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

September 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-09455

Subject: **Response to the NRC Request for Additional Information on "Thermal Design Methodology", MUAP-07009 Rev. 0**

Reference: 1) "Request for Additional Information Topical Report Thermal Design Methodology MAUP-07009 Rev. 0, dated August 20, 2009

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") a document entitled "Response to the NRC Request for Additional Information on topical report: "Thermal Design Methodology", MUAP-07009 Rev. 0".

Enclosed are the responses to 23 RAIs out of 40 contained within Reference 1. Additional proprietary supporting material for the RAI response is provided on an Optical Storage Medium ("OSM"). The OSM contains VIPRE-01M input data. The specific file contained on the OSM is listed on the associated enclosure cover sheet.

As indicated in the enclosed materials, the documents contain 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 non-proprietary version (Enclosure 3), additional proprietary VIPRE-01M supporting documentation provided on an OSM (Enclosure 4), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 and 4 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.

D081
NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. Response to the NRC Request for Additional Information on "Thermal Design Methodology", MUAP-07009 Rev. 0 (proprietary version)

3. Response to the NRC Request for Additional Information on "Thermal Design Methodology", MUAP-07009 Rev. 0 (non-proprietary version)

4. OSM: Supporting material: VIPRE -01M input file (proprietary)

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-09455

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, being duly sworn according to law, deposes and states 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 "Response to the NRC Request for Additional Information on "Thermal Design Methodology", MUAP-07009 Rev. 0" and the enclosed Optical Storage Medium (OSM), both dated September 30, 2009, and have determined that the documents 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). The OSM contains a proprietary VIPRE-01M input file, Enclosure 4 File 1. The label of the OSM has been marked "Proprietary" to indicate that the entire content of the OSM should be withheld from public disclosure pursuant to 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 thermal and hydraulic design developed by MHI and not being used in the exact form by any 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 the development of the thermal and hydraulic 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 and hydraulic 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 the 30th day of September, 2009.

Y. Ogata

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

Enclosure 3

UAP-HF-09455, Rev.0

Response to the NRC Request for Additional Information on "Thermal Design Methodology", MAUP-07009 Rev. 0

September 2009
(Non Proprietary)

Response to the NRC Request for Additional Information on

"THERMAL DESIGN METHODOLOGY", MUAP-07009-P Rev.0

1.0 Requests for Additional Information on the Evaluation Model

1.1 Identify the specific accident/transient scenarios and plant configurations for each application in which VIPRE-01M will be used. Include the parameters of interest, figures of merit, and limiting transients.

Response:

This will be responded to within 75 days.

1.2 For the scenarios identified in RAI 1.1 (above), provide justification that the specific accident/transient scenarios can be adequately modeled with VIPRE-01M. This means that models must be present in the VIPRE-01M code to capture the phenomena and components that have been determined to be important or necessary to simulate the accident/transient under consideration. The chosen mathematical models and numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view.

Response:

This will be responded to within 75 days.

1.3 Provide the following listed in Chapter 9.0 of MUAP-07009-P to the NRC: References 1,2,3,4,5, and 18.

Response:

References 1, 2, 3, 4, and 5 are documents published by Electric Power Research Institute (EPRI). Due to MHI's license agreement with EPRI, we are not allowed to provide these documents directly to the NRC. MHI has contacted EPRI, and they suggest that the NRC contact them directly for obtaining the documents.

Reference 18 is an internal document, and it can be replaced by the Topical Report PQD-HD-19005 Revision 3 "Quality Assurance Program (QAP) Description For Design Certification of the US-APWR". PQD-HD-19005 Revision 3 has been submitted to the NRC by MHI letter UAP-HF-09445, dated September 14, 2009. MHI will revise the description for this reference in MUAP-07009-P.

1.4 Provide an overview of the thermal design methodology which provides a clear roadmap describing all parts of the thermal design methodology and the relationships between the separate parts.

In this overview, describe where VIPRE-01M receives its inputs, which codes interact with VIPRE-01M (provides input to VIPRE-01M or receives output from VIPRE-01M), and what is done with the output from VIPRE-01M. References should be supplied for any interacting codes, as well as verification that the codes are approved by the NRC and are being used within their conditions and limitations.

Response:

This will be responded to within 75 days.

1.5 *Identify which MOD of VIPRE-01 was used to create VIPRE-01M and which revision of documentation was used. If something other than MOD-01 or MOD-02 were used (for example, MOD-2.1) MHI may need to provide additional justification because even a "small" change to the evaluation model can have unintended consequences on calculation results that were through to not be impacted by the changes.*

Response:

NRC issued Safety Evaluations for EPRI VIPRE-01 versions MOD-1.0 and MOD-2.0. MHI's VIPRE-01M code was created from MOD-2.2.1 of the EPRI VIPRE-01 code. MOD-2.2.1 was the latest version of VIPRE-01 when MHI obtained it from Electric Power Research Institute (EPRI) and Computer Simulation & Analysis, Inc. (CSA) in 2006.

VIPRE-01, MOD-2.2.1 incorporated fixes to minor editorial errors which were found after MOD-2.0 was approved by the NRC and some features for user convenience. All of the modifications are administered by EPRI/CSA according to their Quality Assurance Program, and it was confirmed by CSA that the modifications did not impact the calculation results.

