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

MFN 08-347

Docket No. 52-010

May 9, 2008

U.S. Nuclear Regulatory Commission

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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 110 – Related to ESBWR Design Certification Application – RAI Numbers 4.2-2 Supplement 3, 4.2-4 Supplement 2 and 4.8-6 Supplement 1**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) sent by the Reference 1 NRC letter. GEH response to RAI Numbers 4.2-2 S03, 4.2-4 S02 and 4.8-6 S01 is addressed in Enclosures 1, 2, 3 and 4.

Enclosure 1 contains GEH proprietary information as defined by 10 CFR 2.390. GEH 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 4 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GEH. GEH hereby requests that the information of Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 10 CFR 9.17.

Verified DCD changes associated with this RAI response are identified in the Enclosure 3 DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markups may not be fully developed and approved for inclusion in DCD Revision 5.

D068
NRO

The subject LTR (NEDC-33242P) will be revised by July 31, 2008 to show compliance for GE14E fuel with the proposed cladding strain, oxide and hydrogen limits.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

Reference:

1. MFN 07-510, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 110 Related To ESBWR Design Certification Application*, dated September 19, 2007

Enclosures:

1. MFN 08-347 – Response to Portion of NRC Request for Additional Information Letter No. 110 – Related to ESBWR Design Certification Application – RAI Numbers 4.2-2 S03, 4.2-4 S02 and 4.8-6 S01 – GEH Proprietary Information
2. MFN 08-347 – Response to Portion of NRC Request for Additional Information Letter No. 110 – Related to ESBWR Design Certification Application – RAI Numbers 4.2-2 S03, 4.2-4 S02 and 4.8-6 S01 – Non-Proprietary Version
3. MFN 08-347 – Response to Portion of NRC Request for Additional Information Letter No. 110 – Related to ESBWR Design Certification Application – DCD Markups from the Response to RAI Numbers 4.2-2 S03, 4.2-4 S02 and 4.8-6 S01
4. MFN 08-347 – Affidavit

cc:	AE Cubbage	USNRC (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)

eDRF 0000-0057-1360/R4, 0000-0082-4005/R1 and 0000-0085-3745

Enclosure 2

MFN 08-347

Response to Portion of NRC Request for

Additional Information Letter No. 110

Related to ESBWR Design Certification Application

RAI Numbers 4.2-2 S03, 4.2-4 S02, and 4.8-6 S01

Non-Proprietary Version

NRC RAI 4.2-2 S03 and 4.2-4 S02

Fuel rod cladding strain Tier 2 requirement.*

As discussed during the July 2007 GEH Control Blade and Fuel Assembly Design Audit, please revise the fuel rod cladding strain Tier 2 requirement, Appendix 4B, and provide supporting documentation.*

GEH Response to RAI 4.2-2 S03 and 4.2-4 S02 (Combined)

The intent of the statement that the [[

]]

For each GEH fuel design, GEH specifies operating limits (LHGR versus pellet exposure limits) to assure compliance with all thermal-mechanical SAFDLs. GEH maintains [[

]] Thus the operating limits reflect the current corrosion performance of GEH fuel cladding during BWR operation. However, [[
]]

During the subject July 2007 GEH Control Blade and Fuel Assembly Design Audit the NRC expressed concern about the [[ ]] The NRC also expressed a related concern about the [[ ]] The concerns are related to potential oxide spalling that could (1) result directly in fuel failure due to local through wall corrosion or (2) lead to fuel failure due to formation of hydride localizations that result in nonuniform cladding ductility. On the basis of the expressed concerns and subsequent discussions with the NRC, most recently during a meeting on February 12-13, 2008, [[

]]

[[

]]

As discussed in Attachment A, the proposed corrosion limit is based upon [[

]] Additionally, as noted in Attachment A, for plants in which the water chemistry has exceeded current EPRI water chemistry guidelines for Zn, the formation of tenacious crud resulting in spallation has occurred. Based upon lift-off measurements in plants that have experienced high lift-off but not failures, [[

]]

In addition to the concerns about the lack of [[ ]], the NRC has also expressed concern that sufficient data does not currently exist to support the current [[

]] This is different than the 1% total (elastic plus permanent) strain limit at all exposures included in the Section 4.2 of the US NRC Standard Review Plan. GEH has ongoing programs [[

]]

[[

]] The basis for the proposed limit, including its non-applicability to non-AOO power increases such as those that occur during normal operation, is discussed in more detail in Attachment B.

