GE Energy

James C. Kinsey
Project Manager, ESBWR Licensing

PO Box 780 M/C J-70
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

Proprietary Notice
This letter forwards GNF proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 1, the balance of this letter may be considered non-proprietary.

MFN 06-297, Supplement 4

Docket No. 52-010

January 26, 2007

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

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01 - Supplement**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

Enclosure 1 contains GNF proprietary information as defined by 10 CFR 2.390. GNF customarily maintains this information in confidence and withholds it from public disclosure. A non-proprietary version is provided in Enclosure 2.

The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GNF. GE hereby requests that the information of Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

D068

If you have any questions about the information provided here, please let me know.

Sincerely,

James C. Kinsey

James C. Kinsey
Project Manager, ESBWR Licensing

Reference:

1. MFN 06-288, Letter from U. S. Nuclear Regulatory Commission to Mr. David H. Hinds, *Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application,* August 16, 2006

Enclosures:

1. MFN 06-297, Supplement 4 - Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01 – Supplement – GNF Proprietary Information
2. MFN 06-297, Supplement 3 - Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01 – Supplement – Non Proprietary Version
3. Affidavit – Jens G. M. Andersen – dated January 26, 2007

cc: AE Cubbage USNRC (with enclosures)
AA Lingenfelter GNF/Wilmington (w/o enclosures)
GB Stramback GE/San Jose (with enclosures)
eDRFs 0063-2208 for 4.2-5-4.2-7 S01
0063-1801 for 4.4-46 S01

Enclosure 2

MFN 06-297, Supplement 4

Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application

DCD Chapter 4 and GNF Topical Reports

RAI Numbers 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01 Supplement

Non-Proprietary Version

NRC RAI 4.2-5 S01:

DCD Tier 2, Appendix 4B.2 should define the specific Tier 2 and Tier 2 thermal mechanical fuel design requirements. These requirements would then be addressed within a separate fuel assembly mechanical design topical report to demonstrate, using approved models and methods, the acceptability of a proposed fuel assembly design to the ESBWR. The specific thermal-mechanical design requirements may be patterned after the standard review plan. The current text appears to be an overview of a fuel design change process and should be removed.*

GE Response:

The current Appendix 4B will be revised to remove all of the design process information. Three sections remain in Appendix 4B: 4B.1 Thermal-Mechanical; 4B.2 Nuclear; and 4B.3 Critical Power Correlation. The current Section 4B.2 becomes Section 4B.1. While Appendix 4B is referenced only by Section 4.2, Fuel Design, the change criteria for the nuclear core design, and critical power correlation should also be defined as Tier 2* parameters. Thus, Section 4B.2 and 4B.3 provide the appropriate Tier 2* criteria for core design and critical power correlation changes prior to the plant first achieving full power.

DCD Impact:

The proposed Appendix 4B is attached.

Proposed Appendix 4B Changes

4B. FUEL LICENSING ACCEPTANCE CRITERIA

The fuel licensing acceptance criteria are presented in the following subsections.

4B.1 THERMAL-MECHANICAL

A set of design limits are defined, and applied in the fuel rod thermal-mechanical design analyses, to ensure that fuel rod mechanical integrity is maintained throughout the fuel rod design lifetime. The design criteria were developed by GNF and other specific industry groups to focus on the parameters most significant to fuel performance and operating occurrences that can realistically limit fuel performance. The specific criteria are patterned after ANSI/ANS-57.5-1981 (Reference 4B-1) and NUREG-0800 Rev. 2 (Reference 4B-2). Table 4B.1-1 presents a summary of the design criteria. The bases for the design criteria listed in Table 4B.1-1 are presented below.

