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

June 28, 2017

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

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (CAC NO. MF8166)

Dear Mr. Sena:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 205 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station. This amendment consists of changes to the technical specifications (TSs) in response to your application dated July 20, 2016.[1]

The amendment approves adoption of NRC-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015.[2]

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

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 205 to Renewed License No. NPF-57
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

---

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16203A006
[2] ADAMS Accession No. ML15294A555

PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 205
Renewed License No. NPF-57

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

   A. The application for amendment filed by PSEG Nuclear LLC dated July 20, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of 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. 205, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: June 28, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 205

HOPE CREEK GENERATING STATION

RENEWED FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the Renewed Facility Operating License with the attached 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
i	i
1-3	1-3
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 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	6-16e

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. 205, 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.

SECTION

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

1.15 DELETED

1.16 DELETED

FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

INSERVICE TESTING PROGRAM

1.18.1 The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program 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 INSERVICE TESTING 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. In accordance with the Surveillance Frequency Control Program 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 accordance with the Surveillance Frequency Control Program.

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.

---

* 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 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 INSERVICE TESTING 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.

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 INSERVICE TESTING PROGRAM.

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

a. In accordance with the Surveillance Frequency Control Program:

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 INSERVICE TESTING 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. In accordance with the Surveillance Frequency Control Program:

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.

---

* 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. In accordance with the Surveillance Frequency Control Program, performing a CHANNEL FUNCTIONAL TEST of the Primary Containment Instrument Gas System low-low pressure alarm system.

2. In accordance with the Surveillance Frequency Control Program:

   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 INSERVICE TESTING 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. In accordance with the Surveillance Frequency Control Program 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 INSERVICE TESTING PROGRAM.

---

* 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. In accordance with the Surveillance Frequency Control Program 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 INSERVICE TESTING PROGRAM.

---

* 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 in accordance with the Surveillance Frequency Control Program 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 INSERVICE TESTING PROGRAM.

4.6.3.4 In accordance with the Surveillance Frequency Control Program, 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. In accordance with the Surveillance Frequency Control Program by verifying the continuity of the explosive charge.

b. In accordance with the Surveillance Frequency Control Program by removing the explosive squib from at least one explosive valve, 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.

---

* 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. In accordance with the Surveillance Frequency Control Program 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 INSERVICE TESTING 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.*

---

* 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 Deleted

6.8.4.j Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency 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. 205

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 July 20, 2016,[1] PSEG Nuclear LLC (PSEG or the licensee) requested changes to the Technical Specifications (TSs) for the Hope Creek Generating Station (Hope Creek). Specifically, the licensee requested to adopt U.S. Nuclear Regulatory Commission (NRC or the Commission)-approved Technical Specifications Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-545, Revision 3, "TS Inservice Testing Program Removal & Clarify SR [Surveillance Requirement] Usage Rule Application to Section 5.5 Testing," dated October 21, 2015.[2]

The proposed change requests a revision to the TSs to eliminate TS 6.8.4.i, "Inservice Testing Program." A new defined term, "Inservice Testing Program," is requested to be added to the TSs "Definitions" section. The licensee stated that the license amendment request is consistent with NRC-approved TSTF-545, Revision 3. This TS improvement was made available by letter to the TSTF dated December 11, 2015,[3] as part of the consolidated line item improvement process (CLIIP). A notice of availability was published in the *Federal Register* (FR) on March 28, 2016 (81 FR 17208).

The licensee's letter dated December 18, 2015,[4] included a request to use American Society of Mechanical Engineers (ASME) Code Case OMN-20, "Inservice Test Frequency," as an alternative to certain ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) requirements at Hope Creek. The NRC considered this request separately from the proposed license amendment and authorized the licensee's use of this alternative by letter dated December 20, 2016.[5]

---

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16203A006
[2] ADAMS Accession No. ML15294A555
[3] ADAMS Package Accession No. ML15317A071
[4] ADAMS Accession No. ML15352A127
[5] ADAMS Accession No. ML16343A057

2.0 REGULATORY EVALUATION

2.1 Background

An inservice test is a test to assess the operational readiness of a structure, system, or component after first electrical generation by nuclear heat. The ASME OM Code provides requirements for inservice testing (IST) of certain components in light-water nuclear power plants. The ASME OM Code identifies the components subject to the testing (i.e., pumps, valves, pressure relief devices, and dynamic restraints), responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating results, corrective actions, personnel qualification, and recordkeeping.