Summary

[[

]]

DCD Impact

The fuel rod cladding strain Tier 2* requirement in Appendix 4B of the ESBWR DCD Tier 2 will be revised to reflect this new limit. Also DCD Tier 2 Subsections 4.2.1.1.3, 4.2.1.1.4, 4.2.1.1.5, 4.2.3.1.1, 4.2.3.2, 4.2.3.8 and 4.2.7 will be updated based on this RAI response. The DCD markup pages are provided in enclosure 3.

The subject LTR (NEDC-33242P) will be revised by July 31, 2008 to show compliance for GE14E fuel with the proposed cladding strain, oxide and hydrogen limits. Page markups showing proposed changes to Subsection 3.3.1 and Table 3-1 are provided below.

3.3 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).

3.3.1 High Strain Rate (Anticipated Operational Occurrences, Item 3 of **Table 3-1**~~Table 3-1~~)

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%.~~ **ensure mechanical integrity. The cladding strain limit is defined for 2 exposure regions. The strain limit and the corresponding exposure regions are defined as follows:**

Region 1 – When the Peak-Pellet-Exposure (PPE) is less than or equal to [[

]]

Region 2 – When the PPE is greater than [[

]]

This strain limit is based on [[ ]] cladding oxide [[ ]] hydrogen content. Cladding oxide and hydrogen limit are intended to preclude damaging oxide spalling that could lead to cladding failure, and localized reduction in cladding ductility that could reduce the cladding failure strain significantly.

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

Table 3-1 Fuel Rod Thermal-Mechanical Design Criteria

Criterion	Governing Equation
1. The cladding creepout rate ($\dot{\varepsilon}_{cladding\_creepout}$), due to fuel rod internal pressure, shall not exceed the fuel pellet irradiation swelling rate ($\dot{\varepsilon}_{cladding\_creepout}$). Satisfied if design ratio (of internal pressure to critical pressure) is less than 1.00 (Sections 4.2 and 5.1).	$\dot{\varepsilon}_{cladding\_creepout} \leq \dot{\varepsilon}_{fuel\_swelling}$
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~~ ($\cancel{\varepsilon_{\theta}^{P}}$) ~~during an anticipated operational occurrence shall not exceed 1.00%.~~	$\cancel{\varepsilon_{\theta}^{P} \leq 1.00\%}$
3. Region 1 – When the Peak-Pellet-Exposure (PPE) is less than or equal to [[ ]]	**Region 1: [[ ]]**
~~3.~~ **Region 2 – When the PPE is greater than [[ ]]**	**Region 2: [[ ]]**
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).	$\sigma_e < \sigma_f$, $\varepsilon_e < \varepsilon_f$
7. The as-fabricated fuel pellet evolved hydrogen (C_H is content of hydrogen) at greater than 1800 °C shall not exceed prescribed limits.	[[ ]]

Attachment A

(Bases for Proposed Cladding Corrosion and Hydrogen Limits)

The derivation of the proposed corrosion and hydrogen limits are based on GEH's operating experience. [[

]] Both eddy current liftoff and profilometry measure the combined thickness due to corrosion and crud build up on the cladding surface; eddy current liftoff measures the combined oxide and crud thickness under the measuring probe and profilometry measures the clad diameter between two probes located diametrically across the cladding. In addition, the thicknesses of oxide and crud and hydrogen concentration for selected fuel rods have been measured in the hot cell. For the purpose of assessing fuel corrosion performance, a number of options have been considered. Specifically, detailed considerations are given to the possibility of applying a "no-spall" criterion and the use of poolside EC liftoff data for the purpose of fuel design and performance monitoring.

