MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

December 14, 2011

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-11427

Subject: **MHI's Revised Responses to US-APWR DCD RAI No.129-1673 Revision 1 (SRP 04.02)**

Reference: 1) "MHI's Responses to US-APWR DCD RAI No.129-1673 Revision 1" (MHI Ref: UAP-HF-09024) dated on January 30, 2009

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Revised Responses to Request for Additional Information No.129-1673 Revision 1".

In the enclosed document, MHI provides revised responses to RAI No.10, 18, 19 & 20 within Reference 1, in accordance with the NRC staff discussion.

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 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 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. Revised Responses to Request for Additional Information No.129-1673 Revision 1 (Proprietary)

3. Revised Responses to Request for Additional Information No.129-1673 Revision 1 (Non-Proprietary)

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

Contact Information
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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-11427

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, depose and 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 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 "Revised Responses to Request for Additional Information 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 14th day of December, 2011.

Y. Ogata

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

Enclosure 3

UAP-HF-11427
Docket Number 52-021

Revised Responses to Request for Additional Information
No.129-1673 Revision 1

December, 2011
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/14/2011

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. : 04.02-10

In Section 5.1.6.2 of technical report MUAP-07016 it is stated that there is no significant control rod cladding wear over its lifetime. Is there measured data to support this statement? Do the control rods sit in the assembly such that the control tip could wear against the softer Zicraloy-4 guide tube? If so, has guide tube wear due to control vibration been observed or evaluated?

ANSWER (REVISION 1):

Control rod cladding and guide thimble wear due to control rod vibration in Mitsubishi fuel assemblies has been evaluated as described below.

Eddy current testing for RCC

- Measured RCCs: 151 RCCs loaded in Japanese 4-loop plants
- Measured data: 274 RCCs
- Maximum time: 6.9 calendar years
- Maximum wear depth: 160 microns (34 % of control rod cladding wall thickness)
- Estimated maximum wear depth at the end of lifetime of 15 years assuming that wear volume is proportional to time: 230 microns (49 % of control rod cladding wall thickness)

Observations for guide thimbles

[ ]

Assessment of flow induced vibration impact on RCC

Upper head and plenum geometry may induce control rod drive vibration which could vibrate the control rod tips wearing down the softer guide thimble material. The amount of vibration is related to the flow geometry of the lower RCCA guide tube (GT), the upper core plate (UCP), at which point vertical coolant flow will become turbulent and may induce cross flow.

The flow geometries of GT and UCP are the same between the US-APWR and the existing Japanese 4-loop plant designs. In the US-APWR, specifics of GT and UCP are designed to provide the same flow path as that of the Japanese plants. In addition, the axial clearance between GT and UCP is the same.

Increased coolant flow rate may also cause flow-induced RCC vibration. As stated in DCD (MUAP-DC004) Rev.3, Page 4.4-30, the US-APWR has a core mass velocity of 2.25×10^6 lbm/hr-ft^2 as compared to the existing Japanese 4-loop plants flow rate of 2.41×10^6 lbm/hr-ft^2. From this qualitative point of view, lower flow rate of the US-APWR provides lower potential for vibration. Based on these design features, performance of control rod and RCC guide thimble against wear in the US-APWR will be equivalent to or improved from that of existing Japanese 4-loop plants.

Based on the evaluations described above, the design differences between the US-APWR and the existing Japanese 4-loop plants will not result in an increased likelihood of control rod and guide thimble wear from flow-induced vibration.

Impact on DCD

There is no impact on the DCD.

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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/14/2011

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 1):

In order to clarify inspections to be performed, MHI will revise the current description of Inservice Surveillance in DCD 4.2.4.5 to incorporate the following sentences:

"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.

During normal refueling outages, the irradiated fuel assemblies will be visually inspected and the results of the inspections will be evaluated against criteria for performing additional inspection requirements if unusual characteristics are identified in the visual inspection. 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.

Besides the above described inspections at normal refueling outages, for verification of the fuel performance and additional validation of the design bases, close inspection of selected post irradiated fuel assemblies 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.

The number of cycles the fuel assemblies will experience may vary with the operating cycle length chosen by the operator. To provide a consistent reference point for the inspection description, the very last cycle that the fuel assemblies in the initial core loading group will experience will be referred to as cycle N. The cycle preceding the inspection activities will be referred to as N-1. Intermediate inspection will take place during the outage between cycle N-1 and cycle N. The intermediate inspections of cladding oxide thickness will only be performed on peripheral rods to preclude the need to remove rods from the assembly.

Post-irradiated fuel assembly close inspections for assembly dimension (assembly growth, assembly bow, total gap, and rod bow), cladding oxide thickness and crud buildup for the selected fuel assemblies from the initial core will be performed at the end of operating cycle N. A comprehensive 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.

Furthermore, to provide assurance that the oxide thickness limit will not be exceeded during cycle N operation, a separate acceptance criterion will be developed for the N-1 oxide measurements. Compliance with this N-1 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 acceptance criterion. An Intermediate evaluation report with the oxide thickness measurements and the predictions will be available for NRC review before the reactor start-up for the cycle N.

The close inspections and documentation are only applicable to the first US-APWR operator that reaches the refueling outage when the assemblies selected for the inspections have reached the end of the cycle N-1 and/or N."

Impact on DCD

DCD Chapter 4.2.4.5 will be changed as described above response. (See Attachment-1)

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

MUAP-07016(R3) will be changed to include the oxide thickness limit.

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/14/2011

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 1):

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 to incorporate the following sentences:

"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.

During normal refueling outages, the irradiated fuel assemblies will be visually inspected and the results of the inspections will be evaluated against criteria for performing additional inspection requirements if unusual characteristics are identified in the visual inspection. 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.

