Enclosure 1 Contains Proprietary Information to be Withheld from Public Disclosure Pursuant to 10 CFR 2.390

PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG Nuclear LLC

10 CFR 50.90

LR-N17-0163
LAR H17-07

NOV 09 2017

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

Hope Creek Generating Station
Renewed Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: **License Amendment Request – Safety Limit Minimum Critical Power Ratio Change**

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Renewed Facility Operating License No. NPF-57 for Hope Creek Generating Station (HCGS). In accordance with 10 CFR 50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed license amendment request (LAR) modifies Technical Specifications (TS) Section 2.1 ("Safety Limits"). Specifically, this change incorporates a revised Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loop operation (TLO) and single recirculation loop operation (SLO) due to the cycle specific analysis performed by Global Nuclear Fuel (GNF) for HCGS Cycle 22.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and it has been determined that this request involves no significant hazards considerations.

There are no regulatory commitments contained in this letter.

There are three attachments and three enclosures to this letter. Attachment 1 provides an evaluation supporting the proposed changes. The marked-up TS page, with the proposed changes indicated, is provided in Attachment 2. Attachment 3 provides, for information only, proposed changes to the TS Bases. Enclosure 1, GNF Report 004N5379-R0-P, specifies the required SLMCPRs for HCGS Cycle 22. Enclosure 1 contains information proprietary to GNF. GNF requests that the document be withheld from public disclosure in accordance with 10 CFR

Enclosure 1 transmitted herewith contains SUNSI. When separated from enclosure 1, this transmittal document is decontrolled.

10 CFR 50.90

Enclosure 1 Contains Proprietary Information to be Withheld from Public Disclosure Pursuant to 10 CFR 2.390

2.390(a)(4). Enclosure 2 contains a non-proprietary version of the GNF Report, 004N5379-R0-NP. An affidavit supporting this request is contained in Enclosure 3.

These proposed changes have been reviewed by the Plant Operations Review Committee.

PSEG requests NRC approval of the proposed LAR by April 13, 2018 to support the HCGS refueling outage in Spring 2018 (Reload 21). Once approved, the amendment shall be implemented prior to startup from the refueling outage.

If you have any questions or require additional information, please contact Mr. Lee Marabella at (856) 339-1208.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 11/9/17
(Date)

Respectfully,

Eric Carr
Site Vice President
Hope Creek Generating Station

Attachments:

1. Request for Changes to Technical Specifications
2. Technical Specification Pages with Proposed Changes
3. Technical Specification Bases Pages with Proposed Changes (For Information Only)

Enclosures:

1. Proprietary Version of GNF Report 004N5379-R0-P
2. Non-Proprietary Version of GNF Report 004N5379-R0-NP
3. GNF Affidavit in Support of Request to Withhold Information

cc: Administrator, Region I, NRC
Project Manager, NRC
NRC Senior Resident Inspector, Hope Creek
Mr. P. Mulligan, Chief, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. MacEwen, Hope Creek Commitment Tracking Coordinator

Attachment 1

Request for Changes to Technical Specifications

HOPE CREEK GENERATING STATION
RENEWED FACILITY OPERATING LICENSE NO. NPF-57
DOCKET NO. 50-354

License Amendment Request – Safety Limit Minimum Critical Power Ratio Change

Table of Contents

1.0 DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating License No. NPF-57 for Hope Creek Generating Station (HCGS).

The proposed change modifies Technical Specification (TS) Section 2.1 ("Safety Limits"). Specifically, this change incorporates a revised Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loop operation (TLO) and single recirculation loop operation (SLO) due to the cycle specific analysis performed by Global Nuclear Fuel (GNF) for HCGS Cycle 22.

2.0 PROPOSED CHANGE

The proposed change involves revising the SLMCPR contained in TS Section 2.1 for TLO and SLO. The SLMCPR for TLO is being changed from ≥ 1.08 to ≥ 1.09. The SLMCPR for SLO is being changed from ≥ 1.11 to ≥ 1.12.