Potential use of a "no-spall" criterion

[[

]] The no-spall criterion loosely translates to an oxide thickness limit, since corrosion-formed oxide tends to spall once a thickness of approximately [[ ]] is reached. [[

]]

For BWR fuel, a tenacious crud is commonly deposited on top of corrosion-formed zirconium oxide. [[

]] With the introduction of Zn injection combined with hydrogen water chemistry in a majority of plants to minimize occupational dose and to mitigate IGSCC of plant materials, [[

]] Based on visual inspection results (Figure A-1) from a number of plants and operating cycles, [[

]] As a consequence, GEH has collaborated with EPRI and industry partners to develop water chemistry guidelines (Reference A-1) on feed water Zn levels to minimize crud spallation.

[[

]]

Figure A-1: Impact of Plant Water Chemistry on Crud Deposition and Spalling

[[

]] As with crud spalling, the observed spalling occurred during operation or handling. Some fuel rods that exhibit limited oxide spalling from the [[ ]] extended exposure program (to [[ ]]) have been examined in the hot cell. Figure A-2 shows the [[

]]

[[

]]

Figure A-2: Hydrogen and Hydride Morphology in the Vicinity of Crud Spalled Locations for the [[ ]]

[[

]] GEH understands NRC's concern on non-uniformity in cladding mechanical properties potentially resulting from oxide or crud spalling. [[

]]

Design limit for cladding corrosion

[[

]]

[[

]]

Figure A-3: Oxide Model Used for Fuel Design and Licensing Calculations

[[

]]

[[

]]

Figure A-4: Comparison of Hot Cell Oxide with the Pool Side Liftoff Measurements

Action level based on poolside inspections

[[

]] The net result was a reduction in liftoff measurements in subsequent cycles and the affected fuel successfully operated to the planned discharge exposure. The second type of elevated liftoff was related to non-typical cladding corrosion. In [[ ]], three third-cycle bundles were each found to contain one rod that had failed by corrosion. Poolside inspections showed that [[ ]]. Inspections of bundles from the same reload but discharged after one and two cycles of operation also showed liftoff significantly higher than expected for the discharge exposure, and rods from other reloads did not show elevated corrosion. The inspection results are consistent with a non-typical corrosion condition during the first cycle operation of the affected reload. The cycle-to-cycle liftoff data from sound rods from the affected reload at [[ ]] thus provided bounding examples for successful operation of cladding with elevated corrosion. There have been other examples from elevated corrosion events in which [[

]]

The prior experiences with liftoff higher than the proposed corrosion limit provide confidence that fuel can be [[

]]

[[

]]

[[

]]

Figure A-5: Action Limit Based on Successful Operation Database

Under the current practice, [[

]]

Cladding Hydrogen Limit

As there is considerable variability in the hydrogen database, a number of factors have been reviewed to determine [[

]]

[[

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Figure A-6: Typical Fuel Rod Power, Exposure & Fluence Profiles

[[

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[[

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[[ ]]

[[

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[[

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[[

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Figure A-7: GEH Cladding Hydrogen Database (limited to 12-130 inch axial elevation of fuel rods)

[[

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Figure A-8: GEH Cladding Hydrogen Database (ASTM Zircaloy-2 Spec, limited to 12-130 inch axial elevation)

[[

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Figure A-9: GEH Cladding Hydrogen Database (Controlled Chemistry, limited to 12-130 inch axial elevation)

[[

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[[
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[[
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[[
]]

References

[A-1] BWRVIP-130: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2004

[A-2] T. Miyashita et. al, "Corrosion And Hydrogen Pick-Up Behaviors Of Cladding And Structural Components In BWR High Burnup 9x9 Lead Use Assemblies", Proceedings of the 2007 International LWR Fuel Performance Meeting, San Francisco, California, September 30 – October 3, 2007, Paper 1015

