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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-11272

Subject: MHI's Response to US-APWR DCD RAI No. 787-5882 Revision 3 (SRP 15.01.01-15.01.04)

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. 787-5882 Revision 3 (SRP 15.01.01-15.01.04)". The material in Enclosure 1 provides MHI's response to the NRC's "Request for Additional Information (RAI) 787-5882 Revision 3," dated July 26, 2011.

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

Sincerely,

Y. Ogata

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

Enclosures:

1. MHI's Response to US-APWR DCD RAI No. 787-5882 Revision 3 (SRP 15.01.01-15.01.04) (non-proprietary)

CC: J. A. Ciocco
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ENCLOSURE 1

UAP-HF-11272
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 787-5882 Revision 3
(SRP 15.01.01-15.01.04)

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. 787-5882 REVISION 3**
SRP SECTION:	**15.01.01 - 15.01.04 - DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE**
APPLICATION SECTION:	**15.01.01 - 15.01.04**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.01.01 - 15.01.04-7

The analysis of the inadvertent opening of a steam generator relief or safety valve in DCD 15.1.4 credits the low pressurizer pressure signal to actuate emergency core cooling and emergency feedwater (EFW) isolation, but the sequence of events does not include when the pumps start or when the EFW is isolated. In order for the staff to determine if the mitigating systems are actuated at setpoints with allowance for instrument inaccuracy as required per Item 3 of the SRP 15.1.1-15.1.4 Acceptance Criteria on parameters used in the analytical model, include the start of the pumps and time of EFW isolation in the sequence of events for this analysis.

ANSWER:

As indicated in DCD Subsection 15.1.4.3.2, the analysis credits automatic EFW isolation of the affected steam generator by the low main steam line pressure signal. Table 15.0-4 provides the analytical limit for the low main steam line pressure signal and the assumed signal delay time (3 seconds), while Table 15.0-5 provides the delay associated with the closure of the isolation valves (20 seconds). A total time delay of 30 seconds, which includes 7 seconds margin, is assumed in the analysis.

Table 15.0-4 provides the analytical limit for the low pressurizer pressure ECCS signal and the assumed signal delay time (3 seconds), while Table 15.0-5 provides the time delay between SI pump initiation and when the pumps reach full flow capability (18 seconds). However, as noted in the response to RAI 769-5797 Question 15-26 (MHI letter UAP-HF-11224 dated July 15, 2011), the time provided in Table 15.0-5 already includes the signal delay. (See the response to RAI 769-5797 Question 15-26 for a detailed breakdown of 18 second delay.) Therefore, a total time delay of 20 seconds, which includes an additional 2 seconds margin, is assumed in the analysis.

Table 15.1.4-1 is revised as indicated in the "Impact on DCD" section below to include the time at which the SI pumps start and the time at which the automatic EFW isolation occurs.

Impact on DCD

DCD Table 15.1.4-1 is revised 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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

8/25/2011

US-APWR Design Certification

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RAI NO.:	**NO. 787-5882 REVISION 3**
SRP SECTION:	**15.01.01 - 15.01.04 - DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE**
APPLICATION SECTION:	**15.01.01 - 15.01.04**
DATE OF RAI ISSUE:	**7/26/2011**

QUESTION NO.: 15.01.01 - 15.01.04-8
DCD 15.1.2.2 states that an increase in feedwater flow event initiated at power will cause the reactor to stabilize at a new, higher, power level. This is inconsistent with the supporting analysis in DCD 15.1.2.3, which includes a reactor trip on high-high steam generator water level. Please explain this discrepancy.

ANSWER:

The description in DCD 15.1.2.2 was meant to describe the sequence of events in very general terms. In this event, the increase in reactor power is driven by the increased heat transfer due to the higher feedwater flow rate. The purpose of the statement was to generally imply that the reactor power would increase only until it reaches an equilibrium power consistent with the increased secondary heat transfer (reactor trip is not necessary to terminate the power increase). The reviewer is correct that the reactor will eventually trip due to the increase in SG water level. This is clearly described later in DCD 15.1.2.2 as well as in DCD 15.1.2.3 and shown in the analysis results. MHI did not intend for this sentence to imply that reactor trip did not occur during the analysis.

