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TOKYO, JAPAN

September 29, 2010

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-10259

Subject: MHI's Response to US-APWR DCD RAI No. 614-4853

References: 1) "Request for Additional Information No. 614-4853 Revision 0, SRP Section: 03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components, Application Section: DCD, Tier 2 – Section 3.9.2," dated 8/13/2010.

2) UAP-HF-10254, "MHI's Responses to US-APWR DCD RAI No. 614-4853", dated 9/16/2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to Request for Additional Information No. 614-4853 Revision 0."

Enclosed are the responses to questions 88-90 of the RAI (Reference 1) with a 45-day response time. The responses to questions 85-87 of this RAI with a 30-day response time have been issued by Reference (2). The remaining response to question 91 of this RAI has a 60-day response time, as agreed between the NRC and MHI, and will be issued at a later date by a separate transmittal.

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. Response to Request for Additional Information No. 614-4853, Revision 0

D081
NRO

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

Contact Information

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Enclosure

UAP-HF-10259
Docket No. 52-021

Response to Request for Additional Information No. 614-4853,
Revision 0

(Questions 03.09.02-88, 89 and 90)

September, 2010

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

09/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 614-4853**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2**
DATE OF RAI ISSUE:	**08/13/2010**

QUESTION NO.: RAI 03.09.02-88

In the response to the US-APWR DCD RAI 498-3782, question 03.09.02-65, MHI provided tables and figures explaining the differences between the J-APWR, which is in operation in Japan, and the US-APWR. The effects of these differences on the FIV of the reactor internals were also clarified. The only remaining concern related to this RAI is the effect of increasing the weight of the fuel assembly and the neutron reflector on the stresses at the junction between the core barrel and the lower core support plate and also on the vibration of the fuel assemblies.

Therefore, the applicant is requested to confirm that the effect of this increase in weight is considered in the analysis of the fuel assembly and the core barrel.

Reference: MHI's Response to US-APWR DCD RAI No. 498-3782; MHI Ref: UAP-HF10031; dated February 3, 2010; ML100470583.

ANSWER:

The weight of the neutron reflector and the fuel assembly are directly simulated in the mass vibration analysis model. The results of the vibration analysis are used as input to the stress analysis. The weights are also used for the dead weight loads. Therefore, the stress analysis results take into account the increase in weight of the neutron reflector and the fuel assembly both in vibration loads and dead weights. The results (i.e., the primary stress on the core barrel) have been reported in Table 10-1 of Reference (1). The stress at the core barrel / lower core support plate junction is less than half the allowable limit in design conditions and all operating conditions.

The weight increase is also considered in the vibration analysis model of the fuel assembly. Please refer to Reference (2) for further information about the analysis model and its results. In addition, significant flow-induced vibration was not observed in the hydraulic test with a full-scaled mockup fuel assembly.

As discussed above, there is no structural integrity concern regarding the weight increase of the neutron reflector and the fuel assembly.

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.

Reference (1) :MUAP-09004-P(R0) "Summary of Stress Analysis for the US-APWR Core Support Structures"

Reference (2) : MUAP-08007-P (R1) "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads"

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

09/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 614-4853**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2**
DATE OF RAI ISSUE:	**08/13/2010**

QUESTION NO.: RAI 03.09.02-89

In the response to the US-APWR DCD RAI 498-3782, question 03.09.02-71, the applicant confirmed how the dynamic analysis of the reactor internals was benchmarked by means of the scale model test, and whether the dynamic analysis was performed on the size and flow conditions of the small scale model or the full-scale J-APWR. The applicant's response explains the source of confusion described in the question, that is, although the analysis model and the measurements were performed on the 1/5 scale model, the comparison of the results are made after the results were scaled to the full size reactor.

In order to eliminate this source of confusion, the applicant is requested to modify the Technical Report MUAP-07023-P(R1) to include the information provided in the response to RAI 03.09.02-71.

Reference: MHI's Response to US-APWR DCD RAI No. 498-3782; MHI Ref: UAP-HF10031; dated February 3, 2010; ML100470583.

ANSWER:

As requested, the technical report MUAP-07023-P (R1) will be modified to include the information provided in the response to RAI 03.09.02-71.

Impact on DCD

1. DCD document
The revision to the technical report MUAP-07023-P will be reflected in Reference 1.5-2 in DCD Subsection 1.5.10 and Reference 3.9-24 in DCD Subsection 3.9.10.

