GE ENERGY

Louis M. Quintana
Manager, Licensing

P.O. Box 780, M/C A-30
Wilmington, NC 28401
USA

T 910.675.6616
F 910.675-5549
Louis.Quintana@ge.com

<u>*Proprietary Notice*</u>
This letter forwards proprietary information in accordance with 10CFR2.390. The balance of this letter may be considered non-proprietary upon the removal of Enclosure 1.

MFN 06-195
June 23, 2006

U.S Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: Responses to Methods RAIs - Interim Methods LTR

By Reference 1, GE submitted GE Licensing Topical Report (LTR), NEDC-33173P, Applicability of GE Methods to Expanded Operating Domains. During recent discussions with the NRC staff, additional information was requested to support the NRC's review of the LTR. The requested information is provided in Enclosures 1 and 2.

Please note that the information enclosed supports any application that references the subject LTR.

If you have any questions, please contact, Mike Lalor at (408) 925-2443 or myself.

Sincerely,

Louis M. Quintana

Louis Quintana
Manager, Licensing

Project No. 710

Reference:

1. MFN 06-056, Letter from Louis M. Quintana to U.S. Nuclear Regulatory Commission Document Control Desk, February 10, 2006, *GE Licensing Topical Report NEDE-33173P, Applicability of GE Methods to Extended Operating Domains.*

D065

Enclosures:

1. GE Responses to RAIs Related to NEDE-33173P, proprietary.
2. GE Responses to RAIs Related to NEDE-33173P, nonproprietary.
3. Affidavit, dated June 23, 2006.

cc: JS Bowman (GNF/Wilmington)
MJ Colby (GNF/Wilmington)
JF Harrison (GE/Wilmington)
MA Lalor (GE/San Jose)
PT Tran (GE/San Jose)
Michelle Honcharik (NRC)
Tony Langley (TVA)
eDRF 0000-0055-0354

ENCLOSURE 1

MFN 06-195

Responses to RAIs Related to NEDE-33173P

GE Proprietary Information

PROPRIETARY INFORMATION NOTICE

This enclosure contains proprietary information of the General Electric Company (GE) and is furnished in confidence solely for the purpose(s) stated in the transmittal letter. No other use, direct or indirect, of the document or the information it contains is authorized. Furnishing this enclosure does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GE disclosed herein or any right to publish or make copies of the enclosure without prior written permission of GE. The header of each page in this enclosure carries the notation "GE Proprietary Information."

GE proprietary information is identified by a double underline inside double square brackets. In each case, the superscript notation{3} refers to Paragraph (3) of the affidavit provided in Enclosure 3, which documents the basis for the proprietary determination. [[This sentence is an example.{3}]] Specific information that is not so marked is not GE proprietary.

ENCLOSURE 2

MFN 06-195

Responses to RAIs Related to NEDE-33173P

Non-Proprietary Version

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1 to MFN 06-195, which has the proprietary information removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

NRC RAI 2.0

Section 2.3 addresses the adequacy of the 0.0038 Δk/k in the calculation of SDM.

2.1 The demonstration of the shutdown margin is dependent on the cold critical measurement performed at the plant and the eigenvalue for the core with all rods inserted, but with the strongest rod out (Ksro). The code critical measurements are performed after each outage and can be used to demonstrate the adequacy of the neutronics methods for this "distributed" criticality. However, the Ksro value requires experiments to be performed with single rods out, which represent "local" criticality experiments. These local experiments are not performed very frequently, yet the prediction of the SDM relies on the accurate calculation of the Ksro value. The data provided does not distinguish between local cold critical and in-sequence cold critical measurements.

(a) Local cold critical measurements are a more physical demonstration of the stuck rod out (SRO) condition enforced by the 0.0038 Δk/k technical specification limit. Please separate out this data and provide an assessment of the methods accuracy for prediction of the local critical states demonstrating that the bias and uncertainties that are currently applied are adequate for expanded operating domains.

