ENCLOSURE 2

MFN 09-106 Supplement 2

Request for Additional Information Response for the PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance

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 [[ ]].

NRC RAI 39, Supplement 3 (A)

Please provide sensitivity analysis results for small break (SB) loss-of-coolant accident (LOCA) peak cladding temperature (PCT) and oxidation similar to the results provided for the large break (LB) LOCA analyses. It is preferable to perform this analysis for a boiling water reactor (BWR) plant that is small break limited.

GNF Response

The large break (LB) LOCA analyses results that were provided by the responses to RAI–39 in MFN 09-106 were performed using TRACG code to generate a realistic response. The approved ECCS evaluation model based on SAFER/GESTR-LOCA is a conservative Appendix K methodology. The TRACG calculations presented in MFN 09-106 showed varying increase in fuel stored energy when GSTRM fuel conductivity model is replaced with PRIME03, with [[ ]]. For the current RAI response, additional evaluations are performed using SAFER/GESTR-LOCA. These evaluations showed that the impact from [[ ]] upon the PCT for small break limited plants is negligible. This outcome is consistent with the expected response based on the first principles: The primary effect of increased fuel stored energy is to increase the first peak PCT. Small break limited plants do not exhibit the first peak PCT, which is created by early boiling transition and quenched by lower plenum flashing, because small breaks remain in nuclear boiling which removes the initial fuel stored energy. The following table summarizes the results for small break (SB) LOCA for a SBLOCA-limited plant. The plant used in the analysis is a BWR 5/6-representative, however, the results are applicable to both BWR 3/4 and BWR 5/6-type plants.

	Original			[[ ]]			Difference		
Plant Type	Peak PCT (°F)	Max Oxide (%)	Metal Water (%)	Peak PCT (°F)	Max Oxide (%)	Metal Water (%)	Peak PCT (°F)	Max Oxide (%)	Metal Water (%)
BWR 5/6	[[								]]

NRC RAI 39, Supplement 3(B)

The response provides the sensitivity of the TRACG04 predicted second peak PCT. However, the second peak PCT is not necessarily the limiting PCT for all plant designs. Please provide separate results for the first peak PCT.

GNF Response

As it is mentioned in response to RAI 39, Supplement 3(A), the large break (LB) LOCA analyses results that were provided by the responses to RAI 39 in MFN 09-106 were performed using TRACG code to generate a realistic response. The response provided the sensitivity of second peak PCT because [[

]]. For the current RAI response, additional evaluations are performed using SAFER/GESTR-LOCA for two types of plants with first-peak limited PCTs. The following table summarizes the results for first peak PCT impact. [[

]].

	Original			[[ ]]			Difference		
Plant Type	First Peak (°F)	Max Oxide (%)	Metal Water (%)	First Peak (°F)	Max Oxide (%)	Metal Water (%)	First Peak (°F)	Max Oxide (%)	Metal Water (%)
BWR 3/4	[[								
BWR 5/6									]]

It should be also noted that [[

]]. The impact evaluation for PRIME on the licensing basis PCT's per 10 CFR 50.46 reporting requirements will address these impacts according to the approved Appendix K methodology basis.

NRC RAI 39, Supplement 3 (C)

For extended power uprate license applications, an analysis is performed for both mid-peaked and top-peaked power shapes at various points in the allowable operating domain. Similarly, plants may be either top-peaked or bottom-peaked in terms of PCT. Please clarify what power shapes were considered in the subject analyses. Please address both power shapes in the response.

GNF Response

The evaluations presented in responses to RAI 39, Supplements 3(A) and 3(B) covers both mid-peaked and top-peaked axial power shapes. A top-peaked axial power shape results in higher PCTs in small-break LOCA analysis compared to a mid-peak shape, since the higher elevations in the core uncover earlier and recover later than the lower elevations. For large-break analysis, this effect is not dominant and mid-peaked axial shapes remain limiting. The results shown in response (A) are calculated using top-peaked axial power shapes, and the results shown in response (B) are calculated using mid-peaked axial shapes.

NRC RAI 39, Supplement 3(D)

The original response did not provide specific disposition of the coolability requirements of 10 CFR 50.46. Please address core coolability in the response.

