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DUKE POWER

February 21, 1996

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
Startup Report, Unit 2 Cycle 8

In accordance with Section 6.9.1 of the Catawba Nuclear Station Technical Specifications, find attached the Unit 2 Startup Report for Cycle 8 core design.

Any questions concerning this report may be directed to Kay Nicholson at (803) 831-3237.

Very truly yours,

Mark E. Patrick for

W. R. McCollum

KEN/U2C8.SR

Attachment

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Duke Power Company
Catawba Nuclear Station
Unit 2 Cycle 8

STARTUP REPORT

February 1996

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1.0 INTRODUCTION

Catawba Unit Two Cycle 8 includes a feed batch of 80 MkBW fuel assemblies manufactured by B&W Fuel Company (BWFC). The feed batch was enriched to 3.98% (w/o). Burnable poison rod assemblies used in the feed batch were also manufactured by BWFC.

Catawba Unit Two Cycle 8 core loading began at 1046 on November 1, 1995 and ended at 1534 on November 4, 1995. Initial criticality for cycle 8 occurred at 1810 on November 28, 1995. Zero Power Physics Testing ended at 0200 on November 30, 1995. The unit reached full power at 0041 on December 7, 1995. Power escalation testing, including testing at full power, was completed by December 9, 1995.

Table 1 contains some important characteristics of the Catawba 2 Cycle 8 core design.

TABLE 1
C2C8 CORE DESIGN DATA

1. C2C7 end of cycle burnup: 428 EFPD
2. C2C8 design length: 445 ± 10 EFPD

Region	Fuel Type	Number of Assemblies	Enrichment, w/o U^{235}	Loading, MTU**	Cycles Burned
8B	MkBW	25	3.75	11.405	2
9A	MkBW	40	4.00/0.71*	18.248	1
9B	MkBW	8	3.60/0.71*	3.650	1
9C	MkBW	40	3.50	18.248	1
10A	MkBW	80	3.98	36.496	0
Totals		193		88.0466	

* Natural U blanketed fuel assemblies (0.71 w/o enrichment - 6 inches top and bottom)

** Design MTU loadings which were used in all design calculations.

2.0 PRECRITICAL TESTING

Precritical testing includes:

- core loading

- preliminary calibration of nuclear instrumentation

- dilution of reactor coolant system to estimated critical boron concentration

Sections 2.1 through 2.3 describe results of precritical testing for Catawba 2 Cycle 8.

2.1 Total Core Reloading

The cycle 8 core was loaded under the direction of PT/0/A/4150/22, Total Core Reloading. Plots of Inverse Count Rate Ratio (ICRR) versus number of fuel assemblies loaded were kept for each applicable source range and boron dilution mitigation system (BDMS) channel.

Core loading began at 1046 on November 1, 1995 and concluded at 1534 on November 4, 1995. Core loading was verified by PT/0/A/4550/03C, Core Verification, which was completed by 2100 on November 4, 1995.

Figure 1 shows the core loading pattern for Catawba 2 Cycle 8.

2.2 Preliminary NIS Calibration

Periodic test procedure PT/0/A/4600/05E, Preliminary NIS Calibration, is performed before initial criticality for each new fuel cycle. Intermediate range reactor trip and rod stop setpoints are adjusted using measured power distribution from the previous fuel cycle and predicted power distribution for the upcoming fuel cycle. Power range full power currents are similarly adjusted. Intermediate range calibration data is checked and revised as necessary during power escalation testing. Due to the T-hot reduction an added conservatism was used to account for any uncertainties that may have been introduced by the T-hot reduction.

Table 2 shows the calibration data calculated by PT/0/A/4600/05E. Calculations were performed on November 13, 1995. Calibrations were complete by November 26, 1995.

2.3 Reactor Coolant System Dilution

The reactor coolant system boron concentration was diluted from the refueling boron concentration to the estimated critical boron concentration per PT/0/A/4150/19, 1/M Approach to Criticality. Inverse Count Rate Ratio (ICRR) was plotted versus gallons of demineralized water added.

Initial reactor coolant boron concentration was 2577 ppmB. The estimated critical boron concentration was calculated to be 1870 ppmB. The calculated volume of demineralized water required was 19489 gallons. This change in boron concentration was expected to decrease ICRR from 1.0 to 0.4.

Reactor coolant system dilution began at 2215 on November 27, 1995 and concluded at 0400 on November 28, 1995. The final reactor coolant system boron concentration, after allowing system to mix, was 1863 ppmB. Figure 2 shows ICRR versus volume of water used.

FIGURE 1
CORE LOADING PATTERN, CATAWBA UNIT 2 CYCLE 8

North
^
|

180

FUEL TRANSFER CANAL ----->

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					T27 PD	V06 PD	U87 PD	T31 PD	U81 PD	V32 PD	T02 PD				
2			T03 PD	US22 RCCA	V38 BP	US43 RCCA	V57 BP	US08 RCCA	V59 BP	US48 RCCA	V03 BP	US09 RCCA	T36 PD		
3		T20 PD	US02 PD	V75 BP	U63 RCCA	V13 BP	U72 RCCA	V50 BP	U50 RCCA	V52 BP	U77 RCCA	V43 BP	US14 PD	T69 PD	
4		US21 RCCA	V48 BP	US32 RCCA	V31 BP	US10 PD	V34 BP	US23 RCCA	V01 BP	US17 PD	V56 BP	US27 RCCA	V29 BP	US20 RCCA	
5	T19 PD	V15 BP	U62 RCCA	V67 BP	US29 PD	V65 BP	U58 PD	U60 PD	U61 PD	V54 BP	US28 PD	V80 BP	U65 RCCA	V53 BP	T29 SS
6	V49 PD	US41 RCCA	V82 BP	US11 PD	V14 BP	T59 RCCA	V70 BP	U83 RCCA	V12 BP	T25 RCCA	V33 BP	US06 PD	V61 BP	US47 RCCA	V41 PD
7	U57 PD	V66 BP	U64 RCCA	V68 BP	U53 PD	V79 BP	US40 PD	V07 BP	US39 PD	V78 BP	U59 PD	V08 BP	U78 RCCA	V22 BP	U79 PD
8	T51 PD	US24 RCCA	V11 BP	US03 RCCA	U88 PD	U74 RCCA	V23 BP	T23 RCCA	V40 BP	U80 RCCA	U82 PD	US12 RCCA	V39 BP	US19 RCCA	T68 PD
9	U75 PD	V16 BP	U68 RCCA	V58 BP	U66 PD	V46 BP	US37 PD	V71 BP	US34 PD	V42 BP	U76 PD	V17 BP	U69 RCCA	V30 BP	U49 PD
10	V10 PD	US46 RCCA	V09 BP	US15 PD	V19 BP	T06 RCCA	V55 BP	U56 RCCA	V72 BP	T40 RCCA	V20 BP	US18 PD	V04 BP	US45 RCCA	V02 PD
11	T12 SS	V28 BP	U84 RCCA	V77 BP	US05 PD	V73 BP	U70 PD	U51 PD	U73 PD	V76 BP	US38 PD	V27 BP	U55 RCCA	V24 BP	T60 PD
12		US30 RCCA	V51 BP	US31 RCCA	V47 BP	US36 PD	V37 BP	US33 RCCA	V18 BP	US25 PD	V05 BP	US26 RCCA	V63 BP	US16 RCCA	
13		T28 PD	US07 PD	V45 BP	U86 RCCA	V36 BP	U52 RCCA	V21 BP	U67 RCCA	V44 BP	U54 RCCA	V74 BP	US04 PD	T76 PD	
14			T70 PD	US01 RCCA	V60 BP	US44 RCCA	V26 BP	US35 RCCA	V69 BP	US42 RCCA	V64 BP	US13 RCCA	T34 PD		
15					T62 PD	V25 PD	U71 PD	T58 PD	U85 PD	V35 PD	T64 PD				

90

270

0

X## Fuel Assembly Region Reference Number

XX Fuel component (PD = plugging device, BP = burnable poison rod assembly)

