May 9, 2005

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

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000354/2005002

Dear Mr. Levis:

On March 31, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Hope Creek Nuclear Generating Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 1, 2005, 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 two NRC-identified findings and three self-revealing findings of very low safety significance (Green). Three of 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 of the NRC Enforcement Policy. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. 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/

Eugene W. Cobey, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000354/2005002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	050000354
License No:	NPF-57
Report No:	05000354/2005002
Licensee:	Public Service Enterprise Group Nuclear LLC
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	January 1 - March 31, 2005
Inspectors:	M. Ferdas, Acting Senior Resident Inspector T. Wingfield, Acting Resident Inspector M. Gray, Senior Resident Inspector J. Schoppy, Senior Project Engineer S. Hansell, Senior Resident Inspector H. Gray, Senior Reactor Inspector S. Unikewicz, Mechanical Engineer (NRR) J. Wiebe, Project Engineer L. Scholl, Senior Reactor Inspector B. Welling, Senior Project Engineer J. Furia, Senior Health Physicist R. Bhatia, Reactor Inspector
Approved By:	Eugene W. Cobey, Chief Projects Branch 3 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000354/2005002; 01/01/2005 - 03/31/2005; Public Service Enterprise Group (PSEG) Nuclear LLC, Hope Creek Generating Station; Flood Protection Measures, Maintenance Effectiveness, Non-routine Plant Evolutions, ALARA Planning and Controls, Radioactive Material Processing and Transportation.

The report covered a 13-week period of inspection by resident inspectors, and an announced inspection by a regional radiation specialist and reactor inspectors. Three Green non-cited violations (NCVs) and two green findings 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: Initiating Events

- Green. A self-revealing non-cited violation was identified for a recurring failure of the 'A' control area chill water (CACW) pump. The 'A' CACW pump malfunctioned about three weeks prior for a similar reason (air binding), but corrective actions were not effective at preventing recurrence. This finding was determined to be a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

  Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements. The finding was more than minor because it was associated with the equipment performance attribute of the initiating events cornerstone and affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase I SDP screening and determined that the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This required that a Phase 2 SDP analysis be performed. The Region I senior risk analyst (SRA) performed a modified Phase 2 analysis and determined that the issue was of very low safety significance (Green). The performance deficiency had a problem identification and resolution (evaluation and corrective actions) cross cutting aspect. (Section 1R12)

Cornerstone: Mitigating Systems

- Green. The inspectors identified that PSEG did not correct a degraded condition associated with the control rod drive (CRD) pump room floor access hatches and floor drains after the condition resulted in water leaking onto the ‘B’ and ‘D’ core spray pumps in December 2004. This finding was determined to be a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.”

  Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC’s regulatory function, and was not the result of any willful violation of NRC requirements. The finding was more than minor because it was associated with the protection against external factors (flood hazard) attribute of the mitigating systems cornerstone and affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspector Manual Chapter (IMC) 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a Phase I SDP screening and determined the finding to be of very low safety significance (Green). The finding was of very low safety significance because the issue was not a design or qualification deficiency that resulted in a loss of function, did not result in an actual loss of safety function of a single train of equipment for greater than its Technical Specification allowed outage time, did not result in an actual loss of safety function of equipment considered risk significant in the maintenance rule program for greater than 24 hours, and the finding does not screen as potentially risk significant due to external events. The “Seismic, Flooding and Severe Weather Screening Criteria” worksheet in the SDP Phase 1 worksheet was used to determine that the finding was not risk significant due to flooding. The finding does involve the degradation of equipment designed to mitigate flooding events, but it would not cause an initiating event, does not degrade more than one train of the core spray system, and does not degrade a support system. The performance deficiency had a problem identification and resolution (identification and evaluation) cross cutting aspect. (Section 1R06)

- Green. The inspectors identified that control room operators were not able to properly operate the reactor recirculation pump vibration monitoring equipment used to respond to vibration alarms and implement commitments to NRC Confirmatory Action Letter (CAL) 1-05-001. The finding was not a violation of NRC requirements, in that, the performance deficiency was related to operation of non-safety related equipment.

  Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC’s regulatory function and was not the result of any willful violation of NRC requirements. The finding was more than minor because if the condition was left uncorrected the finding would become a more significant safety concern. The finding is not suitable for SDP evaluation because it did not have an actual impact on the initiating events, mitigating systems, or barrier integrity cornerstone. This finding has been reviewed by NRC management and was determined to be a finding of very low safety significance (Green). The performance deficiency had a problem identification and resolution (corrective action) cross cutting aspect. (Section 1R14)

Cornerstone: Occupational Radiation Safety

C Green. A self-revealing finding was identified when work activities during refueling outage 12 (RF12) in the general torus and torus room exceeded their collective dose estimate by 312 percent (%). PSEG failed to evaluate the expanded work scope that occurred in these areas for dose minimization. The finding was not a violation of NRC requirements, because overall exposure performance of the nuclear power plant is used to determine compliance with the as low as reasonably achievable (ALARA) rule.

Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements. This finding was more than minor because it was associated with the ALARA planning attribute of the occupational radiation safety cornerstone and affected the objective to ensure the adequate protection of worker health and safety from exposure of radiation from radioactive material during routine civilian nuclear reactor operations. This finding was also similar to more than minor example 6.a in NRC Inspection Manual Chapter (IMC) 0612, Appendix E, "Examples of Minor Issues," in that, the actual dose achieved exceeded the planned, intended dose by more than fifty percent. This finding was evaluated using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," because the issue involved ALARA. The inspectors determined the finding to be of very low safety significance (Green) because Hope Creek's three-year-rolling-average (2001-2003) is 126 person-rem, which is below the SDP criteria of 240 person-rem for Boiling Water Reactors (BWRs). The performance deficiency had a problem identification and resolution (identification) cross cutting aspect. (Section 2OS2)

Cornerstone: Public Radiation Safety

C Green. A self-revealing non-cited violation was identified when a PSEG shipment of outage related equipment received by a vendor had external radiation levels in excess of regulatory limits. This finding was determined to be a violation of 10 CFR 71.5, "Transportation of Licensed Material," and 49 CFR 173.441(a), "Radiation Level Limitations."

Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements. This finding was more than minor because it was associated with the program and process attribute of the public radiation safety cornerstone and affected the objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operations. This finding was evaluated using

Inspection Manual Chapter (IMC) 0609, Appendix D, "Public Radiation Safety Significance Determination Process," because it was a radiation material control (radioactive material packaging and transportation) issue. The inspectors determined the finding to be of very low safety significance (Green) because the transportation issue resulted in a radiation limit being exceeded that involved external radiation levels that was not readily accessible by the public and not more than two times the federal limit. The inspectors also determined that the finding did not involve a breach in the package, a certificate of non-compliance issue, a low-level burial ground non-conformance, and that surface contamination limits were not exceeded. (Section 2PS2)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The Hope Creek Nuclear Generating Station started the inspection period in refueling outage 12 (RF12) that began on October 18, 2004. Following completion of RF12, operators took the reactor critical on January 18, 2005, and synchronized the main generator to the grid on January 26, 2005.

On January 29, 2005, operators reduced power to 16 percent (%) and removed the main turbine from service to repair a steam leak from a turbine drain line clean-out flange to a pipe weld connection. Following the repair, the turbine was synchronized to the grid on January 30 and the plant achieved 100% (full) power on February 8, 2005.

On February 8, 2005, operators experienced a high vibration alarm on the 'B' reactor recirculation pump. The alarm was experienced after the 'B' reactor recirculation pump speed had been increased to maintain 100% power. Operators, in accordance with alarm response procedure HC.OP-AR.ZZ-0008, "Overhead Annunciator Window Box C1," and abnormal operating procedure HC.OP-AB.RPV-0003, "Recirculation System," reduced pump speed to clear the vibration alarm and power was maintained at 99%. Engineering personnel evaluated the conditions in accordance with engineering procedure HC.ER-AP.BB-0001, "Reactor Recirculation Pump/Motors Vibration Monitoring." and concluded the condition was not representative of shaft cracking. Operators further reduced power several hours later to 90% in accordance with engineering procedure HC.ER-AP.BB-0002, "Hope Creek Recirculation Piping Vibration Monitoring," when vibration levels on the 'B' reactor recirculation ring header exceeded vibration level action limits. PSEG assessed the vibration data and re-analyzed the action limit values. On February 10, 2005, operators began to raise power at 2% increments and collected vibration data on the 'B' reactor recirculation ring header. The plant reached full power on February 11, 2005.

On March 11, 2005, the ‘A’ reactor feed pump experienced flow oscillations. Operators removed the ‘A’ reactor feed pump from service and reduced reactor power to 95%. Operators performed a further power reduction to 90% when they received indication of elevated bearing temperatures on the ‘B’ and ‘C’ reactor feed pumps. The flow oscillations were attributed to a degraded control oil actuator (EG-10P) that resulted in sluggish control valve response. The actuator was replaced, the feed pump control system was re-calibrated, and the plant returned to full power on March 13, 2005.

On March 26, 2005, operators commenced a planned down power to investigate the source of unidentified leakage in the drywell. The unidentified leakage rate had been slowly increasing over several weeks and remained below PSEG’s administrative limit, 1.5 gallons per minute (gpm) and Technical Specification limit, 5 gpm. On March 27, 2005, operators reduced power to 5% and entered operational condition 2, “Startup,” to perform walkdowns of the drywell. During the drywell walkdowns PSEG personnel identified a steam plume from the ‘B’ reactor recirculation system decontamination connection piping. Upon discovery of the leak PSEG took the plant to operational condition 4, “Cold Shutdown.” PSEG determined that the leak was coming from a through wall crack on a weld that connected the decontamination connection to the ‘B’ reactor recirculation piping. At the end of the inspection period, PSEG personnel were evaluating the cause of the leak, conducting repairs to the decontamination connections, and preparing to return the unit to operation.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope (1 site specific sample)

The inspectors reviewed PSEG's response to one site specific weather-related condition. The inspectors reviewed applicable documents associated with adverse weather as listed in the Supplemental Information attachment to this report.

Cold Weather Conditions. During cold weather conditions in January 2005 the inspectors verified that heat tracing, insulation and space heaters were properly protecting equipment susceptible to damage from freezing conditions. The inspectors walked down portions of the station service water system (SSWS) and the fire water pump house. The inspectors also verified that condensate storage tank (CST) temperatures were being properly maintained. On January 21, 2005, the inspectors reviewed PSEG’s response to an impending snow storm to ensure PSEG’s response was in accordance with procedure NC.OP-DG.ZZ-0002, “Severe Weather Guide.”

On August 23, 2004, the NRC’s Executive Director for Operations approved a deviation from the NRC’s Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20192246, 20173153, 20182701, 20216226, and 20202031 related to adverse weather preparation issues. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope (5 safety evaluations and 21 safety screening samples)

The inspectors reviewed five safety evaluations to verify that changes and tests were reviewed and documented in accordance with 10 CFR 50.59; and, when required, PSEG obtained NRC approval prior to implementation. The inspectors assessed the adequacy of the safety evaluations through interviews with PSEG personnel and review of supporting information, such as calculations, engineering analyses, design change documentation, the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS) and plant drawings. In addition, the inspectors reviewed the administrative procedures that control the screening, preparation and issuance of the safety evaluations to ensure that procedures adequately implemented the requirements of 10 CFR 50.59, “Changes, Tests, and Experiments.”

The inspectors also reviewed a sample of twenty-one changes that PSEG had evaluated (using a screening process) and determined to be outside the scope of 10 CFR 50.59, therefore not requiring a full safety evaluation. The inspectors performed this review to assess PSEG’s conclusions with respect to 10 CFR 50.59 applicability. A sample of issues not needing a full safety evaluation (design changes, procedure changes, FSAR changes, temporary modifications, and a calculation revision) were reviewed.

The inspectors also reviewed issues that had been entered into the corrective action program to determine if PSEG had been effective in identifying problems associated with the 10 CFR 50.59 safety evaluation process. A sample of these issues were selected for additional review to assess the adequacy of the corrective actions which had been implemented.