GNF Response

The original response to RAI 39, provided by MFN 09-106, did not explicitly provide specific disposition of the coolability requirements of 10 CFR 50.46, however, with no significant changes in PCT and oxidation results, these were implied. With the latest set of evaluations using SAFER/GESTR-LOCA, the 10CFR50.46 coolability requirements presented in LTR NEDE-20566-P-A, Volume 2, are also unaffected by [[

]]. Responses in (A) and (B) address PCT, oxidation, and hydrogen generation requirements; this response addresses all five 10 CFR 50.46 criteria.

NRC RAI 39, Supplement 3(E)

Please clarify the source of the gap gas composition information used in the sensitivity analyses. Please confirm that the TRACG04/PRIME03 calculations were performed with gas gap compositions generated using the PRIME methodology. Likewise, please confirm that the TRACG04/GSTRM calculations were performed with gas gap compositions generated using the GSTRM methodology.

GNF Response

The understanding of the NRC reviewer is correct. TRACG04 calculations that were performed using the fuel thermal conductivity model compatible with PRIME03 used gap and fission gas parameters supplied through files produced using the PRIME methodology. Similarly, TRACG04 calculations that were performed using the fuel thermal conductivity model compatible with GSTRM used gap and fission gas parameters supplied through files produced using the GSTRM methodology.

NRC RAI 39, Supplement 3(F)

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GNF Response

The parameter referred to in the question was historically introduced to account for minor fuel temperature increases that are expected during AOO transients. It remains valid for its intended purpose relative to NRC-approved applications of TRACG for AOO transients. The parameter is also appropriate for stability applications where no significant increases in fuel temperatures are expected. The parameter is not intended for modeling higher and longer-term fuel temperature increases such as those that might be expected for LOCA applications for operating BWRs. For these applications the uncertainty associated with fission gas released into the gap is addressed as described in the response to RAI 46.

Note that the subject parameter in no way influences how radiological dose evaluations are performed. For fuel rods where the cladding has been determined to be perforated, the amount of fission gas release is stipulated either according to 10 CFR 100 or 10 CFR 50.67 depending on the licensing basis for the plant.

NRC RAI 42

Page 23 of RAI-32 states that GNF will implement the modified creep relation in the PRIME code, and the modified critical pressure correlation. Please provide details of the modified creep relation and the modified critical pressure correlation. Both of these correlations should use thick wall formulas.

GNF Response

As discussed in Attachment A, the current PRIME low stress irradiation creep model is based upon the assumption that the [[

]] However, as stated in the response to RAI-32, to assure consistency between the PRIME code and the critical pressure calculation, and to eliminate the assumptions inherent in applying thin shell relations to current GNF fuel geometries, GNF will use [[

]] The thick shell relations used to derive the modified creep relation and the critical pressure calculation are summarized below.

The component stresses are given by

$$\sigma_r = \frac{P_i r_i^2 - P_o r_o^2 + \dfrac{r_i^2 r_o^2 (P_o - P_i)}{r^2}}{r_0^2 - r_i^2} \qquad \text{Equation 1}$$

$$\sigma_\theta = \frac{P_i r_i^2 - P_o r_o^2 - \dfrac{r_i^2 r_o^2 (P_o - P_i)}{r^2}}{r_0^2 - r_i^2} \qquad \text{Equation 2}$$

$$\sigma_z = \frac{P_i r_i^2 - P_o r_o^2}{r_0^2 - r_i^2} \qquad \text{Equation 3}$$

where:

P_i/P_o = internal/external pressure
r_i/r_o = inner/outer radius
r = midwall radius $(r_i+r_o)/2$

The generalized stress is given by

$$\sigma_g = \sqrt{0.5\left((\sigma_\theta - \sigma_z)^2 + (\sigma_z - \sigma_r)^2 + (\sigma_r - \sigma_\theta)^2\right)} \qquad \text{Equation 4}$$

The relationship between $\dot{\varepsilon}_\theta$ and $\dot{\varepsilon}_g$ is given by

$$\dot{\varepsilon}_i = \dot{\varepsilon}_g \frac{\partial \sigma_g}{\partial \sigma_i} \quad ; \; i = r, \theta, z$$ Equation 5

Taking the partial derivative of σ_g with respect to σ_θ, equation 5 yields

$$\dot{\varepsilon}_\theta = \frac{\dot{\varepsilon}_g}{\sigma_g}\left(\tfrac{1}{2}(\sigma_\theta - \sigma_z) - \tfrac{1}{2}(\sigma_r - \sigma_\theta)\right)$$ Equation 6

From equation 6 the generalized strain is taken as [[
]] in Attachment B.

