MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU
TOKYO, JAPAN

March 27, 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-09116

Subject: MHI's Response to US-APWR DCD RAI No. 206-1576

References: 1) "Request for Additional Information No. 206-1576 Revision 0, SRP Section: 03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components, Application Section: 3.9.2.4," dated 2/25/2009.

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. 206-1576 Revision 0."

Enclosed are the responses to the four questions (40-43) of the RAI (Reference 1). This submittal completes the response to RAI 206-1576.

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

D081
NRO

Mitsubishi Heavy Industries, LTD.
Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 206-1576, Revision 0 (Proprietary)
3. Response to Request for Additional Information No. 206-1576, Revision 0 (Non-Proprietary)

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

Contact Information
C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-09116

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. 206-1576, Revision 0", dated March 25, 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 is that it describes the unique design parameters developed by MHI.

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

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with the development of the unique design parameters.

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.

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 27th day of March 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09116
Docket No. 52-021

Response to Request for Additional Information No. 206-1576,
Revision 0

March, 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

3/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**No. 206-1576, Rev. 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.4**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-40

In DCD Tier 2, Subsection 3.9.2.4, the applicant made a commitment to performing preoperational vibration testing and provided details of the sensors to be used.

The staff reviewed Subsection 3.9.2.4 and found that the DCD did not meet some of the expectations recommended in RG 1.20 and SRP 3.9.2. The applicant did not include any information about the pre-operational and startup test program of the steam generator internals. According to SRP 3.9.2 and RG 1.20, the applicant is expected to perform preoperational and initial start-up testing 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, describe the pre-operational and startup test program to demonstrate that adverse flow effects will not cause unanticipated excessive flow-induced vibrations or structural damage. The test program description should include a list of test flow modes, a list of sensor types and locations, a description of test procedures and methods to be used to process and interpret the measured data, including bias errors and uncertainties, a description of the visual inspections to be made, a comparison of test results with the analytical predictions, and the acceptance criteria for stress levels and for comparison with the analysis results. If the steam generators are classified as non-prototypes, provide the requested information for the components with deviations from the prototype design or operating conditions. If the steam generator internal structures are a non-prototype design, provide reference to the tests of the prototype steam generator and give a brief summary of the results.

The staff needs this information to assure conformance with GDC-1 and 4. Revise Subsection 3.9.2.4 of the DCD to include a detailed description of the pre-operational and startup test program 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

See Attachment 2 for the mark-up of DCD Section 5.4, Revision 2, changes to be incorporated:

- Replace the second paragraph in Subsection 5.4.2.1.2.10 to:

  "RG 1.20 (Ref. 5.4-21), recommends that the potential adverse effects from pressure fluctuations and vibration be considered for PWR SG internals. Although there are instances where similar components in boiling water reactors experienced excessive vibration, no such experience has been reported for PWR SG designs.

  The design of the US-APWR SG upper internals and the flow conditions they experience are similar to the existing and currently operating SGs in the United States and around the world. MHI designed SG upper internals using structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR SGs has been operating in the USA for more than 20 years in SGs of sizes and flow rates that bound those of the US-APWR SGs. Based on an extensive record of vibration-free operation, MHI concludes that the structural and vibration design bases are proven. These non-safety-related SG internals will not experience excessive vibration. Therefore, no startup testing is planned for these components."

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

3/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**No. 206-1576, Rev. 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.4**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-41

According to Section 3.9.2 of the SRP, a preoperational test program for the steam delivery system should be described in Subsection 3.9.2.4 of the DCD.

In DCD Tier 2, Subsection 3.9.2.4, the applicant did not describe the preoperational and start-up vibration test program for the steam delivery systems. The staff needs this information to confirm that appropriate vibration test program is planned to ensure that adverse flow effects will neither cause unanticipated flow-induced vibrations of significant magnitude nor structural damage of the steam delivery systems. The applicant is requested to provide additional details about the flow-induced vibration measuring and monitoring program for the preoperational and start-up tests of the steam delivery system, including the steam separator, the safety relief valves and the steam lines. The requested additional information should address the measurement locations with diagrams, test conditions, hold points to allow data acquisition and analysis, and inspection and monitoring program. This is necessary to assure conformance with GDC-1 and 4. Revise subsection 3.9.2.4 of the DCD to include additional details about the preoperational and start-up vibration test program of the steam delivery systems.

