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

July 17, 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-09387

Subject: MHI's Response to US-APWR DCD RAI No. 374-2446

References: 1) "Request for Additional Information No. 374-2446 Revision 0, SRP Section: 03.09.05 – Reactor Pressure Vessel Internals, Application Section: DCD, Tier 2 – Section 3.9.5," dated 5/21/2009.

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. 374-2446 Revision 0."

Enclosed are the responses to questions 2-4, 6-12, 14-20, 22-24 26 and 27.of the RAI (Reference 1). The responses to the other questions (1, 5, 13, 21 and 25) of this RAI with 30 days are already submitted by a separate transmittal MHI Ref. UAP-HF-09335 dated June 19.

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).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

Sincerely,

Y. Ogata

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

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata
2. Response to Request for Additional Information No. 374-2446, Revision 0 (Proprietary)
3. Response to Request for Additional Information No. 374-2446, 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-09387

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. 374-2446, Revision 0", dated May 21, 2009, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

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

   A. They include the know-how and outputs obtained from analyses or testing which required 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 directly referred from documents or 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 17th day of July 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09387
Docket No. 52-021

Response to Request for Additional Information No. 374-2446,
Revision 0

July, 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-2

The applicant stated in Subsection 3.9.5.1.1 of the DCD that the upper core support assembly is restrained vertically in the upward direction by the RV head flange and in the downward direction by the reactor internals hold-down spring. The preload in the hold-down spring during installation, is controlled by a fixed distance between the bottom of the upper core support flange and the top of the core barrel flange. The horizontal loads on the upper core support assembly due to flow, vibration, and seismic and pipe rupture events are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange; head and vessel alignment pins also transmit some of the horizontal loads.

The staff's review of the DCD indicated that the applicant did not discuss the potential loss of preload in the hold-down spring due to stress relaxation during service and its potential effect on the functional and structural integrity of upper core support assembly. The applicant is requested to provide an assessment of the potential loss of preload of the hold-down spring due to stress relaxation during the design lifetime, and discuss its effect on the horizontal and vertical restraints of the upper core support and core barrel assemblies. Alternately, provide a reference document where this information is available. The staff needs this evaluation for the above mentioned plant components to assure conformance with GDC-1 and 4. Revise the DCD to include the requested information.

ANSWER:

The hold-down spring in PWR operating plants have been observed in the industry to have some loss of preload from inelastic deformation of the contact surfaces during initial bolt-up and subsequent stress relaxation from plant operation. Since the material and design of the US-APWR hold-down spring is similar to that which has been successfully used in many operating PWR plants, it is not expected that the loss of preload will affect the hold-down spring functionality.

Impact on DCD

The following change will be made to the DCD:

In DCD Section 3.9.5.1.1, Upper Reactor Internals Assembly Design Arrangement, change the third paragraph as shown below to read:

"For loads in the upward vertical direction, the upper core support assembly is vertically restrained by the RV head flange, and in the downward direction by the RV flange. A toroidal-shaped hold-down spring is sandwiched between the upper core support flange and the core barrel flange. The primary function of the hold-down spring is to accommodate the thermal expansion differences between the RV and the reactor internals upper core support flange and core barrel flange. A vertical preload in the hold-down spring is developed during installation of the upper internals and is controlled by a fixed distance between the bottom of the upper core support flange and the top of the core barrel flange. Vertical loads on the upper core support assembly come from dead weights less buoyant forces, the upper core support and the upper core plate differential pressure loads, vibration loads on the components, fuel assembly spring and lift loads, and seismic and postulated LOCA loads. There is a designed radial gap between the upper core support flange and the RV inside diameter. The gap is large enough to prevent contact from thermal expansion of the upper core support flange relative to the RV flange during operation. Horizontal loadings from flow loads, vibration loads, and seismic and pipe-rupture loads are transmitted from the upper core support flange to the RV head and hold-down spring by friction or direct contact with the RV flange. Head and vessel alignment pins also transmit some of the horizontal loads to the RV head and RV flanges."

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-3

In DCD Tier 2, Subsection 3.9.5.1.1 the applicant provided a description of the USAPWR upper reactor internals assembly design arrangement, including the manner of positioning and securing of these items and providing for axial and lateral retention and support.

The staff reviewed Subsection 3.9.5.1.1 of the DCD and found that the applicant did not provide sufficient information to allow the review of the upper core plate design and its interfaces with other reactor components. The applicant is requested to provide sufficient details about the design of the upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes. Also, explain any differences from the existing 4-loop design, and how these differences are evaluated against possible excitation mechanisms of flow-induced vibration. Review of any design differences from the 4-loop design and consequent effects on potential adverse flow effects is needed to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to include sufficient information about the design arrangement of the upper core plate and a discussion of the differences, if there are any, in its loading conditions from the 4-loop reactor.

ANSWER:

The US-APWR upper core plate and its interface with the fuel assemblies, core barrel, upper support columns, and lower guide tubes are similar to those of the existing 4 loop design. So, there is expected to be little impact on the flow-induced vibration due to the structural design changes around the upper core plate.

More detail of discussions about the design differences of the US-APWR reactor internals from current 4 loop and effects on the flow-induced vibration are described in Chapter 2.1 of Reference (1).

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

Reference (1): MUAP-070027 R1 "The comprehensive vibration assessment program for the US-APWR Reactor Internals"

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-4

The applicant stated in Subsection 3.9.5.1.1 of the DCD that the guide tubes consist of two main assemblies, an upper and a lower guide tube, that provide horizontal restraint and guidance to the control rods and drive rod assembly, and allow parking of the drive rod during removal and installation after refueling. The upper and lower guide tubes have plates that guide the control rod spider during insertion and retraction of the rod cluster control assembly (RCCA).

The staff's review indicated that the applicant did not provide sufficient information about the control rod guide inside the upper and lower guide tubes. The applicant is requested to provide design details together with relevant flow-induced vibration analysis (if they are needed) for the plates that guide the control rod spider inside the upper and lower guide tubes. In particular, the applicant is requested to explain, with the aid of technical drawing/sketches, the design of the control rod guide and to clarify any differences of this design from that of the existing 4-loop reactor. Also, explain the effects of any design differences on potential flow excitation mechanisms. This information is needed to assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to provide the requested information.

