ENCLOSURE 2

MFN 09-234

Comment Summary and Safety Evaluation Markup

Non-Proprietary Information

IMPORTANT NOTICE

Enclosure 2 is a non-proprietary version of the Comment Summary and Safety Evaluation Markup from Enclosure 1, which has the proprietary information removed. Portions that have been removed are indicated by open and closed double brackets as shown here [[ ]].

Location	Comment
Generic	The SE uses the terminology AOOs and ATWS overpressure. Please make sure that it is clear that the NRC approval includes the ASME overpressure event (MSIV closure with flux scram), which is a variation of an AOO with an additional failure and is clearly within the scope of applicability of the methodology.
Section 3.3.1.3, Lines 31-34	Suggest simplifying complex sentence. "TRACG04 in transient mode will calculate the transient fuel temperature, but the Doppler coefficient is based on the PANAC11-predicted Doppler response to temperature, which is based on the PANAC11-predicted steady-state temperature." to The Doppler response is based on the transient fuel temperature from TRACG04 using the coefficients developed from PANAC11.
Section 3.3.1.5 Line 28	"The infinite eigenvalue used for………." Should be The infinite lattice eigenvalue used for……….
Section 3.4.1.1 Lines 29-32	May want to clarify. "For power calculations for AOOs, the contribution from the decay heat is conservatively increased, therefore, increasing the integrated thermal energy deposited into the reactor coolant system (RCS) for AOO evaluations terminated by a SCRAM." The use of the May Witt accomplishes this, but there is no standard adjustment that does this.
Section 3.4.1.6 Lines 20 -21	The heat from metal-water reaction is included in the energy balance for the core. It is not include in the edit for the total power, which is limited to fission and decay heat power. The edit of energy generation from metal-water reaction is included for each channel component.
Section 3.4.1.7 Footnote 2	The energy release by fission is calculated by TGBLA when the kinetics solver is activated. TRACG alternatively uses NEDO-23739 values for the energy released per fission when the kinetics solver is disabled (see response to RAI 22). The 4 parent chains are always determined from NEDO-23739.
Section 3.7.4 Line 8	Typo Berenson
Section 3.9.1 Lines 20-22	Potentially Confusing. The models for flow regime transitions in TRACG02 had only been qualified at high pressure. GEH qualified TRACG against low pressure data to extend the applicability of TRACG to LOCA applications. The TRACG qualification report NEDE-32177 includes high and low pressure simulations.
Section 3.10.5.1 Lines 8-10	The dynamic gas gap conductance inputs for both TRACG02 and TRACG04 are based on upstream GESTR-M calculations. Suggest also mentioning PRIME, as we will be transitioning to PRIME inputs.
Section 3.10.5.2 Pg 40 Lines 8-11	According to Reference 22, [[ ]] Ref 22, pg. 1-44 says that the correlation parameters consider this, but not that PANAC11 solves it. P11 correlates the fuel temperature to the nodal power

Location	Comment
Section 3.10.5.2 Pg 40 Lines 25-34	The NRC staff found that the new fuel conductivity model in TRACG04 predicts lower fuel thermal conductivities with increasing fuel exposure and agrees to a large extent with the FRAPCON3 model in terms of variation with temperature and exposure. However, the NRC staff finds that the improved model appears to misrepresent the impact of gadolinia on fuel thermal conductivity at high exposure. The NRC staff cannot conclude that the improved thermal conductivity model represents a best estimate of the fuel thermal conductivity over the full range of gadolinia loadings and exposures. These conclusions from the PRIME review may reflect preliminary conclusions given that the PRIME review is not completed. This seems like evaluation verbiage that would be better placed in the PRIME SE. Especially with the PRIME review not yet complete. Other limitations and conditions direct us to use PRIME when it is approved.
Section 3.10.5.3	Similar statements from 10.5.3. Same comment. The NRC staff similarly reviewed the PRIME03-based thermal conductivity model and found that gadolinia bearing pellet temperatures will be under predicted at high exposure. Furthermore, while the NRC staff does not find that the improved model accurately captures the impact of gadolinia on fuel thermal conductivity, particularly for high exposure, the NRC staff notes (based on its preliminary review)……….
3.12.3 Pg 46 Lines 9-12 Also, Lines 39-40	While the oxide layer thickness affects cladding heat transfer characteristics, the NRC staff notes that the initial oxide layer thickness in TRACG04 is either directly input for bounding calculations or is calculated according to the same empirical model based on plant data as in TRACG02. The NRC staff notes that the option to predict the initial clad oxide thickness in TRACG02 and TRACG04 remains unchanged. The TRACG04 model is similar to TRACG02 but different. It has been updated to reflect current data.
3.13.3 Lines 23-35	The second set of curves is based on the Bingham pump curves and is consistent with the curves used as the default in RETRAN02 and RELAP/5-MOD1. The second set of curves is based on Westinghouse not Bingham.
3.13.4 Lines 32-33	The NRC staff will require that plant-specific pump data be verified and input for transient calculations. It should be clear that the pump data is the rated pump information not plant specific homologous curves. Also, the rated pump information is normally provided by the utility for a specific project or reload. GEH and GNF must assume that the utility has provided the data according to their quality procedures. We obviously consistency check the information based on our experience, but we don't literally verify the utility data.
3.15.2 Lines 7-8	These TRACG04 cases were run using program library Version 40. The version is not relevant to the approval of TRACG04 for AOOs and ATWS overpressure. Also, the cases were repeated several times with newer versions and the final validated code.
3.15.5.3 Lines 25-26	While the ICS is not credited in Appendix K LOCA analyses, it is an important system for mitigating AOOs. The LOFW is the only BWR2 AOO in which the ICS is important. The actuation logic is such that the transient duration for other AOOs is not long enough to see actuation.
3.19.2.1 Line 3	Shouldn't (Reference 3) go up by ATWS overpressure instead of being at the end of the sentence. Also, Reference 2 should be cited after the AOO.

Location	Comment
3.19.2.1-X	Generic question. Reference 3 is cited in numerous locations at the end of paragraphs. We do not see a specific relationship back to Reference 3 in these instances.
3.19.2.2 Lines 19-21	Updates to the TRACG nuclear methods to ensure compatibility with the NRC-approved PANACEA family of steady-state nuclear methods (e.g., PANAC11) **would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes** may be used for AOO or ATWS Suggest adding words consistent with 50.59 as was done in the ODYSY-1D SE.
3.19.2.3 Lines 30-33	Changes in the numerical methods to improve code **convergence would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes** may be used in AOO and ATWS overpressure licensing calculations without NRC staff review and approval.
3.19.2.4	Features that support effective code input/output **would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes may** be added without NRC staff review and approval. (Reference 3)
3.19.2.6	As a result, changes to the statistical methodology directly affect the results of safety **analyses and constitute a departure from a method of evaluation used in establishing the design bases or in the safety analysis**.
3.20.1 Lines 47 -48	While the power distribution in the core is flatter for EPU, the SLMCPR continues to be based on 0.1% of the number of rods in the core experiencing BT.
3.20.1 Page 62 Line 1	The NRC staff notes that interfacial phenomena have not been extensively studied……. This is not a precise statement. The two-phase industry has studied phase interface phenomena extensively. It is true that the two-phase data is largely limited to engineering level measurements that do not provide information that can uniquely qualify phasic models.
3.20.1 Page 62 Line 35	The TRACG04 analysis ~~stream~~**initialization**, however, is based on steady-state power distribution calculations performed using PANAC11. Editorial suggestion.
3.20.3 Page 67	Clarification. While the TOP limit does ensure the fuel melt limit is met, the TOP does not correspond to the limit. The TOP limit is a screening criterion that is generically established on a fuel product basis to provide a conservative method to evaluate transient response. The same thing is true of the MOP limit. It does not correspond to the strain limit but to a conservative screening criterion to assure the limit is met.
3.20.3 Page 69 Line 10	Grammar correction.
4.6 Lines 30-33	Until the NRC staff approves PRIME03 ~~and~~ **or** approves the GESTR-M benchmarks **per Appendix F of Reference 5**, the NRC staff will require ATWS overpressure analyses and AOO analyses be performed using the GESTR-M model. 3 comments. (1) Should and be or, (2) clarify where the subject benchmarks are coming from, and (3) the performance of the GESTR-M benchmarks per the Reference 5 Appendix F are not related to the GESTR-M parameters that are used in TRACG04.
4.8	See Comment 3.13.4 Lines 32-33 above.
4.13	Suggest using the same wording as 3.19.2.1.
4.14	Suggest using the same wording as 3.19.2.2.

Location	Comment
4.15	Suggest using the same wording as 3.19.2.3.
4.16	Suggest using the same wording as 3.19.2.4.
4.18	Suggest using the same wording as 3.19.2.6.
4.21	This statement from 3.20.3 clearly states the condition for releasing the 10% margin requirement or penalty when the conditions of 4.21 and 4.22 are met. The conditions specified in Section 4.21 and Section 4.22 of this SE complement Transient LHGR Limitation 3. Therefore, a 10 percent penalty is not required for TRACG04 methods when the conditions specified in Section 4.21 and Section 4.22 of this SE are met. Also, this is a quote from the Methods LTR SE, "If the void history bias is incorporated into the coupled neutronic and transient code set, then the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding strain is no longer required." It is suggested that the release, including the strain term, be added to the limitations and conditions for clarity.
	Related to the previous comment. The Methods SE has this statement concluding Limitation 10, "However, if GE does not adequately address the methodology deficiencies identified in LTR NEDC-33173P in the review of Supplement 3 of NEDE-32906P, the additional margins as described in this SE apply as appropriate." For clarity, Is it not appropriate to reference limitation 10 of the Methods SE and state in the TRACG04 SE, that the subject deficiencies have been met and no margins as described in the SE for NEDC-33173 apply. The limitation should be that the 10% margin should be applied if the historical void is not used. This is an input option.
4.23, 24, and 26	Why is it considered necessary to repeat these limitations from Reference 5? Limitation 4.2 reminds us generally that the SEs for Methods LTR and the MELLLA+ LTR must be met.
4.28	Same as above. 4.25 requires the use of the PRIME models when approved. 4.28 is different than 4.25 in that it applies a condition that, if necessary, should be specified in the PRIME SE. The specification of when PRIME must be used should be in the PRIME SE.
4.30	This is a pure TGBLA methods limitation stated in the SE for the Methods LTR. There is no need to repeat it here.

<table>
<tr><th>Location</th><th>Comment</th></tr>
<tr><td>4.31</td><td>Discussion
As of December 2008, the GEH standard production analysis TGBLA06 has been updated to TGBLA06AE6. The first suggestion is to modify the limitation such that E5 or later versions are defined to be acceptable.
The specific version used for a particular application could span back to TGBLA06AE4 because of the lead time for the design work and schedule delays. The core design work on some EPU and MELLLA+ projects was performed with a TGBLA06AE4 basis. In these cases, while the power uprate SAR is based on an equilibrium core with a TGBLA06AE4 basis, the lattices and core design activities for the cycle implementing the EPU or MELLLA+ change will have a lattice basis of TGBLA06AE5 or later. The core and safety analyses performed with TRACG04 for the SRLR will have a TGBLA06AE5 or later basis.
In the time frame of implementation of EPU or MELLLA+ projects, in NRC review or soon to be in review, that implement TRACG04 it is highly unlikely that any TGBLA06AE4 based bundles will remain. However, there are remote situations where a plant may need to install an older bundle which may have a TGBLA06AE4 basis.
Proposed
The application of TRACG04/PANAC11 is restricted from application to EPU or MELLLA+ plants until TGBLA06 is updated to TGBLA06AE5 or later in the GEH standard production analysis techniques. TGBLA06AE5 or later should be used for all new lattice physics constant generation. For core design activities in support of EPU or MELLLA+ submittals initiated after the date of this SE, TGBLA06AE5 or later should be used. TGBLA06AE5 or later should be used for all cycles implementing EPU or MELLLA+ based on application of TRACG04/PANAC11 for transient or ATWS overpressure analysis. (Appendix A. RAI 1)</td></tr>
<tr><td>Executive Summary</td><td>Section 4 Comments also apply to the Executive Summary.</td></tr>
<tr><td>References</td><td>Added MFN numbers for some GEH to NRC references where they were missing.</td></tr>
<tr><td>Appendix A
RAI-1
Lines 37-39</td><td>Same comment as 4.31 above</td></tr>
<tr><td>Appendix A
RAI-16
Pg A-10
Lines 27-30</td><td>The NRC staff finds that at high exposure, the TRACG04 model does not predict any influence on thermal conductivity by the gadolinia, whereas the FRAPCON3 model consistently predicts degradation in thermal conductivity with increasing gadolinia concentration.
The figures in MFN 08-053 show the gadolinia dependency.</td></tr>
<tr><td>Appendix A
RAI-16
Pg A-10
Lines 35-37</td><td>Therefore, the NRC staff does not accept the conclusion that gadolinia depletion under irradiation results in a negligible impact on fuel thermal conductivity at the end of life for the fuel,…..
We did not make such a concluding statement.</td></tr>
<tr><td>Appendix A
RAI-16
Pg A-10
Lines 42-43</td><td>(2) there is evidence that the new fuel thermal conductivity model remains non-conservative in the prediction of pellet temperature for gadolinia loaded fuel pins.
The evidence is only that FRAPCON shows a steeper reduction in thermal conductivity as gadolinia increases.</td></tr>
<tr><td>Appendix A
RAI-22
Pg A-16
Lines 1-2</td><td>For evaluations where the kinetics solver is disabled, the decay heat fission energy release values are based on historical values reported in GEH LTR NEDO-23729.
The decay heat values are always based on values from NEDO-23729. The reason for this was provided in the RAI response.</td></tr>
</table>

Location	Comment
Appendix A RAI-28 Pg A-18 Line 13	(Bingham) The TRACG04 default homologous pump curves are for a large Westinghouse pump manufactured in Cheswick, PA. The Bingham pump is only slightly different.
Appendix A RAI-29 Pg A-18 Line 39	The NRC staff evaluation of the applicability of TRACG04 to EPU and MELLLA+ mixed core analysis was reviewed separately and is documented in Section 4.20.5 of the subject LTR. May want to clarify the “subject LTR” more specifically.
Appendix A RAI-32b	PANCEA should be PANACEA
Appendix A RAI-32c Pg A-26 Line 39-43	………, the NRC staff is concerned that the inter bundle nuclear coupling may amplify the impact of errors in the predicted nodal reactivity feedback characteristics. The bundles are coupled by internodal neutron leakage. Potentially increased errors in neighboring bundle void reactivity feedback will have a direct effect on the efficacy of the code to accurately determine the limiting bundle transient response. This concern leaves the reader with no closure with respect to the importance of the concern. This discussion is tangential to the issue of the RAI regarding TRACG initialization. The concern by the NRC is new and should not be written into the SE. BWR operation is strongly coupled between the nuclear and thermal-hydraulics field. If the staff concern is retained, the reviewer should document its resolution by consideration that steady-state methods qualification (TIPs, gamma scans, eigenvalues) indicate that the coupled nuclear/thermal-hydraulics solution in GEH/GNF methods is satisfactory.
Appendix A RAI-32c Pg A-29 Line 14	Therefore, the NRC staff maintains that a threshold of significance of 0.005 remains appropriate when evaluating a potential bias. The cited 0.005 is the steady state level of significance. The transient basis has always been 0.01. The [[ ]] change in D/I for CPR is what was agreed to by the NRC staff in NEDE-32906P-A as being the appropriate level of significance for triggering additional NRC review.
Appendix A RAI-32c Pg A-30 Line 40-41	The NRC staff’s conclusions here are predicated on pressurization transients being the limiting transients in reload licensing analyses. While pressurization transients are often the most limiting, it is not always the case. The Page: 1 conclusions based on the example where the pressurization transient is limiting also apply for other AOO transients.
Appendix A References	Added MFN numbers for some GEH to NRC references where they were missing. Also, Reference 9 and 21 are the same.
Appendix B	If we are including the list why not include all through 57. Put an A,P after TRACG04 or eliminate the A.

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT (LTR) NEDE-32906P, SUPPLEMENT 3

“MIGRATION TO TRACG04/PANAC11 FROM TRACG02/PANAC10 FOR TRACG AOO AND

ATWS OVERPRESSURE TRANSIENTS"

GE-HITACHI NUCLEAR ENERGY AMERICAS, LLC (GEH)

PROJECT NO. 710

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Executive Summary

The U.S. Nuclear Regulatory Commission (NRC) staff has performed a safety evaluation (SE) of GE Hitachi Nuclear Energy Americas' (GEH's) licensing topical report (LTR) NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO [anticipated operational occurrences] and ATWS [anticipated transient without SCRAM] Overpressure Transients." The NRC staff conducted its review in accordance with NUREG-0800 "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants." In the course of its review the NRC staff identified areas where additional information was required to complete the review, and issued requests for additional information (RAIs) accordingly.

The NRC staff reviewed the current application for operating boiling water reactor (BWR) plant designs (BWR/2-6) over the current range of plant operating conditions, including extended power uprate (EPU) and maximum extended load line limit analysis plus (MELLLA+) operating domains. The NRC staff has found the methodology acceptable when exercised within a set of limitations and conditions. These limitations and conditions, and their technical bases are described at length in the body of this SE and are summarized in this Executive Summary. The limitations and conditions fall into five general categories: (1) applicability of historical limitations, (2) range of qualification, (3) code maintenance, (4) obsolescence of historical models, and (5) applicability to modern core operating strategies.

The NRC staff leveraged experience in its review of TRACG02/PANAC10 to complete the subject review. Therefore, several conditions regarding the previous application were found to equally apply to the current application for TRACG04.

The approval of methods is limited by the range over which any method is qualified. Extension of analytical codes beyond the scope of their qualification results in un-quantified uncertainties that may have significant ramifications on safety analyses. The range of applicability refers to plant designs, operating conditions, transient conditions, and the design of core internals (e.g., fuel bundle designs). It also takes into account specific modeling capabilities that may or may not be required for a specific set of transients.

In the maintenance of a code, the owner may make several adjustments and corrections to the code (e.g., input/output functions or numerical techniques to improve execution time) without impacting the basic solution technique. Therefore, while code updates are required periodically, special care must be taken to ensure that any changes do not adversely impact the code's ability to execute the methodology as the NRC staff has approved it.

It is common in codes that are continuously being improved, such as TRACG, to retain old models in updated code versions. In some cases these models may not accurately represent phenomena for changes in modern core designs or operating strategies. In these cases, the NRC staff imposes limitations and conditions on the use of certain models to address concerns given the entire scope of its generic approval.

The NRC staff has considered operational circumstances particular to EPU and MELLLA+ conditions in regard to specified acceptable fuel design limits and compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 Appendix A: General

Design Criteria for Nuclear Power Plants (GDC-10). In its consideration, the NRC staff determined conditions for licensing analyses performed for these plants.

Therefore, the NRC staff imposes the following limitations and conditions:

1 Historical Limitations and Conditions

All limitations and conditions imposed on TRACG02/PANAC10 documented in the NRC staff SEs attached to approved revisions to NEDE-32906P-A are considered applicable for TRACG04/PANAC11 unless otherwise specified in this SE. (References 2 and 3)

2 Interim Methods Limitations and Conditions

All limitations and conditions imposed on the TGBLA06/PANAC11 code system documented in the NRC staff SE for NEDC-33173P and the SEs for supplements to NEDC-33173P are applicable to their use in the TRACG04 code stream for AOO and ATWS overpressure calculations for EPU and MELLLA+ applications unless otherwise specified in this SE. (Reference 5)

3 Scope of Applicability Limitation

The approval of TRACG04/PANAC11 is limited to those specific applications reviewed by the NRC staff. The scope of review delineates those plant designs and conditions that the NRC staff considers to be the bounds of applicability. (Section 1.1)

4 Main Condenser Condition

Analyses performed for BWR/2-6 designs that include specific modeling of the condenser will require a plant-specific justification for its use. (Section 1.1)

5 Decay Heat Model Limitation

The NRC staff's acceptance of the TRACG04 decay heat model for simulating AOOs and ATWS overpressure does not constitute NRC staff acceptance of this model for loss-of-coolant accident (LOCA) applications. (Section 3.4.5)

6 Fuel Thermal Conductivity and Gap Conductance Condition

Until the NRC staff approves PRIME03 and approves the GESTR-M benchmarks, the NRC staff will require ATWS overpressure analyses and AOO analyses be performed using the GESTR-M model. Should the NRC staff subsequently approve PRIME03, this approval will constitute approval of the PRIME03 improved thermal conductivity model for use in TRACG04 for AOO and ATWS overpressure analyses when used with PRIME03 dynamic gap conductance input. (Section 3.10.5.3)

7 ATWS Instability During Pressurization Limitation

The NRC staff has not reviewed the TRACG04 code for modeling density wave instabilities during ATWS events. Therefore, while it is not expected for typically limiting ATWS overpressure scenarios, should TRACG04 predict the onset of an instability event for a plant-specific application, the peak pressure analysis must be separately reviewed by the NRC staff. (Section 3.10.5.3)

8 Plant-Specific Recirculation Parameters Condition

Licensing calculations require plant-specific recirculation parameters to be confirmed against plant data and input into the TRACG model. (Section 3.13.4)

9 Isolation Condenser System (ICS) Restriction

On a plant-specific basis, any licensee referencing TRACG04 for ICS BWR/2 plant transient analyses will submit justification of the applicability of the Kuhn-Schrock-Peterson (KSP) Correlation to model condensation in the ICS for pertinent transient analyses. This justification will include an appropriate sensitivity analysis to account for known uncertainties in the KSP Correlation when compared to pure steam data. The sensitivity of the plant transient response to the ICS performance is expected to depend on plant operating conditions, in particular the steam production rate. At EPU conditions the transient response is expected to be more sensitive to the ICS capacity given the increased steam flow rate at the same reactor core flow rate. The sensitivity is expected to be exacerbated at MELLLA+ conditions where the core flow rate is reduced. Therefore, licensees providing ICS BWR/2 plant-specific justification must provide such justification for each operating domain condition for which analyses are performed. (Section 3.15.5.3)

10 ATWS Transient Analyses Limitation

TRACG04 is not approved for analyses of reactor vessel ATWS overpressure after the point of boron injection. (Section 3.17.2 and Reference 3)

11 TRACG02 for EPU and MELLLA+ Limitation

The NRC staff has not generically reviewed the PANAC10 neutronic methods for application to EPU and MELLLA+ conditions. The NRC staff notes that initial comparisons between TRACG04 and TRACG02 for a representative EPU core (Section 3.18) indicate the TRACG02/PANAC10 methods are less conservative. Therefore, the NRC staff generic approval of TRACG04 for EPU and MELLLA+ licensing analyses does not constitute generic approval of TRACG02 for this purpose. (Section 3.18.9)

12 Quality Assurance and Level 2 Condition

TRACG04 must be maintained under the quality assurance process that was audited by the NRC staff as documented in References 25, 27, and 28 or a subsequent NRC-approved quality assurance process for engineering computer programs (ECPs) in order for licensees referencing the subject LTR to comply with the requirements of 10 CFR Part 50 Appendix B. (Section 3.19)

13 Code Changes to Basic Models Condition

Modifications to the basic models described in Reference 26 may not be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval. (Section 3.19.2)

14 Code Changes for Compatibility with Nuclear Design Codes Condition

Updates to the TRACG nuclear methods to ensure compatibility with the NRC-approved PANACEA family of steady-state nuclear methods (e.g., PANAC11) may be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval as long as the ratio of transient change in critical power ratio to initial critical power ratio (ΔCPR/ICPR), peak vessel pressure, and minimum water level show less than one standard deviation difference compared to the results presented in NEDE-32906P, Supplement 3. If the nuclear methods are updated, the event scenarios described in Sections 3.18.1 through 3.18.7 of this SE will be compared and the results from the comparison will be transmitted to the NRC staff for information. (Section 3.19.2)

15 Code Changes in Numerical Methods Condition

Changes in the numerical methods to improve code convergence may be used in AOO and ATWS overpressure licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50 Appendix B. (Section 3.19.2)

16 Code Changes for Input/Output Condition

Features that support effective code input/output may be added without NRC staff review and approval. (Section 3.19.2)

17 Updating Uncertainties Condition

New data may become available with which the specific model uncertainties described may be reassessed. If the reassessment results in a need to change a specific model uncertainty, the specific model uncertainty may be revised for AOO licensing calculations without NRC staff review and approval as long as the process for determining the uncertainty is unchanged and the change is transmitted to the NRC staff for information. (Section 3.19.2)

The nuclear uncertainties (void coefficient, Doppler coefficient, and SCRAM coefficient) are expected to be revised, as would be the case for the introduction of a new fuel design. These uncertainties may be revised without review and approval as long as the process for determining the uncertainty is unchanged from the method approved in this SE. In all cases, changes made to model uncertainties done without review and approval will be transmitted to the NRC staff for information. (Sections 3.19.2 and 3.20.2)

18 Statistical Methodology Limitation

Revisions to the TRACG statistical method may not be used for AOO licensing calculations without NRC staff review and approval. (Section 3.19.2)

19 Event-Specific Biases and Uncertainties Condition

Event-specific ΔCPR/ICPR, peak pressure, and water level biases and uncertainties will be developed for AOO licensing applications based on generic groupings by BWR type and fuel type. These biases and uncertainties do not require NRC staff review and approval. The generic uncertainties will be transmitted to the NRC staff for information. (Section 3.19.2)

20 Interfacial Shear Model Qualification Condition

Any EPU or MELLLA+ plant licensing analyses referencing TRACG04 methods for future Global Nuclear Fuel – Americas (GNF) fuel products shall verify the applicability of the interfacial shear model using void fraction measurements or an alternative, indirect qualification approach found acceptable by the NRC staff. (Section 3.20.1)

21 Void Reactivity Coefficient Correction Model Condition

When performing transient analyses with TRACG04, the revised void reactivity coefficient correction model must be activated. (Section 3.20.2 and Appendix A: RAIs 29 and 30)

22 Void Reactivity Coefficient Correction Model Basis Condition

Licensees referencing NEDC-32906P, Supplement 3, for licensing applications must confirm that the lattices used in the void coefficient correction are representative of the plant's fuel or update the lattices such that they are representative. (Section 3.20.2 and Appendix A: RAIs 29 and 30)

23 Transient Linear Heat Generation Rate (LHGR) Limitation 1

Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the thermal-mechanical (T-M) acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel melting and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for the both the UO_2 and the limiting GdO_2 rods. (Section 3.20.3)

24 Transient LHGR Limitation 2

Each EPU and MELLLA+ fuel reload will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the supplemental reload licensing report (SRLR) or core operating limit report (COLR), or it will be reported directly to the NRC staff as an attachment to the SRLR or COLR. (Section 3.20.3)

25 Fuel Thermal Conductivity for LHGR Condition

When TRACG04 is used to determine the limiting LHGR for transients, the GESTR-M thermal conductivity model must be used unless the NRC staff subsequently approves the PRIME03 models in a separate review. The fuel thermal conductivity and gap conductance models must be consistent. (Section 3.20.3)

26 Application of 10 Weight Percent Gadolinia Condition

Before applying 10 weight percent gadolinia bearing fuel to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for thermal and mechanical overpower conditions that bound the response of plants operating at EPU and expanded operating domains at the most limiting state points, considering the operating flexibilities (e.g., equipment out of service). (Section 3.20.3)

27 10 CFR Part 21 Evaluation of GESTR-M Fuel Temperature Calculation Limitation

Any conclusions drawn by the NRC staff evaluation of GEH's Part 21 report (Reference 41) or subsequent benchmarking of GESTR-M is applicable to this SE. (Section 3.20.3)

28 LHGR and Exposure Qualification Limitation

The conclusions of the plenum fission gas and fuel exposure gamma scans will be submitted for NRC staff review and approval, and revisions to the T-M methods will be included in the T-M licensing process. This revision will be accomplished through an Amendment to GESTAR II or in a T-M LTR review. Once the T-M LTR is approved, future license applications for EPU and MELLLA+ referencing LTR NEDE-32906P, Supplement 3 must utilize these revised T-M methods to determine acceptable LHGR limits. (Section 3.20.3)

29 Mixed Cores Limitation

Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GEH's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in NEDC-33173P (Reference 31) and additional subjects relevant to application of GEH's methods to legacy fuel. Alternatively, GEH must supplement NEDC-33173P (Reference 31) for application to mixed cores. (Section 3.20.5)

30 Fuel Lattices Limitation

The NRC staff did not assess the TGBLA06 upgrade for use with 11x11 and higher lattices, water crosses, water boxes, gadolinia concentrations greater than 8 weight percent, or mixed oxide (MOX) fuels at EPU or MELLLA+ conditions. For any plant-specific applications of TGBLA06 with the above fuel types, GEH needs to provide assessment data similar to that provided for the GNF fuels for EPU or MELLLA+ licensing analyses. (Section 3.20.5)

31 Modified TGBLA06 Condition

Application of TRACG04/PANAC11 is restricted from application to EPU or MELLLA+ plants until TGBLA06 is updated to TGBLA06AE5 in the GEH standard production analysis techniques. (Appendix A: RAI 1)

32 Transient Critical Power Ratio (CPR) Method Condition

Transient licensing calculations initiated from conditions where the minimum critical power ratio (MCPR) exceeds 1.5 require evaluation of the adequacy of the transient CPR method and justification if the improved transient CPR method is not used. (Appendix A: RAI 3)

33 Direct Moderator Heating Condition

Application of the TRACG04/PANAC11 methodology to fuel designs beyond the GE14 fuel design will require confirmation of the DMHZERO value. (Appendix A: RAI 5)

34 Specifying the Initial Core Power Level Condition

For each application of the TRACG ATWS methodology, it must be made clear exactly what power level is being used, not only the percentage of licensed power, but the actual power level. (Reference 3)

35 Submittal Requirements Condition

The staff also notes that a generic topical report describing a code such as TRACG cannot provide full justification for each specific individual plant application. When a licensee proposes to reference the TRACG-based ATWS methodology for use in a license amendment, the individual licensee or applicant must provide justification for the specific application of the code in its request which is expected to include:

1. <u>Nodalization</u>: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

2. <u>Chosen Parameters and Conservative Nature of Input Parameters</u>: A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.

3. <u>Calculated Results</u>: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure. (Reference 3)

36 MELLLA+ Limitations

The NRC staff imposes all limitations specific to transient analyses documented in its SE (Reference 49) for the review of NEDC-33006P (Reference 46) for the application of the TRACG04 method to EPU and MELLLA+ conditions. Some of the limitations from Reference 49 pertinent to MELLLA+ transient analyses include, but are not limited to: 12.1, 12.2, 12.4, 12.18.d, 12.18.e, 12.23.2, 12.23.3, 12.23.8, and 12.24.1. For reference, the complete list of MELLLA+ limitations is provided in Appendix D: SE Limitations for NEDC-33006P from Reference 49.

Conclusion

When the TRACG04/PANAC11 code stream is exercised within these limitations and conditions, the NRC staff has found that the code stream is acceptable for performing licensing calculations of AOO and ATWS overpressure events for the current operating fleet considering current expanded operating domains.

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1 INTRODUCTION

By letter dated May 25, 2006, General Electric Nuclear Energy (now GE-Hitachi Nuclear Energy Americas, hereafter GEH) submitted licensing topical report (LTR) NEDE-32906P, Supplement 3, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO [anticipated operational occurrence] and ATWS [anticipated transient without SCRAM] Overpressure Transients” (Reference 1), for review and approval.

The NRC staff has previously reviewed the TRACG02/PANAC10 code system for AOO and ATWS overpressure analyses (References 2, 3, and 4). In the conduct of its review the NRC staff leveraged experience in related reviews of the TRACG code for thermal-hydraulic and coupled neutron kinetic analyses.

In its review of NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains” (Reference 5), the NRC staff deferred conclusions regarding the applicability of TRACG to Extended Power Uprate (EPU) or Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating conditions to the subject review.

The NRC staff requested additional information to complete its review. GEH supplemented the content of the application with responses to this request by letters dated August 15 and December 20, 2007, and May 30, June 6, June 30, and July 30, 2008 (References 6, 7, 8, 9, 10, and 11, respectively).

1.1 Scope of Review

The NRC staff’s review of TRACG04 is limited to those changes in the TRACG04/PANAC11 methodology relative to the previously approved TRACG02/PANAC10 methodology. Similarly the NRC staff review is limited to the application of the methodology to AOO and ATWS overpressure transient analyses. Therefore, the NRC staff approval of the subject LTR does not constitute generic approval of the TRACG04/PANAC11 methodology to all transient applications. The NRC staff’s review, specifically, does not imply approval of the TRACG04/PANAC11 methodology for reactivity insertion accident analysis, time domain stability analysis, or ATWS evaluations following initiation of the standby liquid control system (other than to benchmark ODYN) or for ATWS events other than overpressure.

The NRC staff conducted its review according to the framework previously adopted for TRACG02/PANAC10 in accordance with the following NRC staff review guidance documents: NUREG-0800, “Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 15.0.2 (Reference 12), Draft Regulatory Guide DG-1096, “Transient and Accident Analysis Methods” (Reference 13), and NUREG/CR-5249, “Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident" (Reference 14).

As in any reactor analysis code package, models are implemented to analyze particular phenomena and components. In the current review the NRC staff performed a review of the TRACG04/PANAC11 methodology to perform calculations for boiling water reactor

(BWR)/2-6 plant designs. Therefore, the NRC staff approval of the subject LTR does not constitute generic approval of the TRACG04/PANAC11 methodology for all reactor types. Furthermore, the NRC staff notes from previous reviews that the condenser model in TRACG02 was found unacceptable by the NRC staff, therefore, analyses performed for BWR/2-6 designs that include specific modeling of the condenser will require a plant-specific justification for its use.

The NRC staff reviewed the applicability of the TRACG04 model description and qualification for the intended use to model AOOs and ATWS overpressure events for the range of current operating fleet conditions. These conditions are limited by the allowable operating domains for the operating fleet and, generically, include conditions of normal operation such as extended load line limit analysis (ELLLA), maximum extended load line limit analysis (MELLLA), increased core flow (ICF), maximum extended operating domain (MEOD), stretch power uprate (SPU), EPU, and MELLLA+. Therefore, the NRC staff reviewed the TRACG04/PANAC11 methodology as an alternative to the ODYN methodology currently approved for EPU and MELLLA+ AOO analysis.

TRACG04 includes several models that the NRC staff determined are not required to conduct the AOO and ATWS overpressure safety analyses, as stated in this safety evaluation (SE) (e.g., quench front model, hot rod model, relevant models for control rod drop accident (CRDA), loss-of-coolant accident (LOCA), stability, or ATWS/instability analyses, cladding oxidation rate model, and the revised uncertainties model). As such, these models were not reviewed in depth for these applications in this SE and approval of TRACG04 for AOO and ATWS overpressure analysis does not constitute approval of these models for any conditions or analyses other than AOO or ATWS overpressure analyses.

The NRC staff's conclusions regarding the acceptability of the TRACG04/PANAC11 are limited to those plant conditions bounded by the aforementioned expanded operating domains. The models and their qualification are limited in terms of the range of applicability based on the thermal-hydraulic and neutronic characteristics of the available data and plant conditions. The applicability of TRACG04/PANAC11 to analyze transients initiated from initial conditions for operating strategies outside of the expanded operating domains currently employed by the operating fleet will require specific justification.

2 REGULATORY EVALUATION

To establish a licensing basis, applicants must analyze transients in accordance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria for Nuclear Power Plants (GDC-10) "Reactor Design" and 10 CFR 50.34, "Contents of construction permit and operating license applications; technical information;" and, where applicable, should address NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," issued November 1980 (Reference 15). The NRC staff reviews the evaluation model to ensure that it is adequate to simulate the transient or accident under consideration. This includes a review of methods to estimate the uncertainty in the calculation.

