MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

January 15, 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-10008

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 59, 60, 63, 67, 72 through 74, 76 through 79, 81 and 83 of the RAI (Reference 1). MHI requests to change the response time to 60 days for questions 61, 62 and 82 of this RAI. The responses to questions 61, 62, 64 through 66, 68 through 71, 75, 80, 82, and 84 will be issued at a later date (ie-60 days) 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 "[ ]".

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.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Response to Request for Additional Information No. 498-3782, Revision 0 (45-day response, Proprietary Version)

3. Response to Request for Additional Information No. 498-3782, Revision 0 (45-day response, Non-Proprietary Version)

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

Contact Information

C. Keith Paulson, Senior Technical Manager
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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-10008

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 (45-day response)," dated January 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.

   B. 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 15th day of January 2010.

Y. Ogata

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

Docket No. 52-021
MHI Ref: UAP-HF-10008

Enclosure 3

UAP-HF-10008
Docket No. 52-021

Response to Request for Additional Information No. 498-3782, Revision 0 (45-day response)

January 2010
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/15/2010

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 03.09.02-59:

The staff reviewed the response to RAI 3.9.2-10 (#205-1584, dated 4/30/2009, ML091240113, MHI Ref: UAP-HF-09184) and noted that the equivalent static load method of analysis is the preferred method for use in seismic analysis of subsystems such as equipment and piping anchorages. The staff also notes that SRP Section 3.9.2, Revision 3, SRP Acceptance Criteria 2.A.(ii) states, "An equivalent static load method is acceptable if: (1) There is a justification that the system can be realistically represented by a simple model and the method produces conservative results in responses. ... (2) The design and simplified analysis account for the relative motion between all points of supports. (3) To obtain an equivalent static load of equipment or components which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used with adequate justification." The applicant did not provide detailed technical information to demonstrate how these three criteria were being satisfied.

The applicant is requested to provide detailed technical information to demonstrate how the three SRP Section 3.9.2, SRP Acceptance Criteria 2.A.(ii) are satisfied. In accordance with the MHI commitment in the response to Question 3.9.2-10, include in a future DCD Revision a list which would summarize the method for determining the stiffness, the related assumptions, and the procedure for verification of the assumptions for all of the anchorage types considered for use on the US-APWR. A copy of the list should be provided, and the DCD revision which will contain this list should be identified, in your response.

ANSWER:

The NRC Staff submitted similar questions relating to "Equivalent Static Analysis" methods in Questions 3.7.3-02, 3.7.3-03, 3.7.3-04 and 3.7.3-15 of RAI 213-1951 and Questions 3.8.4-29 Part 6, 3.8.4-30 Part 6b, and 3.8.4-31 Part 7b of RAI 342-2000. As stated in MHI's responses to these questions and as stated in DCD Subsections 3.7.3.1 and 3.12.3.6, MHI has made commitments to comply with the three above quoted SRP Acceptance Criteria, which are stated in SRP Acceptance Criteria Subsections 3.7.2.II.1.B, 3.7.3.II.1 and 3.9.2.II.2.A(ii). Please refer to the responses to these questions in RAI 213-1951 and in RAI 342-2000 for further discussions.

The anchorage systems used for the US-APWR consists of various combinations of anchor types and base plate arrangements as discussed and listed in MHI's original response to RAI 3.9.2-10. The types of anchorage systems used may be altered or expanded as detailed subsystem design is finalized, and the calculated stiffnesses are also dependent of the finalized details. The list which summarizes the anchorage types, the method for determining their stiffnesses, the related assumptions and the verification of the assumptions can only be prepared after all anchorage systems are finalized. Upon completion, this list will be incorporated into a future revision of the DCD.

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

1/15/2010

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 03.09.02-60:

In its review of the MHI response to RAI 3.9.2-12 (#205-1584, dated 4/30/2009, ML091240113, MHI Ref: UAP-HF-09184) the staff finds the applicant has presented a reasonable explanation of how the coupled lumped mass model of the RCL was validated, including comparison of analytical calculations and with the results of the testing done at the large seismic shake table at the well known Tadotsu Engineering Laboratory in Japan. However, the applicant did not give any specific data as to the frequency inputs for these tests and did not present any rationale that shows that these comparisons will apply to sites with high frequency inputs. Furthermore, the applicant did not provide any technical information to validate the lumped mass stick models. For the staff to accept fully the response to this question, the applicant is requested to explicitly state that this coupled model has been validated for these high frequencies.