Table 4B.1-1 Fuel Rod Thermal-Mechanical Design Criteria

Criterion	Governing Equation
1. [*The cladding creepout rate (&cladding creepout), due to fuel rod internal pressure, shall not exceed the fuel pellet irradiation swelling rate (&fuel swelling).*]*	$\&_{cladding\ creepout} \leq \&_{fuel\ swelling}$
2. [*The maximum fuel center temperature (T_{center}) shall remain below the fuel melting point (T_{melt}).*]*	$T_{center} < T_{melt}$
3. [*The cladding circumferential plastic strain (ε_θ^P) during an anticipated operational occurrence shall not exceed 1.00%*]*.	$\varepsilon_\theta^P \leq 1.00\%$
4. [*The fuel rod cladding fatigue life usage ($\sum_i \frac{n_i}{n_f}$ where n_i=number of applied strain cycles at amplitude ε_i and n_f=number of cycles to failure at amplitude ε_i) shall not exceed the material fatigue capability.*]*	$\sum_i \frac{n_i}{n_f} \leq 1.0$
5. [*Cladding structural instability, as evidenced by rapid ovality changes, shall not occur.*]*	No creep collapse
6. [*Cladding effective stresses (σ_e)/strains(ε_e) shall not exceed the failure stress(σ_f)/strain(ε_f).*]*	$\sigma_e < \sigma_f, \quad \varepsilon_e < \varepsilon_f$
7. [*The as-fabricated fuel pellet evolved hydrogen (C_H is content of hydrogen) at greater than 1800 ℃ shall not exceed prescribed limits.*]*	$C_H \leq$ *Manufacturing Specifications*

Cladding Lift-Off / Fuel Rod Internal Pressure (Item 1 of Table 4B.1-1)

The fuel rod is filled with helium during manufacture to a specified fill gas pressure. With the initial rise to power, this fuel rod internal pressure increases due to the corresponding increase in the gas average temperature and the reduction in the fuel rod void volume due to fuel pellet expansion and inward cladding elastic deflection due to the higher reactor coolant pressure. With continued irradiation, the fuel rod internal pressure will progressively increase further due to the release of gaseous fission products from the fuel pellets to the fuel rod void volume. With further irradiation, a potential adverse thermal feedback condition may arise due to excessive fuel rod internal pressure.

In this case, the tensile cladding stress resulting from a fuel rod internal pressure greater than the coolant pressure causes the cladding to deform outward (cladding creep-out). If the rate of the cladding outward deformation (cladding creep-out rate) exceeds the rate at which the fuel pellet expands due to irradiation swelling (fuel swelling rate), the pellet-cladding gap will begin to open (or increase if already open). An increase in the pellet-cladding gap will reduce the pellet-cladding thermal conductance thereby increasing fuel temperatures. The increased fuel temperatures will result in further fuel pellet fission gas release, greater fuel rod internal pressure, and correspondingly a faster rate of cladding creep-out and gap opening.

This potential adverse thermal feedback condition is avoided by limiting the cladding creep-out rate, due to fuel rod internal pressure, to less than or equal to the fuel pellet irradiation swelling rate. This is confirmed through the calculation of a design ratio (of internal pressure to critical pressure) and ensuring that the calculated design ratio is less than 1.00 at any point in time for all fuel rod types.

Fuel Temperature (Melting, Item 2 of Table 4B.1-1)

Numerous irradiation experiments have demonstrated that extended operation with significant fuel pellet central melting does not result in damage to the fuel rod cladding. However, the fuel rod performance is evaluated to ensure that fuel melting will not occur. To achieve this objective, the fuel rod is evaluated to ensure that fuel melting during normal steady-state operation and whole core anticipated operational occurrences are not expected to occur.

Cladding Strain

After the initial rise to power and the establishment of steady-state operating conditions, the pellet-cladding gap will eventually close due to the combined effects of cladding creep-down, fuel pellet irradiation swelling, and fuel pellet fragment outward relocation. Once hard pellet-cladding contact (PCMI) has occurred, cladding outward diametral deformation can occur. The consequences of this cladding deformation are dependent on the deformation rate (strain rate).

High Strain Rate (Anticipated Operational Occurrences, Item 3 of Table 4B.1-1)

Depending on the extent of irradiation exposure, the magnitude of the power increase, and the final peak power level, the cladding can be strained due to the fuel pellet thermal expansion occurring during rapid power ramps. This high strain rate deformation can be a combination of (a) plastic deformation during the power increase due to the cladding stress exceeding the cladding material yield strength, and (b) creep deformation during the elevated power hold time due to creep-assisted relaxation of the high cladding stresses. This cladding permanent (plastic plus creep) deformation during anticipated operational occurrences is limited to a maximum of 1.00%.