The NRC staff provided guidance for applicants to meet general requirements of a thermal-hydraulic analysis computer code in Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," (Reference 16) and NUREG-0800, Section 15.0.2

(Reference 12). References 12 and 16 describe acceptable approaches by which the calculated uncertainty in the analysis methodology can be assessed. They express a preference for the code scaling, applicability, and uncertainty (CSAU) methodology (Reference 14) as the means for applicants to determine the uncertainty in a code calculation. Specific regulatory criterion for AOO analysis is described below.

GDC-10 requires:

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

GEH uses the TRACG code to ensure that safety limits—such as minimum critical power ratio (MCPR), maximum linear heat generation rate (MLHGR), and downcomer water level—are met during anticipated transients.

Specific regulatory criteria for ATWS include 10 CFR 50.62 and numerous GDC specified in SRP Section 15.8. Insofar as they pertain to the subject review, the specific applicable regulatory criterion is described below.

GDC-14 requires:

> The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GEH uses the TRACG code to calculate the peak vessel pressure to ensure vessel integrity during ATWS pressurization events.

3 <u>TECHNICAL EVALUATION</u>

3.1 Overview of the TRACG Methodology

TRACG is a transient analysis code derived from the original TRAC family of codes. TRACG is a coupled thermal-hydraulic neutron kinetics analysis system developed by GE for BWR applications. The basic thermal-hydraulic model is a two-fluid model explicitly represented in the code with six conservation equations and appropriate closure relationships. The thermal-hydraulic model is coupled with a three-dimensional neutron kinetics engine based on the PANAC11 nuclear design code.

TRACG is initiated by inputting the PANAC11 generated wrap-up file, which includes the steady-state power distribution on a nodal basis as well as the nodal response surfaces for nuclear parameters (infinite eigenvalue, lattice rod powers, migration area, etc.). The basic code structure is based on an eleven step iterative process that couples the neutronic and thermal-hydraulic solvers, as follows:

1. Obtain the initial static flux and nodal conditions from a steady-state PANACEA calculation. Obtain a converged thermal-hydraulic solution based on the fixed static power distribution.

2. Calculate steady-state nodal delayed neutron precursor concentrations.
3. Increment the time step and calculate the thermal-hydraulic response.
4. Update the nodal void and fuel temperature values in the neutronic model based on the thermal-hydraulic calculation.
5. Move control rods consistent with the new time (in cases of SCRAM).
6. Determine nodal nuclear parameters based on updated thermal-hydraulic and control state based on PANAC11 response surfaces. Determine the source distribution given the previous time step flux distributions and delayed neutron precursor concentrations.
7. Solve the neutron diffusion equation at this time step based on fixed thermal-hydraulic conditions.
8. Solve the delayed neutron precursor equations at this time step.
9. Determine the nodal powers at this time step.
10. Calculate decay heat to determine the total power for next iteration of the thermal-hydraulic calculation.
11. Return to 3 to continue the transient evaluation.

The TRACG kinetic solver calculates the nodal powers for the same nodalization as the steady-state PANAC11 wrap-up file; however, TRACG solves the thermal-hydraulic conditions based on a coarser radial nodalization by lumping fuel channels into groups. The NRC staff has previously reviewed the approach for radial channel grouping and the assignment of nodal powers to the groups and found that the TRACG model adequately represents the core and bundle conditions during transient evaluations.

3.2 Summary of Previous Review Findings

3.2.1 Phenomena Identification and Ranking Table (PIRT)

During a nuclear power plant accident or transient, not all phenomena that occur influence the behavior of the plant in an equal manner. A determination must be made to establish those phenomena that are important for each event and various phases within an event. The phenomena are compared to the modeling capability of the code to assess whether the code has the necessary models to simulate the phenomena. Most importantly, the range of the identified phenomena covered in experiments or test data is compared to the corresponding range of the intended application to ensure that the code has been qualified for the highly ranked phenomena over the appropriate range. Development of a PIRT establishes those phases and phenomena that are significant to the progress of the event being evaluated.

The PIRT for TRACG04/PANAC11 is the same as employed for the CSAU based review for TRACG02/PANAC10. The PIRT is independent of the code system. The NRC staff has previously reviewed and approved the AOO PIRT (Reference 17). Therefore, the NRC staff finds that the PIRT is acceptable for reference in the subject LTR.

3.2.2 Code Applicability

TRACG is a two-fluid code capable of one-dimensional and three-dimensional thermal-hydraulic representation along with three-dimensional neutronic representation. The code is designed to perform in a realistic manner with conservatism added, where appropriate, via the input specifications. An analysis code used to calculate a scenario in a nuclear power plant should use many models to represent the thermal-hydraulics and components. Those models should include the following four elements:

(1) Field equations provide code capability to address global processes.

(2) Closure equations provide code capability to model and scale particular processes.

(3) Numerics provide code capability to perform efficient and reliable calculations.

(4) Structure and nodalization address code capability to model plant geometry and perform efficient and accurate plant calculations.

The NRC staff performed an extensive review of the thermal-hydraulics models and their applicability to the GEH passive, natural circulation BWR design (ESBWR) for LOCA events and containment analysis in Reference 18 and ESBWR stability in References 19 and 20. During its review of TRACG04/PANAC11 for application to AOO events for the operating fleet, the NRC staff leveraged this previous review experience and focused on models that were not previously reviewed or that have been updated since previous reviews. The TRACG neutron kinetics models have been updated since the review of TRACG for AOOs in the BWR/2-6, and the models are now based on PANAC11 methods. In addition, the NRC staff focused on the review of cross-section generation using TGBLA06, and items related to expanded operating domain applicability.

3.2.3 Statistical Methodology

The methodology for the statistical combination of model uncertainties in the uncertainty determination for TRACG04 remains unchanged from those methods used in the determination of uncertainty for TRACG02. The NRC staff has previously reviewed this methodology (Reference 17) and found it to be acceptable. The code update from PANAC10/TRACG02 to PANAC11/TRACG04 is not a significant enough deviation to invalidate the basis of the statistical method. Therefore, the NRC staff finds this methodology acceptable for TRACG04.

3.3 PANAC11 Kinetics Model

3.3.1 Description of the Model

The TRACG04 3D kinetics model is based on the PANAC11 nodal diffusion code. The PANAC11 code structure is exactly reproduced in the TRACG04 code. PANAC11 was originally reviewed by the NRC staff as part of an audit at Global Nuclear Fuel – Americas (GNF) (Reference 19). Subsequently, a description of the PANAC11 nuclear methods was submitted to the NRC staff as part of the ESBWR design certification application (Reference 22). The NRC staff reviewed the description of the model in Reference 22 to determine the applicability of the PANAC11 based 3D kinetics solver in TRACG04 to BWR AOO and ATWS overpressure applications.

3.3.1.1 Neutronic Model

The nuclear model in PANAC11 is a static, one-and-a-half group, coarse mesh, nodal diffusion model. The nuclear model begins with three-group theory. The three-group equation is collapsed to a one-and-a-half group equation by assuming that each group has the same buckling. For each node, the one-and-a-half group equation is integrated and solved. A piecewise linear approach is used to determine the nodal flux in terms of the six surface currents. Current continuity and nodal diffusion coefficients are used to eliminate the surface currents and solve for the nodal flux in terms of the neighboring nodal fluxes.

The integrated surface currents are incorporated into the nodal spectral collapsing in order to account for spectrum hardening or softening as a result of neutronic coupling between nodes. The ratio of the infinite thermal to fast flux is corrected according to the integrated neutron balance for each group (Reference 19).

The node size is selected to account for the nuclear coupling between nodes as it relates to neutron transport. In general, the mean free path for a thermal neutron is very short, so the nodal size is selected based on the mean free path for fast neutrons and is about six inches.

Aside from the solution for the flux, there are various feedback mechanisms that must be accounted for within the nuclear model to determine nodal nuclear parameters. These include: void effects, Doppler effects, exposure effects, control rod effects, xenon effects, and reflector effects. GEH provided specific details as to how the simulator accounts for each of these effects in Reference 22, and they are described separately below.

3.3.1.2 Instantaneous Void and Void Exposure History Effects

The solution to the coarse mesh, one-and-a-half group, nodal diffusion equation depends on the converged thermal-hydraulic solution as well as the nuclear parameters in each node as determined by the state point in each node. Each node is characterized by its exposure, its moderator density history, and instantaneous void fraction. These parameters characterize the spectrum and spectral history and burnup for the fuel in each node.

The thermal-hydraulic model is substantially similar to the TRACG02 thermal-hydraulic model. The nuclear parameters for each node are based on the results of the lattice physics analyses and collapsed nodal cross-sections; however, the lattice physics calculations are carried out for three depletion histories with branch cases. These lattice parameters are stored in a table, and extrapolation techniques are used to predict the nodal parameters for node conditions other than those used in the lattice depletion analyses.

GEH describes the technique for accounting for the neutron spectrum and spectral history in Section 1.4.2 of Reference 22. The node's neutronic properties are characterized by four parameters: migration area, diffusion coefficient, infinite eigenvalue, and infinite lattice epithermal fission migration area correction. These properties are collapsed from three-group input parameters that are fit with polynomials and by Lagrangian interpolation of lattice physics analytical results.

The lattice physics infinite eigenvalue inputs are fit with several parameters, including the instantaneous relative water density, the integrated water density history (or the exposure weighted average relative water density), and the exposure. The remaining parameters are fit according to the instantaneous relative water density.

PANAC11 uses a spectral correction term to account for leakage effects. In the one-and-a-half group formulation of the diffusion equation, [[

]] This effect is most pronounced near the core periphery, where epithermal neutrons preferentially leak out of the core.

3.3.1.3 Doppler or Fuel Temperature Effects

The Doppler effect accounts for changes in nodal reactivity based on changes in fuel temperature. The Doppler effect is taken into account in the PANAC11 core simulator by fitting the lattice parameter infinite eigenvalue as a function of the fuel temperature based on branch case lattice analyses. The PANAC11 steady-state predicted fuel temperature is translated with the PANACEA wrap-up file to the TRACG04 calculation during initialization. [[

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TRACG04 in transient mode will calculate the transient fuel temperature, but the Doppler coefficient is based on the PANAC11-predicted Doppler response to temperature, which is based on the PANAC11-predicted steady-state temperature. The TRACG04 model for fuel conductivity has been updated. Therefore, the NRC staff review of the transient Doppler effect is documented in Section 3.10 of this SE.

3.3.1.4 Control Rod Effects

Control rod effects are taken into account by tabulating collapsed three-group lattice data for the controlled and uncontrolled states. If a node is uncontrolled through its exposure history, then the uncontrolled lattice data are used, and the inverse is true for fully controlled nodes. In the cases when a node has a partially inserted control rod, linear interpolation is used to determine the nodal infinite eigenvalue, diffusion coefficient, and migration area. The epithermal fission migration area correction is not sensitive to the control state, and is therefore not tabulated separately for the controlled and uncontrolled state.

The effects of the control history are also accounted for within PANAC11. The control blade history over exposure affects the nodal nuclear properties and is accounted for in PANAC11 by using a procedure for combining lattice parameters that were generated for both controlled (or bladed) and uncontrolled (or unbladed) depletion calculations. TGBLA06 is used to calculate the standard void depletion histories as well as bladed depletion histories, at each exposure point TGBLA06 is used to calculate a branch case where the control state is switched. These data form a basis for calculating the nodal nuclear characteristics considering both the historical effect of the control history as well as the instantaneous effect.

For nodes within PANAC11 that are exposed while in the controlled state, the nodal nuclear parameters are determined by weighted averaging the bladed and unbladed lattice parameters from TGBLA06 based on empirically derived constants and the exposure averaged control state.

The constants are determined by comparing PANAC11 nodal parameters with explicit modeling of a control history within TGBLA06 and comparing the eigenvalue and other nuclear parameters.

3.3.1.5 Spatial Xenon Effects

PANAC11 specifically tracks xenon concentration because it is a very strong thermal neutron absorber. The method for xenon tracking employed in PANAC11 is to use the neutron flux solution to predict the steady-state xenon concentration based on a reference concentration for a given neutron flux. The production and loss terms are balanced to determine the equilibrium xenon concentration based on different neutron fluxes. The ratio of the nodal xenon steady-state concentration to the reference (at nominal power density) concentration is weighted by a reactivity worth factor.

The infinite lattice eigenvalue used for the nodal diffusion calculation is then adjusted by a fractional amount to account for this deviation in xenon concentration for the reference value. The xenon reactivity worth factor is evaluated at rated power density and represented as a function of exposure, water density, control state, and fuel type.

The xenon concentration is predicted based on the assumption of steady-state operation, and therefore, the standard PANAC11 method (NITER=0) cannot be used to predict the transient xenon evolution for plant conditions such as startup. PANAC11 has a separate option for transient xenon calculations (NITER=17). When performing transient evaluations of the xenon concentration during slow plant transients the code must be run in NITER=17 mode. The PANAC11 engine in TRACG04 does not include a transient xenon model; however, standard AOO and ATWS overpressure analyses do not persist for sufficient duration for the evolution of the xenon distribution to appreciably impact transient results. Therefore, the NRC staff did not review the application of TRACG04 to simulate transients that are of a long duration when compared to the xenon half life (~6.7 hours) as part of the subject review.

3.3.1.6 Reflector Boundary Conditions

Mixed type boundary conditions are employed for the radial and axial reflectors. [[

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3.3.2 Qualification of the Model

GEH qualified the three-dimensional depletion method against data obtained from numerical benchmarks and operating BWRs. The qualification studies conducted consist of:

(1) Simulation comparisons to fine mesh three-dimensional diffusion models;
(2) Comparisons to gamma scan data;
(3) Simulation and tracking of nine operating cycles of five plants;
(4) Cold critical measurements taken during seven cycles at two plants; and
(5) Comparisons to traversing in-core probe (TIP) data.

3.3.2.1 Fine Mesh Three-Dimensional Model

As described in Section 1.6.2 of Reference 22, GEH performed 23 separate core calculations using the PANAC11 core simulator and the DIF3D code. DIF3D is a finite difference, multi-group, diffusion theory code developed by Argonne National Laboratory. The comparisons are meant to illustrate the efficacy of the diffusion theory models implemented in the PANAC11 code; therefore, both PANAC11 and DIF3D draw nuclear data from TGBLA06 output, for consistency. In the case of PANAC11, TGBLA06 branch cases and depletion histories are used to construct parametric fitting functions for lattice cross-sections based on void and exposure history as well as local environmental conditions within the node. In DIF3D the core is modeled such that there is a mesh cell for each corresponding TGBLA06 homogenized pin cell.

PANAC11 and DIF3D were used to calculate power distributions and eigenvalues for 23 core configurations. These 23 cases represented five cores (BWR/4, BWR/5, or BWR/6) with a variety of lattice types and core sizes ranging from 240 to 748 bundles. The plants, labeled A through H in Reference 22, are representative of: [[

]]

GEH compared eigenvalues between the two codes and found very good agreement between the two approaches, with a very small standard deviation between calculated results. The comparison of the nodal power distribution predicted with PANAC11 and DIF3D show root mean square (RMS) differences of [[ ]] percent and the peak-to-peak nodal power differences averaged over all cases is approximately [[ ]] percent. Comparisons of all cases show that for Plant A [[ ]] the difference between the two codes is greatest - [[
]]

GEH also demonstrates the efficacy of the lattice homogenization by comparing nodal powers. In this case the nodal power calculated by DIF3D is the summation of the power produced in each mesh within a corresponding PANAC11 node. As the peak-to-peak nodal power and nodal powers compare very well between the two codes, the comparison indicates that the method for homogenization of the assembly in PANAC11 captures the effect of the lattice flux distribution on nodal parameters.

3.3.2.2 Gamma Scan Measurements

As described in Section 1.6.3 of Reference 22, GEH performed a cycle analysis for the Hatch reactor plant for its first and third cycles using specific input into the TGBLA06 code and PANAC11 core simulator. The purpose of these calculations was to calculate the concentration of barium-140 in the fuel assemblies. The barium-140 concentration was calculated based on the PANAC11 predicted exposure history and power distribution over the last 60 days of the cycle for the first and third cycles at Hatch.

For these cycles, gamma scan measurements were made on the fuel assemblies at Hatch. Gamma scanning is a technique that measures the gamma decay of lanthanum-140. Lanthanum-140 comes from the beta decay of the fission product barium-140. By measuring the relative signal along the axial length of the bundle, the lanthanum-140 concentration at the time of the scan can be determined. The lanthanum-140 concentration is then used to determine the concentrations of barium-140 that were present at the end of cycle (EOC) based on the half life. The barium-140 concentrations are then compared to the concentrations derived from calculations based on the PANAC11 power distribution. The results of these comparisons are used to determine the difference in the PANAC11 predicted power distribution and the actual power distribution near the EOC.

Gamma scans afford qualification of the nuclear methods capability for calculating the radial distribution of power in the four bundles surrounding a TIP string.

GEH corrected the Hatch gamma scan data for the time between the EOC and the measurement and compared the code calculated barium-140 concentration to the concentration determined from the gamma scan analysis. The results showed excellent agreement. The nodal RMS differences for Cycle 1 and Cycle 3 are less than [[

]] (Reference 22). The nodal RMS differences based on the Hatch gamma scans are consistent with the nodal RMS differences calculated according to core follow TIP comparisons reported in Reference 19.

3.3.2.3 Critical Eigenvalue

As described in Section 1.6.4 of Reference 22, plant tracking calculations were also performed for five plants over several cycles. These calculations were used to determine the predicted core eigenvalue based on input boundary conditions taken from plant instrumentation, specifically the reactor power, flow, and pressure. The comparisons were made over the course of operating cycles, and therefore, the code prediction of the eigenvalue is compared to unity. A summary of the plant tracking cases is provided in Table 3.3.2.1.

For all of the plants considered, the PANAC11 code predicted core eigenvalues that were near unity. However, a consistent trend for all of these plants was observed where the eigenvalue was over-predicted at the beginning of cycle (BOC). This trend is linear and consistently linear across a large variety of plants ranging in size and power level. Therefore, while the eigenvalue is not exactly predicted, the trend is consistent and easily taken into account. Additionally, the error, despite the trend observed, is still only a slight deviation from unity.

The consistency of the eigenvalue trend confirms that PANAC11 methodology performs similarly for a variety of BWR cores. It also confirms that PANAC11 can predict core eigenvalues for operating plants within a small, predictable error band.

The TRACG04 eigenvalue is based on normalization to the steady-state design basis eigenvalue. In general, a design basis eigenvalue is determined prior to fuel load to characterize the bias in core eigenvalue predicted by PANAC11. The bias is incorporated similarly in TRACG04, such that the TRACG04 model will calculate a core eigenvalue of unity when the PANAC11 predicted eigenvalue is equal to the design basis value at hot conditions.

3.3.2.4 Cold Critical Measurements

As described in Section 1.6.5 of Reference 22, GEH provided PANAC11 calculations for cold critical conditions and comparisons to measurements for the plants and cycles shown in Table 3.3.2.1.

Cold critical data is used from operating plants at each point in the cycle where a cold critical test was performed. Cold critical eigenvalue data for each of the cycles studied is provided in Table 1-16 of Section 1.6.5 of Reference 22.

Calculated cold critical eigenvalues are obtained by running PANAC11 at the same exposure and with the critical rod patterns used in the test. The eigenvalue calculated by the simulator is then corrected for the positive period measured during the test. The data in Table 1-16 of Reference 22 includes both distributed control rod patterns (as would occur during normal startup or shutdown) and local criticals where control rod(s) are withdrawn in a particular core location.

The results of this sample of cold critical results are summarized in Table 1-16 of Reference 22. The results of the cold critical comparisons provided in this section are indicative of the core simulator code's predictive capability over a wide range of plants and core designs. The uncertainty in the results is consistent with expectations and in addition to the nuclear methods uncertainty, includes all other uncertainties (i.e., plant instrumentation, manufacturing, etc.) associated with the design and operation of a nuclear reactor.

The cold critical testing also indicates a consistent bias that is captured by a design basis cold critical eigenvalue. The cold critical eigenvalue is not necessary to compensate in the TRACG04 model for AOO and ATWS overpressure transients which are initiated from hot full power conditions. The shutdown margin is verified by PANAC11 such that subcriticality following a SCRAM is ensured based on technical specification (TS) requirements; therefore, the cold critical measurements provide qualification of the PANAC11 calculational efficacy for determining control blade worth under heavily bladed and high water density conditions. These conditions encompass the conditions in TRACG for a SCRAM during pressurization events and provide an adequate basis to accept the PANAC11 calculation of control blade worth over a large range of plant spectral conditions.

3.3.2.5 TIP Measurements

As described in Section 1.6.6 of Reference 22, GEH used the PANAC11 core simulator to simulate the TIP measurements for four plants and eight cycles (Plants A, B, C, and E from Table 3.3.2.1). GEH provided a table showing the RMS differences between the TIP measurements for axial power shapes as well as bundle (or radial) power shape. The difference was approximately [[ ]] for the bundle RMS. These calculations were performed without any kind of adaption in PANAC11, and therefore indicate both good agreement in static calculations, but also indicate that there is essentially no degradation in modeling performance during cycle exposure for a large range of BWR operating conditions.

3.3.2.6 Updated Experience Database

In order to qualify the current TGBLA06/PANAC11 codes for expanded operating domains GEH has provided references to an updated experience database. This information contains additional qualification comparisons for the nuclear methods. These qualifications were documented in GEH's response to the NRC staff's RAI during the LTR NEDC-33173P review, specifically responses to NRC RAI numbers 25 and 27 in the letter dated April 8, 2005 (Reference 23). Based on the date of this submittal, these qualification calculations were performed using PANAC11AE7 and TGBLA06AE4. The purpose of these calculations was to illustrate the ability of the nuclear design codes to predict cycle follow data for EPU plants. The response to NRC RAI 25 in Reference 20 compares eigenvalue tracking and TIP data for five EPU BWR plants to predictions (without adaption) made with the TGBLA06AE4/PANAC11AE7 code suite. The response to NRC RAI 27 in Reference 23 compares calculated and measured TIP readings based on collections of limiting four bundle locations for each of the plants and cycles considered.

Both of these RAI responses are summarized in this section as they relate to the qualification of the nuclear steady-state code system against operating plant data. Additionally, they provide a basis for the applicability of the nuclear design methods to the power and flow range of operation for EPU and MELLLA+ plants. A subset of these data is reproduced in Reference 22.

3.3.2.6.1 Summarized Response to NRC RAI 25

The nuclear design methods (TGBLA06/PANAC11) were evaluated for high power-to-flow ratio cores and the results were compared to plant data on the bases of hot critical eigenvalue tracking, cold critical eigenvalue, and unadapted comparison to TIP measurements. Five plants over various cycles were considered as part of the study. These plants are described in Table 3.3.2.6.1. The power densities for these plants range from 51.7 kW/liter to 62.9 kW/liter.

In each reference plant study, a cycle follow analysis was performed using TGBLA06AE4 coupled with PANAC11AE7. The calculations were performed with plant adaption disabled in order to compare purely predictive PANAC11AE7 results with TIP measurements. The hot eigenvalue results are shown in Tables 25-2 through 25-10 of Reference 23. The RMS difference between the calculated hot critical eigenvalue and unity was shown to be approximately [[ ]]. The value of [[ ]] is consistent with the predictive capability shown in Reference 19. For most of the studied cycles the eigenvalue trends are fully consistent with the trends and biases expected for non-extended range operating BWRs. [[ ]]

The results of the hot critical eigenvalue comparison indicate that the trend in eigenvalue through exposure does not appear to be a function of the power density, power-to-flow ratio, or core average void fraction. However, [[ ]]

The cold critical eigenvalues were also compared. The cold critical eigenvalue comparisons were carried out against plant configurations where enough control blades were withdrawn at cold conditions for the reactor to be critical, or to have a very large positive period. In cases where the period was positive, the eigenvalue is period corrected to constitute the measured quantity. The measured quantities are in turn compared for the predicted quantities for each plant and cycle. The results indicate good agreement with an RMS difference of [[ ]]. To evaluate the effect of power density, the three highest power density plants were considered separately. For the three high power density cases, the RMS difference was found to be [[ ]]

To evaluate the effect of cycle length (or cycle energy), the plant operating on a one-year cycle (Plant C) was considered separately. The RMS difference in cold critical eigenvalue was found to be [[ ]] (essentially the same as considering all of the reference plants). Therefore, the cold critical eigenvalue comparisons indicate that the predictive capability of PANAC11 does not appear to be a function of the power density or cycle length. Also, the calculational accuracy is essentially consistent with the expected accuracy for non-extended range operating BWRs.

Finally, direct comparisons with TIP measurements were conducted. Plants A, B, C, and D have gamma TIP instruments whereas Plant E has thermal neutron TIP instruments. For the gamma TIP plants, the comparisons of fully predictive calculated TIP responses (CALTIP) when compared to the measured TIP response (PCTIP) indicate that the TIP uncertainty increases with increasing power-to-flow ratios. A linear trend line through the gamma TIP comparison study results appears to indicate [[ ]] The weighted RMS differences from the current study indicate good agreement with data [[ ]] with only a few exposure points [[ ]] for the cases considered. This is consistent with the improvement in calculational accuracy over TGBLA04/PANAC10 described in Reference 19.

3.3.2.6.2 Summarized Response to NRC RAI 27

In addition to the eigenvalue and TIP calculations provided in response to NRC RAI 25 (Reference 23), GE performed a series of predictive calculations with TGBLA06AE4/PANAC11AE7 to illustrate the efficacy of the code system to predict cycle characteristics for EPU plants. The same plants as referenced above were considered in the study.

GE provided the calculations of the predicted TIP readings without adaption for various exposure points during the cycle where TIP data were available. At each exposure point, the predicted integrated radial response, the axial response, and the nodal response were compared to the data. At each exposure point the highest power four bundle instrumented cell was determined. Table 3.3.2.6.2.1 below provides a summary of the differences in the calculated and measured TIP responses for the highest power instrumented four bundle cell, for each plant, at each exposure point. The four bundle power (P4B) listed in Table 3.3.2.6.2.1 is the highest relative four bundle power, where the core average bundle power is unity.

Only the highest power four bundle cells were considered in the comparison, though data was provided for each TIP string. The nodal RMS difference for the highest power four bundle cell is the metric of interest as it can be directly compared to the nodal RMS difference in TIP response quoted in the original submittal dated July 2, 1996 (Reference 34) for the improved physics methodology (Reference 21).

When the nodal RMS differences are averaged over the expanded operating domain plants, the result is approximately[[
]] quoted in Reference 19. This appears to indicate that the accuracy of predictive core follow analysis is essentially the same for expanded operating domain plants as for the plants considered in the original qualification basis for the improved steady-state methods.

Table 3.3.2.1: Plants and Identification for PANAC11 Qualification

Plant	ID	Thermal Power [MWth]	Cycles
[[	A	[[	18 and 19
	B		9 and 10
	C		30 and 31
	D		15
]]	E	]]	9 and 10

Table 3.3.2.6.1: Reference Plants in the MELLLA+ Methods Study

Plant	Cycle	Thermal Power	%OLTP	Core Size	Flow Range	Cycle Length	Loaded Fuel Type	Average Enrichment
		[MWth]		[bundles]	[Mlbm/hr]	[years]		[w/o]
A	18	[[	120	[[			GE14	4.02
A	19		120				GE14	4.11
B	9		105				GE14	4.16
B	10		105				GE14	4.13
C	30		110				GE14	4.19
C	31		110				GE14	4.19
D	15		120				GE14	4.21
E	9		120				GE14	3.89

E	10	]]	120			]]	GE14	4.21

Table 3.3.2.6.2.1: Nodal TIP Prediction vs. TIP Data for EPU Plants

Plant	Cycle	Exposure	P4B	Radial Difference (integrated)	Axial Difference RMS	Nodal Difference RMS	Approximate Power Shape
		[MWD/ST]		[%]	[%]	[%]	
A	18	2344	[[				Bottom Peaked
		4184.2					Bottom Peaked
	19	239.6					Middle Peaked
		4505.5					Bottom Peaked
		9015.6					Double Humped[1]
B	9	541					Middle Peaked
		10336					Bottom Peaked
		15990					Top Peaked
	10	191					Middle Peaked
		5774					Bottom Peaked
		8681					Bottom Peaked
C	30	191					Bottom Peaked
		4006					Bottom Peaked
		6914					Middle Peaked
	31	496					Bottom Peaked
		3916					Bottom Peaked
		7277					Middle Peaked
D	13	130					Bottom Peaked
		8150					Bottom Peaked
		14032					Middle Peaked
	14	246					Bottom Peaked
		5569					Bottom Peaked
		10850					Bottom Peaked
E	9	248					Bottom Peaked
		9314					Double Humped
		15043					Middle Peaked
	10	137					Bottom Peaked
		3579					Bottom Peaked
		8449				]]	Bottom Peaked

[1] Double Humped here does not refer to specific determination against the double humped power shape criterion. This description of the axial power shape refers to axial TIP traces where there are two local peaks in the power of approximately the same magnitude above and below the core mid-plane based on visual inference.

3.3.3 Implementation of the PANAC11 Method in TRACG04

Nuclear data generated from TGBLA06 and PANAC11 are used to perform transient analyses by providing input to the TRACG transient reactor analysis code. Therefore, this section of the SE addresses the adequacy of the PANACEA generated nuclear data for performing transient analyses.

In response to RAI 21.6-85 on the ESBWR in the letter dated June 21, 2007, (Reference 24), GEH provided a table of contents to a PANACEA wrap-up file. The NRC staff reviewed the contents to determine if the PANACEA wrap-up file contained sufficiently detailed parameters to allow for the initialization of the TRACG power distribution, while maintaining a sufficiently detailed characterization of the nuclear parameters to allow the TRACG kinetics solver to model the neutronic feedback. The wrap-up file contains both the functional cross-sections and power distribution; therefore, in the initialization procedure the functional cross-sections are preserved, allowing for accurate feedback modeling. Therefore the NRC staff determined that sufficiently detailed nuclear information is conveyed from the PANACEA wrap-up file to TRACG to both initialize the model and provide for acceptable kinetic feedback modeling.

TRACG analyses are initialized to the PANACEA calculated steady-state conditions through the wrap-up file. During the steady-state initializing calculation with TRACG, updates to the core power distribution are disabled such that TRACG converges on a thermal-hydraulic condition that matches the PANACEA wrap-up file power distribution (Reference 25). The wrap-up file contains nuclear parameters for each neutronic node. Each neutronic node is assigned to thermal-hydraulic channels through user specification and specific TRACG channel grouping. The TRACG 3D kinetics model is based on the same neutronic nodalization as present in PANACEA (Reference 26).

In the initialization process there are several differences in the TRACG thermal-hydraulic model and the PANAC11 model. Additionally, the nodalization for the neutronic model is not the same as the TRACG thermal-hydraulic model. Due to these differences the TRACG initialization process to develop the steady-state condition for stability evaluation employs means for adjusting the neutronic model to accommodate the steady-state thermal-hydraulic solution.

PANACEA calculations are performed such that the neutronic solution is for a predetermined hot critical eigenvalue that is often different from unity to account for modeling biases. The hot critical eigenvalue is taken into account by adjusting the TRACG predicted eigenvalue with the predetermined hot critical eigenvalue for PANACEA. The static effective multiplication factor is the same as the hot critical eigenvalue used in the cycle analyses using PANACEA. This allows the TRACG steady-state solution to converge to the same eigenvalue as PANACEA (Reference 26).

[[

]] The transient response for AOO and ATWS overpressure calculations, however, is a very strong function of the void reactivity feedback.

The TRACG thermal-hydraulic solution for the nodal relative water density solves the bypass, in-channel, and water rod flow and void conditions separately. The flow paths are combined in TRACG to determine the nodal average relative water density based on the flow areas and individual densities. In its review of the LTR NEDC-33173P, where, under some conditions significant bypass voiding may occur, the NRC staff evaluated the impact of the TGBLA06 assumption that the bypass and water rods are purely liquid on the calculation of key parameters such as nodal reactivity and peak pin power. The NRC staff found that the representation, while coarse, does not have a significant impact on the transient analysis given the size of the node relative to an epithermal neutron mean free path and is sufficient to have a negligible impact on the uncertainty analysis associated with the determination of SAFDLs, and is therefore acceptable (Reference 5).

However, the current production method and the extrapolation technique are not able to adequately capture the effect of the plutonium on the void coefficient because the second order fitting inherently assumes that the void coefficient is a linear function of the instantaneous void. For the high void exposure bundles, typical of conditions at EPU or MELLLA+ conditions, the void coefficient behaves non-linearly and the calculation results in a bias. Therefore, when TRACG is used to perform transient analyses there is an exposure-dependent bias in the nodal void feedback. The bias can be quantified and calculated using additional TGBLA06 calculations with higher void depletions. Additionally, TRACG has the functionality of [[

]] in TRACG calculations of the transient LHGR. The NRC staff requested additional information regarding the void coefficient correction model in RAI 7. The NRC staff's review of RAI 7 is included in Appendix A: Staff Evaluation of RAI Responses.

The void reactivity feedback, as calculated, is based on the change in [[

]] This process is performed merely to assess the uncertainty and bias in the void coefficient to be applied to TRACG calculations through the PIRT.

The NRC staff has previously reviewed the impact of the [[ ]] assumption on transient analyses during the review of Reference 31. In that review the NRC staff determined that the transient response predicted by TRACG must include biases and uncertainties that are representative of the lattices in the core design and must be representative of the expected operating strategy. The NRC staff observed biases in TRACG void coefficient for MELLLA+ operation during the review of Reference 31. This is of particular concern for the application of TRACG to EOC isolation ATWS and pressurization AOO analyses where the transient power is a strong function of the void reactivity effect following void collapse. The EOC condition is of particular concern to the NRC staff since the axial power shape is typically top-peaked as is the flux adjoint, thus increasing the reactivity worth of void collapse in that part of the core. The NRC staff determined that explicit TGBLA06AE5 calculations would adequately predict the void coefficient bias at higher void fractions if it were exercised with higher in-channel void depletion histories [[ ]] to account for the influence of plutonium buildup under high void or controlled exposure conditions (i.e., hard spectrum exposure) (Reference 5).

Additionally, the NRC staff is aware of the capability of TRACG to accept void coefficient bias input parameters through the PIRT options in TRACG for uncertainty analyses. Therefore, the NRC staff acceptance of the use of PANACEA generated nuclear data for ATWS calculations in particular will require incorporation of void coefficient biases and uncertainties. The NRC staff requested that the void history bias be quantified and accounted for in RAI 30. The NRC staff review of the response to RAI 30 is documented in Appendix A: Staff Evaluation of RAI Responses. The final disposition of the void history correction is discussed in Section 3.20.2 of this SE.

3.3.4 Related PIRT Parameters

The NRC staff reviewed the PIRT to identify those PIRT parameters affected by the change in the jet pump model. The associated PIRTs are given in Table 3.3.4.1.

Table 3.3.4.1: Kinetics Related PIRT Parameters and Ranking

PIRT		Rank
C1AX	Void Reactivity Coefficient	H
C1BX	Doppler Coefficient	H
C1CX	SCRAM Reactivity	H
C1DX	3D Kinetics	H
C3DX	Prompt Neutron Heating	M

3.3.5 Comparison to the Previously Approved Model

Comparisons of PANAC10 nuclear design methods to PANAC11 were provided to the NRC staff as part of the application for GE improved nuclear design methods (Reference 31). There are a significant number of improvements to the TGBLA06 and PANAC11 models over the TGBLA04 and PANAC10 models. The NRC staff has highlighted some of the improved models below.

