January 26, 2006

EA-05-229
EA-06-001

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

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000354/2005005 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Levis:

On December 31, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at the Hope Creek Nuclear Generating Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 9, 2006, with Mr. Massaro 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 of very low safety significance (Green). The finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the NCV, 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 Hope Creek.

Additionally, the inspectors reviewed the events relating to the discovery of reactor coolant pressure boundary leakage (self-revealing) in the 'B' reactor recirculation decontamination port, and, on a separate occasion, the FO50A residual heat removal check valve position indicator. Although these issues constitute violations of NRC requirements, in that any reactor coolant system boundary leakage at power constitutes a violation, the NRC concluded that the degraded conditions were not avoidable by reasonable quality assurance measures or management controls, and thus, no performance deficiency was identified for either case. Based on these facts, I have been authorized, after consultation with the Director, Office of

Enforcement, and the Regional Administrator, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement for these violations. Such discretion is routinely allowed for cases like this. A regional Senior Risk Analyst evaluated the risk associated with each event and determined that both events were of very low safety significance.

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/ David Lew signed for

Brian E. Holian, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000354/2005005
w/Attachment: Supplemental Information

cc w/encl:
G. Barnes, Site Vice President
D. Winchester, Vice President - Nuclear Assessments
W. F. Sperry, Director - Finance
D. Benyak, Director - Regulatory Assurance
M. Massaro, Hope Creek Plant Manager
J. J. Keenan, Esquire
M. Wetterhahn, Esquire
Consumer Advocate, Office of Consumer Advocate
F. Pompper, Chief of Police and Emergency Management Coordinator
P. Baldauf, Assistant Director, Radiation Protection and Release Prevention, State of New Jersey
K. Tosch, Chief, Bureau of Nuclear Engineering, NJ Dept. of Environmental Protection
H. Otto, Ph.D., DNREC Division of Water Resources, State of Delaware
N. Cohen, Coordinator - Unplug Salem Campaign
W. Costanzo, Technical Advisor - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

Distribution w/encl:
S. Collins, RA
M. Dapas, DRA
B. Holian, DRP
M. Gray, DRP
B. Welling, DRP
G. Malone, DRP, Senior Resident Inspector
K. Venuto, DRP, Resident OA
S. Lee, RI OEDO
R. Shane, OCA
D. Roberts, NRR
S. Bailey, PM, NRR
R. Urban, ORA
D. Screnci, OPA
N. Sheehan, OPA
J. White, DRS
D. Tifft, DRP
Region I Docket Room (with concurrences)
ROPreports@nrc.gov

DOCUMENT NAME: E:\Filenet\ML060270171.wpd

SISP Review Complete: <u>MKG</u> **(Reviewer's Initials)**
After declaring this document "An Official Agency Record" it **will/~~will not~~** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP	RI/DRP	RI/ORA	RI/DRS	RI/DRP
NAME	GMalone/BDW	BWelling	RUrban	WSchmidt	MGray
DATE	01/24/06	01/24/06	01/24/06	01/24/06	01/25/06

OFFICE	RI/DRS	RI/DRP	
NAME	JWhite	Holian/DCL	
DATE	01/24/06	01/25/06	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	050000354
License No:	NPF-57
Report No:	05000354/2005005
Licensee:	Public Service Enterprise Group Nuclear LLC
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	October 1 - December 31, 2005
Inspectors:	M. Gray, Senior Resident Inspector G. Malone, Senior Resident Inspector J. Schoppy, Senior Reactor Inspector J. Wiebe, Project Engineer J. Furia, Senior Health Physicist J. Josey, Reactor Inspector M. Snell, Reactor Engineer A. DeFrancisco, Reactor Inspector T. Fish, Sr., Operations Engineer
Approved By:	Mel K. Gray, Acting Chief Projects Branch 3 Division of Reactor Projects

Enclosure 1

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000354/2005005; 10/01/2005 - 12/31/2005; Hope Creek Generating Station; Maintenance Effectiveness.

The report covered a 13-week period of inspection by resident inspectors, and an announced inspection by a regional radiation specialist and reactor inspectors. One Green non-cited violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, 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: Barrier Integrity

C Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," in that work performed in April 2000 and October 2001 for the 'A' through 'H' suppression pool to drywell vacuum breakers did not include instructions with appropriate acceptance criteria. The licensee entered the deficiency into their corrective action program, performed an extent of condition review on the remaining seven suppression pool to drywell vacuum breakers, enhanced maintenance procedures, and provided training to maintenance technicians on testing and overhaul of these valves.

This finding was more than minor because the performance deficiency was associated with the procedure quality attribute of the containment barrier integrity cornerstone and affected the cornerstone's objective to provide reasonable assurance that physical design barriers protect the public from radionuclide release. Specifically, vacuum breaker sub-components were not replaced or refurbished in intervals evaluated and specified in the mechanical equipment qualification program. The inspectors completed a Phase 1 screening using Appendix A of Inspection Manual Chapter 0609, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," and determined the safety significance of the issue was of very low risk (Green) because it did not represent an actual open pathway in the physical integrity of reactor containment or result in an actual reduction in defense-in-depth for the atmospheric control or hydrogen control of the reactor containment. (Section 1R12)

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

On October 5, 2005, operators reduced power to 95% to remove the 6A feedwater heater from service due to concerns associated with vibration on the 6A feedwater heater drain line and associated piping. Operators increased reactor power to approximately 98% after the feedwater heater was removed from service. Hope Creek returned to 100% power on October 7, 2005, after evaluation of the condition. Hope Creek reduced power to approximately 70% on October 7, 2005, for a planned downpower for control rod adjustments and maintenance work, and increased power to 100% on October 9, 2005. Power was reduced the same day to approximately 94% to remove the 6A feedwater heater from service for further review of feedwater pipe vibration issues. Power was increased the same day to approximately 98% until December 7, 2005, when power was raised to 100%. Reactor power was 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 reviewed the scope of PSEG's cold weather preparations to verify they adequately prepared equipment to operate reliably in freezing conditions. Specifically, inspectors performed a detailed review of PSEG's adverse weather procedures for seasonal extremes, interviewed engineering and operations personnel, and walked down portions of the service water, condensate storage, and fire protection systems that can be impacted by cold temperatures. The inspectors verified that heat tracing and insulation used to protect these systems were functional and that system conditions were adequate to support operation in cold weather. The documents reviewed during this inspection are listed in the attachment. This inspection satisfied one inspection sample for the onset of cold weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown (2 samples)

a. Inspection Scope

The inspectors performed a partial walkdown of the following two systems to verify the operability of redundant or diverse trains and components when safety equipment was inoperable. The inspectors focused their review on potential discrepancies that could impact the function of the system, and therefore, potentially increase risk. 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.

- ‘B’ residual heat removal (RHR) train with ‘A’ RHR pump out-of-service for maintenance on November 7, 2005; and
- ‘B’ Technical Support Center Chilled Water System on December 5, 2005.

.2 Complete Walkdown (IP 71111.04S - 1 sample)

a. Inspection Scope

The inspectors conducted one complete walkdown of accessible portions of the 'A' train of the RHR system. The inspectors used station procedures and drawings to verify proper system lineup for existing plant conditions. The inspectors examined pipe hangers and other component support devices to verify there were not signs of degradation or previous waterhammer conditions. The inspectors also examined RHR and jockey pumps for oil and water leaks to determine whether these components were maintained capable of performing their safety function. The inspectors reviewed the corrective action database to verify that RHR alignment problems were being identified and appropriately resolved. Documents reviewed during the inspection are listed in the attachment.

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 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 properly controlled; fire detection and suppression equipment were available for use; and compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with PSEG’s fire protection plan. The inspectors reviewed Hope Creek’s Individual Plant Examination for External Events (IPEEE) for risk insights and design features credited in these areas. The inspectors reviewed notifications documenting fire protection deficiencies to verify identified problems were being evaluated and appropriately addressed. Documents reviewed are listed in the attachment.

- Reactor core isolation cooling pump room on October 12, 2005;
- ‘A’ and ‘C’ Service Water Bay on October 13, 2005;
- Control equipment room mezzanine on October 20, 2005;
- Cable spreading room on October 26, 2005;
- ‘A’ Control Area HVAC Equipment Room on November 3, 2005;
- ‘B’ Control Area HVAC Equipment Room on November 3, 2005;
- Control and Diesel Generator Building HVAC Equipment Rooms on November 3, 2005;
- Electrical Access Area Room on November 9, 2005; and
- Accessible turbine building rooms containing offsite power source bus ducts to safety-related 4kV busses (rooms 1315, 1316, and 1317) on October 25, 2005.

b. Findings

No findings of significance were identified.

