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

September 19, 2019

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

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT NO. 219 REGARDING REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-564, "SAFETY LIMIT MCPR [MINIMUM CRITICAL POWER RATIO]" (EPID L-2019-LLA-0084)

Dear Mr. Carr:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 219 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station in response to your application dated April 22, 2019.

The amendment adopts Technical Specifications Task Force (TSTF) Traveler TSTF-564, Revision 2, "Safety Limit MCPR [Minimum Critical Power Ratio]," which revises the Hope Creek Generating Station Technical Specification safety limit on minimum critical power ratio to reduce the need for cycle-specific changes to the value, while still meeting the regulatory requirement for a safety limit.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

James S. Kim

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. Amendment No. 219 to
   Renewed License No. NPF-57
2. Safety Evaluation

cc: Listserv

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

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 219
Renewed License No. NPF-57

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by PSEG Nuclear LLC dated April 22, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-57 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 219, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to restart following Refueling Outage H1R22.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: September 19, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 219

HOPE CREEK GENERATING STATION

RENEWED FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2-1	2-1
3/4 4-1	3/4 4-1
6-20	6-20

reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

(7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3902 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 219, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a) Place the recirculation flow control system in the Local Manual mode, and
      b) Reduce THERMAL POWER to ≤ 59.89% of RATED THERMAL POWER, and
      c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
      d) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
      e) Reduce the LINEAR HEAT GENERATION RATE (LHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
      f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
      g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow.

   2. Within 4 hours, reduce the Average Power Range Monitor (APRM) Scram Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 2.2.1; otherwise declare the APRM channel INOPERABLE and take the action of RPS Instrumentation TS 3.3.1 ACTION a.

   3. Within 4 hours, reduce the APRM Control Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise declare the APRM channel INOPERABLE and take the action of Control Rod Block Instrumentation TS 3.3.6 ACTION a and b.

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* See Special Test Exception 3.10.4.

6.9.1.8 Deleted

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the PSEG Nuclear LLC generated CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following Technical Specifications:

2.2	Reactor Protection System Instrumentation Setpoints
3/4.1.4.3	Rod Block Monitor
3/4.2.1	Average Planar Linear Heat Generation Rate
3/4.2.3	Minimum Critical Power Ratio*
3/4.2.4	Linear Heat Generation Rate
3/4.3.1	Reactor Protection System Instrumentation
3/4.3.6	Control Rod Block Instrumentation

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC as applicable in the following document:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number title, revision, date, and any supplements).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The COLR, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.10

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Condition for Operation Section 3.4.6, "RCS Pressure/Temperature Limits"

   2. Surveillance Requirement Section 4.4.6, "RCS Pressure/Temperature Limits"

* The $MCPR_{99.9\%}$ value, for both Two Recirculation Loop Operation and Single Recirculation Loop Operation, used to calculate the LCO 3.2.3, Minimum Critical Power Ratio, limit shall be specified in the COLR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 219

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION AND BACKGROUND

By application dated April 22, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19112A214), PSEG Nuclear LLC (PSEG or the licensee) requested U.S. Nuclear Regulatory Commission (NRC or the Commission) approval for a license amendment request to revise the Hope Creek Generating Station (Hope Creek) Technical Specifications (TSs).

The license amendment request proposed to revise the technical basis, calculational method, and the value of TS Safety Limit (SL) 2.1.2, which protects against boiling transition on the fuel rods in the core. The current basis ensures that 99.9 percent of the fuel rods in the core are not susceptible to boiling transition. The revised basis would ensure that there is a 95 percent probability at a 95 percent confidence level that no fuel rods will be susceptible to boiling transition using an SL based on critical power ratio (CPR) data statistics. As a result of the change to TS 2.1.2, the licensee proposed to delete the reference to Specification 2.1.2 in Limiting Condition for Operation (LCO) 3.4.1.1 ACTION a.1.c), as the minimum critical power ratio $(MCPR)_{95/95}$ is not dependent on the number of recirculation loops in operation, and replaced with a reference to the MCPR limit listed in the COLR for single loop operation. TS 6.9.1.9, "Core Operating Limits Report (COLR)," would also be modified to require $MCPR_{99.9\%}$ value to be included in the cycle-specific COLR.

