U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-354/89-16

License NPF-57

Licensee: Public Service Electric and Gas Company
P. O. Box 236
Hancocks Bridge, New Jersey 08038

Facility: Hope Creek Generating Station

Dates: August 15, 1989 - September 25, 1989

Inspectors: Glenn W. Meyer, Senior Resident Inspector

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Approved: Paul P. Swetland 11/1/89
P. D. Swetland, Chief, Projects Section 2A Date

Inspection Summary:
Inspection 50-354/89-16 on August 15, 1989 - September 25, 1989

Areas Inspected: Resident safety inspection of the following areas: operations, radiological controls, maintenance & surveillance testing, emergency preparedness, security, engineering/technical support, safety assessment/assurance of quality, and Licensee Event Report and open item followup.

Results: The inspectors did not identify any violations. There were three lack of attention to detail incidents within the operations department. An executive summary follows.

EXECUTIVE SUMMARY

Hope Creek Inspection Report 50-354/89-16

August 15, 1989 to September 25, 1989

Operations: Operators safety conscious attitude and controlled pace during the recent reactor startup and shutdown were noteworthy. However, three incidents involving operations department attention to detail errors were identified in this report period indicating the need for more cautious implementation of outage activities.

Radiological Control: The inspectors found an unidentified and unmarked contaminated area on the refuel floor. Evaluations concluded that the areas had been contaminated for only five minutes. The corrective actions were effective. PSE&G found an unlocked radiation door in the solid radwaste area, which should have been closed as specified in administrative procedures. The inspectors noted prompt cleanup efforts in response to contamination incidents associated with spills and backed up drains.

Maintenance/Surveillance: An unqualified valve position limit switch was discovered on one primary containment instrument gas system isolation valve.

Emergency Preparedness: The inspector reviewed the declaration and notification generated when the emergency information system became inoperable and concluded that actions were acceptable.

Security: The inspectors discovered an empty vodka bottle in the reactor building which was left there during construction activities and concluded that Hope Creek's response was appropriate.

Engineering/Technical Support: An open item was closed regarding hydrogen recombiner trips. Engineering support toward locating a packing leak in the drywell contributed to the prompt isolation of the leak.

Safety Assessment/Assurance of Quality: Three incidents occurred due to attention to detail errors in the operations department which indicate that continued management attention is warranted in this area.

## Details

## 1.0 SUMMARY OF OPERATIONS

The unit entered this report period (August 15 - September 25) at approximately 90% power due to the end of cycle coastdown. The reactor continued power operations setting a new continuous run record of 175 days which ended on August 30, 1989 when a failure of a scram air header soldered joint caused a reactor scram. The unit was returned to power on September 1 and continued power operations until shutting down on September 16 to commence the station's second refueling outage.

## 2.0 OPERATIONS (71707, 93702)

### 2.1 Inspection Activities

The inspectors verified that the facility was operated safely and in conformance with regulatory requirements. Public Service Electric and Gas (PSE&G) Company management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, independent verification of safety system status and Limiting Conditions for Operation, and review of facility records. These inspection activities were conducted in accordance with NRC inspection procedure 71707. The inspectors performed 212 hours of normal and back shift inspection including weekend and holiday inspection on:

- September 2, 1989 9:30 a.m. - 3:30 p.m.
- September 24, 1989 1:05 p.m. - 5:35 p.m.

### 2.2 Inspection Findings and Significant Plant Events

A. On August 30, Hope Creek scrammed from 81% power due to a failed soldered joint in the scram air header connection to control rod drive (CRD) 34-59. The 1/2 inch copper pipe supplying air to CRD 34-59 completely separated from the copper T joint/reducer (1 1/2 + 1/2 inch) causing a rapid depressurization of scram air header pressure, and resulting in half of the control rods inserting. The resultant multiple rod insertions caused reactor vessel level to shrink below level 3, and an automatic scram actuation occurred on low vessel level signal. The response of plant equipment and plant operators to the scram was as expected and without additional problems.

