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

February 25, 2013

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REQUEST TO CORRECT TECHNICAL SPECIFICATION AND FACILITY OPERATING LICENSE EDITORIAL ITEMS (TAC NO. ME8113)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 193 to Renewed Facility Operating License (FOL) No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications and FOL in response to your application dated March 1, 2012, as supplemented by letter dated December 21, 2012.[1]

The amendment makes various changes that fall into one of four categories: (1) correct typographical errors, (2) delete historical requirements that have expired, (3) make editorial changes to correct errors and/or omissions from previous license amendment requests, or (4) update component lists to reflect current plant design.

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

Sincerely,

John D. Hughey

John D. Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 193 to
   Renewed License No. NPF-57
2. Safety Evaluation

cc w/encls: Distribution via ListServ

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[1] Agencywide Documents Access and Management System Accession Nos. ML12062A148 and ML12356A474, respectively.

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

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 193
Renewed License No. NPF-57

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

    A. The application for amendment filed by PSEG Nuclear LLC dated March 1, 2012, as supplemented by letter dated December 21, 2012, 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 as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed 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. 193, 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

Meena K. Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License
and the Technical Specifications

Date of Issuance: February 25, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 193

RENEWED FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following pages of the Renewed Facility Operating License with the revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Page 3	Page 3
Page 4	Page 4
Page 5	Page 5
Page 6	Page 6
Page 10	Page 10
Page 11	Page 11
Page 12	Page 12
Page 13	Page 13

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
xi	xi
3/4 1-9	3/4 1-9
3/4 3-11	3/4 3-11
3/4 4-12	3/4 4-12
3/4 6-13	3/4 6-13
3/4 8-25	3/4 8-25
3/4 8-28	3/4 8-28
3/4 9-11	3/4 9-11
6-21	6-21

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. 193, 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) Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No. 4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

(4) Inservice Inspection (Section 6.6, SER; Sections 5.2.4.3 and 6.6.3, SSER No. 5)

a. DELETED

b. Pursuant to 10 CFR 50.55a(a)(3) and for the reasons set forth in Sections 5.2.4.3 and 6.6.3 of SSER No. 5, the relief identified in the PSE&G submittal dated November 18, 1985, as revised by the submittal dated January 20, 1986, requesting relief from certain requirements of 10 CFR 50.55a(g) for the preservice inspection program, is granted.

(5) Solid State Logic Modules

PSEG Nuclear LLC shall continue, for the life of the plant, a reliability program to monitor the performance of the Bailey 862 SSLMs installed at Hope Creek Generating Station. This program should obtain reliability data, failure characteristics, and root cause of failure of both safety-related and non-safety-related Bailey 862 SSLMs. The results of the reliability program shall be maintained on-site and made available to the NRC upon request.

(6) Fuel Storage and Handling (Section 9.1, SSER No. 5)

a. No more than a total of three (3) fuel assemblies shall be out of approved shipping containers, NRC-approved dry spent fuel storage systems, fuel assembly storage racks or the reactor at any one time.

b. The above three (3) fuel assemblies as a group shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and the storage rack array.

c. Fresh Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three (3) containers high.

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(7) Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)

DELETED

(9) Emergency Planning (Section 13.3, SSER No. 5)

DELETED

(10) Initial Startup Test Program (Section 14, SSER No. 5)

DELETED

(11) Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)

The facility shall not be operated with a rated thermal power feedwater temperature less than 329.6°F for the purpose of extending the normal fuel cycle.

(12) Detailed Control Room Design Review (Section 18.1, SSER No. 5)

DELETED

(13) Safety Parameter Display System (Section 18.2, SSER No. 5)

DELETED

(14) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 135, are hereby incorporated into this renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(15) PSE&G to PSEG Nuclear LLC License Transfer Conditions

a. PSEG Nuclear LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application, the requirements of the Order Approving Transfer of License and Conforming Amendment, dated February 16, 2000, and the related Safety Evaluation dated February 16, 2000.

b. The decommissioning trust agreement shall provide that:

1) The use of assets in both the qualified and non-qualified funds shall be limited to expenses related to decommissioning of the unit as defined by the NRC in its regulations and issuances, and as provided in the unit's renewed license and any amendments thereto. However, upon completion of decommissioning, as defined above, the assets may be used for any purpose authorized by law.

2) Investments in the securities or other obligations of PSE&G or affiliates thereof, or their successors or assigns,

(21) Vibration Acceptance Criteria for SRVs

DELETED

(22) Steam Dryer

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the stream dryer).