The safety evaluations and safety screenings were selected based on the safety significance of the affected structures, systems, and components (SSCs). A listing of the safety evaluations, safety screenings, and other documents reviewed are listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC’s Executive Director for Operations approved a deviation from the NRC’s Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for in the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20217171, 20149572, 20192409, and 20221391. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope (4 partial walkdowns and 1 complete walkdown)

The inspectors performed four partial equipment alignment inspections and one complete alignment inspection. Partial alignment inspections were performed on the instrument air, standby liquid control (SLC), emergency diesel generator (EDG) and residual heat removal (RHR) systems. A complete equipment alignment inspection was performed on the core spray system. The inspectors reviewed applicable documents associated with equipment alignments as listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20211151, 20210271, 20177551, and 20206669 related to equipment alignment issues. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

Partial System Alignments. On January 11, 2005, both of the service air compressors were removed from service for a planned service air outage. During this period of time instrument air which is normally supplied from service air was supplied by temporary air compressors located outside the turbine building. The inspectors performed a field walkdown of accessible portions of the service air system from the temporary air compressors through the temporary air dryer and instrument air dryer 1AF-104. The inspectors verified that the temporary air compressors were operating properly and that the temporary piping was properly connected and supplying instrument air to Hope Creek in accordance with plant procedures.

On February 16, 2005, the 'B' SLC pump was removed from service for planned maintenance. The inspectors reviewed the applicable 'A' SLC system operating procedures and drawings to verify that the system was correctly aligned to perform its safety function during unavailability of the 'B' SLC pump. The inspectors verified by plant walkdowns and main control room tours that the redundant 'A' SLC components were adequately protected.

On February 16, 2005, the 'B' EDG was declared inoperable during performance of surveillance procedure HC.OP-ST.KJ-0015, "B EDG 24 Hour Run and Hot Restart Test." The inspectors reviewed the EDG operating procedure and walked down portions of the 'A', 'C', and 'D' EDGs and their associated control panels to verify the EDGs were correctly aligned and maintained to ensure the EDGs were operable.

On, March 29, 2005, the ‘B’ RHR pump was in service in shutdown cooling mode in accordance with HC.OP-SO.BC-0002, “Decay Heat Removal Operation.” The inspectors reviewed the procedure, interviewed operators, observed control room panel lineup, and verified process parameters. The inspectors performed a field walkdown of non-contaminated portions of the ‘A’ and ‘B’ RHR pump and heat exchanger rooms, the ‘A’ and ‘B’ RHR instrument racks, and the safety auxiliaries cooling system to verify that the ‘B’ RHR system was correctly aligned and maintained to ensure reliable operation in the shutdown cooling mode.

Complete System Alignment. The inspectors performed one complete system alignment inspection on the core spray (CS) system to determine whether the system was aligned and capable of providing reactor vessel inventory makeup and spray cooling in accordance with design basis requirements. The inspectors reviewed operating procedures, the surveillance test procedure and equipment lineup lists to determine the required equipment alignment. The inspectors reviewed applicable documents associated with these equipment alignments as listed in the Supplemental Information attachment to this report.

The inspectors verified that the CS system components (piping, valves, and pumps) up to the drywell penetration were aligned to support CS system operation during accident conditions. Specifically, the inspectors verified: valves were locked or maintained in the required position; the system components and supporting equipment were correctly installed and operational; major components were correctly labeled, lubricated, cooled, ventilated and properly maintained; local indication for the keep-fill system provided reasonable assurance that the system would run properly upon an actuation; all major component electrical power supplies were available and aligned for standby readiness; and control room indication and controls were aligned for use by the facility operators. The inspectors also reviewed engineering system health reports to determine if equipment alignment problems for the system were being identified and corrected at an appropriate threshold. Additionally, the inspectors reviewed notifications documenting equipment issues associated with the CS system to verify identified problems were being evaluated and corrected. The overall system performance and reliability were discussed with the system engineer.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope (12 samples)

The inspectors walked down twelve plant areas to assess their vulnerability to fire. During plant walkdowns the inspectors observed combustible material control, fire detection and suppression equipment availability and compensatory measures. The inspectors reviewed Hope Creek’s Individual Plant Examination for External Events (IPEEE) for risk insights and design features credited in these areas. Additionally, the inspectors reviewed notifications documenting fire protection deficiencies to verify identified problems were being evaluated and corrected. The following plant areas were inspected:

- Control rod drive pump room on January 5;
- Reactor auxiliaries cooling system pump and heat exchanger room and adjacent safeguards instrumentation rooms on January 5;
- ‘B’/’D’ safety auxiliaries cooling system pump room on January 13;
- Main control room on January 14;
- ‘A’ recirculation pump motor generator set room on January 15;
- Electro hydraulic control pump skid on January 26;
- ‘A’ and ‘C’ class 1E switchgear rooms on February 14;
- ‘B’ and ‘D’ class 1E switchgear rooms on February 14;
- Electrical equipment room 5501 on February 14;
- Steam tunnel and reactor core isolation cooling and high pressure coolant injection pipe chase rooms, on March 10;
- Drywell pad and torus area on March 11; and
- Core spray pump rooms on March 10.

On August 23, 2004, the NRC’s Executive Director for Operations approved a deviation from the NRC’s Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notification 20219158 to determine if a 55 gallon drum of oil, located in the reactor building near the control rod drive room, was moved to an acceptable location per the fire protection program.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope (1 internal sample)

The inspectors performed one internal flood protection inspection activity in the control rod drive (CRD) pump room. The inspectors performed a walkdown of the flood barriers, floor drains, flood detection detectors, and procedures related to flooding of the CRD pump room. The inspectors evaluated these items to determine if internal flood vulnerabilities existed and to assess the physical condition of the equipment and components in the CRD pump room. Documents associated with these reviews are listed in the Supplemental Information attachment to this report.

During a walkdown of the area the inspectors noted the floor drains were marked with tape indicating they were clogged. The inspectors verified that the condition was documented in PSEG’s corrective action program and identified that corrective action notification 20216883 documented this condition. The notification stated that on December 22, 2004, a planned system drain operation using the CRD room floor drains, resulted in water overflowing the floor drains and spilling onto the ‘B’ and ‘D’ CS pumps located below the CRD pump room in the reactor building. At the time of the event, the CS pumps were not required to be operable because the plant was shutdown. The inspectors reviewed Hope Creek’s corrective actions related to notification 20216883, which included testing of the motor windings to verify water did not impact the operation of the pump motors.

On August 23, 2004, the NRC’s Executive Director for Operations approved a deviation from the NRC’s Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20196105, 20210062, 20221918, and 20224132 related to floor drains that were found clogged or overflowed. The notifications were reviewed to determine if the drain problems resulted in an impact on risk significant and or safety related equipment. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

Introduction. The inspectors identified that PSEG did not correct a degraded condition associated with the CRD pump room floor access hatches and floor drains after the condition resulted in water leaking onto the ‘B’ and ‘D’ CS pumps in December 2004. This finding was of very low safety significance (Green) and determined to be a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.”

Description. On December 22, 2004, PSEG personnel noted that the CRD pump room floor drains were clogged during a planned system draining evolution. During the evolution water overflowed onto the CRD pump room floor and past the unsealed gaps of the CRD pump room floor access hatches located above the CS pump room, instead of draining to the reactor building sump. The water leaked onto the ‘B’ and ‘D’ CS pump motors. At the time of the evolution, the Hope Creek plant was shutdown and the CS pumps were not required to be operable by TS. Maintenance personnel immediately absorbed the water from the pump motors and performed motor insulation resistance checks to confirm that the pump motors were not damaged. PSEG entered the condition into the corrective action program in notification 20216883.

On February 16, 2005, while the plant was operating at power, the inspectors reviewed PSEG’s corrective actions for notification 20216883. The notification listed a corrective action to clear the CRD pump room floor drains. The action was assigned to the maintenance department to coordinate a contractor to clear the drains. The inspectors identified that the task was not assigned or scheduled. In addition, the inspectors identified that the floor access hatches to the CS pump rooms were not sealed or assigned to be sealed to protect the CS pumps from an internal flood until questioned by the inspectors. PSEG entered this condition into the corrective action program in notification 20224587. The floor access hatches were repaired on February 24, 2005, when PSEG maintenance personnel placed a sealant material around the openings between the floor and floor access hatch.

Analysis. The performance deficiency involved a failure to correct a degraded condition that existed for an extended period of time associated with the CRD pump room floor access hatches and floor drains. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements.

The finding was more than minor because it was associated with the protection against external factors (flood hazard) attribute of the mitigating systems cornerstone and affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The degraded equipment (floor hatch and floor drains) provided the potential for an internal flood in the CRD room to impact safety related equipment, 'B' and 'D' CS pumps, when they are required to be available. In accordance with Inspector Manual Chapter (IMC) 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase I SDP screening and determined the finding to be of very low safety significance (Green). The finding was of very low safety significance because the issue was not a design or qualification deficiency that resulted in a loss of function, did not result in an actual loss of safety function of a single train of equipment for greater then allowed by Technical Specifications, did not result in an actual loss of safety function of equipment considered risk significant in the maintenance rule program for greater than 24 hours, and the finding does not screen as potentially risk significant due to external events (i.e., seismic, flood, fire, or severe weather).

The "Seismic, Flooding and Severe Weather Screening Criteria" worksheet in the SDP Phase 1 worksheet was used to determine that the finding was not risk significant due to flooding. The finding does involve the degradation of equipment designed to mitigate flooding events, but it would not cause an initiating event, does not degrade more than one train of the core spray system, and it does not degrade a support system. The performance deficiency had a problem identification and resolution (identification and evaluation) cross cutting aspect.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to the above, between December 22, 2004, and February 24, 2005, PSEG did not promptly correct a condition adverse to quality associated with the control rod drive pump room unsealed floor access hatches and clogged floor drains . Because this issue is of very low safety significance and has been entered into PSEG's corrective action program in notifications 20216883 and 20224587, this violation is being treated as an NCV, consistent with

Section VI.A of the NRC Enforcement Policy. **(NCV 05000354/2005002-01, Control Rod Drive Pump Room Degraded Flood Barrier and Drains)**

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope (2 samples)

Annual Review. The inspectors verified acceptable heat exchanger performance by reviewing the results of two heat exchanger functional tests. Documents reviewed are listed in the Supplemental Information attachment to this report.

The inspectors verified that the ‘B2' safety auxiliaries cooling system (SACS) heat exchanger was within test acceptance criteria after routine testing was performed on February 20, 2005. The inspectors verified that a degraded condition did not exist by comparing the test results to previous test results.

On March 15, 2005, operations personnel identified degraded performance on the station service water system (SSWS) tube side of the ‘B1' SACS heat exchanger. The condition was identified during troubleshooting activities that required closing of isolation valve (EAHV-2204) for the ‘B’ SSWS supply to the reactor auxiliary cooling system (RACS) heat exchanger. The condition was validated through performance of the SACS heat exchanger performance test HC.OP-FT.EA-0001, “Validating SSWS Flow Through SACS Heat Exchangers.” The inspectors reviewed PSEG’s operability evaluation (70045601) of the issue to verify that the ‘B1' SACS heat exchanger remained operable. The inspectors verified that the compensatory measures listed in the evaluation were being appropriately performed and controlled. The inspectors also reviewed the test results of the ‘A1', ‘A2', and ‘B2' SACS heat exchanger which were performed between March 16 and 17, 2005, to ensure that a degraded condition did not exist in these heat exchangers.

On August 23, 2004, the NRC’s Executive Director for Operations approved a deviation from the NRC’s Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20218968, 20220454, 20229173, and 20217939 related to heat exchangers and chlorination issues. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope (1 resident staff sample)

Requalification Activities Review By Resident Staff. The inspectors observed one simulator training scenario on February 1, 2005, to assess operator performance and training effectiveness. The scenario involved a degraded main condenser vacuum due to in-leakage, followed by a main condenser isolation and reactor scram. The inspectors assessed simulator fidelity and observed the simulator instructor's critique of operator performance. The inspectors reviewed applicable documents associated with licensed operator requalification as listed in the Supplemental Information attachment to this report

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20228885, 20195458, 20194278, and 20210905 related to operator training issues. The notifications were reviewed to determine if the training issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (2 samples)

The inspectors reviewed two degraded conditions to assess the effectiveness of PSEG's performance monitoring and maintenance activities. Specifically, the inspectors reviewed the effectiveness of maintenance for the 'A' reactor feed pump flow oscillations and 'A' control area chill water (CACW) pump low flow trips. The inspectors also verified the affected systems were effectively being monitored in accordance with PSEG maintenance rule program requirements. The inspectors compared documented functional failure determinations and unavailable hours to those being tracked by PSEG to evaluate the effectiveness of PSEG's condition monitoring activities and determine whether performance criteria were met. The inspectors reviewed applicable work orders, corrective action notifications, preventive maintenance tasks, systems health reports, vendor documents, and operating procedures. Documents reviewed are listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20173664, 202007020, 20206786, 20212933, 20218982, 2022845, 20223692, and 20225686 related to maintenance effectiveness and maintenance rule issues. The notifications were reviewed to determine if the issues resulted in an impact to risk significant and or safety related equipment. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

Introduction. A self-revealing non-cited violation was identified for a recurring failure of the ‘A’ CACW pump. The ‘A’ CACW pump had malfunctioned about three weeks prior for a similar reason (air binding), but corrective actions were not effective at maintaining the pump’s reliability and availability. This finding was of very low safety significance (Green) and determined to be a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.”

