ENCLOSURE

POST REFUELING OUTAGE STARTUP TEST REPORT UNIT 1 CYCLE 13

EDWIN I. HATCH NUCLEAR POWER PLANT
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PLANT E. I. HATCH
UNIT 1
STARTUP TEST REPORT

1.0 INTRODUCTION

1.1 PURPOSE

This report consists of a summary of selected static and dynamic reactor core performance tests conducted prior to and during the beginning-of-cycle startup of Hatch Unit 1 Cycle 13.

1.2 PLANT DESCRIPTION

The Edwin I. Hatch Nuclear Power Plant Unit 1 is a General Electric design single-cycle boiling water reactor (BWR/4). Hatch Unit 1 is rated at 2436 MW(th) with a generator rating at this power of 810 MW(e). The plant is located on the south side of the Altamaha River, Southeast of the intersection of the river with U.S. Highway #1 in the Northwestern sector of Appling County, Georgia.

1.3 POST-REFUELING OUTAGE STARTUP TEST DESCRIPTION

The Edwin I. Hatch Nuclear Power Plant Unit 1 resumed commercial operation on 06/06/90 after completing a 109 day refueling/maintenance outage. The following core performance tests were performed as part of the post-refueling outage startup test program:

- Core Verification
- Control Rod Drive Friction Testing
- Control Rod Drive Timing
- Full Core Shutdown Margin Demonstration
- Cold Critical Eigenvalue Comparison
- Whole Core LPRM Calibration
- APRM Calibration
- Control Rod Scram Time Testing
- Reactivity Anomaly Calculation

The purpose for, a brief description of, and acceptance criterion for each of the tests listed above is enumerated in Section 3 of this report.

1.4 POST-REFUELING OUTAGE STARTUP TEST ACCEPTANCE CRITERIA

Where applicable, a definition of the relevant acceptance criteria for the test is given and is designated either "Level 1" or "Level 2". A Level 1 criterion normally relates to the value of a process variable which is used as the basis for the reload safety analysis with supplements previously submitted to the Nuclear Regulatory Commission and/or which are affected by the limiting condition for operation in the Unit's Technical Specifications.

A Level 2 criterion is associated with expectations related to the design performance of systems or components. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical technique used for the prediction would be initiated.

2.0 CYCLE DESIGN SUMMARY

2.1 CORE DESIGN SUMMARY

Cycle 13 was designed to operate approximately 412 effective full power days (EFPDs) at rated conditions, with an additional 8 EFPDs available from increased core flow. One hundred and eighty fresh fuel bundles were loaded in a conventional core configuration. With the exception of the four ANF 9x9 Lead Fuel Assemblies, all fuel assemblies loaded in the interior of the core in Cycle 13 have barrier cladding which permits the elimination of PCIOMRs. Control rod sequence exchanges are to be performed at core exposure increments of 2000 MWd/ST.

2.2 REACTIVITY/THERMAL LIMIT MARGINS

The two parameters which describe the global behavior of the core reactivity throughout the cycle are hot excess reactivity (HER) and cold shutdown margin (CSDM). The beginning-of-cycle (BOC) hot excess reactivity is 2.07%. This is also the peak HER of the cycle. The minimum predicted cold shutdown margin of 1.97% occurs at BOC.

LHGR design margins were relaxed slightly in an attempt to maximize cycle energy generation. Standard thermal margin design goals can be readily demonstrated at the expense of some cycle energy. Cycle 13 is the first cycle of operation at Plant Hatch to incorporate channel bow effects on thermal limits per NRC Bulletin 90-02. These effects will be implemented by adjusting the bundle R-factors such that the calculated bundle CPR will be increased by approximately 0.1 delta CPR.

2.3 FUEL SUMMARY/CORE LOADING PATTERN DESCRIPTION

Hatch 1 Cycle 13 is a conventional core loading which was designed to achieve 8900 MWd/ST. The loading pattern is quadrant mirror symmetric. The Reload 12 batch of fresh fuel contains a total of 180 bundles loaded in the interior of the core. These bundles are GE9B with 3.15 weight percent U-235. New features of the GE9B design (relative to GE7B) are:

2.3.1 A redesigned spacer for greater MCPR margin and reduced pressure drop,

2.3.2 A higher limit for linear heat generation rate,

2.3.3 A large central water rod for more efficient fuel utilization, 2.3.4 A greater fuel rod prepressurization and enrichment for higher discharge burnup,

2.3.5 Axial zoning of uranium enrichment and gadolinia concentration for power shaping and improved fuel efficiency, and,

2.3.6 Redesigned upper and lower tie plates for improved bundle flow.

Design features such as axial uranium enrichment and gadolinia concentration are optional and are not included in the design of the Reload 12 bundles utilized in Cycle 13.