ANSWER:

The lower RCCA guide tube of the US-APWR is similar in design with that of the current 4 loop reactor in the layouts and dimensions of the enclosure and guide plates.

The upper RCCA guide tube is extended about 1 ft from that for the 12 ft core design as the 14 ft core 4 loop reactors existing in US.

Because both the flow condition inside the guide and mechanical interface with the RCCA spider is equivalent with existing 4-loop. No impact is expected on the vibration of the RCCA spider.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-6

A description of the US-APWR upper reactor internals assembly design arrangement, including the manner of positioning and securing of these items and coolant flow through the reactor internal assemblies is presented in Subsection 3.9.5.1.1 of the DCD. The applicant stated that the exit flow core pressure difference between the fuel assemblies is limited by the design to an acceptable cross-flow velocity to prevent vibratory damage to the fuel rods, thimbles, or RCCAs.

The staff's review of Subsection 3.9.5.1.1 showed that the applicant did not explain how the thermal-hydraulic design requirement regarding the fuel assembly exit core flow would be verified. As stated in Subsection 3.9.5.3.2 of the DCD, the thermal-hydraulic performance criteria require that the core outlet flows from the fuel assemblies are to be designed to minimize horizontal velocities that may contribute to vibration of the RCCA rodlets. The applicant is requested to describe the procedure that is to be used to verify that the exit flow from the fuel assemblies does not lead to unacceptable cross-flow velocities that may cause vibration of the fuel rods, thimbles, or RCCAs. This information is needed to review the safety analysis design requirements and thereby assure conformance with GDC-1 and 4. Revise Subsection 3.9.5.1 of the DCD to include the requested information.

ANSWER:

It is verified that the fuel assembly exit cross-flow velocity is acceptable for the US-APWR by operating plants with similar design features to the US-APWR fuel assemblies and upper internals.

The design of the upper core plate flow holes, fuel assembly loss coefficients, and the fuel assembly design of the US-APWR is not significantly different from those of existing 4 loop plants. So the cross flow velocities at the core outlet are expected to be similar to the existing 4 loop plants. From the experience of existing 4 loop plants, the adverse flow effects on the vibration of the fuel rods or RCCA is acceptably limited.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-7

The DCD Tier 2, Subsection 3.9.5.1.2 describes the lower core support plate assembly. The applicant stated that the lower core support plate has orificed flow holes to reduce mal-distribution of the flow into the core. The safety analysis design requirements for US-APWR internals listed in Subsection 3.9.5.3.1 of the DCD state that mal-distribution of the flow into the core should be limited so as not to impact core safety limits in Chapter 15 of the DCD. However, the applicant did not refer to any safety analysis that would ensure compliance with this safety requirement for the design of US-APWR core support structure and reactor internals. The applicant is therefore requested to discuss the analysis performed and the measures undertaken to make sure that the mal-distribution of the flow into the core shall be limited so as not to impact the US-APWR core safety limits. Alternately, provide a reference document where this information is available. This information is needed to review the safety analysis design requirements and thereby assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to provide the requested information.

ANSWER:

(This answer includes the response to QUESTION NO.: 03.09.05-23 of this RAI)

Mal-distribution of flow into the core is limited by meeting several reactor internals design requirements. These design requirements include the allowable minimum and maximum fuel assembly inlet flow rate, and the allowable difference in inlet flow rates between adjacent fuel assemblies. The design target values are ≥ [ ]% of nominal flow rate for the minimum flow rate; ≥ [ ]% of nominal for the maximum flow rate; and ≤ [ ]% for difference between adjacent fuel assemblies. These design requirements are similar to those in operating 4-loop US plants. Confirmatory testing was performed for the US-APWR, Reference (1), and the results show that the minimum assembly flow was [ ]%; the maximum assembly flow was [ ]%; and the difference between adjacent fuel assemblies was [ ]%. From the test results, it is concluded that the inlet core flow distribution is such as to preclude adverse effects such as core tilt, flow starvation, or undesirable inlet cross-flow distribution.

The US-APWR core safety limit from mal-distribution of the flow into the core is the DNB margin. As long as the reference target for the reactor vessel internals design in Subsection 1.3.1 of MUAP-07022-P is met, the DNB margin design is not affected as described in RAI NO. 377-2629 R1 QUESTION NO. 04.04-1.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-8

The applicant stated in Subsection 3.9.5.1.2 of the DCD that the energy absorber system and base plate have traditionally been used in PWR internals. Its purpose is to preclude overstressing the RV in the unlikely event of a failed core barrel weld. The drop distance between the bottom of the base plate and the energy absorber system RV bottom is carefully controlled to minimize the impact load and stresses on the RV bottom head. In Subsection 3.9.5.3.1 the applicant further stated that the safety analysis of this issue is a design requirement for the reactor vessel.

The staff reviewed Section 3.9.5 of the DCD and found that the applicant did not refer to any analysis of the impact load, which would result from a postulated core drop event.
The applicant is requested to:
(a) characterize the postulated core drop event as either a design basis accident required by NRC regulation, or as a beyond-design-basis event not required by regulation;
(b) if the core drop event is considered a design basis accident, discuss the analysis performed and the measures undertaken to make sure that the impact load on the RV bottom head from a postulated core drop event would not adversely affect the integrity of the RV bottom head.
This information is needed to review the safety analysis design requirements and thereby assure conformance with GDC-1 and 4. Revise Section 3.9.5 of the DCD to provide the requested information.

ANSWER:

(a) The core drop event is considered a postulated accident condition and considered as a beyond-design-basis-event not required by regulation.

