U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-318/84-21

Docket No. 50-318

License No. DPR-69 Priority -- Category C

Licensee: Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Facility Name: Calvert Cliffs 2

Inspection At: Lusby, Maryland

Inspection Conducted: July 30 - August 3, 1984

Inspectors: Peter C. Wen 8/24/84
P. C. Wen, Reactor Engineer date

Approved by: L. H. Bettenhausen 9/9/84
L. H. Bettenhausen, Chief date
Test Program Section

Inspection Summary: Test Program Section.

Areas Inspected: Routine, unannounced inspection of start-up testing following refueling of Unit 2, Cycle 6 and the water hammer event (April 22, 1984) follow-up. The inspection involved 36 hours on-site by one region based inspector.

Results: In the areas inspected, no items of noncompliance were identified.

DETAILS

1. Persons Contacted

R. Androsik, Electric and Controls Surveillance Test Coordinator
*M. E. Bowman, Principal Engineer, Incore Fuel Management
*J. T. Carroll, General Supervisor, Operations
M. L. Coon, Training Instructor
*J. B. Couch, Engineer, Incore Fuel Management
L. Decker, Engineer, Technical Support
K. M. Hoffman, ISI Engineer, Technical Support
J. Luhr, Operations Surveillance Test Coordinator
G. S. Pavis, Supervisor, Electric Engineering Department
W. E. Putman, Quality Assurance Specialist
*R. P. Heibel, Principal Engineer, Technical Support

*Denotes those present at the exit interview of August 3, 1984.

2. Unit 2, Cycle 6, Start-up Testing Program

The start-up test program was conducted according to test procedures (i) PSTP-2, "Unit 2, Cycle 6, Initial Approach to criticality and Low Power Physics Testing", Rev. 6 and (ii) PSTP-3, "Unit 2, Cycle 6, Escalation to Power Test Procedure", Rev. 6. The test sequence outlined the steps in the test program, set initial conditions and prerequisites, specified calibration or surveillance procedures at appropriate points in the sequence and referenced detailed test procedures and data collections in appendices.

Initial criticality of Unit 2, Cycle 6, was achieved on June 29, 1984. The power ascension testing was completed about July 23, 1984. The inspector independently verified that the predicted values and acceptance criteria were obtained from "Calvert Cliffs Unit 2, Cycle 6, Start-up Test Predictions and Core Data", BG&E-84-219, dated June 18, 1984. The inspector reviewed test results and documents described in this report to ascertain that post start-up testing was conducted in accordance with technically adequate procedures and as required by Technical Specification (TS). The details and findings of the review are described in Sections 3 and 4.

3. Unit 2, Cycle 6, Start-up Testing - Precritical Tests

The inspector reviewed calibration and functional test results to verify the following:

-- Procedures were provided with detailed instructions;

-- Technical content of procedure was sufficient to result in satisfactory component calibration and test;

-- Instruments and calibration equipment used were traceable to the National Bureau of Standards;

-- Acceptance and operability criteria were observed in compliance with TS.

The following tests were reviewed:

3.1 CEA/CEDM Performance Test

CEA/CEDM performance tests were performed in accordance with procedure PSTP-2, Appendix B "CEA/CEDM Performance Test" on June 27-28, 1984. CEA drop times were measured at hot full flow conditions. All CEAs reached a 90% insertion in less than 3.1 seconds as required by the TS. The inspector reviewed several visicorder traces and verified that the drop times had been interpreted correctly.

The distinct CEA traveling path time history was not observed on CEA-52. However, the measured 100% insertion time was 2.50 seconds which is less than that required for 90% insertion time of 3.1 seconds. This anomaly was attributed to a faulty reed switch stack.

Indicators and alarms associated with CEA withdrawal and position indication/deviation were also tested. CEA Position Indication from Computer Pulse Counting System and both Metras' agreed within ± 4.5 inches.

No items of noncompliance were identified.

3.2 Reactivity Computer Setup/Verification

The reactivity computers were set up and calibrated according to procedure PSTP-11, Revision 7, on June 15-19, 1984. Both reactivity computers (Westinghouse and General Atomic) were adjusted with the corrected inputs of delayed neutron fractions (betas) and decay constants (lambdas). A step change signal was fed into the reactivity computers. The output signal was then compared with predicted values which were derived from point reactor kinetics. The results of this calibration check were satisfactory.

