APPENDIX K – SAFETY EVALUATION OF SUPPLEMENT 3 TO NEDC-33173P

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NEDC-33173P, SUPPLEMENT 3

"APPLICABILITY OF GE METHODS TO EXPANDED OPERATING DOMAINS –

SUPPLEMENT FOR GNF2 FUEL"

GE-HITACHI NUCLEAR ENERGY AMERICAS, LLC

PROJECT NO. 710

1.0 INTRODUCTION AND BACKGROUND

The interim methods licensing topical report (NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains", hereafter "IMLTR") provides the basis for the application of the suite of GE-Hitachi (GEH) and Global Nuclear Fuel (GNF) computational methods to perform safety analyses relevant to extended power uprate (EPU) and maximum extended load line limit analysis plus (MELLLA+) licensing (Reference 1). The U.S. Nuclear Regulatory Commission (NRC) staff approved the IMLTR with a limitation in its safety evaluation (SE) that the NRC staff's review was applicable only to GE14 and earlier GE fuel designs (Reference 2).

Recently GNF has developed an advanced fuel design, GNF2 (Reference 3). By letter dated July 31, 2009, GEH requested that the NRC staff review and approve Supplement 3 to the IMLTR, "Supplement for GNF2 Fuel" (Reference 4). This IMLTR supplement (hereafter Supplement 3) provides the basis for the extension of the applicability of the suite of GEH/GNF methods to analyze cores operating at EPU and MELLLA+ conditions with GNF2 fuel.

The NRC staff has previously audited the GNF2 fuel design to ensure compliance with the General Electric Standard Application for Reload Fuel (GESTAR II) process (Reference 5). The NRC staff's audit findings are documented in References 6 and 7. This audit addressed the topics of fuel thermal-mechanical (T-M) performance, neutronic performance, and critical power performance. During this audit, the NRC staff identified several open items in the area of T-M design and analysis. To this end, GNF has addressed the NRC staff open items on an interim basis through Amendment 32 to GESTAR II (Reference 8). To address the NRC staff open items regarding the T-M design and analysis, GNF has imposed an exposure limit for the GNF2 fuel design. The NRC staff reviewed this exposure limit and found that the limit adequately addresses the NRC staff concerns regarding the T-M performance (Reference 9).

ENCLOSURE 2

However, this exposure limit is established to address open items and technical concerns regarding the continued applicability of the GSTRM T-M analysis methodology to the advanced GNF2 fuel design. The NRC staff has previously imposed Limitation 12 on the IMLTR through its approving SE, which requires, in part, that future EPU and MELLLA+ licensing analyses be performed using updated, approved T-M methods. The NRC staff reviewed the PRIME T-M methodology and documented its approval in its SE dated January 22, 2010 (Reference 10).

Consistent with IMLTR Limitation 12 and IMLTR Supplement 4 (Reference 11), it is the understanding of the NRC staff that since PRIME has been approved, future licensing evaluations for GNF2 in EPU and MELLLA+ cores will be performed using the updated PRIME T-M methods. GNF documented its agreement with this understanding in a letter to the NRC dated May 27, 2010 (Reference 12). Noting this expectation, but given that the PRIME T-M methodology was still under NRC review when the GNF2 methods applicability supplement to the IMLTR (Reference 4) was submitted, the NRC staff understands that this IMLTR supplement needed to address the interim GESTAR II Amendment 32 approach as well as an approach that accounts for the use of updated T-M methods now that PRIME has been approved by the NRC staff.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.34, "Contents of applications; technical information" provides requirements for the content of safety analysis reports for operating reactors. The purpose of the IMLTR is to provide a licensing basis that allows the NRC to issue SEs for expanded operating domains including constant pressure, EPU, and MELLLA+ applications. The SE for the IMLTR approves the use of GEH/GNF methods for expanded operating domains. Licensee's applying for EPU or MELLLA+ licensing amendments may refer to the IMLTR as a basis for the license change request regarding the applicability of GEH/GNF methods to the requested changes.

In its SE the NRC staff included several limitations and conditions to specify its approval of the IMLTR. Licensees referencing the IMLTR must demonstrate compliance with the limitations and conditions to ensure that the licensee-specific application of the IMLTR is within the scope of the NRC staff's approval.

Limitation 22 from the NRC staff SE for the IMLTR states that the review of the IMLTR is only applicable to GE fuel designs up to GE14. Therefore, the introduction of the GNF2 fuel design requires NRC review of the applicability of the IMLTR to the GNF2 fuel design. The NRC staff reviewed Supplement 3 only insofar as it justifies a revision to Limitation 22. The NRC staff review in this matter does not impact any other aspects of the original review of the IMLTR. Therefore, all other NRC staff guidance, limitations, and conclusions documented in the SE for the IMLTR remain applicable as originally stated.

3.0 TECHNICAL EVALUATION

Supplement 3 follows the same format as the original IMLTR. This ensures consistency and completeness in the GNF2-specific documentation relative to the original information submitted for NRC review and approval for the earlier GNF fuel designs (e.g., GE14). Therefore, the NRC staff has documented its review of the relevant topics following the same format as the SE for the IMLTR. The review topics consider: (1) extrapolation of the neutronic methods to high void fractions, (2) the 40 percent void fraction depletion assumption, (3) bypass and water rod voiding, (4) stability, and (5) applicability of the thermal-hydraulic models.

Where applicable the NRC staff compared the GNF2 fuel design to the GE14 fuel design to gauge the applicability of previous review findings so as to leverage its experience in reviewing the original IMLTR. Additionally, the NRC staff leveraged experience from its audit of the GNF2 GESTAR II compliance documentation.

3.1 Comparison of GNF2 to GE14

The major differences between the GNF2 and GE14 fuel designs are the part-length rod (PLR) placement and design and the placement and design of the grid spacers. In terms of the PLRs, the GNF2 design includes two different lengths of PLRs, whereas GE14 PLRs are of uniform axial length. Additionally, some of the GNF2 PLRs are included at the lattice edge, which is a novel feature of the GNF2 design. In terms of the grid spacers, the GNF2 design is made entirely of Alloy X-750, whereas the GE14 spacer design is a zircaloy ferrule design with Alloy X-750 springs.

Another difference is in the GNF2 fuel pin design, which incorporates a slightly larger fuel pellet and thinner cladding relative to GE14. Additionally, the geometric stacking factor of GNF2 is slightly higher than GE14 fuel. This results in a slightly higher overall heavy metal loading for the GNF2 fuel design relative to GE14 ([ ] for GNF2 relative to [ ] for GE14).

3.2 Extrapolation of Neutronic Methods to High Void Fractions

3.2.1 Neutronic Methods Assessment

The NRC staff reviewed the relevant assessment of the neutronic methods for applicability to GNF2 fuel. The NRC staff notes that several design features of GNF2 are expected to affect neutronic performance relative to the GE14 fuel design. The most prominent of these design changes are the design of the PLRs, placement of the PLRs, and the change in the fuel rod dimensions. Therefore, the NRC staff considered assessment data similar to those data provided in the IMLTR for GE14 fuel to determine the acceptability of applying the current GEH/GNF methods to neutronic and systems analysis of GNF2 at EPU and MELLLA+ conditions.

3.2.1.1 Cold Eigenvalue

Cold eigenvalue calculations are performed to determine the shutdown margin (SDM) on a cycle-specific basis. The uncertainties in the calculation of the lattice nuclear parameters affect the ability of the core simulator (PANAC11) to predict the reactivity of the core under various conditions, such as control state and temperature.

Supplement 3 provides the results of a series of local cold critical eigenvalue measurements performed for a 240-bundle boiling water reactor (BWR) operating with annual cycles (Plant A). The NRC staff has previously audited the Plant A cold critical tests as part of the GNF2 GESTAR II compliance audit. Several of these tests were local cold critical tests. Under these conditions, the core is predominantly fully controlled and the control blade is withdrawn from one location until the locally uncontrolled region approaches criticality. Several of the tests performed at Plant A were conducted with the local blade withdrawn at the location of GNF2 lead use assemblies (LUAs). These tests provide a direct qualification of the capability of PANAC11 to predict the eigenvalue under cold conditions with one control blade withdrawn. These are essentially the calculations that are performed to determine the SDM.

The NRC staff has reviewed these qualification data and confirmed that the uncertainty in the cold eigenvalue predictions is not sensitive to the presence of the GNF2 bundle. Since the Plant A tests were local cold critical tests, they provide direct relevant qualification of the cold SDM calculation capability of PANAC11 with GNF2 fuel. Therefore, the NRC staff concludes that the uncertainties identified for GE14 cold critical eigenvalue determination remain applicable to analyses performed for GNF2 fuel. The consistency between the GE14 and GNF2 local cold critical results is shown in Figure 2-12 of Reference 4. GEH has adequately demonstrated that the performance of the methods in terms of predicting the cold critical eigenvalue is essentially the same for GE14 and GNF2.

The NRC staff requested that GEH confirm that Plant A from Supplement 3 is equivalent to Plant C from the IMLTR in RAI-1. The NRC staff requested this information to confirm that the local cold critical measurements were performed for the EPU plant (operating at 110 percent originally licensed thermal power (%OLTP)). The response to RAI-1 confirms that Plant A is the same as Plant C from the expanded database (Reference 13). The Plant C core is an EPU core and thus confirms the local cold critical eigenvalue calculation for GNF2 fuel at EPU conditions.

3.2.1.2 Hot Eigenvalue

The hot critical eigenvalue is a measure of the bias in the PANAC11-predicted core steady-state multiplication factor. When performing core tracking evaluations, the reactor remains in a critical state (steady-state); however, the core simulator may predict an eigenvalue that differs from unity. To account for methodology biases, a design basis hot critical eigenvalue is established. When performing cycle depletion calculations, the design basis hot critical eigenvalue is used to bias the core simulator to impose a critical condition at a multiplication factor that differs from unity. These biases are established based on calculational benchmarking and operating experience.

Supplement 3 addresses the adequacy of the design basis hot critical eigenvalue by providing qualification of the core simulator method to predict the eigenvalue consistent with known critical conditions for a BWR plant operating with a reload of GNF2 fuel. The design basis hot critical eigenvalue curve provided in Figure 2-3 of Reference 4 is typical of the current operating fleet and modern fuel designs. The core tracking calculations performed using PANAC11 for known critical conditions indicate that the trend in eigenvalue and the magnitude of the eigenvalue are fully consistent with the imposed design basis bias for the early portion of core exposure. This includes data obtained with a large fraction of GNF2 in the core loading. The consistency through the early portion of cycle exposure confirms that the expected trends in hot critical eigenvalue are insensitive to the presence of large batch quantities of GNF2 fuel. On this basis, the NRC staff is reasonably assured that the design differences between GE14 and GNF2 are sufficiently subtle that the accuracy of the methods used to predict the hot critical eigenvalue and hot critical design basis eigenvalue is not compromised for GNF2 fuel relative to GE14.

3.2.1.3 Traversing In-core Probe (TIP) Measurements

During its audit of the GNF2 GESTAR II compliance, the NRC staff reviewed several TIP data collected near GNF2 LUAs. The NRC staff review of these data is documented in Reference 6. The NRC staff found that the neutronic differences between GE14 and GNF2 were sufficiently small that the axial power shape predictive capability of PANAC11 was not challenged.

These TIP data, however, were limited in scope as only local data were useful in categorizing the calculational efficacy of PANAC11 in terms of GNF2 modeling. Supplement 3 provides additional qualification data for a BWR/4 plant with a reload of GNF2. The GNF2 batch fraction for this plant was 29 percent. Three TIP measurements were performed during the early part of the cycle with GNF2 fuel loaded in the core. These three measurements were analyzed by GNF.

GNF considered separately the TIP data collected for four-bundle cells that contain only GE14 fuel, two GNF2 bundles, and three GNF2 bundles. Comparison of the results for these three cases is shown in Table 2-3 of Supplement 3. The results confirm that the TIP radial biases and uncertainties are not sensitive to the number of GNF2 bundles in the TIP cell. When considered with the global TIP statistics provided in Table 2-2 of Supplement 3, the four-bundle power biases and errors are well within those established for GE14 during the methods qualification provided in the IMLTR. The integrated radial TIP root mean squared (RMS) difference was found to be [ ] percent when all three TIP measurements are considered. This value is well below the [ ] percent σ_{P4B} (four-bundle power uncertainty) established in the IMLTR for the expanded EPU database (Reference 1) and below the [ ] percent used in the development of the safety limit minimum critical power ratio (SLMCPR) (Reference 14).

Table 3.2.1.3.1 provides a summary of the TIP data comparison to historically determined uncertainties. These include the original uncertainties reported in NEDC-32694 (Reference 14) for TGBLA04/PANAC10 (T4/P10) methods as well as subsequent requalification in

NEDC-32773P, Revision 1 (Reference 15) for TGBLA06/PANAC11 (T6/P11) methods, IMLTR (Reference 1) for expanded operating domains, and Supplement 3. The Supplement 3 radial RMS differences are reported for TIP data for the three cases mentioned above (i.e., all GE14 bundles, two GNF2 bundles, and three GNF2 bundles per string). The results confirm that the predictive capability for GNF2 is demonstrated to at least match the predictive capability for GE14. The NRC staff notes that these radial TIP data represent a small sample, and therefore cannot be used to definitively show improved accuracy. However, the NRC staff is reasonably assured based on the good agreement between the predictions and measurements that σ_{P4B} for GNF2 fuel is not greater than the uncertainty for GE14 fuel.

Table 3.2.1.3.1 Radial Power Shape GNF2 Qualification and Comparison

Document	Nuclear Model	# TIP Sets	Weighted RMS Differences [%]
NEDC-32694	T4/P10	[	
NEDC-32773 Rev. 1	T6/P11		
NEDC-33173	T6/P11		
NEDC-33173 Supplement 3	T6/P11		
NEDC-33173 Supplement 3 (2 GNF2 bundles)	T6/P11		
NEDC-33173 Supplement 3 (3 GNF2 bundles)	T6/P11		]

In terms of the axial power shape modeling, the axial RMS TIP differences were also provided in Supplement 3. The NRC staff compared the axial RMS differences to the qualification data audited by the NRC staff during its review of the LUA experience as part of the GNF2 GESTAR II compliance audit (Reference 6). The NRC staff found that the axial RMS differences were consistent. Table 3-6 of Reference 6 provides a direct comparison of axial TIP statistics for GNF2 LUAs with core average axial TIP statistics. Table 3-6 shows that the presence of a GNF2 LUA does not affect the axial TIP RMS differences – the average axial TIP RMS difference for a GNF2 LUA TIP string is reported as [ ] percent as compared to a core average value of [ ] percent. The NRC staff reviewed the expanded TIP data in

Supplement 3 and found that it demonstrates consistent performance of PANAC11 to model the axial power shape with increased quantities of GNF2 fuel bundles loaded in the core. The axial TIP RMS difference based on the three TIP measurements is reported in Supplement 3 as [ ] percent for a core with a 29 percent GNF2 batch reload. This is fully consistent with the GNF2 LUA string values collected over longer cycle durations and is consistent with core average quantities reported for predominantly GE14 loaded cores.