(b) Although the core drop event is not a design basis accident condition, the impact load and contact surface area is calculated using the sizing of the secondary core support structure and the gap clearance between the RV bottom head and base plate. The RV allowable impact load and contact surface area is provided by the RV Designer and is to be

complied with as an RI interface design requirement.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-9

In Subsection 3.9.5.3.4 of the DCD the applicant stated that corrosion, stress corrosion cracking (SCC), radiation embrittlement, and degradation of fatigue strength are considered to be not an issue for the operating conditions of the US-APWR and that the potential for irradiation assisted stress corrosion cracking (IASCC) of the US-APWR core internals is very low. The applicant further stated that void swelling from neutron DRAFT REQUEST FOR ADDITIONAL INFORMATION EMB1 2446 REVISION 0 irradiation was a concern for components with high dose of neutron fluence, e.g., neutron reflector ring blocks, but the ring blocks were cooled to keep metal temperature low and minimize void swelling.

However, the applicant did not provide estimates of temperature and end-of-life neutron fluence for the various reactor internal components, and has not identified the components where void swelling, radiation embrittlement, IASCC, or degradation in fatigue strength is likely to be significant. The primary water stress corrosion cracking (PWSCC) of Ni-alloys, such as X-750, is not addressed in the DCD. Also, the DCD does not provide an assessment of environmental effects on the structural and functional integrity of reactor internals. The applicant is requested to (a) describe the environmental conditions, including estimates of the temperature and end-of-life neutron fluence, for the various reactor internal components, and (b) either provide an evaluation to verify that, under the operating conditions of the US-APWR, the effects of corrosion, SCC, IASCC, PWSCC, degradation of fatigue strength, radiation embrittlement, and void swelling, on the structural and functional integrity of the reactor internal components are not a concern during the design life of 60 years, or define an acceptable program for investigating and managing these environmental effects on reactor internals.
Alternately, provide a reference document where this information is available. This information is needed to assure conformance with GDC-1 and 4. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

Environmental conditions from PWR water chemistry and irradiation fluence have been

assessed for their effects on the functional and structural integrity of the reactor internal components.

The PWR water chemistry environment and the effects from corrosion, and PWSCC are not expected to be a concern for reactor internals because the water chemistry is tightly controlled and high stressed parts within enclosed confinement are properly vented. Furthermore, the reactor internals are subjected to very small differential pressure stresses which should exempt them from environmental fatigue degradation since RG 1.207 addresses rules only Class 1 components in a pressure boundary. In addition, most of the reactor internals are fabricated from austenitic stainless steel material which has shown to be robust in a PWR water chemistry environment to the extent that plant life extension evaluations have allowed an increase in design life up to 60 years.

Materials other than authentic stainless steel, such as water quenched Inconel X-750 for support pins, and Type 403 material for the hold-down spring have been successfully used in 4-Loop operating plants under similar stress and environmental conditions for the US-APWR design.

In operating 4-loop plants, baffle plates are supported by baffle bolts and are used in the most severe conditions from the viewpoint of IASCC. The baffle plate design has been replaced by a neutron reflector in the US-APWR, so that IASCC conditions are less severe.

Irradiation environmental effects such as IASCC, embrittlement, and void swelling have been evaluated for those components subjected to high dose rates, such as the neutron reflector block; neutron reflector block alignment pin; neutron reflector tie-rod, and core barrel.

In response to RAI 4.5.2-2, we evaluated the effect of the neutron reflector void swelling according as Materials Reliability Program (MRP-175), the void swelling is evaluated [ ]%. MRP-175 states that void swelling is smaller than [ ]%, no functionality evaluation is required. The void swelling evaluation of the all reactor internals can be represented by the neutron reflector which surrounds the core.

The assessment of IASCC was also performed in accordance with MRP-175. MRP-175 Appendix B states that the threshold value is [ ] dpa for PWRs. For the structures at neutron doses more than [ ] dpa, there is a proposed screening curve for IASCC initiation of austenitic stainless steels to account for the observed baffle bolt failures in Europe.

The irradiation level and the metal temperature of the reactor internals are shown in table A. This table states that the neutron reflector alignment pin, the tie-rod, and the core barrel are over [ ] dpa. Then, the exposure screening criteria are calculated below according as the screening curve for IASCC and are also shown in Table A.

The stress evaluation was performed for J-APWR, the stress level was equivalent to Sy (18.0 ksi for 650°F). The US-APWR core power density is lower than J-APWR, so the US-APWR reactor internals metal temperature are not hotter than J-APWR. As a result, the stress level for US-APWR is lower than J-APWR, so these components in Table A shall be acceptable for the stress criteria.

Table A

	Irradiation Level (dpa)	Temperature (°F)	Stress Threshold for IASCC (ksi)	Stress Threshold for IASCC (MPa)
Neutron refractor Block alignment pin				
Neutron refractor Tie-rod				
Core barrel				

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-10

The DCD Tier 2, Subsection 3.9.5.3.4 includes the potential effects of irradiation stress relaxation in the list of environmental effects on reactor core internal materials caused by long term exposure to fast neutron irradiation. The applicant stated that neutron fluence and temperature limits are imposed on the tie-rods to preclude excessive loss of preload from irradiation stress relaxation.

The staff reviewed the DCD but did not find where the applicant had provided an evaluation of the loss of preload in various threaded fasteners due to irradiation stress relaxation or a reference where this information is available. The applicant did not identify the fasteners where the effect of irradiation stress relaxation is expected to be significant. Also, it is not clear how the pre-stress will be maintained in the preloaded components such as the ring block tie-rods or guide tube hold-down bolts. The applicant is requested to provide an assessment of the potential loss of preload due to irradiation stress relaxation in various threaded fasteners, in particular the guide tube hold-down bolts, guide tube support pins and the flexible leaves, and the neutron reflector tie-rods, and examine its effect on the structural and functional integrity of the components. Alternately, provide a reference document where this information is available. The requested information will assure conformance with GDC-1 and 4. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

In response to RAI 3.9.5-9, the irradiation level of various threaded fasteners is much smaller than tie-rod, and the volume of the potential loss is small.

The neutron level of the tie-rod is comparable to the neutron reflector block, and the preload of the tie-rod have been decreased for 60 years.

