HITACHI

GE Hitachi Nuclear Energy

**Richard E. Kingston**
Vice President, ESBWR Licensing

PO Box 780 M/C A-65
Wilmington, NC 28402-0780
USA

T 910 819 6192
F 910 362 6192
rick.kingston@ge.com

**Proprietary Notice**
This letter forwards proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosures 1 and 4, the balance of this letter may be considered non-proprietary.

MFN 06-297 Supplement 9 Docket No. 52-010

April 19, 2010

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

Subject: **Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8**

The purpose of this letter is to submit a supplement to the GE Hitachi Nuclear Energy (GEH) response (Reference 2) to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by the Reference 1 NRC letter. GEH supplemental response to RAI Number 4.8-8 is addressed in Enclosure 1.

Enclosures 1 and 4 contain Global Nuclear Fuels (GNF) proprietary information as defined by 10 CFR 2.390. GNF customarily maintains this information in confidence and withholds it from public disclosure. Enclosures 2 and 5 are the respective non-proprietary versions, which do not contain proprietary information and are suitable for public disclosure.

The affidavit contained in Enclosure 6 identifies that the information contained in Enclosures 1 and 4 has been handled and classified as proprietary to GNF. GNF hereby requests that the information of Enclosures 1 and 4 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

If you have any questions or require additional information, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

D068
NRO

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

2. MFN 06-297, Letter from David H. Hinds 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 4.2-2through 4.2-7, 4.3-3, 4.3-4, 4.4-2, 4.4-5, 4.4-6, 4.4-15 through 4.4-17,4.4-19, 4.4-24, 4.4-27, 4.4-31 through 4.4-34, 4.4-36, through 4.4-38,4.4-42 through 4.4-50, 4.4-52 through 4.4-56, 4.8-1 through 4.8-16,* August 23, 2006

Enclosures:

1. MFN 06-297 Supplement 9 - Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – GNF-A Proprietary Information
2. MFN 06-297 Supplement 9 - Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – Public Version
3. MFN 06-297 Supplement 9 - Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – DCD Tier 1 and Tier 2 Markups
4. MFN 06-297 Supplement 9 - Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – LTR NEDC-33240P Markups – GNF-A Proprietary Information
5. MFN 06-297 Supplement 9 - Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – LTR NEDO-33240 Markups – Public Version
6. MFN 06-297 Supplement 9 - Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – Affidavit

| | | |
|---|---|---|
| cc: | AE Cubbage | USNRC (with enclosures) |
| | JG Head | GEH (with enclosures) |
| | DH Hinds | GEH (with enclosures) |
| | SC Moen | GEH (with enclosures) |
| | eDRF | 0000-0116-6195 |

# Enclosure 2

# MFN 06-297 Supplement 9

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

# Reactor

# RAI Number 4.8-8

# Public Version

## NON-PROPRIETARY INFORMATION 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 an open and closed bracket as shown here [[ ]].

**Update for Fuel Lift and Seismic and Dynamic Loads**

GE14 fuel assemblies for BWR/4-6 have been demonstrated to be acceptable for the following peak seismic and dynamic accelerations: [[ ]] (see NRC RAI 4.8-8, MFN 06-297 Enclosure 1). Due to the similarities between GE14E and GE14, and due to the shorter ESBWR fuel assembly length, the GE14E fuel assemblies are capable of withstanding accelerations greater than those for GE14 fuel assemblies.

ESBWR standard plant seismic analysis shows peak SSE accelerations of [[ ]]. These accelerations are less than the demonstrated capability of the GE14 fuel. The shorter ESBWR fuel assembly length results in additional margin to the seismic and dynamic load criteria for GE14E fuel. It is concluded that GE14E fuel assemblies, including spacers, are qualified for the seismic and dynamic loads defined by the ESBWR standard plant seismic analysis.