.2 Fire Protection - Drill Observation

a. Inspection Scope (1 sample)

The inspectors observed a fire drill on October 26, 2005, to assess readiness of PSEG’s fire brigade to respond to fires. The drill scenario involved a fire in a 480 VAC bus in the ‘B’ Class 1E switchgear room. The inspectors observed the drill to verify that fire personnel responded in the time frame required with proper turnout gear and equipment, that protective equipment and self-contained breathing apparatus were brought to the scene and utilized, that fire fighting techniques suitable for an electrical fire were employed, that effective command and control was established, and that there was an effective interface with the operator fire liaison. The inspectors also observed the post-drill critique to determine whether the brigade was appropriately self-critical and that deficiencies were identified and entered into the corrective action program at the proper level. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope (1 sample)

The inspectors reviewed selected risk-important plant design features and station procedures to protect the plant from internal flood events. The inspectors focused on mitigation strategies and equipment in the high pressure coolant injection pump room. The inspectors reviewed the UFSAR, flooding analyses, engineering calculations and abnormal alarm response procedures identify mitigation strategies, vulnerabilities, and licensee commitments. The inspectors observed the condition of watertight doors, wall penetrations, flood alarm switches and drains to assess their readiness to contain flow from an internal flood in accordance with the design basis. Work orders to periodically exercise floor drain check valves and flood alarm indications were also reviewed. The inspectors reviewed PSEG's corrective action program database to verify internal flooding issues were identified at a proper threshold and corrective actions for deficiencies were adequate. Documents associated with these reviews are listed in the attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

.1 Resident Inspector Quarterly Review

a. Inspection Scope (1 sample)

On October 17, 2005, the inspectors observed one simulator training scenario to assess operator performance, training effectiveness, and evaluator's critique of the operating crew. The scenario involved an earthquake which caused damage to instrument tubing in the drywell and an inadvertent 'D' channel emergency core cooling system initiation and primary containment isolation. The scenario was complicated with a large-break loss of coolant accident. The inspectors assessed simulator fidelity and observed the simulator instructor's critique of operator performance to verify whether crew performance issues were properly identified and addressed. Documents associated with this inspection activity are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Biennial Review

a. Inspection Scope (1 sample)

On December 30, 2005, the inspectors conducted an in-office review of PSEG annual operating test results for 2005. The comprehensive written exam was not administered, nor required, because PSEG's two year requalification cycle will not conclude until December 2006. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process." The inspectors verified that the operating crew failure rate was less than 20%, individual failure rate on the dynamic simulator test was less than or equal to 20%, individual failure rate on the walk-through test was less than or equal to 20%, and the overall pass rate among individuals for all portions of the exam was greater than or equal to 75%.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope (3 samples)

The inspectors reviewed performance monitoring and maintenance activities for the three systems or component issues identified below to determine whether PSEG was adequately monitoring equipment performance to ensure their maintenance activities were effective to maintain the equipment reliable. Specifically, the inspectors reviewed the 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 classified as (a)(2). Documents reviewed are listed in the attachment.

- ‘B’ suppression pool to drywell vacuum breaker failure on August 28, 2005;
- HPCI flow oscillations and minimum flow valve issues on September 15, 2005; and
- ‘B’ emergency diesel generator T-19 alarm relay failure on September 28, 2005.

b. Findings

Introduction: The inspectors identified a Green finding regarding inadequate planning and implementation of work orders to refurbish the ‘A' through ‘H' suppression pool to drywell vacuum breakers. This finding was determined to be a non-cited violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.”

Description: The inspectors reviewed the apparent cause evaluation in order 70050000 for the ‘B' suppression pool to drywell vacuum breaker failure to close on August 28, 2005. The evaluation concluded the ‘B' vacuum breaker did not close because of inadequate maintenance practices and quality control for work completed in October 2001 under work order 40001162. The work was performed by completing certain sections of Procedure HC.MD-CM.GS-0002, “Drywell to Torus and Torus to Reactor Building Vacuum Relief Valve Overhaul.” Specifically, sections 5.7 and 5.8 were completed for vacuum breaker pallet seal replacement, section 5.19 to adjust the magnetic closure assemblies, and section 5.20 to replace the pallet o-rings.

The inspectors reviewed the apparent cause evaluation and identified that work order 40001162 was to be planned to refurbish the ‘B' vacuum breaker in accordance with equipment qualification maintenance and surveillance information sheet (EQSMIS) MISM150A-PSV-001, “Containment Atmospheric Control System Relief Valves.” The inspectors reviewed this document and determined it required, on a ten year frequency, that all set screws needed for proper vacuum breaker actuation be replaced or reassembled and locking material re-applied. Consistent with this program requirement, the suppression pool to drywell vacuum breaker drawing (PM150AQ-0002) identified multiple fasteners, including some in the latch plate and magnetic assemblies, that were listed with locking material. The inspectors concluded the work performed under order 40001162 was of inadequate scope because it did not ensure that fasteners required for proper vacuum breaker actuation were replaced or reassembled, and locking material re-applied, as specified by the mechanical environmental qualification program.

The inspectors further determined the completed work scope was inadequate because certain rubber type sub-components were not replaced. Mechanical equipment qualification maintenance sheet M150A-PSV-001 directed that all vacuum breaker parts made of ethylene propylene or viton be replaced every ten years. Consistent with this, Procedure HC.MD-CM.GS-0002 contained procedure steps marked as ‘EQ' to replace non-metallic components. However, the inspectors identified that ‘EQ' procedure steps 5.16.11 and 5.16.20 to replace switch o-rings and junction box o-rings were marked as "not applicable" by maintenance personnel. In response to the inspectors observations, PSEG initiated 20258660 to enter these issues in their corrective action process. PSEG identified this problem included work orders completed on all eight suppression pool to drywell vacuum breakers for their ten year refurbishments. The inspectors reviewed the applicable work orders and determined they were completed during refueling outages in April 2000 (for the ‘A,' 'F' and ‘H' vacuum breakers) and October 2001 (for the ‘B,' ‘C,' ‘D,' ‘E,' and ‘G' vacuum breakers).

Analysis: The inspectors concluded there was a performance deficiency regarding inadequate planning and implementation of refurbishment work orders completed for the eight suppression pool to drywell vacuum breakers completed in April 2000 and October 2001. These work orders did not ensure that all fasteners needed for proper vacuum breaker actuation were replaced or reassembled with locking material re-applied as specified in the equipment qualification program. Additionally, some procedure steps to replace rubber type components identified as required for equipment qualification were marked "not applicable" without technical justification and appropriate procedure changes.

This finding was more than minor because the performance deficiency was associated with the procedure quality attribute of the containment barrier integrity cornerstone and affected the cornerstone’s objective to provide reasonable assurance that physical design barriers protect the public from radionuclide release. Specifically, vacuum breaker sub-components were not replaced or refurbished in intervals evaluated and specified in the mechanical equipment qualification program. The inspectors completed a Phase 1 screening using Appendix A of Inspection Manual Chapter 0609, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” and determined the safety significance of the issue was of very low risk (Green) because it did not represent an actual open pathway in the physical integrity of reactor containment or result in an actual reduction in defense-in-depth for the atmospheric control or hydrogen control of the reactor containment. PSEG entered this problem into their corrective action program (notification 20258660) and performed an extent of condition review of the other seven drywell to suppression pool vacuum breakers. PSEG's corrective actions included immediate repairs to affected components, procedural enhancements, and training for technicians on the testing and maintenance overhaul procedures.

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 this requirement, work orders 40001157, 40001162, 40001167, 40001172, 40001177, 40001182, 40001185 and 40001190 performed in April 2000 and October 2001 for the 'A' through 'H' suppression pool to drywell vacuum breakers did not include instructions with appropriate acceptance criteria to implement environmental qualification refurbishment requirements to replace and re-apply locking material to all setscrews required for proper vacuum breaker operation. Additionally, some steps to replace rubber type components, identified as required to implement the environmental qualification program, were marked as not applicable without evaluating the change and revising the applicable environmental program and procedure requirements. Because the finding was of very low safety significance and has been entered into PSEG's corrective action program (notification 20258660) this deficiency is being treated as a NCV, consistent with section VI.A of the NRC Enforcement Policy and is identified as **NCV 05000354/2005005-01, Vacuum Breaker Mechanical Environmental Qualification Implementation**.

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

a. Inspection Scope (3 samples)

The inspectors reviewed the following three activities to verify that appropriate risk assessments were performed prior to removing equipment from service. The inspectors reviewed 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. Finally, the inspectors reviewed notifications documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed are listed in the attachment.

