MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

January 30, 2009
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-09024

Subject: MHI's Responses to US-APWR DCD RAI No.129-1673 Revision 1

References: 1) "Request for Additional Information No. 129-1673 Revision 1, SRP Section: 04.02, Application Section: Chapter 4.2," dated December 17, 2008

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "MHI's Responses to US-APWR DCD RAI No.129-1673 Revision 1"

Enclosed are the responses to 20 RAIs contained within 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. 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 below.

Sincerely,

Y. Ogata

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

Enclosures:

1. Affidavit of Yoshiki Ogata

D081
NRO

2. MHI's Responses to the NRC's Requests for Additional Information on DCD RAI No.129-1673 Revision 1" (proprietary)

3. MHI's Responses to the NRC's Requests for Additional Information on DCD RAI 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-09024

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 "MHI's Responses to US-APWR DCD RAI No.129-1673 Revision 1" dated January 2009, and have determined that portions of the document 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 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 the unique design and manufacturing technology developed by MHI for the fuel of the US-APWR. These technologies 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 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 establishments of design and manufacturing technology, and development of the fuel system. Providing public access to such information permits competitors to duplicate or mimic the technology without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced safety and reliability and reduced manufacturing costs of the fuel system.

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 January, 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09024
Docket Number 52-021

MHI's Responses to US-APWR DCD RAI No.129-1673 Revision 1

January, 2009

(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/30/2009

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. : 04.02-1

Chapter 4.2 makes no reference to a ZIRLO topical report for material properties and performance. What is the source for the ZIRLO clad properties and performance?

ANSWER:

ZIRLO™ material properties and performance are summarized in Reference (1). Reference (1) Appendix A describes the ZIRLO irradiation experience and Reference (1) Appendix B provides the ZIRLO material properties and performance. Some of the performance data are given in Reference (1) Chapter 4, which describes the FINE code models for ZIRLO cladding.

<u>REFERENCE</u>

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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SRP SECTION:	**04.02 - Fuel System Design**
APPLICATION SECTION:	**04.02**
DATE OF RAI ISSUE:	**12/17/2008**

QUESTION NO. : 04.02-2

In Table 4.1-2 should the FINDS code be referenced in Loads and Stresses row?

ANSWER:

In the next revision of DCD, the FINDS code will be added to Table 4.1-2 under "Fuel".

Impact on DCD
The DCD will be changed to incorporate the following:

Page 4.1-9 Table 4.1-2
The FINDS code will be listed in the "Primary Code(s) Used" column in the row titled "Loads, stress and deflection of fuel assembly components".

Impact on COLA
There is no impact on the COLA.

Impact on PRA
There is no impact on the PRA.

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

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DATE OF RAI ISSUE:	**12/17/2008**

QUESTION NO. : 04.02-3

Section 4.4.1 of MUAP-07016 describes oxide thicknesses at an assembly burnup of 60 GWD/MTU. The section is titled Grid Spacer Irradiation Behavior but Appendix B only provides data for guide tube oxidation thicknesses and states that the behavior should be the same for grid spacers because both use recrystallized Zircaloy-4. Hydrogen distribution can be impacted by stress that can significantly reduce ductility. Spacer grids have different stress distributions than guide tubes. Has this assumption of similar behavior been verified by metalograhic examination in terms of hydrogen distribution or mechanical testing of spacer grids? Is the elongation data provided in Figure B.1.3.6-1 total or uniform elongation and what were the hydrogen levels for this data? If the data is total elongation please provide the uniform elongation data because total elongation is not a good measure of ductility to prevent brittle failure.

ANSWER:

MHI has not performed detailed metallographic examinations to investigate the hydrogen distribution in the grid spacers nor, to date, mechanical testing of the grid spacers. However, the spring force of the Zircaloy grid spacer, which causes the stress distribution, relaxes rapidly early in life, as shown in Figures 4.4-2 and 4.4-3 of Reference (1), and hence the stress magnitude and distribution within the grid spacers decreases under irradiation. Conversely, the hydrogen buildup is increasing slowly over lifetime. Therefore, the effect of the stress distribution in the Zircaloy grid spacer on the hydrogen distribution is small and data from guide tubes can be referenced for evaluation of the grid spacers during normal operation.

