October 1, 1999

Mr. Harold W. Keiser
President and Chief Nuclear Officer
PSEG Nuclear LLC
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: NRC INSPECTION REPORT 50-354/99-05

Dear Mr. Keiser:

On August 29, 1999, 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 management, led by Mr. Dave Garchow, in an exit meeting on September 8, 1999.

This inspection was an examination of activities conducted under your license as they related to reactor safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The attached report documents the results of seven weeks of resident inspection and a one week inspection focused on radiologically controlled area access, ALARA planning, and radiation monitoring instrumentation. Within these areas the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. Negative findings were assessed using the significance determination process; all findings either "screened out" of the process or were determined to be within the licensee response band (Green).

We determined that four violations of NRC requirements occurred regarding the areas of fire protection, operation at reduced feedwater inlet temperature and safety-related battery charging operations. These violations are being treated as non-cited violations (NCVs), consistent with the Interim Enforcement Policy for pilot plants. These violations are described in the subject inspection report and have been entered into your corrective action program.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

Original Signed By:

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

Enclosure: Inspection Report 50-354/99-05

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E. Simpson, Senior Vice President and Chief Administrative Officer
M. Bezilla, Vice President - Nuclear Operations
D. Garchow, Vice President - Technical Support
M. Trum, Vice President - Maintenance
T. O'Connor, Vice President - Plant Support
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J. McMahon, Director - QA/Nuclear Training/Emergency Preparedness
G. Salamon, Manager - Licensing
R. Kankus, Joint Owner Affairs
A. Tapert, Program Administrator
J. J. Keenan, Esquire
Consumer Advocate, Office of Consumer Advocate
W. Conklin, Public Safety Consultant, Lower Alloways Creek Township
M. Wetterhahn, Esquire
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-354
License No:	NPF-57
Report No:	50-354/99-05
Licensee:	PSEG Nuclear LLC.
Facility:	Hope Creek Nuclear Generating Station
Location:	P.O. Box 236 Hancocks Bridge, NJ 08038
Dates:	July 11 - August 29, 1999
Inspectors:	J. D. Orr, Senior Resident Inspector J. T. Furia, Senior Radiation Specialist
Approved By:	Glenn W. Meyer, Chief, Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

Hope Creek Generating Station
NRC Inspection Report 50-354/99-05

The report covers a 7-week period of resident inspection using the guidance contained in NRC Inspection Manual Chapter 2515*.

Inspection findings were assessed according to potential risk significance and were assigned colors of *green, white, yellow, or red.* The inspection found only green findings, which were indicative of issues that, while not necessarily desirable, represented little risk to safety. *White* findings would have indicated issues with some increased risk to safety and which may have required additional NRC inspections. *Yellow* findings would have indicated more serious issues with higher potential risk to safety and would have required the NRC to take additional actions. *Red* findings would have represented an unacceptable loss of margin to safety and would have resulted in the NRC taking significant actions that could have included ordering the plant to shut down. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

Cornerstone: Mitigating Systems

- Green. NRC inspectors identified a long-standing degraded fire protection barrier in the 117' elevation cable spreading room (CSR). The inspectors identified an open 4 inch floor drain valve that provided a vent path and would have degraded the effectiveness of the automatic CO2 fire suppression system. The NRC staff used the significance determination process (SDP) and determined that this longstanding problem had a minimal impact on safety due to the alternative safe shutdown and additional firefighting capabilities which existed, a conservative assumption for medium degradation of the automatic CO2 suppression system, and the low likelihood of a fire in the CSR. This issue was treated as a non-cited violation. (Section 1R05)

- Green. NRC inspectors identified improper fire protection compensatory actions for a degraded condition in the 117' elevation cable spreading room (CSR). PSEG had implemented an hourly firewatch for a degraded fire protection alarm in the 117' elevation CSR, but the Hope Creek fire protection procedures specified a continuous fire watch. This issue had a minimal impact on safety due to the frequency of the existing fire watch and the low likelihood of a fire in the CSR. (Section 1R05)

