Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Unit 1
Docket No. STN 50-498
Unit 1 Cycle 10 Startup Testing Summary Report

South Texas Project Technical Specification 6.9.1.1 requires a summary report of appropriate plant startup and power escalation testing results following a) the installation of fuel that has a different design, and b) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. During the recent Cycle 9 to Cycle 10 refueling outage, South Texas Project Unit 1 installed 80 feed fuel assemblies, each with reduced-enrichment annular axial blanket pellets in the top and bottom seven inches of the fuel stack. In addition, all four Model E Steam Generators were replaced with Model Delta 94 Steam Generators, and the full power Reactor Coolant System average temperature was raised from 589 °F to 592 °F.

Attachment A to this letter is a summary report of the startup physics test results obtained during startup and power ascension. Attachment B to this letter is a summary report of the specific tests performed for the Replacement Steam Generators. No corrective actions were required to obtain satisfactory operation.

There are no new licensing commitments contained in this letter. If there are any questions, please contact Mr. D. E. Gore at (361) 972-8909 or me at (361) 972-7795.

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Attachment A) South Texas Unit 1 Cycle 10 Startup Physics Testing Summary Report
Attachment B) South Texas Unit 1 Cycle 10 Steam Generator Replacement Return to Service Testing Summary Report

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ATTACHMENT A
SOUTH TEXAS PROJECT
UNIT 1 CYCLE 10
STARTUP PHYSICS TESTING SUMMARY REPORT

I. Hot Rod Drop Time (seconds):

Acceptance Criteria (AC): ≤ 2.8 seconds

Measured (M)	Pass/Fail AC
1.7*	P

* Maximum value for 57 control rods

II. Rod Worth Measurements (Dynamic Rod Worth Measurement Method Used):

Design Review Criteria (DRC): Each bank within 15% or 100 pcm of the predicted value (whichever is greater)
Total rod worth within 8% of predicted

Acceptance Criteria (AC): Total rod worth ≥ 90% of Predicted

RCCA Bank	Measured Worth (pcm)	Predicted Worth (pcm)	Delta (M-P) (pcm)	Percent Difference (%)	Pass/Fail DRC	Pass/Fail AC
Shutdown A	329.9	320.0	9.9	3.1	P	-
Shutdown B	911.5	894.3	17.2	1.9	P	-
Shutdown C	384.9	380.1	4.8	1.3	P	-
Shutdown D	404.2	403.2	1.0	0.3	P	-
Shutdown E	503.2	496.5	6.7	1.4	P	-
Control A	808.4	788.5	19.9	2.5	P	-
Control B	666.2	650.9	15.3	2.4	P	-
Control C	862.1	835.5	26.6	3.2	P	-
Control D	596.1	557.9	38.2	6.9	P	-
Total	5466.5	5326.9	139.6	2.6	P	P

ARO: All Rods Out
% Difference = 100 × (M - P) / P

III. Hot Zero Power (HZP) Critical Boron Concentration (ppm):

Design Review Criteria (DRC): ±50 ppm

Acceptance Criteria (AC): ±1000 pcm (143.1 ppm)

Measured (M)	Predicted (P)	(M-P)	Pass/Fail DRC	Pass/Fail AC
1921	1943	-22	P	P

IV. HZP, ARO Isothermal Temperature Coefficient (ITC) (pcm/°F):

Design Review Criteria (DRC): ±2 pcm/°F

Acceptance Criteria (AC): none

Measured (M)	Predicted (P)	(M-P)	Pass/Fail DRC	Pass/Fail AC
-0.97	-2.27	1.3	P	-

V. Inferred HZP, ARO Moderator Temperature Coefficient (pcm/°F)*:

Design Review Criteria (DRC): none

Acceptance Criteria (AC): < +5 pcm/°F, or rod withdrawal limits established

Measured	Predicted	Adjusted	Pass/Fail DRC	Pass/Fail AC
0.83	-0.47	1.60**	-	P

* Inferred MTC is obtained by subtracting the design Doppler Temperature Coefficient (-1.8 pcm/°F) from the measured Isothermal Temperature Coefficient.

** Adjusted MTC includes measurement uncertainty and Integral Fuel Burnable Absorber burnout correction.

VI. POWER DISTRIBUTION MEASUREMENTS:

Design Review Criteria (DRC): Incore Quadrant Power Tilt ≤ 1.02
Assembly Power Error (M-P) ≤ ±0.1

Acceptance Criteria (AC): FDHN < Technical Specification (TS) 3.2.3 Limit
F_{xy} ≤ TS 3.2.2 Limit

Reactor Power	Incore Quadrant Power Tilts		Limiting FDHN	FDHN Limit	Limiting F_{xy}	F_{xy} Limit	Largest Assembly Power Error
Low Power	0.992	1.001	1.4843	1.8573	1.6552	2.1104	0.076
(28.7%)	1.006	1.001					
Intermediate Power	0.999	1.009	1.4370	1.6333	1.6159	1.9322	0.085
(77.5%)	0.994	0.997					
Full Power	1.000	1.011	1.4481	1.5300	1.6174	1.8740	0.091
(100.0%)	0.991	0.998					

FDHN: Nuclear Enthalpy Rise Hot Channel Factor
Incore Tilt: Measured Incore Tilt in Excess of Designed Core Asymmetry

VII. Reactor Coolant System Flow Measurement (gpm):

Design Review Criteria (DRC): none

Acceptance Criteria (AC): ≥ 403,000 gpm

Reactor Power	Measured Flow	Pass/Fail DRC	Pass/Fail AC
100.0%	418,074	-	P

VIII. Full Power Critical Boron (ppm):

Design Review Criteria (DRC): ±50 ppm

Acceptance Criteria (AC): ±1000 pcm (148.6 ppm)

Burnup (EFPD)	Measured (M)	Predicted (P)	(M-P)	Pass/Fail DRC	Pass/Fail AC
5.5	1302.0	1292.9	9.1	P	P

ATTACHMENT B
SOUTH TEXAS PROJECT
UNIT 1 CYCLE 10
STEAM GENERATOR REPLACEMENT RETURN TO SERVICE TESTING SUMMARY REPORT

I. Thermal Expansion Test

The Objective of this test was to verify by visual observation, measurement, and evaluation that specified Replacement Steam Generator components and connected piping are free to expand without restriction of movement.

The Acceptance Criteria were that the equipment, piping and components addressed in the procedure are verified to expand during heat-up without obstructions or restrictions. All piping and components shall not cause interferences with surrounding equipment, supports, restraints, or structures. Thermal movements for each support, restraint, and/or component shall be within the anticipated ranges or evaluated as acceptable.

Observations, measurements, and evaluation of specified Replacement Steam Generator components and connected piping were made at ambient conditions prior to heatup of the Reactor Coolant System, at a Reactor Coolant System temperature of approximately 180 °F on May 9, 2000 and at a Reactor Coolant System temperature of approximately 567 °F on May 11, 2000. All Acceptance Criteria were met.

II. Vibration Monitoring Test

The Objective of this test was to demonstrate that vibration of specified Replacement Steam Generator components and connected piping are within acceptable limits at operating conditions.

The Acceptance Criteria were that equipment, piping and components addressed in the procedure have vibration levels within limits specified in applicable codes.

Observation and evaluation of vibration of Steam Generator Blowdown System piping was performed on May 11, 2000 while operating each Steam Generator Blowdown subsystem at its normal flowrate. All Acceptance Criteria were met.

Measurement and evaluation of vibration of each Steam Generator's Feedwater piping was performed on May 22, 2000. All Acceptance Criteria were met.

III. Steam Generator Blowdown Recirculation Test (0TEP04-SG-0007)

The Objective of this test was to demonstrate that the Steam Generator Blowdown Recirculation system operates as designed following the changes in piping made due to Steam Generator Replacement.

The Acceptance Criteria was that the Steam Generator Blowdown Recirculation system operated as designed.

Data was collected during operation of each Steam Generator's Blowdown Recirculation system between April 30, 2000 and May 8, 2000 and evaluated to verify that the system can be operated as designed. All Acceptance Criteria were met.

IV. Reactor Coolant System Flow Verification (0TEP04-SG-0001)

The Objective of this test was to measure the Reactor Coolant System flow rate prior to criticality using data obtained from installed elbow tap differential pressure (ΔP) instrumentation.

Acceptance Criteria was that Reactor Coolant System flow rate is greater than the minimum required.

Reactor Coolant System flow rate was determined to be 452,854 gallons per minute on May 10, 2000. This was greater than the Thermal Design flow rate of 392,000 gallons per minute in FSAR Table 5.1-1. In addition, this flow rate was greater than the Reactor Coolant System flow determined using the same method during Cycle 1, which was expected. All Acceptance Criteria were met.

