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

January 10, 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-12004

Subject: MHI's Response to US-APWR DCD RAI No. 874-6101 Revision 0 (SRP 04.03)

Reference: 1) "REQUEST FOR ADDITIONAL INFORMATION 874-6101 REVISION 0, SRP Section: 04.03 - Nuclear Design, Application Section: 4.3, QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)."

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document as listed in Enclosure.

Enclosed is the response to the Non-Public RAI 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 response (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 submittal. His contact information is below.

D081
NRO

Sincerely,

Y. Ogata

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

Enclosure:

1. Affidavit of Yoshiki Ogata

2. Response to Request for Additional Information No. 874-6101 Revision 0 (proprietary version)

3. Response to Request for Additional Information No. 874-6101 Revision 0 (non-proprietary version)

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

Contact Information
C. Keith Paulson, Senior Technical Manager
Mitsubishi Nuclear Energy Systems, Inc.
300 Oxford Drive, Suite 301
Monroeville, PA 15146
E-mail: ck_paulson@mnes-us.com
Telephone: (412) 373-6466

Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-12004

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, 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 public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 874-6101 Revision 0" dated January 10, 2012, 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 it describes the unique design and methodology developed by MHI for performing the nuclear design of the US-APWR reactor.

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 development of methodology related to the analysis.

B. Loss of competitive advantage of the US-APWR created by benefits of modeling information.

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 10th day of January, 2012.

Y. Ogata

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

Enclosure 3

UAP-HF-12004
Docket No. 52-021

Response to Request for Additional Information No. 874-6101
Revision 0

January 2012
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

01/10/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	NO. 874 - 6101 REVISION 0
SRP SECTION:	4.3 – Nuclear Design
APPLICATION SECTION:	4.3
DATE OF RAI ISSUE:	12/05/2011

QUESTION NO.: 04.03-68

A combined license (COL) information item should be added to Table 1.8-2 of the FSAR that plant specific surveillance capsule data shall be compared to US-APWR fluence predictions. If the capsule data indicates a more limiting End of License (EOL) fluence value the prediction shall be modified to reflect the measured results. If measured data is used to adjust the prediction the adjustment and its basis shall also be documented.

ANSWER:

MHI will add the following sentence to Subsection 4.3.2.8 to ensure that surveillance results are reflected in EOL fluence predictions:

"An End-of-Life (EOL) neutron fluence verification program will be established prior to the reactor vessel material surveillance program. The program will contain adjustments to be made to the EOL fluence predictions shown in Table 4.3-5 if the surveillance capsule data indicates more limiting EOL fluence values. The results and bases of adjustments will be documented."

Impact on DCD

DCD Chapter 4, Subsection 4.3.2.8 is to be revised as indicated in 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 Reports

There is no impact on the Technical / Topical Reports.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

01/10/2012

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No.52-021

RAI NO.:	NO. 874-6101 REVISION 0
SRP SECTION:	4.3 – Nuclear Design
APPLICATION SECTION:	4.3
DATE OF RAI ISSUE:	12/5/2011

QUESTION NO. : 04.03-69

An uncertainty analysis was performed in technical report MUAP-07021 for five failed incore instrumentation thimbles. Additional measurement uncertainties of [ ] for Fq and [ ] for FDH were determined for five failed thimbles with no failed thimbles within the square root of 10 assembly pitches from each other. Based on this analysis either add a statement to the DCD Chapter 4.3 such as,
"The US-APWR is limited to a maximum of five failed instrument thimbles with no adjacent failed thimbles within the square root of 10 assembly pitches from each other when performing incore flux measurements. Greater than 5 failed thimbles or failed thimbles within the square root of 10 assembly pitches of each other is considered a change to the approved methodology."
or justify why such as statement is not necessary.

ANSWER:

MHI revises the DCD Chapter 4.3 to include a statement which specifies that the thimble failure conditions exceeding the assumed conditions used in the MUAP-07021 Appendix A are considered a change to the approved methodology.

