UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

November 7, 2019

Mr. Eric Carr
President and Chief Nuclear Officer
PSEG Nuclear, LLC
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION, UNIT 1 – INTEGRATED INSPECTION REPORT 05000354/2019003

Dear Mr. Carr:

On September 30, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Hope Creek Generating Station in Hancocks Bridge, New Jersey. On October 9, 2019, the NRC inspectors discussed the results of this inspection with Mr. Steve Poorman, Plant Manager and other members of your staff. The results of this inspection are documented in the enclosed report.

One finding of very low safety significance (Green) is documented in this report. This finding did not involve a violation of NRC requirements.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; and the NRC Resident Inspector at Hope Creek.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* (10CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

Sincerely,

/RA/

Brice A. Bickett, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 05000354
License No. NPF-57

Enclosure:
Inspection Report 05000354/2019003

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SUBJECT: HOPE CREEK GENERATING STATION – INTEGRATED INSPECTION REPORT 05000354/2019003 DATED NOVEMBER 7, 2019

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	05000354
License Number:	NPF-57
Report Number:	05000354/2019003
Enterprise Identifier:	I-2019-003-0044
Licensee:	PSEG Nuclear, LLC
Facility:	Hope Creek Generating Station
Location:	Hancocks Bridge, NJ 08038
Inspection Dates:	July 01, 2019 to September 30, 2019
Inspectors:	A. Ziedonis, Senior Resident Inspector J. Patel, Resident Inspector J. Hawkins, Senior Resident Inspector A. Patel, Senior Reactor Inspector J. Schoppy, Senior Reactor Inspector L. Dumont, Reactor Inspector
Approved By:	Brice A. Bickett, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting an integrated inspection at Hope Creek Generating Station in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

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Open/Closed</td><td>[P.5] - Operating Experience</td><td>71152</td></tr>
<tr><td colspan="4">The inspectors identified a Green finding (FIN) because PSEG did not adequately monitor and analyze safety relief valve (SRV) performance data, in accordance with ER-AA-2003, "System Performance Monitoring and Analysis," Revision 11. Specifically, PSEG's main steam system performance monitoring plan did not identify appropriate SRV tailpipe temperature (TPT) values to detect critical leakage degradation and establish an adequate action plan in anticipation of exceeding action levels. As a result, the 'H' SRV operated for extended periods of elevated TPT, with mischaracterized leakage rates and an inadequate action plan to address the degraded conditions.</td></tr>
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Additional Tracking Items

None.

PLANT STATUS

The Hope Creek Generating Station (Hope Creek) began the inspection period performing power ascension from approximately 85 percent rated thermal power (RTP) on July 1, due to a planned removal of selected feedwater heating to support cycle extension methods in accordance with the operating license. The station returned to RTP later the same day. On July 21, power was reduced to approximately 87 percent, due to ambient weather conditions and corresponding limitations in main condenser backpressure. The station returned to approximately 95 percent on July 23, limited by ambient weather conditions and cycle-specific plant conditions.

On August 3, the station performed a rapid load reduction and inserted a manual reactor scram from approximately 37 percent power, after operators identified degraded vacuum conditions due to an equipment failure. On August 5, operators performed a reactor startup and the station achieved 95 percent power on August 10, limited by cycle-specific plant conditions. The station remained at or near 95 percent until August 15, when power was reduced to approximately 80 percent, to perform planned removal from service of selected feedwater heating. The station returned to approximately 93 percent on August 16, and began an end-of-cycle coastdown period.

On September 3, from approximately 86 percent power, operators performed a manual trip of the 'D' circulating water pump, in response to degraded indications, which resulted in an expected automatic recirculation pump runback to approximately 60 percent thermal power. Operators raised power to approximately 87% later the same day, limited by cycle-specific plant conditions. The station remained in an end-of-cycle coastdown period until the end of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors performed plant status activities described in IMC 2515 Appendix D, "Plant Status" and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.04Q - Equipment Alignment

Partial Walkdown Sample (IP Section 03.01) (3 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) 'B' residual heat removal train with 'A' train out of service for planned maintenance on July 1
(2) Station service water system with the 'D' traveling water screen out of service on July 16
(3) Reactor core isolation cooling system, after realignment to standby conditions following manual reactor vessel level control on August 5