The Hope Creek Quality Assurance Department verified the leak test and pull test which were conducted on all 185 scram air header/CRD joints. In addition, approximately 20 joints were radiographed to verify complete pipe insertion. The failed air header joint was resoldered and five other soldered joints which were identified as containing pin hole solder leaks or questionable voiding in the soldered joints. A packing leak on the reactor core isolation cooling system steam supply bypass valve was repaired and the unit was returned to power on October 1.

The inspectors observed the unit startup including portions of the following procedures:

- OP-IO.ZZ-003 Startup from cold shutdown to rated power
- OP-SO.AE-001 Feedwater system operation
- OP-SO.GS-001 Containment atmosphere operation
- OP-SO.AC-001 Main turbine operation
- OP-SO.MA-001 Main generator and exciter operation and switching

The inspectors concluded that the operators approached reactor operations at a controlled pace and with a safety conscious attitude. The operators were well trained and knowledgeable on plant equipment and procedures.

B. On September 16, a unit shutdown was conducted and Hope Creek entered its second refueling outage. The inspector witnessed portions of the reactor shutdown and the transition to cold shutdown and refueling operations.

The inspector determined that the shutdown activities were well controlled and conducted in accordance with station procedures. Compliance with Technical Specifications was also verified during the various operational condition changes from power operations to refueling.

C. On September 17, a personnel error resulted in approximately 800 gallons of water being spilled from the reactor vessel and causing contamination of the 54 foot elevation of the reactor building. While attempting to drain the D residual heat removal (RHR) loop, 800 gallons of water from the B RHR loop were inadvertently directed to the reactor building drain system which backed up and contaminated the reactor building 54 foot elevation.

A work control operator gave a detailed preparatory briefing to equipment operators (EOs) which included physically pointing out the D RHR vent valve would be opened to vent the D RHR header, directing that donning protective clothing (PC) would be needed to perform the work, and providing the EOs with written confirmation of the vent valve to be used. However, the written instructions incorrectly identified the B RHR loop valve. The EOs deferred to the written instructions even though this valve was not the one pointed out to them and did not require donning PCs. The EOs opened the B RHR vent valve and commenced draining the B RHR loop (which was operating for shutdown cooling) to the reactor building drain system. When personnel detected the drain system backing up at a floor drain, the draining was promptly stopped.

The operations department conducted a detailed event analysis and root cause determination. The operators involved were counseled and the operations department was admonished to proceed at an appropriate safe pace to support outage activities. The inspector concluded that these corrective actions were appropriate and acceptable.

D. On September 23, the D core spray pump was operated for approximately 45 minutes with both the minimum flow and the full flow test lines isolated. This occurred due to inadequate system restoration following the removal of a tagged maintenance boundary. The test isolation valves were left closed following maintenance and the operations department did not resolve the documented abnormal condition prior to running a surveillance test. The D core spray pump started when the D diesel generator sequencing network was tested and was turned off at the conclusion of the test. The D core spray pump had not been declared operable following its maintenance, however operations personnel thought an adequate flow path for the pump was available prior to allowing the pump to start.

The operations department reviewed the event in detail and counseled the individuals involved. Appropriate tagging procedure enhancements were implemented. Outage activities were halted for the operations department to discuss this event, implement tagging procedure changes, and caution personnel to take time to accomplish activities correctly. The D core spray pump operated normally and successfully during an inservice test (IST) run within 48 hours of the incident. The D core spray pump is currently awaiting a review by an IST engineer to confirm pump operability. The inspector reviewed the IST test data and concluded that no pump degradation had occurred and that corrective action was appropriate and acceptable. Attention to detail errors are further discussed in Section 9.A.

## 3.0 RADIOLOGICAL CONTROLS (71707)

### 3.1 Inspection Activities

PSE&G's compliance with the radiological protection program was verified on a periodic basis. These inspection activities were conducted in accordance with NRC inspection procedure 71707.

