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

October 30, 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-09507

Subject: MHI's Response to US-APWR DCD RAI No. 457-3305

Reference: 1) "Request for Additional Information No. 457-3305 Revision 0, SRP Section: 04.05.01 - Control Rod Drive Structural 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. 457-3305, Revision 0" (Proprietary Version)

3. "Response to Request for Additional Information No. 457-3305, Revision 0" (Non-Proprietary Version)

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

<u>Contact Information</u>
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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-09507

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. 457-3305, Revision 0", dated October 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 30th day of October 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09507
Docket No. 52-021

Response to Request for Additional Information No. 457-3305,
Revision 0

October 2009
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

10/29/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 457-3305 REVISION 0**
SRP SECTION:	**04.05.01 - CONTROL ROD DRIVE STRUCTURAL MATERIALS**
APPLICATION SECTION:	**4.5.1**
DATE OF RAI ISSUE:	**09/14/2009**

QUESTION NO.: 04.05.01-8

1. In RAI 04.05.01-1 (3) [RAI 2181-4.5.1-1], the staff requested that the applicant list weld filler classifications for CRDM welds. The applicant provided a proposed revision to Table 4.5-1 by letter dated April 28, 2009. The applicant's proposed revision lists SFA 5.9 ER309/EC316L to weld the latch housing to the rod travel housing. However, the staff is unable to locate EC316L in SFA 5.9. Also, the sketch of the CRDM, provided by the applicant in its response to RAI 4.5.1-1, lists SFA 5.14 ERNiFe-7A weld filler material. The staff notes that ERNiFe-7A is not listed in SFA 5.14 and neither FSAR Section 4.5.1 nor Section 5.2.3 identifies the weld filler material used to join the CRDM pressure housing to the RPV head CRDM nozzle. The staff also notes that FSAR Subection 4.5.1.1 states that a detailed description of the austenitic stainless steel for the CRDM pressure housing is given in Subsection 5.2.3 but the staff is unable to locate a listing for CRDM materials in Subsection 5.2.3 or Table 5.2.3-1. The staff requests that the applicant address the above and modify the FSAR accordingly.

2. In RAI 04.05.01-1(5) [RAI 2181-4.5.1-1] the staff requested that the applicant identify the materials in Table 4.5-1 that are exposed to reactor coolant. The applicant provided a proposed revision of Table 4.5-1 that identifies which CRDM materials are exposed to reactor coolant. The staff notes that ER309L/EC316L is identified as not being exposed to reactor coolant. The staff requests that the applicant verify that the weld between the rod travel housing and the latch housing is not exposed to reactor coolant.

3. In RAI 04.05.01-1(6) [RAI 2181-4.5.1-1] the staff requested that the applicant modify Subsection 4.5.1 to address its compliance with GDC 26. In the applicant's response, it indicated that it will modify FSAR Section 4.5.1 to address GDC 26 but the applicant did not provide a proposed version of it's modification. The staff requests that the applicant provided it's proposed modification to FSAR Section 4.5.1 to address RAI 04.05.01-1 (6)

ANSWER:

1. Control rod drive system is categorized in the reactor system. Therefore, materials of the Control Rod Drive System are not included in subsection 5.2. However, requirements for austenite stainless steel described in subsection 5.2.3 can be applied to the housing material of the control rod drive mechanism.
   Applicable sentences in the subsection 5.2.3 are inserted in subsection 4.5.1.2.

2. Environment condition for the welding material used in CRDMs housing in Table 4.5-1 shall be revised to "Inside surface exposed to reactor coolant water."

3. GDC 26 shall be addressed in subsection 4.5.1.

Impact on DCD

DCD Revision 3 will incorporate the following changes:

- 1st paragraph of Subsection 4.5.1.2 shall be changed as follows:

~~Discussions of fabrication and processing of austenitic stainless steel are provided in Subsection 5.2.3. The processes for control of welding described in Subsection 5.2.3 are applicable to the pressure housing of the CRDM.~~ Austenitic stainless steel base materials for CRDM applications are the solution heat that is treated to prevent sensitization and stress corrosion cracking (SCC).
The welding materials used for joining the austenitic stainless steels meet the requirements of the welding material specification SFA 5.9 in ASME Code Section II. In addition, the above welding materials meet the requirements of ASME Code Section III.
For design temperatures up to and including 800 °F, the minimum acceptance delta ferrite is 5FN (Ferrite Number).

