ENCLOSURE 2

M180059

GNF3 Generic Compliance with NEDE-24011-P-A (GESTAR II), NEDO-33879, Revision 2, March 2018

Non-Proprietary Information – Class I (Public)

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Global Nuclear Fuel

NEDO-33879
Revision 2
March 2018

Non-Proprietary Information - Class I (Public)

GNF3 Generic Compliance with NEDE-24011-P-A (GESTAR II)

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EXECUTIVE SUMMARY

This report presents generic information relative to the GNF3 fuel design and analyses of GE Boiling Water Reactors (BWRs) for which GNF provides fuel. The scope of assessments is in accordance with the fuel licensing acceptance criteria as specified in NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (Reference 1) and is often called the Amendment 22 process. The criteria in GESTAR II establish the basis for evaluating new fuel designs, developing the critical power correlation for these designs, and determining the applicability of generic analyses. This process has been applied in the licensing of the GNF2, GE14, GE12, GE13, and GE11 fuel designs, References 2 through 6, respectively.

In addition to the generic information documented herein, the fuel introduction process includes two additional activities: plant-specific cycle-independent New Fuel Introduction (NFI) analyses, and cycle-unique analyses. The NFI report documents the cycle-independent plant-specific analyses for use by the Licensee as input to the plant's 10 Code of Federal Regulations (CFR) 50.59 evaluation of the NFI. The cycle-unique analyses, which are part of the normal reload process, are documented in the Supplemental Reload Licensing Report (SRLR).

GNF3 was designed for mechanical, nuclear, and thermal-hydraulic compatibility with the other GNF fuel designs. The design has features of GE14 and GNF2 fuel including Pellet-Cladding Interaction (PCI) resistant barrier cladding, high performance spacers, Part Length Rods (PLRs), thick corner/thin wall channel, and axially varying distributions of enrichment and burnable absorber. The GNF3 design is a 10x10 array of fuel rods, [[ ]] and includes both Long Part Length Fuel Rods (LPLRs) and Short Part Length Fuel Rods (SPLRs).

The GNF3 design and licensing is based on the PRIME (Reference 7) Thermal-Mechanical (T-M) methodology. The downstream analyses use PRIME inputs in accordance with NEDO-33173, Supplement 4-A, Revision 1, *Implementation of PRIME Models and Data in Downstream Methods*, November 2012 (Reference 8).

As stated in GESTAR II, "Fuel design compliance with the fuel licensing acceptance criteria constitutes US NRC acceptance and approval of the fuel design without specific US NRC review." All of the criteria defined in GESTAR II have been met for the GNF3 fuel design.

REVISIONS

Number	Purpose of Revision
0	Initial documentation of GNF3 compliance with GESTAR II.
1	Deletion of an inaccurate statement in Section 3.3.2 pertaining to low void conditions. Editorial corrections to one sentence in Section 3.2.10, the location of a vertical line in Figure 3-16 for the lower bound of the mass flux range, and the titles of Tables 3-25 and 3-26.
2	Deletion of a statement in Section 3.2.7 pertaining to fuel lift behavior that is applicable to GNF2 but not to GNF3.

ACRONYMS AND ABBREVIATIONS

Term	Definition
1D	One Dimensional
3D	Three Dimensional
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARTS	APRM, RBM, Technical Specifications
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
ATWSI	ATWS Instability
[[	]]
[[	]]
BOC	Beginning of Cycle
BOEC	Beginning of Equilibrium Cycle
BOL	Beginning of Life
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	BWR Owners' Group
CFR	Code of Federal Regulations
CPR	Critical Power Ratio
ΔCPR	Delta Critical Power Ratio
CRDA	Control Rod Drop Accident
DRC	Doppler Reactivity Coefficient
ECCS	Emergency Core Cooling System
ECP	Engineering Computer Program
ECPR	Calculated bundle critical power divided by experimentally determined bundle critical power
EOC	End of Cycle
EOEC	End of Equilibrium Cycle
EOL	End of Life
EOOS	Equipment Out-of-Service
EPG	Emergency Procedure Guideline
EPU	Extended Power Uprate
FFRO	Fast Recirculation Flow Runout

Term	Definition
FIV	Flow-Induced Vibration
FLE	Fuel Loading Error
FLR	Full-Length Rod
FSAR	Final Safety Analysis Report
GE	General Electric Company
GEH	GE Hitachi Nuclear Energy
GESTAR II	General Electric Standard Application for Reload Fuel, (Current Revision: NEDE-24011-P-A-24, March 2017)
GESTR	General Electric Stress and Thermal Analysis of Fuel Rods
GEXL	General Electric Critical Quality versus Boiling Length Correlation
GNF / GNF-A	Global Nuclear Fuel – Americas, LLC
GSF	Geometric Stacking Factor
HFCL	High Flow Control Line
HPCI	High Pressure Coolant Injection
ICPR	Initial Critical Power Ratio (CPR value for the GNF3 hot channel at the initial core flow)
IRLS	Idle Recirculation Loop Startup
LHGR	Linear Heat Generation Rate
LHGRFAC	Linear Heat Generation Rate Factors
$LHGRFAC_f$	Linear Heat Generation Rate Flow Factor
$LHGRFAC_p$	Linear Heat Generation Rate Power Factor
LOCA	Loss-of-Coolant Accident
LPLR	Long Part Length Fuel Rod
LTP	Lower Tie Plate
LUA	Lead Use Assembly
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCNP	A General Monte Carlo N-Particle Transport Code
MCPR	Minimum Critical Power Ratio
$MCPR_f$	Minimum Critical Power Ratio Flow Factor
$MCPR_p/K_p$	Minimum Critical Power Ratio Power Factor
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
Methods LTR	Applicability of GE Methods to Expanded Operating Domains Licensing Topical Report
MFLE	Mislocated Fuel Loading Error

Term	Definition
MIP	MCPR Importance Parameter
MOC	Middle of Cycle
MOEC	Middle of Equilibrium Cycle
MOP	Mechanical Overpower
MSIV	Main Steam Isolation Valve
NCL	Natural Circulation Line
NFI	New Fuel Introduction
NRC	Nuclear Regulatory Commission
NSF	Zirconium Alloy Containing Niobium, Tin, and Iron
ODYN	1D Transient Model
ODYSY	GE Best-Estimate Frequency Domain Stability Code
OLMCPR	Operating Limit MCPR
PANACEA	GNF BWR Core Simulator
PCI	Pellet-Cladding Interaction
PCT	Peak Cladding Temperature
PD	Pellet Density
PLR	Part Length Rod
PRFO	Pressure Regulator Failure – Open
PRIME	The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance, NEDC-33256P, NEDC-33257P, and NEDC-33258P
PWR	Pressurized Water Reactor
RMS	Root Mean Square
RPT	Recirculation Pump Trip
SE	Safety Evaluation
SER	Safety Evaluation Report
SFRO	Slow Flow Runout
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit MCPR
SPLR	Short Part Length Fuel Rod
SRLR	Supplemental Reload Licensing Report
T-M	Thermal-Mechanical
TD	Theoretical Density
TGBLA	GNF BWR lattice physics code
TIP	Traversing In-Core Probe

Term	Definition
TMOL	Thermal Mechanical Operating Limit
TOP	Thermal Overpower
TRACG	Transient Reactor Analysis Code (GE proprietary version)
US	United States
UTP	Upper Tie Plate

1.0 INTRODUCTION

This report presents generic information relative to the GNF3 fuel design and analyses of GE BWRs for which GNF provides fuel. The organization and scope of assessments is in accordance with the fuel licensing acceptance criteria as specified in GESTAR II (NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel) and often called the Amendment 22 process. The Amendment 22 process was approved by the Nuclear Regulatory Commission (NRC) in July 1990 (Reference 9). The fuel licensing acceptance criteria included in GESTAR II establishes the basis for evaluating new fuel designs, developing the critical power correlation for these designs, and determining the applicability of generic analyses to these new designs. Compliance with the fuel licensing acceptance criteria constitutes United States (US) NRC acceptance of the fuel design without specific US NRC review. This process has been previously applied to the GNF2 fuel design (Reference 2), GE14 fuel design (Reference 3), GE12 fuel design (Reference 4), GE13 fuel design (Reference 5), and the GE11 fuel design (Reference 6).

The GNF3 design and licensing is based on the PRIME (Reference 7) T-M methodology. The downstream analyses use PRIME inputs in accordance with NEDO-33173, Supplement 4-A, Revision 1, *Implementation of PRIME Models and Data in Downstream Methods*, November 2012 (Reference 8).

In addition to the generic information documented herein, the fuel introduction process includes plant-specific cycle-independent NFI analyses. The NFI report (Section 4.2) documents the cycle-independent plant-specific analyses for use by the Licensee as input to the plant's 10 CFR 50.59 evaluation of the NFI. The cycle-specific analyses, which are part of the normal reload process, are documented in the SRLR.

The fuel licensing criteria from GESTAR II are included in the applicable sections. The features and design characteristics of the GNF3 fuel bundle are described in Section 2.0. The evaluations, meeting the requirements of GESTAR II, are presented in Section 3.0. Each section or sub-section of Section 3.0 includes the requirement from GESTAR II. Section 4.0, Licensing Application, describes the manner in which the Licensees use this report.

2.0 GNF3 Fuel Design Description

GNF3 is an evolution from the GNF2 design licensed under the GESTAR II process (Reference 1). Table 2-1 provides a summary of the GNF3 design as compared to the GNF2 design.

A GNF3 bundle schematic is shown in Figure 2-1. The GNF3 design consists of [[ ]] fuel rods and [[

]] fuel rods terminate just past the [[ ]] and are designated as SPLRs. Additionally, [[ ]] fuel rods terminate just past the [[ ]] and are designated as LPLRs. Eight peripheral fuel rods are used as tie rods. The GNF3 lattice arrangement is shown in Figure 2-2. The rods are spaced and supported by the Upper Tie Plate (UTP) and Lower Tie Plate (LTP) and [[ ]] spacers over the length of the fuel rods. The GNF3 channel has a [[

]] The GNF3 channel interacts with the LTP [[ ]] to control leakage flow.

The fuel rods consist of high-density ceramic UO_2 or (U, Gd)O_2 fuel pellets stacked within Zircaloy-2 cladding. The cladding contains an inner zirconium liner providing PCI resistance. The fuel rod is evacuated and backfilled with helium to [[ ]] atm. Fuel rod dimensions are given in Table 2-1.

2.1 NEW DESIGN FEATURES

GNF3 was designed for mechanical, nuclear, and thermal-hydraulic compatibility with previous GE/GNF fuel designs. The design includes many features of the GE10, GE11/13, GE12/14, and GNF2 fuel designs including PCI resistant barrier cladding, high performance spacers, PLRs, thick corner/thin wall channel, and axial variation of gadolinia and/or enrichment. New or improved features included in GNF3 are:

- [[

]]

A discussion of the GNF3 fuel design features is provided below.

2.2 FUEL ASSEMBLY CONFIGURATION

The GNF3 design consists of a total of [[ ]] fuel rods [[ ]] and [[ ]] The rods are spaced and supported by the UTP and LTP and [[ ]] spacers over the length of the fuel rods. The GNF3 design uses a [[ ]] similar to the GNF2 design. The GNF3 fuel rod cladding includes the zirconium barrier liner that has been incorporated in GE6 through GNF2 designs [[ ]] The GNF3 [[ ]] defines the external envelope of the rodded bundle.

2.2.1 Water Rod

The [[ ]] provides neutron moderation in the top region of the bundle where in-channel void fractions are higher while [[ ]] as shown in Figure 2-1 and Figure 2-2. [[ ]] has also been improved in the bundle [[ ]] provides for a [[

]]

The GNF3 [[ ]] to the UTP. [[ ]] The [[ ]] is connected to the LTP [[ ]] as shown in Figure 2-3. [[ ]]

2.2.2 Channel

[[ ]] The GNF3 [[ ]] has a [[ ]] as shown in Figure 2-4. [[ ]] thus providing similar structural and in-reactor behavioral characteristics as prior channel designs. Thus, the GNF3 [[ ]] can be considered a [[ ]] The [[ ]] provides for a more effective and efficient [[ ]] The interface between the [[ ]] and the LTP [[ ]] channel-LTP interface. Thus, [[ ]] the leakage flow holes in the LTP are designed to provide similar overall intra-assembly bypass flow as GE14 and GNF2.

The material of the GNF3 [[ ]] is a zirconium alloy containing niobium (1%), tin (1%), and iron (0.35%) (NSF) (Reference 10). Traditional Zr-2 and Zr-4 channel materials are also optional. NSF is effectively resistant to fluence gradient and shadow corrosion-induced channel bow while maintaining equivalent creep bulge characteristics as Zircaloy. The NSF alloy therefore provides for enhanced resistance to channel-control blade interference.

2.2.3 Spacer

The GNF3 spacer is a similar design as the GNF2 spacer. The GNF3 spacer is a [[ ]] and the spacer structural material is Alloy X-750. The spacer structure is improved for [[ ]] The spacer grid has [[ ]] The spacer design also includes [[

]]

The GNF3 spacer is shown in Figure 2-5. [[
]]

2.2.4 Fuel Pellets and Fuel Rods

[[
]] The GNF3 FLRs and LPLRs are [[

]] A GNF3 fuel rod schematic is shown in Figure 2-6.

2.2.5 Tie Plates

The GNF3 UTP has been [[
]] This was accomplished by [[ ]] as shown in Figure 2-7.

The LTP for GNF3 is a similar design as the GNF2 LTP. The GNF3 LTP is an assembly consisting [[ ]] The LTP [[ ]] reduces the probability of foreign-material-related-fuel-rod failures by limiting foreign material (e.g., wire, springs, drill turnings) that can enter the fuel assembly from the bottom plenum of the reactor vessel. The LTP [[ ]] has been designed to have equivalent hydraulic resistance as previous GE/GNF LTPs. A schematic of the GNF3 LTP is shown in Figure 2-8.

2.2.6 Enrichment and Gadolinia Loading

The GNF3 fuel bundle consists of fuel rods of varying levels of U-235 enrichment and gadolinia concentration. The GNF3 fuel rod enrichment loading consists of varying enrichment(s) from natural uranium up to 5.0% U-235. The GNF3 fuel rod gadolinia loading consists of varying concentration(s) from no gadolinia up to [[ ]] The complexity and variation of the overall enrichment and gadolinia loading for GNF3 is developed considering the intended fuel reload application and manufacturing requirements.