MHI uses two features added between MOD 2.0 and MOD 2.2.1. They have been verified as follows:

[ ]

- Model to read RETRAN-3D generated boundary condition file

This model was modified so as to read the boundary condition file generated by MHI MARVEL-M code. This feature has been verified internally and in the response to RAI 3.10.

1.6 The NRC generic safety evaluation report for VIPRE-01 MOD-01 includes five conditions for use of the code. The NRC generic safety evaluation report for VIPRE-01 MOD-02 includes four additional conditions for use of the code. Explicitly state the SE conditions and discuss how MHI will meet each of these conditions using VIPRE-01M.

Response:

MHI is in compliance with the five conditions required in the generic safety evaluation for VIPRE-01 MOD-1.0 as described in Section 3.2 of MUAP-07009-P.

The safety evaluation for VIPRE-01 MOD-2.0 includes four additional conditions. The first two of them are for the BWR applications, and therefore, are not applicable to the PWR design applications described in MUAP-07009-P. MHI meets the last two conditions as follows:

Condition 3

"Section 2.2 of Volume 5 of the submittal identifies a spectrum of limitations of the code. Each user, should ensure that the code is not being used in violation of these limitations."

MHI compliance

In summary, Section 2.2 of Volume 5 of NP-2511-CCN-A describes that the VIPRE-01 code cannot be used for the following situations:

- Specific two phase flow conditions that are characterized by large relative velocity between the phases or radical changes in flow regime, such as low-flow boil-off, annular flow, stratified two phase flow, or countercurrent flow.
- Phenomena dominated by local pressure such as blow-down transient, boiling inception at low pressure, or BWR transient flow instability.
- Free-field situation not dominated by wall friction
- Out of the applicable range of the constitutive correlations

MHI uses VIPRE-01M for the non-LOCA thermal-hydraulic analyses for the PWR core. Those applications do not involve phenomena that are dominated by large relative velocity between liquid and vapor phase and/or local pressure effects as mentioned above. The flow in the PWR core is highly dominated by the axial rod friction.

The correlations used in MHI VIPRE-01M analyses are described in MUAP-07009-P and are justified for the non-LOCA thermal-hydraulic analyses of the PWR core.

Condition 4

"By acceptance of this code version, we do not necessarily endorse procedures and uses of this code described in Volume 5 as appropriate for licensing applications. As the code developer stated in Reference 5, the materials were provided by the code developers as their non-binding advice on efficient use of the code.

Each user is advised to note that values of input recommended by the code developers are for best-estimate use only and do not necessarily incorporate the conservatism appropriate for licensing type analysis. Therefore, the user is expected to justify or qualify input selections for licensing applications."

MHI compliance

MHI selected conservative input options of VIPRE-01M for the licensing applications. Those were described in MUAP-07009-P and justified to be conservative and in compliance with the safety evaluation for VIPRE-01 MOD-1.0.

1.7 *MHI should provide an assessment of the differences between Pressurized Water Reactors which the NRC staff reviewers are accustomed to dealing with and which the VIPRE-01 computer code was licensed for, and MHI-designed Pressurized Water Reactors. MHI should provide justification as the applicability of VIPRE-01 (and subsequently VIPRE-01M) to MHI-designed Pressurized Water Reactors.*

Response:

The MHI-designed PWR (US-APWR) has plant configurations essentially identical to those of typical 4-loop PWRs currently in operation in the United States.

The core thermal-hydraulic parameters shown in Table 1.7-1 demonstrate the similarity between them. It is obvious that the core thermal hydraulic characteristics of the US-APWR are not different from those of the PWR designs that the NRC staff reviewers are accustomed to dealing with.

The VIPRE-01 code has various models and correlations presented for use as options, and the core thermal hydraulics of the typical PWRs can be modeled using appropriate options as approved by the NRC. VIPRE-01M is the MHI version of VIPRE-01 in which no substantial modifications that impact the calculations have been made.

For the above mentioned reasons, the use of VIPRE-01M to model the US-APWR core thermal hydraulics is justified.

Table 1.7-1 Comparison of Design Parameters

Design Parameters	US-APWR	Typical 12-ft 4-loop PWR [Ref. 1.7-1]	Typical 14-ft 4-loop PWR [Ref. 1.7-2]
Core thermal output (MWt)	4,451	3,565	3,853
System pressure (psia)	2,250	2,250	2,250
Thermal design flow rate (10^6lbm/hr)	168.2	139.4	145.0
Core average coolant mass velocity (10^6lbm/hr-ft^2)	2.25	2.41	2.59
Coolant Temperature			
RV/core inlet (°F)	550.6	556.8	549.8 to 561.2
Average rise in RV (°F)	66.4	63.2	63.6 to 65.0
Heat flux (10^6Btu/hr-ft^2)			
Core average	0.162	0.206	0.181
Local peak	0.421	0.515	0.489
Minimum DNBR at nominal condition			
Typical hot channel	2.05	2.47	2.19
Thimble hot channel	1.98	2.33	2.11

References

1.7-1 Vogtle Electric Generating Plant, Updated Final Safety Analysis Report Revision 12, November 5, 2004.