The reactivity computers were further checked in accordance with procedure PSTP-2, Appendix G "Reactivity Computer Calibration Check" when the reactor was critical. Comparisons of predicted and measured reactivities based on doubling and e-folding times were acceptable.

No unacceptable conditions were identified.

4. Unit 2, Cycle 6, Start-up Testing - Post-Critical Tests

4.1 The inspector reviewed selected test programs to verify the following:

-- The test programs were implemented in accordance with Cycle Refueling Sequencing Procedures;

-- Step-wise instructions of test procedures were adequately provided including Precautions, Limitations and Acceptance Criteria in conformance with the requirements of the TS;

-- Provisions for recovering from anomalous conditions were provided;

-- Methods and calculations were clearly specified and the tests were performed accordingly;

-- Review, approval, and documentation of the results were in accordance with the requirements of the TS and the licensee's administrative controls.

4.2 Low Power Physics Tests

4.2.1 Critical Boron Measurements

The licensee measured the critical boron concentration in accordance with test procedure PSTP-2. The inspector reviewed the test data and noted the following results:

Configuration	Predicted Value (ppm)	Measured Value (ppm)
All Rods Out (ARO)	1590 ± 50	1630
CEA Groups 5, 4, 3, 2, 1 Full In	1336 ± 50	1361

Test results were within acceptance criteria.

4.2.2 Isothermal Temperature Coefficient

Isothermal temperature coefficients were measured in accordance with the procedure specified in PSTP-2, Appendix C. The inspector noted the following results.

Configuration	Predicted Value (10^{-4} delta Rho/°F	Measured Value (10^{-4} delta Rho/°F
CEA Group 5 at 105"	0.29 ± 0.30	0.3885
CEA Groups 5, 4, 3, 2, 1 Full In	-0.50 ± 0.30	-0.3806

The predicted values are based on a slightly different boron concentration than the measured condition. the overall impact on the above comparison is very minor, In the case of CEA Group 5 at 105 inches, the estimated impact is only about 0.06 x 10^{-4} delta Rho/°F.

The Isothermal Temperature Coefficient (ITC) is defined as the change in reactivity for a unit change in the moderator, clad and fuel pellet temperatures. Thus, the ITC can be interpreted as the sum of the moderator and Doppler coefficients. The Doppler coefficient is difficult to measure in normal operation. A value of -0.15×10^{-4} delta Rho/°F was obtained from CE letter BG&E-84-219. Thus, the Moderator Temperature Coefficient (MTC) were determined as follows:

Configuration	Measured MTC (10^{-4} delta Rho/°F)	TS Limits (10^{-4} delta Rho/°F)
CEA Group 5 at 105"	0.539	-2.5 < MTC < 0.5
CEA Groups 5, 4, 3, 2, 1 Full In	-0.231	-2.5 < MTC < 0.5

The measured MTC (0.539×10^{-4} delta Rho/°F) at Essentially All Rod Out (EARO) condition, exceeded the TS limit (0.50×10^{-4} delta Rho/°F). However, the requirements for the TS special test exceptions were met, and the MTC TS limits were suspended during the performance of Low Power Physics Testing. The high MTC value was anticipated since Cycle 6 was designed for extended burnup operation with no poison included in the fresh fuel assemblies. The licensee incore fuel management performed an evaluation, and established an administrative operational limit on boron concentration. The result was presented to the PCSRC (Meeting #84-94, June 30, 1984) and subsequently received its approval.

The inspector reviewed the calculation and determined that the licensee actions were technically adequate to mitigate the high positive MTC condition. The inspector also toured the control room and verified that this new administrative limit was in use by the reactor operators.

The inspector had no further questions.

4.2.3 CEA Worth Measurement

The CEA worth measurements were performed in accordance with the procedure PSTP-2, Appendix D. The licensee conducted integral non-overlapped group worth measurements using conventional boron dilution techniques and overlapped group worth measurement using the boration method. The following results were obtained:

Non-overlapped Measurement

Group	Predicted Worth (% delta Rho)	Measured Worth (% delta Rho)
5	0.420 ± 0.063	0.394
4	0.233 ± 0.035	0.221
3	0.615 ± 0.092	0.596
2	0.403 ± 0.060	0.380
1	0.754 ± 0.113	0.687
Total	2.425 ± 0.242	2.278

Overlapped Measurement

Group	Predicted Worth (% delta Rho)	Measured Worth (% delta Rho)
5, 4, 3, 2, 1	2.425 ± 0.242	2.296

Test results were within acceptance criteria.

