CCN 93-14044

PHILADELPHIA ELECTRIC COMPANY

PEAC. BOTTOM ATOMIC POWER STATION

R.D. 1, Box 208

Delta, Pennsylvania 17314

PEACH BOTTOM—THE POWER OF EXCELLENCE

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Vice President

April 12, 1993

Docket Nos.: 50-277
License No. DPR-44

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Peach Bottom Atomic Power Station Unit No. 2
Report of Plant Startup Following Ninth Refueling Outage

Gentlemen:

Enclosed is the Peach Bottom Atomic Power Station Unit No. 2 Report of Plant Startup Following Ninth Refueling Outage. The report is submitted persuant to reporting requirement 6.9.1.a in Appendix A to License No. DPR-44.

Sincerely,

DBM\MDM\DJF:aus
Enclosure

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PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION
UNIT NO. 2
DOCKET NUMBER 50-277

CYCLE 10
STARTUP REPORT

SUBMITTED TO
THE U.S. NUCLEAR REGULATORY COMMISSION
PURSUANT TO
FACILITY OPERATING LICENSE DPR-44

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APRIL 5, 1993

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INTRODUCTION

PBAPS Unit 2 Technical Specifications section 6.9.1.a requires submittal of a Startup Report following an outage in which fuel of a different design was installed. This report summarizes the plant startup and power ascension testing performed to ensure that no operating conditions or system characteristic changes occurred during the ninth refueling outage of Unit 2 which diminished the safe operation of the plant.

Startup testing was performed in accordance with the Updated Final Safety Analysis Report (UFSAR) section 13.5 "Startup and Power Test Program". This report will address each of the applicable tests identified in UFSAR section 13.5.2.2. UFSAR tests that were only required to be performed during the initial plant startup (Cycle 1) are not included in this report. A description of the measured values of the operating conditions or characteristics obtained during startup testing and a comparison of these values with design predictions and specifications will also be included in this report.

Level 1 and Level 2 test acceptance criteria are described in UFSAR section 13.5.2.1. For each applicable test identified in UFSAR section 13.5.5.2, all Level 1 criteria were met, and all Level 2 criteria were either met, or discrepancies were investigated and determined to have no effect on safety, reliability, operability, and pressure integrity of the systems tested. Any corrective actions that were required to obtain satisfactory operation will also be described.

Peach Bottom Unit 2 was out of service from 9-12-92 to 12-10-92 to accommodate its ninth refueling outage. During this 90 day outage, 41 non-barrier fuel bundles of the P8DRB design and 227 barrier fuel bundles of the GE7B design were replaced with 268 barrier fuel bundles of the GE11 design. In addition to the use of ferrule spacers, thin-walled channels with flow trippers, and axial fuel enrichment and gadolinia loading, GE11 fuel includes the following new design features:

* 9x9 assembly configuration
* 2 large central water rods
* 8 part length rods (PLRs), terminating just past the top of the fifth spacer

The 9x9 assembly configuration allows operation with lower fuel pin powers and therefore leads to improvement in LHGR margin.

The two large central water rod design improves neutron thermalization, critical power, and pressure drop performance.

The 8 part length rods are selectively located in the lattice to reduce the 2 phase pressure drop across the bundle. Because of this large 2 phase pressure drop reduction, it is possible to increase the single phase pressure drop across the lower tie plate, and thus achieve additional core and channel stability benefits. The PLRs also increase the moderator-to-fuel ratio in the top of the core in the cold state, which significantly improves cold shutdown margin.

Many of the GE11 fuel design features, such as increased pre-pressurization (10 atm. He gas), are key to providing overall burnup capability. This allows for improved fuel cycle economics, greater flexibility, and longer cycle length capability.

Major accomplishments during the outage included:

- Completion of 27 Unit 2 & Common modifications
- Performance of approximately 5100 corrective and preventative maintenance tasks
- Replacement of 4 feedwater heaters
- RHR pump impeller and motor replacement
- 160 LLRTS performed
- Digital Feedwater Control System Installation
- Torus hardened vent modification
- Replacement of 33 control rod drives
- Replacement of 19 control rod blades
- Replacement of 14 LPRM strings
- 62 control rod drive HCUs rebuilt
- Bottom head drain inspection/cleaning

Unit 2 returned to service on 12-10-92 and reached full power for the first time in Cycle 10 on 1-18-93. Startup testing was completed on 4-2-93.

The successfully implemented startup test program ensures that the ninth refueling outage of Unit 2 has resulted in no conditions or system characteristics that in any way diminish the safe operation of the plant.

This report is required to be submitted within 90 days following resumption of commercial full power operation. All tests and data referenced in this report are on file at Peach Bottom Atomic Power Station.

2.1 Chemical and Radiochemical

Objectives

Chemical and radiochemical analyses were performed in accordance with UFSAR section 13.5.2.2.(1). The objectives of these analyses were: (1) to maintain control of and knowledge about the reactor water chemistry, and (2) to determine that the sampling equipment, procedures, and analytic techniques are adequate to demonstrate that the coolant chemistry meets water quality specifications and process requirements. In addition, this testing also allowed evaluations to be made of fuel performance, filter demineralizer operation, condenser integrity, offgas system operation, and calibration of certain process instruments.

Description

During the refueling outage and subsequent startup and power ascension, samples were taken and measurements were made to determine the chemical and radiochemical quality of the reactor water, feedwater, amount of radiolytic gas in the steam, gaseous activities leaving the air ejectors, delay times in the offgas lines, and performance of filters and demineralizers. Calibrations were also made of monitors in the stack, liquid waste system, and liquid process lines.

Acceptance Criteria

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 must conform to license limitations. Chemical factors defined in the Technical Specifications must be maintained within those limits specified.

Results

Prior to and during core alterations, chemistry values were verified to be within daily limits per CH-10 "Chemistry Goals".

