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TOKYO, JAPAN

March 4, 2013

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC20555-0001

Attention: Mr. Jeffrey A. Ciocco

Docket No. 52-021
MHI Ref: UAP-HF-13037

Subject: MHI's Response to ACRS Subcommittee Questions on Jan. 15, 2013

Reference: 1)"Thermal Design Methodology", MUAP-07009, Revision 0, May 2007.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to ACRS Subcommittee Questions on Jan. 15, 2013".

Enclosed are the responses to the ACRS subcommittee questions related to Reference 1 which were raised in the meeting on January 15, 2013.

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) of the response, a copy of the non-proprietary version (Enclosure 3) of the response, and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Mr. Joseph Tapia, General Manager of Licensing Department, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittal. His contact information is 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 ACRS Subcommittee Questions on Jan. 15, 2013 (Proprietary version)

3. Response to ACRS Subcommittee Questions on Jan. 15, 2013 (Non-proprietary version)

CC: J. A. Ciocco
J. Tapia

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ENCLOSURE1

Docket No. 52-021
MHI Ref: UAP-HF-13037

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 ACRS Subcommittee Questions on Jan. 15, 2013" dated March 2013, 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 as 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 information of thermal design methodology developed by MHI and not used in the exact form by any of MHI's competitors. This information was developed at significant cost to MHI, since it required the performance of research and development and detailed design for its software and hardware extending over several years.

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 the thermal design. Providing public access to such information permits competitors to duplicate or mimic the methodology without incurring the associated costs.

B. Loss of competitive advantage of the US-APWR created by benefits of enhanced plant safety, and reduced operation and maintenance costs associated with the thermal design.

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 4th day of March, 2013.

Y. Ogata

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

Enclosure 3

UAP-HF-13037
Docket No. 52-021

Response to ACRS Subcommittee Questions on Jan. 15, 2013

March 2013
(Non-Proprietary)

March 4, 2013

RESPONSE TO ACRS SUBCOMMITTEE QUESTIONS ON JAN. 15, 2013

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

TOPICAL REPORT NUMBER:	**MUAP-07009**
TOPICAL REPORT TITLE:	**THERMAL DESIGN METHODOLOGY**
DATE OF MEETING:	**01/15/13**

QUESTION: Item 1

With regards to CHF testing and uncertainties in the measured results, MHI should confirm that the code is still conservatively predicting the results.

ANSWER:

A measured-to-predicted CHF ratio (M/P) uncertainty was derived based on the measurement uncertainties for the KATHY loop and the contribution of each parameter uncertainty to the M/P uncertainty. As shown in Table 1, the instrumentation uncertainties for the KATHY loop are comparable to those in the HTRF loop. The contribution of each parameter uncertainty is evaluated by sensitivity factor based on the VIPRE-01M analyses, using nominal and biased input for the corresponding parameter. The results are combined into the M/P uncertainty by using the root-sum-square (RSS) method; the resulting M/P uncertainty (U_{95}) is estimated to be 1.6%, which corresponds to a 95% upper limit. Detailed numbers in this evaluation are presented in Table 2. As shown in the table, it should be noted that the M/P uncertainty is significantly smaller than the 17% uncertainty in the 95/95 DNBR limit, which is statistically estimated to cover the M/P variation due to measurement uncertainties as well as uncertainty of the DNB phenomenon itself.

In addition, the measurement uncertainty that causes M/P variation is taken into account by the 95/95 DNBR limit (Correlation Limit: CL) which has the following form:

$$CL = \frac{1}{m - k_p s}$$

Here m is sample (M/P) mean, and k_p is Owen's k-factor, which gives a 95% confidence interval of the 95% probability limit for a population. The sample standard deviation s includes the effect of all the uncertainties that cause M/P variation; it includes not only the measurement uncertainties that cause random M/P scatterings but also the ones that cause M/P variation depending on the measurement conditions (e.g. M/P dependency on mass flow). Thus the only type of uncertainty that remains and is not taken into account by s is the bias on m, which is constant throughout the M/P samples. Since most of the measurement uncertainties are considered to be parameter dependent, the portion of uncertainty that biases m is considerably smaller than overall measurement uncertainty.

In past industry practice, the bias on *m* has been empirically ignored when determining the 95/95 DNBR limit because this type of uncertainty occupies only a small part of the overall measurement uncertainty, which is, in turn, considerably smaller than the uncertainty considered by the 95/95 DNBR limit.

Table 1: Comparison of Instrumentation Errors between KATHY and HTRF

(The same as Table 3.0-2 of DNB test report MUAP-11010-R2)

Table 2: Calculation of M/P uncertainty for KATHY loop

RESPONSE TO ACRS SUBCOMMITTEE QUESTIONS ON JAN. 15, 2013

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

TOPICAL REPORT NUMBER:	**MUAP-07009**
TOPICAL REPORT TITLE:	**THERMAL DESIGN METHODOLOGY**
DATE OF MEETING:	**01/15/13**

QUESTION: Item 2

In an RAI response, a typographical error was found in a standard deviation listing ([ ]). MHI should confirm the correct value and verify that the rest of the numbers in that calculation and tables are correct.