Description. During RF12 PSEG personnel performed maintenance on the ‘A’ control area ventilation system. Part of the maintenance involved replacing the ‘A’ CACW evaporator vent valve (H1GJ-1-GJ-V027). This maintenance required the chilled water side of the evaporator be isolated and drained. The ‘A’ CACW pump was restored and placed in service on January 12, 2005, after experiencing many issues with the ‘A’ and ‘B’ CACW pumps tripping due to air intrusion, which was documented in corrective action program notification 20225777. The ‘A’ pump was placed in standby on January 16, 2005.

On February 1, 2005, operators performed a swap of control area ventilation trains. The ‘A’ CACW pump started normally, attained rated flow, experienced signs of cavitation and air-binding (fluctuating flow and cavitation noise) and tripped on low flow approximately 60 seconds after its initial start. PSEG corrective actions for this failure involved performing a fill and vent on the system in accordance with operating procedure HC.OP-SO.GJ-0001, “Control Area Chilled Water System Operation.” PSEG personnel completed a fill and vent of the system; however, the ‘A’ CACW pump again tripped on low flow. A second fill and vent of the system was performed and the ‘A’ CACW pump started without incident. The ‘A’ CACW pump was placed in standby on February 8, 2005. PSEG documented this equipment malfunction in corrective action program notification 20222457.

On February 24, 2005, operators performed a swap of control area ventilation trains. The ‘A’ CACW pump started normally, attained rated flow, experienced signs of cavitation and air-binding (fluctuating flow and cavitation noise) and tripped on low flow approximately 30 seconds after the initial start. PSEG engineers investigated this recurring issue further and identified that the system needed to be more throughly vented using all the available high point vents. PSEG operators performed a fill and vent of the system based on the revised guidance provided by engineering personnel.

The ‘A’ CACW pump was placed in service and declared “operable, but degraded” on February 26. PSEG documented this equipment malfunction in corrective action program notification 20225777.

PSEG’s evaluation (order 70045062) identified the cause of the pump trips to be attributed to not “aggressively” filling and venting the system after maintenance during RF12. The inadequate filling and venting of the system during RF12 allowed air to remain in the system and eventually come out of solution and collect in various high points throughout the system during periods when the ‘A’ train was in a standby alignment and impacting pump operations. PSEG’s review of the operating procedure concluded that the procedure contained insufficient fill and vent guidance to ensure air did not remain in the system following maintenance. PSEG’s corrective action included revising the fill and vent section of the operating procedure.

Analysis. The performance deficiency involved a failure to properly evaluate previous CACW pump trips due to a low flow condition and develop effective corrective actions. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC’s regulatory function and was not the result of any willful violation of NRC requirements.

The finding was more than minor because it was associated with the equipment performance attribute of the initiating events cornerstone and affected the objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a Phase I SDP screening and determined that the finding contributed to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This required that a Phase 2 SDP analysis be performed.

The Region I Senior Risk Analyst (SRA) performed a modified Phase 2 analysis and determined that the issue was of very low safety significance (Green). The chilled water system consists of two redundant trains. Both trains provide cooling to the ventilation system for each 4160 volt safety-related switchgear room and to the control room emergency filtration (CREF) system. If both trains of chilled water fail the TS requires a plant shutdown, because of the lack of cooling to the CREF system and switchgear room ventilation operating procedures directs the opening of doors and installation of temporary fans within 12 hours. The SRA assumed that the loss of chilled water would act as a plant transient initiator at a frequency of 1 in 1,000 years of operation, based on the redundancy in the system. Using the transient initiating event table from the Hope Creek Risk Informed Inspection Notebook. The SRA assumed the following:

- The power conversion system safety function would not be affected by the chilled water issue;
- If the power conversion system failed independently, along with the loss of both trains of chilled water the doors would need to be open to ensure the

functionality of the 4160 volt dependent systems (containment heat removal and low pressure injection); and

- The failure probability of containment heat removal and low pressure injection would be limited by a 1 in 100 chance that operators do not open doors to ensure switchgear ventilation.

Given the assumptions and an exposure time of 23 days, the SRA estimated an increase in core damage frequency several factors of 10 below the threshold of 1 in 10,000,000 years, above which external initiating events and the increase in large early release frequence need to be addressed. The performance deficiency had a problem identification and resolution (evaluation and corrective action) cross cutting aspect.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires that measures shall be established to assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to the above, from February 1 to 24, 2005, PSEG failed to properly identify and correct a malfunction associated with the ‘A’ CACW pump to maintain the pump reliable and available when needed. However, because this finding is of very low safety significance and has been entered into the PSEG’s corrective action program (notifications 20222457 and 20225777), this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000354/2005002-02, Inadequate Corrective Action for A Control Area Chilled Water Pump)**

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

a. Inspection Scope (5 samples)

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

- ‘A’ residual heat removal unavailable during planned maintenance on SACS valve 1EGHV-2512A on January 23, 2005;
- ‘B’ standby liquid control and ‘B’ SACS water pump out of service for planned maintenance on February 16, 2005;
- Unplanned unavailability of the ‘A’ control room emergency filtration train and planned unavailability of an instrument air dryer (1AF104) on February 25, 2005;
- ‘B’ SSWS pump and ‘D’ switchgear room cooler on March 21, 2005; and
- Planned maintenance on the ‘B1' SACS heat exchanger on March 24, 2005.

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 corrective action program notifications documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed are listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20223905, 20225461, 20225836, 20229384, 20177359, 20173664, 20212932, 20200720, 20218982, 20206786, 2026845, 20223692, and 20225686 related to risk assessment issues. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (3 samples)

The inspectors evaluated PSEG's performance and response during three non-routine evolutions to determine whether the operators' response was consistent with applicable procedures and training. The inspectors observed control room activities and/or reviewed control room logs to assess operator performance. PSEG's evaluations of operator performance were also reviewed. The inspectors walked down control room displays and portions of plant systems to verify status of risk significant equipment and interviewed operators and engineers. Documents reviewed are listed in the Supplemental Information attachment to this report. Operator performance during the following non-routine evolutions were reviewed.

Turbine Digital Electro Hydraulic Control (DEHC) Testing. During RF12 PSEG upgraded the electro hydraulic control (EHC) system for the main turbine to a DEHC system. The DEHC system modulates turbine control/bypass valves to maintain turbine speed/load and reactor pressure. The DEHC system also provides turbine protective trips when conditions require. An infrequently performed test or evolution plan (IPTE) was utilized to control the post-maintenance testing of the DEHC system by PSEG, because it created a potential reactivity concern if the system did not operate as designed. The inspectors observed portions of this IPTE from the control room during startup activities from RF12.

'A' Reactor Feed Pump (RFP) Flow Oscillations. On March 11, 2005, operators experienced reactor power and water level fluctuations. The inspectors responded to the control room when they became aware of the condition. The inspectors validated that operators appropriately entered applicable abnormal operating procedures. In response to the condition, operators reduced reactor power to 95% and placed the ‘A’ RFP in manual control in accordance with applicable abnormal operating procedures. The ‘A’ RFP speed continued to fluctuate while in manual control and operations personnel manually reduced speed until the RFP was no longer supplying water to the reactor. Power was further reduced to 90% in response to elevated bearing temperatures on the ‘B’ and ‘C’ RFPs. PSEG convened an operational challenge response (OCR) team to investigate the issue. The inspectors reviewed PSEG’s work activities, OCR team documentation, corrective action program notifications, and control room narrative logs.

Reactor Recirculation Pump Vibration Monitoring. During routine plant status activities the inspectors monitored reactor recirculation pump performance; and verified that operators were able to properly operate the reactor recirculation pump vibration monitoring equipment used to respond to vibration alarms and implement commitments to NRC Confirmatory Action Letter (CAL) 1-05-001. The inspectors also reviewed operations and engineering department personnel’s response to seven vibration alarms on the ‘B’ reactor recirculation pump and five vibration alarms on the ‘A’ reactor recirculation pump between January 1 and March 31, 2005. The alarm conditions were documented in notifications 20223468, 20228138, 20230224, 20230304, and 20230303. The inspectors verified that operators properly responded to these alarms in accordance with alarm response procedure HC.OP-AR.ZZ-0008, “Overhead Annunciator Window Box C1,” and abnormal procedure HC.OP-AB.RPV-0003, “Recirculation System.” The inspectors also verified that engineering personnel evaluated the conditions in accordance with engineering procedure HC.ER-AP.BB-0001, “Reactor Recirculation Pump/Motors Vibration Monitoring.” The inspectors, with assistance from a mechanical engineer in the Office of Nuclear Reactor Regulations (NRR), Division of Engineering, reviewed PSEG’s evaluation of the alarm conditions which concluded, in each case, the condition experienced was not representative of shaft cracking.

On August 23, 2004, the NRC’s Executive Director for Operations approved a deviation from the NRC’s Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20184412, 20188103, 20220948, and 20221648 to determine whether issues identified during nonroutine plant evolutions were being properly evaluated and corrected.

b. Findings

Introduction. The inspectors identified that the control room operators were not able to properly operate the reactor recirculation pump vibration monitoring equipment used to respond to vibration alarms and implement commitments in NRC Confirmatory Action Letter (CAL) 1-05-001. The finding was not a violation of NRC requirements, in that, the performance deficiency was related to operation of non-safety related equipment.

Description. On January 11, 2005, the NRC issued CAL 1-05-001 to PSEG which outlined actions that PSEG committed to perform to ensure acceptable operation of the 'B' reactor recirculation pump. The CAL specified that PSEG would implement a vibration monitoring program to continuously monitor the 'B' reactor recirculation pump's primary and secondary harmonic parameters (total amplitude, 1x and 2x amplitude and phase angle). Operational experience has shown that the onset of shaft cracking can be identified by changes in the primary (1x) and secondary (2x) harmonic parameters. In addition, the CAL stated that the monitoring program would include objective criteria that would demonstrate monitored parameters were within acceptable range and procedures would be established which specified actions to be taken if the monitored parameters were outside of the specified pre-established alarm points.

PSEG installed a new vibration monitoring system per design change package (DCP) 80077512 to continuously monitor reactor recirculation pump parameters. The DCP included the addition of digital outputs to allow for continuous monitoring and provide alarms in the control room for necessary operator actions (lower pump speed or remove the pump from service) in accordance with alarm response and abnormal operating procedures. As part of the DCP a computer (System I) was installed in the control room which allows operators to verify and validate pump parameters.

On January 28, 2005, during plant status activities the inspectors asked operators to display the 'B' reactor recirculation pump parameters on the System I computer in the control room to assess the operators' proficiency in utilizing the new equipment that was installed and required to be accessed in alarm response procedure HC.OP-AR.ZZ-0008. The inspectors identified that operators were unfamiliar with operating the System I computer and unclear on how to directly read pump parameters to confirm the validity of alarms, if needed. The inspectors discussed this observation with operations personnel and questioned the adequacy of the training that the operators previously received and their ability to perform the necessary steps contained in alarm response and abnormal operating procedures. PSEG operations personnel investigated this issue and instituted written directions on operating the System I computer for the operators.

On February 3, 2005, the inspectors asked operators to display reactor recirculation pump parameters on the System I computer in order for the inspectors to review reactor recirculation pump data. The inspectors identified for a second time that operators were unfamiliar with operating the System I computer and unclear on how to directly obtain the pump parameters from System I. The inspectors discussed this observation with operations personnel and questioned the adequacy of the corrective actions that were implemented from the issue identified on January 28, 2005. PSEG operations personnel investigated this issue and confirmed the written directions on operating the System I computer were not effective in addressing the issue from January 28, 2005, and documented this in corrective action notification 20222800.

In response to the inspectors concerns, PSEG issued night order HC-2005-28, which provided operators with enhanced guidance on how to obtain the pump's harmonic parameters; and PSEG performed "just-in-time" training for the operating crews.