On the basis of the previously audited GNF2 LUA TIP measurements and the few TIP measurements collected for a core operating with a reload quantity of GNF2, the NRC staff concludes that there are no discernable biases in the predictive capabilities of the neutronic methods for GNF2 relative to GE14.

3.2.1.4 Monte Carlo N Particle Transport Code (MCNP) Comparisons

Supplement 3 provides a comparison of TGBLA06 lattice physics calculations to MCNP calculations at two exposures (0 and 65 gigawatt-days per metric tonne uranium (GWD/MTU)). These calculations were performed to demonstrate the performance of the TGBLA06 method to model GNF2 lattices relative to its modeling of GE14 lattices. MCNP serves as a higher order method to quantify uncertainties and biases attributed to the solution technique of TGBLA06. The NRC staff accepts the use of MCNP to provide a detailed transport solution such that uncertainties in the TGBLA06 method may be assessed. Therefore, these code-to-code comparisons become a gauge of the uncertainty in the calculation introduced by the assumptions, approximations, and spatial discretization of TGBLA06. The purpose of these comparisons is to test if the design features of GNF2 result in exacerbated uncertainties associated with the method.

The basis for comparison includes the infinite lattice reactivity and the fission density distribution. The infinite lattice reactivity serves as a surrogate metric to quantify any biases or uncertainties in the predictive capability in terms of downstream nodal reactivity calculations. Likewise, the fission density comparisons serve as a surrogate for pin power distribution. These quantities may be directly compared and are closely related to those parameters considered in the safety analysis. The pin power distribution uncertainties, for instance, are propagated to determine uncertainties in the linear heat generation rate (LHGR) and the R-factor. These parameters are utilized in assessing the margin to the LHGR and critical power ratio (CPR) thermal limits.

Supplement 3 compares the GNF2 MCNP/TGBLA06 infinite eigenvalue and fission density calculations to the standard deviation predicted for GE14 lattices. According to the response to RAI-5 (Reference 13), the exposure calculations were performed for a consistent void history of 40 percent. The intent of these comparisons is to demonstrate that the performance of TGBLA06 in terms of modeling capability for GNF2 is essentially identical to the capability for GE14. To this end, the RMS differences in GNF2 lattice calculations at various exposures and void fractions are compared to the one-standard-deviation band of previous results for GE14. The collection of these code-to-code comparisons is provided in Figures 2-1, 2-2, 2-4, and 2-5 of Supplement 3. In response to RAI-6, GEH revised these figures to correct the location of the data points for consistency with the independent axis (relative water density) (Reference 13).

These figures demonstrate that the trends in, and magnitude of, uncertainties for GE14 and GNF2 are fully consistent and essentially equivalent.

On these bases, the NRC staff concludes that the design differences of GNF2 relative to GE14 do not present a challenge to the TGBLA06 lattice physics method that would incur increased uncertainties in the relevant nuclear data calculations over the range of void conditions where TGBLA06 is exercised. However, the NRC staff notes that only uncontrolled conditions were considered in the code-to-code comparisons. Therefore, the NRC staff requested additional information regarding the relative performance under controlled conditions in RAI-2. The response to RAI-2 provides Figures 2-5 and 2-6 (Reference 13). These figures show the difference between TGBLA06 and MCNP for beginning-of-life (BOL) controlled conditions. Figure 2-5 compares the infinite eigenvalue difference between TGBLA06 and MCNP for GNF2 lattices to the GE14 average standard deviation. The NRC staff notes that at high void fraction (70 percent) the TGBLA06 calculations for the GNF2 lattices indicate a slightly higher eigenvalue compared to the GE14 calculations at the same void fraction. Void fractions of 90 percent were not considered as part of the analysis. The NRC staff notes that controlled conditions with very high void fraction (90 percent) are not expected due to the power suppression induced by the control blade. The NRC staff reviewed the differences at high void fraction and found that the standard deviation in the GNF2 calculations was somewhat smaller than for the GE14 lattices. This is depicted in the difference in range of the dashed curves between Figures 2-5 and 2-6 from the response to RAI-2. The NRC staff notes that the small bias in the high void fraction TGBLA06 GNF2 calculations is bounded by the two standard deviation range of the GE14 lattices and further notes that these biases do not impact calculations of shutdown margin (since these calculations are performed at cold conditions).

The NRC staff reviewed the GNF2 TGBLA06/MCNP qualification for controlled conditions and found that the calculations demonstrate essentially equivalent performance for GNF2 and GE14 lattices. Therefore, the NRC staff finds that TGBLA06 controlled calculations have been adequately demonstrated for the GNF2 fuel design.

The NRC staff reviewed the relative performance of the extrapolation of the neutronic methods to higher void fractions. The NRC staff requested in RAI-2 that the polynomial TGBLA06 fit for GNF2 be compared to MCNP calculations at high void fraction and compared to similar results for GE14 to demonstrate consistent extrapolation uncertainties. The response to RAI-2 provides Figures 2-1 through 2-4 (Reference 13). These figures are substantially similar to lattice infinite eigenvalue figures shown in Supplement 3. However, these figures include a comparison of the extrapolated eigenvalue to MCNP calculations at 90 percent void fraction. Since the TGBLA06 results are utilized in PANAC11 by means of a response surface that extrapolates nuclear data beyond 70 percent void fraction, the NRC staff finds that this comparison is useful in assessing the accuracy of the nuclear design methods in determining the nuclear characteristics of nodes at high void fractions.

These comparisons considered BOL conditions and exposure to 65 GWD/MTU at 40 percent void fraction. The NRC staff reviewed the trend in the eigenvalue differences between TGBLA06 and MCNP. In each case, the GNF2 results were within the range of accuracy previously demonstrated for GE14 lattices. Therefore, these figures demonstrate the continued adequate performance of TGBLA06 to generate nuclear data for GNF2, even considering the extrapolation to very high void fractions (90 percent). It is worth noting that the GNF2 lattice results indicate a smaller standard deviation at higher void fractions. The results for the GNF2 VAN1 lattice (vanished region above the short PLRs (SPLRs)) indicate a larger bias than the other lattices; however, this single case remains bounded by the two standard deviation range based on the GE14 qualification.

In its review, the NRC staff considered TGBLA06 calculations that were performed as part of the GESTAR II licensing for GNF2. These calculations are provided in the GESTAR II Compliance Report for GNF2 (Reference 16). The NRC staff found subtle differences in the predicted results and requested additional information in RAI-9 regarding the inconsistency between the Supplement 3 calculations and those provided in the GESTAR II Compliance Report. The response to RAI-9 states that the calculations in the compliance report were performed with an earlier version of TGBLA06 that did not include two modifications that were implemented to improve the accuracy of the code - namely the corrected Dancoff factor calculation and the improvement to the low-lying resonance treatment for plutonium (Reference 13). The magnitude of the differences observed between the calculations provided in the GESTAR II Compliance Report and Supplement 3 was consistent with the NRC staff's expected deviation on the basis of these code modifications. Additionally, the Dancoff factor correction is necessary to adequately model the GNF2 fuel lattices with edge PLRs. The RAI-9 response confirms that Supplement 3 calculations were performed with the most recent standard production version of TGBLA06. Therefore, the NRC staff relied on the calculations provided in Supplement 3 to reach its conclusions.

On the basis of these assessments, the NRC staff concludes that the performance of TGBLA06, including extrapolation to very high void fraction, remains consistent for GNF2 fuel lattices relative to GE14 fuel lattices.

3.2.1.5 Uncertainties

On the basis of the qualification provided in Supplement 3 and the GNF2 GESTAR II Compliance Report, the NRC staff considered those power distribution uncertainties that are treated in the calculation of the SLMCPR to confirm the continued applicability of the interim approach to analyses performed on GNF2 fueled EPU or MELLLA+ cores.

3.2.1.5.1 Pin Power Peaking Uncertainty

The pin power peaking uncertainty, also referred to as the infinite lattice pin power peaking uncertainty, in the interim approach is determined according to a [ ] (Reference 1). The NRC staff has reviewed this interim approach in its review of the IMLTR and found that this approach is acceptable to account for potentially increased uncertainties in the local power distribution at high void conditions typical of EPU or MELLLA+ conditions. The [ ] value was then propagated into the SLMCPR uncertainty analysis to determine a conservative SLMCPR penalty. [

]

Confirmatory calculations performed for GNF2 lattices using TGBLA06 and MCNP confirm that the pin-wise fission density uncertainty is consistent with those for GE14 lattices. To a certain extent, the accuracy in the TGBLA06 calculations is attributed to code updates that have enabled the accurate calculation of Dancoff factors for edge rods. This modification is necessary to accurately calculate the pin power distribution for the GNF2 lattice noting the presence of PLRs at the lattice edge. The NRC staff has previously audited the TGBLA06 updates that have enabled this calculation and found these code modifications acceptable (Reference 6). Therefore, the NRC staff finds that GEH has adequately demonstrated that the pin peaking uncertainties for GNF2 are essentially the same as those for GE14. Therefore, the [ ] remains a valid basis for bounding the impact of potentially increased power distribution uncertainties.

The pin power peaking uncertainty also affects the LHGR limit. The NRC staff found that use of the uncertainty determined by the [ ] approach remains applicable to GNF2 fuel. Therefore, the NRC staff finds that the [ ] assumed in the GSTRM analysis remains bounding of the uncertainty for GNF2 fuel.

3.2.1.5.2 Four-Bundle Power Uncertainty

The four-bundle power uncertainty (σ_{P4B}) used in the SLMCPR calculation has been justified for GNF2 fuel for EPU and MELLLA+ licensing evaluations. TIP measurements were performed for GNF2 LUAs and GNF2 core reloads. The results of the comparison of these TIP data to PANAC11 calculations confirm that the radial uncertainties are consistent with the radial uncertainties for earlier GNF fuel products (e.g., GE14). Therefore, the NRC staff finds that the value of σ_{P4B} remains acceptable for GNF2.

3.2.1.5.3 [ ]

[

]
approach to quantify the SLMCPR impact associated with potentially increased power distribution uncertainty at EPU or MELLLA+ conditions.

The NRC staff reviewed the continued applicability of the [ ] approach to GNF2 fuel. GEH did not specifically provide a GNF2 qualification with regard to [ ]. However, calculations performed for relevant nuclear parameters (infinite eigenvalue) using MCNP and TGBLA06 confirm that uncertainties in the nodal reactivity for GNF2 fuel are essentially the same as for GE14 fuel. Additionally, the assessment of the radial TIP data indicates that the four-bundle power calculation is not sensitive to the number of GNF2 bundles present in the four-bundle set. The NRC staff reached a similar conclusion during its review of the GNF2 LUA

TIP data when the NRC staff assessed the four-bundle power measurements as a function of the GNF2 calculated relative power distribution (Reference 6).

When the TIP radial data are considered with regard to the presence of different numbers of GNF2 bundles and the relative power of those bundles, these data indicates insensitivity in the four-bundle power. This provides assurance that there are no significant biases introduced in the calculation of the [ ] associated with the GNF2 bundle. When considered in concert with the computational benchmark using MCNP, which confirms consistent performance of TGBLA06 relative to GE14 calculations, the NRC staff is reasonably assured that GNF2 is sufficiently similar to GE14 that the [ ] does not increase. Therefore, the NRC staff finds that the [ ] remains equally applicable for GNF2 fuel.

3.2.2 Interim Approach

3.2.2.1 Safety Limit Minimum Critical Power Ratio

The neutronic qualification provided in Supplement 3 for GNF2 fuel includes eigenvalue data, TIP data, and MCNP comparisons. On the basis of its review of these qualification data, the NRC staff has confirmed that the nuclear uncertainties and biases for GNF2 are consistent in magnitude and trend with those for GE14. Therefore, the NRC staff finds that the interim methods approach for assigning uncertainties in the SLMCPR determination as described in the IMLTR is equally applicable to GNF2.

Currently, the SLMCPR for IMLTR plants is determined according to a treatment of the [ ] and R-factor uncertainty based on a [ ]. The values used in these uncertainties are based on historical qualification data and were originally justified based on qualification against an expanded database that includes EPU plants with GE14 fuel. The NRC staff finds that the basis for this approach is acceptably extended to include GNF2 fuel.

Therefore, the NRC staff finds that compliance with IMLTR SE limitations “SLMCPR 1” and “SLMCPR 2” (Limitations 4 and 5, respectively from the IMLTR SE – Reference 2) provides adequate assurance that the nuclear uncertainties are acceptably treated in the safety limit determinations for EPU and MELLLA+ licensing evaluations. Appendix A of Supplement 3 states that for GNF2 fuel these limitations are unchanged for the GNF2 specific application and shall be met.

Appendix A of Supplement 3 also states that GEH has committed to provide additional qualification data to address nuclear methods uncertainties related to the [ ] and R-factor. These data have not been provided as of the time of the subject review. The NRC staff intends to review the applicability of these data to GNF2 applications when they are submitted for NRC review and approval.

On the basis that the [ ] quoted in the IMLTR remain applicable to GNF2 (which is based on the qualification provided in Supplement 3), and that no changes are proposed to the NRC staff's SLMCPR 1 and SLMCPR 2 limitations for the GNF2 specific application, the NRC staff finds that the treatment of power distribution uncertainties for GNF2 applications is acceptable.

However, the NRC staff notes that in the evaluation of the minimum CPR and transient change in CPR, the CPR is calculated according to the GEXL17 correlation. The GEXL17 correlation has biases and uncertainties distinct from the corresponding correlation for GE14 fuel (GEXL14). The NRC staff understands that the uncertainty in the critical power correlation is captured in the SLMCPR analysis according to the approved method. The NRC staff review of the GEXL17 correlation is provided in Section 3.6.1 of this SE.

3.2.2.2 R-factor

In its review of the IMLTR, the NRC staff imposed Limitation 6 on the R-factor calculation (Reference 2). Historically, fuel product specific R-factors were calculated based on [ ]. These [ ] were consistent with operating conditions for plants at OLTP. At EPU or MELLLA+ conditions, the bundle power and void fraction increase. The NRC staff evaluated the impact of correcting the R-factor [ ] for consistency with the limiting bundles and found the impact on the minimum critical power ratio (MCPR) to be significant.

IMLTR Limitation 6 requires that the plant specific R-factor be calculated consistent with the axial void conditions expected for the hot channel operating state. The NRC staff notes that the LHGR rod power limit for GNF2 exceeds the LHGR limit for GE14 at low exposure. The NRC staff postulates that the bundle powers or lattice rod peaking for GNF2 bundles operated near thermal limits may exceed those experienced for GE14 bundles. Therefore, either (1) rod-to-rod power peaking, or (2) gross bundle power for GNF2 bundles operating in an EPU core may exceed those experienced for limiting GE14 bundles. To address this concern, the NRC staff requested in RAI-16 that GEH demonstrate how Limitation 6 is met for GNF2, noting that the allowable LHGR is higher than for GE14.

