PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 3

STARTUP REPORT

CYCLE 9

SUBMITTED TO

THE UNITED STATES NUCLEAR REGULATORY COMMISSION

PURSUANT TO

FACILITY OPERATING LICENSE DPR-56

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# SECTION 1

# INTRODUCTION/SUMMARY

## 1.1 REPORT ABSTRACT

This Startup Report, written to comply with Technical Specification paragraph 6.9.1.a, consists of a summary of the Startup and Power Escalation Testing performed at Peach Bottom Unit 3. During this refueling outage, 256 new bundles of GE9B were added, while the remaining bundles were shuffled. All reused bundles were inspected and cleaned as necessary, to remove debris. Additionally, the Bottom Head and the Bottom Head Drain line were cleaned and inspected.

The report addresses the Startup Tests identified in chapter 13.5 of the FSAR and includes a description of the measured values of the operating conditions or characteristics obtained during the test program with a comparison of these values to the Acceptance Criteria. Pertinent portions of the Digital Feedwater System (Mod 1843) testing are also addressed in this report.

## 1.2 SUMMARY

Peach Bottom Unit 3 was out-of-service from September 14, 1991 to January 8, 1992 to accommodate a refueling outage. The unit returned to service on January 9, 1992 and reached full power operation January 20, 1992.

The successfully implemented Peach Bottom 3 Startup Program ensures that the eighth refueling outage of Peach Bottom 3 has resulted in no conditions or system characteristics that diminishes the safe operation of the plant. The tests and data referenced in this report are on file at the Peach Bottom Atomic Power Plant.

## 1.3 PEACH BOTTOM PLANT DESCRIPTION

The Peach Bottom Atomic Power Station is a two unit nuclear power plant. The two units share a common control room, turbine operating deck, radwaste system, and other auxiliary systems. Each unit has a separate refueling floor.

The PBAPS is located partly in Peach Bottom Township, York County, partly in Drumore Township, Lancaster County, and partly in Fulton Township, Lancaster County, in southeastern Pennsylvania, on the westerly shore of Conowingo Pond at the mouth of Rock Run Creek. The plant is about 38 mi north-northeast of Baltimore, Maryland, and 63 mi west-southwest of Philadelphia, Pennsylvania. Conowingo Pond is formed by the backwater of Conowingo Dam on the Susquehanna River; the dam is located about 9 mi downstream from the Unit 3 reactor. The nearest communities are Delta, Pennsylvania and Cardiff, Maryland approximately 4 mi southwest of Unit 3.

PECO owns the 620-acre property lying within the site boundary except that the immediate area on which Unit 2 and 3 stand is owned by PECO, Public Service Electric and Gas Company, Delmarva Power & Light Company, and Atlantic Electric Company as tenants in common. The adjoining area lying along the discharge canal in a downstream direction, was owned by the Philadelphia Electric Power Company, a wholly-owned subsidiary of PECO, as part of the Federal Power Commission Project No. 405 (Conowingo). An application dated February 25, 1969, was filed with the Federal Power Commission to alter the Conowingo Project boundary so as to exclude from that project this 26-acre tract and a littoral strip of land about 7,000 ft in length (upstream of the 26-acre tract) which was subject to flowage rights in favor of the Conowingo Project. On October 13, 1970, the Commission issued an order approving the application.

Although not a part of the Peach Bottom project, the remaining land along both sides of Conowingo Pond, from below Holtwood Dam to Conowingo Dam ranging up to 300 ft back from the waterline, is owned by PECO's wholly owned subsidiaries.

Each of the Peach Bottom units employs a General Electric Company boiling water reactor (BWR) designed to operate at a rated core thermal power of 3293 MWt (100% steam flow) with a corresponding gross electrical output of 1092 MWe. Approximately 37 MWe are used for auxiliary power, resulting in a net electrical output of 1055 MWe. See Table 1.2-1 for Peach Bottom Plant Parameters.

The primary containment is a pressure suppression system and houses the reactor vessel, the reactor coolant recirculation systems, and other primary system piping. The primary containment system consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and the suppression pool, isolation valves, vacuum breakers, containment cooling systems, and other service equipment.
The drywell is a light bulb-shaped steel pressure vessel with a spherical lower portion.
The drywell is enclosed in reinforced concrete for shielding purposes.