No unacceptable conditions were identified.

4.3 Power Ascension Tests

4.3.1 Core Thermal Power Evaluation

The licensee's procedure, PSTP-3, Appendix E "Core Calorimetric Verification" was reviewed for technical adequacy. The inspector reviewed the data from measurements performed on July 3, 1984 (46% RTP) and July 22, 1984 (97% RTP). In both cases, the plant computer calculated values (XC009) were in agreement with the manual calculation results (per procedure OI-30) with difference of less than 1.5%.

The inspector also reviewed the control room operator log for thermal power surveillance performed from July 2 to July 31, 1984, and verified that the frequency of evaluation and excore power range channel calibrations were performed within the requirements as prescribed by the TS.

No discrepancies were identified.

4.3.2 Power Coefficient

Power coefficient measurements were made in accordance with PSTP-3, Appendix B "Variable TAVG Test". Measurements were made at 50% and 97% RTP. Results as identified by the inspector were:

RTP	Predicted Value (10^{-4} delta Rho/%Power)	Measured Value (10^{-4} delta Rho/%Power)
50%	-1.00 ± 0.2	-1.092
97%	-0.88 ± 0.2	-0.932

Isothermal temperature coefficients were also measured at 50% and 97% RTP by maintaining power level constant while varying tavg. The results are as follows:

RTP	Predicted Value (10^{-4} delta Rho/°F)	Measured Value (10^{-4} delta Rho/°F)
50%	-0.03 ± 0.3	-0.023
97%	-0.23 ± 0.3	-0.298

Moderator temperature coefficients were calculated based on ITC measurements and information from CE letter BG&E-84-219.

RTP	Measured MTC Value (10^{-4} delta Rho/°F)	TS Limits (10^{-4} delta Rho/°F)
50%	0.097	-2.5 < MTC < 0.5
97%	-0.168	-2.5 < MTC < 0.2

No unacceptable conditions were identified.

4.3.3 Core Power Distribution

The procedure and method used by the licensee to verify that the plant is operating within the power distribution limits defined in TS were reviewed and discussed with cognizant licensee personnel. Forty-five (45) fuel assemblies are instrumented with self-powered neutron flux detectors. Each of the 45 detector strings is composed of four rhodium detectors. The data taken by the Incore Detector System was analyzed by the plant computer using the CE "INCA" code.

The licensee performed 12 successful test cases prior to entering INCA into the plant computer for Cycle 6 operation. The inspector reviewed portions of INCA library inputs and verified that various uncertainty factors and flux peaking augmentation factors were included in the setting of the incore detector local power density alarms as required by the TS.

The results from power maps which were taken to support the Cycle 6 start-up physics testing are tabulated below:

	50% RTP		100% RTP	
	Measured	Acceptance	Measured	Acceptance
Total Planar Radial Peaking Factor, F^T_{xy}	1.680	<1.785	1.599	<1.700
Total Integrated Radial Peaking Factor, F^T_r	1.597	<1.720	1.520	<1.650
Azimuthal Power Tilt T_q	0.011	<0.030	0.004	<0.030
Peak Linear Heat Rate, PLHR, KW/ft	6.01	15.50	11.36	15.50

No items of noncompliance were identified.

4.3.4 Excore/Incore Calibration

Excore/Incore calibration was performed at 48% RTP on July 3, 1984. I&C personnel performed the calibration per surveillance test procedure STP-M-213-2. The calibration was performed by comparing INCA output with responses from power range detectors. The second calibration in I&C's monthly surveillance program was performed at 100% RTP on July 30, 1984.

No unacceptable conditions were identified.

5. QA Role in Cycle 6 Start-up Testing

The inspector discussed the subject of QA's role in Cycle 6 start-up testing with station QA engineer. The inspector was told that QA independently verified the core loading activities and audited the post start-up testing program. The inspector reviewed the draft audit report and noted that there were no outstanding open items left as a result of these QA activities.

The inspector had no further question.

6. Water Hammer Event of April 22, 1984

6.1 Steam Generator Water Hammer

During normal plant operation, feedwater enters the steam generator (SG) through the feedwater nozzle where it is distributed via a feedring sparger. Water exits the feedring into SG downcomer through Aperture in the top ("J tubes"). The purpose of these "J tubes" is to keep the feedring from draining, and thus reduce the potential of water hammer. However, "J tubes" alone can not completely prevent drainage because there is a small clearance between the thermal sleeve and the SG feedwater inlet nozzle.

