ENCLOSURE 2

MFN 10-002

Comment Summary Table and Draft SE Markup

Non-Proprietary Information

IMPORTANT NOTICE

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

Comment Summary for Draft Safety Evaluation for Global Nuclear Fuel Americas Topical Reports NEDC-33256P, NEDC-33257P, and NEDC-33258P, The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance (TAC No. MD4114)

Location	Comment
Section 3.1.5 Cladding Thermal Expansion	Conclusions in this section are inconsistent with the Section 3.9.8 of the SER and Section 9 Limitation 3.b of the TER. Maximum cladding temperature is limited to [[ ]] in these sections. To avoid future misinterpretation of this limitation, GNF recommends making the Section 3.1.5 limitation consistent, i.e., the maximum cladding temperature to [[ ]] Suggested Changes in Markup.
Section 3.1.10 Integral Temperature Assessment	Page: 1 Section 3.9.6 of the SER notes the fuel temperature calculations in PRIME are qualified up to [[ ]] To be consistent with these sections, GNF recommend limiting the fuel temperature calculations to [[ ]] GNF understand that PRIME application will be limited to [[ ]] due to staff's concern about other PRIME models. Suggested Changes in Markup.
Section 3.3.1 Cladding Corrosion	The [[ ]] oxide value in Figure 3-1 of NEDC-33258P is not a limit. Rather, it is the [[ ]] used for the oxide perturbation in the PRIME application methodology. The [[ ]] During the ESBWR fuel review, GNF established an [[ ]] limit for GNF fuel designs based on GNF experience with successful operation of fuel with limited spalling. This limit is intended to protect fuel from extensive spalling or unusual corrosion/crud events and thus maintain uniform material properties. To be consistent with this [[ ]] in all licensing calculations. In cases where higher cladding oxidation is observed compared to GNF's experience base [[ ]] Suggested Changes in Markup.
Section 3.5.2 Young's Modulus and Poisson's Ratio	Conclusions in this section are inconsistent with the Section 3.9.8 of the SER and Section 9 limitation 3.b of the TER. Maximum cladding temperature is limited to [[ ]] in these sections. To avoid future misinterpretation of this limitation, GNF recommends to limit the maximum cladding temperature to [[ ]] Suggested Changes in Markup.
Section 3.7.2 Plenum Temperature	Due to the complexity of BWR fuel rod plenum designs [[ ]] and the elevation of the plenum in the core (particularly for different part length rod designs), [[ ]] As discussed in the RAI-41 response, the [[ ]] plenum gas temperature was calculated for a plenum including a [[ ]]While it was not especially clear in the RAI-41 response, this was intended to be an example, not a fixed number for all designs. [[ ]] For the GE14 IMLTR LHGR limit revision, for a fuel rod [[

Location	Comment
	]]the plenum temperature is recalculated to be [[ ]] using the same methodology as in RAI-41. The NRC staff reviewed the plenum temperature calculation methodology (the same as in the RAI-41 response) in detail as part of the GE14 compliance report audit. For the GNF2 design, the plenum gas temperature for the full, short and long part length rods are conservatively calculated using the RAI-41 methodology. The values for the GNF2 full, long PLR, and short PLR are approximately [[ ]] respectively. As noted above, plenum temperature is a function of the specific plenum design [[ ]] and their location in the core (differences in the gamma heating & power on top of the fuel column). Application of any predefined value for plenum gas temperature may be inappropriate for particular fuel designs and in some cases may produce non-conservative results. Based on this discussion, GNF recommends revising Section 3.7.2 as shown with revision tracking. Suggested Changes in Markup.
Section 4 Limitation 2 and Section 3.3.3	Although the EPRI Water Chemistry Guidelines are generally followed by US Utilities, there is a concern that referencing the EPRI Water Chemistry Guidelines may have unintended consequences or complications. This concern results from the fact that the Guidelines document is quite large, and at locations may involve parameters or suggestions that are not directly related with fuel performance parameters of interest for oxide and crud thickness. In addition, as GNF doesn't own this Guideline nor determine its revisions or changes, it is not possible to ensure compliance with future revisions. Therefore, GNF suggests revising Section 4 Limitation 2 and Section 3.3.3 as marked. The recommended values for the cycle average feedwater iron and zinc, (as well as the copper values, for the small number of plants with significant feedwater copper sources) are generally consistent with those suggested in the EPRI BWR Water Chemistry Guidelines, and the EPRI BWR Fuel Crud and Corrosion Guidelines. These recommendations are consistent with GNF fuel inspection experience and supported by the experience base used in developing Figure 3-1.

Markup of Draft Safety Evaluation for Global Nuclear Fuel Americas Topical Reports NEDC-33256P, NEDC-33257P, and NEDC-33258P, The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance (TAC No. MD4114)

The following markup illustrates the GNF proprietary content and suggestions per the comment summary table.

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NEDC-33256P, NEDC-33257P, and NEDC-33258P

THE PRIME MODEL FOR ANALYSIS OF FUEL ROD

THERMAL-MECHANICAL PERFORMANCE

GLOBAL NUCLEAR FUEL – AMERICAS, LLC

1.0 INTRODUCTION

By letter dated January 19, 2007 (Reference 1), as supplemented by letters dated February 27, 2009 (References 2 and 3) and August 11, 2009 (Reference 4), Global Nuclear Fuel – Americas, LLC (GNF) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Reports (TR) NEDC-33256P, NEDC-33257P, and NEDC-33258P, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical [T-M] Performance." These TRs describe the technical basis, qualification, and application methodology for the PRIME03 (hereafter PRIME) T-M fuel rod performance model.

The NRC staff's review was assisted by Pacific Northwest National Laboratory (PNNL). The NRC staff's conclusions on the acceptability of the PRIME model's technical basis, qualification, and application methodology are supported by PNNL's Technical Evaluation Report (TER), which is provided as a separate enclosure with this safety evaluation (SE).

The NRC staff assessed the impact on downstream calculations performed using the General Electric Stress and Thermal Analysis of Fuel Rods (GESTR)-Mechanical (GSTRM) fuel model and GSTRM gas gap conductivity files while the legacy safety analysis methods are migrated to the updated PRIME models. This assessment is documented in Appendix A of this SE. In this interim period, the thermal-mechanical operating limits (TMOL) will be determined using PRIME; however, transient safety analyses will be performed using the GSTRM inputs. The NRC staff notes that the GSTRM models do not account for the physical phenomenon of fuel pellet conductivity degradation with pellet exposure. The NRC staff refers to this process to be used during the period of time between PRIME approval and the eventual update of the legacy methods as the interim process.

2.0 REGULATORY EVALUATION

Regulatory guidance for the review of fuel system designs and adherence to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC-10 "Reactor Design," GDC-27 "Combined Reactivity Control Systems Capability," and GDC-35 "Emergency Core Cooling" 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" (Reference 5). In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

a. The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),

b. Fuel system damage is never so severe as to prevent control rod insertion when it is required,

c. The number of fuel rod failures is not underestimated for postulated accidents, and

d. Coolability is always maintained.

In addition to licensed reload methodologies, an approved fuel rod T-M model and application methodology is utilized to demonstrate compliance with SRP Section 4.2 fuel design and performance criteria. NEDC-33256P, NEDC-33257P, and NEDC-33258P describe the technical basis, qualification, and application methodology for the PRIME T-M fuel rod performance model. The NRC staff reviewed these TRs to: (1) ensure that the PRIME models are capable of accurately (or conservatively) predicting the in-reactor performance of fuel rods, (2) identify any limitations on the code's ability to perform this task, and (3) ensure that the application methodology conservatively accounts for model uncertainties and is capable of ensuring compliance with SRP Section 4.2 criteria.

3.0 TECHNICAL EVALUATION

The NRC staff's review of the PRIME fuel T-M performance model is summarized below:

- Verify material properties based on existing material property databases and supporting mechanical testing database.

- Verify each model (e.g., fuel temperature, creep, etc.) based on separate effects testing and measurements.

- Verify synergistic interaction of coupled models based on comparisons to instrumented in-pile test programs.

- Verify predicted in-reactor performance based on pool-side and hot-cell irradiation database.

- Verify application methodology properly accounts for model uncertainties to provide high confidence compliance to SRP Section 4.2 criteria.

In addition to comparing the computer model predictions to the supporting database, the NRC staff's contractor, PNNL, performed extensive computational comparisons of PRIME against the NRC audit code FRAPCON-3. The fuel performance models in FRAPCON-3 have been validated against an extensive database and are continually assessed against newer data as it becomes available (see References 6 and 7).

In addition to reviewing the material presented in the three PRIME TRs and in response to requests for additional information (RAIs), the NRC staff, along with contractors from PNNL, met with GNF to discuss unresolved issues associated with the ongoing PRIME review on February 12-13, 2008 (GEH - Washington DC), May 1-2, 2008 (GEH – Wilmington, NC), and June 30-July 1, 2009 (GEH - Wilmington, NC).

The NRC staff's review follows the logic of previous SEs for boiling water reactor (BWR) fuel rod performance codes such as Westinghouse's STAV7.2 and AREVA NP's RODEX4 models and methods. (References 8 and 9, respectively)

3.1 Thermal Modeling

Section 3 of NEDC-33256P describes the analytical techniques employed within PRIME to solve the heat generation and temperature distribution across the fuel pellet, fuel-to-cladding gap thermal conductivity, and heat transfer and temperature distribution across the cladding and into the coolant. The qualification of these thermal models against empirical data is provided in NEDC-33257P.

3.1.1 Pellet Heat Generation and Heat Transfer Methods

Fuel and cladding temperatures are calculated assuming steady-state, radial-only heat transfer from the pellet, across the pellet-cladding gap, through the cladding base metal, across the oxide and crud layers, and across the water film to the coolant. PNNL's technical assessment of the heat generation and heat transfer solution methods is provided in Section 2.1 of the TER. FRAPCON-3 comparison calculations were performed at different exposure levels for both uranium oxide (UO_2) and gadolinia bearing uranium oxide (UO_2–Gd_2O_3) fuel rod designs. Based upon NRC staff review of this assessment, the NRC staff finds the pellet heat generation and heat transfer solution methods in PRIME acceptable.

3.1.2 Fuel Thermal Conductivity

Unlike its predecessor GSTRM (see Reference 10), PRIME specifically accounts for the degradation in UO_2 thermal conductivity with increasing exposure. PNNL's technical assessment of the fuel thermal conductivity model is provided in Section 2.2 of the TER. Based upon FRAPCON-3 confirmatory analyses and comparisons to relevant empirical data, PNNL concluded that the [[

]] An assessment of the UO_2-Gd_2O_3 pellet thermal conductivity model, up to the requested [[ ]] weight percent (wt%) gadolinia level, yielded similar results.

Thermal conductivity is one piece of the overall fuel temperature solution. As will be shown below, [[ ]] the integral fuel temperature assessment concludes that PRIME is acceptable.

In Section 2 of NEDC-33256P, GNF requests approval of PRIME for fuel designs that include specified additives to the fuel pellet to achieve specific objectives (e.g., large grain size). However, no data comparisons were provided to justify PRIME models for additive fuel, such as thermal conductivity. In response to RAI 24 (Reference 2) regarding the licensing of PRIME for this application, GNF withdrew its request for NRC staff approval of PRIME for additive fuel. As such, approval for PRIME will be limited to UO_2 and UO_2-Gd_2O_3 fuel pellets with no additives beyond nominal trace elements (in accordance with ASTM[1] specifications).

[1] American Society for Testing and Materials

3.1.3 Fuel-to-Cladding Gap Conductivity

The fuel-to-cladding gap total conductivity consists of three components: (1) solid/solid contact conductance, (2) gap gas conductance, and (3) radiation heat transfer. PNNL's technical assessment of the gap conductivity model is provided in Section 2.3 of the TER. In their assessment, PNNL compared each of the three conductivity components to the corresponding representations in FRAPCON-3.

