U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-354/86-32

Docket No. 50-354

License No. NPF-50

Licensee: Public Service Electric and Gas Company

80 Park Plaza

Newark, New Jersey 07101

Facility Name: Hope Creek Generating Station, Unit 1

Inspection At: Hancocks Bridge, New Jersey

Inspection Conducted: June 23 - July 3, 1986

Inspectors: D. Florek — 7/21/86
D. Florek, Lead Reactor Engineer — date

D. Florek for — 7/21/86
L. Wink, Reactor Engineer — date

Approved by: P. Eselgroth — 7-21-86
P. Eselgroth, Chief, Test Programs Section, OB, DRS — date

Inspection Summary: Inspection on June 23 - July 3, 1986 (Inspection Report No. 50-354/86-32)

Areas Inspected: Routine, unannounced inspection of the power ascension test program covering the initial criticality including test procedure review, test witnessing, test results evaluation, reactor water level indications, independent measurements, calculations and verifications and QA/QC interfaces.

Results: One violation was identified for failure to follow startup test procedures. (see Section 3.3)

NOTE: For acronyms not defined refer to NUREG 0544 "Handbook of Acronyms and Initialisms"

DETAILS

1.0 Persons Contacted

Public Service Electric and Gas Company (PSE&G)

*R. Beckwith, Station Licensing Engineer
G. Chew, Power Ascension Results Coordinator
*G. Connor, Operations Manager
P. Dempsey, Shift Test Coordinator
M. Dick, Startup Test Engineer
*R. Donges, Licensing Engineer
*M. Farshon, Power Ascension Manager
S. Funsten, Instrument and Controls (I&C) Supervisor
*A. Giardino, Manager Station QA
J. Ghu, Startup Test Engineer
*R. Griffith, Principal QA Engineer
J. Head, Reactor Engineer
T. Hersum, Power Ascension Engineer
D. Hosmer, Lead Shift Test Coordinator
R. Hovey, Senior Nuclear Shift Supervisor
*C. Johnson, General Manager Nuclear QA
*P. Krishna, Assistant to General Manager - HCO
*P. Kudless, Planning Manager
*S. LaBruna, Assistant General Manager Hope Creek Operations
M. Lafatta, I&C Supervisor
*M. Metcalf, Principal QA Engineer
D. Moon, Shift Test Coordinator
L. Neuman, Senior Nuclear Shift Supervisor
W. Ott, Startup Test Engineer
E. Rush, I&C Surveillance Coordinator
*R. Salvesen, General Manager, Hope Creek Operations
W. Schell, Power Ascension Technical Director
R. Schmidt, Senior Reactor Supervisor
W. Thomas, Shift Test Coordinator
M. Trum, Senior Operations Engineer
R. Van Alstine, Startup I&C Engineer
L. Zull, Lead Startup Test, Design and Analysis Engineer

U.S. Nuclear Regulatory Commission

*D. Allsopp, Resident Inspector
*R. Borchardt, Senior Resident Inspector
*R. Summers, Project Engineer

The inspector also contacted other members of the licensee's staff including Senior Nuclear Shift Supervisors, Reactor Operators, Test Engineers, Technicians and members of the technical staff.

2.0 Surveillance Test Activities

Scope

The inspector examined the surveillance procedures listed in Attachment A to determine whether the licensee was meeting the technical specification requirements for entry into Operational Condition 2. The examination consisted of reviews of approved surveillance procedures for technical adequacy, reviews of completed surveillance tests to determine whether technical specification requirements were met and witnessing of the performance of surveillance tests. In addition the inspector reviewed the licensee's surveillance logs to determine the overall status of the surveillance program.

Discussion

The inspector determined that the approved surveillance procedures listed in Attachment A were technically adequate. The review of completed surveillance results indicated that technical specification requirements were being satisfied. The field performance of the surveillance tests witnessed conformed to all procedural and administrative requirements.

All technical specification surveillance requirements for entry into Operational Condition 2 were satisfied on June 28, 1986 and the plant entered Operational Condition 2 at 10:38 a.m. in preparation for initial criticality.

Findings

No unacceptable conditions were identified.

