Peach Bottom Atomic Power Station

PECO ENERGY

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February 10, 1994

Docket No.: 50-278
License No. DPR-56

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

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

Gentlemen:

Enclosed is the Peach Bottom Atomic Power Station Unit No. 3 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-56.

Sincerely,

GRRainey

Gerald R. Rainey
Vice President

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PECO ENERGY COMPANY

PEACH BOTTOM ATOMIC POWER STATION
UNIT NO. 3
DOCKET NUMBER 50-278

CYCLE 10
STARTUP REPORT

SUBMITTED TO
THE U.S. NUCLEAR REGULATORY COMMISSION
PURSUANT TO
FACILITY OPERATING LICENSE DPR-56

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INTRODUCTION

PBAPS Unit 3 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 3 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 3 was out of service from 9-18-93 to 11-15-93 to accommodate its ninth refueling outage. During this 59 day outage, 38 non-barrier fuel bundles of the P8DRB design and 217 barrier fuel bundles of the GE7B design and one barrier GE8B bundle were replaced with 4 Siemens 9x9-A lead use assemblies (LUAs) and 252 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:

- Recirc Pump Controller Replacement (MOD 887)
- Reactor Water Level Continuous Backfill System (MOD P128)
- Replacement of 34 control rod drives
- Replacement of 8 control rod blades
- 62 control rod drive HCUs rebuilt
- Bottom head drain / Reactor annulus debris inspection
- Inspection of a statistical sampling of fuel bundles for debris intrusion
- Installation of MOD 5374 (ARTS/MELLLA)

Unit 3 returned to service on 11-15-93 and reached full power for the first time in Cycle 10 on 11-19-93. Startup testing was completed on 2-8-94.

The successfully implemented startup test program ensures that the ninth refueling outage of Unit 3 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-C-095-886-3 "Chemistry Preparation for Reactor Startup" on 11-19-93. The Shift Chemist also verified that reactor water dose equivalent I-131, chloride concentration, and sulfate concentration were within specification per CH-10.

During power ascension, coolant chemistry was verified to meet water quality specifications and process requirements by ST-C-095-824-3 "Reactor Startup Chemistry With Steaming Rates

Less Than 100,000 Lbs/Hr", performed on 11-13-93. At high steaming rates, ST-C-095-823-3 "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-095-857-3, ST-C-095-859-3, and ST-C-095-860-3.

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

The Offgas system was placed in service on 11-14-93. The steam jet air ejector discharge activity indicated that Unit 3 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 9-16-93. 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 310 man-rem of exposure as measured by direct reading dosimeter.

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 10-30-93. The final loading pattern includes 252 new GE11 fuel bundles, 4 new Siemens SPC 9x9A Lead Use Assemblies (modelled as GE9B fuel), 256 GE9B bundles, 188 GE8B bundles, and 64 GE7B bundles. 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-004-970-3 "Core Verification", completed on 10-30-93.

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-003-465-3 "Control Rod Withdraw Tests", completed on 11-11-93.and 11-16-93. This test was competed satisfactorily for all rods with the exception of rod 54-35, which was found to be uncoupled. Several attempts were made to re-couple the rod per ON-105 "Control Rod Uncoupled - Procedure".

When these re-coupling attempts proved to be unsuccessful, the decision was made to declare the rod inoperable and disarm it in the fully inserted position per Technical Specifications. The startup and shutdown rod movement instructions were revised to reflect this inoperable rod condition.

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-002-910-3, performed on 11-12-93.

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-002-910-3 "Shutdown Margin (Unit 3- Cycle 10)" on 11-12-93. 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 1417 on 11-12-93 on step 33 of the RWM startup sequence (rod 50-39 @ position 34). Reactor water temperature was 148 degrees F, count rate doubling time was 105 seconds, and the calculated reactor period was 151.2 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 1.27% delta K/K. This value was verified to be greater than (.38% delta K/K + R), which is (.38% + .141%) delta K/K or 0.521% delta K/K.

To allow a minimum reactor water temperature of 40 degrees F (from a SDM concern) throughout cycle 10, an additional SDM adder of .21% delta K/K was applied, which required the calculated SDM value to exceed 0.731% delta K/K. The calculated SDM value exceeded 0.731% delta K/K by (1.27 - 0.731)% delta K/K, or 0.539% delta K/K.

