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

May 14, 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-09243

Subject: MHI's Response to US-APWR DCD RAI No. 269-2155 Revision 1

Reference: 1) "Request for Additional Information No. 269-2155 Revision 1, 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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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. "Response to Request for Additional Information No. 269-2155, Revision 1" (Proprietary Version)

3. "Response to Request for Additional Information No. 269-2155, Revision 1" (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-09243

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. 269-2155, Revision 1", dated May 13, 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 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 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. Loss of competitive advantage of the US-APWR created by the benefits of the Control Rod Drive Mechanism operation.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 14th day of May 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09243
Docket No. 52-021

Response to Request for Additional Information No. 269-2155,
Revision 1

May 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 268-2155 REVISION 1**
SRP SECTION:	**04.05.02 - REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**4/9/09**

US-APWR Design Certification- 04.05.02 - Reactor Internal and Core Support Structure Materials
[Review performed against revision 0 of the US-APWR DCD Tier 2.]

QUESTION NO. : RAI 2155-4.5.2-1

Tables 4.5-1 and 4.5-2 specify weld material specifications SFA-5.9 or SFA-5.4 for the welding of CRDMs and reactor vessel internals. In addition, please provide the weld material types (e.g. ER308) to be used for welding CRDMs and reactor vessel internals

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ANSWER:

Welding material for the CRDM housing is SFA-5.9 ER316L and/or EC316L.
Welding materials for the reactor vessel internals are SFA-5.9 ER308L, SFA-5.9 ER316L and/or SFA-5.4 E308L.

Table 4.5-1 and Table 4.5-2 will be revised as attachment-1.
Table 4.5-1 will be reflected NRC comments in RAI 2181, too.

Additionally, note for alloy X-750 will be revised to correct reference code number and to add applicable part in Table 4.5-2.

Impact on DCD

DCD Revision 2 will incorporate change as attachment-1:

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 268-2155 REVISION 1**
SRP SECTION:	**04.05.02 - REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**4/9/09**

QUESTION NO. : RAI 2155-4.5.2-2

Table 4.5-2 provides material specifications used for reactor internals. FSAR Section 3.9.5 provides a description of the neutron reflector and states that it replaces the baffle structure used in existing reactors.

a) Provide in the FSAR a detailed drawing of the neutron reflector and its components. Confirm that the materials for the neutron reflector and its components are listed in Table 4.5-2 or, preferably, list the neutron reflector material as a separate item.

b) Discuss whether or not sharp notches will be fabricated into neutron reflector components (such as ring blocks and alignment pins).

c) Discuss your plans to address the effects of void swelling, if any on the neutron reflector.

d) What will be the end of life cumulative peak neutron fluence at the neutron reflector? At what point will the cumulative peak neutron fluence at the neutron reflector cross the threshold of $5x10^{20}$ n/cm^2 (E>1.0 MeV)? What is your plan when the neutron fluence crosses this threshold?

ANSWER:

a) The sketch in the attachment-2 which identifies components of the neutron reflector will be attached in the Subsection 3.9.5 of the DCD.

b) Sharp notches which lead to fail by fatigue shall be avoided not only the neutron reflector but also reactor internal structures.

c) Effect of void swelling such as displacement will be replied in June 2009.

d) Maximum fluence is [ ](E>1.0 MeV) at the corner of the neutron reflector at the end of design life, 60 years. The maximum fluence region, such as inner surface of the neutron reflector can be performed view inspection. The neutron reflector can be replaced, if there is a large damage which can not be repaired.

Impact on DCD

DCD Revision 2 will incorporate the following change:

Attachment-2 will be add in the Subsection 3.9 as Figure 3.9-9 and following sentence will be add 4th paragraph of the Subsection 3.9.5.1.2. "Figure 3.9-9 shows the components of the neutron reflector."