[A-3] A.-M. Alverez-Holsten, "Studies Of Hydrogen Assisted Failures Initiating At The Cladding Outer Surface Of High Burn-Up Fuel Using A Modified Ring Tensile Technique", Proceedings of the 2007 International LWR Fuel Performance Meeting, San Francisco, California, September 30 – October 3, 2007, Paper 1080

Attachment B

(Basis for Proposed Cladding Strain Limit)

Factors considered in the determination of the [[

]]

[[

]]

Figure B-1: Impact of Dislocation Channeling on Tensile Testing of Irradiated Zircaloy (Reference B-1)

[[

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Figure B-2: Typical Plane Strain Test Result for Highly Irradiated (~65 GWd/MTU bundle average) Zircaloy-2 Cladding Material

During irradiation the ductility of fuel rod cladding is reduced by the fast neutron fluence and by the pickup of corrosion hydrogen. Tests conducted by GEH indicate that the effects of neutron fluence saturate at ~1 x 10^{21} nvt (>1 MeV), as shown in Figure B-3. [[

]]

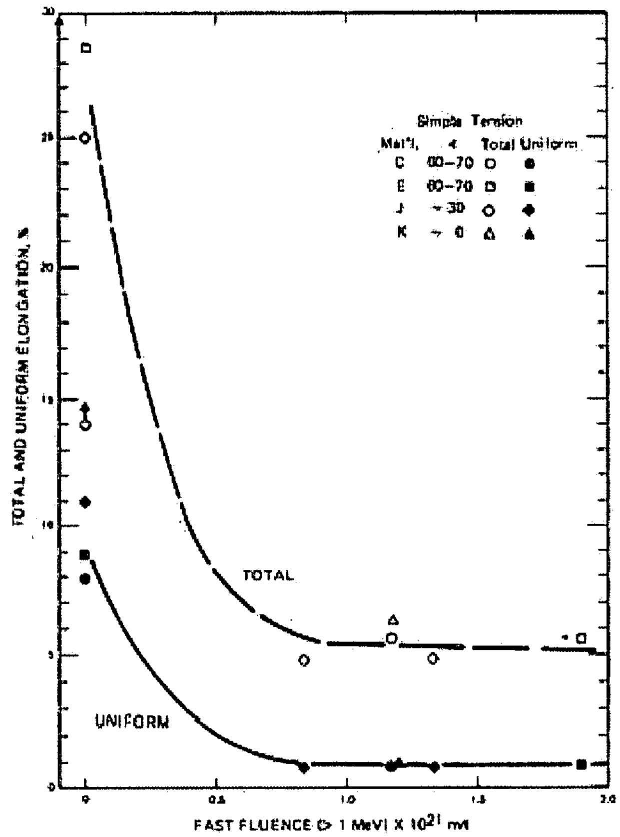

Figure B-3: Impact of Fluence on Uniform and Total Elongation (Reference B-2)

[[

]]

Figure B-4: Burst Test Results for Irradiated Cladding [[
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[[

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[[

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[[

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References

[B-1] Fregonese, M. et. al, "Failure Mechanisms of Irradiated Zr Alloys Related to PCI: Activated Slip Systems, Localized Strain, and Iodine-Induced Stress Corrosion Cracking"STP 1354, pp 377-398, January 2000.

[B-2] Rieger, G. F. and Lee, D., *Zirconium in Nuclear Applications,* ASTM STP 551, Am. Soc. Testing Mater. (1974) 335-369.

NRC RAI 4.8-6

Section 3.3.1 of NEDC-33240P states, "These limits are typically applied to unirradiated material conditions because irradiation increases the material strength properties."

While it is true that irradiation hardening increases the material yield strength, it may not increase the overall strength of a component such as a spacer. Describe the steps taken (e.g. irradiated material testing) to ensure that the beginning-of-life evaluations are most limiting.

NRC RAI 4.8-6 Supplement No. 1

As discussed during the July 2007 GEH Control Blade and Fuel Assembly Design Audit, the revised fuel rod cladding strain limit (RAI 4.2-2 Supplement) will need to be captured in a revision to the original 4.8-6 response.

