ENCLOSURE 2

MFN 11-170

Applicability of GE Methods to Expanded Operating Domains-Supplement for GNF2 Fuel
NEDO-33173 Supplement 3-A Revision 1,
July 2011

Non-Proprietary Information

IMPORTANT NOTICE

Enclosure 2 is a non-proprietary version of the Applicability of GE Methods to Expanded Operating Domains-Supplement for GNF2 Fuel, NEDC-33173 Supplement 3P-A Revision 1, July 2011 from Enclosure 1, which has the proprietary information removed. Portions that have been removed are indicated by open and closed double brackets as shown here [[ ]].

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GE Hitachi Nuclear Energy

NEDO-33173
Supplement 3-A
Revision 1
DRF 0000-0012-1297
DRF Section 0000-0129-0237-R0
July 2011

Non-Proprietary Information – Class I (Public)

Licensing Topical Report

Applicability of GE Methods to Expanded Operating Domains - Supplement for GNF2 Fuel

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please read carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining NRC approval of the Applicability of GE Methods to Expanded Operating Domains - Supplement for GNF2 Fuel. The only undertakings of GE Hitachi Nuclear Energy with respect to information in this document are contained in the contracts between GE Hitachi Nuclear Energy and its customers or participating utilities, and nothing contained in this document will be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GE Hitachi Nuclear Energy makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

December 28, 2010

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
P.O. Box 780, M/C A-18
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SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY AMERICAS TOPICAL REPORT NEDC-33173P, SUPPLEMENT 3, "APPLICABILITY OF GE METHODS TO EXPANDED OPERATING DOMAINS – SUPPLEMENT FOR GNF2 FUEL" (TAC NO. ME1815)

Dear Mr. Head:

By letter dated July 31, 2009 (Agencywide Documents Access and Management System Accession No. ML092151079), GE Hitachi Nuclear Energy Americas, LLC. (GEH) submitted Topical Report (TR) NEDC-33173P, Supplement 3, "Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel" to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated March 23, 2010, an NRC draft safety evaluation (SE) regarding our approval of TR NEDC-33173P, Supplement 3, was provided for your review and comment. By letter dated June 21, 2010, GEH commented on the draft SE. The NRC staff's disposition of GEH's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR NEDC-33173P, Supplement 3, is acceptable for referencing in licensing applications for GEH-designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

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

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

Enclosure 1 and its Attachment transmitted herewith contain proprietary information. When separated from Enclosure 1 and its Attachment, this document is decontrolled.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

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

Sincerely,

/RA/

John R. Jolicoeur, Acting Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710

Enclosures:
1. Proprietary Final SE with Proprietary Attachment
2. Non-Proprietary Final SE with Non-Proprietary Attachment

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NRR-043

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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.

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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. [

].

K-15

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 Stability

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 <u>In-core Liquid Entrainment</u>

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

1. TR NEDC-33173P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains," dated September 2010. (ADAMS Package Accession No. ML102920129)
2. Final Safety Evaluation by the Office of Nuclear Reactor Regulation for NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated July 21, 2009. (ADAMS Package Accession No. ML092020255)
3. TR NEDC-33270P, Revision 2, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," dated June 2009. (ADAMS Package Accession No. ML091830644)
4. TR NEDC-33173P, Supplement 3, "Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel," dated July 2009. (ADAMS Package Accession No. ML092151094)
5. TR NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)," Revision 16, dated October 2007. (ADAMS Package Accession No. ML091340076)
6. Audit Report for Global Nuclear Fuel (GNF) GNF2 Advanced Fuel Assembly Design GESTAR II Compliance Audit, dated January 2008 (ADAMS Accession No. ML081630625).
7. Mendiola, A., NRC, memorandum to Rosenberg, S., NRC, "Staff Findings Regarding Supplemental Information Pertaining to the Compliance of the GNF2 Fuel Design to the General Electric Standard Application for Reactor Fuel (GESTAR II)," August 4, 2009. (ADAMS Accession No. ML092080499).
8. Letter from GNF to NRC, FLN-2008-011, "Amendment 32 to NEDE-24011-P, General Electric Standard Application for Reactor Fuel (GESTAR II)," dated October 15, 2008. (ADAMS Accession No. ML08291051)
9. Final Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment 32 to NEDE-24011-P-A, "General Electric Standard Application for Reload (GESTAR II)," dated July 30, 2009. (ADAMS Package Accession No. ML091680754).
10. TRs NEDC-33256P-A, NEDC-33257P-A, and NEDC-33258P-A, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance" Parts 1-3, dated September 2010. (ADAMS Package Accession No. ML102600259)
11. TR NEDO-33173, Supplement 4, "Implementation of PRIME Models and Data in Downstream Methods," GEH, July 2009. (ADAMS Accession No. ML091910491)
12. Letter from GNF to NRC, MFN 10-045 Supplement 1, "Amendment 33 to NEDE-24011-P, General Electric Standard Application for Reactor Fuel (GESTAR II)," dated May 27, 2010. (ADAMS Package Accession No. ML101481067)

13. Letter from GEH to NRC, MFN 09-647, "Response to NRC RAIs – NEDC-33173P, Supplement 3," dated October 20, 2009. (ADAMS Package Accession No. ML092990416)
14. TR NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," dated August 1999. (ADAMS Accession No. ML003740151)
15. TR NEDC-32773P, Revision 1, "Advanced Methods Power Distribution Uncertainties for Core Monitoring," dated January 1999.
16. TR NEDC-33292P, Revision 3, "GEXL17 Correlation for GNF2 Fuel," dated June 2009. (ADAMS Package Accession No. ML091830644)
17. Letter from Northern States Power Company - Minnesota to NRC, L-MT-09-017, "Monticello Extended Power Uprate: Response to NRC Reactor Systems Branch and Nuclear Performance & Code Review Branch Request for Additional Information (RAI) dated January 16, 2009 (TAC No. MD9990)," dated March 19, 2009. (ADAMS Accession No. ML090790388)
18. Letter from GEH to NRC, MFN 08-087, "Response to Portion of NRC Request for Additional Information Letter No. 137 – Related to ESBWR Design Certification Application – RAI Numbers 4.3-11 and 4.4-68," dated February 4, 2008. (ADAMS Accession No. ML080380296)
19. TR NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated May 2006. (ADAMS Package Accession No. ML061500182)
20. Final Safety Evaluation by the Office of Nuclear Reactor Regulation of NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated January 8, 2008. (ADAMS Package Accession No. ML091890758)
21. Letter from GEH to NRC, MFN 09-647, Supplement 1, "Response to NRC Request for Additional Information RAI 8 – Re: GE-Hitachi Nuclear Energy Americas, LLC (GEH) Topical Report (TR) NEDC-33173P, Supplement 3, 'Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel' (TAC No. ME1815)," dated November 30, 2009. (ADAMS Package Accession No. ML093360138)
22. Letter from GNF to NRC, MFN 09-106, "Request for Additional Information Response for the PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance (TAC # MD4114)," dated February 27, 2009. (ADAMS Package Accession No. ML090620336)
23. TR NEDE-33213P-A, "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," dated April 2009. (ADAMS Package Accession No. ML091100207)
24. TR NEDO-32465 "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated August 1996. (ADAMS Accession No. ML072260045)
25. Letter from GEH to NRC, MFN 08-693, "Implementation of Methods Limitations – NEDC-33173P (TAC No. MD0277)," dated September 18, 2008. (ADAMS Package Accession No. ML082630471)
26. TR NEDC-33075P-A, Revision 6, "General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density," dated January 2008. (ADAMS Package Accession No. ML080310384)

27. TR NEDE-32177P, Revision 3, "TRACG Qualification," dated August 2007. (ADAMS Package Accession No. ML072480007)
28. Letter from GE to NRC, MFN 06-435, "Commitment to Update GE's Void Fraction Data," dated November 3, 2006. (ADAMS Accession No. ML063110299)
29. TR NEDE-32199P, Revision 1, "COBRAG Subchannel Code Model Description Report," dated July 2007. (ADAMS Package Accession No. ML071910320)
30. TR NEDE-21565, "BWR Void Fraction Correlation and Data," dated January 1977.
31. TR NEDE-30996P-A, Volumes 1 and 2, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," dated October 1987. (ADAMS Package Accession No. ML091210245)
32. TR NEDC-33006P-A, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," dated January 2002. (ADAMS Package Accession No. ML091800530)

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Date: December 28, 2010

Comment Resolution Table for Appendix K – Safety Evaluation of NEDC-33173P, Supplement 3, "Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel."

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
1	Section 1.0, Pg K-2, and Section 3.2.2.5.3, Pg K-19 note: GEH identified the section number in error. Sections 3.2.2.5.1, 3.2.2.5.2, and 3.2.2.6 require similarly updated verbiage. No changes made in Section 3.2.2.5.3.	<u>Section 1.0</u>: … 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. Currently, the NRC staff is reviewing the PRIME T-M methodology (References 10, 11, and 12). Consistent with Limitation 12 and IMLTR Supplement 4 (Reference 13), it is the understanding of the NRC staff that if PRIME is approved, then future licensing evaluations for GNF2 in EPU and MELLLA+ cores will be performed using the updated PRIME T-M methods.	The verbiage regarding the status of the PRIME review should be updated to reflect the current approved status. The highlighted portions deserve reconsideration.	Comment accepted. <u>Highlighted portion of Section 1.0 revised to read</u>: … 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

ATTACHMENT

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		Noting this expectation, the NRC staff understands that the GNF2 methods applicability supplement to the IMLTR (Reference 4) must address the interim GESTAR II Amendment 32 approach as well as an approach that accounts for the use of updated T-M methods if PRIME is approved by the NRC staff. <u>Section 3.2.2.5.1</u>: ... However, Supplement 3 provides that, if the NRC staff approves PRIME, future T-M calculations will be performed using the PRIME T-M methodology. … 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 if the PRIME T-M methodology is approved by 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. <u>Section 3.2.2.5.1 revised to read</u>: … 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. … 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.

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		Section 3.2.2.5.2: … Noting that, in accordance with Limitation 12 and Supplement 3, GEH intends to use PRIME T-M methods for future applications once those methods are approved by the NRC staff, the NRC staff reviewed the operating history parameters assumed in the analysis according to the PRIME Application Methodology LTR (Reference 12). 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 will address the adequacy of its predictions of rod internal pressure. Therefore, the NRC staff finds that the potential migration to the PRIME T-M method, once		Section 3.2.2.5.2 revised to read: 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

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		approved, does not invalidate the basis for the acceptance of the T-M method for extension to application to EPU or MELLLA+ conditions. <u>Section 3.2.2.6</u>: Limitation 12 from the NRC staff SE approving the IMLTR requires that future licensing evaluations be performed using updated T-M methods (Reference 2). PRIME is currently under review by the NRC staff to replace the GSTRM T-M methodology (References 10, 11, and 12). IMLTR, Supplement 4 (Reference 13) provides the implementation plan to update the methods for compatibility with PRIME if PRIME is approved by the NRC. Therefore, the NRC staff expects that the exposure limit will be revised for GNF2 fuel. Supplement 3 provides for this possible outcome and discusses revising the peak pellet exposure limit if PRIME is approved. The NRC staff reviewed the proposed alternative limit for use with the PRIME methodology. In RAI-3, the NRC staff requested that the Supplement 3 language be revised to reflect the status of the		acceptance of the T-M method for extension to application to EPU or MELLLA+ conditions. <u>Section 3.2.2.6 revised to read</u>: 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

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		PRIME review. The response to RAI-3 provides a revision to Supplement 3 that removes the specific exposure limit (Reference 14). The exposure limit for GNF2 is expected to be revised, but must be revised consistent with the NRC staff's approval of the T-M methods. Specifying the exposure limit presumes the outcome of the NRC staff's ongoing review of PRIME and is not necessary to describe the process by which this limit would be revised with the approval of a T-M method. The revised Supplement 3 is consistent with this process and the status of the NRC staff's review of PRIME. 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. Supplement 3 states that once PRIME is approved, the new		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-

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		method will be adopted and the exposure limit will be revised through the GESTAR II licensing process. The NRC staff requested additional information in RAI-3 to ensure this limit is consistent with the NRC staff's approval of the T-M methods. On these bases, that NRC staff finds that the exposure limit for GNF2 is acceptable.		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,

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
				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

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		<u>Footnote 4 in Section 4.0</u>: 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. The NRC staff anticipates that this exposure limit will be revised with the approval of the advanced PRIME T-M methodology.		exposure limit for GNF2, as revised by the review and approval of GESTAR II Amendment 33, is acceptable. <u>Footnote 4 in Section 4.0 revised to read</u>: 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.

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
2	Section 3.2.2.4 Pg K-14	Therefore, the ECCS [emergency core cooling system]-LOCA analysis…	Generic Editorial. Use of square brackets vs. parentheses, e.g., ...ECCS [emergency core cooling system]....	Comment rejected. NRC convention is as follows: When an acronym contained within a quoted citation has not been previously defined in the current document, the acronym definition is inserted into the citation text set off by square brackets.
3	Section 3.2.2.8 Pg K-20	The prediction of soluble boron worth is confirmed by the comparison of TGBLA 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 TGBLA generated boron libraries.	Suggest adding 06 to the acronym for TGBLA in the last paragraph.	Comment accepted. Sentences revised as: 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.
4	Section 3.4.3 Pg K-27		Correct spelling of homogenous to homogeneous.	Comment accepted. Spelling changed.

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
5	Sections 3.5.3, 3.5.4, and 3.5.5 Pg K-35	In response to RAI-4 (Reference 14), GEH confirmed that the bypass void fraction will be limited to five percent based on cycle-specific analyses. (repeated in each section)	Suggest adding clarifying expression “at steady state conditions” as noted in the markup.	Comment accepted. Sentence revised in each of the three sections to read: In response to RAI-4 (Reference 14), GEH confirmed that the bypass void fraction at steady state conditions will be limited to five percent based on cycle-specific analyses.
6	Section 3.5.5 Pg K-36	The BSP determines an exclusion region in the power-to-flow map similar to Option I-D and EIA.	Last Paragraph. Suggest corrections regarding the BSP as follows and as included in the markup. Current: The BSP determines an exclusion region in the power-to-flow map similar to Option ID and EIA. Proposed: The BSP determines a scram region in the power-to-flow map similar to the exclusion region in Option I-D and EIA.	Comment accepted. Sentence revised to read: The BSP determines a scram region in the power-to-flow map similar to the exclusion region in Options I-D and EIA.
7	Section 3.5.6 Pg K-36	The DSS-CD LTS is an evolutionary solution based on the Option III detect and suppress strategy with modifications. The first is the use of the PBDA without a specific oscillation magnitude specified for reactor suppression. That is, the PBDA in	The first and second paragraphs seek to explain the design of the DSS-CD in general terms by comparing it to Option III. It may be better to describe the DSS-CD design directly. We suggest replacing the first and second paragraph with something like	Comment accepted. Section revised to read: The Confirmation Density Algorithm (CDA) is the licensing basis protection function of the DSS-CD. The CDA is designed to recognize a developing coherent

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
		DSS-CD calls for reactor SCRAM on any detected coherent power oscillations of any magnitude. The implementation of the PBDA in DSS-CD may be considered similar to the Option III implementation of the PBDA with a very conservative setpoint. To prevent spurious SCRAMs, the DSS-CD solution uses the confirmation density algorithm (CDA). The CDA has only one setpoint, which is the fraction of active OPRM cells that must confirm unstable oscillations before a SCRAM is initiated (Reference 27). The second primary difference is the BSP. BSP is provided for instances where the DSS-CD is declared inoperable, such that automatic suppression will occur under conditions adverse to stability. This feature is necessary for MELLLA+ operation where a dual recirculation pump trip (2RPT) event may result in rapidly growing power oscillations.	the following. 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 Period Based Detection (PBDA). The CDA employs an amplitude OPRM signal discriminator to minimize unnecessary spurious reactor scrams from neutron flux oscillations at or close to the Oscillation Power Range Monitor (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	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.

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
			amplitude discriminator setpoint, an OPRM channel trip signal is generated. The amplitude discriminator setpoint is generically provided in the DSS-CD Licensing Topical Report or can be established as a plant-specific parameter that is set to bound the inherent plant-specific noise. The DSS-CD Backup Stability Protection (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	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

#	Location in Draft SE	Draft SE Text	GEH Comment and Basis	NRC Staff Resolution
			Boundary so that two-recirculation pump trip 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.	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.
8	Section 5 Pg K-45		The date for Reference 2 should be the date of the final SE which is July 21, 2009. The ML number may need to be changed as well.	Comment Accepted. Reference information updated.
9	Section 5 Pg K-45		Reference 9 appears to be an internal draft of the Amendment 32 SE. It should be changed to the final SE which is dated July 30, 2009. The ML number may need to be changed as well.	Comment Accepted. Reference information updated.

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REVISION SUMMARY

Revision Number	Page(s)	Description of Change
0	--	Initial Issue
1	vii	Added this Revision Summary
1	--	Updated Cover Page consistent with Publication Handbook
1	ii	Added NRC Proprietary markup paragraph in the Proprietary Information Notice in the back of cover page.
1	2-11	A typographical error in Table 2-1 was corrected in response to RAI 7.
1	2-13, 2-14, 2-15 and 2-17	Figures 2-1, 2-2, 2-4 and 2-5 have been modified to reflect the correct average density variation of the vanished rod lattices in response to RAI 6.
1	2-31	Section 2.7: Discussion of peak pellet exposure limit for GE14 fuel was deleted in response to RAI 3
1	4-1	Section 4.2: Correction to Table in Section 4.2 in response to RAI 14.a, 14.b and 14.d.
1	B1-B76	Added Appendix B – GEH Responses to RAIs

EXECUTIVE SUMMARY

NEDC-33173P, *Applicability of GE Methods to Expanded Operating Domains* (Methods LTR) (Reference 1), documents the adequacy of the GEH analytical methods at expanded operating domains (e.g., Extended Power Uprate and MELLLA+). The NRC approved the Methods LTR as documented in its Safety Evaluation dated July 21, 2009 (Reference 3). NEDC-33173P, Section 4.2, "Applicability," states, in part, that the Methods LTR is applicable to current GE BWR fuel designs (i.e., GE14 and earlier). The NRC SE states in Section 8.2 and Limitation 22 that the NRC's review of the Methods LTR is limited to the current GEH fuel designs (i.e., GE14 and earlier). GNF has developed a new fuel design, GNF2, which is described in GNF Report NEDC-33270P, Revision 2, June 2009, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)", (Reference 4). The purpose of this supplement is to document the adequacy of the GEH analytical methods relative to GNF2 fuel when used for expanded operating domains. The GNF2 fuel product design is based on the proven GE12 and GE14 10x10 lattice, water rod and fuel rod design. The major differences between GE14 and GNF2 are an advanced fuel rod spacer design and changes in part length rod placement and length.

