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

SUBJECT: Peach Bottom Atomic Power Station Startup Report of Plant Startup following the 13th Refueling Outage of Unit 2

Gentlemen:

Enclosed is the Peach Bottom Atomic Power Station Startup Report for Unit No. 2 Cycle 14. The report is submitted pursuant to Unit 2 Technical Requirements Manual Appendix A.

Sincerely,

John Doering, Jr.
Vice President,
Peach Bottom Atomic Power Station

JD/PJD/CHM/SRH/JWH/dag

Enclosure

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PEACH BOTTOM ATOMIC POWER STATION

UNIT 2
CYCLE 14
STARTUP REPORT

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

DECEMBER
2000

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INTRODUCTION

Peach Bottom Atomic Power Station (PBAPS) Unit 2 Technical Requirements Manual Appendix A requires submittal of a Startup Report following an outage in which: 1) modifications were installed that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant, or 2) amendment to the license involving a planned increase in power level, or 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier. This report is being submitted due to the installation of several modifications during the 2R13 refueling outage, namely, installation of the Power Range Neutron Monitoring system and upgrade of the Rod Block Monitoring System(Mod P00507), Flow Instrumentation Upgrade (ECR 00-00138), and installation of Recirc MG Set Bailey Scoop Tube Positioners, also known as the Jordan Mod (Mod P00797).

This report summarizes the plant startup and power ascension testing performed to ensure that no operating conditions or system characteristic changes occurred during the thirteenth refueling outage of PBAPS Unit 2 which diminished the safe operation of the plant.

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

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

Peach Bottom Unit 2 was out of service from 9-15-00 to 10-05-00 to accommodate its thirteenth refueling outage. During this 20 day, 4 hr, 36 minute outage, 292 new GE14 fuel bundles were loaded into the core, with the balance of the core load being comprised of once and twice burned GE13 fuel bundles. The Cycle 14 core consists entirely of GE barrier fuel.

INTRODUCTION (continued)

This is the first application of the GE14 product line at PBAPS Unit 2. The GE14 fuel type has been approved for use by the NRC. GE14 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. There are some differences between GE14 and the GE13 fuel which makes up the balance of the loaded fuel. These differences as detailed in section 2.3 (Fuel Loading). GE14 fuel complies with all required fuel design and licensing bases during steady-state, transient, and accident conditions.

Other in-vessel maintenance performed during the outage included:

- Replacement of 14 control rod drives.
- Replacement of 14 Local Power Range Monitors (LPRMs).
- Replacement of 15 control rod blades.

Also, an inspection campaign was undertaken to determine the effect of previous cycle's Noble Chemical addition on nuclear fuel corrosion. 5 bundles were inspected using visual and eddy current testing. Crud scrapings were also taken. The details and results of the poolside examination are documented on page 06, section 2.1. The data indicates that Unit 2 may continue operations with no impact to fuel rod thermal-mechanical design and license limits. The unit may continue operation in a safe manner, within current design assumptions.

Unit 2 returned to service on 10-05-00 and reached steady-state full power for the first time in Cycle 14 on 10-08-00. Startup testing was completed on 10/13/00.

The successfully implemented startup test program ensures that the thirteenth refueling outage of Unit 2 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-095-886-2 "Chemistry Preparation for Reactor Startup" on 10-06-00. 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-095-824-2 "Reactor Startup Chemistry With Steaming Rates Less Than 100,000 Lbs/Hr", performed on 10-05-00.

2.1 Chemical and Radiochemical (continued)

At high steaming rates, ST-C-095-823-2 "Conductivity and Chloride Ion Content in Primary Coolant During Normal Operation" was performed at least every 4 days after reaching 850 psig reactor pressure. This test verified that the conductivity was less than or equal to 5 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 10-04-00. The steam jet air ejector discharge activity indicated that Unit 2 began Cycle 13 with no fuel failures. Subsequent analysis of radioisotopic 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.