For seismic/LOCA accident conditions, the stresses in the grid straps can be high and the reduced ductility due to irradiation hardening and hydrogen uptake needs to be considered. Grid strength and deformation characteristics are routinely obtained by testing non-irradiated grids as opposed to FEM analysis. To account for irradiation effects on ductility, lower ductility can be appropriately simulated during grid impact testing by charging the grids to very high ppm levels of hydrogen. These tests will be conducted in the first quarter 2009 and will be described in the report, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads", to be submitted in March 2009. They are expected to verify that lower ductility does not have a significant effect on peak grid strength and on the deformation characteristics over the anticipated load/deflection range.

Figure B.1.3.6-1 of Reference (1) shows the total elongation data. Hydrogen levels have been measured for the data enclosed by rectangles in Figure 4.02-3-1 and range from [ ] ppm.

Total and uniform elongation data for irradiated control rod guide thimbles are shown in Figure 4.02-3-2.

REFERENCE

(1) "US-APWR FUEL SYSTEM AND DESIGN EVALUATION", MUAP-07016-P (Proprietary) and MUAP-07016-NP (Non-proprietary), February 2008.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

Total Elong. (%)

Fast Neutron Fluence (>=1MeV,m^{-2})

Figure 4.02-3-1 Mechanical Properties of the Control Rod Guide Thimble

Elongation (%)

Fast Neutron Fluence (>=1Mev, m-2)

Figure 4.02-3-2 Total and Uniform Elongation of Irradiated Control Rod Guide Thimble

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QUESTION NO. : 04.02-4

Do the spacer grid impact test results include the effect of oxide thickness and/or hydrogen pickup and if not why not?

ANSWER:

The effect of lower ductility in irradiated grid spacer material on grid spacer impact behavior will be simulated in the grid spacer impact tests MHI is planning to perform in the first quarter of 2009 and will be described in the report, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads", to be submitted in March 2009. The relaxed spring force and oxidation of irradiated grid spacers will be simulated in the tests. The lower ductility of the irradiated grid spacer material will be simulated by hydrogen charging to high ppm levels. The results will be compared with those in which only spring force relaxation is considered.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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QUESTION NO. : 04.02-5

Are guide tube stresses evaluated accounting for oxidation and/or hydrogen pick-up? What oxide thickness and/or hydrogen pick-up values are assumed?

ANSWER:

Guide tube stresses are evaluated at the beginning of life condition and are compared with the acceptance limit which is calculated based on the non-irradiated yield strength and ultimate tensile strength. Oxidation and hydrogen pick-up effects have been evaluated and the BOL analysis is conservative or appropriate due to the following reasons:

As shown in Figure B.1.2.2-1 of Appendix B of Reference (1), the double sided oxide thickness of the guide tube at assembly burnup of 60 GWd/MTU is conservatively estimated to be less than [ ]mil ([ ] µm) which corresponds to [ ] mil ([ ] µm) of base material thickness loss. This is approximately [ ]% of the wall thickness of the guide tube. Considering the [ ] % reduction of the guide tube's initial thickness, the stress would increase by [ ] % compared to the beginning of life condition. However, the yield strength and ultimate tensile strength of the guide tubes increase by more than 300 % during irradiation, as shown in Figure 4.02-5-1 (Reference (1) Appendix B Figure B.1.3.6-1). Another benefit is that corrosion is a slow process, with the peak oxide thicknesses occurring at EOL, whereas the material strengthening due to irradiation occurs at relatively low fluence levels. The irradiation-induced increase in the guide tube material strength results in more margin to the allowable stress after the effects of corrosion have been considered.

