MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

February 3, 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-10031

Subject: MHI's Response to US-APWR DCD RAI No. 498-3782

References: 1) "Request for Additional Information No. 498-3782 Revision 0, SRP Section: 03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components, Application Section: 3.9.2," dated 12/1/2009.

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. 498-3782 Revision 0."

Enclosed are the responses to questions 61, 62, 64 through 66, 68 through 71, 75, 80, 82 and 84 of the RAI (Reference 1). The responses to questions 59, 60, 63, 67, 72 through 74, 76 through 79, 81 and 83 of this RAI had already provided as the 45-day response by MHI transmittal UAP-HF-10008.

As indicated in the enclosed materials, this submittal contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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.

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Response to Request for Additional Information No. 498-3782, Revision 0, February 2010 (60-day response, Proprietary Version)

3. Response to Request for Additional Information No. 498-3782, Revision 0, February 2010 (60-day response, Non-Proprietary Version)

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

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-10031

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 498-3782, Revision 0 (60-day response)", dated February 2010, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential are as follows:

   A. They include the know-how and outputs of tests or analyses which required significant cost to MHI. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without the costs or risks associated with the design of new systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI and the Licensors in the U.S. nuclear plant market.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 3rd day of February 2010.

Y. Ogata

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

Enclosure 3

UAP-HF-10031
Docket No. 52-021

Response to Request for Additional Information No. 498-3782, Revision 0 (60-day response)

February 2010
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-61:

In its review of the MHI response to RAI 3.9.2-15 (#205-1584, dated 4/30/2009, ML091240113, MHI Ref: UAP-HF-09184) the staff noted that the applicant provided a generic answer to the question regarding the methodology for response spectrum broadening and smoothing and the clarification for "filling the valleys between all peaks". The applicant indicated the "filling valleys approach is used selectively" and the word "all" will be deleted from the DCD. The staff finds that although the applicant's response may resolve the staff's concerns in question 3.7.3-05 of RAI 213-1951, it did not clearly answer the staff's concerns in RAI 3.9.2-15. Specifically, it is not clear whether the ISRS presented in Fig. 3.7.2-13 of the DCD and Fig. 8.1, 8.2, 8.3 of MHI technical report MUAP-08005 is representative of broadening "all" valleys or for just "selected" valleys. If they do not, will these ISRS need to be changed in Revision 2 of the DCD. Or, does the COL Applicant need to know which ISRS are the "selected" valleys before they are compared with the site-specific ISRS. Therefore, the applicant is requested to provide information that addresses issues raised in the above evaluation. In addition, a confirmatory action will be needed to assure that as stated by the applicant in its response the necessary information would be included in Revision 2 of the DCD.

ANSWER:

Figure 3.7.2-12 in current version of DCD (Revision 2) is an example developed using the procedure shown in Figure 3.7.2-11 and therefore, does not include the "selected" valleys. ISRS which include the "selected valleys" will be included in the upcoming revisions of MUAP-08005, "Dynamic analysis of the Coupled RCL-R/B-PCCV-CIS Lumped Mass Stick Model", and MUAP-08002, "Enhanced Information for PS/B Design".

In order to address the NRC concerns regarding the out-of-plane response of walls and slabs, the coupled lumped mass stick models of the R/B complex are enhanced to incorporate single degree of freedom (SDOF) models representing the out-of-plane response of flexible slabs and walls. The SDOF models are developed based on the results of the modal analyses of detailed FE models, which are extracted from the detailed FE model of the R/B complex. For the PS/Bs, the modeling approach which previously used a dynamic lumped mass stick model is revised to utilize a finite element (FE) model of the bounding PS/B configuration which directly accounts for local floor, wall, and basemat flexibilities. An technical report scheduled to be issued in February 2010 will provide the detailed descriptions regarding these modeling and analysis enhancements, which will be used as a basis for the re-runs of the R/B complex and PS/B dynamic analyses, and will also serve as the basis for the revised reports MUAP-08005 and MUAP-08002. The two revised reports (MUAP-08005 and MUAP-08002) will present the re-run results of SSI analyses and provide a revised set of ISRS for structural design and equipment qualification. Both the broadened and the unbroadened ISRS will be included in the two revised reports, and therefore the selected valleys which have been filled and broadened ISRS will be available. The COL Applicant is expected to compare the site-specific ISRS to the standard plant design (broadened and selectively filled) ISRS as per COL Item 3.7(23). The new set of ISRS will be incorporated into the next revision of DCD.

Impact on DCD

A future DCD revision will incorporate the new set of ISRS shown in the two revised Technical Reports MUAP-08002 and MUAP-08005 cited above.

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

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 498-3782 REVISION 0

SRP SECTION: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components

APPLICATION SECTION: 3.9.2

DATE OF RAI ISSUE: 12/01/2009

QUESTION NO. RAI 03.09.02-62:

In response to this RAI 3.9.2-17 (#205-1584, dated 4/30/2009, ML091240113, MHI Ref: UAP-HF-09184) the applicant stated that in APWR DCD Tier 2, Revision 2, consideration of the effects of wall and floor slab flexibility on seismic anchor motions will be addressed. However, the applicant did not present any data to demonstrate that its approach is conservative. Therefore, the applicant's response to this question is not acceptable. The applicant is requested to provide data and rationale that shows that the approach used is conservative.

ANSWER:

MHI's response to this question is included in MHI's response to Question RAI-03.09.02-61. As stated in that response, an Technical Report scheduled to be issued in February 2010 will address the enhanced modeling and analysis approaches that are implemented in order to address the effects of local wall and floor slab flexibility.

Impact on DCD

A new set of ISRS will be incorporated into a future DCD revision.

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

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 498-3782 REVISION 0

SRP SECTION: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components

APPLICATION SECTION: 3.9.2

DATE OF RAI ISSUE: 12/01/2009

QUESTION NO. RAI 03.09.02-64:

The applicant is requested to revise the DCD to include and to refer to information provided in the response to RAI 3.9.2-20 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), concerning the similarity between the steam delivery system of the US-APWR and existing plants which have been in operation for more than 20 years. Provide specific references from existing operating reactors.

ANSWER:

The DCD will be revised to include the information in the response to RAI 3.9.2-20 as requested.

MHI confirmed the similarity between the steam delivery system of the US-APWR and existing plants, which have been in operation for about 20 years, as follows.

For example, vortical frequencies of acoustic vibration are generally proportional to the ratio of the main steam supply system (MSS) steam delivery velocity to sonic speed with Strouhal number. Both the existing A-plant in the USA and US-APWR have almost the same value of velocity ratio for each vertical frequency. Additionally, the difference between the steam velocities of existing A-Plant in the USA and that of US-APWR is sufficiently small.

