Duke Power Company
McGuire Nuclear Station
Unit 1 Cycle 12

STARTUP REPORT

August 1997

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

McGuire Unit One Cycle 12 includes a feed batch of 68 MkBW fuel assemblies manufactured by Framatome Cogema Fuels (FCF). The feed batch enrichments are 3.67% (w/o). Burnable poison rod assemblies used in the feed batch were also manufactured by FCF.

McGuire Unit One Cycle 12 core loading began at 1208 on April 24, 1997 and ended at 1758 on April 26, 1997. Initial criticality for Cycle 12 occurred at 0244 on May 18, 1997. Zero Power Physics Testing was completed at 0155 on May 19, 1997. The unit reached full power at 2306 on May 31, 1997. Power Escalation testing, including testing at full power, was completed by June 9, 1997.

Table 1 contains some important characteristics of the McGuire 1 Cycle 12 core design.

TABLE 1
M1C12 CORE DESIGN DATA

1. M1C11 end of cycle burnup: 361 EFPD

2. M1C12 design length: 370 ± 10 EFPD

Region	Fuel Type	Number of Assemblies	Enrichment, w/o U^{235}	Loading, MTU**	Cycles Burned
11A	MkBW	1	3.45	0.4562	3
12A	MkBW	52	3.60	23.7224	2
13A	MkBW	48	3.40	21.8976	1
13B	MkBW	24	3.55	10.9488	1
14A	MkBW	68	3.67/2.00*	31.0216	0
Totals		193		88.0466	

* 2.00 w/o enriched U blanketed fuel assemblies (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

Sections 2.1 through 2.2 describe results of precritical testing for McGuire 1 Cycle 12.

2.1 Total Core Reloading

The Cycle 12 core was loaded under the direction of PT/0/A/4150/33, Total Core Reloading. Plots of Inverse Count Rate Ratio (ICRR) versus number of fuel assemblies loaded were maintained for each source range channel.

Core loading commenced at 1208 on April 24, 1997 and concluded at 1758 on April 26, 1997. Core loading was verified by PT/0/A/4550/03C, Core Verification, which was completed by 2200 on April 26, 1997.

2.2 Preliminary NIS Calibration

Periodic test procedure PT/0/A/4600/78, Prestartup NIS Realignment Following Refueling, 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 NIS full power currents are similarly adjusted. Intermediate Range NIS Rod Stop and Reactor Trip setpoints are checked and revised as necessary for initial power ascension. Added conservatism was applied to setpoints to account for any uncertainties that may have been introduced by the T-AVG reduction resulting from replacement of the Steam Generators during the outage.

Table 4 shows the calibration data calculated by PT/0/A/4600/78. Calculations were performed on February 19, 1997. Calibrations were complete by April 30, 1997.

Evaluation of the setpoints at approximately 30% power in accordance with PT/0/A/4150/21, Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing, indicated these calculated setpoints were conservative.

TABLE 2
PRELIMINARY NIS CALIBRATION DATA

Intermediate Range

Channel	Ratio (BOC 12 ÷ Cycle 11)	BOC 12 Reactor Trip Setpoint, µAmps	BOC 12 Rod Stop Setpoint, µAmps
N35	0.698	41.88	33.51
N36	0.694	43.29	34.63

Power Range

Channel	Ratio (BOC 12 ÷ Cycle 11)	Axial Offset, %	Cycle 11 Full Power Current, µAmps		BOC 12 Full Power Current, µAmps	
			Upper	Lower	Upper	Lower
N41	0.893	+30	262.7	183.8	187.9	131.4
		0	217.5	233.2	155.5	166.8
		-30	172.4	282.6	123.3	202.1
N42	0.893	+30	279.2	190.8	199.4	136.3
		0	231.2	240.7	165.1	171.9
		-30	183.3	290.6	130.9	207.5
N43	0.893	+30	264.8	191.2	189.3	136.7
		0	221.9	245.1	158.6	175.2
		-30	179.0	298.9	127.9	213.7
N44	0.885	+30	259.3	185.2	183.7	131.2
		0	213.0	238.1	150.9	168.7
		-30	166.7	291.0	118.1	206.2

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/4150/21, Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing. Test measurements are made below the Point of Nuclear Heat using the output of one Power Range NIS 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 (Boron Endpoint)
- All Rods Out Isothermal Temperature Coefficient measurement
- Measurement of Reference Bank worth by dilution
- Control Rod Worth Measurement by Rod Swap

Zero power physics testing for McGuire 1 Cycle 12 began at 0114 on May 18, 1997 commencing with rod withdrawal for approach to criticality. ZPPT ended at 0155 on May 19, 1997 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 McGuire 1 Cycle 12 was achieved per PT/0/A/4150/28, Criticality Following a Change in Core Nuclear Characteristics. In this procedure, an Estimated Critical Rod Position (ECP) is calculated based on latest available Reactor Coolant boron concentration. Control rods are withdrawn 50 to 60 steps at a time while monitoring source range channel response. Inverse Count Rate Ratio (ICRR) is plotted for each source range channel. ICRR data is used to project critical rod position. If projected critical rod position is acceptable, rod withdrawal may continue.

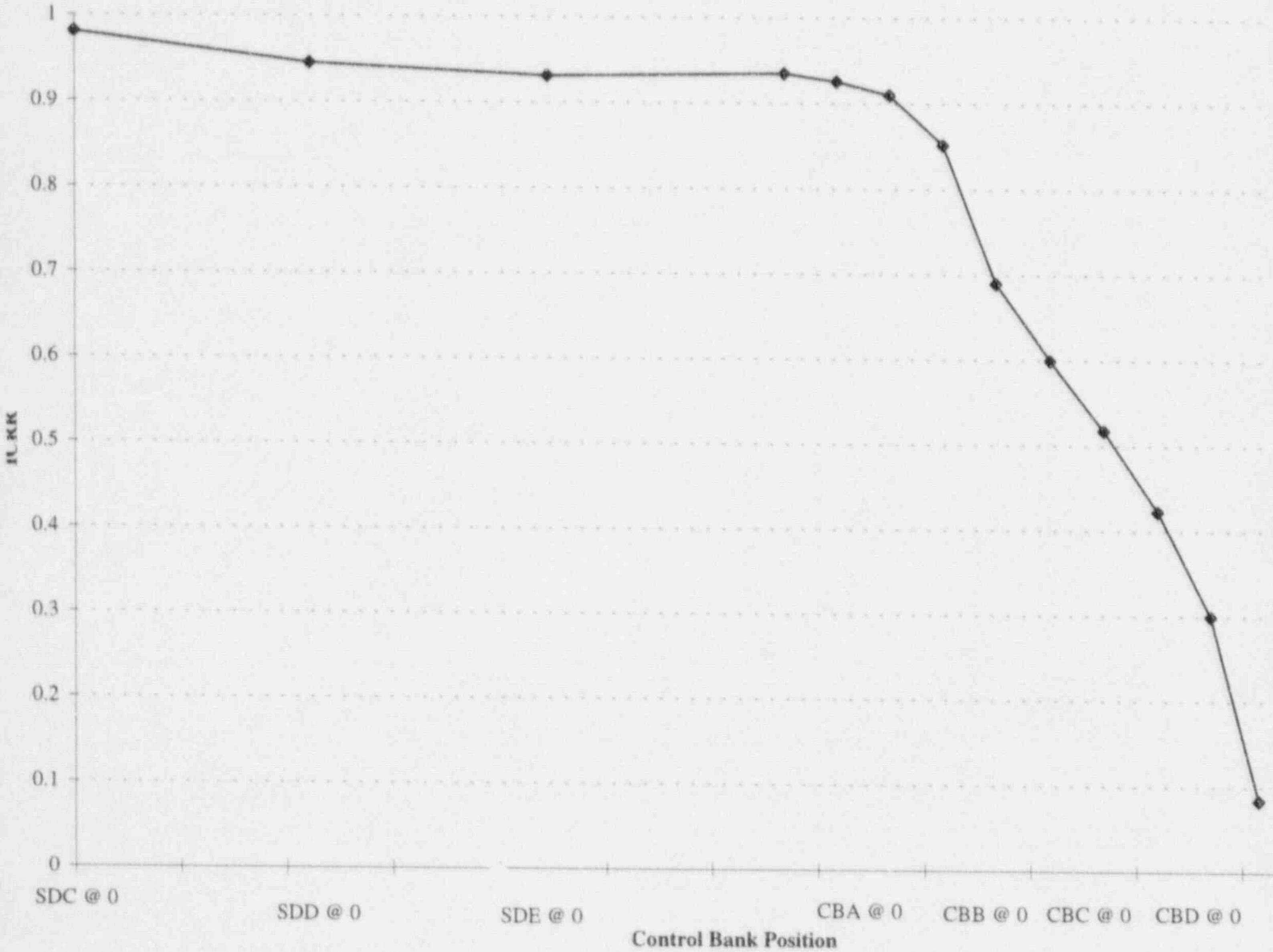
Rod withdrawal for the approach to criticality began at 0114 on May 18, 1997. Criticality was achieved at 0244 on May 18, 1997 with Control Bank D at 111 steps withdrawn.

Figure 3 shows the ICRR behavior during the approach to criticality. All acceptance criteria of PT/0/A/4150/28 were met.

TABLE 3
SUMMARY OF ZPPT RESULTS

PARAMETER	MEASURED VALUE	PREDICTED VALUE OR ACCEPTANCE CRITERIA
Nuclear Heat	4.75×10^{-7} amps (N41)	N/A
ZPPT Test Band	10^{-8} to 10^{-7} amps (N41)	N/A
ARO Critical Boron	1803 ppmB	1778 ± 50 ppmB
ARO ITC	-1.24 pcm/°F	-1.37 ± 2 pcm/°F
ARO MTC	+0.44 pcm/°F	+0.31 pcm/°F
Reference Bank (Shutdown Bank B) Worth	922.0 pcm	881 ± 132 pcm
Ref. Bank in Critical Boron	1683 ppmB	1662 ppmB
Differential Boron Worth	-7.75 pcm/ppmB	-7.55 pcm/ppmB
Control Bank D Worth	607.5 pcm	614 ± 200 pcm
Control Bank C Worth	727.8 pcm	730 ± 219 pcm
Control Bank B Worth	655.0 pcm	622 ± 200 pcm
Control Bank A Worth	385.1 pcm	390 ± 200 pcm
Shutdown Bank E Worth	524.7 pcm	508 ± 200 pcm
Shutdown Bank D Worth	425.7 pcm	424 ± 200 pcm
Shutdown Bank C Worth	418.7 pcm	423 ± 200 pcm
Shutdown Bank A Worth	266.2 pcm	268 ± 200 pcm
Total Rod Worth	4932.6 pcm	4860 ± 486 pcm

FIGURE 1
ICRR vs. CONTROL ROD POSITION DURING APPROACH TO CRITICALITY



3.2 Source Range/Intermediate Range Overlap Data

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

Overlap data for Cycle 12 was obtained per PT/0/A/4150/28, Criticality Following a Change in Core Nuclear Characteristics, on May 8, 1997. 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	1100	850	2×10^{-11}	2×10^{-11}
OAC	1268	835	2×10^{-11}	2×10^{-11}
FINAL DATA: NIS Meters	14,000	8,000	2.0×10^{-10}	2.0×10^{-10}
OAC	10586	7692	1.3×10^{-10}	1.4×10^{-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 12, the Point of Nuclear Heat Addition was determined per PT/0/A/4150/21, Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing, on May 18, 1997. Table 5 summarizes the data obtained.

The Zero Power Physics Test Band was set at 10^{-8} to 10^{-7} amps on Power Range channel N41 (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 (N41), amps	Intermediate Range Channel N35, amps	Intermediate Range Channel N36, amps
RUN #1	3.5×10^{-7}	3×10^{-7}	2×10^{-7}
RUN #2	6×10^{-7}	6×10^{-7}	6×10^{-7}

3.4 Reactivity Computer Checkout

The reactivity computer checkout was performed per PT/0/A/4150/21, Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing, to verify that the Power Range channel connected to the reactivity computer can provide reliable reactivity data. Reactivity Insertions of approximately +25 and +40 pcm are made. The resulting Periods are measured and used to determine the corresponding theoretical reactivities. The measured reactivity is compared to the theoretical reactivity and verified to be within 4.0%.

The checkout was performed for Cycle 12 on May 18, 1997. Table 6 lists the results of the reactivity insertion. The acceptance criterion was met.

TABLE 6
REACTIVITY COMPUTER CHECKOUT

Period, seconds	Theoretical Reactivity, pcm	Measured Reactivity, pcm	Absolute Error, pcm	Percent Error,%
186.11	33.37	33.04	0..33	-0.97
223.86	28.42	28.24	0.18	-0.61

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 12 beginning of cycle, hot zero power, all rods out, critical boron concentration was measured on May 18, 1997 per PT/0/A/4150/10, Boron Endpoint Measurement. The ARO, HZP boron concentration was measured to be 1803 ppmB. Predicted ARO critical boron concentration was 1778 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, state point data is obtained prior to cooldown. A Reactor Coolant System cooldown is initiated, within administrative cooldown limits. When sufficient data (at least 4 °F) is obtained, state point 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 12 ITC was measured per PT/0/A/4150/12, Isothermal Temperature Coefficient Measurement, on May 18, 1997. No additional cooldown/heatup cycles were required because of good agreement between initial heatup and cooldown results. Table 7 summarizes the data obtained during the measurement.

Average ITC was determined to be -1.24 pcm/°F. Predicted ITC was -1.37 pcm/°F. Measured ITC was therefore within acceptance criterion of predicted ITC ± 2 pcm/°F.

