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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-08067

Subject: Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07009-P, "Thermal Design Methodology"

With this letter, Mitsubishi Heavy Industries, LTD. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") the document entitled "Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07009-P, Thermal Design Methodology". In the enclosed document, MHI provides responses to NRC's "Request for Additional Information on US-APWR Topical Report MUAP-07009-P, Thermal Design Methodology" dated March 5, 2008.

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) and 10 C.F.R § 9.17 (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 in this package (Enclosure 3). In the non-proprietary version, the proprietary information, bracketed in the proprietary version, is replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of 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) and 10 C.F.R.§ 9.17(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.

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Enclosures:
Enclosure1 - Affidavit of Yoshiki Ogata
Enclosure2 - Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07009-P, "Thermal Design Methodology" (proprietary)
Enclosure3 - Response to NRC's Request for Additional Information on US-APWR Topical Report MUAP-07009-P, "Thermal Design Methodology" (non-proprietary)

CC: L J. Burkhart
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ENCLOSURE 1

Docket No. 52-021

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) and 10 C.F.R.§ 9.17(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 NRC's Request for Additional Information on US-APWR Topical Report MUAP-07009-P, "Thermal Design Methodology"" dated April 2008, 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 of the Thermal Design, 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 April, 2008.

Y. Ogata

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

Enclosure 3

UAP-HF-08067
Docket No. 52-021

Response to NRC's Request for Additional Information

on

US-APWR Topical Report MUAP-07009-P, "Thermal Design Methodology"

April 2008
(Non-Proprietary)

Response to the NRC Request for Additional Information on

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

Non Proprietary Version

General Comments

G-1. On page 3-2, MHI references the MARVEL code. Reference 17 appears to be a non-loss-of-coolant-accident (LOCA) methodology topical report. Provide a description of all changes made to the MARVEL code from the previously approved version.

Response:

Topical Report MUAP-07010-P describes the methodology used by MHI for analyzing the non-LOCA events presented in Chapter 15 of the DCD. Part of MUAP-07010-P describes the MARVEL-M NSSS non-LOCA code with an emphasis on the changes made by MHI to the previously approved MARVEL code and validations of these changes.

As stated in Topical Report MUAP-07010-P, the MARVEL-M code is the same as the original MARVEL code from the viewpoint of constitutive and principal models. The main differences between the original MARVEL code and MARVEL-M are the extension from 2-loop simulation to 4-loop simulation and the incorporation of the built-in reactor coolant pump model that used to be run externally. MARVEL-M contains some other refinements such as the addition of a pressurizer surge line node, a hot-spot heat flux simulation, core void simulation, and a feedline break blowdown model, as well as an improved pressure convergence algorithm.

Detailed explanation of the modifications and refinements made in MARVEL-M are described in Section 2.1 of Topical Report (MUAP-007010-P), and code-to-code validations for the changes are provided in Section 3.1 of the same topical report.

If further clarifications regarding the MARVEL-M changes are needed during the review of Topical Report MUAP-07010-P, MHI will provide them at that time.

G-2. The statement in section 6.4 on page 6-1, needs clarification on the term "conservative results," as compared to what, or how to select the value to be conservative.

Response:

The purpose of Section 6 in Topical Report MUAP-07009-P is to describe the capabilities of VIPRE-01M, and in the case of Subsection 6.4, how the gap conductance is modeled. Section 6.4 clearly states that the gap conductance is specified as an input in VIPRE-01M, and that the input value can be adjusted to predict fuel pellet temperatures consistent with fuel design codes.

The following simple example illustrates the effect of varying the gap conductance from a lower-than-expected value to a higher-than-expected value. If the gap conductance is lowered from its normal value, the overall heat flux from the fuel rod to the coolant channel is reduced and fuel centerline temperature increases. Similarly, if the gap conductance is increased from its normal value, heat flux increases and fuel centerline temperature decreases. Over-predicting overall heat flux is conservative when calculating minimum DNB ratios and over-predicting gap heat flux is conservative when calculating peak cladding temperature. On the other hand, under-predicting heat flux is conservative when calculating peak fuel temperatures. This simple example illustrates that the conservative value for gap conductance varies for different combinations of initiating events and acceptance criteria. In all these cases, "conservative" means in the adverse direction relative to the normal value of the calculated parameter compared to the acceptance criterion.