1. DELETED

2. PSEG Nuclear LLC shall implement the following actions for the initial power ascension at power levels above 3339 MWt to 3840 MWt:

   a. DELETED

   b. DELETED

   c. DELETED

   d. DELETED

   e. PSEG Nuclear LLC shall revise plant procedures to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure, and to reflect consistency of the facility's steam dryer inspection program with BWRVIP-139.

f. DELETED

g. DELETED

3. DELETED

4. DELETED

5. DELETED

6. DELETED

(23) Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

(24) PSEG Nuclear LLC may make changes to the programs and activities described in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(25) Appendix A of NUREG-2102, "Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station," dated June 2011, and the licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on May 19, 2011, describes certain future programs and activities to be completed before the period of extended operation. PSEG Nuclear LLC shall complete these activities no later than April 11, 2026, and shall notify the NRC in writing when implementation of these activities is complete.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

---NOTE---
Separate condition entry is allowed for each control rod
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a. In OPERATIONAL CONDITIONS 1 or 2:

   1. With one control rod scram accumulator inoperable and reactor pressure ≥ 900 psig, within 8 hours,

      a) Restore the inoperable accumulator to OPERABLE status, or

      b) Declare the associated control rod scram time "slow"***, or

      c) Insert the associated control rod, declare the associated control rod inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. With two or more control rod scram accumulators inoperable and reactor pressure ≥ 900 psig,

      a) Within 20 minutes of discovery of this condition concurrent with charging water pressure < 940 psig, restore charging water header pressure to ≥ 940 psig otherwise place the mode switch in the shutdown position**, and

      b) Within one hour, declare the associated control rod scram time "slow"***, or

      c) Within one hour insert the associated control rods, declare the associated control rods inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

      Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

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* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

*** Only applicable if the associated control rod scram time was within the limits of Table 3.1.3.3-1 during the last scram time Surveillance. Rods that are already considered "slow" should be declared inoperable and fully inserted.

TABLE 3.3.2-1
ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [a]	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, Level 2	2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2[j]	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	8, 9, 10 11, 12, 13, 14, 15, 16, 17, 18	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level – Low Low, Level 2	19[c]	2	1, 2, 3 and *	26
b. Drywell Pressure - High	19[c]	2[j]	1, 2, 3	26
c. Refueling Floor Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	29
d. Reactor Building Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	28
e. Manual Initiation	19[c]	1	1, 2, 3 and *	26

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric gaseous radioactivity in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate in accordance with the Surveillance Frequency Control Program, and

c. Monitoring the drywell air coolers condensate flow rate in accordance with the Surveillance Frequency Control Program, and

d. Monitoring the drywell pressure in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage), and

e. Monitoring the reactor vessel head flange leak detection system in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage), and

f. Monitoring the drywell temperature in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to the IST Program and verifying the leakage of each valve to be within the specified limit:

a. In accordance with the Surveillance Frequency Control Program, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program.

LIMITING CONDITION FOR OPERATION (continued)

ACTION: (Continued)

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.

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.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and verifying that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-252063	10B252	IM TM	Drywell Equip Drain Sump Pump 1AP267
52-252064	10B252	IM TM	Drywell Floor Drain Sump Pump 1CP267
52-252073	10B252	IM TM	Feedwater Inlet A Shutoff 1AE-HV-F011A
52-262021	10B262	IM TM	Drywell Cooler A Fan 1A2V212
52-262022	10B262	IM TM	Drywell Cooler B Fan 1B2V212
52-262031	10B262	IM TM	Drywell Cooler C Fan 1C2V212
52-262032	10B262	IM TM	Drywell Cooler D Fan 1D2V212
52-262041	10B262	IM TM	Drywell Cooler E Fan 1E2V212
52-262042	10B262	IM TM	Drywell Cooler F Fan 1F2V212
52-262051	10B262	IM TM	Drywell Cooler G Fan 1G2V212
52-262052	10B262	IM TM	Drywell Cooler H Fan 1H2V212
52-262063	10B262	IM TM	Drywell Equip Drain Sump Pump 1BP267
52-262064	10B262	IM TM	Drywell Floor Drain Sump Pump 1DP267
52-253021	10B253	IM TM	Recirc Pump 1BP201 Suction Valve 1BB-HV-F023B
52-253031	10B253	IM TM	Recirc Pump 1BP201 Discharge Valve 1BB-HV-F031B
52-253053	10B253	IM TM	Reactor Vessel Head Vent Inboard Isolation 1BB-HV-F001

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 2 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least at its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during handling of fuel assemblies or control rods within the reactor pressure vessel.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 DELETED

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193

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 March 1, 2012, as supplemented by letter dated December 21, 2012,[1] PSEG Nuclear, LLC (the licensee) submitted a license amendment request (LAR) for changes to the Hope Creek Generating Station (Hope Creek), Technical Specifications (TSs) and Facility Operating License (FOL). The supplement 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 or Commission) staff's original proposed no significant hazards determination as published in the *Federal Register* on April 3, 2012 (77 FR 20075).