71111.04S - Equipment Alignment

Complete Walkdown Sample (IP Section 03.02) (1 Sample)

(1) The inspectors evaluated system configuration during a complete walkdown of the core spray system on September 23

71111.05Q - Fire Protection

Quarterly Inspection (IP Section 03.01) (5 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Circulating water pump house, following receipt of a fire alarm on September 4
(2) Service water intake structure room 213 & 313 on September 9
(3) Reactor building corridor RB2 on September 12
(4) Reactor auxiliary cooling system pumps and heat exchangers room RB1 on September 18
(5) Lower control equipment room CD26 on September 18

71111.11Q - Licensed Operator Requalification Program and Licensed Operator Performance

Licensed Operator Performance in the Actual Plant/Main Control Room (IP Section 03.01) (1 Sample)

(1) The inspectors observed and evaluated licensed operator performance in the main control room during reactor and plant startup from a manual scram on August 5

Licensed Operator Requalification Training/Examinations (IP Section 03.02) (1 Sample)

(1) The inspectors observed and evaluated a crew of licensed operators in the plant's simulator during licensed operator re-qualification training that involved a seismic event, loss of containment integrity, a safety relief valve failing open, and torus leak that required emergency depressurization on August 6

71111.12 - Maintenance Effectiveness

Routine Maintenance Effectiveness Inspection (IP Section 02.01) (3 Samples)

The inspectors evaluated the effectiveness of routine maintenance activities associated with the following equipment and/or safety significant functions:

(1) Safety relief valve maintenance effectiveness on September 3

(2) Reactor core isolation cooling system maintenance rule (a)(1) evaluation on September 30
(3) 'H' safety relief valve discharge vacuum breaker maintenance effectiveness on September 24

Quality Control (IP Section 02.02) (1 Sample)

The inspectors evaluated maintenance and quality control activities associated with the following equipment performance activities:

(1) 'B' residual heat removal heat exchanger leak temporary repair during the week of July 29

71111.13 - Maintenance Risk Assessments and Emergent Work Control

Risk Assessment and Management Sample (IP Section 03.01) (4 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Emergent work associated with 'C' emergency diesel generator expanded maintenance scope and extent of condition on July 16
(2) Emergent work associated with unplanned maintenance of 'B' residual heat removal heat exchanger during week of July 22
(3) Emergent work associated with 'C' emergency diesel generator jacket water pump seal repair on September 4
(4) 'B' emergency diesel generator extent of condition repairs on September 10

71111.15 - Operability Determinations and Functionality Assessments

Operability Determination or Functionality Assessment (IP Section 02.02) (4 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) 'C' emergency diesel generator with low fuel oil storage tank level on July 16
(2) 'B' residual heat removal train following heat exchanger gasket leakage on July 30
(3) New fuel shipment, following discovery of manufacturing process imperfections and evaluation to "accept as-is," on August 14
(4) Feedwater heater 4A removal from service, and associated core operating limit report change, to achieve final feedwater temperature reduction limit for cycle 22 on August 15

71111.18 - Plant Modifications

Temporary Modifications and/or Permanent Modifications (IP Section 03.01 and/or 03.02) (1 Sample)

The inspectors evaluated the following temporary or permanent modifications:

(1) 'B' residual heat removal heat exchanger leak temporary repair on July 30

71111.19 - Post-Maintenance Testing

Post-Maintenance Test Sample (IP Section 03.01) (6 Samples)

The inspectors evaluated the following post maintenance tests:

(1) Residual heat removal test return motor operated valve F024A, following motor control center cubicle replacement on July 3
(2) 'C' emergency diesel generator protective relays on July 14
(3) 'B' residual heat removal heat exchanger, following leak repairs on August 6
(4) 'A' residual heat removal pump breaker protective relay, following test failure on August 13
(5) 'C' emergency diesel generator jacket water and intercooler pump seal replacement on September 9
(6) 'D' emergency diesel generator load sequencer card A3-7 on September 25

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

Surveillance Tests (other) (IP Section 03.01) (3 Samples)

(1) HC.OP-ST.BF-0002, control rod drive accumulator operability check on July 2
(2) HC.OP-ST.KJ-0002, 'C' emergency diesel generator monthly operability test on July 15
(3) HC.IC-CC.SE-0006, channel 'B' intermediate range monitor channel calibration on September 27

