July 21, 2006

Mr. William Levis
Senior Vice President and Chief Nuclear Officer
PSEG LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000354/2006003

Dear Mr. Levis:

On June 30, 2006, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Hope Creek Generating Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 6, 2006, 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.

This report documents one NRC-identified finding and two self-revealing findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these three 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Hope Creek Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) 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 Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mel Gray, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000354/2006003
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	050000354
License No:	NPF-57
Report No:	05000354/2006003
Licensee:	Public Service Enterprise Group Nuclear LLC
Facility:	Hope Creek Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	April 1, 2006 through June 30, 2006
Inspectors:	G. Malone, Senior Resident Inspector T. Wingfield, Resident Inspector F. Arner, Senior Reactor Engineer J. Furia, Senior Health Physicist E. Gray, Senior Reactor Engineer T. O'Hara, Reactor Engineer M. Patel, Reactor Engineer M. Snell, Reactor Engineer D. Tifft, Reactor Engineer
Approved By:	Mel Gray, Chief Projects Branch 3 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000354/2006003; 04/01/2006 - 06/30/2006; Hope Creek Generating Station; Maintenance Risk Assessments and Emergent Work Control, Refueling and Other Outage Activities, and Access Control to Radiologically Significant Areas.

The report covered a 3 month period of inspection by resident inspectors and announced inspections by a regional senior health physics inspector, two regional senior reactor inspectors and three regional reactor inspectors. Three Green non-cited violations (NCVs) 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 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

C Green. Inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," when the 'A' service water strainer was rendered unavailable on April 18, 2006. On November 25, 2004, the 'C' service water strainer backwash arm motor experienced elevated running current and multiple thermal overload trips. PSEG performed design change and corrective maintenance activities to increase the size of the thermal overloads for the 'C' strainer motor. This condition adverse to quality was not entered into PSEG's corrective action program (CAP) for evaluation and extent of condition review. On April 18, 2006, PSEG experienced elevated running current and multiple thermal overload trips on the 'A' strainer motor which resulted in unplanned unavailability. PSEG's corrective actions included corrective maintenance to increase the size of the thermal overloads on the 'A', 'B', and 'D' strainer motors and evaluations of the elevated motor currents and the CAP oversight issue.

This performance deficiency is more than minor because it is associated with the equipment performance attribute and affected the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operation Significance Determination Process," the inspectors conducted a Phase 1 SDP screening and determined that, since adequate mitigation capability was maintained and a quantitative assessment was not required, the finding was of very low safety significance (Green). The performance deficiency had a cross-cutting aspect in the area of problem identification and resolution because PSEG did not evaluate and implement corrective action for a condition adverse to quality. (Section 1R13)

C Green. A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified when the single source of shutdown reactor water level indication was rendered inaccurate during reactor vessel reassembly. PSEG’s refueling maintenance procedure directed the installation of blank flanges on all reactor vessel head penetrations during reactor disassembly. This resulted in the reactor being placed in an unvented condition when the head was reinstalled on the vessel which caused the shutdown reactor water level indication to be inaccurate and invalid. PSEG’s corrective actions included changes to the refueling maintenance procedures to install vented flanges and changes to the integrated operations procedures to ensure that the reactor is vented prior to changing vessel level in Operational Condition 4 or 5.

This performance deficiency is more than minor because it is associated with the equipment performance attribute and affected the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operation Significance Determination Process," the inspectors conducted a Phase 1 SDP screening and determined that, since adequate mitigation capability was maintained and a quantitative assessment was not required, the finding was of very low safety significance (Green). The performance deficiency had a cross-cutting aspect in the area of human performance because PSEG did not provide adequate procedure resources to prevent the loss of all shutdown range reactor water level indication. (Section 1R20)

Cornerstone: Occupational Radiation Safety

C Green. A self-revealing non-cited violation of 10 CFR 20.1501, “Surveys and Monitoring: General”, was identified when a worker’s electronic dosimeter alarmed due to dose rates in the ‘A’ steam jet air ejector (SJAE) room exceeding the preset alarm setpoint. During power ascension at the end of the refueling outage, the worker entered the ‘A’ SJAE room and received a dose rate alarm due to the presence of dose rates in excess of 100 millirem per hour measured 30 centimeters from the source of radiation although the rooms were not identified, posted or controlled as a high radiation area. Changing radiological conditions caused by changes in reactor power level and increased steam flow in the plant required that a new radiological survey of the ‘A’ SJAE room be conducted in accordance with 10 CFR 20.1501 to support compliance with 10 CFR 20.1201, “Occupational Dose Limits for Adults,” and plant technical specification 6.12.1, prior to personnel entry. PSEG’s corrective actions included implementing process controls requiring the posting of select steam affected areas upon reactor criticality.

The failure to survey an area subject to changing radiological conditions in accordance with 10 CFR 20.1501 to ensure compliance with the requirements of 10 CFR 20.1201, and to accurately brief workers entering a posted high radiation

area (Plant Technical Specification 6.12) on the radiological conditions was determined to be a performance deficiency and a finding. The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute of exposure control and affected the cornerstone objective of providing adequate protection of workers from exposure to radiation. Because the performance deficiency involved a worker entering an uncontrolled high radiation area, the finding was evaluated using Inspection Manual Chapter (IMC) 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process.” The inspectors determined that the finding was of very low safety significance (Green), because it did not involve (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for an overexposure, or (4) an impaired ability to assess dose. The performance deficiency had a cross-cutting aspect related to human performance. Specifically, PSEG did not correctly coordinate surveys and postings of the ‘A’ SJAE rooms following reactor criticality and startup. (Section 2OS1)

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The Hope Creek Generating Station began the inspection period operating at 100% power. On April 6, 2006, the reactor was shutdown to begin Hope Creek's thirteenth refueling outage (RF13). Hope Creek completed the refueling outage and returned to 100% power on May 12, 2006. Hope Creek operated at 100% power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope (1 sample)

The inspectors performed a detailed review of PSEG's seasonal readiness procedures and reviews associated with hot weather conditions. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, and station procedures to identify system operation in extreme hot weather conditions. Station procedures and 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, control area chilled water, circulating water, and electrical switch-yard system health. This inspection sample satisfied the inspection requirement to review 2 - 4 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.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown (3 samples)

a. Inspection Scope

The inspectors reviewed the status of the following three systems to verify the operability of redundant or diverse trains and components when other safety equipment was inoperable. The inspectors also selected single-train systems to verify operability following periods of maintenance or plant conditions that increased the risk worth of the system. The inspectors reviewed applicable operating procedures, walked down control 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.

- ‘D’ residual heat removal train when ‘B’ train was aligned for shutdown cooling on April 26, 2006
- ‘B’ & ‘D’ service water trains when ‘C’ service water train was out-of-service for maintenance on June 1, 2006
- High pressure coolant injection (HPCI) system on June 7, 2006

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Protection - Tours

a. Inspection Scope (9 samples)

The inspectors conducted a tour of the nine 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; that 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. Documents reviewed are listed in the attachment.

- Motor control center area, elevation 102'
- ‘B’ reactor water recirculation pump motor generator set room
- Standby liquid control area
- 'C' residual heat removal heat (RHR) pump room
- 'D' residual heat removal heat (RHR) pump room
- ‘A’ containment instrument gas compressor room
- ‘A’ and ‘C’ 125V battery and battery charger rooms
- Lower control equipment room
- Remote shutdown facility

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 External Flooding

a. Inspection Scope (1 sample)

The inspectors reviewed the design, material condition, and procedures for coping with the design basis probable maximum flood. The inspectors reviewed the UFSAR to determine the barriers required to mitigate flooding in the emergency diesel generator (EDG) areas. The inspectors also reviewed procedures, walked down affected areas and inspected the water tight doors which are required to ensure the EDGs and other safety-related equipment would remain available following the probable maximum flood. Additionally, the inspectors reviewed the maintenance history of the water tight doors in the area to determine whether they were adequately maintained to protect safety-related equipment during postulated external flood conditions.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope (1 sample)

The inspectors reviewed PSEG's program for maintenance and testing of risk-important heat exchangers in the safety auxiliary cooling system (SACS). Specifically, the review included the residual heat removal (RHR) pump motor bearing coolers and seal coolers. The inspectors reviewed calculations, procedures, test results, and vendor documentation to ensure that the coolers would provide adequate heat removal from the motor thrust bearings and the RHR pump seals. The inspectors also reviewed the results of recent SACS chemistry samples. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope (1 sample)

The inspectors observed selected samples of in-process nondestructive examination (NDE) activities. The inspectors also reviewed documentation of additional samples of NDE and component replacement activities which involved welding processes. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The observations and documentation review were performed to

verify activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspectors reviewed a sample of inspection reports initiated as a result of nonconforming conditions identified during Inservice Inspection (ISI) examinations. Also, the inspectors evaluated effectiveness in the resolution of problems identified during ISI activities.

The inspectors observed remote visual (VT) inspection of the steam dryer. The inspectors also witnessed the installation of a jet pump clamp on jet pump #6. The inspectors reviewed the records of liquid penetrant (LP) examinations, ultrasonic (UT) examinations and visual examinations (VT). Additionally, the inspectors witnessed the testing of several hydraulic snubbers to verify effectiveness of the examiner, test equipment and process in identifying degradation of risk significant systems, structures and components and to evaluate those activities for compliance with the requirements of ASME Section XI of the Boiler and Pressure Vessel Code.

The inspectors selected a sample of notifications for review as representative of a nonconforming condition that was evaluated and dispositioned “accept as is” for continued service without repair. Five crack indications on the steam dryer were recordable and dispositioned “accept as-is” for continued service without repair. All of these indications have reinspection requirements during the next refueling outage. The inspectors assessed PSEG’s evaluation and disposition for continued service without repair of a non-conforming condition identified during ISI activities.

PSEG replaced the ‘B’ reactor recirculation pump rotating element during refueling outage 13. The rotating element primarily consisted of the pump shaft, pump impeller, and parts of the pump seal package. The inspectors reviewed the video-recorded visual examination of the interior of the pump volute. No abnormal indication of wear or any other anomalies were noted. PSEG has accepted this component as acceptable for further use. The inspectors concluded that this remote visual examination met the requirements of ASME Section XI.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope (1 sample)

Resident Inspector Quarterly Review

On June 11, 2006, the inspectors observed a simulator training scenario to assess operator performance and training effectiveness. The scenario involved a reactor recirculation pump trip, a reactor coolant leak in the reactor water clean up system, a loss of the primary containment instrument gas system, and a failure of the reactor protection system to scram the reactor. 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 identified by PSEG self-assessments and third-party assessments. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (2 samples)

The inspectors reviewed the two samples listed below for items such as: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) classification and reclassification in accordance with 10 CFR 50.65(a)(1) or (a)(2); and (8) 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 structures, systems, and components (SSCs)/functions classified as (a)(1). In addition, the inspectors specifically reviewed events where ineffective equipment maintenance has resulted in invalid automatic actuations of Engineered Safeguards Systems affecting the operating units. Documents reviewed are listed in the Attachment. Items reviewed included the following:

- 125 Volt inverter system based on failure of the 1AD482 inverter section on March 27, 2006
- ‘C’ emergency diesel generator based on failure of the associated lube oil keepwarm pump mechanical seal on April 23, 2006

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (7 samples)

The inspectors reviewed seven on-line risk management evaluations through direct observation and document reviews for the following configurations:

C ‘D’ EDG inoperable, ‘B’ filtration, recirculation and ventilation system (FRVS) fan inoperable and plant cooldown in progress on April 9, 2006

C Natural circulation operations concurrent with ‘B’ & ‘D’ channel outage work windows on April 13, 2006

C ‘A’ and ‘C’ channel outage work windows with ‘A’ (Loss of Power/Loss of Coolant Accident (LOP/LOCA) test in progress on April 23, 2006
C Loss of the ‘A’ service water train while ‘B’ and ‘D’ service water trains were tagged out for outage related maintenance on April 18, 2006
C ‘B’ service water train unavailable with one source of offsite power unavailable due to work on the 13kV 1-2 breaker on May 9, 2006
C ‘C’ service water pump out-of-service with degraded service water ventilation train performance in the ‘B’ and ‘D’ service water pump bays on June 1, 2006
C Diesel fire pump inoperable on June 11, 2006

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 and associated evaluations documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed are listed in the attachment.

b. Findings

Introduction: A Green self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified when the ‘A’ service water strainer motor tripped on thermal overload (TOL) twice resulting in the ‘A’ strainer being removed from service for emergent repair.