The Rod Ejection and RCP Locked Rotor / Sheared Shaft events are two examples where the gap conductance is a key parameter. For these events, the capability to define a conservative gap conductance value in VIPRE-01M is used as described in Section 5.3 (5) of the Non-LOCA Methodology Topical Report (MUAP-07010-P) and Section 15.3.3.3.2 of the US-APWR Design Control Document, respectively.

In summary, Topical Report MUAP-07009-P describes the VIPRE-01M capability for the user to specify the value of gap conductance, and event-specific gap conductance assumptions are described elsewhere in the Non-LOCA Methodology Topical Report or applicable DCD Chapter 15 events.

G-3. The reference number for item (1) on page 8-1 should be 11 not 12.

Response

The reference number for item (1) on page 8-1 should read "reference 11", instead of "reference 12". MHI stands corrected.

G-4. In Appendix A, the number of axial nodes for cases 3 and 4 (Table A.1-2) does not agree with either the nodalization diagram (Figure A.1-4), or the result keys (Figures A1.5 thru 7). Please clarify this discrepancy.

Response:

The number of axial nodes for case 3 of Table A.1-2 should be [ ] instead of [ ]; the number of axial nodes for case 4 should be [ ] instead of [ ]. MHI stands corrected. In addition, the presented numbers of nodes in case 1 and case 2 as shown in Figure A.1-4 of the Topical Report will also be updated to accurately reflect the sensitivity cases performed. The updated Figure A.1-4 is attached.

Figure A.1-4 Sensitivity Study Cases for Axial Nodalization

G-5. In Appendix B, the database for the WRB-2 DNB correlation does not appear to support the lower grid spacing value (10 inches, database 22 inches). Please explain the effect of lower grid spacing on the departure from nucleate boiling (DNB) correlation.

Response:

The database for the WRB-2 does support the lower grid spacing value.

The database that covers grid spacing values (10 inches to 26 inches) is embedded in Westinghouse report WCAP-10444-P-A (Ref. G-5-1), which is Westinghouse proprietary document. Although MHI is entitled to using this database and the data were examined in its internal review for the compatibility between WRB-2 and VIPRE-01M, MHI does not feel appropriate to incorporate them into the MHI proprietary report.

The EPRI DNB data (Ref. G-5-2) that are included in Table B.3-2 and those of Westinghouse proprietary DNB data mentioned above justified the validity of WRB-2 correlation, which is now part of the MHI VIPRE-01M code.

Regarding the effect of lower grid spacing on DNB correlation, it is observed that the lower grid spacing tends to suppress DNB in the existing test data. The effect is caused by a local enhancement of turbulent mixing downstream of the grid spacer. This effect can not be modeled by the subchannel analysis code, but is reflected directly in the DNB correlation.

Figure G-5-1 shows the effect of grid spacing on DNBR at an over-power condition which is corresponding to case 2 in Table7-1.

References

G-5-1 S.L. Davidson, "Reference Core Report Vantage 5 Fuel Assembly", WCAP-10444-P-A, 1985

G-5-2 C. F. Fighetti & D.G. Reddy, “Parametric Study of CHF Data, Volume 3, Part 1: Critical Heat Flux Data Compilation”, EPRI NP-2609, 1982

Figure G-5-1 Typical effect of grid spacing on DNBR by WRB-2

G-6. Please confirm that the sensitivity studies directly relate to the MHI fuel to be used in the US-APWR.

Response:

All the sensitivity studies described in Appendix-A are performed based on US-APWR fuel design except that in Section A.6 TIME STEP SIZE are the sensitivity studies based on a generic PWR core.