C 'B' station service water system pump and 'D' safety auxiliaries cooling system pump inoperable due to scheduled maintenance on October 6, 2005;

- ‘B’ and ‘D’ service water intake structure fans unavailable due to scheduled maintenance on November 1, 2005; and
- ‘D’ station service water system pump unavailable due to scheduled maintenance and two service water pumps running on November 2, 2005.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (2 samples)

For the two non-routine events described below, the inspectors reviewed operator logs, applicable procedure requirements, and observed portions of the work activities involved to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures. The inspectors also reviewed corrective action program notifications and evaluations associated with these two non-routine events to verify PSEG had identified and implemented appropriate corrective actions for these off-normal conditions:

- Operators were inspecting three fuel bundles from previous cores that were suspected of having fuel rod leaks to identify and correct the causes of this problem. The bundles were to be moved from their location in the spent fuel pool to the submerged fuel preparation machine for detailed inspection. On September 26, 2005, operators performing these inspections checked the identification markings of a fuel bundle in the preparation machine and determined they had moved an adjacent, incorrect fuel bundle. Upon identification of the error, the fuel bundle was returned to its location in the spent fuel pool and further work was suspended pending an investigation.

  The inspectors reviewed operator response to this condition and the follow-up investigation by PSEG to determine whether the response was in accordance with procedures and the causes were identified and corrected. The evaluation identified human performance issues regarding improper verification techniques and inadequate procedure implementation. Additional causal factors related to acceptance of lighting limitations in the area were also identified. The refueling bridge was determined to have worked correctly within its limitations. The inspectors reviewed the evaluation and the corrective actions for effectiveness. The issue was further reviewed to verify it did not impact the integrity of fuel cladding and that associated procedure adherence issues were minor.

- On October 6, 2005, operators had removed an individual fuel rod from a spent fuel bundle to locate the existence of a clad defect. The individual fuel rod was placed in a "capture rod" device to perform eddy current inspections of the fuel clad. The capture rod included a funnel and enclosed tube to surround the spent fuel rod. All tools and equipment remained submerged in the spent fuel pool.

When eddy current measurements were completed from the top of the fuel rod to approximately 18 inches from the bottom of the rod, the technician observed the eddy current readings appeared "not valid," and the scanning was stopped. As the fuel rod emerged from the capture rod device, technicians observed the fuel rod had separated, with the top portion remaining over the capture rod device and funnel. The technicians reported they did not observe any falling pieces or bubbles from the area, indicating there was not a release of gaseous fission products. The equipment remained in place as the refueling floor was evacuated in accordance with procedures. Radiation protection personnel confirmed there had not been a release of fission products into the refueling floor atmosphere and personnel returned to the refueling floor.

Subsequently, the upper piece of the fuel rod was re-inserted into the capture rod device, and the capture rod device and sub-assembly in the east fuel prep machine were lowered to the bottom of the spent fuel pool. On October 12, 2005, after procedures were reviewed and approved, technicians unbolted the capture rod funnel from the capture rod device and placed the capture rod device, with the failed fuel rod segments, into a defective fuel storage container, or “quiver” designed for this purpose, and placed the quiver in a spent fuel storage rack. The quiver was labeled and open, such that it allowed for visual observation of the contents.

The inspectors reviewed the operator response to the failed fuel pin to determine whether procedure requirements were followed. The inspectors reviewed the subsequent evaluation of the issue, the procedures prepared and implemented to store the failed fuel pin and contents, and verification actions taken by PSEG to ensure that all fuel rod and material remained in the capture rod and quiver. These verification actions included scanning of the spent fuel pool floor via camera and a radiation detector to verify fuel material was fully accounted for and controlled.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (5 samples)

The inspectors reviewed notifications and evaluations associated with the five conditions listed below. The inspectors reviewed the technical adequacy of the operability determinations to ensure the conclusions were technically justified. The inspectors also walked down accessible equipment to assess the adequacy of PSEG’s operability determinations and reviewed other PSEG identified safety-related equipment deficiencies during this report period to assess the adequacy of their operability screens. Notifications and documents reviewed are listed in the attachment.

- Notification 20258065, Unexpected standby liquid control (SLC) tank level increase on October 24 - 26, 2005;
- Notification 20257617, Potential latent design deficiency associated with the safety relief valves (SRVs) on October 24 - 28, 2005;
- Operability Determination 70051858, 1E panel chiller (1A-K-403) guide vane configuration on November 24, 2005;
- Notification 20262911, Crossflow correction factor error on November 27, 2005; and
- Notification 20265323, 'B' emergency diesel generator with oil from starting air system vent.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope (2 samples)

The inspectors reviewed two selected operator workaround conditions to determine if the functional capability of the system or human reliability in responding to an initiating event was affected. This included reviews of corrective action notifications that tracked items listed in the Hope Creek operations workaround list and concerns list to ensure there were not unidentified impacts due to combinations of issues. The inspectors reviewed operator logs, control room instrument panels, and station procedures to evaluate potential impacts on the operators' ability to implement abnormal or emergency operating procedures. Documents reviewed are listed in the attachment.

- 'A' filtration recirculation ventilation system (FRVS) fan's instrumentation lines (low flow sensing lines) require periodic blowdown of the lines to remove accumulated moisture; and
- Control room operators not using normal operating procedures to add makeup water to the suppression pool.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (7 samples)

The inspectors observed portions of and reviewed the results of post- maintenance tests (PMT) for the seven work activities listed below. The inspectors reviewed test procedures to verify that the procedure adequately tested the safety function that may have been affected by the maintenance activity and that the acceptance criteria in the procedure were consistent with information in the applicable licensing basis and design basis documents. Documents reviewed for this inspection activity are listed in the attachment.

- ‘A’ residual heat removal minimum flow valve on October 11, 2005;
- ‘B’ filtration, recirculation and ventilation system vent fan on October 27, 2005;
- ‘B’ safety auxiliaries cooling system pump on October 27, 2005;
- Core spray subsystem ‘B’ Valves F005B, F005B, and F031B on November 2, 2005;
- ‘B’ control room emergency filtration and support systems on November 1 through 3, 2005;
- reactor building to torus vacuum breaker 1GSPSV-5032 on November 4, 2005; and
- ‘A’ residual heat removal pump motor breaker on November 8, 2005.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (8 samples)

The inspectors observed portions and/or reviewed the results of the following eight surveillance tests:

- ‘A’ emergency diesel generator on October 12, 2005;
- Reactor core isolation cooling pump inservice test (IST) on October 24, 2005;
- Standby liquid control system daily surveillance tests on October 26, 2005;
- Leak rate surveillance tests completed during increasing unidentified drywell leak rates during the weeks of October and November 2005;
- ‘B’ Control Room Emergency Filtration System on November 1 through 3, 2005;
- Reactor building to suppression pool vacuum breakers PSV-5030 and PSV-5032 and associated isolation valves on November 3, 2005;
- Drywell leak detection radiation monitoring system functional test completed during elevated unidentified drywell leak rates on November 13, 2005; and
- ‘C’ service water pump - IST on November 16, 2005.

The inspectors evaluated the test procedures to verify that applicable system requirements for operability were adequately incorporated into the procedures and that test acceptance criteria were consistent with the technical specification (TS) requirements and the UFSAR. The inspectors also reviewed notifications documenting deficiencies identified during these surveillance tests. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (2 samples)

The inspectors reviewed the two temporary plant modifications listed below and the associated 10 CFR 50.59 screening, and compared each against the UFSAR and applicable technical specifications to verify that the modification did not affect operability or availability of the affected system. The inspectors walked down each modification to ensure that it was installed in accordance with the modification documents and reviewed post-installation and removal testing to verify the actual impact on permanent systems was adequately verified by the tests. Documents reviewed are listed in the attachment.

- 'B' station service water system ultrasonic flow device (TMOD 03-032) to replace the 1EAFE-2218B flow element and 1EAFIT-2218 flow transmitter on December 5, 2005; and
- Temporary instrument air compressor installation to augment the emergency instrument air system on December 21, 2005.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (2 samples)

The inspectors observed two licensed operator requalification scenario exams that were included as inputs into the emergency drill and exercise performance indicator. These observations were made in the simulator on November 22, 2005. The first exam involved a single reactor recirculation pump loss of speed control, a fuel leak, and a steam leak in the HPCI steam supply. The second exam involved an APRM malfunction (upscale), a single rod scram, loss of cooling water to the main generator stator, and an anticipated transient without scram (ATWS) event. The inspectors observed the exams and PSEG's post-exam critique to verify that weaknesses or deficiencies were adequately identified. The inspectors specifically focused on ensuring PSEG identified operator performance issues and problems with event classification and notification activities.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope (6 Samples)

The inspectors reviewed the solid radioactive waste system description in the UFSAR and the recent radiological effluent release report for information on the types and amounts of radioactive waste disposed, and reviewed the scope of PSEG's audit program to verify that it meets the requirements of 10 CFR 20.1101(c).

