U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-354/85-06

Docket No. 50-354

License No. CPPR-120 Priority -- Category B

Licensee: Public Service Electric and Gas Company

P.O. Box 236

Hancock's Bridge, New Jersey 08038

Facility Name: Hope Creek Generating Station, Unit 1

Inspection At: Hancock's Bridge, New Jersey

Inspection Conducted: February 12-15, 1985

Inspectors: L E Briggs 3/20/85
L. Briggs, Lead Reactor Engineer date

M Gaudino 3/14/85
M. Gaudino, Reactor Engineer date

L H Bettenhausen for 4/8/85
A. Alba, Reactor Engineer date

J. Flores, IAEA Oberserver

Approved by: L H Bettenhausen 4/8/85
L. Bettenhausen, Chief, date
Operations Branch, DRS

Inspection Summary: Inspection on February 12-15, 1985 (Report No. 50-354/85-06)

Areas Inspected: Routine, unannounced inspection (75 hours) of preoperational test procedure review and verification, observation of system flushes, as-built system comparison, review of hanger/support installation, QA/QC interface with the preoperational test program and plant tours by three region-based inspectors.

Results: No violations were identified.

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DETAILS

1.0 Persons Contacted

*S. Casnocha, Senior Designer
G. Champion, QA Engineer
*J. Cicconi, Startup Manager
*R. Donges, QA (Bechtel)
*M. Drucker, QA (Bechtel)
*A. Giardino, Manager QA, Engineering and Construction
*W. Goebel, QA Engineer (Bechtel)
*C. Jaffee, Startup Engineer
J. Jamieson, Startup Engineer Electrical
*H. Jolly, Supervising Engineer
*C. Lambert, Site Engineering
*E. Logan, General Manager Construction/Site Manager
*M. Metcalf, Principal QA Startup Engineer
*W. Moberly, Construction Completion Engineer
C. Moore, Startup Engineer (RHR)
*B. Preston, Licensing Manager
*R. Salvesen, General Manager, Hope Creek Operations
*G. Tenen Baum, Supervising Engineer
*M. Woloski, Lead Engineer

Other NRC Personnel Present

*R. Blough, Senior Resident Inspector
*S. Chaudhary, Senior Resident Inspector, Limerick Unit 2
*J. Flores, Comision Nacional De Seguridad Nuclear y Salvaguardis (Mexican NRC)

The inspector also contacted other QA/QC and technical members of the licensee's startup and construction organizations.

*Denotes those present at the exit meeting conducted on February 15, 1985.

2.0 Preoperational Test Procedure Review

2.1 PTP Review and Verification

The four PTP's listed below were reviewed in preparation for test witnessing, for technical and administrative adequacy and for verification that testing is planned to adequately satisfy regulatory guidance and licensee commitments. They were also reviewed to verify licensee review and approval, proper format, test objectives, prerequisites, initial conditions, test data recording requirements and system return to normal.

- KJ-4, Emergency Diesel Generator, Revision 0, approved on December 21, 1984;

- JE-1, Diesel Fuel Oil Storage and Transfer, Revision 0, approved on July 16, 1984;

- ED-1, Reactor Auxiliary Cooling System, Revision 0, approved on March 20, 1984; and

- SC-1, Loose Parts Monitor, Revision 0, approved on November 9, 1984.

2.2 Findings

A detailed review of the above procedures identified several problems (listed below) that were discussed with licensee.

During review of KJ-4 the inspector noted that to satisfy R.G. 1.108 revision 1, position C.2.a.(4), load rejection of the largest single load, the licensee would open the diesel generator output breaker while loaded to 1000 KW. The largest single load is an RHR pump which is 991 KW. The inspector questioned the licensee about the type of instrumentation and recorders that would be used for this test and their accuracy to ensure that at least 991 KW would be rejected, considering total instrument error. The inspector also noted that special recorders to be used were noted under Section 7, Special Precautions and Notes vice Section 6, Special Test Equipment. Section 7 did not specify required instrument accuracy. The licensee agreed to include required recorders and their accuracy in Section 6 of the KJ procedure series.

During further in-office review of Preoperational Test Procedure KJ-4, the inspector noted that performance of the largest single load reject test as written did not appear to meet the intent of RG 1.108. Although RG 1.108 does not specify the method of conducting the largest single load shedding test, it does by the use of the word "shedding" vice "reject" imply other loads are maintained on the DG. A load shed would simulate securing the RHR pump when not required after an upset condition. The licensee, during a telephone conversation on February 21, 1985, agreed to reevaluate their method of testing. This concern was discussed with NRR reviewers by the inspector. This item is unresolved pending resolution between NRC:RI, NRR and the licensee (354/85-06-01).

The inspector also noted that, during the 24 hour full load run of the diesel, auxiliary data (oil pressure, bearing temperatures, etc.) were required to be taken only one time, at the end of the 24 hour run. The licensee stated that such was not the intent and that the procedure would be revised to clearly require specified time intervals between readings for auxiliary data.