1.7-2 South Texas Project Electric Generating Station (STPEGS) Units 1 and 2, Updated Final Safety Analysis Report (UFSAR), Revision 12, September, 2004.

1.8 Provide a discussion on rod bow. Include how rod bow will be accounted for (especially in any thermal limits) and the basis for that decision.

Response:

This will be responded to within 75 days.

1.9 Provide a discussion on transition cores. Include how the transition cores will be accounted for (especially in any thermal limits) and the basis for that decision.

Response:

This will be responded to within 75 days.

1.10 Provide a discussion and data or analysis on the selection of the constant ABETA, specifically, what is the physical justification for the constant's value?

Response:

This will be responded to within 75 days.

1.11 Provide a discussion and data or analysis on the selection of the form loss coefficients for grid spacers specifically, verify that the form loss coefficient is applicable to the fuel assembly which will be put in the MHI-designed Pressurized Water Reactors.

Response:

This will be responded to within 75 days.

1.12 Which radial noding scheme will be used to perform licensing analysis on MHI-designed Pressurized Water Reactors?

Response:

As described in Subsection 4.1 of MUAP-07009-P, MHI intends to use the radial noding scheme presented in Figure 4-1 for licensing analysis. In this scheme, the fine noding scheme is adopted around the hot sub-channels and the peripheral core region is modeled by large lumped channels.

The various noding schemes presented in Appendix A of MUAP-07009-P are used

for the sensitivity studies to justify the use of the noding scheme presented in Figure 4-1.

1.13 *Provide a discussion on the mode which will be used in VIPRE-01M (UPFLOW or RECIRC). Discuss any associated limitations, and why they will be consistent with the accident/transient scenarios modeled with VIPRE-01M in MHI-designed Pressurized Water Reactors. Specifically which input conditions are needed and where the input conditions will be obtained from.*

Response:

The VIPRE-01M code has two computational modes; UPFLOW and RECIRC. These two modes provide the same solution results because they solve the identical discretized field equations. The Safety Evaluation for the VIPRE-01 code indicated that both of them are applicable for the PWR analysis. MHI uses the RECIRC mode for typical design analysis.

Since the RECIRC mode does not have any computational restrictions for the applied flow conditions for which the UPFLOW mode has, it is applicable for all the accident/transient scenarios of the MHI-designed PWR without any specific consideration for the input conditions.

1.14 *What values of the core inlet distribution factors will be used to perform licensing analysis on MHI-designed Pressurized Water Reactors?*

Response:

In MUAP-07009-P, it is stated that the inlet mass velocity of the hot assembly could be approximately 5 to 10% lower than the average core inlet mass velocity. For the US-APWR design analysis, a 10% reduction for the hot assembly is assumed.

In the past, as a conservative engineering practice, MHI had assumed that the hot assembly inlet flow is 5% lower than the core average in DNB analysis for the conventional type PWR reactor in Japan. For the recent plant design, including the US-APWR, 10% reduction is assumed for the practice.

As a matter of fact, various subchannel analyses all indicated that minimum DNBR is really insensitive to such a prescribed inlet flow distribution. Sensitivity studies in Appendix A of MUAP-07009-P show that the effect of the hot assembly inlet flow rate on the minimum DNBR is negligibly small even if it is 20% lower than the core

average flow rate, because the flow redistribution in the lower core swiftly flattens out the channel inlet mass flux disparity.

1.15 How many radial nodes are used in the clad for both steady state and transient applications? Why was this number chosen.

Response:

Two radial nodes are used in the cladding for VIPRE-01 fuel rod analyses. This noding setup is fixed in the code by the original code developers, and the user cannot change it.

We believe that this model is acceptable, because the cladding is extremely thin and the temperature distribution inside the fuel cladding can be reasonably approximated by a linear distribution.

1.16 Provide a further description for the phrase 'initial pellet heat up'.

Response:

The term "initial pellet heat up" in Subsection 6.1 of MUAP-07009-P means the initial temperature or enthalpy of the pellet used in the transient analyses. In many safety analyses cases, the reactor will be tripped at an early stage of the transient. After that, the fuel releases the energy stored in it. Therefore, the "initial pellet heat up" is an important condition to predict the thermal behavior of the fuel rods during the transient relevant to DNB and fuel temperature.

1.17 Describe the axial nodalization for transient analyses.

Response:

The VIPRE-01 transient fuel rod model is applied at each fluid node to provide the heat input to the fluid. The axial nodalization for the fuel rod transient analysis is the same as the one for the fluid analysis described in Subsection 4.1 of MUAP-07009-P. That is, the channels are divided at the axial mesh size less than

[ ], which keeps the Courant number greater than 1 under the condition of [ ].

1.18 Confirm that the thermal properties for the fuel have been previously reviewed and approved the NRC.

Response:

The fuel thermal properties used in the VIPRE-01M analyses are described in Appendix D of MUAP-07009-P.

