Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

NEDO-33242-A
Revision 2
Class I
eDRFSection 0000-0049-8127
September 2010

Licensing Topical Report

GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report

PROPRIETARY INFORMATION NOTICE

This is a public version of NEDC-33242P-A, Revision 2, from which the proprietary information has been removed. Portions of the document that have been removed are indicated by white space within double square brackets, as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The information contained in this document is furnished as reference material for GE14E Fuel Assembly Mechanical Design. The only undertakings of Global Nuclear Fuel–Americas, LLC respecting information in this document are contained in the contracts between Global Nuclear Fuel–Americas, LLC and the participating utilities in effect at the time this report is issued, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone other than that for which it is intended is not authorized; and with respect to any unauthorized use, Global Nuclear Fuel–Americas, LLC makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

CONTENTS

FIGURES

TABLES

ABSTRACT

The GE4 fuel assembly for use in ESBWR power stations, denoted GE14E, is similar to the GE14 fuel assembly for use in BWR/3-6 and ABWR power stations and the design analyses performed for GE14E are similar to those performed for GE14 and documented in Reference 9. The analyses for UO_2 and $(U,Gd)O_2$ fuel rods for the GE14E fuel assembly are summarized in this report. The analyses results demonstrate that all design criteria applicable to fuel rod thermal-mechanical design are satisfied for operation of the GE14E fuel design to a peak pellet exposure of [[ ]] and a maximum operating time of [[ ]]. The specific design criteria that are addressed by this report include:

1) Fuel rod internal pressure
2) Fuel melting
3) Pellet-cladding mechanical interaction (PCMI)
4) Cladding fatigue
5) Cladding collapse
6) Fuel rod stresses

REVISIONS

NEDO-33242, Revision 2 incorporates the NRC letter describing the acceptance of this revision of this Licensing Topical Report as well as Enclosure 1 of the letter, which contains the Final Safety Evaluation for this Licensing Topical Report. These items have been added at the end of this report as Attachment 1.

NEDO-33242, Revision 2 replaces NEDO-33242, Revision 1 that was submitted for NRC review on February 2007. Revision 2 replaces Revision 1 in its entirety and should be the sole basis for NRC review and approval. The following notes summarize the key changes. Editorial and clarification changes are also included.

1. Section 3.2 has been updated to indicate that fuel melting will not occur during steady state operation and anticipated operational occurrences ("core wide" terminology has been removed).
2. Section 3.3.1 has been updated to reflect the revised cladding strain, oxide and hydrogen limits.
3. Table 3-1 has been updated to reflect the revised cladding strain limit.
4. Section 4.1 has been updated to reflect the revised cladding strain limit.
5. Section 4.5 has been updated to reflect the revised cladding strain limit.
6. Section 5.2.2 has been updated to reflect the revised cladding strain limit.
7. Table 5-2 has been updated to reflect the change to Section 3.2 noted above.
8. Section 5.3 has been updated to reflect the revised cladding oxide limit.
9. Section 5.4 has been updated to reflect the revised cladding hydrogen limit.
10. Table 6.1 and Table 6.2 have been updated to reflect operational experiences up to May 2009.
11. Table B-1 has been updated to reflect the correct tube shell alloy composition and oxygen concentration.

NEDC-33242P, Revision 1 replaces NEDC-33242P, Revision 0 which was submitted for NRC review on January 31, 2006. Revision 1 replaces Revision 0 in its entirety and should be the sole basis for NRC review and approval. The following notes summarize the key changes. Editorial and clarification changes are also included.

1. Proprietary Markings were removed from the 1% plastic strain limit in Table 3-1 and in Sections 3.3, 4.1 and 5.2.2.

2. Sections 3.2, 4.5, 5.2.2 were updated per RAI 4.8-16. Also, included are changes in these sections to make this LTR consistent with the ESBWR DCD Appendix 4B, after updates were made to the appendix per RAI 4.2-6.
3. Reference 1 was changed to a reference that better clarifies the approval of GSTRM.

ACRONYMS AND ABBREVIATIONS

Term	Definition
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
BOC	Beginning of Cycle
BOL	Beginning of Life (bundle)
BWR	Boiling Water Reactor
EOC	End of Cycle
EOL	End of Life (bundle)
GNF	Global Nuclear Fuel
GSTRM	GESTR – Mechanical Fuel Rod Model
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
LFWH	Loss of Feed Water Heating
LWR	Light Water Reactor
MOC	Middle of Cycle
MOP	Mechanical Overpower
PCI	Pellet/Cladding Interaction (failure)
TOP	Thermal Overpower
PCMI	Pellet/Cladding Mechanical Interaction
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RWE	Rod Withdrawal Error
SCC	Stress Corrosion Cracking
TIG	Tungsten Inert Gas
USNRC	United States Nuclear Regulatory Commission

1. INTRODUCTION

A primary consideration in the design and operation of nuclear power plants is the limitation of radioactive species release from the power plant site. Radioactive species are generated within the fuel rod uranium (and uranium-gadolinium) dioxide pellets as a normal product of the nuclear fission process. Therefore, the fuel rod cladding surrounding the uranium dioxide fuel pellets represents an important barrier to the release of radioactive fission products to the reactor coolant. Although the nuclear power plant system is designed to accommodate a level of activity release that may result from defective fuel rods, while conforming to authorized site activity release limits, the GNF fuel rod design objective is to preclude systematic defects arising under the conditions of authorized operation including normal steady-state operation and anticipated operational occurrences.

This fuel rod design objective is achieved by the imposition of mechanistic limits on the predicted performance of the fuel under the conditions of authorized operation. The GNF GESTR-Mechanical (GSTRM) fuel rod thermal-mechanical performance model (Reference 1) is applied to provide conservative fuel performance predictions for comparison against the specified performance limits. These design and licensing basis analyses are described in detail in this report. Results of the analyses for the GE14 design for operation in BWR/3-6 and ABWR power stations are summarized in Reference 9. The GE14 fuel assembly for operation in ESBWR power stations, denoted GE14E in this report, is similar to the GE14 design. The term GE14 refers, in this report, to the GE14 design for used in BWR/3-6 and ABWR power stations unless otherwise specified. The major difference in terms of fuel rod thermal-mechanical analyses is in total rod length and in active fuel length and plenum volume for each rod type. This report summarizes the GE14 thermal-mechanical licensing analyses and limits as they conservatively apply to GE14E fuel design.

The fuel rod design analysis methodology is comprised of three elements:

1. Design criteria - Mechanistic design criteria are applied to those fuel rod parameters that realistically represent fuel rod integrity limitations,

2. The analytical GSTRM model (Reference 1) - This fuel rod model calculates the thermal-mechanical changes within the fuel rod which occur during reactor operation and provides a realistic assessment of the response of each design parameter. GSTRM has been developed and qualified based on an extensive experimental fuel rod data base which enables clear quantification of the model prediction uncertainty, and

3. Statistical and worst tolerance analysis procedures – The statistical analysis methodology, in conjunction with the GSTRM model, enables a realistic assessment of statistical uncertainties of the characteristic fuel rod behavior parameters, e.g. fuel rod pressure and pellet temperature as a function of the statistical model parameter input distribution, e.g. pellet diameter and pellet density. The statistical analysis methodology enables direct quantitative assessment of the conservatism of the analysis results. The worst tolerance analysis methodology, in conjunction with the GSTRM model, enables a bounding

assessment of the cladding circumferential strain during an anticipated operational occurrence. 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 design criteria and analysis procedures are described in Sections 3 and 4. The results of application to the GE14E fuel design are summarized in Section 5. These results demonstrate that all criteria are met by the GE14E fuel design to a peak pellet exposure of [[ ]], corresponding to a fuel rod average exposure of approximately [[ ]] for the UO_2 rods.

2. FUEL ROD DESCRIPTION

The basic GE14E fuel rod is comprised of a column of right circular cylinder fuel pellets enclosed by a cladding tube and sealed gas-tight by plugs inserted in each end of the cladding tube. The plugs are TIG or resistance welded after insertion. The fuel pellets consist of sintered uranium-dioxide (UO_2) or UO_2-gadolinia solid solution ($(U, Gd)O_2$) with a ground cylindrical surface, flat ends, and chamfered edges. Each full-length UO_2 fuel rod may include natural enrichment UO_2 pellets at each end of the fuel pellet column. The fuel rod cladding tube is comprised of Zircaloy-2 with a metallurgically bonded inner zirconium layer.

Each fuel rod includes a plenum at the top of the fuel rod to accommodate the release of gaseous fission products from the fuel pellets. This gas plenum includes a compression spring to minimize fuel column movement during fuel assembly shipping and handling operations while permitting fuel column axial expansion during operation. The GE14E fuel assembly contains 14 fuel rods, which are reduced in length relative to the remaining fuel rods. Fuel rods are internally pressurized with helium to [[ ]] bar to reduce the compressive hoop (and radial) stress induced in the cladding tube by the coolant pressure and to improve the fuel-to-cladding heat transfer.

Figure 2 –1 shows a sketch of the GE14E fuel rods while Figure 2-2 shows a sketch of the GE14E fuel pellet. The characteristic data of the pellet, fuel rod and the cladding are listed in Table 2-1, Table 2-2 and Table 2-3. Materials properties of the pellets and the cladding can be found in Reference 5. Additional details concerning cladding fabrication processing are included in Appendix B.

Table 2-1 Fuel Pellet Characteristics[1]

Item	Value
Material	UO_2, $(U, Gd)O_2$
Melting Temperature[2] UO_2 [3] $(U, Gd)O_2$	[[

]]

[1] Valid at 20 °C
[2] Values shown are valid at beginning-of-life. The melting temperature decreases with exposure at the rate of [[ ]]
[3] The value shown is a conservative estimate of the UO_2 melting temperature.
[4] In-reactor fuel densification is exposure dependent. The value shown represents the fabrication maximum based on a 1700 °C 24-hour resinter test.

Table 2-2 Fuel Rod Characteristics[5]

Item	Value
Fuel Rod Length (shoulder to shoulder)	[[
Full-Length Rod (Basic + Gadolinia)	
Part-Length Rod	

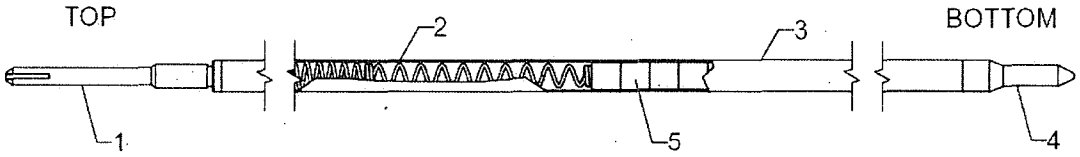
]]

[5] Valid at 20 °C

Table 2-3 Cladding Tube Characteristics[6]

Item	Value
Material	Zircaloy-2, [[ ]] with zirconium liner
Density	[[ ]]

[6] Valid at 20 °C



ITEM	TITLE	MATERIAL
1	PLUG, UPPER	ZIRCALOY
2	PLENUM SPRING	STAINLESS STEEL
3	TUBE	ZIRCALOY-2 WITH ZIRCONIUM LINER (BARRIER)
4	PLUG, LOWER	ZIRCALOY
5	PELLET	UO_2 ENRICHED

Figure 2-1 Fuel Rod

[[

]]

Figure 2-2 Fuel Pellet Sketch

3. DESIGN CRITERIA

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 2) and NUREG-0800 Rev. 2 (Reference 3). Table 3-1 presents a summary of the design criteria. The bases for the design criteria listed in Table 3-1 are presented below.

3.1 Cladding Lift-Off / Fuel Rod Internal Pressure (Item 1 of Table 3-1)

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) as described in Sections 4.2 and 5.1 and ensuring that the calculated design ratio is less than 1.00 at any point in time for all fuel rod types.

3.2 Fuel Temperature (Melting, Item 2 of Table 3-1)

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 anticipated operational occurrences is not expected to occur. This fuel temperature limit is specified to ensure that sudden shifting of molten fuel in the interior of fuel rods, and subsequent potential cladding damage, can be positively precluded.

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)

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%. During review of this LTR, the NRC has expressed concern that sufficient data does not currently exist to support application of the current cladding strain limit of 1% permanent (plastic plus creep) strain at all exposures. GEH has ongoing programs [[

]]

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.

3.3.2 Low Strain Rate (Steady-State Operation, no limit in Table 3-1)

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. For example, Reference 4 reports circumferential tensile creep strains as high as 18% without fracture. For comparison, the imposition of fuel pellet irradiation swelling stresses beginning at the start of irradiation and continuing throughout lifetime to 100 MWd/kgU will result in a low-stress tensile circumferential creep strain of less than [[ ]]. Therefore, no specific limit is applied to low-strain rate cladding deformation.

3.4 Dynamic Loads / Cladding Fatigue (Item 4 of Table 3-1)

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).

3.5 Elastic Buckling / Cladding Creep Collapse (Item 5 of Table 3-1)

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.