However, the function of the tie-rod is to restrain the axial response of the neutron reflector blocks in SSE and LOCA events if the tie-rod is not fastened, the function requirement of

tie-rod is acceptable.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-11

In Subsection 3.9.5.2 of the DCD the applicant has identified the loading conditions that have been considered in the design of US-APWR core support structures and internals components. The list includes pressure differences due to coolant flow.

However, the applicant did not provide any details regarding the method used to determine the pressure differences for reactor internal components during different operating conditions or to validate the calculated values. The applicant is requested to provide a description and validation of the method for determining the maximum pressure differences for reactor internals during ASME Code, Section III, Level A, B, C, and D service conditions. Alternately, provide a reference document where this information is available. The requested information is needed to assure conformance with GDC-1, 2, 4, and 10. Revise the DCD to include the requested information or provide a reference where this information is available.

ANSWER:

For the input to the stress analysis of the core support structures, the maximum pressure difference across the core barrel thickness is determined from the total pressure loss from the vessel inlet to the outlet by following formula.

$$\text{Delta } P = \sum k\,i\,(1/2)\rho\, vi^{2}$$

Delta P: maximum pressure deference across the core barrel thickness.
k I : irrecoverable pressure loss coefficient for each part*.
vi : local flow velocity for the each part*.
ρ : fluid mass density.

*Each part means the downcomer, the lower plenum. the lower core support plate, the fuel assembly, the upper core plate and the upper plenum.

The formula and pressure loss coefficients are determined based on the design hand books

and verified with the experience of the current plants.

The Design Difference Pressure [ ] psi is determined based on [ ] which can represent the Level A (normal operation) and Level B (up-set) service conditions as described in Table 7-7 of Reference (1). The pressure difference in Level C service conditions is also represented by the design conditions..

For the Level D service condition, time histories of dynamic pressure differences in a LOCA event were obtained in the blow-down analysis as answered to RAI 3.9.2-54.(RAI No.207-1577 revision 0). Further information about the analysis method and the computer code is described in Section 4.2.1.1 of Reference (2)

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.

Reference (1): MUAP-090004 R0 "Summary of Stress Analysis Results for the US-APWR Core Support Structures", Mach 2009.

Reference (2): MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping", January 2009 (DCD Ref. 3.9-58)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-12

In Subsection 3.9.5.2 of the DCD the applicant stated that pressure differences due to the coolant flow have been taken into account in designing the US-APWR core support and internal structures. The complete list of loading conditions that have been considered in the reactor internals design is given in Table 3.9-11 of the DCD. The applicant further stated in Subsection 3.9.5.3.2 of the DCD that the thermal-hydraulic performance criteria require the pressure drops across the reactor internals to meet system requirements for all Level A and B service conditions.

The staff reviewed Section 3.9.5 of the DCD but did not find where the applicant had provided estimates of the maximum pressure differentials for the reactor internals, and verified that they meet the thermal-hydraulics performance requirements of Subsection 3.9.5.3.2. The applicant is therefore requested to describe the system requirements for pressure differentials across reactor internals, and provide an assessment of the maximum pressure differentials for the reactor internals with respect to the design basis system requirements. The requested information will assure conformance with GDC-1, 2, 4, and 10. Revise the DCD to include the requested information.

ANSWER:

For the base of the thermal hydraulic design of the reactor system, total pressure loss of the reactor vessel shall be 48.2±4.8 psi as described in Table 4.4-1 in 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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-14

DCD Tier 2, Subsection 3.9.5.2 identifies the loading conditions that have been considered in the design of US-APWR core support structure and reactor internals components. However, Subsection 3.9.5.2 does not include any error analysis.

The applicant is therefore requested to provide detailed analysis of expected bias errors and random uncertainties included in predicting the vibration responses of reactor core support and internal structure, steam generator internal components, and of other plant systems and components. In response to this RAI, the applicant is expected to provide the total (or end-to-end) bias error and random uncertainties, and to substantiate the contributions of each of the following tasks to the total bias and uncertainties :

1. Modelling and validation of the forcing functions
2. Modelling and validation of the acoustic environment using SYSNOISE
3. FE modelling and validation of structural dynamic characteristics
4. Combining the forcing functions and system dynamic characteristics to estimate the dynamic response of structures and components
5. Experimental measurements which are used to validate models and analysis, whether these measurements are performed in-plant or in the laboratory by means of scale model testing.

The applicant is also expected to explain how the bias and uncertainties are implemented in the calculation of the minimum safety margin.

The Staff needs this information to evaluate the (minimum) margin of safety for the dynamic stress of various core support and reactor internals components and thereby assure conformance with GDC-1, 2, and 4. Revise Subsection 3.9.5.2 of the DCD to include analysis of bias errors and random uncertainties.

ANSWER:

The bias error and uncertainties in the FIV assessment analysis was included in the verification process with the bench mark analysis of the APWR1/5 Scale Model Test, as

discussed in Chapter 3.2 of Reference (1).

Typical values of the bias and uncertainties are as follows.

1. The flow-induced forcing functions: The down-comer flow turbulence forcing function had [ ]% bias and [ ]% uncertainties. Cross-flow induced loads had a factor of [ ] uncertainties.

2. RCP induced forcing functions: [ ]% uncertainty was assumed on the acoustic resonance frequency by the SYSNOISE analysis. In the case of acoustic resonance with the RCP pulsations, the pressure amplitudes from the SYSNOISE analysis were conservative because energy dissipation due to structural damping was not included in the analysis.

3. FE Modeling : Typical uncertainty in the computed natural frequencies is [ ] % based on comparisons of the analysis and measured results. The uncertainties in the computed structural natural frequencies were caused by the uncertainties in the mass and the stiffness. For the same forcing function, structural responses depend only on the stiffness. Thus uncertainties in the mass do not directly affect the computed structural response. Since modal frequencies vary as the square root of modal stiffness, if we assume the entire [ ]% uncertainty in the natural frequencies were caused by uncertainties in the modal stiffness (which is a very conservative assumption), this uncertainty in the natural frequencies would lead to [ ] % uncertainty in the computed vibration amplitude.