The properties described here are basically the same as those reviewed in WCAP-8301 "LOCTA-IV Program: Loss-of-coolant Transient Analysis", except for the thermal conductivity of the fuel pellet and the heat capacity of ZIRLO™.

The thermal conductivity of the fuel pellet is the same as that in the FINE code, which is described in MUAP-07008-P "Fuel System Design Criteria and Methodology" and is under review by the NRC.

ZIRLO™ properties are based on WCAP-12610 "VANTAGE+ Fuel Assembly Reference Core Report", which has been reviewed by the NRC. The properties are mostly the same as those of Zircaloy-4, except the phase transformation temperature in the heat capacity shows a small difference from that of Zircaloy-4.

1.19 Provide a discussion identifying each accident/transient scenario and how the gap conductance will be conservative for each of the scenarios modeled with VIPRE-01M.

Response:

The following discussion illustrates the effect of gap conductance varying from a lower-than-expected value to a higher-than-expected value for modeled scenarios. If the gap conductance is reduced from its normal value, the overall heat flux from the fuel rod to the coolant channel is reduced which will result in a higher fuel centerline temperature. On the other hand, if the gap conductance is increased from its normal value, the heat flux will increase which will result in a lower fuel centerline temperature.

Over-predicting the overall heat flux is conservative in calculating minimum DNB ratios and over-predicting gap heat flux is conservative in calculating peak cladding

temperature. Under-predicting the gap heat flux will result in a conservative peak fuel temperature calculation. This illustrates how the selection of a gap conductance model could result in different conservative predictions.

Table 1.19-1 below shows the actual gap conductance behavior during the transient and the analytical assumption used in the safety analysis only for events that use transient VIPRE-01M.

The actual gap conductance behavior in each of the transients is as follows:

- For the partial and complete loss of forced coolant flow events, no DNB is expected to occur and thus the cladding temperature will stay almost constant. The core thermal power and pellet temperature will be decreasing due to the reactor trip; therefore, the pellet-cladding gap width will increase. As a result, for those transients, the gap conductance will actually decrease.

- The RCP rotor seizure event is almost the same as above. When the DNB occurs, the cladding temperature will increase and the pellet-cladding gap width will increase by a larger amount. Then in this transient, the gap conductance will actually decrease.

- For the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition event and the spectrum of rod ejection accidents, the pellet-cladding gap width will decrease due to the rapid core power increase and pellet temperature increase. Then for those transients, the gap conductance will actually increase.

The gap conductance between the fuel pellet and the cladding for each transient is assumed to be conservative according to the purpose of the evaluation. The initial condition of the fuel temperature at the hot spot is consistent with the result of the fuel design code FINE:

- Remains constant after event initiation for DNBR analyses in the case of the partial and complete loss of forced coolant flow and RCP rotor seizure events

- Remains constant (at the maximum value) after event initiation for DNBR analyses in the case of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition event

- Remains constant after event initiation for fuel temperature and enthalpy analyses in the case of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition event and spectrum of rod ejection accidents

- Instantaneously decreases to zero for the adiabatic fuel enthalpy analysis in the case of the spectrum of rod ejection accidents

- Rapidly increases to the maximum value for the cladding temperature analyses in the case of the RCP rotor seizure event and spectrum of rod ejection accidents

- Realistic* increases for the rods in DNB and the RCS pressure analyses in the case of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition event and spectrum of rod ejection accidents

* The realistic gap conductance model is justified in the response to Question No.15.4.8-8 of US-APWR DCD RAI 313-2361, which was submitted by MHI letter UAP-HF-09346, dated July 3, 2009. In this analysis, which has a large power distortion during transient, a large safety margin is included in the power distribution assumption

Table 1.19-1 Actual Gap Conductance (Hgap) Behavior during Transients and Analytical Assumptions in the Chapter 15 Safety Analysis

Section	Event	Actual Hgap Behavior	Analytical Assumption
15.3.1.1	Partial loss of forced reactor coolant flow	Decrease	DNB analysis: Constant value from initial
15.3.1.2	Complete loss of forced reactor coolant flow		
15.3.3	Reactor coolant pump rotor seizure	Decrease	DNB analysis: Constant value from initial PCT analysis: Rapidly increases to the maximum value
15.4.1	Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition	Increase	Fuel temperature analysis: Constant value from initial DNB analysis: Constant value (maximum) from initial RCS pressure analysis: Realistic gap conductance
15.4.8	Spectrum of rod ejection accidents	Increase	Fuel temperature analysis: Constant value from initial Adiabatic fuel enthalpy analysis: Instantly decreases to zero PCT analysis: Rapidly increases to the maximum value Rods in DNB and RCS pressure analysis: Realistic gap conductance

1.20 Provide a discussion of how the VIPRE-01M implementation of the Zr-Water model is derived from the formulas in the source documents. Include the original continuous equations, the discrete approximation to these, the VIPRE-01M variables, the units of all physical quantities, and the energy released by the reaction. Step-by-step descriptions and all the equations must be presented in sufficient detail for replication by independent reviewers.