V. Low Power Steam Generator Water Level Control Test (0TEP04-SG-0003)

The Objective of this test was to demonstrate the ability of the low power steam generator level control system to control at steady state power and to demonstrate the ability of the low power steam generator level control system to respond to a mismatch between steam generator level and setpoint.

The Acceptance Criteria was that the actual steam generator levels remain within specified limits of the programmed values, and that steam generator levels automatically returned to and remained within design limits of the level setpoint following a level setpoint change.

This test was performed on May 14, 2000 at a reactor power level of approximately 12%. Data was collected and evaluated during steady state operation. For each steam generator , a –5% level setpoint change was initiated and response of the level control system was monitored. This was followed by a +5% level setpoint change and response of the level control system was monitored. Figure 1 shows a typical response of Steam Generator level and Low Power Feedwater Regulating Valve position demand. All Acceptance Criteria were met.

VI. Calibration of Steam Flow Transmitters (0TEP04-SG-0001)

The Objective of this test was to verify the calibration of steam flow transmitters.

The Acceptance Criteria was that the difference between transmitter steam flow and actual steam flow is within the specified limits.

Data was collected and used to verify proper scaling of steam generator steam flow instrumentation at a reactor power level of approximately 47% on May 15, 2000, at a reactor power level of approximately 77% on May 18, 2000 and at 100% power on May 20, 2000. No transmitter calibrations were required at 50% and 75% power. At 100% power, five transmitters were calibrated to more closely normalize steam flow with feed flow. All calibrations were completed on May 23, 2000.

VII. Steam Generator Water Level Control Test (0TEP04-SG-0004)

The Objective of this test was to demonstrate proper operation of the turbine–driven feedwater pumps and the pumps speed controllers at steady state power, to demonstrate the ability of the steam generator level control system to control at steady state power and to demonstrate the ability of the steam generator level control system to respond to a mismatch between steam generator level and setpoint.

The Acceptance Criteria were that actual steam generator levels and feedwater to steam header delta pressure are within specified limits of the programmed values, main feedwater regulating valve positions are between the maximum and minimum valve position curves specified for the test, and steam generator level automatically returns to and remains within design limits of the level setpoint following a level setpoint change.

This test was initially performed on May 16, 2000 at a reactor power level of approximately 47%. Data was collected and evaluated during steady state operation. For each steam generator, a –5% level setpoint change was initiated and response of the level control system was monitored. This was followed by a +5% level setpoint change and response of the level control system was monitored. Figure 2 shows a typical response of Steam Generator level and Main Feedwater Regulating Valve position demand. Figure 3 shows a typical response of Steam Generator Feedwater and Steam Flow. All Acceptance Criteria were met.

While preparing to test the Main Feedwater Regulating Valve for Steam Generator A, a circuit board in the level control circuit failed. A new card was calibrated and installed and the test for Steam Generator A was completed satisfactorily.

The steady state operation portion of this test was performed again on May 18, 2000 at a reactor power level of approximately 77%. Data was collected and evaluated during steady state operation. All Acceptance Criteria were met.

The steady state operation portion of this test was performed again on May 20, 2000 at a reactor power level of 100%. Data was collected and evaluated during steady state operation. All Acceptance Criteria were met.

VIII. Load Swing Test (0TEP04-SG-0005)

The Objective of this test was to demonstrate the ability of the plant to sustain an approximate 10% power load reduction.

The Acceptance Criteria was that response of plant systems to the step load change is as follows:

No reactor trip.

No safety injection initiation.

No steam line safety or relief valve operation.

No pressurizer safety valve operation and no pressurizer relief valve operation.

Nuclear power undershoot is less than 3 percent for load decrease.

No manual intervention required to stabilize plant systems.

Plant variables (i.e., Tavg, pressure, feed flow, steam flow, etc.) do not incur sustained or diverging oscillations.

On May 23, 2000, a turbine step load decrease of approximately 10 percent power was initiated at 200 percent per minute from a reactor power level of approximately 95%. Figures 4 through 9 show the response of plant parameters to the step load decrease. Plant variables were stable 13 minutes after initiation of the step load decrease. All Acceptance Criteria were met.

IX. Large Load Reduction Test (0TEP04-SG-0006)

The Objective of this test was to demonstrate the ability of the plant to sustain an approximate 25% power load reduction.

The Acceptance Criteria was that response of plant systems to the step load change is as follows:

No reactor trip.

No safety injection initiation.

No steam line safety or relief valve operation.

No pressurizer safety valve operation.

Nuclear power undershoot is less than 3 percent for load decrease.

No manual intervention required to stabilize plant systems.

Plant variables (i.e., Tavg, pressure, feed flow, steam flow, etc.) do not incur sustained or diverging oscillations.

On May 23, 2000, a turbine step load decrease of approximately 25 percent power was initiated at 200 percent per minute from a reactor power level of approximately 95%. Figures 10 through 15 show the response of plant parameters to the step load decrease. As allowed by the test procedure, the Reactor Coolant System was borated to maintain control rods above the control rod insertion limit. Plant variables were stable 13 minutes after initiation of the step load decrease. All Acceptance Criteria were met.

X. Steam Generator Thermal Performance Test (0PEP07-SG-0003)

The Objective of this test was to verify the performance of the Replacement Steam Generators at or near full power.

The Acceptance Criteria was that measured parameters meet or exceed the values specified in the test procedure.

Parameters measured included Steam Generator Outlet Steam Pressure, Level Stability, and Reactor Coolant System Loop Flow. All Acceptance Criteria were met.