Section 3.2.2 of NEDC-33256P describes the gap gas thermal conductivity model and gas constants. Based upon [[ ]] PNNL concludes that the PRIME gap gas conductance is acceptable (Section 2.7 of the TER). Based upon NRC staff review of this assessment, the NRC staff finds the fuel-to-cladding gap conductivity model in PRIME acceptable.

3.1.4 Fuel Thermal Expansion

Section 5.1 of NEDC-33256P describes the fuel thermal expansion model including the additional thermal strain resulting from the phase change volumetric increase for those regions of the pellet experiencing temperature greater than the melting temperature. PNNL's technical assessment of the fuel thermal expansion model is provided in Section 2.4 of the TER. Based upon comparison to the latest version of FRAPCON-3, PNNL concluded that the fuel thermal expansion model, while acceptable below melt conditions, under predicts phase change volumetric increase. In response to RAI 33 (Reference 2) regarding PRIME's future application, GNF stated that PRIME would not be used to assess cladding strain during fuel melt conditions. Based upon NRC staff review of this assessment, the NRC staff finds the fuel thermal expansion model in PRIME acceptable for fuel temperature below the fuel melting point (including the effects of burnup and gadolinia).

3.1.5 Cladding Thermal Expansion

Section 5.1 of NEDC-33256P describes the cladding thermal expansion model. PNNL's technical assessment of this model is provided in Section 2.5 of the TER. Based upon FRAPCON-3 confirmatory analyses and comparisons to relevant empirical data, PNNL concluded that the PRIME cladding thermal expansion model is acceptable up to [[ ]] Based upon NRC staff review of this assessment, the NRC staff finds the cladding thermal expansion model in PRIME acceptable up to [[ ]]

3.1.6 Fuel Relocation

Section 5.5 of NEDC-33256P describes the fuel pellet relocation model. PNNL's technical assessment of this model is provided in Section 2.6 of the TER. FRAPCON-3 predictions of gap closure and relocation recovery (prior to hard contact) were compared to those in PRIME. PNNL concluded that the [[ ]] and therefore acceptable. Based upon NRC staff review of this assessment, the NRC staff finds the fuel relocation model in PRIME acceptable.

3.1.7 Cladding Thermal Conductivity

As identified in Section 3.1 of NEDC-33256P and discussed in Section 2.7 of the TER, PRIME contains several options for determining cladding surface temperature. The option which calculates the cladding surface temperature based upon coolant temperature and the thermal resistances through the [[ ]] should be used in all licensing analyses.

PNNL's technical assessment of the cladding thermal conductivity (both Zircaloy and pure zirconium barrier) is provided in Section 2.7 of the TER. Based upon FRAPCON-3 confirmatory analyses and comparisons to relevant empirical data, PNNL concluded that the PRIME cladding thermal conductivity is acceptable over the range [[ ]]. Based upon NRC staff review of this assessment, the NRC staff finds the cladding thermal conductivity model in PRIME acceptable over the range [[ ]].

3.1.8 Zirconium Dioxide (ZrO_2) Thermal Conductivity

The PRIME TRs specify [[ ]] In response to RAI 6 (Reference 2), GNF specified that [[ ]]. PNNL's technical assessment of this thermal conductivity is provided in Section 2.7 of the TER. PNNL concludes that [[ ]] Based upon NRC staff review of this assessment, the NRC staff finds the ZrO_2 thermal conductivity acceptable.

3.1.9 Crud Thermal Conductivity

Section 3.1 of NEDC-33256P describes the crud layer thermal conductivity model. See Section 3.3 of this SE for resolution of this item.

3.1.10 Integral Temperature Assessment

Section 2 of NEDC-33257P provides a comparison of PRIME predicted fuel temperature to measured fuel temperature over a wide empirical database. PNNL's technical assessment of the fuel temperature qualification is provided in Section 2.8 of the TER. PNNL requested further validation against specific measured data from Halden instrumented fuel assemblies (IFA). These comparisons resolved an earlier concern regarding fuel thermal conductivity. Figure 2.8.1 of the TER illustrates the extent of the original PRIME fuel temperature qualification database and the additional comparisons requested by PNNL. In addition, PNNL performed comparison calculations with FRAPCON-3. Based upon comparisons to measured centerline temperatures and FRAPCON-3 predictions, PNNL concluded that the overall prediction of fuel temperature in PRIME is acceptable for UO_2 and UO_2-Gd_2O_3 fuel pellets up to a peak pellet burnup of [[ ]] Based upon NRC staff review of this assessment, the NRC staff finds PRIME's overall fuel temperature solution acceptable.

3.2 Fission Gas Release (FGR) Model

Section 8 of NEDC-33256P describes the analytical techniques employed within PRIME to predict FGR from the pellet to the fuel rod free volume. The qualification of these thermal models against empirical data is provided in NEDC-33257P.

3.2.1 Fuel Grain Growth

Section 3.3.4 of NEDC-33256P describes the fuel grain growth model within PRIME. The grain growth model [[ ]]. PNNL's technical assessment of the grain growth model is provided in Section 3.2 of the TER. Based upon comparisons against the Khorushii grain growth model and against empirical data (RAI 5, Reference 2), PNNL concluded that the grain growth model was acceptable given the empirical nature of the FGR model. Based upon NRC staff review of this assessment, the NRC staff finds grain growth model in PRIME acceptable.

3.2.2 Helium Generation and Release

Section 8.2 of NEDC-33256P describes the helium generation and release model within PRIME. The helium generation and release model [[ ]] PNNL's technical assessment of the helium generation and release model is provided in Section 3.3 of the TER. Based upon comparison against empirical data, PNNL concluded that the helium generation and release model was acceptable. Based upon NRC staff review of this assessment, the NRC staff finds the helium generation and release model in PRIME acceptable.

3.2.3 FGR Model and Assessment

The FGR model is comprised of three terms: [[ ]] PNNL's technical assessment of the FGR model is provided in Sections 3.1 and 3.4 of the TER. Based upon independent calculations, PNNL concluded that the PRIME model predicts [[ ]]

The qualification database consists of rod puncture data of [[ ]] fuel rods from commercial and test reactors ([[ ]] more rods than the original GSTRM qualification database). Based upon comparison of PRIME predictions to this database and FRAPCON comparison calculations, PNNL concluded that the fission gas model is acceptable for steady-state and transient FGR up to a rod average burnup of [[ ]] for both UO2 and UO_2-Gd_2O_3 fuel (up to [[ ]] gadolinia). Based upon NRC staff review of this assessment, the NRC staff finds the FGR model in PRIME acceptable up to these limitations.

3.3 Cladding Corrosion and Crud Deposition Models

Section 3.1 of NEDC-33256P describes the analytical techniques employed within PRIME to predict heat transfer across cladding oxide and crud layers.

3.3.1 Cladding Corrosion

NEDC-33256P [[ ]] PNNL's technical assessment of the treatment of cladding corrosion is provided in Section 4.1 of the TER. GNF provided more detail regarding the treatment of corrosion in response to RAI 34 (Reference 2). In their response, GNF stated, "[[ ]]." The NRC staff accepts the [[ ]]

In Figure 3-1 of NEDC-33258P, [[ ]]. In addition to the GNF oxide thickness database, PNNL compared the PRIME best fit and upper 95% bounding line against corrosion data from different fuel vendors. Based upon these comparisons, PNNL concludes that the use [[ ]] was acceptable for PRIME licensing calculations.

For each fuel rod design, [[ ]] The fuel T-M analyses should consider all potential effects of an oxide layer up to the design oxide limit. The corrosion model depicted in Figure 3-1 of NEDC-33258P provides [[ ]]

3.3.2 Cladding Hydrogen Uptake

PRIME [[ ]]

3.3.3 Crud Deposition

In response to RAI 34 (Reference 2), GNF provided more detail regarding the treatment of crud during normal and abnormal corrosion events. The thermal resistance of the cladding oxide layer is [[ ]] The equation (Eqn.) numbers provided refer to corresponding equations in NEDC-33256P (Reference 1).

Option #1: [[ ]]

Where, ΔT_{film} Eqn. 3-4, ΔT_{crud} Eqn. 3-5, ΔT_{oxide} Eqn. 3-6

Option #2: [[ ]]

Where, ΔT_{film} Eqn. 3-4, $\Delta T'_{crud}$ Eqn. 3-10, ΔT_{oxide} Eqn. 3-6

In response to RAI 34 (Reference 2), GNF stated that the normal “soft” and “fluffy” crud [[ ]]

The problem with the standard approach, Option #2, is [[ ]]

The deposition rate of crud on fuel rods depends on the concentration of source material in the reactor coolant system (RCS) (e.g., Fe_2O_3 from piping corrosion) and RCS water chemistry - both of which are plant-specific and potentially cycle-specific. SRP Section 4.2 states that both oxide and crud need to be accounted for in fuel rod T-M design analyses and in calculating inputs to downstream safety analyses (e.g., stored heat for loss-of-coolant accident (LOCA) analyses). To resolve NRC staff concerns, the following analytical process must be followed:

1. To properly account for the thermal resistance of cladding corrosion and crud deposits, [[ ]] The overall thermal resistance should not be underestimated.

    a. Treatment of ZrO_2 layer [[ ]]

        1) This term accounts for both [[ ]].

        2) The [[ ]] of cladding oxidation depicted in Figure 3-1 of NEDC-33258P shall be used for plants not experiencing abnormal cladding oxidation or crud deposition. The oxide thermal conductivity should be set at [[ ]]

        3) For plants operating [[ ]] must be verified.

            - For plants experiencing abnormal cladding oxidation or crud deposition: (1) the Figure 3-1 oxide model must be adjusted to account for potential thermal feedback effects on oxide growth, and (2) the oxide thermal conductivity should be decreased to account for a potentially larger contribution of tenacious crud. An appropriate weighted conductivity should be used based upon the relative thicknesses of oxide and tenacious crud. Unless further data is available to justify specific conductivities for the corrosion/crud layer, an oxide thermal conductivity of [[ ]] and a crud thermal conductivity of [[ ]] should be used to calculate the weighted value.

b. Treatment of loose, fluffy crud deposits [[ ]]

The [[ ]], and thermal conductivity of loose, fluffy crud deposits should be selected based on plant operating experience. The temperature drop across the fluffy crud should not be underestimated.

c. Treatment of heat transfer across liquid film:

The NRC staff finds the film temperature drop calculation in PRIME (Eqn. 3-3 and Eqn. 3-4 of NEDC-33256P) acceptable.

d. Uncertainty in cladding oxide thickness and crud deposits should be applied in accordance with approved statistical and worst case methods.

3.4 Fuel Densification and Swelling Model

Section 5.3 of NEDC-33256P describes the analytical techniques employed within PRIME to predict fuel irradiation swelling. Section 5.4 of NEDC-33256P describes the fuel densification model. PNNL's technical assessment of the fuel densification and swelling models is provided in Section 5.0 of the TER. Comparison between PRIME and FRAPCON-3 [[ ]] However, these comparisons also showed that the [[ ]] In response to RAI 10 (Reference 2) regarding qualification of the fuel swelling model, GNF provided a comparison of their model to measured data and noted good agreement between PRIME and cladding profilometry data in Section 3 of NEDC-33257P. Based upon the FRAPCON-3 analyses and comparison with the empirical database, PNNL concluded that the fuel densification and swelling models are acceptable. Based upon NRC staff review of this assessment, the NRC staff finds these models in PRIME acceptable.

