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August 19, 2013

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-13210

Subject: MHI's Response to US-APWR DCD RAI No. 1041-7066 (SRP Section 07.08)

Reference: 1) "Request for Additional Information No. 1041-7066, SRP Section: 07.08 - Diverse Instrumentation and Control Systems, Application Section: MUAP-07014 (Rev 5) - D3 Coping Analysis," dated July 3, 2013.

With this letter, Mitsubishi Heavy Industries, Ltd. (MHI) transmits to the U.S. Nuclear Regulatory Commission (NRC) a document entitled "Response to Request for Additional Information No. 1041-7066."

Enclosed are the responses to five RAI questions contained within Reference 1.

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 with the information identified as proprietary redacted and replaced by the designation "[ ]."

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, 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,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, Ltd.

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Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 1041-7066 (Proprietary version)
3. Response to Request for Additional Information No. 1041-7066 (Non-proprietary version)

CC: J. A. Ciocco
J. Tapia

Contact Information
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ENCLOSURE 1

Docket No. 52-021
MHI Ref: UAP-HF-13210

MITSUBISHI HEAVY INDUSTRIES, LTD.
AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Executive Vice President of Mitsubishi Nuclear Energy Systems, Inc., and have been delegated the function of reviewing Mitsubishi Heavy Industries, Ltd.'s (MHI) US-APWR documentation to determine whether it contains information that 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.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 1041-7066" dated August 2013, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]." The first page of the 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 information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design approach for the Defense-in-Depth (D3) of the Instrumentation and Control (I&C) system and the unique post-accident response of the US-APWR crediting design features for D3, developed by MHI and 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.

5. 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.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their

design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information identified as proprietary would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with development of the D3 approach for the I&C system and the post-accident plant response of the US-APWR to the design features for D3. Providing public access to such information permits competitors to duplicate or mimic the D3 design approach without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced plant safety, and reduced operation and maintenance costs associated with the D3 approach.

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 19th day of August, 2013.

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.

Enclosure 3

UAP-HF-13210
Docket No. 52-021

Response to Request for Additional Information No. 1041-7066

August 2013
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/19/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1041-7066**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**MUAP-07014 (REV 5) - D3 COPING ANALYSIS**
DATE OF RAI ISSUE:	**07/03/2013**

QUESTION NO. : 07-1

BTP 7-19, Rev 5, acceptance criteria states that for AOO/PA events occurring in conjunction with each single postulated CCF, the plant response calculated using best-estimate analyses should not result in radiation release exceeding 10 percent of the 10 CFR 100 guideline value or violation of the integrity of the primary coolant pressure boundary. In order to meet this criteria, the staff requests the applicant to provide additional information for the five questions in this RAI.

<u>Reactor Coolant Pump Rotor Seizure (MUAP-07014-P, Revision 5, Section 5.3.3)</u>

In D3 coping analysis Section 5.3.3 of MUAP-07014-P technical report, the applicant provided a best-estimate analysis for the reactor coolant pump rotor seizure event. In its D3 coping analysis, the applicant assumed the instantaneous seizure of only one RCP to be the initiating event. The evaluation of this event concurrent with a CCF assumes the same case as DCD Chapter 15 Section 15.3.3 event analysis. According to the report, the assumptions, input parameters, and initial conditions assumed in the D3 coping analysis are the same as the Chapter 15 safety analysis, except the few that are listed in D3 analysis Section 5.3.3(1)(a). The results of the analysis are discussed and also presented in Figures 5.3.3-1 through 5.3.3-5 for plots of key system parameters versus time. The staff reviewed the results of the analysis and the associated plots. Accordingly, the core coolability and dose associated with this event are within the NRC guidelines.

However, regarding pressure integrity for the D3 analysis, one of the exceptions from Chapter 15 assumptions is that any reactor trip actuation by the RTS is ignored. In addition, no reactor trip actuation by DAS is assumed. Whereas, in the D3 results section under "Pressure Boundary Integrity,", section 5.3.3(1)(b), it states that the RCS pressure increase is mitigated by the pressurizer safety valve and the DAS high pressurizer pressure reactor trip actuation and EFWS actuation. The staff views this as a discrepancy between the assumptions and the event results presented in the D3 technical report. Therefore, the staff requests the applicant explain this discrepancy, and therefore this portion of the analysis remains as an open item until resolved.