3.0 SUMMARY OF POST-REFUELING OUTAGE STARTUP TEST RESULTS

3.1 CORE VERIFICATION

3.1.1 Purpose

To verify that all fuel assemblies have been properly loaded into the reactor core as per the licensed final loading pattern including fuel bundle location, orientation, and seating.

3.1.2 Acceptance Criteria

3.1.2.1 Level 1 criteria: Each fuel assembly must be verified to be in its proper location as specified by the General Electric final loading pattern (Licensed Core) and be correctly seated in its respective cell.

3.1.2.2 Level 2 criteria: N/A

3.1.3 Test Description

The Hatch Unit 1 Cycle 13 core verification was performed by use of an underwater TV camera to visually inspect the location (by bundle serial number identification), orientation, and seating of each of the 560 fuel assemblies that comprise the as loaded core.

3.1.4 Test Results

The core verification was performed on 4/18/90 in accordance with engineering procedure 42FH-ERP-014-OS, Fuel Movement Operation. Videotapes of the core loading indicated all fuel assemblies were in the proper location with proper orientation. Fuel bundle seating verification indicated that all bundles were properly seated.

3.2 CONTROL ROD DRIVE FRICTION TESTING

3.2.1 Purpose

To demonstrate that the control rod drive system operates properly following the completion of a core alteration. In particular, this functional test demonstrates the absence of excessive friction in the control rod drive from internal drive obstructions following extensive control rod drive maintenance/replacement.

3.2.2 Acceptance Criteria

3.2.2.1 Level 1 criteria: The differential pressure variation of all control rod drives to be tested must be less than or equal to 15 psid for continuous insertion. If this criterion cannot be satisfied, then a settling test must be performed in which case the differential settling pressure should not be less than 30 psid over the full stroke. Lower differential pressures in the settling test are indicative of excessive friction.

3.2.2.2 Level 2 criteria: N/A

3.2.3 Test Description

Control rod drive friction testing is normally performed on all control drives that have been replaced or have undergone extensive maintenance repair during the refueling outage. In essence, the functional test measures the differential pressure across the drive piston during a normal insertion stroke. If necessary, a settle test, which measures the differential settling pressure of each notch, is performed on a control rod drive during a withdrawal or insertion stroke.

3.2.4 Test Results

Control rod friction testing was performed on 4/18/90 for twenty-four control rod drive units. Twenty of these drive units were replaced during the outage. The testing was performed under engineering procedure 42IT-C11-001-0S, Control Rod Friction Testing. The test results indicated that all of the control rod drives were satisfactory either by the normal insertion differential pressure test or the settle test. A summary of the results of the control rod friction testing is given in Attachment 1.

3.3 CONTROL ROD DRIVE TIMING

3.3.1 Purpose

To demonstrate that the control rod drive system operates properly following the completion of a core alteration. In particular, this functional test verifies that the insert and withdrawal capability of the control rod drive system is within acceptable limits.

3.3.2 Acceptance Criteria

3.3.2.1 Level 1 Criteria: The insert and withdraw drive time for each control rod drive must be between 38.4 and 57.6 seconds. In the event that a control rod drive fails to meet this criteria, then the applicable drive must be adjusted and a new criteria of 43.2 to 52.8 seconds is applied to the adjusted drive.

3.3.2.2 Level 2 Criteria: N/A

3.3.3 Test Description

Control rod drive timing is performed once per operating cycle on all control rod drives. Normal withdrawal and insertion times are recorded for each of the drives under normal drive water pressure. If acceptable withdrawal and/or insertion cannot be obtained for normal drive water pressure, then the respective needle valve for the applicable withdrawal and/or insertion stroke must be adjusted until an acceptable drive time is achieved in accordance with the above criteria.