The design predicted Keff value was compared to the actual value calculated at initial criticality in accordance with ST-R-002-920-3 "Critical Eigenvalue Comparison", performed on 11-12-93. Using the shutdown margin test data, the predicted Keff value was 1.0045 and the actual Keff value was calculated to be 1.0008. Therefore, the difference between the predicted and the actual Keff values was 0.37% 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

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

Results

Prior to startup, during the RPV pressure test, each CRD was scram timed in accordance with ST-R-003-460-3 "CRD Scram Insertion Timing, Full In and Full Out Position Indication Check, and Rod Coupling Integrity Check for All Operable Control Rods", completed on 11-9-93. All 185 rods met with two-by-two array insertion time criteria. Subsequent HCU valve Maintenance and the replacement of CRD 54-35 (uncoupled rod) required rods 54-35, 46-39, 22-47, and 30-19 to be retested during power ascension at 35% power in accordance with ST-R-003-485-3 "Scram Insertion Timing of Selected Control Rods", on 12-22-93.

Each CRD had its normal insert speeds, withdraw speeds, and coupling integrity checked by ST-O-003-465-3 "Control Rod Withdraw Tests". It was during the performance of this test that rod 54-35 was discovered to be uncoupled. This rod was declared inoperable for the initial BOC10 startup, and had its CRD replaced during a subsequent maintenance outage.

During power ascension, ST-O-003-470-3 "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 for all rods on 1-17-94.

During power ascension, when reactor power was above the RWM LPSP (approximately 20%), ST-O-003-560-3 "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-2-3 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-2-3 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 Core Monitoring Instructions".

Results

Cold criticality was achieved on 11-12-93 by withdrawing rods in accordance with GP-2-3 Appendix 1. This same sequence (Startup 1) had previously been verified in the RWM in accordance with ST-R-62A-220-3 "RWM Sequence Verification" prior to rod withdrawal per GP-2. Prior to withdrawing the first rod, ST-O-62A-210-3 "RWM Operability Check" was performed on 11-12-93. Criticality occurred on sequence step 33 in RWM group 2. The critical rod pattern is recorded in GP-2-3 Appendix 1 and ST-R-002-910-3 "Shutdown Margin (Unit 3 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 (flux shaping) 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 rod patterns while staying within licensed core limits meets the requirements of this test.

Results

The flux shaping rod pattern was developed for a reactor statepoint of 75% power and 65% core flow. This rod pattern was achieved on 11-18-93. The full power target rod pattern was developed for a reactor statepoint of 100% power and 92% core flow, placing the core on the 106% load line. This rod pattern was achieved on 11-19-93. Rod movements were governed by RE-31 "Reactor Engineering Core Monitoring Instructions".

Rated 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-250-3 "SRM Operability and Neutron Response Check".

Prior to startup, SRM performance was verified by SI3N-60D-SRM-A2CZ "SRM Channel A Calibration/Functional Check" and SI3N-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 11-12-93. 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-002-910-3.

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

Prior to startup, IRM performance was tested and the scram setpoints were verified by performing SI3N-60C-IRM-A(B)4CZ "Intermediate Range Monitor Channel A(B) Calibration /Functional Check" (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-230-3 "LPRM Gain Calibration" was performed twice during power ascension. The first calibration was performed on 11-17-93 at 29% 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.

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:

* SI3N-60A-APRM-A1CM(thru F1CM) "Average Power Range Monitor Calibration/Functional Check"

* SI3N-60A-APRM-A(B)3FW "Average Power Range Monitor Channel A(B) Functional Check"

APRM calibrations were performed in accordance with ST-O-60A-210-3 "APRM System Calibration During Two Loop Operation" throughout power ascension. The first APRM gain calibration was performed on 11-14-93 at 8% power and the last APRM gain calibration was performed on 11-21-93 at 100% power.

All 6 APRMs were operable for the initial BOC startup.

The APRM channels were calibrated to core thermal power 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 plant personnel. 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. The BOC10 databank was installed and verified in accordance with RE-38 "Process Computer NSSS BOC Databank Update". During power ascension, the PMS heat balance was verified to be correct by performing RT-R-059-500-3 "Checkout of the NSSS Computer Calculation of Core Thermal Power" at approximately 100% power on 11-24-93. 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 with PMS plant data obtained at 100% power steady state conditions. The P1 and BUCLE thermal limit values all agreed to within 1%. 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-013-200-3 on 11-13-93. A cold quick start at rated reactor pressure was performed in accordance with ST-O-013-301-3 on 11-15-93.

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 outage, the HPCI turbine overspeed test was performed (on aux steam from the boilers) on 10-29-93 in accordance with RT-X-023-240-3.

During the startup, a controlled start was performed at 150 psig reactor pressure in accordance with ST-O-023-200-3 on 11-13-93. A cold quick start at rated pressure was performed in accordance with ST-O-023-301-3 on 11-14-93.

