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

September 7, 2021

Mr. Eric Carr
President and Chief Nuclear Officer
PSEG Nuclear LLC – N09
Hope Creek Generating Station
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION − CORRECTION TO SAFETY EVALUATION FOR AMENDMENT NO. 229 RE: REVISE LOW PRESSURE SAFETY LIMIT TO ADDRESS GENERAL ELECTRIC NUCLEAR ENERGY PART-21 SAFETY COMMUNICATION SC05-03 (EPID L-2020-LLA-0210)

Dear Mr. Carr:

By letter dated August 17, 2021 (Agencywide Documents and Access System (ADAMS) Accession No. ML21181A056), the U.S. Nuclear Regulatory Commission (NRC, the Commission) issued the Amendment No. 229 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station. The amendment revised Technical Specification 2.1, ''SAFETY LIMITS,'' specifically, Safety Limits 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 to address General Electric Nuclear Energy 10 CFR Part 21 Safety Communication SC05-03, ''10 CFR [Part] 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit,'' dated March 29, 2005, regarding the potential to violate the low pressure safety limit following a pressure regulator failure-open transient.

Subsequent to the issuance of the amendment, the licensee identified concerns in certain sections of the safety evaluation (SE) issued with the amendment that may, as written, impact licensee's ability to implement the amendment. The licensee also identified some minor errors/oversights. Upon further review, the NRC staff determined that some statements in the SE could potentially be misinterpreted and thus should be clarified. The clarifications do not change any of the conclusions associated with the issuance of Amendment No. 229 and do not affect the associated notice to the public. Enclosed are the corrected proprietary and non-proprietary SEs. The revised pages contain marginal lines indicating the areas of change. Clarifications and corrections were made on pages 4, 5, 7, 8, 17, 18, 19, 20, 21, and Reference 6 on page 23 of the SE.

Enclosure 1 to this letter contains proprietary information. When separated from Enclosure 1, this document is DECONTROLLED.

Please replace the NRC Safety Evaluation to Amendment No. 229 for Hope Creek Generating Station with the enclosed corrected safety evaluation.

If you have any questions, please contact me at 301-415-4125 or James.Kim @nrc.gov.

Sincerely,

/RA/

James S. Kim, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Corrected Safety Evaluation (Proprietary)
2. Corrected Safety Evaluation (Non-Proprietary)

cc without Enclosure 1: Listserv

ENCLOSURE 2

(NON-PROPRIETARY)

CORRECTED SAFETY EVALUATION

RELATED TO AMENDMENT NO. 229

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

Proprietary information has been redacted from this document pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations*.

Redacted information is identified by blank space enclosed within [[double brackets]].

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

CORRECTED SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 229

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated September 24, 2020 (Reference 1) with enclosure (Reference 2), as supplemented by letters dated April 29, 2021 (Reference 3) with enclosure (Reference 4) and May 27, 2021 (Reference 5) with enclosure (Reference 6), PSEG Nuclear LLC (PSEG, the licensee) submitted a license amendment request for the Hope Creek Generating Station (Hope Creek). The amendment would revise Technical Specification (TS) Safety Limit (SL) 2.1.1, "THERMAL POWER, Low Pressure or Low Flow," and SL 2.1.2, "THERMAL POWER, High Pressure and High Flow." The proposed changes would address the issue identified in a notification pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 21, in General Electric (GE) Safety Communication (SC)05-03 (Reference 7). Specifically, the proposed changes would reduce the low steam dome pressure safety limit (LPSL) specified in SLs 2.1.1 and 2.1.2 from 785 pounds per square inch gauge (psig) to 585 psig. As described in GE SC05-03, the issue concerns a potential to momentarily violate this limit during a Pressure Regulator Failure Maximum Demand - Open (PRFO) transient event. The PRFO event is an analyzed transient in Chapter 15 of the Hope Creek Updated Final Safety Analysis Report (UFSAR) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20142A521). The GE-Hitachi (GEH) report NEDC-33928P, Revision 0 (Reference 2), enclosed with Reference 1, refers to the LPSL as the Thermal Power Safety Limit (TPSL) Pressure Boundary (TPSLPB). The licensee stated that LPSL and TPSLPB describe the same parameter and are interchangeable. Consistent with NEDC-33928P (Reference 2), this Safety Evaluation (SE) uses the abbreviation TPSLPB instead of LPSL.