2. Technical Report MUAP-07023-P

The following will be inserted after Section 7.0 as an additional section.

8.0 INFORMATION FOR FIV BENCH-MARK ANALYSIS

The following data is applied to the bench-mark analysis to verify the analysis methodology including the FE modeling and forcing functions. The same methodology is applied to the US-APWR design analysis.

- The natural frequencies of the 1/5 scale J-APWR reactor internals are shown in Table 6-1. Note that these data have been scaled up to actual plant size by the scaling law shown in Table 3-1.
- The damping ratio in water shown in Table 6-1 is directly applied to the bench mark analysis as the best estimated value (For the US-APWR, 1% damping ratio is applied as the design value based on R.G. 1.20).
- The measured vibration (rms) responses under nominal flow rate are shown in Table 6-2 and 6-3. Note that these data has been scaled up to actual plant size by the scaling law in Table 3-1.

The bench-mark analysis is performed with a 1/5 scale model of the J-APWR configuration. The material properties are determined at room temperature as the test conditions. The flow velocities are also determined at nominal test flow conditions. To compare with test results, the analysis results of natural frequencies and vibration responses are scaled to actual plant size in the same manner as the test data shown in Table 3-1.

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

09/29/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 614-4853**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2**
DATE OF RAI ISSUE:	**08/13/2010**

QUESTION NO.: RAI 03.09.02-90

In the response to the US-APWR DCD RAI 498-3782, Question 03.09.02-75, the applicant explained in detail why, despite the existing high degree of uncertainty, a margin of safety of 30 percent is considered conservative for the high cycle fatigue analysis as indicated in Table 3.3.3-4 of the revised vibration assessment report MUAP07027-P-R1. The staff reviewed the applicant's response and found it acceptable because the margin of safety already covers conservatively estimated bias errors and uncertainties associated with determining the loading functions due to cross flow and RCP pulsation. However, this information should be included in MHI report MUAP07027-P-R1.

Therefore, the applicant is requested to indicate clearly in the Technical Report MUAP07027-P(R1) that the safety margin of 0.3 covers all the uncertainties and bias errors which are associated with determining the loading functions.

Reference: MHI's Response to US-APWR DCD RAI No. 498-3782; MHI Ref: UAP-HF10031; dated February 3, 2010; ML100470583.

ANSWER:

As requested, the technical report MUAP-07027-P (R1) will be modified to include the information provided in the response to RAI 03.09.02-75, as shown in the attachment.

Impact on DCD

1. DCD document

The revision to the technical report MUAP-07027-P will be reflected in Reference 3.9-22 in DCD Subsection 3.9.10 References.

2. Technical Report MUAP-07023-P

The first paragraph of 'b. Evaluation Results' in '3.3.3.2 (2) High Cycle Fatigue', on Page 83, will be replaced with the following.

Through the results of the high cycle fatigue analysis for the US-APWR reactor internals, minimum margins of safety were predicted for the components in the upper plenum, the RCC guide tube (GT), the upper support column (USC), and the top slotted column (TSC). Both cross flow and RCP pulsation were taken into account in these results. MHI has verified that these results are acceptable based on the following considerations:

1. The uncertainty of the analysis is determined by the ratio to the best estimated value without bias errors if identified.
2. The alternating stresses due to the RCP pulsation have large uncertainty, but the absolute values of these are lower than those due to the cross flow by one order of magnitude.
3. The RCP pulsation loads include a conservative bias by neglecting the acoustic damping due to structural flexibility. Because this effect is also the main part of the uncertainty in the acoustic resonance analysis, the magnitude of bias error is approximately the same as the uncertainty (by a factor of 5). Therefore, the analysis results due to RCP pulsation may be 10 times larger than the actual values, but not smaller.
4. The cross flow loads on the upper and lower plenum structures are determined with peak cross flow velocity along the entire length of structures. The bias due to neglecting the cross flow distribution is approximately a factor of 2, which is comparable to the assumed uncertainty in the flow-induced loads.

From the above discussions, the minimum margin of safety 0.3 for the upper plenum structures due to cross flow loads includes a conservative bias of approximately 2 due to non-uniform cross flow distribution. Because this bias is comparable to the assumed uncertainty in the flow-induced loads (a factor of 2), the margin of safety 0.3 is considered acceptable.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

This completes MHI's responses to NRC's questions.