ANSWER:

As described in MUAP-07014-P (R5) Section 5.0 (1) "Pressure Boundary Integrity", DAS high pressurizer pressure reactor trip and pressurizer safety valve are available to maintain reactor coolant pressure boundary if overpressurization occurs during an event concurrent with digital CCF. A similar sentence in MUAP-07014-P (R5) Section 5.3.3(1)(b) seems to indicate that DAS reactor trip and pressurizer safety valves are both credited for the RCP rotor seizure event.

However, as the NRC pointed out, the analysis in MUAP-07014-P (R5) Section 5.3.3(1)(a) does not credit the DAS reactor trip. Even without the DAS reactor trip or safety valve opening, the peak RCP outlet pressure, which is the highest pressure in the entire RCS, is below 3200 psig as shown in Figure 5.3.3-3 of MUAP-07014-P (R5). Therefore, the following sentence in Section 5.3.3(1)(b) will be deleted to remove the inconsistency in MUAP-07014-P (R5).

"This demonstrates that the RCS pressure increase is mitigated by the pressurizer safety valve and the DAS high pressurizer pressure reactor trip actuation and EFWS actuation."

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
MUAP-07014-P (R5) Section 5.0(1) will be clarified as Attachment-1.

MUAP-07014-P (R5) Section 5.3.3(1)(b) will be revised as Attachment-2.

This completes MHI's response to the NRC's question.

5.0 D3 COPING ANALYSIS RESULTS

The results of each event are evaluated according to the following criteria as described in Table 4.3-3:

- Pressure boundary integrity
  - Reactor Coolant Pressure Boundary (RCPB)
  - Containment Vessel (C/V)
- Core coolability
- Dose

Additional background on the analysis approach and event screening common to all events for each of the criteria is provided below.

(1) Pressure Boundary Integrity

For RCPB integrity, the capacity of the pressurizer safety valve is designed so that this valve is able to release the maximum surge flow to the pressurizer assuming a turbine trip without a reactor trip, as long as the steam generator secondary side has sufficient water inventory. The DAS includes reactor trips and EFW actuation from the low steam generator water level signal. The reactor trips and EFW actuate from this signal before steam generator dry-out for events assuming a concurrent CCF. Therefore, the RCS pressure increase ~~is~~ can be mitigated by the DAS and the pressurizer safety valve which is not affected by CCF. Therefore, all DCD Chapter 15 safety analysis events assuming CCF are "expertly judged" events for the RCPB criterion. Section 5.2.1 provides a representative D3 coping analysis for the loss of load event to assure that the RCS pressure increase can be successfully mitigated by the pressurizer safety valve and the DAS. Note that for some events, the RCS pressure increase is such that the DAS and pressurizer safety valve are not necessary.

The C/V integrity for initiating events which breach the RCPB is described in each applicable event section.

(2) Core Coolability

For most events, core coolability is demonstrated by evaluating departure from nucleate boiling (DNB). Each event subsection describes the evaluation of core coolability.

(3) Dose

Dose evaluations are not necessary if core coolability is maintained except for the events which lead to release of primary coolant from RCS outside the C/V. For most events concurrent with CCF, core coolability is maintained and an analysis is not performed.

5.1 Increase in Heat Removal by the Secondary System

5.1.1 Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions

A decrease in feedwater temperature causes a reduction in steam generator secondary temperature, resulting in an increase in primary-to-secondary heat transfer. In the

A limiting case is defined for the locked rotor accident that also bounds the plant response to the RCP shaft break event discussed in DCD Section 15.3.4. The bounding case in DCD Section 15.3.3 is defined by assuming the RCP rotor is stopped prior to flow reversal, and that the pump resistance is changed to zero after the flow reverses in the affected loop. The evaluation of this event concurrent with a CCF assumes the same case as DCD Section 15.3.3.

