U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-354/88-15

Docket No. 50-354

License No. NPF-57

Licensee: Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Facility Name: Hope Creek Nuclear Generating Station

Inspection At: Hancock Bridge, New Jersey

Inspection Conducted: April 18-22, 1988

Inspector: H. van Kessel 5.20.88
Henri F. van Kessel, Reactor Engineer — date

Approved by: P. K. Eapen 6/2/88
Dr. P. K. Eapen, Chief Special Test Programs Section, EB, DRS — date

Inspection Summary: Inspection on April 18-22, 1988 (Inspection Number 50-354/88-15)

Areas Inspected: Routine Unannounced Inspection of the startup test program following the first refueling, including the review of startup test procedures and test results evaluation, an independent calculation of the core thermal power balance, and the review of activities in the QA/QC interface with the afore said startup test program.

Inspection Results: No violations or deviations were identified.

See Attachment C for acronyms used in this report.

## 1.0 Persons Contacted

Public Service Electric and Gas (PSE&G)

*R. W. Beckwith, Station Licensing Engineer
R. T. Brown, Nuclear Technical Engineer
*J. DeFebo, Quality Assurance Engineer
J. M. Haun, Senior Nuclear Supervisor
*S. L. Funsten, Maintenance Controls Engineer
*R. T. Griffith, Sr., Principal QA Engineer
*S. LaBruna, General Manager Hope Creek Operations
*M. LaVecchia, Principal QA Engineer
*J. E. Metro, Reactor Staff Engineer
*J. Nichols, Technical Manager HCO
J. O'Brien, Reactor Engineer
R. J. Schmidt, Technical Engineer
C. Vondra, Operations Manager

U.S. Nuclear Regulatory Commission

*G. W. Meyer, Senior Resident Inspector

*Denotes those present during exit meeting held on April 22, 1988

## 2.0 Startup Test Program (72700)

### 2.1 Startup Test Procedure Review

The startup test procedures as listed in Attachment A were reviewed for:

- Management review and approval
- Procedure format
- Clarity of stated objectives
- Prerequisites
- Environmental conditions
- Acceptance criteria and their sources
- References
- Initial conditions
- Attainment of test objectives

- Test performance documentation and verification
- Degree of detail for test instructions
- Restoration of system to normal after testing
- Identification of test personnel
- Evaluation of test data
- Independent verification of critical steps or parameters
- Quality control and assurance involvement

Findings

No noncompliances were identified by the inspector within the scope of this inspection.

2.2 Test Result Evaluation

The test procedures listed in Attachment A (with asterisk) were reviewed to verify that adequate testing was accomplished in order to satisfy regulatory guidance and licensee commitments and to ascertain whether uniform criteria were being applied for evaluating completed preoperational tests in order to assure their technical and administrative adequacy.

The test results were reviewed for:

- Test changes
- Test exceptions
- Test deficiencies
- Acceptance criteria
- Performance verification
- Recording of conduct of test
- QC inspection records
- System restoration to normal
- Independent verification of critical steps or parameters
- Identification of test personnel
- Verification that the test results have been approved

The following observations were made for the procedures within the scope of this inspection:

Core Power Distribution Limits: RE-ST.ZZ-001

(1) The plant was operated within the licensed power distribution limits. A P-1, the periodic NSS Core Performance Log, had been printed out from the process computer every hour. In all of these cases, the MFLCPR, MFLPD, and MAPRAT were smaller than unity indicating that the corresponding values in the Technical Specification for the Critical Power Ratio, the Linear Heat Generation Rate and the Maximum Average Planar Linear Heat Generation Rate, respectively, had not been exceeded.

(2) The means used to confirm operation within the limits mentioned under (1) above consisted of the standard General Electric Company (GE) computer software as supplied for BWRs 4/5 and, as such, had been reviewed by NRR some time ago.

(3) The licensee is preparing a backup software program. A 10 CFR 50.59 type evaluation will be made for this backup program upon completion.

(4) An OD-1 program, "Whole Core LPRM Calibration and Base Distributions," had been run at the 75% and the 100% power plateau. The TIP machine normalization factors were obtained by traversing each probe, one at a time, through the common calibration tube. There are 5 TIP machines and 43 channels. A flux chart was produced for each channel at 100% power on April 17, 1988. The normalization of the TIP readings is done by the OD-1 software program. These data were used for APRM calibration. When the LPRM goes out of calibration, Base Crit Code will be printed on the P-1 printout.