Plant Name	A-Plant in the USA Operation Start on 1987 (Up-rating Approbation on 1993)	US-APWR
1. Main steam pipe		
Steam pressure at rated power operation	965 psig (SG outlet)	972.0 psig

Plant Name	A-Plant in the USA Operation Start on 1987 (Up-rating Approbation on 1993)	US-APWR
Steam temperature at rated power operation	542 degree F (SG outlet)	541.2 degree F
Steam flow rate at rated power operation	3,980,000 lb/h	5,050,000 lb/h
Bore diameter	SG to CV bore dia. NPS 26 in. CV to Safety Valve bore dia. NPS 28 in. Safety Valve to MSIV bore dia. 29.5 in.	NPS 32 in.
Flow velocity	SG to CV 161.1ft/s CV to MSIV 162.1ft/s	~145 ft/s (SG inlet) ~153 ft/s (turbine inlet)
2. Main safety valve inlet pipe		
Main safety valve inlet pipe diameter	NPS 6 in.	NPS 6 in.

Where: Sonic speed in water in main steam piping: 1950 ft/s
Ratio = MSS Steam Velocity / Sonic Speed
~7.4% (SG inlet)
~7.8% (Turbine inlet)

Impact on DCD

See Attachment 1 for the mark-up of DCD Tier 2, Section 3.9, changes to be incorporated.

- Add as last paragraph in Subsection 3.9.2.3

  "The design of the US-APWR steam delivery system (including the safety relief valves and the steam separator) and the flow conditions they experience are similar to the existing and currently operating steam delivery systems in the United States and around the world. The US-APWR steam delivery system is designed using the structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR steam delivery system has been operating in the USA for more than 20 years with sizes and flow rates that bound those of the US-APWR steam delivery system. Based on an extensive record of vibration-free operation, the structural and vibration design bases are proven. This non-safety-related steam delivery system will not experience excessive vibration; therefore, the analysis of the flow excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind standpipes) is not expected."

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

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-65:

In the response to RAI 3.9.2-21 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the applicant addressed the global differences between the US-APWR and the 1/5 SMT. In doing so, they did not clarify other differences which may appear small but can have important effect on the test results. For example, Fig. 3-2 in Report MUAP-07023-P indicates that the scale model has a lower core plate and a lower support plate, whereas Fig. 2.1-1 in Report MUAP-07027-P shows that the US-APWR has one plate only (lower core support plate). Such discrepancies between documents are not acceptable. The applicant is therefore requested to explain the reasons for these differences (and others which may not be apparent in the above noted figures), and to clarify the effect of these differences on the test results, structural modeling and forcing functions. The applicant is also requested to update SMT Report MUAP-07023-P so that it reflects the true geometry of the tested model and to include in the report any differences from the US-APWR and the effects of these differences on the test results.

ANSWER:

(Response to RAI 03-09-02-84 is also included in this answer.)

The US-APWR is a 14 ft-core version derived from the J-APWR design, which has a 12 ft core. Specifications of the reactor internals of the J-APWR and the US-APWR are compared in Table 1. Comparisons of the schematic drawings are shown in Fig. 1 to 6.

Although the core length of the US-APWR is extended from that of the J-APWR, the dimensions of the reactor vessel and core barrel are not changed as shown in rows 1 and 2 in Table 1. This is obtained by eliminating the space between the lower support plate and the lower core plate and replaced these two plates with an integrated LCSP design. Therefore, the vibration characteristics of the core barrel can be simulated by integrating the LCP and the LSP as stated in row of 4 in Table 1.

The neutron reflector of the US-APWR is longer than that of the J-APWR. The fundamental natural frequency is reduced from [ ] Hz (in the J-APWR) to [ ] Hz (in the US-APWR). But the flow-induced vibration response obtained from analysis is still sufficiently small as stated in MUAP-07027-P(R1).

The secondary core support assemblies in the lower plenum of the US-APWR are simplified in the number of columns from those in the J-APWR because of the elimination of the columns for the bottom-mounted ICIS. By optimization of the diameters of the support columns, the fundamental modal frequencies of the assemblies are maintained to be the same as those in the J-APWR.

For the upper plenum structures such as the lower part of the RCCA guide tube or the Upper Support Column, specifications are not changed from those of the J-APWR.

In the US-APWR, the in-core instrumentation system (ICIS) is inserted from the top of the vessel, instead of from the bottom of the vessel as in the J-APWR. In both designs, the ICIS is guided and protected from the cross flow loads by guide tubes and support columns. This function is also replaced from the BMI columns in the lower plenum to the upper support columns in the upper plenum.

The information of this RAI response will be included in revised version of the Vibration Assessment Program Report MUAP-07027-P (not J-APWR SMT Report MUAP-07023-P).

Table 1 (1/2) J-APWR & US-APWR Comparison of Reactor Internals

	J−APWR	US-APWR	Effects on FIV
1. Reactor Vessel (Fig. 1) Inlet Dia. Inside Dia. Outlet Dia.			Unchanged
2. Core Barrel (Fig. 1) Length*1 x Outer Dia. Thickness (Upper / Lower)			Unchanged *1: From lower surface of flange to LSP bottom *2: in SMT
3. Secondary Core Support Assembly (Fig 3)			
Upper Assembly Outer Dia. x Inner Dia. x t Column Dia. x Number Column Length			Unchanged because the fundamental frequency is about [ ] Hz in both designs.
Lower Assembly Outer Dia. x t Column Dia. x Number SCS Dia. x Number Column Length			Unchanged because the fundamental frequency is about [ ] Hz in both designs.
4. LCP/LSP (Fig. 2)			
4.1 LSP/LCSP Thickness Flow Hole			Unchanged as the core barrel assembly.
4.2 LSC Height x Dia. x Number			In spite of the extended core in US-APWR, core barrel length is identical to that of the J-APWR by means of the integrated LCSP design.
4.3 LCP Thickness Flow hole			
5. Fuel Assembly			Little impact on the reactor internals vibration although the fundamental mode frequency of the fuel is reduced.
UCP-LCSP height			
Number of Assembly			
6. Nuetron Reflector (Fig. 4)			Natural frequency of fundamental beam mode is reduced from [ ] Hz to [ ] Hz.
Height x Dia.			
Number of Blocks	8	10	

Table 1 (2/2) J-APWR & US-APWR Comparison of Reactor Internals

	J−APWR	US-APWR	Effects on FIV
7. Upper Internals (Fig 5)			
7.1 UCP Outer Dia. x t			Identical
7.2 UCS Outer Dia. x t Skirt Thickness Height			Identical
7.3 USC Height x Outer Dia.			Identical
7.4 TSC Height x Outer Dia.			Identical
7.5 RCC Guide Tube			
Lower Height x Width Guide Plates			Unchanged
Upper Height x Dia. Guide Plates			Natural frequency is reduced from approximately [ ] Hz to [ ] Hz. There is no impact because flow velocity is lower in one order than upper plenum.
8. ICIS Guide and protection from flow (Fig. 6)	(Lower plenum) Guide and flow protection by BMI Column	(Upper plenum) Guide and flow protection by USC	In both design, ICIS are guided and protected from cross flow.

