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

RAI #	Reviewer*	Full Text
CHAPTER 2: MARVEL-M Code Description		
2.1-1	GAG	The evolution of MARVEL-M 4-loop code from the MARVEL 2-loop code was discussed. Please provide details of the significant modeling changes between the two codes.
2.1-2	GAG	Please provide a description of the simplified DNBR model in MARVEL-M.
2.1-3	GAG	Have any MARVEL/MARVEL-M code comparisons been performed? Please provide a MARVEL/MARVEL-M code comparison for a typical 2-loop calculation.
2.1-4	GAG	If two-phase homogeneous flow is not applicable, what model does the code use for two-phase flow for $\alpha > \alpha_{homogeneous}$? How does the user deal with conditions in which homogeneous two-phase flow is not applicable?
2.1-5	GAG	Have any of the mixing models been compared to scaled tests? If so, please provide the comparisons.
2.1-6	GAG	Please provide the methodology for the natural circulation flow modeling.
2.1-7	GAG	Please provide the details of the reactor coolant pump model.
2.1-8	GAG	Describe the modeling of the four major thermal resistances in the calculation of the overall heat transfer coefficient for the steam generators.
2.1-9	GAG	Please provide the details of over-temperature ΔT and over-power ΔT trip protection methodologies and specifications, including uncertainties.
2.1-10	GAG	Please provide the details of cold leg injection by the safety injection system and its role in non-LOCA events.
2.1-11	GAG	It is stated that the MARVEL-M algorithm for core mixing is changed from the original MARVEL model. Are the changes due to the expansion of the capability to four loops, or are there fundamental changes in the mixing phenomenology?
2.1-12	GAG	Mixing is assumed to occur in the reactor vessel lower plenum as specified in the code input. How are the mixing factors (FMXI) established by the user? What guidance is provided to the user?
2.1-13	GAG	Some 1/7-scale mixing tests were carried out in the 1970's. It is suggested that mixing assumptions can be inferred from the published results. What justification can be provided to support these claims for evaluating FMXI? Provide documentation of the 1/7-scale tests.
2.1-14	GAG	The methodology for reactor vessel upper plenum mixing is exactly the same as for mixing in the lower plenum. The mixing

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		factor is FMXO. How are the mixing factors (FMXO) established by the user? What guidance is provided to the user?
2.1-15	GAG	Are there any scaled experimental data to use for guidance in determining FMXO? If so, please provide the data or guidance.
2.1-16	GAG	A reactor coolant pump model has been added to MARVEL during the evolution to MARVEL-M. Has this model been tested against pump vendor test data? If so, please provide the comparisons.
2.1-17	GAG	Transition to natural circulation flow is modeled in MARVEL-M and the elevation head equations are typical. Has this model been tested against scaled experimental data? If so, please provide comparisons.
2.1-18	GAG	How does the transition to natural circulation depend upon the pump coast down model? Have sensitivity calculations been performed over a range of conditions to demonstrate confidence in the end state? If so, please provide the results.
2.1-19	GAG	Previously proposed RAI has been withdrawn.
2.1-20	GAG	"Realistic models" have been added to MARVEL-M "to simulate real plant transient behavior." Although they are code options, they are not used for licensing evaluations of reactor plants. What controls are in place to ensure that this is the case?
2.2-1	DD	There are several differences between TWINKLE and TWINKLE-M that are mentioned but not discussed in any detail. The introduction of more spatial points and discontinuity factors suggests that the numerical algorithm for solving the diffusion equations may have also been changed although it is not stated. Please list all changes and provide additional description of the differences between the codes.
CHAPTER 3: MARVEL-M Code Validation		
3.1-1	GAG	Previously proposed RAI has been withdrawn.
3.1-2	GAG	Please provide documentation of MARVEL-M/LOFTRAN code comparisons that may not be in agreement (if available), along with any explanations for the deviations.
3.1-3	GAG	Please provide the DNBR vs. t for the partial loss of forced reactor coolant flow analyses for both MARVEL-M and LOFTRAN.
3.1-4	GAG	Please provide the DNBR vs. t for the complete loss of forced reactor coolant flow analyses for both MARVEL-M and LOFTRAN.
3.1-5	GAG	Please provide the DNBR vs. t for the reactor coolant pump shaft seizure analysis for both MARVEL-M and LOFTRAN.
3.2-1	DD	The text says that constitutive models have not changed in TWINKLE-M but the introduction of discontinuity factors can be considered as such. Please respond to this comment.
3.2-2	DD	The TWINKLE code as approved for use by Westinghouse

