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

August 25, 2011

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-11271

Subject: MHI's Response to US-APWR DCD RAI No. 786-5881 Revision 3 (SRP 15.0)

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Response to US-APWR DCD RAI No. 786-5881 Revision 3 (SRP 15.0)". The enclosed material provides MHI's response to the NRC's "Request for Additional Information (RAI) 786-5881 Revision 3," dated July 26, 2011.

As indicated in the enclosed materials, Enclosure 2 contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]".

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

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc., if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

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

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Response to US-APWR DCD RAI No. 786-5881 Revision 3 (SRP 15.0) (proprietary)
3. MHI's Response to US-APWR DCD RAI No. 786-5881 Revision 3 (SRP 15.0) (non-proprietary)

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

<u>Contact Information</u>

C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11271

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. 786-5881 Revision 3 (SRP 15.0)", dated August 24, 2011, and have determined that the document contains proprietary information that should be withheld from public disclosure. Those pages containing proprietary information are identified with the label "Proprietary" on the top of the page and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The basis for holding the referenced information confidential is that it describes the unique design of the safety analysis, developed by MHI (the "MHI Information").

4. The MHI Information is not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years. Therefore public disclosure of the materials would adversely affect MHI's competitive position.

5. The referenced information has in the past been, and will continue to be, held in confidence by MHI and is always subject to suitable measures to protect it from unauthorized use or disclosure.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information.

7. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's application for certification of its US-APWR Standard Plant Design.

8. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design and testing of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI in the U.S. nuclear plant market.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 25th day of August, 2011.

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

ENCLOSURE 3

UAP-HF-11271
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 786-5881 Revision 3
(SRP 15.0)

August 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/25/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 786-5881 REVISION 3**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.0.0-30

The Doppler reactivity used in the non-LOCA analysis in Chapter 15 includes a Doppler power coefficient of reactivity and a Doppler fuel temperature coefficient of reactivity. In order for the staff to determine that the values selected for these parameters in the Chapter 15 analysis are suitably conservative, please address the following:

a. DCD Table 15.0-1 identifies the feedback extreme (maximum or minimum) assumed for the Doppler reactivity coefficient for each event and references a figure showing the Doppler power coefficient. It is not clear from this description which extreme is assumed for the Doppler fuel temperature coefficient of reactivity and the numerical values of this parameter are not provided. Please add the minimum and maximum values assumed for the Doppler fuel temperature coefficients of reactivity to the DCD and clarify that the Doppler reactivity coefficients described in Table 15.0-1 include both Doppler power and Doppler fuel temperature coefficients, except as noted.

b. For all 15.2 and 15.3 events, discuss the basis used to determine which Doppler feedback extreme was assumed and demonstrate that the assumed value is suitably conservative or bounding.

ANSWER:

a.

In the US-APWR Chapter 15 analyses the maximum (or minimum) value of the Doppler power coefficient is chosen and the corresponding maximum (or minimum) Doppler fuel temperature coefficient is chosen in most MARVEL-M analyses (except the over cooling events from hot zero power). The basis for this convention is described in Attachment 1. For the over cooling events from hot zero power (Subsections 15.1.4 and 15.1.5), the maximum negative value of the Doppler temperature coefficient is used as an additional conservatism. This is conservative because the maximum negative value for Doppler temperature coefficient will add more reactivity to the core due to the coolant temperature decrease (whereas the minimum Doppler power coefficient is chosen since power is increasing during the return to criticality).

DCD Subsection 15.0.0.2.4 generally describes the reactivity coefficient assumptions utilized in the safety analyses for the US-APWR. DCD Table 15.0-1 provides a summary of reactivity coefficient assumptions for each event; however, neither of these places specifically describes the assumptions related to the Doppler fuel temperature coefficient. Therefore, DCD Subsection

15.0.0.2.4 is revised to describe the values used for Doppler fuel temperature coefficient as shown in the "Impact on DCD" section below. In addition, DCD Table 15.0-1 is revised to include additional clarification regarding the event specific assumptions related to the Doppler fuel temperature coefficient as described in the "Impact on DCD" section below.

b.

