MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

June 18, 2012

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

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-12153

Subject: Amended Responses to US-APWR DCD RAI No.893-6232 REVISION 3 (SRP 04.02) and US-APWR DCD RAI No.129-1673 REVISION 1 (SRP 04.02)

Reference:
1) **"REQUEST FOR ADDITIONAL INFORMATION 129-1673 REVISION 1" dated on December 17, 2008**
2) **"REQUEST FOR ADDITIONAL INFORMATION 893-6232 REVISION 3, SRP Section:04.02, Application Section: Chapter 4.2" dated on January 24, 2012**
3) **"Response to US-APWR DCD RAI No.893-6232 REVISION 3 (SRP 04.02) and Revised Response to US-APWR DCD RAI No.129-1673 REVISION 1 (SRP 04.02)" UAP-HF-12046 dated on February 23, 2012**

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") documents entitled "Amended Response to US-APWR DCD RAI No.893-6232 REVISION 3" and "Amended Response to US-APWR DCD RAI No.129-1673 REVISION 1".

In the enclosed documents, MHI provides amended responses to Questions 4.2-18, 4.2-19, and 4.2-51 (contained in References 1 and 2) and previously transmitted in Reference 3. Through discussions with the NRC staff, MHI agreed to clarify descriptions of the inservice surveillances that were added to Section 4.2.4.5 of US-APWR DCD in response to those RAI questions.

As indicated in the enclosed materials, this document 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. The proprietary information is bracketed by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3 and Enclosure 4) 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 Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of this submittal. His contact information is provided below.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Amended Response to US-APWR DCD RAI No.129-1673 REVISION 1 (Proprietary)

3. Amended Response to US-APWR DCD RAI No.129-1673 REVISION 1 (Non-Proprietary)

4. Amended Response to US-APWR DCD RAI No.893-6232 REVISION 3

CC: J. A. Ciocco
J. Tapia

Contact Information
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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-12153

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and state as follows:

1. I am Director, 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 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 "Amended response to US-APWR DCD RAI No.129-1673 REVISION 1" and have determined that portions of the report contain proprietary information that should be withheld from public disclosure. Those pages containing 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 technical report indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a).

3. The information in the report identified as proprietary by MHI 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 code and files developed by MHI for the fuel of the US-APWR. These were developed at significant cost to MHI, since they required the performance of detailed calculations, 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.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of supporting the NRC staff's review of MHI's Application for certification of its US-APWR Standard Plant Design.

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 fuel systems and components. Disclosure of the information identified as proprietary would therefore have negative impacts on the competitive position of MHI 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 18th day of June, 2012.

Y. Ogata

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

Enclosure 3

UAP-HF-12153
Docket No. 52-021

Amended response to US-APWR DCD RAI No.129-1673
REVISION 1

June 2012
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/18/2012

US-APWR Design Certification
Mitsubishi Heavy Industries
Docket No. 52-021

RAI NO.:	**NO.129-1673 REVISION 1**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/17/2008**

QUESTION NO. : 04.02-18

In Section 4.2.4.5 it is stated that during normal refueling outages some assemblies will be dimensionally checked. What type of dimensional check will be performed and what criteria will be used to determine re-insertion acceptability?

ANSWER (REVISION 3):

In order to clarify inspections to be performed, MHI will revise the current description of Inservice Surveillance in DCD 4.2.4.5 as shown in Attachment A

Impact on DCD

DCD Chapter 4.2.4.5 will be changed as described above response. (See Attachment-A, Markup from Revision 3.)

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on Technical/Topical Report.

This completes MHI's response to the NRC's question.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

6/18/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.129-1673 REVISION 1**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/17/2008**

QUESTION NO. : 04.02-19

In Section 4.2.4.5 it is stated that some fuel assemblies loaded in the initial core will be closely examined to confirm their performance. What type of measurements/tests will be performed and what acceptance criteria will be used to judge acceptable performance?

ANSWER (REVISION 3):

In order to clarify inspections to be performed, and also to reflect the discussion associated with FINE topical report (MUAP-07008) and DCD RAI 05.02-03-18, MHI will revise the current description of Inservice Surveillance in DCD 4.2.4.5 as shown in Attachment-A

In addition, the minimum scope of the fuel surveillance plan is described in Table 04.02-19-1 to show types of measurements and inspections, the number of assemblies to be inspected, and acceptance criteria. Industry standard methods will be selected for these inspections.

