February 6, 2009

Mr. Yoshiki Ogata, General Manager
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16-5, Konan 2-Chome, Minato-Ku
Tokyo, 108-8215 JAPAN

SUBJECT: MITSUBISHI HEAVY INDUSTRIES, LTD. - REQUEST FOR ADDITIONAL INFORMATION ON TOPICAL REPORT MUAP-07010-P, "NON-LOCA METHODOLOGY"

Dear Mr. Ogata:

On January 14, 2009, Mitsubishi Heavy Industries, Ltd. (MHI) participated in a conference call with the U.S. Nuclear Regulatory Commission (NRC) staff regarding a third set of Requests for Additional Information (RAIs) for Topical Report MUAP-07010-P, "Non-LOCA Methodology." These RAIs were sent to you in draft form for a proprietary review. Enclosure 1 consists of new RAIs while Enclosure 2 pertains to follow-up RAIs. The expected response time to these RAIs is 30 days from the date of this letter, to the NRC Document Control Desk. If you have any questions or comments concerning this matter, you may contact me at 301-415-7871, or via email at Michael.Takacs@nrc.gov.

Sincerely,

/RA/

Michael Takacs, Project Manager
US-APWR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-021

Enclosures:
As stated

cc: See next page

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Requests for Additional Information (RAIs) Rev.1
US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P

Licensing Basis for RAIs: 10 CFR 50 App. A

Criterion 10--Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 15--Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

RAI #	Reviewer	Full Text
SECTION 3.1: MARVEL-M Code		
3.1-6		Present validation data of MARVEL-M for the steam generator tube rupture event.
3.1-7 (Revised)		Give the history of the version of LOFTRAN used in MARVEL-M validation. Is the version of LOFTRAN employed by MHI for validation the same as that which was previously approved by the NRC? Detail any changes MHI has made to LOFTRAN. This is necessary because MHI is using LOFTRAN to validate the results from MARVEL-M, and it must be clear that the version of LOFTRAN employed has received licensing approval in the United States.
3.1-8 (Revised)		MHI uses LOFTRAN as one of the means to demonstrate that the MARVEL-M code behaves as expected. For the comparison to be meaningful, the algorithms, numerical methods and, if used, correlations should be sufficiently different. Does MARVEL-M share any significant algorithms, numerical methods or correlations with the version of LOFTRAN used for the comparison? If yes, please describe the similarities. If no, provide examples where there are fundamental differences.
APPENDICES		
App E-2		Show the scale on the vertical axis of Figure E-1.

Follow-up on previous Requests for Additional Information (RAIs) Rev.1
US-APWR Topical Report: Non-LOCA Methodology, MUAP-07010-P

Licensing Basis for RAIs: 10 CFR 50 App. A

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RAI #	Reviewer	Full Text
CHAPTER 2: MARVEL-M Code Description		
2.1-2-1		Please confirm the following understanding of the DNBR lookup table methodology: The MARVEL-M lookup table is a database of DNBR, core inlet temperatures, system pressures, and core heat flux data generated from VIPRE-01M steady state calculations. Using system analyses to determine the core inlet temperature and system pressure during a transient, an interpolative scheme is used to determine the normalized heat flux and thus the DNBR at each time step through the analysis. This simplified DNBR lookup table methodology is only used for non-LOCA events that have constant core flow rate and “are bounded by the applicable power distribution.” In the event that the parameters of the calculation exceed the limitations of the lookup table, the analyst is flagged to use VIPRE-01M to directly calculate DNBR based on a DNB correlation in VIPRE-01M.
2.1-2-2		How is the number of DNBRs evaluated in the generating the DNBR lookup tables (N) chosen? What degree of interpolation is used when using the table? Verify that the DNBR lookup tables cover the full operating space (pressures, temperatures, flow rates) that the methodology is used for. Explain what is meant by “applicable power distribution” with regard to the use of the simplified DNBR tables.
2.1-3-1		[DELETED]
2.1-8-1 (Revised)		The discussion provided in response to RAI 2.1-8 by MHI contained an error and was inadequate for the reviewer to evaluate the modeling of the four major thermal resistances in the calculation of the overall heat transfer coefficient for the steam generators. Provide clarification on the calculation of the initial values of the four thermal resistances, R_{pf}^0, R_{tube}^0, R_{bo}^0, and R_{foul}^0, and define all parameters and variables. Provide clarification on their calculation during the transient, and their combination to yield the overall