Based on re-crystallized annealed Zircaloy-4 sheet test data, the effect of hydrogen pick-up on yield strength and ultimate tensile strength is negligible up to [ ] ppm, as shown in Figure 4.02-5-2. A value of [ ] ppm hydrogen is the design limit used by MHI. These data confirm the insensitivity of the guide tube material properties to hydrogen level, and it is therefore appropriate to not explicitly consider the effect of hydrogen pick-up in the guide tube stress evaluation.

REFERENCES

(1) "US-APWR FUEL SYSTEM AND DESIGN EVALUATION", MUAP-07016-P (Proprietary) and MUAP-07016-NP (Non-proprietary), February 2008.

(2) J. Bai, et. al., "Effect of hydrides on the ductile-brittle transition in stress-relieved, recrystallised and beta-treated Zircaloy-4", International Topical Meeting on LWR Fuel Performance, Avignon, France, April 21 – 24, 1991.

Impact on DCD

There is no impact on the DCD.

Impact on COLA

There is no impact on the COLA.

Impact on PRA

There is no impact on the PRA.

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

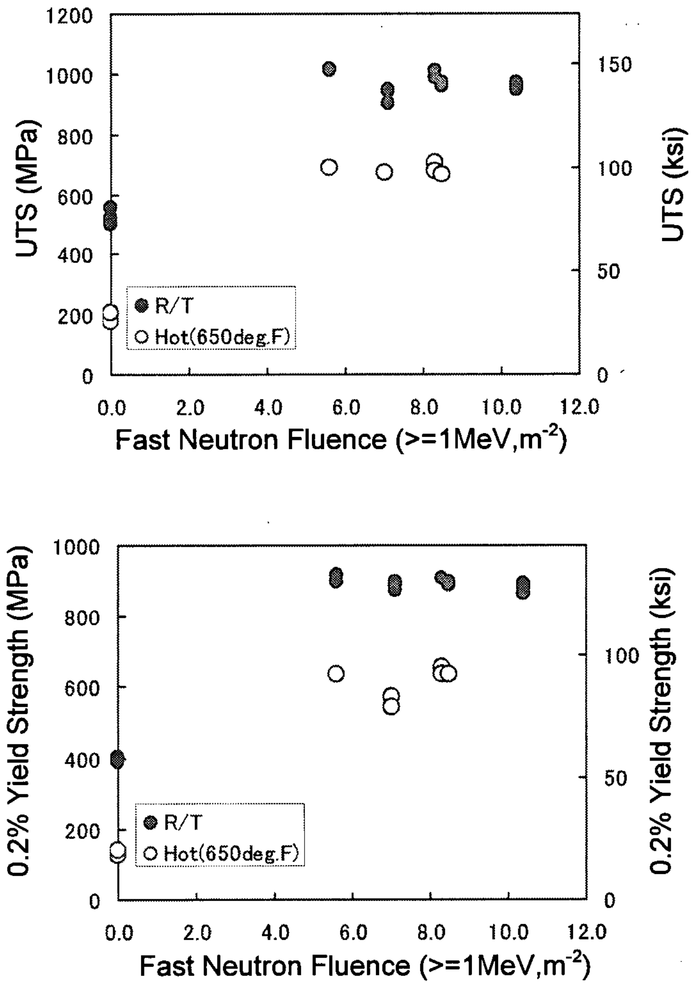

Figure 4.02-5-1 Mechanical Properties of the Guide Tube[1]