The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-564, Revision 2, "Safety Limit MCPR [Minimum Critical Power Ratio]," dated October 24, 2018 (ADAMS Accession No. ML18297A361). TSTF-564 proposed a method for determining a revised, cycle-independent MCPR SL. The NRC issued a final safety evaluation (SE) approving TSTF-564, Revision 2, on November 16, 2018 (ADAMS Accession No. ML18299A069).

1.1 Background on Boiling Transition

During steady-state operation in a boiling-water reactor (BWR), most of the coolant in the core is in a flow regime known as annular flow. In this flow regime, a thin liquid film is pushed up the surface of the fuel rod cladding by the bulk coolant flow, which is mostly water vapor with some liquid water droplets. This provides effective heat removal from the surface of the fuel cladding. However, under certain conditions, the annular film may dissipate, which reduces the heat transfer and results in an increase in fuel cladding surface temperature. This phenomenon is known as boiling transition or dryout. The elevated surface temperatures resulting from dryout may cause damage or failure of the fuel cladding.

1.2 Background on Critical Power Correlations

For a given set of reactor operating conditions (pressure, flow, etc.), dryout will occur on a fuel assembly at a certain power, known as the critical power. Because the phenomena associated with boiling transition are complex and difficult to model purely mechanistically, thermal-hydraulic test campaigns are undertaken using electrically heated prototypical fuel bundles to establish a comprehensive database of critical power measurements for each BWR fuel product.

These data are then used to develop a critical power correlation that can be used to predict the critical power for assemblies in operating reactors. This prediction is usually expressed as the ratio of the actual assembly power to the critical power predicted using the correlation, known as the CPR.

One measure of the correlation's predictive capability is based on its validation relative to the test data. For each point *j* in a correlation's test database, the experimental critical power ratio (ECPR) is defined as the ratio of the measured critical power to the calculated critical power, or:

$$ECPR_j = \frac{Measured\ Critical\ Power_i}{Calculated\ Critical\ Power_j}$$

For ECPR values less than or equal to one, the calculated critical power is greater than or equal to the measured critical power, and the prediction is considered to be non-conservative. Because the measured critical power includes random variations due to various uncertainties, evaluating the ECPR for all the points in the dataset (or, ideally, a subset of points that was not used in the correlation's development) results in a probability distribution. This ECPR distribution allows the predictive uncertainty of the correlation to be determined. This uncertainty can then be used to establish a limit above which it can be assumed that boiling transition will not occur (with a certain probability and confidence level).

1.3 Background on Thermal-Hydraulic Safety Limits

To protect against boiling transition, BWRs have implemented an SL on the CPR, known as the minimum critical power ratio (MCPR) SL. Consistent with the current BWR Standard Technical Specifications (STSs),[1] the current calculation of the MCPR SL for the licensee's facility is to prevent 99.9 percent of the fuel in the core from being susceptible to boiling transition. This limit is typically developed by considering various cycle-specific power distributions and uncertainties and is highly dependent on the cycle-specific radial power distribution in the core. As such, the limit may need to be updated as frequently as every cycle.

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[1] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, General Electric Plants BWR/4," NUREG-1433, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, dated April 2012 (ADAMS Accession Nos. ML12104A192 and ML12104A193).

The fuel cladding SL for pressurized-water reactor (PWR) designs described in the STSs for Babcock & Wilcox, Westinghouse, and Combustion Engineering [2] plants in NUREG-1430, NUREG-1431 and NUREG-1432,[3] respectively, correspond to a 95 percent probability at a 95 percent confidence level that departure from nucleate boiling will not occur. As a result of the overall approach taken in developing the PWR limits, they are only dependent on the fuel type(s) in the reactor and the corresponding departure from nucleate boiling ratio (DNBR) correlations. The limits are not cycle-dependent and are typically only updated when new fuel types are inserted in the reactor.

The TSs for Hope Creek also include an LCO that governs MCPR, known as the MCPR operating limit (OL). The OL on MCPR is an LCO that must be met to ensure that anticipated operational occurrences (AOOs) do not result in fuel damage. The current MCPR OL is calculated by combining the largest change in CPR from all analyzed transients, also known as the ΔCPR, with the MCPR SL.

2.0 REGULATORY EVALUATION

2.1 Hope Creek Current TS Sections

SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs.