The combined fuel lift and seismic and dynamic load analysis is identified in NEDC-33240P Rev 1 as an analysis that needs to be performed prior to fuel release for application. An ITAAC is added as shown in the attached DCD markups to ensure this evaluation is completed.

Section 3.4.1.11 is changed to include the seismic and dynamic loading acceptance criteria, clarify the applicability of the criteria to GE14E, describe the GE14 seismic design basis loads, and their conservative application to GE14E. Additional changes are made to compare the currently defined GE14E seismic and dynamic loads to those used to qualify the GE14E fuel assembly components and to clarify that all GE14E components have been qualified for the GE14 design basis loads.

**DCD Impact**

ESBWR DCD Tier 1 and Tier 2 Chapter 4 will be revised as shown in the attached markups.

**LTR Update (NEDC-33240P, Rev 2)**

Updates to section 3.4.1.11 and section 5 are attached.

# Enclosure 3

# MFN 06-297 Supplement 9

# Supplement to Response to Portion of NRC Request for

# Additional Information Letter No. 53

# Related to ESBWR Design Certification Application

# Reactor

# RAI Number 4.8-8

# DCD Tier 1 and Tier 2 Markups

are fabricated, installed, and inspected in accordance with ASME Code Section III requirements.

(1) Pressure boundary welds in the RPV meet ASME Code Section III non-destructive examination requirements.

(2) The RPV retains its pressure boundary integrity at its design pressure.

(3) The equipment identified in Table 2.1.1-1 as Seismic Category I can withstand Seismic Category I loads without loss of safety function.

(4) RPV surveillance specimens are provided from the forging material of the beltline region and the weld and heat affected zone of a weld typical of those adjacent to the beltline region. Brackets welded to the vessel cladding at the location of the calculated peak fluence are provided to hold the removable specimen holders and a neutron dosimeter in place.

(5) a. The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) must meet the limited provisions of ASME Code Section III regarding certification that these components maintain structural integrity so as not to adversely affect RPV core support structure.

b. The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the requirements of ASME B&PV Code, Subsection NG-3000, except for the weld quality and fatigue factors for secondary structural non-load bearing welds.

(6) The initial fuel to be loaded into the core will withstand flow-induced vibration and maintain fuel cladding integrity during operation.

(7) The fuel bundles and control rods intended for initial core load have been fabricated in accordance with the approved fuel and control rod design.

(8) The reactor internals arrangement conforms to the fuel bundle, instrumentation, neutron sources, and control rod locations shown on Figure 2.1.1-2.

(9) The number and locations of pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations.

(10) The number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components.

(11) The number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking and of measuring the accelerations resulting from support and vessel movements.

(12) The number and locations of pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations.

(13) The number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components.

(14) The number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking and of measuring the accelerations resulting from support and vessel movements.

(15) The initial fuel to be loaded into the core will be able to withstand fuel lift and seismic and dynamic loads under normal operation and design basis conditions.

**Table 2.1.1-3**

**ITAAC For The Reactor Pressure Vessel and Internals**

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 1. The functional arrangement of the RPV and Internals is as described in the Design Description of Subsection 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1. | Inspections of the as-built RPV and Internals will be conducted. | The RPV and Internals and core arrangement conforms to the functional arrangement described in the Design Description of this Subsection 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1. |
| 2. The key dimensions (and acceptable variations) of the as-built RPV are as described in Table 2.1.1-2. | Inspection of the as-built RPV key dimensions (and acceptable variations thereof) will be conducted. | The RPV conforms to the key dimensions (and acceptable variations) described in Table 2.1.1-2. |
| 3a1 The RPV and its components identified in Table 2.1.1-1 (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) as ASME Code Section III are designed in accordance with ASME Code Section III requirements. | Inspection of ASME Code Design Reports (NCA-3550) and required documents will be conducted. | ASME Code Design Report(s) (NCA-3550) (certified, when required by ASME Code) exist and conclude that the design of the RPV and its components identified in Table 2.1.1-1 (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) as ASME Code Section III complies with the requirements of the ASME Code, Section III, including those stresses applicable to loads related to fatigue (including environmental effects), thermal expansion, seismic, and combined. |