The supplemental letters dated April 29, 2021, and May 27, 2021, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 1, 2020 (85 FR 77264).

2.0 REGULATORY EVALUATION

2.1 Description of Proposed Changes

The licensee proposed to revise TS 2.1, "Safety Limits," to lower the TPSLPB from 785 psig to 585 psig. The proposed changes are as follows (deletions in strike-out; additions in underline):

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 $\geq$ 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.

2.2 Applicable Regulatory Requirements and Guidance

As discussed in the Hope Creek UFSAR, Section 3.1, "Conformance with the NRC General Design Criteria," for each of the 64 General Design Criteria (GDCs) in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," a specific assessment of the plant design has been made. The Hope Creek UFSAR Section 3.1 also provides a list of UFSAR sections that include further information related to each GDC.

The NRC staff considered the following regulatory requirements and guidance during its review of the application:

- The Atomic Energy Act of 1954, as amended, Section 182a requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The TSs ensure the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36.

- Safety limits are described in 10 CFR 50.36(c)(1)(i)(A) as follows:

  > Safety limits for nuclear reactors are 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. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.

- Compliance with GDC 10, "Reactor design," is achieved by preventing the violation of fuel design limits. GDC 10 states:

  > 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.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," provides guidance on the acceptability of the reactivity control systems, the reactor core, and fuel system design. Specifically, Section 4.2, "Fuel System Design" (Reference 19), provides assurance that the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs) and meets the specified acceptable fuel design limits (SAFDLs). Section 4.4, "Thermal and Hydraulic Design" (Reference 8), provides guidance on the review of thermal-hydraulic design in meeting the requirement of GDC 10 and the fuel design criteria of Section 4.2. Acceptable approaches to meeting these criteria are as follows:

  A. For departure from nucleate boiling ratio (DNBR), CHFR [critical heat flux ratio] or CPR [critical power ratio] correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB [departure from nucleate boiling] or boiling transition condition during normal operation or AOOs.

  B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs.

3.0 TECHNICAL EVALUATION

3.1 Background

The PRFO transient event is an AOO described in Hope Creek UFSAR Section 15.1.3. During this event, failure of the reactor pressure regulator in the open position would lead to a loss of reactor pressure control. In this condition, the turbine control valve (TCV) could be in a fully open position and the turbine bypass valves (TBVs) could be in a partially open position until the maximum flow is established by the maximum combined flow limiter (MCFL).

In GE SC05-03 (Reference 7), by using NRC-approved methods, GE identified that during a PRFO event, there could potentially be a momentary decrease in the reactor steam dome pressure to below 785 psig while the thermal power is greater than 24 percent of the rated thermal power (RTP) specified in the current TS 2.1.1. Therefore, a PRFO event could potentially cause a violation of the current TS 2.1.1.

To resolve this issue, the licensee proposed to lower the TPSLPB from its current value of 785 psig to 585 psig. The licensee stated that revising the TPSLPB in SLs 2.1.1 and 2.1.2 to 585 psig would resolve the issue identified in GE SC05-03 regarding the potential to violate the SL during a PRFO transient. To confirm the resolution of this issue, the licensee evaluated the following:

- The use of the GEXL14 and GEXL17 correlations for the GE14 and GNF2 fuels, respectively, up to the proposed TPSLPB of 585 psig.
- Normal plant operation and AOOs to confirm that the PRFO event is limiting with respect to challenging the TPSLPB for low pressure or low core flow.
- The PRFO event to confirm that exceeding the proposed TPSLPB of 585 psig for low pressure or low core flow will be avoided.

