UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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KING OF PRUSSIA, PA 19406-1415

August 11, 2009

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

SUBJECT: HOPE CREEK GENERATING STATION – NRC INTEGRATED INSPECTION REPORT 05000354/2009003

Dear Mr. Joyce:

On June 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at the Hope Creek Generating Station. The enclosed inspection report documents the inspection results discussed on July 8, 2009, with Mr. George Barnes and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

The report documents one NRC-identified and two self-revealing findings of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, 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, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Hope Creek Generating Station. In addition, if you disagree with the characterization of any finding 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 Regional Administrator, Region I, and the NRC Resident Inspector at the Hope Creek Generating Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Arthur L. Burritt, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000354/2009003
w/Attachment: Supplemental Information

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Sincerely,
/RA/
Arthur L. Burritt, Chief
Projects Branch 3
Division of Reactor Projects

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-354
License No:	NPF-57
Report No:	05000354/2009003
Licensee:	PSEG Nuclear LLC
Facility:	Hope Creek Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	April 1, 2009 through June 30, 2009
Inspectors:	B. Welling, Senior Resident Inspector A. Patel, Resident Inspector J. Furia, Senior Health Physicist E. Gray, Senior Reactor Inspector E. Burket, Reactor Inspector E. Torres, Project Engineer G. Newman, Project Engineer
Approved By:	Arthur L. Burritt, Chief Projects Branch 3 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000354/2009003; 04/01/2009 – 06/30/2009; Hope Creek Generating Station; Flood Protection Measures, Event Follow-up.

This report covers a three-month period of inspection by resident inspectors and announced inspections by regional reactor inspectors, project engineers and a regional health physicist. Two Green non-cited violations (NCVs) and one Green finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006. The alpha-numeric references to cross-cutting aspects are described in IMC 0305, "Operating Reactor Assessment Program."

Cornerstone: Initiating Events

- Green. A finding was self-revealed because PSEG discovered an air leak at a soldered joint on the scram air header in September 2008, but did not enter the degraded condition in the corrective action program. As a result, PSEG did not evaluate the leak or take corrective actions prior to the joint separating, causing an automatic reactor scram. Following the event, PSEG repaired the affected joint, performed an extent-of-condition inspection of the corresponding joints on all other hydraulic control units, and placed this issue in the corrective action program.

  This issue was more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions. Specifically, by not identifying the air leak in the corrective action program, PSEG did not evaluate the degraded condition and its impact on the reliability of the scram air header. The inspectors determined that the finding was of very low safety significance (Green) based on a Phase I analysis. The finding increased the likelihood of a reactor scram, but did not contribute to the likelihood that mitigating equipment would not be available. The finding had a cross-cutting aspect in the area of problem identification and resolution because the station did not identify the scram air header leak completely, accurately, and in a timely manner commensurate with its safety significance. (P.1(a)) (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because safety auxiliary cooling system (SACS) water-tight door 4309A was blocked open without necessary compensatory measures, as a result of inadequate work instructions. Consequently, flood protection measures for the SACS system were degraded, which affected the capability of both SACS trains to perform their safety function during a flooding event. PSEG entered this issue into the corrective action program and promptly closed the water-tight door.

The issue was more than minor because it is associated with the external factors attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the capability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not having the required flooding compensatory measures in place when the water-tight door 4309A was open affected the reliability and capability of the SACS system during a postulated internal flooding event. The inspectors used Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," to determine the significance of the finding. Based upon the finding not involving a loss of control or thermal margin, this finding does not require a quantitative assessment and screens as having very low safety significance (Green). This finding had a cross-cutting aspect in the area of human performance because PSEG did not define and effectively communicate expectations regarding procedural compliance, and PSEG personnel did not follow procedures. Specifically, PSEG did not adequately follow PSEG procedure CC-AA-201, "Plant Barrier Control Program," to impair the water-tight door. (H.4(b)) (Section 1R06)

- Green. A self-revealing, non-cited violation of 10 CFR Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified because technicians did not have adequate work instructions for troubleshooting a high pressure coolant injection (HPCI) system instrumentation drawer. The instructions did not include appropriate steps to prevent or bypass a HPCI turbine trip signal, thereby leading to an unplanned period of unavailability of the HPCI system. PSEG's corrective actions included providing communications to all supervisors on adequate technical rigor when preparing for troubleshooting and revising a reference document used for the work instructions.

  The issue was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the inadequate work instructions resulted in unplanned unavailability of the HPCI system. The finding was of very low safety significance (Green) based on a Phase 2 analysis. The finding had a cross-cutting aspect in the area of human performance because PSEG did not appropriately plan work activities by incorporating the need for compensatory actions. Specifically, PSEG's work instructions did not incorporate the need for compensatory actions to preclude a HPCI turbine trip. (H.3(a)) (Section 4OA3.1)

Licensee-Identified Violations

- A violation of very low safety significance, which was identified by PSEG, has been reviewed by the inspectors. Corrective actions taken or planned by PSEG have been entered into PSEG's corrective action program. This violation and the corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The Hope Creek Generating Station began the inspection period at approximately 97 percent power, in end-of-cycle coastdown. On April 10, the unit was taken offline for refueling outage RF15. On May 2, the reactor was taken critical following the refueling outage, and the unit achieved 100 percent power on May 7. On May 17, the unit automatically scrammed due to low reactor vessel water level as result of multiple control rods drifting into the core. Operators re-started the unit on May 18, and restored the unit to 100 percent power on May 20. The unit remained at or near 100 percent power for the rest of the period, with the exception of planned power reductions for testing or maintenance.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01 - 2 samples)

.1 Onset of Seasonal Extreme Weather

a. Inspection Scope

The inspectors completed one seasonal weather preparation sample for the onset of hot summer weather. The inspectors performed a review of PSEG's seasonal readiness procedures and reviews associated with hot weather conditions. System health reports were reviewed and systems that could be subject to increased heat conditions were walked down to assess reliability and availability during periods of extreme heat. The inspectors focused on the readiness of the station service water system, emergency diesel generators and safety auxiliary cooling system. This inspection sample satisfied the inspection requirement to review two to four risk-significant systems prior to the onset of hot weather. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Review of Offsite and Alternate AC Power System Readiness

a. Inspection Scope

The inspectors completed one inspection sample to evaluate the readiness of PSEG's offsite and alternate AC power systems for adverse weather. Inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate AC power systems during adverse weather are appropriate. The inspectors reviewed station procedures affecting these areas and communications protocols with the transmission system operator to verify that the appropriate information is exchanged when issues arise that could impact the offsite power system. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 4 samples)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdown inspection samples for the four systems listed below to verify the operability of redundant or diverse trains and components when safety equipment was unavailable. The inspectors completed walkdowns to determine whether there were discrepancies in the system's alignment that could impact the function of the system, and therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, walked down system components, and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The inspectors also verified that PSEG had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program. Documents reviewed are listed in the Attachment.

- A and C emergency diesel generators (EDGs) during preventive and corrective maintenance on B EDG on April 14, 2009
- B 4kV vital bus during preventive maintenance on A 4kV bus on April 24, 2009
- A and C EDGs while B and D EDGs were out of service for unplanned maintenance on May 12, 2009
- A service water system during planned maintenance on C service water system on June 1, 2009

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 6 samples)

.1 Fire Protection – Tours

a. Inspection Scope

The inspectors completed six quarterly fire protection inspection samples. The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with PSEG's administrative procedures; fire detection and suppression equipment was available for use; that passive fire barriers were maintained in good material condition; and that compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with PSEG's fire plan. The six areas toured are listed below with their associated pre-fire plan designator. Other documents reviewed are listed in the Attachment.