(b) As in Figure 2-5, provide the predicted (e.g., design basis) and measured eigenvalues. Compare the performance versus the distributed cold critical measurements and discuss any other biases or uncertainties that are applied to the Ksro values in the SDM demonstration.

GE Response:

Of the plant data provided in Figure 2-5 of Reference 1, plant C contains both in-sequence (distributed) and local cold critical demonstrations. The following table includes the local critical data of the figures, plus additional information from prior cycles for plant C. The table includes both the demonstrated cold critical eigenvalue and the Nuclear Design Basis (NDB) reference eigenvalue for cold shutdown margin and local critical experiments.

The design basis eigenvalue includes [[

]] By comparison with the data reported in Reference 1 which indicates that the standard deviation of all differences (both local and distributed) is [[ ]], one may conclude that the predictive performance for local criticals is essentially the same. Additionally, the procedure to [[ ]] is effective. This performance again supports the margin discussion contained in Reference 1.

Finally, one must note that this database of local critical data for plant C is applicable to other plants primarily because the localized nature of the experiment, which consists of only a small number of withdrawn or partially withdrawn control blades, isolates the event to a very small portion of the core. So, the predictive accuracy for a local critical experiment in any core is readily transferable to other plants and cycles. Additional discussion on the insensitivity of cold critical data to power rating or operational strategy is provided in the response to RAI 2.2.

Table R2.1-1 Plant C Local Critical Eigenvalue Performance

Plant C	Cycle	Test Data	NDB	Delta
Local 1	[[			
Local 2				
Local 3				
Local 1				
Local 2				
Local 3				
Local 4				
Local 1				
Local 2				
Local 3				
Local 1				
Local 2				
Local 1				
Local 2				
Local 3				
Local 4				
Local 5				
Local 6				
Local 7				
Local 8				
Local 1				
Local 2				
Local 3				
Local 1				
Local 2				
Local 3				}]]

NRC RAI 2.2

The LTR states that the same SDM Technical Specification value used for non-EPU core designs is adequate for EPU and expanded operating domain conditions. Provide the basis as to why cold SDM is not a strong function of the current operating strategies by comparing cold critical data before and after EPU. Include in the discussion the impact of core designs necessary to achieve EPU and maintain extended cycle lengths (e.g., larger batch fractions, higher bundle enrichments and different core loading patterns).

GE Response:

Cold shutdown margin (SDM) calculations by their nature are not directly evaluated at EPU conditions. Being a calculation (and a subsequent demonstration) performed at the most reactive core conditions, it is evaluated in a cold, unvoided, xenon-free state; not at the rated power/flow conditions. However, as noted, changes in core and fuel designs resulting from design requirements needed to support EPU could potentially impact the calculational accuracy of the SDM analysis. Provided below is a brief discussion of the purpose and limitations of the SDM demonstration itself, followed by a brief discussion of the impact of EPU related design changes on SDM calculations.

During the design and licensing of a reload core, SDM is calculated to provide assurance that the reactor can remain subcritical in the most reactive condition with the highest worth control rod fully withdrawn. The plant Technical Specifications (Tech Specs) further require that a SDM demonstration be performed prior to startup after any core reconfiguration (i.e., at the start of a new cycle) to demonstrate that the plant does indeed remain subcritical with the calculated strongest worth control rod fully withdrawn.

Tech Specs typically require a SDM value of 0.38% $\Delta k/k$ be demonstrated. This demonstration requirement has been put in place so that predictive calculations are not the sole basis of this Tech Spec. By doing so, the bulk of the uncertainties associated with the modeling of SDM are minimized. The Tech Spec requirement has been established because the SDM demonstration itself is subject to variations regarding the core and fuel that cannot be reasonably eliminated. Among these are fuel manufacturing tolerances in ^{235}U enrichment, gadolinia enrichment and component dimensionalities; and control blade reactivity uncertainties due to manufacturing tolerances and control blade burnup variations. These demonstration uncertainties are not dependent primarily on calculational methods or rated power level (i.e., EPU versus non-EPU), but on manufacturing and operational variations.