Application of VIPRE-01M to Peak Cladding Temperature (PCT) Transients

MHI intends to replace FACTRAN with VIPRE-01M for the performance of Chapter 15 (condition III or IV) non-LOCA events transients, which are PCT limited instead of DNB limited.
Options for forced film boiling at the hot spot and the Zr-H2O reaction heat generation within the clad were added to the code.
Sensitivity studies reported only address DNB results.

S-1. Please provide sensitivity studies affecting PCT results, such as nodalization, time step, properties, etc.

Response:

Additional sensitivity studies focused on PCT analysis are shown in Figure S-1-1 through S-1-3.

Figure S-1-1 shows the sensitivity study on the radial nodalization in a fuel pellet. The sensitivity study with [ ] radial nodes provides good agreement with the result of the [ ] radial nodes case.

Figure S-1-2 shows the sensitivity study on the time step size. In order to maintain courant number larger than unity, appropriate axial mesh size was selected for the time step size sensitivity study. The number of axial nodes in the core region is [ ]. After that, time step size is refined to [ ]. All the cases show good agreement, in spite of the difference of axial mesh size and/or time step size.

Figure S-1-3 shows the sensitivity study on the fuel material properties. In the sensitivity study, UO_2 and Zircaloy properties that have been originally installed in the EPRI version of VIPRE-01 are used. Since the differences in material properties between MHI's and EPRI's are unnoticeably small for the case of un-irradiated fuel as described in Appendix D, the results show good agreement. Please see the response to TP-1, for the effect of the burnup in MHI's fuel properties.

Figure S-1-1 Sensitivity study on radial nodalization in a fuel pellet

Figure S-1-2 Sensitivity study on time step size and axial nodalization

Figure S-1-3 Sensitivity study on fuel material properties

S-2. *Section 4.1(2) Axial Nodalization, states that the number of axial meshes does not have significant impact on DNB ratio results; however, an increase in elevational discrepancy between cell center and grid location may have an adverse effect. Please discuss the magnitude of the effect and the user guidance/control established to offset this effect.*

Response:

Based on the sensitivity studies in Section A.1, it is confirmed that the number of axial meshes does not have significant impacts on DNB ratio results; however, increases in elevation discrepancy between cell center and grid location may have an adverse effect on the minimum DNBR. Figure A.1-7 in Appendix-A indicates that the magnitude of the effect [

].

Therefore, Topical Report recommends that [

], in Section A.1.

This has been reflected in the MHI's internal design procedure manual. For the axial noding used in standard DNBR analyses, the analyst is clearly required to meet the following guidelines;

- To use the axial mesh size smaller than [ ]
- To setup the axial mesh configuration in which [

]

Thermal properties in Section 6.2

TP-1. It is stated in this section that a degradation effect, due to burnup, is applied to the thermal conductivity. Please describe the burnup degradation model and provide results of analysis of the effect at various burnups up to the limiting fuel burnup.

Response:

The fuel thermal conductivity that takes into consideration the degradation effect due to burnup is explained in page D-1 of Appendix-D. Figure TP-1-1 illustrates the burnup effect on the fuel conductivity.

Fuel temperatures with the effect were calculated and compared with those calculated with the fuel design code FINE in Figure 7-17 and 7-18. The fuel temperatures at EOL (Figure 7-18) show larger temperature rise in the fuel than at BOL (Figure 7-17). This EOL condition, assuming 71 GWd/t of pellet burnup, virtually covers the design fuel burnup conditions of the US-APWR, in which the maximum rod burnup is limited up to 62 GWd/t.

Figure TP-1-2 shows the effect of fuel conductivity on the fuel temperatures at EOL. In this sensitivity study, thermal conductivity is simply changed from the EOL condition to other local burnup conditions. All other parameters including gap conductivity are kept unchanged.