4. In turbulence-induced response analysis: About [ ] % bias included in the vibration response caused by the difference of the damping ratios between the bench mark analysis of 1/5 Scale Model Test ([ ] % damping ratio to the critical damping ratio) and the design analysis of the US-APWR ([ ] % damping ratio to the critical damping ratio). This is because turbulence-induced vibration amplitudes vary as the inverse square root of the damping ratios.

5. Experimental measurements: As an example of the uncertainties in scale model flow-induced vibration test, the results of the uncertainty study in the US-APWR Lower Plenum Test in Ref.(3) are shown below.

Table A3-4 Uncertainty in Flow-induced Vibration Measurement

Item	$u_i/\lvert Y\rvert$ (%)	$u_C/\lvert Y\rvert$ (%)
Flow rate setting		
Measurement system		
Measured data		

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.

Reference (1):MUAP-070023 R1 "APWR Reactor Internals 1/5 Scale Model Flow Test Report" ,May 2009,

Reference (2): MUAP-070027 R1 "The comprehensive vibration assessment program for the US-APWR Reactor Internals", May 2009

Reference (3): MUAP-070022 R0 "Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report", June 2008

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-15

In Subsection 3.9.5.2.2 of the DCD the applicant stated that the service limits for reactor internals other than the core support structures (CSSs) are not addressed in the ASME Code, Section III. However, because the structural integrity of the reactor internals are important-to-safety, the stress limits for CSSs are also applied to the reactor internals. If the stress limits for the internal structure do not meet the ASME Code, Section III limits for the CSSs, then the applicant proposes to utilize alternate acceptance criteria based on validation by testing, sound engineering judgment, and experience with similar design. The staff's review of the DCD showed that the applicant neither provided sufficient information about the proposed alternate acceptance criteria nor on the resulting safety margin.

The applicant is requested to explain in more detail the meaning of the following statement, which is given in Subsection 3.9.5.2.2 of the DCD: "However, if the stress limits for the internal structure do not meet the ASME Code, Section III (Reference 3.9-1) limits for the core support structures, then alternate acceptance criteria are employed based on validation by testing, sound engineering judgment, and experience with similar designs." Provide a list of all components, which did not meet the ASME Code for stress limits and explain the alternate design criteria used for these components. Information about these alternate acceptance criteria is needed to assure conformance with GDC-1, 2, 4, and 10.
Revise Section 3.9.5 of the DCD to provide the requested information.

ANSWER:

The loading conditions and stress limit for the Class CS were applied for the reactor internals except the secondary core support structures.

The function of the secondary core support assemblies was to limit the stroke of the drop and the impact force on the lower vessel head in the postulated core drop event. Therefore, the design of the secondary core support structures including the lower diffuser plate are determined with the impact force in the core drop event as a beyond-design basis accidents. For the stress limit, those for Level D of Class CS are applied as shown in Table 1.

Table 1 Load combination and acceptance criteria for the secondary core support structures

Operating conditions	Events	Occurrence	Stress Limit	Remarks
Design				
Beyond design basis accidents				

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-16

The applicant has stated in Subsection 3.9.5.3 of the DCD that the rules for design of the US-APWR core support structures (CSSs) and internal structures follow those in Section III, Subsection NG of the ASME Boiler and pressure Vessel Code (2001 Edition up to and including 2003 Addenda). Also, in DCD Section 3.9.3 the applicant stated that the environmental effects on fatigue of ASME Code, Section III, Class 1 components follow the guidance delineated in RG 1.207.

The staff's review of Section 3.9.5 of the DCD showed that the stress categories and stress intensity limits for CSS given in Table 3.9-12 do not include fatigue. The applicant should explain why fatigue evaluation was excluded from the design bases for CSSs and reactor internals, and provide a technical basis for the exclusion. The applicant is therefore requested to provide the reason and technical justification why fatigue evaluation, including the effects of PWR coolant environment, is not included in the list of CSS stress categories and stress intensity limits given in Table 3.9-12 of the DCD.
Review of the requested information regarding the reactor internals design is necessary to assure conformance with GDC-1, 2, 4, and 10. Revise Subsection 3.9.5.3 of the DCD to include the requested information.

ANSWER:

The category of fatigue was included in the Stress Category in Table 3.9-12 Chapter 3 of the DCD which is repeated below. The ASME "Hopper Diagram" in Subsection NG, Fig. NG-3221-1 does not list fatigue as a category but as a service limit stated as $Pm + Pb + Q + F < S_a$, where F is the peak stress, and S_a is the allowable alternating stress. The industry practice is to consider the fatigue service limit as $\sum U$, which is defined as the cumulative fatigue usage factor. In order to clarify the fatigue limit, Table 3.9-12 will be changed in the DCD as shown below.

Compliance with ASME Code fatigue requirements has always been one of the most important structural calculations in reactor internals. It should be noted though, that the

effects of the PWR coolant environment in regards to RG 1.207 is considered not applicable to the reactor internals since the requirements in Reg. Guide 1.207 applies to Class 1 components under pressure boundary conditions. The core support structures are not under pressure boundary conditions and furthermore, are classified as Class CS, and therefore are considered exempt from the PWR environmental fatigue penalty.

Table 3.9-12 Core Support Structures Stress Categories and Stress Intensity Limits

Stress Category		Service Limits			
		Design	Level A + Level B	Level C	Level D
Primary Membrane	P_m	S_m	S_m	$1.5S_m$	Lesser of $2.4S_m$ and $0.7S_u$
Primary Membrane + Bending	$P_m + P_b$	$1.5S_m$	$1.5S_m$	$2.25S_m$	Lesser of $3.6S_m$ and $1.05S_u$
Primary Membrane + Bending + Secondary	$P_m + P_b + Q$	---	$3.0S_m$	---	---
Peak	$P_m + P_b + Q + F$	---	S_a	---	---
Fatigue Usage Factor	$\sum U$ (2)	---	< 1.0	---	---
Average Primary Shear		$0.6S_m$	$0.6S_m$ (1)	$0.9S_m$	$1.2S_m$
Bearing		S_y ($1.5S_y$)	S_y (1) ($1.5S_y$) (1)	$1.5S_y$ ($2.25S_y$)	$2S_y$ ($3S_y$)

Note:

1. Applied only as Level A service limits
2. $\sum U$ is defined as actual service condition cycles divided by allowable cycles, based on calculated alternating stress and fatigue curve. Alternating stress is adjusted by the modulus of elasticity.