As summarized in Attachment B, application of these relation results in a relation for generalized (low stress, fast flux activated) steady state creep rate for annealed Zircaloy cladding given by

[[ ]]

[[

]]

Where P_c is in ksia

For a typical 10x10 barrier UO_2 rod assuming

[[

]] and taking nominal values of r_p, r_i and r_o as

[[

]]

the critical pressure relation yields

[[ ]]

The critical pressure relation above is used to estimate the uncertainty in critical pressure [[

]] Using the results from these perturbations and adding those by the standard error propagation statistical methodology, the uncertainty for the critical pressure is estimated to be

[[ ]]

This calculation neglects the effects of other uncertainties, which are generally smaller relative to contributions of cladding creep rate and pellet swelling rate uncertainties, as noted in the combined response to RAI-32 and S01. As noted in the response to RAI-32, GNF calculates the internal pressure design ratio DR given by

[[ ]]

where

P_i, P_c = nominal rod internal and critical pressures
σ_{pi}, σ_{pc} = internal and critical pressure standard deviations

A value of DR < 1.0 assures with 95% confidence that the internal pressure will not exceed the pressure at which the cladding creepout rate is equal to the pellet fission product swelling rate. In the formulation of the DR, the denominator is an effective lower 95 critical pressure.

[[

]]

This results in a DR of

[[ ]]

In summary, the modified creep relation derived in Attachment B will be incorporated into the PRIME code and the PRIME qualification will be repeated. New model perturbations will be derived based upon the results of the re-qualification. Section 5.6.1 of NEDC-33256P (PRIME Technical Basis), which describes the PRIME low stress creep relations for annealed cladding, will be revised to reflect the modified creep relation in the –A version of the PRIME LTR. Sections 2.0, 3.0, 4.0 and 5.0 of NEDC-33257P (PRIME Qualification) will be revised to document the results of the re-qualification in the –A version of the LTR. Section 3.2.4 of NEDC-33258P (PRIME Application Methodology) will be revised to reflect the new model perturbations in the –A version of the LTR. All revisions to figures and text in the LTR will be included in the -A version of the LTR.

GNF will apply this modified PRIME code and resulting model uncertainties in fuel design and licensing calculations. GNF will also apply the modified critical pressure relation in the liftoff calculation.

As discussed above, results of [[

]]

A detailed description of the inputs for the critical pressure calculations is shown in Tables 42.1 and 42.2.

Table 42.1: Typical Input Parameters for the Critical Pressure Calculation

Parameters	Source/Reference	Value	Comment
Cladding inner radius, r_i	Fuel Rod Drawings	Varies with fuel design	For typical 10x10 design [[ ]]
Cladding outer radius, r_o	Fuel Rod Drawings	Varies with fuel design	For typical 10x10 design [[ ]]
Cladding midwall radius, r	Calculated as $r = (r_i + r_o)/2$	Varies with fuel design	For typical 10x10 design [[ ]]
Pellet outer radius, r_p	Fuel Rod Drawings	Varies with fuel design	For typical 10x10 design [[ ]]
Rod Internal Pressure, P_i	Calculated by PRIME	Varies with LHGR & fuel design	For typical 10x10 design [[ ]]
Reactor Pressure, P_o	Reactor system pressure	Typical value =1040 psi	
Pellet swelling rate, α =	As documented in the PRIME RAI response	[[ ]]	
LHGR	Instantaneous nodal LHGR	Varies with fuel design & TMOL	[[ ]]
Cladding average temperature, T_K	Calculated by PRIME	Varies with fuel design & TMOL	Typical 10x10 (at high exposure) [[ ]]
Fast Flux, ϕ	Instantaneous nodal fast flux calculated as documented in the Attachment C	[[ ]]	

Table 42.2: Typical Parameters used in the Critical Pressure Uncertainty Calculations

Parameters	Source/Reference	Value	Comment
Cladding Creep Rate Uncertainty	As documented in the PRIME RAI response	[[ ]]	Perturbed 2 sigma
Pellet Swelling Rate Uncertainty	As documented in the PRIME RAI response	[[ ]]	Perturbed 2 sigma
Rod Internal Pressure Uncertainty	Calculated by the statistical error propagation method as described in Section 1.3.1 of NEDC-33258P	Typical value for 10x10 design is [[ ]]	