The response to RAI-16 provides the results of analyses for four reactor cores (Reference 13). These analyses were performed to evaluate the void conditions present in GNF2 bundles that are potentially limiting in terms of low CPR. The approach described in the response is to determine an appropriate void fraction for the calculation of the R-factor. In general, the response describes the process by which a generic R-factor is calculated for GNF2 based on the expected [ ] for the limiting conditions. Cycle-specific confirmations are performed to ensure that the [ ] assumptions are representative for the safety analysis. The NRC staff finds this approach acceptable and consistent with IMLTR Limitation 6. The NRC staff notes that for cases where the generic GNF2-generated R-factor is not consistent with the expected void conditions in the limiting bundle, the approved R-factor methodology may be employed with an appropriate [ ] for the cycle-specific case.

As to the generic GNF2 R-factor, four cores were considered with a range of power densities up to [ ], which is consistent with EPU power densities. The distribution of CPR and channel void fractions was considered in the analyzed cases. The results are provided in a series of figures (Reference 13). These figures illustrate that the low CPR bundles have void fractions of approximately [ ] and that this condition is consistent between the various core designs. These analyses are consistent with similar analyses performed to demonstrate the applicability of the R-factor used in safety analyses for GE14 fuel and have been accepted by the NRC staff (References 17 and 18).

Therefore, the NRC staff finds that the [ ] channel void fraction is appropriate for generating the R-factor. Given its consistency over multiple core designs there is an expectation that this profile will be applicable to various EPU and MELLLA+ cycle- and plant-specific applications. However, the NRC staff notes that IMLTR Limitation 6 will require a cycle-specific verification of the consistency between the R-factor void profile and the limiting channel conditions for each cycle analysis.

3.2.2.3 Operating Limit Minimum Critical Power Ratio (OLMCPR)

The fuel parameters affecting the transient analysis include: local pin power peaking, void reactivity coefficient, and the three-dimensional power distribution. In terms of the local pin power peaking, GEH has performed evaluations using TGBLA06 and MCNP to compare the local pin power uncertainties calculated for GNF2 fuel lattices to equivalent uncertainties calculated for GE14. The results of these comparisons were reviewed by the NRC staff as documented in Section 3.2.1.4 of this SE. The results of these comparisons demonstrate that the GNF2 fuel design is sufficiently similar to GE14 that there is no observed degradation in the predictive capabilities of the lattice physics code to calculate the infinite pin power distribution. As this distribution forms the basis for the calculated local pin power distribution when combined with the PANAC11 pin power reconstruction methodology, the NRC staff is reasonably assured that the accuracy in the prediction of the local pin powers for GNF2 fuel is essentially as accurate as equivalent predictions for GE14 fuel.

The three-dimensional power distribution uncertainty is a combination of the [ ], the four-bundle power uncertainty, and the uncertainty associated with the axial power shape adaption. GEH has provided qualification of the core simulator against TIP data collected at early cycle exposure for a plant loaded with a full reload of GNF2 fuel. The limited qualification is briefly summarized by Table 3.2.1.3.1. The data indicate that the TIP statistics are not sensitive to the GNF2 fuel design. The NRC staff has reviewed these reload data as well as data from various LUAs, including LUAs that were loaded in EPU cores. These data were provided for NRC staff audit as part of the GESTAR II process. The NRC staff found that the TIP statistics for strings near GNF2 bundles did not indicate errors in the four-bundle powers or axial TIP traces that exceeded those for previous GNF fuel designs such as GE14.

The NRC staff documented the findings of its audit in Reference 6. On these bases, the NRC staff finds that the capability of the nuclear design codes (TGBLA06/PANAC11) to predict the power distribution for GNF2 fuel is essentially the same as its capability to predict the power distribution for GE14 fuel.

The NRC staff performed a review of the capability of the methods to accurately predict the void reactivity feedback for transient evaluations. The NRC staff review addressed two potential factors affecting the accurate prediction of the void reactivity: void history assumptions in determining the void reactivity bias and uncertainty, as well as any impact of errors in the prediction of the instantaneous void fraction arising from potentially increased uncertainties in the void-quality correlation.

In terms of the void reactivity coefficient, the NRC staff requested that GEH evaluate the sensitivity of the predicted void reactivity coefficient to the void depletion history in RAI-8. The NRC staff reviewed the impact of the 40 percent void depletion history assumption on the void reactivity coefficient biases and uncertainties in Section 3.3 of this SE.

The NRC staff conducted a review of the qualification of the void-quality correlation for GNF2 fuel. The NRC staff previously imposed a penalty requiring that the calculated OLMCPR be increased with a thermal margin enhancement of 0.01 as stated in Limitation 19 in the NRC staff SE for the IMLTR (Reference 2). Appendix A to Supplement 3 states that licensing analyses performed for EPU and MELLLA+ applications with GNF2 fuel will adhere to this limitation. However, the NRC staff reviewed the supporting qualification data provided in Supplement 3 to justify the continued applicability of the Findlay-Dix void-quality correlation to the GNF2 fuel design. The NRC staff review of the void-quality correlation is provided in Section 3.6.2 of this SE.

On the basis of its review the NRC staff has determined that those uncertainties affecting the transient analysis for GNF2 fuel remain essentially the same as for GE14. Therefore, the IMLTR alternative process for performing transient analyses is applicable to GNF2 fuel.

3.2.2.4 Loss-of-Coolant-Accident (LOCA) Related Nodal Power Limits

The maximum average planar linear heat generation rate (MAPLHGR) limit is established to ensure that peak clad temperature (PCT) does not exceed 2200ºF for the design basis LOCA. The neutronic methods uncertainties affecting the calculation of the MAPLHGR limit include the local power distribution uncertainties. The void reactivity coefficient has only a minor impact on LOCA consequences and the SAFER/GESTR calculations include a conservative power history assumption.

In terms of the affect of power distribution uncertainties on the LOCA results, GEH has previously evaluated the conservatism in the analysis method and concluded that sufficient conservatism was included in the characterization of the limiting rod and bundle powers to bound any potentially increased uncertainty in the local power distribution arising from EPU or MELLLA+ operation. The NRC staff reviewed these conservatisms and agreed with the GEH conclusion (Reference 2).

GEH cites the following conservative assumptions in the SAFER/GESTR LOCA methodology in terms of local pin and bundle powers:

1. [

].

2. In the 10 CFR 50, Appendix K calculation, a 2 percent core thermal power uncertainty is applied to the hot rod in order to account for plant core thermal power uncertainty. Note that some plants implemented improved feedwater measurement instrumentation and apply a lower power uncertainty. In the measurement uncertainty uprates, some plants operate at higher powers equivalent to the increased accuracy of the feedwater flow measurement instrumentation. However, for plants that implement EPUs up to 20 percent, additional power measurement uprate due increased accuracy of the feedwater flow measurement uncertainty is not allowed. The EPU is limited to 20 percent above the OLTP. Therefore, the ECCS [emergency core cooling system]-LOCA analysis will continue to assume 2 percent above the EPU power level.
3. In order to ensure that the SAFER analysis is bounding for all exposures, the hot rod of the hot bundle is placed at the exposure corresponding to the [ ]
4. The plants' core simulator calculates the margins to the fuel design limits (OLMCPR, SLMCPR, LHGR, and MAPLHGR). As a general practice, plants operated with margins to the MAPLHGR limit for most of the cycle operation.
5. Since the total bundle power is important to the severity of the ECCS-LOCA response, higher bundle power is therefore conservative. The SAFER/GESTR methodology [ ]. In an iterative calculation assuming different ECCS-LOCA basis MCPRs with bounding (low) R-factors, the bundle power peaking is maximized.
6. The full spectrum base ECCS-LOCA analysis is performed during initial implementation of SAFER methodology or transition to GE methodology and fuel. For new fuel introduction, or if new operating conditions are implemented, the limiting areas of the full spectrum base ECCS-LOCA analysis are reanalyzed to assure continued compliance with the 10 CFR 50.46 acceptance criteria for the new fuel or operating conditions. Depending on the specific licensing topical report [LTR], full break spectrum analysis may be performed for implementation of new operating strategies. During standard reloads, the assumptions in the ECCS-LOCA analysis-of-record are confirmed to remain applicable in terms of assumed OLMCPR and bundle LGHR and MAPLHGR limits. Therefore, the hot bundle operating power is maximized such that the ECCS-LOCA OLMCPR bounds the OLMCPR calculated from the limiting cycle- and core-specific AOO [anticipated operational occurrence] analyses.
7. To ensure that the ECCS-LOCA results are bounding, the pin power-peaking for the hot rod is also set to a [ ]

The NRC staff has reviewed various TIP data and the computational MCNP/TGBLA06 benchmarking provided in Supplement 3 for GNF2. As discussed in Section 3.2.1, the NRC staff has found that the neutronic methods are capable of predicting the nuclear parameters for GNF2 with essentially the same degree of accuracy as for GE14. On this basis, the NRC staff conclusions regarding the conservatism in the MAPLHGR analysis relative to the local power distribution remain equally applicable to GNF2.

However, the NRC staff requested additional information in RAI-12 regarding the conservatism afforded by the initialization in SAFER. In particular, the NRC staff notes that at early exposure, the GNF2 LHGR limit is much higher than for GE14. Additionally, similarity in the bundle geometry between GNF2 and GE14 and the results of the critical power tests appear to indicate that these two designs have similar critical power performance. Therefore, the NRC staff requested additional information regarding the degree of conservatism in the maximization of the bundle power according to the ECCS-LOCA basis MCPR with low R-factors for GNF2 fuel.

The response to RAI-12 provides additional descriptive details of the ECCS-LOCA analysis methodology initialization process (Reference 13). The response states that the higher LHGR limit for GNF2 does not change the ECCS-LOCA analysis methodology. In particular, the response describes the process by which the limiting bundle is modeled in SAFER to represent a conservative, simplified core condition. [

] On the basis of the response, the NRC staff agrees that the SAFER initialization process is acceptable to account for: (1) the different LHGR limits, (2) the thermal hydraulic conditions for expanded operating domains, and (3) the operational flexibility afforded by the thermal limits in a conservative manner, and therefore is acceptable for ECCS-LOCA analyses performed for plants with GNF2 fuel.

The NRC staff notes that analyses must be performed for multiple axial power shapes (top- and mid-peaked shapes) for both large and small break LOCA. Appendix A of Supplement 3 confirms that LOCA analyses performed for EPU and MELLLA+ licensing evaluations with GNF2 fuel will adhere to the NRC staff's limitations and conditions regarding ECCS-LOCA analyses (Limitations 7 and 8 from the NRC staff's SE for the IMLTR (Reference 2)).

3.2.2.5 Fuel Rod Thermal-Mechanical Performance

The NRC staff reviewed aspects of the Fuel Rod T-M Performance relevant to the application of the GEH/GNF analysis methods to GNF2 at EPU and MELLLA+ conditions. The NRC staff notes that GNF2 T-M operating limits (TMOLs) were reviewed and approved by the NRC staff using the GSTRM analysis method (Reference 9). The calculation to determine the TMOL is not dependent on the reactor power level. Therefore, the NRC staff did not revisit the applicability of the TMOLs to EPU or MELLLA+ conditions.

However, the NRC staff notes that aspects of the T-M analysis require particular inputs to address power distribution uncertainties and assumptions regarding the rod operating history. To this end, the NRC staff reviewed these input parameters to ensure continued applicability to GNF2 fuel and to ensure that the potential migration to the PRIME T-M methodology does not invalidate the basis for the NRC staff acceptance of the GEH/GNF T-M analysis approach for application to EPU and MELLLA+.

Lastly, the NRC staff reviewed the aspects of the methodology related to transient LHGR calculations. These calculations are performed on a cycle-specific basis to ensure that the relevant T-M acceptance criteria are met during AOOs.

3.2.2.5.1 Power Distribution Uncertainties

The power distribution uncertainty assumed in T-M analysis, also referred to as the monitoring uncertainty, accounts for nuclear methods and core monitor uncertainties in the prediction of the LHGR. During its review of the IMLTR, the NRC staff identified concerns regarding the adequacy of existing benchmark data to characterize the efficacy of the nuclear design methods to calculate the local rod powers. To address this concern, an interim approach was adopted to increase the pin power peaking uncertainty (see Section 3.2.1.5.1 of this SE) according to a [ ] approach.

The NRC staff has reviewed the relevant code-to-code qualification of TGBLA06 against MCNP calculations for GNF2 lattices and confirmed, given code modifications, that the local pin power distribution calculations performed using the TGBLA06/PANAC11 code system at EPU and MELLLA+ conditions are essentially the same as those reported in the IMLTR.

The IMLTR provides a summary of the calculated pin power distribution uncertainty based on the component uncertainties. Taking the [ ] power peaking uncertainty, the power distribution uncertainty for T-M analyses was determined to be [ ] percent (Reference 2). When corrected for the update uncertainty of [ ] percent reported in NEDC-32694P-A (Reference 14), the power distribution uncertainty is [ ] percent. This value is bounded by the [ ] percent that is used in GSTRM calculations. Therefore, the NRC staff finds that the power distribution uncertainties used in the GSTRM calculations are acceptable.

However, Supplement 3 provides that, since the NRC staff has approved PRIME, future T-M calculations will be performed using the PRIME T-M methodology. This is consistent with Limitation 12 from the NRC staff's SE for the IMLTR (Reference 2). Therefore, the NRC staff reviewed the PRIME Application Methodology LTR (Reference 10) to ensure that the power distribution uncertainties were adequate for application to GNF2 analyses.

The PRIME Application Methodology LTR (Reference 10) states that the IMLTR power distribution uncertainty is treated to account for monitoring uncertainty and is conservatively increased from [ ] percent in the analysis to [ ] percent (consistent with GSTRM) to account for "future concerns." The NRC staff has approved this magnitude for the monitoring uncertainty for use in GSTRM calculations and on the same basis finds that it is acceptable for PRIME calculations.

3.2.2.5.2 Operating History

At EPU and MELLLA+ conditions, the increase in core power requires bundles to operate at higher powers, or to operate closer to the LHGR limits for longer duration relative to cores operating at OLTP. In its review of the applicability of the T-M methods to EPU and MELLLA+ applications, the NRC staff specifically considered the possibility of operating fuel at the [ ] To this end, sensitivity calculations were performed to quantify the "operating history" conservatism in the analysis. Here the "operating history" conservatism refers to an analytical assumption in the calculation of the LHGR limit that requires that the peak nodal power be equal to the limit at each exposure point in the T-M analysis. The sensitivity study confirms that when fuel operates at the LHGR limit for reasonable durations early in life the "operating history" conservatism bounds the predicted internal rod pressure with a small margin [ ] (Reference 2).

