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TOKYO, JAPAN

July 05, 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-13156

Subject: MHI's Amended Response to US-APWR DCD RAI No.994-7007 (SRP 04.04)

References: 1) "Request for Additional Information No. 994-7007, SRP Section: 04.04 – Thermal and Hydraulic Design" dated February 21, 2013.
2) Letter from Y. Ogata (MHI) to J. A. Ciocco (NRC), UAP-HF-13065, "MHI's Response to US-APWR DCD RAI No.994-7007 (SRP 04.04)," dated March 19, 2013.

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "MHI's Amended Response to US-APWR DCD RAI No. 994-7007 (SRP 04.04)."

Enclosed is an amended response to the question contained in Reference 1. The original response has been transmitted with Reference 2.

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,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, LTD.

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Enclosures:

1. Affidavit of Yoshiki Ogata

2. Amended Response to US-APWR DCD RAI No.994-7007 (Proprietary version)

3. Amended Response to US-APWR DCD RAI No.994-7007 (Non-proprietary version)

CC: J. A. Ciocco
J. Tapia

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ENCLOSURE1

Docket No. 52-021
MHI Ref: UAP-HF-13156

MITSUBISHI HEAVY INDUSTRIES, LTD.
AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am Executive Vice President of Mitsubishi Nuclear Energy Systems, Inc. and have been delegated the function of reviewing Mitsubishi Heavy Industries, LTD's ("MHI") 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 "Amended Response to US-APWR DCD RAI No.994-7007" dated June 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 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 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 5th day of July, 2013.

Y. Ogata

Yoshiki Ogata,
Executive Vice President
Mitsubishi Nuclear Energy Systems, Inc.
On behalf of Mitsubishi Heavy Industries, LTD.

Enclosure 3

UAP-HF-13156
Docket No. 52-021

Amended Response to US-APWR DCD RAI No.994-7007

July 2013
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

07/04/2013

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 994-7007**
SRP SECTION:	**04.04 – THERMAL AND HYDRAULIC DESIGN**
APPLICATION SECTION:	
DATE OF RAI ISSUE:	**2/21/2013**

QUESTION NO.: 04.04-43

In the VIPRE-01M Topical Report MUAP-07009 the applicant requests a DNBR correlation limit of 1.17 for fuel with the Z2 and Z3 grid design for both the WRB-1 and WRB-2 CHF correlations. This limit of 1.17 the previously approved value for the WRB-1 and WRB-2 CHF correlations and in their draft SER for the VIPRE-01M topical, the staff concluded that this value was conservative for MHI fuels. The DNBR correlation limit is used to bound the 95/95 statistic the measured-to-predicted CHF values and is obtained by determining the 95/95 statistic and adding a small conservative bias. The 95/95 statistic is obtained from the mean and standard deviation of the measured-to-predicted CHF values.

In instances where MHI would use the correlation limit, it would be expected that the value of 1.17 would be used. However, the staff is aware that in the previously approved RTDP methodology the DNBR correlation limit is not used. Instead, the mean and standard deviation of the measured-to-predicted data are combined with other uncertainties to obtain a total DNBR limit. While this methodology is approved, the RTDP methodology does not change the approved DNBR correlation limit. Simply using the mean and standard deviation of the measured-to-predicted data ignores this previously approved bias.

Demonstrate that the mean and standard deviation of the measured-to-predicted data which is used in the RTDP methodology will produce a 95/95 statistic which is equal or conservative compared to the approved DNBR correlation limit of 1.17 for both the WRB-1 and WRB-2 CHF correlations.

ANSWER:

The design basis for MHI core thermal design is to prevent a core from experiencing departure from nucleate boiling (DNB) during normal operation and anticipated operational occurrences (AOOs). The 95/95 basis is applied for dealing with the DNB correlation uncertainty and any input parameter uncertainties associated with the analyzed plant conditions.

Two types of DNBR limits are used in the US-APWR analyses, in which the uncertainties are treated in a different manner. One is Correlation Limit (CL), which includes only the correlation uncertainty and is used for analyses that use Standard Thermal Design Procedure (STDP). The input parameter uncertainties are accounted for in conservatively determined input values. The other is Design Limit (DL), which includes both correlation and input parameter uncertainties and is used for Revised Thermal Design Procedure (RTDP) (Reference 1).

The CL is determined based on the 95/95 statistics from the DNB test data analysis by using the following equation:

$$CL = \frac{1}{m_{M/P} - k_{95/95} s_{M/P}} \qquad (1)$$

where,

M/P:	Measured-to-Predicted DNB heat flux ratio
$m_{M/P}$:	Sample mean value of M/P
$s_{M/P}$:	Sample standard deviation of M/P
$k_{95/95}$:	Owen's 95/95 factor.

Since the CL is uniquely determined for each combination of DNB correlation and corresponding fuel type (DNB test data set), it is common practice for the CL to be submitted for NRC approval in topical reports prior to plant application.