Fuel Bundle Inspection Campaign:

Background:
An inspection campaign at Peach Bottom Unit 2 End of Cycle (EOC) 13 was undertaken to determine the effect of NobleChem on nuclear fuel corrosion performance. This was pursued at Peach Bottom Unit 2 EOC 13 due to the relatively high NobleChem injection applied at End of Cycle 12 (approximately 4.5 kg) and the high duty that Peach Bottom fuel experiences as a result of high power density, two year cycles and high capacity factors. This inspection campaign was sponsored by the BWRVIP.

Inspection Campaign Scope:
The final scope consisted of 5 fuel bundles. Each bundle had a peripheral visual examination; at least four rods (in each bundle) were brushed and visually examined; at least four fuel rods per bundle had oxide thickness measurements via eddy current. The first three fuel bundles (the original scope) also had crud scrapings taken for later analysis.

The initial three fuel bundles consisted of:
- one cycle fuel bundle discharged at the end of Cycle 13, inspected to determine effect of re-deposition of NobleChem

2.1 Chemical and Radiochemical (continued)

•third cycle fuel bundle discharged at the end of Cycle 12, to benchmark of high burnup non-NobleChem fuel performance at Peach Bottom
•third cycle fuel bundle discharged at the end of Cycle 13. bundle representative of high burnup NobleChem fuel performance at Peach Bottom

The expanded scope bundles included:
•second cycle fuel bundle discharged at the end of Cycle 13, inspected to determine state of third cycle fuel at Beginning of Cycle (BOC)14
•third cycle fuel bundle discharged at the end of Cycle 13. verify results of earlier third cycle bundle inspection

Poolside Examination Observations:
The first cycle fuel bundle crud loading appeared normal with no unusual features. The first cycle bundle corrosion was noted to be uniform and well within the GNF data base for non-NobleChem conditions. The third cycle fuel bundle that was discharged at EOC 12 had an observed crud loading that also appeared normal with no unusual features. The measured corrosion levels were also noted to be well within the GNF data base for non-NobleChem conditions. The third cycle fuel bundle that was discharged at EOC 13 also exhibited a normal appearing crud layer with no unusual crud features. However, the corrosion on this third cycle bundle was notably thicker than the other bundles measured and exhibited limited localized oxide delamination and separation in several areas. Although this advanced corrosion condition was observed at only a few limited locations, this observation was not expected and represents a departure from normal experience. Even though the corrosion values measured on the EOC 13 discharged bundle were beyond the normal experience base, the distribution of oxide thickness measurements is generally consistent with that applied in the reference fuel rod thermal-mechanical design and licensing analyses. These observations and supporting eddy current information were the drivers for expanding the inspection campaign.

In the follow-on examinations, the second cycle fuel bundle exhibited normal crud characteristics with corrosion performance consistent with the GNF data base. The inspection of the second third cycle bundle was performed to confirm to the observations made from the examination of the first third cycle bundle discharged at the end of Cycle 13. The inspection results were very much like that previously noted: no unusual crud features but thicker corrosion with limited oxide delamination and separation. Again, although the second third cycle bundle corrosion values were beyond normal experience, the distribution of oxide thickness values is generally consistent with that applied in the reference fuel rod thermal-mechanical design and licensing analyses for this fuel.

Conclusions that can be drawn from the Data:

The exhibited condition may be considered an exposure related phenomenon and thus it can be safely assumed that there is no immediate impact regarding Peach Bottom fuel.

The following positions can be made:

All fuel reached End of Cycle 13 without failure.

All fuel associated with the bundles exhibiting accelerated corrosion (i.e. limited oxide delamination and separation) has been discharged.

Eddy current measurements have demonstrated margin to the fuel rod thermal-mechanical design and licensing analyses for the affected fuel.

With the data available, the indications are that successful operation will continue for P2C14 with no specific impact on fuel rod thermal-mechanical design and license limits.

Thus, current operation of the PBAPS Unit 2 Cycle 14 core is safe and within current design assumptions and inputs.

Further Actions:

The BWRVIP and GNF will be reviewing the poolside examination data in detail. The crud scraping data will be analyzed and incorporated with the poolside data by the end of 2000. It is through the actions of the review task force that the root cause of the third cycle fuel cladding accelerated corrosion is expected to be determined. A comprehensive report will be available in early 2001.

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-215. During the refueling outage and subsequent plant startup, appropriate radiation surveys were performed to generate Radiation Work Permits per HP-C-210 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 130.0 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 WRNMs 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 at all times. 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.