**Table 2.1.1-3**

**ITAAC For The Reactor Pressure Vessel and Internals**

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 3a2. The RPV and its components identified in Table 2.1.1-1 (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) as ASME Code Section III shall be reconciled with the design requirements. | A reconciliation analysis of the components using as-designed and as-built information and ASME Code Design Reports (NCA-3550) will be performed. | ASME Code Design Report(s) (certified, when required by ASME Code) exist and conclude that design reconciliation has been completed in accordance with the ASME Code for as-built reconciliation of the RPV and its components identified in Table 2.1.1-1 (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) as ASME Code Section III. The report documents the results of the reconciliation analysis. |
| 3a3. The RPV and its components identified in Table 2.1.1-1 (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) as ASME Code Section III are fabricated, installed, and inspected in accordance with ASME Code Section III requirements. | Inspection of the RPV and its components identified in Table 2.1-1 as ASME Code Section III will be conducted. | ASME Code Data Report(s) (including N-1/N-1A Data reports, where applicable) (certified, when required by ASME Code) and inspection reports exist and conclude that the RPV and its components identified in Table 2.1.1-1 (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) as ASME Code Section III are fabricated, installed, and inspected in accordance with ASME Code Section III requirements. |

## Table 2.1.1-3

## ITAAC For The Reactor Pressure Vessel and Internals

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 4. Pressure boundary welds in the RPV meet ASME Code Section III non-destructive examination requirements. | Inspection of as-built pressure boundary welds in the RPV will be performed in accordance with the ASME Code Section III. | ASME Code Report(s) exist and conclude that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds in the RPV. |
| 5. The RPV retains its pressure boundary integrity at its design pressure. | A hydrostatic test will be conducted on the RPV as it is required to be hydrostatically tested by the ASME Code. | ASME Code Report(s) exist and conclude that the results of the hydrostatic test of the RPV comply with the requirements of the ASME Code Section III. |
| 6. The equipment identified in Table 2.1.1-1 as Seismic Category I can withstand Seismic Category I loads without loss of safety function. | i. Inspection will be performed to verify that the Seismic Category I equipment identified in Table 2.1.1-1 is located in a Seismic Category I structure.<br>ii. Type tests, analyses, or a combination of type tests and analyses of equipment identified in Table 2.1.1-1 as Seismic Category I will be performed using analytical assumptions, or will be performed under conditions which bound the Seismic Category I design requirements. | i. The equipment identified in Table 2.1.1-1 as Seismic Category I is located in a Seismic Category I structure.<br>ii. The equipment identified in Table 2.1.1-1 as Seismic Category I can withstand Seismic Category I loads without loss of safety function. |

## Table 2.1.1-3

## ITAAC For The Reactor Pressure Vessel and Internals

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| | iii. Inspection and analyses will be performed to verify that the as-built equipment identified in Table 2.1.1-1 as Seismic Category I, including anchorage, is bounded by the tested or analyzed conditions. | iii. The as-built equipment, identified in Table 2.1.1-1 as Seismic Category I, including anchorage, can withstand Seismic Category I loads without loss of safety function. |
| 7. RPV surveillance specimens are provided from the forging material of the beltline region and the weld and heat affected zone of a weld typical of those adjacent to the beltline region. Brackets welded to the vessel cladding at the location of the calculated peak fluence are provided to hold the removable specimen holders and a neutron dosimeter in place. | Inspections of the as-built RPV and Internals will be conducted for implementation of the RPV surveillance specimens, neutron dosimeter, and brackets. An analysis is performed to determine the location of the peak fluence. | The RPV surveillance specimens and neutron dosimeters are provided and brackets are installed at the location(s) of calculated peak fluence determined by an analysis of the as-built configuration. |
| 8a. The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) must meet the limited provisions of ASME Code Section III regarding certification that these components maintain structural integrity so as not to adversely affect RPV core support structure. | Inspections will be conducted of the as-built internal structures as documented in the ASME Code design reports. | The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the limited provisions of ASME Code Section III, NG-1122 (c), regarding certification that these components maintain structural integrity so as not to adversely affect RPV core support structure. |