Cornerstone: Barrier Integrity

- Green. Control room operators failed to appropriately identify abnormal lineups in the primary containment instrument gas (PCIG) and feedwater heating systems after a reactor recirculation runback. The operators' failure to promptly correct these abnormal lineups placed the plant outside of its licensing basis. In the case of the feedwater heating system abnormal lineup, the plant was returned to 100% power with feedwater inlet temperature was at a reduced temperature. The reduced feedwater inlet temperature affected the core thermal performance and placed additional strain on the fuel barrier during the recovery to full power. Reactor engineers did not effectively monitor the plant recovery and contributed to the error in operation with reduced feedwater inlet temperature. The NRC inspectors noted that operation at reduced feedwater temperature is prohibited to protect the fuel barrier integrity. This problem related to the fuel barrier had a minimal impact on safety as determined by the SDP because no immediate or long-termed degradation of the fuel barrier occurred. This problem was treated as a non-cited violation.

Operators were indirectly alerted to the abnormal PCIG lineup by a different alarm 45 minutes after the fact. The abnormal PCIG lineup was then promptly corrected by the operators. An NRC risk analyst conducted an assessment of the risk associated with the abnormal PCIG lineup and concluded that the overall plant risk was minimal. (Section 1R14).

Performance Indicator Verification

- The inspectors identified several errors in historical data and one error in recent data (since the start of the pilot program and NRC PI submittal) for the Safety System Unavailability, Residual Heat Removal System performance indicator (PI). The NRC inspectors determined that the RHR unavailability remained green (less than 2%) and changed to about 1.3% from 0.8%. The historical errors were carried forward from an old PSEG performance indicator database and were submitted to the NRC on a "best faith effort." The one error in recent data was a failure to include a support system unavailability, specifically station service water, into the RHR unavailability. PSEG initiated Notification 20003722 to correct the RHR unavailability PI, verify all previous NRC PI submittals, and improve the verification processes and validity of future PIs. (Section 4OA2)

Other

- Green. Technicians did not provide adequate fuse protection and isolation for a non-safety-related single cell battery charger installed on the safety-related batteries. This self-identified violation was reported in LER 99-007-00 and was treated as a non-cited violation. This issue had a minimal impact on safety as determined by the SDP because the batteries were able to properly perform their safety function. (Section 4OA4)

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Report Details

SUMMARY OF PLANT STATUS

Hope Creek operated at or near full power until August 27, 1999. A reactor recirculation runback to about 70% power occurred on August 20th as a result of an electrical transient from a lightning strike in the Salem 500kV switchyard. The plant was returned to full power on August 21st. A scheduled plant shutdown was initiated on August 27th to replace the A reactor recirculation pump seal and reduce main turbine bearing vibrations with a balance weight.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R03 Emergent Work

a. Inspection Scope

The inspectors reviewed troubleshooting activities and work controls associated with an A station service water pump inservice test failure.

b. Observations and Findings

There were no findings identified.

1R04 Equipment Alignments

a. Inspection Scope

The inspectors performed partial redundant equipment alignment verifications during system outages on the instrument air system, A 1E switchgear room cooler, A/C core spray subsystem, and D emergency diesel generator H fuel oil storage tank.

b. Observations and Findings

There were no findings identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed a walkdown of the 77' and 117' elevation cable spreading rooms' (CSRs) fire protection systems. The inspectors also reviewed fire impairments associated with the CSRs.

b. Observations and Findings

The control building 117' elevation CSR has an automatic carbon dioxide (CO_2) suppression system and a fire and smoke detection system. The CO_2 is the primary means to suppress a fire in the 117' CSR. A manual pre-action system is an available backup fire suppression system for the 117' CSR. The CSR boundaries must be sealed for the CO_2 to be completely effective.

During a walkdown on August 17, 1999, the NRC inspectors identified a 117' CSR 4-inch floor drain isolation valve that was open. Updated Final Safety Analysis Report (USFAR) section 9.5.1.1.9 described the floor drain valve as normally closed to prevent CO_2 fire suppression agent dissipation. The inspectors informed the PSEG safety and loss prevention unit, who restored the drain valve to a closed position. These personnel suspected that the drain valve may have been open for a very long time, since the valve was difficult to operate, was not labeled and was not included in the valve configuration data base. PSEG determined that the open drain valve was not in accordance with Hope Creek operating license condition 2.C.(7) and reported the license condition violation to the NRC on August 18, 1999. PSEG entered the problem into its corrective action program as Notification 70000673.

