ENCLOSURE 2

MFN 10-140

Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients NEDO-32906 Supplement 3-A Revision 1, April 2010

Non-Proprietary Information

IMPORTANT NOTICE

Enclosure 2 is a non-proprietary version of the Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients, NEDE-32906P Supplement 3-A Revision 1, April 2010 from Enclosure 1, which has the proprietary information removed. Portions that have been removed are indicated by open and closed double brackets as shown here [[ ]].

GE Hitachi Nuclear Energy

NEDO-32906 Supplement 3-A
Revision 1
eDRFSection 0000-0110-0988-R0
Class I
April 2010

Non-Proprietary Information

Licensing Topical Report

Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients

NON-PROPRIETARY INFORMATION NOTICE

This is a non-proprietary version of the document NEDE-32906P Supplement 3-A Revision 1, from which the GEH proprietary information has been removed. Portions of the document that have been removed are identified by white space within double square brackets, as shown here [[ ]].

Regarding the NRC's SE, which is enclosed in NEDE-32906P Supplement 3-A Revision 1, from which the GEH proprietary information has been removed. Portions of the document that have been removed are identified by white space within single square brackets, as shown here [ ].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY

The design, engineering, and other information contained in this document is furnished for the purposes of obtaining NRC approval of the migration to PANAC11 / TRACG04 from PANAC10 / TRACG02 for TRACG AOO and ATWS overpressure transients. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

July 10, 2009

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
P. O. Box 780, M/C A-18
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION OF GE HITACHI NUCLEAR ENERGY AMERICAS, LLC LICENSING TOPICAL REPORT NEDE-32906P, SUPPLEMENT 3, “MIGRATION TO TRACG04/PANAC11 FROM TRACG02/PANAC10 FOR TRACG AOO AND ATWS OVERPRESSURE TRANSIENTS” (TAC NO. MD2569)

Dear Mr. Head:

By letter dated May 25, 2006, GE Hitachi Nuclear Energy Americas, LLC (GEH) submitted licensing topical report (LTR) NEDC-32906P, Supplement 3, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients” to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated January 8, 2009, an NRC draft safety evaluation (SE) regarding our approval of NEDC-32906P, Supplement 3 was provided for your review and comment. By letter dated April 27, 2009, GEH commented on the draft SE. The NRC staff's disposition of GEH’s comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that NEDC-32906P, Supplement 3, is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified and under the limitations delineated in the LTR and in the enclosed final SE. The final SE defines the basis for our acceptance of the LTR.

Our acceptance applies only to material provided in the subject LTR. We do not intend to repeat our review of the acceptable material described in the LTR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GEH publish accepted proprietary and non-proprietary versions of this LTR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the LTR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this LTR, GEH and/or licensees referencing it will be expected to revise the LTR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710

Enclosures:
1. Proprietary version of the Final SE
2. Non-proprietary version of the Final SE

cc w/encl 2 only: See next page

If future changes to the NRC's regulatory requirements affect the acceptability of this LTR, GEH and/or licensees referencing it will be expected to revise the LTR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710
Enclosures:
1. Proprietary version of the Final SE
2. Non-proprietary version of the Final SE

cc w/encl 2 only: See next page

DISTRIBUTION:
SRosenberg (Hardcopy)
NON-PUBLIC
PUBLIC (Public documents only)
PSPB Reading File
RidsNrrDpr
RidsNrrDprPspb
RidsNrrPMMHoncharik
RidsNrrLADBaxley
RidsOgcMailCenter
RidsAcrs_AcnwMailCTR
Peter Yarsky, NRR
Anthony Mendiola, NRR
Stephen Philpott, NRR
RidsNrrDss
RidsNrrDssSnpb
SRosenberg (HardCopy)

ADAMS ACCESSION Numbers:
PUBLIC documents:
Package ML091890758
Cover Letter ML091520088
Final SE (non-proprietary version) ML091751102
Attachment (comment resolution table) (non-proprietary version) ML091820659
Appendix A (non-proprietary version) ML091820494
Appendix B (non-proprietary version) ML091820549
Appendix C ML091490412
Appendix D ML091490416
Appendix E ML091410266
NON-PUBLIC documents:
Final SE ML091400057 (proprietary version)
Attachment (comment resolution table) ML091410281 (proprietary version)
Appendix A ML091410245 (proprietary version)
Appendix B ML091490300 (proprietary version)

NRR-043

OFFICE	PSPB/PM	PSPB/PM	PSPB/LA	SNPB/BC*	PSPB/BC	DPR/DD
NAME	MHoncharik	SPhilpott	DBaxley (GKL for)	AMendiola	SRosenberg	TBlount
DATE	7/8/09	7/2/09	7/6/09	7/2/09	7/10/09	7/10/09

OFFICIAL RECORD COPY

GE-Hitachi Nuclear Energy Americas Project No. 710

cc:

Mr. James F. Harrison
GE-Hitachi Nuclear Energy Americas LLC
Vice President - Fuel Licensing
P.O. Box 780, M/C J-70
Wilmington, NC 28401-0780
james.harrison@ge.com

Ms. Patricia L. Campbell
Vice President, Washington Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
1299 Pennsylvania Avenue, NW
9th Floor
Washington, DC 20004
patricia.campbell@ge.com

Mr. Andrew A. Lingenfelter
Vice President, Fuel Engineering
Global Nuclear Fuel–Americas, LLC
P.O. Box 780, M/C J-70
Wilmington, NC 28401-0780
Andy.Lingenfelter@gnf.com

Mr. Richard E. Kingston
GE-Hitachi Nuclear Energy Americas LLC
Vice President, ESBWR Licensing
PO Box 780, M/C A-55
Wilmington, NC 28401-0780
rick.kingston@ge.com

Mr. James A Ross
Vice President, Nuclear Licensing
GE-Hitachi Nuclear Energy Americas LLC
1299 Pennsylvania Ave, NW
9th Floor
Washington, DC 20004
james2.ross@ge.com

Location	Comment	NRC Resolution
Generic	The SE uses the terminology AOOs and ATWS overpressure. Please make sure that it is clear that the NRC approval includes the American Society of Mechanical Engineers (ASME) overpressure event (MSIV closure with flux scram), which is a variation of an AOO with an additional failure and is clearly within the scope of applicability of the methodology.	Comment accepted. <u>Staff comment:</u> The ASME overpressure transient evaluation is much akin to pressurization AOO evaluations. The ASME overpressure event is a variation of an AOO with an additional failure of the MSIV position SCRAM and conservative input assumptions. Therefore, the staff finds that its conclusions regarding the acceptability of TRACG04 for AOOs and ATWS overpressure clearly form a valid basis for the staff acceptance of the use of TRACG04 for ASME overpressure calculations. <u>SE Revisions:</u> Section 1.1 "Similarly the NRC staff review is limited to the application of the methodology to AOO and ATWS overpressure transient analyses" Revised as: "Similarly the NRC staff review is limited to the application of the methodology to AOO and, American Society of Mechanical Engineers (ASME) overpressure, and ATWS overpressure transient analyses"

		Section 2 "GEH uses the TRACG code to calculate the peak vessel pressure to ensure vessel integrity during ATWS pressurization events" Revised as: "GEH uses the TRACG code to calculate the peak vessel pressure to ensure vessel integrity during ASME and ATWS overpressure events" Section 5 Added following paragraph at beginning of Section 5: "On the basis of its review the staff has found that the TRACG04 methodology is acceptable for use in licensing evaluations of AOOs, ASME overpressure events, and ATWS overpressure events." Appendix E Add to acronym list: ASME – American Society of Mechanical Engineers
Section 3.3.1.3, Lines 31-34	Suggest simplifying complex sentence. "TRACG04 in transient mode will calculate the transient fuel temperature, but the Doppler coefficient is based on the PANAC11-predicted Doppler response to temperature, which is based on the PANAC11-predicted steady-state temperature." to	Comment partially accepted. Section revised as: 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 Doppler response is based on the transient fuel temperature from TRACG04 using the coefficients developed from PANAC11.	
Section 3.3.1.5 Line 28	"The infinite eigenvalue used for………" Should be The infinite lattice eigenvalue used for………	Comment incorporated.
Section 3.4.1.1 Lines 29-32	May want to clarify. "For power calculations for AOOs, the contribution from the decay heat is conservatively increased, therefore, increasing the integrated thermal energy deposited into the reactor coolant system (RCS) for AOO evaluations terminated by a SCRAM." The use of the May Witt accomplishes this, but there is no standard adjustment that does this.	Comment not applicable in final SE. Sentence has been deleted.
Section 3.4.1.6 Lines 20 - 21	The heat from metal-water reaction is included in the energy balance for the core. It is not include in the edit for the total power, which is limited to fission and decay heat power. The edit of energy generation from metal-water reaction is included for each channel component.	Comment accepted. First sentence revised as: 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.
Section 3.4.1.7 Footnote 2	The energy release by fission is calculated by TGBLA when the kinetics solver is activated. TRACG alternatively uses NEDO-23739 values for the energy released per fission when the kinetics solver is disabled (see response to RAI 22). The 4 parent chains are always determined from NEDO-23739.	Comment incorporated. Section revised as: TRACG uses NEDO-23739 values for the energy released per fission (see response to RAI 22)
Section 3.7.4 Line 8	Typo Berenson	Comment incorporated.
Section 3.9.1 Lines 20-22	Potentially Confusing. The models for flow regime transitions in TRACG02 had only been qualified at high pressure. GEH	Comment accepted. First sentence has been deleted.

	qualified TRACG against low pressure data to extend the applicability of TRACG to LOCA applications. The TRACG qualification report NEDE-32177 includes high and low pressure simulations.	
Section 3.10.5.1 Lines 8-10	The dynamic gas gap conductance inputs for both TRACG02 and TRACG04 are based on upstream GESTR-M calculations. Suggest also mentioning PRIME, as we will be transitioning to PRIME inputs.	Comment accepted. Section revised to include a footnote at end of sentence. GESTR-M is used for this purpose currently. It is the understanding of the NRC staff that if PRIME is approved that the PRIME method will be used for this purpose.
Section 3.10.5.2 Pg 40 Lines 8-11	According to Reference 22, [ ] Ref 22, pg. 1-44 says that the correlation parameters consider this, but not that PANAC11 solves it. P11 correlates the fuel temperature to the nodal power	Comment accepted. Section revised as: According to Reference 22, [ ]
Section 3.10.5.2 Pg 40 Lines 25-34	The NRC staff found that the new fuel conductivity model in TRACG04 predicts lower fuel thermal conductivities with increasing fuel exposure and agrees to a large extent with the FRAPCON3 model in terms of variation with temperature and exposure. However, the NRC staff finds that the improved model appears to misrepresent the impact of gadolinia on fuel thermal conductivity at high exposure. The NRC staff cannot conclude that the improved thermal conductivity model represents a best estimate of the fuel thermal conductivity over the full range of gadolinia loadings and exposures.	Comment accepted. "However, the NRC staff finds that the improved model appears to misrepresent the impact of gadolinia on fuel thermal conductivity at high exposure. The NRC staff cannot conclude that the improved thermal conductivity model represents a best estimate of the fuel thermal conductivity over the full range of gadolinia loadings and exposures." Replaced with:

	These conclusions from the PRIME review may reflect preliminary conclusions given that the PRIME review is not completed. This seems like evaluation verbiage that would be better placed in the PRIME SE. Especially with the PRIME review not yet complete. Other limitations and conditions direct us to use PRIME when it is approved.	"However, the staff defers its detailed review of the PRIME thermal conductivity model to its separate review of PRIME"
Section 3.10.5.3	Similar statements from 10.5.3. Same comment. The NRC staff similarly reviewed the PRIME03-based thermal conductivity model and found that gadolinia bearing pellet temperatures will be under predicted at high exposure. Furthermore, while the NRC staff does not find that the improved model accurately captures the impact of gadolinia on fuel thermal conductivity, particularly for high exposure, the NRC staff notes (based on its preliminary review)……….	Comment accepted. Both sentences deleted.
3.12.3 Pg 46 Lines 9-12 Also, Lines 39-40	While the oxide layer thickness affects cladding heat transfer characteristics, the NRC staff notes that the initial oxide layer thickness in TRACG04 is either directly input for bounding calculations or is calculated according to the same empirical model based on plant data as in TRACG02. The NRC staff notes that the option to predict the initial clad oxide thickness in TRACG02 and TRACG04 remains unchanged. The TRACG04 model is similar to TRACG02 but different. It has been updated to reflect current data.	Comment accepted. First sentence revised as: 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. Second sentence revised as: 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.

3.13.3 Lines 23-35	The second set of curves is based on the Bingham pump curves and is consistent with the curves used as the default in RETRAN02 and RELAP/5-MOD1. The second set of curves is based on Westinghouse not Bingham.	Comment incorporated.
3.13.4 Lines 32-33	The NRC staff will require that plant-specific pump data be verified and input for transient calculations. It should be clear that the pump data is the rated pump information not plant specific homologous curves. Also, the rated pump information is normally provided by the utility for a specific project or reload. GEH and GNF must assume that the utility has provided the data according to their quality procedures. We obviously consistency check the information based on our experience, but we don't literally verify the utility data.	Comment accepted. Section revised as: The NRC staff will require that plant-specific rated pump data be used for transient calculations
3.15.2 Lines 7-8	These TRACG04 cases were run using program library Version 40. The version is not relevant to the approval of TRACG04 for AOOs and ATWS overpressure. Also, the cases were repeated several times with newer versions and the final validated code.	Comment accepted. Sentence deleted.
3.15.5.3 Lines 25-26	While the ICS is not credited in Appendix K LOCA analyses, it is an important system for mitigating AOOs. The LOFW is the only BWR2 AOO in which the ICS is important. The actuation logic is such that the transient duration for other AOOs is not be long enough to see actuation.	Comment accepted. Section revised as: While ICS is not credited in Appendix K LOCA analyses, it is an important system for mitigating the loss of feedwater AOO.
3.19.2.1 Line 3	Shouldn't (Reference 3) go up by ATWS overpressure instead of being at the end of the sentence. Also, Reference 2 should be cited after the AOO.	Comment incorporated
3.19.2.1-X	Generic question. Reference 3 is cited in numerous	Comment accepted.

	locations at the end of paragraphs. We do not see a specific relationship back to Reference 3 in these instances.	Staff Comment: Reference 3 is a typographical error in these instances. The actual reference is Reference 2 (NEDE-32906P-A Rev. 3). In particular the staff cites Section 2.6 "Review Requirements for Updates." Revised as: "Reference 3" has been replaced with "Section 2.6 of Reference 2."
3.19.2.2 Lines 19-21	Updates to the TRACG nuclear methods to ensure compatibility with the NRC-approved PANACEA family of steady-state nuclear methods (e.g., PANAC11) **would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes** may be used for AOO or ATWS Suggest adding words consistent with 50.59 as was done in the ODYSY-1D SE.	Comment incorporated.
3.19.2.3 Lines 30-33	Changes in the numerical methods to improve code **convergence would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes** may be used in AOO and ATWS overpressure licensing calculations without NRC staff review and approval.	Comment incorporated.
3.19.2.4	Features that support effective code input/output **would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes may** be added without NRC staff review and approval. (Reference 3)	Comment incorporated.
3.19.2.6	As a result, changes to the statistical methodology	Comment incorporated.

	directly affect the results of safety **analyses and constitute a departure from a method of evaluation used in establishing the design bases or in the safety analysis**.	
3.20.1 Lines 47 - 48	While the power distribution in the core is flatter for EPU, the SLMCPR continues to be based on 0.1% of the number of rods in the core experiencing BT.	Comment accepted. Section revised as: 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 staff notes that when compared to pre-EPU core designs that EPU cores generally contain a higher number of higher powered bundles. Therefore the thermal margin may be degraded for a significant number of bundles.
3.20.1 Page 62 Line 1	The NRC staff notes that interfacial phenomena have not been extensively studied……. This is not a precise statement. The two-phase industry has studied phase interface phenomena extensively. It is true that the two-phase data is largely limited to engineering level measurements that do not provide information that can uniquely qualify phasic models.	Comment accepted. Section revised as: 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
3.20.1 Page 62 Line 35	The TRACG04 analysis initialization, however, is based on steady-state power distribution calculations performed using PANAC11. Editorial suggestion.	Comment incorporated.
3.20.3 Page 67	Clarification. While the TOP limit does ensure the fuel melt limit is met, the TOP does not correspond to the limit. The TOP limit is a screening criterion that is generically established on a fuel product basis to provide a conservative method to evaluate transient response. The same thing is true of the MOP limit. It does not correspond to the strain limit but to a conservative	Comment accepted. Staff comment: The staff agrees that the TOP and MOP limits are conservative screening criteria that are used in lieu of detailed evaluations of the associated figures of merit against the applicable regulatory criteria.

	screening criterion to assure the limit is met.	Section revised as: Original: “As stated in Section 3.2.6 of Reference 5, the fuel T-M design criteria require, in part, that:” Revised: “Consistent with Section 3.2.6 of Reference 5, the fuel T-M design criteria require, in part, that:” Original: [ ] Revised as: [ ] Original: [ ] Revised as: [ ]
3.20.3	Grammar correction.	Comment incorporated

Page 69 Line 10		
4.6 Lines 30-33	Until the NRC staff approves PRIME03 or approves the GESTR-M benchmarks per Appendix F of Reference 5, the NRC staff will require ATWS overpressure analyses and AOO analyses be performed using the GESTR-M model. 3 comments. (1) Should and be or, (2) clarify where the subject benchmarks are coming from, and (3) the performance of the GESTR-M benchmarks per the Reference 5 Appendix F are not related to the GESTR-M parameters that are used in TRACG04.	Comment accepted. Section revised as: Until the NRC staff approves PRIME03, the NRC staff will require ASME overpressure analyses, ATWS overpressure analyses and AOO analyses be performed using the GESTR-M model
4.8	See Comment 3.13.4 Lines 32-33 above.	Comment accepted. Section revised as: Licensing calculations require plant-specific rated pump data to be used in the TRACG model
4.13	Suggest using the same wording as 3.19.2.1.	Comment incorporated.
4.14	Suggest using the same wording as 3.19.2.2.	Comment incorporated.
4.15	Suggest using the same wording as 3.19.2.3.	Comment incorporated.
4.16	Suggest using the same wording as 3.19.2.4.	Comment incorporated.
4.18	Suggest using the same wording as 3.19.2.6.	Comment incorporated.
4.21	This statement from 3.20.3 clearly states the condition for releasing the 10% margin requirement or penalty when the conditions of 4.21 and 4.22 are met. The conditions specified in Section 4.21 and Section 4.22 of this SE complement Transient LHGR Limitation 3. Therefore, a 10 percent penalty is not required for TRACG04 methods when the conditions specified in Section 4.21 and Section 4.22 of this SE are met. Also, this is a quote from the Methods LTR SE, "If the void history bias is incorporated into the coupled neutronic and transient code set, then the additional 10 percent margin to the fuel centerline melt and the 1	Comment accepted. Added Section below Section 4.22 and revised numbering of conditions and limitations in Section 4, added a corresponding section in the executive summary, and renumbered the conditions. The section added: 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

	percent cladding strain is no longer required." It is suggested that the release, including the strain term, be added to the limitations and conditions for clarity.	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)
	Related to the previous comment. The Methods SE has this statement concluding Limitation 10, "However, if GE does not adequately address the methodology deficiencies identified in LTR NEDC-33173P in the review of Supplement 3 of NEDE-32906P, the additional margins as described in this SE apply as appropriate." For clarity, Is it not appropriate to reference limitation 10 of the Methods SE and state in the TRACG04 SE, that the subject deficiencies have been met and no margins as described in the SE for NEDC-33173 apply. The limitation should be that the 10% margin should be applied if the historical void is not used. This is an input option.	Comment not incorporated. Staff comment: Conditions 4.21 and 4.22 in Section 4 of the draft SE specifically require the historical void to be used. For additional clarification on mixed-core analyses see NRC resolution for comments on Section 4.30
4.23, 24, and 26	Why is it considered necessary to repeat these limitations from Reference 5? Limitation 4.2 reminds us generally that the SEs for Methods LTR and the MELLLA+ LTR must be met.	Comment accepted. Sections 4.23, 4.24, and 4.26 deleted from Section 4 of the SE. Conditions 23, 24, and 26 deleted from executive summary. Other sections and conditions are renumbered accordingly.

4.28	Same as above. 4.25 requires the use of the PRIME models when approved. 4.28 is different than 4.25 in that it applies a condition that, if necessary, should be specified in the PRIME SE. The specification of when PRIME must be used should be in the PRIME SE.	Comment not incorporated. Staff comment: The intent of the condition is to specify that future EPU and MELLLA+ applications referencing TRACG04 methods that utilize TOP/MOP limits in lieu of detailed transient LHGR analyses that directly compare the figure of merit against the applicable specified acceptable fuel design limits must utilize TOP/MOP limits generated using the revised T-M methods or conservative limits. Section 4.28 and corresponding Condition 28 in the executive summary revised as: 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.
4.30	This is a pure TGBLA methods limitation stated in the SE for the Methods LTR. There is no need to repeat it here.	Comment not incorporated. Staff comment: The condition is maintained in the TRACG04 SE as the staff's consideration of the information supplied to support the mixed core application must include the use of MCNP/TGBLA calculations within the framework of the TRACG04 transient analysis methodology to establish the void reactivity coefficient biases and uncertainties. In short, the staff would review this information to ensure that the Void Reactivity Coefficient Correction Model Condition (Section 4.21) and the Void Reactivity Coefficient Correction Model Basis Condition (Section 4.22) are met when legacy mixed-vendor fuel is loaded in the core. The staff understands that implementing the Void Reactivity

		Coefficient Correction Model with lattice bases to reflect the other fuel designs may not be justified based on the information provided to satisfy this condition. However, in these cases licensees may elect to account for the impact of the void history bias for plant-specific EPU and MELLLA+ applications using TRACG04 by demonstrating 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 legacy fuel for limiting AOO transient events, including equipment out-of-service. Section revised as: Original: 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. Revised: 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.
4.31	**Discussion** As of December 2008, the GEH standard production analysis TGBLA06 has been updated to TGBLA06AE6. The first suggestion is to modify the limitation such that E5 or later versions are defined to be acceptable. The specific version used for a particular application could span back to TGBLA06AE4 because of the lead time for the design work and schedule delays. The core design work on some EPU and MELLLA+ projects was performed with a TGBLA06AE4 basis. In these cases, while the power uprate SAR is based on an equilibrium core with a TGBLA06AE4 basis, the lattices and core design activities for the cycle implementing the EPU or MELLLA+ change will have a lattice basis of TGBLA06AE5 or later. The core and safety analyses performed with TRACG04 for the SRLR will have a TGBLA06AE5 or later basis. In the time frame of implementation of EPU or MELLLA+ projects, in NRC review or soon to be in	Comment partially accepted. Staff comment: Use of TGBLA06AE4 for generating lattice physics data for EPU applications has been justified on a plant specific basis for GE14 fuel (see References below). Section revised as: 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 their use to the NRC. (Appendix A: RAI 1) References 1. Letter from O'Connor, T. J. (NSPM) to US NRC,

	review, that implement TRACG04 it is highly unlikely that any TGBLA06AE4 based bundles will remain. However, there are remote situations where a plant may need to install an older bundle which may have a TGBLA06AE4 basis. **Proposed** The application of TRACG04/PANAC11 is restricted from application to EPU or MELLLA+ plants until TGBLA06 is updated to TGBLA06AE5 or later in the GEH standard production analysis techniques. TGBLA06AE5 or later should be used for all new lattice physics constant generation. For core design activities in support of EPU or MELLLA+ submittals initiated after the date of this SE, TGBLA06AE5 or later should be used. TGBLA06AE5 or later should be used for all cycles implementing EPU or MELLLA+ based on application of TRACG04/PANAC11 for transient or ATWS overpressure analysis. (Appendix A. RAI 1)	"Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Performance & Code Review Branch Request for Additional Information (RAI) dated January 16, 2009 (TAC No. MD9990)," dated March 19, 2009. 2. Letter from GEH to USNRC, MFN 06-297 Supplement 1, "Supplemental Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Number 4.3-3," November 8, 2006 (ADAMS Accession No. ML063400067).
Executive Summary	Section 4 Comments also apply to the Executive Summary.	See previous NRC resolutions
References	Added MFN numbers for some GEH to NRC references where they were missing.	Comment incorporated.
Appendix A RAI-1 Lines 37-39	Same comment as 4.31 above	Comment accepted. Following text added to end of Section RAI 1: On a cycle specific basis the use of TGBLA06AE4 nuclear parameters for legacy GEH/GNF fuel may be justified. This justification may be provided to the NRC on an application specific basis to demonstrate for the fuel design that the nodal parameters are negligibly impacted by the code differences between TGBLA06AE4 and TGBLA06AE5. GEH

		has previously provided similar justification to the NRC for GE14 fuel lattices in Reference 9. It is expected that licensees or applicants that reference historical TGBLA06AE4 calculations likely utilize GE14 fuel and will reference those calculations previously reviewed by the NRC. A new reference 9 is added to the references section of Appendix A 9. Letter from GEH to USNRC, MFN 06-297 Supplement 1, "Supplemental Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Number 4.3-3," November 8, 2006 (ADAMS Accession No. ML063400067)
Appendix A RAI-16 Pg A-10 Lines 27-30	The NRC staff finds that at high exposure, the TRACG04 model does not predict any influence on thermal conductivity by the gadolinia, whereas the FRAPCON3 model consistently predicts degradation in thermal conductivity with increasing gadolinia concentration. The figures in MFN 08-053 show the gadolinia dependency.	Comment accepted. Section revised as: The NRC staff finds that at very high exposure, the TRACG04 model predicts only a minor influence on thermal conductivity by the gadolinia, whereas the FRAPCON3 model consistently predicts a much greater degradation in thermal conductivity with increasing gadolina concentration.
Appendix A RAI-16 Pg A-10 Lines 35-37	Therefore, the NRC staff does not accept the conclusion that gadolinia depletion under irradiation results in a negligible impact on fuel thermal conductivity at the end of life for the fuel,….. We did not make such a concluding statement.	Comment accepted. Section revised as: Therefore, the NRC staff has deferred the review of the PRIME thermal conductivity model to the specific review of PRIME and herein makes no statements regarding the veracity of the model for gadolinia bearing fuel near the end of life,…
Appendix A RAI-16 Pg A-10	(2) there is evidence that the new fuel thermal conductivity model remains non-conservative in the prediction of pellet temperature for gadolinia loaded	Comment accepted. Section revised as: (2) when compared to the staff's FRAPCON model, the

Lines 42-43	fuel pins. The evidence is only that FRAPCON shows a steeper reduction in thermal conductivity as gadolinia increases.	PRIME thermal conductivity model predicts a lesser degree of degradation with increasing gadolinia concentration.
Appendix A RAI-22 Pg A-16 Lines 1-2	For evaluations where the kinetics solver is disabled, the decay heat fission energy release values are based on historical values reported in GEH LTR NEDO-23729. The decay heat values are always based on values from NEDO-23729. The reason for this was provided in the RAI response.	Comment accepted. Section revised as: For evaluations the decay heat fission energy release values are based on historical values reported in GEH LTR NEDO-23729.
Appendix A RAI-28 Pg A-18 Line 13	(Bingham) The TRACG04 default homologous pump curves are for a large Westinghouse pump manufactured in Cheswick, PA. The Bingham pump is only slightly different.	Comment accepted. "(Bingham)" is deleted.
Appendix A RAI-29 Pg A-18 Line 39	The NRC staff evaluation of the applicability of TRACG04 to EPU and MELLLA+ mixed core analysis was reviewed separately and is documented in Section 4.20.5 of the subject LTR. May want to clarify the "subject LTR" more specifically.	Comment accepted. Section revised as: The NRC staff evaluation of the applicability of TRACG04 to EPU and MELLLA+ mixed core analysis was reviewed separately and is documented in Section 3.20.5 of this SE.
Appendix A RAI-32b	PANCEA should be PANACEA	Comment incorporated.
Appendix A RAI-32c Pg A-26 Line 39-43	………, the NRC staff is concerned that the inter bundle nuclear coupling may amplify the impact of errors in the predicted nodal reactivity feedback characteristics. The bundles are coupled by internodal neutron leakage. Potentially increased errors in neighboring bundle void reactivity feedback will have a direct effect on the efficacy of the code to accurately determine the limiting bundle transient response. This concern leaves the reader with no closure with respect to the importance of the concern. This	Comment accepted. Section revised as: …, the NRC staff was concerned that the inter bundle nuclear coupling may amplify the impact of errors in the predicted nodal reactivity feedback characteristics at EPU or MELLLA+ conditions. The bundles are coupled by internodal neutron leakage. Potentially increased errors in neighboring bundle void reactivity feedback will have a direct effect on the efficacy of the code to accurately determine the limiting bundle transient response. Therefore, the staff requested in

	discussion is tangential to the issue of the RAI regarding TRACG initialization. The concern by the NRC is new and should not be written into the SE. BWR operation is strongly coupled between the nuclear and thermal-hydraulics field. If the staff concern is retained, the reviewer should document its resolution by consideration that steady-state methods qualification (TIPs, gamma scans, eigenvalues) indicate that the coupled nuclear/thermal-hydraulics solution in GEH/GNF methods is satisfactory.	RAI 32 that GEH specifically evaluate the impact of the void fraction mismatch at MELLLA+ conditions.
Appendix A RAI-32c Pg A-29 Line 14	Therefore, the NRC staff maintains that a threshold of significance of 0.005 remains appropriate when evaluating a potential bias. The cited 0.005 is the steady state level of significance. The transient basis has always been 0.01. The [ ] change in D/I for CPR is what was agreed to by the NRC staff in NEDE-32906P-A as being the appropriate level of significance for triggering additional NRC review.	Comment accepted. Sentence revised as: Therefore, the NRC staff considered a threshold of significance in its review of the current RAI response of 0.005. Values greater than 0.005 approach the one sigma deviation difference considered significant in Section 2.6.1 of NEDE-32906P-A.
Appendix A RAI-32c Pg A-30 Line 40-41	The NRC staff's conclusions here are predicated on pressurization transients being the limiting transients in reload licensing analyses. While pressurization transients are often the most limiting, it is not always the case. The conclusions based on the example where the pressurization transient is limiting also apply for other AOO transients.	Comment accepted. Section revised as: Paragraph deleted. Following added above the former second to last paragraph: The staff considered the relevancy of the sensitivity studies to the broad range of anticipated operational occurrences that may occur for operating BWR plants. Licensees analyze a host of transients each operating cycle to determine thermal operating limits. The potentially limiting AOO events are determined and analyzed. The potentially limiting transient events analyzed on a cycle specific basis include: generator load rejection or turbine trip without bypass, loss of feedwater

		heat or inadvertent high pressure coolant injection (HPCI), control rod withdrawal error, feedwater controller failure to maximum demand, and pressure regulator failure (for BWR/6 plants). For the operating fleet of BWR plants these events are generally the limiting events. Of these the generator load rejection without bypass, turbine trip without bypass, feedwater controller failure, and pressure regulator failure events are pressurization transients. The sensitivity studies provided in the RAI response provide details of the sensitivity of the transient response to pressurization transients. The staff expects that the sensitivity demonstrated for the pressurization transients would bound that for the other potentially limiting events: control rod withdrawal error, loss of feedwater heat, and inadvertent HPCI. The control rod withdrawal error is a postulated AOO whereby the operator erroneously, continuously withdraws the highest worth control blade above 75 percent of power. The event is terminated by the rod block monitor (RBM). During the transient the local reactor power increases due to the reactivity insertion from the withdrawal. The increased local power is sensed by the LPRMs. The RBM will prohibit further withdrawal of the rod as the power increase because increasingly severe. The negative reactivity feedback from any void formation is modeled in TRACG04; however, the bundle power history is a much stronger function of the control blade reactivity and withdrawal rate. Therefore, the staff finds that the CPR sensitivity to any void mismatch for a control rod withdrawal error would be bound by the pressurization transient results.