The Architect Engineer and constructor was Bechtel Power Corporation.

TABLE 1.3-1
PEACH BOTTOM UNIT 3 PLANT PARAMETERS

| Parameter | Value |
|---|---|
| Rated Power (MWt) | 3293 |
| Rated Core Flow (Mlb/hr) | 102.5 (1) |
| Reactor Dome Pressure (psia) | 1020 |
| Rated Feedwater Temperature (Deg. F.) | 376.1 (2) |
| Total Steam Flow (Mlb/hr) | 13.37 |
| Vessel Diameter (in) | 251 |
| Total Number of Jet Pumps | 20 |
| Core Operating Strategy | Control Cell Core |
| Number of Control Rods | 185 |
| Number of Fuel Bundles | 764 |
| Fuel Type | 8 X 8 (Barrier) |
| Core Active Fuel Length (in) | 150 |
| Cladding Thickness (in) | 0.032 |
| Channel Thickness (in) | 0.080 |
| MCPR Operating Limit | 1.27 for BP/P8X8R & GE8X8EB 1.29 for GE8X8NB (3) |
| Maximum LHGR (KW/ft) | 13.4 for BP/P8X8R & GE8X8EB<br>14.4 FOR GE8X8NB |
| Turbine Control Valve Mode | Modified Partial Arc |
| Turbine Bypass Valve Capacity (% NBR) | 25 |
| Relief Valve Capacity (% NBR) | 87.4 |
| Number of Relief Valves | 13 |
| Recirculation Flow Control Mode | Variable Speed M/G Sets |

NOTES FOR TABLE 1.3-1

(1) Unit 3 is analyzed for increased core flow to 105%.

(2) Unit 3 is analyzed for a 60 degrees F final Feedwater temperature reduction.

(3) See Core Operating Limits Report for Peach Bottom Reload 8, Cycle 9 for specifics.

## 2.0 Test Objectives, Description, Acceptance Criteria, and Results

The tests comprising the startup and power test program are discussed with reference to the particular test purpose, brief description, statement of acceptance criteria where applicable, and the test results.

Where applicable, a definition of the relevant acceptance criteria for the test is given and is designated either "Level 1 " or "Level 2." A Level 1 criterion normally relates to the value of a process variable assigned in the design of the plant, component systems, or associated equipment. If a Level 1 criterion is not satisfied, the plant will be placed in a suitable hold-condition until resolution is obtained. Tests compatible with this hold-condition may be continued. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 criterion are satisfied.

A Level 2 criterion is associated with expectations relating to the performance of systems. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical techniques used for the predictions would be started.

## 2.1 Chemical and Radiochemical

### Objectives

The principal objectives of this test are (1) to maintain control of and knowledge about the quality of the reactor coolant chemistry, and (2) to determine that the sampling equipment, procedures, and analytic techniques are adequate to supply the data required to demonstrate that the coolant chemistry meets water quality specifications and process requirements.

Secondary objectives of the test program include data to evaluate the performance of the fuel, operation of the demineralizers and filters, condenser integrity, operation of the off-gas system, and calibration of certain process instruments.

### Description

Subsequent to fuel loading during reactor heatup and at each major power level change, samples will be taken and measurements will be made to determine the chemical and radiochemical quality of reactor water and reactor feedwater, gaseous activities leaving the air ejectors, decay times in the off-gas lines, and performance of filters and demineralizers. Calibrations will be made of monitors in the stack, liquid waste system, and liquid process lines.

### Acceptance Criteria

#### Level 1

Water quality must be known and must conform to the Water Quality specifications at all times.

The activities of gaseous and liquid effluents must be known and they must conform to license limitations.

Chemical factors defined in the Technical Specifications must be maintained within the limits specified.

#### Level 2

Chemical factors in the Fuel Warranty must be maintained within the specified limits.