In performing SDM licensing calculations, a design criterion considerably greater than the Tech Spec requirement is imposed so that there will be a high assurance of success when the demonstration is actually performed. This high assurance of success is desirable from both a safety and a commercial standpoint. At GNF, a SDM design criterion of 1% $\Delta k/k$ has always been required.

Given that a demonstration is always required, the inaccuracies associated with the analytical determination of SDM will always have a built-in confirmation; however, the potential impact of EPU designs on SDM calculations is nevertheless expected to be minimal. The primary influence of EPU designs is the consequence that a higher operating power level (at a similar capacity factor) will require that the core produce more energy for a given cycle length. This higher energy requirement necessitates the loading of fuel of higher enrichment and/or a higher batch fraction of fresh fuel.
As for batch fraction, there continues to be a variety of cycle lengths supported by GNF as utilities continue to request designs for annual, eighteen month, and two year cycles, with accompanying variations in batch size. This has allowed GNF to gain considerable experience with both small, intermediate and large batch sizes for both high and low power density cores. The cold critical information previously provided demonstrates that the cold critical calculational accuracy of GNF methods has not suffered a degradation with increasing batch size.

As for enrichment (and discharge exposure), discharge exposure is currently constrained to a maximum value of 70 GWd/MT peak pellet exposure. Many of GNF's non-EPU designs already approach this licensing limit. Thus the ability for EPU fuel designs to increase enrichment and discharge exposure is limited by the constraints already imposed on peak exposure (as well as peak pellet ^{235}U enrichment). Given this, bundle designs for EPU applications are expected to be very similar in enrichment and gadolinia content to non-EPU designs. Batch fractions, however, are proportionally greater than pre-EPU designs. Since somewhat larger batch fraction designs do not result in fuel of higher discharge exposure or significantly different isotopic content, these proportionally larger fresh fuel batch fractions are not viewed as increasing the cold reactivity calculational uncertainties. The validity of this conclusion will be confirmed in the beginning-of-cycle SDM demonstration for EPU cores prior to startup of the initial cycle. Further confirmation will occur as subsequent cycles are operated.

As a final demonstration of these concepts, the trending of the cold eigenvalues for a BWR/4 through a 120% EPU transition is provided in Figure R2.2-1. The scale of the data is consistent with that given in Figure 2-4 of NEDC-33173P. There is no identifiable aberration with the trend because of EPU.