Marked up TS page 2-1 showing the requested change is provided in Attachment 2.

Proposed changes to the TS Bases are provided in Attachment 3 of this submittal for information only. Changes to the affected TS Bases pages will be incorporated in accordance with TS 6.15, "Technical Specifications (TS) Bases Control Program."

3.0 BACKGROUND

The SLMCPR analysis establishes SLMCPR values that ensure at least 99.9% of all fuel rods in the core do not experience transition boiling during normal operation and analyzed transients. The SLMCPRs are re-evaluated for each reload using NRC-approved methodology to incorporate plant and cycle specific parameters for the current core design. As such, the calculated SLMCPR values may change on a cycle specific basis.

The proposed change involves revising the SLMCPR contained in TS Section 2.1 for TLO and SLO due to the cycle specific analysis performed by GNF for HCGS Cycle 22. Operation in accordance with the revised SLMCPRs will continue to preserve the existing margin to transition boiling and therefore protect the integrity of the fuel cladding barrier.

4.0 TECHNICAL ANALYSIS

The proposed TS change revises the SLMCPR contained in TS Section 2.1 for TLO and SLO due to the cycle specific analysis performed by GNF for HCGS Cycle 22.

The SLMCPRs are calculated using NRC-approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Revision 24 (Reference 1). Information supporting the cycle specific SLMCPRs is included in Enclosure 1. That enclosure summarizes the methodology, inputs, and results for the calculated SLMCPRs.

The SLMCPR analysis establishes SLMCPR values that ensure at least 99.9% of all fuel rods in the core do not experience transition boiling during normal operation and analyzed transients. The SLMCPRs are calculated to include cycle specific parameters and in general, are dominated by two key parameters: (1) flatness of the core bundle-by-bundle MCPR distribution, and (2) flatness of the bundle pin-by-pin power/R-factor distribution.

The HCGS Cycle 22 fresh fuel pin-by-pin power/R-Factor distribution is flatter than the previous cycle fresh fuel pin-by-pin power/R-Factor distribution. The overall core power distribution flatness along with the cycle-to-cycle variation in the core loading tends to produce an increase in the calculated SLMCPR.

No plant hardware or operational changes are required with this proposed change.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Renewed Facility Operating License No. NPF-57 for Hope Creek Generating Station (HCGS).

The proposed change modifies Technical Specifications (TS) Section 2.1 ("Safety Limits"). Specifically, this change incorporates a revised Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loop operation (TLO) and single recirculation loop operation (SLO) due to the cycle specific analysis for HCGS Cycle 22.

PSEG has evaluated whether or not a Significant Hazards Consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The required SLMCPRs for HCGS Cycle 22 are calculated using NRC-approved methodology. The SLMCPR values, contained in TS Section 2.1 ("Safety Limits"), ensure at least 99.9% of all fuel rods in the core do not experience transition boiling during normal operation and analyzed transients, preserving fuel cladding integrity. The proposed change to the SLMCPR values ensures this criterion continues to be met, and therefore does not increase the probability or consequences of an accident previously evaluated. In addition, no plant hardware or operational changes are required with this proposed change.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The required SLMCPRs for HCGS Cycle 22 are calculated using NRC-approved methodology. The SLMCPR values, contained in TS Section 2.1, ensure at least 99.9% of all fuel rods in the core do not experience transition boiling during normal operation and analyzed transients. The proposed change to the SLMCPR values does not involve any plant hardware or operational changes and does not create any new precursors to an accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The required SLMCPRs for HCGS Cycle 22 are calculated using NRC-approved methodology. The SLMCPR values, contained in TS Section 2.1, ensure at least 99.9% of all fuel rods in the core do not experience transition boiling during normal operation and analyzed transients, preserving fuel cladding integrity. The revised SLMCPR values ensure this criterion continues to be met. In addition, the proposed change to the SLMCPR values does not adversely affect the design basis function or performance of a structure, system, or component as described in the HCGS UFSAR.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based upon the above, PSEG Nuclear concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements and Criteria

