MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

April 9, 2009

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-09149

Subject: MHI's Response to US-APWR DCD RAI No. 272-1585

References: 1) "Request for Additional Information No. 272-1585 Revision 0, SRP Section: 03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components, Application Section: DCD, Tier 1 – Section 3.9.2.3," dated 3/10/2009.

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. 272-1585 Revision 0."

Enclosed are the responses to questions 19-32 and 34-35 of the RAI (Reference 1). The response to question 33 of this RAI has a 60-day response time, as agreed to between the NRC and MHI, and will be issued at a later date by a separate transmittal.

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 "[ ]" (brackets).

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) and 10 C.F.R.§ 9.17 (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.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 272-1585, Revision 0 (Proprietary)
3. Response to Request for Additional Information No. 272-1585, Revision 0 (Non- Proprietary)

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

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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-09149

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. 272-1585, Revision 0", dated April 9, 2009, 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 analyses used by mathematical models developed at significant cost to MHI. Since it required the performance of detailed design calculations, supporting analyses and testing extending over several years. 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.

   B. They include the information that is provided to MHI pursuant to licensing agreements with third parties (the "Licensors") for MHI's use and under the obligation to maintain their confidentiality. Furthermore, MHI has an ownership interest in the referenced information by having paid significant sums of money to the Licensors for the rights to the intellectual property therein such that public disclosure of the materials would adversely affect MHI's competitive position.

   C. They include the information directly referred from books the copyrights of which are

reserved.

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 9th day of April 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09149
Docket No. 52-021

Response to Request for Additional Information No. 272-1585,
Revision 0

April, 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-19

In DCD Tier 2, Subsection 3.9.2.3, the applicant made a commitment to ensure the structural and functional integrity of the reactor and steam generator internals under vibratory loadings and thereby assure conformance with GDC 1 and 4.

The staff's review of Subsection 3.9.2.3 indicated that not all the information and documents are provided. Most notably the DCD does not include any dynamic analysis of the steam generator internals, the steam separator, and the safety relief valves, nor does it indicate the design lifetime of the steam generators. According to SRP 3.9.2 and RG 1.20, the applicant is expected to evaluate potential adverse flow effects for the steam generator internals, including the steam separator. The applicant is therefore requested to provide the following:

If the steam generators for the MHI US APWR are classified as prototypes, provide a complete analysis of dynamic responses of structural components within the steam generator caused by steady and operational transient flow conditions. A detailed summary of the assessment of the potential of any adverse flow effects, such as flow- induced vibrations and acoustic resonances, should also be provided. If the steam generators are classified as non-prototypes, provide the dynamic analysis for the components with deviations from the prototype design or operating conditions. The analysis should be accompanied by the expected bias and uncertainty errors. If the steam generator internal structures are a non-prototype design, provide reference to the analysis of the prototype steam generator and give a brief summary of the results. Alternatively, the applicant may prefer to provide a reference document that describes the details of the vibration analysis of the steam generator and includes a brief summary of the results in Subsection 3.9.2.3 of the DCD.
The staff needs this information to assure conformance with GDC-1 and 4. Revise Subsection 3.9.2.3 of the DCD to include the dynamic analysis of the steam generator internals.

ANSWER:

The topic of steam generator (SG) upper internals vibration is addressed in MHI US-APWR DCD Subsection 5.4.2.1.2.10 (via a cross reference from Subsection 3.9.2.4.1). MHI will revise DCD Subsection 5.4.2.1.2.10 to more fully address the structural adequacy of the SG internals.

Impact on DCD

DCD Revision 2 will incorporate the following changes:

- Subsection 5.4.2.1.2 will be revised in DCD Revision 2 as part of the response to RAI 206-1576, Question 3.9.2-40. Refer to Response to Request for Additional Information RAI 206-1576, Question 3.9.2-40, for changes to Subsection 5.4.2.1.2 that are applicable to this response for RAI 272-1585, Question 3.9.2-19.

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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-20

Previous experiences from boiling water reactors (BWRs) indicated that excitation of acoustic resonances in the standpipes of safety valves are likely to be detrimental to the safe operation of the plant. The potential for a similar occurrence also exists in the standpipes of PWRs.

According to SRP 3.9.2 and RG 1.20, the applicant is expected to evaluate potential adverse flow effects for the steam delivery system, including the safety relief valves and the steam separator. The applicant is requested to analyze the potential of flow-excited acoustic resonance occurring in the standpipes of the safety relief valves (or in any other blind standpipes), which are mounted on the main steam lines exiting the steam generators. If any acoustic resonance is anticipated, explain the countermeasure(s) that will be implemented to avoid or mitigate the resonance. Alternatively, the applicant may provide a reference document that includes the requested information and refer to that document in Subsection 3.9.2.3 of the DCD. The staff needs this information to assure conformance with GDC-1 and 4. Revise Subsection 3.9.2.3 of the DCD to include the analysis of the potential of flow-excited acoustic resonance occurring in the standpipes of safety relief valves.

ANSWER:

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. MHI designed the US-APWR steam delivery system 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, MHI concludes that 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 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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-21

In MHI technical report MUAP-07023-P, the applicant described the methodology used to analyze the dynamic responses of the reactor internals. The methodology consists of performing 1/5 scale model tests to characterize the dynamic fluid forces and validate the computational tools used to simulate the dynamic responses. The details of the vibration analysis of the reactor internals are presented in the technical report MUAP- 07027-P. The applicant has used the scale model tests to validate the forcing function definitions and the structural models for dynamic analysis, the details of the model geometry and test conditions, in comparison to the US APWR.

The staff reviewed these documents and found that neither the DCD nor the scale model test report (*1/5 Scale Model Flow Test Report MUAP-07023-P*) discussed the geometry differences between the scale model and the US APWR. The staff needs this information to complete the review of adverse flow effects and to evaluate the validation procedure of the models used to describe the forcing functions and the structural response. The applicant is requested to provide details of any deviations in the 1/5 scale model geometry from the US APWR geometry and operating conditions.