Results

The fuel shuffle was performed in accordance with FH-6C "Core Component Movement - Core Transfers" and was completed on 10-28-00. The final loading pattern includes 292 new GE14 fuel bundles, 292 once-burned GE13 bundles, and 180 twice-burned GE13 bundles. Unit 2 no longer uses GE11 fuel. The complete Cycle 14 core consists of barrier fuel.

To ensure 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 was performed on 9-15-00, 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 was performed on 9-29-00, in accordance with M-C-797-020 "Core Verification."

Serial number and location verification of the 2R13 discharged fuel in the fuel pool were performed prior to reaching 25% power on 10-03-00, in accordance with M-004-117 "Post Refuel Outage Spent Fuel Verification."

2.3 Fuel Loading (continued)

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-0-003-465-2 "Control Rod Withdraw Tests", completed on 10-04-00. The acceptance criteria for this test was met when the actual shutdown margin was demonstrated with a fully loaded core in accordance with ST-R-002-910-2, performed on 10-05-00.

292 bundles of GE-14 advanced 10X10 matrix fuel bundles were installed in the reactor. ECR-99-02682 justified and documented the technical acceptability of using GE14 fuel in PBAPS reactors. The potential impact on the following systems was analyzed: loose parts, accident and transient analysis results, thermal limits, off rated multipliers (ARTS/MELLA), spent fuel pool racks, spent fuel pool cooling, fuel handling equipment, receipt inspection, Appendix R, and EPGs, TRIP procedures, heavy loads, operator training, RECT lesson plans, noble metal impact, dose rate, impact, post LOCA hydrogen generation feedwater heater asymmetry, containment response, vessel fluence. The ECR 10CFR50.59 further addressed the acceptability of GE-14 fuel in PBAPS reactors.

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-002-910-2 "Shutdown Margin" on 10-05-00. Control rods were withdrawn according to the startup sequence per GP-2-2 Appendix A2. WRNM count rates were recorded after each control rod withdrawal. The reactor was declared critical at 2226 on 10-04-00 with RWM Group 2 control rod 26-23 at position 18, RWM step 44. Reactor water temperature was 185 degrees F. Count rate doubling time was 186.6 seconds, and the calculated reactor period was 268.7 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 1.34% 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 14, a SDM adder of 0.15% ΔK/K was applied; therefore, the SDM value for reactor temperatures down to 38 degrees F. is (1.34 - 0.15)%, or 1.19% ΔK/K. The difference between the predicted and actual SDM values is calculated as $\Delta SDM = SDM_{actual} - ((SDM_{predicted} - R))$, where R is the maximum decrease in SDM from BOC: (1.34 - (1.63-.42)), or 0.013% 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.

Results

Each CRD had its normal insert speeds, withdraw speeds and coupling integrity checked by ST-0-003-465-2 "Control Rod Withdraw Tests", completed on 10-04-00. 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-003-460-2 "CRD Scram Insertion Timing for All Operable Control Rods", completed on 10-10-00. All 185 rods had satisfactory scram times prior to exceeding 40% power.

During power ascension, ST-0-003-470-2 "CRD Coupling Integrity Test" was performed to verify CRD/blade coupling integrity for each

2.6 Control Rod Drives (cont.)

control rod when it is fully withdrawn. This test was completed on 10-15-00.

During power ascension, when reactor power was above the RWM LPSP (approximately 23%), ST-0-003-560-2 "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-0-003-561-2 "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-2-2 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-2-2 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-C-01 "Reactor Engineering General Instructions".

Results

Cold criticality was achieved on 10-04-00 by withdrawing rods in accordance with GP-2-2 Appendix A2. This same sequence (Startup 2) had previously been verified in the RWM in accordance with ST-R-62A-220-2 "RWM Sequence Verification", performed on 8-17-00. Prior to withdrawing the first rod, ST-0-62A-210-2 "RWM Operability Check" was performed on 10-04-00. Criticality occurred on rwm sequence step 44 in RWM Group 2. The critical rod pattern is recorded in GP-2-2 Appendix A2 and ST-R-002-910-2 "Shutdown Margin".

