December 19, 2001

Mr. Harold W. Keiser
Chief Nuclear Officer and President
PSEG Nuclear LLC - X04
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - NRC INSPECTION REPORT 50-354/01-10

Dear Mr. Keiser:

On November 11, 2001, the NRC completed an inspection of your Hope Creek facility. The enclosed report documents the inspection findings which were discussed on November 19, 2001, with Mr. Dave Garchow 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. Specifically, this inspection involved six weeks of resident inspection and two region-based inspections of occupational radiation safety and inservice inspection activities.

No findings of significance were identified.

Since September 11, 2001, Hope Creek Nuclear Generating Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to PSEG Nuclear LLC. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA/

William A. Cook, Chief
Projects Branch 3
Division of Reactor Projects

Enclosure: Inspection Report 50-354/2001-10
Attachment: Supplemental Information

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

cc w/encl: E. Simpson, Senior Vice President and Chief Administrative Officer
M. Bezilla, Vice President - Technical support
D. Garchow, Vice President - Operations
G. Salamon, Manager - Licensing
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M. Wetterhahn, Esquire
N. Cohen, Coordinator - Unplug Salem Campaign
E. Gbur, Coordinator - Jersey Shore Nuclear Watch
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-354
License No:	NPF-57
Report No:	50-354/01-10
Licensee:	PSEG Nuclear LLC
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	October 1 - November 11, 2001
Inspectors:	J. G. Schoppy, Jr., Senior Resident Inspector C. G. Cahill, PE, Resident Inspector A. Lohmeier, Reactor Inspector J. T. Furia, Senior Health Physicist
Approved By:	William A. Cook, Chief, Projects Branch 3 Division of Reactor Projects

Summary of Findings

IR 05000354-01-10, on 10/1 - 11/11/01, Public Service Electric Gas Nuclear LLC, Hope Creek Generating Station. Resident inspector report.

The inspection was conducted by resident inspectors, a regional radiation specialist, and a regional reactor inspector. No findings of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, Significance Determination Process (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Violations

The inspectors reviewed four violations of very low significance which were identified by PSEG Nuclear. Corrective actions taken or planned by PSEG Nuclear appeared reasonable. These violations were dispositioned as Non-Cited Violations and are listed in Section 40A7 of this report.

If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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; and the NRC Resident Inspector at the Hope Creek facility.

Report Details

SUMMARY OF PLANT STATUS

At the beginning of the period, operators maintained the unit at 95 percent power as the A feedwater heater string was out of service for planned maintenance. On October 1, operators performed an emergent power reduction to 80 percent in response to high solar magnetic disturbance (SMD). At 2:14 p.m. on October 2, operators increased reactor to 95 percent. At 8:36 p.m. on October 2, operators performed an emergent power reduction to 80 percent in response to high SMD. On October 4, operators increased reactor to 95 percent. The unit remained at 95 percent power until operators commenced a planned shutdown for refueling outage No. 10 (RF10) on October 9. At 1:05 a.m. on October 10 operators performed a planned manual scram from 18 percent power to place the unit in Hot Shutdown.

At 3:34 p.m. on November 1, operators took the mode switch to Startup and commenced a reactor startup. At 5:25 p.m. on November 1, operators declared the reactor critical and at 8:13 a.m. on November 3, entered Mode 1 (Power Operation). At 3:28 p.m. on November 3, operators synchronized the main generator to the grid and on November 10, increased power to 100 percent. The Hope Creek plant operated continuously at or near full power for the duration of the inspection period.

1. REACTOR SAFETY
Initiating Events, Mitigating Systems, and Barrier Integrity [REACTOR - R]

R04 Equipment Alignment

.1 A Emergency Diesel Generator Outage

a. Inspection Scope

The inspectors performed equipment alignment verifications on redundant equipment during an A emergency diesel generator (EDG) outage. The inspectors verified by reviewing the technical specifications (TS), plant walkdowns, and main control room tours that the planned equipment outage on the A EDG did not adversely affect the redundant AC electrical sources during plant operations in operational condition 4. In particular, the inspectors performed walkdowns of the following equipment and areas:

- B, C, and D EDGs.
- Contol room instrumentation and control panels.
- 4160 V vital switchgear rooms and 480V vital motor control centers.
- Safety-related 125Vdc battery rooms.

Additionally, the inspectors reviewed various corrective action Notifications associated with equipment alignment deficiencies (Nos. 20078774, 20078788, 20079990, 20080366, 20080494, 20080501, 20082644, 20082672, 20082703, and 20082725).

b. Findings

No findings of significance were identified.

