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

November 14, 2018

EA-18-044

Mr. Peter P. Sena, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION UNIT 1 – INTEGRATED INSPECTION REPORT AND EXERCISE OF ENFORCEMENT DISCRETION 05000354/2018003

Dear Mr. Sena:

On September 30, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Hope Creek Generating Station (HCGS). On October 10, 2018, the NRC inspectors discussed the preliminary results of this inspection with Mr. Eric Carr, Site Vice President, and other members of your staff. After additional review of specific items, the inspectors discussed the final results of this inspection with Mr. Ed Casuli, Site Plant Manager, and other members of your staff on November 14, 2018. The results of this inspection are documented in the enclosed report.

NRC inspectors documented two findings of very low safety significance (Green) in this report. One of these findings involved a violation of NRC requirements. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the Enforcement Policy.

Separately, a violation involving not setting secondary containment during operations with a potential for draining the reactor vessel (OPDRVs) was identified during the HCGS refueling outage (H1R21). Specifically, from April 19, 2018 to April 29, 2018, while all other Technical Specifications (TSs) were met, HCGS conducted several OPDRVs without maintaining secondary containment integrity, which is a violation of TS 3.6.5.1, “Secondary Containment Integrity.” NRC issued Enforcement Guidance Memorandum (EGM) 11-003, “EGM on Dispositioning Boiling Water Reactor Licensee Noncompliance with TS Containment Requirements during Operations with a Potential for Draining the Reactor Vessel,” on October 4, 2011, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. The EGM, which was most recently revised on January 15, 2016, also required that licensees receiving discretion must submit a license amendment request (LAR) to accept the NRC’s generic change to the Standard TS that will allow a graded approach to OPDRV requirements. The LAR was required to have been submitted and accepted for review by December 20, 2017, in order to continue receiving enforcement discretion while the LAR is being reviewed. By letter dated September 21, 2017, PSEG submitted the required license amendment request (ADAMS accession: ML17265A847). Because the NRC has determined that PSEG has met all criteria, and enforcement discretion was previously authorized for the site as EA-17-071, and the violation occurred during the period while the LAR described in the EGM was under NRC review, the NRC is exercising enforcement discretion and will not issue enforcement action for this violation.

If you contest the violations or the significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at HCGS. In addition, 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 HCGS.

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 the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Fred L. Bower, III, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket No. 50-354
License No. NPF-57

Enclosure:
Inspection Report 05000354/2018003

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SUBJECT: HOPE CREEK GENERATING STATION UNIT 1 – INTEGRATED INSPECTION REPORT 05000354/2018003 DATED NOVEMBER 14, 2018

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Number:	50-354
License Number:	NPF-57
Report Number:	05000354/2018003
Enterprise Identifier:	I-2018-003-0066
Licensee:	PSEG Nuclear LLC (PSEG)
Facility:	Hope Creek Generating Station (HCGS)
Location:	Hancocks Bridge, NJ 08038
Inspection Dates:	July 1, 2018 to September 30, 2018
Inspectors:	J. Hawkins, Senior Resident Inspector S. Haney, Resident Inspector J. Furia, Senior Health Physicist
Approved By:	Fred L. Bower, III, Chief Reactor Projects Branch 3 Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring PSEG's performance at Hope Creek Generating Station (HCGS) Unit 1 by conducting the baseline inspections described in this report 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. NRC identified and self-revealed findings, violations, and additional items are summarized in the table below. No licensee-identified non-cited violations are documented in this report.

List of Findings and Violations

Inadequate Procedures for Fuel Conditioning Results in Multiple Fuel Leaks			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Reactor Safety – Barrier Integrity	Green NCV 05000354/2018003-03 Closed	H.6 – Human Performance – Design Margins	71153 (a.1)
The inspectors documented a self-revealing Green NCV of TS 6.8.1, Procedures and Programs, when PSEG did not maintain adequate procedures for fuel conditioning. Specifically, PSEG's procedure for selecting the appropriate fuel pellet-cladding interaction (PCI) rules, NF-AB-440, BWR Fuel Conditioning, did not provide adequate guidance for protection of the fuel coming out of the April 2018 refueling outage (RF21). As a result, PSEG's selection of non-conservative PCI rules resulted in three PCI fuel cladding leaks.			

Inadequate Procedures for Restoration of the 'A' Reactor Feed Pump Turbine Following Maintenance			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Reactor Safety – Initiating Events	Green FIN 05000354/2018003-02 Closed	H.3 – Human Performance – Change Managment	71153 (a.2)
A self-revealing Green finding (FIN) was identified for PSEG's inadequate procedures that controlled the restoration of the 'A' reactor feedwater pump turbine (RFPT) trip instrumentation following system maintenance. Specifically, the pump's axial position instrumentation was not re-zeroed following a rotor replacement. As a result, on May 21, 2018, the 'A' RFPT tripped while HCGS was operating at approximately 97 percent rated thermal power (RTP), which led to an unplanned automatic recirculation runback to approximately 70 percent of RTP.			

Additional Tracking Items

Type	Issue number	Title	Inspection Results Section	Status
LER	05000354/2018-001-00	Operations with a Potential to Drain the Reactor Vessel (OPDRV) without Secondary Containment	71153 (b.1)	Closed
LER	05000354/2018-002-00 and -01	Safety Relief Valve (SRV) As-found Setpoint Failure	71153 (b.2)	Closed
LER	05000354/2018-003-00 and -01	Feedwater Isolation Valve Leakage Exceeded Technical Specification Limit	71153 (b.3)	Closed

PLANT STATUS

Hope Creek Generating Station (HCGS) began the inspection period at 100 percent RTP and operated at full power until July 19, 2018, when HCGS conducted a planned down power to 65 percent RTP to support power suppression testing. Operators returned the unit to 100 percent RTP on July 21, 2018. On September 20, 2018, HCGS operators conducted a planned down power to 55 percent RTP to support turbine valve testing, rod pattern adjustments, and feedwater heater corrective maintenance. Operators returned the unit to 100 percent RTP on September 22, 2018, and remained at or near 100 percent RTP for the remainder 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 PSEG performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards."

REACTOR SAFETY

71111.01 - Adverse Weather Protection

External Flooding (1 Sample)

The inspectors evaluated readiness to cope with external flooding (walkdowns of all external areas of the plant, including the auxiliary building, emergency diesel generators, and service water intake structure (SWIS) between September 3 and 10, 2018).

71111.04 - Equipment Alignment

Partial Walkdown (3 Samples)

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

(1) High pressure coolant injection (HPCI) system while the reactor core isolation cooling (RCIC) system was out of service on August 22, 2018
(2) 'D' RHR system while 'C' RHR system was out of service on September 6, 2018
(3) 'B' standby liquid control system while performing inservice testing on September 12, 2018

Complete Walkdown (1 Sample)

The inspectors evaluated system configurations during a complete walkdown of the RCIC system on August 16, 2018.