- Table 4.5-1 will be revised as shown in attachment-1 to this RAI.

- 2nd paragraph of subsection 4.5.1 will be changed as follows:

Information in this subsection addresses relevant requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A (Reference 4.5-1):

- GDC 1, as it relates to quality standards for structures, systems, and components important to safety

- GDC 14, as it relates to low probability of abnormal leakage, rapidly propagating failure, or gross rupture

- GDC26, as it relates to reactivity control system redundancy and capability

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

10/29/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 457-3305 REVISION 0**
SRP SECTION:	**04.05.01 - CONTROL ROD DRIVE STRUCTURAL MATERIALS**
APPLICATION SECTION:	**4.5.1**
DATE OF RAI ISSUE:	**09/14/2009**

QUESTION NO.: 04.05.01-9

In RAI 04.05.01-2 (RAI 2181-01-2), the staff requested that the applicant address inconsistencies between FSAR Subsections 4.5.1.1 and 5.2.3 regarding cold worked materials. In the applicant's response, dated April 28, 2009, the applicant stated that strain hardened and/or cold worked material is not specified for CRDM materials. The applicant also stated that it will delete references to cold worked materials in FSAR Subsection 4.5.1.1. In order to provide clarity in the FSAR, the staff requests that the applicant additionally modify FSAR Subsection 4.5.1.1 to state that strain hardened and/or cold worked austenitic stainless steels are not used in CRDM components.

ANSWER:

Following sentence shall be added in subsection 4.5.1.1.
Strain hardened and/or cold worked austenitic stainless steels are not used in CRDM components.

Impact on DCD

DCD Revision 3 will incorporate the following change:

- 2nd paragraph of subsection 4.5.1.1 will be revised as follows:

The properties of the materials selected for the CRDM are found in Section III, Appendix I, Division 1 of the ASME Boiler and Pressure Vessel Code (ASME Code) (Reference 4.5-3) or Section II, Parts D of the ASME Code. <u>Strain hardened and/or cold worked austenitic stainless steels are not used in CRDM components.</u>

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

10/29/2009

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 457-3305 REVISION 0**
SRP SECTION:	**04.05.01 - CONTROL ROD DRIVE STRUCTURAL MATERIALS**
APPLICATION SECTION:	**4.5.1**
DATE OF RAI ISSUE:	**09/14/2009**

QUESTION NO.: 04.05.01-10

In RAI 04.05.01-3 (RAI 2181-01-03) the staff requested that the applicant specify a maximum carbon content limit of 0.03% for austenitic stainless steels. In addition, the staff requested that the applicant list FSAR Subsection 4.5.1 in Table 1.9.1-1 under the line item for RG 1.44. The applicant responded by letter dated April 28, 2009 and stated that because PWR primary water is controlled under 0.1 ppm, stress corrosion cracking is not a concern. The applicant did not address the staff's request to modify Table 1.9.1-1. In order to complete its review of FSAR Subsection 4.5.1, the staff requests the following:

1. The staff requests that the applicant list FSAR Section 4.5.1 in Table 1.9.1-1 under the line item for RG 1.44.

2. The staff requests that the applicant verify, and state in the FSAR, that a stagnant primary coolant environment will not exist in any portion of CRDMs that could result in elevated dissolved oxygen above 0.05ppm. If elevated dissolved oxygen is possible, state that the dissolved oxygen content will be less than 0.10%. The staff notes that elevated dissolved oxygen content in a stagnant primary coolant environment has contributed to stress corrosion cracking in operating PWRs and the applicant should address, in the FSAR, how the US-APWR CRDM design addresses potential elevated dissolved oxygen.