Table 2-1 GNF3 and GNF2 Nominal Dimensions

Fuel Assembly	**GNF2[1]**	**GNF3[1]**
Lattice array	10x10	same
Total number of fuel rods	92	[[
Number of FLRs / LPLRs / SPLRs	78 / 8 / 6	
Number of water rods	2	
Rod-to-rod pitch (cm)	[[	
Typical assembly fuel mass (kgU)		
Fuel Rod		
Active fuel length (cm), FLR / LPLR / SPLR		
Plenum length (cm), FLR / LPLR / SPLR		
Rod length (cm), FLR / LPLR / SPLR	]]	]]
Cladding material	Zr-2 with zirconium inner liner	same
Cladding tube diameter, outer (cm)	[[	[[
Cladding tube wall thickness [[ ]] (cm)		
Zirconium barrier inner liner thickness (cm)		
Pellet diameter, outer (cm)		
Fuel Pellet Density (PD)		
Fuel column Geometric Stacking Factor (GSF)		
Fuel column stack density (g/cm^3)		
Helium backfill pressure (atm)		
Water Rod		
Tube material		
Tube diameter, outer (cm), [[ ]]		
Tube wall thickness (cm), [[ ]]	]]	]]
Spacer		
Material	Alloy X-750	same
Number of spacers, [[ ]]	8	[[ ]]
Spacer height and axial locations	Figure 2-9	Figure 2-9
Tie Plates		
LTP bypass flow control	[[	[[
UTP height (cm)		
UTP grid height (cm)		
UTP grapple clearance (cm)	]]	]]

Table 2-1 GNF3 and GNF2 Nominal Dimensions, continued

Channel	GNF2[1]	GNF3[1]
Material	Zr-2 / Zr-4 / NSF	NSF
Inside width (cm)	[[	[[
Inside corner radius (cm)		
Thickness (cm), [[ ]]		
Thickness (cm), [[ ]]	]]	]]

[1] Values provided to show relative differences between GNF3 and GNF2 and are not intended as a source for detailed design input
[2] Theoretical Density (TD)
[3] Gd_2O_3 Concentration, percent by weight (GC)

[[

]]

Figure 2-1 GNF3 Fuel Bundle Assembly

[[

]]

Figure 2-2 GNF3 Lattice Arrangement

[[

]]

Figure 2-3 GNF3 Water Rod End Plug Connection

[[

]]

Figure 2-4 GNF3 Channel

[[

]]

Figure 2-5 GNF3 Spacers

[[

]]

Figure 2-6 GNF3 Fuel Rod

[[

]]

Figure 2-7 GNF3 and GNF2 Upper Tie Plates

[[

]]

Figure 2-8 GNF3 Lower Tie Plate

[[

]]

Figure 2-9 GNF3 and GNF2 Axial Spacer Locations

3.0 EVALUATION

The fuel licensing acceptance criteria included within GESTAR II are established for evaluating new fuel designs, developing critical power correlation for these designs, and determining the applicability of generic analyses to these new designs. GNF3 fuel design compliance with the fuel licensing acceptance criteria constitutes US NRC acceptance and approval of the fuel design without specific US NRC review.

This process has been previously applied to the GNF2, GE14, and GE12 10x10 fuel designs, and GE13 and GE11 9x9 fuel designs. NRC audits of the previous applications of the GESTAR II new fuel process are documented in References 11, 12, and 13. Subsequent to the GNF2 compliance report audit, changes to the compliance criteria in GESTAR II were completed via Amendment 33 to GESTAR II. (Reference 14)

The GESTAR II fuel licensing acceptance criteria and the bases for compliance of GNF3 fuel with these criteria are presented in the following sections.

3.1 GENERAL CRITERIA

3.1.1 NRC-Approved Models

GESTAR II Section 1.1.1.A: "NRC-approved analytical models and analysis procedures will be applied."

NRC approved methodologies as documented in GESTAR II have been used to demonstrate compliance for each of the analyses required in Subsection 3.2 through Subsection 3.15. Analytical models and analysis procedures for the evaluation of each criterion are described in each respective section of this report.

This section addresses the applicability of the current methods and methodologies to the GNF3 fuel design. Most approved methodologies include an Engineering Computer Program (ECP) that encodes part or all of each methodology within an algorithmic framework. For the most commonly used ECPs, discussion of any effects of the unique characteristics of GNF3 has been included. Generally, the codes are unaffected by these characteristics given flexibility of their modeling, input structure, or representation of these characteristics in the approved methodology.

3.1.1.1 Nuclear Methods

Lattice Physics

TGBLA06 (References 15 and 16) is the two-dimensional transport corrected diffusion theory model used to model the details of nuclear transport at the lattice level. No changes to the methodology were required to properly model GNF3 beyond those described in the GNF2 compliance report. Figure 3-1 to Figure 3-4 demonstrate the applicability of TGBLA06 using direct comparisons to Monte Carlo (Reference 17) at 0.0 and 65.0 GWd/ST exposure. The average reactivity bias for all cases at each moderator density is provided in Figure 3-1 and Figure 3-3. Figure 3-2 and Figure 3-4 extend this comparison to standard deviations of pin-by-pin fission density differences between MCNP and TGBLA06.

[[

]] The GE14 reference values are expressed as the average ± the standard deviation of the differences of the variable under consideration for the given reference fuel product (e.g., the green dotted lines in Figure 3-1 represent the average ± 1 standard deviation of the reactivity differences between TGBLA and MCNP at Beginning of Cycle (BOC)).

The effect of analyzing the GNF3 designs is within the accepted levels of uncertainty in lattice physics modeling and is consistent with the corresponding reference GNF2 and GE14 analyses. The introduction of GNF3 is thus not significant to the TGBLA06 lattice physics methodology.

[[

]]

Figure 3-1 TGBLA06 Reactivity Benchmark for GNF3 at BOC

[[

]]

Figure 3-2 TGBLA06 Fission Density Benchmark for GNF3 at BOC

[[

]]

Figure 3-3 TGBLA06 Reactivity Benchmark for GNF3 at 65 GWd/ST

[[

]]

Figure 3-4 TGBLA06 Fission Density Benchmark for GNF3 at 65 GWd/ST

Steady-State Core Simulator

PANAC11 (Reference 16) is the Three Dimensional (3D) core simulator utilized for design, licensing, and core monitoring. PANAC11 correctly handles varying axial geometry in nuclear and thermal-hydraulic modeling through use of its lattice dependent geometry, nodal thermal-hydraulic properties, and axial meshing routines. This allows PANAC11 to handle multiple PLR, [[ ]] features when modeled at the bundle/lattice library level.

PANAC11, like other GNF thermal-hydraulic codes, uses the "New Dix" void-quality correlation in its thermal-hydraulics treatment and accounts for bundle leakage and water rod flow by parameterized input from ISCOR simulations. As explained further in Section 3.1.1.2, the New Dix void quality correlation has been shown to be applicable to GNF3.

3.1.1.2 Thermal-Hydraulic Methods

ISCOR09 (Reference 18) is a thermal-hydraulic core analysis program wherein different fuel types can be designated to represent various types of bundles within a core. The introduction of various PLR rod heights or other features in GNF3, such as [[

]]

The GE void correlation "New Dix" is applicable for all GE BWR fuel designs, including 10x10 lattices [[

]] Qualification of advanced designs like GE12, GE14, GNF2, and GNF3 has been evaluated with full-scale experimental pressure drop data described in Section 3.5. Correct prediction of the pressure drop requires accurate prediction of the void fraction throughout the length of the bundle. In addition, the void fraction correlation is indirectly qualified via comparison with sub-channel analysis methods as shown in Figure 3-5. Therefore, the GE void fraction correlation forming the basis for all currently approved methodologies is applicable to GNF3 fuel designs.

[[

]]

Figure 3-5 Axial Void Calculation on GNF3 at High Power Conditions from the Dix Correlation and Sub-channel Based Calculation

3.1.1.3 Safety Limit

The facets of the Safety Limit MCPR (SLMCPR) calculation include discussion of the adaptive technology, establishment of uncertainties, and method of SLMCPR calculation. (References 19 and 20)

Adaption

The adaptive methodology is applied within PANAC11. There are no changes to this methodology as a result of introduction of GNF3.

Uncertainties

There is no change to the SLMCPR uncertainties for GNF3 application. In accordance with the Safety Evaluation (SE) for the SLMCPR methodology, the uncertainties must be verified for new fuel designs. These restrictions are evaluated in Section 3.6. The verification of the pin power/R-factor uncertainty is also supported by the analysis in Section 3.1.1.1.

SLMCPR Calculation

GESAM02 embodies the implementation of the revised SLMCPR methodology using PANAC11 physics models to calculate Critical Power Ratio (CPR) distribution (References 19, 20, and 21). There are no changes required to determine the SLMCPR for GNF3.

3.1.1.4 Transient Analysis

Interface and Collapse

CRNC-06 (References 22 and 23) collapses the 3D cross sections supplied by PANAC11 into One Dimensional (1D) cross section fits acceptable for ODYNM10 or ODYNV09 (References 22, 23, and 24). The resulting cross section fits and thermal-hydraulic information is collected and stored on the CRNC-06 output file for ODYNM10/ODYNV09 and other codes to read. Detailed GNF3 geometry information is written under auxiliary dataset names. The capabilities of both PANAC11 and CRNC-06 to perform this function are adequate for the modeling of GNF3.

Transient Simulator

ODYNM10/ODYNV09 (References 22, 23, and 24) retrieves cross section information and thermal-hydraulic information from the CRNC-06 output file. The thermal-hydraulics and void correlation implemented in ODYN is applicable to GNF3.

TRACG02 (References 25, 26, and 27) and TRACG04 (Reference 28) is also approved for use for transients (Anticipated Operational Occurrences (AOOs)). TRACG02 will not be used for GNF3. Simulation of GNF3 does not pose challenges to the modeling capabilities of this technology, and all fuel type specific limitations have been addressed for this method.

Hot Bundle Simulation

TASC-03 (Reference 29) is a single hot channel thermal hydraulic analysis code and requires detailed bundle geometry input that designates different types of rod groups within the bundle. The two types of PLR within GNF3 can be handled in TASC-03 by designating an additional PLR rod group and giving the required geometry inputs. TASC-03 also uses the "New Dix" void quality correlation. This void correlation has been shown to be acceptable for application to GNF3 fuel bundles. The approved methodology is applicable to GNF3.

3.1.1.5 Stability

ODYSY05 (Reference 30) is capable of modeling axially varying bundle designs. This is accomplished by requiring axial geometry to be specified through input on a nodal basis.

ODYSY05 also uses the "New Dix" void quality correlation. This void correlation has been shown to be acceptable for application to GNF3 fuel bundles. The approved methodology is applicable to GNF3.

3.1.1.6 Channel Bow

The methodology used to assess the effect of channel bow on R-Factor (Reference 31), and thus critical power continues to be applicable because during normal operation the shape and degree to which the channel is bowed is dependent upon the material and the fluence gradient across the channel and is generally independent of channel design. The effect on individual rod power peaking continues to be evaluated as a function of the degree of channel bow. While numerical sensitivities of the critical power will differ between the various fuel types, GNF3 included, the process continues to be applicable.

3.1.1.7 Thermal-Mechanical Methods

An important part of the GNF3 design and licensing bases is the fuel rod T-M design and licensing analyses. The PRIME model (Reference 7) has been applied to the GNF3 design consistent with its current application with GNF2. A discussion of the PRIME T-M compliance is provided in Section 3.2.

3.1.1.8 LOCA Analysis Methods

The LOCA analysis models (GESTAR II-US Supplement Section S.2.2.3.2) and application methodology with respect to GNF3 are discussed in Section 3.11. No modifications are needed for application to the GNF3 fuel design.

3.1.2 Lead Use Assemblies

GESTAR II Section 1.1.1.B: "New design features will be included in lead use assemblies."

The new design features of GNF3 relative to previously approved designs are described in detail in Section 2.0. Three Lead Use Assembly (LUA) programs have been initiated to gain operating experience for GNF3 prior to insertion of reload quantities; these LUA programs are described below.

Four (4) prototypical GNF3 LUAs have been loaded into the LaSalle Unit 2 plant in the US and began operation in February 2015. NEDC-33862P, "GNF3 Lead Use Assembly for LaSalle County Station, Unit 2," September 2014, has been provided to the US NRC per the requirements of Letter, T.A. Ippolito (NRC) to R.E. Engel (GE), "Lead Test Assembly Licensing," MFN 182-81, September 23, 1981.

In addition, four (4) prototypical GNF3 LUAs have been loaded into the River Bend plant in the US and began operation in February 2015. NEDC-33863P, "GNF3 Lead Use Assembly for River

Bend Station, Unit 1," October 2014, has been provided to the US NRC per the requirements of Letter, T.A. Ippolito (NRC) to R.E. Engel (GE), "Lead Test Assembly Licensing," MFN 182-81, September 23, 1981.

Finally, four (4) prototypical GNF3 LUAs have been loaded into the Hatch 2 plant in the US and began operation in February 2017. NEDC-33874P, "GNF3 Lead Use Assembly for Edwin I. Hatch Nuclear Plant, Unit 2," September 2016, has been provided to the US NRC per the requirements of Letter, T.A. Ippolito (NRC) to R.E. Engel (GE), "Lead Test Assembly Licensing," MFN 182-81, September 23, 1981.

These lead use programs constitute compliance with this criterion.

3.1.3 Post-Irradiation Fuel Examination

GESTAR II Section 1.1.1.C: "The generic post–irradiation fuel examination program approved by the NRC will be maintained (GESTAR References 1–3 and 1–4)."

The generic post-irradiation fuel examination program approved by the NRC for previous fuel designs will be maintained for GNF3. Descriptions of the NRC-approved fuel examination program required for new fuel designs, and subsequent revisions to the program, were documented in correspondence between GE and the NRC listed below.

1. Letter, J. S. Charnley (GE) to C. H. Berlinger (NRC), "Post Irradiation Fuel Surveillance Program," MFN 218-83, November 23, 1983. (GESTAR Reference 1-3)

2. Letter, L. S. Rubenstein (NRC) to R. L. Gridley (GE), "Post Irradiation Fuel Surveillance," MFN 014-84, January 18, 1984.

3. Letter, J. S. Charnley (GE) to L. S. Rubenstein (NRC), "Fuel Surveillance Program," MFN 024-84, February 29, 1984.

4. Letter, J. S. Charnley (GE) to L. S. Rubenstein (NRC), "Additional Details Regarding Fuel Surveillance Program," MFN 065-84, May 25, 1984.

5. Letter, L. S. Rubenstein (NRC) to R. L. Gridley (GE), "Acceptance of GE Proposed Fuel Surveillance Program," MFN 087-84, June 27, 1984. (GESTAR Reference 1-4)

6. Letter, G. A. Watford (GNF-A) to R. Pulsifer/R. Caruso (NRC), "GNF Fuel Surveillance Plan," FLN-2001-009, May 7, 2001.

These requirements will be carried out for the GNF3 fuel design. A summary of those requirements is provided below:

- LUAs of new designs are placed in operation before commercial applications (i.e., full reloads).

- LUAs are pre-characterized before operation.
- The new features in LUAs will be selectively inspected based on availability of the fuel as influenced by plant operation.
- A visual inspection of exterior surfaces will be performed after discharge on at least 12 bundles of the new product from at least two early reloads.
- As required (e.g., an anomaly is observed or a fuel failure occurs), more extensive evaluations may be performed, including destructive testing.

The fuel examination program meets GNF's commitment to the NRC and provides evidence that the bundles have performed as expected.

3.1.4 New Fuel-Related Licensing Issues

GESTAR II Section 1.1.1.D: "New fuel related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval."

On August 31, 1994, the NRC issued Information Notice 94-64 (Reference 32) that discussed information obtained on the performance of high burnup fuel. The notice expresses concern that the data does not support the current licensing limits for certain accidents and beyond design basis events (Anticipated Transients Without Scram (ATWS)). In the GE11 NRC audit report (Reference 11), the auditors commented: "...the ATWS evaluation did not include consideration of the new issues regarding power oscillations that have been identified by the ATWS/stability studies currently in progress. The criteria and the GE11 design should be reexamined for adequacy with respect to fuel related effects on the conclusions of these studies when they are complete."

Studies have been completed by the BWR Owners' Group (BWROG) to assess the effect of oscillations on the consequences of an ATWS and to evaluate the effectiveness of operator actions to mitigate the effects of oscillations (Reference 33). The studies were based on a bounding 8x8 fuel design and showed that "...the level of safety expected from the requirements of 10 CFR 50.62 is not compromised because of stability" and that "Operator actions to inject boron and reduce reactor water level were...the best options for mitigating oscillations in ATWS events."