MHI performed sensitivity studies with maximum Doppler power and temperature coefficients and minimum Doppler power and temperature coefficients in order to demonstrate that a suitably conservative combination was selected for use in the DCD Section 15.2 (heat-up) and 15.3 (low RCS flow) analyses. The sensitivity analyses demonstrate that Doppler feedback is not a key parameter for any of the events in DCD Section 15.2 or 15.3.

In general, the maximum value of the Doppler power coefficient is chosen and the corresponding maximum Doppler fuel temperature coefficient is chosen. The objective is to use the same set of assumptions in all events within the same event category, e.g. 15.2 or 15.3. The exception to this general assumption is the loss of load event. A sensitivity analysis confirmed that the minimum Doppler power and fuel temperature coefficients result in a slightly more severe peak RCS pressure. Although the Doppler coefficients are not key parameters, the DCD analysis assumes the minimum Doppler coefficients for this event since this event is the limiting AOO with respect to RCS pressure.

The results of the sensitivity studies for DCD Section 15.2 (heat-up) and 15.3 (low RCS flow) analyses are shown in Table 15.0.0-30.1 and Figures 15.0.0-30.1 through 15.0.0-30.30. The sensitivity analyses summarized in the table demonstrate that Doppler feedback is not a key parameter for any of the events in DCD Section 15.2 or 15.3. The table also demonstrates that all the results, regardless of the Doppler reactivity feedback assumption, have sufficient margin above (or below) the applicable safety criteria. In fact, the difference in the results is virtually indistinguishable in the plots of the key parameters, although the detailed numerical values do show some small variations.

In conclusion, MHI consistently selected the maximum Doppler feedback assumption for each category of events (heat-up events in Section 15.2 and low flow events in 15.3) to ensure suitably conservative key parameter results. The one exception to this is the loss of load event, which utilizes minimum Doppler feedback for the reason that was previously described.

Table 15.0.0-30.1 Summary of Sensitivity Study Results

Section	Event	DCD Assumptions		Sensitivity Case		DCD case	Sensitivity case
		Doppler Power Coefficient	Doppler Fuel Temperature Coefficient	Doppler Power Coefficient	Doppler Fuel Temperature Coefficient		
15.2.1	Loss of external load	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)		
15.2.6	Loss of non-emergency AC power to the station auxiliaries	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)		
15.2.7	Loss of normal feedwater flow	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)		
15.2.8	Feedwater system pipe break	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)		
15.3.1.1	Partial loss of forced reactor coolant flow	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)		
15.3.1.2	Complete loss of forced reactor coolant flow	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)		
15.3.3	Reactor coolant pump rotor seizure	Max. feedback Fig.15.0-2 in Ch.15	Max. (-2.9E-5Δk/k/°F)	Min. feedback Fig.15.0-2 in Ch.15	Min. (-1.0E-5Δk/k/°F)		

Figure 15.0.0-30.1 **DNBR versus Time**
Doppler Feedback Sensitivity Study
Loss of External Load - DNBR Analysis

Figure 15.0.0-30.2 **Reactor Power versus Time**
Doppler Feedback Sensitivity Study
Loss of External Load - RCS & Main Steam Pressure Analysis

Figure 15.0.0-30.3 **RCP Outlet Pressure versus Time**
Doppler Feedback Sensitivity Study
Loss of External Load - RCS & Main Steam Pressure Analysis

Figure 15.0.0-30.4 **Steam Generator Pressure versus Time**
Doppler Feedback Sensitivity Study
Loss of External Load - RCS & Main Steam Pressure Analysis

Figure 15.0.0-30.5 **Reactor Power versus Time**
Doppler Feedback Sensitivity Study
Loss of Non-Emergency AC Power to the Station Auxiliaries