ANSWER:

The design of the US-APWR steam delivery system (including the steam separator, safety relief valves, and steam lines) 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 in the steam delivery system of 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, no startup testing is planned for the steam delivery 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

3/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**No. 206-1576, Rev. 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.4**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-42

The vibration assessment report MUAP-07027-P provides details about the types and locations of the transducers that will be used in the preoperational vibration test program, the test conditions, and inspection program.

The staff reviewed the technical report MUAP-07027-P and found that although the overall concept of this test program seems reasonable, it is not clear what provisions are made to ensure that adequate data will be obtained even if several sensors fail during the preoperational test. Subsection 3.9.2.4 of the DCD document does not address the issue of instrumentation redundancy. The applicant is requested to discuss the provisions made in the vibration test program to ensure sufficient redundancy in the instrumentation such that adequate information is obtained from the preoperational and start-up vibration test program. It is essential that sufficient information is obtained from the vibration tests to be able to assess the margin of safety of the critical components of the reactor internals and the steam delivery system. This is necessary to assure conformance with GDC-1 and 4. Revise subsection 3.9.2.4 of the DCD to include additional details about provisions made to ensure sufficient redundancy in the instrumentation.

ANSWER:

As described in Technical Report MUAP-07027-P R0, the vibration measurement program contains redundant sensors to safe guard against premature sensor failure. This is further clarified below.

a. Core Barrel / Lower Core Support Plate

To measure the beam mode response of the lower internals, theoretically three sensors are needed: one for each of the two horizontal directions and a third for the vertical direction. However, a total of four strain gages will be installed to measure the strains due to beam mode vibration of the lower internals. The fourth strain gage is a redundant sensor serving as back up in case one of them fails prematurely. In addition, two additional strain gages will be installed on the inner surface of the core barrel flange to measure the strains caused by local bending.

b. Neutron Reflector / Tie Rod

Since the neutron reflector is a first-of-a-kind design, more data on this component will be acquired to ensure that no unexpected vibrations occur. Taking into consideration the symmetry of this component, MHI believes that a total of four accelerometers with axes in the radial direction will provide sufficient redundancy to define the vibration characteristics of this component, even if one fails prematurely. In addition, an additional accelerometer will be installed to measure the vertical motion.

c. Upper Plenum Structures

The top slotted column is another first-of-a-kind design in US-APWR. For this component, three strain gages will be installed to measure the vibration responses in the two horizontal directions with the third sensor serving as the redundant back up sensor.

d. RCC Guide Tube

Because the rod cluster control (RCC) guide tube has a safety related function and is one of the most important subassembly of the reactor internals, three strain gages will be installed on one of the RCC guide tubes to measure the responses in three directions and two strain gages will be installed in the upper guide tube of this same RCC guide tube. For redundancy, two strain gages will be installed on the lower guide tube of another RCC guide tube.

Additional information about the measurement redundancy in the Lower Plenum Structures

In regards to lower plenum structures, these structures consist of two sub assemblies, the upper diffuser plate assembly and the lower diffuser plate assembly. All of diffuser plate support columns (here after support columns) are tightly connected by the diffuser plate; so the expected vibration response of each support column should be similar. In addition, the natural frequency of the assembly and the stress of support columns can be confirmed by the strain measurement on a support column for the each assembly. Considering the original design of these structures, the strain on two support columns will be measured for each assembly to maintain sufficient redundancy.

Clarifications consistent with the above information will be made to the future Revision 1 of Technical Report, MUAP-07027.