Attachment A
Clarification of GNF Thin Shell Stress Relations

In the RAI-32, S01, S02, S03 and S04 responses, and in this response, GNF-A has referred to the set of equations below (Set 1) as the 'thin shell stress relations':

$$\sigma_\theta = \frac{P_i r_i - P_o r_o}{r_o - r_i}$$

$$\sigma_z = \frac{\sigma_\theta}{2}$$

$$\sigma_R = 0$$

The GNF-A nomenclature is based upon evolving nomenclature, primarily in the design of large pressure vessels. Traditionally, however, the set of equations below (Set 2) is referred to as the 'thin shell stress relations' for a pressurized cylindrical shell:

$$\sigma_\theta = \frac{\left(P_i - P_o\right) r_{avg}}{r_o - r_i}$$

$$\sigma_z = \frac{\sigma_\theta}{2}$$

$$\sigma_R = 0$$

For typical GNF-A cladding geometries and loading conditions, circumferential stresses calculated using the Set 1 equations are very similar to those calculated using the thick shell stress equations for both low exposure (compressive) and high exposure (tensile) conditions. Circumferential stresses calculated using the Set 2 equations are approximately 10% lower (less compressive) for low exposure operation and approximately 10% higher (more tensile) for liftoff conditions. For this reason [[ ]], which is primarily cladding circumferential creep data. For the same reason, and for consistency, [[ ]] As noted in this response, the cladding low stress creep relation and the critical pressure calculation have been rederived using thick shell stress relations.

Attachment B
Derivation of Modified Creep Relation

The current PRIME creep relations for annealed Zircaloy cladding consist of creep relations for [[

]] This data is the newest, most prototypical and best characterized creep data for GNF annealed Zircaloy cladding under liftoff conditions. The radial, tangential and axial stress components, generalized stress, and ratio of tangential/generalized creep are calculated for each [[ ]] point using thick shell relations. The results are shown in Table 1, together with the corresponding quantities calculated using thin shell relations.

The [[

]]

Thus the current steady state [[

]] points yields

[[ ]]

and the low stress, flux activated creep correlation becomes

[[ ]]

Table 1

Stress and Strain Calculations for K5 Creep Data

[[

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Table 2
Ratios of Generalized Stress and Strain to Tangential Stress and Strain for the K5 Creep Data based upon Thick Shell Relations

[[

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Attachment C

(Instantaneous Fast Flux Calculation for the Critical Pressure Calculation)

To calculate fast, flux, the thermal flux must first be obtained. The volume-average fission rate is given in terms of LHGR by:

[[

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[[

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Figure 42.1: Void Fraction as a Function of Axial Elevation

NRC RAI 43

In response to RAI-9, most of the growth data provided above [[ ]] are from water rods and channels, and these data appear to be underpredicted. Please provide additional data above [[ ]] and propose an application limit of the growth model based on the available data.

GNF Response

A comparison of PRIME predicted cladding permanent axial strain with measured fuel rod permanent axial strain data is shown in Figures 3.3 and 3.4 of NEDC-33257P. These figures include measured data to very high exposure (>80 GWd/MTU rod average). These figures include data from low exposure rods where pellet-cladding interaction is minimal and from high exposure rods where pellet-cladding interaction and the resulting permanent strain due to plasticity and creep can be significant contributors to the measured strains. Based upon the results in these figures, PRIME predicts the axial growth of fueled rods due to the combined effects of irradiation growth and pellet-clad axial interaction very well up to 80 GWd/MTU rod average exposure on a best estimate basis. For BWR conditions, this exposure is equivalent to a rod average fast fluence of 15.0×10^{21} n/cm^2. On this basis, GNF proposes a rod average fast fluence limit of 15.0×10^{21} n/cm^2 for PRIME application.

NRC RAI 44

Please provide fuel rod void volume data that can be used to justify the PRIME credit for stacking/chamfer volume.