GEXL Correlations

In boiling-water reactors (BWRs) such as Hope Creek, the thermal-hydraulic conditions resulting in a DNB have been used as limiting conditions to avoid entering a region where fuel damage could occur. Although it is recognized that a DNB would not necessarily result in damage to the BWR fuel rods, the critical heat flux (CHF) at which the onset of transition boiling (OTB) is calculated to occur has been adopted as a limiting condition. GE has developed a critical steam quality versus boiling length correlation, termed the GEXL correlation, for accurately predicting the OTB in BWR fuel assemblies during both steady-state operation and reactor transient conditions. NEDO-32851P-A, Revision 5 (Reference 9) and (Reference 10), Section 5.4 defines the following 6 input parameters to the GEXL correlation for the calculation of bundle critical power: boiling length, thermal diameter, mass flux, system pressure, R-factor, and annular flow length. The GEXL correlation is an integral part of the transient analysis used to determine the MCPR operating limits resulting from transient analysis, the MCPR safety limit analysis, and core operating performance and design. As stated in NEDO-32851P-A, the development of the correlation was based on test data from full-scale simulations of 7x7, 8x8, 9x9, and 10x10 fuel assemblies. The tests were performed to demonstrate that the correlation

can be used to predict the OTB during postulated transient conditions that are analyzed in the safety analysis.

The GEXL correlations for the 10x10 GE14, GNF2, and GNF3 fuel designs, denominated as GEXL14, GEXL17, and GEXL21, are provided in NEDO-32851P-A, NEDC-33292P (Reference 11) and (Reference 12), and NEDC-33880P (Reference 13) and (Reference 14), respectively. In the core reload design analysis, these correlations are used in determining the thermal margin for the operating cycle. In the safety analysis, these correlations are used in determining the change in CPR during postulated transients and an acceptable MCPR safety limit (SLMCPR).

As stated in NEDO-32851P-A, Section 7, to facilitate the statistical evaluation of the predictive capability of the GEXL correlation, the concept of an experimental critical power ratio (ECPR) is used, which is determined from the following relationship:

ECPR = Predicted Critical Power / Measured Critical Power

As stated in NEDO-32851P-A, Section 5.4.5, the R-factor, an input parameter to the GEXL correlation, accounts for the effects of the fuel rod power distributions as well as the fuel assembly local spacer and lattice critical power characteristics. Its formulation for a given fuel rod location depends on the power of that fuel rod, as well as the power of the surrounding fuel rods. In addition, there is an additive constant applied to each fuel rod location **[[**
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3.2 <u>Licensee's Evaluation of the GEXL14 and GEXL17 Correlations to Justify their Application up to TPSLPB of 585 psig (≅ 600 pounds per square inch absolute (psia)).</u>

NEDO-32851P-A, NEDC-33292P, and NEDC-33880P provide the following application range for the GEXL correlations based on pressure as determined from their test data:

- GEXL14 correlation for GE14 fuel - **[[** **]]** (NEDC-32851P-A, Table 2)
- GEXL17 correlation for GNF2 fuel - **[[** **]]** (NEDC-33292P, Table 5-4)
- GEXL21 correlation for GNF3 fuel - **[[** **]]** (NEDC-33880P, Table 5-4)

Since the application of the GEXL14 and GEXL17 correlations does not extend up to the proposed TPSLPB of 585 psig (600 psia), the licensee performed the following to justify the application of these correlations up to the proposed TPSLPB of 585 psig:

(a) Analysis to evaluate the performance of the GEXL14 and GEXL17 correlations against the GNF3 test data to confirm that these correlations can predict the GNF3 critical power down to a pressure of 585 psig (600 psia).

(b) Develop measured critical power trend with pressure for GE14, GNF2, and GNF3 to confirm GEXL14 and GEXL17 can predict critical power down to a pressure of 585 psig (600 psia)

For items (a) and (b) above, the licensee used the following method described in NEDC-33928P, Sections 3.0 and 3.1, and evaluated the performance of the GEXL14 and GEXL17

correlations against the GNF3 test data to confirm their accuracy to predict critical power from these correlations down to 600 psia:

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The licensee selected **[[** **]]** for analyzing the performance of GEXL14 and GEXL17 correlations. These included test points **[[**

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In a letter dated April 29, 2021 (References 3 and 4), the licensee provided additional detail to justify its approach to adjust the R-factor additive constants for the GEXL14 and GEXL17 correlations and the data selected to analyze the performance of the resulting correlation at lower pressures.