- FRH-II-412, A RHR pump room
- FRH-II-422, A RHR heat exchanger room
- FRH-II-413, B RHR pump room
- FRH-II-423, B RHR heat exchanger room
- FRH-II-415, drywell and torus compartment
- FRH-II-435, steam tunnel, reactor core isolation cooling and high pressure coolant injection pipe chases

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors completed one flood protection measure inspection sample. The inspectors reviewed selected risk-important plant design features and PSEG procedures intended to protect the plant and its safety-related equipment from internal flooding events. Specifically, the inspectors focused on internal flood mitigation features for the 102' elevation of the reactor building that contains significant portions of the safety auxiliary cooling system. The inspectors reviewed flood analysis and design documents, including the Updated Final Safety Analysis Report (UFSAR), engineering calculations, and abnormal operating procedures. The inspectors observed the condition of wall penetrations, watertight doors, flood alarm switches, and drains to assess their readiness to contain flow from an internal flood in accordance with the design basis. Other documents reviewed are listed in the Attachment.

b. Findings

Introduction: The inspectors identified a Green non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because a safety auxiliary cooling system (SACS) water-tight door was blocked open without necessary compensatory measures, as a result of inadequate work instructions. Consequently, flood protection measures for the SACS system were degraded, which affected the capability of both SACS trains to perform their safety function during a flooding event.

Description: On April 27, 2009, during refueling outage RF15, the inspectors observed SACS water-tight door 4309A blocked open and questioned whether proper compensatory measures for flood protection were in place. Following discussions with plant personnel, the inspectors determined that the required compensatory measures were to have sand bags staged 1.5 feet high, or to have an individual standing by the door. Neither of these compensatory measures was in place.

The SACS consists of two independent closed loop cooling water systems designed to remove heat from safety-related equipment such as the residual heat removal system (RHR). In plant shutdown operations, the RHR system removes decay heat from the reactor via the RHR heat exchangers, which are cooled by SACS. Water-tight door 4309A functions as both a fire door and an internal flood barrier between the two SACS trains.

The inspectors reviewed the activities leading to PSEG's failure to implement the required compensatory measures for blocking open the water-tight door. In early April, engineering personnel performed an evaluation of the station's internal flooding analysis in accordance with procedure CC-AA-201, "Plant Barrier Control Program," and determined that sandbags 1.5 feet high or an individual standing by the door were the necessary compensatory measures. On April 22, 2009, maintenance technicians blocked open the door to allow temporary power cables to pass through the door during the refueling outage. However, due to a work planning error, the work instructions used by the technicians for blocking open the water-tight door did not incorporate the required compensatory measures.

Additionally, the inspectors determined that PSEG personnel did not reference procedure CC-AA-201 when blocking open the door. Procedure CC-AA-201 specifies that when a barrier is impaired, such as a water-tight door, the work group is responsible to arrange and implement required compensatory actions. The inspectors noted that the procedure provided an additional barrier to prevent this occurrence, because it would have prompted the work group to question what compensatory actions were needed.

Following the identification of this issue, PSEG closed the water-tight door and entered this issue into the corrective action program in notification 20412177.

Analysis: The performance deficiency was PSEG's failure to implement required compensatory measures when blocking open SACS water-tight door 4309A, due to inadequate work instructions. The finding was more than minor because it is associated with the external factors attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the capability and reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, not having the required flooding compensatory measures in place when the water-tight door 4309A was open affected the reliability and capability of the SACS system to function during a postulated internal flooding event. The inspectors used Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," to determine the significance of the finding. Based upon the finding not involving a loss of control or thermal margin, this finding does not require a quantitative assessment and screens as having very low risk significance (Green).

This finding has a cross-cutting aspect in the area of human performance because PSEG did not define and effectively communicate expectations regarding procedural compliance, and PSEG personnel did not follow procedures. Specifically, PSEG did not follow or refer to PSEG procedure CC-AA-201, "Plant Barrier Control Program," to impair the water-tight door. As a result, PSEG did not implement the compensatory measures required to mitigate a postulated internal flooding event. (H.4(b))

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances. Contrary to the above, on April 22, 2009, PSEG did not establish adequate work instructions for impairing the flood protection function of SACS water-tight door 4309A. As a result, PSEG did not have adequate measures to mitigate a postulated internal flooding event between April 22 and April 27, 2009. Operations personnel closed the water-tight door 4309A on April 27, 2009. Because this finding was of very low safety significance and was entered into the corrective action program in notification 20412177, this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000354/2009003-01, Inadequate Work Instructions for Impairing the Flood Protection Function of the Safety Auxiliary Cooling System Water-Tight Door)**

1R07 Heat Sink Performance (71111.07 - 1 sample)

a. Inspection Scope

The inspectors selected the B1 and B2 SACS heat exchangers for review. The inspectors verified that biofouling programs existed and were managed in accordance with PSEG procedures and commitments to Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” and that heat exchanger performance data demonstrated satisfactory performance. The inspectors walked down the B1 and B2 SACS heat exchangers, while they were open for inspection, to identify any potential fouling or degraded conditions. The inspectors also reviewed notifications in the corrective action program to verify that PSEG was identifying SACS heat exchanger problems at the appropriate threshold and that corrective actions addressed the identified problem and were effective. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection (71111.08 - 1 sample)

a. Inspection Scope

A sample of nondestructive evaluation (NDE) activities was inspected during refueling outage RF15, which included a review of ultrasonic testing (UT) analysis of data results using both manual UT techniques and the General Electric (GE) Smart computer-based phased array UT system. This included dissimilar metal nozzle to safe end welds on three recirculation inlet nozzles, RPV1-N2CSE, RPV1-N2GSE, and RPV1-N2HSE; the reactor vessel head to flange weld, RPV1-W20; and the 28” diameter pipe to elbow weld 1-BB-28VCA-012-2. The inspectors also observed the calibration technique used for the manual phased array UT performed on the 20” diameter pipe to pipe weld 1-BC-20CCA-114-6. A sample of in-vessel visual inspection (IVVI) video records for jet pump components, core spray components, top guide beams and the steam dryer were reviewed. Test data for several ultrasonic and visually identified indications were assessed and confirmed to be evaluated by PSEG as part of the in-service inspection (ISI) process.

The inspectors walked down portions of the inside of the drywell and the torus with a PSEG visual examiner to confirm the acceptance of a sample of the visual examinations was in accordance with site procedures and ASME requirements. External portions of the containment boundary were also observed at the location of one of the 4” diameter drain lines from the air gap between the drywell steel and concrete to the torus floor.

The inspectors compared PSEG’s Dissimilar Metal Weld program with the Electric Power Research Institute (EPRI) Boiling Water Reactor Vessel and Internals Projects 75A, Technical Basis for Revisions to NRC Generic Letter 88-01 Inspection Schedules.

The inspectors verified that previously identified embedded indications were analyzed for acceptable condition and continued use. The inspectors interviewed UT examination personnel and reviewed the NDE qualifications, including EPRI Performance Demonstration Initiative certifications, for the technicians responsible for the data collection, review and interpretation of the inspection results.

The inspection included a discussion with the Flow Accelerated Corrosion and Buried Pipe Program Engineer. The inspectors concluded that tools, including operating experience are in place to adequately characterize and inspect susceptible components.

The extent of oversight of ISI/NDE activities, including the topics of current ISI oversight and assessments and audits were reviewed. The inspectors reviewed a sample of notifications shown in the Attachment to confirm that identified problems were being documented for evaluation and proper resolution.

The inspectors reviewed the inspection plan for the top guide beams, consistent with Section 5.0, Recommended Areas for Inspection, of the Safety Evaluation for License Amendment 174 (Extended Power Uprate).

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11 - 1 sample)

.1 Requalification Activities Review By Resident Staff

a. Inspection Scope

The inspectors observed a licensed operator annual requalification simulator scenario on June 2, 2009, to assess operator performance and training effectiveness. The scenario involved a seismic event, a high radiation reading at the independent spent fuel storage installation, and event classification. The inspectors verified that control room staff correctly identified and declared emergency action levels in a timely manner. The inspectors assessed simulator fidelity and observed the simulator instructor's critique of operator performance. The inspectors also observed control room activities with emphasis on simulator-identified areas for improvement. Finally, the inspectors reviewed applicable documents associated with licensed operator requalification as listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 2 samples)

a. Inspection Scope

The inspectors completed two maintenance effectiveness inspection samples. The inspectors evaluated items such as: appropriate work practices; identifying and addressing common cause failures; scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); characterizing reliability issues for performance; trending key parameters for condition monitoring; charging unavailability for performance; classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); and appropriateness of performance criteria for structures, systems, and components (SSCs) functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as (a)(1). Documents reviewed are listed in the Attachment.