It is noted in this PWR experience that, although complete cladding collapse was observed in some cases, cladding fracture did not occur in any case, therefore fuel rod failure by this mechanism is not expected. However, the GNF design basis includes ensuring that fuel rod failure will not occur due to cladding collapse into a fuel column axial gap. The origin of the creep collapse analysis procedure applied by GNF to the GE14 fuel design is the USAEC staff technical report on densification of light water reactor fuels issued in 1972 (Reference 6). In response, GNF produced a number of documents that included the creep collapse analysis procedure detailed in Reference 7. The analysis is performed to confirm that creep collapse of free standing cladding (cladding unsupported by fuel pellets) will not occur. The basic procedure detailed in Reference 7 has been applied by GNF to the GE14 fuel design to demonstrate that creep collapse of the cladding will not occur (Reference 9). The procedure includes deliberately conservative assumptions; including the assumption that fuel densification can result in large axial gaps in the fuel column. GNF has recognized since its introduction that the procedure is very conservative. This is particularly the case for modern GNF fuel designs with current fabrication processes and controls on fuel pellet density and densification.

3.6 Fuel Rod Stresses (Item 6 of Table 3-1)

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 2). The figure of merit employed is termed the Design Ratio where

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

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

3.7 Fuel Rod Hydrogen (Item 7 of Table 3-1)

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 (less than [[ ]] evolved hydrogen above 1800 °C) in conjunction with fabrication practices that eliminate hydrogenous contaminants from all sources during the manufacturing process.

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 ($\dot{\varepsilon}_{fuel\ swelling}$). 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. Range 1 – [[ ]]	Range 1: [[ ]]
[[ ]]	Range 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.	[[ ]]

4. DESIGN ANALYSES DESCRIPTION

Most of the fuel rod thermal-mechanical design analyses are performed using the GSTRM fuel rod thermal-mechanical performance model. The GSTRM fuel rod thermal-mechanical model provides best estimate predictions of fuel rod thermal and mechanical performance. The GSTRM analyses are performed for the following conditions:

1. For the fuel rod design analyses under consideration, the input parameters selected for such analyses are based on the most unfavorable manufacturing tolerances ('worst case' analyses) or by using statistical distributions of the input values. Calculations are then performed to provide either a 'worst case' or statistically bounding tolerance limit for the resulting parameters.

2. Operating conditions, in the form of maximum power verses exposure envelopes for each fuel type, are postulated which cover the conditions anticipated during normal steady-state operation and anticipated operational occurrences.

[[

]] An example power-exposure envelope is shown in Figure 4-1. This maximum power versus exposure envelope is then used for all fuel rod thermal-mechanical design analyses to evaluate the fuel rod design features and demonstrate conformance to the design criteria. This maximum steady-state power versus exposure envelope is applied as a design constraint to the reference core loading nuclear design analyses. This maximum steady-state power versus exposure envelope is also applied as an operating constraint to ensure that actual operation is maintained within the fuel rod thermal and mechanical design bases.

With this maximum steady-state power versus exposure envelope, the GSTRM analyses are conservatively performed [[

]] The fuel rod axial power shape is changed three times during each cycle (BOC, MOC, EOC) and simulates the power distribution effects of Burnup Shape Optimization. The relative axial power distributions used for a full length fuel rod are presented in Figure 4-2.

As discussed above, two types of GSTRM analyses are performed, (1) worst tolerance, or (2) probabilistic.

[[

]]

Figure 4-1 Design Basis Power versus Exposure Envelope (Typical)

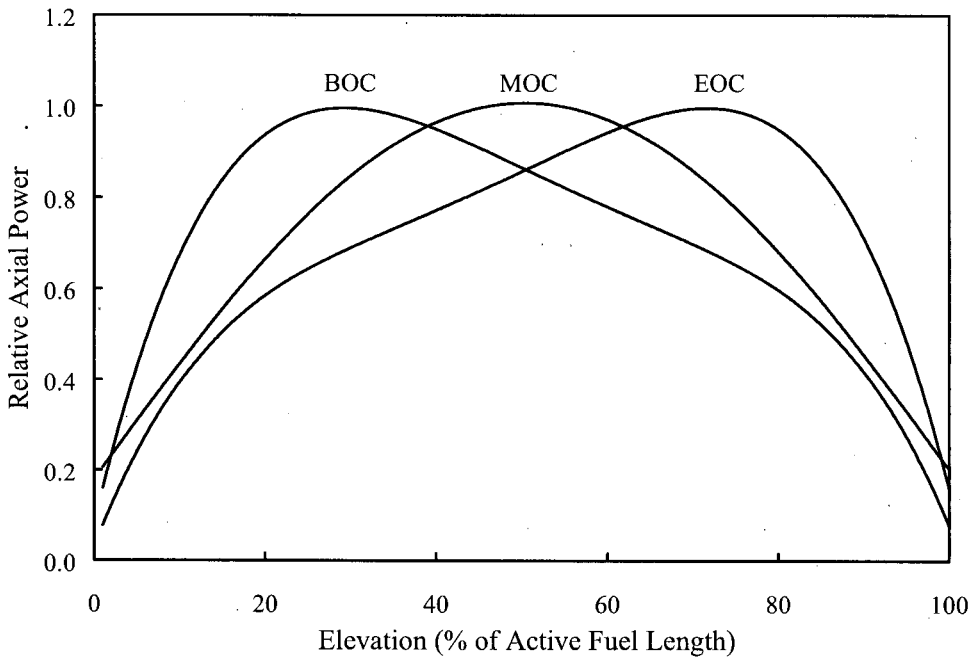


Figure 4-2 Axial Power Distributions (Full Length Fuel Rod)

4.1 Worst Tolerance Analyses

The GSTRM analysis 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. Table 4-1 presents the analysis fabrication parameter biases and bases for those biases. Other input parameters conservatively biased for this analysis include (a) cladding corrosion (2 sigma), and (b) corrosion product (crud) buildup on the cladding outer surface (2 sigma).

The evaluation reflects continuous operation along the maximum power history according to Figure 4-1, followed by an instantaneous overpower due to an anticipated operational occurrence. The analyses to determine the circumferential strain is performed at several exposure points during the fuel rod lifetime. At the exposure point resulting in the highest circumferential strain the overpower event is increased to determine the maximum permissible overpower that will not exceed the cladding 1.00% circumferential strain criterion.

The result from this analysis is used to establish the Mechanical Overpower (MOP) discussed in Section 4.5.

Table 4-1 Worst Tolerance Analysis Manufacturing Parameter Biases

Parameter	Bias Direction	Basis

[[

]]

4.2 Statistical Analyses

The remaining GSTRM analyses are performed using standard error propagation statistical methods. The statistical analysis procedure is presented below:

1. The mean value (x_{nom}) and standard deviation (σ_x) of each GSTRM input parameter is determined as discussed in Appendix A. For the manufacturing parameters, these statistical distribution parameter values are determined from the fuel rod drawing tolerances and manufacturing specifications. Certain manufacturing parameters such as pellet density, pellet densification, pellet surface roughness, and cladding surface roughness are controlled by statistical specifications as discussed further in Appendix A. A GSTRM analysis using the limiting power history is performed using the average values of all input parameters. This analysis represents the reference base case analysis and provides the mean values of the output parameters of interest ($y_{reference}$).

2. Then partial derivatives of the resulting parameters as a function of the input parameters are calculated, by first varying independently each input parameter to the ($x_{nom} + 2\sigma_x$) or ($x_{nom} - 2\sigma_x$) value. The direction of the perturbation ($\pm 2\sigma_x$) is taken to increase the severity of the result relative to the performance parameter of interest. These perturbation analyses provide the perturbed values of the output parameters of interest ($y_{perturbation}$). The specific parameters perturbed are specified in Table 4-2.

Table 4-2 GSTRM Parameters Varied Statistically

[[

]]

The nominal values and standard deviations associated with these parameters are derived as discussed in Appendix A. Values are given in Reference 8.

3. The partial derivative of the GSTRM output parameter of interest, with respect to each of the input parameters, is approximately determined as

$$\frac{\partial y}{\partial x} = \frac{y_{perturbation} - y_{reference}}{2\sigma_{x_i}}$$

where

y	=	GSTRM output parameter of interest (e.g., fuel rod internal pressure)
x_i	=	GSTRM input parameter (e.g., cladding thickness)
σ_{x_i}	=	standard deviation of input parameter, x_i

4. The standard deviation of the GSTRM output parameter of interest is then calculated by standard error propagation methods as

$$\sigma_y^2 = \sum_{i=1}^{n}\left[\frac{\partial y}{\partial x_i}\right]^2 \sigma_{x_i}^2 + 2\sum_{i=1}^{n-1}\sum_{j=i+1}^{n}\frac{\partial P}{\partial x_i}\frac{\partial P}{\partial x_j}\sigma_{x_i}\sigma_{x_j}\rho_{x_i x_j}$$

where,

σ_y	Standard deviation of output parameter being analyzed (internal pressure, etc.)
i, j	Index for input variables perturbed in the error propagation analysis
n	Total number of input variables x_i,x_j perturbed in the error propagation analysis
x_i, x_j	Input variable perturbed in the GESTR-Mechanical analysis
$\frac{\partial P}{\partial x_i}$, $\frac{\partial P}{\partial x_j}$	Partial derivative of output parameter P with respect to perturbed input variable x_i, x_j
σ_{x_i}, σ_{x_j}	Standard deviation of input parameters x_i, x_j
ρ_{x_i,x_j}	Correlation coefficients for variables x_i, x_j

5. [[

]]

The fuel rod internal pressure analysis, the fuel temperature analysis, and the cladding fatigue analysis are all performed statistically in this manner.

4.2.1 Fuel Rod Internal Pressure

For the fuel rod cladding lift-off analysis, the fuel rod internal pressure reflects continuous operation along the maximum steady-state power-exposure envelope throughout lifetime. The standard error propagation analysis results in a mean and standard deviation for the fuel rod internal pressure at various points throughout the design lifetime. 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 standard error propagation method. A design ratio is formed based on these two distributions such that, when the design ratio is less than or equal to 1.00, it is assured with at least [[ ]] confidence that the fuel rod cladding will not creep out at a rate greater than the fuel pellet irradiation swelling rate.

4.2.2 Fuel Pellet Temperature

The fuel temperature analysis also reflects continuous operation along the maximum steady-state power-exposure envelope, but is then followed by an instantaneous overpower due to an anticipated operational occurrence. This analysis is performed at several exposure points during the fuel rod lifetime to determine the most limiting time in life. At the most limiting time in life, the magnitude of the overpower event is increased to determine the maximum permissible overpower that will not exceed the incipient fuel center-melting criterion. The result from this analysis establishes the Thermal Overpower (TOP) discussed in Section 4.5.

4.2.3 Cladding Fatigue Analysis

The cladding fatigue analysis also reflects operation along the maximum steady-state power-exposure envelope. However, superimposed on the power-exposure history are power and coolant pressure/temperature changes. The power change spectrum used is listed in Table 4-3.

The fuel duty cycles shown in Table 4-3 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. The cladding strain cycles are analyzed using the "rainflow" cycle counting method. The fractional fatigue life expended for each strain cycle is determined and summed over the total number of cycles to determine the total

fatigue life expended over the fuel design lifetime. The material fatigue capability is taken as a lower bound to the available experimental measurements of Zircaloy fatigue capability. The statistical calculation determines the mean and standard deviation of total fatigue life expended. The upper [[ ]] value of fatigue life expended is required to be < 1.00.

Table 4-3 Fatigue Analysis Power Cycles

Power Cycle, (% Rated)	Frequency, (#/yr.)	Duration
[[		
		]]

4.3 Cladding Creep Collapse

This analysis consists of a detailed finite element mechanics analysis of the cladding. The cladding is assumed initially oval shaped. The amount of the initial ovality of the tube may either be assumed to be the allowance for maximum ovality as specified by the design drawings or may be assumed to be the two sigma deviation from roundness based on actual manufacturing data. The specific loading conditions consist of the system coolant pressure applied to the outside of the cladding and the minimum internal as-fabricated pre-pressurization level, as corrected for operating conditions, applied to the inside surface of the cladding. In the GE14 analysis, no support is assumed to be provided from contact of the cladding with the fuel pellets. The creep properties employed are the same as are used in GSTRM. After the condition of maximum ovality is reached at end of life, an overpressure transient is assumed to occur. The magnitude of this overpressure transient is taken to bound the conditions expected during pressurization event anticipated operational occurrences. Application and removal of this overpressure is performed to confirm that collapse due to elastic or plastic instability does not occur.

4.4 Fuel Rod Stress Analysis

The fuel rod stress analysis is performed using the Monte Carlo statistical method. The effects of pressure differential, cladding ovality, radial thermal gradients, spacer contact, thermal bow and circumferential thermal gradients are determined for a specific Monte Carlo trial using classical linear elastic mechanics formulations. For each trial calculation, the stresses are combined into an effective stress using the Von Mises method and compared with the appropriate design limit to produce a design ratio. Design ratios are calculated at the cladding inside and outside diameter, at the spacer and away from the spacer. A large number of trials are performed and the [[ ]] percentile design ratio is determined. Separate analyses are performed to address normal operation and overpower transient conditions, beginning and end-of-life conditions considering both UO_2 and gadolinia fuel rods. In the area of the endplug welds, a finite element mechanics analysis is performed, reflecting the combined effects of the internal-external pressure difference, thermal gradients and axial stresses caused by the differential expansion of the fuel and the cladding.