Results

a. Prior to Fuel Load:

Chemistry Limits per CH-10 (Chemistry Goals) were verified on a daily basis.

b. Prior to Startup:

Chemistry requirements were verified by RT 7.8 (Chemistry preparation for Reactor Startup) on January 10, 1992. The Shift Chemist also verified that chemistry limits were within specification per CH-10.

c. During Startup:

Coolant chemistry was determined to meet water quality specifications and process requirements via ST-C-095-824-3 (Reactor Startup Chemistry {<100Klbs/hr} complete' auary 3, 1992.

## 2.2 Radiations Measurements

Objectives

To determine the background gamma and neutron radiation levels in the plant environs prior to operation in order to provide base data on activity buildup. Also to monitor radiation at selected power levels to assure the protection of personnel and continuous compliance with the guideline standards of 10CFR20 during plant operation.

Description

Subsequent to fuel loading, during reactor heatup and at various power levels, gamma radiation level measurements and, where appropriate, thermal and fast neutron dose rate measurements, will be made at significant locations throughout the plant. All potentially high radiation areas will be surveyed.

Acceptance Criteria

Level 1

The radiation doses of plant origin and occupancy times shall be controlled consistent with the guidelines of the standards for protection against radiation outlined in 10CFR20 NRC General Design Criteria.

Results

a. Prior to Fuel Load:

   Routine surveys were taken throughout the protected area to assure personnel safety and to maintain Activity Buildup base data via HP 200 (Routine Survey Program).

b. During Startup:

   Radiation was monitored to assure the protection of personnel and continuous compliance with the guidelines of 10CFR20 during plant operation.

## 2.3 Fuel Loading

### Objectives

The purpose of this test is to load new fuel and shuffle existing fuel safely and efficiently.

### Description

Prior to loading a control cell, control rods and control blades will be installed. Fuel loading/shuffling will be such that each bundle loaded will be neutronically coupled to an operable SRM, in that quadrant.

### Acceptance Criteria

#### Level 1

The criteria for successful completion of this test are that (1) the core is fully loaded, and (2) the core shutdown margin demonstration has been completed.

### Results

Fuel loading was completed on December 8, 1991 via FH-6C (Core Component Movement and Core Alteration Procedure During a Fuel Handling Outage). Bundle locations and orientation were verified via RT-R-004-970-3 (Core Post-Alteration Verification) and completed on December 9, 1991. Each Control Rod was withdrawn and inserted to verify rod coupling integrity, proper rod withdrawal and insertion speeds, and subcriticality.

Level 1 criteria were met when core shutdown margin was demonstrated with a fully loaded core on January 2, 1992. Control Rod Test data is documented in ST 10.8 (Control Rod Performance Test) completed on December 30, 1991.

## 2.4 Shutdown Margin

Objectives

The purpose of this test is to demonstrate that the reactor will be subcritical throughout the fuel cycle with any single control rod fully withdrawn.

Description

Subcriticality was demonstrated with the "in-sequence method".

Acceptance Criteria

Level 1

a) The fully loaded core must be subcritical by at least 0.38%$\Delta$K throughout the fuel cycle with any rod fully withdrawn.

Results

An "in-sequence" shutdown margin of 1.915% delta K/K was obtained during the initial reactor startup in the A sequence. This satisfies the Level 1 criteria that the core must be subcritical by at least 0.38% delta K/K with any rod fully withdrawn. Test data is documented in ST-R-002-910-3 (Shutdown Margin) completed January 2, 1992.

The design predicted core Keff was compared to the measured value at initial startup on January 2, 1992. The predicted Keff was 1.00415 as compared to the measured Keff of 1.0033. The difference between predicted and measured values was 0.00085, which meets the acceptance criteria of ± 1%. The test data is documented in ST 3.9 (Critical Eigenvalue Comparison) completed January 2, 1992.

## 2.5 Control Rod Drives

Objectives

To demonstrate that the CRDS operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movements. Also, to determine the initial operating characteristics of the entire CRDS.

Description

The CRD tests performed during the startup are designed as an extension of the preoperational CRDS tests.

Acceptance Criteria

Level 1

Each CRD must have a normal withdraw speed less than or equal to 3.6 in/sec (9.14 cm/sec), indicated by a full 12-ft stroke in greater than or equal to 40 sec.

