U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-354/85-42 Docket 50-354 License CPPR-120

Licensee: Public Service Electric and Gas Company

Facility: Hope Creek Generating Station

Inspection at: Hancock's Bridge, New Jersey

Conducted: August 12 - September 23, 1985

Inspectors:

for A. R. Blough, Senior Resident Inspector — 10/17/85 Date

for S. K. Chaudhary, Senior Resident Inspector — 10/17/85 Date

for J. J. Lyash, Reactor Engineer — 10/17/85 Date

Approved:

J. Strosn der, Chief, Projects Section 1B — 10/17/85 Date

Summary:

August 12 - September 23, 1985 (Report No. 50-354/85-42): A routine onsite resident inspection (176 hours) of work in progress, preoperational testing, and new fuel receipt was conducted. The inspector also made tours of the site and reviewed licensee action on previous inspection findings, Construction Deficiencies and TMI Action Plan Items.

One violation was noted involving inadequate design control for DC control power alarm circuits (Detail 2).

DETAILS

1. Persons Contacted

Public Service Electric and Gas Company

*A. Barnabei, Principal QC Engineer
J. Carter, Startup Manager
G. C. Connor, Operation Manager
G. Daves, Senior Engineer, Operations
R. Donges, QA Engineer
*A. E. Giardino, Station QE Engineer
R. Griffith, Principal Staff QA Engineer
P. Kudless, Maintenance Manager
*S. LaBruna, Assistant General Manager
C. Lambert, Site Engineering
R. Lovell, Radiation Protection Manager
P. Landrieu, Project Manager
*M. Metcalf, Principal Startup QA Engineer
*J. A. Nichols, Technical Engineer
T. K. Ram, Site Engineering
J. M. Rucki, Maintenance Engineer
R. S. Salvesen, General Manager, Hope Creek Operations
C. Vondra, Operating Engineer

Bechtel

*W. Goebel, QA Engineer
*C. Jaffee, Startup Engineer
D. Long, Field Construction Manager
W. Mourer, Construction Manager
G. Moulton, QA Manager
K. Wilson, PTP Group Supervisor
*J. Zeruca, Startup Director

2. Previous Inspection Item Update

(Open) Unresolved Item (85-19-01), various containment isolation valve issues. All issues from the previous inspection remain open; this update expands the scope of this item. During this inspection the inspector found that FSAR Amendment 11 had deleted from containment isolation valve table 6.2-16 the third feedwater isolation valve, as well the isolation valves in RCIC, HPCI, and Reactor Water Cleanup lines injecting through the feedwater system. The inspector pointed out that the deletions are not only inconsistent with the FSAR text and the SER but also represent an additional departure from General Design Criterion 55. The applicant stated that an FSAR change notice was being prepared to reinstate the valves. The inspector also noted that the applicant plans to relax some of the valve closure times listed in FSAR Table 6.2-16 and to delete closure times on certain valves which receive no containment isolation

signal. The inspector cautioned the applicant that any such changes need to (1) have a sound basis relative to health and safety of both the public and plant workers, (2) be reviewed by NRC:NRR based on the Standard Review Plan, and (3) be accurately reflected in all documentation, including valve testing criteria.

(Closed) Unresolved Item (85-27-05), adequacy of alarm circuits on loss of 125 VDC control power to Class 1E 480 VAC Unit Substations (USS). The inspector reviewed logic diagrams and electrical schematics and discussed the design with PSE&G Site Engineering group.

Design logic diagram E-3132-0(Q) Revision 7 indicates that on loss of 125 VDC control power to each of the Class 1E 480 VAC USS a common main control room annunciator and specific computer alarm points will be actuated. Electrical Schematic Diagram E-0097-0(Q) Sheet 2, Revision 5, and the "as-built" configuration, show power for the annunciator and computer points supplied by the same 125 VDC source being lost. Consequently, on loss of 125 VDC control power, power to corresponding annunciator and computer points is also lost, so that no alarms are generated. The inspector informed the applicant that the above failure to properly translate system design features into installation drawings and a properly functioning system constitutes a violation of 10 CFR 50, Appendix B, Criterion III. Criterion III, Design Control, states in part that measures shall be established to assure that the design basis are correctly translated into specifications and drawings, and that design control measures provide for verifying or checking the adequacy of design (85-42-01).