Model improvements to the TGBLA code in Version 6 include:

- Inter-resonance self shielding model
- Water rod epithermal slowing down cross-section model
- Non-thermal diffusion coefficient weighting factors
- Thermal diffusion coefficient correction
- Gadolinia rod flux renormalization
- Sub-channel void distribution model

- Low lying inter-resonance self shielding thermal cross-section correction model
- Plutonium fission spectrum adjustment on fast fission cross-sections
- Improved epithermal slowing down near control rod tips
- S and D lattice thermal diffusion coefficient under bladed conditions correction
- Improved convergence technique for fission gas plena above part length rods

Model improvements to the PANACEA code in Version 11 include:
- One-and-half group diffusion theory solution
- Spectral history tracking
- Improved pin power reconstruction
- Improved transient xenon model
- Control blade history reactivity model
- Control blade history rod power and exposure peaking models
- Improved axial meshing
- Improved cold temperature model

The NRC staff audited these specific code changes and these results are documented in References 19, 27, 28, and 29. In its review of the application of PANAC11 for nuclear design analyses for the operating fleet the NRC staff reviewed comparisons of the PANAC10 methodology to the PANAC11 methodology in terms of its efficacy to predict important neutronic parameters. The comparison is based on qualification against a plant tracking database. The original plant tracking database described in Reference 31 is shown in Table 3.3.5.1. The results of both benchmark calculations and comparison to plant tracking results using PANAC10 methods and PANAC11 methods are shown in Table 3.3.5.2. The results indicate a significant improvement in neutronic modeling using the improved TGBLA06/PANAC11 code stream. There is a significant reduction in nodal TIP errors as well as eigenvalue prediction errors. The results confirm that the model updates provide a more robust calculational capability relative to the previously approved PANAC10 methods.

Table 3.3.5.1: Plant Tracking Database for T4/P10 to T6/P11 Migration

Plant	Lattice	Cycle	Fuel Type
[[	D	8	GE8
		9	GE9
		10	GE11
	C	1	GE8
		2	GE8
		3	GE9
		4	GE11
	D	10	GE8
		11	GE10
		12	GE10
	D	11	GE7
		12	GE8
		13	GE11
	S	1	BJ
		2	BJ
	D	8	GE8
		9	GE8
		10	GE10
		11	GE11
	S	5	GE7
		6	GE7
		7	GE10
		8	GE11
	C	1	GE6
		2	GE8
		3	GE7/GE8
		4	GE9
		5	GE11
		6	GE11
]]	D	13	GE9

Table 3.3.5.2: Comparison of T4/P10 to T6/P11 Qualification

	PANAC10	PANAC11
DIF3D Eigenvalue Differences - total standard deviation (Δk)	[[	
DIF3D Nodal Power Differences - total RMS		
DIF3D Peak-to-Peak error		
Plant Tracking EOC hot eigenvalue uncertainty (Δk)		
Plant Tracking EOC-BOC hot eigenvalue discontinuity (Δk)		
Plant Tracking BOC cold eigenvalue uncertainty (Δk)		
Plant Tracking BOCn-1-BOCn cold eigenvalue discontinuity (Δk)		
Plant Tracking hot eigenvalue drift over cycle (Δk)		
Plant Tracking nodal TIP RMS		]]

1

3.3.6 Conclusions

The NRC staff finds that the TGBLA06/PANAC11 methodology provides significant advantages in terms of computational accuracy compared to the TGBLA04/PANAC10 methods. Therefore, the NRC staff agrees that implementing the PANAC11 solver in TRACG04 confers a greater degree of accuracy in the transient modeling compared to the TRACG02 kinetics solver. Therefore, the NRC staff finds that the PANAC11 kinetics solver is acceptable when appropriate measures are taken to address modeling concerns at EPU and MELLLA+ conditions.

The NRC staff has previously approved PANAC11 for nuclear design analyses for the operating fleet (References 5 and 35). However, the NRC staff notes that during its review of the applicability of the TGBLA06/PANAC11 code system for EPU and MELLLA+ plants, the NRC staff identified concerns regarding the efficacy of the code to accurately capture the effects of hard spectrum exposure on nodal nuclear parameters. The NRC staff has previously reviewed the capability of the TGBLA06/PANAC11 codes to accurately predict steady-state nuclear characteristics and found that, in the absence of relevant qualification data, the use of the code for EPU and MELLLA+ conditions required additional conservatism in the safety limit MCPR (SLMCPR) to address adequately predicting the core power distribution.

In the subject review, the NRC staff primarily considered the impact of EPU and MELLLA+ operating conditions on the codes' ability to accurately model the void reactivity feedback. The void reactivity feedback is a key parameter dictating the transient fuel rod power, and hence, a highly important parameter in evaluating the fuel thermal-mechanical (T-M) performance during transients.

The NRC staff requested additional information regarding the use of correction factors to the PANAC11 predicted void reactivity coefficient to improve accuracy in RAI 7. The NRC staff review of the void reactivity coefficient correction model as implemented for TRACG04 is documented in Appendix A: Staff Evaluation of RAI Responses under RAI numbers 7 and 30. The NRC staff separately reviewed the use of the void coefficient correction model for EPU and MELLLA+ conditions in Section 3.20.2 of this SE.

The NRC staff separately reviewed the use of the PANAC11 solver in the TRACG04 code for EPU and MELLLA+ T-M performance analyses in Section 3.20.3 of this SE.

3.4 Decay Heat Model

The American Nuclear Society (ANS) standard decay heat model is implemented in TRACG04 as an optional model in addition to the existing May-Witt model. The five-decay-group May-Witt model is retained as a user option in TRACG04 and the default values are also retained for the group constants. The ANS decay heat model includes both the 1979 and the 1994 standards (References 32 and 33, respectively). The 1994 ANS Standard is slightly more accurate than the 1979 ANS Standard, but is substantially similar.

3.4.1 Description of the Model

The shutdown power following a SCRAM signal during a design-basis LOCA includes many heat sources. These sources include:

- Transient fission power during the signal processing and logic delay
- Transient fission power during hydraulic control unit valve deenergization and stroke
- Transient fission power during control blade insertion
- Power from delayed neutron induced fission
- Decay of radioactive fission products
- Decay of activated fission products
- Decay of actinides in the fuel
- Stored energy in the fuel, cladding, vessel, and vessel internals
- Decay of activated nuclides in the cladding and other structural materials
- Exothermic energy release from water-zirconium reactions

The specific means employed by GE for calculating each of these contributions to the total shutdown power are each described in the following sections.

3.4.1.1 Transient Fission Power

The transient fission power is explicitly calculated by TRACG04 using the PANAC11 3D-kinetics engine. This method was reviewed by the NRC staff and documented in Section 3.3 of this SE. The fission power included in the model includes both the transient power from prompt and delayed neutrons. In periods of reactor SCRAM the transient fission power is determined according to the 3D kinetics equations for residual delayed neutrons captured in the analysis. The weight of the fission power is a normalization factor that forces the total fractional contribution of all power sources to equal one. For power calculations for AOOs, the contribution from the decay heat is conservatively increased, therefore, increasing the integrated thermal energy deposited into the reactor coolant system (RCS) for AOO evaluations terminated by a SCRAM.

The NRC staff has reviewed the kinetics engine and found that the PANAC11 method encoded in TRACG04 is acceptable for performing the transient fission power calculation.

3.4.1.2 Fission Products

The contribution to the shutdown power from fission products can be divided into two subsets. First, there is a heat source from the decay of radioactive fission products. Second, there is a heat source associated with the activation of stable fission products, or fission product daughters, as they are exposed to neutron flux during power operation. The first source can be analytically determined using the 1994 ANS Standard (Reference 33) and predicted core isotopic inventory. The fission products considered in the GE analysis include those from the fission of uranium-235, uranium-238, plutonium-239, and plutonium-241. For each of these parent chains, the decay heat is divided into 23 groups. The summation of the decay heat groups is shown in Equation 3-1.

$$F_i(t,T) = \sum_{j=1}^{23} \frac{\alpha_{ij}}{\lambda_{ij}} e^{-\lambda_{ij} t}\left(1 - e^{-\lambda_{ij} T}\right)$$

Equation 3-1

Where T is the irradiation time
α is the amplitude (specified by the standard)
λ is the decay constant (specified by the standard)
j denotes the decay group
i denotes the parent chain

For the 1994 ANS Standard, only four parent chains are considered. All other fissions are treated as occurring for uranium-235 as it has the highest power fission product chain of the four parents considered.

The second source is based on an adjustment to the first source to account for activation of fission fragments in the fuel. The adjustment is based on the G-factor method to account for neutron capture effects.

The G-factor is a ratio of the fission product decay heat calculated based on an infinite flux exposure to the fission product decay heat calculated based on a zero flux exposure; it does not account for transmutation and neutron capture effects for actinides or structural material activation products.

The G-factor is a function of the fuel and core design, irradiation history, neutron flux magnitude, and spectrum. The G-factors reported in the 1994 ANS Standard are based on cross-section data in the evaluated nuclear data file, ENDF-IV, averaged in a typical light water reactor (LWR) spectrum, operating at a constant power for four effective full power years (EFPYs) with a thermal neutron flux of 1.75 x 10^{14} n/sq-cm/sec. For times less than 10^4 seconds, the G-factor can be expressed as shown in Equation 3-2. For longer times, linear interpolation between tabular values is used.

$$G(t) = 1 + (3.24\times10^{-6} + 5.23\times10^{-10}\, t)T^{0.4}\psi$$

Equation 3-2

Where ψ is the G-factor multiplier (the number of fissions per initial fissile atom)

The G-factor in the 1994 ANS Standard is specifically designed to be a conservative estimate. For typical operating BWRs, the flux levels tend to be an order of magnitude smaller than those used in the standard LWR analysis. Therefore, the user of the standard has the option of employing a customizable G-factor based on core-specific calculations. TRACG04 uses a standard multiplier that is representative of BWR fuels. The multiplier is provided as a function of exposure and energy release per fission in Equation 9.3-25 of Reference 26.

3.4.1.3 Actinide Contribution

The heat source from actinides in the fuel is divided into two subsets. The first subset is considered the major actinides and the second subset is considered miscellaneous actinides. The first set includes the heat source from uranium-239 and neptunium-239.

The second set includes a host of actinides, particularly: curium-242, neptunium-238, uranium-237, plutonium-237 and americium-241.

The actinides are divided into these two groups because the major actinides dominate the decay heat for the early part of the accident. The total integrated power from the minor actinides is only approximately one-tenth of the contribution from the major actinides. After approximately 10^3 seconds, the contributions from the major and miscellaneous actinides are equal. After 10^6 seconds the miscellaneous actinides tend to dominate the decay heat calculation.

The major actinide contribution is calculated according to Equation 3-3.

$$F_{^{239}U}(t,T) = E_{^{239}U} R\left(1 - e^{-\lambda_{^{239}U} T}\right) e^{-\lambda_{^{239}U} t}$$

$$F_{^{239}Np}(t,T) = E_{^{239}Np} R \left[\frac{\lambda_{^{239}U}}{\lambda_{^{239}U} - \lambda_{^{239}Np}} \left(1 - e^{-\lambda_{^{239}Np} T}\right) e^{-\lambda_{^{239}Np} t} - \frac{\lambda_{^{239}Np}}{\lambda_{^{239}U} - \lambda_{^{239}Np}} \left(1 - e^{-\lambda_{^{239}U} T}\right) e^{-\lambda_{^{239}U} t} \right]$$

Equation 3-3

Where R is the ratio of uranium-238 captures to total fission
E is the recoverable energy

ORIGEN2 is used to calculate the heat contribution from the miscellaneous actinides to be included in the shutdown power (Reference 25). The ORIGEN2 results are normalized to TGBLA06 calculated fluxes and stored in tabular form as a function of the irradiation time and the time following the accident. A two parameter linear interpolation technique is used to calculate the miscellaneous actinide contribution based on these parameters.

Several enrichment cases were analyzed, for conservatism; the lowest enrichment case in the study was used to develop the shutdown power table (3.10 percent). By selecting a lower enrichment the contribution from the longer lived actinide sources is artificially increased, thereby increasing the integrated thermal load.

3.4.1.4 Stored Energy

The shutdown power curve that is calculated and input into TRACG for transient analysis does not explicitly include the heat from stored energy in structural materials or the fuel. However, TRACG explicitly accounts for these sources during the transient calculation. TRACG calculates the fuel temperature based on fuel, clad, and gap conductance and heat transfer models. For the vessel and vessel internals, TRACG has a heat slab model which models the heat transfer from the structures to the vessel water inventory. For subcooled or nucleate boiling heat transfer, a Chen Correlation is used to calculate the heat transfer to the water. For single phase convection, a Dittus-Boelter Correlation is used, as described in Reference 25. The TRACG calculated transient heat transfer affects the predicted fuel and cladding temperatures, thereby implicitly accounting for the stored energy being transferred to the water.

3.4.1.5 Structural Activation Product Contribution

The activation of structural materials was calculated using ORIGEN2 and normalized to TGBLA06 calculated fluxes. The process uses ORIGEN2 calculations at various enrichments. In general, lower enrichments lead to a greater degree of activation in the structural materials for a given exposure, since, for these cases, the flux is higher. At a lower enrichment, for the same power level, a higher flux is required, which leads to increased activation. The activation products include those activated nuclei in the cladding, channel box, spacers, as well as the activation of the gadolinia in gadolinia bearing fuel pins. Control materials are not included in the structural activation product contribution and it is neglected in the analysis.

For conservatism, a lower enrichment of 3.10 percent is assumed in the determination of the structural activation product contribution. The normalized ORIGEN2 results are correlated to both the time following the accident as well as the irradiation time. A two parameter interpolation is employed in much the same manner as for the miscellaneous actinides.

3.4.1.6 Chemical Reaction Contribution

The heat produced as a result of water-zirconium reactions in the core during transients is not included in the power calculation. However, this reaction does not become exothermic unless there is significant fuel heat up. As calculations for AOO and ATWS overpressure analyses that demonstrate compliance to SAFDLs do not show any heat up of the reactor fuel during the transient, it is acceptable to neglect this heat source for AOO and ATWS overpressure analyses that demonstrate compliance with SAFDLs.

3.4.1.7 Solution Technique

The total decay power is calculated by summing the normalized contributions for each phenomenon. The summation technique is shown in Equation 3-4.

$$H(t,T) = \frac{G(t)\sum_{i=1}^{4} f_i F_i(t,T) + F_{^{239}U}(t,T) + F_{^{239}Np}(t,T)}{Q} + A(t,T) + AP(t,T) + f_{DN} DN(t)$$

Equation 3-4

Where G is the G-Factor
f is the fission fraction for Parent i
Q is the MeV/Fission calculated by TGBLA[2]
A is the contribution from miscellaneous actinides
AP is the contribution from activation products
f_{DN} is a normalization constant to force H(0,T) to unity
DN is the transient fission power from both prompt and delayed neutrons.

[2] The energy release by fission is calculated by TGBLA when the kinetics solver is activated. TRACG alternatively uses NEDO-23739 values for the energy released per fission when the kinetics solver is disabled (see response to RAI 22).

3.4.2 Qualification of the Model

The method is based on the ANS standards. The method is considered best estimate with an uncertainty based on the uncertainty in the nuclear data used to develop the standard. The uncertainty analysis procedure is documented in Reference 26.

3.4.3 Related PIRT Parameters

The NRC staff reviewed the PIRT to identify those PIRT parameters affected by the change to the ANS Standard decay heat model. The associated PIRTs are given in Table 3.4.3.1.

Table 3.4.3.1: Decay Heat Related PIRT Parameters and Ranking

PIRT		Rank
C25	Decay Heat	H

The decay heat is a highly ranked PIRT for AOOs [[
]]

3.4.4 Comparison to the Previously Approved Model

The TRACG02 method for calculating the contribution of decay heat to the transient power is based on the May-Witt model. The May-Witt model is a five-group decay heat model with fixed decay constants and relative power contributions. The 1994 ANS Standard is considered a best estimate method representative of BWR fuel compositions. Conservatisms are included where appropriate to bound potential uncertainties in applying the model in TRACG04. The 1994 ANS Standard includes specific model improvements relative to the May-Witt model, in particular the capability to account for the differences in decay heat due to fission of isotopes other than uranium-235.

3.4.5 Conclusions

The NRC staff finds that the inclusion of the ANS standard decay heat model improves the calculational accuracy of the TRACG04 code relative to TRACG02. The NRC staff has previously reviewed the use of the ANS Standard decay heat models for BWR LOCA analyses during the review of TRACG04 for ESBWR LOCA (Reference 17). In its review the NRC staff found that the input power curve for ESBWR LOCA applications was acceptable. The decay heat curves were based on a combination of offline calculations using generic fission power curves and the 1994 ANS Standard.

The ESBWR shutdown power, however, included specific factors for the G-factor multiplier. The TRACG04 model is based on a more general representation of the G-factor multiplier.

The TRACG04 model, however, includes relevant conservatisms to compensate for any additional uncertainty potentially afforded by small deviations in the G-factor multiplier for specific lattice designs. Namely, the contribution to the power from the actinides and structural activation products is artificially increased by assuming a very low enrichment relative to modern fuel designs (3.10 percent). Therefore, for performing transient power calculations the TRACG04 will artificially predict higher thermal powers following negative reactivity insertion. Additionally, the contribution of increased thermal power due to the neutron capture effect in fission products is a very small contributor to the total thermal power for AOO and ATWS overpressure analyses. The primary contribution to the uncertainty is the uncertainty in the decay constants, which has been captured in the analysis.

Therefore, the NRC staff finds that the inclusion of the ANS standard decay heat models represents an improvement in calculational accuracy compared to the TRACG02 method. The NRC staff also finds that the approach for determining the ANS standards' uncertainty is sufficient to capture those terms that dominate the total uncertainty, and that conservatisms inherent in the method are acceptable to bound any additional uncertainties introduced by the use of generically evaluated parameters for lattice specific quantities (such as the G-factor multiplier).

The NRC staff notes, however, that acceptance of the ANS standard model for AOO and ATWS overpressure analyses does not constitute approval of the method as implemented in TRACG04 for LOCA analyses. Under LOCA conditions the decay heat represents a much larger fraction of the total thermal load and uncertainties in evaluating the neutron capture effect may not be negligible or adequately conservative. Therefore, the NRC staff notes this conclusion in the limitations and conditions section of this SE.

3.5 Quench Front Model

3.5.1 Description of the Model

As part of TRACG04, GEH enhanced and activated the quench front model within the TRACG04 code. This model is used during the initialization of the reflood phase of a LOCA. The quench front model is based on tracking the velocity of a quenching front resulting either from core reflood from the bottom or from downward flow of a liquid film. The quench front temperature is based on the SAFER model.

The quench front velocity is correlated using the heat transfer coefficient. For quenching from below, the correlation is based on FLECHT reflood data. For a falling liquid film the quench front heat transfer correlation is based on an empirically determined value.

3.5.2 Qualification of the Model

The quench front model is qualified against data collected at the GE core spray heat transfer (CSHT) test facility. The CSHT data was previously used in the steady-state experiments to qualify the radiation heat transfer models in TRACG02 (Reference 36). Transient experimental results were compared to the TRACG04 model in Reference 37. Transient tests were performed where emergency core cooling systems (ECCS) were activated and a transient reflood test was performed for an electrically heated test bundle. The flows, pressures, and cladding temperatures were measured during the test and compared against transient TRACG04 calculations for the cladding temperature of the hot rod during the transient reflood. For the transient spray tests, TRACG04

predicted the cladding temperature with a [[
]]

3.5.3 Conclusions

The qualification against CSHT provides validation of the quench front model for core spray ECCS evaluation. In the test, the rods dryout and activation of the core spray initially reduces the vapor superheat before cooling the vapor sufficiently to reach the cladding surface. The quench front then traverses in a film downward as shown by experimental measurement of rod surface temperature.

For AOO transient licensing calculations, the analyses must demonstrate margin to dryout, and therefore, the quench model is not required for AOO licensing analyses. Similarly for ATWS overpressure calculations, the peak pressure is reached early in the ATWS event and terminated prior to standby liquid control system (SLCS) injection. Therefore, the heat transfer characteristics beyond the point of cladding surface dryout are not required for the subject application. Therefore, the NRC staff did not thoroughly review the improved TRACG04 quench front model and approval of TRACG04 for AOO and ATWS overpressure analysis does not constitute approval of the quench front model.

3.6 Hot Rod Model

3.6.1 Description of the Model

GEH implemented a hot rod model in TRACG04 [[
]]

The hot rod model is used in prediction of cladding temperature for cases where a rod is presumed to be near boiling transition or uncovered (as in the case of reflood during LOCA calculations prior to quenching). The model allows for accurate modeling of the peak cladding temperature (PCT) in conditions where the rod may dryout.

3.6.2 Qualification of the Model

The hot rod model was qualified in Reference 37 by comparison to LOCA test facilities. The CSHT test data indicate close agreement between measured PCT values and hot rod model predicted PCT for a core spray reflood test. The Two-Loop Test Apparatus (TLTA) data was also compared against TRACG04 predictions using the hot rod model with good agreement.

3.6.3 Conclusions

The hot rod model improves the prediction of PCT, and while demonstrating low PCT is required for demonstrating core coolable geometry for LOCA and ATWS calculations, the current application is limited to the use of TRACG04 for AOO and ATWS overpressure calculations. In the case of AOO, the post dryout heat transfer analysis is not required since: (1) critical power is determined according to the GEXL correlation and (2) analyses demonstrate margin to the SLMCPR, therefore the hot rod model is not required to demonstrate acceptable fuel performance during transient calculations.

For ATWS overpressure calculations, the figure of merit is the vessel pressure and the accurate modeling of fuel temperature to demonstrate core coolability is not required to demonstrate compliance with the overpressure protection criterion.

Since the hot rod model is an optional model and does not impact the calculation of those figures of merit relevant to the subject review, the NRC staff did not review the hot rod model. Should GEH seek approval of TRACG04 for ATWS (beyond overpressure) or LOCA, the NRC staff will review the applicability of the hot rod model to determine PCT.

3.7 Minimum Stable Film Boiling Temperature Model

3.7.1 Description of the Model

The boundary between the transition boiling regime and the film boiling regime is defined by the minimum stable film boiling temperature. Transition boiling occurs once the wall temperature has dropped below the minimum stable film boiling temperature if in the film boiling regime. In addition to the Iloeje Correlation and the Homogeneous Nucleation Correlation, GEH implemented an additional option for calculating the minimum stable film boiling temperature in TRACG04, the Shumway Correlation.

3.7.2 Qualification of the Model

The Shumway Correlation is based on a parametric fit of experimental data covering the range of BWR operating conditions in terms of pressure, flow, and quality. A comparison of the Shumway Correlation to data indicates that there is a mean error of – 30K and a standard deviation of 35K.

3.7.3 Related PIRT Parameters

The NRC staff reviewed the PIRT to identify those PIRT parameters affected by the change from the Iloeje Correlation to the Shumway Correlation. The associated PIRTs are given in Table 3.7.3.1.

Table 3.7.3.1: Minimum Stable Film Boiling Temperature Related PIRT Parameters and Ranking

PIRT		Rank
C19X	Minimum Stable Film Boiling Temperature	L

3.7.4 Comparison to the Previously Approved Model

Previous versions of TRACG have used the Iloeje Correlation; however, the Iloeje Correlation was based on a limited data set that did not allow for capturing the pressure and flow dependencies of the minimum stable film boiling temperature. The Iloeje Correlation data restricts application of the correlation to equilibrium qualities between 0.3 and 0.8 and mass fluxes between 54.4 and 135.9 kg/sq-m/s. Extrapolation to different pressures (other than 6.9 MPa) is achieved by using the Berenson pool film boiling temperature difference correlation.

The “TRACG Model Description” LTR (Reference 26) provides a comparison of the Iloeje Correlation to other correlations such as the Cheng and Groeneveld Correlations. The comparisons indicate that the Iloeje Correlation tends to over predict the minimum stable film boiling temperature relative to other correlations. The trend is attributed to scale deposits, wall roughness, and axial conduction.

The Shumway Correlation is based on a much greater dataset that includes variations in both the pressure and flow rate, allowing the correlation to capture the variation in the minimum stable film boiling temperature with these parameters. The form of the Shumway Correlation is provided in Equation 6.6-52 of Reference 26. The Shumway experiment covered pressures ranging from 0.4 MPa to 9.0 MPa and a range of Reynold’s numbers from 0.1×10^5 to 6.7×10^5. This range covers the range of operating BWR flows and pressures.

3.7.5 Conclusions

The NRC staff notes that the minimum stable film boiling temperature is used to predict the boundary between the film boiling and transition boiling flow regimes. For AOO and ATWS overpressure analyses this model is of little importance. AOO transient evaluations are performed using TRACG04 to demonstrate compliance with SAFDLs, including the requirement that fewer than 0.1 percent of the rods will enter transition boiling as a result of AOOs. The boiling transition determination is based on a combination of the SLMCPR and an approved critical power correlation, such as GEXL. Therefore, the boundary between these flow regimes is not necessarily breached for those AOOs showing compliance with SAFDLs.

Since the analyses predicting the onset of transition boiling according to the critical power correlation do not rely on the minimum stable film boiling temperature, and those analyses are intended to demonstrate compliance with SAFDLs, the NRC staff does not find that this model is important in the prediction of AOO transients using TRACG04.

The NRC staff furthermore notes that that the Shumway Correlation provides more realistic results in the prediction of the minimum stable film boiling temperature than the previously adopted Iloeje Correlation. The inclusion of terms capturing the pressure and flow dependencies in the Shumway Correlation relative to the Iloeje Correlation improves the prediction accuracy.

Based on the range of application of TRACG04 for AOOs (to those transient analyses indicating acceptable margin to the boiling transition SAFDL) and the demonstrated performance of the Shumway Correlation against relevant test data, the NRC staff finds that its use for AOO transient evaluations is acceptable.

Certain ATWS scenarios may involve rods entering transition boiling. Particularly ATWS instability events under non-isolation conditions may result in rods entering transition boiling and becoming rewetted during power oscillations. The NRC staff notes that the scope of the subject LTR is to evaluate the overpressure response during ATWS. The NRC staff finds that the prediction of boiling transition during this phase of a pressurization ATWS event does not impact the pressure response calculation, and therefore, the model is acceptable for use when determining the ATWS overpressure response prior to boron injection for limiting pressurization transients.

3.8 Entrainment Model

3.8.1 Description of the Model

GEH modified the entrainment model to better match low pressure data in the migration from TRACG02 to TRACG04. TRACG04 uses an entrainment correlation developed by Mishima and Ishii (see Section 5.1.2 of Reference 26). GEH modified the model for entrainment in the case where only a fraction of the wall surface has gone into film boiling. GEH assumes that the liquid will only flow on the fraction of the wall that has not experienced boiling transition and can be wetted. The TRACG02 model uses a linear model that directly modifies the entrainment fraction in terms of the fraction of rod groups in boiling transition. The model in TRACG04 incorporates the wetted perimeter in the calculation of the hydraulic diameter in the entrainment correlation such that the entrainment fraction has a non-linear relationship with the wetted perimeter.

Both the TRACG02 and TRACG04 models impose the condition that if there are no rod groups in boiling transition, then there is no modification to the entrainment fraction. In TRACG02 the entrainment fraction is unity if all rod groups are in boiling transition. In TRACG04 the entrainment fraction approaches unity based on the hyperbolic tangent formulation of the entrainment fraction as a function of the hydraulic diameter (as all rods enter boiling transition, the wetter perimeter becomes zero, and the hydraulic diameter becomes infinite).

In the TRACG04-specific application, GEH modified the Mishima and Ishii Correlation based on void fraction assessment data. GEH found that the correlation over-predicted the void fraction for large entrainments. The over prediction was due to the rapid increase in the hyperbolic tangent form of the correlation. To better match the data the TRACG04 formulation uses a piece-wise formula with a dimensionless parameter that maintains the same relative dependencies as the Ishii Correlation. The modified TRACG04 model predicts a lower entrainment fraction for low values of the dimensionless parameter η, slightly higher in the intermediate range, and again slightly lower for large values of η. The parameter is a function of the superficial velocity, Reynold's number, and hydraulic diameter.

3.8.2 Qualification of the Model

Figure 5-3 in Reference 26 shows the TRACG04 entrainment correlation compared to data (Cousins, et al., Cousins & Hewitt, Steen & Wallis). The correlation predicts the data well with an average error in the entrainment fraction of +0.0008 and a standard deviation of 0.056. The TRACG04 model uncertainties bound the applicable data set for all ranges of the dimensionless parameter.

The Ishii data, however, is limited to low pressures. The qualification basis of the model is indirect qualification of the entrainment by comparison of the TRACG04 predicted void fraction to measurement void fractions in pipes and rod bundles. The entrainment is particularly relevant to the void fraction modeling in the annular flow regime. Qualification of the TRACG04 interfacial shear model to void fractions at various pressures (encompassing normal BWR operating pressures and flow regimes) and showing small void fraction errors [[ ]] provides the basis for extension of the entrainment model to BWR system pressures.

3.8.3 Related PIRT Parameters

The drift velocity used to calculate interfacial shear in the dispersed annular flow regime is based on the entrainment fraction. Therefore, the entrainment model affects calculation of the void fraction through the interfacial shear model. The related PIRT is shown in Table 3.8.3.1. Since the interfacial shear affects the void fraction, it is a highly ranked PIRT for all AOOs and ATWS overpressure analyses.

Table 3.8.3.1: Entrainment Related PIRT Parameters and Ranking

PIRT		Rank
C2AX	Interfacial Shear	H

3.8.4 Comparison to the Previously Approved Model

The NRC staff reviewed the entrainment model and found that the TRACG04 model is a slight modification to the TRACG02 model, which already includes a correction to the Ishii Correlation to address the rapid rise in predicted entrainment fraction.

The TRACG04 model was slightly modified based on the inclusion of void fraction measurements performed for the Toshiba low pressure tests. The comparison of TRACG04 calculations to the Toshiba tests are documented in Section 3.1.6 of Reference 37. The Toshiba low pressure tests were performed between 0.5 and 1.0 MPa. Three tests were performed for a 4x4 rod bundle. The flow regimes included bubbly, churn, transition, and annular. The NRC staff has previously audited the comparisons between TRACG04 and the low pressure Toshiba test data. The results of the audit are documented in Reference 25.

The low pressure data extend to void fractions of [[ ]] percent. TRACG04 calculations, when compared to the low pressure data indicated a mean bias of 0 percent and a standard deviation of [[ ]] percent (Reference 37).

3.8.5 Conclusions

The piece wise TRACG04 entrainment model formulation is based on tuning the TRACG04 model to void fraction data that encompasses tests performed at pressures ranging from 0.5 MPa to nearly 7.0 MPa. The qualification demonstrates robustness of the model for various pressures when compared against void fraction data (based on a larger dataset relative to TRACG02). Furthermore, the comparison of the modified entrainment model to the original Ishii database indicates that the model predicts the data within the uncertainty range. Therefore, the NRC staff finds that the modified entrainment model is acceptable.

The NRC staff separately reviewed the interfacial shear model for EPU and MELLLA+ applications as documented in Section 3.20.1 of this SE.

3.9 Flow Regime Map

3.9.1 Description of the Model

The constitutive correlations for interfacial shear and heat transfer in TRACG are dependent upon the flow regime in each hydraulic cell. Therefore the flow regime for each cell must be identified before the flow equations are solved for that cell. Transition between annular flow and dispersed droplet flow is given by the onset of entrainment. For low vapor flow, annular flow will exist and, as the vapor flux is increased, more and more entrainment will occur causing a gradual transition to droplet flow.

The models for flow regime transitions in TRACG02 had only been qualified at high pressure. GEH qualified TRACG against low pressure data to extend the applicability of TRACG to LOCA applications. In TRACG04, GEH made changes to the model for transition from churn turbulent to annular flow to better match this data. The criterion for transition to annular flow is when the liquid film can be lifted by the vapor flow relative to the liquid in the churn turbulent regime. This is satisfied at the void fraction where the same vapor velocity is predicted for churn turbulent flow as it is for annular flow. GEH sets the vapor velocity in the churn regime equal to that in the annular regime and solves for the transition void fraction. GEH modified the distribution parameter used to calculate the vapor velocity in the churn turbulent regime.

3.9.2 Qualification of the Model

The flow regime map was compared against Bergles and Suo data (1966) and the Wallis transition criterion with good agreement (Reference 26), However, the NRC staff agrees that flow regime identification based on visual inference is somewhat subjective and furthermore agrees that the model should be indirectly qualified against void fraction predictions using the related interfacial shear model. The NRC staff separately reviewed the interfacial shear model for application to EPU and MELLLA+ conditions and documented the results of that review in Section 3.20.1 of this SE. The NRC staff notes that based on data provided in response to RAI 31 (See Appendix A: Staff Evaluation of RAI Responses), the void fraction predictions are accurate [[ ]] based on a variety of assessment cases.

3.9.3 Related PIRT Parameters

Many PIRTs are related to the accurate prediction of the flow regime. The interfacial characteristics are determined by closure relationships that are specific to the flow regime determined by TRACG04, therefore, changes to the flow regime map have downstream calculational impacts on many PIRTs. The NRC staff selected a sample of highly ranked PIRT parameters to highlight the importance of the flow regime map to AOO and ATWS overpressure calculations, but did not consider all affected PIRTs given the nature of the model change, as described in the following section.

Table 3.9.3.1: Sample of Flow Regime Related PIRT Parameters and Ranking

PIRT		Rank
C2AX	Interfacial Shear	H
C8X	Void Collapse	H
C10	Void Distribution	H
F1	Void Distribution / Two Phase Level	H

3.9.4 Comparison to the Previously Approved Model

The primary difference between the TRACG04 and TRACG02 models is the assumption used to determine the transition void fraction for the churn-turbulent to annular flow regime. The TRACG02 model assumes that the drift velocity is negligible compared to the superficial velocity. In the TRACG04 model, the vapor velocity terms are equated as in TRACG02; however, the dependence of the transition void fraction on the drift velocity in either flow regime is carried through the equality equation to arrive at the TRACG04 transition void fraction shown in Equation 5.1-6 of Reference 26.

For a pressure of 1050 psia, the NRC staff compared Equation 5.1-6 of Reference 26 to the distribution parameter calculated according to Equation 5.1-9 of Reference 38. The dependence of the distribution parameter on the Reynold's number is the same for TRACG02 and TRACG04. The TRACG04 model includes the ratio of the densities, thus making the distribution parameter sensitive to the pressure. The TRACG04 leading term for the churn turbulent infinite distribution parameter would be approximately [[ ]] for a pressure of 1050 psia, which compares well with the [[ ]] value assumed for all pressures in TRACG02.

3.9.5 Conclusions

The NRC staff reviewed the model and found that the TRACG04 model provides a more accurate assessment of the transition void fraction for churn-turbulent to annular flow by accurately carrying through the drift velocity in the transition criterion and is more robust for application to higher or lower system pressures by explicitly applying the density variation in the infinite distribution parameter calculation relative to TRACG02. The NRC staff compared the TRACG04 and TRACG02 models and found they are substantially similar given the relative magnitude of the superficial and drift velocities and the magnitude of the pressure correction term. As described in response to RAI 31 and discussed in Section 3.20.1 of this SE, the NRC staff finds that the update to the flow regime map does not adversely impact TRACG04's ability to predict void fraction and is therefore acceptable for use in AOO and ATWS overpressure transient calculations.

3.10 Fuel Rod Thermal Conductivity

3.10.1 Description of the Model

The TRACG04 improved thermal conductivity model has been updated to be compatible with the formulation in the advanced T-M PRIME03 code. The TRACG04 formulation is somewhat simplified by neglecting the presence of any fuel additives and thereby reducing the conductivity correlation to a function of temperature, density, gadolinia concentration, and exposure.

3.10.2 Qualification of the Model

GEH has submitted the PRIME03 code for review and approval separately. As such, GEH has not provided specific comparison of the PRIME03 fuel thermal conductivity model to data as part of the subject LTR. The NRC staff has requested that GEH provide data to support the improved model in RAI 6.3-54S1 on the ESBWR Docket. The NRC staff, in its review of the subject LTR, however, has based its review of the model on comparisons of the improved model to both the previously approved model and the FRAPCON3 conductivity model. The FRAPCON3 model has been qualified against data collected at the Halden Ultra-High Burnup Experiment and Chalk River National Laboratory (Reference 39).

