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

July 29, 2014

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

SUBJECT: HOPE CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: REQUEST TO RELOCATE SAFETY RELIEF VALVE POSITION INSTRUMENTATION TO TECHNICAL REQUIREMENTS MANUAL (TAC NO. MF2508)

Dear Mr. Joyce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 195 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station (Hope Creek). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 30, 2013.

The amendment deletes the operability and surveillance requirements (SRs) for the reactor coolant system safety/relief valve (SRV) position instrumentation from the Hope Creek TSs. The operability and SRs for the SRV position instrumentation will be relocated into the Hope Creek Technical Requirements Manual (TRM) by the licensee. The Hope Creek TRM is controlled by the licensee in a manner consistent with procedures described in the Hope Creek Updated Final Safety Analysis Report (UFSAR), and under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, tests, and experiments." Future changes to the operability and SRs for the SRV position instrumentation will be performed by the licensee pursuant to 10 CFR 50.59.

The affected TSs are: Accident Monitoring Instrumentation, TS 3.3.7.5 and SR 4.3.7.5; and Safety/Relief Valves, TS 3.4.2.1 and SR 4.4.2.1.

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

Sincerely,

John G. Lamb, 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. 195 to
   Renewed 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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 195
Renewed License No. NPF-57

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

    A. The application for amendment filed by PSEG Nuclear LLC dated July 30, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 195, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert G. Schaaf, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License
and the Technical Specifications

Date of Issuance: July 29, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 195

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
Page 3	Page 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 3-85	3/4 3-85
3/4 3-87	3/4 3-87
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8

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. 195, 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.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2,3	80
2. Reactor Vessel Water level	2	1	1,2,3	80
3. Suppression Chamber Water level	2	1	1,2,3	80
4. Suppression Chamber Water Temperature*	2	1	1,2,3	80
5. Suppression Chamber Pressure	2	1	1,2,3	80
6. Drywell Pressure	2	1	1,2,3	80
7. Drywell Air Temperature	2	1	1,2,3	80
8. Deleted				
9. Deleted				
10. Drywell Atmosphere Post-Accident Radiation Monitor	2	1	1,2,3	80
11. North Plant Vent Radiation Monitor #	1	1	1,2,3	81
12. South Plant Vent Radiation Monitor #	1	1	1,2,3	81
13. FRVS Vent Radiation Monitor #	1	1	1,2,3	81
14. Primary Containment Isolation Valve Position Indication ##	2/valve	1/valve	1,2,3	82

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* Average bulk pool temperature.

\# High range noble gas monitors.

\## One channel consists of the open limit switch, and the other channel consists of the closed limit switch.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK[(a)]	CHANNEL CALIBRATION[(a)]	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure			1,2,3
2. Reactor Vessel Water Level			1,2,3
3. Suppression Chamber Water Level			1,2,3
4. Suppression Chamber Water Temperature			1,2,3
5. Suppression Chamber Pressure			1,2,3
6. Drywell Pressure			1,2,3
7. Drywell Air Temperature			1,2,3
8. Deleted			
9. Deleted			
10. Drywell Atmosphere Post-Accident Radiation Monitor		**	1,2,3
11. North Plant Vent Radiation Monitor#			1,2,3
12. South Plant Vent Radiation Monitor#			1,2,3
13. FRVS Vent Radiation Monitor#			1,2,3
14. Primary Containment Isolation Valve Position Indication			1,2,3

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

** CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

High range noble gas monitors.

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

c. Deleted

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.2.1 Deleted

4.4.2.2 At least 1/2 of the safety relief valve pilot stage assemblies shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program, and they shall be rotated such that all 14 safety relief valve pilot stage assemblies are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program. All safety relief valves will be re-certified to meet a ±1% tolerance prior to returning the valves to service after setpoint testing.

4.4.2.3 The safety relief valve main (mechanical) stage assemblies shall be set pressure tested, reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195

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 30, 2013,[1] PSEG Nuclear LLC (PSEG or the licensee) submitted a request for changes to the Hope Creek Generating Station (Hope Creek) Technical Specifications (TSs).

The proposed amendment would delete the operability and surveillance requirements (SRs) for the reactor coolant system safety/relief valve (SRV) position instrumentation from the Hope Creek TSs. The operability and SRs for the SRV position instrumentation would be relocated into the Hope Creek Technical Requirements Manual (TRM) by the licensee. The Hope Creek TRM is controlled by the licensee in a manner consistent with procedures described in the Hope Creek Updated Final Safety Analysis Report (UFSAR), and under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.59, "Changes, tests, and experiments." Future changes to the operability and SRs for the SRV position instrumentation would be performed by the licensee pursuant to 10 CFR 50.59.

The affected TSs are: Accident Monitoring Instrumentation, TS 3.3.7.5 and SR 4.3.7.5; and Safety/Relief Valves, TS 3.4.2.1 and SR 4.4.2.1.

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff used the following regulatory bases for its evaluation of the licensee's amendment request:

- The regulations in 10 CFR 50.36(c)(2) state, in part, that "[w]hen a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met . . ."

