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PEACH BOTTOM—THE POWER OF EXCELLENCE

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April 2, 1990
Docket Nos. 50-278
License No. DPR-56

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

SUBJECT: Peach Bottom Atomic Power Station Unit No. 3
Report of Plant Startup Following Seventh Refueling Outage

Dear Sir:

Attached is the Peach Bottom Atomic Power Station Unit No. 3 Report of Plant Startup Following Seventh Refueling Outage. The report is submitted pursuant to reporting requirement 6.9.1.a in Appendix A to License No. DPR 56.

Sincerely,

DMS/TEC/MJB:gh
MJB
TEC

Enclosures

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PHILADELPHIA ELECTRIC COMPANY
Peach Bottom Atomic Power Station
Unit No. 3
Docket Number 50-278

REPORT OF PLANT STARTUP FOLLOWING SEVENTH REFUELING OUTAGE

March 31, 1987
TO
February 2, 1990

SUBMITTED TO
THE UNITED STATES NUCLEAR REGULATORY COMMISSION
PURSUANT TO
FACILITY OPERATING LICENSEE DPR-56

Preparation Directed By:
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INTRODUCTION

Unit 3 Peach Bottom Technical Specification Section 6.9.1.a Routine Reports requires submittal of a Startup Report following an outage in which changes that may effect Nuclear Safety have occurred.

During the Seventh Refueling Outage (RFO #7):

Plant Modification 1536, Recirculation System, RHR System and RWCU System Piping Replacement, was performed.

Some GE fuel of a newer design was loaded into the core.

This report summarizes the Plant Startup and Power Ascension testing performed to assure that no conditions or system characteristic changes have been created by Unit 3's RFO #7 which diminish the safe operation of the plant.

SUMMARY

Startup Testing was performed in accordance with the Final Safety Analysis Report (FSAR) Section 13.5 Startup and Power Test Program. Measured and calculated values of operating conditions and characteristics obtained during the Startup Test Program were compared to design predictions and specifications. Level 1 criterion were either met, or discrepancies were investigated and determined to have no effect on safety, reliability, operability and pressure integrity of the systems tested.

Peach Bottom Unit 3 was out of service from March 31, 1987 to November 19, 1989 to accommodate the Seventh Refueling Outage (RFO-7), the Recirc Pipe replacement, maintenance and the NRC Shutdown Order.

During this 964 day outage:

*192 P8X8R - P8DRB299 fuel bundles were replaced with 48 GE8B - P8DQB319 and 144 GE8B - P8DQB321 fuel bundles.

*196 Unit 3 Modifications were completed.

The Unit commenced control rod withdraw and reached criticality on November 20, 1989.

The Unit returned to commercial operation on December 11, 1989 and reached full power on January 5, 1990. Startup Testing was completed on February 2, 1990.

The successfully implemented Startup Test Program insures that Unit 3 RFO #7 has resulted in no conditions or system characteristics that diminish the safe operation of the plant.

The tests and data referenced in this report are on file at the Peach Bottom Atomic Power Station.

STARTUP PROGRAM
Peach Bottom Atomic Power Station
Unit No. 3

1. Chemical and Radiochemical

Chemical and Radiochemical analyses were performed in accordance with FSAR Section 13.5.2.2.(1):

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 December 1, 1989 and December 12, 1989. 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 7.2.3B (Reactor Startup Chemistry [<100Klbs/hr]) on November 24, 1989. For high steaming rates [>100 Klbs/hr] ST 7.2.3A (Reactor Startup Chemistry) was performed on December 12, 1989.

2. Radiation Measurements

Radiation measurements were made in accordance with FSAR Section 13.5.2.2.(2):

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 at 20% Power on December 12, 1989, via ST 7.9.1 (Radiation Survey After Refueling).

3. Fuel Loading

Fuel loading, Control Rod Functional and Subcriticality Checks were performed in accordance with FSAR Section 13.5.2.2.(3). Fuel loading was completed on September 20, 1989 via FH-6C (Fuel Movement and Core

Alteration Procedure During a Fuel Handling Outage). Bundle locations and orientation were verified via ST 12.10 (Core Post-Alteration Verification) and completed on October 24, 1989. 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 November 20, 1989. Control Rod Test data is documented in ST 10.8 (Control Rod Performance Test) completed November 15, 1989.

4. Shutdown Margin

Core shutdown margin was demonstrated in accordance with FSAR Section 13.5.2.2.(4). An "in-sequence" shutdown margin of 2.491 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 3.8.3 (Shutdown Margin) completed November 20, 1989.

The design predicted core Keff was compared to the measured value at initial startup on November 20, 1989. The predicted Keff was 1.00469 as compared to the measured Keff of 1.0064. The difference between predicted and measured values was -0.171%, which meets the acceptance criteria of ±1%. The test data is documented in ST 3.9 (Critical Eigenvalue Comparison) completed November 20, 1989.