3.5 Cladding Material and Mechanical Properties

3.5.1 <u>Creep</u>

Section 5.6 of NEDC-33256P describes the cladding creep model. Portions of this model are derived from experimental measurements. During their review (documented in Section 6.2.1 of the TER), PNNL identified a discrepancy in the use of the experimental data to tune the creep model. In response to RAI 42 (Reference 4) regarding a potential discrepancy, GNF provided parameters for an updated creep model.

Section 3 of NEDC-33257P describes the qualification of PRIME's creep model. In RAI 21 (Reference 2), PNNL requested that GNF provide further qualification against in-reactor data (and separate specific data sets). In addition, FRAPCON-3 comparison calculations were compared to PRIME creep model predictions. Based upon comparisons to FRAPCON-3 and against empirical data, PNNL concluded that the PRIME creep model is acceptable [[ ]]. Based upon NRC staff review of this assessment, the NRC staff finds the cladding irradiation creep model in PRIME acceptable.

3.5.2 Young's Modulus and Poisson's Ratio

Section 4.1 of NEDC-33256P describes the cladding elastic and plastic properties. PRIME's correlations were derived analytically from X-ray texture measurements. PNNL's technical assessment of Young's modulus and Poisson's ratio is provided in Section 6.2.2 of the TER. Based upon comparisons to FRAPCON-3 and published data, PNNL concluded that the model for cladding elastic (Young's) modulus in PRIME is acceptable within the [[ ]]. Based upon comparison to FRAPCON-3, PNNL concluded that the model for Poisson's ratio in PRIME is acceptable within the [[ ]] Based upon NRC staff review of this assessment, the NRC staff finds these models in PRIME acceptable.

3.5.3 Yield Strength

Section 4.1.3 of NEDC-33256P describes the yield strength correlation in PRIME. Section 4.2 of NEDC-33256P describes a model for annealing of irradiation hardening. PNNL's technical assessment of these models is provided in Section 6.2.3 of the TER. During their review, PNNL identified a discrepancy in the model for cold-worked and stress-relieved (CWSR) Zircaloy that was addressed in response to RAI 7 (Reference 2). Based upon comparisons to FRAPCON-3 and data provided in the RAI 7 response (Reference 2), PNNL concluded that the yield strength and annealing models in PRIME are acceptable for [[ ]]. Based upon NRC staff review of this assessment, the NRC staff finds these models in PRIME acceptable.

3.6 Fuel Rod Cladding Deformation During Power Ramps

Section 3 of NEDC-33257P describes the qualification of PRIME's ability to predict cladding diametral and axial strains during normal operation and under power ramp conditions. PNNL's technical assessment of these models is provided in Section 6.3 of the TER. In order to evaluate each model's capability, RAI 21 (Reference 2) requested that GNF provide separate plots of cladding creepdown and power ramp strains. Based upon a comparison to the data provided in the RAI response, PNNL concluded that PRIME [[ ]] Based upon NRC staff review of this assessment, the NRC staff finds PRIME's ability to predict cladding diametral and axial strain during power ramps acceptable.

3.7 Fuel Rod Void Volume Model

Section 9 of NEDC-33256P describes the analytical techniques employed within PRIME to calculate fuel rod void volume and internal gas pressure. The qualification of these models against empirical data is provided in NEDC-33257P.

The fuel rod void volume consists of the [[ ]] PNNL's technical assessment of the void volume calculation is provided in Section 7.0 of the TER. During their review, PNNL requested further information regarding the stacking factor in RAI 18 (Reference 2) and its qualification database in RAI 44 (Reference 4). Based upon comparisons to the data provided in these RAI responses, PNNL concluded that the stacking factor was acceptable. Based upon NRC staff review of this assessment, the NRC staff finds the void volume model and stacking factor acceptable.

3.7.1 <u>Rod Growth</u>

Section 5.2 of NEDC-33256P describes the cladding irradiation growth model. PNNL's technical assessment of the rod growth model is provided in Section 7.1 of the TER. Based upon the comparison of growth predictions between FRAPCON-3 and PRIME, PNNL requested further justification for this model in RAI 9 (Reference 2). By comparison with measured data, PNNL concluded that the PRIME irradiation growth model [[ ]] Based upon NRC staff review of this assessment, the NRC staff finds the cladding irradiation growth model in PRIME acceptable for [[ ]]

3.7.2 <u>Plenum Gas Temperature</u>

[[ ]] PNNL requested that further information be provided describing the selection of this code input. Based upon their review of GNF's response to RAI 41 (Reference 2), PNNL concluded that the bounding plenum gas temperature of [[ ]] was acceptable for future licensing calculations on full length fuel rods that include a [[ ]] Part-length fuel rods would have a higher plenum gas temperature due to the location of the plenum further down the core in a region of higher gamma heating and the top of the fuel column being in a region of higher power. These effects are specifically addressed for each part-length rod design using the same methodology used to calculate the [[ ]] value. In addition, design features (such as, [[ ]] new plenum spring design, change of elevation, etc.) that may impact plenum gas temperature are also addressed using the same methodology that was used to calculate the [[ ]] value. Based upon NRC staff review of this assessment, the NRC staff finds the methodology for the selection of plenum gas temperature acceptable.

3.7.3 <u>Void Volume and Rod Internal Pressure Assessment</u>

The void volume and rod internal pressure calculations in PRIME are assessed by comparing these PRIME predictions to end-of-life pressure measurements. The qualification of PRIME's calculations is detailed in Section 5 of NEDC-33257P. During their review, PNNL requested further qualification by comparison with data from commercial fuel rods (shown in Figures 7.0.1 and 7.0.2 of the TER). Based upon this comparison with pressure measurements, PNNL concluded that the void volume calculations in PRIME were acceptable. Based upon NRC staff review of this assessment, the NRC staff finds these models in PRIME acceptable.

3.8 Licensing Application Methodology

NEDC-33258P presents a description of the application methodology for the PRIME code in licensing and design applications. As described above, the NRC's review was focused on ensuring that PRIME's algorithms accurately predict, on a best-estimate or conservative basis, the material and mechanical behavior of fuel rods in-reactor during normal and upset conditions and that the qualification database supports its targeted range of applicability. The application methodology defines how rod power history, modeling uncertainties, and manufacturing tolerances are applied in the design and licensing analyses required to demonstrate compliance with regulatory requirements at a high confidence level.

As stated in Section 1.1 of NEDC-33258P, [[ ]]

3.8.1 Cladding Liftoff Analysis (Rod Internal Pressure)

As listed in Table 2-1 of NEDC-33258P, GNF's design criteria is that the cladding creepout rate, due to fuel rod internal pressure, shall not exceed the fuel pellet irradiation swelling rate. This design requirement, commonly referred to as no clad liftoff (NCLO), is consistent with Section 4.2 of the SRP and therefore acceptable.

Section 3.4.1 of NEDC-33258P describes cladding liftoff analysis procedures. The statistical methodology for assessing manufacturing tolerances and operating conditions, described in Section 3.2.4 of NEDC-33258P, [[ ]]. Section 3.2.4 of NEDC-33258P describes the application of model uncertainties in the statistical analysis.

For the licensing analyses, GNF assumes that a [[ ]] PNNL's technical assessment of the fuel rod power history is provided in Section 8.7 of the TER.

PNNL's technical assessment of the cladding liftoff analysis application methodology is provided in Section 8.1 of the TER. PNNL concluded that the determination of partial derivatives of pressure variation with respect to fabrication and model uncertainties and the statistical error propagation was acceptable and yielded a 95/95 probability/confidence estimate of rod internal pressure.

Section 8.6 of the TER documents PNNL's assessment of the application of uncertainties in licensing calculations. PNNL concluded that the application of fabrication tolerances and operating conditions in the cladding liftoff analysis is conservative. Based upon comparisons to empirical data, PNNL concluded that the stated [[ ]] were sufficient to bound the spread in the empirical database and produce high confidence predictions.

PNNL's assessment of the critical pressure calculation (and associated uncertainty) is documented in Section 8.1 of the TER. In response to RAI 32 (Reference 2), GNF provided an

example critical pressure calculation and justification for their modeling uncertainties. PNNL was unwilling to accept the approach described by GNF in their submittal due to an inconsistent use of the thin wall and thick wall formulae throughout the derivation of both the creep model and the formula for critical pressure. PNNL also had concerns that the proposed uncertainties used were not sufficient to bound the data. In addition, PNNL and NRC staff believed the swelling rate used for determining nominal critical pressure was much lower than that measured in Halden reactor tests and the uncertainty in creep was too small based on PRIME creep model comparisons to data. After several iterations between PNNL, NRC staff, and GNF, the creep equation and the equation for critical pressure were reformulated by GNF and documented in RAI 42 (Reference 4). After reviewing the reformulated critical pressure calculation, PNNL still had concerns with the method for selecting key inputs to this calculation. GNF provided further justification on the selection of inputs (e.g., fast neutron flux) to these equations.
FRAPCON-3 comparison calculations were completed using the information provided by GNF in response to RAI 38 (Reference 3). The comparison calculations demonstrate that PRIME [[ ]]

Based on a consistent use of the thick wall formula throughout the derivations of the creep equation and the critical pressure formula, and the use of the uncertainties discussed in the above mentioned RAI responses, PNNL concluded that the PRIME code is acceptable for application to fuel rod pressure analyses. PNNL also concluded that the GNF methodology for calculating rod internal pressure limit and combining uncertainties to determine the margin to the rod internal pressure limit is acceptable. Based upon NRC staff review of this assessment, the NRC staff finds the rod internal pressure application methodology acceptable.

3.8.2 <u>Fuel Melt Analysis (Transient Thermal Overpower)</u>

As listed in Table 2-1 of NEDC-33258P, GNF's design criteria is that the maximum fuel centerline temperature shall remain below the fuel melting point. This design requirement is consistent with Section 4.2 of the SRP and therefore acceptable.

Section 3.4.2 of NEDC-33258P describes fuel melt analysis procedures. PNNL's technical assessment of the fuel melt analysis application methodology is provided in Section 8.2 of the TER. [[ ]] Section 8.6 of the TER documents PNNL's assessment of the application of these uncertainties in the fuel melt analysis licensing calculations.

FRAPCON-3 comparison calculations were completed using the information provided by GNF in response to RAI 38 (Reference 3). The comparison calculations demonstrated that the [[ ]]

Based upon the application of uncertainties and the FRAPCON-3 comparison, PNNL concluded that the PRIME model and combination of uncertainties for fuel melt analyses are acceptable. Based upon NRC staff review of this assessment, the NRC staff finds the fuel melt application methodology acceptable.

3.8.3 <u>Cladding Strain (Transient Mechanical Overpower)</u>

As listed in Table 2-1 of NEDC-33258P, GNF's design criteria is that the cladding circumferential plastic strain during an overpower transient shall not exceed 1.0%. The capability of the fuel rod cladding to withstand circumferential strain during an overpower AOO is strongly influenced by the fuel design and the cladding alloy. As such, the design strain limit is considered design and/or cladding alloy specific. It is important to note that the 1.0% permanent cladding strain criterion [[ ]]

Section 3.4.3 of NEDC-33258P describes cladding strain analysis procedures. The section title includes the text [[ ]] Similar to the [[ ]], the [[ ]] subset of AOOs has been challenged by the NRC staff in recent fuel design reviews. The PRIME application methodology needs to demonstrate that cladding strain criteria are never exceeded during all AOOs, not just the selected subset.