- The regulations in 10 CFR 50.36(c)(2)(ii) which state that, "[a] technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML13211A205.

(A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

- The regulations in 10 CFR 50.36(c)(3) which state 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."

- On July 22, 1993, the Commission issued its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," indicating that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.36, "Technical Specifications" (58 FR 39132; July 22, 1993). The policy statement described the safety benefits of the improved Standard Technical Specifications (STS) and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments and for complete conversions to the improved STS. Further, the policy statement gave guidance for evaluating the required scope of the licensee's improved TS and defined guidance criteria for determining which limiting conditions for operation (LCOs) and associated SRs should remain in its improved TS. Using this approach, licensees should keep existing LCO requirements that fall within or satisfy any of the policy statement criteria in the TS. Those LCO requirements that do not fall within or satisfy any of these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36(c)(2)(ii) (60 FR 36953; July 19, 1995). NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," dated April 2012, (ADAMS Accession No. ML12104A192) was developed based on the criteria in the Final Policy Statement.

- Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an

Accident," Revision 2, issued December 1980, lists five types (Types A–E) of variables to help designers select the accident monitoring instrumentation and applicable criteria. Categories 1, 2, and 3 separate the type criteria into groups for a graded approach to requirements, depending on the importance to safety or the measurement of a specific variable.

- The NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications, dated, May 9, 1988 (ADAMS Accession No. ML11264A057).

- Precedents: The NRC has approved the relocation of operability and SRs from TS to TRM for the following plants:

  (A) Hope Creek Generating Station, "Relocate Component Lists for Primary Containment Isolation Valves from Technical Specifications (TAC No. MD3600)," dated August 27, 2007. (ADAMS Accession No. ML071430403)

  (B) Limerick Generating Station, Units 1 and 2, "Relocation of Operability and Surveillance Requirements for the Safety/Relief Valve Position Instrumentation (TAC Nos. MC3454 and MC3455)," dated September 27, 2005. (ADAMS Accession No. ML052550369)

  (C) Fort Calhoun Station, Unit No. 1, "Revision of Technical Specifications to Relocate Power-Operated Relief Valve/Safety Valve Position and Tail Pipe Temperature Instrumentation (TAC No. ME4542)," dated September 30, 2011. (ADAMS Accession No. ML112620402)

3.0 TECHNICAL EVALUATION

3.1 Background

The Hope Creek Nuclear Pressure Relief System, including the main steam line safety/relief valves, provides overpressure protection for the reactor coolant pressure boundary (RCPB). Each of the 14 SRVs provide two main protective functions: (1) overpressure safety operation - the valves open automatically to limit a pressure rise; and (2) depressurization operation - the Automatic Depressurization System (ADS) valves open automatically as part of the Emergency Core Cooling System for events involving small breaks in the RCPB. Low-low set relief logic is provided for two of the SRVs. This automatic control system ensures that containment, SRV discharge lines, and reactor overpressure protection design bases are not exceeded. It accomplishes this by providing these valves with altered setpoints that are lower than the normal SRV spring-set opening and closing pressure setpoints.

The Hope Creek Safety Relief Valve Position Indication System (SRVPIS) provides the control room operator with an OPEN/CLOSED indication for all 14 SRVs, and provides an alarm to alert the operator to abnormal SRV position. SRVPIS is an acoustic monitoring system consisting of accelerometers that are strap mounted to the discharge piping downstream of each SRV, as close as possible to the valve. When an SRV is open, the accelerometer senses the flow noise created by the steam passing through the discharge piping and produces a signal proportional to the flow through the pipe. This signal is amplified by a preamplifier located outside the primary containment and transmitted to a signal conditioning unit located in the SRVPIS control panel in the main control room. The signal conditioning unit processes the signal and provides the operator with an OPEN/CLOSED SRV position indication and an alarm, if the valve is determined to be open. The SRVPIS does not provide input to a control or trip function.

The SRVPIS is provided in TS 3.4.2.1/SR 4.4.2.1 for the safety/relief valve acoustic monitors. It is also provided in TS 3.3.7.5/SR 4.3.7.5, along with the SRV tailpipe temperature indicators.

The Hope Creek SRV tailpipe temperature indication consists of one thermocouple installed in the discharge line of each SRV, mounted several feet downstream from the valve body. Thermocouple output is indicated on a multipoint recorder in the main control room. A temperature increase in the discharge piping can indicate that the SRV is leaking or open. Temperature increases above ambient are alarmed in the main control room. The SRV tailpipe temperature indicators do not provide input to a control or trip function. The SRV acoustic monitors and SRV tailpipe temperature indicators are collectively referred to as "SRV position instrumentation" or "SRV position indication."

The Hope Creek Suppression Pool Temperature Monitoring System (SPTMS) monitors bulk suppression pool temperature, providing indication that one or more SRVs may not be closed. The SPTMS consists of 16 sensors that are divided into two redundant channels. Each channel consists of eight sensors located symmetrically around the suppression pool in order to provide a reasonable measure of bulk temperature.