The evaluations presented in Sections 2 and 3 demonstrate the adequacy of the GEH methods for GNF2 when used in the expanded operating domains. Further, the assessment in Appendix A documents the applicability the existing limitations in the NRC SE for the Methods LTR (Reference 3) for GNF2 fuel.

The outline and format of the report is identical to the original document NEDC-33173P (Reference 1), in which the methods uncertainty impact on the key core safety parameters is evaluated. This consistent format is chosen to facilitate the clarity and completeness of the supporting information. This Supplement 3 does not depend on other Supplements to NEDC-33173P. Other supplements to NEDC-33173P will support GNF2 fuel, as necessary.

ACRONYMS AND ABBREVIATIONS

Term	Definition
AOO	Anticipated Operational Occurrence
Methods LTR	Applicability of GE Methods to Expanded Operating Domains Licensing Topical Report
APRM	Average Power Range Monitor
ATWS	Anticipated Transient Without Scram
BOC	Beginning Of Cycle
BT	Boiling Transition
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CDA	Confirmation Density Algorithm
CPPU	Constant Pressure Power Uprate
CPR	Critical Power Ratio
DIVOM	Delta over Initial MCPR Versus Oscillation Magnitude
DSS-CD	Detect and Suppression Solution – Confirmation Density
ΔCPR	Delta Critical Power Ratio
ECCS	Emergency Core Coolant System
EOC	End Of Cycle
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
FMCPR	Final Minimum Critical Power Ratio
FWCF	Feedwater Controller Failure Event
FWHOOS	Feedwater Heating Out-of-Service
FSAR	Final Safety Analysis Report
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy
GESTAR	General Electric Standard Application for Reload Fuel
GEXL	GE Boiling Transition Correlation
GSTRM	GESTR Mechanical
HBB	Hard Bottom Burn
HCOM	Hot Channel Oscillation Magnitude
ICPR	Initial Critical Power Ratio
IV	Instantaneous Void
LHGR	Linear Heat Generation Rate
LOCA	Loss Of Coolant Accident
LTR	Licensing Topical Report
LPRM	Local Power Range Monitor
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate

Term	Definition
MCPR	Minimum Critical Power Ratio
MELLLA+, M+	Maximum Extended Load Line Limit Analysis Plus
MNCP	A General Monte Carlo N-Particle Transport Code
NRC	Nuclear Regulatory Commission
ODYN	1-D Transient Model
ODYSY	GE Best-Estimate Frequency Domain Stability Code
OLMCPR	Operating Limit MCPR
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
Option II	Stability Detect and Suppress LTS for BWR/2
Option III	Stability OPRM-Based Detect and Suppress LTS
PANACEA	Current GE BWR Core Simulator
PCT	Peak Cladding Temperature
PHE	Peak-Hot Excess
PLR	Part Length Rod
PU	Power Uprate
RAI	Request for Additional Information
RPS	Reactor Protection System
SAFDLs	Specified Acceptable Fuel Design Limits
SDM	Shutdown Margin
SE, SER	Safety Evaluation Report
SLMCPR	Safety Limit MCPR
SLO	Single Loop Operation
TGBLA	Current GE BWR lattice physics code
TIP	Traversing In-Core Probes
TRACG	Transient Reactor Analysis Code (GE proprietary version)
UB	Under Burn
UTL	Upper Tolerance Limit
VH	Void History
1-D	One Dimensional
3-D	Three Dimensional

1.0 INTRODUCTION

1.1 BACKGROUND

NEDC-33173P, *Applicability of GE Methods to Expanded Operating Domains* (Methods LTR) (Reference 1), documents the adequacy of the GEH analytical methods at expanded operating domains (e.g., Extended Power Uprate and MELLLA+). The NRC approved the Methods LTR as documented in its Safety Evaluation dated July 21, 2009 (Reference 3). NEDC-33173P, Section 4.2, "Applicability," states, in part, that the Methods LTR is applicable to current GNF BWR fuel designs (i.e., GE14 and earlier). The NRC SE states in Section 8.2 and Limitation 22, the NRC's review of the Methods LTR is limited to the current GNF fuel designs (i.e., GE14 and earlier). GNF has developed a new fuel design, GNF2, which is described in GNF Report NEDC-33270P, Revision 2, June 2009, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)", (Reference 4). The purpose of this supplement is to document the adequacy of the GEH analytical methods relative to the GNF2 fuel when used at expanded operating domains.

GNF has introduced a new fuel design, known as GNF2, described in Reference 4, based on the GE14 design. The GNF2 and GE14 design parameters are compared in Reference 4 and summarized in Table 1-1. The major differences between GE14 and GNF2 are:

- **Part length rod placement and design.** The GNF2 design contains fourteen part length rods, identical to GE14. However, six of the part length rods are short, about one third of the full rod length, and eight are longer, about two thirds of the full rod length. The positions of the part length rods have changed, with the six short part length rods clustered in the center of the lattice and the eight long part length rods located adjacent to the fuel channel.

- **Fuel rod grid spacer design and placement.** Whereas the GE14 spacer grid is a zircaloy ferrule design with Alloy X-750 springs, the GNF2 spacer grid is based on an egg crate configuration and is made entirely of Alloy X-750. The axial spacer locations have been altered to accommodate the change in part length rod lengths.

Reference 4 provides a description of the GNF2 design and analyses that demonstrate GNF2 meets the requirements specified GESTAR II. GNF2 compliance with GESTAR II has been audited by the NRC staff (Reference 20).

1.2 PURPOSE

The purpose of NEDC-33173P Supplement 3 is to provide the document the adequacy of GEH analytical methods to GNF2 fuel when used in expanded operating domains (e.g., extended power uprates and MELLLA+). This supplement is limited to the application of GEH analytical methods as documented in NEDC-33173P and not to the GNF2 fuel design itself, which meets the GESTAR II requirements for introduction of the fuel product (Reference 4). The applicability of NEDC-33173P to GNF2 fuel is supported by the following technical evaluations:

- The lattice physics code TGBLA06 has been modified to accommodate the changes in part length rod location, with a negligibly small impact on core eigenvalue and pin power predictions. The TGBLA06 changes were reviewed in a NRC Audit Report (Reference 20) and found to be consistent with the conclusions stated above.

- The modified TGBLA06 code has been compared to MCNP Monte Carlo results and exhibits similar pin power, criticality, and void coefficient biases as established for previous 9x9 and 10x10 lattice designs. These comparisons support the continued use of current Interim Methods biases for pin power and void coefficient for GNF2 applications.

- The accuracy of the ISCOR and TASC thermal hydraulic models, which are relevant to methods based analyses and embedded in all the GEH thermal hydraulic steady state and transient models, is supported by full-scale critical power and pressure drop tests. The correlation uncertainties are incorporated into SLMCPR evaluations in accordance with NRC-approved procedures. (Reference 5)

- Cold shutdown measurements and analysis carried out on a core containing four GNF2 lead use assemblies (LUA) have shown prediction accuracy consistent with past experience. Results obtained for local critical experiments (i.e., in-reactor demonstrations) near the LUA are consistent with past experience. The cold Critical results were also reviewed and found to be adequate in the NRC Audit Report (Reference 20).

- Full GNF2 reloads are operating in two BWR/4s and a BWR/3. Three TIP measurements have been completed over the first 4000 MWD/MT of cycle exposure for the BWR/4. The simulation of these TIP measurements have been completed and show agreement between calculation and measurement, with both radial and axial root mean square (RMS) values well below values in Reference 1.

1.3 ANALYSIS PROCESS

The approach used to confirm the applicability of GEH Methods to the GNF2 fuel design follows the same process used in the original Methods LTR (Reference 1).

The subsequent sections of this supplement to the Methods LTR provide a review of GEH methodologies, uncertainties, and biases for acceptability to GNF2 applications for expanded operating domains (e.g., CPPU, EPU, and MELLLA+). This format and outline is identical to the original Methods LTR (Reference 1). The impact of uncertainty parameters of interest is identified and their applicability to GNF2 analysis is evaluated. The adequacy of the existing margin, and, as applicable, augmented margin for each of these safety parameters is provided.

The GEH Nuclear Methods are based on three levels of detail, as indicated below:

- **The Individual Fuel Rod:** Individual fuel rod analysis concerns heat transfer, stress conditions, and fission gas buildup in an individual fuel rod. The GNF2 fuel rod design is nearly identical to GE14. The pellet diameter is slightly larger and the cladding slightly thinner. The current design basis for GNF2 fuel included in Reference 4 is based on the GSTRM methodology (Reference 1). Consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the GNF2 design basis as documented in Reference 26 pending the approval of the PRIME methodology currently under review (Reference 18).

- **The Bundle Lattice:** The most significant differences between GNF2 and GE14 occur at the lattice level. The first, which involves the lattice physics code, TGBLA, is that there are two part length rod lengths, and these part length rods (vanished rods) are in different positions in the lattice. The TGBLA06 code has been updated to accommodate the change in vanished rod locations. The output of the TGBLA code is transferred to the core-wide simulation programs in the form of lattice average nuclear parameters and pin power peaking factors. The second significant change involves the design and location of the fuel spacer grids. Fuel spacer design and location have affect on bundle pressure drop and critical power performance. Both pressure drop and critical power performance have been measured at the Stern Laboratory full-scale thermal-hydraulic test facility and correlated with NRC-approved correlations. The thermal-hydraulic output consists of pressure drop and critical power correlations based on the above-mentioned Stern Laboratory tests. The information characterizing the nuclear and thermal-hydraulic differences are incorporated in TGBLA (lattice physics).

- **Core Wide Models:** The core wide models use the lattice average nuclear parameters, critical power correlation, pressure drop correlation, and limits established by the fuel rod performance models to construct a three dimensional power distribution and establish overall core wide margin to limits. The steady state core simulator model (PANACEA), transient models (ODYN and TRACG), and stability model (ODYSY) all use lattice average outputs from TGBLA and thermal-hydraulic correlations. The overall uncertainties assigned to the steady-state core-wide models, the transient models, and stability models are entirely determined by the uncertainties in the detailed lattice and fuel rod models.

The justification for using GEH analytical methods in GNF2 applications at expanded operating domains focuses on the physics and thermal-hydraulic impact of the GNF2 design changes reflected in the lattice model TGBLA and the thermal-hydraulic correlations.

Section 2 focuses on the evaluation of the effect of the TGBLA and thermal-hydraulic uncertainties in the determination of safety parameters for CPPU and EPU applications.

Section 3 extends the Section 2 basis to the MELLLA+ operating domain. The analysis presented in Sections 2 and 3 of this document confirm that the Fuel Design Limits and Associated Methods for GNF2 analysis are identical to Table 1-1 of Reference 1 with one clarification. The current thermal-mechanical design basis for GNF2 fuel included in Reference 4 is based on the GSTRM methodology. Consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the thermal-mechanical design basis as documented in Reference 26 pending the approval of the PRIME methodology currently under review (Reference 18). The conclusions regarding the applicability of the revised Limits and Methods table appears below as Table 1-2. Appendix A provides an assessment of the limitations in the NRC SE (Reference 3) relative to GNF2 fuel.

Table 1-1 GE14 and GNF2 Parameters

Fuel Assembly	GE14	GNF2
Total number of fuel rods	92	No Change
Full length	78	No Change
Partial length	14 total, Single Length	14 total, Two Lengths
Long Part Length Rod (LPLR)	14	8
Short Part Length Rod (SPLR)	0	6
Lattice Array	Figure 2-2	Figure 2-2
Rod to rod pitch (cm)	[[	]]
Number of water rods	2	No Change
Typical Assembly weight (kgU)	[[	
BWR/2-3 Full Length Rod (mm)		
BWR/4-6 Full Length Rod (mm)		
Long Part Length Rod (LPLR) (mm)		
Short Part Length Rod (SPLR) (mm)		
Fuel Rod		
Cladding material		
Typical BWR/2-3 Assembly active fuel length (mm)		
Typical BWR/4-6 Assembly active fuel length (mm)		
LPLR Active Fuel Length (mm)		
SPLR Active Fuel Length (mm)		
Cladding tube diameter, outer (cm)		
Cladding tube wall thickness (cm)		
Pellet diameter, outer (mm)		
Fuel pellet density (PD) standard		
Fuel column Geometric Staking Factor (GSF) standard		
Helium Backfill Pressure BWR/2		
Helium Backfill Pressure BWR/3-6		
Fuel column stack density (g/cc)		
Water Rod		
Cladding material		
Cladding diameter, outer (cm)		
Cladding wall thickness (cm)		]]
Spacer		
Number of spacers	8	No Change
Axial locations	See Reference 22 Page 14	See Figure 2-5 of Reference 4
Material	Zircaloy ferrule and bands with Alloy X-750 springs	Alloy X-750

[1] [[ ]]
[2] Gd_2O_3 Concentration, percent by weight (GC)

Table 1-2 Fuel Design Limits and Associated Methods

Limit	Primary Technology	Description	Evaluation Frequency & Notes
SLMCPR	SLMCPR, PANACEA	The SLMCPR is a MCPR value at which 99.9% of the fuel rods in the core are expected to avoid BT. This value considers the core power distribution and uncertainties.	The limit is evaluated on a plant/cycle specific basis (i.e., each core design).
OLMCPR	ODYN, TRACG, PANACEA	The OLMCPR is additional margin above the SLMCPR to account for the MCPR change due to AOOs. Adherence to the limit assures that in the event of an AOO, 99.9% of the fuel rods are expected to avoid BT.	The limit is evaluated on a plant/cycle specific basis. The FSAR transients that are limiting or potentially limiting with respect to pressure and fuel thermal limits are analyzed for each reload. Transients are confirmed to be within the LHGR basis.
SDM	PANACEA	SDM is maintained regardless of the core design (the value of the limit does not vary with core characteristics like SLMCPR or OLMCPR). The shutdown margin requirement assures that the reactor can be brought and held subcritical with the control system alone. Most BWRs have a TS value of 0.38%. The "working definition" of SDM is the quantity of reactivity needed to reach criticality in a xenon free core with the strongest worth control rod fully withdrawn and all other control rods inserted.	Each core is designed to conform to this limit. SDM margin is demonstrated on a plant/cycle specific basis.
LHGR	GSTRM (current)/ PRIME (future)	LHGR Operating Limits represent an envelope of acceptable linear heat generation rates, as a function of exposure, designed to maintain fuel integrity during normal operation, including Anticipated Operational Occurrences. The LHGR limits reflect the application of SAFDLs on the following fuel performance parameters: • Fuel temperature • Cladding stress • Cladding strain • Cladding fatigue usage • Fuel rod internal pressure • Cladding creep	LHGR Operating Limits are developed generically for each fuel product line (e.g., GE14). They are determined from thermal-mechanical considerations and independent of any particular core design. The current LHGR operating limits for GNF2 fuel included in Reference 4 are based on the GSTRM methodology. Consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the LHGR operating limits for GNF2 fuel as documented in Reference 26 pending the approval of the PRIME methodology currently under review (Reference 18).
MAPLHGR	SAFER	MAPLHGR is a an average planar linear heat generation rate limit that is a product of the plant ECCS-LOCA evaluation performed to demonstrate compliance with 10CFR50.46 acceptance criteria.	ECCS-LOCA evaluations are performed as plant specific, cycle independent analyses. These analyses are typically performed for each initial introduction of new fuel product lines. The analysis output is a Licensing Basis PCT and a set of parameters that are confirmed every cycle to ensure applicability of the analysis.
Stability	ODYSY TRACG	There are several accepted stability solutions, each designed to protect the SLMCPR. The solutions include prevention and detect and suppress strategies, as well as combinations of both elements.	The stability methodologies are applied and/or confirmed for every reload (every cycle).
Exposure	GSTRM (current)/ PRIME (future)	The licensed exposure limit is a result of the LHGR evaluation methodology discussed above.	The exposure limit is developed generically for each fuel product line from thermal-mechanical considerations. It is independent of the core design. The current LHGR operating limits for GNF2 fuel included in Reference 4 are based on the GSTRM methodology. Consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the LHGR operating limits for GNF2 fuel as documented in Reference 26 pending the approval of the PRIME methodology currently under review (Reference 18).

2.0 SAFETY PARAMETERS INFLUENCED BY UNCERTAINTIES

2.1 INTRODUCTION

Section 2 of Reference 1 listed the safety parameters influenced by nuclear, thermal hydraulic, and thermal mechanical methods uncertainties and biases. These safety parameters are unchanged for GNF2 fuel design.

The analysis presented in Section 2 of Reference 1 showed that the allowances for methods uncertainties are adequate to ensure that the fuel design limits are met for fuel designs up to and including GE14 for power uprate conditions. The analysis presented in this section extends this conclusion to the GNF2 fuel design and that Table 1-2 is applicable for power uprate conditions.

2.2 CRITICAL POWER

The components of the critical power (SLMCPR and OLMCPR) are unchanged for GNF2 fuel design.

2.2.1 Safety Limit Critical Power Ratio (SLMCPR)

The methods and uncertainties used to evaluate the SLMCPR have been validated in Reference 1 by considering pin and bundle power combined with critical power, void fraction, and pressure drop correlations. These topics are covered below, with emphasis on GNF2 results.

2.2.1.1 Fuel Parameters That Affect SLMCPR

Table 2-1 and Table 2-2 of Reference 1 contain a summary of the uncertainties relevant to the evaluation the SLMCPR. These parameters are unchanged for GNF2.

2.2.1.2 Treatment of Fuel Parameter Uncertainties

The bundle power is a combination of [[ ]] Uncertainties in local pin power peaking, [[ ]] are explicitly included in the SLMCPR determination and considered separately, then cumulatively in Section 2.2.1.2 of Reference 1. The extension of these uncertainties to the GNF2 design is discussed below.