<u>NRC Staff Conclusion on the Method</u>

The NRC staff determined that the licensee's method of evaluating the performance of the adjusted GEXL14 and GEXL17 correlations against the GNF3 test data and confirming their accuracy to predict critical power from these correlations down to 600 psia is acceptable because:

- The licensee justified the **[[**

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- The licensee's selection of **[[**

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GEXL14 Results

The licensee provided results for the GEXL14 correlation in the following tables and figures of NEDC-33928P:

- Table 3-2 provides the Mean ECPR (Mean (GEXL14 calculation for critical power / GNF3 critical power data)) obtained using the GEXL14 correlation and GNF3 test data at pressures from 600 psia to 1400 psia.

- Figure 3-3 provides the plot of data points of ECPR versus pressure.

- Figure 3-4 provides the plots of calculated critical power by the GEXL14 correlation versus the GNF3 test critical power for pressures from 600 psia to 1400 psia.

- Figure 3-6 shows the measured critical power as a function of pressure for GE14 fuel at different combinations of G (mass flux) and H (inlet subcooling).

- Figure 3-7 shows the GE14, GNF2, and GNF3 fuels' available test data trends with respect to decreasing pressure.

Table 3-2 in NEDC-33928P shows that **[[**

]] Therefore, GEXL14 can predict the critical power of GE14 fuel bundles down to 600 psia **[[**

]] Since the GEXL14 correlation slightly overpredicts GNF3 data at 600 psia compared to the overall GEXL14 statistics, the licensee provided that the remaining GE14 fuel bundles at the current cycle (cycle 23) of Hope Creek are 4th and 5th cycle bundles located on the outer edge of the core with high exposure and large margin to the operating limit minimum critical power ratio (OLMCPR). These GE14 fuel bundles are most likely to be discharged in the next cycle (cycle 24) or they will reside on the periphery or other low power locations. Therefore, the GE14 fuel bundles at the current cycle or future cycles will have significant margin preventing them from being the limiting bundles.

The NRC staff determined that the GEXL14 correlation with adjusted R-factor additive constants can predict the CPR down to the proposed TPSLPB of 585 psig (600 psia) for the GNF3 fuel because the licensee adjusted the additive constants using an acceptable method and addressed the slight overprediction. Therefore, the staff concludes that the GEXL14 correlation is acceptable for predicting the CPR for the GE14 fuel down to the proposed TPSLPB of 585 psig (600 psia).

GEXL17 Results

The licensee provided results for the GEXL17 correlation in the following tables and figures of NEDC-33928P:

- Table 3-1 provides the Mean ECPR (Mean (GEXL17 calculation for critical power / GNF3 critical power data)) obtained using the GEXL17 correlation and GNF3 test data at pressures from 600 psia to 1400 psia.

- Figure 3-1 provides the plot of data points of ECPR versus pressure.

- Figure 3-2 provides the plots of calculated critical power by the GEXL17 correlation versus the GNF3 test critical power for pressures from 600 psia to 1400 psia.

- Figure 3-5 shows the measured critical power as a function of pressure for GNF2 fuel at different combinations of G (mass flux) and H (inlet subcooling).

- Figure 3-7 shows the GE14, GNF2, and GNF3 fuels' available test data trends with respect to decreasing pressure.

Table 3-1 in NEDC-33928P show that **[[**

]] Therefore, GEXL17 can predict the critical power of GNF2 fuel bundles down to 600 psia with **[[** **]]**

The NRC staff determined that the GEXL17 correlation with adjusted R-factor additive constants can predict the CPR down to the proposed TPSLPB of 585 psig (600 psia) for the GNF3 fuel because the licensee adjusted the additive constants using an acceptable method. Therefore, the staff concludes that the GEXL17 correlation is acceptable for predicting the CPR for the GNF2 fuel down to the proposed TPSLPB of 585 psig (600 psia).

3.3 Evaluation of Normal Plant Operation and AOOs

Normal Plant Operation

In NEDC-33928P, Section 4.1, for normal plant operation the licensee states:

> 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 [low pressure isolation setpoint] and the TPSLPB. With the pressure regulator system operating properly there is no possibility that pressure would reduce below the TPSLPB.

The NRC staff determined that the licensee's evaluation for normal plant operation is acceptable because with the pressure regulator in normal operation, there is no possibility for the reactor pressure to drop down to the proposed TPSLPB of 585 psig (600 psia).