- Feedwater start-up level control valve software problem
- Safety auxiliary cooling system (SACS) / turbine auxiliary cooling system

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 4 samples)

a. Inspection Scope

The inspectors completed four maintenance risk assessment and emergent work control inspection samples. The inspectors reviewed on-line risk management evaluations through direct observation and document reviews for the following four configurations:

- A EDG and Salem Unit 3 out of service on May 26, 2009;
- B EDG, B service water (SW) system, and D EDG out of service on May 12, 2009;
- D SW system with 5023 offsite power line out of service on May 21, 2009; and
- E filtration recirculation ventilation system and D SW out of service on June 24, 2009.

The inspectors reviewed the applicable risk evaluations, work schedules and control room logs for these configurations to verify that concurrent planned and emergent maintenance and test activities did not adversely affect the plant risk already incurred with these configurations. PSEG's risk management actions were reviewed during shift turnover meetings, control room tours, and plant walkdowns. The inspectors also used PSEG's on-line risk monitor (Equipment Out-Of-Service workstation) to gain insights into the risk associated with these plant configurations. Finally, the inspectors reviewed notifications documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors completed five operability evaluation inspection samples. The inspectors reviewed the operability determinations for degraded or non-conforming conditions associated with:

- C EDG lube oil strainer cover leak;
- Safety relief valves A, C, F, G, K, and L lift setpoint drift;
- B and D EDG high voltage conditions;
- A SACS pump motor coupling degraded condition; and
- A EDG SACS cooling water valve failure to close.

The inspectors reviewed the technical adequacy of the operability determinations to ensure the conclusions were justified. The inspectors also walked down accessible equipment to corroborate the adequacy of PSEG's operability determinations. Additionally, the inspectors reviewed other PSEG identified safety-related equipment deficiencies during this report period and assessed the adequacy of their operability screenings. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18 - 2 samples)

.1 Temporary Modification

a. Inspection Scope

The inspectors completed a review of one temporary plant modification package that eliminated the rectifier selector switch from the B emergency diesel generator voltage regulator. This modification was performed after the B EDG experienced a voltage spike of 4900 volts for 48 seconds. High contact resistance in the internal switches of the rectifier selector switch was determined to be the probable cause of the occurrence. The inspectors verified that the design bases, licensing bases, and performance capability of the EDG was not degraded by the modification. The inspectors verified the new configuration was accurately reflected in the design documentation, and the post-modification testing was adequate to ensure the structures, systems, and components would function properly. The 10 CFR 50.59 evaluation associated with this temporary modification was also reviewed. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Permanent Modification

a. Inspection Scope

The inspectors completed a review of one permanent plant modification package for the replacement of the B residual heat removal pump motor. This review verified that the design bases, licensing bases, and performance capability of the system was not degraded by the modification. The inspectors verified the new configuration was accurately reflected in the design documentation, and the post-modification testing was adequate to ensure the structures, systems, and components would function properly. The inspectors interviewed plant staff, and reviewed issues that had been entered into the corrective action program to determine whether PSEG had been effective in identifying and resolving problems associated with plant modifications. The 10 CFR 50.59 evaluation associated with this modification was also reviewed. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 6 samples)

a. Inspection Scope

The inspectors completed six post-maintenance testing inspection samples. The inspectors reviewed the post-maintenance tests for the maintenance listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed test procedures to verify the procedure adequately tested the safety functions that may have been affected by the maintenance activity and the acceptance criteria in the procedure were consistent with the UFSAR and other design documentation. The inspectors witnessed the test or reviewed the test data to verify test results adequately demonstrated restoration of the affected safety functions. The inspectors verified that the post-maintenance tests conducted were adequate for the scope of the maintenance performed. Documents reviewed are listed in the Attachment.

- B EDG preventive and corrective maintenance
- Main steam isolation valve F028A actuator replacement
- High pressure coolant injection pump seal replacement
- Reactor core isolation cooling system preventive maintenance
- Safety relief valve K and M accumulator check valve replacements
- B EDG following removal of rectifier selector switch

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

PSEG shut down Hope Creek on April 10, 2009, to begin its fifteenth refueling outage (RF15). The inspectors reviewed the schedule and risk assessment documents associated with the Hope Creek RF15 refueling outage to verify that PSEG appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing an outage plan that maintained a defense-in-depth strategy. Prior to the refueling outage the inspectors reviewed PSEG's outage risk assessment to identify risk significant equipment configurations and to determine whether planned risk management actions were adequate. The inspectors verified that technical specification cooldown restrictions were adhered to by observing portions of the reactor shutdown and plant cooldown evolutions from the control room. The inspectors walked-down the drywell following the reactor shutdown to identify possible sources of unidentified leakage and observe general equipment condition. The inspectors monitored PSEG's control of the additional outage activities listed below. Documents reviewed for these activities are listed in the Attachment.

The inspectors verified that PSEG managed the outage risk in accordance with their outage plan. Refueling floor activities were observed periodically to verify whether refueling gates and seals were properly installed and determine whether foreign material exclusion boundaries were established around the reactor cavity. The inspectors observed portions of new nuclear fuel receipt, inspection, and placement into new fuel racks. Core offload, reload, and shuffle activities were periodically observed from the control room and refueling bridge to verify that operators controlled fuel movements in accordance with station procedures.

The inspectors confirmed, on a sampling basis, that equipment clearance tags were hung or removed properly and that associated equipment was appropriately configured to support the function of the work activity. Equipment work areas were periodically observed to determine whether foreign material exclusion boundaries were adequate. During control room walkdowns and observations of plant evolutions, the inspectors verified that the instrumentation to measure reactor vessel level and temperature were within the expected range for the operating mode and that they were configured correctly to provide accurate indication. The inspectors periodically verified throughout the outage that electrical power sources were maintained in accordance with technical specification (TS) requirements and consistent with the outage risk assessment. Walkdowns of control room panels, onsite electrical buses, and EDGs were conducted during risk significant electrical configurations to confirm the equipment alignments met requirements.

Risk significant plant evolutions were observed on a sampling basis during the outage, including reactor cavity flood up and drain down, installation and removal of main steam line plugs, installation and removal of the fuel pool gates, and residual heat removal system transition to shutdown cooling mode of operation to verify adherence to station procedures and outage risk management plans.

The inspectors verified through daily plant status activities that the decay heat removal safety function was maintained with appropriate redundancy as required by TS and consistent with PSEG's outage risk assessment. Contingency plans, procedures and staged equipment for a potential loss of decay heat removal were reviewed and compared to actual plant conditions to verify the effectiveness of mitigation strategies. During core offload conditions, the inspectors periodically determined whether the fuel pool cooling system was performing in accordance with applicable TS requirements and consistent with PSEG's risk assessment for the refueling outage. Reactor vessel water inventory controls and contingency plans were reviewed by the inspectors to determine whether they met TS requirements and provided for adequate inventory control. Secondary containment status and procedure controls were reviewed by the inspectors to verify that TS requirements and procedure requirements were met for secondary containment. The inspectors walked down the containment drywell prior to reactor startup to verify no evidence of reactor coolant system (RCS) leakage and that debris was not left behind from outage work activities that could adversely impact suppression pool suction strainers. The inspectors verified on a sampling basis that technical specifications, license conditions, other requirements, and procedure prerequisites for mode changes were met prior to plant mode changes. The inspectors reviewed RCS leakage surveillance tests following plant startup to verify RCS integrity.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 7 samples)

a. Inspection Scope

The inspectors completed seven surveillance testing (ST) inspection samples. The inspectors witnessed performance of and/or reviewed test data for the risk-significant STs to assess whether the SSCs tested satisfied technical specification, UFSAR, and procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design documentation; that test instrumentation had current calibrations and the range and accuracy for the application; and that tests were performed, as written, with applicable prerequisites satisfied. Upon ST completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. Documents reviewed are listed in the Attachment.