On August 23, 2012, the NRC issued Regulatory Issue Summary 2012-10, "NRC Staff Position on Applying Surveillance Requirements 3.0.2 and 3.0.3 to Administrative Controls Program Tests."[6] The regulatory issue summary states that the NRC staff had determined that restructuring TS chapters during the development of the improved Standard Technical Specifications (STSs) resulted in unintended consequences when SRs 3.0.2 and 3.0.3 provisions were made applicable to the IST program. The NRC staff concluded that SRs 3.0.2 and 3.0.3 cannot be applied to TS Section 5.5 tests that are not associated with an SR.

TSTF-545, Revision 3, describes how to request license amendments that would eliminate conflicting requirements between Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and standards," and the TSs. TSTF-545, Revision 3, describes elimination of the IST from the Administrative Controls section of the TSs. The TSs contain surveillances that require testing or test intervals in accordance with the IST. TSTF 545, Revision 3, describes adding a new definition, "INSERVICE TESTING PROGRAM," to the TSs, which would be defined as "the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." TSTF-545, Revision 3, describes replacement of existing uses of the term, "Inservice Testing Program," with the defined term, as denoted by capital letters, throughout the TSs.

2.2 Proposed Technical Specification Changes

The licensee requested to delete TS 6.8.4.i from the Administrative Controls section of the Hope Creek TSs and replace it with the word "Deleted." TS 6.8.4.i currently states:

> 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

[6] ADAMS Accession No. ML12079A393

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.

TS 6.8.4.i, which references Specification 4.0.2 (SR 4.0.2), allows an extension of IST intervals by up to 25 percent of the specified surveillance interval. If it is discovered that a surveillance associated with an IST activity was not performed within the required interval, Specification 4.0.3 (SR 4.0.3) allows the licensee to delay declaring the associated limiting condition for operation (LCO) not met in order to perform the missed surveillance. The licensee did not request changes to SRs 4.0.2 or 4.0.3.

The licensee requested to revise the Definitions section of the TSs by adding Definition 1.18.1, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The licensee also requested that all existing occurrences of "IST Program" in TS SRs be replaced with "INSERVICE TESTING PROGRAM," so that the SRs refer to the new definition, in lieu of the deleted program. In addition, the licensee proposed conforming changes to the TS index pages denoting the addition of the new definition.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulatory requirements, guidance, and licensing information during its review of the proposed changes.

As described in 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," whenever a holder of an operating license desires to amend the license, application for an amendment must be filed with the Commission fully describing the changes desired, and following, as far as applicable, the form prescribed for original applications. For TSs, 10 CFR 50.36(a)(1) states that each applicant for an operating license shall include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. Also, 10 CFR 50.36(a)(1) states that a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TSs.

Pursuant to 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. The issuance of operating licenses is addressed by 10 CFR 50.57(a) and requires the Commission to find, among other things, that "[t]here is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter." It also requires a finding that "[t]he issuance of the license will not be inimical to the common defense and security or to the health and safety of the public."

Technical Specifications

Per 10 CFR 50.36(b), each license authorizing operation of a utilization facility will include TSs. The TSs will be derived from the analyses and evaluations included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34 (describing the technical information to be included in applications for an operating license). The Commission may include such additional TSs as the Commission finds appropriate.

Paragraph 50.36(c) of 10 CFR requires TSs to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. Paragraph 50.36(c)(3) of 10 CFR states that "[s]urveillance requirements 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." Paragraph 50.36(c)(5) of 10 CFR states that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

The NRC staff's guidance for review of the TSs is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," dated March 2010.[7] As described therein, as part of the regulatory standardization effort, the NRC staff has prepared improved STSs (NUREG 1430 through NUREG 1434) for each of the LWR nuclear steam supply systems and associated balance-of-plant equipment systems. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with TSTF-545, Revision 3. Special attention is given to TS provisions that depart from the improved STSs, as modified by NRC-approved TSTF travelers, to determine whether proposed differences are justified by uniqueness in plant design or other considerations so that 10 CFR 50.36 is met. In addition, the guidance states that comparing the change to previous STSs can help clarify the intent of the TSs.

Inservice Testing

Pursuant to 10 CFR 50.54, "Conditions of licenses," the applicable requirements of 10 CFR 50.55a are conditions of every nuclear power reactor operating license issued under 10 CFR Part 50. The regulation at 10 CFR 50.55a(f) addresses IST and requires that systems and components of boiling and pressurized water-cooled nuclear power reactors meet the requirements of the ASME Boiler and Pressure Vessel (BPV) Code and ASME OM Code as specified in 10 CFR 50.55a(f)(1) through (f)(6).