ANSWER:

The test results of the seismic proving test for the RCL at Tadotsu Engineering Laboratory contains the high frequency response up to 50 Hz, while the input excitation waves are selected with three kinds of floor response target waves at reactor building (R/B) from the typical seismic design conditions in Japan with the time pitch of 7 msec in each wave. The summary on the seismic reliability of the test was reported in the SMiRT paper [1)]. The more detailed description for the RCL seismic proving test was presented in the NUPEC report [2)].

Reference

1). H. Akiyama et al., "Proving Test on the Seismic Reliability for the PWR Primary Coolant Loop System", 11th SMiRT, Vol. K, 1991.

2). "Proving Tests on the Seismic Reliability for Nuclear Power Plant – PWR Primary Coolant Loop System", Nuclear Power Engineering Center, March, 1991.

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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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 03.09.02-63:

In the response to RAI 3.9.2-19 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the applicant dealt with the dynamic analysis of the steam generator upper internals and did not address the lower components such as the tube bundles and the U-tubes. MHI is therefore requested to provide appropriate vibration analysis for the steam generator lower internals, including the tube bundles and the U-tubes which are exposed to cross and axial flows. If the design of the SG lower internals is not prototypical, it suffices to refer to in-service SGs with similar design, size and flow conditions.

ANSWER:

The topic of steam generator (SG) Tube Bundle is addressed in MHI US-APWR DCD Subsection 5.4.2.1.2.6.

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

1/15/2010

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 03.09.02-67:

In the response to RAI 3.9.2-21 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the applicant provided a comparison between the empirical normalized forcing function (PSD) in the downcomer and that of the 1/5 SMT. In this comparison, the turbulence PSD in the upper portion of the downcomer is about an order of magnitude higher than the upper bound of the empirical PSD. The applicant is requested to explain how this large difference is accounted for in estimating the forcing function of the US-APWR. In particular, the applicant is requested to elaborate on the axial and circumferential distributions of the forcing function.

ANSWER:

(Note: Response to RAI 3.9.2-76 is also included from the third paragraph of this answer.)

The empirical normalized forcing functions (PSD) in the downcomer by Au-Yang (Ref.1) are shown in Figure 1. Please note that both the empirical functions and the field data are based on a measured point in the middle of the downcomer. "Upper bound" does not mean the upper bound of the entire dowmcomer but the envelope of spectral peaks at that single measuring point.

The inlet nozzles are located in the upper portion of the downcomer, where the magnitude of pressure PSD is 3 -4 times larger than that at the mid-section of the downcomer. This difference is reasonable because the local flow velocity in the inlet nozzle is about twice the average velocity in the downcomer, resulting in 4 times larger in the dynamic pressure level.

As for the discussion of the axial and circumferential distributions of the forcing function for US-APWR, please refer to Figures 3 and 4. The PSDs of the downcomer pressure fluctuation measured in the US-APWR 1/7 Scale Model Test are shown in Figure 3. Four PSDs identified with symbols "A" to "D" associated with the pressure transducer locations are shown in Figure 4.

Location "A" is nearest to the inlet nozzle so the PSD is much higher than the others for reasons discussed above.

The mapping of the pressure forcing functions is also show in Figure 4. The entire surface of the core barrel facing the downcomer was divided into 16 segments with 2 elevations and 8 around the circumference, which correspond to the nozzle layouts. For example, the forcing functions for the upper segments facing the inlet nozzles were generated from the same PSD "A" by the inverse Fourier Transform Method. But the time histories for the 4 segments with this PSD were statistically independent of one another and with random phase. In the same manner, a total of 16 time histories were defined from the 4 PSDs.

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

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

Figure 3 Down comer Pressure PSD from US-APWR1/7 Scale Lower Plenum Test
(Figure 3.2.3.4 of MUAP-07027-P (R1))

Figure 4 Downcomer Forcing Function Mapping

Reference

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

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 03.09.02-72:

In the response to RAI 3.9.2-33, several issues are still unclear. Table 3.1 of the revised Report MUAP-07023-P (R1) indicates that no scaling is needed to convert the strain and stress from the SMT measurements to the J-APWR This does not seem appropriate since the SMT.and J-APWR are not identical in size or flow conditions. In Tables 6.8 to 6.14 of the same report, the method of strain and stress conversion is not clear, and in Tables 6.2 and 6.3, the conversion of measured displacement to the J-APWR is not explained. In addition, the source of the stress equation for high cycle fatigue, which is cited in page 4, is not given. The applicant is requested to substantiate the methods used to convert/scale the displacement, the strain and the stress from the SMT data to the full-scale J-APWR.