The applicant's site engineering group agreed that the alarm circuit would not function as designed and installed. The licensee stated, however, that at least one Class 1E to non-class 1E isolation circuit breaker on each of the eight 480 VAC USS was equipped with a component level alarm which would be actuated on loss of 125 VDC control power to that breaker. The inspector pointed out that this component level alarm would not function to alert the operator to loss of control power to a particular USS if the single breaker in question were disabled. This condition can exist when non-class 1E equipment fed by the breakers in question is removed for maintenance. This component level alarm is therefore not redundant to the originally designed USS alarm. The licensee initiated Engineering Change Notice 1770, and Design Change Package 581 was issued to alter the power supply so that the alarm circuit in question is powered from an independent source.

Preoperational Test Procedure PTP-PG-1, Revision 0, Class 1E 480V Unit Substations, had contained acceptance criteria and steps to test the above described alarm. These steps and criterion were deleted from the PTP (See report 85-27). These test steps would not have worked as written, and the design problem might have been identified. Public Service Startup Group personnel stated that verbal concurrence had been received from site engineering prior to deletion of the test steps. Based on the information provided and issuance of the noncompliance described above, the original unresolved item (85-27-05) is closed.

(Open) Unresolved Item (85-27-01), excessive unsupported length of cable. In response to the inspectors questions regarding the adequacy of 40 inches of unsupported cable, the licensee cited Bechtel electrical calculation 30.0Q. This calculation demonstrates that a cable span of 60 inches would not exceed the Hope Creek cable physical characteristics. The response, however, did not address the effect of the weight of unsupported cable on the last rung of the cable tray, and the potential for overloading the rung.

In addition to the above, the inspector had questions regarding a raceway extension which had been added to a similar cable tray, and was not shown on corresponding drawings. Licensee investigation indicates that the addition of the raceway extension was unauthorized rework. Nonconformance report 8358 was initiated to track the problem. NCR 8358 was dispositioned to "use as is", and the installation card was reissued to show the rework. Subsequent to the disposition of NCR 8358, the inspector noted that no grounding strap between the main raceway and the extension was visible. This question was referred to the applicant for review. The applicant believes the unauthorized rework was performed prior to March 1982. In January 1984, the applicant opened Quality Assurance Report (QAR) 240 in response to an increasing number of cases of unauthorized rework. Actions taken as a result of QAR 240 included: 1) meetings with management stressing the importance of rework control, 2) procedure revisions to ensure clear definition of discipline interfaces, 3) review, investigation and corrective actions in response to all unauthorized rework NCR's. The rework identified by the inspector appears to have occurred prior to the vigorous corrective actions taken in response to QAR-240. Because this individual instance of unauthorized rework appears to be part of a programmatic problem previously identified and addressed by the applicant, no Notice of Violation is issued. This item remains open pending completion of licensee investigation/analysis of the acceptability of the unsupported cable length and of the raceway extension.

(Open) Inspection Follow Item (83-14-10), unauthorized rework. As discussed above (see item 85-27-01), the applicant addressed this item in QAR 240. Pursuit of each case of unauthorized rework was aggressive. The architect-engineer's (AE) QA trended unauthorized rework NCR's from February 1984 to June 1985. Initially, the rate of unauthorized rework appeared stable. Upon closer evaluation of the 1985 NCR's, the AE's QA found that actual unauthorized rework rate was low (zero to four cases per month), and classification of NCR's by QC was very conservative (about 20 NCR's per month classified as unauthorized rework). The inspector reviewed examples of NCRs which were classified as "unauthorized rework" but actually involved improper work (e.g., accidental cutting of rebar) or QC holdpoint violations. Since these items do not have potential for permanent avoidance of QC, they are not as significant as unauthorized rework. Most of the actual cases of unauthorized rework involve missing or broken torque paint. The inspector had no further generic questions regarding applicant response to the programmatic problem of unauthorized rework. However, one specific concern was identified relative to fire barrier penetration seal rework. Several cases of unauthorized seal

rework were identified during pulling of safety related cables in early 1985. Since no QC inspection is required for non-safety related cable pulls, the inspector questioned whether some cases of unauthorized fire barrier seal rework might have gone unnoticed. The applicant is evaluating this concern; therefore, this item remains open.