Prior to startup, chemistry requirements were verified by RT 7.8 "Chemistry Preparation for Reactor Startup" on 11-29-92. The Shift Chemist also verified that reactor water dose equivalent I-131, chloride concentration, and sulfate concentration were within specification per CH-10 on 12-3-92.

During power ascension, coolant chemistry was verified to meet water quality specifications and process requirements by ST-C-095-824-2 "Reactor Startup Chemistry With Steaming Rates Less Than 100,000 Lbs/Hr", performed on 12-5-92.

At high steaming rates, ST-C-095-823-2 "Conductivity and Chloride Ion Content in Primary Coolant During Normal Operation" was performed at least every 4 days after reaching 850 psig reactor pressure. This test verified that the conductivity was less than or equal to 5 umhos/cm and the chloride concentration was less than or equal to 200 ppb in all samples.

Gaseous and liquid effluent activities were checked by Chemistry Department surveillance tests and round sheets. The chemistry values required by the Technical Specifications were checked daily in accordance with CH-10 and were verified to be within the specified limits. Gaseous and particulate release dose rates from the main stack and roof vents were checked weekly in accordance with ST-C-095-857-2, ST-C-095-859-2, and ST-C-095-860-2.

Condensate filter demineralizers were backwashed and precoated based on Chemistry recommendations.

The Offgas system was placed in service on 12-8-92. The steam jet air ejector discharge activity indicated that Unit 2 was started up with no fuel failures. Subsequent analysis of chemistry samples using a fuel reliability code confirmed that no fuel failures exist.

Radiation monitors and chemistry sampling equipment were also calibrated during power ascension for the main offgas stack, liquid waste system, and liquid process lines.

2.2 Radiation Measurements

Objectives

Radiation measurements were performed in accordance with UFSAR section 13.5.2.2.(2). The objectives of these measurements were to determine the background gamma and neutron radiation levels in the plant and to monitor radiation levels during power ascension to assure protection of personnel and continuous compliance with 10CFR20 requirements.

Description

A survey of natural background radiation throughout the plant site will be made. During the refueling outage, startup, and power ascension, gamma radiation measurements and neutron dose rate measurements (where appropriate) will be made at significant locations throughout the plant. All potentially high radiation areas will be surveyed.

Acceptance Criteria

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

Routine surveys were performed throughout the protected area in accordance with HP 200 "Routine Survey Program" to determine background radiation levels and assure personnel safety.

The initial survey of the drywell was performed per HP-315 on 9-13-92. During the refueling outage and subsequent plant startup, appropriate radiation surveys were performed to generate Radiation Work Permits per HP-310 and properly post plant radiation areas per HP-215 to maintain compliance with 10CFR20 requirements.

During the refueling outage, several plant areas were continuously manned by Health Physics Personnel. These areas included the Refuel Floor, Drywell Access, and Personnel Access areas.

During the refueling outage, workers received 268.69 man-rem of exposure.

2.3 Fuel Loading

Objective

Fuel loading was performed in accordance with UFSAR section 13.5.2.2(3). The objective was to load new fuel and shuffle the existing fuel safely and efficiently to the final loading pattern.

Description

During fuel movement activities, all control rods must be fully inserted. At least 2 SRMs must be operable, one in the quadrant that fuel movement is being performed in, and one in an adjacent quadrant. Each fuel bundle must remain neutronically coupled to an operable SRM at all times. SRM count rates will be recorded before and after each core component move.

Each control rod will be functionally tested by being completely withdrawn and reinserted. A subcriticality check will be performed by verifying that the core remains subcritical when any single rod is fully withdrawn and all other rods are fully inserted.

Acceptance Criteria

The core is fully loaded in its final loading pattern and the core shutdown margin demonstration has been completed.

Results

The fuel shuffle was performed in accordance with FH-6C "Core Component Movement - Core Transfers" and was completed on 10-24-92. The final loading pattern includes 268 new GE11 fuel bundles, 12 once-burned Qualification Fuel Bundles (4 GE11, 4 ABB/SVEA96, & 4 ANF9X), 152 GE9B bundles, 268 GE8B bundles, and 64 GE7B bundles. Except for the ABB and ANF QFBs, the complete Cycle 10 core consists of barrier fuel.

Fuel bundle serial numbers, core locations, orientations, and seating positions were verified in accordance with RT-R-004-970-2 "Core Verification", completed on 10-26-92.

Each control rod was withdrawn and inserted to verify coupling integrity, position indication, proper rod withdrawal and insertion speeds, and core subcriticality. This test data is documented in ST-O-003-465-2 "Control Rod Withdraw Tests", completed on 12-1-92.

The acceptance criteria for this test was met when the actual shutdown margin was demonstrated with a fully loaded core in accordance with ST-R-002-910-2, performed on 12-5-92.

2.4 Shutdown Margin

Objective

Core shutdown margin was demonstrated in accordance with UFSAR section 13.5.2.2.(4). The objective of this test is to demonstrate that the reactor will be subcritical throughout the fuel cycle with any single control rod fully withdrawn.

Description

Core shutdown margin was demonstrated with the "In-Sequence Critical" method. At criticality, correction factors were applied for moderator temperature, reactor period, worth of the "strongest" rod, and the "R" value for the cycle.

Acceptance Criteria

The fully loaded core must be subcritical by at least 0.38% delta K/K throughout the fuel cycle with any single control rod fully withdrawn.

Results

Core shutdown margin was demonstrated by performing ST-R-002-910-2 "Shutdown Margin (Unit 2- Cycle 10)" on 12-5-92. Control rods were withdrawn according to the startup sequence. SRM count rates were recorded after each control rod withdrawal. The reactor was declared critical at 0706 on 12-5-92 on step 43 of the RWM startup sequence (rod 34-39 @ position 12). Reactor water temperature was 105 degrees F, count rate doubling time was 100 seconds, and the calculated reactor period was 144 seconds.

