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

January 18, 2010

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffery A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-10009

Subject: MHI's Response to US-APWR DCD RAI No. 502-3979 Revision 2

Reference: 1) "Request for Additional Information No. 502-3979 Revision 2, SRP Section: 04.05.02 – Reactor Internal and Core Support Structure Materials

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") documents as listed in Enclosure.

Enclosed is the response to 1 RAI contained within Reference 1.

As indicated in the enclosed materials, this submittal contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

Sincerely,

Y. Ogata

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

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Enclosures:

1. Affidavit of Yoshiki Ogata
2. "Response to Request for Additional Information No. 502-3979, Revision 2" (Proprietary Version)
3. "Response to Request for Additional Information No. 502-3979, Revision 2" (Non-Proprietary Version)

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

Contact Information
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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-10009

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. 502-3979, Revision 2", dated January 18, 2010, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

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

4. The basis for holding the referenced information confidential is that it describes the unique design parameters developed by MHI for the Reactor Internals and Core Support Structures.

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 impact on the competitive position of MHI in the U.S. nuclear plant market:

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

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 18th day of January 2010.

Y. Ogata

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

Enclosure 3

UAP-HF-10009
Docket No. 52-021

Response to Request for Additional Information No. 502-3979, Revision 2

January 2010
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/18/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 502-3979R2**
SRP SECTION:	**04.05.02 – Reactor Internal and Core Support Structure Materials**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**12/01/2009**

QUESTION NO.: RAI 4.5.2-17

In its response to US-APWR DCD RAI 414-3102 Question 04.05-11, MHI discusses welding of the radial supports to the reactor vessel. However, additional information is needed to ensure compliance with 10CFR 50 Appendix A General Design Criterion (GDC) 31 as it relates to the reactor coolant pressure boundary behaving in a nonbrittle manner.

What controls (e.g. weld heat input limits, post weld heat treatments) will be imposed during the manufacturing process to ensure that welding of the radial supports to the reactor vessel does not embrittle the reactor vessel?

Will the reactor vessel be heat-treated after welding of the radial supports?

ANSWER:

Welding of the radial supports to the reactor vessel will be carried out in accordance with welding qualification procedures that satisfy the requirements of ASME Code Section IX.

The qualification procedures control the essential parameters (weld heat input temperatures, heat treatment temperatures and duration, etc.) that ensure the reactor vessel will not experience excessive embrittlement by the welding of the radial supports.

Furthermore, the reactor vessel will be post weld heat treated after the radial supports are welded which will reduce embrittlement.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/18/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 502-3979R2**
SRP SECTION:	**04.05.02 – Reactor Internal and Core Support Structure Materials**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**12/01/2009**

QUESTION NO.: RAI 4.5.2-18

In its response to US-APWR DCD RAI 414-3102 Question 04.05.02-13, MHI stated that the electron-beam welding process is used for the core-barrel welding, and that this welding is performed without adding weld materials. The staff needs additional information to determine compliance with GDC 1 as it relates to structures, systems, and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its importance to safety.

Please specify the codes and standards that will be used to qualify the welding procedures and welders/welding operators for the core-barrel welds?

ANSWER:

Welding locations on the core barrel and welding methods are summarized in Table 1.

Electron Beam Welding (EBW) process is applied for the weld of the core barrel flange to the upper barrel , axial welding of upper core barrel and welds for lower barrel segments.

Gas Tungsten Arc Weld (GTAW) (= Tungsten Inert Gas (TIG) weld) is applied for the weld of core barrel to LCSP. GTAW process is also applied for fix of attachments to the core barrel such as base pads for the Irradiation specimen guides.

The codes and standards that will be used to qualify the welding procedures and welders/welding operators for the core-barrel welds are as follows.

-ASME BOILER & PRESSURE VESSEL CODE Section III, Division 1- Subsection NG, NG-2400 for Welding material and NG-4300 for Welding procedure and Welders/Welding operators.

-ASME BOILER & PRESSURE VESSEL CODE Section IX, Part QW-200 for welding procedure and Part QW-300 for Welders/Welding operators.

Refer to QW-256 for GTAW and QW-260 for EBW.

