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Multiloop Integral System Test (MIST): Final Report

Test Group 34, Steam Generator Tube Rupture

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Prepared for
U.S. Nuclear Regulatory Commission
and
Electric Power Research Institute
and
Babcock & Wilcox Owners Group

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ABSTRACT

The multiloop integral system test (MIST) was part of a multiphase program started in 1983 to address small-break loss-of-coolant accidents (SBLOCAs) specific to Babcock & Wilcox-designed plants. MIST was sponsored by the U.S. Nuclear Regulatory Commission, the Babcock & Wilcox Owners Group, the Electric Power Research Institute, and Babcock & Wilcox. The unique features of the Babcock & Wilcox design, specifically the hot leg U-bends and steam generators, prevented the use of existing integral system data or existing integral system facilities to address the thermal-hydraulic SBLOCA questions. MIST and two other supporting facilities were specifically designed and constructed for this program, and an existing facility -- the once-through integral system (OTIS) -- was also used. Data from MIST and the other facilities will be used to benchmark the adequacy of system codes, such as RELAP-5 and TRAC, for predicting abnormal plant transients. The MIST program is reported in 11 volumes. The program is summarized in Volume 1; Volumes 2 through 8 describe groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5 MOD2 and MIST observations; and Volume 11 presents the later, "Phase 4" tests. This is Volume 6 pertaining to Test Group 34. The seven tests of Test Group 34 dealt with steam generator tube rupture. The specifications, conduct, observations, and results of these tests are described herein.

CONTENTS

Contents (Cont'd)

List of Tables

Tables (Cont'd)

List of Figures

Figures (Cont'd)

Figures (Cont'd)

Figure Page

Figures 5.2.32 through 5.2.41: Test 5 Versus Test 1

Figures 5.2.32 through 5.2.38: 0 to 40 Minutes

Figures 5.2.39 through 5.2.41: 0 to 215 Minutes

Figures 5.3.1 through 5.3.17: Test 3, Low-Elevation Rupture

Figures 5.3.1 through 5.3.9: 0 to 8 Minutes

Figures 5.3.10 through 5.3.11: 0 to 16 Minutes

Figures 5.3.12 through 5.3.14: 8 to 24 Minutes

Figures 5.3.15 through 5.3.17: 0 to End of Test

Figures (Cont'd)

Figures (Cont'd)

Figure Page

Figures (Cont'd)

Figures (Cont'd)

EXECUTIVE SUMMARY

Introduction

The multiloop integral system test (MIST) was a scaled 2-by-4 (2 hot legs and 4 cold legs) physical model of a Babcock & Wilcox (B&W), lowered-loop, nuclear steam supply system (NSSS). MIST was designed to operate at typical plant pressures and temperatures. Experimental data obtained from this facility during post-small-break loss-of-coolant accident (SBLOCA) testing are used for computer code benchmarking. The MIST interactions are of intrinsic interest because they may provide insight into expected plant behavior. MIST was necessarily atypical of a plant in certain important respects, however. The MIST interactions therefore are not to be applied directly to a plant.

MIST consisted of two 19-tube, once-through steam generators, a reactor, a pressurizer, 2 hot legs, and 4 cold legs with scaled reactor coolant pumps. Other loop components included a closed secondary system, 4 simulated reactor vessel vent valves (RVVVs), a pressurizer power-operated relief valve (PORV), hot leg and reactor vessel upper head vents, high-pressure injection (HPI), core flood system, and critical flow orifices for scaled leak simulation. Guard heaters, used in conjunction with passive insulation to reduce model heat loss, were used on all primary system components as well as the steam generator secondaries. MIST is illustrated in Figure 1.

Boundary Systems

The MIST boundary systems were sized to power-scale the plant boundary conditions. HPI and auxiliary feedwater (AFW) characteristics were based on composite plant characteristics. Scaled model vents were included in both hot legs and in the reactor vessel upper head. Leaks were located in the cold leg suction and discharge piping, and the upper and lower elevations of steam generator B (for tube rupture simulation). The desired vent and leak flows were obtained using power-scaled restrictors.

Heat Losses and Guard Heaters

MIST was designed to minimize heat losses from the reactor coolant system. Fin effects (instrument penetrations through the insulation) were minimized by using 1/4-inch penetrations for most of the instrumentation. Heat losses due to conduction through component supports were minimized by designing the supports to reduce the cross-sectional area and placing insulating blocks between load-bearing surfaces. The reactor coolant system piping and vessels were covered with passive insulation, active insulation (or guard heaters), and an outer-sealed jacket (to prevent chimney effects). The guard heaters were divided into 42 zones, each controlled by a zonal temperature difference and pipe metal temperature.