The GNF2 TMOL is higher than for GE14 fuel. Also, the NRC staff identified a deficiency in the GSTRM code in terms of its ability to predict the fission gas release at high exposure, leading the NRC staff to impose a penalty in Appendix F of its IMLTR SE that requires a 350 psi reduction in the critical pressure (Reference 2). Therefore, the NRC staff considered the extension of the GNF2 T-M analyses to EPU or MELLLA+ conditions where the fuel may be operated at higher powers for longer exposure durations relative to OLTP conditions. Concerns regarding the adequacy of the prediction of the rod internal pressure for GNF2 fuel are addressed by the exposure limit of [ ] for the GSTRM analysis of the TMOL (References 8 and 9). However, the NRC staff notes that its acceptance of the TMOL requires that the assumed operating history must bound cases where the rods are assumed to operate at the peak LHGR for EPU or MELLLA+. Noting that, in accordance with IMLTR Limitation 12 and Supplement 3, GEH intends to use PRIME T-M methods for future applications, the NRC staff reviewed the operating history parameters assumed in the analysis according to the PRIME Application Methodology LTR (Reference 10).

Section 3.3.2 of the PRIME Application Methodology LTR states that the PRIME analyses are conservatively performed assuming that the peak power node of the fuel rod operates on the limiting power-exposure envelope throughout the fuel rod lifetime. This sweeping of the axial profile is consistent with the "operating history" conservatism in GSTRM. Further, the NRC staff review of PRIME (Reference 10) addressed the adequacy of its predictions of rod internal pressure.

Therefore, the NRC staff finds that the planned migration to the PRIME T-M method does not invalidate the basis for the acceptance of the T-M method for extension to application to EPU or MELLLA+ conditions.

3.2.2.5.3 Transient Linear Heat Generation Rate

During its review of the IMLTR, the NRC staff identified biases in the predicted transient LHGR resulting from 40 percent void history depletion assumption in the calculation of the void reactivity coefficient biases and uncertainties. The NRC staff review of the 40 percent void history depletion assumption and its impact on analyses performed for GNF2 fuel is documented in Section 3.3 of this SE.

When performing AOO calculations using the TRACG or ODYN codes, GEH must demonstrate an equivalent 10 percent margin to the fuel centerline melt and one percent plastic strain T-M acceptance criteria for AOOs. The requirement for this additional margin is provided by Limitation 11 in the IMLTR SE (Reference 2). This additional margin is based on sensitivity analyses documented in the IMLTR that show [ ] in the thermal and mechanical overpower predicted by TRACG when the void history affect on the void reactivity coefficient bias is corrected (Reference 1).

In its review of TRACG04 (NEDE-32906P, Supplement 3 - Reference 19) the NRC staff reviewed an update of the void reactivity coefficient biases and uncertainties model. The NRC staff found that the revised model was acceptable in terms of accounting for the impact of the void exposure history on the void reactivity coefficient (Reference 20). However, the application of TRACG04 to future GNF fuel products, such as GNF2, requires verification of the void reactivity coefficient correction model basis and verification of the applicability of the interfacial shear model prior to being applied (Reference 20). IMLTR Supplement 3 does not address the use of TRACG04; therefore, the NRC staff did not consider the applicability of TRACG04 to perform the LHGR transient analysis.

However, the NRC staff notes that if the limitations and conditions specified in the NRC staff SE for NEDE-32906P, Supplement 3 are met, TRACG04 may be used to perform the transient analysis for GNF2 loaded EPU or MELLLA+ cores. Consistent with IMLTR Limitation 11, when TRACG04 is used with the modified void reactivity coefficient correction model, it is not necessary to demonstrate the additional 10 percent margin to the fuel centerline melt or one percent plastic strain criteria.

Appendix A of Supplement 3 dispositions the implementation of the IMLTR SE limitations for GNF2 fuel applications. Appendix A states that IMLTR Limitation 11 remains applicable for GNF2 fuel. On the basis of its review of the 40 percent void history depletion assumption for GNF2 fuel, and that Supplement 3 confirms that an additional 10 percent margin will be demonstrated for licensing evaluations for AOOs; the NRC staff finds that the extension of the GEH/GNF methods to transient LHGR calculations for GNF2 applications at EPU and MELLLA+ conditions is acceptable.

3.2.2.6 Fuel Rod Exposure

The fuel rod exposure limit was established for GNF2 according to GESTAR II, Amendment 32 (Reference 8). This was an interim exposure limit to address methodology concerns regarding the applicability of the GSTRM T-M methods to GNF2. The exposure limit documented in Amendment 32 to GESTAR II was reviewed and approved by the NRC staff (Reference 9). This peak pellet exposure limit [ ] than the GE14 peak pellet exposure limit of 70 GWD/MTU. In addition, Limitation 12 from the NRC staff SE approving the IMLTR requires that future licensing evaluations be performed using updated T-M methods (Reference 2). GNF submitted the PRIME T-M methodology for NRC staff review to replace the GSTRM T-M methodology. The NRC staff reviewed and approved the PRIME T-M methodology in its SE dated January 22, 2010 (Reference 10). IMLTR, Supplement 4 (Reference 11) provides the implementation plan to update GEH's methods for compatibility with PRIME. Since PRIME was still under NRC staff review when Supplement 3 was submitted, Supplement 3 needed to address the interim GESTAR II Amendment 32 approach, but also provided for the anticipated approval of PRIME and discussed revising the peak pellet exposure limit if PRIME were to be approved. Following the NRC staff approval of PRIME, GNF submitted GESTAR II Amendment 33 to incorporate the use of PRIME into the GESTAR II process and address these limitations related to GNF2 and the use of GSTRM. In its SE approving GESTAR II Amendment 33, the NRC staff approved the removal of the Amendment 32 exposure limit for GNF2 fuel.

The NRC staff imposed a condition on the use of GSTRM to calculate T-M operating limits in Appendix F of its SE for the IMLTR. This condition requires that the critical pressure limit be adjusted by 350 psi to address potential non-conservatism in the method in terms of predicting the rod internal pressure. Supplement 3 states that this penalty does not apply to GNF2. The NRC staff agrees with this assessment on the basis that the rod internal pressure limits are not challenged until high bundle exposures have been reached, much later than the exposure limit imposed in GESTAR II, Amendment 32. Therefore, the NRC staff finds that the GSTRM T-M operating limits remain acceptable up to the exposure limit of [ ] peak pellet exposure. Since the NRC staff did not evaluate the effectiveness of GSTRM for predicting the rod internal pressure for GNF2 beyond [ ] peak pellet exposure, the use of GSTRM to calculate T-M operating limits for GNF2 fuel beyond the peak pellet exposure limit of [ ] would require that the 350 psi critical pressure adjustment described in Appendix F of the SE for the IMLTR be applied. However, consistent with IMLTR Limitation 12 and Supplement 4 to the IMLTR (Reference 11), it is the understanding of the NRC staff that since PRIME has been approved, future licensing evaluations for GNF2 in EPU and MELLLA+ cores will be performed using the updated PRIME T-M methods. GNF documented its agreement with and commitment to this understanding in a letter to the NRC dated May 27, 2010 (Reference 12). The 350 psi critical pressure adjustment does not apply if the PRIME T-M methods are used.

The NRC staff finds that Supplement 3 is consistent with GESTAR II, Amendment 32 and provides an acceptable peak pellet exposure limit when GSTRM T-M operating limits are utilized. The nature of this exposure limit is such that additional consideration of potential non-conservatism in the predicted rod internal pressure is not required to assure adequate safety. Now that PRIME has been approved, Supplement 3 states that the new method will be adopted and the exposure limit will be revised through the GESTAR II licensing process. This was accomplished through the review and approval of GESTAR II Amendment 33. On these bases, that NRC staff finds that the exposure limit for GNF2, as revised by the review and approval of GESTAR II Amendment 33, is acceptable.

3.2.2.7 Shutdown Margin

Supplement 3 provides specific qualification of cold critical eigenvalue calculations against data collected for an EPU core (Plant A) with GNF2 LUAs. These data provide direct confirmation that the uncertainties in the predicted local cold critical eigenvalue are fully consistent with the GE14 experience base. On this basis, the NRC staff concludes that shutdown margin methods are equally applicable to GNF2.

3.2.2.8 Standby Liquid Control System

The standby liquid control system (SLCS) efficacy is evaluated by calculating the core multiplication factor under cold, borated conditions, with all rods out. These calculations are performed by determining the cold cross section variation with boron concentration using TGBLA06 and calculating the core multiplication factor using the PANAC11 cold model with response surfaces from TGBLA06.

The impact of operation at EPU and MELLLA+ on SLCS margins is related to the overall ability of the methods to compute the core reactivity. Such cores may have higher reload batch fractions and the burned fuel may have differing isotopic compositions than non-EPU cores. Since the soluble boron is distributed throughout the core, the SLCS SDM is determined by core-wide reactivity effects rather than local effects (exposure and isotopic content). Therefore, the assessment of the ability of the nuclear methods to predict the SLCS margin is based on their ability to compute the core reactivity along with the ability to predict soluble boron worth. Based on the results provided for the cold critical demonstration (Section 3.2.1.1 of this SE), the biases and uncertainties for the cold critical calculations for GNF2-loaded core designs are similar to those for non-GNF2-loaded core designs.

The prediction of soluble boron worth is confirmed by the comparison of TGBLA06 with MCNP code results. The accuracy of lattice physics data generated at different boron conditions will factor into the calculation of the SLCS SDM. However, in this review the NRC staff did not perform code-to-code comparisons to assess TGBLA06-generated boron libraries. In terms of predicting the boron worth, the GNF2 lattice design is substantially similar to the GE14 design and these calculations are performed under cold (liquid water) conditions. Therefore, two dimensional coupling is minimized and the effect of differences in the lattices is minimal.

Based on this assessment and the additional level of conservatism resulting from the all rods out assumption, the SLCS calculational procedure remains applicable to EPU and MELLLA+ cores with GNF2 fuel.

3.3 40 Percent Void Fraction Depletion Assumption

When determining the void reactivity coefficient for ODYN analyses, GNF will generate nuclear data assuming a 40 percent void fraction history in TGBLA06 with branch cases calculated at 0 percent and 70 percent in-channel void fraction. These TGBLA06 calculations are used to assess the void reactivity coefficient as a function of exposure.

However, at EPU or MELLLA+ conditions, the core average void fraction increases relative to OLTP conditions. Exposure under these higher void conditions results in more aggressive buildup of plutonium, and as such, the assumption that a void history of 40 percent is representative begins to introduce substantial bias in the void reactivity coefficient at high exposure. Independent calculations performed by the NRC's contractors have indicated that this bias may reach [ ].

The NRC staff requested additional information regarding the sensitivity of the void reactivity coefficient biases and uncertainties to the 40 percent void fraction depletion assumption in RAI-8. GEH provided a response to RAI-8 in Reference 21. RAI-8 provides several alternative approaches to address the NRC staff concern regarding differences in the spectral hardness between GE14 and GNF2 fuel designs. In the response, GEH has elected to provide a comparison of the void reactivity coefficient data between GE14 and GNF2 to justify the continued applicability of the bias and uncertainty used in ODYN.

RAI-8 references a model for void history exposure correction to the void reactivity coefficient in TRACG04. The NRC staff reviewed this model as part of its review of Reference 19. The NRC staff SE provides the basis for the NRC staff acceptance of this model (Reference 20). In the NRC staff's previous review, the set of lattices used in developing the inputs for the void reactivity coefficient uncertainties and biases were not sufficient to be representative of the full range of lattices in the GNF2 bundle design. The response to RAI-8 expands the initial set of lattices to incorporate GNF2 specific lattice designs (Reference 21). The NRC staff reviewed the information provided in Table 8-2 of the response. This table describes the set of lattices included in the expanded database. These lattices are representative of GNF2 fuel and also represent a significant increase in the overall amount of TGBLA06/MCNP comparison data included in the correction model database. Therefore, the NRC staff finds this approach acceptable to address the GNF2 fuel design.

Statistical tests (t-tests) were performed to determine the viability of combining the initial database with the expanded GNF2 database (Reference 21). The results of these statistical tests confirm that the reactivity coefficient biases and uncertainties were essentially indistinguishable between the historical basis (GE14 lattices) and the expanded set (GNF2 lattices). This provides reasonable assurances that the differing geometric configurations and loadings between the fuel designs do not result in significant differences in the void reactivity characteristics between the two designs. An overall statistical test for the normality of the reactivity coefficient biases and uncertainties was performed. The results of this statistical test are provided in the response to RAI-8 and demonstrate that the mean is essentially zero (which is consistent with the conclusions reached during the NRC staff review of the IMLTR for GE14 lattices, see Reference 2). The standard deviation is slightly less than unity when normalized indicating that the data are slightly less variable than expected for a normal distribution, however, treatment of these uncertainties as if they were normal is conservative. Therefore, the NRC staff finds that the results of the assessment demonstrate consistency with the previously approved basis and ensure continued conservatism in the application of the correction model within TRACG04.

To demonstrate the continued applicability of ODYN, GEH provided the results of a transient analysis performed for an equilibrium core of GNF2 fuel. This is similar in many regards to analyses supplied to the NRC staff during its review of PRIME. In particular, the response to RAI-39 associated with the PRIME review documented transient analysis sensitivity to the fuel thermal conductivity model (Reference 22). The approach described in the response to RAI-8 of this review is analogous to the PRIME RAI-39 approach. In the subject analyses in the response to RAI-8, GEH provides the results of sensitivity studies performed using TRACG04 and the results of an ODYN analysis. The figures of merit considered in the response include: peak power, peak vessel pressure, transient critical power ratio, peak centerline temperature, hoop stress, and water level. In these calculations, the peak power and vessel water level are critical parameters that describe the gross transient event progression. The peak pressure, critical power ratio and peak centerline temperature are directly related to safety limits. The hoop stress serves as a surrogate parameter to the safety limit associated with the cladding plastic deformation. Therefore, the NRC staff finds that the parameters considered for comparison are relevant and address the full scope of transient analysis figures of merit.

A typically limiting transient was considered (a turbine trip without turbine bypass) for a BWR/4 plant. This basis is identical to the basis provided during the PRIME review in the response to PRIME RAI-39 (References 21 and 22). As the purpose of these analyses is to demonstrate conservatism in the ODYN modeling for GNF2, the NRC staff accepts this representative case as a sufficient basis to identify dominant trends, but also agrees that the specific sensitivity will depend on the core loading and exposure distribution in the core being analyzed.

The peak pressure, peak centerline temperature, hoop stress, and water level decrease results indicate that either using or not using the void reactivity coefficient correction in TRACG04 leads to essentially identical results. When relevant parameters could be compared with ODYN, the response indicates that the calculation results are essentially the same. On the basis that the transient results are not sensitive to the void reactivity coefficient correction, the NRC staff finds that the use of ODYN to perform those transient analyses associated with the aforementioned acceptance criteria and critical parameters remains acceptable.

Differences are observed between ODYN and TRACG04 in terms of the peak total power and the limiting transient change in CPR per initial CPR (ΔCPR/ICPR). The results indicate an approximate [ ] sensitivity in the ΔCPR/ICPR when the void reactivity coefficient correction model is implemented in TRACG04. These results are fully consistent with the sensitivity demonstrated for GE14 in response to RAI-30 associated with the TRACG04 review (References 19 and 20). These results confirm that the sensitivity of the transient analysis results for GE14 and GNF2 are essentially the same.