Additional quantitative TRACG ATWS Instability (ATWSI) studies have been performed on a GNF3 core in a representative BWR/4 and BWR/6 plant on the Maximum Extended Load Line Limit Analysis (MELLLA) boundary. The analyses of GNF3 fuel for operation on the MELLLA rod line demonstrate that mitigation using water level reduction is effective to suppress ATWSI thermal-hydraulic oscillations. The Peak Cladding Temperature (PCT) compliance to the ATWS PCT acceptance criterion of 2,200°F and the fuel and containment integrity are met for GNF3 on the MELLLA boundary.

The GNF3 ATWSI analyses support the conclusions in NEDO-32047 (Reference 33) in the MELLLA domain that the thermal-hydraulic oscillations do not impose additional containment duty, the current Emergency Procedure Guideline (EPG) actions provide an adequate level of protection regarding containment integrity, and it is not expected to distort the core, impede core cooling or prevent shutdown.

ATWSI studies will be included with the GNF3 NFI report for a plant licensed for operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain. (See Section 4.2)

3.1.5 NRC Separate Review

GESTAR II Section 1.1.1.E: "If any of the criteria in Subsection 1.1 are not met for a new fuel design, that aspect will be submitted for review by the NRC separately."

All of the criteria specified in Subsections 1.1 of GESTAR II are met by the GNF3 fuel design as documented in this report. Therefore, there are no aspects of the GNF3 design that require a separate review by the NRC.

3.2 THERMAL-MECHANICAL

The T-M analysis of the GNF3 fuel assembly, including the fuel rod and assembly components, is performed to demonstrate compliance with the criteria identified in Subsection 1.1.2 of GESTAR II.

The GNF3 analyses utilize the following two processes from Section 1.1.2.A. of GESTAR II:

1. Either worst tolerance assumptions are applied or probabilistic analyses are performed to determine statistically bounding results (i.e., upper 95% confidence).

2. Operating conditions are taken to bound the conditions anticipated during normal steady–state operation and AOOs.

The GNF3 fuel rod and assembly component analyses were performed in accordance with the above guidance to demonstrate compliance to the fuel design criteria in Section 1.1.2.B of GESTAR II. The T-M design criteria from GESTAR II are identified in Table 3-1 together with the corresponding section of this document. The criteria and sections that apply to fuel rod T-M design are identified in Table 3-2.

The GNF3 fuel rod definition includes three variable application parameters, which may vary for different plants and for different energy utilization plans. The following table illustrates the application parameters and an example of a set that may be applied for a specific design.

Fuel Rod Variable Application Parameters:	Example Values
Fuel rod as fabricated internal fill-gas pressure	[[ ]] MPa
Active fuel column length	Table 2-1
Local fuel Linear Heat Generation Rate (LHGR) [[ ]]	Figure 3-6

Compliance of the fuel rod response with the requirements of Sections 3.2.1, 3.2.6, 3.2.9, and 3.2.10 in Table 3-1 is confirmed for specific sets of the three application parameters by performing exposure-dependent T-M analyses with appropriate consideration of anticipated operational overpower occurrences. The compliance analyses confirm that the criteria are satisfied for all exposures from Beginning of Life (BOL) to design discharge exposure. Compliance with the requirement of Section 3.2.8 is confirmed generically for the most limiting set of application parameters. GNF3 compliance with the fuel rod T-M acceptance criteria shall be reconfirmed for each set of application parameters utilized for GNF3 core designs.

The GNF3 fuel rod T-M analyses, including transient analyses, are performed using NRC-approved T-M performance analytical models. The model applied to the fuel rod analyses in this section was the PRIME (Reference 7) T-M performance model as documented in Section 2.2 of NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," GESTAR II. As discussed in Reference 7, the PRIME T-M analysis methodology includes [[ ]] for GNF3.

Table 3-1 GESTAR Fuel Thermal-Mechanical Design Criteria

Section	GESTAR Subsection	GESTAR Criteria
3.2.1 Stress, Strain, Fatigue	1.1.2.B.i	The fuel rod and fuel assembly component stresses, strains, and fatigue life usage shall not exceed the material ultimate stress or strain and the material fatigue capability.
3.2.2 Fretting	1.1.2.B.ii	Mechanical testing will be performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear when operating in an environment free of foreign material.
3.2.3 Metal Thinning	1.1.2.B.iii	The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these effects influence the material properties and structural strength of the components.
3.2.4 Fuel Rod Internal Hydrogen Content	1.1.2.B.iv	The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with American Society for Testing and Materials (ASTM) standards C776-83 and C934-85 to assure that loss of fuel rod mechanical integrity will not occur due to internal cladding hydriding.
3.2.5 Fuel Rod/Channel Bow	1.1.2.B.v	The fuel rod is evaluated to ensure that fuel rod or channel bowing does not result in loss of fuel rod mechanical integrity due to boiling transition.
3.2.6 Cladding Pressure Loading	1.1.2.B.vi	Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading.
3.2.7 Control Rod Insertion	1.1.2.B.vii	The fuel assembly (including channel box), control rod and control rod drive are evaluated to assure control rods can be inserted when required.
3.2.8 Cladding Creep Collapse	1.1.2.B.viii	Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel column axial gap.
3.2.9 Fuel Center Temperature	1.1.2.B.ix	Loss of fuel rod mechanical integrity will not occur due to fuel melting.
3.2.10 Cladding Plastic Strain During AOOs	1.1.2.B.x	Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction.

Table 3-2 Fuel Rod Thermal-Mechanical Design Criteria

Criterion	Section	Governing Equation
The cladding creepout rate ($\dot{\varepsilon}_{cladding\_creepout}$), due to fuel rod internal pressure, shall not exceed the fuel pellet irradiation swelling rate ($\dot{\varepsilon}_{fuel\_swelling}$).	3.2.6	$\dot{\varepsilon}_{cladding\_creepout} \leq \dot{\varepsilon}_{fuel\_swelling}$
The maximum fuel center temperature (T_{center}) shall remain below the fuel melting point (T_{melt}).	3.2.9	$T_{center} < T_{melt}$
Range 1 – [[ ]] Range 2 – [[ ]]	3.2.10	Range 1: [[ ]] Range 2: [[ ]]
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.	3.2.1	$\sum_i \frac{n_i}{n_f} \leq 1.0$
Cladding structural instability, as evidenced by rapid ovality changes, shall not occur.	3.2.8	No creep collapse
Cladding effective stresses (σ_e) shall not exceed the failure stress (σ_f) and cladding effective strains (ε_e) shall not exceed the failure stress strain (ε_f).	3.2.1	$\sigma_e < \sigma_f$, $\varepsilon_e < \varepsilon_f$
The as-fabricated fuel pellet evolved hydrogen (C_H is content of hydrogen) at greater than 1800 °C shall not exceed prescribed limits.	3.2.4	[[ ]]

3.2.1 Stress, Strain, Fatigue

GESTAR II Section 1.1.2.B.i: "The fuel rod and fuel assembly component stresses, strains, and fatigue life usage shall not exceed the material ultimate stress or strain and the material fatigue capability."

Fuel Rods

The fuel rod stress analysis was performed for the limiting application parameters as defined in Section 3.2. The analysis was performed using a Monte Carlo statistical method to calculate the effects of [[

]]

For each calculation, the stresses are combined into an effective stress using the Von Mises theory and compared with the appropriate design limit to produce a design ratio. [[

]] as shown in the reference power-exposure envelope of Figure 3-6. Table 3-3 summarizes the calculated cladding stress design ratios for the power versus exposure envelopes for the UO_2 rod listed in Appendix A.

[[

]]

Figure 3-6 GNF3 for BWR/4-6 Power-Exposure Envelope

Table 3-3 Results of Cladding Stress Analysis for GNF3 BWR/4-6 Fuel Rod

Fuel Rod Type	Period	Design Ratio at Rated Power	Design Ratio at Overpower
[[			
			]]

These analyses demonstrated that the GNF3 fuel rod stresses do not exceed the failure strength of the material.

Inputs to these fuel rod cladding statistical stress analyses are obtained from the fuel rod T-M performance model PRIME as documented in GESTAR II.

Fatigue evaluations of fuel rod designs are performed for the application parameters for each design using the analysis methodology as defined in Section 3.2. These evaluations demonstrate with large conservatism that the cladding fatigue usage does not exceed the cladding fatigue capability. Therefore, loss of fuel rod mechanical integrity due to cladding fatigue will not occur.

[[

]]

Channels

The GNF3 fuel channel (Figure 3-7) is open at the bottom and makes a sliding seal fit on the LTP surface. At the top of the channel, two opposite corners have welded clips. These clips support the weight of the channel on the UTP posts. One of the clips has a hole for attaching the channel to the bundle. [[

]]

[[

]]

The GNF3 channel has been evaluated by finite element analyses. These analyses demonstrate that the stresses and strains are well below the failure strength at operating conditions. The channel wall pressure differential required to cause material yielding is [[

]] For each new channel application, it is confirmed that the specific plant pressures do not exceed the channel capability. A fatigue analysis that addressed the cyclic pressure duty due to normal and transient operation was performed, and the predicted fatigue usage was less than the failure criterion provided in Table 3-2.

[[

]]

Figure 3-7 GNF3 Fuel Channel

Spacers

The GNF3 spacers have been tested with respect to stresses, strains and fatigue to determine their seismic and fatigue life capabilities. The spacers were tested for both seismic (high load, low cycle) and fatigue (low load, high cycle) conditions.

Seismic capability was tested for both Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) loading. [[

]] For the SSE testing, the load equates to the dynamic loading from a [[ ]] The load is increased by [[ ]] to provide design margin and it is also scaled to account for the temperature difference between the testing and the operating conditions. The OBE load was set to be [[ ]]

[[

]]

Thus, the GNF3 spacer is demonstrated to be capable of withstanding a [[ ]] earthquake without compromising safe shutdown and a [[ ]] earthquake without compromising fuel operability.

Fatigue adequacy is demonstrated by testing [[

]] GNF3 spacers were tested in two configurations [[

]] Each spacer was tested for a number of cycles and then examined for failure. This process was repeated until the fatigue capability in terms of cycles to failure was determined. The same procedure was used for spacers of the proven GNF2 design with a similar configuration and loading. The GNF3 spacers were found to have significantly improved fatigue life thus demonstrating compliance with the requirements of this Subsection.

Water Rod

The GNF3 assembly incorporates [[

]] shown in Figure 3-8.

[[

]]

Demonstration that water rod stresses and strains do not exceed failure strength and that the fatigue capability will not be exceeded is shown by stress analysis of the pressure loading. [[

]]

[[ ]].
The Zircaloy material properties at operating conditions appropriate for this analysis are:

Yield Strength = [[ ]] at 288ºC

Tensile Strength = [[ ]] at 288ºC

[[

]]

The maximum stress occurs in the [[ ]]

Therefore; [[ ]]

Because all stresses are well below yield strength and because there is no significant cyclic loading, the fatigue capability is not exceeded.

[[

]]

Figure 3-8 Water Rod

Tie Plates

Demonstration that the GNF3 UTP and LTP do not exceed failure strength was shown by stress analyses that addressed the maximum handling loads. The loads are the largest loads on these components except for seismic and fuel lift loadings that are addressed in Subsection 3.2.7. The UTP and LTP are not subjected to any significant cyclic loadings and fatigue capability is therefore not exceeded.

[[

]]

Appropriate material properties for Type-304 Stainless Steel for the UTP stress evaluations are:

- Yield Strength = [[ ]] at 38º C
- Tensile Strength = [[ ]] at 38º C

The limiting loading on the UTP occurs during fuel handling when the fuel assembly is lifted by the grapple that is attached to the UTP handle. The loads that are evaluated are [[ ]] For this analysis, the GNF3 fuel assembly weight, which includes the fuel bundle, channel, and channel fastener weights, is assumed to be [[ ]] in air. Therefore, the upward loading on the UTP is [[ ]] for this condition.

The UTP was evaluated using the ANSYS® finite element code. The model utilizes [[ ]]

An upward vertical load of [[ ]] was applied at the edge of the grapple interface with the UTP handle (20 mm from the center of the handle). The downward load from the channel of [[ ]] was applied at the channel post locations. Note that this is conservative relative to the channel weight of [[ ]]. The upward loading from the expansion springs is also modeled ([[ ]]). The remainder of the upward vertical load was [[ ]]

The maximum bending stress in the grid portion of the tie plate (corrected for minimum dimensions) based on this loading was determined to be [[ ]].

A finite element analysis, using three dimensional solid elements, was also used to evaluate the stresses in the handle. The maximum stress in the handle occurs at the handle, grid interface. Correcting the stresses for minimum dimensions results in a stress equal to [[ ]]

The limiting loading condition on the LTP is due to seating of the fuel assembly into the core or into the fuel storage racks. The GNF3 design basis load, as distributed over the grid surface, is 4.2 times the assembly weight minus the LTP weight (i.e., [[ ]]. The GNF2 LTP was evaluated at [[ ]]. Relative to GNF2, the only changes to the GNF3 tie plate grid are 1) replacement of two threaded holes with two flow holes of similar size and 2) [[ ]]. Therefore, the GNF3 LTP grid stiffness is comparable to GNF2. For GNF2, the maximum bending stress (corrected for minimum section dimensions) was determined to be [[ ]]. An accurate estimate for GNF3 is obtained from the GNF2 maximum bending stress value times the ratio of the loading, [[ ]]. This result demonstrates that the LTP stresses are well below the yield strength.

The above analyses demonstrate that the GNF3 UTP and LTP are not expected to experience excessive deformation or failure during bundle handling and service.

3.2.2 Fretting

GESTAR II Section 1.1.2.B.ii: “Mechanical testing will be performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear when operating in an environment free of foreign material.”

The GNF3 fuel assembly was tested to assure that the design features do not result in a significant increase in Flow-Induced Vibration (FIV) response and thereby do not increase the potential for fretting. The method used to demonstrate the adequacy of the fuel assembly from a FIV perspective was to compare the vibration response of the GNF3 design with the GNF2 design during FIV tests. The response comparison was based on accelerometer data from various locations in the fuel assemblies. The GNF2 fuel assembly's performance is considered acceptable based upon its reliable performance in reactor operation.

[[

]] The acceleration signals were recorded and then analyzed to perform direct comparisons of Root Mean Square (RMS) and maximum response between GNF2 and GNF3 over a range of flow conditions. Each configuration was tested over a range of flow rates, from [[ ]] to approximately [[ ]] of in-reactor rated mass flow.

The results of the FIV tests demonstrate that there are no significant differences in the peak acceleration response of the GNF3 fuel and water rods compared to the performance of the GNF2 fuel and water rods. The GNF3 FIV test results also demonstrate the acceptable performance of the [[ ]] The differences in fuel rod, LTP, channel-LTP interface and spacer designs show no significant effect on FIV performance when compared to the GNF2 design.

A [[ ]] endurance test [[ ]] further confirmed that the new GNF3 features [[ ]] perform adequately with respect to vibration response. Based on the FIV test program, the performance of the GNF3 fuel design meets the fretting design requirements.

3.2.3 Metal Thinning

GESTAR II Section 1.1.2.B.iii: "The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these effects influence the material properties and structural strength of the components."

Metal thinning of the Zircaloy components due to corrosion will result in higher stresses being calculated at EOL if the loading conditions do not change. The increase in stress is more than offset, in this case, by the increase in material strength due to irradiation. However, the fatigue strength of the Zircaloy components is not increased with irradiation. Where the load cycling is potentially significant, the effects of corrosion are explicitly addressed. Corrosion thinning effects were consequently addressed in the fuel rod stress and fatigue analyses and in the channel fatigue analysis described in Section 3.2.1. Sections 3.2.3.1 and 3.2.3.2 describe the methods applied for consideration of metal thinning.