Table 04.02-19-1 Surveillance Program (Minimum Scope)
Measurements and Inspections for Fuel Assemblies loaded into the initial Core
(Close examination)

Measurement and Inspection	Number of Assemblies	Acceptance Criteria

REFERENCES

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P(R2) (Proprietary) and MUAP-07008-NP(R2) (Non-proprietary), July 2010.

(2) "US-APWR FUEL SYSTEM AND DESIGN EVALUATION", MUAP-07016-P(R3) (Proprietary) and MUAP-07016-NP(R3) (Non-proprietary), August 2010.

(3) “MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(R0) “Mitsubishi Fuel Design Criteria and Methodology”, UAP-HF-08299-P (R0),December 19, 2008.

Impact on DCD

DCD Chapter 4.2.4.5 will be changed as described above response. (See Attachment-A, Markup from Revision 3.)

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on Technical/Topical report.

This completes MHI's response to the NRC's question.

The new fuel assemblies and the in-core control components are then moved to the new fuel storage area inside the plant. The in-core control components are stored in the fuel assemblies in the storage area.

The control rod assembly is functionally tested at the plant site after core loading. Each control rod assembly is dropped at the full flow/hot condition to confirm that the drop time is within the specified limit. Since the control rod is a movable component which must move freely to control reactivity, the control rod capability for partial movement is also inspected. The rod drop test is periodically performed at each refueling outage to confirm rod capability to meet its functional requirements.

4.2.4.4 Coolant Radiation Monitoring

Radioactivity in the reactor coolant is monitored by periodic sampling of the coolant. Analysis is performed for iodine, noble gases and cesium. If any anomaly is found sampling is done more frequently. The US-APWR technical specification limits the radiation level for continued plant operation, but the plant will be shutdown at much lower radiation level, set in each plant operation control document. Detailed radiological monitoring and sampling systems are described in Sections 9.3 and 11.5.

4.2.4.5 Inservice Surveillance

Several monitoring systems are used during plant operation to obtain information related to core reactivity, radiation levels, and water chemistry. If the radiation level increases, it is monitored to determine the degree of fuel degradation and whether a plant shutdown is required. ~~At normal refueling outages the irradiated fuel assemblies are visually inspected and some of them are dimensionally checked to confirm their integrity and verify consistency with the assembly condition assumed for their subsequent irradiation. Some of the US-APWR fuel assemblies loaded in the initial core will be closely examined to confirm their performance.~~ DCD_04.02-18 DCD_04.02-19 DCD_04.02-51

If a coolant radiation level change suggests leakage in the loaded fuel, at the beginning of the fuel inspection the fuel assembly containing the defective rod(s) will be identified by a sipping method. After the leaking fuel assembly is identified, techniques such as ultrasonic testing will be used to identify the leaking rod(s). Additional efforts to identify the cause of the fuel failure and determine countermeasures to eliminate the failure mechanism will continue inside and outside the plant.

The fuel assembly surveillance program for the US-APWR described in Section 4.2.1.7 will be based on the substantial data available from testing and fuel surveillance programs on Mitsubishi conventional fuel assemblies. The surveillance program will include inspection of post-irradiated assemblies. DCD_04.02-18 DCD_04.02-19 DCD_04.02-51

For additional verification of the fuel performance and additional validation of the design bases, close inspection of selected post irradiated fuel assemblies loaded in the initial core of the first operating US-APWR will be performed to determine assembly dimensions (assembly growth, assembly bow, total gap, and rod bow, cladding oxide thickness and crud buildup). DCD_04.02-18 S01 DCD-04.02-19 S01 DCD_04.02-51 S01

The number of cycles that the fuel assemblies will experience may vary with the operating cycle length chosen by the COL holders. To provide a consistent reference point for the inspection description in the following subsections, the very last cycle that the selected fuel assemblies loaded in the initial core will experience will be referred to as cycle N. The cycle preceding cycle N will be referred to as N-1. For further clarification, N is defined as 2 for consecutive 24 month fuel cycles, and 3 for consecutive 18 month fuel cycles. DCD_04.02-51 DCD_04.02-18 S01 DCD_04.02-19 S01

4.2.4.5.1 Fuel Inspections during Normal Refueling Outages

DCD_04.02-51 S01

During normal refueling outages, irradiated fuel assemblies will be visually inspected. The results of the visual inspection will be evaluated against criteria for performing additional inspection requirements if unusual characteristics are identified. Irradiated fuel assemblies receiving unsatisfactory visual inspection results will be dimensionally checked for assembly growth, assembly bow, total gap, and rod bow, as appropriate, to confirm their integrity and re-insertion capability. Those fuel assemblies receiving unsatisfactory dimensional check results will not be re-inserted into the core unless a more detailed inspection and/or evaluation can be performed.