List of Figures

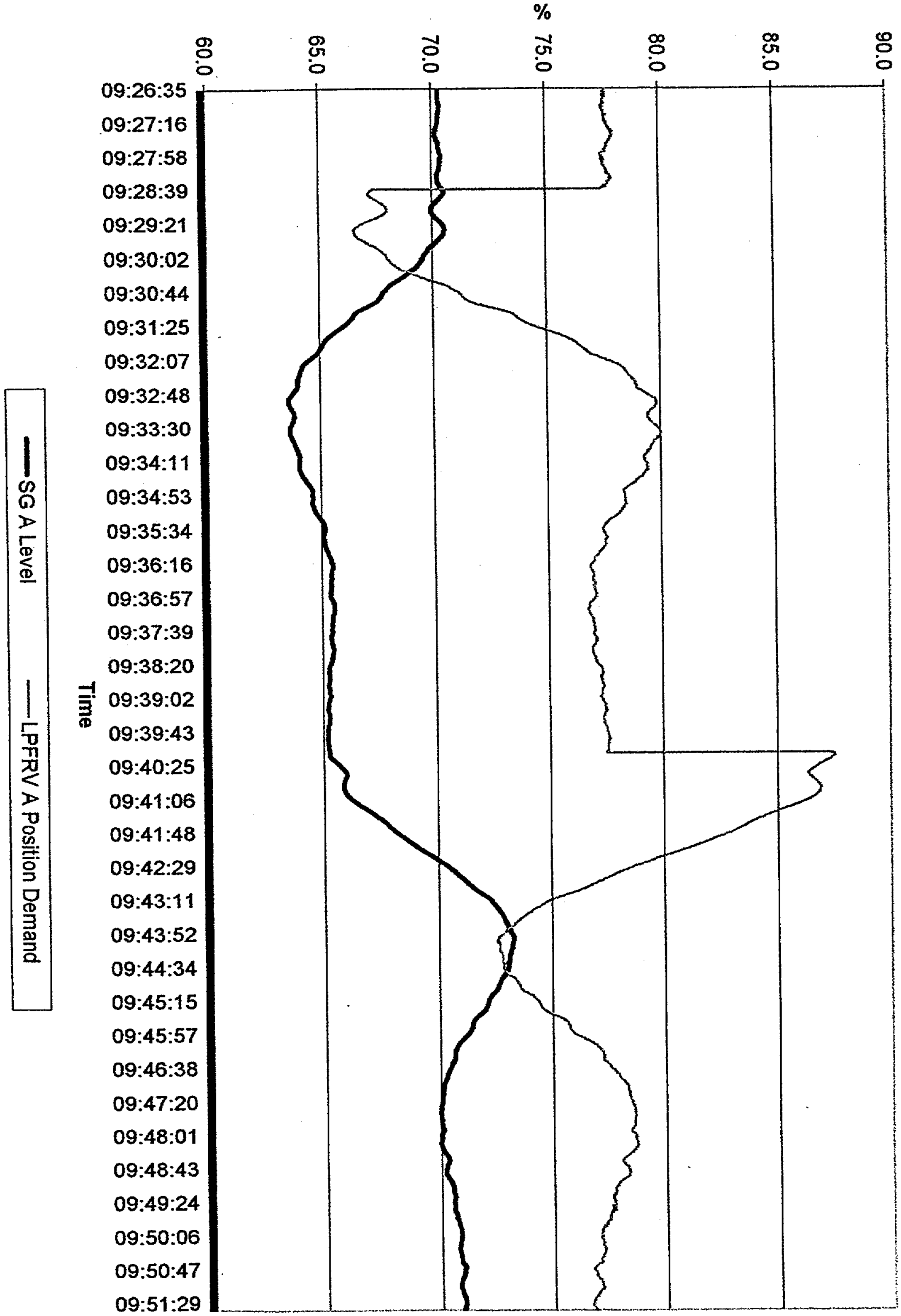


FIGURE 1
LPFRV A Level Swing Test

FIGURE 2
MFRV A Level Swing Test

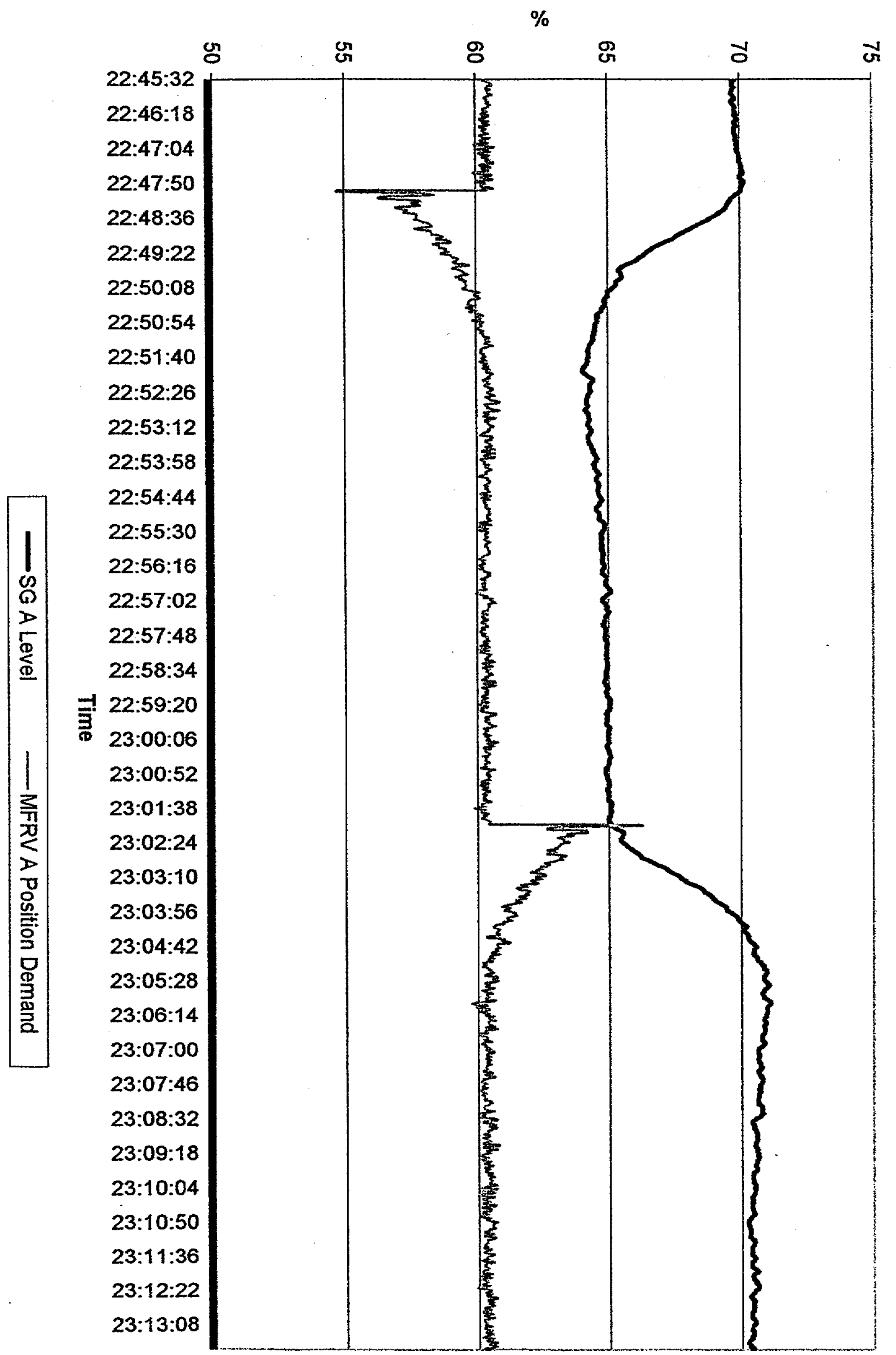
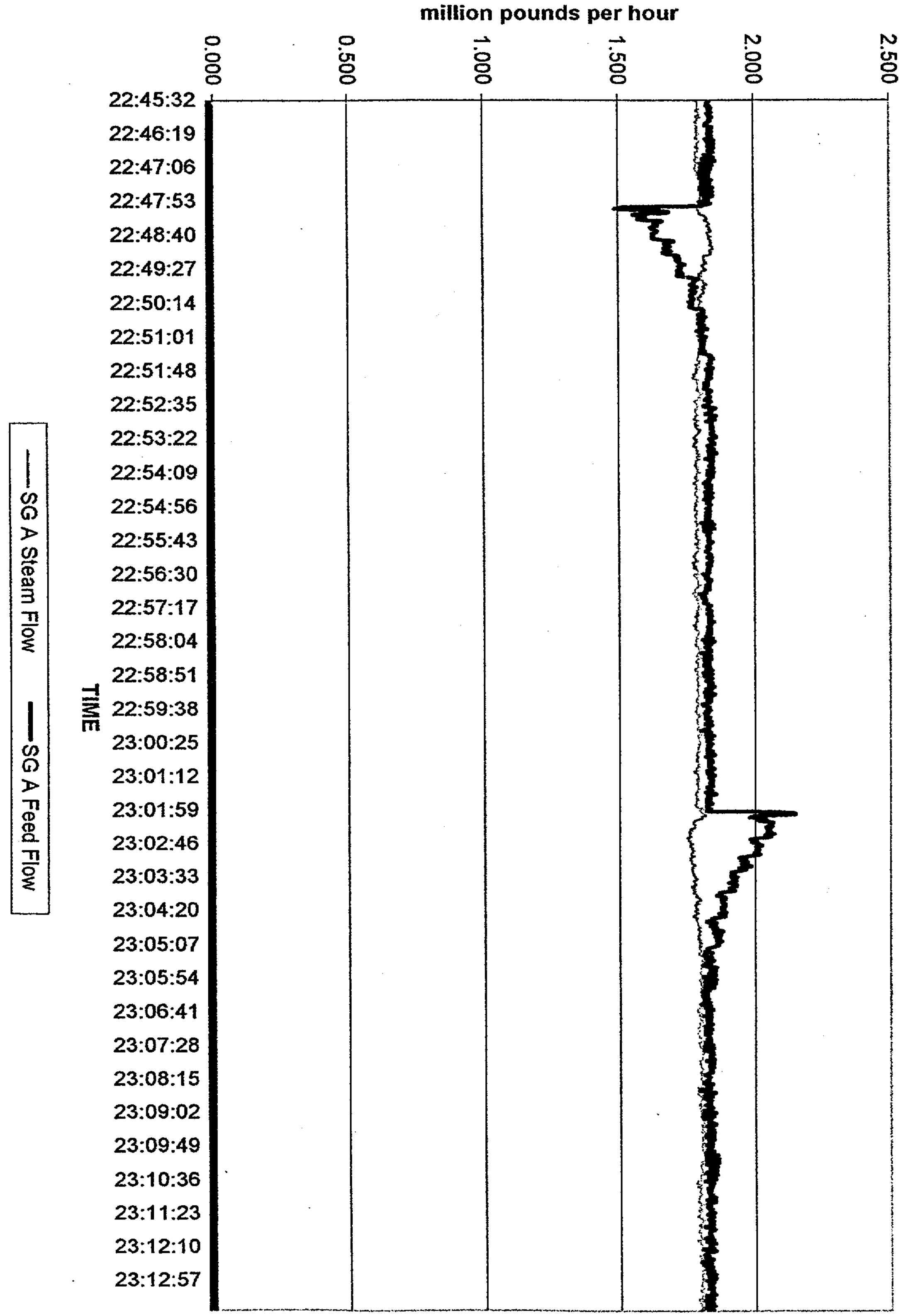