3.10.3 Related PIRT Parameters

The NRC staff reviewed the PIRT to identify those PIRT parameters affected by the change from the GESTR-M fuel thermal conductivity model to the PRIME03 thermal conductivity model in TRACG04. The associated PIRTs are given in Table 3.10.3.1.

Table 3.10.3.1: Fuel Thermal Conductivity Related PIRT Parameters and Ranking

PIRT		Rank
C3BX	Pellet Heat Transfer Parameters	H

3.10.4 Comparison to the Previously Approved Model

A comparison between the TRACG04 and TRACG02 thermal conductivity models was provided in response to RAI 16. The NRC staff review of the response is documented in Appendix A: Staff Evaluation of RAI Responses.

3.10.5 Conclusions

The NRC staff reviewed the fuel thermal conductivity model in the context of its application for AOO and ATWS overpressure transients. The AOO and ATWS overpressure PIRT lists the pellet heat transfer parameters as a highly ranked PIRT. The fuel pellet thermal resistance is a key parameter in predicting the transient heat flux as a result of changes in the neutron power and affects the transient flow of heat from the pellet to the fluid in the RCS. Therefore, the pellet heat transfer characteristics affect the dynamic interaction between the fluid conditions and the neutron flux.

3.10.5.1 Heat Flux and Neutron Flux Coupling

When performing transient calculations of AOOs, the transient neutron power response will be more conservative if the neutron flux and the fluid conditions are less tightly coupled. The total fuel thermal time constant, which is a measure of the coupling between the fluid response and the fission power, is based on the integral thermal resistance of the cladding, gas gap, and the pellet. The cladding conduction models are unchanged between TRACG02 and TRACG04. The dynamic gas gap conductance inputs for both TRACG02 and TRACG04 are based on upstream GESTR-M calculations. However, TRACG [[ ]] Therefore, the fuel thermal conductivity model will affect the calculation of the thermal resistance of the gas gap.

TRACG (both TRACG02 and TRACG04) calculates the fuel pellet dimensions based on pellet swelling models that consider the fuel pellet cold dimensions and operating history. To a certain extent, the calculation of the gap size compensates for any change in the fuel thermal conductivity. When the predicted fuel conductivity is low, the fuel pellet swells to a greater extent, closing the gas gap; thereby reducing the gap thermal resistance while increasing the pellet thermal resistance. This results in a competing effect in terms of the total thermal resistance. Therefore, the NRC staff expects that the increase in thermal time constant associated with the improved model will be partially, if not largely, offset by the gap reduction due to swelling.

The NRC staff considered the coupling of the neutron flux and fluid conditions for AOO transient evaluations for both a reduced thermal time constant and an increased thermal time constant. When the time constant is over predicted, the fluid response to changing neutron power is lagged. Therefore, a pressurization transient would result in an increase in the reactor power that is not impeded by subsequent rapid void formation due to hold up of the heat flux in the pellet. An over prediction of the time constant will tend to increase the fission power for such a transient. However, the same effect of holding the heat up in the fuel pellet has the dual effect of reducing the cladding heat flux response. Therefore, the ultimate effect on the transient CPR is a combination of the conservative prediction of peak neutron flux with the non-conservative prediction of the transient cladding heat flux. For the case where the time constant is under-predicted the inverse is true. The gross reactor power increase due to pressurization is limited due to more rapid void formation in response to the increasing neutron flux, but this is countered by a prediction of higher cladding surface heat flux relative to the pin power throughout the transient.

Based on competing effects in fuel and gap conductance, the improved thermal conductivity model may increase or decrease the thermal resistance. Similarly, an increase or decrease in the thermal resistance does not have a clear impact on the transient predicted CPR due to competing effects in the cladding heat flux and void reactivity.

For ATWS overpressure transient evaluations the peak pressure will be driven by the integrated power deposited during the pressurization transient. As evidenced by direct comparison of TRACG04 to TRACG02, and the conclusion that TRACG04 generally predicts higher pressures as a result of the eventual conduction and convection of the higher neutron power response to the pressurization, the NRC staff finds that the looser coupling of the fluid response and neutron flux would result in a higher predicted peak neutron flux, neglecting all other feedback mechanisms besides void reactivity. The higher flux as a result of the transient would result in a conservative heat load to the reactor pressure vessel (RPV) and subsequently a conservative estimate of the peak vessel pressure for a fixed safety relief valve (SRV) capacity. As the thermal time constant will be slightly greater using the improved model, the NRC staff finds that its use will lead to slightly more conservative results for ATWS overpressure analyses relative to analyses performed using the GESTR-M based thermal conductivity model, particularly for higher core average bundle exposures if one neglects all other reactivity feedback mechanisms.

The NRC staff notes, as stated in Section 3.3.4 of this SE, that the Doppler reactivity feedback is also a highly ranked PIRT. Therefore, while the NRC staff considered the effects of fluid and neutron coupling, the NRC staff's review has also considered the effects of Doppler reactivity calculations.

3.10.5.2 Doppler Worth and Fuel Temperature

The pellet heat transfer characteristics also affect the Doppler kinetic feedback effect. The dynamic prediction of the fuel temperature is used in the PANAC11 solver to predict the nodal reactivity effect of changing fuel temperature. Therefore, changes to the fuel thermal conductivity similarly have a direct impact on the coupling between the pellet heat generation and the nodal reactivity.

In regards to the Doppler effect, the Doppler coefficient is calculated according to lattice parameters generated by TGBLA06. Fuel temperature branch case analysis is used to develop response surfaces for nodal parameters that are tracked in the PANACEA wrap-up file and passed to the TRACG04 kinetics solver. However, the TRACG04 [[

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[[

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The Doppler coefficient itself decreases in magnitude (becomes less negative) with increasing fuel temperature due to reduced energy self shielding within broadened resonances. Therefore, under-predicting the initial fuel temperature will result in over-predicting the magnitude of the Doppler reactivity coefficient. The Doppler worth is related to the magnitude of the coefficient and the magnitude of the temperature change during a transient condition. Over predicting the temperature change will result in over predicting the total Doppler worth in terms of nodal reactivity.

In TGBLA06, the Doppler effect is inherently captured by inputting the fuel temperatures to determine the change in lattice reactivity and other nodal parameters according to direct transport theory solution. The PANAC11 fuel temperatures are calculated on a nodal level based on the neutron flux, fission power, direct moderator heating fraction, rod diameter, and rod thermal resistance. According to Reference 22, [[ ]] TRACG04 similarly solves detailed thermal heat conduction equations for the transient evaluation using updated fuel thermal conductivity models and explicit dynamic gas gap conductance models imported from upstream GESTR-M calculations for the fuel rods.

The fuel temperature solver in PANAC11, as reported in NEDC-33239P (Reference 22), is unchanged from the approved fuel temperature solver reported in NEDO-20953A (Reference 40). However, the NRC staff notes that recently the fuel thermal conductivity model in GESTR-M was found to under-predict fuel temperatures at high exposure and for gadolinia loaded fuel pins (Reference 41). Therefore, the NRC staff finds that the PANAC11 predicted fuel temperature for high exposure bundles typical of modern fuel duties is under predicted. The improved TRACG04 fuel conductivity model was evaluated by the NRC staff and compared to the FRAPCON3 model as discussed in the NRC staff's evaluation of the response to RAI 16 included in Appendix A: Staff Evaluation of RAI Responses. The NRC staff found that the new fuel conductivity model in TRACG04 predicts lower fuel thermal conductivities with increasing fuel exposure and agrees to a large extent with the FRAPCON3 model in terms of variation with temperature and exposure. However, the NRC staff finds that the improved model appears to misrepresent the impact of gadolinia on fuel thermal conductivity at high exposure.

The NRC staff cannot conclude that the improved thermal conductivity model represents a best estimate of the fuel thermal conductivity over the full range of gadolinia loadings and exposures. However, the NRC staff concludes that the improved model will consistently predict reduced thermal conductivity relative to the previous model based on the GESTR-M code.

Based on its evaluation, the NRC staff has found that: (1) the TGBLA06 calculated Doppler coefficient directly accounts for changes in reactivity as a result of fuel temperature change by explicitly accounting for the resonance broadening in the detailed lattice transport calculations, (2) the PANAC11 fuel temperature model has not been updated to reflect recent findings regarding the efficacy of historical models to capture changes in thermal conductivity for modern fuel designs and high exposure typical of modern fuel loadings and therefore will over predict the fuel thermal conductivity and under predict the fuel temperature, and (3) the TRACG04 thermal conductivity model based on GESTR-M consistently over predicts the fuel thermal conductivity, and while the model based on PRIME03 (improved model) predicts a lower fuel thermal conductivity, there is not sufficient evidence to conclude that this model accurately predicts the fuel thermal conductivity at high exposure for gadolinia loaded fuel pins.

Based on review of the available TRACG04 thermal conductivity models, the NRC staff finds that the fuel thermal conductivity is likely to be over predicted in many cases. Over predicting the fuel thermal conductivity results in a more rapid transfer of heat from the pellet to the fluid during transient evaluations, and therefore, will result in a lower predicted change in fuel temperature during the course of a transient calculation. The NRC staff notes that the improved model temperature change during transient calculations is expected to be more representative of the actual change in fuel temperature. The GESTR-M based model was retained in TRACG04 and is expected to consistently under predict the change in fuel temperature during analyses of AOOs and ATWS overpressure transients.

The NRC staff finds that while the functional Doppler coefficient is accurately predicted by TGBLA06, the PANAC11 kinetics solver will evaluate the magnitude of the coefficient at a temperature that is under predicted, and will, therefore, over predict the Doppler coefficient. The TRACG04 improved thermal conductivity model results in higher predicted changes in fuel temperature during transient calculations, and will therefore, enhance the nodal Doppler feedback (which is non-conservative). The previously approved TRACG02 thermal conductivity model under predicts the change in fuel temperature during the transients, and therefore, when considered with an over predicted Doppler coefficient would result in a cancellation of errors when considering the impact on the nodal reactivity response.

The NRC staff notes that the primary reactivity feedback mechanism driving the transient response for limiting AOO and ATWS overpressure transients is the void reactivity feedback. At normal operating conditions, the nodal reactivity response to void changes will be one to two orders of magnitude greater than the nodal response to fuel temperature change. In its review of the comparison of the TRACG02 and TRACG04 calculations presented in the subject LTR, as documented in Section 3.18 of this SE, the NRC staff found that the fuel thermal conductivity model did not substantially affect the transient calculation of core power, flow, or level. Therefore, the NRC staff concludes that the inclusion of the improved thermal conductivity model is not expected to significantly impact the performance of the TRACG04 code for transient analyses. However, the NRC staff expects that the use of the improved model will non-conservatively predict the Doppler reactivity feedback, which is ranked as a highly important PIRT (Section 3.3.4).

3.10.5.3 Model Applicability

The NRC staff finds that the transient CPR evaluation for AOO analyses will be relatively insensitive to the selected fuel thermal conductivity model. The NRC staff has carefully reviewed the effect of the fuel thermal conductivity on the transient calculation of the heat flux, fuel temperature, and nodal reactivity. In the review, the NRC staff found that several competing effects result in cancellation of errors. This conclusion is further supported by the comparison of the TRACG04 model to the TRACG02 model as discussed in Section 3.18 of this SE.

The NRC staff notes, however, that the GESTR-M based thermal conductivity model will under predict fuel temperatures as the model does not account for the decrease in pellet conductivity with increased gadolinia concentrations or exposure. The NRC staff similarly reviewed the PRIME03-based thermal conductivity model and found that gadolinia bearing pellet temperatures will be under predicted at high exposure. When considered in concert with the PANACEA initialization and fuel temperature accommodation factor, use of a reduced thermal conductivity may result in non-conservative prediction of the Doppler feedback during transient evaluations. The NRC staff, however, notes that the primary feedback mechanism affecting transient BWR analyses is the void reactivity feedback and small errors in the Doppler feedback will have a second order impact on the assessment of margin to SAFDLs based on the relative order of magnitude of the void reactivity coefficient to the Doppler coefficient. This is further evidenced by the TRACG04 void coefficient model described in Reference 1. A sensitivity study deactivating the void coefficient bias correction resulted in a change in the ΔCPR/ICPR of approximately [[ ]] for a relatively large [[ ]] change in the void reactivity coefficient.

Since the GESTR-M fuel thermal conductivity model 10 CFR Part 21 evaluation has been reviewed by the NRC staff (References 41 and 42) and benchmarking activities are on-going, the NRC staff defers conclusions regarding this model to the outcome of its review of these benchmarks. The PRIME03 code review has not been completed by the NRC staff, therefore, the NRC staff defers approval of the improved thermal conductivity model to the PRIME03 review. In the subject review the NRC staff finds that there is sufficient technical basis to determine that the use of either model will not significantly impact the results of transient calculations demonstrating margin to critical power due to competing physical effects. The NRC staff reviewed the use of TRACG04 for evaluating margins to T-M limits in Section 3.20.3 of this SE. Furthermore, while the NRC staff does not find that the improved model accurately captures the impact of gadolinia on fuel thermal conductivity, particularly for high exposure, the NRC staff notes (based on its preliminary review) that the model is expected to more accurately predict the fuel temperature for non-gadolinia bearing fuel rods as it incorporates a correction for fuel exposure.

The NRC staff has also considered the applicability of a gas gap composition predicted by GESTR-M and its compatibility with the PRIME03 model for thermal conductivity. The fission gas release predicted by GESTR-M is a function of the pellet duty during exposure analysis. Therefore, while TRACG internally calculates the gas gap size, the gas gap composition is based on fission gas release predictions evaluated at significantly different temperatures. In general, the gas gap and pellet thermal conductivity are difficult to assess separately based on available data (i.e., pellet centerline temperature). T-M codes are typically tuned to experimental results of measured temperature, and therefore, either model is subject to empirical adjustments and deemed acceptable when considered in concert. The NRC staff has not previously approved a single model in an integral T-M code as the results of the qualification analyses may not be reproducible when different thermal conductivity and gas gap models are exchanged in the code.

The NRC staff considered the impact of the thermal conductivity model on ATWS overpressure analyses and found that integral vessel heat load following a pressurization transient is greatest when the thermal time constant is greater. Therefore, the improved fuel thermal conductivity model is expected to produce slightly conservative estimates of the peak vessel pressure for ATWS overpressure events for a fixed power transient. The TRACG04 model, when compared to TRACG02 using a fixed transient power response in Reference 1, confirms that the use of the improved thermal conductivity model results in slightly higher predicted vessel pressures. These analyses were reviewed by the NRC staff and documented in Section 3.18.8 of this SE. Similarly, the NRC staff considered the impact on the neutronic response in light of a non-conservative prediction of the Doppler reactivity worth. The NRC staff has observed a certain degree of conservatism in adopting the PRIME03 model for a fixed power transient. However, the NRC staff has determined that the Doppler reactivity worth may have a greater impact on the overall conservatism of the analysis. The NRC staff reviewed the results of sensitivity analyses performed using TRACG02 for the medium and high ranked PIRT parameters detailed in Reference 43. Figure 8-10 of Reference 43 provides the uncertainty screening of a main steam isolation valve closure (MSIVC) ATWS overpressure event. The results of the uncertainty analysis indicate that over its range of uncertainty the peak pressure calculated by TRACG02 is much more sensitive to the Doppler coefficient (PIRT C1BX) uncertainty than the fuel heat transfer (PIRT C3BX) uncertainty. The sensitivity analysis confirms the NRC staff's understanding of the driving phenomena: over predicting the Doppler reactivity is non-conservative and under predicting the fuel heat transfer is non-conservative.

The NRC staff has reviewed the relative ranking of these sensitivities and found that while including the PRIME03 thermal conductivity will confer some degree of conservatism due to reduced thermal conductance, the impact on the estimation of the Doppler worth results in an overall overpressure result that is non-conservative relative to the previously approved models (considering how these models are used in the code system).

Based on its review, the NRC staff expects the GESTR-M model to predict slightly more conservative estimates of the peak vessel pressure for ATWS overpressure analyses and expects the GESTR-M model to predict slightly more conservative estimates of the transient CPR for AOO analyses (due to reduced Doppler feedback). Therefore, until the NRC staff completes its review of PRIME03 and review of the GESTR-M 10 CFR Part 21 evaluation (References 41 and 42) and benchmarking, the NRC staff will require ATWS overpressure analyses and AOO analyses be performed using the GESTR-M model.

Furthermore, the NRC staff restricts the use of the PRIME03 model because the pairing of the PRIME03 thermal conductivity with gas gap compositions predicted by GESTR-M may result in uncertainties that yield unintended non-conservatisms in the calculation of transient heat conduction that were not intended and may not be representative of the actual state of the fuel rods.

The NRC staff notes that TRACG04 shares models with upstream GEH analytical codes, for example GESTR-M, PANACEA, and PRIME. The NRC staff requires that the TRACG04 fuel thermal conductivity model used in licensing analysis be consistent with the model in an approved T-M code. Furthermore, as the analyses are suspect when the gas gap conductance is generated using a T-M method with a different thermal conductivity than TRACG04, the NRC staff requires that TRACG04 thermal conductivity be set to be consistent with the gas conductance file provided.

In its approval of T-M codes, the NRC staff notes that certain aspects of the modeling are conservative from the standpoint of establishing a MLHGR limit. As evidenced by the NRC staff review, there are several competing effects in the use of these models in transient analyses. Therefore, the NRC staff requires that use of models in TRACG04

consistent with T-M codes be evaluated to determine the impact on AOO and ATWS overpressure analyses.
Therefore, should the NRC staff subsequently review PRIME03, including the use of particular models in transient calculations, and approve this methodology, the use of the PRIME03 thermal conductivity model will be acceptable when the gas gap conductance files are provided by PRIME03.

However, the NRC staff must note that at cold conditions the primary reactivity feedback mechanism is the fuel Doppler coefficient. Therefore, application of TRACG04 to analyze transients initiated from a cold initial condition will require specific justification. The NRC staff specifically notes that the Doppler feedback is highly important in the analysis of CRDAs. The NRC staff is not reviewing TRACG04 for application to CRDA analysis and therefore defers any conclusions regarding the adequacy of the fuel thermal conductivity model for this purpose. In its review of the thermal conductivity model, the NRC staff reviewed the special power reactor test (SPERT) III E qualification documented in References 36 and 37. The NRC staff found that TRACG04 predicts transient power and integral power that is in much closer agreement with experimental results for the test than the TRACG02 code. However, at this stage, the NRC staff cannot discern to what extent the improvement is driven by an improvement in the kinetics modeling (PANAC11 as opposed to PANAC10 diffusion solvers) as opposed to more accurate modeling of the transient fuel temperature and Doppler worth.

Furthermore, the NRC staff notes that the prediction of the fuel conductivity is an important factor determining the stored energy in the fuel. Therefore, prediction of the fuel conductivity is important in evaluating LOCA response as it is a contributor to the total energy that must be removed by the ECCS. The stored energy does not significantly impact the transient response for AOO and ATWS overpressure analyses, which reach peak conditions of power and pressure very early in the transient. The NRC staff is not reviewing TRACG04 for application to LOCA and therefore defers any conclusions regarding the adequacy of the thermal conductivity model for this purpose.

Lastly, since the fuel thermal conductivity is a factor in the fuel thermal resistance, it will impact the coupling between the neutron flux response and the fluid conditions. Thus the improved model would impact stability analyses. The response to RAI 16 states that the decay ratio is not expected to be impacted by the change in thermal conductivity; however, the NRC staff does not agree with the basis for the determination. The transient heat flux has a direct effect on the transient movement of the boiling boundary and is an important feedback mechanism in the open-loop-transfer-function in stability analysis with an impact on the decay ratio directly observed in the frequency domain. Therefore, the NRC staff cannot conclude that such a physical feedback mechanism would not translate directly to the time-domain analog. The NRC staff, however, is not reviewing TRACG04 for application to stability or ATWS/instability analyses and therefore defers any conclusions regarding the adequacy of the thermal conductivity model for this purpose.

3.11 Rod Internal Pressure, Cladding Yield Stress, and Cladding Rupture Stress Uncertainty Model

3.11.1 Description of the Model

GEH implemented models in TRACG04 to model uncertainties in the rod internal pressure, cladding yield stress, and cladding rupture stress. These models were included in TRACG04 to perform uncertainty analyses for LOCA applications.

3.11.2 Conclusions

The NRC staff did not perform a review of these uncertainty models. The figures of merit for AOO and ATWS overpressure transient calculations are the CPR, LHGR, level, and the peak vessel pressure. These calculated parameters are not affected by the implemented models, nor are their uncertainties assessed based on the subject models. Therefore, the NRC staff finds that the inclusion of these models does not affect the subject LTR review.

3.12 Cladding Oxidation Rate Model

3.12.1 Description of the Model

GEH modified the cladding oxidation model to be consistent with the latest Cathcart and Pawel Correlation. Section 6.6.14 of Reference 37 describes how the Cathcart Correlation for the metal water reaction rate is directly integrated to determine the heat released and hydrogen produced by the zirconium water reaction. The reaction rate is a function of the oxide layer thickness. TRACG04 allows the user to specify the initial oxide thickness, but also has the capability of calculating the oxide layer thickness as described in Section 7.5.8 of Reference 37.

The initial oxide layer thickness and the uncertainty are predicted based on a fit to plant data based on the nodal exposure. For transient applications when the cladding temperature is sufficiently large, the metal water reaction is predicted according to the Cathcart Correlation.

3.12.2 Comparison to the Previously Approved Model

The form of the oxidation rate is unchanged between TRACG02 and TRACG04; however, the constants in the oxidation rate correlation have been updated. The change in the coefficients is relatively small. The TRACG02 model is based on the Cathcart Correlation developed in 1976. The TRACG04 model is based on the revised Cathcart Correlation developed by Cathcart and Pawel in 1977. The TRACG02 and TRACG04 cladding oxidation rate models are repeated here for comparison.

TRACG02: $$\frac{ds}{dt} = \frac{3.217\times10^{-6}}{s}\exp\left(-\frac{2.007\times10^{4}}{T}\right)$$

Equation 3-5

Where s is the oxide layer thickness and
T is the cladding temperature

TRACG04: $$\frac{ds}{dt} = \frac{3.473\times10^{-6}}{s}\exp\left(-\frac{2.010\times10^{4}}{T}\right)$$

Equation 3-6

The correlations for the oxidation rate are similar except the TRACG04 is based on slightly more recent evaluation.

3.12.3 Conclusions

While the oxide layer thickness affects cladding heat transfer characteristics, the NRC staff notes that the initial oxide layer thickness in TRACG04 is either directly input for bounding calculations or is calculated according to the same empirical model based on plant data as in TRACG02. The update to the cladding oxidation rate model is to account for the metal water reaction at high temperatures. Under AOO conditions the analyses demonstrate margin to the SLMCPR, therefore, no appreciable cladding heat up occurs, and the metal water reaction models are not required to predict the total heat generation.

For ATWS overpressure transient evaluations, the transient is terminated after reaching peak pressure prior to initiation of the SLCS. Therefore, for ATWS evaluations the scope of the current application does not require NRC staff review of post peak-pressure ATWS evaluation. However, during ATWS events, appreciable fuel heat up may occur during the initial part of the pressurization transient as some fuel rods enter transition boiling. However, the NRC staff notes that the increase in reactor thermal power will largely dominate the thermal load on the vessel. While the TRACG04 model may in some cases predict exothermic metal water reactions for ATWS events, the contribution to the total thermal power is minimal and the peak pressure response will be negligibly affected by any heat released by the few rods that experience significant heat up over the early part of the transient. Therefore, the NRC staff notes the use of either oxidation rate model will negligibly impact the peak pressure analysis.

Based on its evaluation, the NRC staff finds that the update to the cladding oxidation model does not impact the AOO and ATWS overpressure analyses, and therefore, the NRC staff did not conduct a more thorough review of the cladding oxidation rate models. Since these models affect the prediction of heat released by the metal water reaction and the total hydrogen production, this modification will impact LOCA analyses. Approval of TRACG04 for AOO and ATWS overpressure transient evaluations does not constitute NRC staff approval of TRACG04 for LOCA applications.

The NRC staff notes that the option to predict the initial clad oxide thickness in TRACG02 and TRACG04 remains unchanged.

3.13 Pump Homologous Curves

3.13.1 Description of the Model

TRACG models pumps in a flow path as a momentum source to the fluid. TRACG uses pump homologous curves to describe the pump head and torque response as a function of fluid volumetric flow rate and pump speed. GEH has supplemented default pump homologous curves in TRACG04 with representative curves for large pumps.

3.13.2 Related PIRT Parameters

The pump homologous curves are used to model the recirculation pumps for BWR transient evaluations. The related PIRT parameters and rankings are provided in Table 3.13.2.1.

Table 3.13.2.1: Recirculation Pump Related PIRT Parameters and Ranking

PIRT		Rank
H1	Pump Characteristics / Steady-State	L
H2	Pump Characteristics / Coastdown	H
H3	Pump Two-phase Degradation	N/A

The two-phase degradation PIRT is ranked as N/A because flashing does not occur in the recirculation line for AOOs or ATWS overpressure transients.

3.13.3 Comparison to the Previously Approved Model

In TRACG02, pump homologous curves could be specified in the input or default values could be used. In TRACG04, the TRACG02 default pump curves have been maintained as “set 1.” The set 1 curves are based on the MOD-1 Semiscale system pump tests performed in the early to mid-1970s. A second set, “set 2,” is included as the default pump curves in TRACG04. The second set of curves fully specifies the single-phase head, the fully degraded two-phase head, the head degradation multiplier, the single-phase torque, the fully degraded two-phase torque, and the torque degradation multiplier as functions of dimensionless quantities. The second set of curves is based on the Bingham pump curves and is consistent with the curves used as the default in RETRAN02 and RELAP/5-MOD1.

3.13.4 Conclusions

The TRACG04 default pump homologous curves are based on full scale data measurements, are widely used by the industry for similar applications, and require use of plant-specific input prior to transient evaluation, as specified in response to RAI 28. Therefore, the NRC staff finds that these curves are acceptable for BWR AOO and ATWS overpressure transient analyses. The NRC staff will require that plant-specific pump data be verified and input for transient calculations.

3.14 McAdams Convection Heat Transfer Model

3.14.1 Description of the Model

GEH implemented the McAdams Correlation for free convection heat transfer used in three-dimensional and free surface heat transfer calculations. The Nusselt number is evaluated based on the McAdams Correlation and the Prandtl and Grashof numbers. The form of the correlation is given in Equation 6.5-51 of Reference 26 (also see

Equation 6.6-29). For a flat plate, the heat transfer coefficient characteristic length is given as the average of the length and the width. To account for degradation due to non-condensable gases the Sparrow-Uchida degradation factors are applied consistently with the TRACG02 formulation.

3.14.2 Qualification of the Model

The McAdams heat transfer correlation in TRACG is applied to stratified flows. However, the heat transfer characteristics of such a regime are highly sensitive to the surface conditions – such as rippling of the interface. The TRACG model has been qualified against relevant data for qualification for the ESBWR containment analyses. The application is limited to the heat transfer across a stratified surface and the ESBWR qualification performed at PANTHERS and PANDA indicate that (1) the TRACG model under predicts the free convection heat transfer coefficient, and (2) the pressure and temperature for containment analysis is insensitive to the heat transfer coefficient.

3.14.3 Related PIRT Parameters

The McAdams Correlation is used for stratified interfacial heat transfer calculations. This is most relevant for suppression pool or containment analyses. Therefore, no medium or highly ranked PIRT parameters are related to the use of this model.

3.14.4 Comparison to the Previously Approved Model

The McAdams heat transfer correlation replaces the simplified Holman Correlation. The Holman Correlation is a normalized heat transfer coefficient based on a scaling factor from air at room temperature.

3.14.5 Conclusions

The NRC staff acknowledges that interfacial heat transfer in general is a complex phenomenon and the available physical models are subject to substantial uncertainties. Reference 26 estimates the uncertainty in the degradation factor at 16 percent based on the testing and development of the Kuhn-Schrock-Peterson (KSP) Correlation. The McAdams free convection heat transfer model is widely used and accepted in the scientific and engineering practices. The NRC staff has previously accepted the use of the McAdams free convection heat transfer model in TRACG for modeling the heat transfer across a stratified interface for the ESBWR in Reference 17. Therefore, the NRC staff finds that the TRACG04 use of the McAdams free convection correlation in place of the Holman Correlation is acceptable.

3.15 Condensation Heat Transfer

3.15.1 Description of the Model

For the condensation model, a Nusselt condensation correlation can be used with multiplicative factors for shear enhancement and degradation by noncondensibles. In these equations, the liquid film Reynolds number is calculated based on the condensate flow rate per unit perimeter of surface and the liquid viscosity. However, the recommended (default) TRACG method is the KSP Correlation with the shear enhancement factor set to 1. As a lower bound, when the noncondensible fraction is below about 0.1, the Uchida Correlation is available. For this option, the minimum of the Uchida and KSP Correlations is used.

3.15.2 Qualification of the Model

The PANDA tests were originally used for Simplified Boiling Water Reactor (SBWR) qualification. They were updated for ESBWR qualification. These TRACG04 cases were run using program library Version 40. The NRC staff originally reviewed the PANDA qualification during an audit of TRACG04 for ESBWR LOCA (References 25 and 30). The NRC staff revisited the audit findings to determine applicability of the NRC staff's findings for the ESBWR to the operating fleet.

3.15.2.1 M-Series Tests

The original M-series tests were performed for SBWR, but still have all of the features needed for simulation for ESBWR LOCA, including detailed passive containment core cooling system (PCCS), RPV, dry well (DW), wet well (WW), isolation condenser system (ICS), and gravity driven cooling system (GDCS). Test M3 was a simulation of long-term cooling phase following LOCA caused by guillotine rupture of the main steam line (MSL). Test M10B had all steam directed to DW1 and PCC1 was out of service. Test M10B also examined the influence of asymmetric distributions of the DW steam-air mixture on the startup and long-term performance of the PCCS. M3 and M10B were the two tests compared against TRACG04.

The NRC staff reviewed plots of WW and DW pressure and PCCS mass flow rates for Tests M3 and M10B. The only notable difference between TRACG and the data is that the M3 data shows that the flow in PCC3 decreases and drops to zero at 50000 seconds. TRACG's prediction of the PCC flow is comparable to the other 2 PCCs. The NRC staff believes that something happened in the experiment and the other 2 PCCs are compensating for the one out of service. TRACG comparison to data supports the conclusion that an anomaly occurred during the experiment at that the remaining PCCs are compensating for the decreased flow in PCC3.

3.15.2.2 P-Series Tests

The PANDA P-series tests were run to incorporate changes in the early ESBWR design (GDCS airspace connected to wet well). These tests are not as applicable to the current ESBWR design, for which the M-series tests are more applicable. However, these were the original basis for the TRACG02 ESBWR qualification during the approval of TRACG for ESBWR LOCA (Reference 44) so GEH updated the comparisons with TRACG04.

GEH simulated Tests P4 and P6 with TRACG04. Test P4 addressed long-term cooling performance with the delayed release of non-condensable gas in the DW. Test P6 addressed parallel operation of the ICS and PCCS and the direct bypass of DW steam to the WW gas space. The TRACG input model for the P-series tests differed from that used for the M-series primarily by the inclusion of the RPV and the PCC and IC secondary-side pools in the vessel (VSSL) component along with the DW, WW, and GDCS pool, and it was modified to include the connection of the GDCS gas space to the

WW. Test P4 has delayed injection of DW non-condensable gas. Test P6 had parallel operation of the ICS and PCCS and DW-to-WW steam bypass.

For Test P4, the NRC staff reviewed plots of DW and WW pressure. The TRACG04 predictions are comparable to data. The NRC staff notes that at approximately 8000 seconds, the TRACG prediction and data show a pressure transient associated with the opening of the vacuum breaker (VB). TRACG predicts this happening at a slightly earlier time, roughly 1000 seconds earlier. Mass flow through the PCC is generally comparable. There are some numerical spikes in TRACG that are not seen in the data, even with the data being somewhat noisy. However, the overall trend is the same.

For Test P6, the NRC staff reviewed plots of DW and WW pressures. The TRACG04 predictions were comparable to the data. The pressure comparison is affected by the more rapid purging of the initial DW air inventory in the TRACG calculation. This leads to an earlier VB opening in the calculation and a larger DW-to-WW pressure difference at the time the VB was opened. This resulted in a larger initial leakage flow and an earlier rise in the WW pressure in the calculation. GEH also plotted PCC and IC mass flow rates. The TRACG predictions compared well with data, indicating consistent trends. However the data include significant noise, limiting its use for rigorous qualification.

3.15.3 Comparisons to the Previously Approved Model

The previously approved condensation heat transfer model was the Vierow-Schrock (VS) model. The VS model similarly includes multiplicative factors for shearing and non-condensable gases to adjust the Nusselt number. The TRACG04 KSP model assumes the same form as the VS model; however, it includes two multiplicative terms to account for enhancement. The first term accounts for heat transfer enhancement due to thinning of the film, the second factor ($f1_{other}$) is a correction factor that is based on an approximation of smooth interface laminar film theory and is an adjustment to the shear term to bring better agreement with experimental data.

The VS Correlation predicts very high heat transfer coefficients relative to the KSP model when the Reynold’s number is large. The KSP Correlation is based on a larger data set and therefore, extrapolation beyond the experimentally verified range of Reynold’s numbers does not result in sharp changes in predicted heat transfer coefficients. For pure steam data the KSP Correlation was shown to have a standard deviation of only 7.4 percent.

3.15.4 Related PIRT Parameters

The PIRT related to condensation heat transfer is reported in Table 3.15.4.1. The isolation condenser is not considered an important parameter because of the limited number of plants with isolation condenser systems. The NRC staff finds that this parameter may be important for certain plant-specific applications as documented in Section 3.15.5.3 of this SE.

Table 3.15.4.1: Convection Heat Transfer Related PIRT Parameters and Ranking

PIRT		Rank
Q2	Isolation Condenser Capacity	L

3.15.5 Conclusions

3.15.5.1 General Discussion

The KSP Correlation was developed specifically for PCCS-like conditions based on limited, small scale experiments. As applied in the TRACG methodology, the KSP Correlation was successfully tested against SBWR-specific experiments performed at the PANDA test facility. The comparison with the test data was favorable, at least on a global parameters level. Therefore, the NRC staff finds the heat transfer models to be acceptable for similar design configurations.

3.15.5.2 BWR/3-6 Designs

Wall and tube condensation are low ranked PIRTs for BWR/3-6 designs, and as such, the AOO and ATWS overpressure transient evaluations performed for these plant designs using the modified condensation heat transfer correlation will be minimally impacted. Therefore, the NRC staff finds the KSP Correlation adequate for use in the modeling of these events for BWR/3-6 designs.

3.15.5.3 Oyster Creek and Nine Mile Point Unit 1

The Oyster Creek and Nine Mile Point Unit 1 plants include isolation condensers for maintaining liquid inventory during LOCA and providing core cooling during pressurization events, such as MSIVC. The ICS is a passive high pressure system which consists of two independent natural circulation heat exchangers that are automatically initiated by reactor vessel high pressure or low-low water level. While the ICS is not credited in Appendix K LOCA analyses, it is an important system for mitigating AOOs.

The BWR/2 plant ICS design is substantially different from the ESBWR/SBWR ICS designs. Therefore, the NRC staff cannot determine the acceptability of the KSP Correlation for application to these plants, and the NRC staff will impose a restriction on BWR designs with an ICS. On a plant-specific basis, the licensee referencing TRACG04 for ICS BWR plant transient analyses will submit justification of the applicability of the KSP Correlation to model condensation in the ICS for AOOs. This justification will include, but is not limited to, an appropriate sensitivity analysis to account for known uncertainties in the KSP Correlation when compared to pure steam data.