Pin Power Peaking

A key method related uncertainty is the local (pin) peaking factor uncertainty. This value is primarily associated with the lattice code TGBLA (Reference 8). The 1σ uncertainty was evaluated to be [[ ]] in NEDE-32601P-A (Reference 5), based on comparisons with MCNP Monte Carlo evaluations. The overall pin peaking uncertainty, including operational, flux gradient, and manufacturing effects was confirmed by comparison to pin gamma scan measurements performed in an 8x8 lead use assembly. The data presented in NEDE-32601P and in the RAI responses were for the most part based on GE14 and earlier fuel designs. TGBLA06-MCNP (Reference 9) comparisons carried out on other vendor's fuel designs show results consistent with those obtained with the GE designs. The results in NEDE-32601P-A show the overall TGBLA06 pin power accuracy to be similar for the Non-GE designs and the GE 9x9 and 10x10 designs.

While the fundamental methodology for TGBLA06 is not changed from that approved by the NRC, the TGBLA06 ECP required a modification to model the GNF2 part length rod configuration. [[

]] The change in the Dancoff factor and the impact on the qualification basis has been audited by the NRC staff, and documented in Reference 20.

Figure 2-1 and Figure 2-2 demonstrate the applicability of TGBLA06 to GNF2 using direct comparisons to Monte Carlo (MCNP) at 0.0 and 65 GWD/MT lattice exposure. The RMS deviation (see Reference 1 for definition) between the TGBLA06 and Monte Carlo fission density distribution is plotted vs. lattice moderator density. [[

]] For reference, the average difference range is provided for a set of GE14 10x10 lattices. The TGBLA/ MCNP RMS differences are computed for each GE14 lattice and for each moderator density. For each density, the differences are averaged and the standard deviation is evaluated. The dashed lines in the graph represent the average GE14 difference with the standard deviation added and subtracted. The small impact of analyzing the GNF2 designs is demonstrated by the fact that the GNF2 differences are within or less than the differences calculated for the GE14 lattices. As stated in Reference 1, GEH uses [[

]] The consistency of the GNF2 TGBLA06 to MCNP comparisons with previous designs justify the use of GE14 pin power uncertainties for GNF2 R-factor and LHGR evaluations.

Four Bundle Power

The second component of power uncertainties affecting the SLMCPR is the four-bundle power surrounding a TIP string. GNF has continued to provide the NRC with BWR fleet information on the consistency of integral TIP comparisons on periodic basis, e.g., in fuel technology updates. These comparisons provide the basis for the [[ ]] in Table 2-2 of Reference 1. In 2005, GNF provided a data for uprated plants loaded primarily with 10x10 fuel in methods related RAI responses on the MELLLA+ docket (Reference 1). The results of plant tracking studies performed with the current methods are summarized in Table 2-5 of Reference 1, which yield an overall [[ ]]. The TIP RMS metric is defined in Reference 1. Examination of this data confirms the applicability and conservatism of the original [[ ]] uncertainty documented in GEH's approved topical reports (Reference 5, NEDC-32601P-A and Reference 6, NEDC-32694P-A) describing the SLMCPR methodology, for uprated power densities as high as 62 KW/liter.

GNF2 lead use assemblies have been operating in three BWR/4s, and in a European BWR design for up to three cycles and three plants in reload quantities. The TIP data for the four lead use assemblies in two BWR/4 plants have been analyzed in an NRC audit report (Reference 20) and revealed no unusual behavior. The first two full GNF2 reloads are currently operating in BWR/4 and BWR/3 non-EPU plants. Currently, a total of three TIP measurements at the BWR/4 are available for analysis through the first [[ ]] of the cycle. The results are summarized in Table 2-2. This table shows the TIP comparisons, indicating agreement with an average radial RMS difference of [[ ]], which is less than the [[ ]] average in Table 2-5 of Reference 1.

This particular BWR/4 has 31 TIP strings and 560 bundles. The batch fraction for this GNF2 reload is 164 bundles, or 29%. Twelve of the TIP strings have no GNF2 bundles immediately adjacent to them, sixteen TIP strings have two GNF2 adjacent bundles, and three strings have three GNF2 adjacent bundles. It is instructive to look at the radial bias in each of these three groups to see if the GNF2 bundles are influencing the radial bias. The results summarized in Table 2-3 show that there is a small range in mean bias [[ ]] between the three groups of TIP strings, indicating that the simulation of GNF2 bundles is quite consistent with simulation of GE14 bundles, which constitute the remaining bundles in the core. Figure 2-3 provides further evidence of the consistency of the GNF2 simulation, showing the BWR/4 critical eigenvalue and the projected eigenvalue, based on previous GE14 experience. The tracking eigenvalue is within 0.1% of the projected value. The design allowance for the difference between actual and projected critical eigenvalue is [[ ]] which indicates consistent performance for this first reload introduction of GNF2 into an operating reactor.

Bundle Power

[[ ]] is a component of the total bundle power uncertainty. The total bundle power uncertainty for application within GEH's approved SLMCPR determination process consists of the component uncertainties in Table 4.2, page 4-2 in NEDC-32694P-A. [[

]] The bundle power allocation factor for a new fuel design is most sensitive to changes in the reactivity of each lattice as a function of moderator density and fuel exposure. Reference 1 contains a significant amount of data comparing TGBLA06 and MCNP reactivity response to a variety of moderator density and exposures. These same comparisons have been completed for GNF2 lattices. The comparisons are displayed in Figure 2-4 at BOL exposure and in Figure 2-5 at 60 GWD/MTU exposure. The reactivity difference between TGBLA06 and MCNP are plotted versus moderator density. The TGBLA/ MCNP reactivity differences are computed for each GE14 lattice and for each moderator density. For each density, the differences are averaged and the standard deviation is evaluated. The dashed lines in the graph represent the average GE14 difference with the 1σ standard deviation added and subtracted from the mean. The results for the five GNF2 lattice types are plotted individually. The results show that the GNF2 biases are consistent with the other 10x10 results including the trends with void fraction. This consistent behavior justifies the use of the current methods procedures and uncertainties for the GNF2 fuel design.

Thermal-Hydraulic Methods

The introduction of various PLR rod heights, such as in GNF2, or other axially varying features, such as axially varying thick/thin channels, can be readily handled by the steady-state and transient analysis programs because model parameters can be varied axially to account for changes in the number of rods, water rod diameters, etc. in the lattice at different axial locations. The single bundle thermal-hydraulic code, ISCOR09, employs both the void correlation and pressure drop correlation combined with the mass and energy solution to the heat transfer equations. The ISCOR09 methods are embedded in the PANACEA steady state three-dimensional simulator and the stability analysis tools. [[

]] This difference is also accommodated within the core methods methodology.

Void Correlation

The GEH void correlation has been shown to be applicable for existing GNF BWR fuel designs, including 10x10 lattices with part length rods (Reference 1). [[

]]

Qualification of GNF2 has been evaluated with full-scale experimental pressure drop data (Reference 4) Correct prediction of the pressure drop requires an accurate prediction of the void fraction throughout the length of the bundle. In addition, the void fraction correlation is indirectly qualified via comparison with sub-channel analysis methods as show in Figure 2-6. Therefore, the GEH Findlay-Dix void fraction correlation (Reference 7), which forms the basis for currently approved methodologies, is applicable to GNF2 fuel designs.

Pressure Drop

The GNF2 fuel assembly design incorporates the use of nickel-based, Ni-Cr-Ti alloy grid type spacers with flow wings to improve critical power performance. The pressure drop characteristics of the GNF2 spacers are based on the pressure drop data from full-scale testing of the GNF2 fuel assembly as documented in Reference 4. Production spacers were used in the full-scale test assembly with no modifications. The measured pressure drops include static head, wall friction, acceleration pressure drop, and form losses. The loss coefficients were evaluated in a manner consistent with the steady-state thermal-hydraulic analysis methodology documented in Section 4.2 of GESTAR II. The test assembly and the measurement scheme for obtaining differential pressures are shown in Figure 2-7. Test data were obtained at [[

]]

Measured pressure drops across the bundle height from [[ ]] are compared to the predictions in Figure 2-8. The comparison of the predicted versus measured pressure drop for [[ ]] tests over a range of thermal-hydraulic conditions resulted in a mean error for the [[

]] It is instructive to note from Figure 2-8 that the same small pressure drop error is maintained over the entire range of bundle powers. The zero bundle power results, shown as the green diamonds in the Figure, represent the single-phase portion of the pressure drop, are consistent with all the data. The pressure drop correlation is able to accurately model the split between single phase and two phase pressure drop, which is an important characteristic in the thermal hydraulic stability. The ISCOR09 model with the pressure drop correlation also predicts the axial pressure profile in the bundle. Figure 2-9 compares the measured and calculated accumulated pressure drop for a high power and moderate flow condition. The intermediate pressures are taken from the pressure taps shown in Figure 2-7. The pressure profile shows that the effects of the part length rods and advanced spacers are accurately simulated by the ISCOR09 model, the steady state, stability, and transient analysis tools.

The GNF2 fuel assembly hydraulic characteristics have been developed and confirmed by the test comparisons discussed above. These GNF2 hydraulic characteristics are used in all analysis models and methods where the fuel assembly hydraulics are needed. For cores of mixed assembly types, the hydraulics are uniquely represented for each assembly type. Therefore, the flow-pressure drop characteristics for each fuel assembly type (including GNF2) present in a plant are included in all plant cycle-specific analyses for the calculation of the Operating Limit Minimum Critical Power Ratio.

Critical Power Correlation

The GNF2 fuel assembly has a different part length rod configuration and spacer design relative to previous fuel designs. The new correlation, GEXL17, has been established based on significant new data for the GNF2 fuel design.

The GEXL17 (Reference 21) database was obtained from Stern Laboratory tests of full-scale GNF2 bundle simulations. A statistical analysis has been performed for the GNF2 database used to develop the GEXL17 correlation, consisting of [[ ]] data points for [[ ]] different local peaking patterns. This correlation statistics were based on [[ ]] data points. The GEXL17 correlation is valid for GNF2 fuel over the following range of state conditions:

- Pressure: [[ ]]
- Mass Flux*: [[ ]]
- Inlet Subcooling: [[ ]]
- R-factor*: [[ ]]

The GEXL17 Application Range is documented in Figure 2-10.

In addition, there is an additive constant applied to each fuel rod location [[ ]] For GNF2, the additive constants used in the design process are provided in Reference 4. The terms that comprise the form of the correlation have been previously approved by the NRC and have been in use for the past seven GE fuel product designs.

Based on the [[ ]] data points used to develop and verify the GEXL17 correlation statistics, the mean ECPR, μ, was determined to be [[ ]], with a standard deviation, σ, of [[ ]]. In addition to the overall statistic mentioned above the GEXL17 correlation is accurate over the entire flow range. The ECPR statistics are shown as a function of bundle flow in Figure 2-11. The average ECPR is within [[ ]] over the entire flow range expected in EPU and MELLLA+ operation, ensuring accurate CPR modeling of both steady state and transient operation.

2.2.1.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of SLMCPR as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. This conclusion is based on the following:

- The TGBLA06 nuclear evaluations have been shown to yield similar pin power and reactivity behavior relative to MCNP benchmark calculations as the previously documented GE14 analyses (Reference 1).

- Initial TIP data for the first GNF2 application shows agreement with current GEH methods. This agreement with operating TIP data and consistent eigenvalue behavior relative to GE14 experience for the BWR/4 indicates that no change in methods or procedures is required for GNF2 analysis.

- Full-scale thermal-hydraulic pressure drop and critical power tests have been performed and correlated with NRC-approved correlations. The GNF2 GEXL17 critical power correlation uncertainty is incorporated into the determination of the SLMCPR. The range of the pressure drop and critical power test data is sufficient to cover thermal-hydraulic conditions present during EPU and MELLLA+ operations. The correlation forms and implementation methods remain unchanged for GNF2.

2.2.2 Operating Limit Critical Power Ratio (OLMCPR)

The analysis of anticipated operational occurrences (AOOs) examines the change in critical power ratio relative to the initial conditions and determines the most limiting event. The definition of the OLMCPR is unchanged for GNF2.

2.2.2.1 Fuel Parameters That Affect OLMCPR

Reference 1 contains a detailed discussion of the fuel parameters that affect OLMCPR. These parameters are unchanged for GNF2.

2.2.2.2 Treatment of Fuel Parameter Uncertainties

A new fuel design can potentially affect transient response. The three most important parameters are:

- **Core Axial Power Shape:** As stated in Reference 1, the core axial power shape can influence the transient response. Uncertainties in the axial power shape are not directly included in the transient response uncertainty. Rather, the input conditions for the transient are developed in a way that ensures that the axial shape is conservative and is not influenced by the GNF2 design.

- **Void and Moderator Density Reactivity Response:** Both the ODYN and TRACG transient methodologies (References 10, 11, and 12) have established application ranges for void coefficient uncertainty. The basis for these methodologies rests upon a comparison of calculations for a wide variety of plant transients in which the nominal void coefficient is used. The acceptable performance of these codes relative to the data justifies that no large errors in void coefficient exist. As described in Section 2.2.1.2 above, TGBLA06 and MCNP have been utilized to generate reactivity differences for representative GE14 and GNF2 10x10 lattices for the full range of instantaneous void conditions. Differences have also been evaluated for cold conditions. Figure 2-4 and Figure 2-5 show the TGBLA06/MCNP bias as a function of moderator density. The GNF2 results follow the same trend with moderator density as the GE14 results, and therefore yield similar void coefficient biases. The consistent moderator density behavior between hot zero void and cold conditions ensure consistent behavior for cold water events as well.

- **Thermal-Hydraulic Behavior:** Transient conditions require both the critical power and pressure drop correlations be accurate for the full range of flow conditions. This accuracy is demonstrated in Figure 2-8 for the GNF2 pressure drop correlation and in Figure 2-11 for the GEXL17 critical power correlation.

The Reference 1 assumption of [[ ]] void coefficient bias and a 2σ void coefficient uncertainty of [[ ]] is justified for GNF2, given the similarity of GNF2 to GE14 and the consistency of the TGBLA06/MCNP comparisons shown above.

Because inputs to the OLMCPR analysis are conservative, and the pressurization transients that typically establish the limiting ΔCPRs are conservatively analyzed by TRACG or ODYN, the conservatisms in the process of determining OLMCPRs is appropriate and sufficient for application to GNF2.

2.2.2.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of OLMCPR as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. For applications that utilize TGBLA06 based modeling (PANAC11, ODYN, TRACG, and ODYSY), the TGBLA06/MCNP GNF2 comparisons showed a behavior consistent with GE14 behavior. The GNF2 thermal-hydraulic correlations are robust and accurately describe pressure drop and critical power margins over the entire flow range.

Table 2-1 GNF2 Axial Regions

Name	Description	Axial Zone Length
[[		
		]]

Table 2-2 TIP Comparisons for BWR/4 With GNF2 Reload

[[					
					]]

Table 2-3 Effect of GNF2 Bundles on TIP Radial Bias

[[									
									]]

Figure 2-1 TGBLA06 Fission Density Benchmark for GNF2, at BOC

[[

]]

Figure 2-2 TGBLA06 Fission Density Benchmark for GNF2, at 65 GWD/MT

[[

]]

Figure 2-3 Core Eigenvalue tracking for BWR/4 Containing GNF2 Reload

[[

]]

Figure 2-4 TGBLA06 Reactivity Benchmark for GNF2, at BOC (GE14 1σ uncertainty band, dashed line)

[[

]]

Figure 2-5 TGBLA06 Reactivity Benchmark for GNF2, at high exposure (GE14 1σ uncertainty band, dashed line)

[[

]]

Figure 2-6 Axial Void Calculation on GNF2 at High Power Conditions from the Findlay-Dix Correlation and Sub-channel Based Calculation

[[

]]

Figure 2-7 Spacer Test Configuration

[[

]]

Figure 2-8 GNF2 Calculated vs. Measured Delta –P

[[

]]

Figure 2-9 GNF2 ΔP (Calculated or Measured) Versus Elevation

[[

]]

Figure 2-10 Mass Flux vs. R-Factor Plane

[[

]]

Figure 2-11 GEXL17 ECPR as a Function of Bundle Flow

[[

]]

2.3 SHUTDOWN MARGIN (SDM)

The required Technical Specifications for Shutdown Margin are unchanged for GNF2.

2.3.1 Fuel Parameters That Affect SDM

The fuel parameters that affect SDM are unchanged for GNF2.

2.3.2 Treatment of Fuel Parameter Uncertainties

A shutdown margin demonstration experiment is performed at the beginning of each operating cycle. This demonstration is performed in the cold, or most reactive criticality condition. The demonstration configuration attempts to simulate the most reactive rod out condition. In order to obtain a critical condition, other rods are also withdrawn. The 3D simulator (Reference 8) is used to calculate the demonstration condition. Reference 1 presented the results of 39 critical experiments performed over five cores, for which multiple cold critical experiments were performed on the same core. The standard deviation of the critical eigenvalues for the cores in Reference 1 relative to the average obtained for the same core is [[ ]]. This standard deviation can be compared to the Technical Specification allowance of 0.38% Δk/k., indicating that for application to high power density cores, the data supports the continued use of the current Technical Specification limit.

The ability to predict shutdown margin for GNF2 applications has been evaluated through a series of local critical measurements in a 240 bundle BWR/4 operating with annual cycles. Four GNF2 lead use assemblies were inserted at the beginning of cycle 33. In all, a series of 22 local cold critical measurements were performed in cycles 32 through 35. Results from Cycles 33 and 34 have been previously audited by the NRC staff as part of the generic Amendment 22 Audit for GNF2 (Reference 20). The results are summarized in Table 2-4.

Local critical results where the fully withdrawn rod is adjacent to a GNF2 bundle are shown in the shaded rows. An important cold shutdown methods metric is the difference between the projected keff and the actual keff evaluated from the measurement. For these data, the average difference between the projected and actual keff for the non-GNF2 criticals is [[ ]] with a standard deviation of [[ ]]. The GNF2 criticals yield an average difference of [[ ]] with a standard deviation of [[ ]]. These results are well within the range of projected–measured results detailed in Reference 1. The standard deviation of the 22 differences is [[ ]] essentially equal to the value of [[ ]] obtained in Table 2-10 of Reference 1. The distribution of differences is illustrated graphically in Figure 2-12. The red part of the bar represents the GNF2 results and the blue part represents the remaining criticals. These results show the consistency between the two sets of criticals and that there is no significant cold critical bias change for GNF2 bundles.

2.3.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of Shutdown Margin as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. This evaluation is based on the consistent shutdown predictions for the 240-bundle BWR/4, in which local critical experiments have been carried out near GNF2 lead use assemblies. Consistent TGBLA06/MCNP reactivity data have also been obtained for cold conditions.

