U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-354/86-18

Docket No. 50-354

License No. CPPR-120 Priority - Category -

Licensee: Public Service Electric & Gas Company
80 Park Plaza
Hancocks Bridge, New Jersey 08038

Facility Name: Hope Creek Generating Station, Unit 1

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: March 3-14, 1986

Inspectors: D. Florek 4/9/86
D. Florek, Lead Reactor Engineer date

J. Golla 4/9/86
J. Golla, Reactor Engineer date

Fred Paulitz 4/10/86
F. Paulitz, Reactor Engineer date

L. Wink 4/9/86
L. Wink, Reactor Engineer date

Approved by: P. Eselgroth 4/15/86
P. Eselgroth, Chief, Test Programs Section, OB, DRS date

Inspection Summary: Inspection on March 3-14, 1986, (Inspection Report No. 50-354/86-18)

Areas Inspected: Routine unannounced inspection of preoperational test results review, power ascension test program, technical specification surveillance activities for operational condition 5, independent measurements and calculations, QA/QC interfaces, licensee action on previous inspection findings, and tours of the facility.

Results: No violations were identified. NOTE: For acronyms not defined refer to NUREG-0544 "Handbook of Acronyms and Initialisms."

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DETAILS

## 1.0 Persons Contacted

Public Service Electric and Gas Company (PSE&G)

M. Azzard, Operations Shift Supervisor
D. Benway, Startup Test Engineer
*J. Carter, Startup Manager
G. Chew, Power Ascension Technical Support
*J. Coven, Radiation Protection Manager
*R. Donges, Lead Quality Assurance (QA) Engineer
*M. Farshon, Power Ascension Manager
B. Forward, Power Ascension Administration
*A. Giardino, Manager Station QA
R. Griffith, Principal QA Engineer
J. Issacs, Setpoint Program Manager
C. Jaffee, Startup Engineer
P. Kudless, Maintenance Manager
*S. LaBruna, Assistant General Manager
J. Llelwellyn, I&C Technical Specification Coordinator
*C. McNeill, Vice President Nuclear
*M. Metcalf, Principal QA Engineer
*J. Molner, Senior Radiation Protection Supervision
J. Montgomery, Maintenance Technical Specification Coordinator
T. Moose, Chemistry Technical Specification Coordinator
*J. Nichols, Technical Manager
E. Parker, Reactor Operator
W. Ryder, Operations Technical Specification Coordinator
*R. Salvesen, General Manager
W. Schell, Power Ascension Technical Director
D. Shipman, Startup Test Engineer
*M. Sullivan, Site Engineer
K. Wilson, Preop Procedure Development Supervisor

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*D. Allsopp Resident Inspector
*R. Borchardt, Senior Resident Inspector
J. Lyash, Reactor Engineer
*R. Nimitz, Senior Radiation Specialist
*M. Shanbaky, Chief Facilities Radiation Protection Section

The inspector also contacted other members of the licensee's staff including senior nuclear shift supervisors, reactor operators, test engineers and members of the technical staff.

*Denotes those present at exit meeting on March 14, 1986.

## 2.0 Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (354/85-58-04) This item dealt with the incorporation of the as-built elevation data into the reactor vessel water level instrument calibration calculations and the subsequent re-calibration of the affected instruments. The inspector reviewed representative calculations for a narrow range instrument (LT-N080B-Calculation No. SC-BB-0184) and a wide range instrument (LT-N081B-Calculation No. SC-BB-0199-1) and verified that the as-built elevation data had been incorporated into the calibration calculations. The inspector also established that calibration calculations had been used to produce instrument calibration data sheets which were used to re-calibrate the affected instruments. This item is closed.

(Closed) Unresolved Item (354/86-03-02) This item dealt with having practical BWR reactor engineering experience in the reactor engineering staff. Inspection report 50-354/86-10 reviewed the licensee plans and considered them to be acceptable pending the actual hiring of the new reactor engineer. The inspector discussed this item with the Senior Reactor Engineer Supervisor and the new hire reactor engineer and ascertained that the individual hired has sufficient prior practical BWR reactor engineering experience and will be utilized within the group to direct and guide reactor engineering activities. This item is closed.