.2 Post-Outage System Walkdowns

a. Inspection Scope

The inspectors performed equipment alignment walkdowns of the service water and high pressure coolant injection systems prior to plant restart following RF10. The inspectors verified by reviewing the technical specifications and corrective action notifications, plant walkdowns, and main control room tours that these systems were fully operable. The inspectors verified proper system alignment using the following documents:

- Independent Verification Service Water System Operation (HC.OP-SO.EA-0001, Attachment 1).
- Independent Verification High Pressure Coolant Injection System Operation (HC.OP-SO.BJ-0001, Attachment 1).
- Independent Verification 250VDC Electrical Distribution (HC.OP-SO.PJ-0001, Attachment 1).
- Service Water Flow Path Verification - Monthly (HC.OP-ST.EA-0001, Attachments 2 and 3).
- HPCI System Piping and Flow Path Verification - Monthly (HC.OP-ST.BJ-0001, Attachment 2).
- Service Water System TRIS Valve Lineup.
- High Pressure Coolant Injection System TRIS Valve Lineup.

b. Findings

No findings of significance were identified.

R05 Fire Protection

a. Inspection Scope

The inspectors performed walkdowns of the drywell, torus room, diesel/control building, and the refueling floor during RF10. Plant walkdowns included observations of combustible material control, fire detection and suppression equipment availability, and compensatory measures. The inspectors performed fire protection inspections in these areas due to the potential for outage work to impact mitigating systems or initiate a fire. The inspectors reviewed Hope Creek’s Individual Plant Examination for External Events for risk insights concerning these areas. Additionally, the inspectors reviewed Quality Assurance Assessment Report 2001-0358 and several Notifications associated with fire protection deficiencies (Nos. 20078669, 20078697, 20078946, 20080210, 20081120, 20081389, and 20081460).

b. Findings

No findings of significance were identified.

R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed corrective action Notification No. 20080083 involving flood protection issues.

b. Findings

No findings of significance were identified.

R08 Inservice Inspection Activities

a. Inspection Scope

The inspector reviewed the PSEG Nuclear inservice inspection (ISI) Refueling Outage (RFO) Examination Plan for the Second Interval, Second Period, First Outage ISI in satisfaction of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC) Section XI, Nuclear Components. The inspector compared the examination plan schedules with the required ASME Section XI inspection program to determine whether the plan appropriately followed the scheduler requirements of IWB-2400 of the ASME Code.

The inspector observed manual ultrasonic (UT) and magnetic particle testing (MT) activities to verify the qualifications and effectiveness of the examiner in identifying degradation of safety significant systems, structures, and components (SSCs) and to evaluate the activities for compliance with the requirements of ASME Section XI. The inspector observed the performance of inspection personnel in MT and UT of valve to pipe, pipe to elbow, and elbow to pipe full penetration circumferential welds 1, 2, and 3 (summary numbers 213000, 212005, and 213010) of the high pressure coolant injection (HPCI) pipe chase line No. 1-FD-10DBB-002B.

The inspector reviewed and evaluated results of examination summary records of 23 selected system welds, including MT examination data, liquid penetrant test (PT) results, UT calibration data sheets, UT indications recorded, and UT determination of profile/thickness measurements to determine whether the test implementation procedures were in accordance with ASME Section XI requirements. Included in these observations were residual heat removal pump welds, reactor coolant system pump welds, HPCI line welds, and core spray system welds.

The inspector observed samples of video recordings of the remote in-vessel visual inspection (IVVI) of jet pump, core spray piping, sparger, and core shroud support welds. The inspector also reviewed video recordings of the condition of reactor internal surfaces for degraded conditions for the purpose of determining whether the residue on the surfaces could potentially cover defect indications. The inspector reviewed PSEG

Nuclear's disposition of observed findings and visual acceptance criteria in satisfaction of ASME requirements.

The inspector reviewed PSEG Nuclear corrective actions in response to a recent finding of a leaking crack in the socket weld connection at the reactor recirculation pressure tap line BB321. The inspector reviewed the historical antecedents to the socket weld connection failure of this line to the recirculation pipe elbow at this plant and the extent of engineering investigation of the experience with failed socket welds throughout the industry included in reports by the Electric Power Research Institute (EPRI), MPR Associates, Structural Integrity Associates (SIA), and Westinghouse Electric Corporation (WEC). In interviews with ISI personnel, the inspector reviewed the bases for belief that the root cause of cracking was fatigue of the weld material due to extended vibratory stress. The inspector reviewed the possibilities of cracking due to other causes and the historical preponderance of socket weld failures due to vibratory fatigue.

The inspector reviewed radiographs of the line BB321 cracked weld in comparison with radiographs of similar socket weld connections that had not cracked. The inspector reviewed the cracked weld configuration to determine the location of the weld crack with respect to any geometric anomalies that would lead to crack formation. Also reviewed was PSEG Nuclear's assessment of the difficulties in obtaining a metallurgical (boat) sample that could be used for characterization of the crack.