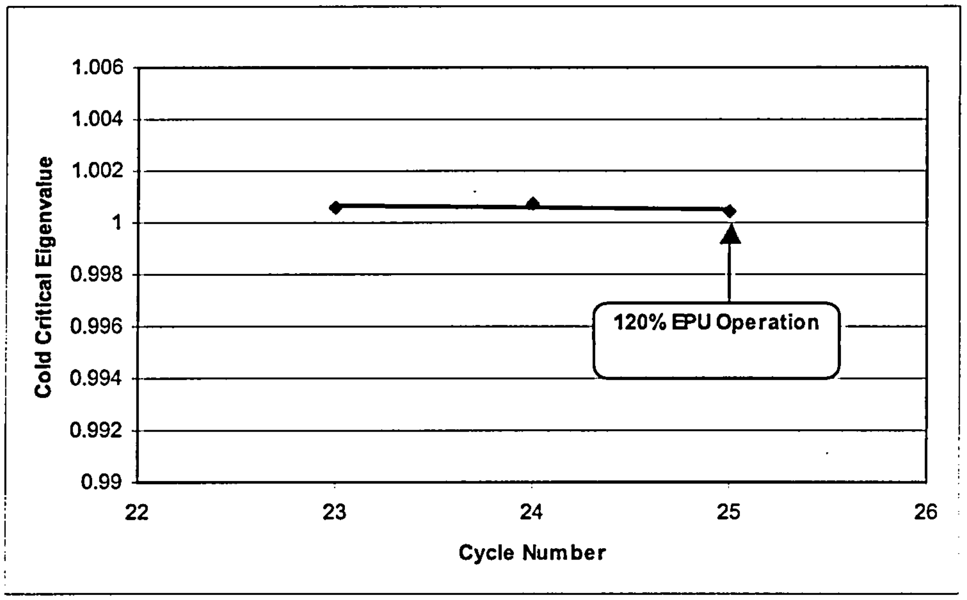

Figure R2.2-1 Sensitivity of Cold Critical Eigenvalue to EPU Operation

NRC RAI 2.3

An equation is provided in Section 2.3.2 stating what the technical specification for cold shutdown requires in terms of ksro and kdemo. Explain the basis for this equation and describe its relationship to the equation relating the SDM calculation to kcrit, ksro, and the period and temperature corrections (e.g., startup control rod withdrawal sequence).

GE Response

The text of interest from Reference 1 states the following:

> The cold shutdown technical specification requires that
>
> $$k_{sro} \leq k_{demo}(1. - 0.0038)$$
>
> where k_{sro} is the calculated criticality for the strongest rod withdrawn condition and 0.0038 is the required shutdown margin.

The derivation is different than that provided previously.

$$\text{SDM} = k_{crit} - k_{sro} - \text{R} + \Delta k_{temp} - \Delta k_{per}$$

Where:
k_{crit} is the multiplication factor for the critical rod pattern;
k_{sro} is the multiplication factor for the strongest rod out;
R is the maximum decrease in SDM throughout the cycle ;
Δk_{temp} is the temperature correction; and
Δk_{per} is the period correction.

The relationship between the two components may be established. SDM at the point in the cycle where the demonstration is performed is

$$\text{SDM} = k_{crit} - k_{sro} - \text{R} + \Delta k_{temp} - \Delta k_{per},$$

and the demonstration of plant criticality is

$$k_{demo} = k_{crit} - \text{R} + \Delta k_{temp} - \Delta k_{per}.$$

Then,

$$\text{SDM} = k_{demo} - k_{sro}$$

$$\frac{\text{SDM}}{k_{demo}} = 1 - \frac{k_{sro}}{k_{demo}}$$

Interpreting the SDM requirement as $\frac{\Delta k}{k} \geq 0.0038$, the following must be true:

$$\frac{\text{SDM}}{k_{demo}} \geq 0.0038$$

Equating the last two relationships results in the original equation.

$$\frac{\text{SDM}}{k_{demo}} = 1 - \frac{k_{sro}}{k_{demo}} \geq 0.0038$$
$$k_{demo} - k_{sro} \geq 0.0038 k_{demo}$$
$$k_{sro} \leq k_{demo}(1 - 0.0038)$$

Considering that $k_{demo} \cong 1$, either interpretation of the SDM requirement is that the strongest rod out is more than 0.38% subcritical.

Reference
28-2. NEDC-33173P "Applicability of GE Methods to Expanded Operating Domains" February 2006.

NRC RAI 3.1, Neutronic Methods

(a) Provide a short description of the methodology used to account for the bypass thermal-hydraulic conditions for transient and stability calculations.

(b) Discuss the accuracy of the assumption that the lattice physics parameters can be characterized as a function of the lattice average moderator density. Discuss the impact of bypass and water rod voiding on lattice depletion. Discuss what impact the presence of bypass voiding (E.g., during RPT) not accounted for in the neutronic methods will have on the core thermal-hydraulic conditions (e.g. power distribution). Discuss the effects of bypass and water rod voiding on lattice power distribution for the exposed fuel.

GE Response

Response to Part (a)

The regular cross section generation process creates homogenized cross sections at many depleted and instantaneous conditions. The effects of reduced moderation due to voiding are calculated by performing lattice physic statepoint analysis of different in-channel void conditions. During this process, the out-channel water and water rod are assumed to have the density of saturated water for hot conditions ($>100^oC$) and the density of solid sub-cooled water for temperatures $<100^oC$.

To accommodate changes in the water rod and bypass water density, the cross sections are then parameterized as a function of node-average relative water density.

