July 27, 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 50-354/01-07

Dear Mr. Keiser:

On June 30, 2001, the NRC completed an inspection of your Hope Creek facility. The enclosed report presents the results of that inspection. The preliminary findings were presented to PSEG Nuclear management led by Mr. Tim O'Connor in an exit meeting on July 5, 2001.

NRC inspectors examined numerous activities as they related to reactor safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspection consisted of selective review of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection involved seven weeks of resident inspection and three region-based inspections of occupational radiation safety, physical security, and plant modifications and evaluation of changes.

Based on the results of this inspection no findings of significance were identified.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA R. Barkley for/

Glenn W. Meyer, Chief
Projects Branch 3
Division of Reactor Projects

Enclosure: Inspection Report 50-354/01-07
Attachment: Supplemental Information

Docket No. 50-354
License No. NPF-57
cc w/encl:

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-354
License No:	NPF-57
Report No:	50-354/2001-007
Licensee:	PSEG Nuclear LLC
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	May 13 - June 30, 2001
Inspectors:	J. G. Schoppy, Jr., Senior Resident Inspector C. G. Cahill, PE, Resident Inspector R. S. Barkley, Senior Project Engineer J. T. Furia, Senior Health Physicist G. C. Smith, Sr. Physical Security Inspector P. R. Frechette, Physical Security Inspector G. W. Morris, Reactor Engineer A. Lohmeier, Reactor Inspector B. S. Norris, Reactor Inspector
Approved By:	Glenn W. Meyer, Chief, Projects Branch 3 Division of Reactor Projects

Summary of Findings

IR 05000354-01-07, on 05/13 - 06/30/01, Public Service Electric Gas Nuclear LLC, Hope Creek Generating Station. Resident inspector report.

The inspection was conducted by resident inspectors, a regional radiation specialist, two security specialists, a regional projects inspector, and three region-based inspectors. This inspection identified no significant findings. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Findings

The inspectors reviewed a violation of very low significance which was identified by PSEG Nuclear. Corrective actions, taken or planned by PSEG Nuclear, appeared reasonable. This violation is described in Section 4OA3.1 of this report.

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Report Details

SUMMARY OF PLANT STATUS

At the beginning of the inspection period, operators maintained the unit in Cold Shutdown following a Technical Specification (TS) required shutdown on May 8 due to two inoperable main steam isolation valve sealing systems. At 9:22 p.m. on May 14, operators took the mode switch to Startup and commenced a reactor startup. At 3:12 a.m. on May 15, operators declared the reactor critical and at 00:56 a.m. on May 16 entered Mode 1 (Power Operation). At 1:13 p.m. on May 16, operators synchronized the main generator to the grid and on May 21 increased power to 100 percent. The Hope Creek plant operated continuously at or near full power for the duration of the inspection period except for planned maintenance power reductions on May 30 for a Salem 500KV line (5021) outage, on June 24 for a rod pattern adjustment, and on June 30 for another Salem 500KV line (5021) outage.

1. REACTOR SAFETY

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

R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed procedure NC.NA-AS.ZZ-0059(Q), Revision 4, *10CFR50.59 Program Guidance*, and procedure NC.NA-AP.ZZ-0059(Q), Revisions 8 and 9, *10CFR Applicability Reviews and Safety Evaluations*.

The inspectors reviewed 24 selected 10CFR50.59 safety evaluations (SE) representing the three cornerstones: initiating events, mitigating systems and barrier integrity. The objectives of this review was to verify that (1) changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), and tests or experiments, not described in the UFSAR, were reviewed and documented in accordance with 10 CFR 50.59 and (2) that approval had been obtained from the NRC prior to implementing those changes that required such approval.

The inspectors interviewed engineering personnel engaged in the preparation and the review of the selected 10CFR50.59 SEs. Throughout the reviews of the selected SEs, the inspectors conducted meetings with PSEG Nuclear to resolve questions and observations made during the course of the review. The 10CFR50.59 safety evaluations that were reviewed are listed in the supplemental information.

The inspectors also reviewed 11 applicability reviews of change items (e.g., procedure, calculation, and UFSAR changes), that were screened out of the 10 CFR 50.59 evaluation process, to verify that such screenings were appropriate.

b. Findings

No findings of significance were identified.

R04 Equipment Alignment

.1 Diesel Fuel Oil Storage and Transfer System Walkdown

a. Inspection Scope

The inspectors performed a complete equipment alignment check on the diesel fuel oil storage and transfer (DFOST) system to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system. The alignment check included a review of documents to determine the correct system lineup and performance of a field walkdown to identify any discrepancies between the existing lineup and the prescribed lineup. The inspectors also monitored A emergency diesel generator (EDG) fuel oil pump pressure and filter differential pressure during a 24-hour run on the A EDG. Specifically the following documents and procedures were reviewed:

- *Acts of Nature* (HC.OP-AB.ZZ-0139)
- *Diesel Fuel Oil Storage and Transfer System Operation* (HC.OP-SO.JE-001)
- *System Health Report Diesel Fuel Oil Storage and Transfer System - JE*, Period 10/1/00 to 12/31/00
- *EDG 1AG400 - 24 Hour Operability Run and Hot Restart Test* (HC.OP-ST.KJ-0014)
- *Updated Final Safety Analysis Report*, Section 9.5.4
- A DFOST IST (HC.OP-IS.JE-0001), dated 4/04/01
- B DFOST IST (HC.OP-IS.JE-0002), dated 4/04/01
- C DFOST IST (HC.OP-IS.JE-0003), dated 5/17/01
- D DFOST IST (HC.OP-IS.JE-0004), dated 5/17/01
- E DFOST IST (HC.OP-IS.JE-0005), dated 6/07/01
- F DFOST IST (HC.OP-IS.JE-0006), dated 6/08/01
- G DFOST IST (HC.OP-IS.JE-0007), dated 4/25/01
- H DFOST IST (HC.OP-IS.JE-0008), dated 4/25/01

The inspectors also reviewed various corrective action notifications associated with DFOST system (20045585, 20059532, 20061335, and 20069400).

b. Findings

No findings of significance were identified.