The inspector reviewed the planning for the weld repair on line BB321 with respect to the RF10 restart schedule and the need to provide a repair consistent with ASME design requirements. The inspector attended meetings of the root cause determination group, the repair/replacement installation group, and the Station Operations Review Committee (SORC) that provided screening for the 10CFR50.59 safety evaluation. The inspector reviewed the root cause committee's determination of the root cause of the weld crack, the significance of any of the minor changes in design details of the socket weld connection, the effectiveness of the repair procedure planning, and the structural adequacy of the socket weld repair.

b. Findings

No findings of significance were identified.

R11 Licensed Operator Requalification

a. The inspectors observed one simulator training scenario to assess operator performance and training effectiveness. The scenario involved the plant startup operations specifically focusing on reactor vessel level control during operations on the condensate pumps and transfer to the feed pumps. The inspectors assessed simulator fidelity and observed the simulator instructor's critique of operator performance. The inspector also observed just-in-time training dealing with level control issues during the plant shutdown for RF10. The inspectors also observed control room activities with emphasis on simulator identified areas for improvement.

b. Findings

No findings of significance were identified.

R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed all corrective action notifications initiated between July 1, 2001, and August 15, 2001, for maintenance rule screening. The inspectors further reviewed six Notifications that included system engineer functional failure determinations (Nos. 2007331, 20073321, 20073543, 20073771, 20073965, and 20074656). The inspectors also reviewed Hope Creek Expert Panel Meeting Minutes (HCEP 01-008).

To assess PSEG Nuclear's implementation of 10CFR 50.65 Maintenance Rule requirements, the inspectors reviewed the following documents:

- System Function Level Maintenance Rule VS Risk Reference (SE.MR.HC.02).
- 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.

b. Findings

No findings of significance were identified.

R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated risk management for the following configurations: (1) a leak on the reactor recirculation pressure tap line BB321 to the main recirculation piping; (2) the emergent on-line corrective maintenance on the D EDG due to an engine-driven jacketwater pump seal failure; (3) shutdown cooling suction header penetration planned work during RF10; and, (4) the concurrent planned outage of the A EDG and the AX501 transformer (offiste power supply to the A and C vital buses) during RF10. The inspectors reviewed maintenance risk evaluations, Outage Risk Assessment and Management (ORAM) status, work schedules, recent corrective action notifications, and control room logs to verify that other concurrent planned and emergent maintenance or surveillance activities did not adversely affect the plant risk already incurred with the out of service components. The inspectors also used PSEG Nuclear’s on-line risk monitor (Equipment Out Of Service workstation), the ORAM Sentinel Logic Database Report, and outage risk assessment procedure (NC.OM-AP.ZZ-0001) to evaluate the risk associated with the plant configurations and to assess PSEG Nuclear’s risk management. In addition, the inspectors reviewed other Notifications involving risk assessment and emergent work (Nos. 20078602, 20078739, 20079762, 20078898, 20079336, 20079500, 20079713, 20082704, and 20082775).

To assess PSEG Nuclear's risk management, the inspectors reviewed the following documents:

- System Function Level Maintenance Rule VS Risk Reference (SE.MR.HC.02).
- On-Line Risk Assessment (SH.OP-AP.ZZ-108).
- NRC Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.
- Section 11, Assessment of Risk Resulting from Performance of Maintenance Activities, dated February 11, 2000, of NUMARC 93-01, Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.
- October 12, 2001, SORC presentation "A" Reactor Recirc. Loop Leakage Outage Impact.
- Outage Management Program (NC.NA-AP.ZZ-0055).
- Outage Risk Assessment (NC.OM-AP.ZZ-0001)

b. Findings

No findings of significance were identified.

R14 Personnel Performance During Nonroutine Plant Evolutions

.1 Emergent Power Reductions

a. Inspection Scope

The inspectors reviewed control room operator actions in response to an emergent power reduction on October 1, due to the potential adverse impact of SMD on the C main power transformer. On October 2, the inspectors observed control room operator actions in response to another SMD induced power reduction. In both circumstances, operators entered HC.OP-AB.ZZ-0152, Loss of Feedwater Heating, and reduced power from 95 percent to 80 percent. The inspectors reviewed the operations logs, abnormal procedure HC.OP-AB.ZZ-0152, the Transient Assessment Response Plan (TARP) report, dated 10/01/01, and the associated Notifications (Nos. 20078762, 20078739, 20078713, 20078714, and 20078898).

b. Findings

No findings of significance were identified.