		The loss of feedwater heat and the inadvertent HPCI AOOs are similar. These AOOs are postulated events where the core flow inlet subcooling is increased due to cooler water injection to the vessel. These events tend to be slowly evolving transients where the core approaches a new steady-state condition where the power increases to compensate for positive reactivity insertion. Generally, the core will approach a condition where the adjoint-weighted core average void fraction remains the essentially the same. Therefore, the staff does not expect the dynamic response to be sensitive to mild variation in the local void fraction due to void-model differences. On this basis, the staff finds that the CPR sensitivity calculated for the pressurization transients would bound any CPR sensitivity for the loss of feedwater heat or inadvertent HPCI AOOs. Revised Section 3.20.1 Page 65 Lines 20-24 "The NRC staff's conclusions here are predicated on pressurization transients being the limiting transients in reload licensing analyses. This is true for the operating fleet of BWR/2-6 reactors. Therefore, the NRC staff's findings in this matter may not be applicable to other BWR designs where pressurization transients are not the limiting transients." Revised as: "The NRC staff's conclusions here are predicated on consideration of those transients that are typically limiting transients in reload licensing analyses. The 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." Appendix E Added to acronym list HPCI – High Pressure Coolant Injection
Appendix A References	Added MFN numbers for some GEH to NRC references where they were missing. Also, Reference 9 and 21 are the same.	Comment incorporated
Appendix B	If we are including the list why not include all through 57. Put an A,P after TRACG04 or eliminate the A.	Comment partially accepted. Section revised as: TRACG04A,P Staff comment: The staff considered the audit of GEH quality assurance procedures related to TRACG04. The staff did not base its review of the quality assurance (QA) program on subsequent information provided, audited, or inspected by the staff. The Appendix is consistent with the scope of the audit referenced in the SE. The Appendix is included to demonstrate that code changes have not impacted the analysis methodology while the method was maintained under the GEH QA program.
Appendix E	INEL should be INL	Comment partially accepted. Section revised as: INEL (currently INL) – Idaho National Engineering Laboratory (currently Idaho National Laboratory)

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDE-32906P, SUPPLEMENT 3

“MIGRATION TO TRACG04/PANAC11 FROM TRACG02/PANAC10 FOR TRACG AOO AND ATWS OVERPRESSURE TRANSIENTS"

GE HITACHI NUCLEAR ENERGY AMERICAS, LLC

PROJECT NO. 710

ENCLOSURE 2

Table of Contents

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 <u>TECHNICAL EVALUATION</u>

3.1 Overview of the TRACG Methodology

TRACG is a transient analysis code derived from the original TRAC family of codes. TRACG is a coupled thermal-hydraulic neutron kinetics analysis system developed by 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 x 10^{14} n/sq-cm/sec. For times less than 10^4 seconds, the G-factor can be expressed as shown in Equation 3-2. For longer times, linear interpolation between tabular values is used.

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

Equation 3-2

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

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

3.4.1.3 Actinide Contribution

The heat source from actinides in the fuel is divided into two subsets. The first subset is considered the major actinides and the second subset is considered miscellaneous actinides. The first set includes the heat source from uranium-239 and neptunium-239. The second set includes a host of actinides, particularly: curium-242, neptunium-238, uranium-237, plutonium-237 and americium-241.

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

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

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

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

Equation 3-3

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

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

Equation 3-5

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

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

Equation 3-6

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

3.12.3 Conclusions

While the oxide layer thickness affects cladding heat transfer characteristics, the NRC staff notes that the initial oxide layer thickness in TRACG04 is either directly input for bounding calculations or is calculated according to 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. [

].

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. [

].

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. [
>
>].

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

1. Letter from GEH to USNRC, MFN-06-155, LTR NEDE-32906P, Supplement 3, "Migration to TRACG04 / PANACII from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated May 25, 2006. (ADAMS Package Accession No. ML061500182)
2. Letter from GEH to USNRC, MFN-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)
3. Letter from GEH to USNRC, MFN 03-148, LTR NEDE-32906P, Supplement 1-A, "TRACG for Anticipated Transients Without SCRAM Overpressure Analysis," dated November 26, 2003. (ADAMS Package Accession No. ML033381073)
4. Letter from GEH to USNRC, MFN 06-079, LTR NEDE-32906P, Supplement 2-A, "TRACG for Anticipated Operational Occurrences Transient Analysis," dated March 16, 2006. (ADAMS Package Accession No. ML060800312)
5. Final Safety Evaluation of NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated January 17, 2008. (ADAMS Accession No. ML073340214)
6. Letter from GEH to USNRC, MFN-07-455, "Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated August 15, 2007. (ADAMS Package Accession No. ML072330520)
7. Letter from GEH to USNRC, MFN-07-445, Supplement 1, "Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569), Supplement 1," dated December 20, 2007. (ADAMS Package Accession No. ML073650365)
8. Letter from GEH to USNRC, MFN-08-483, "Response to Request for Additional Information (RAI) 30 RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," dated May 30, 2008. (ADAMS Accession No. ML081550192)
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)
10. Letter from GEH to USNRC, MFN-08-547, "Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, 'Migration to TRACG04/PANAC11

from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,' (TAC No. MD2569)," dated June 30, 2008. (ADAMS Accession No. ML081840270)
11. Letter from GEH to USNRC, MFN-08-604, "Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated (TAC No. MD2569)," dated July 30, 2008. (ADAMS Accession No. ML082140580)
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)
13. Draft Regulatory Guide DG-1096, "Transient and Accident Analysis Methods," dated December 2000. (ADAMS Accession No. ML003770849)
14. NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," dated, December 1989. (ADAMS Package Accession No. ML030380503)
15. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980 (ADAMS Accession No. ML051400209).
16. Regulatory Guide 1.203, "Transient and Accident Analysis Methods," dated December 2005. (ADAMS Accession No. ML053500170)
17. Final Safety Evaluation of NEDE-32906P, Revision 2, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis," dated August 29, 2006. (ADAMS Accession No. ML062210315)
18. Safety Evaluation of NEDC-33083P "TRACG Application for the ESBWR," dated October 28, 2004. (ADAMS Package Accession No. ML043000285)
19. Letter from GEH to USNRC, MFN 04-131, LTR NEDE-33083P, Supplement 1, "TRACG Application for ESBWR Stability Analysis," dated December 9, 2004. (ADAMS Accession No. ML050060160)
20. Safety Evaluation Report by the Office of Nuclear Reactor Regulation for Licensing Topical Report NEDE-33083, Supplement 1 "TRACG Application for ESBWR Stability Analysis," dated March 28, 2006. (ADAMS Package Accession No. ML072270138)
21. USNRC to GEH (C. P. Kipp), "NRC Inspection Report 99900003/95-01," dated March 5, 1996. (ADAMS Accession No. ML070400521) Package ML070400485
22. Letter from GEH to USNRC, MFN 07-223, LTR NEDC-33239P, Revision 2, "GE14 for ESBWR Nuclear Design Report," dated April 2007. (ADAMS Accession No. ML072841058)
23. Letter from GEH to USNRC, MFN-05-029, "Responses to RAIs – Methods Interim Process (TAC No. MC5780)." (GE Responses to RAIs 5,25,26,27, and 29), dated April 8, 2005. (ADAMS Accession No. ML051050023)
24. Letter from GEH to USNRC, MFN-07-347, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – RAI Numbers 21.6-65 and 21.6-85," dated June 21, 2007 (ADAMS Package Accession No. ML071930527)
25. USNRC Staff Audit Results Summary Report "ESBWR DCD Chapters 6 and 21 TRACG for LOCA," dated January 2007. (ADAMS Accession No. ML082900619)
26. Letter from GEH to USNRC, MFN 06-109, LTR NEDE-32176P, Revision 3, "TRACG Model Description," dated April 20, 2006. (ADAMS Accession No. ML061160236)
27. Results Summary Report of USNRC Staff Audit in December 2006 "ESBWR DCD Section 4.3 Nuclear Codes," dated October 15, 2008. (ADAMS Accession No. ML082890853)
28. Addendum 1 to Results Summary Report of USNRC Staff Audit in December 2006 "ESBWR DCD Section 4.3 Nuclear Codes," dated February 2007 (ADAMS Accession No. ML082890853)

29. USNRC Staff Audit Summary, "Summary of Audit for Nuclear Design Codes October/November 2006," dated July 19, 2007. (ADAMS Accession No. ML071700037)
30. USNRC Staff Audit Summary, "Summary of Exit Meeting Held on December 15, 2006, to Discuss NRC staff's Audit of TRACG Loss-of-Coolant-Accident Analyses," dated January 4, 2007. (ADAMS Accession No. ML063540388)
31. Letter from GEH to USNRC, MFN 06-056, LTR NEDE-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated February 2006 (ADAMS Accession No. ML060450677)
32. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, La Grange Park, Illinois: American Nuclear Society, 1979.
33. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1994, La Grange Park, Illinois: American Nuclear Society, 1994.
34. Letter from GEH to USNRC, MFN-098-96, "Implementation of Improved GE Steady-State Nuclear Methods," dated July 2, 1996. (ADAMS Accession No. ML070400507)
35. USNRC to GEH, "Amendment 26 to GE Licensing Topical Report NEDE-2401 1-P-A, GESTAR II - Implementing Improved GE Steady-State Methods," dated November 10, 1999. (ADAMS Package Accession No. ML993230387).
36. Letter from GEH to USNRC, MFN 00-002, LTR NEDE-32177P, Revision 2, "TRACG Qualification," dated January 31, 2000. (ADAMS Package Accession No. ML003682927)
37. Letter from GEH to USNRC, MFN 07-452, LTR NEDE-32177P, Revision 3, "TRACG Qualification," dated August 29, 2007. (ADAMS Accession No. ML072480007)
38. Letter from GEH to USNRC, MFN 99-040, LTR NEDE-32176P Revision 2, "TRACG Model Description," dated December 15, 1999. (ADAMS Accession No. ML993630096)
39. NUREG/CR-6534, Volume 1, "Modifications to Fuel Rod Materials Properties and Performance Models for High-Burnup Application," Lanning, D.D., Beyer, C.E., Painter, C.L., FRAPCON-3, October 1997 PNNL.
40. Letter from GEH to USNRC, LTR NEDO-20953-A, "Three-Dimensional BWR Core Simulator," dated January 1977. (ADAMS Accession No. ML070730687).
41. Letter from GEH to USNRC, MFN 07-040, "Part 21 Notification: Adequacy of GE thermal-Mechanical Methodology, GSTRM," dated January 21, 2007. (ADAMS Accession No. ML072290203).
42. USNRC Staff Evaluation of GEH Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GSTRM (ADAMS Accession No. ML073340722) - Issued as Appendix F to SE of NEDC-33173 dated January 17, 2008. (ADAMS Accession No. ML073340214).
43. Letter from GEH to USNRC, MFN 03-148, LTR NEDE-32906P, Supplement 1-A, "TRACG Application for Anticipated Transients Without SCRAM Overpressure Analysis," dated November 26, 2003. (ADAMS Package Accession No. ML033381073)
44. Letter from GEH to USNRC, MFN 05-017, LTR NEDC-33083P-A, "TRACG Application for ESBWR," dated April 8, 2005. (ADAMS Accession No. ML051390257)
45. Letter from GEH to USNRC, MFN 00-010, LTR NEDC-32956P, Revision 0, "TRACG User's Manual," dated February 28, 2000. (ADAMS Accession No. ML003688152)
46. Letter from GEH to USNRC, MFN 02-003, LTR NEDC-33006P, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," dated January 15, 2002. (ADAMS Package Accession No. ML020330034)
47. Letter from GEH to USNRC, MFN-06-435, "Commitment to Update GE's Void Fraction Data," dated November 3, 2006. (ADAMS Accession No. ML063110299)
48. Letter from GEH to USNRC, MFN-07-452, "Transmittal of GEH Topical Report, NEDE-32177P, Revision 3, TRACG Qualification, August 2007," dated August 29, 2007. (ADAMS Accession No. ML072480007)

49. Safety Evaluation of NEDC-33006P "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," dated October 15, 2008. (ADAMS Package Accession No. ML082830769)
50. Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 229 to Facility Operating License No. DPR-28 Regarding the Vermont Yankee Extended Power Uprate, dated March 2, 2006. (ADAMS Package Accession No. ML060050024)
51. Letter from GEH to USNRC, MFN-06-297, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-2 through 4.2-7, 4.3-3, 4.3-4, 4.4-2, 4.4-5, 4.4-6, 4.4-15 through 4.4-17, 4.4-19, 4.4-24, 4.4-27, 4.4-31 through 4.4-34, 4.4-36 through 4.4-38, 4.4-42 through 4.4-50, 4.4-52 through 4.4-56, 4.8-1 through 4.8-16," dated August 23, 2006. (ADAMS Accession No. ML062480252) As supplemented by: MFN-06-297 Supplement 1, November 13, 2006 (ADAMS Accession No. ML070600044); MFN-06-297 Supplement 2, December 21, 2006 (ADAMS Package Accession No. ML070110123); MFN-06-297 Supplement 4, January 26, 2007 (ADAMS Accession No. ML070380108); MFN-06-297 Supplement 5, February 8, 2007 (ADAMS Accession No. ML070470629); MFN-06-297 Supplement 7, April 10, 2007 (ADAMS Package Accession No. ML071210061); and MFN-06-297 Supplement 8, June 21, 2007 (ADAMS Package Accession No. ML071930214).
52. LTR NEDO-23729, "Nuclear Basis for ECCS (Appendix K) Calculations," dated November 1977. (ADAMS Accession No. ML073650369)
53. Letter from GEH to USNRC, MFN-06-467, "Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89," dated November 29, 2006. (ADAMS Accession No. ML063450251) As supplemented by: MFN-06-467 Supplement 1, March 6, 2007. (ADAMS Package Accession No. ML070720705)
54. Letter from GEH to USNRC, MFN-07-352, "Response to Portion of NRC Request for Additional Information Letter No. 66 - Related to ESBWR Design Certification Application - RAI Numbers 21.6-66 through 21.6-68, 21.6-80, 21.6-82, 21.6-84," dated June 8, 2007. (ADAMS Accession No. ML0719200980)
55. Letter from GEH to USNRC, MFN 06-046, LTR NEDE-32906P, Revision 2, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," dated February 14, 2006. (ADAMS Package Accession No. ML060530560)

Principal Contributor: P. Yarsky

Date: July 10, 2009

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

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

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

RAI 1

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

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

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

On a cycle specific basis the use of TGBLA06AE4 nuclear parameters for legacy GEH/GNF fuel may be justified. This justification may be provided to the NRC on an application specific basis to demonstrate for the fuel design that the nodal parameters are negligibly impacted by the code differences between TGBLA06AE4 and TGBLA06AE5. GEH has previously provided similar justification to the NRC for GE14 fuel lattices in Reference 9. It is expected that licensees or applicants that reference historical TGBLA06AE4 calculations likely utilize GE14 fuel and will reference those calculations previously reviewed by the NRC.

RAI 2

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

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

RAI 3

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

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

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

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

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

RAI 4

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

RAI 5

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

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

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

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

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

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

] and the results are provided in Figure 5-11 of Reference 21. The response indicates that the value of DMHZERO in the user's manual is based on an assessment [

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

In Part (d), the NRC staff requested that GEH explain the direct heat model when a control blade is present. The control blade in the bypass will displace water and reduce the nodal bypass water content. The TRACG04 model, according to the response, [ ]. The NRC staff finds that the approximation is acceptable for AOO and ATWS overpressure analyses because the bundles that respond with the greatest change in CPR for AOOs are the uncontrolled bundles [ ].

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

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

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

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

<u>RAI 6</u>

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

RAI 7

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

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

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

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

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

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

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

].

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

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

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

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

The void reactivity coefficient biases are based on uncontrolled lattices depleted at 40 percent in-channel void fraction [

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

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

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

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

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

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

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

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

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

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

RAI 8

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

RAI 9

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

RAI 10

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

RAI 11 (deleted)

RAI 12

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

RAI 13

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

RAI 14

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

RAI 15

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

RAI 16

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

For the current application (AOO and ATWS overpressure) the fuel temperature prediction affects the analyses in the coupling between the fluid conditions and the neutron flux. In particular, calculation of the fuel thermal conductivity will impact the fuel thermal time constant and the predicted transient fuel temperature. In cases where the predicted thermal conductivity is large, the fluid condition and the neutron flux are more tightly coupled via the heat flux through the pellet, gap, and cladding.

Similarly, the calculation of the transient change in fuel temperature is used to predict the nodal Doppler reactivity worth, which in turn, is assessed in the neutronic model to determine the transient reactivity feedback and neutron flux.

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

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

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

The NRC staff also considered the impact of gadolinia on the thermal conductivity. The NRC staff finds that at very high exposure, the TRACG04 model predicts only a minor influence on thermal conductivity by the gadolinia, whereas the FRAPCON3 model consistently predicts a much greater degradation in thermal conductivity with increasing gadolina concentration. The PRIME03 model is compared to the FRAPCON3 model for zero exposure and for high exposure in Figures A.16-3 and A.16-4, below respectively. The NRC staff notes that gadolinia isotopes are naturally stable, and expects that the depletion of gadolinia 155 and 157 under irradiation will result in the production of the stable gadolinia 156 and 158 isotopes (with small absorption cross sections). Therefore, the NRC staff has deferred the review of the PRIME thermal conductivity model to the specific review of PRIME and herein makes no statements regarding the veracity of the model for gadolinia bearing fuel near the end of life, because the NRC staff expects that the concentration of gadolinia itself does not appreciably change during irradiation.

Therefore, the NRC staff finds that: (1) the new fuel thermal conductivity model captures the effect of exposure on fuel thermal conductivity and agrees well with the FRAPCON3 model and (2) when compared to the NRC staff's FRAPCON model, the PRIME thermal conductivity model predicts a lesser degree of degradation with increasing gadolinia concentration.

[ ]

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

[ ]

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

[ ]

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

[ ]

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

RAI 17

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

].

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

RAI 18-20 (deleted)

RAI 21

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

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

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

RAI 22

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

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

RAI 23

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

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

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

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

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

<u>RAI 24</u>

The TRACG04 flow regime map and entrainment model were modified to improve agreement with low pressure Toshiba data. The purpose of the update was to improve calculational accuracy for the ESBWR LOCA calculations. In response to this RAI, the NRC staff requested information regarding the uncertainty analysis with respect to the void fraction. The TRACG02 and TRACG04 codes were assessed against the full-scale bundle test facility (FRIGG) void fraction data measurements of [ ]. The assessment indicates that the TRACG02 code predicts the void fraction with a mean error of [ ] and a standard deviation of [ ]. TRACG04 predicts the void fraction with a mean error of [ ] and a standard deviation of 2.4 percent. The bias in TRACG04 [ ] is conservative for AOO analyses. Therefore, GEH concluded that the TRACG02 uncertainty analysis is applicable to the TRACG04 code. The NRC staff agrees that the data most relevant to AOO applications is the high pressure FRIGG OF64 data and furthermore agrees that based on similarities in the qualification [ ] that the TRACG02 uncertainty analysis is applicable.

<u>RAI 25</u>

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

<u>RAI 26</u>

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

RAI 27

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

RAI 28

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

RAI 29

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

The NRC staff evaluation of the applicability of TRACG04 to EPU and MELLLA+ mixed core analysis was reviewed separately and is documented in Section 3.20.5 of this SE.

RAI 30

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

The response provides descriptive details of the implementation of the void history correction model. This model is implemented to account for biases and uncertainties in the TRACG04 void reactivity feedback as calculated by the PANAC11 kinetics engine. The historical void reactivity coefficient correction has been evaluated by the NRC staff in the response to RAI 7 and found unacceptable for application to EPU and MELLLA+

application as the pervious model was based on [ ].

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

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

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

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

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

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

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

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

].

The void reactivity coefficient ratio is fitted based on the [

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

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

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

[

].

RAI 31

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

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

The application of the interfacial shear model to high pressure experienced during AOOs was qualified against high pressure FRIGG OF36 data to justify the application of the model to the subject application. The FRIGG OF36 test includes a full-scale 6x6 bundle with a hydraulic diameter similar to current fuel designs [

]. The calculated and measured void fractions are provided in the response.

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

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

].

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

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

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

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

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

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

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

Table A.31.1: Interfacial Shear Qualification Database and Results

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

RAI 32

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

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

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

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

RAI 32(a)

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

RAI 32(b)

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

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

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

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

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

].

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

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

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

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

Using the relative moderator density differences as a surrogate to approximate the void fraction mismatch, the NRC staff finds that there is adequate indication to find that the differences between the Findlay-Dix predictions and the interfacial shear model tend to be within the uncertainty in the void fraction used to establish the statistical ΔCPR/ICPR for AOO analyses. This result is expected as stated previously as the two models share the same development basis data. However, the NRC staff notes that the interfacial shear model directly models the interphasic friction and is a detailed mechanistic model of the two-phase flow, while the robustness of the Findlay-Dix correlation is limited by the data used in its development and qualification.

Therefore, when considered in concert with the qualification of the interfacial shear model against the GE14 pressure drop data described in the response to RAI 31, the NRC staff has reasonable assurance that potential errors in the PANCEA predicted void fraction during TRACG initialization will have a negligible impact on the predicted ΔCPR/ICPR for limiting transients. The NRC staff also notes that the void fraction uncertainty is considered explicitly in the TRACG statistical evaluation to determine the ΔCPR/ICPR uncertainty and that the sensitivity study indicates that considering an additional uncertainty due to the mismatch between the PANACEA and TRACG at EPU conditions does not have a sufficient impact on the calculated ΔCPR/ICPR to merit a thermal margin enhancement to ensure adequate safety.

RAI 32(c)

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

The response to RAI 31 concludes that the MELLLA+ application does not warrant further consideration as the bundle conditions during normal operation and during transients must demonstrate margin to the onset of transition boiling. Therefore, GEH concludes that the void fraction predictions are adequately qualified. The NRC staff, as stated previously, is concerned that uncertainty in the prediction of the PANACEA steady-state void fraction may result in the miscalculation of the nodal reactivity feedback in response to void changes during transient evaluations. While the NRC staff has reviewed the interfacial shear qualification, including the updated qualification against GE14 critical power data, the NRC staff was concerned that the inter bundle nuclear coupling may amplify the impact of errors in the predicted nodal reactivity feedback characteristics at EPU or MELLLA+ conditions. The bundles are coupled by internodal neutron leakage. Potentially increased errors in neighboring bundle void reactivity feedback will have a direct effect on the efficacy of the code to accurately determine the limiting bundle transient response. Therefore, the NRC staff requested in RAI 32 that GEH specifically evaluate the impact of the void fraction mismatch at MELLLA+ conditions.

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

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

The response also provides the calculated void fraction mismatch based on the relative water density mismatch. The equation provided in the response relates the nodal water density mismatch to the nodal void fraction mismatch. The NRC staff notes that this void fraction mismatch should not be construed as the mismatch between the in-channel void fractions predicted by TRACG04 and PANACEA, [ ].
However, for the purpose of responding to the NRC staff's RAI this parameter serves as an adequate metric to quantify the mismatch between the steady-state (Findlay-Dix) and transient (interfacial shear) void fraction models.

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

The response correctly states that a bias introduced in the initial void fraction is not expected to significantly impact the change in void fraction predicted by TRACG04 in response to a transient pressurization event. The NRC staff agrees with this statement; however, the NRC staff notes that the nodal response surfaces passed from PANACEA to TRACG04 in the initialization accommodate [ ].
The void reactivity coefficient is known to increase in magnitude and become more negative with increasing instantaneous void conditions.

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

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

].

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

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

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

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

[

], which is well beneath the threshold of significance.

As the purpose of the analysis in the response is to quantify the bias introduced in the transient response as a result of the deterministic nodal void fraction mismatch, the NRC staff does not find the one standard deviation uncertainty band to be an acceptable basis for the threshold of significance. The interdependence of the void reactivity coefficient uncertainty and void fraction uncertainty is not explicitly accounted for in the uncertainty analysis. Therefore, the NRC staff considered a threshold of significance in its review of the current RAI response of 0.005. Values greater than 0.005 approach the one sigma deviation difference considered significant in Section 2.6.1 of NEDE-32906P-A. The NRC staff reiterates that the void fraction mismatch is a deterministic evaluation of the differences in two void fraction models used analytically.

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

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

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

].

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

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

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

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

The NRC staff considered the relevancy of the sensitivity studies to the broad range of anticipated operational occurrences that may occur for operating BWR plants. Licensees analyze a host of transients each operating cycle to determine thermal operating limits. The potentially limiting AOO events are determined and analyzed. The potentially limiting transient events analyzed on a cycle specific basis include: generator load rejection or turbine trip without bypass, loss of feedwater heat or inadvertent high pressure coolant injection (HPCI), control rod withdrawal error, feedwater controller failure to maximum demand, and pressure regulator failure (for BWR/6 plants).

For the operating fleet of BWR plants these events are generally the limiting events. Of these the generator load rejection without bypass, turbine trip without bypass, feedwater controller failure, and pressure regulator failure events are pressurization transients. The sensitivity studies provided in the RAI response provide details of the sensitivity of the transient response to pressurization transients.

The NRC staff expects that the sensitivity demonstrated for the pressurization transients would bound that for the other potentially limiting events: control rod withdrawal error, loss of feedwater heat, and inadvertent HPCI.

The control rod withdrawal error is a postulated AOO whereby the operator erroneously, continuously withdraws the highest worth control blade above 75 percent of power. The event is terminated by the rod block monitor (RBM). During the transient the local reactor power increases due to the reactivity insertion from the withdrawal. The increased local power is sensed by the LPRMs. The RBM will prohibit further withdrawal of the rod as the power increase because increasingly severe. The negative reactivity feedback from any void formation is modeled in TRACG04; however, the bundle power history is a much stronger function of the control blade reactivity and withdrawal rate. Therefore, the NRC staff finds that the CPR sensitivity to any void mismatch for a control rod withdrawal error would be bound by the pressurization transient results.

The loss of feedwater heat and the inadvertent HPCI AOOs are similar. These AOOs are postulated events where the core flow inlet subcooling is increased due to cooler water injection to the vessel. These events tend to be slowly evolving transients where the core approaches a new steady-state condition where the power increases to compensate for positive reactivity insertion. Generally, the core will approach a condition where the adjoint-weighted core average void fraction remains the essentially the same. Therefore, the NRC staff does not expect the dynamic response to be sensitive to mild variation in the local void fraction due to void-model differences. On this basis, the NRC staff finds that the CPR sensitivity calculated for the pressurization transients would bound any CPR sensitivity for the loss of feedwater heat or inadvertent HPCI AOOs.

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

RAI 33

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

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

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

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

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

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

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

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

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

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

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

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

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

A.33.1 Axial Power Shape

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

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

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

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

A.33.2 Control Rod Pattern

The NRC staff requested information regarding the limiting control rod pattern. The response directed the NRC staff to Table 8-10 of Reference 21. In the original application of TRACG02 for AOOs, GE performed sensitivity analyses to determine the sensitivity of the thermal margin to initial plant parameters. An analysis was performed for a turbine trip with no-bypass (TTNB) using nominal EOC axial power shapes with a B&W control rod pattern and a nominal control rod pattern (with several rods partially inserted).

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

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

The NRC staff found that there is [ ] (Reference 8).

A.33.3 Transient Varying Axial Power

The NRC staff reviewed the response and agrees that, generally, the TVAPS is most severe for bottom-peaked power shapes, but is compensated for an increased SCRAM reactivity due to a down-skewed flux adjoint. The sensitivity analyses performed using TRACG02 indicate that the sensitivity of the transient CPR to power shape is on the [ ] (Reference 21).

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

EPU plants typically operate with a very limited operational flow window. Therefore, the NRC staff does not expect that BOC to MOC UB analyses will be sensitive to the variation in flow associated with the burn strategy for EPU plants and the general results of the ODYN sensitivity analysis can be applied. MELLLA+ operation allows variation in core flow to control excess reactivity. In the BOC to MOC the strategy may involve reduction in core flow. The net combined effect [ ]. At the MELLLA+ flow corner the SCRAM worth is further reduced due to an increase in core average void fraction and hence a hardened neutron spectrum. The TRACG02 sensitivity analyses only consider an UB bottom-peaked power shape with a peak in node 3 (Figure 8-35 of Reference 21).

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

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

A.33.4 Supplemental Information

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

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

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

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

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

The BOC to MOC UB B&W pattern cases indicated very mild transient responses for TTNB indicating that the enhanced SCRAM reactivity continues to dominate the TVAPS effect at MELLLA+ conditions. The BOC to MOC UB B&W pattern as well as nominal pattern FWCF transients indicate larger ΔCPR results. This is due mostly to the effect of the increased feed flow prior to turbine trip as well as the availability of the turbine bypass to limit the pressurization. The analysis results are summarized in Table A.33.4.1. The TRACG02 analyses predict a [ ].

The NRC staff finds that this is consistent with the trends observed in ODYN analyses and trends observed for pre-EPU plants referencing ODYN and TRACG analyses. Therefore, the NRC staff is reasonably assured that the analysis assumptions imposed in reload licensing to calculate the cycle-specific OLMCPR remain adequately conservative for application to MELLLA+ conditions.