In non-barrier cladding, fast power ramps can also cause a chemical/mechanical pellet cladding interaction commonly known as PCI/SCC. To prevent PCI/SCC failures in non-barrier cladding, reactor operational restrictions must be imposed. To eliminate PCI/SCC failures without imposing reactor operational restrictions, GNF invented and developed barrier cladding. Barrier cladding utilizes a thin zirconium layer on the inner surface of Zircaloy tubes. The minimum thickness of the zirconium layer is specified to ensure that small cracks which are known to initiate on the inner surface of barrier cladding (the surface layer subject to hardening by absorption of fission products during irradiation) will not propagate through the zirconium barrier into the Zircaloy tube. The barrier concept has been demonstrated by experimental irradiation testing and extensive commercial reactor operation to be an effective preventive measure for PCI/SCC failure without imposing reactor operating restrictions.

Low Strain Rate (Steady-State Operation, no limit in Table 4B.1-1)

During normal steady-state operation, once the cladding has come into hard contact with the fuel, subsequent fuel pellet irradiation swelling causes the cladding to deform gradually outward. The fuel pellet swelling rate is very slow. The effect of this slow fuel pellet expansion is the relaxation of low stresses imposed by the fuel swelling, resulting in a low strain-rate outward creep deformation of the cladding. Similarly, when the fuel rod internal pressure exceeds the external pressure exerted by the reactor coolant, the cladding will also slowly creep outward. Under both of these conditions, irradiated Zircaloy exhibits substantial creep ductility. Therefore, no specific limit is applied to low-strain rate cladding deformation.

Dynamic Loads / Cladding Fatigue (Item 4 of Table 4B.1-1)

As a result of normal operational variations, cyclic loadings are applied to the fuel rod cladding by the fuel pellet. Therefore, the fuel rod is evaluated to ensure that the cumulative duty from cladding strains due to these cyclic loadings will not exceed the cladding fatigue capability. The Zircaloy fatigue curve employed represents a statistical lower bound to the existing fatigue experimental measurements. The design limit for fatigue cycling, to assure that the design basis is met, is that the value of calculated fatigue usage must be less than the material fatigue capability (fatigue usage < 1.0).

Elastic Buckling / Cladding Creep Collapse (Item 5 of Table 4B.1-1)

The condition of an external coolant pressure greater than the fuel rod internal pressure provides the potential for elastic buckling or possibly even plastic deformation if the stresses exceed the material yield strength. Fuel rod failure due to elastic buckling or plastic collapse has never been observed in commercial nuclear reactors. However, a more limiting condition that has been observed in commercial nuclear reactors is cladding creep collapse. This condition occurs at cladding stress levels far below that required for elastic buckling or plastic deformation. In the early 1970s, excessive in-reactor fuel pellet densification resulted in the production of large fuel column axial gaps in some PWR fuel rods. The high PWR coolant pressure in conjunction with thin cladding tubes and low helium fill gas pressure resulted in excessive fuel rod cladding creep and subsequent cladding collapse over fuel column axial gaps. Such collapse occurs due to a slow increase of cladding initial ovality due to creep resulting from the combined effect of reactor coolant pressure, temperature and fast neutron flux on the cladding over the axial gap. Since the cladding is unsupported by fuel pellets in the axial gap region, the ovality can become large enough to result in elastic instability and cladding collapse.

Fuel Rod Stresses (Item 6 of Table 4B.1-1)

The fuel rod is evaluated to ensure that fuel rod failure will not occur due to stresses or strains exceeding the fuel rod mechanical capability. In addition to the loads imposed by the difference between the external coolant pressure and the fuel rod internal gas pressure, a number of other stresses or strains can occur in the cladding tube. These stresses or strains are combined through application of the distortion energy theory to determine an effective stress or strain. The applied limit is patterned after ANSI/ANS-57.5-1981 (Reference 4B-1). The figure of merit employed is termed the Design Ratio where:

$$Design\ Ratio = \frac{Effective\ Stress}{Stress\ Limit}\ or\ \frac{Effective\ Strain}{Strain\ Limit}$$

The stress or strain limit is the failure stress or strain. The value of the Design Ratio must be less than 1.00.

Fuel Rod Hydrogen (Item 7 of Table 4B.1-1)

GNF experience has demonstrated that excessive fuel rod internal hydrogen content due to hydrogenous impurities can result in fuel rod failure due to localized hydriding. The potential for primary hydriding fuel rod failure is limited by the application of specification limits on the fuel pellets in conjunction with fabrication practices that eliminate hydrogenous contaminants from all sources during the manufacturing process.