Table 2-4 Summary of Local Cold Critical Measurement for Plant A

Cycle	Case	Cycle Exposure	Temp (ºF)	Period (sec)	Projected keff	Actual keff
35	COLD_L01	0	86	24	[[	
35	COLD_L02	0	86	55		
35	COLD_L03	0	86	70		
35	COLD_L04	0	86	40		
34	COLD_L01	0	92	60		
34	COLD_L02	0	92	50		
34	COLD_L03	0	92	60		
34	COLD_L04*	7567	97	80		
34	COLD_L05	7567	97	55		
34	COLD_L06*	7567	95	55		
33	COLD_L01	0	86	70		
33	COLD_L02	0	86	40		
33	COLD_L03	0	86	70		
33	COLD_L04	7738	108	40		
33	COLD_L05*	7738	108	28		
33	COLD_L06*	7738	108	30		
32	COLD_L01	0	91	80		
32	COLD_L02	0	91	20		
32	COLD_L03	0	91	30		
32	COLD_L04	7718	99	40		
32	COLD_L05	7718	99	60		]]

* Local critical results where the fully withdrawn rod is adjacent to a GNF2 bundle are shown in the shaded rows.

Figure 2-12 Frequency Distribution of Cold Critical Eigenvalue Differences

[[

]]

2.4 Fuel Rod Thermal-Mechanical Performance

For each GNF fuel design, thermal-mechanical based linear heat generation rate limits (LHGR Operating Limits) are specified for each fuel rod type (for both UO_2 and gadolinia-bearing rods) such that, if each rod type is operated within its LHGR limit, the thermal-mechanical design and licensing criteria, including those which address response to anticipated operational occurrences (AOOs), are explicitly satisfied and fuel rod integrity is maintained. The licensing criteria for determining thermal-mechanical design have not changed for GNF2.

2.4.1 Fuel Parameters That Affect Thermal-Mechanical Limits

The fuel parameters that affect thermal-mechanics limits have not changed for GNF2.

2.4.2 Treatment of Fuel Parameter Uncertainties

The impact of the GNF2 design on the uncertainty in local peaking and three-dimensional power distribution is discussed in Section 2.2.1.2 of this document, where the revised uncertainties as shown in Table 2-11 of Reference 1 are shown to be appropriate for GNF2 analysis. The GNF2 fuel pellet and rod diameter design is almost identical to the GE14 fuel rod design. The differences are summarized in Table 1-1. GNF2 fuel rods, however, operate at a higher peak power, while still maintaining the same peak discharge exposure. The current design basis for GNF2 fuel included in Reference 4 is based on the GSTRM methodology. Consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the LHGR operating limits for GNF2 fuel as documented in Reference 26 pending the approval of the PRIME methodology currently under review (Reference 18).

2.4.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of thermal-mechanical limits as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. The standard GE methodology for determining LHGR limits includes conservative consideration for, and provides reasonable assurance of adequate margin to address, the power uncertainties in question and is not affected by the GNF2 design. The current approved GSTRM provide an appropriate basis for the use of GNF2 in the EPU and MELLLA+ extended operating domains; although PRIME will be used as the basis for GNF2 thermal-mechanical design basis consistent with Limitation 9.12 (See Appendix A). The GSTRM basis for GNF2 (Reference 26) does not require the incremental penalty applied to the GE14 design by Appendix F of Reference 3 (See Appendix A).

2.5 LOCA RELATED NODAL POWER LIMITS

The purpose of the maximum average planar linear heat generation rate (MAPLHGR) limits is to assure adequate protection of the fuel during a postulated loss-of-coolant accident (LOCA) with the defined operation of the emergency core cooling system (ECCS). This is unchanged for GNF2.

2.5.1 Fuel Parameters That Affect LOCA Related Nodal Power Limits

The fuel parameters that affect LOCA related nodal power limits are unchanged for GNF2.

2.5.2 Treatment of Fuel Parameter Uncertainties

The ECCS-LOCA analysis follows the NRC-approved SAFER/GESTR application methodology documented in Volume III of NEDE-23785-1-PA (Reference 13). The analytical models used to perform ECCS-LOCA analyses are documented in Volume II of NEDE-23785-1-PA (Reference 14) together with NEDE-30996P-A (Reference 15) and NEDC-32950P (Reference 16). Reference 1 contains a discussion of the relationship of peak power uncertainties and their application to fuel parameter analysis. The analysis presented in Section 2.2.1.2, showing the uncertainty in pin and bundle power for GNF2 is the same as for GE14 and previous designs.

2.5.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of thermal-mechanical limits as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. The conservatism of the present ECCS-LOCA methodology used to determine MAPLGHR limits adequately considers the effects of the uncertainties in local and bundle power and provides adequate and reasonable assurance that those limits provide adequate margin to protect the fuel. This assurance is extended to GNF2. The ECCS-LOCA methodology is fully capable of simulating the necessary features of the GNF2 fuel design and design basis uncertainties for the design GE14 fuel design are adequate and applicable to GNF2 analyses.

2.6 STABILITY

Thermal-hydraulic stability analyses are performed to assure that the SLMCPR is protected in the event of a thermal-hydraulic instability event. Specific analyses are associated with each of the long-term stability solutions. These long-term solutions include Option I-D, Option II, Option III, and Enhanced Option I-A. The stability analyses and the applicability of these stability solution Options remain unchanged for GNF2.

2.6.1 Fuel Parameters That Affect Stability

The fuel parameters identified previously in Reference 1 are unchanged for GNF2.

2.6.2 Treatment of Fuel Parameter Uncertainties

Reference 1 provides the treatment of the fuel parameter uncertainties for each of the long-term stability solutions is unchanged for GNF2. Sections 2.6.2.1 through 2.6.2.4 of Reference 1 discuss the stability impact of nuclear and thermal hydraulic uncertainties for each of the four stability long-term solutions listed above, namely Option 1-D, Option II, Option III, and Enhanced Option I-A. In general, the stability models used to evaluate the options and issues described above imbed the basic bundle nuclear and thermal hydraulic models from the TGBLA, ISCOR and PANACEA programs. Other transient models are consistent with these basic models. Stability performance depends on the following parameters:

- **Moderator void coefficient**: The TGBLA06/MCNP comparisons for the GNF2 design show the same bias with moderator density as previous 10x10 designs. There is no change in moderator void coefficient bias and uncertainty with GNF2.

- **Local pin power peaking**: The TGBLA06/MCNP comparisons for the GNF2 design also show the same pin power accuracy for GNF2 as previous 10x10 designs, and the same stability uncertainty impact as previous designs.

- [[ ]]: The GNF2 reactivity biases relative to Monte Carlo results are consistent with previous 10x10 designs, showing no change needed in stability impact for [[ ]].

- **Bundle pressure drop**: The bundle pressure drop model is based on GNF2 full-scale pressure drop measurements. In addition to the total bundle pressure drop, the axial pressure profile is accurately modeled (see Figure 2-9) by the ISCOR model, which is embedded in the stability evaluations.

2.6.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of Stability as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. All models related to stability have the same uncertainties for the GNF2 design as the GE14 design, and are acceptable for GNF2-related stability analysis.

2.7 LICENSED EXPOSURE

The GNF2 fuel design is licensed to a peak pellet exposure limit of [[ ]] (Reference 4), based on the existing GSTRM methodology basis. GEH anticipates updating the peak pellet exposure limit for GNF2 fuel when the new PRIME methodology is applied (Reference 18) (See Appendix A).

This licensed peak pellet exposure limit is specified and applied in the process computer to assure that fuel is not operated beyond its analyzed basis. In this application, the best estimate value of the local exposure condition is monitored against the specified exposure limit.

2.7.1 Fuel Parameters That Affect Pellet Exposure

The fuel parameters that affect pellet exposure are unchanged for GNF2.

2.7.2 Treatment of Fuel Parameter Uncertainties

The overall pin power uncertainties are unchanged for GNF2 (Section 2.2.1.2).

2.7.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of licensed exposure as specified in Table 1-2 can be used for the GNF2 design under expanded operating domains. As noted previously, the current approved GSTRM (Reference 26) provide an appropriate basis for the use of GNF2 in the EPU and MELLLA+ extended operating domains. However, consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the GNF2 design basis once PRIME is approved.

3.0 EXTENSION OF SAFETY PARAMETER BASES TO THE MELLLA+ OPERATING DOMAIN

3.1 INTRODUCTION

MELLLA+ operation allows the reactor to be at full power down to 80% flow (Reference 2). Like extended power uprate (EPU), these conditions increase the amount of steam voids in the core. The total steam void level in a given bundle is a direct function of the power to flow ratio. Raising the average bundle power (EPU) or lowering the flow (MELLLA+) have the same effect, and for the most part raise similar technical issues. The use of GNF2 fuel does not change the application of the GEH methods for MELLLA+.

3.2 CRITICAL POWER

3.2.1 Safety Limit Critical Power Ratio (SLMCPR)

Section 3.2.1 of Reference 1 describes the process for determining the SLMCPR for MELLLA+ operating conditions. This analysis has shown that use of uncertainties at rated conditions is appropriate for MELLLA+ conditions. Design limits and methods associated with evaluation of SLMCPR as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions. The justification for the use of GEH Methods for GNF2 SLMCPR evaluations is given in Section 2.2.1.

3.2.2 Operating Limit Critical Power Ratio (OLMCPR)

MELLLA+ evaluation procedures require consideration of OLMCPR values for each limiting corner of the power flow map. If changes are required to account for OLMCPR at different flow points, this change is reflected in the process computer algorithm for MFLCPR (ratio of bundle critical power to OLMCPR) for each bundle. The same conservatisms apply for the nuclear inputs to the transient evaluations. The sensitivities remain the same as those evaluated at the full power conditions and are unaffected by GNF2 fuel. Design limits and methods associated with evaluation of OLMCPR as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions.

3.3 SHUTDOWN MARGIN

The data in Section 2.3 of Reference 1 supports a 2σ demonstration margin criteria of 0.38% Δk/k. A series of cold critical experiments performed on a BWR/4 containing GNF2 lead use assemblies appears in Section 2.3.2 of this report shows that this shutdown margin accuracy is maintained with local critical measurements near GNF2 lead use assemblies. Design limits and methods associated with evaluation of shutdown margin as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions.

3.4 FUEL ROD THERMAL MECHANICAL PERFORMANCE

The fuel rod thermal-mechanical analyses explicitly address the variation in the axial power distribution that may occur as a result of spectral shift operation, and therefore the specified LHGR operating limits and exposure limit are directly applicable to MELLLA+ operation and unaffected by GNF2 fuel design. Design limits and methods associated with evaluation of Fuel Rod Thermal Mechanical Performance as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions.

3.5 LOCA RELATED NODAL POWER LIMITS

There are no differences in the ECCS-LOCA methodology between EPU and MELLLA+ except that for MELLLA+ the ECCS-LOCA analyses are performed for at least two additional state points. These are unchanged for GNF2. Design limits and methods associated with evaluation of LOCA related Nodal Power Limits as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions.

3.6 STABILITY

The GE BWR Detect and Suppress Solution – Confirmation Density (DSS-CD) (NEDC-33075P) is a licensed stability solution for operation in the MELLLA+ domain (Reference 17). The GNF2 pressure drop and critical power correlations described in Section 2.2.1.2 are accurate to low flow conditions and accurately represent the pressure profile in the fuel bundle. Design limits and methods associated with evaluation of Stability as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions.

3.7 LICENSED EXPOSURE

The current approved GSTRM (Reference 26) and provide an appropriate basis for the use of GNF2 in the MELLLA+ operating domain. However, consistent with Limitation 9.12 (See Appendix A), GEH anticipates updating the GNF2 design basis once PRIME is approved. Design limits and methods associated with evaluation of Licensed Exposure as specified in Table 1-2 can be used for the GNF2 design under MELLLA+ conditions.

4.0 LICENSING APPLICATION

4.1 OVERVIEW

The purpose of this supplement is to extend the application of Reference 1 to GNF2 fuel.

4.2 APPLICABILITY

The Applicability of GE Methods to Expanded Operating Domains LTR basis is applicable to current GEH BWR product lines licensed with GEH nuclear and safety analysis methods. The Methods LTR is applicable to plants that include current GNF fuels including GNF2. The application of these codes complies with the limitations, restrictions and conditions specified in the approving NRC SER for each code.

The parameters establishing the Applicability of GEH Methods to Expanded Operating Domains applicability envelope are:

Parameter	Generic Value
BWR Product Line	BWR/2-6*
Fuel Product Line	GE fuel designs using square arrays of fuel rods, including 7x7, 8x8, 9x9, and 10x10 designs and GNF2
Licensing Methodology	GEH Nuclear and Safety Analysis Methods
Operating Domain	CPPU, EPU, with MELLLA+ including currently licensed operating domains (e.g., ELLLA, MELLLA) and operational flexibility features
Maximum Rated Power Level	120% OLTP
Stability Solution	GE Stability Solutions

***MELLLA+ is not applicable to BWR/2 plants consistent with NEDC-33006P-A (Reference 2)**

4.3 PLANT SPECIFIC APPLICATION PROCESS

Each plant seeking to apply the Methods LTR must provide information supporting the application that demonstrates that the plant parameters are within the applicability definition in Section 4.2.

5.0 SUMMARY AND CONCLUSION

The evaluations presented in Sections 2 and 3 demonstrate the adequacy of the GEH methods for GNF2 when used in the expanded operating domains. Further, the assessment in Appendix A documents the applicability of the limitations in the NRC SE for the Methods LTR (Reference 3) for GNF2 fuel.

Safety Limit Critical Power Ratio (SLMCPR)

SLMCPR evaluation procedure and methods are not changed due to introduction of GNF2 fuel.

Operating Limit Critical Power Ratio (OLMCPR)

OLMCPR evaluation procedure and methods are not changed due to introduction of GNF2 fuel.

Shutdown Margin (SDM)

The Technical Specification (TS) limit for the SDM of 0.38 % $\Delta k/k$ is not increased for CPPU or EPU and MELLLA+ applications where GNF2 is utilized. The SDM evaluation procedure and methods are unchanged due to the introduction of GNF2 fuel.

Fuel Rod Thermal-Mechanical Performance

The licensing criteria for fuel rod thermal-mechanical performance are unchanged. The current approved GSTRM (Reference 26) fuel methodology provides an appropriate basis for the use of GNF2. However, consistent with Limitation 9.12 (See Appendix A), the GNF2 design basis will be updated once PRIME is approved.

LOCA Related Nodal Power Limits

The LOCA evaluation procedure and methods are unchanged due to introduction of GNF2 fuel.

Stability

The stability evaluation procedure and methods are unchanged due to introduction of GNF2 fuel.

Licensed Exposure

The licensing criteria for fuel rod maximum licensed exposure are unchanged. The current approved GSTRM (Reference 26) fuel methodology provides an appropriate basis for the use of GNF2. However, consistent with Limitation 9.12 (See Appendix A), the GNF2 design basis will be updated once PRIME is approved.

6.0 REFERENCES

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21. GE Nuclear Energy, "GEXL17 Correlation for GNF2 Fuel," NEDC-33292P, Revision 3, June 2009.

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Appendix – A

Limitations from Safety Evaluation for LTR NEDC-33173P

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
9.1	TGBLA/PANAC Version	The neutronic methods used to simulate the reactor core response and that feed into the downstream safety analyses supporting operation at EPU/MELLLA+ will apply TGBLA06/PANAC11 or later NRC-approved version of neutronic method.	Unchanged	EPU/MELLLA+ applications utilizing GNF2 fuel will use TGBLA06/PANAC11 or later NRC-approved version of neutronic methods.
9.2	3D Monicore	For EPU/MELLLA+ applications, relying on TGBLA04/PANAC10 methods, the bundle RMS difference uncertainty will be established from plant-specific core-tracking data, based on TGBLA04/PANAC10. The use of plant-specific trend line based on the neutronic method employed will capture the actual bundle power uncertainty of the core monitoring system.	Not Applicable	EPU/MELLLA+ applications utilizing GNF2 fuel will not use TGBLA04/PANAC10 as the neutronic methods. See Limitation 9.1.
9.3	Power to Flow Ratio	Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any state point in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.	Unchanged	This limitation is not dependent on fuel type. Consistent with Reference 23.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
9.4	SLMCPR1	For EPU operation, a 0.02 value shall be added to the cycle-specific SLMCPR value. This adder is applicable to SLO, which is derived from the dual loop SLMCPR value.	Unchanged	In Reference 24, GEH committed to justify the use of GE's analytical methods for expanded operating domains, up to and including MELLLA+, without the use of the temporary adders based on specific gamma scan data. This data is applicable to the GNF2 fuel type. Until such time the data is reviewed and approved by the NRC, the SLMCPR adder remains applicable to GNF2 fuel.
9.5	SLMCPR2	For operation at MELLLA+, including operation at the EPU power levels at the achievable core flow state point, a 0.03 value shall be added to the cycle-specific SLMCPR value.	Unchanged	In Reference 24, GEH committed to justify the use of GE's analytical

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
				methods for expanded operating domains, up to and including MELLLA+, without the use of the temporary adders based on specific gamma scan data. This data is applicable to GNF2 fuel type. Until such time the data is reviewed and approved by the NRC, the SLMCPR adder remains applicable to GNF2 fuel.
9.6	R-Factor	The plant specific R-factor calculation at a bundle level will be consistent with lattice axial void conditions expected for the hot channel operating state. The plant-specific EPU/MELLLA+ application will confirm that the R-factor calculation is consistent with the hot channel axial void conditions.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.7	ECCS-LOCA 1	For applications requesting implementation of EPU or expanded operating domains, including	Unchanged	This limitation will be implemented for