Analysis. The performance deficiency involved a failure of operators to demonstrate they could properly operate the reactor recirculation pump vibration monitoring equipment, if needed, to respond to vibration alarms and implement procedures listed in NRC CAL 1-05-001. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements.

The finding was more than minor because if the condition was left uncorrected the finding would become a more significant safety concern. A failure to retrieve recirculation pump data in a timely manner would result in a delay in executing "time critical" alarm response and abnormal operating procedure actions. These actions are designed to detect a potential crack in the reactor recirculation pump shaft in time to take appropriate actions to reduce pump speed and/or remove the pump from service prior to a complete shaft failure. The finding is not suitable for SDP evaluation because it did not have an actual impact on the initiating events, mitigating systems, or barrier integrity cornerstone. This finding has been reviewed by NRC management and was determined to be a finding of very low safety significance (Green). The finding is of very low safety significance because there were no alarm conditions present at the time of the discovery. The performance deficiency had a problem identification and resolution (corrective action) cross cutting aspect.

Enforcement. The finding was not a violation of NRC requirements, in that, the performance deficiency was related to operation of non-safety related equipment. **(FIN 05000354/2005002-03, Inability to Properly Operate Reactor Recirculation Vibration Monitoring Equipment)**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (7 samples)

The inspectors reviewed seven operability evaluations associated with:

C Control rod 38-55 fast withdraw speed (70044113);
- Potential over-pressurization of reactor core isolation cooling discharge piping (20220879) ;
- 'B' reactor recirculation pump high vibration alarm (20223468);
- Reactor recirculation piping displacement exceeded Level I monitoring threshold for the 'B' reactor recirculation ring header between N2E jet pump riser and end cap (node 39) (20223538);
- 'B' EDG over excitation trip during twenty-four hour endurance surveillance test (20224537);
- Susceptibility review of emergency bus design with respect to a Crystal River event report #41362 (20222533); and
- Scram pilot solenoid valve nonconforming diaphragm material (70044669).

The inspectors reviewed operability evaluations to ensure the conclusions were technically justified to support operability of the system and/or component. When the operability evaluation involved compensatory measures, the inspectors verified the measures were in place and appropriately controlled. The inspectors also walked down accessible equipment to corroborate the adequacy of PSEG's operability evaluations. Documents reviewed are listed in the Supplemental Information attachment to this report.

Additionally, the inspectors reviewed the operability evaluation (20223538) performed by design engineering, including the analysis of piping vibration measurements, when vibration levels on the 'B' reactor recirculation ring header exceeded a vibration action limit for displacement on node 39. The inspectors verified that the conclusions were technically justified. The inspectors reviewed the actions taken by PSEG in response to displacement measurements at node 39, the revised acceptance criteria guidance provided by General Electric (GE), and how this guidance was incorporated into procedure HC.ER-AP.BB-0002, "Hope Creek Reactor Recirculation Piping Vibration Monitoring." The inspectors also reviewed the vibration data collected between February 11 to 13, 2005, from the reactor recirculation piping monitoring program, to ensure that the data was within the revised acceptance criteria provided by GE.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20222533, 20225957, 20229031, and 20229497 related to operability evaluation issues. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope (1 cumulative sample)

The inspectors performed one cumulative review of PSEG's identified operator workaround (OWA) conditions during the week of February 28 and March 21, 2005. The inspectors reviewed PSEG's lists of operator burdens/concerns, temporary modifications, and operability determinations to assess the potential for these to impact the operators ability to properly respond to a plant transient or accident condition. In addition, the inspectors reviewed PSEG's lists of deficient control room computer points and locked in overhead annunciators to determine if operators were able to adequately identify degraded plant equipment and plant conditions based on a reduction of indication in the control room. The inspectors reviewed operator logs and control room instrument panels to evaluate potential impacts on operators' ability to implement abnormal and emergency operating procedures. The inspectors also toured the plant and control room to identify potential workarounds or deficiencies not previously identified by PSEG. Documents reviewed are listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20228820, 20229038, 20175556, 20221313, 20064153, 20223753, and 20221209 related to issues that are operator concerns and/or burdens. The notifications were reviewed to determine if the issues could result in an impact to the operation of the plant. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a Inspection Scope (13 samples)

Biennial Review. The inspectors reviewed selected permanent plant modification packages to verify that the design bases, licensing bases, and performance capability of risk significant structures, systems and components (SSCs) had not been degraded through plant modifications.

Plant changes were selected for review based on risk insights for the plant. The inspectors performed walkdowns of selected plant systems and components, interviewed plant staff, and reviewed applicable documents, including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action program documents, the UFSAR, and TSs.

The inspectors verified that selected attributes (component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument set-points, uncertainty calculations, electrical coordination, electrical loads analysis, and equipment environmental qualification) were consistent with the design and licensing bases. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For each modification, the 50.59 evaluations or safety evaluation screenings were reviewed as described in section 1R02 of this report. The inspectors verified that procedures, calculations, and the UFSAR were properly updated with revised design information and operating guidance. The inspectors also verified that the as-built configuration was accurately reflected in the design documentation and that post-modification testing was adequate to ensure the SSC would function properly.

The inspectors also reviewed issues that had been entered into the corrective action program to determine if PSEG had been effective in identifying problems associated with the plant modification process and activities. Samples of these issues were selected for further review during which the inspectors assessed the adequacy of the corrective actions which had been implemented for the selected issues. A listing of documents reviewed is provided in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for in the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed additional corrective action notifications 20199446, 20196273, and 20156362, related to problems associated with implementing permanent plant modifications. The completed and planned corrective actions were reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (4 samples)

The inspectors observed portions of and/or reviewed the results of four post-maintenance tests (PMTs) for the following equipment:

- 'H' EDG fuel oil transfer pump following scheduled maintenance on February 2;
- 'B' EDG following repair of combustion air o-ring leak on February 18;
- 'A' CACW pump following unexpected trip during routine swap on February 24; and
- 'C' SSWS pump strainer backwash valve on March 12.

The inspectors verified that the PMTs conducted were adequate for the scope of the maintenance performed. The inspectors also reviewed applicable documents associated with PMTs as listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20184011, 20202506, 20219756, 20223238, and 20225625 to determine whether issues identified during PMTs were being properly evaluated and corrected.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope (2 samples)

The inspectors monitored PSEG's activities associated with the refueling and planned outage activities described below. Documents reviewed for these activities are listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors periodically reviewed the following work orders that were deferred from RF12 to assess the technical justification for deferral of the work and assessed potential impact on the associated equipment. The work orders reviewed are listed in the Supplemental Information attachment to this report.

Refueling Outage. Hope Creek started the inspection period in operational condition 4, "Cold Shutdown," with the plant in RF12 that began on October 18, 2004. PSEG management decided to transition directly from the Hope Creek forced outage that began on October 10, 2004, to the refueling outage. The inspectors reviewed the refueling outage plan, monitored shutdown activities, PSEG's control of outage activities, and observed refueling activities in NRC Inspection Report 05000354/2004005. During the remainder of the refueling outage the inspectors reviewed PSEG's control of outage activities listed below and monitored heatup and startup activities. In addition, the inspectors verified that PSEG managed the outage risk commensurate with their outage plan.

The inspectors confirmed on a sampling basis that tagged equipment was properly controlled and equipment configured to safely support maintenance work. Equipment work areas were periodically observed to determine whether foreign material exclusion boundaries were adequate. During control room tours, the inspectors verified that operators maintained adequate reactor vessel level and temperature instruments and that indications were within the expected range for the operating mode.

The inspectors determined whether offsite and onsite electrical power sources were maintained in accordance with TS requirements and consistent with the outage risk assessment. Periodic walkdowns of portions of the onsite electrical buses and the EDG were conducted during risk significant electrical configurations to confirm the equipment alignments met requirements. The inspectors verified through routine plant status activities whether the decay heat removal safety function was maintained with appropriate redundancy as required by TS and consistent with PSEG's outage risk assessment. The inspectors verified that flow paths, configurations, and alternative means for inventory control were consistent with the outage risk assessment.

The inspectors performed an inspection and walkdown of the drywell (containment) prior to containment closure on January 11, 2005. The inspectors walked down the drywell to verify there were no evidence of leakage, tags were released, no evidence of obvious damage to passive systems, and no debris was left which could affect equipment in the drywell.

The inspectors monitored restart activities to ensure that required equipment was available for mode changes, including verifying that TSs, licensed conditions, and procedural requirements for mode changes were met prior to changing modes. The inspectors observed portions of startup activities from the control room to assess operator performance. The inspectors verified that unidentified and identified leakage values were within expected values and TS requirements. The inspectors verified that containment integrity was established prior to entering operational condition 3, "Hot Shutdown."

Finally, the inspectors monitored the operation of 'B' reactor recirculation pump during restart of Hope Creek. During plant status activities the inspectors validated vibration parameters, including 1x and 2x amplitude and phase angle, were within expected/predicted values. The inspectors verified that PSEG was properly implementing procedure HC. ER-AP.BB-0001, "Hope Creek Reactor Recirculation Pumps/Motors Vibration." The inspectors, with assistance from a mechanical engineer in NRR, reviewed PSEG's setpoint calculations and evaluations associated with the determination of reactor recirculation critical operating speeds.

Planned Outage. On March 26, 2005, operators commenced a planned down power to remove the main turbine from service to investigate the source of unidentified leakage in the drywell. During drywell walkdowns PSEG personnel identified a steam plume from the 'B' reactor recirculation decontamination connection piping. Upon discovery of this leak PSEG took the plant to cold shutdown. The inspectors observed the shutdown and portions of the cooldown process from the control room. The inspectors also monitored PSEG's controls over the outage activities listed below.

The inspectors determined whether cooldown rates during the plant shutdown met TS requirements. The inspectors performed a walkdown of the drywell on March 29, 2005, to inspect the 'A' and 'B' reactor recirculation decontamination connections as well as other equipment and piping for indications of leakage. The inspectors verified that PSEG managed the outage risk commensurate with their outage plan.

The inspectors confirmed on a sampling basis that tagged equipment was properly controlled and equipment configured to safely support maintenance work. Equipment work areas were periodically observed to determine whether foreign material exclusion boundaries were adequate. During control room tours, the inspectors verified that operators maintained adequate reactor vessel level and temperature instruments and that indication were within the expected range for the operating condition. The inspectors verified through routine plant status activities that the decay heat removal safety function was being maintained.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (5 samples)

The inspectors observed portions of and/or reviewed test data from the following five surveillance tests:

- A residual heat removal (RHR) pump in-service test (IST) on January 10, 2005;
- 18-month reactor building integrity functional test on January 13, 2005;
- CRD insertion and withdraw speed test, adjustment and stall flows functional test on January 16, 2005;
- 18-month high pressure coolant injection low pressure functional test on January 19, 2005; and
- TS unidentified (floor drain) leakage monitoring on February 8, 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 TS requirements and the UFSAR. The inspectors verified that test acceptance criteria were achieved. The inspectors also reviewed applicable documents and notifications documenting deficiencies associated with surveillance testing as listed in the Supplemental Information attachment to this report.

The inspectors monitored PSEG's investigation into an increase in unidentified drywell leakage between February 8, 2005, and March 26, 2005. The inspectors independently trended unidentified leak rates to assess PSEG's actions in response to an increasing level in unidentified leakage in accordance with NRC Inspection Manual 2515, Appendix D, "Plant Status." The inspectors verified that PSEG took the appropriate actions in accordance with abnormal procedure HC.OP-AB.CONT-0006, "Drywell Leakage," and leak investigation procedure HC.OP-GP.ZZ-0005, "Drywell Leakage Source Detection," when unidentified drywell leakage exhibited an increasing adverse trend. The inspectors also reviewed drywell floor drain sump and atmospheric samples to assess the potential source of the leakage.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20172803, 20183468,20203823, 20211229, 20225073, 20220759 and 20220760 to determine whether issues identified during surveillance and/or functional testing were being properly evaluated and corrected.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

The inspectors reviewed the following temporary plant modification (TM):

- 'C' SSWS pump strainer backwash valve thermal overload change (TM 05-010).