(1) Pressure Boundary Integrity

(a) Analysis Assumptions, Input Parameters and Initial Conditions

Unless specifically listed below, the assumptions, input parameters, and initial conditions assumed in the D3 coping analysis are the same as the DCD Chapter 15 safety analysis.

- Any reactor trip actuation by the RTS is ignored. In addition, no reactor trip actuation by the DAS is assumed.
- The moderator temperature coefficient is assumed to be -6 pcm/°F (this value is a realistic negative value consistent with the moderator temperature coefficient of 0 pcm/°F at the BOC HZP condition).
- The Doppler power coefficient and the Doppler temperature coefficient are assumed to be the DCD Ch. 15 maximum coefficient times 0.8. This assumed power coefficient and temperature coefficient are close to the Doppler power coefficient and the Doppler temperature coefficient in the US-APWR first core and equilibrium core.

(b) Results

Figures 5.3.3-1 through 5.3.3-4 are plots of key system parameters versus time. The peak RCP outlet pressure, which is the highest pressure in the entire RCS, is below 3200 pisg as shown in Figure 5.3.3-3. ~~This demonstrates that the RCS pressure increase is mitigated by the pressurizer safety valve and the DAS high pressurizer pressure reactor trip actuation and EFWS actuation.~~ Therefore, the integrity of the RCPB is maintained for this event concurrent with a CCF.

(2) Core Coolability

(a) Analysis Assumptions, Input Parameters and Initial Conditions

Unless specifically listed below, the assumptions, input parameters, and initial conditions assumed in the D3 coping analysis are the same as the DCD Chapter 15 safety analysis.

- Any reactor trip actuation by the RTS is ignored. In addition, no reactor trip actuation by the DAS is assumed.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/19/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1041-7066**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**MUAP-07014 (REV 5) - D3 COPING ANALYSIS**
DATE OF RAI ISSUE:	**07/03/2013**

QUESTION NO. : 07-2

Uncontrolled control rod assembly (RCCA) withdrawal from a subcritical or low power (MUAP-
07014-P, Section 5.4.1)

In response to NRC staff's RAI 753-5742, Question 07.08-20 to analyze this event, the applicant performed a D3 coping analysis for this RCCA bank withdrawal event from subcritical condition concurrent with a CCF, and reflected the analysis results in Revision 5 of the MUAP-07014-P report. The analysis was performed in accordance with the acceptance criteria in Table 4.3-3 in the technical report for pressure boundary integrity, core coolability, and dose evaluations. Further, the assumptions, input parameters, and initial conditions for the D3 analysis for pressure integrity are the same as those used in DCD Chapter 15 safety analysis, except for those listed under item 5.4.1(2) "core coolability." For core coolability, the applicant identified specific assumptions and initial conditions in addition to those used in the Chapter 15 analysis, the details of which are described in the MUAP-07014-P (R5) report.

The applicant presented the results of the analysis for core coolability in Figure 5.4.1-1, "RCP Outlet Pressure versus Time," and Figure 5.4.1-2, "Reactor Power versus Time." According to the applicant, core coolability is maintained. Further, the applicant stated that the reactor trip (RT) occurs on high pressurizer pressure RT set-point by the DAS.

However, it appears that only BOC cases were run. Therefore, the staff requests further details on whether BOC results bounds EOC, considering that Beff and Hgap both change with burnup. Also, address whether the assumed positive reactivity insertion rate (4th bullet on page 5-36) bounds EOC. Results on page 5-36 of MUAP-07014 Rev. 5 state that both BOC and EOC are acceptable. The staff needs further clarifications in this regard and therefore this remains as an Open Item until resolved.

ANSWER:

In RAI 753-5742, Question 07.08-20, the analysis assumes BOC core conditions. For the power distribution in the DNBR analysis, the feedback effect due to the power increase is conservatively not assumed. The value of $F^N_{\Delta H}$ is assumed to cover the actual core and is a conservative value. This analysis bounds any realistic condition since the feedback due to the power increase will occur under realistic conditions.

In this response, the analysis assumes EOC core conditions under realistic conditions to show the conservatism of the previous analysis. For the power distribution in the DNBR analysis, the feedback effect due to the power increase is assumed and the value of $F^N_{\Delta H}$ corresponding to the power distribution is assumed.