- Main steam isolation valve stroke time test on 4/11/09
- EDG fuel oil transfer pump test on 4/11/09
- D EDG loss of power/loss of coolant accident test on 4/11/09
- A RHR heat exchanger test on 4/9/09
- Primary containment integrated leak rate test on 5/1/09
- B low pressure coolant injection response time in-service test on 4/19/09
- B SACS pump in-service test on 6/12/09

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors completed one drill evaluation inspection sample. The inspectors observed control room operator emergency plan response actions during a licensed operator requalification training scenario on June 2, 2009. The inspectors verified that emergency classification declarations and notifications were completed in accordance with 10 CFR 50.72, 10 CFR 50, Appendix E, and the Hope Creek emergency plan implementing procedures. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 8 samples)

a. Inspection Scope

The inspectors identified exposure significant work areas within radiation areas, high radiation areas (<1 R/hr), or airborne radioactivity areas in the plant and reviewed associated PSEG controls and surveys of these areas to determine if controls (e.g., surveys, postings, barricades) were acceptable.

With a survey instrument, the inspectors walked down these areas or their perimeters to determine: whether prescribed radiation work permits, procedure, and engineering controls were in place, whether PSEG surveys and postings were complete and accurate, and whether air samplers were properly located.

The inspectors reviewed radiation work permits used to access these and other high radiation areas and identify what work control instructions or control barriers had been specified. The inspectors used plant-specific technical specification high radiation area requirements as the standard for the necessary barriers. The inspectors reviewed electronic personal dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. The inspectors verified that workers know what actions were required when their electronic personal dosimeter noticeably malfunctions or alarms.

Based on PSEG's schedule of work activities, the inspectors selected three jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation (drywell ISI, drywell safety relief valve maintenance, and IVVI). The inspectors observed work that was estimated to result in the highest collective doses, involved diving activities in or around spent fuel or highly activated material, or that involved potentially changing (deteriorating) radiological conditions. The inspectors reviewed all radiological job requirements (radiation work permit requirements and work procedure requirements). The inspectors observed job performance with respect to these requirements. The inspectors determined that radiological conditions in the work area were adequately communicated to workers through briefings and postings.

During job performance observations, the inspectors verified the adequacy of radiological controls, such as: required surveys (including system breach radiation, contamination, and airborne surveys), radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls.

For high radiation work areas with significant dose rate gradients (factor of 5 or more), the inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel.

During job performance observations, the inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors determined that they were aware of the significant radiological conditions in their workplace, and the radiation work permit controls/limits in place, and that their performance took into consideration the level of radiological hazards present.

During job performance observations, the inspectors observed radiation protection technician performance with respect to all radiation protection work requirements. The inspectors determined that they were aware of the radiological conditions in their workplace and the radiation work permit controls/limits, and if their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors evaluated PSEG performance against the requirements contained in 10 CFR 20.1601, Plant Technical Specifications 6.12, and Updated Final Safety Analysis Report (UFSAR) Chapter 12.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 7 samples)

a. Inspection Scope

The inspectors reviewed the As Low As Is Reasonably Achievable (ALARA) work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined that PSEG had established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that were ALARA. The inspectors determined that PSEG had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in PSEG's ALARA planning for these work activities.

Based on scheduled work activities and associated exposure estimates, the inspectors selected work activities in radiation areas, airborne radioactivity areas, or high radiation areas for observation (see Section 2OS1 above). The inspectors concentrated on work activities that present the greatest radiological risk to workers. The inspectors evaluated PSEG's use of ALARA controls for these work activities by evaluating PSEG's use of engineering controls to achieve dose reductions.

The inspectors evaluated PSEG performance against the requirements contained in 10 CFR 20.1101 and UFSAR Section 12.1.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (771121.03 - 1 sample)

a. Inspection Scope

The inspectors verified the calibration expiration and source response check currency on radiation detection instruments staged for use. The inspectors observed radiation protection technicians for appropriate instrument selection and self-verification of instruments operability prior to use.

The inspectors evaluated PSEG performance against the requirements contained in 10 CFR 20.1501, 10 CFR 20.1703 and 10 CFR 20.1704.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 2 samples)

a. Inspection Scope

The inspectors reviewed PSEG's program for gathering, evaluating and reporting information for the performance indicators (PIs) listed below. The inspectors used the definitions and guidance contained in NEI (Nuclear Energy Institute) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, to assess the accuracy of PSEG's collection and reporting of PI data. The documents reviewed by the inspectors are listed in the Attachment.

Cornerstone: Barrier Integrity

- Reactor Coolant System Leakage
- Reactor Coolant System Activity

The inspectors reviewed the data reported for these PIs for the period April 1, 2008, through March 31, 2009. The records reviewed included PI data summary reports, licensee event reports, daily logs, and operator narrative logs.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 2 samples)

.1 Review of Items Entered into the Corrective Action Program

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into PSEG's corrective action program. This was accomplished by reviewing the description of each new notification and attending management review committee meetings.

.2 Semi-Annual Review to Identify Trends: Human Performance - Procedure Use and Adherence

a. Inspection Scope

The inspectors performed a semi-annual review of notifications in PSEG's corrective action program to identify trends that may indicate a more significant safety issue. The inspectors also examined other sources of information, such as entries for PSEG's

Human Performance Fundamentals Management System, NRC inspection reports, PSEG self assessments, and documented observations by station personnel. Additionally, the inspectors discussed issues with plant staff and management. The inspectors' review covered the six-month period from January through June 2009.

The inspectors reviewed trends in human performance issues, with a focus on those related to procedure use and adherence. Documents reviewed are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified.

The inspectors identified a potential adverse trend in consequential issues related to procedure use and adherence. During the six-month period from January to June 2009, there were four NRC findings with cross-cutting aspects in procedure use and adherence. Two of these involved physical security activities and were documented in NRC inspection report 05000354/2009402. The third finding (05000354/2009006-03) was identified for personnel not following an emergency diesel generator test procedure with respect to writing corrective action notifications. The fourth finding was associated with a SACS water-tight door and was documented in Section 1R06 of this inspection report.

PSEG completed a common cause evaluation for the identification of four findings in the last 12 months with cross-cutting aspects in procedure use and adherence. This evaluation indicated that the common cause for these issues was ineffective supervisory actions to correct behaviors and hold people accountable. PSEG's corrective action for this common cause was the development of a change management plan to reinforce standards for procedure adherence. Additionally, PSEG implemented some interim measures to focus personnel on procedure adherence, such as daily, post-usage reviews of procedures by supervisors and periodic "procedure in-hand" days to draw focus on procedure adherence, even when the procedure is not required to be in-hand.

The inspectors determined that PSEG appropriately identified the adverse trend in procedure adherence and placed this item in the corrective action program. The common cause evaluation examined many notifications in the corrective action program, including the four findings and numerous other lower-level issues in various areas related to procedure adherence, in order to reach a conclusion on the common cause.

The inspectors also noted that PSEG identified an adverse trend in human performance due to various causes during refueling outage RF15. In response, PSEG conducted stand-downs to highlight the gaps in performance and reinforce the use of appropriate human performance tools, including proper use of procedures. The issues of minor significance contributing to this adverse trend were: 1) valve maintenance that caused the inadvertent isolation of service water discharge flow paths, 2) the cycling of a breaker that led to the unexpected start of an emergency diesel generator, and 3) concurrent maintenance and testing on turbine valve control logic that caused a half-scram. These issues were attributed to problems in work planning and coordination and to less than adequate verification practices.

The inspectors concluded that PSEG appropriately identified adverse trends in human performance and procedure adherence, and station management took action to identify the causes and develop corrective actions.

