MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

January 11, 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-12006

Subject: Response to US-APWR DCD RAI No.877-5918 REVISION 3 (SRP 04.02)

Reference: 1) "REQUEST FOR ADDITIONAL INFORMATION 877-5918 REVISION 3, SRP Section: 04.02, Application Section: Chapter 4.2" dated on December 12, 2011

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to US-APWR DCD RAI No.877-5918 REVISION 3".

In the enclosed document, MHI provides responses to RAIs in Reference 1.

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

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Enclosures:

1. Affidavit of Yoshiki Ogata

2. Response to US-APWR DCD RAI No.877-5918 REVISION 3 (Proprietary)

3. Response to US-APWR DCD RAI No.877-5918 REVISION 3 (Non-Proprietary)

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

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ENCLOSURE 1

Docket No.52-021
MHI Ref: UAP-HF-12006

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 "Response to US-APWR DCD RAI No.877-5918 REVISION 3" 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 11th day of January, 2012.

Y. Ogata

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

Enclosure 3

UAP-HF-12006
Docket No.52-021

Response to US-APWR DCD RAI No.877-5918 REVISION 3

January 2012
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION NO.5918 (R3)

1/11/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.5918 Revision 3**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/12/2011**

QUESTION NO. : 04.02-46

Question contains proprietary information.

The following analyses presented in MUAP-07016 or in response to the RAIs (UAP-HF-09024 MHI's Responses to US-APWR DCD RAI No.129-1673 Revision 1, dated January, 2009) do not reflect the modifications made to MHI fuel performance analysesas agreed to in the review of MUAP-07008.

a. Section 3.5 does not appear to include the cladding oxide [ ] that was agreed to in MUAP-07008 Safety Evaluation.

ANSWER:

As stated in the response to RAI No.48 for MUAP-07008 (Ref: UAP-HF-09538, dated November 2009), MHI recognizes that [

]

In order to implement this commitment, response to RAI No.19 for DCD Chapter 4.2 with regard to "In-service surveillance" of DCD 4.2.5.2 has been revised and submitted to NRC (Ref: UAP-HF-11427, dated December 2011). In addition to implementing the oxide thickness surveillance program, a limit is imposed that the FINE corrosion model maximum (peak) oxide thickness shall be less than or equal to [ ] based on current corrosion database.

Technical Report MUAP-07016 will be revised as attached to reflect the limitation of oxide thickness (See Attachment-1).

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

Technical report MUAP-07016 is revised as attached (See Attachment-1).

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION NO.5918 (R3)

1/11/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.5918 Revision 3**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/12/2011**

QUESTION NO. : 04.02-47

Load limits are defined Table 4.2-1 (c) of technical report MUAP-07016, for the top grid spacer joint and the intermediate grid space joint. How are these load limits determined, are they determined by testing? If so, provide a description.

ANSWER:

As shown in Figure 2.2-5 to of MUAP-07016, the top grid spacer, made of Inconel-718, mates with a reconstitutable top nozzle joint mechanism via 304L stainless steel sleeves which are brazed to the grid spacer. Therefore, the load limit at the joints between the grid spacer and the control rod guide thimble is determined by the minimum value among the following four components:

i) Reconstitutable top nozzle joint mechanism

ii) Braze joint between the grid spacer strap and the sleeve

iii) Cross-sectional area of the sleeve

iv) Bulge joint between the sleeve and the control rod guide thimble

Table 04.02-47-1 shows the load capability of the above components. [

]

The intermediate spacer grids and sleeves, made of Zircaloy-4, are connected together by welding. The load limit at the joint between the intermediate grid spacer and the control rod guide thimble is determined by the minimum value among the following three components:

i) Grid spacer strap to sleeve joint

ii) Cross-sectional area of the sleeve

iii) Bulge joint between the sleeve and the control rod guide thimble

Table 04.02-47-2 shows the load capability of the above components. [

]

Table 04.02-47-1 Top Grid Spacer Joint Component Load Capabilities

Reconstitutable top nozzle joint mechanism (*1)	Braze joint between the grid spacer strap and the sleeve (*2)	Cross-sectional area of the sleeve (*3)	Bulge joint between the sleeve and the control rod guide thimble (*4)
[ ] lbf ([ ] N)	[ ] lbf ([ ] N)	[ ] lbf ([ ] N)	[ ] lbf ([ ] N)

