ENCLOSURE 2

M170048

Comment Summary Table and Draft SE Markup

Non-Proprietary Information - Class I (Public)

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1, from which has the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

Comment Summary for Draft Safety Evaluation for Amendment 37 to Topical Report NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement (CAC No. MF0743)

Note: Page numbers shown in this table reflect the page numbers in this enclosure. Due to suggested changes in the Safety Evaluation (SE) and the addition of proprietary marking, these page numbers differ from the page numbers in the draft SE sent to GNF for review.

Location	Comment
Pg 1 Line 15	Added two additional letters that modified the original Amendment 37 request. The references were added at the end of the references list and no renumbering was included. Suggested addition included in markup and references were added at the end of Section 7.
Pg 1 Line 25	Because of the subsequent withdrawal of two of the original parts of the Amendment 37 changes, we suggest changing the word from incorporates to proposed. This proposed change was subsequently withdrawn by MFN 16-013, February 26, 2016. Section S.2.2.3.3.1 was not modified except by the reordering and renumbering the references. Comment included in markup.
Pg 1 Lines 41-43	The TRACG cycle coastdown was withdrawn from the Amendment 37 package by MFN 13-074, September 13, 2013 and submitted as a standalone change via Amendment 39. It was reviewed and approved by the NRC by Letter from Mirela Gavrilas (NRC) to Jerald G. Head (GEH), “Final Safety Evaluation for Amendment 39 to Global Nuclear Fuel - Americas Topical Report NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, "General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement" (TAC No. MF2797),” MFN 15-026, April 7, 2015. Suggested addition included in markup.
Pg 1 Line 47-Pg 2 Line 1-2	MFN 16-082, November 3, 2016, proposed some modifications to the original Section 1.4 wording. The modifications allow GNF to add fuel product line compliance reports to the list in Section 1.4 without the submittal of an amendment request to the NRC for review and approval. Suggested addition included in markup.
Pg 4 Line 35-40	Proprietary content identified and marked.
Pg 5 Line 41	The word “reduction” should be “increase.” Suggested change included in markup.

Location	Comment
Pg 5 Line 47-50	The sentence starting ….The flow dependent…. Needs to be broken into two sentences for clarity. Suggested changes included in markup.
Pg 6 Line 4-22	Section 3.4.1 is correct but the TRACG coastdown modification has been approved and incorporated into GESTAR as noted in the comment on Pg 1 Lines 41-43 above. Suggest removing Section 3.4.1.
Pg 6 Lines 42-43 And 46	Slight modifications for clarity. Suggested changes included in markup.
Pg 7 Line 32	The Reference 3 citation should be Reference 5. Suggested change included in markup
Pg 9 Lines 8-9	Add "of Reference 5" to clarify the location of the citation. Suggested addition included in markup
Pg 9 Lines 38-40	Limitation/Condition 2 points back to the criteria stated in the original approved Amendment 22 (1990) as being the basis for the current new fuel introduction acceptance criteria. The Amendment 22 criteria were incorporated into GESTAR II when approved in 1990; however, since that time, the criteria have been modified in GESTAR II to reflect changes in processes and methodology over the years. RAI 1 (See SE Reference 3) was a question specifically about the approved GEXL correlation form. The GEXL criteria are expressed in Section 1.1.7 and 1.2.7 of GESTAR II and by reference to References 1-5 and 1-6 of GESTAR II. Section 1.2.7 C. of GESTAR II explicitly states that ….. "To assure that no unreviewed safety question exists, the functional form of the current correlations must be maintained. A correlation with a different form must be approved by the NRC prior to use." These words have not changed since Amendment 22 was incorporated into Revision 10 of GESTAR II. Given the explicit statements in GESTAR II, Limitation/Condition 2 does not seem to be necessary. If it is retained, the scope should be limited to changes in the GEXL form relative to the criteria currently in GESTAR II for the correlation development.
Pg 11 Lines 5-8	Add Reference 12 Letter from Brian R. Moore (GNF) to Document Control Desk (US NRC), "Amendment 39 to NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement," MFN 13-074, September 13, 2013. Suggested addition included in markup.

Location	Comment
Pg 11 Lines 9-12	Add Reference 13 Letter from Brian R. Moore (GNF) to Document Control Desk (US NRC), “Modification to Proposed Amendment 37 to NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement (TAC No. MF0743),” MFN 16-082, November 3, 2016. Suggested addition included in markup.