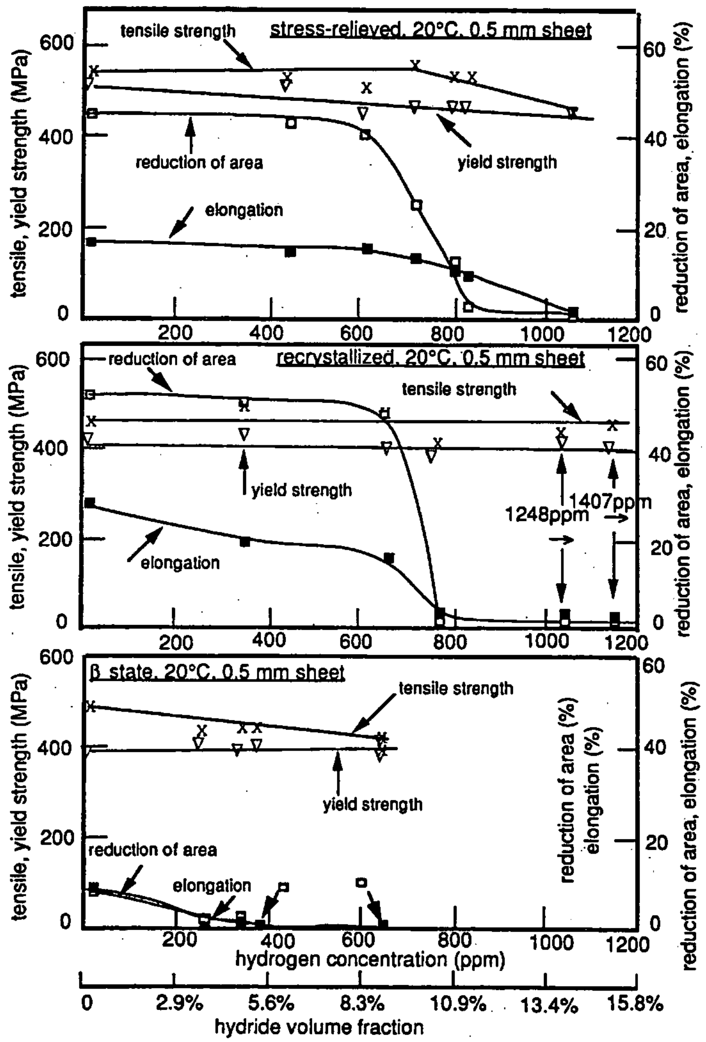

Fig.3: Influence of hydrogen concentration on the mechanical properties of Zircaloy-4 at room temperature.