Comparison of Fig. 3-2 in the technical report MUAP-07023-P with Fig. 2.1-1 in the technical report MUAP-07027-P suggests that the small-scale model reflects the geometry of the 4-loop reactor rather than that of the US APWR, with the exception of including the neutron reflector. Explain the effects of these differences on the validity of the model test results and their applicability to the US APWR. Discuss how the effects of using a shorter core in the scale model [corresponding to 3.66 m (12 ft) instead of 4.27 m (14 ft)] and effects of other differences are accounted for in assessing potential adverse flow effects, in defining the fluid forcing functions and in the scaling of the forcing functions from the model to the full-size reactor. Review of these issues is necessary to assure conformance with GDC-1 and 4. Revise the 1/5 scale model test report to include the requested information.

ANSWER:

1. Difference between J-APWR and US-APWR

   The differences among the current 4-loop plant, the J-APWR and the US-APWR have

been discussed in the response to RAI 3.9-43, item (a).

2. Analysis procedure, modeling and forcing functions

The analysis procedure, modeling and forcing functions for the US-APWR prototype are discussed in the responses to RAI-3.9.2-21 and 3.9.2-26. The following answer is provided as a consolidated response to both RAIs.

As described in Subsection 3.1.1 of the vibration assessment program report MUAP-07027-P, the FIV analysis program consists of two separate tasks. One was to estimate the responses of the model used in the J-APWR scale model test (SMT) and the other was the prototypical analysis of the US-APWR. The modeling and forcing functions were customized for each task as follows.

Task 1: J-APWR SMT simulation analysis

This task was performed to validate the analysis method.

All properties in the SMT simulation analysis were adjusted to 1/5 scale. The structural model of the reactor internals was developed based on the full scale J-APWR drawings and scaled down following the scaling laws for each parameter. The stiffness of the test vessel support, which was not simulated in the actual plant, was determined based on the measured natural frequency in the tapping test.

The forcing functions for the SMT simulation analysis were determined at the test flow rate at room temperature consistent with the SMT conditions.

Task 2. Analysis for the US-APWR prototype

The analysis of the US-APWR was performed with a full scale model and forcing functions for the US-APWR. Tables 1 and 2 give a side-by-side comparison of the US-APWR and the J-APWR SMT simulation analyses.

In the US-APWR analysis, the model properties are developed based on the drawings of US-APWR in the same manner of J-APWR SMT simulation analysis.

The flow-induced forcing functions in the horizontal direction were scaled from the J-APWR SMT as shown in Table 2.

For the down-comer turbulence and the cross flow loads in the upper plenum, the force time history data generated for the 1/5 SMT in the J-APWR analysis were directly converted to those of US-APWR prototype in the following manner. Conversion ratios C1, C2 and C3 were factors for the force area, the dynamic pressure and time (or frequency), as shown in Table 3. Here, the constants of proportionality are assumed to be the same because the structural configurations and flow conditions are the same in these regions in both plants. .

$$Fuspro(t) = C1\ C2\ F\ jsmt(t/C3)$$

Here,

Fuspro: force for analysis of the US-APWR prototype

Fjsmt : force for simulation of 1/5 scale model test of the J-APWR

t : time

This scaling process has been validated by comparison of the normalized PSDs measured in a PWR scale model test under room temperature and that of plant field data by Au-Yang as shown in the Fig 8.17 in Reference 1. Similarity between the US-APWR SMT and the Au-Yang data was also shown in Fig 2.

For the lower plenum cross flow loads, the original forcing functions in the US-APWR prototype were re-constructed for the diffuser plate columns, because the diameters of these columns are different from those of the BMI columns in the J-APWR.

The vertical loads and the RCP pulsation loads were also developed specifically for the US-APWR prototype.

3. Validation of the analysis method

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.

Validation of the forcing functions was confirmed by comparing the computed responses in the model in the J-APWR SMT with the measured responses as discussed and verified in Subsection 3.4.1 of the Vibration Assessment Program Report MUAP-07027-P. In addition, the scaling of the forcing function from a SMT to that in a PWR under operating condition has been validated through the comparison of the normalized PSD measured in a PWR scale model test and the corresponding field test data from a full scale plant by Au-Yang as shown in Figure 1 (Fig. 8.17 in Reference 1). Agreement between the normalized data from the US-APWR SMT and the Au-Yang data is also shown in Figure 2, except J-APWR SMT data of 90 degrees upper and middle. It is reasonable that the two data are larger than the others because these two measure points are closed the inlet nozzle.

Table 1 Comparison of the models in the simulation analysis of the J-APWR SMT and the US-APWR prototype vibration assessment

	J-APWR SMT model	US-APWR Proto-type model
Configurations and dimensions	1/5 of J-APWR	1/1 of US-APWR
Properties of Vessel support stiffness	Test vessel support	Plant design value
Reference temp. for metal / fluid material properties	Room temp.	Temp in normal operation
Vessel Inlet Flow rate for flow velocity definition	28,200 / 25 m^3/h/loop (= test condition)	29,600 m^3/h/loop

Table 2 Conversion of Flow- Induced Forcing Functions from J-APWR SMT into US-APWR prototype

	Fjsmt: Forcing function for J-AWR SMT	C1: Scale effect on force area	C2: Ratio Of ρv^2	C3: Scale effect on time
Down comer Turbulence	Measured data in J-APWR 1/5SMT	25	[ ]	5
Cross flow loads in Lower Plenum	Ref.1 Fig.9-5	Defined with the US-APWR configuration		
Cross flow loads in Upper Plenum structures	Ref.1 Fig.9-5	25	[ ]	5

Figure 1 Comparison of empirical normalized PSD equation with field measured data (Au-Yang and Jordan,1980, Fig.8-17 of Ref.1)

Figure 2 J-APWR 1/5 SMT D/C NORMARIZED PSD with Au-Yang's empirical equation

Reference

1. "Flow-Induced Vibration of Power and Process Plant Components: A Practical Workbook", M.K.Au-Yang ,2001, ASME Press.