The ASME OM Code is a consensus standard that is incorporated by reference into 10 CFR 50.55a. During the incorporation process, the NRC staff reviewed the ASME OM Code requirements for technical sufficiency and found that the ASME OM Code IST program requirements were suitable for incorporation into the NRC's rules.

---

[7] ADAMS Accession No. ML100351425

Since Hope Creek was issued a construction permit (CP) on November 4, 1974, the provisions of 10 CFR 50.55a(f)(3) apply, which state:

> *Design and accessibility requirements for performing inservice testing in plants with CPs issued after 1974.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit under this part or design approval, design certification, combined license, or manufacturing license under part 52 of this chapter was issued on or after July 1, 1974:
>
> (i)–(ii) [Reserved]
>
> (iii) *IST design and accessibility requirements: Class 1 pumps and valves.*
> (A) *Class 1 pumps and valves: First provision.* In facilities whose construction permit was issued before November 22, 1999, pumps and valves that are classified as ASME Code Class 1 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, or Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraphs (a)(3)(ii) and (iii) of this section, respectively) applied to the construction of the particular pump or valve or the summer 1973 Addenda, whichever is later.
>
> (B) *Class 1 pumps and valves: Second provision.* In facilities whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under part 52 of this chapter, issued on or after November 22, 1999, pumps and valves that are classified as ASME Code Class 1 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraph (a)(3)(iii) of this section), incorporated by reference in paragraph (a)(1)(iv) of this section at the time the construction permit, combined license, manufacturing license, design certification, or design approval is issued.
>
> (iv) *IST design and accessibility requirements: Class 2 and 3 pumps and valves.*
> (A) *Class 2 and 3 pumps and valves: First provision.* In facilities whose construction permit was issued before November 22, 1999, pumps and valves that are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, that are incorporated by reference in paragraph (a)(3)(ii) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) *Class 2 and 3 pumps and valves: Second provision.* In facilities whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under part 52 of this chapter, issued on or after November 22, 1999, pumps and valves that are classified as ASME Code Class 2 and 3 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME OM Code Cases listed in NRC Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraph (a)(3)(iii) of this section), incorporated by reference in paragraph (a)(1)(iv) of this section at the time the construction permit, combined license, or design certification is issued.

(v) *IST design and accessibility requirements: Meeting later IST requirements.* All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed in paragraph (b) of this section.

The regulation in 10 CFR 50.55(a)(f)(5)(ii) states, in part, "If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program."

The NRC staff's guidance for functional design, qualification, and IST programs for pumps, valves, and dynamic restraints is in Section 3.9.6 of NUREG-0800, Revision 3, dated March 2007.[8] As part of the review for the IST program for pumps, the NRC staff will review the IST frequencies and test parameters. The frequency of ISTs and test parameters are acceptable if the provisions of Subsection ISTB-3000 of the ASME OM Code are met. As described therein, the licensee's IST program is acceptable if the program meets the requirements of the ASME Code, Section XI, or the ASME OM Code, as incorporated by reference in 10 CFR 50.55a.

The NRC staff's guidance for complying with the codes and standards in 10 CFR 50.55a is in Section 5.2.1.1 of NUREG-0800, Revision 3, dated March 2007. [9] As stated therein, acceptance criteria are based in part on meeting 10 CFR 50.55a.

The NRC staff's guidance for review of inservice inspection and testing of Class 2 and Class 3 components is in Section 6.6 of NUREG-0800, Revision 2, dated March 2007.[10] As stated therein, acceptance criteria are based on meeting the relevant parts of 10 CFR 50.55a as they pertain to the specification of the preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and Class 3 systems and components.

NUREG-1482, Revision 2, "Guidelines for Inservice Testing at Nuclear Power Plants: Inservice Testing of Pumps and Valves and Inservice Examination and Testing of Dynamic Restraints

[8] ADAMS Accession No. ML070720041
[9] ADAMS Accession No. ML070040003
[10] ADAMS Accession No. ML070550071

(Snubbers) at Nuclear Power Plants, Final Report," dated October 2013,[11] provides guidance for the IST of pumps and valves.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine whether the proposed changes are consistent with the regulations, guidance, and licensing information discussed in Section 2.3 of this safety evaluation. In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Among the considerations is whether the TSs, as amended, would provide the necessary administrative controls per 10 CFR 50.36(c)(5).