The sensitivity of the ICS is expected to depend on plant operating conditions, in particular the steam production rate. At EPU or MELLLA conditions the transient response is expected to be more sensitive to the ICS capacity given the relative increase in steam flow rate to reactor core flow rate. The sensitivity is expected to be exacerbated at MELLLA+ conditions where the core flow rate is reduced. Therefore, licensees providing ICS BWR plant-specific justification must provide such justification for each expanded operating domain condition for which analyses are performed.

3.16 6-Cell Jet Pump Model

3.16.1 Description of the Model

The jet pump model in TRACG is based on the TEE component with a momentum source term in the junction. The jet pump component model internally includes loss coefficients for inefficient mixing and pressure losses due to abrupt flow area changes. TRACG02 currently uses a 5-cell jet pump model. TRACG04 has an option to subdivide the straight section between the suction inlet and the diffuser into 2 cells for a 6-cell jet pump model.

3.16.2 Qualification of the Model

The TRACG04 jet pump model was qualified against the 1/6 scale Idaho National Laboratory (INEL) test jet pump test, the full scale Cooper BWR/4 jet pump test, and the full scale LaSalle BWR/5 jet pump test. The basis for the comparison is the calculated and measured relationship between the M-ratio and the N-ratio. The M-ratio is the ratio of the suction to discharge flow, and the N-ratio is the ratio of the pressure difference between the suction and discharge to the pressure difference between the drive flow and the discharge. The jet pump efficiency is the product of the N-ratio and M-ratio.

The INEL 1/6 scale test included both positive and negative driveline flows. The range of scaled M-ratio encompasses all operating BWR/3-6s. The data comparison was provided in Reference 37. The standard deviation in N-ratio between the TRACG prediction and the measurement data for positive flow was [[ ]]. For negative drive flow, the standard deviation was [[ ]], indicating good agreement between the TRACG04 model and test data for both positive and negative drive flow.

The full scale Cooper and LaSalle tests were conducted with positive drive flows only. The standard deviation based on the Cooper test was [[ ]], which compares well with the INEL scaled test results. The LaSalle test was performed at [[ ]] rated drive flow. The standard deviation based on the LaSalle test was [[ ]], which is within the measurement uncertainty of [[ ]] for the test.

The loss coefficient for the nozzle inlet for the 6-cell jet pump was reevaluated in response to RAI 26. The 6-cell jet pump with modified loss coefficients was compared against the INEL 1/6 scale test. The modified inlet loss coefficient indicates a greater degree of agreement between the test data and the TRACG04 model for negative drive flows and large M-ratios.

3.16.3 Related PIRT Parameters

The NRC staff reviewed the PIRT to identify those PIRT parameters affected by the change in the jet pump model. The associated PIRTs are given in Table 3.16.3.1.

Table 3.16.3.1: Jet Pump Related PIRT Parameters and Ranking

PIRT		Rank
G1	Jet Pump Characteristics: Steady-State	H
G2	Jet Pump Characteristics: Coastdown	H
G3	Jet Pump Characteristics: Reverse Flow	H
G7	Jet Pump Pressure Drop	H

3.16.4 Comparison to the Previously Approved Model

The TRACG02 model for typical operating conditions (positive drive flow) indicated uncertainties in the N-ratio on the order of [[ ]]. The TRACG02 model indicated a greater variation for negative drive flow and large M-ratios. The TRACG04 6-cell model with modified loss coefficients, as shown in Figure 26-1 of Reference 7 indicates that the TRACG04 jet pump model confers a greater degree of agreement with the INEL 1/6 test for the most challenging modeling conditions. The NRC staff notes that the modified loss coefficients also result in greater agreement between the TRACG04 model and the INEL test data for positive drive flows and negative M-ratios. The NRC staff finds that the inclusion of the 6-cell model does not adversely impact the jet pump model uncertainties when used with the modified inlet loss coefficients.

The 6-cell jet pump model with modified loss coefficients was compared against the 6-cell jet pump model with historical loss coefficients and full scale data from the Cooper and LaSalle tests. The 6-cell model with modified coefficients indicated greater agreement in N-ratio, particularly for low M-ratios. The improvement in the average N-ratio is on the order of [[ ]] and the improvement in standard deviation on the order of [[ ]].

3.16.5 Conclusions

The qualification of the 6-cell jet pump model with modified loss coefficients provided in response to RAI 26 demonstrates an improvement in the uncertainties associated with the jet pump. The qualification database includes full scale tests as well as a scaled experiment with reverse drive flow. The qualification illustrates an improvement in the prediction of the N-ratio, even under reverse flow conditions. Therefore, the NRC staff finds that the uncertainty analysis is not adversely impacted and the 6-cell jet pump model with modified loss coefficients is acceptable. The NRC staff finds that the sensitivity analysis provided in the response to RAI 26, whereby the loss coefficients were changed using TRACG04 PIRT parameters 70 and 71, provides an adequate technical basis for acceptance of the model.

3.17 Boron Model

3.17.1 Description of the Model

TRACG04 includes a model for the solubility of sodium pentaborate and a model for the boron cross-section. The TRACG04 kinetics solver does not include boron branch cases in the nodal response surface. Therefore, TRACG04 uses an adjustment to the nodal reactivity based on an internal approximation of the boron worth. The NRC staff did not perform a review of this model because it is not applied for AOO analyses, and the subject LTR does not request approval of TRACG04 for ATWS overpressure analysis post boron injection by the SLCS.

3.17.2 Conclusions

The NRC staff finds that the inclusion of the boron solubility models and the boron cross-section model in TRACG04 does not affect the applicability of the methodology to AOO and ATWS overpressure analyses. The NRC staff approval of the subject LTR does not constitute review and approval of the boron models in TRACG04. Should GEH seek approval of TRACG04 for ATWS transients including boron injection, the NRC staff will review the boron models for acceptability.

3.18 Comparison of TRACG02 to TRACG04

The LTR contains comparative analyses performed with TRACG02, TRACG04, and TRACG04 with input options specifying that older TRACG02 models be used in the calculation (TRACG04+). GEH provided qualification of the TRACG02 and TRACG04 against the EOC2 Peach Bottom turbine trip test 1 and test 3. GEH also provided analyses for three pressurization AOOs, one flow increase AOO, one cold water injection AOO, and one ATWS event. The comparison results were evaluated by the NRC staff and discussed separately in the following sections.

The five AOO and one ATWS calculations were performed using a full core model representative of a BWR/4 plant. The core size is 560 bundles and the rated reactor thermal power is 2923 MW_{th}, [[ ]].

3.18.1 Peach Bottom Turbine Trip Tests

TRACG02 and TRACG04 were used to model the first and third EOC2 Peach Bottom turbine trip tests. Both codes predicted the pressure response within the uncertainty in the plant measurements, indicating equally acceptable modeling performance for TRACG04 relative to the previous method. The power responses continue to be over-predicted by the codes; however, this trend is consistent between both codes and is conservative.

3.18.2 Turbine Trip Without Bypass (TTNB)

The TTNB event is characterized by a closure of the turbine stop valve with a concurrent failure of the turbine bypass valves to open. The result is a pressurization of the steam line and consequently the reactor vessel. The increase in pressure results in void collapse and a subsequent increase in neutron power. Pressure increase in response to the increased power is mitigated by SRV actuation and reactor SCRAM. TRACG02 and TRACG04 were used to model a typical TTNB event. The neutron power response predicted by TRACG04 indicates a greater sensitivity to the void collapse and a higher peak power response. TRACG04 predicts a peak power of [[ ]] predicted by TRACG02. This [[ ]] power response is attributable to the improved kinetic solver (PANAC11). The remaining transient response differences between the codes are solely attributed to the [[ ]]

power response predicted by TRACG04. [[ ]]

3.18.3 Feedwater Flow Controller Failure to Maximum Demand (FWCF)

The FWCF is characterized by a failure in the feedwater control system to signal maximum demand. The increased feed flow to the vessel causes the level to rise. When the level reaches the turbine trip level, the turbine trips and the vessel pressurizes similar to the TTNB event. TRACG02 and TRACG04 were used to analyze a typical FWCF event initiated from the same point as the previously described TTNB event. The calculational results indicate similar trends between the codes to the FWCF event. The TRACG04 predicted neutron power response is [[ ]] than the TRACG02 predicted power response. This results in a [[ ]] transient dome pressure and [[ ]] ΔCPR. The differences in the results are primarily driven by the update to the kinetic solver in TRACG04.

3.18.4 MSIVC with Flux SCRAM (MSIVF)

The MSIVF event is characterized by closure of the MSIV with a concurrent failure to SCRAM on MSIV position switch signal. The MSIVC results in pressurization, rapid power increase, and subsequent SCRAM due to high reactor power. TRACG02 and TRACG04 were used to analyze the event. The transient results trend consistently with both the TTNB and FWCF events, indicating similar differences in predicted transient dome pressure attributed to the kinetic solver improvement.

3.18.5 Recirculation Flow Controller Failure (RFCF)

The RFCF event is characterized by a rapid increase in the recirculation pump speed of one recirculation loop. The increase in pump speed results in an increase in reactor flow, and hence reactor power in response to increased moderation. The event is modeled with the APRM high flux trip disabled. The TRACG04 predicted transient peak power reaches [[ ]] percent of rated, while the TRACG02 predicted transient peak reaches [[ ]] percent. The TRACG04 calculated ΔCPR is consequently [[ ]].

3.18.6 Loss of Feedwater Heating (LFWH)

The LFWH event is characterized by a failure in one feedwater heater resulting in an increase in the feedwater temperature. The increased inlet subcooling results in a power increase and downward shift in the axial power. A reactor SCRAM does not terminate this event. TRACG02 and TRACG04 were both used to model a LWFH event for the same reactor conditions. The TRACG04 results indicated a [[ ]]

3.18.7 MSIVC without SCRAM (MSIV/ATWS)

The MSIV/ATWS event is characterized by a closure of the MSIVs without a reactor SCRAM. The rapid pressurization results in a power excursion that is tempered by increased void production in response to the increase in core power. The SRVs relieve reactor dome pressure. Differences in the TRACG02 and TRACG04 responses were observed. The TRACG04 power response to the pressurization is [[

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3.18.8 TRACG04+

TRACG04 was compared to TRACG02 and TRACG04+. TRACG04+ refers to TRACG04 run with several optional models retained from TRACG02 activated in place of new default models. The models deactivated in TRACG04+ are: 6-cell jet pump model, the McAdams convection heat transfer model, the KSP condensation model, and the PRIME03 fuel thermal conductivity model. The models were set to the retained TRACG02 models (5-cell jet pump, Holman convection heat transfer correlation, VS condensation model, and GESTR-M fuel thermal conductivity model). The purpose of the comparison is to demonstrate the impact on transient results of the kinetics solver update relative to the update in the other models considered in the current application.

To determine the impact of all other model changes besides the update to the kinetic solver, the transient power response for a TTNB event analyzed in the comparison of TRACG02 and TRACG04 was used as an input table in the subsequent comparisons. Therefore, the kinetic solver is disabled and all three codes are run with an identical reactor power. Therefore, any changes in the transient response in pressure or flow are attributable to updates in the models. TRACG02 and TRACG04+ trend very closely as the TRACG04+ employs many of the TRACG02 models that have been retained as optional models in TRACG04. The results for TRACG04 predict a [[

]]. The TRACG02 and TRACG04+ results are nearly identical for all transient parameters.

A slight increase in TRACG04 predicted pressure is expected based on the fuel thermal conductivity model, which would result in a slight increase in stored energy for the TRACG04 transient relative to TRACG04+. However, the general agreement between the transient responses provides further evidence that the transient response differences observed for the original TRACG02/TRACG04 comparisons is driven predominantly by the update to the kinetics solver.

3.18.9 Conclusions

The NRC staff concludes that the TRACG04 tends to predict more conservative transient responses based on the update to the kinetics solver to the PANAC11 method. The primary differences in TRACG02 and TRACG04 calculational results are attributable to the updated kinetics solver.

The comparisons highlight that the other model revisions and updates have not had a significant or adverse impact on TRACG modeling capabilities for AOO or ATWS overpressure analyses. Therefore, the NRC staff agrees with GEH's conclusion that most of the TRACG02 component uncertainty parameters are applicable to TRACG04, noting that the void reactivity coefficient uncertainty is revised based on the implementation of the advanced PANAC11 kinetics solver.

The NRC staff has not reviewed the PANAC10 neutronic methods for application to EPU and MELLLA+ transient analysis as part of its review of the subject LTR. The NRC staff notes that initial comparisons between TRACG04 and TRACG02 for a representative EPU core indicate the TRACG02/PANAC10 methods are less conservative. Therefore, the NRC staff approval of TRACG04 for EPU and MELLLA+ licensing analyses does not constitute approval of TRACG02 for this purpose.

3.19 TRACG04 Code Documentation

NUREG-0800, Section 15.0.2, specifies the documents required to describe an analysis methodology. This documentation includes/covers (a) the evaluation model, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the quality assurance program.

3.19.1 Provision of Documents

GEH submitted the evaluation model (Reference 26), accident scenario identification process (Reference 2), code assessment (Reference 37), uncertainty analysis (Reference 2), theory manual (Reference 26), and user manual (Reference 45) as part of the TRACG application.

3.19.2 Quality Assurance

The NRC staff has previously performed audits of the TRACG04 and PANAC11 code documentation under the ESBWR Docket, including the quality assurance program, and documented the results of those audits in internal NRC staff documents (References 25, 27, 29, and 30). The NRC staff has found that the procedures for maintaining a Level 2 engineering computer program (ECP) are acceptable to meet the requirements of 10 CFR Part 50 Appendix B. The NRC staff conclusions relevant to the current application are summarized in this section. The NRC staff included the audit findings regarding code changes in Appendix B: TGBLA06/PANAC11/TRACG04 Code Changes.

The NRC staff has reviewed the changes made to the code and found that they do not have an impact on the methodology (as coded), indicating acceptable quality control through the Level 2 process. The relevant conclusions are documented in the associated audit reports. Furthermore, based on its audit findings, the NRC staff furthermore concludes that those model changes addressed in the subject LTR provide a complete list of the significant code updates between TRACG02 and TRACG04.

To meet the quality assurance criteria of 10 CFR Part 50 Appendix B, GEH must maintain TRACG04 under the Level 2 process or a subsequently NRC-approved process. Therefore, the NRC staff will require that TRACG04 be maintained as a Level 2 ECP under the appropriate procedures or maintained in accordance with any subsequently approved quality assurance processes.

Under the Level 2 process, certain code changes may be made, as evidenced by Appendix B: TGBLA06/PANAC11/TRACG04 Code Changes. Licensees referencing NEDE-32906P Supplement 3 must evaluate all changes to the method in accordance with the criteria of 10 CFR 50.59(c)(2). The NRC staff has considered the potential for future code updates and imposes conditions on these allowable changes consistent with the definition of a methodology change in 10 CFR 50.59(a)(1) and the criteria of 10 CFR 50.59(c)(2)(viii) to ensure that the methodology is not adversely impacted as described in the following sections.

3.19.2.1 Code Changes to Basic Models

Changes to the code models constitute a departure from a method of evaluation used in establishing the design bases or in the safety analysis. Therefore, modifications to the basic models described in Reference 26 may not be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval. (Reference 3)

3.19.2.2 Code Changes for Compatibility with Nuclear Design Codes

Updates to the TRACG nuclear methods to ensure compatibility with the NRC-approved PANACEA family of steady-state nuclear methods (e.g., PANAC11) would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes may be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval as long as the ΔCPR/ICPR, peak vessel pressure, and minimum water level shows less than one standard deviation difference compared to the results presented in NEDE-32906P, Supplement 3. If the nuclear methods are updated, the event scenarios described in Sections 3.18.1 through 3.18.7 of this SE will be compared and the results from the comparison will be transmitted to the NRC staff for information. (Reference 3)

3.19.2.3 Code Changes in Numerical Methods

Changes in the numerical methods to improve code convergence would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes may be used in AOO and ATWS overpressure licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50 Appendix B. (Reference 3)

3.19.2.4 Code Changes for Input/Output

Features that support effective code input/output would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes may be added without NRC staff review and approval. (Reference 3)

3.19.2.5 Updating Uncertainties

New data may become available with which the specific model uncertainties described may be reassessed. If the reassessment results in a need to change specific model uncertainty, the specific model uncertainty may be revised for AOO licensing calculations without NRC staff review and approval as long as the process for determining the uncertainty is unchanged and the change is transmitted to the NRC staff for information. (Reference 3)

The nuclear uncertainties (void coefficient, Doppler coefficient, and SCRAM coefficient) are expected to be revised, as would be the case for the introduction of a new fuel design. These uncertainties may be revised without review and approval as long as the process for determining the uncertainty is unchanged from the method approved in this SE. In all cases, changes made to model uncertainties done without review and approval will be transmitted to the NRC staff for information (Reference 3). This requirement would include those uncertainty changes discussed in Section 3.20.2 of this SE.

3.19.2.6 Statistical Methodology

The statistical methodology is used to determine SAFDLs to account for uncertainties in the analytical transient methodology. As a result, changes to the statistical methodology directly affect the results of safety analyses and constitute a departure from a method of evaluation used in establishing the design bases or in the safety analysis. Therefore, revisions to the TRACG statistical method may not be used for AOO licensing calculations without NRC staff review and approval. (Reference 3)

3.19.2.7 Event Specific Biases and Uncertainties

Event specific ΔCPR/ICPR, peak pressure, and water level biases and uncertainties will be developed for AOO licensing applications based on generic groupings by BWR type and fuel type. These biases and uncertainties do not require NRC staff review and approval. The generic uncertainties will be transmitted to the NRC staff for information. (Reference 3)

3.20 Considerations for EPU and MELLLA+

The NEDC-33173P SE (Reference 5) deferred the review and conclusions of certain topics to the subject TRACG supplemental LTR (Reference 1). Therefore, there are additional margins such as the 10 percent thermal and mechanical overpower margins and the 0.01 operating limit MCPR (OLMCPR) adder for EPU and MELLLA+ applications that have not been applied to the TRACG application. The bases of this approach was to investigate the potential to implement modeling changes in TRACG (e.g., increase in void reactivity biases), which has the capability to simulate three-dimensional reactor core models rather than requiring specific margins to be added to plant-specific applications. In addition, it is appropriate to investigate the adequacy of the supporting data in the review of a specific code for application to EPU and MELLLA+.

The NRC staff has reviewed the information in Reference 1, the supporting LTRs, and RAI responses to determine the applicability of Interim Methods penalties based on the ODYN methodology for the TRACG04 application to EPU and MELLLA+ conditions. These topics include the OLMCPR adder to address concerns regarding potentially increased uncertainty in the application of the Findlay-Dix Correlation for EPU and MELLLA+ transient calculations, considering the more robust interfacial shear model

used in TRACG04 (Section 3.20.1); the void reactivity-void history biases and uncertainties (Section 3.20.2); the thermal and mechanical overpower margin enhancement (Section 3.20.3); the transient varying axial power and control rod pattern input (Section 3.20.4); and the application to mixed core EPU and MELLLA+ licensing evaluations (Section 3.20.5).
The NRC staff will impose all limitations specific to analyses documented in its SE for the review of NEDC-33006P (Reference 46) for the application of the TRACG04 method to MELLLA+ conditions.

3.20.1 Void-Quality Correlation and TRACG04 Interfacial Shear Model

The void-quality correlation implemented in PANAC11 is the Findlay-Dix void quality correlation. In its review of NEDC-33173P LTR (References 5 and 31), the NRC staff found that the correlation basis is not sufficient to categorically extend the application of the correlation to pure steam conditions. To address concerns regarding the void fraction calculations, the NRC staff imposed an interim margin enhancement via an OLMCPR adder of 0.01. Documented specifically in the NRC staff's SE (Reference 5) as follows:

> **Void-Quality Correlation Limitation 1**
> For applications involving PANCEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady-state, transient, and accident conditions.
>
> **Void-Quality Correlation Limitation 2**
> The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," dated May 2006 (Reference [1]). The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to LTR NEDC-32906P (Reference [1]) will be applicable as approved.

The adder is intended to add margin to address uncertainties in the predicted transient behavior for pressurization events (which tend to be limiting from a boiling transition perspective). Under normal steady-state conditions at EPU and MELLLA+ conditions the core average, outlet, and hot channel void fractions are expected to increase. This is a result of core loading patterns that include a larger number of high powered bundles with a flattened radial power profile to achieve the higher core power with the same dome pressure, and bundles with higher bundle powers than were previously loaded. In many cases the hot bundle in-channel outlet void fractions approach 90 percent or potentially higher. These void fractions are close to void fractions predicted for critical power tests indicating that the margin to boiling transitions may be degraded for the hot bundles and that a larger number of bundles in EPU cores may be subject to boiling transition considering the effects of AOOs.

In its review, the NRC staff found that the Findlay-Dix void quality correlation is not well qualified for high void fractions or for modern fuel bundle designs. The correlation directly relates the void quality to the void fraction, and therefore may be sensitive to particular features of the bundle geometry such as part length rods or fuel spacer arrangement. While explicit modeling of the transition to film boiling will require detailed modeling of the flow behavior near fuel spacers (since the limiting point in the bundle from a critical heat flux perspective is directly beneath a spacer where the liquid film thickness is thinnest), the NRC staff notes that the thermal margin is determined according to a critical quality correlation developed based on fuel geometry specific full scale test data for the GEH code system.

The NRC staff notes that the ODYN code is currently used to perform transient AOO and ATWS overpressure calculations for EPU and MELLLA+ licensing calculations currently. The ODYN void-quality correlation is the same Findlay-Dix correlation used in PANAC11. In its review of the ODYN method for EPU and MELLLA+ the NRC staff determined that the potential consequences of pressurization AOOs were increased given the higher bundle powers, higher initial void fractions (hence enhanced void reactivity feedback), and a greater number of high powered bundles at these conditions. To address potentially increased errors for current fuel designs and high void fractions beyond the scope of the Findlay-Dix qualification database, the NRC staff imposed a 0.01 OLMCPR adder. The penalty was imposed to conservatively bound any uncertainty in the transient response to a pressurization transient and ensure that adequate margin exists to boiling transition. The NRC staff determined that 0.01 margin was adequate noting particular features of the void-quality correlation; notably that the correlation is well behaved for annular flow void fraction predictions and that the variation in void fraction for high qualities is relatively insensitive.

The TRACG04 void fraction calculations are based on a more robust interfacial shear model. In response to RAIs 24 and 31, GEH provided additional details of the void fraction qualification. The NRC staff review of these RAIs is provided in Appendix A: Staff Evaluation of RAI Responses.

The TRACG04 interfacial shear model is described in Section 6.1 of Reference 26. The model uses separate correlations for the interfacial shear based on the flow regime. Separate correlations are developed for bubbly/churn flow, annular flow, droplet flow, and annular/droplet flow. The NRC staff reviewed the flow regime map as documented in Section 3.9 of this SE and the entrainment model in Section 3.8 of this SE. The modified flow map and entrainment models dictate the specific correlations used in the interfacial shear model to determine the void fraction. The separate correlations are required as the nature of the interface depends on the flow conditions. For example, the surface area of the interface is different for a liquid film in the annular flow regime as opposed to the surface for bubbly or dispersed droplet flow.

In the Findlay-Dix Correlation, the void fraction and the quality are directly correlated. In the TRACG04 model a more mechanistic approach is developed whereby the two fluid model explicitly determines the phase slip according to the two momentum equations. The phase slip is based on the interfacial shear term in the momentum equations and is determined according to a correlated interfacial friction factor and the relative velocities in the TRACG model.

The NRC staff notes that interfacial phenomena have not been extensively studied and previous experimental data has been aimed at assessing the prediction of gross parameters, such as void fraction and pressure. The experimental data has historically

been developed for correlation development or assessment for previous system codes that have not explicitly tracked interfacial phenomena. Therefore, the interfacial shear model is based on drift flux mechanisms inferred from void fraction data by Ishii. For adiabatic steady-state conditions, the interfacial shear model will collapse to the drift flux model proposed by Ishii.
The interfacial shear model has been qualified according to available data from void fraction and pressure drop measurements. The NRC staff reviewed the database used in the assessment of the void fraction model in its review of the response to RAIs 24 and 31 as documented in Appendix A: Staff Evaluation of RAI Responses. The NRC staff found that the interfacial shear model illustrates robustness in that the errors in the prediction of the void fraction are not sensitive to the pressure, flow regime, or geometry. The errors in the void fraction are less than [[ ]] percent and the interfacial shear model (based on available data) does not exhibit an appreciable bias in the void fraction prediction [[ ]].

To address the applicability of the interfacial shear model to modern fuel designs, as void fraction data is unavailable, GEH used data collected during critical power testing of 10x10 fuel. During these tests GEH collects pressure drop data. For very low flows the dominant pressure drop term is the buoyancy term and the exit void fraction is high. For these data GEH performed an uncertainty analysis by comparing the predicted and measured pressure drop and assigning all uncertainty to an equivalent uncertainty in the nodal void fraction. The results indicate that the conservatively estimated void fraction error is consistent with the error based on direct measurement. The 10x10 GE14 test indicates a [[ ]] mean error and a [[ ]] standard deviation. These results are very similar to the FRIGG OF64 6.8 MPa test qualification results of a [[ ]] mean error and a [[ ]] standard deviation. This indicates stability in the model in its application to modern fuel bundle designs. For AOO and ATWS overpressure transient calculations the modeling of post critical heat flux heat transfer or flow is not important, therefore, critical heat flux tests provide adequate demonstration of the modeling capabilities for the range of application considered in the subject LTR.

The TRACG04 analysis initialization, however, is based on steady-state power distribution calculations performed using PANAC11. The NRC staff described the process for TRACG04 initialization in Section 3.3.3 of this SE. Therefore, the transient calculations still require use of the Findlay-Dix void quality correlation for the prediction of the initial power distribution.

In terms of predicting the transient thermal margin during AOOs, the code will first initialize the TRACG thermal-hydraulic solution to the PANAC11 power distribution. The initial fluid condition in TRACG prior to the AOO is therefore based on the TRACG thermal-hydraulic model, while the initial power distribution is based on the PANAC11 model. Accommodation is performed on a nodal basis to ensure that the thermal-hydraulic solution is stable. [[

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In its review of the thermal conductivity model (Section 3.10 of this SE), the NRC staff described this aspect of the code in regards to the Doppler coefficient. The same aspect holds true for the nodal void reactivity. The TRACG04 void reactivity feedback is slightly different in its application in that it includes a correction model to incorporate known biases. The NRC staff reviewed the void coefficient correction model as discussed in Section 3.20.2 of this SE.

The NRC staff has previously reviewed the application of TGBLA06/PANAC11 to calculate the steady-state conditions. Uncertainties in regards to these methods are addressed in the NRC staff's SE regarding their application as documented in Reference 5. The NRC staff's review in this area is related to the downstream impact in TRACG04 of calculating the nodal reactivity void feedback as modeled in the PANAC11 response surfaces for AOO and ATWS overpressure calculations at EPU and MELLLA+ conditions.

For these conditions, the bundle powers are higher, the flow rates are lower, and the void fraction is increased relative to pre-EPU conditions. The NRC staff found that the Findlay-Dix void quality correlation was not adequately qualified to reasonably assure the NRC staff of its accuracy for high void fraction steady-state calculations. Therefore, the NRC staff finds that the use of this model in the PANAC11 code may result in errors in the PANAC11 predicted nodal conditions at the initiation of the transient. However, the NRC staff notes that the TRACG04 interfacial shear model will accurately calculate the thermal-hydraulic initial condition during the initialization process of the calculation and will model the void collapse during pressurization.

The transient power response will be driven by the TRACG04 calculated thermal-hydraulic conditions as they are translated to the PANAC11 engine through accommodation factors for water density and fuel temperature. The PANAC11 response surface for the nodal reactivity will be based on instantaneous void conditions predicted by PANAC11. For high void fractions, the reactivity void coefficient generally increases. However, the sensitivity of the nodal reactivity void response is damped by the presence of the bypass and water rods. During transient pressurization events (which tend to be the limiting AOO events) the bypass and water rods provide a fixed slowing down source within the node. At increasingly high void fractions, a greater percentage of the slowing down power is provided by the bypass and water rods and the nodal response to transient increasing in-channel void conditions is effectively damped.

The NRC staff evaluated the order of magnitude of the sensitivity of the nodal void reactivity coefficient sensitivity to an error in the prediction of the nodal void fraction in Appendix C: Sample Calculation of Void Reactivity Sensitivity. The PANAC11 nodal response surface is based on its predicted nodal void fraction, which will differ from the TRACG04 calculated in-channel void distribution. In the initialization process in TRACG04, the effect of using the Findlay-Dix void quality correlation is to bias the nodal void reactivity coefficient. [[

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The NRC staff notes that its assessment of the sensitivity of the void reactivity includes a large degree of conservatism. First, the sensitivity is based on the linear fit of the nodal eigenvalue, at high void fractions the spectral shift with changing void fraction is damped by the bypass slowing down source. Second, the percentage change is based on a typical core-wide void reactivity coefficient at cold conditions. At higher void conditions the magnitude of the void reactivity coefficient will increase. Third, the NRC staff considered a bounding bias in the Findlay-Dix void quality correlation. Fourth, the high void nodes comprise only a fraction of the entire core. Considering these conservatisms, the NRC staff's sensitivity analysis when considered with GEH's sensitivity analysis [[ ]] indicates that the residual nodal void reactivity bias in the PANAC11 solver will have an impact on calculation of ΔCPR that is smaller than the threshold of significance (0.005).

In response to RAI 32, GEH provided a detailed sensitivity analysis to address the transient effect of a void fraction mismatch between PANACEA and TRACG. The NRC staff reviewed the results of this analysis as described in Appendix A: Staff Evaluation of RAI Responses. The NRC staff found that the relative moderator density mismatch for a large BWR/4 at EPU conditions was calculated to be on the order of [[ ]] percent, indicating good agreement between the PANACEA and TRACG thermal-hydraulic solutions. The response also includes an analysis performed using a modified version of TRACG04 that allows for convergence of the steady-state solution using the PANAC11 nuclear method with the TRACG interfacial shear model as opposed to direct initialization to the PANACEA solution. The results of analyses performed using the original and modified TRACG04 versions indicate a sensitivity in the limiting channel ΔCPR/ICPR that is well below the threshold of significance [[ ]]. Therefore, the NRC staff is reasonably assured that for EPU analyses the interface between the two codes during the initialization will not introduce significant errors in the predicted transient response.

The NRC staff has also reviewed any conservatism in the application of interfacial shear model to transient applications, noting that the use of the interfacial shear model is expected to yield greater accuracy up to void fractions of [[ ]] or higher for transient evaluations. The NRC staff reviewed the transient responses predicted by TRACG02 and TRACG04 for several transients in Section 3.18 of this SE. Generally the NRC staff found that the PANAC11 neutronic model in TRACG04 predicted a greater flux response to void collapse, indicating that the PANAC11 predicted void reactivity feedback is greater for TRACG04 than TRACG02. The NRC staff finds that predicting a stronger coupling will produce more limiting results for pressurization transients, and is conservative relative to the previously approved method.

In response to RAI 7 and RAI 30, GEH provided details of the uncertainty and biases calculated for the void reactivity coefficient. The biases are captured in the TRACG04 void coefficient correction model, which the NRC staff reviewed in Appendix A: Staff Evaluation of RAI Responses and describes in greater detail in Section 3.20.2 of this SE. The NRC staff found that the void reactivity coefficient uncertainties were conservatively determined by assessing the error using MCNP comparisons based on only uncontrolled lattices. The controlled lattice void reactivity coefficient is less sensitive to the geometric modeling, and including controlled lattices in the assessment would reduce the calculated uncertainty in void reactivity coefficient.

Therefore, the NRC staff finds that the use of the TRACG04/PANAC11 code stream will allow more accurate and reliable modeling of void collapse from EPU and MELLLA+ initial conditions in the determination of transient CPR for limiting pressurization AOO events. The qualification of the interfacial shear model and the sensitivity analyses performed by GEH and the NRC staff indicate a potential bias that is below the threshold of significance for the OLMCPR. Therefore, the NRC staff finds that transient calculations for EPU plants using TRACG04 do not require the 0.01 OLMCPR thermal margin enhancement.

The NRC staff reviewed information provide by GEH in the response to RAI 32 regarding the sensitivity of the transient analyses to the void fraction uncertainties in the Findlay-Dix Correlation (see Appendix A: Staff Evaluation of RAI Responses). While the NRC staff finds that void fraction uncertainty under certain conditions (such as the transition corner of the MELLLA+ operating domain) may have an impact on the calculated transient CPR in excess of the threshold of significance, the NRC staff finds that a thermal margin enhancement is not necessary to address reload licensing applications. The response adequately demonstrates that for the magnitude of the void fraction mismatch that the limiting transient responses are negligibly affected.

The NRC staff's conclusions here are predicated on pressurization transients being the limiting transients in reload licensing analyses. This is true for the operating fleet of BWR/2-6 reactors. Therefore, the NRC staff's findings in this matter may not be applicable to other BWR designs where pressurization transients are not the limiting transients.

The NRC staff based its review findings on the demonstrated applicability of the interfacial shear model to modern bundle designs. Specifically, the NRC staff's review referenced indirect qualification of the interfacial shear model to pressure drop data collected for GE14 fuel during critical power testing. In Reference 47, GEH committed to provide qualification of the Findlay-Dix void quality correlation against similar pressure drop data. The method for using the pressure drop data to qualify the void fraction modeling was exercised in a prototypical manner for the interfacial shear model in response to RAI 31. The method is based on low flow measurements that yield the greatest sensitivity to void fraction because the pressure drop is driven primarily by buoyancy.

GEH has committed to provide the details of this method and data for comparison to the Findlay-Dix Correlation. The NRC staff will review the methodology as a supplement to LTR NEDC-33173P. Should the NRC staff find this methodology acceptable, a parallel method for assessing the interfacial shear model will likewise be acceptable. The NRC staff will require that any EPU or MELLLA+ plant licensing analyses referencing TRACG04 methods for future GNF fuel products shall verify the applicability of the interfacial shear model using void fraction data, or the aforementioned interim approach (if accepted by the NRC staff).

3.20.2 Void History Void Reactivity Coefficient Biases and Uncertainties

GEH provided descriptive details of the void reactivity coefficient correction model in response to RAI 7. The NRC staff has reviewed the response and documented this review in Appendix A: Staff Evaluation of RAI Responses. The NRC staff found that the harder spectrum conditions present in EPU and MELLLA+ cores call into question the validity of the constant void exposure assumption inherent in the void reactivity coefficient correction model. In response to RAI 30, GEH has revised the void reactivity coefficient correction model to explicitly account for the historical void conditions under which a node is exposed. Accounting for the void history allows for accurate characterization of the bias for hard spectrum exposure conditions. The NRC staff reviewed the revised model as documented under RAI 30 in Appendix A: Staff Evaluation of RAI Responses.

The NRC staff has found that the TGBLA06 to MCNP comparisons were adequate to determine the [[

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In its review of the void reactivity coefficient correction model, the NRC staff notes that the acceptance of TRACG04 for AOO and ATWS overpressure transient analysis at EPU or MELLLA+ conditions requires that this correction model be activated.

Furthermore, the NRC staff notes that the void coefficient correction model is based on specific lattice calculations performed using TGBLA06 and MCNP. Lattice designs vary with fuel bundle design, and therefore, a set of lattices may not be representative of all future fuel designs. The current lattice set is based on representative modern fuel designs (10x10 rod arrays). The NRC staff will require that licensees referencing NEDC-32906P, Supplement 3 either: (1) confirm that the void coefficient correction model includes lattice information that is representative of the licensee's fuel, or (2) update the void reactivity coefficient correction model lattice database for consistency and evaluate the uncertainties and biases.