4B.2 NUCLEAR

[*A negative Doppler reactivity coefficient is maintained for any operating condition.*]* The Doppler reactivity coefficient is of high importance in reactor safety. The Doppler coefficient of the core is a measure of the reactivity change associated with an increase in the absorption of resonance-energy neutrons caused by a change in the temperature of the material and is a function of the average of the bundle Doppler coefficients. A negative Doppler coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on a gross or local basis and thus assures the tendency of self-control.

[*A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels is maintained for any operating conditions.*]* The core moderator void coefficient resulting from boiling in the active flow channels is maintained negative over the complete range of ESBWR operation. This flattens the radial power distribution and provides ease of reactor control due to the negative void feedback mechanism.

[*A negative moderator temperature reactivity coefficient is maintained for temperatures equal to or greater than hot standby.*]* The moderator temperature coefficient is associated with a change in the moderating capability of the water. Once the reactor reaches the power producing range, boiling begins and the moderator temperature remains essentially constant. The moderator temperature reactivity coefficient is negative during power operation.

[*To prevent a super prompt critical reactivity insertion accident originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel is negative.*]* The mechanical and nuclear designs of the fuel are such that the prompt reactivity feedback (requiring no conductive or convective heat transfer and no operator action) provides an automatic shutdown mechanism in the event of a super prompt reactivity incident. This characteristic ensures rapid termination of super prompt critical accidents, with additional long-term shutdown capability due to negative void coefficient, for those cases where conductive heat transfer from the fuel to the water results in boiling in the active channel region.

[*A negative power reactivity coefficient (as determined by calculating the reactivity change due to an incremental power change from a steady-state base power level) is maintained for all operating power levels above hot standby.*]* A negative power coefficient provides an inherent negative feedback mechanism to provide more reliable control of the plant as the operator performs power maneuvers. It is particularly effective in preventing xenon initiated power oscillations in the core. The power coefficient is effectively the combination of Doppler, void and moderator temperature reactivity coefficients.

[*The core is capable of being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive control rod, or rod pair, in the full-out position and all other rods fully inserted.*]* This parameter is dependent upon the core loading and is calculated for each plant cycle prior to plant operation of that cycle.

4B.3 CRITICAL POWER CORRELATION

[*The currently approved critical power correlation will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry in the active region of the assembly; this specifically includes fuel and water rod diameter, channel sizing and spacer design.*]*

The criteria for establishing the new correlation are as follows:

- The new correlation shall be based on full-scale prototypical test assemblies.
- Tests shall be performed on assemblies with typical rod-to-rod peaking factors.
- The functional form of the currently approved correlations shall be maintained.
- Correlation fit to data shall be best fit.
- The correlation's range of application shall be determined.
- One or more additional assemblies must be tested to verify correlation accuracy (i.e., test data not used to determine the new correlation coefficients).
- The uncertainty of the resulting correlation shall be determined and included in establishing the operating limits.

The basis of the correlation is a best fit of data taken of prototypical test assemblies with typical rod-to-rod peaking factors.

4B.4 COL INFORMATION

None.

4B.5 REFERENCES

4B-1 American National Standard for Light Water Reactors Fuel Assembly Mechanical Design and Evaluation, American Nuclear Society Standards Committee Working Group ANS 57.5, ANSI/ANS-57.5-1981.

4B-2 US Nuclear Regulatory Commission Standard Review Plan 4.2 – Fuel System Design, (USNRC SRP 4.2), NUREG-0800 Rev. 2, July 1981.

NRC RAI 4.2-6 S01:

DCD Tier 2, Appendix 4B.2 states, "For local AOOs such as rod withdrawal error, a small amount of calculated fuel pellet centerline melting may occur, but is limited by the 1% cladding circumferential plastic strain criterion." The staff has concerns with the ability to accurately model fuel volumetric expansion as fuel enthalpy approached incipient melt temperatures, and the ability to accurately model the evolved fuel pellets in future operation.