4.5 Thermal and Mechanical Overpowers

As discussed in Sections 4.1 and 4.2, analyses are performed to determine the values of the maximum overpower magnitudes that would not exceed the cladding circumferential strain criterion (MOP-Mechanical Overpower) and the incipient fuel center-melting criterion (TOP-Thermal Overpower). Conformance to these MOP and TOP 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 schematically in Figure 4-3, to the mechanical and thermal overpower limits determined by the GSTRM analyses.

The concept of TOP and MOP limits as summarized above was developed to provide parameters that are easily evaluated in terms of LHGR or surface heat flux and that can be used

as computational limits during the design of a core. TOP and MOP limits are intended to prevent exceedance of actual licensing limits (no fuel melting and cladding strain less than 1%) and to provide an initial screen during the nuclear design of a core or an upcoming cycle. Violation of TOP or MOP limits does not indicate violation of actual licensing limits, only that additional analyses are required to confirm compliance with the actual SAFDLs. The analyses are performed with currently approved methodologies.

Although not explicitly addressed in the licensing analyses, similar overpower analyses are performed to confirm that control blade maneuvers will not result in exceedance of temperature or cladding strain limits. [[

]]

Figure 4-3 Thermal and Mechanical Overpowers (Schematic)

5. DESIGN ANALYSIS RESULTS

5.1 Cladding Lift-Off / Fuel Rod Internal Pressure

The fuel rod internal pressure and (cladding lift-off) design ratio are determined statistically using GSTRM. The analysis is performed for each fuel type to assure with [[ ]] confidence that the fuel rod cladding will not creep outward at a rate greater than the fuel pellet irradiation swelling rate. As discussed in Section 3.1, the fuel rod internal pressure is proportional to the fission gas released from the fuel, which in turn for specified operating limits is approximately proportional to the ratio of fuel volume, and the rod free volume, which consists of the plenum volume plus the pellet-cladding gap and the fuel column volumes. For a specified fuel rod geometry, the free volume at any exposure is dependent upon the initial rod free volume. Then the internal pressure is approximately proportional to the ratio of fuel volume to initial rod free volume. On this basis, the full length GE14 UO_2 rod is determined to be limiting in terms of internal pressure and design ratio for the GE14E fuel design (Reference 10). Results for the full length UO_2 rod are summarized in Table 5-1 (from Reference 9).

Table 5-1 Fuel Rod Internal Pressure and Design Ratio

	Value	Exposure MWd/kgU
Maximum Design Ratio	[[	
Nominal EOL Rod Internal Pressure (bar)		]]

Although the results in Table 5-1 were obtained with inputs applicable to current GE14 fuel operating in BWR/3-6 and ABWR plants, the assumed nominal values and uncertainties in operation dependent parameters (such as oxidation rate and axial power shape), and fabrication dependent parameters (such as pellet density and densification) are anticipated to bound GE14E fuel operating in ESBWR plants. Also, as noted in Reference 9, in addition to the conservatism inherent in the assumption of operation on a [[ ]] operating envelope, the design ratio in Table 5-1 is based upon conservative assumptions in the calculations of critical pressure (pressure required to result in the cladding creepout rate being equal to the pellet swelling rate), specifically in the assumed pellet swelling rate uncertainty. Considering these conservatisms, and the large reduction in the ratio of fuel volume to plenum volume for GE14E relative to GE14, the results in Table 5-1 confirm that the GE14E design meets the rod internal pressure criterion for the maximum power versus exposure envelopes specified in Reference 8.

5.2 Thermal and Mechanical Overpowers

5.2.1 Fuel Temperature

The fuel pellet centerline temperature for the maximum duty fuel rod is statistically determined using GSTRM. Evaluations are performed for each fuel rod type over a range of exposures and overpowers to simulate various AOOs. The evaluations reflect operation on the bounding power-exposure operating envelope prior to the AOO. Based upon the results of these evaluations, the thermal overpower limits in Table 5-2 (from Reference 9) are applied to the GE14 fuel design to prevent centerline melting for the maximum power envelopes specified in Reference 8.

Since the maximum power-exposure envelopes for GE14E are identical to those for GE14, if it is assumed, as in Section 5.1, that analysis inputs applicable to current GE14 fuel operating in BWR/3-6 and ABWR plants bound operation of GE14E fuel in ESBWR plants, the thermal overpower limits in Table 5-2 are directly applicable to the GE14E fuel design. The application is slightly conservative for the limiting rod due to the slightly improved thermal performance resulting from the reduced fuel volume to rod free volume ratio for the GE14E design relative to the GE14 design discussed in Section 5.1. Thus the thermal overpower limits in Table 5-2 are applied to the GE14E fuel design to prevent centerline melting for the maximum power envelopes specified in Reference 8.

5.2.2 Cladding Strain

The fuel rod cladding circumferential plastic strain is a 'worst case' analysis (see Section 4.1). The parameters, which according to their consequences on the result, that were set at the extremes in the manufacturing tolerance bands or operation dependent characterizations include: [[

]]. Evaluations are performed for each fuel rod type over a range of exposures and overpowers to simulate various AOOs. The evaluations reflect continuous operation on the bounding power-exposure operating envelope prior to the AOO. Based upon the results of these evaluations, the mechanical overpower limits in Table 5-2 (from Reference 9) are applied to the GE14 fuel design to prevent cladding strain equal to or greater than 1.00% for the maximum power envelopes specified in Reference 8.

As in the case of fuel temperature, since the maximum power-exposure envelopes for GE14E are identical to those for GE14, if it is assumed, as in Section 5.1, that analysis inputs applicable to current GE14 fuel operating in BWR/4-6 and ABWR plants bound operation of GE14E fuel in ESBWR plants, the mechanical overpower limits in Table 5-2 are directly applicable to the GE14E fuel design. Again, as in the case of fuel temperature, the application is slightly conservative for the limiting rod due to the slightly improved thermal performance resulting from the reduced fuel volume to rod free volume ratio for the GE14E design relative to the GE14 design discussed in Section 5.1. Thus the mechanical overpower limits in Table 5-2 are applied to the GE14E fuel design to prevent [[

]] for the maximum power envelopes specified in Reference 8.

Table 5-2 Maximum Allowable Overpowers for the Anticipated Operational Occurrences

Events	Maximum Allowable Overpower, %	
	Thermal Overpower (TOP)	Mechanical Overpower (MOP)
Anticipated Operational Occurrences	[[	]]

The thermal overpower (TOP) and mechanical overpower (MOP) limits in Table 5-2 apply to pressurization transients. For bundles impacted by rod withdrawal events, the TOP limit in Table 5-2 is applied, but a reduced MOP limit of 19% is applied. The TOP and MOP limits in Table 5-2 are determined by the limiting [[ ]] rod at its limiting exposure and are applied to all rods.

5.3 Cladding Corrosion

In the responses to RAI-4.2-2 and 4.2-4 an [[

]]

The effects of cladding oxidation and corrosion product buildup (crud) on the fuel rod surface are included in the fuel rod thermal-mechanical design evaluations (see Section 4). The initial value and growth rate of the crud and the oxide thickness are input parameters for the statistical analyses. The mean value and standard deviation for corrosion thickness as a function of time is provided in Appendix A. The results for cladding corrosion are derived from data collected from plants with a range of saturation temperatures and from fuel operating over a wide range of powers. Thus input parameters derived from the data and the statistical methodology explicitly address small changes in saturation temperature due to small changes

in coolant pressure, such as might occur due to a power uprate or operation of the GE14E fuel design in ESBWR plants.

GEH maintains [[

]]

5.4 Cladding Hydrogen Content

This evaluation relative to hydriding of the fuel rod cladding is based on the substantial operating and manufacturing experience to date with fuel designs fabricated to the same specification limit on the amount of hydrogen permitted in a manufactured fuel rod. This operating experience is summarized in Section 6. The experience with fuel manufactured since 1972 demonstrates that hydriding is not an active failure mechanism for current GNF fuel designs, including the GE14E fuel design. During review of this LTR, the NRC also expressed concern about the [[ ]] On the basis of the expressed concern and subsequent discussions with the NRC, [[

]]

5.5 Cladding Creep Collapse

The results of the analysis described in Section 4.3 confirm that the GE14 design will not experience cladding creep collapse (Reference 9). Since the cladding for the GE14E and GE14 fuel designs are identical, since the power-exposure envelopes for GE14E are identical to those for GE14, and since no fission gas release is assumed in the analysis, the results of the GE14 creep collapse analysis are directly applicable to GE14E. Thus the GE14E design will not experience cladding creep collapse for the maximum power envelopes specified in Reference 8.

5.6 Fuel Rod Stresses/Strains

Table 5-3 (from Reference 9) presents the limiting values of the cladding stress design ratio described in Section 4.4 for rated power and for 30% overpower for the GE14 fuel design. The maximum design ratios for both 100% and 130% power occur at BOL. Cladding stresses are calculated under the spacer and at midspan between the spacers. For the GE14 design, the maxmum design ratios are calculated between spacers. The values in Table 5-3 are the upper

95% values of the design ratios between spacers from the Monte Carlo analysis. These results confirm conformance to the cladding stress design criterion.

Since the fuel rod and spacer geometries for the GE14E and GE14 fuel designs are identical, with the exception of rod length and spacer pitch, and since the power-exposure envelopes for GE14E are identical to those for GE14, the calculated stresses under the spacer will be identical for GE14E and GE14. The reduced minimum spacer pitch for the GE14E design relative to the GE14 design will increase the effective cladding stiffness of the span and possibly increase the axial stress components and calculated design ratios between spacers due to circumferential temperature variation and flow induced vibration. However, these components are small relative to components due to coolant overpressure and cladding ovality, and the net change in calculated design ratios will be small. Additionally, the loads assumed for the cladding stress analysis are deliberately conservative. For these reasons, and considering the large margin to the design limit for the upper 95% case presented in Table 5-3, it is concluded that the GE14E design will meet the cladding stress criterion for the maximum power envelopes specified in Reference 8.

Table 5-3 Results of Cladding Stress Analysis

	Design Ratio	
Period	Rated Power (100%)	Overpower (130%)
BOL	[[	]]

The maximum effective plastic strain in the lower end plug weld zone, determined by the finite element mechanics analysis described in Section 4.4, is [[ ]] for the GE14 fuel design. This value occurs at BOL. The limit for this strain is [[ ]]. Thus this result confirms conformance to the end plug weld plastic strain design criterion.

The weld zone applied loading is determined by the axial interaction (locking) of fuel pellet and cladding in the lower portion of the rod. Since the fuel rod geometries for the GE14E and GE14 fuel designs are identical in the region of the lower endplug and since the power-exposure envelopes for GE14E are identical to those for GE14, the applied loading will be identical. Then the calculated effective plastic strain in the lower end plug weld zone will also be identical for GE14E and GE14. Additionally, the loads are deliberately conservative. For these reasons, and considering the large margin to the design limit to the strain limit, it is concluded that the GE14E design will meet the lower end plug weld plastic strain cladding criterion for the maximum power envelopes specified in Reference 8.

5.7 Dynamic Loads / Cladding Fatigue

Table 5-4 (from Reference 9) shows the results of the cladding fatigue analysis as performed according to Section 4.2.3 for the GE14 fuel design. The [[ ]] tolerance limit of the calculated distribution is listed for the full length UO_2 rod and the limiting gadolinia rod. These results confirm conformance to the cladding fatigue design criterion.

The results in Table 5-4 are at the axial location of maximum fuel duty. Since the fuel rod geometry for the GE14E and GE14 fuel designs are identical, with the exception of rod length, and since the power-exposure envelopes for GE14E are identical to those for GE14, the results in Table 5-4 are directly applicable to GE14E, provided the assumed loading spectrum is adequate for ESBWR operation. The assumed loading spectrum is summarized in Table 4-3. This loading spectrum was developed considering all operating modes and AOOs anticipated for operation in BWR/3-6 and ABWR plants. Although such a spectrum has not been developed for ESBWR operation, the simplified configuration of the ESBWR plant relative to the BWR/3-6 and ABWR plants and the use of fine motion control rod drives is expected to make the assumed loading spectrum conservatively applicable to ESBWR operation. For this reason, and considering the large margin to the fatigue limit in Table 5-4, even for the upper 95% case, it is concluded that the GE14E design will meet the cladding fatigue criterion for the maximum power envelopes specified in Reference 8.

Table 5-4 Cladding Fatigue Usage

		Fatigue Usage	
Rodtype	Nominal	Upper [[ ]]-Tolerance Limit	Limit for upper [[ ]]-Tolerance Limit
UO_2	[[		
Gad			]]

6. FUEL OPERATING EXPERIENCE UPDATE

A summary of GNF fuel experience with recent designs is presented below. The fuel experience summary addresses GE11/13 (9x9) and GE12/14 (10x10) designs, as summarized below in Table 6-1 and Table 6-2.

Table 6-1 GE11/13 (9X9) Experience Summary as of 05/2009

Item	GE11 9x9	GE13 9x9
Fuel Operated		
Reloads	72	32
Bundles	13110	6776
Fuel Rods	973840	502016
Lead Exposure, MWd/kgU		
Batch average	53	50
Peak bundle average	64.8	52

Table 6-2 GE12/14 (10X10) Experience Summary as of 05/2009

Item	GE12 10x10	GE14 10x10
Fuel Operated		
Reloads	31	130
Bundles	4,252	25,523
Fuel Rods	396,152	2,351,980
Lead Exposure, MWd/kgU		
Batch average	50	49
Peak bundle average	68	67

REFERENCES

1. MFN-036-85, Letter, C. O. Thomas (NRC) to J. S. Charnley (GE), Acceptance for Referencing of Licensing Topical Report NEDE-24011-P, Amendment 7 to Revision 6, "GE Standard Application for Reactor Fuel," March 1, 1985.