The inspectors walked-down the liquid and solid radioactive waste processing systems to verify and assess that the current system configuration and operation agree with the descriptions contained in the UFSAR and in the Process Control Program (PCP); reviewed the status of any radioactive waste process equipment that is not operational and/or is abandoned in place; verified that the changes were reviewed and documented in accordance with 10 CFR 50.59, as appropriate; and, reviewed current processes for transferring radioactive waste resin and sludge discharges into shipping/disposal containers to determine if appropriate waste stream mixing and/or sampling procedures, and methodology for waste concentration averaging provide representative samples of the waste product for the purposes of waste classification as specified in 10CFR 61.55 for waste disposal.

The inspectors reviewed the radiochemical sample analysis results for each of PSEG's radioactive waste streams; reviewed PSEG's use of scaling factors and calculations used to account for difficult-to-measure radionuclides; verified that PSEG's program assures compliance with 10 CFR 61.55 and 10 CFR 61.56 as required by Appendix G of 10 CFR Part 20; and, reviewed PSEG's program to ensure that the waste stream composition data accounts for changing operational parameters and thus remains valid between the annual or biennial sample analysis update.

The inspectors observed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and PSEG verification of shipment readiness; verified that the requirements of any applicable transport cask Certificate of Compliance have been met; verified that the receiving licensee is authorized to receive the shipment packages; and, observed radiation workers during the conduct of radioactive waste processing and radioactive material shipment preparation activities. The inspectors determined that the shippers were knowledgeable of the shipping regulations and that shipping personnel demonstrate adequate skills to accomplish the package preparation requirements for public transport with respect to NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H, and verified that PSEG's training program provides training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

The inspectors sampled non-excepted package shipment records and reviewed these records for compliance with NRC and DOT requirements.

The inspectors reviewed PSEG's Licensee Event Reports, Special Reports, audits, State agency reports, and self assessments related to the radioactive material and transportation programs performed since the last inspection and determined that identified problems are entered into the corrective action program for resolution. The inspectors also reviewed corrective action reports written against the radioactive material and shipping programs since the previous inspection.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

p. Inspection Scope

Cornerstone: Occupational Radiation Safety (1 sample)

The inspectors reviewed a listing of PSEG action reports for the period January 1, 2005 through December 16, 2005, for issues related to the occupational radiation safety performance indicator, which measures non-conformances with high radiation areas greater than 1R/hr and unplanned personnel exposures greater than 100 mrem total effective dose equivalent (TEDE), 5 rem skin dose equivalent (SDE), 1.5 rem lens dose equivalent (LDE), or 100 mrem to an unborn child.

The inspectors determined if any of these performance indicator events involved dose rates >25 R/hr at 30 centimeters or >500 R/hr at 1 meter. If so, the inspectors determined what barriers had failed and if there were any barriers left to prevent personnel access. For unintended exposures >100 mrem TEDE (or >5 rem SDE or >1.5 rem LDE), the inspectors determined if there were any overexposures or substantial potential for overexposure.

Cornerstone: Public Radiation Safety (1 sample)

The inspectors reviewed a listing of PSEG action reports for the period January 1, 2005 through December 16, 2005, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/quarter whole body or 5 mrem/quarter organ dose for liquid effluents; or 5 mrads/quarter gamma air dose, 10 mrads/quarter beta air dose; or 7.5 mrems/quarter organ doses from I-131, I-133, H-3 and particulates for gaseous effluents.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Review of Items Entered into the Corrective Action Program:

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

.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 and associated documents to identify trends that could indicate the existence of a more significant safety issue. The review was focused on repetitive unexpected annunciator alarms in the main control room including those that were invalid. The alarms often require an operator to be dispatched to investigate. The inspectors interviewed PSEG engineers and operations staff to determine if the repetitive failures were trended. The inspectors reviewed corrective action notifications and samples of control room operating logs from July through December, 2005. Based on these reviews, the inspectors selected notifications 20226707, 20245966, 20254673, 20260667, 20255674, and 20257063 for detailed review based on the frequency of recurrence of the associated alarms. The inspectors reviewed corrective actions associated with these notifications to determine if the actions taken were appropriate.

b. Assessment and Observations

No findings of significance were identified. The inspectors determined that PSEG did not trend the number or frequency of some erroneous control room overhead annunciator alarms. Although PSEG formally tracks 'nuisance alarms,' many errant alarms did not meet thresholds for tracking. Furthermore, PSEG did not evaluate the impact of these alarms on the operations staff with respect to the time spent by equipment operators responding to errant alarms. The inspectors also identified some weaknesses in PSEG's evaluation and implementation of corrective actions associated with the repetitive alarms. Specifically, a number of notifications were closed without corrective actions to address the anomalous alarms. The inspectors concluded that the combination of weaknesses in corrective actions and lack of trending by engineering were minor because they did not impact the operation of safety-related components.

However, the errant alarms were a control room distraction and required that an equipment operator investigate the issue.

.3 Annual Sample: Review of RCS Leak from Check Valve Position Indicator

a. Inspection Scope (1 sample)

The inspectors reviewed PSEG's evaluations and corrective actions associated with a leak from the FO50A residual heat removal (RHR) check valve position indicator that caused drywell leakage to increase to greater than 10 gpm on June 7, 2005. This leak also resulted in a plant shutdown in accordance with technical specifications requirements and declaration of an Unusual Event.

The inspectors reviewed notifications, all revisions of the root cause report, the associated licensee event report (LER) and supplemental LER, and previous independent recommendations during evaluation of the issue. The inspectors evaluated the adequacy of PSEG's evaluations and verified that corrective actions had been appropriately identified and prioritized. The inspectors interviewed PSEG's cognizant department staff. Documents reviewed are listed in the attachment.

b. Findings and Observations

On June 7, 2005, the plant was shutdown due to a possible reactor coolant system leak indicated by an increasing drywell floor drain leak rate and increasing drywell pressure. The plant was shutdown and PSEG personnel subsequently entered the drywell and observed the leakage was from a through-wall crack in an enclosure tube that was mounted on the bonnet of the 'A' RHR shutdown cooling return testable check valve (FO50A). This enclosure tube was part of the mechanism that provided control room remote position indication for the FO50A valve. The mechanism consisted of an actuator rod connected to the check valve disc via a clevis and pin. The actuator rod traveled up through the enclosure tube to track movement of the check valve disc. An attraction sleeve was mounted on the end of the actuator rod to operate external magnetic switches that provided check valve open and closed indication.

PSEG performed a root cause evaluation and concluded the physical cause of the FO50A enclosure tube through wall crack was caused by vibration of the attraction sleeve in the presence of switch magnetic force. This resulted in the attraction sleeve wearing though the enclosure tube wall to the point where it cracked. PSEG further concluded the root cause of this problem was that the position indication designers did not consider the consequences of vibration. PSEG identified a contributing cause was the FO50A valve and internal actuator rod experienced multiple, elevated vibration modes, at least since February 8, 2005, which likely accelerated the rate of enclosure tube wear. PSEG took initial corrective action to remove the enclosure tubes and position indication assemblies from the FO50A and 'B' RHR (FO50B) check valves and ultrasonically determined that other valves with this type of position indication had adequate thickness in the enclosure tubes.

The inspectors reviewed procedure HC.ER-AP.BB-0002(Q) Rev. 4, "Hope Creek Reactor Recirculation Piping Vibration Monitoring," and its application prior to the June 7, 2005 event. PSEG noted during discussions with the inspectors that the vendor-provided acceptance criteria values in the vibration monitoring procedure were based on large bore piping stresses only, and there would not have been recorded data processed prior to June 2005, that would have led PSEG to inspect the FO50A position indicating device.

Operation of the plant with reactor coolant system (RCS) pressure boundary leakage is prohibited by TS 3.4.3.2.a and constitutes a violation of TS requirements. Although PSEG did not identify the position indicator leak until June 7, 2005, and met the associated TS action statement from the point of discovery, it is reasonable to conclude that the pressure boundary leak existed for some in-determinant period of time prior to discovery and during plant operations, contrary to the requirements of the technical specifications, which prohibits plant operation with pressure boundary leakage. However, this fact alone does not constitute a performance deficiency.

The inspectors determined the FO50A valve position indicator pressure boundary leakage was of very low safety significance. Assuming worst case degradation, the leakage would not have increased due to internal restriction in the position indicator tube. The leakage would have remained within the capacity of the control rod drive pumps and would not have likely affected other mitigation systems resulting in a total loss of their safety function.

