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

September 6, 2019

Mr. Eric Carr
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 NO. 218 TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-551, "REVISE SECONDARY CONTAINMENT SURVEILLANCE REQUIREMENTS" (EPID L-2019-LLA-0083)

Dear Mr. Carr:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 218 to Renewed Facility Operating License No. NPF-57 for the Hope Creek Generating Station in response to your application dated April 18, 2019.

The amendment revises Hope Creek Generating Station Technical Specification (TS) 3.6.5.1, "Secondary Containment Integrity," Surveillance Requirements (SRs) 4.6.5.1.a and 4.6.5.1.b.2.a. SR 4.6.5.1.a is being revised to address conditions during which the secondary containment pressure may not meet the SR pressure requirements. SR 4.6.5.1.b.2.a is being modified to acknowledge that more than one secondary containment access opening may be simultaneously open for entry and exit. Additionally, TS Definitions 1.39.d and 1.39.g are being revised to conform to the changes to these two SRs.

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,

James S. Kim, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-354

Enclosures:
1. Amendment No. 218 to Renewed License No. NPF-57
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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PSEG NUCLEAR LLC

DOCKET NO. 50-354

HOPE CREEK GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 218
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 April 18, 2019, 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. 218, 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 of the date of issuance.

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 Facility Operating License and Technical Specifications

Date of Issuance: September 6, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 218

HOPE CREEK GENERATING STATION

RENEWED FACILITY OPERATING LICENSE NO. NPF-57

DOCKET NO. 50-354

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1-7	1-7
3/4 6-47	3/4 6-47

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 3902 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. 218, 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.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The filtration, recirculation and ventilation system is in compliance with the requirements of Specification 3.6.5.3.

d. For double door arrangements, at least one door in each access to the secondary containment is closed, except when the access opening is being used for entry and exit.

e. For single door arrangements, the door in each access to the secondary containment is closed, except for normal entry and exit.

f. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a, except as indicted by the footnote for Specification 4.6.5.1.a

SHUTDOWN MARGIN (SDM)

1.40 SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

a. The reactor is xenon free;

b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled, by the licensee.

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying in accordance with the Surveillance Frequency Control Program that the reactor building is at a negative pressure**.

b. Verifying in accordance with the Surveillance Frequency Control Program that:

1. All secondary containment equipment hatches and blowout panels are closed and sealed.

2. a. For double door arrangements, at least one door in each access to the secondary containment is closed except when the access opening is being used for entry and exit.

b. For single door arrangements, the door in each access to the secondary containment is closed except for routine entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

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* When recently irradiated fuel is being handled in the secondary containment.

** Not required to be met for 4 hours if analysis demonstrates four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the FRVS are capable of establishing the required secondary containment vacuum.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 218

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-57

PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

1.0 INTRODUCTION

By application dated April 18, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19108A143), PSEG Nuclear LLC (the licensee) requested U.S. Nuclear Regulatory Commission (NRC or the Commission) approval for a license amendment request (LAR) to revise the Hope Creek Generating Station (Hope Creek) Technical Specifications (TSs). The LAR proposes to adopt Technical Specifications Task Force (TSTF) Traveler, TSTF-551, Revision 3, "Revise Secondary Containment Surveillance Requirements," dated October 3, 2016 (ADAMS Accession No. ML16277A226). The NRC approved the traveler on September 21, 2017 (ADAMS Package Accession No. ML17236A365).

The proposed changes would allow the TS secondary containment (i.e., the reactor building) pressure requirements to not be met, provided that the filtration recirculation and ventilation system (FRVS) units remain capable of establishing the required secondary containment vacuum and allow for the brief, simultaneous opening of redundant secondary containment access doors during normal entry and exit conditions.

2.0 REGULATORY EVALUATION

2.1 System Description

The secondary containment is a structure that encloses the primary containment, including components that may contain primary system fluid. The safety function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a design-basis accident (DBA) to ensure the control room operator and offsite doses are within the regulatory limits. There is no redundant train or system that can perform the secondary containment function, should the secondary containment be inoperable.