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated: add a new sentence to the end of the third paragraph of Subsection 3.9.2.3.3, "Discussion about the dimensionless parameters and the ratio of forcing functions in the scale model test and those under the plant operating condition are included in Appendix XX of Reference 3.9-22."

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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-22

Apart from the requirement of geometric similarity, when flow-induced vibration mechanisms are investigated by means of model tests, the model and prototype must also be dynamically similar. The applicant used the 1/5 scale model to confirm the reactor structural integrity against flow-induced vibration and static loading. It is therefore essential that the model tests be dynamically similar to the prototype. Deviations of the model parameters from dynamic similarity should be shown to be conservative; otherwise flow excitation mechanisms that occur under the prototype test conditions may not be reproduced in the model tests.

The staff's review indicated that neither the DCD nor the scale model test report (*1/5 Scale Model Flow Test Report MUAP-07023-P*) discussed the dynamic similarity of the scale model tests despite its importance to the reproduction of flow-induced vibration mechanisms. The applicant is requested to compare the relevant dimensionless parameters for the 1/5 scale model tests and the full size reactor at normal operation conditions to demonstrate dynamic similarity between the small scale model and the prototype. Examples of dimensionless parameters include, but are not limited to, fluid-elastic parameter, Strouhal number, reduced velocity, and the ratio of excitation to resonance frequencies. The staff needs this information to complete the review of adverse flow effects and evaluate conformance with GDC-1 and 4. Revise the 1/5 scale model test report to demonstrate the dynamic similarity of the scale model tests and refer to the satisfaction of dynamic similarity criteria in Subsection 3.9.2.3 of the DCD.

ANSWER:

Detailed comparisons of dimensionless parameters between the J-APWR SMT and the US-APWR plant are listed in the Response to RAI3.9-24 under Item (b). This information will be included in an Appendix in the vibration assessment report MUAP-07027-P and will be referred to in the third paragraph in Subsection 3.1.1 of the same report, as following for the mark-up with under-line:

"The Scale Model Test was conducted using a 1/5 scale model that simulated the reactor vessel and the reactor internals of the 12 ft-core APWR (J-APWR). In this test, the vibration characteristics of each component, the pressure fluctuations due to flow turbulence, and the vibration responses were measured (Reference (7)). The test was performed under ambient temperature and pressure. Comparisons of dimensionless parameters between the J-APWR SMT and the US-APWR plant are discussed in Appendix XX."

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated: add a new sentence to the end of the third paragraph of Subsection 3.9.2.3.3, "Discussion about the dimensionless parameters and the ratio of forcing functions in the scale model test and those under the plant operating condition are included in Appendix XX of Reference 3.9-22."

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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-23

In the technical report MUAP-07027-P the applicant used the resonance frequencies of the small-scale model to validate the FE dynamic simulation of the small-scale model. However, for the US-APWR, it is not clear whether the FE simulation has already been validated (for example, by comparison with the results of the 4-loop reactors). SRP Section 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 reviewed the Subsection 3.9.2.3 of the DCD and the MHI technical reports and found that the applicant did not include sufficient information about the experiments performed to validate the structural models nor did the applicant provide any validation results, except a comparison of resonance frequencies for the 1/5 SMT which is included in report MUAP-07027-P. In particular, full-scale validation tests of reactor internals (e.g. tests of existing 4-loop reactors) are not mentioned. Validation of the structural models is needed to complete the review of the dynamic responses of reactor internals, including the potential of adverse flow effects. The applicant is requested to explain the methodology used to validate the structural models of the prototype reactor internals and provide typical results of the validation tests together with the bias errors and uncertainties which are expected in the results of structural modeling. Briefly describe the measurements performed to determine the structural resonance frequencies, the mode shapes and the frequency response functions (FRF). Accurate structural modeling is essential for reliable evaluation of the dynamic responses of the reactor internal structures. Review of these issues is necessary to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to include the methodology used to validate the structural models and refer to this methodology in Subsection 3.9.2.3 of the DCD.

ANSWER:

The validation of the structure modeling was conducted by the comparison of 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.

A typical value of the uncertainties and bias errors of the structural resonance frequencies is [ ]%, which is obtained from averaging the errors in a total of 12 pairs of natural frequencies as shown in Table 3.2-1 of P. The effect of the [ ] % error in the structural resonance frequencies on the vibration response is estimated by considering the mode shapes and the frequency response functions (FRF). It leads to [ ]% error in the random vibration response, with a conservative assumption that the [ ]% error in the natural frequency is totally caused by the uncertainty in the stiffness of structures. However the error in the mass of the model has little effect on the vibration response.

Comparison with measured data from a currently operating plant is not a better method of validating the analysis methods used in the US-APWR due to the following reasons.

(1) The vibration characteristics of the US-APWR reactor internals are close to those of the J-APWR rather than to the current plant, as discussed in the response to RAI3.9-43.

(2) This method cannot be applied to fist-of-a-kind design with significantly different dimensions or configurations, such as the neutron reflector or the core barrel.

Impact on DCD

See attachment 1 for the changes to be incorporated to DCD Section 3.9.

A new sentence will be added to the end of third paragraph in Subsection 3.9.2.3.3:"Detail of the model validation including the bias error and uncertainties is discussed in Subsection 3.2.1 of Reference 3.9-22. "

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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

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SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-24

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 the 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 that this "validation" approach is 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:

Using test data from the the 1/5 scale model of the APWR to validate the acoustic analysis procedure is not adequate because they do not simulate the acoustic pressure pulsations

induced by the RCPs. MHI instead selected comparison with theoretical results as the validation procedure due to lack of measured data from the existing 4-loop reactors.
The downcomer and the upper plenum were selected for the validation of the SYSNOISE acoustic analysis. These regions have high possibilities of acoustic resonance induced by the RCP. The downcomer was analyzed as an annulus while the upper plenum was analyzed as a cylinder to compare with the theoretical results. The results are reported in Table 3.3-5 of MUAP07027-P.
In the design analysis model, the lower plenum and the reactor core connecting the above region were added. The head plenum was excluded from the model because it is an acoustically-isolated closed space. In addition, a sensitivity analysis was conducted for the effect of reactor internals in the lower plenum and the upper plenum. The internals of the lower plenum was omitted because the presence of the internal components there has insignificant effect on its acoustic characteristics. On the other hand, in the upper plenum, the presence of internal components alters its acoustic characteristics. Therefore they were included in the model to keep the uncertainty and bias errors of the calculated resonance frequencies to within [ ]%, even with uncertainties in the sound speed..