ANSWER:

The same typographical error was found in Table 4.2-1 and 4.2-2 of the RAI response in UAP-HF-09500 and Table 04.04.4-1 and 04.04.4-2 in UAP-HF-09546. Here, UAP-HF-09500 contains RAI responses on the Topical Report MUAP-07009, and UAP-HF-09546 contains the amended response to US-APWR DCD RAI No. 377-2629 Rev. 1, originally submitted in UAP-HF-09336. UAP-HF-09500 Table 4.2-1 values are identical to UAP-HF-09546 Table 04.04.4-1. UAP-HF-09500 Table 4.2-2 values are identical to UAP-HF-09546 Table 04.04.4-2.

It was confirmed that the correct value of [ ] was used in the actual calculations, and there are no other errors in the tables. The correct tables are shown below. Amendments for these RAI responses will be submitted in a separate letter.

Table.4.2-1 Design Limit of Minimum DNBR for US-APWR based on RTDP (Typical Cell)

Parameters (x_i)	μ_i	σ_i	σ_i/μ_i	S_i	$S_i^2(\sigma_i/\mu_i)^2$
Power (fraction)					
Tin (°F)					
Pressure (psia)					
Flow (fraction)					
Effective Core Flow (fraction)					
$F^N_{\Delta H}$					
$F^E_{\Delta H,1}$					
Subchannel Code					
Transient Code					

- Uncertainties of input parameters and code predictions:

  [ ]

- Uncertainty of DNB correlation prediction (See Table B.3-6 of MUAP-07009-P):

  [ ]

  [ ]

- Design limit of minimum DNBR (DL)

Table.4.2-2 Design Limit of Minimum DNBR for US-APWR based on RTDP (Thimble Cell)

Parameters (x_i)	μ_i	σ_i	σ_i/μ_i	S_i	$S_i^2(\sigma_i/\mu_i)^2$
Power (fraction)					
Tin (°F)					
Pressure (psia)					
Flow (fraction)					
Effective Core Flow (fraction)					
$F^N_{\Delta H}$					
$F^E_{\Delta H,1}$					
Subchannel Code					
Transient Code					

- Uncertainties of input parameters and code predictions:

  [ ]

- Uncertainty of DNB correlation prediction (See Table B.3-6 of MUAP-07009-P):

  [ ]

  [ ]

- Design limit of minimum DNBR (DL)

RESPONSE TO ACRS SUBCOMMITTEE QUESTIONS ON JAN. 15, 2013

US-APWR Design Control Document

Mitsubishi Heavy Industries, Ltd.

TOPICAL REPORT NUMBER:	**MUAP-07009**
TOPICAL REPORT TITLE:	**THERMAL DESIGN METHODOLOGY**
DATE OF MEETING:	**01/15/13**

QUESTION: Item 3

Provide a list of items which need to be validated in VIPRE-01M with justification for scaling if it is used (e.g.: use of 12 ft test data for turbulent mixing, use of 1/7 scale inlet mixing test data).

ANSWER:

Qualifications for the VIPRE-01M model and input options are summarized in Table 3, shown below. These options are discussed as well in Topical Report MUAP-07009. Each model option selected for US-APWR analysis is one of the selectable models included in the original VIPRE-01 code (Ref. 1), which was generically approved by the NRC. Each of these options chosen was selected to use conservative and/or well accepted classical models. Finally, qualification of the total effect of the selected options was done by benchmark analyses of VIPRE-01M results with NRC approved code results which provide inherently conservative results.

Additionally, experiments were done to support the US-APWR application.

Full scale (1/1 ratio) rod bundle tests at PWR conditions provide the 95/95 limit value of the mixing parameter in the rod bundle and DNBR correlation limit. Although the mixing test was performed using a 12-ft partial-length rod bundle, the mixing parameter is a "local parameter" and the dominant geometries for this phenomena, such as subchannel dimension and grid spacing, were modeled in full scale. Furthermore, a sensitivity study shows that the mixing parameter effect is negligibly small and does not have an effect in actual plant analysis. DNB tests were performed using both full length (14-ft) and partial-length (12-ft) rod bundle test sections, which provided comparable results.

A1/7 scale hydraulic test was performed at room temperature for the downcomer and lower plenum mixing. This model partially simulates the reactor vessel (RV) of the US-APWR from the RV inlet nozzle to the core inlet. The scaling effect on the inlet flow distribution is justified by the Reynolds scaling law. The core inlet flow distribution obtained from the test is comparable to the assumption used in the plant analysis. In addition, the effect of this parameter is negligibly small and does not have an effect on actual plant analysis.