71111.05AQ - Fire Protection Annual/Quarterly

Quarterly Inspection (5 Samples)

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

(1) HCGS fire protection system jockey pump failure and Salem fire protection system cross-tie opening on July 2, 2018
(2) Technical Support Center fire penetration seal found empty and nonfunctional on July 10, 2018
(3) Remote shutdown panel room on July 12, 2018
(4) ‘A’ Class 1E safety-related switchgear room on August 1, 2018
(5) HPCI transfer/isolation switch on September 11, 2018

71111.06 - Flood Protection Measures

Internal Flooding (1 Sample)

The inspectors evaluated internal flooding mitigation protections in the (SWIS) on August 31, 2018

71111.11 - Licensed Operator Requalification Program and Licensed Operator Performance

Operator Requalification (1 Sample)

The inspectors observed and evaluated a crew of licensed operators in the plant’s simulator during licensed operator requalification training that involved a failed drywell pressure transmitter, a safety auxiliaries cooling system (SACS) pump trip, a safety relief valve (SRV) failing open, a loss of all reactor feed pumps, a SRV tailpipe break resulting in high drywell pressure, an overspeed trip of RCIC, and a failure of the HPCI auxiliary oil pump to auto start when required on August 20, 2018.

Operator Performance (1 Sample)

The inspectors observed and evaluated a planned down power and the performance of power suppression testing (PST) for a suspected fuel leak on July 20, 2018.

71111.12 - Maintenance Effectiveness

Routine Maintenance Effectiveness (2 Samples)

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

(1) Reactor auxiliary cooling system (RACS) flow controller failure July 10, 2018
(2) Scram discharge volume inboard vent isolation valve stroke time testing failures on September 13, 2018

Quality Control (1 Sample)

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

(1) RCIC and 'B' RHR nuclear measurement analysis and control (NUMAC) leak detection monitoring circuit card failures on July 23, 2018

71111.13 - Maintenance Risk Assessments and Emergent Work Control (5 Samples)

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

(1) Planned maintenance on the FLEX Godwin pumps (2 out of 3) resulting in loss of capability July 11, 2018
(2) RACS flow control replacement due to failure July 12, 2018
(3) Planned open phase modification on the 10A404 safety-related bus, breaker 08, on July 25, 2018
(4) Planned fire protection system maintenance on August 2, 2018
(5) 'C' emergency diesel generator functional testing on August 8, 2018

71111.15 - Operability Determinations and Functionality Assessments (5 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) RHR shutdown cooling suction header pressure high alarms increasing and abnormal trend on July 5, 2018
(2) 'B' feedwater supply isolation check valve (F032B) excessive leakby on August 14, and September 12, 2018
(3) RHR safety-related snubber technical evaluation supporting past operability on July 17, 2018
(4) Measurement uncertainty recapture modification produces less gain than expected on August 21, 2018
(5) RCIC 250 Volt (V) direct current (DC) battery low cell number 4 voltage on September 11, 2018

71111.19 - Post Maintenance Testing (7 Samples)

The inspectors evaluated post maintenance testing for the following maintenance/repair activities:

(1) RCIC flow controller circuit card failure troubleshooting and repairs on July 16, 2018
(2) RCIC and 'B' RHR NUMAC leak detection monitoring circuit card failure troubleshooting and repairs on July 18, 2018
(3) PST, fuel leak troubleshooting, and multiple control rod suppressions on July 20, 2018

(4) HPCI 250 VDC battery surveillance following cell cleaning and maintenance on July 31, 2018
(5) Class 1E safety-related battery, CD411, cell number 24 replacement on August 6, 2018
(6) RHR breaker troubleshooting and repair on September 6, 2018
(7) Reactor water cleanup and main steam isolation valve Division I NUMAC leak detection monitor troubleshooting and repairs on September 18, 2018

71111.22 - Surveillance Testing

The inspectors evaluated the following surveillance tests:

Routine (2 Samples)

(1) Review of HC.OP-ST.GS-0004, Suppression Chamber/Drywell Vacuum Breaker monthly operability surveillance testing, and surveillance test interval evaluation (STI-17-004) on September 6, 2018
(2) HC.OP-IS.BC-0003, 'B' RHR pump in-service test on September 12, 2018

In-service (2 Samples)

(1) HC.OP.IS.BJ-0101, HPCI valves in-service testing on July 11, 2018
(2) HC.OP-IS.BD-0001, RCIC in-service testing on August 22, 2018

71114.06 - Drill Evaluation

Drill/Training Evolution (1 Sample)

The inspectors observed a simulator training evolution for licensed operators that involved raising reactor power, a failed drywell pressure transmitter, a SACS pump trip, an SRV failing open, a loss of all reactor feed pumps, a SRV tailpipe break resulting in high drywell pressure, an overspeed trip of RCIC, and a failure of the HPCI auxiliary oil pump to auto start when required on September 11, 2018.

RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

71124.07 - Radiological Environmental Monitoring Program

Site Inspection (1 sample)

The inspectors walked down various thermoluminescent dosimeter and air and water sampling locations and reviewed associated calibration and maintenance records. The inspectors observed the sampling of various environmental media as specified in the offsite dose calculation manual. The inspectors reviewed the groundwater monitoring program as it applies to selected potential leaking structures, systems, and components , and 10 CFR 50.75(g) records of leaks, spills, and remediation since the previous inspection.

Groundwater Protection Initiative Implementation (1 sample)

The inspectors reviewed: groundwater monitoring results; changes to the Groundwater Protection Initiative program since the last inspection; anomalous results or missed groundwater samples; leakage or spill events including entries made into the decommissioning files (10 CFR 50.75(g)); evaluations of surface water discharges; and PSEG's evaluation of any positive groundwater sample results including appropriate stakeholder notifications and effluent reporting requirements.

OTHER ACTIVITIES – BASELINE

71151 - Performance Indicator Verification (5 Samples)

The inspectors verified PSEG's performance indicator submittals for the Mitigating Systems Performance Index (MSPI) listed below.

(1) Emergency AC power systems (MS06; July 1, 2017 through June 2018)
(2) High pressure injection system (MS07; July 1, 2017 through June 2018)
(3) Heat removal system (MS08; July 1, 2017 through June 2018)
(4) Residual heat removal system (MS09; July 1, 2017 through June 2018)
(5) Cooling water system (MS10; July 1, 2017 through June 2018)

71152 - Problem Identification and Resolution

Annual Follow-up of Selected Issues (3 Samples)

The inspectors reviewed PSEG's implementation of its CAP related to the following issues:

(1) Notification (NOTFs) 20797582, 20797038, 20800580, Adverse trend of SWIS degraded structural components
(2) NOTFs 20659947, 20794237, 20792630, and 20794371, Recent equipment issues experienced on the 'H' main steam SRV and SRV discharge line
(3) NOTFs 20799124 and 20799402, Recent inspector questions involving RCIC system preventive maintenance

71153 - Follow-up of Events and Notices of Enforcement Discretion

(a) Events (2 Samples)

The inspectors evaluated PSEG's response to the following event:

(1) Downpower, power suppression testing, fuel leak troubleshooting performed by PSEG on July 20, 2018. This also included a review of PSEG's root cause evaluation (70202192) completed on September 7, 2018.
(2) Trip of the 'A' reactor feedwater pump and subsequent reactor recirculation pump automatic runback to 70 percent RTP on May 21, 2018. This included a review of PSEG's causal evaluation (70201021) completed on July 12, 2018.

