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

October 19, 2010

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: SURVEILLANCE REQUIREMENTS FOR INSERVICE INSPECTION AND INSERVICE TESTING (TAC NO. ME2560)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 185 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications (TSs) and Facility Operating License in response to your application dated November 4, 2009.

The amendment revises the TSs to: (1) delete TS 4.0.5, which pertains to surveillance requirements for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable; and (4) revise TS 6.10.3.h to reflect the deletion of the ISI Program from the TSs.

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

Sincerely,

Richard B. Ennis, Senior 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. 185 to
   License No. NPF-57
2. Safety Evaluation

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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. 185
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 November 4, 2009, 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. 185, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the 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

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

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: October 19, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the 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 0-3	3/4 0-3
3/4 0-4	- - -
3/4 1-20	3/4 1-20
3/4 4-12	3/4 4-12
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 5-4	3/4 5-4
3/4 5-5	3/4 5-5
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 6-18	3/4 6-18
3/4 7-11	3/4 7-11
- - -	6-16e
6-22	6-22

(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 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. 185, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the 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.

* 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.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified frequency shall be a failure to meet the Limiting Condition for Operation, except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within its specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met and the applicable Actions must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 4.0.3. When an LCO is not met due to Surveillances not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by:

   1. Verifying the continuity of the explosive charge.

   2. Determining that the available weight of sodium pentaborate is greater than or equal to 5,776 lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

   3. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to the IST Program, the minimum flow requirement of 41.2 gpm, per pump, at a pressure of greater than or equal to 1255 psig is met.

d. At least once per 18 months by:

   1. Initiating one of the standby liquid control system subsystem, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel and verifying that the relief valve does not actuate. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection subsystems shall be tested in 36 months.

   2. **Demonstrating that all heat traced piping between the storage tank and the injection pumps is unblocked and then draining and flushing the piping with demineralized water.

   3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized.

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* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

** This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

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 at least once per 8 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate at least once per 8 hours, and

c. Monitoring the drywell air coolers condensate flow rate at least once per 8 hours, and

d. Monitoring the drywell pressure at least once per 8 hours (not a means of quantifying leakage), and

e. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours (not a means of quantifying leakage), and

f. Monitoring the drywell temperature at least once per 24 hours (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. At least once per 18 months,** 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:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

** P.I.V. leak test extension to the first refueling outage is permissible for each RCS P.I.V. listed in Table 3.4.3.2-1, that is identified in Public Service Electric & Gas Company's letter to the NRC (letter No. NLR-N87047), dated April 3, 1987, as needing a plant outage to test. For this one time test interval, the requirements of Section 4.0.2 are not applicable.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to the IST Program.

REACTOR COOLANT SYSTEM

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than the ISI Program.

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the core spray system, the LPCI system, and the HPCI system:

a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

c) Verify the RHR System cross tie valves on the discharge side of the pumps are closed and power, if any, is removed from the valve operators.

2. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to the IST Program:

1. The two core spray system pumps in each subsystem together develop a flow of at least 6150 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥105 psi above suppression pool pressure.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥20 psid.

3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of 1000 psig when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. At least once per 18 months:

1. For the core spray system, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

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* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

2. For the HPCI system, verifying that:

   a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥200 psig, when steam is being supplied to the turbine at 200 + 15, -0 psig.**

   b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.

3. Performing a CHANNEL CALIBRATION of the CSS, and LPCI system discharge line "keep filled" alarm instrumentation.

4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 4.4 psid.

5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 1.0 psid.

d. For the ADS:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the Primary Containment Instrument Gas System low-low pressure alarm system.

2. At least once per 18 months:

   a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b) Verify that when tested pursuant to the IST Program, that each ADS valve is capable of being opened.

   c) Performing a CHANNEL CALIBRATION of the Primary Containment Instrument Gas System low-low pressure alarm system and verifying an alarm setpoint of 85 ± 2 psig on decreasing pressure.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 540 gpm on recirculation flow through the RHR heat exchanger (after consideration of flow through the closed bypass valve) and suppression pool spray sparger when tested pursuant to the IST Program.

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* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required RHR pumps develops a flow of at least 10,160 gpm on recirculation flow through the RHR heat exchanger (after consideration of flow through the closed bypass valve) and the suppression pool when tested pursuant to the IST Program.

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* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shall be determined to be within its limit when tested pursuant to the IST Program.