FIGURE 3
MFRV A Level Swing Test



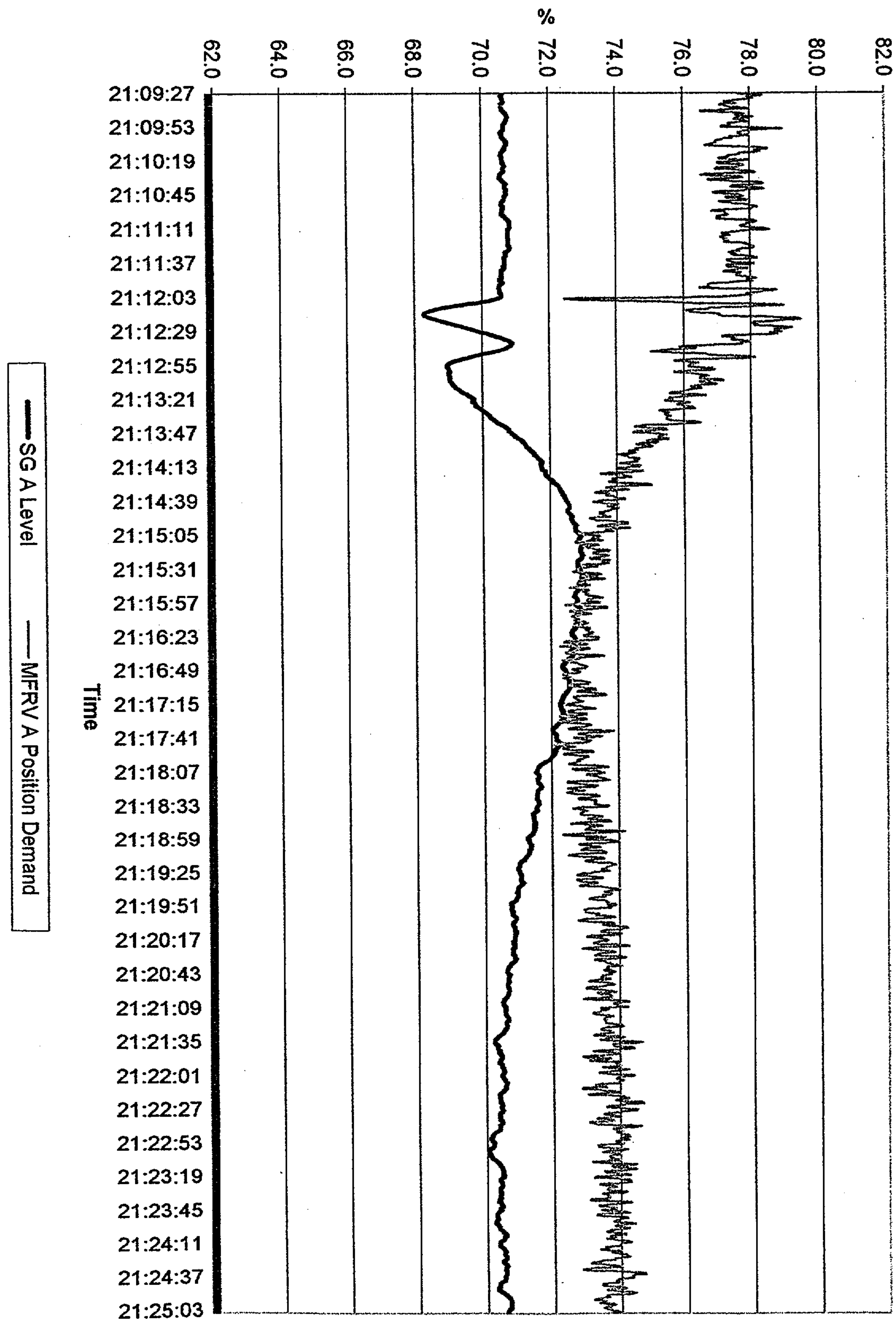


FIGURE 4
10% Step Load Reduction

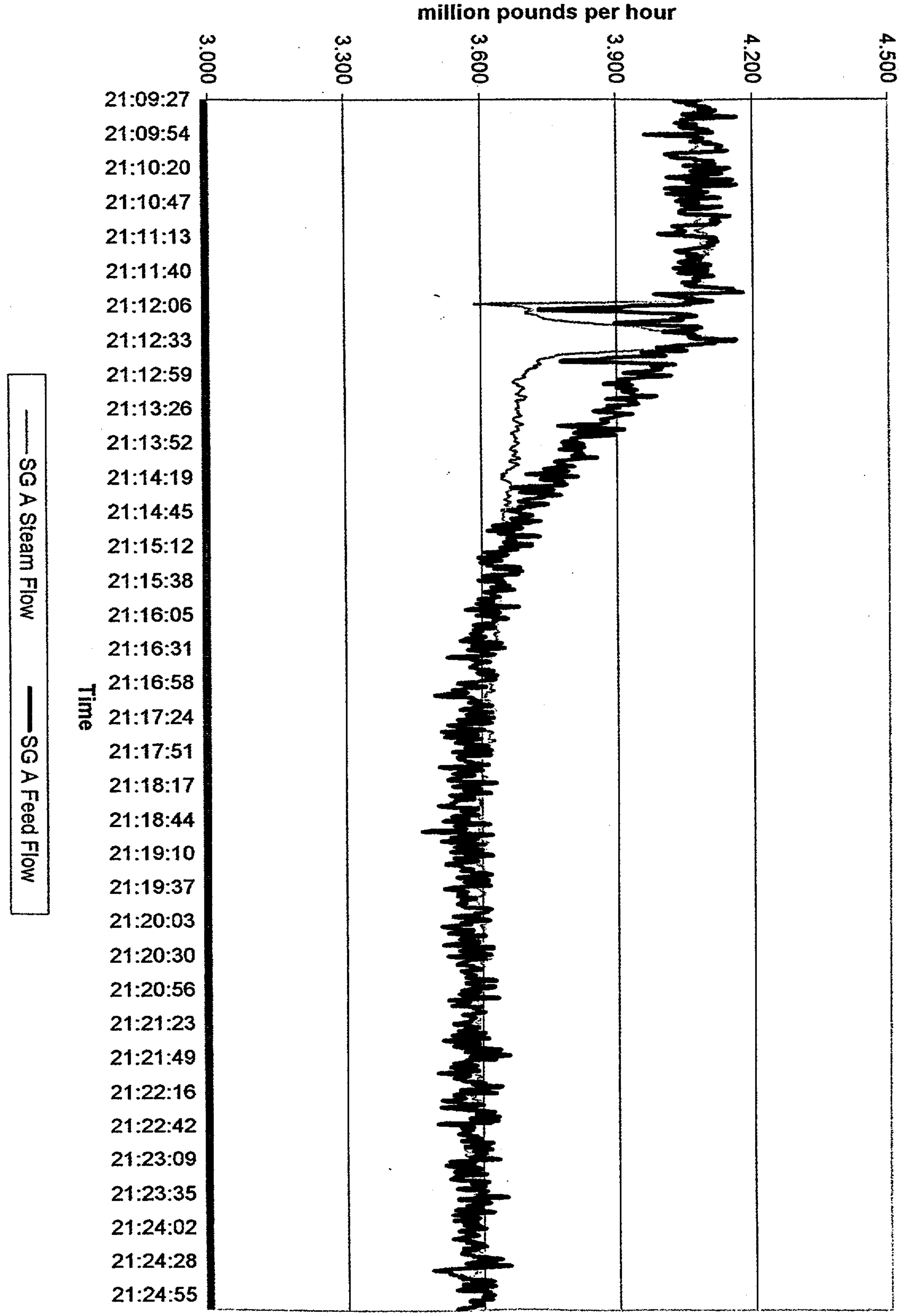


FIGURE 5
10% Step Load Reduction

FIGURE 6
10% Step Load Reduction

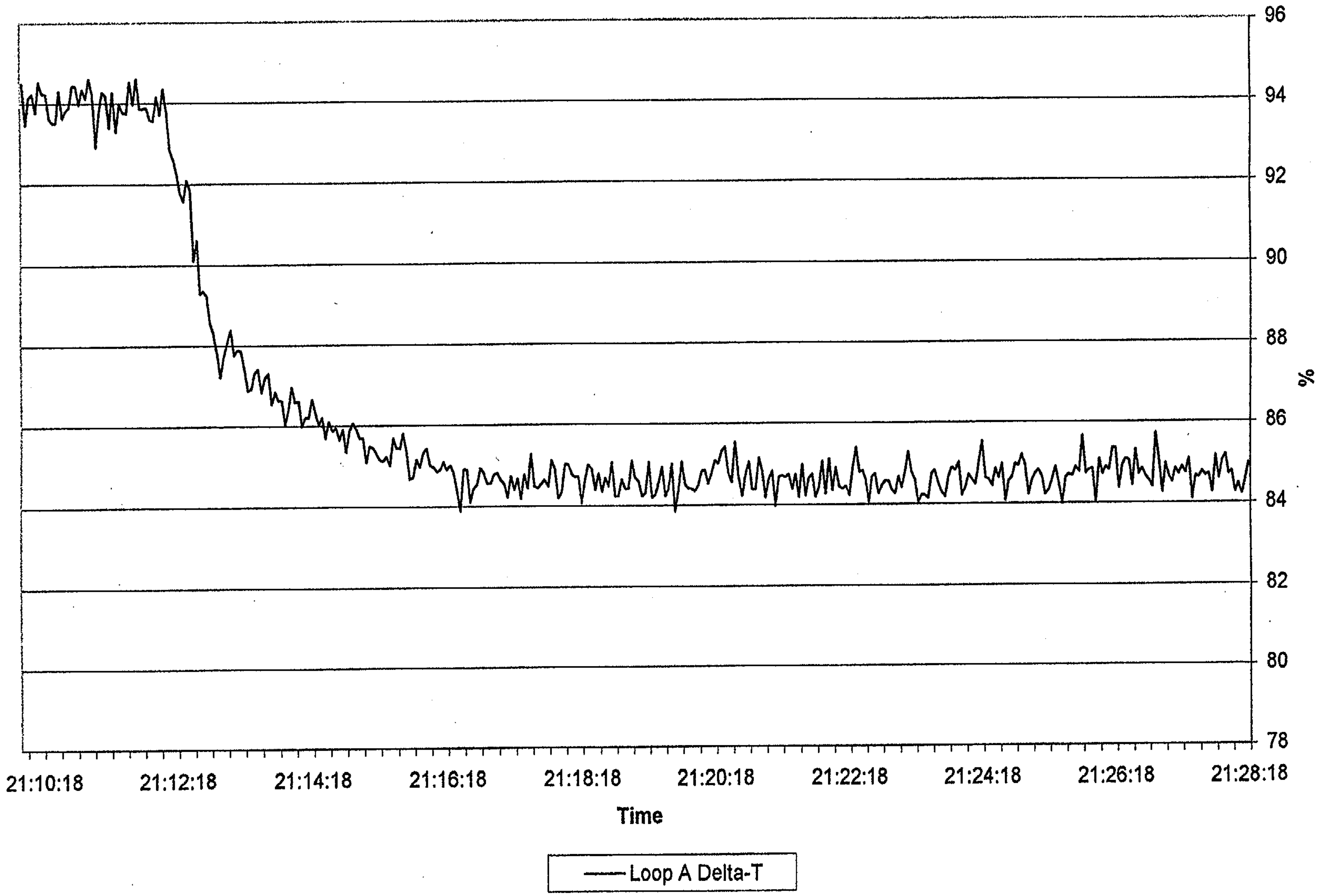


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FIGURE 7
10% Step Load Reduction