However, in the supplement, the licensee withdrew 2 of the requests submitted in the LAR. The first involved Item 9 located on Page 4 of Attachment 1 to the LAR which requested removal of the requirement for an annual report for main steamline safety/relief valve challenges located in TS 6.9, "Reporting Requirements," on pages 6-17 and 6-18. The second involved Item 18 located on Page 8 of Attachment 1 to the LAR which requested removal of FOL condition 2.C.(22), which includes FOL condition 2.C.(22).2.e, which is located on Page 11 of the FOL. These changes are explained in more detail in Sections 3.9 and 3.12 of this safety evaluation.

The requested changes would make various editorial and other changes to the TSs and FOL, including correction of typographical and format errors, and deletion of outdated requirements that have expired.

2.0 REGULATORY EVALUATION

The NRC'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." This regulation requires that the TSs include items in the following five specific categories: 1) safety limits, limiting safety system settings, and limiting control settings; 2) limiting conditions

[1] Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12062A148 and ML12356A474, respectively.

Enclosure

for operation (LCOs); 3) surveillance requirements (SRs); 4) design features, and 5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," (Final Policy Statement) which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953, July 19, 1995). Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

> Criterion 1
> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> 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.
>
> Criterion 3
> A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 4
> A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As discussed in the *Federal Register* notice for the final rule dated July 19, 1995 (60 FR 36955):

> LCOs that do not meet any of the criteria, and their associated actions and surveillance requirements, may be proposed for relocation from the technical specifications to licensee-controlled documents, such as the FSAR [Final Safety Analysis Report]. The criteria may be applied to either standard or custom technical specifications.

3.0 TECHNICAL EVALUATION

3.1 TS Index - Delete word BASES from Header

The word "BASES" was added to the header of page xi of the Hope Creek TS in error with the issuance of Amendment 134, dated October 3, 2011.[2] The NRC staff concludes that since page

[2] ADAMS Accession No. ML012600176.

xi is not a Bases page, the word should be removed from the header. The NRC staff concludes that this change is corrective in nature, and therefore is acceptable.

3.2 TS LCO 3.1.3.5 Action a - Correct Typographical Error

Currently, LCO 3.1.3.5 Action a concludes with,

> Otherwise be in at least HOT SHUTDOWN with the next 12 hours.

The licensee proposes to change the word "with" in the previous sentence, to "within" to correct the typographical error. The NRC staff concludes that this editorial change is corrective in nature, and therefore is acceptable.

3.3 TS Table 3.3.2-1 – Delete Reference to Valve Actuation Group 1

Valve Actuation Group 1 is listed as one of the Valve Actuation Groups Operated by Signal from Trip Function 1, in Table 3.3.2-1 in the Hope Creek TS. Valve Activation Group 1 refers to the Main Steam Line Isolation Valve (MSIV) Seal System inboard supply valves. The requirement for the MSIV Seal System was deleted when the NRC staff issued Amendment 134, and Valve Actuation Group 1 should have been deleted with this amendment. The NRC staff concludes that since this reference should have been removed previously, this change is corrective in nature, and is therefore acceptable.

3.4 SR 4.4.3.2.2 - Delete Outdated Footnote

SR 4.4.3.2.2 contains a footnote that references a leak test extension to be applied during the first refueling outage. It goes on to state that during that specific one time test interval, the requirements of Section 4.0.2 are not applicable. Since the first refueling outage has long since been completed, the licensee proposes to remove this footnote.

The NRC staff concludes that this change is acceptable because the footnote does not contain any requirements applicable to current or future operations.

3.5 TS LCO 3.6.2.1 Action c – Correct Typographical Error

TS LCO 3.6.2.1 Action c currently states, in part,

> With one drywell-to-suppression chamber bypass leakage in excess of the limit

The licensee proposes to change the word "one", in the previous sentence, to the word "the" to correct the typographical error. Drywell-to-suppression chamber bypass leakage is not an item but a type of leakage, therefore, the action statement does not make sense as currently written. This error was introduced into the Hope Creek TS with the implementation of Amendment 133, dated October 3, 2001.[3] The NRC staff concludes that this change is corrective in nature, and therefore is acceptable.

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[3] ADAMS Accession No. ML011730396.

3.6 SR 4.8.4.1.a.2 – Correct Typographical Error

The fourth sentence of surveillance requirement a.2 states,

> The instantaneous element shall be tested by injecting a current in excess of 120% the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay.