The NRC inspectors used the fire protection significance determination process (SDP) to understand the potential risk significance on safe shutdown (SSD) capability. Phase 2 of the SDP was performed to determine the significance of this finding. The inspectors assumed that the CO_2 suppression system as well as the fire barrier had a medium level of degradation and had existed in that condition for more than 30 days. The inspectors compared the initiating event likelihood for the CSR with the remaining SSD mitigating capability and determined that the issue had a minimal impact on safety. Therefore, the inspectors considered this license violation and problem within the licensee response band (green). This violation is being treated as a non-cited violation, consistent with the Interim Enforcement Policy for pilot plants. This violation is in the PSEG corrective action program as Notification 70000673. **(NCV 50-354/99-05-01)**.

The NRC inspectors reviewed PSEG's immediate corrective actions for the open drain valve problem and determined that had the valve not been closeable, PSEG would have implemented a continuous fire watch for an open drain valve in the 117' elevation CSR. The inspectors noted that this was inconsistent with an hourly fire watch for another known degraded fire barrier condition in the 117' CSR. PSEG reviewed the compensatory actions at the inspectors' request and determined that the appropriate compensatory actions were not in place for this degraded fire barrier in accordance with Hope Creek fire protection procedure, *HC.FP-AP.ZZ-0004(Q).* A continuous fire watch was specified, not an hourly fire watch as implemented. PSEG entered the problem into its corrective action program as Notification 70000692. This failure to correctly implement the fire impairment compensatory action represented a procedure violation. This issue was a green finding and had minimal impact on safety due to the frequency of the existing fire watch and the low likelihood of a fire in the CSR. This violation is being treated as a non-cited violation, consistent with the Interim Enforcement Policy for pilot plants. This violation is in the licensee's corrective action program as Notification 70000692. **(NCV 50-354/99-05-02)**.

1R09 Inservice Testing

a. Inspection Scope

The inspectors reviewed inservice test results and the adequacy of inservice test procedures associated with the D safety auxiliaries cooling system (SACS) pump, the reactor core isolation cooling (RCIC) system pump, and the D emergency diesel generator H fuel oil storage tank transfer pump.

b. Observations and Findings

There were no findings identified.

1R10 Large Containment Valves

a. Inspection Scope

The inspectors reviewed administrative controls for the drywell and suppression chamber purge system isolation valves.

b. Observations and Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed maintenance rule implementation for three high risk significant equipment failures; Action Requests (AR) 990401439, ‘C’ station service water (SSW) strainer failure; AR 990609245 ‘C’ SSW strainer breaker tripped; and AR 990601157, 'B' control room ventilation train trip.

b. Observations and Findings

Each of the ARs was initially screened as a system functional failure (SFFs). System engineers had not yet screened ARs 990609245 and 990601157 for maintenance preventable system functional failures (MPFFs). The NRC inspectors asked the Hope Creek maintenance rule program manager for the results on the MPFF determinations. The maintenance rule program manager subsequently discovered that MPFF determinations were not yet made and in all likelihood would have gone unexplored because of a recent conversion to a new PSEG administrative computer system. PSEG performed a search and found eight other system functional failures which could have gone unnoticed for maintenance rule implementation.

The third AR, 990401439, had been reviewed by the system engineer for MPFF determination, who had determined that the specific SSW strainer equipment failure was not a system functional failure and was reclassified as such. The NRC inspectors reviewed the circumstances of the strainer failure and disagreed with the system engineers decision. The system engineer reconsidered the failure, agreed with the inspectors, and reclassified the failure as a SFF and a MPFF.

1R13 Maintenance Work Prioritization

a. Inspection Scope

The inspectors evaluated PSEG's on-line risk management for an instrument air system outage on July 19, 1999.

b. Observations and Findings

There were no findings identified.