The BOC SDM value was calculated by subtracting the worth of the analytically determined strongest rod from the worth of all withdrawn rods and then applying the temperature and period correction factors. This calculated SDM value was equal to 2.348% delta K/K. This value was verified to be greater than (.38% delta K/K + R), which is (.38% + .984%) delta K/K or 1.364% delta K/K.

To allow a minimum reactor water temperature of 38 degrees F throughout cycle 10, an additional SDM adder of .15% delta K/K was applied, which required the calculated SDM value to exceed 1.514% delta K/K. The calculated SDM value exceeded 1.514% delta K/K by (2.348 - 1.514)% delta K/K, or 0.834% delta K/K. The test data was sent to Fuel Management Section for analysis using a licensed three dimensional core physics code. The BOC SDM value calculated by Fuel Management Section was 2.343% delta K/K.

The design predicted Keff value was compared to the actual value calculated at initial criticality in accordance with ST-R-002-920-2 "Critical Eigenvalue Comparison". Using the shutdown margin test data, the predicted Keff value was 1.0019 and the actual Keff value was calculated to be 1.0027. Therefore, the difference between the predicted and the actual Keff values was -0.08% delta K, which meets the acceptance criteria of +/- 1% delta K.

2.5 Control Rod Drives

Objectives

Control rod drive testing was performed in accordance with UFSAR section 13.5.2.2.(5). The objectives of this testing were to demonstrate that the CRD system operates properly over the full range of primary coolant temperatures and pressures and that thermal expansion of core components does not bind or significantly slow the control rod movements.

Description

The CRD system was tested at rated reactor pressure to verify that there was no significant binding caused by thermal expansion of core components. The withdraw and insert speeds were checked for each control rod, and each rod was individually scram-timed at rated reactor pressure.

Acceptance Criteria

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 foot stroke in 40 to 60 seconds.

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	FSAR Insertion Time (sec)	T.S. Adjusted Insertion Time (sec)
5	0.375	.359
20	0.900	.920
50	2.000	1.990
90	5.000	3.670

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	FSAR Insertion Time (sec)	T.S. Adjusted Insertion Time (sec)
5	0.398	.382
20	0.950	.974
50	2.120	2.110
90	5.300	3.970

Note: Scram time is measured from time pilot scram valve solenoids are de-energized.

Results

Each CRD had its normal insert speeds, withdraw speeds, and coupling integrity checked by ST-O-003-465-2 "Control Rod Withdraw Tests", completed on 12-1-92. All insert and withdraw speeds fell within the acceptance criteria of 45-51 sec/ full stroke. This test also checked CRD stall flows and rod position indication, and verified core subcriticality.

Prior to startup, during the RPV pressure test, each CRD was scram timed in accordance with ST-R-003-460-2 "CRD Scram Insertion Timing, Full In and Full Out Position Indication Check, and Rod Coupling Integrity Check for All Operable Control Rods", completed on 11-19-92. All 185 rods met with two-by-two array insertion time criteria, but six control rods were retested per ST-R-003-485-2 "CRD Scram Insertion Timing of Selected Control Rods", on 11-20-92 in order to reverify their individual insertion times. Subsequent HCU valve maintenance was performed on rods 10-35 and 30-11. In addition, rod 46-15 was required to be retested due to a position indication problem. These three control rods were retested during power ascension at 40% power in accordance with ST-R-003-485-2, performed on 12-14-92.

During the refueling outage, ST 9.2.2 "Control Rod Exercise when the Reactor is in Refuel" was performed weekly while the Reactor Mode Switch was in Refuel. This test required a full stroke of each operable, coupled control rod. Drive pressures and withdraw stall flows were recorded for each rod.

During power ascension, ST-O-003-470-2 "CRD Coupling Integrity Test" was performed to verify coupling integrity, full-out position indication, and neutron response for each control rod. This test was completed on 12-20-92.

During power ascension, when reactor power was above the RWM LPSP (approximately 20%), ST-O-003-560-2 "Control Rod Exercise" was performed weekly. This test required each fully or partially withdrawn rod to be inserted and withdrawn one notch.

2.6 Control Rod Sequence

Objectives

Control rod sequence testing was performed in accordance with UFSAR section 13.5.2.2(6). The objectives of this testing were to achieve criticality in a safe and efficient manner using the approved rod withdrawal sequence, and to determine the effect on reactor power of control rod motion at various operating conditions.

Description

The approved rod withdrawal sequence implements the BPWS (Banked Position Withdrawal Sequence) methodology and CCC (Control Cell Core) operation. This sequence is contained in GP-2-2 Appendix 1 (Startup Rod Withdrawal Sequence Instructions), which is used by Operations personnel when rod movement is enforced by the RWM.

At power levels below the RWM LPSP, the RWM will prevent an out of sequence rod withdrawal and will not allow more than two rods to be inserted out of sequence. The GP-2-2 Appendix 1 sequence is programmed into the RWM and is designated as "Startup 1". This sequence specifies rod withdrawal from the all-rods-in condition to the rod pattern in which all CCC rods are fully inserted and all other rods are fully withdrawn. Rod withdrawals beyond this pattern are governed by RE-31 "Reactor Engineering Startup/Load Drop Instructions".

Results

Cold criticality was achieved on 12-5-92 by withdrawing rods in accordance with GP-2-2 Appendix 1. This same sequence (Startup 1) had previously been verified in the RWM in accordance with ST-R-62A-220-2 "RWM Sequence Verification", performed on 12-1-92. Prior to withdrawing the first rod, ST-O-62A-210-2 "RWM Operability Check" was performed on 12-5-92. Criticality occurred on sequence step 43 in RWM group 2. The critical rod pattern is recorded in GP-2-2 Appendix 1 and ST-R-002-910-2 "Shutdown Margin (Unit 2 Cycle 10)".