(b) Licensee Event Reports (3 Samples)

The inspectors evaluated the following licensee event reports (LERs):

(1) LER 05000354/2018-001-00, Operation with a Potential to Drain the Reactor Vessel (OPDRV) Without Secondary Containment (ADAMS Accession: ML18169A198). The circumstances surrounding this LER are documented in the 'Inspection Results' section of this report.

(2) LER 05000354/2018-002-00 and -01, Safety Relief Valve (SRV) As-found Setpoint Failure (ADAMS Accession: ML18169A199 and ML18276A022). The circumstances surrounding this LER are documented in the 'Inspection Results' section of this report.

(3) LER 05000354/2018-003-00 and -01, Feedwater Isolation Valve Leakage Exceeded Technical Specification Limit (ADAMS Accession: ML18169A307 and ML18255A232). The circumstances surrounding this LER are documented in the 'Inspection Results' section of this report.

INSPECTION RESULTS

Inadequate Procedures for Fuel Conditioning Results in Multiple Fuel Leaks			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Reactor Safety – Barrier Integrity	Green NCV 05000354/2018003-03 Closed	H.6 – Human Performance – Design Margins	71153 (a.1)
The inspectors documented a self-revealing Green NCV of TS 6.8.1, Procedures and Programs, when PSEG did not maintain adequate procedures for fuel conditioning. Specifically, PSEG's procedure for selecting the appropriate fuel PCI rules, NF-AB-440, BWR Fuel Conditioning, did not provide adequate guidance for protection of the fuel during restart from the April 2018 refueling outage (RF21). As a result, PSEG's selection non-conservative PCI rules resulted in three PCI fuel leaks.			

Description: HCGS is currently in operating Cycle 22 with a modified control cell core design strategy. During RF21, a large number of GE14 fuel assemblies were replaced with new GNF2 fuel assemblies. Each of the four Group 10A banked position withdrawal sequence control cells are at the center of two twice-burned GE14 (new fuel in Cycle 20) fuel assemblies and two once-burned GNF2 (new fuel in operating Cycle 21) fuel assemblies that form a control cell. During Cycle 21, prior to RF21, the GE14 fuel assemblies were in the second row from the periphery (lower power), or outer edge of the reactor core. In RF21, some of these fuel assemblies were moved from the outer edge, inward toward the core center (higher power) into the cells surrounding the four Group 10A control cells. This type of movement is known to create a configuration that may reduce margin to pellet-clad interation related failures.

Pellet-clad interaction (PCI) is degradation mechanism that occurs when fuel pellets within the fuel rod swell and contact the fuel cladding. PCI related fuel leaks occur due to stress-corrosion cracking of the fuel cladding. In a new fuel rod, there is a gap between the fuel pellet and the fuel cladding. As the fuel pellet is irradiated, it expands and the gap closes, putting the fuel pellet in direct contact with the fuel cladding. High cladding stress can result prior to the fuel cladding becoming conditioned to the contact forces. This stress is highly dependent on the rate of change of local power, which is the basis for establishing fuel conditioning or PCI rules, thresholds, and ramp rates to control the rate of local power change.

The fuel vendor's PCI guidance is specified in procedure GNF-0142-5151, "GNF Fuel Operating Guidelines," and is used by plants for both core design and core operation. PSEG initially incorporated these guidelines as part of NF-AB-440, BWR Fuel Conditioning. There are two standard options specifying thresholds and ramp rates for the fuel: Option A used by most plants; and, Option B used for situations when there is heightened concern for PCI suseptibility, and therefore treats the fuel more gently by slowing the local increase in power, primarily by controlling factors such as ramp rates, control rod movement and sequence exchanges more restrictively. The choice of which option to use is left up to the utility, based on their calculated PCI risk or suspetibility. Revision 12 of NF-AB-440, which occurred on March 20, 2018, removed the detailed guidance that was provided in GNF-0142-5151, and therefore did not have adequate means of determining appropriate option to select for controlling PCI strategy.

On June 11, 2018, a few weeks after the reactor was started up from RF21, PSEG performed a planned power reduction to fully withdraw banked position withdrawal sequence Group 10A control rods. After PSEG completed the rod pattern adjustment, chemistry reported a sample that indicated a potential fuel leak. Power suppression testing (PST) was performed by PSEG on June 29, 2018, and identified a fuel leak in the cell with control rod 22-31, one of the four Group 10A control rods. This control rod was fully inserted to suppress the local power around the leaking fuel bundle and minimize further degradation during the cycle.

On July 13, 2018, there was an increase in chemistry samples that were consistent with a new fuel leak. PST was again performed by PSEG on July 20, 2018, which identified fuel leaks in the cells corresponding to control rods 30-39 and 38-31, two of the other three remaining Group 10A control rods. PSEG fully inserted these two control rods to suppress the local power around the leaking fuel bundle and minimize further degradation during the cycle.

With input from the fuel vendor (GNF), PSEG's causal evaluation (RCE 70202192) determined that the fuel leaks were PCI induced by the selection of the incorrect option for fuel conditioning. Specifically, on June 11, 2018, during the power reduction to fully withdraw the four Group 10A control rods, PSEG did not apply PCI rules from GNF-0142-5151 since they were no longer incorporated as part of NF-AB-440. Instead, PSEG applied their own legacy rules based on historical operating experience. These PCI rules did not account for the promotion of the GE14 fuel assemblies from the second row from the periphery (low power) into a high power area of the reactor core. This type of promotion is known to create a configuration that may reduce margin to PCI-related leak. When promotions occur, GNF PCI guidelines provide information on the proper selection of PCI rules to reduce the possibility of PCI leaks when the fuel assemblies are exposed to significantly higher power than they experienced in their previous position. PSEG did not recognize the risks associated with the removal of the detailed guidance referenced in GNF-0142-5151 from their own operating procedures. Consequently, Revision 12 of the procedure for BWR Fuel Conditioning, NF-AB-440, provided inadequate guidance for the selection of the appropriate PCI rules, and resulted in the PCI-related fuel leaks of promoted GE14 fuel assemblies in three of the Group 10A control cells between June 11, and July 13, 2018.

Corrective Actions: PSEG's immediate corrective actions included conducting PST and fully inserting the three control rods in cells that contain fuel leaks to suppress the local power around the leaks and minimize further degradation during the operating cycle. PSEG also revised their fuel conditioning procedures to return the detailed guidance for determining PCI rules and strategy. PSEG also updated the core monitoring system with the new PCI rule requirements, established a more robust reload design process to include vendor reviews, and developed more oversight of nuclear fuels to ensure the right level of reviews are performed.

Corrective Action References: 20801980 and RCE 70202192.

Performance Assessment:

Performance Deficiency: The inspectors determined that PSEG not maintaining adequate procedural guidance for fuel conditioning, specifically for selecting the appropriate fuel PCI rules was a performance deficiency within PSEG's ability to foresee and correct, and should have been prevented. Specifically, Revision 12 to PSEG's procedure for selecting the appropriate fuel PCI rules, NF-AB-440, BWR Fuel Conditioning, did not provide adequate guidance for the selection of the appropriate PCI rules coming out of RF21, and on June 11, 2018, during a power reduction to fully withdraw the four Group 10A control rods, the selection of non-conservative PCI rules resulted in three PCI fuel leaks occurring between June 11, and July 13, 2018.