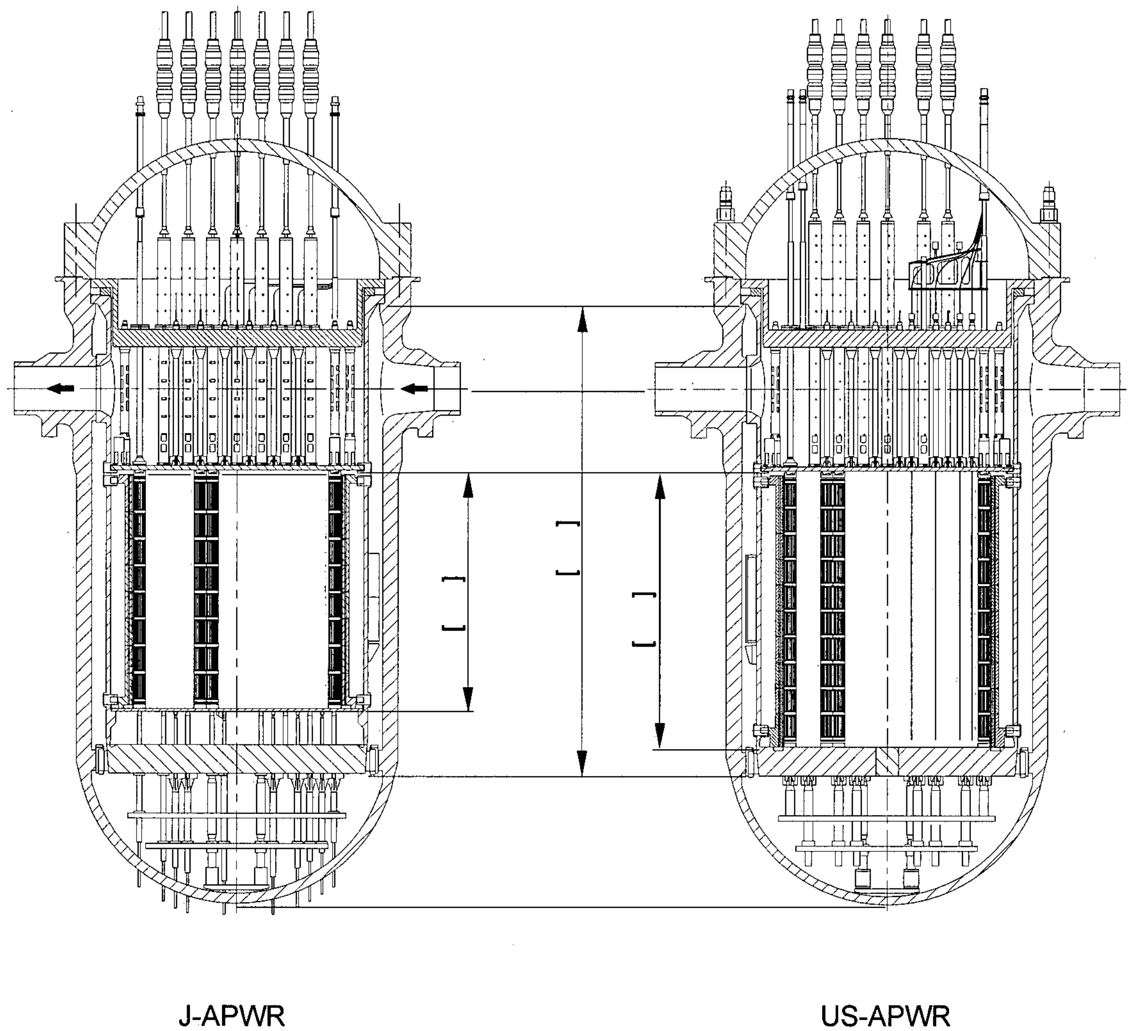
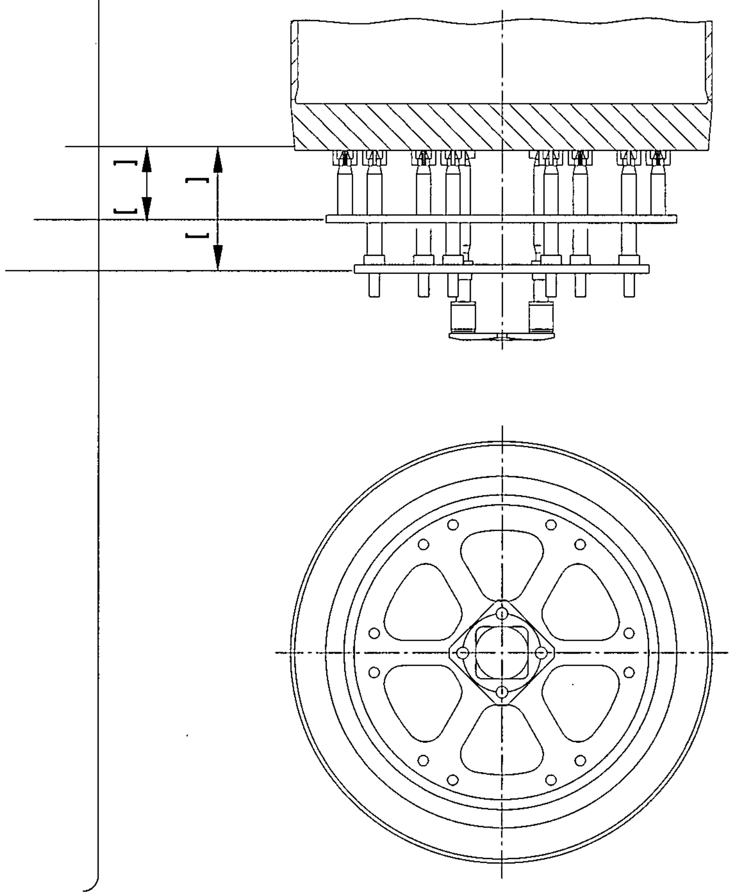