.3 Annual Sample: Operator Workarounds

a. Inspection Scope

The inspectors performed a cumulative review of PSEG's identified operator workaround conditions. The inspectors reviewed PSEG's list of operator burdens and concerns, temporary modifications, and operability determinations to assess the potential for these issues to impact the operators' ability to properly respond to plant transients or postulated accident conditions. In addition, the inspectors reviewed PSEG's list of deficient control room computer points and locked-in overhead annunciators to determine whether operators could adequately identify degraded plant equipment. The inspectors also reviewed operator logs and control room instrument panels to evaluate potential impacts on operator ability to implement abnormal and emergency operating procedures. Finally, the inspectors toured the plant and control room to identify potential workaround conditions not previously identified by PSEG. Documents reviewed for this inspection activity are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified.

The inspectors determined that PSEG appropriately identified the issues and entered them into the corrective action program. Operations personnel reviewed the cumulative impact of operator burdens, concerns, and workarounds on a periodic basis.

.4 Annual Sample: Corrective Actions for Refueling Outage Reactor Vessel Level Control Issues

a. Inspection Scope

NRC inspection report 05000354/2007005 documented three findings for reactor vessel level control issues during refueling outage RF14 in the fall of 2007. The findings included reactor vessel inventory losses due to safety relief valve testing and open steam line drain valves, and a reactor vessel level control problem during digital feedwater control system testing.

The inspectors reviewed PSEG's causal evaluations and corrective actions for these issues. Additionally, the inspectors considered PSEG's performance during refueling outage RF15 in April/May 2009, as a means to evaluate the effectiveness of the corrective actions. The inspectors discussed the actions with plant personnel and reviewed associated corrective action notifications. Documents reviewed are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified.

The inspectors noted that PSEG's causal evaluations were thorough and were completed in a timely manner. PSEG assigned appropriate corrective actions for the identified causal factors. The key corrective actions included improvements in the work control and tagging areas and revisions to station administrative procedures.

The inspectors observed that there were no similar reactor vessel inventory or level control problems during RF15. The inspectors concluded that PSEG's corrective actions for these findings were adequate.

4OA3 Event Followup (71153 - 2 samples)

.1 (Closed) LER 05000354/2008-003, HPCI Inoperability Due to Instrument Failure Initiated Turbine Trip

a. Inspection Scope

The inspectors reviewed the circumstances related to an unintended period of unavailability of the HPCI system in October 2008. The inspectors discussed the issue with station personnel and reviewed the LER and supporting documentation. Documents reviewed are listed in the Attachment. This LER is closed.

b. Findings

Introduction: A Green self-revealing, non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified because technicians did not have adequate work instructions for troubleshooting a HPCI system instrumentation drawer. The instructions did not include appropriate steps to prevent or bypass a HPCI turbine trip signal, thereby leading to an unplanned period of unavailability for the HPCI system.

Description: On October 5, 2008, technicians performed troubleshooting of a HPCI steam leak detection instrumentation drawer using a Troubleshooting Log prepared in accordance with MA-AA-716-004, "Conduct of Troubleshooting." The technicians down-powered and then re-energized the drawer, which caused a HPCI isolation signal and an unexpected HPCI turbine trip signal.

The Troubleshooting Log for this activity included instructions to prevent an isolation of the system by opening the breaker to the HPCI isolation valve, thereby maintaining the valve open. However, the Troubleshooting Log did not include instructions to prevent a HPCI turbine trip signal. The Troubleshooting Log should have included procedure steps to preclude a turbine trip signal from affecting the HPCI system, such as ensuring that a NORMAL/BYPASS switch on the drawer was in the BYPASS position.

PSEG performed an apparent cause evaluation of the event and identified two causal factors: an improper mindset on the part of the technicians performing the activity and incomplete planning/documentation. PSEG noted that the technicians were knowledgeable of the protection functions provided by the drawer, but did not fully consider or question the possibility that a turbine trip signal may be generated. PSEG also determined the Troubleshooting Log should have included steps to prevent a turbine trip, but it did not. Additionally, the evaluation identified that a key reference for this Troubleshooting Log, PSEG channel calibration procedure HC.IC-CC.SK-0004,

Exhibit 1, “Operations Information Sheet,” was incomplete in that it did not list the HPCI turbine trip as one of the functions of the drawer. PSEG’s corrective actions included providing communications to all supervisors on adequate rigor and questioning attitude when preparing for troubleshooting and adding the HPCI turbine trip signal to the Operations Information Sheet.

The inspectors reviewed the event and PSEG’s apparent cause evaluation and concluded that PSEG’s evaluation was adequate. The inspectors also noted that PSEG’s procedure for developing a Troubleshooting Log, MA-AA-716-004, “Conduct of Troubleshooting,” contains specific guidance to include actions to limit the impact on the plant and prevent creating an undesired or unanalyzed equipment condition, such as “placing a component in bypass.” The inspectors discussed this guidance with the maintenance supervisor who was involved in the troubleshooting activities. He stated that the guidance was followed, but he did not rigorously challenge the work planning documentation with respect to placing the channel in bypass.

Analysis: The inspectors determined the work instructions in the Troubleshooting Log for this activity did not meet the standard established by PSEG procedure MA-AA-716-004 and was a performance deficiency. The issue was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone, and it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the inadequate work instructions in the Troubleshooting Log resulted in unplanned unavailability of the HPCI system. The inspectors performed a Phase I screening of the finding in accordance with Inspection Manual Chapter (IMC) 0609, Attachment 0609.04, Table 4a, Mitigating Systems Cornerstone column. The inspectors concluded, consistent with LER 05000354/2008-003, that there was a loss of safety function of the HPCI system for 29 minutes. Therefore, the inspectors used IMC 0609, Appendix A, and performed a Phase 2 analysis using the Hope Creek plant-specific pre-solved table. The inspectors accessed the pre-solved table from the NRC internal web-page, as described in IMC 0609, Appendix A, and used an exposure period of < 3 days. The dominant sequence leading to core damage was a transient with a loss of the power conversion system followed by a failure of high pressure injection and a failure to depressurize the reactor coolant system. The remaining mitigating capability while the HPCI system was unavailable included the feedwater and reactor core isolation cooling systems, and the ability to depressurize the reactor coolant system using safety relief valves. The Phase 2 analysis determined that the finding was of very low safety significance (Green).

The finding had a cross-cutting aspect in the area of human performance because PSEG did not appropriately plan work activities by incorporating the need for compensatory actions. Specifically, PSEG’s work instructions did not incorporate the need for compensatory actions to preclude a HPCI turbine trip. (H.3(a))

Enforcement: 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings,” requires, in part, that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances. Contrary to the above, on October 5, 2008, PSEG did not adequately establish documented instructions appropriate to the circumstances of troubleshooting a safety-related HPCI steam test detection instrumentation drawer. Specifically, the Troubleshooting Log for this activity did not include appropriate written instructions to prevent a HPCI turbine trip signal from causing the unintended unavailability of the HPCI system for a 29-minute period.

Because this finding was of very low safety significance and was entered into the corrective action program in notification 20385744, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000354/2009003-002, Unplanned HPCI Unavailability Due to Troubleshooting)**

.2 Automatic Reactor Scram Due to Scram Air Header Leak

a. Inspection Scope

On May 17, 2009, at 3:35 am, operators observed indications of control rods drifting into the reactor core. Operators placed the mode switch in shutdown in accordance with station procedures. However, reactor vessel water level had already dropped to below the low-level scram setpoint, causing an automatic reactor scram.

The inspectors responded to the site and verified that plant systems performed as designed following the transient and that operator response was consistent with plant procedures. The inspectors reviewed control board indications, plant logs, computer alarm data, and other post-transient records and data. The inspectors also reviewed PSEG's prompt investigation, technical evaluations, and root cause evaluation for this event.

b. Findings

Introduction: A Green finding was self-revealed because PSEG identified an air leak at a soldered joint on the scram air header in September 2008, but did not enter the degraded condition in the corrective action program. As a result, PSEG did not evaluate the leak or take corrective actions prior to the joint separating, causing an automatic reactor scram.

Description: On May 17, 2009, a soldered copper joint on the scram air header at hydraulic control unit (HCU) 22-11 separated, causing the scram header to depressurize and as a result, multiple control rods began drifting into the core. The reactor automatically scrammed on low reactor vessel water level due to the resulting shrink in level.