.2 Hazardous Material Control

a. Inspection Scope

The inspectors observed PSEG Nuclear's response to an unknown substance discovered inside the protected area on October 16, 2001. The inspectors observed the hazardous material brief, recovery of the unknown substance, and the decontamination of the affected area in accordance with NC.FP-ED.ZZ-0002, Fire Department Hazardous Material Response. Upon further investigation and chemical analysis, the unknown substance was determined to be non-hazardous. The inspectors also reviewed PSEG Nuclear's formal notification to the NRC (Event Number 38392.)

b. Findings

No findings of significance were identified.

.3 Post-Outage Hydrostatic Testing

On October 27, operators performed a plant hydrostatic test. The inspectors observed operators' preparations for the plant hydrostatic test, the pre-evolution control room briefing, and control room operations associated with the hydrostatic test. The inspectors performed walkdowns of the drywell, torus room, HPCI pipe chase, and the steam tunnel while the plant was at test pressure. Additionally, the inspectors reviewed HC.OP-IS.ZZ-0001, Inservice System Leakage Test of the Reactor Coolant Pressure Boundary, and HC.OP-DL.ZZ-0026, Attachment 3, Minimum Reactor Pressure Vessel Metal Temperature vs. Reactor Pressure Logs, to ascertain operator compliance and conformance to these procedures.

b. Findings

No findings of significance were identified.

R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the operability determination for a safety auxiliaries cooling heat exchanger bypass valve failure (Order No. 70020041) and the source range monitor detector configuration non-conformance (Order No. 70020375). The inspectors also reviewed all other PSEG Nuclear identified safety-related equipment deficiencies during this report period and assessed the adequacy of the operability screening.

b. Findings

No findings of significance were identified.

R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed corrective action notifications, operator logs, and instrument panel status to evaluate potential impacts on the operators' ability to implement abnormal or emergency operating procedures. The inspectors also reviewed the following documents:

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

b. Findings

No findings of significance were identified.

R19 Post Maintenance Testing

a. Inspection Scope

The inspectors witnessed post maintenance testing (PMT) on the B EDG and reviewed the test data on D EDG engine driven jacket water cooling pump. The inspectors observed portions of the A recirculation loop instrument line tap repair, reviewed the leak repair paperwork, and independently visually examined the repair during the post-outage plant hydrostatic test. The inspectors also reviewed the HPCI 250V battery replacement PMT paperwork and performed a post-installation HPCI battery walkdown. The inspectors reviewed NC.NA-TS.ZZ-0050, Maintenance Testing Program Matrix, and verified that the PMTs were adequate for the scope of maintenance performed. The inspectors also reviewed Notifications concerning problems associated with PMTs (Nos. 20080437, 20080460, 20080553, 20080601, 20081100, 20081578, and 20082798).

The inspectors reviewed the following documents:

- Emergency Diesel Generator BG 400 Operability Test - Monthly (HC.OP-ST.KJ-0002).
- IPTE 01-020, Pre-Evolution Briefing for Reactor Recirculation Pressure Tap BB321 Pipe Repair.
- Framatome Traveler No. 50-5015123-00, Repair of the Hope creek Recirc Pipe Instrument Line.
- Drawing 1-P-BB-321, Revision 5, Engineered Small Piping/Drywell Bldg. Instrument Line from Recirc Loop A Suction to Drywell Penetration.
- BCT-2000 Load Test Report, dated 10/25/01.
- Full Battery Bank Replacement (VHC.MD-GP.ZZ-0100).
- 250 Volt Quarterly Battery Surveillance (HC.MD-ST.PJ-0002), dated 10/18/01.
- 250 Volt Quarterly Battery Surveillance (HC.MD-ST.PJ-0002), dated 10/29/01.
- Battery Equalizing Charge (HC.MD-GP.ZZ-0015).
- 18-Month Surveillance and Service Test of 250 Volt Batteries using BCT-2000.

b. Findings

No findings of significance were identified.

R20 Refueling and Outage Activities

a. <u>Inspection Scope</u>

On October 9, operators commenced a planned shutdown for RF10. At 1:05 a.m. on October 10, operators performed a planned manual scram from 18 percent power to place the unit in Hot Shutdown. The inspectors observed operators' preparations for the plant shutdown, portions of the power reduction, the pre-evolution scram briefing, control room operations associated with the manual scram initiated to place the plant in Hot Shutdown, and portions of the plant cool-down.

During the outage, the inspectors performed verifications of the cooldown rate, shutdown cooling flow paths, inventory control, offsite power availability, reactivity control, containment integrity, and equipment tagging. The inspectors evaluated PSEG Nuclear's shutdown risk management and configuration control. The inspectors observed fuel handling activities from the refueling bridge and the control room. The inspectors reviewed a risk-informed sample of Outage Scope Deferral Requests and Outage Scope Addition Requests. The inspectors also reviewed Notifications concerning problems related to the refueling outage (Nos. 20079415, 20079560, 20079469, 20079480, 20079539, 20079706, 20079792, 20080112, 20080346, 20080350, 20080567, 20080598, 20080860, 20080915, and 20080939).

