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TOKYO, JAPAN

September 9, 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-11305

Subject: MHI's Revised Response to US-APWR DCD RAI No. 769-5797 Revision 3 (SRP 15.0)

References:
1) "REQUEST FOR ADDITIONAL INFORMATION 769-5797 REVISION 3", dated June 14, 2011 (ML111662035)
2) "MHI's Response to US-APWR DCD RAI No. 769-5797 Revision 3 (SRP 15.0)", MHI Letter UAP-HF-11224, dated July 15, 2011 (ML11206A421)

Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "MHI's Revised Response to US-APWR DCD RAI No. 769-5797 Revision 3 (SRP 15.0)". In Reference 2, MHI provided the response to the NRC's Request for Additional Information ("RAI") in Reference 1. MHI has revised the response to Questions 15-26, 15-27, and 15-29 of Reference 2 based on feedback provided in conference calls between MHI and the NRC. The enclosed material provides MHI's revised response to the NRC's "Request for Additional Information (RAI) 769-5797 Revision 3," dated June 14, 2011. This letter transmits only the revised responses for Questions 15-26, 15-27, and 15-29; the responses to the other questions transmitted by Reference 2 have not been revised.

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.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata
2. MHI's Revised Response to US-APWR DCD RAI No. 769-5797 Revision 3 (SRP 15.0) (proprietary)
3. MHI's Revised Response to US-APWR DCD RAI No. 769-5797 Revision 3 (SRP 15.0) (non-proprietary)

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

<u>Contact Information</u>

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ENCLOSURE 1

Docket No. 52-021

MHI Ref: UAP-HF-11305

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 documents entitled "MHI's Revised Response to US-APWR DCD RAI No. 769-5797 Revision 3 (SRP 15.0)", dated September 9, 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 9th day of September, 2011.

Y. Ogata

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

ENCLOSURE 3

UAP-HF-11305
Docket No. 52-021

MHI's Revised Response to US-APWR DCD RAI No. 769-5797
Revision 3 (SRP 15.0)

September 2011

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/09/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 769-5797 REVISION 3**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0**
DATE OF RAI ISSUE:	**6/14/2011**

QUESTION NO.: 15-26

Table 15.0-5 identifies the time delays for mitigation systems. Provide the source for each of these values. Are these time delays measured from when the instrumentation reaches the actuation limit (i.e. time delays in Table 15.0-5 already include the signal time delay identified in Table 15.0-4) or are they measured from when the mitigation system receives the signal (i.e. signal time delays from Table 15.0-4 should be added to the mitigation system delays from Table 15.0-5)?

ANSWER:

DCD Table 15.0-5 provides the mitigation system time delays for two general types of mitigative features: (1) valve motion only and (2) systems involving pump startup. Since the source of the values and their relation to the values in DCD Table 15.0-4 are different, each of these general types is discussed separately.

(1) Valve motion only

This general type covers the first four rows of Table 15.0-5. In each of these four cases, the mitigation "system" is only a valve that needs to be closed when it receives the appropriate signal. Therefore, the time in Table 15.0-5 for these four entries is simply the amount of time that it would take for the valve to mechanically close after it receives a signal to close. Since the time only depends on a mechanical action, the basis for each of the values is the past operating experience for the type of valve. The valve closure times will be used to create specifications during procurement to ensure that these times used in the safety analysis are valid. For the main feedwater isolation valve and main feedwater regulation valve, the 5 second valve closure time is further controlled by the Chapter 16 Technical Specification (TS) Surveillance Requirement (SR) 3.7.3.1. Likewise, the main steam isolation valve closure time of 5 seconds is controlled by TS SR 3.7.2.1. The main steam relief valve block valve closure time (30 seconds) and the emergency feedwater isolation valve closure time (20 seconds) are not specified in the TS but are based on experience and procurement specifications.

Since each of these times is only the mechanical valve closure times following a valid signal, they do not include any signal delays. The signal delays are provided in Table 15.0-4.

