

Enclosure 6 Contains Proprietary Information to be Withheld from
Public Disclosure Pursuant to 10 CFR 2.390

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10 CFR 50.90

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LAR H20-04

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

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

Subject: **License Amendment Request: Revise Hope Creek Generating Station Low Pressure Safety Limit to Address General Electric Nuclear Energy Part-21 Safety Communication SC05-03**

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) is submitting a request for an amendment to the Technical Specifications (TS) for Hope Creek Generating Station (HCGS).

The proposed amendment will revise Hope Creek Technical Specification 2.1 SAFETY LIMITS, specifically 2.1.1, "THERMAL POWER, Low Pressure or Low Flow," and 2.1.2, "THERMAL POWER, High Pressure and High Flow," to reduce the reactor vessel steam dome pressure value. These changes will address General Electric Nuclear Energy 10 CFR Part 21 Safety Communication SC05-03 regarding the potential to violate the low pressure safety limit following a Pressure Regulator Failure - Open transient for HCGS.

Enclosures 1 through 6 include a description of the proposed changes and the supporting documentation as follows:

- Enclosure 1 Provides a description and evaluation of the proposed change that requires NRC approval, and a No Significant Hazards Consideration (NSHC) analysis.
- Enclosure 2 Provides a markup of the proposed TS changes.
- Enclosure 3 Provides a markup of the affected TS Bases pages. These marked-up pages are provided for information only and do not require NRC approval.

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- Enclosure 4 Provides a non-proprietary version of the GE Hitachi Nuclear Energy Americas LLC (GEH) report NEDO-33928 – SC05-03 Evaluation for Hope Creek Generating Station. This report provides the analytical basis supporting the requested change.
- Enclosure 5 Contains affidavits for withholding information executed by GEH.
- Enclosure 6 Provides a proprietary version of the GEH report NEDC-33928P - SC05-03 Evaluation for Hope Creek Generating Station.

PSEG requests approval of this license amendment request (LAR) in accordance with standard NRC approval process and schedule. Once approved, the amendment will be implemented within 60 days from the date of issuance.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated State of New Jersey Official.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Mr. Michael Wiwel at 856-339-7907.

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

Executed on 9/24/20
(Date)

Respectfully,



Edward T. Casulli
Vice President – Hope Creek Generating Station
PSEG Nuclear LLC

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

Enclosure 1

Evaluation of the Proposed Changes

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1.0 SUMMARY DESCRIPTION

The proposed amendment will revise the Hope Creek Technical Specification (TS) Safety Limit (SL) 2.1.1, THERMAL POWER, Low Pressure or Low Flow, and TS SL 2.1.2, THERMAL POWER, High Pressure and High Flow, and their associated Actions, to lower the low (steam dome) pressure safety limit (LPSL) to address General Electric (GE) 10 CFR Part 21 Safety Communication SC05-03 (Reference-1). This Part 21 condition identifies the potential to exceed the TS 2.1.1 Safety Limit for a brief moment of time during a Pressure Regulator Failure - Open (PRFO) plant transient condition which is an analyzed transient in Chapter 15 of the Hope Creek Generating Station (HCGS) Updated Final Safety Analysis Report (UFSAR). This brief exceedance of the LPSL does not present a challenge to the integrity of the fuel cladding as assessed by GE and recognized by the NRC in Reference 3. Technical justification for lowering the Hope Creek LPSL is contained in General Electric-Hitachi (GEH) report "SC05-03 Evaluation for Hope Creek Generating Station," which is included, in both non-proprietary (NEDO-33928) and proprietary (NEDC-33928P) versions, in Enclosures 4 and 6 respectively of this License Amendment Request (LAR). Note that the enclosed GEH report refers to the LPSL as the Thermal Power Safety Limit Pressure Boundary (TPSLPB). The two terms describe the same parameter and are interchangeable.

2.0 DETAILED DESCRIPTION

In 2005, GE Nuclear Energy issued 10 CFR Part 21 Safety Communication SC05-03 identifying the potential vulnerability for the PRFO transient event to result in a condition in which the TS 2.1.1 Safety Limit may be momentarily exceeded. This potential condition does not challenge the fuel cladding integrity or constitute a safety hazard; however, there exists a potential for violation of a TS SL as a result of a PRFO event. To address this condition, PSEG proposes to revise the reactor vessel steam dome pressure specified in TS SL 2.1.1 and SL 2.1.2, based on the HCGS-specific GEH analysis that justifies the use of the GE critical quality – boiling length (GEXL)17 and GEXL14 correlations, and confirms that HCGS will avoid exceeding the TS 2.1.1 Safety Limit with the proposed TS LPSL during a PRFO event.

2.1 Current Technical Specification Requirements

The applicable current Hope Creek TS SAFETY LIMITS are as follows:

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:

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be ≥ 1.07 .

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 value 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.

2.2 Reason for Proposed Change

The proposed change to TS SL 2.1.1 and SL 2.1.2 to reduce the vessel steam dome pressure value (i.e. the LPSL) is necessary to address GE Part 21 Safety Communication SC05-03 (Reference 1) that identifies a potential to exceed the low pressure technical specification safety limit in response to the PRFO transient, which is one of the analyzed transients in Chapter 15 of the HCGS UFSAR.

The PRFO transient analysis performed by GE for UFSAR Chapter 15 during initial and subsequent licensing evaluations described the event being terminated by a turbine trip and associated reactor scram due to high reactor level caused by void induced swell from the depressurization of the RPV through the full open turbine control and bypass valves. Subsequent analyses by GEH using enhanced computer models identified the potential for a less severe swell response from the PRFO depressurization and the event ultimately being terminated by closure of the main steam isolation valves (MSIVs) based on low steam line pressure and a reactor scram from MSIV position switch logic. Depending on specific plant design characteristics such as main steam line pressure drop, depressurization to the low main steam line pressure isolation setpoint (LPIS) may result in a corresponding reactor steam dome pressure falling below the LPSL while still above the thermal power safety limit (TPSL). This potential to violate a TS SL during a PRFO transient led GE to issue the Reference 1 Part 21 Safety Communication.

