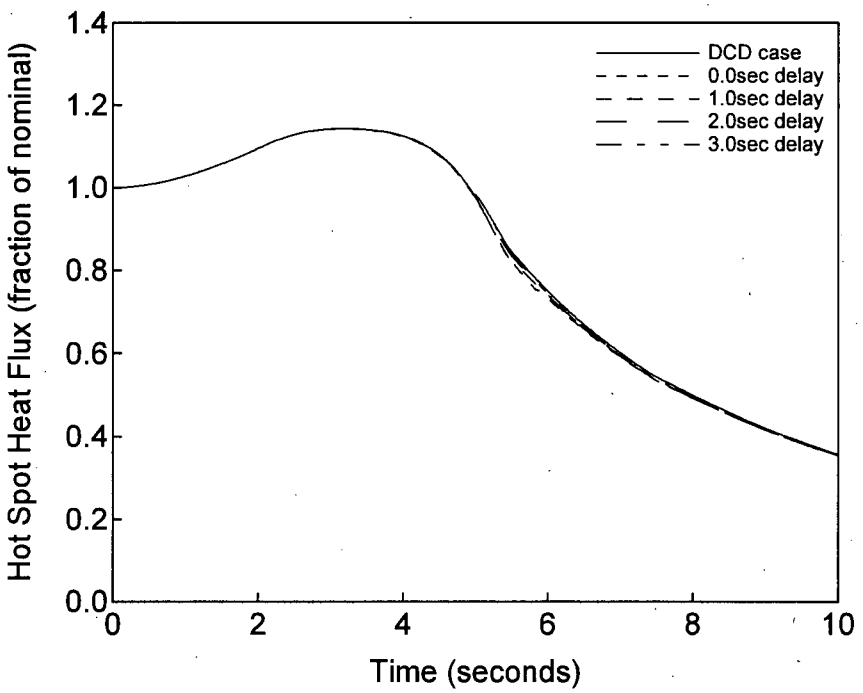

Figure 15.0.0-3.2 **Hot Spot Heat Flux versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

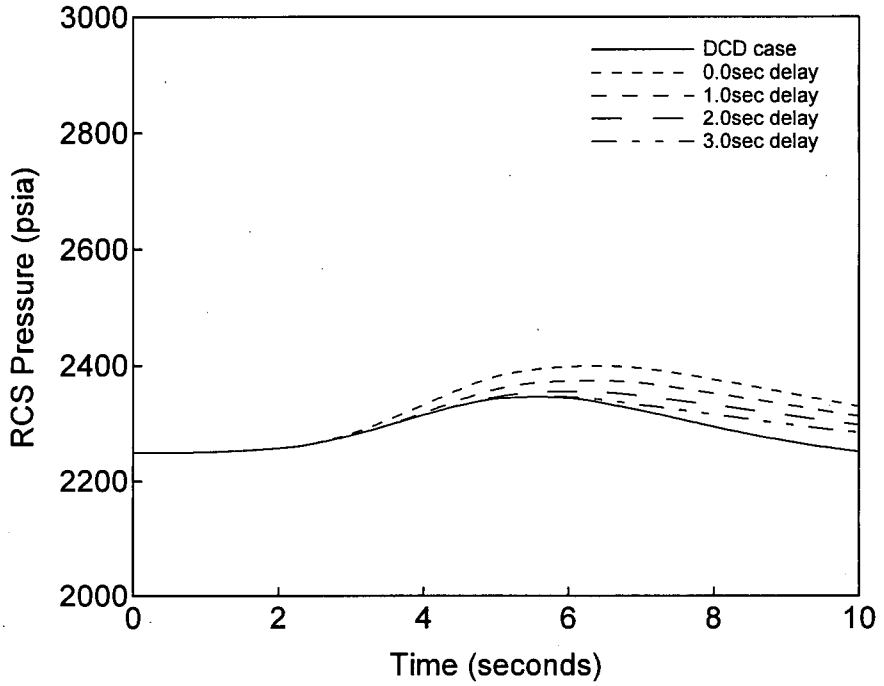

Figure 15.0.0-3.3 **RCS Pressure versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

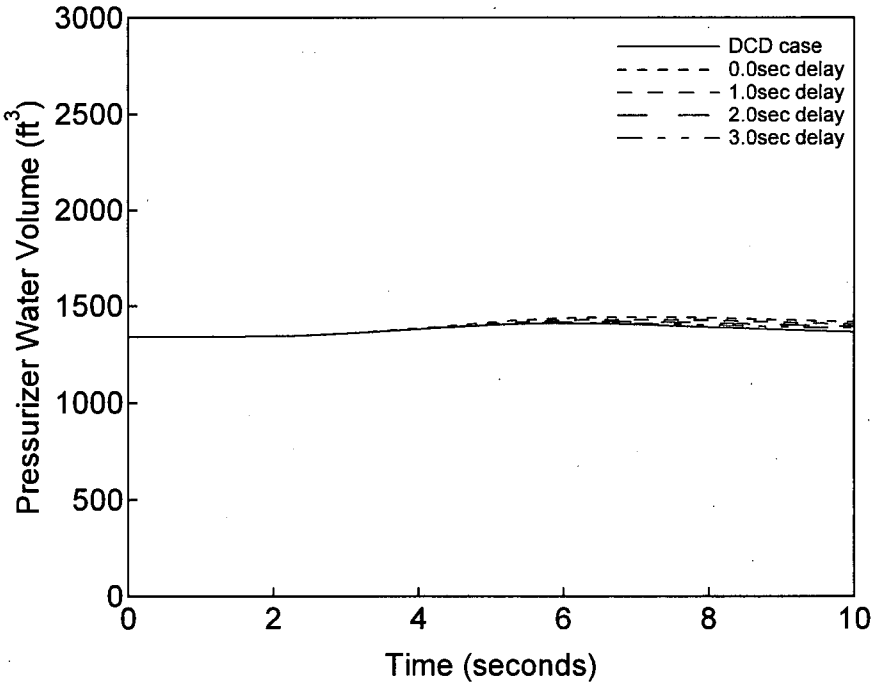

Figure 15.0.0-3.4 **Pressurizer Water Volume versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

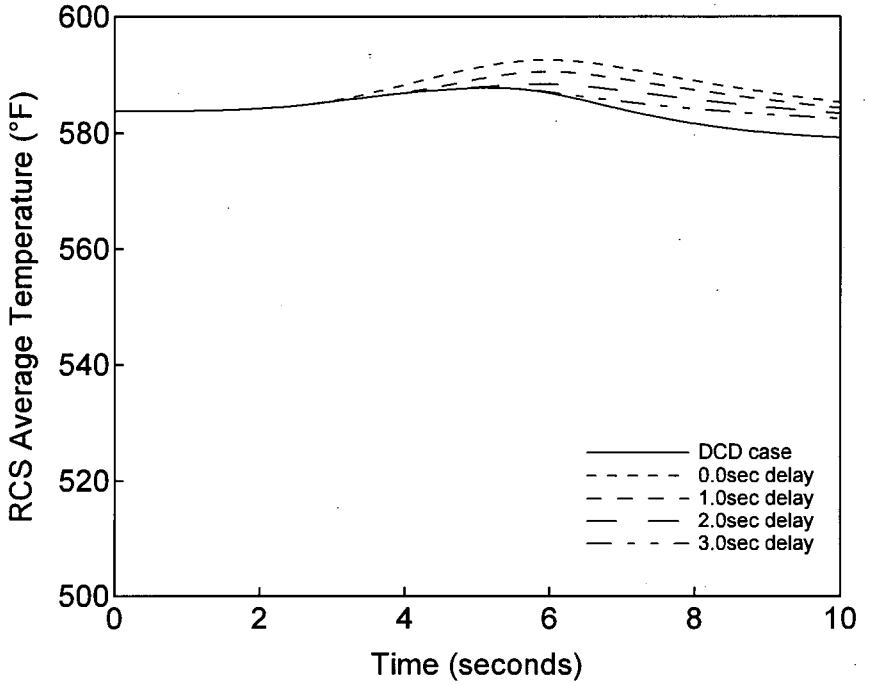

Figure 15.0.0-3.5 **RCS Average Temperature versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

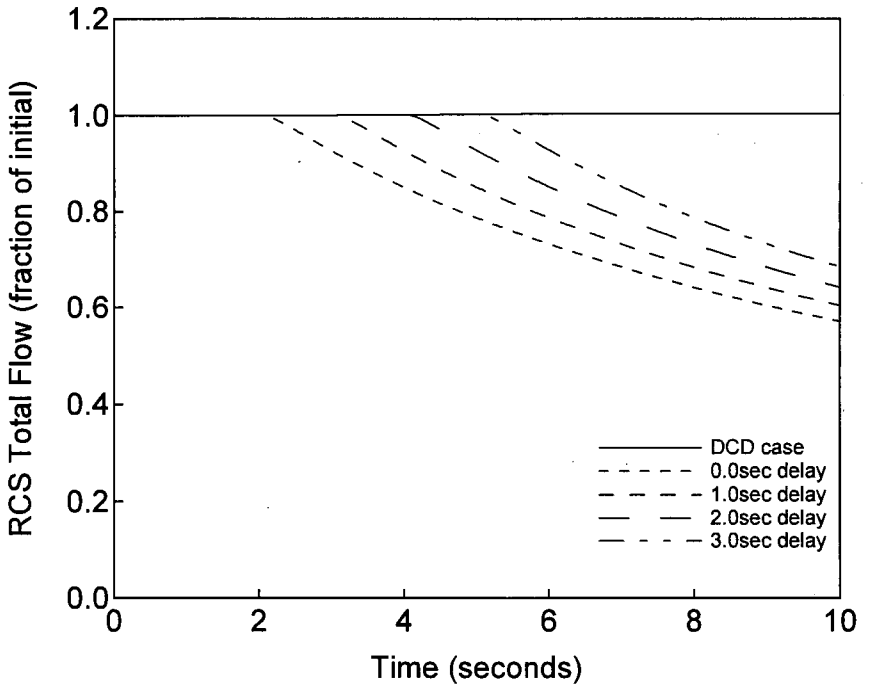

Figure 15.0.0-3.6 **RCS Total Flow versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

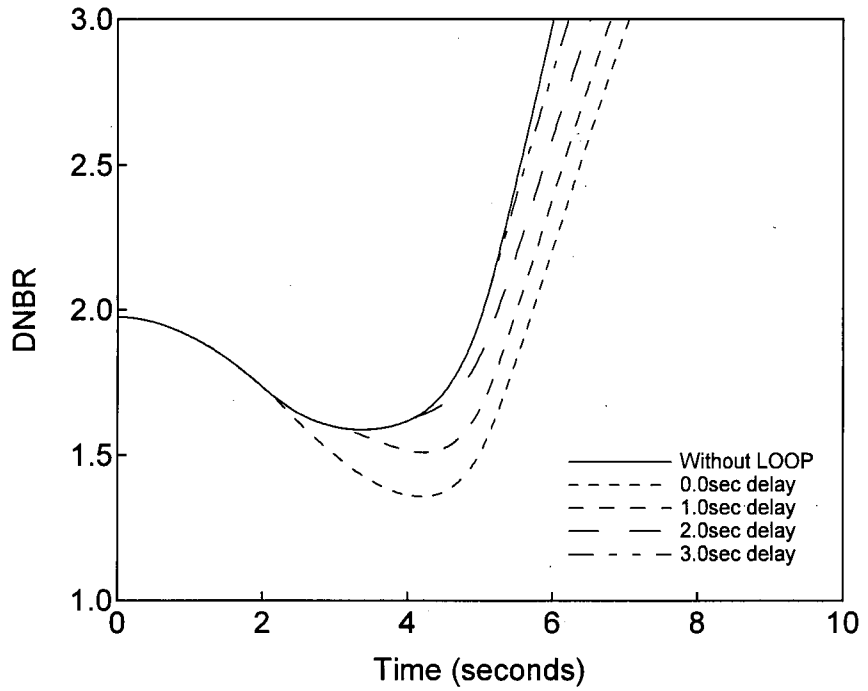

Figure 15.0.0-3.7 **DNBR versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 75 pcm/sec)**

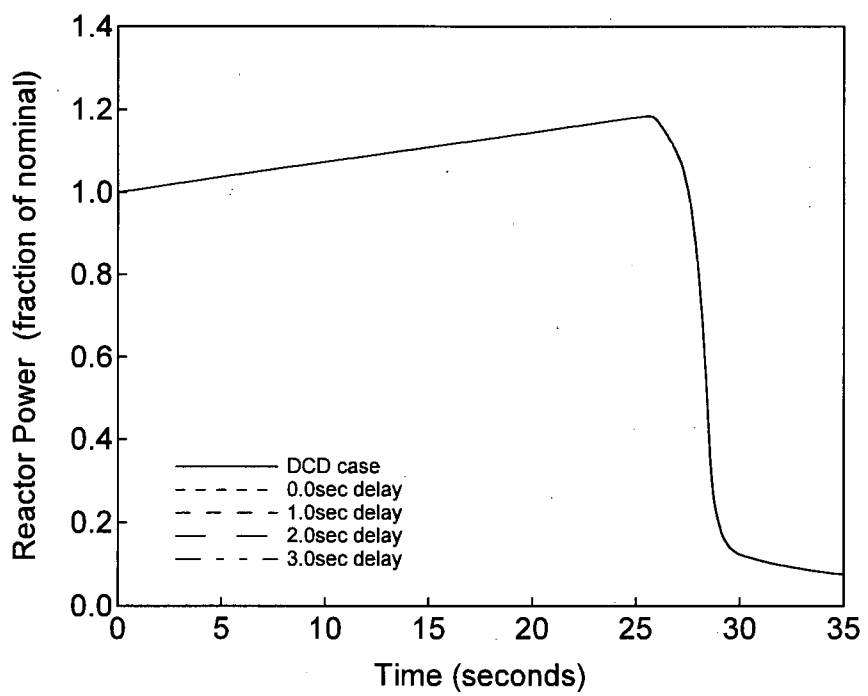

Figure 15.0.0-3.8 **Reactor Power versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)**

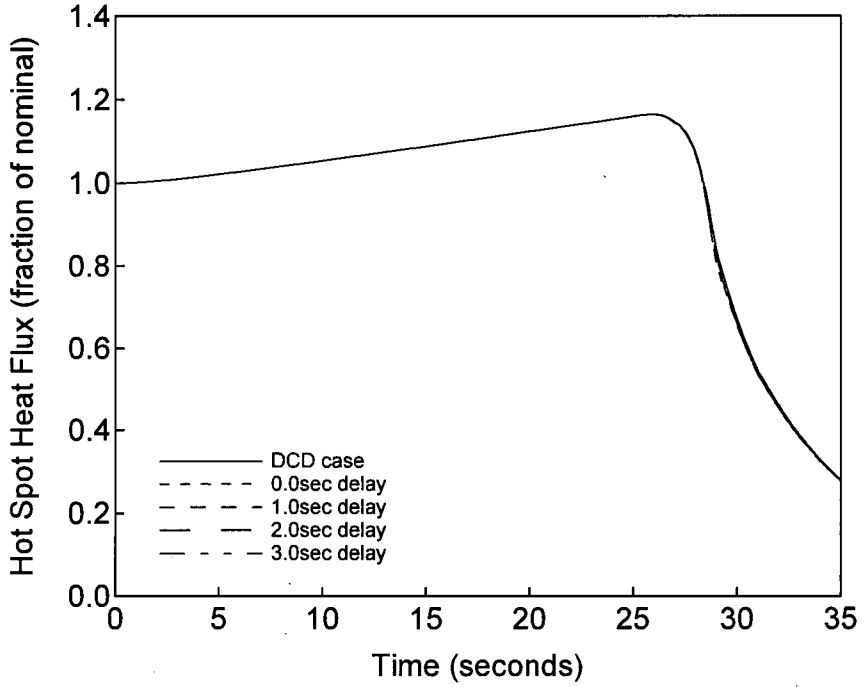

Figure 15.0.0-3.9 Hot Spot Heat Flux versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

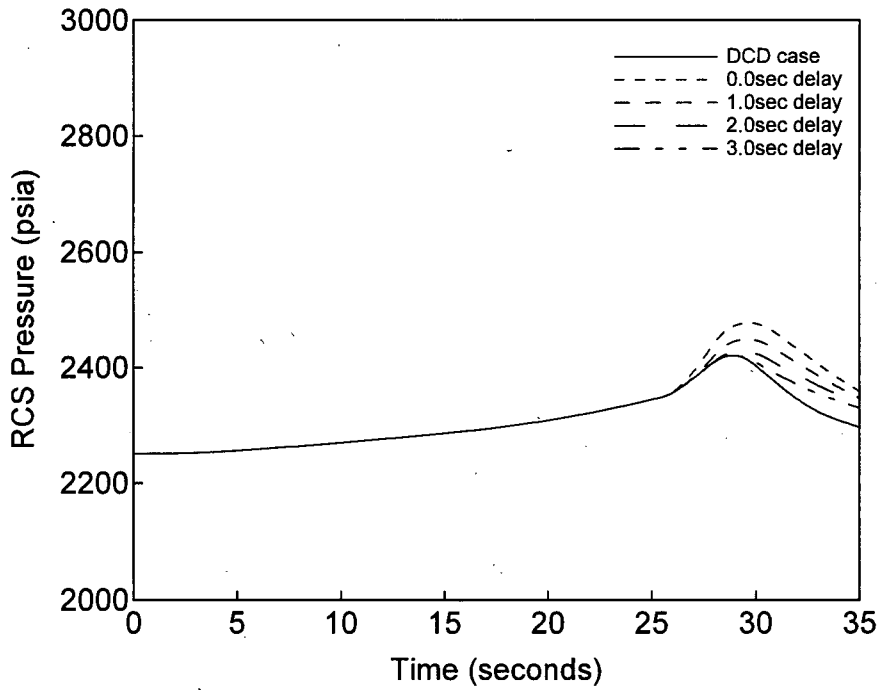

Figure 15.0.0-3.10 RCS Pressure versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

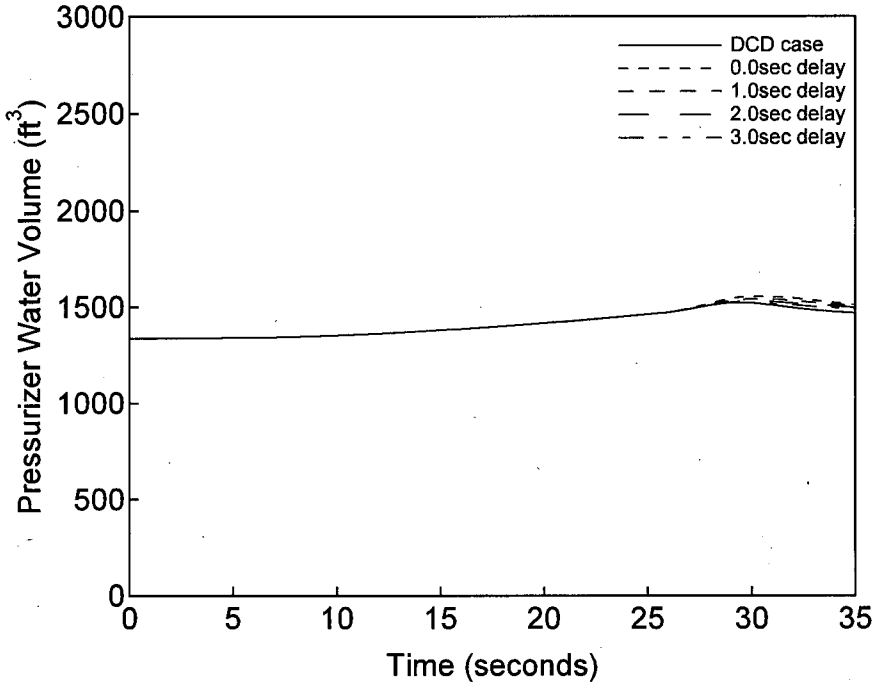

Figure 15.0.0-3.11 Pressurizer Water Volume versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

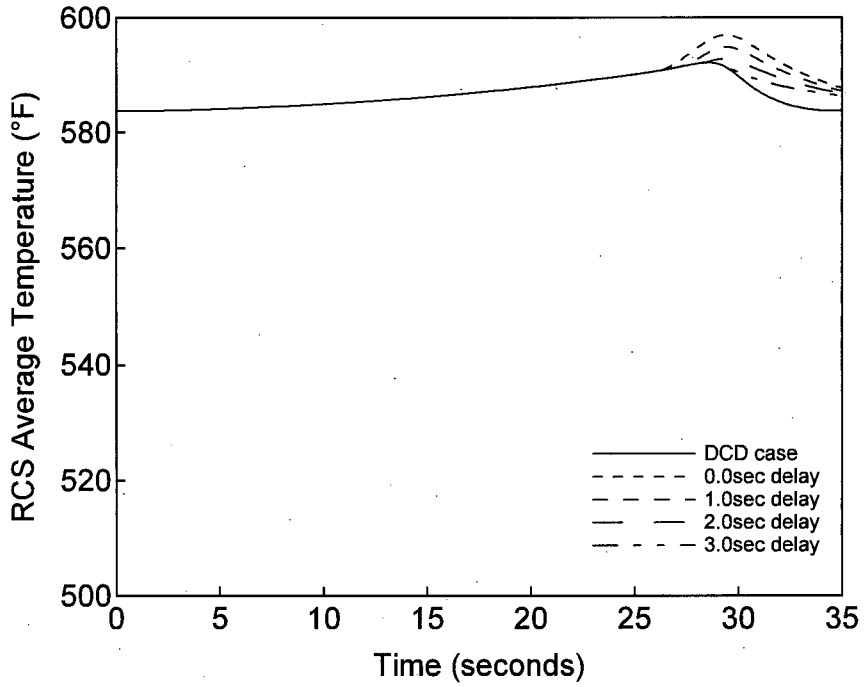

Figure 15.0.0-3.12 RCS Average Temperature versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