The inspectors concluded that the RCS pressure boundary leak resulted from an equipment failure that was not avoidable by the implementation of reasonable quality measures or management controls. The inspectors also concluded that PSEG took appropriate actions to correct the condition and adequately characterized the extent of condition and safety significance. Accordingly, although RCS pressure boundary leakage is a violation of NRC requirements, the NRC has decided to exercise enforcement discretion in accordance with VII.B.6 of the NRC Enforcement Policy and refrain from issuing enforcement action for the violation (EA-05-229).

.4 Annual Sample: Review of 'B' Reactor Recirculation Decontamination Connection Leakage

a. Inspection Scope (1 sample)

The inspectors reviewed PSEG's evaluations and corrective actions associated with the 'B' reactor recirculation loop decontamination connection leak. Specifically, the inspectors reviewed issues associated with recirculation piping vibration monitoring and the effects of pump operation on components attached to the recirculation system that may be affected by vibration induced high-cycle wear, fretting, or fatigue.

The inspectors reviewed "'B' Reactor Recirculation Decontamination Connection Leak Root Cause Evaluation Report," dated May 13, 2005, to verify the scope of the analysis was adequate and recommended corrective actions were appropriate and completed in a timely manner. In addition, the inspectors reviewed notifications documenting increased piping vibration in February 2005, and the discovery of a steam leak from the ‘B’ reactor recirculation decontamination port in March 2005. The inspectors also reviewed vibration monitoring procedures, common cause recirculation system vibration evaluation (70037702), a review of risk-informed ISI activities (70046376), samples of plant data from PSEG’s recirculation system vibration monitoring activities, and interviewed cognizant plant personnel. Documents reviewed are listed in the attachment.

b. Findings and Observations

A leak from the decontamination port was identified during a drywell walkdown by PSEG on March 27, 2005. The walkdown was performed to identify the source of increased unidentified drywell leakage, which was monitored since February 8, 2005. Following identification, the decontamination port was removed for metallurgical examination. The results of this examination suggest that the crack had existed for an extended period of time and propagated to through-wall, then failing by fatigue. PSEG shortened the length of the 4" decontamination pipe on both the 'A' and 'B' recirculation pipe loops to eliminate vulnerability to reactor recirculation pump vane passing frequency-induced vibration. PSEG systematically performed non-destructive examinations on a number of other piping welds in the drywell before restarting the unit.

PSEG collected vibration data in accordance with procedure HC.ER-AP.BB-0002(Q) Rev. 4, “Hope Creek Reactor Recirculation Piping Vibration Monitoring.” Through interviews with plant personnel and review of collected data, the inspectors learned that PSEG was collecting recirculation piping data on a monthly basis, but did not analyze (to calculate acceleration, displacement and vibration) several months of data for 2005, contrary to the above-mentioned procedure, which states that collected data should be analyzed "in a timely manner whenever it is collected." The inspectors determined that although the analysis of the data was untimely, it did not result in an inability to detect high vibrations on the recirculation piping.

Operation of the plant with RCS pressure boundary leakage is prohibited by TS 3.4.3.2.a and constitutes a violation of TS requirements. PSEG identified the decontamination port leak on March 27, 2005, and complied with the requirements of TS 3.4.3.2.a at the time of discovery; however, it is reasonable to conclude that the pressure boundary leak existed for an indeterminate period of time prior to discovery while the plant was in an operating condition in which TS 3.4.3.2 prohibits operation with RCS pressure boundary leakage. Nonetheless, this fact alone does not constitute a performance deficiency.

The inspectors determined that the decontamination port pressure boundary leakage was of very low safety significance. The decontamination port leakage resulted in an increase in RCS leakage, and operators took action prior to exceeding Technical Specification limits for unidentified RCS leakage. Assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for identified RCS leakage and would not have likely affected other mitigation systems resulting in a total loss of their safety function.

The inspectors concluded that the RCS pressure boundary leak resulted from an equipment failure that was not avoidable by the implementation of reasonable quality measures or management controls. The inspectors also concluded that PSEG took appropriate actions to correct the condition and adequately characterized the extent of condition and safety significance. Accordingly, although RCS pressure boundary leakage is a violation of technical specification requirements, the NRC has decided to exercise enforcement discretion in accordance with VII.B.6 of the NRC Enforcement Policy and refrain from issuing enforcement action for this violation (EA 06-001).

.5 (Closed) Unresolved Item 05000354/2004005-04, Station Service Water System De-Silting Criteria

The inspectors reviewed PSEG's evaluation of station service water system (SSWS) silt level acceptance criteria and silt level trending. This was an unresolved item from a previously completed problem identification and resolution annual sample. The inspectors determined that based on the inspectors questions, PSEG had expanded the areas surveyed for silt on an annual basis to include surveys and de-silting in front of the trash racks (known as area 7) and in front of the inactive SSWS bays from the cancelled Hope Creek Unit 2 pump intakes. Additionally, PSEG increased the frequency, from semi-annual to quarterly, for silt survey and de-silting tasks for the areas from the trash racks to the pump intakes.

The inspectors concluded these actions would likely reduce silting levels at the Hope Creek intake structure and better ensure the SSWS would remain capable of performing its safety function during postulated low river level conditions. The inspectors further concluded the as-found historic silt levels did not impact SSWS system availability and reliability because the levels were not consistently elevated across the face of the intake structure, the majority of silt that may be entrained would likely be very fine and pass through the SSWS, and plant procedures required periodic swapping of SSWS pumps and strainers to ensure that silt build-up had not occurred that would affect the idle pumps. Also, procedures were in place to backwash SSWS strainers in the event there was an increased differential pressure on a pump start. This unresolved item is closed.

4OA3 Event Followup (71153)

.1 (Closed) LER 05000354/2005-002-00, Through-Wall Leak on 'B' Reactor Recirculation System Decontamination Port

On March 27, 2005, the plant was shutdown to investigate an increasing unidentified reactor coolant leak rate that remained below technical specification leak rate limit requirements. PSEG discovered RCS pressure boundary leakage through a 'B' reactor recirculation decontamination port. PSEG performed a root cause analysis of the failure and repaired the failed piping with a modified design. The LER and associated evaluations were reviewed by inspectors. Inspection efforts and enforcement actions associated with this event are documented in section 4OA2.4 of this report. The LER provided an accurate description of the event and PSEG's followup actions. This LER is closed.

Inspectors opened unresolved item (URI) 05000354/2005002-07 in NRC Integrated Inspection Report 2005002, Section 4OA3.3 pending further NRC review of PSEG's evaluation of the causes of 'B' reactor recirculation decontamination connection leak and PSEG's classification of the leak. The NRC completed that review. Inspection efforts and enforcement actions associated with this event are documented in section 4OA2.4 of this report. URI 05000354/2005002-07 is closed.

.2 (Closed) LER 05000354/2005-003-00 and Supplement -01, Reactor Coolant System Leak from Check Valve Position Indicator

On June 7, 2005, the plant was shutdown to comply with Technical Specification 3.4.3.2 when the 'A' residual heat removal (RHR) check valve (FO50A) position indicator failed and caused drywell leakage to increase greater than allowed technical specification limits. PSEG performed a root cause analysis of the failure, modified the valve design, and repaired the leak. The LER 05000354/2005-003-01 and associated evaluations were reviewed by the inspectors. Inspection efforts and enforcement actions associated with this event are documented in section 4OA2.3 of this report. The LER provided an accurate description of the event and PSEG's followup actions. This LER is closed.

Inspectors opened URI 05000354/2005003-05, "'A' Residual Heat Removal Shutdown Cooling Return Testable Check Valve Leak" in NRC Integrated Inspection Report 2005003, Section 4OA3.2 pending review of PSEG's root cause analysis of the event. The NRC completed that review. Inspection efforts and enforcement actions associated with this event are documented in section 4OA2.3 of this report. URI 05000354/2005003-05 is closed.