To further justify the continued applicability of ODYN, the RAI-8 response provides comparison of ODYN transient calculations to the TRACG04 calculations. The results of these analyses indicate that ODYN consistently predicts a higher peak power and higher ΔCPR/ICPR relative to TRACG04. The comparison indicates that the difference between the ODYN and TRACG04 predictions are much greater than the [ ] sensitivity in ΔCPR/ICPR associated with the correction to the void reactivity coefficient to account for void exposure history.

On the basis that the results of detailed calculations using the approved TRACG04 void reactivity coefficient void history correction model indicate consistent results for GE14 and GNF2, the NRC staff concludes that the implications in the safety analysis associated with the 40 percent depletion assumption are identical between these two fuel designs. On the basis of the demonstration of the conservatism in the ODYN analysis method relative to the TRACG04 method, the NRC staff finds that the conclusions reached regarding the ODYN transient analysis methods for GE14 are likewise applicable to GNF2 without modification.

Therefore, the NRC staff concludes that the specific limitations and conditions specified in its SE for the IMLTR (Reference 2) to address concerns regarding the 40 percent depletion assumption in the transient analyses remain fully applicable to GNF2 without modification. Appendix A of Supplement 3 states that these conditions will be met for safety analyses performed for GNF2 loaded cores (Reference 4). Therefore, the NRC staff finds that the continued use of ODYN within the framework of the interim methods process is acceptable for application to GNF2-loaded cores.

3.4 Bypass and Water Rod Voiding

At EPU and MELLLA+ operating conditions, the reactor power-to-flow ratio is increased relative to OLTP operation. Under these conditions, it is expected that voids may form in the bypass regions (intra- and inter-assembly). The formation of bypass voids affects several key uncertainties in various safety analyses. At the extremes of high power-to-flow ratio, stability becomes a limiting phenomenon. Therefore, the impacts of bypass void formation on the various stability solutions must be evaluated. In its IMLTR, GEH provided various assessments of the impact of bypass void formation on local power range monitor (LPRM) indications during steady state operation and under conditions of small margin to instability.

The NRC staff has postulated that the higher LHGR limits for GNF2 may allow for higher powered bundles in EPU or MELLLA+ core designs loaded with GNF2 fuel. Therefore, the local bypass void fraction near the higher powered bundles may exceed those void fractions evaluated for GE14 fuel as part of the IMLTR.

3.4.1 Power Distribution

The NRC staff notes that the nodal diffusion code PANAC11 and the equivalent engine in TRACG04 [

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The NRC staff has evaluated this assumption for high in-channel void fractions and relatively large bypass void fractions for GE14 during its review of the IMLTR. In its assessment, the NRC staff found that the approach does not introduce any appreciable error in the nodal reactivity or R-factor calculations.

In RAI-4, the NRC staff requested that GEH evaluate the effect of bypass void formation at high in-channel void fraction on the radial power distribution for GNF2. The NRC staff notes that the GNF2 fuel design includes PLRs at the lattice edge; therefore, the NRC staff requested the evaluation to compare the radial power redistribution for GNF2 fuel to GE14 fuel to assess the continued applicability of the previous NRC staff findings.

The response to RAI-4 provides the results of power distribution calculations with a bypass void fraction of 5 percent at an in-channel void fraction of 90 percent (Reference 13). The NRC staff agrees that 90 percent in-channel void fraction is an appropriate analysis condition as this takes into account: (1) the increased sensitivity of the rod powers to the bypass at high void conditions and (2) a realistic combination of bypass and in-channel void conditions.

The response explicitly compares the radial power redistribution and finds that for the potentially limiting rods (non-gadolinia-bearing rods) the effect of bypass voiding for GE14 and GNF2 is largely similar. The maximum change in rod power for non-gadolinia-bearing fuel for GNF2 is slightly lower than for GE14. The radial power shape redistributes in largely the same way with power shifting slightly away from the lattice edge and water rods. The largest increase in rod power for the GNF2 lattice was [ ] percent (compared to [ ] percent for the GE14 lattice)[1]. The maximum increase occurred for a low power rod and this rod is not likely to be a peak or limiting rod during the life of the bundle.

With regard to the basis for the calculations provided in the response to RAI-4, the NRC staff concludes that the effect of bypass void formation for GNF2 is largely similar to that for GE14. The difference in the lattice geometry was explicitly considered in the analysis. For the two fuel designs, the maximum change in rod powers between the two designs was essentially the same with the largest increases occurring in rods that were not likely to be the limiting rods. Therefore, the NRC staff finds that the previous review findings regarding GE14 fuel power distribution under bypass void conditions remain equally applicable to GNF2 fuel.

In terms of the axial power shape, the formation of bypass voids will have the effect of lowering the nodal reactivity of affected axial extremes of the bundle, and thereby result in a downward shift in the axial power shape. In the NRC staff audit of the detailed TIP trace data provided for GNF2 LUAs, downward biases in the axial power shape were not observed (Reference 6). However, these LUAs were not operated in limiting bundle locations. Further, the TIP data did not include EPU plants operating at 120 percent of the OLTP or MELLLA+ plants. Under higher power-to-flow conditions typical of MELLLA+ with spectral shift control or higher power density EPU plants, inter- and intra-assembly bypass void fractions are expected to be higher. Therefore, the NRC staff cannot conclude that this effect would not be observed if the database included higher power density plants.

In terms of the safety analysis, however, neglecting the bypass void formation would conservatively result in higher axial power peaks. This is generally conservative for the transient safety analysis and forms the basis for the [ ] assumed in the cycle-specific safety analysis. Therefore, coarse treatment of the bypass void in PANAC11 and ODYN is expected to confer some degree of conservatism, in terms of the initial conditions, for the limiting bundle calculation in the transient safety analysis. Transient calculations are addressed in Section 3.4.3 of this SE.

3.4.2 Instrumentation and Power Distribution Uncertainties

Limitations imposed through the NRC staff's SE for the IMLTR restrict steady-state bypass void fraction at the LPRM Level-D location to five percent. This limitation assures that the LPRM indications are not significantly impaired by LPRM sensitivity to the local fluid conditions. Limitation 17 from the SE for the IMLTR documents the steady-state bypass void limit of five percent. Appendix A of Supplement 3 provides that Limitation 17 will be met. Therefore, the degree of bypass void formation will be evaluated each cycle and the results documented in the supplemental reload licensing report (SRLR). Compliance with IMLTR Limitation 17 provides the NRC staff with reasonable assurance that the introduction of GNF2 fuel to EPU or MELLLA+

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[1] The values quoted neglect the gadolinia-bearing fuel rods.

cores does not degrade the LPRM Level-D indications, and is therefore acceptable in terms of steady-state monitoring capabilities.

The NRC staff requested additional information regarding the impact on gamma TIP and neutron-sensitive TIP (thermal TIP) instruments to the presence of bypass voids in RAI-15. TIP data are used to periodically update the core monitor axial power shape. The updated axial power shape is then used in the core monitor to determine the margin to thermal limits. The axial power shape monitoring and adaption is credited in the safety limit analyses.

When adapting the axial power shape, [

]. The presence of bypass voids would affect the core monitoring capability to predict the local power. The NRC staff has already reviewed the effect of bypass void formation on the local radial pin power distribution and found that this generally flattens the radial distribution for those nodes experiencing substantial bypass void formation.

When considered in total, the formation of bypass voids: (1) reduces nodal power due to lower moderation, (2) reduces instrument response by shifting radial power distribution away from the instrument, and (3) reduces neutron sensitive instrument response by decreasing moderating effect near the fission chamber. Therefore, at substantially high intra-assembly void fraction, the axial power shape monitoring may indicate significant biases. When the axial power shape is adapted, it may be biased towards the bottom of the core relative to actual power distribution if these effects are significant and unaccounted. When determining the minimum CPR (MCPR) for the assessment of operational margin to the OLMCPR, the core monitor may non-conservatively calculate the bundle MCPR.

To address this concern, GEH utilized the results of the calculations performed in the response to RAI-4 to determine the potential impact of bypass void formation on TIP instrument response. The response to RAI-15 states that the channel box geometry and the location of the corner rod relative to the instrument tube is identical between GNF2 and GE14 (Reference 13). Further, the response provides comparison of the GE14 and GNF2 corner rod power sensitivity to bypass void formation. A limiting case of 90 percent in-channel void fraction and 5 percent bypass void fraction was considered. The results indicate that the corner rods at the wide-wide (WW) and narrow-narrow (NN) corners were essentially the same for both fuel products (Reference 13).

The NRC staff reviewed the results of these calculations and confirmed that both fuel products exhibit essentially identical sensitivities to the presence of bypass voids. As the other parameters affecting the predicted TIP reading (instrument tube and channel geometry) are identical between the two designs, the NRC staff accepts the power distribution calculation as an adequate surrogate analysis parameter to address potential biases in the TIP reading. The power distribution errors introduced by bypass voids are minimal [
] As the TIP readings are most sensitive to the corner rod power for both gamma and thermal TIP instruments, the NRC staff is reasonably assured that significant errors would not be introduced that are specific to the GNF2 fuel.

Further, the NRC staff reiterates that the bypass void fraction is limited to 5 percent at the LPRM Level D elevation by Limitation 17 of the IMLTR SE (Reference 2). This limitation ensures that power peaking factors are constrained such that significant bypass voids do not form. Therefore, the NRC staff finds that the basis for the analysis (5 percent bypass void fraction) is acceptable. The results provided at 90 percent in-channel void fraction present the maximum predicted change in the corner rod power, as the higher the in-channel void fraction, the more sensitive the rod power distribution is to the bypass voids. On these bases, the NRC staff has found that GEH has demonstrated that the performance of the analysis methods to analyze GNF2 considering the range of allowable bypass void formation is essentially the same as the performance for GE14 fuel. On this basis, the NRC staff finds that application of the methods to GNF2 fuel is acceptable when Limitation 17 imposed by the NRC staff on the IMLTR is met.

3.4.3 Transient Response

Given that the GNF2 LHGR limit is much higher at low exposure than the associated limit for GE14, the NRC staff expects that the introduction of GNF2 in EPU or MELLLA+ core designs may allow for increased radial power peaking and even higher bundle powers relative to a homogeneous GE14 EPU or MELLLA+ core. Therefore the NRC staff considered the degree of bypass void formation for GNF2 fuel operating at or near the LHGR limit.

As a bounding case, one might consider a limiting GNF2 fuel assembly, operating near the LHGR limit, with an isolated bypass channel around the bundle. In this case, the instantaneous void fraction around the high powered GNF2 bundle is expected to be over-estimated. The effect of a pressurization event may add additional local reactivity due to an increase in the reactivity addition from void collapse in the bypass. However, when compared to an analysis where the GNF2 bypass is not isolated, the initial nodal powers will be lower. So the limiting nodal location would be such that the transient would initiate from a lower power level, but the differential nodal reactivity added in response to the pressurization would be higher.

When a code such as ODYN is used to perform transient calculations, the bypass is treated as a single channel. For this single channel bypass, the void fraction will be representative of the entire core and, as such, be low. Under conditions of pressurization, which are typically limiting for transient calculations, the core wide response accounts for the collapse of the bypass voids to an essentially solid water condition. In the case where the bypass is treated as either an isolated channel or a core-wide bypass channel is used, the absolute nodal reactivity prediction for the limiting bundle will be essentially identical in response to the void collapse.

If the pressurization is sustained, then the final power predicted using either method would be essentially identical. However, the transient is terminated by a SCRAM for transient analyses. Therefore, initiating the transient response from a higher power would be conservative. On this basis, the NRC staff finds that it remains acceptable to model the bypass as a lumped channel even though there is the potential for local bypass void conditions to be higher around GNF2 bundles.

3.4.4 <u>Stability</u>

The NRC staff specifically reviewed the applicability of the stability methods to GNF2 fuel in Section 3.5 of this SE. This section provides a discussion of those phenomena relevant to the stability evaluation in the context of the GNF2 fuel design. Conditions that must be evaluated to determine the margin to instability generally are high power-to-flow conditions. Under these conditions, the bypass void fraction is expected to be much greater than at steady state conditions. However, significant margins are typically applied to stability calculations when determining exclusion regions, for instance. An analysis provided by GEH in response to RAI-3.2(a)(iii) during the NRC staff's review of the IMLTR provides the results of calculations that demonstrate small bypass void fractions along an exclusion region boundary.

However, the NRC staff notes that the GNF2 fuel design includes features that are expected to enhance the stability performance of the design relative to GE14 or earlier fuel designs. These features are described in greater detail in Section 3.5 of this SE. However, on the basis of these design differences, the NRC staff could not reach the same conclusion that the bypass void fraction is expected to be small along an exclusion region since the NRC staff would expect the power-to-flow ratio for GNF2 fuel to be higher at the same decay ratio relative to GE14 or earlier fuel designs.

While the effect is expected to be negligible, the NRC staff notes that it has approved the use of an alternative exclusion region shape function per Reference 23. The modified shape function provides a mildly less restrictive exclusion region, thus populating the allowable operating domain with a region of slightly higher power-to-flow ratio. The NRC staff requested that GEH confirm the limiting conditions for GNF2 in RAI-11.

The response to RAI-11 provides the results of analyses for GNF2 and GE14 fuel for comparison. The response shows the exclusion region calculated for both fuel products (Reference 13). Consistent with the NRC staff's expectations, the GNF2 exclusion region is smaller than the GE14 region and the intersection of the exclusion region along the natural circulation line (NCL) for GNF2 fuel occurs at a higher power than for the analogous GE14 case. The response to RAI-11 computes the bypass void fraction at the exclusion region boundary for GNF2 using the limiting power-to-flow conditions and conservative ISCOR assumptions for direct moderator heating. The calculations indicate higher bypass void fractions for the GNF2 bundle – though similar in-channel void fractions when compared to the GE14 bundle. However, these higher bypass void fractions remain within the range of void fractions computed for EPU and MELLLA+ plants as part of the original IMLTR submittal [ ] (References 2 and 13). Therefore, the NRC staff finds that the bypass conditions analyzed remain within the previously established basis in the IMLTR.

Bypass void formation has the potential to affect stability analyses by impacting the nodal reactivity feedback mechanisms due to fluctuation in the bypass void fraction and also has the potential to impact the instrument response. The LPRMs are neutron sensitive and therefore the sensitivity of the instrument is a function of the local moderating effectiveness of the bypass water.

In terms of the first phenomenon, void formation and collapse in the bypass is treated to the "first order" in TRACG and ODYSY. These two codes include a bypass channel, in the case of TRACG several bypass channels may be modeled, but this capability is not typically utilized. The axial variation in the bypass void fraction (core-average) is calculated according to the thermal-hydraulic models and the nuclear feedback is captured by tracking the nodal water content as discussed in Section 3.4.1 of this SE.