2.7 Rod Pattern Exchange

Objective

A rod pattern exchange was performed in accordance with UFSAR section 13.5.2.2.(7). The objective was to perform a representative change in basic rod pattern at a reasonably high reactor power level.

Description

The control rod pattern was adjusted by rod withdrawals in a planned sequence in order to ultimately achieve the full power target rod pattern. An intermediate rod pattern (short shallow) was used as an interim step to the final rod pattern. The short shallow and final rod patterns were developed by running cases on a licensed 3-dimensional core physics code.

Acceptance Criteria

The achievement of the final target rod pattern by the use of the intermediate short shallow rod patterns while staying within licensed core limits meets the requirements of this test.

Results

The short shallow rod pattern was developed for a reactor statepoint of 60% power and 55% core flow. This rod pattern was achieved on 12-15-92. The full power target rod pattern was developed for a reactor statepoint of 100% power and 100% core flow. This rod pattern was achieved on 1-18-93. Rod movements were governed by RE-31 "Reactor Engineering Startup/Load Drop Instructions".

Core thermal power, rated core flow and core thermal limits were not exceeded at any time during the power ascension.

2.8 SRM Performance

Objective

SRM performance was monitored in accordance with UFSAR section 13.5.2.2.(8). The objective was to demonstrate that SRM instrumentation provided adequate information to the operator during startup.

Description

Source Range Monitor count rate data was taken during rod withdrawals to criticality and was compared with stated operability criteria.

Acceptance Criteria

There must be a neutron signal-to-noise ratio of at least 2 to 1 on the required operable SRMs, and a minimum count rate of 3 counts per second on the required operable SRMs.

Results

SRM operability was verified daily during the outage by performing ST-O-60D-250-2 "SRM Operability and Neutron Response Check". On 9-21-92, SRM D failed due to a detector short. The detector was replaced by 9-22-92.

Prior to startup, SRM performance was verified by SI2N-60D-SRM-A2CZ "SRM Channel A Calibration/Functional Check" and SI2N-60D-SRM-B2CZ "SRM Channel B Calibration/Functional Check".

Minimum SRM count rate was determined to be greater than 3 CPS prior to control rod withdrawal on 12-5-92. The signal-to-noise ratio check is only required to be performed in accordance with Tech Specs if the SRM count rate is less than 3.0 CPS. Since the SRM count rate was never less than 3.0 CPS at any time during the startup, this verification was not performed.

During startup, SRM operability was verified in accordance with GP-2 "Normal Plant Startup". All 4 SRMs were operable for the initial BOC startup. SRM count rate data following each rod withdrawal to criticality was recorded in ST-R-002-910-2.

2.9 IRM Performance

Objective

IRM performance was monitored in accordance with UFSAR section 13.5.2.2.(9). The objective was to adjust the IRMs to obtain a optimum overlap with the SRMs and APRMs.

Description

IRM calibration and functional checks were performed to ensure adequate overlap with the SRMs and APRMs. In addition, IRM response was monitored during startup in accordance with GP-2 to assure that the IRMs were properly indicating the increasing neutron flux levels during the power ascension.

Acceptance Criteria

Each IRM channel must be adjusted so the overlap with the SRMs and APRMs is assured. The IRMs must produce a scram signal at 120 on a full scale of 125.

Results

During the outage, cables and connectors were replaced for IRMs C, D, E, and F on 11-27-92. Prior to startup, IRM performance was tested and the scram setpoints were verified by performing SI2N-60C-IRM-A(B)4CZ "Intermediate Range Monitor Channel A(B) Calibration /Functional Check" on 12-1-92 (for both the "A" and "B" channels). All 8 IRMs were operable for the initial BOC startup.

During startup, SRM/IRM overlap was verified in accordance with GP-2, and all IRMs were verified to have on-scale increasing indication prior to switching to Range 4 on any IRM.

In addition, proper IRM response to power increases was verified in accordance with GP-2 during power ascension.

Prior to withdrawing the IRMs, all APRM downscale lights were verified to be cleared prior to exceeding the scram setpoint of 120/125 IRM scale. This verified proper IRM/APRM overlap.

2.10 LPRM Calibration

Objective

To calibrate the Local Power Range Monitor (LPRM) system in accordance with UFSAR section 13.5.2.2.(10).

Description

The LPRM channels were calibrated to make the LPRM readings proportional to the neutron flux in the narrow-narrow water gap at the LPRM detector elevation. Calibration and gain adjustment information was obtained by using the Plant Monitoring System to relate the LPRM reading to the average fuel assembly power at the detector location.

Acceptance Criteria

With the reactor in the rod pattern and at the power level which the calibration is to be performed, the LPRM meter readings will be proportional to the average flux in the four adjacent fuel assemblies at the LPRM detector elevation.

Results

ST-R-60A-230-2 "LPRM Gain Calibration" was performed twice during power ascension. The first calibration was performed on 12-14-92 at 27% power and the final calibration was performed on 1-26-93 at 100% power. The Gain Adjustment Factor (GAF) acceptance criteria in the test ensured that the LPRM detectors were adjusted to be proportional to the neutron flux at the detector locations.

2.11 APRM Calibration

Objective

To calibrate the Average Power Range Monitor (APRM) system in accordance with UFSAR section 13.5.2.2.(11).

Description

During power ascension, the APRM channel readings were adjusted to be consistent with core thermal power as determined from the Plant Monitoring System heat balance. When required by Technical Specifications, the APRM channel readings were set higher than core thermal power, and were set equal to the leading MFLPD thermal limit value in the core.