The licensee proposes to add the word "of" to the above sentence after 120% to correct the typographical error. This typographical error has been present since the original Hope Creek TSs were issued. The NRC staff concludes that this change is administrative in nature, and therefore is acceptable.

3.7 TS Table 3.8.4.1-1 – Remove Reference to Disconnected Equipment

Table 3.8.4.1-1 contains a reference to Circuit Breaker No. 52-253012, which powered a Recirculation Pump Motor Hoist that the licensee replaced with a manually actuated model in 2007. When the hoist was replaced, the breaker was spared and determed. Since the circuit breaker has been disconnected and will remain disconnected, the LCO for primary containment penetration conductor overcurrent protection devices is no longer applicable and the licensee proposes to remove all references to it from the table.

The NRC staff concludes that this change is acceptable because the LCO no longer applies to Circuit Breaker No. 52-253012.

3.8 SR 4.9.8 – Correct Typographical Error

SR 4.9.8 states, in part,

> The reactor vessel water level shall be determined to be at least its minimum required depth

The licensee proposes to add the word "at" to the above sentence after "least" to correct the typographical error. This typographical error has been present since the original Hope Creek TSs were issued. The NRC staff concludes that this clarifying change is corrective in nature, and therefore is acceptable.

3.9 TS 6.9 – Delete Reporting Requirements

On page 4 of Attachment 1 to letter dated March 1, 2012, the licensee requests to remove the requirement for an annual report for main steamline safety/relief valve challenges from TS Administrative Controls Section 6.9.1.5. The licensee stated that the requirement should have been deleted during the implementation of Amendment 161, which was issued with letter dated

January 11, 2006,[4] but it was not due to an oversight. However, after the licensee's review of the RAI dated December 5, 2012, this specific proposed change was withdrawn by supplement dated December 21, 2012.

3.10 TS 6.10 – Delete Second "Special Report" Heading

TS 6.10 currently contains the heading "Special Reports" twice on page 6-21. The first "Special Reports" heading on page 6-21 is placed above TS 6.9.2, which outlines how special reports shall be submitted to the NRC. The second heading is above TS 6.10.2, which outlines which records shall be retained for at least 5 years. The licensee proposes to remove the second "Special Reports" heading because it appears that the heading was placed there by mistake.

This error has been present since the original Hope Creek TSs were issued. The NRC staff concludes that this change is corrective in nature, and therefore is acceptable.

3.11 FOL Condition 2.C.(13) – Delete Outdated License Condition

In Item 16 of Attachment 1 on page 7 of the LAR, the licensee requests to remove FOL License Condition 2.C.(13), "Safety Parameter Display System [SPDS] (Section 18.2, SSER No.5)," from the license for Hope Creek. Specifically, FOL License Condition 2.C.(13) requires PSEG to add four certain parameters to the SPDS and have them operational prior to the earlier of 90 days after restart from the first refueling outage, or July 12, 1988.

By letter dated December 5, 2012, the NRC staff asked the licensee to provide the rational for requesting this change with this LAR because the change did not appear to be editorial in nature as characterized in the submittal. In the supplement dated December 21, 2012, the licensee stated that the NRC Staff Safety Evaluation for Amendment No. 13, by letter dated November 24, 1987,[5] identifies FOL Condition 2.C.(13) as a scheduler requirement to make specific parameters operational in SPDS within a specified time frame. Subsequently, the parameters were made operational in the SPDS in 1988. The NRC staff has confirmed that the Safety Evaluation associated with Amendment No.13 characterizes FOL Condition 2.C.(13) as a scheduler requirement. The NRC staff concludes that this change is an update to the FOL that reflects the current licensing and design basis of the facility, and therefore is acceptable.

3.12 FOL Condition 2.C.(22) – Delete Outdated License Condition

In Item 18 of Attachment 1 on page 8 of the LAR, the licensee requested to remove FOL Condition 2.C.(22), "Steam Dryer," from the license for Hope Creek. The individual license conditions listed in FOL 2.C.(22) placed specific requirements for actions to be completed during power accession for the initial operation of the facility at power levels associated with an extended power uprate (EPU). The actions related to the development of startup test procedures as well as monitoring, data collection and analysis requirements. All but one of these license conditions actions were completed by the end of the second refueling outage

[4] ADAMS Accession No. ML060050496.
[5] ADAMS Accession No. ML011760027.

following the implementation of the EPU, and are now outdated. The exception is FOL License Condition 2.C.(22).2.e, which states:

> PSEG Nuclear LLC shall revise plant procedures to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with BWRVIP-139 [Steam Dryer Inspection and Flaw Evaluation Guidelines];[6] and to identify the NRC Project Manager for the facility as the point of contact for providing power ascension testing information during power ascension.