### 3.2 Inspection Findings and Review of Events

During an August 28 inspection of the refuel floor, the inspector's shoes became contaminated and were detected as such by contamination monitors in the reactor building. The contamination was minimal and was removed by cleaning the soles of the shoes. However, the areas in which the inspector had walked were not marked off as contamination areas. A subsequent survey found a contaminated area adjacent to a step off pad from the dryer/separator pit. The inspector had observed two personnel remove protective clothing (PC) while exiting across the step-off pad. This additional area was subsequently controlled and decontaminated. A PSE&G evaluation concluded that the contamination near the step-off pad resulted from loose surface contamination released when personnel exiting the dryer/separator pit removed their outer PC. The radiation work permit for working in the dryer/separator pit was modified to require the outer set of PCs be removed in the pit prior to ascending up the ladder to the step-off pad. PSE&G concluded that the contaminated area had most likely just occurred and that no further spreading of this contamination occurred. The inspector subsequently inspected the refuel floor and found no uncontrolled contaminated areas. The inspector concluded that PSE&G's response to the contamination had been acceptable and appropriate.

On August 21, the inspector found a location in the radiologically controlled area (RCA) in which someone had set up a resting area and had eaten fried chicken. PSE&G utilized random security patrols and video camera monitoring of the access route to the location in attempts to discover the person violating PSE&G policy, procedures, and training. These attempts were unsuccessful over the next four weeks and were subsequently abandoned, and the area was cleaned up. The General Manager - Hope Creek issued a letter to all department managers to reemphasize to all personnel the PSE&G policy prohibiting food in the RCA. The inspector concluded that these corrective actions were appropriate and acceptable.

At approximately 1:30 a.m. on September 20, an administratively locked radiation door was discovered unlocked and ajar. The door controls access to the capper/seamer room which stores radiological waste barrels awaiting shipment offsite. At the time of the incident, the highest radiation readings in this room was about 350 mrem/hr, 18 inches from the most radioactive barrel. The door had been unlocked by a radiation waste equipment operator who failed to lock the door upon his exit. Since the radiation levels involved are below Technical Specification requirements for a locked door this is not being cited as a violation. However, it does indicate another example of an attention to detail personnel error which has contributed to three other administratively controlled locked radiation doors being discovered unsecured. PSE&G reviewed the event and is pursuing disciplinary action against the individual involved.

The inspector noted prompt radiation protection (RP) department response to several room contamination events. Radiation protection personnel quickly decontaminated a spill in the service/radioactive waste building. Rapid RP actions minimized the severity of the contamination of the reactor building 54 foot elevation when the B RHR loop was inadvertently drained. Contamination levels reached 1.6 rad smearable and 1.36 minimum permissible concentration (MPC) airborne activity in the B core spray room. One personnel contamination resulted from the incident.

## 4.0 MAINTENANCE/SURVEILLANCE TESTING (62703, 61726)

### 4.1 Maintenance Inspection Activity

The inspectors observed selected maintenance activities on safety-related equipment to ascertain that these activities were conducted in accordance with approved procedures, Technical Specifications, and appropriate industrial codes and standards. These inspections were conducted in accordance with NRC inspection procedure 62703.

Portions of the following activities were observed by the inspector:

| Work Order | Procedure | Description |
|---|---|---|
| 891006021 | M9-ILP-03H | Type C local leak rate test on D residual heat removal suction valve |
| 890830124 | Work Standard | Pull test of air header solder joints |
| 890828081 | 1C-GP.ZZ-053 | Replacement of reactor pressure transmitter 1BBPT-N079D-B21 |
| 890901113 | Work Standard | Perform 30 day spectrum analysis of Rosemount transmitters |

| Work Order | Procedure | Description |
|---|---|---|
| 890811135 | NA | Replace high pressure coolant injection room cooler supply valve actuator |
| 890815213 | MD-GM.EG-001 | Replace seal on the A safety auxiliary coolant system (SACS) pump |

The maintenance activities inspected were effective with respect to meeting the safety objectives of the maintenance program.

## 4.2 Surveillance Testing Inspection Activity

The inspectors performed detailed technical procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspectors verified that the surveillance tests were performed in accordance with Technical Specifications, approved procedures, and NRC regulations. These inspection activities were conducted in accordance with NRC inspection procedure 61726.