Therefore, the total delay time is obtained by adding the values from Table 15.0-4 and Table 15.0-5. For example, the main steam line isolation signal results in the automatic closure of the main steam isolation valves. Table 15.0-4 specifies that the signal delay time is 3.0 seconds after the main steam line isolation setpoint is reached. Table 15.0-5 specifies that the main steam isolation valve closure time is 5 seconds. Therefore, the full closure of the main steam isolation valves will not occur until 8.0 seconds after the main steam line isolation setpoint is reached. This example case can be verified in the time sequence of events for the steam line break in Table 15.1.5-1: the setpoint is reached at 1.5 seconds and full valve closure occurs at 10.0 seconds (8.0 seconds of delay time plus 0.5 seconds of additional margin).

(2) Systems involving pump startup

This general type covers the last three rows of Table 15.0-5. In each of these three cases, the mitigation system includes pumps that need to be started and reach full flow. The time in Table 15.0-5 for these three entries consists of several components. The detailed breakdown of the components is provided as the basis as follows:

EFW Pump	Motor-Driven	Turbine-Driven	
Response time of sensor and digital controller	3.0 seconds	3.0 seconds	(from Table 15.0-4)
GTG start and load delay time	100.0 seconds	NA	(for LOOP cases only)
Sequence time delay and margin	22.0 seconds	NA	
Response time of digital controller and electrical circuit	3.0 seconds	3.0 seconds	
EFW pump time to full flow	5.0 seconds	<54.0 seconds	
Total*	60.0 seconds 133.0 seconds		 (for LOOP cases only)

*Note that the total time is the time at which all EFW pumps would be running, so it is determined from the longer of either the motor-driven or turbine-driven EFW pump delays.

SI Pump		
Response time of sensor and digital controller	3.0 seconds	(from Table 15.0-4)
GTG start and load delay time	100.0 seconds	(for LOOP cases only)
Sequence time delay and margin	7.0 seconds	
Response time of digital controller and electrical circuit	3.0 seconds	
SI pump time to full flow	5.0 seconds	
Total	18.0 seconds 118.0 seconds	 (for LOOP cases only)

Containment Spray System		
Response time of sensor and digital controller	3.0 seconds	(from Table 15.0-4)
GTG start and load delay time	100.0 seconds	
Sequence time delay and margin	32.0 seconds	
Response time of digital controller and electrical circuit	3.0 seconds	
Pump motor starting time to full speed	5.0 seconds	
Time for the water to reach the CS nozzles through the CS lines	100.0 seconds	
Total*	243.0 seconds	

*Note that all analysis cases that consider the CS pumps are LOOP cases, so only the time for LOOP cases is shown here.

As indicated by the tables above, each of the total mitigation system delay times already includes the signal delay time from Table 15.0-4. Therefore no other addition is required since the values in Table 15.0-5 for these three items reflect the time from when the setpoint is reached until the system is fully functional (pumps at full flow). [Note that this is treatment is different than what was previously described for the valve closures.] The delay times vary based on the availability of offsite power. If offsite power is not available (LOOP), then the delay time has to consider the additional delay for the GTGs to start and load in order to power the pumps. As an example case, the ECCS low main steam line pressure setpoint will result in SI pumps being at full flow after 18 seconds for the case with offsite power and 118 seconds for the case without offsite power. This example can be verified in the time sequence of events for the steam line break in Table 15.1.5-1: the setpoint is reached at 1.5 seconds and SI pumps are running at 21.5 seconds for the case with offsite power (18 seconds of delay time plus 2 seconds of additional margin) and 121.5 seconds for the case without offsite power (118 seconds of delay time plus 2 seconds of additional margin).

Note: ECCS pump start-up and MSIV closure are actuated by a lead/lag compensated main steam line pressure signal, as shown in DCD Figure 7.2-2 Sheet 9. The main steam line pressure signal used to actuate EFW does NOT include lead/lag compensation, as shown in DCD Figure 7.2-2 Sheet 8.

Since some of the time delays in Table 15.0-5 already include the delays from Table 15.0-4 but others do not, the DCD will be revised as indicated in the "Impact on DCD" section below in order to clearly indicate the relationship between the values in the two tables.

Impact on DCD

In order to clarify how the values in Table 15.0-4 and 15.0-5 are related, DCD Subsection 15.0.0.2 will be modified as indicated in the mark-up in Attachment 1.