TABLE 2
PRELIMINARY NIS CALIBRATION DATA

Intermediate Range

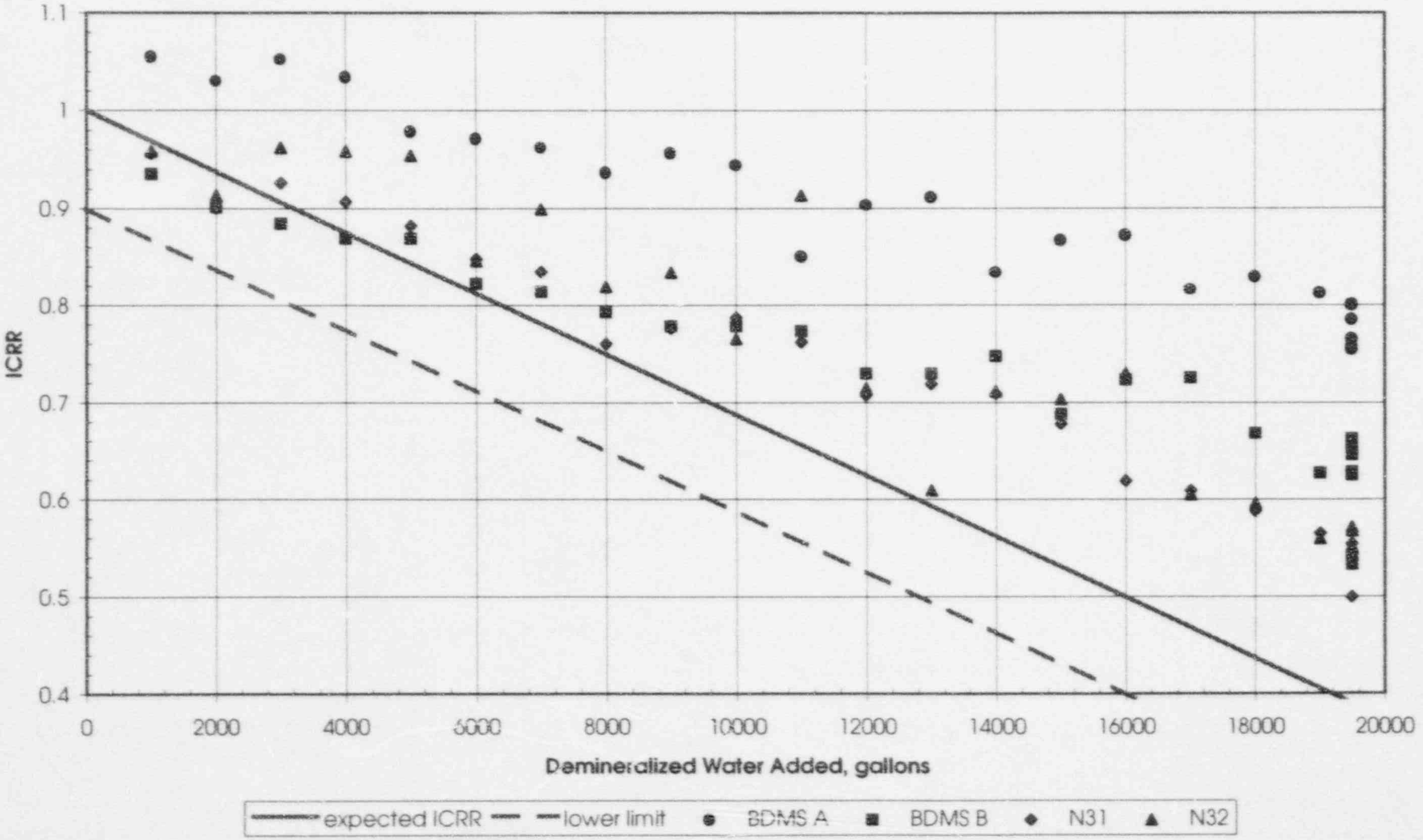
Channel	Ratio (BOC 8 ÷ Cycle 7)	Cycle 7 Reactor Trip Setpoint, μAmps	BOC 8 Reactor Trip Setpoint, μAmps	BOC 8 Rod Stop Setpoint, μAmps
N35	0.8	84.8	64.4	51.5
N36	0.8	68.5	52.1	41.7

Power Range

Channel	Ratio (BOC 8 ÷ Cycle 7)	Axial Offset, %	Cycle 7 Full Power Current, μAmps		BOC 8 Full Power Current, μAmps	
			Upper	Lower	Upper	Lower
N41	0.854	+20	330.2	262.2	282.0	223.9
		0	290.4	308.8	248.0	263.7
		-20	250.5	355.2	213.9	303.3
N42	0.857	+20	264.1	215.0	226.3	184.3
		0	233.0	250.9	199.7	215.0
		-20	202.0	286.7	173.1	245.7
N43	0.860	+20	301.6	239.7	259.4	206.1
		0	261.4	279.2	224.8	240.1
		-20	221.2	318.7	190.2	274.1
N44	0.859	+20	280.0	225.9	240.5	194.0
		0	244.5	262.6	210.0	225.6
		-20	208.9	299.3	179.4	257.1

FIGURE 2

ICRR vs. DEMIN WATER ADDED DURING REACTOR COOLANT SYSTEM DILUTION



3.0 ZERO POWER PHYSICS TESTING

Zero Power Physics Testing (ZPPT) is performed at the beginning of each cycle and is controlled by PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing. Test measurements are made below the point of nuclear heat using the output of one power range detector connected to a reactivity computer. Measurements are compared to predicted data to verify core design. The following tests/measurements are included in the ZPPT program:

- 1/M Approach to Criticality
- Measurement of point of adding heat
- Reactivity computer checkout
- All Rods Out critical boron concentration measurement
- All Rods Out isothermal temperature coefficient measurement
- Measurement of reference bank worth by dilution
- Reference bank in critical boron concentration measurement
- Differential boron worth determination
- Control rod worths measurement by Rod Swap

Zero power physics testing for Catawba 2 Cycle 8 began at 1447 on November 28, 1995 with the beginning of rod withdrawal for approach to criticality. ZPPT ended at 0200 on November 30, 1995 following analysis of rod swap data. Table 3 summarizes results from ZPPT. All acceptance criteria were met.

Sections 3.1 through 3.10 describe ZPPT measurements and results.

3.1 1/M Approach to Criticality

Initial criticality for Catawba 2 Cycle 8 was achieved per PT/0/A/4150/19, 1/M Approach to Criticality. In this procedure, Estimated Critical Rod Position (ECP) is calculated based on latest available reactor coolant boron concentration. Control rods are withdrawn until Boron Dilution Mitigation System (BDMS) or Source Range count rate doubles. Inverse Count Rate Ratio (ICRR) is plotted for each source range and BDMS channel. ICRR data is used to project critical rod position. If projected critical rod position is acceptable, rod withdrawal may continue.

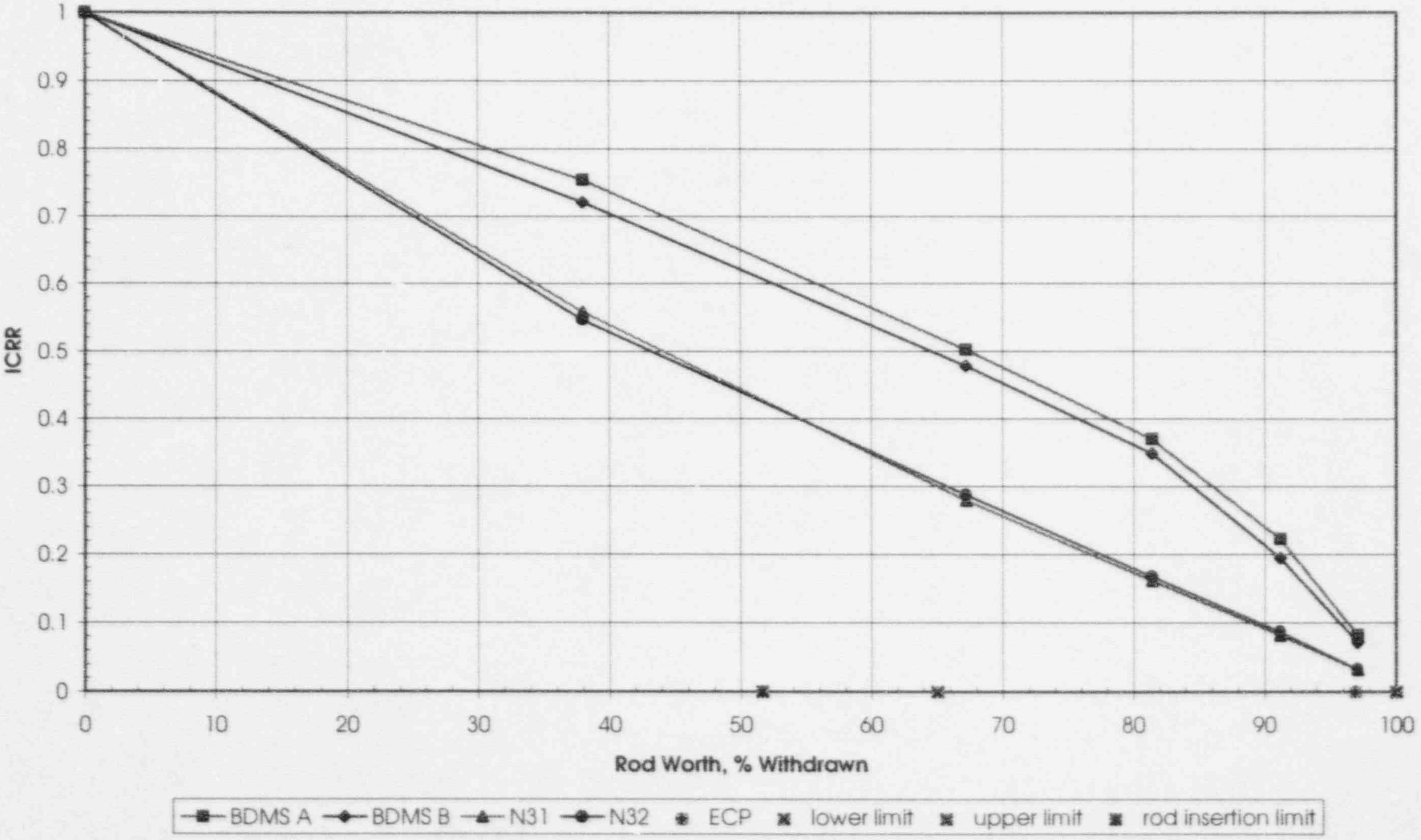
Rod withdrawal for the approach to criticality began at 1451 on November 28, 1995. Criticality was achieved at 1811 on November 28, 1995 with Control Bank D at 208 steps withdrawn.

Figure 3 shows the ICRR plots that were used during the approach to criticality. All acceptance criteria of PT/0/A/4150/19 were met.

TABLE 3
SUMMARY OF ZPPT RESULTS

PARAMETER	MEASURED VALUE	PREDICTED VALUE/ACCEPTANCE CRITERIA
Nuclear Heat	3.4 × 10^{7} amps (N44)	N/A
ZPPT Test Band	10^{8} to 1.7x10^{7} amps (N44)	N/A
ARO Critical Boron	1869 ppmB	1876 ± 50 ppmB
ARO ITC	-3.14 pcm/°F	-3.47 ± 2 pcm/°F
ARO MTC	-1.37 pcm/°F	-1.70 pcm/°F
Reference Bank (Shutdown Bank B) Worth	999.5 pcm	900 ± 135 pcm
Ref. Bank in Critical Boron	1731 ppmB	1746 ppmB
Differential Boron Worth	-7.24 pcm/ppmB	-6.92 ± 1.04 pcm/ppmB
Control Bank D Worth	593.5 pcm	584 ± 200 pcm
Control Bank C Worth	809.0 pcm	754 ± 226 pcm
Control Bank B Worth	803.4 pcm	713 ± 214 pcm
Control Bank A Worth	373.6 pcm	360 ± 200 pcm
Shutdown Bank E Worth	570.1 pcm	516 ± 200 pcm
Shutdown Bank D Worth	468.8 pcm	436 ± 200 pcm
Shutdown Bank C Worth	469.9 pcm	437 ± 200 pcm
Shutdown Bank A Worth	283.7 pcm	269 ± 200 pcm
Total Rod Worth	5372.4 pcm	4969 ± 497 pcm

FIGURE 3
ICRR vs. CONTROL ROD WORTH DURING APPROACH TO CRITICALITY



3.2 Source Range/Intermediate Range Overlap Data

During the initial approach to criticality, source range and intermediate range data was obtained to verify that at least one decade of overlap existed. If one decade of overlap did not exist, intermediate range compensation voltage would have been adjusted to provide the overlap.