The NRC staff notes that this first order treatment does not consider: (1) the affect of local void distribution on pin power distribution, or (2) local radial variation in bypass void formation. The NRC staff has previously concluded that the impact of bypass void formation is generally beneficial from an R-factor perspective (as described in Section 5.2 of the NRC staff SE for the IMLTR (Reference 2)). Therefore, the approximate nature of the first order treatment does not result in significant or non-conservative errors in the calculation of the R-factor used in the CPR response in certain stability calculations.

In terms of the radial void distribution within the intra-assembly bypass, the NRC staff does not expect a significant analytical impact for two reasons: (1) the bypass is open to radial thermal-hydraulic communication and (2) EPU and MELLLA+ cores are generally designed with flattened radial power shapes relative to OLTP core designs. However, the NRC staff notes that the GNF2 fuel TMOL is substantially higher for low exposure than the corresponding GE14 limit. Therefore, the NRC staff expects that the introduction of GNF2 fuel to an EPU or MELLLA+ core design may allow for higher radial power peaking for the low exposure GNF2 fuel bundles than would be considered conventional for a more homogeneous core design. Therefore, the NRC staff postulates that the local bypass void conditions for GNF2 fuel under natural circulation conditions may be higher than for GE14 fuel. The NRC staff requested, in RAI-4, that GEH assess the bypass void fraction under natural circulation conditions. The NRC staff notes that the response to RAI-11 provides a relevant analysis of the bypass void fraction at the exclusion region boundary along the NCL.

In response to RAI-4, GEH states that Limitation 17 from the NRC staff's SE for the IMLTR requires that the cycle-specific loading be evaluated to ensure that bypass void fraction remains below five percent at the LPRM Level-D elevation (References 2 and 13). The response states that the cycle-specific analysis must consider all operating conditions within the upper boundary of the expanded operating domain. The response states that the peaking factors, among other factors affecting initial conditions, are inherently limited such that the five percent bypass void limit is met during normal operation. Therefore, while the LHGR limit for the GNF2 fuel design exceeds the LHGR limit for GE14, cycle-specific analyses are performed that ensure that the bundle power peaking is limited to ensure that the bypass void fraction remains within the five percent limit imposed by Limitation 17. On this basis, the NRC staff agrees that the cycle-specific reload licensing analyses ensure that the bypass void conditions are not exacerbated for GNF2 fuel relative to GE14 fuel. Therefore, the NRC staff finds that the setpoint setdown imposed by Limitation 18 from the NRC staff's SE for the IMLTR (Reference 2) remains appropriate and applicable to GNF2 since the degree of bypass void formation is constrained to the same degree by Limitation 17.

The NRC staff has previously reviewed the potential for bypass void formation to introduce a calibration error in the oscillation power range monitor (OPRM) or the average power range monitor (APRM). The potential for GE14 bypass void formation under natural circulation conditions was conservatively evaluated by GEH during the IMLTR review. The NRC staff found that a setpoint setdown of 5 percent for the OPRM and 2 percent for the APRM was sufficient (based on the nature of the stability solution) to address any calibration error associated with bypass void fractions of [ ]

The response to RAI-11 states that the setpoint setdown for the OPRM is conservatively applied in that the attenuation of the average signal is not credited (Reference 13). This conservatively increases the importance of the five percent attenuation of the OPRM.

On the basis that ODYSY is applied within the bypass and in-channel void fraction range for GNF2 previously considered for EPU and MELLLA+ conditions as part of the IMLTR and that the OPRM setpoint setdown is conservatively applied, the NRC staff concludes that the stability methods and associated acceptance criteria remain acceptable and applicable to address bypass void formation for the GNF2 fuel product.

3.5 Stability

Stability calculations are performed to assure that the SLMCPR is protected in the event of a thermal-hydraulic instability. A variety of stability long term solutions (LTSs) have been developed and implemented. These stability LTSs are based on: (1) prevention, (2) detection and suppression, or (3) a combination of these two aspects. For EPU plants, the candidate LTSs include the following BWR Owners' Group (BWROG) stability LTSs: Enhanced Option I-A, Option I-D, Option II, and Option III. A specific stability LTS was developed by GEH for MELLLA+ plants. This is the detect and suppress solution – confirmation density (DSS-CD) solution. DSS-CD is an evolutionary modification of the Option III solution.

As these stability LTSs implement various strategies in terms of prevention and/or detection and suppression of thermal-hydraulic instabilities, the cycle-specific licensing strategy and implementation relies on varied analyses. Therefore, the NRC staff conducted its review of the applicability of the stability methods to GNF2 on a solution-specific basis for EPU and MELLLA+ operating domains. The NRC staff review of DSS-CD is limited to the MELLLA+ domain, whereas the review of the other LTSs is limited to consideration of EPU operation.

The NRC staff review addresses the applicability of the methods to analyze GNF2 fuel at conditions that are representative of the likely application of the specific stability calculations that are performed for each LTS. In its review, the NRC staff has identified that the GNF2 fuel design incorporates several design changes relative to GE14 that affect the stability performance. In particular, the NRC staff notes that GNF2 includes a number of SPLRs. These SPLRs are expected to enhance the stability performance of GNF2 fuel as they contribute to increasing the single phase to two phase pressure drop ratio.

Another important design difference between GNF2 and GE14 is the fuel pellet thickness. The GNF2 fuel pellets are slightly thicker than GE14 fuel pellets. This will likely have the effect of increasing the fuel thermal time constant. Increasing the fuel thermal time constant likewise has a stabilizing effect as it "decouples" the fluid state and neutronic flux response to a greater degree than for GE14 fuel.

Given consideration of these two design differences, it is the expectation of the NRC staff that the onset of instability for GNF2 fuel will occur at more adverse reactor operating conditions than it would for GE14 fuel. Namely, the onset of core-wide or fuel channel instability for GNF2 fueled cores is expected to occur at higher power-to-flow ratio conditions than for GE14 fuel designs. Tables 3-24 and 3-25 of Reference 3 provide results of stability analyses for representative tight and loose orifice plants. The results provided in these tables confirm that, generally, GNF2 fuel is more stable than GE14 or earlier fuel designs (e.g., P8x8R).

The NRC staff requested, in RAI-11, that GEH evaluate the difference in thermal-hydraulic conditions predicted for GE14 and GNF2 fuel at an equivalent decay ratio. As exclusion regions are typically defined with an analytical decay ratio of 0.8, the NRC staff requested that these analyses be performed to determine the thermal-hydraulic condition of the fuel at this decay ratio.

The response to RAI-11 compares the void fraction and the power-to-flow ratios calculated for the GNF2 limiting points on the exclusion region boundary to the qualification database for ODYSY (Reference 13). This database includes the Nine Mile Point 2 (NMP2) instability event, the Perry instability event, and high decay ratio tests performed for Vermont Yankee. The response confirms that the GNF2 analysis conditions along the exclusion region boundary remain within those thermal-hydraulic conditions present in the qualification data. Therefore, the NRC staff finds that the response provides an adequate basis for the NRC staff to conclude that ODYSY is being applied within the range of its qualification for the GNF2 fuel product.

The NRC staff identified those phenomena generally important to reactor stability and considered the qualification of the analysis methods for GNF2 fuel. Supplement 3 states that the stability performance depends on the following parameters: (1) void reactivity coefficient, (2) local pin power peaking, (3) [ ], and (4) bundle pressure drop. The NRC staff reviewed each of these parameters and the uncertainties associated with GNF2 generally before reviewing the specific ramifications for each stability LTS.

Bypass void formation, as discussed in Section 3.4.4 of this SE, may affect the stability analyses and LTS performance. The NRC staff reviewed the ramifications associated with bypass void formation strictly on a LTS-specific basis.

3.5.1 General Review of Stability Performance Parameters

3.5.1.1 Void Reactivity Coefficient

The void reactivity coefficient is a highly important parameter affecting the stability performance. The NRC staff compared the uncertainties in the calculated void reactivity coefficient for GNF2 relative to GE14 fuel to determine if the extension of the nuclear methods to higher void fractions would introduce additional uncertainty based on the specific consideration of the GNF2 fuel design.

In RAI-8, the NRC staff requested that GEH evaluate the void reactivity coefficient biases and uncertainties associated with the 40 percent void history depletion assumption. As described in greater detail in Section 3.3 of this SE, the NRC staff finds that the sensitivity of the transient analysis to the void reactivity coefficient void exposure history effect is essentially identical for GE14 and GNF2 fuel lattices.

In its review of the IMLTR the NRC staff determined that errors in the void reactivity coefficient (core average) of approximately two percent were essentially negligible when assessing the core stability performance (Reference 2). The NRC staff compared the errors in the void reactivity coefficient for GNF2 and GE14 fuel attributed to the 40 percent void history depletion assumption and found that the errors are essentially consistent. The TRACG calculations performed for this magnitude of error indicate that the stability methods are unaffected. Therefore, the NRC staff finds that these methods are acceptable for application to GNF2 without additional consideration of the void reactivity coefficient uncertainties or biases introduced by the 40 percent depletion assumption.

In RAI-11, the NRC staff requested that GEH evaluate the bundle conditions near the onset of thermal-hydraulic instability for GNF2 fuel and evaluate the impact of potential biases and uncertainties in the void reactivity coefficient on the stability calculations. The response to RAI-11 compares the application range of ODYSY for GNF2 fuel to the qualification range of the code. The response demonstrates that ODYSY is applied within the range of its qualification. Comparison of the ODYSY code predictions to the high decay ratio test data collected at Vermont Yankee confirm its accuracy to analyze plant conditions at high power-to-flow ratios. On this basis, that NRC staff agrees that the uncertainties applied to the ODYSY acceptance criteria remain adequate and acceptable for GNF2 applications.

3.5.1.2 Power Distribution Uncertainties

The NRC staff reviewed the impact of the power distribution uncertainties generically for various elements of stability solutions and the associated calculations that support the licensing of those solutions. The NRC staff considered the uncertainties in the local pin power peaking and the [ ] These uncertainties affect the axial and radial power distribution and therefore have an impact on the calculation of either the decay ratio or the detect and suppress solution setpoint.

3.5.1.2.1 Local Pin Power Peaking

The local pin power peaking uncertainties for GNF2 have been compared to GE14 based on detailed MCNP comparisons. On the basis of these comparisons, the NRC staff determined that the uncertainties for GNF2 are consistent with those for GE14 (see Section 3.2.1.5.1 of this SE). These uncertainties are captured in the SLMCPR, and inherently in the OLMCPR.

3.5.1.2.2 [ ]

The NRC staff reviewed the neutronic qualification for GNF2, including TIP measurements for LUAs and reload quantities, as well as MCNP comparisons to GE14. On the basis of these comparisons, the NRC staff determined that the uncertainties for GNF2 are consistent with those for GE14 (see Section 3.2.1.5.3 of this SE). These uncertainties are captured in the SLMCPR, and inherently in the OLMCPR.

3.5.1.2.3 Decay Ratio

Decay ratio analyses are performed for plants incorporating a LTS with a prevention element. In its review of the IMLTR, the NRC staff determined that TRACG and ODYSY were qualified against a variety of plant data with high decay ratios. The qualification cases were reported in Section 6.1.1 of the NRC staff's SE for the IMLTR (Reference 2). In RAI-11, the NRC staff requested that GEH compare the thermal-hydraulic conditions where GNF2 is predicted to become marginally unstable to those conditions included in the ODYSY and TRACG qualification database.

The response to RAI-11 compares the void fraction and the power-to-flow ratios to the qualification database for ODYSY and confirms that the GNF2 analysis conditions remain within the range of the qualification data. In addition, the response to RAI-11 states that a conservative Haling axial power shape is used to perform the decay ratio analysis; therefore, axial power shape uncertainties do not affect the analysis (Reference 13). The Haling power shape is a limiting "flat" axial power shape compared to expected power shapes during normal depletion and this assumption in the decay ratio calculations affords additional conservatism in terms of the power distribution. On these bases, the NRC staff concludes that the power distribution uncertainties are adequately treated through qualification, acceptance criteria, and analytical conservatism.

3.5.1.2.4 Change in CPR per Initial CPR versus Oscillation Magnitude (DIVOM)

When the power distribution uncertainties are included as an adder to the SLMCPR, the uncertainties affect the allowable hot bundle oscillation magnitude, and hence protection system SCRAM setpoints on a cycle-specific basis for plants implementing a LTS with a detect and suppress element.

A ΔCPR/ICPR versus oscillation magnitude, or DIVOM, curve is calculated on a cycle-specific basis. When performing licensing evaluations, the CPR response to an oscillation of given magnitude is determined from the DIVOM and the CPR is compared to the SLMCPR. Setpoints in suppression features of the LTS are determined to ensure that the oscillation magnitude is sufficiently small as to meet the SLMCPR (Reference 24). The influence of the increased bundle power uncertainties on the detect and suppress solution is apparent when comparing the maximum allowable ΔCPR/ICPR[2] with and without the increase in the SLMCPR. Increasing the SLMCPR (and hence the OLMCPR) by an equivalent amount reduces the allowable ΔCPR/ICPR on a cycle-specific basis. The result is that the SCRAM setpoint must be reduced to ensure a smaller hot bundle oscillation magnitude during a potential instability.

The NRC staff notes that the OLMCPR penalty of 0.01 applied by IMLTR Limitation 19 is not used in establishing stability setpoints in order to be conservative (Reference 25). On this basis, the NRC staff finds that the detect and suppress solutions, or the detect and suppress features of the various solutions, inherently account for the increased power distribution uncertainties through the DIVOM curve and setpoint determination process by reducing allowable ΔCPR/ICPR.

3.5.1.3 Pressure Drop

The bundle pressure drop is an important parameter for stability as it affects the core flow distribution and hence has an influence on the bundle flow characteristics and power. To illustrate, core pressure drop equalization for a mixed core of GNF2 and earlier fuel designs at EPU conditions will affect the distribution of core flow to the various bundles, in turn, affecting the radial power distribution and the appropriate characterization of the power-to-flow feedback mechanisms during thermal-hydraulic oscillations.

The NRC staff reviewed the pressure drop qualification for GNF2 fuel. Pressure drop measurements were made for various power levels and power shapes. Figure 2-9 of Supplement 3 provides a comparison of the ISCOR predicted axial pressure profile to pressure tap measurements collected during full-scale testing. In addition, total bundle pressure drops were compared to ISCOR predictions and the comparison is summarized in Figure 2-8 of Supplement 3. The ISCOR pressure drop calculations are consistent with the calculations performed throughout the suite of GEH stability analysis methods (PANACEA and ODYSY). On the basis of these qualification data, the NRC staff concludes that the capability of the analysis methods in terms of predicting the pressure drop is essentially as accurate when applied to GNF2 as with GE14 fuel. Therefore, the NRC staff concludes that the interim approach basis for stability is acceptable for GNF2 fueled EPU and MELLLA+ core applications.