The MTC was determined to be +0.44 pcm/°F. This value was used with procedure PT/0/A/4150/31, Determination of Temporary Rod Withdrawal Limits to Ensure Moderator Temperature Coefficient Within Limits of Technical Specifications, 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

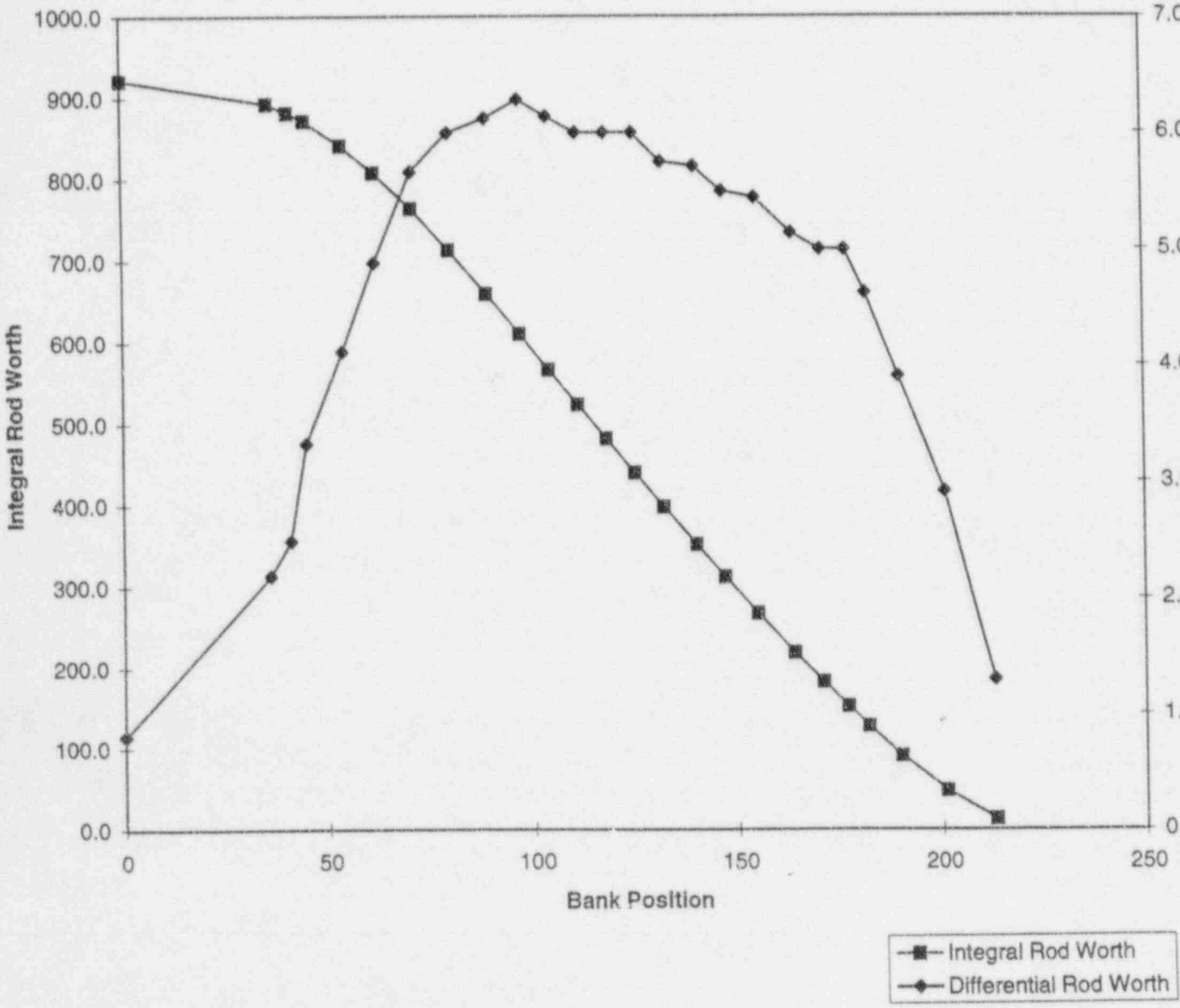
	ΔT, °F	$\Delta\rho$, pcm	T_{avg}, °F	ITC, pcm/°F
Cooldown	-4.8	+5.2	554.6	-1.08
Heatup	+5.0	-7.0	554.6	-1.40
				Average: -1.24

3.7 Reference Bank Worth Measurement by Dilution

The control rod bank predicted to have the highest worth is designated the Reference Bank. This RCCA bank is measured by inserting the bank (with all other rod banks fully withdrawn) in discrete steps while slowly diluting the Reactor Coolant System (at rate < 500 pcm/hr). 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 12 Reference Bank (Shutdown Bank B) worth was measured on May 18, 1997 per PT/0/A/4150/11, Control Rod Worth Measurement. Figure 3 shows integral worth of Reference Bank versus bank position. The Reference Bank was measured to be worth 922 pcm; predicted worth was 881 pcm. The acceptance criterion, measured worth within ± 15% of predicted, was met.

FIGURE 2
INTEGRAL AND DIFFERENTIAL WORTH OF REFERENCE BANK



3.8 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 measured 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 12, Hot Zero Power differential boron worth was calculated to be -7.75 pcm/ppmB per PT/0/A/4150/11, Control Rod Worth Measurement. The predicted value was -7.55 pcm/ppmB. There is no acceptance criterion on this measurement.

3.9 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 12 rod worth measurement by Rod Swap was performed on May 18 - 19, 1997 per PT/0/A/4150/11A, Control Rod Worth Measurement - 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	922.0	881	-41	4.7
Shutdown A	882	100	592	1.040	266.2	268	+2	-0.7
Control A	882	122	458	1.085	385.1	390	+5	-1.3
Shutdown C	882	123	452	1.025	418.7	423	+4	-1.0
Shutdown D	882	124	446	1.023	425.7	424	-2	+0.4
Shutdown E	882	129	416	0.859	524.7	508	-17	+3.3
Control D	882	162	230	1.194	607.5	614	+6	-1.1
Control B	882	154	274	0.830	655.0	622	-33	+5.3
Control C	882	174	168	0.918	727.8	730	+2	-0.3
				Total	4932.6	4860	-73	+1.5

4.0 POWER ESCALATION TESTING

Power Escalation Testing is performed during the initial power ascension to full power for each cycle and is controlled by PT/0/A/4150/21, Post Refueling Controlling Procedure for Criticality, Zero Power Physics, and Power Escalation Testing. Tests are performed from 0% through 100% power with major testing plateaus at ~30%,~75%, and 100% power.

Significant tests performed during McGuire 1 Cycle 12 Power Escalation were:

- Core Power Distribution (at ~30%, ~78%, and 100% power)
- One-Point Incore/Excore Calibration (at ~30% power)
- Reactor Coolant Delta Temperature Measurement (at 90% 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/4150/21)
- Unit Load Steady State - at 30%, 78%, 90%, 98%, and 100% (Steam Generator Replacement Post-Mod testing)
- Unit Load Transient Test - at 38% and 78% (Steam Generator Replacement Post-Mod testing)
- Replacement S/G Functional Tuning and Testing of DFCS - at 10%, 30%, 78%, and 100% (Steam Generator Replacement Post-Mod testing)
- Evaluation of Intermediate Range NIS Rod Stop and Rx Trip Setpoints

Power Escalation Testing for McGuire 1 Cycle 12 began on May 19, 1997. Full power was reached on May 29, 1997. Full power testing was completed on June 11, 1997. Sections 4.1 through 4.9 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 75%), 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 NIS 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 COMET code (adapted from Shangstrom Nuclear Associates' CORE package and FCF's MONITOR code, respectively).

The McGuire 1 Cycle 12 Core Power Distribution measurements were performed on May 21, 1997 (30% power), May 25, 1997 (78% power), and June 9, 1997 (100% power). Tables 9 through 11 summarize the results. All acceptance criteria were met.

TABLE 9
CORE POWER DISTRIBUTION RESULTS
30% POWER

Plant Data

Map ID:	m1c12f001
Date of Map:	May 21, 1997
Cycle Burnup:	0.21 EFPD
Power Level:	29.14% F.P.
Control Rod Position:	Control Bank D at 203 Steps Wd
Reactor Coolant System Boron Concentration:	1773 ppmB

COMET Results

Core Average Axial Offset:	12.328%
Tilting Factors for Entire Core Height: Quadrant 1:	0.99450
Quadrant 2:	1.00481
Quadrant 3:	0.99569
Quadrant 4:	1.00499
Maximum F_Q (nuclear):	1.983
Maximum $F_{\Delta H}$ (nuclear):	1.459
Maximum Error between Pred. and Meas $F_{\Delta H}$:	4.22%
Average Error between Pred. and Meas. $F_{\Delta H}$:	1.20%
Maximum Error between Expected and Measured Detector Response:	4.39%
RMS of Errors between Expected and Measured Detector Response:	1.62%

MONITOR Results

Minimum F_Q Operational Margin:	22.04%
Minimum F_Q RPS Margin:	14.80%
Minimum F_Q LCO Margin:	57.11%
Minimum $F_{\Delta H}$ Surveillance Margin:	37.40%
Minimum $F_{\Delta H}$ LCO Margin:	25.15%

TABLE 10
CORE POWER DISTRIBUTION RESULTS
78% POWER

Plant Data

Map ID:	m1c12f002
Date of Map:	May 25, 1997
Cycle Burnup:	1.78 EFPD
Power Level:	78.39% F.P.
Control Rod Position:	Control Bank D at 200 Steps Wd
Reactor Coolant System Boron Concentration:	1488 ppmB

COMET Results

Core Average Axial Offset:	0.383%
Tilting Factors for Entire Core Height: Quadrant 1:	0.99691
Quadrant 2:	1.00009
Quadrant 3:	1.00235
Quadrant 4:	1.00064
Maximum F_Q (nuclear):	1.782
Maximum $F_{\Delta H}$ (nuclear):	1.417
Maximum Error between Pred. and Meas $F_{\Delta H}$:	3.57%
Average Error between Pred. and Meas. $F_{\Delta H}$:	1.02%
Maximum Error between Expected and Measured Detector Response:	3.74%
RMS of Errors between Expected and Measured Detector Response:	1.40%

MONITOR Results

Minimum F_Q Operational Margin:	9.18%
Minimum F_Q RPS Margin:	14.83%
Minimum F_Q LCO Margin:	39.57%
Minimum $F_{\Delta H}$ Surveillance Margin:	16.65%
Minimum $F_{\Delta H}$ LCO Margin:	20.31%

TABLE 11
CORE POWER DISTRIBUTION RESULTS
100% POWER

Plant Data

Map ID:	m1c12f005
Date of Map:	June 8, 1997
Cycle Burnup:	12.46 EFPD
Power Level:	99.82% F.P.
Control Rod Position:	Control Bank D at 214 Steps Wd
Reactor Coolant System Boron Concentration:	1211 ppmB

COMET Results

Core Average Axial Offset:	-0.685%
Tilting Factors for Entire Core Height: Quadrant 1:	0.99515
Quadrant 2:	1.00413
Quadrant 3:	0.99874
Quadrant 4:	1.00198
Maximum F_Q (nuclear):	1.737
Maximum $F_{\Delta H}$ (nuclear):	1.405
Maximum Error between Pred. and Meas $F_{\Delta H}$:	2.82%
Average Error between Pred. and Meas. $F_{\Delta H}$:	0.78%
Maximum Error between Expected and Measured Detector Response:	3.07%
RMS of Errors between Expected and Measured Detector Response:	1.04%

MONITOR Results

Minimum F_Q Operational Margin:	10.04%
Minimum F_Q RPS Margin:	17.61%
Minimum F_Q LCO Margin:	24.99%
Minimum $F_{\Delta H}$ Surveillance Margin:	6.30%
Minimum $F_{\Delta H}$ LCO Margin:	15.50%

4.2 One-Point Incore/Excore Calibration

PT/0/A/4600/02FD, One-Point Incore/Excore Calibration, is performed using results of Power Range NIS data taken at 30% power and the incore axial offset measured at 30%. Power Range channels are calibrated before exceeding 50% in order to have valid indications of Axial Flux Difference and Quadrant Power Tilt Ratio for subsequent power ascension. The calibration is checked using the intermediate power level flux map (78% F.P. for M1C12). If necessary, Power Range NIS is recalibrated per PT/0/A/4600/02F or PT/0/A/4600/02G, Incore and NIS Recalibration.

Data for McGuire 1 Cycle 12 was obtained on May 21, 1997 and all Power Range NIS calibrations were completed on May 23, 1997. All acceptance criteria were met.

4.3 Reactor Coolant Loop Delta Temperature Measurement

Reactor Coolant System (NC) Hot Leg and Cold Leg temperature data is normally obtained at approximately 90% and 100% power per PT/0/A/150/40, NC Loop Delta-T, RTAS, and OPDT &OTDT Channel Check Criteria Evaluation, to ensure that full power delta temperature constants (ΔT_0) are valid. ΔT_0 is used in the Over-power and Over-temperature Delta Temperature reactor protection functions.

In the case of M1C12, the four loop ΔT_0's were each preliminarily established at 59.97°F, 57.82°F, 58.97°F, and 57.49°F for Loops A-D, per Steam Generator Replacement Project and previously observed biases. Portions of PT/0/A/4150/40 were completed at 90% power on May 26, 1997 to verify proper conservatism in the calibrations, and the entire procedure was performed after 100% power equilibrium conditions were achieved. All four NC Loop ΔT_0's were adjusted using full power results. Table 12 summarizes the test results.

TABLE 12
REACTOR COOLANT DELTA TEMPERATURE DATA

Reactor Power = 99.67%

	Loop A	Loop B	Loop C	Loop D
Meas. T_{HOT}, °F	614.5	613.5	614.6	612.9
Meas. T_{COLD}, °F	554.6	555.4	555.5	554.8
Calc. ΔT_0, °F	59.9	58.1	59.1	58.1

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 McGuire 1 Cycle 12, the Hot Full Power critical boron concentration was measured on June 11, 1997. The measured critical boron concentration was 1195.9 ppmB. Predicted critical boron concentration was 1179.2 ppmB; when adjusted for difference at zero power, the adjusted predicted critical boron concentration was 1170.9 ppmB. The difference between measured and adjusted predicted critical boron concentration was -8.3 ppmB, which met the acceptance criterion.

4.5 Incore/Excore Calibration

Excore NIS Power Range channels are calibrated at full power per PT/0/A/4600/02G, Incore and NIS Recalibration. Incore data (flux maps) and Power Range NIS 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 McGuire 1 Cycle 12 on June 7 and 8, 1997. All Power Range NIS calibrations were completed on June 9. Eight flux maps, with axial offset ranging from -6.568% to +3.989% were used. Table 13 summarizes the results. All acceptance criteria were met.