Impact on DCD

Table 3.9-12 will be revised.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-17

The load and displacement limits for the reactor internals that affect the safety and operability of the interface components are summarized in Table 3.9-2 of the DCD. However, the DCD does not give any details how the deformation limits were determined, or provide the technical basis for these deformation limits. As stated in SRP Section 3.9.5, SRP Acceptance Criteria, deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report, and the basis for these limits should be included. Also, the stresses for these displacements should not exceed the specified limits.

The applicant is requested to provide the technical basis for defining the displacement limits listed in Table 3.9-2 of the DCD. Alternately, provide a reference document where this information is available. Review of the requested information regarding the reactor internals design is necessary to assure conformance with GDC-1, 2, 4, and 10. Revise Subsection 3.9.5.2.3 of the DCD to DRAFT REQUEST FOR ADDITIONAL INFORMATION EMB1 2446 REVISION 0 include the requested information or provide a reference where this information is available.

ANSWER:

The technical basis of the loads and deformation limits in Table 3.9-2 of DCD are explained as follows.

(a) Allowable horizontal load of the RCCA guide tube should not impede insertion of the RCCA after the LOCA event.

Technical Basis: The horizontal load limit provides assurance that after a SSE + LOCA combined event, the inelastic deformation of the guide tube is such that the control rods will be unimpeded during rod drop insertion. The horizontal load or displacement limit is determined from testing.

(b) Upper core barrel radial displacement to prevent impeding emergency core cooling flow in RV downcomer.

Technical Basis: The limit of the radial outward deformation of the upper core barrel, 60 mm, is determined such that the flow area of the connection part of the inlet nozzle to the downcomer is not smaller than the inlet pipe section area.

(c) RV and upper head flange loads
Lower radial key loads
Postulated core drop bottom of RV impact load and bearing area

Technical Basis: Lower radial key loads are limited by the reactor vessel radial restraints. Postulated core drop bottom of RV impact load and bearing area are also limited by the reactor vessel bottom head stresses.

(d) The maximum vertical displacement of the upper core plate relative to the upper support plate should preclude buckling of the guide tube.

Technical Basis: The maximum relative displacement between the upper core plate and the upper core support plate 3mm is based on the axial clearance of the shoulder of GT support pin and the upper core plate to avoid the axial loading on the guide tube as shown in Figure 1.

(e) Upper core barrel permanent displacement should not prevent loss of function of the RCCA by radial inwardly deforming the upper guide tube.

Technical Basis: The maximum inward radial deformation of the upper core barrel of 270mm is determined based on the horizontal distance between the lower guide tube and the core barrel inside wall to prevent the interaction with the guide tube.

Figure 1 Close-UP of the RCCA guide tube shoulder and the upper core plate

---

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-19

According to the recommendations in Appendix A of SRP 3.9.5, the applicant is expected to evaluate potential adverse flow effects on piping and components of plant systems.

The staff reviewed Section 3.9.5 of the DCD and found that the applicant did not include an evaluation of these effects. The applicant is requested to provide a detailed evaluation of potential adverse effects from flow-induced vibrations and acoustic resonances on piping and components of plant systems, including the reactor coolant, steam, feedwater, and condensate systems. Flow-induced vibrations of various sampling probes should also be evaluated. Also, substantiate any assumptions made in the analysis, particularly for damping coefficients of structural elements. The staff needs this evaluation for the above mentioned plant components to assure conformance with GDC-1 and GDC-4. Revise Subsection 3.9.5.2 of the DCD to include a detailed evaluation of potential adverse flow effects on piping and components of plant systems, including the sampling probes, or refer to this evaluation if it is included elsewhere in the DCD.

ANSWER:

MHI designed the US-APWR piping using the structural design rules based on years of empirical experience with similar piping. Therefore, flow-induced vibrations or acoustic resonances do not affect the US-APWR piping adversely.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-20

Appendix A of SRP 3.9.5 also recommends that the applicant should maintain monitoring of potential adverse flow effects on plant systems and components for a sufficient time period to verify that adverse flow effects are not occurring (See Appendix A, Item 7 of SRP 3.9.5 for more details).

The staff reviewed Section 3.9.5 of the DCD and found that the applicant did not discuss monitoring of potential adverse flow effects. The applicant is therefore requested to discuss the plans for monitoring potential adverse flow effects in the plant after the initial start-up period. Previous plant experience has shown that adverse flow effects might not appear for an extended period of time following initial start-up. The staff needs information about the monitoring program to complete the review and to evaluate conformance with GDC-1 and GDC-4. Revise Subsection 3.9.5.2 of the DCD to include adequate information about monitoring of potential adverse flow effects on plant systems and components.

ANSWER:

In accordance with the RG.1.20 the vibration monitoring will be performed in the pre-operational testing of the first US-APWR reactor. Based on the vibration response analyses, the vibration measurements without the core will give the conservative responses than those with the core. Therefore, the vibration measurement will be conducted before core lading as discussed in Subsection 3.9.2.3 and 3.9.2.4 of DCD. More details of this assessment are described in Section 3.3 of Reference (1) and the vibration measurement plan is included in Section 4 of the same reference.

The needs of vibration measurement after the core loading have been also discussed in RAI 206-1576 Question 3.9.2-43 on DCD 3.9.2.4.

In addition, as discussed in DCD Subsection 4.4.6.3, ex-core neutron detectors can be used to provide continuous monitoring of the core vibration. The detected signals are recorded and analyzed based on a spectrum analyzer. Since the vibration frequencies and amplitudes are

measured in a preoperational test, the correlation between the detected signals and the core vibration characteristics can be established.

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.