3.0 Power Ascension Test Program (PATP)

3.1 References

- Regulatory Guide 1.68, Revision 2, August 1978, "Initial Test Programs for Water-Cooled Nuclear Power Plants"
- ANSI N18.1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"
- Hope Creek Generating Station (HCGS) Technical Specifications, Revision 0, April 11, 1986
- HCGS Final Safety Analysis Report (FSAR), Chapter 14, "Initial Test Program"

- HCGS Safety Evaluation Report (SER), Chapter 14, "Initial Test Program"

- Station Administrative Procedure, SA-AP.ZZ-036, Revision 3, "Phase III Startup Test Program"

- Specification NEBO 23A4137, Revision 0, "Hope Creek Startup Test Specification"

- HCGS Power Ascension Test Matrix, Revision 7

3.2 Power Ascension Test Procedure Review

Scope

The power ascension test procedures listed in Attachment B were reviewed for conformance to the requirements and guidelines of the references listed above and the attributes previously defined in Inspection Report No. 50-354/86-03.

Discussion

- TE-SU.ZZ-041, TE-SU.SE-062 and TE-SU.SE-101 are revisions to previously reviewed procedures governing the initial criticality testing sequence. The procedures were revised to accommodate the performance of initial criticality in Operational Condition 2. In addition the acceptance criterion for SRM/IRM overlap in TE-SU.SE-101 was made more stringent.

- TE-SU.CH-272, verifies that, within the capacity of the main steam bypass valves, a main turbine trip will not result in a reactor scram.

- TE-SU.BB-191, verifies that core thermal limits are not exceeded during any operational condition. This procedure employs normal station reactor engineering surveillance procedures to calculate core thermal power and core flow and verify thermal limits adherence. The inspector reviewed the following reactor engineering procedures to assess their adequacy:

  - RE-RA.ZZ-001, Core Thermal Power Evaluation, Rev. 0

  - RE-ST.ZZ-001, Core Thermal Limits Evaluation - Process Computer Method, Rev. 0

  - RE-ST.ZZ-002, Core Thermal Limits Evaluation - BUCLE Method, Rev. 0

Findings

No unacceptable conditions were identified.

3.3 Initial Criticality Witnessing

Scope

The inspector witnessed portions of the initial criticality testing sequence to assess that:

- Licensee was complying with the technical specification requirements for initial criticality
- SRM/IRM instruments were properly calibrated and had appropriate outputs and response
- RPS shorting links were removed
- Adequate staffing existed
- Current revisions of procedures were in use and being followed.
- Procedure prerequisites were satisfied.
- Required data was obtained.
- Acceptance criteria were satisfied.

Discussion

During the current and two previous inspections (50-354/86-27 and 50-354/86-31) the inspector monitored the licensee's surveillance program to verify compliance with technical specification requirements for entry into Operational Condition 2. On June 28, 1986 the inspector attended the daily plant management meeting and the shift turnover meetings for the day and swing shifts to witness management review of activities and to assess overall readiness for initial criticality. The inspector also witnessed two shift test briefings by power ascension test engineers covering the planned initial criticality testing sequence. All preparatory activities were adequately accomplished.

Prior to the commencement of test activities the inspector reviewed the Reactor Engineering Data Book and the Control Rod Pull Sheets and performed an independent calculation of the expected critical position and the ±1% reactivity anomaly limits. The inspector verified that official working copies of the following procedures were available in the control area:

- OP-TE-IO-003, Startup from Cold Shutdown to Initial Criticality
- TE-SU.ZZ-041, Full Core Shutdown Margin Demonstration
- TE-SU.SE-062, Source Range Monitor Response to Rod Withdrawal
- TE-SU.SE-064, SRM Non-Saturation Demonstration
- TE-SU.SE-101, Source Range Monitor/Intermediate Range Monitor Overlap Verification
- RE-ST.ZZ-007, Shutdown Margin Demonstration
- RE-RA.ZZ-005, Reactor Period Measurement

By independent observations and review of records the inspector verified procedural prerequisites such as removal of the RPS shorting links, lowering of the SRM rod block and scram setpoints by one decade, SRM minimum counts and signal-to-noise ratio and loading of the Rod Worth Minimizer (RWM) sequence. All prerequisites were satisfied.

During the course of the initial criticality sequence the inspector also monitored control room staffing levels and the conduct of the operations shift personnel. Control room manning was more than adequate to support testing and satisfy technical specification requirements. The senior nuclear shift supervisors were very effective in maintaining the proper control room atmosphere including positive control of the numbers and conduct of non-participating observers.