5. Control Rod Drives

Control Rod Drive (CRD) testing was performed in accordance with FSAR Section 13.5.2.2.(5). In cold shutdown, each CRD was tested for position indication, normal insert/withdrawal times and coupling via ST 10.8 (Control Rod Withdrawal Tests). At rated reactor pressure, Position Indication (GP-2 Normal Plant Startup), Coupling Checks (ST 10.8-1 CRD Coupling Integrity Test), and Scram Insertion Times (ST 10.13 CRD Scram Insertion Timing of Selected Control Rods) were tested. The testing performed at cold shutdown conditions satisfied Level 1 and 2 criteria.

6. Control Rod Sequence

The control rod sequence was followed in accordance with FSAR Section 13.5.2.2.(6). 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 10.5-1 (RWM Sequence Loading Verification) on November 18, 1989. ST 10.5 (RWM Operability Check) was performed and ST 3.8.3 (Shutdown Margin) recorded the critical rod pattern on November 20, 1989.

7. Rod Pattern Exchange

Rod pattern adjustments were performed in accordance with FSAR Section 13.5.2.2.(7). Rod pattern adjustments were guided by RE-31 (Reactor Engineering Startup/Load Drop Instructions) throughout the Power Ascension Program.

Due to the Unit 3 Cycle 7 not being fully completed the Cycle 8 loading resulted in a more reactive core than is usual. Consequently, thermal limits were higher than normally expected at beginning of cycle. To provide continuity throughout the operating shifts, and to ensure thermal limit integrity, an administrative guideline of 0.995 was established for the Core Maximum fraction of limiting power density (CMFLPD).

8. SRM Performance

Source Range Monitor (SRM) instrumentation operability was checked during performance of startup procedure GP-2. FSAR Section 13.5.2.2.(8) 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 November 17, 1989.

9. IRM Performance

Intermediate Range Monitor (IRM) performance was tested in accordance with FSAR Section 13.5.2.2.(9). 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 November 19, 1989.

10. LPRM Calibration

Local Power Range Monitor (LPRM) calibrations were completed in accordance with FSAR Section 13.5.2.2.(10). Calibrations were performed at 31% rated thermal power on December 18, 1989 and at 100% rated thermal power on January 16, 1990 per ST 3.4.1 (LPRM Gain Calibration).

11. APRM Calibration

Numerous Average Power Range Monitor (APRM) calibrations were completed during Power Ascension in accordance with FSAR Section 13.5.2.2(11). Test data is documented in ST 3.3.2's completed from December 14, 1989 at 30% rated thermal power to February 1, 1990 at 100% rated thermal power.

12. Process Computer

The Process Computer was tested in accordance with FSAR Section 13.5.2.2.(12). A manual calculation was performed via ST 3.11 (checkout of the NSS Computer Calculation of Core Thermal Power) at approximately 100% power on January 16, 1990.

The Thermal Limit calculations were verified by General Electric via BUCKLE with full-power data provided by the Process Computer.

Computer operational problems were experienced during portions of the Power Ascension and are being tracked under Unit 3 Lessons Learned - PORC Open Items, to determine the cause, evaluate its effects and correct or repair as necessary.

13. Reactor Core Isolation Cooling (RCIC) System

The RCIC System was tested in accordance with FSAR Section 13.5.2.2.(13). A controlled start was performed at 150 psig via ST 10.2 (RCIC Flow Rate at 150 psig) on November 26, 1989. A Quick start at rated pressure was performed via ST 6.11F-3 (RCIC Pump, Valve, Flow and Cooler) on December 7, 1989. RCIC Controller Stability was checked by RT 8.18 (RCIC Flow Controller Stability) at 150 psig on November 26, 1989 and at 920 psig on December 7, 1989. No adjustments were required.

14. High Pressure Coolant Injection (HPCI) System

The HPCI System was tested in accordance with FSAR Section 13.5.2.2(14). A controlled start was performed at 150 psig via ST 10.1 (HPCI Flow Rate at 150 psig) on November 26, 1989. A quick start at rated pressure was performed via ST 6.5F-3 (HPCI Pump, Valve, Flow and Cooler) on December 8, 1989. HPCI Controller Stability was checked by RT 8.17 (HPCI Flow Controller Stability) at 150 psig on November 26, 1989 and at 920 psig on December 7, 1989. No adjustments were required.

15. Recirc Pump Trip Process Temperatures

Recirc Pump Trip Process Temperatures were tested in accordance with FSAR Section 13.5.2.2.(15). Temperature data during each Recirc Pump trip and restart was taken during the performance of MAT 1536M (Recirc Pump Trips at 75% Power). Level 1 acceptance criteria was met and is documented in MAT 1536P (Recirc Pump Trip Process Temps) dated December 30, 1989 for the "A"Recirc Pump and December 31, 1989 for the "B" Recirc Pump.

16. System Expansion

System expansion was determined in accordance with FSAR Section 13.5.2.2.(16). Pipe expansion data was taken on applicable portions of the Recirc, RHR and RWCU systems.

AT 160 psig:

(1) A level 1 violation was reported on the "A" RHR Discharge piping based on vibration data. The data was sent to GE Engineering for review and analysis. GE issued FDDR# HE-3-0544 Rev. 0, stating that the pipe growth due to thermal expansion was within their analytical limits and was acceptable.