2. 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.

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

4. E. F. Ibrahim, Creep Ductility of Cold-Worked Zr-2.5 w/o Nb and Zircaloy-2 Tubes In-Reactor, Journal of Nuclear Materials, Vol. 96 (1981), pgs. 297-304.

5. Y1002C001-600, BWR Fuel and Control Materials Properties Handbook.

6. "USAEC Technical Report on Densification of Light Water Reactor Fuels", November 14, 1972.

7. "Creep Collapse Analysis of BWR Fuel Using CLAPS Model", NEDE-20606-P-A, August 1976.

8. "GE14, UO2 and Gad Rod, Design and Licensing Analysis", DRF J11-03057 Study 199, September 1999.

9. "GE14 Fuel Rod Thermal-Mechanical Design Report", NEDC-33241P, November 2005.

10. "ESBWR GE14 Limits", eDRF 0000-0042-3749, June 2005.

Appendix A Statistical Distribution Parameters

The GSTRM statistical fuel rod thermal-mechanical performance analyses require the definition of a mean value and standard deviation for each input parameter. These input parameters can be separated into three categories:

Manufacturing parameters

Model prediction uncertainty

External parameters

The derivation of the input variable statistical distribution parameters is described below for each of these categories.

A.1 Manufacturing Parameters

The statistical analysis input values for the fuel rod manufacturing parameters are determined from the applicable engineering drawings and fabrication specifications. The manufacturing parameter limits may be specified as either in the form of (a) classical design nominal ± a tolerance or as minimum/maximum parameter values, or (b) statistical specifications.

For the case of the classical design nominal ± a tolerance or minimum/maximum specifications, the best estimate (mean) value is taken as the mid-point between the upper and lower tolerance values. The standard deviation of the parameter distribution is determined by assuming that the total range represented by the manufacturing tolerances corresponds to two standard deviations on both sides of the best estimate value.

Certain manufacturing parameters are controlled by the application of statistical specifications. In this case, the distribution parameters are specified and controlled explicitly. Limit values are specified for both the upper and lower 95% confidence interval on the distribution mean. Limit values are also specified for the upper and lower 95/95 distribution limits. [[

]]

[[

]]

Figure A-1 UO_2 Pellet Density Statistical Specification and Sampling Results

A.2 Model Prediction Uncertainty

The GSTRM fuel rod thermal-mechanical performance model has been developed as a best estimate predictor of fuel performance. Verification of the best estimate prediction capability is provided by the extensive experimental qualification documented in Reference 1. Therefore, the best estimate value of a given output parameter, such as fuel center temperature, is provided by GSTRM when all input parameters are set at their best estimate values.

The GSTRM model prediction uncertainty has been derived through recognition that the fuel rod is a highly thermally driven system. Figure A-2 has been extracted from Reference 1 and presents the comparison of GSTRM fuel temperature predictions to experimentally determined temperatures obtained by direct in-reactor measurement by fuel central thermocouples. As

indicated by Figure A-2, the magnitude of the uncertainty in predicted fuel temperatures increases in proportion to the magnitude of the temperature, indicating a constant percentage uncertainty. Since the fuel pellet temperature drop is directly proportional to the fuel rod power level, a constant percentage uncertainty in fuel temperature is equivalent to a constant percentage uncertainty in effective power level. [[

]]

Again, recognizing that the fuel rod is a highly thermally driven system, [[

]]

[[

]]

Figure A-2 GSTRM Fuel Temperature Experimental Qualification

Figure A-3 presents the GSTRM experimental qualification to the available fission gas release measurements. The variability in Figure A-3 is comprised of (1) the uncertainty in the actual operating power history used for the GSTRM fission gas release prediction, (2) the uncertainty in the fuel and cladding fabrication parameters as compared to the nominal values used in the GSTRM fission gas release prediction, (3) the uncertainty in the fuel rod puncture/gas collection measurement of the released fission gas inventory, (4) the uncertainty in the accumulated fuel exposure used to define the total generated fission gas inventory, and (5) the true inherent fission gas release model prediction uncertainty. The degree of conservatism introduced by the applied model prediction uncertainty alone [[ ]] is demonstrated in Figure A-4. Figure A-4 presents a comparison of the fission gas release measurements to the GSTRM predictions for the case of a $+2\sigma$ model uncertainty perturbation. Figure A-4 demonstrates that the model uncertainty perturbation alone results in an overprediction of [[ ]] of the fission gas release measurements.

A.3 External Parameters

The external parameter inputs to GSTRM include the reactor coolant pressure, the cladding corrosion rate, and the corrosion product (crud) buildup rate. The reactor coolant pressure mean and standard deviation are derived from the operational tolerances specified for this parameter at the full rated power condition. The mean value is taken as equal to the nominal specified coolant pressure. The coolant pressure standard deviation is derived from the coolant pressure operational tolerances by assuming that the total range corresponds to two standard deviations on both sides of the best estimate value.

The cladding corrosion rate and corrosion product (crud) buildup rate statistical distribution values are derived from characterization measurements taken on production fuel rods operating in commercial nuclear reactors. For example, Figure A-5 presents a comparison of the design corrosion model to the available GNF corrosion-resistant cladding oxide thickness measurements as determined by eddy current probe lift-off measurements.

[[

]]

Figure A-3 GSTRM Fission Gas Release Experimental Qualification

[[

]]

Figure A-4 Effect of +2 Sigma Bias in Model Prediction Uncertainty on Fission Gas Release Predictions

[[

]]

Figure A-5 Cladding Corrosion Model Statistical Parameters

Appendix B Fuel Rod Processing

GE14 fuel rods are, and GE14E fuel rods will be, fabricated in accordance with materials and processing specifications current at the time of fabrication. Currently, the fuel rod is specified to include [[ ]] Zircaloy-2 barrier cladding. This alloy has been used by GNF since before the introduction of reload quantities of barrier fuel in the early 1980s. The cladding process current at the date of this report is denoted P8. Details of the P8 process, including specifications for finished tubes, are as follows.

[[

]]

The alloy composition plus allowable oxygen level for the Zircaloy-2 and zirconium portions of the tube shell are defined in the table below. Other requirements are currently specified in GNF material specification 26A5757 Rev. 4.

Table B-1 Tube Shell Alloy Composition and Oxygen Concentration

	Concentration (weight %)	
Element	Zircaloy-2	Zirconium
Tin	[[	
Iron		
Chromium		
Nickel		
Iron + Chromium + Nickel		
Oxygen		]]

The tube shell is reduced to tubing on Pilger tube reducers. [[

]] The tube is then polished, inspected, cut to size, given a final NaOH clean and a final inspection.

The Zircaloy-2 portion of the finished tube must meet the chemistry and texture requirements in the tables below. In addition, the finished tube must meet requirements on strength, surface finish, corrosion resistance and other aspects that may impact in-reactor performance. All requirements are currently specified in GNF material specification 26A5798 Rev. 5.

Table B-2 Finished Tube Chemistry - Zircaloy-2 Portion

Element	Maximum Concentration (ppm)
Oxygen	[[
Hydrogen	
Nitrogen	]]

Table B-3 Finished Tube Crystallographic Texture - Zircaloy-2 Portion

Direction	Texture Factor
Longitudinal	[[
Radial	
Transverse	]]

Note: f_l is the fraction of basal poles in the l-direction

Periodically, GNF revises the processing of the cladding, primarily to obtain optimum PCI resistance and corrosion performance as fuel operating strategies and plant water chemistries evolve. The impact of such changes on fuel rod thermal-mechanical design and licensing analyses are assessed as follows.

The material properties of Zircaloy based LWR fuel cladding used in thermal-mechanical design and licensing analyses include:

1. Elastic properties (elastic modulus and Poison's ratio)
2. Thermal expansion coefficients
3. Plastic properties (yield and ultimate stress and failure strain)
4. Creep properties
5. Fatigue properties
6. Irradiation growth properties
7. Corrosion properties

The elastic properties and thermal expansion coefficients are only weakly dependent upon alloy composition and more dependent upon fabrication process, specifically the reduction process and the resulting texture. Since GNF has maintained essentially unchanged texture specifications on fuel rods, the periodic process changes will have negligible impact on these properties.

Likewise, the plastic and creep properties are only weakly dependent upon alloy composition. However, these properties are strongly dependent upon the fabrication process, specifically the final heat treatment. Since GNF tubes are [[ ]] at the end of the fabrication process, the periodic process changes will have negligible impact on these properties.

Also, the fatigue and irradiation growth properties are only weakly dependent upon alloy composition and strongly dependent upon the fabrication process, specifically the final heat treatment and texture. Since GNF tubes are [[ ]] at the end of the fabrication process and the texture specifications are essentially unchanged, the periodic process changes will have negligible impact on irradiation growth properties.

Finally, the corrosion properties have a strong dependency on fabrication process, and specifically on the in-process heat treatments. GNF has recognized this dependency and maintains an on-going program to measure and characterize corrosion (and crud) performance for a variety of operating conditions and plant water chemistries. These characterizations are used to determine corrosion and crud statistical distributions for thermal-mechanical analyses of GNF fuel rods and are updated when the data indicates an update is necessary. Thus the potential changes in corrosion performance of GNF fuel rods due to both periodic process

changes and changing water chemistries in the plants are directly addressed by the GNF design and licensing process.

In summary, the material properties used in GNF fuel rod design and licensing analyses adequately address periodic minor changes in the cladding fabrication process that may be made for GE14E (and GE14) cladding to optimize PCI resistance and corrosion performance. If more significant process changes are made, the applicability and adequacy of the properties will be confirmed. It will also be confirmed that the impact on in-reactor performance and reliability will be acceptable.

NEDO-33242-A Revision 2
Attachment 1

NRC SAFETY EVALUATION

GE14 FOR ESBWR FUEL ROD THERMAL-MECHANICAL DESIGN REPORT

9-13-10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 7, 2010

10-252

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-18
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY LICENSING TOPICAL REPORTS NEDC-33240P, REVISION 01, "GE14E FUEL ASSEMBLY MECHANICAL DESIGN REPORT" AND NEDC-33242P, REVISION 02, "GE14 FOR ECONOMIC SIMPLIFIED BOILING WATER REACTOR FUEL ROD THERMAL-MECHANICAL DESIGN REPORT"

Dear Mr. Head:

On August 24, 2005, GE Hitachi (GEH) Nuclear Energy submitted the Economic Simplified Boiling Water Reactor (ESBWR) design certification application to the staff of the U.S. Nuclear Regulatory Commission. Subsequently, in support of the design certification, GEH submitted the license topical reports (LTRs) NEDC-33240P, Revision 01, "GE14E Fuel Assembly Mechanical Design Report" and NEDC-33242P, Revision 02, "GE14 for Economic Simplified Boiling Water Reactor Fuel Rod Thermal-Mechanical Design Report." The staff has now completed its review of NEDC-33240P, Revision 01 and NEDC-33242P, Revision 02.

The staff finds NEDC-33240P, Revision 01, "GE14E Fuel Assembly Mechanical Design Report" and NEDC-33242P, Revision 02, "GE14 for Economic Simplified Boiling Water Reactor Fuel Rod Thermal-Mechanical Design Report," acceptable for referencing for the ESBWR design certification to the extent specified and under the limitations delineated in the LTRs and in the associated safety evaluation (SE). The SE, which is enclosed, defines the basis for acceptance of the LTR.

The staff requests that GEH publish the revised version of the LTRs listed above within 1 month of receipt of this letter. The accepted version of NEDC-33240P and NEDC-33242P shall incorporate this letter and the enclosed SE and add an "-A" (designated accepted) following the report identification number.

If NRC's criteria or regulations change, so that its conclusion that the LTR is acceptable is invalidated, GEH and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the LTR without revision of the respective documentation.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosures, this document is "DECONTROLLED."

Pursuant to 10 CFR 2.390, we have determined that the enclosed SE contains proprietary information. We will delay placing the non-proprietary version of this document in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any additional information in Enclosure 1 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

The Advisory Committee on Reactor Safeguards (ACRS) subcommittee, having reviewed the subject LTR and supporting documentation, agreed with the staff's recommendation for approval following the May 18, 2010, ACRS subcommittee meeting.