		needs to be shown to provide acceptable results when used by MHI. This becomes more important when the code has also been modified. The comparisons in Section 3.2.1 show reasonable agreement with ANC for reactor conditions of interest in rod ejection accident analysis. However, no information is given to show how well TWINKLE-M models steady state power and reactivity measurements at an operating PWR. as well as how it models neutron kinetics. Please provide additional information which helps validate the code as used at MHI.
3.2-3	DD	The treatment of the core-reflector boundary condition is important especially when the ejected rod is near the core periphery. Please explain the algorithm by which the diffusion coefficient in the reflector is modified.
3.2-4	DD	The comparisons between TWINKLE-M and ANC at HZP show the largest differences in the assemblies with control rods inserted. What are the differences in modeling between the two codes that are claimed to be causing the larger errors?
3.2-5	DD	The largest differences between TWINKLE-M and ANC for the HFP case cannot be due to the effect of modeling control rods as all rods are withdrawn. Please comment on the cause of the differences for the HFP case.
3.2-6	DD	Table 3.2.1-1 provides a comparison between TWINKLE-M and ANC for key parameters. A key parameter for the REA that can be calculated with the two codes is the Doppler reactivity coefficient. Please provide a comparison of the Doppler coefficient for at least one core configuration.
3.2-7	DD	Finite difference codes were originally meant to have mesh sizes on the order of a transport mfp. The mesh spacing used in the comparisons to ANC is 11 cm in all directions. In order to understand why the coarse mesh 2x2 results are in agreement with the 4x4 results please address the following: Is the ejected rod worth calculated for each case (Table 3.2.2-1) or is the worth for one case fixed somehow to be in agreement (within 5 pcm) of the other case? What is the maximum fuel enthalpy (or temperature) for the two cases? It is assumed that the maximum hot channel factor is for an entire assembly. Please provide a comparison of additional hot channel factors (at a minimum for the assemblies surrounding the ejected rod). What would be the result if the axial mesh were also changed?
3.2-8	DD	It is not clear why the neutron lifetime is given in Table 3.2.2-1. If it is meant to be a given reactor condition then how does it enter into the calculation? Whether it is a reactor condition or the result of an edit from the calculation, how is it obtained?
3.2-9	DD	How is the delayed neutron fraction calculated for use in TWINKLE-M? If a single number is used for all fuel

		assemblies, please explain how results would compare if using a number generated for each assembly.
3.2-10	DD	Please provide discussion of the effect of axial mesh size on the axial power distribution.
CHAPTER 5: Event Specific Methodology		
5.2-1	GAG	Are the calculated results for the complete loss of reactor coolant flow AOO sensitive to the reactor coolant pump coast down assumptions? Have any sensitivity calculations been performed?
5.2-2	GAG	For the complete loss of reactor coolant flow AOO, it is stated, "a suitable rod bundle DNB correlation…are used." Provide details of the DNB correlation.
5.3-1	DD	In the three- and one-dimensional analysis of the REA, a design limit for the ejected rod worth is used by adjusting the eigenvalue. How is the design limit determined? How is the eigenvalue changed in the calculation to simulate the ejection?
5.3-2	DD	In the HZP analysis only the hot channel factor and core average power are passed on to VIPRE-M from TWINKLE-M. How is any detailed description of assembly powers and axial power distributions passed on to VIPRE-M? What is the advantage of using VIPRE-M when all the information needed to ascertain whether or not the acceptance criteria are met is contained in the TWINKLE-M results?
5.3-3	DD	Since hot channel design limits usually refer to a steady state condition, please explain the definition of the hot channel design limit for a transient.
5.3-4	DD	Since only one hot channel factor (the maximum) is extracted from the TWINKLE-M calculation, how is it applied in VIPRE-M where an entire 1/8 core is represented? Why doesn't VIPRE-M do a single channel calculation?
5.3-5	DD	It is recognized that increasing the hot channel factor will make the calculation of fuel temperature and departure-from-nucleate-boiling ratio (DNBR) more conservative in the hot channel. However, assuming that the total power comes from the TWINKLE-M calculation, other channels will have lower powers in the VIPRE-M calculation relative to what was calculated in TWINKLE-M and this will make the calculation of temperature and DNBR in those channels lower. Since one acceptance criterion is a function of clad oxide thickness, the limiting channel may not be the hottest one. Is this taken into account by using the original TWINKLE-M distribution of fuel enthalpy rise rather than the distribution from VIPRE-M?
5.3-6	DD	The VIPRE-M model uses a 1/8-core representation. This assumes symmetry that is not present in the neutronics calculation. However, this will be acceptable if there is no cross-flow out of that sector or if the time frame is too short for