The following surveillance tests were reviewed, with portions witnessed by the inspector:

- RE-ST.SE-002 APRM calibration calculations
- IC-FT.AB-003 Functional test of division 3 main steam flow
- IC-FT.FD-001 Functional test of high pressure coolant injection steam line flow isolation
- IC-FT.FD-005 Functional test of high pressure coolant injection turbine exhaust diaphragm
- IC-FT.BB-004 Functional test nuclear boiler division 2 channel B21-N691F reactor vessel level
- IC-FT.BB-011 Functional test of division 3 drywell pressure

The surveillance testing activities inspected were effective with respect to meeting the safety objectives of the surveillance testing program.

### 4.3 Inspection Findings

On August 2, PSE&G determined that a limit switch assembly on primary containment instrument gas (PCIG) system valve HV-5154 did not meet environmental qualification requirements. This limit switch provides position indication for the outboard torus isolation valve on the PCIG supply to the torus vacuum relief valves. Upon discovery, this valve was declared inoperable until valve position was verified locally and a properly qualified limit switch was installed. An investigation determined that this valve had been replaced during the pre-commercial construction period by the plant architect/engineering firm (Bechtel). Bechtel performed all work and follow-up inspections under their construction programs and procedures. Although the field instructions specified the use of conax connectors, Bechtel installed the valve without conax connectors and Bechtel's post- maintenance quality control inspection did not detect this deviation. This deviation was identified by an alert station planner who was in the field assessing work on an adjacent similar valve. Corrective action consisted of replacing the non-EQ limit switch with an environmentally qualified switch, and reviewing all similar valves to ensure EQ compliance. Action by the station planner to identify and pursue questions on the EQ qualification of this valve were noteworthy. This deviation was self-identified by Hope Creek and testing subsequent to the incident verified that the subject valve would have performed its intended isolation function without position indication. If the position indication of the valve had failed to operate, alternate methods of valve position were available including local valve indication and indirect valve position indication from remote process line parameters. Additionally, the inboard containment isolation valve is routinely shut isolating this penetration except during testing. This is considered an isolated occurrence based on the results of three earlier Hope Creek EQ inspections of drywell valves and NRC reviews of the EQ program. The inspector determined that this occurrence is a licensee-identified violation of 10 CFR 50.49 for which no citation will be issued in accordance with 10 CFR 2. (354/89-16-01)

## 5.0 ENGINEERING SAFETY FEATURE (ESF) SYSTEM WALKDOWN (71710)

### 5.1 Inspection Activity

The reactor core isolation cooling (RCIC) system was reviewed in accordance with NRC inspection procedure 71710 to identify equipment conditions that might degrade performance, to determine that instrumentation was calibrated and functioning, and to verify that valves were properly positioned and locked as appropriate. Also, the inspectors reviewed the appropriate operating, alarm response, and abnormal procedures and the instrumentation logic drawings.

5.2 Inspection Findings

The RCIC system was inspected and in-plant conditions found to be acceptable. The inspector observed the RCIC turbine overhaul activities and surveyed the drywell contamination level near valve F076 which recently had significant packing leakage (see Section 8.B). Surveillance procedures were reviewed and found to adequately fulfill the testing requirements of Technical Specifications. No deficiencies were detected which had not previously been identified by Hope Creek and were being tracked for correction. The RCIC system was found to be fully functional with equipment in good physical condition.

6.0 EMERGENCY PREPAREDNESS

The inspector reviewed records to ensure appropriate and timely notification of the NRC operations center when the Emergency Notification System (ENS) became inoperable on September 22, 1989. The inspector concluded Hope Creek's response was appropriate and the ENS was restored to service in approximately an hour and a half.

7.0 SECURITY

7.1 Inspection Activity

PSE&G's compliance with the security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries. These inspection activities were conducted in accordance with NRC inspection procedure 71881.