.3 (Closed) LER 05000354/2005-008-00, Technical Specification Shutdown Due to 'B' Suppression Chamber to Drywell Vacuum Breaker Not Closed

On August 28, 2005, the 'B' drywell to suppression chamber vacuum breaker indication was observed to cycle from closed to an intermediate open position during surveillance testing. Attempts to close the vacuum breaker from the control room were unsuccessful. Because the vacuum breaker could not be closed, PSEG shut down the plant in accordance with Technical Specification 3.6.4.1.b which states, in part, with one or more suppression chamber-drywell vacuum breaker not closed, close the vacuum breaker within 2 hours; or be in at least hot shutdown within the next 12 hours. PSEG concluded the cause of the failure to be a loose locknut on the latch plate self-aligning screw due to historical poor maintenance practices. PSEG repaired the vacuum breaker and inspected the remaining vacuum breakers. PSEG entered this deficiency into their corrective action program as notification 20251271. The inspectors determined that the 'B' vacuum breaker would likely have closed under postulated accident conditions as evidenced by its closure under its own weight on August 28, 2005, after slight mechanical agitation by maintenance personnel. The inspectors concluded the plant was shutdown in accordance with technical specification requirements before further normal vacuum breaker cycling occurred which could have resulted in further latch plate displacement and prevent vacuum breaker closure under postulated accident conditions. One finding was identified as a result of inspecting the vacuum breaker failure and is documented in Section 1R12 of this report. This LER is closed.

4OA5 Other

.1 (Closed) TI 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

The inspectors examined site specific records pertaining to PSEG's use of DOT Specification 7A Type 'A' packaging for the shipment of Control Rod Drive Mechanisms (CRDM) for the period between January, 2002 and the present. The inspectors examined records for the purpose of determining PSEG's compliance with DOT transportation requirements contained in 49 CFR Parts 173.412 and 173.415. In addition to reviewing documents, inspectors interviewed cognizant PSEG personnel. The inspectors determined that Hope Creek had undergone refueling activities between January 1, 2002, and the present; and that irradiated control rod drives were shipped in DOT Specification 7A, Type 'A' packaging.

b. Findings

No findings of significance were identified.

.2 Response to Contingency Events (92709)

a. Inspection Scope

NRC Region I staff and inspectors reviewed PSEG's strike contingency plan prior to the site's security force initiating a possible strike on October 12 - 26, 2005. The staff utilized Inspection Procedure 92709 to determine if PSEG was properly implementing their safeguards contingency plan. Specifically, inspectors verified that the minimum number of qualified personnel were available for proper operation of the facility, reactor operation and facility security were maintained as required, and the strike contingency plan complied with the requirements in Hope Creek and Salem Technical Specifications and the Code of Federal Regulations. NRC staff discussed specific strike provisions with PSEG management regarding the effectiveness of site security, previous and potential safeguards threats, and plans to counter these threats. Ultimately, no strike actions were initiated because a new contract was approved.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 05000354/2005004-02, Maintenance Rule Performance Monitoring of Service Air and Instrument Air System

The inspectors opened this URI in NRC Inspection Report 2005004, Section 1R12.2 pending review of PSEG evaluation 70049655 on the adequacy of performance monitoring of the service and instrument air systems. PSEG reviewed risk insights, current practices of monitoring reliability and system condition, and engineering evaluations of the emergency instrument air compressor's design loads and capacity. PSEG concluded that current monitoring practices were appropriate. Enhancements were made to the definition of a system functional failure to reduce the likelihood of human error in declaring increased system air leakage as a system functional failure. Results were reviewed and approved by PSEG's Maintenance Rule Expert Panel on October 20, 2005.

The inspectors reviewed evaluation 70049655 and determined that performance monitoring of the emergency instrument air compressor was adequate and met the requirements of 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. URI 05000354/2005004-02 is closed.

4OA6 Meetings, Including Exit

NRC/PSEG Management Meeting. The NRC conducted a meeting with PSEG on November 17, 2005 to discuss PSEG's actions to improve performance in problem identification and resolution, and the safety conscious work environment at the Salem and Hope Creek stations. The meeting occurred at the Holiday Inn Select Bridgeport, New Jersey and was open for public observation. A copy of the slide presentations and other background documents can be found in ADAMS under accession number ML053270463.

Exit Meeting Summary

On January 9, 2006, the resident inspectors presented the inspection results to Mr. M. Massaro and other members of his staff, who acknowledged the findings. None of the information reviewed by the inspectors was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

R. Cavalieri, Outage Services Sr Manager
G. Gellrich, Plant Support Manager
H. Hanson, Hope Creek Operations Director
M. Jesse, Regulatory Affairs Manager
M. Massaro, Hope Creek Plant Manager
J. Perry, Hope Creek Maintenance Director
B. Sebastian, Radiation Protection Manager
B. Thomas, Sr. Licensing Engineer
P. Tocci, Hope Creek Training Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000354/2005005-01	NCV	Vacuum Breaker Mechanical Environmental Qualification Implementation (Section 1R12)
05000354/2005-002-00	LER	Through-Wall Leak on ‘B’ Reactor Recirculation System Decontamination Port (Section 4OA3.1)
05000354/2005-003-00; 05000354/2005-003-01	LER	Reactor Coolant System Leak from Check Valve Position Indicator (Section 4OA3.2)
05000354/2005-008-00	LER	Technical Specification Shutdown Due to ‘B’ Suppression Chamber to Drywell Vacuum Breaker Not Closed (Section 4OA3.3)

Closed

05000354/2004005-04	URI	Station Service Water System De-Silting Criteria (Section 4OA2.5)
05000354/2005002-07	URI	‘B’ Reactor Recirculation Decontamination Connection Leakage (Section 4OA3.1)
05000354/2005003-05	URI	‘A’ RHR Shutdown Cooling Return Testable Check Valve Leak (Section 4OA3.2)

05000354/2005004-02	URI	Maintenance Rule Performance Monitoring of Service Air and Instrument Air System (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 (SH.OP-AP.ZZ-0108(Q), Rev. 21)
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
Station Preparations for Winter Conditions (HC.OP-GP.ZZ-0003(Q), Rev. 18)
Station Seasonal Readiness Guide (SH.OP-DG.ZZ-0011, Rev. 4)

Drawings
Service Water (M-10-1, Sheet 2)

Notifications
20260516, 20260517, 20260518, 20260105, 20260373

Orders
60059035, 60059115, 60059116, 80077567

Other Documents
NRC IN 98-02, Nuclear Power Plant Cold Weather Problems and Protective Measures
NRC IN 96-36, Degradation of Cooling Water Systems Due to Icing
UFSAR Section 9.2.6
UFSAR Section 9.2.1
Service Water Technical Specifications 3/4.7.1.2

Section 1R04: Equipment Alignment

Procedures
Operability Assessment and Equipment Control Program (SH.OP-AP.ZZ-0108(Q), Rev. 21)
Chilled Water System Operation (HC.OP-SO.GB-0001(Q), Rev. 35)
Control Area Chilled Water System Operation (HC.OP-SO.GJ-0001(Q), Rev. 41)
Residual Heat Removal System Operation (HC.OP-SO.BC-0001(Q), Rev. 41)
Shift Turnover Responsibilities Equipment Status Checklist (SH.OP-AP.ZZ-0107(Q), Rev. 14)

Drawings
Chiller Water System Turbine Bldg. Chilled Water Drawing (M-87-1)
Residual Heat Removal Drawing (M-51-1)

Notifications
20240257, 20223074, 20219918, 20253676, 20166549, 20192648, 20209090, 20219319, 20140238, 20241360

Orders
60052408, 60057613, 60046626, 70034995, 60051244, 60055350

Other Documents
UFSAR Section 6.3
UFSAR Section 9.2.7.2
Technical Specifications 3/4.5.1
RHR System Piping and Flowpath Configuration (HC.OP-ST.BC-0001(Q), Rev. 14)

Section 1R05: Fire Protection

Procedures
Hope Creek Pre-Fire Plan FRH-II-331, Service & Radwaste Area, Rev. 4
Hope Creek Pre-Fire Plan FRH-II-5221, Cable Spreading Room, Rev. 5
Hope Creek Pre-Fire Plan FRH-III-133, Turbine Building, Rev. 5
Actions For Inoperable Fire Protection - Hope Creek Station, (HC.FP-AP.ZZ-0004(Q), Rev. 9)
Precautions Against Fire, (NC.FP-AP.ZZ-0025(Q), Rev. 4)
Fire Protection Impairment Tracking Report, dated 10/27/05
Fire Drill S4AD102605, Hope Creek 130 elevation, Room 5413, ‘B’ 1E Switchgear Room
Hope Creek Fire and Medical Emergency Response for Area FRH-II-541, Class 1E Switchgear Rooms
Annual Fire Extinguisher Inspection (HC.FP-PM.KC-0038 (Q)

Drawings
Hope Creek Safety Analysis Report Figure 9A-35, Fire Area Boundaries for Auxiliary Building Elevation 130 and 124 feet.