OFFICE OF NUCLEAR REACTOR REGULATION DRAFT SAFETY EVALUATION OF TOPICAL REPORT AMENDMENT 37 TO NEDE-24011-P-A-19 AND NEDE-24011-P-A-19-US, GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL (GESTAR II) AND THE US SUPPLEMENT (CAC NO. MF0743)

1.0 INTRODUCTION AND BACKGROUND

By letter dated February 13, 2013 (Reference 1) Global Nuclear Fuel (GNF) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review, Topical Report (TR) "Amendment 37 to NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement." The information in Reference 1 was supplemented by additional information (Reference 2) in response to staff's request for additional information (RAI) (Reference 3). References 4**,** ~~and~~ 5**, 12 and 13** were submitted by GNF either modifying the original Amendment 37 or adding modification to the original Amendment 37 to GESTAR II.

The GESTAR report provides information and description of fuel design and licensing criteria and fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases. The report also provides information and approved methods used to determine reactor operating limits, both independent of plant specific as well as plant specific, and the transient and accident analysis methods that are used in the country specific supplements.

Amendment 37 ~~incorporates~~ **proposed** several changes to the previous versions of GESTAR II. A brief summary of these changes are listed below:

- Modification of reference to NEDE-31152P, *General Electric Fuel Bundle Design Report* and addition of plant and cycle specific *Fuel Bundle Information Report (FBIR)*
- Addition of a new Section 1.4 on compliance report references
- Clarification of actions that plants must take if they deviate from generically analyzed Banked Position Withdrawal Sequence (BPWS) bank notch positions
- Addition of end-of-cycle (EOC) coastdown for reloads analyzed with TRACG
- Incorporation of SAFER/PRIME in to US supplements for loss-of-coolant accident (LOCA) analysis and removed SAFER/GEST~~E~~R report list and associated references
- Incorporation of TRACG-LOCA methodology for emergency core cooling system (ECCS) performance analysis

Global Nuclear Fuel submitted a letter dated February 26, 2016 (Reference 4) withdrawing the GESTAR II supplement for banked position withdrawal sequence changes from Amendment 37. **The proposed TRACG cycle coastdown change was withdrawn from the Amendment 37 package by MFN 13-074, September 13, 2013 and submitted and approved as a standalone change via Amendment 39. (Reference 12)**
In addition to the above, GNF submitted a supplement to Reference 1, by letter dated July 18, 2016, modification of Amendment 37 for GESTAR II USA supplement to support the use of approved TRACG-LOCA TR (Reference 5).
GNF further proposed modifications to the original Section 1.4 content in MFN 16-082, November 3, 2016, (Reference 13) to allow GNF to add fuel product line compliance

reports to the list in Section 1.4 without the submittal of an amendment request to the NRC for review and approval.

The Nuclear Performance and Code Review Branch (SNPB) staff has reviewed the GNF Amendment 37 to GESTAR II from GNF. The draft safety evaluation (SE) for the Amendment 37 to GESTAR II follows.

2.0 REGULATORY EVALUATION

GESTAR II report, NEDE-24011-P-A/NEDO-24011-A, provides an NRC-approved fuel design and core reload process. The approved methodology and acceptance criteria detailed within TR NEDE-24011 are cited within many boiling water reactor (BWR) technical specifications as references in the core operating limits report.

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs and adherence to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria 10, 27, and 35 is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design." In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- the number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

For LOCA evaluations, two methods are listed in GESTAR II. These two separate ECCS evaluation methodologies to determine the effects of loss-of-coolant accident are in accordance with the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

3.0 TECHNICAL EVALUATION – BWRs WITH GE FUEL (NEDE-24011-P)

This section presents the evaluation of changes in Amendment 37 to GESTAR II for BWRs for which the General Electric Company (GE) provides fuel.

3.1 Fuel Bundle Design and Fuel Licensing Acceptance Criteria

Fuel bundle design information on specific fuel bundles for each cycle will be listed in the FBIR that is given in Appendix A, *Standard Supplemental Reload Licensing Report and Fuel Bundle Information Report,* of the country-specific supplement to the country specific part of the TR, NEDE-24011-P.

