<u>FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION</u>

<u>LICENSING TOPICAL REPORT NEDE-32906P, SUPPLEMENT 3</u>

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

<u>ATWS OVERPRESSURE TRANSIENTS"</u>

<u>GE HITACHI NUCLEAR ENERGY AMERICAS, LLC</u>

<u>PROJECT NO. 710</u>

ENCLOSURE 2

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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, LLC's (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 the PRIME03, the NRC staff will require an American Society of Mechanical Engineers (ASME) overpressure analyses, ATWS overpressure analyses and AOO analyses be performed using the GSTR-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 rated pump data to be used in 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

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 (Reference 2) or ATWS overpressure (Reference 3) licensing calculations without NRC staff review and approval. (Section 3.19.2.1)

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) would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis. Such changes may be used for AOO or ATWS overpressure licensing calculations without NRC staff review and approval as long as the ratio of the transient change in critical power ratio to the initial critical power ratio (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 3.19.2.2)

15 Code Changes in Numerical Methods Condition

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. (Section 3.19.2.3)

16 Code Changes for Input/Output Condition

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. (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)

18 Statistical Methodology Limitation

The statistical methodology is used to determine specified acceptable fuel design limits (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. (Section 3.19.2.6)

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, LLC (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 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 one 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).

When the Void Reactivity Coefficient Correction Model Condition (Section 4.21) and the Void Reactivity Coefficient Correction Model Basis Condition (Section 4.22) specified in this SE are met, the additional 10 percent margin to the fuel centerline melt and the one percent cladding circumferential plastic strain criteria is no longer required for TRACG04. (Section 3.20.3)

24 Fuel Thermal Conductivity for LHGR Condition

When TRACG04 is used to determine the limiting LHGR for transients, the GSTR-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)

25 10 CFR Part 21 Evaluation of GSTR-M Temperature Calculation Limitation

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

26 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 thermal-mechanical (T-M) methods will be included in the T-M licensing process. This revision will be accomplished through an Amendment to the General Electric Standard Application for Reactor Fuel (GESTAR II) or in a T-M LTR review. If PRIME is approved, future license applications for EPU and MELLLA+ referencing LTR NEDE-32906P, Supplement 3, must utilize these revised T-M methods to determine, or confirm, conservative thermal overpower (TOP) and mechanical overpower (MOP) limits as applicable. (Section 3.20.3)

27 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)

28 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.

If the Void Reactivity Coefficient Correction basis is not updated to include these lattices, and the information provided to meet this condition is insufficient to justify the applicability of the Void Reactivity Coefficient Correction Model basis (i.e., Condition 4.22 is not met for these fuel types), then the plant-specific EPU or MELLLA+ application using TRACG04 must demonstrate an equivalent to 10 percent margin to the fuel centerline melt and the one percent cladding circumferential plastic strain acceptance criteria due to pellet-cladding mechanical interaction for these fuel types for limiting AOO transient events, including equipment out-of-service. (Section 3.20.5)

29 Modified TGBLA06 Condition

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. Should an applicant or licensee reference historical nuclear data generated using TGBLA06AE4 or earlier, the applicant or licensee shall submit justification for its use to the NRC. (Appendix A: RAI 1)

30 Transient 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)

31 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)

32 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)

33 Submittal Requirements Condition

The NRC staff also notes that a generic LTR 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. Nodalization: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

2. Chosen Parameters and Conservative Nature of Input Parameters: 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. Calculated Results: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure. (Reference 3)

34 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.

1 INTRODUCTION

By letter dated May 25, 2006, now GE Hitachi Nuclear Energy Americas LLC, (GEH) submitted LTR NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS 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 EPU or 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, ASME overpressure, 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 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 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, stretch power uprate, 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 SE (e.g., quench front model, hot rod model, relevant models for control rod drop accident (CRDA), 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 10 CFR Part 50, Appendix A, 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 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 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 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 ASME and ATWS overpressure events.

