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

June 2, 2008

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-08099

**Subject: Supplemental Information Committed by MHI RAI Response Letter UAP-HF-08070 Related to Defense-in-Depth and Diversity Topical Report**

With this letter, Mitsubishi Heavy Industries, LTD. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Supplemental Information Committed by MHI RAI Response Letter UAP-HF-08070 Related to Defense-in-Depth and Diversity Topical Report," dated April 25, 2008. In the enclosed document, MHI provides supplemental information related to three specific initial responses provided in MHI letter UAP-HF-08070 (RAI-03, RAI-15 and RAI-26) to the NRC's Request for Additional Information on US-APWR Topical Report MUAP-07006-P(R1), "Defense-in-Depth and Diversity," dated March 25, 2008.

As noted in the enclosure, some changes or additional information will be included in future revisions of the "Defense-in-Depth and Diversity" Topical Report (MUAP-07006), "Defense-in-Depth and Diversity Coping Analysis" Technical Report (MUAP-07014), and US-APWR Design Control Document. The revised versions of these reports will be submitted to the NRC separately.

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

Sincerely,

Y. Ogata

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

Enclosure: Supplemental Information Committed by MHI RAI Response Letter UAP-HF-08070 Related to Defense-in-Depth and Diversity Topical Report

CC: L. J. Burkhart
C. K. Paulson

D081
NRO

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

Enclosure

UAP-HF-08099
Docket No. 52-021

# Supplemental Information Committed by MHI RAI Response Letter UAP-HF-08070 Related to Defense-in-Depth and Diversity Topical Report

**June 2008**

## INTRODUCTION

Mitsubishi Heavy Industries (MHI) and the U.S. Nuclear Regulatory Commission (NRC) met on April 23, 2008 to discuss MHI's draft responses to three (References 1 through 3) NRC requests for additional information (RAIs) on MHI Topical Report, MUAP-07006-P(R1), "Defense-in-Depth and Diversity". Following the meeting, MHI formally provided RAI responses in the Reference 4 letter. The RAI responses in that letter committed to provide supplemental information related to the responses to RAI numbers 3, 15, and 26 from the Reference 1 NRC request.

Two supplemental information responses are included in this letter. The MHI responses for both RAI-3 and RAI-15 committed to provide the same table showing the functional diversity of reactor trip functions for each AOO and PA analyzed for the US-APWR. These are combined in a single supplemental information response. The MHI response for RAI-26 committed to provide a detailed evaluation of the feedwater control behavior based on operating plant data to support MHI's previous position that SG water level control could be credited during events in the D3 Coping Analysis. As the supplemental response indicates, the D3 Coping Analysis no longer credits SG water level control during the events and the detailed evaluation is not required to support the analysis and is not provided as originally planned.

## REFERENCES

1 NRC letter dated March 25, 2008, "Mitsubishi Nuclear Energy Systems, Inc. - Request for Additional Information on US-APWR Topical Report MUAP-07006-P, Defense-in-Depth and Diversity" (ML080790297).

2 NRC letter dated April 2, 2008, "Second Request for Additional Information on US-APWR Topical Report MUAP-07006-P, Defense-in-Depth and Diversity" (ML080880164).

3 NRC informal request provided to MHI on April 22, 2008, "Human-Factors Engineering-Related Requests for Additional Information for MHI Topical Report MUAP-07006-P (R1), Defense-in-Depth and Diversity"

4 MHI letter UAP-HF-08070-P(R0) and UAP-HF-08070-NP(R0) (proprietary and non-proprietary versions) dated April 25, 2008, "MHI's Responses to NRC's Requests for Additional Information on Topical Report MUAP-07006-P(R1) Defense-in-Depth and Diversity" (ML081200218-NP)