PNNL's technical assessment of the cladding strain analysis application methodology is provided in Section 8.3 of the TER. Section 8.6 of the TER documents PNNL's assessment of the application of uncertainties in the cladding strain licensing calculations. In response to RAI 36 (Reference 2) regarding [[ ]] GNF provided a comparison of PRIME predictions at the [[ ]] against measured strains which [[ ]]. In addition, PNNL performed comparison calculations of cladding strains at several different overpower conditions which showed that PRIME predicts [[ ]] Based upon the comparisons to power ramp data and FRAPCON-3 analyses, PNNL concluded that the application methodology was acceptable. Based upon NRC staff review of this assessment, the NRC staff finds these methods acceptable.

[[ ]] The NRC staff has developed a requirement (See Section 4) for periodic assessment of manufacturing tolerances and confirmation against power ramp data.

3.8.4 Cladding Fatigue

As listed in Table 2-1 of NEDC-33258P, GNF's design criteria is that the fuel rod cladding fatigue life usage shall not exceed the material fatigue capability. This design requirement is consistent with Section 4.2 of the SRP and therefore acceptable.

Section 3.4.4 of NEDC-33258P describes cladding fatigue analysis procedures. PNNL's technical assessment of the cladding fatigue analysis application methodology is provided in Section 8.4 of the TER. In response to RAI 40 (Reference 2) regarding the fatigue analysis, GNF provided the basis of their Zircaloy fatigue curve and provided further detail on the rain flow fatigue cycle. Based on the information presented in NEDC-33258P and in response to RAI 40, PNNL concluded that the cladding fatigue application methodology was acceptable. Based upon NRC staff review of this assessment, the NRC staff finds these methods acceptable.

3.8.5 T-M Inputs to Downstream Analyses

Section 3.4.5 of NEDC-33258P states that PRIME will replace the GESTR-LOCA and GSTRM codes in analyses performed to generate inputs for other analyses, including LOCA, core transient, and stability analyses. PNNL's technical assessment of the downstream application methodology is provided in Section 8.5 of the TER. In response to RAI 39 (Reference 2) regarding the use of PRIME models and the continued use of GSTRM models during an interim period, GNF provided details of the impacts of PRIME and their plans to update downstream methods. See Appendix A for the NRC staff's evaluation.

3.9 Range of Applicability

Table 2.1 of NEDC-33256P specifies the range of applicability for various dimensional and performance parameters. The range of applicability of individual fuel performance models is governed by the extent of the qualification database. As part of its review of the calibration and validation of individual fuel performance models, PNNL assessed the range of applicability. Differences relative to Table 2.1 of NEDC-33256P are identified below.

3.9.1 Pellet Inner Diameter (Annular Pellets)

Table 2.1 of NEDC-33256P specifies a pellet inner diameter range of [[ ]] of pellet outer diameter (OD). Based upon pellet stability and operating experience, this manufacturing parameter is limited to [[ ]] of pellet OD.

3.9.2 Pellet Length-to–Diameter (L/D) Ratio

Table 2.1 of NEDC-33256P specifies a pellet L/D ratio of [[ ]] The NRC staff interprets this specification to be less than [[ ]]

3.9.3 Pellet Enrichment

Table 2.1 of NEDC-33256P specifies a pellet enrichment range of [[ ]]. Since commercial enrichment facilities are limited to 5.0 wt% U^{235}, the availability of irradiated data on commercial fuel rods beyond this limit to validate fuel performance models is minimal. As such, the range of applicability for PRIME is [[ ]]

3.9.4 Pellet Density

Table 2.1 of NEDC-33256P specifies a pellet density range of [[ ]] theoretical density (TD). Based upon the qualification database and manufacturing specifications, this range was [[ ]]

3.9.5 Peak Linear Power

Table 2.1 of NEDC-33256P specifies an [[ ]] PNNL's technical assessment of the fuel rod power envelope is provided in Section 8.7 of the TER. Based upon an assessment of the qualification database provided in response to RAI 1e (Reference 2), PNNL proposed a [[ ]]

3.9.6 Peak Pellet Exposure

Table 2.1 of NEDC-33256P specifies a peak pellet exposure range [[ ]] Based upon its respective qualification database, each individual fuel performance model may have a unique limit on peak pellet exposure. Limits on PRIME's qualification database are identified below:

[[

•

•

•

•

•

•

]]

Based primarily on lack of [[ ]] PNNL recommended approval of PRIME to [[ ]] Based upon NRC staff review of PNNL's assessment, the NRC staff finds the [[ ]]

3.9.7 Fuel Temperature

Table 2.1 of NEDC-33256P specifies an [[ ]] In Section 3.1, the NRC staff determined that [[ ]]

3.9.8 Cladding Temperature

Table 2.1 of NEDC-33256P specifies a cladding temperature range of [[ ]] Based upon comparisons to empirical data and FRAPCON-3 comparison calculations, PNNL concluded that several models were limited to cladding temperatures below the upper limit in Table 2.1 of NEDC-33256P. Based on NRC staff review of these assessments, approval for PRIME will be limited to [[ ]]

3.9.9 Fuel Grain Size

Table 2.1 of NEDC-33256P specifies a grain size range of [[ ]]. Based on the extent of the qualification database, the range of [[ ]]

3.9.10 Fuel Pellet Additive Concentration, Weight Percentage

Table 2.1 of NEDC-33256P specifies an additive concentration range of [[ ]] However, no data comparisons were provided to justify PRIME models for additive fuel, such as thermal conductivity. In response to RAI 24 (Reference 2) regarding the licensing of PRIME for this application, GNF withdrew its request for NRC staff approval of PRIME for additive fuel. As such, approval for PRIME [[ ]]

Although not specified in Table 2.1 of NEDC-33256P, the range of applicability must be further limited to [[ ]]

4.0 LIMITATIONS AND CONDITIONS

Licensees referencing the PRIME fuel rod T-M performance model license TRs (NEDC-33256P, NEDC-33257P, and NEDC-33258P) must ensure compliance with the following limitations and conditions:

1. The PRIME fuel rod T-M model and application methodology are approved and applicable within the range of parameters specified in Table 2.1 of NEDC-33256P as amended by Section 3.9 of this report. An additional limitation on the applicability of PRIME is listed below.

    a. Applicability is limited to approved [[

    ]] (Section 3.9)

2. To properly account for the thermal resistance of cladding corrosion and crud deposits, set [[

    ]] Licensees should be careful to ensure that the overall thermal resistance is not underestimated. (Section 3.3.3)

    a. Treatment of ZrO_2 layer [[ ]]

        1) This term accounts for both [[ ]]

        2) The [[

        ]]

        3) [[

        ]]

            - For plants experiencing abnormal cladding oxidation or crud deposition: (1) the Figure 3-1 oxide model must be adjusted to account for potential thermal feedback effects on oxide growth, and (2) the oxide thermal conductivity should be decreased to account for a potentially larger contribution of tenacious crud. An appropriate weighted conductivity should be used based upon the relative thicknesses of oxide and tenacious crud. Unless further data is available to justify specific conductivities for the corrosion/crud layer, an oxide thermal conductivity of [[ ]] and a crud thermal conductivity of [[ ]] should be used to calculate the weighted value.

b. Treatment of loose, fluffy crud [[ ]]

[[ ]]

The temperature drop across the fluffy crud should not be underestimated.

c. Treatment of heat transfer across liquid film:

The NRC finds the film temperature drop calculation in PRIME (Eqn. 3-3 and Eqn. 3-4 of NEDC-33256P) acceptable.

d. Uncertainty in cladding oxide thickness and crud deposits should be applied in accordance with approved statistical and worst case methods.

3. Due to the empirical nature of the PRIME calibration and validation processes, the specific values of equation constants and tuning parameters derived in NEDC-33256P (as updated by the RAI responses submitted as part of this review) become inherently part of the approved models. Thus, these values may not be updated without necessitating further NRC review.

   a. Computer code revisions, necessitated by errors discovered in the source code, needed to return the algorithms to those described in NEDC-33256P (as updated by RAIs) are acceptable.

   b. Changes in the numerical methods to improve code convergence or speed of convergence, or transfer of the methodology to a different computing platform to facilitate utilization, would not be considered to constitute a departure from a method of evaluation in the safety analysis. Such changes may be used in PRIME licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B. Features that support effective code input/output would not be considered by the NRC staff to constitute a departure from a method of evaluation in the safety analysis and such changes may be made without NRC staff review and approval.

4. PRIME models have been calibrated and validated by direct comparison to the existing empirical database. Further, model uncertainties described within the application methodology were derived by direct comparison of model predictions to the existing empirical database. To ensure PRIME's best-estimate predictions and applied uncertainties remain valid, GNF must demonstrate and document, in a letter addressed to the Director, Division of Safety Systems, Office of Nuclear Reactor Regulation, the continued applicability of PRIME every five years starting in 2015.

   a. In preparation of this letter, GNF must review available sources for applicable commercial and research reactor fuel performance data which may augment the existing PRIME qualification database (e.g., international research activities, pool-side examinations, hot-cell programs, power ramp programs).

b. In the letter, sources for new data should be clearly identified. If no new data for a particular model (e.g., FGR model) has been discovered, the letter should state this fact and identify which sources were investigated.

c. PRIME model predictions and uncertainties should be compared against the augmented database. New data should be easily differentiated on the plots. At a minimum, the letter should separately address the following model predictions and their respective uncertainties: (1) fuel temperature, (2) FGR, (3) fuel irradiation swelling, (4) cladding creep, (5) cladding strain (due to over power conditions), and (6) void volume/rod internal pressure.

d. Any data discarded from the augmented qualification database should be identified and dispositioned.

e. The letter should identify and disposition any bias on model predictions or increase in uncertainty.

f. Since the worst case methodology employed in the [[

]]

5. Interim Process Thermal Overpower Condition (see Appendix A, Section A.2.2.2): (This limitation will be implemented for future plant- and cycle-specific analyses):

a. TOP screening limits for GNF fuel products currently used in operating plants shall be confirmed to be conservative using the PRIME methodology, or revised to be consistent with the PRIME results.

b. If the TOP screening limit has been exceeded, detailed cycle-specific calculations (if they are required) must be performed using transient fuel performance models that are fully consistent with the approved PRIME models. Implementation of this condition will be consistent with the schedule proposed in MFN 09-466 (Reference 11).

5.0 CONCLUSION

Based upon its review of TRs NEDC-33256P, NEDC-33257P, and NEDC-33258P and technical support provided by PNNL, the NRC staff finds GNF's PRIME fuel rod T-M performance model and application methodology acceptable. Licensees referencing these TRs will need to comply with the limitations and conditions (L&Cs) listed in Section 4.

The NRC staff has completed its review of the downstream impact of the PRIME model to steady-state, transient, and accident analysis methods that comprise the GNF standard set of reload licensing methods and calculations. On the basis of its review, the NRC staff has found that GNF has adequately addressed each downstream analysis method. The NRC staff primarily reviewed the information provided in response to RAI 39 (Reference 2) to assess the downstream impact.

When the “Interim Process Thermal Overpower Condition” (L&C 5) is met, the NRC staff finds that the use of legacy transient analysis methods during the interim process is acceptable.

Based on the results of peak cladding temperature sensitivity calculations, the NRC staff found that PRIME is not expected to significantly impact the downstream ECCS-LOCA evaluations. However, the NRC staff notes that the reporting requirements of 10 CFR 50.46 are explicit. The responses to the NRC staff RAIs have confirmed that the approach to meeting the reporting requirements of 10 CFR 50.46 is acceptable. Therefore, the NRC staff finds that use of legacy accident methods during the interim process has been adequately addressed by GNF’s RAI responses.