Overlap data for Cycle 8 was obtained per PT/0/A/4150/01, Controlling procedure for Startup Physics Testing, on November 28, 1995. Table 4 contains the overlap data. The acceptance criterion was met.

TABLE 4
SOURCE RANGE/ INTERMEDIATE RANGE OVERLAP DATA

	SOURCE RANGE		INTERMEDIATE RANGE	
	N31, cps	N32, cps	N35, amps	N36, amps
INITIAL DATA: NIS Meters	550	500	1×10^{-11}	1×10^{-11}
OAC	478.4	509.0	1.044×10^{11}	1.082×10^{11}
FINAL DATA: NIS Meters	15,000	15,000	1.5×10^{-10}	1.5×10^{-10}
OAC	12,340	12,130	1.350×10^{-10}	1.197×10^{-10}

3.3 Point of Nuclear Heat Addition

The point of nuclear heat addition is measured by trending reactor coolant system temperature, pressurizer level, flux level, and reactivity while slowly increasing reactor power. A slow, constant startup rate is initiated by rod withdrawal. An increase in reactor coolant system temperature and/or pressurizer level accompanied by a change in reactivity and/or rate of flux increase indicates the addition of nuclear heat.

For Cycle 8, the point of nuclear heat addition was determined per PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing, on November 28, 1995. Table 5 summarizes the data obtained.

The zero power physics test band was set at 10^{8} to $1.7x10^{7}$ amps on power range channel N44 (connected to reactivity computer). This test band provided more than a factor of two margin to nuclear heat for zero power physics testing. Acceptance criterion was satisfied.

TABLE 5
NUCLEAR HEAT DETERMINATION

	Reactivity Computer (N44), amps	Intermediate Range Channel N35, amps	Intermediate Range Channel N36, amps
RUN #1	3.4 x 10^{-7}	4.515 x 10^{-7}	3.893 x 10^{-7}
RUN #2	4.5 × 10^{-7}	3.6 × 10^{-7}	3.1 × 10^{-7}

3.4 Reactivity Computer Checkout

The reactivity computer checkout was performed per PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing, to verify that the power range channel connected to the reactivity computer can provide reliable reactivity data. A reactivity insertion of approximately +20, was made. The period is measured and used to determine the theoretical reactivity. The measured reactivity is compared to the theoretical reactivity and verified to be within 4.0%.

The checkout was performed for Cycle 8 on November 28, 1995. Table 6 lists the results of the reactivity insortion. The acceptance criterion was met.

TABLE 6
REACTIVITY COMPUTER CHECKOUT

Period, seconds	Theoretical Reactivity, pcm	Measured Reactivity, pcm	Absolute Error, pcm	Percent Error,%
390.24	17.69	17.77	0.08	0.45

3.5 ARO Boron Endpoint Measurement

This test is performed at the beginning of each cycle to verify that measured and predicted total core reactivity are consistent. The test is performed near the all rods out (ARO) configuration. Reactor coolant system boron samples are obtained while control bank D is pulled to the fully withdrawn position. The reactivity difference from criticality to the ARO configuration is measured and converted to an equivalent boron worth using the predicted differential boron worth. The average measured boron concentration is adjusted accordingly to obtain the ARO critical boron concentration.

The Cycle 8 beginning of cycle, hot zero power, all rods out, critical boron concentration was measured on November 29, 1995 per PT/0/A/4150/10, Boron Endpoint measurement. The ARO, HZP boron concentration was measured to be 1869 ppmB. Predicted ARO critical boron concentration was 1876 ppmB. The acceptance criterion, measured boron within 50 ppmB of predicted, was met.

3.6 ARO Isothermal Temperature Coefficient Measurement

The all rods out (ARO) isothermal temperature coefficient (ITC) is measured at the beginning of each cycle to verify consistency with predicted value. In addition, the moderator temperature coefficient (MTC) is obtained by subtracting the doppler temperature coefficient from the ITC. The MTC is used to ensure compliance with Technical Specification limits.

To measure the ITC, statepoint data is obtained prior to cooldown. A reactor coolant system cooldown is initiated, within administrative cooldown limits. When sufficient data (at least 5 °F) is obtained, statepoint data is again obtained. A heatup is performed while again maintaining administrative limits. The Delta Reactivity divided by the Delta Temperature (for each cooldown/heatup) are used to determine the ITC. The cooldown/heatup cycle is repeated if additional data is required.

The beginning of cycle 8 ITC was measured per PT/0/A/4150/12A, Isothermal Coefficient of Reactivity Measurement, on November 29, 1995. No additional cooldown/heatup cycles were required because of good agreement between the heatup and cooldown results. Table 7 summarizes the data obtained during the measurement.

Average ITC was -3.14 pcm/°F. Predicted ITC was -3.47 pcm/°F. Measured ITC was within acceptance criterion of predicted ITC ± 2 pcm/°F.

The MTC was determined to be -1.37 pcm/°F. This value was used with procedure PT/0/A/4150/21, Temporary Rod Withdrawal Limits Determination, to ensure that MTC would remain within Technical Specification limits at all power levels. No rod withdrawal limits were required.

TABLE 7
ITC MEASUREMENT RESULTS

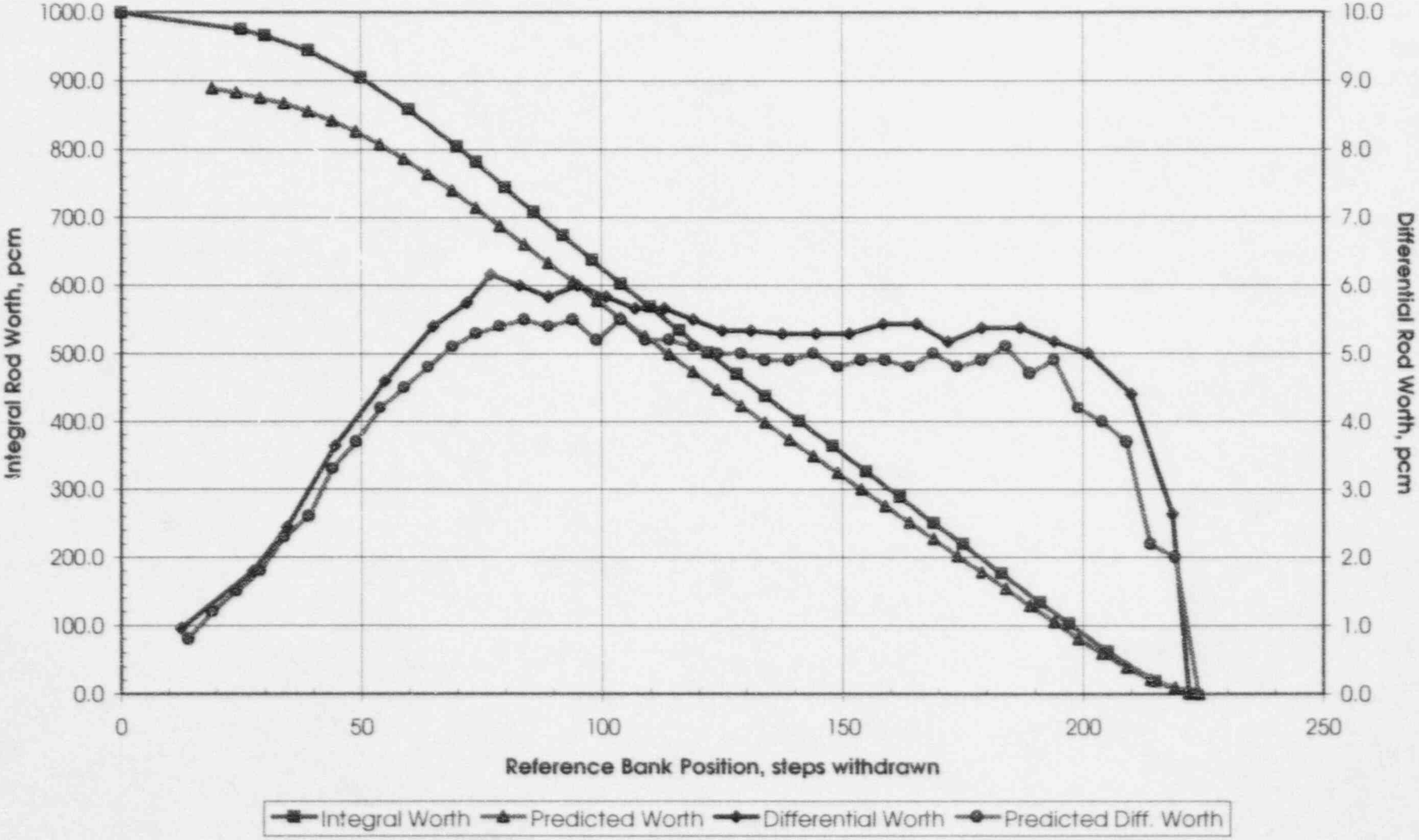
	ΔT, °F	$\Delta\rho$, pcm	T_{avg}, °F	ITC, pcm/°F
Cooldown	-7	+22.0	554.8	-3.14
Heatup	+7	-22.0	554.8	-3.14
				Average: -3.14

3.7 Reference Bank Worth Measurement by Dilution

The control rod bank predicted to have the highest worth is designated the reference bank and is measured by inserting the bank (with all other rod banks fully withdrawn) in discrete steps while slowly diluting the reactor coolant boron concentration. The reactivity worths of the discrete steps of rod insertion are measured using the reactivity computer and summed to obtain the integral worth of the reference bank.