Impact on DCD

See Attachment 1 for the mark-up of DCD Section 3.9, Revision 2. Changes to be incorporated:

- Add a new paragraph to the end of Subsection 3.9.2.4.2:

  "Details for the data acquisition and reduction system, including redundancy, are described in Section 4.1 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

3/27/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**No. 206-1576, Rev. 0**
SRP Section:	**03.09.02 – Dynamic Testing and Analysis of Systems Structures and Components**
APPLICATION SECTION:	**3.9.2.4**
DATE OF RAI ISSUE:	**02/25/09**

QUESTION NO.: RAI 3.9.2-43

A major conclusion, based on the results of the vibration assessment program described in technical report MUAP-07027-P, is that the vibration responses of the reactor internals without the core are the same or slightly larger than those with the core. Therefore, the applicant proposes to conduct the preoperational and start-up vibration testing (cold hydraulic test and hot functional test) only before loading the core. It is argued that the vibration levels after loading the core will be bounded by those measured without the core.

The staff has reviewed the technical report MUAP-07027-P and is concerned about the validity of this conclusion, and also about other undesirable/safety consequences that may arise if the preoperational tests are performed without the core. The applicant is requested to substantiate the validity of the argument that the dynamic response of the reactor internals under normal and operational flow transient conditions with fuel assemblies in the core is the same or slightly smaller than that under hot functional test conditions without the core. Verification of this argument is needed to assess the proposal made by the applicant to confine data acquisition during the initial start-up tests to the hot functional tests before core loading. In responding to this RAI, the applicant is requested to address the following issues and their influence on the effect of core loading on the dynamic response of the reactor internals:

(a) The scale model tests were performed on a J-APWR, which has a shorter core than that of the US-APWR. In addition, the geometry of the scale model seems to represent the 4-loop reactor rather than the US-APWR.
(b) Dynamic similarity of the scale model tests and the reactor prototype.
(c) Effect of fuel assembly on flow distribution and pattern within the reactor, including the cross-flow velocity in the upper and lower plenums of the reactor vessel.
(d) Since the pressure drop will be lower without the core, the bypass/leakage flow will be smaller than with the core. This may affect leakage flow-induced vibration, especially at the exit nozzles of the core barrel.
(e) The operating point on the Q-H characteristic curve of the RCP will be different from that with the core.
(f) Vibration tests of the fuel assemblies can not be performed unless the core is present in the reactor.

The DCD and the vibration assessment report do not discuss the effect of the above mentioned issues. The applicant is requested to adequately address these issues so that the staff can assess conformance with GDC-1 and 4. Revise the vibration assessment report to address the staff concerns and refer to these additions in Subsection 3.9.2.4 of the DCD.

ANSWER:

The J-APWR SMT data is used for simulation analysis input or as a comparison reference. The results of the reactor internals flow induced vibration integrity evaluation for the J-APWR Scale Model Test(SMT) are not directly applied for US-APWR assessment as follows.

As shown in the analysis flow diagram, Figure 3.1-1, from MUAP-07027-P, Revision 0, the measured natural frequencies and vibration response levels were used as reference data for the validation of analysis methodology through the simulation of the J-APWR SMT results.

In addition, the normalized pressure power spectral densities (PSDs) measured in the downcomer of the J-APWR SMT are applied for both J-APWR SMT simulation analysis and US-APWR prototype analysis. This is based on the similarity of downcomer geometrical dimensions and flow rate as discussion in answer (a) (below).

Responses to issues (a) through (f) can be found below.

(a) Comparison of the Reactor - Current- 4-loop/J-APWR/US-APWR:

Key specifications of the reactor for the current 4-loop, J-APWR, and US-APWR are summarized in Table 1 (below).

The core length between the J-APWR and US-APWR is different, but the number of fuel assemblies is the same. The dimensions of reactor vessel and the core barrel are also the same as is the flow rate.These parameters do differ from the current 4 loop, except core barrel (downcomer) length.

Therefore, the flow-induced vibration response characteristics of the US-APWR reactor internals are equivalent to those of the J-APWR.