.2 Partial System Walkdowns

a. Inspection Scope

The inspectors performed equipment alignment verifications on redundant equipment during system outages on the B service water (SW) pump and the reactor core isolation cooling (RCIC) system. The inspectors verified by plant walkdowns and main control room tours that planned equipment outages on the B SW pump and RCIC did not adversely affect the redundant SW subsystems and high pressure coolant injection (HPCI) system, respectively. The inspectors also verified that the B SW pump and RCIC system were restored to an operable condition after the planned maintenance was complete. Additionally, the inspectors reviewed various corrective action notifications associated with equipment alignment deficiencies (20065676, 20068101, 20068151, 20068162, 20068169, 20069043, 20069082, and 20070127).

b. Findings

No findings of significance were identified.

R05 Fire Protection

a. Inspection Scope

The inspectors reviewed Hope Creek's Individual Plant Examination for External Events for risk insights concerning fire areas. The inspectors performed walkdowns of the following risk significant fire areas: the upper control equipment room, the cable spreading room, auxiliary building 124' elevation electrical access area, and auxiliary building 137' elevation electrical access area. The inspectors also reviewed NRC NUREG 1742, *Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program*, for risk insights relative to the Hope Creek Generating Station (HCGS). Additionally, the inspectors reviewed several notifications associated with fire protection deficiencies (20065766, 20069568, 20027561, 20069658, and 20070537).

b. Findings

No findings of significance were identified.

R06 Flood Protection Measures

The inspectors reviewed several notifications involving flood protection (20060621, 20060623, 20067621, and 20069577).

b. Findings

No findings of significance were identified.

R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed all corrective action notifications initiated from February 16 to March 31, 2001, for Maintenance Rule screening. The inspectors further reviewed five notifications that included system engineer functional failure determinations (20059524, 20059534, 20059842, 20060084, and 20060711); one maintenance preventable functional failure evaluation (70015338); and three notifications involving PSEG Nuclear's implementation of their Maintenance Rule program (20057806, 20059348, and 20070476). The inspectors reviewed an (a)(2) and an (a)(1) system health report (EDG and SW systems, respectively) and discussed system reliability and availability monitoring with the respective performance engineers. The inspectors also reviewed Hope Creek Expert Panel Meeting Minutes (HCEP 01-007).

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

- SE.MR.HC.02, *System Function Level Maintenance Rule VS Risk Reference*
- 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 on-line risk management for the following configurations: (1) the concurrent planned outage of the B SW pump, the B residual heat removal (RHR) pump, and the D RHR pump; (2) emergent corrective maintenance issues associated with the C RHR pump and B EDG concurrent with an extended outage on C EDG; and (3) the concurrent planned outage of RCIC and the emergent corrective maintenance issues associated with the C SW spray wash booster pump. The inspectors reviewed maintenance risk evaluations, 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 or inoperable components. The inspectors also used PSEG Nuclear's on-line risk monitor (Equipment Out Of Service workstation) to evaluate the risk associated with the plant configuration and to assess PSEG Nuclear's risk management. In addition, the inspectors reviewed other notifications involving risk assessment and emergent work (20066512, 20066565, 20066757, 20067256, 20069706, and 20070471).

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

- SE.MR.HC.02, *System Function Level Maintenance Rule VS Risk Reference*
- HCGS PSA Risk Evaluation Forms for Work Week Nos. 19-24
- SH.OP-AP.ZZ-108, *On-Line Risk Assessment*
- 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*

b. Findings

No findings of significance were identified.

R14 Personnel Performance During Nonroutine Plant Evolutions

.1 Safety Auxiliaries Cooling System Pressure Relief Valve Lifting

a. Inspection Scope

On June 22, 2001, operators identified that a safety auxiliaries cooling system (SACS) relief valve (1EGPSV-2409C) had unexpectedly lifted during operation with two SACS pumps running in the A SACS loop (see also Sections 1R15.1 and 4OA3.1 of this report). Operators removed the C SACS pump from service to reseat the relief valve and entered TS limiting condition for operation (LCO 3.7.1.1.a.1.a) which provided an allowed outage time (AOT) of 30 days in this condition. The single failure criterion did not apply to the redundant A SACS loop pump as technical specifications allow an exception to the General Design Criteria for the brief period of the TS AOT. Operators noted that the installed relief valve was set at and lifted at 120 psig while the design lift setpoint for the valve was 150 psig. Operators implemented compensatory measures for the degraded relief valve in accordance with engineering evaluation 70018021(see Section 1R15.1). Operators restored the A SACS loop to its normal configuration following the relief valve replacement on June 24.

Inspectors evaluated operator response to this condition and the initiating causes regarding personnel error contribution. Specifically, the inspectors reviewed the reactor operator narrative log, the TS Action Statement Log, alarm response procedures, SACS system operating procedure, SACS relief valve 1EGPSV-2409C maintenance work order No. 60014470 dated 6/5/01, post maintenance testing (PMT) for work order No. 60014470, and engineering design change package (DCP) 4HZ-04225 dated 3/31/93. In addition, the inspectors discussed the event with operators, maintenance technicians, quality assessment (QA) personnel, valve engineers, and PSEG Nuclear management.

b. Findings

No findings of significance were identified in this area. An associated finding concerning the identification and control of parts is described and assessed for significance in Section 4OA3.1 of this report.

.2 Control Rod Pattern Adjustment

a. Inspection Scope

On June 24, operators performed a deep/shallow control rod exchange to evenly distribute fuel exposure and control blade history throughout the core. The rod exchange required operators to reduce reactor power to 60 percent using core flow reductions and control rod insertions. The inspectors reviewed reactor engineering's Maneuver Sequence guidance, attended the pre-job brief, and observed portions of the control rod exchange. Additionally, the inspectors reviewed several notifications associated with the power reduction (20070324, 20070344, 20070345, and 20070366).

b. Findings

No findings of significance were identified.