Screening: This performance deficiency was considered more than minor because the performance deficiency was associated with the Procedure Quality attribute of the Barrier Integrity Cornerstone and adversely impacted the cornerstone objective to provide reasonable assurance that the fuel cladding physical design barrier protects the public from radionuclide releases caused by accidents or events. Specifically, PSEG not appropriately incorporating GNF fuel conditioning limits into HCGS's procedures resulted in fuel clad damage and fuel leaks as a result of pellet-clad interaction that increased the possibility of adversely impacting nuclear safety.

Significance: The inspectors evaluated the issue in accordance with Exhibit 1 and 3 of IMC 0609, Appendix A, "The SDP for Findings At-Power," and determined to be of very low safety significance (Green) because the issue did not involve pressurized thermal shock, or a reactor trip and the loss of mitigation equipment, or a significant amount of fuel leaks.

Cross-Cutting Aspect: This finding has a cross-cutting aspect in the area of Human Performance, Design Margin, because PSEG did not operate and maintain equipment within design margins, did not carefully guard those design margins and change them only through a systematic and rigorous process, and did not ensure that special attention was placed on maintaining fission product barriers, defense-in-depth, and safety related equipment. Specifically, the inspectors found that PSEG did not carefully guard revisions made to their fuel conditioning and PCI rules through a systematic and rigorous process, leading to multiple fuel leaks. (H.6)

Enforcement:

Violation: TS 6.8.1, "Procedures and Programs," states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix 'A' of RG 1.33, Revision 2, February 1978. RG 1.33, Appendix A, Section 2, General Plant Operating Procedures, lists procedures for power operation and refueling and core alterations. The purpose of PSEG's procedure for BWR Fuel Conditioning, NF-AB-440, was to review, approve and implement fuel vendor fuel conditioning instructions, develop PSEG specific fuel conditioning instructions, and establish requirements for the control of these instructions.

Contrary to this requirement, between March 20, 2018 and July 13, 2018, Revision 12 of PSEG's procedure for selecting the appropriate fuel PCI rules, NF-AB-440, did not provide adequate guidance for the selection of the appropriate PCI rules. Specifically, on June 11, 2018, during a power reduction to fully withdraw the four Group 10A control rods, PSEG's selection of non-conservative PCI rules due to their inadequate procedure, resulted in three PCI fuel leaks occurring between June 11, and July 13, 2018.

Disposition: This violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy.

Inadequate Procedures for Restoration of the 'A' Reactor Feed Pump Turbine (RFPT) Following Maintenance

Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Reactor Safety – Initiating Events	Green FIN 05000354/2018003-02 Closed	H.3 – Human Performance – Change Managment	71153 (a.2)

A self revealing Green finding (FIN) was identified for PSEG's inadequate procedures for controlling the restoration of the 'A' RFPT trip instrumentation following maintenance. Specifically, on May 21, 2018, the 'A' RFPT tripped while HCGS was operating at approximately 97 percent RTP due to the pump's axial position instrumentation not being re-zeroed following a rotor replacement. The trip of the 'A' RFPT resulted in an unplanned automatic recirculation runback to approximately 70 percent RTP.

Description: HCGS has three steam-driven RFPTs that are part of the feedwater system, which purifies and preheats condensed steam from the main condenser before returning it to the reactor vessel. The RFPTs are operated by a digital control system that monitors reactor water level. All three RFPTs must be in-service to support 100 percent RTP.
On May 21, 2018, the 'A' RFPT tripped while HCGS was operating at approximately 97 percent RTP that resulted in an unplanned automatic recirculation runback to approximately 70 percent RTP. All of HCGS's systems responded appropriately to the runback.

PSEG's investigation (NOTF 20795822) and follow-up causal evaluation (ACE 70201021) determined that the 'A' RFPT automatically tripped due to exceeding the rotor axial position trip setpoint of 0.025 inches. The 'A' RFPT was overhauled during the refueling outage in May 2018, and the rotor was replaced per WO 30305396. The three rotor axial position instrumentation probes were re-installed using guidance in maintenance procedures, HC.MD-CM.FW-0002 - RFPT Overhaul and WO 30305396. The guidance was expected to re-zero the rotor axial position instrumentation probes for the newly installed rotor.

PSEG determined that during the restoration of the RFPT, the rotor axial position instrumentation probes were only re-zeroed in one of the two 'A' RFPT instrumentation and control cabinets. One of these cabinets provides rotor axial position instrumentation display data and input to the MCR OHAs, and the other cabinet provides input to the RFPT trip function. Because the second control cabinet was not re-zeroed and synchronized with the first, the indicated position trip setpoint within the second cabinet was offset in the direction of the trip value resulting in reduction of margin between the indicated position and the actual trip setpoint. Combined with small differences in axial dimensions between the original ‘A’ RFPT rotor and the newly installed rotor, and normal amount of rotor deflection that occurs between the unloaded and fully loaded conditions led to the RFPT exceeding its axial position offset trip setpoint. PSEG also determined that a 2015 design change package (DCP 80105383), which affected the RFPT instrumentation and control, did not identify the need to revise the procedure controlling and configuring the RFPT axial probe information after an overhaul.

Based on the information above, the inspectors determined that the maintenance procedures, HC.MD-CM.FW-0002 for RFPT Overhaul and WO 30305396, that included steps re-zero the rotor axial position instrumentation probes did not contain sufficient guidance to ensure that both RFPT instrumentation and control cabinets were re-zeroed and synchronized following RFPT maintenance.

Corrective Actions: PSEG’s corrective actions included performing an extent of condition on the other two RFPT instrumentation and control cabinets and revising the RFPT maintenance procedures to ensure clear guidance about re-zeroing and synchronizing the two control cabinets during system restoration.

Corrective Action References: 20795822

Performance Assessment:

Performance Deficiency: PSEG procedure HC.MD-CM.FW-0002 for RFPT Overhaul and WO 30305396 were inadequate to control the restoration of the ‘A’ RFPT trip instrumentation following system maintenance. Specifically, on May 21, 2018, the ‘A’ RFPT tripped while HCGS was operating at approximately 97 percent of RTP because the turbine’s axial position instrumentation was zeroed in the first instrument and control cabinet that provides the display function and was not re-zeroed in the second cabinet that provides the trip function and the two cabinets were synchronized following the RFPT’s rotor replacement.

Screening: The performance deficiency is more than minor because it was associated with the Equipment 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.

Significance: The inspectors evaluated the finding in accordance with Exhibit 1 of IMC 0609, Appendix A, “SDP for Findings At-Power,” and determined the finding was of very low safety significance (Green) because it did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. Specifically, the trip of the ‘A’ RFPT resulted in an unplanned automatic recirculation runback to approximately 70 percent RTP, but did not result in a reactor trip or a complete loss of reactor feedwater.

Cross-Cutting Aspect: This finding has a cross-cutting aspect in the area of Human Performance, Change Management, because PSEG did not use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. Specifically, when a 2015 design change was implemented for the RFPT instrumentation and

control cabinets, PSEG did not identify the need to revise the procedure controlling and configuring the RFPT axial probe information after an overhaul. (H.3)	
Enforcement: This finding does not involve enforcement action because no violation of regulatory requirements was identified. Because the finding does not involve a violation of regulatory requirements and has very low safety significance, it is identified as a finding.	