Sincerely,

David B. Matthews, Director
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure:
1. Safety Evaluation (Non-Proprietary)
2. Safety Evaluation (Proprietary)

cc: See next page (w/o enclosure)

DC GEH - ESBWR Mailing List (Revised 08/11/2010)

cc:

Ms. Michele Boyd
Legislative Director
Energy Program
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

Mr. Tom Sliva
7207 IBM Drive
Charlotte, NC 28262

OFFICIAL USE ONLY – ENCLOSURE 2 CONTAINS PROPRIETARY INFORMATION

DC GEH - ESBWR Mailing List

Email

aec@nrc.gov (Amy Cubbage)
APH@NEI.org (Adrian Heymer)
awc@nei.org (Anne W. Cottingham)
bevans@enercon.com (Bob Evans)
bgattoni@roe.com (William (Bill) Gattoni))
BrinkmCB@westinghouse.com (Charles Brinkman)
cberger@energetics.com (Carl Berger)
charles.bagnal@ge.com
charles@blackburncarter.com (Charles Irvine)
chris.maslak@ge.com (Chris Maslak)
CumminWE@Westinghouse.com (Edward W. Cummins)
cwaltman@roe.com (C. Waltman)
Daniel.Chalk@nuclear.energy.gov (Daniel Chalk
david.hinds@ge.com (David Hinds)
david.lewis@pillsburylaw.com (David Lewis)
David.piepmeyer@ge.com (David Piepmeyer)
donaldf.taylor@ge.com (Don Taylor)
erg-xl@cox.net (Eddie R. Grant)
gcesare@enercon.com (Guy Cesare)
GEH-NRC@hse.gsi.gov.uk (Geoff Grint)
GovePA@BV.com (Patrick Gove)
gzinke@entergy.com (George Alan Zinke)
hickste@earthlink.net (Thomas Hicks)
hugh.upton@ge.com (Hugh Upton)
james.beard@gene.ge.com (James Beard)
jerald.head@ge.com (Jerald G. Head)
Jerold.Marks@ge.com (Jerold Marks)
jgutierrez@morganlewis.com (Jay M. Gutierrez)
Jim.Kinsey@inl.gov (James Kinsey)
jim.riccio@wdc.greenpeace.org (James Riccio)
joel.Friday@ge.com (Joel Friday)
Joseph_Hegner@dom.com (Joseph Hegner)
junichi_uchiyama@mnes-us.com (Junichi Uchiyama)
kimberly.milchuck@ge.com (Kimberly Milchuck)
KSutton@morganlewis.com (Kathryn M. Sutton)
kwaugh@impact-net.org (Kenneth O. Waugh)
lchandler@morganlewis.com (Lawrence J. Chandler)
lee.dougherty@ge.com
Marc.Brooks@dhs.gov (Marc Brooks)
maria.webb@pillsburylaw.com (Maria Webb)
mark.beaumont@wsms.com (Mark Beaumont)
matias.travieso-diaz@pillsburylaw.com (Matias Travieso-Diaz)
media@nei.org (Scott Peterson)
mike_moran@fpl.com (Mike Moran)

DC GEH - ESBWR Mailing List

MSF@nei.org (Marvin Fertel)
mwetterhahn@winston.com (M. Wetterhahn)
nirsnet@nirs.org (Michael Mariotte)
Nuclaw@mindspring.com (Robert Temple)
patriciaL.campbell@ge.com (Patricia L. Campbell)
Paul@beyondnuclear.org (Paul Gunter)
peter.yandow@ge.com (Peter Yandow)
pshastings@duke-energy.com (Peter Hastings)
rick.kingston@ge.com (Rick Kingston)
RJB@NEI.org (Russell Bell)
Russell.Wells@Areva.com (Russell Wells)
sabinski@suddenlink.net (Steve A. Bennett)
sandra.sloan@areva.com (Sandra Sloan)
sara.andersen@ge.com (Sara Anderson)
sfrantz@morganlewis.com (Stephen P. Frantz)
stephan.moen@ge.com (Stephan Moen)
steven.hucik@ge.com (Steven Hucik)
strambgb@westinghouse.com (George Stramback)
tdurkin@energetics.com (Tim Durkin)
timothy1.enfinger@ge.com (Tim Enfinger)
tom.miller@hq.doe.gov (Tom Miller)
trsmith@winston.com (Tyson Smith)
Vanessa.quinn@dhs.gov (Vanessa Quinn)
Wanda.K.Marshall@dom.com (Wanda K. Marshall)
wayne.marquino@ge.com (Wayne Marquino)
whorin@winston.com (W. Horin)

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
NEDC-33240P, REVISION 01, "GE14E FUEL ASSEMBLY MECHANICAL DESIGN REPORT"
AND
NEDC-33242P, REVISION 02, "GE14 FOR ESBWR FUEL ROD THERMAL-MECHANICAL DESIGN REPORT"
GLOBAL NUCLEAR FUEL

1.0 INTRODUCTION

By letters dated February 3, 2009 (Reference 1), and June 10, 2009 (Reference 2), General Electric Hitachi (GEH), asked the U.S. Nuclear Regulatory Commission (NRC) to review and approve NEDC-33240P, Revision 01, "GE14E Fuel Assembly Mechanical Design Report," and NEDC-33242P, Revision 02, "GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report." These licensing topical reports, provided by Global Nuclear Fuel (GNF) to GEH, describe the GE14E fuel assembly and fuel rod design, including mechanical specifications and performance aspects, which will serve as the initial fuel design for the Economic Simplified Boiling-Water Reactor (ESBWR). The applicant provided supplemental information in response to staff requests for additional information (RAIs) in letters dated August 23, 2006 (Reference 3), January 21, 2007 (Reference 4), January 26, 2007 (Reference 5), January 4, 2008 (Reference 6), April 18, 2008 (Reference 7), May 9, 2008 (Reference 8), October 8, 2008 (Reference 9), October 24, 2008 (Reference 10), November 12, 2008 (Reference 11), March 30, 2009 (Reference 12), August 13, 2009 (Reference 13), and August 17, 2009 (Reference 14).

NEDC-33240P, Revision 01, and NEDC-33242P, Revision 02, supersede earlier revisions of these licensing topical reports, which did not receive NRC approval.

2.0 REGULATORY EVALUATION

Regulatory guidance for the review of fuel rod cladding materials and fuel system designs appears in Section 4.2, "Fuel System Design," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," issued March 2007 (hereafter referred to as the SRP). The SRP also provides guidance for adhering to General Design Criterion (GDC) 10, "Reactor Design"; GDC 27, "Combined Reactivity Control Systems Capability"; and GDC 35, "Emergency Core Cooling," in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance of the following:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs).
- Fuel system damage is never so severe as to prevent control rod insertion when it is required.

Enclosure 1

- The number of fuel rod failures is not underestimated for postulated accidents.

- Coolability is always maintained.

Using currently approved fuel design requirements and mechanical design methodology, GEH provided the GE14E fuel assembly and fuel rod thermal-mechanical design analyses in NEDC-33240P and NEDC-33242P, respectively. The staff is reviewing these licensing topical reports to ensure that the fuel design criteria and mechanical design methodology remain valid and that the GE14E design adequately addresses the regulatory requirements identified in SRP Section 4.2.

3.0 TECHNICAL EVALUATION

In its review of NEDC-33240P and NEDC-33242P, the staff did the following:

- Verified that the fuel assembly components and fuel rod design criteria are consistent with the regulatory criteria identified in SRP Section 4.2.

- Verified that the fuel mechanical design methodology is capable of accurately or conservatively evaluating each component with respect to its applicable design criteria.

- Verified that the reference GE14E fuel assembly design satisfies the regulatory requirements.

- Verified that the reference GE14E fuel assembly design satisfies all Tier 1, Tier 2, and Tier 2* design requirements specified in the ESBWR design certification document (DCD).

- Verified that the GEH experience database supports the requested operating limits.

In addition to reviewing the material presented in NEDC-33240P, NEDC-33242P, and responses to staff RAIs, the staff conducted two separate audits at the GE-Wilmington offices and performed independent calculations. The staff's audit reports (Reference 15) document the scope of these audits which included reviewing GEH's finite element analysis (FEA) models and methods as well as the GE14E fuel assembly component structural evaluations.

3.1 GE14E Fuel Assembly Design

Section 2 of NEDC-33240P provides a detailed description of the GE14E fuel rod and fuel assembly design, illustrated in Figures 2-1 through 2-9. It is important to note that the GE14E design pertains solely to the ESBWR design; hence, it does not include design variances that would address the differences among the reactors in the boiling-water reactor (BWR) fleet.

In its response to RAI 4.8-1 (Reference 3), regarding the debris resistance of GE14E, GEH provided further details about the debris filter lower tieplate and its effectiveness relative to earlier assembly designs. Based on the applicant's response, RAI 4.8-1 was resolved.

Section 2.2 of NEDC-33240P states, "GE14E fuel assemblies are fabricated in accordance with materials and processing specifications and assembly processes specifications current at the time of fabrication." Appendix B includes similar statements. Changes in component design, materials, or processing specifications may alter the in-reactor behavior of the fuel assembly. GEH currently does not have an approved fuel design change process (similar to GESTAR II) applicable to the ESBWR design. As such, modifications to the fuel assembly design may invalidate the staff's approval of the GE14E fuel design.

In its response to RAI 4.8-2 (Reference 3), regarding GEH's quality control procedures, GEH provided details of the process and process control steps taken to ensure that mechanical properties are not inadvertently altered. Based on the applicant's response, RAI 4.8-2 was resolved.

Process changes must be limited to those that do not impact the physical or mechanical properties of the assembly components or fuel rods. As defined in the Conditions and Limitations (Section 5.0 of this safety evaluation), the staff's approval of GE14E is limited to the detailed description, without deviation, provided in Section 2 of NEDC-33240P. Changes to assembly component design or materials are not permitted without prior staff approval.

3.1.1 Dimensional Compatibility

Section 3.1 of NEDC-33240P describes design analyses performed to ensure that the GE14E fuel assembly remains mechanically compatible with reactor core components, including the top guide, fuel supports, and control blades. In addition, fuel rod and assembly design allowances must address dimensional changes and differential growth throughout the operating lifetime (e.g., irradiation growth and creep). These design requirements, which ensure mechanical compatibility and sufficient design allowances, are consistent with SRP Section 4.2 and therefore are acceptable for application to GE14E.

Relative to earlier GE designs, design allowances for GE14E have increased to accommodate differential growth among assembly components. Figures 3-1 through 3-10 (Reference 1) illustrate design allowances between assembly components and measured growth data from previous poolside measurement campaigns. In its response to RAI 4.8-5 (Reference 3), regarding the applicability of prior measurements and the assumed linearity of the data, GEH stated that the materials and fabrication processes for the fuel rods, water rods, and channels measurements are consistent with those for GE14. Based upon similarities in component design, materials, and processing specifications, the staff accepts the applicability of these growth measurements to the GE14E design. Based on the applicant's response, RAI 4.8-5 was resolved.

GEH provided mechanical design analyses supporting the following design requirements related to differential growth among assembly components:

- Fuel rod upper end plug engagement into the upper tieplate must accommodate differential fuel rod (and tie rod) growth.

- The distance between the top of the fuel rod and the upper tieplate (expansion spring) must accommodate differential fuel rod (and tie rod) growth.

- Water rod upper end plug engagement into the upper tieplate must accommodate differential growth between tie rods and water rods.

- Water rod lower end plug engagement into the lower tieplate must accommodate differential growth between tie rods and water rods.

- Fuel channel overlap with the finger springs must accommodate differential growth between the tie rods and the fuel channel.

In its response to RAI 4.8-3 (Reference 3), regarding part-length fuel rod upper plug engagement with grid spacers and differential growth between part-length fuel rods and water rods, GEH provided a calculation demonstrating sufficient engagement for the part-length fuel rods. Based on the applicant's response, RAI 4.8-3 was resolved.

In its response to RAI 4.8-4 (Reference 3), regarding channel spring engagement and differential assembly growth, GEH provided a calculation demonstrating sufficient channel fastener spring overlap. Based on the applicant's response, RAI 4.8-4 was resolved.

Based upon the staff's review of Section 3.1 and in response to RAIs, the staff finds that the GE14E design satisfies design and regulatory criteria related to dimensional compatibility.

3.1.2 Assembly Component Structural Evaluation

Section 3.3 of NEDC-33240P describes the design criteria for the assembly component structural evaluation. Specifically, for structural components, the combined effective stress may not exceed the material tensile strength. Further, if the combined stresses exceed the material yield strength, then the applicant must justify why the resulting distortion is not significant to component performance and that cyclic loading will not cause fatigue failure. In addition to stress, design criteria include limits on fatigue not exceeding material capability and vibration not resulting in fretting wear. These design criteria are consistent with past GEH fuel designs.

Section 3.4 of NEDC-33240P describes the assembly component structural evaluations given below.

<u>Upper Tieplate</u>

The maximum loading on the upper tieplate occurs during fuel handling when the grapple that is attached to the upper tieplate handle lifts the fuel assembly. The structural evaluation included both finite element analysis (FEA) using the ANSYS code and mechanical testing. ANSYS is an industry-standard FEA code widely applied both within and outside of the nuclear industry. FEA calculations identified that the limiting stress slightly exceeded the yield strength of the material. Staff experienced with ANSYS and FEA conducted an onsite audit of the GEH engineering calculations supporting the upper tieplate structural evaluation. [[

]] The staff reviewed the engineering calculations and supporting tests and found them to be acceptable.

Lower Tieplate

The maximum loading on the lower tieplate occurs during fuel handling when the fuel assembly is seated into the core or into fuel storage racks. The structural evaluation, based upon FEA using the ANSYS code, demonstrated that the maximum loads remained below the material yield strength. The staff conducted an onsite audit of the GEH engineering calculations supporting the lower tieplate structural evaluation and found these calculations acceptable.

Based upon an audit of the structural evaluation which demonstrated that the maximum loads remain below the material yield strength, the staff finds the GE14E lower tieplate design acceptable.