This response also presents the analysis result for the BOC case assuming the feedback effect due to the power increase for comparison. The purpose is to show that the conservative BOC case results provided in MUAP-07014-P (R5) are bounding for both BOC and EOC conditions.

The DNBR analyses demonstrate that minimum DNBR for both EOC and BOC assuming feedback effect are well above the 95/95 DNBR limit. Additionally these are bounded by the minimum DNBR in RAI 753-5742, Question 07.08-20, which was previously incorporated into MUAP-07014-P (R5) as Figure 5.4.1-3.

(1) Pressure Boundary Integrity

(a) Analysis Assumptions, Input Parameters, and Initial Conditions

Other than the exceptions specifically listed in the description of (2) Core Coolability below, the assumptions, input parameters and initial conditions assumed for the D3 coping analysis are the same as for the corresponding DCD Chapter 15 safety analysis.

(b) Results

The time sequence of events is provided in Table 07-2.1-2. The peak RCP outlet pressure, which is the highest pressure in the entire RCS, is well below 3200 psig as shown in Figure 07-2.1-1. Thus, the DAS reactor trip and the pressurizer safety valve maintain the integrity of the RCPB for this event concurrent with a CCF in the digital I&C system.

(2) Core Coolability

(a) Analysis Assumptions, Input Parameters, and Initial Conditions

The assumptions, input parameters, and initial conditions assumed for the D3 coping analysis are the same as for the DCD Chapter 15 safety analysis, with the following exceptions (note that these assumptions cover the 24 month equilibrium core conditions and unless otherwise noted are for the EOC case):

- High pressurizer pressure reactor trip actuation by the DAS is assumed. In addition to the time delay listed in Table 4.4-1 of MUAP-07014-P (R5), the time delay from when MG-set is de-energized until rod motion is assumed to be 5 seconds.

- The moderator temperature coefficient (MTC) is assumed to be [ ] as the maximum (minimum negative) value at EOC.

- The Doppler feedback effect is evaluated using the design value with no additional margin or multiplier. (The DCD analysis assumed a Doppler feedback multiplier of -20%.)

- The initial values of reactor coolant average temperature and RCS pressure are assumed to be the nominal values corresponding to hot standby conditions without uncertainty.

- The positive reactivity insertion rate is assumed to be [ ] This value covers the maximum reactivity insertion rate during the withdrawal of control bank D with the maximum speed (45 inches per minute). The simultaneous withdrawal of two RCCA banks is not assumed since only control bank D is inserted when the core is near criticality during the normal startup process.

- In this D3 coping analysis the smaller delayed neutron fraction (β_{eff}) gives more conservative analysis results because the timing of the reactor trip by the high pressurizer pressure signal is later than that of the DCD case. So β_{eff} is assumed to be [ ] as a minimum value at EOC.

- In the evaluation of the neutron flux transient of the DCD case using the TWINKLE-M code, the heat transfer rate from the fuel pellet surface to the clad inner surface (H_{gap}) is assumed to be the maximum value. This assumption results in a smaller Doppler negative reactivity effect and a larger moderator positive reactivity effect. In this D3 coping analysis, H_{gap} is calculated by considering the fuel pellet expansion/contraction caused by the fuel pellet temperature increasing/decreasing.

- When the power distribution for DNBR analysis in the VIPRE-01M code is calculated, the feedback effect corresponding to the power increase by rod withdrawal is assumed. For the EOC case, the value of $F_{\Delta H}^{N}$ corresponding to the power distribution is [ ]

- A sensitivity study for the BOC case is also performed to compare with the EOC case. Both analytical conditions are listed in Table 07-2.1-1. Other assumptions for the BOC case are the same as the EOC case.