10 CFR 50.36 Technical Specifications

10 CFR 50.36, "Technical specifications" identifies the requirements for the Technical Specification categories for operating power plants: (1) *Safety limits, limiting safety system settings, and limiting control settings*, (2) *Limiting conditions for operation,* (3) *Surveillance requirements*, (4) *Design features,* (5) *Administrative controls,* (6) *Decommissioning,* (7) *Initial notification, and* (8) *Written Reports*. Specifically, 10 CFR 50.36(c)(1)(i)(A) states: *Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.*

The SLMCPR values, contained in TS Section 2.1 ("Safety Limits"), ensure at least 99.9% of all fuel rods in the core do not experience transition boiling during normal operation and analyzed transients, preserving fuel cladding integrity. The proposed change to the SLMCPR values ensures this criterion continues to be met.

In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Revision 24, March 2017.

Attachment 2

Technical Specification Pages with Proposed Changes

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specification for Renewed Facility Operating License No. NPF-57 is affected by this change request:

Technical Specification	Page
2.1.2	2-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 24% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 24% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow:

1.09

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be ≥ ~~1.08~~ for two recirculation loop operation and shall be ≥ ~~1.11~~ for single recirculation loop operation.

1.12

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow and the MCPR below the values for the fuel stated in LCO 2.1.2, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

Attachment 3

Technical Specification Bases Pages with Proposed Changes

(For Information Only)

TECHNICAL SPECIFICATION BASES PAGES WITH PROPOSED CHANGES

The following Technical Specification Bases for Renewed Facility Operating License No. NPF-57 is affected by this change request:

Technical Specification Bases	Page
2.0	B 2-1

2.1 SAFETY LIMITS

not less than the limits specified in Specification 2.1.2

BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is ~~≥ 1.08 for two recirculation loop operation and ≥ 1.11 for single recirculation loop operation~~. These MCPR values represent a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the applicable NRC-approved critical power correlations are not valid for all critical power calculations performed at reduced pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 24% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

Enclosure 2

Non-Proprietary Version of GNF Report 004N5379-R0-NP

October 2017
GNF-004N5379-R0-NP

Non-Proprietary Information – Class I (Public)

GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR

Hope Creek Cycle 22

Information Notice

This is a non-proprietary version of the document GNF-004N5379-R0-P, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

Important Notice Regarding Contents of this Report Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of providing information regarding the requested changes to the Technical Specification SLMCPR for PSEG Hope Creek. The only undertakings of GNF-A with respect to information in this document are contained in the contract between GNF-A and PSEG, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone other than PSEG, or for purposes other than those for which it is intended is not authorized; and with respect to any unauthorized use, GNF-A makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

Table of Contents

List of Tables

1.0 Summary

The requested changes to the Technical Specification (TS) Safety Limit Minimum Critical Power Ratio (SLMCPR) values are 1.09 for Two Recirculation Loop Operation (TLO) and 1.12 for Single Recirculation Loop Operation (SLO) for Hope Creek Cycle 22. This SLMCPR change is applicable to the power level proposed in the Thermal Power Optimization (TPO) license amendment (Reference 1) currently under review as well as to the current power level. Additional details are provided in Table 1.

A main contributor to the change in the limiting case for Cycle 22 is the bundle pin-by-pin power/R-Factor distribution which produces a flatter core power distribution than that of the limiting case in the previous cycle.

2.0 Regulatory Basis

10 Code of Federal Regulations (CFR) 50.36(c)(1), "Technical Specifications," requires that power reactor facility TS include safety limits for process variables that protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environment. The purpose of the SLMCPR is to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during steady state operation and analyzed transients.

General Design Criterion (GDC) 10, "Reactor Design," of Appendix A to 10 CFR 50 states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that the SAFDLs are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Guidance on the acceptability of the reactivity control systems, the reactor core, and fuel system design is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." Specifically, SRP Section 4.2, "Fuel System Design," specifies all fuel damage criteria for evaluation of whether fuel designs meet the SAFDLs. SRP Section 4.4, "Thermal Hydraulic Design," provides guidance on the review of thermal-hydraulic design in meeting the requirement of GDC 10 and the fuel design criteria established in SRP Section 4.2.