6.0 REFERENCES

1. Letter from GNF to USNRC, FLN-2007-001, “GNF Licensing Topical Report, ‘The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance,’ NEDC-33256P, NEDC-33257P, and NEDC-33258P, January 2007,” dated January 19, 2007. (ADAMS Accession No. ML070250417)
2. Letter from GNF to USNRC, 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 Accession No. ML090620312)
3. Letter from GNF to USNRC, MFN 09-106 Supplement 1, “Supporting Information for Request for Additional Information Response for the PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance (TAC # MD4114),” dated February 27, 2009. (ADAMS Accession No. ML090690523)
4. Letter from GNF to USNRC, MFN 09-106 Supplement 2, “Response to Supplement to Request for Additional Information Re: Licensing Topical Reports NEDC-33256P, NEDC-33257P, and NEDC-33258P, ‘The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance (TAC No. MD4114)’,” dated August 11, 2009. (ADAMS Accession No. ML092250348)
5. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 4.2, Revision 3, “Fuel System Design,” dated March 2007. (ADAMS Accession No. ML070740002)
6. NUREG/CR-6534, Volume 1, “FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application,” dated October 1997. (ADAMS Accession No. ML092950544)
7. NUREG/CR-6534, Volume 4, “FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties,” dated May 2005. (ADAMS Accession No. ML051440720)
8. USNRC to Westinghouse, “Final Safety Evaluation for Westinghouse Topical Report WCAP-15836-P, ‘Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1’ (TAC NO. MB5740),” dated September 28, 2005. (ADAMS Accession No. ML052720151)
9. USNRC to AREVA NP, “Final Safety Evaluation Report for AREVA NP, Inc. (AREVA) Topical Report (TR) BAW-10247(P), Revision 0, ‘Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors’ (TAC NO. MC4261),” dated February 12, 2008. (ADAMS Accession No. ML080350138)
10. TR NEDE-23785-1-PA, Revision 1, Volume 1, “The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident: GESTR-LOCA – A Model for the Prediction of Fuel Rod Thermal Performance,” dated October 1984. (ADAMS Accession No. ML090780920)
11. Letter from GE-Hitachi Nuclear Energy to USNRC, MFN 09-466, “Implementation of PRIME Models and Data in Downstream Methods, NEDO-33173, Supplement 4, July 2009,” dated

July 10, 2009. (ADAMS Accession No. ML091910490)

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Date: December 24, 2009

APPENDIX A - REVIEW OF IMPACT OF PRIME ON DOWNSTREAM TRANSIENT AND ACCIDENT ANALYSES

A.1 INTRODUCTION

The purpose of Request for Additional Information (RAI) 39 (References 1 and 2) is to assist the U.S. Nuclear Regulatory Commission (NRC) staff in assessing the impact on downstream calculations performed using the General Electric Stress and Thermal Analysis of Fuel Rods (GESTR)-Mechanical (GSTRM) fuel model and GSTRM gas gap conductivity files (Reference 3) during the interim while the legacy safety analysis methods are migrated to the updated PRIME models (NEDC-33256P, NEDC-33257P, and NEDC-33258P – References 4, 5, and 6, respectively). In this interim period, the thermal-mechanical (T-M) operating limits will be determined using PRIME; however, transient safety analyses will be performed using the GSTRM inputs. The NRC staff notes that the GSTRM models do not account for the physical phenomenon of fuel pellet conductivity degradation with pellet exposure. The NRC staff refers to this process to be used during the period of time between PRIME approval and the eventual update of the legacy methods as the interim process.

During its review of TRACG04 to perform transient calculations as described in the GE-Hitachi Nuclear Energy Americas (GEH, previously GE Energy) topical report (TR) NEDE-32906P, Supplement 3 (Reference 7), the NRC staff identified a concern with utilizing the PRIME thermal conductivity model in TRACG04 with gas gap conductance files based on the GSTRM code. This concern arises because the fuel thermal time constant is a strong function of the pellet thermal conductivity and the gas gap conductance. Combining the GSTRM gas gap conductance file, noting deficiencies in the GSTRM fuel conductivity model, may have an adverse impact on the efficacy of the safety analysis codes.

Therefore, the NRC staff requested that Global Nuclear Fuel – Americas, LLC (GNF) use the TRACG04 code (with both PRIME and GSTRM consistent inputs) to assess the sensitivity of the safety analysis figures of merit. The TRACG04 code was selected to perform this sensitivity analysis in part because the code already includes a capability for utilizing the PRIME thermal conductivity model. TRACG04 was also selected because the NRC staff has reviewed various capabilities of TRACG to perform a wide variety of transient and safety analyses.

The NRC staff accepts the use of TRACG04 for the aforementioned purpose because the TRACG04 models are significantly similar to those models included in the other legacy codes [PANAC11 (Reference 8), ODYN (NEDO-24154 – Reference 9), SAFER (NEDE-23785, NEDE-30996, NEDC-32950 – References 3, 10, 11, and 12), TASC (NEDC-32084 – Reference 13), ODYSY (NEDC-32992, NEDE-33213 – References 14 and 15), and TRACG02 (NEDE-32906 – References 16 - 18)]. Therefore, the NRC staff expects that the TRACG04 code, because it has more detailed modeling capabilities (e.g., three-dimensional kinetics), will yield the most accurate assessment of the physical sensitivity of the transient and accident plant response to differences in the fuel thermal model.

The NRC staff acceptance of the usage of TRACG04 to determine the sensitivity of the relevant figures of merit does not herein constitute NRC approval of TRACG04 to perform licensing safety analyses.

The NRC staff's review considered each safety analysis. These include: anticipated operational occurrences (AOOs), overpressure transients, anticipated transients without SCRAM (ATWS), stability evaluations, and design basis accident (DBA) emergency core cooling system loss-of-coolant accident (ECCS-LOCA) analyses. For each type of analysis, the NRC staff reviewed the sensitivity of the figures of merit to determine if the interim process results in non-conservatism in the safety analysis results.

A.2 TRANSIENTS

Transients refer to those analyses performed to assess the impact of AOOs as well as analyses performed to demonstrate compliance with overpressure criteria, namely American Society of Mechanical Engineers (ASME) Overpressure and ATWS Overpressure. ATWS Overpressure refers to a specific transient analysis where SCRAM is not modeled; however, the transient analysis is performed for the period of time prior to boration.

A.2.1 Critical Power Criterion

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC-10 "Reactor Design," requires that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of AOOs. To demonstrate compliance with GDC 10, critical power ratio (CPR) safety and operating limits are established to preclude fuel cladding failure as a result of boiling transition.

Transient calculations are performed in safety analyses to demonstrate margin to boiling transition. For these calculations the figure of merit is the relative change in CPR (ΔCPR/ICPR). The direct comparison of the boiling water reactor (BWR)/4 turbine trip without bypass (TTNB) AOO indicates that the predictions of ΔCPR/ICPR using GSTRM and PRIME models are essentially identical. The GSTRM result is mildly higher (conservative) relative to the PRIME result. This trend is consistent with the NRC staff's expectation based on its review of NEDE-32906P, Supplement 3 (Reference 19).

Therefore, the NRC staff finds that the use of the GSTRM models in the legacy methods will not adversely affect licensing calculations to demonstrate margin to boiling transition.

A.2.2 Thermal-Mechanical Criteria

GDC 10 requires that SAFDLs are not exceeded during any condition of normal operation. To demonstrate compliance with GDC 10, fuel rod T-M design limits are established to ensure fuel rod integrity in its core lifetime along the licensed power/flow domain, during normal steady-state operation and in the event of an AOO. The T-M acceptance criteria for new fuel product lines are specified in Amendment 22 to the NRC-approved GNF licensing methodology GESTAR II. The linear heat generation rate (LHGR) limit is an exposure-dependent limit placed on the peak pin power that ensures the integrity of the fuel cladding during normal steady-state operation and limits the initial heat generation rate during transient thermal and mechanical overpower conditions. Internal rod pressures during steady-state conditions, maximum fuel temperature, and cladding strain during transients (AOOs) all affect fuel integrity. The fuel T-M design criteria (consistent with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2 – Reference 20) requires, in part, that:

1. Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading.

   The fuel rod internal pressure is limited so that the cladding creep out rate due to internal gas pressure during normal operation will not exceed the instantaneous fuel pellet cladding irradiation swelling rate. In establishing the LHGR limit, at each point of the exposure dependent envelope, the fuel rod internal pressure required to cause the cladding to creep outward at a rate equal to the pellet irradiation swelling is determined.

   The calculated internal rod pressures along the LHGR envelope are statistically treated so that there is assurance with 95 percent confidence that the fuel rod cladding creep rate will not exceed the pellet irradiation swelling rate.

2. Loss of fuel rod mechanical integrity will not occur due to fuel melting.

   The fuel rod is evaluated to ensure that fuel melting will not occur during normal operation and core–wide AOOs. For every fuel product line, the thermal overpower (TOP) limit is established to preclude fuel centerline melting. The acceptable thermal overpower [[

   ]]

3. Loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction.

   The fuel rod is evaluated to ensure that the calculated cladding circumferential plastic strain due to pellet–cladding mechanical interaction does not exceed 1 percent. For every fuel product line, the mechanical overpower (MOP) limit is established to preclude one percent cladding diametric strain during AOOs. The acceptable MOP limit [[

   ]]

A.2.2.1 Clad Liftoff Criterion (Item 1)

No-clad-liftoff (NCLO) is demonstrated using PRIME in a standalone fashion. Therefore, consideration of the NCLO criterion for transient applications is not required.

A.2.2.2 Fuel Centerline Temperature Criterion (Item 2)

The response to RAI 39 (Reference 1) indicates that the use of GSTRM models in the legacy codes may result in the [[

]]

On the basis of the RAI 39 response, the NRC staff cannot conclude that the detailed analyses using legacy methods are conservative with reasonable assurance.

The NRC staff imposes a condition for the interim process:

<u>Interim Process Thermal Overpower Condition</u>

(This limitation will be implemented for future plant- and cycle-specific analyses):

(1) TOP screening limits for GNF fuel products currently used in operating plants shall be confirmed to be conservative using the PRIME methodology, or revised to be consistent with the PRIME results.

(2) If the TOP screening limit has been exceeded, detailed cycle-specific calculations (if they are required) must be performed using transient fuel performance models that are fully consistent with the approved PRIME models. Implementation of this condition will be consistent with the schedule proposed in MFN 09-466 (Reference 21).

When using the generic TOP limits, the figure of merit from the transient calculation is the transient change in LHGR predicted by the systems analysis code. This code may be either ODYN or TRACG. In its review of the TRACG04 methodology for transients (NEDE-32906P, Supplement 3 – Reference 19), the NRC staff found that the use of GSTRM thermal conductivity is conservative for this purpose. This is because the transient LHGR will be over-predicted because the higher GSTRM thermal conductivity will reduce the fuel thermal time constant and result in higher calculated transient cladding heat flux. Additionally, the GSTRM model will result in conservative Doppler worth calculations.

These trends are independent of the analytical code; therefore, the same arguments are applicable to ODYN and TRACG02. On this basis, the NRC staff finds that the use of generic PRIME TOP limits is acceptable when the transient LHGR is calculated using the legacy methods during the interim process.

In terms of meeting the condition for detailed cycle-specific calculations, the NRC staff understands that several approaches may be employed that are acceptable. For example, TRACG04 may be used as it is an approved transient analysis code that includes the PRIME thermal conductivity model and may accept gas gap conductance input from PRIME. NEDO-33173, Supplement 4 (Reference 21) describes the process that GNF will employ in upgrading the other downstream codes to incorporate PRIME T-M models. Therefore, several alternative analyses may be employed to satisfy the Interim Process Thermal Overpower Conditions once this upgrade is complete.

A.2.2.3 One Percent Plastic Strain Criterion (Item 3)

Transient calculations are performed to demonstrate margin to the one percent cladding plastic strain limit, thus ensuring mechanical overpower margin. TRACG04 does not directly output the plastic strain. [[

]] The TRACG04 results using GSTRM and PRIME models are essentially identical. Again, the NRC staff notes that the use of the GSTRM models is slightly conservative relative to using PRIME models.