Acceptance Criteria

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

Results

Prior to startup, the following tests were verified to be within surveillance per GP-2 on 12-2-92:

* SI2N-60A-APRM-A1CM(thru F1CM) "Average Power Range Monitor Calibration/Functional Check"

* SI2N-60A-APRM-A(B)3FW "Average Power Range Monitor Channel A(B) Functional Check"

Numerous APRM calibrations were performed in accordance with ST-O-60A-210-2 "APRM System Calibration During Two Loop Operation" throughout power ascension. The first APRM gain calibration was performed on 12-8-92 at 8% power and the last APRM gain calibration was performed on 1-18-93 at 100% power.

All 6 APRMs were operable for the initial BOC startup.

The APRM channels were calibrated to core thermal power or to the leading MFLPD value, whichever was greater, at all times during the power ascension.

2.12 Process Computer

Objective

The Plant Monitoring System (PMS) was tested in accordance with USFAR section 13.5.2.2.(12). The objective was to verify the performance of the PMS under operating conditions.

Description

During power ascension, the PMS provided NSSS and BOP process variable information to the reactor operator. The NSSS heat balance was verified to be correct and the BOC NSSS databank was installed and verified to be correct.

Acceptance Criteria

NSSS programs OD-1 and P-1 will be considered operational when the thermal limit values calculated by an independent method and the PMS are in the same fuel assembly and do not differ in value by more than 10%, and that the LPRM calibration factors calculated by an independent method and the PMS agree to within 5%. The remaining programs will be considered operational upon successful completion of static testing.

Results

GE11 core monitoring software was installed and tested prior to the refueling outage on 8-15-92 in accordance with NSSS04.TST. The BOC10 databank was installed and verified in accordance with RE-38 "Process Computer NSSS BOC Databank Update", performed on 12-1-92. During power ascension, the PMS heat balance was verified to be correct by performing RT-R-059-500-2 "Checkout of the NSSS Computer Calculation of Core Thermal Power" at approximately 100% power on 1-22-93. The PMS core monitoring output was compared with benchmark cases run on a licensed 3-dimensional core physics code.

Thermal limit calculations were also independently verified by General Electric using the BUCLE code with full power data obtained from the PMS. The BUCLE run was performed on 4-2-93 with PMS plant data obtained at 100% power steady state conditions. The P1 and BUCLE thermal limit values all agreed to within 1%. Actually, the MFLPD and MAPRAT bundles were identical in values and bundle locations, and the P1 MFLCPR value differed from the BUCLE value by 1% due to an administrative penalty applied due to slow scram times on some control rods. The LPRM calibrated readings on the P1 and BUCLE output were within 2%.

2.13 RCIC System

Objective

Reactor Core Isolation Cooling (RCIC) system testing was performed in accordance with UFSAR section 13.5.2.2.(13). The objective was to verify RCIC operation at various reactor pressures during the power ascension.

Description

A controlled start of the RCIC system will be done at a reactor pressure of 150 psig and a quick start will be done at a reactor pressure of 1000 psig. Proper operation of the RCIC system will be verified and the time required to reach rated flow will be determined. These tests will be performed with the system in test mode so that discharge flow will not be routed to the reactor pressure vessel.

Acceptance Criteria

The RCIC system must have the capability to deliver rated flow (600 gpm) in less than or equal to the rated actuation time (30 seconds) against rated reactor pressure.

Results

A controlled start was performed at 150 psig reactor pressure in accordance with ST-O-013-200-2 on 12-5-92. A cold quick start at rated reactor pressure was performed in accordance with ST-O-013-301-2 on 12-8-92.

The RCIC turbine did not trip off during the testing and rated flow was achieved in less than 30 seconds.

2.14 HPCI System

Objective

High Pressure Coolant Injection (HPCI) system testing was performed in accordance with UFSAR section 13.5.2.2.(14). The objective was to verify proper operation of the HPCI system throughout the range of reactor pressure conditions.

Description

Controlled starts of the HPCI system will be performed at reactor pressures near 150 psig and 1000 psig, and a quick start will be initiated at rated pressure. Proper operation of the HPCI system will be verified, the time required to reach rated flow will be determined, and any adjustments to the HPCI flow controller and HPCI turbine overspeed trip will be made. These tests will be performed with the system in test mode so that discharge flow will not be routed to the reactor pressure vessel.

Acceptance Criteria

The time from actuating signal to required flow must be less than 30 seconds with reactor pressure at 1000 psig. With HPCI and discharge pressure at 1220 psig, the flow should be at least 5000 gpm. The HPCI turbine must not trip off during startup.

Results

During the refueling outage, two modifications were made to the HPCI system. Mod 5204 changed the HPCI booster pump impeller from a 4-vane to a 5-vane impeller, and Mod 5344 added vibration monitoring equipment to several HPCI system components.

During the outage, the HPCI turbine overspeed test was performed (on aux steam from the boilers) on 11-19-92 in accordance with RT-X-023-240-2. A minor spring tension adjustment to the overspeed mechanism was required to be made.

During the startup, a controlled start was performed at 150 psig reactor pressure in accordance with ST-O-023-200-2 on 12-6-92. A cold quick start at rated pressure was performed in accordance with ST-O-023-301-2 on 12-9-92.

A minor adjustment to the flow controller ramp generator signal converter was made to decrease the ramp rate.

HPCI response time was checked in accordance with ST 6.5.R-2 on 1-2-93. Initial response time (30.2 sec) slightly exceeded the acceptance value of 30 seconds, but a check valve leak was repaired and the final response time was 23.7 seconds from actuating signal to rated flow. The HPCI turbine did not trip off during the testing.

2.15 Selected Process Temperatures

Objective

Selected temperatures were monitored in accordance with UFSAR section 13.5.2.2.(15). The objective was to ensure that the water temperature in the bottom head of the reactor vessel was within 145 degrees F of the steam dome saturation pressure prior to starting a second Recirc pump.

Description

The applicable reactor parameters were monitored during the power ascension and following Recirc pump trips in order to determine that adequate mixing of the reactor water was occurring in the lower plenum of the pressure vessel. This was done to ensure that thermal stratification of the reactor water was not occurring.