The requirement to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure in plant procedures is an on-going obligation per FOL Condition 2.C.(22).2.e. The NRC staff reminded the licensee of this requirement by RAI letter dated December 5, 2012. In the supplement dated December 21, 2012, the licensee states:

> FOL 2.C.(22) is a historical requirement related to EPU implementation that can be deleted, with the exception of part of 2.C.(22).2.e, which is still applicable. The requirement to identify the NRC project manager as the point of contact for power ascension testing is a historical requirement that was completed and may be removed.

In its supplement dated December 21, 2012, the licensee proposes to remove FOL 2.C.(22), with the exception of 2.C.(22).2.e, to be revised as follows:

> PSEG Nuclear LLC shall revise plant procedures to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure, and to reflect consistency of the facility's steam dryer inspection program with BWRVIP-139.

The licensee's proposed changes to the amendment request, as described above, recognize the requirement to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure in plant procedures is an on-going obligation per FOL Condition 2.C.(22).2.e. Actions required by the other conditions listed in FOL 2.C.(22) were completed, as required, during initial EPU and are now outdated license conditions. Therefore, the NRC staff concludes that the changes to FOL Condition 2.C.(22) update the FOL to: 1) delete outdated requirements; and 2) reflect the current licensing and design basis of the facility, are acceptable.

3.13 <u>FOL Conditions 2.C.(4).a and 2.C.(8) through 2.C.(10), 2.C.(12), and 2.C.(21) – Delete Completed License Conditions</u>

Hope Creek FOL Conditions 2.C.(4).a, 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(12), and 2.C.(21) are all related to actions required within specific time frames or dates that have all now passed. The licensee proposes to delete these license conditions since they are completed and are no longer applicable.

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[6] ADAMS Accession No. ML101270122.

FOL 2.C.(4).a is a one-time requirement to submit the inservice inspection program to the NRC, which was completed in 1986. FOL 2.C.(8) required that NRC approval be granted for the solid waste program and was completed in 1987. FOL 2.C.(9) related to ensuring that timely approval of State off-site emergency plans was addressed. The final State emergency response plan was approved in 1998. Finalization of the initial Startup Test Program in 1987 completed FOL 2.C.(10). FOL 2.C.(12) required submission of the Detailed Control Room Design Review Summary Reports to the NRC and that temporary zone markings be established during initial startup. The submissions required by FOL 2.C.(12) were completed in 1986 and 1987. Finally, 2.C.(21) required that technical information regarding safety relief valves be provided to the NRC specifically related to an extended power uprate. This information was submitted to the NRC in 2008.

The NRC staff concludes that the deletion of Hope Creek FOL Conditions 2.C.(4), 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(12), and 2.C.(21) is acceptable because they do not contain any requirements that are applicable to current or future operations.

3.14 Technical Evaluation Conclusion

Based on the considerations in Sections 3.1 through 3.13 above, the NRC staff has concluded that the proposed amendment makes various changes to the Technical Specifications (TS) and Facility Operating License (FOL) including: 1) correction of typographical errors; 2) deletion of outdated requirements that have expired; 3) corrections of errors or omissions from previous license amendment requests; and 4) updating of component lists to reflect current plant design. Therefore, the NRC staff concludes that the proposed amendment, as supplemented, 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 amendments. 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 (77 FR 20075; April 3, 2012). The amendment also makes editorial, corrective and other minor revisions. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(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 Contributors: J. Whited
J. Hughey

Date: February 25, 2013

February 25, 2012

Mr. Thomas Joyce
President and Chief Nuclear Officer
PSEG Nuclear LLC
P.O. Box 236, N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REQUEST TO CORRECT TECHNICAL SPECIFICATION AND FACILITY OPERATING LICENSE EDITORIAL ITEMS (TAC NO. ME8113)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 193 to Renewed Facility Operating License (FOL) No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the Technical Specifications and FOL in response to your application dated March 1, 2012, as supplemented by letter dated December 21, 2012.[1]

The amendment makes various changes that fall into one of four categories: (1) correct typographical errors, (2) delete historical requirements that have expired, (3) make editorial changes to correct errors and/or omissions from previous license amendment requests, or (4) update component lists to reflect current plant design.

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

Sincerely,
/ra/
John D. Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 193 to
   Renewed License No. NPF-57
2. Safety Evaluation

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[1] Agencywide Documents Access and Management System Accession Nos. ML12062A148 and ML12356A474, respectively.

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