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

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

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

AOO	Initial Condition	ΔCPR	ΔΔCPR
TTNB	[		
FWCF			]

AOO	Initial Condition	ΔCPR	ΔΔCPR
TTNB	[		
FWCF			]

RAI 34

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

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

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

<u>REFERENCES</u>

1. Letter from GEH to USNRC, MFN-06-155, LTR NEDE-32906P, Supplement 3, "Migration to TRACG04 / PANACII from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated May 25, 2006. (ADAMS Package Accession No. ML061500182)
2. Letter from GEH to USNRC, MFN-07-455, "Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated August 15, 2007. (ADAMS Accession No. ML072330518)
3. Letter from GEH to USNRC, MFN-07-445, Supplement 1, "Partial Response to Request for Additional Information RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569), Supplement 1," dated December 20, 2007. (ADAMS Package Accession No. ML073650365)
4. Letter from GEH to USNRC, MFN-08-483, "Response to Request for Additional Information (RAI) 30 RE: GE Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," dated May 30, 2008. (ADAMS Accession No. ML081550192)
5. Letter from GEH to USNRC, MFN-07-455, Supplement 2, "Response to USNRC Follow-up Question on RAI 33 RE: GEH Topical Report NEDE-32906P, Supplement 3, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients, (TAC No. MD2569)," dated June 6, 2008. (ADAMS Package Accession No. ML081630008)
6. Letter from GEH to USNRC, MFN-08-547, "Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, 'Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS

Overpressure Transients,' (TAC No. MD2569)," dated June 30, 2008. (ADAMS Accession No. ML081840270)

7. Letter from GEH to USNRC, MFN-08-604, "Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated (TAC No. MD2569)," dated July 30, 2008. (ADAMS Accession No. ML082140580)
8. Final Safety Evaluation of NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated January 17, 2008. (ADAMS Accession No. ML073340214)
9. Letter from GEH to USNRC, MFN 06-297 Supplement 1, "Supplemental Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Number 4.3-3," November 8, 2006. (ADAMS Accession No. ML063400067)
10. Results Summary Report of USNRC Staff Audit in December 2006 "ESBWR DCD Section 4.3 Nuclear Codes," dated October 15, 2008.
11. Addendum 1 to Results Summary Report of USNRC Staff Audit in December 2006 "ESBWR DCD Section 4.3 Nuclear Codes," dated February 2007.
12. USNRC Staff Audit Summary, "Summary of Audit for Nuclear Design Codes October/November 2006," dated July 19, 2007. (ADAMS Accession No. ML071700037)
13. Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 229 to Facility Operating License No. DPR-28 Regarding the Vermont Yankee Extended Power Uprate, dated March 2, 2006. (ADAMS Package Accession No. ML060050024)
14. Letter from GEH to USNRC, MFN-06-297, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-2 through 4.2-7, 4.3-3, 4.3-4, 4.4-2, 4.4-5, 4.4-6, 4.4-15 through 4.4-17, 4.4-19, 4.4-24, 4.4-27, 4.4-31 through 4.4-34, 4.4-36 through 4.4-38, 4.4-42 through 4.4-50, 4.4-52 through 4.4-56, 4.8-1 through 4.8-16," dated August 23, 2006. (ADAMS Accession No. ML062480252) As supplemented by: MFN-06-297 Supplement 1, November 13, 2006 (ADAMS Accession No. ML070600044); MFN-06-297 Supplement 2, December 21, 2006 (ADAMS Package Accession No. ML070110123); MFN-06-297 Supplement 4, January 26, 2007 (ADAMS Accession No. ML070380108); MFN-06-297 Supplement 5, February 8, 2007 (ADAMS Accession No. ML070470629); MFN-06-297 Supplement 7, April 10, 2007 (ADAMS Package Accession No. ML071210061); and MFN-06-297 Supplement 8, June 21, 2007 (ADAMS Package Accession No. ML071930214).
15. Letter from GEH to USNRC, MFN-07-452, "Transmittal of GEH Topical Report, NEDE-32177P, Revision 3, TRACG Qualification, August 2007," dated August 29, 2007. (ADAMS Accession No. ML072480007)
16. Letter from GEH to USNRC, MFN-08-547, "Transmittal of Response to NRC Request for Additional Information - NEDC-32906P, Supplement 3, 'Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,' (TAC No. MD2569)," dated June 30, 2008. (ADAMS Accession No. ML081840270)
17. LTR NEDO-23729, "Nuclear Basis for ECCS (Appendix K) Calculations," dated November 1977. (ADAMS Accession No. ML073650369)
18. Letter from GEH to USNRC, MFN 06-109, LTR NEDE-32176P, Revision 3, "TRACG Model Description," dated April 20, 2006. (ADAMS Accession No. ML061160236)

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

Appendix B: TGBLA06/PANAC11/TRACG04 Code Changes

Table B.1: Code Changes to TGBLA06A

Version	Item	Code Change
[		
		]

Table B.2: Code Changes to PANAC11A,P

Version	Item	Code Change
[		

Version	Item	Code Change

Version	Item	Code Change
		]

Table B.3: Code Changes to TRACG04A,P

Version	Item	Code Change
[		

Version	Item	Code Change

Version	Item	Code Change

Version	Item	Code Change

Version	Item	Code Change
		]

Appendix C: Sample Calculation of Void Reactivity Sensitivity

This appendix performs a calculation of the impact of void-fraction errors on the value of the void reactivity coefficient. The void reactivity is tracked in TRACG04 according to the PANAC11 calculated nodal response surfaces. The relative water density is used in TRACG04 to determine the nodal reactivity. The relative water density is related to the void fraction according to Equation C-1.

$$U = 1 - VFAT + \frac{VFAT}{\rho_{ref}}\left[(1-\alpha)\rho_f + \alpha\rho_g\right]$$

Equation C-1

Where U is the lattice average relative water density
VFAT is the in-channel to total flow area ratio
α is the in-channel void fraction
ρ is the water density
f denotes saturated liquid water density (0.73749 g/cc)
g denotes saturated liquid vapor density (0.03733 g/cc)
ref denotes the reference water density (0.73751 g/cc)

For typical modern fuel designs, the in-channel to total flow ratio is on the order of 2/3. Therefore, the staff assumed a value of 2/3 for this parameter.

The nodal response surfaces are based on quadratic parametric fits of the TGBLA06 calculated lattice parameters. In this case, the staff considered the fit of the reactivity void coefficient. The nodal reactivity as a function of the nodal relative water density is given according to Equation C-2 and, therefore, the void reactivity coefficient can be calculated according to Equation C-3.

$$k_{nodal} = A + BU + \frac{C}{2}U^2$$

Equation C-2

$$\frac{\partial k_{nodal}}{\partial \alpha} = \frac{\partial k_{nodal}}{\partial U}\frac{\partial U}{\partial \alpha} = (B + CU)\left(\frac{\rho_g - \rho_f}{\rho_{ref}}\right)VFAT$$

$$\frac{\partial k_{nodal}}{\partial \alpha} = B\left(\frac{\rho_g - \rho_f}{\rho_{ref}}\right)VFAT + C\left(\frac{\rho_g - \rho_f}{\rho_{ref}}\right)VFAT\left[1 - VFAT + \frac{VFAT}{\rho_{ref}}\left[(1-\alpha)\rho_f + \alpha\rho_g\right]\right]$$

Equation C-3

Where k_{nodal} is the nodal reactivity
A, B, and C are the fitting constants in the nodal response surface
U is the nodal relative water density

The nodal reactivity coefficient is corrected for known biases according to MCNP comparisons. Therefore, the constant term in the expression is known to be accurate based on the void reactivity coefficient correction model. The variation in the void reactivity coefficient with void fraction is given by the second term in the last expression in Equation C-2. Based on several lattice analyses a typical value of C ranges between 0.1 and 0.2, the staff assumed 0.15 for the current purpose. The change in the nodal void reactivity coefficient for a change in void fraction can be determined based on the second derivative shown in Equation C-3.

$$\frac{\partial^2 k_{nodal}}{\partial^2 \alpha} = C\left[\left(\frac{\rho_g - \rho_f}{\rho_{ref}}\right) VFAT\right]^2 \approx 0.06$$

Equation C-4

Based on the staff's analysis, the sensitivity of the nodal void reactivity coefficient in units of pcm per change in void (% void) per error in void fraction (% void) is approximately 0.6 pcm/%/%. For reference, a typical value of the void reactivity coefficient at limiting cold conditions is approximately –100 pcm/%.

Appendix D: Limitations and Conditions from NRC Staff SE of LTR NEDC-33006P

The following limitations and conditions are excerpted from the safety evaluation (SE) of licensing topical report (LTR) NEDC-33006P, "General Electric Boiling Water Reactor [BWR] Maximum Extended Load Line Limit Analysis [MELLLA] Plus," (MELLLA+). All references, section numbers, appendix, and page numbers cited in the following sections refer to those in the SE of LTR NEDC-33006P. All acronyms stated below are defined in the SE of LTR NEDC-33006P. The SE of LTR NEDC-33006P is in the Agencywide Documents Access and Management System at Package Accession No. ML082830769.

12.1 GEXL-PLUS (SECTION 1.1.4)

The plant-specific application will confirm that for operation within the boundary defined by the MELLLA+ upper boundary and maximum CF range, the GEXL-PLUS experimental database covers the thermal-hydraulic conditions the fuel bundles will experience, including, bundle power, mass flux, void fraction, pressure, and subcooling. If the GEXL-PLUS experimental database does not cover the within bundle thermal-hydraulic conditions, during steady state, transient conditions, and DBA conditions, [GEH] will inform the NRC at the time of submittal and obtain the necessary data for the submittal of the plant-specific MELLLA+ application.

In addition, the plant-specific application will confirm that the experimental pressure drop database for the pressure drop correlation covers the pressure drops anticipated in the MELLLA+ range.

With subsequent fuel designs, the plant-specific applications will confirm that the database supporting the CPR correlations covers the powers, flows and void fractions BWR bundles will experience for operation at and within the MELLLA+ domain, during steady state, transient, and DBA conditions. The plant-specific submittal will also confirm that the NRC staff reviewed and approved the associated CPR correlation if the changes in the correlation are outside the GESTAR II (Amendment 22) process. Similarly, the plant-specific application will confirm that the experimental pressure drop database does cover the range of pressures the fuel bundles will experience for operation within the MELLLA+ domain.

12.2 RELATED LTRS (SECTION 1.1.5)

Plant-specific MELLLA+ applications must comply with the limitations and conditions specified in and be consistent with the purpose and content covered in the NRC staff SEs approving the latest version of the following LTRs: NEDC-33173P, NEDC-33075P, and NEDC-33147 (References 37, 45, and 47).

12.3 CONCURRENT CHANGES (SECTION 1.2.1)

a) The plant-specific analyses supporting MELLLA+ operation will include all operating condition changes that are implemented at the plant at the time of MELLLA+ implementation. Operating condition changes include, but are not limited to, those changes that affect, an increase in the dome pressure, maximum CF, fuel cycle length, or any changes in the licensed operational enhancements. For example, with an increase in dome pressure, the following analyses must be analyzed: the ATWS analysis, the ASME overpressure analyses, the transient analyses, and the ECCS-LOCA analysis. Any changes to the safety system settings or any actuation setpoint changes necessary to operate with the increased dome pressure must be included in the evaluations (e.g., SRV setpoints).

b) For all topics in LTR NEDC-33006P that are reduced in scope or generically dispositioned, the plant-specific application will provide justification that the reduced scope or generic disposition is applicable to the plant. If changes that invalidate the LTR dispositions are to be implemented at the time of MELLLA+ implementation, the plant-specific application will provide analyses and evaluations that demonstrate the cumulative effect with MELLLA+ operation. For example, if the dome pressure is increased, the ECCS performance will be evaluated on a plant-specific basis.

c) Any generic bounding sensitivity analyses provided in LTR NEDC-33006P will be evaluated to ensure that the key plant-specific input parameters and assumptions are applicable and bounded. If these generic sensitivity analyses are not applicable or additional operating condition changes affect the generic sensitivity analyses, a plant specific evaluation will be provided. For example, with an increase in the dome pressure, the ATWS sensitivity analyses that model operator actions (e.g., depressurization if the HCTL is reached) needs to be reanalyzed, using the bounding dome pressure condition.

d) If a new GE fuel product line or another vendor's fuel is loaded at the plant, the applicability of any generic sensitivity analyses supporting the MELLLA+ application shall be justified in the plant-specific application. If the generic sensitivity analyses cannot be demonstrated to be applicable, the analyses will be performed including the new fuel. For example, the ATWVS instability analyses supporting the MELLLA+ condition are based on the GE14 fuel response. New analyses that demonstrate the ATWS instability performance of the new GE fuel or another vendor's fuel for MELLLA+ operation shall be provided to support the plant-specific application.

e) If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the analyses supporting the plant-specific MELLLA+ application will be based on a specific core configuration or bounding core conditions. Any topics that are generically dispositioned or reduced in scope in LTR NEDC-33006P will be demonstrated to be applicable, or new analyses based on the specific core configuration' or bounding core conditions will be provided.

f) If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the plant-specific application will reference an NRC-approved stability method supporting MELLLA+ operation, or provide sufficient plant-specific information to allow the NRC staff to review and approve the stability method supporting MELLLA+ operation. The plant-specific application will demonstrate that the analyses and evaluations supporting the stability method are applicable to the fuel loaded in the core.

g) For MELLLA+ operation, core instability is possible in the event a transient or plant maneuver places the reactor at a high power/low-flow condition. Therefore, plants operating at MELLLA+ conditions must have a NRC-approved instability protection method. In the event the instability protection method is inoperable, the applicant must employ an NRC-approved backup instability method. The licensee will provide technical specification (TS) changes that specify the instability method operability requirements for MELLLA+ operation, including any backup stability protection methods.

12.4 RELOAD ANALYSIS SUBMITTAL (SECTION 1.2.2.3.2)

The plant-specific MELLLA+ application shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel- and cycle-dependent analyses including the plant-specific thermal limits assessment may be submitted by supplementing the initial M+SAR. Additionally, the SRLR for the initial MELLLA+ implementation cycle shall be submitted for NRC staff confirmation.

12.5 OPERATING FLEXIBILITY (SECTION 1.3.3)

a) The licensee will amend the TS LCO for any equipment out-of-service (i.e., SLO) or operating flexibilities prohibited in the plant-specific MELLLA+ application.

b) For an operating flexibility, such as FWHOOS, that is prohibited in the MELLLA+ plant-specific application but is not included in the TS LCO, the licensee will propose and implement a license condition.

c) The power flow map is not specified in the TS; however, it is an important licensed operating domain. Licensees may elect to be licensed and operate the plant under plant-specific-expanded domain that is bounded by the MELLLA+ upper boundary. Plant-specific applications approved for operation within the MELLLA+ domain will include the plant-specific power/flow map specifying the licensed domain in the COLR.

12.6 SLMCPR STATEPOINTS AND CF UNCERTAINTY (SECTION 2.2.1.1)

Until such time when the SLMCPR methodology (References 40 and 41) for off-rated SLMCPR calculation is approved by the staff for MELLLA+ operation, the SLMCPR will be calculated at the rated statepoint (120 percent P/100 percent CF), the plant-specific minimum CF statepoint (e.g., 120 percent P/80 percent CF), and at the 100 percent OLTP at 55 percent CF statepoint. The currently approved off-rated CF uncertainty will be used for the minimum CF and 55 percent CF statepoints. The uncertainty must be consistent with the CF uncertainty currently applied to the SLO operation or as NRC-approved for MELLLA+ operation. The calculated values will be documented in the SRLR.

12.7 STABILITY (SECTION 2.4.1)

Manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. If the primary stability protection system is declared inoperable, a non-manual NRC-approved backup protection system must be provided, or the reactor core must be operated below a NRC-approved backup stability boundary specifically approved for MELLLA+ operation for the stability option employed.

12.8 FLUENCE METHODOLOGY AND FRACTURE TOUGHNESS (SECTION 3.2.1)

The applicant is to provide a plant-specific evaluation of the MELLLA+ RPV fluence using the most up-to-date NRC-approved fluence methodology. This fluence will then be used to provide a plant-specific evaluation of the RPV fracture toughness in accordance with RG 1.99, Revision 2.

12.9 REACTOR COOLANT PRESSURE BOUNDARY (SECTION 3.5.1)

MELLLA+ applicants must identify all other than Category "A" materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the MELLLA+ operation on a plant-specific basis.

12.10 ECCS-LOCA OFF-RATED MULTIPLIER (SECTION 4.3.1.3)

a) The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF, rated EPU power/minimum CF, at the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA-LOCA PCT response for the operation within the MELLLA+ boundary. The M+ SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.

b) LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific off-rated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle-specific off-rated thermal limits applied at the 55 percent CF and/or the transition statepoints are consistent with those assumed in the plant-specific ECCS-LOCA analyses.

c) Off-rated limits will not be applied to the minimum CF statepoint.

d) If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.

12.11 ECCS-LOCA AXIAL POWER DISTRIBUTION EVALUATION (SECTION 4.3.1.4)

For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

12.12 ECCS-LOCA REPORTING (SECTION 4.3.1.5)

a) Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and

b) The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

12.13 SMALL BREAK LOCA (SECTION 4.3.2.4)

Small break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small break LOCA limited based on small break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to [ ] relative to the Appendix K or the licensing basis PCT.

12.14 BREAK SPECTRUM (SECTION 4.3.3)

The scope of small break LOCA analysis for MELLLA+ operation relies upon the EPU small break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.

12.15 BYPASS VOIDING ABOVE THE D-LEVEL (SECTION 5.1.1.5.3)

Plant-specific MELLLA+ applications shall identify where in the MELLLA+ upper boundary the bypass voiding greater than 5 percent will occur above the D-level. The licensee shall provide in the plant-specific submittal the operator actions and procedures that will mitigate the impact of the bypass voiding on the TIPs and the core simulator used to monitor the fuel performance. The plant-specific submittal shall also provide discussion on what impact the bypass voiding greater than 5 percent will have on the NMS as defined in Section 5.1.1.5. The NRC staff will evaluate on plant-specific bases acceptability of bypass voiding above D level.

12.16 RWE (SECTION 9.1.1.2)

Plants operating at the MELLLA+ operating domain shall perform RWE analyses to confirm the adequacy of the generic RBM setpoints. The M+SAR shall provide a discussion of the analyses performed and the results.

12.17 ATWS LOOP (SECTION 9.3.1.1)

As specified in LTR NEDC-33006P, at least two plant-specific ATWS calculations must be performed: MSIVC and PRFO. In addition, if RHR capability is affected by LOOP, then a third plant-specific ATWS calculation must be performed that includes the reduced RHR capability. To evaluate the effect of reduced RHR capacity during LOOP, the plant-specific ATWS calculation must be performed for a sufficiently large period of time after HSBW injection is complete to guarantee that the suppression pool temperature is cooling, indicating that the RHR capacity is greater than the decay heat generation. The plant-specific application should include evaluation of the safety system performance during the long-term cooling phase, in terms of available NPSH.

12.18 ATWS TRACG ANALYSIS (SECTION 9.3.1.3)

a) For plants that do not achieve hot shutdown prior to reaching the heat capacity temperature limit (HCTL) based on the licensing ODYN code calculation, plant-specific MELLLA+ implementations must perform best-estimate TRACG calculations on a plant-specific basis.

b) The TRACG analysis will account for all plant parameters, including water-level control strategy and all plant-specific emergency operating procedure (EOP) actions.

c) The TRACG calculation is not required if the plant increases the boron-10 concentration/enrichment so that the integrated heat load to containment calculated by the licensing ODYN calculation does not change with respect to a reference OLTP/75 percent flow ODYN calculation.

d) Peak cladding temperature (PCT) for both phases of the transient (initial overpressure and emergency depressurization) must be evaluated on a plant-specific basis with the TRACG ATWS calculation.

e) In general, the plant-specific application will ensure that operation in the MELLLA+ domain is consistent with the assumptions used in the ATWS analysis, including equipment out of service (e.g., FWHOOS, SLO, SRVs, SLC pumps, and RHR pumps, etc.). If assumptions are not satisfied, operation in MELLLA+ is not allowed. The SRLR will specify the prohibited flexibility options for plant-specific MELLLA+ operation, where applicable. For key input parameters, systems and engineering safety features that are important to simulating the ATWS analysis and are specified in the Technical Specification (TS) (e.g., SLCS parameters, ATWS RPT, etc.), the calculation assumptions must be consistent with the allowed TS values and the allowed plant configuration. If the analyses deviate from the allowed TS configuration for long term equipment out of service (i.e., beyond the TS LCO), the plant-specific application will specify and justify the deviation. In addition, the licensee must ensure that all operability requirements are met (e.g., NPSH) by equipment assumed operable in the calculations.

f) Nominal input parameters can be used in the ATWS analyses provided the uncertainty treatment and selection of the values of these input parameters are consistent with the input methods used in the original GE ATWS analyses in NEDE-24222. Treatment of key input parameters in terms of uncertainties applied or plant-specific TS value used can differ from the original NEDE-24222 approach, provided the manner in which it is used yields more conservative ATWS results.

g) The plant-specific application will include tabulation and discussion of the key input parameters and the associated uncertainty treatment.

12.19 PLANT-SPECIFIC ATWS INSTABILITY (SECTION 9.3.3.1)

Until such time that NRC approves a generic solution for ATWS instability calculations for MELLLA+ operation, each plant-specific MELLLA+ application must provide ATWS instability analysis that satisfies the ATWS acceptance criteria listed in SRP Section 15.8. The plant-specific ATWS instability calculation must: (1) be based on the peak-reactivity exposure conditions, (2) model the plant-specific configuration important to ATWS instability response including mixed core, if applicable, and (3) use the regional-mode nodalization scheme. In order to improve the fidelity of the analyses, the plant-specific calculations should be based on latest NRC-approved neutronic and thermal-hydraulic codes such as TGBLA06/PANAC11 and TRACG04.

12.20 GENERIC ATWS INSTABILITY (SECTION 9.3.3.1)

Once the generic solution is approved, the plant-specific applications must provide confirmation that the generic instability analyses are relevant and applicable to their plant. Applicability confirmation includes review of any differences in plant design or operation that will result in significantly lower stability margins during ATWS such as:

- turbine bypass capacity,
- fraction of steam-driven feedwater pumps,
- any changes in plant design or operation that will significantly increase core inlet subcooling during ATWS events,
- significant differences in radial and axial power distributions,
- hot-channel power-to-flow ratio,
- fuel design changes beyond GE14.

12.21 INDIVIDUAL PLANT EVALUATION (SECTION 10.5)

Licensees that submit a MELLLA+ application should address the plant-specific risk impacts associated with MELLLA+ implementation, consistent with approved guidance documents (e.g., NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A) and the Matrix 13 of RS-001 and re-address the plant-specific risk impacts consistent with the approved guidance documents that were used in their approved EPU application and Matrix 13 of RS-001. If an EPU and MELLLA+ application come to the NRC in parallel, the expectation is that the EPU submittal will have incorporated the MELLLA+ impacts.

12.22 IASCC (SECTION 10.7)

The applicant is to provide a plant-specific IASCC evaluation when implementing MELLLA+, which includes the components that will exceed the IASCC threshold of 5×10^{20} n/cm^2 (E>1 MeV), the impact of failure of these components on the integrity of the reactor internals and core support structures under licensing design bases conditions, and the inspections that will be performed on components that exceed the IASCC threshold to ensure timely identification of IASCC, should it occur.

12.23 LIMITATIONS FROM THE ATWS RAI EVALUATIONS (APPENDIX A)

12.23.1 Limitation from Appendix A RAI 4-1

See limitation 12.18.d.

12.23.2 Limitation from Appendix A RAI 4-2

The plant-specific ODYN and TRACG key calculation parameters must be provided to the staff so they can verify that all plant-specific automatic settings are modeled properly.

12.23.3 Limitation from Appendix A RAI 11-4

The ATWS peak pressure response would be dependent upon SRVs upper tolerances assumed in the calculations. For each individual SRV, the tolerances used in the analysis must be consistent with or bound the plant-specific SRV performance. The SRV tolerance test data would be statistically treated using the NRC's historical 95/95 approach or any new NRC-approved statistical treatment method. In the event that current EPU experience base shows propensity for valve drift higher than pre-EPU experience base, the plant-specific transient and ATVVS analyses would be based on the higher tolerances or justify the reason why the propensity for the higher drift is not applicable the plant's SRVs.

12.23.4 Limitation from Appendix A RAI 13-1

EPG/SAG parameters must be reviewed for applicability to MELLLA+ operation in a plant-specific basis. The plant-specific MELLLA+ application will include a section that discusses the plant-specific EOPs and confirms that the ATWS calculation is consistent with the operator actions.

12.23.5 Limitation from Appendix A RAI 14-5

The conclusions of this LTR and associated SE are limited to reactors operating with a power density lower than 52.5 MW/MLBM/hr for operation at the minimum allowable CF at 120 percent OLTP. Verification that reactor operation will be maintained below this analysis limit must be performed for all plant-specific applications.

12.23.6 Limitation from Appendix A RAI 14-9

For MELLLA+ applications involving GE fuel types beyond GE14 or other vendor fuels, bounding ATWS Instability analysis will be provided to the staff. Note: this limitation does not apply to special test assemblies.

12.23.7 Limitation from Appendix A RAI 14-10

See limitation 12.23.6.

12.23.8 Limitation from Appendix A RAI 14-11

The plant-specific ATWS calculations must account for all plant- and fuel-design-specific features, such as the debris filters.

12.23.9 Limitation from Appendix A RAI 16-1

Plant-specific applications must review the safety system specifications to ensure that all of the assumptions used for the ATWS SE indeed apply to their plant-specific conditions.

The NRC staff review will give special attention to crucial safety systems like HPCI, and physical limitations like NPSH and maximum vessel pressure that RCIC and HPCI can inject. The plant-specific application will include a discussion on the licensing bases of the plant in terms of NPSH and system performance. It will also include NPSH and system performance evaluation for the duration of the event.

12.23.10 Limitation from Appendix A RAI 16-3

Plant-specific applications must ensure that an increase in containment pressure resulting from ATWS events with EPU/MELLLA+ operation does not affect adversely the operation of safety-grade equipment.

12.23.11 Limitation from Appendix A RAI 17-1

The plant-specific applications must justify the use of plant-specific suppression pool temperature limits for the ODYN and TRACG calculations that are higher than the HCTL limit for emergency depressurization.

12.24 LIMITATIONS FROM FUEL DEPENDENT ANALYSES RAI EVALUATIONS (APPENDIX B)

12.24.1 Limitation from Appendix B RAI 3

For EPU/MELLLA+ plant-specific applications that use TRACG or any code that has the capability to model in-channel water rod flow, the supporting analysis will use the actual flow configuration.

12.24.2 Limitation from Appendix B RAI 7

The EPU/MELLLA+ application would provide the exit void fraction of the high-powered bundles in the comparison between the EPU/MELLLA+ and the pre-MELLLA+ conditions.

12.24.3 Limitation from Appendix B RAI 17

See limitation 12.6.

12.24.4 Limitation from Appendix B RAI 30

See limitation 12.18.d.

Appendix E: Acronym and Abbreviation List

Acronym	Definition
3D	Three Dimensional
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without SCRAM
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CALTIP	Calculated TIP Responses
CLTP	Currently Licensed Thermal Power
CPR	Critical Power Ratio
COLR	Core Operating Limit Report
CRDA	Control Rod Drop Accident
CSAU	Code Scaling Applicability and Uncertainty
CSHT	Core Spray Heat Transfer
ΔCPR	Transient Change in Critical Power Ratio
DIF3D	Argonne National Laboratory Fine Mesh Three Dimensional Diffusion Code
DW	Dry Well
EBWR	Experimental Boiling Water Reactor
ECCS	Emergency Core Cooling System
ECP	Engineering Computer Program
ELLA	Extended Load Line Limit Analysis
ENDF	Evaluated Nuclear Data File
EOC	End of Cycle
EPU	Extended Power Uprate
ESBWR	Economic Simplified Boiling Water Reactor (a GEH Passive, Natural Circulation BWR Design)
FDMH	Fraction of Direct Moderator Heat
FRAPCON	Pacific Northwest National Laboratory Fuel Thermal Mechanical Code
FRIGG	Full Scale Bundle Test Facility
FWCF	Feedwater Flow Controller Failure to Maximum Demand
GDC	General Design Criterion
GDCS	Gravity Driven Cooling System
GE	General Electric
GEH	GE - Hitachi Nuclear Energy Americas, LLC
GESTAR	General Electric Standard Application for Reactor Fuel
GEXL	GE Critical Quality Boiling Length Correlation
GNF	Global Nuclear Fuels
GSTRM	GE Fuel Thermal Mechanical Code
HBB	Hard Bottom Burn
HFCL	High Flow Control Line
HPCI	High Pressure Coolant Injection
IC	Isolation Condenser
ICF	Increased Core Flow
ICPR	Initial Critical Power Ratio
ICS	Isolation Condenser System

Acronym	Definition
INEL (currently INL)	Idaho National Engineering Laboratory (currently Idaho National Laboratory)
K5	Kashiwazaki-Kariwa Unit 5
KKM	Kernkraftwerk Muhleberg
KSP	Kuhn-Schrock-Peterson
Level 2	GNF designation for highest degree of quality assurance
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LOFW	Loss of Feedwater
LTPL	Licensed Thermal Power Line
LTR	Licensing Topical Report
LWR	Light Water Reactor
MCNP	Monte Carlo N Particle
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MLHGR	Maximum Linear Heat Generation Rate
MOC	Middle of Cycle
MOP	Mechanical Overpower
MOX	Mixed Oxide
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure with Flux SCRAM
MSL	Main Steam Line
NITER	PANACEA mode identification
ODYN	GEH Transient Analysis Code
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Originally Licensed Thermal Power
ORIGEN	Oak Ride National Laboratory One Dimensional Depletion Code
P4B	Four Bundle Power
PANAC	Three dimensional BWR core simulator
PANACEA	Three dimensional BWR core simulator
PANDA	small-scale 30 tube ESBWR passive containment core cooling system test
PANTHERS	a full-scale prototype of the ESBWR ICS and a near-full scale PCCS test
PB	Peach Bottom
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCTIP	Measured TIP Response
PIRT	Phenomena Identification and Ranking Table
PRIME	GHNEA Fuel Thermal Mechanical Code
RAI	Request for Additional Information
RCF	Rated Core Flow
RCS	Reactor Coolant System
RELAP	Idaho National Laboratory Transient Analysis Code
RETRAN	Electric Power Research Institute Transient Analysis Code
RFCF	Recirculation Flow Controller Failure

Acronym	Definition
RG	Regulatory Guide
RMS	Root Mean Square
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
SAFDL	Specified Acceptable Fuel Design Limit
SAFER	GE Loss of Coolant Accident Analysis Code
SBWR	Simplified Boiling Water Reactor
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SPERT	Special Power Reactor Test
SRLR	Supplemental Reload Licensing Report
SRP	Standard Review Plan
SRV	Safety Relief Valve
TEE	TRACG component
TGBLA	Toshiba-General Electric Bundle Lattice Analysis Code
TIP	Traversing In-core Probe
TLTA	Two Loop Test Apparatus
T-M	Thermal-Mechanical
TMI	Three Mile Island
TOP	Thermal Overpower
TRACG	GEH Transient Analysis Code
TS	Technical Specifications
TT	Turbine Trip
TTNB	Turbine Trip with No Bypass
TVAPS	Transient Varying Axial Power Shape
UB	Under Burn
VB	Vacuum Breaker
VS	Vierow-Schrock
WW	Wet Well

ABSTRACT

This report addresses Anticipated Operational Occurrences (AOO) and Anticipated Transients Without SCRAM (ATWS) overpressure transients as a result of the transition from the TRACG02 / PANAC10 computer codes to the TRACG04 / PANAC11 computer codes. The TRACG04 / PANAC11 computer codes are the current GE state-of-the-art tools for 3D BWR core physics and reactor transient predictions.

This report demonstrates that this code transition is not an adverse methodology change with respect to the calculated transient behavior for AOO and ATWS overpressure transients. Because no inherent margins are being gained as part of this code transition, GEH plans to use both code streams (TRACG02 / PANAC10 and TRACG04 / PANAC11) on an as-needed basis going forward for AOO and ATWS overpressure transients.