Table 1 Comparison of Reactor of Current 4-loop /J-APWR / US-APWR

	Current 4-loop	J-APWR	US-APWR
Number of RC Loops	4	4	4
Numbers of Fuel Assemblies	193	257	257
Core length (ft)	12	12	14
Downcomer length (inch)	[ ]	328	328
Vessel Inside Diameter (inch)	[ ]	202.8	202.8
Numbers of RCCA/GT	53	69 / 77 / 85	69
Loop flow rate for Mechanical Design (GPM)	[ ]	129,000	130,000
Structure around the core	Core Baffle	Neutron Reflector	Neutron Reflector

(b) Comparison of dimensionless parameters between J-APWR SMT and US-APWR plant:

The dimensionless parameters related to flow-induced vibrations, the Reynolds number (Re), and reduced velocity (Ur) for the J-APWR SMT, J-APWR plant, and US-APWR plant are shown in the Table 2. In addition, the Strouhal numbers (St) are also summarized in the same table for the structures exposed to cross-flow in the lower and upper plenums.

i) Reynolds number – Under operating conditions of PWR, the coolant flow inside the reactor vessel will be in the turbulent flow regime. It is considered that the flow characteristics would remain the same in sufficiently developed turbulent flow regime. The transition from laminar flow to turbulent flow occurs at Reynolds number ($Re = UD/\nu$) around 10^3 in general. For this reason, we selected the scale model test condition to keep Reynolds number greater than 10^4.

   As shown in the Table-2, sufficiently high Reynolds number remained in the downcomer, lower plenum and upper plenum even in the test conditions of the J-APWR SMT. This is also true for the plant operating conditions.

ii) Reduced velocity – The reduced velocity ($Ur = U/(fnD)$) is generally utilized in the dimensional analysis of the flow-induced vibration. Ur represents the ratio of the path length per cycle (U/fn) and the model width (D). From another view point, Ur represents the ratio of the fluid force frequency (proportional to U/D, the vortex shedding frequency (fs) is a typical example) to the natural frequency of the model.

   As shown in Table 2, the reduced velocities in the J-APWR SMT are close to those of J-APWR plant conditions, and also similar with those of US-APWR plant.

iii) Strouhal number – The Strouhal number ($St = fsD/U$) is the non-dimensional parameter for the vortex shedding frequency. As well known, St of a cylinder in cross-flow is almost constant (around 0.2) under the critical Re number based on the cylinder diameter. As shown in Table 2, St in the SMT, under room temperature, is also around 0.2. But for plant operating conditions, St will be around 0.3 because the Re number should be in the super critical region. So the evaluation based on the analysis as shown in the Table 3.2-4 of MUAP-07027-P, Revision 0, is also required for check of vortex shedding lock-in, even if any lock-in is not observed in a SMT.

Table 2. Comparison of Dimensionless Parameters Between J-APWR SMT, J-APWR plant, and the US-APWR Plants

	J-APWR plant	J-APWR 1/5 SMT	US-APWR plant
Downcomer /Core Barrel Flow Velocity U (m/s) Annulus width h (m) Core barrel fn (Hz)			
Re=Uh/ν Vr=U/fnh			
Lower plenum / Lower Diffuser Plate Support Colum Flow Velocity U (m/s) Column diameter D (m) Column fn (Hz)			
Re=Uh/ν Vr=U/fnD St =fs D/U			
Upper Plenum / Top Slotted Colum Flow Velocity U (m/s) Column diameter D (m) Column fn (Hz)			
Re=Uh/ν Vr=U/fnD St =fs D/U			

(c) Effect of fuel assemblies on the flow condition inside the reactor, including the lower and upper plenums:

For the following reasons, the fuel assembly has little effect on reactor vessel flow conditions, including cross flow velocities in the lower and upper plenums.

i) The maximum cross-flow distribution in the upper plenum depends on the outlet nozzle flow velocity and geometries of structures near the outlet. It does not depend on the core outlet flow distribution into the upper plenum. And because of a little bit increase of total flow rate with lower pressure loss in the core, the maximum cross flow velocity in the upper plenum in the hot functional test without core will be higher than the normal operating condition.

ii) The maximum cross flow distribution in the lower plenum depends on the flow velocity in the downcomer and geometries of structures in the peripheral region of the lower plenum. It does not depend on the core inlet flow distribution in the downstream side. And because of the increase of total flow rate with lower pressure loss in the core, the maximum cross flow velocity in the lower plenum during the hot functional test without the core will be higher than the normal operating condition.