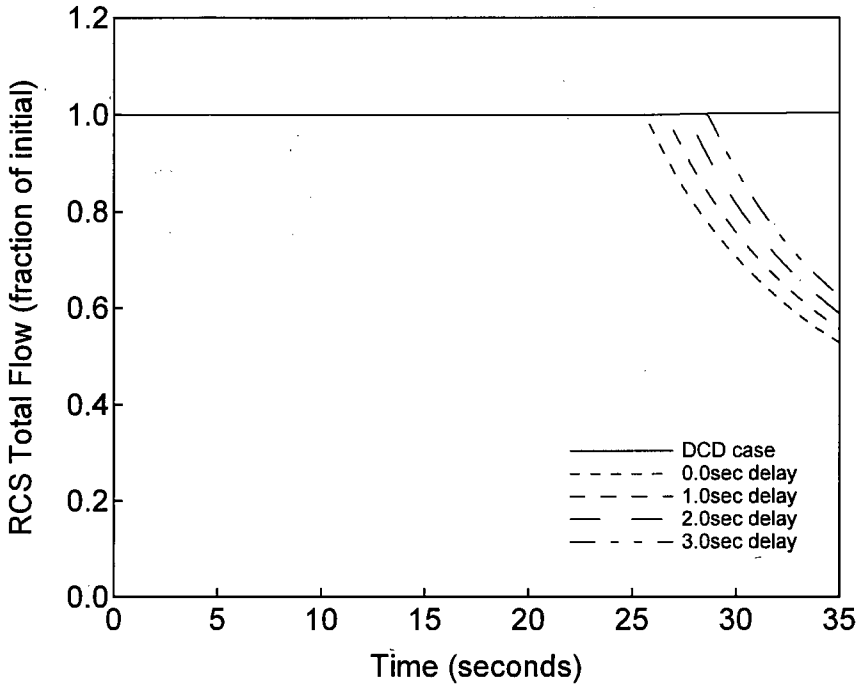

Figure 15.0.0-3.13 RCS Total Flow versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

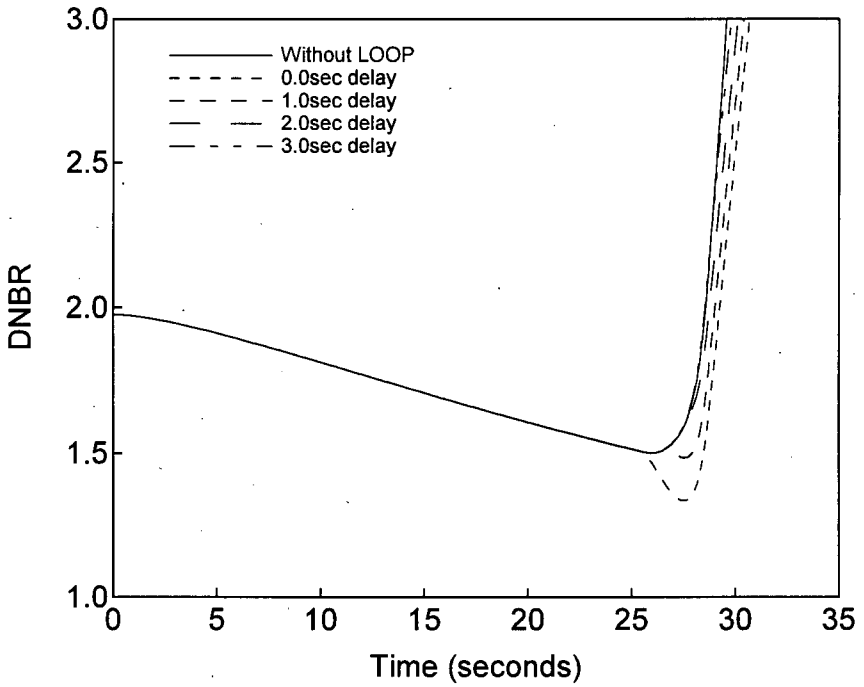

Figure 15.0.0-3.14 DNBR versus Time with and without LOOP Uncontrolled Control Rod Assembly Withdrawal at Power - DNBR Analysis (HFP, BOC, 5.0 pcm/sec)

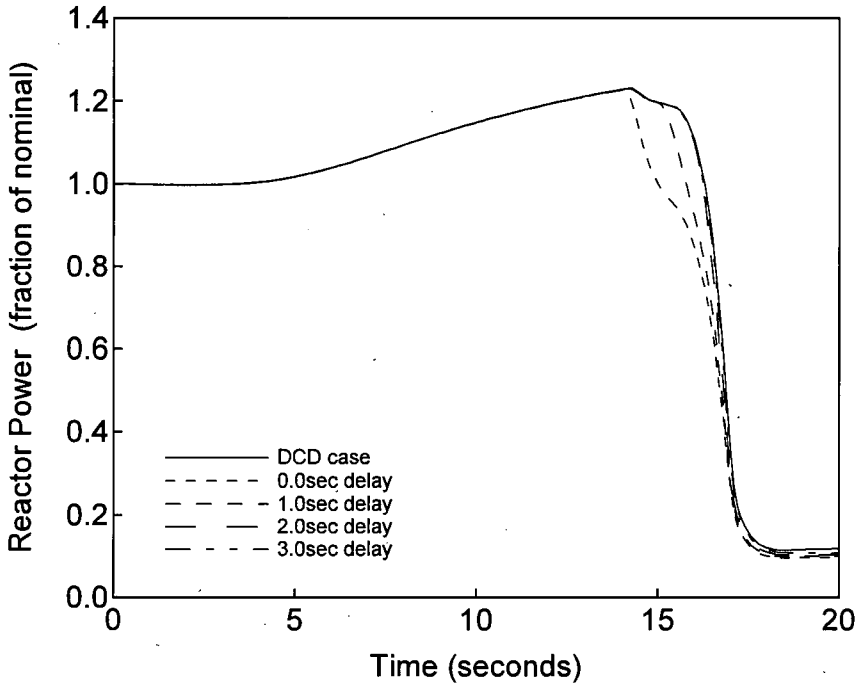

Figure 15.0.0-3.15 Reactor Power versus Time with and without LOOP Steam System Piping Failure - Case C: Limiting case for spectrum of breaks at 100% power

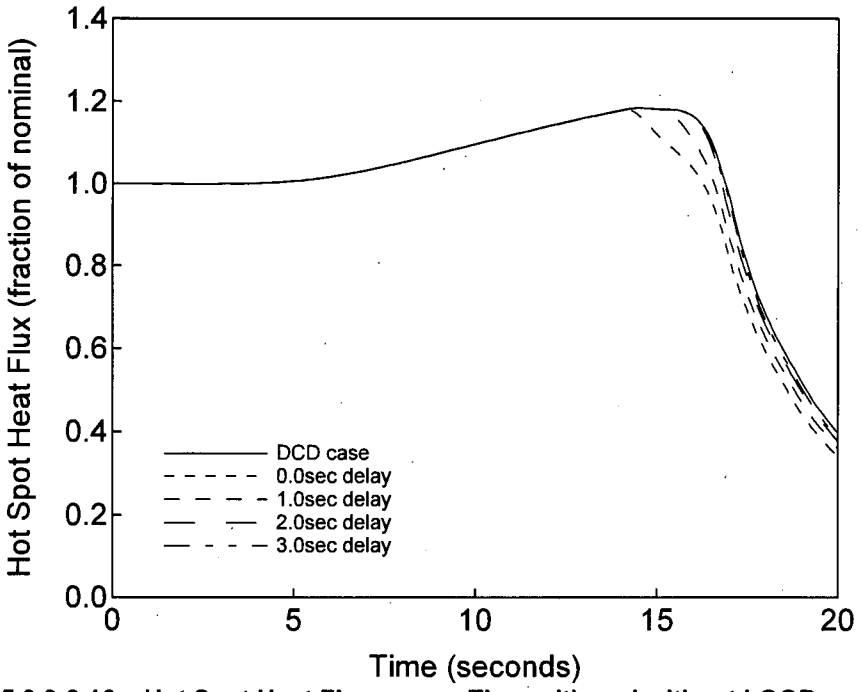

Figure 15.0.0-3.16 Hot Spot Heat Flux versus Time with and without LOOP Steam System Piping Failure - Case C: Limiting case for spectrum of breaks at 100% power

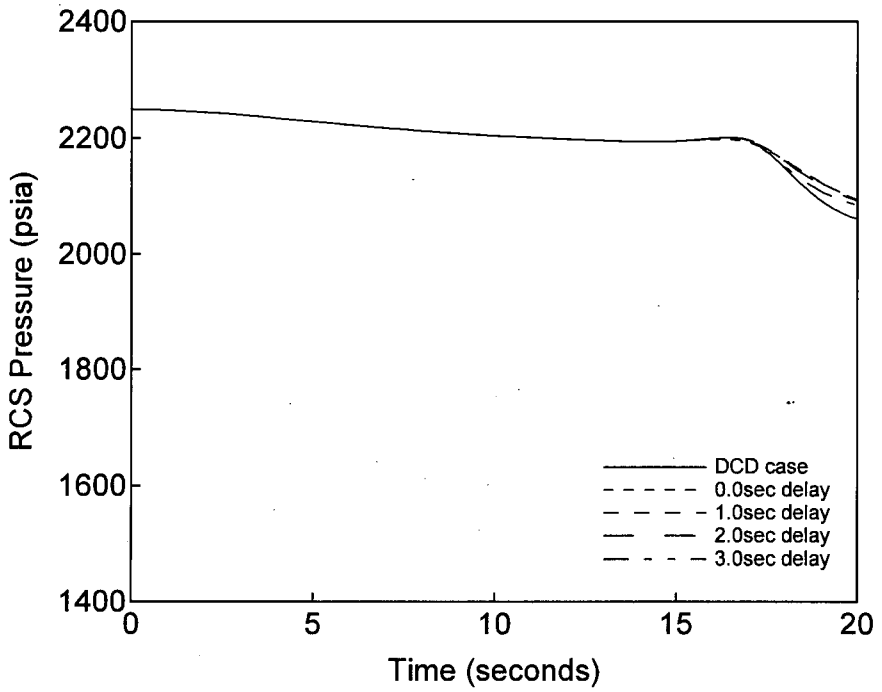

Figure 15.0.0-3.17 RCS Pressure versus Time with and without LOOP Steam System Piping Failure - Case C: Limiting case for spectrum of breaks at 100% power

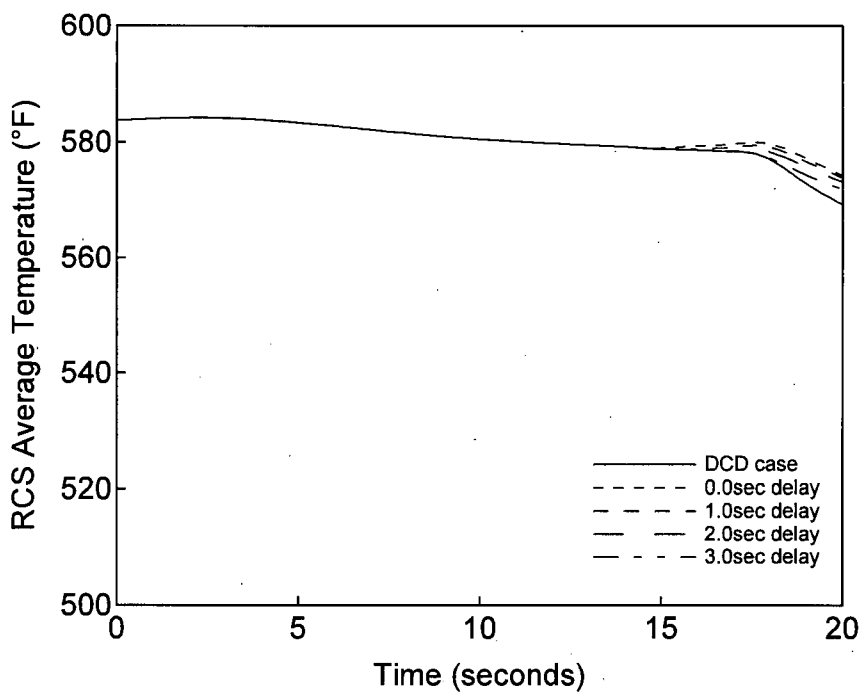

Figure 15.0.0-3.18 RCS Average Temperature versus Time with and without LOOP Steam System Piping Failure - Case C: Limiting case for spectrum of breaks at 100% power

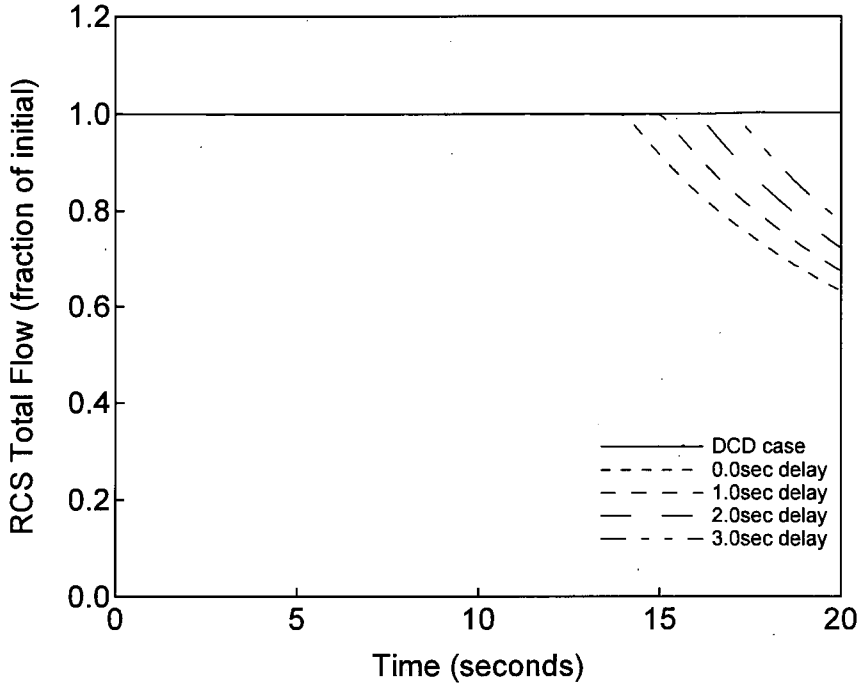

Figure 15.0.0-3.19 RCS Total Flow versus Time with and without LOOP
Steam System Piping Failure
- Case C: Limiting case for spectrum of breaks at 100% power

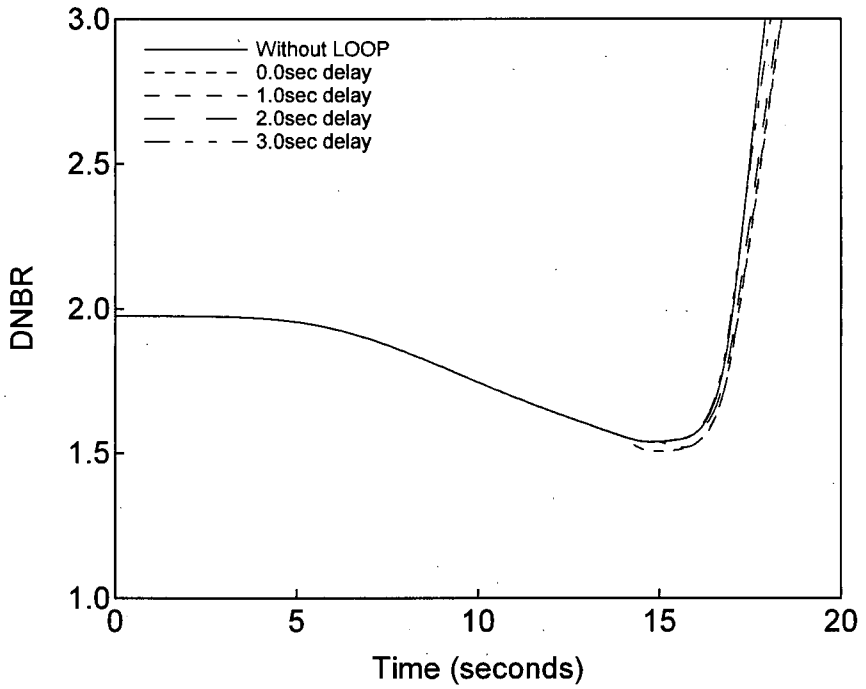

Figure 15.0.0-3.20 DNBR versus Time with and without LOOP
Steam System Piping Failure
- Case C: Limiting case for spectrum of breaks at 100% power

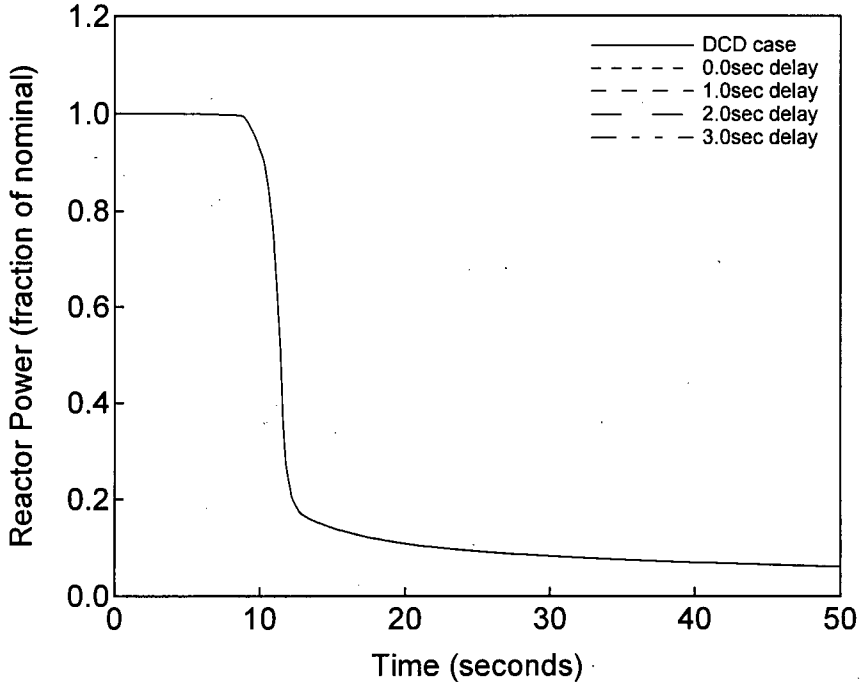

Figure 15.0.0-3.21 Reactor Power versus Time with and without LOOP Loss of External Load - DNBR Analysis

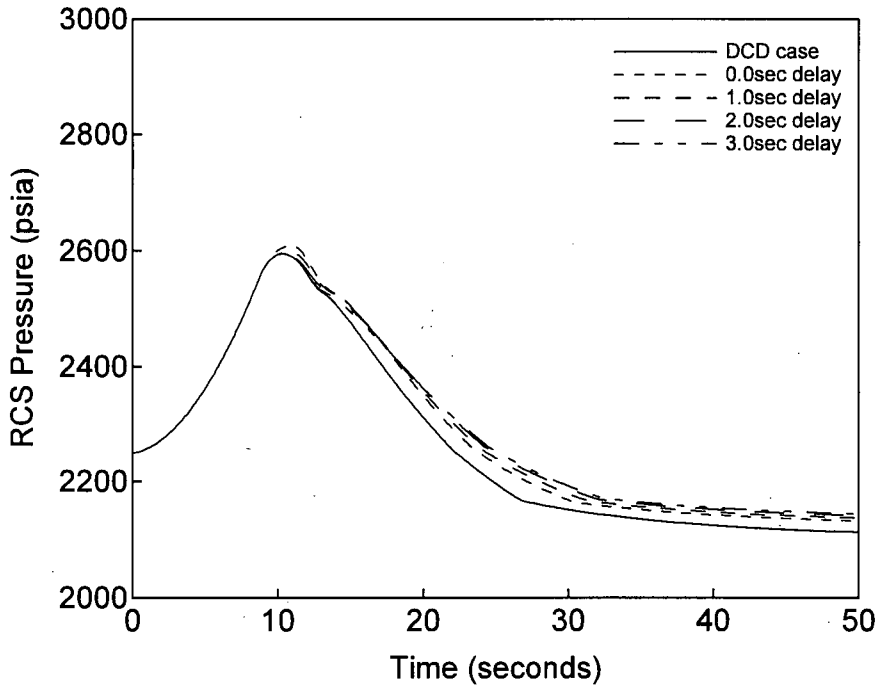

Figure 15.0.0-3.22 RCS Pressure versus Time with and without LOOP Loss of External Load - DNBR Analysis

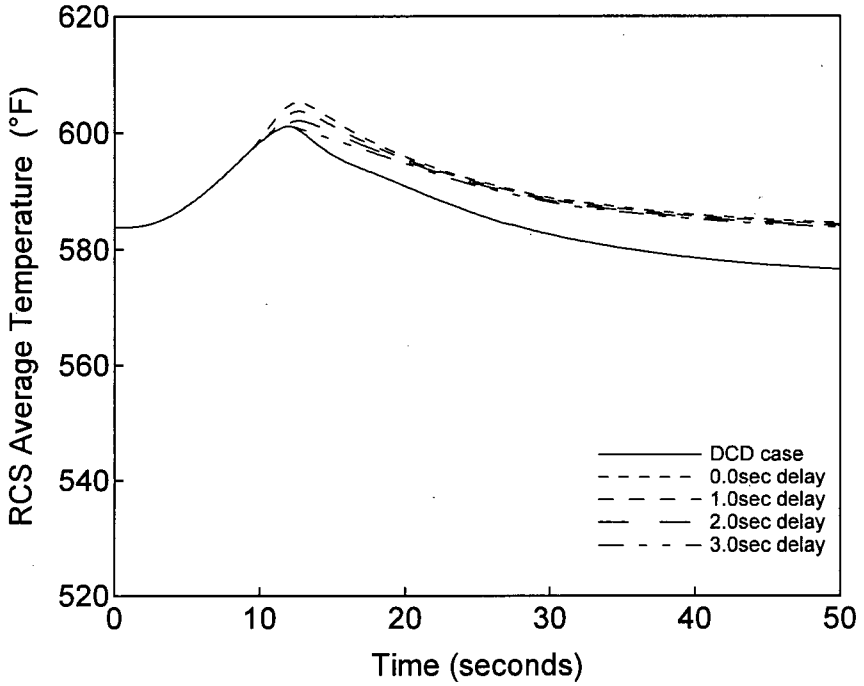

Figure 15.0.0-3.23 RCS Average Temperature versus Time with and without LOOP Loss of External Load - DNBR Analysis

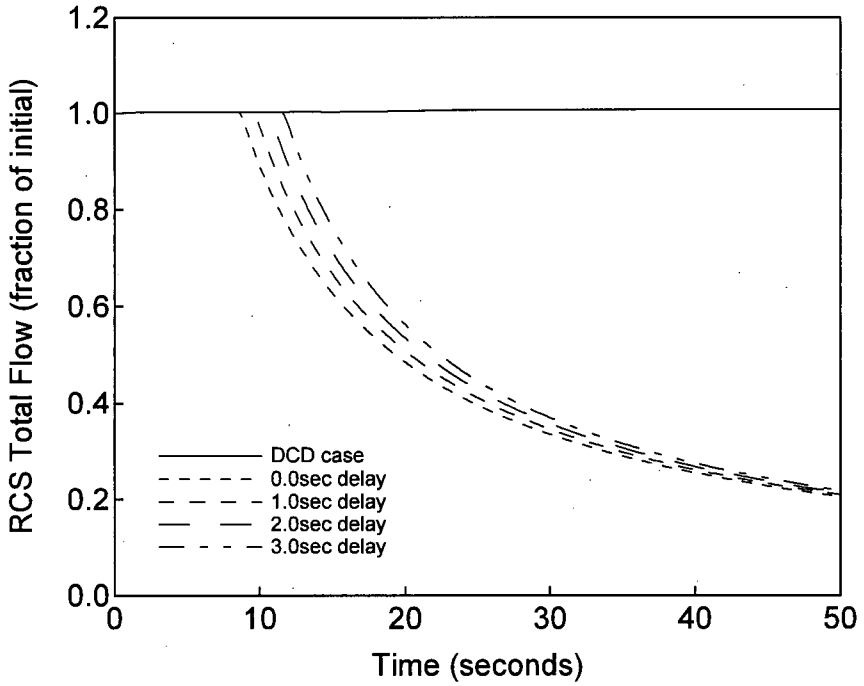

Figure 15.0.0-3.24 RCS Total Flow versus Time with and without LOOP Loss of External Load - DNBR Analysis