In order to clarify how the values in Table 15.0-4 and 15.0-5 are related, DCD Table 15.0-5 will be modified as indicated in the mark-up in Attachment 1.

In order to clarify the timing of reaching the EFW isolation signal, DCD Tables 15.1-5-1 and 15.2.8-1 will be revised as indicated in the mark-up in Attachment 1.

In order to clarify the lead/lag compensation associated with the low main steam line pressure signal, DCD Subsection 15.1.4.2 will be modified as indicated in the mark-up in Attachment 1.

In order to clarify the lead/lag compensation associated with the low main steam line pressure signal, DCD Subsection 15.1.5.2 will be modified as indicated in the mark-up in Attachment 1.

In order to clarify the lead/lag compensation associated with the low main steam line pressure signal, the 7th bullet of DCD Subsection 15.1.4.3.2 will be modified as indicated in the mark-up in Attachment 1.

In order to clarify the lead/lag compensation associated with the low main steam line pressure signal, the 6th bullet of DCD Subsection 15.1.5.3.2 (1) will be modified as indicated in the mark-up in Attachment 1.

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.

the core results from absorption of gamma rays generated by the fission products. The location where gamma rays are absorbed may be some distance from the parent fission product nuclei. Therefore, the distribution of heat generation within the core is changed from that at the steady-state. The fraction of heat generated within the clad and pellet is 97.4 percent at the steady state. This factor reduces for the high-power rod after shutdown. The large-break LOCA analyses account for this effects.

15.0.0.3 Reactor Trip System and Engineered Safety Feature Systems Analytical Limit and Delay Times

The reactor trip system (RTS) initiates signals to open the reactor trip breakers when operating parameters that are monitored by the RTS approach pre-determined limits. This action removes power to the control rod drive mechanism (CRDM) coils permitting the rods to fall by gravity into the core.

Instrumentation system time delays are associated with each of the RTS trip functions. These include delays in sensor, signal generation, opening the reactor trip breakers, and the release of the rods by the CRDM. The total response time delay for a reactor trip is the interval from the time the operating parameter reaches the analytical limit to the time the control rods are released and start to drop into the core. The delay for each trip signal is selected so as to give conservative analysis results. Chapter 7 provides a general discussion of the reactor trip and engineered safety features (ESF) actuation signals.

Table 15.0-4 summarizes the reactor trip and ESF actuation analytical limit and response delay times for functions used in the event analyses. The difference between the trip analytical limit and the nominal trip setpoint specified in the plant technical specifications accounts for instrumentation channel error and setpoint error. Both the availability and range of each type of instrumentation are consistent with the corresponding predicted parameter values in the specific event analyses. The instrumentation and control characteristics used in the specific event analyses (including values used for analytical limit and delay times) are consistent with the information documented in Sections 7.2 and 7.3.

Table 15.0-5 summarizes the time delays associated with accident-mitigating equipment such as valve movement or pump start delays. Except where otherwise noted, the time delays in Table 15.0-5 should be added to the delays in Table 15.0-4. In addition, instrumentation is provided to monitor key plant parameters during events (e.g., EFW flow indication in the main control room) as described in Section 7.5. DCD_15-26

15.0.0.4 Component Failures

The accident analyses documented in this chapter account for certain component failures. This treatment is an element of the defense in depth safety philosophy. The discussion of component failures in this section is based in part on Reference 15.0-11.

Certain events (such as a steam system piping failure) are initiated by component failures. The analysis of these accidents assumes that any equipment that can be damaged as a consequence of the initiating event is not available for mitigation of the accident. In addition, the analysis must demonstrate that the acceptance criteria are met

Table 15.0-5
Mitigation System Time Delays

Component	Time Delay (sec)	
Main feedwater isolation valve closure, main feedwater regulation valve closure	5	
Main steam isolation valve closure	5	
Main steam relief valve block valve closure	30	
Emergency feedwater isolation valve closure	20	
Emergency feedwater pump - initiation to full flow with offsite electrical power without offsite electrical power	 60[*3] 133[*1,*2,*3]	DCD_15-26
Safety injection pump - initiation to full flow with offsite electrical power without offsite electrical power	 18[*2,*3] 118[*1,*2,*3]	DCD_15-26
Containment spray system initiation without offsite electrical power	243[*1,*3]	DCD_15-26

