MITSUBISHI HEAVY INDUSTRIES, LTD.
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June 16, 2009

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

Attention: Mr. Jeffery A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-09322

Subject: MHI's Amended Response to US-APWR DCD RAI No. 316-2296 Revision 0

Reference: 1) "Request for Additional Information No. 316-2296 Revision 0, SRP Section: 04.06 – Functional Design of Control Rod Drive System
2) "MHI's Response to US-APWR DCD RAI No. 316," MHI Ref. UAP-HF-09259

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Amended Response to US-APWR DCD RAI" This amended response is submitted to add discussion of the combined performance of two reactivity control systems which demonstrates compliance with GDCs 27 and 28.

Enclosed is the amended response to Question 2296-4.6.-8. MHI replaces the previous letter reference 2) with this amended response letter.

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

Sincerely,

Y. Ogata

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

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NRO

Enclosures:

1. "Amended Response to Request for Additional Information No. 316-2296, Revision 0"

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

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Enclosure 1

UAP-HF-09322
Docket No. 52-021

Amended Response to Request for Additional Information No. 316-2296,Revision 0

June 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/16/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 316-2296 REVISION 0**
SRP SECTION:	**04.06 – FUNCTIONAL DESIGN OF CONTROL ROD DRIVE SYSTEM**
APPLICATION SECTION:	**4.6**
DATE OF RAI ISSUE:	**4/02/2009**

QUESTION NO.: 2296-4.6.-8

In Section 4.6.5, the applicant restates the description given in FSAR Section 4.6.4 that only a limited number of postulated events assume the availability of two reactivity control systems to prevent or mitigate the accident such as the SLB and LOCA. The applicant did not discuss or provide a reference to demonstrate that the combined performance of the two reactivity control systems is in compliance with GDCs 27 and 28. Provide a discussion or a reference of the analysis that confirms the combined performance of the two reactivity control systems is in compliance with GDCs 27 and 28.

ANSWER:

DCD Sections 4.6.4 and 4.6.5 state that only the main steam line break (SLB) and loss of coolant accident (LOCA) assume the availability of two reactivity control systems in the accident analysis of DCD Chapter 15.

The SLB analysis is described in detail in DCD Section 15.1.5. Three different SLB cases are considered and each case credits the reactivity insertion of both the rod cluster control assemblies (RCCAs) due to a reactor trip (or initial shutdown margin) and the injection of borated water by the emergency core cooling system (ECCS). The conservative assumptions used in the analysis concerning margin for a stuck rod, shutdown margin, boron concentration, the number of ECCS pumps available to inject borated water, and the transport time for borated water to reach the core are described in DCD Section 15.1.5.3.2. The results of the SLB analysis, described in DCD Section 15.1.5.3.3, show that with the two reactivity control systems, the capability to cool the core and the integrity of the reactor coolant pressure boundary are maintained during the entire transient. Therefore, the SLB analysis is in compliance with GDCs 27 and 28.

For the large break LOCA safety analysis, the reactor core fission power can be suppressed by large negative reactivity feedback due to the coolant density decrease following the accident occurrence, although the control rod insertion is not taken into account conservatively. In addition, the borated water from the accumulators and SIS maintains the core subcritical throughout the accident. The chapter 15.6.5 in US-APWR DCD demonstrates the US-APWR is able to result in safe shutdown.

For the small break LOCA safety analysis, the applied model takes credit for negative reactivity insertion due to the control rod insertion (SCRAM), which is activated by the pressurizer pressure low signal. Therefore, the core fission power can be suppressed by SCRAM in addition to the negative coolant density reactivity following the accident occurs. These negative reactivity insertions maintain the core subcritical throughout the accident, and the chapter 15.6.5 shows that the US-APWR is able to result in safe shutdown after SBLOCAs.

Impact on DCD

Editorial: After the second paragraph in the subsection 4.6.5 on page 4.6-3 of the DCD Revision 1, the following paragraph will be added.

With the exception of large break LOCA, no credit is taken for reactivity control systems other than reactor trip to maintain the reactor core subcritical. For large break LOCA, reactor trip is not assumed following the event occurs, and the core power can be suppressed only by the negative reactivity insertion due to increase in core void fraction during the initial phase of the event. Then, the borated water from the accumulators and SIS provides the negative reactivity to maintain the shutdown margin during the refilling of the core. Control rod insertion is credited to provide additional shutdown margin during long-term cooling.

For small break LOCA, the reactor core fission power can be suppressed by the negative reactivity due to the reactor trip in addition to the core void increase, and the core is maintained under the subcritical state throughout the event. Borated water from the accumulators and HHIS is credited to provide addition shutdown margin during long-term cooling.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.