Inservice Testing (IP Section 03.01) (1 Sample)

(1) HC.OP-IS.BD-0001, reactor core isolation cooling pump OP203 inservice test on August 20

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification

The inspectors verified licensee performance indicators submittals listed below:

MS06: Emergency AC Power Systems (IP Section 02.05) (1 Sample)

(1) July 1, 2018 - June 30, 2019

MS07: High Pressure Injection Systems (IP Section 02.06) (1 Sample)

(1) July 1, 2018 - June 30, 2019

MS08: Heat Removal Systems (IP Section 02.07) (1 Sample)

(1) July 1, 2018 - June 30, 2019

MS09: Residual Heat Removal Systems (IP Section 02.08) (1 Sample)

(1) July 1, 2018 - June 30, 2019

MS10: Cooling Water Support Systems (IP Section 02.09) (1 Sample)

(1) July 1, 2018 - June 30, 2019

71152 - Problem Identification and Resolution

Annual Follow-up of Selected Issues (IP Section 02.03) (1 Sample)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) Review of root cause evaluation 70206428 for 'H' safety relief valve leakage on September 16

71153 - Followup of Events and Notices of Enforcement Discretion

Event Followup (IP Section 03.01) (1 Sample)

(1) Event response inspection performed following manual reactor scram (event notification 54198) on August 3

INSPECTION RESULTS

Inadequate SRV Performance Monitoring and Analysis			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000354/2019003-01 Open/Closed	[P.5] - Operating Experience	71152
The inspectors identified a Green finding (FIN) because PSEG did not adequately monitor and analyze safety relief valve (SRV) performance data, in accordance with ER-AA-2003, “System Performance Monitoring and Analysis,” Revision 11. Specifically, PSEG's main steam system performance monitoring plan did not identify appropriate SRV tailpipe temperature (TPT) values to detect critical leakage degradation and establish an adequate action plan in anticipation of exceeding action levels. As a result, the ‘H’ SRV operated for extended periods of elevated TPT, with mischaracterized leakage rates and an inadequate action plan to address the degraded conditions.			
Description: Hope Creek has 14 safety-related main steam SRVs that provide reactor pressure vessel overpressure protection. The ‘H’ SRV consists of two stages: a pilot stage that senses main steam system pressure, and a main stage that actuated by the pilot stage to relieve elevated main steam system pressure. Each SRV discharges through a tailpipe that directs the steam through a downcomer into a suppression pool. Internal SRV leakage has been known in the industry to occur through the pilot and/or main stages, even under normal main steam system operating pressures. Industry operating experience (OE) indicates the consequences of SRV leakage can include relief setpoint drift, failure of the pilot stage to reseat, delayed actuation, spurious opening, suppression pool cooling margin reduction,			

cycling of the SRV tailpipe vacuum breakers, and condensate-induced void collapse stresses on the tailpipe t-quencher.

The inspectors reviewed the main steam system performance monitoring plan, and noted that it required weekly monitoring of SRV TPTs by system engineering. The inspectors also noted that PSEG identified elevated 'H' SRV TPT, beginning in November 2016 at the beginning of cycle 21, and monitored elevated TPT through start-up for cycle 22 operation in the spring of 2018. PSEG replaced the 'H' pilot stage during the refueling outage at the end of cycle 21, but did not replace the main stage. A highlight of elevated 'H' SRV TPT in cycle 21 and 22 is included below:

- On November 21, 2016, approximately one week after start-up for cycle 21, notification (NOTF) 20748846 identified the 'H' SRV TPT rose from 125 degrees to 210 degrees F. The inspectors noted that the main steam performance monitoring plan specified a normal SRV TPT temperature range of 90-190 degrees F.
- On April 12, 2018, NOTF 20789878 identified the 'H' SRV TPT rose above 220 degrees during the planned shutdown at the end of cycle 21, and documented that a TPT of 225 degrees correlated with a leak rate of approximately 5 pound-mass per hour (lbm/hr), in accordance with calculation AB-0076, Tailpipe Temperature versus Leak Rate for the 'H' SRV, Revision 0.
- On October 1, 2018, NOTF 20806034 captured a rising SRV leak rate trend from 155 lbm/hr in June of 2018, to a current value 325 lbm/hr at 223 degrees. The NOTF did not discuss the large increase in correlated leakage for approximately the same TPT reported on April 12, 2018.
- On December 20, 2018, NOTF 20814836 identified a periodic loud banging noise in the torus area, which was a recurrence of a similar banging noise in 2014 due to 'H' SRV leakage and condensate-induced water hammer in the discharge line to the torus (see NCV 2014-005-01).