Impact on DCD

DCD Chapter 4.3 Rev.3 is revised as indicated in mark-up shown in 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 Reports

There is no impact on the Technical / Topical Reports.

trip settings are provided in the technical specifications, while the descriptions of the systems are provided in Sections 7.2 and 7.7.

The comparison between 'measured' and predicted power distributions for the same situation includes some measurement error, and it is necessary to account for such error. The accumulated data on power distributions in actual operation are basically data taken during normal operation in steady-state equilibrium conditions, and data from induced xenon transient conditions. Examples of these data are presented in detail in Reference 4.3-5.

As described in Subsection 4.3.2.2.2, uncertainties must be applied to the limiting power distributions. The appropriate uncertainties are discussed in References 4.3-7 and 4.3-8. A summary of the bases, results and conclusions of the referenced reports is given below.

Using measurements of core power distributions with the incore instrumentation system described in Section 7.7 and Subsection 4.4.6, the following uncertainties are considered:

1- Measurement reproducibility.

2 - Errors in the calculated relationship between detector current and local power generation within the fuel assembly.

3 - Errors due to the inference of power some distance from the measurement thimble, i.e. the extrapolation method.

The appropriate allowance for category 1 above has been quantified by multiple measurements made with several inter-calibrated detectors by using the common thimble features of the in-core detector system. This system allows more than one detector to access any thimble. Control of measurement reproducibility is provided by strict adherence to manufacturing specifications. Errors in category 2 above are quantified by comparisons of analytical results and measurements for critical experiment data on arrays of rods with simulated guide thimbles, control rods, burnable poisons, etc. Errors in category 3 are quantified by multiple comparisons of two 'measured' power distributions of actual plants in which extrapolated results were compared to 'reference' results.

The approach for determining the above uncertainties is discussed in Reference 4.3-7, which describes critical experiments performed and plant measurements taken with incore instrumentation systems. In Reference 4.3-7, the uncertainties listed above are evaluated quantitatively. The Reference 4.3-8 report re-evaluates the uncertainties under the conditions of the US-APWR incore instrumentation system. Both reports conclude that the uncertainty associated with peak linear heat rate (F_Q x P) results in an allowance of five percent, at a 95 percent probability at a 95 percent confidence level. For conservatism, an eight percent uncertainty factor in F_Q x P is adopted for US-APWR nuclear design.

A similar analysis for the uncertainty in $F^N_{\Delta H}$ x P (hot rod integral power) results in an allowance of four percent, at a 95 percent probability at a 95 percent confidence level. A conservative six percent uncertainty factor is applied to $F^N_{\Delta H}$ x P for US-APWR nuclear design. Thimble failure simulations were performed in Reference 4.3-8 by assuming a DCD_04.03-69 maximum of five failed instrument thimbles with no adjacent failed thimbles within the square root of 10 assembly pitches from each other. The results showed that the thimble failure impact can be bounded by the margins contained in the uncertainty factors for FQ and FNΔH shown above. Thimble failure conditions exceeding the assumed conditions are considered a change to the approved methodology. DCD_04.03-69

4.3.2.3 Reactivity Coefficients

In operating reactors, the transient changes in the neutron effective multiplication factor due to the change of some plant conditions, such as density, temperature, void and power, can be expressed in terms of reactivity coefficients. In a PWR, important reactivity coefficients to compensate for the rapid reactivity insertion and the associated increases in reactor power are the fuel temperature coefficient and moderator coefficients due to the change of the temperature and density. The combined effect of fuel and moderator temperature changes due to power level changes can be translated into power coefficients. Other moderator coefficients, related to the pressure and the void, are relatively not very significant in PWRs, but void effects are included for shutdown margin considerations.

The reactivity coefficients are calculated using the analytical methods described in Subsection 4.3.3.1. Comparisons of calculated and experimental reactivity coefficients are included in Reference 4.3-5.