Figure 15.0.0-30.6 **Pressurizer Water Volume versus Time**
Doppler Feedback Sensitivity Study
Loss of Non-Emergency AC Power to the Station Auxiliaries

Figure 15.0.0-30.7 **Steam Generator Pressure versus Time**
Doppler Feedback Sensitivity Study
Loss of Non-Emergency AC Power to the Station Auxiliaries

Figure 15.0.0-30.8 **DNBR versus Time**
Doppler Feedback Sensitivity Study
Loss of Normal Feedwater Flow - DNBR Analysis

Figure 15.0.0-30.9 **Reactor Power versus Time**
Doppler Feedback Sensitivity Study
Loss of Normal Feedwater Flow - RCS Pressure Analysis

Figure 15.0.0-30.10 **RCP Outlet Pressure versus Time**
Doppler Feedback Sensitivity Study
Loss of Normal Feedwater Flow - RCS Pressure Analysis

Figure 15.0.0-30.11 Steam Generator Pressure versus Time
Doppler Feedback Sensitivity Study
Loss of Normal Feedwater Flow - RCS Pressure Analysis

Figure 15.0.0-30.12 Pressurizer Water Volume versus Time
Doppler Feedback Sensitivity Study
Loss of Normal Feedwater Flow
- Pressurizer Water Volume Analysis

Figure 15.0.0-30.13 Reactor Power versus Time
Doppler Feedback Sensitivity Study
Feedwater System Pipe Break - RCS Pressure Analysis

Figure 15.0.0-30.14 RCP Outlet Pressure versus Time
Doppler Feedback Sensitivity Study
Feedwater System Pipe Break - RCS Pressure Analysis

Figure 15.0.0-30.15 Steam Generator Pressure versus Time
Doppler Feedback Sensitivity Study
Feedwater System Pipe Break - RCS Pressure Analysis

Figure 15.0.0-30.16 Temperature of Faulted Loop versus Time
Doppler Feedback Sensitivity Study
Feedwater System Pipe Break - Hot Leg Boiling Analysis

Figure 15.0.0-30.17 Pressurizer Water Volume versus Time
Doppler Feedback Sensitivity Study
Feedwater System Pipe Break
- Pressurizer Water Volume Analysis

Figure 15.0.0-30.18 Reactor Power versus Time
Doppler Feedback Sensitivity Study
Partial Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.19 DNBR versus Time
Doppler Feedback Sensitivity Study
Partial Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.20 Steam Generator Pressure versus Time Doppler Feedback Sensitivity Study Partial Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.21 Reactor Power versus Time Doppler Feedback Sensitivity Study Complete Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.22 DNBR versus Time
Doppler Feedback Sensitivity Study
Complete Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.23 Steam Generator Pressure versus Time
Doppler Feedback Sensitivity Study
Complete Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.24 **Reactor Power versus Time**
Doppler Feedback Sensitivity Study
Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.25 **DNBR versus Time**
Doppler Feedback Sensitivity Study
Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.26 Steam Generator Pressure versus Time
Doppler Feedback Sensitivity Study
Frequency Decay Resulting in a Complete Loss of Forced Reactor Coolant Flow

Figure 15.0.0-30.27 Cladding Inside Temperature versus Time
Doppler Feedback Sensitivity Study
RCP Rotor Seizure - Cladding Temperature Analysis

Figure 15.0.0-30.28 Reactor Power versus Time
Doppler Feedback Sensitivity Study
RCP Rotor Seizure - RCS Pressure Analysis

Figure 15.0.0-30.29 RCP Outlet Pressure versus Time
Doppler Feedback Sensitivity Study
RCP Rotor Seizure - RCS Pressure Analysis

Figure 15.0.0-30.30 **Steam Generator Pressure versus Time**
Doppler Feedback Sensitivity Study
RCP Rotor Seizure - RCS Pressure Analysis

Impact on DCD

DCD Subsection 15.0.0.2.4 and Table 15.0-1 are revised as indicated in the mark-up in Attachment 2.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

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

Attachment 1

Attachment 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/25/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 786-5881 REVISION 3**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.0.0-31

The only reference in the Tier 2 DCD for MUAP-07026-P, "Mitsubishi Reload Evaluation Methodology," is in Chapter 16, TS 5.6.3. The NRC requests MUAP-07026-P be added as a Reference to Chapter 15 in order for the NRC to evaluate this document as part of the Chapter 15 SE.