Description: On April 17, 2006, with the unit shutdown, the ‘A’ service water strainer motor tripped on TOL. PSEG determined the tripping of the strainer was due to improperly sized thermal overloads.

The inspectors reviewed this issue and determined that on September 25, 2004, PSEG installed a new 2.0 amp motor for the ‘B’ service water strainer. On October 7, 2004, PSEG identified the strainer was experiencing slightly elevated running current at 2.05 amps. PSEG evaluated this condition and determined that the slightly higher current was expected due to changes in strainer load while in service. This evaluation also stated that the existing Cutler Hammer H1022 (Lo) TOL size was appropriate.

On November 25, 2004, PSEG installed a new 2.0 amp motor on the ‘C’ service water strainer, identified elevated running currents as high as 2.4 amps, and responded to multiple TOL trips during post-maintenance testing. The remaining strainer motors were scheduled for replacement during other maintenance periods. PSEG performed a design change package (DCP) to increase the TOL size to “H1023.” The ‘C’ strainer tripped again and PSEG revised the DCP to further increase the size to “H1024." The new H1024 TOLs were installed in the ‘C’ strainer motor on December 2, 2004. The inspectors identified that PSEG did not conduct an evaluation and extent of condition

review for this condition adverse to quality as required by their notification and corrective action procedures.

PSEG replaced the ‘A’ and ‘D’ strainer motors on May 8 and 23, 2005, respectively. As of May 23, 2005, PSEG had new 2.0 amp motors in all four strainers, but had lower rated H1022 TOLs in the circuitry for the ‘A’, ‘B’, and ‘D’ strainer motors in contrast to the H1024 TOLs in the ‘C’ strainer motor circuitry.

On April 18, 2006, the ‘A’ strainer motor TOLs tripped twice resulting in unplanned unavailability of the ‘A’ strainer. The H1024 TOLs were evaluated by PSEG to be acceptable for all service water strainer motors. PSEG replaced the ‘A’ strainer TOLs and restored the service water train to an operable status on April 19, 2006. However, as was done in November 2004, PSEG did not conduct an evaluation and extent of condition review to implement corrective action for a condition adverse to quality. The inspectors questioned cognizant PSEG operations, engineering, and maintenance personnel regarding evaluation of this condition and whether an extent of condition review was warranted for the ‘B’ and ‘D’ strainer motors which still had the H1022 TOLs installed. Subsequently PSEG wrote notifications to conduct operability reviews on the ‘B’ and ‘D’ service water strainers and to evaluate the multiple TOL trips of the 'A' strainer. PSEG determined that the apparent cause of the ‘A’ strainer TOL trips in April 2006 was the failure to conduct an evaluation and extent of condition review of the ‘C’ strainer TOL trips in November 2004.

<u>Analysis</u>: The inspectors determined that the failure to evaluate and implement corrective actions for a condition adverse to quality resulted in 7 hours of unplanned unavailability for the ‘A’ service water strainer in April 2006 and constitutes a performance deficiency. Because PSEG did not, in accordance with their procedures, evaluate and perform an extent of condition review for multiple trips of the ‘C’ strainer motor, they did not implement corrective actions to prevent a similar condition in the ‘A’ strainer.

This issue is more than minor because it is associated with the equipment performance attribute of the mitigating systems cornerstone and affected the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

In accordance with NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 7, the inspectors conducted a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green). The inspectors verified PSEG’s shutdown mitigation capability and determined that the finding was not similar to those requiring a Phase 2 or Phase 3 analysis. This finding had a cross-cutting aspect in problem identification and resolution because PSEG did not adequately implement corrective action for a condition adverse to quality. Specifically, PSEG did not conduct an evaluation and implement corrective action for the elevated running current and subsequent multiple TOL trip condition of the ‘C’ service water strainer motor.

Enforcement: 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, PSEG did not implement corrective action for the elevated running current and multiple TOL trip condition of the 'C' service water strainer motor on November 25, 2004. As a result, the 'A' service water train strainer motor experienced elevated running current and multiple TOL trips and accrued 7 hours of unplanned unavailability on April 18, 2006. PSEG' s corrective actions included corrective maintenance activities to increase the size of the thermal overloads on the 'A', 'B', and 'D' strainer motors. Because this finding is of very low safety significance and has been entered into PSEG's corrective action program (evaluations 70058063 and 70059256), this finding is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000354/2006003-01, Corrective Actions to Prevent Repeat Failures of Service Water Strainer Overloads not Implemented**.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope (3 samples)

The inspectors evaluated personnel performance during two planned evolutions and one unplanned plant transient. The inspectors observed control room operator performance to verify that operator actions were consistent with station procedures and that all applicable technical specification action statements were adhered to. The inspectors reviewed trends in applicable plant parameters to verify that plant equipment operated as designed. The inspectors also reviewed evaluations associated with plant transients to verify PSEG identified causes for the plant transient and implemented appropriate corrective actions. The following evolutions and transients were observed:

- Intermediate reactor recirculation pump runback during reactor shutdown on April 6, 2006
- Reactor recirculation pump motor generator mechanical and electrical stop setting on June 9, 2006
- Reactor recirculation loop and shutdown cooling loop vibration test performed June 16, 2006

The Hope Creek plant has operated with a limitation on the maximum recirculation pump speeds that are lower than the plant design of 1680 revolutions per minute (rpm) to minimize system vibration. PSEG instrumented the recirculation system piping and pumps to measure the system vibration at various pump speeds with the objective of selecting pump operating speeds associated with minimizing system component vibration and avoiding speeds that produce resonant vibration. PSEG had previously taken pipe vibration measurements at various pump speeds below 1500 rpm to correlate pump speed to piping system vibration over about a 2 year period. The plant ran with recirculation pump speeds that correlate to minimum and acceptable vibration levels.

The objectives of the reactor recirculation pump test conducted on June 9-16, 2006, included the identification of any higher pump speeds that should be avoided to minimize excess system vibration. The test program included a baseline design analysis, a large array of direct measurement points, computer based evaluation of the data from the measurements at various pump speeds, and in-plant observations by plant operators to monitor reactor building noise and vibration.

Inspection was performed on the testing evolution of the Post ‘B’ Reactor Recirculation Pump Replacement Vibration Evaluation for Core Flows greater than 100 Mlb/hr. On June 8, 2006, the inspectors walked down the test areas and reviewed the test plans with the system engineer responsible for the testing process. On June 9, 2006, inspectors observed the first portion of the test cycle, which was to reduce plant power to 95% by inserting control rods and then separately increasing each of the two recirculation pumps to reach the test level of flow rate. This was achieved at about 1555 rpm pump speed and included the setting of recirculation pump MG sets mechanical and electrical stops. As this was done for both the ‘A’ and ‘B’ pumps, the inspectors observed data vibration measurement and listened for the system sounds in the vicinity of the two pipe tunnels and the jet pump instrument racks. The inspectors observed a meeting in which the testing team debriefed PSEG management on the activities at the conclusion of the first portion of the test cycle.

The inspectors observed the pre-job brief and execution of the second phase of the test on June 16, 2006. This part of the test raised pump speeds on the ‘A’ and ‘B’ pumps simultaneously from 100 Mlbm/hr to 104.5 Mlbm/hr. Vibration data on the recirculation and shutdown cooling system was gathered at various points during the speed increase. The inspectors observed control room activities as well as walked down portions of the reactor building to determine if abnormal vibrations were present. PSEG reviewed vibration data and determined that no alarm thresholds were reached during the performance of the test.

No unusual noise or vibrations were noted by the inspectors during the observed testing and pump speed changes. PSEG had an equipment operator assigned to observe the system conditions of noise and vibration during the test for comparison to normal plant operation. Discussion with the equipment operator confirmed the inspectors observation in regard to noise and vibrations. PSEG engineers analyzed the vibration data collected and concluded that it correlated with field observations in that no abnormal vibrations were present. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (8 samples)

The inspectors reviewed the following eight issues for operability. The inspectors evaluated the technical adequacy of the associated evaluations to verify operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors reviewed the UFSAR and other design basis documents to verify that the system or component remained available to perform its intended function. Interviews were conducted with control room operators and staff engineers. The inspectors walked down plant components and systems to examine their condition and corroborate the adequacy of PSEG's operability assessment. The inspectors also reviewed a sampling of notifications to verify that PSEG was identifying and correcting deficiencies associated with operability determinations. Documents reviewed are listed in the attachment.

- NOTF 20277825, Failure of 'B' control room emergency filtration to produce adequate differential pressure
- NOTF 20274462, High vibrations on 'C' emergency diesel generator lube oil keepwarm pump
- NOTF 20278850, 'D' emergency diesel generator load sequencer failure during surveillance test
- NOTF 20280569, 'A' service water strainer motor trips on thermal overload
- NOTF 20283884, Unexpected gain adjustments on LPRMs following refueling outage
- NOTF 20286560, Low level observed on wide-range torus water level instrument
- NOTF 20288035, 'B' reactor recirculation pump motor-generator voltage regulator oscillations
- NOTF 20280701, Control rod blade 02-138 blistering found during refueling outage

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope (1 sample)

The inspectors reviewed one design change associated with the replacement of the 'B' reactor recirculation pump internals. Specifically, the inspectors reviewed Engineering Change 80076232, Revision 5, which was implemented to provide an upgrade of the 'B' reactor recirculation pump by replacing the pump cover and internals to resolve thermal fatigue cracking concerns. In general, the changes incorporated into the new design were intended to reduce the potential for failed rotating parts. Several of the changes included shaft cracking mitigating features, a welded on impeller and improved maintenance and inspection capabilities.

The inspectors performed a field walkdown of selected portions of the modification to verify that the installation was in accordance with the design requirements. The inspectors reviewed the change to seal purge flow, along with the elimination of one of the two seal coolers and the jacket cooler from the pump, to ensure the changes had been adequately analyzed and incorporated into system procedures. Due to minor configuration changes in the connections of the new pump cover design, the attached piping required minor rerouting. A sample calculation associated with the re-analysis for minor piping modifications was chosen for review to verify that pipe stress remained within acceptable limits. Instrument and Control Calculation, SC-ED-0503, was reviewed to ensure the change in the setpoint for the alarm to the plant computer on low pump seal cooler flow had an adequate engineering basis.

Additionally, the inspectors reviewed the design change determination that the new pump had the same nominal system performance with respect to the original pump capabilities. The reactor recirculation pump vibration monitoring procedure was reviewed to ensure that appropriate revisions were made to incorporate the effects of the modification such as the requirement to determine new critical pump speeds. The proposed revision to Procedure HC.OP-SO.BB-0002(Q), Rev. 59, with field change requests for the modification was reviewed to ensure adequate incorporation of the design changes to the operating procedure. Lastly, PSEG's analyses of recirculation pump startup vibration data was reviewed to evaluate the methodology used in determining the new pump critical speeds.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (8 samples)

The inspectors reviewed the eight post-maintenance tests 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 basis documentation. The inspectors also witnessed the test or reviewed the test data to verify test results adequately demonstrated restoration of the affected safety functions. Documents reviewed are listed in the attachment.