ACRONYMS AND ABBREVIATIONS

3D	Three Dimensional
ABWR	Advanced Boiling Water Reactor
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrences
APRM	Average Power Range Monitor
ATWS	Anticipated Transient Without SCRAM
BWR	Boiling Water Reactor
BWR/*n*	GE BWR product line *n* (*n* can be 2, 3, 4, 5, or 6)
CFR	Code of Federal Regulations
CPR	Critical Power Ratio
ΔCPR	Delta (Change In) Critical Power Ratio
EOC	End-of-Cycle
ESBWR	Economic Simplified BWR
FWCF	Feed Water Controller Failure
GE	General Electric Company
GEH	GE Hitachi Nuclear Energy
GE*n*	GE fuel product line *n* (e.g., GE13, GE14)
GENE	GE Energy, Nuclear
GNF-A	Global Nuclear Fuels - America
ICPR	Initial Critical Power Ratio
K	Kelvin
L8	Level 8
LFWH	Loss of Feed Water Heating

LOCA	Loss of Coolant Accident
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
M/G	Motor Generator
MSIV	Main Steam Isolation Valve
MSIVF	Main Steam Isolation Valve Closure - Flux SCRAM
MWt	Mega-Watt Thermal
Pa	Pascal
PANAC	Three-Dimensional BWR Core Steady State Simulator Code
PB	Peach Bottom
PIRT	Phenomena Identification and Ranking Table
psi	Pounds per Square Inch
RFCF	Recirculation Flow Control Failure
SCRAM	Reactor Trip
sec	Second (Also s)
SRV	Relief Valve
TRAC	Transient Reactor Analysis Code
TRACG	GE Version of TRAC
TSV	Turbine Stop Valves
TT	Turbine Trip
TTNB	Turbine Trip No Bypass
USNRC	United States Nuclear Regulatory Commission

CONTENTS

LIST OF FIGURES

Figure **Page**

LIST OF TABLES

Table **Page**

REVISION SUMMARY

Revision Number	Page(s)	Description of Change
0	--	Initial Issue
1	--	Modified Proprietary markups consistent with the GEH current guidelines.
1	--	Updated Cover Page consistent with Publication Handbook
1	ii	Updated Back of cover page consistent with updated Proprietary Notices per the Regulatory Affairs Web Page
1	iv	Acronyms and Abbreviations – Added GEH – GE Hitachi Nuclear Energy
1	5-1	Fixed formatting of Figure 5-1
1	7-1 through 7-4	Replaced Section 7.1.1 and Figures 7-1 through 7-6 per the MFN 07-445 Supplement 3.
1	8-2 through 8-44	Formatted Figures 8-1 through 8-55 for correct proprietary marking
1	10-1	Updated References 1and 3 to provide accepted (-A) version of Licensing Topical Reports. Combined Reference 10 and 2 to provide -A version of Licensing Topical Report. Added new Reference 10 from Section 7.1.1

1.0 INTRODUCTION

GE Hitachi Nuclear Energy (GEH) is in the process of migrating from the code stream of TRACG02 / PANAC10 to TRACG04 / PANAC11. The TRACG04 / PANAC11 computer codes are the current GEH state-of-the-art tools for 3D BWR core physics and reactor transient predictions.

This report addresses Anticipated Operational Occurrences (AOO) and Anticipated Transients Without SCRAM (ATWS) overpressure transients as a result of the transition from the TRACG02 / PANAC10 computer codes to the TRACG04 / PANAC11 computer codes. It should be noted that [1] is a more recent revision of the approved topical report for AOO using TRACG, and [1] supplies new TRACG calculations that incorporate a correction to the void coefficient model calculations internal to TRACG.

This report demonstrates that the change in computer codes to TRACG04 / PANAC11 is not an adverse methodology change with respect to the previously approved methods for AOO and ATWS overpressure transients using TRACG02 / PANAC10. Numerous transient calculations are presented in a comparative fashion to illustrate the specific sensitivity of the transient results to the changes being implemented. Upon approval by the USNRC of the use of TRACG04 / PANAC11 for simulation of AOO and ATWS overpressure transients, GEH will use TRACG04 / PANAC11 in addition to TRACG02 / PANAC10 for future design analyses.

The general report format used in [1] will be re-used here in a congruent fashion to highlight the differences that require further attention. This should facilitate the review and approval of this document by the USNRC.

2.0 LICENSING REQUIREMENTS AND SCOPE OF APPLICATION

All of the same licensing requirements that applied in [1] will remain applicable for the new code series of PANAC11 / TRACG04. With respect to future updates to the TRACG code beyond TRACG04, the same requirements will apply. The discussion pertaining to updated steady state nuclear methods beyond PANAC11 applies here also. The AOO scenario and nuclear power plant selection specifications still apply to this new code stream.

3.0 PHENOMENA IDENTIFICATION, RANKING

There are no changes to the PIRT tables as a result of the change in code versions. All of the relative importance of specific phenomena remains intact. There are no new phenomena being introduced, and the ranking of phenomena importance remains the same. The data presented in Section 3.0 of [1] still applies here.

4.0 APPLICABILITY OF TRACG TO AOOS

There are no changes to the BWR phenomena and TRACG model capability matrix tables as a result of the change in code versions. A number of the TRACG models have been upgraded to improve or add additional capabilities to TRACG. The main enhancement, which affects the applicability of TRACG to AOO transient and ATWS overpressure analyses, is the implementation of the PANAC11 kinetics. The other enhancements were implemented primarily to extend the applicability of TRACG beyond AOO transient and ATWS overpressure events to applications such as LOCA, ATWS with boron injection and ESBWR applications. These enhancements are summarized in Section 4.1. The same goes for the Qualification assessment matrix tables. Because TRACG04 produces results that are similar in nature to those produced by TRACG02, it can be concluded that data contained in the qualification tables presented in [1] remains applicable for TRACG04 for the application to AOO transient and ATWS overpressure analyses. The data presented in Section 4.0 of [1] still applies here.

4.1 Enhancements to TRACG

The primary enhancement to TRACG for application to AOO transient and ATWS overpressure analyses is the implementation of the PANAC11 kinetics model. The additional enhancements to TRACG expand the scope of TRACG to include the ESBWR in addition to all operating BWRs. Thus, the applicability of TRACG includes BWR/2-6, ABWR, and ESBWR. Finally, TRACG04 includes a number of new models and upgrades to several existing models in order to improve the application of TRACG to LOCA and ATWS. The major new models are:

- Replace the existing PANAC10 kinetics model with the PANAC11 kinetics model [5]. The effect of the PANAC11 kinetics on AOO transient analyses and ATWS overpressure events is evaluated in Section 8.0 of this report.

- The ANS decay heat model [6 and 7]. The ANS decay heat model is implemented as an optional model in addition to the existing May-Witt model. The ANS model improves the simulation of the effect of exposure on the decay heat and was implemented primarily for applications to LOCA. The ANS decay heat model has a negligible effect on AOO transient and ATWS overpressure analyses, where decay heat variations are insignificant compared to the fission power.

- Implement the quench front model for fuel rods and channel box. The quench front model was not activated in the previous version of TRACG. The model has been activated for application to LOCA, where quench front controlled rewetting is important for the calculation of the peak cladding temperature. The quench front model has no effect on AOO transient and ATWS overpressure analyses, where peak cladding temperatures are not calculated.

- A hot rod model for the fuel channel component. The one-dimensional hydraulic model in the TRACG channel component does not simulate the cross sectional variation in void fraction and steam superheat that can exist in a fuel bundle prior to reflooding and quenching during a LOCA. The hot rod model is implemented to capture the effect of cross sectional variations on the peak cladding temperatures. This model has no effect on AOO transient and ATWS overpressure analyses, because the peak cladding temperature is not calculated for AOO transient and ATWS overpressure events.

- The Shumway model for the minimum stable film boiling temperature [8]. The Shumway model is implemented as an optional enhancement to the minimum film boiling temperature correlation. This model primarily effects the rewetting during the reflood phase of a LOCA. The Shumway model has no effect on AOO transient and ATWS overpressure analyses, because the peak cladding temperature is not calculated for these events.

- Enhancement to the entrainment model to give better agreement with data. The models for the interfacial shear in the previous version of TRACG had primarily been qualified for pressure ranges applicable to normal operating conditions and AOO transient and ATWS overpressure analyses. Additional qualification for low pressure was performed to support the expansion of the application of TRACG to LOCA. Minor enhancements to the entrainment model were introduced to improve the application of TRACG at lower pressures. The enhancements affect the onset of entrainment and primarily the calculation of entrainment when some surfaces (e.g., fuel rods in a channel component) have experienced boiling transition. The enhancement to the entrainment model has a negligible effect on AOO transient and ATWS overpressure analyses.

- Enhancement to the flow regime map to give better void fraction predictions for low pressure. The models for the flow regime transitions in the previous version of TRACG had primarily been qualified for pressure ranges applicable to normal operating conditions and AOO transient and ATWS overpressure analyses. Additional qualification for low pressure was performed to support the expansion of the application of TRACG to LOCA. Minor enhancements to the model for transition to annular flow was introduced to improve the application of TRACG at lower pressures. The enhancement to the flow regime transition model has a negligible effect on AOO transient and ATWS overpressure analyses.

- Fuel rod conductivity consistent with PRIME [4]. The fuel conductivity from the PRIME model has been implemented as the default model in TRACG04, while the previous GESTR-based model has been retained as an optional model. The PRIME model improves the effect of temperature, exposure, and Gadolinium on the fuel thermal conductivity. This model does have an effect on the fuel temperature, but has a negligible effect on the hydraulic response. The effect on AOO transient and ATWS overpressure analyses, such as the effect on pressure response and CPR margin, is negligible.

- Models for the uncertainty in fuel rod internal pressure, the cladding yield stress, and the cladding rupture stress. These models were implemented for use in the statistical analysis of a LOCA and are not used for AOO transient and ATWS overpressure analyses. Therefore, these models do not affect the AOO transient and ATWS overpressure analyses.

- Modify the Zircaloy oxidation rate to be consistent with the latest version of the Cathcart & Pawel correlation [9]. This has no effect on AOO transient and ATWS overpressure analyses, as boiling tansition and high fuel temperatures with Zircaloy oxidation do not occur.

- Enhanced default pump homologous curves. The default pump homologous curves, which were based on data from the Semiscale test facility, have been supplemented with curves representative for large pumps. This has no effect on TRACG applications to AOO transient and ATWS overpressure events, because the pump homologous curves are required by procedure to be provided as input (e.g., not left at default values).

- Improved free convection heat transfer. The McAdams correlation correlation for free convection at a liquid surface has been implemented in addition to the current model that was based on Holman. However, the sensitivity of free convection heat transfer on AOO transient and ATWS overpressure events is insignificant.

- Improved condensation heat transfer. The default correlation for condensation heat transfer in the pressence of noncondensibles was changed from the Vierow-Schrock to the Kuhn-Schrock-Peterson correlation. This has an effect on ESBWR applications and has a negligible effect on AOO transient and ATWS overpressure events.

- Optional 6-cell jet pump. One-nozzle jet pumps have a relaively long straight section between the suction inlet and the diffuser. In the standard 5-cell jet pump, a single cell is used for this region. An option to subdivide this region into two cells has been implemented, primarily to improve the accuracy of the calculation of the void profile and static head in the jet pump for low flow two-phase flow conditions such as during the refill/reflood phase of a LOCA. The effect of the nodalization change for single-phase conditions such as during AOO transient and ATWS overpressure events is insignificant.

- Improved boron model. The models for solubility of sodiumpentaborate and the B_{10} absorption cross section have been improved to give better agreement with available data. This has an effect on ATWS events with activation of the standby liquid control system, but has no effect on AOO transient and ATWS overpressure events.

Of these changes, only the PANAC11 kinetics implementation has any significant effect on the previously approved applications for AOO transient and ATWS overpressure events for TRACG [1 and 10]. The remaining changes primarily affect and improve the applications of TRACG for LOCA and ATWS with application of the standby liquid control system. A detailed description of these new models and model enhancements is included in Revision 3 to the TRACG Model Description LTR [4]. Additional discussion of the effect of these model changes is included in Section 8.0.

5.0 MODEL UNCERTAINTIES AND BIASES

Overall model biases and uncertainties for a particular application are assessed for each high and medium ranked phenomena by using a combination of comparisons of calculated results to: (1) separate effects test facility data, (2) integral test facility test data, (3) component qualification test data and (4) BWR plant data. Where data is not available, cross-code comparisons or engineering judgment are used to obtain approximations for the biases and uncertainties. For some phenomena that have little effect on the calculated results, it is appropriate to simply use a nominal value or to conservatively estimate the bias and uncertainty.

The phenomena for BWR AOO transients have already been identified and ranked, as indicated in Section 3.0 of [1]. For the high and medium ranked phenomena, the bases used to establish the nominal value, bias and uncertainty for that parameter are documented in Section 5.1 of [1]. Also, the basis for the selection of the probability density function used to model the uncertainty is provided in Section 5.1 of [1].

5.1 Model Parameters and Uncertainties

This section in [1] discusses the uncertainties associated with each item from Table 3-1 from [1] that has been identified as having an effect on one or more critical safety parameters. Only the void coefficient (C1AX) uncertainty has undergone a significant change due to the change from PANAC10 to PANAC11 kinetics and will be discussed in this section. There are other changes that were implemented into TRACG04 for application to LOCA and ESBWR analyses. However, as discussed in Section 4.0 and demonstrated in Section 8.2, these changes have negligible effect on AOO transient and ATWS overpressure transient analyses. Therefore the model uncertainties for all parameters except the void coefficient have been retained from TRACG02 as described in [1].

C1AX Void Coefficient, H

TRACG04 uses a 3-D neutron kinetics model based on the PANAC11 neutronics parameters. The nodal reactivity is calculated [[ ]]. All of these parameters are correlated in terms of the moderator density. The infinite multiplication factor is also dependent on [[ ]] moderator density and nodal exposure.

The biases and uncertainties in void coefficient as determined from the PANAC11 models are predominantly due to biases and uncertainties in the infinite lattice eigenvalues (k_∞) calculated by the TGBLA06 lattice physics code. Values of k_∞ at [[ ]] points were calculated for a representative set of [[ ]] lattices at [[ ]] different exposures for in-channel voids of [[ ]] using both TGBLA06 and MCNP. The results for each lattice and exposure were fit to a [[ ]] function to determine k_∞ as a function of voids. These functional forms were extrapolated to obtain [[ ]] values of k_∞ corresponding to 100% in-channel voids. The void coefficients at a total of [[ ]] points were defined separately for TGBLA06 and MCNP by evaluating the derivative of k_∞ [[ ]]. Biases and uncertainties in TGBLA06 void coefficients were evaluated by performing [[ ]] comparisons between TGBLA06 and the corresponding MCNP benchmark values. These assessments were made using uncontrolled lattices (lattices without a control blade). An earlier independent set of [[ ]] other TGBLA04 lattices all at zero exposure were evaluated [[ ]] as a check on the process. The check set using TGBLA04 comparisons to MCNP included [[ ]] controlled lattices to confirm that the uncontrolled lattices bound the biases and uncertainties for the controlled lattices. Because of the similarity in the TGBLA04 and TGBLA06 comparisons, the comparisons based on TGBLA06 using uncontrolled lattices are also expected to bound the biases and uncertainties for the controlled lattices.

The set of [[ ]] points was reduced to [[ ]] by eliminating [[ ]] outliers outside the ±2.17 sigma range. The remaining [[ ]] were used to correlate the biases and uncertainties in the void coefficient as a function [[ ]] in order to obtain response surfaces that are modeled in TRACG04. The fraction of the total water volume that is inside the channel box excluding the water rods is given by "*g*". A typical value is g = [[ ]]. For values of ρ_ℓ and ρ_g representative of operating pressures, and for conditions where the void fraction remains zero in the water rods and bypass, a typical value for the relative water density averaged for the lattice is related to the in-channel void fraction [[ ]]. Curves of the percentage biases and standard deviations for void coefficient are shown in Figure 5-1 for the different exposures that were considered in developing the response surfaces.

[[

]]

Figure 5-1: Void Coefficient %Bias and %Standard Deviation

Note: The parametric curves have the units of GWd/ST for each exposure point.

The void coefficient biases and uncertainties are implemented in TRACG04 calculations [[

]]. Consider a representative in-channel void fraction of 40% and a core-average exposure of 15 GWd/ST. For $\alpha = 0.4$, Figure 5-1 indicates that the bias is around [[ ]]. The standard deviation from Figure 5-1 is [[ ]] at this condition. For low exposures, the uncertainties tend to be [[ ]]. As the poison is burned and the bundles approach their peak reactivity and power, the void coefficient bias and uncertainty [[ ]].

TRACG04 internally models the response surfaces for the void coefficient biases and uncertainties in order to account for the known dominant dependencies due to relative moderator density and exposure [[ ]]. Cross sections are generated within TRACG04 using data from the lattice physics code that gets passed through via the PANAC11 wrap-up. Thus, the lattices are explicitly modeled. [[

]]. Thus, the normality of the [[ ]] residual errors can be tested at each of these locations. This is what was done to get the P-values presented in Table 5-1. All the P-values except for one are significantly larger than the 0.05 threshold required to confirm normality and reach the conclusion that it is appropriate to assume that the residual errors are random [[ ]]. The single set of [[ ]] points that fails the normality test produces a low P-value because the sample distribution is more centrally concentrated than what is expected for a normal distribution; therefore, it is conservative to model the sample distribution using an assumed normal distribution because that will predict wider scatter than the sample indicates.

TRACG04 input has been structured to allow the internally calculated uncertainties to be correlated [[

]]. For most fast pressurization events, the impact of not modeling the void coefficient biases is on the order of [[ ]] in calculated values of transient ΔCPR/ICPR. Whether the bias is conservative or not depends on the exposure distribution and the relative water density distribution in the core.

For sensitivity studies, a core-wide bias and uncertainty in void coefficient can be specified through the TRACG04 input. As an example of the importance of the void coefficient uncertainty, consider that for a typical BWR/4 plant an [[ ]] variation in the void coefficient when applied to all nodes in the core corresponds to a sensitivity of [[ ]] in the ΔCPR/ICPR for a turbine trip without bypass.

Table 5-1: Normality Test P-values for the Void Coefficient Residual Errors

Void → **Exp ↓**	**[[**			**]]**	**Avg**	**Stdev**	**Min**
[[							
							]]
Avg	[[						
Stdev							
Min							]]

6.0 APPLICATION UNCERTAINTIES AND BIASES

The descriptions for input, initial conditions, and plant parameters in Section 6.0 of [1] remain applicable for PANAC11 / TRACG04. As a result, no new data is presented here.

7.0 COMBINATION OF UNCERTAINTIES

The change in code streams from PANAC10 / TRACG02 to PANAC11 / TRACG04 does not affect the existing statistical methodology. As a result, the method for statistical combination of uncertainties remains unchanged from that presented in [1].

7.1 Statistical Analysis for Qualification Events

Because the data presented in Section 7.6 of [1] was produced using PANAC10 / TRACG02, a new comparison with the Peach Bottom turbine trip tests using PANAC11 / TRACG04 will be presented here to demonstrate the relative effect of the new code versions on this data comparison.

7.1.1 Peach Bottom Turbine Trip Comparison

The "TRACG Nominal - Revision 2" data from [1] has changed slightly [[

]]. These same inputs are reflected in both the TRACG02 and TRACG04 calculated results that are presented here in order to provide an equivalent basis for comparing the two codes. Note that the TRACG04 results are also the same as those contained in [10]. The 2-sigma error bands from the earlier TRACG02 calculations in [1] are shown here superimposed on the updated TRACG02 results to provide some indication of the level of significance for the differences in the TRACG02 and TRACG04 calculated results. See Figure 7-1 through Figure 7-6 for the results comparisons for each of the three Peach Bottom tests.

[[

]]

Figure 7-1: PB TT Test 1 Power Response

[[

]]

Figure 7-2: PB TT Test 1 Pressure Response

[[

]]

Figure 7-3: PB TT Test 2 Power Response

[[

]]

Figure 7-4: PB TT Test 2 Pressure Response

[[

]]

Figure 7-5: PB TT Test 3 Power Response

[[

]]

Figure 7-6: PB TT Test 3 Pressure Response

The pressure responses are quite similar between TRACG02 and TRACG04. Both codes tend to slightly under predict the longer term pressure responses for the lower power TT1 and TT2 tests. With respect to the power results, both codes conservatively over predict the response for the lowest power TT1 test but do an excellent job of predicting the measured power for the higher power TT2 and TT3 tests. The results show that TRACG04 is capable of accurately modeling the Peach Bottom turbine trip test data. For these comparisons to test data, the TRACG04 code produces essentially the same calculated results as TRACG02.

8.0 DEMONSTRATION ANALYSIS

As was presented in [1] and [2], numerous AOO and ATWS pressurization transients were simulated to demonstrate the capabilities of the TRACG code to accurately predict transient behavior. Because the computer codes used to calculate these demonstration analyses have been modified, comparisons are provided here to illustrate the effect of these computer code changes.

8.1 Baseline Analysis

Six transients (5 AOO and 1 ATWS) are recalculated using both the old code stream and the new code stream to highlight the relative effect of the transition.

This new set of calculations is for a different BWR/4 plant than was used in [1]. The specific plant selected is adequate for comparison purposes. In [1] only a quarter core was represented, and symmetry was assumed. For the new plant selected, a full-core model was developed using more recent fuel types (GE13 9x9 and GE14 10x10) and a higher core power rating (2923 MWt). This plant selection should better illustrate the effect derived from use of the latest fuel types and extended power uprate conditions.

The same general vessel modeling technique as is shown in Figure 8-1 of [1] is used here. Figure 8-1 through Figure 8-4 depict the channel groupings selected for both TRACG02 and TRACG04 calculations and for both initial power levels. The hot assembly channel (GE14 fuel) is highlighted in red for each.

[[

]]

Figure 8-1: 100% Power TRACG02 Channels

[[

]]

Figure 8-2: 100% Power TRACG04 Channels

[[

]]

Figure 8-3: 53.3% Power TRACG02 Channels

[[

]]

Figure 8-4: 53.3% Power TRACG04 Channels

For the TRACG02 channel groupings, channel groups 20 through 24 are GE13 fuel with the remaining channel groups being GE14 fuel. For the TRACG04 channel groupings, channel groups 26 through 30 are GE13 fuel with the remaining channel groups being GE14 fuel.

Given the nature of the changes in the code, the following events are selected to be compared using both TRACG02 and TRACG04.

1. Pressurization: turbine trip without bypass [TTNB], feed water controller failure [FWCF], and main steam line isolation valve closure with the backup (flux) SCRAM [MSIVF]

2. Core flow transient: recirculation flow controller failure [RFCF]

3. Cold water transient: loss of feed water heating [LFWH]

4. ATWS pressurization transient: main steam line isolation valve closure without SCRAM [MSIV ATWS]

8.1.1 Baseline Analysis of Pressurization Transients

8.1.1.1 Turbine Trip No Bypass (TTNB)

The TTNB event is characterized by the fast closure of the turbine stop valves (TSV). The sudden closure of the stop valves causes a rapid pressurization of the steam lines and reactor vessel, resulting in a rapid power excursion. The event is heightened by the assumed failure of the pressure relief function provided by the turbine bypass valves. The turbine stop valve position switches initiate a reactor SCRAM. Power is mitigated with the help of negative reactivity due to the SCRAM and due to void production as the heat flux rises. The safety/relief valves actuate as the steamline pressure rises to the setpoint. This action limits the pressure rise. The event is modeled at 100% power and 104.6% flow with an EOC nominal power shape. The key parameters for both code streams are presented in Figure 8-5 through Figure 8-12 and Table 8-1.

[[

]]

Figure 8-5: TTNB Power

[[

]]

Figure 8-6: TTNB Feed Water Flow

[[

]]

Figure 8-7: TTNB Core Flow

[[

]]

Figure 8-8: TTNB Inlet Subcooling

[[

]]

Figure 8-9: TTNB Dome Pressure Increase

[[

]]

Figure 8-10: TTNB SRV Flow

[[

]]

Figure 8-11: TTNB Vessel Flow

[[

]]

Figure 8-12: TTNB ΔCPR / ICPR

Table 8-1: TTNB Key Transient Parameters

Trip Time Summary (sec)	**TRACG02**	**TRACG04**
Turbine Trip	[[	
Reactor SCRAM initiated on TSV position		
Safety/relief valves start to open		]]
Initial Conditions	**TRACG02**	**TRACG04**
Core power (%)	[[	
Core flow (%)		
Dome pressure (Pa)		
Core Inlet Temperature (K)		]]
Key Transient Parameters	**TRACG02**	**TRACG04**
Peak power (%) and time of max. (sec)	[[	
Maximum core flow (%) and time of max. (sec)		
Maximum dome pressure (Pa) and time of max. (sec)		
Maximum vessel bottom pressure (Pa) and time of max. (sec)		]]
CPR Summary	**TRACG02**	**TRACG04**
Hot Channel ICPR	[[	
Hot Channel MCPR		
Hot Channel ΔCPR/ICPR		]]

8.1.1.2 Feed Water Controller Failure (FWCF)

The FWCF event is characterized by the feedwater flow controller failing to the maximum demand value. This causes an increase in the feedwater flow. The water level rises until the high level trip setpoint (L8) is reached. When L8 is reached, a high level turbine trip is initiated, the feedwater pumps are tripped off, and a reactor SCRAM is initiated. The turbine trip causes a rapid pressurization event that results in a power excursion similar to the TTNB. Power is mitigated with the help of negative reactivity due to the SCRAM and due to void production as the heat flux rises. The safety/relief valves actuate as the steamline pressure rises to the setpoint. This action limits the pressure rise. The event is modeled at 100% power and 104.6% flow with an EOC nominal power shape. The key parameters are presented in Figure 8-13 through Figure 8-20 and Table 8-2 for the FWCF event.

[[

]]

Figure 8-13: FWCF Power

[[

]]

Figure 8-14: FWCF Feed Water Flow

[[

]]

Figure 8-15: FWCF Core Flow

[[

]]

Figure 8-16: FWCF Inlet Subcooling

[[

]]

Figure 8-17: FWCF Dome Pressure Increase

[[

]]

Figure 8-18: FWCF SRV Flow

[[

]]

Figure 8-19: FWCF Vessel Flow

[[

]]

Figure 8-20: FWCF ΔCPR / ICPR

Table 8-2: FWCF Key Transient Parameters

Trip Time Summary (sec)	**TRACG02**	**TRACG04**
Feed water controller at maximum demand	[[	
Level 8 turbine trip and feed water pump trip		
Reactor SCRAM initiated on TSV position		
Safety/relief valves start to open		]]
Initial Conditions	**TRACG02**	**TRACG04**
Core power (%)	[[	
Core flow (%)		
Dome pressure (Pa)		
Core Inlet Temperature (K)		]]
Key Transient Parameters	**TRACG02**	**TRACG04**
Peak power (%) and time of max. (sec)	[[	
Maximum core flow (%) and time of max. (sec)		
Maximum dome pressure (Pa) and time of max. (sec)		
Maximum vessel bottom pressure (Pa) and time of max. (sec)		]]
CPR Summary	**TRACG02**	**TRACG04**
Hot Channel ICPR	[[	
Hot Channel MCPR		
Hot Channel ΔCPR/ICPR		]]

8.1.1.3 MSIV Closure Flux SCRAM (MSIVF)

The MSIV closure is characterized by closure of the main steam isolation valves. The closure causes a rapid pressurization event that leads to a power excursion. The reactor SCRAM is conservatively assumed to occur on high flux rather than the earlier isolation valve position. Power is mitigated with the help of negative reactivity due to the SCRAM and due to void production as the heat flux rises. The safety/relief valves actuate as the steamline pressure rises to the setpoint. This action limits the pressure rise. This is the limiting event for vessel overpressure protection. The primary output is peak pressure response. The event is modeled at 100% power and 104.6% flow with an EOC nominal power shape. The key parameters are presented in Figure 8-21 through Figure 8-27 and Table 8-3 for the MSIVF event.

[[

]]

Figure 8-21: MSIVF Power

[[

]]

Figure 8-22: MSIVF Feed Water Flow

[[

]]

Figure 8-23: MSIVF Core Flow

[[

]]

Figure 8-24: MSIVF Inlet Subcooling

[[

]]

Figure 8-25: MSIVF Dome Pressure Increase

[[

]]

Figure 8-26: MSIVF SRV Flow

[[

]]

Figure 8-27: MSIVF Vessel Flow

Table 8-3: MSIVF Key Transient Parameters

Trip Time Summary (sec)	**TRACG02**	**TRACG04**
Initiate MSIV closure	[[	
Reactor SCRAM initiated on high APRM flux		
Safety/relief valves start to open		
MSIV fully closed		]]
Initial Conditions	**TRACG02**	**TRACG04**
Core power (%)	[[	
Core flow (%)		
Dome pressure (Pa)		
Vessel bottom pressure (Pa)		
Core Inlet Temperature (K)		]]
Key Transient Parameters	**TRACG02**	**TRACG04**
Peak power (%) and time of max. (sec)	[[	
Maximum core flow (%) and time of max. (sec)		
Maximum dome pressure (Pa) and time of max. (sec)		
Maximum vessel bottom pressure (Pa) and time of max. (sec)		]]

8.1.2 Baseline Analysis of a Core Flow Transient

8.1.2.1 Recirculation Flow Controller Failure (RFCF)

The RFCF event is characterized by an upscale failure of the recirculation motor/generator speed controller in one loop. The B loop fluid coupler velocity is assumed to increase at a rate of 25%/sec. The pump speed increases to maximum in about 3 seconds. The APRM high neutron flux trip is assumed to be disabled so that an automatic scram is not initiated for this event. The event is modeled at 53.3% power and 36.1% flow at EOC conditions. The key parameters are presented in Figure 8-28 through Figure 8-35 and Table 8-4 for the RFCF event.

[[

]]

Figure 8-28: RFCF Power

[[

]]

Figure 8-29: RFCF Feed Water Flow

[[

]]

Figure 8-30: RFCF Core Flow

[[

]]

Figure 8-31: RFCF Pump B Coupler Position

[[

]]

Figure 8-32: RFCF Pump B Speed

[[

]]

Figure 8-33: RFCF Dome Pressure Increase

[[

]]

Figure 8-34: RFCF Vessel Flow

[[

]]

Figure 8-35: RFCF ΔCPR / ICPR

Table 8-4: RFCF Key Transient Parameters

Trip Time Summary (sec)	**TRACG02**	**TRACG04**
Initiate M/G controller failure	[[	
M/G coupler at maximum position		]]
Initial Conditions	**TRACG02**	**TRACG04**
Core power (%)	[[	
Core flow (%)		
Dome pressure (Pa)		
Core Inlet Temperature (K)		]]
Key Transient Parameters	**TRACG02**	**TRACG04**
Peak power (%) and time of max. (sec)	[[	
Maximum core flow (%) and time of max. (sec)		]]
CPR Summary	**TRACG02**	**TRACG04**
Hot Channel ICPR	[[	
Hot Channel MCPR		
Hot Channel ΔCPR/ICPR		]]

8.1.3 Baseline Analysis of a Cold Water Transient

8.1.3.1 Loss of Feed Water Heating (LFWH)

The LFWH event is characterized by the reduction in core inlet subcooling caused by a reduction in feedwater heating. The increase of inlet subcooling increases moderation and causes an increase in power. An automatic reactor scram does not occur for this event. The event assumes a 30-second feedwater heater time constant. The event is modeled at 100% power and 104.6% flow with an EOC nominal power shape. The key parameters are presented in Figure 8-36 through Figure 8-42 and Table 8-5 for the LFWH event.