One additional concern identified related to design accident load sequencing of the diesel generator (DG) to verify its ability to maintain frequency and voltage within required limits (position C.2.a.(5) of RG 1.108) immediately following the 24 hour full load test. As presently written, the test procedure starts the diesel with a loss of power signal then synchronizes the DG with the bus and backfeeds to the grid. The inspector informed the licensee that this method of testing did not subject the DG to large starting currents as intended by RG 1.108 and would not test the DG's ability to maintain voltage and frequency within limits for a start under plant upset conditions. The inspector also noted that an exception to this phase of testing was not stated in the FSAR Section 1.8.1.108. Further discussion with the licensee indicated that NRR acceptance of this method of testing had tentatively been obtained through FSAR question 640.10 (Section 14.2.12). A review of question 640.10 and the licensee's response indicated that manual loading was the stated means of testing.

Subsequent discussions between the NRR reviewer and the inspector indicate that the reviewer interpreted manual loading as manually sequencing the emergency loads onto the diesel, not backfeeding the grid. This is part of an item pending resolution between NRC:RI, NRR and the licensee (354/85-06-01) regarding DG testing in accordance with RG 1.108.

During review of SC-1 (Non Q) the inspector noted that the loose parts monitor was being tested to verify that an alarm would be actuated if the pipe was struck with a force of 0.5 ft. lbs. or less three feet from the sensor. RG 1.133, Revision 1, Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors, recommends that alarms actuate at an energy level of 0.5 ft. lbs. or greater. Although the setpoints of SC-1 are more conservative than RG 1.133 it would result in alarms due to normal system noises. This was discussed with the licensee who agreed to revise Procedure SC-1 to provide a limit of sensitivity to more appropriately reflect the guidance of RG 1.133.

This item will be reviewed during a future routine inspection.

Another problem was identified during the review of PTP-ED-1, Rev. 0, Reactor Auxiliary Cooling System (RACS). The test procedure does not require RACS to automatically supply cooling water to the Emergency instrument air compressor heat exchangers upon loss-of-offsite power (LOP) without occurrence of a loss-of-coolant accident (LOCA) (Section 5.4). The PTP does include verification of RACS flow through the heat exchangers during normal operation (Section 8.3.15), however, this did not appear to fulfill all of the requirements of HCGS FSAR Section 9.2.8.2.

After discussions with a licensee representative, it was determined that RACS supplies cooling water manually, not automatically, to the heat exchangers upon LOP without LOCA. The representative agreed to make the appropriate changes to the FSAR to reflect the actual system configuration. The inspector will review this item in a subsequent inspection.

3.0 System Flushes

During this inspection several system flushes were in progress. The inspectors toured the various areas of activity and discussed the different lineups being used with the senior shift supervisor (SSS) and the system startup engineer. Flushes in progress were:

- Integrated system flush (main steam lines and main feedlines);
- Core Spray System;
- Reactor Core Isolation Cooling System; and,
- Residual Heat Removal (RHR).

While observing the RHR flush, the technician assigned to take RHR pump data stated that pump discharge pressure had recently increased from about 175 psi to 325 psi. He did not know the reason but had noted piping noise that indicated the discharge valve had been closed or that some valve lineups had been changed. The on duty SSS during a subsequent discussion was also not aware that the RHR flushing lineup had been changed. The system startup engineer had modified the lineup to flush the min-flow line. The valve lineup change had been properly conducted by a control room operator, but the SSS had not been informed. Corrective action was taken by the SSS immediately.

The inspector also noted that spring hangers on the RHR system still had their keeper plates (preventing hanger movement) installed. When questioned the system startup engineer (SSE) stated that Bechtel hanger engineers had examined the system and provided a written statement that the system was adequately supported for flushing. The written statement was provided the inspector. The hanger adequacy statement also imposed a 120°F temperature restriction on the system. Bechtel piping specification P410 also allows spring hanger keepers to be installed if system temperature remains below 120°F. Further discussion with the SSE indicated that system temperatures are taken but not at regular intervals or at predesignated locations and they are not recorded. The inspector stated a concern that system temperatures in excess of 120°F could impose a system quality problem. This item is being followed by the Senior Resident Inspector and will be documented in NRC:RI Inspection Report 50-354/85-05.

4.0 As-Built System Comparison

4.1 Scope

The "C" line of the main steam system outside the drywell, the main steam bypass system and the "A" Recirculation Loop were inspected by direct observation to determine that the physical installations were in agreement with the corresponding Piping and Instrumentation Diagrams (P&ID's) and Isometric Drawings (ISO's). The system P&ID's and ISO's were used to trace the system to assure that the equipment, supports, hangers and snubbers used in the system were installed in accordance with the applicable documents.