- DCP 80076232, Replacement of 'B' reactor recirculation pump
- WO 60058580, Replacement of 'B' station service water strainer body
- WO 60063300, 'B' control room emergency filtration train damper not maintaining required flow
- WO 50078803, Repair of 'C' low pressure coolant injection valve BCHV-F007C
- WO 60063505, Repair of 'A' core spray minimum flow check valve BE-V028
- WO 60063201, Station service water pump 'A' packing replacement

- WO 60061918, Repair of 'C' emergency diesel generator lube oil keep-warm pump
- WO 30119573, Emergent repair of damaged refueling mast

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

The inspectors reviewed the schedule and risk assessment documents associated with the Hope Creek RF13 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 with a regional Senior Risk Analyst to identify risk significant equipment configurations and 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. Prior to RF13, PSEG postulated through a review of work performed in refueling outage 12 (RF12), observed drywell conditions at the completion of RF12, and radionuclide analysis of drywell sump drains, that most of the measured unidentified leakage during the subsequent operating cycle was likely from the 'C' main steam isolation valve (MSIV) stem-packing. The inspectors confirmed through visual observation that a majority of the unidentified drywell leakage was due to stem packing leakage identified on 'C' MSIV during the drywell walkdown. 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 observe 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 hanged 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, the 500kV switchyard, onsite electrical buses, and EDGs were conducted during risk significant electrical configurations and configuration changes to confirm the equipment alignments met requirements.

Risk significant plant evolutions were observed 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 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 during fuel offload and reload activities to verify that TS requirements and procedure requirements were met for secondary containment. Specifically, the inspectors periodically reviewed control room logs for secondary containment penetrations that were open and verified that materials and equipment were staged to seal these penetrations during fuel movement activities as assumed in the licensing basis.

The inspectors walked down the containment drywell prior to reactor startup to verify no evidence of 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. Inspectors reviewed RCS leakage surveillance tests following plant startup to verify RCS integrity.

The inspectors responded to an unexpected reactor vessel level change condition on April 26, 2006. During reactor reassembly activities, indicated shutdown reactor water level rose by more than 65 inches. Operators ceased main steam line draining activities and investigated the issue. The inspectors discussed the transient with operators, engineers, and plant management to understand the event and assess PSEG’s

evaluation of the cause and followup actions. The inspectors reviewed operator actions, station procedures, and plant response to verify proper actions were taken and plant equipment responded as expected. The inspectors reviewed PSEG's apparent cause evaluation of the condition and equipment issues. PSEG determined that procedural direction to install blank flanges on RPV head penetrations was the apparent cause of the loss of shutdown level indication.

b. Findings

Introduction: A Green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified when the single source of shutdown reactor water level indication was rendered inaccurate for 7 hours during reactor vessel reassembly.

Description: On April 26, 2006, Hope Creek operators were maintaining reactor water level between 210 and 217 inches, which is just below the reactor pressure vessel (RPV) head flange. Shutdown level recorder LI-R605 and visual observation from the refueling floor were the two sources of indication for reactor water level. At 12:45 am on April 26, 2006, the RPV head was set on the vessel head flange leaving LI-R605 as the single indication of reactor water level. However, all penetrations on the RPV head were isolated via bolted blank flanges (for foreign material exclusion control) creating a non-vented condition for the reactor vessel. At 2:17 am, operators began lowering reactor water level to a new band of 80 to 90 inches to allow for draining of the main steam lines which are at 118 inches. Lowering reactor water level rendered LI-R605 inaccurate, because the RPV was not vented. At 7:33 am, operators began draining the main steam lines to support main steam line isolation valve maintenance. A few minutes later, operators observed that reactor water level on LI-R605 had unexpectedly dropped from 86 to 76 inches and stopped the main steam line draining evolution. At 7:48 am, operators had begun restoring reactor water level to the pre-transient level when indicated reactor water level began to rise rapidly from 83 inches to 145 inches. While operators were investigating this condition, at 8:15 am, reactor reassembly personnel informed operations control room personnel that they had removed a foreign material exclusion blank flange cover from the RPV head vent flange at approximately 7:45 am.

PSEG's RPV disassembly procedure directed the installation of blank flanges on the RPV head penetration connections. PSEG's RPV reassembly and RPV head installation procedures did not contain precautions, cautions or instructions to maintain the RPV head vented following reinstallation of the RPV head on the vessel flange. This was necessary to maintain the reactor water level indication (LI-R605) accurate with a changing level in the reactor vessel.

The integrated operations procedure for moving from Refueling to Cold Shutdown also lacked specific guidance to assure that reactor remained vented to maintain accuracy of the single indication of reactor water level in the shutdown range.

Analysis: A performance deficiency was identified in that the shutdown reactor water level indication was rendered inaccurate for 7 hours because PSEG's integrated plant operations and reactor vessel maintenance procedures did not contain sufficient instructions to ensure that the RPV remained vented during reactor reassembly activities. The finding was more than minor because it was associated with the procedure quality and configuration control attributes of the mitigating systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter (IMC) 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, Checklist 8, the inspectors conducted a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green). The inspectors verified PSEG's shutdown mitigation capability and determined that the finding was not similar to those requiring a Phase 2 or Phase 3 analysis. The finding had a cross-cutting aspect in the area of human performance because PSEG did not have adequate procedures to maintain accurate shutdown range reactor water level indication.

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, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. Contrary to the above, the PSEG maintenance and integrated operations procedures did not contain sufficient guidance to ensure that the RPV remained vented. As a result, the single indication of reactor water level in the shutdown range was rendered inaccurate while lowering reactor water level on April 26, 2006. Because the finding was of very low safety significance and has been entered into PSEG's corrective action program (notification 20282029) this deficiency is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000354/2006003-02, Loss of Shutdown Reactor Vessel Level Indication.**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (6 Samples)

The inspectors witnessed 6 surveillance tests and/or reviewed test data of selected surveillance tests listed below to verify that the test met the requirements of the technical specifications, UFSAR, and station procedures. The inspectors also determined whether the testing effectively demonstrated that the systems and components were operationally ready and capable of performing their intended safety functions. Documents reviewed are listed in the attachment.

- WO 50081260, 50082713, Residual heat removal system heat exchanger flow measurement - 18 Month test
- Sample 196492, Reactor coolant system dose equivalent iodine calculation
- WO 50080759, Seat leakage testing of residual heat removal valve 1BCV-113

- WO 50082684, ‘B’ emergency diesel generator LOP/LOCA testing
- WO 50082344, Pressure isolation valve inputs into total identified leakage
- WO 50094668, Drywell floor and equipment drain sump monitor channel functional test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

A temporary plant modification associated with the reactor building polar crane was reviewed by the inspectors. The modification bypassed the load-cell interlock during refueling outage activities. The inspectors verified the modification was consistent with the design and licensing bases of the crane and that the performance capability of the crane was not degraded by the modification. The inspectors reviewed documents to verify PSEG followed their processes for implementing temporary modifications on safety-related equipment. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

Resident inspectors evaluated the conduct of control room operators during simulated emergency condition scenarios on June 12, 2006, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation (PAR) development activities. The inspectors observed emergency response operations in the simulated control room to verify that event classification and notifications were done in accordance with regulations and procedures. The inspectors also attended PSEG’s critique of the drill to compare any inspector-observed weakness with those identified by PSEG in order to verify whether PSEG was properly identifying problems. Documents reviewed 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)

a. Inspection Scope (7 samples)

Based on PSEG's schedule of work activities during the refueling outage (RF13), the inspectors selected three jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation; reviewed radiological job requirements (radiation work permit [RWP] requirements and work procedure requirements); observed job performance with respect to these requirements; and, determined that radiological conditions in the work area were adequately communicated to workers through briefings and postings. The jobs reviewed were: safety relief valve work; in-service inspection; and, control rod drive replacement.

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.

During job performance observations, the inspectors observed radiation worker performance with respect to stated radiation protection work requirements and determined that they were aware of the significant radiological conditions in their workplace, and the RWP 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 radiation protection work requirements; determined that they were aware of the radiological conditions in their workplace and the RWP controls/limits; and, determined that their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

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.

The inspectors walked down these areas or their perimeters to determine: whether prescribed RWP, 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 RWPs used to access these and other high radiation areas and identified what work control instructions or control barriers had been specified.

The inspectors reviewed electronic personal dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy.

In addition, the inspectors reviewed the circumstances surrounding a plant worker receiving a dose rate alarm while working in a radiation area in the turbine building. Investigation of the event by PSEG determined that the work area had radiation levels in excess of 100 millirem per hour measured 30 centimeters from the source of radiation, but was not posted or controlled as a high radiation area.

b. Findings

Introduction. A Green self-revealing non-cited violation of 10CFR20.1501, “Surveys and Monitoring - General,” was identified when a high dose rate alarm was received by a plant worker when working in an improperly controlled high radiation area.

Description. On May 7, 2006, during reactor startup operations at the conclusion of refueling outage RF13, a plant worker entered the ‘A’ steam jet air ejector (SJAE) room. After working in the room for a few minutes, the workers electronic dosimeter began to alarm due to high dose rate. The worker immediately exited the room and notified radiation protection personnel. The electronic dosimeter indicated an exposure of less than 4 millirem, however, the peak dose rate measured by the electronic dosimeter was 122 millirem per hour. The alarm setpoint was set for 10 millirem per hour, which is consistent with entries into some areas in the plant that are not high radiation areas.

PSEG performed a prompt investigation of the situation. The investigation into the cause of the alarm revealed that dose rates in the area were in excess of 100 millirem per hour measured 30 centimeters from the source of radiation. PSEG also determined that the room was not posted or controlled as a high radiation area. The area was subsequently posted and controlled as a high radiation area. PSEG concluded that there was no formal procedural guidance on when to survey or post this area as a high radiation area.

Analysis. The failure to survey an area subject to changing radiological conditions in accordance with 10 CFR 20.1501 to ensure compliance with the requirements of 10 CFR 20.1201, and to accurately brief workers entering a posted high radiation area (Plant technical specification 6.12) on the radiological conditions was determined to be a performance deficiency and a finding. The finding is more than minor because it is associated with the occupational radiation safety cornerstone attribute of exposure control and affected the cornerstone objective of providing adequate protection of workers from exposure to radiation. Specifically, the radiological conditions present in the ‘A’ SJAE required posting and control as a high radiation area, in accordance with plant technical specification 6.12.1. Because the performance deficiency involved a worker entering an uncontrolled high radiation area, the finding was evaluated using Inspection Manual Chapter (IMC) 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process.” The inspectors determined that the finding was of very low safety significance (Green), because it did not involve (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for an overexposure, or (4) an

impaired ability to assess dose. The performance deficiency had a cross-cutting aspect related to human performance associated with it. Specifically, PSEG work controls did not correctly coordinate surveys and postings of the ‘A’ SJAE rooms following reactor criticality and startup.

Enforcement. 10CFR20.1501, “Surveys and Monitoring - General,” requires the licensee to make or cause to be made surveys that are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels to ensure compliance with 10CFR20.1201 and plant technical specification 6.12.1. Contrary to this requirement, PSEG failed to survey the ‘A’ SJAE room on May 3, 2006, when the reactor was made critical. The failure to survey resulted in the ‘A’ SJAE room becoming an uncontrolled high radiation area that was subsequently accessed by a plant worker on May 7, 2006.

Because this finding was of very low safety significance and PSEG entered this finding into the corrective action program as notification 20283666, this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600: **NCV 05000354/2006003-03, Deficiency in Access Control to Radiological Areas.**

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (3 samples)

The inspectors obtained from PSEG a list of work activities ranked by actual or estimated exposure that were in progress during the current refueling outage and selected the 3 work activities of highest exposure significance (listed in paragraph 2OS1 above).