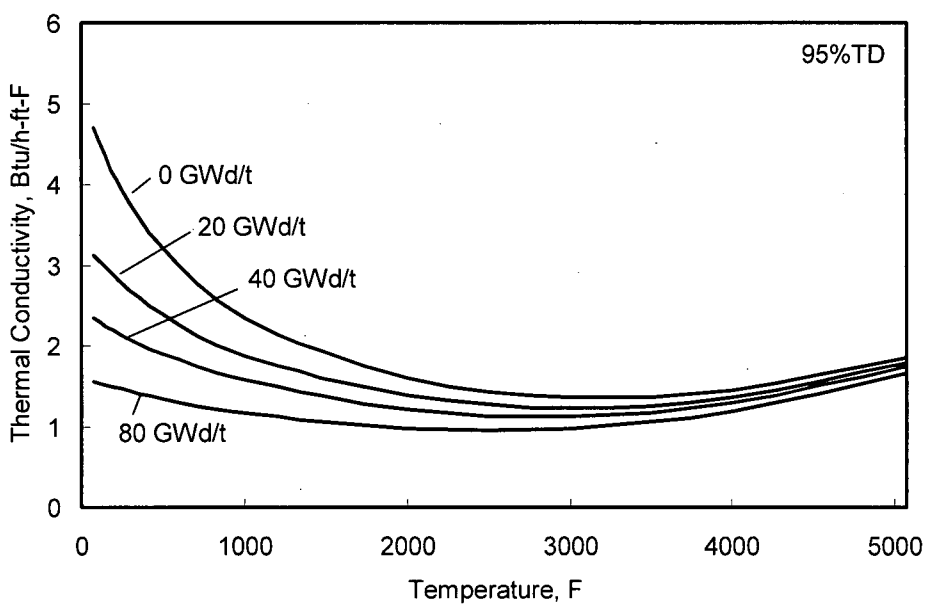

Figure TP-1-1 UO_2 thermal conductivity degradation with local burnup

Figure TP-1-2 Effect of degradation model on thermal conductivity for various local burnup

The metal-water reaction model described in Section 6.6.

Limitations in VIPRE-01M's basic capabilities limit the potential for enhancements. For example, constant material density since the code does not account for changes, and the simplified Zr-H2O layer treatment.

MW-1. Other than the comparison to FACTRAN, what assessment of the model was performed?

Response:

VIPRE-01M does not physically model the oxide layer as part of the cladding surface directly. The thermal properties of the cladding are treated [ ], as described in Appendix-D of the Topical Report. [ ].

The above treatment is justified by the comparisons with the fuel design code FINE. Figure 7-18 of Topical Report shows the comparison of both codes. In VIPRE-01M analyses, [ ]. It provided the same diametrical cladding thickness as in the FINE analysis. Both codes showed a good agreement in the fuel pellet surface temperatures.

MW-2. Can fuel pin failure occur resulting in oxidation on the interior surface of the cladding?

Response :

PCT analyses using VIPRE-01M are conducted focusing on the cladding temperature behavior prior to the cladding failure. Therefore, [ ], thus no appreciable oxidation would result on the interior surface of the cladding during the transient events.

MW-3. Provide a discussion of any differences between the FACTRAN models and the models implemented in VIPRE-O1M. Address the treatment of the oxide layer as a composite material.

Response :

VIPRE-01M treats the cladding as [ ]. Please see the response to MW-1.

FACTRAN does not consider the increase of thermal resistance potentially caused by the oxide layer.

In the non-LOCA PCT analyses, the above treatments do not cause noticeable differences, because the duration of the high cladding temperature is so short that the amount of the reaction is sufficiently small.

System Code (MARVEL) Capability for Asymmetrical Flows

M-1. Can the VIPRE-01M core flow driver code (MARVEL) adequately treat asymmetrical core flow? For example, a locked pump rotor, with loss of flow in 1-of-4 reactor coolant loops resulting in some portion of the core receiving less than the total pumped flow. If the total flow is 400 units, then each core quadrant obtains 100 units. The loss of flow in one reactor coolant loop means a total of 300 units, but the remaining 3-of-4 reactor coolant loops would likely see more than the loop without the pump (not necessarily 75 units each, but, perhaps, 90 units in three and 30 units in the other). How does this impact the applicability of the 1/8-core model, Figure 4.1, page 4-7?