The inspectors verified the modification was consistent with the design and licensing bases of the affected system, and the performance capability of the system was not degraded by the modification. The inspectors reviewed the modification to verify applicable TS operability requirements were met during installation. The inspectors also reviewed applicable documents associated with the TM as listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20210303, 20173605, 20228044, 20224643, 20221234, 20227776, 20228353, and 20221500 related to TMs. The notifications were reviewed to determine if the TM issues resulted in an impact on risk significant and or safety related equipment. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors observed one emergency preparedness (EP) drill from the control room simulator and the emergency operations facility (EOF) on February 2, 2005. The inspectors evaluated the conduct of the drill, performance related to developing classifications, corrective action program notifications, and protective action recommendations by PSEG personnel. The inspectors reviewed EP Training Drill Critique Report H05-01 to evaluate the adequacy of PSEG's drill critique. Corrective action program notification 20221636 documenting EP weaknesses and deficiencies identified during the drill was also reviewed. Additional applicable documents were reviewed associated with the EP drill are listed in the Supplemental Information attachment to this report.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the inspectors reviewed notifications 20203915, 20180600, and 20202267 related to EP issues. The notifications were reviewed to determine if the issues could impact PSEG's response to an event. The completed and planned corrective actions were also reviewed to determine if the problems were being addressed in an appropriate time frame.

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)

The inspectors reviewed radiation work permits (RWPs) for airborne radioactivity areas with the potential for individual worker internal exposures of greater than 50 mrem committed effective dose equivalent (CEDE) (20 DAC-hrs). The inspectors also verified the adequacy of barrier integrity and engineering controls performance in those areas.

The inspectors reviewed and assessed the adequacy of PSEG's internal dose assessment for any actual internal exposure greater than 50 mrem CEDE.

Based on PSEG's schedule of work activities, the inspectors selected two jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (greater than 1 R/hr) for observation. The inspectors reviewed radiological job requirements for RWPs and observed job performance with respect to these requirements. The inspectors verified that radiological conditions in the work areas were being adequately communicated to workers through briefings and postings.

During job performance observations, the inspectors verified the adequacy of radiological controls, such as: required surveys (including system breach radiation, contamination, and airborne surveys), radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. In addition, the inspectors observed radiation protection technicians performance with respect to radiation protection work requirements to ensure that the technicians were aware of the radiological conditions and the RWP controls/limits associated with work activities, and performance was consistent with training and qualifications.

For high radiation work areas with significant dose rate gradients (factor of five or more), the inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel and verified that PSEG's controls were adequate.

b. Findings

No findings of significance were identified.

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/estimated exposure that were in progress or recently completed during RF12, and selected the three highest exposure significant work activities (control rod drive replacement, in-service inspection, and N2K nozzle repair).

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

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

b. Findings

Introduction. A self-revealing finding was identified when work activities during RF12 in the general torus and torus room exceeded its collective dose estimate by 312%. PSEG did not evaluate the expanded work scope that occurred in these areas for dose minimization. The finding was not a violation of NRC requirements, because overall exposure performance of the nuclear power plant is used to determine compliance with the ALARA rule.

Description. During RF12, PSEG exceeded its collective dose estimate for work activities performed in the general torus and torus room by 312%. Personnel working in these areas received a collective exposure of 6.48 person-rem against an estimate of 1.564 person-rem. During RF12, PSEG added additional work activities to be performed in the general torus and torus room without assessing the additional work for potential engineering controls to minimize exposure once the established dose goal had been exceeded.

Analysis. The performance deficiency involved a failure to evaluate the expanded scope of work, eliminating the potential for exposure minimization through the implementation of additional dose mitigation techniques. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements.

This finding was more than minor because it was associated with the ALARA planning attribute of the occupational radiation safety cornerstone and affected the objective to ensure the adequate protection of worker health and safety from exposure of radiation from radioactive material during routine civilian nuclear reactor operations. This finding was similar to more than minor example 6.a in NRC Inspection Manual 0612, Appendix E, "Examples of Minor Issues," in that, the actual dose achieved exceeded the planned, intended dose by more than fifty percent. This finding was evaluated using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process" because the issue involved ALARA. The inspectors determined the finding to be of very low safety significance (Green) because Hope Creek's three-year-rolling-average (2001-2003) is 126 person-rem, which is below the SDP criteria of 240 person-rem for Boiling Water Reactors (BWRs). The performance deficiency had a problem identification and resolution cross cutting (identification) aspect.

Enforcement. The finding was not a violation of NRC requirements. The ALARA rule contained in 10 CFR 20.1101(b) Statements of Consideration indicates that compliance with the ALARA requirement will be judged on whether PSEG has incorporated measures to track and, if necessary, to reduce exposures and not whether exposures and doses represent an absolute minimum or whether PSEG has used all possible methods to reduce exposures. The overall exposure performance of the nuclear power plant is used to determine compliance with the ALARA rule. Since this issue is an isolated occurrence and the remainder of the work performed during the Hope Creek refueling outage was in compliance with the ALARA rule, no violation of 10CFR20.1101(b), "Radiation Protection Programs," has occurred. **(FIN 05000354/2005002-04, Work Activities During RF12 Exceeded Collective Dose Estimate)**

2OS3 Radiation Monitoring Instrumentation (71121.03)

a. Inspection Scope (1 sample)

The inspectors verified the calibration, operability, and alarm setpoints of several types of radiation monitoring instruments and equipment. The inspectors determined what actions were taken by PSEG when during calibration or source checks an instrument was found significantly out of calibration (>50%). The inspectors determined possible consequences of instrument use since last successful calibration or source checks and verified that out of calibration results were entered into the corrective action program.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope (1 Sample)

The inspectors reviewed the circumstances surrounding the shipment of contaminated outage equipment from Hope Creek to Framatome ANP in Lynchburg, Virginia, on December 22, 2004.

b. Findings

Introduction. A self-revealing non-cited violation was identified when a PSEG shipment of outage related equipment received by a vendor had external radiation levels in excess of regulatory limits. This finding was of very low safety significance (Green) and determined to be a violation of 10 CFR 71.5, "Transportation of Licensed Material" and 49 CFR 173.441(a), "Radiation Level Limitations."

Description. On December 22, 2004, PSEG shipped seven boxes containing contaminated RF12 equipment (shipment HC 04-151) via exclusive use on a flat bed trailer to Framatome in Lynchburg, VA. The shipping manifest documented that the maximum contact radiation level measured by PSEG personnel on the exterior of any package was 25 millirem per hour. Additional documentation indicated that the maximum contact radiation level with any individual item in any package did not exceed 60 millirem per hour (survey performed after all items had been secured within the packaging).

On December 23, 2004, personnel at the Framatome ANP facility in Lynchburg, Virginia, conducted a receipt survey of the shipment, and determined that the radiation level on contact with one portion of the bottom of one package was 236 millirem per hour. This level exceeded the Department of Transportation (DOT) regulatory limit of 200 millirem per hour, as specified in 49 CFR 173.441(a). Framatome placed the unopened package in a secured location within its facility, and informed PSEG. At the request of PSEG, the package was opened by Framatome personnel, and the individual contents surveyed. One object was determined to have a contact radiation level of 800 millirem per hour and none of the package contents appeared to have shifted during transport. PSEG documented this issue in the corrective action program under notification 20216990, and temporarily suspended radioactive material shipping activities.

Analysis. The performance deficiency involved transporting a radioactive material package with radiation levels in excess of the regulatory limit while in the public domain. Traditional enforcement does not apply because the issue did not have any actual safety consequence or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements.

This finding was more than minor because it was associated with the program and process attribute of the public radiation safety cornerstone and affected the objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operations. PSEG did not ensure that a shipment had surface radiation levels below DOT regulatory limits while in the public domain during transport. Hope Creek personnel failed to meet the requirements of DOT for ensuring that the radiation level on the surface of the package did not exceed 200 millirem per hour at any time during transport. This finding was evaluated using IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process" because it was a radiation material control (radioactive material packaging and transportation) issue. The inspectors determined the finding to be of very low safety significance (Green) because the transportation issue resulted in a radiation limit being exceeded that involved external radiation levels that were not readily accessible by the public and not more than two times the federal limit. The inspectors also determined that the finding did not involve a breach in the package, a certificate of non-compliance issue, a low level burial ground non-conformance, and that surface contamination limits were not exceeded.

Enforcement. 10 CFR 71.5 requires that NRC licensees ship radioactive materials in accordance with the applicable provisions of the DOT regulations found in 49 CFR 100-177. 49 CFR 173.441(a) requires that each package of radioactive material offered for transport must be designed and prepared for shipment so that under conditions normally incident to transportation, the radiation level does not exceed 200 millirem per hour at any point on the external surface of the package. Contrary to this requirement, a package shipped by Hope Creek on December 22, 2004, arrived at the Framatome ANP facility on December 23, 2004, with contact radiation levels of 236 millirem per hour on the bottom external surface of the package. Because this finding was of very low safety significance (Green), and Hope Creek entered this finding into its corrective action program (notification 20216990), this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000354/2005002-05, PSEG Shipment to Vendor with External Radiation Levels In Excess of Regulatory Limits)**

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

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 review was accomplished by reviewing hard copies of each condition report, attending daily screening meetings, or accessing PSEG's computerized database.

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with this deviation, the following additional inspection samples were performed. The inspectors reviewed PSEG's Business Plan Initiative CAP.02.PS.01.01, "Enter/Verify in Corrective Action Program Corrective Actions From Self Assessments and Assists" and CAP.01.PS.02.02, "Assess Initial Senior Reactor Operator Operability Screening," to verify that adverse conditions were identified and entered into the corrective action program. The inspectors also reviewed progress in initiatives SCWE.01.OPS.02.14, "Review and revise as necessary training effecting the conduct of site activities, i.e., conduct of operations and maintenance, to ensure that the message of safety over production is consistently delivered across site programs" and SCWE.02.OPS.02.14, "Review all current problem identification processes, identify and discontinue any department specific problem reporting systems, ensure all problems identified through separately maintained systems are captured and processed in accordance with the site specific processes." The inspectors reviewed notification 20230806 related to PSEG's business plan on safety conscious work environment (SCWE) item progress. No findings of significance were identified.

1. Annual Sample Review (2 samples)

a. Inspection Scope

The inspectors reviewed PSEG's evaluation and corrective actions associated with the following two issues:

5014 Line Outage Results in Unplanned Downpower. The inspectors reviewed corrective action notification 20203787 associated with an unplanned downpower when the 5014 (Rock Springs - Peach Bottom) offsite power line was removed from service on

September 16, 2004. On September 16, 2004 the control room received alarms due to high temperatures on the main power transformers and main generator field. Shortly after receiving the alarms, operations personnel received communications from the generation desk that a power reduction was necessary for Hope Creek due to grid instabilities as a result of the planned 5014 line outage. Operators incrementally reduced power to 500 MWe at the request of the system operator. The Pennsylvania, New Jersey and Maryland Interconnection (PJM) authorized the removal of the 5014 line from service, however the PJM day-ahead study and PJM Transmission Log summary did not anticipate instabilities and the need for power reductions in the Hope Creek/Salem area. In addition, the real time studies performed by the PJM prior to the removing the 5014 line form service did not indicate the need for power reductions or predict an over voltage condition. As a result, drastic load reductions were required for the Hope Creek and Salem to control grid stability.

The inspectors reviewed PSEG's and PJM's corrective actions for this event to ensure the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors also interviewed the control room operators and visited the PJM facility to confirm they had instituted the necessary corrective actions to resolve this issue.

'B' EDG Load Wandering at Full Load Operations. The inspectors reviewed corrective action notifications and evaluations associated with 'B' EDG load wandering at full load operations. During an August 2000, monthly surveillance test, operators experienced slow load changes of several kilowatts (KW) above and below the operator selected load of 4000 KW. The inspectors noted that the load wandering has only been observed by operators when the EDG is in the droop mode of operations.

The inspectors reviewed PSEG's corrective actions for this issue to ensure the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors reviewed several consultants' and PSEG's evaluations on the potential causes (electrical and mechanical) of the load wandering by the 'B' EDG. The inspectors also reviewed the surveillance test results and the recorded history performance curves before and after the repair that was performed in December 2004 to ensure the issue had been adequately resolved. The inspectors also interviewed control room operators to confirm that the issue with the EDG no longer existed after the repair to the 'B' EDG's fuel rack linkage was completed.

b. Findings and Observations

No findings of significance were identified. The inspectors concluded that the 'B' EDG load wandering issue had been adequately addressed even though the issue existed for approximately four years before it was appropriately resolved.

2. Safety Conscious Work Environment Review

c. Inspection Scope

The inspectors reviewed PSEG's progress in addressing safety conscious work environment (SCWE) issues that were discussed in the NRC's recent annual assessment letter dated March 3, 2005. In that letter, the NRC staff documented a SCWE substantive cross cutting issue and also stated the NRC's intention to continue to monitor progress in this area.