Generally, compliance with the one percent plastic strain criterion is demonstrated by performing transient calculations and demonstrating margin to the generic MOP limit for a specific fuel design. The [[

]] Therefore, the NRC staff finds that GSTRM MOP limits generated for legacy fuel products are conservative.

When the generic MOP limit is not met on a cycle-specific basis, detailed transient analyses are performed. When TRACG04 is used, [[

]] On the basis of this insensitivity, the NRC staff finds that the legacy methods may be used during the interim process. For the legacy methods, the NRC staff finds that no specific thermal margin enhancement is required to address their use in demonstrating compliance with the one percent plastic strain criterion if detailed cycle-specific analyses are required.

A.2.3 Pressure Criteria

According to GDC 14 “Reactor Coolant Pressure Boundary” from 10 CFR 50, Appendix A, the reactor coolant pressure boundary must be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. To demonstrate compliance with GDC 14, transient calculations are performed to ensure that ASME pressure limits are met for the reactor vessel.

The transient calculations performed include ATWS and ASME Overpressure analyses. These calculations are very similar to pressurization transient analyses. Therefore, the NRC staff considered the predicted pressurization for the BWR/4 TTNB AOO as representative for all pressurization transients (including Overpressure) in terms of the pressure sensitivity to the fuel conductivity models and gas gap conductance files.

In the BWR/4 TTNB AOO case, TRACG04 predicts essentially identical peak pressures when using either the PRIME or GSTRM model. The NRC staff notes that the use of GSTRM appears to be slightly conservative. This is consistent with the NRC staff’s expectations based on its review of NEDE-32906P, Supplement 3 (Reference 19).

Further discussion of the Economic Simplified Boiling Water Reactor (ESBWR) ATWS event analyses is provided in a subsequent section. The NRC staff notes that the ESBWR ATWS event provides a comparison of the PRIME and GSTRM predicted peak pressures. These two predicted peak pressures are essentially identical. Therefore, when considered with the BWR/4 TTNB AOO, the NRC staff has reasonable assurance that the calculated peak pressure for transients and ATWS events are insensitive to the fuel thermal modeling.

Therefore, the NRC staff finds that the licensing calculations performed to demonstrate Overpressure margin are not sensitive to which T-M model is used in the analysis.

A.3 ANTICIPATED TRANSIENTS WITHOUT SCRAM

The ATWS acceptance criteria are specified in SRP, Section 15.8 (Reference 22) and are based on meeting the relevant requirements of the following regulations:

1. 10 CFR 50.62 (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via (a) inclusion of prescribed design features and (b) demonstration of their adequacy;

2. 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures (PCT), maximum cladding oxidation, and coolable geometry;

3. GDC 12 "Suppression of Reactor Power Oscillations," as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed;

4. GDC 14 "Reactor Coolant Pressure Boundary," as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary;

5. GDC 16 "Containment Design," as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents;

6. GDC 35 "Emergency Core Cooling," as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reaction must be limited to negligible amounts;
7. GDC 38 "Containment Heat Removal," as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment; and

8. GDC 50 "Containment Design Basis," as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

Insofar as analytical codes are used to demonstrate compliance with the regulatory criteria, calculations are performed for the limiting ATWS event(s) to: (1) determine the vessel pressurization to demonstrate compliance with GDC 14; (2) determine the suppression pool temperature to demonstrate compliance with GDC 16, GDC 38, and GDC 50; (3) determine the PCT and maximum oxidation to demonstrate compliance with 10 CFR 50.46 criteria; and (4) determine whether the core remains in a coolable geometry.

The RAI 39 response (Reference 1) provides the results of sensitivity studies for the ESBWR main steam isolation valve closure (MSIVC) ATWS event. The parameters compared are the maximum neutron flux, the vessel pressure, and the suppression pool bulk temperature. The response states that the sensitivity of the figures of merit of 10 CFR 50.46 (as considered for ATWS) are substantially similar to those sensitivities reported for the ECCS-LOCA calculations. The NRC staff agrees with this assessment and finds that, when considered with the

ECCS-LOCA comparisons, the response adequately addresses the relevant safety figures for ATWS simulations.

Generally, an ATWS event may be described in three distinct phases. In the first phase, the reactor vessel is pressurized by an initiating event (in this particular case, an MSIVC). During this first phase, the reactor power and neutron flux will pulse as the initial void collapse introduces reactivity and a combination of negative void and Doppler worth terminate the power increase. In the second phase, the reactor power stabilizes at a critical configuration that is governed by the core flow rate (natural circulation conditions). In this second phase, the core attains an adjoint-weighted average void fraction that is similar to the initial condition. The reactor power remains relatively steady during this phase (but will change with any variation in the reactor vessel level), and steam is relieved to the suppression pool. In the third phase, boron injection shuts down the reactor and brings the core to a subcritical state.

A.3.1 Power Pulse

The important phenomena that dictate the reactor behavior during the first phase are the intensity of the pressure wave impinging on the core and the void and Doppler reactivity feedback.

Compared to other transients, ATWS events tend to demonstrate a greater sensitivity to the Doppler coefficient due to greater fuel heat up when the event is not terminated with a SCRAM. The RAI 39 response (Reference 1) states that the Doppler feedback is stronger for higher initial temperatures. The NRC staff does not agree with the response in its assessment of the Doppler feedback. The NRC staff conducted a detailed review of the Doppler feedback trends with temperature during its review of NEDE-32906P, Supplement 3 (Reference 19). During this review, the NRC staff found that TRACG04 (as well as legacy codes) will incorporate nodal temperature reactivity feedback response surfaces that are generated at the PANAC11-predicted initial fuel temperature. Therefore, the Doppler coefficient itself is not treated as being sensitive to the initial temperature. The NRC staff noted in its previous review of TRACG04 (Reference 19) that as the temperature increases the magnitude of the Doppler coefficient tends to decrease.

The RAI 39 response may refer to a trend whereby increased initial temperature results in a greater temperature increase during the transient evaluation. This may be a result of increased heat holdup due to a smaller thermal conductivity of the pellet with increasing temperature. However, the dynamics of the power increase are a strong function of the core hydraulics, the void reactivity, and the fuel thermal time constant. Therefore, the NRC staff cannot conclude categorically that higher fuel temperatures result in increased Doppler feedback. The NRC staff would counter that the opposite trend is expected and merely not captured in GNF's analytical methods.

The NRC staff expects that the peak neutron flux would be sensitive to the fuel modeling parameters. The NRC staff expects that the calculated power pulse will be impacted by a combination of the void reactivity and Doppler feedback. These two reactivity effects will likely have a competing effect when the fuel thermal modeling is perturbed between the GSTRM and PRIME models. That is, the void formation that occurs after the pressurization is enhanced when the fuel thermal resistance is lower, thus contributing to a lower flux peak. However, when the fuel thermal resistance is low, the fuel temperature increase is dampened by effective heat transfer and the Doppler effect is lessened. Regardless of the relative magnitude of these two separate effects, the comparison provided in the RAI 39 response demonstrates that the peak flux predicted by either model is essentially identical.

The NRC staff considered the impact of a potential Doppler coefficient bias that is consistent with the predicted difference in average fuel temperature assuming that the Doppler coefficient scales as the square-root of the temperature. Using the values provided in the RAI 39 response, the NRC staff estimated that the temperature difference would indicate a bias in the Doppler coefficient on the order of [[ ]] provided in NEDE-32906P, Supplement 1-A (Reference 17). Figure 8-11 of NEDE-32906P, Supplement 1-A provides the peak pressure sensitivity to a Doppler coefficient variation of [[ ]] The results indicate that the potential sensitivity to the Doppler coefficient bias introduced by the error in the GSTRM temperature prediction is on the order of [[ ]] The NRC staff finds that this potential bias is negligible.

While there may be competing effects, the NRC staff finds that during the initial power pulse, the kinetics solutions remain generally insensitive to the fuel thermal models. Further evidence of this is shown by the high degree of agreement between the peak pressures calculated using either method. These also are essentially the same. Therefore, in terms of demonstrating compliance with GDC 14, the NRC staff finds that it is acceptable to utilize the legacy methods during the interim process.

A.3.2 Natural Circulation

Prediction of the containment performance during an ATWS event is particularly sensitive to the predicted core thermal power during the second phase of the event. The initial power pulse contributes only a small fraction of the total heat load that is deposited in the containment. During the second phase, the reactor is brought to a natural circulation condition by tripping the recirculation pumps for plants in the current operating fleet. In the case of the ESBWR, the reactor core remains in a natural circulation condition where emergency operating procedures dictate the evolution of the core flow. In either case, during this phase of the event, the reactor power is still significant and the steam is routed to the suppression pool. Considering the relatively long duration of this phase relative to the initial power pulse, it is the most significant contributor to the containment heat load.

The RAI 39 response (Reference 1) is correct insofar as the reactor power level is most sensitive to the core hydraulics. The power will stabilize at any given flow rate such that the adjoint-weighted void fraction is essentially the same (with some variations given the magnitude of the negative Doppler worth). Given that the void reactivity coefficient is much greater than the Doppler coefficient, the NRC staff agrees that the heat load to the suppression pool will not be sensitive to the fuel thermal modeling as this phase is dominated by void reactivity effects and only negligibly affected by the Doppler worth.

A.3.3 Boration

During the boration phase, reactor power is governed primarily by the concentration of boron delivered to the active region. This is true for operating reactors and the ESBWR. The boron worth is not sensitive to the fuel thermal modeling, and therefore, use of either model (GSTRM or PRIME) is not expected to have a significant effect on this stage of the simulation. Additionally, the fraction of the total heat deposited in the suppression pool from this phase is small compared to the heat deposited from the second phase. Therefore, the NRC staff finds that close agreement between the two calculated suppression pool temperatures is expected.

Overall, when all phases are considered, the NRC staff finds that either fuel thermal modeling methodology generates essentially identical containment temperature response. Therefore, the NRC staff finds that it is acceptable to utilize the legacy methods for ATWS containment analysis during the interim process.

A.4 STABILITY

GDC 12, "Suppression of Reactor Power Oscillations," requires that reactor systems be designed to assure that power oscillations that may result in the fuel exceeding SAFDLs are either not possible or are readily detected and suppressed. GDC 10 requires that the fuel does not exceed SAFDLs. SRP Section 4.3 (Reference 23) addresses reviews to assure the conformance with the requirements of GDC 10 and GDC 12. To demonstrate compliance with these GDC requirements, various analyses may be required.

To demonstrate that power oscillations are not possible, calculations are performed to determine the channel, core, and, in some cases, regional mode decay ratios. These calculations may be performed with TRACG04 in the case of the ESBWR or with ODYSY in the case of the channel and core decay ratios for the operating fleet.

For cases where the power oscillations are suppressed, analyses must be done to establish appropriate setpoints that ensure that these oscillations do not result in the fuel exceeding SAFDLs. In this case, detailed transient calculations are performed to assess the change in thermal margin with the oscillation magnitude. The NRC staff has approved TRACG02 for this purpose, but also understands that TRACG04 has been applied for this application on a plant-specific basis under the provisions of 10 CFR 50.59 where it was demonstrated to be conservative by comparison or essentially the same as the TRACG02 methodology.

The RAI 39 response (Reference 1) contains comparisons of the core wide growth rate for a BWR/4 and regional mode decay ratio for the ESBWR. Use of the PRIME model results in [[

]] Therefore, the results are expected.