		heat transfer coefficient during the transient. Provide a (hypothetical) numerical example to clarify for the reviewer the overall procedure as modeled in the code from initial conditions through a transient. Discuss the overall heat balance and the fouling resistance. Provide the bases for the steam generator thermal resistances at nominal conditions? Provide comparisons to data if possible.
2.1-10-1		There may be physical effects in the coolant flow not captured by modeling safety injection into the cold leg instead of the Direct Vessel Injection featured in the US-APWR. Justify why cold leg injection is more conservative.
2.1-13-1 (Revised)		In the MHI response to RAI Appx.E-1 (in UAP-HF-08141, Docket No. 52-021, August 22, 2008), it is stated, "It should also be noted that uniformity in vessel inlet mixing can best be approximated by perfect mixing for very low flows such as during natural circulation conditions. Perfect mixing is assumed for the Steam Generator Tube Rupture event due to natural circulation conditions that exist during most of the event." This statement appears to contradict the MHI response to RAI 2.1-13. MHI is requested to clarify and comment on the discussion above.
2.1-16-1		Describe or demonstrate that the MARVEL-M's pump model is adequate for the full range of Chapter 15 events.
2.1-16-2		Explain why the measured and predicted (using MARVEL-M) pump coast down curves are so similar. Have these measured data of pump coast down used to validate MARVEL-M been compared to the LOFTRAN code?
3.1-2-1		Did the Uncontrolled RCCA Bank Withdrawal at Power event have the largest differences between LOFTRAN and MARVEL-M?
3.2-3-1		Please confirm that the approach used in TWINKLE-M modifies the two-group diffusion coefficients for both the radial reflector region and two axial reflector regions.
3.2-3-2		Detail the process by which the diffusion coefficient in the reflector is modified before it is input into TWINKLE-M.
3.2-4-1		MHI asserts that the differences between the two codes in the assemblies with control rods inserted for the HZP case is due to the different modeling of the spatial dependence and the fact that there is a strong spatial gradient near the control rods. This is a reasonable explanation; however, it is not consistent with their response to RAI 3.2-5. The explanation of the differences at the location of a control rod for the HFP case is claimed to be because the burnup is different in the controlled and surrounding assemblies. However, since the reactor is expected to be operated with control rods withdrawn, the burnup differential between controlled assemblies and surrounding assemblies should be no different than between any other set of adjacent assemblies. Please provide further clarification of the responses to RAI 3.2-4 and 3.2-5.
3.2-6-1		Comment on the positive Doppler temperature coefficients presented in Table 3.2-6.1.
3.2-7-1		MHI states that the ejected rod worth is the same with both mesh sizes in part because the steady state power distributions are the same. It is true that the power distributions should be similar as they are both tuned to the same ANC-generated power distributions. The result shown for the