TABLE 13
INCORE/EXCORE CALIBRATION RESULTS

Full Power Currents, Microamps

Axial Offset, %	N41		N42		N43		N44	
	Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
+30%	220.3	153.9	236.9	162.3	218.1	158.7	211.9	150.1
0%	178.3	195.2	193.4	205.6	179.3	202.9	171.9	193.6
-30%	136.2	236.5	150.0	248.9	140.5	247.1	132.0	237.2

Correction (M_J) Factors

N41	N42	N43	N44
1.340	1.379	1.382	1.312

4.6 BOC12 Unit Load Steady State Test

In order to verify satisfactory steady state plant operation with Replacement Steam Generators (NSM MG-19815, TT/1/A/9815/00/02E, BOC12 Unit Load Steady State Test for NSM MG-19815 was performed at approximately 30%, 78%, 89%, 98% 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

- Turbine Impulse Pressure
- S/G Narrow Range Level
- S/G Main Steam Flow
- S/G Steam Pressure
- S/G Feedwater Flow

The test was performed at ~30% power on May 20, 1997, ~78% on May 25, 1997, ~89% on May 26, 1997, ~98% on May 26, 1996, and ~100% on August 13, 1997. All acceptance criteria were met. Tables 14 through 24 document the results.

Table 14
NC Loop Cold Leg Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
29.37	555.35	555.66	555.77	555.16
78.24	553.72	554.46	554.45	553.84
88.92	555.47	556.26	556.27	555.64
98.21	555.18	556.03	556.04	555.43
99.79	554.63	555.32	555.50	554.78

Table 15
NC Loop Hot Leg Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
29.37	573.53	572.99	573.26	572.80
78.24	600.22	600.12	600.71	599.42
88.92	607.63	607.67	608.29	606.78
98.21	612.39	612.54	613.19	611.46
99.79	613.11	612.78	613.48	611.83

Table 16
NC Loop Average Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
29.37	564.74	564.50	564.88	564.40
78.24	577.67	577.61	578.20	577.34
88.92	582.29	582.29	582.90	582.00
98.21	584.59	584.62	585.24	584.29
99.79	584.60	584.40	585.15	584.13

Table 17
NC Loop Delta Temperatures

Power Level	NC Loop A	NC Loop B	NC Loop C	NC Loop D
29.37	18.19	17.32	17.49	17 64
78.24	46.50	45.66	46.27	45.57
88.92	52.16	51.40	52.01	51.13
98.21	57.20	59.49	57.14	56.03
99.79	58.49	57.45	57.96	57.07

Table 18
Pressurizer Level Data

Power Level	PZR Level Channel 1	PZR Level Channel 2	PZR Level Channel 3
29.37	34.23	34.50	33.26
78.24	48.71	47.60	47.88
88.92	53.15	52.03	52.25
98.21	55.73	54.60	54.94
99.79	54.94	53.79	54.13

Table 19
Turbine Governor Valve Positions

Power Level	GV 1	GV 2	GV 3	GV 4
29.37	9.2	11.5	8.5	8.0
78.24	6.6	73.0	70.1	0
88.92	26.1	97.8	94.8	0
98.21	95.4	97.8	94.8	8.2
99.79	95.4	97.8	94.7	15.5

Table 20
Turbine Impulse Pressure
(PSIG)

Power Level	Ch 1	Ch 2
29.37	186.21	185.51
78.24	557.44	560.89
88.92	641.83	643.78
98.21	713.89	713.80
99.79	728.49	728.12

Table 21
S/G Narrow Range Level
(%)

Power Level	S/G A	S/G B	S/G C	S/G D
29.37	45.42	44.67	47.20	46.68
78.24	57.72	58.39	58.90	58.98
88.92	60.88	61.42	62.15	62.07
98.21	63.74	63.77	64.27	64.23
99.79	64.29	64.24	64.50	64.20

Table 22
S/G Main Steam Pressure
(PSIA)

Power Level	S/G A	S/G B	S/G C	S/G D
29.37	1061.8	1058.1	1056.9	1065.5
78.24	1019.1	1016.4	1015.4	1022.9
88.92	1027.6	1024.9	1024.1	1031.5
98.21	1018.6	1016.1	1015.4	1023.3
99.79	1007.0	1004.2	1004.0	1011.1

Table 23
S/G Main Steam Flow
(MLB/HR)

Power Level	S/G A	S/G B	S/G C	S/G D
29.37	1.001	0.877	0.995	0.979
78.24	2.922	2.831	3.022	2.975
88.92	3.378	3.274	3.491	3.434
98.21	3.784	3.670	3.903	3.851
9979	3.899	3.757	3.987	3.944

Table 24
S/G Feedwater Flow
(MLB/HR)

Power Level	S/G A	S/G B	S/G C	S/G D
29.37	0.995	1.014	0.967	1.029
78.24	2.860	2.912	2.907	2.910
88.92	3.290	3.352	3.348	3.346
98.21	3.674	3.744	3.734	3.730
99.79	3.738	3.800	3.794	3.781

4.7 UNIT LOAD 10% TRANSIENT TEST

TT/1/A/9815/00/03E, Unit Load 10% Transient Test for NSM MG-19815, was performed to verify proper operation of the modifications performed on various control systems per NSM MG-19815, Replacement Steam Generator Instrumentation and Control. The purpose of the test was to demonstrate proper plant response, including automatic control system performance, to a ~10% step load change (initiated via Turbine/Generator Control). The test verifies that the control systems work 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 satisfies the transient retest as required for the Post Modification Testing for Replacement Steam Generator Instrumentation and Control.

This test was performed from 38% Reactor Power. on May 23, 1997 and from 78% Reactor Power on May 26,1997. All acceptance criteria for the test were met as follows:

1) Reactor did not trip
2) Turbine did not trip
3) Safety Injection was not initiated
4) No Manual Operator Intervention was required to stabilize the Unit
5) Pressurizer PORV's did not lift
5) Pressurizer Code Safety Valves did not lift
6) Steam Generator PORV's did not lift
7) Steam Generator Code Safety Valves did not lift

8) Monitored plant parameters did not indicate sustained or diverging oscillations

Figures 3 through 21 illustrate response of plant parameters during the 78% Reactor Power test.

FIGURE 3
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - U1 REACTOR THERMAL POWER, BEST

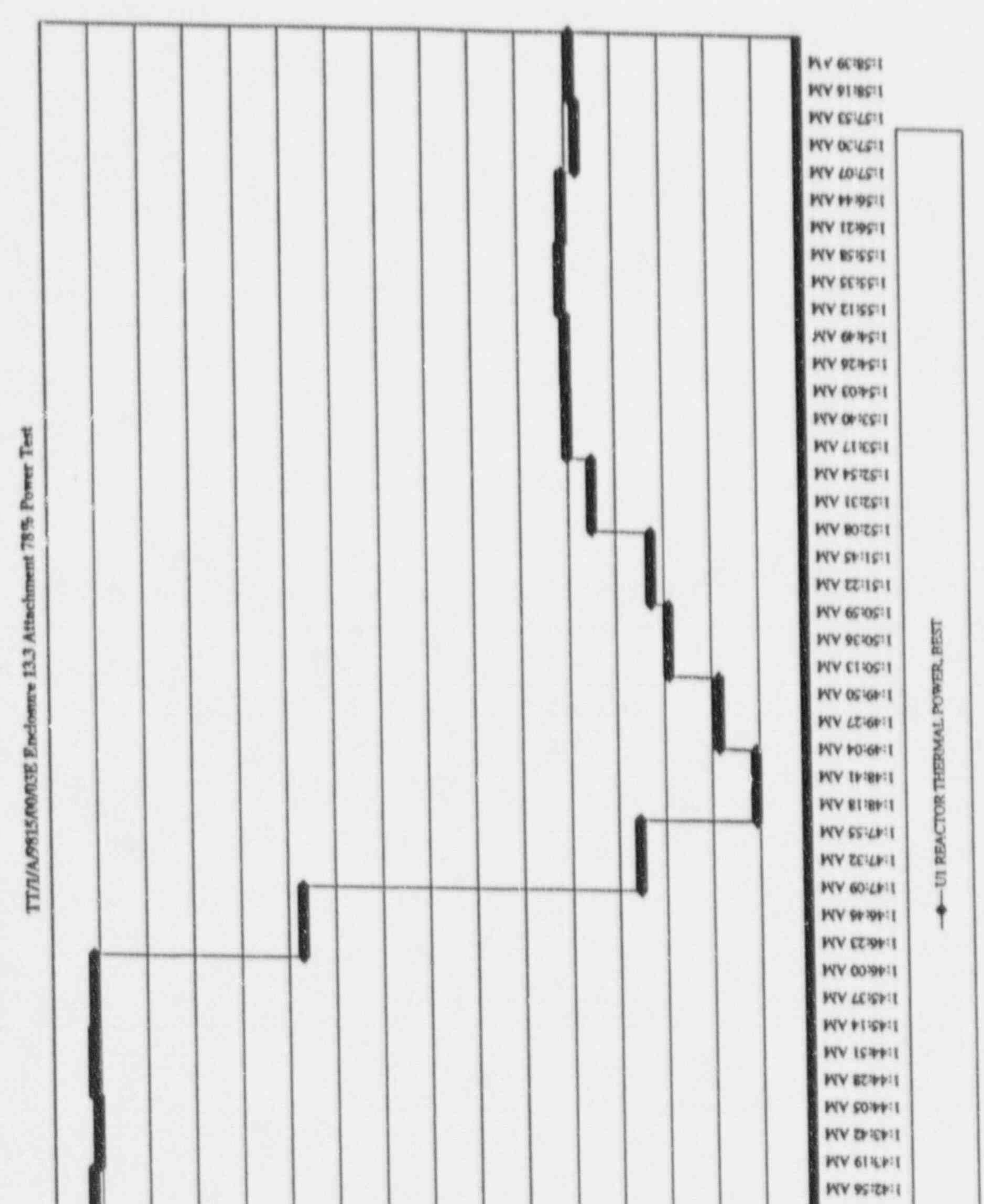


FIGURE 4
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - U1 POWER RANGE AVERAGE LEVEL