3.2.3.1 Metal Thinning Effects On Zircaloy Cladding

Zircaloy cladding tubes undergo oxidation at slow rates during normal reactor operation. This oxidation causes thinning of the cladding tube wall and introduces a resistance to the fuel rod-to-coolant heat transfer. Corrosion products present in the reactor coolant system also tend to deposit on the fuel rod cladding outer heat transfer surface. This corrosion product deposition also introduces a resistance to the fuel rod-to-coolant heat transfer. In the extensive GNF operational history database, fuel rod failures have not occurred due to cladding corrosion without the presence of an augmenting factor such as an aggressive crud-induced localized corrosion environment. Therefore, no specific value of cladding oxide thickness can be identified to correspond to fuel rod failure; however, cladding oxidation does affect the overall strength of the cladding through loss of structural material and reduced material strength due to higher temperature. Therefore, all fuel rod evaluations explicitly include the amount of cladding metal thinning and the cladding temperature increase due to cladding oxidation and the buildup of corrosion products based on the results summarized in Table 3-4.

Table 3-4 Cladding Oxidation and Corrosion Product Buildup

	Radial Thickness (mm)	
	Mean	**Standard Deviation**
Beginning of Life		
Cladding Oxidation	[[	
Corrosion Product Buildup		
End of Life (8 years)		
Cladding Oxidation		
Corrosion Product Buildup		]]

These results are based on numerous field measurements at many plants representing normal GNF experience, excluding cases involving specific water chemistry issues outside of normal operating experience. The data above was generated with a best-fit estimate based on the data set mentioned above.

[[

]]

[[

]]

Figure 3-9 Projected GNF3 Cladding Hydrogen Content vs. Exposure

[[

]]

3.2.3.2 Metal Thinning Effects On Zircaloy Channels

The effects of metal thinning have been considered in a channel fatigue and stress rupture analysis. This analysis shows that the GNF3 channel is structurally adequate, with respect to fatigue and stress rupture, for a bounding design basis pressure differential and a maximum lifetime of [[ ]].

Metal thinning, as a result of oxidation, for the fatigue and stress rupture analysis is modeled by consideration of the thermal and irradiation components in a BWR environment. Metal thinning is modeled according to the following relationship.

[[ ]]

where Z_{total} is the oxidation on each side of the channel wall.

Considering metal thinning, a channel pressure differential of [[ ]] was used to determine the limit of pressure differential that exceeds a total damage of 1.0. The damage is calculated as the sum of the fatigue and rupture stress life consumed under a series of events and conditions. By definition, a damage value of 1.0 indicates failure. The limiting condition for GNF3 occurs in the lower uniform portion of the thinner channel option where the maximum allowable pressure differential is [[ ]].

This analysis demonstrates the adequacy of the GNF3 design and the methods for resisting the effects of metal thinning due to corrosion. The methods are applicable for Zircaloy-2, Zircaloy-4 and NSF variants of the GNF3 product.

3.2.4 Fuel Rod Internal Hydrogen Content

GESTAR II Section 1.1.2.B.iv: "The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with ASTM standards C776-83 and C934-85 to assure that loss of fuel rod mechanical integrity will not occur due to internal cladding hydriding."

The pellet specifications include a requirement that limits the maximum amount of hydrogen that is allowed to be present in the manufactured fuel pellets. This limit is consistent with or less than that specified by ASTM standards C776-83 and C934-85. Manufacturing processes for the fuel rod and its components include controls to ensure that the hydrogen limit is met and are designed to avoid spurious sources of hydrogen in the fuel rod.

3.2.5 Fuel Rod/Channel Bow

GESTAR II Section 1.1.2.B.v: "The fuel rod is evaluated to ensure that fuel rod or channel bowing does not result in loss of fuel rod mechanical integrity due to boiling transition."

Analysis Procedures for Incorporating Channel Bow Effects in Critical Power Evaluations

Channel bow effects are incorporated in critical power evaluations by modifying the bundle R-factor to include changes in local peaking caused by channel bowing. The model is described in the GE report MFN 086-89 submitted by letter to the NRC November 15, 1989 and in additional information contained in MFN 041-90, May 3, 1990, and MFN 109-90, September 26, 1990. The methodology has been approved by the NRC letter, "Acceptance for Referencing of Topical Report Titled "GE-Nuclear Energy Report MFN 086-89," to J. S. Charnley (GE) from A.C. Thadani (NRC), MFN 014-91, January 11, 1991.

Channel Bow Compliance

Loss of mechanical integrity due to boiling transition is prevented because all critical power evaluations in the plant process computer and other licensing analyses include an allowance for channel bow effects according to approved methods described above.

Rod Bow Compliance

Reference 34 describes a large program to characterize the extent of rod bowing in BWR fuel along with full scale thermal hydraulic experiments on 8x8 assemblies to investigate the potential effect on boiling transition due to rod bow. This program included poolside measurements of over 1,000 assemblies and concluded that significant rod bowing did not exist in BWR fuel. Furthermore, the thermal hydraulic testing did not observe any significant effect on critical power.

This original work was supplemented with additional full scale testing of 9x9 assemblies. The results of this testing, described in Reference 35, were verbally communicated to NRC. In summary, a very improbable configuration was tested in which the critical rods in a reference test were bowed to contact just upstream of the onset of boiling transition. This testing again concluded that rod bowing does not degrade the margins to boiling transition even in this highly improbable circumstance. The results of these two programs are considered applicable to 10x10 fuel. As such, standard critical power limits are sufficient to prevent loss of mechanical integrity due to boiling transition even in the presence of rod bow. As stipulated in Reference 34, NRC will be notified if rod-to-rod gap closures greater than 50% are observed.

Compliance with requirement has been met.

3.2.6 Cladding Pressure Loading

GESTAR II Section 1.1.2.B.vi: "Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading."

Evaluations of fuel rod designs are performed for the application parameters using the analysis methodology as referenced in Section 3.2 of this document. These evaluations demonstrate that the cladding creepout rate due to fuel rod internal pressure will not exceed the irradiation-swelling rate of the fuel pellet. Therefore, loss of fuel rod mechanical integrity due to excessive pressure loading will not occur.

In this section, cladding lift-off is defined as the separation of the cladding from the pellet. Cladding lift-off evaluations are used to ensure that the criterion in Item 1 of Table 3-2 is met. For the cladding lift-off evaluation, fuel rod internal pressure for the maximum duty fuel rods is determined using the PRIME T-M performance model in conjunction with the standard error propagation statistical method. [[

]] The standard error propagation analysis results in a mean and standard deviation for the fuel rod internal pressure at uniformly spaced exposure points throughout the design lifetime. [[

]] This design ratio has been calculated at several exposure points for the maximum duty fuel rod for each fuel rod type present in the fuel bundle.

Table 3-5 summarizes the PRIME results for the cladding lift-off evaluation for some of the key rod types for BWR/4-6. Because all design ratios are less than 1.0, it is assured, [[

]]

Table 3-5 GNF3 for BWR/4-6 Fuel Rod Cladding Lift-Off Results

		Rod Internal Pressure (MPa)		Critical Pressure (MPa)		
Fuel Rod Type	**Exposure where Design Ratio is Max., GWd/MTU**	**Mean**	**Standard Deviation**	**Mean**	**Standard Deviation**	**Max. 95% Confidence Design Ratio**
[[						
						]]

3.2.7 Control Rod Insertion

GESTAR II Section 1.1.2.B.vii: "The fuel assembly (including channel box), control rod and control rod drive are evaluated to assure control rods can be inserted when required."

The fuel assembly is evaluated to assure that component deformations are not severe enough to prevent control rod insertion and that vertical uplift forces will not unseat the LTP such that the resultant loss of lateral fuel bundle positioning would prevent control rod insertion. This evaluation is performed considering the combined effects of safe shutdown earthquake and LOCA loadings on fuel assembly deformation and lift-off.

Assurance that component deformations are not excessive is provided by primary load stress analyses and tests of the components. These evaluations are based on un-irradiated material properties at operating temperature. The loads used in the evaluation of the fuel assembly components are derived from enveloping values of combined horizontal and vertical acceleration of the fuel assembly. All component stress evaluations have minimum margins of at least [[ ]] because the limit is specified to be [[ ]] times ultimate. The channel buckling has the same margin as was demonstrated previously in NEDE-21175-3-P-A (Reference 36). The existing plant seismic analysis results for the fuel assembly are checked to assure that fuel loadings do not exceed the enveloping values.

Assurance that vertical uplift forces will not unseat the fuel assembly such that loss of lateral fuel bundle positioning could occur was provided by a nonlinear fuel lift analysis as described in detail in NEDE-21175-3-P-A. The GNF3 fuel design, while visibly different from the previous fuel designs for which the lift analysis was initially performed, is dynamically similar when modeled. Because of this dynamic similarity, there is no significant difference in the fuel lift behavior between GNF3 and previous fuel designs.

Separate from consideration of the combined effects of safe shutdown earthquake and LOCA loads on control rod insertability, considerations also arise for control rod insertability during normal operation due to any channel-control blade interference that may result from irradiation-induced channel bulge and channel bow deformations. The primary control for channel-control blade interference is provided by the plant Technical Specifications surveillance where actions are specified both (1) to ensure control rod drive scram performance is consistent with requirements, and (2) to appropriately disposition instances where control rod operability, including channel control blade interference effects, is less than adequate. These plant technical specification requirements will continue to be applied with GNF3. Additionally, the guidance as documented in MFN 08-420 R1, "Update to GEH Surveillance Program for Channel-Control Blade Interference Monitoring," December 16, 2013, and MFN 10-245 R6, "Update to Part 21 Notification: Failure to Include Seismic Input in Channel-Control Blade Interference Customer Guidance," December 16, 2013, remains applicable and will be similarly applied to operating plants with GNF3 fuel to mitigate any elevated levels of channel-control blade interference.

3.2.8 Cladding Collapse

GESTAR II Section 1.1.2.B.viii: "Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel rod column axial gap."

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 Pressurized Water Reactor (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. Because 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 creep collapse analysis procedure applied to the GNF3 fuel design is documented in NEDC 33139P-A, "Cladding Creep Collapse Licensing Topical Report," July 2005. The analysis consists of a detailed finite element mechanics analysis of the cladding.

An evaluation has been performed using the approved process that confirms creep collapse of freestanding cladding (cladding unsupported by fuel pellets) will not occur for the GNF3 design.

3.2.9 Fuel Melting

GESTAR II Section 1.1.2.B.ix: "Loss of fuel rod mechanical integrity will not occur due to fuel melting."

Evaluations of fuel rod designs are performed for the application parameters using the analysis methodology referenced in Section 3.2 of this document. These evaluations demonstrate that the fuel center temperature will not exceed the fuel melting temperature. Therefore, loss of fuel rod mechanical integrity due to fuel melting will not occur.

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 rod failure due to 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 AOOs 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.

The fuel center temperature evaluation is performed using the PRIME T-M performance model in conjunction with the standard error propagation statistical method [[ ]]. The standard error propagation analysis results in a mean and standard deviation for the fuel center temperature during AOOs at uniformly spaced exposure points throughout the design lifetime. These results are used to specify a Thermal Overpower (TOP) limit that assures with 95% confidence that the fuel center temperature will not exceed the fuel melting temperature for the maximum duty fuel rod during an AOO at any point in the licensed design lifetime of the fuel. [[

]]

3.2.10 Pellet-Cladding Mechanical Interaction

GESTAR II Section 1.1.2.B.x: "Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction."

Evaluations of fuel rod designs are performed for the application parameters using the analysis methodology as defined in Section 3.2 of this document. These evaluations demonstrate that the cladding circumferential strain due to pellet-cladding mechanical interaction during an AOO will not exceed the cladding circumferential strain limit. Therefore, loss of fuel rod mechanical integrity due to pellet-cladding mechanical interaction will not occur.

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 has occurred, a rapid power increase, such as would occur during an AOO, will result in cladding outward diametral deformation due to the fuel pellet thermal expansion. The extent of deformation depends on the extent of irradiation exposure, the magnitude of the power increase, and the final peak power level. 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. [[

]]

The cladding strain evaluation is performed using the PRIME T-M performance model in conjunction with worst tolerance assumptions. The fabrication parameters important to the analysis are all biased to the fabrication tolerance limit in the direction that produces the most severe result. Other input parameters conservatively biased for this analysis include (a) cladding corrosion, and (b) corrosion product (crud) buildup on the cladding outer surface. These analyses result in cladding strain during AOOs at uniformly spaced exposure points throughout the design lifetime. These results are used to specify a Mechanical Overpower (MOP) limit that assures that the fuel circumferential strain will not exceed the specified strain limit for the maximum duty fuel rod during an AOO at any point in the design lifetime of the fuel. [[

]]

Table 3-6 GNF3 for BWR/4-6 Circumferential Cladding Strain Results

[[						
						]]

3.3 NUCLEAR

3.3.1 Doppler Reactivity Coefficient

GESTAR II Section 1.1.3.A: "A negative Doppler reactivity coefficient shall be maintained for any operating conditions."

Analysis Description

The Doppler Reactivity Coefficient (DRC) is of high importance in reactor safety. The DRC is a measure of the reactivity change associated with a change in the temperature of the fuel material. An increase in fuel temperature causes an increase in the absorption of resonance energy neutrons and a decrease in reactivity. The DRC of a core is a function of the average of the bundle DRCs. A negative DRC provides inherent negative reactivity feedback to any rise in fuel temperature, on a gross or local basis, and thus assures the tendency of self-control for the BWR.

The DRC characteristics for GNF3 were determined by using the NRC-approved GEH/GNF lattice physics ECP, TGBLA06 (Reference 16). [[

]] The results of the calculations demonstrate that the DRC becomes more negative as the initial fuel temperature decreases.

The DRC in units of pcm/K is defined as follows:

$$DRC = \frac{10^5\left(k_{T_1} - k_{T_0}\right)}{k_{T_0}\left(T_1 - T_0\right)}$$

where

T_0 = Reference temperature (K)

T_1 = Elevated temperature (K)

k_{T_0} = Eigenvalue at reference temperature

k_{T_1} = Eigenvalue at elevated temperature k_{T_1}

Typical values for varying enrichments are shown in Figure 3-10. The zero void fraction value is illustrated in the figure because it corresponds to the least negative DRC. The least negative DRC calculated is approximately [[ ]]

[[

]]

Figure 3-10 Typical Behavior for Doppler Reactivity Coefficient (Hot, Uncontrolled, Zero Void Fraction)

Conclusion

The GNF3 DRC is negative for any operating condition, thus meeting the requirement of GESTAR II Section 1.1.3.A.

3.3.2 Moderator Void Coefficient

GESTAR II Section 1.1.3.B: "A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels shall be maintained for any operating conditions."

Analysis Description

The moderator void coefficient of reactivity is associated with the change in moderating capability of the in-channel water.

The magnitude of the void coefficient of reactivity is strongly dependent on the amount of moderator present in the bypass region. Conditions that allow for more moderation produce a more positive (or less negative) void coefficient of reactivity. [[

]]

The analysis performed to calculate the moderator void coefficient used the lattice physics code TGBLA06 and the 3D core simulator PANAC11 (Reference 16). [[

]] Thus, this analysis is applicable to BWR types 2 through 6. The ABWR, ESBWR, and non-GE plants will require a separate evaluation.