(Closed) Violation (354/85-65-01) The violation dealt with manual operation of a containment isolation valve (CIV) in preparation for a containment integrated leak rate test (CILRT) in January 1986. The inspector verified the four corrective steps taken to preclude reoccurrence as identified in licensee letter dated February 12, 1986, as described below.

The inspector verified that operations personnel were informed of the contents of the violation and corrective actions. A memo was sent to all operations personnel directing them to read PSE&G's response to the violation and the revised operating procedure OP-AP.ZZ-109(Q). The inspector reviewed 61 of 84 response memos from operations personnel indicating that they had completed the required reviews. The inspector also interviewed two operators and ascertained they were knowledgeable of this issue.

The inspector verified that operating procedure OP-AP.ZZ-109 (Q), "Equipment Operational Control", has been revised incorporating the requirement that manual torquing of motor operated valves is not permitted when performing ILRT lineups.

The inspector reviewed a procedure revision request dated January 18, 1986, initiated to incorporate the same requirement into the next revision of the Hope Creek Station Safety tagging procedure SA-AP.ZZ-015(Q).

The inspector obtained and reviewed a memo from ISI Engineer L. Lake to A. Giardino, QA Station Manager dated January 28, 1986. This memo indicated that the following precaution will be incorporated into inservice Type A Test Procedure M9-ILP-301: "Power actuated containment isolation valves shall not be manually operated. If these valves are manually operated a penalty must be taken and the leakrate in the amount measured during local leak rate testing shall be added to the results of the type A test."

The inspector questioned the licensee as to why a revision request has not been written. The licensee indicated that the inservice type A test procedure has not been written but the above statement will be included. The next CILRT for Hope Creek will take place approximately three years from the date of this report. The inclusion of the above into the inservice type A Test Procedure will be tracked as unresolved Item No. 354/86-18-01. The licensee also committed in their February 12, 1986, response, to using red blocking tags under the station safety tagging procedure for the ILRT valve lineup rather than the standard orange T-Mod tags. This should further bring to the attention of operations personnel that CIVs have been blocked for an ILRT and shouldn't be manually operated. This will not be implemented until the next CILRT and will also be included in unresolved Item No. 354/86-18-01.

Another document reviewed and verified by the inspector was on-the-spot change No. 28 to preoperational CILRT Test Procedure PTP-GP-1. This change provided a leakage penalty of 97 SCCM (measured leak rate through valve No. KL-HV-5148,V-001) to be added to the CILRT results. This penalty will account for any leakage through the valve which was masked by hand tightening. Based on the above this item is considered closed.

(Closed) Construction Deficiency Report (84-00-14) Bailey Controls Company Model 862 Digital Logic Modules. The licensee identified that the input logic isolators were affected by electrical noise from within their respective cables and associated wires. This noise caused malfunction of safety related control circuits. There were 2,248 logic modules, both safety and nonsafety, that required corrective action. Each of these had 8 inputs that were potentially affected. The electrical noise was able to affect these circuits because each individual control circuit was either not assigned its own cable or shielded conductors within a multiconductor cable. The licensee corrective action was to modify the Model 862 logic card to discriminate between valid signals and induced voltages.

During a NRC inspection (50-354/84-21) of the licensee corrective actions the inspector raised questions regarding electrical noise effects not specifically addressed by the licensee's corrective actions. These questions pertained to electrical noise from the following sources:

--Ships Radar
--Electrical welding
--Electrical faults or switching surges
--Large DC motors starting and stopping

--Portable radios (walkie-talkie)
--Changes in conductor capacitance due to cable water immersion

The NRC held a meeting with the licensee on February 15, 1985, (50-354/85-07) to discuss modifications to the logic modules to correct the design deficiencies. As a result of the meeting, the licensee was requested to provide the following additional information:

--A description and summary of the results from the verification tests to demonstrate immunity of the modified Bailey Control system against adverse EMI and RFI effects;

--A summary of final resolution of time delay and in-rush current considerations;

--Additional details regarding the administrative controls to be applied throughout plant life relative to potential RFI sources such as two way radios and inductive sources such as welding machines;

--Verification that other solid state components are not affected by the Hope Creek EMI environment.