3.0 Methodology

GNF performs SLMCPR calculations in accordance with NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel (GESTAR II)" (Reference 2) for plants such as Hope Creek that are equipped with the GNF ACUMEN core monitoring system by using the following Nuclear Regulatory Commission (NRC) approved methodologies and uncertainties:

- NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," (Reference 3).
- NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," (Reference 4).
- NEDC-32505P-A, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," (Reference 5).

These methodologies were used for the Hope Creek Cycle 21 and Cycle 22 SLMCPR calculations.

3.1. Methodology Restrictions

Four restrictions were identified on page 3 of NRC's Safety Evaluation (SE) relating to the General Electric (GE) Licensing Topical Reports (LTRs) NEDC-32601P, NEDC-32694P, and in Amendment 25 to NEDE-24011-P-A (Reference 6).

The four restrictions were addressed for GE14 in FLN-2001-016 "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR" (Reference 7) and FLN-2001-17 "Power Distribution and R-Factor Methodologies" (Reference 8).

The following statement was extracted from the generic compliance report for the GNF2 fuel assembly design (Reference 9) that GNF sent to the NRC in March of 2007:

> "The NRC Safety Evaluation (SE) for NEDC-32694P-A provides four actions to follow whenever a new fuel design is introduced. These four conditions are listed in Section 3 of the SE. In the last paragraph of Section 3.2.2 of the Technical Evaluation Report included in the SE are the statements "GE has evaluated this effect for the 8x8, 9x9, and 10x10 lattices and has indicated that the R-Factor uncertainty will be increased … to account for the correlation of rod power uncertainties" and "it is noted that the effect of the rod-to-rod correlation has a significant dependence on the fuel lattice (e.g., 9x9 versus 10x10). Therefore, in order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice." Therefore, the definition of a new fuel design is based on the lattice array dimensions (e.g., NxN). Because GNF2 is a 10x10, and the evaluations in NEDC-32694P-A include 10x10, then these four actions are not applicable to GNF2."

In an NRC audit report (Reference 10) for this document, Section 3.4.1 page 59 states:

> "The NRC staff's SE of NEDC-32694P-A (Reference 19 of NEDC-33270P) provides four actions to follow whenever a new fuel design is introduced. These four conditions are listed in Section 3.0 of the SE. The analysis and evaluation of the GNF2 fuel design was evaluated in accordance with the limitations and conditions stated in the NRC staff's SE, and is acceptable."

Another methodology restriction is identified on page 4 of the NRC's SE relating to the GE LTR NEDC-32505P (Reference 11). Specifically, it states that "if new fuel is introduced, GENE must confirm that the revised R-factor method is still valid based on new test data." NEDC-32505P addressed the GE12 10x10 lattice design (i.e., how the R-Factor for a rod is calculated based upon its immediate surroundings (fuel rods, water rods or channel wall)). Validation is provided by the fact that the methodology generates accurate predictions of Critical Power Ratio (CPR) with reasonable bias and uncertainty. The applicability of the R-Factor method is coupled and documented (along with fuel specific additive constants) with the GEXL correlation development (References 12 and 13), which is submitted as a part of GESTAR II compliance for each new fuel product line.

4.0 Discussion

In this discussion, the TLO nomenclature is used for two recirculation loops in operation, and the SLO nomenclature is used for one recirculation loop in operation.

Table 2 provides the description of the current cycle and previous cycle for the reference loading pattern as defined by NEDE-24011-P-A (Reference 2).

4.1. Major Contributors to SLMCPR Change

In general, for a given power-flow statepoint, the calculated safety limit is dominated by two key parameters: (1) flatness of the core bundle-by-bundle Minimum Critical Power Ratio (MCPR) distribution, and (2) flatness of the bundle pin-by-pin power/R-Factor distribution. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR. Therefore, the calculated SLMCPR may change whenever there are changes to the core configuration or to the fresh fuel designs. The plant-cycle specific SLMCPR methodology accounts for these factors.