		this to be important. Please comment.
5.3-7	DD	Please explain how the TWINKLE-M one-dimensional model is obtained and how results for the axial power distribution compare with those obtained from a three-dimensional model.
5.3-8	DD	What "conservative multiplier" is applied for the Doppler feedback?
5.3-9	DD	If a three-dimensional model is available in TWINKLE-M for doing the REA analysis at HZP, why isn't this same model used for the HFP case instead of shifting to a one-dimensional model?
5.3-10	DD	For the modeling of reactor trip in TWINKLE-M it is stated that the reactivity insertion curve is "simulated." Does this mean that the reactivity insertion is not explicitly modeled by changing cross sections in fuel assemblies? If it is not explicitly modeled, how is the simulation done?
5.4-1	GAG	For the steam system piping failure, asymmetric power generation in the core occurs due to a non-uniform cool down. Is this accident sensitive to the user-input mixing factors for the lower and upper plena? Have any sensitivity calculations been performed? If so, please provide the results.
5.4-2	GAG	It is stated "flow mixing in the reactor vessel is modeled in the code. The mixing factors for the reactor vessel inlet and outlet plena are defined conservatively by the input referring to the mixing test results by the 1/7 scale reactor vessel model." Please provide the model and substantiate why it is conservative.
5.5-1	GAG	For the feed water system pipe break, the natural circulation model in MARVEL-M is invoked. Are the calculated results sensitive to the timing of the transition to natural circulation flow? Also, are the reactor vessel inlet and outlet plena mixing factor the same during natural circulation as for the case of forced flow, and what values for the mixing factors are used?
5.5-2	GAG	Are the calculated results for the feedwater system pipe break sensitive to the assumed mixing factors, and have any sensitivity calculations been performed? If so, please provide the comparisons.
5.6-1	GAG	For the steam generator tube rupture, "the reactor coolant in the reactor vessel upper head dead volume may flash and form a steam phase at the top…." Does this condition have any effect on upper plenum mixing, and are there any feedbacks to the lower plenum and reactor inlet mixing conditions?
CHAPTER 6: Sample Transient Analysis		
6.2-1	GAG	For the complete loss of reactor coolant flow AOO, please specify how uncertainties in the input parameters, trip set points, and calculated variables are included in the calculations.
6.3-1	DD	In the discussion of the HFP rod ejection accident sample case,

		it is stated that ANC is used to obtain local peaking factors. No mention of the ANC code is found in the section on the rod ejection accident methodology. Please clarify.
6.4-1	GAG	Is the steam system piping failure event sensitive to the location of the break? Is a break between the steam generator and turbine the most limiting break location for all accident conditions?
6.5-1	GAG	The analysis assumes that "all of the steam generators are at the steam generator water level low trip set point." Is the pressurizer water volume sensitive to perturbations in this assumption?
6.6-1	GAG	For the steam generator tube rupture event, manual reactor trip is assumed at time = 900 seconds because operator action is required to recognize the event. Would the transient results be different if the timing of the manual reactor trip were varied, i.e., 1200 seconds and 1500 seconds, instead of 900 seconds? What was the basis for choosing 900 seconds?
APPENDICES		
App. A-1	GAG	Appendix A compares the calculated DNBR from MARVEL-M and VIPRE-01M (steady state) for two cases. The code results are identical in both cases. However, the MARVEL-M DNBR analysis uses look-up tables that were generated by calculations with VIPRE–01 M. Although it is expected that the VIPRE–01 M model is the same as approved by NRC, this comparison does not validate the DNBR model in MARVEL-M for other transient and accident conditions. Please comment.
App. E-1	GAG	Appendix E presents a set of sensitivity analyses based upon reactor inlet plenum mixing for the case of steam system piping failure. Although the case presented indicates that the assumption of no inlet mixing only causes a small reduction in the minimum DNBR, it cannot be extrapolated to other cases on this basis alone. Provide additional comparisons for other AOOs and PAs to substantiate the claim that DNBR is insensitive to the mixing assumptions.
App. F-1	GAG	Please provide the Zaloudek correlation that MHI has used to perform the steam generator tube rupture break flow calculations.
* Reviewers: George Greene (GAG), David Diamond (DD)		