(5) The APRM set points did not have to be adjusted for CMPF peaking factor at core max fraction of limiting power density greater than design value because they could stay within the time limit (6 hrs.) by changing control rod pattern.

(6) APRM gain adjustments were made at 98.8% power on April 17, 1988. An OD-3, "Core Thermal Power and APRM Calibration," was made to provide new Gain Adjustment Factors (GAFs). This was done at the 100, 89, 63, and 24 percent power levels.

(7) TIP traces were made after adjustments to observe the impact on CMPF values at the 75% and 100% power plateaux. In each case proper APRM calibration was obtained as indicated by the absence of Base Crit Codes on the P-1 printout.

(8) Although the process computer has not malfunctioned to date, a GE document (NEDO 25443) can be used to ascertain that the plant is operating within licensed limits when the process computer is down. The GE document (NEDO-25443)* was reviewed by NRR. The licensee is preparing an independent software program which can be used on another available computer. This program will be essentially the same as the GE plant monitoring program but can be run on another computer.

*NEDO 25443, "P-1 Backup (PIB), a manual method for core performance evaluation and LPRM calibration," by G. R. Parkos, November 1981, General Electric.

(9) An OD-1 was made just before LPRM gain changes at 100% power. Gain adjustments were made. A new OD-1 was produced after these gain adjustments. All final GAF values were within the Tech. Spec. requirements.

LPRM Calibration: RE-ST.SE-003

(1) LPRM calibration had been performed at the different power levels in accordance with the procedure.

(2) APRMs had been recalibrated following LPRM gain adjustments in accordance with step 5.1.16 in the procedure. (See, for example, APRM calibration calculation/data sheet, dated 4-17-88, attachment 1 of RE-ST.SE-002).

(3) The identification of LPRM calibrations is tracked by the data on Attachment 1 of the procedure. These data are used to detect loss of sensitivity of the sensors. A program is being written in the Fuel Group to do this on the computer. At present, this work is done manually.

(4) LPRM readings were obtained at core conditions existing prior to amplifier gain adjustments. (See "as found" values on Attachment 1). The performance of a full core flux map, by means of the TIP system, was obtained prior to making amplifier gain changes as can be seen in the OD-1 edit. A P-1 calculation was performed after the full core flux map for the calibrated APRM readings (See P-1 dated 4-17-88, 0256).

(5) Calculations were made for the new input calibration currents to be applied to the respective LPRM amplifiers (See Attachment 1). These calculated input currents were applied to make the proper LPRM calibration adjustments.

(6) Each APRM channel was by passed during its respective LPRM group adjustments in accordance with paragraphs 5.1.7 and 5.1.8 of IC-CC.SE-029, "channel calibration nuclear instrumentation system - LPRM gain calibration."

(7) Reviews were made of the Full Core Flux Map and a P-1 after the LPRM calibration adjustments. All of the GAFs were verified to be within the established limits.

(8) An OD-3 was made following the LPRM calibrations to assure that APRM settings were within Tech. Spec. Limits (see for example the OD-3 made on April 17, 1988)

APRM Calibration: RE-ST.SE-002

(1) The APRM system was properly calibrated to the Core Thermal Power as can be seen, for instance, in Attachment 1 of the procedure, dated April 17, 1988.

(2) As a prerequisite (2.3 paragraph), the core was maintained at steady state operating conditions at the desired power level and recirculation flow rate during the calibration.

(3) Only one APRM channel, per RPS bus, was bypassed at a time. Since there is only one switch per RPS bus, there is no physical possibility for this to happen.

(4) The APRMs were adjusted to read calculated % of rated power as evidenced, for example, by the calculation of Attachment 1 of the procedure, dated April 21, 1988. The same reference also shows the "as left" APRM readings versus % thermal power.

Core Thermal Power Evaluation: RE-RA.ZZ-001

(1) The power range nuclear instruments were properly adjusted to agree with the heat balance results as evidenced by Attachment 1 of the procedure for the 98.7% power level during startup.

(2) An independent hand calculation was made by the inspector using the licensee's procedure (RE-RA.ZZ-001). The results are discussed in this report under Section 4.

(3) The results of this procedure were reviewed and approved by the licensee's designees in accordance with Administrative Procedure AP.12. Approvals were made by the Senior Reactor Supervisor.

Determination of Reactor Shutdown Margin (SDM): RE-ST.ZZ-007

(1) Changes to SDM, due to inoperative control rods, were not experienced during this startup.

(2) The SDM after startup was in agreement with the Technical Specifications as evidenced by the completed attachments of this procedure, dated April 10, 1988, soon after criticality.