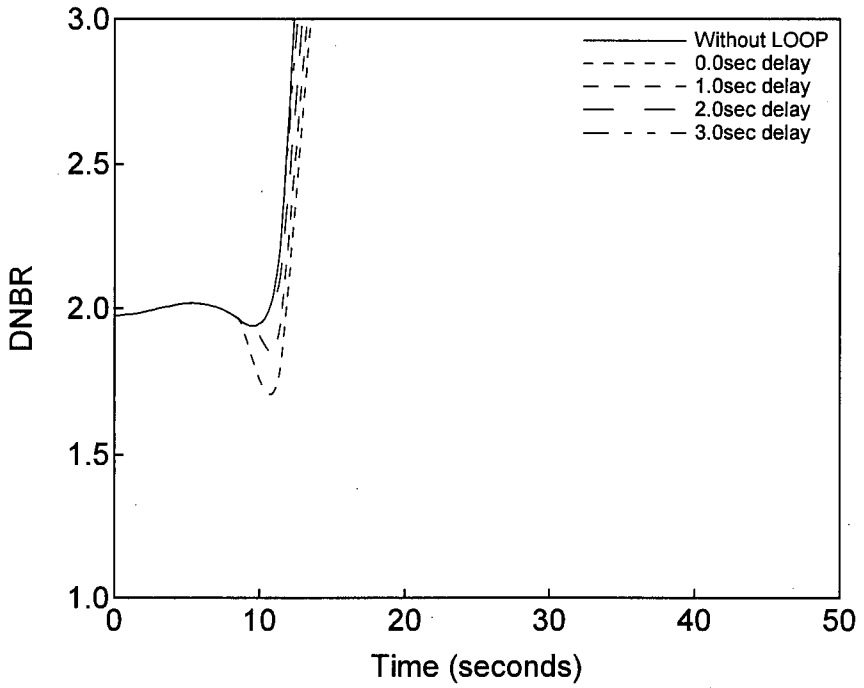

Figure 15.0.0-3.25 DNBR versus Time with and without LOOP
Loss of External Load
- DNBR Analysis

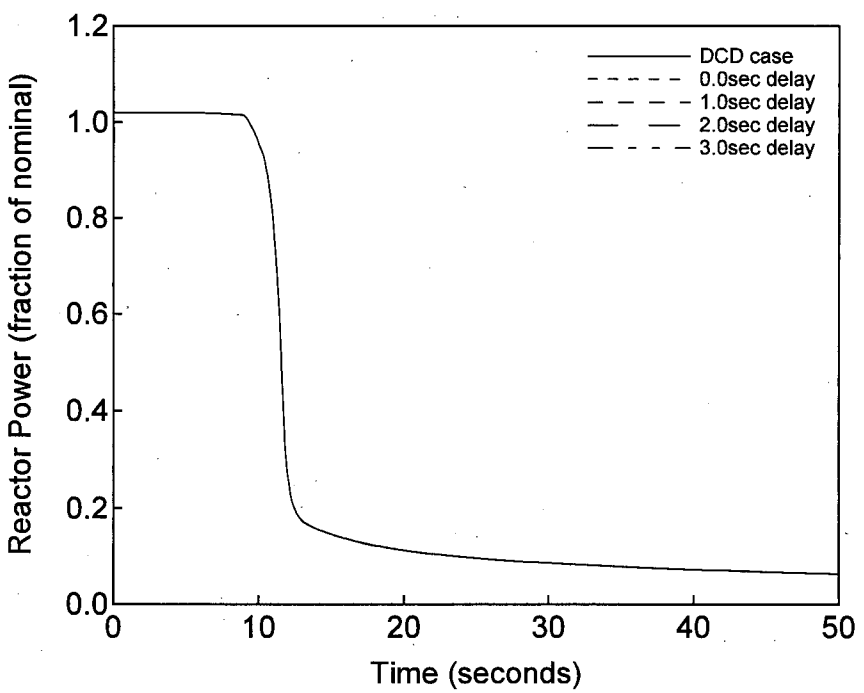

Figure 15.0.0-3.26 Reactor Power versus Time with and without LOOP
Loss of External Load
- RCS & Main Steam Pressure Analysis

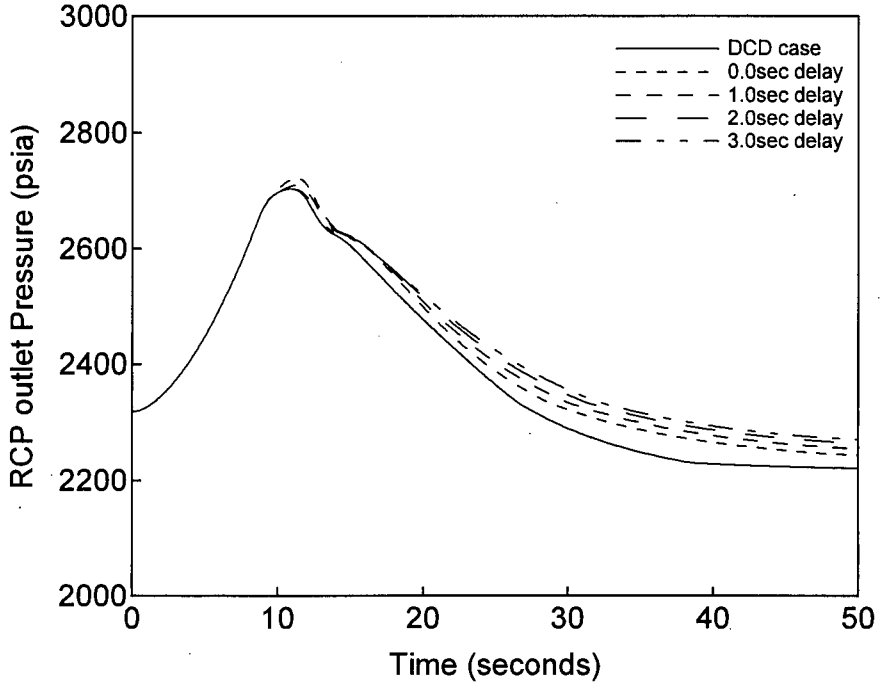

Figure 15.0.0-3.27 RCP Outlet Pressure versus Time with and without LOOP Loss of External Load - RCS & Main Steam Pressure Analysis

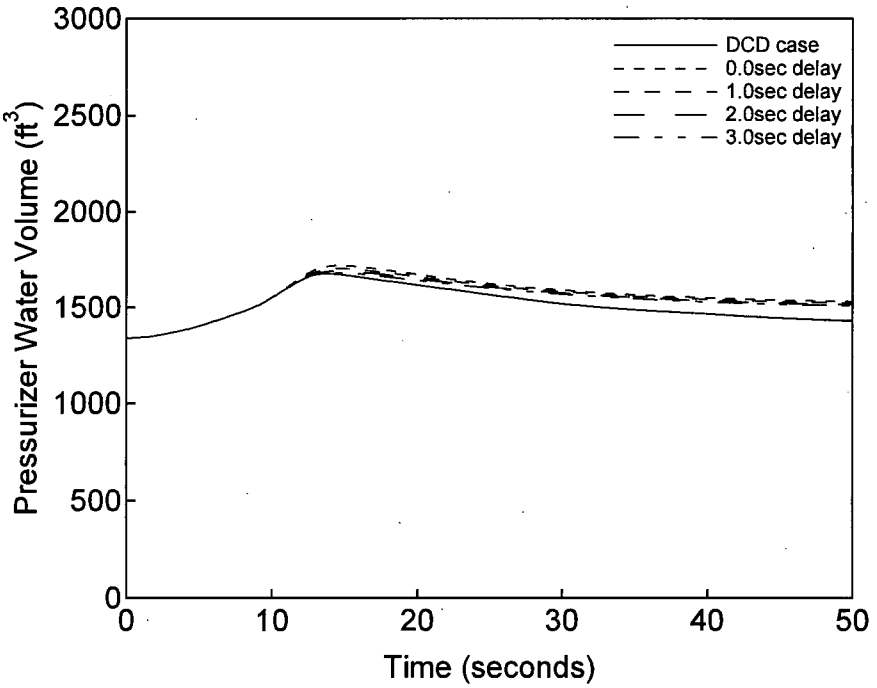

Figure 15.0.0-3.28 Pressurizer Water Volume versus Time with and without LOOP Loss of External Load - RCS & Main Steam Pressure Analysis

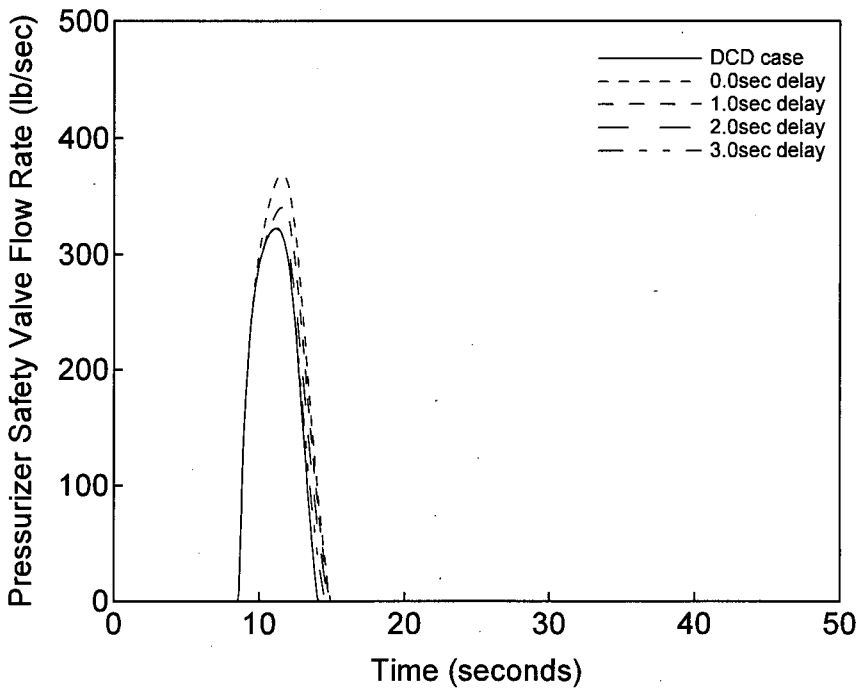

Figure 15.0.0-3.29 Pressurizer Safety Valve Flow Rate versus Time with and without LOOP Loss of External Load - RCS & Main Steam Pressure Analysis

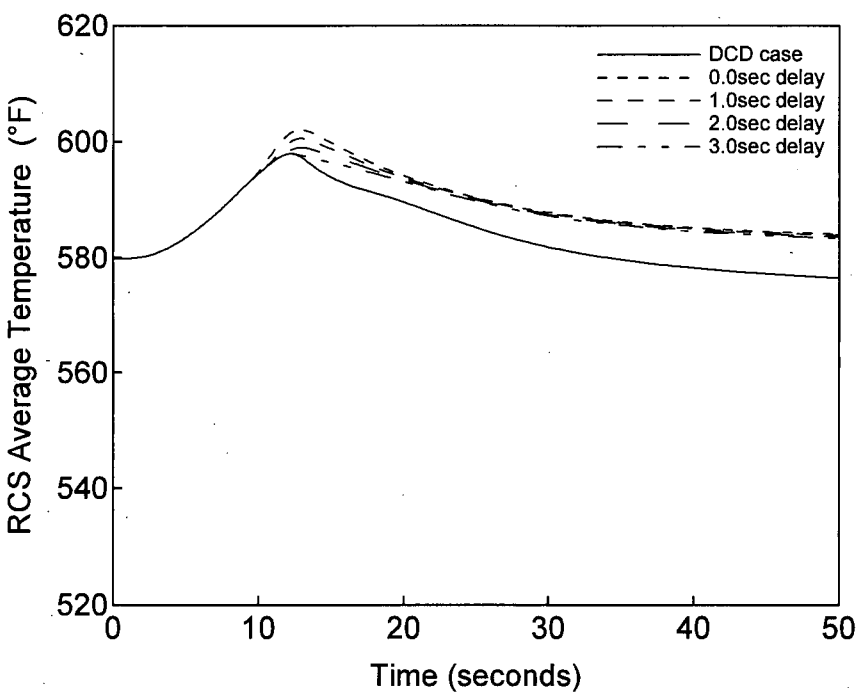

Figure 15.0.0-3.30 RCS Average Temperature versus Time with and without LOOP Loss of External Load - RCS & Main Steam Pressure Analysis

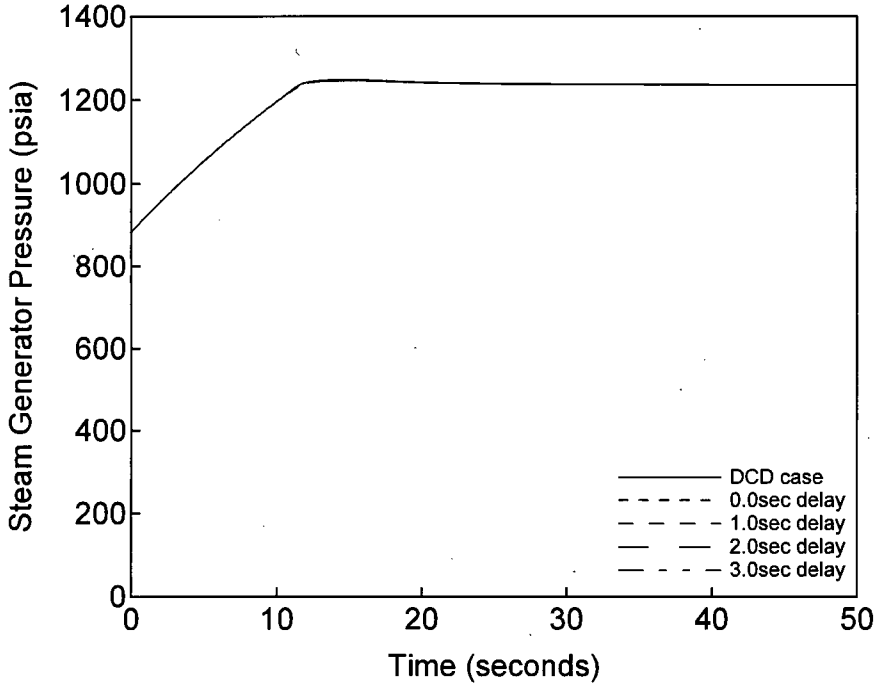

Figure 15.0.0-3.31 Steam Generator Pressure versus Time with and without LOOP Loss of External Load - RCS & Main Steam Pressure Analysis

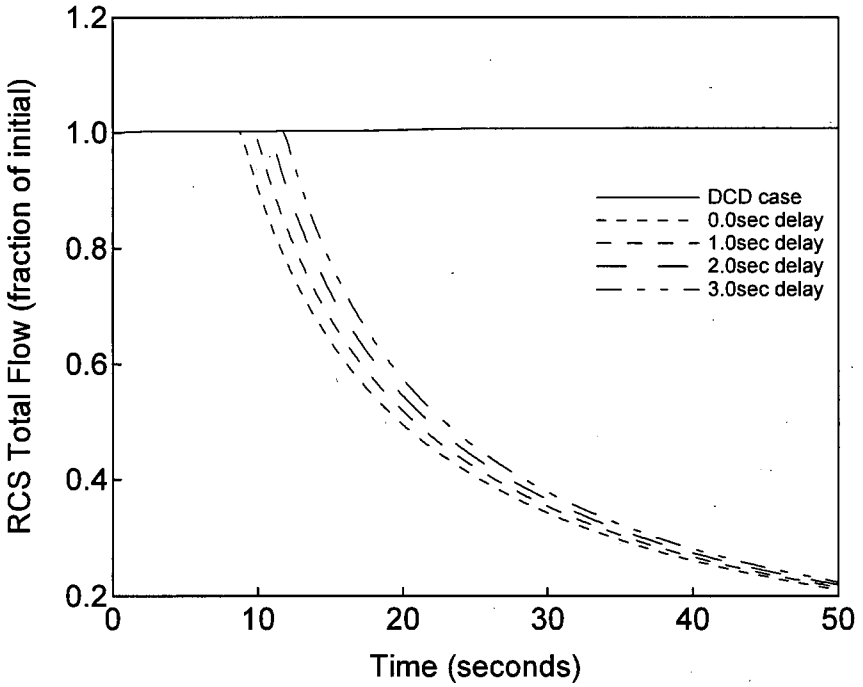

Figure 15.0.0-3.32 RCS Total Flow versus Time with and without LOOP Loss of External Load - RCS & Main Steam Pressure Analysis

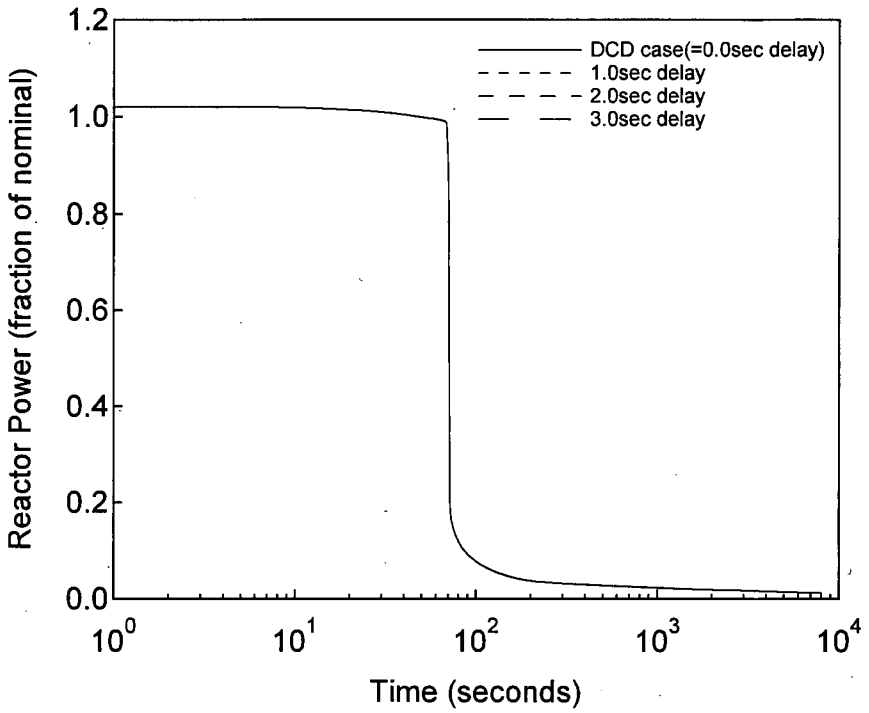

Figure 15.0.0-3.33 Reactor Power versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

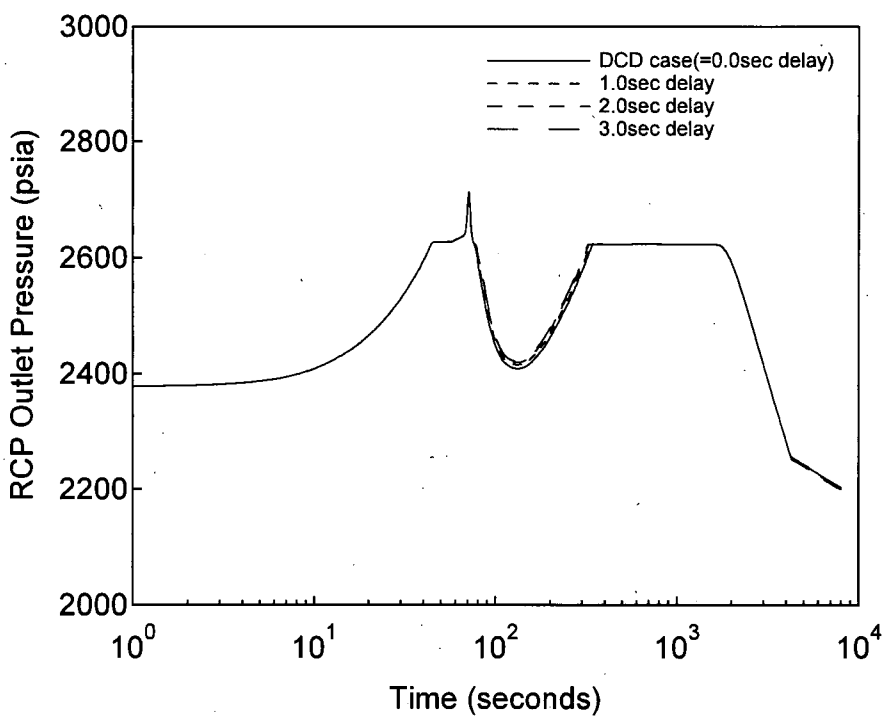

Figure 15.0.0-3.34 RCP Outlet Pressure versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

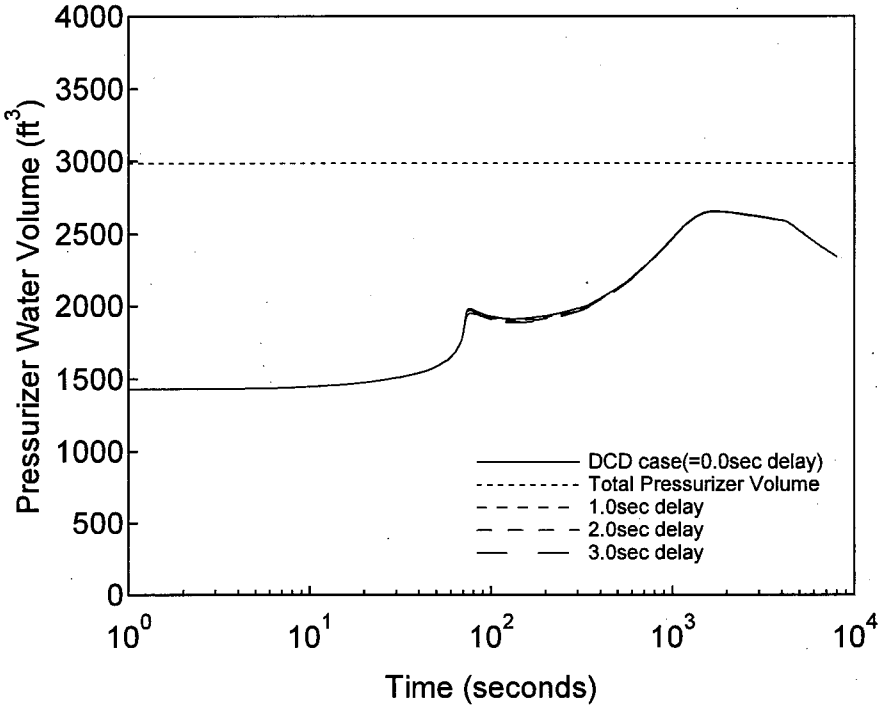

Figure 15.0.0-3.35 Pressurizer Water Volume versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