The secondary containment boundary is the combination of walls, floor, roof, ducting, doors, hatches, penetrations, and equipment that physically forms the secondary containment. Routinely used secondary containment access openings contain at least one inner and one outer door in an airlock configuration. In some cases, secondary containment access openings

are shared such that there are multiple inner or outer doors. All secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit of personnel, equipment, or material.

Secondary containment operability is based on its ability to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. To prevent ground level exfiltration of radioactive material while allowing the secondary containment to be designed as a mostly conventional structure, the secondary containment requires support systems to maintain the pressure at less than atmospheric pressure. During normal operation, non-safety-related systems are used to maintain the secondary containment at a slight negative pressure to ensure any leakage is into the building and that any secondary containment atmosphere exiting is via a pathway monitored for radioactive material. However, during normal operation, it is possible for the secondary containment vacuum to be momentarily less than the required vacuum for a number of reasons, such as during wind gusts or swapping of the normal ventilation subsystems.

During emergency conditions, the FRVS is designed to be capable of drawing down the secondary containment to a required vacuum within a prescribed time and continuing to maintain the negative pressure as assumed in the accident analysis. For Hope Creek, the design basis of the FRVS is to establish the post-accident secondary containment pressure to a negative 0.25 inches of vacuum wager gauge within 375 seconds. The leaktightness of the secondary containment, together with the FRVS, ensure that radioactive material is either contained in the secondary containment or filtered through the FRVS system filter trains before being discharged to the outside environment via the elevated release point.

2.2 Proposed Technical Specifications Changes

The proposed changes would allow the secondary containment vacuum limit to not be met, provided the FRVS system remains capable of establishing the required secondary containment vacuum. The proposed changes would also allow for the brief, simultaneous opening of redundant secondary containment access doors during normal entry and exit conditions.

2.2.1 Revision to Surveillance Requirement 4.6.5.1.a

Surveillance Requirement (SR) 4.6.5.1.a requires verifying, in accordance with the Surveillance Frequency Control Program, that the reactor building is at a negative pressure. This SR would be modified by a footnote that states:

> **Not required to be met for 4 hours if analysis demonstrates four filtration recirculation and ventilation system (FRVS) recirculation units and one ventilation unit of the FRVS are capable of establishing the required secondary containment vacuum.

2.2.2 Revision to Surveillance Requirement 4.6.5.1.b.2.a

SR 4.6.5.1.b.2.a requires verifying, in accordance with the Surveillance Frequency Control Program, that for double door arrangements, at least one door in each access to the secondary containment is closed. This SR would be modified to read:

> For double door arrangements, at least one door in each access to the secondary containment is closed except when the access opening is being used for entry and exit.

2.2.3 Revision to Technical Specification Definition 1.39

TS Definition 1.39, "Secondary Containment Integrity," item 'd' and item 'g' are revised to align with the proposed changes to TS SR 4.6.5.1.a and SR 4.6.5.1.b.2.a, respectively.

Item d would be modified to read:

> Four double door arrangements, at least one door in each access to the secondary containment is closed, except when the access opening is being used for entry and exit.

Item g would be modified to read:

> The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a, except as indicated by the footnote for Specification 4.6.5.1.a.

2.3 Regulatory Requirements and Guidance

The regulation at Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(a)(1) requires an applicant for an operating license to include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must include in the application a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls." However, per 10 CFR 50.36(a)(1), these TS bases "shall not become part of the technical specifications."

Additionally, 10 CFR 50.36(b) requires, in part:

> Each license authorizing operation of a ... utilization facility ... will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34 ["Contents of applications; technical information"]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The categories of items required to be in the TSs are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2)(i), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of SRs, which 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 LCOs will be met.

The regulation at 10 CFR 50.54, "Conditions of licenses," describes conditions in every nuclear power reactor operating license issued under Part 50 of 10 CFR.

The regulation at 10 CFR 50.67, "Accident source term," states:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[1] total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

The NRC staff's guidance for review of 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" (SRP), dated March 2010 (ADAMS Accession No. ML100351425).

NUREG-0800, SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (ADAMS Accession No. ML003734190), provides guidance to the NRC staff for the review of alternative source term (AST) amendment requests. SRP 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000 (ADAMS Accession No. ML003716792).