Above information will be included in Subsection 3.3.4 (2) b, c of the revised version of MUAP-07027-P.

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated: add a new sentence to the end of the 2nd paragraph of Subsection 3.9.2.3.3, "Detailed information, about the acoustic resonance analysis including the discussions about the code verification and assumption of the acoustic damping are included in Subsection 3.3.4 of Reference 3.9-22."

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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

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APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-25

The applicant has used the SYSNOISE model to describe the acoustic forcing function within the reactor vessel of the US APWR. The reliability and associated bias and uncertainty errors of the reactor acoustic model results depend on the sound attenuation coefficient (i.e. acoustic damping) assumed at various locations along the flow path. The use of appropriate sound attenuation coefficients is therefore necessary to ensure that the reactor internal structures are designed to quality standards commensurate with the importance of their safety functions.

The staff reviewed the Subsection 3.9.2.3 of the DCD and found that the applicant did not provide any information about the sound attenuation coefficient used in the acoustic model of SYSNOISE. The value(s) of this coefficient and the method used to validate these values are needed to complete the review of the dynamic responses of reactor internal structures, including the potential of adverse flow effects. The applicant is requested to discuss the value(s) of the sound attenuation coefficient which is used in:

(a) the validation of the SYSNOISE model against the test results of the 1/5 scale model, and

(b) computing the acoustic loading on the reactor internals of the US APWR by means of the SYSNOISE model

Also, in each of the above two cases, substantiate the used value(s) and explain how these values were validated. The staff requests this information to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to address the sound attenuation coefficient and its validation methods and refer to this information in Subsection 3.9.2.3 of the DCD.

ANSWER:

The following is MHI's response to RAI on sound attenuation in the SYSNOISE acoustic loading analysis.

(a) Test data from the 1/5 scale model of the J-APWR is inadequate to validate the sound attenuation in the SYSNOISE model because they do not simulate the RCP pulsation.

(b) The sound attenuation included in the SYSNOISE model is discussed in the analysis of acoustic loading in the reactor internals of the US-APWR.

Table 1 shows the mechanisms causing sound attenuation in the reactor vessel.
Test data for the validation of sound attenuation are limited. Therefore, for obvious reasons, only attenuation through perforated plates (spray nozzles, lower core support plate, and upper core plate) was conservatively included in the model. The derivation of acoustic attenuation through perforated plates is described in Document 25-1. The acoustic resistance was obtained with equivalent hole diameter, plate thickness, opening ratios in the spray nozzle, lower core support plate, and upper core plate respectively. At this time, steady flow along the perforated plates was ignored for a conservative evaluation.

Table 2 shows values for the acoustic resistance in each perforated plate derived as shown above.
The actual sound attenuation will be higher due to other attenuation elements shown in Table 1.

Above information will be included in Subsection 3.3.4 (2) c. of the revised version of MUAP-07027-P.

Table- 1 Mechanism by Sound Attenuation in Reactor

Sound attenuation mechanism	Description	Analysis in DCD
Perforated plate	Attenuation of acoustic energy is caused by resistance during passing through holes of perforated plates.	Considered ;spray nozzle, lower core support plate, and upper core plate
Viscosity of fluid itself	Acoustic energy is converted to thermal energy during shear deformation due to viscosity of fluid itself.	Not considered
Friction between fluid and wall	Acoustic energy is converted to thermal energy when fluid moves contacting the wall.	Not considered
Coupling of eddy	Acoustic energy is converted to eddy energy by eddy in fluid with steady flow.	Not considered
Attenuation due to vibration of internals	Converted to kinetic energy of internals	Not considered

Table-2 Value of Sound Attenuation with SYSNOISE Input

Location	Value of acoustic resistance $\mathrm{Re}[Z/(\rho c)]$	Remarks
Spray nozzle		Acoustic impedance has an effect on frequency response.
Lower core support plate		
Upper core plate		

Reference

1 : A.W.Guess 1975 Journal Sound and Vibration 40(1), pp119-137 Calculation of Perforated Plate Liner Parameters from Specified Acoustic Resistance and Reactance

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated: add a new sentence to the end of the 2nd paragraph of Subsection 3.9.2.3.3, "Detailed information, about the acoustic resonance analysis including the discussions about the code verification and assumption of the acoustic damping are included in Subsection 3.3.4 of Reference 3.9-22."

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

4/09/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-26

The applicant presented the details of the vibration analysis of the reactor internals in the MHI technical report MUAP-07027-P.

The staff reviewed Subsection 3.9.2.3 of the DCD and the technical report MUAP-07027-P, and found the DCD and the vibration assessment report did not give a clear overview of the procedure. For example, it is clear that scale model tests were performed and a finite element model of the scale model geometry was developed to validate the FE simulation. However, it is not clear how the dynamic response of the US APWR was calculated. Is it calculated by scaling up the SMT results/simulation, or by means of a finite element simulation of the prototype geometry and forcing functions? If it was the later case, explain the validation procedure of the prototype simulation. The applicant is requested to explain the procedure used to perform the dynamic analysis of the reactor internal structures and to compute the resulting dynamic stresses. The requested information is needed to complete the review of the dynamic responses of reactor internals caused by steady state and operational flow transient conditions and to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to explain more clearly how the dynamic response of the prototype reactor internals was calculated.

ANSWER:

The response to this question is combined in the response to RAI3.9-21.

Impact on DCD

Refer to the response to RAI-3.9-21.