Upon scramming, the average of the insertion times of all operable control rods, exclusive of circuit response times, must be no greater than:

| Percent Inserted | Insertion Time (sec) | |
|---|---|---|
| 5 | 0.375 | Scram time is measured |
| 20 | 0.90 | from time pilot scram |
| 50 | 2.0 | valve solenoids are |
| 90 | 5.0 | deenergized. |

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

| Percent Inserted | Insertion Time (sec) | |
|---|---|---|
| 5 | 0.398 | Scram time is measured |
| 20 | 0.950 | from time pilot scram |
| 50 | 2.120 | valve solenoids are |
| 90 | 5.300 | deenergized. |

Level 2

Each CRD must have a normal insert or withdraw speed of 3.0 ± 0.6 in/sec (7.62 ± 1.52 cm/sec), indicated by a full 12-ft stroke in 40 to 60 sec.

Results

Scram timing (at rated pressure) was performed on all CRDs as per ST-10.7, completed on December 15, 1991. Scram timing of selected CRDs was completed on January 14, 1992.

In cold shutdown, each CRD was tested for position indication, normal insert/withdrawal times and coupling via ST 10.8 (Control Rod Withdrawal Tests) completed on December 30, 1991.

At rated reactor pressure, Position Indication (GP-2 Normal Plant Startup), and Coupling Checks (ST 10.8-1 CRD Coupling Integrity Test) were performed and completed on January 16, 1992. The testing performed satisfied Level 1 and 2 criteria.

2.6 Control Rod Seque[illegible]

Objectives

To achieve initial criticality in a safe and efficient manner using the startup 1 withdr[illegible] sequence. Also to determine the effect on reactor power of control rod motion at vari[illegible]s operating conditions.

Description

The start up 1 control rod withdraw sequence has been calculated which completely specify control rod withdraws from the all-rods-in condition to the rated power configuration. The sequence will be used to attain cold criticality. Critical rod patterns will be recorded periodically as the reactor is heated to rated temperature.

Movement of rods in a prescribed sequence is monitored by the rod worth m[illegible]nimizer, which will prevent out-of-sequence withdraw. Also, not more than tw[illegible] rods may be inserted out of sequence.

Results

The sequence was defined in GP 2-3 Appendix 1 (Startup Rod Withdrawal Sequence Instructions) and verified for use by the Rod Worth Minimizer (RWM) via ST-R-62A-220-3 (RWM Sequence Loading Verification) on December 12, 16, & 20, 1991. ST-0-62A-210-3 RWM Operability Check) was performed on January 1, 1992 and ST-R-002-910-3 (Shutdown Margin) recorded the critical rod pattern on January 2, 1992.

## 2.7 SRM Performance

Objectives

To demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequence provide adequate information to the operator during startup.

Description

Source range monitor count-rate data will be taken during rod withdrawals to critical and compared with stated criteria on signal count-to-noise count ratio.

Acceptance Criteria

Level 1

There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRM's.

Results

Source Range Monitor (SRM) instrumentation operability was checked during the performance of startup procedure GP-2. The criteria of a minimum count rate of 3 counts/sec. was verified to be met for the SRM's. Data is documented in GP-2 dated January 20, 1992.

## 2.8 IRM Performance

Objectives

The purpose of this test is to adjust the IRMS to obtain an optimum overlap with the SRMS and APRMS.

Acceptance Criteria

Level 1

Each IRM channel must be adjusted so the overlap with the SRM's and APRM's is assured. The IRM's must produce a scram at 120 on a full scale of 125.

Results

Intermediate Range Monitor (IRM) performance was tested. The IRM scram setpoints met Level 1 criteria of SI3N-60C-IRM-A(B)4CW (Intermediate Range Monitor Channel "A"("B") Calibration/Functional Check) dated December 2, 9, 17, 24, and 31, 1992 (for the "A" channel), and December 3, 10, 17, 24, and 31, 1992 (for the "B" channel).

## 2.9 LPRM Calibration

Objectives

To calibrate the Local Power Range Monitoring (LPRM) System.