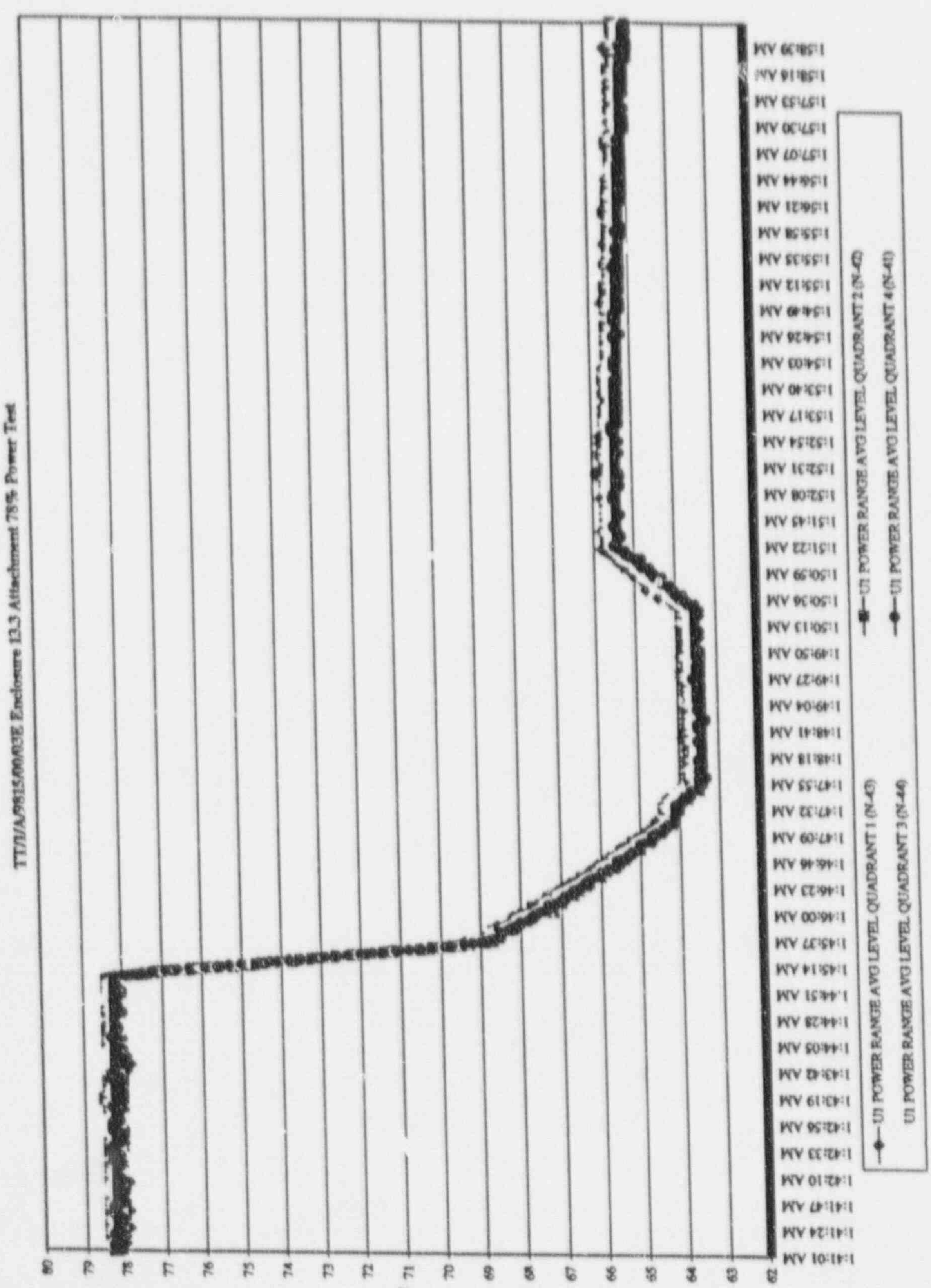


FIGURE 5
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - U1 GENERATOR MW

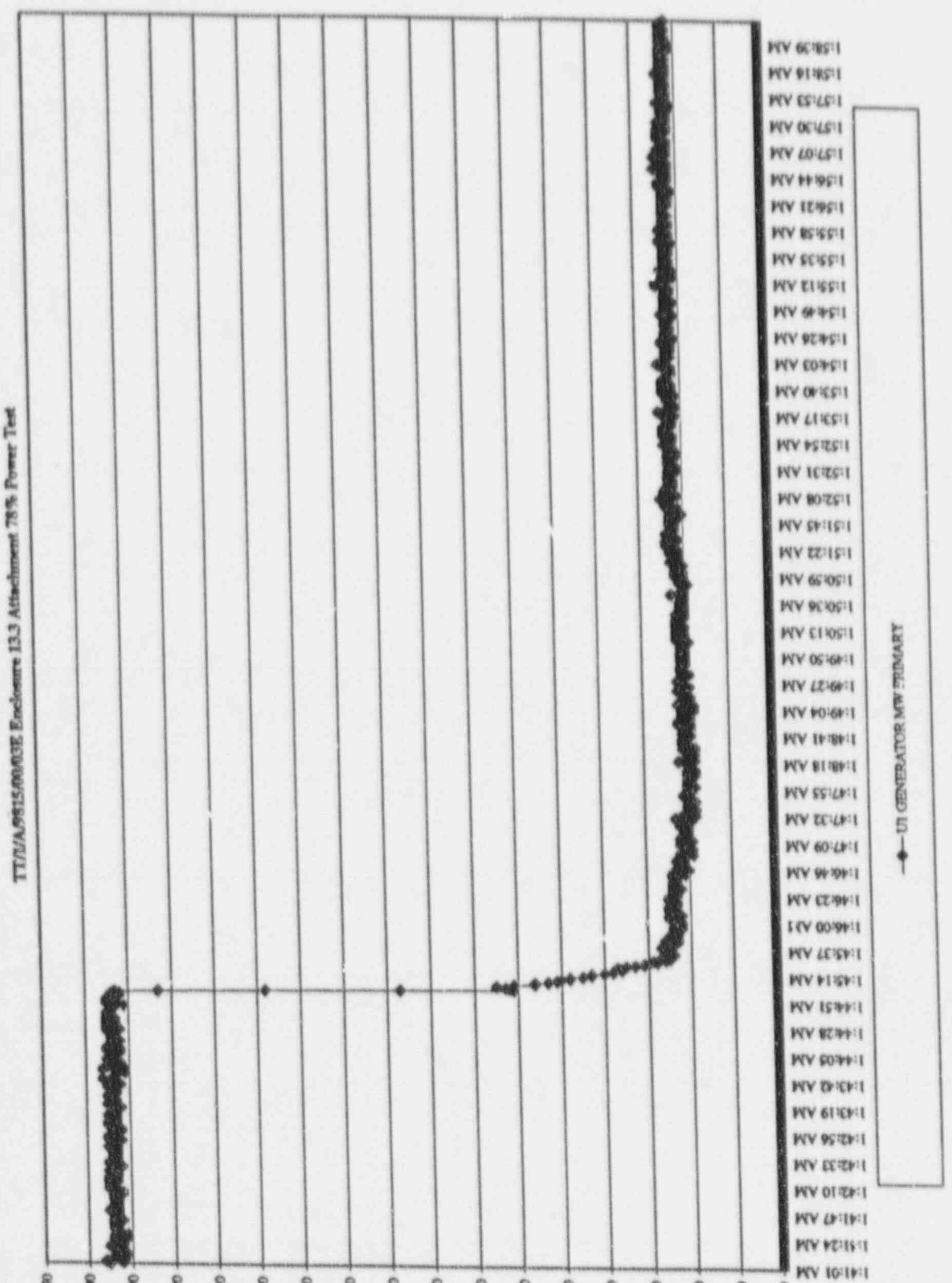


FIGURE 6
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - NC LOOP AVERAGE TEMP

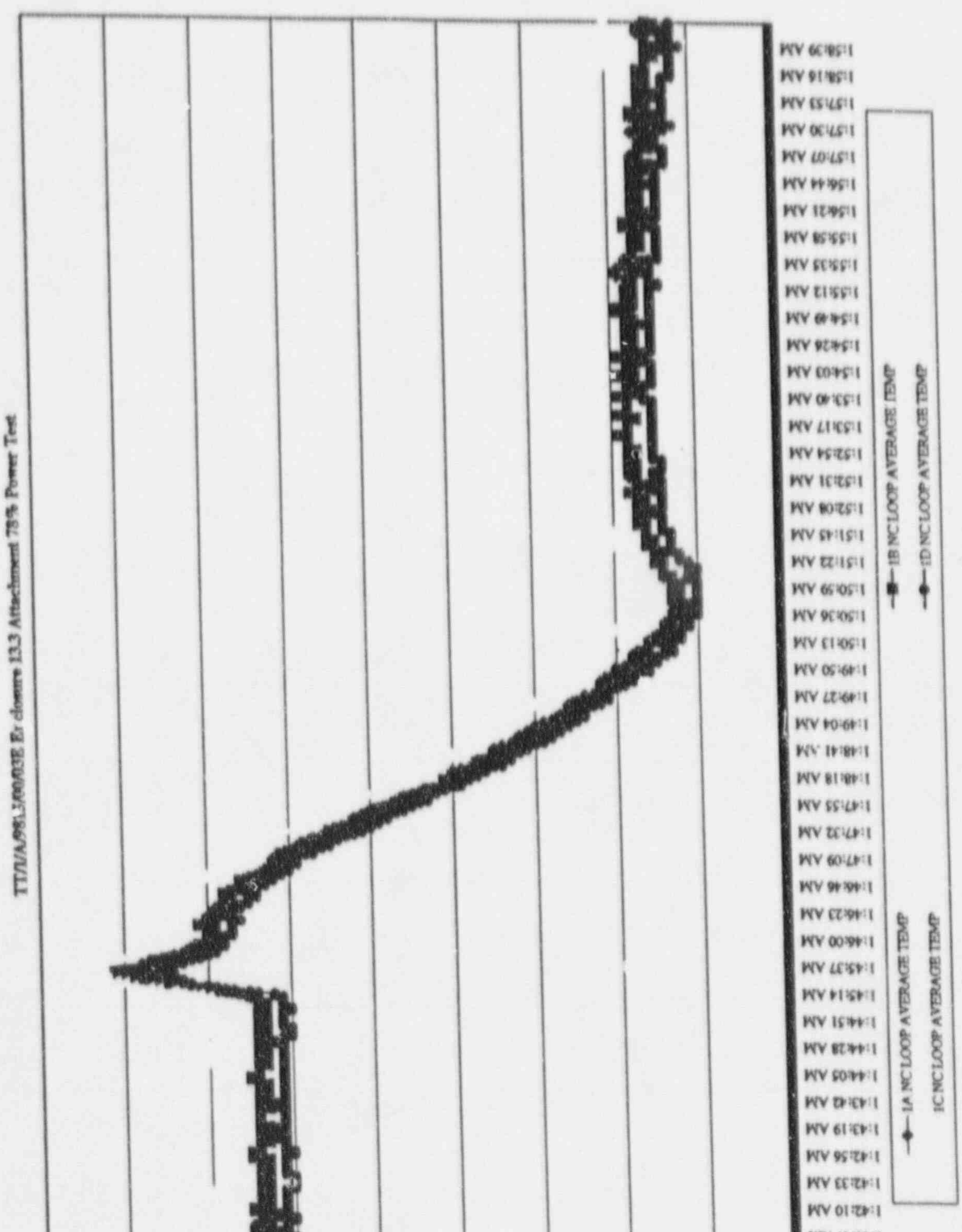


FIGURE 7
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - NC LOOP "ACTUAL" D/T

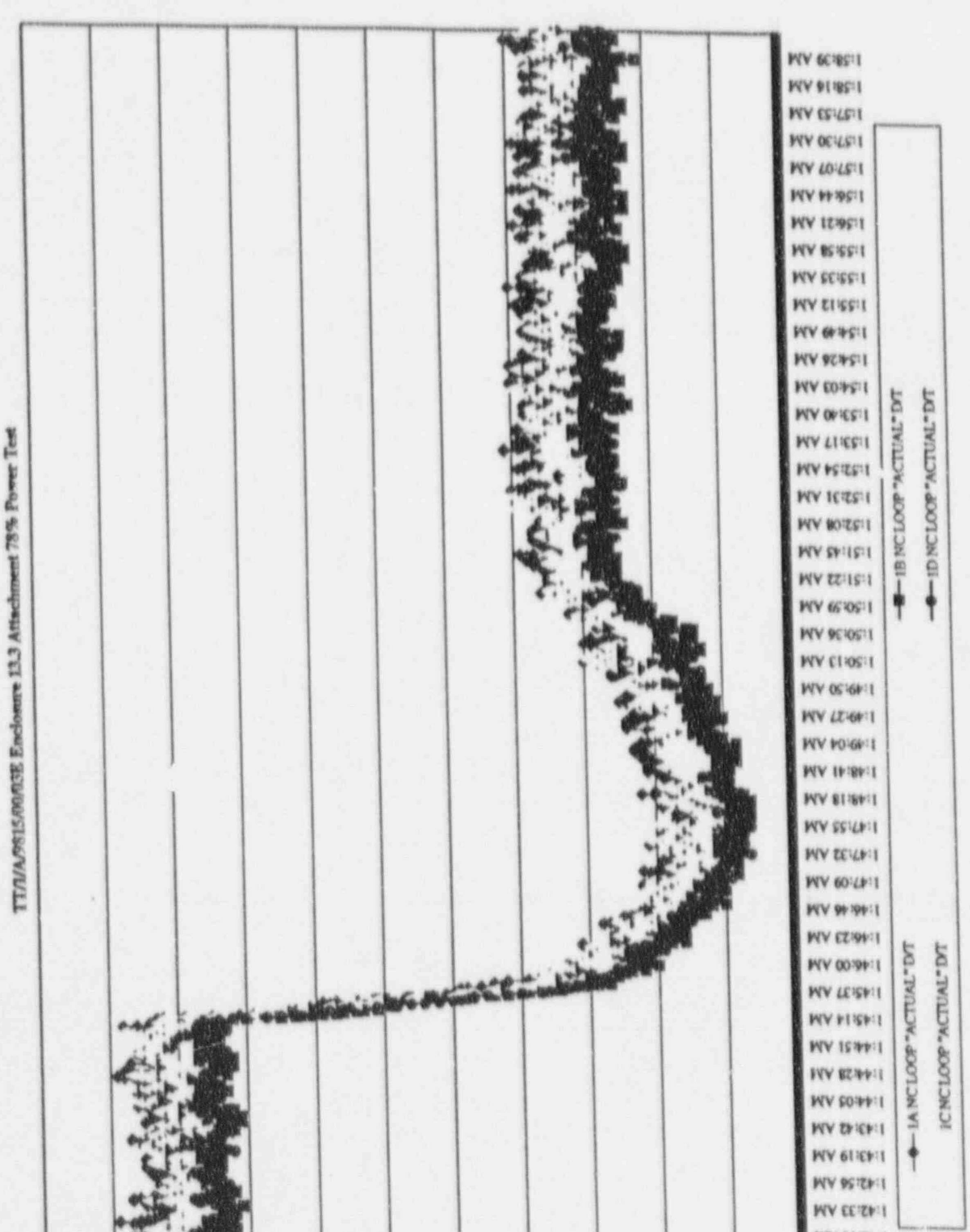


FIGURE 8
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - NC LOOP HIGHEST AVG TEMP AND T-REF

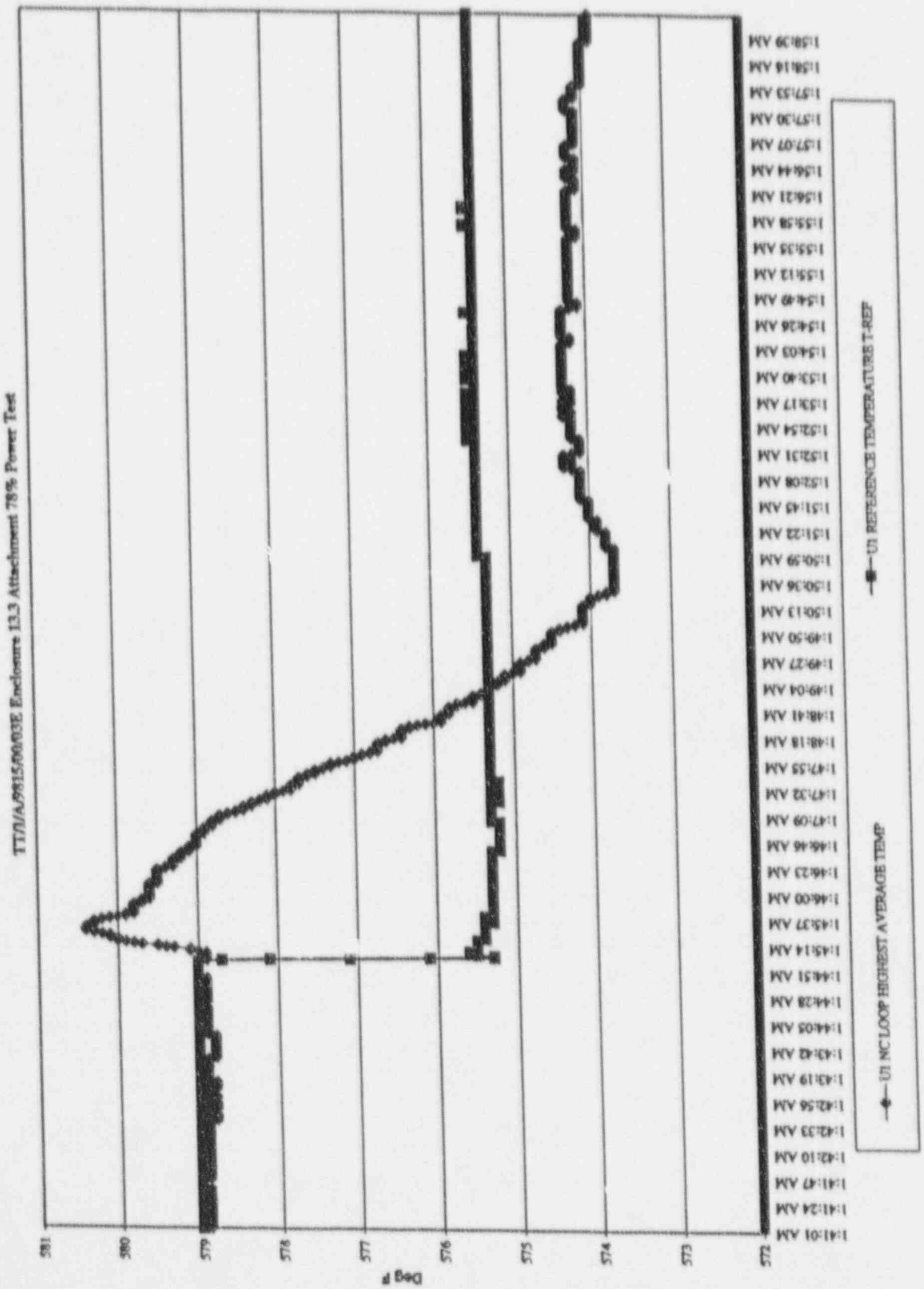


FIGURE 9
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - U1 PRESSURIZER PRESS