## Supplemental Information for RAI-03 and RAI-15

As described in the response to RAI-03 and 15 in UAP-HF-08070-P/NP(R0), "MHI's Responses to NRC's Requests for Additional Information on Topical Report UAP-07006-P(R1) Defense-in-Depth and Diversity", MHI credits functional diversity to minimize the potential for CCF in the PSMS. However, functional diversity within the PSMS is not credited in the D3 Coping Analysis. The D3 Coping Analysis conservatively assumes the CCF affects and disables all digital control and protection systems in their entirety, including those trip functions that are functionally diverse. The defense against CCF through functional diversity and separation of functions within the RPS/ESFAS is described in Section 9.1 of MUAP-07006. Typical examples of this functional diversity are shown in Table 9.1-1 of MUAP-07006 for operating plants. Typical examples of this functional diversity for the US-APWR are shown in DCD Table 7.2-5. Table 1 is provided to show the trip parameters in the two separate controller groups of the US-APWR for each AOO and PA. Although this level of functional diversity within the PSMS is not a design requirement and is not credited in the D3 Coping Analysis, the table shows that this additional functional diversity is reflected in the design and is available for each of the events. The reactor trip signals shown in the table are available to provide protection from each transient, and are described in the appropriate event section of DCD Chapter 15.

| Table 1 : Diverse Reactor Trip Parameters in Two Separate Controller Groups by Chapter 15 Event | | | | |
|---|---|---|---|---|
| DCD Section | Title | Group 1 | Group 2 | Remarks |
| 15.1.1 | Decrease in Feedwater Temperature as a Result of Feedwater System Malfunctions | • Over temperature ΔT<br>• Over power ΔT | • High power range neutron flux | |
| 15.1.2 | Increase in Feedwater Flow as a Result of Feedwater System Malfunctions | • Over temperature ΔT<br>• Over power ΔT<br>• High-high steam generator water level | • High power range neutron flux | |
| 15.1.3 | Increase in Steam Flow as a Result of Steam Pressure Regulator Malfunction | • Over temperature ΔT<br>• Over power ΔT | • High power range neutron flux | |
| 15.1.4 | Inadvertent Opening of a Steam Generator Relief or Safety Valve | • Over temperature ΔT<br>• Over power ΔT | • High power range neutron flux<br>• Low pressurizer pressure | |
| 15.1.5 | Steam System Piping Failures Inside and Outside of Containment | • Over temperature ΔT<br>• Over power ΔT | • High power range neutron flux<br>• Low pressurizer pressure | |
| 15.2.1 | Loss of External Load | • High pressurizer water level<br>• Low steam generator water level<br>• Over temperature ΔT | • High pressurizer pressure | |
| 15.2.2 | Turbine Trip | Same as 15.2.1 above | Same as 15.2.1 above | |
| 15.2.3 | Loss of Condenser Vacuum | Same as 15.2.1 above | Same as 15.2.1 above | |
| 15.2.4 | Closure of Main Steam Isolation Valve | Same as 15.2.1 above | Same as 15.2.1 above | |
| 15.2.5 | Steam Pressure Regulator Failure | N/A | N/A | BWR event |

| Table 1 : Diverse Reactor Trip Parameters in Two Separate Controller Groups by Chapter 15 Event | | | | |
|---|---|---|---|---|
| DCD Section | Title | Group 1 | Group 2 | Remarks |
| 15.2.6 | Loss of Non-Emergency AC Power to the Station Auxiliaries | • High pressurizer water level<br>• Low steam generator water level<br>• Over temperature ΔT<br>• Low reactor coolant pump speed | • High pressurizer pressure<br>• Low reactor coolant flow | |
| 15.2.7 | Loss of Normal Feedwater Flow | • High pressurizer water level<br>• Low steam generator water level<br>• Over temperature ΔT | • High pressurizer pressure | |
| 15.2.8 | Feedwater System Pipe Break Inside and Outside Containment | • High pressurizer water level<br>• Low steam generator water level<br>• Over temperature ΔT | • High pressurizer pressure | |
| 15.3.1.1 | Partial Loss of Forced Reactor Coolant Flow | • Low reactor coolant pump speed | • Low reactor coolant flow | |
| 15.3.1.2 | Complete Loss of Forced Reactor Coolant Flow | • Low reactor coolant pump speed | • Low reactor coolant flow | |
| 15.3.2 | Flow Controller Malfunctions | N/A | N/A | BWR event |
| 15.3.3 | Reactor Coolant Pump Rotor Seizure | • Low reactor coolant pump speed | • Low reactor coolant flow | |
| 15.3.4 | Reactor Coolant Pump Shaft Break | Same as 15.3.3 above | Same as 15.3.3 above | |
| 15.4.1 | Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition | • High power range neutron flux rate<br>• High source range neutron flux<br>• High intermediate range neutron flux | • High power range neutron flux | |