Table 1 Core Barrel welding locations and methods

Locations or Parts	Welding method
CB flange to upper core barrel	EBW
upper core barrel axial	EBW
upper core barrel to lower core barrel	GTAW(TIG)
lower core barrel segments axial / circumferential	EBW
lower core barrel to LCSP	GTAW(TIG)
Radial key to LCSP	GTAW(TIG)
Outlet nozzle to upper core barrel	GTAW(TIG)
UCP alignment pins or NR alignment pins to CB	GTAW(TIG)
protection guides / pads for irradiation specimen guides to CB	GTAW(TIG)
safety Injection pad to CB	GTAW(TIG)

- **Impact on DCD**

There is no impact on the DCD.

- **Impact on COLA**

There is no impact on the COLA.

- **Impact on PRA**

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/18/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 502-3979R2**
SRP SECTION:	**04.05.02 – Reactor Internal and Core Support Structure Materials**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**12/01/2009**

QUESTION NO.: RAI 4.5.2-19

In its response to US-APWR DCD RAI 414-3102 Question 04.05.02-7, MHI stated that the potential for IASCC in the neutron reflector is less than that in the core-baffle structures in existing PWRs, and that in-service inspections based on ASME Code, Section XI requirements are sufficient to assure integrity of the neutron reflector. Please provide the technical bases for concluding that the effects of IASCC in the US-APWR neutron reflector will be less than that found in core-baffle structures in existing PWRs and are, thus, not significant. Discuss specific examples of operating experience in the U.S., Japan or other countries that support this statement.

ANSWER:

In general, the potential for IASCC depends on the irradiation, stress, temperature and material. In conventional PWR plants, the threaded fasteners on the core baffle are supposed as the critical parts for IASCC, because both the irradiation and stress are almost maximum level in reactor internals. In fact, some failures due to the IASCC have been observed on the threaded fasteners of core baffle in France.

As a concept of the neutron reflector design, threaded fasteners applied with high tensile stress are not located in the high influence region.
The maximum influence on the inside surface of the neutron reflector is estimated [ ] n/cm^2 in 60 years as stated in the response to RAI269-2155-4.5.2-2, dated 5/13/2009. This influence is in the same order of that on the core baffle structures in conventional PWR. But there is no high stress on the ring blocks of the neutron reflector like threaded fasteners.
For the tie rods of the neutron reflector, although some tensile stress are applied on them, the influence level is smaller in almost one order of magnitude from that on the core side of the ring block or the baffle fasteners in conventional PWRs.

From the above discussions, in-service inspections based on ASME Code, Section XI as the conventional PWR are sufficient to check the long term reliability of the neutron reflector.

- **Impact on DCD**

There is no impact on the DCD.

- **Impact on COLA**

There is no impact on the COLA.

- **Impact on PRA**

There is no impact on the PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/18/2010

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 502-3979R2**
SRP SECTION:	**04.05.02 – Reactor Internal and Core Support Structure Materials**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**12/01/2009**

QUESTION NO.: RAI 4.5.2-20

In response to US-APWR DCD RAI 414-3102 Question 04.05.02-14, MHI stated that the potential for IASCC in the core barrel is less than that found in existing PWRs, and that in-service inspections based on ASME Code, Section XI requirements are sufficient to assure integrity of the core barrel. Similar to the previous supplemental RAI on the USAPWR neutron reflector, please discuss the technical basis for determining that there will be no adverse effects of IASCC on the core barrel. Discuss any operating experience in the U.S., Japan or other countries that support this statement.

ANSWER:

The maximum influence on the core barrel of US-APWR is estimated [ ] n/cm^2 in 60 years as stated in the response to RAI269-2155-4.5.2-3, dated 5/13/2009. This influence is about one third of that in the conventional 4-loop PWR in same years. The reduction is obtained by the shield effect of the neutron reflector in US-APWR, replacing the conventional core baffle structures between the core and core barrel.

Other factors associated with IASCC, such as stress, temperature and water chemical conditions of the US-APWR core barrel are similar to those in a conventional PWR plant.

As for the plant operating experience, any failure of the core barrel due to IASCC is not identified in PWRs through in US, Europe and Japan.

From the above discussions, in-service inspections based on ASME Code, Section XI requirements same as conventional PWR are sufficient to assure the long term integrity of the US-PAWR 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.

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