Notes:
*1 including emergency standby gas turbine generator start and load delay (100 seconds)
*2 depending on the event, additional time margin may be taken into consideration
*3 the time delays for these mitigative systems already include the signal delays from Table 15.0-4 DCD_15-26

- Over power ΔT
- Low pressurizer pressure
- ECCS actuation

The following signals could actuate the ECCS, which injects borated water into the reactor vessel via the safety injection pumps:

- Low pressurizer pressure
- Low main steam line pressure (any one loop)

An ECCS actuation signal provides feedwater isolation by automatically tripping the main feedwater pumps and fully closing all control valves and feedwater isolation valves in the feedwater system to prevent feedwater from excessively cooling the RCS. The signal also starts the emergency feedwater (EFW). The main steam line pressure signal used to actuate the ECCS function is a lead/lag compensated signal (See Figure 7.2-2 Sheet 9 of 21), which results in this signal occurring early in the transient. In addition to the reactor trips listed above, the following engineered safety feature functions are assumed to be available to mitigate the accident: DCD_15-26 S01

- Steam line isolation
- EFW automatic actuation DCD_15-26 S01
- EFW isolation
- ECCS
- Main feedwater isolation

The automatic reactor coolant pump (RCP) trip will actuate on an ECCS actuation signal generated from low pressurizer pressure or low main steam line pressure. However, the RCP trip is ignored in this analysis to maximize the RCS cooldown and associated reactivity and return-to-power response.

15.1.4.3 Core and System Performance

15.1.4.3.1 Evaluation Model

The MARVEL-M plant transient analysis code is used to calculate transient responses of primary and secondary temperatures and pressures, pressure-dependent relief or safety valve flow rate, reactivity, and boron concentration, as well as the automatic actuation and operation of engineered safety feature functions such as safety injection, emergency feedwater initiation and isolation, and main feedwater isolation. In addition, a non-perfect mixing model is used for the reactor vessel inlet plenum for the purpose of conservatively predicting reactivity for the non-uniform core inlet conditions. Additional information on the use of the MARVEL-M code for analyzing non-LOCA events including secondary steam releases can be found in Reference 15.1-2. This evaluation model is described in Section 15.0.2.2.1.

instantaneously open. The steam release from this valve is assumed to be 485 lb/s at 1200 psia. The valve is assumed to be upstream of the main steam check valve, resulting in the unterminated blowdown of a single steam generator.

- The shutdown margin is assumed to be 1.6% $\Delta k/k$, corresponding to the most limiting condition in the cycle, with the most reactive RCCA in a fully withdrawn position.
- The boron concentration in the refueling water storage pit (RWSP) is assumed to be 4000 ppm. This value is lower than or equal to the minimum allowable Technical Specification value. The accumulators are not actuated for this accident.
- A dry steam blowdown (steam quality = 1.0) is assumed. This assumption maximizes the energy released from the failed-open valve. The Moody curve for f(L/D) = 0 (Ref. 15.1-4) is used for calculating the steam flow from the failed-open valve.
- EFW is assumed to be initiated at time t = 0 and deliver flow at rated capacity for the purpose of maximizing the cooldown. EFW is automatically isolated from the affected steam generator on a low main steam line pressure signal. This signal uses actual main steam line pressure as opposed to the ECCS actuation signal on low main steam line pressure that is lead/lag compensated. This assumption increases the primary-to-secondary heat transfer. DCD_15-26 S01
- Only two pumps operate to inject borated water from the RWSP into the reactor vessel downcomer. This treatment is consistent with the most severe single active failure, assumed to be one train of the ECCS, and allows for future operational flexibility.
- The core and systems performance analysis conservatively ignores decay heat to provide the maximum RCS cooldown during the transient.
- The nominal primary-to-secondary heat transfer coefficient is used to maximize heat transfer to the secondary. In addition, the reverse heat transfer coefficient is set to zero, so that heat cannot be transferred from the secondary to the primary side if the steam generator temperature is warmer than the primary coolant in the steam generator tubes.
- No credit is taken for the heat capacity of the reactor coolant system or steam generator thick metal in attenuating the resulting plant cooldown. This assumption helps to maximize the heat transfer from the primary to the secondary.
- The time required for borated water to reach the core is determined by taking into consideration: (1) the period from the time the ECCS actuation signal is generated to the time the safety injection pumps reach full speed and (2) transport time for the injected water to pass through the reactor coolant piping. These delays and purge volumes are directly modeled in the MARVEL-M code.