Instrumentation

MIST had approximately 850 instruments. These instruments were interfaced to a computer-controlled, high-speed data acquisition system. MIST instrumentation consisted of measurements of temperature, pressure, and differential pressure. Fluid level and phase indications were provided by optical viewports, conductivity probes, differential pressures, and gamma densitometers. Mass flow rates at the system boundaries were measured using Coriolis flowmeters and weigh scales. Loop mass flow rates were measured using venturis or turbines.

Transient Test Program

The MIST transient tests were defined to generate integral system data for code benchmarking. The transient test series was divided into the following seven groups:

- Mapping
- Boundary systems
- Leak-HPI configuration
- HPI-PORV cooling (feed and bleed)
- Steam generator tube rupture
- Noncondensible gas (NCG) and venting
- Reactor coolant pump (RCP) operation

The mapping tests were intended to examine the initial post-SBLOCA transient interactions. In these tests, the primary system inventory was carefully controlled and slowly varied to allow the examination of the normally rapid and overlapping post-SBLOCA events. The boundary system controls, such as guard heating and steam generator secondary level controls, were varied in Test Group 31. The leak size, location, and isolation status, as well as HPI capacity, were varied singly in Test Group 32. Test Group 33 addressed HPI-PORV cooling.

MIST Test Group 34 consisted of 7 tube rupture transients. The single-ended rupture of 1 tube was simulated in Tests 2 and 7, and the remaining 5 tests addressed the double-ended rupture of 10 tubes. The ruptures were nominally located at the top of steam generator B, but this location was shifted to the bottom of steam generator B in Test 3 and its repeat, Test 6. The affected steam generator was isolated in both single-tube rupture tests and in Test 4. Finally, a steam line break and a 10-tube rupture were simulated in Test 5. The tests were generally conducted as specified.

The tests simulating 10 ruptured tubes experienced relatively rapid initial depressurization and mass-depletion transients. Primary system saturation occurred in approximately 1 minute, and the intact loop voided sufficiently to obtain boiler-condenser mode (BCM) depressurization during the initial refill of the intact steam generator secondary. The level in the ruptured steam generator quickly rose to 20 feet, prompting the full opening of the low-capacity steam pressure control valves to control level. The level generally continued to rise, however, exceeding the simulated overfill elevation of 33 feet and stabilizing near 40 feet in the ruptured steam generator secondary.