(3) The reactor went critical on 2152 notches. The demonstrated SDM=1.983% $\underline{\Delta k/k}$ as shown in Attachment 3 of the procedure. The SDM is required to be greater than .769% $\underline{\Delta k/k}$.

(4) The licensee reviewed the data supplied by the vendor (GE) and amongst others used it in their own SDM determination. The data were presented by GE in their Document No. 23A5879, rev. 0, "Hope Creek Cycle 2, Cycle Management Report", by H. H. Yeager., dated March 25, 1988. PSE&Gs fuel department produced, independently, a document by the same title i.e., NFU-0092, dated March 29, 1988. The latter document, however, does not contain any reference to the GE document, raising the impression that the GE data were not used in the preparation of same or for comparison of results. This item was discussed with the representatives of the nuclear fuel department. They concurred that the reference should be made because the comparison of the data of the two reports was made and, in the case of the SDM, the PSE&G data were more conservative. The GE document will be added to the reference listing of the PSE&G report.

(5) No stuck out rod problems were experienced and, therefore, no special SDM determination had to be made.

3. <u>QA/QC Interface</u>

The participation of Station QA in test witnessing of the post refueling startup tests was evaluated by the inspector. The Station QA Surveillance Reports as listed in Attachment B were reviewed.

The inspector observed that none of the QA Surveillance reports contained any unsatisfactory finding. The check lists appeared to cover all of the important acceptance criteria and none of these criteria were violated during the tests.

Station QA Surveillance Report No. 88-246, however, found that testing was performed while the system (in test) was legitimately being worked on under work order 880303065 and had relay E21-K14B missing (in plastic bag) from the system. This finding was not identified as an unsatisfactory item. The inspector discussed this item with the QA personnel responsible for the surveillance. The QA personnel provided the following statement ? an explanation:

"There was a lack of communications between the supervisors in the I&C Department, which permitted 1C-FT-BB-004 to be run; however, the Operations Department had declared this system INOPERABLE prior to performance of 1C-FT-BB-004. Since we were in a refuel outage, this system was not required to be operable in Mode 5 with cavity flooded.

When a system is not required for a particular operating condition or is in an outage declared INOPERATIVE by operations, it is a normal acceptable practice for multiple Station departments to perform several maintenance functions in parallel. Upon completion of the maintenance, a channel calibration/functional test is performed prior to declaring the system operable."

In the case noted above, the I&C miscommunication resulted in an attempt to run the Functional Test prior to completion of maintenance work.

The requirements of the Station retest program and the fact that the system had previously been declared inoperable provide the necessary insurance that return to operable status would be achieved."

The inspector found the above explanation to be adequate.

No discrepancies or unacceptable conditions, other than the one mentioned above, were noted in the review of the QA Surveillance reports.

4. Independent Effort

The inspector, independently, calculated the thermal power balance for the core in accordance with instructions contained in procedure RE-RA.ZZ-001 and for the data of April 16, 1988. Discrepancies were found for formulae line items [6] and [7] where the formulae include a factor of 32 to be subtracted from the mV reading in each case. This factor no longer exists because the transmitter range was changed to (0-166mV). The results for the April 16 data were not affected because an alternate method was used to calculate the pertinent values.

Another discrepancy involved line items [4] and [5] on page 2 of Attachment 1. The mV values versus temperature on the table of page 5 were misread causing the calculated temperatures to be higher for the correct readings. The impact of the latter discrepancy was that the calculated core thermal power was 2 MWth. higher than the actual.

The licensee issued rev. 3 of procedure RE-RA.ZZ-001 containing the corrections to lines A203 and A204 for the new transmitter range (0-160mV).

5. Plant Tours

The inspector made tours of the plant including the control room, the control building, turbine building, and the reactor building to observe housekeeping, testing activities and cleanliness.

The inspector, noted that APRM A and B are in one compartment of panel 10-C608. Both channels have to be bypassed at the same time to avoid the possibility of inadvertent reactor scrams. A warning to this effect is posted on the cabinet door. This design feature is common for BWR 4/5 designs and has been reviewed by the NRC in the past.

6. Exit Interview

At the conclusion of the site inspection, on March 22, 1988, an exit interview was conducted with the licensee's senior site representatives (denoted in Section 1). The findings were identified and previous inspection items were discussed.

At no time during this inspection was written material provided to the licensee by the inspector. Based on the NRC Region I review of this report and discussions held with licensee representatives during this inspection, it was determined that this report does not contain information subject 10 CFR 2.790 restrictions.