3.3.4 Test Results

Control rod drive timing was performed on 4/28/90 for all 137 control rod drives in accordance with operations procedure 34SV-C11-004-1S, CRD Timing. Each control rod drive was determined to have, or adjusted (where necessary) to have, a normal insertion and withdrawal speed as required. A summary of the results of this functional test is given in Attachment 2

3.4 FULL CORE SHUTDOWN MARGIN DEMONSTRATION

3.4.1 Purpose

To demonstrate that the reactor can be made subcritical for any reactivity condition during Cycle 13 operation with the analytically determined highest worth control rod, capable of withdrawal, fully withdrawn and all other rods fully inserted.

3.4.2 Acceptance Criteria

Level 1 Criteria: The fully loaded core must be subcritical by at least 0.38% delta k with the analytically determined highest worth control rod, capable of withdrawal, fully withdrawn and all other rods fully inserted at the most reactive condition during the cycle.

Level 2 Criteria: N/A

3.4.3 Test Description

The full core shutdown margin demonstration was performed analytically during the Hatch Unit 1 Cycle 13 BOC in-sequence critical with the reactor core in a xenon-free state. To account for reactivity effects such as moderator temperature, reactor period, and one rod out criterion, correction factors are used to adjust the startup condition to cold conditions with the highest worth control rod fully withdrawn.

3.4.4 Test Results

The full core shutdown margin demonstration was performed on 06/01/90 in accordance with core calculation procedure 42CC-ERP-010-0S, Shutdown Margin Demonstration. Results of this calculation yielded a cold shutdown margin of 2.06% delta k. The minimum cold shutdown margin was also 2.06% delta k because BOC is the most reactive point in this operating cycle. A summary of the shutdown margin demonstration is given in Attachment 3 of this report.

3.5 COLD CRITICAL EIGENVALUE COMPARISON

3.5.1 Purpose

To compare the critical eigenvalue calculated using the actual cold, xenon-free critical control rod configuration (corrected for moderator temperature and reactor period reactivity effects) to the cold critical eigenvalue assumed in the cycle management analyses.

3.5.2 Acceptance Criteria

3.5.2.1 Level 1 criteria: N/A

3.5.2.2 Level 2 criteria: N/A

3.5.3 Test Description

The cold critical eigenvalue is the assumed value of the PANACEA 3-D simulator model Keff at which criticality is achieved with the reactor in a xenon-free state and the coolant at 68 degrees F. This value is determined based on historical data and used for cycle management analyses by the BWR Core Analysis Group of Southern Company Services in Birmingham, Alabama. Once the actual critical state is achieved during the beginning of cycle startup, the applicable data is sent to the BWR Core Analysis Group and the actual (corrected for moderator temperature and reactor period reactivity effects) cold critical eigenvalue is calculated. This value is then compared to the assumed critical eigenvalue as a method of validating rod worth and shutdown margin calculations throughout the cycle. The actual critical eigenvalue is also entered into a database for predicting future cold critical eigenvalues.

3.5.4 Test Results

The beginning-of-cycle startup for Hatch Unit 1 Cycle 13 was performed on 6/1/90. The following reactor core conditions were observed when a critical state was achieved:

Sequence	A2
RSCS Group 1	Fully withdrawn
RSCS Group 2	Fully withdrawn
RSCS Group 3	5 control rods fully withdrawn and the 6th control rod (34-07) withdrawn to notch 6
Moderator Temperature	175 degrees F
Reactor Period	155.8 sec
Control Rod Density	0.738
Calculated MCSDM	2.06% delta k

A cold critical eigenvalue of 1.00324 was calculated from the actual critical data given above. This compares well to an assumed value of 1.0025.

3.6 WHOLE CORE LOCAL POWER RANGE MONITOR (LPRM) CALIBRATION

3.6.1 Purpose

To determine (1) The LPRM calibration constants such that when multiplied by the actual LPRM readings will produce calibrated LPRM readings proportional to the traversing in-core probe (TIP) signal readings at the LPRM locations and (2) The BASE and BASELP arrays which contain the machine normalized full power adjusted TIP signals at every node and LPRM detector location, respectively.