ANSWER:

In general, in addition to geometric scaling, adjustments due to differences in the fluid mass densities and Young moduli are needed to convert flow-induced dynamic responses from a scale model test at room temperature to those in the full-size reactor under plant operating conditions, as follows:

$$D_P = \frac{D_T}{Sc}\frac{\rho_P}{\rho_T}\frac{E_T}{E_P}$$

$$\varepsilon_P = \varepsilon_T \frac{\rho_P}{\rho_T}\frac{E_T}{E_P}$$

$$\sigma_P = \sigma_T \frac{\rho_P}{\rho_T} = \varepsilon_T E_T \frac{\rho_P}{\rho_T}$$

Where,

D : displacement (mm)

ε : strain (mm/mm)

σ : stress (kgf/mm2)

ρ : fluid density (kg/mm3)

E : Young moduli (kgf/mm2)

suffix P : in plant operating conditions

T : in Test conditions

Sc : geometric scale ratio (= 1/5 for J-APWR SMT)

The dynamic pressures in the room temperature test is 30-40 % higher than those under plant operating conditions as determined by the ratio of fluid mass densities (1000 kg/m3 at room temperature and 660-750 kg/m3 under plant operating conditions). On the other hand the stiffness of the structure under plant operating conditions is reduced by 10% from that at room temperature in accordance with the difference in the Young moduli.

In the data reduction process of the J-APWR 1/5 SMT, the difference in the fluid mass densities and Yong moduli due to temperature difference were intentionally ignored for a conservative bias.

Due to a combination of the dynamic pressure and stiffness effects as discussed above, the dynamic responses such as displacement, acceleration, stress and moment at room temperature are 20-30 % higher than those under plant operating conditions because of the effect of temperature difference.

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

1/15/2010

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 03.09.02-73:

In the response to RAI 3.9.2-33, several issues are still unclear. Several parameters and definitions are not clear in Table 6.4 of the revised Report MUAP-07023-P (R1). The applicant is requested to explain:

(a) The procedure of converting the moment from the SMT to the J-APWR

(b) The meaning of the term "design load", especially when this design load is lower than that measured from the SMT

(c) When and how the design load will be determined for the bottom mounted instrumentation nozzle.

ANSWER:

(a) The relationship between the loading moment and the measured strain was obtained by a series of unit loading tests for the column structures, which have been performed as part of the flow test. The moment at the end of the column was derived from the measured strain and correlation factor determined by the unit loading test.

(b) The design load had been determined by analysis or hand calculation in the process of sizing the structural components. In case a design load is lower than that measured in the SMT, the need for revising the design load should be evaluated. For example the moment of the RV (water) level instrumentation support tube is lower than the measured value as shown in Table 1 (Table 6-4 of MUAP-07023-P (R1)), but the design load is not revised because this component is not located in a high cross flow region in the upper plenum and the corresponding measured stress is much lower than the allowable limit as shown in Table 2 (Table 6-7 of MUAP-07023-P (R1)).

(c) The determination of the design load of the J-APWR bottom mounted instrumentation nozzle was not performed because of its very llimited length exposed to the cross flow. The corresponding measured stress is much lower than the allowable limit as shown in Table 2 (Table 6-7 of MUAP-07023-P (R1)).