Figure 1 Reactor General Assembly

J-APWR US-APWR

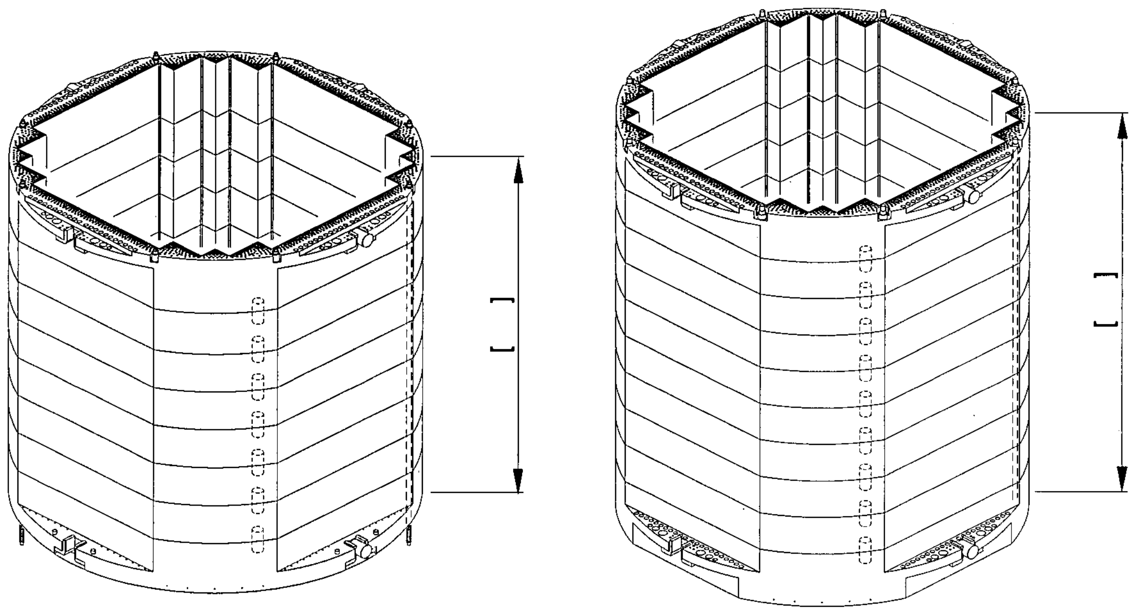
Figure 2 Lower Core Support Plate



J-APWR

US-APWR

Figure 3 Secondary Core Support Assembly



J-APWR   US-APWR

Figure 4 Neutron Reflector

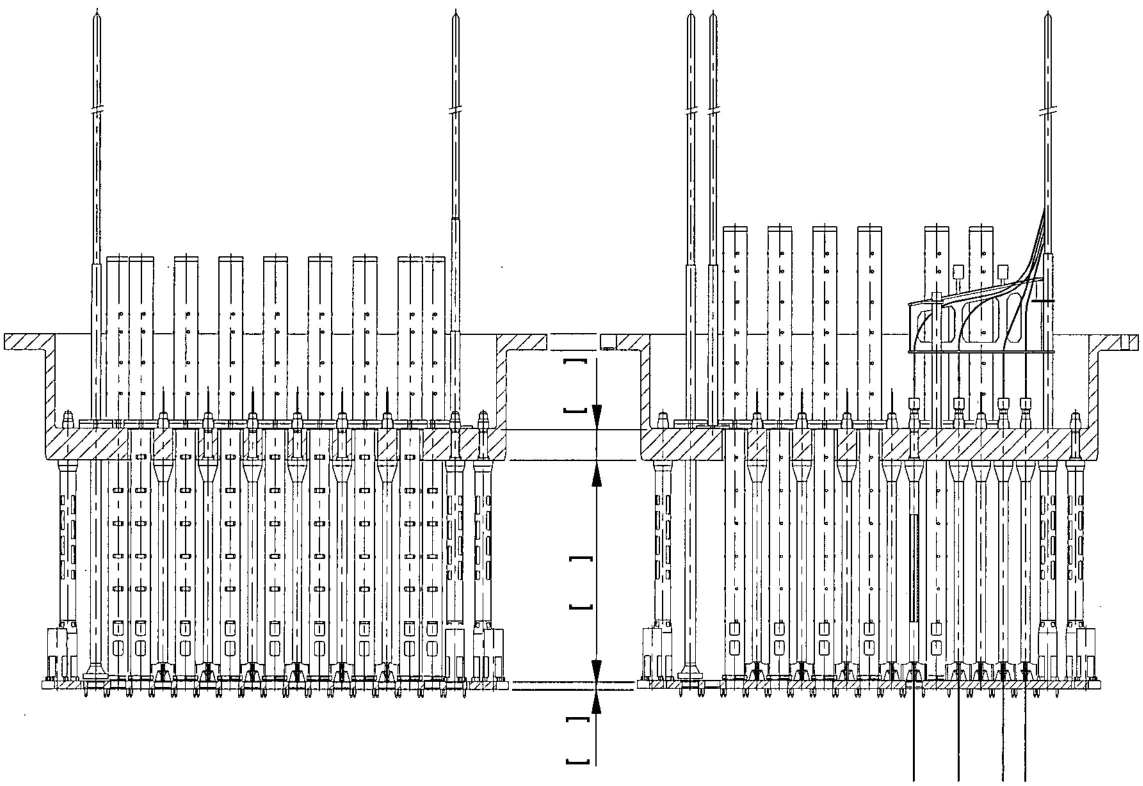
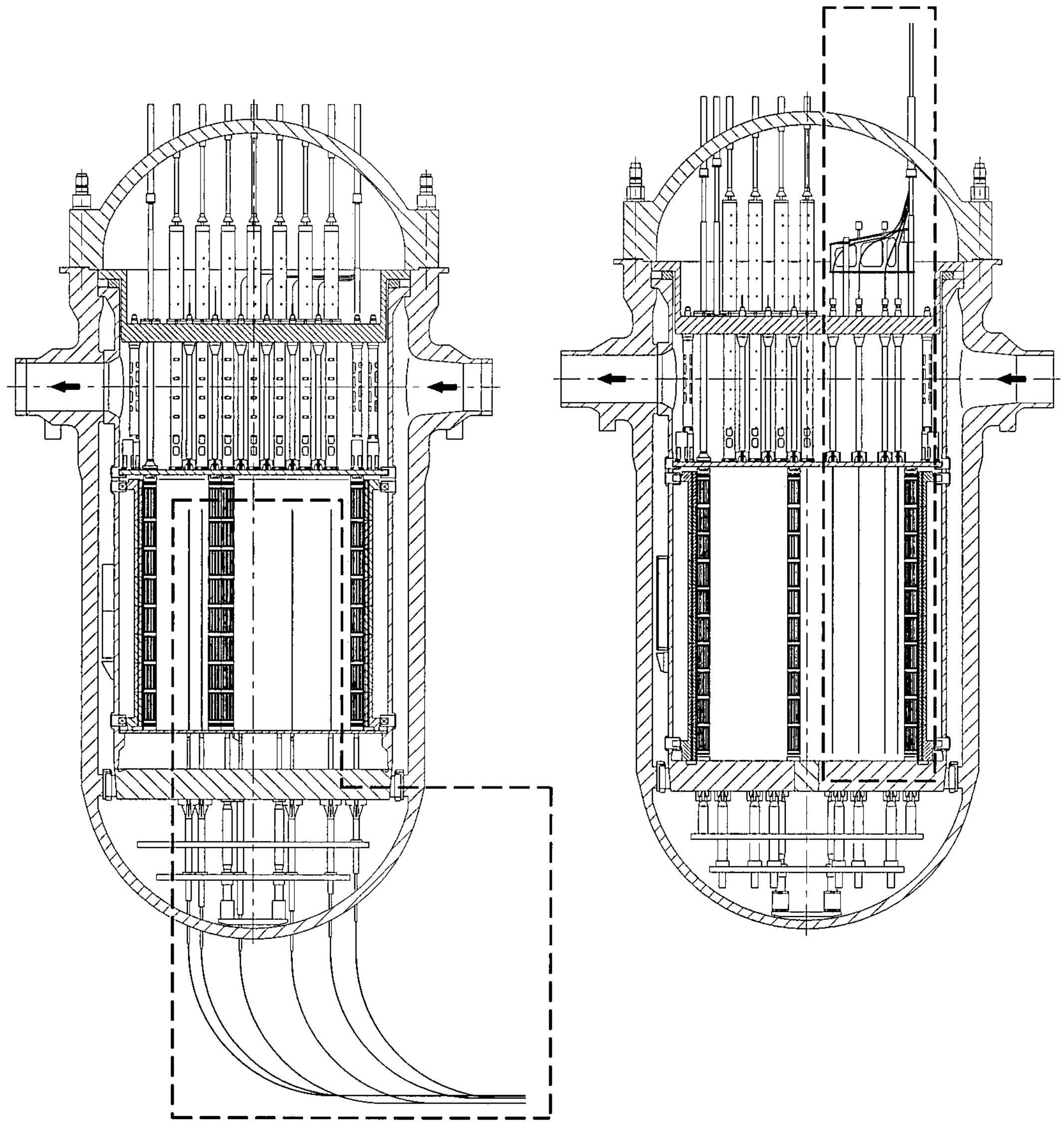