Hope Creek TS 2.1.2 under TS 2.1, "Safety Limits," currently requires that with reactor steam dome pressure greater than 785 psig and core flow greater than 10 percent of rated flow, MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

Hope Creek TS 3/4.2.3, "MINIMUM CRITICAL POWER RATIO," requires that MCPR shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

Hope Creek TS 6.9.1.9, "CORE OPERATING LIMITS REPORT," requires core OLs to be established before each reload cycle or any remaining portion of a reload cycle. These limits are required to be documented in the CORE OPERATING LIMITS REPORT.

2.2 Proposed Changes to the TSs

The licensee proposed to revise the MCPR SL to make it cycle-independent, consistent with the method described in TSTF-564, Revision 2.

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[2] Denotes applicability to Combustion Engineering plants with digital control systems only

[3] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Babcock and Wilcox Plants," NUREG-1430, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, dated April 2012 (ADAMS Accession Nos. ML12100A177 and ML12100A178)

U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, dated April 2012 (ADAMS Accession Nos. ML12100A222 and ML12100A228)

U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Combustion Engineering Plants," NUREG-1432, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, dated April 2012 (ADAMS Accession Nos. ML12102A165 and ML12102A169)

The proposed changes to the Hope Creek TSs revise the value of the MCPR SL in TS 2.1.2 to 1.07, with corresponding changes to the associated bases. The change to TS 2.1.2 replaces the existing separate SLs for single- and two-recirculation loop operation with a single limit, since the revised SL is no longer dependent on the number of recirculation loops in operation.

The $MCPR_{99.9\%}$ (i.e., the current MCPR SL) is an input to the MCPR OL in Limiting Condition for Operation (LCO) 3.2.3, "Minimum Critical Power Ratio (MCPR)." While the definition and method of calculation of both $MCPR_{99.9\%}$ and LCO 3.2.3 MCPR OL remain unchanged, the proposed TS changes include revisions to TS 6.9.1.9 to require the $MCPR_{99.9\%}$ value used in calculating the LCO 3.2.3 MCPR OL to be included in the cycle-specific COLR.

In addition, the licensee proposed variations from the TS changes described in TSTF-564 or the applicable parts of the NRC staff's SE.

Although Hope Creek is a BWR/4 plant, the Hope Creek TSs are based on NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors," and therefore, the wording, numbering, and format vary slightly from NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," shown in TSTF-564, Revision 2, and the applicable parts of the NRC staff's SE. The licensee stated that these variations are editorial changes.

The Hope Creek TSs contain requirements that differ from the STSs on which TSTF-564 was based, such as TS 3/4.4.1, "Recirculation Loops." LCO 3.4.1.1 Action a.1.c) needs revision as a result of the change to TS 2.1.2. Reference to Specification 2.1.2 would be deleted, as the $MCPR_{95/95}$ is not dependent on the number of recirculation loops in operation and would be replaced with a reference to the MCPR limit listed in the COLR for single loop operation. The licensee stated that this change would align the Hope Creek TSs with the current STS wording and does not affect the applicability of the TSTF-564 Revision 2.

2.3 Applicable Regulatory Requirements and Guidance

The regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(a)(1), requires an applicant for an operating license to include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must also include in the application a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls." However, per 10 CFR 50.36(a)(1), these TS bases "shall not become part of the technical specifications."

As required by 10 CFR 50.36(c), TSs will include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings. As required by 10 CFR 50.36(c)(1)(i)(A), safety limits for nuclear reactors are "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. 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."

As required by 10 CFR 50.36(c)(2)(i), the TSs will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. Paragraph 50.36(c)(5) of 10 CFR states that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

"Acceptance Criteria" of Section 4.4, "Thermal and Hydraulic Design," Revision 2 (ADAMS Accession No. ML070550060), of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP) states, in part:

> SRP Section 4.2 specifies the acceptance criteria for the evaluation of fuel design limits. One criterion provides assurance that there be at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs. ...
>
> The following are two examples of acceptable approaches to meeting this criterion:
>
> A. For departure from nucleate boiling ratio (DNBR), CHFR [critical heat flux ratio] or CPR 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 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 <u>TECHNICAL EVALUATION</u>

3.1 <u>MCPR SL Calculation</u>

As discussed in Section 1.3 of this SE, the current MCPR SL (i.e., the $MCPR_{99.9\%}$) is affected by the plant's cycle-specific core design, especially including the core power distribution, fuel type(s) in the reactor, and the power-to-flow operating domain for the plant. As such, it is frequently necessary to change the MCPR SL to accommodate new core designs. Changes to the MCPR SL are usually determined late in the design process and necessitate an accelerated NRC review (i.e., license amendment request) to support the subsequent fuel cycle.