During cycle 22, PSEG engineering was performing informal weekly monitoring of SRV leak rates, and the inspectors periodically requested to review the data as part of routine inspection. During review of leak rate data in late December of 2018, the inspectors identified data from October and November reflected significant changes from data previously reported for the same time period. Specifically, as reported in late November of 2018, the weekly leak rate data from October and November of 2018 was reported to range between 155 lbm/hr and 245 lbm/hr. However, on January 2, 2019, the data from the October and November of 2018 was reported to range between 613 lbm/hr and 755 lbm/hr. The inspectors questioned PSEG as to the reason for the change in the data, and the station ultimately determined the leak rate trend data was incorrectly monitored from August of 2018 through December of 2018, and initiated NOTF 20816775. The inspectors also noted that equipment apparent cause evaluation (EQACE) 70168630, performed in response to 2014 'H' SRV leakage that resulted in torus noise and a maintenance outage, determined that calculation AB-0076 was not accurate, and assigned an action to address it, but the action was subsequently cancelled.

On December 27, 2018, in response to NOTF 20814836, documenting 2018 torus noise as mentioned above, PSEG initiated action to develop adverse condition monitoring plan (ACM) 19-001. On February 4, 2019, PSEG determined the 'H' SRV leak rate was approximately 961 lbm/hr, utilizing ACM 19-001, which established enhanced monitoring and analysis methods similar to ACM 14-0014 following the onset of torus noise in 2014.

On February 12, 2019, PSEG determined that a planned maintenance outage would be scheduled in the Spring of 2019, in response to ‘H’ SRV leakage that was projected to exceed the ACM limit for torus heat-up rate prior to the summer period. The limit was established under Technical Evaluation 80124191-0020, to support compliance with the Technical Specification 3.6.2.1.a.2 suppression pool temperature limit of 95 degrees.

On March 28, 2019, Hope Creek entered a planned maintenance outage to replace the ‘H’ SRV pilot and main stages. As-found bench testing of the ‘H’ SRV determined that the ‘H’ SRV lifted within the Technical Specification limits of +/- 3 percent, identified pilot stage seat leakage to be minimal, and identified main seat leakage was very high at approximately 900 lbm/hr. Hope Creek operations procedures state that spurious opening may occur with leakage as low as 200 lbm/hr. The inspectors also noted that industry owner’s group reports state that industry data shows plants may be effected at approximately 2000 lbm/hr, either spurious opening or relief setpoint drift, though the data can vary from plant to plant. As such, industry owner’s group reports also state that some plants use 1000 lbm/hr as an administrative limit for defining operator actions (e.g., plant shutdown).

The inspectors review PSEG procedure ER-AA-2003. Step 4.2.3 required the system engineer to identify the data required to effectively detect critical degradation that can effect performance, and step 4.4.3 required establishing action plans to address conditions in anticipation of exceeding action levels. An action level is defined as the parameter which, when reached, indicates that preventive or corrective maintenance is required. Within the main steam system performance monitoring plan, there was an action plan established at a specific TPT, which required notification of the shift manager to consider plant shutdown in accordance with the limits established in operations procedure HC.OP-SO.SN-0001, Nuclear Pressure Relief and Automatic Depressurization System Operation (controlled copy), Attachment 2. The inspectors reviewed Attachment 2, and noted that it established unique SRV TPT shutdown limits for each 2-stage SRV. However, the TPT limits were based on calculations that PSEG had previously identified to be inaccurate, as described in 2005 evaluation 70049218 and 2014 EQACE 70168360. The inspectors questioned the adequacy of an action plan and procedure limits based on calculations with known inaccuracies. In response, PSEG wrote NOTF 20836010 to evaluate any necessary changes to the monitoring plan and operations procedure. Establishing proper SRV action plans is important, because industry operating experience (OE) indicates that excessive SRV leakage can potentially result in a plant transient or challenge critical safety functions.