Figure 5 Upper Reactor Internals



J-APWR

US-APWR

Figure 6 ICIS Routing

Impact on DCD

Technical Report MUAP-07027-P will be revised to include this RAI response information.

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

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-66:

In the response to RAI 3.9.2-21 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the applicant stated:

> "The validation of the method of structure modeling was conducted by the comparing the computed natural frequencies of the J-APWR SMT with the measured data, as discussed in Subsection 3.2.1 of the Vibration Assessment Program Report MUAP-07027-P."

The staff finds this validation procedure inadequate because it does not take into account the frequency response functions (FRFs) which express the relationship between the structural response and the forcing functions. In addition, the structural modeling of the US-APWR should be validated from measurements on other full size installations. SRP 3.9.2 recommends that uncertainties and bias errors in FE simulations be estimated from comparisons with measurements made on structures similar in construction to the reactor internals being modeled. The staff appreciates that the validated model will not be that of the US-APWR. However, the procedure for modeling boundary conditions, structural tolerances, damping, welds, etc..., and the resulting bias and uncertainty errors can be validated.

The applicant is therefore requested to provide additional information to assure the staff that:

(a) the structural modeling approach has been adequately validated, and

(b) the bias error and uncertainties have been adequately assessed and incorporated in the dynamic analysis of the reactor internals. In addressing the bias error and uncertainties, the applicant is requested to address how the systematic bias and the random uncertainties are separately estimated.

ANSWER:

(a) Validation of the structural modeling approach

1. Procedure and Referenced data

The vibration analysis of the US-APWR reactor internals consists of the following two Tasks.

Task1: Verification of the analysis model and forcing functions

Task2: Prediction of the US-APWR reactor internals vibrations

In Task 2, as stated in the question, MHI does not use existing plant data as reference. Instead, MHI uses the J-APWR 1/5 scale model test data as reference, for the following reasons:

- The US-APWR reactor internals are more similar to those of the J-APWR than those of the existing plants.
- MHI dose not have measured data from existing plants for use as reference.

Finally, the adequacy of the analysis will be verified with the measured data in the preoperational test of the US-APWR, based on the acceptance criteria described in Section 3.5 of MUAP-07027-P(R1)

2. Acceptance Criteria due to analysis model uncertainty

Based on past experience, MHI assumed a factor of 3 for the uncertainty in flow-induced vibration responses and a factor of 2 for the uncertainty in the flow-induced loads. Therefore the uncertainty of the analysis model itself is expected to be no larger than a factor of 1.5 in the response. To achieve this, the uncertainty in the fundamental modal frequency was limited to be less than 10 percent, which corresponds to 20 percent in the response considering the frequency transfer function as explained below.

- The vibration characteristic of the model can be represented by the natural frequency and the frequency transfer function (FTF). In modal analysis, the FTF of each mode is similar to that of a single spring-mass system, in which a force with a frequency much lower than the structural natural frequency would act like a static force. When the force frequency is close to the natural frequency, the response is amplified with an amplification factor depending on the damping ratio. When the force frequency is higher than 140 % of the natural frequency, the vibration response is reduced from that caused by the static force of the same magnitude.
- For the core barrel or neutron reflector, the flow-induced responses depend on the downcomer flow turbulence. Because a turbulence forcing function have a broad-band spectrum, the response is insensitive to small changes in the structural natural frequency. Therefore the impact of uncertainties in the natural frequency on the response can be estimated by the changes in quasi-static response due to the changes in the stiffness. For example, a 10 percent change in the natural frequency can be represented by a 20 % change in the stiffness or vibration response.
- For structures in the upper plenum or the lower plenum, both cross flow turbulence and vortex shedding load were considered. Because the vortex shedding frequency will be locked in with the natural frequency of structures if it is within 70-130 percent of the vortex shedding frequency, the uncertainty in the natural frequency should be checked

to avoid lock-in with vortex shedding. As shown in Table 3.4.1-1 of MUAP-07023-P(R1), the best estimated natural frequencies in the upper or lower plenum structures are sufficiently high to avoid lock-in with vortex shedding even with 10 % uncertainty. Therefore the sensitivity on the response can be represented by the same discussion as in the case of the core barrel.