R15 Operability Evaluations

.1 Safety Auxiliaries Cooling System Pressure Relief Valve Degraded Condition

a. Inspection Scope

As stated in Section 1R14.1 above, on June 22, 2001, operators identified that SACS relief valve (1EGPSV-2409C) had unexpectedly lifted during operation with two SACS pumps running in the A SACS loop (see Section 4OA3.1 of this report). Operators removed the C SACS pump from service to reseat the relief valve and entered TS LCO 3.7.1.1.a.1.a which provided a 30 day AOT for this condition. Operators noted that the installed relief valve was set at and lifted at 120 psig while the design lift setpoint for the valve was 150 psig. Engineering performed an operability determination to evaluate continued operability of the A SACS loop with the degraded relief valve installed. Engineering determined that the A SACS loop was operable but degraded.

The inspectors reviewed the operability determination for the SACS relief valve degraded condition (evaluation 70018021). The operability evaluation included compensatory measures to maintain one of the two A SACS loop pumps out of service and to administratively control the A SACS loop flowrate (above 10,000 gpm) to ensure that the SACS relief valve did not re-lift. The inspectors performed control room panel and vital switchgear area walkdowns to independently verify that operators adhered to specified compensatory measures. The inspectors noted that the administratively controlled SACS loop flowpath was the normal plant configuration and controlled by procedure HC.OP-SO.EG-0001, *Safety and Turbine Auxiliaries Cooling Water System Operation.* During plant operation, the flowrate via this path is normally maintained above 10,000 gpm.

b. Findings

No findings of significance were identified.

.2 Operability Determination Reviews

a. Inspection Scope

The inspectors reviewed the operability determination for a RCIC high suction pressure alarm (notification 20066604) and the potential impact of loose parts (C RHR pump minimum flow discharge valve 1BCV-130 disc nut, washer, and nut pin) on the RHR system (evaluation 70017602). The inspectors also reviewed all other PSEG Nuclear identified safety-related equipment deficiencies during this report period and assessed the adequacy of the operability screenings.

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 evaluated the cumulative effects of operator workarounds as related to (1) the reliability, availability, and potential for mis-operation of plant systems; (2) the potential to increase an initiating event frequency or to affect multiple mitigating systems; and (3) operator ability to respond in a correct and timely manner to plant transients and accidents. In addition, the inspectors reviewed four notifications involving PSEG Nuclear’s Operator Burden Program (20030720, 20053385, 20065709, and 20067491).

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*
- *Operator Burden Program* (SH.OP-AP.ZZ-0030)

b. Findings

No findings of significance were identified.

R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed procedure NC.NA-AP.ZZ-0008(Q), Revision 15, *Control of Design and Configuration Change, Tests and Experiments*, procedure NC.NA-AP.ZZ-0008(Q), Revision 16, *Configuration Control Program*, procedure NC.NA-AP.ZZ-0035(Q), Revision 12, *Nuclear and Environmental Licensing*, and procedure NC.LR-AP.ZZ-0035(Q)-Rev.1, *Licensing Implementation*.

The inspectors reviewed selected permanent plant changes, design changes, set point changes, procedure changes, equivalency evaluations, suitability analyses, and calculations representing the three cornerstones: initiating events, mitigating systems and barrier integrity. The objectives of this review were to verify that (1) the design bases, licensing bases, and performance capability of risk significant structures systems or components (SSCs) had not been degraded through modifications, and (2) that modifications performed during risk-significant configurations did not place the plant in an unsafe condition.

b. Findings

No findings of significance were identified.

R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the PMT results for the retest of the remote shutdown panel reactor vessel wide range level instrumentation, RHR minimum flow line check valve 1BCV-130, and preventive maintenance on the B SW pump. 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 (20065684, 20066303, 20066840, 20067609, 20070016, 20070481, and 20070699).

The inspectors reviewed the following documents:

- *B Service Water Pump -BP502 In-service Test* (HC.OP-IS.EA-0002)
- *Service Water Subsystem B Valves In-service Test* (HC.OP-IS.EA-0102)
- *B Spray Water Pump -BP507 In-service Test* (HC.OP-IS.EP-0002)
- *Remote Shutdown Monitoring Instrumentation Channel Check* (HC.OP-ST.SV-001)
- *Residual Heat Removal Subsystem C Valves In-service Test* (HC.OP-IS.BC-0103)

b. Findings

No findings of significance were identified.

R20 Refueling and Outage Activities

a. Inspection Scope

During the forced outage the inspectors performed verifications of shutdown cooling flow paths, inventory control, offsite power availability, reactivity control, and containment integrity. The inspectors evaluated PSEG Nuclear's shutdown risk management and configuration control. In preparation for plant restart, the inspectors performed plant equipment walkdowns; observed the startup shift briefing; and reviewed control room deficiency logs, the TS Action Statement Log, and reactor engineering's estimated critical positions for various temperatures. The inspectors observed the reactor startup and criticality from the control room and portions of the power ascension activities. The inspectors also reviewed notifications concerning problems related to the forced outage (20065747, 20065759, 20065839, 20065866, 20065921, 20065964, and 20066416).

The inspectors reviewed the following documents:

- *Decay Heat Removal Operation* (HC.OP-SO.BC-0002)
- *Outage Management Program* (NC.NA-AP.ZZ-0055)
- *Outage Risk Assessment* (NC.OM-AP.ZZ-0001)
- *Post-Trip Data Collection Guidelines* (HC.OP-DG.ZZ-0101)
- *Preparation For Plant Startup* (HC.OP-IO.ZZ-0002)

- *Core Operations Guidelines* (HC.RE-IO.ZZ-0001)
- *Startup From Cold Shutdown to Rated Power* (HC.OP-IO.ZZ-0003)
- *Hope Creek Generating Station Core Operating Limits Report* (NFS-0181)
- *Drywell and Suppression Chamber Oxygen Concentration Verification - Weekly* (HC.OP-ST.GS-0001)

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 A and C core spray pump inservice test (IST). Problems encountered with installed plant instrumentation initially caused unsatisfactory IST results for the D DFOST pump and the D SW pump (notifications 20066364 and 20067412). The inspectors reviewed PSEG Nuclear's troubleshooting and the subsequent IST results for the D DFOST pump and the D SW pump. The inspectors also reviewed notifications concerning problems encountered during surveillance testing (20065684, 20066364, 20067031, 20067412, 20067903, 20067941, 20068247, 20068492, and 20069749).

