Thomas N. Mitchell
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50-278

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February 2, 1998

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 The Eleventh Refueling Outage

Gentlemen:

Enclosed is the Peach Bottom Atomic Power Station Startup Report for Unit No. 3 Cycle 12. The report is submitted pursuant to Unit 3 Technical Requirements Manual Appendix A.

Sincerely,

TNMitchell

Thomas N. Mitchell
Vice President,
Peach Bottom Atomic Power Station

MEW OAL MTC DBH
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PECO
ENERGY

PEACH BOTTOM ATOMIC POWER STATION

CYCLE 12
STARTUP REPORT
UNIT 3

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

JANUARY
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INTRODUCTION

PBAPS Unit 3 Technical Requirements Manual Appendix 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 eleventh 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 10-3-97 to 11-2-97 to accommodate its eleventh refueling outage. During this 29 day outage, 292 new GE13 fuel bundles were loaded into the core, with the balance of the core load being comprised of once and twice burned GE11 fuel bundles and four Siemens Power Corporation (SPC) qualification fuel bundles. The Cycle 12 core consists entirely of barrier fuel.

This is the first application of the GE13 product line at PBAPS Unit 3, but it is currently in use in PBAPS Unit 2 Cycle 12. The GE13 fuel type has been approved for use by the NRC, and incorporates only minor evolutionary changes to the fuel types previously used at PBAPS. GE13 fuel is mechanically, neutronically, and thermal-hydraulically compatible with the co-resident fuel, RPV internals, spent fuel pool internals, refueling equipment, and other interfacing plant systems. GE13 fuel complies with all required fuel design and licensing bases during steady-state, transient, and accident conditions.

INTRODUCTION (continued)

The primary design differences between the GE13 and GE11 designs are as follows:

- Improved critical power performance and cycle economics
- ~~3 part length rods are 12 in. longer than GE11
- 3 has one more fuel pin spacer than GE11
- ~13 bundle has a mass approximately 2 Kg more than GE11

Both GE.. and GE13 fuel designs have a 9 x 9 rod array with two large central water rods and an active fuel length of 146 inches. The GE13 fuel product line has a different Safety Limit Minimum Critical Power Ratio (SLMCPR) value than the fuel designs previously utilized at PBAPS. The new SLMCPR values are 1.11 for operation with two recirculation loops in service and 1.12 for single loop operation. Under ECR 96-03620, a License Change Request was submitted to update Unit 3 Technical Specification Section 2.1.1.2 with the GE13 SLMCPR values. The License Change Request was approved by the NRC, and was effective prior to Cycle 12 Startup.

Other in-vessel maintenance performed during the outage included:
- Replacement of 19 control rod drives.
- Replacement of 13 LPRM strings.
- Replacement of 16 control blades.
- Wide Range Neutron Monitoring (WRNM) system replaced the Source Range Monitoring (SRM) system and the Intermediate Range Monitoring (IRM) system.

Also, during 3R11, in-vessel inspections were performed. During these inspections cracks were identified in three jet pump risers. A 10CFR50.59 review was performed to support Unit 3 operation until repairs can be designed and installed. These repairs will be performed during mid cycle outage 3J12. Information regarding the operating strategy for Unit 3 was transmitted in a letter from T.N. Mitchell to the U.S. Nuclear Regulatory Commission, dated November 19, 1997.

Unit 3 returned to service on 11-2-97 and reached steady-state full power for the first time in Cycle 12 on 11-12-97. Startup testing was completed on 11-27-97.

The successfully implemented startup test program ensures that the eleventh refueling outage of Unit 3 has resulted in no conditions or system characteristics that in any way diminish the safe operation of the plant.

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 demineralized 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 11-5-97. 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 11-2-97.

2.1 Chemical and Radiochemical (continued)

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 µmhos/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 11-2-97. 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 is performed. During the refueling outage, startup, and power ascension, gamma radiation measurements and neutron dose rate measurements (where appropriate) is performed 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-C-200 "Routine Survey Program" to determine background radiation levels and assure personnel safety.

The initial survey of the drywell was performed in accordance with HP-315. During the refueling outage and subsequent plant startup, appropriate radiation surveys were performed to generate Radiation Work Permits per HP-C-310 and properly post plant radiation areas per HP-C-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 282 person-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. The SRM requirements for fuel moves were met until 10-15-97 when all fuel moves were halted and the transition from SRMs to WRNMs began.