Enforcement Discretion	Enforcement Action (EA)-18-044: EGM on Dispositioning BWR Licensee Noncompliance With TS Containment Requirements During Operations With A Potential For Draining The Reactor Vessel (EGM-11-003)	71153 (b.1)

Description: From April 19 through April 29, 2018, HCGS performed OPDRVs without establishing secondary containment integrity. An OPDRV is an activity that could result in the draining or siphoning of the reactor pressure vessel water level below the top of fuel, without crediting the use of mitigating measures to terminate the uncovering of fuel. TS 3.6.5.1, "Secondary Containment Integrity," requires that secondary containment integrity be maintained, and is applicable during OPDRVs. The required action for this specification without secondary containment integrity in this condition of applicability is to suspend OPDRVs. As reported in LER 05000354/2018-001, HCGS conducted the following OPDRVs during the period of secondary containment inoperability:

- Control rod drive mechanism replacements;
- Local power range monitor replacements; and
- Cavity let down via Reactor Water Clean Up system.

Additionally, an unplanned OPDRV occurred due to RHR system relief valves seat leakage.

NRC EGM 11-03, "EGM on Dispositioning BWR Licensee Noncompliance With TS Containment Requirements During Operations With A Potential For Draining The Reactor Vessel," Revision 3, provides, in part, for the exercise of enforcement discretion only if the licensee demonstrates that it has met specific criteria during an OPDRV activity. The inspectors assessed that HCGS adequately implemented these criteria.

In accordance with EGM 11-003, in order to continue to receive enforcement discretion, a license amendment request (LAR) must be submitted and accepted for review within 12 months of the NRC staff's publication of the generic change that occurred on December 20, 2016. The inspectors verified that PSEG submitted the required LAR on September 20, 2017 (ADAMS Accession No. ML17265A847), and that it was subsequently accepted by the NRC for review by a letter dated October 25, 2017 (ADAMS Accession No. ML17299A009).

Corrective Action: PSEG submitted an LAR to adopt TS Task Force Traveler 542, Reactor Pressure Vessel Water Inventory Control, on September 20, 2017, that was subsequently accepted by the NRC for review on October 25, 2017. (After the end of the inspection period, on October 30, 2018, the NRC staff responded (ML18260A203) to PSEG's LAR dated September 20, 2017, and issued License Amendment No. 213 that revised the technical specifications to adopt TSTF-542, Revision 2.

Corrective Action Reference: 20792923

Enforcement:

Violation: TS 3.6.5.1, "Secondary Containment Integrity," requires that secondary containment integrity be maintained, and is applicable during OPDRVs. The required action for this specification without secondary containment integrity in this condition of applicability is to suspend OPDRVs.

Contrary to the above, from April 19 through April 29, 2018, HCGS performed OPDRVs without secondary containment integrity. Therefore, set and maintain secondary containment integrity during OPDRVs without suspending the operation was considered a condition prohibited by TSs as defined by 10 CFR 50.73(a)(2)(i)(B).

Basis for Discretion: The NRC is exercising enforcement discretion in accordance with Section 3.5, "Violations Involving Special Circumstances," of the NRC Enforcement Policy because all criteria described in EGM 11-003 were met and enforcement discretion was previously authorized by EA-2017-071; therefore, no enforcement action will be issued for this violation.

The disposition of this violation closes LER 05000354/2018-001-00.

Observation	71153 (b.2)

Licensee Event Report 05000354/2018-002-00 and -01: Safety Relief Valve (SRV) As-found Setpoint Failure

On April 20, 2018, PSEG received results that the 'as-found' setpoint tests for safety relief valve (SRV) pilot stage assemblies had exceeded the lift setting tolerance prescribed in Technical Specification (TS) 3.4.2.1. The TS requires the SRV lift settings to be within +/- 3 percent of the nominal setpoint value. During the twenty-first refueling outage (H1R21), all fourteen SRV pilot stage assemblies were removed for testing at an offsite facility. Between April 20 and May 11, 2018, HCGS received the test results for all fourteen of the SRV pilot valve assemblies. A total of eight of the fourteen SRV pilot stage assemblies experienced setpoint drift outside of the TS 3.4.2.1 specified values. All of the valves failing to meet the limits were Target Rock Model 7567F two-stage SRVs. As a result, PSEG reported this condition as this is a condition prohibited by plant TS under 10 CFR 50.73(a)(2)(i)(B) to the NRC as Licensee Event Reports 05000354/2018-003-00 and -01, Safety Relief Valve (SRV) As-found Setpoint Failure, on June 18, and October 3, 2018. The inspectors performed inspections documented in Section 4OA2 of this report. The inspectors did not identify any findings or violations of NRC requirements during the review of this Licensee Event Report. This review closes LER 05000354/2018-002-00 to the open unresolved item (URI) 05000354/2018001-02, Concern Regarding As-Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable Limit. This review also closes Supplemental LER 05000354/2018-002-01.

Observation	71153 (b.3)

Licensee Event Report 05000354/2018-003-00 and -01: Feedwater Isolation Valve Leakage Exceeded Technical Specification Limit

On April 18, 2018, during a planned refueling outage, PSEG performed a required surveillance test of the long term seal of the feedwater lines. The test criteria could not be met due to leakage past feedwater isolation valve (F032B). This valve is sealed with a water

seal from the HPCI or RCIC system to form a long-term seal boundary of the feedwater lines. The valve is tested per TS 4.6.1.2.d to verify a maximum leak rate of 10 gallons per minute (gpm) at a test pressure of 55.7 pounds per square inch gauge (psig). During the local leak rate test (LLRT), a test pressure of 44 psig was the highest pressure that could be obtained, which did not meet the acceptance criteria. As a result, PSEG reported this as a condition prohibited by plant TS under 10 CFR 50.73(a)(2)(i)(B) to the NRC as Licensee Event Reports (LER) 05000354/2018-003-00 and -01, Feedwater Isolation Valve Leakage Exceeded Technical Specification Limit, on June 18, and September 12, 2018. The inspectors performed inspections documented in Sections 71111.15 and 4OA3 of this report. The inspectors identified a TS violation and minor performance deficiencies during the review of these LERs that are documented in the Minor Violation and Minor Performance Deficiency Sections of this report. This review closes LER 05000354/2018-003-00 and Supplemental LER 05000354/2018-003-01.

Minor Violation	71153 (b.3)
Minor Violation: During the review of LER 05000354/2018-003-00 and -01, Feedwater Isolation Valve Leakage Exceeded Technical Specification Limit, the inspectors identified a condition prohibited by TS. Specifically, TS 3.6.1.2.d requires that Primary Containment Leakage rates shall be limited to a combined leakage rate of less than or equal to 10 gpm for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines, when tested at 1.10 P_a (1.1 times the calculated peak containment internal pressure related to the design basis accident) or 55.7 psig. TS surveillance requirement (SR) 4.6.1.2.g states that these valves be tested at least once per 18 months. Contrary to this requirement, on April 18, 2018, during the TS required SR for LLRT of the F032B, PSEG was unable to achieve the required test pressure and could not determine a leakage rate. Screening: The inspectors evaluated the issue above in accordance with the guidance in the NRC's Enforcement Policy, IMC 0612, Appendix B, "Issue Screening," and Appendix E, "Examples of Minor Issues," and determined the issue was a minor violation because, although PSEG did not successfully complete the TS required SR because they could not attain the required test pressure, there were no actual safety consequences. Specifically, PSEG's technical evaluation (70200206-0085) estimated the leak rate through the F032B to be approximately 3 gpm, and determined that the potential leakage through the F032B would not have posed a challenge to its ability to establish and maintain the required feedwater seal for 30 days post-LOCA. Enforcement: PSEG has taken actions to restore compliance by repairing and successfully testing the valve, and revising their LLRT procedures to: 1) update administrative limits and actions that are required when limits are exceeded; and, 2) include specify the exact size and length of tubing required for the testing. This inability to comply with TS 3.6.1.2.d constituted a minor violation that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.	