The DL is derived from the same 95/95 statistics as those used to determine the CL:

[ ] (2)

where,

[ ] (3)

[ ] (4)

μ_y, σ_y: Mean and standard deviation of DNBR considering the input parameter uncertainties via sensitivity analysis.

$\mu_{M/P}$, $\sigma_{M/P}$: Mean and standard deviation of M/P population associated with the 95/95 basis.

Since the DL includes input parameter uncertainties which are associated with plant-specific conditions, it is submitted for NRC approval on a plant-by-plant basis.

Thus, the CL and DL values are based on the same M/P statistics and 95/95 basis, while they are obtained through different equations and are reviewed in different processes.

In the Topical Report MUAP-07009 (Reference 4), MHI presented the DNB test data analyses for the original WRB-1 and WRB-2 databases using the VIPRE-01M code. The results show that the MHI-evaluated CLs are slightly lower than the original CL of 1.17, which was derived by Westinghouse using the THINC code for both correlations.

CL for WRB-1 (Reference 4)

(5)

CL for WRB-2 (Reference 4)

(6)

When STDP is applicable, a CL of 1.17 is used as stated in the Topical Report (Reference 4), which conservatively meets the 95/95 design basis as shown in equation (5) and (6). When RTDP is applicable, the DL is derived for the US-APWR design from Equation (2) using the 95/95 statistics in Equation (6), as shown in Reference 5. This approach is identical to the NRC approved method previously used by Westinghouse. In addition, the M/P statistics set from Equation (6) provide the most conservative result among the numbers directly derived for the WRB-2 and VIPRE-01M combination from existing DNB test databases including the DNB test data for the US-APWR fuel presented in the supplemental test report (Reference 6). The resulting DLs of 1.35 and 1.33 are obtained for typical cell and thimble cell, respectively, for US-APWR analysis.

As described above, the DL for the US-APWR and the CL in the topical report MUAP-07009 are both derived from the same 95/95 statistics and meet the required design basis. MHI believes that both DNBR limits are conservatively applicable to the US-APWR fuel design. However, in order to address the NRC concern, MHI agrees to add conservative bias to the statistics used for the US-APWR RTDP analysis so that the statistics are equivalent to a CL of 1.17.

The new DL is derived [

]

(7)

[

]

(8)

The resulting DLs using Equation (2) are 1.37 and 1.35 for typical cell and thimble cell, respectively.

The safety analysis limit (SAL) of 1.45 remains the same because sufficient margin is available between the DLs and SAL when considering the rod bow penalty of 1%.

References:

1. A. J. Friedland, S. Ray, "Revised Thermal Design Procedure," WCAP-11397-P-A, 1989.
2 F. E. Motley, et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, 1984.
3. Edited by S. L. Davidson, "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, 1985.
4. Y. Makino, et al., "Thermal Design Methodology," MUAP-07009, 2007.
5. Letter from Y. Ogata (MHI) to J. Ciocco (NRC), "MHI's Amended Response to US-APWR DCD RAI No.377-2629 Revision 1," UAP-HF-09546, December 2, 2009.
6. M. Tatematsu, et al., "DNB Tests for US-APWR Fuel," MUAP-11010 Rev. 2, July 2012.

Impact on DCD

The DL values described in Chapter 4 and Chapter 16 will be changed. DCD mark-ups are attached.

Impact on COLA

There is no impact on the COLA

Impact on PRA

There is no impact on the PRA

Impact on Technical/Topical Report

There is no impact on Technical/Topical reports.

Table 4.1-1 Comparison of Principal Reactor Design Parameters (Sheet 2 of 3)

Parameter	US-APWR	Typical 12-ft 4-loop PWR (Ref. 4.1-3)	Typical 14-ft 4-loop PWR (Ref. 4.1-4)
Fuel pellet material	Sintered UO_2 Sintered $(U,Gd)O_2$	Sintered UO_2	Sintered UO_2
Fuel pellet diameter (in)	0.322	0.3088	0.3225
Fuel pellet density (%TD)	97	95	95
Number of grids per assembly	11	8	10
Fuel pellet length (in) Blanket pellet length (in)	0.453	0.370 0.462/0.500	0.387 (0.462)[(a)]
Rod Cluster Control Assemblies			
Neutron absorber material	Ag-In-Cd	Ag-In-Cd or Hafnium	Hafnium or Ag-In-Cd
Number of clusters	69	53	57
Number of absorber rods per cluster	24	24	24
Absorber diameter (in)	0.341	0.341	0.366
Cladding material thickness, for Ag-In-Cd (in)	Type 304 SS 0.0185	Type 304 SS 0.0185	Type 304 SS 0.0185
Key Core Design Limits & Conditions			
Total heat flux hot channel factor, F_Q	2.60	2.50	2.70
Fraction of heat generated in the fuel (%)	97.4	97.4	97.4
Maximum fuel centerline temperature during AOOs (°F)	≤4620	≤4700	≤4700
Maximum peak linear heat rate during AOOs[(b),(c)] (kW/ft)	≤21.9 (assuming overpower of 120%)	≤22.4 (assuming overpower of 120%)	≤22.0 (assuming overpower of 118%)
Minimum DNBR during AOOs Typical channel Cold wall (thimble) channel	≥1.3~~6~~7 ≥1.3~~3~~5	≥1.24 ≥1.23	≥1.26 ≥1.24
Correlation used for above DNBR values	WRB-2	WRB-2	WRB-1

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The critical condition for DNB occurrence can be characterized by surface heat flux. The departure from nucleate boiling ratio (DNBR), the ratio of predicted DNB heat flux to actual local heat flux as defined in Subsection 4.4.2.2.1, is used to express the margin to the point of DNB occurrence.