The generic moderator coefficient analyses included the following considerations:

1. [[

]]

The core eigenvalue is calculated at various temperatures from [[

]] The void coefficient of reactivity, which is the change in reactivity divided by the change in void fraction, is calculated for each of these moderator temperatures. This was performed at the following three exposure points of the cycle:

BOC: Zero Exposure

MOC: [[ ]]

EOC: [[

]]

The following characteristics were selected in order to obtain a bounding condition:

1. [[

]]

A GNF3 equilibrium fuel cycle with a [[

]]

All of the nuclear libraries included cold libraries with moderator temperatures at [[

]]

The void coefficient is calculated as follows:

$$\alpha_v \equiv \frac{d\rho}{dv} \approx \frac{1}{k}\frac{dk}{dv} \approx \frac{1}{k_{v_0}}\left(\frac{k_{v_1} - k_{v_0}}{v_1 - v_0}\right)$$

where:

ρ = Reactivity

k_{v_0} k_{v_0} = Eigenvalue at 0% in-channel void fraction

k_{v_1} = Eigenvalue at 5% in-channel void fraction

v_0 = 0% in-channel void fraction

v_1 = [[ ]] in-channel void fraction

In order to obtain a critical control blade configuration, [[

]]

At each exposure and moderator temperature, a critical control blade configuration was established [[ ]] Figure 3-11 summarizes the results.

[[

]]

Figure 3-11 GNF3 Void Coefficient of Reactivity

Conclusion

The GNF3 void coefficient of reactivity is negative for any operating condition, thus meeting the requirement of GESTAR II Section 1.1.3.B.

3.3.3 Moderator Temperature Coefficient

GESTAR II Section 1.1.3.C: "A negative moderator temperature coefficient shall be maintained for temperatures equal to or greater than hot standby."

Analysis Description

The moderator temperature coefficient is associated with the change in moderating capability of the water. A negative moderator temperature coefficient during power operation provides inherent protection against power excursions. Hot standby is the condition under which the BWR core coolant has reached operating pressure and the temperature at which boiling has begun. Once boiling begins, the moderator temperature remains essentially constant in the boiling regions.

The magnitude of the moderator temperature coefficient is strongly dependent on the amount of moderator present in the bypass region. Conditions that allow for more moderation produce a more positive (or less negative) moderator temperature coefficient. [[

]]

The analysis performed to calculate the moderator temperature coefficient used the lattice physics code TGBLA06 and the 3D core simulator PANAC11 (Reference 16). The analysis used to demonstrate that it is negative for temperatures equal to or greater than hot standby was performed [[

]] Thus, this analysis is applicable to BWR types 2 through 6. The ABWR, ESBWR, and non-GE plants will require a separate evaluation.

A GNF3 [[

]]

The core eigenvalue is calculated at various temperatures from [[

]] The moderator temperature coefficient was calculated for each of these temperatures. This was performed at the following exposure points of the cycle:

- BOC: Zero Exposure
- MOC: [[ ]]
- EOC: [[
 ]]

The moderator temperature coefficient analysis includes the following considerations:

1. [[

]]

The moderator temperature coefficient is the change in reactivity divided by the change in moderator temperature and is defined as:

$$\alpha_T \equiv \frac{d\rho}{dT} \approx \frac{1}{k}\frac{dk}{dT}$$

where:

ρ: Reactivity

T: Moderator Temperature

k: Effective Multiplication Factor

The eigenvalue can be described as a quadratic function of the moderator temperature as shown below.

$$k = C_0 + C_1 T + C_2 T^2$$

Differentiating this equation and dividing by the eigenvalue yields the following expression:

$$\alpha_T = \frac{1}{k}\left(C_1 + 2C_2 T\right)$$

The moderator temperature coefficient is calculated by fitting the eigenvalue versus temperature to a quadratic curve and solving the differential equation of the quadratic expression. For each temperature, the [[ ]]

[[

]] Again the results were tabulated and the moderator temperature coefficient calculated. Figure 3-12 summarizes the results.

[[

]]

Figure 3-12 GNF3 Moderator Temperature Coefficient

Conclusion

The GNF3 moderator temperature coefficient of reactivity is negative for moderator temperatures equal to or greater than hot standby thus meeting the requirement of GESTAR II Section 1.1.3.C.

3.3.4 Prompt Reactivity Feedback

GESTAR II Section 1.1.3.D: "For a super prompt critical reactivity insertion accident (e.g., control rod drop accident) originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel shall be negative."

Analysis Description

The mechanical and nuclear design of the fuel shall be such that the prompt reactivity feedback (requiring no conductive or convective heat transfer and no operator action) provides an automatic shutdown mechanism in the event of a super prompt incident such as a Control Rod Drop Accident (CRDA). This characteristic will assure rapid termination of super prompt critical accidents with additional long-term void reactivity shutdown capability provided by the moderator void feedback for those cases where heat transfer from the fuel to the moderator results in boiling in the active flow channel.

A model is developed relating moderator temperature and fuel temperature for a super prompt critical excursion. Enthalpy increases in moderator and fuel are given by the following expressions,

[[

]]

The GNF3 fuel mass [[

]]

A super prompt reactivity excursion occurs in a time frame much too short to allow heat conduction from the fuel through the cladding to the moderator. The only mechanism for moderator heating is through fission neutron slowing and fission gamma absorption. [[

]]

Hence, the heating fractions are calculated as follows:

		Energy Deposition, MeV		
Emission Type	**Recoverable Energy, MeV**	**Fuel**	**Moderator**	**Zircaloy**
Fission Fragments	[[			
Prompt γ - rays				
Fission Neutrons				
Total				
Fraction				]]

Calculations of prompt reactivity insertion are made at the [[

]]

The lattice physics code TGBLA06 (Reference 16) is used to evaluate [[

]]

Figure 3-13 illustrates the change in eigenvalue [[ ]] due to prompt heating of the moderator and fuel for different lattice enrichments. [[
]]

[[

]]

Figure 3-13 Prompt Reactivity Feedback for a Typical GNF3 Lattice

The results demonstrate that the eigenvalues at [[
]]

Conclusion

It is concluded that the net prompt reactivity feedback due to prompt heating of the moderator and fuel is negative thus meeting the requirement of GESTAR II Section 1.1.3.D.

3.3.5 Power Coefficient

GESTAR II Section 1.1.3.E: "A negative power coefficient, as determined by calculating the reactivity change due to an incremental power change from a steady-state base power level, shall be maintained for all operating power levels above hot standby."

Analysis Description

The power coefficient is defined as the rate of change in reactivity as the core power changes while all other core boundary conditions (control rod distribution, core inlet coolant flow, core inlet coolant enthalpy, reactor system pressure) remain constant.

A negative power coefficient provides an inherent negative feedback mechanism to provide more reliable control of the plant during power maneuvers. The power coefficient is effectively the combination of Doppler, moderator void and moderator temperature coefficients of reactivity.

Conclusion

For the GNF3 fuel design, each of these three components has been shown to be negative for all operating power levels above hot standby. Therefore, a negative power coefficient is assured for all operating power levels above hot standby.

3.3.6 Cold Shutdown Margin

GESTAR II Section 1.1.3.F: "The plant shall be calculated to meet the cold shutdown margin requirement for each plant cycle specific analysis."

Analysis Description

The core must be capable of being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive control rod in its full out position and all other control rods fully inserted. The typical values of cold shutdown margin required by plant Technical Specifications are 0.38% $\Delta k/k$ or 0.25% $\Delta k/k$, depending on the specific plant. Shutdown margin is dependent upon the core loading. It is calculated for each plant cycle prior to the operation of that cycle.

Conclusion

The calculations demonstrating compliance with this requirement will be performed for every reload of GNF3 fuel. The results of the cycle-specific calculations will be documented in the reload license report for that cycle.

3.3.7 Fuel Storage

GESTAR II Section 1.1.3.G: "The effective multiplication factor for new fuel designs stored under normal and abnormal conditions shall be shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k-infinity calculated in a normal reactor core configuration meets the limits provided in Section 3 of GESTAR II (Reference 1) for GE-designed regular or high density storage racks."

Compliance of GNF3 fuel with the k_∞ limits specified in GESTAR II Section 3.5 (Reference 1) will be confirmed and documented for each GNF3 lattice as part of the design process.

3.4 NEW FUEL DESIGN LICENSING EVALUATION

Section 2.4 from US NRC SE: "Licensing evaluations of new fuel designs will include generic analyses of a large BWR/4 or BWR/5 plant at limiting points of the cycle for an equilibrium loading of the new fuel design to assure that (1) nuclear design criteria are satisfied, and (2) safety limit MCPR values are correct. In addition, Chapter 15 safety analyses are performed for each reload application on a cycle-specific basis for (3) limiting anticipated operational occurrences and (4) bounding accidents. The cycle-specific plant (5) operating limit MCPR is determined and the effect of the new fuel design on previously evaluated accidents must be reconfirmed or reanalyzed."

Compliance with each of these criteria are performed in accordance with the methodologies as described in NEDE-24011-P-A-24 and are documented in the sections of this report as well as other licensing documentation supporting NFI and cycle operation.

(1) **Nuclear Design Criteria:** Compliance with this criterion is documented in Section 3.3. In addition, cycle-specific nuclear design criteria are confirmed for each operating cycle.

(2) **Safety Limit Minimum Critical Power Ratio (MCPR):** Safety Limit MCPR is now calculated for each unique core loading. (Reference 21) This criterion is no longer meaningful for the generic new fuel design evaluations, but is satisfied by performing the cycle-specific SLMCPR calculation.

(3) **Anticipated Operational Occurrences:** Compliance with this criterion is documented in Section 3.7. Per GESTAR II, limiting AOOs are analyzed on a cycle-specific basis.

(4) **Accidents:** Compliance with this criterion is documented in Section 3.3.4 (super prompt critical feedback), Section 3.11 (LOCA), Section 3.12 (rod drop accident), and Section 3.14 (ATWS). Plant-specific accident evaluations are performed when the GNF3 fuel product is first installed in reload quantities (Section 4.2).

(5) **Operating Limit MCPR (OLMCPR):** Compliance with this criterion is documented in Section 3.7. The plant OLMCPR is established by considering the limiting AOOs for each operating cycle.

3.5 THERMAL-HYDRAULIC

GESTAR II Section 1.1.4: "Flow pressure drop characteristics shall be included in plant cycle specific analyses for the calculation of the Operating Limit Minimum Critical Power Ratio."

Because of the channeled configuration of BWR fuel assemblies, there is no bundle-to-bundle cross flow inside the core and the only issue of hydraulic compatibility of the various bundle types in a core is the bundle inlet flow rate variation and its effect on margin to the OLMCPR. The coupled thermal-hydraulic-nuclear analyses performed each cycle for each plant to determine fuel bundle flow and power distribution use the various bundle pressure loss coefficients to determine the flow distribution required to maintain total core pressure drop boundary conditions to be applied to all fuel bundles. The margin to the thermal limits of each fuel bundle is determined using this consistent set of bundle flow and power.

The GNF3 fuel assembly design incorporates the use of nickel-based, Ni-Cr-Ti alloy grid type spacers with special flow wings designed for improved critical power performance. The pressure drop characteristics of the GNF3 spacers are based on the pressure drop data from full-scale testing of the GNF3 fuel assembly. Production spacers were used in the full-scale test assembly with no modifications. The measured pressure drops include static head, wall friction, acceleration pressure drop, and form losses. The loss coefficients were evaluated in a manner consistent with the steady state thermal hydraulic analysis methodology documented in Section 4.2 of GESTAR II (Reference 1). The test assembly and the measurement scheme for obtaining differential pressures are shown in Figure 3-14. Test data were obtained at [[

]]

Table 3-7 provides measured pressure drops across the bundle height from [[ ]] to [[ ]] cm ([[ ]] to [[ ]] inches) as well as comparisons to the predictions. Figure 3-15 summarizes the results graphically. The comparison of the predicted vs. measured pressure drop for [[ ]] tests over a range of thermal-hydraulic conditions resulted in a mean error for the [[ ]]
Therefore, it is concluded that the models and methods used for the determination of pressure drop in the GNF3 fuel assembly accurately predict the test data over a wide range of power and flow conditions.

Conclusion: The unique GNF3 fuel assembly hydraulic characteristics have been developed and confirmed by the test comparisons discussed above. These unique GNF3 hydraulic characteristics are used in all analysis models and methods where the fuel assembly hydraulics are needed. For cores of mixed assembly types, the hydraulics are uniquely represented for each assembly type. Therefore, the flow-pressure drop characteristics for each fuel assembly type (including GNF3) present in a plant are included in all plant cycle specific analyses for the calculation of the OLMCPR.

Table 3-7 Spacer Test Results

Run	Pressure (psia)	Mass Flux (Mlb/hr-ft^2)	Bundle Power (MW)	Inlet Temp (°F)	Measured ΔP (psid)	Predicted ΔP (psid)	Predicted-Measured ΔP (psid)
[[							

Table 3-7 Spacer Test Results (continued)

Run	Pressure (psia)	Mass Flux (Mlb/ hr-ft^2)	Bundle Power (MW)	Inlet Temp (°F)	Measured ΔP (psid)	Predicted ΔP (psid)	Predicted-Measured ΔP (psid)
							]]
				Average			[[
				Standard Deviation			]]

[[

]]

Figure 3-14 Spacer Test Configuration

[[

]]

Figure 3-15 Spacer Test Results and Predictions

3.6 SAFETY LIMIT MCPR

3.6.1 Confirmation of Applicability

GESTAR II Section 1.1.5.A: "A cycle-specific Safety Limit MCPR will be calculated on a cycle-specific basis following the steps in 1.1.5.B (of GESTAR II)."

The SLMCPR will be established on a cycle-specific basis following the calculational process steps in Section 1.1.5.B of GESTAR II. It will be calculated prior to the operation of that cycle to confirm that the SLMCPR value to be used for that cycle, which is incorporated into the SRLR, is applicable.

Reference 21 approves GNF methodologies associated with the SLMCPR and power distribution uncertainty. The NRC Safety Evaluation Report (SER) for Reference 21 provides conditions that require confirmation of the methodologies when new fuel designs are introduced. The following summary provides the confirmation of the GNF3 design to the requirements of the SER. Each requirement is quoted from the SER and the GNF response is then provided.

Requirement 1

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table 3.1 of NEDC-32601P, since changes in fuel design can have a significant effect on calculation accuracy. (Reference 21, page 3, Conclusions section)

GNF Response

The fidelity of the TGBLA06 lattice physics calculations for fuel rod powers depend on the lattice designs. Table 3.1 of NEDC-32601P includes GE12 10x10 lattices. GNF3 is also a 10x10 lattice, with the same fuel rod pitch and outside rod diameter as GE12.

No changes to the methodology (Reference 16) beyond that developed for GNF2 (described in Section 3.1.1.1 of Reference 2) were required to properly model GNF3 fuel. Calculations demonstrated the applicability of TGBLA06 using direct comparisons to Monte Carlo. The analyzed GNF3 dataset includes a total of 75 lattices and it is representative of any GNF3 design.

The results indicate that GNF3 designs are within the accepted levels of uncertainty in lattice physics modeling. The introduction of GNF3 is thus not significant to the TGBLA06 lattice physics methodology.

Requirement 2

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice. (Reference 21, page 3, Conclusions section)

GNF Response

The NRC approved overall R-Factor uncertainty for use in evaluating the SLMCPR is 1.6%. The derivation of this value for 10x10 lattices is presented in Appendix C of NEDC-32601P-A (Reference 19). It incorporates coverage of a core-average cell-average bow uncertainty up to 30 mils. In a letter addressed to the NRC (Reference 38), the R-Factor uncertainty was later increased to 2.0% to account for the potential effect of control blade shadow corrosion, which accounts for a bow uncertainty of up to 62 mils.