Section 1.4 of the submitted document for Amendment 37 lists the compliance reports for the GNF fuel product line. Section 1.4 of NEDE-24011-P lists compliance reports for the fuel product line for GE11, GE13, GE12, GE14, and GNF2 fuel designs from GNF.

Through the RAI, the NRC staff, based on Sections 1.2.7 and 1.2.7 C, requested clarification whether General Electric-Hitachi (GEH) has explicit assurance from NRC staff that a critical power ratio (CPR) correlation for a new fuel design needs NRC staff approval. The applicant stated that the Amendment 22 process proposed in 1989 and approved by NRC in 1990 provided a framework of criteria and guidelines for a new fuel design to be effectively licensed without explicit NRC review and approval. The 1990 SE assured the applicant that *if a fuel design complies with the fuel acceptance criteria, it is acceptable for licensing applications without the explicit review* (Reference 11). Sections 1.1.7 and 1.2.7 list the characteristics of the

GEXL model, its requirements and constraints. The 1990 SE approving Amendment 22 also approved the GEXL process.

The staff notes that, if a fuel design deviates from the acceptance criteria as specified in the 1990 SE, the licensing of such fuel will require NRC staff review and approval for the new fuel design and revision to GEXL process.

3.2 Fuel Mechanical Design

Amendment 37 request does not contain any major changes in Section 2, *Fuel Mechanical Design,* except a few additions of references and editorial changes.

The staff, in an RAI, requested to provide a description as to how the lattice-dependent maximum average peak linear heat generation rate (MAPLHGR)/or linear heat generation rate (LHGR) is different from the current methodology by which MAPLHGR is calculated. In a response to the RAI, GNF stated that though for the majority of plants the MAPLHGR limit is set at the limiting exposure so to assure compliance by emergency core cooling system – loss-of-coolant accident (ECCS-LOCA) analysis to the acceptance criteria, the ECCS-LOCA evaluation model used with the SAFER model uses the axial power distribution conservatively by considering a limiting bundle with full length fuel rods and with the limits set to extremes for that single bundle. The power distribution is applied so that it is consistent with the integral power traversing up through the bundle. This is the general condition where the LHGR limits are monitored directly. For plants that are challenged by the ECCS-LOCA criteria the CORCL code offers a more detailed assessment of core/bundles by taking into account fuel rod groupings with peaking variation as affected by part-length rods and radiation components to heat transfer. This application identifies spans between lattices and fuel rod condition as a function of exposure and assigns a MAPLHGR value for each node. These details are coupled with output from SAFER code in calculating the peak clad temperature (PCT) and local oxidation.

Section 2.2.2.7.2 specifies calculated cladding circumferential plastic strain limit during AOOs before and after the PRIME code implementation. For fuel product lines prior to PRIME implementation, as defined by the compliance reports in Section 1.4 of MFN 13-006, the fuel rod was evaluated to ensure that the calculated cladding circumferential plastic strain would not exceed 1 percent. [[

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The staff reviewed the minor changes to the fuel mechanical design part of Amendment 37 to the GESTAR II report and determined that the changes are acceptable.

3.3 Nuclear Design

The changes requested in Amendment 37 under this section are related to reactivity criterion associated with the storage of both irradiated (spent) fuel and new fuel and associated effective multiplication factor. The basic criterion in 10 CFR 50.68 for the storage of both irradiated and new fuel is that the effective multiplication factor of fuel stored under normal and abnormal conditions will be ≤ 0.950 for GE low-density and high-density racks over a temperature range of 4° C to 100° C. For cases where optimum moderation is credible for fresh unburned fuel in GE low-density racks, the maximum k-effective for optimum moderation condition shall be ≤ 0.98 per 10 CFR 50.68. These storage criteria will be satisfied if the cold uncontrolled in-core k-infinity for a lattice calculated for GE designed fuel storage has the following conditions:

(a) $k_\infty \leq 1.28$ for low-density spent fuel storage racks with an inter-rack spacing ≥ 11.70 inches (as revised through Reference 3).
(b) $k_\infty \leq 1.33$ for high-density spent fuel storage racks with an inter-rack spacing ≥ 6.563 inches.
(c) $k_\infty \leq 1.31$ for low-density new fuel vault storage racks with an inter-rack spacing ≥ 10.50 inches.