3 FSAR Subsection 4.5.1.2 states that furnace sensitized material is allowed. The staff requests that the applicant identify components that will be furnace sensitized and discuss the maximum carbon content allowed for those components. In addition, discuss the operating experience with furnace sensitized material in operating plants and verify that the carbon content, of the material used to produce the figure provided in response to RAI 04.05.01-3, is greater than that which will be used for all austenitic stainless steel CRDM components. The FSAR should be modified to include a discussion that specifically addresses furnace sensitized components and the potential for the susceptibility of these components to stress corrosion cracking.

ANSWER:

1. Subsection 4.5.1 shall be added in Table 1.9.1-1 under the line item for RG 1.44.
2. Inside surface of the welding portions of the CRDM housing comes in contact with primary water during operation. Dissolved oxygen level of primary coolant water is controlled less than

0.1%. This low dissolved oxygen water flows into the CRDM housing by thermal siphoning flow. Thus, environment condition of inside surface near welding portion of CRDM housing is low dissolved oxygen condition.

3. Maximum carbon contents of CRDM housing material is [0.06] %. MHI will use verified welding process to avoid sensitization at heat affect zone. Thus, MHI does not use sensitized material for the CRDM housing.
   The test data by M. O. Speidel shows that SCC potential of type 304 austenite stainless steel is very low in the PWR primary water condition, and furthermore, MHI controls welding to avoid sensitizing. Thus, the risk of SCC using type 304 stainless steel, which carbon contents is over 0.03%, is very low.

On the other hand, MHI decides to use low carbon material for the CRDM housing. Table 4.5-1 shall be revised.

Impact on DCD

DCD Revision 3 will incorporate the following changes:

Add subsection 4.5.1 in column of RG 1.44 of Table 1.9.1-1 as attachment-2.

Table 4.5-1 will be revised as shown in attachment-1 to this RAI.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the 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 LN	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	SA-479 Type 304 SA-213 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	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
Welding material used in CRDMs housing	SFA-5.9 ER316L EC316L	Inside surface 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-S-23192, is applied.
(3) Equivalent material is a substitute material having the same chemistry and properties as the preferred material.

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 4 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section/ Subsection
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments (Rev. 0, March 1973)	Conformance with no exceptions identified.	8.1.5.3, 14.2.7
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.2.3.3.2, 5.3.1.4
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	3.6.3.3.4, 4.5.1, 5.2.3.4.1, 5.2.3.4.2, 6.1.1
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 1, May 2008)	Conformance with no exceptions Identified.	5.2.5
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	Conformance with no exceptions identified.	8.1.5.3, table 8.1-1, 18.7.3.2, table 18.7-1
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	This RG has been withdrawn by NRC.	N/A
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	Conformance with no exceptions identified.	5.3.1.2, 5.3.1.4, 5.2.3.3.2, 6.1.1
1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	Conformance with no exceptions identified.	6.4.2, 6.4.6, Table 6.4-2, 6.5.1, Table 6.5-3, 9.4.1, 9.4.5, 12.3.3, 14.2.7
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 2, November 2003)	Conformance with no exceptions identified.	7.1.3. 2, 7.1.3.3, 8.1.5.3
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	Conformance with exceptions. Programmatic/operational and site-specific aspects are not applicable to US-APWR design certification. ASTM standard revision levels may differ from RG 1.54 as specifically referenced in the "Corresponding Chapter/Section/Subsection"	6.1.2 11.2 11.4
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	Not applicable. US-APWR has a concrete containment.	N/A
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	Conformance with exceptions. RG applies to a site-specific characterization for flooding.	2.4, 3.4.1.2
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	Conformance with no exceptions identified. Note: COL Applicant will verify site-specific data is bounded by data used in DCD analyses.	2.3, 2.5, 3.7
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, March 2007)	Conformance with no exceptions identified.	3.7, 3.9.2, 3.12.3, 3.12.5.4, 3.12.6.8