MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 14, 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-10332

Subject: Amended Response to US-APWR DCD RAI No. 646-5065 (SRP 03.09.02)

References: 1) "Request for Additional Information No. 646-5065 Revision 0, SRP Section: 03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components, Application Section: 3.9.2" dated 10/7/2010.
2) "MHI's Response to US-APWR DCD RAI No.646-5065 (SRP 03.09.02)', MHI Ref. UAP-HF-10310, dated 11/11/2010.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Amended Response to Request for Additional Information No. 646-5085."

Enclosed are the amended responses to the question 03.09.02-92 of the RAI (Reference 1). In this response, Section 2 is added on the original response transmitted by Reference 2 to include the requested information by NRC in the confirmation call in November 17.

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.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata
2. Amended Response to Request for Additional Information No.646-5065 Revision 0 (Proprietary)
3. Amended Response to Request for Additional Information No.646-5065 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-10332

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 "Amended Response to Request for Additional Information No. 646-5065, Revision 0", dated December, 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 analyses used by mathematical models developed at significant cost to MHI. 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.

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 14th day of December 2010.

Y. Ogata

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

Enclosure 3

UAP-HF-10332
Docket No. 52-021

Amended Response to Request for Additional Information
No. 646-5065, Revision 0

December, 2010
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/14/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 646-5065 R0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2**
DATE OF RAI ISSUE:	**10/07/2010**

QUESTION : 03.09.02-92

The applicant states in DCD Tier 2, Subsection 3.9.5.3.2 that some percentage of the main coolant flow is bypass flow, which is either for cooling metal or leakage between gaps. The bypass flows from gap leakages are as follows: small gap between the core-barrel outlet nozzle and RV outlet nozzle, neutron-reflector ring block inside surface and peripheral fuel assembly grids and nozzles, and neutron-reflector small gaps between the ring blocks. However, the applicant did not assess the liability of the core barrel flange to leakage flow-induced vibration.

In RAI 374-2446, Question # 03.09.05-25, the staff requested the applicant to discuss the liability of the core barrel flange to flow-induced vibration caused by the leakage (or bypass) flow between the outlet nozzle of the core barrel flange and the RV exit nozzle.

Since the diameter of the core barrel flange is larger than that of the current 4-loop reactors, its shell modes have lower frequencies. In addition, the leakage flow rate is higher in the US-APWR than in the current 4-loop reactors. The applicant was also requested to provide evidence showing that the leakage flow between the outlet nozzle of the core barrel flange and the RV exit nozzle will not cause excessive vibration of the core barrel flange. Lastly, the applicant was requested to revise Section 3.9.5 of the DCD to include an assessment of the leakage flow effects on the core barrel flange.

In its response MHI stated the following:

There has been no reported evidence of nozzle gap by-pass flow being a major contributor to the core barrel vibration response through the experience of previous plants operation or testing. The bypass flow from the outlet nozzle gap between the Core Barrel / RV has little effect on the core barrel vibration because the flow rate and the flow contact area of the gap are much smaller than those of the downcomer as discussed in the response to RAI 206-1576 (QUSTION NO.: RAI 3.9.2-43) and Appendix-A of MUAP-07027-R1:"Comprehensive Vibration Assessment Program for US-APWR Reactor Internals".

The applicant argues that the US-APWR is not expected to experience leakage flow-induced vibration because such vibration has not been experienced by other in-service reactors. However, the in-service reactors are smaller in size, operating at lower flow rates, and experience smaller pressure drops than the US-APWR. This argument is therefore unacceptable, and the applicant is requested in this supplementary question to provide evidence or a basis for stating that leakage flow vibration is not a concern in the US-APWR.

References:
MHI's Response to US-APWR DCD RAI No. 374-2446; MHI Ref: UAP-HF-09335; dated June 19, 2009; ML091751096.MHI's Response to US-APWR DCD RAI No. 206-1576; MHI Ref: UAP-HF-09116; dated March 27, 2009; ML090910123.

ANSWER (Revision 1):

In this response, MHI will discuss two points on the potential core barrel vibration due to the outlet nozzle bypass flow. One is the forced vibration due to the flow turbulence, in which we have estimated the turbulence force due to the outlet nozzle in comparison with that due to the inlet nozzle. The other is the instability vibration due to the leakage flow.

1.0 Forced vibration due to the flow turbulence

This section addresses the potential core barrel vibration due to the outlet nozzle bypass flow by making a comparison to the potential vibration due to the reactor vessel inlet nozzle flow.

The core barrel flange stress due to the flow vibration depends both on the magnitude of the flow-induced force on the core barrel and the distance from the forced point to the core barrel flange. The reactor vessel inlet nozzle and outlet nozzle are located at the same elevation, in which the distances from the inlet and the outlet nozzle centers to the core barrel flange are the same. Therefore, insights into the effects on the core barrel flange stress by the outlet nozzle bypass flow can be based on a comparison of the vibration forces by the inlet flow to the forces associated with the outlet nozzle bypass flow.

Table 1 summarizes the key parameters of the comparison, such as flow rate, flow velocity and forced area on the core barrel. It is noted that, in the US-APWR reactor design, the outlet nozzle bypass flow velocity at the mating surface gap is approximately the same as the inlet nozzle flow velocity. Because the turbulent pressure fluctuation is generally proportional to the square of flow velocity, the pressure fluctuation amplitude of the outlet nozzle bypass flow has the same magnitude as the inlet nozzle flow.