The current cycle core design has produced similar results to the previous cycle core design, that is, the SLMCPR values are within 0.005 for the TLO limiting case. The change in the calculated SLMCPR can be attributed to cycle-to-cycle variation. A key component of this variation is both the bundle-by-bundle MCPR distribution as well as the pin-by-pin power/R-Factor distribution. The only effect the increase in power level has is a difference in the flow uncertainty, which decreases due to the increase in the minimum core flow.

For the limiting TLO case, the current fresh fuel pin-by-pin power/R-Factor distribution is flatter than the previous cycle fresh fuel pin-by-pin power/R-Factor distribution while the core bundle-by-bundle MCPR distribution is slightly more peaked than the previous cycle. While the current cycle core bundle-by-bundle MCPR distribution is slightly more peaked, the change in the fresh fuel pin-by-pin power/R-Factor is more significant, thus the combination of the two distributions produces a flatter core power distribution. The overall core power distribution flatness along with the cycle-to-cycle variation in the core loading tends to produce an increase in the calculated SLMCPR.

The current cycle's change in the Monte Carlo SLO SLMCPR from the previous cycle is consistent with the Monte Carlo TLO SLMCPR change between the two cycles. The SLO values are greater than the TLO values as expected due to the increase in uncertainties used for the SLO case.

4.2. Deviations from Standard Uncertainties

Table 3 provides a list of deviations from NRC-approved uncertainties (References 3 and 4). A discussion of deviations from these NRC-approved values follows, all of which are conservative relative to NRC-approved values.

4.2.1. R-Factor

GNF has generically increased the GEXL R-Factor uncertainty from [[ ]] to account for an increase in channel bow due to the phenomena called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of the approved R-Factor uncertainty. Reference 14 technically justifies that a GEXL R-Factor uncertainty of [[ ]] accounts for a channel bow uncertainty of up to [[ ]]. The Hope Creek Cycle 22 analysis shows an expected channel bow uncertainty of [[ ]], which is bounded by a GEXL R-Factor uncertainty of [[ ]]. Thus, the use of a GEXL R-Factor uncertainty of [[ ]] adequately accounts for the expected control blade shadow corrosion-induced channel bow. The effect of this change is considered not significant (i.e., < 0.005 increase on SLMCPR).

4.2.2. Core Flow Rate and Random Effective TIP Reading

In Reference 15, GNF committed to the expansion of the state points used in the determination of the SLMCPR. Consistent with the Reference 15 commitments, GNF performs analyses at the rated core power and minimum licensed core flow point in addition to analyses at the rated core power and rated core flow point. The approved SLMCPR methodology is applied at each state point that is analyzed.

For the TLO calculations performed at 97.1% core flow, the approved uncertainty values for the core flow rate (2.5%) and the random effective Traversing In-Core Probe (TIP) reading (1.2%) are conservatively adjusted by dividing them by 97.1/100.

The core flow and random TIP reading uncertainties used in the SLO minimum core flow SLMCPR analysis remain the same as in the rated core flow SLO SLMCPR analysis because these uncertainties (which are substantially larger than used in the TLO analysis) already account for the effects of operating at reduced core flow.

4.2.3. Flow Area Uncertainty

GNF has calculated the flow area uncertainty for GNF2 and GE14 using the process described in Section 2.7 of Reference 3. It was determined that the flow area uncertainty for GNF2 and GE14 is conservatively bounded by a value of [[ ]]. Because this is larger than the Reference 3 value of [[ ]] the bounding value was used in the SLMCPR calculations. The effect of this change is considered not significant (i.e., < 0.005 increase on SLMCPR).

4.2.4. Fuel Axial Power Shape Penalty

The GEXL correlation critical power uncertainty and bias are established for each fuel product line according to a process described in NEDE-24011-P-A (Reference 2).