Table 1 (Table 6-4 of MUAP-07023-P (R1))

Conversion of Test Flow Loads to J-APWR Reactor Internals Flow Loads (77 GTs, 120% Flow)

Components	Location	Test results			Conversion to J-APWR	Design loads
		Static moment (kgf-mm)	Dynamic Moment[(3)] (kgf-mm)	Total moment (kgf-mm)	Moment (kgf-mm)	Moment (kgf-mm)
RCC guide tube	D-2					
Upper support column	F-3					
Top slotted column	F-1					
Mixing device	E-1					
RV level instrumentation support tube	A-5					
Secondary core support column	L-9					
Base of bottom mounted instrumentation guide tube	A-9					
	S-8					
Bottom mounted instrumentation nozzle	A-9					

(1): Impossible to evaluate because of very minor strain below measurement limit
(2): Design load is not determined
(3): Axial component of dynamic load was also estimated (bending component conservative estimate because of larger variation)

Table 2 (Table 6-7 of MUAP-07023-P (R1))

Conversion of Test Strains to J-APWR Stresses for Evaluation of the Reactor Internals (77 GTs, 120% Flow)

Evaluated components(1)		Measured strain (μs)			Conversion to J-APWR Stresses	
		Static	rms	Maximum strain(1) Static + rms × 3	Stress (2) (kg/mm²)	Allowable stress (kg/mm²)
Core barrel	Outer surface					
Upper core support skirt	Inner surface					
RCC guide tube	D-2					
Upper support column	F-3					17.6
Top slotted column	F-1					
RV level instrumentation support tube	A-5					
Secondary core support column	L-9					
Bottom mounted instrumentation guide tube	A-9					
Bottom mounted instrumentation nozzle	A-9					25

(1): In case of the J-APWR evaluation, factor (ratio of rms to peck) 3.0 was used to calculate the maximum strain

(2): Young's modulus of 304 stainless steel at ambient temperature=19900kg/mm²
Young's modulus of Alloy 690=21000kg/mm²

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

1/15/2010

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 03.09.02-74:

In the response to RAI 3.9.2-33, several issues are still unclear. In Section 6.1 of the revised Report MUAP-07023-P (R1), MHI states:

> "These natural frequencies, after scaling up to the J-APWR reactor internals in water were shown in Table 6-1, then test results were compared with the J-APWR pre-analysis results to confirm the adequacy of the J-APWR 1/5 test models".

The NRC Staff believes that one of the objectives of the SMT is to validate the dynamic analysis, and not to use the dynamic analysis to confirm the adequacy of the small-scale test models. The applicant is requested to explain what is meant by the above cited statement.

ANSWER:

1. Test Objective for the J-APWR design

First, we must clarify the back-ground of the J-APWR 1/5 SMT and the historic evolution of the validation process.

The J-APWR 1/5 scale model was performed in 1996 as the design confirmation of the J-APWR. MUAP-07023-P (R1) is the English translation version of the original Japanese report written in 1996. The pre-analysis was performed before the test with actual plant dimensions and with a FEM code developed in Japan. In the test procedure, it was true that the scale model test results were cross-checked by comparison with the predicted natural frequencies from the FE analysis as described in Section 6.1 of MUAP-07023-P (R1).

2. Use of J-APWR SMT results for US-APWR design

In the design process of US-APWR, the J-APWR SMT results are used for the verification of dynamic analysis method through the benchmark analysis described in Chapter 3 of MUAP-07027-P (R1).

The benchmark analysis was performed in 2006. For this purpose, a 1/5 scale dimensions model was re-constructed from the original J-APWR actual plant model described above. Any fine tuning with the test results was not included. The FEM code was changed to ANSYS which was also used for the US-APWR analysis.

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 03.09.02-76:

In the response to RAI 3.9.2-33, several issues are still unclear. The power spectral density (PSD) of the turbulence excitation in the downcomer is stronger near the inlet nozzles and becomes weaker as the flow progresses along the downcomer, as illustrated, for example, in Fig. 3.2.2-3 of the revised Report MUAP-07027-P (R1). The applicant is requested to explain the axial and circumferential distributions of the turbulence excitation PSD which are used in the dynamic analysis of the reactor internals.

ANSWER:

Response to this RAI is included in the ANSWER to RAI 3.9.2-67.

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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1/15/2010

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 03.09.02-77:

In the response to RAI 3.9.2-33, several issues are still unclear. In the introduction of the revised Report MUAP-07027-P (R1), as well as at several other sections of the report, the applicant states:

> In the first version of the report "measured data in the J-APWR scale model test was used for the forcing functions due to the downcomer flow turbulence. After the completion of Revision 0 of this report at the end of 2007, new data pertinent to the US-APWR configuration was obtained in the US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test".

The applicant is requested to explain why the SMT of the J-APWR in the revised reports was not entirely replaced by the available SMT of the US-APWR. It is also appropriate to revise the DCD document to include and to refer to the results of the 1/7 scale model tests of the US-APWR.