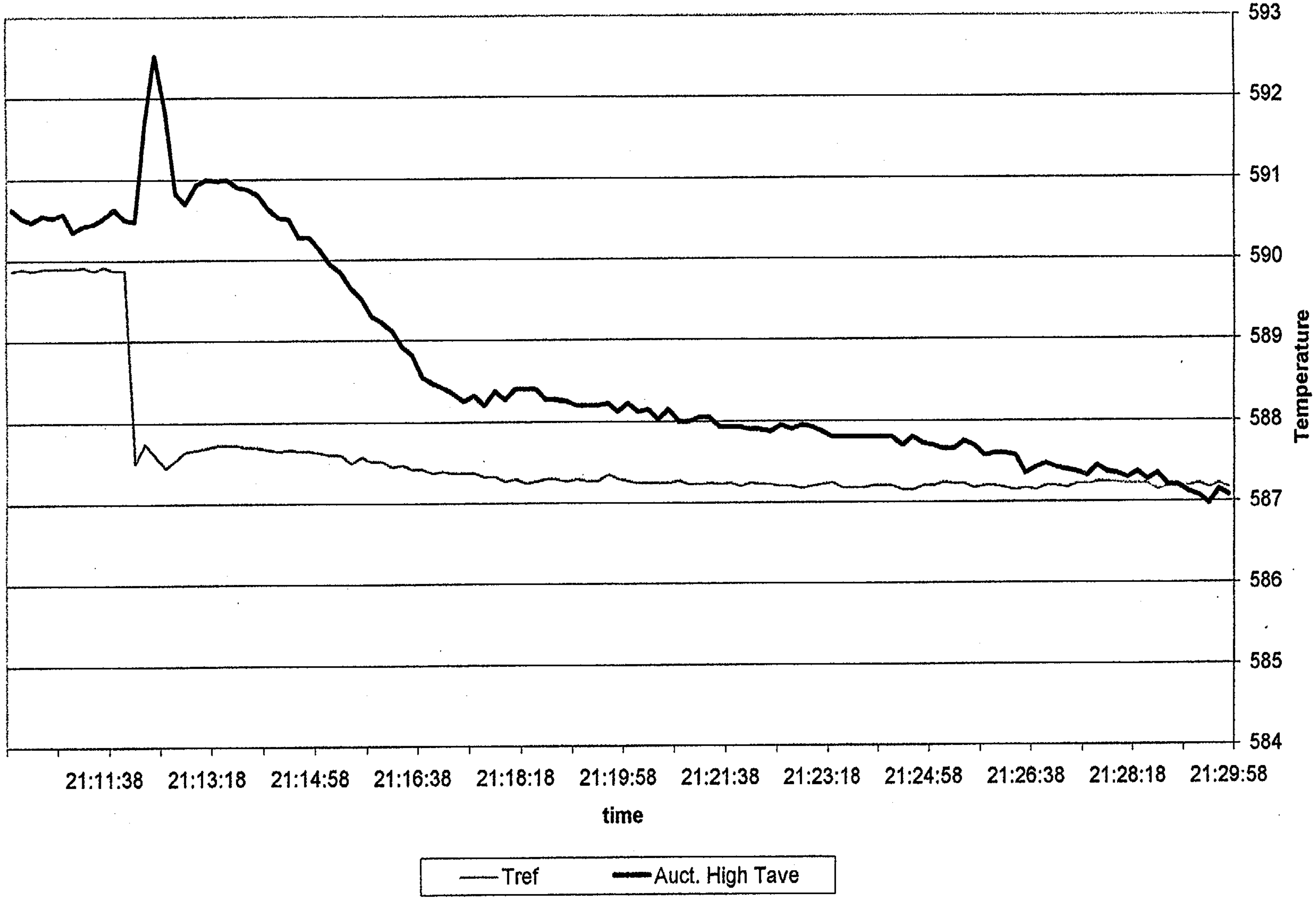


FIGURE 8
10% Step Load Reduction

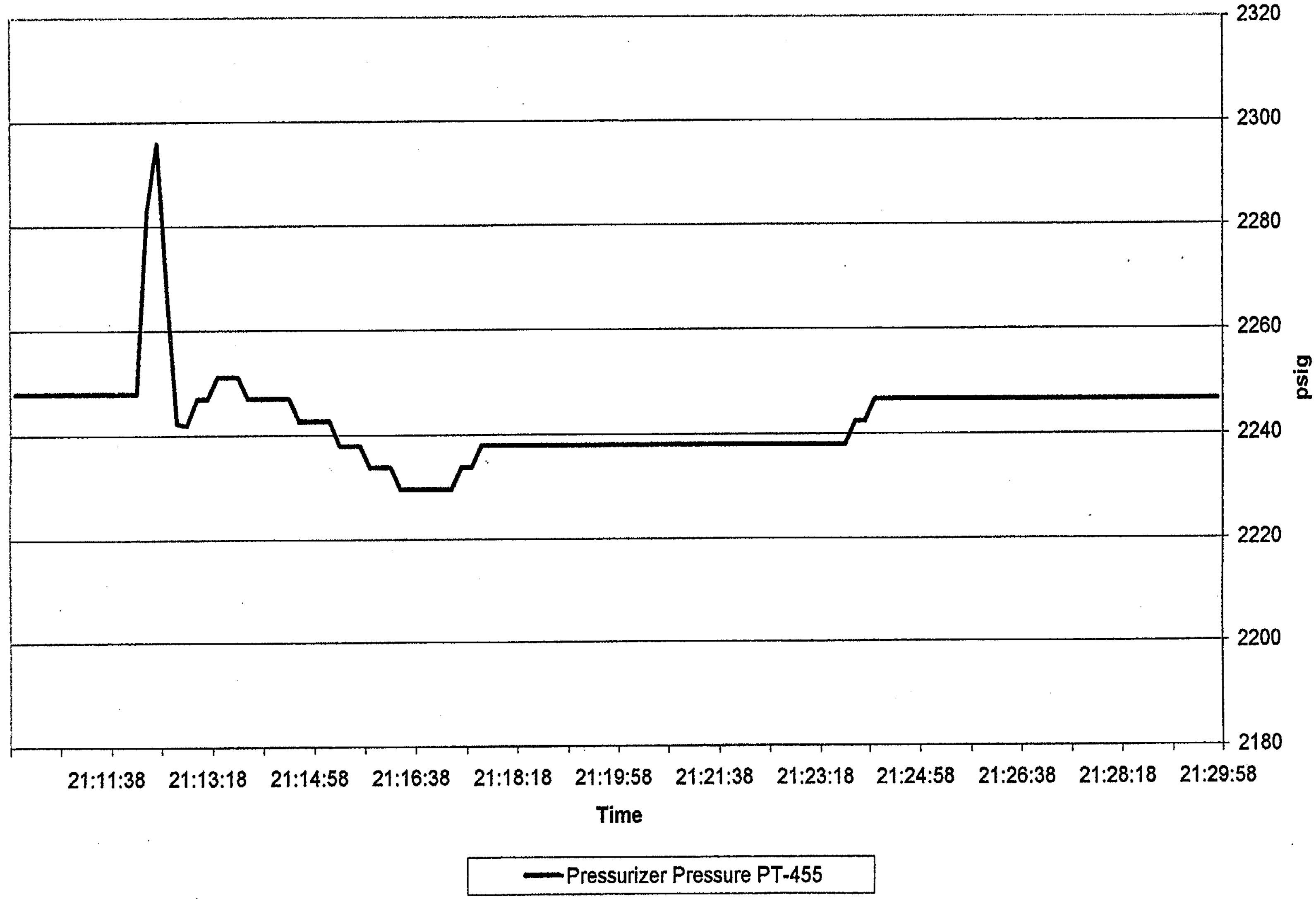