To predict DNB heat flux for the US-APWR fuel design, the WRB-2 DNB correlation as described in Subsection 4.4.2.2.1 is adopted. The local coolant conditions utilized by the WRB-2 correlation are provided by the VIPRE-01M code as described in Subsection 4.4.2.2.1. The compatibility of WRB-2 with VIPRE-01M for the Mitsubishi fuel design has been verified, as described in Reference 4.4-2.

The uncertainties of several parameters that affect DNBR, such as those associated with plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions, are considered statistically in the revised thermal design procedure (RTDP, Reference.4.4-3) to obtain design limits of the minimum DNBR value in the core (Min. DNBR) described in Subsection 4.4.2.2.1. With the uncertainties, the design limits of Min. DNBR are determined such that there is at least a 95-percent probability at a 95-percent confidence level that the hot fuel rod in the core does not experience a DNB.

The input parameters to the VIPRE-01M code are adopted at their nominal values. The Min. DNBR is maintained above the design limit during normal operation and AOOs.

The design limits of Min. DNBR for the US-APWR that are obtained by the VIPRE-01M code and the WRB-2 DNB correlation are ≥1.3~~5~~7 for a typical channel and ≥1.3~~3~~7 for a thimble channel, respectively. Those values were determined based on sensitivity analyses for US-APWR core conditions and uncertainties with VIPRE-01M and WRB-2. DCD_04.04-43

The safety analysis limit of Min. DNBR is determined as 1.45 for both the channel types, accommodating the DNBR penalties incurred due to rod bows described in Subsection 4.4.2.2.4 and transition core geometry, and/or reserving more core operational flexibilities.

For the analyses where the RTDP is not applicable, all the uncertainties except DNB correlation uncertainty are deterministically taken into account. The input parameters to VIPRE-01M are applied in a conservative way. The DNBR limit covers the DNB correlation uncertainty, and also provides necessary margin to offset the DNBR penalties, if needed.

4.4.1.2 Fuel Temperature

4.4.1.2.1 Design Basis

There is at least a 95-percent probability at a 95-percent confidence level that the fuel rod with the most limiting linear heat rate (kW/ft) does not cause the fuel pellet to melt during normal operation and AOOs.

Table 4.4-1 Thermal-Hydraulic Comparison between US-APWR and Other Designs (Sheet 2 of 2)

Design Parameters	US-APWR	Typical 12-ft 4-loop PWR (Ref. 4.4-20)	Typical 14-ft 4-loop PWR (Ref. 4.4-21)
Local peak	12.1 [(d)]	14.2	14.0
Power density [(e)] (kW/l)	89.2	109.2	98.8
Specific power (kW/kg uranium)	32.0 [(f)]	42.5	36.4
Minimum DNBR at nominal condition			
Typical hot channel	2.05	2.47	2.19
Thimble hot channel	1.98	2.33	2.11
Minimum DNBR during AOOs			
Typical hot channel	≥1.3~~5~~7	≥1.24	≥1.26
Thimble hot channel	≥1.3~~3~~5	≥1.23	≥1.24
Correlation used for above DNBR values	WRB-2	WRB-2	WRB-1
Maximum peak linear heat rate during AOOs [(c)] [(g)] (kW/ft)	≤ 21.9 (assuming overpower of 120%)	≤ 22.4 (assuming overpower of 120%)	≤22.0 (assuming overpower of 118%)
Maximum fuel centerline temperature during AOOs (°F)	≤ 4,620	≤ 4,700	≤ 4,700
Pressure drop [(h)] (psi)			
Across core	32.1±3.2	25.8±2.6	39.78±4.0
Across RV	48.2±4.8	48.5±4.9	62.68±8.9

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Notes:

(a) Based on thermal design flow and design core bypass flow (9.0%)

(b) Based on average enthalpy

(c) Based on densified active heated length

(d) Based on heat flux hot channel factor, F_Q = 2.60

(e) Based on cold dimensions

(f) Based on 97% of theoretical density fuel

(g) See Subsection 4.4.2.11.5

(h) Based on best estimate flow rate

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained
≥ 1.3~~5~~7 for typical hot channel
≥ 1.3~~3~~5 for thimble hot channel
with WRB-2 DNB correlation and revised thermal design procedure (RTDP) .

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2.1.1.2 The peak fuel centerline temperature shall be maintained < 5072°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2733.5 psig.

2.2 SAFETY LIMIT VIOLATIONS

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.