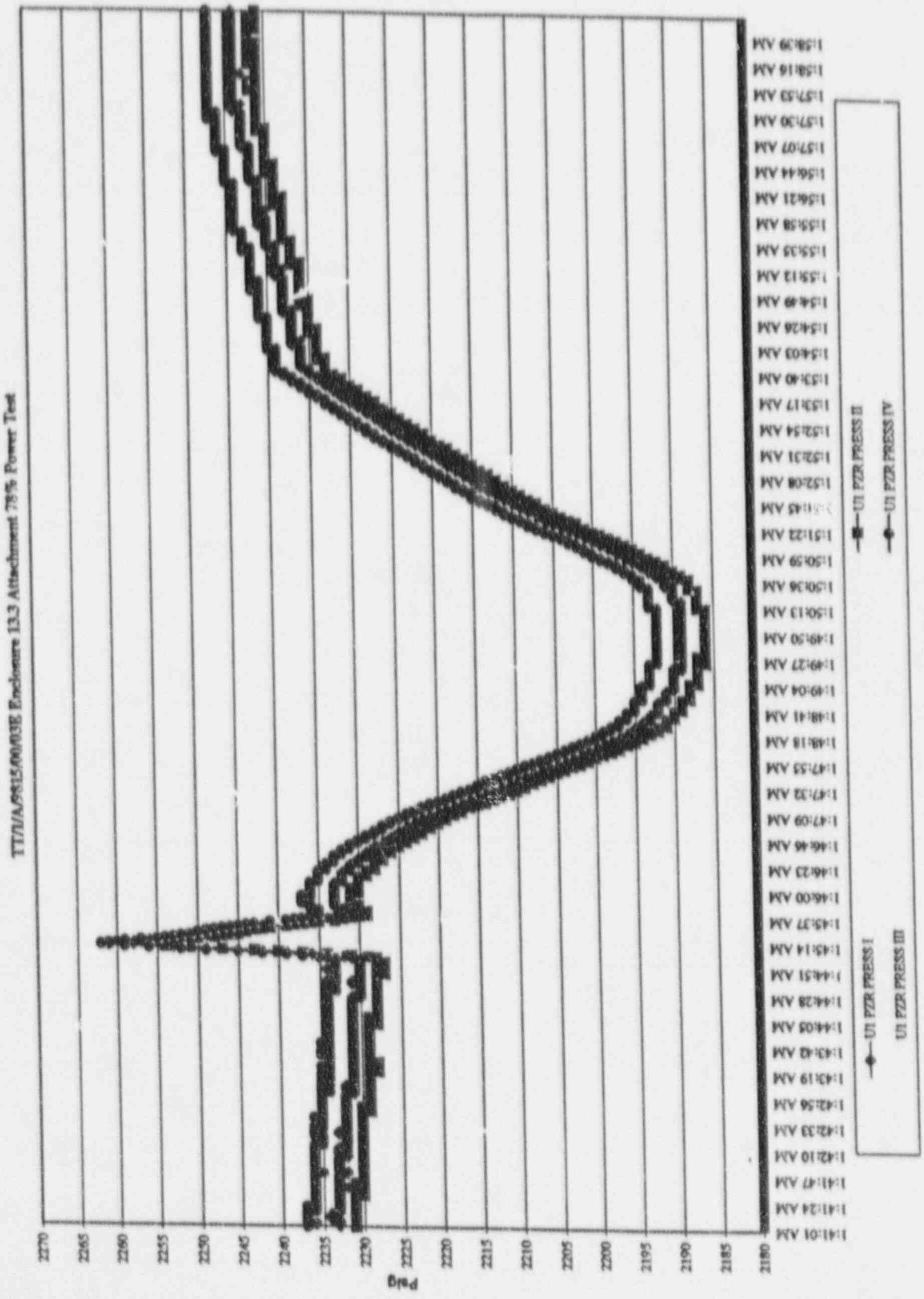


FIGURE 10
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - U1 PRESSURIZER LEVEL

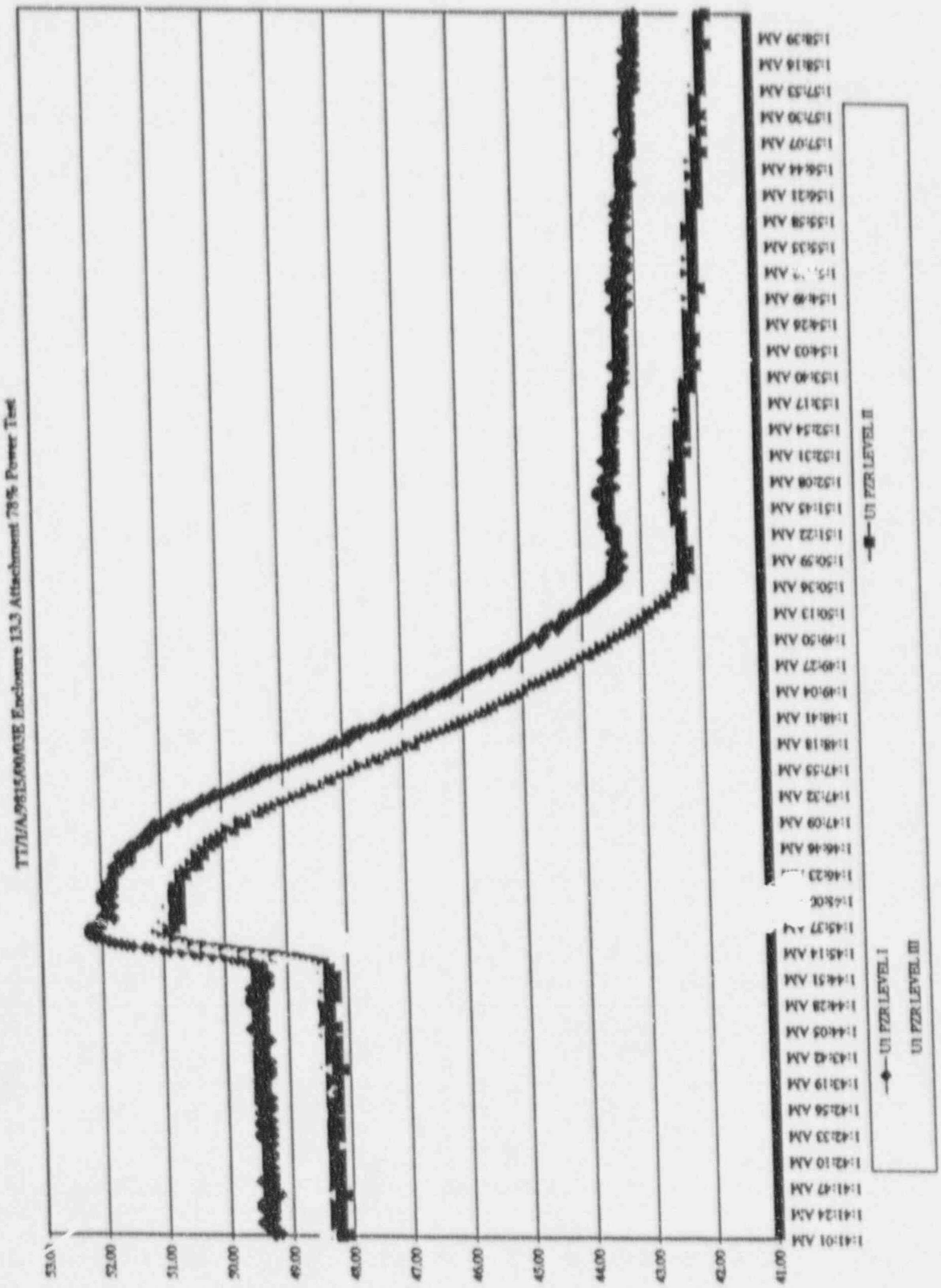


FIGURE 11
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1A S/G NARROW RANGE LEVEL

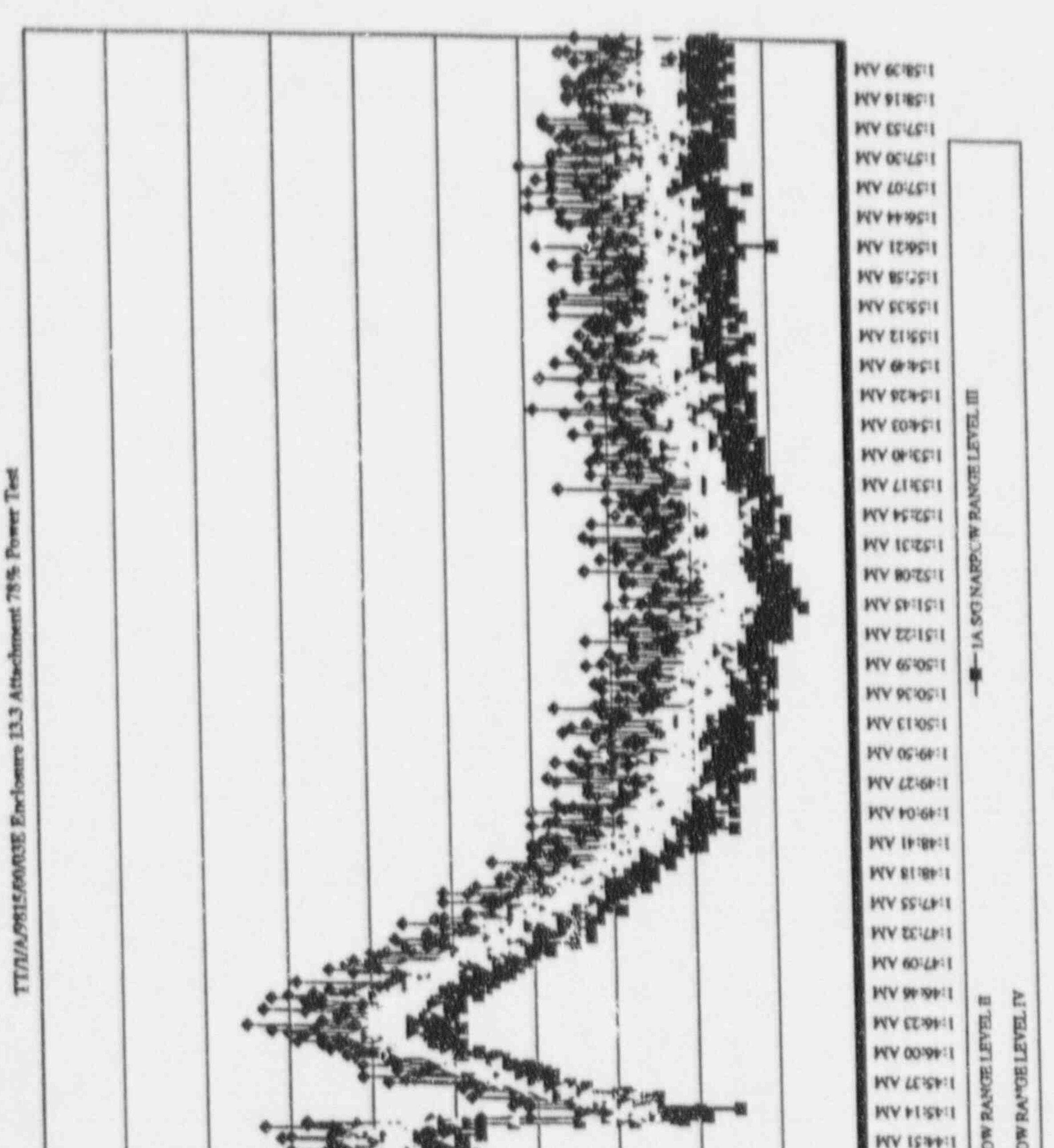


FIGURE 12
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1B S/G NARROW RANGE LEVEL