In making its determination as to whether to amend the license, the NRC staff considered those regulatory requirements that are automatically conditions of the license pursuant to 10 CFR 50.54. Where the regulations already condition the license, there is no need for a duplicative requirement in the TSs; the regulations provide the necessary reasonable assurance of the health and safety of the public.

3.1 Assessment of Requested Deletion of TS 6.8.4.i, "Inservice Testing Program"

Hope Creek TS 6.8.4.i, which is in the Administrative Controls section, requires the licensee to have an IST program that provides controls for IST of ASME Code Class 1, 2, and 3 components. The NRC staff notes that the licensee's IST, which is required by 10 CFR 50.54 and 50.55a(f), and which is outside the scope of this amendment request, already contains requirements and considerations similar to those of TS 6.8.4.i. Therefore, requiring the licensee to have an IST program in TSs is duplicative of the license condition in 10 CFR 50.54. Thus, with the proposed TS changes, the licensee will still be required to maintain an IST program in accordance with the ASME OM Code, as specified in 10 CFR 50.55a(f). For the reasons explained further in this safety evaluation, it is not necessary to have additional administrative controls in the TSs for Hope Creek relating to the IST program to assure operation of the facility in a safe manner.

Consideration of TS 6.8.4.i.a

The ASME OM Code requires testing to normally be performed within certain time periods. TS 6.8.4.i.a more precisely defines those time periods specified in the ASME OM Code and applicable addenda (e.g., states "at least once per 31 days" for the ASME OM Code frequency of "monthly"). However, the NRC staff has determined that the ASME OM Code frequencies are sufficient to assure operation of the facility in a safe manner. Therefore, the more precise definitions in TS 6.8.4.i.a are not necessary to assure operation of the facility in a safe manner.

Consideration of TS 6.8.4.i.b

TS 6.8.4.i.b allows the licensee to extend, by up to 25 percent, the interval between IST activities, as required by TS 6.8.4.i.a, and for other normal and accelerated frequencies specified as 2 years or less in the IST program. Similar to TS 6.8.4.i.b, the NRC authorization of ASME Code Case OMN 20, by letter dated December 20, 2016,[12] also permits the licensee to extend the IST intervals specified in the ASME OM Code by up to 25 percent.

---

[11] ADAMS Accession No. ML13295A020
[12] ADAMS Accession No. ML16343A057

The NRC staff determined that the TS 6.8.4.i.b allowance to extend IST intervals is not needed to assure operation of the facility in a safe manner. Therefore, the NRC staff determined that deletion of TS 6.8.4.i.b is acceptable. Moreover, the deletion of TS 6.8.4.i.b does not impact the licensee's ability to extend IST intervals using ASME Code Case OMN-20, as authorized by the NRC.

Consideration of TS 6.8.4.i.c

TS 6.8.4.i.c allows the licensee to use Specification 4.0.3 when it discovers that an SR associated with an inservice test was not performed within its specified frequency. Specification 4.0.3 allows the licensee to delay declaring an LCO not met in order to perform the missed surveillance. The use of Specification 4.0.3 for inservice tests is limited to those inservice tests required by an SR. In accordance with 10 CFR 50.55a, the licensee may also request relief from the ASME OM Code requirements to address issues associated with a missed inservice test. The deletion of TS 6.8.4.i.c does not change any of these requirements, and Specification 4.0.3 will continue to apply to those inservice tests required by SRs. Based on the above, the NRC staff determined that deletion of TS 6.8.4.i.c is acceptable.

Consideration of TS 6.8.4.i.d

TS 6.8.4.i.d states that nothing in the ASME OM Code shall be construed to supersede the requirements of any TSs. However, the regulations in 10 CFR 50.55a(f)(5)(ii) address what to do if a revised IST program for a facility conflicts with the TSs for the facility. The regulations require the licensee to apply for an amendment to the TSs to conform the TSs to the revised program at least 6 months prior to the start of the period for which the provisions become applicable. Accordingly, there is no need for a TS stating how to address conflicts between the TSs and the IST program because the regulations specify how conflicts must be resolved.

Conclusion Regarding Deletion of TS 6.8.4.i

As explained above, the NRC staff determined that the requirements currently in TS 6.8.4.i are not necessary to assure operation of the facility in a safe manner. Based on this evaluation, the staff concludes that the deletion of TS 6.8.4.i from the licensee's TSs is acceptable because TS 6.8.4.i. is not required by 10 CFR 50.36(c)(5).