ANSWER:

Only data pertaining to the lower plenum from the US-APWR 1/7 scale lower plenum test was included in the vibration assessment report because of the following reasons:

1. The test model was set up-side down to give easy accessibility to the lower plenum structures and flow visualization. In addition, the fuel assemblies, the radial reflector and the upper core internals were not included in this model. Thus flow-induced vibrations of these structures were not within the scope of this test, which was designed specifically for the lower plenum structures.

2. The flow paths from the vessel inlet nozzle to the downcomer and the lower plenum were simulated in the Lower Plenum 1/7-scale Model Test, and the layout and dimensions of these configurations were not changed from J-APWR. Therefore, the pressure fluctuation data of the downcomer measured in US-APWR lower plenum test can be applied both to the simulation analysis of the J-APWR SMT and the US-APWR prediction analysis.

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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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 03.09.02-78:

In RAI 3.9.2-35, the applicant was asked 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.* In the response (#272-1585, dated4/9/2009, ML091040693, MHI Ref: UAP-HF-09149) to this RAI, MHI addressed the upper core plenum internals only and did not discuss the reactor piping systems. Therefore, RAI 3.9.3-35 still stands for the piping system and MHI is requested to discuss the analysis performed to assess adverse flow effects on the reactor piping system due to the increase in the flow velocity represented by the vessel outlet nozzle as identified in Table 2.1-1 of MUAP-07027-P (R1). The applicant may refer to other sections of the DCD or to technical reports which address the concerns expressed in this RAI.

ANSWER:

The report of MUAP-07027-P (R1) describes that the flow velocity in the vessel exit nozzle of the US-APWR will be increased in comparison with the current 4-loop reactors. The effects of higher flow velocity on the piping system have been confirmed based on the analysis of the current 4-loop reactors. The result shows the structures in the piping have sufficient margins of safety for the vortex shedding lock-in and fluid elastic instability. Therefore, it is concluded that the flow velocity increase has no impact on the instability of vibration of the piping system.

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

1/15/2010

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 03.09.02-79:

The staff finds this response to RAI 3.9.2-40 (#206-1576, dated 3/27/2009, ML090910123, MHI Ref: UAP-HF-09116) acceptable and agrees that since similar steam generators have been in use in existing plants for many years without any vibration problems, there is no need to perform startup testing of the steam generators.

However, contrary to the statement in the response, the staff could not find any reference in DCD Subsection 3.9.2.4.1 to Subsection 5.4.2.1.2.10 that addresses the SG dynamic response. The applicant is, therefore, requested to add the cross reference to Subsection 3.9.2.4.1 mentioned above.

The applicant is also requested to include a reference to the statement that the design of the US-APWR steam delivery system, including the SG upper internals, the safety relief valves, and the steam lines, has been operating in the USA for more than 20 years in sizes and flow rates that bound those of the US-APWR. These additions are requested so that the DCD document meets the expectations of US NRC RG 1.20 and SRP 3.9.2.

ANSWER:

The cross reference from Subsection 3.9.2.4.1 to Subsection 5.4.2.1.2.10 is described in the last paragraph of DCD Rev. 2, Subsection 3.9.2.4.1.

The similar design of the US-APWR steam delivery system has been operating almost 20 years as shown in FSARs (e.g., Comanche Peak and Alvin Vogtle).

Flow velocity is an important factor relative to vibration, and the flow velocity of the US-APWR steam delivery system is described below:

- Main steam piping: approximately 150 ft/s
- Main steam safety valve inlet piping: approximately 500 ft/s

Main steam piping and main steam safety valve inlet piping have almost the same flow velocity as current existing, 4-loop plants in U.S.

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

1/15/2010

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 03.09.02-81:

In its review of the applicant's response to RAI 03.09.02-35 (#272-1585, dated 4/9/2009, ML091040693, MHI Ref: UAP-HF-09149), the staff noted that the description of the models is adequate. There are two items that need further discussion: (a) how local SG shell flexibility at piping nozzles is considered in the model for SG component supports, and (b) how the decoupling criteria in SRP 3.7.2 was considered and applied to the separate analysis of the upper structure from the steam generator (SG) model.

Therefore, the staff requests the applicant to provide the following information:

(a) Provide a description of how the steam generator (SG) shell flexibility was considered in the analysis in the vicinity of piping nozzle penetrations.