"Sharp notches which lead to fail by fatigue shall be avoided not only the neutron reflector but also reactor internal structures" will be added end of 4th paragraph of the Subsection 3.9.5.1.2.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 268-2155 REVISION 1**
SRP SECTION:	**04.05.02 - REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**4/9/09**

QUESTION NO. : RAI 2155-4.5.2-3

Section 4.5.2.1 states that Alloy 690 is used for the radial-support clevis insert. Section4.5.2.5 states that the Alloy 690 radial-support clevis inserts are bolted to the radial support keys. Section 3.9.5.2.1 states that the radial support keys are attached to the outside rim of the lower, core-support plate.

a) Describe how the Alloy 690 radial-support clevis inserts are attached to the reactor vessel. If welding processes are used, discuss what weld materials are used, preheat and interpass controls, and subsequent heat treatments to the reactor vessel.

b) Are radial support keys and the lower-core support plate considered "reactor vessel internals primary material" as described in Table 4.5-2? Describe how the radial support keys are attached to the outer rim of the lower-core support plate. Discuss welding processes, if welding is used.

c) Are radial-support keys and radial-support clevis inserts subjected to a fluence of $5x10^{20}$ n/cm^2 (E>1.0 MeV) or greater in the proposed 60-year life of the USAPWR, thereby making them susceptible to irradiated-assisted, stress-corrosion cracking (IASCC)?

ANSWER:

a) Alloy 690 radial support clevis inserts are attached to the radial support in the reactor vessel by the bolts made of Alloy X-750. These bolts are described in Subsection 4.5.2.5.

b) In Subsection 4.5.2.5, the radial support keys will be revised radial supports to comply with Subsection 5.3.3.1. The radial supports are one of the parts of the reactor vessel, which material including welding material are identified in Table 5.2.3-1.

c) The fluence of the radial supports are under [ ]n/cm^2 (E>1.0 MeV) at the end of the design life, 60 years. There is not any concern for the IASCC.

Impact on DCD

DCD Revision 2 will incorporate the following change:

2nd paragraph in Subsection 4.5.2.5 will be changed as following.

Alloy 690 radial-support clevis inserts are bolted to the radial support ~~keys~~.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 268-2155 REVISION 1**
SRP SECTION:	**04.05.02 - REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**4/9/09**

QUESTION NO. : RAI 2155-4.5.2-4

Section 3.9.5.2.1 states that the following welds are used in the core-barrel assembly:

-Forged-flange-to-upper-core-barrel welds
-Upper-core-barrel-to-lower-core-barrel welds
-Lower-core-barrel-to-lower-core-support-plate welds
a) Are any other welds used in the core-barrel assembly?
b) Will any of the core-barrel welds be subjected to a fluence of 5 $x10^{20}$ n/cm^2 (E>1.0 MeV) or greater in the proposed 60-year life of the USAPWR, thereby making the core barrel susceptible to irradiated-assisted stress-corrosion cracking (IASCC)?

ANSWER:

a) Yes. The following welds were not included because they are not structural welds or are used for attachment of internal structures:
- Upper core barrel -to– core barrel nozzle welds
- Upper core barrel –to- safety injection pad welds
- Lower core barrel –to– pad welds for alignment pins which align neutron reflector and upper core plate.
- Lower core barrel –to- irradiation specimen guide welds (this is the guide to protect the irradiation specimen guide during insertion of the core barrel)

b) Maximum fluence on the core barrel is [ ]n/cm^2 (E>1.0 MeV) at the end of design life, 60 years. The core barrel can be inspected.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

5/13/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 268-2155 REVISION 1**
SRP SECTION:	**04.05.02 - REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS**
APPLICATION SECTION:	**4.5.2**
DATE OF RAI ISSUE:	**4/9/09**

QUESTION NO. : RAI 2155-4.5.2-5

Section 4.5.2.2 provides a description of controls on welding of reactor internals and core-support structures.

a) Section 4.5.2.2 states, "For design temperatures up to and including 800F, the minimum acceptance delta ferrite is 5FN." Provide the upper delta ferrite limit for this temperature range.

b) Section 4.5.2.2 also states, "For design temperatures exceeding 800F, the delta ferrite is limited to the range FN3-FN10." Since RG1.31 specifies a lower ferrite limit of 5FN to avoid micro fissuring in welds, provide the technical basis for specifying a lower delta ferrite limit of 3FN for this temperature range.

ANSWER:

a) The upper delta ferrite is 20FN.

b) Reactor internals are used under 800 °F condition. MHI will delete the sentence to discuss over 800 °F condition.