- Low pressurizer pressure
- High power range neutron flux

The following signals could actuate the ECCS, which injects borated water into the reactor vessel via the safety injection pumps:

- Low pressurizer pressure
- Low main steam line pressure (any one loop)
- High containment pressure

An ECCS actuation signal provides feedwater isolation by automatically tripping the main feedwater pumps and fully closing all control valves and feedwater isolation valves in the feedwater system. The signal also starts the EFW. The main steam line pressure signal used to actuate the ECCS function is a lead/lag compensated signal (See Figure 7.2-2 Sheet 9 of 21), which results in this signal occurring early in the transient. In addition to the reactor trips listed above, the following engineered safety feature functions are assumed to be available to mitigate the accident: DCD_15-26 S01

- Main steam line isolation
- EFW automatic actuation DCD_15-26 S01
- EFW isolation
- ECCS
- Main feedwater isolation

The automatic reactor coolant pump (RCP) trip will actuate on an ECCS actuation signal generated from low pressurizer pressure, low main steam line pressure, or high containment pressure. The core response for the limiting steam system piping failure event is analyzed with and without offsite power as described in Section 15.1.5.3 below (Cases A and B). The RCP trip is ignored for the case with offsite power available to maximize the RCS cooldown and associated reactivity and return-to-power response. For the case assuming loss of offsite power, the RCP trip is assumed to occur on the ECCS actuation signal as designed. No operator action is required to trip the RCPs for this event.

The availability and adequacy of instrumentation and controls is described in Section 15.0.0.3.

To prevent excessive cooldown of the reactor coolant, the main feedwater regulation valves are fully closed by the reactor trip coincident with a low reactor coolant average temperature (P-4) signal. Also, the ECCS actuation signal automatically trips the main feedwater pumps, and fully closes all the control valves and main feedwater isolation valves.

- EFW is assumed to be initiated at time t = 0 and deliver flow at rated capacity for the purpose of maximizing the cooldown. EFW is automatically isolated from the affected steam generator on a low main steam line pressure signal. This signal uses actual main steam line pressure as opposed to the ECCS actuation signal on low main steam line pressure that is lead/lag compensated. This assumption increases the primary-to-secondary heat transfer. DCD_15-26 S01

- Only two pumps operate to inject borated water from the RWSP into the reactor vessel downcomer. This treatment is consistent with the most severe single active failure, assumed to be one train of the ECCS, and allows for future operational flexibility.

- The core and systems performance analysis conservatively ignores decay heat to provide the maximum RCS cooldown during the transient.

- The nominal primary-to-secondary heat transfer coefficient is used to maximize heat transfer to the secondary. In addition, the reverse heat transfer coefficient is set to zero, so that heat cannot be transferred from the secondary to the primary side if the steam generator temperature is warmer than the primary coolant in the steam generator tubes.

- The time required for borated water to reach the core is determined by taking into consideration: (1) the period between the time the ECCS actuation signal is generated and the time the safety injection pumps reach full speed and (2) the time for the injected water to pass through the reactor coolant piping. These delays and purge volumes are directly modeled in the MARVEL-M code.

- No credit is taken for the heat capacity of the reactor coolant system or steam generator thick metal in attenuating the resulting plant cooldown. This assumption helps to maximize the heat transfer from the primary to the secondary.

- The pressurizer is modeled on one of the intact loops.

- Conservative axial power profile and radial power distribution are assumed in the analysis as described in Section 15.0.0.2.3.