In response to SC05-03, the Boiling Water Reactor Owners Group (BWROG) developed a methodology for plants to assess the adequacy of their current LPIS setting and to provide a set of recommendations for what actions could be taken based on the outcome of their assessment including lowering the LPSL and raising the LPIS. The methodology and recommendations are documented in the Reference 2 GEH-BWROG report. The methodology was developed by analyzing a limiting plant, assessing uncertainties, and determining a method to conservatively scale the limiting plant's results to other plant configurations and operating flexibility options through sensitivity studies.

Based on the results of the studies documented in Reference 2, PSEG evaluated lowering the LPSL to 685 psig as recommended in Reference 2 and raising the current Hope Creek LPIS to preclude reactor vessel steam dome pressure from falling below the LPSL while above the TPSL of 24% power for current operation during a PRFO event. PSEG concluded that increasing the Hope Creek LPIS to a level that would maintain steam dome pressure above the reduced LPSL recommended in Reference 2, will result in an increased likelihood for a plant scram to be complicated by an automatic closure of the main steam isolation valves (MSIVs) following the scram. The potential for main steam isolation is dependent on the nature of the scram and the response time of the operator to bypass the automatic MSIV closure logic after the scram via the reactor mode switch. This conclusion was reached based on scram modeling of increased LPIS settings performed in the Hope Creek simulator and is supported by the PRFO transient modeling documented in the enclosed GEH Report. Isolation of the reactor from the normal heat sink would unnecessarily complicate post-scram actions for the reactor operator resulting in a challenge to overall reactor safety. Therefore, raising the LPIS is not a suitable option for HCGS to address a condition that does not threaten fuel clad integrity, as acknowledged by the NRC in the Reference 3 letter.

The subject Part 21 condition, although not considered a safety hazard, does pose a regulatory risk in the event of a PRFO event at HCGS. Based on the options available to PSEG to resolve the issue, lowering of the LPSL in the Hope Creek TS provides the least challenge to plant operators and systems while permanently addressing the Reference 1 GE Part 21 Safety Communication SC05-03.

2.3 Description of Proposed Change

The proposed changes to Hope Creek TS are described below and are indicated on the marked up TS page provided in Enclosure 2. Deletions are indicated with a strike through and additions are marked in double underlines.

Revise TS 2.1.1 and 2.1.2 to lower the LPSL from 785 psig to a value of 585 psig.

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~~ 585 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~~ 585 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~~ 585 psig and core flow greater than 10% of rated flow:

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be ≥ 1.07 .

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With reactor steam dome pressure greater than ~~785~~ 585 psig or core flow greater than 10% of rated flow and the MCPR below the value 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.

Enclosure 3 includes a markup of TS Bases changes for information only.

3.0 TECHNICAL EVALUATION

3.1 Overview of Low Pressure Safety Limit

The TS SLs are established to protect the integrity of principal barriers to the release of radioactive materials to the environment, specifically the fuel cladding, reactor pressure vessel (RPV) and primary system piping. The SAFETY LIMITS associated with protecting the fuel cladding are set such that no fuel damage is calculated to occur if the limit is not violated. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Fuel cladding perforations can result from thermal stresses which occur from reactor operation significantly above design conditions and Limiting Safety System Settings. Thermally caused cladding perforations signal a threshold beyond which still greater thermal stress may cause gross rather than incremental cladding deterioration. The fuel clad Safety Limit Minimum Critical Power Ratio (SLMCPR) is defined, via NRC approved correlations, to represent a limit during a significant departure from the condition intended by design for planned operation, where transition boiling will not occur. The associated SLs impacted by this proposal and their intended purpose are as follows:

- TS Safety Limit 2.1.1 for THERMAL POWER, Low Pressure or Low Flow

The NRC approved critical power correlations are established with NRC reviewed limits of applicability on reactor pressure and core flow. For the fuel product lines in use at HCGS, these limits are conservatively established at 785 psig and 10% core flow. The thermal power safety limit (TPSL) of 24% RATED THERMAL POWER is conservatively established to ensure that bundle power is guaranteed to be below critical power based on full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia including applicable scaling for station power uprates.

- Safety Limit 2.1.2 for THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity safety limit is set such that no significant fuel damage is calculated to occur as long as the limit is not violated. The thermal and hydraulic conditions resulting in the onset of transition boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power ratio has been adopted as a convenient limit for protecting the fuel clad integrity. The TS SLMCPR is analytically determined based on MCPR correlations established from empirical test data (i.e. GEXL correlations) for the fuel product lines used at HCGS. Hope Creek TS conservatively apply a minimum reactor pressure of 785 psig to the applicable MCPR correlations for the fuel assemblies in use at HCGS.

3.2 Basis for Lowering Hope Creek LPSL

The current HCGS value of 785 psig in TS SL 2.1.1 and 2.1.2 corresponds to the range of pressures over which the GE critical power (GEXL) correlation was originally tested for the legacy fuel lines used at HCGS. Despite the lower pressure CPR testing performed for the GE14 and GNF2 fuel product lines currently in use at HCGS, the current TS LPSL was conservatively left in place. In Reference 2, GE stated that licensees could consider a change to their TS SL to align with a lower pressure GEXL correlation range in addressing the compliance issue outlined in SC 05-03. This approach takes advantage of the fact that more recent critical power correlations have been tested over a wider range of pressure. The most recent NRC-approved GNF GEXL correlation (GEXL21) has been tested to pressure values well below the current Hope Creek TS SL value of 785 psig (800 psia) used for initial licensing of HCGS.

PSEG is proposing to lower the reactor vessel steam dome pressure specified in TS SL 2.1.1 and SL 2.1.2 due to the risks associated with increasing Hope Creek's LPIS to an adequate value to address Safety Communication SC05-03. The proposed LPSL is based on the conclusions in the enclosed GEH Report (Enclosures 4 (non-proprietary, NEDO-33928) and 6 (proprietary, NEDC-33928P) of this application) that was developed specifically for HCGS to address Safety Communication SC05-03. The GEH report evaluates the ability of the GEXL14 and GEXL17 correlations (established for the GE14 and GNF2 fuel product lines respectively) to adequately predict newer GNF3 critical power (CP) test data at lower pressures. This ability to predict the low pressure GNF3 data demonstrates the ability to extrapolate the GEXL14 and GEXL17 correlations to the pressure proposed for the LPSL. The GNF3 data was utilized in a manner that excluded the effects of the GNF3 specific critical power performance characteristics.