(2) A visual inspection of the Drywell found that a cable feed terminal box mounted on the side of the "B" Recirc pump motor was making contact with a portion of the floor grating. A MRF (#8910262) was issued and the impacting portion of the floor grating was removed.

At 920 psig:

(1) 3 Level 1 violations and 19 Level 2 violations were reported. The data was sent to GE Engineering for review and analysis. At the same time the station wrote Nonconformance Report #P89986. The disposition of the NRC was to accept the recommendations of the GE Piping Engineer. GE issued FDDR# HE-3-0545 Rev. 0, stating that the thermal expansion would not create thermal stressed beyond the ASME III stress limits and to accept as is. GE's justification was that no safety, reliability and pressure integrity problems are created by accepting the above violations. The plant can continue operation.

17. Core Power Distribution

Core power symmetry and Transversing Incore Probe (TIP) reproducibility were tested in accordance with FSAR Section 13.5.2.2.(17). Two full sets of TIP traces were obtained at rated core thermal power (100%) via ST 3.4.1 (LPRM Gain Calibration) and RE-27 (Peach Bottom 2 & 3 Core Power Symmetry and TIP Reproducibility) over January 16 & 17, 1990. 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.

18. Core Performance

Core performance was evaluated in accordance with FSAR (Section 17.5.2.2.(18). The core thermal limits were verified daily above 25% power via the Process Computer. ST 3.7-2 (Reactor Anomalies) verified the full Power Control Rod Pattern provided by the PECO Fuel Management Section and General Electric and was completed on January 17, 1990.

19. Pressure Regulator

Pressure Regulator Control response was verified in accordance with FSAR Section 13.5.2.2.(21). At 20% core Thermal Power, positive and negative step changes of 3 psi and 5 psi were introduced into each pressure regulator control circuit. Decay ratios were less than 0.25 and met both FSAR Level 1 & 2 criteria. Test data is documented in RT 8.15 (Pressure Regulator Stability Test) dated December 12, 1989.

20. Feedwater System

Feedwater Controller Stability testing was performed in accordance with FSAR Section 13.5.2.2.(22) 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 at 50% power (12/16/39), 67% power (12/20/89), 83% power (01/04/90) and 100% power (01/09/90). The overall Feedwater Control system tested in three element mode displayed satisfactory system response. Feedwater test data is documented in RT 8.16 (Feedwater Controller Stability) completed on the dates indicated above.

Individual Reactor Recirc Pump trips were performed at 75% Core Thermal Power, 100% Core Flow. The Feedwater Control System satisfactorily controlled the water level, avoiding a Turbine Trip on high water level. Test data is documented in MAT 1536M (Recirc Pump 75% Power Trips) dated December 30 & 31, 1989.

21. Relief Valves

Relief Valves were tested in accordance with FSAR Section 13.5.2.2.(25). Each Safety Relief Valve (SRV) was manually cycled at 168 psig Reactor Pressure. Test data is documented in ST 10.4 (Relief Valve Manual Actuation) dated November 28, 1989.

22. Flow Control

Plant response to changes in recirculation flow was tested in accordance with FSAR Section 13.5.2.2.(28). At 69% Core Thermal Power while preparing to perform RT 8.19 (Recirculation Controller Stability), the "C" Reactor Feed Pump tripped due to a coupling failure. Consequently, a Recirc Runback occurred and all Feedwater Flow was picked up by the "B" Reactor Feed Pump for approximately 10 minutes until the plant had stabilized. After analyzing the data, it was determined that this transient was more severe than the 10% flow step required by RT 8.19. The decay ratios on the transient data were less than 0.25 for oscilatory variables, and met both FSAR Level 1 & 2 criteria. The transient data is documented in the Peach Bottom Unit 3 Restart Testing Final Summary Report submitted by GE.

23. Recirculation System

The Recirculation system was tested in accordance with FSAR Section 13.5.2.2(29). Each Recirc Pump was tripped at 75% Core Thermal Power, 100% core flow. This configuration was utilized in order to maximize the effect of the Recirc pump trip. Both pumps were initially operating at a nominal 82% speed. The decay ratios on the test data were less than 0.25 for oscillatory variables, and met both FSAR Level 1 & 2 criteria. Test data is documented in MAT 1536M (Recirc Pump 75% Power Trips) dated December 30 & 31, 1989.

Jet Pump Operability was checked during the performance of Startup Procedure GP-2, and documented in ST 9.21-3 (Jet Pump Operability) dated November 21, 1989. Jet pump calibration was verified at 100% Core Thermal Power in ST 13.30-2 (Core Flow Calibration Verification U/3) dated January 10, 1990.

24. Vibration

Recirc pipe vibration was measured in accordance with FSAR Section 13.5.2.2.(32). Vibration data was taken at 920 psi, 50%, 75% and 100% Core Thermal Powers. At 75% & 100% Core Thermal Powers Bi-Stable flow was exhibited. The existence of Bi-Stable does not limit reactor operation in any way. This data is documented in the Peach Bottom Unit 3 Restart Testing Final Summary Report submitted by General Electric.