The inspectors reviewed the as low as is reasonably achievable (ALARA) work activity evaluations, exposure estimates, and exposure mitigation requirements and determined that PSEG had established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that are ALARA.

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.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

a. Inspection Scope (1 sample)

The inspectors verified the calibration expiration date and validated that the source response check was current on radiation detection instruments staged for use.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

g. Inspection Scope (5 samples)

Cornerstone: Initiating Events

The inspectors reviewed PSEG's program to gather, evaluate and report information on the following performance indicators (PIs). The inspectors used the guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 3, to assess the accuracy of PSEG's collection and reporting of PI data. The documents reviewed by the inspectors are listed in the attachment.

- Unplanned SCRAMS per 7,000 Critical Hours
- Unplanned SCRAMS with Loss of Normal Heat Removal
- Unplanned Power Changes per 7,000 Critical Hours

The inspectors verified the accuracy and completeness of reported manual and automatic unplanned scrams during the period of October 1, 2004 through March 31, 2006 for the "Unplanned Scrams per 7,000 Critical Hours" PI.

The inspectors reviewed and verified PSEG's basis for including or excluding an unplanned reactor scrams for the "Unplanned Scrams with Loss of Normal Heat Removal" PI during the period of October 1, 2004 through March 31, 2006.

The inspectors verified the accuracy and completeness of reported transients that resulted in unplanned changes and fluctuations in reactor power of greater than 20 percent power for the "Unplanned Power Changes per 7,000 Critical Hours" PI during the period of October 1, 2004 through March 31, 2006.

Cornerstone: Barrier Integrity

- Reactor Coolant System Specific Activity

- Reactor Coolant System Leakage

The inspectors verified the methods used to calculate the reactor coolant system specific activity PI and reviewed the accuracy of the PI data submitted during for the period July 1, 2004 through March 31, 2006.

The inspectors verified the methods used to calculate the reactor coolant system leakage PI. The inspectors verified the accuracy of PI data submitted for the period July 1, 2004 through March 31, 2006.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the Corrective Action Program

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a daily screening of all items entered into PSEG's corrective action program to identify repetitive equipment failures or specific human performance issues for additional review. This was accomplished by reviewing the description of each new notification and attending management review committee meetings. Risk significant issues were reviewed further by inspectors through Plant Status or were selected as a sample for inspection under Reactor Safety inspection attachments.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope (1 sample)

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of PSEG's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.1. The review also included issues documented outside the normal CAP in system health reports, corrective maintenance work orders, component status reports, site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the six-month period of December 1, 2005, through June 1, 2006, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors specifically trended events affecting reactivity management reactivity events as defined in PSEG procedure NC.NA-AP.ZZ-0089. The inspectors compared and contrasted their results with the results contained in PSEG's latest monthly Reactivity Management Performance Indicator and station reactivity management procedure. Corrective actions associated with a sample of the issues identified in PSEG's performance indicator were reviewed for adequacy. Documents reviewed are listed in the attachment.

b. Assessment and Observations

No findings of significance were identified.

PSEG's Reactivity Management performance indicator identified three reactivity management challenges which correlated with the issues identified by the inspectors through plant status and CAP reviews.

.3 Annual Sample: Station Service Water Deicing Line Degradation

a. Inspection Scope

The inspectors reviewed PSEG's actions to resolve repetitive degraded conditions identified on the deicing system for the service water intake structure. Specifically, flooding of a number of underground valve pits containing motor-operated valves used to operate the non-safety related deicing system was identified a number of times in the CAP. This issue was selected due to its potential to impact the operability of risk significant equipment, including the potential for common cause failure of all four trains of service water due to frazil ice buildup on the service water intake trash racks and traveling screens.

The deicing system is not identified as a safety-related system; however, it is described in the UFSAR and used in station emergency procedures to deliver warming water to the service water intake to mitigate both frazil ice buildup and potential blockage of the service water trash racks and traveling screens.

The deicing system draws water from either the circulating water system at the outlet of the main condenser or from the service water system discharge header servicing the cooling tower basin. Both deicing system warm water supplies are normally isolated by a single motor-operated valve in each supply header. The valves are normally controlled remotely from the control room when needed, but have the capability of being operated manually inside the valve pits.

The inspectors reviewed notifications, evaluations, design documentation and interviewed cognizant engineers and operators to determine if the system was capable of performing its design function. The inspectors also reviewed PSEG's plans to address and correct the degraded conditions.

b. Findings and Observations

No findings of significance were identified.

The inspectors found that PSEG generally entered degraded conditions into the corrective action program. PSEG had entered degraded conditions associated with the flooded valve pits and the potential for the valves in the valve pit to fail a number of times over several years. However, PSEG did not thoroughly evaluate the impact of the degraded conditions on the ability of the deicing system to perform its design function.

Also, PSEG did not effect corrective actions or maintenance activities to repair known degraded conditions of the motor operated valves described above. Additionally, PSEG determined through a review of maintenance history that the valves were tagged out in the closed position from at least February 1992 until December 2005.

Following questioning from inspectors, PSEG evaluated the condition of the service water deicing system. PSEG's evaluation included corrective actions that developed a deicing system restoration plan to improve the material condition of the system and systematically inspect and test system components prior to the onset of cold weather in 2006. Improvements include sealing valve pit penetrations, repair or installation of new sump pumps in the valve pits, repair electrical supplies to valve pit motor operated valves, repair or replacement of trash racks and support components, and replacement of the deicing header and downcomer piping.

The inspectors determined that PSEG had the ability to place the system in service manually, if required, at all times. The inspectors also concluded that the corrective actions developed by PSEG were appropriate to the extent it would return the system to a fully functional condition and adequately address known deficiencies.

.4 Safety Conscious Work Environment Metric Review

a. Inspection Scope

The inspectors reviewed PSEG's progress in addressing safety conscious work environment (SCWE) issues that were discussed in the NRC's annual assessment letter dated March 3, 2006. In that letter, the NRC staff documented a SCWE substantive cross-cutting issue and stated the NRC's intention to continue to monitor progress in this area.

On May 10, 2006, the inspectors conducted a sampling review of PSEG's SCWE metrics, or PIs, for first quarter 2006. Documents reviewed are listed in the attachment.

b. Findings and Observations

No findings of significance were identified.

In first quarter 2006, PSEG identified twenty-four PIs as being green or satisfactory while six PIs were identified as red or needing improvement. An additional PI documenting the results of a recent Synergy Consulting Services Corporation survey of the Salem/Hope Creek workforce was added in the first quarter 2006 PIs. This was an improvement from the fourth quarter 2005 results of twenty-one green PIs and eight red PIs.

4OA5 Other Activities

.1 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the INPO plant assessment of the Hope Creek Generating Station conducted in March 2006. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC assessment of PSEG's performance and to verify if any significant safety issues were identified that required further NRC review.

b. Findings

No findings of significance were identified.

.2 Implementation of Temporary Instruction (TI) 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

a. Inspection Scope

The objective of TI 2515/165, "Operational Readiness of Offsite Power and Impact on Plant Risk," was to gather information to support the assessment of nuclear power plant operation readiness of offsite power systems and impact on plant risk. The inspectors evaluated PSEG procedures against the specific offsite power, risk assessment, and system grid reliability requirements of TI 2515/165. The inspectors also discussed the attributes with PSEG personnel.

The information gathered while completing this TI was forwarded to the Office of Nuclear Reactor Regulation (NRR) for further review and evaluation on April 3, 2006. The NRR review was completed with no further action required with respect to TI 2515/165.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 2006002-02, Additional NRC Review Required to Further Evaluate RHR Heat Exchanger (HX) Flow Testing Methodology

URI 2006002-02 was opened in NRC Inspection Report 05000354/2006002 Section 1R07.2 because inspectors identified issues with the methodology PSEG used to perform residual heat removal (RHR) HX flow testing. Specifically, the inspectors identified that: (1) the 18-month ST did not provide direction on how to calculate RHR HX and bypass flows; (2) the 18-month ST did not provide direction on placement of ultrasonic flow instruments, calibration of these instruments, or required accuracy and range of these instruments; (3) PSEG used temporarily installed measuring and test

equipment having a minimum accuracy of ± 0.5% for the RHR combined (HX & bypass) flow rate during the quarterly RHR pump ST, but used the less accurate installed plant instrumentation for the 18 month ST; (4) PSEG did not use the recorded ultrasonic flow instrument data on the RHR HX outlet lines in their calculation of HX flow (this temporary instrument was specifically installed for this flow test); and (5) the 35 sets of recorded data for each HX appeared erratic.

The inspectors reviewed notifications 20272419, 20288825, and evaluation 70054151 that documents PSEG's response to the above issues. The inspectors also reviewed the results of the 'A' and 'B' RHR HX flow testing surveillance tests during the refueling outage as listed in Section 1R22 of this report. As a corrective action from evaluation 70054151, PSEG changed the surveillance test procedure and testing methodology prior to the refueling outage to improve the direction provided to calculate RHR HX bypass flow and place the ultrasonic detector at a fixed location on the HX discharge line to ensure accurate and consistent test results. The ultrasonic measurement device that measured bypass flow previously was removed altogether to eliminate large measurement fluctuations due to low flow conditions in the bypass line. The test results achieved during the refueling outage demonstrated that the RHR HXs were operable.

The inspectors determined that the procedure and methodology changes made by PSEG addressed the issues identified in URI 2006002-02 satisfactorily. This URI is closed.

4OA6 Meetings, Including Exit

NRC/PSEG Management Meeting - Reactor Oversight Process Annual Assessment. The NRC conducted a meeting with PSEG on May 17, 2006, to discuss the NRC's annual assessment of safety performance at Salem and Hope Creek for calendar year 2005 and PSEG actions to improve the safety conscious work environment. The meeting occurred at the Holiday Inn Select in Bridgeport, New Jersey and was open for public observation. A copy of slide presentations and other background documents can be found in ADAMS under accession number ML060680412.

Exit Meeting. On June 6, 2006, the inspectors presented their overall findings to members of PSEG management led by Messrs. Barnes and Massaro. None of the information reviewed by the inspectors was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Barnes, Site Vice President
M. Massaro, Hope Creek Plant Manager
H. Hanson, Operations Director
Paul Davison, Engineering Director
Mark Pfizenmeier, Senior Manager Plant Engineering
Joan Glunt, Work Management Director
M. Davis, Radiation Protection Supervisor
T. O'Hare, Radiation Protection Supervisor
B. Sebastian, Radiation Protection Manager
J. Barstow, Regulatory Affairs/Compliance Engineer
J. Williams, Hope Creek Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000354/2006003-01	NCV	Corrective Actions to Prevent Repeat Failures of Service Water Strainer Overloads not Implemented (Section 1R13)
05000354/2006003-02	NCV	Loss of Shutdown Reactor Pressure Vessel Level Indication (Section 1R20)
05000354/2006003-03	NCV	Deficiency in Access Control to Radiological Areas (Section 2OS1)

Closed

05000354/2006002-02	URI	Additional NRC Review Required to Further Evaluate RHR HX Flow Testing Methodology (Section 4OA5.3)

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 (HC.OP-AP.ZZ-0108)
HCGS NCO Narrative Logs
HCGS Plant Status Reports
Weekly Reactor Engineering Guidance to Hope Creek Operations
Hope Creek Operations Night Orders and Temporary Standing Orders

Section 1R01: Adverse Weather Protection

Procedures
SH.OP-DG.ZZ-0011, Rev. 4, Station Seasonal Readiness Guide
WC-AA-107, Rev. 2, Seasonal Readiness
NC.OP-DG.ZZ-0002, Rev. 6, Severe Weather Guide
HC.OP-AB.COOL-0001, Rev. 7, Station Service Water
HC.OP-AB.HVAC-0001, Rev. 3, HVAC