Since reactivity coefficients change with core parameters such as cycle burnup and control rod insertion ratio, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout its life. The results of such simulations and the reactivity coefficients used are presented in Chapter 15.

4.3.2.3.1 Fuel Doppler Coefficient

The fuel Doppler coefficient is defined as the change in reactivity per degree change in effective fuel temperature. As a consequence of the Doppler effect, resonance cross sections are broadened and the neutron absorption in the resonance region is increased and, thus, the neutron escape probability decreases by the increase in temperature. U-238 has several narrow resonance peaks, so in PWRs, which are fueled with low enriched uranium, the Doppler temperature coefficients are negative. The Pu-240 has a large neutron absorption resonance peak and its concentration increases with cycle burnup as shown in Figure 4.3-3, so the Doppler coefficient becomes more negative with cycle burnup. The influence of other isotopes is considered, however, they are relatively small compared with U-238 and Pu-240.

The effective fuel temperature and the reactivity change are calculated with the analytical method described in Subsection 4.3.3.1. Reactivity is calculated by slightly varying the effective fuel temperature around the reference value. The moderator temperature is held constant, in order to specifically exclude the moderator temperature effect. The Doppler temperature coefficient is a weak function of the boron concentration. Typical Doppler temperature coefficients for BOC and EOC as a function of the fuel effective temperature are shown in Figure 4.3-21.

"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 4.3-9). The evaluation methodology is described in Reference 4.3-20. The summary of the methodology is as follows.

The DORT code (Reference 4.3-10) is used to evaluate neutron flux distributions. DORT is widely used in the nuclear industry for flux-distribution evaluations of reactor vessels. DORT is a discrete ordinates Sn code, and can perform calculations in (X, Y), (R, θ) and (R, Z) geometry. Each of these geometric capabilities is used as described below.

To determine the reactor vessel neutron flux distribution, (R, θ) geometry is selected, modeling the circular shape of the reactor vessel and core barrel. Likewise, the irregular shapes of the core and the neutron reflector are "smoothed" and are modeled as circular regions. Calculations of the perpendicular neutron flux distribution in the reactor vessel are performed with an (R, Z) geometric model. On the other hand, in the evaluation of the neutron reflector neutron flux distribution, greater local spatial accuracy is desired. Therefore, (X, Y) geometry is selected for this region, and the "polygon" shape of the side facing the reactor core of the neutron-reflector is modeled by straight lines.

For the DORT calculations, the BUGLE-96 (Reference 4.3-11) cross section library is used. This library is generated from ENDF/B-VI data collapsed to 47 neutron energy groups.

Fuel assembly and pinwise power distributions are obtained by standard reactor core calculations and are used as the fission source terms for the neutron flux distribution calculations outside the reactor core. The average power distribution in a representative reactor core operating at full power is used. This distribution is considered to be representative of the core average power distribution during the plant lifetime. This approach is considered appropriate for reactor vessel irradiation calculations, because the integrated flux (neutron fluence) during operation is considered most important. A typical neutron flux inside the reactor vessel obtained by the above evaluation is shown in Table 4.3-4. In addition, fast neutron fluence (time integrated neutron flux) at reactor vessel is shown in Table 4.3-5. An End-of-Life (EOL) neutron fluence verification program will be established prior to the reactor vessel material surveillance program. The program will contain adjustments to be made to the EOL fluence predictions shown in Table 4.3-5 if the surveillance capsule data indicates more limiting EOL fluence values. The results and bases of adjustments will be documented. DCD_04.03-68

Calculation uncertainty of this flux estimation methodology is within 20 %, which was estimated by uncertainty analysis. The reactor vessel surveillance program is discussed in Subsection 5.3.1.6.

4.3.3 Analytical Methods

4.3.3.1 Nuclear Design Methods

A lattice physics code and a core simulator have been mainly used for US-APWR nuclear design. The lattice physics code is used for generating group constants for core simulation and the core simulator is then used for calculating the main nuclear parameters, such as power distribution, exposure, critical boron concentration, and reactivity coefficients.