4.6.3.4 At least once per 18 months, verify that a representative sample of reactor instrumentation line excess flow check valves# actuates to the isolation position on a simulated instrument line break signal.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE*:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

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* Exemption to Appendix J of 10 CFR Part 50.

\# The reactor vessel head seal leak detection line (penetration J5C) is not required to be tested pursuant to this requirement.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

Note: LCO 3.0.4.b is not applicable to RCIC.

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to the IST Program by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

PROCEDURES AND PROGRAMS (Continued)

6.8.4.i INSERVICE TESTING PROGRAM

This Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

RECORD RETENTION (Continued)

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections.

i. Records of quality assurance activities required by the Quality Assurance Program.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff).

l. DELETED

m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATIONAL MANUAL and the PROCESS CONTROL PROGRAM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 185 TO FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By letter dated November 4, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093210172), PSEG Nuclear, LLC (PSEG or the licensee) submitted a request for changes to the Hope Creek Generating Station (HCGS) Technical Specifications (TSs). The proposed amendment would revise the TSs to: (1) delete TS 4.0.5, which pertains to surveillance requirements (SRs) for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable; and (4) revise TS 6.10.3.h to reflect the deletion of the ISI Program from the TSs. The new TS for the IST Program, TS 6.8.4.i, will indicate that the program will include testing frequencies applicable to the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), replacing the current reference to Section XI of the ASME Code specified in TS 4.0.5. In addition, TS 6.8.4.i would revise the requirements, currently contained in TS 4.0.5, regarding the applicability of the surveillance interval extension provisions of SR 4.0.2.

The licensee's application dated November 4, 2009, stated that the license amendment request proposes changes for consistency with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55(a)(f)(4) for IST of pumps and valves and removes TS requirements that are redundant to the requirements of 10 CFR 50.55a, "Codes and standards." The licensee's application also stated that the proposed changes are consistent with Nuclear Regulatory Commission (NRC or the Commission)-approved Technical Specification Task Force (TSTF) Travelers TSTF-479, "Changes to Reflect Revision to 10 CFR 50.55a," and TSTF-497, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less;" and NUREG-1433, Revision 3.0, "Standard Technical Specifications - General Electric Plants, BWR [Boiling Water Reactors]/4."

Enclosure

2.0 REGULATORY EVALUATION

The Commission's regulatory requirements related to the content of the TSs are set forth in 10 CFR 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 for operation; (3) SRs; (4) design features; and (5) administrative controls. Paragraph (c)(3) of 10 CFR 50.36 states that SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulations in 10 CFR 50.55a(f)(4) and (g)(4) establish the ASME Code edition and addenda to be used by licensees for performing IST of pumps and valves and ISI of components (including supports). Paragraphs 50.55a(f)(4)(ii) and (g)(4)(ii) require the use of the latest edition and addenda that has been incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the beginning of each 120-month interval. With respect to the requirements for IST, the ASME OM Code was initially incorporated by reference in 10 CFR 50.55a(b) in a final rule dated September 22, 1999 (64 FR 51370). Prior to the final rule, IST programs were required to meet the requirements of Section XI of the ASME Code.

Paragraph (f)(5)(ii) of 10 CFR 50.55a requires that, if a revised IST program for a facility conflicts with the TSs for that facility, the licensee shall apply to the Commission for amendment of the TSs to conform the TSs to the revised program. As discussed in a PSEG letter dated October 12, 2006, "Inservice Testing (IST) Program - Third Ten Year Interval" (ADAMS Accession No. ML062970433), the third 10-year interval of the IST program at HCGS, which began on December 23, 2006, was developed in accordance with the 2001 Edition through the 2003 Addenda of the OM Code. HCGS TS 4.0.5 currently references Section XI of the ASME Code for IST requirements. As such, a revision to the TSs is needed in accordance with 10 CFR 50.55a(f)(5)(ii) to conform the TSs to the revised program.

NUREG-1433, Revision 3.0, "Standard Technical Specifications - General Electric Plants, BWR/4," dated June 2004 (ADAMS Accession No. ML041910194), contains the improved Standard Technical Specifications (STS) for General Electric BWR/4 plants.

TSTF-479, Revision 0, "Changes to Reflect Revisions of 10 CFR 50.55a," dated December 2, 2004 (ADAMS Accession No. ML052990317), proposed changes to the improved STS, NUREG-1430 through 1434, to reflect the current edition of the ASME Code specified in 10 CFR 50.55a(b) (i.e., ASME OM Code rather than Section XI of the ASME Code). The NRC staff approved TSTF-479, Revision 0, in a letter dated December 6, 2005 (ADAMS Accession No. ML053460302).

TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less," dated July 12, 2006 (ADAMS Accession No. ML061930221), proposed changes to the improved STS, NUREG-1430 through 1434, to revise the STS section regarding the IST program by clarifying that the application of the 25% IST interval extension allowed by STS SR 3.0.2 was for IST frequencies of 2 years or less. The NRC staff approved TSTF-497, Revision 0, in a letter dated October 4, 2006 (ADAMS Accession No. ML062780321).

NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," dated January 2005 (ADAMS Accession No. ML050550290), provides guidelines and recommendations for developing and implementing programs for the IST of pumps and valves at commercial nuclear power plants.

3.0 TECHNICAL EVALUATION

3.1 Specific Proposed TS Changes

As discussed above in Safety Evaluation (SE) Section 1.0, the proposed amendment would delete TS 4.0.5 which pertains to SRs for ISI and IST of ASME Code Class 1, 2 and 3 components. The IST portion of TS 4.0.5 would be relocated to new TS 6.8.4.i, "Inservice Testing Program," consistent with NUREG-1433, TS 5.5.7, "Inservice Testing Program." Consistent with TSTF-479, the new TS for the IST Program will indicate that the program will include testing frequencies applicable to the ASME OM Code, replacing the current reference to Section XI of the ASME Code specified in TS 4.0.5. In addition, TS 6.8.4.i would revise the requirements, currently contained in TS 4.0.5, regarding the applicability of the surveillance interval extension provisions of SR 4.0.2 for consistency with TSTF-497. The ISI portion of TS 4.0.5 would be removed from the HCGS TSs consistent with NUREG-1433.

The new TS 6.8.4.i would read as follows:

6.8.4.i INSERVICE TESTING PROGRAM

This Program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

The following TSs would be revised to remove references to TS 4.0.5 and to add references to either the IST Program or the ISI Program, as applicable:

SR 4.1.5.c	Standby Liquid Control System
SR 4.4.3.2.2	Reactor Coolant System Operational Leakage
SR 4.4.7	Main Steam Isolation Valves
SR 4.4.8	Reactor Coolant System Structural Integrity
SR 4.5.1.b	Emergency Core Cooling Systems - Operating
SR 4.5.1.d.2.b	Emergency Core Cooling Systems - Operating
SR 4.6.2.2.b	Suppression Pool Spray
SR 4.6.2.3.b	Suppression Pool Cooling
SR 4.6.3.3	Primary Containment Isolation Valves
SR 4.7.4.b	Reactor Core Isolation Cooling System

In addition, TS 6.10.3.h, which relates to record retention requirements for ISI, would be revised to reflect the deletion of the ISI Program from the TSs.

3.2 <u>Evaluation of Proposed Changes Related to IST Program</u>

In 1990, the ASME published the initial edition of the OM Code, which provides rules for IST of pumps and valves. The OM Code was developed and is maintained by the ASME Committee on Operation and Maintenance of Nuclear Power Plants. The OM Code was developed in response to the ASME Board on Nuclear Codes and Standards directive that transferred responsibility for development and maintenance of rules for the IST of pumps and valves from the ASME Code, Section XI, Subcommittee on Nuclear Inservice Inspection to the ASME OM Committee. The ASME intended the OM Code to replace Section XI rules for IST of pumps and valves, and the rules for IST of pumps and valves have been deleted from Section XI of the ASME Code.

Section 50.55a(f) of 10 CFR, "Inservice Testing Requirements," requires, in part, that ASME Code Class 1, 2, and 3 pumps and valves meet the testing requirements of the OM Code. The third 10-year interval of the HCGS IST Program was updated to comply with the 2001 Edition through the 2003 Addenda of the OM Code as required by 10 CFR 50.55a(f)(4)(ii). As a consequence, the reference in TS 4.0.5 to Section XI of the ASME Code for IST requirements results in a reference to a deleted portion of the ASME Code, and a revision to the TS is required. The proposed amendment was submitted, in part, to revise the TSs to reference the current OM Code requirements.