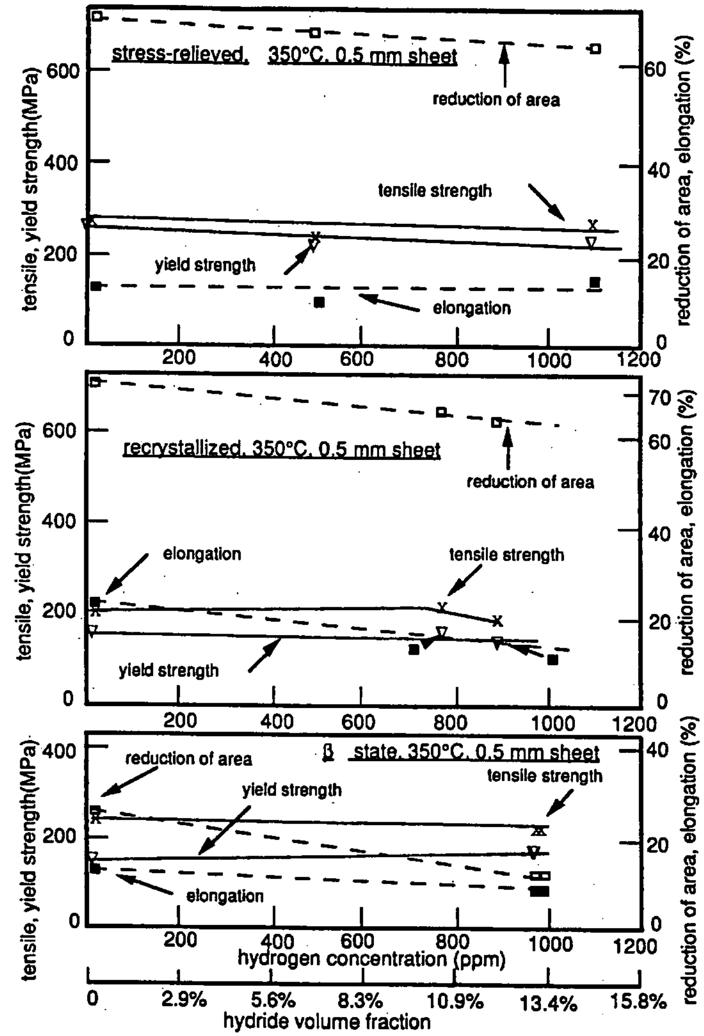

Fig.4: Influence of hydrogen concentration on the mechanical properties of Zircaloy-4 at 350°C.

Figure 4.02-5-2 Effect of Hydrogen pick up on Material Property of Zircaloy-4[2]

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QUESTION NO. : 04.02-6

In Section 4.6 of MUAP-07016 the document seems to indicate that the holdown spring undergoes plastic deformation under cold conditions. Is this the correct interpretation? Please clarify.

ANSWER:

Yes, it is the correct interpretation. Plastic deformation does occur at cold fit-up conditions where the spring experiences its maximum deflection. At normal operating temperatures, the spring operates at a lower deflection (due to the much lower thermal expansion of Zr-4 versus stainless steel) and on a linear portion of the load deflection curve.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

1/30/2009

US-APWR Design Certification

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QUESTION NO. : 04.02-7

Section 4.2.3.5.2 states that the guide tubes have a longer region with an enlarged inner diameter in the dashpot to prevent Incomplete Rod Insertion (IRI). This would appear to only move the weak point lower in the assembly. Has a comprehensive analysis for the potential of IRI been performed which includes assembly growth, holdown spring force and assembly lateral stiffness? This is especially important as it appears the holdown spring is designed to undergo plastic deformation. Has this design been compared to assembly designs which have experienced IRI?

ANSWER:

The control rod guide thimbles have a longer region with an enlarged inner diameter in the dashpot region, which provides more clearance between the control rod OD and the guide thimble ID to reduce the drag force on the rod cluster control assembly (RCCA). Fig 4.02-7-1 shows a schematic of the dashpot region. The larger clearance also decreases the slowing of the RCCA during a scram, which is compensated for by reducing the flow through the hole at the bottom of the dashpot in the guide tube screw. The additional diametrical clearance provides a substantial benefit by reducing the mechanical interaction between the control rod OD and the guide thimble ID.

The change from the guide thimble diameter to the dashpot diameter creates a discontinuity which can lead to local guide thimble assembly distortion. Also the smaller diameter dashpot reduces the local bending resistance. By replacing a section of the smaller diameter dashpot with the larger guide thimble diameter there is locally less potential for distortion due to the higher stiffness. There is another discontinuity before the end of the guide thimble assembly, as shown in Fig 4.02-7-1 (at the bottom nozzle), but the smaller dashpot diameter is now a shorter length and the transition is below the end of the RCCA travel.

The holddown springs for the US-APWR are designed to result in equal axial forces in the dashpot compared to that in MHI conventional fuel assemblies, which have not experienced IRI in Japanese PWR plants.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

Figure 4.02-7-1 The schematic of the control rod guide thimble dashpot region

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QUESTION NO. : 04.02-8

In Section 4.2.2.3.1, control rod lifetime is given as 15 years. Is this in calendar years or effective-full-power-years (EFPY)? What data and assumptions were used in determining the 15 year lifetime. (e.g., plant capacity factor, clad fluence values, absorber swelling rate etc.)?

ANSWER:

The control rod lifetime of 15 years given in Section 4.2.2.3.1 of the DCD is specified in calendar years.
Cladding wear, irradiation swelling of cladding and absorber material, cladding fatigue, compatibility between cladding and absorber material, and cladding corrosion are dependent on time. In addition, the accumulated fluence is important in assessing the swelling of the absorber and cladding and the resultant increase in clad diameter.