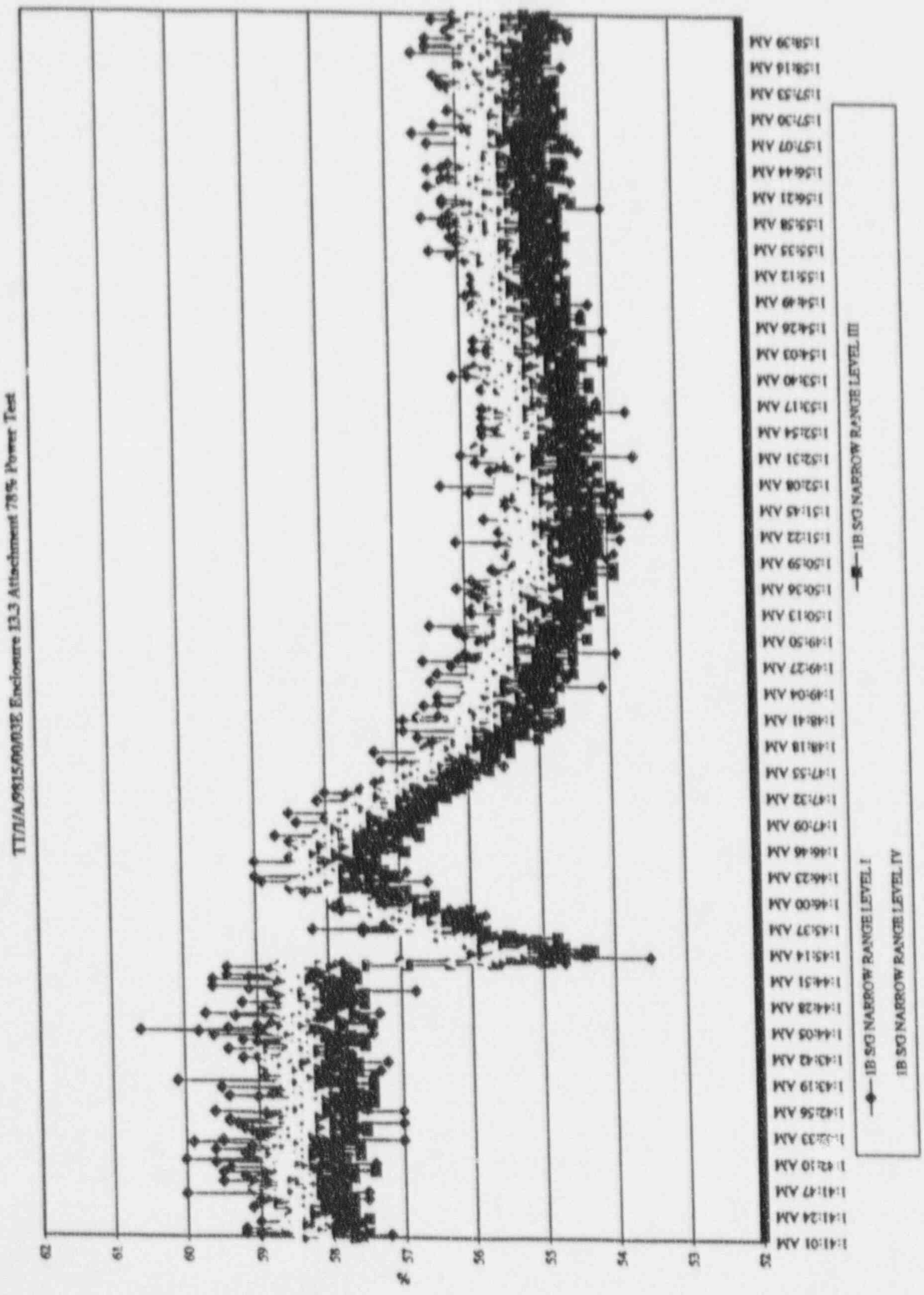


FIGURE 13

UNIT LOAD 10% TRANSIE*IT TEST, 78% PWR - 1C S/G NARROW RANGE LEVEL

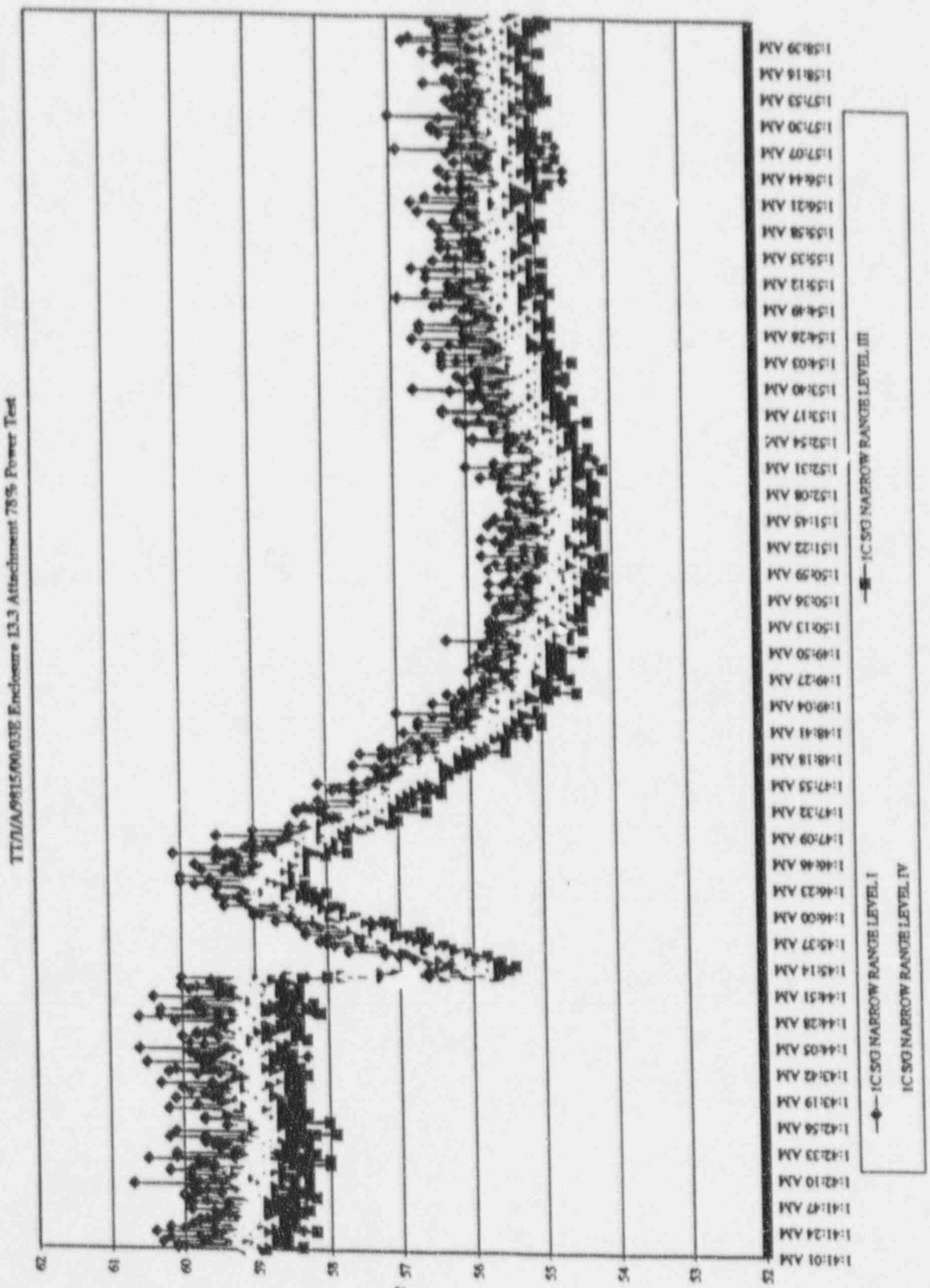


FIGURE 14
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1D S/G NARROW RANGE LEVEL

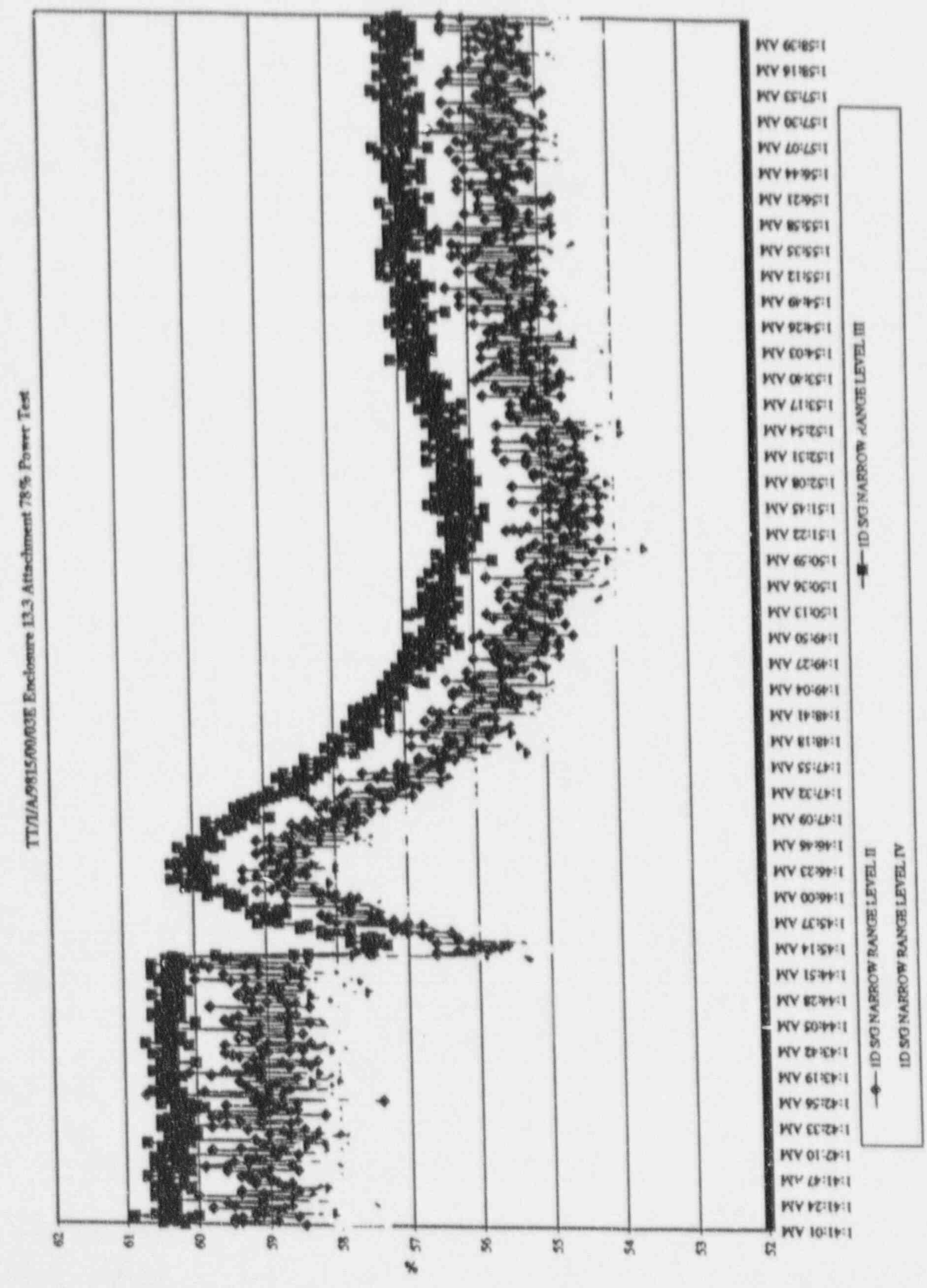


FIGURE 15
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - S/G AVERAGE STEAM PRESSURE