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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RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-27

In DCD Tier 2, Subsection 3.9.2.3, the applicant made a commitment to performing a comprehensive vibration analysis program for the first US-APWR. The DCD and the vibration analysis report MUAP-07027-P, address various excitation mechanisms, including vortex shedding, flow turbulence, and fluid-elastic instability excitations. The strength of these excitation mechanisms is obviously dependent on the value of the cross-flow velocity component.

The staff's review showed that the DCD included only qualitative information about the velocity distributions, and the methodology used to determine the cross-flow velocities was described rather vaguely. Accurate predictions of the cross-flow velocity distributions over various structures of the reactor internals are essential for reliable evaluation of the effects of flow-induced vibration. Also, validation of the cross-flow velocities used in the flow-induced vibration analysis is needed to complete the review of the dynamic responses of reactor internals, including the potential of adverse flow effects. The applicant is requested to explain the methodology used to determine the cross-flow velocity over various components of the reactor internals, such as the cross-flow velocities near the exit nozzles and over the core supporting structures (upper and lower support columns, guide tubes and instrumentation support structures). If the flow velocity distributions were assumed to be similar to the existing 4-loop reactors, provide evidence to substantiate this assumption. Identify the critical structural components with regard to cross-flow induced vibrations (FIV) and explain the method used to deal with non-uniform velocity distribution along the lengths of these components (e.g. whether partial admission factors were used or uniform velocity distributions were assumed). Review of these issues is needed to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to include the methodology used to determine the cross-flow velocity distributions and refer to this methodology in Subsection 3.9.2.3 of the DCD.

ANSWER:

The cross flow velocity around the structure in the lower and upper plenum of the reactor vessel was evaluated in the following manner.

(1) Upper plenum

a. The cross flow velocities in the upper plenum were calculated based on the potential flow theory without structures in the plenum.

b. The Cross flow velocity distribution between the structures were determined based on the equation of continuity and the pitch-to-diameter ratio of the structures

c. When the cross flow is not uniform along the axis of the structure in the upper plenum, the maximum cross flow was used for vortex shedding and fluid elastic instability evaluation.

(2) Lower plenum

a. The cross flow velocity in the lower plenum was assumed to be equal to the downcomer average velocity..

b. The cross flow velocity distribution between the diffuser plate support columns was determined based on the equation of continuity and the pitch-to-diameter ratio of the support columns.

c. When the cross flow is not uniform along the axis, the maximum cross flow was used in vortex shedding and fluid elastic instability evaluations.

The Above information will be included in Subsection 3.2.3 of the revised version of MUAP-07027-P.

Impact on DCD

See attachment 1 for the changes to be incorporated to DCD Section 3.9.

Add the new paragraph at the end of "General" in Subsection 3.9.2.3.3 : "The detail of the evaluation including the determination of the cross flow velocity is discussed in Subsection 3.2.3 of Reference 3.9-22. "

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-28

The applicant stated that the RCPs of the US APWR deliver 27% higher flow rate at 10% higher head than the existing 4-loop reactor coolant pumps. It is not clear how this increase in the pump capacity is accounted for in the acoustic forcing function, or in the acoustic source representing the pump excitation. Section 3.3.4 of the vibration assessment report MUAP-07027-P stated that: "*The RCP pulsation amplitudes at each rotation speed are assumed as shown in Table 3.3-3.*"

The staff reviewed the relevant documents and found that the DCD did not provide sufficient information regarding the effect of increasing the capacity of the RCP on the acoustic excitation generated by the pumps, and that the vibration assessment report MUAP-07027-P did not explain the basis for the assumed pulsation amplitudes. The applicant should substantiate the assumed values of pressure pulsations. Appropriate modeling of the pump acoustic excitation is essential to evaluate the dynamic responses of the reactor internal structures. This information is needed to ensure proper modeling of the pump acoustic excitation, which is used to compute the acoustic and structural responses of the reactor internals. The applicant is requested to describe the effect of using RCP with higher flow rates and delivery heads on the acoustic excitation generated by the pumps. Review of the acoustic sources generated by the RCP is required to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to describe how the higher capacity of the pumps is accounted for in the acoustic excitation source and to substantiate the assumed values of pulsation amplitudes. Refer to this information in Subsection 3.9.2.3 of the DCD.

ANSWER:

The effect of increasing the capacity of the RCP from the current 4-loop was encountered on the evaluation of the US-APWR RCP pulsation load as follows.

(1) The RCP forcing function was determined based on a model test of the RCP for the J-APWR ,the flow rate and the head of which are higher than the current 4-loop plant. The

design flow rate for the J-APWR is the same with the US-APWR as discussed in the response to RAI 3.9.2-43.

(2) The head of the US-APWR RCP is also encountered to the forcing function.

Above information will be included the Subsection 3.3.4 (1) "RCP Characteristics " in the revised version of MUAP-07027-P.

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated to add the new sentence after the 1st sentence of the 2nd paragraph of Subsection 3.9.2.3.3, "The effect of increasing the capacity of the RCP from the current 4 loop was encountered on. the evaluation of the US-APWR RCP pulsation loads. The details about the characteristics of US-APWR RCP pulsation loads are described in Subsection 3.3.4(1) of Ref.3.9-22."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-29

In Subsection 3.3.1 of the vibration assessment report MUAP-07027-P the applicant stated that: "The methodology of the turbulence force generation proposed by Au-Yang (Reference 4) is applied for the down comer forcing function with some modifications."

The staff's review indicated that neither the technical report nor the DCD gave any information as to what were these modifications. Clarification of these modifications is necessary to ensure that an appropriate methodology has been used to develop the flow-induced forcing function in the down comer. The requested information will help the staff evaluate the flow-induced forcing function in the down comer, which is used to compute the dynamic responses of the reactor internal structures. The applicant is requested to describe the modifications made in the methodology suggested by Au-Yang to define the turbulence excitation forces. The reason for introducing these modifications should also be clarified. Review of these methodology modifications is therefore necessary to assure conformance with GDC-1 and 4. Revise the vibration assessment report to include a description of these modifications as well as the reasons for introducing them. Refer to these additions in Subsection 3.9.2.3 of the DCD.