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		MELLLA+, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.		all fuel types, including GNF2.
9.8	ECCS-LOCA 2	The ECCS-LOCA will be performed for all state points in the upper boundary of the expanded operating domain, including the minimum core flow state points, the transition state point as defined in Reference 2 and the 55 percent core flow state point. The plant-specific application will report the limiting ECCS-LOCA results as well as the rated power and flow results. The SRLR will include both the limiting state point ECCS-LOCA results and the rated conditions ECCS-LOCA results.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.9	Transient LHGR 1	Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the T-M acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel	Unchanged	This limitation will be implemented for all fuel types, including GNF2.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		melting and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for the both the UO_2 and the limiting GdO_2 rods.		
9.10	Transient LHGR 2	Each EPU and MELLLA+ fuel reload will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the SRLR or COLR, or it will be reported directly to the NRC as an attachment to the SRLR or COLR.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.11	Transient LHGR 3	To account for the impact of the void history bias, plant-specific EPU and MELLLA+ applications using either TRACG or ODYN will demonstrate an equivalent to 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain acceptance criteria due to pellet-cladding mechanical interaction for all of limiting AOO transient events, including equipment out-of-service. Limiting transients in this case, refers to transients where the void reactivity coefficient plays a significant role (such as pressurization events). If the void history bias is incorporated into the transient model within the code, then the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding	Unchanged	This limitation will be implemented for all fuel types, including GNF2.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		circumferential plastic strain is no longer required.		
9.12	LHGR and Exposure Qualification	In MFN 06-481, GE committed to submit plenum fission gas and fuel exposure gamma scans as part of the revision to the T-M licensing process. The conclusions of the plenum fission gas and fuel exposure gamma scans of GE 10x10 fuel designs as operated will be submitted for NRC staff review and approval. This revision will be accomplished through Amendment to GESTAR II or in a T-M licensing LTR. PRIME (a newly developed T-M code) has been submitted to the NRC staff for review (Reference 18). Once the PRIME LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDC-33173P must utilize the PRIME T-M methods.	Unchanged	This limitation will be implemented for GNF2. In Reference 25, GEH provided supplement to NEDC-33173P describing the implementation of the PRIME code models into the downstream safety analysis codes, and the schedule to complete the implementation. That supplement is applicable to GNF2 fuel.
9.13	Application of 10 Weight Percent Gd	Before applying 10 weight percent Gd to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for TOP and MOP conditions that bounds the response of plants	Unchanged	This limitation will be implemented for all fuel types, including GNF2. GEH has no current plans to apply 10 weight percent Gd to GNF2 fuel.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		operating at EPU and expanded operating domains at the most limiting state points, considering the operating flexibilities (e.g., equipment out-of-service). Before the use of 10 weight percent Gd for modern fuel designs, NRC must review and approve TGBLA06 qualification submittal. Where a fuel design refers to a design with Gd-bearing rods adjacent to vanished or water rods, the submittal should include specific information regarding acceptance criteria for the qualification and address any downstream impacts in terms of the safety analysis. The 10 weight percent Gd qualifications submittal can supplement this report.		
9.14	Part 21 Evaluation of GESTR-M Fuel Temperature Calculation	Any conclusions drawn from the NRC staff evaluation of the GE's Part 21 report will be applicable to the GESTR-M T-M assessment of this SE for future license application. GE submitted the T-M Part 21 evaluation, which is currently under NRC staff review. Upon completion of its review, NRC staff will inform GE of its conclusions.	Not applicable.	The current approved GSTRM and the future PRIME fuel methodologies both provide an appropriate basis for the use of GNF2 in the EPU and MELLLA+ extended operating domains. The GSTRM basis for GNF2 (Reference 26) does not require the incremental penalty applied to the

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
				GE14 design by Appendix F of the Methods LTR SE (Reference 3).
9.15	Void Reactivity 1	The void reactivity coefficient bias and uncertainties in TRACG for EPU and MELLLA+ must be representative of the lattice designs of the fuel loaded in the core	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.16	Void Reactivity 2	A supplement to TRACG /PANAC11 for AOO is under NRC staff review (Reference 19). TRACG internally models the response surface for the void coefficient biases and uncertainties for known dependencies due to the relative moderator density and exposure on nodal basis. Therefore, the void history bias determined through the methods review can be incorporated into the response surface "known" bias or through changes in lattice physics/core simulator methods for establishing the instantaneous cross-sections. Including the bias in the calculations negates the need for ensuring that plant-specific applications show sufficient margin. For application of TRACG to EPU and MELLLA+ applications, the TRACG methodology must incorporate the void history bias. The manner in which this void history bias is accounted for will be established by the NRC staff SE approving NEDE-32906P, Supplement 3, "Migration to	Unchanged	This limitation will be implemented for all fuel types, including GNF2.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		TRACG04/PANAC11 from TRACG02/PANAC10," May 2006 (Reference 19). This limitation applies until the new TRACG/PANAC methodology is approved by the NRC staff.		
9.17	Steady-State 5 Percent Bypass Voiding	The instrumentation specification design bases limit the presence of bypass voiding to 5 percent (LRPM levels). Limiting the bypass voiding to less than 5 percent for long-term steady operation ensures that instrumentation is operated within the specification. For EPU and MELLLA+ operation, the bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions within the MELLLA+ upper boundary. The highest calculated bypass voiding at any LPRM level will be provided with the plant-specific SRLR.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.18	Stability Set points Adjustment	The NRC staff concludes that the presence bypass voiding at the low-flow conditions where instabilities are likely can result in calibration errors of less than 5 percent for OPRM cells and less than 2 percent for APRM signals. These calibration errors must be accounted for while determining the set points for any detect and suppress long-term methodology. The calibration values for the different long-term solutions are	Unchanged	This limitation will be implemented for all fuel types, including GNF2.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		specified in the associated sections of this SE, discussing the stability methodology.		
9.19	Void-Quality Correlation 1	For applications involving PANACEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady-state, transient, and accident conditions.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.20	Void-Quality Correlation 2	The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," dated May 2006. The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to LTR NEDC-32906P will be applicable as approved.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.
9.21	Mixed Core Method 1	Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GE's analytical methods or codes. The content of the plant-specific application will cover the topics	Not applicable	The purpose of the supplement is to address GNF2 fuel, not mixed fuel vendor cores. Therefore, this limitation

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		addressed in this SE as well as subjects relevant to application of GE's methods to legacy fuel. Alternatively, GE may supplement or revise LTR NEDC-33173P for mixed core application.		is not applicable to this supplement.
9.22	Mixed Core Method 2	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 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	Comply	This supplement is intended to address this limitation for GNF2 fuel. GNF2 does not use: • Square internal water channels water crosses • Gd rods simultaneously adjacent to water and vanished rods • 11x11 lattices • MOX fuel GNF2 fuel does not have significant changes in the Gd rod optical thickness or in Gd loading that would result in nodal reactivity bias beyond those previously established for GE14 fuel.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		review before the GE methods may be applied.		
9.23	MELLLA+ Eigenvalue Tracking	In the first plant-specific implementation of MELLLA+, the cycle-specific eigenvalue tracking data will be evaluated and submitted to NRC to establish the performance of nuclear methods under the operation in the new operating domain. The following data will be analyzed: • Hot critical eigenvalue, • Cold critical eigenvalue, • Nodal power distribution (measured and calculated TIP comparison), • Bundle power distribution (measured and calculated TIP comparison), • Thermal margin, • Core flow and pressure drop uncertainties, and • The MIP Criterion (e.g., determine if core and fuel design selected is expected to produce a plant response outside the prior experience base). Provision of evaluation of the core-tracking data will provide the NRC staff with bases to establish if operation at the expanded operating domain indicates: (1) changes in the performance of nuclear methods outside the EPU experience base; (2) changes in the available thermal margins; (3) need for changes in the uncertainties and NRC-approved criterion used in the	Unchanged	This limitation will be implemented for the first plant-specific implementation of MELLLA+, independent of fuel type.

Limitation Number from NRC SER	Limitation Title	Limitation Text	Disposition	Comments
		SLMCPR methodology; or (4) any anomaly that may require corrective actions.		
9.24	Plant Specific Application	The plant-specific applications will provide prediction of key parameters for cycle exposures for operation at EPU (and MELLLA+ for MELLLA+ applications). The plant-specific prediction of these key parameters will be plotted against the EPU Reference Plant experience base and MELLLA+ operating experience, if available. For evaluation of the margins available in the fuel design limits, plant-specific applications will also provide quarter core map (assuming core symmetry) showing bundle power, bundle operating LHGR, and MCPR for BOC, MOC, and EOC. Since the minimum margins to specific limits may occur at exposures other than the traditional BOC, MOC, and EOC, the data will be provided at these exposures.	Unchanged	This limitation will be implemented for all fuel types, including GNF2.

Appendix B

GEH Responses to RAIs

NRC RAI 1

Please confirm that Plant A from NEDC-33173P, Supplement 3 "Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel," dated July, 2009 (hereafter Supplement 3) is equivalent to Plant C from NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," (hereafter the interim methods licensing topical report (IMLTR)) Appendix B.

GEH Response

Plant A noted in Supplement 3 is equivalent to Plant C (denoted in Figure 2-4 and Table 2-10 in NEDC-33173P)

NRC RAI 2

Please provide information similar to that depicted in Figures 2-1, 2-2, 2-4, and 2-5 of Supplement 3 that addresses the relative performance of TGBLA06 and MCNP for GNF2 under controlled conditions.

Also, please demonstrate consistent performance in terms of the nuclear data extrapolation to higher void fractions between GNF2 and GE14. Please provide a comparison of the extrapolated infinite lattice multiplication factor (k_{inf}) to MCNP calculations at high void conditions. For example, please use the polynomial TGBLA06 fit for k_{inf} at 90 percent void fraction to compare to MCNP calculations (or an alternative higher order transport method) for GNF2 and GE14 fuel. Compare the trends in uncertainty with the extrapolation to higher void conditions.

GEH Response

Figures 2-1 through 2-6 provide comparisons of TGBLA06 and MCNP reactivity values for GE14 and GNF2 bundles. Figures 2-1 and 2-2 present uncontrolled lattice reactivity at beginning of life, where the TGBLA06 results have been extrapolated to 90% void fraction. Figure 2-1 shows the GNF2 lattices explicitly compared to the average $\pm\sigma$ for GE14. Figure 2-2 shows the GE14 lattices explicitly compared to the average $\pm\sigma$ for GNF2. Results are shown for seven GE14 lattices, four C Lattices numbered C1 through C4, and three D lattices numbered D1 through D3. Figures 2-3 and 2-4 show the same data at a lattice exposure of 65 GWD/MT. Reactivity comparisons for the beginning of life controlled state are presented in Figures 2- and 2-6. Controlled comparisons have not been generated for 90% void fraction.

All of the figures show good consistency between the GE14 lattice comparisons and the GNF2 lattice comparisons, considering the fact that the lattices chosen for the two products contain a varying degree of enrichments and Gadolinium loadings. Both GNF2 and GE14 results show a more negative TGBLA06/MCNP difference for the vanished zone lattices. The controlled comparisons show a slight difference in reactivity results at 70% voids. The remainder of the data is very consistent.

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Figure 2-1 – Beginning of Life TGBLA06/MCNP Reactivity Comparisons-GNF2 Data

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Figure 2-2 – Beginning of Life TGBLA06/MCNP Reactivity Comparisons-GE14 Data

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Figure 2-3 - 65GWD/MT Exposure TGBLA06/MCNP Reactivity Comparisons-GNF2 Data

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Figure 2-4 - 65GWD/MT Exposure TGBLA06/MCNP Reactivity Comparisons-GE14 Data

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Figure 2-5 - Beginning of Life Controlled TGBLA06/MCNP Reactivity Comparison-GNF2 Data

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Figure 2-6 - Beginning of Life Controlled TGBLA06/MCNP Reactivity Comparison-GE14 Data

NRC RAI-3

Please revise Supplement 3 to provide more clarity regarding the PRIME peak pellet exposure limit. As PRIME has not been approved by the Nuclear Regulatory Commission (NRC) staff, please delete the peak pellet exposure limit for consistency with the status of the NRC staff's ongoing review.

GEH Response

The purpose of the paragraph is to discuss the current licensed peak pellet exposure limit for GNF2 and that the limit will be evaluated using PRIME once it is approved by the NRC. Further, the PRIME evaluation would be consistent with Limitation 12 of the NRC's Safety Evaluation approving NEDC-33173P.

Therefore, the discussion of the peak pellet exposure limit for GE14 is extraneous and has been deleted as shown in the attached. The update will be incorporated into the "-A' version of the supplement.

- [[ ]]: The GNF2 reactivity biases relative to Monte Carlo results are consistent with previous 10x10 designs, showing no change needed in stability impact for [[ ]].
- **Bundle pressure drop**: The bundle pressure drop model is based on GNF2 full-scale pressure drop measurements. In addition to the total bundle pressure drop, the axial pressure profile is accurately modeled (see Figure 2-9) by the ISCOR model, which is embedded in the stability evaluations.

2.6.3 Adequacy of Existing Treatment and Alternate Approach

The design limits and methods associated with evaluation of Stability as specified in Table 1-2 can be used for the GNF2 design under EPU conditions. All models related to stability have the same uncertainties for the GNF2 design as the GE14 design, and are acceptable for GNF2-related stability analysis.

2.7 LICENSED EXPOSURE

The GNF2 fuel design is licensed to a peak pellet exposure limit of [[ ]] (Reference 4), based on the existing GSTRM methodology basis. GEH anticipates updating the peak pellet exposure limit for GNF2 fuel when the new PRIME methodology is applied (Reference 18) (See Appendix A).

This licensed peak pellet exposure limit is specified and applied in the process computer to assure that fuel is not operated beyond its analyzed basis. In this application, the best estimate value of the local exposure condition is monitored against the specified exposure limit.

2.7.1 Fuel Parameters That Affect Pellet Exposure

The fuel parameters that affect pellet exposure are unchanged for GNF2.

2.7.2 Treatment of Fuel Parameter Uncertainties

The overall pin power uncertainties are unchanged for GNF2 (Section 2.2.1.2).

NRC RAI 4

The GNF2 peak linear heat generation rate (LHGR) is higher than that for GE14. Therefore, the NRC staff expects that cores designed with GNF2 fuel may include higher powered bundles. This will have the affect of potentially increasing the degree of expected bypass void formation for these assemblies early in life.

Low Flow Conditions

Please evaluate the expected degree of bypass void formation under dual recirculation pump trip (2RPT) conditions for GNF2 assuming that the GNF2 was operating at or near the peak LHGR prior to the 2RPT. Compare these results to those obtained for GE14. Evaluate any adverse impact the GNF2 bypass void formation may have on: local power range monitor detector response, stability calculations, and power shape. Please provide justification that the stability setpoint setdown limitation provides a sufficiently large conservatism in terms of long term stability solution performance to bound GNF2 relative to GE14 noting that at higher LHGR, the bypass void formation is expected to be higher.

Radial Power Shape

The NRC staff notes that GNF2 includes part length rods (PLRs) at the lattice edge. Therefore, the effect of bypass void formation at high in-channel void fractions may not have the same impact for GNF2 as GE14 – or possibly the same impact but to a different extent. Please compare the degree of power shape flattening expected for bypass void conditions for these two fuel types at high in-channel void fraction. Please compare the redistributed power shape to the location of pins that are typically limiting in terms of boiling transition.

GEH Response

Low Flow Conditions

In compliance with Limitation 17 of the Interim Methods SER for operation under EPU and MELLLA+ conditions, GNF2 fuel will be designed in such a way as to preclude operation with bypass voids greater than 5%. Parameters related to bypass void formation, such as bundle power, are therefore constrained and will be limited to ensure that the 5% limit is met at all LPRM levels during steady state conditions within the licensed operating domain [Ref. 4-1]. Therefore, with respect to potential bypass void formation, local peaking or bundle power may be different, but the envelope of initial conditions that would exist prior to an AOO or a stability event (e.g., two recirculation pump trip) will be the consistent between GNF2 and GE14. The highest calculated bypass voiding at any LPRM level will continue to be provided with the plant specific SRLR.

Even if the bypass void fraction is initially 5% at lower flow conditions and the in channel and bypass voids can increase further under 2RPT conditions, the two recirculation pump trip (2RPT) is an AOO that results in very small MCPR changes (i.e., power margin is retained at the reduced flow rates), offering ample margin to OLMCPR limits throughout the transient with either GNF2 or GE14 fuel.

The loss of recirculation pumps also results in a Limiting Condition of Operation (LCO) for the plant [Ref. 4-2, 4-3] in a region of the power-flow map where stability protection and limits are considered. The stability setdown will be designed to ensure compliance with Limitation 18, which requires consideration of LPRM and APRM calibration errors (including a provision for bypass voids). Note that the setdown is not necessary for MELLLA+ plants employing DSS-CD.

In summary, given the relevant limitations, GNF2 is not expected to result in adverse impacts relative to bypass void formation. The remainder of our response to the staff's request for additional information considers the calculated impact of a 5% bypass void fraction.

<u>Radial Power shape</u>

Bypass voiding, while uncommon, will affect neutron moderation and alter the pin power distribution in the bundle. Tables 4-1 and 4-2 evaluate the change in pin power caused by a 5% bypass void fraction in a GE14 and GNF2 lattice. The lattice parameters follow:

- Both the GE14 and the GNF2 lattices are D lattices with average enrichments of 4.51 and 4.30%. Both lattices come from the region with 14 vanished rods near the top of the fuel bundle. The GE14 lattice contains 17 gadolinium rods and the GNF2 lattice contains 16 gadolinium rods. A D lattice was chosen because the wide gap corner rod will experience a larger perturbation from a change in bypass water density.

- An in-channel void fraction of 90% is used, which is at the upper end of a range of in-channel conditions (upper elevations) in a high power bundle. For the purpose of the analysis, both the bypass and water rod are assumed to be at 5% void fraction. The actual in channel void fraction corresponding to a 5% bypass and water rod void fraction depends on the flow split between the bypass, water rod, and active channel. This flow split depends on actual operating conditions and GNF2 application. The 90% void fraction is used because it yields the highest change in rod power due to a given change in bypass/ water rod void fraction.

Figures 4-1 and 4-2 show the percent change in rod peaking due to 5% bypass and water rod void fraction for the GE14 and GNF2 lattice as a function of position in the lattice. The wide gap location is denoted by the words "control blade" in the figure and is only there to locate the position of the control blade part on the interchannel gap. All calculations are performed in the uncontrolled configuration. The gadolinium rod locations are shaded grey and the top six peaking locations in the upper right half of the lattice are identified by bold, italic font. (The peaking in the lower left half is symmetric with the upper right half.)

- The percentage change in pin power is mainly a function of pin position and is quite similar for both lattice designs.

- All of the high peaked fuel rods in both designs are located next to the bypass channel and therefore suffer a decrease in power due to bypass voiding, lowering the overall lattice peak pin power. Placing the maximum enrichment rods near the bypass enhances lattice reactivity and lowers fuel cycle cost. This behavior exists in practically all modern bundle designs.

- Larger percentage changes [[ ]] are observed for gadolinium rods. This is because the gadolinium rods start out at very low initial peaking, so the percentage change is increased, but the absolute power never approaches the level of the non-gadolinium rods.