The inspectors reviewed the following documents:

- *D Service Water Pump -DP502 In-service Test* (HC.OP-IS.EA-0004) dated 1/2/01, 3/26/01,4/27/01, 5/26/01, and 5/27/01
- *Service Water Pump Troubleshooting at Different Flow Rates* (SE H99-053)
- *Operations Troubleshooting and Evolutions Plan Development - D SW Pump Troubleshooter* (SH.OP-ZP.ZZ-0008)
- *C Diesel Fuel Oil Transfer Pump-CP401 - In-service Test* (HC.OP-IS.JE-0003)
- *D Diesel Fuel Oil Transfer Pump-DP401 - In-service Test* (HC.OP-IS.JE-0004)
- *A & C Core Spray Pumps - AP206 and CP206 - In-service Test* (HC.OP-IS.BE-0001)

b. Findings

No findings of significance were identified.

Emergency Preparedness [EP]

EP6 Drill Evaluation

a. Inspection Scope

The inspector observed an emergency preparedness drill from the control room simulator and the emergency operations facility on May 23, 2001. The control room portions of the drill were credited toward the Drill and Exercise Performance performance indicator. The inspector evaluated the conduct of the drill and adequacy of PSEG Nuclear's critique of performance to identify weaknesses and deficiencies. The inspectors also reviewed notifications concerning problems related to the emergency preparedness drill (20066573, 20066574, 20066575, 20066576, and 20066702).

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupation Radiation Safety [OS]

OS1 Access Control (7112101)

a. Inspection Scope

The inspector evaluated 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 controls (i.e., surveys, postings, barricades) were acceptable. For these areas, the inspector reviewed all 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 controls; 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; surveys and postings were complete and accurate; and air samplers were properly located. Reviews of RWPs used to access these and other high radiation areas and to identify what work control instructions or control barriers had been specified were conducted. Areas examined were determined by the work being performed. Observation of work activities occurred in the reactor, turbine, radwaste, and support buildings. Plant TS 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. The inspector also examined PSEG Nuclear's programmatic controls for highly activated/contaminated materials (non-fuel) stored within the spent fuel pool.

The inspector reviewed notifications concerning problems related to accessing radiologically significant areas (20062388 and 20062545). The inspector also reviewed quality assurance assessment reports (QAAR) and quality assurance monitoring feedback (QAMF) related to the access control program (QAAR 2000-0238, QAAR 2000-0351, QAAR 2001-0003, QAAR 2001-0013, QAMF 2000-0172, QAMF 2000-0178, QAMF 2001-0083, and QAMF 2001-0102).

b. Findings

No findings of significance were identified.

OS2 ALARA Planning and Controls (7112102)

a. Inspection Scope

The inspector reviewed work performance during the current operating cycle. Areas reviewed included a review of the use of low dose waiting areas; review of on-the-job supervision provided to workers; and a review of individual exposures from selected work groups. An evaluation of engineering controls utilized to achieve dose reductions, and analysis of PSEG Nuclear source term reduction plans was also conducted.

The inspector observed radiation worker and radiation protection technician performance during high dose rate or high exposure jobs and determined if workers demonstrated the ALARA philosophy in practice. The inspector 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 ALARA plans, which were compared with the results achieved. A review of the integration of ALARA requirements into work procedures and RWP documents; the accuracy of person-hour estimates and person-hour tracking; and generated shielding requests and their effectiveness to dose rate reduction was also conducted. The inspector also reviewed PSEG Nuclear planning for RF10, scheduled to commence in the Fall of 2001.

A review of actual exposure results versus initial exposure estimates 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 notifications, related to maintaining occupational exposures as low as is reasonably achievable (20060684 and 20067342). The inspector also reviewed quality assurance assessment report, QAAR 2001-0081. Additionally, the inspector reviewed focused self-assessment reports (FSRs) performed by the radiation protection staff. These FSRs included review of work week management dose estimates, and performance indicators.

b. Findings

No findings of significance were identified.

OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity, including portable field survey instruments, friskers, portal monitors and small article monitors. The inspector identified types of portable radiation detection instrumentation used for job coverage of high radiation area work, other temporary area radiation monitors currently used in the plant, and continuous air monitors associated with jobs with the potential for workers to receive 100 mrem committed effective dose equivalent (CEDE). The inspector conducted a review of instruments observed, specifically 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.

The inspector reviewed the following notifications, related to radiation monitoring instrumentation, to ensure that problems were being identified, characterized, prioritized, entered into a corrective action system, and resolved: 20061451 and 20061409. Additionally, the inspector reviewed FSRs performed by the radiation protection staff, including external exposure control and passive internal monitoring.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Physical Protection [PP]

PP1 Access Authorization

a. Inspection Scope

The following activities were conducted to determine the effectiveness of PSEG Nuclear's behavior observation portion of the personnel screening and fitness-for-duty programs as measured against the requirements of 10CFR26.22 and PSEG Nuclear's Fitness-for-Duty Program documents.

Five supervisors representing the maintenance, radiation protection, chemistry and security organizations were interviewed, on June 12, 2001, regarding their understanding of behavior observation responsibilities and the ability to recognize aberrant behavior traits. Two Access Authorization/ Fitness-for-Duty self-assessments, an audit, and event reports and loggable events for the four previous quarters were reviewed during June 11-13, 2001. On June 12, 2001, five individuals who perform escort duties were interviewed to establish their knowledge level of those duties. Behavior observation training procedures and records were reviewed on June 11, 2001.

b. Findings

No findings of significance were identified.

PP2 Access Control

a. Inspection Scope

The following activities were conducted during the period June 11-13, 2001, to verify that PSEG Nuclear has effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area as measured against 10CFR73.55(d) and the Physical Security Plan and Procedures.