Table 07-2.1-1
Analysis Condition for RCCA Bank Withdrawal from Subcritical (BOC/EOC Case)

Parameters	BOC	EOC
MTC	[	[
Positive reactivity insertion rate		
β_{eff}		
$F_{\Delta H}^{N}$	]	]

(b) Results
The time sequence of events is provided in Table 07-2.1-2. Figures 07-2.1-2 and 07-2.1-3 are plots of key system parameters versus time. Reactor trip occurs on the high pressurizer pressure reactor trip setpoint by the DAS. The minimum DNBR is above the 95/95 DNBR limit. Therefore, the peak cladding temperature does not exceed 2200°F and the core coolability is maintained for this event concurrent with a CCF in the digital I&C system. Figure 07-2.1-4 shows the DNBR comparison for the BOC and EOC case assuming feedback effect as additional information. All minimum DNBRs are above the 95/95 DNBR limit. This figure also shows that the case for RAI 753-5742, Question 07.08-20, which is now included in MUAP-07014-P (R5) is the limiting case.

(3) Dose
Core coolability is maintained for this event concurrent with a CCF. Therefore, the dose associated with this event does not exceed 10 percent of the 10 CFR 100 dose guidelines for AOOs. No explicit analysis is performed here.

Table 07-2.1-2
Time Sequence of Events for RCCA Bank Withdrawal from Subcritical (EOC Case)

Event	Time (sec)
Initiation of rod withdrawal	
Initial peak of neutron flux (limited by Doppler feedback effect) occurs	
High pressurizer pressure diverse reactor trip setpoint (2440 psia) reached	
Pressurizer safety valves open	
Peak RCP outlet pressure occurs	
Rod motion begins	
Peak reactor power occurs	
Minimum DNBR occurs	

Figure 07-2.1-1 RCP Outlet Pressure versus Time
RCCA Bank Withdrawal from Subcritical for EOC Case
(Pressure Analysis)

Figure 07-2.1-2 Reactor Power versus Time
RCCA Bank Withdrawal from Subcritical for EOC Case
(DNBR Analysis)

Figure 07-2.1-3 DNBR versus Time
RCCA Bank Withdrawal from Subcritical for EOC Case
(DNBR Analysis)

Figure 07-2.1-4 DNBR versus Time
RCCA Bank Withdrawal from Subcritical for BOC and EOC Case
(DNBR Analysis)

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/19/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 1041-7066**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**MUAP-07014 (REV 5) - D3 COPING ANALYSIS**
DATE OF RAI ISSUE:	**07/03/2013**

QUESTION NO. : 07-3

Uncontrolled control rod assembly (RCCA) withdrawal at power (MUAP-07014-P, Section 5.4.2)

Section 5.4.2 of MUAP-07014 describes the RCCA bank withdrawal event at power. Accordingly, this event is caused by a failure of a control system or rod control system, which results in an increase in core heat flux, which, in turn tends to result in reactor coolant temperature and steam generator pressure increase. Without a manual or automatic reactor trip, this could result in a DNB. According to DCD Section 15.4.2, this event may occur from various initial power levels. However, this event was analyzed assuming full power initial conditions at both beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions. Further, for pressure boundary integrity, according to the applicant, this event results in an increase in core heat flux, and the heat extracted from steam generator (SG) lags behind the core power until the SG pressure reaches the main steam safety valve setpoint, and the reactor coolant temperature tends to increase. Also, the RCS pressure increase is mitigated by the pressurizer safety valve and the DAS, and therefore the integrity of the RCPB is maintained for this event concurrent with a CCF. However, based on staff's review, it does not seem that there was a trip and hence a DAS actuation. Also, provide the setpoint(s) for the pressurizer safety valve actuation, and explain if the pressure depicted in Figure 5.4.2-2, "RCS Pressure versus Time", reaches a setpoint. The staff requests the applicant provide additional information and clarification in this regard.

ANSWER:

This RAI asked about an apparent inconsistency in which the description in Section 5.4.2 of MUAP-07014-P (R5) describes a DAS reactor trip but the analysis does not appear to credit one. MHI would like to clarify that there is no inconsistency because the description and the analyses are for different cases. No analysis is performed for the pressure boundary integrity case in Section 5.4.2(1). Instead, the DAS reactor trip and pressurizer safety valves are available to limit the RCS pressure increase as also described in general in Section

5.0(1). On the other hand, an explicit analysis is performed for the core coolability case in Section 5.4.2(2). The full power analysis does not assume a DAS reactor trip as described in the first bullet of 5.4.2(2)(a).