The scram air header, which is supplied by the instrument air system, maintains pressure on the scram pilot valves for each HCU. If the scram air header is depressurized, the scram pilot valves will re-position, allowing the scram inlet and outlet valves to open and the control rods to be inserted into the core.

Following the event, PSEG repaired the affected joint and performed an extent-of-condition inspection of the corresponding joints on all other HCUs. No other leaks on soldered joints were identified. PSEG repaired several other minor leaks that were identified during this inspection.

In September 2008, technicians identified a significant air leak on the scram air header at HCU 22-11; however, this condition was not placed in the corrective action program. Specifically, on September 23, 2008, vendor technicians performed a preventive maintenance work order to identify air leaks on the scram air header. This activity identified a total of 39 air leaks, which were documented in the work order completion remarks. The leak at HCU 22-11 was documented as a “large leak” and was separated from others in the work order list, with the intent of highlighting the significance. The work order information was sent to engineering with the expectation that the system manager would evaluate the deficiencies and place them in the corrective action program (CAP), as necessary. The engineering department entered 38 of 39 leaks in the CAP, but missed the item for HCU 22-11.

PSEG's root cause evaluation of the event identified two root causes: the failure of the soldered joint due to incomplete soldering during original construction; and inconsistent expectations for vendor technicians, resulting in the leak not being correctly identified in the CAP. PSEG also determined that communication deficiencies between the technicians and the engineering system manager contributed to the event. The inspectors concluded that PSEG accurately identified the root and contributing causes for the event.

Analysis: PSEG did not enter identify an air leak at a soldered joint on the scram air header, a degraded condition, in the CAP. This oversight was a performance deficiency because it was contrary to PSEG procedure LS-AA-120, "Issue Identification and Screening Process," which specifies that problems and equipment deficiencies are to be entered in the CAP. This issue was more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions. Specifically, by not identifying the air leak in the corrective action program, PSEG did not evaluate the degraded condition and its impact on the reliability of the scram air header. Consequently, PSEG did not take corrective actions or perform repairs prior to the condition degrading further and causing an automatic reactor scram. To evaluate the significance of the finding, the inspectors performed a Phase I screening using IMC 0609, Attachment 0609.04, Table 4a, Initiating Events Cornerstone column, Transient Initiators. The inspectors determined that the finding increased the likelihood of a reactor scram, but did not contribute to the likelihood that mitigating equipment would not be available. Therefore, the finding screens as Green.

The finding had a cross-cutting aspect in the area of problem identification and resolution because the station did not identify the scram air header leak completely, accurately, and in a timely manner commensurate with its safety significance. (P.1(a))

Enforcement: The scram air header system is not covered by 10 CFR 50, Appendix B or Technical Specifications. Therefore, while a performance deficiency existed, no violation of regulatory requirements occurred. **(FIN 05000354/2009003-003, Automatic Reactor Scram Due to Leak on Scram Air Header)**

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with PSEG security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours. These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 Extended Power Uprate Closeout (IP 71004)

a. Inspection Scope

On May 14, 2008, the NRC approved Hope Creek License Amendment 174 for a 15-percent Extended Power Uprate (EPU) and issued the associated Safety Evaluation (ADAMS package ML081230648). The inspectors have observed and reviewed selected activities throughout the phased EPU implementation. The inspectors have determined, based on a sample review of these activities and comparison of records and tests with the current licensing documents, that PSEG's commitments have been met regarding the Hope Creek EPU and that PSEG has fully implemented the EPU within its approved implementation timeline.

The review of the top guide beams inspection plan, as discussed in Section 1R08 of this inspection report, represents the final activity in the 15-percent EPU inspection. A consolidated list of EPU-related inspection reports are listed in the Attachment. This completes the 15-percent EPU inspection.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The resident inspectors presented the inspection results to Mr. George Barnes and other members of PSEG staff on July 8, 2009. The inspectors asked PSEG whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by PSEG and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation:

Technical Specification 6.12, "High Radiation Area," requires, in part, that each entryway to a high radiation area be barricaded and conspicuously posted. Contrary to this, on April 19, 2009, two Health Physics technicians observed that a high radiation area swing gate barricade located at the entrance to the torus room (Azimuth 135) on the 77' elevation of the reactor building had been taped open, leaving the area unbarricaded. This issue is not greater than green because it involved a high radiation area, and no unauthorized personnel entered the area while the swing gate was left opened. This issue was documented in PSEG's corrective action program as notification 20410956.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Barnes, Site Vice President
B. Booth, Operations Director
R. Canziani, Maintenance Director
E. Casulli, Shift Operations Superintendent
K. Chambliss, Assistant Plant Manager
P. Duca, Senior Engineer, Regulatory Assurance
M. Gaffney, Regulatory Assurance Manager
K. Knaide, Engineering Director
W. Kopchick, Plant Engineering Manager
A. Oliveri, NDE Services Superintendent
J. Perry, Plant Manager
H. Trimble, Radiation Protection Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000354/2009003-01	NCV	Inadequate Work Instructions for Impairing the Flood Protection Function of the Safety Auxiliary Cooling System Water-Tight Door (Section 1R06)
05000354/2009003-02	NCV	Unplanned High Pressure Coolant Injection Unavailability Due to Troubleshooting (Section 4OA3.1)
05000354/2009003-03	FIN	Automatic Reactor Scram Due to Leak on Scram Air Header (Section 4OA3.2)

Closed

05000354/2008-003	LER	HPCI Inoperability Due to Instrument Failure Initiated Turbine Trip (Section 4OA3.1)

LIST OF DOCUMENTS REVIEWED

In addition to the documents identified in the body of this report, the inspectors reviewed the following documents and records:

Hope Creek Generating Station (HCGS) Updated Final Safety Analysis Report
Technical Specification Action Statement Log
HCGS Narrative Logs
HCGS Plant Status Reports
Hope Creek Operations Night Orders and Temporary Standing Orders

Section 1R01: Adverse Weather Protection

Procedures
HC.OP-AB.BOP-0004, Grid Disturbances, Revision 14
OP-SH-101-112-1002, Online Risk Assessment, Revision 3
OP-AA-108-107-1001, Electric System Emergency Operations and Electrical Systems Operator Interface, Revision 3
WC-AA-107, Seasonal Readiness, Revision 8
HC.OP-AB.COOL-0001, Station Service Water, Revision 17
HC.OP-AB.BOP-0004, Grid Disturbances, Revision 16
HC.OP-AB.COOL-0002, Safety/Turbine Auxiliaries Cooling System, Revision 4
HC.OP-AB.MISC-0001, Acts of Nature, Revision 13

Notifications
20408624

Orders
70086413 60080900

Other Documents
Hope Creek Generating Station Site Summer Readiness Memo, dated May 22, 2009
System Summer Readiness Challenge Documents

Section 1R04: Equipment Alignment

Procedures
HC.OP-SO.KJ-0001, Emergency Diesel Generators Operation, Revision 47
HC.OP-SO.EA-0001, Service Water System Operation, Revision 34
HC.OP-SO.BC-0001, Residual Heat Removal Operation, Revision 44

Drawings
M-51-1, Residual Heat Removal, Revision 38
M-10-1, Service Water System, Revision 52

Section 1R05: Fire Protection

Procedures
NC.FP-AP.ZZ-0005, Fire Protection Surveillance and Periodic Test Program, Revision 14
NC.FP-AP.ZZ-0025, Operational Fire Protection Program, Revision 7
OP-AA-201-009, Control of Transient Combustible Material, Revision 1

Notifications
20412930 20409937

Other Documents
FRH-II-412, RCIC Pump & Turbine Room, RHR Pump and Heat Exchanger Rooms & Electrical Equipment Room, Revision 3
FRH-II-422, RHR Heat Exchanger & MCC Area, Revision 3
FRH-II-413, HPCI Pump & Turbine Room, RHR Pump and Heat Exchanger Rooms, Revision 3
FRH-II-423, MCC Area, RHR Heat Exchanger Room, Safeguard Instrument Rooms & RACS Pumps & Heat Exchanger Area, Revision 4
FRH-II-415, Drywell Pad & Torus Area, Revision 4
FRH-II-435, Steam Tunnel, RCIC, HPCI, Pipe Chases, CRD Removal & Repair Area, Revision 4
FRH-II-532, Lower Control Equipment Room, Revision 6