Site access control activities were observed, including personnel and package processing through the search equipment during peak ingress periods on June 11, 12, and 13, 2001, and vehicle searches on June 11, 2001. On June 12, 2001, testing of all access control equipment; including metal detectors, explosive material detectors, and X-ray examination equipment, was observed. The Access Control Event Log, an audit, and three maintenance work requests were also reviewed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

OA1 Performance Indicator Verification

.1 Heat Removal System Unavailability

a. Inspection Scope

The inspectors verified the methods used to calculate the *Heat Removal System Unavailability* (RCIC) performance indicator and reviewed the data for the period April 1, 2000, through March 31, 2001. The inspectors reviewed LCO logs, control room operating logs, corrective action program notifications, and Maintenance Rule electronic data bases.

b. Findings

No findings of significance were identified.

.2 Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment

a. Inspection Scope

The inspector reviewed PSEG Nuclear's programs for gathering and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment performance indicators. The review included PSEG Nuclear's tracking and trending reports, personnel interviews and security event reports for the Performance Indicator data collected from the 1st quarter of 2000 through the 1st quarter of 2001.

b. Findings

No findings of significance were identified.

OA2 Identification and Resolution of Problems

a. Inspection Scope

The finding in Section 4OA3.1 of this report also had implications regarding PSEG Nuclear's identification, evaluation, and resolution of problems, as follows:

- The work order planner, the maintenance technicians, maintenance supervisors, the equipment operator who performed the PMT, and engineers involved in the 1993 DCP all missed potential opportunities to identify a degraded SACS relief valve prior to it impacting the plant. This demonstrated weak identification of a configuration control deficiency.

Additional items associated with PSEG Nuclear's corrective action program were reviewed without findings and are listed in Sections 1R02, 1R04, 1RO5, 1R06, 1R12, 1R13, 1R14.21R15, 1R16, 1R19, 1R20, 1R22, 1EP6, 2OS1, 2OS2, and 2OS3 of this report.

b. Findings

No findings of significance were identified.

OA3 Event Follow-up

.1 Safety Auxiliaries Cooling System Pressure Relief Valve Lifting

a. Inspection Scope

As stated in Sections 1R14.1and 1R15.1above, on June 22, 2001, operators identified that SACS relief valve (1EGPSV-2409C) had unexpectedly lifted during operation with two SACS pumps running in the A SACS loop. Operators removed the C SACS pump from service to reseat the relief valve and entered TS LCO 3.7.1.1.a.1.a which provided a 30 day AOT for this condition. Operators noted that the installed relief valve was set at and lifted at 120 psig while the design lift setpoint for the valve was 150 psig. Operators initiated corrective action notification 20070275 and made a non-emergency eight-hour event report (EN 38087) in accordance with 10CFR50.72(b)(3)(v)(D). Engineering performed an operability determination to evaluate continued operability of the A SACS loop with the degraded relief valve installed.

The inspectors reviewed operator actions in response to the lifted SACS relief in the C EDG room. The inspectors reviewed the reactor operator narrative log, the TS Action Statement Log, alarm response procedures, SACS system operating procedure, SACS relief valve 1EGPSV-2409C maintenance work order No. 60014470 dated 6/5/01, post maintenance testing (PMT) for work order No. 60014470, engineering design change package (DCP) 4HZ-04225 dated 3/31/93, and engineering evaluation 70018021 (see Section 1R15.1 of this report). The inspectors independently inspected the C EDG to verify that the SACS relief valve leakage (approximately 400 gallons) did not impact EDG operability. Additionally, the inspectors performed independent walkdowns of the normal SACS expansion tank makeup valve and the SACS emergency SW makeup valves, and verified the electrical power sources for these valves.

The inspectors reviewed the following documents:

- *Safety Auxiliaries Cooling System Malfunction* (HC.OP-AB.ZZ-0124)
- *Safety and Turbine Auxiliary Cooling Water System Operation* (HC.OP-SO.EG-0001)
- *Overhead Annunciator Window A1-E4, SACS LOOP A TROUBLE* (HC.OP-AR.ZZ-0011)
- *Safety Auxiliaries Cooling System - Subsystem A Valves - InService Test* (HC.OP-IS.EG-0101), dated 4/12/01 and 7/4/01
- *Service Water Subsystem A Valves - InService Test* (HC.OP-IS.Ea-0101), dated 4/15/01
- *Process Setpoints for the SACS Expansion Tanks* (Calculation EG-0009)
- *Demineralized Water makeup Storage & Transfer* (P&ID M-18-0)

Work order No. 50002421, *1EGPSV-2409C CAT C Relief Valve 10YR PM*

- Transient Assessment Response Plan (TARP) investigation report dated June 22, 2001

b. Findings

PSEG Nuclear identified that they had failed to adequately identify and control a SACS relief valve replacement activity resulting in the use of an incorrect component that adversely impacted SACS system operability. This finding was determined to be of very low safety significance (Green).

A PSEG Nuclear review identified that during initial plant startup, 120 psig relief valves were replaced with 150 psig relief valves in the SACS loops (FCR J-50067) but the bill of materials (BOM) material master, vendor drawings, and instrument calibration data (ICD) cards were not updated. In 1993 engineering implemented DCP 4HZ-04225 and updated the vendor drawings and ICD cards to reflect the updated relief valve setpoint (150 psig). On June 5, 2001, maintenance technicians replaced the installed 150 psig relief valve with a 120 psig relief valve under work order No. 60014470. The work order planner had reviewed the functional location for relief valve 1EGPSV-2409C in the work planning database (SAP - Systems, Applications, & Processes) and found that SAP listed relief valve 1EGPSV-2409C as 150 psig with an associated folio number of PM141Q-0048. However, spare relief valves in folio PM141Q-0048 all had 120 psig springs installed as the BOM was never updated to create distinct BOMs and folio numbers for each particular valve setpoint.

The PMT for work order No. 60014470, completed and documented on June 10, 2001, specified that there was no visible signs of leakage from the relief valve. There was no requirement to lift test the relief valve prior to installation or to configure the system to test above normal operating pressures after installation. There was no QA hold points or QA signatures in the work order. The inspector determined that there was no requirement for QA approval of the work.

On June 24 maintenance technicians replaced the degraded relief valve under work order 60020367. Prior to installation, technicians lift tested the upgraded relief valve (an old 120 psig relief valve modified to lift at 150 psig) to validate the actual lift pressure.