FIGURE 9
10% Step Load Reduction

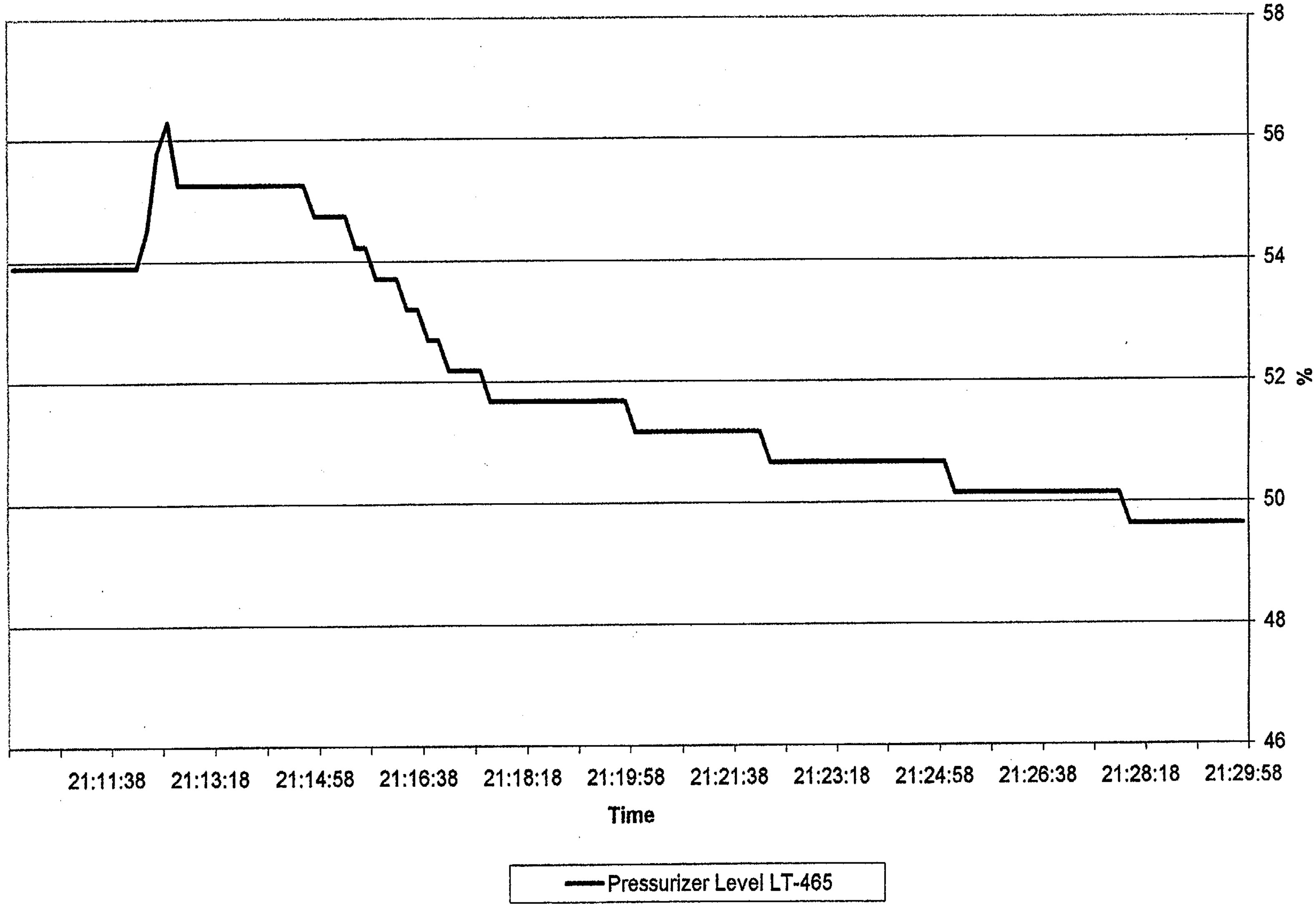
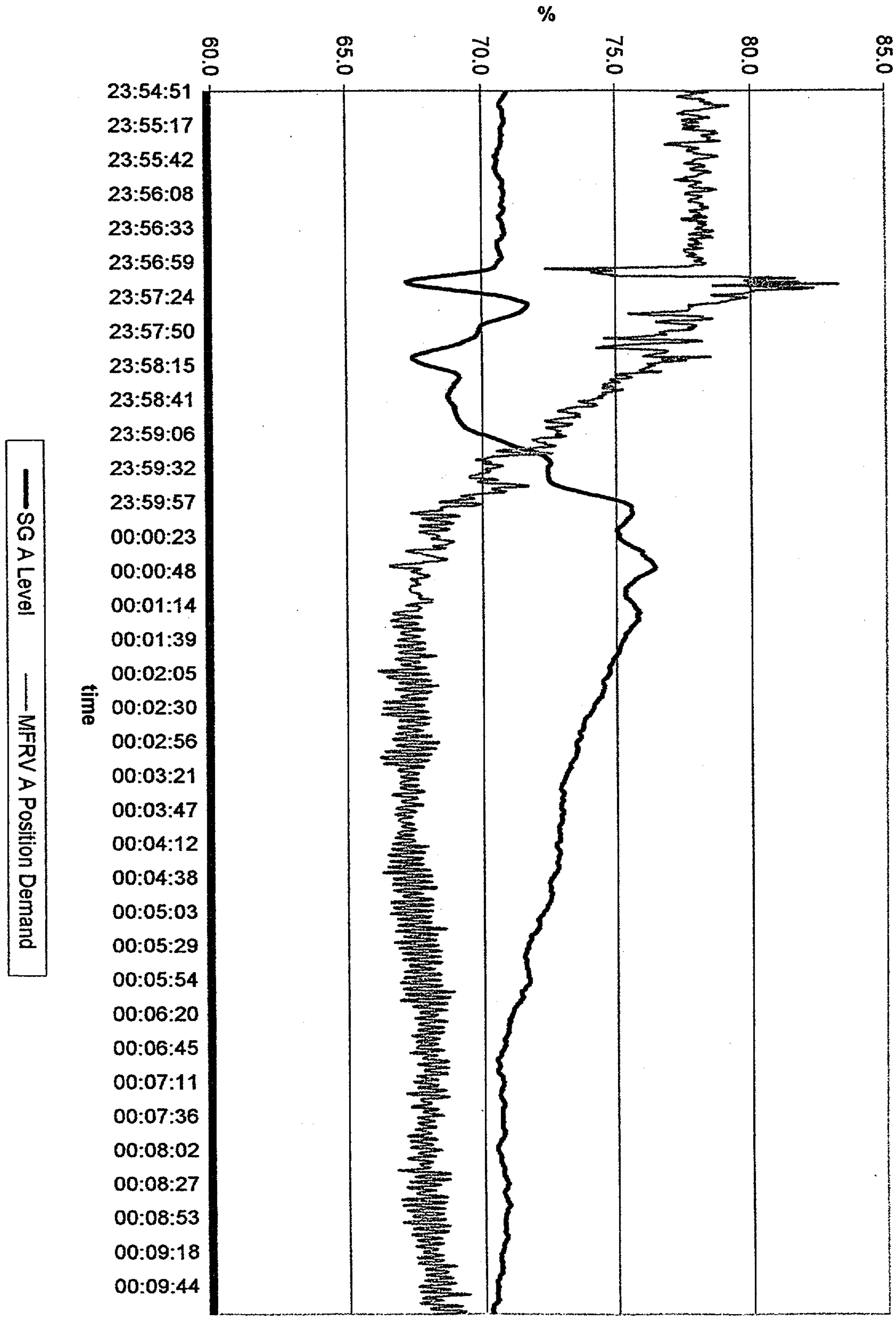