Response:

Total inlet flow for the VIPRE-01M analysis is provided from MARVEL, and the azimuthally uniform core inlet flow distribution is used.

The coolant flows from each RCS loop encounter at the bottom of lower plenum and are mixed there, even in the partial loss of flow event. Only small distribution is left at the core inlet. Such a small distribution does not affect the DNBR analyses as shown in A.5 of Appendix-A. MHI's flow test experiences also endorse the applicability of that treatment.

The preliminary test results indicated that the flow from the remaining running loops are mixed very quickly by the time it reached to the entrance of the core. Flow distribution was rather uniform than the deviation of flow between different quadrants within [ ].

DNB Correlations

DNB-1. General: There are no 14 foot data referenced in the Topical Report MUAP-07009-P. All of the Columbia test data are for 12 foot Westinghouse/MHI fuel. MHI is assuming that the grid spacer studies conducted at Columbia for the 12 foot fuel are applicable to the MHI 14 foot fuel. Provide quantitative and qualitative technical justification for this assumption.

Response:

MHI followed the Westinghouse protocol (Ref.DNB-1-1) to demonstrate that WRB-1 and WRB-2 correlations can be used for available (including new designs) grid designs. MHI grid designs are essentially the same as the Westinghouse designs, which are characterized by egg crate type structure, spring and dimple rod holding mechanism, and mixing vane type flow mixing structures.

MHI has demonstrated that MHI grid designs have essentially the same DNB characteristics as those of the Westinghouse grid designs and can be soundly covered by WRB-1 and WRB-2 as shown in Appendix C.

WRB-1 and WRB-2 correlations properly reflect the effect of the heated length based on its database including 8 and 14 ft heated length data, as shown in Figure DNB-1-1.

Therefore, the WRB-1 and WRB-2 correlations are applicable for the MHI fuel designs with grid spacers such as Z2 and Z3.

Westinghouse modified WRB-2 correlation to become WRB-2M in order to account for its recent grid changes for the purpose of securing more DNBR margins. MHI could in fact gain extra DNBR margins from its improved grid spacers, Z2 and Z3, as well. However, MHI decided not to take credit for it. MHI will stay with the original WRB-1 and WRB-2 correlations for mainly the reason that the US-APWR core has a lower core power density, thus, more thermal margins in hand.

Reference:

DNB-1-1 L. D. Smith, III, et al., "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids" WCAP-15025-P-A, 1999.

Figure DNB-1-1 Effect of Heated Length up to Min. DNBR elevation on WRB-1 and WRB-2 correlations

DNB-2. Based on figures in Appendix C, it appears that the grid spacing in the MHI fuel design is less than the minimum used in the test programs (database). In addition, a grid spacer is located 0.7 inches above the start of the heated length of the core. Two grid spacers occur before the 13 inch or 10 inch lower limit of the database.

Response:

In the DNB test bundle set up, "simple support grids" are used to stand against the strong magnetic force induced by large electric current. The simple support grids will not have appreciable impact on the DNB test (Ref. DNB-2-1). Each simple support grid is placed at mid-span between two consecutive mixing vane grids and is not used in the actual fuel assemblies. The mixing vane grids in test bundles simulate the typical grid spacing of the US-APWR fuel assemblies. The grid spacing is of 18 inches and is greater than the lower limits of the databases (10 inch or 13 inch).

In the current MHI test programs, the mixing vane grids are located [

]. MHI does not have any specific intention to improve or degrade the DNB test results by adjusting the location of the lowest grid spacer.