(a) *Demonstrate that the fuel thermal expansion/swelling model is capable of accurately predicting volumetric expansion during rapid power changes and at temperatures (1) approaching Tmelt and (2) exceeding Tmelt. Include a discussion of the models ability to predict fission-product-induced swelling. Provide supporting empirical database, especially test results on irradiated fuel rods.*

(b) *Demonstrate that all of the fuel performance models (e.g. conductivity, expansion, relocation, FGR, grain growth, etc.) remain valid and within their original accuracy for simulating evolved fuel (having undergone partial melt) during future operation including AOOs. Provide supporting empirical database, especially test results on irradiated fuel rods.*

GE Response:

10 CFR 50 Appendix A provides an explicit definition of an AOO. 10 CFR 50 Appendix A states "Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power." The ESBWR design life is 60 years, and thus, any abnormal event with a probability ≥ 1/60 per year must be classified as an AOO, and conversely, any abnormal event with a probability < 1/60 per year should not be classified as an AOO. However, Subsection 15.0.1.2 conservatively defines an AOO "any abnormal event that has an event probability of ≥ 1/100 per year."

From Table 15A-3, the most likely RWE has a probability of 1/1000 per year (1 Event in 1,000 yrs). Therefore, the RWE is correctly classified as an infrequent event in Chapter 15, and Tables 15.0-2, 15.0-7 and 15A-3 in Chapter 15 are correct.

DCD Tier 2, Section 4.2 will be modified as noted in the response to RAI 4.2-5. The proposed response will remove the allowance of fuel melting in steady state and AOOs.

Because of the changes above, the data requested in (a) and (b) is not required and is not included in this response.

DCD Impact:

DCD Tier 2, Section 4.2 will be modified as noted in the response to RAI 4.2-5 S01.

NRC RAI 4.2-7 S01:

DCD Tier 2, Appendix 4B.5 states, "99.9% of the rods in the core must be expected to avoid boiling transition for core-wide incidents of moderate frequency..." This criteria differs from GESTAR-II which states, "Ninety-nine point nine percent (99.9%) of the rods in the core must be expected to avoid boiling transition."

(a) *Discuss the basis for this change.*

(b) *Identify AOOs not characterized as "core-wide" and the criteria used to evaluate each.*

(c) *Distinguish between events classified as moderate frequency and those classified as less frequent.*

GE Response:

Please see the response to RAI 4.2-6 for discussion of the characterization of events. With the response to RAI 4.2-5, Appendix 4B will be rewritten. The language above will be completely removed, because it is already covered in Chapter 15 of the DCD.

DCD Impact:

DCD Tier 2, Appendix 4B will be rewritten as described in the revised response to RAI 4.2-5 S01.

NRC RAI 4.4-46 S01:

From Fuels Audit 10/23 - 10/31

The response to RAI 4.4-45 included a qualitative discussion providing the reason for the observed trend in calculated hot eigenvalue. Similarly, these phenomena result in calculated trends for cold eigenvalue as discussed in the response to RAI 4.4-46. The response indicated that the trends are consistent across several reactor cores and cycles. The response also indicates that a database of mid-cycle plant data is used to predict cold eigenvalue trends. Provide more descriptive details of this database, including the range of core sizes, fuel types, exposure, and power levels. Additionally, provide the calculation models employed, and a description of their implementation in cycle calculations to account for the trends in both hot and cold eigenvalue predictions. Explain how BOC eigenvalue is incorporated into the best estimate prediction of the cold eigenvalue trend. Explain the nature of any conservatism in the applied methods for both hot and cold eigenvalue trends. Clarify if the accounting methods for these trends were implemented in the calculations provided for the ESBWR in Tier 2, Section 4.3, including cycle tracking calculations, ratio of operating limit critical power ration (CPR) to CPR (CPRRAT) predictions, maximum fraction of limiting power density (MFLD) predictions and shutdown margin. Clarify whether the trend accounting methods are an integral part of the PANACEA code.

GE Response:

The cold critical eigenvalue database of [[ ]] is an accumulation of operating reactor data from a variety of core sizes and power levels, and encompasses BWR/2-6 reactor operation. The data is predominately 10x10 fuel since this has been GE/GNF's fuel design for the past several years. The database represents cycle exposures from [[ ]] of cycle exposure.

All the data used in this assessment of cold eigenvalue trends, as well as that presented in Figures 1-26 and 1-27 of NEDC-33239P, utilized the PANAC11 version of the PANACEA three-dimensional core simulator. The simulation (or core tracking) of actual reactor operation is performed utilizing hot operating statepoints to step through the operating cycle. The cold critical states are simulated by restarting from the hot operating simulation at the appropriate cycle exposure and evaluating the cold critical condition. The resultant eigenvalues obtained from the PANACEA simulation of the critical conditions (both hot and cold) form the basis for prediction of expected eigenvalue trends.

As was stated in the original response, the plant will always perform a BOC critical, both to satisfy the Technical Specification shutdown margin demonstration requirement and as the initial step in bringing the reactor to power. The predicted cold eigenvalue at BOC is established relying heavily on this previous cycle(s) BOC information. Having established the BOC cold eigenvalue based on plant data, the change in this cold eigenvalue with cycle exposure is determined by applying the generic trend established from the mid-cycle cold critical database discussed in the first paragraph.

Both hot and cold eigenvalues are established on a best estimate basis. That is, the design engineer, using the operating plant data, selects a set of eigenvalues that best reflect expected performance. Reactivity design margins for both hot operating reactivity and cold shutdown margin are to be met when establishing the core loading to provide adequate assurance that the plant will be able to operate safely and meet the shutdown margin demonstration requirements with a high degree of confidence, even accounting for the uncertainties associated with eigenvalue selections. Having stated that eigenvalues are selected on a best estimate basis, two areas on modest conservatism are often included in the selection. First, the cold eigenvalue trend through the cycle discussed above has been established as a best estimate, but slightly conservative, trend based on the operating database. Secondly, although the design engineer selects a best estimate set of eigenvalues, there is usually a degree of uncertainty in the interpretation of the operating data [[ ]]. In such situations the engineer will often gravitate to the more conservative value within the best estimate selection range. For this reason it is not uncommon for a small conservative bias [[ ]] to exist when comparing predicted eigenvalues to actual plant criticality.

The selection of eigenvalues for the ESBWR design work was done using the best estimate approach discussed above. Because operating data for ESBWR plants do not exist, the eigenvalue selections were based on typical operating BWR/2-6 data. Extra design margin for shutdown margin was maintained in the Tier 2, Section 4.3 work, in part to account for the added uncertainty inherent in the ESBWR eigenvalue selection. The rod pattern depletions through the cycle and the associated thermal limits results (critical power ratio and linear heat generation rate) were based on these best estimate eigenvalue selections. Variations in hot eigenvalues from nominal will result in rod pattern adjustments to maintain criticality. For eigenvalue differences on the order of [[ ]], the rod pattern adjustments and associated changes in thermal margins are generally not a major impact. As more modern initial core reactor startups are achieved (involving ABWR and 10x10 fuel designs) GE/GNF will have opportunity to revisit the eigenvalue selections made for the ESBWR design work.

The method of selecting hot and cold critical eigenvalues is external to PANACEA. Although PANACEA simulation of operating plant performance is the key input to selecting critical eigenvalues, the establishment of critical eigenvalues for future design work is done by the design engineer in conjunction with his or her engineering peers by using the PANACEA results and engineering judgment.

DCD Impact:

No changes to the Tier 1 or Tier 2 sections of the DCD are required.

Enclosure 3

MFN 06-297, Supplement 4

Affidavit

Affidavit

I, Jens G. M. Andersen, state as follows:

(1) I am Consulting Engineer, Thermal Hydraulic Methods, Global Nuclear Fuel – Americas, L.L.C. ("GNF-A") and 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 of GE letter MFN 06-297, Supplement 4, James C. Kinsey to U. S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports - RAI Numbers 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01 – Supplement* dated January 26, 2007. The proprietary information in Enclosure 1, *MFN 06-297, Supplement 4 Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports - RAI Numbers 4.2-5 S01, 4.2-6 S01, 4.2-7 S01 and 4.4-46 S01 – Supplement – GNF Proprietary Information,* is delineated by double underlined dark red font text and is enclosed inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. 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 or licensee, GNF-A 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 GNF-A's competitors without license from GNF-A 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 GNF-A customer–funded development plans and programs, of potential commercial value to GNF-A;

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 the 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 GNF-A, and is in fact so held. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in (6) and (7) following. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, 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.

(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, or subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A 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 GNF-A 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) is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology.

The development of the methods used in these analyses, along with the testing, development and approval of the supporting methodology was achieved at a significant cost, on the order of several million dollars, to GNF-A or its licensor.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit–making opportunities. The fuel design and licensing methodology is part of GNF-A'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 GNF-A or its licensor.

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.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A 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 GNF-A 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 GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining 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 at Wilmington, North Carolina this 26th day of January 2007.

Jens GM Andersen

Jens G. M. Andersen
Global Nuclear Fuels – Americas, LLC