Besides the above described inspections at normal refueling outages, for verification of the fuel performance and additional validation of the design bases, close inspection of selected post irradiated fuel assemblies 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.

The number of cycles the fuel assemblies will experience may vary with the operating cycle length chosen by the operator. To provide a consistent reference point for the inspection description, the very last cycle that the fuel assemblies in the initial core loading group will experience will be referred to as cycle N. The cycle preceding the inspection activities will be referred to as N-1. Intermediate inspection will take place during the outage between cycle N-1

and cycle N. The intermediate inspections of cladding oxide thickness will only be performed on peripheral rods to preclude the need to remove rods from the assembly.

Post-irradiated fuel assembly close inspections for assembly dimension (assembly growth, assembly bow, total gap, and rod bow), cladding oxide thickness and crud buildup for the selected fuel assemblies from the initial core will be performed at the end of operating cycle N. A comprehensive 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.

Furthermore, to provide assurance that the oxide thickness limit will not be exceeded during cycle N operation, a separate acceptance criterion will be developed for the N-1 oxide measurements. Compliance with this N-1 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 acceptance criterion. An Intermediate evaluation report with the oxide thickness measurements and the predictions will be available for NRC review before the reactor start-up for the cycle N.

The close inspections and documentation are only applicable to the first US-APWR operator that reaches the refueling outage when the assemblies selected for the inspections have reached the end of the cycle N-1 and/or N."

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-1)

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

MUAP-07016(R3) will be changed to include the oxide thickness limit.

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/14/2011

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. : 04.02-20

It is unclear how the rod power histories in Section 3.3.2 of MUAP-07016 are used to determine rod internal pressure. A sentence in the third paragraph states that actual core analysis determines rod histories. Within that same paragraph another statement is made that if the clad liftoff pressure limit is exceeded than a rod specific power history is used to show acceptability. Clarify whether a bounding or cycle specific power history is used to determine peak rod internal pressure.

ANSWER (REVISION 1):

Cycle specific fuel rod power histories are used to determine the peak rod internal pressure. The following description of the power histories used for the US-APWR fuel rod design evaluation is given in Section 3.1.2 of Reference (1):

> Some characteristic power histories are known to be most limiting with respect to margin to the fuel rod design limits. These limiting histories may be the highest or lowest power rods in a cycle or the highest burnup fuel rods in a cycle, depending on the fuel rod design criterion to be assessed. In general, a single fuel rod power history is not limiting for all fuel criteria, so a set of limiting characteristic power histories are typically assessed in the fuel rod design. These power histories bracket the range of fuel rod power histories for the fuel region, and provide the basis for assessing fuel rod performance relative to the established specified acceptable fuel design limits (SAFDLs).
>
> Based on the Mitsubishi fuel rod design evaluation experience, the following power history types for each fuel type are assessed as part of the process for defining the limiting rods for each criterion.

[ ]

[ ]

The time-dependant power histories based on the reactor core analysis are the interface information from the core design. The US-APWR fuel rod design evaluations given in this report use the core conditions, such as a 24 month cycle length and the equilibrium core described in Appendix A, to define the typical operating conditions for the US-APWR.

Appendix A of Reference (1) describes the US-APWR 24-month equilibrium core nuclear design. The above description of the power histories is given in the RAI 1a and 9b responses of Reference (2). FINE input and output for the rod pressure evaluations are provided on the CDROM for the response to RAI 9d of Reference (2).

REFERENCES

(1) "US-APWR Fuel System Design Evaluation", MUAP-07016-P(R3) (Proprietary) and MUAP-07016-NP(R3) (Non-Proprietary), August 2010.

(2) "MHI's Partial Responses to the NRC's Requests for Additional Information on Topical Report MUAP-07008-P(0) "Mitsubishi Fuel Design Criteria and Methodology", UAP-HF-08299-P (R0), December 19, 2008.

Impact on DCD

There is no impact on the DCD.

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.

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

DCD_04.02-18
DCD_04.02-19

~~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.~~

~~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.

During normal refueling outages, the irradiated fuel assemblies will be visually inspected and the results of the inspections will be evaluated against criteria for performing additional inspection requirements if unusual characteristics are identified in the visual inspection. 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.

Besides the above described inspections at normal refueling outages, for verification of the fuel performance and additional validation of the design bases, close inspection of selected post irradiated fuel assemblies 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 DCD_04.02-19

The number of cycles the fuel assemblies will experience may vary with the operating cycle length chosen by the operator. To provide a consistent reference point for the inspection description, the very last cycle that the fuel assemblies in the initial core loading group will experience will be referred to as cycle N. The cycle preceding the inspection activities will be referred to as N-1. Intermediate inspection will take place during the outage between cycle N-1 and cycle N. The intermediate inspections of cladding oxide thickness will only be performed on peripheral rods to preclude the need to remove rods from the assembly.

Post-irradiated fuel assembly close inspections for assembly dimension (assembly growth, assembly bow, total gap, and rod bow), cladding oxide thickness and crud buildup for the selected fuel assemblies from the initial core will be performed at the end of operating cycle N. A comprehensive 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.

Furthermore, to provide assurance that the oxide thickness limit will not be exceeded during cycle N operation, a separate acceptance criterion will be developed for the N-1 oxide measurements. Compliance with this N-1 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 acceptance criterion. An Intermediate evaluation report with the oxide thickness measurements and the predictions will be available for NRC review before the reactor start-up for the cycle N.

The close inspections and documentation are only applicable to the first US-APWR operator that reaches the refueling outage when the assemblies selected for the inspections have reached the end of the cycle N-1 and/or N.

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.