GNF Response

Comparison of the PRIME void volume predictions with the available measurements are shown below in Figure 44.1. As shown in this Figure, PRIME predicts the available free volume measurements very well. Also, as described in the RAI-18 response, based on manufacturing data a 'stacking factor' is used in the calculation of cold and hot free volume in the fuel rod. This factor quantifies the contributions of [[

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[[

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Fig. 44.1: Predicted versus Measured Fuel Rod Free Volume

For each GNF fuel design, the stacking factor is determined from [[

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For application in the PRIME code, the stack factor is converted to a stack density. [[

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As V_{stack} is difficult to measure directly in production fuel, it is calculated based on an estimate of the Geometric Stacking Factor (GSF). The GSF is the volume fraction of a pellet column that is void space due to enrichment marks, chamfers, pellet chips and other defects, and is expressed as:

[[

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In production rods, the actual GSF for any axial fuel zone can be estimated using the following relation:

[[

]]

The data sets in Figures 44.2 and 44.3 on the following page show production runs for GE-14 and GNF2 fuel from early 2008, each data point showing the GSF of an enriched axial zone from a single rod design, totaling 11,184 data points (rod-zones). The dark lines in the plots represent the GSF calculated using design basis values, which are also used for PRIME inputs. This design basis GSF is shown in Table 44.1 below, along with GSF values determined by averaging the plotted production data points:

[[

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Table 44.1: GSF Summary

The production data supports the design basis GSF (and therefore V_{stack}), with the average GSF for this as-built fuel closely matching the design value. [[

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[[

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NRC RAI 45

PNNL has noted that PRIME underpredicts fuel centerline temperature at high rod powers and overpredicts fuel conductivity between [[ ]]. Please provide an example calculation that shows how uncertainties will be applied to bound this non-conservatism for power to melt analyses assuming [[ ]] over-power for UO2.

GNF Response

Basis for the PRIME thermal conductivity model is provided in RAI-20 S01 and RAI-17 S01 responses. As shown in these responses, the PRIME model predicts the measured diffusivity data of GNF fuel pellets very well. In addition, in the PRIME application methodology, the fuel melting analysis is performed statistically, where uncertainties in the manufacturing and operating parameters as well as in the PRIME model prediction are explicitly addressed. In the statistical analysis each of the manufacturing and operating parameters is individually perturbed two standard deviations (with some exceptions) from its best estimate value in the direction that worsens the output parameter being analyzed. The specific parameters perturbed include:

[[

]]

In addition to the parameters specified above, the statistical analysis methodology includes a model uncertainty perturbation to address fundamental uncertainties in the PRIME qualification results and known product and operating uncertainties. This model uncertainty perturbation consists of a [[ ]] (2σ) increase in power plus a perturbation of the fission gas release threshold as described in Section 3.2.4 of NEDC-33258P.

The results from these perturbation analyses are then utilized with the best estimate results to calculate finite difference approximations of the partial derivatives as follows for use in the error propagation analysis:

$$\frac{\partial P}{\partial x_i} = \frac{P_{perturbed} - P_{nom}}{2\sigma_{x_i}}$$

where

$P_{perturbed}$ = Value of the PRIME output parameter for the case where PRIME input variable xi is perturbed

P_{nom} = Best estimate value of PRIME output parameter being analyzed

The PRIME output parameter standard deviation is then calculated using the standard error propagation method. The 95% confidence value for the output parameter P is given by:

$$P_{95} = P_{nom} \pm K_{95}\sigma_p$$

where:

K_{95} = 95% confidence statistical tolerance factor
= [[ ]]

For the thermal overpower analyses, evaluations are performed for each fuel rod type (UO_2 full and part length rods, gadolinia rods, barrier or non-barrier cladding) over a range of exposures and overpowers to simulate various AOOs. The evaluations reflect operation on the bounding power-exposure operating envelope prior to the AOO. Based upon the results of these evaluations, fuel rod type specific maximum thermal overpower limits are defined to prevent centerline melting. Compliance with these maximum overpower (Thermal Overpower) limits assures with 95 percent confidence that violation of the no fuel (centerline) melting criterion does not occur for operation on the LHGR limits.