Notifications
20256218, 20258512, 20258609, 20258617, 20246392

Orders
60056402

Section 1R06: Flood Protection Measures

Procedures
Overhead Annunciator Window Box B1 (HC.OP-AR.ZZ-0006 (Q), Rev. 2)
Device/Equipment Calibration: Fluid Components Inc. (HC.IC-DC.ZZ-0212 (Q), Rev. 3)

Drawings
Building and Equipment Drain - Reactor Building (M-97-1, sheet 2)

Notifications
20162207, 20214375, 20247022

Orders
30095949, 30106773, 40005000, 40005582, 60056297

Other Documents
Calculation SC-SK-0075, Room and Structure Flooding Alarm, Revision 6, March 17, 2000

Section 1R11: Licensed Operator Requalification Program

Procedures
Seismic Instrumentation (HC.OP-SO.SG-0001(Z), Rev. 6)
Transient Plant Conditions (HC.OP-AB.ZZ-0001(Q), Rev. 5)
Acts of Nature (HC.OP-AB.MISC-0001(Q), Rev. 6)
Drywell Pressure (HC.OP-AB.CONT-0001(Q), Rev. 0)
Reactor Auxiliary Cooling (HC.OP-AB.COOL-0003(Q), Rev. 2)
Reactor/Pressure Vessel (RPV) Control (HC.OP-EO.ZZ-0101(Q), Rev. 1)
RPV Flooding (HC.OP-EO.ZZ-0206(Q), Rev. 7)
Emergency RPV Depressurization (HC.OP-EO.ZZ-0202(Q), Rev. 7)

Section 1R12: Maintenance Implementation

Procedures
Drywell to Torus and Torus to Reactor Building Vacuum Relief Valve Overhaul (HC.MD-CM.GS-0002(Q), Rev. 12)
HPCI System Valves - Inservice Test (HC.OP-IS.BJ-0101, Rev. 27)
System Function Level Maintenance Rule Scoping vs. Risk Reference (HC.ER-DG.ZZ-0002(Z), Rev. 0

Drawings
PSEG Vendor Technical Document PM150AQ-0002, 24 Inch Slimline Vacuum Breaker
J-55-0 sht 4 and 4a - HPCI Logic Diagram
E-6075-0 sht 4 - Electrical Schematic Diagram - HPCI Min flow bypass valve F012
D8010029E - Bailey Controls vendor diagram HV-F012
791E420AC shts 6, 8, 12, - General Electric HPCI System schematic diagram

Notifications
20258660, 20254514, 20253733, 20253009, 20252921, 20253472, 20254427

Orders
40001157, 40001162, 40001167, 40001172, 40001177, 40001182, 40001185, 40001190, 70050000, 60057465, 30016391

Other Documents
Equipment Qualification Maintenance and Surveillance Information Sheet MISM150A-PSV-001, Containment Atmospheric Control System Relief Valves, Revision 4.
General Electric Service Information Letter 321, Increasing Wetwell to Drywell Vacuum Breaker Reliability, December 1979
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
Event 41995 - Event Report
HPCI System Health report - September 2005
Control Room Narrative Logs - 9/14/05 through 9/16/05
Event 41995 - Event Retraction Report
HPCI Lesson Plan - 0301-000.00H-000026
HPCI Hunting Troubleshooting Plan

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
On-Line Risk Assessment (SH.OP-AP.ZZ-0108(Q), Rev. 21)
PRA Weekly Risk Assessment (a)(4) Desktop Guide (NC.CC-DG.ZZ-0003(Z), Rev. 2)
Alternate Injection Using Condensate Transfer (HC.OP-EO.ZZ-0309(Q), Rev. 4)
Condensate Storage and Transfer System Operation (HC.OP-SO.AP-0001(Q), Rev. 2)

Notifications
20255099, 200221702, 20201633, 20214446, 20251208, 20258988, 20259394, 20260414

Orders
60048166, 70041138

Other Documents
HCGS PSA Risk Evaluation Forms for affected work weeks
Evaluation of 12 Week On Line Maintenance Schedule for Hope Creek (H-1-ZZ-RZZ-0025)
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

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

Procedures
Hope Creek Conduct of Fuel Handling (HC.RE-AP.ZZ-0049(Q), Rev. 2)
Refueling Platform and Fuel Grapple Operation (HC.OP-SO.KE-0001(Q), Rev. 3)
Irradiated Fuel Damage (HC.OP-AB.CONT-0005(Q), Rev. 1)

Westinghouse Boiling Water Reactor Failed Fuel Inspection and Repair (VHC.RE-FR.ZZ-0027(Q), Rev. 3)
Special Nuclear Material Control and Accounting (HC.RE-FR.ZZ-0017(Q), Rev. 5)

Notifications
20254206, 20255511, 20256254

Orders
70050693, 70046610

Other Documents
Pre-Job Brief Sheet for Irrradiated Fuel Handling Within the Spent Fuel Pool
Pre-Job Brief for September 2005 Failed Fuel Inspection
PSEG memo to file from J. Stavely - Hope Creek Reactor Engineering, dated October 12, 2005, approving intermediate location of failed fuel rod in spent fuel pool

Section 1R15: Operability Evaluations

Procedures
Operability Assessment and Equipment Control Program (SH.OP-AP.ZZ-0108(Q), Rev. 21)
Notification Process (NC.WM-AP.ZZ-0000(Q), Rev. 12)
Hope Creek Carrier Centrifugal Chiller Frequent & Periodic Inspections (HC.MD-GP.ZZ-0245(Q), Rev. 1)

Notifications
20260490, 20262694

Orders
70051858, 60045681, 60057725, 70050651, 70047411

Operating Experience
NRC Information Notice No. 80-40: Excessive Nitrogen Supply Pressure Actuates Safety-Relief Valve Operation to Cause Reactor Depressurization, dated 11/7/80
NRC IE Bulletin No. 80-25: Operating Problems with Target Rock Safety-relief Valves at BWRs, dated 12/19/80
GE SIL No. 196S8, SRV Inadvertent Opening Caused By High Pneumatic Supply Pressure, dated 11/4/80
NRC Information Notice No. 86-48: Inadequate Testing of Boron Solution Concentration in the Standby Liquid Control System, dated 6/13/86
Target Rock Safety/Relief Valve Model 7567F Technical Manual, dated April 1987

Other Documents
NRC Inspection Manual Part 9900: Technical Guidance Operability Determinations & Functionality Assessments For Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety, dated 9/26/05
DCR No. 4EC-3102 Closure Notification, dated 11/22/94

DCR No. 4EC-3315 Closure Notification, dated 7/6/94
Hope Creek UFSAR - 9.2.7.2 - Control Area Chilled Water System

Section 1R16: Operator Workarounds

Notifications
20155744, 20143377, 20247527, 20249279

Other Documents

Condition Resolution Operability Determination Notebook
Inoperable Instrument/Alarm/Indicators/Lamps/Device Log
Inoperable Computer Point Log
Hope Creek Operator Workaround List
Hope Creek Operator Concerns List

Section 1R19: Post-Maintenance Testing

Procedures
Maintenance Testing Program Matrix (NC.MD-AP.ZZ-0050(Q), Rev. 4)
Residual Heat Removal Subsystem A Valves - Inservice Test (HC.OP-IS.BC-0101(Q), Rev. 23)
FRVS Operability Test (Single Vent Fan Method) - Monthly, dated 10/27/05 (HC.OP-ST.GU-0007(Q), Rev. 2)
B SACS PUMP - BP210 - Inservice Test, dated 10/27/05 (HC.OP-IS.EG-0002(Q), Rev. 31)
Core Spray Subsystem ‘B’ Valves Inservice Test (HC.OP-IS.BE-0102(Q), Rev. 21)
Control Room Emergency Filtration System Functional Test - Monthly (HC.OP-ST.GK-0003(Q), Rev. 5)
Control Area Chilled Water System Operation (HC.OP-SO.GJ-0001(Q))
Reactor Building/Suppression Chamber Vacuum Breaker Operability Test - Monthly (HC.OP-ST.GS-0003(Q), Rev. 5)

Notifications
200258443, 200258751, 20259662, 20261266

Orders
60057946, 60058011, 50072758, 50072757, 50072756, 50090452, 30107172, 30106796, 50084400, 50086880, 50075326, 70047411, 30127136, 70051459

Section 1R22: Surveillance Testing

Procedures
EDG 1AG400 - 24 Hour Operability Run and Hot Restart Test (HC.OP-ST.KJ-0014(Q), Rev. 22)
Emergency Diesel Generator 1AG400 Operability Test-Monthly (HC.OP-ST.KJ-0001(Q), Rev. 57)