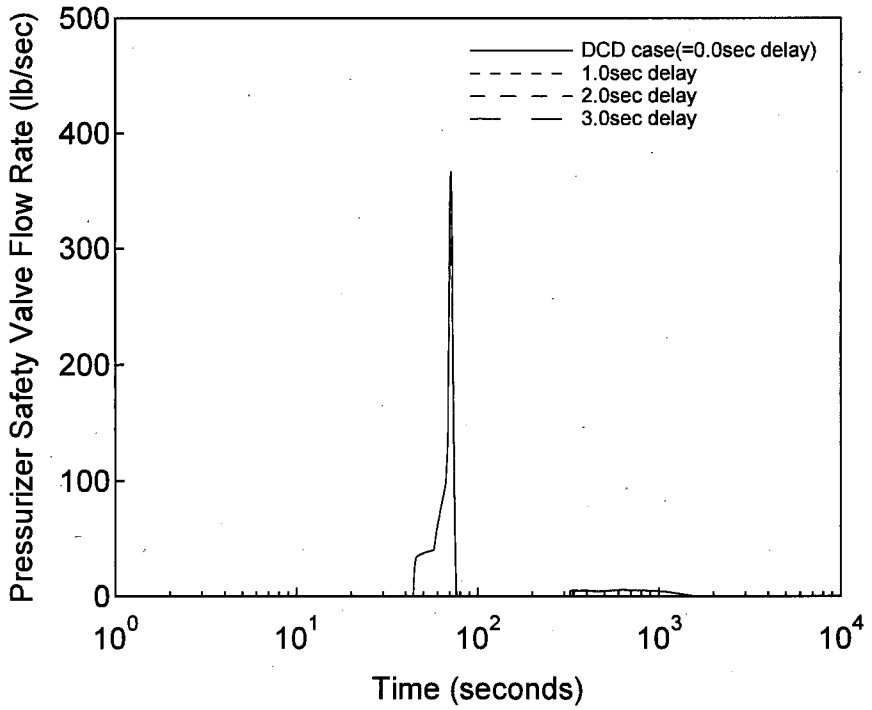

Figure 15.0.0-3.36 Pressurizer Safety Valve Flow Rate versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

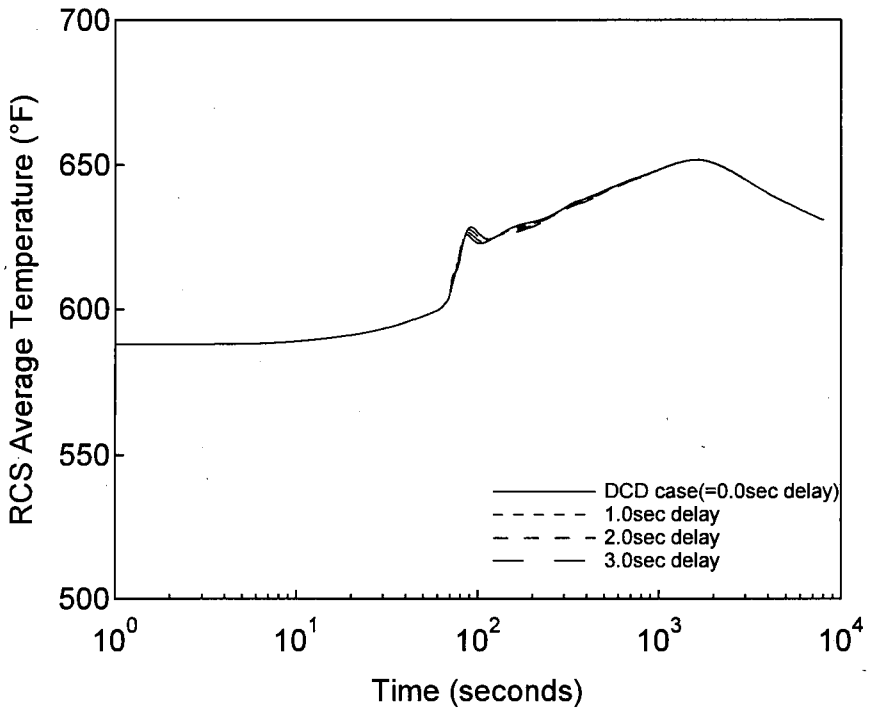

Figure 15.0.0-3.37 RCS Average Temperature versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

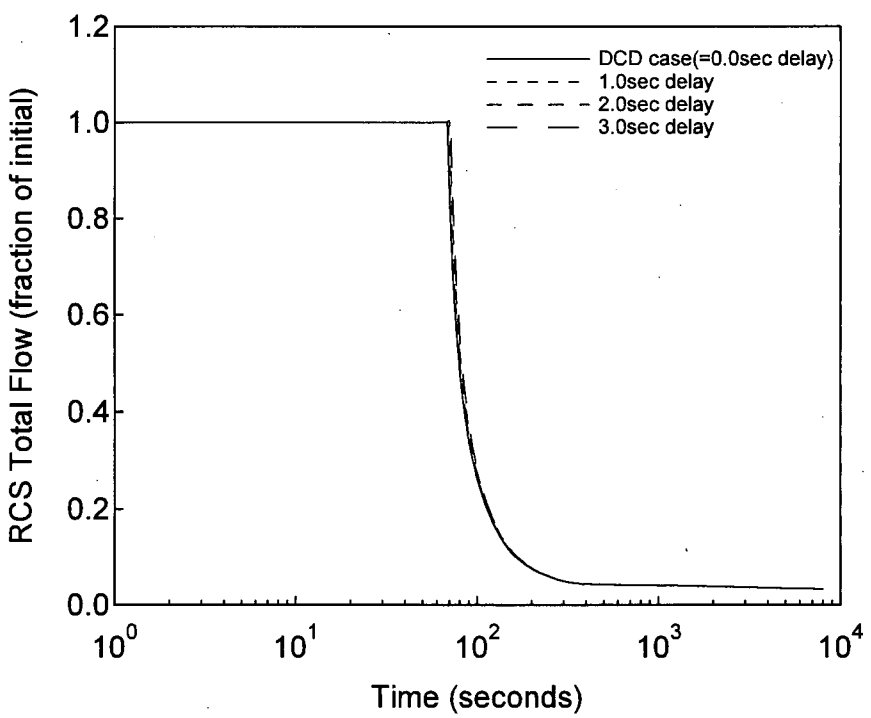

Figure 15.0.0-3.38 RCS Total Flow versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

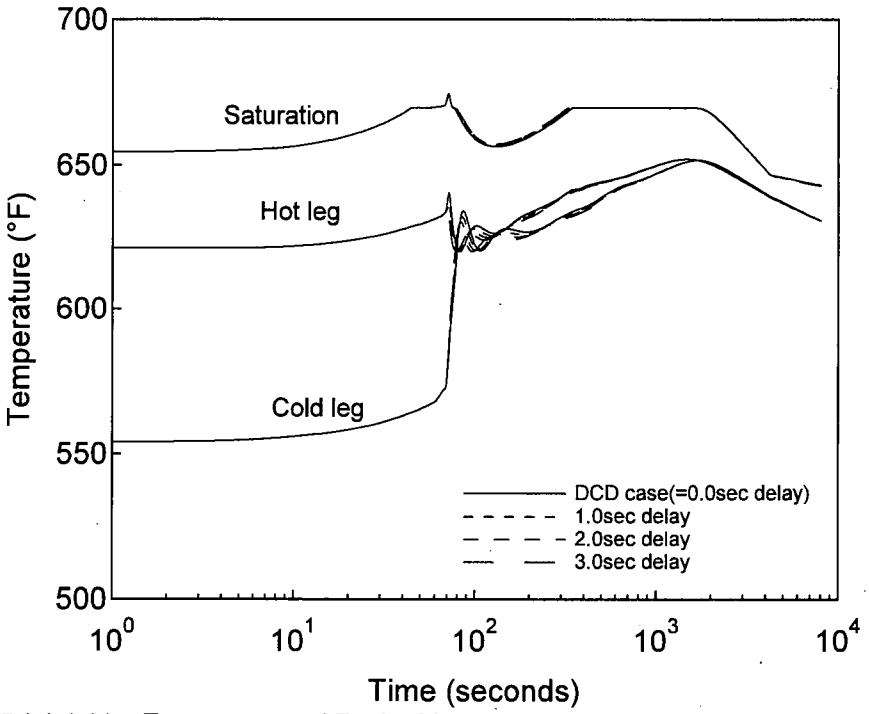

Figure 15.0.0-3.39 Temperature of Faulted Loop versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

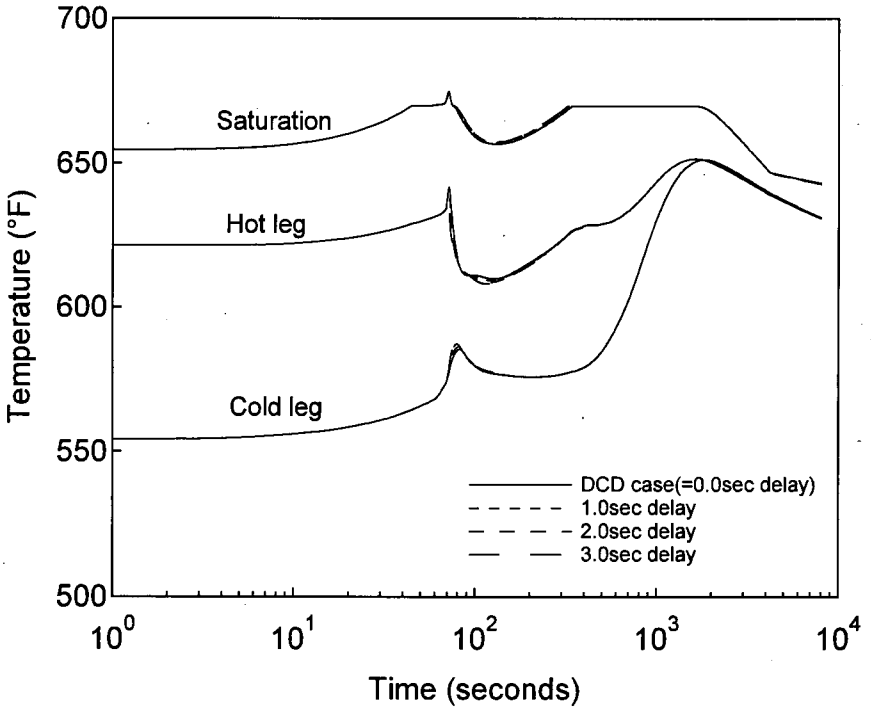

Figure 15.0.0-3.40 Temperature of Intact Loop without EFW versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

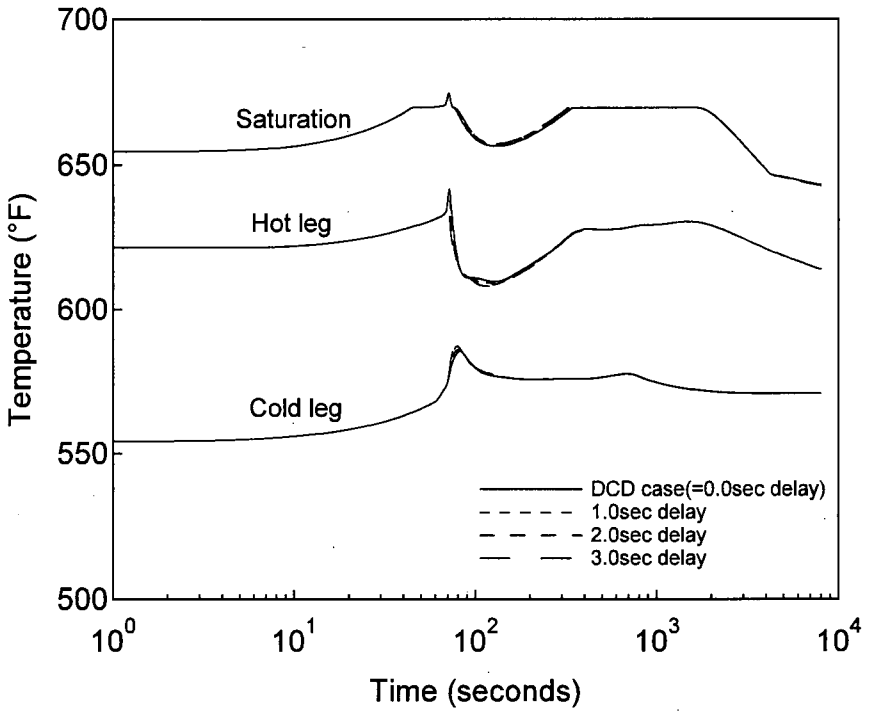

Figure 15.0.0-3.41 Temperature of Intact Loop with EFW versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

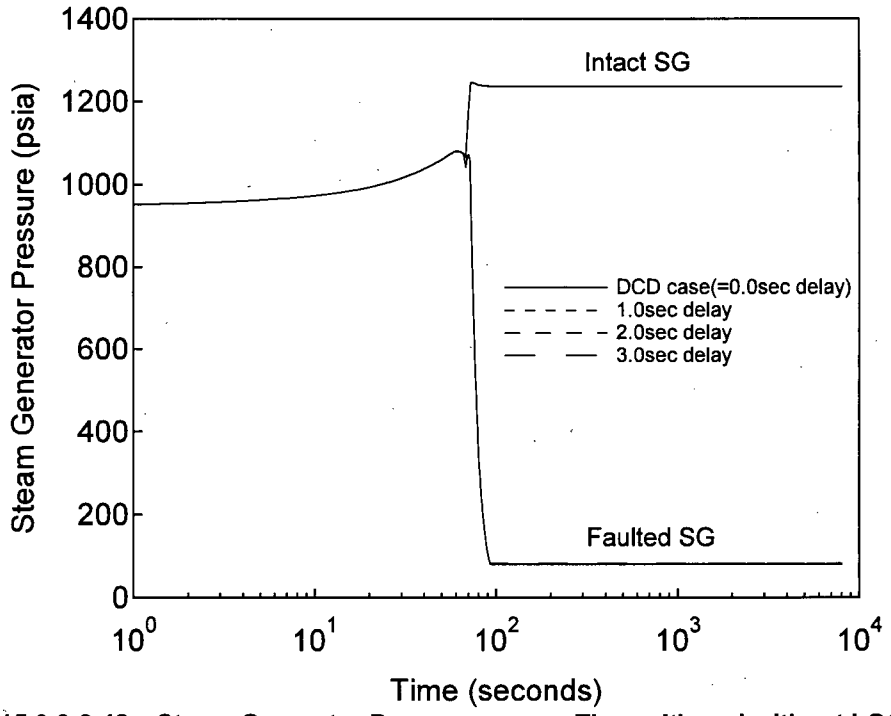

Figure 15.0.0-3.42 Steam Generator Pressure versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

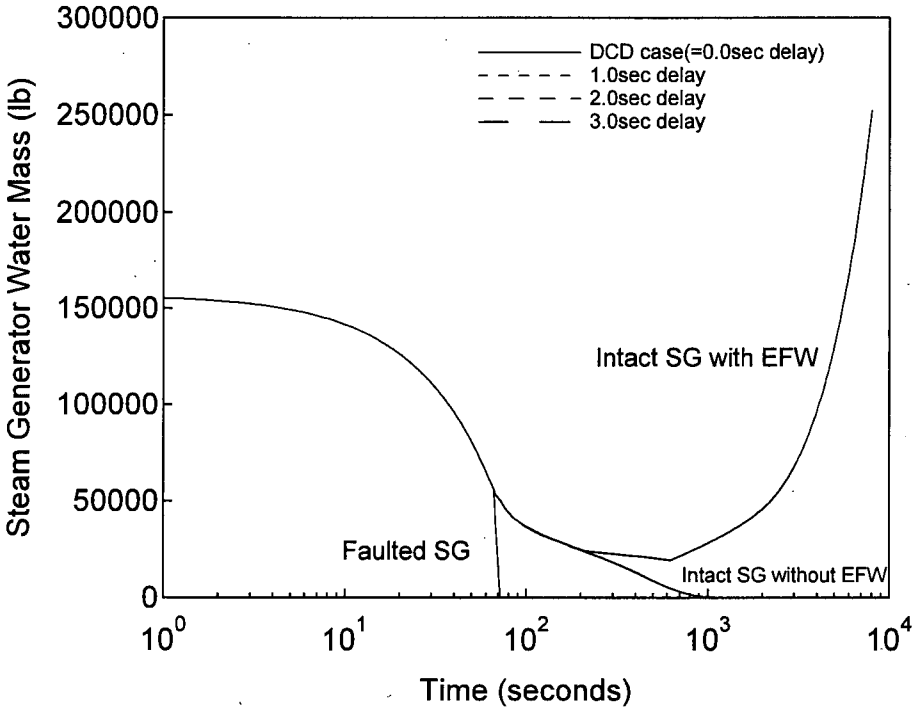

Figure 15.0.0-3.43 Steam Generator Water Mass versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

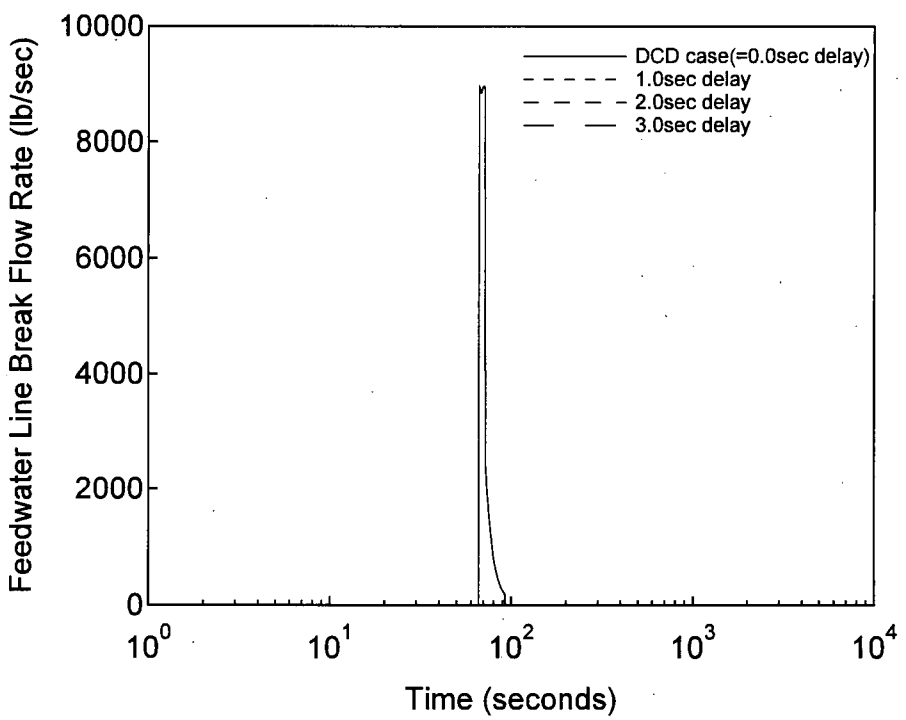

Figure 15.0.0-3.44 Feedwater Line Break versus Time with and without LOOP Feedwater System Pipe Break - RCS Pressure Analysis

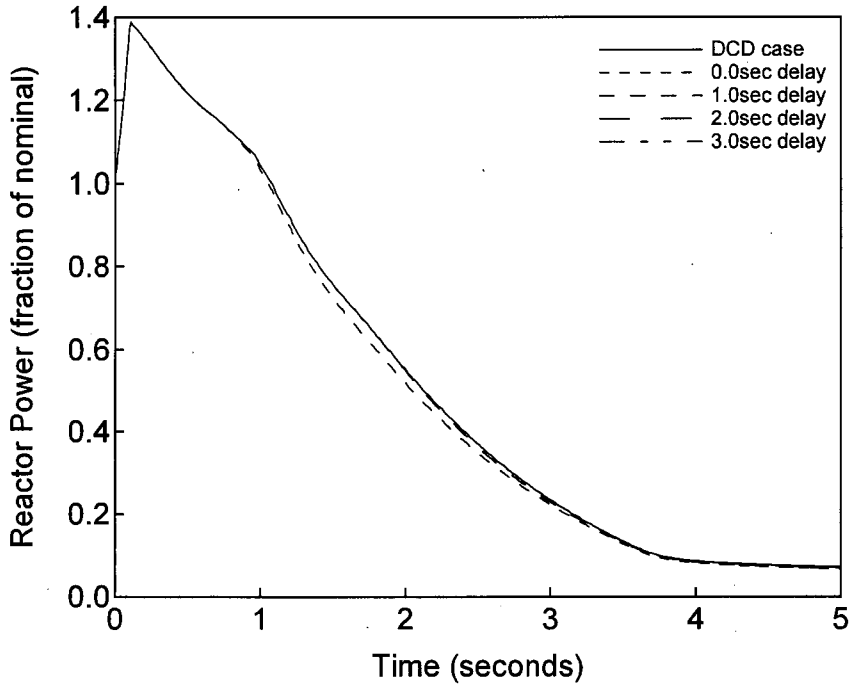

Figure 15.0.0-3.45 Reactor Power versus Time with and without LOOP Rod Ejection (HFP, EOC)

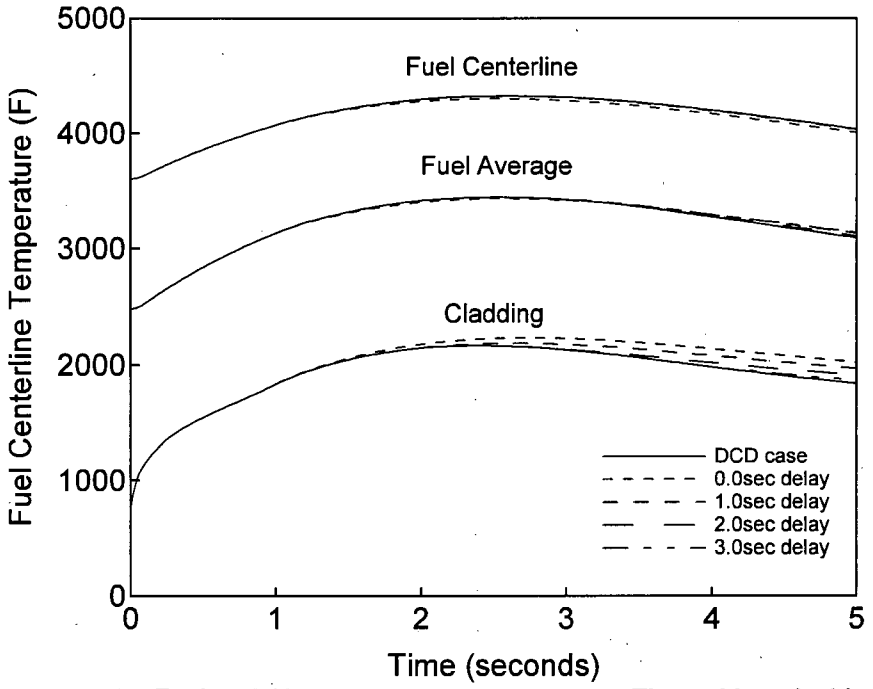

Figure 15.0.0-3.46 Fuel and Cladding Temperature versus Time with and without LOOP Rod Ejection (HFP, EOC)