Minor Performance Deficiency	71153 (b.3)
Minor Performance Deficiency: During the review of LER 05000354/2018-003-00 and -01, Feedwater Isolation Valve Leakage Exceeded Technical Specification Limit, the inspectors identified the following two minor PDs: 1. PSEG did not follow their procedure, ER-HC-380-1005, HCGS Specific Appendix J Program Information, Step 2.2.1, which states that "leakage rates above the administrative	

limit but less than the action limit are considered indicative of potential component sealing performance degradation, warranting troubleshooting of test results, and removal of extended interval testing eligibility." Contrary to this, on October 22, 2016, PSEG did not troubleshoot the F032B when it exceeded its administrative limit for LLRT.

2. PSEG did not follow their F032B LLRT procedure, HC.OP-LR.AE-0003, which requires the use of "sufficient size and length of tubing" as part of the LLRT test rig, in order to successfully complete the LLRT. Contrary to this, on April 18, 2018, PSEG did not use sufficient size or the proper length of tubing required to complete the F032B LLRT, and was unable to achieve the TS required test pressure.

Screening: The inspectors evaluated the issue above in accordance with the guidance in IMC 0612, Appendix B, "Issue Screening," and Appendix E, "Examples of Minor Issues," and determined the issue was of minor significance because the limit exceeded was an administrative limit, and based on testing history, the F032B leakage rates have always been low when compared to the TS limit of 10 gpm. In addition to this, PSEG's technical evaluation (70200206-0085) estimated the leak rate through the F032B to be approximately 3 gpm, and determined that the potential leakage through the F032B would not have posed a challenge to its ability to establish and maintain the required feedwater seal for 30 days post-LOCA.

Observation	71152 (2)

Review of Equipment Issues Associated with the 'H' SRV and SRV Discharge Line:

The inspectors performed an in-depth review of PSEG's evaluation and corrective actions associated with multiple equipment issues experienced on the 'H' main steam safety relief valve (SRV) and SRV discharge line. Specifically:

1. 'H' SRV Main Seat Leakage
   August 2014 (NOTF 20659947; ACE 70168360) documented loud cyclic banging noises coming from the TORUS area. PSEG determined that there was significant leakage past the 'H' SRV main seat due to the existence of cold spring in the tailpipe during installation of the valve (NOTF 20661387 and NCV 05000354/2014005-01);

2. 'H' SRV High Tailpipe Temperature
   April 2018 and May 2018 (NOTF 20789878, 20794091 and 20794237) documented that during down power for and the start up from RF21, the 'H' SRV tailpipe temperature spiked up to 220 degrees Fahrenheit which is indicative of potential SRV main and/or pilot valve leakage;

3. 'H' SRV Vacuum Breaker Failure
   April 2018 (NOTF 20792630 and ERE 70199676) documents that one of the 'H' SRV discharge line vacuum breakers (F037H) failed open due to a missing locknut and damage caused by high vibrations and poor maintenance practices from item #1 above; and,

4. 'H' SRV Pilot As-Found Lift Test Failures
   May 2018 (NOTF 20794371, 70200658, and LERs 05000354/2018-002-00 and -01) documented the 'H' SRV pilot as-found setpoint testing. Eight of HCGS's fourteen SRV pilots lifted high (above the 3 percent TS limit). The 'H' SRV pilot was the only valve that lifted high on the first and second as-found lift testing (8.3 and 3.3 percent).

[*Note that the 2-stage SRVs, manufactured by Target Rock, of which HCGS has 13 2-stage and 1 3-stage SRVs, have been subject to setpoint drift, typically in the increased setpoint direction at a number of boiling water reactor nuclear power plants, and that the specific setpoint drift issue will be addressed by the unresolved item (URI) opened in NRC Inspection Report, URI 05000354/2018001-02, Concern Regarding As-Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable Limit*.]

The inspectors reviewed associated documents and interviewed personnel to assess the adequacy of PSEG's actions. The inspectors also reviewed SRV main and pilot testing results, tailpipe temperature, main steam vibration records, and acoustic monitoring data. The inspectors found the following issues during their review of the events listed above:

The inspectors found that PSEG had an extended timeline (6 months) and a lack of prioritization and ownership of the disassembly of the 'H' SRV pilot due to it lifting high twice (NOTF 20799218*). Based on the inspector's questions regarding timeliness, PSEG initiated a NOTF and actions to disassemble and inspect the pilot four months ahead of its original schedule. As a result of the disassembly, PSEG's determined that the pilot disc and valve body were severely steam cut and worn, with unknown impurities on the valve pilot disc. PSEG initiated work group evaluation (WGE) 70200658 to evaluate these unexpected conditions;

WGE 70200658 was completed on September 21, 2018, for the 'H' SRV failed setpoint lift test high twice in which PSEG determined that the first high test lift was due to corrosion bonding, and the second high test lift was due to pilot valve wear between the disc and liner caused by steam cutting from a pilot leak during the last operating cycle. PSEG's WGE found that some of the unknown impurities were cobalt and nickel oxide due to the corrosion bonding experienced by the valve. The WGE did not determine the source of the lead (Pb) in the impurities but pointed to the valve material test report that cites 0.5 percent of the total valve disc material being from 'OTHER' material. The inspectors reviewed PSEG's conclusion and discussed with PSEG on September 27, 2018, that the site is still awaiting feedback from the vendor and BWROG about the potential source of the lead in the impurities.