*1 : Load obtained by the tensile test of a mockup of the top nozzle mechanism

*2 : Load obtained by the tensile test of braze joint specimen

*3 : (Yield strength of SUS 304L Stainless Steel specified in ASME code) x (cross-section area of the sleeve)

[ ]

*4 : Load obtained by the tensile test for the bulge joint specimen

Table 04.02-47-2 Intermediate Grid Spacer Joint Component Load Capabilities

Welding joint between the grid spacer strap and the sleeve (*1)	Cross-sectional area of the sleeve (*2)	Bulge joint between the sleeve and the control rod guide thimble (*3)
[ ] lbf ([ ] N)	[ ] lbf ([ ] N)	[ ] lbf ([ ] N)

*1 : Minimum requirement specified by the manufacturing drawing

*2 : (Yield strength of Zircaloy-4 specified in ASTM B353) x (cross-section area of the sleeve)

[ ]

*3 : Load obtained by the tensile test for the bulge joint specimen

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.

1/11/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.5918 Revision 3**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/12/2011**

QUESTION NO. : 04.02-48

Provide the equation used to define the specific wear ratio, S, in Equation C.2.3-1, of technical report MUAP-07016, along with the basis for the coefficients used to define the S term including the dependency with oxide thickness.

ANSWER:

The specific wear ratio S is determined in the wear test with fuel cladding and a grid spacer. The test system is shown in Figure C.3.2-2 of MUAP-07016. In the test, the fuel cladding specimen is inserted into a unit grid spacer cell, which is cut from the full-sized grid spacer. The cladding specimen and the unit grid spacer cell are sinusoidally vibrated, at a constant frequency, at operating temperatures throughout the test period. The vibration causes wear of the cladding at the contact points, the springs and dimples in the unit grid spacer cell.

The wear ratio S is given by

$$S = \frac{V}{FL}$$

Where

S: Specific wear ratio

F: Spring force in 1 cell of the grid spacer

L: Accumulated sliding distance, $L = 4A \cdot f \cdot T$

A: Vibration amplitude (0 to peak)

f: Vibration frequency

T: Test duration

From experience with wear tests for the Japanese conventional fuel assemblies, the specific wear ratios are approximately as follows:

- Specific wear ratio between fuel cladding metal and the grid spacer metal: [ ]
- Specific wear ratio between fuel cladding oxide layer and the grid spacer metal: [ ]

There is no difference between the specific wear ratios measured for Inconel-718 and Zircaloy-4 grid spacers.

In addition to wear test measurements, information has been obtained from hydraulic tests for the fuel cladding wear, and results for the adjusted specific wear ratios are:

- Specific wear ratio between the fuel cladding metal and the grid spacer metal: [ ]
- Specific wear ratio between the fuel cladding oxide layer and the grid spacer metal: [ ]

The above specific wear ratios are used for the US-APWR evaluation. These specific wear ratios are the same as those used in the evaluation for Japanese conventional fuel assemblies.

With regard to the depth dependence of the wear ratio when there is an oxide thickness, an equivalent specific wear ratio is calculated and used for wear volume calculation when both growth of oxide layer and wear are occurring concurrently on the surface of the cladding. A schematic figure for the equivalent specific wear ratio calculation is shown in Figure 04.02-48-1

The adequacy of the cladding wear evaluation for the US-APWR fuel assembly is confirmed by the comparison with the long term flow test results described in MUAP-11017-P/NP (R0), which has been submitted to NRC in May, 2011.

Figure 04.02-48 - 1 Calculation of Equivalent Wear Specific Ratio

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 NO.5918 (R3)

1/11/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.5918 Revision 3**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/12/2011**

QUESTION NO. : 04.02-49

The following relates to the performance of the control rods as discussed in Section 5.1 of technical report MUAP-07016:

a. Provide the swelling values at end-of-life (EOL) for 304 stainless steel and Ag-In-Cd and include the fast fluence and temperatures used to determine these values.

b. Provide details of how the maximum control rod diameter will be tracked to EOL.

c. Summarize past experience with control rods similar to the APWR design. What is the maximum fluence level experienced by previous MHI control rods with Ag-In-Cd? How many control rods have experienced this fluence level? What are the design differences between past designs and the APWR of the Ag-In-Cd control rods?