Fuel Rod End Plug

The maximum loading on the fuel rod end plugs occurs during fuel handling. Using conservative assumptions (e.g., less than the full complement of tie rods carrying the weight of the assembly), design calculations demonstrate that loads remain below the material yield strength. Based upon review of the design calculations within Section 3.4 of NEDC-33240P which demonstrate that the maximum loads remain below the material yield strength, the staff finds the GE14E fuel rod end plug design acceptable.

Plenum Spring

The plenum spring is designed to (1) resist an acceleration load during transportation and (2) exert a preload on the pellet column. The only safety function that the plenum spring serves is to ensure that the pellet stack remains undisturbed during transportation. The GEH calculations show that the plenum spring design is capable of performing this function up to the design loads. Based upon review of the design calculations within Section 3.4 of NEDC-33240P which demonstrate that the plenum spring design is capable of satisfying transportation design requirements, the staff finds the GE14E plenum spring design acceptable.

Expansion Spring

The expansion spring is designed to exert a downward force on the fuel rods while allowing for axial growth. The design calculations demonstrate that loads remain below the material tensile strength. Based upon review of the design calculations within Section 3.4 of NEDC-33240P which demonstrate that the maximum loads remain below the material tensile strength, the staff finds the GE14E fuel rod expansion spring design acceptable.

Water Rod

The water rod design is evaluated to accommodate a differential wall pressure and the effects of spacer lift forces from flow or differential component growth.

The GEH calculations show significant design margin for this structure. The staff has reviewed these calculations and finds them acceptable. During an audit at the GE-Wilmington offices (Reference 15), the staff identified a more limiting design requirement for the water rod involving

handling loads during fuel movement. Upon review of the supporting GEH engineering calculations, the staff identified that the combined loading calculations did not consider the water holes (present at the top and bottom of the water rod). GEH postulated that conservative analytical assumptions offset any stress concentration associated with the water holes. The staff still had concerns and in RAI 4.2-33 requested that GEH perform more detailed calculations. In response to RAI 4.2-33 (Reference 12), GEH provided more detailed FEA of the GE14E water rod (including specific modeling of the water holes) which demonstrate that the water rod will not buckle under handling loads. Based upon the results of the more recent FEA calculations, the staff finds the GE14E water rod design acceptable. Based on the applicant's response, RAI 4.2-33 was resolved.

Spacer

The grid spacer designs (including part-length fuel rod configurations) were mechanically tested to measure lateral load before distortion. GEH relied upon testing and analyses previously completed for the GE14 design. In its response to RAI 4.8-8 (Reference 3), regarding seismic and dynamic loads, GEH stated that the GE14 fuel assemblies for BWR/4-6 have been demonstrated to be acceptable for the following peak seismic and dynamic accelerations: [[ ]] in the horizontal direction and [[ ]] in the vertical direction and would bound the shorter GE14E design. ESBWR standard plant seismic analysis shows peak safe shutdown earthquake (SSE) accelerations of [[ ]] in the horizontal direction and [[ ]] in the vertical direction. 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. Based on the applicant's response, RAI 4.8-8 was resolved.

Consistent with past practice, testing was performed on unirradiated fuel assembly components to simulate beginning-of-life conditions (i.e., before irradiation hardening). In its response to RAI 4.8-6 (Reference 8), on the use of unirradiated material conditions, GEH discussed the potential embrittlement of spacer grids as a result of hydrogen uptake. Testing on spacers precharged with hydrogen was completed to simulate the effects of in-reactor corrosion. These tests confirm that the spacers maintain fracture resistance up to very high hydrogen levels. While these impact tests were completed to evaluate handling loads, they provide evidence of end-of-life performance during postulated accidents. Based on the applicant's response, RAI 4.8-6 was resolved.

Channel

In addition to its inclusion in the LOCA and seismic lateral load testing, the channel is designed to withstand steady-state and transient differential pressure. The structural evaluation, based upon FEA using the ANSYS code, demonstrated that the maximum loads remained below the material yield strength. The staff conducted an onsite audit of the GEH engineering calculations supporting the channel structural evaluation and found these calculations acceptable. Based upon an audit of the structural evaluation which demonstrated that the maximum loads remain below the material yield strength, the staff finds the GE14E channel design acceptable.

3.1.3 Assembly Design Evaluation

Flow-Induced Vibration

Section 3.4.1.10 of NEDC-33240P describes flow-induced vibration (FIV) testing performed on the GE14 assembly design. Based on a comparison of these results to earlier testing, GEH concludes that assembly differences do not have a significant effect on FIV performance. The staff was not entirely convinced by this qualitative argument and had concerns regarding which aspects of GE14E (e.g., spacer elevations) necessitate FIV testing. In its response to RAI 4.8-7 (Reference 3), regarding FIV, GEH stated that FIV testing will be performed on GE14E before fuel loading. While the staff strongly endorses validation by testing, the specifics of the proposed FIV testing raised concerns.

During an audit at the GE-Wilmington offices (Reference 15), the staff reviewed several internal GE documents comparing the response to RAI 4.8-7 to past detailed FIV test programs on different legacy fuel bundle designs. Review of these FIV test reports revealed sensitivities in measured acceleration and displacement that challenge the limited FIV testing proposed for GE14E.

The older FIV tests were broader in scope—investigating a range of temperatures, flow rates, and steam quality on Root Mean Squared (RMS) acceleration and displacement. Steam quality refers to the proportion of saturated steam in a saturated water/steam mixture. A steam quality of 0 indicates 100% water while a steam quality of 1 indicates 100% steam. Notable observations from the staff's audit include the following (see Figure 3-1):

- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

[[

]]

Figure 3-1 RMS Acceleration versus Steam Quality

After several iterations (as documented in GEH's response to RAI 4.8-7, Supplements 1 and 2 (Refs. 3 and 8), the staff agreed to the required FIV testing for GE14E. GEH's response to

RAI 4.8-7, Supplement 3 (Reference 11), documents the basis of the proposed FIV testing and acceptance criteria. As requested by the staff, GEH also discussed past FIV test results in its response to RAI 4.8-7.

Based on the proven in-reactor performance of GE14 and the lower flow rate of the ESBWR, the staff found the limited-scope FIV testing (outlined in the response to RAI 4.8-7, Supplement 3) for GE14E to be acceptable, with conservative penalties on the acceptance criteria to account for known sensitivities to temperature and quality. However, the staff did not accept this limited-scope FIV test program to justify more substantial fuel design changes and/or new fuel design features that may influence the sensitivity of RMS acceleration to rod location, flow rate, temperature, and steam quality.

The last sentence of the proposed FIV acceptance criteria stated, "Any GE14E locations that exceed the adjusted peak GE14 value at the respective elevation will be dispositioned individually." This acceptance criterion was too broad and was not in compliance with the level of specificity expected within DCD Tier 1 inspections, tests, analyses, and acceptance criteria (ITAAC). GEH retracted this statement in its response to RAI 4.8-7, Supplement 4 (Reference 15) and GEH proposed a revision to section 3.3.3 of NEDC-33240P to specify testing requirements for GE14E to satisfy the ESBWR FIV ITAAC. The staff finds the proposed acceptance criteria to be acceptable for inclusion in the approved version of the LTR. Based on the applicant's response, RAI 4.8-7 was resolved.

Seismic/Dynamic Loading

Section 3.4.1.11 of NEDC-33240P (Reference 1) describes the structural capability of the GE14E assembly and assembly components to withstand seismic/dynamic loading. GEH relied upon testing and analyses previously completed for the GE14 design. As described in section 3.1.2 of this report under the heading "Spacer" it was concluded in the response to RAI 4.8-8 that GE14E fuel assemblies are qualified for the seismic and dynamic loads defined by the ESBWR standard plant seismic analysis. Therefore, based on the applicant's response, RAI 4.8-8 was resolved.

With respect to assembly lift, GEH has incorporated acceptance criteria in DCD Tier 1, Table 2.1.1-3 stating 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.

Channel Bow and Control Blade Insertion

Section 4 of NEDC-33240P (Reference 1) describes the fuel assembly channel and compatibility with the control blades. Figure 4-4 of NEDC-33240P provides dimensions, including gaps between fuel channels and control blades. Compared with current designs, the ESBWR N-lattice design includes a larger gap at both the corner and midwall relative to the C-lattice and S-lattice plants. Operating experience has shown that control blade friction occurs only at the C- and S-lattice plants. D-lattice plants (which have a larger gap than the N-lattice design) have experienced minimal issues with control blade friction.

In its response to RAI 4.8-9 (Reference 3), regarding channel bow and control blade insertion, GEH discussed margin to control blade interference relative to the current fleet. In addition to physical differences, the ESBWR control rod drive system would actively detect any control

blade hangup resulting from channel-to-blade friction. The same fuel management and operating guidelines used to minimize control blade interference in the current fleet will be applied to the ESBWR. The ESBWR will maintain the same technical specification surveillance requirements and actions as the current fleet.

Based upon the improved design margins of the ESBWR N-lattice (relative to the C- and S-lattice), along with fuel management guidance and surveillance, the staff finds that GEH has adequately addressed control blade interference. Based on the applicant's response, RAI 4.8-9 was resolved.

3.2 GE14E Fuel Rod Design Evaluation

Section 2 of NEDC-33242P (Reference 2) describes in detail the GE14E fuel rod and fuel pellet design. Section 3 of NEDC-33242P identifies the design criteria used to evaluate the adequacy of the GE14E fuel rod design. The fuel rod thermal-mechanical design criteria are consistent with past GE designs (e.g., GE14).

In its response to RAI 4.8-10 (Reference 3), regarding deviations from approved methodology, GEH stated that the methodology, including the treatment of model uncertainties and manufacturing tolerances, is identical to that used to confirm compliance of the GE14 design with GESTAR for BWR/3-6 and the advanced BWR. Based on the applicant's response, RAI 4.8-10 was resolved.

In its response to RAI 4.8-11 (Reference 3), regarding the continued applicability of approved methods to ESBWR conditions, GEH demonstrated that the currently approved methods are within the qualification database for the GE14E fuel rod design and ESBWR operating conditions. Based on the applicant's response, RAI 4.8-11 was resolved.

Fuel Rod Internal Pressure

The design criterion for rod internal pressure, as defined in Section 3.1 of NEDC-33242P, is that the outward creep rate of the cladding will not exceed the fuel pellet irradiation swelling rate. This design requirement for no cladding liftoff is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E.

In addition to reviewing Sections 3 and 4 of NEDC-33242P, the NRC staff completed independent calculations using the fuel rod thermal-mechanical performance code FRAPCON-3. In support of the staff's calculations, GEH provided fuel specifications, manufacturing tolerances, and limiting axial and nodal power histories. FRAPCON-3 is a best estimate fuel rod performance code that is calibrated against a wide range of applicable empirical data. In order to obtain design calculations with the best estimate FRAPCON-3 code, manufacturing tolerances were deterministically modeled and rod power penalties were employed in lieu of modeling uncertainties (e.g., cladding creep, cladding strain, fuel swelling). Based upon engineering judgment, a 10-percent rod power penalty conservatively bounds the modeling uncertainties related to fission gas release. With respect to cladding creep prediction uncertainties, the cladding creep model in FRAPCON-3 (based upon Zry-4) was conservative for modeling GE14E's Zry-2 cladding.

The NRC staff performed FRAPCON-3 sensitivity studies to determine the worst set of initial conditions (e.g., pellet diameter, stack height) to minimize plenum volume, maximize fission gas release, and maximize rod internal pressure. Following these sensitivity studies, the staff performed several FRAPCON-3 calculations to evaluate the GE14E fuel rod design with respect to rod internal pressure and void volume. Table 3.2-1 lists the results of these calculations.

Table 3.2-1 FRAPCON-3 Calculations—Rod Internal Pressure

Case	Description	FRAPCON-3 Calculated Results	
		Fission Gas Release (%)	Rod Internal Pressure (psia)
1	UO_2—Worst case inputs along TMOL rod power curve	[[ ]]	[[ ]]
2	UO_2—Worst case inputs along TMOL+10% rod power curve	[[ ]]	[[ ]]
3	UO_2—Worst case inputs along TMOL+10% with extended knee (at [[ ]] GWd/MTU)	[[ ]]	[[ ]]
4	UO_2—Worst case inputs along TMOL+10% with extended knee (at [[ ]] GWd/MTU), power history more aggressive to achieve licensed burnup in shorter duration (i.e., extended power uprate fuel usage)	[[ ]]	[[ ]]
5	UO_2—Worst case inputs along TMOL with three AOO excursions (1 hour +25% power) at 10, 15, and 59 GWd/MTU exposure	[[ ]]	[[ ]]
6	$UGdO_2$—Worst case inputs along TMOL+10% with extended knee	[[ ]]	[[ ]]
7	UO_2 PLR—Worst case inputs along TMOL+10% with extended knee	[[ ]]	[[ ]]

The internal pressures calculated with FRAPCON-3 remain below the critical pressure that would cause an outward creep of the cladding. Best estimate critical pressure is approximately 3,000 pounds-force per square inch absolute (psia). However, large modeling uncertainties associated with cladding creep and fuel swelling rate conservatively set an upper tolerance critical pressure at 2,050 psia (1,000 psia over system pressure). The maximum calculated rod internal pressure (Case 3, uranium dioxide (UO_2)) remains below this conservative estimate of critical pressure. Furthermore, none of the FRAPCON-3 cases predict a widening of the fuel-to-cladding gap—indicative of cladding liftoff. The UO_2 Part Length Rod (PLR) (Case 7) is less limiting because of a greater plenum volume, relative to the UO_2 rod. The $UGdO_2$ rod (Case 6) is less limiting because of a greater plenum volume and a more benign power curve, relative to the UO_2 rod.