This RAI also asked the setpoints for the pressurizer safety valves and whether the safety valves actuated. The pressurizer safety valves are assumed to operate at 2525 psia. As shown in MUAP-07014-P (R5) Figure 5.4.2-2, the maximum RCS pressure is such that the pressurizer safety valves actuate for the BOC case, but do not actuate for the EOC case.

Finally, this RAI also asked for the reason why the event was only analyzed for full power conditions. The analysis in Section 5.4.2(2) is for full power cases at both BOC and EOC.

A sensitivity study at partial power conditions (10% and 75%) is provided below.

Core Coolability

(a) Analysis Assumptions, Input Parameters and Initial Conditions

Unless specifically listed below, the assumptions, input parameters and initial conditions assumed in the D3 coping analysis are the same as the DCD Chapter 15 safety analysis.

- Any reactor trip actuation by the RTS is ignored
- High pressurizer pressure reactor trip actuation by the DAS is assumed.
- The reactivity inserted into the core is assumed in Table 07-3.1 for the BOC case and the end-of-cycle (EOC) case consistent with the available reactivity of the RCCAs being withdrawn from the insertion limit to the all rods fully withdrawn position.
- The withdrawal of the RCCA is assumed to be at possible maximum speed. The time to withdraw from the insertion limit to the all rods fully withdrawn position is listed in Table 07-3.1. The withdrawal time is calculated as follows: Time = Control rod step / maximum rod speed.
- The moderator temperature coefficient (MTC) is assumed in Table 07-3.1 for the BOC case and the EOC case as the maximum (minimum negative) value at each case.
- The Doppler power coefficient and the Doppler temperature coefficient are assumed to be 1.2 times the minimum coefficient used in DCD Chapter 15 for both the BOC and the EOC case. This assumed power coefficient and the Doppler temperature coefficient are close to the Doppler power coefficient in the US-APWR first core and equilibrium core.

The power distribution is assumed to be the limiting design power distribution used in the DCD Chapter 15 safety analysis. The axial power distribution for the BOC case may be mitigated by assuming the power shape consistent with the core burn-up, but is not adopted in this analysis.

Table 07-3.1 Analysis Conditions

Power		Reactivity Insertion (pcm)	Time (Seconds)	MTC (pcm/ °F)
10%	BOC			
	EOC			
75%	BOC			
	EOC			

(b) Results

Figures 07-3.1-1 through 07-3.1-4 are plots of key system parameters versus time for the BOC/EOC, 10% power case. Figures 07-3.2-1 through 07-3.2-4 are plots of key system parameters versus time for the BOC/EOC, 75% power case. The reactivity insertion results in increase in core heat flux, RCS temperature, and decrease in DNBR. However, the reactor is automatically tripped by the high pressurizer pressure DAS reactor trip. The DNBR figures for the partial power analysis (Figures 07-3.1-4 and 07-3.2-4) show that the minimum DNBR in both 10% and 75% cases are above the 95/95 DNBR limit and are bounded by the results in MUAP-07014-P (R5) Figure 5.4.2-4. Therefore, core coolability is maintained for this event concurrent with a CCF in the digital I&C system and only the full power case is shown in MUAP-07014-P (R5).

Figure 07-3.1-1 **Reactor Power versus Time**
Uncontrolled Control Rod Assembly Withdrawal at Power (10%)

Figure 07-3.1-2 **RCS Pressure versus Time**
Uncontrolled Control Rod Assembly Withdrawal at Power (10%)

Figure 07-3.1-3 RCS Average Temperature versus Time
Uncontrolled Control Rod Assembly Withdrawal at Power (10%)

Figure 07-3.1-4 DNBR versus Time
Uncontrolled Control Rod Assembly Withdrawal at Power (10%)

Figure 07-3.2-1 Reactor Power versus Time
Uncontrolled Control Rod Assembly Withdrawal at Power (75%)

Figure 07-3.2-2 RCS Pressure versus Time
Uncontrolled Control Rod Assembly Withdrawal at Power (75%)