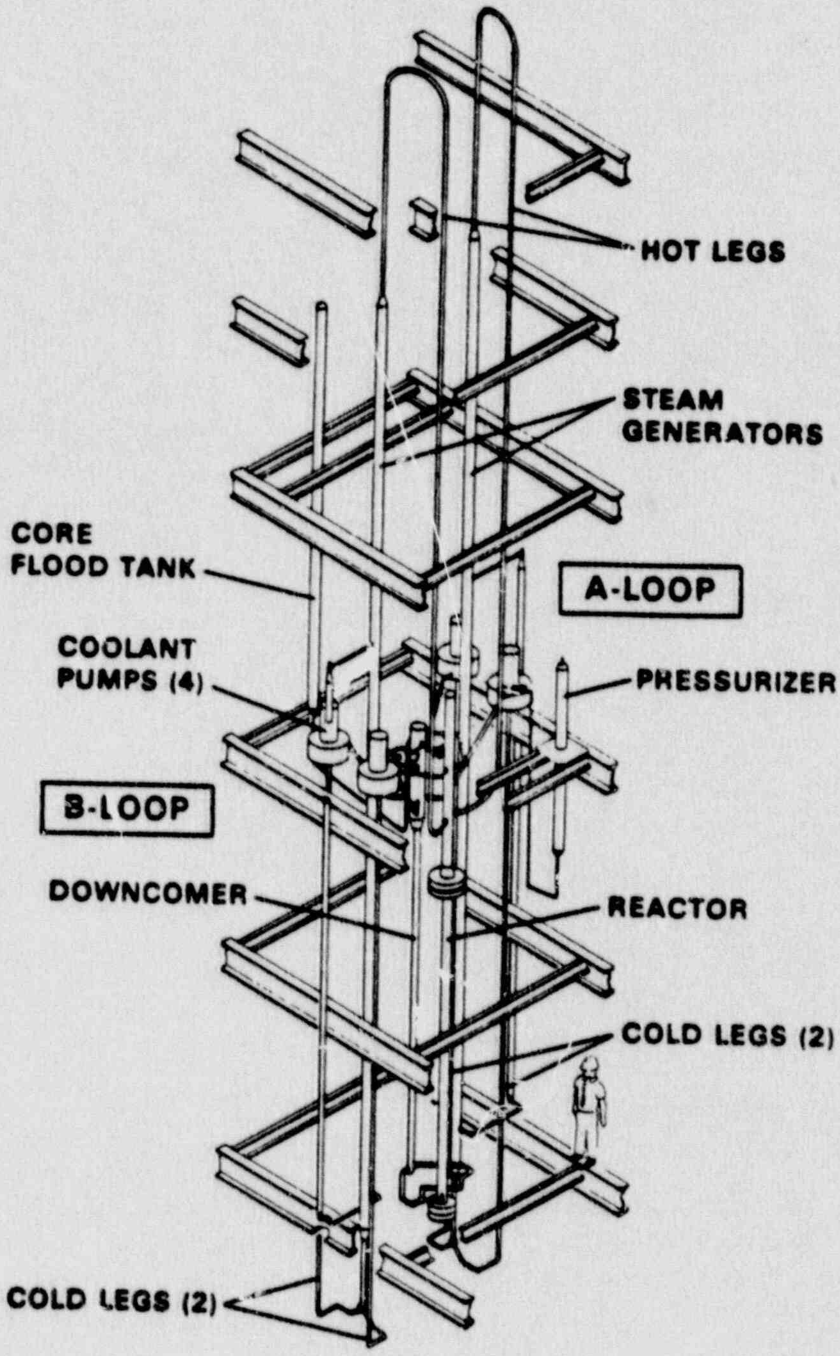
The mid-term transients of the 10-tube rupture tests were characterized by extreme inter-loop asymmetries. Because the primary system depressurization caused reverse heat transfer in the intact steam generator, the intact loop voided extensively, helping to sustain the levels in the affected loop containing the ruptured steam generator. The hot leg levels of the affected loop intermittently achieved the U-bend spillover elevation, reinitiating flow in the affected loop and obtaining primary-to-ruptured steam generator heat transfer. These interactions augmented the depressurization of the primary system and thus helped to mitigate the rupture flow rate. The inter-loop asymmetries abated as the controlled depressurization of the intact steam generator took effect (at about 1 hour).

The 10-tube tests with altered break location (low versus high) and with the simulated steam line break obtained variations in timing and degree of interactions rather than causing singular interactions. Minor changes in initial conditions caused marked differences of the accumulated conditions between nominally identical tests in the 2 tests having low-elevation tube ruptures.

The steam generator secondary repressurized beyond the secondary safety lift pressure following the isolation of the ruptured generator. The intact steam generator was operable, but primary-to-secondary heat transfer was inhibited by intermediate hot leg levels -- above the steam generator but below the U-bend spillover.

The high-elevation, single-ended rupture of 1 steam generator tube was simulated in Tests 2 and 7. Power-operated relief valve (PORV) depressurizations were used in Test 2, and continuous pressurizer venting was used in Test 7. Depressurization differences notwithstanding, the subcooling margin (SCM) was controlled and the two-loop cooldown was maintained in both tests. Upon isolation of the affected steam generator, the cooldowns were continued using one loop. Whereas the hot leg in the affected loop voided intermittently in the test using PORV actuations for depressurization, the affected loop voided continuously and extensively in the continuous-venting test. The loop voiding altered the pressurizer level, but the single loop cooldown continued and the primary system was readily depressurized. The long-term primary fluid conditions were set mainly by the pressure in the active steam generator secondary and by core power. Abrupt changes of the boundary system controls only temporarily altered the quiescent primary system fluid conditions.

Figure 1. Reactor Coolant System -- Multiloop Integral System Test (MIST)



1. INTRODUCTION

The multi-loop integral system test (MIST) was a scaled 2-by-4 (2 hot legs and 4 cold legs) physical model of a Babcock & Wilcox (B&W), lowered-loop, nuclear steam supply system (NSSS). MIST was sponsored by the U.S. Nuclear Regulatory Commission, the B&W Owners Group, the Electric Power Research Institute, and B&W. The MIST results are presented in the following eleven volumes:

1. Summary
2. Mapping Tests, Group 30
3. SBLOCA Tests With Varied Boundary Conditions, Group 31
4. SBLOCA Tests With Altered Leak and HPI Configurations, Group 32
5. HPI-PORV Cooling Tests, Group 33
6. Steam Generator Tube Rupture Tests, Group 34
7. Noncondensible Gas and Venting Tests, Group 35
8. Pump Operation Tests, Group 36, and Core Uncovery Test 3801
9. Inter-Group Comparisons
10. RELAP5/MOD2 Calculations Versus MIST Observations
11. Phase 4 Tests

Test Group 34 is reported herein.