GEH Response

During operation, the strength of Zircaloy structural components increases due to hardening resulting from fast neutron fluence. The only currently identified mechanisms that could reduce the overall strength of fuel assembly components with in-core operation are [[

]] GEH structural evaluations explicitly address the reduction in strength [[
]], so a reduction in strength of irradiated components relative to the strength at beginning-of-life would be due [[ ]].

For fuel rods, GEH provides data that supports a [[
]] as discussed in the combined response to RAIs 4.2-2 S03 and 4.2-4 S02.

For other fuel assembly components, [[ ]] Embrittlement due to hydrogen has been extensively investigated, for example in an EPRI sponsored program (Yagnik et al, ASTM STP 1467, p605-631). The EPRI program tested irradiated and unirradiated Zircaloy with hydrogen concentrations up to about 2000 ppm using a variety of test geometries, including uniaxial tension, burst and slotted-arc specimens. The results showed that, for hydrogen concentrations up to about 2000 ppm, the primary effect of hydrogen was a reduction in the ductility particularly in irradiated Zircaloy; the results showed there was little hydrogen effect on the strength of the Zircaloy. These results confirm that evaluations [[ ]]

For spacers, GEH has [[
]] For reference, [[

]]

[[

]]

[[

]] These results show that the fracture resistance of the spacer material and structure remains high enough to avoid failure [[
]]

DCD Impact

No DCD changes will be made in response to this RAI

[[

]]

Figure 4.8-6-1: Hydride Spacer Test Fixture

[[

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Figure 4.8-6-2: Typical Force (in lb) Time History

[[

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Figure 4.8-6-3: Force (in lb) Time History with Crack Development

[[

]]

Figure 4.8-6-4: Hydrided Spacer Test Results

Enclosure 3

MFN 08-347

Response to Portion of NRC Request for

Additional Information Letter No. 110

Related to ESBWR Design Certification Application

DCD Markups from the Response to RAI Numbers 4.2-2 S03, 4.2-4 S02, and 4.8-6 S01

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markups may not be fully developed and approved for inclusion in DCD Revision 5.

The detailed design bases for each of the fuel assembly damage, fuel rod failure and fuel assembly cooling criteria, as defined in Section II.A of NRC Standard Review Plan 4.2 (except control rod reactivity; see Subsection 4.2.1.2) are provided in Section 4B.2 of Appendix 4B.

4.2.1.1.1 Fuel Temperature

The fuel rod centerline temperature is limited to ensure with high probability that fuel melting does not occur during normal operation, including AOOs.

4.2.1.1.2 Fuel Rod Internal Pressure

During fabrication, the fuel rod is filled with helium to a specified 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 sufficient irradiation, a potential adverse thermal feedback condition may arise due to excessive fuel rod internal pressure.

When the internal pressure exceeds the reactor coolant pressure, the cladding begins to deform outward (cladding creep out). If the rate of this cladding outward deformation exceeds the rate at which the fuel pellet expands due to irradiation (fission product) swelling (fuel swelling rate), the pellet-cladding gap begins to open (or increase if the gap is already open). An increase in the pellet-cladding gap reduces the pellet-cladding thermal conductance thereby increasing fuel temperatures. The increased fuel temperatures results in further fuel pellet fission gas release, greater fuel rod internal pressure, and correspondingly a faster rate of cladding outward deformation and gap opening.

This potential 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.

4.2.1.1.3 Cladding Strain

The fuel rod cladding strain is limited to ensure that fuel rod failure due to pellet-clad mechanical interaction does not occur. To achieve this objective the calculated cladding circumferential ~~plastic~~ strain is limited as described in Reference 4.2-5 during anticipated operational occurrences.