The beginning of cycle 8 reference bank (Control Bank C) worth was measured on November 29, 1995 per PT/0/A/4150/11A, Control Rod Worth Measurement by Boration/Dilution. Figure 4 shows integral worth of reference bank versus bank position. The reference bank was measured to be worth 999.5 pcm; predicted worth was 900 pcm. The acceptance criterion, measured worth within ± 15% of predicted, was met.

FIGURE 4
INTEGRAL AND DIFFERENTIAL WORTH OF REFERENCE BANK



3.8 Reference Bank In Boron Endpoint Measurement

This test is performed at the beginning of each cycle to measure the critical boron concentration with the reference bank fully inserted and all other control rod banks fully withdrawn. The measured boron concentration is used with the measured ARO critical boron concentration and the measured worth of the reference bank to calculate the differential boron worth. Reactor coolant system boron samples are obtained while control rods are inserted or withdrawn to the "Reference Bank in" configuration. The reactivity difference from criticality to the "Reference Bank in" configuration is measured and converted to an equivalent boron worth using the predicted differential boron worth. The average measured boron concentration is adjusted accordingly to obtain the "Reference Bank in" critical boron concentration.

The Cycle 8 beginning of cycle, hot zero power, reference bank in, critical boron concentration was measured on November 29, 1995 per PT/0/A/4150/10, Boron Endpoint measurement. The Reference Bank In Boron Endpoint boron concentration was measure to be 1731 ppmB. Predicted "Reference Bank in" critical boron concentration was 1746 ppmB. There is no quantitative acceptance criteria directly associated with this test.

3.9 Differential Boron Worth Determination

The differential boron worth is calculated from the measured ARO critical boron concentration, Reference Bank in critical boron concentration, and total reactivity worth of reference bank. The calculated value is compared to predicted value to verify consistency. This calculation also provides an indirect check of measured reference bank worth and of the boron endpoint measurements.

The beginning of Cycle 8, hot zero power differential boron worth was calculated to be -7.24 pcm/ppmB per PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing. The predicted value was -6.92 pcm/ppmB. The acceptance criterion (measured within ± 15% of predicted), was met.

3.10 Control Rod Worth Measurement by Rod Swap

The worths of all control rod banks except the reference bank are measured by inserting each bank while withdrawing the reference bank and/or previously measured bank to maintain near critical conditions. When the bank being measured is fully inserted, the reference bank is positioned to achieve critical conditions with all other rod banks fully withdrawn. The worth of the fully inserted bank is determined from the critical position of the reference bank. The measured worth is compared to predicted worth to verify consistency. The sum of the worths of all banks, including the reference bank, is also compared to predicted total.

The beginning of cycle 8 rod worth measurement by rod swap was performed on November 29, 1995 per PT/0/A/4150/11B, Control Rod Worth Measurement by Rod Swap. Table 8 summarizes the results. All acceptance criteria were met.

TABLE 8
CONTROL ROD WORTH MEASUREMENT DATA

Bank	Adjusted Reference Bank Worth	Critical Position of Ref. Bank	Remaining Worth of Ref. Bank	Alpha	Measured Worth, pcm	Predicted Worth, pcm	Difference (Predicted - Measured)	% Diff. (Pred - Meas)/Pred × 100
Shutdown B (ref. bank)	N/A	N/A	N/A	N/A	999.5	900	-99.5	-11.1
Shutdown A	1003.5	88	697	1.033	283.7	269	-14.7	-5.5
Control A	1003.2	102.5	611	1.030	373.6	360	-13.6	-3.8
Shutdown D	1002.9	117.5	526	1.013	469.8	436	-33.8	-7.8
Shutdown C	1002.6	118.5	521	1.023	469.9	437	-32.9	-7.5
Shutdown E	1002.4	119.5	515	0.839	570.1	516	-54.1	-10.5
Control D	1002.1	147	369	1.108	593.5	584	-9.5	-1.6
Control B	1001.8	174	225	0.883	803.4	713	-90.4	-12.7
Control C	1001.5	174.5	222	0.867	809.0	754	-55.0	-7.3
				Total	5372.4	4969	-403.4	-8.1

4.0 POWER ESCALATION TESTING

Power escalation testing is performed during the initial power increase to full power for each cycle and is controlled by PT/0/A/4150/01, Controlling Procedure for Startup Physics Testing. Tests are performed from 0% through 100% power with major testing plateaus at ~30%,~65%, and 100% power.

Significant tests performed during power escalation are:

- Core Power Distribution (at ~30%, ~65%, and 100% power)
- One-Point Incore/Excore Calibration (at 30% power)
- Reactor Coolant Delta Temperature Measurement (at 65% and 100% power)
- Hot Full Power Critical Boron Concentration Measurement (at 100% power)
- Incore/Excore Calibration (at 100% power)
- Calorimetric Reactor Coolant Flow Measurement (at 100% power, This test is not under the control of PT/0/A/4150/01)
- Unit Load Steady State (at 10%,20%,30%,50%,75%,90% and 100%, This test is not under the control of PT/0/A/4150/01)
- Unit Load Transient Test (This test is not under the control of PT/0/A/4150/01)

In addition to the tests listed above, PT/0/A/4150/01 performs checks on the incore detector system, on-line thermal power program, intermediate range setpoints, etc. The results of these checks are not included in this report.

Power escalation testing for Catawba 2 Cycle 8 began on November 30, 1995. Full power was reached on December 7, 1995. Full power testing was completed on December 9, 1995.

Sections 4.1 through 4.6 describe the significant tests performed during power escalation and their results.

4.1 Core Power Distribution

Core power distribution measurements are performed during power escalation at low power (approximately 30%), intermediate power (approximately 65%), and full power. Measurements are made to verify flux symmetry and to verify core peaking factors are within limits. Data obtained during this test are also used to check calibration of power range channels and to calibrate them if required (see sections 4.2 and 4.6). Measurements are made using the moveable incore detector system and analyzed using Duke Power's CORE and MONITOR codes (adapted from Shangstrom Nuclear Associates' CORE package and BWFC's MONITOR code, respectively).

The Catawba 2 Cycle 8 core power distribution measurements were performed on December 1, 1995 (30% power), December 2, 1995 (60% power), and December 7, 1995 (100% power). Table 9 through 11 summarize the results. All acceptance criteria were met.

TABLE 9
CORE POWER DISTRIBUTION RESULTS
30% POWER

Plant Data

Map ID:	FCM/2/08/01
Date of Map:	December 1, 1995
Cycle Burnup:	0.210 EFPD
Power Level:	30.41%
Control Rod Position:	Control Bank D at 216 steps withdrawn
Reactor Coolant System Boron Concentration:	1672 ppmB

CORE Results

Core Average Axial Offset:	14.589%
Tilt Ratios for Entire Core Height: Quadrant 1:	0.99223
Quadrant 2:	0.99981
Quadrant 3:	1.00661
Quadrant 4:	1.00135
Maximum F_Q (nuclear):	1.869
Maximum $F_{\Delta H}$ (nuclear):	1.433
Maximum Error between Pred. and Meas $F_{\Delta H}$:	5.70%
Average Error between Pred. and Meas. $F_{\Delta H}$:	1.53%
Maximum Error between Expected and Measured Detector Response:	5.80%
RMS of Errors between Expected and Measured Detector Response:	2.1%

MONITOR Results

Minimum F_Q Operational Margin:	23.67%
Minimum F_Q RPS Margin:	15.21%
Minimum F_Q LCO Margin:	54.84%
Minimum $F_{\Delta H}$ Surveillance Margin:	40.08%
Minimum $F_{\Delta H}$ LCO Margin:	23.87%

TABLE 10
CORE POWER DISTRIBUTION RESULTS
60% POWER

Plant Data

Map ID:	FCM/2/08/02
Date of Map:	December 2, 1995
Cycle Burnup:	0.697 EFPD
Power Level:	59.71%
Control Rod Position:	Control Bank D at 216 steps withdrawn
Reactor Coolant System Boron Concentration:	1510 ppmB

CORE Results

Core Average Axial Offset:	7.648 %
Tilt Ratios for Entire Core Height: Quadrant 1:	0.99820
Quadrant 2:	0.99588
Quadrant 3:	1.00347
Quadrant 4:	1.00246
Maximum F_Q (nuclear):	1.706
Maximum $F_{\Delta H}$ (nuclear):	1.409
Maximum Error between Pred. and Meas $F_{\Delta H}$:	6.4%
Average Error between Pred. and Meas. $F_{\Delta H}$:	1.44%
Maximum Error between Expected and Measured Detector Response:	6.20%
RMS of Errors between Expected and Measured Detector Response:	2.30%

MONITOR Results

Minimum F_Q Operational Margin:	24.99%
Minimum F_Q RPS Margin:	20.38%
Minimum F_Q LCO Margin:	52.19%
Minimum $F_{\Delta H}$ Surveillance Margin:	34.03%
Minimum $F_{\Delta H}$ LCO Margin:	21.44%

TABLE 11
CORE POWER DISTRIBUTION RESULTS
100% POWER

Plant Data

Map ID:	FCM/2/08/03
Date of Map:	December 7, 1995
Cycle Burnup:	4.843 EFPD
Power Level:	99.87%
Control Rod Position:	Control Bank D at 211.5 steps withdrawn
Reactor Coolant System Boron Concentration:	1262 ppmB

CORE Results

Core Average Axial Offset:	-1.733%
Tilt Ratios for Entire Core Height: Quadrant 1:	0.99563
Quadrant 2:	1.00168
Quadrant 3:	0.99840
Quadrant 4:	1.00429
Maximum F_Q (nuclear):	1.669
Maximum $F_{\Delta H}$ (nuclear):	1.387
Maximum Error between Pred. and Meas $F_{\Delta H}$:	3.90%
Average Error between Pred. and Meas. $F_{\Delta H}$:	1.02%
Maximum Error between Expected and Measured Detector Response:	3.8%
RMS of Errors between Expected and Measured Detector Response:	1.4%

MONITOR Results

Minimum F_Q Operational Margin:	2.70 %
Minimum F_Q RPS Margin:	22.21%
Minimum F_Q LCO Margin:	22.27%
Minimum $F_{\Delta H}$ Surveillance Margin:	4.20%
Minimum $F_{\Delta H}$ LCO Margin:	13.33%

4.2 One-Point Incore/Excore Calibration

PT/0/A/4600/05D, One-Point Incore/Excore Calibration, is performed using results of power range data taken at 30% power and the incore axial offset measured at 30%. Power ranges are calibrated before exceeding 50% in order to have valid indications of axial flux difference and quadrant power tilt ratio for subsequent power increase. The calibration is checked at 60% power. If necessary, power ranges are calibrated again per PT/0/A/4600/05D or PT/0/A/4600/05A, Incore/Excore Calibration.