The inspectors conducted a sampling review of PSEG's actions to improve the work environment on February 15 through 17, 2005. During the inspection, a limited number of interviews with PSEG personnel and 32 SCWE performance indicators (PIs) from the fourth quarter of 2004 were reviewed to assess progress since the last quarterly review. In the fourth quarter 2004, PSEG identified 14 PIs as being green (satisfactory) while 15 were identified as red (needs improvement) compared to the third quarter 2004 when 17 PIs were identified as green and 12 PIs as red. The inspectors did not identify any discernable performance improvement from the third quarter to the fourth quarter of 2004.

The inspectors review of the fourth quarter 2004 PIs showed that certain individual PIs were red at both Hope Creek and Salem stations. The PIs in this category included repeat maintenance and emergency diesel generator unavailability. The auxiliary feedwater and chemical volume control/safety injection system unavailability PIs were red for both Salem units. The inspectors reviewed PSEG's corrective actions to improve performance in these areas and noted that PSEG was applying additional resources in those areas in an attempt to improve performance.

Discussions between the inspectors and PSEG personnel focused on uncertainty about how the management changes implemented under the January 17, 2005, Nuclear Operating Services Contract would affect the work environment. Specifically, PSEG personnel acknowledged questions about the continued effectiveness of previous commitments to address work environment issues such as the use of the "People Team" and the Executive Review Board (ERB) to review certain employment actions.

d. Findings

No findings of significance were identified.

3. Executive Review Board Commitments

a. Inspection Scope

On August 23, 2004, the NRC's Executive Director for Operations approved a deviation from the NRC's Action Matrix to provide a greater level of oversight for the Hope Creek station than would typically be called for by the Action Matrix. One provision of the deviation memorandum provided for the enhancement of existing reactor oversight process baseline inspections. In accordance with the deviation memorandum, the inspectors performed a review to evaluate the effect the Nuclear Operations Service Contract (NOSC) between Exelon and PSEG had on the work environment at Salem and Hope Creek stations. Specifically, the inspectors reviewed corrective action program notification 20221830 which documents a failure to implement the ERB process.

b. Findings

The failure to implement the ERB process is unresolved pending further review by NRC staff.

In a January 28, 2004 letter to PSEG, NRC published interim results from its review of work environment issues at the Salem and Hope Creek Generating Stations. During subsequent public meetings with the NRC in March and June 2004, PSEG described its plan to address the work environment issues at the stations. PSEG further described this plan and committed to taking a number of actions to improve the work environment at the stations in a June 25, 2004 letter to the NRC.

In that letter, PSEG stated that an ERB had been established to review PSEG and contractor personnel actions to preclude retaliation and/or chilling effect at the stations. This action was taken to improve management effectiveness in detecting and preventing retaliation and the creation of a chilling effect. In addition, in this letter PSEG committed to providing to the NRC, on a quarterly basis, selected performance metrics related to safety conscious work environment. These metrics include a metric on ERB effectiveness. On July 30, 2004, in a letter to PSEG, NRC published the final results from its review of work environment issues at the stations and acknowledged that PSEG's June 25, 2004 letter appeared to address the key findings of both the NRC and PSEG assessments.

In December 2004, PSEG announced that it had entered into a Nuclear Operating Services Contract (NOSC) with Exelon to provide management services for plant operations at the Salem and Hope Creek Generating Stations. Prior to implementation of the NOSC on January 17, 2005, PSEG, in cooperation with Exelon, identified a number of personnel changes that would be necessary to implement the Exelon management model at the stations.

While onsite on January 7, 2005, an NRC Region I manager learned that the initial set of personnel actions associated with the NOSC had not been reviewed by the ERB. NRC management requested that PSEG explain why the personnel actions had been taken without being reviewed by the ERB. The NRC also requested that PSEG describe what actions they intended to take in order to accomplish the intended function of the ERB. During follow-up discussions with PSEG management, the NRC learned that several other personnel actions, not associated with implementation of the NOSC, had also occurred without being subjected to the ERB process.

In a letter dated January 31, 2005, PSEG notified the NRC of its intent to commission an independent review of those personnel actions related to the implementation of the NOSC to ensure that they complied with 10 CFR 50.7 "Employee Protection" requirements. While the NRC acknowledged PSEG's intention to perform this review, the NRC, in a letter dated February 17, 2005 requested a written response to specific items detailed in the enclosure to the letter. In a letter dated March 21, 2005, PSEG provided their response.

At the end of the inspection period the inspectors had performed an initial review of PSEG's response and concluded that a more detailed review of the information referenced in the PSEG's response was necessary. This issue is unresolved pending NRC's review of the information referenced in the PSEG response. **(URI 50-354/2005002-06, Failure to Implement the ERB Process)**

4. Cross-References to PI&R Findings Documented Elsewhere

Section 1R06 of this report describes a finding in which PSEG did not identify that the CRD pump room floor hatches needed to be sealed until identified by the inspectors. Additionally, PSEG did not properly apply the lessons learned from recent operating experience related to a similar event that occurred at the Susquehanna Nuclear Power Plant in August 2004 and recognize the condition as a potential internal flooding hazard which could impact safety related equipment.

Section 1R12 of this report describes a finding in which PSEG did not adequately evaluate low flow trips of the 'A' CACW pump due to air in the system after maintenance. Additionally, PSEG's initial corrective actions were ineffective because a similar condition occurred approximately three weeks later.

Section 1R14 of this report describes a finding in which PSEG did not implement adequate corrective actions to ensure operators were able to properly operate the reactor recirculation vibration monitoring equipment (System I) after PSEG personnel were made aware that operators could not effectively operate the System I computer.

Section 2OS2 of this report describes a finding in which PSEG failed to identify that activities being performed in the torus room exceeded pre-established radiation exposure goals.

4OA3 Event Followup (71153 - 3 samples)

5. (Closed) Licensee Event Report (LER) 05000354/2004009-00, As Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable

On November 9 and 19, 2004, PSEG determined that the as-found lift setpoint for five of fourteen main steam safety relief valves (SRV) failed to open within the required TS actuation pressure setpoint tolerance. TS 3.4.2.1 provides an allowable pressure band of +/- three percent for an individual SRV. All five of the SRVs opened above the required pressure band (actual range was +3.6 to +6.8 percent). PSEG determined that the apparent cause for the B, D, and F SRV setpoint failures was due to corrosion bonding/sticking of the pilot disc. The apparent cause of the 'A' and 'C' SRVs is still being currently reviewed by PSEG. The 'A' and 'C' SRVs are scheduled to be disassembled and inspected prior to the next refueling outage and are being tracked in PSEG's corrective action program under work order 70042439. All fourteen SRVs were replaced with tested and certified rebuilt spare pilot assemblies.

The performance deficiency involved a failure to ensure the SRV lift setpoints were within TS requirements. The finding was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and affected the objective to maintain the reliability of the SRVs to lift at their design setpoint. The LER described that the self actuating lift pressures for all SRVs were below the bounding analysis (1250 psig); therefore, the as-found test results identified that the SRVs would not have challenged the maximum analyzed pressure value, and there was no loss of safety function. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase I SDP screening and determined the finding to be of very low safety significance (Green). The finding was of very low safety significance because the issue was not a design or qualification deficiency that resulted in a loss of function, did not result in an actual loss of safety function of a single train of equipment for greater than allowed by technical specification, did not result in an actual loss of safety function of equipment considered risk significant in the maintenance rule program for greater than 24 hours, and the finding does not screen as potentially risk significant due to external events (i.e., seismic, flood, fire, or severe weather). This licensee-identified finding involved a violation of TS 3.4.2.1. The enforcement of licensee identified violations is discussed in Section 4OA7 of this report. This LER is closed.

6. (Closed) LER 05000354/2004010-00, Manual Reactor Scram Due to Moisture Separator Dump Line Failure

This LER described a manual reactor scram that occurred on October 10, 2004, due to a failure of an eight inch diameter moisture separator drain line to the main condenser. The NRC performed a special inspection to independently investigate operator and equipment performance during the event and assess PSEG's evaluations and corrective actions. The results of the inspection are presented in NRC Inspection Report 05000354/2004013 dated February 4, 2005. No new issues of significance were identified. This LER is closed.

7. 'B' Reactor Recirculation Decontamination Connection Leakage

a. Inspection Scope

On March 26, 2005, operators commenced a planned down power to investigate the source of unidentified leakage in the drywell. The unidentified leakage rate had been slowly increasing over several weeks and remained below PSEG's administrative limit and TS limit requiring a plant shutdown. On March 27, 2005, operators reduced power to 5% and entered operational condition 2, “Startup,” to perform walkdowns of the drywell. During the drywell walkdowns, PSEG personnel identified a steam plume from the ‘B’ reactor recirculation system decontamination connection piping. Upon discovery of the leak PSEG took the plant to cold shutdown. PSEG determined that the leak was coming from a through wall crack on a weld that connected the decontamination connection to the ‘B’ reactor recirculation piping. Hope Creek ended the inspection period preparing to return the plant to operations, conducting repairs to decontamination connections, and completing an evaluation into the cause of the leak.

PSEG submitted two event notifications which described PSEG’s actions associated with this issue. The inspectors reviewed event notification 41530, “Plant Shutdown to Repair Small Reactor Coolant System Leak,” and event notification 41536, “RCS Through Wall Leakage Identified at Welded Piping Junction,” on March 28, 2005. The inspectors reviewed these event notifications to ensure that PSEG properly characterized and operated the plant in accordance with TS 3.4.3.2, “Reactor Coolant System Operational Leakage.”

b. Findings

This issue is unresolved pending further NRC review of PSEG’s evaluation (70045989) of the causes of ‘B’ reactor recirculation decontamination connection leak and PSEG’s classification of the leak. **(URI 05000354/2005002-7, ‘B’ Reactor Recirculation Decontamination Connection Leakage)**

4OA5 Other

1. Reactor Recirculation Pump Vibration Monitoring Procedure Review

a. Inspection Scope

The inspectors reviewed the revisions to the reactor recirculation pump vibration monitoring procedures listed in the Supplemental Information attachment to this report to ensure that the proposed changes were consistent with maintaining the vibration monitoring program as described in NRC Confirmatory Action Letter 1-05-001.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

NRC/PSEG Management Meeting To Discuss Results of Special Team Inspection & Other Technical Isssues. The NRC conducted a meeting with PSEG on January 12, 2005, to discuss the results of a NRC’s special inspection into the circumstances surrounding the steam leak that occurred on October 10, 2004. The meeting also included a discussion of the issues associated with the B reactor recirculation pump and exhaust piping snubber failures for the high pressure coolant injection (HPCI) pump. PSEG provided a synopsis and status of corrective actions taken to address the issues discussed. The meeting occurred at the Bridgeport Holiday Inn (Swedesboro, New Jersey) and was open for public observation. A copy of the slide presentations can be found in ADAMS under accession number ML050120400.

Resident Inspector Exit Meeting. On April 1, 2005, the inspectors presented inspection results to members of PSEG management led by Mr. George Barnes. None of the information reviewed by the inspectors was considered proprietary.

4OA7 Licensee-Identified Violations.