HPCI response time was checked in accordance with ST 6.5.R-3. 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 startup preparations prior to rod withdrawal, the appropriate reactor vessel temperatures were monitored to verify that thermal stratification did not exist. After verifying that the dome-to-bottom head and loop-to-loop temperature criteria were met, the second Recirc pump was placed in service in accordance with SO 2A.1.B-3 "Starting the Second Recirculation Pump".

No Recirc pump trips occurred during power ascension.

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 during power ascension once an octant-symmetric rod pattern was established. The TIP traverses were reproducible within 3.5% relative error. Data Set 1 had a total TIP uncertainty of 2.09% and Data Set 2 had a total TIP uncertainty of 2.53%, both of which are within the 7.1% acceptance criteria.

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-059-500-3 and RT-R-002-250-3 "Core Flow Verification" on 11-24-93.

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-002-900-3 "Reactivity Anomalies" at 100% power on 11-24-93 and 12-21-93.

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 and evaluate and adjust feedwater controls.

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.

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.

Results

ST-O-02B-250-3 "Reactor Water Level Instrument Perturbation Test", a monthly test, was performed satisfactorily during the startup on 12-23-93 and 1-14-94.

Proper Reactor Feed Pump and Feedwater system response was verified during power ascension per GP-2.

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-001-409-3, performed on 11-10-93 and 12-16-93. 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 stroked satisfactorily in accordance with RT-M-01A-471-3, performed on 10-23-93.

During the initial startup, each MSIV was opened in accordance with GP-2 and SO 1.A.1.A-3.

MSIV individual closure timing and continuity checks were performed on 10-27-93 per ST-O-07G-470-3. 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-440-3 "Relief Valve Manual Actuation". This test was performed on 11-13-93 at approximately 150 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 during power ascension:

* ST-O-60F-420-3 "Turbine Control Valve Fast Closure Scram Functional"
* ST-O-001-200-3 "Turbine Main Stop Valve Closure Functional"

In addition, the TSVs are tested weekly in accordance with RT-O-001-400-3.

Following the BOC 10 startup, it was necessary to reduce power and remove the generator from service in order to perform a maintenance outage. Power was reduced to approximately 20% in accordance with GP-5 "Power Operations", and 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.

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 45% 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 installation of MOD 887 (Recirc Pump controller Replacement) and MOD 5374 (ARTS/MELLLA) required testing of the flow control loops. The value for the upper Recirc flow limiter was change from 60% speed to 45% speed. This change was necessary since the Recirc pumps will be operating at a lower core flow (approximately 82% vice 100%) while in the MELLLA region of the power/flow map.

In-place calibrations of the 30% and 45% Recirc pump speed limiters were performed during the power ascension. During the initial BOC10 startup, the Moore controller on the 3B Recirc pump (installed by MOD 887) reached its maximum output prematurely at 92% core flow. Additional control rods were withdrawn to achieve full power, placing the core on the 106% load line instead of the 100% load line.

During a subsequent maintenance outage, a cam in the 3B Recirc M/G set scoop tube positioner was re-cut to allow pump speed to be varied over the desired range with the controller.

Using data collected during the power ascension, the Recirc M/G high speed mechanical and electrical stops were also set.

MOD 5374 installed ARTS/MELLLA on Unit 3. This MOD involved making hardware changes to the APRM and RBM systems, and analytical changes to allow operation in an extended region of the power/flow map. ARTS/MELLLA allows reactor operation with load lines as high as 121%, which allows full power operation at core flows as low as 75%. This allows the flow-enhanced spectral shift operating strategy to be implemented, which in turn improves cycle energy output. The hardware changes were made during the refueling outage, and the at-power testing was performed on 2-8-94 in accordance with MAT 5374B. This testing involved verifying that several plant parameters exhibited stable behavior (quarter-wave dampening) following step changes in core flow. The test also mapped the rod line behavior while in the MELLLA region. All test acceptance criteria were met.

2.25 Recirculation System

Objectives

Recirc system testing was performed in accordance with UFSAR section 13.5.2.2.(29). The objectives were to obtain jet pump performance data and to calibrate the jet pump and flow instrumentation.

Description

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

Acceptance Criteria

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-550-3 "Jet Pump Operability". Jet pump performance was trended weekly in accordance with RT-R-02F-550-3 "Jet Pump Performance Trending".

The flow instrumentation calibration was checked by performing RT-R-002-250-3 "Core Flow Verification" at 100% power steady-state conditions on 12-2-93.