In preparation for plant restart, the inspectors reviewed the control room deficiency logs and the TS Action Statement Log, and performed plant equipment walkdowns. The inspectors observed the reactor startup and criticality from the control room and portions of the power ascension activities. The inspectors reviewed the following documents:

- Shutdown From Rated Power To Cold Shutdown (HC.OP-IO.ZZ-0004).
- Reactor Scram (HC.OP-AB.ZZ-0000).
- Transient Plant Conditions (HC.OP-AB.ZZ-0001).
- Post-Transient Response Requirements (SH.OP-AP.ZZ-0101).
- Reactor Mode Switch Functional Test - 18 Months (HC.OP-ST.SF-0001).
- Refuel Interlock Operability Functional Test (HC.OP-ST.KE-0001).
- Refuel Platform and Fuel Grapple Operation (HC.OP-SO.KE-0001).
- Refueling Operations (HC.OP-IO.ZZ-0009).
- Power Distribution Lineup - Weekly (HC.OP-ST.ZZ-0001).
- Primary Containment Airlock Operability Test (HC.OP-ST.ZZ-0004).
- Main Turbine Operation (HC.OP-SO.AC-0001).
- Decay Heat Removal Operation (HC.OP-SO.BC-0002).
- Loss of Shutdown Cooling (HC.OP-AB.ZZ-0142).
- Irradiated Fuel Damage (HC.OP-AB.ZZ-0101).
- SRM/IRM Malfunction (HC.OP-AB.ZZ-0107).
- Outage Management Program (NC.NA-AP.ZZ-0055).
- Outage Risk Assessment (NC.OM-AP.ZZ-0001).
- Conduct of Fuel Handling (NC.NA-AP.ZZ-0049).
- Fuel Handling Controls (HC.RE-FR.ZZ-001).
- Loss of Fuel Pool Inventory Cooling (HC.OP-AB.ZZ-0144).
- Preparation For Plant Startup (HC.OP-IO.ZZ-0002).
- Startup From Cold Shutdown to Rated Power (HC.OP-IO.ZZ-0003).

b. Findings

No findings of significance were identified.

R22 Surveillance Testing

a. Inspection Scope

The inspectors observed portions of and reviewed the results of the following tests: (1) secondary containment pressure; (2) as-found local leak rate testing for the inboard and outboard main steam isolation valves; (3) reactor building/secondary containment integrity verification; (4) stroke of the hard torus vent valve (HV-11541); and, (5) end of cycle recirculation pump trip breaker testing. The inspectors reviewed the test procedures to verify that applicable system requirements for operability were incorporated correctly into the test procedures, test acceptance criteria were consistent with the TS and UFSAR requirements, and the systems were capable of performing their intended safety functions. The inspectors also reviewed Notifications concerning problems encountered during surveillance testing (Nos. 20078549, 20078633, 20078646, 20078772, 20079711, 20079769, 20080028, 20080058, 20080449, 20082778, 20082844, and 20082874).

The inspectors reviewed the following documents:

- Reactor Building Integrity Functional Test - 18 Months (HC.OP-ST.GU-0002).
- Containment Isolation Valve Type C Leak Rate Test (HC.RA-IS.ZZ-0010).
- Reactor Building / Secondary Containment Verification (HC.OP-ST.ZZ-0003).
- MSIV Leakage Rate Conversion for Testing (CALC No. AB-0067).
- Containment Atmosphere Control System Valves - 18 Months (HC.OP-IS.GS-0102).
- Time Response Test Reactor Protection System Division 2 End of Cycle Recirculation Pump Trip Breaker Arc Suppression (HC.IC-TR.SB-0010).

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupation Radiation Safety [OS]

OS1 Access Control

a. Inspection Scope

The inspector identified exposure significant work areas, high radiation areas, and airborne radioactivity areas in the plant and reviewed associated controls and surveys of these areas to determine if the controls (i.e., surveys, postings, barricades) were acceptable. For these areas, the inspector: reviewed radiological job requirements and attended job briefings; determined if radiological conditions in the work area were adequately communicated to workers through briefings and postings; verified radiological controls, radiological job coverage and contamination control; and verified the accuracy of surveys and applicable posting and barricade requirements. The inspector determined if prescribed radiation work permits (RWPs), procedure and engineering controls were in place, whether surveys and postings were complete and accurate, and if air samplers were properly located. Observation of work activities inside the radiologically controlled area (RCA) occurred in the reactor, turbine, service, and radwaste buildings. Plant technical specification 6.12 and 10 CFR 20, Subpart G were utilized as the standard for necessary barriers. The inspector reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy.