On 10-21-97, the Wide Range Neutron Monitoring (WRNM) system testing was complete, the system was put into service, and the WRNM TS requirements were applicable. The balance of the core alterations were performed within the WRNM TS requirements. During the balance of the fuel movement activities, at least 2 WRMNs 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 WRNM in the quadrant where the alteration is performed. WRNM 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.

2.3 Fuel Loading (continued)

Results

The fuel shuffle was performed in accordance with FH-6C "Core Component Movement - Core Transfers" and was completed on 10-25-97. The final loading pattern includes 292 new GE13 fuel bundles, 284 once-burned GE11 bundles, 184 twice-burned GE11 bundles and four SPC qualification fuel bundles. The complete Cycle 12 core consists of barrier fuel.

Ensuring proper fuel loading into the core, the following three steps were performed:

Serial number and location verification of all of the new fuel in the fuel pool prior to core load were performed on 8-19-97, in accordance with M-004-116 "Pre-Refuel Outage Spent Fuel Pool Verification."

Proper fuel bundle orientation and seating verification and debris inspection of the final loaded core were performed on 10-26-97, in accordance with M-C-797-020 "Core Verification."

Serial number and location verification of the 3R11 discharged fuel in the fuel pool were performed prior to reaching 25% power on 10-27-97, in accordance with M-004-117 "Post Refuel Outage Spent Fuel Verification."

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-0-003-465-3 "Control Rod Withdraw Tests", completed on 11-1-97. 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 11-1-97.

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, the bias between local and distributed eigenvalue, and the "R" value for the cycle.

Acceptance Criteria

The fully loaded core must be subcritical by at least 0.38% ΔK/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" on 11-1-97. Control rods were withdrawn according to the startup sequence. WRNM count rates were recorded after each control rod withdrawal. The reactor was declared critical at 1740 on 11-1-97 with RWM group 2 control rod 18-15 at position 28. Reactor water temperature was 147 degrees F. count rate doubling time was 190 seconds, and the calculated reactor period was 273.6 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, period, local versus distributed eigenvalue, and 'R' correction factors. This calculated SDM value was equal to 1.348% delta K/K. This value was verified to be greater than 0.38% ΔK/K.

To allow a minimum reactor water temperature of 38 degrees F throughout Cycle 12, a SDM adder of 0.15% ΔK/K was applied; therefore, the SDM value for reactor temperatures down to 38 degrees F. is (1.348 - 0.15)%, or 1.198% ΔK/K. The difference between the predicted and actual SDM values was (1.348% - 1.04%), or 0.308% ΔK/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	.44 to pos 46
20	0.900	1.08 to pos 36
50	2.000	1.83 to pos 26
90	5.000	3.35 to pos 06

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

2.5 Control Rod Drives (continued)

Results

Each CRD had its normal insert speeds, withdraw speeds and coupling integrity checked by ST-0-003-465-3 "Control Rod Withdraw Tests", completed on 11-1-97. All insert and withdraw speeds fell within the acceptance criteria of 45-51 sec/ full stroke, or an Action Request was generated to investigate the problem. This test also checked CRD stall flows and rod position indication, and verified core subcriticality.

Prior to exceeding 40% power during the BOC startup, each CRD was scram timed in accordance with ST-R-003-460-3 "CRD Scram Insertion Timing for All Operable Control Rods", completed on 11-3-97. All 185 rods had satisfactory scram times prior to exceeding 40% power.

During power ascension, ST-0-003-470-3 "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 on 11-5-97.

During power ascension, when reactor power was above the RWM LPSP (approximately 23%), ST-0-003-560-3 "Control Rod Exercise - Fully Withdrawn" was performed weekly. This test required each fully withdrawn rod to be inserted and withdrawn one notch.

In addition, ST-0-003-561-3 "Control Rod Exercise - All Rods" was performed monthly, and required every control rod to be exercised 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 used for startup implemented the BPWS (Banked Position Withdrawal Sequence) methodology with the A2 sequence control rods. This sequence is contained in GP-2-3 Appendix A2 (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 A2 sequence is programmed into the RWM and is designated as "Startup A2". This sequence specifies rod withdrawal from the all-rods-in condition to the rod pattern in which all A2 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 11-1-97 by withdrawing rods in accordance with GP-2-3 Appendix A2. This same sequence (Startup 2) had previously been verified in the RWM in accordance with ST-R-62A-220-3 "RWM Sequence Verification", performed on 10-31-97. Prior to withdrawing the first rod, ST-0-62A-210-3 "RWM Operability Check" was performed on 11-1-97. Criticality occurred on sequence step 44 in RWM group 2. The critical rod pattern is recorded in GP-2-3 Appendix A2 and ST-R-002-910-3 "Shutdown Margin".