		Doppler fuel temperature is similar for the two meshes with the 2x2 mesh giving a higher value. MHI also supplies radial power distributions at different times during a rod ejection accident to show that the effect of mesh size is small. Several comparisons of the effect of axial mesh are also shown to be insensitive to mesh size. The conclusions implicitly assume that the 4x4 mesh yields a converged solution. What changes would occur if a smaller mesh (6x6 or 8x8 radially or >76 mesh points axially) were used?
3.2-7-2		What is the time step size used in the TWINKLE-M simulation of the rod ejection accident? How is this time step size determined?
3.2-7-3		In Figure 3.2-7.6, why is no adjustment made of the diffusion coefficient in the reflector region?
3.2-8-1		Were the adjustments made to neutron lifetime and delayed neutron fraction adjusted to minimize the difference when changing mesh size? Were these only made for the comparison of mesh sizes or is this part of the procedure for calculating transients?
3.2-9-1		Explain how the adjustments made in the responses to RAIs 3.2-7 through 3.2-9 relate to the way in which analysis is generally done with TWINKLE-M.
CHAPTER 5: Event Specific Methodology		
5.3-1-1		Do operating procedures allow for fully or partially inserted misaligned or inoperable control rods and if so, was this taken into account in the analysis of the design limit?
5.3-1-2		The TWINKLE-M model is adjusted to give the same design limit worth as from the ANC calculation. Is the identical configuration represented in the TWINKLE-M three-dimensional model?
5.3-1-3		The adjustment is done by “changing the eigenvalue” in both the three- and one-dimensional models. In the former, this would mean that a multiplier is applied to the fission rate throughout the reactor rather than to a local property of the controlled fuel assembly (e.g., absorption cross section). What is the impact on the analysis of this approximation?
5.3-2-1		According to Figure 5.2-1, the VIPRE-M calculation needs rod power factors for 23 rods; TWINKLE-M presumably provides the “hot channel factor” for the hottest quarter (based on a 2x2 mesh) of a fuel assembly, i.e., not even for one fuel rod. Please explain what information is used by VIPRE-M to analyze the hot channel.
5.3-2-2		The response to this RAI indicates that whatever the model is, it is not necessary to do a detailed 1/8 core calculation; all information is obtained by focusing on the hot channel. Does this allow you to make the assumption that the VIPRE-M model can place the ejected rod at the center of the core rather than at its realistic position?
5.3-2-3		The third sentence of the second paragraph in response to RAI 5.3-2 is grammatically incorrect; clarify.
5.3-3-1		The response suggests that the design limit is actually just the ANC generated peak/average fuel assembly power (with some unknown adjustment for uncertainty and “safety margin”) for the case with the ejected rod out and no other change in rod configuration. It is used to add a conservative factor to the analysis only because it turns out to be larger than the same ratio calculated by TWINKLE-M. Please confirm this understanding of the hot channel design limit, mathematically describe the hot channel factor, uncertainty and safety

		margin, and explain how the uncertainty and safety margin are applied to the design limit.
5.3-6-1		In MUAP-07009-P and in the response to RAI 5.3-6, it is noted that mixing between assemblies is “conservatively ignored.” Yet later in the response to RAI 5.3-6, it is stated that “the power of the hot assembly is assumed to be higher than the power of the surrounding assemblies, which would cause *larger flow redistribution from the hot assembly to the surrounding assemblies* and result in a more limiting coolant condition in the hot assembly.” As written, these two statements are contradictory. Clarify.
5.3-7-1		The response explains how cross sections are averaged radially, but does not address other parameters such as diffusion coefficients and delayed neutron data. Please provide the missing information.
5.3-7-2 (revised)		In response to RAI 5.3-7 of UAP-HF-08141-P MHI provided a comparison between the TWINKLE-M 3-D and 1-D core average axial power in Figure 5.3-7.1. Is the difference given in Figure 5.3-7.1 a representative or maximum expected difference? If representative, please explain if a larger difference would have a significant impact on the number of rods in DNBR.
5.3-10-1		According to the response, the control rods are modeled explicitly. However, the report says that the trip reactivity is the design limit, i.e., not what is calculated by TWINKLE-M. Please explain further.
5.5-2-1		[DELETED]
APPENDICES		
App. E-1-1		[DELETED]
App. F-1-1		Although the final form of the “modified Zaloudek” correlation is attractive due to its simple dependence on ($P – P_{sat}$), comparison to experimental data is necessary to show that the correlation is accurate under the conditions for which it was developed. Please provide additional validation for the “modified Zaloudek” correlation.