The GNF3 R-Factor uncertainty has been evaluated by considering 224 lattices with bow uncertainties ranging from 30 mils to 70 mils. Results showed that the overall GNF3 R-Factor uncertainty meets the licensing acceptance criteria of 2.0% corresponding to 62 mils channel bow uncertainty. It should be noted that the GNF3 product line is designed to be equipped with NSF channels. Per Section 3.1 of NEDE-33798P-A (Reference 10), NSF channels do not bow significantly as a function of exposure. Thus, the core-average cell-average bow for a GNF3/NSF core is expected to remain well below the 30 mils bow uncertainty threshold that was the basis for the original NRC approved 1.6% R-Factor uncertainty.

Requirement 3

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies. (Reference 21, page 3, Conclusions section)

GNF Response

The use of MCPR Importance Parameter (MIP) as a constraint on the development of the limiting rod pattern is required by the GNF technical design procedure for each cycle-specific SLMCPR analysis. GNF continues to monitor MIP and periodically assess it as part of the procedural review process. There is no indication for any current 10x10 reload designs (GE14 and GNF2) that suggest that the MIP criterion is not applicable. Scoping analyses performed for GNF3 equilibrium core designs have given no indications that suggests that the MIP values from these calculations are statistically distinct from historical data. Thus, there is no indication that the MIP criterion should be changed.

Requirement 4

(4) The 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons in Tables 3.1 and 3.2 of NEDC-32694P. (Reference 21, page 3, Conclusions section)

GNF Response

NEDC-32694P-A did not include 10x10 fuel in the evaluation of bundle power uncertainty. The approved values were documented as applicable for GE12 / GE14

10x10 in Reference 39. The approved values were demonstrated as applicable at expanded operating domains for GE14 in NEDC-33173P-A (Reference 40) and for GNF2 in NEDC-33173P-A Supplement 3 (Reference 41).

The continued applicability of the approved bundle power uncertainties to GNF3 was assessed by using Traversing In-Core Probe (TIP) comparisons and by performing TGBLA to MCNP infinite lattice reactivity calculations.

An evaluation of 3D-MONICORE generated TIP data measurements from two plants with one cycle of operation with GNF3 LUAs was used to verify that GNF3 bundle power modeling uncertainties are within those approved by the NRC. TIP comparisons of calculated versus predicted powers for instrument strings adjacent to the LUAs demonstrate no unusual or adverse trends at any point in the cycles which would be indicative of a method issue for the new fuel type.

TGBLA06 infinite lattice reactivity comparisons confirm that the performance for GNF3 fuel (versus MCNP) is essentially equivalent to the performance seen for GNF2 fuel (and for GE14 fuel).

The results of the TIP comparisons and TGBLA/MCNP comparisons leads to the conclusion that the bundle power calculational uncertainty of NEDC-32694P-A applies to GNF3 fuel.

Based on the above responses to the conditions of the SER, it is concluded that the GNF3 fuel design satisfies the conditions of the SER.

3.7 Operating Limit MCPR Evaluation

Section 3.7 summarizes the analyses performed for GNF3 fuel to demonstrate the applicability of cycle/plant-specific and generic MCPR and LHGR analyses described in Section 4 (of GESTAR II).

3.7.1 Cycle-Specific Analysis

GESTAR II Section 1.1.6.A: "Plant operating limit MCPR is established by considering the limiting anticipated operational occurrences for each operating cycle."

AOOs are classified as transient events of moderate frequency and must be analyzed with NRC approved methods. AOO events are analyzed to establish the reactor system response, including the calculation of OLMCPR.

The OLMCPR is established by adding (with appropriate statistical adjustment factors) the change in the MCPR (ΔCPR) for the limiting analyzed AOO to the SLMCPR, or calculated (with appropriate statistical adjustment) such that < 0.1% of the fuel rods will be subject to boiling transition during the transient. The calculational process for determining the SLMCPR is documented in Section 3.6.

The limiting AOO events analyzed are documented in Section S.2.2.1 of the US Supplement to GESTAR II (Reference 1).

3.7.1.1 Cycle Specific OLMCPR Analytical Models and Analysis Procedures

The primary NRC-approved methods used in the calculation process of the CPR response during a pressurization AOO include: (1) lattice physics models (TGBLA, Reference 16); (2) 3D core simulator (PANACEA, Reference 16); (3) 1D transient model (ODYN, References 22, 23, and 24) in conjunction with (4) transient hot channel model (TASC, Reference 29); or with (5) an advanced realistic combination 1D and 3D method (TRACG, References 25, 26, and 28) and (6) GEXL critical power correlation (described in Section 3.8).

The nuclear libraries for GNF3 fuel are generated by TGBLA and then are used as input to PANACEA. PANACEA, based on the cycle-specific reference core loading pattern, calculates the core state and the nuclear parameters for input to the plant transient model, ODYN or TRACG.

Loss of Feedwater Heating is analyzed using the steady-state nuclear methods (PANACEA), but may also be analyzed using the system transient models, ODYN and TASC or TRACG. The inadvertent High Pressure Coolant Injection (HPCI) startup may be bounded by the Loss of Feedwater Heating event. When necessary, it is analyzed using ODYN and TASC or TRACG.

The design process assures that an inadvertent rotation of a fuel bundle will not result in violation of the SLMCPR by calculating nominal and rotated bundle average R-factors as a function of exposure for each new bundle. From these results, delta R-factors and delta powers are constructed, and the maximum delta R-factors and the corresponding delta power are input to the analysis that determines the OLMCPR.

The Rod Withdrawal Error and Mislocated Fuel Loading Error (MFLE) events are also evaluated on a cycle-specific basis (Section 3.7.2).

A description of the OLMCPR calculational process is contained in Section S.2.2.1 of the US Supplement to GESTAR II.

Cycle Specific Operating Limit Compliance: The OLMCPR is dependent upon the cycle-specific core loading pattern. The OLMCPR is calculated prior to operation of that cycle and the results are incorporated into the SRLR.

3.7.2 Generic Analysis

GESTAR II Section 1.1.6.B: "For each new fuel design, the applicability of generic MCPR analyses described in Section 4 (of GESTAR II) or in the country specific supplement to this base document shall be confirmed for each operating cycle or a plant-specific analysis will be performed."

In addition to the MCPR statement in GESTAR II above, GEH/GNF confirms the applicability of generic LHGR analyses for each operating cycle or a plant-specific analysis will be performed.

3.7.2.1 Rod Withdrawal Error

Generic event analysis results have been calculated for the Rod Withdrawal Error. A plant cycle-specific evaluation will be performed for the GNF3 fuel design using NRC approved methods. The plant/cycle-specific result is then compared to the generic event analyses. If the calculated limit is less than the generic event analyses, then the generic limit is applied. If the calculated limit is greater than the generic event analyses, the calculated value is considered in the determination of the rated OLMCPR.

A description of the cycle-specific Rod Withdrawal Error analysis process is contained in Section S.2.2.1.5 of the US Supplement to GESTAR II.

3.7.2.2 Mislocated Fuel Loading Error

A mislocated bundle analysis to determine the potential influence of the GNF3 critical power correlation and the GNF3 fuel design will be performed for the first introduction of a reload batch of GNF3 into a BWR. This may justify the generic disposition of this AOO.

3.7.2.3 Off-Rated (Partial Power/Flow) Thermal Limits

The OLMCPR must be increased for the low core flow and low core power conditions to provide assurance that the fuel will not approach boiling transition in the event of an AOO at a low flow/power condition. Fuel LHGR operating limits are typically decreased for the low core flow and low core power conditions to provide assurance that the fuel rod T-M design and safety bases are not exceeded in the event of an AOO at the low flow/power condition. Extensive analyses have been performed for the low flow/power condition for many fuel designs and many plant/cycles. From the resulting database, a generic partial flow/power set of thermal limits has been established which is termed generic. Applicability of the generic partial flow/power thermal limits to the GNF3 fuel design, such as the generic limits for Average Power Range Monitor (APRM), RBM, Technical Specifications (ARTS) plants (Minimum Critical Power Ratio Flow Factor ($MCPR_f$), Minimum Critical Power Ratio Power Factor ($MCPR_p/K_p$), Linear Heat Generation Rate Flow Factor ($LHGRFAC_f$), and Linear Heat Generation Rate Power Factor ($LHGRFAC_p$)) will be confirmed by comparing the off-rated conditions thermal limits calculated for a GNF3 core with the generic limits and confirm the existence of margin.

3.7.2.3.1 Power Dependent Limits

ARTS and BWR/6 plants operate with generic power dependent MCPR and LHGR limits. Extensive transient analyses at various power and flow conditions are performed in determining these limits. The OLMCPR must be increased for low core power conditions. The power dependent LHGR operating limits are decreased for the low core power conditions.

[[

]]

3.7.2.3.2 Flow Dependent Limits

[[

]]

The plants chosen for these analyses are described below. These were selected due to their high power density as these plants and core designs incorporate the latest Extended Power Uprate (EPU) and the extended operating domain features of MELLLA+.

Model	BWR/6	BWR/4
Number of Bundles	[[	
Thermal Power, MWt		
Rated Core Flow, kg/sec		
Core Flow Range, % of rated		
Power Density, kW/l		]]

The generic partial power and flow dependent thermal limits established with the introduction of ARTS and used in these analyses are summarized in Tables 3-8, 3-9, 3-10, and 3-11.

3.7.2.3.2.1 $MCPR_f$

[[

]] In all cases, the GNF3 results are bounded by the generic limits that have been employed with the introduction of ARTS power and flow dependent limits. [[

]] and the results are also bounded by the generic limits.

3.7.2.3.2.2 $LHGRFAC_f$

[[

]]

3.7.2.3.2.3 Idle Recirculation Loop Startup and Fast Recirculation Flow Runout

The IRLS and FFRO are rapid flow increase transients and both are analyzed with ODYN for both the BWR/4 and BWR/6 at several power/flow points. [[

]]

3.7.2.3.2.4 Flow Dependent Limits Summary

[[

]]

Table 3-8 Generic Flow-Dependent $MCPR_f$ Limits

Flow [1] (%)	BWR/2-BWR/4				BWR/5-6			
	$MCPR_f$ 102.5% Max Flow	$MCPR_f$ 107% Max Flow	$MCPR_f$ 112% Max Flow	$MCPR_f$ 117% Max Flow	$MCPR_f$ 102.5% Max Flow	$MCPR_f$ 107% Max Flow	$MCPR_f$ 112% Max Flow	$MCPR_f$ 117% Max Flow
[[								
								]]

Note:

1. [[ ]]

Table 3-9 Generic Flow Dependent $LHGRFAC_f$ Limits

Flow [1] (%)	$LHGRFAC_f$ 102.5% Max Flow	$LHGRFAC_f$ 107% Max Flow	$LHGRFAC_f$ 112% Max Flow	$LHGRFAC_f$ 117% Max Flow
[[				
				]]

Note:

1. [[ ]]

Table 3-10 Generic Power Dependent K_p Limits

Power (%)	Generic K_p
[[	
	]]

Table 3-11 Generic Power Dependent $LHGRFAC_p$ Limits

Power (%)	Generic $LHGRFAC_p$
[[	
	]]

Table 3-12 BWR/4 Equilibrium Core Limiting MCPR Results for SFRO Transient

Flow (%)	Calculated	Generic	Calculated	Generic	Calculated	Generic	Calculated	Generic
	$MCPR_f$ 102.5% Max Flow	$MCPR_f$ 102.5% Max Flow	$MCPR_f$ 107% Max Flow	$MCPR_f$ 107% Max Flow	$MCPR_f$ 112% Max Flow	$MCPR_f$ 112% Max Flow	$MCPR_f$ 117% Max Flow	$MCPR_f$ 117% Max Flow
[[								
								]]

Table 3-13 BWR/6 Equilibrium Core Limiting MCPR Results for SFRO Transient

Flow (%)	Calculated	Generic	Calculated	Generic	Calculated	Generic	Calculated	Generic
	$MCPR_f$ 102.5% Max Flow	$MCPR_f$ 102.5% Max Flow	$MCPR_f$ 107% Max Flow	$MCPR_f$ 107% Max Flow	$MCPR_f$ - 112% Max Flow	$MCPR_f$ 112% Max Flow	$MCPR_f$ 117% Max Flow	$MCPR_f$ 117% Max Flow
[[								
								]]

Table 3-14 BWR/4 Equilibrium Core LHGRFAC$_f$ SFRO Results

Flow (%)	Calculated [1]	Generic	Calculated [1]	Generic	Calculated [1]	Generic	Calculated [1]	Generic
	LHGRFAC$_f$- 102.5% Max Flow		LHGRFAC$_f$- 107% Max Flow		LHGRFAC$_f$- 112% Max Flow		LHGRFAC$_f$- 117% Max Flow	
[[								
								]]

Note:

1. Includes a [[ ]] conservatism factor.

Table 3-15 BWR/4 Transition Core $LHGRFAC_f$ SFRO Results

	Calculated [1]	Generic	Calculated [1]	Generic	Calculated [1]	Generic	Calculated [1]	Generic
Flow (%)	$LHGRFAC_f$ 102.5% Max Flow		$LHGRFAC_f$ 107% Max Flow		$LHGRFAC_f$ 112% Max Flow		$LHGRFAC_f$ 117% Max Flow	
[[								
								]]

Note:

1. Includes a [[ ]] conservatism factor.

Table 3-16 BWR/6 Equilibrium Core $LHGRFAC_f$ SFRO Results

	Calculated [1]	Generic	Calculated [1]	Generic	Calculated [1]	Generic	Calculated [1]	Generic
Flow (%)	**$LHGRFAC_f$ 102.5% Max Flow**		**$LHGRFAC_f$ 107% Max Flow**		**$LHGRFAC_f$ 112% Max Flow**		**$LHGRFAC_f$ 117% Max Flow**	
[[								
								]]

Note:

1. Includes a [[ ]] conservatism factor.

Table 3-17 CPR Results for IRLS for BWR/4 Equilibrium Core

Power (%)	Flow(%)	ΔCPR	Required MCPR [1]	Generic $MCPR_f$ (102.5% Max Flow)	Required K_p [2]	Generic K_p
[[						
						]]

Notes:

1. Includes a [[ ]] conservatism factor.
2. The required K_p is conservatively calculated [[
   ]]

Table 3-18 TOP and MOP Results for IRLS for BWR/4 Equilibrium Core

Power (%)	Flow(%)	TOP	MOP	Calculated LHGRFAC [1]	Generic $LHGRFAC_f$ (102.5% Max Flow)	Generic $LHGRFAC_p$
[[						
						]]

Note:

1. Includes a [[ ]] conservatism factor.

Table 3-19 CPR Results for IRLS for BWR/6 Equilibrium Core

Power (%)	Flow(%)	ΔCPR	Required MCPR [1]	Generic $MCPR_f$ (102.5% Max Flow)	Required K_p [2]	Generic K_p
[[						
						]]

Notes:

1. Includes a [[ ]] conservatism factor.
2. The required K_p is conservatively calculated [[ ]]

Table 3-20 TOP and MOP Results for IRLS for BWR/6 Equilibrium Core

Power (%)	Flow(%)	TOP	MOP	Calculated LHGRFAC [1]	Generic $LHGRFAC_f$ (102.5% Max Flow)	Generic $LHGRFAC_p$
[[						
						]]

Note:

1. Includes a [[ ]] conservatism factor.

Table 3-21 CPR Results for FFRO for BWR/4 Equilibrium Core

Power (%)	Flow(%)	ΔCPR	Required MCPR [1]	Generic $MCPR_f$ (102.5% Max Flow)	Required K_p [2]	Generic K_p
[[						
						]]

Notes:

1. Includes a [[ ]] conservatism factor.
2. The required K_p is conservatively calculated [[
   ]]

Table 3-22 TOP and MOP Results for FFRO for BWR/4 Equilibrium Core

Power (%)	Flow(%)	TOP	MOP	Calculated LHGRFAC [1]	Generic $LHGRFAC_f$ (102.5% Max Flow)	Generic $LHGRFAC_p$
[[						
						]]

Note:

1. Includes a [[ ]] conservatism factor.

Table 3-23 CPR Results for FFRO BWR/6 Equilibrium Core

Power (%)	Flow(%)	ΔCPR	Required MCPR [1]	Generic $MCPR_f$ (102.5% Max Flow)	Required K_p [2]	Generic K_p
[[						
						]]

Notes:

1. Includes a [[ ]] conservatism factor.
2. The required K_p is conservatively calculated [[ ]]

Table 3-24 TOP and MOP Results for FFRO for BWR/6 Equilibrium

Power (%)	Flow(%)	TOP	MOP	Calculated LHGRFAC [1]	Generic $LHGRFAC_f$ (102.5% Max Flow)	Generic $LHGRFAC_p$
[[						
						]]

Note:

1. Includes a [[ ]] conservatism factor.

3.8 CRITICAL POWER CORRELATION

3.8.1 New Fuel Design Features

GESTAR II Section 1.1.7.A: "The currently approved critical power correlation will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design."