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

where

ρ_f is the in-channel density with radial (bundle or channel) and axial dependence,

ρ_{byp} is the axially dependent bypass density,

ρ_o is a standard base density,

A_f is the in-channel flow area

A_{byp} is the out-channel (bypass) flow area

A_{wr} is the water rod flow area

and

the subscripts of *f*, *byp* and *wr* indicate the in-channel, bypass, and water rod regions of the lattice.

During the steady-state or kinetics simulator application, the calculated conditions in the bypass, water rod, and active region are combined to calculate the observed node average relative water density and inquire appropriate cross sections.

$$U_{ijk} = \left(\frac{A_f}{A_f + A_{byp} + A_{wr}}\right)\frac{\rho_{ijk}}{\rho_o} + \left(\frac{A_{byp}}{A_f + A_{byp} + A_{wr}}\right)\frac{\rho_{byp,k}}{\rho_o} + \left(\frac{A_{wr}}{A_f + A_{byp} + A_{wr}}\right)\frac{\rho_{wr,k}}{\rho_o}$$

where

$\rho_{f,ijk}$ is the in-channel density with radial (bundle or channel) and axial dependence,

$\rho_{byp,k}$ is the axially dependent bypass density,

$\rho_{wr,k}$ is the axially dependent water rod density for each bundle modeled.

and

A_f is the in-channel flow area

A_{byp} is the out-channel (bypass) flow area

A_{wr} is the water rod flow area

In the 3D simulator PANACEA, the bypass regions and the water rod regions are combined into a single axial nodalized channel for purposes of modeling moderator density. The in-channel, bypass and water rod regions are then combined as described in the equation above to form the nodal average lattice moderator density.

In the plant transient simulator TRACG, the bypass and water rod regions are treated separately and are nodalized in the axial direction as specified by application. The in-channel, bypass and water rod regions are then combined as described in the equation above to form the nodal average lattice moderator density.

Thus, by the use of the lattice average water density parameter, potential changes in the bypass and water rod voiding (water density) are accurately modeled in the core steady-state and transient simulators.

Response to Part (b)

The presence of bypass and water rod voiding is accounted for in the neutronic methods through the process discussed in the response to RAI 3.1(a). The accuracy of the nodal, axial and radial power distribution is directly related to the ability of the 3D simulators to model the nodal reactivity accurately. In the following discussion, it is shown that the nominal impacts for bypass and water rod voids on the axial power distribution are accurately accounted for in the 3-dimensional steady state and transient methods.

In MFN 05-31 RAI 1.4, the adequacy of the polynomial fitting process under high in-channel voids with and without bypass and water rod voiding was addressed using MCNP. The review in MFN 05-31, RAI 1.4 covered the ability to extrapolate to either the 90% in-channel void without water rod and bypass void state or to the 85% in-channel void with 25% water rod and 10% bypass void state. In this RAI, the error in the reactivity (k-infinity) fit extrapolation from the 0,40, and 70% void fraction base data to the 90% void fraction level was shown to be less than 0.7% for the lattice evaluated. The error associated with the presence of bypass and water rod voiding is less than 0.5% and will not contribute to a significant decrease in the ability of the 3D simulators to predict the axial power distributions. Figure 3.1-1is taken from MFN 05-31, RAI 1.4 for completeness.

Additional evaluations of the accuracy of this assumption for the components of k-infinity are provided to support the accuracy of the lattice average moderator assumption. The component cross sections evaluated are macroscopic thermal absorption (capture + fission), macroscopic thermal fission, macroscopic fast to epi-thermal scattering cross section and the epi-thermal to thermal scattering cross section. The calculated flux ratios are also presented to demonstrate the overall effectiveness of this assumption.