Regulatory Guide 1.183 provides an acceptable methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. Regulatory Guide 1.183 provides guidance to licensees for providing acceptable AST (also known as accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and licensing information discussed in Section 2.3 of this safety evaluation and in NRC-approved TSTF-551, Revision 3. In determining whether an amendment to a license will be issued, the NRC staff is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. In making its

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[1] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design-basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

determination as to whether to amend the license, the NRC staff considered those regulatory requirements that are automatically conditions of the license through 10 CFR 50.54.
The regulation at 10 CFR 50.36(a)(1) states, in part, "A summary statement of the bases or reasons for such specifications ... shall also be included in the application, but shall not become part of the technical specifications." Accordingly, along with the proposed TS changes, the licensee also submitted TS bases changes that correspond to the proposed TS changes for information only.

3.1 Variations from the TSTF-551, Revision 3

The licensee proposed the following variations from the TS changes described in TSTF-551 or in the NRC staff's safety evaluation for TSTF-551 (ADAMS Accession No. ML17236A366).

1. The Hope Creek TSs utilize different numbering than the Standard Technical Specifications (STS) on which TSTF-551 was based. Specifically, secondary containment is governed by Hope Creek TS 3.6.5.1, which is titled "Secondary Containment Integrity." STS SR 3.6.4.1.1 is equivalent to Hope Creek SR 4.6.5.1.a, and STS SR 3.6.4.1.3 is equivalent to Hope Creek SR 4.6.5.1.b.2.a. These numbering differences are administrative and do not affect the technical applicability of TSTF-551 to Hope Creek.

2. The safety evaluation and change to STS SR 3.6.4.1.1 provided by TSTF-551 refers to one standby gas treatment subsystem capable of establishing the required secondary containment vacuum. Hope Creek maintains post-accident secondary containment vacuum via FRVS units. The combination of four (of six) FRVS recirculation units and one FRVS ventilation unit (out of two) performs the equivalent pressure control and air filtration functions of the standby gas treatment subsystem described in TSTF-551.

3. STS SR 3.6.4.1.1 requires verification that the secondary containment vacuum is greater than or equal to a specific value of vacuum water gauge, whereas Hope Creek TS SR 4.6.5.1.a. requires verification that the reactor building is at a negative pressure. The reactor building is synonymous with secondary containment, and the SR to verify that the reactor building is maintained at a negative pressure in lieu of a specific gauge value ensures the reactor building is sufficiently leaktight, which is consistent with STS SR 3.6.4.1.1.

4. The Hope Creek TSs include Definition 1.39 for secondary containment integrity. Item 'd' within Definition 1.39 describes the closure requirements for double door access to secondary containment. Item 'g' within Definition 1.39 describes the pressure within secondary containment relative to TS 4.6.5.1.a. These definitions are revised to align with the proposed changes to TS SR 4.6.5.1.b.2.a and SR 4.6.5.1.a.

5. The bases changes in TSTF-551 are incorporated with applicable changes to substitute the Hope Creek FRVS in place of the standby gas treatment subsystem described in the TSTF traveler. In addition, supporting text is provided describing the analyzed secondary containment drawdown capability of the FRVS.

The staff finds these variations do not affect the applicability of TSTF-551 or the staff safety evaluation for TSTF-551 to the proposed license amendment and are, therefore, acceptable.

3.2 Proposed Changes to Surveillance Requirement 4.6.5.1.a

A note is being added to SR 4.6.5.1.a. The note allows the SR to not be met for up to 4 hours if an analysis demonstrates that one FRVS subsystem is capable of establishing the required secondary containment vacuum. The FRVS subsystem is the combination of four (of six) FRVS recirculation units and one FRVS ventilation unit (out of two) that performs the equivalent pressure control and air filtration functions of the standby gas treatment subsystem described in TSTF-551. During normal operation, conditions may occur that result in SR 4.6.5.1.a not being met for short durations. For example, wind gusts that lower external pressure or loss of the normal ventilation system that maintains secondary containment vacuum may affect secondary containment vacuum. These conditions may not be indicative of degradations of the secondary containment boundary or of the ability of the FRVS system to perform its specified safety function.