The following references were used in the walkdown:

- Hope Creek Generating Station (HCGS), Final Safety Analysis Report (FSAR), Chapter 5
- HCGS FSAR, Chapter 10
- 1-P-AB-01, Rev. 15, System Isometric/Turbine Building Main Steam Lead
- 1-P-AB-02, Rev. 12, System Isometric/Turbine Building Main Steam Bypass
- M-01-1, Rev. 10, P&ID, Main Steam Supply System
- M-43-1, Rev. 9, P&ID, Reactor Recirculation System

4.2 Findings

The components, pipespools, reducers, hangers, snubbers and other miscellaneous equipment were readily identiifed and easily tracable to the applicable P&ID's and/or ISO's.

5.0 Inspection of the Hangers, Snubbers and Supports required for Hydro Test and Flushing of the Main Steam and Recirculation System

5.1 Scope

Hangers which were required for hydro test and flushing of the main steam and recirculation systems were chosen from General Electric Company's (GE's) required hanger list. The status of the selected hangers was determined by Bechtel Power Corporation Design Documenting Register. The hangers selected were examined to verify that the hangers were constructed according to the corresponding hanger drawing. Dimension, stability, identification numbers, thread engagement and fasteners, such as, lock nuts and interlocks were examined to ensure compliance with design drawings and specifications. The documents listed in Enclosure A were used for this inspection.

5.2 Findings

The hangers which were required for supporting the main steam and recirculation system during flushing and hydrotesting of both systems were installed according to design drawings and specifications. However, the inspector noted hangers not required for structural support by GE for hydrotesting and flushing of these systems which were incomplete (e.g., a missing bolt or nut). The hangers with the missing parts were verified as still under construction or not yet released for quality control inspection.

6.0 QA/QC Interface with Preoperational Test Program

The inspector reviewed recent QA surveillance reports (QASR) regarding different activities of the licensee's startup group. The following QASR's were reviewed:

- QASR-1563, RHR Flush of PSV-F025D and PSV-F025B, completed February 4, 1985. The QA inspector witnessed the two flushes identified in General Test Instruction (GTI)-01M-BC01. The observed flush cloths were clean and the sections flushed were found acceptable.

- QASR-1572, Motor operated valve functional checkout of 1BD-HV-F012 (Reactor Core Isolation Cooling), completed on February 4, 1985. The QA inspector noted that valve stroke times were 4.98 seconds to close and 4.72 seconds to open. Specification for valve timing was 15 seconds to open and 15 seconds to close. This discrepancy was noted in test package BDE-0014.

- QASR-1573, Cardox Concentration test of diesel generator rooms A through D completed on February 3, 1985. This test was conducted by the Chemetron Vendor Representative and the licensee. Several minor problems were corrected during the test and satisfactory results were obtained on diesel generator rooms A, B and D. A deficiency in the insulation around an HVAC duct in the "C" DG room prevented a satisfactory test. The "C" DG room will be tested at a later date when the fuel oil storage rooms are tested.

No unacceptable conditions were identified.

7.0 Plant Tours

The inspector made tours of the facility to become familiar with plant design and location of various structures and components. The inspector observed work in progress, housekeeping, cleanliness control and status of construction activities. The flushing activities discussed in Paragraph 3 of this report were observed.

8.0 Exit Interview

A management meeting was held at the conclusion of the inspection on February 15, 1985 to discuss the inspection scope and findings as detailed in this report (see Paragraph 1 for attendees). No written information was provided to the licensee at any time during the inspection.

ENCLOSURE A

Drawings Used for Hanger/Support Verification

1. FSK-P-169, Rev. 0, Recirculation Piping Loop "A"
2. FSK-P-215, Rev. 0, Main Steam Inside Drywell Lines C and D
3. 1-P-BB-012-H01, Rev. 4, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
4. 1-P-BB-012-H04, Rev. 3, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
5. 1-P-BB-012-H05, Rev. 3, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
6. 1-P-BB-012-H06, Rev. 7, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
7. 1-P-BB-012--H12, Rev. 3, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
8. 1-P-BB-012-H13, Rev. 4, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
9. 1-P-BB-012-H14, Rev. 3, Pipe Support, Reactor Building Recirc Loop "A"/ Inside Drywell
10. 1-P-BB-013-H03, Rev. 5, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
11. 1-P-BB-013-H05, Rev. 5, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
12. 1-P-BB-013-H10, Rev. 3, Pipe Support, Reactor Building, Recirc Loop "A"/ Inside Drywell
13. 1-P-AB-033-H01, Rev. 5, Pipe Support, Reactor Building, Main Steam Line "D"/ Inside Drywell
14. 1-P-AB-033-H06, Rev. 8, Pipe Support, Reactor Building, Main Steam Line "D"/ Inside Drywell
15. 1-P-AB-032-H01, Rev. 4, Pipe Support, Reactor Building, Main Steam Line "C"/ Inside Drywell

16. 1-P-AB-032-H06, Rev. 7, Pipe Support, Reactor Building, Main Steam Line "C"/ Inside Drywell

17. 1-P-AB-032-H11, Rev. 4, Pipe Support, Reactor Building, Main Steam Line "C"/ Inside Drywell

18. Specification 10855-P-410(Q), Technical Specification for Installation, Inspection, and Documentation of Pipe Supports in Nuclear Service for HCGS