It is not clear if the database referenced is complete or just a part of the database used to develop the actual DNB correlations. MHI omitted some data, see Appendix B. The database is used to establish a 95/95 DNB limit and then show the MHI fuel database supports the 1.17 minimum DNB value.

Response:

The database referenced in the topical report is complete. They are covered in Appendices B and C. These data enable us to confirm the applicability of WRB-1 and WRB-2 for MHI fuel designs.

The entire database for MHI fuel design is used to establish a 95/95 DNB limit and then show the MHI fuel database supports the 1.17 minimum DNB value.

a. *What is the expected impact of the grid spacers near the core entrance? Would they improve or degrade the grid spacer enhancements measured higher in the bundle? If the spacers streamline the flow then the turbulence enhancement seen higher up may actually be less - the DNB location needs to be considered. What does the "goodness" of the predicted-to-measured ratio mean?*

Response:

Grid spacers near the core entrance do not have noticeable impacts on DNB test results. They would not either improve or degrade the grid spacer enhancement functions at the locations downstream of the bundle.

Any effects created by the grid spacers near the entrance would dissipate in a short distance since all grid spacer effects on flow mixing are local. Typically, DNB is detected rightly before reaching the grid spacer of interest.

Please elaborate on the question 'What does the "goodness" of the predicted-to-measured ratio mean?'. We have not been successful trying to respond to the question.

b. *Westinghouse, for example, typically does not credit improvements below the first grid spacer, and instead uses a different DNB correlation (no enhancements) for that region. It would seem MHI uses the enhanced DNB correlations over the full height, perhaps conceptually correct but not obvious from the database. Provide justification that this approach is appropriate.*

Response:

MHI does not take credit for the first grid spacer effect either. Since the lowest grid spacer does not have mixing vanes in the MHI fuel design, MHI will use the W-3 correlation without spacer factor for the span between the first and the second grids as Westinghouse does.

Reference

DNB-2-1 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

DNB-3. On page 5-1 of Topical Report MUAP-07009-P, the first paragraph of section 5.1 alludes to the use of the W-3 correlation in regions outside the ranges of the WRB-1/2 correlations. No detail was provided as to where exactly the W-3 correlations apply. The applicable ranges of the W-3 correlation needs to be stated and demonstrated to be valid as per the NRC approved SER for this correlation.

Response:

W-3 correlation without spacer factor was originally approved by the NRC for the Westinghouse fuel bundle DNB analyses and has been permitted to be used for open-lattice fuel bundle design outside the applicability ranges of WRB-1 and WRB-2 correlations. This design practice has been documented in the Westinghouse Thermal Hydraulic Design Procedures.

The applicable ranges are stated below.

Pressure:
Local mass flux:
Local quality:

(*DNBR correlation limits related to 95/95 basis are 1.45 for pressure range below 1000 psia, and 1.30 above 1000 psia)

The typical applications are expected in the DNBR analyses of low pressure events such as Steam System Piping Failures. In addition, W-3 is also used when the DNBR analysis at locations upstream of the first mixing vane grid is needed.

Since the W-3 correlation database consists of test results mostly from single tube tests. Its applicability and its correlation limits are not affected by the subchannel analysis code in use. MHI follows the same practice as prescribed in the Westinghouse procedures that were concurred by NRC.

DNB-4. On page 7-4 of Topical Report MUAP-07009-P, it is not clear from Tables 7-1 and 7-2 whether the stated thermal hydraulic parameters cover the expected ranges of operation (steady state and transient domains), for the US-APWR. Please relate the thermal hydraulic parameters to the expected operational ranges.

Response:

Those operating conditions in Table 7-1 are selected to represent the normal operating condition and the design limit conditions of US-APWR. Those conditions are as follows:

case 1:
case 2:
case 3 - 6:

case 7:
case 8:

The above conditions cover wider operation ranges than the safety analysis conditions described in the response to DNB-9.

The operating condition in Table 7-2 is [ ], which provide more severe transient conditions than the US-APWR.