3. Follow-up on Events Occurring During the Inspection - Carbon Dioxide Injection

On September 4, the Diesel Fuel Oil Tank Bay 'A' carbon dioxide (CO_2) fire protection system spuriously actuated, overfilling the room with CO_2 while workers were doing cleanup in the adjacent hallway. The applicant, the architect engineer, and OSHA are performing investigations and evaluations of the industrial safety event. The inspectors reviewed this event and CO_2 system design relative to potential for nuclear safety impact of similar events during the operations phase. The following potential concerns were reviewed:

(1) Potential for overpressurization of rooms with resultant structural damage to the room, components within, and adjacent components. The applicant stated that his commitment to NFPA-12 requires sufficient room venting or overpressure protection to prevent exceeding 1.0 psi room pressure in event of a complete storage tank injection to any CO_2-served room. Preliminary calculations associated with injection tests in Spring 1985 indicated a peak postulated pressure of 0.6 psi in the current configuration. However, additional sealing of the room door frames is planned and calculations are needed to confirm adequate remaining vent area.

(2) Potential for deleterious effects due to dispersal of CO_2 through the plant, especially into the control room. The inspector reviewed ventilation system designs and flowpaths and did not identify any ventilation-related concerns. Additional inspector review is required regarding potential for leakage into the control room from the control equipment mezzanine directly below, in the event of excessive CO_2 injection into the mezzanine.

(3) Potential for common-mode failure of diesels due to injections associated with seismic events. The inspector determined that, even though not all the piping is seismically qualified, the control cabinets and the shutoff valve are qualified not to actuate falsely in seismic events. The inspector had no further questions on this item.

The following items will be reviewed after completion of applicant investigations and evaluations:

(1) Final calculations regarding room overpressure protection,

(2) CO_2 leakage potential from control equipment mezzanine to the control room, and

(3) Applicant determination of cause and corrective actions.

This is an inspector follow item (85-42-02).

4. TMI Action Plan (TAP) Items

The inspector reviewed the following TMI Action Plan items to verify that the applicant is meeting his commitments to NUREG-0737

(Closed) TAP Item I.A.1.3 Shift Manning. The inspector reviewed Draft Technical Specifications and procedure OP-AP-ZZ-002(Q), Revision 1, regarding (1) shift manning requirements, (2) overtime limits, and (3) overtime approval. These conform to NUREG-0737 and Generic Letter 82-12.

(Closed) TAP Item II.B.1, Reactor Coolant Vents. The SER accepted this item based on the fact that use of HPCI, RCIC and/or Safety/Relief Valves (SRV's) to control reactor parameters during accidents would assure adequate venting. The inspector verified that applicant emergency operating procedures address use of HPCI, RCIC and SRVs for appropriate accident situations.

(Closed) Relief and Safety Valve Testing Program. In SER, Supplement 2, the NRC:NRR staff accepted previously reviewed generic test data as being applicable to Hope Creek. Therefore, the inspector has no further questions on this matter.

(Open) Post-Accident Sampling System (PASS). SER Section 9.3.2 accepts the PASS design and states that PASS must be operable before 5 per cent power is exceeded. Before fuel load, a plant specific procedure for estimating core damage is required. The inspector reviewed procedure CH-TI.ZZ-011(Q) and verified that it provides for estimating core damage. The PASS system hardware and procedures will be inspected at a later date.

5. Construction Deficiency Reports (CDR's)

(Closed) CDR (85-00-01), fatigue cracking of fuel injection pump delivery valve holders. The applicant replaced all holders on all diesels with acceptable (redesigned) replacement. After reviewing the associated nonconformance report, start-up deviation reports and QC inspection reports, the inspector had no further questions.

6. Construction

6.1 Plant Tour and Walk Through Inspection

The inspector periodically toured the plant and performed walk-through inspection during this inspection period. In the walk-through inspection, special emphasis was placed in the areas of drywell, reactor building torus/wetwell, and diesel generator buildings. These inspections were carried-out to assess the level of general workmanship in the areas of piping and pipe support; effectiveness of cleanliness and housekeeping program; and general

conformance to project procedures in the work-in-progress and the completed work. During a tour of the control room on September 9, the inspector noted a fire barrier penetration seal partially removed. Further, the sides of the seal appeared to have a convex curvature such that they would not provide good sealing. The applicant investigated and determined that (1) the seal removal had been properly authorized and (2) the abnormal seal shape was apparently related to the efforts to remove the seal. After reviewing the seal rework authorization card and interviewing the cognizant QA engineer, the inspector had no further questions.

6.2 Spent Fuel Racks

The inspector reviewed design and procurement documents, held discussions with cognizant personnel, and visually examined the installed racks to assess their conformance to the design and regulatory requirements. The design of the racks was also examined and reviewed to ascertain the technical validity of the design parameters, and the procurement documents were reviewed to determine the adequacy of imposed quality requirements in design documents for fabrication materials, inspection, and shipping and handling of the racks. The inspector reviewed the following documents:

a. Bechtel Purchase Order: 10855-M-178-Q-AC.