Response:

Reaction between Zirconium and water is modeled based on the Baker-Just correlation (Reference 1.20-1):

$$W^2 = K_1 \, exp\left(-\frac{K_2}{R'T}\right)t \tag{1}$$

where,

W: Amount of Zirconium reacted (kg/m^2)
K_1: Empirical coefficient (33.3×10^2 $(kg/m^2)^2/s$ = $140(lb_m/ft^2)^2/s$)
K_2: Activation energy (45,500 cal/mol)
R': Gas constant (1.987 cal/mol-K)
T: Temperature (K)
t: Time (s)

In the VIPRE-01M code, the growth of the oxide layer and heat generation due to the reaction are incorporated into the thermal analysis of the fuel rod cladding as follows:

(1) Growth of oxide layer

The reaction rate can be derived from Equation (1).

$$\frac{dW}{dt} = \frac{K_1}{2W} exp\left(-\frac{K_2}{R'T}\right) \tag{2}$$

Converting to British unit, and dividing by the density of Zircaloy, ρ_{Zr} =410 lb_m/ft^3, leads to:

$$\frac{dx}{dt} = \frac{C_1}{2x} exp\left[-\frac{C_2}{T(t)}\right] \tag{3}$$

where,

x: Thickness of Zircaloy reacted (ft)
C_1: Constant ($=K_1/\rho_{Zr}$ =8.33x10^{-4} ft^2/s)
C_2: Constant ($=K_2/R'$ =41,218 R)

Since the reaction rate can be strongly influenced by the temperature, it is preferable to consider the temperature variation during the time step in the discretization process. Assuming the temperature variation during the time step between $t=t_0$ and $t=t_0+\Delta t$ to be expressed by a linear function of time:

[ ] (4)

Equation (4) is a proper approximation, [ ]. Then, Δt is limited, in this model, by:

[ ] (5)

Equation (4) is substituted into Equation (3).

[ ] (6)

The bracketed term can be expanded as follows:

[ ] (7)

Substituting Equation (7) into Equation (6) gives:

[ ] (8)

Integrating Equation (8) after the separation of variables

(9)

Putting:

(10)

then, Equation (9) becomes:

(11)

(12)

Finally, x can be obtained from following equation:

(13)

(2) Heat generation

Heat generated by the above reaction is defined as follows:

$$\int_0^t q''dt = x\rho_{Zr}h_r \tag{14}$$

where,

q'': Heat generation rate per unit area (Btu/h-ft^2)

h_r: Heat of reaction per unit mass of Zircaloy reacted (Btu/lb$_m$)

The heat of reaction is obtained from Reference 1.20-1:

$$2H_2O + Zr \rightarrow H_2 + ZrO_2 + 140{,}500\,(cal/Mol - Zr) \tag{15}$$

Using the atomic weight of Zirconium, 91, h_r is conservatively estimated:

$$h_r = \frac{141 \times 10^3}{91} = 2790 \text{ (Btu/lb}_m\text{)} \tag{16}$$

Then, Equation (14) becomes:

$$\int_0^t q''dt = (410 \times 2790)x = (1.144 \times 10^6)x \tag{17}$$

Integrating Equation (17) gives the heat generation rate due to the reaction:

$$q'' = 1.144 \times 10^6 \frac{x - x_0}{\Delta t} \quad (\text{Btu/ft}^2\text{-s}) \tag{18}$$

(3) Incorporation into VIPRE-01 fuel rod model

The above models have been incorporated into the VIPRE-01M fuel rod model.

VIPRE-01M considers Zirconium-Water reaction at the outer surface of the fuel cladding, because the VIPRE-01M code is applied only to the condition without cladding rupture.

The calculation is carried out at each VIPRE-01M time step as follows:

1. If the VIPRE-01M time step size is larger than the maximum value in Equation (4), define a new time step size so as to satisfy Equation (4).
2. Calculate the reacted layer thickness, x, at the end of the (divided) time step using Equation (12) and the conditions at the previous time step.
3. Substitute x into Equation (17), and then, obtain the heat generation during the (divided) time step.
4. Repeat the above Steps 2 and 3 until the end of the time step for the VIPRE-01M calculation is reached.

Reference:

1.20-1 L. Baker Jr. and L. C. Just, "Studies of Metal-Water Reactions at High Temperatures," ANL-6548, Argonne National Laboratories, May, 1962.

1.21 What fraction of power is deposited directly in the coolant? What fraction of power is deposited directly in the clad? Provide an adequate justification for each.

Response:

In the VIPRE-01M analysis, it is assumed that 97.4% of the total power (heat energy) is generated in the fuel rods, and 2.6% is deposited directly in the coolant.

The heat energy deposition is due to several mechanisms:

- Kinetic energy of fission fragments
- Kinetic energy of newly born fast neutrons
- γ (gamma) ray energy released at time of fission
- Kinetic energy of delayed neutrons (very small contribution)
- β (beta) decay energy of fission products
- γ (gamma) ray decay energy of fission products
- Neutrinos associated with β decay energy of fission products
- Non-fission reactions due to capture of excess neutrons

Each mechanism deposits energy in various reactor and vessel components primarily in the following manner:

All fission product energy is deposited in the fuel rod, as well as β-decay energy. Neutron and γ ray energy is primarily deposited in the fuel and moderator. A small amount of the γ ray and neutron energy is also deposited in the reactor vessel, fuel components, and internals (which are recoverable), as well as shielding and other components outside of the vessel (which will not contribute to recoverable power); this represents less than 1% of total fission power. Neutrino energy is not recoverable.