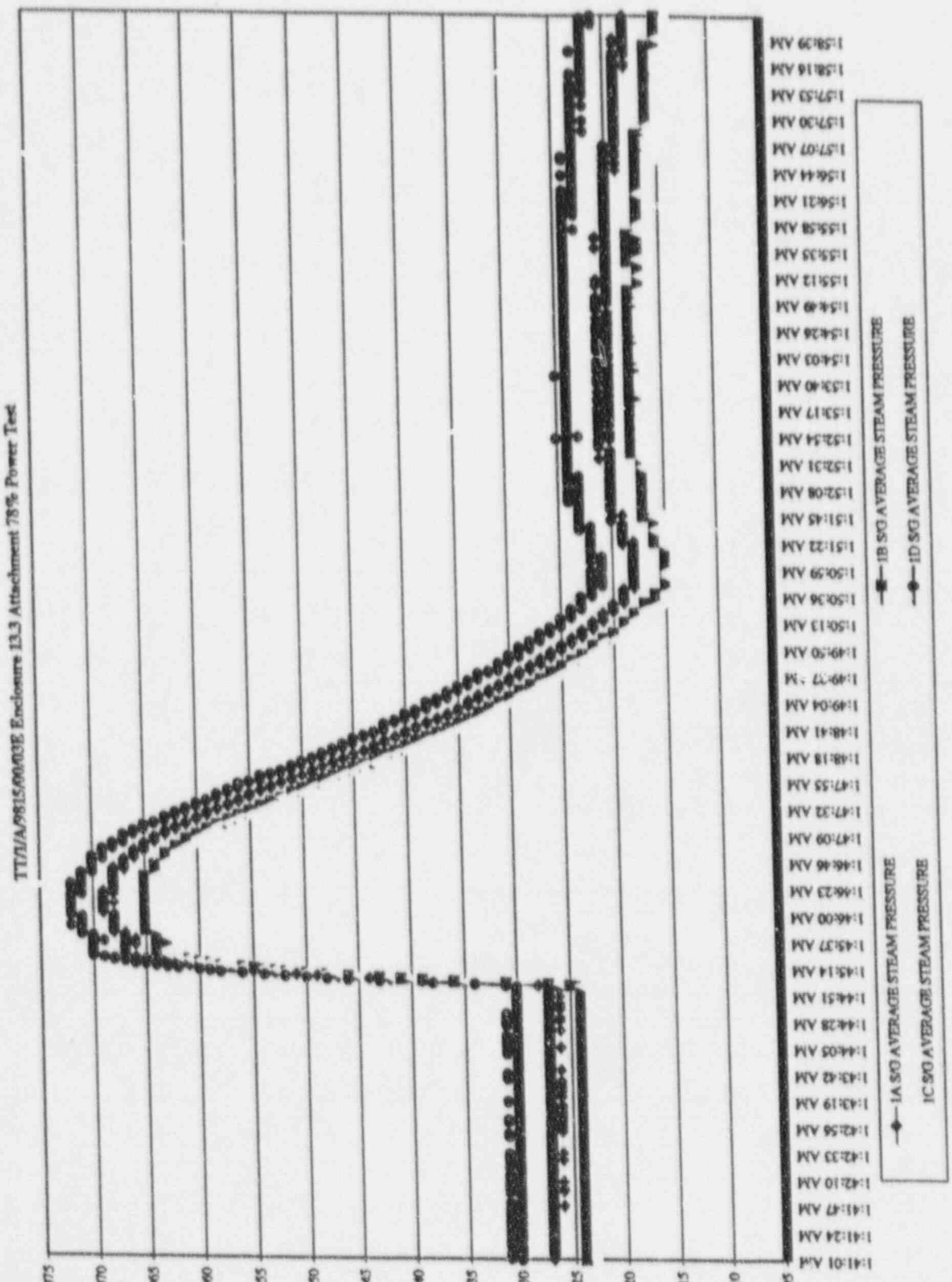


FIGURE 16
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - CF PUMP SPEED

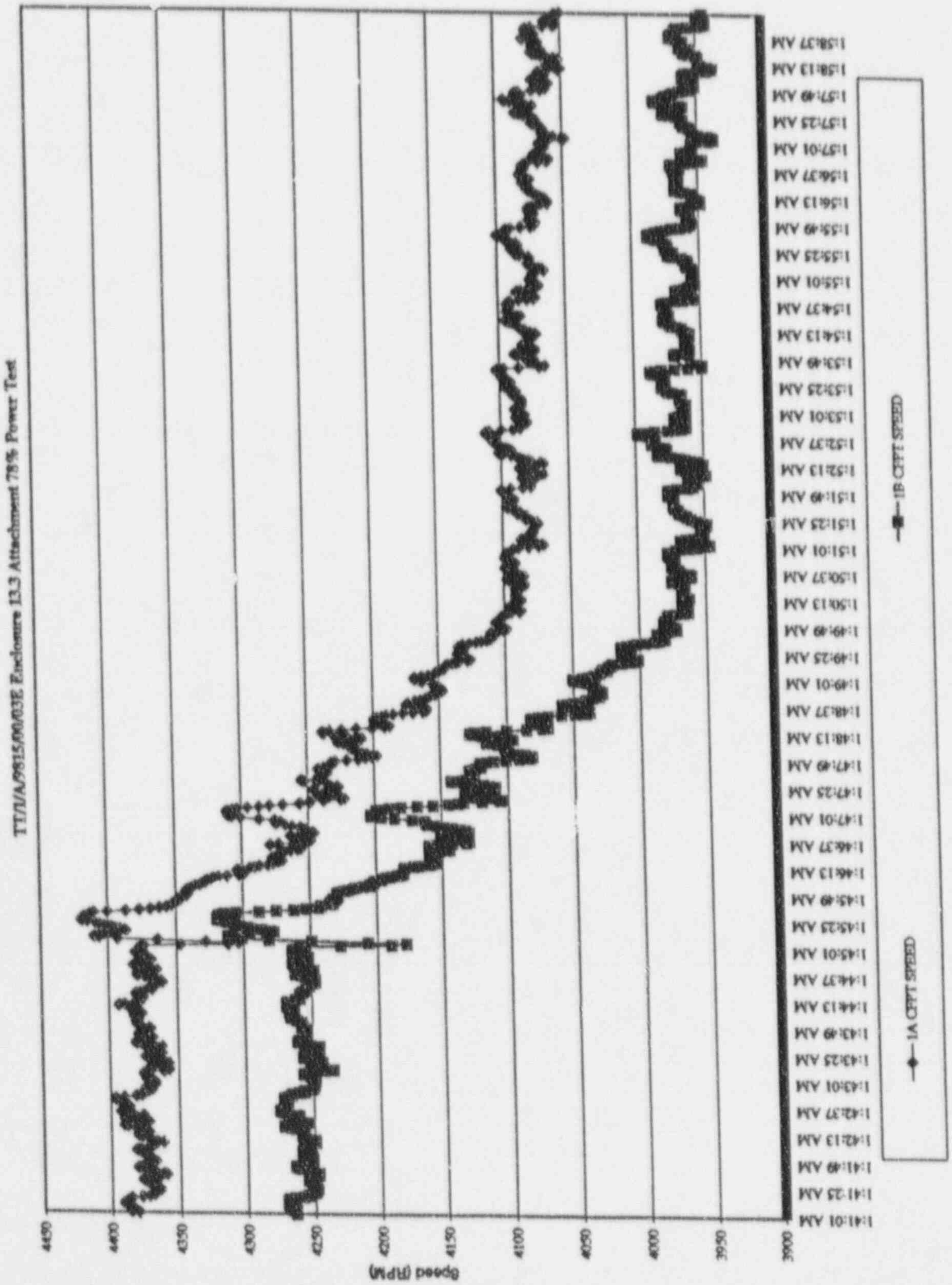


FIGURE 17
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - U1 TURBINE IMPULSE CHAMBER PRESSURE

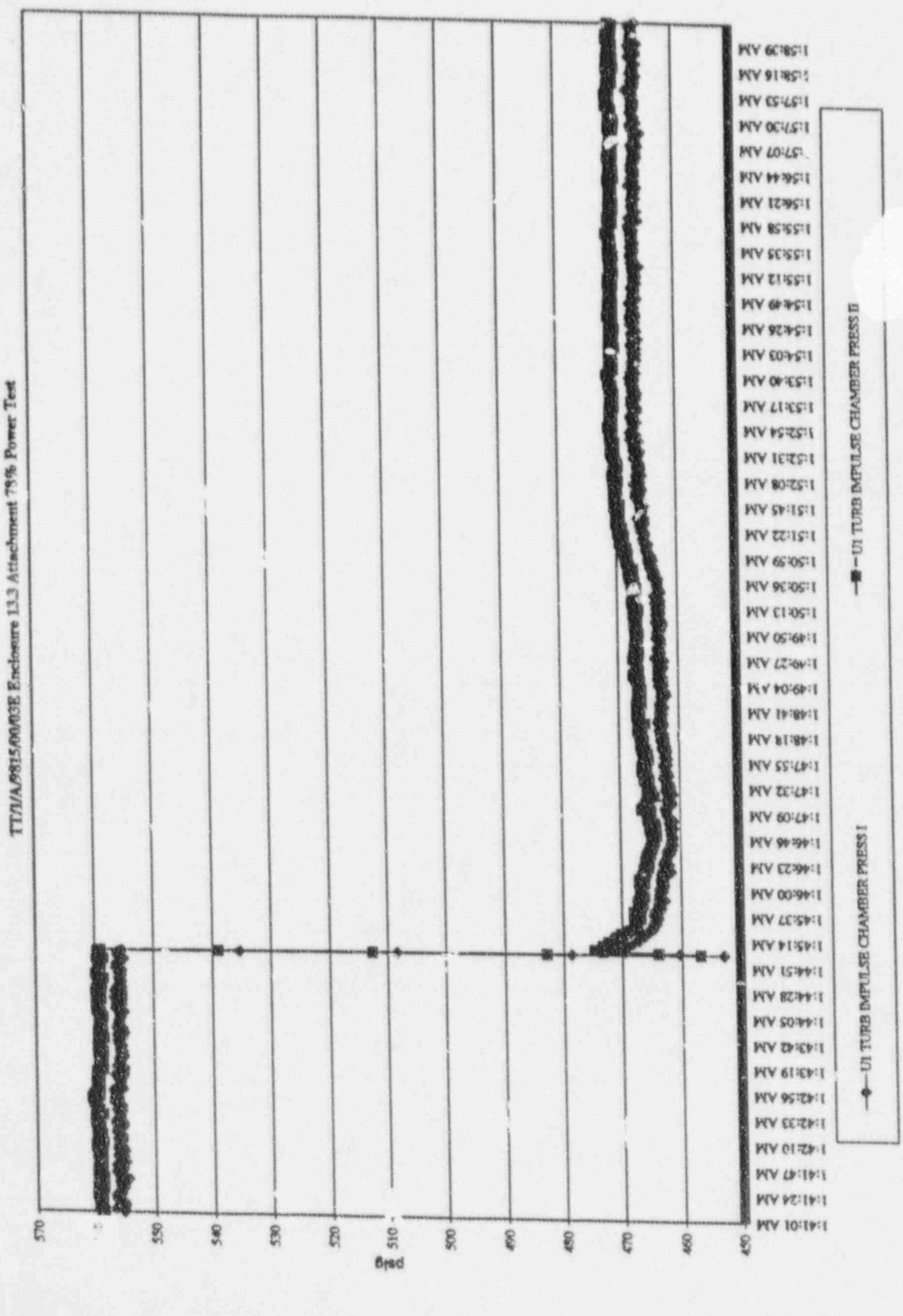


FIGURE 18
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1A S/G FEEDWATER FLOW

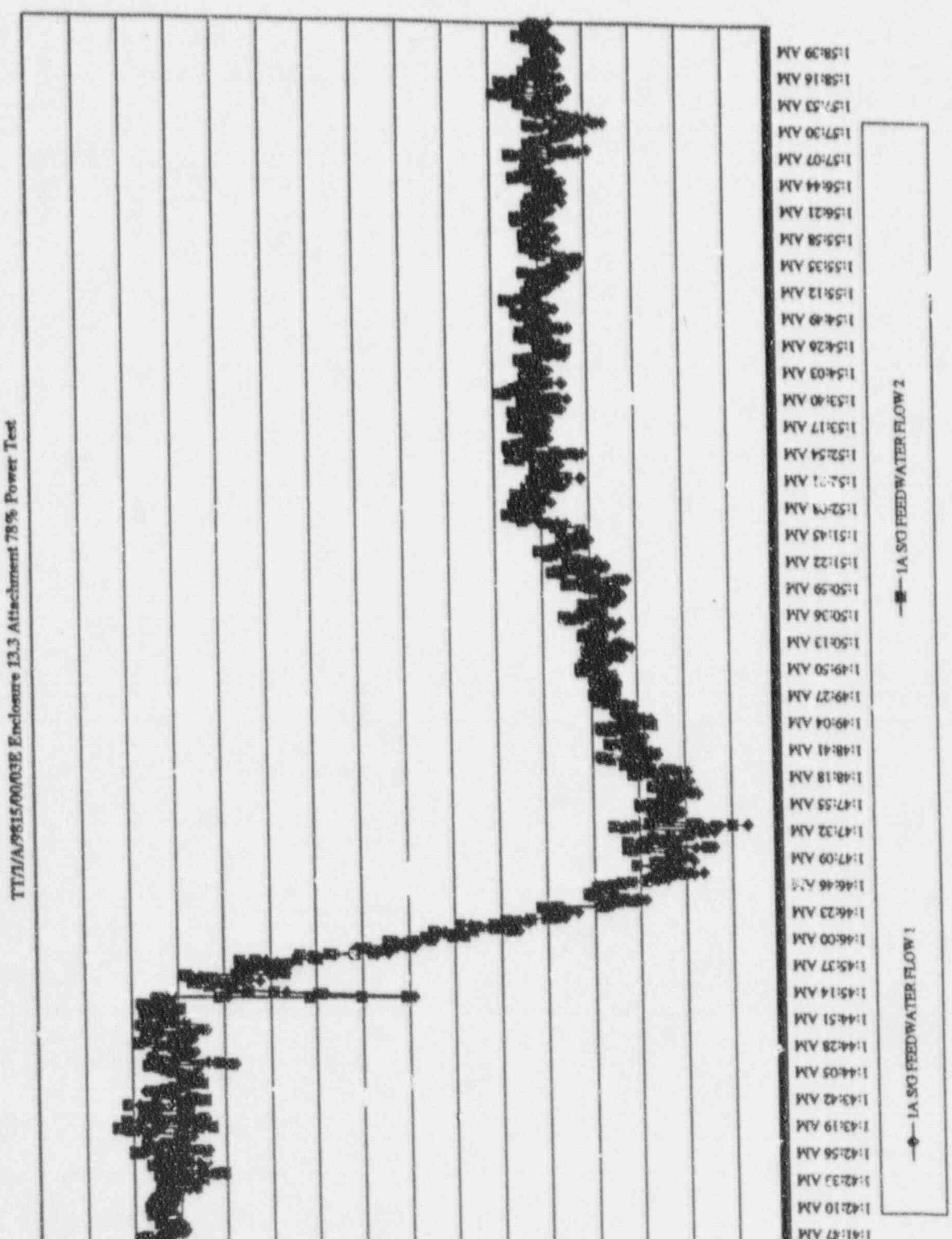


FIGURE 19
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1B S/G FEEDWATER FLOW

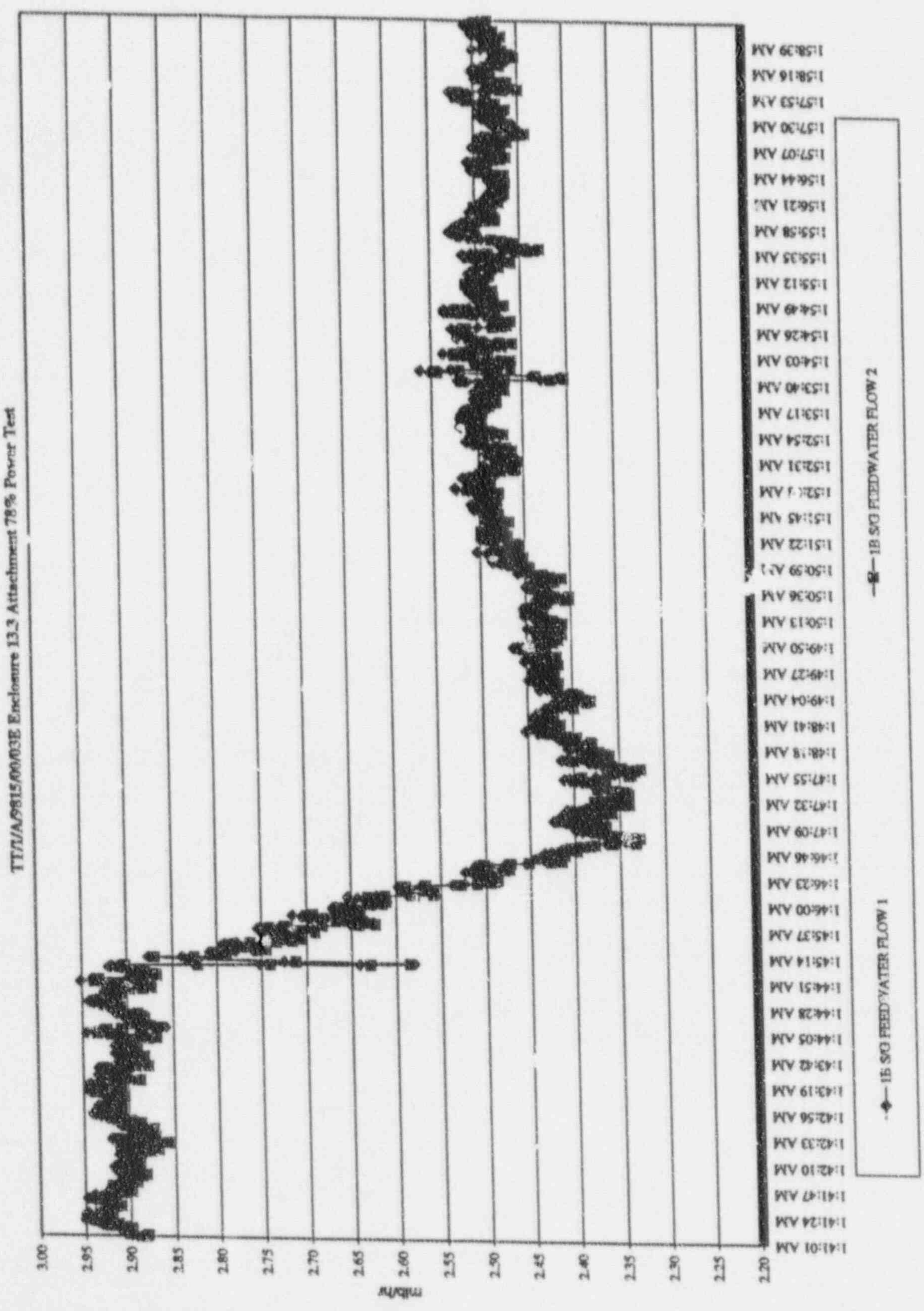


FIGURE 20
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1C S/G FEEDWATER FLOW