The enclosed analysis shows that GEXL17 and GEXL14 can predict the pressure trend of the GNF3 critical power data with acceptable accuracy. HCGS currently operates a mixed core comprised of GE14 and GNF2 fuel lines. Considering the conservatism used in the GEH analysis and the margins available to the operating limit MCPR, the results of the GEH analysis support the proposed change to the SL for the fuel lines in use at HCGS.

The report also compares the measured pressure trend of Critical Power (the power where transition boiling is projected to occur within the fuel assembly) for the GE14, GNF2, and GNF3 fuel lines. The enclosed GE analysis shows that the three fuel product lines have similar trends of increasing CP as reactor pressure decreases. It can therefore be concluded that the GEXL14 and GEXL17 correlations that were empirically established for the GE14 and GNF2

fuel product lines respectively would have similar trends of increasing CP versus decreasing reactor pressure as the empirically backed GEXL21 correlation for GNF3, including reactor pressures down through 585 psig. MCPR inherently rises as CP increases when reactor pressure (and power) decreases. Therefore, margin to the SLMCPR, that the LPSL is established to protect, increases as reactor steam dome pressure decreases in response to a PRFO.

Based on GE's assessment of the consistent CP versus pressure trend established across multiple fuel lines and the accuracy of the GEXL14 and GEXL17 correlations in predicting the established GNF3 CP data, a reduction in the TS low pressure safety limit from 785 psig to 585 psig for the fuel product lines in use at HCGS is appropriate.

3.3 Assessment of PRFO Event for Hope Creek Generating Station

The Reference 1 GE Safety Communication describes the potential for a licensee to violate TS SL 2.1.1 in response to a PRFO event. The PRFO event can potentially cause the reactor pressure to decrease below the Hope Creek LPSL value of 785 psig while reactor power is above the 24% rated thermal power safety limit (TPSL). GE identified that plants with an MSIV low pressure isolation setpoint less than 785 psig may experience a PRFO event that could potentially violate the LPSL. The PRFO event and depressurization transients in general are non-limiting for fuel cladding integrity because CPR increases during a depressurization as discussed above in Section 3.2. Therefore, even though the PRFO event could result in the plant entering a condition that will challenge the LPSL, there is no safety concern with this condition.

The redundant pressure regulators within the turbine electro-hydraulic control system ensure that reactor steam dome pressure is maintained well above the LPSL during normal power operation, therefore, a challenge to the LPSL during power operation can only occur as a result of a plant transient. The enclosed GEH report assessed all anticipated operational occurrences (AOO's) analyzed in Chapter 15 of the Hope Creek Updated Final Safety Analysis Report (UFSAR). The evaluation concluded that the PRFO event is the only AOO that can credibly challenge the LPSL. The report also documents the results of a HCGS-specific analysis of the PRFO event under limiting conditions. The analysis concludes that the existing LPIS for Hope Creek can potentially result in a condition where the steam dome pressure momentarily falls below either the current HCGS LPSL, or the lower 685 psig LPSL recommended as an option in Reference 2, with power above the 24% TPSL prior to the PRFO event being terminated by MSIV closure and associated scram.

Increasing the Hope Creek LPIS to a level that would maintain steam dome pressure above either the current TS LPSL, or the reduced LPSL recommended in Reference 2, will result in a potential increase in scrams with complications due to an automatic closure of MSIVs. The potential for MSIV closure and subsequent loss of the normal heat sink is dependent on the nature of the scram and the response time of the operator to bypass the MSIV logic via the reactor mode switch. This conclusion was reached based on scram modeling of increased in-plant LPIS settings performed in the Hope Creek simulator and is supported by the PRFO transient modeling documented in the enclosed GEH Report.

Given the limited potential to raise the LPIS to a suitable value without complications, PSEG is proposing to lower the LPSL to 585 psig (600 psia) to ensure the safety limit is not exceeded in response to the limiting PRFO event. As part of the enclosed report, GEH performed PRFO transient modeling using TRACG04 which has been approved by the NRC for transient

simulation. The GEH analysis conservatively took into account uncertainties in the transient modeling to make the analysis cycle independent. The analysis confirmed that lowering the LPSL to 600 psia (585 psig) provides acceptable reactor steam dome pressure margin such that the LPSL is not violated in response to a PRFO event.

A non-proprietary version of GEH report (NEDO-33928) is included as Enclosure 4 of this LAR and a proprietary version (NEDC-33928P) is included as Enclosure 6. An affidavit for withholding the proprietary version from public disclosure is included as Enclosure 5.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36(c) provides that TS will include Safety Limits which are “the limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.” The proposed reduction in the Low Pressure Safety Limit for TS 2.1.1, Low Pressure-Low Flow, and TS 2.1.2, High Pressure-High Flow, establishes the reactor core safety limits that will continue to protect the integrity of the fuel cladding barrier and guard against an uncontrolled release of radioactivity. Therefore, the proposed changes are consistent with current regulations.

Although not the direct subject matter of this requested amendment, the following 10 CFR 50, Appendix A, General Design Criteria apply to the safety limit impacted by the proposed change in this amendment application:

CRITERION 10 – REACTOR DESIGN

“The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.”

4.2 Precedents

Most BWR licensees have submitted LARs to lower the LPSL and/or raise the LPIS to address the Reference 1 Safety Communication based on each plant’s specific design and licensing basis. The precedent identified below is similar to the approach proposed by PSEG in this submittal.

1. Letter from F. Saba NRC to J. Shea Tennessee Valley Authority, December 16, 2015, Issuance of Amendments Regarding Technical Specification Changes to Reactor Core Safety Limits (CAC Nos. MF5412, MF5413 AND MF5414 (ADAMS Accession Number ML15287A213)

4.3 No Significant Hazards Consideration

The Hope Creek Generating Station Technical Specification (TS) SAFETY LIMIT (SL) 2.1.1 for Low Pressure or Low Flow operating conditions and SL 2.1.2 for High Pressure and High Flow operating conditions are revised to lower the Low Pressure Safety Limit (LPSL). The proposed change addresses General Electric (GE) Part 21 Safety Communication SC05-03 regarding

potential violation of the low pressure safety limit following a Pressure Regulator Failure - Open (PRFO) analyzed event.