(b) Provide a description of how the decoupling criteria in SRP 3.7.2 SRP Acceptance Criteria II.3.B was considered and applied to justify the analysis of the upper internal structure separately from the SG shell model. If any deviations were made from the SRP 3.7.2 criteria provide the rationale for such deviations.

ANSWER:

(a) Local flexibility of the SG shell at the primary nozzle connections is considered as a six degrees of freedom spring connecting between the end node of Hot Leg elbow and the tip node of SG inlet nozzle, and between the tip node of SG outlet nozzle and the node of outlet elbow of Cross Over Leg, in the whole stick mass spring model for the Reactor Coolant Loop as shown in Figure 1 of Technical Report MUAP-08005, Rev. 0.

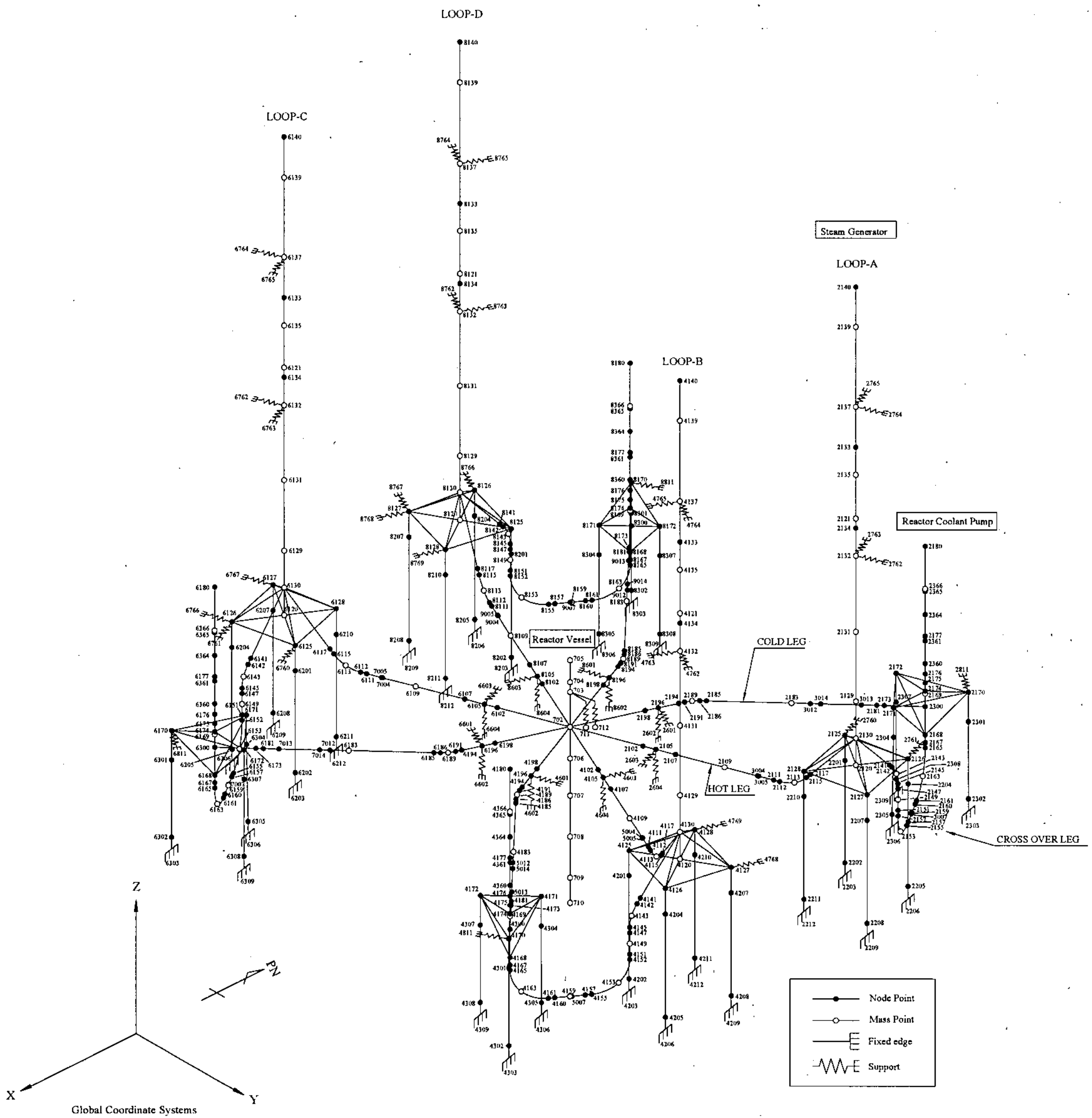

Figure 1 Stick Mass Spring Model for Reactor Coolant Loop

The SG shell flexibility of six degrees of freedom spring is evaluated by the local translational or the rotational flexibilities obtained by the finite element model of SG lower shell as follows;