## Table 2.1.1-3

## ITAAC For The Reactor Pressure Vessel and Internals

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 8b. The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the requirements of ASME B&PV Code, Subsection NG-3000, except for the weld quality and fatigue factors for secondary structural non-load bearing welds. | Inspections will be conducted of the as-built internal structures as documented in the ASME Code design reports. | The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the requirements of ASME B&PV Code, Subsection NG-3000, except for the weld quality and fatigue factors for secondary structural non-load bearing welds. |
| 9. The initial fuel to be loaded into the core will withstand flow-induced vibration and maintain fuel cladding integrity during operation. | Flow-Induced Vibration (FIV) testing will be performed on the fuel bundle design that will be loaded into the ESBWR initial core and on the reference fuel design in reactor use during the time of the tests. Bundle and rod responses at various elevations between the ESBWR design and the fuel design with the most similar design features will be compared. | The initial fuel to be loaded into the core will withstand flow-induced vibration and maintain fuel cladding integrity during operation. |
| 10. The fuel bundles and control rods intended for initial core load have been fabricated in accordance with the approved fuel and control rod design. | An inspection of the fuel bundles and control rods will be performed. | The fuel bundles and control rods intended for the initial core load have been inspected upon receipt to verify that they have been fabricated in accordance with the approved fuel and control rod design. |

## Table 2.1.1-3

## ITAAC For The Reactor Pressure Vessel and Internals

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 11. The reactor internals arrangement conforms to the fuel bundle, instrumentation, neutron sources, and control rod locations shown on Figure 2.1.1-2. | An inspection of the as-built system will be performed. | The as-built reactor system fuel bundle, control rod, instrumentation, and neutron source locations conform to the locations shown on Figure 2.1.1-2. |
| 12. The number and locations of pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations. | An analysis of the number and locations of pressure sensors installed on the steam dryer for startup testing will be performed. | The number and locations of pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations. |
| 13. The number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components. | An analysis of the number and locations of strain gages and accelerometers installed on the steam dryer for startup testing will be performed. | The number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components. |
| 14. The number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking and of measuring the accelerations resulting from support and vessel movements. | An analysis of the number and locations of accelerometers installed on the steam dryer for startup testing will be performed. | The number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking of and measuring the accelerations resulting from support and vessel movements. |

## Table 2.1.1-3

## ITAAC For The Reactor Pressure Vessel and Internals

| Design Commitment | Inspections, Tests, Analyses | Acceptance Criteria |
|---|---|---|
| 15. The initial fuel to be loaded into the core will be able to withstand fuel lift and seismic and dynamic loads under normal operation and design basis conditions. | An analysis of the fuel lift and seismic and dynamic loads will be performed on the fuel bundle design that will be loaded into the ESBWR initial core. | The initial fuel to be loaded into the core will have primary stresses and maximum fuel bundle lift out of the fuel support piece that do not exceed the allowable values provided in the approved Fuel Assembly Mechanical Design Report. |

In the GSTRM analyses it is assumed that during the fuel rod operating lifetime that the fuel rod (axial) node with the highest power operates on the limiting power-exposure envelope during its entire operating lifetime. The axial power distribution is changed three times during each operating cycle Beginning of Cycle (BOC), Middle of Cycle (MOC), and End of Cycle (EOC), to assure conservative prediction of the release of gaseous fission products from the fuel pellets to the rod free volume. The relative axial power distributions used for a standard fuel rod are shown in Figure 4.2-1.

#### 4.2.3.1.1 Worst Tolerance Analyses

The analyses performed to evaluate the cladding circumferential strain during an anticipated operational occurrence applies worst tolerance assumptions. In this case, the GSTRM inputs important to this analysis are all biased to the fabrication tolerance extreme in the direction that produces the most severe result. The biases are discussed in detail in Reference 4.2-5.

#### 4.2.3.1.2 Statistical Analyses

The remaining GSTRM analyses are performed using standard error propagation statistical methods. The statistical analysis procedure is presented in Reference 4.2-5.

#### 4.2.3.1.3 <u>Fuel Lift and Seismic and Dynamic Load Analyses</u>

<u>The fuel lift and seismic and dynamic load analyses will be completed prior to fuel release as described in Reference 4.2-4.</u>

### *4.2.3.2 Cladding Strain*

The cladding strain analysis is performed using the GSTRM code and the worst-tolerance methodology noted above. For each fuel rod type the cladding strain is calculated at different exposure points, whereby an overpower is assumed relative to the limiting power history. At the most limiting exposure point, the magnitude of the overpower event is further increased until the cladding strain approaches limits described in Reference 4.2-5. The result from this analysis is used to establish the mechanical overpower (MOP) discussed below.

### *4.2.3.3 Fuel Rod Internal Pressure*

The fuel rod internal pressure analysis is performed using the GSTRM code and the statistical methodology noted above. Values for the fuel rod internal pressure average value and standard deviation are determined at different fuel rod exposure points. At each of these exposure points, the fuel rod internal pressure required to cause the cladding to creep outward at a rate equal to the fuel pellet irradiation swelling rate is also determined using the same method. Based on the two calculated distributions a design ratio defined as the ratio of 'cladding creep out rate-to-fuel swelling rate' is determined such that, with at least 95% confidence, the fuel rod cladding does not creep out at a rate greater than the fuel pellet irradiation swelling rate.

### *4.2.3.4 Fuel Pellet Temperature*

The fuel pellet temperature analysis is performed statistically using the GSTRM code. For each fuel rod type the fuel pellet center temperature is statistically calculated at different exposure points, whereby an overpower is assumed relative to the limiting power history. At the most limiting exposure point, the magnitude of the overpower event is further increased until incipient fuel center-melting occurs. The result from this analysis establishes the thermal overpower (TOP) discussed below.

**Enclosure 5**

**MFN 06-297 Supplement 9**

**Supplement to Response to Portion of NRC Request for**

**Additional Information Letter No. 53**

**Related to ESBWR Design Certification Application**

**Reactor**

**RAI Number 4.8-8**

**LTR NEDO-33240 Markups**

**Public Version**

**NON-PROPRIETARY INFORMATION NOTICE**

This is a non-proprietary version of Enclosure 4, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

3.4.1.11 Seismic/Dynamic Loading

The GE14E fuel assembly has been designed to comply with the loading envelope and methods requirements stipulated in NEDE 21175–3–P–A, *BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings (Amendment No. 3)*, Ref. 2. The acceptance criteria, as given in Ref. 2, are that the primary stresses in the fuel are less than 70% of the material ultimate strength and the fuel bundle lift out of the fuel support piece does not exceed [[ ]]. This limit is established by the fitup between the lower tie plate and the fuel support piece. The fitup for ESBWR is identical to that for earlier fuel and reactor designs.

GE14 fuel assemblies for BWR/4-6 have been demonstrated to be acceptable for the following peak seismic and dynamic accelerations: [[

]]. Due the similarities between GE14E and GE14, and due to the shorter ESBWR fuel assembly length, the GE14E fuel assemblies are capable of withstanding accelerations greater than those for GE14 fuel assemblies.

From Ref. 3, ESBWR standard plant seismic and dynamic analysis shows peak SSE accelerations of [[ ]]. These accelerations are less than the demonstrated capability of the GE14 fuel. The shorter ESBWR fuel assembly length results in additional margin to the seismic and dynamic load criteria for GE14E fuel.

