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

March 21, 2017

Mr. Peter P. Sena, III
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 TO DELETE TECHNICAL SPECIFICATION ACTION STATEMENT 3.4.2.1.B ASSOCIATED WITH STUCK OPEN SAFETY/RELIEF VALVES (CAC NO. MF7709)

Dear Mr. Sena:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 203 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 11, 2016, as supplemented by letter dated December 13, 2016.

The amendment revises the Hope Creek Generating Station TS requirements by deleting TS Action Statement 3.4.2.1.b concerning stuck open safety/relief valves. In addition, TS 3.6.2.1 Action Statements regarding suppression chamber water temperature are revised to align with NUREG-1433, Revision 4.0, "Standard Technical Specifications – General Electric BWR/4 Plants."

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Carleen J. Parker

Carleen J. Parker, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 203 to Renewed License No. NPF-57
2. Safety Evaluation

cc w/enclosures: Distribution via 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. 203
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 May 11, 2016, as supplemented by letter dated December 13, 2016, 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. 203, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James Danna

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

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: March 21, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 203

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
3/4 4-7	3/4 4-7
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13

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) Maximum Power Level

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 203, 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.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 13 of the following reactor coolant system safety/relief valves shall be OPERABLE*# with the specified code safety valve function lift settings:**

4	safety-relief valves @ 1108 psig ±3%
5	safety-relief valves @ 1120 psig ±3%
5	safety-relief valves @ 1130 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety valve function of two or more of the above listed fourteen safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. Deleted

c. Deleted

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* SRVs which perform as ADS function must also satisfy the OPERABILITY requirements of Specification 3.5.1, ECCS-Operating.

** The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

\# SRVs which perform a low-low set function must also satisfy the OPERABILITY requirements of Specification 3.4.2.2, Safety/Relief Valves Low-Low Set Function.

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. With an indicated water level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. 1. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER and testing that adds heat to the suppression pool is not being performed, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 105°F and THERMAL POWER greater than 1% of RATED THERMAL POWER during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber.

3. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

LIMITING CONDITION FOR OPERATION (continued)

ACTION: (Continued)

4. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits in accordance with the Surveillance Frequency Control Program.

b. In accordance with the Surveillance Frequency Control Program in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

   1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

   2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

      a) Suppression chamber average water temperature to be less than or equal to 110°F.

c. At least once per 30 minutes in OPERATIONAL CONDITION 3 following a scram with suppression chamber average water temperature greater than 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

d. By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 177°F and reactor coolant system pressure greater than 100 psig.

e. In accordance with the Surveillance Frequency Control Program by a visual inspection of the accessible interior and exterior of the suppression chamber.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By application dated May 11, 2016,[1] as supplemented by letter dated December 13, 2016,[2] PSEG Nuclear LLC (PSEG or the licensee) requested changes to the Hope Creek Generating Station (Hope Creek) Technical Specifications (TSs).

The requested changes would revise the Hope Creek TS requirements by deleting TS Action Statement 3.4.2.1.b concerning stuck open safety/relief valves. In addition, TS 3.6.2.1 Action Statements regarding suppression chamber water temperature would be revised to align with NUREG-1433, Revision 4.0, "Standard Technical Specifications – General Electric BWR/4 Plants."[3]

On July 19, 2016, the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (81 FR 46965) for the proposed amendment. Subsequently, by letter dated December 13, 2016, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register.* Accordingly, the NRC published a second proposed NSHC determination in the *Federal Register* on January 17, 2017 (82 FR 4932), which superseded the original notice in its entirety.

2.0 REGULATORY EVALUATION

The NRC staff reviewed the proposed TS changes against the regulatory requirements and guidance listed below to ensure that there is reasonable assurance that the systems and components affected by the proposed TS changes will perform their safety functions.

2.1 Applicable Regulatory Requirements

The Commission's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications."

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[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16132A374
[2] ADAMS Accession No. ML16348A017
[3] ADAMS Accession No. ML12104A192

The regulation requires that the TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements, (4) design features, and (5) administrative controls. The regulation does not specify the particular requirements to be included in plant TSs.

The regulations in 10 CFR 50.36(c)(2)(i), state, in part, that "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." More specifically, 10 CFR 50.36(c)(2)(ii) states:

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> ...
>
> (B) *Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

2.2 Regulatory Guidance

The NRC staff identified the following regulatory guidance as being applicable to the proposed amendments:

- NUREG-1433, Revision 4, "Standard Technical Specifications – General Electric BWR/4 Plants," and
- NUREG-0123, Revision 4, "Standard Technical Specifications for General Electric Boiling Water Reactors."