3 TECHNICAL EVALUATION

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 GEH 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 (3D) 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 3D thermal-hydraulic representation along with 3D 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 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. [

]. The Doppler response is based on the transient fuel temperature from TRACG04 and the reactivity coefficients developed from PANAC11. The PANAC11 reactivity coefficients are calculated at 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. [

].

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 DIF3D. 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 RAIs 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

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

[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.

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 GEH 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		]

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 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 GEH 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.

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 GEH 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 with a thermal neutron flux of 1.75×10^{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.

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

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.

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

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

Equation 3-3

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

The Oak Ridge National Laboratory one dimensional depletion code, 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 edit. It is treated separately for each channel component. 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] TRACG uses NEDO-23739 values for the energy released per fission (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 GEH 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 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.

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 Pacific Northwest National Laboratory fuel thermal mechanical code, 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 GSTR-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 reactor coolant system (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 GSTR-M calculations.[3] 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

[3] GSTR-M is used for this purpose currently. It is the understanding of the NRC staff that if PRIME is approved the PRIME method will be used for this purpose.

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 GSTR-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 [

].

[

].

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 GSTR-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-20953-A (Reference 40). However, the NRC staff notes that recently the fuel thermal conductivity model in GSTR-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 defers its detailed review of the PRIME thermal conductivity model to its separate review of PRIME. The NRC staff concludes that the improved model will consistently predict reduced thermal conductivity relative to the previous model based on the GSTR-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 GSTR-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 GSTR-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 GSTR-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. 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 GSTR-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.

The NRC staff has also considered the applicability of a gas gap composition predicted by GSTR-M and its compatibility with the PRIME03 model for thermal conductivity. The fission gas release predicted by GSTR-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 GSTR-M model to predict slightly more conservative estimates of the peak vessel pressure for ATWS overpressure analyses and expects the GSTR-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 GSTR-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 GSTR-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 GSTR-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 GSTR-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 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: $$\frac{ds}{dt} = \frac{3.217 \times 10^{-6}}{s} \exp\left(-\frac{2.007 \times 10^{4}}{T}\right)$$

Equation 3-5

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

TRACG04: $$\frac{ds}{dt} = \frac{3.473 \times 10^{-6}}{s} \exp\left(- \frac{2.010 \times 10^{4}}{T} \right)$$

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 an empirical model based on plant data. 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 TRACG04 remains similar to TRACG02 except that it has been updated to reflect current plant data.

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 Westinghouse 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 rated pump data be used 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 3D 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 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. 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 cooling system (PCCS), RPV, dry well (DW), wet well (WW), 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 passive core cooling (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 WW). 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 isolation condenser (IC) secondary-side pools in the vessel 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 IC is not considered an important parameter because of the limited number of plants with ICSs. 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	IC 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 ICS is not credited in Appendix K LOCA analyses, it is an important system for mitigating the LOFW AOO.

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 (PB) turbine trip (TT) 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 MWth, [ ].

3.18.1 PB TT Tests

TRACG02 and TRACG04 were used to model the first and third EOC2 PB TT 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 With No 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 TT 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 Main Steam Isolation Valve Closure with Flux SCRAM (MSIVF)

The MSIVF event is characterized by closure of the main steam isolation valve (MSIV) with a concurrent failure to SCRAM on MSIV position switch signal. The main steam isolation valve closure (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 average power range monitor 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

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 [ ].

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 GSTR-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

SRP 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 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 (Reference 2) or ATWS overpressure (Reference 3) licensing calculations without NRC staff review and approval. (Section 2.6 of Reference 2)

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. 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. (Section 2.6 of Reference 2)

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. (Section 2.6 of Reference 2)

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. (Section 2.6 of Reference 2)

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. (Section 2.6 of Reference 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. (Section 2.6 of Reference 2) 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. (Section 2.6 of Reference 2)

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. (Section 2.6 of Reference 2)

3.20 Considerations for EPU and MELLLA+

The NEDC-33173P SE (Interim Methods, 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 3D 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 [GEH] 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. The NRC staff notes that when compared to pre-EPU core designs EPU cores generally contain a higher number of higher powered bundles. Therefore, the thermal margin may be degraded for a significant number of bundles.