Based on the above considerations, an "effective" (recoverable) average total energy from a fission event is approximately 195 MeV. As described above, most of this fission energy is released as the kinetic and decay heat energy of fission products and prompt γ energy which is captured within the fuel rods (about 190 MeV), while about 5 MeV is transmitted outside the fuel rods via γ rays and neutrons, resulting in power deposition in the coolant. Therefore it is appropriate to assume that 2.6% (=5 MeV/195 MeV) of fission energy is deposited in the coolant without heat transfer through the fuel rods.

For the direct power deposition in the fuel rod, the clad itself may also generate a very small fraction of the power due to gamma heating, as described above; however in the VIPRE-01M analysis all the heat generated inside the fuel rod is assumed to be generated in the fuel pellet. This assumption is conservative for the transient safety analyses, because it leads to a higher initial fuel temperature and consequently increases the heat release during the transient.

1.22 *Section 6.5 of MUAP-07009-P discusses a correlation that is used to calculate film boiling heat transfer. This correlation was accepted by NRC staff for use in the FRACTRAN code. Provide a list of transients and accidents for which MHI will use VIPRE-01M in the post CHF heat transfer region and discuss the licensing requirement that the calculation is designed to meet. As stated in the NRC generic SE for VIPRE-01, "Post CHF analysis aspects of the code dealing with post-CHF phenomena were excluded from this review," the use of VIPRE for this type of analysis will therefore require additional justification. Please provide justification that VIPRE can conservatively model the mechanical, physical and chemical changes that might occur in the fuel rods at elevated temperature including comparisons with applicable reactor fuel rod test data.*

Response:

This will be responded to within 75 days.

2.0 Requests for Additional Information on the Accident Scenario Identification Process

2.1 Provide a complete description of the accident/transient scenarios which will be analyzed by VIPRE-01M including plant initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident.

Response:

This will be responded to within 75 days.

3.0 Requests for Additional Information on the Code Assessment

3.1 Provide a code assessment of VIPRE-01M. Assessments performed with other versions of VIPRE-01M (such as EPRI's VIPRE-01) require additional justification because even a "small" change to the evaluation model can have unintended consequences on calculation results that were thought to not be impacted by the changes.

Response:

This will be responded to within 75 days.

3.2 Along with the code assessment of VIPRE-01M, MHI will need to provide an assessment of correct implementation of the code. Comparisons of VIPRE-01M by MHI to other NRC approved codes provide some assessment of the difference between codes, but such a comparison can not provide adequate regulatory basis to justify the evaluation model which VIPRE-01M is a part. Two possible ways to provide adequate regulatory basis for the evaluation model are VIPRE-01M comparisons to data or VIPRE-01M comparisons to analysis performed by an organization with an approved evaluation model. Such analysis should be performed to verify that VIPRE-01M accurately captures the physical phenomena of the accidents/transients of interest for MHI-designed Pressurized Water Reactors.

Response:

This will be responded to within 75 days.

3.3 Confirm that the code assessment (both the assessment already submitted and any additional assessment) was performed with a frozen version of the evaluation model that has been submitted for review?

Response:

The VIPRE-01M code assessment was performed with a frozen version of the evaluation model that has been described in MUAP-07009-P for review. The code

version is strictly controlled under the Quality Assurance Program, and can be confirmed by the printed information in the output file.

3.4 *Provide verification and documentation that as a result of the Accident Scenario Identification Process, no new accident/transient scenarios were identified that contain a physical phenomenon that was previously unimportant in the VIPRE-01 code assessment. If a new accident/transient scenario was identified, provide appropriate justification (for example, comparison to separate effects test data) for VIPRE-01M's modeling of that particular physical phenomenon.*

Response:

No new accident/transient scenarios were identified for the US-APWR that contain a physical phenomenon that was previously unimportant in the VIPRE-01 code assessment.

3.5 *Provide verification and documentation that as a result of the Accident Scenario Identification Process no new accident/transient scenarios were identified that that were not previously identified in the VIPRE-01 code assessment. If a new accident/transient scenario was identified, provide appropriate justification (for example, comparison to integral effects test data) for VIPRE-01M's modeling of that particular accident/transient scenario.*

Response:

No new accident/transient scenarios were identified for the US-APWR that were not previously identified in the VIPRE-01 code assessment.

3.6 *Provide verification and documentation that as a result of the Accident Scenario Identification Process no new accident/transient scenarios were identified which exceed the parameter range of previously identified accident/transient scenarios in VIPRE-01. If a new accident/transient scenario was identified, confirm that VIPRE-01's models can adequately model the physical phenomena in the new range.*

Response:

No new accident/transient scenarios were identified for the US-APWR which exceed the parameter range of previously identified accident/transient scenarios in VIPRE-01.

3.7 *Confirm that the code options used in the code assessment calculations will be the same as those used in plant accident calculations.*

Response:

For the fuel temperature, heat flux and DNBR, and peak cladding temperature analyses in the US-APWR DCD Chapter 15, the same code options as described in MUAP-07009-P are used.