The results of the EMI/RFI tests and a response to the other NRC concerns were sent to the NRC. The NRC review of these issues will be the subject of a supplemental safety evaluation report to be issued by NRR.

The licensee addressed the following EMI/RFI potential effects:

--Ships radar will not cause RFI within the plant because of the shielding provided by concrete reinforcement bars.

--Electric welding will not cause RFI because of administrative controls placed on welding.

--Electrical faults and switching surges will not be an RFI concern because the modified logic module is not affected by this RFI.

--Large DC motors have not affected the control system.

--Two way radios will not cause RFI because of administrative controls to control their use.

--Changes in conductor capacitance could occur, because of the exchange of gasses through the cable jacket, if the cable were immersed in a water filled conduit. However, the modification was designed to include additional capacitance effects from conductors other than the interfering pair and the disturbed pair. Further the maximum cable length used for the redesign was 2500 feet. The longest circuit length identified by the licensee was 1500 feet. Most cables would average 800 feet in length.

The inspector reviewed the Station Fire Protection Program SA-AP.ZZ-025 dated January 31, 1986 which addresses the prerequisites to electric arc welding for EMI/RFI. This is a cross reference on the Hot Work Authorization Form to the Station Organization and Operating Practices SA-AP.ZZ-002(Q) approved January 25, 1986. Section 5.16 provides guidance to prevent EMI/RFI in specified areas. The inspector did not identify any problems with the licensee's administrative controls. These areas are also the areas that must be maintained free of two way radios. The inspector verified that these areas are posted with instructions such that within these areas two way radio use is not permitted.

Region I of the NRC has no further questions regarding this issue. In consideration of the ongoing review of the Bailey Logic Modules by NRR, the Region I open item is closed.

## 3.0 Pre-Operational Test Results Evaluation Review

### 3.1 Scope

The completed test procedures discussed below were reviewed to verify that adequate testing was performed to satisfy regulatory guidance, licensee commitments and FSAR requirements and to assure that uniform criteria were being applied for evaluation of the technical and administrative adequacy of completed test results.

The inspector reviewed the test results and verified the licensee's evaluation of the test results by review of the "As-Run" copy of the test, procedure test changes, test exceptions, test deficiencies, acceptance criteria, performance verification, restoration of systems to normal after test, independent verification of critical steps/parameters/calculations and identification of test personnel conducting and evaluating the test and verified that test results had been approved.

### 3.2 Discussion

PTP-EG-1, Safety and Turbine Auxiliary Cooling, Revision 1, Results Approved February 21, 1986

The objective of this test was to demonstrate the proper operation of the Safety and Turbine Auxiliaries Cooling Systems. The inspector reviewed the test results and performed independent calculations to verify the pump head and flow (from venturi data) for pumps AP210, BP210, CP210 and DP210 had been correctly evaluated. In addition, independent measurements were made of TACS Isolation Valves (HV-2522 E and F) fast closure times from strip chart traces of valve performance.

In two instances, the inspector identified changes to the test that appeared to reduce the scope of the test and to result in the failure to test certain valve interlocks and bypasses. Following discussions with the test engineer and startup manager and the review of applicable design documents, the inspector was satisfied that the test changes were justified and that the valve interlocks and bypasses had been adequately tested.

The approved test results included nine open test exceptions. These open test exceptions have been included, for tracking purposes, in the SDR list in section 3.3.

DTP-SB-0005, Main Steam Line Pressure Low Response Time Test, Revision 0, Results approved March 3, 1986

This detailed test procedure was performed in support of PTP-SB-2, Response Time Testing. The purpose of PTP-SB-2 is to demonstrate the response times of the instrument loops and logics for the Reactor Protection system and the Nuclear Boiler and Main Steam Line Radiation Monitoring Subsystems.

DTP-SB-0005 determined the response time of the Nuclear Boiler system, from the occurrence of a low pressure condition in the Main Steam Lines to the de-energization of the MSIV pilot solenoid valves. The response time was determined in two segments:

1) The response time of the pressure transmitters, and

2) The response time of the Automatic Trip Units (ATUs) and logic to the de-energization of the MSIV pilot solenoids.