(b) Bias errors and uncertainties on the vibration analysis model of reactor internals

For the reactor internals models, model dimensions and relating properties (length, area and section modulus etc.), were determined based on the design drawings of the actual plant. Because the dimensions in the drawings are given under room temperature conditions, the thermal expansion under operating condition is one of the bias errors. In general, the effects of thermal expansions are negligibly small for the base dimensions, but not for the small clearances such as key supports. Therefore the clearance properties in key supports are specified under plant operating temperature, taking into account thermal expansions. Uncertainties with tolerance in manufacturing or alignment are not considered because they are smaller than thermal expansions.

Material properties such as mass density and Young modulus are specified under the temperature during plant operating conditions. Because the coolant temperatures are precisely controlled during plant operations, both the bias and uncertainties of the material properties can be neglected.

For support conditions at the mating surface with friction such as the bottom of the neutron reflector, or close fitting like key supports, realistic analysis conditions depend on the magnitude of the displacement or acting loads at that point. The damping ratio to the critical damping is also depends on the vibration magnitude. Therefore, two kinds of models are made based on the same basic dimensions and material properties. One is used for flow-induced vibration with small responses and the another is used for seismic / LOCA analysis with larger responses. Bias error and uncertainties on the FIV analysis model are summarized in Table 1 and those for the Seismic/ LOCA analysis model are shown in Table 2.

Table 1 Bias errors and uncertainties on the Vibration analysis model of Reactor Internals(FIV)

	Bias	Uncertainty	Validation
Structure dimensions			
Clearance of closed fitting parts			
Friction on mating surface			
Material • Mass density • Young modulus			
Damping ratio			

Table 2 Bias errors and uncertainties on the Vibration model of Reactor Internals (Seismic and LOCA)

	Bias	Uncertainty	Validation
Structure dimensions			
Clearance of closed fitting parts such as radial key			
Friction on mating surface			
Material • Mass density • Young modulus			
Damping ratio			

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.

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-68:

In the response to RAI 3.9.2-24 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the applicant provided inadequate information. As a result, the original request for information is repeated. As previously stated, the applicant has used the SYSNOISE model to describe the acoustic forcing function within the reactor vessel of the US APWR. Therefore, additional information about the validation of this model and its associated uncertainty and bias errors is needed to complete the review process.

In MHI Technical Report MUAP-07027-P, "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals," the applicant used very simple geometries (an annulus and a cylinder) to validate the SYSNOISE model. The staff reviewed the technical report and found this "validation" approach inadequate because the geometry of the reactor and cooling system is much more complex than an annulus or a cylinder. According to SRP 3.9.2 and RG 1.20, the applicant is expected to validate the analytical tools by measurements made on structures similar in construction to the reactor internals being modeled. The staff needs this information to complete the review of the models that are used to describe the acoustic forcing functions and the resulting acoustic and structural responses. The applicant is requested to explain the method used to validate the SYSNOISE model of the reactor acoustic environment. Discuss the bias and uncertainty errors in the model predictions. The validation procedure may include comparisons of SYSNOISE predictions with in-plant measurements of existing 4-loop reactors and with tests of the 1/5 scale model of the APWR. Clarify any differences between the predicted and measured values of acoustic resonance frequencies and frequency response functions. Provide the requested comparisons for various locations within the reactor vessel. Review of these issues is needed to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to include the requested information.

ANSWER:

An acoustic analysis code, SYSNOISE, was used in the assessment of the US-APWR reactor internals. MHI performed the verification of SYSNOISE by benchmark analysis of a simple cylindrical annulus system. The model dimensions were selected to represent the downcomer or the upper plenum of the actual reactor respectively. As stated in the question, the upper plenum of an actual reactor is more complicated because of the many components, such as the RCCA Guide tubes, inside. MHI assumed that these internal components may act as an acoustic resistance (damper of pressure) but do not have significant effects on the basic acoustic modes, because the diameters and their pitches (0.1-0.2m) are much smaller than the wave lengths of RCP pulsation ([ ] m for [ ] Hz and [ ] m for [ ] Hz), although it is difficult to verify this estimation. The 1/5 SMT test data also provided no information on this issue because the RCP characteristics were not simulated in this test. Therefore MHI used the simple models without internal structures for the benchmark problem, where theoretical values of acoustic resonance modes can be calculated.

To allow for the above small uncertainty, it is assured that the analysis results with the SYSNOISE code will have sufficient conservatism by neglecting the acoustic damping effects due to the structural flexibility. MHI performed a sensitivity study using the ANSYS code and a scaled vessel model filled with water as shown in Figure 1. Case1 is an analysis with a rigid wall like in the SYSNOISE model and case 2 with the actual flexible reactor vessel wall. With the flexible wall, the resonance peak is about one order of magnitude smaller than that with a rigid wall. This is the uncertainty in the analysis but the bias is on the conservative side.

From the above discussions, we assumed a factor of 5 for the uncertainty in the RCP pulsation loads and with the same magnitude of bias error on the conservative side. In other words, the RCP pulsation loads may be 10 times larger than the actual value, but not smaller than that.

Figure 1 : Effect of vessel wall flexibility on the acoustic resonance
(Note: The frequency is approximately 15 times of that in actual plant.)

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

2/3/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-69:

In the response to RAI 3.9.2-32 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the applicant explained the acceptance criteria to be 10 percent in the natural frequency for the fundamental mode and the lowest shell mode and a factor of 3 in random response displacement and stress. Regarding the factor of 3 in the random response, the applicant is requested to clarify the implication that the actual stresses of the reactor internals can be up to a factor of 3 higher than the computed stresses. If this is indeed the case, how is this factor accounted for in the bias error and random uncertainties? With respect to the acceptance criterion of 10 percent in the resonance frequency, the applicant is requested to explain how the analysis accounts for an unanticipated coincidence between a resonance frequency and an excitation frequency that are within 10 percent of each other.

ANSWER:

In Section 3.5 of the technical report MUAP07027-P(R1), the acceptance criteria for the measured responses in the pre-operational test were determined. Two categories were defined. Category 1 is the criteria on the structural integrity such as the alternating stress amplitude for the high cycle fatigue evaluation. Category 2 criteria are on the comparison between the analysis and measured results, which are defined to check the prediction analysis reliability.

In general, Category 1 criteria have higher priority and the higher values of the measured data in the Category 2 criteria may be limited to satisfy the Category 1 criteria.

Further discussions based on the predicted stress for US-APWR are included in the answer for RAI 03.09.02-75.