PSEG has evaluated the proposed changes to the TS using the criteria in 10 CFR 50.92, and determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards:

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

Response: No

The proposed change to the TS will not alter the way any structure, system, or component (SSC) functions, and will not alter the manner in which the plant is operated. The proposed change does not alter the design of any SSC. Since no design or operational changes are being made to any SSC, the proposed TS change does not change the accident initiation capability of any plant system. Therefore, the probability of an accident previously evaluated is not significantly increased.

The proposed change has no physical or operational impact to any Reactor Protection System (RPS) or Emergency Core Cooling System (ECCS) trip or initiation function or associated setpoints. There are no changes to any accidents or transients analyzed in the Hope Creek Safety Analyses; therefore the consequences of an accident previously evaluated are not increased.

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

Response: No

The proposed change does not involve a modification to the physical configuration of the plant or a change in the methods governing normal plant operation. The proposed change does not impose any new or different requirements or introduce a new accident initiator, accident precursor, or mechanism for equipment malfunction.

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

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

Response: No

The proposed change to the Hope Creek TS will not result in changes to system design or setpoints that are credited in the safety analyses. The proposed change does not impact systems or indications intended to ensure timely identification of plant conditions that could be precursors to accidents or potential degradation of accident mitigation systems.

The proposed amendment reduces the TS low pressure safety limit based on the capability of the GEXL17 and GEXL14 critical power correlations to accurately predict critical power at lower pressures. The proposed change is within the capabilities of the critical power correlations applicable to fuel product lines currently in use. Therefore, since the proposed

change does not impact the analyzed response of the plant to a design basis accident and is analytically supported for the fuel in use at Hope Creek, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, PSEG 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.

4.4 Conclusions

Based on the considerations discussed above, (1) there is 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 Commission'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.

5.0 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.

6.0 REFERENCES

1. General Electric Part-21 Safety Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit", March 29, 2005 (ADAMS Accession No. ML050950428).
2. GE-Hitachi Nuclear Energy (GEH) Report NEDC-33743P, "BWR Owners' Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," Revision 0, dated April 2012.
3. Letter from T.J. Kobetz to the Technical Specification Task Force, Denial of TSTF-495. Revision 0, "Bases change to address GE Part 21 SC05-03." Docket No: PROJ0753 (TAC MD2672) (ADAMS Accession No. ML072340113).

Enclosure 2

Mark-up of Proposed Technical Specification Pages

The following Technical Specifications page for Renewed Facility Operating License NPF-57 is affected by this change request:

<u>Technical Specification</u>	<u>Page</u>
SAFETY LIMIT 2.1.1	2-1
SAFETY LIMIT 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:

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be ≥ 1.07 .

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 value 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.

Enclosure 3

**Mark-up of Proposed Technical Specification Bases Pages
For Information Only**

The following Technical Specifications Bases pages for Renewed Facility Operating License NPF-57 are affected by this change request:

<u>Technical Specification Bases</u>	<u>Page</u>
2.1.1	B2-1
2.1.1	B2-2

2.1 SAFETY LIMITS

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 \geq the limit specified in Specification 2.1.2. 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. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as SLMCPR_{95/95}, which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur.

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 28×10^3 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 28×10^3 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.

585

585

Reference 2 shows that the critical power correlations established for the fuel designs currently in use at Hope Creek may be considered valid for pressures down to 600 psia.

SAFETY LIMITS

BASES

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in the onset of transition boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that the onset of transition boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification Safety Limit value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with a 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e. most limiting) of the MCPR values for the fuel product lines that are fresh or once burnt at the start of the cycle.

Reference:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)

2. GE-Hitachi Nuclear Energy Report NEDC-33928P, SC5-03 Evaluation for Hope Creek Generating Station, September 2020

**LR-N20-0023
Enclosure 4**

LAR H20-04

Enclosure 4

**GE-Hitachi Nuclear Energy (GEH) Report NEDO-33928
SC05-03 Evaluation for Hope Creek Generating Station, Revision 0
(Non-Proprietary Version)**



HITACHI

GE Hitachi Nuclear Energy

NEDO-33928
Revision 0
September 2020

Non - Proprietary Information

SC05-03 Evaluation for Hope Creek Generating Station

PROPRIETARY INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33928P, Revision 0, which has the proprietary information removed. Portions of the document that have been removed are indicated by open and closed brackets as shown here [[]].

**IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT**

Please Read Carefully

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GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, **Michelle P. Catts**, state as follows:

- (1) I am the Senior Vice President of Nuclear Programs, GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.
- (2) The information sought to be withheld is contained in GEH proprietary report NEDC-33928P, "SC05-03 Evaluation for Hope Creek Generating Station," Revision 0, dated September 2020. GEH proprietary information in NEDC-33928P Revision 0 is identified by a dotted underline inside double square brackets. [[This sentence is an example.^{3}]]. GEH proprietary information in figures and large objects is identified by double square brackets before and after the object. In each case, the superscript notation ^{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.
- (3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. §552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. §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 qualifies 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 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).
- (4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:
 - a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without a license from GEH constitutes a competitive economic advantage over other companies;
 - b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;
 - c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;
 - d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

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- (5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions for proprietary or confidentiality agreements or both that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).
- (6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH.
- (7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.
- (8) The information identified in paragraph (2) is classified as proprietary because it contains the detailed GEH methodology for analyzing and applying GEXL correlations to determine appropriate turbine low-pressure setpoint requirements for the GEH Boiling Water Reactor (BWR). These methods, techniques, and data along with their application to the design, modification, and analyses associated with the setpoint requirements were achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience databases that constitute a major GEH asset.

- (9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply

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the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

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

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

Executed on this 18th day of September 2020.

Michelle P. Catts

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NEDO-33928 Revision 0
Non - Proprietary Information

REVISION SUMMARY

Revision	Section	Revision Summary
0	-	Initial release.