Section 1R06: Flood Protection Measures

Procedures
OP-HC-103-102-1005, High Energy and Internal Flooding Barrier Control Program, Revision 0
CC-AA-201, Plant Barrier Control Program, Revision 0

Calculations
D7.5, Hope Creek Generating Station Environmental Design Criteria (Reactor Building Internal Flooding Depths), Revision 21

Drawings
A-0702-0, Door & Hardware Schedule Pressure-Tight Doors, Revision 17
A-0703-0, Door & Hardware Schedule Pressure-Tight Doors, Revision 10

Notifications (*NRC-identified)
20420281* 20412447* 20412177*

Orders
70097332 70097468 30158652

Other Documents
H-1-ZZ-FEE-1803, Separation Barrier Control Aid for Hope Creek, Revision 0
ND.DE-PS.ZZ-0010, Internal Hazards Program, Revision 1

Section 1R07: Heat Sink Performance

Procedures
ER-AA-340-1002, Service Water Heat Exchanger and Component Inspection Guide, Revision 3
HC.OP-AB.COOL-0002, Safety/Turbine Auxiliaries Cooling System, Revision 3
HC.OP-SO.EG-0001, Safety and Turbine Auxiliaries Cooling Water System Operation, Revision 39
ER-AA-340, GL 89-13 Program Implementing Procedure, Revision 3
HC.OP-FT.EA-0001(Q), Validating SSWS Flow Through SACS HXs, Revision 7
HC.SE-PR.EG-0001(Q), Safety and Auxiliary Cooling System Annual Biofouling Monitoring, Revision 5

Drawings
M-11-1 Sh. 1, Safety Auxiliaries Cooling Reactor Building, Revision 29
M-12-1 Sh. 1, Safety Auxiliaries Cooling Auxiliary Building, Revision 31

Orders
30138487 30138488

Other Documents
H-1-EG-MEE-1301, 100ºF SACS Design Temperature Limit Evaluation, Revision 2
Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment

Section 1R08: Inservice Inspection Activities

Drawings
C-0786-1, Rev 7, Reactor Building Drywell Foundation
C-0794-0, Rev 8, Reactor Building Drywell Shield Wall Air Gap Details, Sh. 1
C-0795-0, Rev 9, Reactor Building Drywell Shield Wall Air Gap Details, Sh. 2
C-0931-0, Rev 18, Containment Vessel Requirements Suppression Chamber Penetrations
C-0932-0, Rev 16, Containment Vessel Requirements Suppression Chamber-Internal Piping
C-0940-0, Rev 5, Primary Containment RPV Pedestal Floor El 86'-11"
FSAR, Figure 3.8-1, Primary Containment Elevation
P-8142-1, Rev 7, Plumbing & Drainage Reactor Building Plan at EL. 77'-0", Area 14
P-8162-1, Rev 8, Plumbing & Drainage Reactor Building Plan at EL. 77'-0", Area 16
P-8172-1, Rev 10, Plumbing & Drainage Reactor Building Plan at EL. 77'-0" Area 17
P-8182-1, Rev 8, Plumbing & Drainage Reactor Building Plan at EL. 77'-0" Area 18
P-8202-1, Rev 8, Plumbing & Drainage Reactor Building Plan at EL. 77'-0" Area 20
P-9000-0, Rev 11, Plumbing & Drainage
VTD PC155Q-0089, Rev 8, Pipe Support Modifications for Penetrations 212A&B Field Installations

Notifications
20348616 20410286 20381114 20271701 20410504 20410931
20411220 20411396 20411423

NDE Inspection Reports and Data Sheets
NOS Assessment Plan NOSPA-HC-09-1C, dated 12/31/2008. SAP 3 80094644
Assessment NOSPA-HC-09-1C (EN-1C09-040) on ISI Program readiness
Audit of ISI/NDE dated 4/15/2009 for orders 60052865, 30154681, and 30138487
Summary No. 105610 report for UT of pipe to elbow weld 1-BB-28VCA-012-2
Drywell thickness measurements at 2" above 86'-11" floor, per order 30154681
Summary No. 850200 report for VT of accessible internal torus surfaces
BWRVIP-75-A: BWR Vessel and Internals Project, 1012621, dated Oct 2005
BWRVIP-26-A: BWR Vessel and Internals Project, 1009946, dated 2004

Procedures
EPRI-DMW-PA-1, Rev 0, Procedure for Manual Phased Array Ultrasonic Examination of Dissimilar Metal Welds
GE-PDI-UT-2, Rev 4, PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds
GE-UT-247, Ver 1, Procedure for Phased Array Ultrasonic Examination of Dissimilar Metal Welds

GE-UT-300, Ver 10, Procedure for Manual Examination of Reactor Vessel Assembly Welds in accordance with PDI
GEH-VT-204, Ver 12, Procedure for Invessel Visual Inspection (IVVI) of BWR 4 RPV Internals
OU-AA-335-018, Detailed and General, VT-1 and VT-3 Visual Examination of ASME Class MC and CC Containment Surfaces and Components

Other Documents
ASME Section XI
ASME Section XI, Sub-Section IWE
NRC IN 2006-001, Torus Cracking, Order 000070054239
Paragraph 3.3.1, Page 47 of Power Uprate Conditions for EVT-1 of Top Guide Beams
Acceptance Memo per IWA 2240 for use of GE-UT-300 for RPV Flange Weld UT
NRC NUREG 0313, Rev 2, BWR Coolant Pressure Boundary Piping
Hope Creek RF15 UT examination scan plans for 1-BC-20CCA-114-6, 1-BB-28VCA-012-2 and RPV1-N2CSE
FP-08-046, Hope Creek Generating Station – Design Margin for Drywell Shell

Section 1R11: Licensed Operator Requalification Program

Procedures
HC.OP-AB.MISC-0001, Acts of Nature, Revision 13

Other Documents
Hope Creek Generating Station Emergency Classification Guide
Simulator Scenario Guide SG-656

Section 1R12: Maintenance Effectiveness

Procedures
ER-AA-301-1001, Maintenance Rule – Scoping, Revision 3
ER-AA-301-1003, Maintenance Rule – Performance Criteria Selection, Revision 3
ER-HC-301-1009, Maintenance Rule system Function and Risk Significant Guide, Revision 2

Notifications
20398278 20412833 20413430

Orders
60083496 70080064 70093613 70094991 70097677

Other Documents
Hope Creek Expert Panel Meeting Minutes, dated June 2, 2009
Hope Creek Maintenance Rule Status & Projections, dated June 4, 2009
Hope Creek Narrative Log, dated May 2, 2009

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
OP-AA-101-112-1002, On-Line Risk Assessment, Revision 2
WC-AA-101, On-Line Work Management Process, Revision 16

Other Documents
R15 Shutdown Risk Assessment, Revision 4

Section 1R15: Operability Evaluations

Procedures
HC.MD-PM.KJ-0005(Q), Standby (Emergency) Diesel Generator – Inspection, Revision 21
SH.OP-DL.ZZ-0027, Temporary Routine Sheet (1X Daily), Revision 5

Notifications

20411387	20415123	20414118	20414295	20414018	20414454
20403906	20414636	20411328	20417725	20411330	20411244
20409221					

Orders

80098699	60082832	60082950	80098805	60083385	60083290
30158799	30150385	30158471	30117013	70096933	70098323
60082759					

Other Documents
Technical Evaluation - Standby Diesel Generators B and D Overvoltage Impact
Prompt Investigation - D EDG Did Not Achieve Proper Voltage
Complex Troubleshooting Data Sheet - B EDG Overvoltage
Failure Analysis of 2 Rectifier Selector Switches for Hope Creek