ANSWER:

In the original definition by Au-Yang, the Joint acceptance is a function to determine the relation between the turbulent pressure forcing function and the displacement response. As a result, the joint acceptance integral involves both the coherence function of the pressure field and the structural mode shapes. The coherence function of the pressure field includes a convection velocity term with flow velocities in x and y directions. When the flow is in one direction (eg, x-direction), the convection term disappears in the cross-stream direction (in this case the y-direction).

MHI simplified the Joint acceptance integral as follows:

1. Assumed constant mode shape functions inside the acceptance integral.

2. Assume the downcomer flow is purely axial so that the convection term in the pressure coherence function in the circumferential direction could be eliminated.

Because the joint acceptance involves integration over the entire mode shape, Assumption 1 has only a secondary effect on the joint acceptance. An example of this is shown in Figure 8.5 in Au-Yang's book "Flow-Induced Vibration of Power and Process Plant Components". Assumption 2 is generally valid over most of the downcomer flow surface. Since neither assumption involves the modal frequencies, the modal transfer functions are not affected. Therefore, the above two assumptions have no significant impact on the validity of the original method .

The above discussion will be included in 3.3.1 (1) of the revised version of the Vibration Assessment report MUAP-07027-P.

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated to add the new words after the 2nd sentence of the 1st paragraph of Subsection 3.9.2.3.3, "with some simplifications (more details in 3.3.1 (1) of Refermore3.9-22) ."

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-30

In Subsection 3.4.2 of the vibration assessment report MUAP-07027-P the applicant discussed the alternating stress SaFIV resulting from flow-induced forces. A stress index factor K is used to account for structural discontinuity. The value of the stress index K clearly affects the maximum alternating stress level and the safety margin with respect to the ASME Code of fatigue limit.

The staff review of the relevant documents showed that the DCD did not discuss the stress index K, and the vibration assessment report did not give the values that were used for welds and structural discontinuities. The applicant is requested to discuss the assumed values of the stress index for structural discontinuities (factor K) that are used in the equation of the alternating stress (SaFIV) given in page 46 of the vibration assessment report MUAP-07027-P. In particular, provide the values used for welds and for joints of components with different thicknesses. Review of this safety margin is essential to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to include the value of K used in the alternating stress equation.

ANSWER:

A constant value [5.0] was used as the stress index (stress concentration factor K) for structural discontinuities in the equation of alternating stresses (SaFIV) in accordance with ASME Boiler and Pressure Vessel Code. Section III,.

This value will be added in the high cycle fatigue evaluation in Subsection 3.4.2 (2) in the vibration assessment report MUAP-07027-P, by adding the following new sentence follows.

" (2) High Cycle Fatigue Evaluation
The alternating peak stress due to the flow turbulent and the RCP pulsation is determined from the FEM response as shown in the following equations.

PROPRIETARY

Sa_{FIV} = [4.5] (□rms_{FIV}) K (E/E p)
Sa_{RCP} = (□ 0-P_{RCP}) K (E/E p)

where,
Sa_{FIV} : alternating peak stress due to the flow
Sa_{RCP} : alternating peak stress due to the RCP pulsation
σ rms_{FIV}: RMS amplitude of alternating stress due to the flow
σ 0-p_{RCP} : zero to peak amplitude of the alternating stress due to the RCP
K : stress index for the structural discontinuous ([5] is used)
E : young modulus in the room temperature
Ep : young modulus in the plant operating condition
[4.5] : conversion coefficients for the peak stress from the RMS value

Total of alternating peak stress is assumed as the simple sum of those due to the flow turbulence and the RCP-induced pulsation as following equation.

Sa_{total} = Sa_{FIV} + Sa_{RCP} "

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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Mitsubishi Heavy Industries

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RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-31

In Sub-section 3.3.3 of the vibration assessment report MUAP-07027-P the applicant stated that the power spectral density (PSD) of the down comer is used to calculate the flow-induced vertical forces generated by the flow through the holes in the lower core support plate and in the upper core plate, but did not clarify why this approach was used.

The applicant should elaborate on the differences that may result in the flow-induced vertical forces if the PSD of a jet flow issuing from an orifice is used (instead of that of the down comer), and explain the physical mechanism that causes the atypical discontinuities depicted in the force PSD given in Fig. 3.3-8 of the vibration assessment report. Also, discuss and substantiate the correlation lengths and the joint acceptance coefficients that are used to compute the total integrated vertical forces on the reactor internals. The applicant is requested to explain the reasons for using the power spectral density (PSD) function which is measured in the down comer, rather than the PSD of orifice flow, to calculate the flow-induced vertical forces generated by the flow through the holes in the lower core support plate and in the upper core plate.

The staff recognizes the difficulties involved in responding to this RAI and is therefore prepared to review a response that states the current value of safety margin and gives a realistic estimate of the change in this safety margin that will result by considering a more representative PSD to calculate the flow-induced vertical force. The requested information will help the staff evaluating the flow-induced vertical forces, which are used to compute the dynamic responses of the reactor internal structures. Review of the effect of using a more representative PSD of the pressure fluctuations is therefore necessary to assure conformance with GDC-1 and 4. Revise the vibration assessment report to include a brief description of the effect of the PSD characteristics on the margin of safety.

ANSWER:

MHI applied the same measured pressure data in the downcomer to the lower core support and upper core plate flow holes. The justification for this premise is based on the assumption that the pressure fluctuation close to the RPV inlet nozzle is caused by jet flow turbulence exiting from the inlet nozzle and, therefore is assumed to be similar to the jet flow turbulence through the lower core support plate and upper core plate flow holes.

A measured pressure PSD of .the circular jet as shown in Figure1., comparing with the US-APWR forcing function based on downcomer data as shown in Figure 2. Both PSD is similar in the shape and absolute values. From this, it is adequate that the downcomer PSD close to the inlet nozzle is used for the flow hole forcing function.

The joint acceptance or correlation length is not defined. The total force on the plate was calculated as the SRSS of all flow holes in the plate, because the Jet flow turbulence in each flow hole should be independent from the others.