Description

The LPRM channels will be calibrated to make the LPRM readings proportional to the neutron flux in the narrow-narrow water gap at the chamber elevation. Calibration factors will be obtained through the use of a process computer calculation that relates the LPRM reading to average fuel assembly power at the chamber height.

Acceptance Criteria

Level 1

With the reactor in the rod pattern and at the power level at which the calibration is to be performed, the meter reading of each LPRM chamber will be proportional to the average heat flux in the four adjacent fuel rods at the height of the chamber.

Results

Local Power Range Monitor (LPRM) calibrations were performed at 30% rated thermal power on January 13, 1992, and at 100% rated thermal power on January 28, 1992 per ST-R-60A-230-3 (LPRM Gain Calibration).

## 2.10 APRM Calibration

Objectives

To calibrate the Average Power Range Monitor (APRM) channels.

Description

A heat balance will be made at least once each shift and after each major power level change. Each APRM channel reading will be adjusted to be consistent with the core thermal power as determined from the heat balance.

Acceptance Criteria

Level 1

The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

Results

Numerous Average Power Range Monitor (APRM) calibrations were completed during Power Ascension. Test data is documented in ST-R-60A-210-3 completed from January 7, 1992 at 6% rated thermal power to January 21, 1992 at 100% rated thermal power.

## 2.11 Process Computer

### Objectives

To verify the performance of the process computer under operating conditions.

### Description

Following fuel loading, during plant heatup and the ascension to rated power, the nuclear steam supply system and the balance-of-plant system process variables sensed by the computer as digital or analog signals will become available. Verify that the computer is receiving results of performance calculations of the nuclear steam supply system and the balance-of-plant are correct. Verify proper operation of all computer functions at rated power operating conditions.

### Acceptance Criteria

#### Level 2

Program OD-1 and P-1 will be considered operational when the Thermal Limits calculated by an independent method and the process computer are in the same fuel assembly and do not differ in value by more than 10 percent, and when the LPRM calibration factors calculated by the independent method and the process computer agree to within 5 percent.

### Results

A manual calculation was performed via RT-R-059-500-3 (checkout of the NSS Computer Calculation of Core Thermal Power) at approximately 100% power on January 23, 1992.

The Thermal Limit calculations were verified by General Electric via BUCKLE with full-power data provided by the Process Computer. The BUCLE run was performed on February 23, 1992 with plant data obtained on February 11, 1992 while Unit 3 was operating at steady state full power conditions.

2.12 Reactor Core Isolation Coolant System

Objectives

To verify the operation of the RCICS at operation reactor pressure conditions.

Description

A controlled start of the RCICS will be done at a reactor pressure of 150 psig and a quick start will be done at a reactor pressure of 1,000 psig. Verify proper operation of the RCICS and determine time to reach rated flow. These tests may be performed with the system in the test mode so that discharge flow will not be routed to the reactor pressure vessel.

Acceptance Criteria

Level 1

The RCICS must have the capability to deliver rated flow, 600 gpm, in less than or equal to the rated actuation time, 30 sec, against rated reactor pressure.

Results

A controlled start was performed at 150 psig via ST 10.7 (RCIC Flow Rate at 150 psig) on January 4, 1992; Although RCIC was unable to meet the required flow of 600 gpm at 100 psid over reactor pressure, it did meet that flow rate requirement at 80 psid over reactor pressure. A 10CFR50.59 review was performed by Nuclear Engineering and RCIC was declared operable by PORC 92-002 on January 4, 1992. A Quick start at rated pressure was performed via ST 6.11F-3 (RCIC Pump, Valve, Flow and Cooler) on January 7, 1992.

## 2.13 High Pressure Coolant Injection System

Objectives

To verify the proper operation of the HPCIS throughout the range of reactor pressure conditions.

Description

Controlled starts of the HPCIS will be done at reactor pressures near 150, and 1,000 psig during the heatup phase, and a quick start will be initiated during Phase 3. Verify proper operation of the HPCIS, determine time to reach rated flow, adjust flow controller in HPCIS for proper flow rate and adjust overspeed trip of HPCI turbine. These tests will be performed with the system in the test mode so that discharge flow will not be routed to the reactor pressure vessel.