The licensee shall use a checkerboard array where only one out of every three storage locations in either linear direction contains a fuel bundle if a new fuel rack is used where there is no administrative control and/or design features to prevent optimum moderation **from** occurring.

The NRC staff, through an RAI, requested details of the k-effective and k-infinity calculations performed by the applicant. The applicant responded that the methodology used was the recent Peach Bottom spent fuel pool criticality submittal and accepted by the NRC staff in the Peach Bottom SE in Reference 7. In-core eigenvalues and exposure dependent, pin-by-pin isotopics are generated using the GEH/GNF lattice physics code TGBLA. TGBLA solves two-dimensional diffusion equations with diffusion parameters corrected by transport theory to provide system multiplication factors and perform burnup calculations. The in-rack k-effective calculations were performed using MCNP code using a robust geometry representation that can correctly model complex components in two or three dimensions. The applicant's methodology has been consistent with the most current NRC guidance for performing spent fuel pool criticality analyses listed in DSS-ISG-2010-01(Reference 7).

The NRC staff has reviewed the fresh and irradiated fuel storage criteria proposed and the methodology used in the above calculations in Amendment 37 to GESTAR II and has determined that they are acceptable.

3.4 Thermal-Hydraulic Design

The applicant, in the submittal for Amendment 37 to GESTAR II, has requested minor changes in Section 4.3.1.2.7, *Low Flow and Low Power Effects on MCPR [minimum critical power ratio].* The operating limit MCPR (OLMCPR) must be increased for low flow conditions and the OLMCPR must be increased for BWR/6 plants and plants with anticipatory reactor trip system (ARTS) (plants licensed for average power range monitor, rod block monitor and technical specification) at low flow and low power conditions. The ~~reduction~~ **increase** in the required OLMCPR for BWR/2-6 plants is accomplished by specifying an absolute MCPR as a function of core flow ($MCPR_f$) or as multiplier on the rated OLMCPR.

Both power and flow dependent limits on OLMCPR are imposed on plants licensed for the ARTS improvement program. The flow dependent OLMCPR ($MCPR_f$) is defined as a function of the core flow rate and **maximum core flow. The maximum core flow may be based on the** positioning of the scoop tube **for MG set plants** ~~on the recirculation pump motor~~ or the maximum core flow **capability for** ~~runout for plants with the~~ recirculation flow control valve~~s~~ or adjustable speed drive~~s~~ **plants**. $MCPR_f$ are provided in the cycle-specific Supplemental Reload Licensing Report (SRLR). The power dependent MCPR limits ($MCPR_p$) are also provided in the cycle-specific SRLR. The power–dependent OLMCPR, $MCPR_p$, is determined from the product of the OLMCPR at 100% of rated and a power–dependent multiplier, K*p*.

~~3.4.1 End–of–Cycle Coastdown Considerations~~

~~Section 4.3.1.2.8 of Reference 1 provides a methodology to analyze end-of-cycle coastdown conditions using the TRACG code.~~

~~Once an individual plant reaches end-of-rated (EOR), it may shutdown for refueling or it may be placed in a coastdown mode of operation. In the end–of–cycle coastdown type of operation the control rods are normally held in the all–rods–out position and the plant is allowed to coastdown to a lower percent of rated core power while maintaining rated core flow. Analysis of coastdown mode shows that the pressure and MCPR from the limiting pressurization AOO exhibit a larger margin for each of these points than the EOR condition. LHGR limits for the EOR condition are conservative for the coastdown period, since the core power will be decreasing and rated core flow will be maintained. The analysis shows that the coastdown operation beyond the EOR condition is conservatively bounded by the analysis at the EOR conditions. This is confirmed by the analysis performed with the ODYN code. Analyses with the TRACG code show the same trend and same conclusion for the coastdown operation.~~

~~The NRC staff concludes that the TRACG code can be used in the analyses related to plant coastdown operation.~~

4.0 TECHNICAL EVALUATION – GE BWRs IN THE UNITED STATES (NEDE-24011-P-US)

This section evaluates the changes requested in Amendment 37 to GESTAR II for BWRs that are operated in the United States.