Under certain plant transients, if main feedwater is interrupted and no auxiliary feedwater flow is established, the SG water level may drop below the bottom of the feedring. This allows steam to enter the feedring and feedline. Upon recommencement of main feedwater flow, the steam in the line could be trapped due to some piping configurations. Once trapped, the cooler feedwater inlet flow would cause the steam bubble to collapse and result in water hammer. The pressure waves generated as a result of water hammer then propagates through the piping system. Depending on the type and magnitude of the pressure wave, localized stressing of the system may occur and, in severe cases, may cause failure of the system boundary and/or damage to adjacent supports.

6.2 Event Description

The SG is normally operating at 0 inch indicated level. This level is above the feedring (center line at about (-)47 inches) and thus keeps it full. The water level is maintained by utilizing feedwater regulating valve or by-pass valve.

At midnight April 21, 1984, the unit was in hot standby (Mode 3) with RCS pressure of 1400 psia, RCS temperature of 405°F and steam generator pressure of 260 psia. The unit was cooling down in preparation for its fifth refueling outage. Due to a leakage problem in both feedwater regulating valve (2-CV-1111) and by-pass valve (2-CV-1105) the SG level was being maintained by opening and closing the feedwater isolation valve (2-MOV-4516). In preparation for testing of the Auxiliary Feedwater System (TSP-165) and because of the need to add chemicals to the SG, the SG level was later lowered to -40 inches. When the desired level (-40 inches) was attained, the feedwater isolation valve was shut. Approximately 30 minutes later, the operator discovered that the SG level dropped to -65 inches and the feedring was completely uncovered. In attempt to recover the water level, the operator opened the feedwater isolation valve and the water hammer occurred immediately.

6.3 Licensee Actions

Following the event, the licensee conducted an investigation, examined components, assessed the damages and took corrective actions. This information was sent to the NRC on July 20, 1984. (letter from R.E. Denton (BG&E) to D. H. Jaffe (NRC)).

The inspector independently verified the extent of the problem and licensee's actions as follows:

-- The root cause of this event was attributed to operator failure to follow the procedure OI-12A for the Feedwater System. The Auxiliary Feedwater (AFW) system has a separate nozzle connected to SG. OI-12A instructs the operator to use AFW for feeding the SG any time when SG level is below -26 inches. The inspector reviewed procedures OI-12A Rev. 8 and EOP-1 (Reactor Trip), Rev. 15, and noted that the instruction has been emphasized in this area. In conjunction, a memorandum from the General Supervisor - Operations to licensed operation personnel was issued to reiterate the need for procedural compliance. The inspector also discussed the event subject with an operation training instructor. This event and related water hammer phenomena will be taught at the upcoming operator requalification training sessions scheduled in October and November, 1984.

-- Water hammer damage appeared limited to the feedwater system from the SG21 to the Feedwater Regulating Valve (2-CV-1111). Main Feedwater Isolation Valve (2-MOV-4516) motor operator was found broken off and the hand wheel was shattered. Other plant damage included a broken air actuator in the Main Feed Regulating Valve (2-CV-1111) and a cracked yoke at the top of Main Feed By-pass Valve (2-CV-1105). The licensee performed various tests including ultrasonic, liquid penetrant, pressure and visual examination on various system components and locations of high piping stress. No further damage was found. The licensee and his SG vendor also conducted SG21 entry for feedring and feedwater nozzle inspection. No damage was revealed. The damaged feedwater system components have been repaired since then. The inspector walked down the system and observed no abnormal conditions.

The inspector also discussed the subject of water hammer loading on the SG component design evaluation with a cognizant licensee engineer. The inspector was provided with the information that the fatigue usage factors for feedwater nozzle and feedring were 0.26 and 0.54, respectively. These values were calculated by CE based on plant design life operation conditions. In view of these low values and associated damages observed following the event, the additional dynamic loading introduced by the water hammer is not likely to jeopardize the plant safe operation. The inspector had no further question.

7. <u>Exit Interview</u>

Licensee management was informed of the purpose and scope of the inspection at the entrance interview. The findings of the inspection were periodically discussed and were summarized at the conclusion of the inspection on August 3, 1984. Attendees at the exit interview are denoted in paragraph 1.

At no time during this inspection was written material provided to the licensee by the inspector.