Data for Catawba 2 Cycle 8 was obtained on December 1, 1995 and all power range calibrations were completed on December 1, 1995. Results are listed in Table 12. All acceptance criteria were met.

TABLE 12
ONE-POINT INCORE/EXCORE CALIBRATION RESULTS

Reactor Power = 30.41%   Axial Offset = 14.589%

Measured Power Range Currents, μAmps

	N41	N42	N43	N44
Upper	75.0	56.2	67.2	58.0
Lower	65.7	51.2	61.3	52.5

Ratio, Extrapolated (from measured) Currents to "Expected" (from last calibration) Currents

	N41	N42	N43	N44
Upper	0.7689	0.7113	0.7320	0.7003
Lower	0.7953	0.7409	0.7680	0.7258

New Calibration Currents, μAmps

Axial Offset, %	N41		N42		N43		N44	
	Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
+20	254.9	206.1	190.9	161.1	229.3	193.0	197.5	165.3
0	224.1	242.7	168.4	188.0	198.7	224.8	172.4	192.2
-20	193.4	279.2	146.0	214.9	168.1	256.6	147.3	219.0

4.3 Reactor Coolant Loop Delta Temperature Measurement

Reactor coolant system (NC) hot leg and cold leg temperature data is obtained between 50% and 80% power and at 100% power per PT/0/A/4600/26, NC Temperature Calibration, to ensure that full power delta temperature constants (ΔT_0) are valid. ΔT_0 is used in the overpower and overtemperature delta temperature reactor protection functions.

PT/0/A/4600/26 was performed at 75% power on December 4, 1995 and at 100% power on December 5, 1995. No adjustments were required based on results obtained at 75% power. Loops A, B, C, and D were calibrated using full power results. Table 13 summarizes the test results.

TABLE 13
REACTOR COOLANT DELTA TEMPERATURE DATA

Reactor Power = 75.36%

	Loop A	Loop B	Loop C	Loop D
Meas. T_{HOT}, °F	599.9	602.4	599.9	601.3
Meas. T_{COLD}, °F	553.8	554.7	554.2	555.1
Calc. Δh, BTU/lb	61.22	63.67	60.60	61.61
Calc. Δh_0, BTU/lb	81.24	84.49	80.42	81.75
Calc. ΔT_0, °F	59.0	61.1	58.5	59.3
Current ΔT_0, °F	58.1	60.7	58.6	58.9
Difference, °F	0.9	0.4	-0.1	0.4

Reactor Power = 98.08%

	Loop A	Loop B	Loop C	Loop D
Meas. T_{HOT}, °F	613.9	616.9	614.1	615.8
Meas. T_{COLD}, °F	555.3	556.3	555.8	557.0
Calc. Δh, BTU/lb	79.92	83.09	79.49	80.64
Calc. Δh_0, BTU/lb	81.49	84.72	81.05	82.22
Calc. ΔT_0, °F	59.7	61.6	59.3	59.9
Current ΔT_0, °F	58.1	60.7	58.6	58.9
Difference, °F	1.6	0.9	0.7	1.0

4.4 Hot Full Power Critical Boron Concentration Measurement

The hot full power critical boron concentration is measured using PT/0/A/4150/04, Reactivity Anomaly Calculation. Reactor coolant boron concentration is measured (average of three samples) with reactor at essentially all rods out, hot full power, equilibrium xenon conditions. The measured boron is corrected for any off-reference condition (e.g. inserted rod worth, temperature error, difference from equilibrium xenon) and compared to predicted value.

For the purposes of Startup Physics testing, the predicted critical boron concentration is adjusted for the difference between predicted and measured critical boron concentration measured at zero power. The difference between measured boron concentration and adjusted predicted value is used to compare to acceptance criterion(±50 ppmB).

For Catawba 2 Cycle 8, the hot full power critical boron concentration was measured on December 6, 1995. The measured critical boron concentration was 1259 ppmB. Predicted critical boron concentration was 1270 ppmB; when adjusted for difference at zero power, the adjusted predicted critical boron concentration was 1263 ppmB. The difference between measured and adjusted predicted critical boron concentration was -4 ppmB, which met the acceptance criterion.

4.5 Incore/Excore Calibration

Excore power range channels are calibrated at full power per PT/0/A/4600/05A, Incore/Excore Calibration. Incore data (flux maps) and power range currents are obtained at various axial power distributions. A least squares fit of the output of each detector (upper and lower chambers) as a function of measured incore axial offset is determined. The slopes and intercepts of the fit for the upper and lower chamber for each channel are used to determine calibration data for that channel.

This test was performed for Catawba 2 Cycle 8 on December 8 and December 9, 1995. All power range calibrations were completed on December 9. Eight flux maps, with axial offset ranging from -8.570% to +3.561% were used. Table 14 summarizes the results. All acceptance criteria were met.

TABLE 14
INCORE/EXCORE CALIBRATION RESULTS

Full Power Currents, Microamps

Axial Offset, %	N41		N42		N43		N44	
	Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
+20%	291.8	229.6	220.0	180.7	263.1	216.0	227.0	184.7
0%	253.8	265.9	193.6	210.8	228.8	250.9	197.6	213.3
-20%	215.8	302.3	167.3	240.9	194.5	285.9	168.2	241.8

Correction (M_j) Factors

N41	N42	N43	N44
1.396	1.434	1.383	1.415

4.6 Calorimetric Reactor Coolant Flow Measurement

With clean venturis, PT/2/A/4150/13B, Calorimetric Reactor Coolant Flow Measurement, is performed to establish a Primary Loop Delta T correction value to correct Primary power to Secondary Power. This is needed due to the fact that the NC Loop Elbow Tap Correction Factors are now fixed constants per Tech Specifications.

This test was performed on December 5, 1995. The Primary Loop D/T Correction was calculated to be 0.98011. Table 15 summarizes the results. All acceptance criteria were met.

TABLE 15
CALORIMETRIC REACTOR COOLANT FLOW MEASUREMENT

Run Number	Secondary Thermal Output %	Primary Thermal Output %
1	98.1885	100.1737
2	98.1309	100.1279
3	98.0303	100.0166
Average	98.1166	100.1077
	Primary Loop D/T Correction (Secondary Power/ Primary)	0.98011

4.7 Unit Load Steady State Test

In order to verify satisfactory steady state plant operation with reduced T-hot implemented (T-hot reduced ~ 3° to extend Steam Generator Life) per NSM CN-21367, TT/2/A/9200/092, Unit Load Steady State for NSM CN-21367 was performed at approximately 10%,20%, 30%,50%,75%,90% and 100%. With the plant at steady power level data on the following parameters was obtained.

- Reactor Power
- NC Loop Cold Leg Temperature
- NC Loop Hot Leg Temperature
- NC Loop Average Temperature
- NC Loop Delta Temperature
- Pressurizer Level
- Turbine Control Valve Positon
- NIS Power Range Indication
- NIS Intermediate Range Indication

The test was performed at ~10% power on November 30, 1995, ~20% on November 30, 1995, ~30% on December 1, 1995, ~50% on December 1, 1995, ~75% on December 4, 1995, ~90% on December 4, 1995 and ~100% on December 5, 1995. All acceptance criteria were met. Tables 16 through 23 document the results.