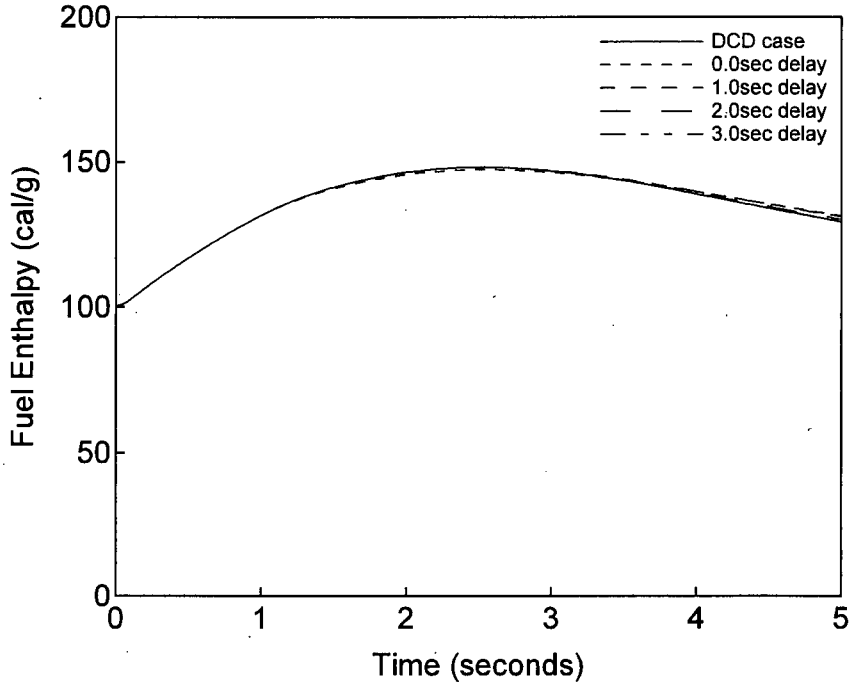

Figure 15.0.0-3.47 Radial Average Fuel Enthalpy versus Time with and without LOOP Rod Ejection (HFP, EOC)

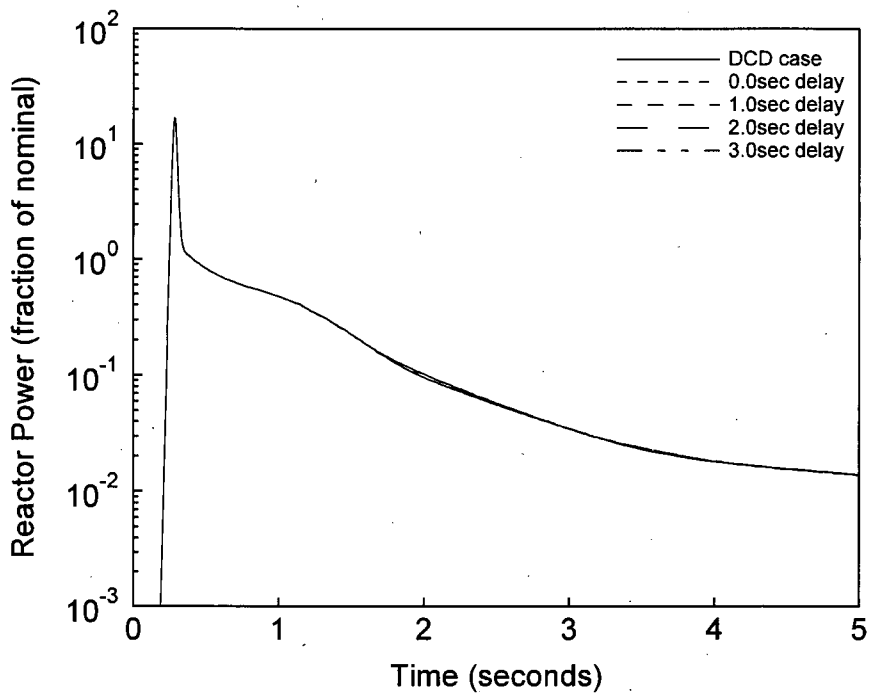

Figure 15.0.0-3.48 Reactor Power versus Time with and without LOOP Rod Ejection (HZP, BOC)

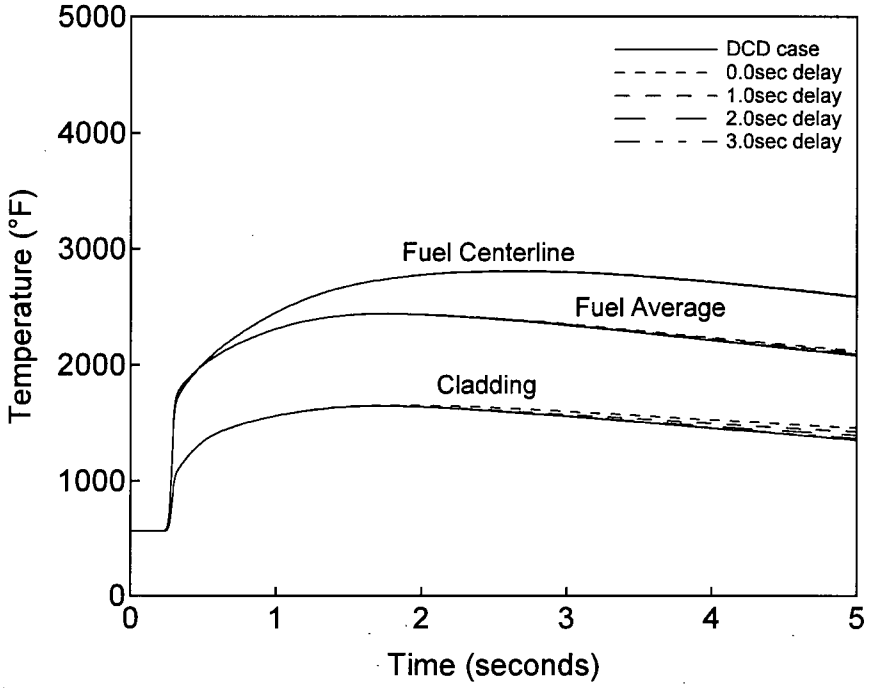

Figure 15.0.0-3.49 Fuel and Cladding Temperature versus Time with and without LOOP Rod Ejection (HZP, BOC)

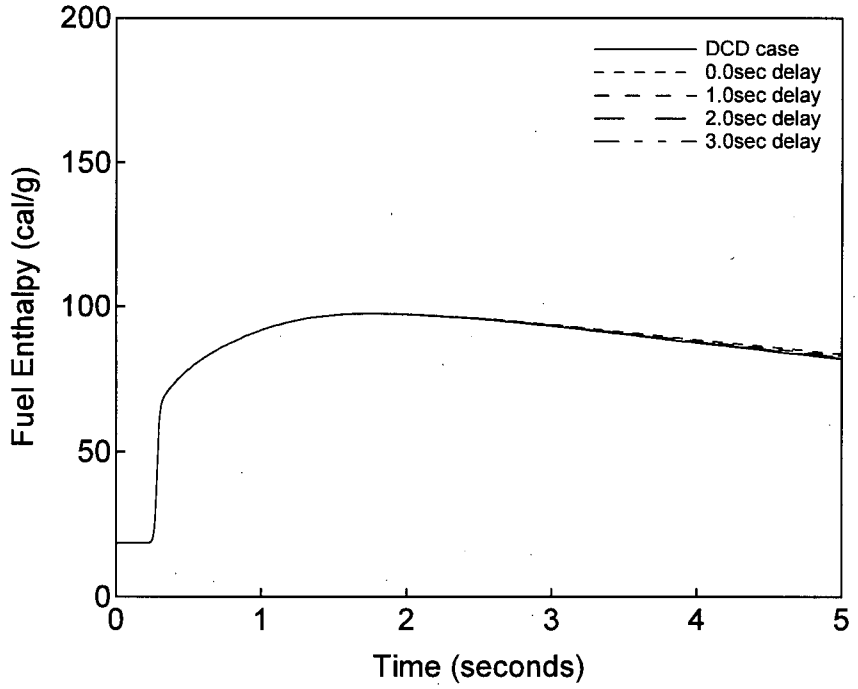

Figure 15.0.0-3.50 Radial Average Fuel Enthalpy versus Time with and without LOOP Rod Ejection (HZP, BOC)

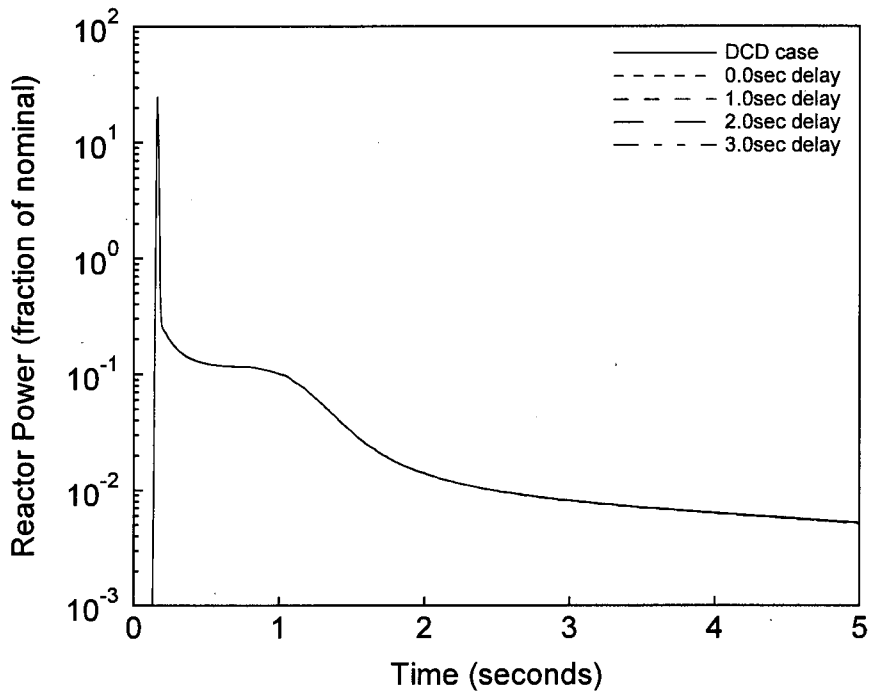

Figure 15.0.0-3.51 Reactor Power versus Time with and without LOOP Rod Ejection (HZP, EOC)

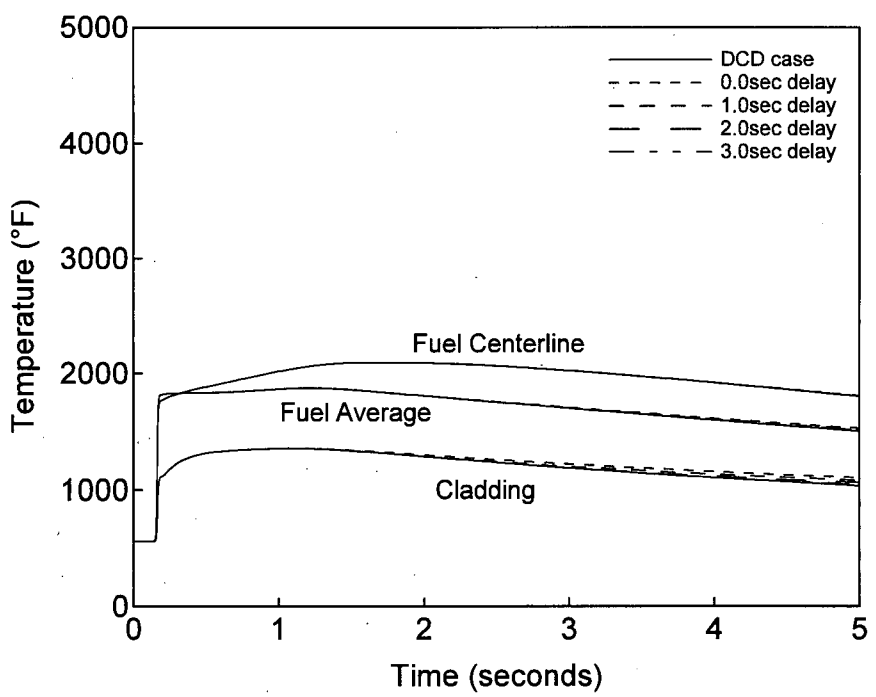

Figure 15.0.0-3.52 Fuel and Cladding Temperature versus Time with and without LOOP Rod Ejection (HZP, EOC)

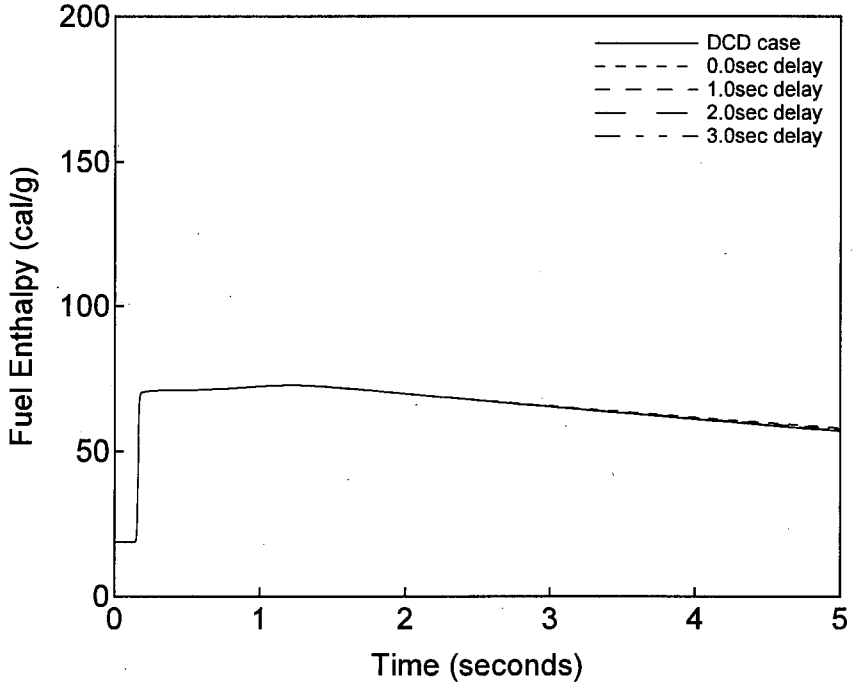

Figure 15.0.0-3.53 Radial Average Fuel Enthalpy versus Time with and without LOOP Rod Ejection (HZP, EOC)

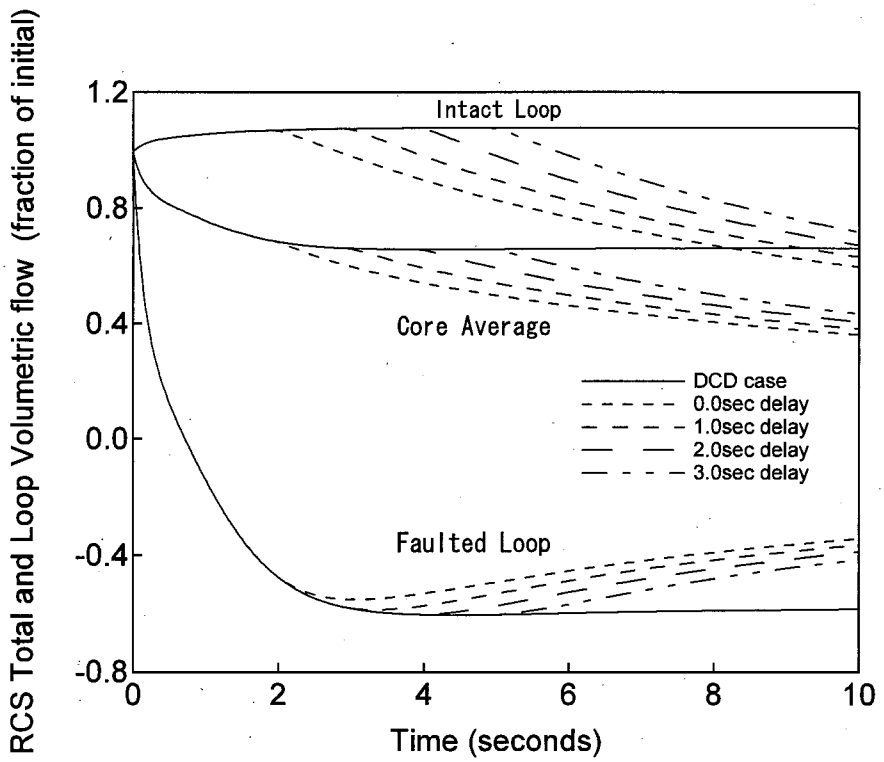

Figure 15.0.0-3.54 RCS Total and Loop Volumetric Flow versus Time with and without LOOP
RCP Rotor Seizure
- Cladding Temperature Analysis

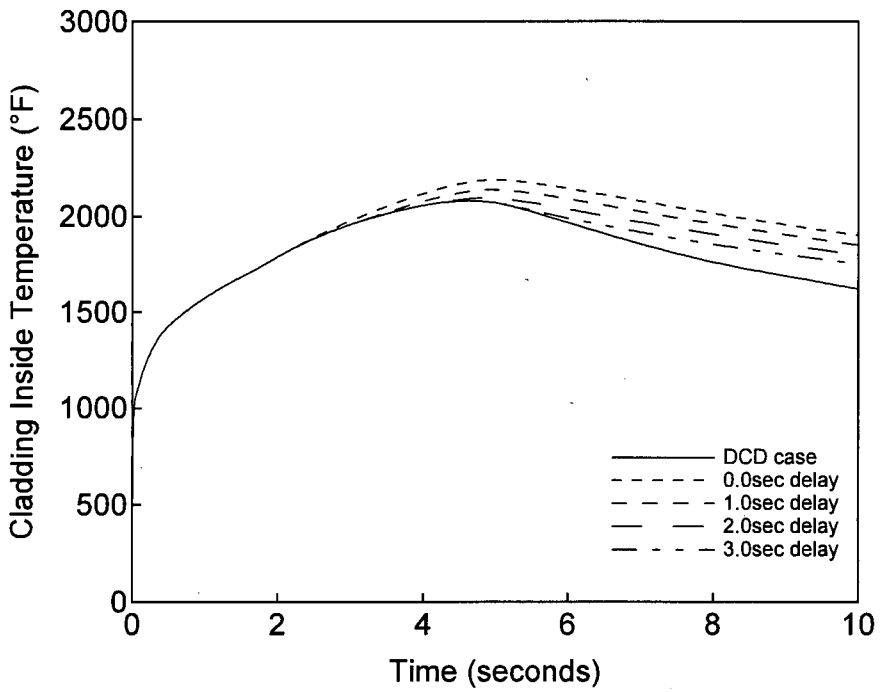

Figure 15.0.0-3.55 Cladding Inside Temperature versus Time with and without LOOP RCP Rotor Seizure - Cladding Temperature Analysis

Impact on DCD

DCD Section 15.0.0.7 will be revised to more accurately describe the LOOP assumptions used for pressure analyses. This DCD change is shown in the "Impact on DCD" section of the response to Question 15.0.0-2 of this RAI.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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.: 15.0.0-4

Does the applicant intend to conduct a grid stability analysis to demonstrate that the grid will remain stable, as assumed in the Chapter 15 analyses, for a minimum of three seconds following a turbine trip, thus delaying the initiation of LOOP and RCP coast down for at least three seconds following a turbine trip event? Or is this issue deferred to COL activities?

ANSWER:

DCD Subsection 8.2.3 states that the COL applicant is to perform a grid stability analysis to confirm the DCD Subsection 15.0.0.7 assumption that the grid will remain stable for a minimum of three seconds following a turbine trip. COL Item 8.2(11) captures this requirement.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

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QUESTION NO.: 15.0.0-5

The applicant presents arguments that suggest that the three-second delay to the initiation of LOOP following turbine trip is considered a constant parameter in all applicable analyses. How was the three-second delay determined, what is the calculated uncertainty in this value, how was the uncertainty calculated, and has this uncertainty been applied in the analyses?

ANSWER:

The basis for the three-second delay assumed in DCD Subsection 15.0.0.7 is (1) that it is acceptable for the accident analysis and (2) it can be confirmed to be applicable for potential new plant locations in the US based on a grid stability analysis. In addition, since the time for the control rods to reach the dashpot following any reactor trip is on the order of 3 seconds, the DNBR and power transients will be essentially terminated by the time the assumed grid-response time has elapsed.

The margin between the results using a three-second delay and a safety analysis analytical limit determines the uncertainty in this parameter. Sensitivity analyses have confirmed that even with a RCP coastdown delay as small as 1 second, the accident analysis results will remain acceptable. The transient analyses performed to confirm this sensitivity analyses utilize the normal uncertainties. The results of these sensitivity analyses are described in the response to Question 15.0.0-3 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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QUESTION NO.: 15.0.0-6

Provide the results of an assessment for the design basis events analyzed in Chapter 15 of the FSAR that the analyzed event scenario in each of the transients and accidents will bound the results of the specific event occurring from various power level and all modes of plant operation.

ANSWER:

MHI previously performed an analysis to document that the events presented in Chapter 15 of the US-APWR DCD bound the results for other modes and initial power levels. This analysis was submitted to the NRC by MHI letter UAP-HF-08045, dated February 27, 2008, in response to NRC questions during the Acceptance Review of the US-APWR DCD. A revised version of this analysis, with a few minor editorial changes, is shown in Table 15.0.0-6.1 below. The conclusion remains the same; the cases presented in the DCD are the most limiting.

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.