1R14 Nonroutine Plant Evolutions

Full Power Operation at Reduced Feedwater Temperature

a. Inspection Scope

The inspectors reviewed the circumstances surrounding an August 20 reactor recirculation runback and recovery.

b. Observations and Findings

On August 20, 1999 at 11:58 p.m., a lightning strike in the Salem switchyard induced a voltage transient in the Hope Creek plant. (The Hope Creek and Salem switchyards are normally tied through a 500 kv transmission line.) The Salem plants were unaffected, but an incoming Salem switchyard transmission line was tripped by the lightning strike. No Hope Creek electrical busses were lost, however several pieces of equipment tripped; back-up equipment auto-started as designed due to these equipment trips. Four plant systems were affected that had an immediate effect on the continued operation of the Hope Creek plant: all drywell coolers tripped, the A reactor feed pump turbine tripped, each motor operated extraction steam valve to the high pressure feedwater heaters closed, and both primary containment instrument gas (PCIG) compressors locked out.

Through interviews and log review, the NRC inspectors determined that operators executed the appropriate abnormal procedures. However, the inspectors determined that operators failed to appropriately notice abnormal lineups in the primary containment instrument gas and feedwater heating systems. The operators' failure to promptly correct these abnormal lineups placed the plant outside of its licensing basis. These problems are further discussed in detail in the following paragraphs.

The trip of the A RFPT and subsequent reactor water level lowering to 30 inches caused, as designed, an intermediate reactor recirculation runback from 100%. The plant was stabilized at about 70% power. Operators promptly restarted all drywell coolers to maintain drywell temperatures normal. The high pressure feedwater heaters extraction steam motor operated valves (MOV) closed during the electrical transient.

However, control room operators did not notice the abnormal feedwater heating lineup until after the plant had been recovered to 100% power. The operators began a power ascension at about 12:52 a.m. The operators believed that plant conditions were stable, the cause of the runback was understood, and that equipment lineups had been restored to a point of allowing a plant recovery to 100%. The operators also understood that continued plant operation below about 85% power was potentially undesirable because of known main turbine bearing vibration problems at reduced load. Since restart from the refuel outage ending in April 1999, the number 7 main turbine bearing had experienced increased vibrations at reduced loads. The vibration problem also tended to increase with time at reduced load. See report section 1R16 for further details about the main turbine bearing vibrations. The operators completed the ascension to 100% power at 4:19 a.m.

At about 06:00 a.m., the operators became concerned that some plant parameters were not the expected normal full power values: reactor pressure was about 10 psig less, feed flow was about 7% less, and RFPT controller demand was about 3% less. Operators had also been concerned about core thermal values (core maximum fraction of critical power, core maximum average planar linear heat generation rate, and core maximum fraction of limiting power density) throughout the duration of the transient. The operators updated reactor engineers via telephone on the core thermal behavior since shortly after the runback occurred. In their investigation of the unusual plant parameters, the operators discovered that feedwater inlet temperature was low and the high pressure feedwater heaters extraction steam MOVs were closed. Feedwater temperature was at about 365 degrees Fahrenheit during the event compared to the normal 420 degrees Fahrenheit. Hope Creek operating license condition 2.C.(11) prohibits full power operation with a feedwater inlet temperature below 400 degrees Fahrenheit without prior NRC approval. The operators reported the operating license condition violation to the NRC Operations Center on August 21, 1999.

The operators entered the abnormal procedure for loss of feedwater heating and recovered the high pressure feedwater heaters. Reactor engineers and nuclear fuels engineers were called on-site to investigate the immediate and long-term effects on the fuel. The cooler water entering the reactor affects the strain induced on the fuel rod cladding, potentially challenging the integrity of this barrier. The nuclear fuels engineers determined that one PSEG administrative power ramp rate limit was exceeded, but that no technical specification core thermal limits had been exceeded. The nuclear fuels engineers also determined that sufficient margins existed on core thermal values and the plant was not operated outside the design basis. Normal reactor coolant activity samples and offgas system activity during subsequent plant operation confirmed that in all likelihood, no long-term fuel defects were induced during the plant transient.

PSEG initiated a level 1 root cause investigation for the equipment responses, operator errors and reactor engineer weaknesses that occurred related to the lightning strike transient, reactor recirculation runback and subsequent plant recovery at reduced feedwater temperature. The corrective actions were initiated as Notification 20003538.