4.1 Rod Withdrawal Error

Control rod withdrawal error (RWE) occurs when the reactor is operating at a power level above 75 percent of rated power. During the RWE, the reactor operator makes a procedural error and withdraws the maximum worth control rod to its rod block position. This causes a positive reactivity increase and resulting increase in average core power. The local power in the vicinity of the withdrawn control rod increases and potentially cause cladding damage due to overheating and possibly boiling transition. The resulting events consist of a local power range monitor (LPRM) alarm and rod block by rod block monitor (RBM) or rod withdrawal limit (RWL) or for some plants a full withdrawal. Under most normal operating conditions, no operator action is required except for responding to an LPRM alarm. If RWE is severe, the RBM will sound alarms and the operator will take appropriate corrective actions.

For BWR/2-6 plants, the ΔCPR from a control rod ~~ejection~~ **withdrawl error** (~~C~~R**W**E) is reported ~~for each fuel type~~ in the SRLR **for the specific analysis type**. The plant/cycle-independent generic bounding analysis for ARTS based systems will be developed for the first application of the ARTS based system for the specific plant. If ΔCPR is a limiting value for a particular fuel type, a plant**/cycle**-specific analysis will be performed for that fuel type.

For non-ARTS plants, operating at rated power with control rod (CR) pattern in thermal design limits, ensures conservative results. For ARTS basis and BWR/6 plants, the core is assumed operating at rated power with a control rod pattern that is adjusted to maximize the worth of the targeted error rod, or group of rods. For ARTS based RWE a statistical analysis is performed to determine the initial MCPR necessary to provide 95 percent confidence that the safety limit for minimum critical power will not be violated in 95 percent of the RWEs initiated. The BWR/6 RWE result is calculated based on the worst RWL-distance limited withdrawal segment occurring during a full withdrawal.

The staff reviewed the RWE part of Amendment 37 to GESTAR II and associated new references listed in Reference 3 and found the modifications acceptable.

4.2 Control Rod Drop Accident Evaluation

Control Rod Drop Accident (CRDA) consists of a rapid removal of a high worth control rod that results in a high local reactivity insertion in a relatively small region of the core. For large loosely coupled cores, the CRDA can cause significant shifts in the spatial power generation during the course of the excursion.

In the original submittal for Amendment 37 to GESTAR II (Reference 1) US Supplement Section S.2.2.3.1 clarified the actions that a plant must take if the plant deviates from the generically analyzed BPWS bank notch positions. GNF has withdrawn those changes from Amendment 37 to GESTAR II (Reference 4). Reference 4 lists new References S-16 of Reference 1 that lists the bank position withdrawal sequences and the improved BPWS control rod insertion process defined in Reference S-17 of Reference 1. For group notch plants, the references are S-14 and S-15 in Reference 1.

The NRC staff has reviewed the CRDA part of Amendment 37 to GESTAR II and found the references acceptable.

4.3 Effect of Fuel Densification

The effect of axial gap formation due to fuel densification on the rod drop accident results is discussed in Reference S–24 of Reference 3. The radiological consequences of the CRDA, assuming a full core of more recent GE fuel designs, are discussed in Reference S-25 of Reference ~~3~~5. Amendment 37 adds that for the GE14 and GNF2 product lines, and for future fuel products, the number of fuel rods that would reach 170 cal/gm is provided in the GESTAR II compliance report for the fuel product line.

4.4 Loss-Of-Coolant Accident

Reference 5 requested the NRC to incorporate the TRACG-LOCA methodology into Amendment 37 to GESTAR II.

4.4.1 SAFER/GESTR Methodology for ECCS Evaluation

Historically, there were two separate ECCS evaluation methodologies available to determine the effects of the LOCA in accordance with the requirements of 10 CFR 50.46 and Appendix K. The first methodology designated as SAFE/REFLOOD is now replaced by the SAFER/GESTR or SAFER/PRIME methodology in all US plants utilizing GEH LOCA evaluation methodology. The Amendment 37 submittal deleted the section on SAFE/REFLOOD and renumbered sections such that the SAFER/GESTR or SAFER/PRIME methodology comes first. The SAFER/GESTR methodology in Sections S.2.2.3.2.4.1 and S.2.2.3.2.2 of References 1 and 5 utilizes improved ECCS evaluation models as indicated in References S-26, S-27, and S-28 of References 1 and 5 to calculate a licensing PCT with margin as substantiated by statistical considerations. Appendix K required inputs are utilized only for the limiting break in order to establish a licensing margin to 10CFR50.46 limits. This input is revised as per Reference S-29 listed in References 1 and 5.