Corrective Action Notifications
20246072 20272916 20276127

Orders
70055389 70054394

Other Documents
2006 Hope Creek Summer Readiness Matrix
Service Water SHIP Action Plan for Red and Yellow Systems dated February 3, 2006 and February 28, 2006
Chilled Water SHIP Action Plan for Red and Yellow Systems

Section 1R04: Equipment Alignment

Procedures
HC.OP-SO.BC-0001, Rev 41, Residual Heat Removal System Operation
HC.OP-SO.BC-0002, Rev 18, Decay Heat Removal Operation
HC.OP-ST.BJ-0001, Rev 11, HPCI System Piping and Flow Path Verification - Monthly
HC.OP-SO.BJ-0001, Rev 32, High Pressure Coolant Injection System Operation
HC.OP-SO.EA-0001, Rev. 28, Service Water System Operation

Drawings
M-55-1, Rev. 24, High Pressure Coolant Injection
M-56-1, Rev. 16, HPCI Pump Turbine

Corrective Action Notifications
20264759

Other Documents
EC-0074, Rev. 11, HCGS Decay Heat-up Rates and Curves

Section 1R05: Fire Protection

Procedures
Hope Creek Pre-Fire Plan FRH-II-413, Rev. 3, HPCI Pump & Turbine Room, RHR Pump & Heat Exchanger Rooms, Elevation 54'
Hope Creek Pre-Fire Plan FRH-II-412, Rev.3, RCIC Pump & Turbine Room, RHR Pump & Heat Exchanger Rooms & Electrical Equipment Room, Elevation 54'
Hope Creek Pre-Fire Plan FRH-II-151, Rev. 3, Turbine Building Elevation 137'
Hope Creek Pre-Fire Plan FRH-II-434, Rev. 3, Motor Control Center Area Elevation 102'
Hope Creek Pre-Fire Plan FRH-II-461, Rev. 3, FRVS Rooms, MCC Area, Recombiner Areas, Spent Fuel Pool & Gamma Scan Detector Area
Hope Creek Pre-Fire Plan FRH-II-351, Rev. 6, Service and Radwaste Area Elevation 137'
Hope Creek Pre-Fire Plan FRH-II-442, Rev. 4, Inert Gases Compressor Rooms, FRVS Re-Circulating Unit Area, Steam Vent & Equipment Area Elevation 132'
Hope Creek Pre-Fire Plan FRH-II-551, Rev. 6, Battery Rooms & Cable Chases Elevation 146'
Hope Creek Pre-Fire Plan FRH-II-532, Rev. 5, Lower Control Equipment Room, Elevation 102'
HC.FP-SV.ZZ-0026, Rev. 4, Flood and Fire Barrier Penetration Seal Inspection
HC.FP-AP.ZZ-0004, Rev. 9, Actions for Inoperable Fire Protection - Hope Creek Station
HC.FP-AP.ZZ-0025, Rev. 4, Precautions Against Fire

Corrective Action Notifications
20269258 20278166 20248868

Orders
50069540

Other Documents
Hope Creek Hourly Fire Watch Patrol Inspection Log dated May 24, 2006

Section 1R06: Flood Protection Measures

Procedures
HC.OP-AB.MISC-0001 Rev. 6, Acts of Nature
NC.OP-DG.ZZ-0002 Rev. 6, Severe Weather Guide

Drawings
A-0203-0, Rev. 11, Hope Creek Generating Station, General Plant Floor Plan, Level 3 - Elevation 102'-0"

Corrective Action Notifications
20287013 20268230 20227388 20212131

Orders
70042746

Other Documents
UFSAR Section 2.4, Hydrologic Engineering
Individual Plant Examination for External Events (IPEEE) sections 5.3.4 and 5.5
Technical Specifications 3/4.7.3, Flood Protection Measures

Section 1R07: Heat Sink Performance

Procedures
HC.SE-PR.EG-0001, Rev. 5, Safety and Auxiliary Cooling System Annual Biofouling Monitoring
NC.MCT-DG.ZZ-0001, Rev. 1, Balance of Plant Heat Exchanger Condition Assessment Program

Drawings
M-II-I Sheet 1, Rev 17, Safety Auxiliaries Cooling Reactor Building

Corrective Action Notifications
20050728 20060607 20051221 20050702

Other Documents
Chemistry Measurements of SACS from June 1, 2005 to June 20, 2006
Information Notice 89-71, Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a Loss-of-Coolant Accident
Calculation Number EG-0020(Q), Rev 8
Calculation Number SC-EG-0159, Rev 1
Calculation Number EG-0011(Q), Rev 1
Design, Installation and Test Specification for Safety and Turbine Auxiliaries Cooling System for the Hope Creek Generating Station
PN1-E11-C002-0051, Vendor Information for the Residual Heat Removal System Pump
EPRI NP-7552, Heat Exchanger Performance Monitoring Guidelines
DE-CB.EG-0054, Rev. 2, Configuration Baseline Documentation for Safety and Turbine Auxiliaries Cooling System

Section 1R08: Inservice Inspection Activities

Procedures
SH.RA-IS.ZZ-0116, Revision 10, 4/16/04; Inservice Inspection Visual Examination of Nuclear Class I Bolting, Nuclear Class 3 Integral Welded Attachments, Nuclear Class I Pump/Valve Internal Surfaces, Nuclear Class I External Pump Casing Weld Surfaces
Hope Creek Nuclear Generating Station, Inservice Inspection Program Second 10 Year Interval
IWE Long Term Plan, Revision 0, January 2000
HC.CH-SA.ZZ-0004, Revision 1, 2/86/06; Determination Of Reactor Percent Moisture Carryover

Drawings
L002040, Revision C, Pump Sectional Type DVSS
2F-1435, Revision 0, Outline Reactor Recirculating Pump
2F-1437, Revision H, Outline Reactor Recirculating Pump
1-P-ED-220, Revision 2; Engineered Small Piping/Drywell Building; RACS Water From Pump BP-201 Internal Heat Exchanger
1-P-ED-221, Revision 2; Engineered Small Piping/Drywell Building; RACS Water From Pump BP-201 Internal Heat Exchanger

Calculations
H-1-ZZXX-SEE-0166-0, 3/12/87; NRC IE Information Notice No. 86-99 Degradation of Steel Containments

Corrective Action Notifications

20211152	20255245	20210893	20212311	20251588	20280760
20213688	20213925	20211135	20212272	20215541	20280742
20209438	20208591	20212346	20212353	20207276	20280574
20214041	20141931	20212271	20175906	20280110	20280904
20230960	20211910	20211638	20215470	20280861	20212346
20260038	20175906				

Design Change Packages
DCP 80089168, Revision 0, 4/17/06; Jet Pump Slip Joint Clamp Repair
DCP 80040594, 12/16/04; HIBB-10-S-201 piping, ASME III, Class 2 Pressure Boundary Repair
DCP 80076232, ANSI B31.1 Repair; modify RACS cooling water lines to new recirc pump cooler

NDT Examination Reports
Data Sheet 50082844/101212; Recirc Pump B Flange Surface; 1-BP-201-PIS
Data Sheet 50082874/160010; VT-3 B Recirc Pump
Data Sheet 105521, Liquid Penetrant Exam, Component ID: 1-BB-1CCA-225-1, instrument line to nozzle
Data Sheet 105522, Liquid Penetrant Exam, Component ID: 1-BB-1CCA-223-1, instrument line to nozzle
Data Sheet 105732, Liquid Penetrant Exam, Component ID: 1-BB-1CCA-220-1, instrument line to nozzle
Data Sheet 105731, Liquid Penetrant Exam, Component ID: 1-BB-1CCA-218-1, instrument line to nozzle
Data Sheet 105523, Liquid Penetrant Exam, Component ID: 1-BB-1CCA-319-FW1, instrument line to nozzle
Data Sheet 250150, Liquid Penetrant Exam, Component ID: 1-CP-206-CSP-W4, Pump Casing Weld
Data Sheet 107565, Liquid Penetrant Exam, Component ID: 1-BB-12VCA-014E-5, Pipe To Safe End
Data Sheet 100145, Ultrasonic Exam, Component ID: RPV1-W20,Head to Flange
Data Sheet 100690, Ultrasonic Exam, Component ID: RPV1-N2KSE,Weld Overlay

Data Sheet 101212, VT-1 Visual Exam of Nuclear Class I Bolting, order 50082874, B Recirculation Pump Drywell; Component ID: RCPB-1BLT,(Flange 16)
Data Sheet 160010, VT-3 Visual Exam of Nuclear Class I Pump/Valve Internal Surfaces; Component ID: BP-201-PIS, Pump Casing, 'B' Recirc Pump
INR HCR13-IVVI-06-03 Steam Dryer
INR HCR13-IVVI-06-04, Revision 1. Steam Dryer Partition Plate
INR HCR13-IVVI-06-04, Steam Dryer Partition Plate
INR HCR13-IVVI-06-05, Steam Dryer Ring
INR HCR13-IVVI-06-06, Steam Dryer Lifting Assembly
INR HCR13-IVVI-06-07 Steam Dryer Lower Guide
INR HCR13-IVVI-06-01 Jet Pump WD-1 Wedge Rev. 1
INR HCR13-IVVI-06-02 Jet Pump AS-1 Gap Measurement
IWE Visual Inspection Report #830900, Component ID PEN-HC-J9
IWE Visual Inspection Report #830400, Component ID PEN-HC-J4
IWE Visual Inspection Report #827500, Component ID PEN-HC-P23
IWE Visual Inspection Report #835400, Component ID PEN-HC-J1351
IWE Visual Inspection Report #830600, Component ID PEN-HC-J6
IWE Visual Inspection Report #827000, Component ID PEN-HC-P18
IWE Visual Inspection Report #829500, Component ID PEN-HC-P39
IWE Visual Inspection Report #830300, Component ID PEN-HC-J3
IWE Visual Inspection Report #829510, Component ID PEN-HC-P13
IWE Visual Inspection Report #835700, Component ID PEN-HC-J1354
IWE Visual Inspection Report #827310, Component ID BLT-HC-P21 Flange Bolting
IWE Visual Inspection Report #827300, Component ID PEN-HC-P21 Blind Flange
IWE Visual Inspection Report #830700, Component ID PEN-HC-J7
IWE Visual Inspection Report #822200, Component ID PEN-HC-W102C
IWE Visual Inspection Report #822500, Component ID PEN-HC-W103B
IWE Visual Inspection Report #823800, Component ID PEN-HC-W105C
IWE Visual Inspection Report #829700, Component ID HCH-HC-C2 Equipment Hatch
IWE Visual Inspection Report #829710, Component ID HCH-HC-C2 Equipment Hatch Bolting
IWE Visual Inspection Report #829720, Component ID ALK-HC-C2 Personnel Airlock
IWE Visual Inspection Report #829730, Component ID BLT-HC-C2 Pers. Airlock Bolting
IWE Visual Inspection Report #821200, Component ID PEN-HC-W100C
IWE Visual Inspection Report #824600, Component ID PEN-HC-W106C
IWE Visual Inspection Report #829800, Component ID HCH-HC-C3 CRD Hatch
IWE Visual Inspection Report #829810, Component ID BLT-HC-C3 CRD Hatch Bolting
IWE Visual Inspection Report #822000, Component ID PEN-HC-W102A
IWE Visual Inspection Report #821000, Component ID PEN-HC-W100A6C
IWE Visual Inspection Report #823500, Component ID PEN-HC-W104K
IWE Visual Inspection Report #824000, Component ID PEN-HC-W105E
IWE Visual Inspection Report #823600, Component ID PEN-HC-W105A
IWE Visual Inspection Report #824100, Component ID PEN-HC-W105F
IWE Visual Inspection Report #824400, Component ID PEN-HC-W106A
IWE Visual Inspection Report #823400, Component ID PEN-HC-W104J
IWE Visual Inspection Report #823300, Component ID PEN-HC-W104H
IWE Visual Inspection Report #826200, Component ID PEN-HC-P7
IWE Visual Inspection Report #823200, Component ID PEN-HC-W104G