The structural capability of the GE14E fuel assembly for withstanding seismic/dynamic loading is primarily determined by the channel and spacer designs. The channel and spacer design have been tested to assure adequate capability. The remaining fuel assembly components have been demonstrated by analysis to meet the above acceptance criteria.

~~The horizontal dynamic response of the core is controlled primarily by the mass and stiffness of the fuel assemblies. The mass and stiffness properties of the GE14E fuel assembly design are improved over earlier GNF fuel designs with respect to horizontal seismic loading as a result of the shorter overall length, and corresponding reduction in mass. This will result in improved horizontal dynamic response versus previous GNF fuel designs.~~

The GE Nuclear Energy fuel lift procedure calculates the net vertical force acting on the fuel assembly and the direction of that force, upward or downward. The vertical loads on the RPV internals resulting from fuel lift are also calculated. The following loads act on the fuel assembly for the normal condition and the accident condition:

1) Vertical and horizontal seismic inertia loads, obtained from the seismic analysis of the primary structure analytical model, with a detailed representation of the RPV and internals. These loads for ESBWR are expected to be comparable to other BWR vertical and horizontal seismic loads.

2) Vertical and horizontal dynamic inertia loads (SRV and LOCA loads), obtained from dynamic analyses of the detailed RPV and internals portion of the primary structure seismic/dynamic model. These loads for ESBWR are expected to be comparable to other BWR dynamic inertia (SRV/LOCA) loads.

3) The fuel lift margin force acting downward (from thermal hydraulic analysis of the reactor coolant flow through the reactor core). The fuel lift margin force will be less than other BWR plants since fuel bundle weight for GE14E is reduced relative to previous GE fuel designs.

4) Control rod guide tube forces acting upward (from thermal hydraulic analysis of the reactor coolant flow through the reactor core). This force for ESBWR is significantly less then other BWR plants that will offset reduced fuel lift margin.

Based on above it is concluded that for ESBWR maximum fuel lift will be comparable to other plants. The seismic and dynamic loads on the RPV internals due to fuel lift for ESBWR will be comparable to other plants. Per GE procedure, application specific fuel lift and seismic load evaluations will be performed before fuel release for application.

# 5. REFERENCES

1. Zirconium Alloys Fatigue Design Curve, Y1002C200 Rev. 6, GE Fuel & Control Materials Properties Handbook

2. NEDE-21175-3-P-A, BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant (LOCA) Loadings (Amendment No. 3), October 1984

3. 26A6647 Rev. 4, Seismic Analysis of Reactor/Fuel Building Complex, June 2008

Enclosure 6

MFN 06-297 Supplement 9

Supplement to Response to Portion of NRC Request for

Additional Information Letter No. 53

Related to ESBWR Design Certification Application

Reactor

RAI Number 4.8-8

Affidavit

# Global Nuclear Fuel – Americas, L.L.C.

## Affidavit

**I, Andrew A. Lingenfelter,** state as follows:

(1) I am Vice President, Fuel Engineering, 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 Enclosures 1 and 4 of GEH letter MFN 06-297 Supplement 9, Richard E. Kingston to U. S. Nuclear Regulatory Commission, *Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8,* dated April 19, 2010. The proprietary information in Enclosure 1, *MFN 06-297 Supplement 9 Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8,* and Enclosure 4, *MFN 06-297 Supplement 9 Supplement to Response to Portion of NRC Request for Additional Information Letter No. 53 Related to ESBWR Design Certification Application – Reactor – RAI Number 4.8-8 – LTR NEDC-33240P Markups*, is delineated by dotted underlined 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, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 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;

## Global Nuclear Fuel – Americas, L.L.C.

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,

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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 19th day of April 2010.

Andrew A. Lingenfelter

Andrew A. Lingenfelter
Global Nuclear Fuels – Americas, LLC