A larger gap between the cladding ID and absorber OD is used in the lowermost section of the control rod ([ ] in. ([ ] mm) from the bottom of the absorber). This compensates for the additional irradiation swelling of absorber material, that is shown in Section B.4.3.3 of Reference(1), due to it being closest to the active fuel.

Cladding wear is low due to chromium plating of the cladding surface. This design feature has been used in the RCCAs used by Mitsubishi in conventional PWRs, and Mitsubishi irradiation experience has shown that it effectively eliminates control rod clad wear. Cladding corrosion and fatigue are negligible. Compatibility is maintained by controlling the mechanical interaction between the absorber due to thermal expansion and swelling versus cladding due to creep and swelling. The two materials are also chemically stable with one another.

As described in Reference (1) Section 5.1, the most dominant phenomenon limiting control rod operational lifetime is irradiation swelling of the cladding.
The maximum increase of the control rod diameter during its lifetime is[ ]mil ([ ] mm). The minimum clearance between the control rod and the control rod guide thimble, at the guide thimble dash pot region, is[ ] mil ([ ] mm), taking into account manufacturing tolerances.

In addition, while the maximum allowable fluence exposure to the lower portion of the RCCA rodlet is 5×10^{22} (n/cm^2) to assure only a small reduction in neutron absorbing capability, It is been concluded that there will be no significant control rod cladding wear over its design lifetime of 15 years, as described in answer to QUESTION NO. 04.02-10.

The control rod lifetime of 15 calendar years has been shown to be met with sufficient margin.

<u>REFERENCE</u>

(1) "US-APWR FUEL SYSTEM AND DESIGN EVALUATION", MUAP-07016-P (Proprietary) and MUAP-07016-NP (Non-proprietary), February 2008.

Impact on DCD
There is no impact on the DCD

Impact on COLA
There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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QUESTION NO. : 04.02-9

Have control rod neutron-absorbing capabilities been evaluated over the projected lifetime? If so, where is this discussed? If not, what assumptions are made regarding neutron-absorbing capabilities and explain why they are conservative.

ANSWER:

The control rod neutron absorbing capabilities have been evaluated and a limit on fluence exposure has been set. The fluence criterion for control rod replacement is 5×10^{22} (n/cm^2). Using this value, an analysis was performed to determine the effect on neutron absorption capability. This analysis used the conservative assumption that[

] Based on this analysis, the reduction in control rod neutron-absorbing capability is less than [ ] at an exposure of 5×10^{22} n/cm^2.

However, this value only applies to the tip of the control rod. For typical operation of the US-APWR, the control rods are normally almost completely withdrawn and therefore only the lower portion of the control rod experiences significant exposure to the neutron flux. On a control-rod average basis, the reduction of control rod neutron-absorbing capability is less than [ ] of total control rod worth, and therefore the impact on reactivity control and shutdown margin is negligible over the control rod lifetime.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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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:

It is been concluded that there will be no significant control rod cladding wear over its design lifetime based on the following data:

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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QUESTION NO. : 04.02-11

Over what extent (dimension) does the chrome plating cover the control rod?

ANSWER:

[ ]

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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QUESTION NO. : 04.02-12

Has an evaluation been performed examining the potential for guide tube water boiling when the control rods are present? If not, has the impact of possible boiling induced corrosion been evaluated on guide tube integrity?

ANSWER:

A bounding calculation is performed to verify that no boiling will occur in a guide thimble when an In-Core Control Component (ICCC) rod is present. The heat generation, when a control rod is in the guide thimble, is smaller than that for a burnable absorber rod. Of course, during normal operation the control rods are almost fully withdrawn from the core. Therefore an evaluation of a burnable absorber rod inserted into the guide thimble is performed as part of the ICCC analysis to determine whether boiling occurs in a guide thimble or not. This is a conservative analysis for the following reasons:

- fuel rods surrounding the guide thimble are assumed to generate maximum power, and,
- the axial power distribution used maximizes the water temperature in the guide thimble.