4.2.1.1.4 Cladding Corrosion and Corrosion Product Buildup

Zircaloy cladding tubes undergo oxidation at slow rates during normal reactor operation and reactor water corrosion products (crud) are deposited on the cladding outside surface (see Reference 4.2-10). The cladding oxidation causes thinning of the cladding tube wall and introduces a resistance to the fuel rod-to-coolant heat transfer. Crud buildup can also introduce a resistance to heat transfer. The expected extent of the oxidation and the buildup of the corrosion products is specifically considered in the fuel rod design analyses. Thus the impacts of the temperature increase, the correspondingly altered material properties and the thinning of the cladding wall resulting from cladding corrosion on fuel rod behavior relative to impacted design criteria (such as fuel temperature and cladding strain) are explicitly addressed. The ~~oxide~~

~~thickness itself is not separately limiting and no direct~~ design limit on cladding oxide thickness is ~~therefore~~ specified in Reference 4.2-5.

4.2.1.1.5 Fuel Rod Hydrogen Absorption

There are two considerations relative to fuel rod hydrogen absorption. The first consideration involves the potential for hydrogenous impurity evolution, historically from the fuel pellets, resulting in primary hydriding and fuel rod failure. This consideration is addressed by the application of a specification limit on the as-fabricated fuel pellets. The absence of primary-hydriding induced fuel rod failures demonstrates the effectiveness of this limit since its first application in 1972. The second consideration is the partial absorption by the fuel rod cladding of hydrogen liberated by the cladding waterside corrosion reaction. Mechanical properties testing demonstrates that the cladding mechanical properties are negligibly affected by hydrogen contents far in excess of that experienced during normal operation. Based on available mechanical properties test data of the irradiated cladding, a design basis hydrogen limit is specified in Reference 4.2-5. ~~On this basis, there is no specific design criterion applied to the cladding hydrogen content.~~

4.2.1.1.6 Cladding Creep Collapse

The fuel rod is evaluated to ensure that fuel rod failure due to cladding collapse into a fuel column axial gap does not occur. This criterion is discussed in detail in Reference 4.2-3.

4.2.1.1.7 Fuel Rod Stresses

Based upon the limits specified in ANSI/ANS 57.5-1981, the fuel rod is evaluated to ensure that the fuel does not fail due to cladding stresses or strains exceeding the cladding ultimate stress or strain capability. The figure of merit employed is termed the Design Ratio, where:

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

The effective stress or strain is determined by applying the distortion energy theory. The limit is the material ultimate stress or strain. To be within the limit, the Design Ratio must be less than or equal to 1.0.

4.2.1.1.8 Dynamic Loads / Cladding Fatigue

The fuel rod is evaluated to ensure that cladding strains due to cyclic loadings do not exceed the cladding material fatigue capability. The design limit for fatigue cycling is determined from Zircaloy fatigue experiments and is conservatively specified to ensure with high confidence that failure by cladding fatigue does not occur. Based on the LWR cyclic design basis presented in Reference 4.2-5, the cladding fatigue life usage is calculated and maintained below the cladding material fatigue limit.

As noted in Subsection 4.2.1.1, for each fuel design, steady-state operating limits are established to ensure that actual fuel operation, including AOOs, complies with the fuel rod thermal–mechanical design and safety analysis bases above. These operating limits define the maximum allowable fuel operating power level as a function of fuel exposure. Lattice local power and exposure peaking factors may be applied to transform the maximum allowable fuel power level into maximum linear heat generation rate (MLHGR) limits for individual fuel bundle designs.

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 (BOC, MOC and 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 ~~plastic~~ 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.

4.2.3.2 Cladding ~~Plastic~~ Strain

The cladding ~~plastic~~ strain analysis is performed using the GSTRM code and the worst-tolerance methodology noted above. For each fuel rod type the cladding ~~plastic~~ 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 ~~plastic~~ 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.

4.2.3.5 Cladding Fatigue Analysis

The cladding fatigue analysis is performed statistically using the GSTRM code. For calculating the cladding fatigue, variations in power and coolant pressure, as well as coolant temperature, are superimposed on the limiting power history.