3.8.2 New Correlation Data

GESTAR II Section 1.1.7.B: "A new correlation may be established if significant new data exists for a fuel design(s)."

3.8.3 Critical Power Correlation Calculation

GESTAR II Section 1.1.7.C: "The criteria for establishing the new correlation are as follows:

A. The new correlation shall be based on full-scale prototypical test assemblies.

B. Tests shall be performed on assemblies with typical rod-to-rod peaking factors.

C. The functional form of the currently approved correlations shall be maintained.

D. Correlation fit to data shall be best fit.

E. One or more additional assemblies will be tested to verify correlation accuracy (i.e., test data not used to determine the new correlation coefficients).

F. Coefficients in the correlation shall be determined as described in Reference 1-5 or 1-6 of GESTAR II.

G. The uncertainty of the resulting correlation shall be determined by:

$$\sigma^2 = \frac{1}{N-1}\sum_{i=1}^{N}\left(\mu - ECPR_i\right)^2$$

Where:

σ = standard deviation

$\mu = \frac{1}{N}\sum_{i=1}^{N} ECPR_i$ = mean ECPR

N = Total number of data in both the data set used to determine the coefficients and the set used for verification

$ECPR$ = Calculated bundle critical power divided by experimentally determined bundle critical power."

Critical Power Correlation Results

The GEXL21 (NEDC-33880P, Revision 0, "*GEXL21 Correlation for GNF3 Fuel*," March 2017) database was obtained from Stern Laboratory tests of full-scale GNF3 bundle simulations. A statistical analysis has been performed for the GNF3 database used to develop the GEXL21 correlation, consisting of [[ ]] data points for [[ ]] different local peaking patterns. The GEXL21 correlation statistics were based on [[ ]] data points.

The GEXL21 correlation is valid for GNF3 fuel over the following range of state conditions:

- Pressure: [[ ]]
- Mass Flux*: [[ ]]
- Inlet Subcooling: [[ ]]
- R-factor: [[ ]]

[[

]]. Refer to Figure 3-16.

[[

]]

Figure 3-16 Mass Flux vs. R-Factor Plane

In addition, there is an additive constant applied to each fuel rod location [[ ]] For GNF3, the additive constants used in the design process are provided in Table 3-25. [[ ]]

Table 3-25 GEXL21 Additive Constants for GNF3

Fuel Rod Lattice Position	Fuel Rod Additive Constant
[[	
	]]

Notes:

1. [[ ]]

The resulting GEXL21 correlation for the critical quality (dimensionless) applicable to GNF3 fuel is of the form: $X_C = \sum_{i=1}^{18} A_i V_i$ where the variables and their coefficients are defined in Table 3-26:

Table 3-26 GEXL21 Variables and Coefficients

i	V_i		A_i
1	[[		
2			
3			
4			
5			
6			
7			
8			
9			
10			
11			
12			
13			
14			
15			
16			
17			
18			]]

Where:

G = Mass flux in 10^6 pounds per hour per square foot (Mlb/hr-ft^2)

P = Pressure in pounds per square inch (psia)

DQ = Thermal diameter in inches

LB = Boiling length in inches

LA = Annular flow length in inches

R = R-factor

The terms that comprise the form of the correlation have been previously approved by the NRC. These terms are specifically identified in References 42 and 43.

Conclusion

The GNF3 fuel assembly has a different PLR configuration and spacer design relative to previous fuel designs. Therefore, a new correlation has been established which is based on the same terms and form as the previous correlation. The new correlation, GEXL21, has been established based on significant new data for the GNF3 fuel design. Criteria a through g defined above have been used in the development of the GEXL21 correlation.

Based on the [[ ]] data points used to develop and verify the GEXL21 correlation statistics, the mean ECPR, μ, was determined to be [[ ]], with a standard deviation, σ, of [[ ]]. [[ ]]

3.9 STABILITY

GESTAR II Section 1.1.8: "New fuel designs must satisfy either criterion A or B below:

A. The stability behavior, as indicated by core and limiting channel decay ratios, must be equal to or better than a previously approved GE BWR fuel design.

B. If the core and limiting channel decay ratios are not equal to or better than a previously approved GE fuel design, it must be demonstrated that there is no change to the exclusion zone."

The GNF3 fuel design was analyzed against both Criterion A and Criterion B of GESTAR II Section 1.1.8. Acceptance of the GNF3 fuel design is based on Criterion B.

Stability Analytical Models and Analysis Procedures

The stability compliance calculations utilize the approved ODYSY methodology (Reference 30). ODYSY is a frequency domain program that calculates both the core and channel decay ratios for a prescribed fuel design, plant configuration, and plant operating state.

Analysis with Respect to Criterion A

Previous fuel designs have demonstrated acceptable stability performance, therefore, a comparison to these fuels ensures that the new fuel designs also have acceptable performance. The fuel design comparative evaluation in Criterion A will be performed as follows:

1. [[

]]

The core and channel decay ratios for both fuel designs shall be calculated using identical operating state conditions for power, flow, inlet subcooling, axial and radial core power shapes, and core pressure.

The power-flow condition selected shall be on the rated power control rod line and near the point of minimum recirculation pump speed. The methods and procedures used to analyze both fuel designs shall be identical.

Calculations were performed to compare the GNF3 design with the earlier, NRC-approved P8x8R fuel design. [[

]] The cycle exposure dependent decay ratio results for GNF3 and P8x8R are presented below in Table 3-27.

Table 3-27 Decay Ratios for Loose Inlet Orifice Plant

	Core			**Channel**
	Beginning of Equilibrium Cycle (BOEC)	**Middle of Equilibrium Cycle (MOEC)**	**End of Equilibrium Cycle (EOEC)**	
GNF3	[[			
P8x8R				]]

The plant analyzed had relatively "loose" inlet flow orifices. The GNF3 and P8x8R decay ratios for relatively "tight" inlet flow orifices are presented in Table 3-28.

Table 3-28 Decay Ratios for Tight Inlet Orifice Plant

	Core			**Channel**
	BOEC	**MOEC**	**EOEC**	
GNF3	[[			
P8x8R				]]

[[

]]

Analysis with Respect to Criterion B

For the same plant, the Exclusion Region analysis was performed for both GNF3 and P8x8R and both "loose" and "tight" inlet flow orifices. The Exclusion Region intercepts with the High Flow Control Line (HFCL) and Natural Circulation Line (NCL) are given in Tables 3-29 and 3-30.

Table 3-29 Exclusion Regions for Loose Inlet Orifice Plant

	P8x8R		GNF3	
	%P	%F	%P	%F
HFCL	[[			
NCL				]]

Table 3-30 Exclusion Regions for Tight Inlet Orifice Plant

	P8x8R		GNF3	
	%P	%F	%P	%F
HFCL	[[			
NCL				]]

Stability Compliance

As demonstrated in Tables 3-29 and 3-30, introduction of GNF3 to a plant operating with P8x8R does not cause the Exclusion Region to become larger. This validates the GNF3 stability performance under Criterion B of GESTAR II Section 1.1.8.

3.10 OVERPRESSURE PROTECTION ANALYSIS

GESTAR II Section 1.1.9: "Adherence to the ASME overpressure protection criteria shall be demonstrated on plant cycle specific analysis."

Overpressure Protection Analysis Acceptance Criterion

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Class I, permits pressure transients up to 10% over design pressure for "upset conditions". Section III to the Code allows credit to be taken for the scram protection system as a pressure protection device when determining the required safety valve capacities for nuclear vessels.

The GEH analysis to demonstrate vessel overpressure protection is performed assuming that all Main Steam Isolation Valves (MSIVs) close inadvertently and that the MSIV position switch fails to initiate a scram. Using this low probability event definition, application of an "emergency condition" limit is considered appropriate. However, GEH conservatively applies the "upset" code requirements.

Overpressure Protection Analytical Models and Analysis Procedures

The primary methods of the transient analysis process used in the calculation of the vessel overpressure during an AOO include: (1) lattice physics models (TGBLA, Reference 16); (2) 3D core simulator (PANACEA, Reference 16); and (3) 1D transient model (ODYN, References 22, 23, and 24) or a combination 1D/3D method (TRACG, References 25, 26, and 28). All of these models are NRC-approved.

The nuclear behavioral libraries, for GNF3 fuel, are generated by TGBLA and then are used as input to PANACEA. PANACEA, based on the cycle-specific reference core loading pattern, calculates the core state and the nuclear parameters for input to the plant transient model, ODYN or TRACG. ODYN calculates the time-dependent plant response to the prescribed transient using a 1D (axial) representation of the core and TRACG uses an advanced realistic combination 1D and 3D method. The output of ODYN and TRACG includes the vessel pressure.

Overpressure Protection Analysis Compliance

The calculated vessel pressure for MSIV inadvertent closure may be dependent upon the fuel design and core loading pattern. Compliance with the overpressure protection criterion is demonstrated by cycle-dependent analysis prior to the operation of that cycle.

A description of the criteria, models and procedure for vessel overpressure protection analysis is contained in Section S.3 of the US Supplement to GESTAR II (Reference 1).

3.11 Loss-of-Coolant Accident Analysis

The SAFER/PRIME Emergency Core Cooling System (ECCS) evaluation methodology is used to determine the effects of the postulated Loss-of-Coolant Accident (LOCA) in accordance with the requirements of 10 CFR 50.46 and Appendix K. This methodology is NRC-approved and is described in Section S.2.2.3.2 of the US Supplement to GESTAR II (Reference 1) and its references. The SAFER/PRIME evaluation methodology is used for all GE BWRs.

In addition to the SAFER/PRIME methodology, a best-estimate plus uncertainties method is also available for ECCS performance evaluation. This methodology, designated as TRACG-LOCA, (Reference 44) is described Section S.2.2.3.2 of the US Supplement to GESTAR II (Reference 1) and its references.

Either SAFER/PRIME or TRACG-LOCA methodology can be used for ECCS performance evaluation calculations for postulated LOCAs.

The SAFER/PRIME methodology uses improved ECCS evaluation models along with a realistic application approach to calculate a licensing PCT with margin substantiated by statistical considerations. Nominal values are used for most inputs, and Appendix K required inputs are utilized to identify the limiting break in order to establish the licensing basis values for comparison to the 10 CFR 50.46 limits. A description of the SAFER/PRIME methodology is contained in Sections S.2.2.3.2.4 and S.2.2.3.2.5 of the US Supplement to GESTAR II and its references. Four different GEH computer codes are utilized to calculate LOCA analyses results. These models are briefly described below.

1. Short-Term Thermal-Hydraulic Model (LAMB)

   The LAMB model (Reference 45) is used to analyze the short-term thermodynamic and thermal-hydraulic behavior of the coolant in the vessel during a postulated LOCA. In particular, this model predicts the core flow, core inlet enthalpy and core pressure during the blowdown prior to the end of lower plenum flashing. The detailed features of the fuel design do not significantly affect the system response; therefore, no modifications of this model are required for application to the GNF3 fuel design.

2. Transient Boiling Transition Model (TASC)

   This model is used to evaluate the short-term thermal-hydraulic response of the coolant in the hot channel of the core during a postulated LOCA. In particular, the calculated time of boiling transition (the onset of loss of nucleate boiling) is used as input to the core heatup model of SAFER described later in this section. The details of the fuel design can affect the calculated time to boiling transition. The TASC code (Reference 29) is a single hot channel thermal hydraulic analysis code which accepts detailed bundle geometry input that designates different types of rod groups within the bundle to explicitly model axially varying flow areas and heat transfer areas while incorporating the bundle specific critical power correlation described in Section 3.8. This model is the same one used for calculating the hot channel behavior during AOOs as described in Section 3.7. No modifications of this model are required for application to the GNF3 fuel design.

3. Long-Term Thermal-Hydraulic Model (SAFER)

   This model is used to analyze the long-term thermal-hydraulic behavior of the coolant in the vessel for all breaks. The SAFER code (References 46 to 50) calculates the uncovery and reflooding of the fuel and the duration of spray cooling. This code provides a realistic nodal representation of the counter current flow limiting phenomena at all flow restrictions between the core and adjacent regions and a realistic representation of the numerous leakage paths that exist in a BWR between the core and bypass regions. These leakage paths serve the important function of helping to refill the lower plenum and subsequently reflood the core region. Counter current flow limiting modeling in the SAFER code for the GNF3 configuration will be validated prior to plant-specific application. The SAFER code also calculates realistic core heat transfer coefficients. The SAFER code employs a heatup model with a simplified radiation heat transfer correlation to calculate PCT and local maximum oxidation. For calculated events in which the PCT is substantially below design limits and no cladding perforations are expected to occur, the PCT and local maximum oxidation fraction from SAFER can be used directly without recourse to additional calculations using the CORCOOL code. Detailed axial bundle geometry is not used in the SAFER methodology; therefore, no modifications of this model are required for application to the GNF3 fuel design.

4. Core Heatup Model (CORCOOL)

   The CORCOOL model (References 46 to 50) solves the transient heat transfer equations for the highest power assembly, for the entire LOCA transient. The various heat transfer modes considered include nucleate boiling, film boiling (flow and pool), core spray heat transfer and thermal radiation. The introduction of GNF3 and its multiple PLR rod heights can be handled by the CORCOOL code because the CORCOOL model accounts for changes in the number of rods in the lattice at different axial locations by axially varying active flow within the channel. CORCOOL can accommodate the designation of separate PLR groups such that different PLR lengths can be input. PLR height for specific rod groupings within the bundle can be specified in the CORCOOL input. No modifications of this model are required for application to the GNF3 fuel design.

5. Best Estimate Fuel Rod T-M Model (PRIME)

   The PRIME model (Reference 7) has been developed to provide best estimate predictions of the thermal performance of GEH/GNF nuclear fuel rods experiencing variable power histories. For ECCS analyses, the PRIME model is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA. No modifications of this model are required for application to the GNF3 fuel design.

3.11.1 Emergency Core Cooling System Criteria

GESTAR II Section 1.1.10.A: "The criteria in 10 CFR 50.46 shall be met on plant-specific or bounding analyses."

The ECCS criteria in 10 CFR 50.46 are met by the exposure-dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit in plant-specific or bounding analyses. GEH demonstrates compliance with these ECCS criteria for any new fuel designs using NRC-approved analytical models and analysis procedures.