- The maximum increase in rod power (excluding Gadolinium rods) is [[ ]]for the GE14 lattice. This increase is located at the third row in from the edge of the bundle and occurs at a low power rod having a low probability of becoming a peak rod any time in the life of the bundle.

In summary, the impact of 5% bypass voiding on lattice pin power peaking is minimal, and generally results in a decrease in lattice pin power peaking. The minimal effect is also independent of product design, being slightly less for the GNF2 case than the GE14 case. The impact on bundle R-factor is also minimal, because the bypass voiding occurs at most over the top 20% of the bundle axial height. A further discussion of the effect of bypass voiding on the bundle R-factor can be found in Reference 4-4.

References:

4-1 GEH letter, J. Harrison (GEH) to NRC, "Implementation of Methods Limitations - NEDC-33173P (TAC No. MD0277)," MFN 08-693, September 18, 2008.

4-2 NUREG-1434, Standard Technical Specifications General Electric Plants, BWR/6, Vol. 1, Rev. 2, June 2001.

4-3 NUREG-1433, Standard Technical Specifications General Electric Plants, BWR/4, Vol. 1, Rev. 2, June 2001.

4-4 GEH Letter G. Stramback (GEH) to NRC, Responses to DSS-CD LTR RAIs (See RAI 18), MFN 05-133, November 11, 2005.

Control Blade

Conntrol Blade

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									]]

LPRM

Figure 4-1 - Per Cent Difference Between 5% Bypass Void Pin Power and No Bypass Void Pin Power – GNF2 Design

Control Blade

Control Blade

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LPRM

Figure 4-2 - Ratio of 5% Bypass Void Pin Power to No Bypass Void Pin Power - GE14 Design

NRC RAI 5

The NRC staff has questions regarding Figures 2-2 and 2-5 of Supplement 3:

a. Please provide the void history or void histories used to perform the TGBLA06 depletion calculations.

b. The NRC staff is aware that the version of TGBLA06 used to generate these nuclear data is corrected for the edge rod Dancoff factor calculation, but does this version also include the updates to the low-lying plutonium resonance correction?

GEH Response

The depletion history used to generated the isotopics for the 65 GWD/MT TGBLA06/MCNP reactivity comparisons carried out at 40% void fraction. The version of TGBLA06 used to generate the nuclear data includes the Dancoff correction as well as the updates to the low-lying resonance correction.

NRC RAI 6

Please clarify Figures 2-1, 2-2, 2-4, and 2-5 of Supplement 3. Specifically, clarify what is meant by relative water density. Please address that points appear for GE14 and GNF2 at the same "relative water density," however, given different geometries and arrangements of PLRs, it is not expected that identical void fractions would yield identical relative water densities, depending on how this quantity is defined.

For example, if relative water density is defined according to equation (1), the relative water

$$U = \left(\frac{A_f}{A_f + A_{byp} + A_{wr}} \right) \frac{\rho_f}{\rho_o} + \left(\frac{A_{byp} + A_{wr}}{A_f + A_{byp} + A_{wr}} \right) \frac{\rho_{byp}}{\rho_o}$$

density appears to be lattice geometry dependent for a given void fraction.

Where:	U	is the relative water density,
	P	is the static density,
	A	is the flow area,
	0	denotes reference,
	f	denotes in-channel,
	byp	denotes external bypass, and
	wr	denotes water rod

At a given void fraction, the relative water density appears to vary between lattices as a function of the in-channel flow area. Please clarify why several lattices appear on these figures at the same relative water densities.

In the –A version of the TR, please revise these figures by adjusting the label of the independent axis or shifting the points to a relative water density that is consistent with the definition provided in equation (1).

<u>GEH Response</u>

Figures 2-1, 2-2, 2-4, and 2-5 have been modified to reflect the correct average density variation of the vanished rod lattices. The revised figures appear below and will be incorporated in the –A version.

Figure 2-1 TGBLA06 Fission Density Benchmark for GNF2 , at BOC

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Figure 2-2 TGBLA06 Fission Density Benchmark for GNF2, at 65 GWD/MT

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Figure 2-4 TGBLA06 Reactivity Benchmark for GNF2, at BOC (GE14 1σ uncertainty band, dashed line)

[[

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Figure 2-5 TGBLA06 Reactivity Benchmark for GNF2, at high exposure (GE14 1σ uncertainty band, dashed line)

[[

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NRC RAI 7

Table 2-1 of Supplement 3 appears to be in error, particularly the second entry in the bottom row. Please correct this table in the –A version of the LTR.

GEH Response

Table 2-1 contains a typographical error. The revised table is attached and will be incorporated in the –A version of the supplement.

Revised Table 2-1

GNF2 Axial Regions

Name	Description	Axial Zone Length
[[		
		]]

NRC RAI 8

Void history exposure reactivity coefficient biases and uncertainties predicted for GE14 may not be applicable to GNF2. The staff notes that the GNF2 heavy metal loading is higher than for GE14 and, as such, at equivalent void conditions the GNF2 spectrum is expected to be harder than the GE14 spectrum on this basis.

Please provide a limited demonstration that is similar to Table 2-11 from the IMLTR for the GNF2 lattices presented in IMLTR Supplement 3. It is not necessary to provide an equally comprehensive table, but please consider the higher exposure range and please focus on lattices expected to experience higher void fractions located near the top of the core (e.g., PLN2, VAN2, etc.).

Alternatively, the staff is aware of a higher order transport based lattice method under development by GNF, LANCER 2. It would be acceptable to address this RAI with a table similar to Table 2-11 that compares the TGBLA06 void reactivity coefficient biases and uncertainties for GNF2 compared to LANCER 2.

Alternatively, the staff is aware that a void history exposure reactivity coefficient biases and uncertainties were incorporated in TRACG04. This model requires a database generated using MCNP and TGBLA06 for GE14 and GNF2 lattices. Please provide a comparison of these void reactivity coefficient data between the two fuel designs. To justify the continued applicability of the bias and uncertainty used in ODYN.

Alternatively, using a GNF2 MELLLA+ core design, provide sensitivity studies using TRACG04 (with and without the void history exposure reactivity coefficient biases and uncertainties model) to generate a table similar to Table 2-10 of the IMLTR to demonstrate that the sensitivities for GNF2 are essentially the same or conservative relative to GE14.

GEH Response

The response to NRC RAI 8 will be provided at a later date.

Overview

This response addresses the RAI by means of the approach suggested in paragraph 4 of the RAI. This response is an update to the previous RAI responses related to the void coefficient correction model in TRACG04. The void coefficient corrections have been updated based on extensive TGBLA06/MCNP comparisons for GNF2 lattices. The method to account for the biases and the uncertainties in the void coefficient model had previously been modified to include the effects due to void history (VH). Section *C1AX* in Reference [8-1] describes the TRACG methodology with the void history effects included. Calculations had previously been performed including the void history effects as part of the void coefficient correction model. By comparison to similar calculations performed with the model deactivated, these calculations reveal that correcting for biases in the void coefficient can result in small changes to the key AOO calculated parameter of ΔCPR/ICPR. A similar comparison updated to include the GNF2 lattices is indicated here as Figure 8-1. The figure shows a typical calculated CPR response for the most limiting channel for the usually limiting pressurization event, a turbine trip with no bypass (TTNB). [[

]] These impacts may vary by core and cycle since the model depends on core and cycle-specific elements such as exposure, instantaneous voids and void history. One key point is that the impacts, either positive or negative, are incorporated in the TRACG AOO methodology as amended in Reference [8-1] to incorporate the effects due to void history in determining the biases and uncertainties in the void coefficient on a plant and cycle-specific basis. [[

]] Both key points were previously supported in Reference [8-1] and are by this response also shown to continue to be supported for applications involving GNF2 fuel.

In addition to the ΔCPR/ICPR value tabulated for the limiting channel at the limiting point in time (as plotted in Figure 8-1), Table 8-1 shows how the void coefficient correction model impacts other key transient quantities. [[

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Figure 8-1 Typical CPR Impact of Updated Void Coefficient Correction Model

Table 8-1 Typical Impact on Other Key Transient Outputs (turbine trip with no bypass)

Description	TRACG04P model on	TRACG04P model off	ODYN
peak total power (%)	392	357	426
peak vessel pressure (MPa)	8.909	8.892	8.842
limiting ΔCPR/ICPR	0.165	0.155	0.200
peak centerline temperature (K) (UO_2 melting occurs at ~3000 K)	1568	1565	not available
max. hoop / yield stress ratio	0.0904	0.0898	not available
water level decrease (inch)	44.7	44.7	45.0

Additional Details

The technical basis for the TRACG04 model was previously provided in Reference [8-1] so it will not be duplicated here. This response will simply compare how the model has been updated to incorporate additional information for GNF2 lattices.

As previously described in Reference [8-1], TRACG04 uses a 3-D neutron kinetics model based on the PANAC11 model[8-2] that uses neutronics parameters provided by TGBLA06. The nodal reactivity is calculated[8-3] [[ ]]. All of these parameters are expressed in terms of the instantaneous moderator density and also include a dependency on moderator density history and nodal exposure. Consequently, the infinite multiplication factor also has these same dependencies.

The biases and uncertainties in void coefficient as determined from the PANAC11 originate in the biases and uncertainties in the infinite lattice eigenvalues (k_∞) calculated by the TGBLA06 lattice physics code [[ ]] Values of k_∞ at a number of points were calculated for a representative set of lattices with 10x10 geometry at [[ ]] different exposures of [[ ]] and at different void histories (VH) of [[ ]] for in-channel instantaneous voids (IV) of [[ ]] using both TGBLA06 and MCNP. The number of lattices of each type and other details related to the previous and current datasets are provided in Table 8-2. The processing of the k_∞ point values to determine the void coefficient values is the same as used previously so the details provided previously in Reference [8-1] will not be repeated here.

In the previous evaluations described in Reference [8-1], a number of 10x10 lattices (set “a”) were considered, but none represented the exact GNF2 partially-rodded lattices. However, the previously-considered fully-rodded lattices were representative of those found in the lower part of GNF2 bundles. Many additional lattices (set “b”) representative of those used in GNF2 bundles have been evaluated. Table 8-2 provides details about the number of lattices in sets “a” and "b". The additional new lattices in set "b" were used together with the lattices previously evaluated for set "a" in order to extend the validity of the TRACG04 model to GNF2 lattices.

Table 8-2 Details of Previous and Current Databases

Description or Quantity	Set "a"	Set "b"	Combined "a" and "b"
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As previously observed in Reference [8-1], the implementation of void history effects into the TRACG04 model has allowed us to demonstrate (see Figure 8-1 and Table 8-1) that the CPR response with the complete model produces a ΔCPR/ICPR value that is [[ ]] resulting in a slightly [[ ]] minimum CPR value than when the model is turned off. For comparison purposes, the CPR response calculated by ODYN for the same core and conditions is also shown. [[

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Several statistical tests were performed to see how the new lattices for GNF2 bundles in set "b" were different or similar to those in set "a". By performing two-sample t-tests it was determined that it was appropriate to make the following combinations. [[

]] The resulting composition of the combined dataset is indicated in the rightmost column and bottom four rows of Table 8-2.

Like before, the response surfaces for the biases and uncertainties in the void coefficient that are modeled in TRACG04 are obtained from the derived void coefficient values by characterizing the response surfaces as a function [[ ]]. The response surfaces from the previous evaluation were shown in Reference [8-1] so they are not shown here; however, a visual comparison of the figures from Reference [8-1] to the updated ones shown here reveals that they are quite similar [[ ]].

The updated response surfaces for the relative biases are shown in Figure 8-2 and the updated response surfaces for the relative standard deviations are shown in Figure 8-3. In both figures there are [[ ]] surfaces corresponding to different void histories. For each surface the vertical axis is the in-channel instantaneous void fraction and the horizontal axis is the nodal exposure. The color scheme shown in the legends at the top of the figures denotes the ranges for the biases in Figure 8-2 and the ranges for the standard deviations in Figure 8-3. A negative bias means that the TGBLA06 void coefficient is smaller in absolute magnitude than the corresponding MCNP value.

The response surfaces for the biases in Figure 8-2 and the uncertainties in Figure 8-3 show that in the exposure range from about 15 to 25 GWd/STU that corresponds to the limiting CPR bundle for AOO analyses that the void coefficient bias [[ ]] For exposures less than 15 GWd/STU the PANAC11 standard process as supplied with TGBLA06 nuclear information [[ ]] Also for low exposures, the uncertainties tend to be [[ ]]. As the poison is *burned* and the bundles approach their peak reactivity and power, the void coefficient biases and uncertainties [[ ]]. Void history does not begin to make any discernable differences until the exposure has exceeded about 25 GWd/STU as previously noted in Reference [8-1]. At exposures above this point the standard process tends to [[

]] A larger void coefficient (in the absolute sense) is conservative because it tends to produce a more dynamic power response and a less favorable CPR response. [[

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The relative biases in Figure 8-2 are higher for exposures less than 15 GWd/STU simply because the absolute void coefficient values to which the relative values are normalized are smaller for these exposures. The same statement applies to the relative uncertainties shown in Figure 8-3. The *standard process* used in PANAC11 to capture these trends is based on void coefficient dependencies with respect to IV that were established at a void history of 40%. As previously noted in Reference [8-1], at exposures above 25 GWd/STU the standard process tends to [[

]] The model used in TRACG04 to correct the standard process remains unchanged from what was described previously in Reference [8-1]; therefore, those details are not repeated here.

As previously explained in Reference [8-1], [[

]]. The normality of these normalized residual errors for the entire population was analyzed to determine whether it is appropriate to assume that the residual errors are normally distributed. The histogram for the [[ ]] normalized standard residual errors is shown in Figure 8-4 together with the red normal curve and a statistical summary for the residuals.

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Figure 8-2 Void Coefficient Relative Bias Updated for GNF2 Lattices

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Figure 8-3 Void Coefficient Relative Standard Deviation Updated for GNF2 Lattices

Because this population of residuals is in standard form, it should theoretically have a mean of zero and a standard deviation of unity. The actual mean of the residuals is essentially zero but the standard deviation is 0.976 which means that modeling the residuals with an assumed normal distribution conservatively yields a larger variability. [[

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Figure 8-4. Histogram and Statistical Summary of the Standard Residual Errors

How TRACG04 applies the uncertainties and biases has not changed from what was reviewed and approved by the NRC staff in connection with Reference [8-4]. [[

]] As stated previously in Reference [8-1], the impact of not modeling the void coefficient biases is on the order of [[ ]] in the TRACG calculated values of transient ΔCPR/ICPR for most fast pressurization events. The current results shown for GNF2 in Table 8-1 are consistent with this generalization. Whether the bias is conservative or not depends on the exposure distribution and the relative water density distribution in the core and that is why it is important for a best-estimate calculation like TRACG to model the bias as a function of the nodal conditions. On the other hand, the model used in ODYN (where the bias is not considered) is seen from the comparisons presented in Figure 8-1 and Table 8-1 to be adequately conservative even without considering the bias. This is the justification for continuing to use ODYN for transient applications involving GNF2 fuel.

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Figure 8-5. Normality Probability Plot of the Standard Residual Errors

For sensitivity studies, a core-wide bias and uncertainty in void coefficient can be specified through the TRACG04 input in a way that is comparable to how ODYN would apply this uncertainty. As an example of the importance of the void coefficient uncertainty, consider that for a typical BWR/4 plant a variation at the one-sigma level of [[ ]] in the void coefficient when applied to all nodes in the core corresponds to a sensitivity of [[ ]] in the ΔCPR/ICPR for a turbine trip without bypass. Since the turbine trip without bypass tends to be the most limiting AOO transient for purposes of calculating ΔCPR/ICPR, this uncertainty value can be bounded by the conservative ODYN methodology Reference [8-5] at greater than two sigma [[ ]].

Because it has not changed, the detailed Technical Description of the TRACG void coefficient correction model previously provided in Reference [8-1] in the latter part of the RAI response has not been duplicated here.

References

[8-1] Response to Request for Additional Information (RAI) 30, RE: NEDE-32906P, Supplement 3, *Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,* (TAC No. MD2569), Letter from R. E. Kingston (GEH) to M. C. Honcharik (USNRC) and USNRC Document Control Desk, MFN 08-483, May 30, 2008.

[8-2] *Steady-State Nuclear Methods*, NEDE-30130-P-A and NEDO-30130-A, April 1985, and for TGBLA Version 06 and PANACEA Version 11, Letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

[8-3] J. G. M. Andersen, et al., *TRACG Model Description*, NEDE-32176P, Revision 4, January 2008.

[8-4] Final Safety Evaluation for General Electric Nuclear Energy (GENE) topical Report (TR) NEDE-32906P, Revision 2, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses" (TAC No. MD0249), Letter from Ho K. Nieh (NRC) to Bob E. Brown (GE), August 29, 2006.

[8-5] *Qualification of the One-dimensional Core Transient Model for Boiling Water Reactors*, NEDO-24154-A and NEDE-24154-P-A, Volumes I, II and III, August 1986.

NRC RAI 9

Please compare Figures 2-1, 2-2, 2-3, and 2-4 of Supplement 3 to Figures 3-1, 3-2, 3-3, and 3-4 of GNF TR NEDC-33270P, Revision 2, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)" (ADAMS Package Accession No. ML091830644). These figures appear inconsistent. Please explain.

GEH Response

The calculations documented in NEDC-33270P, Rev. 2, were performed with a version of TGBLA06, which did not contain the corrections for Dancoff factor or the updates to the low-lying plutonium resonance correction. The results shown in Figures 2-1, 2-2, 2-3, and 2-4 of Supplement 3 to NEDC-33173P were regenerated using the most up to date version of TGBLA06.

NRC RAI 10

Please clarify the version of COBRA used to do the sub-channel analysis. Is this version of COBRA consistent with the COBRAG model description that was submitted to the NRC staff in LTR NEDE-32199P, Revision 1, "COBRAG Subchannel Code – Model Description Report" (ADAMS Package Accession No. ML071910320)?

GEH Response

The version of COBRA used in the GNF2 sub-channel analysis to support the void correlation is consistent with LTR NEDE-32199P.

NRC RAI -11

Certain features of the GNF2 fuel assembly make the bundle more stable than GE14 bundles in terms of core, regional, and channel instability modes. These include a population of shorter PLRs to increase single phase pressure drop to two phase pressure drop ratio, and a thicker fuel pellet that increases the fuel thermal time constant. Therefore, it is expected that the exclusion and back-up stability protection (BSP) regions analyzed for GNF2 fueled cores must be analyzed at increased power to flow ratios relative to the analysis conditions for GE14 fuel.