Reactor Core Isolation Cooling Pump - OP203 - Inservice Test, dated 4/19/05, 7/12/05, 8/5/05, and 10/24/05 (HC.OP-IS.BD-0001(Q))
Attachment 1a, Surveillance Log - Control Room, dated 10/26/05 (HC.OP-DL.ZZ-0026(Q), Rev. 101)
Attachment 1c, Reactor Building, dated 10/26/05 (HC.OP-DL.ZZ-0026(Q), Rev. 101)
SLC Valve Operability Test - Monthly, dated 10/10/05 (HC.OP-ST.BH-0001(Q), Rev. 5)
'B' Control Room Emergency Filtration System Functional Test (HC.OP-ST.GK-0003(Q), Rev. 5)
Reactor Building to Suppression Chamber Vacuum Breaker Operability Test (HC.OP-ST.GS-0003(Q), Rev. 5)
Process Radiation Monitoring - Non Divisional Monitor HISK-ISKRY-4991, Drywell Leak Detection Noble Gas (HC.IC-CC.SK-0013(Q), Rev. 16)
Radiation Monitoring - Non Divisional Channel 1SK-RY-4991, Drywell Noble Gas (HC.IC-FT.SK-0013(Q), Rev. 21

Calculations
SC-BH-0501, Loop Tolerance Calculations for 1-BH-TSHL-4106 A/B, Rev. 1

Notifications
200257739, 200258225, 20257109, 20253779, 20249242

Orders
50077200, 50090062, 60058556, 50090452, 50090456, 50091582, 50090726, 50071218, 70050376

Other Documents
Instrument Calibration Data Report for 1BHTSHL-4106A & 1BHTSHL-4106B
Technical Specifications 3/4.4.3 "Reactor Coolant System Leakage"
Surveillance Log - Control Room, Page 6 of 17
Plant Leak Detection (M-25-1) of Drywell atmosphere since 8/16/05, drywell leak detection (DLD) Iodine since 2/17/05, DLD Particulate Filters since 2/17/05

Section 1R23: Temporary Plant Modifications

Procedures
Control of Temporary Modifications (NC.DE-AP.ZZ-0030(Q), Rev. 4)
Replace 1EAFE-2218B & 1EAFIT-2218B with Panametrics DF868 Ultrasonic Flowmeter (TM# 03-032, Rev. 2)
Replace 1EAFE-2218B & 1EAFIT-2218B with Panametrics DF868 Ultrasonic Flowmeter (TM# 03-032, Rev. 3)
Instrument Air System Operation (HC.OP-SO.KB-0001, Rev. 17)
Installation of Temporary Air Compressors (HC.MD-GP.ZZ-0099(Z), Rev. 6)
Service Air System Operation (HC.OP-SO.KA-0001(Z), Rev. 16)

Drawings
Service Water (M-10-1, Sheet 2)
Breathing Air (M-15-1)

Notifications
20197987, 20241153, 20263403, 20216054, 20261295, 20253353, 20253354

Orders
60046808, 60037424, 70028374, 80077269, 70051709

Other Documents
Service Water Technical Specifications 3/4.7.1.2
UFSAR Section 9.2.1
UFSAR 9.3.1

Section 1EP6: Drill Evaluation

Other Documents
Exam Scenario Guide ESG-008 Single Recirc Runaway, Fuel Leak, HPCI Steam Leak
Exam Scenario Guide ESG-073 APRM Upscale, Single Rod Scram, Loss of Stator Water Cooling, ATWS

Section 2PS2: Radioactive Material Processing and Transporation

Procedures
Environmental Training Program (NC.TQ-TC.ZZ-0220(Z), Rev 6)

Other Documents
Radioactive waste shipments: 05-50; 05-51; 05-52; 05-62; 05-81; 05-86; 05-103
Training Module "Radioactive Material Shipping (79-19)"
Quality Assurance Assessment Reports:
2004-0084, Process Control Program for Processing and Packaging of Radioactive Wastes
2003-0229, Solid Radioactive Waste Packaging and Transportation
Nuclear Utilities Procurement Issues Council Audit # SA05-007, Subject: Duratek
Framatome ANP Environmental Laboratory Reports: Radwaste Bead Resin; Dry Active Waste; Condensate Bead Resin; Waste Sludge
Nuclear Training Center Lesson Plan, HSENDOT-HMRC, DOT Hazmat Employee for NP&MM

Section 4OA1: Performance Indicator Verification

Notifications
All HP and RETS related notifications from January 1, 2005 through December 16, 2005.

Section 4OA2: Identification and Resolution of Problems

Procedures
Service Water Intake Bay Silt Survey and Silt Removal (HC.MD-PM.EA-0002(Q), Rev. 13)
Hope Creek Reactor Recirculation Piping Vibration Monitoring (HC-ER-AP.BB-0002(Q), Rev. 4)
Attachment 1, various dates (HC-ER-AP.BB-0002(Q), Rev. 4)

Form 2 Ultrasonic Thickness Examination Record, 1BE-HV-F041D, 1BE-HV-F041B, 1BE-HV-F041A, 1BE-HV-F041C, 1BE-HV-F006A, 1BE-HV-F006B (SH.RA-AP.ZZ-0101(Q), Rev. 14)
Hope Creek Reactor Recirculation Piping Vibration Monitoring (HC.ER-AP.BB-0002(Q), Rev. 4)
Plant Data Acquisition, Attachments 1 & 2 (HC.ER-AP.BB-0002(Q), Rev. 4)
Atwood & Morrill Testable Check Valve Overhaul (HC.MD-GP.ZZ-0042(Q), Rev. 6)
Operations Standards (SH.OP-AS.ZZ-0001(Q), Rev. 11)

Notifications
20223538, 20230319, 20257644, 20214019, 20242287, 20245966, 20226707, 20254673, 20255674, 20257063, 20260667,

Orders
70042421, 70037702, 70046376, 70037702

Other Documents
Station Service Water System Desilting Maintenance Plans HC213816, HC213817, HC213818, HC213819
LR-N04-0599, December 29, 2004, Startup Readiness
LR-05-0017, January 9, 2005, PSEG Actions in Response to NRC Concerns Regarding 'B' Reactor Recirculation Pump
Sargent and Lundy LLC, November 12, 2004 Independent Assessment of Hope Creek Reactor Recirculation System and Pump Vibration Issues
'B' Reactor Recirculation Decontamination Connection Leak Root Cause Evaluation Report, May 13, 2005
'A' Residual Heat Removal Check Valve Position Indicator Leak Cause Determination Report, 6/12/2005
'A' Residual Heat Removal Check Valve Position Indicator Leak Root Cause Evaluation Report, 8/26/2005, 9/30/2005
MPR-2826, Rev. 0, August 2005, Cause Determination of Leak in Hope Creek RHR Check Valve F050A Magnitrol Enclosing Tube
LR-N04-0599, Dec. 29, 2004, Startup Readiness Hope Creek Generating Station "B" Reactor Recirculation Pump, Action 9, SL recommendations status.xls
LR-05-0383, Aug. 8, 2005, LER 354/2005-003-00
Atwood & Morrill Co., Inc. Seismic Analysis For 12" - 900# Testable Check Valves, Reference Drawing 14053-01-H
Independent Assessment of Hope Creek Reactor Recirculation System and Pump Vibration Issues, Nov. 12, 2004, Sargent & Lundy LLC
LR-N05-0477, Sept. 29, 2005, Supplemental LER 354/2005-003-01
Hope Creek Control Room Operator Logs
Evaluation of Hope Creek In-Drywell Pipe Vibration, 4/05/2004 (H-1-BB-CEE-1830)
Hope Creek Recirc/RHR Pipe Vibration Common Cause Analysis, 7/27/2004 (H-1-BB-CEE-1862)

Section 4OA3: Event Followup

References used for the three LERS documented in 4OA3 were listed in section 4OA2 of this report where the inspection effort was documented.

Section 4OA5: Other Activities

Notifications
20258185

Orders
70049655

Other Documents

UFSAR Section 9.3.1

LIST OF ACRONYMS

ATWS	Anticipated Transient Without SCRAM
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
DOT	Department of Transportation
EDG	Emergency Diesel Generator
FRVS	Filtration, Recirculation and Ventilation System
HCGS	Hope Creek Generating Station
HPCI	High Pressure Coolant Injection
IPEEE	Individual Plant Examination For External Events
IST	Inservice Test
LDE	Lens Dose Equivalent
LER	Licensee Event Report
MR	Maintenance Rule
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PCP	Process Control Program
PMT	Post Maintenance Testing
PSEG	Public Service Enterprise Group
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SACS	Safety Auxiliaries Cooling System
SDE	Skin Dose Equivalent
SDP	Significance Determination Process
SIL	Service Information Letter
SLC	Standby Liquid Control
SRV	Safety Relief Valve

SSWS	Station Service Water System
TEDE	Total Effective Dose Equivalent
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item