The inspector independently verified the response times by measurements made from the response time strip chart traces. The longest measured response time for the ATUs was 151 msec. The longest pressure transmitter response times were 80 msec for a slow pressure ramp and 13 msec for a fast ramp. The longest overall response times were 23 msec for the slow ramp and 164 msec for the fast ramp, well within the acceptance criterion of 1.0 sec.

The approved test results did not contain any open test exceptions.

DTP-SB-0013, Main Steam Line Flow Response Time Testing, Revision 0, Results approved March 3, 1986.

This detailed test procedure was performed in support of PTP-SB-2, Response Time Testing. DTP-SB-0013 determined the response time of the Nuclear Boiler System, from the occurrence of a high flow condition in the Main Steam Lines to the de-energization of the MSIV pilot solenoid valves. The response was determined in two segments:

1) The response time of the differential pressure transmitters and

2) The response time of the ATUs and logic to the de-energization of the MSIV pilot solenoids.

   The inspector independently verified the response times by measurements made from the response time strip chart traces. The longest transmitter response times were 25 msec for a slow ramp and 22.5 msec for a fast ramp.

   The longest measured response time for the ATUs was 153 msec. The longest overall response times were 173 msec for a slow ramp and 170.5 msec for a fast ramp, well within the acceptance criterion of 0.5 sec.

   The inspector found that the longest overall response time identified in the test results for a fast ramp was 166.5 msec, which did not agree with the determination made independently.

   During discussions with the test engineer it was determined that a transcription error had been made when the response time was transferred from the strip chart to the data sheets. The test engineer issued a test exception against PTP-SB-2 to document this error. Since the mistake did not affect the acceptability of the system, the inspector had no further concerns.

   The approved test results did not contain any open test exceptions.

3.3. Findings

No violations were identified. However, several open test exceptions require resolution by the licensee. The inspector routinely assigns an unresolved item number to the open test exceptions for tracking purposes. The following exceptions identified in previous NRC reports, along with those open test exceptions identified in the above review, are being consolidated into one unresolved item (354/86-18-02). Unresolved item 354/86-12-03 is closed.

| Procedure No. | Short Title | SDR No. |
|---|---|---|
| PTP-AN-2 | Demin. Wtr Storage & Transfer | AN-0039 |
| PTP-PK-1 | 125 VDC Class IE | PK-0117, 0119 and 0120 |
| PTP-PJ-1 | 250 VDC Class IE | PJ-0026, 0033 and 0129. |
| PTP-BC-1 | RHR | BC-915, 1042, 1043, 1143, 1144, 1146, 1147, 1148. RL-736, 738. |
| PTP-SV-1 | Remote Shutdown Panel | BB-1011 and 1019; BC-1046, 1080, 1141 and 1142; BD-411, 482 and 496; EG-562,577, 665 and 666; FC-17; GJ-129, 185 and 195; RL-942, 944 and 950; SV-36, 39, 43, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55 and 57; ZZ-996 |
| PTP-SM-2 | NSSS and PCIS Equipment Actuation | AB-0470, BB-1011 and 1019, BG-0367, GS-0459, HB-829, KL-259 RJ-129, SK-113, SM-0096, 0100, 0101, 0102, 0106, 0107, 0108 and 0109 |

| | | |
|---|---|---|
| PTP-GU-1 | FRVS | GU-528, 529, 558,574, 576, 572, 530, 575, 573, 577, 574, 568, 556 and 581 |
| PTP-EG-1 | STACS | EG-687, 709, 722, 724, 727, 781, 783 |

4. Power Ascension Test Program (PATP)

The inspector held several discussions with the Power Ascension Manager and members of his staff regarding overall PATP activities. The licensee is aggressively pursuing the technical review board (TRB) approach to assuring PATP procedures are technically adequate and consistent with licensee commitments. Revisions of PATP procedures reflecting the TRB review are beginning to be issued by the licensee. Future inspections will again review the procedures for implementing the PATP.

The licensee inquired regarding the need for NRC review prior to proceeding from one test condition into the next. The inspector indicated that unless a license condition was imposed, no specific NRC concurrence was required prior to proceeding into the next test condition. The licensee activities would be subjected to NRC inspection to assure that the licensee conducted the reviews in accordance with license commitments. The inspector also indicated that no testing activities should be delayed only to permit NRC inspector witnessing unless this was specified as a license condition.