ANSWER:

MUAP-07026-P will be added as a reference to Chapter 15 as requested by the NRC.

Impact on DCD

DCD Subsection 15.0.0.2 and Subsection 15.0.5 are revised as indicated in the mark-up in Attachment 2.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/25/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 786-5881 REVISION 3**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.0.0-32
In order to find that the acceptance criteria on fuel cladding integrity is maintained for the event specific analysis in Chapter 15, identify the numerical value of the 95/95 departure from nucleate boiling ratio (DNBR) in the DCD.

ANSWER:

DCD Revision 3 Subsection 4.4.1.1.2 currently provides the numerical value of the 95/95 DNBR safety analysis limit utilized for the safety analyses. A quote of the relevant portion of Subsection 4.4.1.1.2 is provided below for convenience.

> "The safety analysis limit of Min. DNBR is determined as 1.45 for both the channel types, accommodating the DNBR penalties incurred due to rod bows described in Subsection 4.4.2.2.4 and transition core geometry, and/or reserving more core operational flexibilities."

Impact on DCD

There is no impact on the DCD.

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

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

- Several additional event-specific criteria, which are described in Section 15.4.8.2.5, are applied to the rod ejection accidents.

- For loss-of-coolant accidents (LOCA), the analysis criteria of 10 CFR 50.46 also apply (SRP 15.0):

- The calculated maximum fuel clad temperature shall not exceed 2200°F.

- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Calculated changes in core geometry shall be such that the core remains amenable to cooling.

- After successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

The SRPs provide additional criteria for certain initiating events, which are described on a case-by-case basis in each respective event analysis section.

The third column of Table 15.0-1 indicates which initiating events are classified as PAs.

15.0.0.2 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

This subsection describes the plant characteristics and initial conditions assumed in the analysis of AOO and PA events. The nuclear key parameters described in this subsection will be confirmed in each reload core based on Reference 15.0-21. DCD_15.0.0-31

15.0.0.2.1 Design Plant Conditions

Table 15.0-2 lists key rated (nominal) power conditions. Two power ratings are considered:

- The design core thermal power output.

- The design nuclear steam supply system (NSSS) thermal power output, which includes the thermal power generated by the reactor coolant pumps (RCPs).

values used in the transient analyses are provided in Figure 15.0-2. For the Doppler fuel temperature coefficient, one of two constant values is used. The minimum value is -1.0E-5 Δk/k/°F and the maximum value is -2.9E-5 Δk/k/°F. For most accidents analyzed using the MARVEL-M code (described in Section 15.0.2.2.1), one of two constant values of the moderator density coefficient is used. The minimum value is 0.0 (Δk/k)/(g/cc) and the maximum value is 0.51 (Δk/k)/(g/cc). The justification for the use of specific values of these coefficients is described on a case-by-case basis in the respective analysis section. A summary of the reactivity coefficient assumptions used for each event is provided in Table 15.0-1. DCD_15.0.0-30

15.0.0.2.5 Rod Cluster Control Assembly Insertion Characteristics

A reactor trip signal causes all of the RCCAs to be inserted by gravity to the bottom of the active fuel region. In the analyses, the single highest-reactivity-worth RCCA is conservatively assumed to fail to insert (i.e., to remain fully withdrawn).

Figure 15.0-3 is a plot displaying the conservative RCCA displacement as a function of time that is used in the analyses for the RCCA insertion following a reactor trip until RCCAs are fully inserted. A time of 3 seconds is used for insertion time to dashpot entry.