Table 16
NC Loop Cold Leg Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
13.94	554.7	555.3	555.0	555.2
17.69	554.5	555.1	554.8	555.1
30.01	553.1	553.8	553.4	553.8
46.96	553.5	554.2	553.9	554.5
75.58	554.1	555.0	554.6	555.5
85.71	553.7	554.6	554.2	555.2
98.54	555.8	556.9	556.4	557.5

Table 17
NC Loop Hot Leg Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
13.94	563.4	564.1	563.3	563.7
17.69	565.5	566.3	565.5	565.9
30.01	571.3	572.8	571.4	572.2
46.96	581.9	583.9	582.2	583.2
75.58	599.1	602.2	599.7	601.1
85.71	604.5	607.6	605.2	606.8
98.54	613.2	616.6	614.0	615.6

Table 18
NC Loop Average Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
13.94	559.5	559.8	559.0	559.4
17.69	560.4	560.8	560.1	560.4
30.01	562.8	563.5	562.4	563.0
46.96	569.4	569.3	568.1	568.9
75.58	577.4	579.0	577.3	578.5
85.71	580.0	581.5	579.9	581.2
98.54	585.4	587.3	585.4	586.9

Table 19
NC Loop Delta Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
13.94	8.7	8.9	8.3	8.5
17.69	10.9	11.2	10.7	10.8
30.01	18.2	19.0	18.0	18.3
46.96	28.3	29.7	28.3	28.8
75.58	44.9	47.1	45.2	45.7
85.71	50.8	53.0	51.0	51.6
98.54	57.4	59.7	57.7	58.1

Table 20
Pressurizer Level Data

Power Level	PZR Level Setpoint	PZR Level Channel 1	PZR Level Channel 2	PZR Level Channel 3
13.94	27.7	26.8	27.4	25.5
17.69	28.7	27.5	28.1	26.2
30.01	31.3	30.7	31.2	29.3
46.96	37.1	36.6	37.0	35.2
75.58	46.6	45.9	46.4	44.8
85.71	49.1	48.4	48.9	47.2
98.54	54.7	54.3	54.7	53.1

Table 21
Turbine Control Valve Positions

Power Level	CV 1	CV 2	CV 3	CV 4
13.97	10.40%	10.70%	0.0%	0.0%
17.69	12.5%	12.8%	0.0%	0.0%
30.01	20.5%	20.7%	0.0%	0.0%
46.96	32.0%	32.3%	0.0%	0.0%
75.58	100.0%	100.0%	17.5%	0.0%
85.71	100.0%	100.0%	46.6%	0.1%
98.66	100.0%	100.0%	100.0%	35.30%

Table 22
NIS Power Range Data
(μAmps)

Power Level	N-41	N-42	N-43	N-44
13.97	65	49	58	51
17.69	77	58	70	59
30.01	140	105	127	110
46.96	226	173	206	176
75.53	378	292	347	297
85.71	432	336	400	343
98.70	509	397	471	402

Table 23
NIS Intermediate Range Data
(μAmps)

Power Level	N-35	N-36
13.94	42	37
17.69	54	47
30.01	88	77
46.96	144	128
75.58	239	214
85.71	273	245
98.54	328	293

4.8 Unit Load Transient Test

The Unit Load Transient Test, TT/2/A/9200/91, was performed to verify proper operation of the modifications performed on various control systems for NSM CN-21367, T-Hot Reduction. The purpose of the test was to demonstrate proper plant response,including automatic control system performance, to a ~10% step load change. The test verified that the control systems worked as designed to prevent the following plant transients (in response to a ~10% step load change):

- Reactor Trip
- Turbine Trip
- Actuation of Safety Injection
- Pressurizer and Steam Safeties or PORVs Lifting

This test satisfied the transient retest as required for the Post Modification Testing for T-Hot Reduction.

This test was performed at ~75% power on December 4, 1995. All acceptance criteria for the test were met. Figures 5 through 12 show plant parameters during the test.