   -- Miscellaneous Hardware Book #1

   -- Section III, Material traceability records

   -- Boral Summary Sheets for Drawing/Spec-BP10053QAP

   -- SS-CMTR #79-21263-4, dated 10/8/84 for Heat #A14999

   -- CMTR for Heat #1G-4124 (Total of 21 pages)

b. Spent Fuel Storage Racks - Module Fabrication Book-II, Serial #AD-37892-D-01.

   -- Section III - Base Plate Fabrication - 4 sheets

   -- Section IV - Can Fabrication 30 sheets

   -- Section V - Module Fabrication 16 sheets

   -- Section VI - Checkout Procedures - 23 sheets

c. Certification of Module - Book-I.

   -- Section I - Welders Qualification - 35 sheets

   -- Section III - Certified Material Test Reports (MTR) - 35 sheets

-- Section IV - Boral Test Reports -30 sheets, (MRR-H1001 through H0007)

-- Section V - Quality Control Procedures - 10 sheets

-- CMRTs for Boron Carbide Powder.

d. Final Module and Installation Tool check-out Procedure for Serial #AD-37891-D-01; Book II.

e. Bechtel Quality Control Inspection Reports.

-- QCIR-178-128295

Based on the review of the above documents, discussions with responsible engineers and QC personnel and the visual examination of the installed racks, the inspector determined that the spent fuel racks for Hope Creeek Generating Station were adequately designed and manufactured. The installation and checkout was properly carried out, and the racks as-installed were acceptable.

No violations were identified.

7. Preoperational Phase Activities

7.1 Plant Tour

The inspector toured the control room on regular and backshifts. He interviewed operations personnel regarding testing scheduled or in progress, reviewed logs and night orders, and observed alignment and indications of systems undergoing tests. Operators and supervisors were knowledgeable regarding plant status and test plans. The inspector toured areas of the plant, including drywell, reactor building, and the control building. He checked on tests and operations in progress, observed equipment and housekeeping conditions, and interviewed personnel involved in ongoing activities.

7.2 Preoperational Test Procedure (PTP) Verification.

The following procedures were reviewed to verify proper (1) administrative review and approval, (2) proper format, and (3) general agreement with Section 14.2 of the FSAR. The inspector did not perform a detailed technical review. (These procedures are classified in the NRC inpsection program as neither "mandatory" nor "primal" tests.)

-- AC-1, Revision 0, Main Turbine;

-- GC-1, Revision 1, Service Area HVAC;

-- AF-1, Revision 0, Extraction Steam;

-- GE-1, Revision 0, Turbine Building HVAC;

-- KF-1, Revision 0, Cranes & Hoists;

-- AE-2, Revision 0, Feedwater System;

-- ED-1, Revision 1, Reactor Auxiliarie Cooling System; and

-- [illegible]-1, Revision 0, Aux Bldg. Control Area Chilled Water System.

The inspector noted on September 17 that procedure AE-2, Feedwater System, had been classified as non-Q, even though the procedure included stroke time testing of containment isolation valves. The applicant apparently had previously identified the problem, and a revision which upgraded the procedure to 'Q' classification was being prepared.

The inspector noted that, for HVAC preops, the air balancing is done as a General Test Procedure (GTP) rather than as an integral portion of the PTP. The GTP is referenced in section 9 of the associated PTP. The inspector asked if the completed GTP results would be reviewed along with the PTP. The applicant pointed out a recent change to Startup Administrative Procedure 24 that requires any procedure which is referenced in Section 9 of the PTP to be attached for results review.

The inspector noted that PTP KF-1, Cranes and Hoists, requires load testing of all monorails and hoists in FSAR Table 9.1-10. FSAR Table 9.1-10 lists three monorail locations without hoists installed. The intent is to borrow a hoist from another location as needed. The inspector asked how the preoperational testing of these monorails would be done. In attempting to answer this question, the applicant found that no existing hoist was compatible with the vacant locations. The applicant wrote a test exception and a Startup Deviation Report (SDR) to resolve the compatibility problem and track the monorail testing. After reviewing these documents, the inspector had no further questions.

7.3 Preoperational Test Procedure (PTP) Review

The inspector reviewed the following preoperational test procedures for verification that planned testing fully demonstrates that system operation and response meets regulatory requirements and applicant commitments. The review also verfied proper format and content, and applicant review and approval of the subject procedures.