At 1038 on June 28, 1986 the Mode Switch was placed in "Startup" and the unit entered Operational Condition 2. The RWM was verified operational but the Rod Sequence Control System (RSCS) failed its operational check when a required alarm (rod drift) failed to be received. The initial criticality testing sequence was officially placed in a "Hold" condition. Trouble shooting located the problem in a failed card. The decision was made to replace the card with a slightly different (older) version, a safety evaluation was performed and a temporary modification was approved by the Station Operations Review Committee (SORC). The card was replaced and successfully retested. The test "Hold" was lifted.

At 1647 the reactor startup commenced. At 1723 while withdrawing the 15th control rod (18-39) a Rod Position Information System (RPIS) "data fault" was received when the rod settled at notch position 48 (full out). Under reactor engineering guidance the rod was exercised several times by moving it between notch position 46

and 48 but the "data fault" remained when the rod was at position 48. The senior nuclear shift supervisor entered the appropriate action statement for technical specification 3.1.3.7 and a substitute position was entered for the rod and it was bypassed in RSCS. Rod pulls were then resumed.

At 1845 all Group I rods (a total of 24 control rods) had been withdrawn to the full out position. The inspector made a spot check of the performance of TE-SU.SE-062 which was being conducted at the SRM drawers at the back panels of the control room. This procedure was being conducted in parallel with the primary test, TE-SU.ZZ-041, with communications maintained between the main control area and the back panels by means of sound powered phones. The inspector observed that the test personnel for TE-SU.SE-062 had only obtained data for 23 control rods. When questioned on this point they expressed the belief that only 23 control rods had been withdrawn when, in fact, the correct number was 24. The inspector brought this discrepancy to the attention of the shift test coordinator and the senior nuclear shift supervisor (SNSS). The SNSS immediately halted testing and directed the shift test coordinator to investigate. A determination was made that data had not been obtained or plotted for the 15th control rod withdrawn. The inspector informed the licensee representatives that this was contrary to 10 CFR 50 Appendix B Criterion XI and was considered a violation (354/86-32-01). Criterion XI requires in part that "a test program shall be performed in accordance with written test procedures." Procedure TE-SU.ZZ-041, step 5.2.1, requires that "When each individual control rod reaches the notch position identified in the Rod Pull Listing, record SRM readings in accordance with TE-SU.SE-062." Procedure TE-SU.SE-062, step 5.2.1, requires that SRM response data be recorded, the Inverse Multiplication Data Sheet be updated, the results plotted and a 1/M criticality prediction made. This was not accomplished for the 15th control rod withdrawn (rod 18-39).

The SNSS took immediate steps to address this identified deficiency. The tests were officially placed in "Hold" and the shift test personnel were counseled by the SNSS who instituted strict communication requirements to insure adequate test coordination and to prevent a reoccurrence. The Lead Shift Test Coordinator also undertook to provide additional counseling to the shift test personnel. The inspector was satisfied that these actions were appropriate and sufficient to correct the problem.

At 1945 rod pulls resumed and at 2318 the reactor was declared critical on control rod 46-35 at notch position 06 with a moderator temperature of 123°F. Following initial criticality the inspector independently verified SRM/IRM overlap sufficient to satisfy technical specification requirements when 7 of 8 IRMs came on scale prior to the SRMs exceeding their rod block setpoints. The "A" IRM failed to respond to the increasing neutron flux. This constituted a level 1 acceptance criterion failure for test TE-SU.SE-101.

Witnessing of the initial criticality testing sequence was also performed by the Resident Inspectors and is discussed in their inspection report covering this period.

Findings

One violation was identified for failure to perform testing in accordance with approved procedures.

3.4 Other Startup Test Witnessing

The inspector witnessed performance of the shutdown margin demonstration portion of TE-SU.ZZ-041 and performance of the SRM non saturation of TE-SU.SE-064 on June 29, 1986. The inspector witnessed the coordination of reactor engineers, test engineers and control room operators in the performance of data gathering during the reactor period for shutdown margin. Good coordination and communication were maintained. The inspector independently calculated the shutdown margin using the data obtained during the test using the licensee's method and calculated a 2.4% $\Delta k/k$. shutdown margin, which agreed with the licensee. The required shutdown margin was greater than 0.38%$\Delta k/k$.