[[

]]

Figure 8-36: LFWH Power

[[

]]

Figure 8-37: LFWH Feedwater Flow

[[

]]

Figure 8-38: LFWH Core Flow

[[

]]

Figure 8-39: LFWH Inlet Subcooling

[[

]]

Figure 8-40: LFWH Dome Pressure Increase

[[

]]

Figure 8-41: LFWH Vessel Flow

[[

]]

Figure 8-42: LFWH ΔCPR / ICPR

Table 8-5: LFWH Key Transient Parameters

Trip Time Summary (sec)	**TRACG02**	**TRACG04**
Loss of feed water heating	[[	]]
Initial Conditions	**TRACG02**	**TRACG04**
Core power (%)	[[	
Core flow (%)		
Dome pressure (Pa)		
Feed water temperature (K)		
Core inlet temperature (K)		]]
Key Transient Parameters	**TRACG02**	**TRACG04**
Peak power (%) and time of max. (sec)	[[	
Maximum core flow (%) and time of max. (sec)		
Maximum dome pressure (Pa) and time of max. (sec)		
Minimum feed water temperature (K) and time of min. (sec)		
Maximum core inlet subcooling (%) and time of max. (sec)		]]
CPR Summary	**TRACG02**	**TRACG04**
Hot Channel ICPR	[[	
Hot Channel MCPR		
Hot Channel ΔCPR/ICPR		]]

8.1.4 Baseline Analysis of an ATWS Pressurization Transient

8.1.4.1 MSIV Closure ATWS (MSIV ATWS)

The MSIV closure ATWS is characterized by closure of the main steam isolation valves. The closure causes a rapid pressurization event that leads to a power excursion. No reactor SCRAM is assumed to occur. Power is mitigated with the help of negative reactivity due to void production as the heat flux rises. The safety/relief valves actuate as the steamline pressure rises to the setpoint. This action limits the pressure rise. The primary output is peak pressure response. The event is modeled at 100% power and 104.6% flow with an EOC nominal power shape. The key parameters are presented in Figure 8-43 through Figure 8-49 and Table 8-6 for the MSIV ATWS event.

[[

]]

Figure 8-43: MSIV ATWS Power

[[

]]

Figure 8-44: MSIV ATWS Feed Water Flow

[[

]]

Figure 8-45: MSIV ATWS Core Flow

[[

]]

Figure 8-46: MSIV ATWS Inlet Subcooling

[[

]]

Figure 8-47: MSIV ATWS Dome Pressure Increase

[[

]]

Figure 8-48: MSIV ATWS SRV Flow

[[

]]

Figure 8-49: MSIV ATWS Vessel Flow

Table 8-6: MSIV ATWS Key Transient Parameters

Trip Time Summary (sec)	**TRACG02**	**TRACG04**
Initiate MSIV closure	[[	
Safety/relief valves start to open		
MSIV fully closed		]]
Initial Conditions	**TRACG02**	**TRACG04**
Core power (%)	[[	
Core flow (%)		
Dome pressure (Pa)		
Vessel bottom pressure (Pa)		
Core Inlet Temperature (K)		]]
Key Transient Parameters	**TRACG02**	**TRACG04**
Peak power (%) and time of max. (sec)	[[	
Maximum core flow (%) and time of max. (sec)		
Maximum dome pressure (Pa) and time of max. (sec)		
Maximum vessel bottom pressure (Pa) and time of max. (sec)		]]

8.2 Effect of Kinetics and Thermal Hydraulic Model Changes

The comparisons shown in Sections 7.1 and 8.1 all show that the power response is higher in magnitude with PANAC11 / TRACG04 than with PANAC10 / TRACG02. The initial vessel steam flow and pressure responses are generally very close as seen in Figure 8-9 and Figure 8-11. This would indicate that the thermal hydraulic response of TRACG04 is essentially the same as TRACG02. The propagation of the pressure wave through the steam line following the closure of the turbine stop valve is the same, and the initial pressurization of the reactor pressure vessel is the same. However, the response to this pressurization is higher in magnitude for PANAC11 / TRACG04 than for PANAC10 / TRACG02. Once this higher power response for PANAC11 / TRACG04 is propagated primarily through conductive and convective heat transfer to the coolant, the resulting increased vapor generation leads to a higher pressure for PANAC11 / TRACG04 than for PANAC10 / TRACG02. Similarly, the increased heat flux for PANAC11 / TRACG04 leads to an increased ΔCPR for PANAC11 / TRACG04 relative to PANAC10 / TRACG02. This would indicate that the main cause of the differences between PANAC11 / TRACG04 and PANAC10 / TRACG02 is the change from the PANAC10 to the PANAC11 kinetics and that the effect of the other changes as summarized in Section 4.1 is negligible.

To verify the above assumption, a comparison was made for the turbine trip with no bypass [Section 8.1.1.1] where the power responses were forced to be identical. The same power response versus time based on the TRACG04 calculation was used as input to both TRACG02 and TRACG04. The TRACG04 channel grouping and power distribution was also used for TRACG02. This way, the power versus time and the spatial power distribution were identical for TRACG02 and TRACG04. In addition, the power of the limiting CPR channel in the core was increased to yield an MCPR for the transient that was close to one. Note, the purpose of this comparison was to compare the thermal hydraulic response of TRACG04 and TRACG02 for the an identical power response. Therefore, the results in this section are not directly comparable to the results in Section 8.1.1.1.

An additional case is represented in the following comparisons by the green curves labeled TRACG04+. This case contains the 5-cell TRACG02 jet pump model, the Holman free convection heat transfer, the Vierow-Schrock condensation heat transfer, and the GESTR fuel conductivity models. As described in the following paragraphs, these models were turned back to the TRACG02 default to demonstrate the source of some of the very small differences shown in Figure 8-51 through Figure 8-55.

Figure 8-50 shows the power response for the TTNB comparison. It is seen that the power responses are the same, by design.

[[

]]

Figure 8-50: TTNB TRACG02 and TRACG04 Power

Figure 8-51 shows the pressure response. It is seen that the pressure responses are virtually identical until the opening of the safety relief valves (SRV). TRACG04 depressurizes slightly slower than TRACG02 following the opening of the SRV.

[[

]]

Figure 8-51: TTNB TRACG02 and TRACG04 Dome Pressure Increase

The SRV flows are shown in Figure 8-52. The small difference in SRV flow is a direct result of the small difference in pressure between TRACG04 and TRACG02. The difference in pressure occurs after the peak vessel pressure and after the minimum CPR.

[[

]]

Figure 8-52: TTNB TRACG02 and TRACG04 SRV Flow

Figure 8-53 shows the core flow. It is seen that the core flows are similar for TRACG04 and TRACG02, however there are small differences. TRACG04 used the 6-cell jet pump nodalization as the default while TRACG02 used the 5-cell nodalization. To evaluate the sensitivity to the jet pump nodalization, TRACG04 was also run with the same 5-cell nodalization as TRACG02. This case is represented by the green curves labeled TRACG04+ on the Figure 8-51 through Figure 8-55. The TRACG04 core flow response is nearly identical to that from TRACG02 when the same 5-cell nodalization is used for both cases, but the 6-cell model has an insignificant effect on the results. In addition, three other known differences were eliminated between TRACG04 and TRACG02 in the TRACG04+ calculation. The Holman free convection heat transfer, the Vierow-Schrock condensation heat transfer, and the GESTR fuel conductivity models were used in this TRACG04+ calculation as in the TRACG02 calculation.

[[

]]

Figure 8-53: TTNB TRACG02 and TRACG04 Core Flow

Figure 8-54 shows the vessel steam flow. The vessel steam flows are virtually the same for TRACG04 and TRACG02.

[[

]]

Figure 8-54: TTNB TRACG02 and TRACG04 Vessel Flow

Figure 8-55 shows the CPR response for the most limiting channel in the core. The CPR responses are very similar for TRACG04 and TRACG02. The difference in the minimum CPR is [[ ]]. When the 5-cell jet pump model and the same correlations for the free convection heat transfer, condensation heat transfer, and fuel conductivity are used in both calculations, the results are very close, and the difference in the minimum CPR is [[ ]]. In both cases the differences are small, less than [[ ]], which is a general threshold of significance for the minimum CPR. Similarly, the differences in the other parameters as shown in Figure 8-51 through Figure 8-55 are also negligibly small.

Due to the closeness of the thermal hydraulic response between TRACG04 and TRACG02 when the same power response is used, it can be concluded that the major differences between TRACG04 and TRACG02 are due to the difference between the PANAC10 and PANAC11 kinetics models.

[[

]]

Figure 8-55: TTNB TRACG02 and TRACG04 CPR

8.3 Conclusions

This report documents a comparison of TRACG04 and TRACG02 for AOO transient and ATWS overpressure events. The following observations and conclusions can be made from these comparisons:

- The major difference between TRACG04 and TRACG02 is due to the difference between the PANAC11 kinetics in TRACG04 and the PANAC10 kinetics in TRACG02.
- The other model improvements implemented into TRACG04, primarily to improve the applicability to LOCA and ATWS events and to improve some models significant for the ESBWR, have a negligible effect on AOO transient and ATWS overpressure analyses.
- TRACG04 generally produces more conservative results than TRACG02 when applied to AOO transient and ATWS overpressure events.
- Use of PANAC11 / TRACG04 for AOO transient and ATWS overpressure transients by GE going forward has been shown here to be acceptable.

9.0 TECHNICAL SPECIFICATION MODIFICATIONS

The text presented in Section 9.0 of [1] in relation to the Technical Specifications and Technical Specification Bases remains applicable for the new code stream of PANAC11 / TRACG04.

10.0 REFERENCES

[1] J. G. M. Andersen, et. al., *TRACG Application for Anticipated Operation Occurrences Transient Analysis*, NEDE-32906P-A, Revision 3, September 2006.

[2] F. T. Bolger and M. A. Holmes, "*TRACG Application for Anticipated Transients without Scram Overpressure Transient Analyses*," NEDE-32906P, Supplement 1-A, November 2003.

[3] MFN-115, Letter from G. B. Stramback (GE) to M. B. Fields (USNRC), *Transient CPR Calculation for TRACG (TRACG Application for Anticipated Operational Occurrences Transient Analyses, NEDE-32906P, Supplement 2-A, March*, 2006.

[4] J. G. M. Andersen, et. al., *TRACG Model Description*, NEDE-32176P, Revision 4, January 2008.

[5] Letter from S. A. Richards (NRC) to G. A. Watford (GE), *Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II" – Implementing Improved GE Steady-State Methods (TAC No. MA6481)*, FLN-1999-011, November 10, 1999.

[6] *American National Standard for Decay Heat Power in Light Water Reactors*, ANSI/ANS-5.1-1979.

[7] *American National Standard for Decay Heat Power in Light Water Reactors*, ANSI/ANS-5.1-1994.

[8] R. W. Shumway, *TRAC-BWR Heat Transfer: Assessment of Tmin*, EGG-RST-6781, October 1984.

[9] J. V. Cathcart and R. E. Pawel, *Zirconium Metal-Water Oxidation Kinetics: IV. Reaction Rate Studies*, ORNL/NUREG-17, August 1977.

[10] J. G. M. Andersen, et al., *TRACG Qualification*, NEDE-32177P, Revision 3, August 2007.

Appendix A: GEH RESPONSES TO NRC RAIs

RAI Table of Contents

NRC RAI 1

Do you intend to use TGBLA06-Modified as part of this application?

GEH Response:

The codes used to generate NEDE-32906P, Supplement 3 are consistent with the current NRC approved methodologies. In particular, the version of TGBLA06 applied does not contain the resonance modeling modification discussed in responses to NRC questions on the ESBWR docket [1] and extended operational ranges [2]. The modified TGBLA06 will be applied once the quality assurance procedures for the error correction are complete and the final determination of any impacts are assessed in accordance with 10 CFR 50.59 rules.

[1] Letter from George Stramback to US NRC Document Control Desk, " Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-2 through 4.2-7, 4.3-3, 4.3-4, 4.4-2, 4.4-5, 4.4-6, 4.4-15 through 4.4-17, 4.4-19, 4.4-24, 4.4-27, 4.4-31 through 4.4-34, 4.4-36, through 4.4-38, 4.4-42 through 4.4-50, 4.4-52 through 4.4-56, 4.8-1 through 4.8-16",MFN-06-297, August 23, 2006. (See response to RAI 4.3-3.)

[2] Letter from George Stramback to Herbert Berkow (NRC), "Responses to RAIs - Methods Interim Process (TAC No. MC5780)", MFN-05-022, May 31, 2005. (See response to RAI 3-1.)

NRC RAI 2

Provide a qualitative discussion on the differences seen in the transient analysis time traces between TRACG02/PANAC10 and TRACG04/PANAC11 in the thermal hydraulic parameters such as pressure, core flow, inlet subcooling, etc.

GEH Response

To some extent, this qualitative comparison has already been provided in a general sense in Section 8.2 on Page 8-37 for the case where the nuclear kinetics differences were removed from the equation. However, this general description of the calculated trends will be further detailed here describing the effects of the nuclear kinetics on the overall transient response comparisons.

The turbine trip no bypass (TTNB) calculation comparison found in Section 8.1.1.1 will be used here for illustrative purposes. The same trends can be observed when looking at the TRACG04/PANAC11 (T4/P11) calculations in comparison to the TRACG02/PANAC10 (T2/P10) calculations for the other pressurization events.

The pressure responses of T4/P11 and T2/P10 track in quite good agreement until the peak pressure is achieved at around [[ ]] (Figure 8-9). However, the power response (Figure 8-5) does not track quite so closely. The power trends are the same, but the magnitudes of the peaks are very different. For T4/P11, the first power peak is found to be [[ ]] at [[ ]], while the T2/P10 power peak is found to be [[ ]] at [[ ]].

This deviation in peak powers is a direct result of using the P11 kinetics rather than the P10 kinetics. For a similar pressure transient, the P11 kinetics produce a much more responsive neutronic feedback. With this increased nuclear feedback, the extra power is deposited in the fuel and subsequently the coolant and manifests in an internal energy increase in the bulk coolant thereby yielding higher transient CPR values (Figure 8-12) and higher system pressures (Figure 8-9) downstream of the peak powers for T4/P11.

As the pressures are higher in T4/P11, these higher pressures result in higher inlet subcooling (Figure 8-8) for a given fluid temperature, because the saturation temperature at the inlet of the core has been increased.

In the case of TTNB, the two transient peaks in ΔCPR/ICPR are found at roughly [[ ]]. Because T4/P11 results in higher integrated power responses in these time periods, the ΔCPR/ICPR values are found to be higher in both instances as compared to the results using T2/P10.

In general, the core flow responses (Figure 8-7) are quite similar between T4/P11 and T2/P10. (See also the discussion on Page 8-41 and Figure 8-53.) Core flow is generally more controlled by the forced flow from the recirculation system and less by the more responsive P11 neutronic feedback. The feedwater flow (Figure 8-6), on the other hand, is more dynamic in its response. As the pressure in the vessel increases, the feedwater system is less capable of delivering flow at this higher backpressure. Additionally, the feedwater flow is modulated based upon the transient level response, which in turn is dependent upon the pressure in the vessel. As the pressure responses diverge between T4/P11 and T2/P10, the feedwater flow will potentially deviate in a manner consistent with the transient pressure and level differences.

<u>NRC RAI 3</u>

(RAI 21.6-78 on the Economic Simplified Boiling Water Reactor (ESBWR) DCD Docket)

On Page 7-47 of NEDE-32176P, Revision 3 (Reference 1), you state: "Two options exist for the calculation of the CPR [critical power ratio] for transient conditions." Why do you have two options for calculation of transient CPR? Is one method more conservative than the other? What are your guidelines for when to use which method for transient CPR calculations? Which method is used during an anticipated operational occurrence (AOO) calculation and during an anticipated transient without scram (ATWS) calculation? On Page 7-48 of the same document you state: "The assessment of the critical power calculation can be found in Section 3.6 of the TRACG Qualification LTR." The NRC staff has not received Revision 3 of the TRACG Qualification LTR which you state was to be published in June 2006. Provide the information from this document that may answer the above questions on the CPR calculation options for transient conditions.

GEH Response

Two options exist for the calculation of the transient CPR response in TRACG. In the first option the transient CPR are calculated using the traditional [[ ]]. In the second option the transient CPR is calculated by performing an [[ ]] in the calculation. The second method is [[ ]], but is also more compute intensive. The two transient CPR methods are both approved and are described in detail in the approved LTR supplement "TRACG Application for Anticipated Operational Occurrences Transient Analyses", NEDE-32906P Supplement 2-A, March 2006.

NRC RAI 4

(RAI 21.6-80 on the ESBWR Docket) The variable f in Eq. 9.3-2 in NEDE-32176P, Revision 3, is described as the sum of the five decay heat group fractions, fk. However, in the preceding paragraph you state that TRACG04 allowsfor a variable number (Nd) of decay heat groups. Please update your documentation to reflect this change.

GEH Response

Answered on the ESBWR DCD Docket, GEH letter MFN 07-352

NRC RAI 5

(RAI 21.6-81 on the ESBWR Docket) Please address the following questions related to distribution of channel power:

a. Equation 9.4-11 in NEDE-32176P, Rev. 3, includes Fco, which is the fraction of direct moderator heating that appears in the coolant in the bypass, water rod, and bundle coolant. In TRACG, the water rod coolant, the core bypass coolant, and the bundle coolant are simulated as separate flow paths. How is the direct moderator heating associated with Fco split up for these three different coolant regions within the BWR core? Please describe the basis of the model.

b. Page 62 of NEDC-32965P, Rev. 0 (UM-0149, Rev. 0), describes the user input fractions for fission power and decay heat for direct moderator heating, fuel clad gamma heating and water rod(s) clad gamma heating as described in NEDC-32176, Rev. 3, page 9-35. The description for FDMN2 (direct moderator heating fraction for decay heat power) states "The prior practice of setting FDMH2=FDMH1 is discouraged since it is non-conservative with respect to post-scram evaluations of peak clad temperature." Where FDMH1 is the direct moderator heating fraction for fission power. Please explain why you have set FDMH1=FDMH2 for all of the CHANs in the ESBWR TRACG decks for LOCA, AOO, ATWS and Stability given this statement in the user's guide.

c. [[

]]

d. How does the direct moderator heating model change based on the control fraction for a given CHAN component? How specifically is the user input for BPAPC (bypass area per channel) used in the direct moderator heating model?

e. The fission power distribution model presented in section 9.4 in NEDE-32176P, Rev. 3, appears to assume no gamma heat of the pressure vessel walls. Explain how gamma heating of the pressure vessel walls is considered.

f. In Equation 9.4-13 of NEDE-32176P, Rev. 3, a and b are assumed constant for calculating the fractional deposition of fission power in the fuel clad, water rod clad, control blades, and channel wall. [[

]]

g. What is the normalization formula used to normalize Eq. 9.4-11 in NEDE-32176P, Rev. 3? If the energy distribution fraction Fco is decreasing because the moderator density is decreasing, how are the other fractions in Eq. 9.4-11 in NEDE-32176P, Rev. 3, adjusted to ensure that they sum to one?

h. Does the TRACG uncertainty analysis include uncertainty associated with a and b for c, f, w, bl, ch, and co?

GEH Response

The letters denoting the paragraphs in the request will be followed in the responses that follow.

a. Additional detail for the direct moderator heating (DMH) model is available in subsection C3DX of Section 5.1 of NEDE-32906P-A, Revision 3. The total DMH fraction for a kinetics node kij is calcualted from Equation (9.4-14) of NEDE-32176P, Rev. 3 using a nodal density ρ_m that is calculated in the way indicated in the response to RAI 21-b for NEDE-32906P-A, Revision 3. Each node can contain three regions J denoted by the subscripts AC for active channel, BP for bypass, and WR for water rod. The defining equation for ρ_m is repeated here.

$$\rho_m = F_{AC}\rho_{AC} + F_{BP}\rho_{BP} + F_{WR}\rho_{WR} \quad (21.6\text{-}81.1)$$

where

$$\rho_J = \left[(1-\alpha)\rho_\ell + \alpha\rho_v\right]_J \text{ for } J \in \{AC, BP, WR\}, \quad (21.6\text{-}81.2)$$

and F_J is the volume fraction in region J,

α_J is the void fraction in region J,

ρ_ℓ is the liquid density in region J,

ρ_v is the vapor density in region J.

When the dynamic water rod model is active, the value for ρ_{WR} is calculated from the TRACG hydraulic solution, otherwise $\rho_{WR} = \rho_{BP}$ for each axial location. For all cases, all quantities are defined for each kij kinetics node. For each such node the volume fractions satisfy the relationship

$$F_{AC} + F_{BP} + F_{WR} = 1.0 \quad (21.6\text{-}81.3)$$

Equations (21.6-81.2) and (21.6-81.3) apply for either a controlled or uncontrolled node since the value of F_{BP} depends on the whether a control blade is present or absent in determining the nodal values for ρ_m used to drive the cross section model.

Using ρ_m from Equation (21.6-81.1), the total DMH fraction $F_{co}(t)$ is calculated from Equation (9.4-14) of NEDE-32176P, Rev. 3 and then split among the three regions proportional to the water density fraction from the uncontrolled condition so that

$$F_{co} = \left[\gamma_{AC} + \gamma_{BP} + \gamma_{WR}\right]_{uncontrolled} \quad (21.6\text{-}81.4)$$

where $\gamma_J = \dfrac{\rho_J}{\rho_m}\left[F'_J\right]_{uncontrolled}$ for $J \in \{AC, BP, WR\}$. (21.6-81.5)

Please see the response to part **H** for additional discussion related to controlled conditions.

The TRACG DMH model is based on the fact that the largest component of DMH is due to neutron scattering off of hydrogen atoms in water molecules and that this effect is proportional to the number density of hydrogen atoms and thus proportional to the water density. This fact was supported by detailed MCNP analyses that assessed both the neutron and photon components of DMH in each of the three regions for different fuel types. The results are shown in Figures 5-11, 5-12 and 5-13 of NEDE-32906P-A, Revision 3.

b. The context for which setting FDMH2=FDMH1 is nonconservative is with respect to calculating the peak clad temperature (PCT) during a LOCA event in an operating BWR. That was the purpose for which the comment in the User's Manual (UM) was made. The UM comment does not apply for AOO, ATWS and stability scenarios. For a postulated LOCA in an operating BWR, it would be conservative to assume that most or all of the DMH component attributed to decay heat is due to gamma heating in the fuel since this will result in the maximum heat flux through the cladding and maximize the calculated PCT. In other words, with respect to impact on the calculated PCT, setting FDMH2=0.0 is the most conservative choice. PCT is not a key parameter except for LOCA scenarios in operating BWRs. For LOCA scenarios in the ESBWR, PCT is only nominally a key parameter since fuel heatup does not occur for the design basis accident; therefore, setting FDMH1=FDMH2 is acceptable.

c. [[

]]

d. [[

]]

e. TRACG does not explicitly account for gamma heating in the vessel wall.

f. The constants *a* and *b* are used to account for the fact that fission gammas and decay heat gammas have different energies that may impact how their energies are deposited. Gamma energy is primarily deposited into materials with larger atomic numbers like fuel and structural materials so their deposition is insensitive to the moderator density. In any case, gamma energy primarily gets redeposited into the fuel itself. The other major component of directly deposited energy is energy from neutrons. Unlike gamma energy, neutron energy is primarily deposited in the moderator as neutrons scatter with hydrogen in water. Eventually most of the neutrons are moderated to thermal energy and end up being absorbed in the fuel. [[

]] simulations confirm that total energy deposition in the moderator is modeled well within an uncertainty of [[ ]] as indicated in Figure 5-11 of NEDE-32906P-A, Revision 3.

g. The value for $F_f(t)$ is calculated as unity minus the sum of all the other fractions. As $F_{co}(t)$ decreases the value of $F_f(t)$ increases. Similarly, changes in the other fractions with time will also result in a change in $F_f(t)$ so that all the fractions will continue to sum to unity.

h. The TRACG uncertainty does not explicitly consider the uncertainties in all the components of the model. The total uncertainty of [[ ]] in the total DMH is sufficient to encompass all of these other minor uncertainties. To put everything in the correct perspective, a [[ ]] change in the total DMH results in less than a [[ ]] impact in the calculated ΔCPR/ICPR. A change of [[ ]] in CPR is considered to be negligible.

<u>NRC RAI 6</u>

(RAI 21.6-82 on the ESBWR DCD Docket) Section 9.1.3 in NEDE-32176P, Rev. 3, indicates that at the beginning of the calculation with the PANCEA wrapup, that the TRACG cross sections include the presence of Xenon. However, the transient calculation procedure does not indicate that the Xenon concentration is updated. The staff is aware that TRACG is capable of simulating transients with transient Xenon conditions, but is unable to locate any details about your models and calculation procedures. Please provide these details. Are transient Xenon conditions used in the simulation of any AOO and ATWS events? Include information on how the treatment of Xenon is conservative for these events.

GEH Response

Answered on the ESBWR DCD Docket, GEH letter MFN 07-352

NRC RAI 7

(RAI 21-6-84 on the ESBWR DCD Docket) In discussing the biases and uncertainties for the void coefficient in NEDE-32906P "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," in response to Staff RAI 12 you state: "When the PANAC11 model is implemented in TRACG it will be necessary to make a similar assessment TGBLA06 and MCNP and change the TRACG void coefficient model accordingly." Please state if this has been done and provide the staff with the documentation that includes the details of the new evaluation.

GEH Response

Answered on the ESBWR DCD Docket, GEH letter MFN 07-352.

NRC RAI 8

(RAI 21.6-85 on the ESBWR DCD Docket) Describe the computational procedure used to generate a PANACEA Wrap up file for use with TRACG. Specifically explain what calculations are performed with PANAC11 and how these results are captured numerically in the PANACEA Wrap up file.

GEH Response

Answered on the ESBWR DCD Docket, GEH MFN 07-347

NRC RAI 9

(RAI 21.6-86 on the ESBWR Docket) The isotopic tracking in the PANAC11 code is discussed in NEDC-33239P (Reference 4). Please provide a prototypical calculational model (e.g. the differential equations) for the determination of plutonium content based on the nodal power, exposure, and moderator density history.

GEH Response

Answered on the ESBWR DCD Docket, GEH letter MFN 06-467. After the review of RAI 21.6-86, additional information was requested by the NRC under RAI 21.6-86 S01 and RAI 21.6-94. Both of these RAIs have been addressed respectively in MFN 06-467, Supplement 1 and MFN 07-079.

NRC RAI 10

(RAI 21.6-87 on the ESBWR DCD Docket) PANAC11 uses the GEXL correlation to determine critical quality for the purpose of calculating the minimum CPR. Describe how PANAC11 calculates the bundle power where boiling transition occurs.

GEH Response

Answered on the ESBWR DCD Docket, GEH letter MFN 06-467.

NRC RAI 11

Deleted

NRC RAI 12

(RAI 21.6-68 on the ESBWR DCD Docket) On Page 6-135 of Reference 1, should the reference to Figure 6-34 actually be to Figure 6-37?

GEH Response

Yes. This will be corrected in the next revision of the TRACG Model Description (NEDE-32177P). This was also answered on ESBWR Docket, GEH Letter MFN 07-352.

NRC RAI 13

(RAI 21.6-75 on the ESBWR DCD Docket) Please submit Reference 5.

GEH Response

Transmittal of GEH Topical Report, NEDE 32177P, Revision 3, *TRACG Qualification*, August 2007. This transmittal provides the subject document requested by RAI 13 of Reference 1 and RAI 21.6-75 of Reference 2.

NRC RAI 14

(RAI 4.3-3 on the ESBWR DCD Docket) In DCD Tier 2, Page 4.3-3, reference is made to the lattice code TGBLA06, which has recently been modified to accommodate a minor correction in the programming of analytical formulation in the code. Please submit the modification(s) to

TGBLA06. The submittal should include the changes made to the code and validation of the code as it pertains to recent application(s) since the modification of the code, and any natural circulation database, as it pertains to the analysis of the ESBWR steady-state neutronic performance. The contents of the submittal should include before and after calculational results with technical justification(s) in support of the changed results. Also provide a comparison between the modified TGBLA and MCNP results in Section 1.3 of NEDC-33239P, "GE14 for ESBWR Nuclear Design Report" (Reference 4).

GEH Response

Answered on the ESBWR DCD Docket, GEH letter MFN 06-297, Supplement 1

NRC RAI 15

(RAI 4.3-4 on the ESBWR DCD Docket) Discuss any recent changes made to PANACEA since the NRC staff's last approval. Provide similar information to that requested in RAI 4.3-3. It is presumed that this version of the code is the NRC-approved version of record.

GEH Response:

Answered on the ESBWR DCD Docket, GEH letter MFN 06-297. Additional information in response to NRC questions was provided in MFN 06-297 Supplement 2, and Supplement 8.

NRC RAI 16

(RAI 6.3-54 on the ESBWR DCD Docket) Section C.1.4.1 of NEDE-32176P, "TRACG Model Description,"Revision 3, states that the correlation for thermal conductivity used in TRACG04 for UO2 with and without Gadolinia has been updated to be compatible with the model used in PRIME03. PRIME03 has not been reviewed and approved by the NRC staff. Provide justification for using this model.

GEH Response

The PRIME03 computer code is not currently being used to provide input to any of the TRACG04 ESBWR applications. GEH understands and has acknowledged that the NRC staff wish to review PRIME03 and that NRC approval for licensing applications of PRIME03 is required. Fuel file inputs to TRACG04 are being provided by the approved GSTRM computer code.

Section C.1.4.1 of NEDE-32176P, Revision 3 serves to document the improved fuel thermal conductivity model in TRACG04. The fact that this model is compatible with the PRIME03 model does not constitute a request or a requirement that PRIME03 be reviewed and approved by the NRC. The PRIME03 code is not being used to provide input to any TRACG04 ESBWR calculations.

The improved thermal conductivity model in TRACG04 introduces two real dependencies that are not present in the TRACG02 model: (1) degradation of thermal conductivity due to the presence of gadolinium; (2) change in thermal conductivity with exposure. At zero exposure and when there is no gadolinium, the TRACG04 thermal conductivity is [[ ]] as the thermal conductivity from the TRACG02 model as illustrated in Figure 6.3-54. 1. All of the figures that follow are based on the TRACG04 model because the TRACG02 model is dominated by the temperature dependency shown in Figure 6.3-54. 1. The TRACG02 model has no dependency on gadolinium [[

]].

Increasing gadolinium in the TRACG04 model results in lower thermal conductivity as shown in Figure 6.3-54. 2. Increasing fuel exposure also results in lower thermal conductivity as seen in Figure 6.3-54. 3. Thus the new model will produce higher, more conservative fuel temperatures relative to the model previously approved. The combination of high amounts of gadolinium and increasing exposure produces the lowest fuel thermal conductivity as seen in Figure 6.3-54. 4. The three previous figures all show that the effects on fuel thermal conductivity [[ ]] are negligibly small at the higher temperatures where one might postulate an impact on the design basis.

The greater impact of gadolinium and exposure on fuel thermal conductivity occur for lower fuel temperatures that are expected for normal operation. Figure 6.3-54. 5 depicts the calculated thermal conductivities versus exposure for the two extremes of gadolinium for two temperatures in the normal operational range. The figure shows that the largest variation in fuel thermal conductivity occurs for the lower exposures [[ ]] before there has been an appreciable release of fission gases from the fuel pellet. The exposure range from between 10 to 30 GWd/t is the range of most interest with regards to transient, stability, ATWS and LOCA analyses for which TRACG is applied because these are the exposure ranges where the limiting cases occur. The impact of fission gas release and thermal conductivity of those gases in the pellet-clad gap that is captured via the fuel files is of [[ ]] importance. The [[ ]] impact is on the calculated temperature for the fuel pellet since the temperature gradient will vary inversely proportional to the thermal conductivity. In other words, pellet average temperature will increase when the fuel thermal conductivity is decreased. For steady conditions, the fuel pellet temperature impacts the stored energy and the gap size. For transient conditions the gap size impacts the overall thermal time constant for the fuel rod in addition to the direct impact on time constant that results from thermal conductivity.

In all the events of interest, the fuel thermal conductivity will tend to be lower when the effects of gadolinia and exposure are considered. The following paragraphs discuss how the realistic treatment of these effects in the improved TRACG04 model impact licensing calculations.

One common factor to all the event scenarios is the determination of the initial gap size. A lower thermal conductivity results in a higher fuel temperature and results in a smaller gap or results in pellet-clad gap closure at a lower exposure. [[

]]

For transients, a smaller gap produces a more dynamic response which tends to increase the calculated change in CPR. [[

]]

For stability events, the decreased initial gap size resulting from a higher overall pellet temperature provides initially for a larger mismatch in the heat flux relative to the flow reduction making it easier to trigger the oscillation. This may result in a larger amplitude power response to a particular flow reduction [[

]]

For ATWS events, the increased dynamic response changes the oscillation signature early in the event [[

]]

For LOCA analyses in operating BWRs, more stored energy initially in the fuel pellet together with a smaller calculated gap for lower exposures tends to increase the probability of an initial boiling transition as the core flow decreases rapidly during the first few seconds of a DBA LOCA. [[

]] The ESBWR LOCA calculations used a constant gap conductance so the dynamic gap model in TRACG was not used. The justification for this simplification for ESBWR LOCA calculations was provided in the response to RAI 6.3-53.

The improved TRACG04 fuel thermal conductivity model realistically accounts for the degradation of thermal conductivity due to the presence of gadolinium and the change in thermal conductivity with exposure. Accounting for these dependencies removes the bias in TRACG02 modeling that would be present except when the fuel was at zero exposure and contained no gadolinium.

In summary, the principle justifications for using the improved thermal conductivity model in TRACG04 are as follows: (1) the TRACG04 model is technically more correct in that it accounts for known dependencies that are not modeled in TRACG02; (2) the TRACG04 model will produce calculated results for all applications that are [[

]] relative to the TRACG02 model; (3) the model can be and is being used with the GSTRM fuel files and does not require NRC review and approval of PRIME03 since the calculated results from PRIME03 are not being used in licensing calculations.

[[ ]]

Figure 6.3-54. 1 Comparison of TRACG04 and TRACG02 Models

[[ ]]

Figure 6.3-54. 2 Impact of Gadolinium in the TRACG04 Model

[[
]]

Figure 6.3-54. 3 Impact of Exposure in the TRACG04 Model with 0% Gadolinium

[[
]]

Figure 6.3-54. 4 Impact of Exposure in the TRACG04 Model with 10% Gadolinium

[[
]]

Figure 6.3-54. 5 Thermal Conductivity Variations for Operational Fuel Temperatures

(RAI 6.3-54 S01 on the ESBWR Docket) Provide experimental data and benchmarks to support the improved TRACG04 pellet conductivity model. Provide a sensitivity study using the TRACG02 and TRACG04 conductivity models with fuel files.

<u>GEH Response</u>

The TRACG04 fuel pellet conductivity model is based on the PRIME03 fuel pellet conductivity model. The implementation of this model in TRACG04 is described in Appendix C of NEDC-33256P (Reference 6.3-54-1). A detailed description of this model is provided in Section 3.3.1 of Reference 6.3-54-2. In addition, the detailed basis of this model and comparisons with NRC identified open literature data has been discussed with the NRC related to the response to PRIME RAI-20 Supplement 1. The response to PRIME RAI-20 Supplement 1 will be provided separately.

A sensitivity study is performed on the two UO_2 (fuel pellet) conductivity models available in TRACG04, and the gap conductance fuel files. Please note that the conductivity model referred to as the TRACG02 model in the RAI above is referred to as the GESTR UO_2 conductivity model in this response. Likewise, the TRACG04 default model is referred to as the PRIME UO_2 conductivity model. TRACG04 is used for all the cases in this study. The base case for the sensitivity study is the analysis of record in DCD Tier 2 Revision 5. These cases are run with the PRIME UO_2 conductivity model and with fuel rod gap conductance inputs from the GESTR computer code. Details can be found the TRACG Model Description (Reference 6.3-54-1). For each event evaluated two sensitivity cases are performed. Case 1: This case is the same as the base case except that the GESTR UO_2 conductivity model is selected. Case 2: This case is the same as the base case except that the fuel rod gap conductance inputs are from the PRIME computer code.

The details of the effect of the PRIME versus GESTR UO_2 conductivity model can be seen in the response to RAI 6.3-54 (Reference 6.3-54-3). [[

]]

Three events are selected for the study: 1) The generator load rejection with total bypass failure (traditionally abbreviated LRNBP), in DCD Tier 2 Subsection 15.3.5. This is the transient event (Anticipated Operational Occurrences [AOO] and Infrequent Events) expected to be the event most affected by UO_2 conductivity and fuel rod gap conductance changes, because it is a fast event with the most severe flux peak. 2) The Anticipated Transients without Scram (ATWS) Main Steam Isolation Valve Closure (MSIVC) bounding case in DCD Tier 2 Subsection 15.5.2. This is the ATWS event expected to be the event most affected by UO_2 conductivity and fuel rod gap conductance changes. The effect on the ATWS loss of condenser vacuum event is expected to be similar; however, only one event is needed to demonstrate the sensitivity. 3) The stability Loss of Feedwater Heating (LOFWH) regional stability evaluation at (Middle of Cycle) MOC exposure in DCD Tier 2 Subsection 4D.1.5. This event is selected because it is the limiting stability event. No Loss-of-Coolant Accident (LOCA) event is evaluated. Because the water level remains above top of active fuel in the LOCA analysis there is no fuel heat-up; therefore, the UO_2 conductivity and fuel rod gap conductance changes are much less important than in cases where fuel heat-up is calculated, as discussed in the response to RAI 6.3-53 (Reference 6.3-54-4).

LRNBP Study: Table 6.3-54-1 shows the results of the LRNBP study. [[

]]

Table 6.3-54-1

	Base Case – LRNBP*	Case 1 – LRNBP	Case 2 – LRNBP
UO_2 Conductivity Model	PRIME	GESTR	PRIME
Gap Conductance Input	GESTR	GESTR	PRIME
Maximum Neutron Flux, % NBR	425	[[	
Maximum Dome Pressure, MPaG (psig)	8.23 (1194)		
Maximum Vessel Bottom Pressure, MPaG (psig)	8.36 (1212)		
Maximum Steamline Pressure, MPaG (psig)	8.22 (1192)		
Maximum Simulated Thermal Power, % of Initial	108.7		
ΔCPR/ICPR	0.138		]]
*Data From DCD Tier 2, Table 15.3-1a.			

ATWS MSIVC Study: Table 6.3-54-2 shows the results of the ATWS MSIVC study. [[

]]

Table 6.3-54-2

	Base Case – ATWS MSIVC*	Case 1 – ATWS MSIVC	Case 2 – ATWS MSIVC
UO_2 Conductivity Model	PRIME	GESTR	PRIME
Gap Conductance Input	GESTR	GESTR	PRIME
Sensed Maximum Neutron Flux, %	265.3	[[	
Maximum Vessel Bottom Pressure, MPaG (psig)	9.47 (1374)		
Maximum Bulk Suppression Pool Temperature, ºC (ºF)	72.8 (163)		]]

	Base Case – ATWS MSIVC*	Case 1 – ATWS MSIVC	Case 2 – ATWS MSIVC
Associated Containment Pressure, kPaG (psig)	205.6 (29.8)	[[	
Peak Cladding Temperature, ºC (ºF)	849.1 (1560)		]]
*Data From DCD Tier 2, Table 15.5-5a.			

Stability LOFWH Study: Table 6.3-54-3 shows the results of the stability LOFWH study. All cases are run at MOC (peak hot excess). [[

]]

Table 6.3-54-3

	Base Case – LOFWH Regional Stability*	Case 1 – LOFWH Regional Stability	Case 2 – LOFWH Regional Stability
UO_2 Conductivity Model	PRIME	GESTR	PRIME
Gap Conductance Input	GESTR	GESTR	PRIME
Power (% of Rated)	106	[[	
Regional Decay Ratio	0.66		]]
*Data From DCD Tier 2, Table 4D-4.			

Figure 6.3-54-1

GE14 UO2 Peak Gap Conductance at 7.5 kW/ft Power Level

[[

]]

Figure 6.3-54-2

GE14 UO2 Peak Gap Conductance at 13.4 kW/ft Power Level

[[

]]

References

6.3-54-1 GE Hitachi Nuclear Energy Letter # MFN 08-072 dated February 6, 2008, to USNRC, "Transmittal of GE Hitachi Nuclear Energy (GEH) Licensing Topical Report, NEDE-32176P, Revision 4, TRACG Model Description, January 2008."

6.3-54-2 Global Nuclear Fuel Letter # FLN-2007-001, "GNF Licensing Topical Report, "The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance," NEDC-33256P, NEDC-33257P, and NEDC-33258P, January 2007.," January 19, 2007.

6.3-54-3 GE Hitachi Nuclear Energy Letter # MFN 08-053 dated January 25, 2008, to USNRC, "Response to Portion of NRC Request for Additional Information Letter No. 68 – Emergency Core Cooling Systems - RAI Numbers 6.3-54, 6.3-55"

6.3-54-4 GE Energy Letter # MFN 07-218 dated May 15, 2007, to USNRC, "Response to Portion of NRC Request for Additional Information Letter No. 68 - Engineered Safety Features - RAI Numbers 6.2-100, 6.3-44, 6.3-53, 6.3-58, and 6.3-59"

NRC RAI 17

(RAI 6.3-55 on the ESBWR Docket) Provide justification for using gas gap conductivity and fuel thermal conductivity from two different analysis codes (GSTRM for gap conductivity and PRIME03 for fuel thermal conductivity).

GEH Response

The requested justification was provided in the response to RAI 6.3-54. The most salient portion of the justification is repeated here. [[

]]

NRC RAI 18

Deleted

NRC RAI 19

Deleted

NRC RAI 20

Deleted

NRC RAI 21

Provide additional information demonstrating that the uncertainty in the Doppler coefficient and in scram reactivity (phenomena identification ranking table (PIRT) items C1BX and C1CX, respectively) cited in Table 5-5 in the TRACG AOO analysis for BWR/2-6 (Reference 3) is still applicable or bounding when applying the new PANAC11 physics methods.

GEH Response

The scram reactivity (C1CX) uncertainty is dominated by the uncertainty in determining the scram speed. The scram speed uncertainty is determined based on plant data obtained from scram speed tests at BWR plants and does not depend on the lattice or core physics methods.

The remainder of this response is concerned with the Doppler coefficient uncertainty (C1BX).

The basis for the [[ ]] value for the Doppler coefficient uncertainty (C1BX) that is cited in Reference [21-1] traces back to several RAI responses scattered in the different volumes of Reference [21-2]. The [[ ]] value for C1BX can be traced to question 38 filed in Volume 1 of Reference [21-2]. The response to that question indicates that the value is conservatively determined based on calculated responses for the SPERT tests. [[

]]

To re-establish use of the [[ ]] value for TRACG04, test 43 in the SPERT-III series was evaluated using this value. Test 43 in the SPERT-III series is the most relevant test for this purpose because it is the cold test with the highest control blade worth. The cold tests are not complicated by early voiding in the fluid thus the power pulse is controlled by only two factors: (1) control blade reactivity insertion, (2) Doppler negative feedback. The primary impact due to the blade worth was removed by adjusting the simulation to precisely match the stated value of $1.21 from the test similar to what was done in creating Figure 3.8-2 of Reference [21-4]. Two additional calculations were then performed on either side of this reference calculation by multiplying the Doppler coefficient for all nodes by factors of [[ ]] in one case and [[ ]] in the other case to account for the presumed one-sigma variation of [[ ]]. The calculated power responses are shown in Figure 21-1 superimposed on the results presented previously in Figure 3.8-2 of Reference [21-4]. Note that the scales have been modified to allow the detail in the vicinity of the power peak to be seen. The results in Figure 21-1 show that the nominal measured peak power value is bracketed by a [[ ]] variation in Doppler coefficient that corresponds to a variation of about [[ ]] in the calculated peak power value.

Additional support for a [[ ]] uncertainty in the Doppler coefficient is provided by the AOO transient analyses for cases where comparisons to plant data are possible. These evaluations were updated using TRACG04 and have been provided to the NRC staff in Chapter 7 of Reference [21-4]. Selected results for the analyses of the Peach Bottom turbine trip tests obtained prior to completion of Reference [21-4] had been provided in Figures 7-1 through 7-6 of Reference [21-5], the LTR that is the focus of this RAI. For consistency with the final results in Reference [21-4], Figures 7-1 through 7-6 for the LTR under review (Ref. [21-5]) are being updated to reflect input modifications [[

]]. The replacement figures are provided separately and will be incorporated in the approved LTR containing a copy of the NRC Safety Evaluation Report.

Comparisons between calculated and measured power responses (some of which have been cited here) justify continued use of [[ ]] for AOO applications using our three-dimensional process. The uncertainty in Doppler coefficient for our three-dimensional neutronics models is relatively unimportant compared to the larger range of values already accounted for spatially across the core. For example, the section for C1BX in Chapter 5 of Reference [21-1] documented that a [[ ]] change in the Doppler coefficient applied to each node in the core simultaneously in the same direction would change the calculated ΔCPR/iCPR for the limiting turbine trip without bypass event by only about [[ ]].

The uncertainty in Doppler coefficient would have to increase by about an order of magnitude before the expected impact on the calculated ΔCPR/ICPR would start to become important relative to the overall transient uncertainty in ΔCPR/iCPR that is considered in the licensing basis for AOO events. To demonstrate this point, the PB2TT3 case for which the nominal results are shown in Figures 7.5 and 7.6 of Reference [21-5] was recalculated for ±40% core-wide perturbations in Doppler coefficient. The calculated nominal ΔCPR/iCPR result together with the results for the ±40% perturbations in Doppler coefficient are shown in Figure 21-2. Even at 10 times the expected uncertainty, the Doppler coefficient perturbations conservatively applied to each node in the core simultaneously produce a change in the calculated ΔCPR/iCPR that is only slightly greater than the [[ ]] threshold that is considered to be negligible.

The [[ ]] uncertainty for Doppler coefficient is adequate for AOO applications and its use in TRACG04 AOO calculations has been justified.

References

[21-1] *TRACG Application for Anticipated Operational Occurrences Transient Analyses*, NEDE-32906P-A, Revision 1, Class III, April 2003.

[21-2] *Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors*, NEDO-24154P-A (Volumes 1 and 2) and NEDE-24154-P-A (Volumes

3 and 4), August 1986 for Volumes 1-3, February 2000 for Revision 1 for Volume 4.

[21-3] *Generation of Void and Doppler Reactivity Feedback for Application to BWR Design*, NEDO-20964, Class I, December 1975.

[21-4] *TRACG Qualification*, NEDE-32177P, Revision 3, Class III, August 2007.

[21-5] *Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients*, NEDE-32906P Supplement 3, Class III, May 2006.

[[

]]

Figure 21-1: SPERT III Test 43 Power Responses

[[ ]]

Figure 21-2: PBTT3 ΔCPR/iCPR Sensitivity to ±40% Core-Wide Doppler Coefficient Change

NRC RAI 22

Regarding the decay heat model, provide additional information on how you determine fission fractions of U-238 and Pu-239 as a function of exposure and the MeV/fission values by submitting the following reference (Reference 203 in Reference 1): C. L. Martin, *Nuclear Basis for ECCS (Appendix K) Calculations*, NEDO-23729, Class 1 GE Report, November 1977. In addition, clarify if the MeV/fission value cited on Page 9-24 of Reference 1 (TRACG04 Model Description) is used in the nodal power calculation (ε_i in Equation 9.1-59 in Reference 1).

GEH Response.

The requested report is transmitted along with this response.

The MeV / fission values cited in Equation (9.3-18) are those from Table 2-11 of NEDO-23729. These are generic, historical values used to assure consistency between TRACG and the decay heat evaluations performed previously. [[

]]

The value for ε_i in Equation (9.1-59) is determined for each neutronics node in the 3D kinetics model based on the evaluation of the energy per fission supplied in functional form by the lattice physics. For TRACG04 the relevant lattice physics code is TGBLA06. In this sense, ε_i is evaluated like any other nuclear parameter used with the 3D kinetics model so it includes the dependency on lattice design, exposure, moderator density, and moderator density history.

NRC RAI 23

Provide additional information on the specific implementation and/or guidelines for using the American Nuclear Society (ANS) decay heat curves for use in BWR/2-6 AOO analyses. Include user input details such as number of irradiation periods selected and number of decay heat groups.

GEH Response

Change in CPR for AOO events has been shown to be insensitive to variations in decay heat. Only the level response for a loss-of-feedwater (LOFW) event justifies decay heat being ranked as "high" in the Phenomena Identification Ranking Table (PIRT). The ANS decay heat models were implemented primarily for LOCA applications where decay heat is more important.

AOO applications use the 3D kinetics model (IRPOP=9) for which the default decay heat model is May-Witt (IDHOP=1, IDHG=5). See the discussion in Section 9.3.2 of NEDE-32176P, Rev. 3. No additional code inputs are required to select the May-Witt model. This model employs five decay heat groups and the decay group parameters are constants that are independent of exposure and the irradiation periods. Use of this model is acceptable for AOO applications since it is quite conservative. Use of the May-Witt model will result in an under prediction of the minimum level for the LOFW event because an over-prediction of the decay heat by about 15% produces a boil off rate that is too large by about the same amount.

Use of either of the ANS models is allowed for AOO analyses. Selection of the 1979 or 1994 ANS models is clearly described in the *TRACG04A,P User's Manual*. The 1979 ANS model is selected using the single input value IDHOP=2. Similarly, the 1994 ANS model is selected using the single input value IDHOP=3. The 1979 ANS model is preferred for the shorter-duration AOO events because it is slightly more conservative than the 1994 ANS model for the decay times analyzed in AOO events. Either choice will automatically select the correct default value for the number of decay groups (IDHG) and establish the default values for the decay heat group constants associated with each model. Initialization of the ANS models depends on exposure. By default, the selected

ANS model is initialized and applied based on the CHAN group exposure individually for each CHAN group. The CHAN group exposures are determined from the PANAC11 wrapup when the 3D kinetics option is active as is the case for AOO analyses. The ability to specify the CHAN group exposure is a code feature that was primarily intended for LOCA applications where the 3D kinetics model is usually not used. Because of the relative unimportant role of decay heat for AOO analyses, a single irradiation period at the initial CHAN power is assumed prior to the transient and the time of irradiation is estimated assuming a constant power over that interval in order to produce the CHAN group exposure. Other special features of the ANS decay heat model implementation are not needed or used for AOO application.

The recent fission power history prior to the scram is not important for AOO events but it may be important for other analyses such as ATWS and small-break LOCAs where the scram can be delayed. The ANS models depend on this recent history as well as the extended history. The extended history is accounted for in the initialization as described in the previous paragraph. The recent history is accounted for as described in Section 9.3.1 of NEDE-32176P, Rev. 3.

NRC RAI 24

In the TRACG04 application for ESBWR AOO's (Reference 6), you increased the uncertainty in interfacial shear based upon comparisons to the Toshiba data (PIRT Item C2AX in Table 4.4-1, Reference 6, PIRT22 in TRACG04). This value was increased from the value for BWR/2-6 AOO's cited in Table 5-5 as PIRT item C2AX in Reference 3, which was based upon comparisons to FRIGG data. In addition, in the ESBWR application (Reference 6) you included an uncertainty for the entrainment multiplier to account for the data in the transition and annular flow regimes (included with PIRT Item C2AX in Table 4.4-1, Reference 6, PIRT52 in TRACG04). Explain why you do not increase/include these uncertainties in the TRACG04 application for BWR/2-6 AOO events.

GEH Response

The report "TRACG Application for ESBWR", NEDE-33083P-A, March 2005 (Reference 6) in the request for additional information describes the TRACG application for ESBWR LOCA. The Toshiba tests were added to the TRACG qualification basis in order to expand the void fraction qualification for low pressures. The Toshiba tests were conducted at pressures of 0.5-1 MPa. The increased uncertainty in the interfacial shear (PIRT22) and the added uncertainty in the entrainment (PIRT52) were introduced to cover the larger uncertainty observed in the comparison to void fraction data at low pressure and to cover the wider pressure range needed for LOCA applications.

The uncertainty in the void fraction is essentially unchanged for applications at rated pressure such as anticipated operational occurrences. The basis for the void fraction uncertainty for AOO applications is the comparisons to the FRIGG tests. The qualification of TRACG02 (NEDE-32177P, Revision 2, January 2000) against the FRIGG data at pressure of [[ ]] showed a mean error of [[ ]] and a standard deviation of [[ ]]. The qualification of TRACG04 against the same data shows a mean error of [[ ]] and a standard deviation of [[ ]]. The uncertainty is unchanged and the bias has increased by [[ ]], which leads to slightly increased [[ ]] for typically limiting AOO events. Therefore the same void fraction uncertainty as documented in the LTR for the TRACG02 application to AOOs (NEDE-32906P-A, Revision 3, September 2006) can be applied for TRACG04.

NRC RAI 25

Provide the implementation details of the optional 6-cell jet pump model. Please update the TRACG04 User's Manual (Reference 2) and the TRACG Model Description (Reference 1) with these details.

GEH Response

The following sentence will be added to Section 7.6.1 in the next revision of TRACG Model Description (NEDE-32176P, revision 3, April 2006):

"The TRACG jet pump model uses a standard nodalization with 3 or 4 nodes in the primary branch consisting of the region from the suction inlet to the end of the diffuser and 2 nodes in the secondary branch simulating the nozzle. Figure 7-22 shows the nodalization with 3 nodes in the primary branch. For applications where two-phase conditions may exist in the jet pump and where an accurate characterization of the void profile is important, the first node in the primary branch may be subdivided into two nodes."

The TRACG04 Users Manual (UM-0136, December 2006) correctly describes the jet pump nodalization options.

The extra node in the primary branch was introduced to get a more accurate characterization of the void profile and static head in the jet pump for scenarios such as a LOCA where two-phase conditions may exist in the jet pump. For applications to AOOs where single-phase conditions exist in the jet pump, the extra node is not needed and the two options produce similar results.

NRC RAI 26

Provide additional information demonstrating that the bias and standard deviation in the jet pump N-ratio (PIRT parameters G1 and G3) cited on Pages 5-35 and 5-36 of Reference 3 are applicable or bounding when using the 6-cell jet pump model in BWR/2-6 AOO analysis.

GEH Response

The implementation of the 6-cell nodalization for the jet pump affected the pressure drop calculation between the nozzle and the and the junction cell. Therefore the recommended jet pump input (loss coefficients) and PIRT parameters have been re-evaluated for the 6-cell jet pump. The recommended values for the 6-cell jet pump are given in Table 26-1 below.

Table 26-1 Additional Nozzle and Inlet Losses for 6-cell Jet Pump

	Input Parameter	1-Nozzle Jet Pump	5-Nozzle Jet Pump
Nozzle Loss (Normal drive flow*)	EPSNOZ	[[	
Nozzle Loss (Reverse drive flow*)	EPSNZR		
Inlet Loss	FINLET		]]

* Internally in TRACG, the driveline velocity is positive in the direction from the nozzle toward the driveline. Thus internally in TRACG the driveline velocity is negative for normal drive flow, and positive for reverse drive line flow.

The TRACG04 Users Manual will be updated to incorporate these recommended values.

The comparisons of TRACG04 with the 6-cell jet pump against the INEL 1/6 scale jet pump data, Cooper one-Nozzle jet pump and the LaSalle five-nozzle jet pump are shown in Figures 26-1 through 26-3.

[[
]]

Figure 26-1 Comparison to INEL 1/6 Scale Jet pump Data

[[
]]

Figure 26-2 Comparison to Full Scale One-Nozzle Jet Pump Data

[[
]]

Figure 26-3 Comparison to Full Scale Five-Nozzle Jet Pump Data

These comparisons show that the impact of the 6-cell jet pump with the modified jet pump losses is a small improvement on the order of the uncertainty in the data. The improvement in the average N-Ratio ranges from [[ ]]. The improvement in the standard deviation ranges from [[ ]].

The statistics for the comparison of the 6-cell jet pump TRACG04 model with the One-Nozzle jet pump data for normal drive flow are shown in Figure 26-4. The sensitivity to the variation in the inlet loss (PIRT71) is shown in Figure 26-5.

[[
]]

Figure 26-4 Statistics for TRACG04 Comparison to Full Scale One-Nozzle Jet Pump Data

[[
]]

Figure 26-5 Sensitivity to Inlet Loss for a Full Scale One-Nozzle Jet Pump

There is essentially no bias in the nominal calculation (PIRT71 =1.0) and a variation in PIRT71 of [[{3}]] corresponds to one standard deviation in the N-ratio. It is observed from this analysis that it is possible to significantly affect the jet pump performance by increasing the jet pump inlet loss, while a limited impact is only possible for reduced inlet loss. For application to transient analyses this is not a significant concern, as the more severe results for limiting transient events are obtained for increased inlet loss [NEDE-32906P-A, Revision 3].

The statistics for the comparison of the 6-cell jet pumpTRACG04 model with the Five-Nozzle jet pump data for normal drive flow are shown in Figure 26-6. The sensitivity to the variation in the inlet loss (PIRT71) is shown in Figure 26-7.

[[
]]

Figure 26-6 Statistics for TRACG04 Comparison to Full Scale Five-Nozzle Jet Pump Data

[[
]]

Figure 26-7 Sensitivity to Inlet Loss for a Full Scale Five-Nozzle Jet Pump

There is no essentially bias in the nominal calculation (PIRT71 =1.0) and a variation of PIRT71 of [[ ]] corresponds to one standard deviation in the N-ratio.

Data is only available for reverse jet pump drive flow for the 1/6 scale INEL jet pump. The statistics for the comparison of TRACG04 with the 6-cell jet pump to the INEL jet pump data for reverse drive flow is shown in Figure 26-8, and the sensitivity to the variation in the reverse flow nozzle loss (PIRT70) is shown in Figure 26-9.

There is no essentially bias in the nominal calculation (PIRT70 =1.0) and a variation of PIRT70 of [[ ]] corresponds to one standard deviation in the N-ratio.

[[
]]

Figure 26-8 Statistics for TRACG04 Comparison to 1/6 Scale INEL Jet Pump Data for Reverse Drive Flow

[[
]]

Figure 26-9 Jet Pump Sensitivity to Nozzle Loss for the 1/6 Scale INEL Jet Pump Data for Reverse Drive Flow

It was stated previously based on the comparisons to the jet pump data that the impact of the re-optimized jet pump losses is insignificant. This statement has been confirmed by performing a comparison for a typical plant transient. The Hatch cycle 14 turbine trip simulation that was the base case in the TRACG Application LTR [NEDE-32906P-A, Revision 3] was analyzed with the original jet pump losses and the revised losses. The impact on the MCPR for the event is [[ ]].

[[
]]

Figure 26-10 Hatch Cycle 14 Turbine Trip Event

NRC RAI 27

(Similar to RAI 21.6-93 on the ESBWR Docket) The TRACG Model Description (Reference 1) states that "The default correlation for thermal conductivity (k) for unmolten UO2 has been updated to be compatible with the model used in PRIME03." For the demonstration calculations presented in your submittal (Reference 7), please state if you used the GSTRM or PRIME03 model for fuel thermal conductivity. In addition, how does a TRACG04 user specify the use of either model in a TRACG04 input deck? Provide the NRC staff the location in the TRACG04 User's Manual (Reference 2) that provides this guidance.

GEH Response

The AOO demonstration calculations presented in the submittal used the TRACG04 default correlation for unmolten UO2 thermal conductivity. The default model is consistent with the model used in PRIME03 and is described in Section C.1.4.1 of Revision 3 of NEDE-32176P dated April 2006. Justification for using this updated model is provided in the response to RAI 6.3-55 on the ESBWR docket. That ESBWR RAI was also included on the docket for this RAI as NEDE-32906P Supplement 3 RAI 17.

As stated in Enclosure 1 to Reference 27-1, the value of PIRT227 determines which model is used. Table D-2 on page 237 of the TRACG04 User's Manual provides this guidance.

References.

[27-1] Response to Portion of NRC Request for Additional Information Letter No. 82 – RAI Number 21.6-93, Docket No. 52-010, MFN 07-258, May 8, 2007.

NRC RAI 28

Provide additional information on the procedures for selecting the pump homologous curve input into TRACG.

GEH Response

The selection of the pump curves is performed as part of the initial base deck creation for a plant-specific application. Each plant has a plant specific TRACG base model. Pertinent pump data (e.g. rated head, torque, and speed) is used in the development of the plant-specific model. A generic set of pump homologous curve data is used. The generic pump curve data represents full scale test data that is appropriate for BWR recirculation pumps. When the TRACG basedeck is generated, the pump inertia is set to represent the plant. This, along with the input for the rated conditions are the key inputs for AOO application.

NRC RAI 29

The void reactivity coefficient bias and uncertainties in TRACG must be representative of the lattice designs of the fuel loaded in the core. State the lattices used to generate the void reactivity coefficient response for TRACG04/PANAC11. Include the restriction that Reference 7 is only applicable for these lattice designs.

GEH Response

The void coefficient was developed based on data for 8x8, 9x9, and 10x10 fuel, representative of GE9, GE10, and GE14 fuel, respectively. To address the restriction in this RAI, the void coefficient bias and uncertainties will be confirmed for new fuel (lattice) types.

NRC RAI 30

TRACG internally models the response surface for the void coefficient biases and uncertainties for known dependencies due to the relative moderator density and exposure on a nodal basis. Section 2.8.7 of the Vermont Yankee extended power uprate (EPU) safety evaluation report (Reference 8) reviewed the impact of the void history bias on the safety analyses. RAI SRXB-A-68 response (Reference 9) quantified the void history bias and discussed its impact. Section 2.2.2.2, "Treatment of Fuel Parameter Uncertainties," of Reference 10 also addressed the void history bias. Based on the quantified void history bias typical for the fuel designs typical of the EPU and the maximum extended load line limit analysis plus (MELLLA+) operating domain, modify the TRACG methodology to account for void history bias. The void history bias can be incorporated into the response surface "known" bias or through changes in lattice physics/core simulator methods for establishing the instantaneous cross-sections. Including the void history bias in the methodology negates the need for ensuring that each plant-specific application has sufficient margin available to account for the impact of the void history bias. Revise the nodal void reactivity coefficient biases and uncertainties an incorporate the void history biases. Provide sufficient technical details for the NRC staff to assess that the void history bias applied on a nodal level will conservatively bound the non-conservatisms in the current assumptions for nodes depleting at high void conditions.

NRC References

8. Vermont Yankee Nuclear Power Station - Draft Safety Evaluation for the Proposed Extended Power Uprate (TAC No. MC0761), October 21, 2005. (ML052910200)
9. BVY 05-088 Letter, J. Thayer (Vermont Yankee) to NRC, Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 35, Extended Power Uprate - Response to Request for Additional Information, September 28, 2005. (ML052770039)
10. NEDC-33173P, Applicability of GE Methods to Expanded Operating Domains, February 2006. (ML060720281)

GEH Response

<u>Overview</u>

The method to account for the biases and the uncertainties in the void coefficient model has been modified to include the effects due to void history (VH). Section *C1AX* has been updated to describe the TRACG methodology with the void history effects included. Calculations have been performed including the void history effects as part of the void coefficient correction model. By comparison to similar calculations performed with the model deactivated, these calculations reveal that correcting for biases in the void coefficient is expected to caused the key AOO calculated parameter of ΔCPR/iCPR to become somewhat more conservative as indicated in Figure 30-1 by a typical response for the usually limiting pressurization event. [[

]] These impacts may vary by core and cycle since the model depends on core and cycle-specific elements such as exposure, instantaneous voids and void history. The key point is that the impacts, either positive or negative, are now incorporated in the TRACG AOO methodology which is amended by incorporating the effects due to void history in determining the biases and uncertainties in the void coefficient on a plant and cycle-specific basis.