Figure 07-3.2-3 RCS Average Temperature versus Time
Uncontrolled Control Rod Assembly Withdrawal at Power (75%)

Figure 07-3.2-4 DNBR versus Time
Uncontrolled Control Rod Assembly Withdrawal at Power (75%)

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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RAI NO.:	**NO. 1041-7066**
SRP SECTION:	**07.08 - DIVERSE INSTRUMENTATION AND CONTROL SYSTEMS**
APPLICATION SECTION:	**MUAP-07014 (REV 5) - D3 COPING ANALYSIS**
DATE OF RAI ISSUE:	**07/03/2013**

QUESTION NO. : 07-4

Control Rod Misoperation (System Malfunction or Operator Error); MUAP-07014-P, Section 5.4.3

Section 5.4.3 of MUAP-07014-P states that for a single rod withdrawal, the RCS pressure increase is mitigated by the pressurizer safety valve and the DAS; and also the reactivity inserted to the core is not more severe than that for the event in Section 5.4.2, "uncontrolled control rod assembly withdrawal at full power." However, explain how this bounds the control bank withdrawal at full power when the single rod withdrawal has higher peaking factor. Provide additional information and justification to support the response. Therefore, this remains an Open Item until resolved.

ANSWER:

MHI's response to US-APWR DCD RAI No. 904-6324 Revision 3 (SRP 15.4.3) Question 15.04.03-13 (UAP-HF-12101, dated April 20, 2012) provides the single RCCA withdrawal analyses at lower, partial power, and hot full power using a detailed analysis method. The TWINKLE-M (3D) and VIPRE-01M codes are used in this analysis as described in that response. In the detailed analysis, all reactor trip signals are ignored. In all cases, the resulting minimum DNBR is above the safety analysis limit and therefore no DNB fuel failure occurs. The results of the analysis in that RAI response are applicable to justify the explanation in MUAP-07014-P (R5) since no reactor trip is assumed.

For the pressure boundary case of the single rod withdrawal in Section 5.4.3(1), the total reactivity inserted to the core is less than for the uncontrolled control rod assembly withdrawal at power in Section 5.4.2. Therefore, the overall plant response for the RCS pressure transient is bounded by the Section 5.4.2 event. Note that analyses are not performed for the pressure boundary case for either Section 5.4.2 or 5.4.3; the DAS reactor trip and pressurizer safety valves are available to limit the RCS pressure increase.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
There is no impact on the technical / topical reports.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

08/19/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

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QUESTION NO. : 07-5

In MUAP-07014, Rev. 5, Section 5.2.8, "Feedwater System Pipe Break Inside and Outside Containment", the applicant stated that in the event of feedwater system pipe break, the RCS pressure increase is mitigated by the pressurizer safety valve and the DAS low steam generator water level trip actuation and initiation of the Emergency Feedwater System. Also, in the core coolability section of the report, it states that DNB is mitigated by the effect of the RCS cooldown because of the discharge of two-phase flow from the feedwater line after the perforated nozzle is uncovered by the secondary water in this event, and therefore the core coolability is maintained for this event concurrent with a CCF. Since no analyses were provided for staff review, the staff requests the applicant to provide following additional information:

· According to the DCD Section 15.2.8, "Feedwater System Pipe Break Inside and Outside Containment" analysis, the liquid water level nears the top of the pressurizer (DCD Figure 15.2.8-16). Since the DAS SG water level trip is lower and the delay time greater, explain if liquid passes through the pressurizer safety valves.

· Also, provide additional information, with justification, on how DNB is not a significant adverse consequence and its mitigation.

ANSWER:

In MUAP-07014-P (R5) Section 5.2.8, MHI did not perform an explicit analysis to determine whether water relief through the pressurizer safety valves occurs. However, due to the differences between the RTS and the DAS (i.e., setpoints and delay times), the pressurizer may fill and relieve water through the safety valves for a feedwater system pipe break concurrent with a CCF in the digital I&C system. There is no significant impact from this potential water relief for the following reasons:

- The water relief will occur at a time well after DAS reactor trip such that the core power will already be at decay heat levels. DCD Table 15.2.8-1 shows that peak pressurizer water volume occurs at more than 1000 seconds after reactor trip. This confirms that the peak pressurizer water volume will occur at a time well after DAS reactor trip even considering that the DAS reactor trip is actuated the with time delay listed in Table 4.4-1 in MUAP-07014-P (R5).