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. 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 studied in a manner to yield qualification data for phasic models. 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 consideration of those transients that are typically limiting transients in reload licensing analyses. The NRC staff considered those potentially limiting events 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.

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 [

].

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 NRC-approved Amendment 22 to 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. Consistent with 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. [

].

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 one 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 (COLR). 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. 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 UO2 and the limiting GdO2 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, 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 GSTR-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 GSTR-M model conservatively predicts a smaller negative Doppler reactivity worth and, therefore, when TRACG04 is used to determine the limiting LHGR for transients, the GSTR-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 GSTR-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 GSTR-M Fuel Temperature Calculation Limitation**
> Any conclusions drawn from the NRC staff evaluation of the [GEH]'s Part 21 report will be applicable to the GSTR-M T-M assessment of this SE for future license application. [GEH] submitted the T-M Part 21 evaluation, which is currently under NRC staff review. Upon completion of its review, NRC staff will inform [GEH] of its conclusions.
>
> **LHGR and Exposure Qualification Limitation**
> In MFN 06-481, [GEH] 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, the NRC staff will require ASME overpressure analyses, ATWS overpressure analyses and AOO analyses be performed using the GSTR-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)

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 rated pump data to be used in 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

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 (Reference 2) or ATWS overpressure (Reference 3) licensing calculations without NRC staff review and approval. (Section 3.19.2.1)

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) would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis. 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. (Section 3.19.2.2)

4.15 Code Changes in Numerical Methods Condition

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. (Section 3.19.2.3)

4.16 Code Changes for Input/Output Condition

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. (Section 3.19.2.4)

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

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. (Section 3.19.2.6)

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 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).

When the Void Reactivity Coefficient Correction Model Condition (Section 4.21) and the Void Reactivity Coefficient Correction Model Basis Condition (Section 4.22) specified in this SE are met, the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain criteria is no longer required for TRACG04. (Section 3.20.3)

4.24 Fuel Thermal Conductivity for LHGR Condition

When TRACG04 is used to determine the limiting LHGR for transients, the GSTR-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)

4.25 10 CFR Part 21 Evaluation of GSTR-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 GSTR-M is applicable to this SE. (Section 3.20.3)

4.26 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. If PRIME is approved, future license applications for EPU and MELLLA+ referencing LTR NEDE-32906P, Supplement 3, must utilize these revised T-M methods to determine, or confirm, conservative TOP and MOP limits as applicable. (Section 3.20.3)

4.27 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)

4.28 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.

If the Void Reactivity Coefficient Correction basis is not updated to include these lattices, and the information provided to meet this condition is insufficient to justify the applicability of the Void Reactivity Coefficient Correction Model basis (i.e., Condition 4.22 is not met for these fuel types), then the plant-specific EPU or MELLLA+ application using TRACG04 must 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 these fuel types for limiting AOO transient events, including equipment out-of-service. (Section 3.20.5)

4.29 Modified TGBLA06 Condition

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. Should an applicant or licensee reference historical nuclear data generated using TGBLA06AE4 or earlier, the applicant or licensee shall submit justification for its use to the NRC. (Appendix A: RAI 1)

4.30 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)

4.31 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)

4.32 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)

4.33 Submittal Requirements Condition

The NRC staff also notes that a generic LTR 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. Nodalization: Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.

2. Chosen Parameters and Conservative Nature of Input Parameters: 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. Calculated Results: The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure. (Reference 3)

4.34 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

On the basis of its review, the NRC staff has found that the TRACG04 methodology is acceptable for use in licensing evaluations of AOOs, ASME overpressure events, and ATWS overpressure events. 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.

6 REFERENCES

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2. 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)
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5. Final Safety Evaluation of NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated January 17, 2008. (ADAMS Accession No. ML073340214)
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9. 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)
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12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.0.2, "Review of Transient and Accident Analysis Methods," dated December 2000 (ADAMS Accession No. ML053550265)
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