References:

1. C. W. Stewart, et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores Volume1–5 (Revision 4)," NP-2511-CCM-A Electric Power Research Institute (EPRI), February 2001.
2. Response to RAI 1.10, Attachment1 of Letter from Y. Ogata to J. A. Ciocco, "Response to the NRC Request for Additional Information on 'Thermal Design Methodology' MUAP-07009 Rev. 0," MUAP-HF-09500, October, 2009.
3. S. Watanabe, et al., "US-APWR Reactor Vessel Lower Plenum1/7 Scale Model Flow Test Report," MUAP-07022, June 2008.
4. J. Ogawa, et al., "Non-LOCA Methodology," MUAP-07010, 2007.
5. Y. Makino, et al., "Thermal Design Methodology," MUAP-07009, May, 2007
6. M. Tatematsu, et al., "DNB Tests for US-APWR Fuel," MUAP-11010, July 2012.
7. T. Ogata, et al., "Hydraulic Test of the Full Scale US-APWR Fuel Assembly," MUAP-11017-P, May 2011.

Table 3: Qualifications for VIPRE-01M Models and Input Options

Phenomena/ Models	Design Consideration	Qualification	Supporting Experiment	Scale Ratio	Modeled Area	Fluid Condition	Parameter to be scaled
4.0 CORE MODELING							
4.1 Nodalization	Sufficiently small mesh sizes in both axial and lateral directions are ensured.	Sensitivity studies show a negligible effect when decreasing the mesh size from that used in actual plant analysis.	-				
4.2 Turbulent Mixing	The mixing effect is minimized in the plant analysis. - The mixing parameter is given based on the 95/95 lower limit obtained from the experimental result. - Core modeling does not allow turbulent thermal mixing between fuel assemblies.	A sensitivity study shows a negligible effect of different mixing parameters in actual plant analysis.	Rod bundle mixing test (Ref. 2)	1/1	5x5 rod bundle 12ft partial length	PWR condition	Subchannel dimension and grid spacing (both are in full scale)
4.3 Hydraulic Resistance	Well accepted correlation is used.	A sensitivity study shows a negligible effect of coefficients for the correlation in actual plant analysis.	-(*)				
4.4 Two-Phase Flow Model	Conservative model is selected from the models incorporated in the approved VIPRE-01 code.	A sensitivity study shows the small effect on minimum DNBR values in actual plant analysis.	-				
4.5 Engineering Factors	Fuel fabrication tolerances on pellet weight and enrichment are taken into account based on 95/95 basis.	95/95 basis uncertainty will be ensured in the fuel fabrication.	-				

(*)For the mechanical design, the total fuel assembly pressure drop was measured in the full scale/full length fuel assembly hydraulic test. (Ref. 7)

Phenomena/ Models	Design Consideration	Qualification	Supporting Experiment	Scale Ratio	Modeled Area	Fluid Condition	Parameter to be scaled
4.6 Core Inlet Flow Distribution	The flow test shows that maximum core inlet flow rate reduction is comparable to 10%. A 10% inlet flow rate reduction is assumed for the hot assembly in the plant analysis.	A sensitivity study shows a negligible effect in actual plant analysis.	Downcomer/ lower plenum hydraulic test (Ref. 3)	1/7	R/V inlet ~ core inlet	Room temperature	Reynolds Number
4.7 Boundary Conditions	MARVEL-generated transient conditions are used for core power, system pressure, core inlet temperature, and core inlet flow.	Transient analysis mode using MARVEL is reviewed in Non-LOCA Topical Report (Ref. 4).	-				
4.8 Calculation Control Parameters	Time step size is maintained so that the Courant number is greater than 1.	It complies with the NRC condition in the VIPRE-01 generic SE.	-				
5.0 DNB Correlation	DNBR limit for the safety analysis accounts for the uncertainty of DNB correlation based on applicable DNB test data.	DNBR limit covers 95/95 uncertainty based on the full scale DNB test results.	Rod bundle DNB test (Ref. 5 and 6)	1/1	5x5 rod bundle 12-ft partial and 14-ft full length	PWR condition	Subchannel dimension and grid spacing (both are in full scale)
6.0 Transient Fuel Rod Modeling							
6.1 Nodalization	Sufficiently small mesh size in radial direction is ensured.	Sensitivity studies show that reducing the mesh size used in actual plant analysis has negligible effects.	-				
6.2 Thermal Properties	Properties consistent to FINE code are incorporated.	Steady state analysis shows good agreement with FINE results.	-				
6.3 Power Distribution	Radial power distribution from FINE results is used as an input.		-				

Phenomena/ Models	Design Consideration	Qualification	Supporting Experiment	Scale Ratio	Modeled Area	Fluid Condition	Parameter to be scaled
6.4 Gap Conductance	Steady state gap conductance is specified [ ]. For transients, the model is selected [ ] give conservative results.	Steady state analysis shows good agreement with FINE results. The gap conductance for transient analysis is reviewed in Non-LOCA Topical Report (Ref. 4).	-				
6.5 Heat Transfer Coefficient	For post-CHF PCT analysis, [ ].	The PCT analysis results show good agreement with the results of approved evaluation model using FACTRAN code, which includes highly conservative assumptions.	-				
6.6 Zr-Water Reaction	Approved Baker-Just correlation is used in PCT analysis.		-				