Impact on DCD

DCD Revision 2 will incorporate the following change:

4.1.1.1 Controls on Welding

Welding of reactor internals and core support structures follows the rules of NG-2000, NG-4000, and NG-5000 and Section IX "Welding and Brazing Qualifications" of the ASME Code. The discussions provided in Subsection 5.2.3 are applicable to the welding of reactor internals and core support components which are in accordance with ASME Code Section III Subsection NG-4000. For design temperatures up to and including 800 °F, the minimum acceptance delta ferrite is 5FN. ~~For design temperatures exceeding 800 °F, the delta ferrite is limited to the range FN3-FN10.~~

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

Table 4.5-1 Summary of Control Rod Drive System Structural Materials

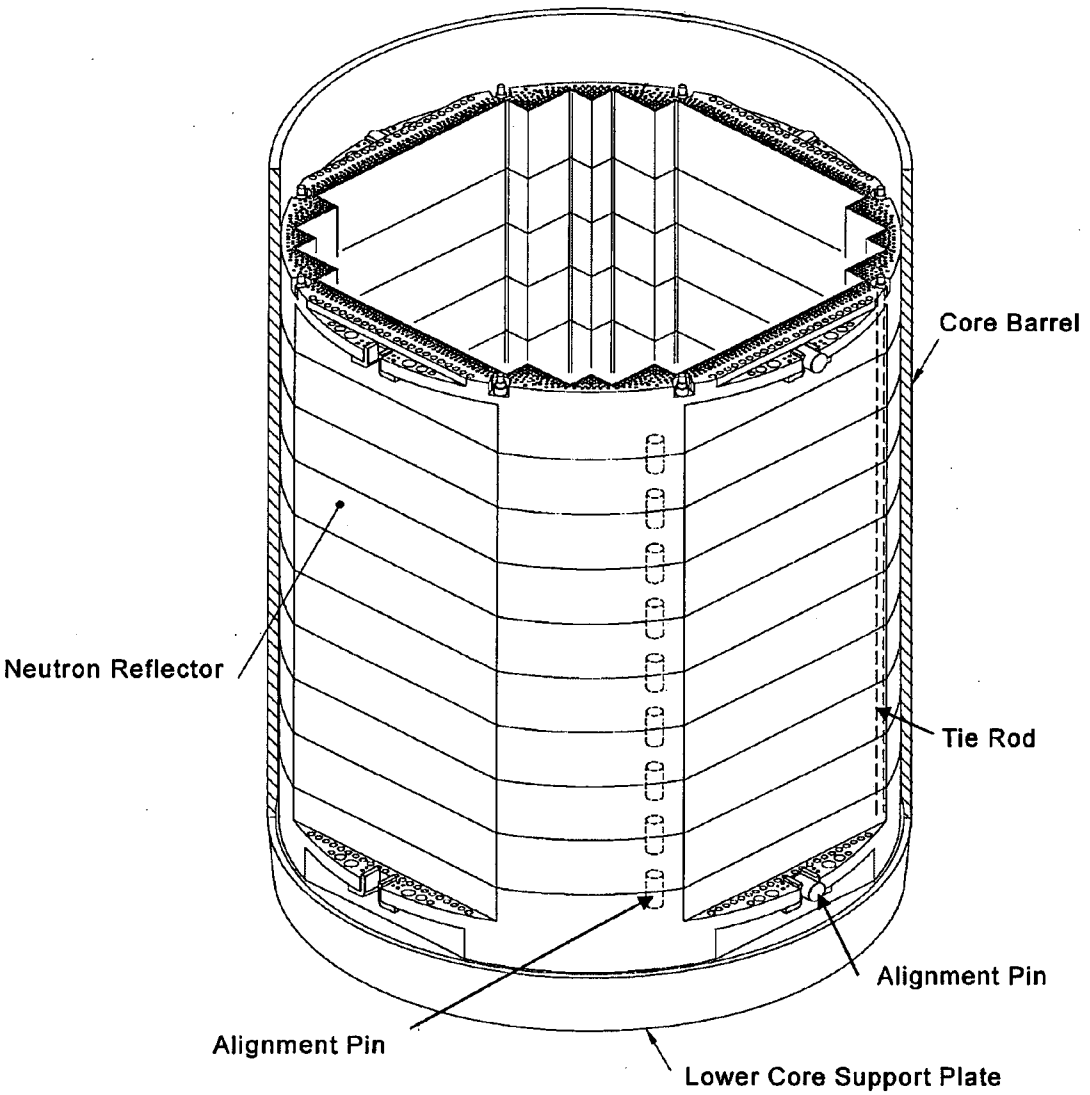
Component	Material Specification[1]	Environment
CRDM pressure housing material in contact with reactor coolant on the inside surface	SA-182 Grade F316	Inside surface exposed to reactor coolant water
Flux Ring	ASTM A519 Gr.1015	Not exposed to reactor coolant water
Latch assembly - magnetic poles, plungers, and keys	SA-479 Type 410	Exposed to reactor coolant water
Latch assembly - springs	Alloy X-750 (ASME SB637 N07750)[2]	Exposed to reactor coolant water
Latch assembly - link pins	Cobalt alloy (HAYNES No. 25 or equivalent material[3])	Exposed to reactor coolant water
Latch assembly - other parts	SA479 Type 304 SA-213 ~~Type~~ Grade TP 304	Exposed to reactor coolant water
Latch assembly - cladding on latch arm tips and pin holes	Cobalt alloy (Stellite No.6 or equivalent material[3])	Exposed to reactor coolant water
Latch assembly - plating on sliding surfaces	Chrome plate	Exposed to reactor coolant water
Latch assembly - coating on tips of latch arms	Chrome carbide	Exposed to reactor coolant water
Drive rod assembly - drive rod,	SA-268 TP410	Exposed to reactor coolant water
Drive rod assembly - unlatch button, protection sleeve	SA-479 Type 410	Exposed to reactor coolant water
Drive rod assembly - coupling	SA-479 Type 403	Exposed to reactor coolant water
Drive rod assembly - springs	Alloy X-750 (ASME SB637 N07750)[2]	Exposed to reactor coolant water
Locking button in the drive rod assembly and pins in the latch assembly	Cobalt alloy (HAYNES No. 25 or equivalent material[3])	Exposed to reactor coolant water
Drive rod assembly other parts	SA-479 Type 304	Exposed to reactor coolant water
Coil assembly - housing	~~SA356~~ ASTM A536 Grade 60-40-18	Not exposed to reactor coolant water
Coil assembly - coil bobbins	Glass silicone resin	Not exposed to reactor coolant water
Coil assembly - wire	Double glass insulated copper	Not exposed to reactor coolant water
Weld material used in CRDMs	SFA 5.9 ER316L EC316L	Not exposed to reactor coolant water