## Attachment A

### Procedure Review

| Proc. No. | Title | Rev. No. | Approval Date |
|---|---|---|---|
| RE-FR.ZZ-008(Q) | Verification of Fuel Location | 2 | 02-18-88 |
| RE-RA.BB-001(Q) | Recirculation Flow Determination | 1 | 05-12-86 |
| RE-RA.BB-002(Q) | Core Flow Determination | 1 | 09-23-86 |
| *RE-RA.ZZ-001(Q) | Core Thermal Power Evaluation | 1 | 09-02-86 |
| RE-RA.ZZ-005(Q) | Reactor Period Measurement | 1 | 03-10-86 |
| RE-RA.ZZ-005(Q) | Reactor Period Measurement | 1 | 03-10-86 |
| RE-SO.RJ-003(Q) | NSS Computer Reinitialization and OD-15 program operation | 1 | 11-21-86 |
| RE-SO.SE-001(Q) | Traversing Incore Probe System Operation | 5 | 10-07-87 |
| *RE-ST.ZZ-001(Q) | Core Thermal Limits Evaluation Process Computer Method | 2 | 07-30-87 |
| RE-ST.BF-001(Q) | Control Rod Drive Scram Time Determination | 1 | 07-16-86 |
| RE-ST.SE-001(Q) | APRM Setpoint Surveillance | 5 | 09-23-87 |
| *RE-ST.SE-002(Q) | APRM Calibration Surveillance | 3 | 11-26-86 |
| *RE-ST.SE-003(Q) | LPRM Calibration | 5 | 01-08-87 |
| RE-ST.ZZ-005(Q) | Reactivity Anamoly Check | 2 | 12-15-86 |
| *RE-ST.ZZ-007(Q) | Shutdown Margin Demonstration | 4 | 06-25-86 |
| IC-CC.SE-029(Q) | Channel Calibration Nuclear Instrumentation System LPRM Gain Calibration | 6 | 09-10-87 |

*Used in test result evaluation.

Attachment B

Review of QA Surveillance Reports

| Q/A Surv. Report No. | Surveillance Activity | Completion Date |
|---|---|---|
| 88-276 | Verification of Fuel Location | 3-20-88 |
| 88-275 | I&C Channel Calibration Redundant Reactivity Control System, Div 1 Channel B, ATWS Recirc. Pump Trip | 3-20-88 |
| 88-273 | Reactor Engineering/Core and Spent Fuel Pool Mapping | 3-20-88 |
| 88-248 | TS Surveillance of LPRMs (TS 4.7.6.1 and 4.7.6.2c) | 3-15-88 |
| 88-223 | Control Rod Removal and Installation | 3-04-88 |
| 88-247 | Functional Test Nuclear Instr. System Channel A Source Range Monitor TS Surveillances | 3-13-88 |
| 88-246 | Functional Test Nuclear Boiler DIV. 2 Channel B21-N691F Reactor Vessel level (trips 1, 2, 8 - CS, RHR, ADS, RCIC) | 3-12-88 |
| 88-264 | Time Response Test for Transmitter 1 BBPT-N078D-B21 | 3-15-88 |
| 88-270 | Time Response Test for PIdsh-686D-B21 | 3-17-88 |
| 88-093 | Fuel Handling, Off Load of Core | 2-25-88 |
| 88-242 | Fuel Movement, Shuffle and Reload | 3-18-88 |

## Attachment C

## Acronyms

| | | |
|---|---|---|
| * | APRM | Average Power Range Monitor |
| | BASE CRIT CODE | LPRM strings which may require new data accumulated by TIP System (shown on P-1 printout) |
| | BWR | Boiling Water Reactor |
| | CMPF | Peaking factor at core MFLPD |
| * | CPR | Critical Power Ratio |
| | GAF | Gain Adjustment Factor |
| * | LPRM | Local Power Range Monitor |
| | MAPRAT | Maximum Fraction of Limiting Average Planar Linear Heat Generation Rate |
| | MFLCPR or MFLCP | Maximum Fraction of Limiting Critical Power Ratio |
| | MFLPD | Maximum Fraction of Limiting Power Density |
| * | NSS | Nuclear Steam System |
| | OD | On Demand (Programmatic Output of PMS) |
| * | PMS | Performance Monitoring System |
| * | RPS | Reactor Protection System |
| * | SDM | Shut Down Margin |
| * | TIP | Traversing Incore Probe |
| | $\Delta k/k$ | Reactivity, fractional change in neutron population per neutron generation. |

* Also defined in NUREG-0544, rev. 2.