3.11.2 Plant MAPLHGR

GESTAR II Section 1.1.10.B: "Plant MAPLHGR adjustment factors must be confirmed when a new fuel design is introduced."

Plant MAPLHGR is sometimes adjusted for a specific operational configuration or region. The GNF3 design accommodates a Thermal Mechanical Operating Limit (TMOL) which varies with exposure. Consistent with current methodology, the analysis is performed using a limiting value exposure and corresponding LHGR/MAPLHGR values coinciding with the TMOL, considering pellet densification, swell and cladding creep down and ballooning, bounding these and other pertinent fuel rod dynamic behavior in operation. GEH will confirm the revised MAPLHGR limit for the GNF3 fuel design for the plant and cycle when it is introduced.

3.12 Rod Drop Accident Analysis

3.12.1 Cycle Specific Analysis

GESTAR II Section 1.1.11.A: "Plant cycle specific analysis results shall not exceed the licensing limit described in the country specific supplement to this base document."

A generic CRDA analysis confirming that peak fuel enthalpy limits are met was performed and documented in NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," March 1972 (Reference 51). NEDO-21231, "Banked Position Withdrawal Sequence," January 1977 (Reference 37) provides specified control rod sequences that maintain the rod worths to such low values that peak fuel enthalpies do not threaten the design or fuel cladding failure threshold. Plant-specific enthalpy calculations are only necessary for plants that do not follow a generically approved CRDA withdrawal sequence. Plant-specific rod worth calculations will be performed for any such reload as part of the reload analysis and will be shown to meet the specified limits. Plant-specific rod worth calculations are also performed based on approved withdrawal sequences to confirm that rod worths are bounded, which ensures that the licensing limit in GESTAR II is met. Therefore, compliance to this criterion for GNF3 fueled cores not having a generically approved CRDA withdrawal sequence will be demonstrated as part of the reload license process.

3.12.2 Bounding BPWS Analysis

GESTAR II Section 1.1.11.B: "Applicability of the bounding BPWS analysis must be confirmed."

Rod drop analyses were performed generically for BPWS plants in Reference 37. R. E. Engel to D. B. Vassallo, "Control Rod Drop Accident," MFN-026-82, February 24, 1982 (Reference 52) eliminates the need for CRDA analyses for plants that implement BPWS. In 2004, an alternate BPWS, "Improved BPWS Control Rod Insertion Process," NEDO-33091-A, Revision 2, July 2004, was approved by the US NRC (Reference 53). The analysis performed for GNF3 compliance consists of performing [[

]] The compliance calculations conform with a modified procedure documented in J. S. Charnley (GE) to M. Wayne Hodges (US NRC), “Revised Generic BPWS CRD Analysis,” MFN-034-087, April 22,1987, (Reference 54) which more accurately predicts the most reactive control rod, results in a more limiting control rod configuration, and takes credit for the BPWS scram function. The peak fuel enthalpy for the bounding analysis is still significantly lower than the design limit.

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This analysis demonstrates the applicability of the generic BPWS analyses. Plant and cycle-specific rod worth calculations will demonstrate that References 37, 51, and 52 are still valid as a part of the reload analysis.

3.12.3 Fuel Enthalpy Analysis

Based on a bounding postulated CRDA analysis, it was conservatively determined for the 8x8 fuel designs that approximately 850 fuel rods would reach a fuel enthalpy of 170 cal/g. This is the enthalpy limit for eventual cladding perforation. For the 9x9 GE11 and GE13 fuel designs, approximately 1,000 fuel rods would reach a fuel enthalpy of 170 cal/g, and for the 10x10 GE12, GE14, and GNF2 fuel designs, approximately 1,200 fuel rods would reach a fuel enthalpy of 170 cal/g.

As with the other 10x10 designs, when the bounding analysis is applied to GNF3, approximately 1,200 fuel rods are calculated to reach a fuel enthalpy of 170 cal/g. The number of bundles damaged is conservatively determined to be 13.47 bundles for GNF3.

3.13 REFUELING ACCIDENT

GESTAR II Section 1.1.12: “The consequences of a refuel accident as presented in the country-specific supplement to this base document or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed (using the methods and assumptions described in the country supplement) and documented when a new fuel design is introduced.”

Accidents that result in the release of radioactive materials directly to the containment can occur when the drywell is open and the reactor vessel head has been removed. The only credible accident that could lead to the release of significant quantities of fission products to the containment is one resulting from the accidental dropping of a fuel bundle onto the top of the core. This results in mechanical damage to the fuel rod cladding both in the dropped bundle and those in the core. This event occurs under non–operating conditions for the fuel with the core in a cold condition.

3.13.1 Fuel Damaged

GEH is now manufacturing a new design of the refueling mast with grapple head (NF–500). The new design has a circular cross–section mast versus the previous triangular cross–section mast. The new design is also more "rugged" and weighs more, 280.8 kg compared to 158.8 kg. Additionally, GNF has made changes in the fuel bundle configurations. The number of fuel rods has increased from the initial 7x7 array to the current GNF3 10x10 array with corresponding dimensional changes as well as the inclusion of PLRs.

A damage analysis is performed, taking into consideration the PLRs in the GNF3 design, based on the equivalence of 88.3 FLRs per GNF3 bundle. It is concluded that 169 and 149 equivalent FLRs failed for plants equipped with the NF-500 mast and the standard triangular refueling mast, respectively. The smallest number of damaged rods documented in any BWR Final Safety Analysis Report (FSAR) refueling accident for a 7x7 array is 111 rods.

3.13.2 Radiological Consequences Comparisons

Assuming all other operating parameters remain unchanged, the relative radiological consequence of a refueling accident can be assessed by comparing the equivalent number of fuel bundles damaged. For most BWRs, the FSAR analysis was based on 7x7 array fuel with a minimum of 111 damaged rods; therefore, the activity released is equivalent to that of (111/49) or 2.27 fuel bundles. The accident involving GNF3 with the NF-500 mast is equivalent to (169/88.3) or 1.91 bundles. With the traditional triangular mast, the damage is equivalent to (149/88.3) or 1.69 bundles. Therefore, the damaged GNF3 bundle equivalent is bounded by the 7x7 array of most original plant designs.

3.13.3 Power Peaking Factors

If the radial peaking factor assumed in the FSAR bounds the plant-specific radial peaking factor for GNF3, and the plant design was based on 7x7 fuel, then the radiological consequence of the GNF3 bundle drop is bounded by the original plant design as shown, and the criterion is met. If the radial peaking factor of the GNF3 core design is greater than that assumed in the FSAR, the effect can be accounted for by taking the ratio of these factors. For example, a GNF3 bundle with a radial peaking factor of 1.7, compared to a 7x7 bundle with a radial peaking factor of 1.5 typically used in the FSAR, is expected to have an activity release of (1.7/1.5)*(1.91/2.27) = 0.95 times the FSAR values. In this example, the criterion is again met.

Plants may have changed or modified the refueling masts, the FSAR or current licensing basis may be based on fuel types other than GE 7x7 fuel, and the plant-specific GNF3 radial peaking factor will depend on the core design. For these reasons, compliance to the refueling accident criterion is confirmed on a plant-specific basis during preparation for the GNF3 fuel transition.

3.14 ANTICIPATED TRANSIENT WITHOUT SCRAM

GESTAR II Section 1.1.13: "The fuel must meet either criteria A or B below:"

A. "A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients provided in GESTAR II References 1–7 and 1–8, shall be maintained for any operating conditions above the startup critical condition."

B. "If criterion 1.1.13.A is not satisfied, the limiting events (as described in GESTAR II References 1–7 and 1–8) will be evaluated to demonstrate that the plant response is within the ATWS criteria specified in GESTAR II References 1–7 and 1–8."

In response to the requirements of Alternate 3, set forth in NUREG–0460, References 55 and 56 present assessments of the capabilities of representative BWR plants to mitigate the consequences of a postulated ATWS event. Sensitivity studies are provided for the key parameters affecting plant response during the most limiting events requiring ATWS consideration. Values of parameters that fall within the range of characteristics studied have been shown to satisfy the ATWS acceptance criteria.

In terms of core and system response to an ATWS event, the core moderator void reactivity coefficient is the key parameter compared to other fuel and nuclear parameters that may change with a change in fuel type. Maintaining this coefficient within the range of point model void coefficients (or equivalent one–dimensional void coefficients) assumed in the sensitivity studies presented in References 55 and 56 when loading new fuel designs, assures that the conclusions reached regarding BWR mitigation of an ATWS event are still valid. Although the methodology used in References 55 and 56 shows some importance to the void coefficient, the more recent approved methodology in Reference 24 does not show the system response to be sensitive to the void coefficient. This evaluation is shown below in Section 3.14.1.

3.14.1 Void Coefficient Evaluation

The point model void coefficient must fall within the range of -8 to -14 cents/% voids in order for the new fuel design to meet the acceptance criterion. A preliminary evaluation of the GNF3 void coefficient indicates that the void coefficient is similar to GNF2, and may not always fall in this range.

Analyses have also been performed with ODYN with a core-wide [[ ]] increase in ODYN void coefficient magnitude. The results are presented in Table 3-31 for BOC and EOC conditions. [[

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In addition, the effect on the peak pool temperature response is also addressed. Sensitivity studies have been performed with a core-wide [[ ]] increase in the ODYN void coefficient magnitude. A sensitivity study was performed for a limiting Pressure Regulator Failure – Open (PRFO) at both BOC and EOC exposure conditions. The results shown in Table 3-32 below show that the peak pool temperature is [[ ]].

Table 3-31 ODYN Peak Vessel Pressure Void Coefficient Study

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

Table 3-32 Suppression Pool Peak Temperature Void Coefficient Study

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

As the GNF3 void coefficient is in a similar range to GNF2 and the sensitivity study above shows very small changes in key results to changes in void coefficient, the introduction of GNF3 will have a small effect on these key ATWS acceptance parameters.

3.14.2 Plant Response Evaluation

Because the GNF3 void coefficient may not always fall within the prescribed range, additional plant-specific ATWS evaluations will be performed for the introduction of GNF3 for a plant. This evaluation will assure that there is acceptable margin to the key ATWS acceptance criteria identified in Table 3-33.

Many plants have implemented power uprates, which have reduced ATWS margins to the acceptance limits and this makes it more difficult to implement a fleet-wide generic analysis.

Therefore, plants where margins are less than [[ ]] psi to the overpressure limit and/or less than [[ ]]°F to the peak suppression pool temperature limit will be re-analyzed with the introduction of GNF3. These criteria provide a substantial margin of conservatism compared to the sensitivity to the void coefficient described in Section 3.14.1 above. The margin criteria were established for the two parameters whose values may be primarily affected and are sometimes close to the ATWS acceptance criteria. The PCT and containment pressure have substantial margin for all plants. For example, the observed PCT for all plants has been at least 600 to 700°F below the acceptance criteria. The factor of three provides conservatism to ensure that plants whose key ATWS parameters are close to the limits will be analyzed. If the analyses are required, they will be performed at the time of plant fuel introduction using the Reference 24 and 29 NRC approved methodology.

Table 3-33 Key ATWS Acceptance Criteria

Acceptance Criteria	Limit
Peak Vessel Pressure (psig)	1,500
Peak Cladding Temperature (°F)	2,200
Peak Local Cladding Oxidation (%)	17
Peak Suppression Pool Temperature	Design Limit
Peak Containment Pressure	Design Limit

3.15 FUEL LOADING ERROR EVENT ANALYSIS

GESTAR II Section 1.1.14: "Section S.5.3 of the country-specific supplement presents the requirements for analyzing the FLE (misloaded or misoriented fuel bundle) as an Infrequent Incident. Should a plant not meet the requirements in Section S.5.3, the event will be analyzed as an AOO."

A. "As an Infrequent Incident, the FLE events are subject to the radiological limits of 10% of 10 CFR 100, or 10% of 10 CFR 50.67 for Alternate Source Term plants. Bounding radiological analysis of these events is referenced in the country-specific supplement to this base document."

B. "As an AOO, the FLE events are subject to the MCPR criteria." (GESTAR II Section 1.1.5 and Section 1.1.6).

3.15.1 FLE Radiological Analysis

The methodology of a Fuel Loading Error (FLE) radiological analysis in the GESTAR II country-specific supplement remains unchanged and applicable. In order to bound the consequences for this event, it is conservatively assumed that all of the fuel rods in five failed fuel assemblies (primary and four adjacent) will experience instantaneous failure. A conservative fuel bundle radial peaking factor of 2.5 was used to ensure that the peak bundle power to average bundle power ratio was bounded. In addition, a safety factor of 1.4 was used to address the variation in fission product inventory over the cycle of the operating fuel. Therefore, the FLE radiological analysis in the GESTAR II country-specific supplement is fuel type independent.

3.15.2 Fuel Loading Error AOO Analysis

As an AOO, the FLE events are analyzed consistent with Section 3.7. The FLE events may be analyzed as cycle-specific analysis or generic analysis.

4.0 LICENSING APPLICATION

4.1 APPLICABILITY

This report documents the completion of the generic portions of the GESTAR II requirements for the introduction of a new GEH or GNF fuel design into GEH BWRs. Revision 0 of the GNF3 compliance report is [[ ]] This document applies the approved PRIME T-M methodology to the GNF3 fuel assembly to the T-M design basis and all downstream analyses.

The ranges of operation that have been investigated include EPU power levels as well as the MELLLA+ operating domain expansion (Reference 57). Currently licensed operating domains and operational flexibility features have been considered where applicable. ATWSI studies will be included with the NFI report for a plant licensed for operation in the MELLLA+ domain that desires to install GNF3.

The evaluations documented in this report demonstrate that the GNF3 fuel design meets the GESTAR requirements for the introduction of a new fuel design.

4.2 PLANT-SPECIFIC APPLICATION PROCESS

In addition to the generic aspect of this GNF3 compliance document, the plant-specific application process will confirm that the plant-specific cycle-independent aspects of the GNF3 fuel introduction meets the design and licensing basis requirements of the plant. The cycle-independent analyses will be defined and evaluated consistent with the plant licensing basis.

The NFI report will document the cycle-independent plant-specific analyses for use by Licensee as input to the plant's 10 CFR 50.59 evaluation of the NFI. A typical table of contents for a plant-specific introduction is shown in Table 4-1.

Table 4-1 Typical Contents of New Fuel Introduction Report

1.0 INTRODUCTION AND SUMMARY
[[
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12.0 REFERENCES

5.0 SUMMARY AND CONCLUSION

This report documents the completion of the requirements for a new fuel design per the criteria defined in GESTAR II. Section 1.1 of GESTAR II defines a set of fuel licensing acceptance criteria for evaluating new fuel designs and for determining the applicability of generic analyses to these new designs. As stated in GESTAR II, "Fuel design compliance with the fuel licensing acceptance criteria constitutes US NRC acceptance and approval of the fuel design without specific US NRC review." All of the criteria defined in GESTAR II have been met for the GNF3 fuel design.

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Appendix A – PRIME Based GNF3 LHGR Envelopes

BWR/4-6

Tables A-1 and A-2 provide the complete set of UO_2 and gadolinia PRIME based thermal-mechanical LHGR limits applicable to GNF3 fuel bundles for BWR/4-6 plants.

Table A-1 BWR/4-6 UO_2 Rod Power Exposure Envelope

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										]]

Note: The power limit at a given exposure between BOL and the first knee is equal to the pellet maximum power. The power limit at a given exposure between the first and second or second and third or between third and EOL is determined by interpolation of power with exposure.

Table A-2 BWR/4-6 (U,Gd)O_2 Rod Power Exposure Envelopes

[[										
										]]