- For Case A with offsite power available, the reactor coolant pumps are assumed to run for the duration of the transient and provide nominal RCS flow. This is conservative in that full flow enhances primary-to-secondary heat transfer, maximizing the adverse effect of reactivity, power, and non-uniform inlet temperature on the core. This assumption bounds the case where the reactor coolant pumps trip as designed on ECCS actuation.

- For Case B without offsite power, the reactor coolant pumps are assumed to trip as designed on ECCS actuation.

(2) Case C – Spectrum of Breaks from Power with Offsite Power

DNBR is calculated the same way as for the RCCA Bank Withdrawal at Power described in Section 15.4.2 (using internal data tables based on RTDP calculations using the

Table 15.1.5-1
Time Sequence of Events for the Steam System Piping Failure

Event Description	Case A Time [sec]	Case B Time [sec]
Steam pipe rupture occurs	0.0	0.0
Low steamline pressure analytical limit reached	1.5	1.5
RCP coastdown begins	N/A	4.5
MSIVs closed	10.0	10.0
EFW isolation signal (low main steam line pressure analytical limit reached) (Case B)	N/A	20.2
Automatic isolation of EFW to faulted SG (Case B)	N/A	50.2
Safety injection pumps start	21.5	121.5
EFW isolation signal (low main steam line pressure analytical limit reached) (Case A)	21.7	N/A
Boron reaches core	44.9	141.4
Automatic isolation of EFW to faulted SG (Case A)	51.7	N/A
Peak core heat flux occurs	89.8	152.8
Faulted SG water mass depleted	330	1420

DCD_15-26 S01

DCD_15-26 S01

DCD_15.01.05-6

Event Description	Case C[*1] Time [sec]
Steam pipe rupture occurs	0.0
Over power ΔT analytical limit reached	8.0
Reactor trip initiated (rod motion begins)	14.0
Minimum DNBR occurs	14.7

*1 Limiting case at power break

Table 15.2.8-1 Time Sequence of Events for Feedwater System Pipe Break - RCS Pressure Analysis

Event	Time (sec)
Loss of feedwater flow occurs	0
Pressurizer safety valves open	44
Low steam generator water level analytical limit reached	66
Feedwater line break initiated	66
Reactor trip initiated (rod motion begins)	68
RCP coastdown begins	68
Peak RCP outlet pressure occurs	71
Main steam safety valves open	72
EFW isolation signal (low main steam line pressure analytical limit reached)	77
EFW isolated to faulted loop	107
EFW pumps start	208
Peak pressurizer water volume occurs	1683

DCD_15-26
S01

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/09/2011

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 769-5797 REVISION 3**
SRP SECTION:	**15 – INTRODUCTION – TRANSIENT AND ACCIDENT ANALYSES**
APPLICATION SECTION:	**15.0**
DATE OF RAI ISSUE:	**6/14/2011**

QUESTION NO.: 15-27

DCD 15.0.0.2.2 states that the steady-state errors of ±4°F for average RCS temperature and ±30 psi for pressurizer pressure are conservative. In accordance with SRP 15.0 I.6.C.ii, please provide the bases for these values so the staff can assess the degree of conservatism.

ANSWER:

The steady-state errors are calculated using the method described in MUAP-09022 Rev. 2 "US-APWR Instrument Setpoint Methodology". A summary of this calculation method and the detailed breakdown of the steady-state errors for the pressurizer pressure and RCS average temperature (T_{avg}) are provided as follows.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

9/09/2011

US-APWR Design Certification

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RAI NO.:	**NO. 769-5797 REVISION 3**
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QUESTION NO.: 15-29

Per Table 15.0.0-10.1 (included in the response to RAI 15.0.0-10), the increase in heat removal transients analyzed in Sections 15.1.1 through 15.1.5 assume 10% of the steam generator tubes are plugged. This appears to be non-conservative because it reduces the heat transfer from the primary to secondary system, lessening the severity of these cool-down events. Please justify this assumption.