ANSWER:

a. <u>Type 304 Stainless Steel</u>:

Accounting for the manufacturing tolerances and thermal expansion, the minimum clearance between the inner diameter of the dashpot region in the control rod guide thimble and the outer diameter of the RCC cladding is [ ] mil ([ ] mm). The RCC cladding outer diameter increase due to swelling is [ ] mil ([ ] mm) at a fast neutron fluence level of [ ] n/cm^2 (E>1MeV) after 15 years operation, assuming a core average linear heat rate of [ ] kW/ft during normal operation, and therefore the increase of the outer diameter of the RCC cladding is much less than the minimum gap clearance. Assuming that the fluence level after 15 years operation is [ ] n/cm^2 (E>1MeV), which considers a safety factor of [ ], the increase in the RCC cladding outer diameter due to swelling is [ ] mil ([ ] mm) and the gap clearance is still maintained. In MUAP-07016, a diameter increase due to swelling of [ ] mil ([ ] mm) has been conservatively used. A conservative cladding temperature of [ ] deg.F ([ ] deg.C) is used in these evaluations.

In addition, the outer diameter profiles along the axis of all the RCC rods for conventional 17x17 fuel assemblies have been measured by the UT method at every periodic inspection in Japanese plants. Figure 04.02-49-1 shows the outer diameter variation data, as of October 2011, for fast neutron fluence levels up to [ ]n/cm^2 (E>1MeV) corresponding to

[ ] years of operation. Since RCC rods are not fully inserted into the fuel assembly at all times, the maximum fluence level for these data is lower than the fluence level of [ ] n/cm^2 (E>1MeV) assumed in the design evaluation for the 15 year residence time as described above. The fluence level assumed in the design evaluation is therefore a conservative value. As shown in Figure 04.02-49-1, no significant swelling is observed for almost 15 years of operation. This result confirms that the RCC swelling evaluation given above is conservative.

<u>Ag-In-Cd:</u>

As shown in the Figure 04.02-49-1, no increase of the outer diameter in RCC cladding is observed after almost 15 years operation. This result indicates that there is no significant contact between the outer diameter of the Ag-In-Cd and the inner diameter of the RCC cladding. Therefore, it is concluded that Ag-In-Cd swelling is not a concern for the performance of the RCC for up to 15 years of operation.

b. The RCC rod diameter is tracked to EOL with the method described in the answer, a.

c. As of October 2011, 739 RCC assemblies fabricated by Mitsubishi have been irradiated in conventional 17 x 17 fuel type plants. Among them, there were [ ]RCC assemblies that achieved a maximum fluence level between [ ] and [ ] $\times 10^{22}$ n/cm^2 (E>1MeV). The US-APWR RCC assembly design is the same as the RCC assembly design for the conventional 17 x 17 fuel plants, except for the overall length.

Outer Diameter Change of RCC Rod (%)

Figure 04.02-49-1 Correlation between Neutron Irradiation Level and Outer Diameter Change of RCC Rods

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 NO.5918 (R3)

1/11/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO.5918 Revision 3**
SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/12/2011**

QUESTION NO. : 04.02-50

The following relates to the performance of the burnable absorber assembly in Section 5.2 of technical report MUAP-07016.

a. A value is provided for the softening point for the borosilicate glass but no information is provided for the basis of this value. Provide the basis for this value. Of concern is the small temperature margin between maximum temperature of the glass and the softening temperature.

b. Section 5.2.4.2 states that maximum absorber pressure is achieved within one cycle of operation but Sections 5.2.2 and 5.2.5.2 state a postulated lifetime of 15 years. Provide details of how the rod pressures are determined up to15 years of operation and include the resultant fluence level and 10B neutron captures.

c. Discuss MHI experience with borosilicate glass as a burnable absorber including maximum fluence level and ^{10}B neutron captures. Summarize past experience with MHI absorber rods similar to the APWR design and ^{10}B concentration. Discuss the differences between past designs and the APWR design.

ANSWER:

a. The softening point for the borosilicate glass is determined to be [ ] deg.F ([ ] deg.C), as given by the manufacturer's specification value of [ ] deg.C.

   Although the margin between the softening point and the evaluation temperature is small, the conditions used in the evaluation are conservative., The conditions are as follows:

   1) The boundary condition at the cladding surface is [ ] deg.F ([ ] deg.C), which is higher than the coolant boiling temperature ([ ] deg.F ([ ] deg.C)) at [ ] psia ([ ] MPa [abs]).