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.

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

Several intermediate rod patterns were developed and attained prior to achieving the target rod pattern. On 11-6-97, a load drop to 70% power was performed to set the final target rod pattern. In accordance with the Jet pump riser crack operating strategy, a nominal 94% power target rod pattern was achieved on 11-7-97. Full power equilibrium conditions in the target rod pattern were achieved on 11-12-97 and was sustained for two days.

During the numerous control rod movements performed during the startup, no thermal limit violations occurred.

2.8 WRNM Performance

Objective

SRM performance (UFSAR section 13.5.2.2.(8)) and IRM performance (UFSAR section 13.5.2.2.(9).) are no longer applicable to Peach Bottom Unit 3 since the SRM and the IRM systems have been replaced with the Wide Range Neutron Monitor (WRNM) system. The WRNM system was installed during 3R11 under Mod P271. Core monitoring and startup and testing was performed in accordance with MAT P271 D-3 and F-3, respectively.

The objective was to demonstrate that WRNM instrumentation provided adequate information to the operator during startup.

Description

WRNM 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 WRNMs as well as a minimum count rate of 3 CPS on the required operable WRNMs. In addition, WRNM indication was monitored throughout the startup range to verify proper period response and correct auto-ranging during power ascension. WRNM power indication was adjusted to match APRM power (as calibrated to BPV position) at the transition from Mode 2 to Mode 1.

Results

Prior to startup, WRNM performance was tested via several MATs and surveillance tests. WRNM scram setpoints were verified by performance of SI3N-60C-WRNM-A(through H)1C2 "WRNM Channel A (through H) Calibration/Functional Check." These surveillance's were conducted as part of MAT P271 C-3. In addition, WRNM signal to noise ratio check was performed per SI3N-60C-WRNM-A(through H)1MX as part of MAT P271 D-3. WRNM minimum count rate was determined to be greater than 3 CPS prior to control rod withdraw on 11-1-[illegible] All 8 WRNM channels were operable for BOC12 startup.

During startup, WRNM operability was verified in accordance with GP-2 "Normal Plant Startup." WRNM count rate data following each rod withdrawal to criticality was recorded in ST-R-002-910-3. WRNM response during power ascension was monitored and verified in accordance with GP-2 and MAT P271 F-3. WRNM gain adjustment to APRM power indication was performed per MAT P271 F-3 following APRM calibration to BPV position per ST-O-60A-210-3. Following this adjustment, Mode 1 was entered and WRNM performance was monitored during the remaining power ascension per MAT P271 F-3.

2.9 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 3D Monicore 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-I-60A-230-3 "LPRM Gain Calibration" was performed on 11-11-97 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.

2.10 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-A1CE(through F1CE) "Calibration/Functional Check of Average Power Range Monitor (APRM) A (thruogh F)"
- SI3N-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-210-3 "APRM System Calibration During Two Loop Operation" throughout power ascension. The first APRM gain calibration was performed on 11-2-97 at ~6.25% power and the last APRM gain calibration was performed on 11-12-97 at 100% power.

The APRMs were calibrated to within plus or minus 2% of core thermal power during the power ascension.

All 6 APRMs were operable for the initial BOC startup.

2.11 Process Commuter

Objection

The Plant Monitoring System (PMS) and 3D Monicore System were tested in accordance with USFAR section 13.5.2.2.(12). The objective was to verify the performance of the these systems under operating conditions.

Description

During power ascension, the PMS provided NSSS and BOP process variable information to the operator. 3D Monicore provided core monitoring and predictor capabilities. The NSSS heat balance was verified to be correct and the BOC NSSS databank was installed and verified to be correct.

Acceptance Criteria

The PMS and 3D Monicore systems will be considered operational when plant sensor information is processed accurately, resulting in a correct thermal heat balance and core power distribution. The calculations shall be independently evaluated by the use of an off-line core physics code.

Results

The BOC12 databank was installed and verified in accordance with RE-38 "NSSS Software BOC Databank Update", and RE-41 "Installation/Verification of the 3D Monicore Thermal Operating Limits". During power ascension, the core heat balance was verified to be correct by performing RT-R-59C-500-3 "Checkout of the NSSS Computer Calculation of Core Thermal Power" at approximately 85% power on 11-6-97.

Thermal limit and power distribution results were also independently evaluated by Fuels & Services Division (FSD) using their off-line PANACEA code. Good agreement was observed between 3D Monicore and PANACEA results.