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Reactivity (K_{eff})	NA	<0.99	<0.99	<0.99		≥0.99	≥0.99	
%Rated Thermal Power	NA	NA	NA	NA		≤5	>5	
Avg. Reactor Coolant Temp., T_{avg} (F)	NA	≤200	350>T_{avg}>200	≥350		NA (No Load)	NA (from No Load to HFP-T_{avg})	
				~350 to No Load	No Load		Partial Power	HFP
Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions (15.1.1)	Non-limiting due to low T_{avg} and margin to criticality.	Same as Mode 6	Same as Mode 6	Same as Mode 6	Under no-load conditions, the rate of energy change is reduced as feedwater flow decreases, making the no-load case less severe than the HFP case.		Same as Mode 2	Discussed in the DCD
Increase in Feedwater Flow as a Result of Feedwater System Malfunctions (15.1.2)	Non-limiting due to low T_{avg} and margin to criticality.	Same as Mode 6	Same as Mode 6	Same as Mode 6	Core reactivity is more severe than at HFP due to increased feedwater supplied at a lower temperature and later Doppler feedback. However, reactivity insertion is bounded by the 15.4.1 event.		Same as Mode 2	Discussed in the DCD
Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction (15.1.3)	Same as 15.1.4 event	Same as 15.1.4 event	Same as 15.1.4 event	Same as 15.1.4 event		The peak power is much less than at HFP because of the neutron flux high trip (low setpoint).	Reactivity insertion is higher in this case due to the higher secondary pressure. However, lower initial power and coolant temperature ensure that the minimum DNBR is bounded by the HFP case and 15.1.4 event.	Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Inadvertent Opening of a Steam generator Relief or Safety Valve (15.1.4)	Non-limiting because secondary system steam release cannot reduce T_{avg} below T_{sat} at atmospheric pressure.	Same as Mode 6	High shutdown margin keeps core subcritical or significantly limits power increase when critical.		Discussed in the DCD	Bounded by 15.1.3 event	Bounded by 15.1.3 event	Bounded by 15.1.3 event
Steam System Piping Failures Inside and Outside of Containment (15.1.5)	Non-limiting because secondary system steam release cannot reduce T_{avg} below T_{sat} at atmospheric pressure.	Same as Mode 6	High shutdown margin keeps core subcritical or significantly limits power increase when critical.		Discussed in the DCD	Bounded by the at-power and hot standby DCD cases	Discussed in the DCD, including sensitivity to initial power (100%, 75%)	
Loss of External Load (15.2.1)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Non-limiting since the reactor is at zero power.			Non-limiting due to smaller load loss and greater margin to DNB and overpower trips relative to HFP case.		Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Loss of Non-Emergency AC Power to the Station Auxiliaries (15.2.6)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Same as Mode 6	Non-limiting since the reactor is at zero power.		When the reactor power is greater than the P-7 setpoint (10%), the same R/T as for HFP occurs from a lower initial power. Therefore, this case is bounded by the HFP case. When the reactor power is less than the P-7 setpoint, low RCS flow, low RCP speed, and TT R/T signals are blocked. However, the loss of non-emergency AC power results in the loss of power to the MG set, which, in turn, causes a R/T. Additionally, the initial power for the partial power cases is sufficiently low so that the pressure response is bounded by the HFP case. The ability to establish sufficient natural circulation to remove nuclear power or decay heat at these power levels is demonstrated for the HFP case.		Discussed in the DCD
Loss of Normal Feedwater (15.2.7)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Same as Mode 6	Non-limiting since the reactor is at zero power.		Same as Mode 1 Partial Power	Bounded by HFP case since initial power is less than at HFP and R/T is initiated by low SG water level.	Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Feedwater System Pipe Break Inside and Outside Containment (15.2.8)	Non-limiting since the core is in RHR cooling.	Same as Mode 6	Same as Mode 6	A FLB at zero power is not limiting.		Same as Mode 1 Partial Power.	Bounded by HFP case since initial power is less than at HFP and R/T is initiated by low SG water level.	Discussed in the DCD
Partial Loss of Forced Reactor Coolant Flow (15.3.1.1)	RCPs are in stand-by; therefore, there is no event initiation.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		When the reactor power is less than the P-7 setpoint, the low RCS flow R/T signal is blocked. However, the low initial power and minimum of two RCPs running results in significant DNB margin to the nominal operating conditions. For initial power levels greater than the P-7 setpoint, the flow transient and time of trip is the same as HFP, but the higher HFP power level is bounding.		Discussed in the DCD
Complete Loss of Forced Reactor Coolant Flow (15.3.1.2)	RCPs are in stand-by; therefore, there is no event initiation.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		For initial power above the P-7 setpoint, the trip and flow response is the same as for HFP and the lower initial power is less limiting than HFP. For initial conditions below the P-7 setpoint where the low flow and low RCP speed trips are blocked, the combination of low core power and the associated natural circulation flow is sufficient to protect the fuel design limits. Therefore, these cases are not limiting.		Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Reactor Coolant Pump Rotor Seizure & Shaft Break (15.3.3 & 15.3.4)	RCPs are in stand-by; therefore, there is no event initiation.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		When the reactor power is less than the P-7 setpoint, low RCS flow R/T signal is blocked. However, the initial power is sufficiently low that this case is not limiting. Above the P-7 setpoint, flow and trip response is the same for all initial power levels as HFP, which is bounding because of the higher initial power.		Discussed in the DCD
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition (15.4.1)	CRDM is in stand-by; therefore, there is no event initiation.	RCCA operation is administratively restricted. (Core power cannot be used for the RCS heating)	Same as Mode 5	Same as Mode 5	Discussed in the DCD	Bounded by the DCD case	Bounded by 15.4.2 event	NA
Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2)	NA	NA	NA	NA		Bounded by 15.4.1 event and Mode 1 spectrum of cases	Discussed in the DCD (100%, 75%, 10%)	
One or More Dropped RCCAs within a Group or Bank (15.4.3)	All RCCAs are fully inserted; therefore, there is no event initiation.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		Same as Mode 1 Partial Power	Non-limiting because the initial power is lower than HFP case.	Discussed in the DCD
Uncontrolled Withdrawal of a Single RCCA (15.4.3)	The core has sufficient criticality margin to prevent an occurrence of criticality caused by a single RCCA withdrawal event.	Same as Mode 6	Same as Mode 6	Same as Mode 6		Same as Mode 1 Partial Power.	Non-limiting because the initial power is lower than HFP case.	Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (15.4.6)	Discussed in the DCD	Discussed in the DCD	Discussed in the DCD	Discussed in the DCD		Discussed in the DCD	Discussed in the DCD	Discussed in the DCD
Spectrum of Rod Ejection Accidents (15.4.8)	The core has sufficient criticality margin to prevent an occurrence of criticality caused by a rod ejection event.	Same as Mode 6	Same as Mode 6	Same as Mode 6		Discussed in the DCD (HZP)	The analysis conditions such as reactivity assumed in the HFP case and the HZP case bound partial power cases.	Discussed in the DCD (HFP)
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (15.5.2)	The reactor vessel head is detached; therefore, there is no pressure increase.	Pressurizer is water-solid. Control of the pressurizer water level and LTOP system prevent RCS and RHR system overpressurization due to the addition of reactor coolant inventory caused by a CVCS malfunction.	Same as Mode 5	Bounded by the HFP case because pressurizer water level is lower than at HFP, resulting in more time to terminate the event.		Same as Mode 3	Same as Mode 3	Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve (15.6.1)	Non-limiting due to low initial RCS pressure.	Non-limiting since the reactor is at zero power.	Same as Mode 5	Same as Mode 5		Same as Mode 1 Partial Power.	When the reactor power is less than the P-7 setpoint, the low RCS pressure R/T signal is blocked. However, a R/T on low RCS pressure SI signal occurs; therefore, this case is not limiting.	Discussed in the DCD
Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (15.6.2)	Same as Mode 1 Partial Power.	Same as Mode 1 Partial Power.	Same as Mode 1 Partial Power.	Same as Mode 1 Partial Power.		Same as Mode 1 Partial Power.	Bounded by HFP case because release due to flashing of leaked coolant is smaller than at HFP.	Discussed in the DCD
Radiological Consequences of Steam Generator Tube Failure (15.6.3)	There is no primary-to-secondary coolant leakage because the primary system is not pressurized.	RCS pressure and temperature are low enough to prevent opening of MSRV and/or MSSV.	Same as Mode 5	Same as Mode 5	The heat storage in the primary system is less than at HFP conditions, thus time needed to attain primary-to secondary pressure balance is shorter than at HFP. This reduces the primary-to-secondary coolant leakage and the steam release. Therefore, this is not a limiting case.		Same as Mode 2	Discussed in the DCD

Table 15.0.0-6.1 DCD Limiting Case Selection Matrix

Event	Mode-6 Refueling	Mode-5 Cold Shutdown	Mode-4 Hot Shutdown	Mode-3 Hot Standby		Mode-2 Startup	Mode-1 Power Operation	
				T < No Load	No Load		Partial Power	HFP
Loss-of- Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (15.6.5)	Bounded by Mode 5	The decrease of inventory is slow, since the RCS pressure is low. Therefore, non-limiting since there is time for manual start of SI.	Bounded by Mode 3	When the RCS pressure is less than the P-11 setpoint, ECCS actuation signal is blocked. However, non-limiting since safety injection and accumulator injection start automatically upon receipt of the High containment pressure ECCS actuation signal.		Bounded by the DCD case, since initial power is lower than HFP case.		Discussed in the DCD
Fuel Handling Accident (15.7.4)	Discussed in the DCD	This accident is not expected to occur during this mode, because in-vessel fuel handling is not carried out during this mode.	Same as Mode 5	Same as Mode 5		Same as Mode 5	Same as Mode 5	

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DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.0.0-7

Confirm that the assumptions used in the transient and accident analyses are consistent with the range of values specified in technical specifications.

ANSWER:

The Technical Specifications (TS) in Chapter 16 of the DCD describe various conditions within which the plant must be operated. Only certain of the TS are directly or indirectly related to the Chapter 15 accident analyses, so an initial screen was made to eliminate certain TS requirements not related to this response. The remaining TS parameters and values are addressed below, relative to the accident analysis. To facilitate review, the parameters are presented by TS number and generally in the order they appear in the TS. Certain parameters that were not eliminated by the initial screen and subsequently determined to be unrelated to the safety analysis are described below for completeness. In the context of this response, only the Chapter 15 transient analysis is addressed. For example, the control room dose analyses and containment response analyses associated with certain Chapter 15 events are separate analyses that are evaluated in Chapter 6.

<u>Parameters to Be Defined in the Core Operating Limits Report (COLR)</u>

In some cases, the TS do not currently give an explicit value. Instead, the specification for these items states that the value is to be within the limits specified by the Core Operating Limits Report (COLR). As defined in Section 1.1 of Chapter 16, the COLR is the document that provides cycle-specific parameter limits. The cycle-specific parameter limits are determined for each cycle in accordance with Specification 5.6.3. As described in Specification 5.6.3, the COLR establishes the limits for the following:

- 2.1.1 - Reactor Core Safety Limits (SLs) (thermal power, RCS average temperature, & pressurizer pressure)
- 3.1.1 - Shutdown Margin
- 3.1.3 - Moderator Temperature Coefficient
- 3.1.5 - Shutdown Bank Insertion Limits
- 3.1.6 - Control Bank Insertion Limits
- 3.2.1 - Heat Flux Hot Channel Factor

- 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor
- 3.2.3 - Axial Flux Difference
- 3.3.1 - Reactor Trip System Instrumentation
- 3.4.1 - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits
- 3.9.1 - Boron Concentration

Item (c) of TS 5.6.3 states that the COLR limits shall be determined such that the applicable limits of the safety analysis are met. As a result, these eleven TS described in the COLR will, by definition, be consistent with the values assumed for the accident analyses in Chapter 15.

The remaining TS from Chapter 16 are addressed individually below.

- 2.1.1.1, DNBR Limit – The TS DNBR limit specified is the 95/95 limit used as the AOO acceptance criterion for DNB cases.
- 2.1.1.2, Peak Fuel Centerline Temperature – The TS peak fuel centerline temperature as described in DCD Subsection 4.2.1.2.1, is used as the AOO acceptance criterion. This definition is consistent with the TS limit.
- 2.1.2, Reactor Coolant System Pressure Safety Limit – The TS limit (≤2735 psig) corresponds to 110% of the design pressure, which is the AOO acceptance criterion used for the Chapter 15 peak pressure cases.
- 3.1.2, Core Reactivity – This TS is to keep the measured and calculated reactivity within a certain range and is not applicable to the Chapter 15 safety analyses.
- 3.1.4, Rod Group Alignment Limits – This TS says that all rods shall be operable. Some of the safety analyses conservatively assume a control rod is not operable (i.e. is stuck). In addition, the total scram reactivity used in the safety analysis does not credit the most reactive rod.
- 3.2.4, Quadrant Power Tilt Ratio – This TS is not applicable to the Chapter 15 safety analyses.
- 3.3.1, Reactor Trip System Instrumentation – TS Table 3.3.1-1 contains the nominal reactor trip setpoint values and an allowable value uncertainty. The analytical limit trip setpoint values used in the accident analyses shown in DCD Table 15.0-4 are consistent with (conservative with respect to) the TS allowable values for events that credit reactor trips. Certain of the trip setpoint parameters such as the Over Temperature ΔT and Over Power ΔT may be cycle-specific and will be specified in the COLR, consistent with the accident analysis.
- 3.3.1, Reactor Trip System Instrumentation – TS Table 3.3.1-1 also contains the reactor trip function required channels (or trains) operable. The assumed single failures shown in DCD Table 15.0-6 are consistent with the TS required channels (or trains) operable in that the required operable channels with an additional single failure will not affect the actuation of the credited RTS trip function. TS Table 3.3.1-1 also shows the applicable requirements for each operating mode. Reactor trips are only credited in the accident analysis if they are available during the analysis conditions for the event, (e.g., power operation, subcritical).
- 3.3.2, Engineered Safety Feature Actuation System Instrumentation – TS Table 3.3.2-1 contains the ESFAS setpoint values and required channels (or trains) operable. As with the RTS instrumentation, the accident analysis analytical limit conservatively bounds the TS allowable value, and the required operable channels with an additional single failure will not affect the actuation of the credited ESFAS function.
- 3.3.3, Post Accident Monitoring Instrumentation – This TS is not applicable to the Chapter 15 safety analyses.
- 3.3.4, Remote Shutdown Console – This TS is not applicable to the Chapter 15 safety analyses.
- 3.3.5, Loss of Power Class 1E Gas Turbine Generator Start Instrumentation – This TS is not applicable to the Chapter 15 safety analyses. Each of the gas turbine generators is a

support system to one or more mitigating systems (e.g., EFW, SI, etc.). A GTG failure to start is modeled in the safety analyses as the loss of the mitigating system(s) it supports.

- 3.3.6, Diverse Actuation System Instrumentation – This TS is not applicable to the Chapter `15 safety analyses.
- 3.4.2, RCS Minimum Temperature for Criticality – The Chapter 15 safety analyses use the nominal RCS temperature, including uncertainties, and are therefore consistent with this TS.
- 3.4.3, RCS Pressure and Temperature Limits – This TS refers to the Pressure and Temperature Limit Report (PTLR) which will be developed according to TS 5.6.4. This is similar to the TS requirements that are deferred to the COLR, and which will, by definition, be consistent with the accident analysis.
- 3.4.4, RCS Loops - Modes 1 and 2 – This TS states that all four loops shall be operable for these modes. For the events that start from these modes, the safety analyses initially assume four loops are operable (four RCPs running). The inactive loop startup event is not analyzed, consistent with this TS. The events analyzed from HZP conditions assume four RCPs running.
- 3.4.5, RCS Loops - Mode 3 – This TS requires one or two RCPs running depending on whether the rod control system is capable of rod withdrawal or not, respectively. The boron dilution event credits boron mixing in the RCS consistent with this TS.
- 3.4.6, RCS Loops - Mode 4 – This TS requires any combination of two CS/RHR and RCS loops to be operable and one loop in operation. The boron dilution event credits boron mixing in the RCS consistent with this TS.
- 3.4.7, RCS Loops - Mode 5 Loops Filled – This TS requires two CS/RHR loops to be operable and in operation. The boron dilution event uses the conservatively small RHR loop volume and credits boron mixing in the RHR loop consistent with this TS.
- 3.4.8, RCS Loops - Mode 5 Loops Not Filled – This TS is not applicable to the Chapter 15 safety analyses.
- 3.4.9, Pressurizer – This TS provides the maximum pressurizer water level (≤92%) and minimum number of operable pressurizer heaters. The safety analyses assume the nominal pressurizer water level (approximately 44%) plus uncertainty for the initial condition. This approach is analogous to and consistent with initial conditions for DNB events. For DNB events that utilize the Revised Thermal Design Procedure, the initial conditions for average temperature, RCS pressure, and power are assumed at the nominal operating conditions. These conditions are pre-defined and maintained by plant control systems including the rod control system and pressurizer pressure control system. When these control systems are unavailable or in manual control, the plant is maintained at the nominal conditions by the operator. The uncertainties for these parameters are statistically treated as part of the approved RTDP. For DNB events that do not use the RTDP, the initial conditions for average temperature, RCS pressure, and power are assumed at the nominal condition with the normal uncertainties applied in the conservative direction for each of the parameters. The uncertainties include both process measurement uncertainty and control system deadbands. Similarly, for events resulting in a challenge to filling the pressurizer, the initial pressurizer water level is the nominal level as predefined and maintained by the pressurizer level control system, with the uncertainty applied in the conservative direction. For Chapter 15 events, the pressurizer heaters are only assumed to operate when their operation has an adverse effect on the results. In these cases, the maximum design heater capacity is assumed independent of the TS.
- 3.4.10, Pressurizer Safety Valves – This TS gives the pressure range for which the safety valves should open. For the DCD Section 15.2 events and RCCA ejection (peak RCS pressure case) that credit the pressurizer safety valves to limit RCS pressure, the safety valves are conservatively assumed to open at a pressure at or above the maximum in the TS.

- 3.4.11, Safety Depressurization Valves – This TS states that two SDVs shall be operable, which is consistent with the Chapter 15 Steam Generator Tube Rupture analyses that credits one of the SDVs.
- 3.4.12, Low Temperature Overpressure Protection System – This TS is not applicable to the Chapter 15 safety analyses.
- 3.4.13, RCS Operational Leakage – This TS is not applicable to the Chapter 15 safety analyses.
- 3.4.14, RCS Pressure Isolation Valve Leakage – This TS is credited in the radiological consequences evaluation.
- 3.4.15, RCS Leakage Detection Instrumentation – This TS is not applicable to the Chapter 15 safety analyses.
- 3.4.16, RCS Specific Activity – This TS is credited in the radiological consequences evaluation.
- 3.4.17, Steam Generator Tube Integrity – This TS is not applicable to the Chapter 15 safety analyses.
- 3.5.1, Accumulators – This TS states that four accumulators shall be operable, which is consistent with the analyses that credit the accumulators (LOCA). The accumulators are not credited for any non-LOCA events. Failure of one train of Safety Injection in the LOCA analyses conservatively includes failure of the associated accumulator. The accumulator borated water volume, nitrogen cover pressure, and boron concentration used in the LOCA analyses as defined in DCD Tables 15.6.5-1, 15.6.5-2, and 15.6.5-3 conservatively bound or fall within the allowable ranges of the TS.
- 3.5.2, Safety Injection System - Operating – This TS states that three SI trains shall be operable. This is consistent with the safety analyses assuming one SI train has a single failure and an additional train out for online maintenance for the events that credit SI (Inadvertent Opening of a Steam Generator Safety or Relief Valve, Main Steam Line Break, and LOCA). However, the SGTR safety analysis conservatively assumes all four SI trains are operable to maximize primary-to-secondary leakage.
- 3.5.3, Safety Injection System - Shutdown – This TS is not applicable to the Chapter 15 safety analyses.
- 3.5.4, Refueling Water Storage Pit – The TS values for the RWSP are consistent with the safety analyses that credit the RWSP. In the case of the cooldown events crediting SI analyzed in DCD Subsections 15.1.4 & 15.1.5, the minimum temperature and minimum boron concentration is assumed, consistent with this TS. The RWSP inventory requirement is not challenged by any of the Chapter 15 events. The LBLOCA ASTRUM statistical analysis samples the temperature range (45°F to 120°F) as described in DCD Table 15.6.5-1. The RWSP temperature rise is modeled in the SBLOCA with the initial condition at the maximum temperature as described in DCD Table 15.6.5-2. Safety injection temperature of post-LOCA long term cooling analysis is assumed to be maximum to maximize the core evaporation rate as described in Table 15.6.5-3.
- 3.5.5, pH Adjustment – This TS is credited in the containment sump pH analysis.
- 3.6, Containment Systems – None of the Technical Specifications in this section are applicable to the Chapter 15 safety analyses.
- 3.7.1, Main Steam Safety Valves (MSSVs) – All of the MSSVs (six per steam line) are credited in the non-LOCA events that credit the MSSVs. The MARVEL-M code models all the valves on each steam line as a single valve using a non-mechanistic flow model. The "valve" is assumed to ramp from zero to full rated flow on a very steep ramp at 103% of steam system design pressure (103% of 1200 psia = 1236 psia). The results of the events that challenge the steam line safety valves confirm that the maximum steam system pressure is well within the acceptance criterion of 110% system design pressure.
- 3.7.2, Main Steam Isolation Valves (MSIVs) – This TS requires four operable MSIVs with a valve closure time of 5 seconds or less, which is consistent with the scenario defined for the limiting Steam System Piping Failure analyzed in DCD Subsection 15.1.5 (one steam generator is unisolatable). For this same analysis, the time of steam line isolation (10 seconds) includes an allowance for valve closure of at least 5 seconds, consistent

with this TS. The same assumptions are used in the containment mass and energy release analyses analyzed in DCD Subsection 6.2.1.4.