PSEG proposed to change the calculation for the MCPR SL for Hope Creek so that it is no longer cycle-dependent, reducing the frequency of revisions and eliminating the need for the NRC's review on an accelerated schedule. The proposed revised calculation for the MCPR SL aligns it with that of the DNBR SL used in PWRs, which, as previously noted in Section 2.3 of this SE, provides a 95 percent probability at a 95 percent confidence level that no fuel rods will experience departure from nucleate boiling.

The NRC staff finds that calculating the revised MCPR SL on a 95/95 criterion is acceptable because it meets SRP Section 4.4, Acceptance Criterion 1. The remainder of this SE evaluates whether the methodology for determining the revised MCPR SL provides the intended result and documents the review to ensure that the revised MCPR SL can be adequately determined in the core using various types of fuel, that the proposed SL continues to fulfil the necessary functions of an SL without unintended consequences, and that the proposed changes have been adequately implemented in the Hope Creek TSs.

3.2 Revised MCPR SL Calculational Method

As discussed in Section 1.2 of this SE, a critical power correlation's ECPR distribution quantifies the uncertainty associated with the correlation. TSTF-564, Revision 2, provides a formula for a limit that bounds 95 percent of a correlation's ECPR distribution at a 95 percent confidence level, according to the following formula:

$$MCPR_{95/95}(i) = \mu_i + \kappa_i\sigma_i$$

where μi is the correlation's mean ECPR, σi is the standard deviation of the correlation's ECPR distribution, and κi is a statistical parameter chosen to provide "95% probability at 95% confidence (95/95) for the one-sided upper tolerance limit that depends on the number of samples (Ni) in the critical power database." This formula is commonly used to determine a 95/95 one-sided upper tolerance limit for a normal distribution, which is appropriate for the situation under consideration. The factor κ is generally attributed to D. B. Owen[4] and was also reported by M. G. Natrella,[5] as referenced in TSTF-564, Revision 2. Example values of κ are provided in Table 2 of TSTF-564, Revision 2. Table 1 of the traveler includes some reference values of the $MCPR_{95/95}$.

As discussed by Piepel and Cuta,[6] for PWR DNBR correlations, the acceptability of this approach is predicated on a variety of assumptions, including the assumptions that the correlation data comes from a common population and that the correlation's population is distributed normally. These assumptions are typically addressed generically when a critical power or critical heat flux correlation is reviewed by the NRC staff, who may apply penalties to the correlation to account for any issues identified. In a letter dated May 28, 2018 (ADAMS Accession No. ML18149A320), the TSTF stated that such penalties applied during the NRC's review of the critical power correlation would be imposed on the mean or standard deviation used in the calculating the $MCPR_{95/95}$ for BWRs. These penalties would also continue to be imposed in the determination of the $MCPR_{99.9\%}$, along with any other penalties associated with the process of (or other inputs used in) determining the $MCPR_{99.9\%}$ (e.g., penalties applied to the $MCPR_{99.9\%}$ SL for operation in the Maximum Extended Load Limit Line Analysis Plus (MELLLA+) operating domain).

The NRC staff reviewed the information provided by the TSTF and determined the formula for $MCPR_{95/95}$ will appropriately establish a 95/95 upper tolerance limit on the critical power correlation and that any issues in the underlying correlation will be addressed through penalties

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[4] D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," Sandia Corporation, SCR-607, dated March 1963, ADAMS Accession No. ML14031A495

[5] M. G. Natrella, "Experimental Statistics," National Bureau of Standards, National Bureau of Standards Handbook 91, dated August 1963

[6] G. F. Piepel and J. M. Cuta, "Statistical Concepts and Techniques for Developing, Evaluating, and Validating CHF Models and Corresponding Fuel Design Limits," SKI Technical Report, 93:46, 1993

on the correlation mean and standard deviation, as necessary. Therefore, the NRC staff concluded that the $MCPR_{95/95}$ formula can be used to establish acceptable fuel design limits.