The licensee was careful in the performance of SRM Non-Saturation testing to avoid a reactor scram since the shorting links were removed. All SRM's satisfied the non-saturation demonstration test.

At 1:12 p.m. on June 29, 1986 the reactor automatically scrammed on high IRM flux. After having just completed the shutdown margin test and SRM non-saturation demonstration test which completed the testing to install the shorting links, the reactor was brought to a subcritical condition. Due to decreasing neutron counts the operator was downranging the IRM's. The operator was attempting to downrange IRM B and selected IRM D. The IRM D exceeded the RPS trip point and due to the non-coincident logic in effect due to removal of the shorting links the reactor scrammed. All rods withdrawn inserted into the core.

The licensee had decided to install the shorting links after having reviewed the completed testing and restart the reactor. During the restart the licensee began experiencing difficulties in the reactor manual control system such that on June 30, 1986 the licensee was unable to either insert or withdraw a control rod using the normal drive system due to rod insert and withdrawal blocks. The control rod scram ability was not affected. After not being successful to solve the problem on day shift on June 30, 1986 the licensee initiated a manual scram at 4:51 p.m. on June 30, 1986 to more easily troubleshoot the problem. All rods withdrawn inserted. The licensee subsequently determined that a power supply was not operating properly and a bad card existed in the Rod Sequence Control System bypass logic. After repairs the licensee resumed the power ascension test program.

The inspector witnessed portions of the licensee efforts to the adjust the Average Power Range Monitors (APRM) after having performed TE-SU.SE-121, APRM Calibration During Heatup. The licensee adjusted the gain on one APRM and was preparing to adjust the second when difficulties were experienced in the zero voltage readings. The licensee stopped APRM calibration to trouble shoot this problem. During the investigation of this problem the licensee observed that several of the LPRM's were not adjusted as required per the startup test procedure. The licensee began an investigation into the LPRM status to determine the cause. This is being followed up by the Resident Inspector.

3.5 Test Results Evaluation

Scope

The inspector reviewed the field run procedures listed in the discussion section prior to completion of the licensee review process to determine the adequacy of the test. The inspector also witnessed portions of the licensee Technical Review Board Meeting on July 1, 1986 wherein several startup results were reviewed.

- TE-SU.SE-101 SRM/IRM Overlap Test performed June 29, 1986 and July 1, 1986.

During the inspector review of the first test results he questioned whether test data recorded indicated the acceptance criteria was satisfied. The licensee repeated the test after having repaired IRM A which did not respond during the first attempt of SRM/IRM overlap. The second test demonstrated adequate SRM/IRM overlap but again the inspector noted that analysis of the data was performed with data not recorded in the official data package. The licensee took corrective action and instructed personnel to only perform analysis steps with data recorded in the procedure and if the data is not conclusive, initiate the necessary changes to correct the situation.

- TE-SU-SE-102 IRM Range 6/7 Overlap Range 6/7 overlap was demonstrated except for IRM A which was subsequently retested.

The inspector witnessed a portion of the Technical Review Board Meeting held on July 1, 1986 which reviewed the test results TE.SU-SE-062, TE.SU-SE-064 and TE.SU-SE-102. The TRB review identified one question similar to that identified by the inspector in the review of TE.SU-SE-101 above, namely, by review of the data recorded during TE.SU-SE-102, one could not assess whether the test was successful. This was also a contributor to direct licensee action to correct the problem identified. The adequacy of licensee corrective actions will be followed in future inspections.

Findings

No violations were identified.

3.5 Reactor Water Level Indications

As the reactor plant heatup progressed to 250°F, the licensee began observing differences in indicated reactor water level. Certain water level indications for feedwater level control and reactor protection were substantially different from others. Inspector inquiry into the differences in the feedwater level control revealed that the licensee had scheduled the calibrations to be completed at a later date and only one of the three had been recalibrated. The inspector inquired about any other control room instruments that were scheduled to be calibrated "later", only necessary for subsequent plant conditions. There were several instruments so categorized and the licensee indicated that some marking would be placed on the instrument in the control room to identify that the instrument was not yet in calibration.