AOOs

Hope Creek UFSAR Chapter 15, Sections 15.1 through 15.6 provide safety analysis of AOOs. In NEDC-33928P, Section 4.2, the licensee provided evaluation of AOOs and confirmed that the PRFO event is most limiting with respect to the reactor dome pressure drop while the reactor power is at 24 percent of RTP or above. Table 1 below provides the licensee's evaluation as documented in Table 4-1 of NEDC-33928P and the NRC staff's evaluation of the AOOs. The acceptance criteria for any AOO to be non-limiting is that when it occurs, the reactor pressure must remain above the proposed TPSLPB while the reactor thermal power is above 24 percent of RTP. For reactor thermal power below 24 percent of RTP, the reactor pressure is not of concern whether it stays below or above the proposed TPSLPB.

Table 1: Licensee's and NRC Staff's Evaluation of AOOs

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
15.1.1	Loss of Feedwater Heating	The pressure controller continues to operate normally; therefore, the pressure will remain well above the TPSLPB. It is possible the event could result in a scram on high neutron flux or STP [simulated thermal power]. 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.	The NRC staff finds the licensee's evaluation acceptable because during this event the reactor pressure regulator is not affected, and a reactor scram will reduce the thermal power below 24% of RTP prior to the reactor pressure reaching the proposed TPSLPB.
15.1.2	Feedwater Controller Failure – Maximum Demand	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	The NRC staff finds the licensee's evaluation acceptable because the reactor scram resulting from turbine trip due to reactor high-water level would reduce the reactor thermal power below 24% of RTP before the reactor pressure reaches the proposed TPSLPB.

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
		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.	
15.1.3	Pressure Regulator Failure – Open	Limiting event with respect to approaching TPSLPB with high reactor power.	The licensee's and the NRC staff's evaluations of this event are given in Section 3.4 below.
15.1.4	Inadvertent Main Steam Relief Valve Opening	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.	The NRC staff finds the licensee's evaluation acceptable because the event would result in a sudden increase in the rate of main steam flow leaving the reactor causing a mild depressurization of the reactor and therefore a decrease in the main steam line pressure drop. The pressure controller will continue to operate and stabilize the reactor pressure at a lower value but above the proposed TPSLPB.
15.1.5	Spectrum of Steam System Piping Failures Inside and Outside of Containment in a Pressurized Water Reactor	Not applicable to BWRs.	The NRC staff agrees that this event is 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.	The NRC staff finds the licensee's evaluation acceptable because this transient would not occur when the reactor is in power operation. High reactor pressure would not permit

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
			operation of the RHR system in its shutdown cooling mode.
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.	The event results in a quick closing of the TCVs, which rapidly increases the turbine inlet pressure and the reactor pressure. The NRC staff finds the licensee's evaluation acceptable because the redundant pressure regulator is expected to control the reactor pressure above the proposed TPSLPB and re-establish a steady state.
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.	The NRC staff finds the licensee's evaluation acceptable because the event will result in turbine trip and reactor scram with rapid power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.
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.	The NRC staff finds the licensee's evaluation acceptable because the event will result in turbine trip and reactor scram with rapid power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.
15.2.4	Main Steam Isolation Valve Closures	This event results in an automatic scram on MSIV [main steam isolation valve] position. There is no concern with pressure approaching	The NRC staff finds the licensee's evaluation acceptable because the event will result in turbine trip and reactor scram with rapid

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
		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.	power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP. Additionally, for a single MSIV closure transient, the pressure controller is not expected to be affected and it will maintain the reactor pressure above the proposed 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.	The NRC staff finds the licensee's evaluation acceptable because the event will result in turbine trip and reactor scram with rapid power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.
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.	The NRC staff finds the licensee's evaluation acceptable because the automatic reactor scram will result in a rapid power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.
15.2.7	Loss of Feedwater Flow	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.	The NRC staff finds the licensee's evaluation acceptable because even though there will be a slight reduction in reactor pressure, the pressure controller will maintain the reactor pressure above the proposed TPSLPB.