Therefore, the inspectors determined that PSEG was not in compliance with ER-AA-2003, steps 4.2.2 and 4.4.3, because PSEG’s SRV performance monitoring plans did not identify appropriate SRV TPT values to detect critical leakage degradation and establish an adequate action plan in anticipation of exceeding action levels.

Corrective Actions: PSEG’s corrective actions included performing a maintenance outage on March 28, 2019, to replace the ‘H’ SRV pilot and main valves, performing a root cause evaluation (RCE) 70206428 to evaluate repeat elevated main seat leakage on the ‘H” SRV.

Corrective Action References: 20748846, 20789878, 20806034, 20814836, 20806034, 20818407*, 20819899*, 20823472*, 70206428, 70205765 and 70205995

Performance Assessment:

Performance Deficiency: The inspectors determined that PSEG’s SRV performance monitoring plans did not identify appropriate SRV TPTs to detect critical leakage degradation,

and did not establish an adequate action plan in anticipation of exceeding action levels, in accordance with ER-AA-2003, steps 4.2.2 and 4.4.3. Screening: The inspectors determined the performance deficiency was more than minor, because it was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, the inspectors determined that not adequately monitoring and analyzing SRV performance during conditions of elevated TPT and leakage affected PSEG's ability to limit likelihood of an events such as spurious SRV opening; and PSEG's ability to limit the likelihood of challenging critical safety functions such as suppression pool cooling and SRV pressure relief at the Technical Specification setpoint. Significance: The inspectors assessed the significance of the finding using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." Cross-Cutting Aspect: P.5 - Operating Experience: The organization systematically and effectively collects, evaluates, and implements relevant internal and external operating experience in a timely manner. P.5 - OE: The organization systematically and effectively collects, evaluates, and implements relevant internal and external OE in a timely manner. This finding, in accordance with IMC 0310, has a cross-cutting aspect in the Problem Identification and Resolution area associated with Operating Experience, in that PSEG did not systematically and effectively collect, evaluate, and implement relevant internal operating experience in a timely manner. Specifically, PSEG's main steam performance monitoring plan did not implement relevant internal operating experience to effectively monitor and analyze critical leakage degradation prior to operation with elevated 'H' SRV TPT in 2018 and 2019. For example, and PSEG identified in 2005 and 2014 that SRV TPT and leak rate calculations were not accurate, and used an ACM to calculate leak rates in 2014, but did not incorporate these learnings into the system monitoring plan. (P.5)
Enforcement: Inspectors did not identify a violation of regulatory requirements associated with this finding.

Observation: Missed opportunities associated with the 'H' safety relief valve (SRV) and its discharge line vacuum breaker (VB)	71152

The inspectors identified the following missed opportunities associated with the 'H' safety relief valve (SRV) and its discharge line vacuum breaker (VB):

- 'H' SRV monitoring and CAP evaluation missed opportunities:
    - PSEG did not download and graph SRV TPT data from operating Cycle 21 (November 2016 through April 2018), which was needed to properly trend the recorder data during periods of elevated 'H' SRV tailpipe temperature (20819899).
    - 2019 RCE 70206428, and 2014 Equipment Apparent Cause Evaluation (EQACE) 70168360, stated that 2014 was the first occurrence of 'H' SRV main seat leakage, but missed an opportunity to include 'H' SRV leakage evaluated in 2011 RCE 70119769.
    - 2019 RCE 70206428 confirmed a previous NRC observation that 2014 EQACE 70168360 cause determination, performed in response to 'H' SRV leakage, was not supported with adequate technical rigor (20803213).

- ‘H’ SRV vacuum breaker missed evaluation opportunities:
  - PSEG did not consider the ‘H’ SRV vacuum breaker as a potential source of elevated drywell leakage during periods of ‘H’ SRV leakage and condensate-induced water hammer in operating cycle 22, after the vacuum breaker was found by to be failed open during preventive maintenance in the spring of 2018 refueling and maintenance outage (20818407 and 20821310).
  - PSEG did not update the ‘H’ SRV failed-open vacuum breaker CAP evaluation in a timely manner once results from the failure analysis were obtained (20823472).

The inspectors did not identify that any of the above observations would be considered more than minor performance issues.

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On October 9, 2019, the inspectors presented the integrated inspection results to Mr. Steve Poorman, Plant Manager and other members of the licensee staff.