-- PTP SM-1, Revision 0, NSSSS and Primary Containment Isolation System Logic Test

The inspector reviewed the test for administrative and technical adequacy and for conformance with FSAR commitments. The inspector

also discussed the test with start-up engineers, reviewed logic diagrams, and reviewed the applicant's internal procedure review comments. No inadequacies were noted. The inspector noted that portions of FSAR commitments for the Primary Containment Isolation System (PCIS) are met in other tests: SM-2 for manual isolation, BB-3 for LOCA Level 1 and High Drywell Pressure signal processing, and numerous other tests for valve stroke timing. Most of these tests have not yet been approved. NRC review of the additional tests is therefore needed in order to verify overal adequacy of PCIS testing.

-- BJ-1, Revision 0, High Pressure Coolant Injection

The inspector reviewed the test for compliance with administrative review and approval requirements, General Electric preoperational test specifications, Bechtel design documents, FSAR commitments and NRC Regulatory Guides. Individual test sections were reviewed in detail to determine the technical adequacy of the procedure.

No unacceptable conditions were identified.

7.4 Preoperational Test Witnessing

The inspector witnessed testing in progress on regular and backshifts and verified that: 1) testing was conducted using the latest revision of the approved procedure by qualified individuals, 2) controlled and calibrated measuring and test equipment was available for required data gathering, 3) adequate quality control coverage was provided, 4) proper coordination between test engineers and operations existed, and 5) test exceptions and changes were documented and dispositioned properly.

During the report period, the inspector witnessed several sections of the following preoperational tests:

1. BE-1, Revision 0, Core Spray System

2. BJ-1, Revision 0, High Pressure Coolant Injection

3. BD-1, Revision 0, Reactor Core Isolation Cooling

Sections of the above tests observed include interlock and logic testing, valve functional checks and stroke timing. The inspector also witnessed flow testing of the Core Spray System including the Core Spray Pattern Test. Full Quality Control coverage was provided during test sections witnessed by the inspectors. Testing was conducted in accordance with program requirements and the criteria listed above. Test Engineers were well informed and test personnel were adequately briefed prior to test start.

No unacceptable conditions were identified.

8. New Fuel Receipt

During August 26-30. 1985, the resident inspectors particpated in a fuel receipt readiness inspection--these activities are documented in inspection 50-354/85-40. New fuel receipt began September 3. The inspectors periodically observed new fuel receipt activities including handling, inspection, storage, and security.

No unacceptable conditions were identified.

9. General Employee Training/Radiation Worker Training

On September 9, the inspector observed the "short-course" General Employee Training and Radiation Worker Training to evaluate adequacy of the training and visual aids and to satisfy site access requirements. The inspector also reviewed the course quizzes and answer keys and interviewed the instructor. No inadequacies were noted.

10. Inspection Program Status

Preoperational Test Program Inspection completion status is approximately as follows:

Area	% Inspection Complete
Overall Program	60
Procedure Reviews -	
Mandatory	50
Primal	100
Test Witness -	
Mandatory	35
Primal	35
Results Review -	
Mandatory	20
Primal	10

Inspection status is consistent with applicant test program progress.

Operational readiness inspection status is approximately as follows:

Area	% Inspection Complete
OPS-Staffing & Procedures	10
Tech Spec Review	10
QA	75
Maintenance	25
Fire Protection	25
Fuel Receipt	90
Surveillance	25

Rad. Controls	30
Rad. Waste	15
Security	15
Emerg. Planning	40

Additional inspection will be done in each area to verify readiness for fuel load.

Open Item inspection approximate status is listed below. Items are considered "backlogged" if the applicant has presented information for closure but the inspector has not begun his review.

Area	% Closed	% Working (NRC)	% Backlogged	#Items Now Open
Inspection Findings*	43	3	5	59
TMI Items				
-for OL	22	12	6	28
-for 100%	21	0	0	11
Bulletins*	17	0	25	44
Circulars	19	0	6	29
CDRs*	50	12	0	16
SER Verfications*	55	0	0	5

*All data subject to change as new items are opened. Start date for this accounting system was April 15, 1985 -- Items closed before then are not reflected in percentages.

The inspector considers the backlogs as minor; NRC inspection status is consistent with applicant progress in closing open items.

11. Exit Interview

The inspectors met with applicant and contractor personnel periodically and at the end of the inspection report to summarize the scope and findings of their inspection activities. Written material was not provided to the applicant.

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