3.0 TECHNICAL EVALUATION

3.1 Proposed Technical Specification Changes

The May 11, 2016, license amendment request (LAR) proposes to delete TS Action Statement 3.4.2.1.b concerning main steam stuck open safety/relief valves. TS Action Statement 3.4.2.1.b currently states:

> With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open safety relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

This language would be deleted in its entirety.

In addition, the licensee's December 13, 2016, supplement proposes to revise TS 3.6.2.1 Action b regarding suppression chamber water temperature to align with NUREG-1433, Revision 4. TS 3.6.2.1 Action b. currently states:

> b. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL

POWER, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

Proposed TS 3.6.2.1 Action b, with bold (added) and strikethrough (deleted) text indicating changes, states:

b. **1.** With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER **and testing that adds heat to the suppression pool is not being performed**, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~, except, as permitted above:~~

2. With the suppression chamber average water temperature greater than 105°F **and THERMAL POWER greater than 1% of RATED THERMAL POWER** during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber ~~and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours~~.

3. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

4. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

Note: NUREG-1433 no longer uses the term "suppression chamber average water temperature." For this safety evaluation, the suppression chamber average water temperature is also referred to as the suppression pool average water temperature.

3.2 NRC Staff Technical Evaluation

3.2.1 TS Action Statement 3.4.2.1.b

The licensee proposed to delete TS Action Statement 3.4.2.1.b, which requires the reactor mode switch be placed in the Shutdown position if a stuck open safety/relief valve(s) cannot be closed within 2 minutes or if the suppression pool average water temperature is 110 degrees Fahrenheit (°F) or greater.

The Hope Creek Updated Final Safety Analysis Report (UFSAR), Section 15.1.4, "Inadvertent Main Steam Relief Valve Opening," discusses the inadvertent opening of a main steam relief valve. Specifically, the UFSAR states the following:

> The plant operator must "reclose" the valve and check that reactor and turbine-generator output return to normal. If the valve cannot be closed, plant shutdown must be initiated.
>
> ...
>
> The opening of an SRV allows steam to be discharged into the suppression chamber. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.
>
> The pressure regulator senses the nuclear system pressure decrease and closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value, and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. Minimum critical power ratio (MCPR) is essentially unchanged, and therefore the safety limit margin is unaffected.

In its May 11, 2016, LAR, the licensee stated that:

> ...the transient resulting from an inadvertent main steam safety/relief valve (SRV) opening is a mild depressurization that is within the range of normal load following, and, therefore, has no significant effect on reactor coolant pressure boundary (RCPB) and containment design pressure limits. While the consequence of this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there is no exposure to operating personnel. This transient does not result in an uncontrolled release to the environment. In addition, there is no impact on the safety function of the SRVs to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code or to provide Automatic Depressurization System (ADS) functions.

The licensee also stated that "[t]he compensatory actions for failure to meet the LCO for suppression pool temperature are provided in TS 3.6.2.1, "Depressurization Systems –

Suppression Chamber" and plant procedures." TS Action 3.4.2.1.b is redundant to TS 3.6.2.1, limiting condition for operation, Action b.2, which also directs the operator to place the reactor mode switch in SHUTDOWN to scram the reactor if the suppression pool average water temperature reaches 110 °F, except for the 2-minute requirement.

The licensee stated that "[t]he two minute requirement represents detailed methods of responding to an event and not necessarily a compensatory action for failure to meet the limiting condition for operation (LCO) for safety/relief valves," and that elimination of "the requirement to scram the reactor after two (2) minutes can allow operators to devote more time to addressing problems with the stuck open valve and preparing for post-scram actions and depressurization."

The licensee stated in its December 13, 2016, supplement that no documentation or references were identified related to Hope Creek's current licensing basis that require inclusion of the 2-minute SRV closure requirement. The current TS Action Statement 3.4.2.1.b requiring the reactor mode switch be placed in the shutdown position if a stuck open main steam SRV is not closed within 2 minutes has always been in the Hope Creek TSs.

NRC staff note that the 2-minute operator action time was in a single revision of the original Standard Technical Specifications (STSs) for General Electric Boiling Water Reactors (BWRs) (NUREG-0123, Revision 4). The current BWR STSs (NUREG-1433, Revision 4) has no time restrictions associated with the inadvertent opening of one or more SRVs. The BWR STSs have Action Statements that ensure plant safety, including if plant operations result in increasing suppression pool temperature.