For the spectrum of rod ejection accidents, a realistic fuel pellet-and-cladding gap conductance model is used for the rods in DNB and RCS pressure analyses. This is justified in the response to Question No. 15.4.8-8 of US-APWR DCD RAI 313-2361, which was submitted by MHI letter UAP-HF-09346, dated July 3, 2009.

3.8 *If scaling was performed, provide a scaling analysis which identifies important non-dimensional parameters related to geometry and key phenomena.*

Response:

No scaling analysis was performed in MUAP-07009-P. DNB tests were performed using the test section which represents the actual fuel assembly geometry including grid spacer type and grid spacing. Since the DNB tests described in MUAP-07009-P were conducted for 12-ft heated length which differs from the actual fuel applied for the US-APWR core (14-ft heated length), the additional DNB tests using 14-ft heated length will be conducted as proposed by MHI letter UAP-HF-09182, dated April 28, 2009. In the DNB tests, fluid conditions such as pressure, mass velocity, and inlet temperature have been carefully selected to cover the spectrum of safety analysis conditions for which DNB correlations are applied as shown in Table 3.8-1.

Table 3.8-1 Test Conditions for MHI DNB Test

	WRB-1 Applicable Range	WRB-2 Applicable Range	US-APWR Safety Analysis Conditions*	DNB Test Conditions
Pressure: P (psia)	$1440 \leq P \leq 2490$	$1440 \leq P \leq 2490$	$1860 \leq P \leq 2490$	$1440 \leq P \leq 2490$
Local mass flux: G (Mlb/hr-ft^2)	$0.9 \leq G \leq 3.7$	$0.9 \leq G \leq 3.7$	$1.3 \leq G \leq 2.0$	$0.9 \leq G \leq 3.7$
Local quality: Xe (%)	$Xe \leq 30$	$Xe \leq 30$	$Xe \leq 12$	$Xe \leq 30$

*Safety analysis conditions related to DNB analysis by WRB-2

3.9 *Confirm and provide documentation that each empirical correlation in VIPRE-01M will be used within its intended range. For the empirical correlations in VIPRE-01M provide a list of those correlations and their ranges.*

Response:

This will be responded to within 75 days.

3.10 *For codes which interact with VIPRE-01M, where applicable, provide the results of a null transient. Include comparison of the parameters of interest between the interacting codes.*

Response:

Transient calculation flow diagrams of the MARVEL-M / VIPRE-01M and the TWINKLE-M / VIPRE-01M methodology are described in Sections 5.2 and 5.3, respectively, of the Non-LOCA Methodology Topical Report (MUAP-07010-P). In both cases, an interface file, which includes time histories of the nuclear power, RCS pressure, core inlet temperature and core inlet flow rate, is used as a boundary condition in these sequence calculations.

A null transient analysis for the VIPRE-01M code is performed by using the

US-APWR nominal conditions and the following null transient boundary condition data. Ordinarily, this interface file is created by the MARVEL-M code.

Figures 3.10-1 and 3.10-2 show the results of the null transient for minimum DNBR and maximum temperature (fuel centerline, fuel average, and cladding), respectively. These results show excellent stability during the transient.

Figure 3.10-2 Maximum Fuel and Cladding Temperature vs. Time in Null Transient

4.0 Requests for Additional Information on the Uncertainty Analysis

4.1 Provide a sample uncertainty analysis evaluation for a typical plant application. This analysis should include discussion of all the engineering factors used as well as justification for their values. Include in this discussion any assumptions made about the engineering factors (such as the heat flux engineering factor) and provide appropriate justification for those assumptions.

Response:

Thermal design analyses should conservatively include uncertainties involving analysis conditions. Major uncertainties are statistically combined into the DNBR design limit of the Revised Thermal Design Procedure (RTDP).

Plant Operating Conditions

Measured plant operating condition parameters, such as reactor power, RCS pressure, RCS temperature, and RCS flow, are monitored and/or controlled for their nominal values. Their measurement uncertainty and control allowance are treated in the thermal design analysis in a conservative way.

All uncertainties and allowances used for the US-APWR design are summarized below. They are the conservatively determined bounding values based on the accumulated experience in the MHI-designed PWRs.

Controlled or Monitored Parameter	Rated Value	Uncertainty Included	Uncertainty Value
Reactor power	4451 MWt	- Calorimetric measurement uncertainty	±2.0 %
Pressurizer pressure	2250 psia	- Measurement and control system uncertainty of the pressurizer pressure	±30 psi
RV average temperature	583.8°F	- Measurement uncertainty for the hot and cold leg temperature - Dead band for rod control	±4.0°F
RCS flow	460,000 gpm (Minimum Measured Flow)	- Calorimetric measurement uncertainty - Elbow tap flow uncertainty	±2.5 %

The methodology of the detailed evaluation for these uncertainties is based on the guidance provided by ANSI/ISA-67.04.01 endorsed by Regulatory Guide 1.105 Revision 3 "Setpoints for Safety-Related Instrumentation".