Figure 15.0-4 shows the negative reactivity addition as a function of time that is used in the analysis for the RCCA insertion following a reactor trip. This curve is based on: (1) the same conservatively slow RCCA insertion rate discussed in the preceding paragraph and (2) a conservative bottom-skewed axial power distribution within the core.

The RCCA negative reactivity insertion versus time shown in Figure 15.0-4 is input into the computer codes used in the analyses. Unless otherwise described in the individual event analysis sections, the scram reactivity is -4%Δk/k for hot full power condition.

15.0.0.2.6 Residual Decay Heat

15.0.0.2.6.1 Total Residual Decay Heat

Residual heat in a subcritical core, including decay heat from fission products and actinides, is calculated for the large break LOCA and the non-LOCA transient in accordance with the methodology of ANSI/ANS-5.1-1979 (Ref. 15.0-15).

For the small break LOCA and post-LOCA long-term cooling analysis, the decay heat from fission products is assumed to be equal to 1.2 times the values for infinite operating time in the ANS standard 5.1-1971, conforming to the requirement of 10 CFR 50 Appendix K (Ref. 15.0-7). The heat from the decay of actinides is calculated in accordance with the methodology of ANSI/ANS-5.1-1979.

Input parameters used with ANSI/ANS-5.1-1979 are selected so as to envelope conceivable core conditions for the US-APWR.

15.0.0.2.6.2 Distribution of Decay Heat Following a Loss of Coolant Accident

Early in a LOCA, the neutron chain reaction in the core is terminated due to void formation or RCCA insertion, or both. After this shutdown, most of the heat generation in

15.0-21 Mitsubishi Reload Evaluation Methodology, MUAP-07026-P Rev. 0 (Proprietary) and MUAP-07026-NP Rev. 0 (Non-Proprietary), December 2007. DCD_15.0.0-31

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 1 of 4)

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)	
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler[*7]		DCD_15.0.0-30
15.1.1	Decrease in feedwater temperature	AOO	MARVEL-M	max	--	min feedback Figure 15.0-2	4466	
15.1.2	Increase in feedwater flow	AOO	MARVEL-M	max	--	min feedback Figure 15.0-2	4466	
15.1.3	Increase in steam flow	AOO	MARVEL-M	min and max	--	min feedback Figure 15.0-2	4466	
15.1.4	Inadvertent opening of a steam generator relief or safety valve	AOO	MARVEL-M, ANC, VIPRE-01M	See Figure 15.1.4-1	--	See Figure 15.1.4-2[*8]	0	DCD_15.0.0-30
15.1.5	Steam system piping failures - Minor/Major	AOO/PA	MARVEL-M, ANC, VIPRE-01M[*1]	Hot standby: Figure 15.1.4-1 HFP: max	--	Hot standby: Figure 15.1.4-2[*8] HFP: min feedback Figure 15.0-2	0%, 75%, & 100% of 4466	DCD_15.0.0-30
15.2.1	Loss of external load	AOO	MARVEL-M	min	--	min feedback Figure 15.0-2	4466 for DNBR 4555[*2] for RCS pressure	
15.2.2	Turbine trip	AOO	Bounded by loss of load	--	--	--	--	
15.2.3	Loss of condenser vacuum	AOO	Bounded by loss of load	--	--	--	--	
15.2.4	Closure of main steam isolation valves	AOO	Bounded by loss of load	--	--	--	--	
15.2.5	Steam pressure regulator failure	N/A to US-APWR						
15.2.6	Loss of non-emergency AC power to the station auxiliaries	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4555[*2]	

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 2 of 4)