The inspector and the licensee discussed means and methods for measuring 5% power limit.

The licensee is also continuing use of the plant simulator to train the PATP personnel. The licensee has a training session planned for new PATP personnel the week of March 17, 1986.

The licensee provided the inspector the recently approved copy of procedure TE-SU.ZZ-041 "Full Core Shutdown Margin Demonstration" Revision 1, Approved March 8, 1986. The inspector preliminarily reviewed this procedure. This procedure would be performed in Operational Condition 5. The inspector questioned the basis for conducting this test in Operational Condition 5. The licensee identified the special test exception of technical specification 3.10.3, Shutdown Margin Demonstrations as the basis. This procedure would have only two source range monitors (SRM) operable for the full core shutdown margin. Since this full core shutdown margin demonstration requires the core to attain criticality and technical specifications for normal plant criticality during startup, require three SRM operable detectors, the licensee agreed to revise the technical specification exceptions to require three operable SRM's for critical shutdown margin demonstrations. The licensee prepared a technical specification change request dated March 13, 1986 to resolve this issue.

5. Technical Specification Surveillances for Operational Conditions 5

Scope

The inspector held discussions with various work group technical specification coordinators to determine if the technical specification surveillances identified by the inspector for tracking to verify entry into operational condition 5 were completed. The inspector also reviewed the surveillance procedures listed in Appendix A to determine if they were consistent with the final draft of technical specifications dated February 14, 1986. This activity will continue over several inspections.

Discussion

The inspector determined based on the licensee records that three of the surveillances out of approximately 80 being tracked by the inspector were completed at the end of the inspection period. (IC-CC.BF-01, OP-ST-GR-001 and OP-ST.GR-002).

The technical specifications surveillance for the Standby Liquid Control (SLC) system procedures of Appendix A were found to contain inconsistencies with the final draft of technical specifications. The licensee was aware of all the deficiencies except the one identified below and had procedure revisions in the final review process. During inspector review of CH-TI-ZZ-012 the inspector noted that the procedure did not specifically note that if control rods are withdrawn in Mode 5 the SLC must be operable and as such the chemical analysis must be verified every 31 days. The licensee had a revision to this procedure in process and indicated that this will be added to the procedure.

The inspector noted that the acceptance criteria for the operations surveillance procedures were not clearly identified. Recent NRC inspections had identified similar concerns and licensee activities to correct this have been initiated. The inspector inquired about the short term solutions to assure that all acceptance criteria are satisfied. The licensee is utilizing the review process with a deliberate review objective to assure that acceptance criteria are satisfied. This was acceptable to the inspector.

Findings

No violations were identified.

6. Independent Measurements/Calculations

The inspector made independent measurements of the response times of several transmitters and ATUs using strip chart traces. The inspector also performed independent calculations of pump head and flow from recorded data and confirmed the licensee's results. The details of these evaluations are discussed in Section 3.2.

7. QA/QC Interfaces

The inspector reviewed the test results review comments provided by QA personnel during the review of test results discussed in section 3.2. The inspector verified that resolution of all comments was properly documented and that commitments made during the comments resolution were being implemented.

The inspector also reviewed 9 QA Surveillance reports performed by QA when the licensee was performing technical specification surveillance activities.

QA/QC involvement in the above activities was acceptable.

8. Tours of the Facility

The inspector made several tours of various areas of the facility to observe work in progress, housekeeping, cleanliness controls and the status of construction and pre-operational test activities.

No unacceptable conditions were noted.

9. Unresolved Items

Unresolved items are matters about which more information is required in order to determine whether they are acceptable, an item of noncompliance or a deviation. Unresolved items disclosed during the inspection are discussed in Section 3.

10. Exit Interview

At the conclusion of the site inspection on March 14, 1986, an exit meeting was conducted with the licensee's senior site representatives (denoted in paragraph 1). The findings were identified and discussed. At no time during the inspection did the inspector provide written inspection findings to the licensee. The licensee did not identify that any proprietary information was contained in the scope of this inspection.