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1.0 INTRODUCTION AND BACKGROUND

Boiling Water Reactor (BWR) Owners' Group (BWROG) safety communication (SC) 05-03 (Reference 1) discusses the effect of having a low pressure isolation setpoint (LPIS), as sensed at the turbine inlet, less than the technical specification (TS) thermal power safety limit pressure boundary (TPSLPB). Evaluations of the pressure regulator failure open (PRFO) event demonstrated that the decrease in the reactor steam dome pressure may result in the condition in which the thermal power safety limit (TPSL) for low pressure or low core flow may be exceeded depending on the plant's LPIS. The PRFO event and depressurization transients in general are non-limiting for fuel cladding integrity because critical power ratio (CPR) increases during a depressurization. Therefore, even though the PRFO event may result in the plant entering a condition below their allowed TPSLPB, there is no safety concern with this condition.

The BWROG SC05-03 analysis report (Reference 2) presents a methodology for plants to assess their current LPIS setting to determine if action needs to be taken to ensure compliance with the TPSLPB. PSEG identified that increasing the LPIS, as recommended by the BWROG SC05-03 analysis report, runs the risk of inadvertent isolation following a reactor scram. Therefore, PSEG requested GEH perform a technical evaluation supporting lowering the TPSLPB in TS 2.1.1 and 2.1.2 from 785 psig (Reference 3) to 585 psig for Hope Creek Generating Station (HCGS) to support a License Amendment Request (LAR).

This evaluation includes three elements:

- An evaluation of the GEXL correlation to justify reduction of the TPSLPB to 585 psig (600 psia) for HCGS.
- An evaluation of normal operations and anticipated operational occurrences (AOOs) to confirm that the PRFO event is limiting with respect to challenge to the TS TPSL for low pressure or low core flow.
- An evaluation of the PRFO event to confirm that HCGS will avoid exceeding the TS TPSL for low pressure or low core flow with a TPSLPB of 585 psig.

2.0 DESIGN INPUTS AND ASSUMPTIONS

Table 2-1 Design Inputs

Item	Design Input
1	Current TPSLPB: 785 psig (800 psia) (Reference 3)
2	Approved GEXL dome pressure lower limit for GE14 and GNF2 fuel: [[]]
3	Proposed TPSLPB: 585 psig (600 psia)
4	Maximum combined steam flow available (MCFA): 126%
5	Maximum combined flow limiter setpoint (MCFL): 130%
6	LPIS analytical limit: 720 psig (735 psia)
7	Steamline pressure drop: 64.6 psid
8	Maximum main steam isolation valve (MSIV) closure time: 5.0 seconds
9	Thermal power scram time constant: 6.6 seconds
10	TPSL low pressure or low flow: 24.0% rated power (Reference 3)
11	High-water level (L8) setpoint: 586.50 inches above vessel zero
12	Licensed feedwater temperature reduction (FWTR): 102°F below rated temperature

Table 2-2: Assumptions

Item	Assumption
1	[[]]
2	The simulated thermal power (STP) is an acceptable model of the core power. The thermal power time constant is provided in Design Input Item 9. The STP time constant used is [[]]. Therefore, use of the STP time constant is conservative for the purpose of the PRFO analysis.

3.0 GEXL CORRELATION EVALUATION

In this Section, the pressure trend of the GEXL17 and GEXL14 correlations is justified down to 600 psia by:

- 1) exercising the GEXL17 and GEXL14 correlations against GNF3 data and
- 2) reviewing the critical power trend of GE14, GNF2, and GNF3.

The low bound of the GEXL17 and GEXL14 pressure application range is [[
]] (References 4 and 5). Full-scale critical power data were collected at [[
]] and the GEXL21 correlation for GNF3 fuel extended the pressure application range [[
]] (Reference 6). For each applicable GNF3 test point, the measured critical power and the corresponding test conditions (axial power shapes, radial peaking distributions, mass flow, and inlet subcooling) are employed in the analysis. Adequacy of GEXL17 and GEXL14 pressure trends between [[
]] were confirmed by virtue of good statistics against full-scale critical power data in References 4 and 5. The present evaluation is to show the GEXL correlation trend with pressure down to 600 psia with GNF3 data. However, GEXL17 and GEXL14 correlations are not expected to accurately calculate the critical power of GNF3 fuel. [[

]]

3.1 GEXL17 and GEXL14 Correlation Applicability to Low Pressure Data

The purpose of this analysis is to evaluate the performance of the GEXL17 or GEXL14 correlation against GNF3 test data in the lower pressure range of 600 psia. Therefore, the selected GNF3 test data points include test points with low pressures down to 600 psia. In addition to the test points [[

]]

3.1.1 GEXL17 Correlation

Figure 3-1 and Figure 3-2 show the GEXL17 ECPR and comparison between the GEXL17 simulated critical power and the measured data, respectively, for GNF3 Stern Test Assembly (STA) as a function of pressure. The average ECPR values and the corresponding standard deviations at different pressures are given in Table 3-1. The purpose of such comparisons is to validate the GEXL17 prediction for low pressure down to 600 psia with the GNF3 Stern test data and to demonstrate the [[

]] (Reference 5).

Table 3-1: Statistics of GEXL17 Prediction for GNF3 Data at Different Pressures

Pressure (psia)	Mean ECPR	Standard Deviation	Number of Test Points
[[
]]

[[

]]

Figure 3-1: GEXL17 Pressure Trend with the GNF3 Test Data

[[

]]

Figure 3-2: GEXL17 Predicted Critical Power Versus GNF3 Test Data

3.1.2 GEXL14 Correlation

The GEXL14 correlation pressure trend for GNF3 test data in terms of ECPR is shown in Figure 3-3. The comparison between the GEXL14 simulated critical power and the measured data for GNF3 STA is demonstrated in Figure 3-4 as a function of pressure. The average ECPR values and the corresponding standard deviations for GEXL14 at different pressures are given in Table 3-2. Again, the purpose of such comparisons is to validate the GEXL14 prediction for low pressure down to 600 psia with the GNF3 Stern test data and to demonstrate the [[

]] However, HCGS's current cycle (cycle 23) is loaded with 618 GNF2 bundles and 146 GE14 bundles (loaded cycles 19 and 20) for a total of 764 fuel bundles. The GE14 bundles at the current cycle (cycle 23) are 4th and 5th cycle bundles located on the periphery with high exposure and large margin to the operating limit minimum critical power ratio (OLMCPR). Furthermore, all GE14 fuels are most likely to be discharged in the next cycle (cycle 24) or reside on the periphery or other low power locations, if any remains. Hence, the GE14 fuel bundles at the current cycle or future cycles will have significant margin preventing them from being limiting.