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
		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.	Additionally, low water level (Level 3) results in a reactor scram.
15.2.9	Failure of RHR Shutdown Cooling	RHR shutdown cooling is used when the reactor is shutdown. The TPSL is not applicable.	The NRC staff agrees with the licensee that shutdown cooling operation does not take place during reactor operation at power.
15.3.1	Reactor Recirculation Pump Trip	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. A trip of two reactor recirculation pumps may result in a scram on high reactor water level. 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 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.	The NRC staff finds the licensee's evaluation acceptable because the pressure controller operation to maintain reactor pressure above the proposed TPSLPB will not be affected. Additionally, if an automatic reactor scram were to occur due to high water level, it would result in a rapid power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
15.3.2	Recirculation Flow Control Failure – Decreasing Flow	A Recirculation Flow Control Failure – Decreasing Flow may result in a scram on high reactor water level. 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. 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.	The NRC staff finds the licensee's evaluation acceptable because a decreasing recirculation flow will increase the reactor water level and the pressure controller will continue to operate and maintain the reactor pressure above the proposed TPSLPB. Additionally, if an automatic reactor scram were to occur due to high water level, it would result in a rapid power drop below 24% of RTP while the reactor pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.
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.	The NRC staff finds the licensee's evaluation acceptable because the event is postulated to occur at power level below 24% of RTP, at which point the reactor pressure is not of concern whether it is above or below the proposed 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.	The NRC staff finds the licensee's evaluation acceptable because the pressure controller is not affected and will continue to maintain the reactor pressure above the proposed TPSLPB.
15.4.3	Control Rod Maloperation (System Malfunction or Operator Error)	Covered by 15.4.1 and 15.4.2.	UFSAR Section 15.4.3 states that this event is less severe than rod withdrawal events 15.4.1 and 15.4.2. Therefore, the NRC staff evaluations for 15.4.1 or 15.4.2 also apply for this event.

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
15.4.4	Abnormal Startup of Idle Recirculation Pump	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. 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.	The NRC staff finds the licensee's evaluation acceptable because, in the case that this transient event leads to a rapid rise in the reactor power level, the high neutron flux will automatically scram the reactor resulting in a rapid power drop below 24% of RTP while the pressure is above the proposed TPSLPB. Therefore, it would not be possible for the reactor pressure to drop below the proposed TPSLPB while the power is above 24% of RTP. Additionally, if an automatic reactor scram does not occur, the pressure controller remains functional and would control the reactor pressure above the proposed TPSLPB.
15.4.5	Recirculation Flow Control Failure with Increasing Flow	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.	The NRC staff finds the licensee's evaluation acceptable because, in the case that this transient event leads to a rapid rise in the reactor power level, the high neutron flux will automatically scram the reactor resulting in a rapid power drop below 24% of RTP while the pressure is above the proposed TPSLPB. Therefore, it would not be possible for the reactor pressure to drop below the proposed TPSLPB while the power is above 24% of RTP. Additionally, if an automatic reactor scram does not occur, the pressure controller remains functional and would control the reactor pressure above the proposed TPSLPB.

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
15.4.6	Chemical and Volume Control System Malfunctions	Not applicable to BWRs.	The NRC staff agrees that this event is 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.	The NRC staff finds the licensee's evaluation acceptable because this event will have no effect on the pressure controller and, therefore, the reactor pressure will remain above the proposed TPSLPB.
15.4.8	Spectrum of Rod Ejection Accidents	Not applicable to BWRs.	The NRC staff agrees that this event is not applicable to BWRs.
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 TPSLPB. 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 TPSLPB with the power above 24%, because the scram will reduce power below 24% before the pressure reduces significantly.	The NRC staff finds the licensee's evaluation acceptable because, as stated in the UFSAR, this event would result in a mild reactor pressurization and corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state and, therefore, the reactor pressure would be stabilized above the proposed TPSLPB. Additionally, in the case of a scram due to high-water level, the reactor power would drop below 24% of RTP while the pressure is above the proposed TPSLPB. There is no chance that the pressure will drop below the proposed TPSLPB while the power is above 24% of RTP.
15.5.2	Chemical Volume Control System Malfunction (or Operator Error)	Not applicable to BWRs.	The NRC staff agrees that this event is not applicable to BWRs.