3.20.3 Thermal and Mechanical Overpower Margin

GDC-10 requires that SAFDLs are not exceeded during any condition of normal operation. To demonstrate compliance with GDC-10, fuel rod T-M design limits are established to ensure fuel rod integrity in its core lifetime along the licensed power/flow domain, during normal steady-state operation and in the event of an AOO. The T-M acceptance criteria for new fuel product lines are specified in Amendment 22 to the NRC-approved GENE licensing methodology GESTAR II. The LHGR limit is an exposure-dependent limit placed on the rod peak pin nodal power that ensures the integrity of the fuel cladding during normal steady-state operation and limits the initial heat generation rate during transient thermal and mechanical overpower conditions. The internal rod pressures during steady-state, the maximum fuel temperature, and the cladding strain during transients (AOOs) all affect the fuel integrity. As stated in Section 3.2.6 of Reference 5, the fuel T-M design criteria require, in part, that:

1. Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading.

> The fuel rod internal pressure is limited so that the cladding creepout rate due to internal gas pressure during normal operation will not exceed the instantaneous

fuel pellet irradiation swelling rate. In establishing the LHGR limit, at each point of the exposure dependent envelope, the fuel rod internal pressure required to cause the cladding to creep outward at rate equal to the pellet irradiation swelling is determined. The calculated internal rod pressures along the LHGR envelope are statistical treated so that there is assurance with 95 percent confidence that the fuel rod cladding creep rate will not exceed the pellet irradiation swelling rate.

2. Loss of fuel rod mechanical integrity will not occur due to fuel melting.

The fuel rod is evaluated to ensure that fuel melting will not occur during normal operation and core-wide AOOs. [[

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3. Loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction.

The fuel rod is evaluated to ensure that the calculated cladding circumferential plastic strain due to pellet–cladding mechanical interaction does not exceed 1 percent during normal operation and AOOs. [[

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Therefore, the fuel rods loaded in the core are monitored to ensure that the exposure-dependent LHGR envelope for each product line is met. The LHGR limit is specified in the TS and/or the core operating limit report. The ratio of the steady-state operating peak nodal LHGR (MLHGR) over the steady-state LHGR limit is referred to as maximum fraction of limiting power density (MFLPD). Fuel parameters that affect the local pin powers such as pin power peaking, void reactivity, and bundle powers all factor into the development of the LHGR limits. Therefore, increases in the power distribution uncertainties affect the prediction and monitoring of the operating LHGR during steady-state operation and transient conditions. Operating experience data show that fuel rods can operate at or near the LHGR limit at some point in the operating cycle; therefore, the accuracy of the prediction of maximum operating LHGR (MLHGR) becomes important. Operation at EPU and the proposed MELLLA+ domain will result in a more limiting transient response since the steam flow increases but the pressure relief capacity remains fixed. In addition, the number of fuel bundles operating at the peak LHGR envelopes is expected to be higher for plants operating with 24-month cycles at EPU and MELLLA+ conditions. Therefore, the thermal and mechanical overpower response during limiting AOO events are expected to be higher for operation at EPU and MELLLA+.

Therefore, the NRC staff imposes a restriction for AOO analyses that reflects the same NRC staff position regarding the licensing process for EPU and MELLLA+ plants referencing the ODYN transient methodology for AOO and ATWS overpressure analyses (Reference 5):

> **Transient LHGR Limitation 1**
> Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the T-M acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel melting and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for the both the UO_2 and the limiting GdO_2 rods.

> **Transient LHGR Limitation 2**
> Each EPU and MELLLA+ fuel reload will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the SRLR [supplemental reload licensing report] or COLR [core operating limit report], or it will be reported directly to the NRC [staff] as an attachment to the SRLR or COLR.

In its review of the ODYN transient analysis code, the NRC staff imposed a restriction for AOO analyses related to demonstrating compliance with TOP and MOP criteria as documented in the NRC staff's SE for NEDC-33173P (Reference 5):

> **Transient LHGR Limitation 3**
> To account for the impact of the void history bias, plant-specific EPU and MELLLA+ applications using either TRACG or ODYN will demonstrate an equivalent to 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain acceptance criteria due to pellet-cladding mechanical interaction for all of limiting AOO transient events, including equipment out-of-service. Limiting transients in this case, refers to transients where the void reactivity coefficient plays a significant role (such as pressurization events). If the void history bias is incorporated into the transient model within the code, then the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain is no longer required.

The NRC staff reviewed the proposed method for incorporating the void history bias in the TRACG04 3D kinetic solver methodology in Section 3.20.2 of this SE. Based on its review of the updated methodology, the NRC staff finds that TRACG04 sufficiently accounts for void coefficient biases for hard spectrum exposure and that the 10 percent margin to the applicable T-M criteria of Transient LHGR Limitation 3 does not apply to EPU or MELLLA+ licensing calculations when TRACG04 methods are referenced.

However, the NRC staff notes that TRACG04 includes an updated thermal conductivity model that is not consistent with the GESTR-M methodology used for calculating the LHGR limits. The TRACG04 thermal conductivity model is based on PRIME03, which includes models to account for thermal conductivity degradation as a result of exposure and gadolinia content. The NRC staff review of this model is documented in Section 3.10 of this SE. In its review the NRC staff found that the updated thermal conductivity model is used for transient evaluations only and has not been used to [[ ]] The NRC staff found that the primary impacts of the improved thermal conductivity model are on the transient cladding heat flux and the Doppler worth. Of these, only the Doppler worth will affect the overpower LHGR. The NRC staff concluded (as documented in Section 3.10.5.2 of this SE) that the GESTR-M model conservatively predicts a smaller negative Doppler reactivity worth and, therefore, when TRACG04 is used to determine the limiting LHGR for transients, the GESTR-M thermal conductivity model must be used unless the NRC staff subsequently approves the PRIME03 thermal conductivity and dynamic gap conductance models in a separate review.

Therefore, the conditions specified in the NRC staff's SE for NEDC-33173P (Reference 5) regarding the adequacy of the GESTR-M methodology similarly apply for the use of TRACG04 to perform AOO analyses for EPU and MELLLA+. Namely, the NRC staff specifically imposes the same conditions for the TRACG04 methodology as follows:

Application of 10 w/o Gadolinia Limitation
Before applying 10 weight percent Gd to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for TOP and MOP conditions that bounds the response of plants operating at EPU and expanded operating domains at the most limiting statepoints, considering the operating flexibilities (e.g., equipment out-of-service). Before the use of 10 weight percent Gd for modern fuel designs, NRC [staff] must review and approve TGBLA06 qualification submittal. Where a fuel design refers to a design with Gd-bearing rods adjacent to vanished or water rods, the submittal should include specific information regarding acceptance criteria for the qualification and address any downstream impacts in terms of the safety analysis. The 10 weight percent Gd qualifications submittal can supplement this report.

Part 21 Evaluation of GESTR-M Fuel Temperature Calculation Limitation
Any conclusions drawn from the NRC staff evaluation of the GE's Part 21 report will be applicable to the GESTR-M T-M assessment of this SE for future license application. GE submitted the T-M Part 21 evaluation, which is currently under NRC staff review. Upon completion of its review, NRC staff will inform GE of its conclusions.

LHGR and Exposure Qualification Limitation
In MFN 06-481, GE committed to submit plenum fission gas and fuel exposure gamma scans as part of the revision to the T-M licensing process. The conclusions of the plenum fission gas and fuel exposure gamma scans of GE

10x10 fuel designs as operated will be submitted for NRC staff review and approval. This revision will be accomplished through Amendment to GESTAR II or in a T-M licensing LTR. PRIME (a newly developed T-M code) has been submitted to the NRC staff for review (Reference 58). Once the PRIME LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDC-33173P must utilize the PRIME T-M methods.

The conditions specified in Section 4.21 and Section 4.22 of this SE complement Transient LHGR Limitation 3. Therefore, a 10 percent penalty is not required for TRACG04 methods when the conditions specified in Section 4.21 and Section 4.22 of this SE are met.

3.20.4 Control Rod Patterns and Transient Varying Axial Power

During the course of cycle operation many control rod patterns and core burn strategies are available to meet cycle operating limits. However, the core power distribution as a function of exposure is a strong function of this operating strategy and a factor influencing the core response to AOOs. Core response to pressurization transients, which tend to be limiting AOO transients in terms of thermal margin, is sensitive to the instantaneous void reactivity coefficient and core adjoint. To this end, top-peaked power shapes tend to be limiting in the assessment of pressurization events as: (1) the core adjoint is up-skewed resulting in lower control rod worth during the early portion of SCRAM, and (2) enhanced reactivity insertion due to void collapse in the high adjoint region of the core, and hence greater neutron flux increase as a result of the pressurization.

Cycle-specific analyses are performed during each reload to establish the OLMCPR by evaluating the thermal margin for limiting exposure points and transients. Therefore, the OLMCPR calculations must account for the sensitivity of the AOO response to control and burn strategies to ensure that the transient ΔCPR/ICPR is conservatively estimated to bound the initial power conditions projected for realistic cycle operation. GEH's methodology does not include specific uncertainties for power shape, but does require analyses using hypothetical burn strategies to maximize axial peaking for bottom-skewed and top-skewed power shapes at EOC.

The hard bottom burn (HBB) and under burn (UB) strategies are used to develop the cycle-specific analyses according to Reference 31. In the HBB strategy, deep control rods are used to suppress excess reactivity (in conjunction with flow reduction early in cycle for MELLLA+ plants). The deep rods and reduced flow result in high depletion in the bottom region of the core allowing the axial power shape at the EOC to become highly top-peaked. In the UB strategy, shallow control rods are used to reduce core reactivity, thereby resulting in bottom peaking when control rods are withdrawn at the EOC due to the low exposure of the bottom region of the core.

In its review of the applicability of ODYN to perform cycle-specific ΔCPR/ICPR calculations, the NRC staff raised concerns regarding the effects of transient varying axial power shape (TVAPS) and double humped power shapes on transient results. To address concerns regarding TVAPS, the NRC staff requested additional information for the TRACG04 application in RAI 33.

Particularly, the NRC staff is concerned about conservatism in transient evaluations in situations between the BOC and middle of cycle (MOC) when a SCRAM occurs as a result of an AOO. In response to the SCRAM, reactor power is reduced initially in the bottom portion of the core, shifting the axial power profile upwards in concert with decreased voiding, resulting in a larger amount of moderation in the upper portion of the core. The increased water density in the upper portion of the core, the upward shifted axial shape, and the harder spectrum exposure (and enhanced plutonium production) in the top of the core could result in a large transient response in neutron flux.

The NRC staff reviewed the response to RAI 33 and documented this review in Appendix A: Staff Evaluation of RAI Responses. In its review the NRC staff found that the analyzed exposure strategies and subsequent power shapes do not necessarily capture the limiting axial power shapes afforded by operational flexibility. The NRC staff also found that the conservatism of the black and white control blade assumption results in a [[ ]]. The NRC staff furthermore found that for bottom-peaked power shapes that are mildly up-skewed (i.e., bottom-peaked with the axial peak occurring above node four), the TVAPS may be more dominant than the increased SCRAM reactivity resulting in a CPR sensitivity that may be greater than 0.03 at MELLLA+ conditions.

The NRC staff considered BOC to MOC UB operation in concert with flow reduction afforded by MELLLA+ operation and found that the combination of the burn and flow control strategy may result in more limiting axial power shapes from the standpoint of TVAPS with a reduced compensating SCRAM reactivity worth. Therefore, the NRC staff requested supplemental information regarding the conservatism of the black and white rod pattern assumption for MELLLA+ conditions relative to BOC to MOC UB operation. This information was provided in Reference 48.

The NRC staff evaluation of the supplemental information is provided in Section A.33.4 of Appendix A. The NRC staff found that the analyses provide reasonable assurance that the black and white rod pattern conservatism is adequate to bound the effect of TVAPS for MELLLA+ plants at the MOC exposure point.

3.20.5 Mixed Core Evaluations

Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GEH's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in NEDC-33173P (Reference 31) and additional subjects relevant to application of GEH's methods to legacy fuel. Alternatively, GEH must supplement NEDC-33173P (Reference 31) for application to mixed cores.

The NRC staff did not assess the TGBLA06 upgrade for use with 11x11 and higher lattices, water crosses, water boxes, gadolinia concentrations greater than 8 w/o, or MOX fuels at EPU or MELLLA+ conditions. For any plant-specific applications of TGBLA06 with the above fuel types, GEH needs to provide assessment data similar to that provided for the GNF fuel products in Reference 31.

4 LIMITATIONS AND CONDITIONS

4.1 Historical Limitations and Conditions

All limitations and conditions imposed on TRACG02/PANAC10 documented in the NRC staff SEs attached to approved revisions to NEDE-32906P-A are considered applicable for TRACG04/PANAC11 unless otherwise specified in this SE. (References 2 and 3)

4.2 Interim Methods Limitations and Conditions

All limitations and conditions imposed on the TGBLA06/PANAC11 code system documented in the NRC staff SE for NEDC-33173P and the SEs for supplements to NEDC-33173P are applicable to their use in the TRACG04 code stream for AOO and ATWS overpressure calculations for EPU and MELLLA+ applications unless otherwise specified in this SE. (Reference 5)

4.3 Scope of Applicability Limitation

The approval of TRACG04/PANAC11 is limited to those specific applications reviewed by the NRC staff. The scope of review delineates those plant designs and conditions that the NRC staff considers to be the bounds of applicability. (Section 1.1)

4.4 Main Condenser Condition

Analyses performed for BWR/2-6 designs that include specific modeling of the condenser will require a plant-specific justification for its use. (Section 1.1)

4.5 Decay Heat Model Limitation

The NRC staff's acceptance of the TRACG04 decay heat model for simulating AOOs and ATWS overpressure does not constitute NRC staff acceptance of this model for LOCA applications. (Section 3.4.5)

4.6 Fuel Thermal Conductivity and Gap Conductance Condition

Until the NRC staff approves PRIME03 and approves the GESTR-M benchmarks, the NRC staff will require ATWS overpressure analyses and AOO analyses be performed using the GESTR-M mode. Should the NRC staff subsequently approve PRIME03, this approval will constitute approval of the PRIME03 improved thermal conductivity model for use in TRACG04 for AOO and ATWS overpressure analyses when used with PRIME03 dynamic gap conductance input. (Section 3.10.5.3)

4.7 ATWS Instability During Pressurization Limitation

The NRC staff has not reviewed the TRACG04 code for modeling density wave instabilities during ATWS events. Therefore, while it is not expected for typically limiting ATWS overpressure scenarios, should TRACG04 predict the onset of an instability event for a plant-specific application, the peak pressure analysis must be separately reviewed by the NRC staff. (Section 3.10.5.3)

4.8 Plant-Specific Recirculation Parameters Condition

Licensing calculations require plant-specific recirculation parameters to be confirmed against plant data and input into the TRACG model. (Section 3.13.4)

4.9 Isolation Condenser Restriction

On a plant-specific basis, any licensee referencing TRACG04 for ICS BWR/2 plant transient analyses will submit justification of the applicability of the KSP Correlation to model condensation in the ICS for pertinent transient analyses. This justification will include an appropriate sensitivity analysis to account for known uncertainties in the KSP Correlation when compared to pure steam data. The sensitivity of the plant transient response to the ICS performance is expected to depend on plant operating conditions, in particular the steam production rate. At EPU conditions the transient response is expected to be more sensitive to the ICS capacity given the increased steam flow rate at the same reactor core flow rate. The sensitivity is expected to be exacerbated at MELLLA+ conditions where the core flow rate is reduced. Therefore, licensees providing ICS BWR/2 plant-specific justification must provide such justification for each operating domain condition for which analyses are performed. (Section 3.15.5.3)

4.10 ATWS Transient Analyses Limitation

TRACG04 is not approved for analyses of reactor vessel ATWS overpressure after the point of boron injection. (Section 3.17.2 and Reference 3)

4.11 TRACG02 for EPU and MELLLA+ Limitation

The NRC staff has not generically reviewed the PANAC10 neutronic methods for application to EPU and MELLLA+ conditions. The NRC staff notes that initial comparisons between TRACG04 and TRACG02 for a representative EPU core (Section 3.18) indicate the TRACG02/PANAC10 methods are less conservative. Therefore, the NRC staff generic approval of TRACG04 for EPU and MELLLA+ licensing analyses does not constitute generic approval of TRACG02 for this purpose. (Section 3.18.9)

4.12 Quality Assurance and Level 2 Condition

TRACG04 must be maintained under the quality assurance process that was audited by the NRC staff as documented in References 25, 27 and 28 or a subsequent NRC-approved quality assurance process for ECPs in order for licensees referencing the subject LTR to comply with the requirements of 10 CFR Part 50 Appendix B. (Section 3.19)

4.13 Code Changes to Basic Models Condition

Modifications to the basic models described in Reference 26 may not be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval. (Section 3.19.2)

4.14 Code Changes for Compatibility with Nuclear Design Codes Condition

Updates to the TRACG nuclear methods to ensure compatibility with the NRC-approved PANACEA family of steady-state nuclear methods (e.g., PANAC11) may be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval as long as the ΔCPR/ICPR, peak vessel pressure, and minimum water level shows less than one standard deviation difference compared to the results presented in NEDE-32906P, Supplement 3. If the nuclear methods are updated, the event scenarios described in Sections 3.18.1 through 3.18.7 of this SE will be compared and the results from the comparison will be transmitted to the NRC staff for information. (Section 3.19.2)

4.15 Code Changes in Numerical Methods Condition

Changes in the numerical methods to improve code convergence may be used in AOO and ATWS overpressure licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50 Appendix B. (Section 3.19.2)

4.16 Code Changes for Input/Output Condition

Features that support effective code input/output may be added without NRC staff review and approval. (Section 3.19.2)

4.17 Updating Uncertainties Condition

New data may become available with which the specific model uncertainties described may be reassessed. If the reassessment results in a need to change a specific model uncertainty, the specific model uncertainty may be revised for AOO licensing calculations without NRC staff review and approval as long as the process for determining the uncertainty is unchanged and the change is transmitted to the NRC staff for information. (Section 3.19.2)

The nuclear uncertainties (void coefficient, Doppler coefficient, and SCRAM coefficient) are expected to be revised, as would be the case for the introduction of a new fuel design. These uncertainties may be revised without review and approval as long as the process for determining the uncertainty is unchanged from the method approved in this SE. In all cases, changes made to model uncertainties done without review and approval will be transmitted to the NRC staff for information. (Sections 3.19.2 and 3.20.2)

4.18 Statistical Methodology Limitation

Revisions to the TRACG statistical method may not be used for AOO licensing calculations without NRC staff review and approval. (Section 3.19.2)

4.19 Event-Specific Biases and Uncertainties Condition

Event-specific ΔCPR/ICPR, peak pressure, and water level biases and uncertainties will be developed for AOO licensing applications based on generic groupings by BWR type and fuel type. These biases and uncertainties do not require NRC staff review and approval. The generic uncertainties will be transmitted to the NRC staff for information. (Section 3.19.2)

4.20 Interfacial Shear Model Qualification Condition

Any EPU or MELLLA+ plant licensing analyses referencing TRACG04 methods for future GNF fuel products shall verify the applicability of the interfacial shear model using void fraction measurements or an alternative, indirect qualification approach found acceptable by the NRC staff. (Section 3.20.1)

4.21 Void Reactivity Coefficient Correction Model Condition

When performing transient analyses with TRACG04, the revised void reactivity coefficient correction model must be activated. (Section 3.20.2 and Appendix A: RAIs 29 and 30)

4.22 Void Reactivity Coefficient Correction Model Basis Condition

Licensees referencing NEDC-32906P, Supplement 3, for licensing applications must confirm that the lattices used in the void coefficient correction are representative of the plant's fuel or update the lattices such that they are representative. (Section 3.20.2 and Appendix A: RAIs 29 and 30)

4.23 Transient LHGR Limitation 1

Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the T-M acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel melting and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for the both the UO_2 and the limiting GdO_2 rods. (Section 3.20.3)

4.24 Transient LHGR Limitation 2

Each EPU and MELLLA+ fuel reload will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the SRLR or COLR, or it will be reported directly to the NRC staff as an attachment to the SRLR or COLR. (Section 3.20.3)

4.25 Fuel Thermal Conductivity for LHGR Condition

When TRACG04 is used to determine the limiting LHGR for transients, the GESTR-M thermal conductivity model must be used unless the NRC staff subsequently approves the PRIME03 models in a separate review. The fuel thermal conductivity and gap conductance models must be consistent. (Section 3.20.3)

4.26 Application of 10 Weight Percent Gadolinia Condition

Before applying 10 weight percent gadolinia bearing fuel to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for thermal and mechanical overpower conditions that bound the response of plants operating at EPU and expanded operating domains at the most limiting state points, considering the operating flexibilities (e.g., equipment out of service). (Section 3.20.3)

4.27 10 CFR Part 21 Evaluation of GESTR-M Fuel Temperature Calculation Limitation

Any conclusions drawn by the NRC staff evaluation of the GEH's Part 21 report (Reference 41) or subsequent benchmarking of GESTR-M is applicable to this SE. (Section 3.20.3)

4.28 LHGR and Exposure Qualification Limitation

The conclusions of the plenum fission gas and fuel exposure gamma scans will be submitted for NRC staff review and approval, and revisions to the T-M methods will be included in the T-M licensing process. This revision will be accomplished through an Amendment to GESTAR II or in T-M LTR review. Once the T-M LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDE-32906P, Supplement 3, must utilize these revised T-M methods to determine acceptable LHGR limits. (Section 3.20.3)

4.29 Mixed Cores Limitation

Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GEH's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in NEDC-33173P (Reference 31) and additional subjects relevant to application of GEH's methods to legacy fuel. Alternatively, GEH must supplement NEDC-33173P (Reference 31) for application to mixed cores. (Section 3.20.5)

4.30 Fuel Lattices Limitation

The NRC staff did not assess the TGBLA06 upgrade for use with 11x11 and higher lattices, water crosses, water boxes, gadolinia concentrations greater than 8 weight percent, or MOX fuels at EPU or MELLLA+ conditions. For any plant-specific applications of TGBLA06 with the above fuel types, GEH needs to provide assessment data similar to that provided for the GNF fuels for EPU or MELLLA+ licensing analyses. (Section 3.20.5)

4.31 Modified TGBLA06 Condition

Application of TRACG04/PANAC11 is restricted from application to EPU or MELLLA+ plants until TGBLA06 is updated to TGBLA06AE5 in the GEH standard production analysis techniques. (Appendix A: RAI 1)

4.32 Transient CPR Method Condition

Transient licensing calculations initiated from conditions where the MCPR exceeds 1.5 require evaluation of the adequacy of the transient CPR method and justification if the improved transient CPR method is not used. (Appendix A: RAI 3)

4.33 Direct Moderator Heating Condition

Application of the TRACG04/PANAC11 methodology to fuel designs beyond the GE14 fuel design will require confirmation of the DMHZERO value. (Appendix A: RAI 5)

4.34 Specifying the Initial Core Power Level Condition

For each application of the TRACG ATWS methodology, it must be made clear exactly what power level is being used, not only the percentage of licensed power, but the actual power level. (Reference 3)

4.35 Submittal Requirements Condition

The staff also notes that a generic topical report describing a code such as TRACG cannot provide full justification for each specific individual plant application. When a licensee proposes to reference the TRACG-based ATWS methodology for use in a license amendment, the individual licensee or applicant must provide justification for the specific application of the code in its request which is expected to include:

1. <u>Nodalization</u>: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

2. <u>Chosen Parameters and Conservative Nature of Input Parameters</u>: A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.

3. <u>Calculated Results</u>: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure. (Reference 3)

4.36 MELLLA+ Limitations

The NRC staff imposes all limitations specific to transient analyses documented in its SE (Reference 49) for the review of NEDC-33006P (Reference 46) for the application of the TRACG04 method to MELLLA+ conditions. Some of the limitations from Reference 49 pertinent to MELLLA+ transient analyses include, but are not limited to: 12.1, 12.2, 12.4, 12.18.d, 12.18.e, 12.23.2, 12.23.3, 12.23.8, and 12.24.1. For reference, the complete list of MELLLA+ limitations is provided in Appendix D: SE Limitations for NEDC-33006P from Reference 49.

5 CONCLUSIONS

Questions regarding the TRACG04 model for fuel thermal conductivity have prompted the NRC staff to specifically note that review of the subject LTR does not constitute an approval of the application of the current TRACG04 methodology to CRDA analysis (where the fuel enthalpy and Doppler feedback phenomena are highly important factors driving the transient response), LOCA analysis (where the stored energy is an important factor in predicting PCT), and time domain stability (where the fuel thermal time constant is an important parameter driving the void/reactivity coupling mechanism). Any future submittal requesting approval of the application of TRACG04 to the aforementioned analyses will require detailed justification and qualification of the thermal conductivity and gap conductance models.

The NRC staff did not review the application for ATWS event simulation post peak pressure or LOCA analysis. In the case of ATWS analyses post peak pressure or LOCA analyses, the uncertainty in time to boiling transition must be taken into account.

The NRC staff finds TRACG04 generically applicable to BWR/3-6 designs. Application of TRACG04 to ICS BWR/2 plants requires justification of the condensation model capabilities on a plant-specific basis.

If the NRC's criteria or regulations change so that its conclusions about the acceptability of the thermal-hydraulic, fuel performance, or nuclear methods or uncertainty analyses are invalidated, the licensee referencing the LTR (Reference 1) will be expected to revise and resubmit its respective documentation, or submit justification for the continued effective applicability of these methodologies without revision of the respective documentation.

The NRC staff has reviewed the TRACG04 code, and does not intend to review the associated LTR when referenced in licensing evaluations, but only finds the methods applicable when exercised in accordance with the limitations and conditions described in Section 4 of this SE. When exercised appropriately, the methods as documented in Reference 1 are acceptable for reference to perform transient AOO and ATWS overpressure licensing analyses.

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Date: January 8, 2009

APPENDIX A: NRC Staff Evaluation of Responses to Requests for Additional Information

By letter dated May 25, 2006, General Electric (GE) Nuclear Energy (now GE-Hitachi Nuclear Energy Americas LLC, hereafter GEH) submitted licensing topical report (LTR) NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO [anticipated operational occurrence] and ATWS [anticipated transient without SCRAM] Overpressure Transients" (Reference 1), for review and approval.

The NRC staff requested additional information to complete its review. GEH supplemented the content of the application with responses to this request by letters dated August 15 and December 20, 2007, and May 30, June 6, June 30, and July 30, 2008 (References 2, 3, 4, 5, 6, and 7, respectively). This appendix provides the NRC staff evaluation of these RAI responses.

RAI 1

The NRC staff requested that GEH specify the TGBLA06 code used for the upstream generation of nuclear parameters for PANAC11 and subsequently translated as a response surface to TRACG04 via the PANAC11 wrap-up file. The response states that the approved methods were used to generate the nuclear parameters in NEDE-32906P, Supplement 3. In particular, the code referenced is TGBLA06AE4. The modified version of TGBLA06AE5 includes improved resonance modeling capabilities to better predict the plutonium vector for hard spectrum (high void fraction) exposure.

The NRC staff has reviewed the modified TGBLA06AE5 code as part of the review of GEH codes and methods for expanded operating domains. In its review for Extended Power Uprate (EPU) and Maximum Extended Load Line Limit Analysis Plus (MELLLA+) applications, the NRC staff found that the modified TGBLA06 code provided more consistent results for hard spectrum exposure typical of operating conditions for EPU plants. Specifically, the NRC staff concluded in its safety evaluation (SE) for NEDC-33173P that "… the code-to-code comparisons provide reasonable assurance that the modified TGBLA06 neutronic methods are acceptable for analyzing the lattices and conditions for EPU and MELLLA+" (Reference 8).

Therefore, the NRC staff will restrict the application of TRACG04/PANAC11 to plants operating at originally licensed thermal power (OLTP) conditions until the TGBLA06 methodology is updated in the standard production analysis techniques to TGBLA06AE5. GEH will notify the NRC staff once relevant 10 CFR 50.59 reviews and quality assurance processes are complete. The NRC staff finds that application of the TRACG04/PANAC11 code system to EPU or MELLLA+ plants is only acceptable when TGBLA06AE5 is used to generate the nuclear parameters.

RAI 2

The NRC staff requested a qualitative discussion of the sensitivity of the thermal hydraulic core conditions to the PANAC11 kinetics solver relative to the PANAC10 kinetics solver. The NRC staff notes that based on the sample calculations provided in NEDE-32906P, Supplement 3, the TRACG04/PANAC11 code system (T4/P11) consistently predicts a higher neutron power than the corresponding transient using TRACG02/PANAC10 (T2/P10) for the pressurization events.

The response indicates that the void reactivity feedback predicted by T4/P11 is higher than the corresponding void feedback predicted by T2/P10 based on the differences in the PANAC11 methodology relative to the PANAC10 methodology. The resultant increase in core neutron power results in higher pressures, the results of which increase core inlet subcooling and impact feedwater flow. The NRC staff finds that the response is reasonable.

<u>RAI 3</u>

The NRC staff requested additional clarification regarding the calculation of the transient critical power ratio (CPR) in TRACG04. The response indicates that two methods are available. The first method predicts the CPR based on [[ ]]. The second method performs a rigorous calculation of the CPR [[ ]] is more computationally intensive, though [[ ]].

The revised methodology was submitted for NRC staff review and approval as Supplement 2 to NEDE-32906P. Sensitivity studies documented in the supplement have found that under certain conditions, the thermal margin to CPR relationship can result in errors in the ratio of transient change in critical power ratio to initial critical power ratio (ΔCPR/ICPR). The process for calculation of the transient CPR has been modified to reduce the error. The new process uses actual calculated parameters rather than a pre-defined relationship to get the instantaneous conditions. In so doing, the calculation of the transient CPR yields less error in the ΔCPR/ICPR ratio.

The NRC staff reviewed the method change and found that the improved method results in much more consistent results for the ΔCPR/ICPR for transient evaluations with both large and small margin to the safety limit minimum CPR (SLMCPR). [[ ]].

The NRC staff notes that the improved method predicts transient CPR more consistently than the base case; however, the NRC staff agrees that the base model is adequate to predict transient CPR for those transient conditions approaching the SLMCPR.

As a condition, the NRC staff will require that licensees referencing the subject LTR for anticipated operational occurrences (AOO) analyses [[ ]]

<u>RAI 4</u>

The NRC staff requested a correction to the model description regarding the number of decay heat groups. The NRC staff reviewed the response referencing the economic simplified boiling water reactor (ESBWR) documentation and found that the documentation had been appropriately revised.

<u>RAI 5</u>

In Part (a), the NRC staff requested that GEH explain how direct moderator heat is assigned to the coolant in the active channel, water rod, and bypass on a nodal level. The response indicates that the direct moderator heat is assigned based on weighting factors by the flow area and density of the coolant in each respective region. Since the heat is deposited predominantly by neutron scattering in the water, the NRC staff finds that the weighting is appropriate to capture the direct heat deposition in the coolant.

In Part (b), the NRC staff requested that GEH justify the use of the FDMH2=FDMH1 option. The user's manual specifies that this option may be non-conservative for post SCRAM loss-of-coolant accident (LOCA) evaluations of the peak cladding temperature (PCT). FDMH1 is the fraction of fission power that is directly deposited in the coolant at a reference density of 1.0 g/cc. FDMH2 is the fraction of decay heat that is directly deposited in the coolant. The default value for FDMH2 is set to 0.0. The NRC staff agrees that setting this value to 0.0 is conservative for LOCA analyses since the decay heat is then deposited in the fuel element and will result in a limiting calculation of the PCT following a SCRAM. The current practice for transient calculations, however, is to equate the direct heat fractions.

For anticipated transients without SCRAM (ATWS) overpressure analyses, the dome pressure is a strong function of the gross core thermal power during the pressurization transient, so these analyses are not sensitive to the means of heat deposition (either direct or through cladding heat flux) to the coolant. Therefore, assigning the same fraction of direct moderator heating from the decay heat is acceptable for ATWS overpressure analysis.

For AOO transient analyses, the figure of merit is the transient determination of the CPR. The CPR correlation is the GEXL (GE critical quality boiling length) correlation. The CPR performance is driven by the integrated heat deposited in the coolant below the point of boiling transition and is not inherently sensitive to the local cladding heat flux at a given axial elevation. Additionally, the direct moderator heat from decay heat will represent approximately [[

]] The resultant impact on CPR calculations from the fraction of direct heat from decay heat is negligible. Therefore, assigning equal fractions for the decay heat deposited from the fission and decay power is acceptable for AOO calculations.

In Part (c), the NRC staff requested justification of the default value for DMHZERO. DMHZERO is a parameter that describes the relationship between direct moderator heating and the water density in the bundle. DMHZERO was calculated [[

]] and the results are provided in Figure 5-11 of Reference 9. The response indicates that the value of DMHZERO in the user's manual is based on an assessment [[

]], the default DMHZERO value is applicable for GE14. Advanced fuel lattice designs, however, may include significant changes in the two dimensional lattice that affects the fraction of direct moderator heating. These changes may include changes to the fuel pin radii, part length rods, bundle pitch (i.e., N-lattice), or other geometry differences. The NRC staff will include a restriction that application of the default DMHZERO value to new fuel designs will require confirmation of its acceptability.

In Part (d), the NRC staff requested that GEH explain the direct heat model when a control blade is present. The control blade in the bypass will displace water and reduce the nodal bypass water content. The TRACG04 model, according to the response, [[

]] The NRC staff finds that the approximation is acceptable for AOO and ATWS overpressure analyses because the bundles that respond with the greatest change in CPR for AOOs are the uncontrolled bundles [[ ]]

In Part (e), the NRC staff requested that GEH describe the method for assigning direct gamma heat to the pressure vessel walls. In the RAI response, GEH indicated that TRACG does not assign direct gamma heat to the vessel walls. The NRC staff finds this approach acceptable because: (1) the amount of gamma heating in the vessel wall will be small due to shielding by the bypass and downcomer and (2) the heat deposited in the vessel wall will be effectively removed by the coolant flow in the annulus and therefore, effectively transferred to the coolant regardless.

In Part (f), the NRC staff requested that GEH explain the constants “a” and “b” in Equation 9.4-14. The constants characterize the distribution of direct gamma heat among the fuel clad, channel wall, coolant channels, and control blades. Since the gamma heat is predominantly deposited in high Z materials, the model is [[ ]] The formulation for neutron direct heating [[ ]] is, therefore, acceptable.

In Part (g), the NRC staff requested clarification of the model normalization. GEH responded by describing the transient $F_f(t)$ term (the fraction of direct heat deposited in the fuel) that ensures power fractions sum to unity. The response is acceptable.

In Part (h), the NRC staff requested details of the TRACG uncertainty analysis regarding the direct heat model. The response states that a total uncertainty of [[ ]] is applied to account for all individual component uncertainties. GEH performed a sensitivity analysis by perturbing the direct moderator heating by [[ ]] and found that the CPR change is on the order of [[ ]]. Therefore, the NRC staff finds that a more accurate assessment of the uncertainties is not required and will not impact the use of TRACG04 for modeling AOOs or ATWS overpressure transients.

<u>RAI 6</u>

GEH responded to RAI 6 by referencing the response to RAI 21.6-82 on the ESBWR Docket, which requested the same information regarding transient xenon for the anticipated operational occurrence/infrequent events (AOO/IE) and ATWS calculations for the ESBWR. The response states that the xenon concentration is assumed constant during AOO and ATWS overpressure transients. Since limiting fuel conditions (i.e., CPR) and vessel conditions (i.e., peak pressure) are achieved within minutes following the initiation of the transient event, the NRC staff agrees with GEH’s response that there is insufficient time for the xenon concentration to evolve during the response to affect the nuclear characteristics within the core appreciably. Therefore, the NRC staff finds that the constant xenon assumption will have a negligible impact on the calculation of margin to pressure, CPR, and heat generation rate limits. Therefore, for application to AOO and ATWS overpressure analyses the NRC staff finds that the neutronic modeling of xenon is acceptable.