Table 3-2: Statistics of GEXL14 Prediction for GNF3 Data at Different Pressures

Pressure (psia)	Mean ECPR	Standard Deviation	Number of Test Points
[[
]]

[[

]]

Figure 3-3: GEXL14 pressure trend with the GNF3 test data

[[

]]

Figure 3-4: GEXL14 Predicted Critical Power Versus GNF3 Test Data

3.2 GNF2, GE14, and GNF3 Critical Power Trends with Pressure

The pressure trends of measured critical powers for GNF2, GE14, and GNF3 were investigated as well. Figure 3-5 and Figure 3-6 show the measured critical power as a function of pressure for GNF2 and GE14, respectively. Here, G represents mass flux in Mlbm/hr-ft^2 and H represents inlet subcooling in BTU/lbm . As shown, the critical power increases as pressure decreases. The available GNF3, GNF2, and GE14 data trends with pressure are compared in Figure 3-7. Clearly, these trends exhibit aligned and consistent behavior with respect to decreasing pressure.

In Section 3.1, it is shown that GEXL17 and GEXL14 can predict the pressure trend of the GNF3 data. Considering that GNF2, GE14, and GNF3 critical power trends with pressure are similar, it can be concluded that GEXL17 can predict GNF2 critical power down to 600 psia and similarly that GEXL14 can predict GE14 critical power down to 600 psia.

[[

]]

Figure 3-5: Measured Critical Power as a Function of Pressure for GNF2

[[

]]

Figure 3-6: Measured Critical Power as a Function of Pressure for GE14

[[

]]

Figure 3-7: Measured Critical Power as a Function of Pressure for GNF3, GNF2, and GE14

3.3 Summary

The accuracy of the GEXL17 and GEXL14 correlation down to 600 psia is evaluated using GNF3 low pressure data. In general, both the GEXL17 and GEXL14 can predict the pressure trend of the GNF3 data down to 600 psia. For selected GNF3 test data, [[

]] The GEXL14 slightly overpredicts GNF3 data at 600 psia compared to the overall GEXL14 statistics. However, the remaining GE14 bundles at the current cycle (cycle 23) of HCGS are 4th and 5th cycle bundles located on the periphery with high exposure and large margin to the OLMCPR. Furthermore, all GE14 fuels are most likely to be discharged in the next cycle (cycle 24) or reside on the periphery or other low power locations, if any remains. Therefore, the GE14 fuel bundles at the current cycle or future cycles will have significant margin preventing them from being limiting. It is also shown that GNF2, GE14, and GNF3 critical power trends with pressure are similar. Therefore, it is concluded that GEXL17 can predict the critical power of GNF2 bundles down to 600 psia with [[

]] Similarly, it is concluded that GEXL14 can predict the critical power of GE14 bundles down to 600 psia [[

]]

4.0 EVALUATION OF NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCURRENCES

Normal operation and AOOs are evaluated below to confirm that the PRFO AOO event is the limiting event that could challenge the TS TPSLPB.

4.1 Normal Operation

During reactor startup, normal pressure control is established via the main turbine Electro-Hydraulic Control (EHC) system prior to power reaching the TPSL. Once established, three identical pressure regulators within EHC are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and use the pressure to control the position of the TCVs. The pressure is controlled well above the LPIS and the TPSLPB. With the pressure regulator system operating properly there is no possibility that pressure would reduce below the TPSLPB.

Any challenge to the TPSLPB would have to come from an AOO initiated from pressure conditions consistent with normal pressure control.

4.2 AOOs

Table 4-1 includes an evaluation of the AOO events contained in the HCGS Updated Final Safety Analysis Report (UFSAR) Chapter 15 (Reference 7). Accidents contained in UFSAR Chapter 15 are not dispositioned because TS safety limits do not apply.

The evaluation concludes that the PRFO event is not only the limiting event but the only AOO that can credibly challenge the TPSLPB.

Table 4-1: Evaluation of UFSAR Chapter 15 AOO Events

UFSAR Section	Event	Event Evaluation
15.1.1	Loss of Feedwater Heating	<p>The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB.</p> <p>It is possible the event could result in a scram on high neutron flux or STP. In this case, there is no concern with pressure approaching the TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.</p>
15.1.2	Feedwater Controller Failure – Maximum Demand	<p>The pressure controller continues to operate normally during the event. The increase in feedwater flow results in an increase in reactor level and a decrease in downcomer / lower plenum enthalpy. This results in an increase in reactor power and a slight increase in reactor pressure. The reactor water level increases until there is a trip on high-water level. This results in a feedwater trip and a turbine trip. The turbine trip results in a scram from turbine stop valve (TSV) closed position. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.</p>
15.1.3	Pressure Regulator Failure – Open	Limiting event with respect to approaching TPSLPB with high reactor power.
15.1.4	Inadvertent Main Steam Relief Valve Opening	<p>This event would result in a small decrease in reactor pressure because a small fraction of steam flow is diverted from the main steam line which decreases the main steam line pressure drop. The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB.</p>
15.1.5	Spectrum of Steam System Piping Failures Inside and Outside of Containment in a Pressurized Water Reactor	Not applicable to BWRs.
15.1.6	Inadvertent Residual Heat Removal (RHR) Shutdown Cooling Operation	RHR cannot inject at pressures above (or near) TPSLPB; therefore, the event is not possible with pressure above TPSLPB.

Table 4-1: Evaluation of UFSAR Chapter 15 AOO Events

UFSAR Section	Event	Event Evaluation
15.2.1	Pressure Regulator Failure – Closed	This event results in a small increase in reactor pressure and power. An independent pressure regulator is expected to take over pressure control at a new steady state well above TPSLPB.
15.2.2	Generator Load Rejection	This event results in a fast closure of the TCVs and automatic scram. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.
15.2.3	Turbine Trip	This event results in a fast closure of TSVs and automatic scram. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.
15.2.4	Main Steam Isolation Valve Closures	This event results in an automatic scram on MSIV position. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly. A closure of a single MSIV at off rated conditions may not result in an automatic scram; however, the pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB.
15.2.5	Loss of Condenser Vacuum	This event results in a fast closure of TSVs and automatic scram. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.
15.2.6	Loss of Alternating Current Power	This event results in a fast closure of the TCVs and automatic scram. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.