3.2 Definition of INSERVICE TESTING PROGRAM and Revision to Surveillance Requirements

The licensee proposed to revise the TS Definitions section to include the term, "INSERVICE TESTING PROGRAM," with the following definition: "The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)." The proposed definition is consistent with the definition in the NRC-approved TSTF-545, Revision 3. The NRC staff finds the definition acceptable because it correctly refers to the IST requirements in 10 CFR 50.55a(f).

The licensee requested that all existing references to the "IST Program" in SRs be revised to "INSERVICE TESTING PROGRAM" to reference the new TS-defined term in lieu of the deleted IST program TSs. The proposed change is consistent with the intent of TSTF-545, Revision 3, to replace the current references in SRs with the new definition. The NRC staff verified that for each SR reference to the "IST Program," the licensee proposed to change the reference to "INSERVICE TESTING PROGRAM." The proposed change does not alter how the SR testing is performed. The IST frequencies could change because the TSs will no longer include the more precise test frequencies in TS 6.8.4.i.a. However, as discussed in Section 3.1 of this safety evaluation, the NRC staff determined that the TSs do not need to include the more precise test frequencies currently in TS 6.8.4.i.a in order to assure operation of the facility in a safe manner.

Based on its review, the NRC staff determined that revising the SRs to refer to the new definition is acceptable because these SRs will continue to be performed in accordance with the requirements of 10 CFR 50.55a(f). The staff also determined that with the proposed changes, 10 CFR 50.36(c)(3) will continue to be met because the SRs will continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

3.3 Conforming Changes and Variations from TSTF-545, Revision 3

The NRC staff also evaluated the following conforming changes and variations from TSTF-545, Revision 3, not previously addressed in this safety evaluation.

a. Hope Creek's TSs have not been converted to the improved STSs on which TSTF-545, Revision 3, is based. As a result, the numbering, format, and content of Hope Creek's TSs vary from TSTF-545, Revision 3. In addition, Hope Creek's TSs use different numbering than the improved STSs on which TSTF-545, Revision 3, is based. The staff finds that the licensee's proposed deviations in numbering, format, and content are editorial in nature and consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

b. The phrase "Inservice Testing Program" may appear in different locations in Hope Creek's TSs when compared to the STSs. Revising this phrase to be capitalized wherever it may appear is within the scope of this proposed change. The NRC staff finds that these proposed deviations are consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

c. An index is included in Hope Creek's TSs. Therefore, the licensee included conforming changes to the index resulting from the addition of the new definition. The staff finds that the proposed deviations are editorial in nature and consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

d. The licensee proposed to replace the content of Hope Creek's IST TSs with the word "Deleted" and retain the existing numbering sequence. The staff finds that these proposed changes are editorial in nature and consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds the licensee's proposed TS changes acceptable.

The NRC staff finds that the proposed changes and variations from TSTF-545, Revision 3, are editorial in nature, and the licensee's proposed TS changes remain consistent with the intent of TSTF-545, Revision 3. Therefore, the staff finds that the licensee's proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State Official was notified of the proposed issuance of the amendment on April 13, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 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 on October 25, 2016 (81 FR 73437). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Tilton

Date: June 28, 2017

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-545, REVISION 3, "TS INSERVICE TESTING PROGRAM REMOVAL & CLARIFY SR USAGE RULE APPLICATION TO SECTION 5.5 TESTING" (CAC NO. MF8166) DATED JUNE 28, 2017

DISTRIBUTION:
PUBLIC
RidsNrrDssStsb Resource
RidsRgn1MailCenter Resource
RidsNrrPMHopeCreek Resource
RidsNrrDssStsb Resource
GBedi, NRR
RidsACRS_MailCTR Resource
RidsNrrDorlLpl1 Resource
RidsNrrDeEpnb Resource
RidsNrrLALRonewicz Resource
CTilton, NRR

ADAMS Accession Number: ML17164A355 *by e-mail

OFFICE	DORL/LPL1/PM	DORL/LPL1/LA	DE/EPNB/BC*	DSS/STSB/BC(A)
NAME	THood	LRonewicz	DAlley	JWhitman
DATE	06/12/2017	06/14/2017	06/15/2017	06/15/2017
OFFICE	OGC	DORL/LPL1/BC	DORL/LPL4/PM	
NAME	JWachutka (w/comment)	JDanna	LRegner	
DATE	06/22/2017	06/27/2017	06/28/2017	

OFFICIAL RECORD COPY