The inspector determined that maintenance activities in June 2001 and operators' response to the degraded condition did not involve personnel error. However, the work order planner, the maintenance technicians, maintenance supervisors, the equipment operator who performed the PMT, and engineers involved in the 1993 DCP all missed potential opportunities to identify the condition prior to it impacting the plant.

The finding had a credible impact on safety as the initial PSEG Nuclear engineering analysis determined that following a postulated design basis accident scenario with two SACS pumps operating in the A SACS loop, the A SACS loop could have reached a pressure causing the relief valve to open resulting in draining the A SACS loop head tank and loop inoperability. SACS is a risk significant mitigating system designed to provide a heat sink for engineered safety feature (ESF) equipment (RHR heat exchangers, RHR pump motor bearing coolers, EDG heat exchangers, RHR pump room coolers, HPCI pump room coolers, RCIC pump room coolers, and core spray pump room coolers). The inspectors assessed the finding for significance, with assistance from the Region I Senior Reactor Analyst (SRA), using a Phase 2 SDP. The risk screening was a conservative estimate based on the following assumptions:

- The A SACS loop would function for all initiating events other than loss of offsite power because the automatic demineralized water makeup valve opens to maintain the SACS head tank. Based on historical data plots (as recent as March 2001), the demineralized water makeup capacity of 50 gpm exceeded the relief valve leakage rate of 13 gpm.

- Loss of the A SACS loop would result in the loss of the A & C EDGs, the A& C RHR pumps, the A & C core spray pumps, and the HPCI pump room coolers.

- Due to minor steam leakage coming from HPCI steam trap drain lines, the HPCI pump was assumed to fail due to a lack of room cooling.

- The resultant estimated likelihood rating (C) was based on a loss of offsite power given the condition existed for 17 days (June 5 through June 22, 2001).

- The demineralized water makeup source function was monitored in the Maintenance Rule under SACS and had no documented functional failures.

- Credit was given for operators ability to recover the A SACS loop (nine out of ten times). Operators would have approximately 4.5 hours (3500 gallons in tank when operators would receive control room alarm/ 13 gpm leakrate) to provide alternate makeup to the tank using the SW emergency makeup valves.

- No credit was given for operator identification of the leakage prior to receiving the control room alarm for low SACS tank level. Equipment operators routinely tour the EDG rooms (where the relief valve was located) and would most likely tour the room following a loss of offsite power with the EDGs running. On June 22 control room operators identified the loss of SACS inventory, equipment operators identified the lifting relief valve, and operators took action to reseat the relief valve prior to receiving the low level alarm.

- Operators demonstrated a good knowledge level concerning use of the SW emergency makeup valves. Makeup via this source is covered by plant procedures. Operators need only operate three valves and check a third valve closed, all from the control room. The makeup valves were safety-related, vital powered, and had been adequately maintained and tested.

- There was no adjustments made for other defense-in-depth design features such as the ability to cross-tie the B SACS loop to supply the ESF equipment noted above or the ability to provide SACS loop makeup from the fire main.

Based on the remaining mitigation capability and the estimated likelihood rating, all sequences screened to Green. The inspectors and the SRA also reviewed the SDP for all other initiating events assuming that 10 percent of the time the demineralized water makeup source would fail to function. All of these scenarios also screened to Green. In addition, the SRA used the Hope Creek, Revision 3, GEM/SPAR model to confirm the Phase 2 results. This analysis supported the Green finding conclusion. Based on the above risk assessment, the finding is characterized as Green by the SDP.

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 measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or traceable to the item, as required throughout fabrication, erection, installation, and use of the item. 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 identify and trace the part number to the proper relief valve (150 psig setpoint) to prevent use of an incorrect relief valve (120 psig setpoint). However, because the violation is of very low significance and PSEG Nuclear entered the deficiency into their corrective action system (notification 20070275), this finding is being treated as a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000354/2001-007-01)**

If you deny this non-cited violation, 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.

.2 (Closed) Event No. 38042 (60-Day Optional Phone Call): Inadvertent loss of A RPS bus. NRC inspection report 05000354/2001-001-06 section 1R14.2 describes the circumstances and PSEG Nuclear actions regarding this event. Inspectors reviewed this event report and did not identify any findings of significance.

.3 (Closed) LER 354/1997-23-01: Core spray nozzle weld through-wall leak. NRC Inspection Reports 354/1997-07 Section M2.4 and 354/1997-09 Section M8.1 describe the circumstances and Public Service Electric & Gas actions regarding this event. LER 354/1997-23-00 was closed in NRC Inspection Report 354/1997-09 section M8.2. The inspectors reviewed this supplemental LER (97-23-01) and identified no additional findings of significance.

OA6 Management Meetings

a. Exit Meeting Summary

On July 5, 2001, the inspectors presented their overall findings to members of PSEG Nuclear management led by Mr. Tim O'Connor. PSEG Nuclear management stated that none of the information reviewed by the inspectors was considered proprietary.

ATTACHMENT

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Andy Caplinger, Loss Control & Insurance Program Manager
Terry Cellmer, Radiation Protection Manager
Matt Conroy, Maintenance Rule Supervisor
Mike Dammann, Maintenance Manager - Controls & Power Distribution
R. Fisher, Security, Access Authorization
G. Gibson, Manager, Security
M. Ivanick, Security
J. Johnson, Security
Kurt Krueger, Operations Manager
Gene Nagy, Plant Engineering Manager
Devon Price, Assistant Operations Manager
R. Ritzman, Licensing
Gabor Salamon, Nuclear Safety & Licensing Manager
T. Straub, Security
Larry Wagner, Director - Site Work Integration & Management

b. List of Items Opened, Closed, and Discussed

Opened/Closed

05000354/2001-07-01	NCV	Failure to adequately identify and control a SACS relief valve resulting in the use of an incorrect component that adversely impacted SACS system operability. (Section 4OA3.1)
Event No. 38042	60-Day Phone Call	Inadvertent Loss of A RPS Bus. (Section 4OA3.2)
05000354/1997-23-01	LER	Core Spray Nozzle Weld Through-wall Leak. (Section 4OA3.3)