Table 4-1: Evaluation of UFSAR Chapter 15 AOO Events

UFSAR Section	Event	Event Evaluation
15.2.7	Loss of Feedwater Flow	<p>This event results in a reduction of reactor water level and an increase in the downcomer enthalpy. This results in a reduction of reactor power and a slight reduction in reactor pressure. A scram is initiated when the reactor water level reaches low level (Level 3). During the event the pressure regulator continues to operate normally. There is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.</p>
15.2.9	Failure of RHR Shutdown Cooling	<p>RHR shutdown cooling is used when the reactor is shutdown. The TPSL is not applicable.</p>
15.3.1	Reactor Recirculation Pump Trip	<p>A trip of one reactor recirculation pump results in a new steady state at a reduce reactor power, pressure, and core flow. The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB.</p> <p>A trip of two reactor recirculation pumps may result in a scram on high reactor water level.</p> <p>If there is no scram, the event results in a new steady state at a reduced reactor power, pressure, and core flow. The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB</p> <p>If there is a scram then there is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.</p>
15.3.2	Recirculation Flow Control Failure – Decreasing Flow	<p>A Recirculation Flow Control Failure – Decreasing Flow may result in a scram on high reactor water level.</p> <p>If there is no scram, the event results in a new steady state at a reduced reactor power, pressure, and core flow. The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB</p> <p>If there is a scram then there is no concern with pressure approaching the TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.</p>

Table 4-1: Evaluation of UFSAR Chapter 15 AOO Events

UFSAR Section	Event	Event Evaluation
15.4.1	Rod Withdrawal Error – Low Power	These events occur well below 24% power; therefore, the pressure may be above or below the TPSLPB.
15.4.2	Rod Withdrawal Error – At Power	The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB.
15.4.3	Control Rod Maloperation (System Malfunction or Operator Error)	Covered by 15.4.1 and 15.4.2.
15.4.4	Abnormal Startup of Idle Recirculation Pump	<p>This event results in a rapid increase in core flow. This results in an increase in core power and pressure. Severe events will scram on high Average Power Range Monitor (APRM) neutron flux; therefore, there is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.</p> <p>Less limiting instances of this event could avoid a high APRM neutron flux scram. In these cases, the normal pressure controller will continue to operate and maintain the reactor pressure well above the TPSLPB.</p>
15.4.5	Recirculation Flow Control Failure with Increasing Flow	<p>This event results in a rapid increase in core flow. This results in an increase in core power and pressure. Depending on the initial conditions and severity of the core flow increase a scram may or may not occur on high APRM neutron flux. If the scram occurs there is no concern with pressure approaching TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly. If the scram does not occur the normal pressure controller will continue to operate and maintain the reactor pressure well above the TPSLPB.</p>
15.4.6	Chemical and Volume Control System Malfunctions	Not applicable to BWRs.
15.4.7	Misplaced Bundle Accident	The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB.
15.4.8	Spectrum of Rod Ejection Accidents	Not applicable to BWRs.

Table 4-1: Evaluation of UFSAR Chapter 15 AOO Events

UFSAR Section	Event	Event Evaluation
15.5.1	Inadvertent High Pressure Coolant Injection Startup	Results in a small reduction in pressure due to a decrease in steam flow; however, the pressure controller continues to control pressure at a new steady state well above the TPSPB. There is potential that inadvertent high pressure coolant injection results in a scram on high-water level. In this case there is no concern with pressure approaching TPSPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.
15.5.2	Chemical Volume Control System Malfunction (or Operator Error)	Not applicable to BWRs.
15.5.3	Increase In Reactor Coolant Inventory BWR Transients	Covered in Section 15.1 and 15.2.
15.6.1	Inadvertent Safety/Relief Valve Opening	See 15.1.4.
15.6.3	Steam Generator Tube Failure	Not applicable to BWRs.

5.0 PRFO EVENT EVALUATION

The PRFO event is modeled in order to ensure the proposed TPSLPB (Design Input Item 3) is not violated.

5.1 Sequence of Events

The following is a brief event sequence description for the PRFO event. All three pressure regulators fail open, resulting in a signal for maximum steam demand, limited by the MCFL value. This signals the TCVs and turbine bypass valves (TBVs) to open at the servo speed to meet the demand. With the opening of the valves, the steam flow increases resulting in dome and turbine pressure decrease, and a water level swell occurs in the reactor as a result of the increased steam void production. As this occurs, either the LPIS for the turbine is reached or the high-water level (L8) setpoint is reached. If the high-water level (L8) setpoint is reached first this results in a turbine trip and scram. If the LPIS is reached first, this results in a MSIV closure. When the MSIV position switch setpoint is reached a scram is signaled. The dome pressure begins to rebound as a result of the MSIV closure.

5.2 Analysis Scope

The analysis is performed with TRACG04, which is approved for transient simulation in References 8 and 9. The PRFO event modeling considers multiple parameters to ensure the bounding behavior is captured. The use of TRACG04 for this evaluation is not to determine an OLMCPR value, but to demonstrate that the dome pressure during the PRFO event remains above the proposed lower TPSLPB. The methods to address uncertainties developed in Reference 8 do not apply and are not used in the PRFO analysis because the objective is different than that of the methods in Reference 8. The following PRFO analysis basis is used to address event uncertainty and ensure the analysis is reasonably conservative.

The following operating parameter considerations are included consistent with Reference 2:

- [[

-

-

-

]]

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The following plant configuration parameter considerations are included consistent with Reference 2:

- [[
-
-

]]

Additional considerations include:

- [[
-
-
-
-

]]

Due to operational considerations, Reference 2 recommends applying [[

]]

Table 5-1 lists all the conditions analyzed for the PRFO event in a compact format.

Table 5-1: HCGS PRFO Event Analysis Initial Conditions

Power (% Rated)	Flow ¹ (% Rated)	Feedwater Temperature ² (°F)	Dome Pressure ³ (psia)
[[

- 1.
- 2.
- 3.
- 4.
- 5.