IWE Visual Inspection Report #823100, Component ID PEN-HC-W104F
IWE Visual Inspection Report #828100, Component ID PEN-HC-P28A
IWE Visual Inspection Report #834200, Component ID PEN-HC-J42
IWE Visual Inspection Report #825800, Component ID PEN-HC-P6A
IWE Visual Inspection Report #837100, Component ID PEN-HC-P35A CRD Insert
IWE Visual Inspection Report #837500, Component ID PEN-HC-P36A CRD Withdraw
IWE Visual Inspection Report #834300, Component ID PEN-HC-J43
IWE Visual Inspection Report #834100, Component ID PEN-HC-J41
IWE Visual Inspection Report #827400, Component ID PEN-HC-P22
IWE Visual Inspection Report #828300, Component ID PEN-HC-P29
IWE Visual Inspection Report #825400, Component ID PEN-HC-P4A
IWE Visual Inspection Report #826300, Component ID PEN-HC-P8A
IWE Visual Inspection Report #828400, Component ID PEN-HC-P30
IWE Visual Inspection Report #826400, Component ID PEN-HC-P8B
IWE Visual Inspection Report #825600, Component ID PEN-HC-P5A
IWE Visual Inspection Report #821400, Component ID PEN-HC-W101A
IWE Visual Inspection Report #834900, Component ID PEN-HC-J49
IWE Visual Inspection Report #834700, Component ID PEN-HC-J47
IWE Visual Inspection Report #825900, Component ID PEN-HC-P6B
IWE Visual Inspection Report #821500, Component ID PEN-HC-P101B
IWE Visual Inspection Report #837200, Component ID PEN-HC-P35B CRD Insert
IWE Visual Inspection Report #837600, Component ID PEN-HC-P36B CRD Withdraw
IWE Visual Inspection Report #821600, Component ID PEN-HC-W101C
IWE Visual Inspection Report #828500, Component ID PEN-HC-P31
IWE Visual Inspection Report #834800, Component ID PEN-HC-J48
IWE Visual Inspection Report #826700, Component ID PEN-HC-P11
IWE Visual Inspection Report #820100, Component ID VSL-HC-Drywell Head Internal
IWE Visual Inspection Report #820200, Component ID VSL-HC-Drywell Head External
IWE Visual Inspection Report #820300, Component ID BLT-HC-Drywell Head Bolting
IWE Visual Inspection Report #829900, Component ID HCH-HC-C5 Drywell Head Hatch
IWE Visual Inspection Report #829910, Component ID BLT-HC-C5 DW Head Hatch Bltg
IWE Visual Inspection Report #825300, Component ID PEN-HC-P3
IWE Visual Inspection Report #837300, Component ID PEN-HC-P35C CRD Insert
IWE Visual Inspection Report #837700, Component ID PEN-HC-P36C CRD Withdraw
IWE Visual Inspection Report #834600, Component ID PEN-HC-J46
IWE Visual Inspection Report #826000, Component ID PEN-HC-P6C
IWE Visual Inspection Report #829300, Component ID PEN-HC-P38A
IWE Visual Inspection Report #834500, Component ID PEN-HC-J45
IWE Visual Inspection Report #835000, Component ID PEN-HC-J50
IWE Visual Inspection Report #828000, Component ID PEN-HC-P27
IWE Visual Inspection Report #829400, Component ID PEN-HC-P38B
IWE Visual Inspection Report #827700, Component ID PEN-HC-P24B
IWE Visual Inspection Report #828200, Component ID PEN-HC-P28B
IWE Visual Inspection Report #835200, Component ID PEN-HC-J52
IWE Visual Inspection Report #825700, Component ID PEN-HC-P5B
IWE Visual Inspection Report #835100, Component ID PEN-HC-J51
IWE Visual Inspection Report #821700, Component ID PEN-HC-W101D

IWE Visual Inspection Report #834400, Component ID PEN-HC-J44
IWE Visual Inspection Report #825500, Component ID PEN-HC-P4B
IWE Visual Inspection Report #837400, Component ID PEN-HC-P35D CRD Insert
IWE Visual Inspection Report #837800, Component ID PEN-HC-P36D CRD Withdraw
IWE Visual Inspection Report #821800, Component ID PEN-HC-W101E
IWE Visual Inspection Report #826100, Component ID PEN-HC-P6D
IWE Visual Inspection Report #821900, Component ID PEN-HC-W101F
IWE Visual Inspection Report #826800, Component ID PEN-HC-P12
IWE Visual Inspection Report #829520, Component ID PEN-HC-P15
IWE Visual Inspection Report #824700, Component ID PEN-HC-P1A
IWE Visual Inspection Report #825100, Component ID PEN-HC-P2A
IWE Visual Inspection Report #824800, Component ID PEN-HC-P1B
IWE Visual Inspection Report #824900, Component ID PEN-HC-P1C
IWE Visual Inspection Report #825200, Component ID PEN-HC-P2B
IWE Visual Inspection Report #825000, Component ID PEN-HC-P1D
IWE Visual Inspection Report #829600, Component ID PEN-HC-C1 Equipment Hatch
IWE Visual Inspection Report #829610, Component ID BLT-HC-C1 Equip Hatch Bltg
IWE Visual Inspection Report #821100, Component ID PEN-HC-W100B
IWE Visual Inspection Report #823700, Component ID PEN-HC-W105B
IWE Visual Inspection Report #822100, Component ID PEN-HC-W102B
IWE Visual Inspection Report #824200, Component ID PEN-HC-W105G
IWE Visual Inspection Report #824300, Component ID PEN-HC-W105H

Qualification Records
NDE Certificate of Qualification, Michael Hicks, 4/5/06; VT-1, 2, 3 Level II

Engineering Evaluations
Steam Dryer Indications, NUCR 70056479, Operation 10

Miscellaneous
NRC Confirmatory Action Letter No. 1-05-001, 1/11/05
PSEG Ltr. LR-N05-0017, 1/9/05; PSEG Actions In Response To NRC Concerns Regarding ‘B’ Reactor Recirculation Pump Hope Creek Generating Station Docket No. 50-354
PSEG Ltr. LR-N06-0053; Actions To Close CAL 1-05-001 ‘B’ Reactor Recirculation Pump Hope Creek Generating Station Facility Operating License No. NPF-57, Docket No. 50-354
BWR-VIP-139; BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines, EPRI 2005, Prepared by GE Nuclear
GE SIL 644, Revision 1
VTD-327395(001); GENE-0000-0034-9350-R1, Revision 1, 12/04; Evaluation of Steam Dryer Indications Hope Creek Generating Station
VTD-328444(001); GENE-0000-0046-8137-R0, DRF-0000-0043-9289, Revision 0, 12/08; Steam Dryer Support Ring Crack Growth Rate Prediction For Hope Creek Generating Station.
VTD-327419(1); Lisega Calculation ER-VR04-0752, 12/23/04
VTD-327693(001); GENE-0000-0036-1606, Revision 0; Technical Safety Evaluation, Reactor Recirculation Pump 4th Generation Modification, Hope Creek Generating Station, 4/14/05

Section 1R11: Licensed Operator Requalification Program

Procedures
HC.OP-AB.CONT-0002, Primary Containment, Rev 2
HC.OP-EO.ZZ-0101, Reactor Pressure Vessel Control, Rev 10
NC.EP-EP.ZZ-0404, Protective Action Recommendations (PARS) Upgrades, Rev 2
HC.OP-AB.COMP-0002, Primary Containment Instrument Gas, Rev 4

Corrective Action Notifications

20287438	20287762

Other Documents
Simulator Scenario Guide SG-263, Reactor Recirc Pump Trip / RWCU Leak / PCIG Leak / Failure of RPS, Rev 3
NC.TQ-WB.ZZ-0003, Attachment 1, Crew Competency Summary Sheet, Rev 6
NC.EP-DG.ZZ-0001, Form 1, DEP Observation Checklist, Rev 6
NC.TQ-AS.ZZ-1003, Attachment 1, Management Observation of Training (MOT) Form, Rev 3

Section 1R12: Maintenance Effectiveness

Procedures
HC.OP-SO.PN-0001, 120 VAC Electrical Distribution, Rev. 16
HC.MD-CM.PN-0001, 20 KVA Inverter Troubleshooting and Repair, Rev. 9
HC.OP-AP.ZZ-0108, Operability Assessment and Equipment Control Program, Rev. 18
HC.OP-AB.ZZ-0136, Loss of 120 VAC Inverter, Rev. 9
HC.OP-AP.ZZ-0031, Control of Alarm Bypass, Rev. 0

Drawings
E-0006-1, 4.16 KV Class 1E Power System, Rev. 11
E-0012-1, 120V AC Instrumentation & Misc. Systems, Rev. 7
E-0018-1, 480 Volt Class 1E Unit Substa. 10B410, 10B420, 10B430, 10B440, 10B450, 10B460, 10B470, 10B480, Rev. 16
E-0019-1, 480 Volt MCC Tabulation Class 1E - Aux Bldg - D/G Area 10B411, 10B421, 10B431, 10B441, Rev. 12
E-0020-1, 480 Volt MCC Tabulation Class 1E - Aux Bldg - D/G Area 10B451, 10B461, 10B471, 10B481 Rev. 14

Corrective Action Notifications

20281682	20278760	20274462	20272434	20277344	20284475
20283474	20278666	20210237	20276416	20279751	20284765
20277990	20287820	20255644	20277183	20282241	

Orders

60062683	60061918	50081764	60061789	70054410	70056832
90002327	70056104	60061044	70042626	70055566	

Other Documents
NRC Regulatory Guide 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2
NUMARC 93-01, Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2
Plant Health Committee System Presentation for Emergency Diesel Generators - 1Q 2006
Emergency Diesel Generator System Health Report 1Q 2006
Emergency Diesel Generator Maintenance Rule Reliability and Unavailability logs
OTDM HC-2006-0009 Emergency Diesel Generator (EDG) Lube Oil Keep Warm Pump

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
HC.OP-AP.ZZ-0108, Rev. 0, On-Line Risk Assessment
HC.FP-ST.KC-0009, Rev. 14, Diesel Driven Fire Pump Operability Test
HC.MD-PM.KC-0001, Rev. 5, Diesel Fire Pump And Diesel Engine P.M.
HC.MD-CM.EA-0003, Rev. 25, Service Water Strainer Overhaul & Repair
HC.MD-CM.EA-0003, Rev. 27, Service Water Strainer Overhaul & Repair
HC.MD-PM.EA-0001, Rev. 18, Service Water Strainer - Clean and Inspect
HC.MD-PM.EA-0001, Rev. 19, Service Water Strainer - Clean and Inspect
HC.MD-PM.EA-0001, Rev. 20, Service Water Strainer - Clean and Inspect
HC.MD-PM.EA-0001, Rev. 21, Service Water Strainer - Clean and Inspect
NC.WM-AP.ZZ-0000, Rev. 10, Notification Process
NC.WM-AP.ZZ-0000, Rev. 13, Notification Process
NC.WM-AP.ZZ-0002, Rev. 8, Corrective Action Process
NC.WM-AP.ZZ-0002, Rev. 9, Corrective Action Process
NC.CC-AP.ZZ-0080, Rev. 19, Engineering Change Process
NC.CC-AP.ZZ-0081, Rev. 8, Engineering Change Implementation and Test Process
HC.OP-AP.ZZ-0108, Rev. 23, Operability Assessment and Equipment Control Program
HC.OP-AP.ZZ-0108, Rev. 24, Operability Assessment and Equipment Control Program
HC.OP-AP.ZZ-0108, Rev. 25, Operability Assessment and Equipment Control Program
LS-AA-120, Rev. 5, Issue Identification and Screening Process
HC.OP-AR.ZZ-0001, Rev. 16, Overhead Annunciator Window Box A1
HC.OP-AP.ZZ-0109, Rev. 14, Equipment Operational Control
HC.OP-AB.COOL-0001, Rev. 4, Station Service Water
HC.OP-AB.COOL-0001, Rev. 8, Station Service Water