The same general description about reactor power reaching a new equilibrium power level was used in the sequence of events in DCD 15.1.1.2 and 15.1.3.2 for the decrease in feedwater temperature and increase in steam flow events, respectively. However, in both of those cases, the description also clearly states that no reactor trip is assumed.

In order to remove any potential confusion, the DCD will be revised to simply delete the sentence which states that the reactor will stabilize at a new, higher, power level since the reliance on the high-high steam generator water level is already clearly described later in Section 15.1.2.2, as well as in Section 15.1.2.3.

Impact on DCD

The last sentence of the second paragraph of DCD Section 15.1.2.2 is revised 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.

15.1.2 Increase in Feedwater Flow as a Result of Feedwater System Malfunctions

15.1.2.1 Identification of Causes and Frequency Classification

An increase in the feedwater flow rate to the secondary side of the steam generator will increase the heat transfer from the primary to the secondary side of the steam generator. This will cause a reduction in the reactor coolant temperature at the reactor vessel inlet. In the presence of a negative moderator temperature coefficient (positive moderator density coefficient), the decrease in primary temperature (and associated increase in density) results in a positive reactivity insertion and core power increase.

This transient is caused by a main feedwater regulation valve that is opened fully due to an operator error or a malfunction of the feedwater control system during rated power or part load operation. The feedwater control system is designed such that a single control system failure will affect only one main feedwater regulation valve, and hence, one steam generator.

This event is classified as an anticipated operational occurrence (AOO). Historically, this was classified as a Condition II event of moderate frequency as defined in ANSI N18.2 (Ref. 15.1-1). Event frequency conditions are described in Section 15.0.0.1.

MHI conservatively adopts an additional acceptance criterion to not allow steam generator overfill.

15.1.2.2 Sequence of Events and Systems Operation

The sequence and timing of major events for the increase in feedwater flow event is described in the results section.

An increased water supply rate to the secondary side of the steam generator will increase the heat transfer from the primary to the secondary side of the steam generator, causing a reduction in the reactor coolant temperature at the reactor vessel inlet. This leads to the introduction of cooler (more dense) water into the core, which adds reactivity as a result of the positive moderator density coefficient (negative moderator temperature coefficient), thereby increasing the reactor power. ~~For transients of this type initiated at power, the reactor will stabilize at a new, higher, power level.~~ DCD_15.01.01-15.01.04-8

The decrease in average temperature can also cause rod cluster control assemblies (RCCAs) to withdraw if the reactor is operating in automatic rod control mode, as the system attempts to restore the selected core average temperature input into the automatic rod control system. Although, this RCCA motion could contribute to the overall increase in core power, the analysis of the 10% steam flow increase event in Section 15.1.3 demonstrates there is no difference in the results for the maximum negative temperature coefficient cases with and without automatic rod control. Therefore, only the manual rod control case is presented for this event.

The temperature decrease, and associated density change, leads to a decrease in reactor coolant system (RCS) pressure. The combination of higher core power and lower RCS pressure can lead to a lower departure from nucleate boiling ratio (DNBR).

Table 15.1.4-1
Time Sequence of Events for Inadvertent Opening of a Steam Generator Relief or Safety Valve

Event Description	Time [sec]
Inadvertent opening of one main steam relief or safety valve	0
Pressurizer empties	135
Safety injection actuation (low pressurizer pressure ECCS actuation analytical limit reached)	169
Safety injection pumps start	189
Boron reaches core	240
EFW isolation signal (low main steam line pressure analytical limit reached)	292
Automatic isolation of EFW to faulted SG	322

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