The results of the evaluation show that no boiling will occur in the guide thimble when any In-Core Control Component rod, including the control rods, are present.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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QUESTION NO. : 04.02-13

Oxidation and hydrogen pick-up reduce clad ductility especially at high fuel duties over an extended time period. Does the cladding 1% strain limit include the effect of the maximum oxide thickness and/or the maximum hydrogen pick-up limit? If so, what values were used? If not, explain why it is conservative to not include oxide thickness and/or hydrogen pick-up effects.

ANSWER:

The cladding strain limit for the US-APWR fuel accounts for the degradation of cladding ductility due to hydrogen pick-up during irradiation. The following basis for the US-APWR cladding oxidation and hydriding design criteria is given in Section 3.2.6 of Reference (1):

> 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.
>
> The test data to support these design limits are presented in Appendix B.

Appendix B of Reference (1) describes the mechanical properties of fuel rod. The following description of the strain criterion is given in Reference (1) Appendix B Section B.3.3.5 and the Reference (2) RAI 5a response:

> The results of high-temperature tensile tests of irradiated cladding show that the total elongation decreases as the hydrogen absorption increases up to approximately [ ] [ ] ppm, but this effect saturates at higher hydrogen concentrations and the total elongation exceeds the 1 % criterion for hydrogen concentrations up to [ ] ppm. This shows that the 1 % total strain criterion is conservative up to hydrogen concentrations of [ ] ppm.

REFERENCES

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

(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 COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/30/2009

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. : 04.02-14

In Section 4.2.4.2, mention is made of visually inspecting fuel pellets. Missing pellet surfaces caused by manufacturing deficiencies have lead to PCMI fuel failures and greatly increase the chance of cladding collapse. Are manufacturing (quality assurance) standards in place which limits the amount (area or volume) of missing pellet material?

ANSWER:

Visual standards for the following conditions are used for inspection on all pellets as quality assurance to prevent use of pellets having unacceptable levels of missing pellet material:

[ ]

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1/30/2009

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. : 04.02-15

Provide the data that substantiates the 95/95 uncertainty values for corrosion and hydrogen provided in Table 3.1-2 of MUAP-07016.

ANSWER:

Reference (1) Section 4.3.5 provides the following basis for the 95/95 uncertainty values for corrosion:

> The model uncertainties have been determined using the ZIRLO™ data with measured oxide thicknesses [ ]. The results for the uncertainties at a 95% probability at a 95% confidence level are:
>
> Number of data : [
> M-P average :
> M-P standard deviation :
> Uncertainty Upper Bound :]
>
> The uncertainty of ZIRLO™ corrosion model is conservatively set as [ ], which is the same value for Zircaloy-4 model.

Reference (1) Section 4.3.6 provides the following basis for the 95/95 uncertainty values for hydrogen:

> Uncertainty is calculated with the data more than [ ]ppm of hydrogen content which is approximately corresponding to hydrogen solubility limit at around operation temperature.
>
> Number of data : [
> M-P average :]

M-P standard deviation : [ ] (ppm)
Uncertainty Upper Bound : [ ] (ppm)

Uncertainty of hydrogen content is conservative set as [ ] ppm which is the same value applied to Zircaloy-4.

REFERENCE

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

REFERENCE

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

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SRP SECTION:	**04.02 - Fuel System Design**
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DATE OF RAI ISSUE:	**12/17/2008**

QUESTION NO. : 04.02-16

Have manufacturing moisture limits been set which eliminate internal hydriding? If so, what are the limits and what manufacturing controls (quality assurance sampling) are in place to limit fuel moisture? If not, explain why no internal moisture limits are necessary to prevent internal hydriding.