Reference (1): MUAP-070027 R1 "The comprehensive vibration assessment program for the US-APWR Reactor Internals", May 2009

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-22

The applicant states in DCD Tier 2, Subsection 3.9.5.3 that the rules for materials, design, fabrication, examination, and preparation of reports for the manufacture and installation of the US-APWR core support structures (CSSs) and internal structure follow those in Section III, Subsection NG of the ASME Boiler and Pressure and Vessel Code, 2001 Edition up to and including 2003 Addenda. Additional codes, standards, regulations, and guidelines from the NRC and the Utility Requirements Document are adhered to, and are listed in the Owner's design specification. However, these additional design codes, code cases, and acceptance criteria are not identified in the DCD. Section 3.9.5 of the SRP states that if other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified.

The applicant is requested to provide a list and justification of the applicable codes, standards, regulations, and guidelines, for the design of US-APWR CSSs and reactor internals, if different from ASME III, Subsection NG requirements. The requested information is needed to assure conformance with GDC-1. Revise the DCD to include the requested information.

ANSWER:

The applicable codes and standards for the US-APWR reactor internals are referenced in the Design Specification which is available at NRC inspection.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-23

The reactor coolant flow path for the reactor internals is described in Subsection 3.9.5.3.2 of the DCD. The applicant states that the main coolant flow enters the bottom of the RV and turns upward, flowing past the diffuser plates and distributing into the lower core support plate orificed holes. The orifices are carefully designed to control the flow into the fuel assemblies and to minimize uneven flow distributions and hot spots.
The thermal-hydraulics performance criteria for the design of the US-APWR core support and internal structure, listed in Subsection 3.9.5.3.2, require that the distribution of main coolant inlet flow into the fuel assemblies during normal operation must meet fuel assembly core inlet requirements. However, the DCD does not provide any details about these requirements or how compliance with the requirements is verified. The applicant is requested to provide additional details regarding the fuel assembly core inlet requirements to explain how compliance with the requirements during service is verified.
The requested information is needed to confirm compliance with the thermal-hydraulics design basis requirements for the design of the US-APWR core support and internal structure, and assure conformance with GDC-4, and -10. Revise the DCD to include the requested information.

ANSWER:

Please refer to the response to QUESTION NO.: 03.09.05-7 of this RAI.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-24

In Subsection 3.9.5.3.2 of the DCD the applicant states that the main coolant flow then mixes in the upper plenum and exits from the core barrel outlet nozzles at an average fluid temperature of Thot. The applicant further states that special flow columns are spaced on the periphery of the upper core plate near the core barrel outlet nozzles in order to improve mixing and minimize outlet fluid temperature mal-distribution.
The thermal-hydraulics performance criteria for the design of core support and internal structures, identified in Subsection 3.9.5.3.2, require that the main coolant flow into the outlet piping during normal operation meets the system requirements, specifically (a) exit fluid temperature striations are minimized, and (b) velocity criteria to prevent erosion are met. The DCD does not provide any details how compliance with these system requirements is verified. The applicant is requested to provide additional details regarding the system requirements for exit fluid velocity and temperature striations, and describe the procedure used to verify compliance with these requirements during service. The requested information is needed to confirm compliance with the thermal-hydraulics design basis requirement for the design of the US-APWR core support and internal structures and assure conformance with GDC-4 and GDC-10. Revise the DCD to include the requested information.

ANSWER:

(a) Exit fluid temperature striations
"Minimize of temperature striation" is only a future of the better design of the upper internals but no design criterion is determined for the exit fluid temperature striations.

(b) Flow velocity of the exit of reactor vessel
The outlet diameter of the outlet was determined to be corresponded with the increase of flow rate from the current 4 loop design to assure the equivalent margin to the erosion.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-26

The applicant states in Subsection 3.9.5.3.12 of the DCD that the pre-service inspection as well as the in-service inspection (ISI) plans follow the rules of ASME Code, Section XI. However, the applicant does not examine or discuss the adequacy of the ASME Code, Section XI ISI plan to detect environmental effects on the structural and functional integrity of the core support and internal structures during their 60-year design life.

For license renewal of operating reactors, the staff has reviewed the aging effects on components and structures, identified the relevant existing aging management programs (AMPs), and evaluated the program attributes to determine where existing programs are adequate without modifications and where existing programs should be augmented for the extended period of operation. The evaluation results documented in the Generic Aging Lessons Learned (GALL) report, NUREG-1801, Rev. 1, indicate that the ASME Code, Section XI ISI program is inadequate to manage aging effects such as cracking due to SCC or IASCC, change in dimensions due to void swelling, loss of fracture toughness due to neutron irradiation embrittlement, or loss of pre-load due to irradiation stress relaxation.
The applicant is requested to provide a commitment to:
(a) Review and evaluate the effect of environmental degradation processes such as SCC, IASCC, PWSCC, degradation of fatigue strength, radiation embrittlement, and void swelling on the structural and functional integrity of the reactor core support and internals components.
(b) Define the range of environmental and service conditions under which these environmental effects can be significant.
(c) Evaluate and implement the results of the industry programs for investigating and managing environmental effects as applicable to reactor core support and internal structures.
(d) Develop an inspection plan for reactor core support and internal structures that addresses these service conditions and environmental degradation issues.
This information is needed for timely detection the effects of environmental effects on the structural and functional integrity of the US-APWR core support and internal structures and components and assure conformance with GDC-1, -4, and -10. Revise Section 3.9.5 of the DCD to include the requested information.

ANSWER:

a) A commitment to addressing environmental conditions is in responses RAI 03.09.05.9; RAI 03.09.05.10; and RAI 03.09.05.16.

b) The definition of the range of environmental and service conditions that are significant will depend largely on the MRP program investigations and resolutions; particularly with regards to reactor internals.

c) As noted in the RAIs mentioned above, evaluation and implementation of industry programs and recommendations will be considered when available.

d) An inspection plan that addresses service conditions and environmental degradation issues is part of the on-going industry investigations in environmental effects and will be considered for implementation when there is consensus throughout the industry on a planned approach.