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 Updated Final Safety Analysis Report
Technical Specification Action Statement Log (SH.OP-AP.ZZ-108)
HCGS NCO Narrative
HCGS Plant Status Report
Service Water System Malfunction (HC.OP-AB-ZZ-0122)
Weekly Reactor Engineering Guidance to Hope Creek Operations

Reactor Coolant System Pressure Isolation Valve Leakage Determination (HC.OP-GP.ZZ-004)
Plant Access Training - Fitness for Duty General Worker, Escort and Supervisor Study Guide, July 20, 2000
QA Assessment 2001-0239, Fitness for Duty Program, May 21, 2001
QA Assessment 2001-0054, Alarm Stations and Communications, March 6, 2001
QA Assessment 2000-038, Security Lock and Key Control, March 31, 2000
QA Assessment 2000-0068, Corrective Action Program Administration, March 17, 2000
QA Assessment 2000-0083, Security Access Control, March 21, 2000
QA Assessment 2000-0114, Security Contingency Response, May 5, 2000
QA Assessment 2000-0440, Access Authorization, December 8, 2000
QA Assessment 2000-0305, Security Plans and Procedures, September 22, 2000
QA Assessment 2000-0193, Fitness For Duty Program, June 28, 2000

Design Packages

DCP 4HE-0390	Removal of SSWS Vacuum Breaker Valves
DCP 80003631	Replacement of Safety Auxiliary Cooling System Fuel Pool Cooling Inlet and Outlet Cross-Tie Valves
DCP 4EE-0435/0436	RCIC/HPCI Condensate Storage Rank Suction Valve Automatic Swapover Instrument Setpoint Changeover to Account for Vortexing
DCP 4EC-03674	Replacement of Class 1E batteries
ECA 4HE-025804	SW Strainer Small Bore Drain Line Replacement
ECA 80004281	Control Rod Drive Pump Low Suction Pressure Trip Time Delay
ECA 80005494	Jet Pump Differential Pressure Indicator Abandoned-in-Place
ECA 80007100	RCIC steam Isolation Time Delay
ECA 80011019	RHR Heat Exchanger SACS Valve Rubber Seat Removal
ECA 80012185	Main Transformer C Phase Replacement
ECA 80018661	Core Spray “B” Loop Injection Line Orifice Replacement/Enlargement

Notifications and Evaluations

20014833	RHR NPSH
20027065	Replacement of SACS Valve Rubber Seats
20038462	Evaluation of Thermal Overload Relays vs. RG 1.106
20043669	Core Spray Orifice
20070795	Incomplete Evaluation of Notification 200007355
80003631	SACS Fuel Pool Cross Tie Valves
80007888	Replacement of SACS Valve Rubber Seats
970717261	SW Vacuum Breakers
981223156	Incorrect Relay in Fuel Oil Transfer Control Circuit (Level Switch Calibration Single Side Tolerance
981228174	RHR Shutdown Cooling During a LOP

UFSAR Changes

HCN 1998-038:	related to DCP 4HE-0390

HCN 1999-013:	Clarification of how RHR Shutdown Cooling Mode Functions during a LOOP
HCN 1999-039:	related to DCP 4EE-0435/0436
HCN-1999-063:	Replacement of Class 1E Batteries
HCN 2000-002:	Revise Net Positive Suction Head for Residual Heat Removal Pumps
HCN 2000-007:	related to DCP 80003631
HCN 2000-064:	Net Positive Suction Head Requirements for Core Spray Pumps
HCN-2001-006:	Commitment to Regulatory Guide 1.106
HCN 2001-016:	Changes to UFSAR Chapter 15 to Reflect Updated Analysis for Line Breaks Outside of Containment

Procedure Changes

HC.IC-EU.KJ-0001(Q), Rev 0: Astro-Med Recorder/Equipment Setup for EDG Related Surveillance Testing
HC.OP-AR.ZZ-0003(Q), Rev 9: Alarm response procedure related to DCP 4EE-0435/0436
HC.OP-AR-ZZ-0006(Q), Rev 14: Alarm response procedure related to DCP 4EE-0435/0436
HC.OP-EO.ZZ-101(Q), Rev 8: Reactor Pressure Vessel Control (EOP)
HC.OP-EO.ZZ-101A(Q), Rev 1: ATWS-RPV Control (EOP)
HC.OP-EO.ZZ-102(Q), Rev 10: Containment Control (EOP)
HC.OP-EO.ZZ-103/4(Q), Rev 3: Conversion Document (EOP)
HC.OP-EO.ZZ-202(Q), Rev 6: Emergency Depressurization (EOP)
HC.OP-SO.BG-0001, Rev 29: Reactor Water Clean-up System Operation
HC.OP-SO.SB-0001(Q), Rev 15: Reactor Protection System Operation
HC.SA-AP.ZZ-0039(Q), Rev 9: Receipt of New Fuel
HC-ODCM, Rev 19: Offsite Dose Calculation Manual

50.59 Safety Evaluations

H1996-037:	SW Strainer Drain Line Pipe Replacement
H1998-036:	related to DCP 4HE-0390, SW Vacuum Breakers
H1998-038:	DCP-4HE-0390
H1999-003:	H98-09B, Primary Containment Integrity Verification
H1999-005:	related to HC.OP-SO.SB-0001(Q)
H1999-020:	related to HCN 1999-020, RHR Shutdown Cooling
H1999-028:	related to EOP changes based on BWROG's EPG/SAG, Rev 1
H1999-036:	related to DCP 4EE-0435/0436, HPCI/RCIC Auto Swap-Over Setpoint
H1999-054:	DCP-4EC-03674, Replacement of Class 1E Batteries
H2000-004:	related to DCP 80003631, SACS Fuel Pool Cross Tie Valves
H2000-008:	DR 80007888, SACS Valve Rubber Seat Removal
H2000-010:	DR 80011019, RHR Heat Exchanger SACS Valves
H2000-032:	ECA-80004281, CD Pump TRIP Time DELAY
H2000-033:	related to HC.OP-SO.BG-0001
H2000-034:	DR 8001-1019, SACS Valve Rubber Seat Removal
H2000-037:	ECA-80012185, Transformer Replacement
H2000-046:	DCP-80007100, RCIC Steam Isolation Time Delay