2.12 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 approximately 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 175 psig reactor pressure in accordance with ST-0-013-200-3 on 11-2-97. A cold quick start at rated reactor pressure was performed on 11-2-97.

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

2.13 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 prig 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 10-24-97 in accordance with RT-N-023-240-3.

A controlled start was performed at 175 psig reactor pressure in accordance with ST-0-023-200-3 on 11-2-97. A cold quick start at rated pressure was performed in accordance with ST-0-023-301-3 on 11-2-97. The HPCI turbine did not trip off during testing, and rated flow was achieved within the required time period.

2.14 Selected Process Temperatures

Objective

Selected temperatures were monitored in accordance with UFSAR section 13.5.2.2.(1S). 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 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

No Recirc pump trips occurred during the BOC12 power ascension. Prior to placing the second Recirc pump in service, all temperature requirements specified in SO 2A.1.B-3 were verified to be met. Throughout power ascension, whenever a heatup or cooldown of the RPV was in progress, the appropriate temperature readings were recorded in accordance with ST-0-080-500-3 "Recording and Monitoring Reactor Vessel Temperatures and Pressure".

2.15 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 between 980 and 1030 psig. During the 3R11 RPV pressure test, reactor pressure dropped below the required 980 psig for a period of 12 minutes. This reduction in pressure was related to Reactor Water Cleanup (RWCU) valve CV-3-12-055 allowing excessive flow, and the need to TC the procedure to throttle the HV-3-12-54. The details of this issue are documented in PEP I0007548. Evaluation #2 documents the satisfactory compliance with the ASME XI code requirements and Evaluation 3 documents ANII acceptance of the PEP disposition.

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

2.16 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 94% power on 11-11-97. The TIP traverses were reproducible within 3.5% relative error. Total TIP uncertainty was 1.37% which is within the 7.1% acceptance criteria. The maximum deviation between symmetrically located pairs (pair 40/12) was 7.76%, at node 10.

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.

2.17 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, 3D Monicore 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 3458 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% power using the 3D Monicore System. The core thermal power heat balance and core flow values were verified by performing RT-R-59C-500-3 on 11-6-97 and RT-I-002-250-3 "Core Flow Verification" on 11-10-97.

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

The proper reactivity behavior of the core as a function of cycle exposure was verified by performing ST-R-002-900-3 "Reactivity Anomalies" on 11-8-97.

2.18 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 to evaluate and adjust feedwater controls, as appropriate.

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

RT-0-02B-250-3 "Reactor Water Level Instrument Perturbation Test", a monthly test, was performed satisfactorily during the startup on 11-27-97.

No Feed Pumps were tripped during the power ascension, so the automatic Recirc runback feature was not observed.

2.19 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 ST-0-001-409-3, performed on 11-4-97. 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.

2.20 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 ST-M-01A-471-3, performed on 10-23-97. 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 are performed quarterly per ST-0-07G-470-3 and was performed on 10-23-97. All MSIVs had a full closure stroke time between 3 and 5 seconds.

2.21 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-01A-440-3 "Main Steam Relief Valve Manual Actuation". This test was performed on 11-2-97.

Each SRV (including the 5 ADS valves) had a satisfactory closure time.

2.22 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 and TCVs will be tripped at a selected reactor power level in order to evaluate the effect on the primary system, pressure control, and the main turbine generator.
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 11-3-97 at ~22% power:
- ST-0-60F-420-3 "Turbine Control Valve Fast Closure Scram Functional"
- ST-0-001-200-3 "Turbine Main Stop Valve Closure Functional"

In addition, the TSVs are tested weekly in accordance with RT-0-001-400-3.

2.23 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 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 11-2-97 in accordance with RT-I-002-230-3 "Recirculation Pump 30 Percent Speed Limiter In-Place Calibration".

The Recirc pump 45% speed limiters were set on 11-4-97 in accordance with RT-I-002-260-3 "Recirculation Pump 45 Percent Speed Limiter In-Place Calibration".

The Recirc M/G set high speed mechanical stops were placed in their final positions on 11-12-97 in accordance with GP-5 "Power Operations", when the target rod pattern/flow conditions were achieved.

2.24 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

No Recirc pump trips occurred during the BOC12 power ascension. During power ascension, jet pu operability was checked daily and performance was trended in accordance with ST-0-02F-550-3 "Jet Pump Operability".

The flow instrumentation calibration was checked by performing RT-I-002-250-3 "Core Flow Verification" on 11-10-97.