3.6.2 Acceptance Criteria

3.6.2.1 Level 1 criteria: N/A

3.6.2.2 Level 2 criteria: N/A

3.6.3 Testing Description

The whole core LPRM calibration and BASE distribution calculation determines the LPRM calibration constants and the BASE and BASLP distributions used in the axial power distribution calculations. In essence, the TIP system is used in conjunction with the process computer to generate the axial distribution of machine-independent/core average power-independent TIP signals. The axial distribution of machine normalized full power adjusted TIP data is used to generate LPRM calibration constants required for TIP normalized LPRM readings.

In addition, machine normalized full power adjusted TIP readings are generated at every axial node (BASE distribution) and at every LPRM detector location (BASLP distribution). These arrays are used as input data in the core calculation/monitoring programs to accurately calculate the power distribution at every node in the core.

3.6.4 Test Results

Whole core LPRM Calibration and BASE distribution was performed in accordance with Engineering procedure 42CC-ERP-015-0S, OD1 and OD2 NUMAC TIP Operation at approximately 24.5%, 37.2%, 50.3%, and 77% power. LPRM calibration constants, BASE and BASLP arrays, were computed by the process computer and subsequently used successfully by the process computer to calculate the nodal power distribution and the core thermal limits.

3.7 APRM CALIBRATION

3.7.1 Purpose

To calibrate the APRM system to actual core thermal power, as determined by a heat balance.

3.7.2 Acceptance Criteria

3.7.2.1 Level 1 criteria: The APRM readings must be within a tolerance of 2% of core thermal power as determined from a heat balance.

3.7.2.2 Level 2 criteria: N/A

3.7.3 Test Description

The APRM gains are adjusted after major power level changes, if required, to read the actual core thermal power as determined by a heat balance in accordance with procedure 34SV-SUV-021-0S, APRM Adjustment to Core Thermal Power. The heat balance required for the calibration process may be obtained from the process computer program P1 (Periodic Core Evaluation) and OD3 (Core Thermal Power and APRM Calibration) or from a manual heat balance in accordance with core calculation procedure 42CC-ERP-001-1S, Core Heat Balance-Power Range.

3.7.4 Test Results

APRM calibration was performed in accordance with plant procedure 34SV-SUV-021-0S, APRM Adjustment to Core Thermal Power at approximately 18%, 24%, 28%, 37%, 50.2% and 85.9% power. Each APRM was calibrated within a 2% tolerance to read core thermal power as calculated by the process computer.

3.8 CONTROL ROD SCRAM TIME TESTING

3.8.1 Purpose

To demonstrate that the control rod drive system functions as designed with respect to scram insertion times following the completion of core alterations.

3.8.2 Acceptance Criteria

3.8.2.1 Level 1 criteria:

(a) The average scram insertion time for all operable control rods from the fully withdrawn position, based on de-energization of the scram pilot solenoids, with reactor steam dome pressure above 950 psig shall not exceed the following:

Notch Position from Fully Withdrawn	Average Insertion Time (secs)
46	0.358
36	1.096
26	1.860
06	3.419

(b) The average scram insertion time, from the fully withdrawn position, for the 3 fastest control rods in each group of four control rods arranged in a 2x2 array, based on the de-energization of the scram pilot solenoids, shall not exceed the following:

Notch Position from Fully Withdrawn	Average Insertion Time (secs)
46	0.379
36	1.162
26	1.972
06	3.624

(c) The maximum scram insertion time of each control rod, from the fully withdrawn position to position 06, based on the de-energization of the scram pilot solenoid, shall not exceed 7.0 seconds.

Level 2 criteria: N/A

3.8.3 Test Description

The control rod drive scram time testing was performed in accordance with engineering procedure 42SV-C11-001-1S, Control Rod Scram Testing, with the steam dome pressure above 950 psig. The test consists of scramming each control rod, collecting the resulting scram time data, and analyzing the data in accordance with the procedure to ensure compliance with the acceptance criteria noted above.

3.8.4 Test Results

All control rod drives were tested in accordance with engineering procedure 42SV-C11-001-1S, Control Rod Scram Testing, with the steam dome pressure greater than 950 psig. A summary of the results is given in Attachment 4 of this report.

3.9 REACTIVITY ANOMALY CALCULATION

3.9.1 Purpose

To check for possible reactivity anomalies as the core excess reactivity changes with exposure.