The fuel duty cycles shown in Reference 4.2-5 represent conservative assumptions regarding power changes anticipated during normal reactor operation including anticipated operational occurrences, planned surveillance testing, normal control blade maneuvers, shutdowns, and special operating modes such as daily load following. Based on these assumptions, the cladding strain cycles are analyzed as shown in Reference 4.2-5.

4.2.3.6 Cladding Creep Collapse

The cladding creep collapse analysis consists of a detailed finite element mechanics analysis of the cladding. This evaluation is described in detail in References 4.2-3 and 4.2-5.

4.2.3.7 Fuel Rod Stress Analysis

The fuel rod stress analysis is performed using the Monte Carlo statistical methodology and addresses local fuel rod stress concerns, such as the stresses at spacer contact points, that are not addressed by the GSTRM code. Results from GSTRM analyses are used to generate inputs for the stress analysis. The cladding stress analysis is described in detail in Reference 4.2-5.

4.2.3.8 Thermal and Mechanical Overpowers

As discussed above, analyses are performed to determine the values of the maximum overpower magnitudes that do not result in violation of the cladding circumferential ~~plastic~~ strain criterion Mechanical Overpower (MOP) and the incipient fuel center-melting criterion (TOP-Thermal Overpower). Conformance to these criteria is demonstrated as a part of the normal core design and transient analysis process by comparison of the calculated core transient mechanical and thermal overpowers, as defined in Reference 4.2-5, to the mechanical and thermal overpower limits determined by the GSTRM analyses.

4.2.3.9 Fretting Wear

Testing is performed to assure that the mechanical features of the design, particularly those related to spacers and tie plates, do not result in significant vibration and consequent fretting wear, particularly at spacer –fuel rod contact points. The vibration response of the new design is compared to a design that has demonstrated satisfactory performance through discharge exposure.

4.2.3.10 Water Rods

Calculations are performed to determine component stresses at the bounding load conditions and compared to applicable criteria, such as yield and ultimate stresses. The load conditions take into account shipping and handling loads, seismic induced bending moment, and the pressure differential across the water rod. The design is also evaluated using finite element analysis to determine the critical buckling load and insure adequacy relative to axial loads resulting from differential growth of water rods and other fuel assembly components.

The control rod is designed to permit coupling and uncoupling of the control rod drive from below the vessel for FMCRD servicing without necessitating the removal of the reactor vessel head. The control rod is also designed to allow uncoupling and coupling from above the vessel using control rod handling tools.

The control rod is positively coupled to the FMCRD and is designed to remain coupled during all scrams and loading conditions, including inoperative buffer scram loads. The control rod withstands the loads induced by the FMCRD without exceeding the structural design criteria as stated in Subsections 4.2.4.1 and 4.2.4.2 above.

The control rod is dimensionally compatible with the fuel assemblies (unirradiated and irradiated). The control rod is guided, rotationally restrained and laterally supported by the adjacent fuel assemblies. The control rod is designed and constructed to establish and maintain the alignment of the control rod drive line (that is, the CRDH, CRGT, and fuel assemblies) so that control rod insertion and withdrawal is predictable. The top of the active absorber of a fully withdrawn control rod is below the Bottom of the Active Fuel (BAF). Absorber gap requirements are placed on the control rod in the operating condition to be compatible with the core nuclear design requirements.

4.2.5 Testing, Inspection, and Surveillance Plans

GEH has an active program for the surveillance of both production and developmental fuel. The NRC has reviewed the GEH program and approved it in Reference 4.2-6.

4.2.6 COL Information

This section contains no requirement for additional information to be provided in support of the combined license. Combined License Applicants referencing the ESBWR certified design will address changes to the reference design of the fuel assembly or control rods from that presented in the DCD.

4.2.7 References

4.2-1 GE Nuclear Energy, "GE Fuel Bundle Designs," NEDE-31152P, Revision 8, April 2001.

4.2-2 GE Nuclear Energy, "Fuel Rod Thermal Analysis Methodology (GSTRM)", NEDC-31959P, April 1991.

4.2-3 GE Nuclear Energy, "Cladding Creep Collapse", NEDC-33139P-A, July 2005.