3.5.2 Enhanced Option I-A

The Enhanced Option I-A (EIA) LTS is a prevention solution. Stability calculations are performed to determine exclusion, restricted, and monitored regions. The exclusion region is defined by an area in the power-to-flow operating map where reactor operation is prevented by an automatic flow-biased APRM SCRAM function. The restricted region is a region outside the exclusion region where flow-biased control rod block functions are relied upon to contain reactor operation. The monitored region is outside both the exclusion and restricted regions and is administratively controlled. To define the boundaries of the respective regions, stability calculations are performed using the ODYSY code. These calculations determine the power

[2] Maximum allowable ΔCPR/ICPR in this case refers to the ΔCPR/ICPR associated with an oscillation initiated from the OLMCPR that results in a final MCPR equal to the SLMCPR.

and flow conditions where the decay ratio is a particular value corresponding to that region. For the exclusion region, the decay ratio is limited to 0.8.

GEH provided additional information regarding the bypass void conditions for GNF2 in response to RAI-4 and RAI-11 (Reference 13). The response to RAI-4 confirms that the bypass void fraction will be analyzed on a cycle-specific basis and confirmed to remain below five percent for GNF2 fuel at the LPRM Level-D elevation. The response to RAI-11 considers the conditions of high decay ratio for GNF2 and confirms that the ODYSY application remains within the previously reviewed range of void conditions. Therefore, the NRC staff finds that the ODYSY calculated exclusion region is determined within the qualification range of the methodology and is acceptable.

3.5.3 Option I-D

The Option I-D LTS has both prevention and detect and suppress elements. In terms of prevention, an administratively controlled exclusion region and a buffer region are calculated on a cycle-specific basis. These regions are defined by points along the NCL and the high flow control line (HFCL) where the decay ratio is calculated to be a certain value. For example, the boundary points for the exclusion region are determined where ODYSY calculations predict a decay ratio of 0.8. The boundary is established according to either a generic shape function (GSF) or a modified shape function (MSF) (Reference 23). The detect and suppress function is provided by a flow-biased APRM SCRAM that initiates a reactor trip when core-wide power oscillations reach a sufficient magnitude.

Supplement 3, Appendix A states that IMLTR Limitation 18 will be met. Limitation 18 requires a setpoint setdown of two percent for the APRM to account for miscalibration of the nuclear instruments under bypass void conditions. In response to RAI-4 (Reference 13), GEH confirmed that the bypass void fraction at steady state conditions will be limited to five percent based on cycle-specific analyses. Therefore, the NRC staff finds that compliance with Limitation 18 ensures adequate stability protection for Option I-D.

3.5.4 Option II

The Option II LTS has both prevention and detect and suppress elements. In terms of prevention, an administratively controlled exclusion region is calculated on a cycle-specific basis. The exclusion region is defined by points along the NCL and HFCL where the decay ratio is calculated to be 0.8. The boundary is established according to either a GSF or MSF (Reference 23). The detect and suppress function is provided by a flow-biased quadrant-based APRM SCRAM.

Supplement 3, Appendix A states that IMLTR Limitation 18 will be met. Limitation 18 requires a setpoint setdown of two percent for the APRM to account for miscalibration of the nuclear instruments under bypass void conditions. In response to RAI-4 (Reference 13), GEH confirmed that the bypass void fraction at steady state conditions will be limited to five percent based on cycle-specific analyses. Therefore, the NRC staff finds that compliance with Limitation 18 ensures adequate stability protection for Option II.

3.5.5 Option III

The Option III LTS is primarily a detect and suppress solution. This LTS operates, in principle, by utilizing LPRM signals in local regions of the core to determine if there are local oscillations. This makes the Option III LTS well suited for large BWR cores where the likelihood of regional mode oscillations is higher. LPRM signals are combined into OPRM cells. Automatic suppression takes place when OPRM signals exceed the trip setpoint (determined on a cycle-specific basis). The OPRM SCRAM is based on the period-based detection algorithm (PBDA), which initiates a reactor SCRAM signal when coherent unstable oscillations of a pre-determined magnitude are detected. The magnitude is determined according to the DIVOM curve based on several parameters, including the cycle-specific OLMCPR and SLMCPR (Reference 24).

Supplement 3, Appendix A states that IMLTR Limitation 18 will be met. Limitation 18 requires a setpoint setdown of five percent for the OPRM to account for miscalibration of the nuclear instruments under bypass void conditions. In response to RAI-4 (Reference 13), GEH confirmed that the bypass void fraction at steady state conditions will be limited to five percent based on cycle-specific analyses. The response to RAI-11 (Reference 13) states that the setpoint setdown is conservatively applied for Option III plants. Therefore, the NRC staff finds that compliance with Limitation 18 ensures adequate stability protection for Option III.

Option III plants have the option of incorporating a backup stability protection (BSP) feature instead of BWROG interim corrective actions (Reference 26). The NRC staff requested additional information regarding BSP in RAI-17. The BSP determines a scram region in the power-to-flow map similar to the exclusion region in Options I-D and EIA. The response to RAI-17 provides a description of the licensing analyses that are performed on a cycle-specific basis and confirmed that they are largely similar to those performed for the other LTSs (Reference 13). The NRC staff reviewed the applicability of ODYSY for performing the necessary decay ratio analyses. The NRC staff concluded that ODYSY is well qualified to analyze the thermal-hydraulic conditions anticipated for its application to GNF2 at the exclusion region boundary. Therefore, the NRC staff finds that its use for BSP analyses for GNF2 fuel is acceptable.

3.5.6 Detect and Suppress Solution – Confirmation Density

The Confirmation Density Algorithm (CDA) is the licensing basis protection function of the DSS-CD. The CDA is designed to recognize a developing coherent instability and initiate control rod insertion before the power oscillations increase much above the noise level. The CDA capability of early detection and suppression of instability events is achieved by relying on the successive confirmation period element of PBDA. The CDA employs an amplitude OPRM signal discriminator to minimize unnecessary spurious reactor scrams from neutron flux oscillations at or close to the OPRM signal noise level. The CDA identifies a confirmation density (CD), which is the fraction of operable OPRM cells in an OPRM channel that reach a target successive oscillation period confirmation count. When the CD exceeds a preset number of OPRM cells, and any of the confirming OPRM cell signals reaches or exceeds the amplitude discriminator setpoint, an OPRM channel trip signal is generated. The amplitude discriminator setpoint is generically provided in the DSS-CD LTR or can be established as a plant-specific parameter that is set to bound the inherent plant-specific noise.

The DSS-CD BSP methodology describes two BSP options that are based on selected elements from three distinct constituents: (a) manual; (b) automated; and (c) BSP boundary. The two BSP options are:

> Option 1: consists of the BSP Manual Regions, BSP Boundary and associated operator actions.
>
> Option 2: consists of the Automated BSP (ABSP) Scram Region, as implemented by the APRM flow-biased scram setpoint and associated rod-block setpoints, and associated operator actions.

For BSP Option 1, the reactor power is reduced below the BSP Boundary so that two-recirculation pump trip (2RPT) does not result in operation inside the Exclusion Region. For BSP Option 2, a scram is automatically generated if the reactor enters the Exclusion Region. Both BSP options rely on calculations to demonstrate that instabilities outside the Exclusion Region are not likely. The sample Technical Specifications (TS) in the DSS-CD LTR delineate specific implementation requirements for both BSP options when the OPRM system is declared inoperable.

Given the similarities between the features of DSS-CD and other stability solutions (namely Options I-D, EIA, and III), the technical basis for the staff's conclusions documented in the preceding sections is applicable to DSS-CD.

The NRC staff requested additional information in RAI-18 regarding the analyses performed to support DSS-CD, particularly in the context of GNF2. The response to RAI-18 provides reference to the evaluation procedures that guide the applicability of DSS-CD to fuel transitions, such as to GNF2, or in cases where GNF2-fueled reactors implement DSS-CD (Reference 13). Tables 6-3, 6-4, and 6-5 of Reference 26 describe the approved evaluation procedure. In particular, scenario 1b listed in Table 6-5 describes the analysis conditions required to support DSS-CD licensing evaluations for GNF2 fuel. Calculations must be performed using TRACG for regional mode oscillations under natural circulation conditions (induced by 2RPT or single recirculation pump trip from the highest core power level) and partial flow reduction. These calculations must be performed using reasonably limiting best-estimate TRACG calculations. Table 6-5 provides a description of the core designs that must be considered in the analysis.

The NRC staff has approved these evaluation procedures and analysis scenarios for various fuel transitions (Reference 26). The response to RAI-18 further clarifies that the analysis sensitivities to the uncertainty parameters for the DSS-CD licensing evaluations is the same as described in Section 2.6 of Supplement 3 (References 4 and 13). In the DSS-CD licensing analysis, plant simulations are performed to directly assess the CPR margin under transient events that evolve into unstable reactor conditions. Due to the best-estimate, one-analysis approach for DSS-CD, the NRC staff agrees with the response insofar as Section 2.6 of Supplement 3 provides a list of these basic phenomena and uncertainties affecting the simulation of an instability event.

However, the Supplement 3 pressure drop qualification for GNF2 considers the performance of the ISCOR methodology. The response to RAI-18 includes documentation of the qualification of TRACG to analyze the pressure drop based on the GNF2 pressure drop tests (Reference 13). The NRC staff compared the pressure drop qualification for GNF2 provided in the RAI-18 response with the GE14 results provided in Figure 3.5-5 of the TRACG qualification LTR (Reference 27). Figure 1 from the RAI-18 response and Figure 3.5-5 from the TRACG qualification LTR are plotted on different bases (mass flux as opposed to bundle power). However, the agreement between the measurements and calculations is consistent. Therefore, the NRC staff concludes that the TRACG methodology is essentially as accurate in the calculation of the pressure drop for GNF2 as for GE14. On this basis, the NRC staff concludes that the pressure drop calculation capability in TRACG has been adequately demonstrated for GNF2 and is acceptable.

The NRC staff has generically reviewed the uncertainties associated with GNF2 in terms of the parameters described in Section 3.5.1 of this SE and found that these uncertainties are essentially the same for GNF2 as for GE14. Therefore, the NRC staff finds that the licensing analyses for the implementation of DSS-CD using TRACG are acceptable for GNF2 application at EPU or MELLLA+ conditions.

3.6 Applicability of Thermal-hydraulic Models

The NRC staff conducted a review of the continued applicability of several thermal-hydraulic models in the GEH/GNF safety analysis methods to GNF2 at EPU or MELLLA+ operating conditions. These models include the critical power correlation (GEXL17), the void-quality correlation, the in-core liquid entrainment model, the counter current flow limitation correlation, and the spray heat transfer models. The NRC staff selected these models based on the potential sensitivity of these phenomena to the GNF2 bundle geometry and/or spacer design.

3.6.1 Critical Power Correlation (GEXL17)

The NRC staff conducted an audit of the GEXL17 critical power correlation for GNF2 fuel as part of the GESTAR II compliance audit (Reference 6). The GEXL17 correlation is described in Reference 16. The NRC staff found that the GEXL17 correlation was acceptable (References 6 and 7). Operation in an expanded operating domain does not inherently imply that the correlation is applied outside its range of validation. At OLTP, EPU, and MELLLA+ conditions, the bundles are required to be operated above the OLMCPR.

For expanded operating domains, the fluid conditions are constrained by the CPR limits to ensure that fuel failures do not occur as a result of boiling transition. The NRC staff reviewed the application range of the GEXL17 correlation reported in Supplement 3 for consistency with the application range audited by the NRC staff as part of the GESTAR II compliance audit and confirmed that these ranges were identical. On this basis, the NRC staff finds that the GEXL17 correlation remains equally acceptable for use in evaluating critical power margins for expanded operating domain applications for GNF2 fuel.

The GEXL17 correlation statistics are utilized in the SLMCPR calculation to account for the uncertainties and biases associated with the correlation. The process for the treatment of these uncertainties is unchanged for EPU or MELLLA+ licensing evaluations. Therefore, the NRC staff finds that the correlation applicability and treatment of the associated uncertainties are adequately addressed in the analysis methodology.

3.6.2 Void Quality Correlation

Void fraction is calculated in the GEH/GNF codes using the Findlay-Dix void-quality correlation. The NRC staff has previously reviewed the basis for the Findlay-Dix correlation and found that the supporting database is limited in that it does not extend to the conditions of modern expanded operating domains, such as EPU or MELLLA+. Additionally, full-scale data has not been collected that is representative of conditions associated with modern fuel design features (such as PLRs or modern spacers) or with power distributions that are consistent with current fuel designs and reactor operating strategies.

The NRC staff concluded that additional qualification was required to support the application of the correlation to EPU or MELLLA+ conditions. In the interim, to assure adequate safety, the NRC staff imposed a penalty to the OLMCPR of 0.01. This requirement is provided in Limitation 19 of the SE for the IMLTR (Reference 2). As discussed in Section 3.2.2.3 of this SE, the same OLMCPR penalty is applied to the GNF2 fuel. To support the adequacy of this penalty, GEH has provided a partial qualification of the Findlay-Dix correlation for application to the GNF2 fuel design.

During its review of the IMLTR, GEH committed to submit a supplement to the IMLTR that will provide qualification of the Findlay-Dix correlation against data collected for modern fuel designs including 10X10 lattices with PLRs. This commitment was communicated to the NRC by letter dated November 3, 2006 (Reference 28). The updated qualification is based on a two-pronged approach. The qualification includes the use of pressure drop data to indirectly qualify the void-quality correlation as well as computational benchmarks using the COBRAG sub-channel thermal-hydraulic code. In Supplement 3, GEH provided a subset of qualification for GNF2 against pressure drop measurements and COBRAG calculations that is generally consistent with the type of information GEH has committed to provide as a supplement to the IMLTR. These qualification data, however, are limited in scope and do not form a sufficient basis to eliminate the OLMCPR penalty.

The NRC staff reviewed the limited scope qualification to determine if features of the GNF2 fuel design result in significant errors or biases in the void-quality correlation such that the magnitude of the OLMCPR penalty established for GE14 would be insufficient. Therefore, the NRC staff reviewed the information to ensure consistency in the predictive capability of the correlation to predict void fraction for GNF2 relative to previous fuel designs.

3.6.2.1 Pressure Drop Data

The pressure drop qualification is depicted in Figure 2-8 of Supplement 3. The data were collected for cosine and inlet peaked power shapes as well as for zero power conditions. Given the contribution of the elevation head to the overall pressure loss, it is difficult to predict consistent pressure drops correlated with measurement data when significant errors or biases are present in the void-quality correlation. The data provided indicate consistency between the predicted and measured pressure drops over a wide range of pressure loss and power conditions. This provides a certain degree of assurance that the void-quality correlation performs well for the GNF2 fuel design.

The qualification data does not provide details regarding trends in the data. Such information should be provided in the committed IMLTR supplement to demonstrate the robustness of the void-quality correlation for high void fraction ranges, low flow conditions, and variation in axial geometry. However, for the current purpose of demonstrating that the correlation predicts results consistently for GNF2 fuel relative to GE14 fuel, the NRC staff finds that the submittal is sufficient.