UFSAR Section	Event	Licensee's Evaluation in NEDC-33928P, Table 4-1	NRC Staff's Evaluation
15.5.3	Increase In Reactor Coolant Inventory BWR Transients	Covered in Section 15.1 and 15.2.	The increase in reactor coolant inventory is because of an increase in density, or decrease in voiding, due to either (a) a decrease in temperature or (b) an increase in its pressure. The NRC staff agrees that the evaluations of events described in UFSAR Section 15.1, "Decrease in Reactor Coolant Temperature," and Section 15.2, "Increase in Reactor Pressure," apply to the events that cause increase in the reactor coolant inventory.
15.6.1	Inadvertent Safety/Relief Valve Opening	See 15.1.4.	The NRC staff's evaluation is the same as for the event in 15.1.4.
15.6.3	Steam Generator Tube Failure	Not applicable to BWRs.	The NRC staff agrees that this event is not applicable to BWRs.

Based on the above, the NRC staff concludes that the PRFO event is limiting with respect to challenging the TPSLPB.

3.4 PRFO Event Evaluation

The purpose of the licensee's evaluation of the PRFO event is to demonstrate that the reactor dome pressure during a PRFO event remains above the proposed TPSLPB of 585 psig. The licensee's analysis is based on the following sequence:

- All three pressure regulators fail open causing a maximum steam demand dictated by the MCFL value.
- The generated signal opens the TCVs and TBVs to meet the increased steam flow demand.

The increase in steam flow would result in: (a) decrease in the reactor dome pressure and (b) increase in the reactor water level due to swelling caused by increased void formation. If the turbine inlet pressure decreased to the proposed TPSLPB first due to item (a), then the MSIVs would have already closed at the Low Pressure Isolation Setpoint (LPSI) and their position switch setpoint signal would scram the reactor. The MSIVs' closure would result in a reactor dome pressure rebound that stops the reactor depressurization and, therefore, terminates the

event. If a reactor high-water level (L8) setpoint is reached first due to item (b), then the turbine trip and reactor scram would occur. The licensee determined that the **[[**

]]

3.4.1 Analysis

The licensee used the NRC-approved TRACG04 methodology for transient analysis documented in NEDE-32906P-A, Revision 3 (Reference 15) and (Reference 16) and in NEDE-32906P, Supplement 3-A, Revision 1 (Reference 17) and (Reference 18). NEDC-33928P, Table 2-1 provides the key values of design inputs. NEDC-33928P, Table 2-2 provides the major assumptions, which are:

1. **[[**

   ]]

2. To calculate the simulated thermal power (STP), the licensee conservatively used the thermal power scram time constant of 6.6 seconds as an input. The licensee stated that the STP time constant used is **[[ ]].** Therefore, the use of the STP time constant is conservative for the purpose of the PRFO analysis.

The licensee provided additional information regarding the development and use of the thermal power scram time constant via letter dated May 27, 2021. Specifically, the licensee provided that:

- The definition of the plant STP time constant is consistent with its classical definition.

- The parameter in consideration for the time constant is the fuel surface heat flux which represents the time delay between the neutron flux response and the fuel surface heat flux response.

- In the PRFO analysis, the time constant provides a conservative approximation of the fuel surface heat flux based upon the neutron flux measured by the APRMs.

- The licensee demonstrated that the time constant value utilized for the PRFO analysis is conservative in predicting the fuel surface heat flux.

Other Considerations

The licensee stated that the key parameters affecting the minimum reactor dome pressure are in the categories of operating parameters and the parameters related to plant configuration.

The input parameters related to plant operation are initial core power, core flow, cycle exposure, feedwater temperature, and main steam flow. For conservative analysis the licensee used **[[**

]]

The input parameters related to plant configuration are LPIS, main steam line (MSL) pressure drop, MSIV closure time, and delay time of the pressure sensor. For conservative analysis, the licensee used the **[[**

]]

Additional considerations for conservative analysis included **[[**

]] because it will lead to a higher main steam flow resulting in a greater steam dome pressure drop during the event.

The licensee analyzed the PRFO event for the following:

- **[[**

]]

The licensee used **[[**

]] The licensee stated that the **[[**
]]

3.4.2 Results

NEDC-33928P, Table 5-2 tabulates the PRFO analysis results for all cases analyzed **[[**
]] NEDC-33928P, Figure 5-1 presents the reactor dome and turbine inlet pressure responses for the initial thermal power of 55 percent and 85 percent of RTP cases.

The key results to be examined for the reactor dome pressure response are at the lower end of the initial power level, which are as follows:

[[

]]

NEDC-33928P, Figure 5-2 presents graphs of **[[**

] The licensee stated that the reactor scram occurred at about 7.5 and 9.5 seconds **[[** **]]** respectively, at which point the CPR has greatly increased.