To perform this evaluation, a lattice depletion at a 40% void fraction was perform to create the base data for the instantaneous void evaluation using TGBLA. Using the isotopics generated by the base depletion case, the state points identified in Table 3.1-1, Instantaneous Void Evaluation Conditions were evaluated with MCNP. The instantaneous void data is fit as a function of lattice average moderator density at several exposure points from beginning of lattice life to assumed end of lattice life (65 Gwd/st). The base fits are performed by use of the 0, 40, and 70% void fraction data and these fits are then evaluated at lattice average moderator density values equivalent from a 0% in-channel state to a 90% in-channel without bypass or water rod voiding state to provide the fitted data representation in Figure 3.1-2 through Figure 3.1-7. Two explicit MCNP calculations for high voids with and without bypass voiding at each of the 4 exposure points were then performed to provide the basis for the comparison.

From Figure 3.1-2 through Figure 3.1-7, it can be seen that these significant nodal parameters can be fit and extrapolated with a high degree of accuracy and that the presence of bypass and water rod voiding can be parameterized as overall lattice average moderator density with a high degree of accuracy. No noticeable degradation in the nodal evaluations can be attributed to the presence of bypass and water rod void.

Since the nominal operating core does not experience bypass and water rod voiding and that the core conditions with bypass and water rod voiding are transitory in nature, there will be no significant impact on core depletion simulation.

The lattice physic state point analysis assumes non-voided bypass regions and water rods; therefore, the local pin power distributions do not account for the voided bypass and water rod effects. Evaluations for the impact of the non-voided bypass and water rod assumption show that the uncertainty in the local pin power distribution is small and that the subsequent impact on the R-factor process is small.

In the reactor core, the probability that a node experiencing bypass and water rod voiding is a maximum powered node is extremely small. However, to review the impact of bypass and water rod voiding, a comparison was made between a lattice at 90% in-channel voids without bypass and water rod voiding and a lattice at 85% in-channel voids with 10% bypass voids and 25% water rod voids. This combination of in-channel, bypass and water rod voids produces essentially identical average moderator density. To perform this comparison, a upper zone lattice from a bundle designed for MELLLA+ operation was chosen and the isotopics are based on a 70% in-channel void fraction without water rod and bypass void depletion case.

In Figure 3.1-8and Figure 3.1-9, the impact on local pin fission density is presented. In Figure 3.1-8, the normalized fission density peaking is presented for the lattice at 90% in-channel void fraction without water rod and bypass voiding and for the lattice at 85% in-channel void fraction with 25% water rod and 10% bypass voiding. Figure 3.1-9 contains the delta normalized fission density for four (4) fuel pins for which at some point in the lattice lifetime are the peak powered rod. The contiguous rod peaking is plotted to demonstrate the impact as the peak powered rod changes location as a function of lattice exposure.

[[

]]This impact will not impact the accuracy of the LHGR evaluation in the neutronic methods.

From MFN 05-31 RAI 18, the Figure 3.1-10 below shows that the impact of the voiding of the bypass and water rods has a minimal impact on the value of the R-factor. A bundle that was designed for use in a MELLLA+ core design was used for this evaluation. This comparison is made by using the standard "production" three void points (0,40, and 70%) without bypass and water rod voiding as the base case for the R-factor generation process. The 90VF_axial-4VP model is generated by using four void points at 0, 40, 70, and 90VF without bypass and water rod voiding. The 90VF_20BP-4VF was generated by using 0, 40, and 70 VF without bypass and water rod voiding and a 90 in-channel void with 20% bypass and water rod voiding case for the fourth data point for the R-factor generation process.

As can be seen below, the magnitude of the perturbed R-factor can vary both positive and negative relative to the base "production" R-factor and hence the modeling of bypass and water rod voiding in the R-factor generation process is neither conservative nor non-conservative.