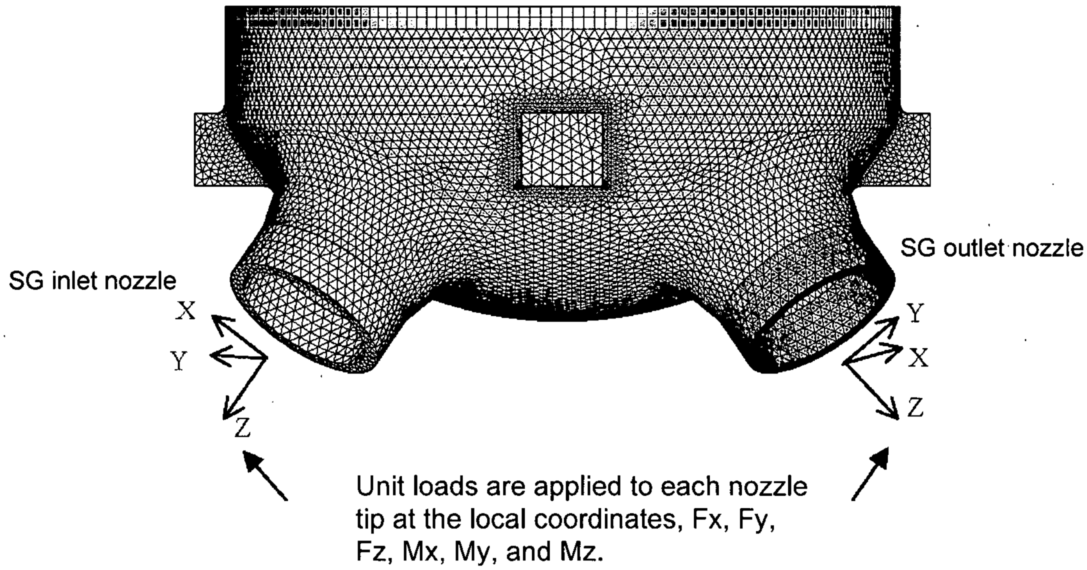

Figure 2 Finite Element Model for SG Lower Shell

(b) The model of the SG upper shell internal structure is decoupled from the SG shell because the mass of the upper internal structure is smaller than the mass of the SG and the dominant frequency of the upper internal structure is higher than the one of SG and their ratios satisfy the decoupling criteria of SRP 3.7.2 as shown in the following table.

	Mass	Frequency	Note
SG Upper Internal Structure			
SG (including Upper Internal Structure)			
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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/15/2010

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 03.09.02-83:

In MHI's response to US-APWR DCD RAI No. 03.09.02-39, 214-1920, dated April 30, 2009 (MHI Ref: UAP-HF-09190, ML091240403), the applicant stated that a list of damping values used for each of the major mechanical components analyzed is provided in US-APWR DCD Tables 3.7.3-1(a) and (b). The SSE analysis for the CRDM used a damping value of 4 percent, and not 5 percent. The staff finds the applicant's response acceptable because the applicant stated that SSE analysis for the CRDM used a damping value of 4 percent, and not 5 percent. However, the applicant did not mention in its response that in the DCD Table 3.7.3-1(a) the damping value for the control rod drive mechanism (CRDM) will be changed. The applicant is requested to revise the CRDM damping value in DCD Table 3.7.3-1(a) and submit the revised DCD for staff review.

ANSWER:

In Table 3.7.3-1 (a) of DCD, 4 percent damping ratio for CRDM is not directly specified but assumed for one of the "Welded and friction bolted steel structures and equipment" in this table. Therefore, there is no need to revise the CRDM damping in Table 3.7.3-1 of DCD.

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.