DNB-5. On page 7-5 of Topical Report MUAP-07009-P, Figures 7-1 thru 7-16 provide a variety of void model behavior as a function of chosen thermal hydraulic parameters. Of particular interest to the NRC staff is the void fraction behavior as a function of axial height. It appears that approximately 2/3 up the core, void effects come into play. Is this void behavior captured in the neutronic models as well as the thermal hydraulic models?

Response:

VIPRE-01M takes the void effects into consideration in its subchannel analyses. No neutronic models are included in the VIPRE-01M code.

In most safety analyses, an axial power distribution is predetermined as the initial condition. Since the effect of voids will suppress the power generation in upper core region due to less available thermal neutrons, it is conservative not to take into consideration the nuclear feedback effect caused by voids.

In the several events, for example, Steam System Piping Failures case, the power distributions are evaluated by nuclear design codes during the transients. In the codes, the subcooled void effect is eliminated for conservatism.

DNB-6. Regarding these same figures, it appears that the operational domain of the USAPWR could entail void fractions on the order of 80 to 85 percent. How do these high void fractions compare to current pressurized water reactors (PWRs), and how are these high voids accounted for in the thermal and neutronic models?

Response:

Those figures represent postulated limiting local conditions, under which the DNBR is close to those for the design limit conditions. The void fractions under the postulated hot sub-channel conditions may look much higher than those under the core-averaging normal operating conditions or the conditions of typical transient events, however, it is in fact not much difference from the core conditions of the current pressurized water reactors.

The effect of the voids on thermal models has been included as part of the VIPRE-01M governing equations. The effect of the voids on neutronics has been discussed in the response to DNB-5 above.

DNB-7. In Appendix B, page B-1 of Topical Report MUAP-07009-P, the first paragraph alludes to the use of the WRB-1 and WRB-2 DNB correlations in the sub-channel code VIPRE-01M to determine the DNB value for the US-APWR. It is not clear to the staff from this paragraph whether the data bases for the WRB-1 and 2 cover the intended operational range of the US-APWR.

Response:

The data bases for the WRB-1 and WRB-2 do cover the intended operational range of the US-APWR. For the operational range of the US-APWR, please see the response to DNB-9.

DNB-8. Appendix C - page C-1, provides a description of the z1 and z2 spacer tests conducted at Columbia University. It appears that these tests were conducted on 12 foot fuel and only with a cosine power profile. No justification was provided in this appendix as to why 12 foot test data is applicable to 14 foot fuel. Please provide qualitative and quantitative technical justification in support of the intended application.

Response:

Please see the response to DNB-1.

Appendix C - page C-1, provides a description of the Z2 and Z3 spacer tests.

DNB-9. In neither Appendix B nor C are the operation ranges, pressure, mass flow, quality, temperatures, etc., stated. Please provide table(s) showing the full Operational ranges of the US-APWR compared to the current PWRs, (12 and 14 foot cores). Explain any differences.

Response:

Table DNB-9-1 shows the operational ranges anticipated in the Safety Analysis of the US-APWR in comparison with current PWR with 12-foot core. MHI does not possess the 14-foot core operation data of current PWRs.

Table DNB-9-1 Comparison of Operational Ranges between US-APWR and current PWR

	WRB-2 Applicable range	US-APWR	4-loop plant
Core Thermal Power (MWt)	-	4451	[ ]
Thermal Design Flow (gpm)		112,000x4	
Coolant Temperature (F) T_{COLD} T_{HOT}		 551 617	
$F_{\Delta H}^{N}$ (*)		1.73	
Design Axial Power Distribution for DNB Analyses		Fig. 4.4-4 in the Design Control Document (Rev.0) of US-APWR	
Pressure (psia) Minimum Maximum	 1440 2490	[ ]	
Local quality (%) Minimum Maximum	 -10 30		
Local mass flux (Mlb/hr-ft^2) Minimum Maximum	 0.9 3.7		

* including measurement uncertainties.