ANSWER:

In order to determine the impact of the SG tube plugging used in the initial conditions for the events in DCD Section 15.1, MHI has performed a sensitivity study. The sensitivity study was performed for the steam line break in Section 15.1.5 since this is the limiting event in this group of events (increase in heat removal). The sensitivity analysis which assumes the 0% SG tube plugging was performed for steam line break Cases A (with offsite power), B (without offsite power), and C (at power). The results of the sensitivity study are shown in Figures 15-29.1 through 15-29.31. These results show that the impact of the SG tube plugging assumption is negligible and all applicable limits continue to be met. Therefore, MHI assumes the 10% SG tube plugging in the increase in heat removal transients (Section 15.1.1 through 15.1.5) as a consistent assumption with the rest of the DCD Chapter 15 transients.

Figure 15-29.1 Core Reactivity versus Time
Steam System Piping Failure (Case A)

Figure 15-29.2 Reactor Power versus Time
Steam System Piping Failure (Case A)

Figure 15-29.3 Core Heat Flux versus Time
Steam System Piping Failure (Case A)

Figure 15-29.4 RCS Pressure versus Time
Steam System Piping Failure (Case A)

Figure 15-29.5 Pressurizer Water Volume versus Time
Steam System Piping Failure (Case A)

Figure 15-29.6 Core Average Temperature versus Time
Steam System Piping Failure (Case A)

Figure 15-29.7 Reactor Vessel Inlet Temperature versus Time
Steam System Piping Failure (Case A)

Figure 15-29.8 Steam Generator Pressure versus Time
Steam System Piping Failure (Case A)

Figure 15-29.9 Steam Generator Water Mass versus Time
Steam System Piping Failure (Case A)

Figure 15-29.10 Steam Flow Rate versus Time
Steam System Piping Failure (Case A)

Figure 15-29.11 Feedwater Flow Rate versus Time
Steam System Piping Failure (Case A)

Figure 15-29.12 Core Boron Concentration versus Time
Steam System Piping Failure (Case A)

Figure 15-29.13 **Core Reactivity versus Time**
Steam System Piping Failure (Case B)

Figure 15-29.14 **Reactor Power versus Time**
Steam System Piping Failure (Case B)

Figure 15-29.15 **Core Heat Flux versus Time**
Steam System Piping Failure (Case B)

Figure 15-29.16 **RCS Pressure versus Time**
Steam System Piping Failure (Case B)

Figure 15-29.17 **Pressurizer Water Volume versus Time Steam System Piping Failure (Case B)**

Figure 15-29.18 **Core Average Temperature versus Time Steam System Piping Failure (Case B)**

Figure 15-29.19 Reactor Vessel Inlet Temperature versus Time
Steam System Piping Failure (Case B)

Figure 15-29.20 RCS Total Flow versus Time
Steam System Piping Failure (Case B)

Figure 15-29.21 Steam Generator Pressure versus Time
Steam System Piping Failure (Case B)

Figure 15-29.22 Steam Generator Water Mass versus Time
Steam System Piping Failure (Case B)

Figure 15-29.23 **Steam Flow Rate versus Time**
Steam System Piping Failure (Case B)

Figure 15-29.24 **Feedwater Flow Rate versus Time**
Steam System Piping Failure (Case B)

Figure 15-29.25 **Core Boron Concentration versus Time Steam System Piping Failure (Case B)**

Figure 15-29.26 **Reactor Power versus Time**
Steam System Piping Failure (Case C)
Limiting Case for Spectrum of Breaks at 100% Power

Figure 15-29.27 **Core Heat Flux versus Time**
Steam System Piping Failure (Case C)
Limiting Case for Spectrum of Breaks at 100% Power

Figure 15-29.28 **RCS Pressure versus Time**
Steam System Piping Failure (Case C)
Limiting Case for Spectrum of Breaks at 100% Power

Figure 15-29.29 **Core Average Temperature versus Time**
Steam System Piping Failure (Case C)
Limiting Case for Spectrum of Breaks at 100% Power

Figure 15-29.30 **Reactor Vessel Inlet Temperature versus Time**
Steam System Piping Failure (Case C)
Limiting Case for Spectrum of Breaks at 100% Power

Figure 15-29.31 **DNBR versus Time**
Steam System Piping Failure (Case C)
Limiting Case for Spectrum of Breaks at 100% Power

Impact on DCD

There is no impact on the DCD.

Impact on 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.