Please provide an analysis at equivalent core and channel decay ratio (0.8) for GE14 and GNF2. The results of this analysis should provide an assessment of the relative degree of in-channel and bypass void for GNF2 at the exclusion or BSP region boundary relative to GE14. Comment on the significance of the difference in these void fractions. In estimating the bypass void fraction, please use the ISCOR code (conservative) at power/flow conditions identified using ODYSY.

Please also consider that the NRC staff has approved the use of the modified shape function (MSF) relative to the generic shape function. Therefore, the limiting conditions analyzed for GE14 fuel in response to RAI 3.2(a)(iii) from the NRC staff's review of the IMLTR are not necessarily the most limiting conditions along the exclusion boundary for GNF2 fuel. In this comparison please consider the MSF an intermediate point between the natural circulation and high flow control lines to demonstrate the limiting condition has been identified.

Please compare the calculated thermal-hydraulic conditions predicted for the stability threshold for GNF2 fuel (i.e., decay ratio ~ 0.8) to the predicted thermal-hydraulic conditions present for the ODYSY high decay ratio benchmarks.

Provide justification that the sensitivity of the ODYSY code to any additional uncertainty introduced by the higher void conditions has been adequately addressed by the IMLTR safety evaluation (SE) conditions and limitations.

GEH Response

There are two applications of the Backup Stability Protection (BSP) – one for Option III and one for Detect and Suppress Solution – Confirmation Density (DSS-CD).

Only the BSP for Option III is considered here. The BSP for Option III covers for operating domain up to the Extended Power Uprate/Maximum Extended Load Line Limit Analysis (EPU/MELLLA) operating domain. The BSP for DSS-CD (which is for Maximum Extended Load Line Limit Analysis Plus (MELLLA+) implementation) has an Automated BSP with a flow-clamp scram feature, which ensures an automatic reactor scram with a two-recirculation pump trip event. Hence as a backup stability solution, there is less concern due to bypass voiding for BSP for DSS-CD. The DSS-CD LTR (Reference 11-1) outlines the requirements to cover for a new fuel product line like GNF2.

BSP for Option III

The calculation of the BSP region boundary is based on a conservative ODYSY (One-Dimensional Dynamic Code for Stability) acceptance criteria map that may be influenced by the core wide axial power distribution calculation. However, the ODYSY methodology requires the use of a conservative Haling power shape, and this is a limiting flat axial power shape compared to actual power shapes throughout the cycle. Therefore, uncertainties in the actual axial power distribution do not affect the calculation of the BSP region. Also, any uncertainties in either local or radial power distribution have no influence on the core-wide decay ratio (Reference 11-2).

Two new ODYSY cases were generated based on the Amendment 22 (A22) GNF2 and GE14 PANACEA wrap-ups. One case was along the MELLLA boundary (or the High Flow Control Line, HFCL) and the other case was along the Natural Circulation Line (NCL). The power/flow search along the NCL and the power/flow search along the HFCL yielded the 0.80 core decay ratio power/flow state points for both GNF2 and GE14 as requested in this RAI. These two comparisons bound the Modified Shape Function (MSF) or Generic Shape Function (GSF) state points in terms of bypass voiding conditions. Hence no additional MSF or GSF state points are presented here. Figure 11-1 illustrates the Controlled Entry Region boundary corresponding to the 0.80 core decay ratio for both GNF2 and GE14 using the GSF. The GNF2 Controlled Entry Region boundary tends to be smaller compared to that of GE14 as was pointed out by the staff. A smaller Controlled Entry Region boundary is conservative for the bypass voiding application since this penetrates deeper into the less stable region of the power/flow map (top left corner), where bypass voiding is more severe.

Please note that the BSP analysis is used a backup solution for Option III and that the BSP Scram Region boundary cannot be smaller than the Base BSP Scram Region (Reference 11-5), with the boundaries generated by applying either the GSF or MSF to Points A and B:

Point A: Intersection of the MELLLA upper boundary and 40% rated core flow,

Point B: Intersection of the NCL and 100% Original Licensed thermal Power (OLTP) load line.

The Base BSP Scram Region with the GSF option is also illustrated in Figure 11-1. Hence, the size of the Controlled Entry Region is also limited by the Base BSP Scram Region.

The conservative ISCOR bypass heating model at these power/flow conditions was used in the ODYSY evaluation. ISCOR computes bounding values of the bypass void fraction because the [[

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The GNF2 and GE14 bypass voiding results are summarized in Tables 11-1a and 1b, respectively. Since the calculated HFCL point is at a lower power/flow point than the corresponding Base BSP Scram Region end point, the results at the Base BSP Scram Region are also included and used in the GNF2/GE14 comparison for HFCL. For the NCL, the calculated Controlled Entry Region boundaries are lower than the Base BSP Scram Region and will be used in the GNF2/GE14 comparison.

In general, the GNF2 bypass flow elevation head is smaller than that of GE14 and hence the bypass flow tends to be lower for GNF2 at the same power/flow conditions. This resulted in a higher bypass exit void fraction (EVF) for GNF2 relative to GE14.

Along the NCL, the GNF2 average bypass EVF is only slightly higher [[

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at the Base BSP Scram Region boundary. The hot channel bypass EVF also shows a similar difference. Please note that the hot channel was applied with a 1.28 radial peaking factor. The hot channel bypass void model in ISCOR provides bounding values of the hot channel bypass voids, but the values are not realistic. Furthermore, the ISCOR hot channel methodology does not account for bypass cross flow that will tend to increase flow in high power zones thus reducing the bypass voids to nearly the core average level (Reference 11-3).

Despite the difference between GNF2 and GE14, these EVFs are in line with the numbers reported to the NRC as shown in Table 11-2 for MELLLA conditions. Hence the GNF2 numbers are within the ODYSY application methodology.

The hot channel (HC) in-channel void fractions at the top of the active fuel are also in line with the numbers provided earlier in the Interim Methods Licensing Topical Report (IMLTR) as shown in Table 11-3. [[

]] Hence the GNF2 numbers are within the ODYSY application methodology.

The thermal-hydraulic conditions calculated for GNF2 and GE14 BSP Controlled Entry Region boundaries are in line with the decay ratios and power/flow conditions observed in Table 11-4. The calculated decay ratios are covered by the Vermont Yankee tests. In addition, the highest core average power/flow ratios for GNF2 (57.3 MW/Mlbm/hr) and GE14 (54.6 MW/Mlbm/hr) are covered by the VY benchmark data (57.8 MW/Mlbm/hr). Hence the GNF2 numbers are within the ODYSY application methodology.

The sensitivity of the ODYSY code to any additional uncertainty introduced by the higher void conditions is adequately addressed by the SE conditions and limitations for NEDC-33173P.

As stated in the Nuclear Regulatory Commission (NRC) Safety Evaluation for NEDC-33173P (SE) (Reference 11-4, Section 6.2), the current Option III penalty in calibration errors (of less than 5 percent) for Oscillation Power Range Monitor (OPRM) cells associated with bypass voiding, is very conservative for the OPRM system since the original basis did not account for the attenuation of the OPRM cell average signal. If an OPRM channel is miscalibrated by a given factor of X percent due to bypass voids, the same bias error magnitude applies to the peak amplitude and to the average. When the peak over average is computed, the bias error (miscalibration) factor cancels out, and the percent oscillation amplitude is maintained regardless of the value (X percent) of the bias error. GEH has not credited the bias error of the average signal in the 5% calibration error penalty. This 5% penalty is adequate to cover for the expected increase in the bypass voiding due to the GNF2.

As noted in the NRC IMLTR SE (Reference 11-4, Section 6.3), the exclusion region calculations are based the following facts:

1. Exclusion regions calculation procedures are well-defined by the approved stability Long Term Solution methodology, and they use mostly prescribed power shapes. Therefore, power distribution uncertainties have a small effect on the size of the exclusion regions.

2. The ±0.2 uncertainty imposed by the DR < 0.8 criterion captures the possible effect of power distribution uncertainties and cross-section methodology errors (including the effect on void reactivity coefficient).

3. The ±0.2 uncertainty level is justified by the ODYSY and TRACG validation database. For these validation analyses, the neutronic methodology included the errors.

The implementation of BSP for Option III is a manual solution. It does not rely on the Average Power Range Monitor (APRM) flow-biased flux scram line as the means of reactor Safety Limit Minimum Critical Power Ratio (SLMCPR) protection. If a plant enters the BSP Scram Region, a manual scram is required. As long as the BSP Controlled Entry Region boundary is generated correctly, the impact due to the bypass voiding on the BSP Controlled Entry Region is minimal. The measured APRM power may be off by 1% to 2% rated power due to the bypass voiding. This is a small uncertainty that is within the typical reactor power uncertainty.

References:

11-1 NEDC–33075P–A, Rev. 6, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," January 2008.

11-2 NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," February 2006.

11-3 MFN 06-209, "Remaining Responses to Methods RAIs - Interim Methods LTR," June 30, 2006.

11-4 Final Safety Evaluation for GE Hitachi Nuclear Energy America, LLC Licensing Topical Report NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" (TAC No. MD0277), July 2009.

11-4 OG 02-0119-260, "Backup Stability Protection (BSP) for Inoperable Option III Solution," July 17, 2002.

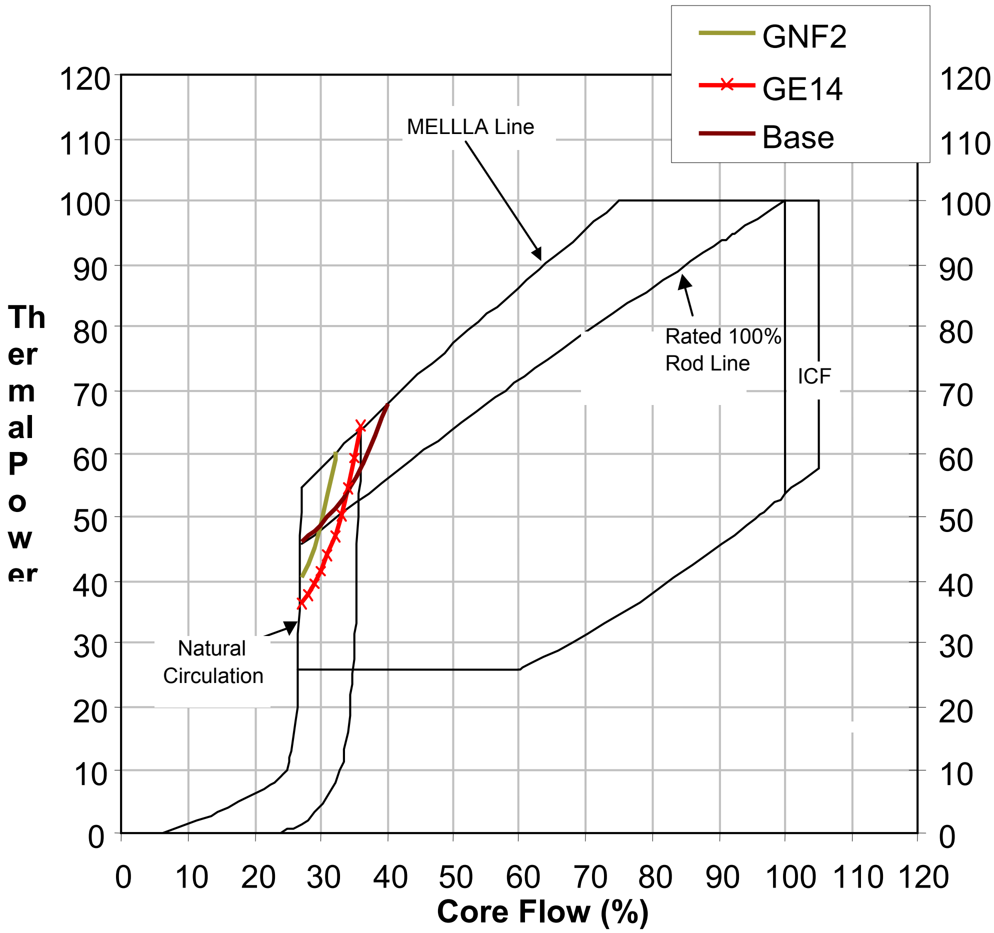

Figure 11-1. Illustration of the BSP Controlled Entry Regions and Base Scram Region

Table 11-1a. GNF2 Results*

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Table 11-1b. GE14 Results*

[[										
										]]

*EVF = Exit Void Fraction, CDR = Core Decay Ratio, HC= Hot Channel, rated power = 3323 MW, rated core flow = 108.5 Mlbm/hr.
** Hot channel applied with the [[ ]] radial peaking factor
*** Void fraction (VF) at the top of active fuel

Table 11-2. ISCOR Bypass Voids (from Reference 11-3, Table 3.2(a)-2)

[[					

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Table 11-3. ISCOR In-Channel Voids for Selected Events and Conditions
(from Reference 11-3, Table 4.1d-4.)

Event/Condition	**ISCOR Core Average In-Channel Voids (Top of Active Fuel)**	**ISCOR Hot Channel In-Channel Voids (Top of Active Fuel)**
NMP-2 Instability Event	73%	81%
Perry Instability Event	75%	86%
VY EPU/MELLLA	76%	85%
Hope Creek EPU/MELLLA	76%	86%

Table 11-4. Summary of ODYSY Results for Vermont Yankee High Decay Ratio Tests (from Reference 11-3, Table 4.1d-1 with power/flow ratio column newly added)

Test	Power/Flow	Power/Flow Ratio*	Test Data		ODYSY Results	
Point	**(% rated)**	**MW/Mlbm/hr**	**Decay Ratio**	**Frequency**	**Decay Ratio**	**Frequency**
6P	57.2/38.5	49.3	0.74	0.44	0.67	0.39
7N	51.2/32.6	52.1	1.00	0.43	0.99	0.38
8P	50.9/32.6	51.8	0.96	0.43	0.97	0.37
9P	48.1/32.4	49.3	0.81	0.42	0.86	0.36
10P	49.8/33.0	50.1	0.90	0.42	0.97	0.37
11P	67.1/38.5	57.8	0.85	0.47	0.85	0.42
12P	63.1/38.5	54.4	0.78	0.47	0.75	0.42

*Based on rated power = 1593 MW and rated core flow of 48 Mlbm/hr

NRC RAI 12

GNF2 LHGR limits are higher at low exposure than GE14 limits. However, the critical power performance as predicted by GEXL14 and GEXL17 indicates similarity between the two designs. To establish conservatism in the emergency core cooling system – loss of coolant accident (ECCS-LOCA) evaluation, it is customary to place the limiting bundle at the peak LHGR with the maximum stored energy in a bundle operating below the operating limit minimum critical power ratio (OLMCPR). This will yield the maximum value peak cladding temperature for the first peak that accounts for the maximum allowable operating space based on thermal limits considerations. Given that the GNF2 LHGR limit is much higher than that of the GE14 design, while the OLMCPR is expected to be similar, how are the ECCS-LOCA analyses initialized for GNF2 loaded cores at extended power uprate (EPU) or maximum extended load line limit analysis plus (MELLLA+) conditions? Please compare the conservatism associated with the ECCS-LOCA basis MCPR iteration for GNF2 fuel to the conservatism for GE14 fuel.

GEH Response

There is no difference in the methodology or initialization for ECCS-LOCA analysis for GNF2 loaded cores at EPU or MELLLA+ conditions as compared to a GE14 loaded core at EPU or MELLLA+ conditions. The difference in the peak LHGR of GNF2 vs. GE14 fuel will not lead to any difference in the ECCS-LOCA methodology or initialization process.

The fact that the LHGR limit for GNF2 fuel is larger than GE14 fuel does not change the ECCS-LOCA analysis modeling, which is set to represent a simplified, yet conservative core condition. The SAFER model considers [[

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Given the assumed EPU or MELLLA+ operating conditions, the above methodology determines and sets the power distribution in a conservative way. The fact that the GNF2 fuel will be able to reach a higher LHGR limit than GE14 fuel, [[ ]] The EPU and MELLLA+ conditions in the initialization will be factored into the bounding power distribution for the GNF2 fuel, just as they would be factored into the power distribution for GE14 fuel under such assumptions. This will yield the maximum first peak PCT that accounts for the maximum allowable operating space based on thermal limits considerations for GNF2 fuel in like manner as the methodology has been previously applied to other fuel such as GE14.

NRC RAI 13

The NRC staff has questions regarding the continued applicability of other relevant thermal-hydraulic models to GNF2 fuel.

In-core Liquid Entrainment

Please describe how liquid entrainment in the core is modeled for GNF2. Modern liquid entrainment correlations such as the one described in NEDE-32176P, Revision 3 appear to have geometry dependence. Please address the GNF2-specific geometry in the response.

Counter-Current Flow Limitation (CCFL)

- Please provide the definition for the characteristic length, also referred to as the effective diameter, used in the calculation of the CCFL.
- Please describe how the axially varying geometry of the GNF2 bundle is treated in SAFER and CORECOOL.
- Please compare the GNF2 geometry to the experiments that were used to develop the CCFL correlation.
- Please describe how the spacers are taken into account when using the CCFL correlation.

Spray Heat Transfer

Please justify the applicability of the CORECOOL core spray heat transfer model to GNF2. Please consider the differences in the qualification data and the GNF2 fuel design.

GEH Response

Response to In-core Liquid Entrainment

The current GEH ECCS/LOCA analysis methodology for BWR/2 to 6 is SAFER, which is not a two-fluid model code like TRACG (NEDE-32176P, Rev. 4). SAFER uses a validated drift-flux model to determine the vapor and liquid volumetric fluxes in terms of the void concentration parameter, C_o, and the void-weighted vapor drift velocity, V_{gj}. These drift-flux parameters, i.e., C_o and V_{gj}, are obtained from proprietary GEH (Findlay – Dix) correlation and do not require any entrainment model. So the entrainment model of TRACG or a similar code is not relevant for SAFER LOCA analysis of core loaded with GNF2 fuel. The same is true for other GEH codes namely ODYN and ODYSY.