<u>RAI 7</u>

The NRC staff requested information regarding the implementation of the void coefficient correction model. The void coefficient correction model is used in TRACG04 to correct the void reactivity predicted by PANAC11 [[ ]] This information was initially requested in the review of TRACG04 for application to ESBWR AOO/IE and ATWS calculations. The void coefficient correction model is reported in the subject LTR in Section 5.1 according to the RAI response.

The void coefficient correction in TRACG04 was revised relative to the correction model in TRACG02 since the kinetics solver in TRACG04 is based on the improved neutronic methods PANAC11 and TGBLA06. To determine the necessary corrections, several TGBLA06 lattices were compared to MCNP analyses. In the correction model, only uncontrolled lattices were considered.

The uncontrolled TGBLA06 calculations were used to correlate the infinite eigenvalue as a function of [[ ]] The correlation was then used to determine the void coefficient as a function of [[ ]] by taking the derivative with respect to the void fraction. These comparisons were performed for [[

{3}]]

The NRC staff notes that TGBLA06 cannot be directly compared against MCNP [[ ]] due to the TGBLA06 exposure chain model [[ ]] Therefore, to directly compare the TGBLA06 and MCNP results, they must be compared [[ ]] In the NRC staff audit (References 10, 11, and 12) of the nuclear design methods for the application to ESBWR, GEH provided details of the TGBLA06/MCNP comparison procedures.

[[ ]] To compare with MCNP, [[ ]] material compositions are taken from TGBLA06 and input into MCNP, and then the [[ ]] MCNP eigenvalues and fission densities are directly compared.

The effect [[ ]] has a minor impact on calculational results. During the NRC audit analyses for representative lattices demonstrated [[ ]] cross sections result in small variations in the pin-wise fission density, resulting in a maximum root mean squared (RMS) difference for very high exposure [[ ]]. Therefore, the NRC staff finds that the use [[ ]] to facilitate direct TGBLA06 and MCNP comparisons provides a reasonable basis for assessing TGBLA06 calculational efficacy.

To cover the range of void fractions from 0 percent to 100 percent, the correlated fit of lattice reactivity according to TGBLA06 branch cases performed at [[

]]

The NRC staff notes that TGBLA06 has recently been revised to incorporate substantial improvements (TGBLA06AE5). The modified TGBLA06AE5 includes a more robust model for absorption in the low-lying [[ ]] plutonium-240 resonance. In addition to the modified resonance absorption model, TGBLA06AE5 also includes an error correction to the thermal scattering matrix normalization. In the NRC staff's review of these changes, the NRC staff found that the TGBLA06AE5 representation was more accurate than that of previous variants of the TGBLA06 code and compared lattice calculation results on a pin-by-pin basis. As documented in Reference 11, the NRC staff found that the corrections resulted in minor impacts on plutonium depletion effects [[ ]], but did have a relatively large impact on the rod fission power for pins near the water rods. This is attributed to corrections to the thermal scattering matrix, given that these rods are adjacent to a strong slowing down source. However, these differences are well within the quoted uncertainty for TGBLA06 and produce more accurate results than the previously qualified variants of TGBLA06.

In reviewing the modified TGBLA06, the NRC staff found that the modified TGBLA06 provided much more consistent comparisons with higher order methods. The NRC staff requested information regarding TGBLA06 modifications in RAI 14 as part of its review of LTR NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated February 2006. In its review of a plant-specific application, the NRC staff performed independent analyses comparing the TGBLA06AE5 modified code against the HELIOS code (Reference 13). These results were incorporated by reference in the NRC staff's review of NEDE-33173P. The NRC staff found that TGBLA06AE5 results in more accurate evaluation of lattice parameters at very high void fractions.

In its review of the application of TGBLA06 to the ESBWR, the NRC staff requested that the TGBLA06AE4 methods being applied in the nuclear design analysis be compared against those approved for the generic application of the nuclear design methods to expanded operating domains (EPU and MELLLA+) plants given similar concerns regarding computational efficacy for high in-channel void fractions. GEH provided comparisons between the TGBLA06AE4 (standard production code) to the modified TGBLA06AE5 (more robust code) using extrapolation from the standard production in-channel void fractions predicted by TGBLA06AE4 against eigenvalues explicitly calculated by TGBLA06AE5 for high in-channel void fractions. The results were transmitted to the NRC staff in response to RAI 4.3-3 on the ESBWR Docket File in Reference 14.

The response indicates that extrapolation from the TGBLA06AE4 standard production void fractions to 90 percent in-channel void fraction results in lattice parameter predictions that are essentially the same as those predicted by the more robust TGBLA06AE5 code. Therefore, the NRC staff accepts the GEH approach of calculating the TGBLA06 eigenvalues by extrapolating from the in-channel void fractions to 100 percent. The NRC staff notes that, given comparisons to the TGBLA06AE5 code, the extrapolation technique is actually expected to confer a greater degree of accuracy than performing explicit TGBLA06AE4 calculations at 100 percent in-channel void fraction and is more representative of how the TGBLA06AE4 lattice parameters are manipulated in the PANAC11 kinetics solver.

The void reactivity coefficient biases are based on uncontrolled lattices depleted at 40 percent in-channel void fraction [[ ]] Controlled lattices were not considered, because evidence from TGBLA04 comparisons to MCNP indicates that the uncontrolled lattices are bounding. The uncontrolled lattices are expected to yield greater biases and uncertainties as the presence of the control blade results in significant spectrum hardening due to strong thermal neutron absorption in the blade. The resulting hardening of the spectrum results in the eigenvalue becoming more sensitive to the resonance escape probability dependence on the void fraction from the fuel utilization, thereby reducing the sensitivity of the void reactivity to local thermal neutron effects – which are more sensitive to geometric modeling assumptions. Therefore, the NRC staff finds that considering the uncontrolled lattices only is expected to bound the void reactivity coefficient uncertainty and result in a larger calculated bias. As the code scaling applicability and uncertainty (CSAU) process accounts for these uncertainties, the NRC staff finds that the current approach will conservatively estimate the uncertainty in transient response to uncertainty in the void reactivity coefficient.

The NRC staff considered the applicability of the void coefficient correction model to EPU and MELLLA+ conditions. EPU cores are generally designed by flattening the radial core power shape relative to a pre-EPU core. In doing so, the highest power bundle tends to remain the most limiting bundle while other non-limiting bundles have increased power. To sustain the higher core power level through the same cycle duration, the core must be a high energy core. A high energy core has significant reactor physics attributes that differentiate such a design from a pre-EPU, pre-extended cycle core.

High energy cores require high burnable poison loadings. The high poison loadings are necessary to compensate for the additional excess reactivity that is required to sustain core criticality for the same cycle duration with a higher thermal power. In addition to these high burnable poison loadings, a larger fraction of assemblies are typically loaded in each cycle to also increase the core cycle energy.

High energy cores also tend to operate with non-standard control strategies. A standard example would be a black and white (B&W) control rod pattern with an aim towards achieving a Haling depletion. High energy cores are typically depleted in a spectral shift manner to maintain core power while achieving the desired duration. The control blade density at the beginning of cycle (BOC) and during the peak reactivity exposure point tends to be larger compared to pre-EPU core designs.

A combination of higher batch reload fraction and a higher loading of neutron poison, both in the form of burnable poisons and control blades tends to harden the neutron spectrum during cycle exposure. Additionally, as the average bundle power is increased, the core average void fraction tends to increase. The combination of higher inventories of thermal neutron absorbers, more fissile content, and higher void fractions may result in a hard spectrum that can result in uncertainties in important neutronic parameters over exposure that have not been previously quantified or accounted for based on operating experience in a much softer exposure-averaged neutron spectrum.

Aside from these effects at the bundle level, the increase in total core power will have an impact on the core bypass conditions. During normal operation a fraction of the fission power is released in the form of radiation, which is directly deposited in the coolant and core structures. The increase in reactor thermal power will result in an increased heat load to the core bypass region, which may result in either lower bypass subcooling, or potentially the formation of significant void in the core bypass. The formation of void in the bypass contributes to spectrum hardening.

The MELLLA+ operating domain exacerbates the spectral hardening effect by maintaining steam flow at reduced core flow conditions, resulting in an increase in core average void fraction at 100 percent currently licensed thermal power (CLTP).

The hardened neutron spectrum at EPU and MELLLA+ conditions has prompted the NRC staff to request information regarding the adequacy of the TRACG04 void coefficient correction model to account for any effects of hard spectrum exposure on the void reactivity coefficient. Under hard spectrum exposure, the fuel has a greater affinity for converting fertile uranium to plutonium. In so doing, the dynamic void reactivity coefficient may be biased as a result of increased plutonium conversion relative to the bias predicated on comparisons at a lattice average in-channel void fraction of 40 percent [[
]]

Operation at EPU and MELLLA+ results in a significantly larger number of bundles accruing exposure under higher void and harder spectrum conditions, resulting in greater degrees of plutonium conversion, and an overall impact on the nodal and core average void reactivity coefficient. As a core becomes increasingly under-moderated the void reactivity coefficient will increase. The presence of low lying plutonium resonances may enhance or damp the void reactivity coefficient depending on the relative production and destruction rates of plutonium-239 and 240 under nodal exposure conditions, however, without accounting for exposure effects the TRACG04 void coefficient correction model may under-predict the void reactivity coefficient for EPU and MELLLA+ conditions, resulting in non-conservative estimates of pressurization power response.

Therefore, the NRC staff does not find that the response to RAI 7 is acceptable based on the scope of the subject review (which includes EPU and MELLLA+ plants). The NRC staff requested consideration of the void reactivity coefficient bias with high void exposure in RAI 30.

RAI 8

GEH provided a table of contents to a PANACEA wrap-up file. The NRC staff reviewed the contents to determine if the PANACEA wrap-up contained sufficiently detailed parameters to allow for the initialization of the TRACG power distribution while maintaining a sufficiently detailed characterization of the nuclear parameters to allow the TRACG kinetics solver to model the neutronic feedback. The wrap-up file contains both the functional cross sections and power distribution, and therefore, in the initialization procedure the functional cross sections are preserved, allowing for accurate feedback modeling. Therefore the NRC staff determined that sufficiently detailed nuclear information is conveyed from the PANACEA wrap-up file to TRACG to both initialize the model and provide for acceptable kinetic feedback modeling.

RAI 9

The NRC staff requested information regarding isotopic tracking. The GEH methodology performs isotopic tracking at the lattice level using TGBLA06 calculations; however, it does not track isotopics in full core modeling. The core simulator and the TRACG04 kinetics are based on evaluating nodal neutronic parameters based on a response surface as a function of exposure and exposure history using quadratic fitting functions. Therefore, no explicit isotopic tracking is required to predict nodal reactivity or buckling. As such, the NRC staff did not review any capability in the code stream to track isotopes.

RAI 10

The NRC staff requested how PANAC11 calculates the power where boiling transition occurs. PANAC11 bundle power is iteratively adjusted to calculate the nodal quality, boiling length, annular length, mass flux, inlet subcooling, and axial power shape. These parameters are input to the GE critical quality boiling length correlation (GEXL) to determine the critical quality. The nodal qualities are compared to the GEXL critical quality. If the nodal qualities are higher than the critical quality the bundle power is reduced until exactly one node has a quality equal to the critical quality. The bundle power where a single nodal quality is equal to the critical quality is the critical power. The critical power ratio is based on the predicted bundle power determined from the normal exposure analysis and the iteratively determined critical power. The NRC staff finds this approach acceptable.

RAI 11 (deleted)

RAI 12

The NRC staff requested clarification and a correction to the documentation. This correction was made.

RAI 13

The NRC staff requested that GEH provide the TRACG04 Qualification LTR. Revision 3 of LTR NEDE-32177P was provided in Reference 15.

RAI 14

The NRC staff requested information regarding the modifications to the TGBLA06 code. This information was supplied to the NRC staff in response to RAI 4.3-3 on the ESBWR Docket File in Reference 14. The NRC staff found this response acceptable for the ESBWR. Since the ESBWR fuel lattices are substantially the same as GE14 lattices for the operating fleet, and TGBLA06 is a lattice physics code, the NRC staff's technical basis for the acceptance of the response for the ESBWR is equally applicable for the operating fleet. Greater discussion of the response is included in the NRC staff's evaluation of the response to RAI 7.

RAI 15

The NRC staff requested information regarding any modifications to PANAC11. During its review of the PANAC11 methods for the ESBWR, the NRC staff conducted an audit of the nuclear design codes TGBLA06 and PANAC11. The results of the audits are documented in References 10, 11, and 12. The code changes are listed and summarized in Appendix B: TGBLA06/PANAC11/TRACG04 Code Changes of the subject SE. The NRC staff found that the code changes did not constitute a methodology change.

RAI 16

The NRC staff requested that GEH provide justification for the use of the improved thermal conductivity model based on PRIME03. The improved model includes corrections for the fuel thermal conductivity to account for the effects of fuel exposure and the presence of gadolinia on the fuel conductivity.

For the current application (AOO and ATWS overpressure) the fuel temperature prediction affects the analyses in the coupling between the fluid conditions and the neutron flux. In particular, calculation of the fuel thermal conductivity will impact the fuel thermal time constant and the predicted transient fuel temperature. In cases where the predicted thermal conductivity

is large, the fluid condition and the neutron flux are more tightly coupled via the heat flux through the pellet, gap, and cladding.
Similarly, the calculation of the transient change in fuel temperature is used to predict the nodal Doppler reactivity worth, which in turn, is assessed in the neutronic model to determine the transient reactivity feedback and neutron flux.

The NRC staff reviewed the information contained in the RAI response to determine the acceptability of using the improved thermal conductivity model for transient analyses. In its review, the NRC staff compared the improved model against the FRAPCON3 fuel thermal conductivity model.

First, the NRC staff notes that in the temperature range between 1000K and 2000K, the PRIME03, GSTR-M, and FRAPCON3 models predict essentially the same fuel thermal conductivity at zero exposure for pure urania fuel. The predicted thermal conductivity as a function of temperature for these models is depicted in Figure A.16-1. The GSTR-M or TRACG02 fuel conductivity model does not consider any effect on the thermal conductivity from exposure [[ ]] or gadolinia.

In comparing the PRIME03 or TRACG04 model against the GSTR-M and FRAPCON3 models, the NRC staff plotted the variation in thermal conductivity as a function of the exposure for zero gadolinia concentration. The results are shown in Figure A.16-2 below. The GSTR-M model shows only a slight variation with exposure. The exposure dependence of the GSTR-M thermal conductivity is based on [[ ]] The FRAPCON3 and PRIME03 models indicate similar trends in thermal conductivity with exposure and show a significantly greater degree of agreement when compared to the GSTR-M model.

The NRC staff also considered the impact of gadolinia on the thermal conductivity. The NRC staff finds that at high exposure, the TRACG04 model does not predict any influence on thermal conductivity by the gadolinia, whereas the FRAPCON3 model consistently predicts degradation in thermal conductivity with increasing gadolinia concentration. The PRIME03 model is compared to the FRAPCON3 model for zero exposure and for high exposure in Figures A.16-3 and A.16-4, below respectively. The NRC staff notes that gadolinia isotopes are naturally stable, and expects that the depletion of gadolinia 155 and 157 under irradiation will result in the production of the stable gadolinia 156 and 158 isotopes (with small absorption cross sections). Therefore, the NRC staff does not accept the conclusion that gadolinia depletion under irradiation results in a negligible impact on fuel thermal conductivity at the end of life for the fuel, because the NRC staff expects that the concentration of gadolinia itself does not appreciably change during irradiation.

Therefore, the NRC staff finds that: (1) the new fuel thermal conductivity model captures the effect of exposure on fuel thermal conductivity and agrees well with the FRAPCON3 model and (2) there is evidence that the new fuel thermal conductivity model remains non-conservative in the prediction of pellet temperature for gadolinia loaded fuel pins.

[[ ]]

Figure A.16-1: Comparison of 0 Exposure, 0 Gadolinia Fuel Conductivity Models as a Function of Temperature

[[ ]]

Figure A.16-2: Comparison of 0 Gadolinia Fuel Conductivity Models as a Function of Exposure at 1000K

[[ ]]

Figure A.16-3: Comparison of 0 Exposure Fuel Conductivity Models as a Function of Gadolinia at 1000K

[[ ]]

Figure A.16-4: Comparison of 65 GWD/ST Exposure Fuel Conductivity Models as a Function of Gadolinia at 1000K

RAI 17

The NRC staff requested that GEH provide justification for the use of inconsistent fuel thermal resistance models, particularly, the GSTR-M predicted gap conductance files and the PRIME03-based TRACG04 improved thermal conductivity model. TRACG04 explicitly calculates the fuel pellet dimensions. [[ ]]

The NRC staff finds that while the increased temperature will affect the fission gas release, changes in the overall rod thermal resistance are counter balanced by the closing or widening of the gas gap consistent with the pin operating history. Therefore, the NRC staff finds that the response is acceptable and will have a negligible impact on the analysis of AOOs or ATWS overpressure events.

RAI 18-20 (deleted)

RAI 21

The NRC staff requested additional information regarding the uncertainty in the Doppler coefficient and SCRAM reactivity. The response was provided in Reference 16. The response states that the SCRAM reactivity uncertainty is driven by the uncertainty in the SCRAM speeds. The SCRAM speeds are based on full-scale tests and are not dependent on the analysis tools. The NRC staff finds that the response is acceptable.

The Doppler coefficient uncertainty was preserved as the TRACG02 value [[ ]], GEH provided analyses of the special power reactor test (SPERT) reactivity insertion test 43 with perturbed Doppler worth. When the nodal Doppler coefficient was multiplied by [[ ]] for all nodes within the core, the TRACG04 and SPERT experimental powers were in very close agreement. The NRC staff notes that the measurement uncertainty bands for the SPERT test are relatively large compared to the sensitivity demonstrated for [[ ]] uncertainty in the Doppler coefficient. The NRC staff finds that the SPERT tests are adequate to justify a [[ ]] in the Doppler coefficient and indicate that a [[ ]] is reasonable.

The NRC staff finds that the available data and technique are acceptable based on the sensitivity of the ΔCPR/ICPR value to Doppler coefficient uncertainty. Further justification is not required as sensitivity analyses performed by GEH confirm that the statistical analysis results are insensitive to this parameter and uncertainties on the order of [[ ]] are required to substantially impact the calculated ΔCPR/ICPR errors.

RAI 22

The NRC staff requested additional information regarding the energy release per fission. When the 3D kinetics model is activated, the energy release per fission is tracked as a function of nodal parameters via lattice parameter input from TGBLA06 and explicitly calculated for each neutronic node in the PANAC11 solver internal to TRACG04 based on the PANACEA wrap-up data file. For evaluations where the kinetics solver is disabled, the decay heat fission energy release values are based on historical values reported in GEH LTR NEDO-23729 (Reference 17).

NEDO-23729 (Reference 17) has been provided to the NRC in response to the NRC staff RAIs regarding the subject LTR. The energy release per fission values for the fissile isotopes is based on the least-squares assessment reported by Sher (October 1976). The fertile isotopes with the exception of thorium-232 are based on a systematic evaluation by Sher (October 1976), thorium-232 energy release per fission is based on the least-squares approach. The least-squares approach combines calculations using the mass defect with experimental observations. The NRC staff finds that the approach appropriately leverages available theoretical data, including evaluated nuclear data file (ENDF) libraries (ENDF-IV) and experimental data and is therefore acceptable.

<u>RAI 23</u>

The NRC staff requested additional information regarding the implementation of the American Nuclear Society (ANS) standard decay heat models in TRACG04. The response indicates that the default decay heat model remains the May-Witt model. The May-Witt five group model is approximately 15 percent conservative relative to the ANS standard, and for AOO and ATWS overpressure analyses, the NRC staff finds that its continued application is acceptable due to the conservatism in the integrated heat load for loss of feedwater (LOFW) and ATWS overpressure.

The ANS standard models (1979 or 1994) represent best estimates of the decay heat energy deposition. The transient power is based on a time integration of the power history to determine the decay heat. As described in the GEH response, and also in Section 9.3.1 of Reference 18, the power history is accounted for in two pieces. The long term exposure history is accounted for by approximating the integrated power history to the start of the transient using the channel powers and integrating over a time duration sufficient to yield the same channel group exposure at the specified channel power. The recent history is captured by performing step integrations of the transient power during the TRACG calculation.

The sensitivity of the power to the decay heat variations over the transient is very limited for AOO analyses given the short time frame prior to SCRAM. Following the SCRAM the reactor power is sufficiently reduced that CPR margins are maintained, and therefore, accurate modeling of the decay heat following the SCRAM is not generally required. However, for ATWS or small break LOCAs, the SCRAM may be delayed and the short term transient neutron power, and downstream calculated decay heat response, may have a greater effect on the thermal margins.

Since the AOO transient prediction of CPR margin is insensitive to the decay heat following SCRAM, and that the SCRAM occurs shortly into the transient, the NRC staff finds that the precise treatment of the decay heat time integration will have only a negligible impact on the licensing calculations. The NRC staff finds that use of either the ANS standard models or the May-Witt model is acceptable since the ANS standard is widely used by the industry for the subject application and the May-Witt model produces bounding conservatism estimates of the integrated thermal load.

For ATWS overpressure, the subject application is only for the prediction of the peak pressure. The peak pressure will occur shortly into the transient. While there is the potential for the short term power history under ATWS conditions to affect the long term decay heat modeling, the TRACG04 calculation for ATWS overpressure is terminated shortly after initiation of the event. During the ATWS overpressure scenario, the transient response is a very strong function of the 3D kinetic behavior and the void reactivity coefficient. The use of any decay heat model will negligibly impact the predicted ATWS peak pressure, since the kinetic power will dominate the response. Therefore, the NRC staff finds that the ANS standard is acceptable for this purpose.

<u>RAI 24</u>

The TRACG04 flow regime map and entrainment model were modified to improve agreement with low pressure Toshiba data. The purpose of the update was to improve calculational accuracy for the ESBWR LOCA calculations. In response to this RAI, the NRC staff requested information regarding the uncertainty analysis with respect to the void fraction. The TRACG02 and TRACG04 codes were assessed against the full-scale bundle test facility (FRIGG) void fraction data measurements of [[ ]] The assessment indicates that the TRACG02 code predicts the void fraction with a mean error of [[ ]] and a standard deviation of [[ ]] TRACG04 predicts the void fraction with a mean error of [[ ]] and a standard deviation of [[ ]] The bias in TRACG04 has [[ ]] is conservative for AOO analyses. Therefore, GEH concluded that the TRACG02 uncertainty analysis is applicable to the TRACG04 code. The NRC staff agrees that the data most relevant to AOO applications is the high pressure FRIGG OF-64 data and furthermore agrees that based on similarities in the qualification [[ ]] that the TRACG02 uncertainty analysis is applicable.

<u>RAI 25</u>

The NRC staff requested that GEH update the model description with implementation details of the optional 6-cell jet pump model. The NRC staff reviewed the proposed revision and finds the revision acceptable.

<u>RAI 26</u>

GEH provided the qualification of the 6-cell jet pump model with modified loss coefficients. The response demonstrates an improvement in the uncertainties associated with the jet pump. The qualification database includes full-scale tests as well as a scaled experiment with reverse drive flow. The qualification illustrates an improvement in the prediction of the N-ratio, even under reverse flow conditions. Therefore, the NRC staff finds that the uncertainty analysis is not adversely impacted and the 6-cell jet pump model with modified loss coefficients is acceptable. The NRC staff finds that the sensitivity analysis, whereby the loss coefficients were changed using TRACG04 phenomena identification and ranking table (PIRT) parameters 70 and 71, provides an adequate technical basis for acceptance of the model.

RAI 27

The NRC staff finds the response acceptable. The requested information was provided in the expanded discussion in response to RAI 16. The NRC staff furthermore notes that the Model Description LTR (Reference 18) provides the means for specifying the fuel thermal conductivity model. This reference had not been provided when the ESBWR related RAI (21.6-93) was issued.

RAI 28

The NRC staff requested information regarding the use of the TRACG04 default pump homologous curves. The homologous curves are based on full-scale test data (Bingham) and are representative of boiling water reactor (BWR) recirculation pumps. For plant-specific applications, plant data regarding the pump rated speed, flow, head, torque, density, and inertia are input in the plant-specific model for rated conditions. The default pump curves are used to model the transient conditions. The NRC staff finds this approach acceptable to capture the plant-specific characteristics of the recirculation pump.

RAI 29

The void reactivity coefficient bias and uncertainties in TRACG must be representative of the lattice designs of the fuel loaded in the core. GEH provided the lattice information describing the lattices used to develop the void reactivity coefficient biases and uncertainties. These lattices include 8x8 through 10x10 (GE9, GE10, and GE14 fuel products). The NRC staff does not find that these biases and uncertainties are generically applicable, but are dependent on lattice features that may affect calculational efficacy or the validity of assumptions in developing the neutronic solutions. Therefore, the NRC staff will impose a condition that for application for fuels other than those included in the data set used to develop the void reactivity biases and uncertainties, the biases and uncertainties must be demonstrated to be applicable. In cases where these biases and uncertainties are determined not to be applicable, they must be updated for the new fuel application. The NRC staff also notes that the representative lattice designs do not include non-GEH fuel designs. The NRC staff will impose a similar condition for legacy fuels in the specific case of mixed core evaluations.

The NRC staff evaluation of the applicability of TRACG04 to EPU and MELLLA+ mixed core analysis was reviewed separately and is documented in Section 4.20.5 of the subject LTR.

RAI 30

In RAI 30, the NRC staff requested that GEH revise the void reactivity coefficient correction model to account for void history effects in the determination of the void reactivity coefficient biases. GEH has developed the revised model and implemented the model in TRACG04. Details regarding the model were provided to the NRC staff in Reference 19.

The response provides descriptive details of the implementation of the void history correction model. This model is implemented to account for biases and uncertainties in the TRACG04 void reactivity feedback as calculated by the PANAC11 kinetics engine.

The historical void reactivity coefficient correction has been evaluated by the NRC staff in the response to RAI 7 and found unacceptable for application to EPU and MELLLA+ application as the pervious model was based on [[ ]]

The revised model is based on comparisons between TGBLA06 and MCNP for [[ ]]

The NRC staff has previously issued RAIs in similar reviews regarding the applicability of the database used to calculate the eigenvalue response surfaces to advanced fuel designs. The response to RAI 30 indicates that the TRACG04 revised [[ ]] Therefore, the NRC staff requires for licensing applications that any licensee referencing NEDC-32906P, Supplement 3, confirm that the lattice database is applicable to the specific cases considered, or revise the database input to ensure that the database is consistent with the fuel being analyzed.

The basis for the correction model is to perform lattice calculations using TGBLA06. The predicted infinite eigenvalue is compared to eigenvalues predicted using a sophisticated MCNP. [[ ]]

The NRC staff has reviewed the basis for the comparison noting that a code-to-code comparison is used. The response states, and the NRC staff agrees, that the MCNP qualification is extensive and indicates very small biases and uncertainties, such that there is a high degree of confidence that any uncertainty in the MCNP prediction is sufficiently small that the code to code comparison will serve as an acceptable indication of any bias or uncertainty in TGBLA06.

Furthermore, the NRC staff notes that the comparisons were performed for uncontrolled lattices. In its evaluation of the response to RAI 7, the NRC staff has concluded that the use of the uncontrolled lattices will bound any uncertainty for similar analyses performed for controlled lattices.

The void reactivity correction model response surface has [[ ]] The NRC staff finds that this approach is acceptable and appropriate because it is characteristic of the means by which the TGBLA06 calculations are used in the PANAC11 code. That is, errors associated with extrapolation of TGBLA06 parameters in PANAC11 are included in the uncertainties and biases by comparing the extrapolated values against MCNP instead of direct TGBLA06 calculations. The intention of the correction model is not to characterize the efficacy of the TGBLA06 code, but rather to normalize the PANAC11 neutronic response to match the more accurate void coefficient predicted by MCNP.

The results of the comparisons for modern fuel designs were evaluated statistically. The NRC staff has reviewed the results of these comparisons and finds that the results indicate normality of the uncertainties.

Equation 17 provides the means by which the correction model is implemented in TRACG. [[

]]

The void reactivity coefficient ratio is fitted based on the [[

]] The NRC staff finds that the extrapolation from higher void conditions is acceptable to characterize the general behavior of the void coefficient. The NRC staff finds this acceptable on the basis that as void fraction increases, the void reactivity coefficient tends to increase in magnitude and become more negative. Therefore, the correction model at low void conditions is providing a correction to a nodal response that is somewhat insensitive, and also to a nodal response that is non-limiting (low void fractions correspond to low power). Generally these nodes do not play a significant role in the transient progression in terms of overall core response.

The NRC staff has reviewed the fitting and interpolation schemes for the discrete points in the database and found these techniques to be acceptable. On the basis of the fitting and interpolation techniques and the range of void fractions covered by the database, the NRC staff finds that the void reactivity coefficient correction model is acceptable to characterize the biases and uncertainties in the void reactivity coefficient in TRACG over a range of instantaneous and exposure-weighted void fractions between 0 percent and 100 percent.

GEH has provided a sample calculation demonstrating the effect of the void reactivity correction model. Two representative pressurization transient analyses were performed using TRACG04. In one case, the void reactivity coefficient correction model was deactivated. The calculations indicate that the ΔCPR/ICPR is sensitive to the void reactivity coefficient correction and the predictions [[ ]] in the maximum ΔCPR/ICPR. The NRC staff finds that [[ ]] and agrees with GEH that the new model continue to be applied for AOO analyses. The NRC staff will impose a condition that transient analyses for licensing applications must be performed with the revised void reactivity coefficient correction model activated.

[[ ]]

RAI 31

The NRC staff requested that GEH justify the use of the interfacial shear model for modern fuel designs for normal operation, transient, and accident conditions at EPU and MELLLA+ conditions. The response provides the aggregate data used to qualify the TRACG04 interfacial shear model, including the low pressure Toshiba data. The response also provides indirect qualification of the interfacial shear model against pressure drop data collected during GE14 critical power testing.

GEH considered the bundle conditions during AOOs initiated from EPU and MELLLA+ conditions to determine the required range of applicability of the interfacial shear model generically. These ranges are specified in terms of bundle power, flow, pressure, and void fraction in the response. The NRC staff agrees with GEH's basis for evaluating these ranges based on conditions at critical power.

The application of the interfacial shear model to high pressure experienced during AOOs was qualified against high pressure FRIGG OF36 data to justify the application of the model to the subject application. The FRIGG OF36 test includes a full-scale 6x6 bundle with a hydraulic diameter similar to current fuel designs [[ ]] The calculated and measured void fractions are provided in the response.

The assessment indicates a mean void fraction error of [[ ]] and a standard deviation of [[ ]] than the error predicted for the FRIGG OF64 or Toshida tests [[ ]]. However, the response quotes a larger experimental uncertainty for the FRIGG OF36 test, and therefore, [[ ]] in the predictive capability of the interfacial shear model at higher pressures.

The total qualification database is summarized in the response. The table is reproduced below in Table A.31.1. The NRC staff reviewed the accompanying details in the TRACG04 Qualification LTR (Reference 15). The range of tests encompasses bundle conditions for the range of AOO and ATWS overpressure applications when considered with the additional FRIGG OF36 high pressure data up to [[ ]] The concert of qualification data indicates stability in the interfacial shear model to predict the void fraction consistently over a large range of pressures, mass fluxes, and hydraulic diameters. The qualification against the Toshiba data at low pressures [[ ]]

The data also includes a wide range of hydraulic diameters. The Ontario Hydro tests were included in the qualification of for the TRACG04 application for the ESBWR. The qualification demonstrates that for very large diameters, the interfacial shear model predicts the void fraction with a mean error of [[ ]] This is consistent with the predictive capability of smaller hydraulic diameters. For very large hydraulic diameters, such as the Bartolomei tests, the void fraction is very reliably predicted.

The reported standard deviation for the experimental BWR (EBWR) test in the response is [[ ]]. The TRACG04 Qualification LTR describes the EBWR qualification in Section 3.1.4 of Reference 15. The test consists of pressure measurements in the EBWR chimney. The inlet quality was determined based on the heat balance. The experimental uncertainty was not reported, but has been inferred to be on the order of 2.0 percent based on data scatter. Several tests were performed for several power levels, reaching 100 MW in order to develop a range of two-phase conditions in the chimney section. The agreement between the TRACG04 prediction and the EBWR data is approximately [[ ]]. The agreement is very good considering the very large conical shaped chimney and that the void fraction in the test varies over the chimney diameter due to wall shear effects. The somewhat increased uncertainty [[ ]] for this test relative to the balance of the database is attributed to the use of a one-dimensional pipe model for the large chimney.

The database also includes the FRIGG and CISE tests, which are typically used across the industry to qualify void fraction models, and their inclusion for the subject application is appropriate.

Considering the range of parameters, sufficient tests have been performed to encompass the expected bundle thermal-hydraulic conditions for AOOs and ATWS overpressure events initiated from EPU and MELLLA+ conditions.

The NRC staff further notes that, considering all of the separate effects tests, the interfacial shear model appears to reliably predict the void fraction with a consistent error. This reliability indicates that the mechanistic approach developed for the interfacial shear model is robust in modeling a range of geometry and fluid conditions. To provide the NRC staff greater assurance of the applicability of the model to modern fuel design, GEH provided additional qualification of the model through indirect qualification against pressure data collected during critical power testing of the GE14 fuel bundle design.

The critical power data collected for low flow was considered in the assessment. For the low flow tests the two-phase pressure drop is minimized and the pressure drop is driven by buoyancy effects. The void fraction is the key driver of buoyancy pressure drop and, therefore, the subset of considered data is particularly relevant in the indirect assessment. The pressure drop qualification conservatively assigns all uncertainty, including experimental uncertainty and spacer loss uncertainty, to an uncertainty in the void fraction calculation. The results of the qualification against these data indicate a [[ ]] These uncertainties are consistent with the trends observed in the remainder of the TRACG04 separate effects qualification database.

Therefore, the NRC staff is reasonably assured that the interfacial shear model will adequately predict transient void fraction for bundle conditions anticipated for AOO and ATWS overpressure events initiated from EPU and MELLLA+ conditions.

Table A.31.1: Interfacial Shear Qualification Database and Results

Test	Pressure	Mass Flux	Inlet Subcooling	Hydraulic Diameter	Void Fraction	Mean Error	Standard Deviation
	MPa	kg/sq-m/s	K	m			
FRIGG OF64	[[						
Christensen							
Wilson							
Bartolomei							
EBWR							
CISE							
Toshiba							
Ontario Hydro							
FRIGG OF36							]]

RAI 32

In its review of the PANAC/ISCOR/ODYN/TASC code system to analyze the transient response of plants operating at EPU and MELLLA+ conditions, the NRC staff identified concerns regarding the adequacy of the Findlay-Dix void quality correlation. To address these concerns the NRC staff imposed two limitations on its approval of NEDC-33173P to address potential uncertainties in the transient response arising from errors in the predicted void fraction. In its SE of NEDC-33173P, the NRC staff states that conclusions regarding the TRACG interfacial-shear model will be applicable to its use at EPU and MELLLA+ conditions.

The NRC staff requested information regarding the qualification of the interfacial shear model in RAI 31 and documented its review of these qualification data in this Appendix. The NRC staff found that the interfacial shear model is a detailed mechanistic model of the interphasic friction phenomena, giving the NRC staff reasonable assurance based on its qualification (including pressure drop data collected during GE14 critical power testing) that it reliably predicts the change in void during transient events characteristic of transients initiated from EPU or MELLLA+ conditions.