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

7/17/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 374-2446 REVISION 0**
SRP SECTION:	**3.9.5 – REACTOR PRESSURE VESEL INTERNALS**
APPLICATION SECTION:	**03.09.05**
DATE OF RAI ISSUE:	**5/21/2009**

QUESTION NO.: 03.09.05-27

A description of the US-APWR upper reactor internals assembly design arrangement, including the classification of the various upper reactor internals components is presented in Subsection 3.9.5.1 of the DCD. The applicant stated that both the upper core support assembly and lower core support assembly are classified as Core Support Structures (CSS). The design bases requirements provided in DCD Section 3.9.5.3 specifies that those reactor internals components classified as CSS conform to the materials, design, fabrication, examination, and documentation requirements of the ASME Boiler and Pressure Vessel Code, Section III (ASME III), Subsection NG, 2001 Edition through the 2003 Addenda.

The staff's review of DCD Subsection 3.9.5.1, together with DCD Subsection 3.9.5.1.3, revealed that the applicant did not clearly define the classification of the reactor internals hold-down spring, which is a load bearing component of the upper core support assembly. The jurisdictional boundaries of the reactor internals are defined in DCD Subsection 3.9.5.1.3. The fourth bullet item defines the boundary between the components classified as CSS and components classified as internal structures, following the guidance for boundaries of jurisdiction in the ASME Code Section III, Subsection NG-1000. The second line item under the fourth bullet states, "Upper core support flange and core barrel flange with the reactor internals hold-down spring." The staff review interprets this DCD statement to mean that the reactor internals hold-down spring is classified by the applicant as internal structure, as opposed to a core support structure, for purposes of specifying the applicable design code requirements for the hold-down spring. Based on the staff's interpretation of the DCD statement, the classification of the reactor internals hold-down spring appears to be inconsistent with the requirements of GDC-1 and 10 CFR 50.55a. ASME III, Article NG-1121 defines core support structures as structures or parts of structures which provide direct support or restraint of the core (fuel and blanket assemblies) within the reactor pressure vessel. The staff considers the reactor internals hold-down spring, together with the upper core support assembly and lower core support assembly, to be complementary parts of the load bearing assembly providing support for the reactor core. The applicant is requested to: (a) provide clarification for the classification of the reactor internals hold-down spring; (b) provide technical justification for any classification which would not require use of the design, fabrication, examination, and documentation requirements of the ASME Code Section III, Subsection NG for design of the hold-down spring; and (c) revise DCD Sections 3.9.5.1 and 3.9.5.1.3, and DCD Table 3.2-2, including the requested information. The staff requires this

information to assure conformance with the regulatory requirements of GDC-1.

ANSWER:

The response to questions on the hold-down spring classification and function are also explained in RAI 03.09.05.2.

A summary response justifying classification of the hold-down spring as an internal structure is discussed below.

(a) Classification

The hold-down spring is classified as an internal structure.

(b) Technical justification

The primary functions of the hold-down spring are to allow compliance for thermal expansion between the reactor vessel and the reactor internals (upper support flange and core barrel flange), and provide sufficient preload to the flanges to prevent excessive vibration or sliding during operation. These are considered to be not core support requirements, but functional requirements.

The reason the hold-down spring has always been classified as an internal structure is because it is not required to directly support the core. The computational modeling of the reactor internals, for example, has the load path of the hold-down spring not connected in series but in parallel with those of the fuel assemblies.

One other point should be made. Even if the hold-down spring loses all its preload from stress relaxation, the shape of the hold-down spring will remain unchanged – and the vertical loads from the core can still be transferred through the hold-down spring to the upper support and core barrel flanges and then to the vessel head and vessel flange. This extreme example of complete loss of preload is undesirable from a functional standpoint because of the potential adverse effects on vibration and sliding of the reactor internals, but does not warrant re-classifying the hold-down spring as a core support structure.

(c) Revise of DCD

Table 3.2-2 of DCD will be revised to including "internal structures", based on the response to RAI 03.02.01-14.

Table 3.2-2 Classification of Mechanical and Fluid Systems, Components, and Equipment **(Sheet 1 of 53)**

System and Components	Equipment Class	Location	Quality Group	10 CFR 50 Appendix B (Reference 3.2-8)	Codes and Standards(3)	Seismic Category	Notes
Primary System							
1. Reactor Systems							
Fuel assemblies	1	PCCV	A	YES	5	I	
Rod control cluster	1	PCCV	A	YES	5	I	
Burnable poison	1	PCCV	A	YES	5	I	
Neutron source assemblies	1	PCCV	A	YES	5	I	
Upper core support **assembly**	1	PCCV	A	YES	ASME III, CS	I	
Lower core support **assembly**	1	PCCV	A	YES	ASME III, CS	I	
Guide tube assemblies	1	PCCV	A	YES	5	I	
Control rod drive mechanism latch housing	1	PCCV	A	YES	1	I	
Control rod drive mechanism rod travel housing	1	PCCV	A	YES	1	I	
Internal structures	**4**	**PCCV**	**D**	**N/A**	**4**	**II**	
2. Reactor Coolant System							
Reactor vessel	1	PCCV	A	YES	1	I	
Reactor vessel head	1	PCCV	A	YES	1	I	
Surveillance capsule guide basket	**5**	**PCCV**	**N/A**	**N/A**	**5**	**II**	
Reactor vessel insulation (shell)	**5**	**PCCV**	**N/A**	**N/A**	**5**	**II**	
Reactor vessel insulation (closure head)	**5**	**PCCV**	**N/A**	**N/A**	**5**	**II**	
Reactor coolant pump casing	1	PCCV	A	YES	1	I	
Reactor coolant pump main flange	1	PCCV	A	YES	1	I	
Reactor coolant pump thermal barrier	1	PCCV	A	YES	1	I	
Reactor coolant pump thermal barrier heat exchanger	1	PCCV	A	YES	1	I	
Reactor coolant pump #1 seal housing	1	PCCV	A	YES	1	I	
Reactor coolant pump #2 seal housing	2	PCCV	B	YES	2	I	

Impact on DCD

DCD will be revised as described in c) of the above answer.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.