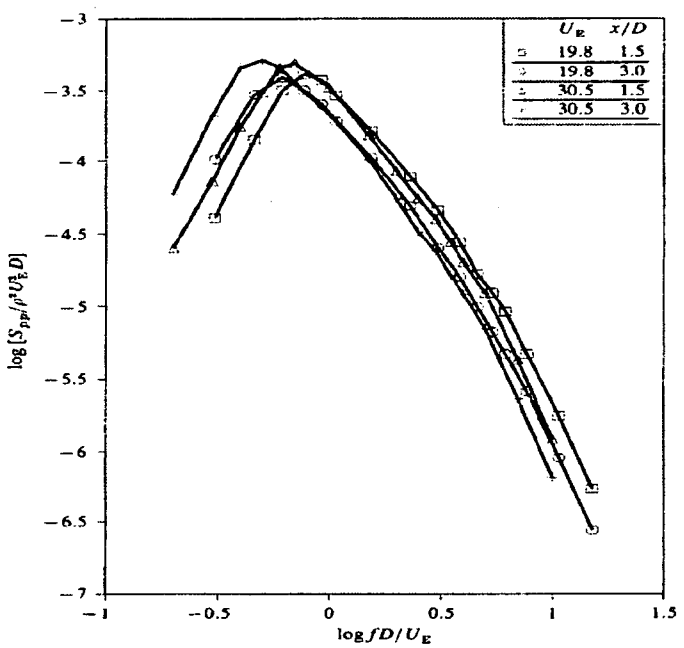

Figure 1 - Normalized pressure PSD measured in a jet flow (Ref.1)

Figure 2 - Normalized Pressure PSD for US-APWR Vertical Loads

Reference

(1) Pressure spectra in turbulent free shear flows, Journal of Fluid Mechanics, 1984, vol. 148, pp. 155-191

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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Docket No. 52-021

RAI NO.:	**NO. 272-1585**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-32

Section 3.9.2 of the SRP and RG 1.20 recommend the validation of all numerical models and forcing functions that are used in the design process. Obviously, the reliability and associated bias and uncertainty errors of these tools depend on the acceptance criteria of their predictions. The staff is concerned about the rigor of the acceptance criteria used by the applicant to validate the computational models and forcing functions. To explain the gist of this RAI, the applicant is referred to Table 3.3-5 and Fig. 3.4-2 of the vibration assessment report MUAP-07027-P. In the table, the validation is judged acceptable although the reactor design is much more complex than the simple annulus/cylinder geometry used to validate the SYSNOISE model, and in Fig. 3.4-2, the analysis predictions of the core barrel and neutron reflector displacements are judged acceptable although their predicted PSDs do not seem related to those measured. The acceptance criteria for the prediction and validation of structural models (FE programs), acoustic excitations (SYSNOISE) and forcing function definitions need to be clarified to ensure that the reactor internal structures are designed to quality standards commensurate with the importance of their safety functions.

The DCD does not provide any information about the acceptance criteria used to validate the computational tools used to analyze the dynamic responses of the reactor internal structures. In addition, the comparisons provided in the report are not adequate to confirm validation of the analysis tools. The applicant is requested to discuss the acceptance criteria used to evaluate the suitability of the analytical and/or computational tools to calculate the dynamic responses of the reactor internal structures. The staff requests this information to assure conformance with GDC-1 and 4. Revise the comprehensive vibration report to adequately address the acceptance criteria of the analysis results and refer to the acceptance criteria in Subsection 3.9.2.3 of the DCD

ANSWER:

For the discussion regarding validation of the SYSNOISE code, please refer to the response to RAI 3.9.2-24.

In the simulation analysis of the J-APWR 1/5 Scale Model Test , the category 2 acceptance criteria discussed in the response to RAI3.9.2-70 was applied as follows.

a. Natural frequency for fundamental beam mode and lowest shell mode :with in 10%
b. Random response (displacement, stress etc) :factor of 3

The analysis results show in the Figure 1 (Fig 3.4-2 of MUAPo70027-P) Were adjudged acceptable based on the following rationale:

(1) The ratio of the analysis RMS response and measured one is [ ](Table 3.4.2 of MUAP-07027-P) . It is in the factor of 3 as the acceptance criteria for the random response.

(2) The dominant frequency of the neutron reflector is observed around [ ] Hz both in the analysis and measured data. The difference is in the 10% as the acceptance criteria for the natural frequency.

(3) The response peak around [ ] Hz in the measurement was identified to be related to the test vessel mode. We accepted the analysis response because this mode should be compressed in actual plant with much higher stiffness of the reactor vessel than test condition.

Above discussion will be included in Subsection 3.4.1 of the revised version of MUAP-07027 -P.

Figure.1 NR Top / CB Relative Displacement Linear Spectral (Test Scale)

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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SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-34

The applicant stated in the MHI technical report MUAP-07023 that the scale model tests were performed with either 77 or 85 Guide Tubes (GTs).

The staff reviewed the technical report and found that the applicant did not explain why these numbers of GTs were chosen for the model tests. The applicant is therefore requested to explain the reason(s) of choosing these numbers of GTs for the model tests and to elaborate on the significance of these numbers of GTs, in comparison to the total numbers of GTs that will be used during normal operation of the US-APWR. The requested information is needed to assure conformance with GDC-1 and 4.

ANSWER:

Both the 77-GT and 85-GT configurations selected for the J-APWR scale model test were based on requests from Japanese customers. In general, increasing the number of GTs leads to an increase in the blockage factor and cross flow loads. Therefore, the cross flow loads on the GTs in the US-APWR, with its 69-GT design, are bounded by the test results of J-PWR SMT.

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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SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.3**
DATE OF RAI ISSUE:	**03/10/09**

QUESTION NO.: RAI 3.9.2-35

The applicant stated in Table 2.1-1 of the vibration assessment report MUAP-07027-P that the flow velocity in the vessel exit nozzle of the US-APWR will be increased, compared with the current 4-loop reactors.