DCD_15.0.0-30

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler*7	
15.2.7	Loss of normal feedwater flow	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4466 for DNBR4555*2 for RCS pressure, Pzr Level
15.2.8	Feedwater system pipe break - Minor/Major	AOO/PA	MARVEL-M	min	--	max feedback Figure 15.0-2	4555*2
15.3.1.1	Partial loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M	min	--	max feedback Figure 15.0-2	4466
15.3.1.2	Complete loss of forced reactor coolant flow	AOO	MARVEL-M, VIPRE-01M	min	--	max feedback Figure 15.0-2	4466
15.3.3	Reactor coolant pump rotor seizure	PA	MARVEL-M, VIPRE-01M	min	--	max feedback Figure 15.0-2	4555*2
15.3.4	Reactor coolant pump shaft break	PA	Bounded by rotor seizure	--	--	--	--
15.4.1	Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition	AOO	TWINKLE-M, VIPRE-01M, MARVEL-M	--	+2	Temperature coefficient -20% from design	0
15.4.2	Uncontrolled control rod assembly withdrawal at power	AOO	MARVEL-M	min and max	--	min and max feedback Figure 15.0-2	10%, 75%, & 100% of 4466
15.4.3	Control rod misoperation	AOO/PA	MARVEL-M, VIPRE-01M*1	min	--	min feedback Figure 15.0-2	4466

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 3 of 4)

DCD_15.0.0-30

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler[*7]	
15.4.4	Startup of an inactive loop or recirculation loop at an incorrect temperature	AOO	N/A	--	--	--	--
15.4.5	Flow controller malfunction causing an increase in BWR recirculation loop	N/A to US-APWR					
15.4.6	Inadvertent decrease in boron concentration in the RCS	AOO	N/A	--	--	--	0 and 4466
15.4.7	Inadvertent loading and operation of a fuel assembly in an improper Position	PA	ANC	--	--	--	--
15.4.8	Spectrum of rod ejection accidents	PA	TWINKLE-M, VIPRE-01M, MARVEL-M	--	Temperature coefficient -20% from design	Temperature coefficient -20% from design	0 and 4540[*3]
15.5.1	Inadvertent operation of ECCS that increases reactor coolant inventory	AOO	N/A	--	--	--	--
15.5.2	CVCS malfunction that increases reactor coolant inventory	AOO	MARVEL-M	min	--	min feedback Figure 15.0-2	4555[*2]
15.6.1	Inadvertent opening of a PWR pressurizer pressure relief valve	AOO	MARVEL-M	min	--	max feedback Figure 15.0-2	4466

Table 15.0-1
Summary of Event Classification, Initial Conditions and Computer Codes (Sheet 4 of 4)

DCD_15.0.0-30

Section	Event	Category	Computer Code(s) Utilized	Reactivity Coefficients Assumed			Initial Power Output (MW_t)
				Moderator Density	Moderator Temperature (pcm/°F)	Doppler*7	
15.6.2	Radiological consequences of the failure of small lines carrying primary coolant outside containment	PA	RADTRAD	--	--	--	4540*3
15.6.3	Radiological consequences of SGTR	PA	MARVEL-M	min	--	max feedback Figure 15.0-2	4555*2
15.6.5	Loss-of-Coolant Accidents	PA	WCOBRA/TRAC, HOTSPOT	*4	--	*4	4466
			M-RELAP5	*5	--	*6	4555*2

Notes:

*1Steady state analysis

*2102% of 4466MW_t (NSSS thermal power)

*3102% of 4451MW_t (core thermal power)

*4Applicability confirmed (Ref.15.0-18).

*5 Conservative Moderator Density Coefficient changes with moderator density assumed (Ref.15.0-20).

*6Conservative Doppler Temperature Coefficient changes with moderator density assumed (Ref.15.0-20).

*7Doppler feedback may be modeled by a Doppler power coefficient as well as a Doppler fuel temperature coefficient. Unless otherwise noted, the two coefficients are assumed to be consistent (i.e. both minimum or both maximum). Values for the Doppler fuel temperature coefficient are provided in Subsection 15.0.0.2.4. DCD_15.0.0-30

*8For the over cooling events from hot zero power (Subsections 15.1.4 and 15.1.5), the maximum negative value of the Doppler fuel temperature coefficient is used to conservatively add more reactivity to the core due to the coolant temperature decrease. The minimum Doppler power coefficient is chosen since power is increasing during the return to criticality.