NEDC-33928P, Table 5-3 shows that **[[**

]] The licensee therefore stated that the effect of uncertainty **[[** **]]** is not significant because there is a significantly large SLMCPR margin during the PRFO event.

The NRC staff determined that the results presented in NEDC-33928P, Tables 5-2 and 5-3, and Figures 5-1 and 5-2 are acceptable because of the following:

- The minimum dome pressure stays above the proposed value of TPSLPB (585 psig) **[[** **]]** with significant margin for the conservatively analyzed **[[** **]]**

- When the thermal power is equal to or less than the TPSL (24 percent of RTP), the proposed TPSLPB (585 psig) does not apply.

- The PRFO event does not affect the fuel integrity as the CPR responses indicate significantly increased margin in CPR as the reactor dome pressure decreases.

- Table 2 of NEDC-33743P recommends a **[[**

  ]] The licensee's analysis results show significant margin in the minimum reactor dome pressure during the analyzed PRFO event.

Based on the technical evaluations presented above, the NRC staff concludes the followings:

- The GEXL14 and GEXL17 correlations can predict the critical power trend with pressure of the GE14 and GNF2 fuels, respectively, to the proposed TPSLPB of 585 psig.

- The proposed TPSLPB is not challenged during normal plant operation.

- The evaluation of AOOs shows that the PRFO event is not only the limiting transient but the only AOO that can credibly challenge the TPSLPB.

- The PRFO event evaluation results show that the MCPR increases substantially as the power and pressure decrease and, therefore, is a non-limiting event for the fuel integrity.

- The PRFO event evaluation confirmed that the lowest reactor dome pressure reached during this event is greater than (bounded by) the proposed TPSLPB of 585 psig with a significant margin while the STP is 24 percent of RTP or greater.

3.5 Technical Conclusions

With respect to the proposed change of the TPSLPB from 785 psig to 585 psig in Hope Creek SL 2.1.1 and SL 2.1.2, the NRC staff concludes as follows:

- For the AOOs described in the Hope Creek UFSAR, including the PRFO transient, the GEXL14 and GEXL17 correlations can be used to predict the critical power versus pressure trend down to the TPSLPB of 585 psig for the GE14 and GNF2 fuels, respectively.

- The issue reported in GE SC05-03 regarding the potential to violate the SLs during a PRFO event is resolved for Hope Creek. This event is non-limiting for fuel cladding integrity and the proposed change will have no negative impact on the MCPR.

- The SAFDLS are not exceeded during normal plant operation and AOOs and the fuel cladding integrity is maintained.

The NRC staff evaluated the proposed change against the applicable regulatory requirements and acceptance criteria and concludes that as long as the reactor pressure and core flow are within the revised range of the approved GEXL14 and GEXL17 correlations, the proposed TPSLPB changes in Hope Creek SL 2.1.1 and SL 2.1.2 will continue to ensure that the MCPR of the fuel rods in the core is within the required limit. Therefore, as required by GDC 10 and 10 CFR 50.36(c)(1)(i)(A), the criteria of the SLs are met and, thus, the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment on July 2, 2021. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, as published in the *Federal Register* (85 FR 77264; December 1, 2020), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

7.0 REFERENCE

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3 PSEG Nuclear LLC, letter LR-N21-0018 to U. S. Nuclear Regulatory Commission (NRC), "Hope Creek Generating Station - Response to Requests for Additional Information SNSB-RAI 2 and SNSB RAI 3 Re: License Amendment Request to Revise Low Pressure Safety Limit to Address General Electric Part 21 Safety Communication," April 29, 2021 (ADAMS Accession No. ML21119A367).

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Date: August 17, 2021

SUBJECT: HOPE CREEK GENERATING STATION − CORRECTION TO SAFETY EVALUATION FOR AMENDMENT NO. 229 RE: REVISE LOW PRESSURE SAFETY LIMIT TO ADDRESS GENERAL ELECTRIC NUCLEAR ENERGY PART-21 SAFETY COMMUNICATION SC05-03 (EPID L-2020-LLA-0210) DATED: SEPTEMBER 7, 2021.

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