- 3.7.3, Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs), Main Feedwater Bypass Regulation Valves (MFBRVs), and Steam Generator Water Filling Control Valves (SGWFCVs) – This TS requires four operable MFIVs, MFRVs, MFBRVs, and SGWFCVs with a valve closure time of 5 seconds or less. This requirement is consistent with the scenario defined for the limiting Steam System Piping Failure analyzed in DCD Subsection 15.1.5. Because there are two isolation valves in series for each feed water flow path, feed water is terminated to all four steam generators by the ECCS Actuation signal. For this analysis, the time of feed water line isolation (10 seconds) includes an allowance for valve closure of at least 5 seconds, consistent with this TS. The same assumptions are used in the containment mass and energy release analyses analyzed in DCD Subsection 6.2.1.4.
- 3.7.4, Main Steam Depressurization Valves (MSDVs) – This TS requires that four MSDVs be operable. The Steam Generator Tube Rupture analysis in DCD Subsection 15.6.3 assumes that two of the intact steam generators are used to cool down the RCS using the MSDVs. This is consistent with this TS if one MSDV is on the affected loop and a single failure occurs on one of the intact steam generator MSDVs.
- 3.7.5, Emergency Feedwater System (EFWS) – This TS requires four operable EFW trains (with all EFW pump discharge cross-connect isolation valves in all trains closed). The Steam Generator Tube Rupture (DCD Subsection 15.6.3), Loss of Non-Emergency AC Power to the Station Auxiliaries (DCD Subsection 15.2.6), Loss of Normal Feedwater Flow (DCD Section 15.2), and Feedwater System Pipe Break (DCD Subsection 15.2.8) all assume two EFW trains, consistent with this TS.
- 3.7.6, EFW Pits – This TS requires two operable EFW pits with at least 204,850 gallons each. This is not directly related to the Chapter 15 accident analysis since none of the events challenge the EFW pit inventory during the short durations of the events. The inventory requirement is a long-term heat removal requirement.
- 3.7.14, Secondary Specific Activity – This TS is credited in the radiological consequences evaluation.
- 3.8, Electrical Power Systems – None of the TS in this section are applicable to the Chapter 15 safety analyses.
- 3.9.2, Unborated Water Source Isolation Valves (During Refueling Operations) – This TS requires that each valve used to isolate unborated water sources be secured in the closed position during Mode 6. The Boron Dilution event described in DCD Subsection 15.4.6 takes credit for this administrative control as the basis for not performing a quantitative analysis during Mode 6.
- 3.9.8, Decay Time – This TS will be newly established for the decay time prior to fuel handling. This TS is credited in the radiological consequences evaluation for fuel handling accidents.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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.: 15.0.0-8

Provide the assessment for the parameters, initial conditions, and single failures used in various transients and accidents to support the values and sequence of events assumed in each event that would lead to the most conservative results with respect to each of the acceptance criteria.

ANSWER:

The parameters, initial conditions, and single failures used in the transients are event-specific. This information is described in the individual subsections of DCD Chapter 15, rather than in the introduction. The assessment of the single failures assumed in each of the events is given in the response to Question 15.0.0-11 of this RAI. The assessment of the initial power and plant operation mode assumed in each of the events is given in response to Question 15.0.0-6 of this RAI. The assessment of other initial conditions and core parameters is given in the discussion and tables below.

MHI performed a study to evaluate the effect of minor changes in the selection of initial conditions and core parameters used to analyze each of the Chapter 15 events. The results of the sensitivity study were compared to the DCD case to determine the impact on the relevant acceptance criteria. Table 15.0.0-8.1 summarizes the specific parameters that were evaluated for this study and the change used in the sensitivity study. The change in initial power, T_{avg}, and pressure are given in terms of their nominal values, which can be obtained from DCD Table 15.0-3. Also, the initial conditions and core parameters assumed for each event are given in DCD Table 15.0-1 and are not repeated in this response. However, the analysis for certain acceptance criteria for some events uses initial conditions based on the revised thermal design procedure (RTDP) and therefore are not varied during this sensitivity study.

Table 15.0.0-8.2 shows the results of the sensitivity analysis for the minimum DNBR evaluation. The results are given in terms of the percent difference based on the equation below. A positive value means that the DNBR for the sensitivity case is higher than the DCD case and is therefore less limiting.

$$\text{Percent difference (\%)} = \frac{MDNBR_{sensitivity} - MDNBR_{DCD}}{MDNBR_{DCD}} \times 100$$

where:

$MDNBR_{sensitivity}$ = Minimum DNBR of the sensitivity analysis case

$MDNBR_{DCD}$ = Minimum DNBR of the DCD case

As shown in the table, all of the values are either zero or positive. Therefore, the initial conditions and core parameters assumed in the DCD lead to the most conservative results with respect to DNBR.

Table 15.0.0-8.3 shows the results of the sensitivity analysis for the maximum RCS pressure and maximum pressurizer water level evaluations. The results are given in terms of the percent difference based on the equations below. A negative value means that the pressure or water level for the sensitivity case is lower than the DCD case and is therefore less limiting.

$$\text{Percent difference (\%)} = \frac{P^{max}_{sensitivity} - P^{max}_{DCD}}{P^{max}_{DCD}} \times 100$$

where:

$P^{max}_{sensitivity}$ = Peak RCS pressure of the sensitivity analysis case

P^{max}_{DCD} = Peak RCS pressure of the DCD case

$$\text{Percent difference (\%)} = \frac{V^{max}_{sensitivity} - V^{max}_{DCD}}{V^{max}_{DCD}} \times 100$$

where:

$V^{max}_{sensitivity}$ = Peak pressurizer water volume of the sensitivity analysis case

V^{max}_{DCD} = Peak pressurizer water volume of the DCD case

As shown in the table, all of the values analyzed for RCS pressure are either zero or negative. Therefore, the initial conditions and core parameters assumed in the DCD lead to the most conservative results with respect to RCS pressure. As shown in the table, the values analyzed for pressurizer water level are either zero or negative, with two exceptions. [

]

Table 15.0.0-8.4 shows the results of the sensitivity analysis evaluated against several different criteria. The results are given in terms of the percent difference. The first set of analyses is for maximum heat flux. A negative value means that the maximum heat flux for the sensitivity case is lower than the DCD case and is therefore less limiting. Since all of the values in the columns are either zero or negative, the initial conditions and core parameters assumed in the DCD lead to the most conservative results with respect to maximum heat flux. The next analysis is for $\Delta T_{subcooling}$. A positive value means that the $\Delta T_{subcooling}$ for the sensitivity case is higher than the DCD case and is therefore less limiting. Since all of the values in the column are either zero or

positive, the initial conditions and core parameters assumed in the DCD lead to the most conservative results with respect to subcooling margin. The next analysis is for Peak Cladding Temperature (PCT). A negative value means that the PCT for the sensitivity case is less than the DCD case and is therefore less limiting. Since all of the values in the column are either zero or negative, the initial conditions and core parameters assumed in the DCD lead to the most conservative results with respect to PCT. The next analysis is for the time from the high pressurizer level alarm to the pressurizer being filled. A positive values means that there will be more time for operator actions and is therefore less limiting. Since all of the values in the column are either zero or positive, the initial conditions and core parameters assumed in the DCD lead to the most conservative results with respect to the amount of time for operator actions. The next analysis is for steam generator level. Since all of the values in this column are zero, this parameter is not sensitive to the initial conditions and core parameters and the DCD case is conservative with respect to steam generator water level. The final case is for maximum primary-to-secondary leakage. A negative value means that there is less leakage and is therefore less limiting. [

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Table 15.0.0-8.5 also shows the results of the sensitivity analysis evaluated for fuel centerline temperature, fuel enthalpy, and adiabatic fuel enthalpy. The results are given in terms of the percent difference. For these three parameters, a negative value means that the temperature or enthalpy is lower than the DCD and is therefore less limiting. All of the values in the table are either zero or negative with one exception. [

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Table 15.0.0-8.1
Sensitivity Analysis to Initial Conditions and Core Parameters

Initial Condition or Core Parameter	DCD Case	Sensitivity Case
Initial Power	Nominal + 2%	
Initial T_{avg}	Nominal + 4°F	
	Nominal – 4°F	
Initial Pressure	Nominal + 30 psi	
	Nominal – 30 psi	
Moderator Density Coefficient	Maximum	
	Minimum	
Moderator Temperature Coefficient	-20% from design	
Doppler Coefficient	Maximum	
	Minimum	

Table 15.0.0-8.2
Sensitivity of Minimum DNBR to Initial Conditions and Core Parameters for Applicable Chapter 15 Events (results shown in percent difference)

DCD Subsection	15.1.1	15.1.2	15.1.3	15.2.1	15.2.7	15.3.1.1	15.3.1.2	15.3.3	15.4.1	15.4.2	15.4.3	15.6.1
Analysis Case	DNB	DNB	DNB[1]	DNB	DNB	DNB	DNB	DNB	DNB	DNB[2]	DNB	DNB
Criteria Analyzed	Minimum DNBR											
Initial Power												
Initial T_{avg}												
Initial Pressure												
Moderator Density Coefficient												
Moderator Temperature Coefficient												
Doppler Coefficient												

Table 15.0.0-8.3
Sensitivity of Peak RCS Pressure and Peak Pressurizer Water Level to Initial Conditions and Core Parameters for Applicable Chapter 15 Events (results shown in percent difference)

DCD Subsection	15.2.1	15.2.7	15.2.8	15.3.3	15.4.1	15.4.8	15.2.6	15.2.7[2]	15.2.8
Analysis Case	Peak Pressure	Peak Pressure	Peak Pressure	Peak Pressure	Peak Pressure	Peak Pressure	Pressurizer Overfill	Pressurizer Overfill	Pressurizer Overfill
Criteria Analyzed	RCS Pressure						Pressurizer Water Level		
Initial Power									
Initial T_{avg}									
Initial Pressure									
Moderator Density Coefficient									
Moderator Temperature Coefficient									
Doppler Coefficient									

Table 15.0.0-8.4
Sensitivity of Other Acceptance Criteria to Initial Conditions for Applicable Chapter 15 Events
(results shown in percent difference)

DCD Subsection	15.1.4	15.1.5	15.1.5	15.2.8	15.3.3	15.5.2	15.6.3	15.6.3[3]
Analysis Case	HZP	HZP Case A	HFP Case C	RCS Boiling	PCT	Pressurizer Level	Steam Generator Overfill	Radiological Consequences
Criteria Analyzed	Maximum Heat Flux			$\Delta T_{subcooling}$	PCT	Time from Alarm to Pressurizer Filled	Steam Generator Level	Maximum Primary-to-Secondary Leakage
Initial Power								
Initial T_{avg}								
Initial Pressure								
Moderator Density Coefficient								
Doppler Coefficient								

Table 15.0.0-8.5
Sensitivity of Other Acceptance Criteria to Initial Conditions for Applicable Chapter 15 Events
(results shown in percent difference)

DCD Subsection	15.4.8	15.4.8	15.4.8	15.4.8	15.4.8	15.4.8
Analysis Case	BOC HFP	EOC HFP	BOC HZP	EOC HZP	BOC HZP	EOC-HZP
Criteria Analyzed	Fuel Centerline Temperature		Fuel Enthalpy		Adiabatic Fuel Enthalpy	
Initial Power						
Initial T_{avg}						
Initial Pressure						
Moderator Temperature Coefficient						
Doppler Coefficient[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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

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**
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QUESTION NO.: 15.0.0-9

Confirm that each of the transients and accidents analyzed has been assessed against multiple acceptance criteria, including specified acceptable fuel design limits (SAFDL), the maximum primary pressure, maximum secondary side pressure, and the minimum departure from nucleate boiling ratio (MDNBR).

ANSWER:

Each of the events analyzed in DCD Chapter 15 has been assessed against all applicable acceptance criteria. The comparison of the results to the acceptance criteria is event-specific and shown in the conclusions of each individual event subsection of DCD Chapter 15. In addition, a summary of the assessment to multiple acceptance criteria (peak primary pressure, peak secondary pressure, and minimum DNBR) for each event is provided in the response to Question 15.0.0-16 of this RAI.

In addition to the three parameters addressed in the response to Question 15.0.0-16 of this RAI, the following events have been evaluated in their applicable DCD Chapter 15 subsection for other specified acceptable fuel design limits (SAFDLs) as summarized in Table 15.0.0-9.1 below.

Table 15.0.0-9.1
Chapter 15 Events Evaluated Against Additional SAFDLs

DCD Subsection & Event Description	Additional SAFDLs Evaluated
15.3.3 Locked Rotor	Peak Clad Temperature
15.4.1 RCCA Withdrawal from Subcritical	Peak Fuel Centerline Temperature
15.4.8 Rod Ejection (HZP)	PCMI (Includes Peak Fuel Enthalpy)
15.4.8 Rod Ejection (HFP)	Peak Fuel Enthalpy, Peak Centerline Temperature
15.6.5 LBLOCA	Peak Clad Temperature, Cladding Oxidation
15.6.5 SBLOCA	Peak Clad Temperature, Cladding Oxidation

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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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 COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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
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
	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

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
15.3.1.2	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
	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

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
15.6.1	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%	
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	
15.2.6	Loss of non-emergency AC power to the station auxiliaries			-4	2525	1236		10%	
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%	

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
15.2.8	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%	
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%	

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
15.4.7	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			
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

7/03/2009

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**
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QUESTION NO.: 15.0.0-11

FSAR Table 15.0-6 lists the assumed most-limiting single failure in each of the analyzed design basis transients and accidents. How were the assumed failures listed in Table 15.0-6 determined? It appears that most of the assumed single failures listed in this table are the failure of one of the redundant trip functions, emergency core cooling systems, or emergency feedwater systems. These assumptions are not likely to affect the results of the safety analyses since the available redundant system will perform the required safety function. The applicant should provide a sensitivity study for various single failures to determine the most limiting assumed single failure that would lead to the most limiting consequences of each event.

ANSWER:

The selection of the limiting single failure for each event evaluated in Chapter 15 of the US-APWR DCD was determined by a systematic event-specific review of the progression of the event and assumed mitigative equipment and its associated function. The primary steps associated with the performance of this review are as follows:

1) Determine the mitigative functions (i.e., systems) assumed operational in the safety analysis of each event. The results of this step are shown in Table 15.0.0-11.1 below. (The legend and footnotes are at the end of the table.)
2) Determine whether the equipment needed to perform the mitigative function have single failure design assumptions. The results of this step are shown in Table 15.0.0-11.2 below.
3) Compare the results of Steps 1 and 2 on an event-specific basis to determine the single failure that results in the most severe analysis result. The results of this step are described as follows:

For the analysis of the feedwater temperature decrease event in Subsection 15.1.1, there is no mitigative equipment for which a single failure should be assumed.

For the analysis of the excessive feedwater flow event in Subsection 15.1.2, the only mitigative equipment is the Reactor Trip System (RTS). Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

For the analysis of the excessive steam flow event in Subsection 15.1.3, there is no mitigative equipment for which a single failure should be assumed.

For the analysis of the inadvertent secondary depressurization event in Subsection 15.1.4, the mitigative equipment is the boron injection of the ECCS and emergency feedwater isolation. A single failure of the boron injection of the ECCS reduces the amount of borated water injection flow and therefore reduces the effectiveness of the reactivity control. A single failure of an emergency feedwater isolation valve does not result in a loss of function. Therefore, the failure of 1 train of ECCS is selected as the most severe single failure.

For the analysis of the steam system piping failure (hot shutdown) event in Subsection 15.1.5, the mitigative equipment is the boron injection of the ECCS, emergency feedwater isolation, and the GTG. A single failure of the boron injection of the ECCS reduces the amount of borated water injection flow and therefore reduces the effectiveness of the reactivity control. A single failure of an emergency feedwater isolation valve does not result in a loss of function. A single failure of the GTG results in the loss of the same train of ECCS, CSS, RHRS, and EFWS, of which, only the ECCS is assumed in the analysis and its single failure is already considered as just described. Therefore, the failure of 1 train of ECCS is selected as the most severe single failure.

For the analysis of the steam system piping failure (hot full power) event in Subsection 15.1.5, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

For the analysis of the loss of external load in Subsection 15.2.1, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

The analyses of the turbine trip, loss of condenser vacuum, and closure of MSIV events in Subsections 15.2.2, 15.2.3, and 15.2.4, respectively, are bounded by the analysis of the loss of external load event in Subsection 15.2.1.

The analysis of the steam pressure regulator failure event in Subsection 15.2.5 is not applicable to the US-APWR.

For the loss of non-emergency AC power event in Subsection 15.2.6, the mitigative equipment is the RTS, the EFWS, and the GTG. A single failure of the RTS does not result in a loss of function. A single failure of the EFWS reduces the amount of EFW flow and therefore reduces the heat removal capability. A single failure of the GTG results in the loss of the same train of ECCS, CSS, RHRS, and EFWS, of which, only the EFWS is assumed in the analysis and its single failure is already considered as just described. Therefore, the failure of 1 train of EFWS is selected as the most severe single failure.

For the analyses of the loss of normal feedwater flow and feedwater system pipe break events in Subsections 15.2.7 and 15.2.8, respectively, the mitigative equipment is the same as for the loss of non-emergency AC power event in Subsection 15.2.6. Therefore, the most severe single failure for these two events is the failure of 1 train of EFWS.

For the analyses of the partial and complete loss of flow events in Subsections 15.3.1.1 and 15.3.1.2, respectively, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analyses. However, a single failure of the RTS does not result in a loss of function and the safety analyses are not affected.

The analysis of the flow controller malfunction event in Subsection 15.3.2 is not applicable to the US-APWR.

For the analysis of the RCP rotor seizure event in Subsection 15.3.3, the mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

The analysis of the RCP shaft break event in Subsection 15.3.4 is bounded by the analysis of the RCP rotor seizure event in Subsection 15.3.3.

For the analyses of the RCCA bank withdrawal from subcritical and at power events in Subsections 15.4.1 and 15.4.2, respectively, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analyses. However, a single failure of the RTS does not result in a loss of function and the safety analyses are not affected.

For the analyses of the one or more dropped RCCAs and single rod withdrawal events in Subsection 15.4.3, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analyses. However, a single failure of the RTS does not result in a loss of function and the safety analyses are not affected. No transient analysis is performed for the misaligned RCCA event in Subsection 15.4.3.

The analyses of the startup of an inactive loop and flow controller malfunction events in Subsections 15.4.4 and 15.4.5, respectively, are not applicable to the US-APWR.

For the analysis of the boron dilution event in Subsection 15.4.6, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

No transient analysis is performed for the improper fuel loading event in Subsection 15.4.7.

For the analysis of the rod ejection event in Subsection 15.4.8, the mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

The analysis of the inadvertent ECCS operation at power event in Subsection 15.5.1 is not applicable to the US-APWR.

For the analysis of the CVCS malfunction event in Subsection 15.5.2, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

For the analysis of the inadvertent RCS depressurization event in Subsection 15.6.1, the only mitigative equipment is the RTS. Therefore, this is the single failure assumed in the analysis. However, a single failure of the RTS does not result in a loss of function and the safety analysis is not affected.

No transient analysis is performed for the analysis of the failure of a small line carrying primary coolant outside containment event in Subsection 15.6.2.

For the radiological consequences analysis of the steam generator tube rupture (SGTR) event in Subsection 15.6.3, the mitigative equipment is the RTS, ECCS, EFWS, EFW isolation, MSDV, SDV, and GTG. A single failure of the RTS does not result in a loss of function. A single failure of the ECCS will reduce the amount of safety injection flow. Since the injection flow of the ECCS increases the primary system pressure, it increases the primary-to-secondary leakage flow and it is therefore more conservative to assume that all trains of the ECCS are operable. A single failure of the EFWS will reduce the amount of EFW flow and therefore reduces the heat removal capability. A single failure of an EFW isolation valve does not result in a loss of function. A single failure of one MSDV will reduce the amount of steam that can be released and thus reduce

the RCS cooldown capacity, but the failure of one MSDV is already assumed as a conservative assumption. A single failure of an SDV does not result in a loss of function. A single failure of the GTG results in the loss of the same train of ECCS, CSS, RHRS, and EFWS, of which, the ECCS and EFWS are assumed in the analysis. The failure of 1 train of the GTG causing a failure of 1 train of both ECCS and EFWS is less severe than just the failure of 1 train of EFWS since it is more conservative to have all ECCS trains operable as described previously. Therefore, the failure of 1 train of EFWS is selected as the most severe single failure. In addition to the single failure of 1 train of EFWS, a stuck open MSRV is assumed as an additional failure.

For the steam generator overfill analysis of the steam generator tube rupture (SGTR) event in Subsection 15.6.3, the mitigative equipment is the RTS, ECCS, EFWS, EFW isolation, MSDV, SDV, and GTG. The analyses of the effect of single failures associated with the mitigative equipment for this event are identical to the radiological consequences SGTR event previously described, except that the additional failure of the MSRV is not assumed for the SG overfill case. Therefore, the failure of 1 train of EFWS is also selected as the most severe single failure.

For the analysis of the large break loss of coolant accident (LB LOCA) event in Subsection 15.6.5, the mitigative equipment is the ECCS, C/V isolation, C/V spray (CSS), and GTG. A single failure of the ECCS will reduce the amount of safety injection flow. A single failure of a C/V isolation valve does not result in a loss of function. CSS is assumed to operate in order to minimize C/V backpressure which will increase PCT and it is therefore more conservative to assume that all trains of CSS are operable. A single failure of the GTG results in the loss of the same train of ECCS, CSS, RHRS, and EFWS, of which, the ECCS and CSS are assumed in the analysis. The failure of 1 train of the GTG causing a failure of 1 train of both ECCS and CSS is less severe than just the failure of 1 train of ECCS since it is more conservative to have all CSS trains operable as described previously. Therefore, the failure of 1 train of ECCS is selected as the most severe single failure.

For the analysis of the small break loss of coolant accident (SB LOCA) event in Subsection 15.6.5, the mitigative equipment is the RT, ECCS, EFWS, and GTG. A single failure of the RTS does not result in a loss of function. A single failure of the ECCS will reduce the amount of safety injection flow. A single failure of the EFWS will reduce the amount of EFW flow and therefore reduces the heat removal capability. A single failure of the GTG results in the loss of the same train of ECCS, CSS, RHRS, and EFWS, of which, the ECCS and EFWS are assumed in the analysis. Since failures of 1 train of the ECCS and EFWS both impact the analysis, the combined failure of both systems will be the worst case. Therefore, the failure of 1 train of GTG is selected as the most severe single failure.

In conclusion, the limiting single failure assumption for each event determined by this analysis are the same as those summarized in DCD Table 15.0-6.