Section 1R18: Plant Modifications

Procedures
CC-AA-112, Temporary Configuration Changes, Revision 11
CC-AA-112-1001, Temporary Configuration Change Implementation, Revision 1
LS-AA-104, Exelon 50.59 Review Process, Revision 5
LS-AA-104-1001, 50.59 Review Coversheet Form, Revision 2
LS-AA-104-1002, 50.59 Applicability Review Form, Revision 2
LS-AA-104-1003, 50.59 Screening Form, Revision 1
HC.OP-SO.KJ-0001, Emergency Diesel Generators Operation, Revision 48
HC.MD-PM.KJ-0005(Q), Standby (Emergency) Diesel Generator – Inspection, Revision 21

Drawings
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Orders
80098662

Other Documents
Temporary Configuration Change Package 09-014
50.59 Screening Form No. HC 09-072
Hope Creek Lesson Plan NOH04EDG000C-00, Emergency Diesel Generators

Section 1R19: Post-Maintenance Testing

Procedures

HC.IC-FT.SN-0009, ADS and Safety Relief Valve Operability Test, Revision 4
HC.RA-IS.ZZ-0011, Leakage Test of Safety/Relief Valve Accumulators, Revision 4
HC.OP-LR.ZZ-0003, Leakage Test of Safety/Relief Valve Accumulators, Revision 1
HC.MD-ST.KJ-0001, Diesel Generator Technical Specification Surveillance and Preventive Maintenance, Revision 39
HC.MD-CM.KJ-0015, Diesel Generator Speed/Load Control System Alignment, Revision 7
HC.OP-FT.KJ-0002, Emergency Diesel Generator 1BG400 – Functional Test, 4/21/2009
HC.OP-IS.BJ-0001, HPCI Main and Booster Pump Set – Inservice Test, Revision 51

Drawings
M-41-1, HCGS Nuclear Boiler, Revision 27

Notifications

20411900	20411901	20410708

Orders

50109515	70096779	60082832	60082344	70096402	30159100

Section 1R20: Refueling and Outage Activities

Drawings
M-51-1, HCGS Residual Heat Removal, Revision 37
I-P-BC-04, System Isometric, RHR Suction, Pumps A, B, C, & D, Revision 16

Procedures
HC.OP-LR.ZZ-0004, Drywell Temperature Survey, Revision 0
HC.OM-AP.ZZ-0001, Shutdown Safety Management Program – Hope Creek Annex, Revision 1
HC.OP-IO.ZZ-0005, Cold Shutdown to Refueling, Revision 29
HC.OP-AB.RPV-0009, Shutdown Cooling, Revision 6
HC.OP-SO.BC-0002, Decay Heat Removal Operation, Revision 22

Notifications (*NRC-identified)

20411450*	20413148*	20412342*	20410505*	20410826*	20408678*
20411753*	20409609*	20421543*	20409472	20411758	20409221
20409812	20409653	20411042	20410708	20410574	20412477
20393176	20411256	20409377	20413851	20413789	20409704
20409372	20409653	20410384	20411382		

Orders

70096492	60083111	70089575

Other Documents
R15 Shutdown Risk Assessment, Revision 4
2009-058, Plant Shutdown for RF15, 4/6/2009

Section 1R22: Surveillance Testing

Procedures
MA-AA-716-040, Control of Portable Measuring and Test Equipment Program, Revision 6
HC.OP-ST.KJ-0011, Diesel Fuel Oil Transfer Operability – 18 Months, Revision 4
ER-AA-380-1002, Integrated Leakage Rate Test Planning and Implementation Guide, Revision 0

HC.OP-IS.AB-0102, MSIV Loss of Power - Cold Shutdown – Inservice Test, Revision 13
HC.OP-LR.ZZ-0004, Primary Containment Integrated Leakage Rate Test, Revision 0

Completed Surveillances
HC.OP-ST.BC-0005, LPCI Subsystem B ECCS Time Response Functional Test – 18 Months, 4/19/2009
HC.OP-IS.AB-0102, MSIV Loss of Power - Cold Shutdown – Inservice Test, 4/11/2009
HC.OP-ST.KJ-0008, Integrated Emergency Diesel Generator 1DG400 Test – 18 Months, 4/11/2009
HC.OP-ST.BC-0009, Residual Heat Removal System RHR Heat Exchanger Flow Measurement - 18 Month, 4/9/2009
HC.OP-ST.KJ-0011, Diesel Fuel Oil Transfer Operability – 18 Months, 4/9/2009
HC.OP-LR.ZZ-0004, Primary Containment Integrated Leakage Rate Test, 5/1/2009
HC.OP-IS.EG-0002, B SACS Pump-BP210 – Inservice Test, 6/12/2009
HC.OP-IS.AB-0102, Main Steam System Valves - Cold Shutdown – Inservice Test, Revision 20, 4/11/2009

Drawings
M-30-1, HCGS Diesel Engine Auxiliary Systems Fuel Oil, Revision 26

Notifications (*NRC-identified)
20407300* 20414338* 20411618* 20412240 20412237 20412348
20408678*

Orders
50109570 30098621

Other Documents
HC-2009-01, Missed TS Surveillance Evaluation for Emergency Diesel Generator Fuel Oil Storage Pump (EDGFOTP) Cross-Tie Function, Revision 0

Section 1EP6: Drill Evaluation

Procedures
HC.OP-AB.MISC-0001, Acts of Nature, Revision 13

Other Documents
Hope Creek Generating Station Emergency Classification Guide
Simulator Scenario Guide SG-656

Section 4OA1: Performance Indicator Verification

Procedures
LS-AA-2001, Collection and Reporting of NRC Performance Indicator Data, Revision 10
LS-AA-2100, Monthly Data Elements for NRC Reactor Coolant System Leakage, Revisions 5, 6

Notifications
20418318

Other Documents
Daily Surveillance Log Data
Daily Dose Equivalent Iodine-131 Sample Data

Section 4OA2: Identification and Resolution of Problems

Procedures
OP-AA-102-103, Operator Work-Around Program, Revision 2
OP-AA-102-103-1001, Operator Burdens Program, Revision 0
HU-AA-104-101, Procedure Use and Adherence, Revision 3

Notifications

20343032	20344944	20342758	20413430	20413210	20420636

Orders

70076318	70076610	70076985	70097637

Other Documents
Hope Creek Narrative Log, dated June 8 through 15, 2009
Hope Creek Plan of the Day, dated June 16, 2009
Quarterly Operator Burden Assessment – 2008 – Fourth Quarter, dated February 3, 2009
Nuclear Communications Article, "Hope Creek to Hold Procedure Use Day," dated June 9, 2009
Nuclear Communications Article, "Hope Creek Resets the Station Event Free Clock for RF15 Human Performance Gaps," dated May 1, 2009

Section 4OA3: Event Followup

Procedures
LS-AA-120, Issue Identification and Screening Process, Revision 8
MA-AA-716-004, Conduct of Troubleshooting, Revisions 6 and 7

Notifications (*NRC-identified)

20411328	20415621	20415534	20385744	20419337*

Orders

70090304	70096933	70098002	30164922	70098100	60078979

Other Documents
Technical Specification Surveillance Tracking Log, dated 10/04/2008
Hope Creek Licensed/Non-Licensed Operator Training Lesson: Control Rod Drive Hydraulics, dated 1/15/08

Section 4OA5: Other Activities

Cross-Reference of Hope Creek Inspection Reports which contain EPU-related Inspection Activities

05000354/2006015
05000354/2007004
05000354/2007005
05000354/2008003
05000354/2008004
05000354/2008007
05000354/2009003

LIST OF ACRONYMS

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
EDGs	Emergency Diesel Generators
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
GE	General Electric
HCGS	Hope Creek Generating Station
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
ISI	In-service Inspection
LER	Licensee Event Report
MR	Maintenance Rule
NCV	Non-cited Violation
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
PIs	Performance Indicators
PSEG	Public Service Enterprise Group Nuclear LLC
RCS	Reactor Coolant System
RF15	Refueling Outage
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SACS	Safety Auxiliary Cooling Systems
SSCs	Structures, Systems, and Components
ST	Surveillance Testing
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing