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.

<u>RAI 7</u>

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

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

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

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.

<u>RAI 8</u>

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.

<u>RAI 9</u>

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.

<u>RAI 23</u>

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

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

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

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

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

<u>RAI 24</u>

The TRACG04 flow regime map and entrainment model were modified to improve agreement with low pressure Toshiba data. The purpose of the update was to improve calculational accuracy for the ESBWR LOCA calculations. In response to this RAI, the NRC staff requested information regarding the uncertainty analysis with respect to the void fraction. The TRACG02 and TRACG04 codes were assessed against the full-scale bundle test facility (FRIGG) void fraction data measurements of [ ]. The assessment indicates that the TRACG02 code predicts the void fraction with a mean error of [ ] and a standard deviation of [ ]. TRACG04 predicts the void fraction with a mean error of [ ] and a standard deviation of 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.

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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.

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

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.

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

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
[

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

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

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

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

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

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

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 [

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

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

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

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

[

], which is well beneath the threshold of significance.

As the purpose of the analysis in the response is to quantify the bias introduced in the transient response as a result of the deterministic nodal void fraction mismatch, the NRC staff does not find the one standard deviation uncertainty band to be an acceptable basis for the threshold of significance. The interdependence of the void reactivity coefficient uncertainty and void fraction uncertainty is not explicitly accounted for in the uncertainty analysis. Therefore, the NRC staff 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.

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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			]

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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.

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