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November 6, 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-12282

Subject: MHI's Amended Response to US-APWR DCD RAI No. 374-2446 Revision 0 (SRP 03.09.05)

Reference: 1) "Request for Additional Information No. 374-2446 Revision 0, SRP Section 03.09.05 – Reactor Pressure Vessel Internals, Application Section: DCD, Tier 2 – Section 3.9.5", dated May 21, 2009

2) "MHI's Response to US-APWR DCD RAI No. 374-2446", UAP-HF-09387, dated July 17, 2009.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Amended Response to Request for Additional Information No. 374-2446 Revision 0 (Question 03.09.05-22)"

Enclosed is the amended response to Question 03.09.05-22 contained within Reference 1. MHI amends the previous response to the question transmitted in Reference 2 to include changes to the DCD based on discussions with the NRC staff.

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, 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,
Director- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure:

1. Amended Response to Request for Additional Information No. 374-2446 Revision 0 (Question 03.09.05-22)

D081
NRO

CC: J. A. Ciocco
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Enclosure 1

UAP-HF-12282
Docket No. 52-021

Amended Response to Request for Additional Information No. 374-2446 Revision 0 (Question 03.09.05-22)

November 2012

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

11/06/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2466 REVISION 0**
SRP SECTION:	**03.09.05 – REACTOR PRESSURE VESSEL INTERNALS**
APPLICATION SECTION:	**3.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-22:

The applicant states in DCD Tier 2, Subsection 3.9.5.3 that the rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures (CSSs) and internal structure follow those in Section III, Subsection NG of the ASME Boiler and Pressure and Vessel Code, 2001 Edition up to and including 2003 Addenda. Additional codes, standards, regulations, and guidelines from the NRC and the Utility Requirements Document are adhered to, and are listed in the Owner's design specification. However, these additional design codes, code cases, and acceptance criteria are not identified in the DCD. Section 3.9.5 of the SRP states that if other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified.

The applicant is requested to provide a list and justification of the applicable codes, standards, regulations, and guidelines, for the design of US-APWR CSSs and reactor internals, if different from ASME III, Subsection NG requirements. The requested information is needed to assure conformance with GDC-1. Revise the DCD to include the requested information.

ANSWER (Revision 1):

As described in DCD Subsection 3.9.5.3, requirements for materials, design fabrication examination and preparation of reports for manufacture and installation of the core support structures and internal structure follow Subsection NG of the ASME Boiler and Pressure and Vessel Code. Note that the portions of external code sections listed below, which are referred from Subsection NG, are also utilized.

- ASME Boiler and Pressure and Vessel Code Section II Material Specification
- ASME Boiler and Pressure and Vessel Code Section V Non Destructive Examination
- ASME Boiler and Pressure and Vessel Code Section IX Welding and Brazing Qualification

Additionally, ASME Boiler and Pressure and Vessel Code Section XI is applied for the pre-service and in-service inspections as identified in DCD Subsection 3.9.5.3.12.

Impact on DCD

DCD Tier 2 Subsection 3.9.5.3 will be revised as shown in the attachment.

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 Topical Report / Technical Report

There is no impact on the Topical Report / Technical Report.

3.9.5.2.3 Interface Load and Displacement Limits

There are certain load and displacement limits for the reactor internals that affect the safety and operability of the interface components. These limits are summarized in Table 3.9-2.

3.9.5.3 Design Bases

The rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures and internal structure follow those in Section III, Subsection NG of the ASME Boiler and Pressure and Vessel Code 2001 Edition up to and including 2003 Addenda (Reference 3.9-1).

~~Additional codes, standards, regulations, and guidelines from the NRC and the Utility Requirements Document are adhered to and are listed in the Owners design specification.~~ DCD_03.09.05-22 S01

The design basis for the operability of the US-APWR internals are listed below and discussed in detail under the following sections:

- Safety analysis
- Thermal-hydraulic performance
- Core loading pattern
- Environmental conditions including irradiation
- RCS transients
- Interface design requirements

3.9.5.3.1 Safety Analysis Design Basis

The safety analysis design requirements and limits for the US-APWR internals are as follows:

- Mal-distribution of flow to the core should be limited so as not impact core safety limits in Chapter 15.
- RCCA drop times or insertion during normal service conditions should not be adversely affected.
- RCCA are to be inserted without impediment after an unanticipated accident, or a seismic and postulated LOCA event.
- There should be no impediment of the reactor internals to the emergency core cooling flow after a seismic and postulated LOCA event.
- The impact load on the RV bottom head from a postulated core drop event should not adversely affect the integrity of the RV bottom head.