Direct observation of significant radiological activities were focused on the work being performed in support of the refueling outage (RF10). Jobs included: in-service inspection of reactor vessel nozzles; safety relief valve change-out; reactor coolant pump seal replacement; in-vessel visual inspections; repairs to the drywell equipment hatch; local power range monitor removal; control rod drive change-out; and fuel movement. Work activities being performed in high and locked high radiation areas included the drywell and on the refueling floor. Other significant work activities were observed in the steam tunnel, condenser bay, and turbine deck.

The inspector also reviewed ten problem reports, nine of which were generated during the current refueling outage (Nos. 20079706, 20079389, 20079641, 20079787, 20079875, 20079989, 20080113, 20080115, 20080347, and 20079058).

b. Findings

No findings of significance were identified.

OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed work to be performed during the refueling outage (RF1O). Areas reviewed included: use of low dose waiting areas; on-job supervision provided to workers; and, individual exposures from selected work groups. An evaluation of engineering controls utilized to achieve dose reductions and an analysis of source term reduction plans were also conducted. PSEG Nuclear's outage goals were 19 days duration and less than 140 person-rem (stretch goal of 125 person-rem).

The inspector observed radiation worker and radiation protection technician performance during high dose rate and/or high exposure jobs (i.e., work in the drywell and on the refueling floor) to determine the extent of workers implementing the as low as is reasonably achievable (ALARA) practices. The inspector also observed radiation worker performance to determine whether the training/skill level was sufficient with respect to the radiological hazards and the work involved.

The inspector reviewed ALARA job evaluations, exposure estimates and exposure mitigation requirements, and the results achieved for the work listed above. The inspector also reviewed: the integration of ALARA requirements into work procedures and RWP documents; the accuracy of person-hour estimates and person-hour tracking; and, the generated shielding requests and their effectiveness to dose rate reduction.

A review of actual exposure results versus initial exposure estimates for outage work was conducted, including: comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, (exposure report timeliness, and exposure report distribution to support control of collective exposures), to determine compliance with the requirements contained in 10 CFR 20.1101(b). The inspector reviewed the planning of five high exposure jobs performed during RF1O, and their associated ALARA packages, including: In-Service Inspection (33 rem); radiation protection (18.99 rem); reactor reassembly (17.5 rem); drywell support (10.106 rem); and cavity decon (6.735 rem).

b. Findings

No findings of significance were identified.

OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed instrumentation utilized by health physics technicians and plant workers to measure radioactivity, including portable field survey instruments, friskers, portal monitors, and small article monitors. Specifically, the inspector conducted a verification of proper function and certification of appropriate source checks for these instruments, which are utilized to ensure that occupational exposures are maintained in accordance with 10 CFR 20.1201.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors verified the methods used to calculate the performance indicator (PI) on Reactor Coolant System Leakage. The inspectors verified the accuracy of PI data submitted through review of the applicable pages in the daily TS surveillance data sheet (HC.OP-DL.ZZ-0026, Surveillance Log - Control Room) for the period October 2000 through September 2001.

b. Findings

No findings of significance were identified.

OA2 Identification and Resolution of Problems

The inspectors reviewed numerous Notifications associated with PSEG Nuclear's identification, evaluation, and resolution of problems. No significant findings were identified. The Notifications are listed in Sections 1R04, 1RO5, 1RO6, 1R12, 1R13, 1R14, 1R15, 1R16, 1R019, 1R20, 1R22, and 2OS1of this report.

OA4 Cross-cutting Issues

The licensee identified findings in Sections 4OA7.1 and 4OA7.2 of this report had implications regarding the cross-cutting area of human performance. On October 25, 2001, PSEG Nuclear management initiated a Tagging Assessment Plan to evaluate safety tagging process barriers and work management interface issues that have resulted in numerous tagging errors at the Salem and Hope Creek stations.

OA6 Management Meetings

a. Exit Meeting Summary

On November 19, the inspectors presented their overall findings to members of PSEG Nuclear management, led by Mr. Dave Garchow. PSEG Nuclear management stated that none of the information reviewed by the inspectors was considered proprietary.

b. PSEG Nuclear/NRC Management Meeting

On October 3 and 4, 2001, Hubert J. Miller, Regional Administrator, Region I, and Mr. A. Randolph Blough, Director, Division of Reactor Projects, Region I, met with PSEG Nuclear management, discussed regulatory issues, and toured the Salem and Hope Creek units.

OA7 Licensee Identified Violations. The following findings of very low significance were identified by PSEG Nuclear and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations (NCV).