The inspectors determined that there was insufficient information provided by PSEG in licensee event report (LER 2018-002) for the as-found testing results of the SRV pilots, specifically, no information on the 'H' SRV pilot lifting high twice was reported As a result, PSEG initiated NOTF 20799025* and took corrective actions (70201546) to change their process for LER reviews to include a technical validation team review prior to submittal to the NRC;

The inspectors found that PSEG's procedure for SRV removal and installation, HC.MD-CM.AB-0006, was not revised in accordance with their causal evaluation (70168360) to include a step to unpin the spring can after installation of the SRV. PSEG initiated a NOTF with actions to revise the procedure and review all completed SRV work packages to ensure all pins were removed (NOTF 20801471*, 20803451*, and 70202115). As a result, PSEG's review found that three SRVs replaced in RF20 did not have any documentation that their spring cans had been unpinned. PSEG has created actions to conduct follow-up inspection of these SRVs ('J', 'K', and 'R') during the next refueling outage;

The inspectors found that PSEG's NOTF 20661387 and 70169063-0010 never validated a questionable spring can setting for the 'H' SRV due to a lack of understanding the issue. Because of this, inspectors also questioned the validity of PSEG's causal evaluation (70168360) conclusions based on the as-found cold spring being expected because of the piping configuration. The inspectors determined that during the development of the evaluation, PSEG did not consult the appropriate resources knowledgeable in pipe stress analysis. As a result, PSEG took action to validate that the spring can setting was correct and initiated NOTF 20803213* with a recommendation from engineering to review the causal evaluation's conclusions based on the inspector's questions and an independent engineering assessment. As of September 12, 2018, this recommendation was not supported by PSEG because the condition on the 'H' SRV is no longer present and there is no perceived value in performing the action. The inspectors noted that as of the end of this inspection period, PSEG initiated NOTF 20806034 on October 1, 2018, for degrading conditions associated with the 'H' SRV main seat leakage increasing from ~155 pound mass per hour (lbm/hr) to approximately 323 lbm/hr since H1R21 (June 2018), which is similar to the conditions that occurred on the 'H' SRV in August 2014, and were the subject of PSEG's causal evaluation (70168360).

The inspectors found that PSEG's WGE 70173184 had not determined a basis for what amount of displacement is considered unacceptable. In addition, PSEG had not performed trending of SRV piping misalignments as discussed in the WGE for RF19 (2015) and RF20 (2016). PSEG initiated NOTFs 20803211* and 20803212* to address the inspector's concerns and plans to perform extent of condition reviews of all SRV main replacements over the last few outages.

The inspectors evaluated all of the issues above in accordance with the guidance in IMC 0612, Appendix B, "Issue Screening," and Appendix E, "Examples of Minor Issues," and determined the issues were of minor significance because the inspectors did not identify any condition adverse to quality that were not appropriately corrected or scheduled for correction in a reasonable period of time as a result of PSEG's administrative delays, lack of prioritization, and insufficient information. Consequently, these issues are not subject to enforcement action in accordance with the NRC's enforcement policy.

Observation	71152 (3)

Review of Recent Inspector Questions Involving RCIC System Preventive Maintenance:

The inspectors performed an in-depth review of PSEG's evaluation and corrective actions involving RCIC system preventive maintenance (PM). Specifically, the inspectors reviewed the basis behind PSEG elimination of the RCIC cooling water pressure control valve (PCV) PM activity in July 2018. At the time, PSEG initiated multiple NOTFs (20799124 and 20799402) and actions (70201434 and 70201667) because the inspectors determined that the elimination of the PM had been performed without an adequate basis for elimination and had not been relocated to the inservice test (IST) procedure.

The inspectors assessed PSEG's problem identification threshold, problem analysis, extent of condition reviews, operating experience, compensatory actions, and the prioritization and timeliness of their corrective actions to determine whether PSEG staff were appropriately identifying, characterizing, and correcting problems associated with this issue, and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of PSEG's CAP, 10 CFR Part 50, Appendix B, and technical specifications. The inspectors reviewed associated documents and interviewed engineering personnel to assess the adequacy of PSEG's actions.

The inspectors determined that PSEG's actions (70201434 and 70201667) had not been prioritized in the CAP and had not been performed as originally planned by PSEG during the August 2018 RCIC IST. The actions involved evaluating whether the PCV discharge pressure readings were required and adding these pressure readings to the IST procedure or the system manager walkdown sheet (performed during IST). PSEG acknowledged the inspector's concerns with respect to the original actions not being performed and not meeting the original intent of the corrective actions.

The inspectors evaluated the issues above in accordance with the guidance in IMC 0612, Appendix B, "Issue Screening," and Appendix E, "Examples of Minor Issues," and determined the issues were of minor significance because the inspectors did not identify any conditions adverse to quality that was not appropriately corrected or scheduled for correction in a reasonable period of time as a result of PSEG's administrative delays, lack of prioritization, insufficient information. Consequently, these issues were not subject to enforcement action in accordance with the NRC's enforcement policy.

EXIT MEETINGS AND DEBRIEFS

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

- On October 10, 2018, the inspectors presented the preliminary quarterly resident inspector inspection results to Mr. Eric Carr, Site Vice President, and other members of the PSEG staff. The inspectors subsequently discussed the final results of this inspection with Mr. Ed Casuli, Site Plant Manager, and other PSEG staff on November 14, 2018.

- On August 31, 2018, the inspector presented the radiation safety inspection results to Mr. Heithwaite, REMP Coordinator, and other members of the licensee staff.

THIRD PARTY REVIEWS

The inspectors reviewed Institute of Nuclear Power Operations reports that were issued during the inspection period.

DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
HC.OP-AB.MISC-0001, Acts of Nature, Revision 31

Section 1R04: Equipment Alignment

Procedures
HC.OP-IS.BD-0001, Reactor Core Isolation Cooling Pump – OP203 – In-service Test, Revision 61
HC.OP-IS.BH-0004, Standby Liquid Control Pump – BP208 – In-service Test, Revision 14

Notifications
20794718 20800401 20801223 20801331

Maintenance Orders/Work Orders
30262548 50193227 50204333 50204874

Section 1R05: Fire Protection

Procedures
FP-HC-004, Actions for Inoperable Fire Protection – Hope Creek Station, Revision 5
FRH-II-362, Hope Creek Pre-Fire Plan: TSC Electrical, Mechanical, HVAC Equipment Rooms and Vent Stack Enclosure Elevations (ROOF), Revision 6
FRH-II-441, Hope Creek Pre-Fire Plan: Monitoring and Common Area, Computer Rooms and Console Room, Revision 3
FRH-II-451, Hoper Creek Pre-Fire Plan: General Work Area, Conference Room and Document Storage, Revision 3
FRH-II-541, Hope Creek Pre-Fire Plan: Class 1E Switchgear Room, Revision 7
FRH-III-714, Hope Creek Pre-Fire Plan: Fire Water Pump House, Revision 4
HC.FP-SV.ZZ-0026, Flood and Fire Barrier Penetration Seal Inspection, Revision 7
HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room, Revision 35

Drawings
M-22-0, Sheet 3, Fire Protection – Fire Water Reactor and Auxiliary Buildings, Revision 38

Notifications
20722147 20799606 20799704 20801105* 20801630* 20803400
20803533

Maintenance Orders/Work Orders
30240015 60136547 80120545

Miscellaneous
HC Fire Protection Impairment Tracking Report dated July 10, 2018
HC Fire Protection FASA 2018

Section 1R06: Flood Protection Measures

Procedures
HC.FP-SV.ZZ-0026, Flood and Fire Barrier Penetration Seal Inspection, Revision 8

Notifications
20790960

Maintenance Orders/Work Orders
50187889 70197453

Miscellaneous
HC.DE-PS.ZZ-0021, Hope Creek Penetration Seal Program, Revision 0

Section 1R11: Licensed Operator Requalification Program

Miscellaneous
SG-772, 2018 CPE Diagnostic Scenario, dated August 10, 2018

Section 1R12: Maintenance Effectiveness

Procedures
ER-AA-321, Administrative Requirements for In-service Testing, Revision 14
HC.OP-IS.BF-0101, Control Rod System Valves – In-service Test, Revision 20
HC.OP-AB.COOL-0003, Reactor Auxiliary Cooling, Revision 8