PBAPS Unit 2 will operate in the A2 control rod sequence for approximately the first four months of Cycle 14 and then will be swapped to the A1 control rod sequence. These two sequences will be alternated every 4 months for the remainder of Cycle 14.

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. Two follow-up load drops were undertaken to set the final rod pattern. The final target rod pattern was set on 10/12/00 at 08:31.

During the numerous control rod movements performed during the startup, no thermal limit violations occurred and thermal limits remained below the administrative limit of 0.98 at all times.

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 were replaced with the Wide Range Neutron Monitor (WRNM) system.

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 surveillance tests. WRNM scram setpoints were verified by performance of SI2N-60C-WRNM-A(through H)1C2 "WRNM Channel A (through H) Calibration/Functional Check. In addition, WRNM signal to noise ratio check was performed per SI2N-60C-WRNM-A(through H)1MX. WRNM minimum count rate was determined to be greater than 3 CPS prior to control rod withdraw on 10-04-00.

WRNM D & A experienced spiking during the outage and startup. Both were INOP independently and concurrently during different portions of the outage and startup. The D WRNM cause a "B" Channel half scram without a valid actuation signal (it spiked). During the outage, 2D WRNM was much more sensitive to noise than the other WRNMs. The installation of ferrite beads to the input wires helped to greatly reduce the spiking. PEP I0011899.

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-002-910-2. WRNM response during power ascension was monitored and verified in accordance with GP-2.

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 readings will be proportional to the average flux in the four adjacent fuel assemblies at the LPRM detector elevation.

Results

ST-I-60A-230-2 "LPRM Gain Calibration" was performed on 10-16-00 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.

There were fourteen LPRM string (56 detectors) replacements performed during 2R13.

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 Core Thermal Power. The absolute difference between the APRM channels and calculated core thermal power is verified to be ≤2% CTP, while operating at ≥25% CTP.

Results

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

- SI2N-60A-APRM-11C2 (through 41C2) "Calibration/Functional Check of Average Power Range Monitor (APRM) 1 (through 4)"
- SI3N-60A-APRM-11FS (through 41FS) "Functional Check of Power Range Monitor (APRM) 1 (through 4)"

Numerous APRM calibrations were performed in accordance with ST-O-60A-210-2 "APRM System Calibration During Two Loop Operation" throughout power ascension. The first APRM gain calibration was performed on 10-05-00 at ~5% power and the last APRM gain calibration was performed on 10-12-00 at 100% power.

The APRMs were calibrated to within plus or minus 2% of indicated core thermal power during the power ascension. PBAPS Tech Specs require that the absolute difference between APRM channels and the calculated thermal power is less than or equal to 2% CTP when operating above 25% core thermal power.

All 4 APRM channels were operable for the initial BOC startup.

Power Range Nuclear Monitor (PRNM) Performance

Objective

Mod P-507 essentially replaced all the APRM, LPRM, and RBM signal processing hardware in the main control room with GE NUMAC based equipment similar to that used for the WRNM system. LPRM detectors and associated field cabling was not replaced however.

Power Range Nuclear Monitor (PRNM) Performance (cont.)

This new PRNM system also employs the approved BWROG Option III Instability Detect and Suppress Algorithm as a long term solution for reactor thermal-hydraulic instability. The trip and rod block logic associated with this feature is bypassed during its first cycle of operation in order to perform tuning and assess the system's operation.

The objective was to demonstrate that the new PRNM instrumentation could track core neutron flux and recirculation drive flow during power ascension and provided adequate information to the operator.

Description

LPRM and APRM flux response was verified as Mode-1 was approached. Following initial APRM gain calibration to BPV position, subsequent APRM gain calibrations were performed against a formal heat balance via direct digital download of core thermal power values from 3D Monicore to the APRMs. LPRM gain calibration was also performed via down load of LPRM GAFT values from 3D Monicore.

Acceptance Criteria

Indicated APRM power must be within 2% absolute of formal heat balance once reactor power is greater than or equal to 25%. In addition, PRNM power and associated Recirc Drive Flow indication was monitored throughout Mode-1 power ascension and verified to conform to historical data trends.