Cornerstone: Initiating Events

.1 **NCV 05000354/2001-010-01**: Technical Specification 6.8.1 requires, in part, that written procedures shall be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33. Regulatory Guide 1.33 requires, in part, that procedures be developed for equipment control (e.g., locking and tagging). PSEG Nuclear procedure SH.OP-AP.ZZ-0015, SAP/WCM Tagging Operations, requires, in part, that personnel ensure the adequacy of the blocking points for the work being performed. On September 30, 2001, plant operators established blocking points on the A feedwater heater string that were not adequate for the scheduled feedwater work resulting in a challenge to the offgas system. PSEG Nuclear entered this issue into their problem identification and corrective action system as Notification No. 20078977. This issue is being treated as a Non-Cited Violation.

.2 **NCV 05000354/2001-010-02**: Technical Specification 6.8.1 requires, in part, that written procedures shall be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33. Regulatory Guide 1.33 requires, in part, that procedures be developed for equipment control (e.g., locking and tagging). PSEG Nuclear procedure SH.OP-AP.ZZ-0015, SAP/WCM Tagging Operations, requires, in part, that operators check the component to be tagged by comparing installed field labels with the tag and the Tagging Working List. On November 4, 2001, plant operators failed to adequately check a feedwater (FW) heater drain valve tag resulting in the tagging of the wrong drain valve (operators tagged open the drain valve to the in-service 5C FW heater, vice the out-of-service 5B FW heater) and potentially impacted FW flow to the reactor vessel. PSEG Nuclear entered this issue into their problem identification and corrective action system as Notification No. 20082644. This issue is being treated as a Non-Cited Violation.

Cornerstone: Mitigating Systems

.3 **<u>NCV 05000354/2001-010-03</u>**: 10CFR50, Appendix B, Criterion VIII, Identification and Control of Materials, Parts, and Components, requires that measures be established for the identification and control of parts and components. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components. Contrary to the above, PSEG Nuclear did not establish adequate measures to preclude installation of a safety relief valve (SRV) with an incorrect relief setpoint during refueling outage No. 9. Although the SRV was installed with the incorrect setpoint (1108 psig vice 1130 psig), the as-found lift pressure (1115 psig) was within the TS 3.4.2.1 acceptance criteria for the 1130 psig setpoint pressure. PSEG Nuclear entered this issue into their problem identification and corrective action system as Notification No. 20081563. This issue is being treated as a Non-Cited Violation.

Cornerstone: Barrier Integrity

.4 **<u>NCV 05000354/2001-010-04</u>**: 10CFR50, Appendix B, Criterion III, Design Control, requires that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, PSEG Nuclear did not establish adequate measures to assure that the reactor building filtration, recirculation, and ventilation system (FRVS) vent fan differential pressure controllers were set properly to allow the FRVS vent fans to adequately establish secondary containment integrity. PSEG Nuclear entered this issue into their problem identification and corrective action system as Notification No. 20079341. This issue is being treated as a Non-Cited Violation.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Howard Berrick, Licensing
John Carlin, Vice President - Nuclear Reliability
Terry Cellmer, Radiation Protection Manager
Matt Conroy, Maintenance Rule Supervisor
Mike Dammann, Maintenance Manager - Controls & Power Distribution
Wayne Denlinger, In-service Inspection
Ali Fakhar, Reliability Programs Manager
Robert Gary, Radiation Protection Operations Superintendent
David Kelly, Support Supervisor - Calibration
Robert Keyes, Radiation Protection Supervisor
Kurt Krueger, Operations Manager
Theodore Neufang, ALARA Supervisor
Kevin O'Hare, ALARA Superintendent
Michael Oliveri, In-service Inspection
Michael Petrowski, Radiation Protection Supervisor
Devon Price, Assistant Operations Manager
Allen Roberts, Nuclear Reliability
Elwood Robinson, Quality Assessment
Gabor Salamon, Nuclear Safety & Licensing Manager
W. Treston, In-service Inspection/Testing
Jay Trombley, Radiation Protection Supervisor
Larry Wagner, Director - Site Work Integration & Management
Tommy Wallender, Radiation Protection Supervisor

b. List of Items Opened, Closed, and Discussed

Opened/Closed

50-354/2001-010-001	NCV	Failure to establish adequate blocking points for the work being performed. (Section 4OA7.1)
50-354/2001-010-002	NCV	Operators failed to adequately check a feedwater heater drain valve tag resulting in the tagging of the wrong drain valve. (Section 4OA7.2)
50-354/2001-010-003	NCV	Failure to establish adequate measures to preclude installation of a safety relief valve with an incorrect relief setpoint during refueling outage No. 9 (Section 4OA7.3)

50-354/2001-010-004	NCV	Failure to establish adequate measures to assure that the reactor building filtration, recirculation, and ventilation system (FRVS) vent fan d/p controllers were set properly to allow the FRVS vent fans to adequately establish secondary containment integrity. (Section 4OA7.4)

c. 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
HCGS Plant Status Report
Weekly Reactor Engineering Guidance to Hope Creek Operations
B & D Core Spray Pumps - BP206 and DP206 - In-service Test (HC.OP-IS.BE-0002)