The note provides an allowance for the licensee to confirm secondary containment operability by confirming that one FRVS subsystem is capable of performing its specified safety function. This confirmation is necessary to apply the exception to meeting the SR acceptance criterion. While the duration of these occurrences is anticipated to be very brief, the allowance is permitted for a maximum of 4 hours, which is consistent with the time permitted for secondary containment to be inoperable per Condition A of LCO 3.6.5.1.

The NRC staff has evaluated the impact of this note on the licensee's design-basis radiological consequence analyses to ensure that the proposed change will not result in an increase in the dose consequences and that the resulting calculated doses remain within the current radiological consequence analyses.

The proposed addition of the note to SR 4.6.5.1.a does not change the TS requirement to meet SR 4.6.5.1. SR 4.6.5.1.a requires verification that the secondary containment is at a negative pressure. In addition, TS LCO 3.6.5.3, "Filtration, Recirculation and Ventilation System (FRVS)," must be met; otherwise, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

As discussed above, secondary containment operability is based on its ability to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. To prevent ground level exfiltration of radioactive material, the secondary containment pressure must be maintained at a pressure that is less than atmospheric pressure. The secondary containment requires support systems to maintain the control volume pressure less than atmospheric pressure. Following an accident, the FRVS ensures the secondary containment pressure is less than the external atmospheric pressure. During normal operation, non-safety-related systems are used to maintain the secondary containment at a negative pressure. However, during normal operation, it is possible for the secondary containment vacuum to be momentarily less than the required vacuum for a number of reasons. These conditions may not be indicative of degradations of the secondary containment boundary or of the ability of the FRVS system to perform its specified safety function. Since the licensee meets the requirements of SR 4.6.5.1 following the actions of TS LCO 3.6.5.1, and the licensee's analysis confirms secondary containment operability by confirming that the combination of four (of six) FRVS recirculation units and one FRVS ventilation unit (out of two) is capable of performing its specified safety function, there is reasonable assurance that the secondary containment and FRVS subsystem will maintain the vacuum requirements during a DBA.

Therefore, the NRC staff has determined that the changes continue to provide (1) the ability to maintain the secondary containment pressure during an accident at a vacuum that is consistent with the accident analyses, and (2) to draw down the secondary containment pressure within the time assumed in the accident analyses. Thus, the secondary containment can perform its safety function and may be considered TS operable. This is evident by being able to successfully perform and meet secondary containment integrity in SR 4.6.5.1.c.1. This SR requires the FRVS to establish and maintain the required vacuum in the secondary containment as assumed in the accident analyses.

Furthermore, because the specified safety functions of the secondary containment and FRVS subsystem can be performed in the time assumed in the licensee's accident analysis, the fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment prior to release to the environment, will be contained and processed as assumed in the licensee's design-basis radiological consequence dose analyses. The NRC staff finds that the proposed changes do not affect the current radiological consequence analyses and concludes that the proposed change is acceptable with respect to the radiological consequences of DBAs.

3.3 Proposed Changes to Surveillance Requirement 4.6.5.1.b.2.a

The NRC staff review was limited to the licensee's request to provide an allowance for the brief, simultaneous opening of redundant secondary containment access doors during normal entry and exit conditions. Normal entry and exit conditions do not include planned activities that could result in the simultaneous opening of redundant secondary containment access openings, such as maintenance of a secondary containment personnel access door or movement of large equipment through the openings that would take longer than the normal transit time.

The NRC staff reviewed the proposed changes to SR 4.6.5.1.b.2.a. The NRC staff determined that the SR continues to provide appropriate confirmation that secondary containment boundary doors are properly positioned and capable of performing their function in preserving the secondary containment boundary. The NRC staff determined that the SRs continue to appropriately verify the operability of the secondary containment and provide assurance that the necessary quality of systems and components are maintained in accordance with 10 CFR 50.36(c)(3).

Additionally, the NRC staff evaluated the impact of modifying the licensee's TSs to allow secondary containment access openings to be open for entry and exit on the licensee's design-basis radiological consequence dose analyses to ensure that the modification will not result in an increase in the radiation dose consequences and that the resulting calculated radiation doses will remain within the design criteria specified in the current radiological consequence analyses. The NRC staff review of these DBAs determined that there is one DBA that takes credit for the secondary containment and is possibly impacted by the brief, simultaneous opening of both an inner and outer access door during normal entry and exit conditions, the loss-of-coolant accident (LOCA).