H2000-047: related to HCN 2000-002, RHR NPSH
H2001-002: related to HCN 2000-064, CS NPSH
H2001-003: related to HC.IC-EU.KJ-0001(Q)
H2001-004: related to ECA 80018661, Core Spray Orifice
H2001-008: related to HC-ODCM

50.59 Applicability Reviews (related to Procedure change)

HC.IC-CC.AB-0041(Q) MS-SRV Indication (Acoustic Monitor)
HC.IC-CC.GS-0008(Q) Containment Atmospheric Control - Hydrogen Control
HC.IC-FT.SM-0021(Q) Logic System Functional Test - NSSS Valve Control
HC.MD-CM.EA-0003(Q) Service Water Strainer Repair
HC.OP-FT.EG-0102(Q) Fuel Pool Heat Exchanger Delta P Test
HC.OP-SO.MA-0001(Q) Main Transformer Operation

50.59 Applicability Reviews (related to Calculation change)

C-0141 Pipe Stress Report for Recirculation Loop A and RHR (inside the drywell)
GU-009 Reactor Building Room Temperature with LOCA Scenarios
SC BF-0355 NSSS Transmitter Uncertainty
SC EA-0023 SACS Heat Exchanger LoLo Flow Uncertainty

Conditions Reports Associated with Modifications and/or 50.59 Reviews

CR 70000944: Multiple Process Failure with SAP, DCPs
CR 70000972: DCP-1EZ-9633 Incorrect
CR 70001442: Equipment Qualification Calculation does not Address the Margin Required in accordance with IEEE 323
CR 70001583: CICP for Freon 12 is a Disapproved Chemical but Is Currently Being Used at Hope Creek
CR 70001804: Evaluate how Abnormal Procedures are Used
CR 70001927: Unplanned Alarm in Control Room
CR 70002065: DCP-1EC-3400 Safety Evaluation Issue
CR 70002259: DCP-WB2-1ER-0058 Less Than Adequate
CR 70003179: related to Calculation BC-0002
CR 70003426: Safety Evaluation Qualification Records Not Processed in Accordance with Procedures
CR 70004420: HC.OP-SO.KF-0001 Confusing
CR 70004750: DCP-4EC-3192-1 Less Than Adequate
CR 70007332: Improper Incorporation of DCP MCR
CR 70011536: related to ECA 80018661
CR 70013929: NAP-59 Procedure Violation
CR 70016308: Improper Procedure Use
CR 70017201: CM970812166 is a Design Issue not a Maintenance Issue
CR 961029064: related to DCP 4HE-0390
CR 961107063: RCIC Time Delay
CR 981223156: Incorrect Relay in Fuel Oil Transfer Control Circuit
CR 981228174: related to HCN 1999-013
CR 990423133: related to DCP 4EE-0435/0436

Calculations

AP-0004(Q), Rev 6: CST Level Setpoints, related to DCP 4EE-0435/0436
BC-0002, Rev 4: NPSH for RHR System Pumps Suction from the Suppression Pool
BE-0016, Rev 0: Core Spray System Hydraulic Analysis
SC-AP-0001, Rev 3: CST Low Level to HPCI, related to DCP 4EE-0436
SC-AP-0003, Rev 5: CST Low Level to RCIC, related to DCP 4EE-0435
E-1.2, Verification of Short Circuit Current of Main Generator Iso-Phase Bus
E-4.1(Q), Rev.13, HC Class 1E 125 Volt Station Battery
E-018(Q), Rev. 1, “Selection of Overload Heaters for AC Motors”

Drawings

M-49-1(Q), Rev. 28, RCIC P&ID
PNI-E51-1030-0061, RCIC Logic Diagram
11-1030-0183, Sht. 6, Control Rod Drive Pump Logic Diagram

Vendor Technical Documents (VTD)

VTD 322543, Artificial Island Operating Guide
GE Design Specification 22A6237, High Pressure Coolant Injection System, February 19, 1982

Procedures Referenced during Inspection

NC.NA-AP.ZZ-0008(Q), Rev 14 and Rev 15, Control of Design and Configuration Change
NC.NA-AP.ZZ-0008(Q), Rev 16, Configuration Control Program
NC.NA-AP.ZZ-0059(Q), Rev 8: 10CFR50.59 Applicability Reviews & Safety Evaluations
NC.NA-AP.ZZ-0059(Q), Rev 9: Regulatory Change Determination and 10CFR50.59 Review Process
NC.NA-AS.ZZ-0059(Q), Rev 4: 10CFR50.59 Program Guidance
Form NC.DE-WB.ZZ-0001-4, Workbook 1, (Standard Design Change) Interface Record
Form NC.DE-WB.ZZ-0006-4, Workbook 6, (Engineering Change Authorization) Interface Record
SH.OP-AP.ZZ-0108(Q), Rev 2: Operability Assessment & Equipment Control Program

d. List of Acronyms

ALARA	As Low As Is Reasonably Achievable
AOT	Allowed Outage Time
BOM	Bill of Materials
CEDE	Committed Effective Dose Equivalent
CR	Condition Report
CST	Condensate Storage Tank
DCP	Design Change Package
DFOST	Diesel Fuel Oil Storage and Transfer
DR	Deficiency Report
ECA	Engineering Change Authorization
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
FSRs	Focused Self-assessment Reports
HCGS	Hope Creek Generating Station
HPCI	High Pressure Coolant Injection
ICD	Instrument Calibration Data
IPEEE	Individual Plant Examination of External Events
IST	Inservice Test
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PMT	Post Maintenance Testing
PSEG	Public Service Electric Gas
QA	Quality Assessment
QAAR	Quality Assurance Assessment Report
QAMF	Quality Assurance Monitoring Feedback
RCIC	Reactor Core Isolation Cooling
RF	Refueling
RG	Regulatory Guide
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SACS	Safety Auxiliaries Cooling System
SAP	Systems, Applications, & Processes
SDP	Significance Determination Process
SE	Safety Evaluation
SRA	Senior Reactor Analyst
SSCs	Structures Systems or Components
SW	Service Water
TARP	Transient Assessment Response Plan
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VTD	Vendor Technical Document