However, the TRACG04 simulation is predicated on the initialization of the transient analysis to the steady-state conditions predicted by PANAC11. The PANAC11 thermal hydraulic solution is based on the Findlay-Dix void quality correlation. Therefore, the prediction of nodal nuclear parameters may be affected by the prediction of the steady-state void fraction. Errors in the void fraction affect the nodal reactivity feedback characteristics and, therefore, may have a significant impact on downstream transient analysis results. To address this concern, the NRC staff requested that GEH evaluate the impact of initialization to the PANAC11 void fraction distribution on limiting transient response for challenging conditions typical of EPU or MELLLA+ operation.

GEH provided a response to the NRC staff RAI in Reference 20. The NRC staff requested that GEH determine: (a) the impact of the void quality correlation uncertainty on the void reactivity coefficient uncertainty, (b) provide a code-to-code comparison illustrating the effects of the void fraction mismatch during initialization on the transient response, and (c) provide additional information regarding the qualification of the Findlay-Dix void quality correlation.

RAI 32(a)

The response states that the nuclear uncertainties are captured in the TRACG04 transient analysis. In particular, the nodal void reactivity coefficient is corrected in accordance to normalization to MCNP results as a function of the void history and instantaneous void conditions. The NRC staff reviewed this approach and found this acceptable to improve void reactivity feedback calculations as described in the NRC staff's evaluation of the response to RAI 30 in this Appendix and in Section 4.20.2 of this SE. The NRC staff agrees that the void reactivity coefficient uncertainty is based on explicit lattice calculations to account for the nuclear methods uncertainty. However, the NRC staff was requesting that GEH evaluate the impact of Findlay-Dix void fraction uncertainty on the transient analysis as the void reactivity feedback is sensitive to the instantaneous void conditions. The NRC staff finds that the response is acceptable as it clarifies the basis for the void reactivity coefficient uncertainty, and the NRC staff's technical concerns are adequately addressed with the information provided in response to item (b) of the NRC staff's RAI 32.

RAI 32(b)

GEH provided an analysis using a modified version of TRACG04 to assess quantitatively the impact of the initial void fraction mismatch between TRACG and PANACEA. The analysis is performed by bypassing the standard TRACG initialization process and running TRACG in a transient mode to allow the PANAC11 nuclear engine to reach a steady-state condition consistent with the TRACG thermal hydraulic models, which include the interfacial shear model to determine the nodal void fraction.

As a result of running TRACG in this manner, the initialization transient results in a reactivity imbalance and subsequently a slightly different initial power level. To account for the power level mismatch resulting from the modified initialization, [[

]] GEH attributes this small deviation to the fact that the interfacial shear model and the void quality correlation share the same development basis. The NRC staff agrees that the magnitude of the deviation is therefore expected. The NRC staff agrees that including [[ ]] the reactor power level without impacting the void fraction mismatch effect on the core power distribution and therefore provides a valid basis for comparison of the transient response to pressurization.

The sensitivity analysis is performed using a large BWR/4 model consistent with EPU operating conditions (Case A). The response compares the axial power distribution in the hot channel between the original and modified TRACG calculations. The results indicate minor deviations in the axial power that are consistent with the magnitude of the deviation in the predicted axial void fractions. Therefore, the NRC staff finds that this approach adequately captures the impact of the void fraction difference on the initial nodal reactivity feedback characteristics with the appropriate magnitude and, furthermore, demonstrates that the modified model predicts expected results.

A pressurization turbine trip without bypass transient is initiated for the original and modified TRACG models. The results of the analysis confirm that the initial power pulse is only mildly affected by the void fraction mismatch. The subsequent reactor power during the SCRAM indicates some minor deviations, but these differences are consistent in magnitude with the void fraction mismatch magnitude. The basis for comparison is the hot channel ΔCPR/ICPR. These results are provided over the course of the simulated transient in the RAI response. The numerical results indicate that the[[

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The NRC staff reviewed the transient CPR curves to ensure that the code predicts consistent physical behavior between the two versions. The NRC staff finds that the thermal margin consistently increases during the initial void collapse and decreases in response to the power pulse and transient cladding heat flux. During the SCRAM a second peak occurs in the transient response that is consistent in magnitude and timing between the two cases due to transverse axial power shape (TVAPS) effects. Therefore, the NRC staff finds that the analyses indicate that the TRACG/PANACEA void fraction mismatch does not significantly impact any of the prevailing transient phenomena important to the prediction of CPR margin.

The response also provides details of the statistical nature of the void fraction mismatch. These data are provided on the basis of nodal relative water density as opposed to void fraction directly. As described in detail in Equation C-1 of Appendix C: Sample Calculation of Void Reactivity Sensitivity, the relative water density and in-channel void fraction are directly related according to the fuel bundle geometry (which is captured in the in-channel to total flow area ratio (VFAT) term). Therefore, this parameter provides an indirect measure of the void fraction mismatch between the two codes.

The NRC staff notes that the statistical nature of the void fraction mismatch is not to be construed as an uncertainty. It is a deterministic bias that occurs for a specific analysis at a nodal level based on the different void fraction models. Statistical information regarding the mismatch is therefore only useful in gauging the performance of the full system model to calculate the reactor conditions during steady state and transient conditions.

The NRC staff notes that the mismatch numerical results based on void fraction are subtly different than the values generated by the TRACG edit; however, the NRC staff notes that these values may be manipulated according to the known relationship between the two parameters to quantify the mismatch between the TRACG and PANACEA predicted void fractions.

Using the relative moderator density differences as a surrogate to approximate the void fraction mismatch, the NRC staff finds that there is adequate indication to find that the differences between the Findlay-Dix predictions and the interfacial shear model tend to

be within the uncertainty in the void fraction used to establish the statistical ΔCPR/ICPR for AOO analyses. This result is expected as stated previously as the two models share the same development basis data. However, the NRC staff notes that the interfacial shear model directly models the interphasic friction and is a detailed mechanistic model of the two-phase flow, while the robustness of the Findlay-Dix correlation is limited by the data used in its development and qualification.
Therefore, when considered in concert with the qualification of the interfacial shear model against the GE14 pressure drop data described in the response to RAI 31, the NRC staff has reasonable assurance that potential errors in the PANCEA predicted void fraction during TRACG initialization will have a negligible impact on the predicted ΔCPR/ICPR for limiting transients. The NRC staff also notes that the void fraction uncertainty is considered explicitly in the TRACG statistical evaluation to determine the ΔCPR/ICPR uncertainty and that the sensitivity study indicates that considering an additional uncertainty due to the mismatch between the PANACEA and TRACG at EPU conditions does not have a sufficient impact on the calculated ΔCPR/ICPR to merit a thermal margin enhancement to ensure adequate safety.

<u>RAI 32(c)</u>

The NRC staff requested information regarding the qualification of the Findlay-Dix model. The response states that the Findlay-Dix model is adequately qualified. The NRC staff, as noted previously, finds that the empirical nature of the Findlay-Dix correlation makes it difficult to determine the uncertainty in its predictions for conditions slightly beyond the scope of its qualification, such as for application to fuel bundles with modern geometric features (such as part length rods, 10x10 arrays, or modern fuel spacers). Likewise, the data used to qualify the Findlay-Dix correlation for prototypical fuel geometries at conditions encountered during transients initiated from EPU or MELLLA+ conditions is limited.

The response to RAI 31 concludes that the MELLLA+ application does not warrant further consideration as the bundle conditions during normal operation and during transients must demonstrate margin to the onset of transition boiling. Therefore, GEH concludes that the void fraction predictions are adequately qualified. The NRC staff, as stated previously, is concerned that uncertainty in the prediction of the PANACEA steady-state void fraction may result in the miscalculation of the nodal reactivity feedback in response to void changes during transient evaluations. While the NRC staff has reviewed the interfacial shear qualification, including the updated qualification against GE14 critical power data, the NRC staff is concerned that the inter bundle nuclear coupling may amplify the impact of errors in the predicted nodal reactivity feedback characteristics. The bundles are coupled by internodal neutron leakage. Potentially increased errors in neighboring bundle void reactivity feedback will have a direct effect on the efficacy of the code to accurately determine the limiting bundle transient response.

The response to RAI 33 correctly states that the channel response is a function of the core environment from which any transient is initiated. To address concerns regarding the impact of the void fraction mismatch, GEH has provided calculations to address the NRC staff's concerns at MELLLA+ conditions. Two cases were considered, B and C. The B case is evaluated using initial conditions established using the original and modified initialization process at the intersection of the high flow control line (HFCL) and the licensed thermal power line (LTPL) of the MELLLA+ domain (100 percent rated thermal power (RTP)/ 85 percent rated core flow (RCF)). The C case considers the impact of the void fraction mismatch at the intersection of the HFCL and the transition line (77.6 percent RTP/55 percent RCF).

The response states that the core average void conditions are expected to be largely similar along the HFCL. As the core reactivity is constant during steady-state operation, and maneuvering along the HFCL is done without movement of the control blades, the NRC staff agrees that the adjoint weighted core average void fraction is not expected to change, since the core remains critical at both points. Comparisons of these cases indicate only a small change in core average void fraction between Points B and C on the MELLLA+ operating map.

The response also provides the calculated void fraction mismatch based on the relative water density mismatch. The equation provided in the response relates the nodal water density mismatch to the nodal void fraction mismatch. The NRC staff notes that this void fraction mismatch should not be construed as the mismatch between the in-channel void fractions predicted by TRACG04 and PANACEA, [[

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However, for the purpose of responding to the NRC staff's RAI, this parameter serves as an adequate metric to quantify the mismatch between the steady-state (Findlay-Dix) and transient (interfacial shear) void fraction models.

The response includes a discussion of the statistical nature of the mismatch and compares this mismatch to the void fraction uncertainty propagated in the statistical analysis. First, the NRC staff notes that the statistical information provided for the calculated nodal mismatch values should not be construed as an uncertainty because it is a deterministic bias at the nodal level. However, as the purpose of the analysis methodology is to evaluate the limiting channel behavior based on core response, the biases introduced by the mismatch appear to impact the analysis similarly to void fraction uncertainties. Second, the NRC staff is interested in the impact of any errors or biases introduced in the steady-state calculation affecting the nodal response to void change in the transient response and the subject methodology is acceptable to address the NRC staff's concerns.

The response correctly states that a bias introduced in the initial void fraction is not expected to significantly impact the change in void fraction predicted by TRACG04 in response to a transient pressurization event. The NRC staff agrees with this statement; however, the NRC staff notes that the nodal response surfaces passed from PANACEA to TRACG04 in the initialization accommodate [[

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The void reactivity coefficient is known to increase in magnitude and become more negative with increasing instantaneous void conditions.

Operation at MELLLA+ conditions, particularly at the transition corner in the domain, may result in a substantial increase in core average void fraction. The void reactivity coefficient tends to increase in magnitude and become more negative with increasing void fraction. Therefore, the NRC staff expects that the transient response to pressurization will be exacerbated due to a higher void fraction at MELLLA+ conditions along the HFCL. Since the nuclear power response of any bundle is governed by the void conditions within the bundle, as well as the internodal leakage from neighboring bundles, the higher core average void fraction may result in an amplification of the limiting bundle power response to the pressurization and void collapse within the limiting bundle and its neighbors. In cases where the void fraction mismatch is exacerbated, the NRC staff would expect that the errors in the limiting bundle ΔCPR/ICPR would increase by a greater amount than indicated by the EPU transient analysis provided in response to RAI 32(b) due to a potentially greater sensitivity in bundle power response to void collapse at higher void conditions.

The NRC staff must note that the effect of a potential [[

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However, the NRC staff agrees with the response insofar as the modified and original initialization procedures allow for a means to quantify the impact on the final transient response arising from the interdependence of the PANACEA calculated initial void distribution and the TRACG04 calculated void distribution.

The response compares the mean values of the nodal relative water density mismatch between the TRACG04 and PANACEA solutions. The purpose of providing this mean bias information is to demonstrate that the biases do not increase with increasing core average void fraction in the MELLLA+ operating domain. The response indicates that the bias in the void mismatch remains consistent between Cases A, B, and C. The comparison demonstrates the efficacy of the Findlay-Dix correlation up to MELLLA+ steady-state operating conditions. Therefore the NRC staff finds that an increase in the sensitivity of the transient response to the mismatch is not expected at MELLLA+ conditions.

The response provides similar figures for Cases B and C as those provided for the EPU case considered in RAI 32(b). The results indicate a more substantial shift in axial void and axial power at these conditions as shown in Figures 32-5, 32-6, 32-9, and 32-10 of the response. The response compares the differences in the nodal powers to the nodal power uncertainty based on the uncertainties in NEDE-32601P-A. The NRC staff finds this comparison to be somewhat misleading. The uncertainties in NEDE-32601P-A are evaluated based on an older TGBLA/PANACEA code system and substantial improvements have been made relative to these methods in the TGBLA06/PANAC11 code system. This comparison is misleading only in the basis of the uncertainty value quoted. The uncertainty quoted in the response is the uncertainty used to develop the cycle-specific SLMCPR so it is valid for comparison, albeit not fully self consistent as it is based on qualification of the historical core monitoring methods.

In all cases, the predicted nodal powers were within the uncertainty used in the SLMCPR analysis. Similarly, in all cases, the RMS void mismatch was approximately [[

]] which is well beneath the threshold of significance.

As the purpose of the analysis in the response is to quantify the bias introduced in the transient response as a result of the deterministic nodal void fraction mismatch, the NRC staff does not find the one standard deviation uncertainty band to be an acceptable basis for the threshold of significance. The interdependence of the void reactivity coefficient uncertainty and void fraction uncertainty is not explicitly accounted for in the uncertainty analysis. Therefore, the NRC staff maintains that a threshold of significance of 0.005 remains appropriate when evaluating a potential bias. The NRC staff reiterates that the void fraction mismatch is a deterministic evaluation of the differences in two void fraction models used analytically.

In regards to this metric for significance, the NRC staff observes that the transient response sensitivity for Case C [[ ]] The difference in the predicted ΔCPR/ICPR is [[ ]]. The analysis indicates that the modified TRACG04 initialization produces the more limiting response. The response states that the initial axial power shape sensitivity to the initialization process becomes more evident at reduced core flows.

For operation at MELLLA+ conditions, the axial power shape tends to shift downward in the core for operation at the transition corner. The reduced flow results in a redistribution of the core void fraction. While the reactor is along the same rod line (the HFCL of the MELLLA+ upper boundary) and the core average void fraction does not appreciably change, the onset of boiling tends to shift downward in the core. Under these low flow conditions, the axial power shape also shifts downward due to the reduced moderation in the upper portions of the core. Figures 32-5 and 32-9 demonstrate the downward shift in reactor power for Case C relative to Case B.

The plots in Figures 32-3, 32-6, and 32-10 indicate [[

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As the reactor power is somewhat shifted downward in the core for the MELLLA+ transition point on the HFCL, the NRC staff expects that a greater sensitivity in the transient response would be observed as the reactor adjoint has shifted to a greater extent into the region of the core where the void fraction mismatch is greatest. This effect is observed in the results of the analyses provided in response to the RAI.

The NRC staff does not agree with GEH that the impact on the numerical result is insignificant. While the resultant ΔCPR/ICPR changes by approximately one standard deviation for the modified TRACG04 initialization case, the NRC staff finds that the standard production TRACG04 analysis at the MELLLA+ transition corner appears to be less conservative than the modified TRACG04 analysis methodology. The NRC staff similarly finds that the degree of sensitivity exceeds the threshold of significance.

However, the NRC staff finds that along the LTPL that the results of the analysis are insensitive to the void fraction mismatch for the limiting initial conditions for the limiting transient analyses. While the NRC staff disagrees that analyses performed at the transition corner exhibit no significant sensitivity to the mismatch, the NRC staff agrees with the basis in the response that the Case C analysis will not be limiting on a cycle-specific basis, and therefore does not contribute to determining the cycle-specific operating limit MCPR (OLMCPR).

The response states that the transition corner is non-limiting relative to the Case B point at the intersection of the HFCL and LTPL for several reasons. First, the reactor is at a lower power level and therefore, the steam flow rate through the main steam line is lower. The lower steam flow rate will result in a milder back pressure wave in response to a pressurization initiating event. The NRC staff agrees with this point. Second, the reactor power shape is downward shifted at the transition corner relative to the high-power low-flow corner of the MELLLA+ domain. The response states that the downward shifted power results in an enhanced SCRAM worth under these conditions. The NRC staff agrees that the SCRAM worth is expected to increase with the downward skewed axial adjoint. But likewise, the NRC staff finds that downward skewed power shapes are less limiting in terms of pressurization transients as the pressure wave is dissipated by void collapse in the upper regions of the core predominantly, therefore, making the up-skewed power shapes the most limiting. Third, the response states that the back pressure effect on the core flow rate is less severe at low flow conditions. The NRC staff likewise agrees with this point. Therefore, the NRC staff agrees that the pressurization transient response of a core operating at the transition corner of the MELLLA+ operating domain is inherently bounded by the high-power low-flow corner state point.

Therefore, while the NRC staff finds that void fraction uncertainty under certain conditions (such as the transition corner of the MELLLA+ operating domain) may have an impact on the calculated transient CPR in excess of the threshold of significance, the NRC staff finds that a thermal margin enhancement is not necessary to address reload licensing applications. The response adequately demonstrates that for the magnitude of the void fraction mismatch that the limiting transient responses are negligibly affected.

The NRC staff's conclusions here are predicated on pressurization transients being the limiting transients in reload licensing analyses. This is true for the operating fleet of BWR/2-6 reactors.

<u>RAI 33</u>

Considering the CPR response benefit from the use of TRACG instead of ODYN code set, the NRC staff expects most licensees will migrate to TRACG for AOO. For operation at EPU and MELLLA+ conditions, where the CPR response will potentially be higher due to the fixed SRV relief capacity relative to the increase in the pressurization response, TRACG, which has the capability to simulate 3D core conditions, is expected to be more attractive to licensees.

Therefore, it is important to evaluate the TRACG AOO methodology for defining the control rod patterns and the corresponding axial power shapes modeled in TRACG applications. LTR NEDE-32906P, Revision 2 (Reference 21) contains the following pertinent information:

(1) Section 7.5.2.7, "High Worth Scram Rods for Pressurization event OLMCPR," describes the initial conditions used to minimize the SCRAM worth.

(2) Section 8.0, "Demonstration Analysis," covers the bases for application of TRACG for AOO, using sensitivity analyses to establish the initial conditions and assumptions that will be applied on plant-specific bases.

(3) Section 8.2, "Initial Conditions and Plant Parameter Review," defines the initial conditions that are demonstrated to have an impact on the AOO response.

(4) Table 8-9, "Allowable Operating Range Characterization Basis," lists the key parameters that influence the AOO response. For the axial power shape, the table states that the cases are analyzed at nominal (top-peaked) end of cycle (EOC) conditions and at EOC bottom-peaked conditions.

(5) For the control rod pattern, Table 8-9 states that cases are analyzed at middle of cycle (MOC) with a nominal rod pattern and with a conservative B&W rod pattern.

From the discussion in Section 8.2 of NEDE-32906P (Reference 21), it is not apparent that the bounding axial power shapes or control rod patterns assumed ensure that the plant operates with some control rod flexibility while ensuring that the assumed axial power shapes bound the power shapes the plant experiences. Therefore, the adequacy of these assumptions in terms of the control rod patterns and the corresponding axial power shapes assumed needs to be reconfirmed for operation at EPU and MELLLA+.

The NRC staff requested that GEH provide additional information regarding the control rod patterns assumed for TRACG04 AOO analyses, namely that GEH:

- For the plant-specific MELLLA+ application of TRACG04 to AOOs, demonstrate that the limiting control rod patterns assumed in the power history envelops and bounds the axial power peaking the plant will experience at different exposure ranges.
- Discuss how the limiting control rod patterns assumed as the core depletes minimizes the scram reactivity worth.
- Provide an assessment of TVAP that would result from the SCRAM during power profiles other than top-peaked.

TVAPS phenomenon is a flow reduction effect caused by the rapid void collapse when the power is suppressed in the bottom part of the fuel bundles as the control rods insert during SCRAM. The channel flow stagnates as it occupies the collapsed void region and then continues to pick up energy as it traverses to the top of the fuel bundle. The fluid enthalpy at the top of the channel may lead to dryout conditions.

Since TVAPS is primarily a flow effect, the fluid transport velocity affects the flow reduction and timing of the maximum impact at the channel exit. At lower channel flows as expected for MELLLA+ operation, the impact on the mass flux is greater. In addition, the timing of the maximum impact will shift to later times in the transient. With a reduction in channel flow, the TVAP change in quality is larger but is also shifted later in the transient. However, the total impact will not be seen in the ΔCPR analysis since the timing is shifted beyond the time of the maximum fluid enthalpy.

The overall sensitivity to channel flow is variable from plant-to-plant and the extent by which TVAPS can cause the ΔCPR to increase is limited. The power-to-flow ratio is not a significant contributor to the TVAPS severity, because the ICPR for the hot channel is set such that the transient MCPR is equal to the SLMCPR. Therefore, the pre-EPU MELLLA hot channel will experience a similar thermal-hydraulic transient as an MELLLA+ hot channel.

A.33.1 Axial Power Shape

Plant-specific licensing analyses are performed using the conservative approach consistent with the TRACG02 application. Both HBB and UB strategies are simulated to develop top-peaked and bottom-peaked EOC power shapes. This is consistent with the ODYN approach. The UB power shape is included to account for the potentially limiting impact of TVAPS. The HBB and UB strategies are intended to bound the operational flexibility in control rod pattern during cycle operation.

The TRACG04 plant-specific transient calculations are performed assuming HBB from BOC to MOC and MOC to EOC, as well as assuming UB from MOC to EOC to ensure a bottom-skewed EOC power shape. To demonstrate the conservatism in the assumed burn strategies, GEH provided sensitivity studies performed using ODYN during the NRC staff review of ODYN for application to EPU and MELLLA+ plants. Particularly, the NRC staff requested information regarding the effect of bottom-skewed or double-humped power shapes during early cycle exposure and the effect on transient analysis.

In its review of the additional information, the NRC staff determined that the BOC to MOC HBB strategy is typically limiting, with some exceptions. The NRC staff found that when the BOC to MOC UB power shape is not highly bottom-skewed that the difference in [[ ]] (Reference 8). The NRC staff notes that these calculations were performed with the ODYN code; however, finds that the results are consistent with expected phenomenological sensitivity in the response to axial power shape variation and are indicative of expected trends for TRACG04.

Therefore, the NRC staff concludes that the assumed burn strategies do not explicitly account for limiting axial power shapes. The NRC staff furthermore concludes that these results are primarily an effect of TVAPS. Double-humped power shapes occur for partially inserted control rods, which enhances the SCRAM reactivity during transient evaluations and tends to result in less limiting CPR evaluations. Therefore, the NRC staff agrees that double-humped power shapes will not result in limiting transient responses and are not required for specific evaluation for cycle operating limit determination.

A.33.2 Control Rod Pattern

The NRC staff requested information regarding the limiting control rod pattern. The response directed the NRC staff to Table 8-10 of Reference 9. In the original application of TRACG02 for AOOs, GE performed sensitivity analyses to determine the sensitivity of the thermal margin to initial plant parameters. An analysis was performed for a turbine trip with no-bypass (TTNB) using nominal EOC axial power shapes with a B&W control rod pattern and a nominal control rod pattern (with several rods partially inserted).

The sensitivity analysis indicates a [[ ]] in predicted ΔCPR for the nominal control rod pattern for a TTNB. The [[ ]] predominantly to a reduced SCRAM reactivity worth. The B&W pattern reduces the SCRAM worth as: (1) the fully inserted control blades do not contribute to the SCRAM worth, (2) the fully withdrawn control rods initially add negative reactivity in the low adjoint bottom of the core, limiting the total negative reactivity insertion, and (3) there are no partially inserted control rods, which would contribute a large negative reactivity insertion early in the SCRAM because of their tip's proximity to the high adjoint region of the core.

For the TTNB sensitivity analysis the [[ ]] condition was selected. The analyses were performed for the B&W control rod pattern and the nominal control rod pattern. Two hot channels were considered (Channels 27 and 29). The results of the sensitivity analyses are provided in Table A.33.2.1.

The NRC staff found that there is [[

]] (Reference 8).

A.33.3 Transient Varying Axial Power

The NRC staff reviewed the response and agrees that, generally, the TVAPS is most severe for bottom-peaked power shapes, but is compensated for an increased SCRAM reactivity due to a down-skewed flux adjoint. The sensitivity analyses performed using TRACG02 indicate that the sensitivity of the transient CPR to power shape is on the [[ ]] (Reference 9).

However, in its review of BOC to MOC for the review of the ODYN code, the NRC staff found that [[

]]

EPU plants typically operate with a very limited operational flow window. Therefore, the NRC staff does not expect that BOC to MOC UB analyses will be sensitive to the variation in flow associated with the burn strategy for EPU plants and the general results of the ODYN sensitivity analysis can be applied. MELLLA+ operation allows variation in core flow to control excess reactivity. In the BOC to MOC the strategy may involve reduction in core flow. The net combined effect [[

]] At the MELLLA+ flow corner the SCRAM worth is further reduced due to an increase in core average void fraction and hence a hardened neutron spectrum. The TRACG02 sensitivity analyses only consider an UB bottom-peaked power shape with a peak in node 3 (Figure 8-35 of Reference 9).

In its review of the application of ODYN to EPU and MELLLA+, the NRC staff concluded that despite non-conservatism in the BOC to MOC power shapes, sufficient conservatism was included in the assumed control rod pattern to ensure that overall analysis results remained conservative. Sensitivity analyses performed with TRACG02 indicate that the [[ ]] While the NRC staff concludes that there is sufficient margin to EPU plants, the NRC staff cannot reach the same conclusion regarding TRACG04 for application to MELLLA+ based on the above discussion.

The NRC staff requested that additional transient analyses for MELLLA+ plants be performed assuming a BOC to MOC UB strategy with flow reduction to ensure that the axial power shape bounds potentially limiting axial power shapes during exposure when determining the cycle OLMCPR.

A.33.4 Supplemental Information

The NRC staff cannot conclude based on the response that the B&W control rod pattern conservatism in the analysis is sufficient to bound the limiting power shapes for BOC to MOC UB exposure. Particularly, the NRC staff is concerned that under MELLLA+ conditions at the low flow/100 percent CLTP MELLLA+ corner that the TVAPS effect may be magnified by the axial power shape at reduced flow conditions and will not be compensated by high SCRAM reactivity because of spectrum hardening at the reduced flow condition.

The NRC staff requested that GEH provide the results of analyses for a large, representative MELLLA+ BWR/4 to demonstrate the effect of BOC to MOC UB at the MELLLA+ corner on (1) axial power shape, (2) TVAPS effect, and (3) ΔCPR/ICPR. Compare these results to BOC to MOC HBB results. In response to the NRC staff's request, GEH evaluated the conservatism of the B&W rod pattern for MELLLA+ conditions. The response transmits a detailed description of those aspects of the analysis assumptions and important phenomena that ensure the most limiting power shapes are bounded by the cycle-specific analyses.

GEH performed explicit calculations using a 560 bundle BWR/4 operating at 2923 MWth (120 percent OLTP). The BOC to MOC UB and HBB depletion strategies were used. At the MOC condition, TTNB and FWCF events were simulated. These events were simulated using TRACG02. The NRC staff has evaluated TRACG04 relative to TRACG02 as discussed in the body of this report. While the NRC staff finds that the TRACG02 kinetics methods are less robust than the PANAC11 based engine in TRACG04, the NRC staff finds that for the purpose of demonstrating the relative effect of TVAPS the analyses are adequate.

Several events were initiated from the MELLLA+ corner (120 percent OLTP/85 percent RCF). Table 1 of the supplemental response provides the results of the transient analysis and Figure 1 provides a depiction of the axial power shapes considered. The axial power shapes range from highly bottom-peaked (BOC to MOC UB B&W pattern) to shapes that are relatively flat (BOC to MOC HBB B&W pattern).

The results of the analysis indicate substantial conservatism is maintained for the MELLLA+ condition assuming the BOC to MOC HBB B&W pattern relative to the BOC to MOC nominal blade pattern. The analysis also considers BOC to MOC UB with a B&W pattern and demonstrates that the BOC to MOC HBB still results in conservative transient results despite a power shape that is relatively flat (as opposed to top-peaked).

The BOC to MOC UB B&W pattern cases indicated very mild transient responses for TTNB indicating that the enhanced SCRAM reactivity continues to dominate the TVAPS effect at MELLLA+ conditions. The BOC to MOC UB B&W pattern as well as nominal pattern FWCF transients indicate larger ΔCPR results. This is due mostly to the effect of the increased feed flow prior to turbine trip as well as the availability of the turbine bypass to limit the pressurization. The analysis results are summarized in Table A.33.4.1. The TRACG02 analyses predict a [[
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The NRC staff finds that this is consistent with the trends observed in ODYN analyses and trends observed for pre-EPU plants referencing ODYN and TRACG analyses. Therefore, the NRC staff is reasonably assured that the analysis assumptions imposed in reload licensing to calculate the cycle-specific OLMCPR remain adequately conservative for application to MELLLA+ conditions.

Table A.33.2.1: Transient Sensitivity of ΔCPR to Rod Pattern (TRACG02)

AOO	Initial Condition	ΔCPR (29)	ΔCPR (27)	Average ΔΔCPR
TTNB	[[			
FWCF				
				]]

Table A.33.4.1: Transient Sensitivity of ΔCPR to Burn Strategy at MELLLA+ conditions (TRACG02)

AOO	Initial Condition	ΔCPR	ΔΔCPR
TTNB	[[		
FWCF			
			]]

AOO	Initial Condition	ΔCPR	ΔΔCPR
TTNB	[[		
FWCF			
			]]

RAI 34

The NRC staff requested information regarding a potential non-conservatism in the TRACG04 calculated time to boiling transition. For AOO calculations, the analyses demonstrate margin to the SLMCPR, and therefore, boiling transition is precluded. For ATWS overpressure transients, bundles may enter boiling transition. However, the current application is limited to the prediction of the peak pressure of the vessel and is not currently under review for the determination of core coolability. The ATWS overpressure response is most sensitive to the gross core thermal power generation and mass balance. TRACG04 may predict the onset of boiling transition for rods earlier or later in the transient. TRACG uses best estimate methods to predict the neutronic power during the evolution of the transient. However, the determination of boiling transition is based on comparison against the GEXL correlation. The integrated thermal load to the RCS will be insensitive to the limited number of bundles experiencing boiling transition and TRACG will effectively account for the increase in reactor pressure due to the energy deposition from these bundles regardless.

The response contains limited qualification data against transient critical power tests. The results indicate that the time to boiling transition predicted by TRACG04 may have an [[ ]] The time to boiling transition measured during the test according to a criterion of boiling transition based on engineering judgment is on the [[ ]] Since the NRC staff agrees that the rods entering boiling transition will not have a significant impact on either the peak vessel pressure during ATWS overpressure analyses or that the TRACG04 code calculation of that pressure is significantly impacted by the bundles in boiling transition, the NRC staff finds that the transient modeling of boiling transition in TRACG04 is acceptable for that purpose.

The NRC staff, however, notes that the time to boiling transition is an important parameter in evaluating core coolability for LOCA and other ATWS scenarios. Therefore, should GEH seek approval of an application of TRACG04 to ATWS analyses besides overpressure and/or for LOCA, the NRC staff will require that the uncertainty in time to boiling transition be accounted for.

<u>REFERENCES</u>

1. Letter from GEH to USNRC, MFN-06-155, LTR NEDE-32906P, Supplement 3, "Migration to TRACG04 / PANACII from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated May 25, 2006. (ADAMS Package Accession No. ML061500182)
2. Letter from GEH to USNRC, MFN-07-455, “Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated August 15, 2007. (ADAMS Package Accession No. ML072330520)
3. Letter from GEH to USNRC, MFN-07-445, Supplement 1, “Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569), Supplement 1," dated December 20, 2007. (ADAMS Package Accession No. ML073650365)

4. Letter from GEH to USNRC, MFN-08-483, “Response to Request for Additional Information (RAI) 30 RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," dated May 30, 2008. (ADAMS Accession No. ML081550192)
5. Letter from GEH to USNRC, MFN-07-455, Supplement 2, “Response to USNRC Follow-up Question on RAI 33 RE: GEH Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," dated June 6, 2008. (ADAMS Package Accession No. ML081630008)
6. Letter from GEH to USNRC, MFN-08-547, “Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, ‘Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,’ (TAC No. MD2569)," dated June 30, 2008. (ADAMS Accession No. ML081840270)
7. Letter from GEH to USNRC, MFN-08-604, “Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated (TAC No. MD2569)," dated July 30, 2008. (ADAMS Accession No. ML082140580)
8. Final Safety Evaluation of NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains," dated January 17, 2008. (ADAMS Accession No. ML073340214)
9. Letter from GEH to USNRC, MFN-06-327, LTR NEDE-32906P-A, Revision 3, “TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," dated September 25, 2006. (ADAMS Package Accession No. ML062720163)
10. Results Summary Report of USNRC Staff Audit in December 2006 “ESBWR DCD Section 4.3 Nuclear Codes," dated October 15, 2008. (ADAMS Accession No. ML082890853)
11. Addendum 1 to Results Summary Report of USNRC Staff Audit in December 2006 “ESBWR DCD Section 4.3 Nuclear Codes," dated February 2007 (ADAMS Accession No. ML082890853)
12. USNRC Staff Audit Summary, “Summary of Audit for Nuclear Design Codes October/November 2006," dated July 19, 2007. (ADAMS Accession No. ML071700037)
13. Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 229 to Facility Operating License No. DPR-28 Regarding the Vermont Yankee Extended Power Uprate, dated March 2, 2006. (ADAMS Package Accession No. ML060050024)
14. Letter from GEH to USNRC, MFN-06-297, “Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-2 through 4.2-7, 4.3-3, 4.3-4, 4.4-2, 4.4-5, 4.4-6, 4.4-15 through 4.4-17, 4.4-19, 4.4-24, 4.4-27, 4.4-31 through 4.4-34, 4.4-36 through 4.4-38, 4.4-42 through 4.4-50, 4.4-52 through 4.4-56, 4.8-1 through 4.8-16," dated August 23, 2006. (ADAMS Accession No. ML062480252) As supplemented by: MFN-06-297 Supplement 1, November 13, 2006 (ADAMS Accession No. ML070600044); MFN-06-297 Supplement 2, December 21, 2006 (ADAMS Package Accession No. ML070110123); MFN-06-297 Supplement 4, January 26, 2007 (ADAMS Accession No. ML070380108); MFN-06-297 Supplement 5, February 8, 2007 (ADAMS Accession

No. ML070470629); MFN-06-297 Supplement 7, April 10, 2007 (ADAMS Package Accession No. ML071210061); and MFN-06-297 Supplement 8, June 21, 2007 (ADAMS Package Accession No. ML071930214).

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18. Letter from GEH to USNRC, MFN 06-109, LTR NEDE-32176P, Revision 3, “TRACG Model Description," dated April 20, 2006. (ADAMS Accession No. ML061160236)
19. Letter from GEH to USNRC, MFN-08-483, “Response to Request for Additional Information (RAI) 30 RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," dated May 30, 2008. (ADAMS Accession No. ML081550192)
20. Letter from GEH to USNRC, MFN-08-604, “Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated (TAC No. MD2569)," dated July 30, 2008. (ADAMS Accession No. ML082140580)
21. Letter from GEH to USNRC, MFN-06-327, LTR NEDE-32906P-A, Revision 3, “TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," dated September 25, 2006. (ADAMS Package Accession No. ML062720163)

Appendix B: TGBLA06/PANAC11/TRACG04 Code Changes

Table B.1: Code Changes to TGBLA06A

Version	Item	Code Change
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Table B.2: Code Changes to PANAC11A,P

Version	Item	Code Change
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Version	Item	Code Change

Version	Item	Code Change
		]]

Table B.3: Code Changes to TRACG04A

Version	Item	Code Change
[[		

Version	Item	Code Change

Version	Item	Code Change

Version	Item	Code Change

Version	Item	Code Change
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