[[

]]

Figure 30-1 Typical Impact of Updated Void Coefficient Correction Model

It is unclear what is meant in the NRC request by "conservatively bound the non-conservatisms" since the purpose of a realistic model is to provide a means to quantify and account for the impact due to biases and uncertainties for the expected applications. That is what the revised TRACG04 model does. Additional details for the model are provided in the *Technical Description* subsection. This updated technical description has been expanded to incorporate details previously contained in a multi-part RAI response associated with the previous model. Those details have also been updated to describe how void history is accounted for in the updated model. Care has been taken to provide the same level of detail and where possible in almost the same order as in the original responses.

C1AX Void Coefficient, H

This section is an update to section by the same title in Reference [30-3].

TRACG04 uses a 3-D neutron kinetics model based on the PANAC11 model that uses neutronics parameters provided by TGBLA06 (see References [30-1] and [30-2]). The nodal reactivity is calculated [[

]]. All of these parameters are expressed in terms of the instantaneous moderator density and also include a dependency on moderator density history and nodal exposure. Consequently, the infinite multiplication factor also has these same dependencies.

The biases and uncertainties in void coefficient as determined from the PANAC11 originate in the biases and uncertainties in the infinite lattice eigenvalues (k_∞) calculated by the TGBLA06 lattice physics code [[

]] Values of k_∞ at a total of [[ ]] points were calculated for a representative set of [[ ]] lattices with 10x10 geometry at [[ ]] different exposures of [[ ]] and at different void histories (VH) of [[ ]] for in-channel instantaneous voids (IV) of [[ ]] using both TGBLA06 and MCNP. The results for each lattice, exposure, and void history were fit to a [[ ]] function to determine k_∞ as a function of instantaneous voids. The functional forms derived separately for TGBLA06 and MCNP were extrapolated to obtain [[ ]] values of k_∞ corresponding to 100% in-channel voids for each code. The void coefficients at a total of [[ ]] points were defined separately for TGBLA06 and MCNP by evaluating the derivative of k_∞ [[

]]. Biases and uncertainties in TGBLA06 void coefficients were evaluated by performing [[ ]] comparisons between TGBLA06 and the corresponding MCNP benchmark values. These assessments were made using uncontrolled lattices (lattices without a control blade). An earlier independent set of [[ ]] other TGBLA04 lattices all at zero exposure were evaluated [[ ]] as a check on the process. The check set using TGBLA04 comparisons to MCNP included [[ ]] controlled lattices to confirm that the uncontrolled lattices bound the biases and uncertainties for the controlled lattices. Because of the similarity in the TGBLA04 and TGBLA06 comparisons, the comparisons based on TGBLA06 using uncontrolled lattices are also expected to bound the biases and uncertainties for the controlled lattices.

To obtain the response surfaces that are modeled in TRACG04, the set of [[ ]] points was used to characterize the biases and uncertainties in the void coefficient as a function [[ ]]. The response surfaces for the relative biases are shown in Figure 30-2 and the response surfaces for the relative standard deviations are shown in Figure 30-3. In both figures there are [[ ]] surfaces corresponding to different void histories. For each surface the vertical axis is the in-channel instantaneous void fraction and the horizontal axis is the nodal exposure. The color scheme shown in the legends at the top of the figures denote the ranges for the biases in Figure 30-2 and the ranges for the standard deviations in Figure 30-3. A negative bias means that the TGBLA06 void coefficient is smaller in absolute magnitude than the corresponding MCNP value.

The response surfaces for the biases and uncertainties shown in Figures 30-2 and 30-3 show that in the exposure range from about 15 to 30 GWd/STU that corresponds to the limiting CPR bundle for AOO analyses that the void coefficient bias [[ ]] For exposures less than 15 GWd/STU the PANAC11 standard process as supplied with TGBLA06 nuclear information [[ ]] Also for low exposures, the uncertainties tend to be [[ ]]. As the poison is *burned* and the bundles approach their peak reactivity and power, the void coefficient biases and uncertainties [[ ]] as indicated in Figure 30-4. Figure 30-4 also shows that void history does not begin to make any discernable differences until the exposure has exceeded about 25 GWd/STU. At exposures above this point the standard process tends to [[ ]] A larger void coefficient (in the absolute sense) is conservative because it tends to produce a more dynamic power response and a less favorable CPR response. [[ ]]

The implementation of void history effects into the TRACG04 model has allowed us to demonstrate (see Figure 30-1) that the CPR response with the complete model is [[

]] The implications are that the importance of the void coefficient correction model for purposes of calculating the CPR response [[

]]

[[
]]

[[
]]

[[
]]

[[
]]

Figure 30-2. Void Coefficient Relative Bias

[[

]]
[[

]]
[[

]]
[[

]]

Figure 30-3. Void Coefficient Relative Standard Deviation

[[

]]

Figure 30-4. Average Absolute Value of Void Coefficients from MCNP

Figure 30-4 is also useful to understand the trends seen in Figures 30-2 and 30-3. Although the results and trends are shown only for the MCNP reference values, TGBLA06 values and trends are similar. In the absolute sense the void coefficient biases of TGBLA06 to MCNP are nearly constant up to about 30 GWd/STU. The relative biases in Figure 30-2 are higher for exposures less than 15 GWd/STU simply because the absolute void coefficient values to which the relative values are normalized are smaller for these exposures. The same statement applies to the relative uncertainties shown in Figure 30-3. At the higher exposures, Figure 30-4 shows that void history begins to make a discernable difference in the calculated void coefficient values from once the lattice exposures have exceeded about 25 GWd/STU (as noted previously). The *standard process* used in PANAC11 to capture these trends is based on void coefficient dependencies with respect to IV that were established at a void history of 40% (solid triangle symbols in Figure 30-4). Figure 30-4 shows that at exposures above 25 GWd/STU the standard process (solid triangles) at all IV values tends to [[

]]

[[

]] Additional detail is provided in this response in the section titled *C1AX Void Coefficient - Technical Description*.

TRACG04 internally models the response surfaces for the void coefficient biases and uncertainties in order to account for the known dominant dependencies due to relative moderator density, exposure, and void history [[ ]]. Lattices also are explicitly modeled on a nodal basis because cross sections are generated within TRACG04 using data from the lattice physics code that gets passed through via the PANAC11 wrap-up. The void coefficient biases and uncertainties are implemented in TRACG04 calculations [[

]]. Thus, the normality of the [[ ]] residual errors can be tested at each of these locations. This is what was done to get the P-values presented in Table 30-1. The Anderson-Darling test for normality was used because it is effective for small sample sizes.

A sample size of [[ ]] is too small to expect that a specific P-value for each lattice state point can be accurately determined; however, the set of all [[ ]] such values can be judged as a whole to support the conclusion that it is reasonable to assume that the residual errors are normally distributed. As shown in Table 30-1, the P-values from all the sets average to 0.432±0.277 which is well above the traditional 0.05 threshold where normality would be rejected. This conclusion has been further supported by creating a composite histogram of the standard residuals errors as shown in Figure 30-5. The composite population shown in the figure contains all [[ ]] standard residual errors that can be obtained from the database. Because this population is in standard form, it should theoretically have a mean of zero and a standard deviation of unity. Actually it does have a mean of zero but its standard deviation is 0.9358 which means that modeling it with an assumed normal distribution conservatively yields a larger variability.

Table 30-1
Normality Test P-values for the Void Coefficient Residual Errors

	%IV	0	40	70	100	%IV	
X	%VH	P Values				Avg	Stdev
[[							
							]]

[[

]]

Figure 30-5. Histogram of Standard Residual Errors with Normal Curve

TRACG04 input has been structured to allow the internally calculated uncertainties to be correlated [[

]] For most fast pressurization events, the impact of not modeling the void coefficient biases is on the order of [[ ]] in calculated values of transient ΔCPR/ICPR. Whether the bias is conservative or not depends on the exposure distribution and the relative water density distribution in the core. That is why it is important to model the bias as a function of the nodal conditions.

For sensitivity studies, a core-wide bias and uncertainty in void coefficient can be specified through the TRACG04 input. As an example of the importance of the void coefficient uncertainty, consider that for a typical BWR/4 plant a variation of [[ ]] in the void coefficient when applied to all nodes in the core corresponds to a sensitivity of [[ ]] in the ΔCPR/ICPR for a turbine trip without bypass.

C1AX Void Coefficient – Technical Description

This section is an updated version of the details that were previously provided in response to RAI 13 in NEDE-32906P-A[30-3].

The *void coefficient* (C_V) is introduced and defined as

$$C_v \equiv \frac{1}{k}\frac{\partial k}{\partial \alpha} \cong \frac{1}{k_\infty}\frac{\partial k_\infty}{\partial \alpha} \tag{6}$$

where α is void fraction, k is the neutron multiplication constant for a spatially finite geometry and k_∞ is the neutron multiplication constant for a spatially infinite geometry. Following the historical approach outlined in response to Q38 given in Volume 1 of NEDO-24154-P-A[30-4], it is instructive to envision a quadratic fit in void fraction (α) to get

$$k_\infty(\alpha) = k_\infty(0.0)\frac{(\alpha-0.4)(\alpha-0.7)}{0.28} - k_\infty(0.4)\frac{\alpha\left(\alpha-0.7\right)}{0.12} + k_\infty(0.7)\frac{\alpha\left(\alpha-0.4\right)}{0.21} \tag{7}$$

where the values of k_∞ at specified values of in-channel void fraction are determined from the lattice physics calculations. For a given TGBLA lattice at a particular exposure, k_∞ is presumed to be a smooth function of in-channel void fraction [[

]].

The essential point is that lattice physics values at discrete moderator conditions are fitted to a continuous function that can be differentiated to define the void coefficient. For example, the expression for $k_\infty\left(\alpha\right)$ in Eq. (7) is differentiated to obtain

$$\frac{\partial k_\infty(\alpha)}{\partial \alpha} = k_\infty\left(0.0\right)\frac{\left(\alpha-0.55\right)}{0.14} - k_\infty\left(0.4\right)\frac{\left(\alpha-0.35\right)}{0.06} + k_\infty\left(0.7\right)\frac{\left(\alpha-0.2\right)}{0.105} \tag{8}$$

which is a linear function of void fraction.

A similar approach is used to determine the bias and uncertainty for the void coefficient. First the calculated values of k_∞ at in-channel instantaneous void (IV) fractions [[
]] are calculated using both TGBLA06 and MCNP. [[

]] Uncertainty and bias for the void coefficient do <u>not</u> refer to errors associated with the polynomial fit rather they refer to how the TGBLA06 and MCNP void coefficient results compare to each other. Generally, the calculated values of k_{∞} from TGBLA06 and MCNP for the same particular lattice, exposure, instantaneous void, and void history point will be different; thus, an equation like Eq. (8) produces different values for TGBLA06 and MCNP. The calculated values for k_{∞} from MCNP are assumed to be the *true* values thus the void coefficients derived from them are also the assumed *true* values. In other words, any bias and uncertainty in the values of k_{∞} from MCNP due to the Monte Carlo process are assumed to be negligible compared to differences between TGBLA06 and MCNP that are presumed to be larger. Table 4.14 from Reference [30-5] was previously provided to show that for UO2 lattices the average bias in the MCNP calculated k_{∞} values compared to critical experiments is –0.0006 and the standard deviation from the mean is 0.0015. These values are much less than the corresponding values expected for the differences between TGBLA06 and MCNP.

Given that the k_{∞} values from MCNP are true, the true void coefficient (C_V) for a particular lattice at a particular exposure, void history (VH) and any specified instantaneous void (IV) is obtained from a function similar to Eq. (6) that fits the specified MCNP calculated k_{∞} values obtained specifically for that lattice at the exposure of interest. An equation like Eq. (8) can be evaluated at any desired value of α to get $\partial k_{\infty} / \partial \alpha$. Similarly, Eq. (7) can be evaluated at any desired value of α to get k_{∞}. Both equations are extrapolated to get values for k_{∞} and $\partial k_{\infty} / \partial \alpha$ at $\alpha = 1$. [[

]]

For a number (N) of different lattices (n) at different conditions m corresponding to an instantaneous void, void history, and exposure condition $\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right)$ obtain

$$C_{V,n}\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right) = \left[\frac{1}{k_{\infty}\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right)} \frac{\partial k_{\infty}\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right)}{\partial \alpha}\right]_n \tag{9}$$

for instantaneous void fractions of $\alpha_{i,m}$ [[ ]] and void history values of $\alpha_{h,m}$ [[{3}]] at every void point. These evaluations are made using separate input sets of k_∞ values from TGBLA06 and MCNP [[

]] for each lattice n and exposure (X_m). One set of evaluations is obtained from MCNP and another set of evaluations is obtained from TGBLA06. At each IV, VH and X condition a ***relative deviate*** is defined as the ratio of the void coefficient predicted by MCNP to the void coefficient predicted by TGBLA06, or mathematically

$$z_{n,m} \equiv \frac{C^M_{V,n,m}}{C^T_{V,n,m}} \tag{10}$$

where the superscript T denotes the TGBLA value and the superscript M denotes the MCNP value. As before, the subscript m denotes a particular point corresponding to $\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right)$. All N lattices are evaluated at the same particular condition m then the $z_{n,m}$ ratios for these lattices at that particular condition are averaged to define the ***mean relative bias*** for condition m as

$$\mu_m \equiv \frac{1}{N}\sum_n^N \left(1 - z_{n,m}\right) \tag{11}$$

Note that the mean relative bias is the average of all the relative biases considering all N lattices at the same condition. The value of μ_m is what we have called the ***relative bias***. By definition this bias is specified at a particular point m in the two-dimensional space defined by all voids and exposures. The $\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right)$ conditions are maintained separately. The ***relative standard deviation*** for the N samples ($\sigma_m \equiv \sigma_m\left(\alpha_{i,m}, \alpha_{h,m}, X_m\right)$) for each condition m is obtained using the common expression

$$\sigma_m \equiv \sqrt{\tfrac{1}{N-1}\sum_n^N \left[\left(1 - z_{n,m}\right) - \left(1 - \overline{z}_m\right)\right]^2} = \sqrt{\tfrac{1}{N-1}\sum_n^N \left[\overline{z}_m - z_{n,m}\right]^2} \tag{12}$$

The value of σ_m is what we have called the ***relative uncertainty***. By definition, this uncertainty is also specified at a particular point m in the two-dimensional space defined by all voids and exposures. The *N-1* form for defining the standard deviation is used because the lattices that are considered are an incomplete sample of essentially an infinite population of lattices that could be evaluated at that particular void and exposure condition.

The relative bias is the mean of the ratios between the estimate and the *truth*. Such a bias or mean of ratios is only meaningful when comparing two lattice evaluations performed for the same lattice at the same conditions, for example, at the same void fraction and exposure. There is no such thing as a "true" lattice; however, for a given lattice the *true* characterization of the lattice is assumed to be that from MCNP. The *estimate* for the lattice is the characterization obtained from TGBLA. For our purposes the key parameter from this characterization is the void coefficient (C_V). It has been shown from Eqs. (3) and (4) that the void coefficient is directly related to the lattice-physics-calculated k_∞ values at in-channel void fractions of 0%, 40% and 70%.

Equation (6) contains the mathematical expression for the relative bias at a particular (instantaneous void, void history, exposure) point. Equation (7) contains the mathematical expression for the relative uncertainty at a particular (instantaneous void, void history, exposure) point. There are [[ ]] such expressions corresponding to the [[ ]] discrete (instantaneous void, void history, exposure) points where TGBLA06 and MCNP comparisons were performed.

Statistics for the relative deviates such as the mean and standard deviation from Eqs. (11) and (12) are directly applicable for modeling the bias and uncertainty in the void coefficient. That is because the statistics as defined in the relative sense account for the fact that the lattices that are evaluated have varying amounts of reactivity. The purpose for characterizing the void coefficient bias and uncertainty is to assure that the correct change in reactivity is obtained from an associated change in void fraction.

The goal of the void coefficient correction model is to cause the reactivity impact of the void coefficient to be the same between TGBLA06 and MCNP. Mathematically, the goal is that

$$\tilde{\rho}^T = \tilde{\rho}^M \tag{13}$$

or equivalently that

$$\left(C_v \Delta\alpha\right)^T = \left(C_v \Delta\alpha\right)^M \tag{14}$$

The void coefficient values are not explicitly defined in the TRACG model, thus the only way to achieve the desired result in Eq. (14) is to modify the change in void fraction ($\Delta\alpha$) calculated by the hydraulics model before it gets applied in the evaluation of the nuclear parameters. These nuclear parameters are evaluated for each neutronics node using the nodal relative water density (u). To define u it is helpful to first define the nodal water density (ρ). In general, the nodal water density is a volume average of the water densities in the water rod, in-channel (excluding the water rod) and the out-channel.

$$\rho = \sum_{\gamma}^{3} g_{\gamma} \rho_{\gamma} \tag{15}$$

where

g_{γ} is the fraction of water volume in region γ where $\gamma = \{w, i, o\}$ corresponding to the water rod (w), in-channel (i) and out-channel (o) regions;

ρ_{γ} is the water density for region γ.

The nodal densities are defined for each axial node of each channel group. Within a particular node, the axial projection is constant over the height of the node so that volume fraction in each region is the same as the axial projection of the area fraction for that region. The water density in each region is related to the void fraction (α_{γ}) in that region by

$$\rho_{\gamma} = \alpha_{\gamma} \rho_{g,\gamma}(P_{\gamma}) + (1-\alpha_{\gamma}) \rho_{\ell,\gamma}(P_{\gamma}) \tag{16}$$

where

$\rho_{g,\gamma}(P_{\gamma})$ is the saturated steam density in region γ at the pressure (P_{γ}) of that region and

$\rho_{\ell,\gamma}(P_{\gamma})$ is the saturated liquid density in region γ at the pressure (P_{γ}) of that region.

The mean deviate and its corresponding standard deviation are modeled respectively by Eqs. (11) and (12) at discrete instantaneous void, void history, and exposure conditions corresponding to each member from the set of all pairs of $(\alpha_{i,m}, \alpha_{h,m}, X_m)$. These discrete points are assumed to be samples from a continuous distribution. The continuous distribution is constructed using the following process. [[

]]

There are minor variations between lattices for the in-channel versus out-channel volumes. These variations are accounted for by evaluating the fits in terms of instantaneous voids so that all the lattices are evaluated for the same relative water density. The functional forms are $f_{\mu_m(X_m)}(u_{\alpha_i})$ and $f_{\sigma_m(X_m)}(u_{\alpha_i})$ in order to facilitate the evaluation of $\mu_m(u)$ and $\sigma_m(u)$ in terms of the relative water density across all lattices.

This form is most convenient for use in the 3D neutron kinetics formulation used in TRACG.

Thus a function ($f_{\mu_m(X_m)}\left(u_{\alpha_i}\right)$) for each exposure that will exactly reproduce the values of $\mu_m\left(u\right)$ corresponding to [[

]] .

(17)

A similar function ($f_{\sigma_m(\alpha_{h,m},X_m)}\left(u_{\alpha_i}\right)$) created for each exposure that will exactly reproduce the values of $\sigma_m\left(u\right)$ corresponding to [[

]]. (18)

Such an equation in terms of u is obtained for each of the exposure conditions thus there is a set of [[ ]] coefficients for each exposure for both $\mu_m(u,\alpha_{h,m},X_m)$ and $\sigma_m(u,\alpha_{h,m},X_m)$. Double linear interpolation [[ ]] is used to define functions for $\mu_m\left(u,\alpha'_{h,m},X'\right)$ and $\sigma_m\left(u,\alpha'_{h,m},X'\right)$ for values of $\alpha'_{h,m}$ and X' that are not at one of specified void history and exposure points where the lattice evaluations were performed.

Next consider how the functional forms for $\mu_m(u,X_m)$ and $\sigma_m(u,X_m)$ are used. For each neutronics node the mean bias $\mu_m(u,X_m)$ and standard deviation $\sigma_m(u,X_m)$ are evaluated in terms of the nodal relative water density ($u_{k,i,j}$) and nodal exposure ($X_{k,i,j}$). The *k,i,j* subscript denotes the node indices for the axial and the two planar directions, respectively. [[

(19)

is defined. [[

]] Note that $N_{i,j}$ is a ***standard normal deviate*** that is randomly determined [[

]]. Note that a *standard normal deviate* (δ) is one from a normal distribution that has been expressed in standard form. The definition of a ***standard normal deviate*** is

$$\delta \equiv \frac{s-\mu}{\sigma} \tag{20}$$

where s is a sample value, μ is the mean for the population and σ is the standard deviation for the population.

[[

]] The trends in mean bias and standard deviation, on the other hand, have been explicitly correlated to independent lattice conditions in terms of voids and exposure so it is appropriate to apply them consistently with how they were derived.

The goal from Eq. (14) is achieved [[

]] (21)

The purpose of the void coefficient correction model is to provide a representation of the ratio C_v^M / C_v^T as a function of the nodal relative water density u and the nodal exposure X. [[

]] (22)

The ratio of C_v^M / C_v^T is available at discrete points from the lattice evaluations as suggested by Eq. (10). Although these ratios are obtained only at the discrete conditions at which the lattice evaluations were performed, the void coefficients themselves are evaluated from a continuous function obtained by substituting the expression from Eq. (8) into Eq. (6). Thus the void coefficients can be evaluated at any in-channel void fraction in the range of [0,1]. Separate such functions for the void coefficients exist for the TGBLA and MCNP lattice evaluations so that the ratio of C_v^M / C_v^T can be constructed at any desired void fraction and exposure. Similarly, the response surface defined by the evaluation of $z_{k,i,j}$ from Eq. (19) models the ratio of C_v^M / C_v^T at any desired void fraction and exposure.

The continuity of C_v^M / C_v^T is an important feature that is useful for calculating the values of C_v^M / C_v^T as $\square_i$ approaches zero. [[

]] Fortunately, the model only requires that we be able to represent the mean and standard deviation of the C_v^M / C_v^T ratio as $\square_i$ approaches zero. [[

]]

The two-dimensional response surface for the void coefficient biases for each void history $\alpha_{h,m}$ is defined from Eq. (17) and is obtained by fitting the [[ ]] discrete μ_m values obtained from Eq. (11). There are [[ ]] values because there are [[ ]] exposures and [[ ]] instantaneous relative water densities corresponding to in-channel void fractions of [[ ]]. The [[ ]] coefficients are generated for each of the [[ ]] exposures where the TGBLA-MCNP comparisons were made. For exposures in between, the [[ ]] coefficients are linearly interpolated so that fitted surface is piece-wise linear in terms of exposure.

The two-dimensional response surface for the void coefficient uncertainties for each void history $\alpha_{h,m}$ is defined from Eq. (18) by fitting the [[ ]] discrete σ_m values obtained from Eq. (12). This surface is also [[ ]] in terms of the instantaneous relative water density. The [[ ]] coefficients are generated for each of the [[ ]] exposures where the comparisons between TGBLA and MCNP were made. Linear interpolation is used to get values for exposures between the know grid lines.

TRACG has been modified so that it can evaluate the fits to these two relative surfaces in order to reproduce the statistics at the known [[ ]] points and interpolate for conditions in between. This process is repeated for each void history and linear interpolation between void histories is used to get the value corresponding to the nodal void history value as determined from the PANAC11 wrapup information. The biases and uncertainties as characterized by the two surfaces are termed ***relative*** because they have been derived as the ratio of the void coefficient predicted by MCNP to the void coefficient predicted by TGBLA as defined by Eq. (5). As such, the surfaces are in dimensionless form and are not dependent on the absolute magnitude of the void coefficient bias and uncertainty that they are used to adjust.

The mean relative bias(es) in the TGBLA-calculated void coefficient values compared to the MCNP-calculated true values have been obtained from Eq. (6) by using a sample size of [[ ]] modern 10x10 lattices. The response surface for this bias has been coded into TRACG [[

]] If all [[ ]] lattices that were sampled showed the same relative bias at each of the (void, void history, exposure) points, then there would be no *residual* error since each sample relative deviate ($z_{n,m}$ from Eq. (5)) would be related to the mean relative deviate μ_m from Eq. (6) by $z_{n,m} = 1 - \mu_m$. It follows for this scenario that the values of σ_m obtained from Eq. (7) would be zero for all [[ ]] (void, exposure) points on the sample grid for a specified void history. Thus the term *residual error* refers to that portion of the error that remains after the mean relative bias between TGBLA and MCNP is removed.

In summary, the *generic TRACG response surface* is actually two surfaces at [[ ]] different exposure histories. One surface is the fit ($\mu_m(u, X_m)$) for the mean relative deviates (or relative biases) from Eq. (15) and the other surface is the fit ($\sigma_m(u, X_m)$) for the relative standard deviations from Eq. (16). Both fits are two-dimensional fits [[ ]] in terms of relative water density (u) and piecewise-linear in terms of exposure (X). These surfaces are obtained for each of [[ ]] different exposure histories so that $\mu_m(u, X_m)$ and $\sigma_m(u, X_m)$ values can be interpolated linearly between adjacent exposure histories that bracket the nodal void history obtained from the PANAC11 wrapup.

Response References

[30-1] *Steady-State Nuclear Methods*, NEDE-30130-P-A and NEDO-30130-A, April 1985, and for TGBLA Version 06 and PANACEA Version 11, Letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: “Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods,” (TAC NO. MA6481), November 10, 1999.

[30-2] J. G. M. Andersen, et al., *TRACG Model Description*, NEDE-32176P, Rev. 4, January 2008.

[30-3] J. G. M. Andersen, et al., *TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses*, NEDE-32906P-A, Revision 3, September 2006.

[30-4] *Qualification of the One-dimensional Core Transient Model for Boiling Water Reactors*, NEDO-24154-A and NEDE-24154-P-A, Volumes I, II and III, August 1986.

[30-5] Sitaraman, S., *MCNP: Light Water Reactor Critical Benchmarks*, NEDO-32028, Class 1, March 1992.

NRC RAI 31

Section 3.1 of NEDE-32177P, "Licensing Topical Report, TRACG Qualification" (Reference 5) discusses the qualification of the TRACG void fraction predictions. FRIGG OF-64 tests simulate a full-scale 64-rod BWR fuel bundle. The test was designed as a full-scale simulation of an Oskarshamn-I fuel assembly, consisting of 64 heated rods placed in a 8x8 array. The test simulated a realistic and somewhat conservative (outlet peaked) BWR heat flux and the TRACG interfacial shear model void fraction prediction is compared against FRIGG OF-64 void fraction data.

Table 3.1-1 of NEDE-32177 (Reference 5) shows the ranges of the FRIGG OF-64 test parameter ranges as follows:

Test Parameter	**Range**
Pressure (MPA)	[[
Inlet Subcooling	
Void Fraction	
Mass Flux	]]

Table 3.1-4 of NEDE-32177 (Reference 5) provides the mean and standard deviation for TRACG Model based on the FRIGG OF-64 tests as follows:

Pressure (MPA)	**Mean**	**Standard Deviation**
[[		
		]]

The following questions relate to the applicability of the TRACG qualification ranges for operation at EPU and MELLLA+ conditions.

1. Tabulate the key thermal-hydraulic parameters (e.g. mass flux, pressure, void fractions, inlet subcooling) for the 10X10 GE14 within bundle conditions for operation at the EPU / MELLLA+ conditions, during steady state, transient and accident conditions.

2. Provide the qualification range and tests used to qualify the TRACG interfacial shear and model, if different than the above data.

Demonstrate that the qualification data supporting the TRACG interfacial shear model is applicable and acceptable for operation at EPU/MELLLA+ condition, during steady state, transient and accident conditions. Justify the accuracy of the void fraction predictions for the ranges in which qualification data is not available.

GEH Response

The information on the qualification of TRACG against the FRIGG OF64 data in the RAI text is based on the qualification of TRACG02 and taken from the TRACG Qualification Report NEDE-32177P Revision 2. The statistics has changed slightly with TRACG04 as documented in the TRACG Qualification Report NEDE-32177P Revision 3. The updated qualification results are summarized in the following response.

The key thermal hydraulic parameters for a 10X10 bundle operating at EPU/MELLLA+ conditions are given by the BWR power flow map. Specifically the flow range of interest is the range where the BWR can operate at maximum licensed power. This flow range typically ranges from 70 to 110% of rated core flow for BWRs licensed to operate at MELLLA and MELLLLA+ conditions. This corresponds to a mass flux range in the limiting fuel bundle of approximately 1000 – 1500 kg/m^2-sec (0.7-1.1 Mlb/ft^2-hr). Non-limiting lower power bundles will have lower two-phase pressure drop and a corresponding higher flow in order to match the core pressure drop. The maximum power a bundle can have is given by the operating limit minimum critical power ratio (OLMCPR) and bundle critical power as function of flow. This will typically put the limiting bundle power in the 8 MW range. It should be noted that this is always the case independent of whether the BWR is operating at uprated power conditions or not, as the core radial power peaking is always maximized in order to minimize the neutron leakage and maximize the fuel economy. EPU affect the core average bundle power and not the maximum bundle power, however the core average bundle power is closer to the maximum bundle power at EPU conditions. For a bundle operating at a mass flux of 1000 kg/m^2-sec and a power of 8 MW, the void fraction can be as high as 95%. Below the minimum MELLLA/MELLLLA+ condition the core power and the corresponding bundle power will decrease with flow along the corresponding rod line down to the natural circulation point. Natural circulation corresponds to a core flow at 30% of rated flow, which is equivalent to a bundle mass flux of 400 kg/m^2-sec (0.3 Mlb/ft^2-hr). The bundle inlet subcooling is typically in the range of 5 – 20 K (10 – 40 Btu/lbm). Rated pressure is 7 MPa (1000 psia), but the pressure can range from 5.5 – 9 MPa (800 –1300 psia) during normal operation and anticipated operational occurrences (AOO). The ranges of key thermal hydraulic parameters for a 10X10 fuel bundle are listed in Table 31-1:

Table 31-1 Typical 10X10 BWR Fuel Bundle Thermal Hydraulic Parameters.

Parameter	Range
Power	0 – 8 MW
Mass flux	400 – 1500 kg/m^2-sec
Pressure	5.5 – 9 MPa
Inlet subcooling	5 – 20 K
Hydraulic diameter	[[ ]]
Bundle void fraction	0 – 95%

The qualification of TRACG against void fraction data is documented in the TRACG Qualification Report, NEDE-32177P Revision 3. This qualification covers a wide range of conditions, but is limited to pressures at and below rated pressure. Additional qualification has been done against the FRIGG OF36 data in order to expand the qualification range to the higher pressure range. The FRIGG OF36 data which were also used in the qualification for the GE design void fraction correlation (NEDE-21565) covers a pressure range from 3 to 9 MPa. The key test parameters for the OF36 test are listed in Table 31-2:

Table 31-2 FRIGG OF36 Test Parameters

Parameter	Range
Bundle type	Full scale 6X6 bundle
Heated length	[[
Hydraulic diameter	
Pressure	
Mass flux	
Inlet subcooling	
Void fraction	]]

Comparisons were made to a total of [[ ]] void fraction data points at pressures of [[ ]]. The comparisons between the measured and calculated void fractions for the individual pressures are shown in Figure 31-1 and the comparison for the combined data set is shown in Figure 31-2

[[ ]] [[ ]]

[[ ]] [[ ]]

Figure 31-1 FRIGG OF36 Void Fractions at 3, 5, 7 and 9 MPa

[[
]]
Figure 31-2 FRIGG OF36 Void Fractions

The mean error in the predicted minus measured void fraction is [[ ]] and the standard deviation is [[ ]]. The uncertainty in the void fraction is slightly lager than the uncertainty reported for the FRIGG OF64 data, however the reported measurement uncertainty for the OF36 tests, which predates the OF64 tests, is larger; [[ ]] for the OF36 tests versus [[ ]] for the later OF64 tests. Considering the difference in the measurement uncertainty, there does not appear to be any inconsistency between the comparisons.