Table 3.1-1: Instantaneous Void Evaluation Conditions

Lattice State	Lattice Exposure (Gwd/st)	In-channel Void (%)	Bypass Void (%)	Water Rod Void (%)
1	0.2,13,65	0	0	0
2	0.2,13,65	40	0	0
3	0.2,13,65	70	0	0
[[				
				]]

[[

Figure 3.1-1: Fit Uncertainty for TGBLA06 Reactivity

]]

Non-Proprietary Version

[[

Figure 3.1-2: Macroscopic Group 3 (thermal) Sigma Absorption

]]

Non-Proprietary Version

[[

Figure 3.1-3: Macroscopic Group 3 (thermal) Sigma Fission

]]

Non-Proprietary Version

[[

Figure 3.1-4: Macroscopic Sigma Slowing Group 1 (Fast) to Group 2 (Epi-thermal)

]]

Non-Proprietary Version

[[

Figure 3.1-5: Macroscopic Sigma Slowing Group 2 (Epi-thermal) to Group 3 (thermal)

]]

Non-Proprietary Version

[[

Figure 3.1-6: Group 1 (Fast) to Group 3 (thermal) Flux Ratio

]]

Non-Proprietary Version

[[

Figure 3.1-7: Group 2 (Epi-thermal) to Group 3 (thermal) Flux Ratio

]]

[[

Figure 3.1-8: Peak Rod Fission Density Impact for Bypass and Water Rod Voiding

]]

Non-Proprietary Version

[[

Figure 3.1-9: Peak Rod Delta Fission Density for Bypass and Water Rod Voids

]]

Non-Proprietary Version

[[

Figure 3.1-10: R-factor Response for 20% Bypass/Water Rod Void Fraction (from MFN 05-133 RAI 18)

]]

NRC RAI 4.2 40 Percent Depletion Assumption and Anticipated Transient Without Scram (ATWS)

a. Address the impact of the 40 percent depletion assumption on the ATWS response.

GE Response

The 40% void depletion assumption can affect the void coefficient. The effect of void coefficient uncertainty has been addressed in previous studies. With respect to ATWS Overpressure results, uncertainty screening was performed in NEDE-32906P Supplement 1-A (MFN 03-148, November 26, 2003). The initial conditions for this study were for a plant at 113% of original rated power and 73% core flow, which are MELLLA+ type of conditions.

For an ATWS event, the steam line isolation causes a rapid increase in reactor vessel pressure, which results in core void reduction. Consequently, power increases with positive void reactivity insertion. For ATWS simulation purposes, the expected MSIV position and high flux scrams do not occur. The power excursion is initially mitigated by void production from the increased core heat flux, as well as negative doppler reactivity from increasing fuel temperature. Soon after the time the MSIVs are fully closed, Recirculation Pump Trip (RPT) is initiated on high pressure, such that core flow begins to decrease. At about this same time, the Safety/Relief Valves (S/RVs) open, reducing the rate of pressure increase. As core flow continues to decrease, core voiding increases, causing the power to decrease in parallel. Finally, the steam production decreases to the point at which the S/RV capacity is sufficient to relieve all of the steam generation, and the pressure begins to fall. Figures 4.2a and 4.2b show the response of key parameters for this event. These figures also contain the results for the void coefficient perturbation.

Analyses have been performed at +/- 1σ level for each of the model uncertainties. The results of the screening are shown in Figure 4.2c.

The analysis results show that the peak pressure results are [[

]]

Analyses have also been performed for another EPU plant with ODYN with a core-wide [ ]] increase in ODYN void coefficient magnitude. The results are presented in

Table 4.2a for BOC and EOC conditions. [[

]]

In addition, the effect on the peak pool temperature response is also addressed. Sensitivity studies have been performed with a core-wide [[ ]] increase in the ODYN void coefficient magnitude. A sensitivity study was performed for a limiting Pressure Regulator Failue – Open (PRFO) at both BOC and EOC exposure conditions. The results shown in Table 4.2b below show that the peak pool temperature is [[

]].