Power Distribution

The radial power distribution can be characterized by the nuclear enthalpy rise hot channel factor, $F_{\Delta H}^N$, which represents the ratio of the maximum value of integral rod power to core-averaged integral rod power. The uncertainty factor of $F_{\Delta H}^N$ is defined as $F_{\Delta H}^U$. It is determined to be 1.06 for the US-APWR core based on the evaluation described in Technical Report MUAP-07021 "US-APWR Incore Power Distribution Evaluation Methodology".

Since the axial power distribution changes during an operating cycle, and its effect on DNBR is rather complicated, the design axial power distribution, which is a bounding curve of the axial power distribution, is used in DNBR analyses. While this design axial power distribution is selected to practically envelop all axial power distributions from normal operation for the DNBR calculations, the conservatism of this design power shape is to be confirmed in the first and reload core designs.

For certain transients such as steam line break and rod ejection in which the design power distribution may not be bounding, conservatively determined event specific power distributions are used.

For the analyses that are dominated by local power, such as the peak cladding temperature (PCT) determination for the postulated non-LOCA accidents, the nuclear heat flux hot channel factor, F_Q^N, is defined to represent the peak ratio of local heat flux or linear heat rate to its core average. The uncertainty factor included in F_Q^N, defined as F_Q^U, is also evaluated in the Technical Report MUAP-07021. The Technical Report demonstrates that a value of 1.08 is conservative.

Engineering Factors

The hot channel factor should include the effect of the fuel rod manufacturing tolerances for the parameters such as fuel pellet diameter, fuel pellet density, fuel pellet enrichment, and fuel cladding diameter, in addition to the nuclear effect described above.

Engineering hot channel factors, $F_{\Delta H}^E$ and F_Q^E for enthalpy rise and heat flux, respectively, are basically obtained from the statistical combination of the standard deviation of related parameters as follows:

$$F_Q^E = 1.0 + k\sigma_Q, \quad \left(\qquad \right)$$

$$F_{\Delta H}^E = 1.0 + k\sigma_{\Delta H}, \quad \left(\qquad \right)$$

where,

k : Owen's factor for 95x95 basis

σ_Q : Standard deviation of local heat flux based on manufacturing data

$\sigma_{\Delta H}$: Standard deviation of rod power based on manufacturing data

Standard deviations for the pellet are also applied to the $F_{\Delta H}^E$, which represent the maximum integrated rod power, conservatively.

Both hot channel factors, $F_{\Delta H}^E$ and F_Q^E, are defined as 1.03, which conservatively bounds the value derived by a statistical analysis of the Japanese manufactured fuel product. Typical results from the recent data are less than [ ].

The manufacturing tolerances for the core bypass flow areas are considered in the core bypass flow evaluation. The core bypass flow rate is estimated so that the pressure drop through the core bypass flow paths equals that through the main flow path. The uncertainty is determined as the difference between the evaluation of each core bypass flow rate with and without considering the uncertainty such as manufacturing tolerances of the core bypass flow area and pressure drop through the main flow path. The uncertainty of the total core bypass flow rate is evaluated as the SRSS (Square Root of the Sum of the Squares) of all core bypass flows. The uncertainty is conservatively determined to be [ ] of RCS flow for the US-APWR.

4.2 Discuss how MHI will implement the RTDP analysis, specifically addresses the uncertainties used and the basis for the uncertainties.

Response:

This will be responded to within 75 days.

5.0 Requests for Additional Information on the Theory Manual

5.1 For VIPRE-01M, provide a theory manual that is a self-contained document and that describes (a) field equations, (b) closure relationships, (c) numerical solution techniques, (d) simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods, (e) pedigree or origin of closure relationships used in the code, and (f) limits of applicability for all models in the code.

Response:

This will be responded to within 75 days.

6.0 Requests for Additional Information on the User Manual

6.1 For VIPRE-01M, provide a user manual that provides (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitation and options that should be avoided for particular accidents, components, or reactor types, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model.

Response:

This will be responded to within 75 days.

6.2 Provide the NRC with a VIPRE-01M executable (preferably PC) and the appropriate input parameters for Runs 1 – 7 on Table 7-1 of the TR such that the NRC may generate their own input deck according to the VIPRE-01M user manual, execute the code, and compare the results to results generated by MHI.

Response:

The VIPRE-01M executable for the PC has been provided by MHI letter UAP-HF-08092, dated May 30, 2008 and MHI letter UAP-HF-09434, dated August 28, 2009. Both letters include the same frozen version of the code. The input data file specified by the NRC is provided in the Optical Storage Media included with this RAI response.

7.0 Requests for Additional Information on the Quality Assurance Program

7.1 Provide the quality assurance plan for VIPRE-01M which describes the procedures and controls under which the code was developed and assessed, and the corrective action procedures that are followed when an error is discovered.

Response:

This will be responded to within 75 days.

7.2 Verify that the quality assurance plan described in response to RAI 7.1 was used when performing all changes which were required to generate VIPRE-01M from VIPRE-01 and performing all analysis submitted to the NRC. Verify that all VIPRE-01M analyses were performed with a specific frozen version of the code.

Response:

This will be responded to within 75 days.