Corrective Action Notifications

20287503	20204814	20206474	20280569	20283448	20286362
20178691	20204953	20212968	20280959	20284983	20289628
20178953	20206335	20213174	20281813	20284984	

Orders

30139954	60038730	60062268	70041614	70057117	80076763
30059866	60043083	60062304	70041902	70058063	80079630
30097091	60048403	60062305	70043287	80032450	80089544
30124011	60050019	70037109	70056583	80068087	
30130802	60050123	70037181	70056729	80076640	

Other Documents
SE.MR.HC.02, System Function Level Maintenance Rule VS Risk Reference
HCGS PSA Risk Evaluation Forms for Work Week Nos. XX to XX
NRC Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants
NUMARC 93-01, Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Section 11- Assessment of Risk Resulting from Performance of Maintenance Activities, dated February 11, 2000
PSA white paper on risk associated with operator actions for a loss of control room chilled water
Hope Creek Shutdown Safety Status, 04/17/2006 and 04/18/2006
PSEG River Grass Data Spreadsheet
E-18 Calculation, Rev. 1 and 2, Selection of Overload Heaters For AC MOVs and Continuous Duty Motors
M-076, Rev. 8, Design Specification for Service Water Self-Cleaning Strainers
DE-CB.EA/EP/EQ-0052, Rev. 2, Configuration Baseline Documentation for Station Service Water System
Hope Creek Maintenance Rule Computer System Service Water Information

Section 1R14: Operator Performance During Non-routine Evolutions and Events

Procedures
HC.OP-SO.AE-0001, Rev. 44, Feedwater System Operation
HC.OP-IO.ZZ-0004, Rev. 66, Shutdown From Rated Power to Cold Shutdown
HC.OP-ST.BB-0001, Rev. 34, Recirculation Jet Pump Operability - Daily
HC.ER-AP.BB-0002, Rev. 7, Hope creek Reactor Recirculation Pump Piping Vibration Monitoring

Corrective Action Notifications
20251352 20282041 20278329 20287950 20287553 20287559

Orders
70056063 70050125

Other Documents
IPTE 06-002, Rev 0,1,2 dated June 2006. Titled - "RFO-13 Post "B" Reactor Recirculation Pump Replacement and Piping Support Modification Vibration Evaluation at Core Flows greater than 100 Mlb/hr."
Recirculation Pump Vibration Monitoring Just-In-Time training package
Feedwater Control System Licensed Operator Lesson Plan
H-1-BB-MDC-4000, Reactor Recirculation System Acoustic Model, dated December 29, 2005
GENE-0000-0035-6906, Rev. 4, Recirculation & RHR Piping Start Up Test Criteria
Results of Reactor Recirculation Acoustic Vibration Testing - June 2006
Plant Historian plots of recirculation pump flow and reactor power vs time April 6, 2006
Prompt Investigation for Recirc runback during reactor shutdown on April 6, 2006

Section 1R15: Operability Evaluations

Procedures
HC.MD-CM.EA-0003, Rev. 25, Service Water Strainer Overhaul & Repair
HC.MD-CM.EA-0003, Rev. 27, Service Water Strainer Overhaul & Repair
HC.MD-PM.EA-0001, Rev. 18, Service Water Strainer - Clean and Inspect
HC.MD-PM.EA-0001, Rev. 19, Service Water Strainer - Clean and Inspect
HC.MD-PM.EA-0001, Rev. 20, Service Water Strainer - Clean and Inspect
HC.MD-PM.EA-0001, Rev. 21, Service Water Strainer - Clean and Inspect
NC.WM-AP.ZZ-0000, Rev. 10, Notification Process
NC.WM-AP.ZZ-0000, Rev. 13, Notification Process
NC.WM-AP.ZZ-0002, Rev. 8, Corrective Action Process
NC.WM-AP.ZZ-0002, Rev. 9, Corrective Action Process
NC.CC-AP.ZZ-0080, Rev. 19, Engineering Change Process
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HC.OP-AP.ZZ-0108, Rev. 23, Operability Assessment and Equipment Control Program
HC.OP-AP.ZZ-0108, Rev. 24, Operability Assessment and Equipment Control Program
HC.OP-AP.ZZ-0108, Rev. 25, Operability Assessment and Equipment Control Program
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HC.OP-AP.ZZ-0109, Rev. 14, Equipment Operational Control
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HC.OP-ST.GK-0002, Rev. 6, Control Room Emergency Filtration System Isolation /Actuation Functional Test - 18 Months
HC.IC-FT.GK-0001, Rev. 6, Control Room Emergency Filtration System Flow Measurements
HC.MD-CM.KJ-0004, Rev. 11, Diesel Generator Lubrication System Maintenance and Repair
HC.IC-FT.PE-0008, Rev. 5, Time Interval Test Emergency Load Sequencer System Diesel Generator D, 1DC428
HC.OP-ST.KJ-0008, Rev. 33, Integrated Emergency Diesel Generator 1DG400 Test - 18 Months
MA-AA-716-004, Rev. 4, Conduct of Troubleshooting
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HC.RE-ST.SE-0003, Rev. 19, LPRM Calibration Surveillance
HC.OP-IS.BE-0001(Q), Rev. 36, A & C Core Spray Pumps-AP206 and CP206 - In-Service Test
HC.OP-IS.BE-0101(Q), Rev. 23, Core Spray Subsystem A Valves - Inservice Test
MA-AB-772-301, Rev. 1, Procedure for Dresden and Quad Cities Recirculating MG Set Voltage Regulator Tuning
HC.IC-SC.BJ-0011, Rev.8, HPCI - Division 1 Channel L-4805-1 Suppression Chamber Water Level

Drawings
PN1-B31-1030-0024, Reactor recirculation pump and MG set schematic
PN1-B31-S0001-0120, General Electric Recirc MG set voltage regulator schematic
M-55-1, High Pressure Coolant Injection

Corrective Action Notifications

20288523	20178953	20280569	20289628	20278068	20282911
20288483	20204814	20280959	20278147	20278533	20284680
20288035	20204953	20281813	20277825	20278666	20285325
20286560	20206335	20283448	20274462	20278760	20279434
20280701	20206474	20284983	20276142	20278899	20278996
20285752	20212968	20284984	20277423	20281682	20278850
20178691	20213174	20286362	20277891	20282499	20283884

Orders

80089685	60048403	70041614	80068087	30084482	70055925
60063505	60050019	70041902	80076640	60061905	70056104
30059866	60050123	70043287	80076763	60061918	70056913
30097091	60062268	70056583	80079630	60061965	90002327
30124011	60062304	70056729	80089544	60062341	60062087
30130802	60062305	70057117	50080604	60062683	70057358
60038730	70037109	70058063	60061841	60062957	
60043083	70037181	80032450	70055864	70054786	

Other Documents

Hope Creek Inservice Testing Program Basis Data Sheets for 1BEV-028
E.Q. Maintenance and Surveillance Information Sheet for Reactor Core Spray
NRC Inspection Manual Part 9900: Technical Guidance Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming conditions Adverse to Quality or Safety
SC-BJ-0008-3, HPCI Suppression Chamber Level Low
SC-BJ-0004, Setpoint Calculation - HPCI (Suppression Pool Level High)
Hope Creek RF13 Refuel Outage CRB 26-59, engineering white paper on blistered control rod 26-59
OTDM HC-2006-0008, Disposition of blistered control rod blade 26-59
Hope Creek Shutdown Safety Status, 04/17/2006 and 04/18/2006
PSEG River Grass Data Spreadsheet
E-18 Calculation, Rev. 1 and 2, Selection of Overload Heaters For AC MOVs and Continuous Duty Motors
M-076, Rev. 8, Design Specification for Service Water Self-Cleaning Strainers
DE-CB.EA/EP/EQ-0052, Rev. 2, Configuration Baseline Documentation for Station Service Water System
Hope Creek Maintenance Rule Computer System Service Water Information
Hope Creek Reactor Engineering Startup Reactivity Plan to Maximum Power Based on Feed Pump Configuration, 05/10/2006

Section 1R17: Permanent Plant Modifications

Procedures

HC.ER-AP.BB-0001(Q), Rev. 6, HC Reactor Recirculation Pumps Vibration Monitoring
HC.OP-SO.BB-0002(Q), Rev. 59, Reactor Recirculation System Operation

HC.OP-DL.ZZ-0004(Q), Rev. 31, Log 4 Reactor Building Log

Other Documents
ECN 80076232, Rev. 5, HC “B” Recirc Pump Internals Replacement
ED-221-1, Rev. 0IR0, Drywell Building RACS Water To Pump BP-201
SC-ED-0503, Rev.1IR0, Loop Tolerance Calculations For 1ED FISL N004A&B
GE-NE-0000-0036-1608, Rev. 0, Seal Purge Setting Procedure Reactor Recirculation Pump 4th Generation Modification
M:\Shared\HC Reactor Recirc Vibrations\Cycle 14 Startup Data, Rev. 0, Reactor Recirculation Critical Speed Review Cycle 14 Startup Review

Section 1R19: Post-Maintenance Testing

Procedures
HC.MD-ST.ZZ-0009(Q), Rev. 17, Motor Operated Valve Thermal Overload Protection Surveillance
HC.OP-IS.BC-0103(Q), Rev. 22, Residual Heat Removal Subsystem C Valves - Inservice Test
NC.NA-TS.ZZ-0050, Maintenance Testing Program Matrix
NC.NA-AP.ZZ-0050(Q), Station Post Maintenance Testing, Rev. 7
HC.MD-CM.EA-0003(Q), Service Water Strainer Overhaul & Repair, Rev. 27
HC.OP-IS.EA-0002(Q), B Service Water Pump-BP502 - Inservice Test, Rev. 44
HC.OP-ST.GK-0003(Q), B - Control Room Emergency Filtration System Functional Test - Monthly, Rev. 5
HC.OP-IS.BE-0001(Q), Rev. 36, A & C Core Spray Pumps-AP206 and CP206 - In-Service Test
HC.OP-IS.BE-0101(Q), Rev. 23, Core Spray Subsytem A Valves - Inservice Test
HC.MD-CM.EA-0002(Q), Rev. 17, Service Water Pump Overhaul Repair
HC.MD-CM.EA-0001(Q), Rev. 22, Service Water Pump & Motor Removal & Replacement
HC.MD-PM.EA-0002(Q), Rev. 13, Service Water Intake Bay Silt Survey and Silt Removal
HC.OP-IS.EA-0001(Q), Rev. 39, A Service Water Pump - AP502 - Inservice Test
HC.MD-CM.KJ-0004, Rev. 11, Diesel Generator Lubrication System Maintenance and Repair

Drawings
E-6231-0 sheet 10, Electrical schematic diagram residual heat removal system RHR pump min flow bypass valves

Corrective Action Notifications

20287404	20234790	20218671	20280019	20277423	20278899
20287060	20231157	20243514	20280080	20277891	20281682
20257550	20229555	20288523	20280229	20278068	20282499
20255075	20227580	20288523	20279864	20278533	20282911
20254827	20287588	20288483	20282499	20278666	20284680
20254760	20287606	20288381	20274462	20278760	20285325
20254634	20287803	20288333	20276142		