The proposed new TS 6.8.4.i would replace the IST portion of TS 4.0.5. The proposed changes do not eliminate any inservice tests and do not relieve the licensee of its responsibility to seek relief from Code test requirements when the licensee determines that conformance with the requirements is impractical. The proposed changes will eliminate the inconsistency between the HCGS TSs and the HCGS IST Program as required by 10 CFR 50.55a(f)(5)(ii). Therefore, the NRC staff finds the proposed changes, related to referencing the OM Code in lieu of the Section XI of the ASME Code, to be acceptable. The change is also consistent with NRC-approved TSTF-479.

The proposed amendment would allow the application of the 25% IST interval extension, provided for in HCGS SR 4.0.2, to specified normal and accelerated SR frequencies in the TSs, but will limit the applicability to test intervals of 2 years or less only. This is consistent with the intent that the extension would provide operational flexibility, but would not significantly degrade the reliability that results from performing the surveillance at a specified frequency. Further, the proposal to limit the applicability to frequencies of 2 years or less limits the maximum incremental time period, between surveillances, which could be added by the 25% extension. Without this limitation, some components, such as safety and relief valves, which may be tested at surveillance intervals greater than 2 years, could have extensions applied which would be much greater than needed for operational flexibility. Based on the above considerations, the NRC staff finds the proposed changes, related to IST interval extensions, to be acceptable. The change is also consistent with NRC-approved TSTF-497 and the guidance in NUREG-1482.

3.3 Evaluation of Proposed Changes Related to ISI Program

The NRC's requirements in 10 CFR 50.55a(g) state, in part, that ASME Code Class 1, 2, and 3 components and their supports must meet the requirements of the ASME Code. The ASME publishes a new edition of the ASME Code every 3 years, and a new addendum every year. The ISI Program is required to comply with the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval per 10 CFR 50.55a(g)(4)(ii). As discussed in a PSEG letter dated December 12, 2007 (ADAMS Accession No. ML073540384), the third 10-year interval of the HCGS ISI Program, which began on December 13, 2007, was developed in accordance with the 2001 Edition, 2003 Addenda of ASME Code, Section XI.

Per the current requirements in TS 4.0.5, the ISI of ASME Code Class 1, 2, and 3 components shall be performed in accordance with applicable editions and addenda of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The NRC staff finds that the removal of the ISI requirements, currently in TS 4.0.5, does not eliminate any ISI not covered by 10 CFR 50.55a, and does not relieve the licensee of its responsibility to seek relief from the ASME Code requirements when they are impractical. The proposed changes will eliminate the regulatory redundancy between the TSs and 10 CFR 50.55a. Based on the above considerations, the NRC staff finds the proposed changes related to removal of the ISI program requirements, currently contained in TS 4.0.5, to be acceptable. The change is also consistent with NUREG-1433.

3.4 Evaluation of Other TS Changes

As discussed above in SE Section 3.1, the proposed amendment would revise a number of TSs to remove references to TS 4.0.5 and to add references to either the IST Program or the ISI Program, as applicable. In addition, TS 6.10.3.h would be revised to reflect the deletion of the ISI Program from the TSs. Due to the proposed deletion of TS 4.0.5, these other TS changes are considered administrative in nature. Therefore, the NRC staff finds these changes to be acceptable.

3.5 Technical Evaluation Conclusion

Based on the considerations in SE Sections 3.2, 3.3 and 3.4, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment. 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 and changes SRs. 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 (75 FR 4118). 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) 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. Huang
M. Audrain
K. Bucholtz
R. Ennis

Date: October 19, 2010

October 19, 2010

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: SURVEILLANCE REQUIREMENTS FOR INSERVICE INSPECTION AND INSERVICE TESTING (TAC NO. ME2560)

Dear Mr. Joyce:

The Commission has issued the enclosed Amendment No. 185 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station (HCGS). This amendment consists of changes to the Technical Specifications (TSs) and Facility Operating License in response to your application dated November 4, 2009.

The amendment revises the TSs to: (1) delete TS 4.0.5, which pertains to surveillance requirements for inservice inspection (ISI) and inservice testing (IST) of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* Class 1, 2 and 3 components; (2) add a new TS for the IST Program to Section 6.0, "Administrative Controls," of the TSs; (3) change TSs that currently reference TS 4.0.5 to reference the IST Program or ISI Program, as applicable; and (4) revise TS 6.10.3.h to reflect the deletion of the ISI Program from the TSs.

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/
Richard B. Ennis, Senior 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. 185 to License No. NPF-57
2. Safety Evaluation

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October 19, 2010

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President and Chief Nuclear Officer
PSEG Nuclear LLC
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Hancocks Bridge, NJ 08038

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