FIGURE 5
Unit Load Transient Test Power Range Level

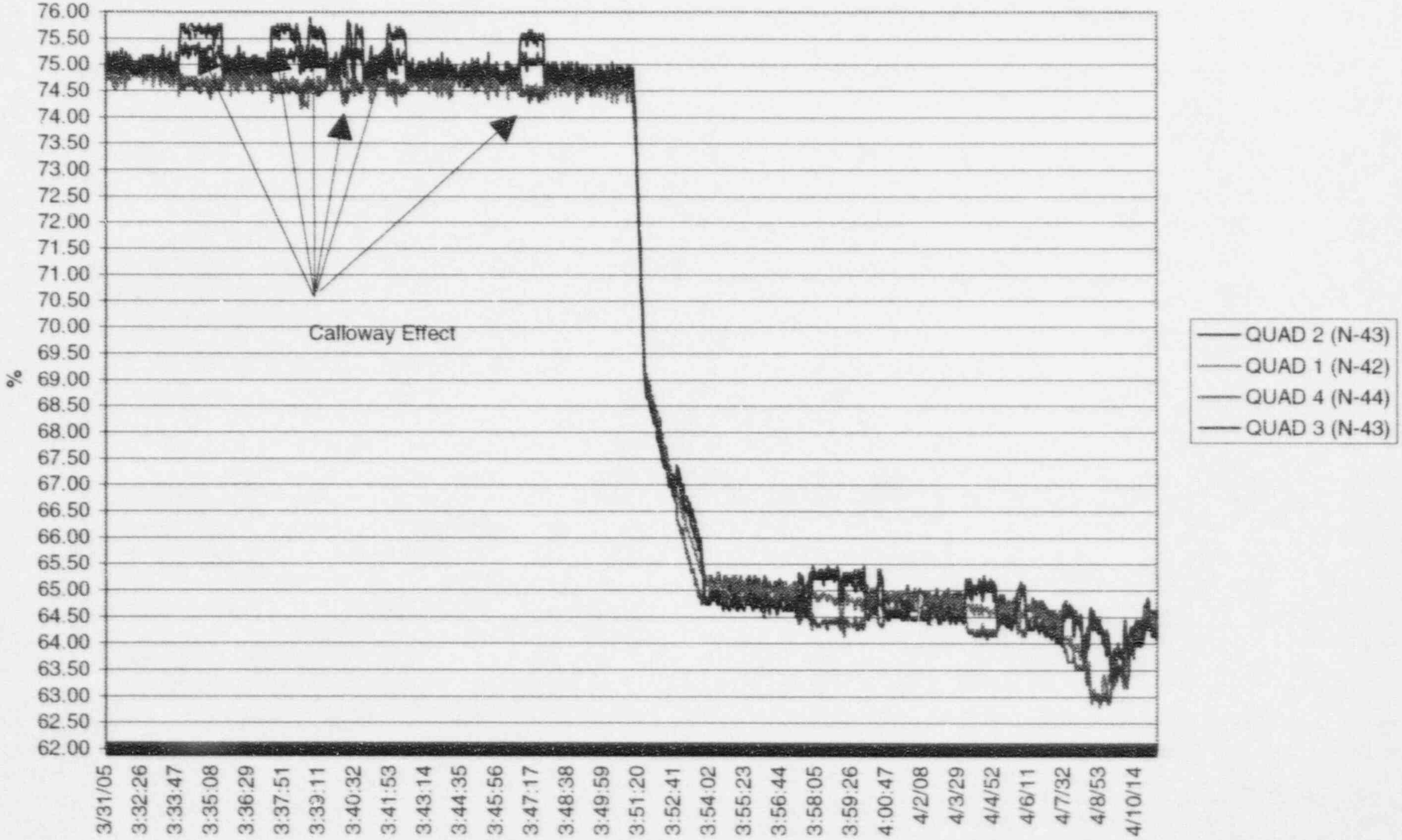


Figure 6
Unit Load Transient Test NC Loop Highest T-avg T-ref

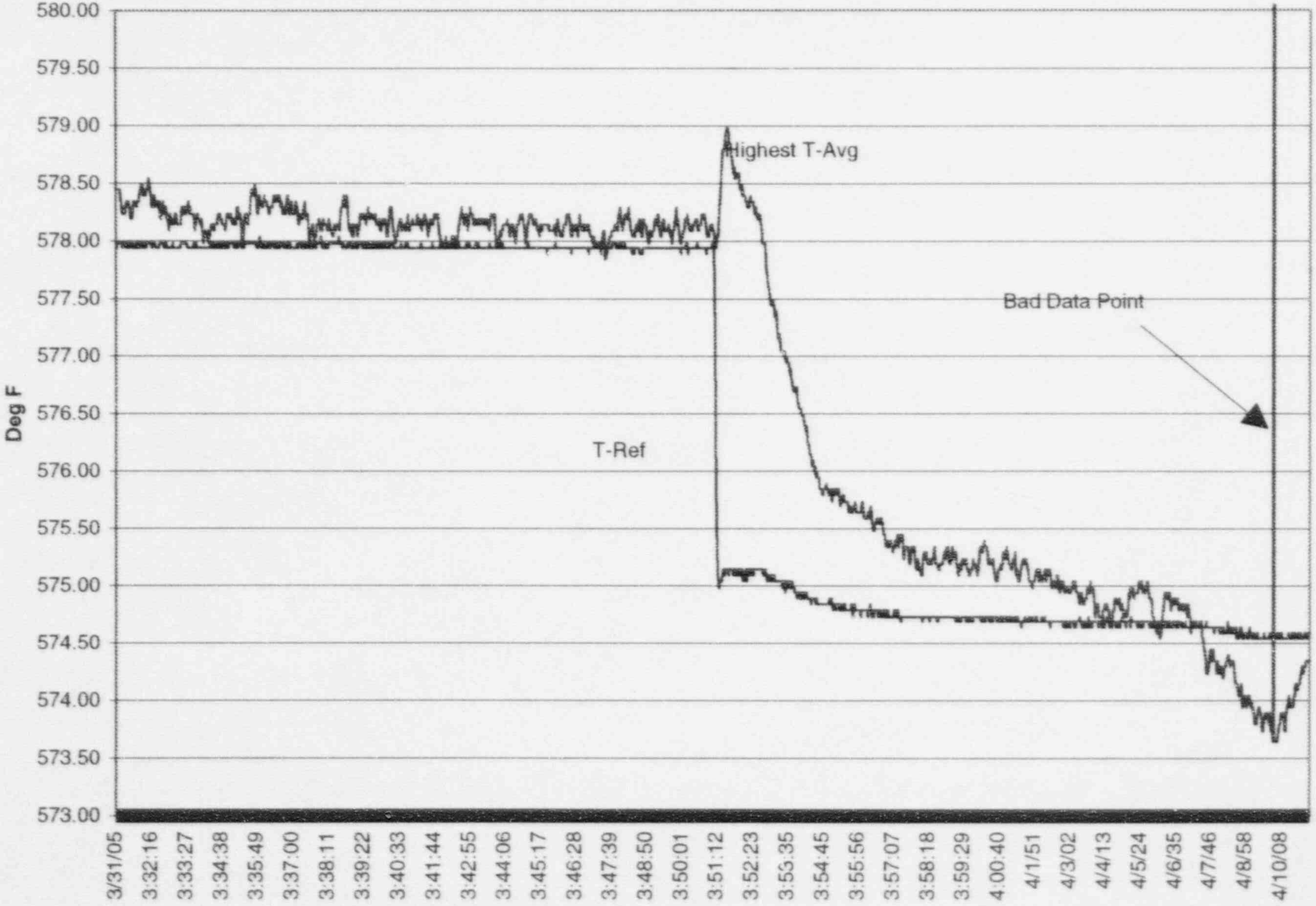


FIGURE 7
Unit Load Transient Test Pressurizer Pressure and Level

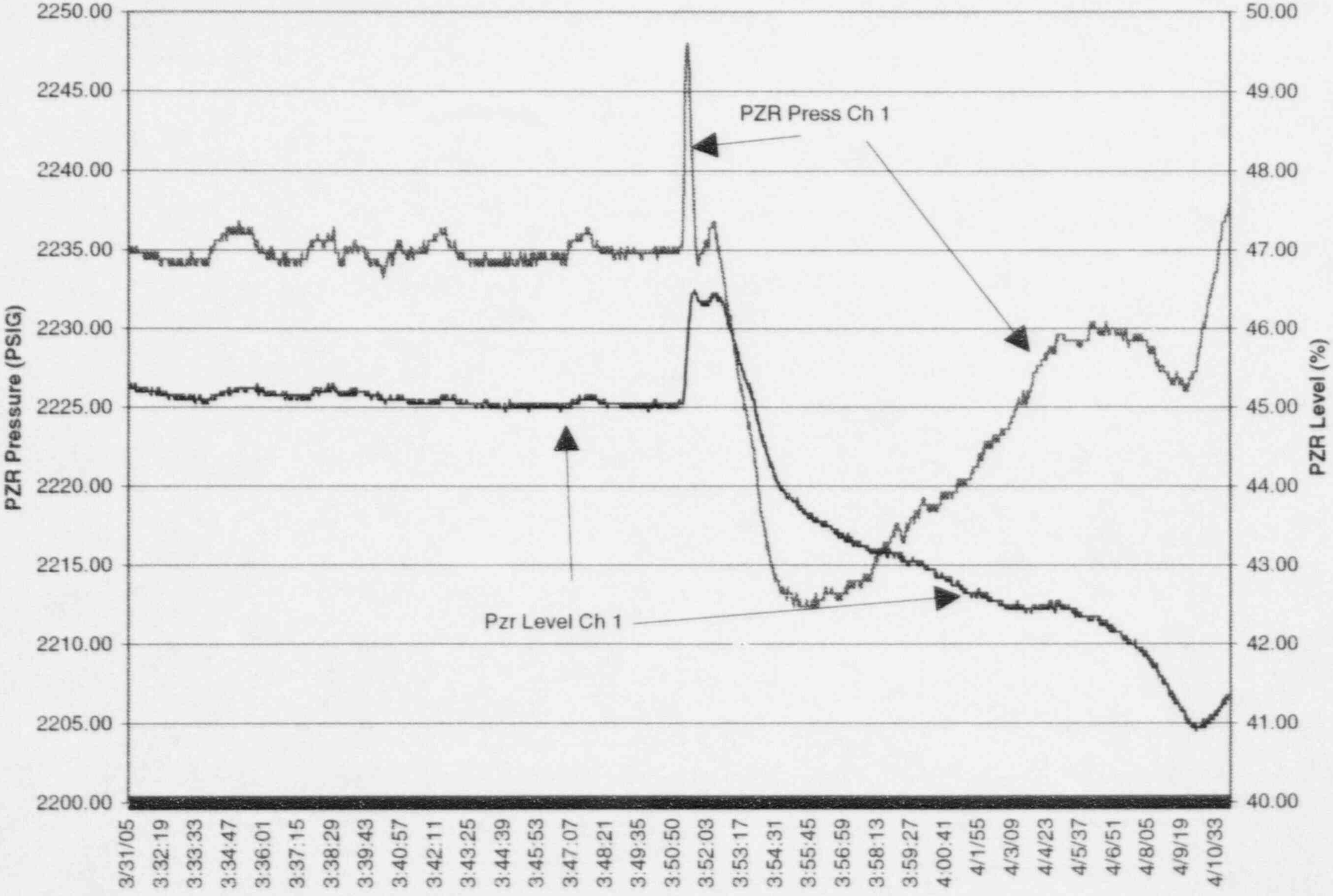


FIGURE 8
Unit Load Transient Test S/G Steam Pressure

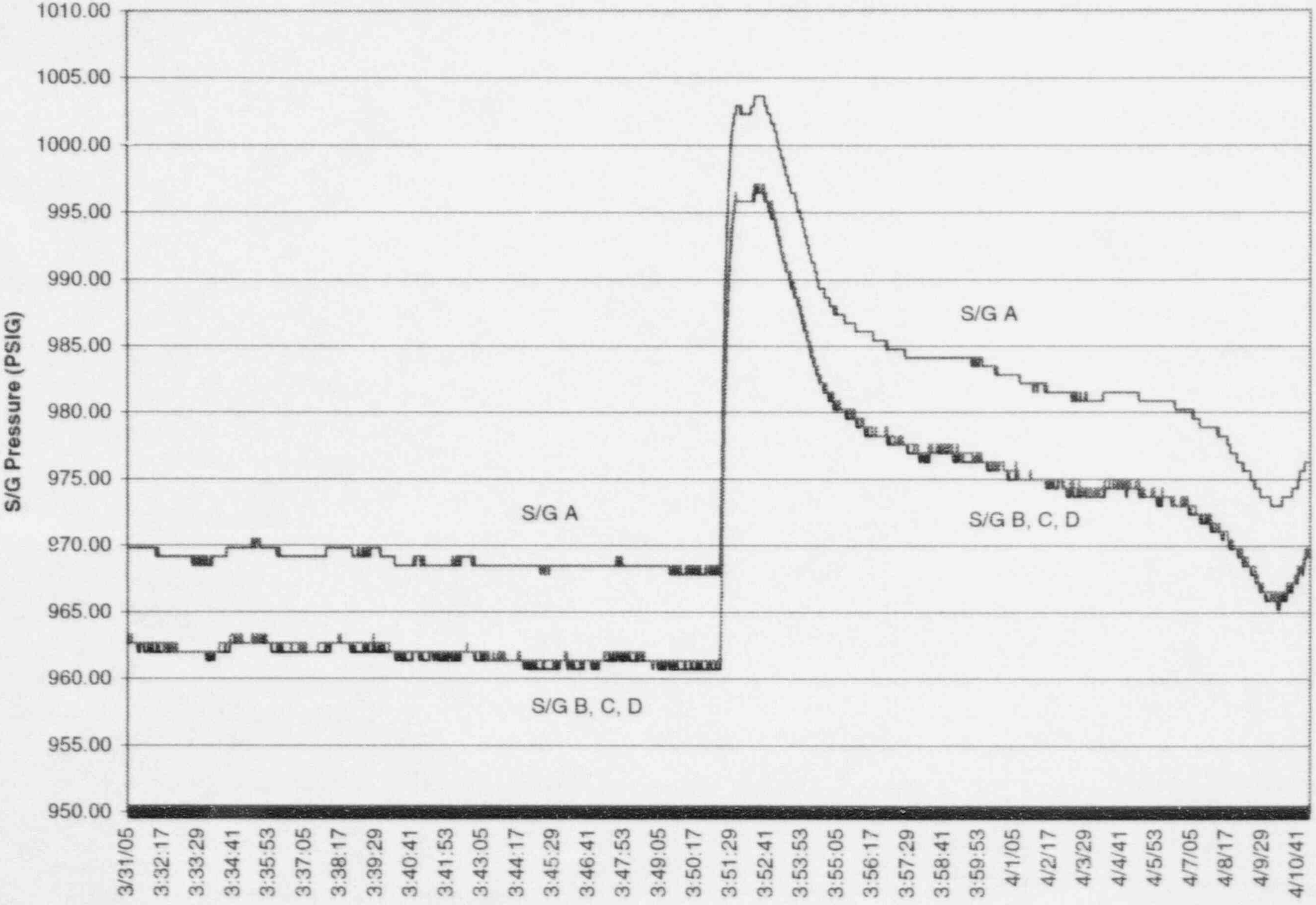