3.9.1 Acceptance Criteria

3.9.1.1 Level 1 criteria: The corrected control rod density shall not differ from a control rod density equivalent by more than plus or minus 1% delta k.

3.9.1.2 Level 2 criteria: N/A

3.9.2 Test Description

During the BOC startup following a refueling outage and every month thereafter, a reactivity anomaly calculation is performed to monitor the core reactivity during the cycle. Since anticipated operation or unanticipated events may place the reactor in a condition other than that for which the baseline anomaly curve was developed, the actual control rod density is corrected for off-rated conditions. The corrected control rod density is then compared to the reactivity anomaly curve provided in the Cycle Management Report to ensure that the corrected control rod density is within a plus or minus 1% delta k acceptance band about the curve.

3.9.3 The reactivity anomaly calculation was performed in accordance with 42CC-ERP-007-0S, Reactivity Anomaly Calculation, on 6/29/90. The corrected control rod density was well within the acceptance criteria range as specified above. The results of this calculation are given in Attachment 5 of this report.

ATTACHMENT 1
CONTROL ROD FRICTION TEST RESULTS

DRIVE LOCATION	RX. PRESS.	CRD DIFF PRESS.	INSERTION DIFF. PRESS			SETTLE TEST REQUIRED
			MAX	MIN	DIFF.	
18-03	0	270	66.6	53.3	13.3	NO
06-15	0	270	66.7	52.7	14.0	NO
06-23	0	270	84.0	60.0	24.0	YES
06-19	0	270	70.0	58.0	12.0	NO
14-15	0	270	70.0	56.7	13.3	NO
42-11	0	270	73.3	58.3	15.0	NO
42-19	0	270	71.7	58.3	13.4	NO
46-23	0	270	71.7	56.7	15.0	NO
50-31	0	270	70.0	56.7	18.3	YES
50-27	0	270	73.0	58.0	15.0	NO
34-43	0	270	70.0	61.5	8.5	NO
38-39	0	270	67.3	58.7	8.6	NO
34-35	0	270	75.0	60.0	15.0	NO
30-35	0	270	66.7	56.7	10.0	NO
34-31	0	270	63.0	54.0	9.0	NO

ATTACHMENT 1
CONTROL ROD FRICTION TEST RESULTS

DRIVE LOCATION	RX. PRESS.	CRD DIFF PRESS.	INSERTION DIFF. PRESS			SETTLE TEST REQUIRED
			MAX	MIN	DIFF.	
34-47	0	270	68.0	56.0	12.0	NO
26-51	0	270	71.0	56.0	15.0	NO
26-35	0	270	69.0	57.0	12.0	NO
22-35	0	270	72.0	58.0	14.0	NO
22-51	0	270	73.0	60.0	13.0	NO
22-47	0	270	70.0	57.0	13.0	NO
22-39	0	270	68.0	56.0	12.0	NO
14-39	0	270	71.0	56.0	15.0	NO
06-35	0	270	71.0	56.0	15.0	NO

DRIVE LOCATION 06-23

NOTCH OPERATION		SETTLE DIFFERENTIAL PRESSURE (PSID)
NOTCHING-IN	NOTCHING-OUT	
48-46	00-02	37.0
46-44	02-04	37.1
44-42	04-06	37.3
42-40	06-08	35.0
40-38	08-10	37.0
38-36	10-12	37.0
36-34	12-14	39.0
34-32	14-16	40.0
32-30	16-18	37.0
30-28	18-20	37.0
28-26	20-22	39.0
26-24	22-24	38.0
24-22	24-26	38.0
22-20	26-28	38.0
20-18	28-30	38.0
18-16	30-32	39.0
16-14	32-34	39.0
14-12	34-36	38.0
12-10	36-38	37.0
10-08	38-40	36.0
08-06	40-42	36.0
06-04	42-44	37.0
04-02	44-46	38.0
02-00	46-48	41.0