The following violation of very low significance (Green) was identified by PSEG and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 3.4.2.1, "Safety/Relief Valves," requires that 13 of the 14 SRVs open within a lift setpoint of +/- 3 percent of the specified code safety valve function lift setting. Contrary to this requirement, PSEG identified that 5 of 14 SRVs experienced setpoint drift outside of the TS limit. PSEG entered this issue into their corrective action program as notification 20210558. This finding is of very low safety significance because the SRVs would have functioned to prevent a reactor vessel over-pressurization.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
G. Barnes, Site Vice President
R. Bhai, System Engineer
J. Bisti, Engineering
D. Boyle, Operation Manager
T. Cellmer, Radiation Protection Manager
J. Clancy, Radiation Protection and Chemistry Support Manager
G. Daves, Electrical & I&C Supervisor, System Engineering
J. Dower, Hope Creek Training Supervisor
T. Fowler, Control Room Supervisor
J. Frick, Shipping Supervisor
T. Hendricks, Control Room Operator
C. Johnson, Valve Engineer
P. Koppel, Component Engineer
E. Martin, System Engineer
M. Massaro, Hope Creek Plant Manager
G. Modi, Engineering
J. Nagle, Regulatory Compliance Supervisor
C. Perino, Director, Regulatory Assurance
B. Sebastian, Radiation Protection Manager
D. Sourber, Control Room Operator
M. Tadjalli, Design Engineering Manager
B. Thomas, Sr. Licensing Engineer
P. Tocci, Hope Creek Maintenance Manager
B. Udall, Licensing
J. Williams, Engineering Director

PJM Personnel
F. Koza, General Manager, Regional Operations
D. Souder, Chief System Operator

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000354/2005002-06	URI	Failure to Implement the ERB Process (Section 4OA2.3)
05000354/2005002-07	URI	'B' Reactor Recirculation Decontamination Connection Leakage (Section 4OA3.3)

Opened/Closed

05000354/2005002-01	NCV	Control Rod Drive Pump Room Degraded Flood Barrier and Drains (Section 1R06)
05000354/2005002-02	NCV	Inadequate Corrective Action For A Control Area Chilled Water Pump (Section 1R12)
05000354/2005002-03	FIN	Inability to Properly Operate Reactor Recirculation Vibration Monitoring Equipment (Section 1R14)
05000354/2005002-04	FIN	Work Activities During RF12 Exceeded Collective Dose Estimate (Section 2OS2)
05000354/2005002-05	NCV	Hope Creek shipped a radioactive material package with radiation levels in excess of 200 millirem per hour (Section 2PS2)

Closed

05000354/2004009-00	LER	As Found Values for Safety Relief Valve Lift Setpoints Exceed Technical Specification Allowable (Section 4OA3.1)
05000354/2004010-00	LER	Manual Reactor Scram Due to Moisture Separator Dump Line Failure (Section 4OA3.2)

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-108)
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

Acts of Nature (HC.OP-AB.MISC-0001)
Station Preparations for Winter Conditions (HC.OP-GP.ZZ-0003)
Severe Weather Guide (NC.OP-DG.ZZ-0002)
Notification: 20215765

Section 1R02: Evaluation of Changes, Tests, or Experiments

For documents reviewed refer to listing under Permanent Plant Modifications (71111.17)

Section 1R04: Equipment Alignment

Instrument Air System Operation (HC.OP-SO.KB-0001)
Service Air System Operation (HC.OP-SO.KA-0001)
P&ID - Compressed Air Service (M-15-0, sheet 1 of 14)
Standby Liquid Control System Operation (HC.OP-SO.BH-0001)
Emergency Diesel Generators Operations (HC.OP-SO.KJ-0001)
Notifications: 20219084, 20226090, 20179640, 20182458, 20182459, 20189530, 20214112, 20221500

Section 1R05: Fire Protection

Notifications: 20219768, 20225058
Orders: 70044557, 70044175

Section 1R06: Flood Protection Measures

Hope Creek Generating Station Individual Plant Examination (IPE)
Notifications: 20216883, 20224587, 20196105, 20210062, 20221918, 20224132

Section 1R07: Heat Sink Performance

Validating SSWS Flow Through SACS HXs (HC.OP-FT.EA-0001)
P&ID -Service Water (M-10-1)
Notification: 20228650
Orders: 30116091, 30116092, 30112253, 30109357, 30118312, 30107703, 30109722, 70045601

Section 1R11: Licensed Operator Requalification

Simulator Scenario Guide 247, "Loss of Vacuum, HPCI Pressure Control/SRV Cooldown"
HPCI System Operation (HC.OP-SO.BJ-0001)
RCIC System Operation (HC.OP-SO.BD-0001)
Plant Transient Condituons (HC.OP-AB.ZZ-0001)
Main Condenser Vacuum (HC.OP-AB.BOP-0001)
Reactor Pressure Vessel Control (HC.OP-EO.ZZ-0101)

Section 1R12: Maintenance Effectiveness

Reactor Power (HC.OP-AB.RPV-0001)
Reactor Power Oscillations (HC.OP-AB.RPV-0002)
Recirculation System (HC.OP-AB.RPV-0003)
Reactor Level Control (HC.OP-AB.RPV-0004)
Loop Calibration RFPT A Control Valve Indication and Limit Switches (HC.IC-LC.AE-0006)
Engineering Technical Issues Fact Sheets (3/11/2005 and 2/02/2005 RFP Oscillation Incidents)
Feedwater System Health Report
Auxiliary Building Chilled Water - Control Room System Health Report
System Function Level Maintenance Rule (MR) VS Risk Reference (SE.MR.HC.02)
Monitoring the Effectiveness of Maintenance, Rev. 5 (NC.NA-AP.ZZ-0016)
Maintenance Rule (a)(1) Evaluations and Goal Monitoring, Rev. 1 (SH.ER-DG.ZZ-0002)
Operability Assessment and Equipment Control Program (SH.OP-AP.ZZ-0108)
10CFR50.59 Program Guidance (NC.NA-AS.ZZ-0059)
Component Configuration Control (SH.OP-AP.ZZ-0103)
Information Distribution (SH.OP-AP.ZZ-0105)
Control Area Chilled Water System Operation (HC.OP-SO.GJ-0001)
A Control Room Chilled Water System Compensatory Measures (Temporary Standing Order HC-2005-11)
Hope Creek Control Room Logs
Hope Creek UFSAR Sections 9.2.7.2 and 9.4.1
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
Notifications: 20091904, 20091905, 20228007, 20228112, 20222668, 20223033, 20185374, 20220360, 20220362, 20220280, 20220732, 20106150, 20109389, 20140358, 20148886, 20173406, 20199196, 20217920, 20222457, 20225777, 20180255, 20172516
Orders: 70021535, 70025358, 70029034, 70029556, 70030879, 70031040, 70031041, 70031253, 70031468, 70034667, 70034992, 70036826, 70036830, 70037472, 70038081, 70039503, 70041238, 70041163, 70044471, 70045372, 70045541, 70044727, 70038352, 70025902, 70026484, 70036163, 70040686, 70044501, 70045062, 70037501, 70036044

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

System Function Level Maintenance Rule VS Risk Reference (SE.MR.HC.02)
On-Line Risk Assessment (SH.OP-AP.ZZ-0027
Work Management Process (NC.WM-AP.ZZ-001
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

Nuclear Procedure Program (NC.NA-AP.ZZ-0001)
IPTE 04-011, Turbine Digital EHC
Reactor Power (HC.OP-AB.RPV-0001)
Reactor Power Oscillations (HC.OP-AB.RPV-0002)
Recirculation System (HC.OP-AB.RPV-0003)
Reactor Level Control (HC.OP-AB.RPV-0004)
Loop Calibration RFPT A Control Valve Indication and Limit Switches (HC.IC-LC.AE-0006)
Engineering Technical Issues Fact Sheets (3/11/2005 and 2/02/2005 RFP Oscillation Incidents)
Reactor Recirculation System Operation (HC.OP-SO.BB-0002)
Control Room Narrative Log, dated February 8, 2005
Control Room Narrative Log, dated March 11 to March 13, 2005
Temporary Standing Order HC-2005-15, “Reactor Recirculation Vibration Alarm Response”
Notifications: 20228814, 20218444, 20228007, 20228112, 20222668, 20223033, 20185374, 20220360, 20220362, 20220280, 20220732
Work Orders: 60053063, 60051872, 60043614
Orders: 70045541, 70044727, 70038352, 80077511

Section 1R15: Operability Evaluations

Operability Assessment and Equipment Control Program (SH.OP-AP.ZZ-0108)
NRC Generic Letter No. 91-18, Revision 1, Resolution of Degraded and Nonconforming Conditions
NRC Generic Letter No. 88-07, Modified Enforcement Policy Relating to 10 CFR 50.49, “Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants”
Notification Process (NC.WM-AP.ZZ-0000)
Analysis of Allowable Pressurization of RCIC Piping Systems (Calc. BD-001)
Over Pressurization of RCIC Discharge Lines (H-1-BDXX-MSE-0436)
Hope Creek Reactor Recirculation Piping Vibration Monitoring (HC.ER-AP.BB-0002)
CRD Hydraulic System Operation (HC.OP-SO.BF-0001)
CRD Insertion and Withdrawl Speed Test, Adjustment, and Stall Flows (HC.OP-FT.BF-0001)
GE Vendor Document - Control Rod Drive System (VTD: PN0-C11-4010-0055)
Operating Experience (OE) Program (NC.LR-AP.ZZ-0054)

Control Room Narrative Logs, dated February 6 to 10, 2005
Potential non-conformance of diaphragm disc used in certain ASCO scram pilot valves and rebuild kits (ML050560237)
P&ID - Control Rod Drive Hydraulic - Part A (M-46-1)
P&ID - High Pressure Coolant Injection (M-55-1)
Drawings: E-0002-1, E-0046-1, E0106-0, E-0049-1, E-0006-1, E-3020-0
Notifications: 20218297, 20220743, 20222533, 20223463, 20223538, 20224182, 20219926, 20219961, 20218930, 20223870, 20229031, 20227172, 20172074, 20191978, 20208718, 20224092, 20225114, 20173913, 20175866, 20178849, 20196819, 20209023, 20217402 , 20222113, 20218297, 20225836
Orders: 70044027, 70044649, 70044113, 70044669, 70039556, 80078753

Section 1R16: Operator Workarounds

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
Notifications: 20108190, 20195723, 20229056, 20197714, 20197711, 20087812, 20222052, 20219041, 20226255, 20195723, 20225626, 20223109, 20220497, 20220981

Section 1R17: Permanent Plant Modifications

Replace “A” Crankcase Lube Oil HI-LO Alarm Switch H1KJ-1KJLSHL-7558A (DCP 80076323)
HPCI-RCIC Exhaust Drain Pot Line Modification (DCP 80076904)
Removal of Hope Creek RPV Head Spray (DCP 80040594)
Add Time Delay to “A” RHR Min Flow Valve H1BC-BC-HV-F007A Close Logic (DCP 80071938)
Replace APRM and RBM Flow Control Trip Reference Card (DCP 80071246)
Upgrade Primary Fuse For H1BC-HV-F009 Control Circuit in the RSP (DCP 80076765)
Hope Creek Main Transformer Replacement-Phase “B” (DCP 80043091)
Hope Creek Main Transformer Replacement-Phase “C” (DCP 80016883)
Hope Creek 500kV Breaker Addition (DCP 80062464)
Hope Creek Switchyard Breaker BS4-2 Control Room Indication (DCP 80043100)
OPRM Reactor Trip Circuit (DCP 80065875)
HPCI Steam Condensing Line Hydrogen Accumulation Mitigation (DCP 80062840)
Modifications of RHR Valves FO60A/B & FO77 (DCP 80072763)

<u>10 CFR 50.59 Safety Evaluations</u>
Localized High Drywell Temps after RF11 (H2003-002)
Evaluation of Compensatory Actions Required for Fast Control Rod Speeds (H2003-003)
Hope Creek 12 Core Design (H2003-001)
Hope Creek Cycle 13 Core Design (H2004-002)
EHC System Upgrades DCP (H2004-001)

10 CFR 50.59 Safety Evaluation Screens

Screenings for DCPs:

80076904, HPCI-RCIC Exhaust Drain Pot Line
80040594, Remove Hope Creek RPV Head Spray
80071938, Add Time Delay to “A” RHR Min Flow Valve H1BC-BC-HV-F007A Close Logic
80043091, Hope Creek Main Transformer Replacement-Phase “B”
80016883, Hope Creek Main Transformer Replacement-Phase “C”
80062464, Hope Creek 500kV Breaker Addition
80043100, Hope Creek Switchyard Breaker BS4-2 Control Room Indication
80065875, OPRM Reactor Trip Circuit
80065875, HCGS Oscillation Power Range Monitor (OPRM) ASF Trip to RPS Function

Procedure Change Screens:

Procedure Change Screening - HPCI Injection System Operation (HC.OP-SO.BJ-0001)
“C” Service Water Pump - CP502 - In Service Test, (HC.OP-IS.EA-0003)
High Pressure Coolant Injection System Operation (HC.OP-SO.BJ-0001)
Reactor Recirculation System Operation (HC.OP-SO.BB-0002)
Reactor Protection System Operation (HC.OP-SO-0001)
Emergency Diesel Generators Operation (HC.OP-SO.KJ-0001)

FSAR Change Screens:

Cumulative Usage Factor for the RPV Skirt (FSAR Table 3.9-4b) (HCN 04-018)
PCT Tracking (HCN 04-023)

Temporary Modification Screens:

T-Mod 03-055, Install a Temporary Heater for Diesel Battery Room 5541
T-Mod 04-021, Jumpering of Motor Hi Temp Trip on CK-111
T-Mod 03-014, Temporary Installation of Gag on H1BF-1BFPSV-FO25A-C11

Calculation Revision Screen:

Screening of Revision to Calculation EG-0047 to Determine Impact to the Ultimate Heat Sink Temperature Limit (RHR Heat Load Transfer) (EG-0047)

Design References and Calculations

HCGS Ultimate Heat Sink Temperature Limits (EG-0047)
Evaluation of Cumulative Usage Factor for RPV Skirt (H-1-GT-CDC-1566)
CALC. E-1.4 (Q), HC Class 1E 125 & 250 VDC Systems: Short Circuit & Voltage Drop Studies, Rev. 5
CALC. E-17D (Q),125 V DC-Voltage Drop From Distribution Panel to Load, Rev. 4
CALC. E-7.7 (Q), Class 1E 480 Volt System Protective Relaying, Rev. 3

Calc EA-0001, Station Service Water System Hydraulic Model, Rev. 3
LCR H02-002, 1/17/03, Request for Change to Technical Specifications (Deactivate Reactor Vessel Head Spray)
Recirculation Jet Pump Operability-Daily, Rev. 30 (HC.OP-ST.BB-001)
NIS Div. 1-OPRM Channel A1 Oscillation Power Range Monitor, Rev. 4 (HC.IC-CC.SE-0048)
BWROG 96113 dated September 17, 1996, Guidelines for Stability Option III “Enabled Region” (TAC M92882) Letter to NRC
E-9017-1-0, Rev. 0, Neutron Monitoring System OPRM-Interconnection Diagram
Electrical Schematic Diagram-Class IE 4.16 kV Station Power System (E-106-0)
Electrical Schematic Diagram-Diesel Generator Regular and Backup Lockout Relaying (E-107-0)
Elementary Diagram, Power Range Neutron Monitoring System (791E411Ac)
Neutron Monitoring system-OPRM Interconnection Wiring Diagram (E-9017-1)

Administrative Procedures

Engineering Change Process (NC.CC-AP.ZZ-0080)
10CFR50.59 Program Guidance (NC.NA-AS.ZZ-0059)
Modification Walkdown Program for Engineering Changes (NC.CC-AP.ZZ-0003)
Corrective Action Process (NC.WM-AP.ZZ-0002)
Regulatory Change Determination and 10CFR50.59 Review Process (NC.NA-AP.ZZ-0059)
CRs, Notifications and Work Orders: 20097671, 20166751, 20199704, 20128443, 20171387, 20201334, 20129161, 20171769, 20203298, 20130791, 20177118, 20212960, 20137894, 20179360, 20213917, 20141289, 20181352, 20217171, 20149572, 20192409, 20221391, 20153393, 20156362, 20196273, 20199446, 60040709, 70024404, 70029524, 70031262, 70032275, 70032770, 70035009, 70035901, 70036006, 70036936, 70037259, 70037614, 70040812, 70041535, 70043048, 70043188, 70043914

Section 1R19: Post-maintenance Testing

Maintenance Testing Program Matrix (NC.NA-TS.ZZ-0050)
B EDG Monthly Surveillance Test (HC.OP-ST.KJ-0002)
B EDG Operability Test Unloaded (HC.OP-ST.KJ-0019)
B EDG 24 Hour Run and Hot Restart Test (HC.OP-ST.KJ-0015)
EDG Operation (HC.OP-SO.KJ-0001)
Diesel Fuel Oil Transfer Operability 18 Months (HC.OP-ST.KJ-0011)
H Diesel Fuel Oil Transfer Pump - HP401 - Inservice Test (HC.OP-IS.JE-0008)
EDG Vendor Manual (PM18Q-0449)
Notifications: 20224978, 20225032, 20224997, 20224731, 20224537, 20225822, 20225823, 20191353, 20188896, 20187846, 20172516, 20174638, 20177715, 20224978, 20225822, 20225777, 20180255, 20187450, 20187511, 20187550, 20187581, 20198472, 20216947, 20217920,
Orders: 60052325, 70039172, 60045313, 60045886, 30084583, 70045049, 60047041, 60038971, 60032137, 60051036, 70045062

Section 1R20: Refueling and Other Outage Activities

Outage Management Program (NC.NA-AP.ZZ-0055)
Outage Risk Assessment (NC.OM-AP.ZZ-0001)
Preparation for Plant Startup (HC.OP-IO.ZZ-0002)
Startup From Cold Shutdown to Rated Power (HC.OP-IO.ZZ-0003)
Shutdown From Rated Power to Cold Shutdown (HC.OP-IO.ZZ-0004)
Shutdown Cooling (HC.OP-AB.RPV-0009)
Startup Reactivity Plan, dated January 13, 2005 and
P&ID Control Rod Drive Hydraulic - Part A (M-46-1)
Notifications: 20218378, 20219019, 20230268, 20230205, 20229405,20229408, 20230234, 20230256, 20219866, 20219824, 20219825, 20219779, 20219869, 20220743, 20219789
Deviation Memo samples: 60040011, 60043156, 80074246, 60023652, 60041435, 60040465, 60040166, 60009622, 60041524, 60043818, 50065556, 60011928, 60024338, 60036961, 60041862, 60028768, 60032006, 60033245, 60033408, 60038414, 60045317, 60044248, 60047176, 60045948, 60007984, 60027083, 60026989, 60044251, 60012611, 50067095, 50067180, 60041956, 40000976, 60028915, 60033915, 60010922, 60010921, 30107891, 30107800, 30107802, 30107801, 60031130, 60031131, 60031132, 60038205, 30111306, 40001624, 50081428, 60040800, 30086841, 30055781, 30055619, 30055608, 30055742, 60022144, 60044487, 60049795, 60049799, 60049800, 60049801, 30096361, 60048643, 60048668, 60049724, 60049297, 20216812, 60049869, 60049870, 30073740, 60022879, 20078633, 60042444, 20165713, 60007982, 60042251, 30055518, 60042251, 60050220, 30094502, 50021880, 50021647, 50030212, 60010921, 60022879, 60036842, 60044109, 60045813, 60048799, 60049297, 60050118, 60050220, 60050297, 60050308, 60050378, 60050559, 60050694, 30111306, 60043159, 20218884, 60051039, 50081980, 20218926, 60020903, 40021746, 60042512, 60050960, 60050288, 60049473, 20219155, 60048493, 60047584, 30055608, 50021880, 50021600, 30106340, 60037017, 2021885, 60051120

Section 1R22: Surveillance Testing

AP202, A residual Heat Removal Pump In-Service Test (HC.OP-IS.BC-0001)
Reactor Building Integrity Functional Test - 18 Months (HC.OP-ST.GU-0002)
CRD Insertion and Withdraw Speed Test, Adjustment and Stall Flows (HC.OP-FT.BF-0001)
HPCI System Functional Test (Low Pressure) - 18 months and HPCI System Response Time Test (High Pressure) (HC.OP-ST.BJ-0002)
Surveillance Log (HC.OP-DL.ZZ-0026)
Drywell Leakage (HC.OP-AB.CONT-0006)
Drywell Leakage Source Detection (HC.OP-GP.ZZ-0005)
Alternate RCS Leakage Determination (HC.OP-ST.SK-0001)
DLD System Alarm/Trbl (HC.OP-AR.ZZ-0011)
Inservice System Leakage Test of the Reactor Coolant Pressure Boundary (HC.OP-IS.ZZ-0001)
Reactor Coolant System Pressure Isolation Valves Seat Leakage Measurement/Test (HC.RA-IS.ZZ-0017)
VT-2 Visual Examination of Nuclear Class 1, 2 and 3 Systems (SH.RA-IS.ZZ-0005)
Temporary Standing Orders HC-2004-22, 2005-07, 12, 13, and 14 (SH.OP-AP.ZZ-0105)

PI&D -High Pressure Coolant Injection (M-55-1)
Hope Creek Technical Specification 3.4.3.2
NRC Bulletin 88-04, “Potential Safety Related Pump Loss”
Notifications: 20220492, 20192929, 20219086, 20219123, 20223525, 20224211, 20224059, 20223646, 20220470, 20223712, 20224643, 20224676, 20227717, 20227756, 20219866
Orders: 70039871, 70044148, 70044802

Section 1R23: Temporary Plant Modifications

Control of Temporary Modifications (NC.DE-AP.ZZ-0030)
Electrical Print - 480 Volt Motor Control Center 10B573 Tabulation - (E-0022-1 Sh. 2)
Order: 60053045

Section 1EP6: Drill Evaluation

Artificial Island Emergency Plan
Hope Creek Emergency Classification Guide
Hope Creek Event Classification Guide Technical Basis

Section 2OS2: ALARA Planning and Controls

ALARA Review (AR) 2004-124, RFO12-CRDM Maintenance Activities and Related Work
AR 2004-151, DCP 80062840-HPCI Steam Condensing Line Hydrogen Mitigation
AR 2004-162, N2K Recirc Inlet Nozzle Repair and Related Work (1-BB-013-V(D)CA-012)
Radiation Protection Instrumentation (RPI) Laboratory Calibration and Quality Control (NC.RS-TI.ZZ-0592)

Section 2OA2: Identification and Resolution of Problems

Emergency Diesel Generator 1BG400 Operability Test-Monthly (HC.OP-ST-KJ-0002)
Engineering Analysis A-5-5000-EEE-1680, Rev. 8
Grid Disturbances (HC.OP-AB.BOP-004)
Notifications: 60047176, 70032821, 70035290, and 70043848
Orders: 20175209, 20177825, 20177854, 20183896, 20184258, 20196826, 20217153

Section 4OA3: Event Followup

Safety Review For Hope Creek Generating Station Safety/Relief Valve Tolerance Analysis (VTD 322869-01)
BWROG SRV Leakage Committee Guide for Addressing Leaking SRVs in BWRs (VTD 325484)
Safety Relief Valve Model 7567F Technical Manual (PN1-B21-F013-0162)

Section 4OA5: Other Activities

Hope Creek Reactor Recirculation Pumps/Motor Vibration Monitoring, Revision 1 (HC.ER-AP.BB-0001)
Hope Creek Reactor Recirculation Pumps/Motor Vibration Monitoring, Revision 2 (HC.ER-AP.BB-0001)
Hope Creek Reactor Recirculation Pumps/Motor Vibration Monitoring, Revision 3 (HC.ER-AP.BB-0001)
Recirculation System, Revision 9 (HC.OP-AB.RPV-0003)
Overhead Annunciator Window Box C1, Revision 28 (HC.OP-AR.ZZ-0008)

LIST OF ACRONYMS

ALARA	As Low As Is Reasonably Achievable
BWRs	Boiling Water Reactors
CACW	Control Area Chilled Water
CAL	Confirmatory Action Letter
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CREF	Control Room Emergency Filtration
CS	Core Spray
CST	Condensate Storage Tank
DCP	Design Change Package
DEHC	Digital Electro Hydraulic Control
DOT	Department of Transportation
EDG	Emergency Diesel Generator
EHC	Electro-Hydraulic Control
EOF	Emergency Operations Facility
EP	Emergency Preparedness
ERB	Executive Review Board
GE	General Electric
HCGS	Hope Creek Generating Station
HPCI	High Pressure Coolant Injection
IPEEE	Individual Plant Examination For External Events
IPTE	Infrequently Performed Test or Evolution
IST	In-service Test
LER	Licensee Event Report
MG	Motor Generator
MR	Maintenance Rule
NCV	Non Cited Violation
NOSC	Nuclear Operations Service Contract
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OCR	Operational Challenge Response

OWA	Operator Work-around
PARS	Publicly Available Records
PI	Performance Indicators
PJM	Pennsylvania, New Jersey and Maryland Interconnection
PMT	Post-maintenance Testing
PSEG	Public Service Enterprise Group
RACS	Reactor Auxiliary Cooling System
RCIC	Reactor Core Isolation Cooling
RF12	Refueling Outage 12
RFP	Reactor Feedwater Pump
RHR	Residual Heat Removal
ROP	Reactor Oversight Process
RWP	Radiation Work Permit
SACS	Safety Auxiliaries Cooling System
SCWE	Safety conscious work environment
SDP	Significance Determination Process
SLC	Standby Liquid Control
SRA	Senior Risk Analyst
SRV	Safety Relief Valves
SSC	Structures, Systems and Components
SSWS	Station Service Water System
T-Mod	Temporary Modification
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report