The results confirm that the use of GSTRM models in the legacy stability codes will predict enhanced coupling relative to the PRIME models. Therefore, licensing calculations performed using the legacy codes will be conservative relative to licensing calculations using the PRIME models during the interim process. The sensitivity of the analysis is consistent with the expectations reported by the NRC staff in its safety evaluation (SE) of NEDC-32906P, Supplement 3 (Reference 19) where the NRC staff states that the use of the PRIME or GSTRM thermal conductivity models is expected to have a significant impact on the use of TRACG04 for stability analyses. Therefore, the NRC staff finds the use of legacy methods for stability calculations during the interim process to be acceptable.

A.5 ECCS-LOCA

ECCS-LOCA evaluation acceptance criteria are specified in 10 CFR 50.46. The criteria are:

(1) the PCT will not exceed 2200°F; (2) the maximum oxide thickness does not exceed 17 percent of the cladding thickness anywhere in the core; (3) the total hydrogen formed does not exceed one percent of the hypothetical amount if the entire cladding inventory (excluding plena) were reacted; (4) the core retains a coolable geometry; and (5) long-term cooling is maintained. For the operating reactor fleet, GESTR/SAFER analyses are performed to calculate the PCT, oxide thickness, and core volume oxidized.

A.5.1 Design Basis LOCA

A.5.1.1 PCT Criterion

GNF provided calculated PCTs for a BWR/4 and BWR/2. The BWR/4 case indicates that the PCT predicted using the PRIME fuel thermal model results in an insignificant increase in PCT of approximately [[ ]] relative to GSTRM. For the BWR/2 case, the difference is even smaller (approximately [[ ]]). The results are summarized in Table 1.

Table 1: TRACG04 PCT Sensitivity Study Results

	GSTRM	PRIME	PRIME-GSTRM
BWR/4 PCT (°F)	[[		
BWR/2 PCT (°F)			]]

The NRC staff notes that the predicted initial fuel temperatures using the PRIME thermal model are significantly higher in both cases (approximately [[ ]]). This will affect the stored energy in the core.

For BWR/3-6 plants, the limiting DBA LOCA break occurs in the recirculation suction line. During the LOCA event, the core flow stagnates after a few seconds due to uncovering of the jet pumps. The stagnation results in two phase separation and the formation of a liquid level in the core. The core is subject to transition boiling during the flow stagnation period. The occurrence of early transition boiling results in an early (or first) peak in the PCT. The level in the core is restored once depressurization occurs when the break uncovers. The level then drops due to mass loss through the break, uncovering the core. The second peak in PCT occurs when the core becomes uncovered. A representative trace of PCT is provided in Figure 1.

The magnitude of the first peak PCT is primarily a function of the stored energy as this dictates when the fuel enters boiling transition during the flow stagnation. The effectiveness of the depressurization to remove the energy makes the second peak PCT magnitude much more dependent on the relative capability of the ECCS to inject coolant and the amount of decay heat being generated within the core.

The analytical results provided in the sensitivity studies appear to confirm that the sensitivity of the second peak PCT to the difference in the stored energy is insignificant. Specifically, for both the BWR/2 and BWR/4 cases, the PCT differences were less than the 50°F significance threshold specified in 10 CFR 50.46.

The NRC staff compared the TRACG04 stored energy sensitivity to the GESTR/SAFER sensitivity reported in Table A.8 of NEDE-23785-1-PA (Reference 3). The NRC staff approximated the change in the stored energy based on the average fuel temperature as represented in Equation (Eqn.) 1. The NRC staff then normalized the PCT sensitivity to the magnitude of the stored energy perturbation. The specific heat is taken from the default urania model in Eqn. C.1-5 of NEDE-32176P, Revision 3 (Reference 24).

$$\delta PCT_N \approx \frac{\Delta PCT_N}{1.645\sigma}\left(\frac{\int_0^{T_{PRIME}} C_p\,dT}{\int_0^{T_{GSTRM}} C_p\,dT} - 1\right) \qquad (1)$$

Where δPCT is the PCT sensitivity to the initial average fuel temperature difference,
ΔPCT is the PCT sensitivity to the initial stored energy [[ ]]
N denotes either first or second peak,
σ is the GESTR stored energy uncertainty [[ ]]
C_p is the specific heat,
T is the initial average fuel temperature,
GSTRM denotes calculated according to the GSTRM models, and
PRIME denotes calculated according to the PRIME models

The NRC staff's approximated second peak PCT sensitivity was calculated to be [[ ]]for a BWR/4. This is consistent with the PCT difference predicted by TRACG04 [[ ]] Therefore, the NRC staff has confidence that the TRACG04 sensitivity studies are consistent with expected trends in the GESTR/SAFER methodology.

For BWR/5-6 plants it is not a forgone conclusion that the limiting PCT occurs for the second peak. These plants and some BWR/4 plants include low pressure injection into the core bypass that results in a more rapid delivery of coolant to the core relative to BWR/3-4 plant designs where the low pressure coolant injection is into the lower plenum.

Using Eqn. 1, the NRC staff estimated the impact of the difference in stored energy on the first peak PCT. The NRC staff calculation indicated a potential non-conservatism on the order of [[ ]], which is greater than the significance threshold according to 10 CFR 50.46. Therefore, the NRC staff could not reach a conclusion regarding the applicability of the interim process to BWR/5-6 plants. To address this concern, the NRC staff requested additional information regarding the first peak PCT sensitivity to the differences in stored energy in RAI 39, Supplement 3 Part B (RAI 39S3-B).

The response to RAI 39S3-B (Reference 2) provides the results of SAFER/GESTR calculations for two representative BWR plant configurations that are first peak PCT limited. The results indicate sensitivity in the first peak PCT of approximately [[ ]], which indicates consistency across the various BWR plant designs. The more detailed SAFER/GESTR calculations are: (1) consistent with the Appendix K analysis method, and (2) representative of the detailed plant response sensitivity to differences in stored energy. Therefore, the NRC staff finds that these results provide a more robust and reasonable basis (relative to the NRC staff's simplistic approach) to determine the PCT impact of the PRIME thermal model.

The RAI 39S3-B response states that any analysis impact for PRIME will be treated in accordance with the reporting requirements of 10 CFR 50.46. Therefore, the NRC staff finds that first and second peak PCT results have been adequately addressed in terms of the sensitivity to the PRIME thermal models. Additionally, the NRC staff finds that the impact on

PCT is not expected to be significant (greater than 50°F). Further, the response assures that any PCT changes will be adequately addressed according to the requirements of 10 CFR 50.46. Therefore, the NRC staff finds this approach acceptable during the interim process.

A.5.1.2 Metal-Water Reaction Criteria

GNF performed cladding oxidation calculations for a BWR/2. The BWR/4 PCT results indicate that the degree of cladding oxide formation would be insignificant based on the low temperatures. BWR/2 plants tend to have more limiting core oxidation during DBA LOCAs based on the nature of the recirculation piping. Therefore, the NRC staff finds it acceptable to compare the oxidation results for the BWR/2 plant without consideration of the BWR/4 plant.

The calculations were done to compare the maximum local oxide layer thickness as well as the fraction of cladding oxidized. The fraction of cladding oxidized is a surrogate metric to ensure that the maximum hydrogen generation criterion of 10 CFR 50.46 is met. The results indicate close agreement between the TRACG04 calculations using both fuel thermal models (PRIME and GSTRM). The NRC staff agrees that the oxidation results are essentially identical.

It is well understood that BWR/2 plant designs are most limiting in terms of the oxidation criteria due to the more aggressive rate and duration of core uncovery during DBA LOCA events. The primary reason is the design of the recirculation system with large lower vessel penetrations. Therefore, the primary phenomena driving cladding oxidation for the BWR/2 design is the period of core uncovery, which is not very sensitive to the initial fuel temperature or stored energy. Since the BWR/3-6 plant designs incorporate jet pumps, the level drop during DBA LOCA is not as severe, leading to significant margin to the cladding oxidation limits in 10 CFR 50.46.

Therefore, the NRC staff finds that the licensing calculations performed to demonstrate compliance with the metal-water reaction acceptance criteria specified in 10 CFR 50.46 are not sensitive to the thermal-mechanical models assumed in the analysis.

A.5.1.3 Coolability Criterion

The original response to RAI 39 (Reference 1) did not address core coolability. To meet the requirements of GDC 27 “Combined Reactivity Control Systems Capability” and GDC 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, or extreme coplanar fuel rod ballooning.

To meet the requirements of 10 CFR 50, Appendix K, “ECCS Evaluation Models” as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution. Burst strain and flow blockage models must be based on applicable data in such a way that: (1) the temperature and differential pressure at which the cladding will rupture are properly estimated, (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.

The NRC staff is aware that the GSTRM fuel thermal conductivity model under-predicts fuel temperature as it does not account for thermal conductivity degradation with exposure. The

PRIME model, [[ ]] The GSTRM method may also under-predict the contact pressure and rod internal pressure. These quantities are used to establish the differential pressure across the cladding during LOCA analyses. The combination of these phenomena may result in [[ ]]

On these bases, the NRC staff was unable to conclude that the use of the legacy methods would not underestimate the degree of clad swelling. Therefore, the NRC staff requested additional information regarding core coolability requirements in RAI 39, Supplement 3 Part D (RAI 39S3-D). The response to RAI 39S3-D (Reference 2) states that the PCT and oxidation were shown to be only mildly sensitive to changes in the PRIME thermal models. Therefore, it was concluded that the core coolability analyses would not be affected. The response states that the coolability requirements presented in NEDE-20566-P-A, Volume 2 (Reference 12) are unaffected because: (1) the PCT and oxidation effect is small, and (2) changes in stored energy do not affect the long-term reflood (and therefore the long-term requirement is unaffected).

The GNF position is consistent with the Atomic Energy Commission determination that meeting the PCT and oxidation requirements of 10 CFR 50.46 ensures that the Zircaloy cladding retains ductility, thus ensuring that the core geometry remains essentially unchanged and amenable to core cooling in the long-term. On this basis, the NRC staff agrees with the GNF assessment and finds that GNF has adequately addressed all of the criteria of 10 CFR 50.46.

A.5.1.4 Long Term Core Cooling Criterion

Long term core cooling is demonstrated by performing ECCS-LOCA evaluations for a sufficient duration to ensure that the capability of the ECCS to deliver coolant to the core exceeds the decay heat load. The decay power is not sensitive to the fuel thermal model, therefore, demonstration of long term core cooling is unaffected by use of either the GSTRM or PRIME fuel thermal models. Therefore, it is acceptable during the interim process to demonstrate adequate long term core cooling using the legacy methods.

A.5.2 Small Break LOCA (SBLOCA)

The response to RAI 39 (Reference 1) did not address SBLOCA sensitivity to the fuel thermal model. The conclusions of the original GESTR/SAFER model qualification and application statement regarding the limiting nature of large break LOCA (LBLOCA) events for the BWR/3-6 operating fleet have been challenged by significant changes in plant operations and other modifications. Therefore, the NRC staff requires that the impact of the sensitivity be quantified for SBLOCAs. The NRC staff requested additional information regarding the sensitivity of small break LOCA analyses in RAI 39, Supplement 3 Part A (RAI 39S3-A).

The response to RAI 39S3-A (Reference 2) provides the results of calculations performed using the SAFER/GESTR analysis methodology for varying initial stored energies. The previous analyses using TRACG indicated approximately [[ ]] The SBLOCA Appendix K calculations indicate that the differences in PRIME and GSTRM PCT, oxidation, and metal water reaction calculational results are negligible. The response states that since core uncovery does not occur during the early stage of the SBLOCA, the nucleate boiling occurring in-core during the event is sufficient to remove the initial stored energy. The response also states that the sensitivity is expected to be small once the transient evaluation period reaches the longer durations when PCT occurs for SBLOCA events. The NRC staff has reviewed these calculations and their interpretation and agrees with the engineering judgment of GNF that SBLOCA calculation results are expected to be negligibly impacted during the interim process.