Typically, the melt margin analyses are done based on a 1-node model. The 1-node case is a transient case run to determine overpower (OP) limits and is based upon the assumption that the node operates on LHGR limits prior to the transient. The perturbation variables are utilized for the 1-node fuel melting analysis as discussed above. [[ ]] is included in the 1-node analysis to account for the power peaking due to pellet densification. The active fuel length and plenum are assumed to be equal to those for the fuel rod, but the gas inventory and pressure are obtained from the results of multi-node cases. This avoids overly conservative fission gas release and gap conductance that would result from calculation of fission gas release based upon the power of the 1-node (peak power) case. This analysis is done at various exposure points and overpowers to determine the most limiting point for fuel melting in the fuel lifetime and the magnitude of allowable overpower at that most limiting time in the fuel lifetime (as defined by rapidly ramping to a specified percentage over the bounding envelope and holding for 10 minutes). The limiting values are determined for the limiting UO_2 and gadolinia rods. The melt margin analysis results with 25% and 45% overpower for a typical 10x10 UO_2 fuel rod is shown in Table 45.1 and Table 45.2, respectively.

The upper 95 results are used to confirm compliance with the no melting criterion. The results in Table 45.1 and Table 45.2 indicate that the PRIME methodology introduces significant conservatism in the analysis to assure compliance with the no melting criterion.

Table 45.1: Melt Margin Analysis Results for a Typical 10x10 UO2 rod (non-barrier) with 25% Overpower

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Table 45.2: Melt Margin Analysis Results for a Typical 10x10 UO_2 rod (non-barrier) with 45% Overpower

[[

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NRC RAI 46

RAI-39, Supplement 1 requested a demonstration of the impact on LOCA PCT due to a 50% lower rod pressure than the LOCA input from PRIME. This was not provided in previous RAI responses. Please provide the impact of 50% lower and 50% higher rod internal pressure than PRIME calculates on LOCA PCT and cladding oxidation.

GNF Response

For LOCA applications in operating BWRs, the primary factor that results in a challenge to the integrity of the cladding is the reduction in strength that occurs as the cladding temperature rises. Above about 1000 K the cladding strength is reduced so that increased rod internal gas pressure may cause the cladding to balloon and fail. There are three main elements of total gas pressure that are relevant for a postulated LOCA in an operating BWR: (1) initial fill gas pressure and fission gases released from the fuel pellets prior to the transient; (2) additional fission gases released from the pellet during the transient; (3) increase in the total gas (fill gas plus fission gases) pressure due to thermal expansion during the transient. Element (1) is considered in both the SAFER and TRACG04 methodologies [[

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TRACG04 is used in this RAI response as a tool to provide information similar to but not exactly the same as what has been requested. The curves in Figure 46-1 are representative for BWR fuel rods from modern commercial 10x10 fuel. The green curves are representative of an average LHGR history at [[ ]] kW/ft that can be expected for a high-powered commercial BWR fuel rod. For rods without gadolinia (dashed green curve) and rods with [[ ]] gadolinia (open green squares) the requested ±50% change in rod internal gas pressure bounds the uncertainty in the fuel rod internal pressure at greater than the 2-sigma level for any exposure below about [[ ]] GWd/MTU. This exposure value is well above the exposure for which the limiting peak clad temperature (PCT) for a LOCA will occur. The uncertainty in the rod internal pressure increases with exposure and average LHGR operating history due to uncertainty associated with fission gas release from the pellet. TRACG04 models these dependencies [[

]] For an uncertainty range of ±2-sigma a high-powered BWR fuel rod operated so that the peak axial node has averaged [[ ]] kW/ft will be modeled in TRACG04 with a perturbation to the total internal rod pressure that is greater than the requested ±50% range when the peak pellet exposure is greater than about [[ ]] GWd/MTU. For lower exposures and LHGR histories the uncertainty in total internal rod pressure decreases consistent with the expected uncertainty characteristics associated with the PRIME fission gas release model. It is the opinion of GEH that this will provide the impact on PCT and oxidation more consistent and appropriate to how TRACG04 will be used in BWR LOCA applications.

For LOCA applications it is important to model the timing of cladding rupture in order to accurately calculate the oxidation rate that can occur on the inside surface of the cladding after rupture occurs. TRACG models the cladding rupture stress as shown in Figure 7-17 of NEDE-32176P, Rev. 4 (previously provided to the NRC). This figure is redrawn on a linear scale in this response as Figure 46-2 [[

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The responses to RAI 39, supplement 2 provided the sensitivity of the LOCA licensing parameters to changes in the thermal conductivity of the gap. Those responses explained why the gap composition has negligible impact on the LOCA licensing parameters. The results in Table 46-1 below show that the changes in the gap pressure also have a negligible impact on peak clad temperature and cladding oxidation. In these calculations the conductivity of gas in the gap has not been changed because that impact was previously evaluated.