Acceptance Criteria

Level 1

The time from actuating signal to required flow must be less than 25 sec with reactor pressure at 1,000 psig. With pump discharge pressure at 1,220 psig the flow should be at least 5,000 gpm. The HPCI turbine must not trip off during startup.

Results

A controlled start was performed at 150 psig via ST 10.1-3 (HPCI Flow Rate at 150 psig) on January 3, 1992. A quick start at rated pressure was performed via ST 6.5F-3 (HPCI Pump, Valve, Flow and Cooler) on Jan  ry 7, 1992.

## 2.14 Core Power Distribution and TIP Uncertainty

Objectives

To (1) confirm the reproducibility of the TIPS readings, (2) determine the core power distribution in three dimensions, and (3) determine core power symmetry.

Description

The check is made with the plant at steady-state condition by producing several TIP traces in the same location with each TIP machine. The traces are evaluated to determine the extent of deviations between traces from the same TIP machine.

Core power distribution including power symmetry will be obtained during the power ascension program. Axial power traces will be obtained at each of the TIP locations. Several TIPS's have been provided to obtain these traces. A common location can be traversed by each TIP chamber to permit intercalibration. The results of the complete set of TIP traces will be evaluated to determine core power symmetry.

Acceptance Criteria

Level 2

In the TIP reproducibility test, the TIP traces shall be reproducible within ± 3.5 percent relative error or ± 0.15 in (3.8 mm) absolute error at each axial position, whichever is greater.

Results

RE-27 (Core Power Symmetry and TIP Reproducibility Test) was performed at 100% rated core thermal power to demonstrate TIP reproducibility and core symmetry. The TIP readings were within the standard deviation used to establish safety limit criteria of 8.7%, per General Electric Document NEDE-24011 Table S.2-1. The maximum deviation between symmetrically located pairs satisfied the 25% acceptance criteria for core power symmetry.

2.15 Core Performance

Objectives

To evaluate the core performance parameters of the core flow rate, core thermal power level, Maximum Fraction of Limiting Critical Power Ratio, maximum fuel rod Linear Heat Generation Rate (LHGR), and Maximum Average Planar Linear Heat Generation Rate (APLHGR).

Description

Core power level, recirculation flow, fuel assembly power, MFCP, MFLPD, and MAPR will be determined. Plant and in-core instrumentation, and conventional heat balance techniques will be used. This will be performed above 25 percent power and at various pumping conditions.

Acceptance Criteria

Level 1

Steady-state reactor power shall be limited to 3,293 MWt.

Maximum Fraction of Limiting Critical Power Ratio (MFCP) shall not exceed 1.00 during steady-state conditions when evaluated at the operating power level.

Maximum Fraction of Limiting Power Density (MFLPD) shall not exceed 1.00

Maximum Average Planar Ratio (MAPR) shall not exceed 1.00.

Results

The core thermal limits were verified daily above 25% power via the Process Computer. ST-R-002-900-3 (Reactor Anomalies) verified the full Power Control Rod Pattern provided by the PECo Fuel Management Section was completed on January 22, February 18, and March 16, 1992.

2.16 Feedwater System

Objectives

To demonstrate acceptable reactor water level control, evaluate and adjust feedwater controls, and demonstrate capability of automatic flow runback feature to prevent low water level scram following trip of one feedwater pump.

Description

Reactor water level set point changes of approximately ± 6 inches will be used to evaluate and acceptably adjust the feedwater control system settings for all power and feedwater pump modes.

One of the three operating feedwater pumps will be tripped at 75 percent and full power while the automatic flow runback circuit acts to drop power to within the capacity of the remaining two pumps.

Various additional test required by the modification to the Feedwater Level Control System (Mod 1843) are described in the results section.

Acceptance Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to feedwater system changes.

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to feedwater system set point changes when the plant is operating above the lower limit of the master flow controller. The automatic flow runback feature will prevent a scram from low water level following a trip of one feedwater pump.

Results

Feedwater Controller Stability testing was performed to demonstrate acceptable reactor water level control. The response of each Reactor Feedpump to changes to the Master Level controller of positive and negative 3 and 6 inch level changes was observed.