Table 1 : Diverse Reactor Trip Parameters in Two Separate Controller Groups by Chapter 15 Event

| DCD Section | Title | Group 1 | Group 2 | Remarks |
|---|---|---|---|---|
| 15.4.2 | Uncontrolled Control Rod Assembly Withdrawal at Power | • High power range neutron flux rate<br>• Over temperature ΔT<br>• Over power ΔT<br>• High pressurizer water level | • High power range neutron flux<br>• High pressurizer pressure | |
| 15.4.3 | Control Rod Misoperation (System Malfunction or Operator Error) | • High power range neutron flux rate<br>• High intermediate range neutron flux<br>• Over temperature ΔT<br>• Over power ΔT | • High power range neutron flux<br>• Low pressurizer pressure | |
| 15.4.4 | Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature | N/A | N/A | N-1 loop operation is not permitted for the US-APWR. |
| 15.4.5 | Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate | N/A | N/A | BWR event |
| 15.4.6 | Inadvertent Decrease in Boron Concentration in the Reactor Coolant System | • Over temperature ΔT | • High power range neutron flux | |
| 15.4.7 | Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position | N/A | N/A | The event does not credit any reactor trip. |
| 15.4.8 | Spectrum of Rod Ejection Accidents | • High power range neutron flux rate | • High power range neutron flux | |
| 15.4.9 | Spectrum of Rod Drop Accidents in a BWR | N/A | N/A | BWR event |

Table 1 : Diverse Reactor Trip Parameters in Two Separate Controller Groups by Chapter 15 Event

| DCD Section | Title | Group 1 | Group 2 | Remarks |
|---|---|---|---|---|
| 15.5.1 | Inadvertent Operation of Emergency Core Cooling System that Increases Reactor Coolant Inventory | N/A | N/A | This event is not applicable to the US-APWR because the safety injection pumps are not able to inject water into the reactor coolant system at normal, at-power operating pressures. |
| 15.5.2 | Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory | • High pressurizer water level | • High pressurizer pressure | |
| 15.6.1 | Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve | • Over temperature ΔT | • Low pressurizer pressure | |
| 15.6.2 | Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment | N/A | N/A | The event does not need any reactor trip. |
| 15.6.3 | Radiological Consequences of Steam Generator Tube Failure | • Over temperature ΔT<br>• High-high steam generator water level | • Low pressurizer pressure | |
| 15.6.4 | Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) | N/A | N/A | BWR event |

| Table 1 : Diverse Reactor Trip Parameters in Two Separate Controller Groups by Chapter 15 Event | | | | |
|---|---|---|---|---|
| DCD Section | Title | Group 1 | Group 2 | Remarks |
| 15.6.5 | Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary | • Over temperature ΔT | • Low pressurizer pressure<br>• Low reactor coolant flow | |

## Supplemental Information for RAI-26

The response to RAI-26 in UAP-HF-08070-P/NP(R0) describes MHI's initial position that SG water level control is assumed to function correctly for all AOOs and PAs, with concurrent CCF in the RPS/ESFAS, as described in the current version of Technical Report MUAP-07014.

MHI continues to believe that a thorough evaluation of plant performance data would support the initial position that a fail-as-is condition in the SG water level control function is self-announcing, hence correctable prior to any AOO or PA. Therefore, it would be reasonable to credit the continued operation of the SG water level control function for all events in the D3 Coping Analysis.

However, MHI has now determined that for all events, there is sufficient time for manual action even with the assumption of a concurrent CCF that results in a fail as-is condition in the SG water level control function. Therefore, MHI no longer plans to conduct that evaluation of plant performance data to support this premise. Instead, MHI will revise the D3 Coping Analysis Report (MUAP-07014) so that continued SG water level control is not credited for any event.