For the discussion about the 10 percent criteria for the uncertainty in the analysis model natural frequency, please refer to the answer for RAI.03.09.02-66 (a).

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

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US-APWR Design Certification

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RAI NO.:	**NO. 498-3782 REVISION 0**
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APPLICATION SECTION:	**3.9.2**
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QUESTION NO. RAI 03.09.02-70:

In the response to RAI 3.9.2-33 (#272-1585, dated 5/13/2009, ML091460116, MHI Ref: UAP-HF-09228), Report No. MUAP-07027-P (R1) was revised with additional information to clarify many aspects of the vibration assessment program. However several issues are still unclear. Section 3 of Report No.MUAP-07023-P (R1) indicates that the designs of the fuel assembly, the radial support of the core barrel, and the holes in the neutron reflector were modified in the scale model for the sake of simplicity. However, the details of these modifications are not addressed. It is requested that these modifications and their effects on the test results be discussed and documented in the DCD.

ANSWER:

The descriptions on the three kinds of model modifications in the J-APWR 1/5 SMT and their effects on the test results are summarized in Table 1.

Because the vibration of the fuel assembly was verified with a full scale mock-up test, fuel assembly in the 1/5 scale model was simplified. The numbers of rods and grids were reduced although the scaled mass and pressure drop were still simulated. The natural frequency of the fuel was not simulated. But its impact on the vibration responses of the reactor internals was small because their natural frequencies are well separated.

For the Neutron Reflector, the numbers and diameter of flow-holes were modified so that the total section area of flow holes was properly scaled (1/25 of that in the actual plant). This modification had no impact on the shell mode stiffness and natural frequencies as confirmed by FE analysis.

The modification of the radial key was not for simplification but to control the test conditions. The shapes and locations of the radial key were modeled to simulate the flow around the radial key. Because the support condition of the core barrel bottom was controlled by additional push bolts, the clearance of the radial key were extend to assure the no contact condition.

Table 1 J-APWR 1/5 Scale Model Test Model Modifications

	Modification	Simulated properties	Effects on test results
Fuel Assemblies			
Clearance between the radial key to support of the core barrel			
Neutron Reflector			

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

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-71:

In the response to RAI 3.9.2-33 (#272-1585, dated 5/13/2009, ML091460116, MHI Ref: UAP-HF-09228), several issues are still unclear. The applicant is requested to confirm how the dynamic analysis of the reactor internals was benchmarked by means of the SMT. Section 6.1 of the revised Report MUAP-07023-P (R1) suggests that the SMT results were scaled up to the J-APWR and the dynamic analysis was performed on the J-APWR. However, in MHI's response to RAI 3.9.2-33, and in the revised version of MHI Report MUAP-07027-P (R1), the applicant explained that in the FIV analysis program, the measured responses of the J-APWR scale model tests were compared with those estimated by the dynamic analysis applied to the SMT size and test conditions. Also, in Figs. 3.2.1-3 to -12 of Report MUAP-07027-P (R1), the figure captions refer to "actual dimensions" without indicating whether these dimensions are those for the SMT or the full-scale reactor. The applicant is therefore requested to explain this apparent contradiction. In particular, was the dynamic analysis performed on the size and flow conditions of the small scale model or the full-scale J-APWR? The applicant is also requested to modify the necessary documents to eliminate this apparent contradiction.

ANSWER:

The benchmark analysis model was developed with the 1/5 scale model dimensions. The material properties of structures and coolant are defined at room temperature as the test conditions.

Comparisons of the analysis response and measured values were performed after scaling up to the plant because the measured results had been scaled in the test report. The scaling factor from test dimensions to actual ones is shown in Table 1 (Table 3.1 of the MUAP-07023-P(R1)).

In addition to the scaling factors, the effect of the temperature difference such as fluid mass density and Young moduli has been discussed in the response to the RAI 03.09-72, dated January 15, 2010.

Table.1 Scaling Law and Flow Condition Comparison with J-APWR and Test Condition
(Table 3.1 of MUAP-07023-P(R1))

Items		J-APWR	Test condition
Flow condition	Flow volume		
	Pressure		
	Temperature		
	Dynamic viscosity		
	Velocity at outlet nozzle		
Scaling law	Length		
	Strain		
	Stress		
	Velocity		
	Acceleration		
	Load		
	Frequency		
Reynolds number	Inlet nozzle		
	Downcomer		
	Outlet nozzle		

(1): Flow rate is mechanical design flow at that time.

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.

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US-APWR Design Certification

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Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-75:

In the response to RAI 3.9.2-33, several issues are still unclear. Report No. MUAP-07027-P (R1) indicates that substantial uncertainties exist in the dynamic analysis. For example, in the revised SYSNOISE analysis, the RCP pulsation amplitude is reduced by a factor of 5, and the response of the reactor internals to this RCP pulsation increases by a factor of 5 when the simulation time step is refined. Moreover, when comparing the SMT random response with the response obtained from the dynamic analysis, a ratio of 3 between the measured and predicted values is considered acceptable.

Despite these substantial uncertainties indicated above, the applicant considers a margin of safety of 30 percent acceptable for the high cycle fatigue analysis as indicated in Table 3.3.3-4 of the above mentioned report. The applicant is requested to explain why this margin of safety (30 percent) is considered conservative despite the existing much wider range of uncertainty.

ANSWER:

The uncertainty of the analysis is determined as the ratio to the best-estimated value without bias errors if they are identified.

Through the results of the high cycle fatigue analysis for the US-APWR reactor internals, minimum margins of safety 0.3 were predicted for the components in the upper plenum, the RCC guide tube (GT), upper support column (USC) and top slotted column (TSC) as shown in Table 1. (Table 3.3.3-4 of MUAP-07027-P(R1). 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 alternating stresses due to the RCP pulsation have large uncertainty (5) but the absolute value of these ([ ] ksi) are lower than those due to the cross flow ([ ]ksi) by one order of magnitude.

2. The RCP pulsation loads include a conservative bias by neglecting the acoustic damping due to structural flexibility as discussed in the response to RAI 03.09.02-68. 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 (factor of 5). Therefore, the analysis results due to RCP pulsation may be 10 times larger than the actual values, but not smaller.

3. 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 estimated to be around a factor of 2, which is comparable to the assumed uncertainty in the flow-induced loads.

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

Table 1 High Cycle Fatigue Evaluation Based on Analysis Responses
(Table 3.3.3-4 of MUAP-07027-P(R1))

Components	Locations or parts	Alternating Stress (ksi)			Limit	Margin of Safety [1]
		Flow	RCP	Total		
Core Barrel	Flange					
Neutron Reflector	Block Alignment Pin					
Diffuser Plate Assembly	Support Column Upper Assembly Lower Assembly				13.6 ksi	
UCS	Flange Skirt					
RCCA GT	Top of Lower GT					
USC						
TSC						

Note
1) Margin of safety = (Allowable Stress Limit) / (Alternating Stress) - 1

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.