So, the mechanical integrity of structures subjected to cross-flow in the lower and upper plenums can be verified without fuel assemblies.

(d) The bypass flow rate from the outlet nozzle gap between the Core Barrel/RV in the plant operating condition is not larger than that in the pre-operational testing. Because the gap clearance is designed to be minimum in the normal operating condition considering the core barrel thermal expansion.

In any case, the bypass flow between the outlet nozzle gap has little effect on the core barrel vibration because the both flow rate and the surface area contact to the flow are much smaller than those of downcomer flow.

(e) The difference of operating point on the reactor coolant pump (RCP) Q-H curve has been considered in the estimation of test flow rate from that for normal operating conditions with the fuel loaded.

(f) There is little need for fuel assembly vibration measurement in pre-operational or start – up testing because of following reasons.

i) Flow induced vibration response of the fuel assembly will be confirmed in a full size mock-up testing.

ii) Vibration of the fuel assemblies in the core can be checked by the Fast Fourier Transform (FFT) analysis of the ex-core nuclear instrumentation signals in the start-up testing, if needed.

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 3.9.2.4.1, "Detailed information, including discussions about other effects with or without the core, is described in Subsection 3.4.3, Structural Responses for Preoperational and Initial Startup Testing Conditions, 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.

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

- responses without the core are also the same or slightly larger than those with the core. This is because the flow rate increases with the elimination of fuel assemblies and the subsequent pressure loss. Thus, in the preoperational test of the prototype plant, the results of vibration measurements after core loading are bounded by the measurements before core loading, and only measurements before core loading are necessary.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

3.9.2.4.1 Background

The first operational US-APWR reactor internals are classified as Prototype in accordance with RG 1.20 (Reference 3.9-21). Upon qualification of the first US-APWR as a Valid Prototype, subsequent plants will be classified as Non-Prototype category I based on the designation of Regulatory Guide 1.20. The first COL Applicant is to commit to implement a pre-operational vibration assessment program and to prepare the final report consistent with guidance of RG 1.20 for a prototype. Subsequent COL Applicant need only provide information in accordance with the applicable portion of position C.3 of RG 1.20 for Non-Prototype internals.

Following the recommendation of Regulatory Guide 1.20 (Reference 3.9-21), a pre-operational vibration measurement program is developed for the first operational US-APWR reactor internals. Data will be acquired only during the hot functional test, before core loading. Analysis (Subsection 3.9.2.3) shows that the responses under normal operating condition with fuel assemblies in the core are almost the same or slightly smaller than those under hot functional test conditions without the core. <u>Detailed information, including discussions about other effects with or without the core, is described in Subsection 3.4.3, Structural Responses for Preoperational and Initial Startup Testing Conditions, of Reference 3.9-22.</u>

Subsequent to the completion of the vibration assessment program for the first US-APWR reactor internals, the vibration analysis program will be used to qualify subsequent US-APWR under the criteria for non-prototype category I.

The needs for flow-induced vibration, measurement testing, of steam generator internals is discussed in Subsection 5.4.2.1.2.10.

3.9.2.4.2 Measurement Program

Measurements will be performed during the pre-operational test to confirm the vibration characteristics and structural integrity of the Prototype US-APWR reactor internals.

The acquired data will be used to confirm that unexpected, abnormal vibrations do not occur, and that the vibration responses are sufficiently small compared to an acceptance criterion based on the design fatigue curves in the ASME Code, Section III.

Instrumentation consisting of strain gages, accelerometers, pressure transducers and displacement transducers will be installed on selected components. Accelerometers and displacement transducers will be used to measure the responses of the reactor internals. Strain gages will be used to directly measure the strains at key connecting points, and dynamic pressure transducers will be used to measure the pressure fluctuations at selected locations. Some of the specific measurement locations are described below.