While performing these calculations, the staff identified a discrepancy between the fission gas release calculated by GSTR-M and FRAPCON-3. Upon further investigation, including benchmarking identical best estimate cases, the staff determined that the GSTR-M [[

]] In response to a staff request, GEH completed an evaluation (Reference 5) in accordance with the provisions of 10 CFR Part 21, "Reporting of Defects and Noncompliance," of a potential nonconservatism in the GE thermal-mechanical methodology, GSTR-M. GEH concluded that the error in GSTR-M was not reportable.

After reviewing the GEH evaluation (Reference 5), the staff continued to have concerns related to fission gas release and rod internal pressure calculated with GSTR-M and proposed a penalty to maintain a conservative cladding liftoff analysis. In an attempt to identify compensating conservatism within the fuel performance methodology and negate the application of a proposed critical pressure penalty, GEH issued a supplement to the 10 CFR Part 21 Notification (Reference 6). Upon further review, the staff maintained its concerns with respect to the GSTR-M cladding liftoff analysis and concluded that a [[ ]] penalty on critical pressure was required (Reference 16).

Although the staff's independent calculations show that the GE14E fuel design satisfies the rod internal pressure design criterion, the accuracy of future reload analyses with GSTR-M may be in question, especially with more aggressive rod power histories than those cited in NEDC-33242P and used in staff calculations. For example, the [[ ]] penalty on critical pressure was based on a maximum linear heat generation rate of [[ ]] kilowatts per foot. A higher rod power may necessitate a larger penalty. The required penalty on critical pressure relates to the use of GSTR-M in the cladding liftoff analysis. Migration to an NRC-approved, up-to-date fuel rod thermal-mechanical code (e.g., PRIME) may eliminate the need for such a penalty.

To resolve staff concerns regarding the use of GSTR-M for fuel rod design analysis, the staff asked GEH to provide the final ESBWR Cycle 1 thermal-mechanical operating limits for all Cycle 1 fuel rod designs. In its response to RAI 4.8-17 (Reference 15), GEH provided the final thermal-mechanical operating limits and power suppression factors for ESBWR Cycle 1. GEH's supporting rod internal pressure design calculations included the [[ ]] of critical pressure.

Based upon GEH's design calculations, including the [[ ]], provided in response to RAI 4.8-17 as confirmed by the independent calculations, the staff finds the GE14E fuel rod design acceptable for the rod power histories specified in Tables 1 and 2 of GEH's response to RAI 4.8-17 (Reference 15). Section 5 of this safety evaluation cites a limitation on GE14E rod power history. Based on the applicant's response, RAI 4.8-17 was resolved.

The NRC staff originally generated RAIs 4.8-15 and 4.8-16 to address the GSTR-M issues. The GEH 10 CFR Part 21 evaluation and staff assessments supersede the information in the responses to these RAIs. Hence, GEH responses to RAIs 4.8-15 and 4.8-16 (Reference 3) were not a factor in the staff's approval.

Fuel Pellet Temperature/Thermal Overpower Limit

Table 3-1 of NEDC-33242P specifies the design criteria, stating "The maximum fuel center temperature (Tcenter) shall remain below the fuel melting point (Tmelt)." However, Section 3.2 of NEDC-33242P states the following:

> To achieve this objective, the fuel rod is evaluated to ensure that fuel melting during normal steady-state operation and core wide anticipated operational occurrences is not expected to occur. This fuel temperature limit is specified to ensure that sudden shifting of molten fuel in the interior of fuel rods, and subsequent potential cladding damage, can be positively precluded.

During review of the ESBWR DCD Tier 2* fuel design criteria, the staff was concerned with allowing fuel melting during any AOO—moderate or infrequent classification, local or core wide—and identified this as an open item within the ESBWR DCD safety evaluation report with open items (Reference 17). GEH subsequently revised the ESBWR DCD to reflect a more restrictive requirement precluding fuel centerline melting during any AOO. Avoidance of fuel melting during AOOs is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E.

In addition to reviewing Sections 3 and 4 of NEDC-33242P, the NRC staff completed independent calculations using the fuel rod thermal-mechanical performance code FRAPCON-3. As was done for the rod internal pressure cases, manufacturing tolerances were deterministically modeled and rod power penalties were employed in lieu of modeling uncertainties.

The staff performed FRAPCON-3 sensitivity studies to determine the worst set of initial conditions (e.g., pellet diameter, cladding oxide thickness) to maximize fuel temperature. Following this sensitivity study, the staff performed several FRAPCON-3 calculations to evaluate the GE14E fuel rod design in combination with proposed thermal overpower (TOP) limits at preventing centerline fuel melt. Table 3.2-2 lists the results of these calculations.

Table 3.2-2 FRAPCON-3 Calculations—Fuel Temperature

Case	Description	FRAPCON-3 Calculated Results	
		Fuel Temp. (°F) before Spike	Fuel Temp. (°F) at Spike
1	UO_2—Worst case inputs along TMOL with [[ ]] TOP at knee	[[ ]]	[[ ]]
2	UO_2—Worst case inputs along TMOL+10% with [[ ]] TOP at knee	[[ ]]	[[ ]]
3	UO_2—Worst case inputs along TMOL+10% with [[ ]] TOP at extended knee	[[ ]]	[[ ]]
4	$UGdO_2$—Worst case inputs along TMOL with [[ ]] TOP at knee	[[ ]]	[[ ]]
5	$UGdO_2$—Worst case inputs along TMOL+10% with [[ ]] TOP at knee	[[ ]]	[[ ]]
6	$UGdO_2$—Worst case inputs along TMOL+10% with [[ ]] TOP at extended knee	[[ ]]	[[ ]]

The fuel temperatures calculated with FRAPCON-3 remain below incipient centerline melting conditions. Fuel melting temperature is defined as follows:

UO_2 Tmelt = 5,080 degrees F—58 degrees F per 10 GWd/MTU
$(U,Gd)O_2$ Tmelt = 5,080 degrees F—58 degrees F to 60 degrees F per 10 GWd/MTU (based on 8 percent Gd)

The FRAPCON-3 calculations included a 10-percent increase in rod power (above Thermal Mechanical Operating Limits (TMOL)) to cover modeling uncertainties and an extended knee to cover more aggressive fuel management. The fuel temperatures calculated with FRAPCON-3 remained below melting temperatures for UO_2 fuel rods with a [[ ]]-percent TOP and $UGdO_2$ fuel rods with a [[ ]]-percent TOP. The $UGdO_2$ rod (Case 6; [[ ]] degrees F) is more limiting because of reduced thermal conductivity of gadolinia fuel pellets, relative to UO_2 pellets (Case 3; [[ ]] degrees F).

Based upon the material presented in NEDC-33242P and the staff's independent calculations, the staff finds that the GE14E fuel rod design and prescribed TOP limits ([[ ]]-percent UO_2, [[ ]]-percent $UGdO_2$) satisfy the fuel temperature design criteria.

Cladding Strain/Mechanical Overpower Limit

The design criterion for fuel cladding strain (high-rate strain during AOOs), as defined in Section 3.3 of NEDC-33242P, is that cladding permanent deformation (plastic plus creep) remain below 1.0 percent. While SRP Section 4.2 defines an allowable total cladding strain limit of 1.0 percent permanent (plastic plus creep), the fuel vendor is responsible for (1) defining the total strain capability of its fuel rod design/cladding alloy, (2) providing evidence supporting this strain capability, and (3) demonstrating that this design criterion is not exceeded during AOOs.

Mechanical testing under prototypical loading on irradiated cladding specimens provides acceptable evidence of a cladding alloy's strain capability. While irradiation damage under normal operation promotes an increase in yield strength (and lower ductility), the formation of zirconium hydrides within the cladding (resulting from the absorption of hydrogen during cladding corrosion) limits the strain capability of the fuel rod cladding. As a result of this relationship, fuel vendors need to specify a design limit on cladding hydrogen content corresponding to the extent of the empirical database supporting the cladding strain specified and acceptable fuel design limit (SAFDL) per 10 CFR 50 Appendix A, GDC-10.

During its review of the ESBWR DCD Tier 2* fuel design criteria, the staff raised issues with the lack of any corrosion limits and the empirical database supporting the cladding strain design limit. The staff identified these as open items in the ESBWR DCD safety evaluation with open items (Reference 17). After several iterations (RAI 4.2-2 and 4.2-4, Refs. 3, 4, 9, and 11), GEH proposed the following cladding strain design limit and cladding hydrogen design limit, along with supporting empirical database:

- cladding strain limit = [[ ]]
- cladding strain limit = [[ ]]
- cladding hydrogen limit = [[ ]]

The applicant explained the basis of these design limits and the breakpoint in cladding strain in its response to RAIs 4.2-2 S03 and 4.2-4 S02 (Reference 8). Although the applicant used various information provided in response to RAIs 4.2-2 and 4.2-4 and during ESBWR audits collectively to justify these design limits, only the above specified strain and hydrogen limits are NRC reviewed and approved.

In addition to the material presented in NEDC-33242P, in response to the RAIs, and in presentations during audits, the staff has access to proprietary mechanical test data on similar BWR Zry-2 cladding (see RAI 4, Reference 18) that reinforce the GE14E cladding strain design limit. Based upon the above review, the staff finds the GE14E cladding strain limits and hydrogen limit acceptable.

In addition to reviewing Sections 3 and 4 of NEDC-33242P, the NRC staff completed independent calculations using the fuel rod thermal-mechanical performance code FRAPCON-3. As was done for the rod internal pressure cases, manufacturing tolerances were deterministically modeled and rod power penalties were employed in lieu of modeling uncertainties (e.g., fuel swelling).

The staff performed FRAPCON-3 sensitivity studies to determine the worst set of initial conditions (e.g., pellet diameter, cladding thickness) to maximize cladding strain. Following this sensitivity study, the staff performed several FRAPCON-3 calculations to evaluate the GE14E fuel rod design in combination with proposed mechanical overpower (MOP) limits to ensure that cladding strains were maintained below empirically based limits. The basis for GEH's specified acceptable fuel design limit (SAFDL) on cladding strain is discussed further below. Table 3.2-3 lists the results of these calculations.

Table 3.2-3 FRAPCON-3 Calculations—Fuel Cladding Strain

Case	Description	FRAPCON-3 Calculated Results	
		Plastic Strain (%)	Total Strain (%)
1	UO_2—Worst case inputs along TMOL with [[ ]] MOP at knee	[[ ]]	[[ ]]
2	UO_2—Worst case inputs along TMOL with [[ ]] MOP at knee	[[ ]]	[[ ]]
3	UO_2—Worst case inputs along TMOL+10% with [[ ]] MOP at knee	[[ ]]	[[ ]]
4	UO_2—Worst case inputs along TMOL+10% with [[ ]] MOP at extended knee	[[ ]]	[[ ]]
5	$UGdO_2$—Worst case inputs along TMOL with [[ ]] MOP at knee	[[ ]]	[[ ]]
6	$UGdO_2$—Worst case inputs along TMOL with [[ ]] MOP at knee	[[ ]]	[[ ]]

The fuel cladding strain calculated with FRAPCON-3 remained below 1.0 percent total, as well as below the GEH empirically based, hydrogen-dependent strain SAFDL. Since the calculated strain remained below the more limiting 1.0-percent total strain, separate cases at varying levels of hydrogen (based on burnup and corresponding to hydrogen-based strain SAFDLs) were not necessary. Cases investigated the impact of a power penalty (to account for fuel swelling modeling uncertainty) applied to the initial power and then separately applied to the power excursion.

Based upon the material presented in NEDC-33242P and the staff's independent calculations, the staff finds that the GE14E fuel rod design and prescribed MOP limits ([[ ]]-percent UO_2,[[ ]]-percent $UGdO_2$) satisfy the fuel cladding strain design criteria.

SAFDLs on fuel rod cladding strain and fuel centerline melting are employed to preclude fuel rod cladding failure because of pellet/cladding mechanical interaction during rapid overpower AOOs. However, as described in SRP Section 4.2, these design limits may not provide sufficient protection to preclude fuel cladding failure because of pellet/cladding interaction stress-corrosion cracking (PCI/SCC) under certain sustained cladding loading conditions. In its response to RAI 4.8-12 (Reference 3), regarding the PCI/SCC resistance of GE14E fuel, GEH provided results from past and recent ramp test programs that are applicable to GE14E's barrier design. This information shows that margin exists between current operating limits and an empirically based lower failure threshold such that PCI/SCC failures would not occur during power maneuvering. Furthermore, GE's barrier cladding design has been proven to reduce PCI/SCC susceptibility during both power maneuvering and AOO-type scenarios. As a result, PCI/SCC fuel cladding failure is unlikely during any AOO scenario that involves a sustained power excursion and does not already predict fuel rod failure from violating other SAFDLs (e.g., Minimum Critical Power Ratio (MCPR), cladding strain, centerline melt). Hence, there is reasonable assurance that fuel cladding failure would not be underestimated. Based on the applicant's response, RAI 4.8-12 was resolved.