4.2.4.5.2 Fuel Assembly Dimension and Crud Buildup Measurement for Additional Design-Basis Verification

During the outage following the cycle N, measurement of fuel assembly dimension and crud buildup will be performed for selected post-irradiated fuel assemblies that were loaded in the initial core. The dimensional check is performed to measure assembly growth, assembly bow, total gap, and rod bow. An evaluation report documenting the results of these inspections will be available for NRC review within 90 days following the off-load of the last fuel assemblies to be inspected.

4.2.4.5.3 Cladding Oxide Thickness Inspections for Additional Design-Basis Verification

During the outage between cycle N-1 and cycle N, intermediate inspections of cladding oxide thickness will be performed. The intermediate inspections will only be performed on peripheral fuel rods of the selected assembly to preclude the need to remove fuel rods from the assembly. The results are to be documented in an intermediate evaluation report.

Additionally, to provide assurance that the cladding oxide thickness limit will not be exceeded during cycle N, a separate acceptance criterion will be developed for the cladding oxide thickness measurement that will take place during the outage between cycle N-1 and N. Compliance with this acceptance criterion will be verified prior to loading those assemblies for their cycle N operations, and will ensure that the predicted oxide at the end of cycle N meets the oxide thickness limit stated in Reference 4.2-7. The oxide thickness predictions will be documented in the intermediate evaluation report described above along with the oxide thickness measurements. The intermediate evaluation report will be available for NRC review before the reactor start-up for the cycle N.

Furthermore, cladding oxide thickness is inspected, following cycle N, for the fuel assemblies that are inspected during the outage between cycle N-1 and cycle N and loaded in the core for cycle N operation. An evaluation report documenting the results of the inspection will be available for NRC review within 90 days following the off-load of the last fuel assemblies to be inspected. DCD_04.02-51

*[To ensure that the cladding oxide thickness inspections and documentation described in this subsection will be completed by a COL holder, the COL holder for the first plant which reaches the outage between cycle N-1 and N is to perform the inspections and documentations. The COL holders for subsequent plants are to confirm that the inspections have been completed by another COL holder and the inspections are not required to be repeated, or to perform the inspections and documentation if required.]** DCD_04.02-18 S01 DCD_04.02-19 S01 DCD_04.02-51 S01

Information in this subsection that is italicized and enclosed in square brackets with an asterisk following the closing bracket is a special category of information designated by the NRC as Tier 2*. Material change to this information requires prior NRC approval.

4.2.5 Combined License Information

No additional information is required to be provided by a COL Applicant in connection with this section.

4.2.6 References

4.2-1 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 4.2, March 2007.

4.2-2 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50, Appendix A.

4.2-3 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR 50.46.

4.2-4 Reactor Site Criteria, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 100.

4.2-5 Combined License Applications for Nuclear Power Plants (LWR Edition), NRC Regulatory Guide 1.206. Section C.I.4.2.

4.2-6 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P Rev.2 (Proprietary) and MUAP-07008-NP Rev.2 (Non-Proprietary), July 2010.

4.2-7 US-APWR Fuel System Design Evaluation, MUAP-07016-P Rev.3 (Proprietary) and MUAP-07016-NP Rev.3 (Non-Proprietary), August 2010.

4.2-8 American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III.

4.2-9 W. J. O'Donnel and B. F. Langer, Fatigue Design Basis for Zircaloy Components, Nuclear Science and Engineering 20, pp.1-12, 1964.

Enclosure 4

UAP-HF-12153
Docket No. 52-021

Amended response to US-APWR DCD RAI No.893-6232
REVISION 3

June 2012

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION 893-6232 (R3)

6/18/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**893-6232 Revision 3**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**1/24/2011**

QUESTION NO. : 04.02-51

In MHI's response to RAI 129-1673, Questions 4.2-18 and 4.2-19, documented in UAPHF-11427 (December 2011), MHI agreed to a fuel inspection campaign and a corresponding DCD wording revision. The staff is requesting that the DCD wording revision given in the response Question 4.2-18 be made Tier 2* in Chap 4.2.

ANSWER(Revision 1):

MHI will modify Subsection 4.2.4.5 as shown in Attachment-A. Subsection 4.2.4.5 is divided into Subsections 4.2.4.5.1, 4.2.4.5.2, and 4.2.4.5.3. Subsections 4.2.4.5.1 and 4.2.4.5.2 describe normal refueling outage inspections and additional design-basis verification, respectively.