The technical report does not provide any discussion regarding possible adverse flow effects on the structural components that will be subjected to flow velocities higher than those in the 4-loop reactor. FIV analysis of the structural components which are exposed to higher flow velocities than in the 4-loop reactors is needed to ensure that these structures are designed to quality standards commensurate with the importance of their safety functions. The applicant is requested to discuss the analysis performed and the tests planned to demonstrate that adverse flow effects will not cause unanticipated excessive flow-induced vibrations or structural damage to the *reactor piping systems and the internal structures in the upper core plenum near the exit nozzles*. The requested information is needed to assure conformance with GDC-1 and 4.

ANSWER:

The technical report tried to make the point that the structural components have large structural and high cycle fatigue margins of safety to accept higher flow-induced vibration loadings and cross-flow loads loadings from the higher flow velocities. Furthermore, DCD 3.9.2.6 states: "The results from the prediction analysis of US-APWR will eventually be compared with the measured data from the pre-operational vibration test, described in Subsection 3.9.2.4, in accordance with the guidelines given in Regulatory Guide 1.20 (Reference 3.9-21). Any discrepancies between the predicted and measured values will be accounted for and fully explained. If necessary the input parameters, such as the turbulent forcing function and the damping coefficient, in the vibration analysis will be adjusted in accordance with the measured data and the analysis repeated to resolve the difference between the analytical and measured results."

Also, it is concluded that the structures in the upper plenum have sufficient margins of safety for the adverse flow effect due to the cross flow such as the vortex shedding lock-in or fluid elastic instability, as discussed in Subsection 3.2.3 of the vibration assessment program report MUAP.07027-P.

It is concluded that the structures in the upper plenum have sufficient margins of safety for the adverse flow effect due to the cross flow such as lock-in vortex induced vibration or fluid elastic instability, as discussed in Subsection 3.2.3 of the vibration assessment program report MUAP-07027-P.

The same conclusion is also described in DCD Subsection 3.9.2.3.2 as follows:

"Upper Reactor Internals Assembly

The reduced velocity U/fnD for the upper plenum structures is slightly higher than that of the current 4-loop plant, but sufficient margins of safety are maintained for cross-flow induced vibrations, such as fluid elastic instability and turbulence-induced vibration. In addition, since the vortex shedding frequencies are lower than 70% of the structural fundamental frequencies, lock-in vortex-induced vibration is avoided per ASME Code, Section III (Reference 3.9-1) Appendix N-1324. "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals" (Reference 3.9-22) provides further details".

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.

The fundamental modal frequencies of the upper support column and the lower RCCA guide tube are the same as those in the current 4-loop plant because the basic dimensions of these structures are not changed. The top slotted column, located at the peripheral part of upper plenum, is another support column with larger diameter and higher fundamental mode frequency. The reduced velocity U/f_nD for the upper plenum structures is slightly higher than that of the current 4-loop plant, but sufficient margins of safety are maintained for cross-flow induced vibrations, such as fluid elastic instability and turbulence-induced vibration. In addition, since the vortex shedding frequencies are lower than 70% of the structural fundamental frequencies, lock-in vortex-induced vibration are avoided per ASME Code, Section III (Reference 3.9-1) Appendix N-1324. "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals" (Reference 3.9-22) provides further details.

3.9.2.3.3 Quantitative FIV Analysis of the US-APWR

Turbulent fluctuating pressure has been identified as the main source of excitation of reactor internals during normal operation. The methodology proposed by Au Yang and Connelly, "A Computerized Method for Flow-Induced Random Vibration Analysis of Nuclear Reactor Internals" (Reference 3.9-23), is used to assess the responses of the core support barrel to the turbulence flow in the down-comer with some simplifications (more detail can be found in Subsection 3.3.1(1) of Reference 3.9-22).[RAI3.9.2-29] The methodology for cross-flow induced vibration analysis of tube structures in the lower and upper plenums is based on the ASME Code, Section III (Reference 3.9-1), Appendix N, "N-1300 Flow-Induced Vibrations of Tubes and Tube Banks." The detail of the evaluation including the determination of the cross flow velocity is discussed in Subsection 3.2.3 of Reference 3.9-22.[RAI3.9.2-27]

Acoustic pressure fluctuations generated by the RCP are assumed to be similar to those generated by RCP used in the current plant, because the key parameters governing RCP-induced pressure pulsations, such as the pump rotational speed and the number of impellers are the same. The effect of increasing the capacity of the RCP from the current 4 loop was encountered on. the evaluation of the US-APWR RCP pulsation loads. The details about the characteristics of US-APWR RCP pulsation loads are described in Subsection 3.3.4(1) of Ref.3.9-22.[3.9.2-28] Forcing functions due to RCP are determined taking into consideration acoustic resonance modes in the reactor vessel. Detailed information, about the acoustic resonance analysis including the discussions about the code verification and assumption of the acoustic damping are included in Subsection 3.3.4 of Reference 3.9-22. [RAI3.9.2-24,25]

FE analysis with computer code ANSYS (Reference 3.9-7) is used to calculate the flow-induced vibration amplitudes of the reactor vessel and internals. The methodology of structural modeling and forcing function assessment have been confirmed by a simulation analysis of the APWR 1/5 scale model flow test (Reference 3.9-24) in the same manner. The computed vibration response of the core barrel with the best estimate damping coefficient agrees with measured results. Detail of the model validation including the bias error and uncertainties is discussed in Subsection 3.2.1 of Reference 3.9-22. [RAI-3.9.2-23] Discussion about the dimensionless parameters and the ratio of

forcing functions in the scale model test and those under plant operating condition are included in Appendix XX of Reference 3.9-22. RAI3.9.2-21,22]

For the prediction analysis of US-APWR reactor internals vibration responses, the damping coefficient smaller than the best estimate value is used to assure a conservative evaluation.

Refer to the "Comprehensive Vibration Assessment Program for US-APWR Reactor Internals" (Reference 3.9-22) for further details.

From the analysis results, the following conclusions are obtained.

- Alternating stress levels of reactor internals due to flow-induced vibrations are acceptably low in comparison with the limit for high cycle fatigue.