3.3.1 Loss-of-Coolant Accident

Following a LOCA, the secondary containment structure is maintained at a negative pressure, ensuring that leakage from primary containment to secondary containment can be collected and filtered prior to release to the environment. The FRVS performs the function of maintaining a negative pressure within the secondary containment, as well as collecting and filtering the

leakage from primary containment. The FRVS consists of two engineered safety feature subsystems, the FRVS-VS and the FRVS-RS. The FRVS-VS processes and filters air from the containment before it is released to the environment. The FRVS-RS cleans contaminated air recirculated through the secondary containment. The FRVS-VS is designed to exhaust sufficient air from the reactor building to maintain a negative pressure in that building and to remove airborne radioactive materials before discharging the air to the environment. The FRVS-VS takes suction only from the discharge duct of the FRVS-RS. The licensee credits the FRVS system for mitigation of the radiological releases from the secondary containment. In the LOCA analysis, the secondary containment drawdown analysis assumes that FRVS system can draw down the secondary containment within 375 seconds to a pressure of 0.25 inches of vacuum water gauge.

The licensee assumes that releases into the secondary containment prior to the 375-second drawdown time leak directly to the environment as a ground level release with no filtration. After the assumed 375-second drawdown, these releases are filtered by the FRVS and released to the environment.

Based on this information, the NRC staff concludes that the licensee's DBA LOCA analysis has sufficient conservatism by assuming a drawdown time of 375 seconds from the start of the DBA LOCA. Margin exists to ensure that the secondary containment can be reestablished during a brief, simultaneous opening of the inner and outer doors, and there is reasonable assurance that a failure of a safety system needed to control the release of radioactive material to the environment will not result. The brief, simultaneous opening of the secondary containment access doors does not impact the design bases and will not result in an increase in any onsite or offsite dose.

Based on the above discussion, the NRC staff finds that the licensee's proposed changes to the TSs do not impact the licensee's design-basis LOCA radiological consequence analysis and will not result in an increase in any onsite or offsite dose. Therefore, the NRC staff concludes that these changes are acceptable with respect to the radiological consequences of the DBAs.

The licensee was approved to use the AST methodology and the radiological dose consequences analyses for DBAs by License Amendment No. 146, dated April 15, 2003 (ADAMS Accession No. ML030760293), for Hope Creek and most recently updated the radiological dose consequences analyses for DBAs by License Amendment No. 184, dated October 7, 2010 (ADAMS Accession No. ML102710156). Chapter 15 of the Hope Creek Updated Final Safety Analysis Report describes the DBAs and their radiological consequence analysis results. The NRC staff reviewed the impact of the proposed changes to the Hope Creek TSs on all DBAs currently analyzed in the Hope Creek Updated Final Safety Analysis Report that could have the potential for significant dose consequences and determined that the proposed TS SR changes will not result in an increase in any onsite or offsite dose.

3.4 Summary

The NRC staff reviewed the proposed changes and determined that changes to the TSs meet the standards for TSs in 10 CFR 50.36(b). The proposed SRs 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 and satisfy 10 CFR 50.36(c)(3). Additionally, the changes to the TSs were reviewed for technical clarity and consistency with customary terminology and format in accordance with SRP Chapter 16.

Additionally, the NRC staff has evaluated the impact of the proposed changes on the design-basis radiological consequence analyses against the regulatory requirements and guidance identified in Section 2.3 of this safety evaluation. The NRC staff finds, with reasonable assurance, that the licensee's changes to the TSs will continue to comply with the requirements of the current radiological consequence analyses. Therefore, the proposed changes are acceptable regarding the radiological consequences of the postulated DBAs.

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 July 22, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements 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 (84 FR 25839). 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: E. Dickson

Date: September 6, 2019

SUBJECT: HOPE CREEK GENERATING STATION – ISSUANCE OF AMENDMENT NO. 218 TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-551, "REVISE SECONDARY CONTAINMENT SURVEILLANCE REQUIREMENTS" (EPID L-2019-LLA-0083) DATED SEPTEMBER 6, 2019

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