GNF determined that higher uncertainties and non-conservative biases in the GEXL correlations for certain types of axial power shapes could exist relative to the NRC-approved methodology values (References 16, 17, 18, and 19). The GE14 and GNF2 product lines are potentially affected in this manner only by Double-Hump (D-H) axial power shapes.

The D-H axial power shape did not occur on any of the limiting bundles (i.e., those contributing to the 0.1% of rods susceptible to transition boiling) in the current and/or prior cycle limiting cases. Therefore, D-H power shape penalties were not applied to the GEXL critical power uncertainty or bias.

4.3. Additional SLMCPR Licensing Conditions

As shown in Table 1, there are no added penalties applied to the calculated SLMCPR. Cycle 22 will be the second cycle in which Hope Creek does not apply an added penalty to the calculated SLMCPR. This is permitted by the approvals presented in Reference 20.

5.0 References

1. Letter, Eric Carr (PSEG) to NRC Document Control Desk, "License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate," LR-N17-0044, LAR H17-03, July 7, 2017.

2. Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, Revision 24, March 2017.

3. GE Nuclear Energy, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," NEDC-32601P-A, August 1999.

4. GE Nuclear Energy, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694P-A, August 1999.

5. GE Nuclear Energy, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," NEDC-32505P-A, Revision 1, July 1999.

6. Letter, Frank Akstulewicz (NRC) to Glen A. Watford (GNF-A), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations; NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle-Specific Safety Limit MCPR (TAC Nos. M97490, M99069 and M97491)," MFN-003-099, March 11, 1999.

7. Letter, Glen A. Watford (GNF-A) to NRC Document Control Desk with attention to R. Pulsifer (NRC), "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies," FLN-2001-016, September 24, 2001.

8. Letter, Glen A. Watford (GNF-A) to NRC Document Control Desk with attention to J. Donoghue (NRC), "Confirmation of the Applicability of the GEXL14 Correlation and Associated R-Factor Methodology for Calculating SLMCPR Values in Cores Containing GE14 Fuel," FLN-2001-017, October 1, 2001.

9. Letter, Andrew A. Lingenfelter (GNF-A) to NRC Document Control Desk with cc to MC Honcharik (NRC), "GNF2 Advantage Generic Compliance with NEDE-24011P-A (GESTAR II), NEDC-33270P, March 2007, and GEXL17 Correlation for GNF2 Fuel, NEDC-33292P, March 2007," FLN-2007-011, March 14, 2007.

10. Memorandum, Michelle C. Honcharik (NRC) to Stacy L. Rosenberg (NRC), "Audit Report for Global Nuclear Fuels GNF2 Advantage Fuel Assembly Design GESTAR II Compliance Audit," September 25, 2008. (ADAMS Accession Number ML081630579)

11. Letter, Thomas H. Essig (NRC) to Glen A. Watford (GNF-A), "Acceptance for Referencing of Licensing Topical Report NEDC-32505P, Revision 1, 'R-factor Calculation Method for GE11, GE12 and GE13 Fuel,'" (TAC Nos. M99070 and M95081)," MFN-046-98, January 11, 1999.

12. Global Nuclear Fuel, "GEXL14 Correlation for GE14 Fuel," NEDC-32851P-A, Revision 5, April 2011.

13. Global Nuclear Fuel, "GEXL17 Correlation for GNF2 Fuel," NEDC-33292P, Revision 3, April 2009.

14. Letter, John F. Schardt (GNF-A) to NRC Document Control Desk with attention to Mel B. Fields (NRC), "Shadow Corrosion Effects on SLMCPR Channel Bow Uncertainty," FLN-2004-030, November 10, 2004.

15. Letter, Jason S. Post (GENE) to NRC Document Control Desk with attention to Chief, Information Management Branch, et al. (NRC), "Part 21 Final Report: Non-Conservative SLMCPR," MFN 04-108, September 29, 2004.

16. Letter, Glen A. Watford (GNF-A) to NRC Document Control Desk with attention to Joseph E. Donoghue (NRC), "Final Presentation Material for GEXL Presentation – February 11, 2002," FLN-2002-004, February 12, 2002.