ANSWER:

The manufacturing control on pellet moisture content is the following:

[ ]

This control assures that the moisture content is kept sufficiently low compared with its limit of 1.3 ppm (hydrogen from all sources for fuel pellets) specified in "Standard Specification for Sintered Uranium Dioxide Pellets" to American Standards and Testing Methods (ASTM) C776-06 and Section 3.3.1 of Reference (1).

REFERENCE

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-Proprietary), May 2007.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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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. : 04.02-17

Section 4.2.1.2.2 states that the "FINE code evaluates cladding stress, cladding strain, fatigue, fuel temperature, rod internal pressure, etc., using applying the fuel densification model and the swelling model." Wording should be modified by deleting either "using" or "applying".

ANSWER:

The word, "applying", is to be deleted in the next revision of the DCD.

Impact on DCD

The DCD will be changed to incorporate the following:

Page 4.2-4 line 1
The FINE code evaluates cladding stress, cladding strain, fatigue, fuel temperature, rod internal pressure, etc., using ~~applying~~ the fuel densification model and the swelling model.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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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:

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. Utility concurrence will be obtained for the items to be measured and the number of fuel assemblies to be measured. The following is a tentative plan.

Video cameras will be used for the visual inspection and image analysis of the irradiated fuel assemblies. Fuel assembly length and gaps between the fuel rod and top/bottom nozzles will be measured by image analysis through video media such as video tape. In addition, fuel assembly bow is measured using ultra-sonic testing equipment. Significant fuel assembly bow effects loading assemblies into the core and can create interference with the handling tool. The re-insertion acceptance criterion is [ ] in.([ ] mm) of bow, which is determined from interference with the handling tool.

Impact on DCD

There is no impact on the DCD

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

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US-APWR Design Certification

Mitsubishi Heavy Industries

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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:

The Surveillance Program for fuel assemblies loaded in the initial core is described as follows (DCD Section 4.2.1.7):.

> The US-APWR fuel assembly is designed and manufactured based on the substantial database obtained from the testing and fuel surveillance programs on Mitsubishi conventional fuel assemblies, which has been used for verification of fuel performance and validation of the design bases, as described in Reference (1).
>
> A surveillance program for the US-APWR fuel assembly will be established for verification of the fuel performance and validation of the design bases. This surveillance program will specify the inspection items, inspection criteria, methodology, schedule, number of fuel assemblies and in-core control components such as burnable absorbers and RCCAs, so as to be sufficient for identifying gross problems of structural integrity, fuel rod failure, rod bowing, dimension changes, or crud deposition. This program will also include criteria for additional inspection requirements for irradiated fuel assemblies or in-core control components, if abnormal behavior is observed during operation or in visual inspections as described in DCD Subsection 4.2.4.5.

Some of the US-APWR fuel assemblies loaded in the initial core will be closely examined to confirm their performance. The tentative surveillance program is shown in Table 04.02-19-1.

Utility concurrence will be obtained for the items to be measured and the number of assemblies to be inspected..

REFERENCES

(1) "Mitsubishi Fuel Design Criteria and Methodology", MUAP-07008-P (Proprietary) and MUAP-07008-NP (Non-proprietary), May 2007.

(2) "US-APWR FUEL SYSTEM AND DESIGN EVALUATION", MUAP-07016-P (Proprietary) and MUAP-07016-NP (Non-proprietary), February 2008.

(3) "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 COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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

Table 04.02-19-1 Tentative Surveillance Program
Measurements and Inspections for Fuel Assemblies loaded into the initial Core

Measurements and Inspections	Number of Assemblies	Inspection Method	Acceptance Criteria

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US-APWR Design Certification

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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:

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 CD-ROM for the response to RAI 9d of Reference (2).

REFERENCES

(1) "US-APWR Fuel System Design Evaluation", MUAP-07016-P (Proprietary) and MUAP-07016-NP (Non-Proprietary), February 2008.

(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 COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

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