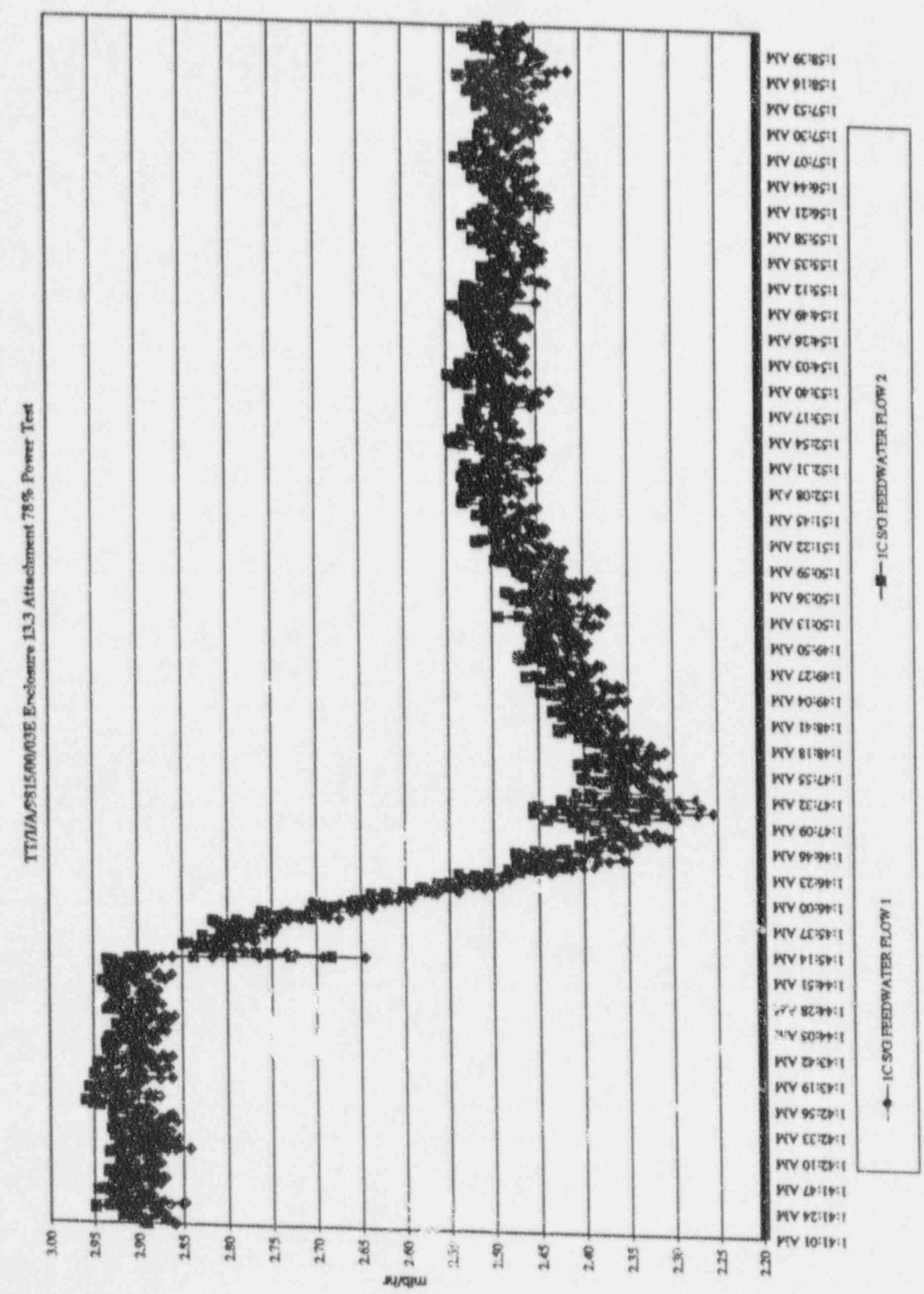
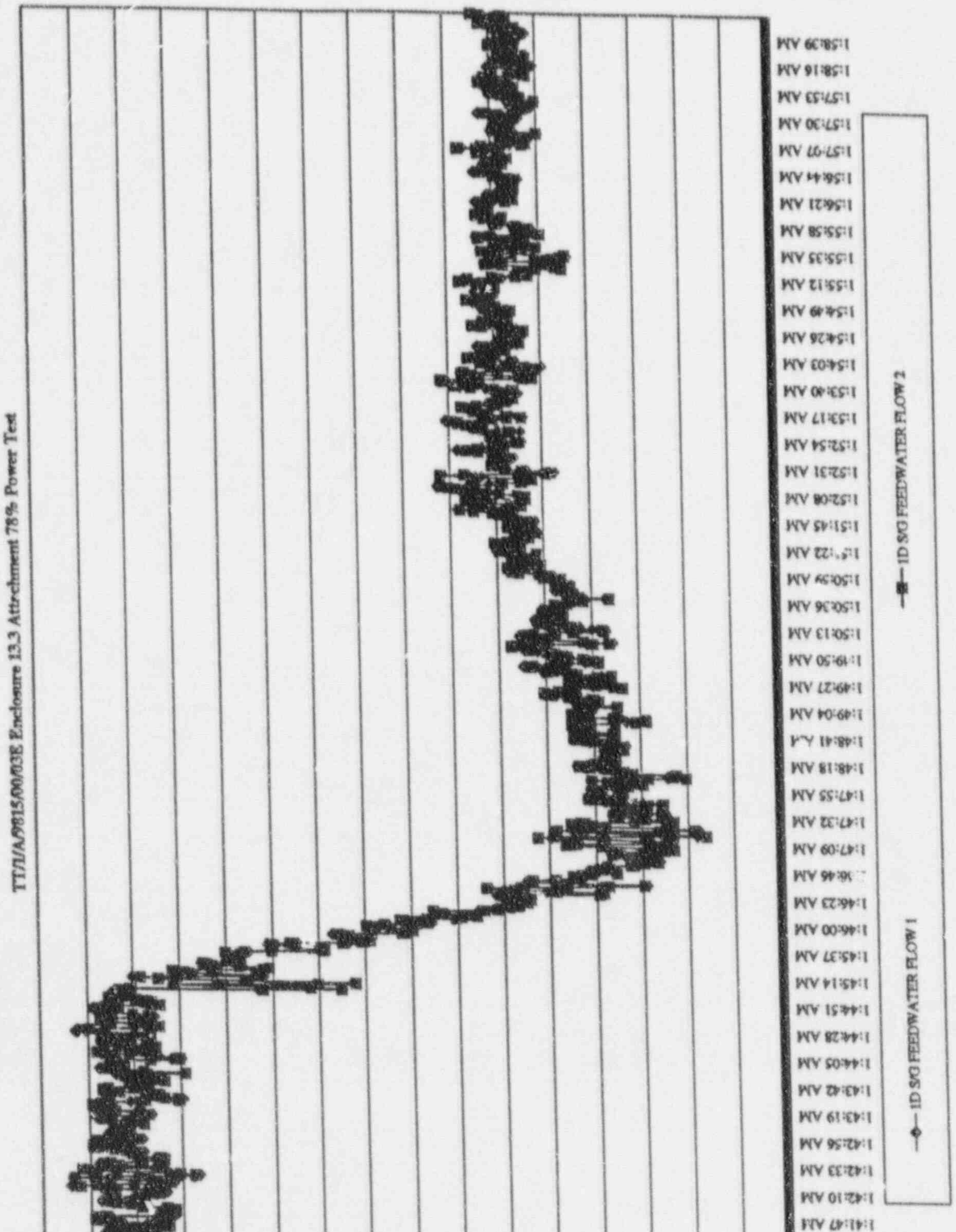


FIGURE 21
UNIT LOAD 10% TRANSIENT TEST, 78% PWR - 1D S/G FEEDWATER FLOW



4.8 Replacement S/G Tuning and Testing of Feedwater Control

TT/1/A/9815//00/04E, Functional Tuning and Testing of the Feedwater Control System, was performed to record the behavior of the S/G Level Controls and the course of action taken to optimally tune the system following replacement of the Westinghouse D3 Steam Generators. Testing was performed prior to Startup and at various power levels during initial power ascension to allow monitoring of S/G Level Controls and Feed Pump Speed Controls. Difficulties with level swings at very low power levels (~3% power) prevented optimization of level control due to inaccuracies in main steam flows and changes in the CF header to MS header differential pressure program for Feed Pump Speed Control Program. These effects were unexpected since CNS did not remove the Nozzle Swap startup evolution while MNS did. The following discussion summarizes the results of the testing.

Monitoring System Being Placed in Automatic Control

The Feedwater Control System Bypass Valves were placed in AUTO 0n May 20, 1997 with the A Main Feed Pump in MANUAL at approximately 130 pounds on the D/P Program. Reactor Power was approximately 3% and level swings of 4 to 8 % narrow range were observed. This instability improved with a reduction to approximately 76 pounds D/P on the Feed Pump Speed Program and a Bypass Valve Controller Gain reduction from 4.7 v/v to 3.0 v/v., however, operations was uncomfortable with testing at these level swings. Testing was canceled at this power level.

Monitoring the Placement of the Main Turbine On-Line

Placing of the Turbine Generator on line at approximately 8% power was uneventful and occurred on May 20, 1997. All S/G levels fluctuated slightly (decreased) and the Feed Pump Speed Program D/P dropped slightly. S/G levels returned to setpoint evenly with no oscillations occurring. The A Feed Pump returned smoothly to program.

Monitoring the Transition onto the Main Feed Reg Valves

At approximately 15% power, the Main Feed Reg Valves were opened in MANUAL with the associated Bypass Valve closing in AUTO. The Feed Pump Program D/P was 130 pounds. Again there were difficulties with stability. Reduction of Feed Pump Speed D/P Program to 76 pounds resulted in improved behavior and the CF System was placed in full automatic operation.

Tuning Tests at 30% Power

Steam Generators D through A (in that order) were subjected to five (5) percent level perturbations on May 21, 1997. These tests were conducted on one generator at a time. An increasing step change was applied to the narrow range level setpoint followed by a decreasing step change of the same magnitude to return the setpoint to its programmed position. The S/G levels responded adequately for the increasing step change , overshooting approximately 1 to 2%. The setpoint decrease produced a similar undershoot. No problems or tuning adjustments were required for this phase of testing.

A Feed Pump Speed Program D/P perturbation was simulated by introducing a five (5) percent Feed Pump Speed demand on the Feed Pump Speed Controller in MANUAL. The five percent speed demand resulted in a CF Header to SM Header D/P change of nearly 100 pounds. The Main Feedwater Reg Valves compensated (closed) and S/G levels changed less than 2 percent and settled within one cycle. The Feed Pump Controller was returned to program by placing the controller back to AUTO. A reverse of the above behavior resulted. No problems encountered or tuning adjustments were required for this phase of testing.

Monitoring the Placement of the B Feed Pump into the Feedwater Header

The second Feed Pump was introduced into service on May 25, 1997 at approximately 40% power. No problems were encountered or tuning adjustments required for this transition.

Tuning Tests at 78% Power

S/G Level control was well behaved through power ascension to 78% power, therefore, no further level perturbation testing was performed. The Feed Pump Speed Perturbation was performed with both Feed Pumps in MANUAL. CF Header to SM Header D/P was raised from approximately 160 pounds to 185 pounds. The A and D S/G Main Feed Reg Valves showed signs of oscillation between 25% and 45% open and the test was terminated.

Observations and Conclusions

1) The elimination of Nozzle Swap at MNS had a significant effect on the Feedwater Control System. The shift in CF/SM Header D/P resulted in excessive Pump sped with the Bypass or Main regulating Valves compensating by positioning at less than optimal throttling positions.
2) The Feedwater Bypass Valve Controllers had their loop gains reduced from 4.7v/v to 3.0v/v to make them less responsive to level swings.
3) The adjusted tuning constants for the Bypass Valve controls were adequate for the Replacement Steam Generators at higher power levels. At startup and lower power levels the level control was poor. Engineering is studying this phenomenon and has plans to thoroughly investigate the associated control circuitry behavior and make further tuning adjustments as warranted. The issue has been deemed an Operator Workaround and been added to the Major Equipment Problem Resolution Program. Work Orders for additional testing/tuning have been added to the Plant Trip List should the opportunity for further investigation arise.

4.9 Reactor Coolant Flow Testing

PT/1/A/4150/13, NC Flow Calculation, was performed on May 28, 1997 at approximately 97.3% power. The average reactor coolant system flow (as determined by four one hour test runs, measured by the elbow taps, is 401,932 gallons per minute. The NC flow prior to Steam Generator replacement was approximately 385,600 gpm, or an increase of 16,332 gpm, or approximately 4.25%.