Within the current GEH ECCS/LOCA analysis methodology, CORECOOL code is sometimes used in conjunction with SAFER to determine a more accurate peak cladding temperature (PCT) for plants where core spray heat transfer is important and the PCT is very high. CORECOOL uses a three-field model comprising of a liquid film on the fuel rods and channel wall, liquid droplets in the vapor core and a (superheated or saturated) steam or vapor core. The decay heat is removed by radiation and convective heat transfer which is enhanced by the presence of liquid droplets formed from the break up of spray water at the upper tie plate and sputtering front of falling liquid films. The upward vapor flow rates are small at low decay heat of interest and no entrainment from the liquid film is predicted. Therefore, GNF2-specific geometry is not relevant even for CORECOOL for in-core liquid entrainment.

TRACG may be used as a best-estimate code in support of upper bound PCT calculation for GNF2 fuel. TRACG can simulate the axially varying geometry of GNF2 fuel assembly and uses mechanistic validated in-core liquid entrainment models and correlations.

Response to Counter-Current Flow Limitation

In the current GEH methodology, the characteristic length or the effective diameter, D, for CCFL is eliminated by multiplying the original Wallis CCFL or "flooding" equation (Reference 13-1) by $D^{0.25}$. Thus, the modified non-dimensional superficial liquid and vapor velocities, j_l^* and j_v^*, in the current GEH methodology do not contain any characteristic length or effective diameter. The constant at the right hand side of the modified CCFL equation, K (defined by $C_{Wallis}D^{0.25}$), is directly obtained from the GNF2-specific experiments. Therefore, the definition of characteristic length or effective diameter for CCFL in the current GEH methodology is irrelevant.

For the GE8 and later fuel, the upper tie plate flow area was opened to reduce pressure drop across the tie plate. As a result, the location where CCFL occurs moved [[

]] This treatment of CCFL at the UTP is conservative since liquid downflow into the bundle is reduced because of higher steam upflow at the UTP compared to that at a spacer below where the CCFL actually occurs.

Confirmatory CCFL testing for GNF2 spacers (for both Long Part Length Rods and Short Part Length Rods) have been performed . The GNF2 spacer CCFL constants are then compared to the experimentally determined StepII (GE10) and StepIII (GE11) spacer CCFL constants and the smallest of all these spacer CCFL constants is [[ ]] and this conservative value is used in SAFER for GNF2 CCFL at UTP.

Since GNF2 spacer CCFL constants are obtained from the confirmatory tests mentioned above, the axially varying geometry of GNF2 is not relevant in SAFER or CORECOOL for CCFL application.

TRACG, when used in support of the upper bound PCT calculation, simulates the axially varying geometry of the GNF2 bundle and CCFL is calculated at spacer and UTP locations as determined by the thermal hydraulic parameters.

Response to Spray Heat Transfer

CORECOOL has mechanistic models for core spray heat transfer (CSHT) as described in Chapter 5 of Reference 13-2. It consists of two basic models: a hydraulic model and a heat transfer model. The hydraulic model simulates steam, liquid droplets and liquid film flow on the fuel rods and channel wall independently and in a mechanistic manner. The heat transfer model is based on the one-dimensional heat conduction in the fuel rods and surface heat transfer including both convective and radiative heat transfers. The convective heat transfer is based on the well-known Dittus Boelter correlation with an enhancement due to liquid droplets in the vapor core. The Dittus Boelter correlation is valid over a wide range of parameters, which cover the GNF2 bundle conditions. The radiative heat transfer utilizes a mechanistic model based on view factors calculated from the actual bundle geometry.

GNF2 fuel assembly consists of eight (8) long part length rods (LPLRs) and six (6) short part length rods (SPLRs). Since the CORECOOL code structure allows a maximum of [[
]] some simplification is needed to model the GNF2 fuel bundle in CORECOOL. Specifically, [[

]]
Both of these modeling treatments act to conservatively increase the PCT and cladding oxidation.

CORECOOL has been qualified with various CSHT experiments as described in Chapter 7 of Reference 13-2. All of these experiments utilized fully-rodded heated bundle simulating a BWR fuel assembly. Although GNF2 fuel assembly consists of two types of part length rods, CORECOOL can simulate such fuel assembly using mechanistic modeling of both hydraulics and heat transfer. However, because of the conservative modeling as discussed above, the CORECOOL prediction of GNF2 core spray heat transfer is expected to be conservative.

CORECOOL is primarily applied to BWR/2 plants where core spray heat transfer plays an important role in evaluating ECCS performance. For other BWRs (BWR/3 to 6), the PCT is lower and CORECOOL is usually not applied.

References

13-1 G. B. Wallis, "One-dimensional Two-phase Flow," pp. 336 – 338, McGraw-Hill Book Co. Inc., New York, 1969.

13-2 NEDO-30996-A, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants, Volume I, SAFER – Long Term Inventory Model for BWR Loss-of-Coolant Analysis," Class I, March 1988.

<u>NRC RAI-14</u>

Section 4.2 is not sufficiently detailed. In reference to the table in Section 4.2, please address the following sections:

a) The “BWR product line” includes BWR/2. Please clarify.

b) Please footnote or otherwise clarify the “fuel product line” applicability statement to make it consistent with the Mixed Core Limitations in the NRC staff’s SE to the IMLTR.

c) Please clarify the “licensing methodology” section. This section refers to GEH nuclear and safety analysis methods. Is it more appropriate to list GNF or a combination of GEH and GNF?

d) In “Operating Domain,” please correct the typographical error “ELLA” to read “ELLLA.”

e) The “Stability Solution” section states “GE Stability Solutions.” Is it more appropriate to identify the solutions as BWR Owners’ Group (BWROG) (for Options EIA, I-D, II, and III) and GEH (for Detect and Suppress Solution – Confirmation Density (DSS-CD)) stability solutions?

<u>GEH Response</u>

<u>Response to Part a</u>

The NRC approved NEDC-33173P with a BWR product line that includes BWR/2 plants. The Methods LTR is applicable to expanded operating domains including EPU and MELLLA+. GEH LTR's NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A address EPU applications and are applicable to BWR/2 plants. However, MELLLA+ applications are addressed by NEDC-33006, which is not applicable to BWR/2 plants. To clarify, a footnote was added to the applicability table in Section 4.2 as shown in the attached. The update will be incorporated into the "-A' version of the supplement.

Response to Part b

The phrase, "non-GE," was deleted from the applicability table in Section 4.2 as shown in the attached. The update will be incorporated into the "-A' version of the supplement.

Response to Part c

Throughout the Methods LTR, as well as Supplement 3, reference is made to GEH methods. These methods include analytical methods developed by GEH and GNF. The use of the term GEH methods is to describe the methods available to GEH and GNF and is not used to define ownership.

Response to Part d

The abbreviation for ELLLA was corrected in the applicability table in Section 4.2 as shown in the attached. The update will be incorporated into the "-A' version of the supplement.

Response to Part e

The use of the phrase, "GE Stability Solutions," was used in the Methods LTR, and was continued as part of Supplement 3 as well, since it is unaffected by the addition of the GNF2 fuel design. The use of GE Stability Solutions is used to describe stability solutions that utilize GEH analytical methods and is not used to define ownership.

4.0 LICENSING APPLICATION

4.1 OVERVIEW

The purpose of this supplement is to extend the application of Reference 1 to GNF2 fuel.

4.2 APPLICABILITY

The Applicability of GE Methods to Expanded Operating Domains LTR basis is applicable to current GEH BWR product lines licensed with GEH nuclear and safety analysis methods. The Methods LTR is applicable to plants that include current GNF fuels including GNF2. The application of these codes complies with the limitations, restrictions and conditions specified in the approving NRC SER for each code.

The parameters establishing the Applicability of GEH Methods to Expanded Operating Domains applicability envelope are:

Parameter	Generic Value
BWR Product Line	BWR/2-6*
Fuel Product Line	GE fuel designs using square arrays of fuel rods, including 7x7, 8x8, 9x9, and 10x10 designs and GNF2
Licensing Methodology	GEH Nuclear and Safety Analysis Methods
Operating Domain	CPPU, EPU, with MELLLA+ including currently licensed operating domains (e.g., ELLLA, MELLLA) and operational flexibility features
Maximum Rated Power Level	120% OLTP
Stability Solution	GE Stability Solutions

***MELLLA+ is not applicable to BWR/2 plants consistent with NEDC-33006P-A (Reference 2)**

4.3 PLANT SPECIFIC APPLICATION PROCESS

Each plant seeking to apply the Methods LTR must provide information supporting the application that demonstrates that the plant parameters are within the applicability definition in Section 4.2.

NRC RAI 15

Please evaluate any additional uncertainty in the power distribution that may be introduced due to the effect of bypass void formation on traversing in-core probe (TIP) instruments. Please consider conditions of bypass voiding expected for GNF2 operating at or near the LHGR limits. Please address thermal and gamma TIP instruments separately. The evaluation should consider the influence of radial power distribution, J-factor, and instrument sensitivity. The power distribution uncertainties should consider integrated TIP (radial) redings near high powered GNF2 assemblies as well as axial power distribution, which may affect the LHGR uncertainty.

GEH Response

There is nothing in the GNF2 design that alters the LPRM, Gamma TIP or Neutron TIP response to changes in the bypass void fraction. The GNF2 channel and LPRM/TIP location is identical to GE14. In the upper part of the bundle, the only difference between the two designs is the location of the part length rods. The similarity of LPRM/TIP change for the two designs is supported by the analysis presented in the response to RAI–4. The change in the corner rod power provides an upper bound for the change in both thermal and gamma TIP response due to the presence of bypass voids. These changes are summarized in the Table 15-1.

The changes are basically the same for GE14 and GNF2. The narrow-narrow corner change is slightly less than the wide-wide corner change. In C lattice plants the narrow and wide inter channel gaps are the same, so the average change would apply. [[

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Reference

15-1 NEDC–32601P–A, Methodology and Uncertainties for Safety Limit MCPR Evaluation, August 1999.

Table 15-1 Change in Corner Rod Power Due to 5% Bypass Void Fraction

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NRC RAI 16

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. Therefore, 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 (hence void fraction) for GNF2 bundles operating in an EPU core may exceed those experienced for limiting GE14 bundles.

Please provide a demonstration calculation of the GNF2 R-factor for an EPU or MELLLA+ transition core application (one reload quantity of GNF2 fuel and the balance GE14 fuel) that illustrates how Limitation 6 is met. Specifically address the higher allowable LHGR for GNF2 fuel.

GEH Response

Reference 1 describes the R-factor parameter and the methodology for computing it for BWR fuel bundles with partial length fuel rods. This same methodology is used in computing the R-factor for use with GEXL17 in critical power predictions for GNF2. As part of verifying GEXL17 for GNF2, the void conditions expected during operation were considered in relation to the calculation of the bundle R-factor. Several GNF2 equilibrium core designs were evaluated to determine the bundle void fractions for limiting bundles. These core designs were developed at a range of power densities, including EPU conditions, and the bundle average void fractions observed. Both the void history and instantaneous void fraction were considered. The R-factor [[ ]]

These designs were prepared to represent typical GNF2 application [[ ]]. The instantaneous void fractions for the limiting bundles throughout the cycle were observed [[ ]] and the most limiting bundles were very well represented by a bundle average instantaneous void fraction of [[ ]] which was selected for use in generating R-factors for GNF2. The observed bundle instantaneous void fractions as a function of MCPR for the four (4) cores considered are provided in the Figures 16-1.

Also, a representative core comprised of one batch of GNF2 with the remainder of the core consisting of GE14 is evaluated as requested in the RAI. This core is a high power density core representative of a reactor that has installed EPU. The [[ ]] relationship is provided in the graph below and the average instantaneous void fraction for the limiting bundles throughout the cycle is [[ ]]. GNF's overall approach in confirming compliance with Limitation 6 is to perform this evaluation on a plant specific basis for plants referencing the IMLTR and confirm that the reference void fraction value [[ ]] for R-factor determination remains applicable based on the cycle average instantaneous void fraction for the limiting fuel.

In summary, a bundle average void fraction of [[ ]] is very representative of limiting GNF2 bundles over a range of conditions that includes EPU and is adequate for use in calculating rod power distributions for the bundle R-factor.

[[

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Figure 6-1a Void Fractions at Operating Conditions

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Figure 16-1b Void Fractions at Operating Conditions

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Figure 16-1c First Reload of GNF2

NRC RAI 17

BSP has been approved by the NRC staff for implementation at Option III plants. However, NEDC-33173P, Revision 1 does not explicitly discuss BSP for Option III. Please provide a discussion similar to those in NEDC-33173P, Revision 1 addressing BSP for Option III. It is expected that the nature of this discussion will be generic, but please give specific consideration to GNF2.

GEH Response

The NRC-approved ODYSY methodology (Reference 17-1) is used in the Backup Stability Protection (BSP) regions calculation for every reload. The BSP regions consist of two regions, I-Scram and II-Controlled Entry. The Base BSP Scram Region and Base BSP Controlled Entry Region are defined by statepoints on the High Flow Control Line (HCFL) and on the Natural Circulation Line (NCL). The bounding plant-specific BSP region state points must enclose the corresponding Base BSP region state points on the High Flow Control Line (HFCL) and on the Natural Circulation Line (NCL). If a calculated BSP region state point is located inside the corresponding base BSP region state point, then it must be replaced by the corresponding base BSP region state point. If a calculated BSP region state point is located outside the corresponding base BSP region state point, this point is acceptable for use. That is, the selected points will result in the largest, or most conservative, region sizes. The proposed BSP Scram and Controlled Entry region boundaries are constructed by connecting the corresponding bounding state points on the HFCL and the NCL using a shape function like the Generic Shape Function (GSF) or the Modified Shape Function (MSF).

The calculation of the BSP region boundary is based on a conservative ODYSY acceptance criteria map that may be influenced by the core wide axial power distribution calculation. [[

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The results of the BSP for Option III analysis are documented in the supplemental reload licensing report. Usually, two sets of BSP regions may be generated for different rated and reduced feedwater temperature ranges. Because the BSP regions are plant- and cycle-specific it is required to calculate or validate them for each core design. Therefore, a core design including GNF2 fuel is still required to satisfy the ODYSY acceptance criteria map in the determination of the cycle-specific BSP regions.

References:

17-1 NEDE-33213P-A, Revision 0, Licensing Topical Report, "ODYSY Application for Stability Licensing Calculations, including Option I-D and II Long Term Solutions," April 2009.

NRC RAI 18

Section 3.6 of the IMLTR refers to the generic applicability envelope for MCPR margin. Section 3.6 of Supplement 3 only discusses the pressure drop and critical power correlation. The NRC staff notes that the generic applicability envelope is only applicable to GE14 and earlier fuel designs.

To assist the NRC staff in its review, please describe the calculations (and specify the methods used) that must be performed to support DSS-CD for (1) GNF2 loaded cores implementing DSS-CD, and (2) plants that utilize DSS-CD that are introducing GNF2 fuel.

Please update Supplement 3 with a discussion regarding the analyses that must be performed to support DSS-CD and address the relevant uncertainties. This discussion should be similar to the discussions provided in the IMLTR for the other stability solutions. It is expected that this discussion will be generic in nature. Please include additional discussion that specifically addresses GNF2 uncertainties.

GEH Response

Section 3.6 of the Methods LTR (Reference 18-1) discusses the use of DSS-CD and that the uncertainties in power distribution calculations and void reactivity are accounted for in the stability analysis. Section 3.6 of NEDC-33173P, Supplement 3 (Reference 18-2) concludes that the stability analysis established for DSS-CD is applicable to GNF2 fuel.

The stability Section 3.6 references Section 2.2.1.2 as a basis for pressure drop correlation. Section 2.2.1.2 describes the pressure drops and the comparison between calculated and measured pressure drops for GNF2. The provided comparison is related to ISCOR calculated pressure drops. The reference to ISCOR is for the leakage flow calculation. This is dominated by the various models for the frictional pressures drop in the leakage paths from the lower plenum and channel to the bypass, and these models are identical between TRACG and ISCOR for normal flow in the leakage paths. The leakage flow models are documented in the TRACG Model Description report (Reference 18-3).

For the active bundle the TRACG pressure drop is evaluated by direct comparisons to pressure drop data from the ATLAS and Stern Lab test bundles. Bundle pressure drop comparisons are documented in the TRACG Qualification report (Reference 18-4) for GE14 fuel. TRACG hydraulic model to calculate pressure drops is not changed for different fuel types, whereas the loss coefficients input in the TRACG channel model typically change for different fuel types. Figure 18-1 represents the comparisons between Stern GNF2 test assembly pressure drops (measured) and TRACG predicted pressure drops (calculated). [[

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The DSS-CD LTR, NEDC-33075P-A (Reference 18-5), specifies the process to extend the applicability of DSS-CD to new fuel such as the GNF2 fuel design. NEDC-33075P-A, Table 6-5 identifies various fuel transitions and the required TRACG cases required for the different transitions. One of the included transitions, Scenario 1b, addresses transitioning from an approved fuel design (e. g., GE14) to an unapproved GEH fuel (e. g., GNF2). Approved/unapproved GE fuel designs are in reference to fuel designs approval for DSS-CD applications.

In such a case, the DSS-CD LTR requires [[

]] This process would apply to both cases where a GNF2 loaded core is implementing DSS-CD and where a DSS-CD core is introducing GNF2 fuel.

The NRC subsequently approved DSS-CD LTR in letter dated November 27, 2006 (Reference 18-5). The NRC reviewed the protocol as documented in SE Section 3.3. Further, Limitations 3 and 5 of the NRC's SE states:

3. For situations where the plant applicability checklist is not satisfied (e.g., introduction of a new fuel type), Tables 6.3 and 6.4 of NEDC-33075P, Revision 5, describe a technically acceptable procedure to extend the future applicability of DSS-CD.

5. Table 6.5 of NEDC-33075P, Revision 5, describes the fuel transition scenarios, which are subject to a plant-specific review for each application.

Section 2.6 of Supplement 3 (Reference 18-2) addresses the treatment of uncertainties relative to fuel parameters that affect stability. That discussion is applicable to DSS-CD as well. The update to address this clarification will be incorporated into the "-A' version of the supplement.

References

18-1 NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains", February 2006.

18-2 NEDC-33173P, Supplement 3, "Applicability of GE Methods to Expanded Operating Domains – Supplement for GNF2 Fuel", July 2009.

18-3 NEDE-32176P, "TRACG Model Description," Rev. 4, January 2008.

18-4 NEDE-32177P, "TRACG Qualification," Rev. 3, August 2007.

18-5 NEDC-33075P-A, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density", Rev. 6, January 2008.

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Figure 18-1 TRACG calculated pressure drops versus measured Stern GNF2 test assembly pressure drops (circle symbols) for GNF2 fuel at different power and mass flux values.