Notifications
20756234 20773926 20799118 20799542 20799692 20799867
20800221

Maintenance Orders/Work Orders
60089284 60124737 60139349 70201778 80122701

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
FP-HC-004, Actions for Inoperable Fire Protection – Hope Creek Station, Revision 5
FRH-II-541, Hope Creek Pre-Fire Plan: Class 1E Switchgear Room, Revision 7
HC.OP-AB.COOL-0003, Reactor Auxiliary Cooling, Revision 8
HC.OP-FT.KJ-0003, Emergency Diesel Generator 1CG400 – Functional Test, Revision 9
OP-HC-108-115-1001, Operability Assessment and Equipment Control Program, Revision 36
WC-AA-101, On-line Work Management Process, Revision 25

Drawings
M-22-0, Sheet 3, Fire Protection – Fire Water Reactor and Auxiliary Buildings, Revision 38

Notifications
20756234 20799692 20799867 20801306 20801307 20801308
20801309 20801494

<u>Maintenance Orders/Work Orders</u>

30303269	30313450	60139349	70179254

Section 1R15: Operability Determinations and Functionality Assessments

<u>Procedures</u>

HC.MD-ST.PK-0002, 125 Volt Quarterly Battery Surveillance, Revision 41
HC.OP-AR.ZZ-0014, Overhead Annunciator Window Box D3, Revision 39
HC.OP-AR.ZZ-0029, CRIDS Computer Points Book 10, Revision 23
HC.OP-GP.ZZ-0004, Reactor Coolant System Pressure Isolation Valve Leakage Determination, Revision 7
HC.OP-FT.ZZ-0006, Measurement Uncertainty Recapture implementation & Power Ascension Testing, Revision 0
HC.OP-SO.BC-0001, Residual Heat Removal System Operation, Revision 54
LS-AA-120, Issue Identification and Screening Process, Revision 13
OP-HC-108-115-1001, Operability Assessment and Equipment Control Program, Revision 32
SH.MD-GP.ZZ-0001, Snubber Removal and Installation, Revision 9
SH.RA-ST.ZZ-0105, Snubber Examination and Testing, Revision 10

<u>Notifications</u>

20740807	20797417*	20797558	20799705	20799842	20802467*
20802468*	20802469*	20803192	20803490	20803660	

<u>Maintenance Orders/Work Orders</u>

50162445	60131225	60139899	70074649	70200694	70201260
70201331	70201459	80116312			

Section 1R19: Post-Maintenance Testing

<u>Procedures</u>

HC.IC-CC.FC-0013, RCIC Turbine Steam – Division 2 Channel F-4158, S-4280 RCIC Pump Turbine Control (RSP), Revision 15
HC.IC-CC.SK-0009, NSSSS, RWCU AMB T, RWCU DT, and RWCU Flow – Division I Leak Detection Monitor H1SK-1SKXR-11497, Revision 23
HC.MD-GP.ZZ-0014, Single Cell Battery Charging, Replacement and Jumpering, Revision 26
HC.MD-GP.ZZ-0015, Battery Equalizing Charge, Revision 24
HC.MD-PM.PB-0001, 4.16 KV Breaker Cleaning and PM, Revision 29
HC.MD-ST.PK-0002, 125 Volt Quarterly Battery Surveillance, Revision 41

<u>Notifications</u>

20796685	20797125	20797958	20800020	20800156	20800224
20800436	20800995	20803261	20803484	20803490	

<u>Maintenance Orders/Work Orders</u>

30238551	50202962	50205332	60139140	60139569	60139876
60139899	70201064	70201508	80122784		

Section 1R22: Surveillance Testing

Procedures
HC.OP-IS.BD-0001, Reactor Core Isolation Cooling Pump – OP203 – In-service Test, Revision 61
HC.OP-IS.BJ-0101, High Pressure Coolant Injection System Valves – In-service Test, Revision 67
HC.OP-IS.BC-0003, 'B' Residual Heat Removal Pump In-service Test, Revision 50

Notifications

20788816	20796841	20800027

Maintenance Orders/Work Orders

50203237	50204333	60138269

Section 1EP6: Drill Evaluation

Procedures
HC.OP-AB.COOL-0002, Safety Turbine Auxiliaries Cooling System, Revision 8
HC.OP-AB.RPV-0006, Safety/Relief Valve, Revision 6
HC.OP-EO.ZZ-0102, Containment Control, Revision 14

Section 4OA1: Performance Indicator Verification

Procedures
LS-AA-2200, Mitigating Systems Performance Index Data Acquisition & Reporting, Revision 4

Notifications

20204119	20772359	20772476	20774620	20776162	20776163
20776165	20780552	20782371	20782624	20782625	20784226
20784837	20786975	20787547	20787622	20788227	20788811
20790870	20793085	20795727	20795825	20797121	20801660

Maintenance Orders/Work Orders

60135784	60137234	60137974	70041885	70195605	70196661
70196662	70196664	70197950	70197951	70197945	70198673
70198964	70201046				

Section 4OA2: Problem Identification and Resolution

Procedures
EP-HC-111-223, Containment Barrier, Revision 2
HC.OP-IS.BD-0001, Reactor Core Isolation Cooling Pump – OP203 – In-service Test, Revision 61
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set – OP204 and OP217 – In-service Test, Revision 65

Calculations/Engineering Evaluations
C-0139, MSL 'D' Pipe Stress Analysis, Revision 9

Notifications

20788227	20795966	20796705	20797038	20797483	20797582
20797762	20798753	20798962	20798963	20799124	20799129
20799145	20799146	20799209	20799402	20800580	

Maintenance Orders/Work Orders

30137511	60138705	70121456	70197831	70198964	70201018
70201079	70201434	70201667	80107006	80115269	80121631
80122776					

71153 - Follow-Up of Events and Notices of Enforcement Discretion

Procedures
CC-AA-309-101, Engineering Technical Evaluations, Revision 11
ER-AA-380, Primary Containment Leakrate Testing Program, Revision 10
ER-AA-380-1005, Determination of Administrative and Action Limits for the PSEG Appendix J Testing Program, Revision 0
HC.OP-FT.ZZ-0006, Measurement Uncertainity Recapture Power Ascension Testing, Revision 0
HC.OP-LR.AE-0101, Containment Isolation Valve Type C Leak Rate Test CIVs 1AEV-007 and 1AEHV-F074A (1AEV-006) Penetration P2B: 'A' Feedwater Line, Revision 3
HC.OP-SO.AE-0001, Feedwater System Operation, Revision 73
NF-AB-430, Failed Fuel Action Plan, Revision 8
NF-AB-440, BWR Fuel Conditioning, Revision 11
OP-HC-108-102, Management of Operations with the Potential to Drain the Reactor Vessel (OPDRV), Revision 6

Notifications/Orders

20757793	20792700	20792923	20793302	20795822	20797726
20799901	20800365	20801980	20802467*	20802468*	20802467*
20803247*					

Maintenance Orders/Work Orders

60138904	60139606	70189240	70192567	70198260	70200206
70200694	70201021	70201260	70201814	70202192	70202552
80105570					

Other Documents
HC 18-001, Cycle 22 Failed Fuel Monitoring Plan dated June 19, 2018