Acceptance Criteria

The second reactor Recirc pump shall not be started unless the coolant temperatures in the upper (steam dome) and lower (bottom head drain) regions of the reactor pressure vessel are within 145 degrees F of each other. The pump in the idle Recirc loop shall not be started unless the temperature of the coolant within the idle loop is within 50 degrees F of the active Recirc loop temperature.

Results

During power ascension, the 2B Recirc pump was manually tripped from 54% power on 12-16-92 in order to repair the M/G set Bailey positioner shaft bearing. The appropriate reactor vessel temperatures were monitored to ensure thermal stratification did not occur. After verifying that the dome-to-bottom head and loop-to-loop temperature criteria were met, the 2B Recirc pump was restarted later on 12-16-92 in accordance with SO 2A.1.B-2 "Starting the Second Recirculation Pump".

In order to support MAT 1843 digital FW controller testing, the 2A Recirc pump was also manually tripped from 95% power on 1-29-93. The same reactor vessel temperatures were monitored and trended, and the 2A Recirc pump was restarted later on 1-29-93 after verifying all of the temperature criteria given in SO 2A.1.B-2 were met.

2.16 System Expansion

Objective

System expansion inspections were performed in accordance with UFSAR section 13.5.2.2.(16). The objective was to verify that the reactor drywell piping system is free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner.

Description

An inspection of the horizontal and vertical movements of major equipment and piping in the nuclear steam supply system and auxiliary systems will be made to assure components are free to move as designed. Any adjustments necessary to assure freedom of movement will be made.

Acceptance Criteria

There shall be no evidence of blocking or the displacement of any system component caused by thermal expansion of the system. Hangers shall not be bottomed out or have the spring fully stretched.

Results

During the refueling outage, snubber inspections were performed in accordance with Tech Specs. A sample of pipe hangers were inspected in accordance with the ISI program.

During the RPV pressure test, drywell piping was visually inspected at 500 psi and 1000 psi. During startup, another drywell inspection was performed at 500 psi reactor pressure.

No blocking or interference of piping due to thermal expansion was observed.

2.17 Core Power Distribution

Objectives

Core power distribution testing was performed in accordance with UFSAR section 13.5.2.2.(17). The objectives were to confirm the reproducibility of the TIP readings, determine the core power distribution in three dimensions, and determine core power symmetry.

Description

TIP reproducibility is checked with the plant at steady-state conditions by running several TIP traverses through the same core location (common channel 32-33) with each TIP detector. The TIP data is then statistically evaluated to determine the extent of deviations between traverses from the same TIP machine.

Core power distribution, including power symmetry, will be determined by running at least two full sets of TIP runs (OD-1s) at steady state conditions, and then statistically evaluating the TIP data from symmetric core locations to determine core power symmetry. This TIP data will also provide the axial and radial flux distribution for the core.

Acceptance Criteria

In the TIP reproducibility test, the TIP traverses shall be reproducible within +/- 3.5% relative error or +/- 0.15 inches (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 78% power on 12-31-92. The TIP traverses were reproducible within 3.5% relative error. Data Set 1 had a total TIP uncertainty of 1.946% and Data Set 2 had a total TIP uncertainty of 2.034%, both of which are within the 7.1% acceptance criteria. The maximum deviation between symmetrically located pairs (pair 40/12) was 8.22%.

The axial and ring relative power distributions that were predicted for the short shallow and full power target rod patterns were compared with the actual power distributions after the rod patterns were set.

2.18 Core Performance

Objectives

Core performance was monitored in accordance with UFSAR section 13.5.2.2.(18). The objectives were to evaluate the core performance parameters of the core flow rate, core thermal power, and the core thermal limit values of Minimum Critical Power Ratio, Linear Heat Generation Rate, and Average Planar Linear Heat Generation Rate.

Description

Core thermal power, core flow, and thermal limit values were determined using the Plant Monitoring System and other plant instrumentation. This was determined at various reactor conditions, and methods independent of the Plant Monitoring System were also used.

Acceptance Criteria

Steady state core thermal power shall not exceed 3293 MWth. The thermal limit values of Maximum Fraction of Limiting Critical Power Ratio (MFLCPR), Maximum Fraction of Limiting Power Density (MFLPD), and Maximum Average Planar Ratio (MAPRAT) shall not exceed 1.00.

Results

The core thermal limit values were checked at least daily above 25% using the Plant Monitoring System. The PMS core thermal power heat balance and core flow values were verified by performing RT-R-059-500-2 and RT-R-02F-250-2 "Core Flow Verification" on 1-22-93.

Core thermal power, core flow, and thermal limit values did not exceed their maximum allowed values at any time during the power ascension.

PMS thermal limit values were checked against output from PANACEA, a licensed 3-dimensional core physics code, and General Electric's BUCLE code.

The proper reactivity behavior of the core as a function of cycle exposure was verified by performing ST-R-002-900-2 "Reactivity Anomalies" at 100% power on 1-20-93.

2.19 Feedwater System

Objectives

Feedwater system testing was performed in accordance with UFSAR section 13.5.2.2.(22). The objectives were to demonstrate acceptable reactor water level control, evaluate and adjust feedwater controls, and to demonstrate capability of the automatic Recirc runback feature to prevent a low water level scram following a trip of a Feedwater pump.

Description

Reactor water level setpoint changes of approximately +/- 6 inches will be used to evaluate and adjust the feedwater control system settings for all power and Feedwater pump modes. One of the operating Feedwater pumps will be tripped at 75% power and above 90% power while the automatic flow runback circuit acts to drop power to within the capability of the remaining two Feedwater pumps. Additional testing required by the installation of MOD 1843 (Digital Feedwater Controller Installation) is described in the "Results" section.