ASME Section XI 2nd Interval 2nd Period 1st Outage In-Service Inspection

2nd Interval 2nd Period 1st Outage - Examination Plan - RFO10
Status of: Manual Ultrasonic Examinations (10/17/01)
Visual Examinations 1 (VT-1) (9/24/01)
Visual Examinations 2 (VT-2) (9/24/01)
Reactor Vessel (RPV) Class 1 Interior Component Examinations (IVVI)

NDE Examination Summary Records

250500	Residual Heat Removal System Pump RHP-W3 Pump Casing Weld 10/12/01 Examination Summary, PT, Surface Examinations, 10/03/01
209645	Reactor Core Spray System Piping Line 1-BE-14GBB-009-16 Pipe to Elbow Weld Examination Summary, MT, UT Calibration, UT Indication Data, Indication Plot Sheet, Wall Thickness Profile, 10/12/01
211605	High Pressure Coolant Injection Line 1-BJ-14DBB-003A -30 Pipe to Elbow Weld Examination Summary, MT, UT Calibration, Indication Plot Sheet, Wall Thickness Profile, 10/12/01
250157	Reactor Core Spray Pump CSP 76 Pump Casing Weld Examination Summary, PT Examinations, 10/05/01

A Reactor Recirculation Pump Suction Pipe 1 Instrument Leaking Pipe Weld Repair

Root Cause Determination

R.N. Coward (MPR)	Hope Creek Reactor Recirculation Pressure Tap BB321 Cracked Connection Evaluation,
R. Shindel	A Reactor Recirculation Loop Leakage Outage Impact
R. Labott	Metallurgical Engineering - Cause of Failure Evaluation - Field Weld No. 46 (Drawing No. 1-P-BB-321)
EPRI TR-107455	Vibration Fatigue of Small Bore Socket-Welded Pipe Joints
SIA Sketches	Two Major Causes for Vibrational Fatigue Failures at Small Bore Piping
H. Malikowski	Vibration Fatigue of Small-Bore Socket Welds
O&MR 424	Small Bore Piping Connection Failures
SIA Report	Industry Fatigue Experience
W R&D Center	Fracture Analysis of a Drain Valve Nipple
PO P1-341912	Hope Creek Weld Fracture Analysis

Corrective Action

Recirc Failure	Long Term Potential Areas Under Review Short Term Corrective Action Matrix Short Term Corrective Action Matrix (Cont'd)

10CFR 50.59 Safety Evaluation

NC.NA-AS.ZZ-0059(Q)	Regulatory Change Determination and 10CFR50.59 Review Process
NC.NA-AS.ZZ-0059(Q)	10CFR50.59 Screening - Repair of A Reactor Recirculation Pump Suction Pipe 1" Instrument Pipe Weld Repair

Drawings

M-43-1 (Q)-26	Hope Creek Generating Station - Reactor Recirculation System P&ID Sheet 1 of 2 (9/29/87) (Formerly Bechtel Power Corporation M-43-1 Sheet 1 Rev 18)
1-P-BB-321 Rev 5	Engineered Small Piping/Drywell Bld3 Instrument Line From Recirculation Loop A Suction To Drywell Penetration (Bechtel - San Francisco)
6011248A Rev 0	Hope Creek Plug Assembly (10/13/01) (Framatome ANP)
P&ID M-55-1 Sheet 1	Hope Creek ISI Weld Identification Figure B-59 - HPCI Turbine Steam Line 1-FD-12DBB-002
NC.DE-WB-ZZ-001-20	Pictograph Plan View of Socket Weld Repair configuration

d. List of Acronyms

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
B&PVC	Boiler and Pressure Vessel Code
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
FRVS	Filtration, Recirculation, and Ventilation System
FW	Feedwater
HCGS	Hope Creek Generating Station
HCEP	Hope Creek Expert Panel
HPCI	High Pressure Coolant Injection
ISI	Inservice Inspection
IVVI	Internal Vessel Visual Inspection
MT	Magnetic Test
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
ORAM	Outage Risk Assessment and Management
PARS	Publicly Available Records
PI	Performance Indicator
PMT	Post Maintenance Testing
PSEG	Public Service Electric Gas
PT	Penetrant Test
RCA	Radiologically Controlled Area
RF10	Refueling Outage No. 10
RWP	Radiation Work Permit
SDP	Significance Determination Process
SIA	Structural Integrity Associates
SMD	Solar Magnetic Disturbance
SORC	Station Operations Review Committee
SSC	Systems, Structures, and Component
SRV	Safety Relief Valve
TS	Technical Specification
UT	Ultrasonic Test
VT	Visual Test
WEC	Westinghouse Electric Corporation