7.2 Inspection Findings

On August 16, the inspector found a small, empty vodka bottle in the reactor building during an inspection tour of the A fuel pool heat exchanger room. The bottle was on top of a piping support approximately ten feet above the floor and was dusty. The heat exchanger room has frequently been locked for radiological reasons. PSE&G security concluded that the bottle had been in this location since the construction period, because the manufacturer's lot number and the retail price tag dated from more than ten years ago. The General Manager - Hope Creek issued a letter to all department managers to reemphasize to all personnel the PSE&G policy prohibiting alcohol onsite and food in the radiologically controlled areas. The inspector concluded that this corrective action was appropriate and acceptable.

## 8.0 ENGINEERING/TECHNICAL SUPPORT

A. (Closed) Unresolved Item (354/87-22-02); Required hydrogen recombiner trips. The inspector had noted that high gas inlet pressure and high blower inlet temperature recombiner trips were not tested consistent with the FSAR. System engineering has determined that these automatic trips are not tested nor do they exist on Hope Creek recombiners. Communication with the vendor confirmed that Hope Creek recombiners had these trips removed prior to installation at Hope Creek. These automatic trips are not required to be tested by Technical Specification (TS) surveillance testing. PSE&G has committed to updating the UFSAR to remove reference to these automatic trips. The inspector reviewed TS, UFSAR, and vendor information and concluded the trips were not required for recombiner operation. This item is closed.

B. The prompt support of the system engineering group in response to increasing unidentified drywell leakage allowed the station to identify and isolate the leak. The leak originated from the packing on RCIC steam valve (F076). The system engineering group made significant contribution to the rapid identification and corrective action to isolate the leaking valve. This prompt action minimized drywell contamination levels and stopped the leak before it progressed to magnitude where a unit shutdown would have been required. The valve packing was replaced after the reactor scrammed on August 30, 1989.

## 9.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

During this assessment period three events involving operations department lack of attention to detail were noted. Similar maintenance department attention to detail errors were noted in the preceding inspection report 50-354/89-14. The operations department events included an unlocked high radiation door (Section 3.2.C), inadequate core spray restoration following tag removal (Section 2.2.C), and the inadvertent draining of reactor vessel water (Section 2.2.B). Although these events had minor safety significance taken in isolation, collectively they indicate the need for continued management involvement in the attention to detail area. It is significant to note that sufficient controls were in place to promptly detect the unlocked high radiation door which was identified and corrected within an hour and a half of the door being left unsecured. Controls provided multiple opportunities to prevent the inadequate core spray restoration and inadvertent reactor vessel draining. The importance of personnel performance was clearly evident in that the errors (multiple errors in some cases) circumvented the satisfactory implementation of administrative control programs. The operations department has concluded that rapid outage activity pace contributed significantly to these two events and stopped outage activities to counsel the department to take the time required to do the job right. The inspectors have sensed no undue schedule pressure from management. The effectiveness of these corrective actions and other efforts in this area will be judged based on the future reduction of attention to detail errors.

## 10.0 LICENSEE EVENT REPORT (LER) AND OPEN ITEM FOLLOWUP (92700)

A. PSE&G submitted the following event reports and periodic reports, which were reviewed for accuracy and adequacy of the evaluation. The asterisked (*) items identify reports which involve licensee identified Technical Specification violations which are not being cited based upon meeting the criteria of 10 CFR 2 Appendix C.

Monthly Operating Report for August 1989

*LER 89-016 Non-Environmentally qualified (EQ) limit switch assembly utilized in an EQ application due to installation deficiencies during pre-commercial construction activities; discussed in Section 4.3.A of this report.

B. The following previous inspection item was followed up during this inspection and is tabulated below for cross reference purposes.

| | | |
|---|---|---|
| Closed | 87-22-02 | Section 8.A |

## 11.0 EXIT INTERVIEW (30703)

The inspectors met with Mr. J. Hagan and other PSE&G personnel periodically and at the end of the inspection report period to summarize the scope and findings of their inspection activities.

Based on Region I review and discussions with PSE&G, it was determined that this report does not contain information subject to 10 CFR 2 restrictions.