]]

5.3 Analysis Results

The following aspects are considered when reviewing the PRFO results for acceptability. The dome pressure should remain above the proposed TPSLPB [[]]. However, if the limit [[]] is reached but the core power drops below the low pressure or low flow TPSL, there is no concern. Additionally, the high-water level (L8) setpoint may be reached prior to the LPIS setpoint. In this case, the event is terminated with a L8 turbine trip and scram, [[]]

Table 5-2 summarizes the minimum dome pressure and corresponding STP for each of the initial power cases analyzed. The STP is used to model the core power per Assumption Item 2. For cases which experience L8 turbine trip and scram (45.0% and 50.0% initial rated power), [[]]

[[

]]

Figure 5-1: [[]] **Initial Rated Power PRFO Event Pressure Response**

[[

]]

Figure 5-2: [[]] **Initial Rated Power PRFO Event Power and CPR Responses**

Table 5-3 provides a comparison of the initial CPR (ICPR) to the minimum CPR (MCPR) during the transient for the case [[

]]

Table 5-3: HCGS PRFO Event CPR Results

Initial Power (% Rated)	ICPR (-)	MCPR @700 psia (-)	MCPR @Minimum Dome Pressure (-)
[[]]

6.0 CONCLUSIONS

An evaluation of the GEXL correlation, evaluation of normal operation and limiting event identification of AOOs, and PRFO event analysis are performed for HCGS.

In general, both the GEXL17 and GEXL14 can predict the pressure trend of the GNF3 data down to 600 psia. [[

]] It is also shown that GNF2, GE14, and GNF3 critical power trends with pressure exhibit similar and consistent behavior. Additionally, the PRFO analysis shows that the MCPR increases substantially as the power and pressure decrease. [[

]] Therefore, the effect of any uncertainty [[]]

is not significant because there is abundant SLMCPR margin during the PRFO event.

A review of normal operation confirmed that any challenge to the TPSLPB would have to come from an AOO initiated from pressure conditions consistent with normal pressure control. The AOO event evaluation concludes that the PRFO event is not only the limiting event but the only AOO that can credibly challenge the TPSLPB.

Finally, the PRFO analysis confirmed the proposed TPSLPB of 600 psia adequately bounds the minimum dome pressure with substantial margin, much more margin than is needed to account for initial condition uncertainty. The analysis considered various parameters and demonstrates significant margin, and thus is cycle-independent and applicable to all future reloads assuming no plant modifications are made that would significantly affect the Table 2-1 design inputs or other design bases. Additionally, it is confirmed that the PRFO event is non-limiting in terms of fuel cladding integrity.

7.0 ACRONYMS AND SYMBOLS

7.1 Acronyms

Acronym	Definition
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
BWR	Boiling Water Reactor
BWROG	BWR Owners' Group
CP	Critical Power
CPR	Critical Power Ratio
ECPR	Experimental Critical Power Ratio
EHC	Electro-Hydraulic Control
EOC	End of Cycle
FA	Full Arc
FWTR	Feedwater Temperature Reduction
GEH	GE-Hitachi Nuclear Energy Americas LLC
GEXL	General Electric Critical Quality versus Boiling Length Correlation
HCGS	Hope Creek Generating Station
ICPR	Initial Critical Power Ratio
LAR	License Amendment Request
LCF	Low Core Flow
L8	Level 8
LPIS	Low Pressure Isolation Setpoint
MCFA	Maximum Combined Steam Flow Available
MCFL	Maximum Combined Flow Limiter Setpoint
MCPR	Minimum Critical Power Ratio

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Acronym	Definition
MSIV	Main Steam Isolation Valve
OLMCPR	Operating Limit Minimum Critical Power Ratio
PA	Partial Arc
PRFO	Pressure Regulator Failure Open
RHR	Residual Heat Removal
SC	Safety Communication
SL	Safety Limit
SLMCPR	Safety Limit Minimum Critical Power Ratio
STA	Stern Test Assembly
STP	Simulated Thermal Power
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TPSL	Thermal Power Safety Limit
TPSLPB	Thermal Power Safety Limit Pressure Boundary
TS	Technical Specification
TSV	Turbine Stop Valve
UFSAR	Updated Final Safety Analysis Report

7.2 Symbols

Symbol	Definition
G	Mass Flux
H	Inlet Subcooling

8.0 REFERENCES

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4. Global Nuclear Fuel. *GEXL14 Correlation for GE14 Fuel*. NEDC-32851P-A Revision 5, April 2011.
5. Global Nuclear Fuel. *GEXL17 Correlation for GNF2 Fuel*. NEDC-33292P Revision 3, June 2009.
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7. *Hope Creek Generating Station Updated Final Safety Analysis Report Chapter 15*. HCGS-UFSAR Revision 24.
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9. GE Hitachi Nuclear Energy. *Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients*. NEDE-32906P, Supplement 3-A Revision 1, April 2010.

Enclosure 5

**Affidavit from GEH Supporting the Withholding of Information in Enclosure 6
from Public Disclosure**

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, **Michelle P. Catts**, state as follows:

- (1) I am the Senior Vice President of Nuclear Programs, GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.
- (2) The information sought to be withheld is contained in GEH proprietary report NEDC-33928P, "SC05-03 Evaluation for Hope Creek Generating Station," Revision 0, dated September 2020. GEH proprietary information in NEDC-33928P Revision 0 is identified by a dotted underline inside double square brackets. [[This sentence is an example.^{3}]]. GEH proprietary information in figures and large objects is identified by double square brackets before and after the object. In each case, the superscript notation ^{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.
- (3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. §552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. §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 qualifies 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 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).
- (4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:
 - a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without a license from GEH constitutes a competitive economic advantage over other companies;
 - b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;
 - c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;
 - d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

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- (5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions for proprietary or confidentiality agreements or both that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).
- (6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH.
- (7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.
- (8) The information identified in paragraph (2) is classified as proprietary because it contains the detailed GEH methodology for analyzing and applying GEXL correlations to determine appropriate turbine low-pressure setpoint requirements for the GEH Boiling Water Reactor (BWR). These methods, techniques, and data along with their application to the design, modification, and analyses associated with the setpoint requirements were achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience databases that constitute a major GEH asset.

- (9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply

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the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

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

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

Executed on this 18th day of September 2020.

Michelle P. Catts

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