FIGURE 10
25% Step Load Reduction



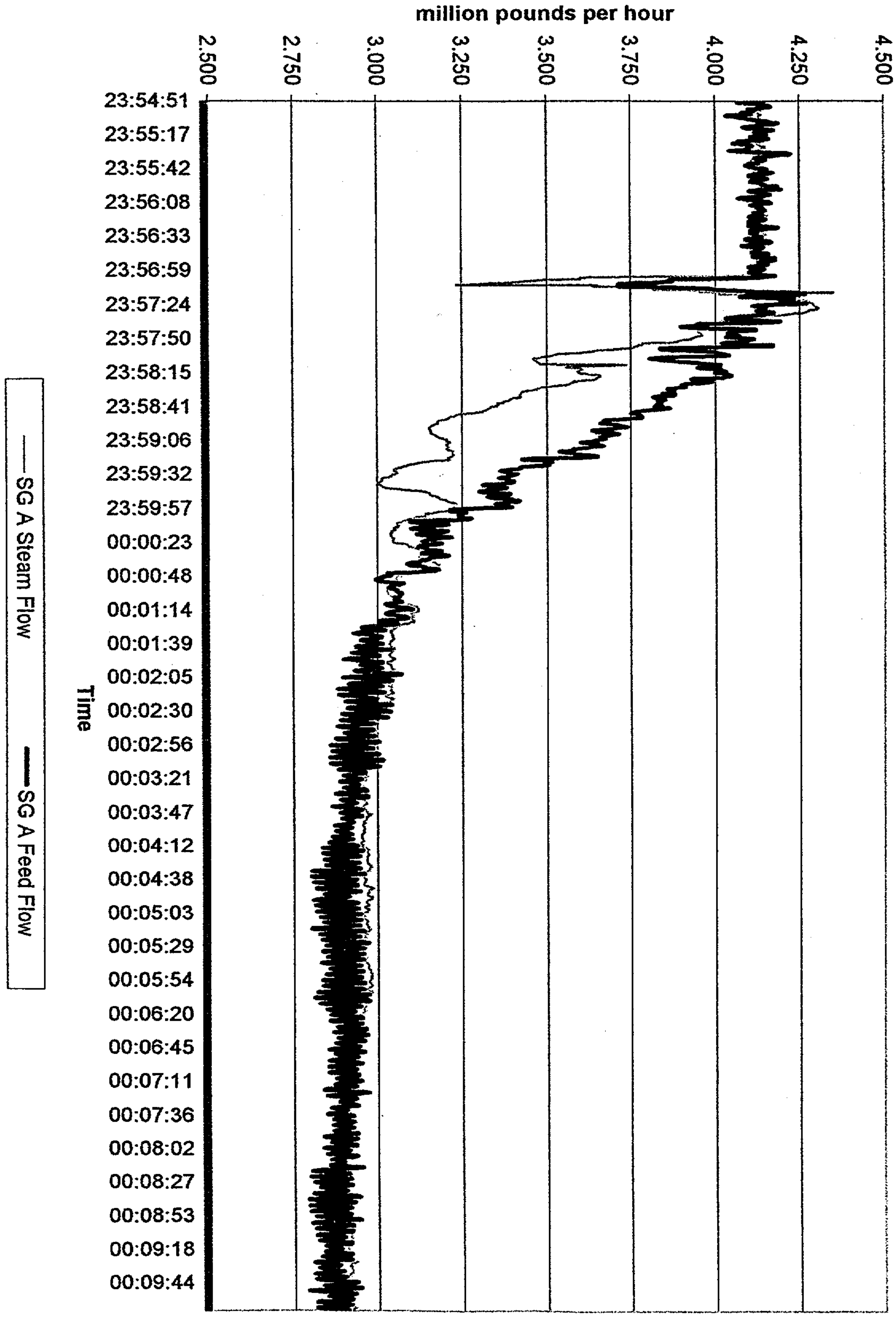


FIGURE 11
25% Step Load Reduction

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FIGURE 12
25% Step Load Reduction

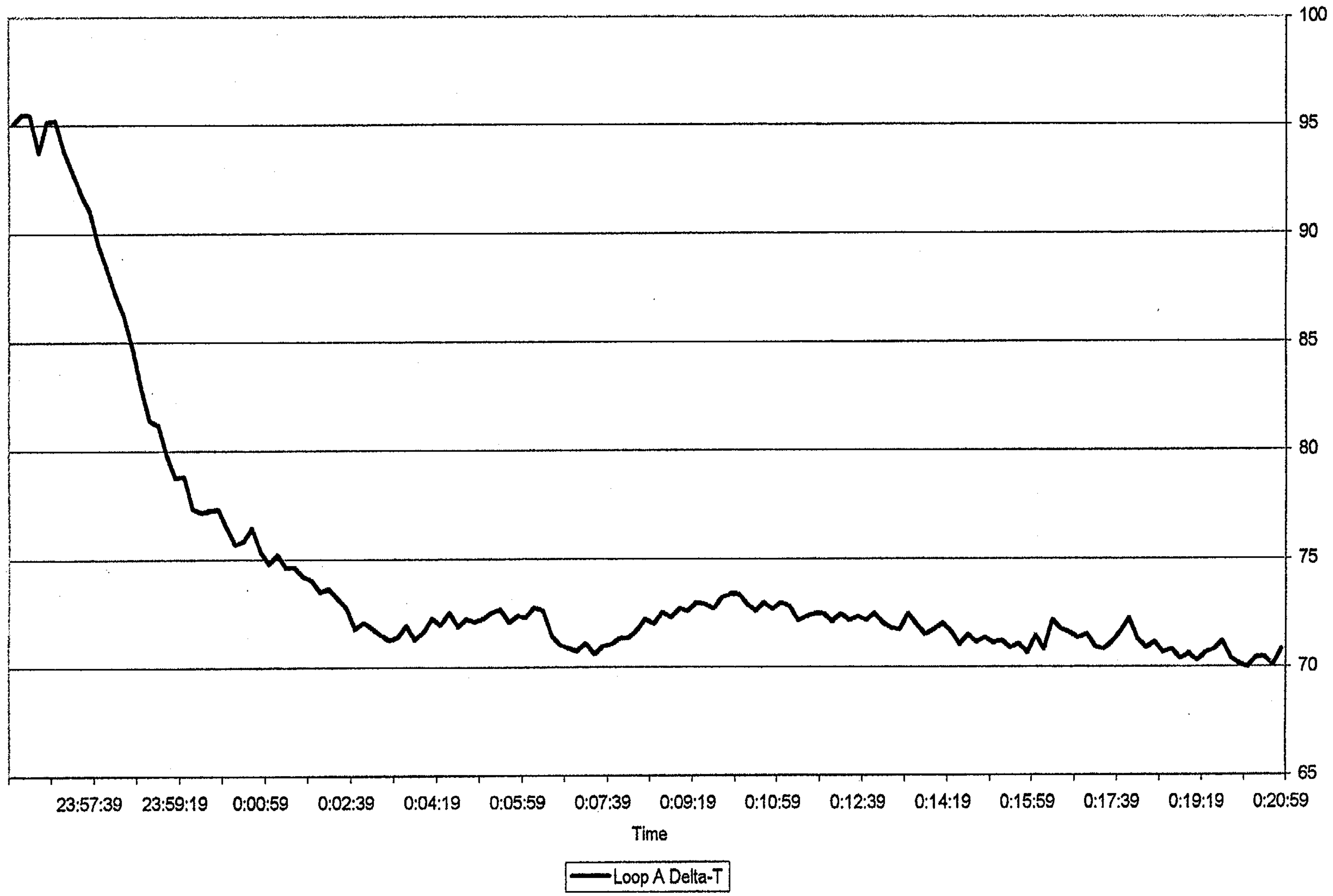

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FIGURE 13
25% Step Load Reduction