DRIVE LOCATION 50-31

NOTCH OPERATION		SETTLE DIFFERENTIAL PRESSURE (PSID)
NOTCHING-IN	NOTCHING-OUT	
48-46	00-02	40.0
46-44	02-04	40.0
44-42	04-06	40.0
42-40	06-08	40.0
40-38	08-10	40.0
38-36	10-12	40.0
36-34	12-14	40.0
34-32	14-16	40.0
32-30	16-18	40.0
30-28	18-20	40.0
28-26	20-22	40.0
26-24	22-24	40.0
24-22	24-26	40.0
22-20	26-28	40.0
20-18	28-30	40.0
18-16	30-32	40.0
16-14	32-34	40.0
14-12	34-36	40.0
12-10	36-38	40.0
10-08	38-40	40.0
08-06	40-42	40.0
06-04	42-44	40.0
04-02	44-46	40.0
02-00	46-48	40.0

ATTACHMENT 2

CRD TIMING RESULTS

DRIVE LOCATION	WITHDRAW TIME	INSERT TIME	DRIVE LOCATION	WITHDRAW TIME	INSERT TIME
18-51	41.1	48.9	10-43	48.7	50.3
22-51	42.6	49.0	14-43	40.4	43.9
26-51	51.8*	47.9*	18-43	44.3	47.8
30-51	44.0	49.5	22-43	39.2	42.1
34-51	51.4	51.4	26-43	41.8	49.2
10-47	49.3	47.6	30-43	40.9	55.9
14-47	50.3	44.1	34-43	44.1*	43.3*
18-47	47.9	50.4	38-43	45.4	50.3
22-47	48.3*	44.3*	42-43	43.5	50.8
26-47	54.0	46.7	46-43	42.9	45.1
30-47	41.8	54.3	06-39	39.8	52.9
34-47	47.2*	49.5*	10-39	46.8	47.4
38-47	47.5	50.2	14-39	43.5*	38.7
42-47	45.6	54.6	18-39	49.3	44.7
06-43	46.3	51.3	22-39	49.3*	45.2*

* ADJUSTED DRIVE

ATTACHMENT 2

CRD TIMING RESULTS

DRIVE LOCATION	WITHDRAW TIME	INSERT TIME	DRIVE LOCATION	WITHDRAW TIME	INSERT TIME
26-39	41.4	52.7	38-35	50.8*	46.9 *
30-39	50.4	54.8	42-35	46.1	47.4
34-39	42.0	56.3	46-35	42.3	47.6*
38-39	49.3*	45.6*	50-35	46.4*	44.9*
42-39	42.8	44.6	02-31	42.8	47.5
46-39	39.2	52.1	06-31	42.4	51.2
02-35	47.2*	51.2*	10-31	39.7	45.1
06-35	53.5	42.9	14-31	49.3	44.7
10-35	41.8	45.8	18-31	42.8	54.4
14-35	47.7	47.7	22-31	47.6	47.7
18-35	39.3	48.9	26-31	49.2	45.7
22-35	38.5	42.8	30-31	51.2*	50.2*
26-35	44.1*	51.9	34-31	43.9	47.0*
30-35	49.2*	54.3	38-31	52.5	48.1
34-35	43.8	51.3	42-31	44.4	46.9*

* ADJUSTED DRIVE

ATTACHMENT 2

CRD TIMING RESULTS

DRIVE LOCATION	WITHDRAW TIME	INSERT TIME	DRIVE LOCATION	WITHDRAW TIME	INSERT TIME
46-31	47.7	57.8	02-23	46.9	54.3
50-31	51.0*	43.8	06-23	49.0	49.1
02-27	39.1	47.5	10-23	42.8	48.[illegible]
06-27	47.7*	44.1*	14-23	50.2	51.4
10-27	42.3	51.5	18-23	40.1	49.9
14-27	53.7	50.6	22-23	41.3	46.9
18-27	38.8	47.4	26-23	51.3	47.1
22-27	54.4	46.5	30-23	43.5*	49.0
26-27	42.8	56.5	34-23	41.9	51.3
30-27	46.6	48.8	38-23	46.6	47.6
34-27	50.9*	57.3	42-23	48.1	43.1
38-27	48.8	48.3	46-23	48.9*	38.4
42-27	50.2	51.2	50-23	46.0	49.6
46-27	44.1	46.6	02-19	40.5	51.8
50-27	45.9*	47.8*	06-19	50.4	46.3