The SAFER/GESTR methodology is updated to include the fuel and gap properties from the approved PRIME methodology and therefore the LOCA analysis methodology is designated as SAFER/PRIME methodology. All other aspects of SAFER/GESTR methodology remain unchanged (Reference 8 which is listed as S-30 in Reference 5).

The SAFER/GESTR LOCA or the SAFER/PRIME 10 CFR Part 50, Appendix K, conformance calculations will be performed only for the limiting break of a nominally calculated break spectrum with a range of break flow multipliers between ~~.60~~**0.6** and 1.0. The licensing PCT is obtained as described in Reference S-28 of Reference 5 of this SE.

The staff has reviewed the parts of Amendment 37 to GESTAR II regarding the SAFER/GESTR or SAFER/PRIME LOCA methodology and the references mentioned in the Amendment 37 document and found that the changes are acceptable for plants to use the methodology.

4.4.2 Total LOCA Analysis

The total LOCA analysis based on the use of the SAFER/GESTR–LOCA or SAFER/PRIME codes (Sections S.2.2.3.2.1 and S.2.2.3.2.2 of Reference 5), is performed using the procedures outlined in Reference S-28 of Reference 5. The total LOCA analysis for each plant is independent of the SRLR, however, the SRLR will contain either the MAPLHGR or PCT as a function of exposure for fuel not previously licensed to operate in the specific reactor. An overview of the LOCA analysis process flow chart with SAFER/GESTR–LOCA or SAFER/PRIME application is given in Figure S-1 of Reference 5.

4.4.3 TRACG-LOCA Application Methodology

The TRACG-LOCA methodology for ECCS performance evaluation is approved by the NRC staff (References 9 and 10).

The TRACG-LOCA evaluation model was developed in accordance with the regulatory requirements established in 10 CFR 50.46, "Acceptance Criteria for ECCS for light water nuclear power reactors." The TRACG-LOCA methodology is based on nominal analysis together with a quantification of the uncertainties in the analysis following the guidelines of Regulatory Guide (RG) 1.157. The methodology for licensing application in the U.S. is structured following code scaling, applicability and uncertainty evaluation methodology.

TRACG-LOCA review considered the GE14 and GNF2 fuel designs with minor upgrades to these fuel designs, but the introduction of new fuel designs may require substantial revision to the evaluation model methodology. The staff review determined that there was general agreement between GEH phenomena identification and ranking table (PIRT) for TRACG-LOCA and other contemporary PIRTs (Phenomena Identification and Rankings). Additional guidance principal criteria for such applications are provided in NRC RG 1.157. GEH has also used the guidance and continues to develop and maintain the TRACG code per RG 1.203, "Transients and Accident Analysis Methods."

The analytic approach in TRACG-LOCA provides an acceptably detailed core model to capture the effects of variation in power distribution, time-in-cycle, and steady-state thermal-hydraulic performance, as recommended by RG 1.157. The staff has determined that GEH appropriately analyzed the break spectrum to determine the limiting breaks. The staff found that GEH treated initial conditions and plant operating parameters in an appropriate fashion. The TRACG-LOCA evaluation model used fuel parameter inputs supplied by the NRC-approved PRIME code. Fission heat is calculated using a point kinetics model, which has been validated against a more detailed, three-dimensional, nodal kinetics model. The TRACG decay heat model is carried forward from SAFER/GESTR-LOCA. The uncertainty quantification, an essential part of best-estimate methods, is achieved by statistical techniques described in Reference S-60 **of Reference 5**. An overview of the LOCA analysis process flow chart with TRACG-LOCA application is given in Figure S-2.

Chapter 10 of the SE for the TRACG-LOCA methodology lists several limitations that must be adhered to by the licensees when they adopt the TRACG-LOCA methodology for their ECCS performance analysis.

The NRC staff has reviewed the part of Amendment 37 to GESTAR II pertaining to the use of approved TRACG-LOCA methodology and determined that TRACG-LOCA methodology for ECCS performance analysis included in GESTAR II Amendment 37 is acceptable.