4.2-4 [*GE Nuclear Energy, "GE14 ~~for ESBWR~~ Fuel Assembly Mechanical Design Report", NEDC-33240P, Class III (Proprietary), January 2006.*]*

4.2-5 [*GE Nuclear Energy, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report", NEDC-33242P, Class III (Proprietary), Revision ~~1~~2, ~~February 2007~~May 2008.*]*

4.2-6 USNRC Letter, L. S. Rubenstein (NRC) to R. L. Gridley (GE), "Acceptance of GE Proposed Fuel Surveillance Program", June 27, 1984.

4.2-7 GE Nuclear Energy, "GE Marathon Control Rod Assembly," NEDE-31758P-A, October 1991.

normal steady-state operation and ~~whole core~~ anticipated operational occurrences ~~is~~ does not ~~expected to~~ occur.

(3) 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%.~~ ensure that loss of fuel rod mechanical integrity (cladding circumferential strain greater than 1%) doesn't occur due to pellet-cladding mechanical interaction. The specific strain criteria, ie, percentage of plastic or total, will be defined in the fuel thermal-mechanical design licensing topical report for each licensed design.

To ensure that uniform properties of the cladding are maintained, a design oxide thickness and hydrogen limit will be specified for each fuel design in the fuel thermal-mechanical design licensing topical for each licensed design.

In non-barrier cladding, fast power ramps can also cause a chemical/mechanical pellet cladding interaction commonly known as Pellet Clad Interaction / Stress Corrosion Cracking (PCI/SCC). To prevent PCI/SCC failures in non-barrier cladding, reactor operational restrictions must be imposed. To reduce the potential for 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 measure for reducing PCI/SCC failures.

Table 4B~~.1~~-1

Fuel Rod Thermal-Mechanical Design Criteria

Criterion	Governing Equation
1. [*The cladding creepout rate ($\dot{\varepsilon}_{cladding\_creepout}$), due to fuel rod internal pressure, shall not exceed the fuel pellet irradiation swelling rate ($\dot{\varepsilon}_{fuel\_swelling}$).*]*	$\dot{\varepsilon}_{cladding\_creepout} \leq \dot{\varepsilon}_{fuel\_swelling}$
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 $\cancel{\varepsilon_\theta^P}$) during an anticipated operational occurrence shall not exceed 1%. The specific strain criteria, ie, percentage of plastic or total, will be defined in the fuel thermal-mechanical design licensing topical report for each licensed design*]*	$\varepsilon_\theta \leq 1\%$
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).*]*	$\sigma_e < \sigma_f$, $\varepsilon_e < \varepsilon_f$
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*

Enclosure 4

MFN 08-347

Affidavit

GE Hitachi Nuclear Energy

AFFIDAVIT

I, David H. Hinds, state as follows:

(1) I am General Manager, New Units Engineering, GE Hitachi Nuclear Energy ("GEH"), 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 GEH's letter, MFN 08-347, Mr. James C. Kinsey to U.S. Nuclear Energy Commission, entitled "*Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application – RAI Numbers 4.2-2 Supplement 3, 4.2-4 Supplement 2 and 4.8-6 Supplement 1,*" dated May 9, 2008. The proprietary information in enclosure 1, which is entitled "*Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application – RAI Numbers 4.2-2 Supplement 3, 4.2-4 Supplement 2 and 4.8-6 Supplement 1 – GEH Proprietary Information,*" is delineated by a [[dotted underline inside double square brackets.{3}]] 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 or licensee, GEH 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 GEH's competitors without license from GEH 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 GEH customer-funded development plans and programs, resulting in potential products to GEH;

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 GEH, 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 GEH, 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, or subject to the terms under which it was licensed to GEH. Access to such documents within GEH 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 for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH 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 details of GEH's evaluation methodology.

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 GEH asset.

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

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.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH 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 GEH 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 GEH 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 on this 9th day of May 2008.

David H. Hinds
GE Hitachi Nuclear Energy