A.5.3 Expanded Operating Domain ECCS-LOCA Analyses

Extended Power Uprate and Maximum Extended Load Line Limit Analysis Plus plants are required to perform ECCS-LOCA analyses at various points on the boundary of the operating domain and at various axial power shapes. The NRC staff requested additional information regarding the sensitivity of mid- and top-peaked power shape LOCA calculations in RAI 39, Supplement 3 Part C (RAI 39S3-C).

The response to RAI 39S3-C (Reference 2) states that SBLOCA calculations are performed using top-peaked power shapes since the higher elevations of the core uncover earlier and recover later than the lower portions of the core. [[

]] Therefore, GNF has addressed the various axial power shapes and confirmed that the appropriate shapes were applied to each analysis to ensure limiting conditions were evaluated using the licensing basis methodology. The NRC staff finds that these selected power shapes are appropriate and that the calculations provided by GNF have addressed the range of power shapes analyzed using the current ECCS-LOCA evaluation methods.

A.5.4 Reporting Requirements of 10 CFR 50.46

The response to RAI 39S3-B (Reference 2) states that evaluation of PRIME's impact on the licensing basis PCTs per 10 CFR 50.46 reporting requirements will be addressed in accordance with the approved Appendix K methodology basis. The NRC staff finds this approach reasonable and acceptable.

NEDO-33173, Supplement 4, "Implementation of PRIME Models and Data in Downstream Methods," July 2009 (Reference 21 and hereafter, IMLTR Supplement 4), describes the process for the implementation of the PRIME thermal models in downstream codes used for transient, stability, and accident analyses. IMLTR Supplement 4, while submitted to address Limitation 12 from the NRC staff's SE for NEDC-33173P, Revision 1, "Applicability of GE Methods to Expanded Operating Domains," (Reference 25) provides a generic plan for the implementation of the PRIME thermal models in the downstream safety analysis codes and is likewise applicable for originally licensed thermal operation applications. Therefore, the NRC staff considers IMTLR Supplement 4 as providing the basis for the implementation of the PRIME thermal models in the downstream analysis codes during the interim process.

IMLTR Supplement 4 states that the impact of using PRIME properties instead of GSTRM properties will be treated as a change in the approved methodology, per the reporting requirements of 10 CFR 50.46. The impact of this change can be conservatively estimated from the stored energy sensitivities that are carried out as a part of the Upper Bound PCT and oxide thickness calculations. These calculations in the SAFER/GESTR methodology adjust the nominal PCT to account for modeling and plant variable biases and uncertainties.

The NRC staff finds this approach acceptable to address the 10 CFR 50.46 reporting requirements during the interim process.

A.6 ADVANCED REACTOR DESIGNS

In the case of analyses performed for the Advanced Boiling Water Reactor (ABWR), General Electric (now GEH) developed modified versions of the safety analysis codes to model specific features of the ABWR design. In the case of the transient analyses, ODYN and REDY were modified and dubbed ODYNA and REDYA. The NRC staff has deferred any conclusions regarding the ABWR specific codes on the basis that no ABWR plants are currently operating.

In the case of analyses performed for the ESBWR, GEH has submitted TRs for NRC review and approval for application to the analysis of the ESBWR transients and accidents. These TRs are grouped in a series of supplements with the designation of NEDE-33083P (References 26 and 27). These TRs are currently under review by the NRC staff as part of the design certification application. While the RAI 39 response (Reference 1) utilizes sensitivity analyses performed for the ESBWR plant design, the NRC staff defers any conclusions regarding the ESBWR safety analysis methods to the ongoing review effort. The NRC staff review of the sensitivity analyses as part of the PRIME review does not herein constitute approval of the results generated for the ESBWR safety analysis of record.

A.7 CONCLUSIONS

The NRC staff has completed its review of the downstream impact of the PRIME model to steady-state, transient, and accident analysis methods that comprise the GNF standard set of reload licensing methods and calculations. On the basis of its review, the NRC staff has found that GNF has adequately addressed each downstream analysis method. The NRC staff primarily reviewed the information provided in response to RAI 39 (References 1 and 2) to assess the downstream impact.

As part of this review, the NRC staff identified one condition that has been documented in the body of this SE. It is repeated here for convenience.

Interim Process Thermal Overpower Condition:

(This limitation will be implemented for future plant- and cycle-specific analyses):

a. TOP screening limits for GNF fuel products currently used in operating plants shall be confirmed to be conservative using the PRIME methodology, or revised to be consistent with the PRIME results.

b. If the TOP screening limit has been exceeded, detailed cycle-specific calculations (if they are required) must be performed using transient fuel performance models that are fully consistent with the approved PRIME models. Implementation of this condition will be consistent with the schedule proposed in MFN 09-466. (Reference 21).

When the “Interim Process Thermal Overpower Condition” is met that NRC staff finds that the use of legacy transient analysis methods during the interim process is acceptable.

Based on the results of PCT sensitivity calculations, the NRC staff found that PRIME is not expected to significantly impact the downstream ECCS-LOCA evaluations. However, the NRC staff notes that the reporting requirements of 10 CFR 50.46 are explicit. The responses to the NRC staff RAIs have confirmed that the approach to meeting the reporting requirements of 10 CFR 50.46 is acceptable. Therefore, the NRC staff finds that use of legacy accident methods during the interim process has been adequately addressed by GNF's RAI responses.

A.8 REFERENCES

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3. TR NEDE-23785-1-PA, Revision 1, Volumes 1, 2, and 3, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident: GESTR-LOCA – A Model for the Prediction of Fuel Rod Thermal Performance," dated October 1984. (ADAMS Accession No. ML090780920)
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12. TR NEDE-20566-P-A, Volume 2, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K," dated September 1986. (ADAMS Accession No. ML092110816)
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15. TR NEDE-33213P-A, "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," dated April 2009. (ADAMS Accession No. ML091100203)

16. TR NEDE-32906P-A, Revision 3, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," dated September 2006. (ADAMS Package Accession No. ML062720163)
17. TR NEDE-32906P, Supplement 1-A, "TRACG for Anticipated Transients Without SCRAM Overpressure Analysis," dated November 2003. (ADAMS Package Accession No. ML033381073)
18. TR NEDE-32906P, Supplement 2-A, "TRACG for Anticipated Operational Occurrences Transient Analyses," dated March 2006. (ADAMS Package Accession No. ML060800312)
19. Final Safety Evaluation of TR NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated July 10, 2009. (ADAMS Package Accession No. ML091890758)
20. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, Revision 3, "Fuel System Design," dated March 2007. (ADAMS Accession No. ML070740002)
21. Letter from GEH to USNRC, MFN 09-466, TR NEDO-33173, Supplement 4, "Implementation of PRIME Models and Data in Downstream Methods," dated July 10, 2009. (ADAMS Accession No. ML091910490)
22. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 15.8, Revision 2, "Anticipated Transients Without SCRAM," dated March 2007. (ADAMS Accession No. ML070570008)
23. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.3, Revision 3, "Nuclear Design," dated March 2007. (ADAMS Accession No. ML070740003)
24. Letter from GEH to USNRC, MFN 06-109, TR NEDE-32176P, Revision 3, "TRACG Model Description," dated April 20, 2006. (ADAMS Accession No. ML061160236)
25. Final SE for NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," dated July 21, 2009. (ADAMS Package Accession No. ML092020255)
26. TR NEDC-33083P, "TRACG Application for ESBWR," dated November 2002. (ADAMS Accession No. ML023260440)
27. Safety Evaluation of NEDC-33083P, "TRACG Application for ESBWR," dated August 19, 2004. (ADAMS Package Accession No. ML041450315)

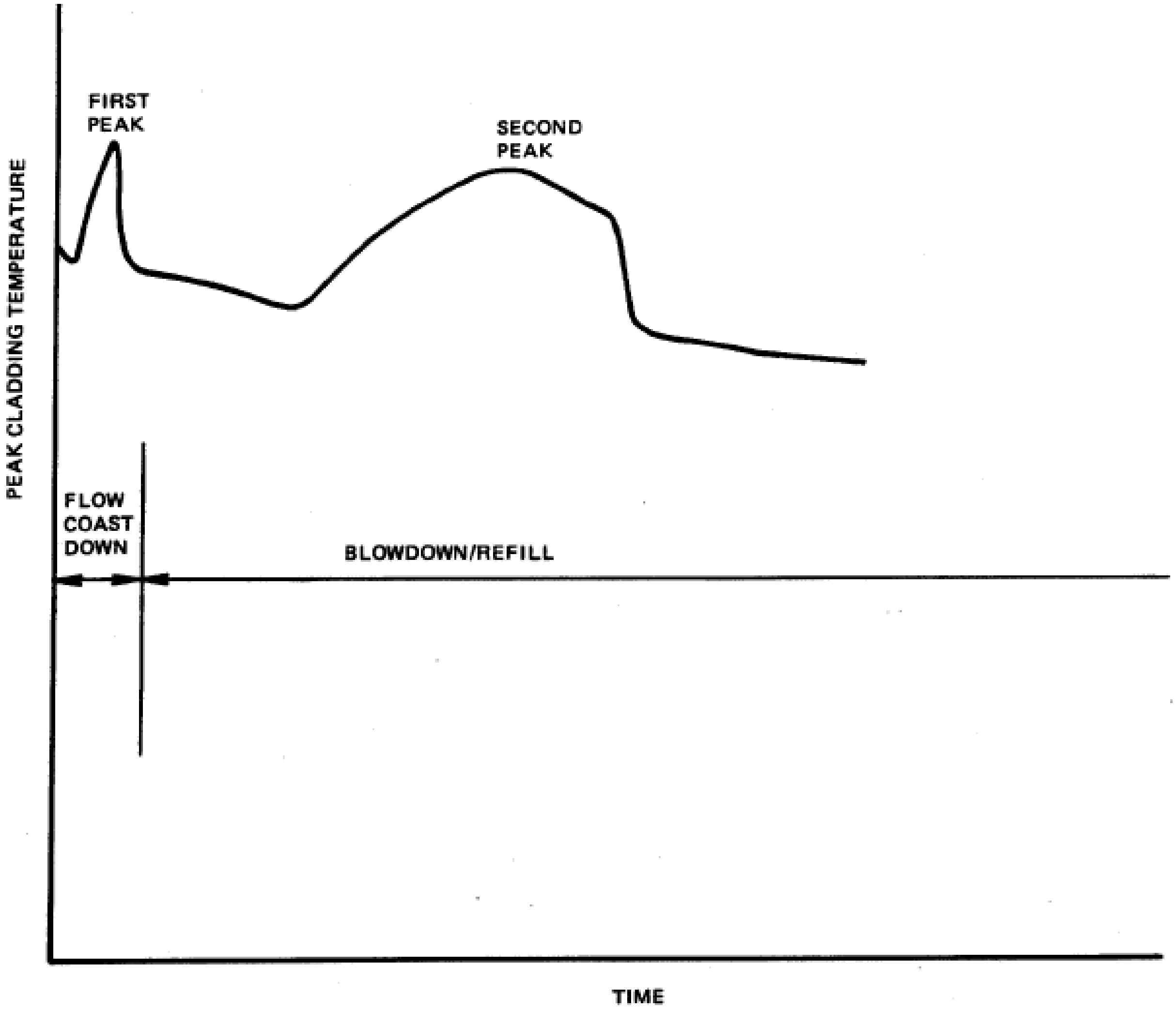

Figure 1: Typical BWR Large Break LOCA PCT Transient (from NEDE-23785-1-PA)