Table 4.2a ODYN Peak Vessel Pressure Void Coefficient Study

Event and Description	Exposure	Peak Vessel Pressure (psig)
PRFO Base Case	BOC	[[
PRFO with 10% void coefficient increase	BOC	
PRFO Base Case	EOC	
PRFO with 10% void coefficient increase	EOC	]]

Table 4.2b Suppression Pool Peak Temperature Void Coefficient Study

Event and Description	Exposure	Peak Suppression Pool Temperature (°F)
PRFO Base Case	BOC	[[
PRFO with 10% void coefficient increase	BOC	
PRFO Base Case	EOC	
PRFO with 10% void coefficient increase	EOC	]]

[[

]]

Figure 4.2a. TRACG Power and Flow Response for MSIVC Event

Non-Proprietary Version

[[

Figure 4.2b. TRACG Pressure and Relief Valve Response for MSIVC Event

]]

[[

]]

Figure 4.2c MSIVC ATWS Peak Vessel Pressure Sensitivity to Individual Uncertainties (Pcase-Pnominal [kPa]

NRC RAI 6

The changes to the LTR proposed by the RAI 6 responses are reflected in the attached LTR labeled Revision 1 RAI 6 Draft and shown by revisions bars (Attachment 1 to Enclosure 1). The LTR will be formally issued, reflecting these and other required changes, approximately 2 weeks after the closure of the methods related RAIs supporting the review of Tennessee's Valley Authority's license change request for an extended power uprate.

NRC RAI 6.1

The LTR summarizes the content of the VY RAIs. However, this eliminates relevant figures and evaluations. For the void fraction correlation, void reactivity coefficient, and Option 1D include the relevant figures and discussions so that the supporting information is integrated in this LTR.

GE Response

The relevant figures, tables, and discussion from the VY RAIs have been incorporated into the body of the LTR. Appropriate references have also been included.

NRC RAI 6.2
Appendix A contains many RAIs not related to the methods review. All EPU SRXB-A RAIs were cited in Appendix A. Many of these RAIs, did not address nor are they relevant to the Methods review. This array of RAIs hampers efficient use of the reference material. Delete the SRXB-A RAIs that were not part of the methods review.

GE Response
The table in Appendix A will be reduced to include only the VY RAIs that are related to the methods review.

In addition, because the VY RAIs in Appendix B are grouped and formatted according to the VY Supplemental submittals, the removal of individual RAIs would result in the section being fragmented and difficult to follow. GE believes that Appendix B is no longer an essential part of the Interim Methods LTR and, therefore, proposes its removal.

See the attached LTR for the revised Appendix A.

NRC RAI 6.3, EPU Maximum Bundle Operating Conditions

Vermont Yankee SRXB-A Figures 6-1 thru 6-6 (Reference 1) show the maximum bundle operating conditions of high density and EPU plants. Each plant specific application should, include the plant-specific data in the plots containing the high density and EPU plants maximum bundle operating conditions (Attachment 3, BVY 05-024)

(a) Therefore, include in the EPU applications the following bundle operating conditions with exposure in the EPU maximum bundle operating condition plots:
maximum bundle power,
maximum bundle power/flow ratio,
exit void fraction of maximum power bundle,
maximum channel exit void fraction,
peak linear heat generation rate and
peak end-of-cycle nodal exposure

(b) Provide quarter core map (assuming core symmetry) showing the bundle operating linear heat generation (MLHGR) and the minimum critical power ratio (MCPR) for beginning-of-cycle (BOC), middle-of-cycle (MOC) and end-of-cycle (EOC). Similarly, show the associated bundle powers.

GE Response

Section 4.3 of the Methods LTR will be modified, to specify that the requested core operating information be included with plant specific applications of the Methods LTR.

See the attached LTR for the revised Section 4.3.

ENCLOSURE 3

MFN 06-195

Affidavit

General Electric Company

AFFIDAVIT

I, Louis M. Quintana, state as follows:

(1) I am Manager, Licensing, General Electric Company ("GE"), have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 to GE letter MFN 06-195, Louis Quintana to NRC Document Control Desk, *Responses to Methods RAIs - Interim Methods LTR*, dated June 23, 2006. The proprietary information in Enclosure 1, *GE Responses to RAIs Related to NEDE-33173P*, is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions regarding GE Methods supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for the expanded power/flow operating domains including Extended Power Uprates, Constant Pressure Power Uprates, and the MELLLA+ domain for a GE BWR, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology

base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 23rd day of June 2006.

Louis M. Quintana

Louis M. Quintana
General Electric Company