The inspections of cladding oxide thickness was discussed in detail in response to Question 4.2-19, and are described in Subsection 4.2.4.5.3. MHI understands the importance of the cladding oxide thickness inspections. Therefore, MHI added a commitment to perform the cladding oxide thickness inspections and designated the commitment as Tier 2* information. Also, a description of Tier 2* information has been added at the end of Subsection 4.2.4.5.3 for clarity.

For consistency, MHI will also revise the responses to RAI129-1673 Question 4.2-18 and 4.2-19 to refer to the RAI 893-6232 response for the DCD wording revision.

Impact on DCD

DCD Chapter 4.2.4.5 will be changed as described above response. (See Attachment-B, Markup from Revision 3)

Impact on R-COLA

There is no impact on the R-COLA.

Impact on S-COLA

There is no impact on the S-COLA.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical Report

There is no impact on a Technical/Topical Report.

This completes MHI's response to the NRC's question.

The new fuel assemblies and the in-core control components are then moved to the new fuel storage area inside the plant. The in-core control components are stored in the fuel assemblies in the storage area.

The control rod assembly is functionally tested at the plant site after core loading. Each control rod assembly is dropped at the full flow/hot condition to confirm that the drop time is within the specified limit. Since the control rod is a movable component which must move freely to control reactivity, the control rod capability for partial movement is also inspected. The rod drop test is periodically performed at each refueling outage to confirm rod capability to meet its functional requirements.

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Several monitoring systems are used during plant operation to obtain information related to core reactivity, radiation levels, and water chemistry. If the radiation level increases, it is monitored to determine the degree of fuel degradation and whether a plant shutdown is required. ~~At normal refueling outages the irradiated fuel assemblies are visually inspected and some of them are dimensionally checked to confirm their integrity and verify consistency with the assembly condition assumed for their subsequent irradiation. Some of the US-APWR fuel assemblies loaded in the initial core will be closely examined to confirm their performance.~~ DCD_04.02-18 DCD_04.02-19 DCD_04.02-51

If a coolant radiation level change suggests leakage in the loaded fuel, at the beginning of the fuel inspection the fuel assembly containing the defective rod(s) will be identified by a sipping method. After the leaking fuel assembly is identified, techniques such as ultrasonic testing will be used to identify the leaking rod(s). Additional efforts to identify the cause of the fuel failure and determine countermeasures to eliminate the failure mechanism will continue inside and outside the plant.

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DCD_04.02-51 S01

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Furthermore, cladding oxide thickness is inspected, following cycle N, for the fuel assemblies that are inspected during the outage between cycle N-1 and cycle N and loaded in the core for cycle N operation. An evaluation report documenting the results of the inspection will be available for NRC review within 90 days following the off-load of the last fuel assemblies to be inspected. DCD_04.02-51

*[To ensure that the cladding oxide thickness inspections and documentation described in this subsection will be completed by a COL holder, the COL holder for the first plant which reaches the outage between cycle N-1 and N is to perform the inspections and documentations. The COL holders for subsequent plants are to confirm that the inspections have been completed by another COL holder and the inspections are not required to be repeated, or to perform the inspections and documentation if required.]** DCD_04.02-18 S01 DCD_04.02-19 S01 DCD_04.02-51 S01

Information in this subsection that is italicized and enclosed in square brackets with an asterisk following the closing bracket is a special category of information designated by the NRC as Tier 2*. Material change to this information requires prior NRC approval.

4.2.5 Combined License Information

No additional information is required to be provided by a COL Applicant in connection with this section.

4.2.6 References

4.2-1 U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 4.2, March 2007.

4.2-2 General Design Criteria for Nuclear Power Plants, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 50, Appendix A.

4.2-3 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR 50.46.

4.2-4 Reactor Site Criteria, NRC Regulations Title 10, Code of Federal Regulations, 10 CFR Part 100.

4.2-5 Combined License Applications for Nuclear Power Plants (LWR Edition), NRC Regulatory Guide 1.206. Section C.I.4.2.

4.2-6 Mitsubishi Fuel Design Criteria and Methodology, MUAP-07008-P Rev.2 (Proprietary) and MUAP-07008-NP Rev.2 (Non-Proprietary), July 2010.

4.2-7 US-APWR Fuel System Design Evaluation, MUAP-07016-P Rev.3 (Proprietary) and MUAP-07016-NP Rev.3 (Non-Proprietary), August 2010.

4.2-8 American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III.

4.2-9 W. J. O'Donnel and B. F. Langer, Fatigue Design Basis for Zircaloy Components, Nuclear Science and Engineering 20, pp.1-12, 1964.