Figure 2-9 of Supplement 3 depicts the comparison of predicted and measured cumulative pressure drop. This figure demonstrates the relative performance of the pressure drop calculational method over the full range of the bundle height. Under the conditions presented, the outlet void fraction is high, nearly 90 percent, which is slightly lower than the maximum void conditions expected for EPU or MELLLA+ operation (e.g., 95 percent). The data indicate that the cumulative pressure drop calculation matches the data well. This provides assurance that the elevation pressure head is being consistently calculated over the length of the fuel bundle. Therefore, this provides additional assurance that the correlation appropriately evaluates the void fraction above the PLRs. The qualification, albeit, is essentially integral in nature; however, it is reasonable to conclude that good agreement between the calculated and measured local pressure drops provides assurance that the individual pressure loss components are adequately treated. The elevation head term requires the accurate prediction of the in-channel void fraction.

3.6.2.2 COBRAG Comparison

Figure 2-6 of Supplement 3 provides a calculational benchmark of the Findlay-Dix correlation for GNF2 fuel. The figure depicts the axial void profile for GNF2 evaluated using the Findlay-Dix correlation and the void profile calculated using the COBRAG sub-channel code[3].

The COBRAG model description has been submitted to the NRC staff and is provided in Reference 29. COBRAG is a sub-channel code that has been used internally by GNF to predict critical power. The code includes a detailed two-fluid, multi-field model. The inter-phase phenomena of shear, heat transfer, entrainment, and deposition are explicitly treated with

[3] The response to RAI-10 confirms that the version of COBRA used to perform the analysis is COBRAG (Reference 13).

detailed constitutive relationships. The code also includes explicit models for inter-channel phenomena such as void drift and mixing (Reference 29).

The NRC staff has not conducted a review of the COBRAG code, but notes, based on the model description document, that the code includes a robust modeling approach to predict the flow characteristics for BWR fuel. The TRACG interfacial shear model is based on the COBRAG model and has been qualified against several void fraction measurement data (Reference 27). The NRC staff, therefore, accepts the use of COBRAG to provide a computational benchmark for the current purposes on the basis that it provides a higher-order calculation.

As shown in Figure 2-6 of Supplement 3, the COBRAG calculations and the predictions of the Findlay-Dix correlation provide fully consistent predictions of the local radially-averaged void fraction through the entire length of the bundle. The calculation is performed to a high outlet in-channel void fraction (approximately 92 percent) that is consistent with the expected maximum outlet void fractions for EPU operation (Reference 2).

Minor differences are observed in the COBRAG and Findlay-Dix correlation in the mid-region of the node where the in-channel void fraction is between 70 and 80 percent. These differences, however, are approximately 1 percent. The NRC staff judged these differences to be negligible based on the quoted uncertainty of the correlation per Reference 30.

3.6.2.3 Void-Quality Correlation Conclusion

A set of qualification data similar to those committed to be provided by Reference 28 was provided in Supplement 3 to justify the applicability of the Findlay-Dix void quality correlation to GNF2 fuel. The NRC staff has previously reviewed this approach to qualify the void-quality correlation and, as documented in its SE for the IMLTR, has found that this approach is acceptable (Reference 2). The NRC staff finds that this set of data is insufficient to fully qualify the correlation as it lacks substantial trend data. However, the NRC staff does find that this set is sufficient for the current review purpose, which is to demonstrate a consistency in the performance of the correlation for GNF2 and GE14 fuel.

The calculated and measured void fractions in the qualification set are similar to the maximum void range expected for EPU operation (89 to 92 percent). Therefore, the NRC staff finds that a sufficient range has been considered for the current purpose. On the basis of the close agreement of the measured and calculated pressure drop for GNF2, the NRC staff concludes that there is reasonable assurance that the Findlay-Dix correlation does not introduce significant bias in the prediction of the void fraction for GNF2 fuel relative to GE14 fuel. The cumulative pressure drop data indicate that no biases are introduced at the geometric variations above PLRs. This provides additional assurance that the GNF2 design features do not pose an inherent challenge to the validity of the correlation.

Calculations performed using the higher-order COBRAG thermal-hydraulics code confirm that the Findlay-Dix correlation performs well for GNF2. The calculations do not indicate any degradation in the correlation relative to the detailed two-fluid, multi-field calculation with either void fraction or axial elevation. Differences in the calculations are negligible compared to the correlation uncertainty reported in Reference 30.

On these bases, the NRC staff can conclude that the GEH basis for the applicability of Findlay-Dix to GE14 applies equally to GNF2. Therefore, the NRC staff finds that the OLMCPR penalty of 0.01 in IMLTR Limitation 19 is adequate to bound any uncertainty in the correlation as it is applied to GNF2 fuel at EPU or MELLLA+ conditions.

3.6.3 In-core Liquid Entrainment

The NRC staff requested additional information in RAI-13 regarding how in-core liquid entrainment is modeled for the GNF2 fuel bundle. Specifically, the NRC staff noted in its RAI that the TRACG code includes geometry-dependent parameters in the treatment of liquid entrainment. The response to RAI-13 states that the GEH ECCS-LOCA method is SAFER and that the SAFER code relies on the Findlay-Dix void quality correlation (Reference 13). The transient code ODYN also relies on the Findlay-Dix correlation for AOO and ATWS analysis; similarly, the ODYSY code, derived from ODYN, relies on the same correlation for stability analysis (Reference 4).

The NRC staff has reviewed the qualification of the Findlay-Dix void quality correlation for GNF2 fuel. This qualification was provided in the form of comparison to pressure drop data and code-to-code comparisons against the detailed two-fluid COBRAG code. The detailed NRC staff review of this qualification is provided in Section 3.6.2 of this SE. The NRC staff has found that the data and code-to-code comparisons indicate equivalent performance of the correlation for GNF2 fuel relative to GE14 fuel. Therefore, the NRC staff concludes that the use of the Findlay-Dix correlation in the transient and accident analysis methods remains equally acceptable for GNF2 fuel relative to GE14 fuel.

The RAI-13 response also addresses CORECOOL. CORECOOL is a detailed three-field model that is commonly used to analyze core heatup for plants with high PCT where core spray heat transfer is important (e.g., BWR/2 plants). Under the conditions where CORECOOL is applied, the vapor upward flow is small and no entrainment is predicted by CORECOOL. Section 5.1.2 of NEDE-30996P-A lists the small steam flow rate as a basic assumption in the CORECOOL method (Reference 31). The RAI-13 response states that the GNF2 geometry is not relevant for CORECOOL from a liquid entrainment perspective since it is not expected or predicted to occur under the relevant LOCA conditions (Reference 13). As liquid entrainment is not expected or predicted to occur with the low vapor upward flows at the plant conditions where CORECOOL is applied, the NRC staff agrees that its treatment in CORECOOL is irrelevant.

On these bases, the NRC staff finds the treatment of the physical process of entrainment is adequately captured in the methods.

3.6.4 Counter Current Flow Limitation

The NRC staff requested additional information in RAI-13 regarding the calculation of the counter current flow limitation (CCFL) for GNF2 fuel. The response clarifies that the CCFL correlation is a modified version of the Wallis correlation. The modified Wallis correlation eliminates that characteristic length from the superficial velocity term and combines this length with the constant "K" on the right hand side of the equation (Reference 13). The response states that the modified constant is directly obtained from GNF2-specific experiments (Reference 13). Therefore, the NRC staff agrees that the GNF2 geometry is inherently captured in the modified Wallis correlation.

The RAI-13 response states that for the GE8 and later fuel designs, the upper tie plate (UTP) was opened to reduce pressure drop. As a consequence for GE8 and later designs, the location where CCFL occurs has moved downward in the bundle to the location of the spacer. Confirmatory CCFL testing for the GNF2 spacers has been performed at Stern labs (Reference 13). The NRC staff finds that the experiments form a valid basis for the justification of the CCFL correlation for the GNF2 design.

Aside from the direct experiments, the response mentions a conservatism in the SAFER methodology for ECCS-LOCA analysis whereby the CCFL constants are scaled to the UTP flow area and the smaller value is used in SAFER (Reference 13). The NRC staff notes that the CCFL will occur at the axial point where the flow is most restricted. For the GNF2 fuel design this occurs at the transition between the fully rodded region and the region above the short PLRs. The depth of this point is below the core midplane. In SAFER, the CCFL is treated as occurring at the UTP. This is a conservative feature of the ECCS-LOCA analysis, particularly for GNF2 where tests have confirmed CCFL to occur much lower in the bundle. Therefore, SAFER will conservatively predict the point of CCFL during design basis LOCA analyses.

On the basis that the CCFL correlation has been experimentally validated for the GNF2-specific bundle design and that the inherent treatment of CCFL in the SAFER methodology is conservative, the NRC staff finds that the continued applicability of the CCFL methodology to GNF2 for expanded operating domains is acceptable. Further, the experimental basis for the GNF2 CCFL correlation ensures that the treatment of this phenomenon is equally valid as for previous fuel designs, such as GE14. Therefore, the CCFL model is acceptable.

3.6.5 Spray Heat Transfer

In its review of the applicability of the analysis methods for GNF2 fuel, the NRC staff considered the detailed treatment of the spray heat transfer. Spray heat transfer is conservatively neglected in SAFER. However, CORECOOL provides a more detailed model of the spray heat transfer and is an optional approach to model core heatup in ECCS-LOCA analyses.

The NRC staff notes that the CORECOOL model is typically not applied for BWR/3-6 plants where large PCT margins exist. However, future use of CORECOOL for BWR/3-6 plants at EPU or MELLLA+ conditions is not precluded. Additionally, the NRC staff notes that while utilized for BWR/2 plants currently, the NRC staff has not approved the generic MELLLA+ LTR (Reference 32) for BWR/2 plants. However, no methodology restrictions have been imposed on the IMLTR regarding application to BWR/2 plants at EPU conditions. Therefore, the NRC staff reviewed these models noting that, while their application is not expected, the use of these methods for GNF2 fuel at EPU conditions is not precluded.

The NRC staff requested additional information in RAI-13 regarding the applicability of the CORECOOL core spray heat transfer model to GNF2. The response refers to the CORECOOL model description provided in NEDE-30996P-A (Reference 31). CORECOOL has been qualified against GE, AB Atomenergi, Toshiba, and Hitachi full-scale core spray heat transfer data. These qualification data are presented in Section 7 of Reference 31. Section 5.1 of Reference 31 provides a description of the CORECOOL model. The NRC staff agrees that the models are mechanistic in nature and may be applied to various configurations within the capabilities of the code.

The GNF2 GESTAR II Compliance Report (Reference 3) has been revised to address the licensing aspects of loading GNF2 fuel in BWR/2 plants. Section 3.11 of the report includes a discussion of the applicability of the CORECOOL methodology to analyze GNF2 fuel. The report states that the CORECOOL method allows for the specification of several rod groups that enable the code to explicitly model varying lengths of the PLRs (Reference 3).

The response to RAI-13, however, states that the GNF2 licensing analyses are performed using a conservative rod grouping in CORECOOL (Reference 13). This rod grouping treats the set of SPLRs as extending above the core midplane and treats the long PLRs (LPLRs) as full bundle height. The current modeling approach conservatively increases the active length of these SPLRs and limits the effectiveness of radiation heat transfer. Therefore, CORECOOL analyzes the SPLRs as if they extend higher into the core and reach the point of peak nodal power for mid-peaked axial power shapes and treats the area above the LPLRs as fully rodded, thus minimizing radiation heat transfer to the coolant. The NRC staff concludes that this approach does not explicitly consider the axially varying geometry, but does conservatively treat the rod grouping so as to increase the calculated PCT by increasing the power in the limiting power locations and limiting the heat transfer from potentially limiting rods. On this basis, the NRC staff finds that the application of CORECOOL to GNF2 is conservative, and therefore acceptable.

4.0 CONCLUSION

The NRC staff has found that the qualification provided in Supplement 3 demonstrates equivalent performance of the GEH methods suite to analyze GNF2 as that demonstrated for GE14 fuel. This includes the neutronic, thermal-hydraulic, and T-M[4] aspects of the methods.

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[4] The T-M review considered the GNF2-specific exposure limit provided by GESTAR II, Amendment 32. This exposure limit does not necessitate the critical pressure penalty imposed on GSTRM calculations for GE14. Now that the advanced PRIME T-M methodology and GESTAR II, Amendment 33 have been approved by the NRC staff, this specific exposure limit has been revised and the critical pressure penalty imposed on GSTRM does not apply to GNF2 when the PRIME methodology is used.

Therefore, the NRC staff finds that the extension of the approval of the interim methods process to GNF2 fuel is acceptable. To this end, the NRC staff has revised IMLTR SE Limitation 22 to extend application of the neutronic methods to GNF2 lattices without further review.

Limitation 22 from the SE for the IMLTR states:

> For any plant-specific applications of TGBLA06 with fuel type characteristics not covered in this review, GE needs to provide assessment data similar to that provided for the GE fuels. The Interim Methods review is applicable to all GE lattices up to GE14. Fuel lattice designs, other than GE lattices up to GE14, with the following characteristics are not covered by this review:
>
> - Square internal water channels water crosses
> - Gd [gadolinia bearing] rods simultaneously adjacent to water and vanished rods
> - 11x11 lattices
> - MOX [mixed oxide] fuel
>
> The acceptability of the modified epithermal slowing down models in TGBLA06 has not been demonstrated for application to these or other geometries for expanded operating domains.
>
> Significant changes in the Gd rod optical thickness will require an evaluation of the TGBLA06 radial flux and Gd depletion modeling before being applied. Increases in the lattice Gd loading that result in nodal reactivity biases beyond those previously established will require review before the GE methods may be applied.

On the basis of the subject review, the NRC staff finds that Supplement 3 addresses the applicability of the GEH analysis methods to GNF2 fuel. Therefore, the NRC staff has revised Limitation 22 in Section 9.22 of the IMLTR SE as follows:

> This Limitation has been revised according to Appendix K of this SE.
>
> For any plant-specific applications of TGBLA06 with fuel type characteristics not covered in this review, GEH needs to provide assessment data similar to that provided for the GEH/GNF fuels. The Interim Methods review is applicable to all GEH/GNF lattices up to GNF2. Fuel lattice designs, other than GEH/GNF lattices up to GNF2, with the following characteristics are not covered by this review:
>
> - Square internal water channels water crosses
> - Gd rods simultaneously adjacent to water and vanished rods
> - 11x11 lattices
> - MOX fuel
>
> The acceptability of the modified epithermal slowing down models in TGBLA06 has not been demonstrated for application to these or other geometries for expanded operating domains.
>
> Significant changes in the Gd rod optical thickness will require an evaluation of the TGBLA06 radial flux and Gd depletion modeling before being applied. Increases in the

lattice Gd loading that result in nodal reactivity biases beyond those previously established will require review before the GEH methods may be applied.

The NRC staff reviewed Supplement 3 only insofar as it justifies a revision to Limitation 22 of the NRC staff SE for the IMLTR. The NRC staff review in this matter does not impact any other aspects of the original review of the IMLTR. Therefore, all other NRC staff guidance, limitations, and conclusions documented in the SE for the IMLTR remain applicable as originally stated.

5.0 REFERENCES

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