- Core barrel: Strains in the core barrel flange will be measured with strain gages. Shell-mode responses will be measured with accelerometers mounted on the wall of the core barrel.
- Lower core support plate: An accelerometer mounted near the center of the lower core support plate will be used to measure the vertical response of this component.
- Neutron reflector: Shell mode responses and vertical motions will be measured by accelerometers. Relative displacement between the core barrel and the neutron reflector will be measured by displacement sensors. The vibration responses of the tie rod will be measured by strain gages.
- Secondary core support assembly: Vibration responses will be measured by strain gages mounted on the diffuser plate support columns.
- RCCA guide tubes and upper support columns: Beam mode responses due to the cross-flow in the upper plenum will be measured by strain gages and accelerometers.
- Upper core support: The vertical response will be measured by an accelerometer mounted near the center of the upper core support plate. Horizontal responses will be measured by strain gages installed on the upper core support skirt.

Details for the data acquisition and reduction system, including redundancy, are described in Subsection 4.1 of Reference 3.9-22.

3.9.2.4.3 Inspection Program

The internal components of all US-APWR plants will be inspected before and after the hot functional test. The reactor internals will not be considered adequate and pass the comprehensive vibration assessment program unless no structural damage or change is observed.

3.9.2.4.4 Acceptance Criteria

The acceptance criteria of the pre-operational flow-induced vibration testing for reactor internals are as follows.

- Vibration measurement

  The measured rms vibration amplitudes will be multiplied by 4.5 to convert them into 0-peak values. The corresponding 0-peak stresses in key connecting components will be calculated from the measured vibration amplitudes or strains. These stresses must show sufficient safety margins based on the design fatigue curves in the ASME Code, Section III, Appendix-I.

- Inspection

  No structural damage or change is observed in the post-hot functional test inspection.

ATTACHMENT 2

to RAI 206-1576

velocities are within operating experience. Also the primary side water che[...] controlled. Therefore, no special corrosion allowance is required for these surfaces.

Most of the materials used on the SG secondary side are low alloy steel or carbon steel that are more susceptible to general corrosion or flow accelerated corrosion. These parts have corrosion allowances based on the operating experience and corrosion test data (Table 5.4.2-3).

5.4.2.1.2.9 Foreign Material Perforated nozzle

The SG feedwater ring is equipped with perforated nozzles that capture foreign materials entering the SG from the feedwater system. Foreign materials, if allowed to reach the tube bundle, can wear against the tubes on the tube bundle periphery at the top of the tubesheet. Small foreign materials that can migrate between the tubes until they reach a low velocity region pose less risk of tube wear. So, the hole size in the feedwater ring perforated nozzles is smaller than the space between the tubes. Larger foreign materials caught inside of feedwater ring pose no risk to the SG and can be retrieved through the feedwater ring inspection port.

5.4.2.1.2.10 Flow Induced Vibration of Secondary Side Internals

The SG internals are analyzed for their vibrational characteristics and structural integrity to confirm their adequacy for long term operation and to minimize the potential for the formation of loose parts.

~~RG 1.20 (Ref. 5.4-21) applies to prototype configurations where field experience is not available. The US-APWR steam generator secondary side internals, including the primary and secondary separators are a proven design used in many operating steam generators. Therefore, because the design is already field proven, the RG 1.20 recommendation for startup vibration measurements does not apply.~~

RG 1.20 (Ref. 5.4-21), recommends that the potential adverse effects from pressure fluctuations and vibration be considered for PWR SG internals. Although there are instances where similar components in boiling water reactors experienced excessive vibration, no such experience has been reported for PWR SG designs.

The design of the US-APWR SG upper internals and the flow conditions they experience are similar to the existing and currently operating SGs in the United States and around the world. MHI designed SG upper internals using structural design rules based on years of empirical experience with similar equipment. The configuration employed in the US-APWR SGs has been operating in the USA for more than 20 years in SGs of sizes and flow rates that bound those of the US-APWR SGs. Based on an extensive record of vibration-free operation, MHI concludes that the structural and vibration design bases are proven. These non-safety-related SG internals will not experience excessive vibration. Therefore, no startup testing is planned for these components.