Notes: (1) Additional information appears in the text of Section 4.5 and Subsection 5.2.3.
(2) Additional stringent specification, ~~MIL-N-24114 Class A No.1~~ MIL-S-239192, is applied.
(3) Equivalent material is a substitute material having the same chemistry and properties as the preferred material. The same requirements would be specified in the procurement requirements.

Table 4.5-2 Summary of Reactor Internals and Core Support Materials

Component	Material Specification(1)
Reactor vessel internals - primary material	SA182 Grade F304 SA336 Grade F304 SA240 Type 304 SA479 Type 304/ Type 304L SA213 Type 304 SA312 Type 304
Reactor vessel internals - radial support clevis inserts	Alloy 690 (SB-166 N06690)
Reactor vessel internals - hold-down spring	ASME Code Case N-4-11 (AISI Type 403 Modified)
Reactor vessel internals - most threaded fasteners and bolts	SA193 Class 2 B8M SA 479 Type 316 Strain hardened
Reactor vessel internals - radial support clevis insert bolts and spring	Alloy X-750 (ASME SB637 N07750)(2)
Reactor vessel internals - guide tube support pins	ASME SB637 N07750 SA193 Class 2 B8M
Incore instrumentation detector guide thimble	SA213 Type 316
Pressure retaining block of the incore instrumentation support column	SA479 Type 316
Guide funnel of the CRDM thermal sleeve	SA182 Type 304 SA351 CF8
Tie rods of neutron reflector	SA 479 Type 316 Strain hardened (with chrome plating on thread)
Alignment pins and clevis inserts	SA182 Grade F304 SA 479 type 304 SA 240 type 304 (with cladding cobalt alloy, Stellite No.6 or equivalent material)
Reactor vessel internals - weld material	SFA-5.9 ER308L SFA-5.9 ER316L SFA-5.4 E308L.

notes: (1) Additional information appears in the text of Section 4.5 and Subsection 5.2.3.
(2) Additional stringent specification, ~~MIL-N-24114 Class A No.1~~ MIL-S-239192 is applied for spring.


Core Barrel
Neutron Reflector
Tie Rod
Alignment Pin
Alignment Pin
Lower Core Support Plate