* ADJUSTED DRIVE

ATTACHMENT 2

CRD TIMING RESULTS

DRIVE LOCATION	WITHDRAW TIME	INSERT TIME	DRIVE LOCATION	WITHDRAW TIME	INSERT TIME
10-19	42.4	46.8*	22-15	49.4	48.9
14-19	48.8*	48.7*	26-15	42.4	52.2
18-19	39.4	55.4	30-15	46.6	46.0
22-19	44.7	54.4	34-15	49.5	54.1
26-19	44.5	50.0	38-15	53.3	42.7
30-19	49.7	56.2	42-15	41.5	41.6
34-19	44.9*	46.0*	46-15	41.4	42.8
38-19	40.1	45.5*	06-11	41.7	48.3*
42-19	49.7*	47.2*	10-11	47.4	57.0
46-19	47.9*	50.0	14-11	46.1	56.2
50-19	50.4*	53.0	18-11	47.6	53.5
06-15	46.5*	46.2*	22-11	50.0*	48.0*
10-15	44.2	44.6*	26-11	56.2	46.0
14-15	50.6*	48.2*	30-11	55.2	51.5
18-15	52.2	46.2*	34-11	40.4	45.2*

* ADJUSTED DRIVE

ATTACHMENT 2

CRD TIMING RESULTS

DRIVE LOCATION	WITHDRAW TIME	INSERT TIME	DRIVE LOCATION	WITHDRAW TIME	INSERT TIME
38-11	42.7	50.3	30-03	46.6	53.9
42-11	51.4	42.8	34-03	40.5	56.1
46-11	39.3	51.2*			
10-07	44.7	53.3			
14-07	44.5	40.7*			
18-07	47.5	47.7*			
22-07	43.5	49.7			
26-07	47.6	45.3			
30-07	45.1	53.7			
34-07	37.8	46.0*			
38-07	43.9	55.8			
42-07	46.9	54.2			
18-03	51.8*	45.8*			
22-03	44.8	45.3			
26-03	41.5	47.5*			

* ADJUSTED DRIVE

ATTACHMENT 3

FULL CORE SHUTDOWN MARGIN DEMONSTRATION

1	K_{SRO}	0.9803
2	K_{CRIT}	1.0030
3	CONTROL ROD DENSITY	0.738
4	RX. COOLANT TEMPERATURE	175 °F
5	REACTIVITY CORRECTION FOR TEMPERATURE	-.0017 Δ K
6	REACTOR PERIOD	155.8 SEC
7	REACTIVITY CORRECTION FOR PERIOD	0.0004 Δ K
8	COLD SHUTDOWN MARGIN (2-1+5-7)	2.06 % Δ K
9	VALUE OF R	0.0 % Δ K
10	MINIMUM COLD SHUTDOWN MARGIN (8-9)	2.06 % Δ K
11	TECH SPEC REQUIRED SHUTDOWN MARGIN	0.38 % Δ K

ATTACHMENT 4

SCRAM TIME TESTING RESULTS - BOC13

NOTCH POSITION	SLOWEST SCRAM INSERTION TIME		AVERAGE SCRAM INSERTION TIME		AVERAGE SCRAM INSERTION TIME CRITERIA	
	SINGLE ROD	2X2 ARRAY 3 FASTEST	SINGLE ROD	2x2 ARRAY 3 FASTEST	SINGLE ROD	2x2 ARRAY 3 FASTEST
46	.399	.332	.305	.298	.358	.379
36	1.001	.850	.825	.815	1.096	1.162
26	1.859	1.406	1.349	1.323	1.860	1.972
06	3.439	2.554	2.484	2.442	3.419	3.624

ATTACHMENT 5

REACTIVITY ANOMALY CALCULATION RESULTS

Date Performed: June 29, 1990

Unit 1 Cycle 13 Sequence A2

CMWT: 2429.9

WT: 80.50 Mlb/hr

Dome Pressure: 998.7 psia

DHS: 21.6 BTU/lb_m

Actual Control Rod Density: 0.0912

Corrected CRD = Actual CRD + Correction

Correction: -3.0187E-1 *(1-(CMWT/2436))

+1.9875E-1 *(1-(WT/RATED CORE FLOW))

+2.6563E-3 *(DESIGN INLET SUBCOOLING - DHS)

+6.5843E-5 *(1020 - PR) = -0.0037

Corrected CRD = 0.0875

Predicted CRD = 0.0951

+1% Value = 0.1415 -1% Value = 0.0487