Cladding Oxidation and Corrosion Product Buildup

As described in Section 5.3 of NEDC-33242P, the fuel rod thermal-mechanical design evaluations include the effects of cladding oxidation and corrosion product buildup (e.g., crud) on the fuel rod surface. The statistical treatment of crud buildup and oxidation within the design analyses is consistent with current fuel designs (e.g., GE14). This approach is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E.

In addition to explicitly accounting for the effects of cladding oxidation and crud, the staff requires that fuel vendors establish a design limitation on cladding oxidation. This upper bound on cladding oxidation defines (1) the limit of oxidation included in the design analyses and (2) the limit of oxidation under which cladding oxide spallation and hydride blisters have not been observed. Currently approved fuel performance models rely on uniform mechanical properties along the axial and circumferential directions of the fuel rod cladding. Localized cladding defects (e.g., spallation, hydride blisters) may significantly impact fuel rod stress and strain calculations and ultimately the ability to accurately predict cladding failure.

Earlier versions of NEDC-33242P did not define a cladding oxidation limit that satisfied the staff position. During its review of the ESBWR DCD Tier 2* fuel design criteria, the staff raised issues with the lack of any corrosion limits. The staff identified this as an open item in the ESBWR DCD safety evaluation with open items (Reference 17). In concert with its discussions of hydrogen and cladding strain (responses to RAIs 4.2-2 and 4.2-4; Refs. 3, 4, 9, and 11), GEH proposed a cladding oxidation limit of [[ ]] along with the supporting empirical database.

GEH detailed the basis of the cladding oxide design limit in its response to RAIs 4.2-2 S03 and 4.2-4 S02 (Reference 8). For pressurized-water reactor fuel designs, cladding oxide limits have been selected to minimize the possibility of spallation (in order to ensure uniform mechanical properties). Attachment A of (Reference 8) describes the difficulty with implementing a similar approach for BWRs. In its response, GEH provides hot cell examinations on medium- and high-burnup fuel rods from Duane Arnold and Limerick Unit 1 that show no evidence of hydride localization at spalled locations. Figure A-3 of Reference 8 provides pool-side cladding liftoff measurements, and Figure A-4 gives confirmatory hot cell metallography. Based upon the information provided in NEDC-33242P and in response to RAIs, the staff finds the proposed cladding corrosion design limit of [[ ]] acceptable for GE14E. Based on the applicant's responses, RAIs 4.2-2 and 4.2-4 were resolved.

GE14E fuel cladding corrosion shall be limited such that cladding oxidation thickness remains less than [[ ]] and cladding hydrogen content remains less than [[ ]]. In Reference 8 Figure A-5, GEH proposed an "action level" on measured lift-off beyond the design limit of [[ ]]. This proposed "action level" is not approved.

Cladding Hydrogen Content

Hydrogen trapped within the fuel rod as a result of manufacturing may be absorbed by the cladding. The design criterion for fuel pellet hydrogen content is intended to prevent fuel rod failure because of localized, internal hydriding. GEH relies on the manufacturing process and controls to restrict hydrogenous contaminants from all sources during the manufacturing

process. This design requirement is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E criterion.

In addition to internal hydrogen sources, a design limitation on absorbed hydrogen in the fuel rod cladding from all possible sources has been established as discussed above.

Cladding Creep Collapse

The design criterion for cladding structural instability is that fuel cladding creep collapse will not occur. This design requirement is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E criterion.

The finite element methods assume a maximum initial ovality, maximum delta-pressure (e.g., minimum fill gas pressure and no fission gas release), and no support provided by fuel pellets. In addition, the maximum overpressure AOO is applied at end-of-life conditions. Because of an identical fuel rod design, the current GE14 creep collapse analysis bounds the GE14E fuel rod design.

Based upon the material presented in NEDC-33242P, the staff finds that the GE14E fuel rod design satisfies the fuel cladding creep collapse criterion.

Fuel Rod Stresses and Strain

The design criterion is that effective cladding stresses and strain will not result in fuel rod failure. This design requirement is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E criterion.

The methods employed to calculate effective stress and strain are consistent with the currently approved methods used for GE14. Relying on GE14 analyses, GEH stated that the large design margins present in GE14 are applicable to GE14E.

In its response to RAI 4.8-14 (Reference 3), regarding the modeling of the barrier liner, GEH noted that certain fuel rod thermal-mechanical analyses explicitly address the impact of the liner on heat transfer and cladding strength. Other applications conservatively neglect the zirconium barrier.

Based upon the material presented in NEDC-33242P, the staff finds that the GE14E fuel rod design satisfies the fuel cladding stress and strain criterion. Based on the applicant's response, RAI 4.8-14 was resolved.

Cladding Fatigue Analysis

The design criterion is that fatigue life usage will not exceed the material fatigue capability resulting in fuel rod failure. This design requirement is consistent with SRP Section 4.2 and therefore acceptable for application to GE14E criterion.

Section 4.2.3 of NEDC-33242P describes the cladding fatigue analysis methodology. The conservative power cycles listed in Table 4-3 of NEDC-33242P as well as the statistical methodology are consistent with current fuel designs. In its response to RAI 4.8-13

(Reference 3), regarding Zircaloy fatigue data, GEH provided the empirical database used to justify the upper and lower 95/95 fatigue curves. The RAI response further justifies the conservatism of the fatigue analysis relative to the SRP guidelines.

Based upon the material presented in NEDC-33242P and the RAI response, the staff finds that the GE14E fuel rod design satisfies the fuel cladding fatigue criterion. Based on the applicant's response, RAI 4.8-13 was resolved.

3.3 Operating Experience

Historically, the staff has relied on lead test assembly programs to generate in-reactor operating experience for new assembly design features or materials in order to validate performance and model predictions. Since no ESBWR designs have been built and the current fleet is incompatible with the 10-foot tall GE14E design, lead test assemblies are not possible. However, since the GE14E assembly component designs and materials are consistent with currently operating designs, insight into the anticipated in-reactor performance of GE14E is achievable.

Section 6 of NEDC-33242P provides information related to GEH's extensive fuel operating experience. GE14E shares the same fuel rod and spacer designs and materials as the GE14 design, with the exception of rod length and spacer pitch. No systematic failures have been reported on the nearly 1.4 million GE14 fuel rods manufactured. This operating experience provides reasonable assurance that normal operational failure modes such as cladding collapse, grid-to-rod fretting, cladding liftoff, cladding stress and strain, excessive corrosion, and cladding fatigue are unlikely for GE14E. In addition, the continued as-anticipated performance of GE14 and ongoing surveillance programs has validated model predictions (e.g., growth, creep, corrosion).

The fuel design limits, operating rod power limits, and projected rod burnups for GE14E are identical to those for GE14. Based upon the information presented in NEDC-33240P and NEDC-33242P, the staff finds the operating experience database supporting the GE14E fuel assembly design review of sufficient breadth to cover the range of burnup and operating conditions under consideration.

4.0 CONCLUSION

Based upon its review of NEDC-33240P and NEDC-33242P, the staff finds the application of GEH's fuel thermal-mechanical design criteria and methodology acceptable for GE14E. Furthermore, the staff finds the GE14E fuel assembly and fuel rod design acceptable for use in the ESBWR. Licensees referencing this topical report will need to comply with the conditions listed in Section 5.0. Furthermore, licensees will need to ensure that the GE14E fuel design criteria are consistent with the final ESBWR Tier 2* fuel design criteria in DCD Section 4.2 and Appendix 4B.

Since the GE14E fuel design meets the criteria and methodology defined in Section 2, the staff has concluded that the GE14E fuel design is acceptable.

5.0 CONDITIONS AND LIMITATIONS

Licensees referencing NEDC-33240P and NEDC-33242P must ensure compliance with the following six conditions and limitations:

1) Following the fuel assembly and fuel rod mechanical design methodology described in NEDC-33240P and NEDC-33242P, the licensee must ensure that all of the design criteria are satisfied for each refueling cycle.

2) The GE14E fuel assembly design is restricted to the design specifications provided in Section 2 of NEDC-33240P and the fuel rod cladding material processing specifications provided in Appendix B to NEDC-33240P. Changes in component design, materials, or processing specifications may alter the in-reactor behavior of the fuel assembly and invalidate the staff's approval.

3) The GE14E fuel assembly design is approved up to a peak pellet exposure of [[ ]] and a maximum operating time of [[ ]].

4) GE14E fuel cladding corrosion shall be limited such that cladding oxidation thickness remains less than [[ ]] and cladding hydrogen content remains less than [[ ]].

5) As described in Section 3.1.3, GE14E must complete the required FIV testing and satisfy the acceptance criteria (e.g., measured GE14E RMS acceleration below adjusted, measured GE14 RMS acceleration at every location) before loading into an ESBWR.

6) GE14E rod power history (peak linear heat generation rate versus peak pellet exposure) must remain at or below the thermal-mechanical operating limits and power suppression factors provided in Tables 1 and 2 of Reference 13.

6.0 REFERENCES

1. GNF Letter MFN 09-095, "NEDC-33240P, Licensing Topical Report, GE14E Fuel Assembly Mechanical Design Report, Revision 1, January 2009," February 3, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML090700714, ML090700717).

2. GNF Letter MFN 09-377, "Submittal of NEDC-33242P Revision 2 and NEDO-33242 Revision 2, 'GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report,'" June 10, 2009 (ADAMS Accession Nos. ML091630213, ML091630214, ML091630215).

3. GNF Letter MFN 06-297, "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 4.2-2 through 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 (ADAMS Accession Nos. ML062480252, ML062480254, ML062480255). MFN 06-297, Supplement 9, RAI 4.8-8, April 19, 2010 (ADAMS Accession No. TBD)

4. GNF Letter MFN 07-040, Jason S. Post (GE) to Document Control Desk (NRC), "Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GSTRM," January 21, 2007 (ADAMS Accession No. ML072290245).

5. GNF Letter 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 4.2-2S01, 4.2-4S01 and 4.8-16S01—Supplement," January 26, 2007 (ADAMS Accession Nos. ML070380118, ML070380120).

6. GNF Letter MFN 07-040, Supplement 1, Dale E. Porter (GEH) to Document Control Desk (NRC), "Part 21 Notification: Adequacy of GE Thermal-Mechanical Methodology, GESTR-M—Supplement 1," January 4, 2008 (ADAMS Accession Nos. ML080100670, ML080100672).

7. GNF Letter MFN 08-391, "Response to Portion of NRC Request for Additional Information Letter No. 110—Related to ESBWR Design Certification Application—RAI Number 4.8-7 Supplement 2," April 18, 2008 (ADAMS Accession Nos. ML081130488, ML081130490).

8. GEH Letter 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 Supplement 3, 4.2-4 Supplement 2 and 4.8.6 Supplement 1," May 9, 2008 (ADAMS Accession Nos. ML081350380, ML081350381).

9. GNF Letter MFN 08-757, "Response to Portion of NRC Request for Additional Information Letter No. 243—Related to ESBWR Design Certification Application—RAI Numbers 4.2-24 Supplement 1,4.2-26 Supplement 1,4.2-31," October 8, 2008 (ADAMS Accession Nos. ML082880090, ML082880089).

10. GNF Letter MFN 08-789, "Response to Portion of NRC Request for Additional Information Letter No. 229—Related To Design Control Document (DCD) Revision 5—RAI Number 4.2-2 Supplement 4," October 24, 2008 (ADAMS Accession No. ML083020523).

11. GEH Letter MFN 08-867, "Response to Portion of NRC Request for Additional Information Letter No. 231—Related to ESBWR Design Certification Application—RAI Number 14.3-398 and NRC Request for Additional Information Letter No. 229 Related to ESBWR Design Certification Application RAI Number 4.8-7S03," November 12, 2008 (ADAMS Accession Nos. ML083190139, ML083190140).

12. GEH Letter MFN 08-946, Revision 1, "Revised Response to Portion of NRC Request for Additional Information Letter No. 243—Related To Design Control Document (DCD) Revision 5—RAI Number 4.2-33," March 30, 2009 (ADAMS Accession Nos. ML090910653, ML090910654).

13. GEH Letter MFN 09-542, "Response to Portion of NRC Request for Additional Information Letter No. 350 Related to ESBWR Design Certification Application—Reactor—RAI Number 4.8-17," August 13, 2009 (ADAMS Accession No. ML092300406).

14. GEH Letter MFN 08-867 Supplement 1, "Response to Portion of NRC Request for Additional Information Letter No. 350 Related to ESBWR Design Certification Application—Reactor—RAI Number 4.8-7 S04," August 17, 2009 (ADAMS Accession Nos. ML092310271, ML092310272).

15. NRC Memorandum, "Audit Report and Summary (2007 & 2008) for Global Nuclear Fuels Control Blade and Fuel Assembly Design," December 22, 2008 (ADAMS Accession No. ML083230072).

16. NRC Memorandum, "Re-Assessment of GEH GSTR-M Part 21 Notification," February 24, 2009 (ADAMS Accession No. ML090510434).

17. NRC Memorandum, "Safety Evaluation with Open Items Report Input for the ESBWR Design Certification Section 4.2, Fuel System Design," March 29, 2007 (ADAMS Accession No. ML070920337).

18. WCAP-15942-NP-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287," March 2006 (ADAMS Accession Nos. ML061110272, ML061110247, ML061110351)

Enclosure 3

MFN 10-274

Affidavit