   2) The linear heat rate at the borosilicate glass is assumed to be the maximum value during AOOs, which corresponds to a fuel rod linear heat rate of [ ] kW/ft ([ ] kW/m).

   3) Heat transfer in the axial direction is not considered.

Due to these conservative conditions, the maximum temperature of the borosilicate glass is evaluated to be near its softening point. However, considering the conservative assumptions used in this evaluation, the calculated margin is sufficient.

b. Since expected function for the burnable absorber is reactivity control during one cycle of operation, it is assumed that released gas reaches its maximum value within one cycle of operation. On the other hand, for conservatism the cladding stress is evaluated at the BOL condition, where there is no gas release, because the cladding stress is due to the difference between the internal and external pressures, and assuming no gas release results in a larger pressure difference.

c. As of October 2011, Mitsubishi has irradiated [ ] burnable absorber assemblies in Japanese conventional 17 x 17 fuel type plants. Of these [ ] burnable absorber assemblies, [ ] have been irradiated for 1 cycle (typically 13-month cycle length), [ ] have been irradiated for 2 cycles and [ ] have been irradiated for 3 cycles. The maximum fast neutron fluence level for these burnable absorber assemblies is [ ] n/cm^2 (E>1MeV).

   The burnable absorber assembly design for the US-APWR is the same as the burnable absorber assembly design for the Japanese conventional 17x17 fuel type plants, except that the US-APWR burnable absorber overall length is 14ft, rather than 12ft.

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.

3.5 Chemical Reaction

3.5.1 Design Bases and Criteria

The fuel system will not be damaged due to excessive cladding corrosion during normal operation and AOOs.

During normal operation and AOOs the cladding surface temperature shall remain below the temperature at which an acceleration of corrosion could occur, as determined by the out of pile Zircaloy-4 cladding corrosion tests. To prevent this acceleration of the cladding corrosion, the calculated cladding metal-oxide interface temperature shall be less than[ ]deg.F during normal operation, and less than[ ]deg.F during AOOs [3-1]. These are the conservative design limits because they are based on Zircaloy-4 cladding corrosion tests, and the temperature required for accelerated corrosion of ZIRLO cladding is higher than that of Zircaloy-4.

The cladding hydrogen content shall remain below the value required to prevent degradation of cladding mechanical properties. Based on mechanical test data for irradiated and un-irradiated cladding material, the hydrogen content limit is established as[ ]ppm evaluated on a best estimate basis.

3.5.2 Evaluation

During normal operation and AOOs, the cladding surface temperature is limited to be less than the temperature at which an accelerated corrosion could occur.

The cladding metal-oxide interface temperature during irradiation is evaluated by the FINE code for both normal operation and AOO conditions. The metal-oxide interface temperature increases as the power increases during AOOs, but the cladding oxidation does not increase during an AOO due to the short duration of the AOOs. The evaluation takes into account the uncertainty in the fuel performance models. The uncertainty is[ ]deg.F ([ ]deg.C) based on [ ][3-1], assuming [ ] kW/ft ([ ] kW/m), determined from[ ]. This uncertainty is considered statistically at a 95 % probability at a 95 % confidence level. The cladding metal-oxide interface temperature of the limiting rod in the US-APWR equilibrium core during normal operation and AOOs is below the limits given above, as shown in Table 3.5-1. In addition to metal oxide interface temperature evaluation, the maximum oxide thickness of the limiting rod is less than or equal to [ ] which is determined as the limitation of FINE code corrosion model usage. For the cladding corrosion criterion, the limiting UO_2 rod is [ ], and the limiting $(U,Gd)O_2$ rod is [ ] using the identification of the region names as given in Appendix A. DCD 04.02-19 04.02-46

Hydrogen generated due to cladding corrosion is partially absorbed by the cladding. To maintain cladding ductility, the criterion requires that the hydrogen absorbed by the cladding is less than the hydrogen absorption limit. High temperature mechanical properties data for un-irradiated cladding show that the cladding retains its ductility up to the hydrogen absorption limit.

The cladding hydrogen content at the end of irradiation is also evaluated by the FINE code for normal operation on a best estimate basis, including best estimate plant conditions. Since oxidation is negligible during AOOs, as described above, hydrogen absorption does not