The calculated peak clad temperature (PCT) responses for BWR/4 and BWR/2 DBA LOCAs are shown in Figure 46-3 and Figure 46-4, respectively. In these figures and those that follow, the green curve is the nominal response, the blue curve corresponds to a 2-sigma reduction in the gap pressure and the red curve corresponds to a 2-sigma increase in the gap pressure. The PCT responses shown in Figure 46-3 and Figure 46-4 [[
]] The PCT responses are not sensitive to the composition or pressure of fission gases in the gap because for the fuel rods and pellet locations where the PCT occurs the gap is essentially closed. For this condition the heat transfer from the fuel pellet to the cladding inner surface is dominated by thermal conduction at the direct contact points between the pellet and the inner cladding surface. The direct contact between the fuel pellet and the cladding is also dominating the outward pressure being exerted on the cladding.

The PCT values for the BWR/4 DBA LOCA remain low enough that the zirc-water reaction is insignificant (see Figure 46-3). That is why oxide thicknesses and volumes for the BWR/4 DBA LOCA are not presented. The BWR/2 the DBA LOCA produces substantially higher PCT values than the BWR/4 DBA as seen when comparing Figure 46-4 to Figure 46-3. The zirc-water reaction rate is an exponential function of temperature so it is not insignificant for the BWR/2 DBA LOCA. The maximum oxide thickness as a fraction of the cladding thickness is plotted for the BWR/2 case in Figure 46-5. The interesting feature in these curves occurs at about 250 seconds where the abrupt increase in the slope corresponds to perforation of the cladding and the beginning of two-sided oxidation. The case with the increased internal fuel rod pressure perforates slightly sooner than the two cases with nominal and reduced internal fuel rod pressure. The differences in the times to perforation is not large because perforation is dominated by the reduction in strength that occurs as the cladding temperature rises rather than the internal forces pushing outward on the cladding. Ultimately, the maximum oxide thicknesses are essentially the same because the temperature profile with time is dominated by the balance between decay heat and cooling neither of which are significantly altered by changing the internal fuel rod pressure. As seen in the earlier sensitivity studies where the gap conductance was varied, changing the characteristics of the pellet-cladding gap will impact the initial stored energy and thus have some impact on the cladding temperature very early during the LOCA but these impacts will be insignificant in the longer term where the balance between decay heat and cooling is the dominant consideration.

The total oxide volume as a fraction of the total zirc volume in the core is shown for the BWR/2 DBA LOCA in Figure 46-6. The oxide volume fraction is a core-wide parameter unlike maximum oxide thickness, which is localized. Note that the plotted volume fraction includes the initial pre-transient oxide whereas the 1% acceptance criterion is based on only the increase in oxide volume during the accident. In other words, there is plenty of margin to the licensing limit. The variation between the three sensitivity calculations is well within the uncertainty of the calculations so one should be careful not to attach too much significance to the trends.

CONCLUSIONS

An increase or decrease in the internal gas pressure within a fuel rod in the range of ±2-sigma has a negligible impact on the LOCA licensing parameters. This is the result of the fact that the timing of perforation of the cladding that will impact when two-sided cladding oxidation can occur is primarily sensitive to the temperature response of the cladding not the gas pressure inside the cladding.

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Figure 46-1 Uncertainty in Rod Internal Pressure

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Figure 46-2 Cladding Rupture Stress versus Temperature

Table 46-1 Fission Gas Pressure Impact on LOCA Licensing Parameters

Fuel conductivity model	PRIME03	PRIME03	PRIME03
Fuel file basis	PRIME03	PRIME03	PRIME03
Gap gas pressure	-2 sigma	nominal	+2 sigma
BWR/4 Design Basis Accident LOCA			
Peak Clad Temperature (K)	[[		]]
BWR/2 Design Basis Accident LOCA			
Peak Clad Temperature (K)	[[		
max. cladding oxide: thickness (%) core volume(%)			]]

[[

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Figure 46-3 Impact on BWR/4 DBA PCT

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Figure 46-4 Impact on BWR/2 DBA PCT

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Figure 46-5 Impact on BWR/2 DBA Maximum Oxide Thickness

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Figure 46-6 Impact on BWR/2 DBA Oxide Volume Fraction