The flow entering through the inlet nozzle is spread into the reactor vessel downcomer after impingement on the core barrel wall. Therefore, the vibration force of the inlet nozzle flow is applied to the upper portion of core barrel. On the other hand, the forced area for the outlet nozzle bypass flow is applied only to the mating surface of the core barrel outlet nozzle. The forced area for an inlet nozzle flow is much larger and can be approximated by a 1/4 sector of the upper half of the core barrel excluding the outlet nozzle. The 1/4 sector is simply derived from the four equally-positioned inlets. As a result, as shown in Table 1, the forced area for the outlet nozzle bypass flow is only 3.8% of that for the inlet nozzle flow.

The flow-induced force is proportional to the multiple of the forced area and the amplitude of pressure fluctuation. As shown in Table 1, the effective vibration force due the outlet nozzle bypass flow is evaluated as 3.5% of that by the inlet flow. The combined effects of inlet flow and outlet nozzle bypass is derived using Square Root Sum of Squares (SRSS) as $(1.0^2+0.035^2)^{1/2}=1.0006$. Therefore, the relative effect of the outlet nozzle bypass flow is evaluated as 0.06 %.

2.0 Potential instability vibration due to the leakage flow

2.1 Fundamental consideration

In general, a fluid-structure system with a small gap potentially has vibration instability if the structural vibration causes the gap width to change resulting in a corresponding gap flow velocity change. In this system, fluid flow has its own inertia. Therefore, the flow oscillation may have a time delay from the structural vibration movement. As a result, the dynamic gap width narrowing will increase the flow velocity because the flow rate reduction is delayed from the decrease of gap flow area. The gap static pressure acts as a force on the gap mating surface. When the gap flow velocity increases due to the narrowed gap, the gap static pressure decreases. The force caused by this static pressure decrease acts to further narrow the gap. In this case, the static pressure force is considered as "negative damping (force)". This negative damping is the fundamental mechanism of instability or self-excited vibration in a fluid-structure system.

On the other hand, the gap between the core barrel and the reactor vessel outlet nozzle forms a different type of fluid-structure interaction system. In this system, due to the core barrel vibration, the gap width also fluctuates. However, the gap flow velocity is determined by the static pressure difference across the gap and is not related to the upstream flow inertia. The pressure force acting on the gap mating surface is influenced by the gap width fluctuation, which is caused by the core barrel vibration. Based on this discussion, the vibration instability is not predicted to occur in the US-APWR.

2.2 Comparison with the existing plants

In addition to the discussion above, quantitative insights are also available by comparison of the key parameters with the existing plants.

The key non-dimensional parameter for flow-induced vibration is the "reduced velocity" defined as U/(fn D), where "U" is the flow velocity, "fn" is the natural frequency of structures and "D" is the representative dimension. In general, a larger reduced velocity provides greater potential for flow instability. In this comparison, the outlet nozzle gap flow velocity was represented by the pressure drop of the reactor vessel. As shown in Table 2, the pressure drop of the US-APWR reactor vessel is equivalent to the 12-ft core 4-loop PWR or approximately 80 % of the 14-ft core 4-loop PWR design.

As for the vibration characteristics, the oval (n=2) mode natural frequency of the core barrel was considered as the most likely mode to interact with the outlet nozzle gap. An evaluation was conducted with the core barrel diameter, length and wall thickness. The result is that the shell mode natural frequency of the US-APWR core barrel is approximately 10% higher than that of the current 12-ft core 4-loop PWR design, as shown in Table 2.

Based on the comparison of both the vessel pressure drop and the core barrel natural frequency, the potential vibration instability due to the outlet nozzle flow for the US-APWR is not greater than the existing PWR plants.

3. Conclusion

Based on the aforementioned discussions, MHI concludes that there is no concern for the core barrel vibration due to the outlet nozzle gap flow for the US-APWR.

Table 1 Comparison of Vibration Forces Associated with Inlet Nozzle and Outlet Nozzle Bypass Flows

	Main Flow from Inlet Nozzle	Outlet Nozzle Bypass
Flow rate: Q(ratio)	1.0	0.003
Flow Velocity: V (ratio)	1.0	0.95
Pressure fluctuation: Prms $\propto \rho V^2$ (ratio)	1.0	0.91
Forced Area per one nozzle: A (ratio)	1.0 (upper core barrel 1/4 sector excluding the outlet nozzle)	0.038 (mating surface)
Effective Vibration Force $\propto$ Prms A (ratio)	1.0	0.035
Effect on Core Barrel Vibration (ratio)	1.0	0.0006

Notes ρ : fluid mass density (same for both flows)
Prms : root mean square of the pressure fluctuation amplitude

Table 2 Comparison between US-APWR and Existing PWR Designs

	US-APWR	Typical 12-ft core 4-loop PWR	Typical 14-ft core 4-loop PWR
Pressure Drops (psi)[(1)]			
Core	32.1 ± 3.2	25.8 ± 2.6	39.78 ± 4.0
Reactor Vessel	48.2 ± 4.8	48.5 ± 4.9	62.68 ± 8.9
Core Barrel Vibration Characteristics			
Outside diameter (in)			-
Wall thickness (in)			-
Axial length of shell portion (in)			-
Oval mode natural frequency (n=2) (relative ratio of the theoretical values)	1.1	1.0	-

[(1)] From Table 4.4-1 of US-APWR 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.