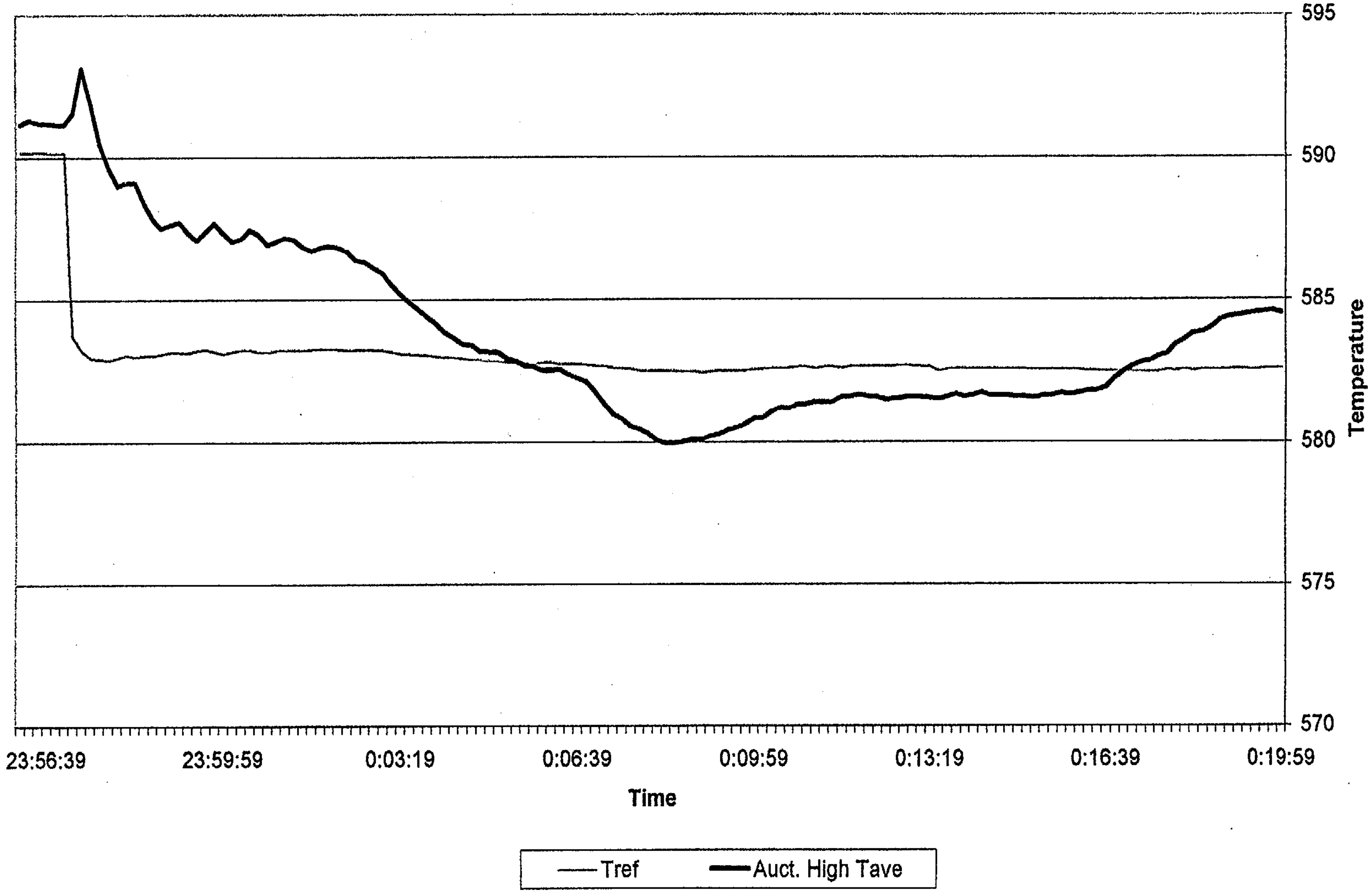

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FIGURE 14
25% Step Load Reduction

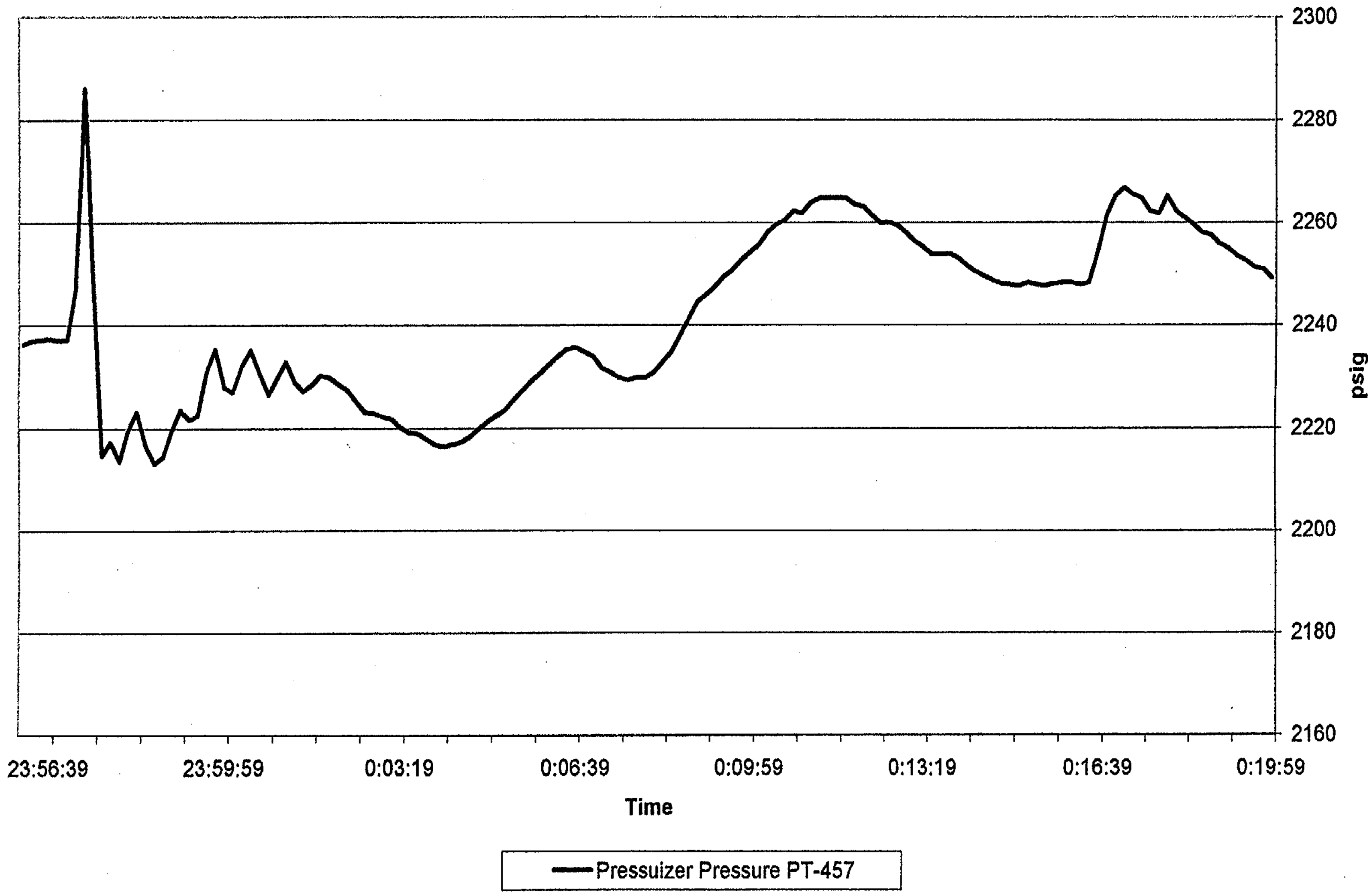

FIGURE 15
25% Step Load Reduction