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 498-3782 REVISION 0

SRP SECTION: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components

APPLICATION SECTION: 3.9.2

DATE OF RAI ISSUE: 12/01/2009

QUESTION NO. RAI 03.09.02-80:

In the response to RAI 3.9.2-70 (#207-1577, dated 3/27/2009, ML090910120, MHI Ref: UAP-HF-09117), MHI states that it is acceptable to have a factor of 2 between the measured damping ratios during the pre-operational tests and the damping values used in the prediction analysis. The staff finds it excessively unconservative if the damping ratios, used in the prediction are twice the actual ratios determined during the preoperational tests. To maintain sufficient conservatism in the analysis, the applicant is requested to use damping ratios which are equal or smaller than those determined from measurements.

ANSWER:

In general, damping ratios in water are larger than those in air because of the additional fluid damping. For example, [ ] percent damping ratio was measured for the core barrel fundamental mode in water although only 1 percent was measured in air in the J-APWR 1/5 Scale Model Testing. Based on these data, we applied [ ] percent as the best estimate value in the scale model test simulation analysis. But for the prediction analysis of the US-APWR, 1 percent damping ratio was applied to maintain sufficient conservatism.

The acceptance criterion of the damping ratio was defined as one of the Category2 criteria which are defined to check the analysis reliability. Therefore, measured damping ratios in the pre-operational test should be compared with best-estimate values without any bias. AS an example, if the measured damping ratio of the core barrel is [ ] percent, which is half that of the predicted best estimate value, the design analysis results with 1 percent damping ratio is still conservative.

From the above discussions, the factor of 2 for the damping ratio is reasonable as the acceptance criteria. It is also noted that the structural integrity of the components will also be assured by other acceptance criteria for the measured responses, such as the stress limit.

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

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US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: **NO. 498-3782 REVISION 0**

SRP SECTION: **03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components**

APPLICATION SECTION: **3.9.2**

DATE OF RAI ISSUE: **12/01/2009**

QUESTION NO. RAI 03.09.02-82:

In MHI's response to US-APWR DCD RAI No. 03.09.02-37, 214-1920, dated April 30, 2009 (MHI Ref: UAP-HF-09190, ML091240403), the applicant stated that this question was answered in the responses to RAI 212-1950, RAI 3.7.2-26 (dated March 30, 2009, MHI Ref: UAP-HF-09113, ML090930727). In its response to RAI 3.7.2-26 the applicant stated that it is the intent of the US-APWR design to always meet the requirements of RG 1.92, Rev.2 or 1 (when permitted) for combining modal responses. The applicant also stated that DCD Section 3.7.2.7 will be revised to clarify this issue. In its review, the staff noted that to resolve the staff's concerns in RAI 3.7.2-26 the applicant needs to provide data to show that their approach is conservative.

ANSWER:

MHI has made the commitment in their response to RAI 3.7.2-26 that the US-APWR design meets the requirements of RG 1.92, Rev.2 or 1 (when permitted) for combining modal responses. The NRC staff has stated in the 1st paragraph of the "Background Discussion" section and the 1st paragraph of the "Regulatory Position" section of Regulatory Guide 1.92, Revision 2 that "The more conservative methods of combining modal responses (as described in revision 1) remain acceptable".

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.

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Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.: NO. 498-3782 REVISION 0

SRP SECTION: 03.09.02 - Dynamic Testing and Analysis of Systems Structures and Components

APPLICATION SECTION: 3.9.2

DATE OF RAI ISSUE: 12/01/2009

QUESTION NO. RAI 03.09.02-84:

Based on the evaluation of the applicant's response to the RAIs on Subsection 3.9.2 of the DCD, the staff is still concerned about the differences between the scale model geometry and the US-APWR. Some of these differences have already been addressed by the applicant, but others seem to exist in the submitted drawings but are not addressed by the applicant, e.g. the second follow-up RAI to question 3.9.2-21, ninth question of this RAI. The staff is also concerned that additional differences may exist which cannot be seen in the scale model drawings. The applicant is therefore requested to provide a list of all the differences between the US-APWR and the geometry of the scale model, which is used in the vibration testing. The applicant is also requested to demonstrate that the effect of each of these differences on the estimated vibration response of the US-APWR is conservative.

ANSWER:

The response to this question is included in the answer to RAI 03.09.02-65.

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.

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

assessment program. Detail of the analysis is described in Reference 3.9-22. The evaluation of the SG is described in Subsection 5.4.2.1.

The design of the US-APWR steam delivery system (including the safety relief valves and the steam separator) and the flow conditions they experience are similar to the existing and currently operating steam delivery systems in the United States and around the world. The US-APWR steam delivery system is designed using the structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR steam delivery system has been operating in the USA for more than 20 years with sizes and flow rates that bound those of the US-APWR steam delivery system. Based on an extensive record of vibration-free operation, the structural and vibration design bases are proven. This non-safety-related steam delivery system will not experience excessive vibration; therefore, the analysis of the flow excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind standpipes) is not expected.

3.9.2.3.1 Classification of Reactor Internals in Accordance with the Comprehensive Vibration Assessment Program

The US-APWR reactor internals components are evolved from that of the well-proven current 4-loop plant design operating in United States and Japan. The differences are as follows:

- Design: the US-APWR uses neutron reflector instead of baffles
- Size: there are increases in the diameters of RV, core barrel and the secondary core support assembly
- Arrangement: RCCA guide tubes and upper support columns in the upper plenum
- Operating conditions: there is an increase in flow rate

The US-APWR reactor internals represent a unique, first of a kind design because of its design, size, arrangements and operating conditions. Therefore, the first US-APWR will be classified as a Prototype in accordance with Regulatory Guide 1.20 (Reference 3.9-21). Upon qualification of the first US-APWR as a valid prototype, subsequent plants will be classified as Non-Prototype Category I.

3.9.2.3.2 Comparative Analysis of the US-APWR and the Current Plant

In this section, flow-induced vibration characteristics of the US-APWR reactor internals are assessed in comparison to those of the current 4-loop plant. Subsection 3.9.5 provides general information on the reactor internals.

- **General**

  The basic design of the US-APWR reactor internals follows that of the current 4-loop plant but features a larger core barrel diameter and a neutron reflector instead of a baffle structure. However, the coolant flow velocities are carefully designed to remain the same as those in the current 4-loop plant so that any increase in the excitation force due to a larger surface area exposed to the coolant