4.4.4 Main Steam Line Break Analysis

The main steam line break (MSLB) accident analysis depends on the operating thermal-hydraulic parameters of the overall reactor (such as pressure) and overall factors affecting the consequences (such as primary coolant activity). Results for the MSLB analysis are usually documented in the plant final safety analysis report. The initial SAFER/GESTR analysis for each plant included a re-analysis of the main steam line break, which establishes the non-limiting response of this break as compared to the other analyzed breaks. During the introduction of a new fuel product line, the differences in fuel design are evaluated with respect to the previous break spectrum and the response of the main steam line break.

The NRC staff reviewed this administrative change to the MSLB accident analysis and found it to be acceptable.

5.0 LIMITATIONS/CONDITIONS

1. When plants use TRACG-LOCA methodology, they shall comply with all the limitations listed in Chapter 10 of the SE for TRACG-LOCA methodology.
2. If a fuel design deviates from the acceptance criteria as specified in the 1990 SE (Reference 11), the licensing of such fuel shall require NRC staff review and approval for the new fuel design and revision to GEXL process

6.0 CONCLUSION

The staff has reviewed all of the administrative, editorial, and methodology changes listed in Amendment 37, Section 3.0 for BWRs with GE fuel and Section 4.0 for GE BWRs in the US. For BWRs with GE fuel, the changes reviewed are in the areas of bundle design and fuel licensing acceptance criteria, fuel mechanical design, nuclear design, and thermal-hydraulic design. For BWRs in the United States, the areas reviewed are RWE, CRDA, effect of fuel densification, LOCA (both SAFER/GESTR methodology for ECCS evaluation and TRACG-LOCA application methodology).

The NRC staff has determined that all of the changes listed in Amendment 37 to GESTAR II are acceptable.

7.0 REFERENCES

1. Letter and Enclosure 1 (MFN 13-006), "Amendment 37 to NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US, General Electric Standard Application for Reactor Fuel (GESTAR II) and the US Supplement," Global Nuclear Fuel, February 13, 2013.
2. Letter from J. A. Golla (US NRC) to J. G. Head (Ge-Hitachi Nuclear Energy), "Request for Additional Information Regarding Review of Amendment 37 to Licensing Topical Reports NEDE-24011-P-A-19 and NEDE-24011-P-A-19-US General ~~electric~~ **Electric** Standard Application for Reactor Fuel and the U.S. Supplement," US NRC, January 11, 2016.
3. Letter and Enclosures (MFN 16-053) from Brian R. Moore (GNF) to USNRC, "Response to NRC Requests for Additional Information Regarding Amendment 37 to GESTAR II," Global Nuclear Fuel, August 8, 2016.
4. Letter and Enclosure 1 (MFN 16-013) from Brian R. Moore (GNF) to USNRC, "Withdrawal of GESTAR II US Supplement Banked Position Withdrawal Sequence Changes from Amendment 37," Global Nuclear Fuel, February 26, 2016.
5. Letter and Enclosure 1 (MFN 16-048) from Brian R. Moore (GNF) to USNRC, "Modification of Amendment 37 for GESTAR II US Supplement to Support the Approval of the TRACG-LOCA Topical Report," Global Nuclear Fuel, July 18, 2016.
6. Letter, Richard B. Ennis (NRC) to Michael J. Pacilio (Exelon Nuclear), Subject: Peach Bottom Atomic Power Station, Units 2 and 3 – "Issuance of Amendments RE: Use of Neutron Absorbing Inserts in Spent Fuel Pool Storage Racks" (TAC Nos. ME7538 and ME7539), May 21 2013 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML13122A423).
7. DSS-ISG-2010-01, Revision 0, Final Division of Safety Systems Interim Staff Guidance, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools.
8. GNF Licensing TR, The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance, Technical Bases - NEDC-33256P-A, Qualification – NEDC-33257P-A, and Application Methodology – NEDC-33258PA, September 2010.
9. Letter from Kevin Hsueh (USNRC) to Jerald G. Head (GEH), "Final Safety Evaluation for GE Hitachi Nuclear Energy Americas, LLC Topical Report NEDE-33005, Revision 0, "Licensing Topical Report TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6," US NRC, February 14, 2017, ADAMS Accession No. ML17032A280.
10. Reports NEDE-33005P (Proprietary) and NEDO-33005 (Publicly Available), and Cover Letter MFN 11-001 "TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6," Project No. 710, January 27, 2011, ADAMS Package No. ML110280321.

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