3.2 Description of Proposed TS Changes

Specifically, the proposed license amendment would remove the following items from the TS and relocate them to the TRM:

A. Item 9, "Safety/Relief Valve Position Indicators," from Table 3.3.7.5-1, "Accident Monitoring Instrumentation," on page 3/4 3-85, in TS 3.3.7.5. This table identifies the operability requirements for this instrumentation. With the number of OPERABLE channels less than the Minimum Number of Channels, the associated action statement requires the inoperable channel to be restored 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. The associated footnote to be relocated reads: "Acoustic monitoring and tail pipe temperature."

B. Item 9, "Safety/Relief Valve Position Indicators," from Table 4.3.7.5-1, "Accident Monitoring Instrumentation Surveillance Requirements," on page 3/4 3-87, in TS 4.3.7.5. This table identifies the SR for this instrumentation. Channel checks and channel calibrations are required to be performed in Operational Conditions 1, 2, and 3 in accordance with the Surveillance Frequency Control Program.

C. TS 3.4.2.1.c on page 3/4 4-7. This specification reads: "With one or more of the above required safety/relief valve acoustic monitors inoperable, restore the inoperable monitors 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."

D. TS 4.4.2.1 on page 3/4 4-8. This specification reads: "The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be ≤ 30% of full open noise level by performance of a

   a. CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program, and a

   b. CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program."

3.3 Evaluation of Proposed TS Changes

The Hope Creek Updated Final Safety Analysis Report (UFSAR) identifies, "Primary System Relief Valves Position, Including ADS (Automatic Depressurization System) or FLOW Through or Pressure in Valve Lines" as Type D, Category 2, Variables in conformance with RG 1.97. Type D variables are those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. Category 2 instruments are designed to less stringent qualifications than Category 1 instruments. Category 2 instruments are not required to meet seismic qualification, redundancy, or continuous display, and are required to have a high reliability power source, but not necessarily standby power. As such, relocation of the SRV position instrumentation from TS to TRM conforms to RG 1.97.

The safety/relief valves themselves are part of the primary success path in the UFSAR accident analysis for Design Basis Accidents (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. The operability of the SRVs is therefore required by TS 3.4.2, "Safety/Relief Valves." The SRV acoustic monitors and tailpipe temperature indicators, however, do not detect or indicate a significant abnormal degradation of the RCPB, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, Criterion 1, which is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 does not include instrumentation installed to identify the source of actual leakage, such as valve position indicators.

SRV position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

While the function of the SRVs to open on high Reactor Coolant System pressure is part of the primary success path in the UFSAR, SRV position indication is not required nor is it necessary to maintain operability of this safety function. The UFSAR accident analysis assumes that the SRVs open as designed to reduce reactor pressure and no operator action based on SRV position indication is required. Therefore, the NRC staff concludes that the SRV position indication is not part of the primary success path as specified in Criterion 3.

The Suppression Pool Temperature Monitoring System is available to monitor SRV position. The NRC staff also finds that the loss of SRV position instrumentation has no effect on the probabilistic risk assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The Hope Creek Emergency Operating Procedures (EOPs) provide symptom-based instruction to the operating staff in mitigating an upset condition of the plant. Individual EOPs using SRV position can be accomplished regardless of whether SRV position instrumentation is available (i.e., for Emergency Depressurization and Reactor Pressure Vessel Flooding).

Based on the above information, SRV position instrumentation does not meet any of the four screening criteria of 10 CFR 50.36(c)(2)(ii). This conclusion is supported by the absence of operability and SRs for the SRV position instrumentation in the improved STS (ISTS) presented in NUREG-1433. Accordingly, the proposed changes conform to the ISTS. The proposed changes also do not involve a modification to the physical configuration of the plant or a change in the methods governing normal plant operation. The proposed changes will not impose any new or different requirement or introduce a new accident initiator, accident precursor, or a malfunction mechanism.

The licensee will place the operability and SRs for the SRV position instrumentation in the Hope Creek TRM to ensure the reliability of the SRV position monitoring capability and all changes to these requirements in the TRM will be subject to the provisions of 10 CFR 50.59. The Hope Creek TRM is controlled by the licensee in a manner consistent with procedures described in the Hope Creek UFSAR. Future changes to the operability and SRs for the SRV position instrumentation will be performed by the licensee pursuant to 10 CFR 50.59, and subject to NRC inspection.

The NRC staff determined that the proposed changes conform to NRC regulations and guidance regarding the content of plant TS, as identified in 10 CFR 50.36, NUREG-1433, and the NRC Final Policy Statement in 58 FR 39132. Also, the NRC staff has approved similar TS changes for Limerick and Fort Calhoun. Therefore, the NRC staff concludes that the deletion of the SRV acoustic position indication and tail pipe temperature indication provisions from the Hope Creek TS is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 (79 FR 18334, dated April 1, 2014). 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: Subinoy Mazumdar

Date: July 29, 2014

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

Sincerely,

/RA/

John G. Lamb, 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. 195 to
   Renewed License No. NPF-57
2. Safety Evaluation

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ADAMS Accession No: **ML14108A312** ***via memo dated**

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