The inspector reviewed the calculations and surveillance for RPS level transmitters listed below to ascertain that the calculations were adequately transformed into the surveillance procedure.

Calculation: SC-BB-0183 Revision 1 and
SC-BB-0184 Revision 1
Surveillance procedure: IC-SC-BB-029 and
IC-SC-BB-028

The surveillance procedures reflected the calculations. The licensee investigation into the level variations is continuing. Power ascension testing is required to assess water level indications once rated temperature is achieved.

4.0 Independent Measurements, Calculations and Verifications

The inspector performed independent measurements and verifications during the witnessing of surveillance test activities. Prior to the occurrence of the initial criticality the inspector made independent calculations of the expected critical position (ECP) and the ±1% Reactivity Anomaly Limits. When the licensee declared the reactor critical the inspector independently verified the determination by examination of nuclear instrumentation indications. During the SRM/IRM Overlap Verification the inspector independently verified SRM/IRM overlap. During the shutdown margin test the inspector independently calculated the shutdown margin.

In all cases the inspector's measurements, calculations and verifications agreed with those of the licensee. No unacceptable conditions were noted.

5.0 QA/QC Interfaces

The Station Quality Assurance organization has selected approximately 25% of the power ascension tests for surveillance. Power Ascension Test, TE-SU.ZZ-041, Full Core Shutdown Margin Determination, had been selected

for coverage and the inspector observed adequate QA monitoring during the performance of this test. QA was also monitoring licensee activities during the initial plant heatup. QA was also reviewing completed startup test results and participated in the TRB meeting witnessed by inspector on July 1, 1986.

6.0 Exit Interview

At the conclusion of the site inspection on July 3, 1986, an exit meeting was conducted with the licensee's senior site representative (denoted in paragraph 1.0). The findings were identified and discussed. At no time during the inspection was written inspection findings provided to the licensee by the inspector. Based on NRC Region I review of this report and discussions held with the licensee representatives during this inspection, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

ATTACHMENT A

SURVEILLANCE PROCEDURES

KEY: PR = Procedure Reviewed for Technical Specification compliance

VC = Verified test complete via surveillance log

CT = Completed Test Results Reviewed

SW = Surveillance Witnessed

IC-CC.BE-014, Core Spray - Division 3 Channel E21A-K22C Pump Start Delay - Normal Power (PR, VC)

IC-CC.BE-018, Core Spray - Division 3 Channel E21A-K21C Pump Start Delay - Emergency Power (PR, VC)

IC-CC.GS-007, Containment Atmosphere Control Channel A, AS205 Hydrogen Recombiner Instrumentaiton (PR, VC, SW)

IC-CC.SE-014, Nuclear Instrumentation System - Channel B Average Power Range Monitor (PR, VC, SW)

IC-CC.SE-042, Nuclear Instrumentation System - Division 4 Source Range Monitor Channel D, Preamplifier Gain, Discriminator and High Voltage Settings (PR, VC, SW)

IC-CC.SP-004, Process Radiation Monitoring - Division 4 Channel D, D11-K610D Main Steam Line Radiation Monitor (PR, VC)

IC-FT.BH-001, SBLC - System A Relay AC652, Slot 7-14-8 RWCU Isolation on SBLC Initiation (PR, VC)

IC-FT.PE-001, Emergency Load Sequencer System - Diesel Generator A, 1AC428 (PR, CT, SW)

IC-FT.SE-015, Nuclear Instrumentation System Division 3 - Channel C Average Power Range Monitor (PR, VC)

IC-FT.SK-013, Radiation Monitoring - Non-Divisional Channel R-4991 Drywell Noble Gas (PR, VCO

OP-ST.KP-001, Functional Test MSIV Sealing System (SW)

RE-ST.BF-001, Control Rod Drive Scram Time Determination (PR)

ATTACHMENT B

Power Ascension Test Procedures Reviewed

TE-SU.ZZ-041, Full Core Shutdown Margin Demonstration, Revision 4,

TE-SU.SE-062, Source Range Monitor Response to Rod Withdrawal, Revision 2,

TE-SU.SE-101, Source Range Monitor/Intermediate Range Monitor Overlap Verification, Revision 3,

TE-SU.BB-191, Core Performance, Revision 1

TE-SU.CH-272, Low Power Turbine Stop Valve Trip Test, Revision 1