With the addition of the FRIGG OF36 comparisons, the total void fraction qualification basis is summarized in Table 31-3:

Table 31-3 TRACG Void Fraction Qualification Basis

Test	Pressure	Mass Flux	Inlet Subcooling	Hydraulic Diameter	Void Fraction	Mean Error	Standard deviation
	MPa	kg/m^2-sec	K	m			
OF64	[[						
Christensen							
Wilson							
Bartolomei							
EBWR							
CISE							
Toshiba							
Ontario Hydro							
OF36							]]

The expected operating range for 10X10 fuel as summarized in Table 31-1 is covered by the TRACG qualification range shown in Table 31-3. This is summarized in Table 31-4.

Table 31-4 Typical 10X10 BWR Fuel Application Versus TRAC Qualification

Parameter	10X10 BWR Fuel	TRACG Qualification
Mass flux	400 – 1500 kg/m^2-sec	[[
Pressure	5.5 – 9 MPa	
Inlet subcooling	5 – 20 K	
Hydraulic diameter	[[ ]]	
Void fraction	0 – 95%	]]

Full-scale pressure drop and critical power data have been obtained from 10X10 Fuel bundles. TRACG qualification against 10x10 pressure drop data (NEDC-32874) is documented in the TRACG Qualification Report (NEDE-32177P Revision 3). The good agreement with the measured pressure drop over a wide range of condition corresponding to the application range in Table 31-1 does not indicate any deficiency in either the void fraction model or the wall friction model. [[

]]. This uncertainty is consistent with the uncertainties from the TRACG void fraction qualification summarized in Table 31-3.

References

- TRACG Qualification, NEDE-32177P, Revision 3, August 2007.
- Critical Power and pressure Drop tests of Simulated 10X10 Bundle Designs Applicable to GE14, NEDC-32874P, March 2000.

- OF-36. Results of Void Measurements, FRIGG – PM-43, June 6, 1969.
- BWR Void fraction Correlation and Data, NEDE-21565, January 1977.

NRC RAI 32

TRACG04 is coupled with PANAC11 for neutronic feedback. Specifically, the TRACG04 steady state power distribution is initialized using the PANAC11 predicted power distribution. PANACEA uses the Findlay-Dix void fraction correlation, while the TRACG thermal-hydraulic analysis relies on the interfacial shear model to predict the void fraction. The NRC staff evaluated the Findlay-Dix correlation and determined that the database supporting the Findlay-Dix correlation is not well supported.

a. The NRC staff is concerned that the uncertainties associated with the correlation will result in additional uncertainty in the void coefficient model. Explain how the uncertainty in this correlation is accounted for in the TRACG04 analyses performed in the methodology described in Reference 3.

b. Propose a means of calculating the initial TRACG04 power and void distribution using the interfacial shear model (i.e., using PANAC11 cross sections but void and power distribution not initialized to the PANAC11 solution) and provide a code to code comparison of the "independent" TRACG04 solution to the TRACG04 solution initialized to the PANAC11 conditions (i.e., using Findlay-Dix void correlation).

c. Provide the data range used to develop the Findlay-Dix correlation and demonstrate that the experimental data covers the range of steady state, transient, EPU and expanded operating domains for which Reference 7 applies.

NRC References

3. NEDE-32906P, Revision 2, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," February 2006.

7. NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," May 2006.

GEH Response

It is true that the PANAC11 and TRACG04 models have different bases; however, both must match the same data. It is evident that PANAC11 must be reasonably successful in predicting the steady state void fraction distribution in the core because otherwise it would not be able to predict the power amplitude or shape and the exposure distribution with time. Such empirical evidence refutes the hypothesis that limited support for Findlay-Dix will translate into some deficiency in the ability of TRACG04 to analyze AOO transients. Admittedly, the question remains as to the consistency between the TRACG04 and PANAC11 void distributions. The NRC concern as we understand it is that the initialization process used in TRACG04 could mask the impact of a mismatch in the void distribution. The concern is not the void distribution per se; rather it is the impact that the steady state void distribution may ultimately have on the axial power shape transient response.

Paragraph (a) of the RAI expresses a concern with how void fraction uncertainty propagates into the void coefficient uncertainty. The uncertainty in void coefficient originates from the uncertainty in the change in reactivity to a specified change in void fraction as calculated in the lattice physics methods. [[

]] Again the concern is not with void coefficient per se; rather it is with how a void fraction uncertainty manifests itself as an uncertainty in the calculated change in power. The void coefficient simply acts as a gain on the void fraction uncertainty. The void fraction uncertainty is evaluated entirely relative to the TRACG04 model. Examples of these assessments were provided in the response to RAI 31. A potential non-conservative bias in the void coefficient due to assumptions regarding how the void fraction impacts the neutron spectrum (void history effects) is addressed separately in the response to RAI 30.

Paragraph (a) of this RAI deals with uncertainty; paragraph (b) deals with a postulated bias. To assess the impact of a potential bias the TRACG04 initialization process was modified [[

]]

For the modified initialization, the initial reference fluid density distribution is different and thus the initial power distribution is potentially different from the distribution obtained in PANAC11. To assess this difference, the modified initialization process was applied to the same EPU core used to produce the demonstration calculations in Chapter 8 of the LTR submittal. The set of initial conditions are at the EPU uprated power and increased core flow (ICF) at end-of-cycle (EOC). For this particular case the modified initialization process produced the same total power for the initiation of the transient as the original initialization process [[ ]]. At the end of the null transient the original and modified initialization processes produce the steady state relative axial power shapes in the limiting channel that are compared in Figure 32-1. The associated steady state axial void profiles for this same limiting channel are compared in Figure 32-2. These comparisons show that any postulated bias that might be inherent to the Findlay-Dix void model relative to the TRACG04 void model [[ ]]

The comparisons in the previous paragraph provide only an indirect indicator of the potential impact on the transient response. Based on the direct assessments against transient plant data provided previously in Chapter 7 of NEDE-32177P, Rev. 3, any adverse impacts from the original initialization process cannot be very significant or one would not expect all the comparisons with transient data to have turned out so well. Nevertheless, the impact of the initialization process on the transient response was directly quantified by performing a specific calculation with the modified initialization and comparing it to the identical calculation made using the original initialization. Such a comparison is made for the transient power responses from a turbine trip with no bypass (TTNB) in Figure 32-3. [[

]] The TTNB event was chosen because it tends to be one of the most limiting transient events for purposes of evaluating the change in CPR. For AOO transients the key parameter is the change in CPR (ΔCPR) over the initial CPR (ICPR). The comparison of ΔCPR/ICPR between the results from the original and modified initialization procedures is shown in Figure 32-4 for the most limiting channel. [[

]]

A minor part of the effort to assess a potential bias due to differences in TRACG04 and PANAC11 was to add a new edit in TRACG04 as illustrated in Table 32-1. This edit allows any potential bias to be assessed for each application. The values in Table 32-1 are for the particular case described above. [[

]]

Paragraph (c) of the RAI is concerned (in part) with the range of application for the Findlay-Dix model. The range-of-application concern for Findlay-Dix was addressed in the RAI responses resulting in the SER for the interim methods LTR NEDC-33173P. The range-of-application concern for the TRACG04 model is addressed in the response to RAI 31. The conclusion from the assessment provided in the response for RAI 31 was that even the hottest channel for EPU and MELLLA+ conditions remains within the qualification range of TRACG04 because the limiting channel must operate with about the same margin as quantified by its critical power ratio. The other part of the NRC concern is that any mismatch in the calculated PANAC11 versus TRACG04 void distribution resulting from the initialization process could be amplified for conditions that produce a higher average core void fraction. This concern is based on the observation that, unlike the conditions for the limiting channel that were addressed specifically in RAI 31, the feedback mechanisms that drive the transient power and flow responses are determined by the conditions of the entire core. For example, at EPU/MELLLA+ conditions more channels may be operating with higher powers and hydraulic conditions similar to those of the limiting channel; so, even if the limiting channel conditions has not changed appreciably the core environment has. To address this point, the calculations presented for rated EPU/ICF condition were repeated (using the same process) for the high-flow and low-flow corners of the power/flow map corresponding to the rod line for the EPU/MELLLA+ boundary.

The EPU/MELLLA+ calculations were performed at end-of-cycle (EOC) for the same core and exposure condition analyzed in the first part of this response and the same EPU core used to produce the demonstration calculations in Chapter 8 of the LTR submittal. The EOC exposure was selected because it is generally most limiting in terms of ΔCPR/ICPR.

The key digital values for all three power/flow state points are summarized in Table 32-2. For convenience they are labeled “A”, “B” and “C”. The figures pertaining to state point “A” corresponding to EPU/ICF at EOC have already been discussed. State points “B” and “C” correspond respectively to the upper and lower flow bounds on the EPU/MELLLA+ rod line. Table 32-2 will be discussed first before mentioning some key points from the figures that have been added for the calculations for the “B” and “C” state points that define the EPU/MELLLA+ boundary.

The upper part of the Table 32-2 simply repeats the information from the edit shown in Table 32-1 for state point "A". A different format is used so that the values can be easily compared to the similar information for the other two state points.

Table 32-2 also contains the calculated values for the nodal void fraction uncertainty. The nodal uncertainty for the mismatch in relative moderator density (u) has been transformed into an uncertainty in nodal void fraction (α) using the relationship

$$\Delta\alpha = \frac{\Delta u \cdot \rho_{ref}}{\left(\rho_g - \rho_\ell\right)}$$

where ρ_{ref} is the constant reference density used by TGBLA and PANAC for all conditions, ρ_g is the density of saturated steam evaluated at the core average pressure, and ρ_ℓ is the density of saturated water evaluated at the core average pressure. For all three analyzed state points the tabulated nodal void fraction uncertainty is less than the value used in the transient statistical analyses as pointed out previously for state point "A". This observation is simply an acknowledgement that the determination of the [[ ]] void fraction uncertainty is largely due to data measurement uncertainty and uncertainty in fitting the data and that these uncertainty elements and the actual data is common to the development of both the PANAC and the TRACG models; so, this magnitude of uncertainty is expected when comparing all the differences in nodal values between PANAC and TRACG. This observation should not be construed to imply a requirement for the following reasons: (a) the nodal void fraction uncertainty reported in Table 32-2 is a conservative estimate of the standard deviation from a set of point-by-point differences in two populations that is a factor of $\sqrt{2}$ larger than the standard deviation for each population (if the differences are random as they appear to be in these applications); (b) the void fraction uncertainty of [[ ]] as determined from TRACG04 comparisons to separate effects test data has conservatively been assigned entirely to modeling uncertainty in interfacial shear whereas the compounded void fraction uncertainty for AOO analyses has larger components due to uncertainties in other parameters such as flow, pressure, heat input, etc. that get treated separately in the transient statistical analyses and thus to some extent are accounted for twice; (c) the steady state uncertainties for initial conditions are already accounted for entirely in the SLMCPR process so the transient analyses for ΔCPR/ICPR needs only to account for how initial conditions will impact the transient response but again many of these component uncertainties are considered twice in determining the OLMCPR uncertainty; (d) the uncertainties given in Table 32-2 increases when fewer CHAN groups are used but such an increase is not correlated to a change in the calculated ΔCPR/ICPR since it is the change in moderator density or void fraction during the transient that is dominant and initial conditions are much less important (one reason for continued successful application of the historically approved single-channel, one-dimensional models). The fact that a coarser CHAN grouping does not significantly change the transient response is demonstrated in Figures 32-7, 32-8, 32-11 and 32-12 that are discussed later.

Table 32-2 also contains other key digital values. The core average void fractions and average in-channel void fractions have been listed to make the point that the lower-flow corner on the MELLLA+ line (state point "C") produces essentially the same average void fraction values as for the higher-flow point on the same rod line (state point "B"). The higher-flow point on the MELLLA+ rod line does have a higher void fraction value than the rated power at ICF (state point "A") simply because the power-to-flow ratio increases as power is maintained and flow decreases. Additional reactivity must be provided by withdrawing control blades (or some other means) to maneuver from state point "A" to state point "B". By contrast, movement from point "B" to point "C" is accomplished entirely with flow without control blade movement so power decreases so that the net reactivity due to voiding in the core does not change. No net change in the total reactivity due to voids simplistically means that the average void fraction values cannot change in any appreciable way when moving from point "B" to point "C".

The calculated limiting ΔCPR/ICPR digital values for both the original and modified initialization processes are provided (as requested) in Table 32-2 along with the changes due to changing the initialization procedure. It is most important to note that the calculated changes in the most limiting values of ΔCPR/ICPR that are shown in the bottom row of Table 32-2 are [[

]] The corresponding transient responses for power/flow state points "A", "B" and "C" in Table 32-2 are shown in Figures 32-4, 32-8 and 32-12, respectively. The figures will be discussed later. For now it is sufficient to point out that the "C" state point is very far from being the most limiting point for purposes of transient ΔCPR/ICPR.

Figures 32-1 through 32-4 corresponding to the calculations at state point "A" were discussed previously in response to part (b) of the RIA. No further discussion is needed.

Figures 32-5 through 32-8 pertain to state point "B" and Figures 32-9 through 32-12 pertain to state point "C". Both groups of figures follow the same format and order that was used for state point "A". The conclusions previously made for state point "A" also apply for state points "B" and "C"; however, there is some additional information in the figures for state points "B" and "C" that warrants additional discussion.

One point worth mentioning is that the impact on the initial axial power shape of the change in the initialization process [[

]] To put this impact into the proper perspective, the one-sigma uncertainty has been shown in these figures. The nodal power uncertainty value already accounted for in the SLMPCR is at least [[ ]] obtained by considering only the component values due to σ_{mdl} and σ_{ran} on page 2-9 of NEDE-32601P-A, revision 0. Even with this smaller value, the changes in the axial power shape due to initialization are essentially within the one-sigma band already addressed in the SLMCPR.

Figures 32-6 and 32-10 show how the steady state void distribution in the limiting channel corresponds to the small change in axial power shape. At the channel exit, the void fractions are essentially the same for the "B" and "C" state points. This observation further supports the argument that was made in RAI 31 regarding the fluid conditions for the limiting channel. A comparison with the results from Figure 32-2 reveals that the exit void fraction in the limiting channel for MELLLA+ has [[ ]] relative to the value for the highest flow at ICF.

Figures 32-5, 32-6, 32-9 and 32-10 show that any postulated bias that might be inherent to the Findlay-Dix void model relative to the TRACG04 void model [[ ]] at EPU/MELLLA+ conditions. This is the same conclusion that was made previously for the EPU/ICF case corresponding to state point "A".

Next consider the transient power responses shown in Figure 32-7 for state point "B" and Figure 32-11 for state point "C". Compare these power responses to the power response for state point "A" that is shown in Figure 32-3. It is clear from such a comparison that the transient power change becomes significantly less severe as the total core flow decreases. This is a typical trend for pressurization events in operating BWRs. Figures 32-7 and 32-11 also show results of a sensitivity study to the number of CHAN groups. These results are in addition to the comparisons between the original and modified initialization processes. The solid lines correspond to the calculations performed with [[{3}]] CHAN groups whereas the open symbols correspond to calculations performed with only [[{3}]] CHAN groups. [[ ]] the core average response dominates all the limiting AOO transient events. This is the reason that the historically approved one-dimensional models that typically model a single channel have continued to be used. The main point is that the initial uncertainty in moderator density (or voids) is largely irrelevant because the dominant influence is the transient change in moderator density and the initial absolute value contributes only in a very minor way.

Figures 32-7 also 32-11 also support the conclusion that [[ ]] This is the same conclusion that was reach for the evaluations at state point "A".

The final objective of these evaluations is to compare the calculated ΔCPR/ICPR responses and values. Digital values at the limiting point have previously been presented in Table 32-2. The transient responses are shown in Figures 32-8 and 32-12 for state points "B" and "C", respectively. Sensitivity to the CHAN grouping is shown as well as the impact due to changing the initialization process. The legends are the same as those previously described for the transient power responses in Figure 32-7 and Figure 32-11. All the ΔCPR/ICPR responses for state point "B" that are shown in Figure 32-8 [[ ]]. For state point "C" the curves in Figure 32-12 show [[ ]] This point was also made previously in discussing the precise digital values shown in Table 32-2. The transient results shown in Figures 32-8 and 32-12 for state points "B" and "C" [[ ]] This is the same conclusion that was previously supported for state point "A".

There is one final point to emphasize that has already been stated briefly several times. The digital values for ΔCPR/ICPR at the limiting point that are presented in Table 32.2 indicate that there is substantial margin at the low-flow MELLLA+ condition corresponding to state point "C". This margin is independent of other substantial process conservatisms described in the RAI 33 response that may be construed to be *reserved* to accommodate flexibility in how the core is operated. The margin described here is attributed to a milder transient power response at lower power/flow conditions that is characteristic of how BWRs respond to pressurization events. The point is made dramatically by the ΔCPR/ICPR responses that are shown in Figure 32-13. Only the results for the preferred original initialization process are shown [[

]]

Table 32-1 Example of New Edit for Rated EPU/ICF Conditions

[[

]]

Table 32-2 Summary for Three EPU/MELLLA+ Calculations

power/flow state point →	A	B	C
Total Core Power (%rated)	100	100	77.6
Total Core Flow (%rated)	104.5	85	55
[[			
			]]
Change in Limiting D/I CPR due to initialization	0.0015	0.0010	0.0121

[[

]]

Figure 32-1 Steady State Relative Axial Power Shapes in Limiting Channel for 100% Power, 104.5% Flow

[[

]]

Figure 32-2 Steady State Axial Void Fraction Profile in Limiting Channel for 100% Power, 104.5% Flow

[[

]]

Figure 32-3 Total Power Responses for Turbine Trip without Bypass from 100% Power, 104.5% Flow

[[

]]

Figure 32-4 ΔCPR/ICPR Comparison for the Limiting Channel for a TTNB from 100% Power, 104.5% Flow

[[

]]

Figure 32-5 Steady State Relative Axial Power Shapes in Limiting Channel for 100% Power, 85% Flow

[[

]]

Figure 32-6 Steady State Axial Void Fraction Profile in Limiting Channel for 100% Power, 85% Flow

[[

]]

Figure 32-7 Total Power Responses for Turbine Trip without Bypass from 100% Power, 85% Flow

[[

]]

Figure 32-8 ΔCPR/ICPR Comparison for the Limiting Channel for a TTNB from 100% Power, 85% Flow

[[

]]

Figure 32-9 Steady State Relative Axial Power Shapes in Limiting Channel for 77.6% Power, 55% Flow

[[

]]

Figure 32-10 Steady State Axial Void Fraction Profile in Limiting Channel for 77.6% Power, 55% Flow

[[

]]

Figure 32-11 Total Power Responses for Turbine Trip without Bypass from 77.6% Power, 55% Flow

[[

]]

Figure 32-12 ΔCPR/ICPR Comparison for the Limiting Channel for TTNB from 77.6% Power, 55% Flow

[[

]]

Figure 32-13 ΔCPR/ICPR Responses for the Limiting Channel for TTNB from Varying Powers and Flows

NRC RAI 33

Section 7.5.2.7, "High Worth Scram Rods for Pressurization event OLMCPR," of NEDC-32906P (Reference 3) describes the initial conditions used to minimize the worth of the scram reactivity. Section 8.0 in Reference 3, "Demonstration Analysis," covers the bases for application of TRACG for AOO, using sensitivity analyses to establish the initial conditions and assumptions that will be applied on plant-specific bases. Section 8.2 in Reference 3, "Initial Conditions and Plant Parameter Review," defines the initial conditions that are demonstrated to have an impact the AOO response.

Table 8-9 in Reference 3, "Allowable Operating Range Characterization Basis," lists the key parameters that influence the AOO response. For the axial power shape, the table states that the cases are analyzed at nominal (top-peaked) end-of-cycle (EOC) conditions and at EOC bottom peaked conditions. For the control rod pattern, Table 8-9 of Reference 3 states that cases are analyzed at middle-of-cycle (MOC) with a nominal rod pattern and with a conservative black and white rod pattern.

From this discussion, it is not apparent that for EPU and MELLLA+ operation, the assumed axialpower shapes with exposure will be conservative relative to the nominal or planned operatingcontrol rod and core flow strategies. Specifically, considering the impact of TVAP, Reference 3 did not discuss why bottom and middle peaked or double hump power profile early in the cycle will not result in higher transient response. The following RAIs relate to the use of TRACG for EPU/MELLLA+ applications.

1. For the plant-specific EPU/MELLLA+ application of TRACG04 to AOOs (References 3and 5), demonstrate that the limiting control rod patterns assumed in the power history envelops and bounds the axial power peaking the plant will experience at differentexposure ranges.
2. Discuss how the limiting control rod patterns assumed as the core depletes minimizes thescram reactivity worth.
3. Provide an assessment of TVAP that would result from the scram during power profilesother than top-peaked.

GEH Response (33-1)

It should be noted that the approach for dealing with analysis of AOOs for plants operating at EPU and MELLLA+ conditions is the same for TRACG04 / PANAC11 as compared to that which is currently approved for analysis of AOOs using TRACG02 / PANAC10. Nothing in thetransition of codes is expected to invalidate the approach used. The general trend for the calculated results for TRACG04 / PANAC11 is conservative with respect to results calculated by TRACG02 / PANAC10.

With respect to the limiting control rod patterns assumed as a function of cycle exposure, aconservative approach is used in TRACG04 / PANAC11 as is already done using TRACG02 /PANAC10 and consistent with the ODYN basis described in Reference 11.

Pressurization events are most limiting at EOC where control rods are full-out and scram

reactivity is minimized. The EOC condition is evaluated using a variation in the axial power shape at EOC through two burn strategies – a Hard Bottom Burn (HBB) and an Under Burn (UB). The main reason UB power shapes are considered is the potential effect from the Time Varying Axial Power Shape (TVAPS).

This range of exposure-dependent operational strategies (HBB to UB) is expected to bound intermediate burn strategies such that the effect of power shape deviations on the EOC power shape will be explicitly verified at both ends of the spectrum if the limiting shape can not be clearly established.

GEH Response (33-2)

At any given exposure point, there are many control rod patterns which will render the core critical and within thermal limits. To ensure that conservative values of the important dynamic parameters are calculated, it is necessary to select special control patterns. Conservative values of both the scram reactivity and dynamic void coefficient result when "black-white" control patterns are used. A black-white control pattern is one in which control rods are either fully inserted (black), or fully withdrawn (white).

The scram reactivity is minimized with black-white patterns because:

1. the fully inserted control rods provide no contribution to the scram reactivity,
2. the fully withdrawn control rods begin their insertion in a region of zero power; thus, their impact during the early portion of the scram is minimized; and
3. there are no partially inserted control rods, which generally provide a major contribution during the early portion of the scram.

The assumption of the black-white control pattern adds significant conservatism to the results. Note, the HBB strategy normally produces a more bottom peaked power shape at MOC compared to the EOC exposure. Control rod configurations with rods in the core at MOC may produce a double humped axial power shape. From review of a number of cores, it was found that double humped axial power shapes occurred for conditions with partially inserted control blades. Potentially limiting double humped power shape bundles are those very near partially inserted rods where local scram reactivity is maximized for transients. However, demonstration analyses have been performed where the partially inserted control rods are in the core and compared to the standard analysis where the "MOC" point uses the HBB with a black-white pattern. For TRACG, the results in Reference 12 Table 8-10 indicate a significant difference in the ΔCPR/ICPR between the standard analysis method (black and white control rod pattern) and the nominal case with partially inserted rods was about 0.05 for a Turbine Trip with No Bypass. Therefore, the standard process of using the HBB burn strategy with the black-white is very

conservative compared to the smaller difference that would be observed between the HBB and UB with nominal rod patterns. The process of analyzing exposure dependent limits is conservative.

GEH Response (33-3)

The principal factors controlling the severity of the TVAPS transient CPR effect are: (a) initial axial shape, (b) initial flow, and (c) plant specific MCPR timing. Cases with a more bottom peaked initial power shape will show a more severe TVAPS effect. However, the resulting operating limit is usually insensitive to the initial power shape because of the compensating effect of the increase in scram worth. Studies documented Reference 12 (see Table 8-10) show the axial power shape sensitivity (axial power shapes shown in Figure 8-35 of Reference 12). This study showed that the sensitivity was very small (0.002 ΔCPR/ICPR). As discussed in Response 33-1 and 33-2, the TRACG04/PANAC11 analysis will include consideration of both the HBB and UB axial power shape when performing the cycle specific analysis.

NRC RAI 34

(ESBWR RAI 21.6-104) Figure 4.4-31 S01-2 in MFN 06-297, Supplement 7, (Reference 11) shows that the time to boiling transition as calculated by TRACG could be non-conservative. Provide additional information demonstrating that this calculation is accurate or conservative. Explain how the uncertainty of the calculation is accounted for in TRACG04 analyses for BWR/2-6 anticipated operational occurrences and anticipated transients without scram over pressure analyses.

Reference 11:

Letter from J. C. Kinsey (GE) to NRC, MFN 06-297, Supplement 7, "Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports RAI Number 4.4-2S01, 4.4-27S01, 4.4-31S01 and 4.4-54S01", April 10, 2007 (ADAMS Accession No. ML7121061).

GEH Response.

TRACG predicts boiling transition when the critical power ratio (CPR) becomes less than unity. The CPR is defined as the critical power predicted by the GEXL critical power correlation divided by the bundle power. The GEXL critical power correlation and its uncertainty are determined from full-scale bundle critical power tests. The uncertainty of the GEXL correlation varies by fuel product line but is typically on the order of [[ ]]. This uncertainty is incorporated in the analysis of anticipated operational occurrences (AOO) as described in approved LTR NEDE-32906P-A, Revision 3, Section 5, C13. This reference quotes an average error of [[ ]] and a variance of [[ ]] in the prediction of transient change in CPR over initial CPR (ΔCPR/ICPR) for simulated AOO transient events. This deviation is covered by the uncertainty in the GEXL correlation. The comparisons to transient tests have been repeated with TRACG04 and are documented in NEDE-32177P, Revision 3. These comparisons showed and average error in the ΔCPR/ICPR of [[ ]] and a standard deviation of [[ ]]. This deviation is covered by the uncertainty in the GEXL correlation.

For application to AOO evaluations the specified acceptable fuel design limit (SAFDL) as required by GDC 10 is that the fuel cladding integrity shall be maintained by ensuring that the minimum CPR (MCPR) remains above the safety limit MCPR (SLMCPR) based on acceptable correlations. Therefore the critical safety parameter is the MCPR or the transient ΔCPR for the event. The qualification of the prediction of transient ΔCPR is documented in NEDE-32177P, Revision 3 as summarized in the previous paragraph. Boiling transition will not occur since the MCPR remains above the SLMCPR and therefore the time to boiling transition is not relevant for AOO events.

Anticipated transients without scram (ATWS) overpressure events are analyzed for the purpose of determining the peak reactor vessel pressure. The approved methodology for these analyses is described in NEDE-32906P Supplement 1-A. The peak reactor vessel pressure depends on the overall system response and on the overall system mass and energy balance. A small fraction of the highly peaked fuel bundles in the core may experience boiling transition, which is acceptable during an ATWS overpressure event, but boiling transition in a few bundles will have no noticeable impact on the overall system response and the peak reactor vessel pressure. This was confirmed by the sensitivity studies documented in NEDE-32906P Supplement 1-A. Therefore ATWS overpressure events are not sensitive to boiling transition or to the time to boiling transition.

The question on the uncertainty in the prediction of the time to boiling transition remains. The GE14 transient test used to quantify the accuracy of the margin to boiling transition can also be used to quantify the accuracy of the time to boiling transition. In the TRACG calculation the time to boiling transition is taken as the time of the MCPR for a test simulation having a calculated MCPR of 1.00. For the experiments it is more difficult to determine the time to boiling transition. During a typical pressurization event the rector pressure increases from 7 to 8.3 MPa (1000 to 1200 psia). The corresponding increase in the saturation temperature and coolant temperature in the two-phase region of the bundle, where boiling transition will occur, is 12 C (22 F). At the same time the surface heat flux increases by 30 – 50 %. This corresponds to an increase in the wall superheat of 14-22% assuming that the heat flux is proportional to the wall superheat squared. The GE14 transient tests show a wall superheat of approximately [[ ]] for the initial conditions prior to the transient event and therefore an additional [[ ]] increase in the wall temperature is expected. In summary, a total increase in the wall temperature of [[ ]] would be expected without experiencing boiling transition. At the time of boiling transition the temperature will start to increase at a faster rate, which will show up as a slight inflection in the slope of the temperature versus time. The times for this change in slope has been estimated from tests and are given in Table 34-1 as the first indication of BT. The exact determination of this time from thermocouple readings, when the temperature excursion is mild, is not very accurate for these pressurization events. The increased rate of change may be difficult to discern from the noise in the readings. As the power in the test is increased, the temperature slope change and additional temperature rise due to BT will become more pronounced, i.e., the effective signal-to-noise ratio improves. In NEDC-32084P-A, which documents the qualification of TASC against the transient tests, the boiling transition was therefore determined as the time when a [[ ]] increase in the surface superheat was observed. [[

]]. These times are also shown in table 34-1. The span from the time for the first indication of BT and the time from NEDC-32084 characterizes the uncertainty in the determination of the experimental time to BT. Finally, the corresponding times to BT as predicted by TRACG are also given in Table 34-1.

Table 34-1. GE14 Transient Test – Time to Boiling Transition

GE14 transient test	First indication of BT	NEDC-32084P-A	TRACG
TTNBT/NRPT	[[		
TTNBP/RPT			
ABWR pump trip			]]

The comparison of the calculated versus measured times to boiling transition is also shown in Figure 34-1. On the basis of these comparisons, the uncertainty in the TRACG prediction of the time to boiling transition is 0.5 sec.

[[
]]
Figure 34-1. Time to Boiling Transition

The key points from the preceding discussion may be summarized as:

- The MCPR remains above the SLMCPR for AOO transients. Therefore the margin to BT is the relevant parameter rather than the time to boiling transition.

- Boiling transition may occur in high power bundles for an ATWS over pressure event, but the peak reactor vessel pressure is insensitive to this phenomenon.
- In addition to the points made on the basis of the above discussion, it should be noted that time to boiling transition is important for the early BT for a BWR/2-6 design basis loss of coolant accident (LOCA). The early boiling transition is caused by the fast core flow coast down. TRACG conservatively under predicts the time to BT for a fast flow reduction as demonstrated by the ABWR all pump trip test. The sensitivity to uncertainty in the time to BT will be addressed in the TRACG BWR/2-6 LOCA Application Methodology report. For the ESBWR LOCA there is no early boiling transition or core uncovery.

References.

- TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, NEDE-32177P-A, Revision 3, September 2006.
- TRACG Qualification, NEDE-32177P, Revision 3, August 2007
- TRACG Application for Anticipated Transient Without Scram Overpressure Transient Analyses, NEDE-32177P Supplement 1-A, November 2003.
- TASC-03A, A Computer Program for Transient Analysis of a Single Channel, NEDC-32084P-A, Revision 2, July 2002.