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

July 3, 2009

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-09343

Subject: MHI's Response to US-APWR DCD RAI No. 305-2331 Revision 2

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. 305-2331 Revision 2". The material in Enclosure 1 provides MHI's response to the NRC's "Request for Additional Information (RAI) 305-2331 Revision 2," dated May 4, 2009.

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. 305-2331 Revision 2 (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

D081
NRO

ENCLOSURE 1

UAP-HF-09343
Docket No. 52-021

MHI's Response to US-APWR DCD RAI No. 305-2331 Revision 2

July 2009

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 305-2331 REVISION 2**
SRP SECTION:	**15.02.08 – FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)**
APPLICATION SECTION:	**15.2.8**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2.8-1

In DCD Section 15.2.8, Feedwater System Pipe Break, the applicant states that only the case without offsite power available is presented, and that the RCPs remain running until the ECCS signal is reached. LOOP and RCP trips are assumed to be concurrent with the turbine trip. This sequence of events as presented is inconsistent with the previous approach (DCD Section 15.0.0.7) that assumes the turbine trip and immediate reactor trip initiate a 3-second delayed LOOP that results in a delayed coast down of the RCPs until the rods and sufficient negative reactivity have been inserted in the core to cause the DNBR to increase; therefore minimum DNBR always occurs before the initiation of a LOOP. This apparent inconsistency in methodology prompts the staff to question MHI to explain why the turbine/reactor trip, LOOP, RCP coast down sequence should differ between various transients. Isn't this a generic sequence of events?

ANSWER:

The generic sequence of events for LOOP is described in DCD Subsection 15.0.0.7. The generic sequence of events includes a 3 second delay between the turbine trip and LOOP. However, MHI uses event-specific assumptions that are described in the DCD subsection for each event and may differ from this generic sequence of events in order to provide more conservative results. For the event in DCD Subsection 15.2.8, Feedwater System Pipe Break, the RCP coastdown following a LOOP is assumed to occur at the same time as the turbine trip (turbine trip is assumed to occur at the same time as reactor trip) as shown in DCD Table 15.2.8-1. This assumption is conservative for the purpose of calculating peak RCS pressure and peak pressurizer water level. However, for the purpose of calculating DNBR for the feedwater system pipe break, assuming no delay between the turbine trip and LOOP is overly conservative. Instead a delay of 3 seconds between the turbine trip and LOOP is assumed, as described in the generic sequence of events in DCD Subsection 15.0.0.7. The DNBR transient curve for the feedwater system break is provided in the response to Question 15.2.8-3 of this RAI. The basis for the LOOP timing assumptions is described in detail for the sensitivity analysis in the response to Question 15.0.0-3 of RAI 297-2287.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 305-2331 REVISION 2**
SRP SECTION:	**15.02.08 – FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)**
APPLICATION SECTION:	**15.2.8**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2.8-2

In FSAR Section 15.2.8, "Feedwater Piping Breaks Inside and Outside Containment", it is indicated that the RCPs coast down at 68 seconds into the event (see Figure 15.2.8-6). Discuss the basis for this assumption.

ANSWER:

DCD Table 15.2.8-1 indicates that the low steam generator water level reactor trip analytical limit is reached at 66 seconds (note that the time values in the table are rounded to the nearest integer). The delay time for this reactor trip is 1.8 seconds, as specified in DCD Table 15.0-4. The reactor trip at 68 seconds causes a turbine trip. As described in DCD Subsection 15.2.8.2 and the response to Question 15.2.8-1 of this RAI, the analysis assumes a loss of offsite power (LOOP) at the time of turbine trip. As a result of the LOOP, the reactor coolant pumps (RCPs) begin to coast down at 68 seconds, as indicated in DCD Table 15.2.8-1. The RCP coastdown can also be seen in DCD Figure 15.2.8-6 which shows RCS total flow.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/03/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 305-2331 REVISION 2**
SRP SECTION:	**15.02.08 – FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)**
APPLICATION SECTION:	**15.2.8**
DATE OF RAI ISSUE:	**5/04/2009**

QUESTION NO.: 15.2.8-3

In FSAR Section 15.2.8, "Feedwater Piping Breaks Inside and Outside Containment", provide the transient curve for DNBR verses time and the calculated amount of fuel failure based on the criterion that all fuel pins with MDNBR below the DNBR limit are assumed to fail.

ANSWER:

The transient curve for DNBR versus time for the analysis in DCD Subsection 15.2.8 is shown below in Figure 15.2.8-3.1. The DNBR case shown here uses slightly different initial conditions than the RCS pressure analysis case described in the DCD. The nominal conditions (power, temperature, and pressure) are assumed as the initial conditions for the DNBR case which is analyzed using the revised thermal design procedure (RTDP). This figure confirms that this event is not a DNB limiting event as described in DCD Subsection 15.2.8.3. Since the minimum DNBR (MDNBR) remains above the DNBR limit during the transient, the calculated amount of fuel failure, based on the criterion that all fuel pins with MDNBR below the DNBR limit are assumed to fail, is 0%.

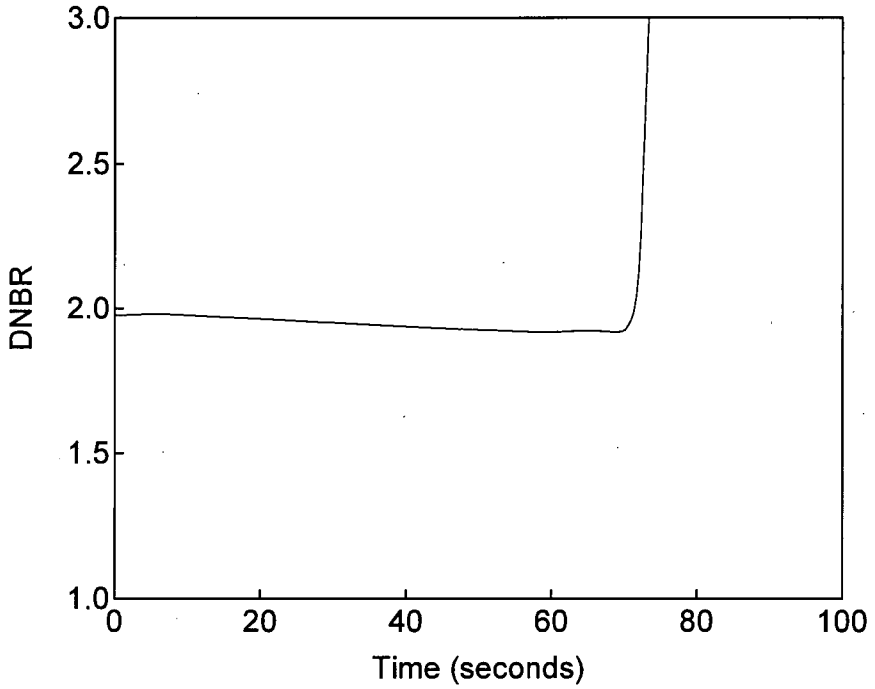

Figure 15.2.8-3.1 DNBR versus Time Feedwater System Pipe Break

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.