- The amount of relieved water will be bounded by the Loss of Coolant Accident analysis provided in Section 5.6.5.
- The DAS can automatically initiate SI pumps or operators can manually start SI pumps to provide RCS makeup if necessary.

In DCD Section 15.2.8, Feedwater System Pipe Break Inside and Outside Containment, the break is assumed to occur concurrent with the low steam generator water level reactor trip signal resulting from the loss of feedwater flow assumed as a precondition. This conservative precondition minimizes the total steam generator inventory available to remove heat from the RCS and makes the RTS response independent of the steam generator pressure and level dynamics of the feedwater line break prior to the reactor trip.

DCD Section 15.2.8 also describes that the minimum DNBR for the pre-trip portion of this event is not calculated because it is bounded by the minimum DNBR for the loss of normal feedwater event analyzed in Section 15.2.7.

MUAP-07014-P (R5), Section 5.2.7 demonstrates that the minimum DNBR is above the 95/95 DNBR limit. Therefore, in the Feedwater System Pipe Break Inside and Outside Containment concurrent with a digital I&C CCF (Section 5.2.8), the minimum DNBR is bounded by the minimum DNBR for the loss of normal feedwater event analyzed in Section 5.2.7.

Impact on DCD
There is no impact on the DCD.

Impact on R-COLA
There is no impact on the R-COLA.

Impact on PRA
There is no impact on the PRA.

Impact on Technical / Topical Reports
MUAP-07014-P (R5) Section 5.2.8(2) will be revised as Attachment-3.

This completes MHI's response to the NRC's question.

5.2.8 Feedwater System Pipe Break Inside and Outside Containment

The feedwater system pipe break is a non-uniform transient that involves modeling the flow from one of the secondary loops. Unlike the secondary piping rupture resulting in RCS cool down analyzed in DCD Section 15.1.5, the feedwater system pipe break analyzed in DCD Section 15.2.8 causes a loss of inventory from the saturated liquid mass in the steam generator resulting in RCS heat-up and pressurization. Unless the heat-up of the RCS is mitigated, there will be a possibility of water relief through the pressurizer safety valve.

(1) Pressure Boundary Integrity

The RCS pressure increase is mitigated by the pressurizer safety valve and the DAS low steam generator water level reactor trip actuation and initiation of the Emergency Feedwater System. Therefore, the integrity of the RCPB is maintained for this event concurrent with a CCF.

(2) Core Coolability

This event in the DCD is bounded by the minimum DNBR for the DCD Section 15.2.7 event in that DNB does not occur ~~by~~ before the low steam generator water level reactor trip. Although the diverse low steam generator water level reactor trip analytical limit is lower and the delay time is greater than that of the RTS, the analysis in Section 5.2.7 shows that the minimum DNBR remains above the DNBR limit. This indicates that the conclusion in DCD Chapter 15 also applies during a CCF: the minimum DNBR for this event concurrent with a CCF is bounded by the DNBR for the event in Section 5.2.7. ~~DNB is not a significant adverse consequence considering the axial power distribution for the BOC. On the other hand, DNB is mitigated by the effect of the RCS cool down because of the discharge of two-phase flow from the feedwater line after the perforated nozzle is uncovered by the secondary water in this event.~~ Therefore, the core coolability is maintained for this event concurrent with a CCF. This event is categorized as an “expertly judged” event for core coolability.

(3) Dose

The core coolability is maintained for this event concurrent with a CCF. Therefore, the dose associated with this event does not exceed 10% of the 10 CFR 100 dose guidelines for AOOs and the 10 CFR 100 dose guidelines for PAs.

5.3 Decrease in Reactor Coolant System Flow Rate

5.3.1 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor

5.3.1.1 Partial Loss of Forced Reactor Coolant Flow