ST-0-02B-250-3 Reactor Water Level Instrument Perturbation Test, Completed Sat. 1/01/92.

MAT 1843 F demonstrated the operability of the Start Up Bypass "C" Feedpump Discharge Bypass Level Controller to Maintain Reactor Level, completed Sat. 1/06/92, and 1/07/92.

MAT 1843 H demonstrated acceptable response to small signal step changes to the Maste Level Controller, completed Sat. on 1/22/92.

MAT 1843 K demonstrated satisfactory response of the Feedwater Level Control System during the trip of one Condensate Pump with Reactor Power greater than 90%, completed Sat on 1/20/92, and 2/07/92.

MAT 1843 J proved the satisfactory response of the Feedwater Level Control System during the trip of one Reactor Feed pump with Reactor Power between 70 to 75% and then again with Reactor Power greater than 90%, completed Sat on 1/29/92.

MAT 1843 L verified the ability of the Feedwater Level Control System to Control Reactor Level following a Recirc Pump trip from 95% power and 100% flow, completed Sat on 1/21/92.

MAT 1843 M tested the response of the Feedwater Level Control System to a loss of its 120 VAC feeds, completed Sat on 12/28/91.

MAT 1843 P demonstrated that the Feedwater Level Control System can control Reactor Level within +/- 6.0 inches of the desired setpoint when increasing and decreasing reactor power at a rate of 10 MWe/min, completed Sat on 1/19/92.

MAT 1843 Q demonstrates the Fault Tolerance of the Feedwater Level Control System; performed, but failed due to Reactor Level dropping more than anticipated. This problem is being worked on by Engineering and the Vendor, and is not expected to be resolved by the time this report is issued.

## 2.17 Relief Valves

Objectives

To verify the proper operation of the dual purpose safety relief valves, to determine their capacity, and to verify their leak tightness following operation.

Description

The main steam relief valves will each be opened manually so that at any time only one is open. Capacity of each relief valve will be determined by the amount the bypass or control valves close to maintain reactor pressure. Proper reseating of each relief valve will be verified by observation of temperatures in the relief valve discharge piping.

Acceptance Criteria

Level 2

Each relief valve is expected to have a capacity of at least 800,000 lb/hr at a pressure setting of 1,080 psig. Relief valve leakage must be low enough that the temperature measured by the thermocouples in the discharge side of the valves falls to within 10°F of the temperature recorded before the valve was opened. Each valve must move from fully closed to fully opened in 0.3 sec.

Results

Each Safety Relief Valve (SRV) was manually cycled at 168 psig Reactor Pressure. Test data is documented in ST-0-01C-440-3 (Relief Valve Manual Actuation) dated January 3, and January 6, 1992.

2.18 Recirculation Flow Control System

Objectives

To de ..nine the plant response to changes in recirculation flow, and to set the 30% and 60% flow limiters.

Descr...on

Various process variables will be recorded while changes are introduced into the reci..ulation flow control system.

Acceptance Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to flow control changes.

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to flow changes when the plant is operating above the lower limit setting of the master flow controller.

Results

Various tests were performed as a result of the Digital Feedwater Control System that was installed under Mod 1843, see Section 2.16 Feedwater System for a list of those tests. The Recirc pump 30% speed limiters were set on January 1, 1992 (RT 3.12). The 60% speed limiters were set as per RT 3.13 on January 17, 1992. The High Speed Stops were set as per RT 3.14 on January 28, 1992.

## 2-19 Recirculation System

Objectives

To calibrate the jet pump flow instrumentation.

Description

The jet pump instrumentation will be calibrated to read total core flow.

Acceptance Criteria

Level 1

None

Level 2

Flow instrumentation has been calibrated such that the reactor jet pump total flow recorder provides correct flow indication.

Results

Jet Pump Operability was checked during the performance of Startup Procedure GP-2, and documented in ST-0-02F-550-3 (Jet Pump Operability). Jet pump calibration was verified at 100% Core Thermal Power in RT-R-02F-250-3 (Core Flow Calibration Verification U/3) dated January 27, 1992.