Results

Prior to startup, PRNM performance was tested via several surveillance tests and mod acceptance tests. APRM electronic calibration, trip and rod block setpoints and logic were verified by SI2N-60A-APRM-1(through 4)1C2, "APRM Channel 1 (through 4) Calibration/Functional Check." Calibration check of all new Recirc drive flow transmitters and associated processing loops was accomplished via performance of SI2N-60A-110-AE(through DH)C2 "Calibration Check of APRM 1 (through 4) Flow Bias Signal." RBM calibration and logic checks were performed via SI2N-60B-RBM-A(B)1C2 "RBM Channel A(B) Calibration/Functional Check." Various plant interface and computer point verifications were made via MAT P507.A-2 "PRNMS Wiring and Logic Checks" and MAT P507.B-2 "Verification of PMS Changes For PRNM System Installation". All four APRM and both RBM channels were operable for BOC13 startup.

Power Range Nuclear Monitor (PRNM) Performance (cont.)

During startup, PRNM operability was verified in accordance with GP-2 "Normal Plant Startup." LPRM flux response and APRM downscale indications were verified to clear as mode-1 was approached. APRM gain adjustments were performed utilizing ST-O-60A-210-2 "APRM Gain Calibration for Two Loop Operation". During mode-1 power ascension, tuning of the Option III Period Based Algorithm (PBA) of the Oscillation Power Range Monitor (OPRM) was performed at various steady-state points on the power to flow map. APRM and LPRM gain calibrations were performed via direct data download from 3D Monicore to the APRM system. All tests were acceptable.

2.11 Process Commuter

Objective

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 BOC14 databank was installed and verified in accordance with FM-UG-270, "Process Computer Databank Review", RE-C-28, "NSSS Software BOC Databank Update", and RE-C-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-500-2 "Checkout of the NSSS Computer Calculation of Core Thermal Power" at approximately 100% power on 10-17-00.

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-0-013-200-2 on 10-05-00. A cold quick start at rated reactor pressure was performed in accordance with ST-O-013-301-2 on 10-11-00.

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 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 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 9-26-00 in accordance with RT-N-023-240-2.

A controlled start was performed at 175 psig reactor pressure in accordance with ST-0-023-200-2 on 10-05-00. A cold quick start at rated pressure was performed in accordance with ST-0-023-301-2 on 10-06-00. 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 BOC14 power ascension. Prior to placing the second Recirc pump in service, all temperature requirements specified in SO 2A.1.B-2 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-0-080-500-2 "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 performed on 10-02-00, drywell piping was visually inspected at between 980 and 1030 psig. No blocking or interference of piping due to thermal expansion was observed.

The Integrated Leak Rate Test (ILRT) was completed SAT on 10/04/00. This test is of the Primary Containment at peak calculated design basis post-accident pressure per tech spec SR 3.6.1.1.1. Under 10 CFR50 App. J., Option B, the Type A test shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. The initial "Final As Left" ILRT Leakage Rate was below the acceptance criteria of 0.375%/day which allowed the unit to startup. The "Performance" ILRT leakage rate was below 0.5%/day (0.3365%/day). Therefore, the surveillance will remain at every 10 years.

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 32-23) 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-C-06 "Core Power Symmetry and TIP Reproducibility Test" was performed at 100% power on 10-13-00. Total TIP uncertainty was less than 1.58%, therefore, the acceptance criteria of RE-C-06. The maximum deviation between symmetrically located pairs (pair 43/33) was 5.66%, at node 11. The random noise component was 0.64% and the geometric noise component was 1.44%.

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 and the results were SAT.

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 in accordance with GP-2 and RE-C-01. The core thermal power heat balance and core flow values were verified by performing RT-R-59C-500-2 on 10-17-00 and RT-I-002-250-2 "Core Flow Verification" on 10-31-00.

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-002-900-2 "Reactivity Anomalies" on 10-13-00.

Flow instrumentation was upgraded as part of ECR 00-00138. Square root extracters, input summers, power supplies, and other equipment of antiquated design were replaced with newer models. Digital signal processors lowered loop uncertainty. Testing of the new equipment was completed SAT and is detailed in A/R# A1274941 (acceptance test plan).