Table 15.0.0-11.1 Mitigative Systems Assumed in the Chapter 15 Safety Analysis

	Function	Reactivity Control		RCS Inventory & Core Cooling		Heat Removal by Secondary System									RCS Integrity		Containment Pressure		Emer. Power Supply
							Isolation of Secondary System					Secondary Depressurization			Primary Depressurization				
DCD Sub-section	Related System → / Event ↓	Reactor Trip	ECCS (Boron Injection)	ECCS	RHRS	EFWS	MSIV	MFIV	MSCV	MFCV	EFW IV	MSSV	MSRV	MSDV	Przr S/V	SDV	Isolation	Spray	GTG
15.1.1	Feedwater temperature reduction																		
15.1.2	Excessive feedwater flow	X																	
15.1.3	Excessive steam flow																		
15.1.4	Inadvertent secondary depressurization		X								X								
15.1.5	SLB (HSD)		X								X								X
	SLB (HFP - prior to trip)	X																	
15.2.1	Loss of load	X																	
15.2.2	Turbine trip[1]																		
15.2.3	Loss of condenser vacuum[1]						#	#	#	#		#			#				
15.2.4	Closure of MSIV[1]																		
15.2.5	Steam pressure regulator failure[2]																		
15.2.6	Loss of AC power	X				X													X
15.2.7	Loss of normal feedwater	X				X													X
15.2.8	Feedwater line break	X				X													X
15.3.1.1	Partial loss of flow	X																	
15.3.1.2	Complete loss of flow	X																	
15.3.2	Flow controller malfunction[2]																		
15.3.3	RCP locked rotor	X																	
15.3.4	RCP shaft break[4]																		

Table 15.0.0-11.1 Mitigative Systems Assumed in the Chapter 15 Safety Analysis

	Function	Reactivity Control		RCS Inventory & Core Cooling		Heat Removal by Secondary System									RCS Integrity		Containment Pressure		Emer. Power Supply
							Isolation of Secondary System					Secondary Depressurization			Primary Depressurization				
DCD Sub-section	Related System → / Event ↓	Reactor Trip	ECCS (Boron Injection)	ECCS	RHRS	EFWS	MSIV	MFIV	MSCV	MFCV	EFW IV	MSSV	MSRV	MSDV	Przr S/V	SDV	Isolation	Spray	GTG
15.4.3	Dropped RCCA	X																	
15.4.1	RCCA bank withdrawal from subcritical	X																	
15.4.2	RCCA bank withdrawal at power	X																	
15.4.3	Dropped RCCA	X																	
	Misaligned RCCA[3]																		
	Single RCCA withdrawal at power	X																	
15.4.4	Startup of an inactive loop[2]																		
15.4.5	Flow controller malfunction[2]																		
15.4.6	Boron dilution	X																	
15.4.7	Improper fuel loading[3]						#	#	#	#		#			#				
15.4.8	RCCA ejection	X																	
15.5.1	Inadvertent ECCS operation at power[2]																		
15.5.2	Increase in reactor coolant inventory (CVCS)	X																	
15.6.1	Inadvertent RCS depressurization	X																	
15.6.2	Failure of a small line carrying primary coolant outside containment[3]																		
15.6.3	SGTR (Radiological)	X		X[5]		X					X		X[6]	X		X			X
	SGTR (Overfill)	X		X[5]		X					X			X		X			X
15.6.5	LB LOCA			X													X	X[7]	X
	SB LOCA	X		X		X													X

Notes:

1: Bounded by the Subsection 15.2.1 analysis
2: Not applicable to the US-APWR
3: No transient analysis
4: Bounded by the Subsection 15.3.3 analysis
5: Not a mitigative system, but assumed to operate in order to increase primary-to-secondary flow
6: Stuck open valve is assumed as an additional failure
7: Not a mitigative system, but assumed to operate in order to minimize C/V backpressure which increases PCT

Legend:
X Mitigation system assumed operable in the safety analysis
Single failure is not assumed because the mitigative system is considered passive

Table 15.0.0-11.2 Potential Effect of Single Failure Assumption

Function	System and Equipment	Assumption of Single Failure	Discussion of Impact of Single Failure Assumption	Impact on Safety Analysis
Reactivity Control	RTS	1 train fails to operate	The system design includes four train redundancy with 2-out-of-4 logic, so a loss of one train does not result in a loss of function.	None
	ECCS (Boron Injection)	1 train fails to operate	The assumption of one inoperable train reduces the amount of the borated water injection flow.	Yes – Modify model input
RCS Inventory & Core Cooling	ECCS	1 train fails to operate	The assumption of one inoperable train reduces the amount of safety injection flow.	Yes – Modify model input
	RHRS	1 train fails to operate	The assumption of one inoperable train reduces the amount of RHR flow.	Yes – Modify model input
Heat Removal by Secondary System	EFWS	1 train fails to operate	The assumption of one inoperable train reduces the amount of EFW flow.	Yes – Modify model input
	MSIV (valve closure function)	None	The MSIV is a normally open air operated check valve that will fail closed on a loss of control air. Additionally, the control air is maintained by redundant solenoid controlled valves. A single failure does not impact the ability of the equipment to perform its required safety function, which is to close.	None
	MFIV (valve closure function)	None	The MFIV is a normally open pneumatic hydraulic gate valve that will fail closed on a loss of power to the control solenoid. Additionally, the solenoid controlled valve has redundant trains. Therefore, a single failure of this valve has no impact.	None
	MSCV (valve closure function)	None	Check valves are passive components for which single failure assumptions do not apply.	None
	MFCV (valve closure function)	None	Check valves are passive components for which single failure assumptions do not apply.	None

Table 15.0.0-11.2 Potential Effect of Single Failure Assumption

Function	System and Equipment	Assumption of Single Failure	Discussion of Impact of Single Failure Assumption	Impact on Safety Analysis
	EFW Isolation	1 train fails to operate or 1 valve fails to close	There are two separate ESFAS trains for the emergency feedwater isolation valves for each SG (there are two redundant isolation valves per SG). Therefore, a single failure does not result in a loss of isolation function.	None
	MSSV (valve opening function)	None	Single failures are not assumed for this type of spring-loaded safety valve.	None
	MSDV (valve opening function)	1 valve fails to open	An assumed failure of one MSDV would reduce the amount of steam that can be released and thus reduce the RCS cooldown capacity.	Yes – Modify model input
RCS Integrity	Pressurizer Safety Valve (valve opening function)	None	Single failures are not assumed for this type of spring-loaded safety valve.	None
	SDV (valve opening function)	1 valve fails to open	There are two redundant flow paths each with a motor-operated SDV and its associated block valve, so a loss of one of the depressurization paths does not result in a loss of function.	None
Containment Pressure	CV Isolation Valves (valve closure function)	1 valve fails to close	Two isolation valves are required in series; one inside and one outside of the containment. This design prevents the loss of isolation function under single failure conditions.	None
	CV Spray Systems (CSS)	1 train of spray fails to operate	The assumption of one inoperable train reduces the amount of spray flow.	Yes – Modify model input
Emergency Power Supply	GTG	1 train fails to operate	The assumption of the loss of one train of GTG results in the loss of the same train of ECCS, CSS, RHRS, and EFWS.	Yes – Impacts number of available trains

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

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SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0.0**
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QUESTION NO.: 15.0.0-12

In the acceptance review of the US-APWR, the staff requested MHI to provide Emergency Response Guidelines (ERG) for operator actions credited in the FSAR Chapter 15 safety analyses so that the NRC could verify that the future plant Emergency Operating Procedures (EOPs) will correspond to operator actions assumed in the safety analyses. In response to the staff request, MHI in its letter to NRC dated February 8, 2008, provided a table listing the operator actions assumed in various safety analyses. MHI also stated that additional information for operator actions assumed in the Chapter 15 safety analyses that is contained in the ERG, such as the operator action criteria in terms of parameter and values as well as the source of the information (instrument or channel). However, the staff has not yet received that additional information up to date. Please provide the following:

a) The applicant should provide planned schedule of APWR ERG development in light of its availability for developing plant specific emergency operating procedures (EOPs) by COL applicants. Identify any potential conflict with APWR new plant deployment schedule.
b) The applicant should expand FSAR Section 15.0 to address the need for supporting analyses (best estimate and licensing analyses) and the need for verification and validation (V&V) of the developed ERGs to demonstrate that the ERGs will achieve their design intentions and be consistent with the operator actions assumed in the transient and accident documented in Chapter 15 of FSAR.

ANSWER:

a)
MHI is currently developing an Emergency Response Guidelines (ERG) document for the US-APWR for the purpose of supporting plant-specific Emergency Operating Procedures (EOPs). The ERGs are composed of both event-based guidelines and functional-based guidelines, and will include the manual actions described in DCD Chapter 15.

The US-APWR ERGs are being developed by MHI in two phases. Phase 1 will develop a draft ERG that reflects the US-APWR design, and will include US industry input. The Phase 1 draft ERG will be completed by the end of 2009. During Phase 2 (January 2010 to December 2012) MHI plans to add the remainder of the detailed design-specific bases, add equipment details such

as MHI component IDs, and develop a draft EOP for use by US-APWR COL Applicants.

b)
ERGs will be developed as described in the response to part a) above. The US-APWR plant-specific ERGs will address all of the operator actions credited for Chapter 15 events as listed in DCD Subsection 15.0.0.6 and described with each applicable event in DCD Chapter 15.

DCD Subsection 13.5.2 (and in particular 13.5.2.1.3) describes the responsibilities of the COL Applicant to develop and implement EOPs. Interim steps include the Procedures Generation Package (PGP) that consists of a plant-specific technical guideline (US-APWR ERG), a plant-specific writer's guide, a description of a program for verification and validation, and a description of the program for training operators on EOPs (COL applicant specific).

When the Phase 2 US-APWR ERGs are completed, they will contain bases documents and references to various supporting analyses as described in this RAI question.

MHI believes that the necessary descriptions of the ERG and EOP development are already contained in DCD Chapter 13, and no further changes to Chapter 15 are proposed.

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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QUESTION NO.: 15.0.0-13

FSAR Section 15.0.0.1 indicates that due to similarities between the APWR and the current generation of operating reactors in the U.S, MHI has determined that no new event type are required to bound the possible initial event. In staff review of Section 4.6.2.4 of MUAP-07016, It seems that a Reactor Coolant Pump (RCP) over-speed at cold condition could result in potential plastic spring deformation and lift off a fuel assembly. Please discuss the need of analyses as an AOO for the RCP over-speed event to address the consequences of fuel performance due to increased cooldown and/or lift of fuel assembly. Also, a RCP over-speed may cause increased primary pressure. Specifically, discuss: a) what temperature and pressure define the hot and cold conditions during a RCP over-speed AOO referred in MUAP-07016, and b) What prohibits a RCP over-speed AOO at cold conditions or makes it less limiting from a fuel assembly lift-off perspective than at hot conditions?

ANSWER:

The reactor coolant pump (RCP) over-speed is not defined by the NRC in the SRP as an AOO and is therefore not evaluated in DCD Chapter 15 as an initiating event. This postulated scenario is not unique to the US-APWR, but is applicable to all PWRs that use synchronous reactor coolant pump motors. Not including this event in Chapter 15 is consistent with the licensing for all PWRs.

As noted in the RAI question, the RCP over-speed is described in MUAP-07016 to address the evaluation of the fuel assembly against its component-specific design criteria. There are many equipment-related design criteria discussed in various sections of the DCD that include limiting parameters that are derived from many sources including the safety analyses. The maximum flow experienced by a fuel assembly during an RCP over-speed is an example of such a component-specific design criteria.

An RCP over-speed condition and resultant flow increase of the magnitude described in MUAP-07016 would increase the pump head. However, the power increase caused by the increased cooling would be less than that of the uncontrolled control rod assembly withdrawal at power. Therefore, even if an RCP over-speed did occur it would be bounded by the uncontrolled control rod assembly withdrawal at power event in DCD Subsection 15.4.2 from the view point of the minimum DNBR. An RCP over-speed would also be bounded by the loss of load event in

DCD Subsection 15.2.1 from the view point of RCS pressure because the increase in RCS pressure would be offset by the decrease in RCS pressure attributed to the increase in the heat transfer in the steam generator (caused by the higher flow rate).

a)
The hot condition during an RCP over-speed is defined as:

- RV inlet temperature =[ ]°F
- Pressure =[ ]psia

Although the cold condition during an RCP over-speed is not considered as the event for determining the lift-off height of the fuel assemblies, it is assumed to be the same as the cold condition for reactor startup, which is defined as:

- RV inlet temperature =[ ]°F
- Pressure =[ ]psia

b)
As discussed in Section 4.6 of MUAP-07016, the lift-off height of the fuel assembly during an RCP over-speed event at hot conditions is limited to meet the criterion that the plastic deformation of the holddown spring of the fuel assembly is not increased.

The hydraulic loads for the fuel assembly lift-off evaluation are assumed as follows:

- Cold startup & hot full power conditions: Mechanical Design Flow (MDF) is used for calculating the hydraulic load. The hydraulic load for each condition is conservatively increased by[ ] which consists of the[ ]uncertainty associated with the hydraulic load evaluation in addition to the[20%]margin for fuel assembly design.

- Pump-over-speed condition: The hot full power coolant condition with [ ] MDF is assumed. The hydraulic load is also increased by[ ] which consists of the[ ] uncertainty associated with the hydraulic load evaluation in addition to the[ ]margin for fuel assembly design.

RCP over-speed in the cold condition can be caused by a change in the external power frequency; however, the incremental change in the flow is estimated to be only[ ]. Since the[ ] flow increment is less than the[ ] hydraulic load margin, the RCP over-speed at cold condition is bounded by the cold startup condition. Therefore, the fuel assembly lift-off does not occur in the cold condition including the RCP over-speed.

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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Docket No. 52-021

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SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0.0**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.0.0-14

FSAR Tables 8.3.1-4 show four divisions of electrical safety equipment (each division being 50%). It is indicated that the two motor driven emergency feedwater pumps (MDEFWP) are powered by Division B and C Class 1E power supplies respectively. Assuming one division of electrical power supply out for maintenance allowed by Technical Specification and a single failure on the other division, both MDEFWP could be inoperable during a design basis event. Please provide discussion on the operability and adequacy of the two turbine driven emergency feedwater pumps (TDEFWP) with respect to the condition of steam supplies and feed water flow arrangement of the system.

ANSWER:

The EFWS has two turbine-driven emergency feedwater pumps (trains A and D). Each turbine-driven emergency feedwater (T/D-EFW) pump is a horizontal multiple-stage centrifugal pump with a mechanical seal. Each T/D-EFW pump is connected to two main steam lines for receiving the supply of driving steam. Considering that one of the two steam lines may not be able to supply the driving steam because of an upstream faulted steam generator, each steam line has a capacity large enough to deliver 100% of the steam flow required to drive the T/D-EFW pump. The electric power required by the related components, including the power required for actuating the valves that start up and control the operation of the T/D-EFW pumps, is supplied by a class 1E DC bus. The valve that starts or terminates the supply of driving steam to the emergency feedwater pump A is powered by the class 1E DC bus A. The valve that starts or terminates the supply of driving steam to the emergency feedwater pump D is powered by the class 1E DC bus D. Each of the class 1E DC busses is powered from class 1E dc batteries, respectively. The batteries are also charged from class 1E GTGs, respectively.

The pump heads of the T/D-EFW pumps are designed to provide adequate flow to the steam generator at design pressure. During long term cooling, as the steam generator pressure and core decay heat decrease, the steam pressure and amount of driving steam for T/D-EFW pump operation will decrease. However, the pump turbine is designed to maintain sufficient flow to the steam generator under these conditions, such as is the case for RHR entry conditions. In conclusion, the T/D-EFW pumps have adequate ability as a mitigation system under accident conditions and during long term cooling.

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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QUESTION NO.: 15.0.0-15

Confirm that for each of the events analyzed both with and without offsite power cases are performed. For DNBR case, a three second time delay of RCP trip may be assumed (provided this time delay is approved by the staff in reviewing FSAR Chapter 8). For peak primary and secondary pressure concern, a LOOP at the time of turbine trip is considered per the criterion set forth in Section 15.0.0.7.

ANSWER:

DCD Subsection 15.0.0.7 describes the loss of offsite AC power (LOOP) assumptions for the Chapter 15 analyses. One of the assumptions discussed is the assumption of a 3 second delay between the time of the turbine-generator trip and the LOOP. For DNBR cases, this 3 second delay results in the minimum DNBR being the same with and without LOOP. Therefore, DCD Subsection 15.0.0.7 states that the LOOP cases are not presented for all events. The response to Question 15.0.0-3 of this RAI provides a sensitivity analysis that supports this DCD statement. Between the sensitivity analysis results presented in the response to Question 15.0.0-3 of this RAI and the responses to other Chapter 15 RAIs the with and without LOOP cases for those cases originally not included in the DCD have now been presented. In addition, confirmation of the applicability of the assumed 3 second delay is confirmed by a grid stability analysis included as a COL item as described in the response to Question 15.0.0-4 of this RAI. For peak pressure analyses, the time assumed for the LOOP is not a key parameter and therefore the 3 second delay is ignored in some cases. This is also discussed in the response to Question 15.0.0-3 of this RAI.

Impact on DCD

DCD Subsection 15.0.0.7 will be revised to clarify the LOOP assumptions used for the safety analysis. This DCD change is shown in the "Impact on DCD" section of the response to Question 15.0.0-2 of this RAI.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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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*1				
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 COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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QUESTION NO.: 15.0.0-17

In Section 15.0.0.2.5 it is stated that a conservative bottom-skewed axial power distribution is used to help define the control rod insertion worth. Discuss how this distribution is defined to assure that it is bounding?

ANSWER:

The scram reactivity curve shown in Figure 15.0-4 of the US-APWR DCD shows the negative reactivity versus time inserted into the core after scram initiation. Since the insertion of negative reactivity reduces the reactor power, it is conservative to assume a slow reactivity insertion in the safety analyses. The reactivity insertion is delayed when the axial power distribution is skewed to the bottom of the core. Therefore, when the scram reactivity curve used for the Chapter 15 safety analyses was determined, the axial power distribution was conservatively skewed to the bottom of the core so as to bound the axial power distribution allowed by the Technical Specifications during normal operational conditions.

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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QUESTION NO.: 15.0.0-18

RE: MUAP-07026-P (R0)
The reload methodology is meant to be applicable to "fuel design changes in dimensions and/or materials, and of thermal design changes." Are there limits to design changes at which the reload methodology would no longer apply? For example, can it be used if MOX fuel is introduced, or if an axial loading is introduced, or if a core with two different types of fuel design is used, or if the enrichment is above a certain value, or if the burnable absorbers are changed? What criteria are used to make the determination?

ANSWER:

As shown in the Mitsubishi Reload Evaluation Methodology Technical Report, MUAP-07026-P (R0), bounding values are used as input parameters for the safety analyses to eliminate the necessity to perform reanalysis for each reload core. It is not necessary to change the reload methodology as long as the safety analysis methodology and/or analysis codes are not changed. However, when significant fuel specifications are changed, such as the introduction of MOX fuel or extended burnup fuel, the current bounding values used for the safety analyses will be evaluated in accordance with the reload methodology to confirm they are sufficiently conservative to use the same values for the safety analysis. If the evaluation results indicate the current values are not bounding, they will be revised.

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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QUESTION NO.: 15.0.0-19

RE: MUAP-07026-P (R0)
Is it stated somewhere that the versions of the codes to be used will have been approved by the NRC? If not, where will this statement appear?

ANSWER:

Subsection 3.2.9 of the Mitsubishi Reload Evaluation Methodology Technical Report, MUAP-07026-P (R0), describes the codes that are used: MARVEL-M, TWINKLE-M, VIPRE-01M, RADTRAD, ANC, WCOBRA/TRAC, and HOTSPOT. Some of these codes have been approved by the NRC and the others are currently being reviewed for approval. The approval status of some of the aforementioned codes is described in Subsection 3.2.9 of the technical report. After all of these codes have been reviewed and approved by the NRC, Subsection 3.2.9 of the report will be revised to state that the codes to be used are approved by the NRC.

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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QUESTION NO.: 15.0.0-20

RE: MUAP-07026-P (R0)
In Section 3.2.9.3 it is stated that the Doppler and moderator effects are taken into account through a change in absorption cross-section. For the moderator feedback, this is insufficient to model the physics and indeed in the TWINKLE manual (WCAP 7979-P-A) the (additional) correction for the removal cross-section is explained. Please correct the wording in Section 3.2.9.3.

ANSWER:

Subsection 3.2.9.3 of the Mitsubishi Reload Evaluation Methodology Technical Report, MUAP-07026-P (R0), describes the TWINKLE-M code. In that subsection, it is stated that TWINKLE-M takes into account the Doppler and moderator feedback effects by absorption cross-section compensation at each mesh point. This statement is not entirely complete, as indicated in this RAI question. As described in the Non-LOCA Methodology Topical Report, MUAP-07010, and its associated RAI responses, TWINKLE-M is based on TWINKLE. The method in which Doppler and moderator feedback effects are accounted for in TWINKLE-M was not modified from the original TWINKLE, which was described in WCAP-7979-P-A. As a result, the description in Subsection 3.2.9 of the Mitsubishi Reload Evaluation Methodology Technical Report will be revised to more accurately state that, "The Doppler feedback effect is taken into account by absorption cross-section compensation at each mesh point, while the moderator feedback effect is taken into account by absorption and removal cross-section compensation at each mesh point."

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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QUESTION NO.: 15.0.0-21

RE: MUAP-07026-P (R0)
In Section 4.2.2.1 it is stated that a conservative factor is applied to the total rod worth. However, in Section 15.0.0.2.5 of Chapter 15, there is no mention of this factor. Provide discussion to clarify. What is the value of this conservative factor and its basis?

ANSWER:

Subsection 4.2.2.1 of the Mitsubishi Reload Evaluation Methodology Technical Report, MUAP-07026-P (R0), discusses how to evaluate the total control rod worth. Because it is conservative to assume a smaller value of the control rod worth when plant shutdown capability is evaluated, the calculated total control rod worth is conservatively multiplied by [ ] and then compared to the value used for the Chapter 15 safety analysis. [

].

On the other hand, Subsection 15.0.0.2.5 of Chapter 15 discusses the RCCA negative reactivity insertion versus time used for Chapter 15 safety analysis. Since the factor is applied to the calculation results of total rod worth for conservatism as described above, there is no mention of this factor in this section.

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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QUESTION NO.: 15.0.0-22

RE: MUAP-07026-P (R0)
Explain the rationale for the specific equation for the hot channel factor cited in Section 5.3.1.1?

ANSWER:

The equation for the hot channel factor shows that the design limit of $F^N_{\Delta H}$ increases with the reduction of reactor power. The design limit of $F^N_{\Delta H}$ is allowed to increase by 30% at HZP compared to the limit at HFP since thermal margin for DNB increases with the reduction of the reactor power. The equation for the design limit of $F^N_{\Delta H}$ is used for the Chapter 15 safety analyses and establishing the Over Temperature ΔT setpoints.

The control rod insertion limit is determined as a function of the reactor power, which allows deeper control rod insertion into the core at lower reactor power. The insertion of control rods in general distorts the radial power distribution, resulting in the increase of $F^N_{\Delta H}$. The equation is determined considering this effect for $F^N_{\Delta H}$, as described in Subsection 4.4.4.3.1 of the US-APWR DCD. In actual operation, the equation for the design limit of $F^N_{\Delta H}$ is ensured by the Technical Specifications.

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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QUESTION NO.: 15.0.0-23

RE: MUAP-07026-P (R0)
In the discussion of axial power distributions (Section 5.3.1.2) and fuel temperature (Section 5.3.2), it is pointed out that if a parameter is bounded by the reference case, then normal operation and AOO analyses previously done are acceptable. Is this meant to also apply to PAs?

ANSWER:

The axial power distributions and fuel temperatures discussed in Subsections 5.3.1.2 and 5.3.2, respectively, of the Mitsubishi Reload Evaluation Technical Report, MUAP-07026-P (R0), are also meant to apply to PAs. However, event-specific axial power distributions are used for certain PAs: steam system piping failure (SLB), rod ejection at hot zero power (R/E HZP), and loss of coolant accident (LOCA). The Mitsubishi Reload Evaluation Topical Report will be revised to discuss PAs and the event-specific axial power distributions used for certain PAs.

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.