The NRC inspectors used the significance determination process (SDP) to understand the potential risk significance of operation at reduced feedwater temperature. The inspectors noted that the basis for the feedwater temperature limit is to protect the fuel barrier integrity due to core analysis uncertainties. This problem related to the fuel barrier screened out in phase 1 of the SDP since the additional strain placed on the fuel barrier during the full power recovery did not cause any immediate or long-term degradation of the clading. Also, the performance indicator (PI) for reactor coolant system specific activity measures the consequences of any fuel barrier problem. This violation of Hope Creek operating license condition 2.C.(11) is being treated as a non-cited violation, consistent with the Interim Enforcement Policy for pilot plants. **(NCV 50-354/99-05-03)**

The safety lockout of the PCIG compressors prevented the compressors from auto cycling on PCIG system receiver pressures. PCIG compressors are necessary to support the main steam isolation valve sealing system and is required by technical specification 3.6.1.4. The operators did not notice the immediate lockout of both PCIG compressors until an instrument gas low pressure overhead annunciator was received about forty five minutes after the transient. The PCIG compressors were made available after the annunciator was received and operators responded locally to reset the safety lockout. An NRC risk analyst conducted an assessment of the risk associated with the locked-out PCIG compressors and concluded that the overall plant risk was minimal.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed all operability determinations initiated during the report period and included degraded conditions on the No. 2 turbine control valve and the 'A' master trip solenoid valve.

b. Observations and Findings

There were no findings identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed all Hope Creek operator workarounds to assess the impact on the operators' ability to effectively respond to plant events.

b. Observations and Findings

Elevated turbine bearing vibrations at reduced loads were present during the plant startup after refuel outage No. 8 in April 1999 and were experienced on every scheduled plant load reduction below 85% power. The vibrations tended to increase with load reduction and the time spent at reduced load. The vibrations could increase to the point where operators would have been required to manually trip the turbine. Hope Creek operators considered this turbine bearing vibration problem an operator workaround. PSEG had scheduled a plant shutdown on or about September 11 to install a weight balance and reduce the main turbine bearing vibrations. In the interim, Hope Creek operators were provided with guidance to recover any unplanned load reduction in an expeditious manner after the initiating problem was corrected and understood. The guidance emphasized plant safety above load recovery. Nonetheless, the turbine bearing vibration problem was an operator workaround and contributed an element of urgency for recovery to full power operation after any plant transient. On August 20, 1999, a reactor recirculation runback to about 70% power occurred. The details of the reactor recirculation runback are described in Section 1R14, *NonRoutine Plant Evolutions.* Some operator errors occurred during the recovery and PSEG considered that this workaround may have been a contributing factor. This and other human error problems during the event were included in PSEG's corrective action program. The scheduled plant shutdown was moved up to August 27, 1999, and the main turbine vibration problems were corrected.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the results and adequacy of post maintenance tests associated with condensate storage tank low level instrument design changes for the reactor core isolation cooling and high pressure coolant injection systems. The inspectors also reviewed post maintenance tests for the D emergency diesel generator H fuel oil storage tank drain and refill.

b. Observations and Findings

There were no findings identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the performance and adequacy of technical specification surveillance tests for the reactor core isolation cooling system steam leak detection, D emergency diesel generator 184 day fast load, and 1E 4kV feeder breaker degraded voltage instrumentation.

b. Observations and Findings

There were no findings identified.

Cornerstone: Emergency Preparedness

1EP1 Drill, Exercise, and Actual Events

a. Inspection Scope

The resident inspectors observed control room simulator and emergency operations facility performance as it related to event classification, notification and protective action recommendations during a facilities emergency drill.

b. Observations and Findings

There were no findings identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

OS1 Access Control

a. Inspection Scope

The inspector reviewed the access control program by examining the controls established for exposure significant areas, including postings, markings, control of access, dosimetry, surveys and alarm set points. Areas selected were located throughout Hope Creek.

b. Observations and Findings

There were no findings identified.

OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed radiological work performance during the refueling outage in February-March 1999 (RF08). Selected jobs which exceeded their exposure estimates were examined relative to: work integration; coordination between working groups; shielding and other engineering controls to minimize exposures; accuracy of person-hour and effective dose rate estimates; post-job reviews; and, ALARA reports. The inspector also examined audits and self-assessments conducted by the licensee of its ALARA program.

b. Observations and Findings

License exposures during RF08 were 15 person-rem above estimated (less than 10% above the total estimated for the outage). The majority of the additional exposure was traceable to significantly higher than anticipated exposure rates under vessel. The licensee anticipates that the recently installed iron reduction system will aid in lowering dose rates in this and other plant areas by reducing the amount of crud in the primary system over time.

OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed and verified licensee records of survey instrument, personnel contamination monitor and whole body counter calibration, daily check, maintenance and repair. Records of calibration source traceability to the National Institute of Standards and Technology (NIST) primary standards were also reviewed and verified.

b. Observations and Findings

There were no findings identified.

4. OTHER ACTIVITIES [OA]

4OA2 Performance Indicator Verification

a. Inspection Scope (71151)

The inspectors verified the accuracy of and methods used to calculate the *Safety System Unavailability, Residual Heat Removal* performance indicator (PI) for the previous 36 months. Limiting Condition for Operation logs, control room operating logs and maintenance rule electronic data bases were reviewed.

b. Observations and Findings

During the inspection, the NRC identified errors in the *Safety System Unavailability, Residual Heat Removal* (RHR) PI data submitted to the NRC. The errors included several omissions of RHR train unavailability throughout the historical data (prior to May 30, 1999) and one error in June 1999. The error in June 1999 did not account for a planned 'B' station service water loop outage that rendered the 'B' RHR train unavailable. The inspectors determined that the RHR unavailability PI remained green and changed to about 1.3% from 0.8% and did not exceed the white threshold (2%). PSEG initiated a corrective action Notification (#20003722) to improve the completeness and accuracy of all future NRC PI submittals.

4OA4 Other

a. (Closed) LER 99-007-00: license condition violation - class-1E battery charging operation. This LER described the improper installation of a non-1E battery charger on single cells of an operable class-1E battery. The inspectors evaluated the improper battery charge installation and the circumstances in the LER. The inspectors determined that the charger installation, although not in accordance with the license requirements, did not render the class-1E battery inoperable. Specifically the battery charger installation did not include redundant class-1E fuses on the charger leads. PSEG promptly removed the improperly installed battery charger and initiated corrective actions (Action Request 990614116) to preclude repetition. This issue was considered a GREEN finding with minimal impact on safety as determined by the significance determination process. Specifically, although the event did not render the class 1E batteries inoperable, it did have the potential to impact the operation of safety-related equipment. The improper battery charger installation was a violation of Hope Creek additional license condition amendment 114. This violation is being treated as a non-cited violation, consistent with the Interim Enforcement Policy for pilot plants. This violation is in the corrective action program as Action Request 990614116. **(NCV 50-354/99-05-04)**

4OA5 Management Meetings

a. Exit Meeting Summary

On September 8, 1999, the inspectors presented their overall findings to members of PSEG Nuclear management led by Mr. Dave Garchow. PSEG Nuclear management acknowledged the findings presented and did not contest any of the inspectors' conclusions. Additionally, they stated that none of the information reviewed by the inspectors was considered proprietary. During this inspection, four non-cited violations were identified as discussed in the report. If PSEG contests these NCVs, a response should be provided within 30 days of the date of this inspection report, with the basis for the denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region I, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Hope Creek facility.

ITEMS OPENED AND CLOSED

Open/Closed

50-354/99-05-01	NCV	License condition violation - degraded fire protection barrier in the 117' elevation cable spreading room. (Section 1RO5)
50-354/99-05-02	NCV	Failure to implement appropriate fire protection impairment compensatory actions. (Section 1RO5)
50-354/99-05-03	NCV	License condition violation - operation at reduced feedwater inlet temperature. (Section 1R14)
50-354/99-005-04	NCV	License condition violation - class-1E battery charging operation. (Section 4OA4)
50-354/99-007-00	LER	License condition violation - class-1E battery charging operation. (Section 4OA4)