Acceptance Criteria

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 setpoint changes. System response for large transients should not be unexplainably worse than pre-analysis. The automatic Recirc flow runback feature will prevent a scram from low water level following a trip of one Feedwater pump.

Results

During the refueling outage, 2 modifications were installed that affected the Feedwater system. MOD 5223 (Feedwater Heater Replacement) removed four feedwater heaters in the neck of the condensers and replaced them with new heaters of the same design. This work was completed on 11-18-92 and the feedwater heater string flush was completed on 11-30-92. During power ascension, the increase in thermal efficiency due to the new heaters was readily observable, since less extraction steam was required to provide the same feedwater temperature at the core inlet.

Mod 1843 (Digital Feedwater Controller Installation) was installed during the outage and tested extensively during the outage and power ascension by performing a series of Modification Acceptance Tests (MATs) at various plant conditions.

ST-O-02B-250-2 "Reactor Water Level Instrument Perturbation Test", a monthly test, was performed satisfactorily during the startup on 12-5-92 and 12-24-92.

MAT 1843F demonstrated the ability of the startup bypass "C" Feedpump discharge bypass level controller to maintain reactor water level. This test was completed satisfactorily on 12-7-92.

MAT 1843H demonstrated acceptable response of each Reactor Feed Pump (RFP) to step changes in the Master Level Controller. Master level controller step changes of +/- 6 inches were performed at 25%, 35%, 40%, 50%, 75%, 90% and 100% reactor power from 12-11-92 through 1-19-93. The digital feedwater control system (operating in 3 element and single element control) displayed satisfactory response.

MAT 1843J demonstrated satisfactory response of the feedwater level control system during the trip of one RFP (2B) with reactor power at 72% on 12-15-92, and then again with reactor power at 92% on 1-28-93. When the RFP was tripped at 92% reactor power, the Recirc runback to 60% speed was received as designed and the remaining two operating RFPs properly responded to control reactor water level.

MAT 1843K demonstrated satisfactory response of the feedwater level control system during the trip of one Condensate pump (2B) with reactor power at 93%. Reactor water level dropped below 17" and the Recirc runback to 60% speed was received as designed. This test was performed satisfactorily on 1-28-93.

MAT 1843L verified the ability of the feedwater level control system to control reactor water level following a Recirc pump trip from 95% power and 100% core flow. This test was performed satisfactorily on 1-29-93.

MAT 1843M tested the response of the feedwater level control system to a loss of its 120 VAC feeds. This test was performed satisfactorily on 11-4-92.

MAT 1843P demonstrated that the feedwater level control system can control reactor water level within +/- 6.0 inches of the desired setpoint when increasing or decreasing reactor power at a rate of 10 MWe/minute. This test was completed satisfactorily on 12-15-92.

MAT 1843Q demonstrated the fault tolerance of the feedwater level control system. In this test, various components of the control system were purposely failed and the appropriate actions, transfers, and alarms were verified. This test was satisfactorily performed on 12-17-92.

2.20 Bypass Valves

Objectives

The main turbine Bypass Valves (BPVs) were tested in accordance with UFSAR section 13.5.2.2.(23). The objectives were to demonstrate the ability of the pressure regulator to minimize the reactor disturbance during a change in reactor steam flow and to demonstrate that a bypass valve can be tested for proper functioning at rated power without causing a high flux scram.

Description

One of the BPVs will be tripped open by a test switch. The pressure transient will be measured and evaluated to aid in making adjustments to the pressure regulator.

Acceptance Criteria

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to BPV position changes. The maximum pressure decrease at the turbine inlet should be less than 50 psig to avoid approaching low steam line pressure isolation or cause excessive water level swell in the reactor.

Results

Each BPV was operationally tested in accordance with RT-O-001-409-2, performed on 12-14-92. This is a monthly test that fully strokes all 9 BPVs. Turbine first stage pressure and reactor water level remained normal during the BPV testing.

During power ascension, the performance of the BPVs were monitored in accordance with GP-2.

2.21 Main Steam Isolation Valves

Objectives

The MSIVs were tested in accordance with UFSAR section 13.5.2.2.(24). The objectives were to functionally check the MSIVs for proper operation at selected power levels and to determine isolation valve closure time.

Description

Functional checks (10% closure) of each isolation valve will be performed at selected power levels. Each MSIV will be individually closed below 75% power and the closure times will be measured.

Acceptance Criteria

MSIV stroke time will be within 3 and 5 seconds, exclusive of electrical delay time. During full closure of individual valves, reactor pressure must remain 20 psi below scram, neutron flux must remain 10% below scram, and steam flow in individual lines must be below the trip point.

Results

During the outage, each MSIV was strokes satisfactorily in accordance with RT-M-01A-471-2, performed on 11-27-92.

During the initial startup, each MSIV was opened in accordance with GP-2 and SO 1.A.1.A-2 on 12-5-92.

Prior to reaching 100% power for the initial time in Cycle 10, it was necessary to take a force maintenance outage on 1-2-93. Unit 2 was being shut down in accordance with GP-3 "Normal Plant Shutdown". At approximately 10 psig reactor pressure, the MSIVs were closed. All MSIVs closed normally except the inboard MSIVs on the "A" and the "C" Main Steam Lines. The "A" inboard MSIV never fully closed and the "C" inboard MSIV had an excessively long closure time of 47 seconds.

During the Maintenance outage, the air manifolds were checked for clogging and rebuilt. Following the rebuild, the MSIVs were stroked satisfactorily.

MSIV individual closure timing and continuity checks are performed monthly per ST-O-07G-470-2 and was performed during the power ascension on 1-11-93. All MSIVs had a full closure stroke time between 3 and 5 seconds.

2.22 Relief Valves

Objective

Relief valve testing was performed in accordance with UFSAR section 13.5.2.2.(25). The objectives were to verify the proper operation of the dual purpose relief safety valves, to determine their capacity, and to verify their leaktightness following operation.