Orders

50078803	80089685	30118573	30084482	60062341	70055925
60058580	60063505	80089186	60061905	60062683	70056104
60061526	60063201	70056913	60061918	60062957	70056913
50096174	80076232	60062341	60061965	70054786	90002327
60063300					

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Hope Creek Inservice Testing Program Basis Data Sheets for 1BEV-028
E.Q. Maintenance and Surveillance Information Sheet for Reactor Core Spray
Reactor Recirculation Pump Plant Data dated April 29, 2006
VTD PN1-A41-8010-0052 GE Refueling Platform
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Section 1R20: Refueling and Outage Activities

Procedures
NC.NA-AP.ZZ-0055, Outage Management Program
NC.OM-AP.ZZ-0001, Outage Risk Assessment
HC.OP-IO.ZZ-0002, Rev. 44, Preparation for Plant Startup
HC.OP-IO.ZZ-0003, Rev. 74, Startup From Cold Shutdown to Rated Power
HC.OP-IO.ZZ-0004, Rev. 66, Shutdown From Rated Power to Cold Shutdown
HC.OP-AB.RPV-0009, Shutdown Cooling
HC.MD-FR.KE-0036, Rev. 11, Reactor Pressure Vessel Assembly
HC.OP-GP.ZZ-0002, Rev. 12, Primary Containment Closeout
HC.RE-FR.ZZ-0001, Rev. 28, Fuel Handling Controls
HC.RE-FR.ZZ-0014, Rev. 7, New Fuel Inspection, Channeling, and Storage
HC.OP-AB.RPV-0009, Rev. 5, Shutdown Cooling
HC.OP-SO.BC-0002, Rev. 18, Decay Heat Removal Operation
HC.OP-IO.ZZ-0001, Rev. 18, Refueling to Cold Shutdown
HC.ER-AP.BB-0001, Rev. 5, Hope Creek Reactor Recirculation Pumps Vibration Monitoring
HC.OP-SP.BF-0001, Rev. 6, Control Rod Drive Mechanism / Blade Simultaneous Removal
HC.RA-IS.ZZ-0010, Rev. 13, Containment Isolation Valve Type C Leak Rate Test
HC.RA-IS.ZZ-0017, Rev. 5, Reactor Coolant System Pressure Isolation Valves Seat Leakage Measurement/Test

Drawings
M-53-1, Sheet 1, Rev. 29, Fuel Pool Cooling & Torus Water Cleanup

Corrective Action Notifications

20278400	20280952	20283892	20278837	20279813	20281129
20278441	20281797	20282550	20281583	20280986	20281330
20282029	20283537	20278486	20276570	20280990	20281397
20282517	20284446	20278447	20283057	20281127	20281439
20280760					

Orders

70057641	80088863	60062347	30115511	50080759	70056836
80087044	60062346	70056677	30115756		

Other Documents
Hope Creek RF13 Outage Risk Assessment Report
NRC Information Notice 2002-26, Failure of Steam Dryer Cover Plate After a Recent Power Uprate
GENE-0000-0053-6264-00-R1, MSIV Body to Bonnet Replacement Stud Use "As-Is" Evaluation
Hope Creek Work Clearance Documents 4154067 and 4165359
Hope Creek Refuel Outage System Preparation Documents
Hope Creek Refuel Outage 13 Turnover Log - ISI/CISI/SPT/Snubbers/Supports

Section 1R22: Surveillance Testing

Procedures
HC.CH-RC.ZZ-0002 Rev. 17, Gross Beta and Tritium by Liquid Scintillation
HC.OP-DL.ZZ-0026 Rev. 104, Surveillance Log
HC.IC-FT.SK-0016, Rev. 17, Radiation Monitoring - Channel D Monitor H1SK-1SKLY-4930 Drywell Leak Detection Sump Monitoring System (DLD-SMS)
HC.RA-IS.ZZ-0010, Rev. 13, Containment Isolation Valve Type C Leak Rate Test
HC.RA-IS.ZZ-0017, Rev. 5, Reactor Coolant System Pressure Isolation Valves Seat Leakage Measurement/Test
HC.OP-ST.KJ-0006, Rev. 31, Integrated Emergency Diesel Generator 1BG400 Test - 18 Months

Corrective Action Notifications

20253347	20252790	20279813	20281127	20281397	20279218
20276310	20287588	20280986	20281129	20281439	20280967
20237445	20289565	20280990	20281330	20279434	

Orders

50094668	60062346	30115756	30062264	50080821	50081843
50081260	60062347	50080759	50080759	50081847	30115511
50082713	70056677	50082684	50080634	30015756	50080664
50080759	30115511	60062152	50081807	50081885	50080675

Section 1R23: Temporary Plant Modifications

Procedures
SH.MD-AP.ZZ-0002, Rev. 9, Maintenance Department Troubleshooting and Repair
MA-AA-716-004, Rev. 4, Conduct of Troubleshooting
MA-HC-716-004-001, Rev. 0, Conduct of Troubleshooting
SH.OP-AP.ZZ-0108, Rev. 22, Operability Assessment and Equipment Control Program
NC.NA-AP.ZZ-0008, Rev. 20, Configuration Control Program
NC.DE-AP.ZZ-0030, Rev. 5, Control of Temporary Modifications
HC.MD-ST.KF-0001, Rev. 12, Polar Crane Periodic Inspection

NC.WM-AP.ZZ-0003, Rev. 5, Regular Maintenance Process
HC.DE-AP.ZZ-0060, Rev. 0, Functional Classification Methodology For Component Data Module Functional Location Within SAP/R3 For Hope Creek Generating Station

Drawings
PM063Q-0065, Sheet 2, Rev. 0, Polar Crane Schematic Diagram - Main Hoist Power

Corrective Action Notifications
20278656 20275285 20278656 20285199 20286440 20278542

Orders
70056229 30118573

Other Documents
ASME B30.2-2005, Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)
Refuel Floor Activity Logs

Section 1EP6: Drill Evaluation

Procedures
HC.OP-AB.CONT-0002(Q), Primary Containment, Rev. 2
HC.OP-EO.ZZ-0101(Q), Reactor Pressure Vessel Control, Rev. 10
NC.EP-EP.ZZ-0404(Q), Protective Action Recommendations (PARS) Upgrades, Rev. 2
HC.OP-AB.COMP-0002(Q), Primary Containment Instrument Gas, Rev 4

Corrective Action Notifications
20287438 20287762

Other Documents
Simulator Scenario Guide SG-263, Reactor Recirc Pump Trip / RWCU Leak / PCIG Leak / Failure of RPS, Rev 3

Section 2OS1: Access Control to Radiologically Significant Areas

Corrective Action Notifications
20279873

Other Documents
Radiation Work Permit #11
Shielding package #2006-036

Section 2OS2: ALARA Planning and Controls

ALARA Reviews: 2006-082; 2006-141

Section 4OA1: Performance Indicator Verification

Procedures
HC.CH-DG.PI-0001, Rev. 0, Hope Creek Chemistry Desk Top Guide NRC Performance Indicator Status Determination
HC.CH-TI.ZZ-0012, Rev. 50, Chemistry Sampling Frequencies, Specifications, and Surveillances
HC.OP-DL.ZZ-0026, Rev. 104, Surveillance Log
LS-AA-2090, Rev. 4, Monthly Data Elements for NRC Reactor Coolant System (RCS) Specific Activity
LS-AA-2001, Rev. 4, Collecting and Reporting of NRC Performance Indicator Data
LS-AA-2100, Rev. 5, Monthly Data Elements for NRC Reactor Coolant System (RCS) Leakage
LS-AA-2010, Rev. 4, Monthly Data Elements for NRC/WANO Unit/Reactor Shutdown Occurrences
LS-AA-2030, Rev. 4, Monthly Data Elements for NRC Unplanned Power Changes per 7000 Critical Hours

Corrective Action Notifications

20200936	20200551	20237921	20245877	20255951	20200553

Other Documents
Monthly Operating Reports for the Months of October 2004 through January 2006
LER 050000354/2004010, Manual Reactor Scram Due to Moisture Separator Dump Line Failure
LER 050000354/2005002, Through-Wall Leak on ‘B’ Reactor Recirculation System Decontamination Port
LER 050000354/2005003, Reactor Coolant System Leak from Check Valve Position Indicator
LER 050000354/2005008, Technical Specification Shutdown due to ‘B’ Suppression Chamber to Drywell Vacuum Breaker Not Closed

Section 4OA2: Identification and Resolution of Problems

Procedures
HU-AA-1081, Rev. 0, Fundamentals Tool Kit
HU-AA-101, Rev. 3, Human Performance Tools and Verification Practices
HU-AA-102, Rev. 1, Technical Human Performance Practices
HU-AA-1212, Rev. 1, Technical Task Risk/Rigor Assessment, Pre-Job Brief, Independent Third Party Review, and Post-Job Brief
HU-AA-1101, Rev. 1, Change Management
HU-AA-104-101, Rev. 1, Procedure Use and Adherence
NC.NA-AP.ZZ-0089, Rev. 0, Reactivity Management
OP-AA-300-1540, Rev. 3, Reactivity Management Administration

Corrective Action Notifications

20265044	20256302	20181292	20203675	20217767	20240045
20216735	20263496	20193148	20214921	20220180	20265806
20235815	20186751	20199679	20214923	20201993	20201991
20250244	20174947	20199983	20215389	20220400	20282581
20233706	20175121	20199984	20215390	20258640	20281397
20261768	20175996	20200795	20216170	20269559	20285708
20259571	20180888	20200796	20216600	20226240	20285326
20260710					

Orders

70038773	70043521	70043522	70051281

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PSEG Metrics for Improving the Work Environment, Salem and Hope Creek Generating Stations, Quartely Report, April 28, 2006
UFSAR Section 9.2, Water Systems
Hope Creek Service Water Deicing System presentation to Plant Health Committee dated June 22, 2006

Section 4OA5: Other Activities

Procedures

HC.OP-AB.BOP-0004(Q), Rev. 11, Grid Disturbances
NC.WM-AP.ZZ-0001(Q), Rev. 12, Work Management Process
NC-CC-DG.ZZ-0003(Z), Rev. 3, PRA Weekly Risk Assessment (a)(4) Desktop Guide
SH.OP-AP.ZZ-0027(Q), Rev. 9, On-Line Risk Assessment
SH.OP-DD.ZZ-0001(Z), Rev. 3, Electric System Emergency Operations and Electric System Operator Interface
WC-AA-101, Rev. 11, On-Line Work Control Process
HC.OP-ST.BC-0009, Rev. 5, Residual Heat Removal System RHR heat Exchanger Flow Measurement - 18 Month

Corrective Action Notifications

20283988	20281777

Orders

50093314	50082713	50081260	70054151

LIST OF ACRONYMS

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CCA	Common Cause Analysis
CFR	Code of Federal Regulations
CM	Corrective Maintenance
DCP	Design Change Package
EDG	Emergency Diesel Generator
HCGS	Hope Creek Generating Station
HPCI	High Pressure Coolant Injection
HX	Heat Exchanger
INPO	Institute of Nuclear Power Operations
ISI	Inservice Inspection
LOP/LOCA	Loss of Power/Loss of Coolant Accident
MR	Maintenance Rule
MSIV	Main Steam Isolation Valve
NCV	Non-cited Violation
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PI	Performance Indicator
PM	Preventive Maintenance
PSEG	Public Service Enterprise Group Nuclear LLC
RHR	Residual Heat Removal
rpm	Revolutions Per Minute
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SACS	Safety Auxiliary Cooling System
SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
SJAE	Steam Jet Air Ejector
STACS	Safety and Turbine Auxiliary Cooling System
TI	Temporary Instruction
TOL	Thermal Overload
TS	Technical Specificaion
UFSAR	Updated Final Safety Analysis Report