An operating experience (OE) review (since September 1992) determined that there are three events related to a stuck open main steam SRV at boiling water reactors similar to Hope Creek: Brunswick Steam Electric Plant, Unit 2; Limerick Generating Station, Unit 1; and LaSalle County Station, Unit 1. A review of the OE indicated that in two instances, a manual shutdown was initiated after the open SRV could not be closed within 2 minutes in accordance with TSs. In one instance, the TS requirements allowed additional time (until the high suppression pool temperature was reached) to attempt to close the SRV and to reduce reactor power before initiating a manual shutdown. In each instance, the safety consequences of the event were negligible.

As stated above, a stuck open main steam SRV is a mild depressurization event that is within the range of normal load following. During this scenario, the main concern is the rising suppression pool temperature. Actions for TS 3.6.2.1 already provide operators with appropriate direction for response to a suppression pool high temperature caused by a stuck open SRV. The NRC staff finds elimination of the 2-minute requirement appropriate because LCO 3.6.2.1 and plant procedures provide operators with appropriate direction for response to a suppression pool high temperature caused by a stuck open SRV. Providing specific direction to place the reactor mode switch in the Shutdown position if the stuck open SRV is not closed within 2 minutes does not provide additional plant protection beyond what is provided for in TS 3.6.2.1. Therefore, the NRC staff finds it acceptable to delete TS Action Statement 3.4.2.1.b.

3.2.2 TS Action Statement 3.6.2.1.b.2

Hope Creek Action Statement TS 3.6.2.1.b.2, directs the operator to place the reactor mode switch in Shutdown if the suppression pool average water temperature reaches 110 °F. The current TS 3.6.2.1, Action Statement b.2 is read to apply when the suppression chamber average water temperature is greater than 95 °F and thermal power is greater than 1 percent of rated thermal power. In its December 13, 2016, supplement, the licensee stated that:

> In Modes 1 and 2, with THERMAL POWER greater than 1% of RATED THERMAL POWER, TS 3.6.2.1 Action b.2 requires the reactor mode switch to be placed in the Shutdown position, regardless of the time elapsed since suppression chamber water temperature exceeded 95°F. The Hope Creek containment control emergency operating procedure requires operators to runback reactor recirculation flow and to initiate a manual scram before suppression pool temperature exceeds 110°F, regardless of power level.

However, the licensee recognizes the current Hope Creek TSs allow for ambiguity in implementing TS Action 3.6.2.1 Action b and, therefore, the licensee proposed in its December 13, 2016, supplement to revise TS 3.6.2.1.b to align with the current BWR STSs (NUREG-1433, Revision 4). The proposed changes will remove any ambiguity.

The NRC staff reviewed the change and determined it was acceptable. Specifically, the proposed changes to TS 3.6.2.1.b are in alignment with the intent of NUREG-1433 in which the reactor mode switch is placed in the Shutdown position before the suppression pool temperature exceeds 110 °F, regardless of rated thermal power while in Modes 1, 2, and 3. NUREG-1433, Revision 4, was previously approved by the NRC. Therefore, the staff finds the proposed TSs meet the regulatory requirements of 10 CFR 50.36 and guidance found in NUREG-1433.

3.3 Summary of NRC Staff Evaluation

The NRC staff finds elimination of TS 3.4.2.1 Action b and the proposed modifications to TS 3.6.2.1 Action b acceptable. Specifically, the NRC staff finds the following:

- The intent of the proposed TS 3.4.2.1 change is to remove the requirement to scram the reactor after 2 minutes of a stuck open main steam SRV when the suppression pool average water temperature in not challenged in Modes 1, 2 and 3.

- Providing specific direction to close the stuck open main steam relief valve or shutdown within 2 minutes does not provide additional plant protection beyond what is provided in TS 3.6.2.1.

- The proposed changes to TS 3.6.2.1 clarify the thermal power requirements related to suppression pool average water temperature heatup in Modes 1, 2, and 3 during normal operation and testing that adds heat to the suppression chamber.

- Actions for TS 3.6.2.1 provide operators with appropriate direction for response to a suppression pool high temperature caused by a stuck open relief valve.

- Reasonable assurance that the proposed changes meet 10 CFR 50.36 and are in alignment with NUREG-1433, Revision 4, which has been previously approved by the NRC.

- The proposed TS changes to delete TS 3.4.2.1 Action b and modify TS 3.6.2.1 Action b meet the regulatory requirements as described in Section 2.0 of this safety evaluation.

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 February 7, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement 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 (82 FR 4932). 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.

Principal Contributors: R. Beaton
L. Wheeler

Date: March 21, 2017

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT TO DELETE TECHNICAL SPECIFICATION ACTION STATEMENT 3.4.2.1.B ASSOCIATED WITH STUCK OPEN SAFETY/RELIEF VALVES (CAC NO. MF7709) DATED MARCH 21, 2017

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