17. Letter, Glen A. Watford (GNF-A) to NRC Document Control Desk with attention to Alan Wang (NRC), "NRC Technology Update – Proprietary Slides – July 31 – August 1, 2002," FLN-2002-015, October 31, 2002.

18. Letter, Jens G. Munthe Andersen (GNF-A) to NRC Document Control Desk with attention to Alan Wang (NRC), "GEXL Correlation for 10X10 Fuel," FLN-2003-005, May 31, 2003.

19. Letter, Andrew A. Lingenfelter (GNF-A) to NRC Document Control Desk with cc to MC Honcharik (NRC), "Removal of Penalty Being Applied to GE14 Critical Power Correlation for Outlet Peaked Axial Power Shapes," FLN-2007-031, September 18, 2007.

20. GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

Table 1. Monte Carlo SLMCPR

	Previous Cycle Limiting Cases		**Current Cycle Limiting Cases**	
Description	**Rated Power Minimum Core Flow**	**Rated Power Rated Core Flow**	**Rated Power Minimum Core Flow**	**Rated Power Rated Core Flow**
Limiting Cycle Exposure Point Beginning of Cycle (BOC) / Middle of Cycle (MOC) / End of Cycle (EOC)	EOC	EOC	EOC	EOC
Cycle Exposure at Limiting Point (MWd/STU)	10,000	10,000	9,600	9,600
[[				
				]]
Requested Change to the TS SLMCPR	N/A		1.09 (TLO) / 1.12 (SLO)	

Table 2. Description of Core

Description	Previous Cycle	Current Cycle
Core Rated Power (MWt)	3,840	3,902
Minimum Flow at Rated Power (% rated core flow)	94.8	97.1
Number of Bundles in the Core	764	764
Batch Sizes and Types: (Number of Bundles in the Core) Fresh Once-Burnt Twice-Burnt Thrice-Burnt or more	 212 GNF2 220 GE14 224 GE14 108 GE14	 200 GNF2 212 GNF2 220 GE14 132 GE14
Fresh Fuel Batch Average Enrichment (Weight%)	3.82	3.82
Core Monitoring System	3DMonicore and ACUMEN	ACUMEN

Table 3. Deviations from Standard Uncertainties

Description	NRC Approved Value ± σ (%)	Previous Cycle	Current Cycle
Power Distribution Uncertainties			
GEXL R-Factor	[[ ]]	[[ ]]	[[ ]]
Random Effective TIP Reading All TLO Cases at Rated Power and Minimum Flow (Non-Maximum Extended Load Line Limit Analysis Plus (MELLLA+))	1.2	1.27	1.24
Non-Power Distribution Uncertainties			
Reactor Pressure Measurement	[[ ]]	[[ ]]	[[ ]]
Channel Flow Area Variation	[[ ]]	[[ ]]	[[ ]]
Total Core Flow Measurement All TLO Cases at Rated Power and Minimum Flow (Non-MELLLA+)	2.5	2.64	2.57

Enclosure 3

GNF Affidavit in Support of Request to Withhold Information

Global Nuclear Fuel – Americas

AFFIDAVIT

I, Brian R. Moore, state as follows:

(1) I am the General Manager, Core & Fuel Engineering, Global Nuclear Fuel – Americas, LLC (GNF-A), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GNF proprietary report GNF-004N5379-R0-P, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR Hope Creek Cycle 22," dated October 2017. GNF proprietary information in GNF-004N5379-R0-P is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] GNF proprietary information in some tables is identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GNF-A customer-funded development plans and programs, resulting in potential products to GNF-A;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GNF-A.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology. The development of this methodology, along with the testing, development and approval was achieved at a significant cost to GNF-A.

The development of the fuel design and licensing methodology along with the interpretation and application of the analytical results is derived from an extensive experience database that constitutes a major GNF-A asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GNF-A.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 19th day of October 2017.

Brian R. Moore
General Manager, Core & Fuel Engineering
Global Nuclear Fuel -- Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401
Brian.Moore@ge.com