3.3 Determination of Revised MCPR SL for Mixed Cores

Traveler TSTF-564, Revision 2, proposed that a core containing a variety of fuel types would evaluate the $MCPR_{95/95}$ for all of the fresh and once-burnt fuel in the core and apply the most limiting (i.e., the largest) value of $MCPR_{95/95}$ for each of the applicable fuel types as the MCPR SL. As stated in Section 3.1 of traveler TSTF-564, Revision 2, this is because bundles that are twice-burnt or more at the beginning of the cycle have significant MCPR margin relative to the fresh and once-burnt fuel. The justification is that the MCPR for twice-burnt and greater fuel is far enough from the MCPR for the limiting bundle that its probability of boiling transition is very small compared to the limiting bundle and it can be neglected in determining the SL. Results of a study provided by the TSTF indicate that this is the case even for fuel operated on short (12-month) reload cycles. As discussed in the traveler, twice-burnt or greater fuel bundles are included in the cycle-specific evaluation of the $MCPR_{99.9\%}$ and the MCPR OL. If a twice-burnt or greater fuel bundle is found to be limiting, it would be governed by the MCPR OL, which will always be more restrictive than both the $MCPR_{95/95}$ and the $MCPR_{99.9\%}$. The NRC staff found this justification to be appropriate and determined that it is acceptable to determine the $MCPR_{95/95}$ SL for the core based on the most limiting value of the $MCPR_{95/95}$ for the fresh and once-burnt fuel in the core.

The NRC staff reviewed the information furnished by the TSTF and determined that the process for establishing the revised MCPR SL for mixed cores ensures that the limiting fuel types in the core will be evaluated and the limiting $MCPR_{95/95}$ will be appropriately applied as the SL. The NRC staff, therefore, found this process to be acceptable.

3.4 Relationship Between MCPR Safety and OLs

As discussed in TSTF letter dated May 29, 2018, the current $MCPR_{99.9\%}$ safety limits are greater than the proposed $MCPR_{95/95}$ safety limits for two reasons. First, because the $MCPR_{99.9\%}$ includes uncertainties not factored into the $MCPR_{95/95}$, and second, because the 99.9 percent probability basis for determining the $MCPR_{99.9\%}$ is more conservative than the 95 percent probability at a 95 percent confidence level used in determining the $MCPR_{95/95}$. The level of conservatism in the $MCPR_{95/95}$ SL is appropriate because the lead fuel rod in the core (i.e., the limiting fuel rod with respect to MCPR) is used to evaluate whether any fuel rods in the core are susceptible to boiling transition). This is consistent with evaluations performed for PWRs using a 95/95 upper tolerance limit on the correlation uncertainty as an SL.

Consistent with TSTF-564, Revision 2, the MCPR OL for STS LCO 3.2.2 would continue to be evaluated using the $MCPR_{99.9\%}$ as an input. This STS corresponds to Hope Creek TS 3.2.3. Per, TS 3.2.3, which requires that the MCPR limit be specified in the COLR, the $MCPR_{99.9\%}$ will continue to be evaluated in the same way as it is currently, using the whole core.

Consistent with TSTF-564, Revision 2, the TSTF also proposed to revise STS 5.6.3 to require the cycle-specific value of the $MCPR_{99.9\%}$ to be included in the COLR. This TS corresponds to Hope Creek TS 6.9.1.9. The methods for determining $MCPR_{99.9\%}$ are included in the list of COLR references contained in Hope Creek TS 6.9.1.9 (COLR). Consistent with the TSTF, the licensee proposed to add a footnote to Hope Creek TS 6.9.1.9 (COLR) that would require $MCPR_{99.9\%}$ value for both single and two recirculation loop operation to be included in the cycle-specific COLR to ensure that the uncertainties being removed from the MCPR SL are still included as part of the MCPR OL. The licensee also inserted "COLR" in place of "CORE OPERATING LIMITS REPORT" after the term is first defined in TS 6.9.1.9. The NRC staff finds these changes are consistent with TSTF-564 and are acceptable.

The NRC staff, therefore, finds that the changes proposed by the licensee will retain an adequate level of conservatism in the MCPR SL in TS 2.1.2, while appropriately ensuring that plant- and cycle-specific uncertainties will be retained in the MCPR OL. The $MCPR_{95/95}$ represents a floor on the value of the $MCPR_{99.9\%}$, which should always be higher, since it accounts for numerous uncertainties that are not included in the $MCPR_{95/95}$.

3.5 Implementation of the Revised MCPR SL in the TSs

The licensee proposed to change the value of the SL in TS 2.1.2 from ≥1.09 and ≥ 1.12 (for two recirculation loop operation and single recirculation loop operation, respectively) to 1.07, consistent with the value from Table 1 of TSTF-564, Revision 2, for the fuel types in use at Hope Creek (i.e., GE14, GNF2, and possible GNF3 in future). As noted in Section 3.3, above, the licensee appropriately evaluated the fresh and once-burnt fuels in use at Hope Creek, and the NRC staff concluded that the limiting $MCPR_{95/95}$ for these fuels was provided for inclusion in TS 2.1.2, consistent with the process described in TSTF-564, Revision 2.

The NRC staff finds that the value reported in Hope Creek TS 2.1.2 is acceptable because it was calculated using Equation 1 from TSTF-564, Revision 2, and reported at a precision of two digits past the decimal point with the hundreds digit rounded up. Thus, the proposed TS change is acceptable.

As a result of the change to TS 2.1.2, the licensee proposed to delete the reference to Specification 2.1.2 in LCO 3.4.1.1 ACTION a.1 .c), as the $MCPR_{95/95}$ is not dependent on the number of recirculation loops in operation, and replaced with a reference to the MCPR limit listed in the COLR for single loop operation. The NRC staff finds this variation acceptable because it is consistent with TSTF-564 and does not affect the applicability of the LCO.

PSEG also proposes to add a footnote to Hope Creek's TS 6.9.1.9 that requires the value of the $MCPR_{99.9\%}$ for both single and two recirculation loop operation, which is used to calculate the MCPR limit in Hope Creek LCO 3.2.3, be specified in the COLR to continue to require that the value be reported in the COLR. Thus, the COLR continues to report the cycle-specific value of the MCPR OL contained in LCO 3.2.3. Hope Creek TS 6.9.1.9 will continue to reference appropriate NRC-approved methodologies for determination of the $MCPR_{99.9\%}$ and the MCPR OL, which will ensure that cycle-specific parameters are determined such that applicable limit are met. Therefore, the NRC staff finds the proposed change acceptable.

The NRC staff reviewed the licensee's proposed TS changes and found that the licensee appropriately implemented the revised MCPR SL as discussed in this SE.

3.6 NRC Staff Conclusion

The NRC staff reviewed the licensee's proposed TS changes and determined that the proposed SL associated with TS 2.1.2 was calculated in a manner consistent with the process described in TSTF-564, Revision 2, and was, therefore, acceptably modified to suit the revised definition of the MCPR SL. Under the new definition, the MCPR SL will continue to protect the fuel cladding against the uncontrolled release of radioactivity by preventing the onset of boiling transition, thereby fulfilling the requirements of 10 CFR 50.36(c)(1) for SLs. The MCPR OL in LCO 3.2.3

remains unchanged and will continue to meet the requirements of 10 CFR 50.36(c)(2), as discussed in Section 2.3 of this SE, by ensuring that no fuel damage results during normal operation and AOOs. The NRC staff finds that the proposed change to Hope Creek LCO 3.4.1.1 is acceptable because referring to the MCPR limits in the COLR instead of specifying the limits in TS does not change the applicability of the LCO. In addition, the changes to Hope Creek TS 6.9.1.9 as proposed in the traveler are acceptable; upon adoption of the revised MCPR SL, the COLR will be required to contain the $MCPR_{99.9\%}$, supporting the determination of the MCPR OL using current methodologies.

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 31, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component 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, and there has been no public comment on such finding (84 FR 23078). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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.

Principal Contributor: F. Forsaty

Date: September 19, 2019

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT NO. 219 REGARDING REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-564, "SAFETY LIMIT MCPR [MINIMUM CRITICAL POWER RATIO]" (EPID L-2019-LLA-0084) DATED SEPTEMBER 19, 2019

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