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-250-2 "Reactor Water Level Instrument Perturbation Test", a monthly test, was performed satisfactorily on 10-04-00.

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-0-001-409-2, performed on 10-03-00. 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

The PBAPS Tech Specs and IST program have been revised such that MSIV full closure testing at power is no longer required. MSIV closure testing is now performed in Cold Shutdown.

During the outage, each MSIV was stroked satisfactorily in accordance with ST-M-OlA-471-2, performed on 09-30-00. During the initial startup, each MSIV was opened in accordance with GP-2 and SO 1.A.1.A-2.

MSIV individual closure timing and continuity checks are performed quarterly when in Cold Shutdown per ST-0-07G-470-2 and were performed on 09-29-00. 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 leak-tightness 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-M-016-220-2 "Main Steam Relief Valve Actuator and Backup N_2 Supply Valve Functional Test". This test was performed on 09-28-00.

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 10-06-00 at ~22% power:

- ST-0-60F-420-2 "Turbine Control Valve Fast Closure Scram Functional"
- ST-0-001-200-2 "Turbine Main Stop Valve Closure Functional"

In addition, the TSVs are tested monthly in accordance with RT-0-001-400-2.

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 10-06-00 in accordance with RT-I-002-230-2 "Recirculation Pump 30 Percent Speed Limiter In-Place Calibration".

The Recirc pump 45% speed limiters were set on 10-07-00 in accordance with RT-I-002-260-2 "Recirculation Pump 45 Percent Speed Limiter In-Place Calibration".

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 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 unexpected Recirc pump trips occurred during the BOC13 power ascension. Dual Recirc pump trips as described in the UFSAR at 50%, 75%, and 100% CTP have not been required since Peach Bottom's 1st cycle startup, since that initial benchmarking data is still valid. The Unit/Cycle specific supplemental reload licensing report (SRLR) describes GE's analysis of the dual pump trip (and other related transients). Recirc pump trips were conducted as part of the RPT mod as described below.

During power ascension, jet pump operability was checked daily and performance was trended in accordance with ST-0-02F-560-2 "Daily Jet Pump Operability".

Recirc system baseline data was obtained during power ascension in accordance with RT-O-02A-210-2 "Recirc System Baseline Data - 2 Loop Operation".

The flow instrumentation calibration was checked by performing RT-I-002-250-2 "Core Flow Verification" on 10-31-00.

2.24 Recirculation System (cont.)

Discussion of Recirc System modifications installed during 2R13:

The 2A(B) Recirc MG Set Bailey Scoop Tube Positioners were replaced with new positioners under Mod P00797 due to chronic problems and unavailable replacement parts. The new positioners were satisfactorily tested prior to start-up and during power ascension to maximum pump speed.

The EOC-RPT System is in operation. The system reduces the severity of the thermal effects on the fuel during plant pressurization transients, e.g. turbine/generator trip or load rejection. Both Recirc pumps are tripped upon receipt of a TSV closure or TCV fast closure signal from RPS. Disconnecting the Recirc pumps from the M/G sets removes system inertia normally provided by the M/G sets and allows the pumps to coast down quicker. The resulting rapid core flow reduction increases core void content and thereby reduces reactivity in conjunction with the control rod scram. The EOC-RPT is composed two ABB 4.16 KV circuit breakers connected in series between each Recirc M/G set generator and Recirc pump motor. These breakers provide a redundant means of tripping each Recirc pump upon receipt of a logic signal initiated from the Reactor Protection System (RPS) on a TSV closure or TCV fast closure signal. An automatic bypass of the EOC-RPT is applied whenever reactor thermal power is less than 30% as sensed by turbine first stage pressure. The system is in effect above 30% power.

Surveillance of the EOC-RPT system was performed by the following procedures:
ST-I-02A-100-2, End Of Cycle RPT I System Logic System Functional Test 9/27/00,
ST-I-02A-105-2, End Of Cycle RPT II System Logic System Functional Test 9/27/00,
ST-O-60F-425-2, Turbine Control Valve Fast Close Scram & EOC-RPT, 10/06/00, and
ST-R-02A-400-2, End of Cycle-Recur Pump Trip Response Time, 10/06/00.