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TOKYO, JAPAN

June 26, 2012

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-12178

Subject: MHI's Amended Response to US-APWR DCD RAI No. 310-2346 Revision 2 (SRP 15.04.03)

References: 1) "Request for Additional Information No. 310-2346 Revision 2, SRP Section: 15.04.03 – Control Rod Misoperation (System Malfunction or Operator Error), Application Section: 15.4.3," dated May 4, 2009.
2) Letter MHI Ref: UAP-HF-09345 from Y. Ogata to U.S. NRC, "MHI's Response to US-APWR DCD RAI No. 310-2346 Revision 2", dated July 3, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Amended Response to Request for Additional Information No. 310-2346 Revision 2 (SRP 15.04.03)".

Enclosed is the amended response to Question 15.4.3-9 contained within Reference 1. Responses to other questions contained within Reference 1 have previously been submitted to the NRC in Reference 2.

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

Y. Ogata

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

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NRO

Enclosure:

1. Amended Response to Request for Additional Information No. 310-2346 Revision 2 (SRP 15.04.03)

CC: J. A. Ciocco
J. Tapia

Contact Information

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Enclosure 1

UAP-HF-12178
Docket No. 52-021

Amended Response to Request for Additional Information No.310-2346 Revision 2 (SRP 15.04.03)

June 2012

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/26/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 310-2346 REVISION 2

SRP SECTION: 15.04.03 – CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

APPLICATION SECTION: 15.4.3

DATE OF RAI ISSUE: 5/04/2009

QUESTION NO.: 15.4.3-9

What is the fuel centerline temperature for the withdrawal of a single RCCA?

ANSWER:

DCD Subsection 15.4.3.3.3.3 states that using a detailed analysis method, the minimum DNBR is above the 95/95 limit and no fuel failures are predicted. In the response to RAI 904-6324 Question 15.04.03-13, submitted by MHI letter UAP-HF-12101 dated April 20, 2012, MHI provided the detailed analysis of the uncontrolled withdrawal of a single RCCA event. The detailed analysis was performed from several initial power levels. In all cases, the resulting minimum DNBR is above the safety analysis limit and therefore no DNB fuel failure occurs. Since the DNB remains above the safety analysis limit and the increase in nuclear power and F_Q is small, a large increase in fuel centerline temperature would not occur. Therefore, the maximum fuel centerline temperature for this analysis is well below the fuel pellet melting temperature. The fuel pellet melting temperature is a function of burnup and is calculated per the methodology described in DCD Subsection 4.2.1.2.1.

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.

Impact on Technical/Topical Report

There is no impact Technical/Topical Report.

This completes MHI's response to the NRC's question.