FIGURE 9
Unit Load Transient Test S/G Narrow Range Level

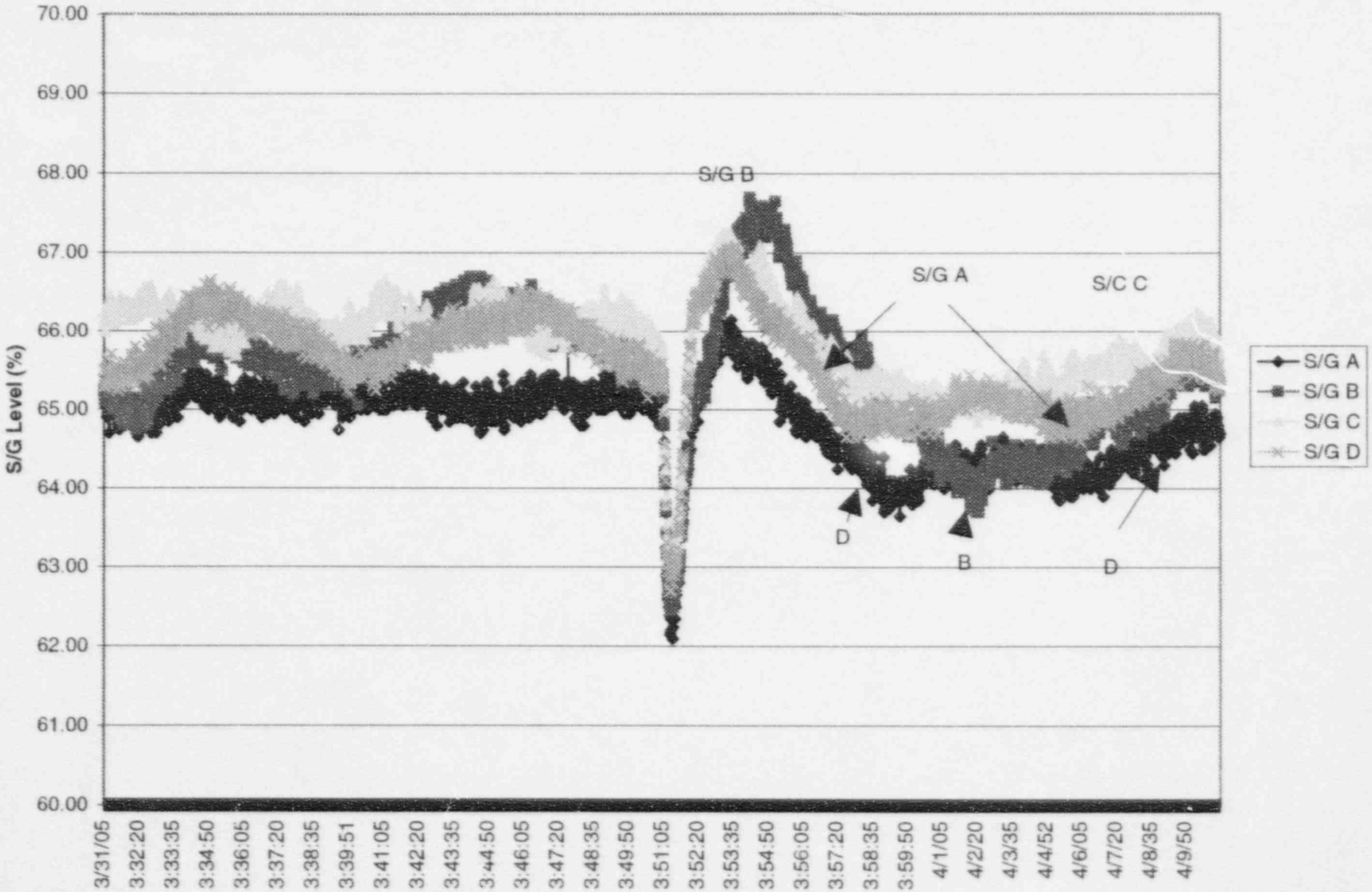


Figure 10
Unit Load Transient Test Generator Megawatts and Turbine 1st Stage Pressure

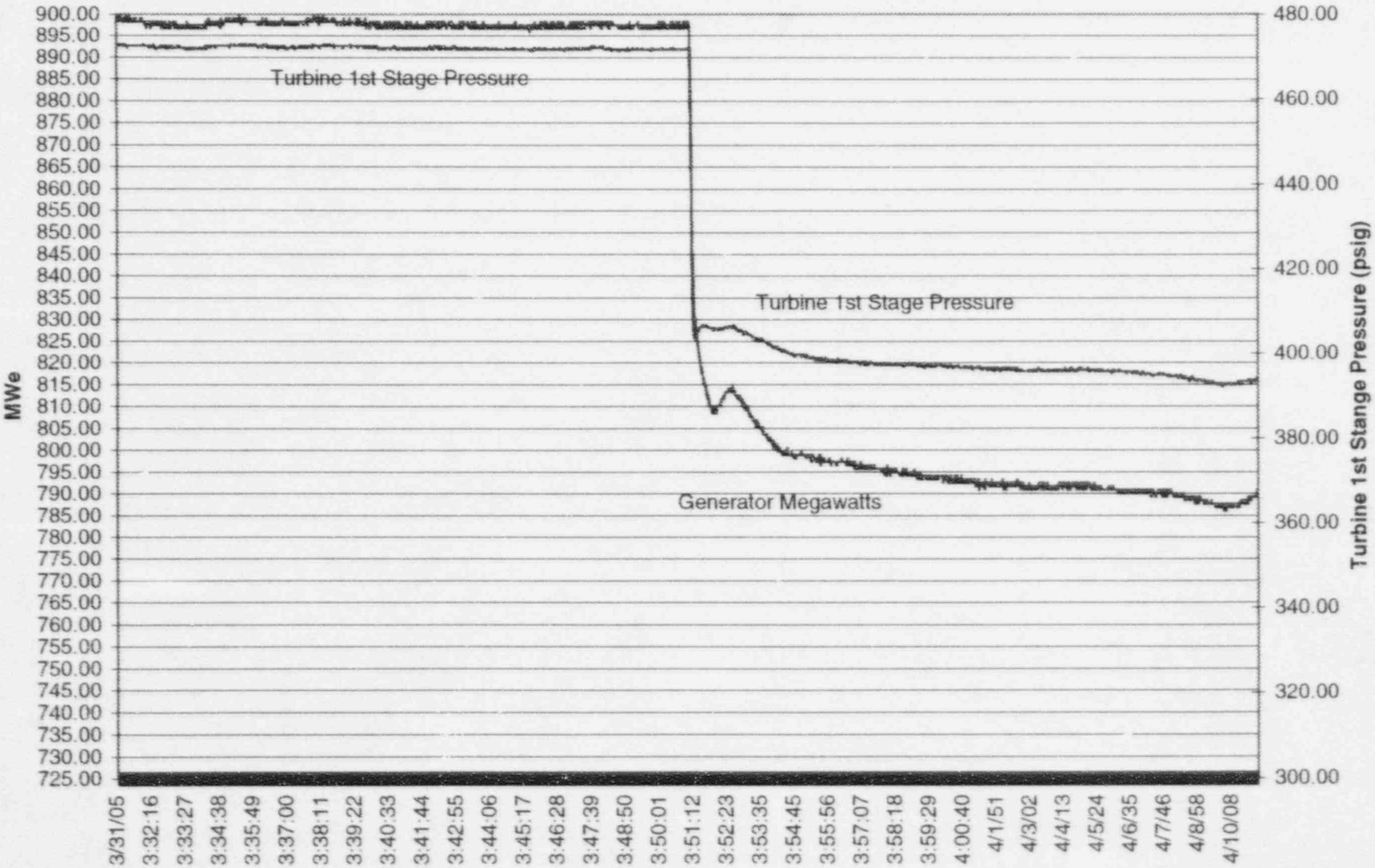


FIGURE 11
Unit Load Transient Test CF Pump Turbine Speed

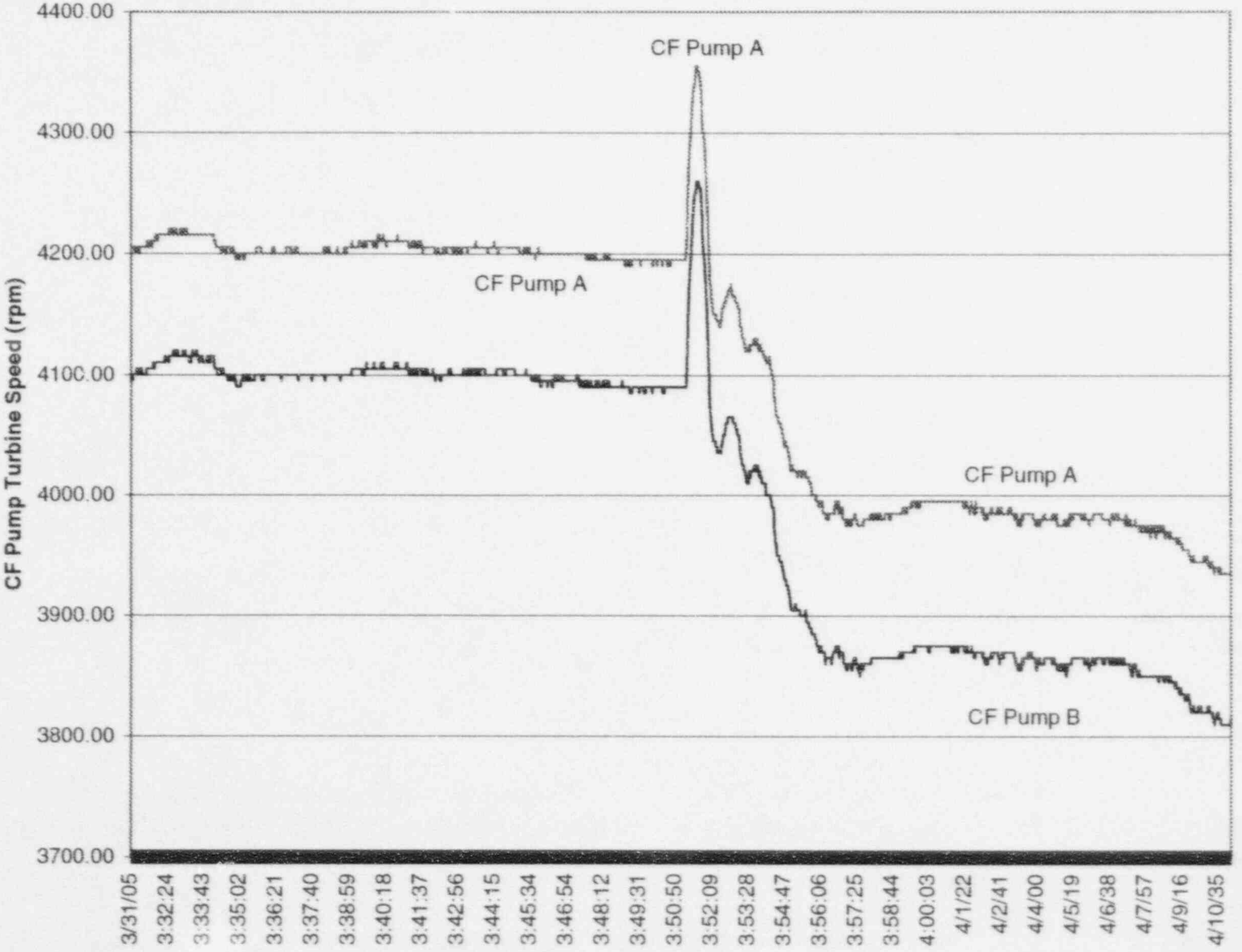


FIGURE 12
Unit Load Transient Test S/G Feedwater Flow