Description

The Main Steam Relief Valves (MSRVs) 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 Turbine 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 tailpipe.

Acceptance Criteria

Each relief valve is expected to have a capacity of at least 800,000 lb/hr at a pressure setting of 1080 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 degrees F of the temperature recorded before the valve was opened. Each valve must move from fully closed to fully opened in 0.3 seconds.

Results

Each Safety Relief Valve (SRV) was manually cycled in accordance with ST-O-01G-440-2 "Relief Valve Manual Actuation". This test was performed on 12-5-92 at 148 psig reactor pressure.
Each SRV (including the 5 ADS valves) had a satisfactory closure time.

2.23 Turbine Stop and Control Valve Trips

Objective

The Turbine Stop Valve (TSV) and Turbine Control Valve (TCV) trips were tested in accordance with UFSAR section 13.5.2.2.(26). The objective of this test was to demonstrate the response of the reactor and its control systems to protective trips in the turbine and the generator.

Description

The TSVs will be tripped at selected reactor power levels and the main generator breaker will be tripped in such a way that a load imbalance trip occurs. Several reactor and turbine operating parameters will be monitored to evaluate the response of the bypass valves, relief valves, RPS, and the effect of Recirc pump overspeed (if any) during the control valve trip. Additionally, peak values and change rates of reactor steam pressure and neutron flux will be determined. The ability to experience a load rejection without a scram will be demonstrated.

Acceptance Criteria

The maximum reactor pressure should be less than 1200 psig, 30 psi below the fast safety valve setpoint, during the transient following first closure of the TSVs and TCVs. Core thermal power must not exceed the safety limit line. The trip at or below 25% power must not cause a scram. Feedwater control adjustments shall prevent low level initiation of the HPCI system and Main Steam isolation as long as feedwater flow remains available.

Results

The following tests were performed on 12-10-92 at 22% power:

* ST-O-60F-420-2 "Turbine Control Valve Fast Closure Scram Functional"
* ST-O-001-200-2 "Turbine Main Stop Valve Closure Functional"

In addition, the TSVs are tested weekly in accordance with RT-O-001-400-2.

During the BOC 10 startup, it was necessary to reduce power and remove the generator from service in order to repair a hydrogen leak and perform EHC system troubleshooting. Power was reduced from 78% in accordance with GP-5 "Power Operations" on 12-18-92. At approximately 18.5% power (200 MWe), the main generator was manually tripped. The turbine bypass, control, and stop valves

performed as designed, and the reactor pressure and neutron flux spikes were well below the trip setpoints. The feedwater control system maintained reactor water level throughout the transient, and the reactor did not scram due to the load rejection.

Following the transient, 5.5 BPVs were open and delivering steam to the main condenser, as designed.

The generator was returned to service on 12-20-92 and power ascension resumed.

2.24 Flow Control

Objective

Flow control testing was performed in accordance with UFSAR section 13.5.2.2.(28). The objective was to determine the plant response to changes in recirculation flow and thereby adjust the local control loops. The Recirc 30% and 60% limiters, and high speed electrical and mechanical stops, will also be set.

Description

Various process variables will be monitored while changes (positive and negative) are introduced into the Recirc flow control system.

Acceptance Criteria

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to flow control changes.

Results

The Recirc pump 30% speed limiters were set on 11-10-92 in accordance with RT 3.12 "Recirculation Pump 30 Percent Speed Limiter In-Place Calibration".

The Recirc pump 60% speed limiters were set on 12-17-92 in accordance with RT 3.13 "Recirculation Pump 60 Percent Speed Limiter In-Place Calibration".

The Recirc M/G set high speed mechanical and electrical stops were set on 2-1-93 in accordance with RT 3.14 "Recirculation Pump High Speed Stop Adjustments".

During the MAT 1843 digital feedwater controller testing (see section 2.20), the Recirc runback to 60% speed performed as designed during the Condensate pump trip and RFP trip.

2.25 Recirculation System

Objectives

Recirc system testing was performed in accordance with UFSAR section 13.5.2.2.(29). The objectives were to determine transient responses and steady state conditions following Recirculation pump trips at selected power levels, to obtain jet pump performance data, and to calibrate the jet pump and flow instrumentation.

Description

Following each Recirc pump trip, process variables such as reactor pressure, steam and feedwater flow, jet pump differential pressure, and neutron flux will be monitored during the transient and at steady state conditions. The jet pump instrumentation will be calibrated to indicate total core flow.

Acceptance Criteria

For each pump trip test, no core limits shall be exceeded. Flow instrumentation shall be calibrated such that the reactor jet pump total flow recorder provides correct flow indication.

Results

During power ascension, jet pump operability was checked daily in accordance with ST-O-02F-550-2 "Jet Pump Operability". Jet pump performance was trended weekly in accordance with RT-R-02F-550-2 "Jet Pump Performance Trending".

On 12-16-92 the 2B Recirc pump was manually tripped from 54% power in order to repair the M/G set Bailey positioner shaft bearing. All process variables responded as expected and single loop Recirc baseline data was obtained for the active loop (A) in accordance with ST-R-02A-210-2 "Recirc System Baseline Data - Single Loop Operation".

On 1-29-93, the 2A Recirc pump was manually tripped from 95% power to support MAT 1843 digital feedwater controller testing (see section 2.20). Core thermal-hydraulic stability monitoring was performed in accordance with OT-112 "Recirc Pump Trip" and Table 1 control rods were inserted to get below the 80% rod line. All process variables responded as expected and Recirc baseline data was obtained for the active loop (B) in accordance with ST-R-02A-210-2.

The flow instrumentation calibration was checked by performing RT-R-02F-250-2 "Core Flow Verification" at 100% power steady-state conditions on 1-22-93.