The MIST design, features, and instruments are outlined in section 2. The Group 34 Test Specifications are provided in section 3; the 7 tests of Group 34 dealt with steam generator tube ruptures. The double-ended rupture of 10 steam generator tubes was simulated in 5 tests, 2 tests imposed the single-ended rupture of a single tube. The 10-tube tests varied rupture location, steam line status, and steam generator isolation. The low-elevation, 10-

tube test was repeated. The 1-tube-ruptured tests varied chiefly the primary depressurization method. A series of long-term depressurization and cooldown steps was also tested.

The control of these tests and instrument performance are described in section 4. Section 5 provides a brief narrative description of each test, inter-test comparisons, and in section 5.6, a summary of major observations. Key data plots are provided with each test. A complete plot set for each test is provided in the enclosed microfiche, and is described and indexed in Appendix A of Volume 9.

Figures 1.1 through 1.7 provide an overview of the tube rupture test transients. Major events and timing are indicated on traces of primary system pressure versus primary system total fluid mass.

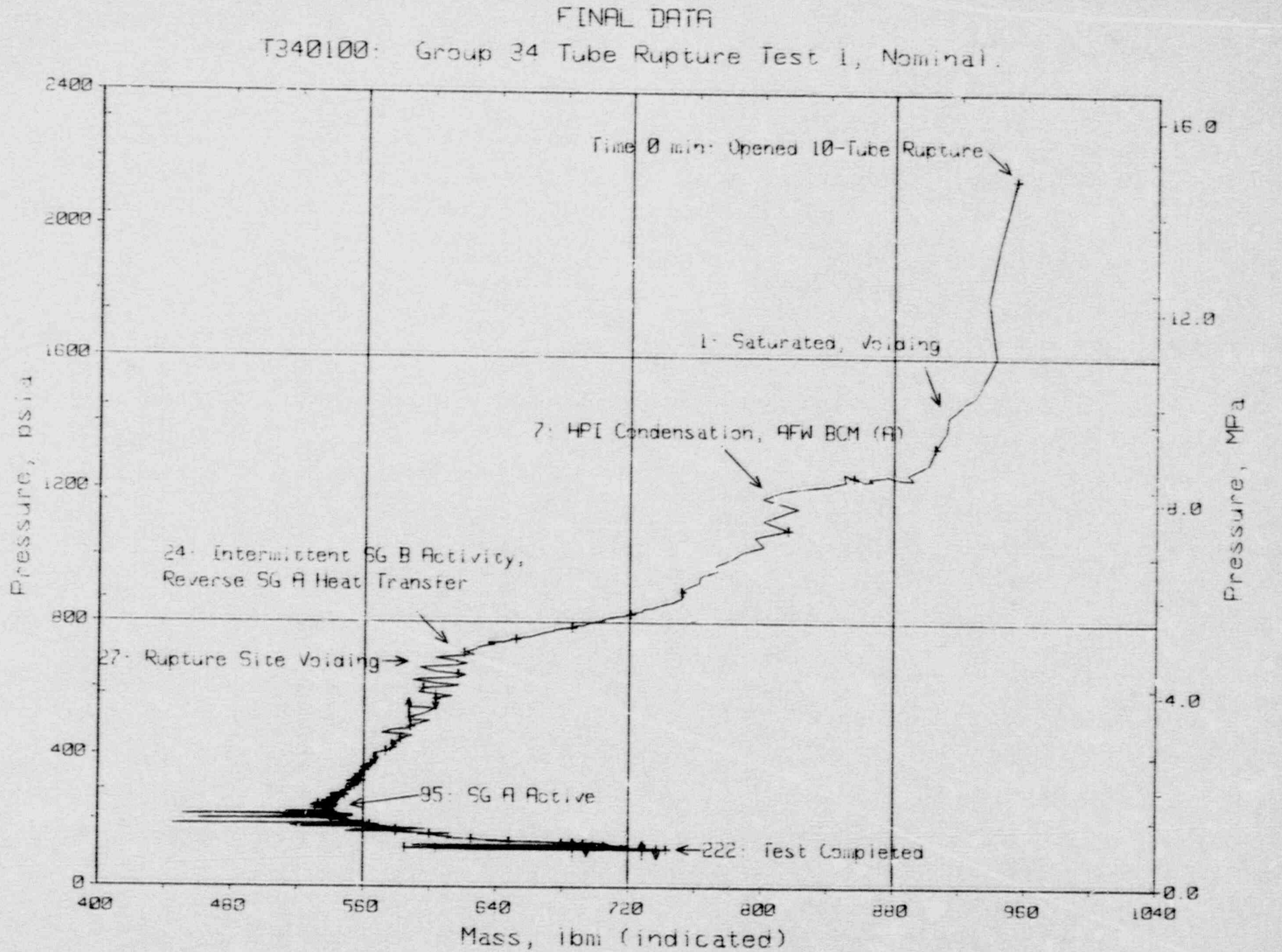


Figure 1.1 Primary System Pressure Vs Primary System Total Fluid Mass

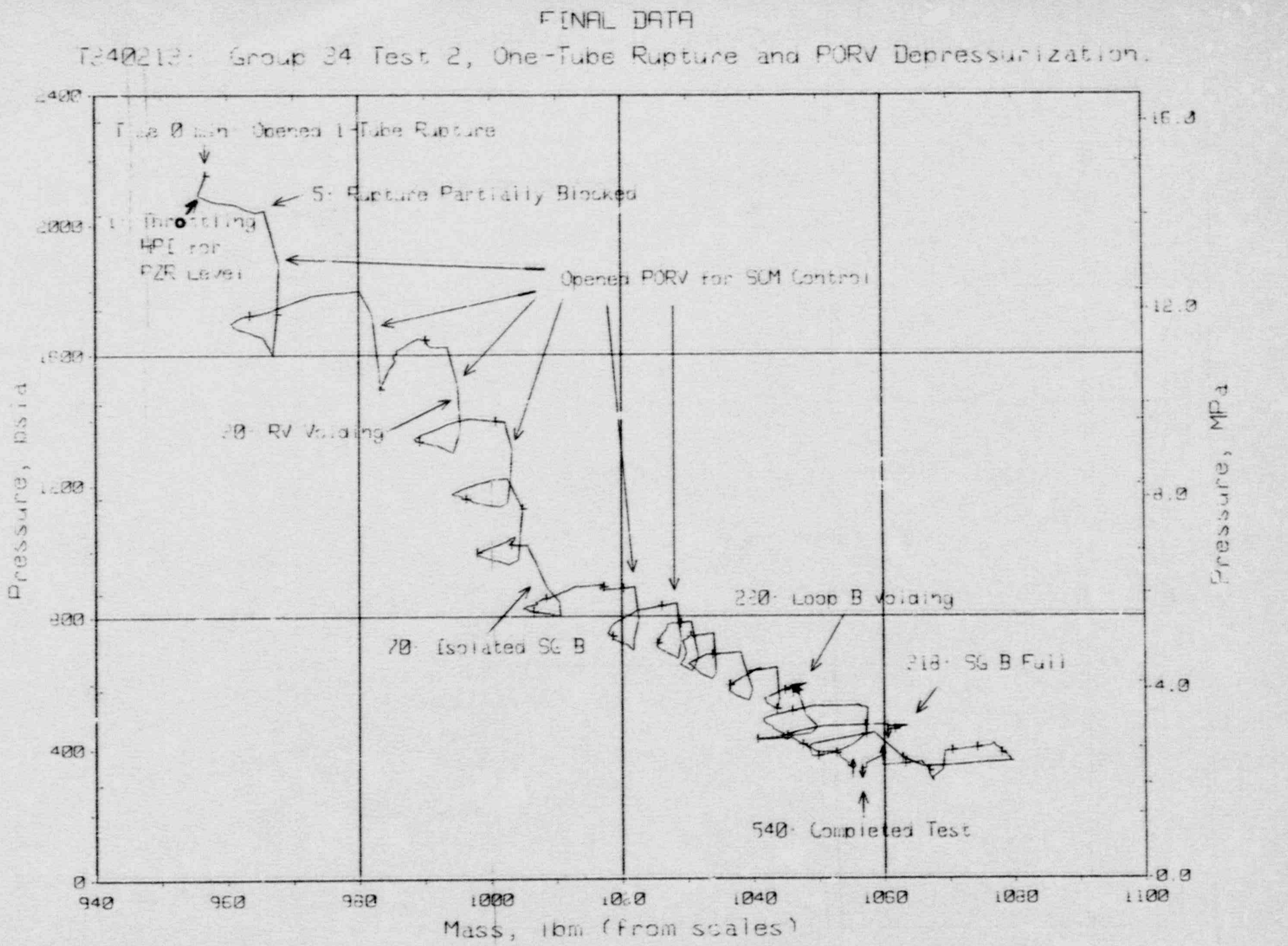


Figure 1.2 Primary System Pressure Vs Primary System Total Fluid Mass

PP/PM

1-5

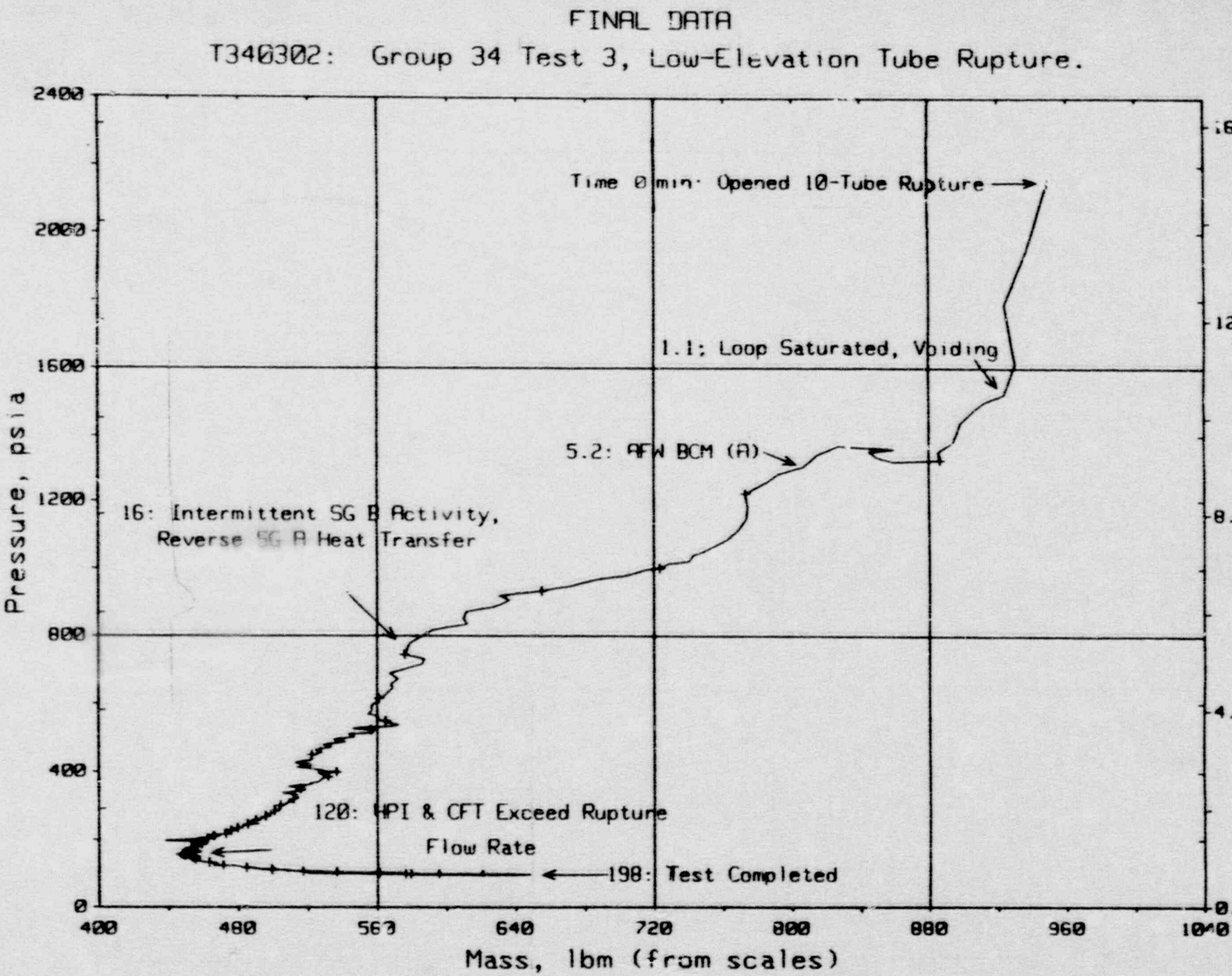


Figure 1.3 Primary System Pressure Vs Primary System Total Fluid Mass

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PPvPM

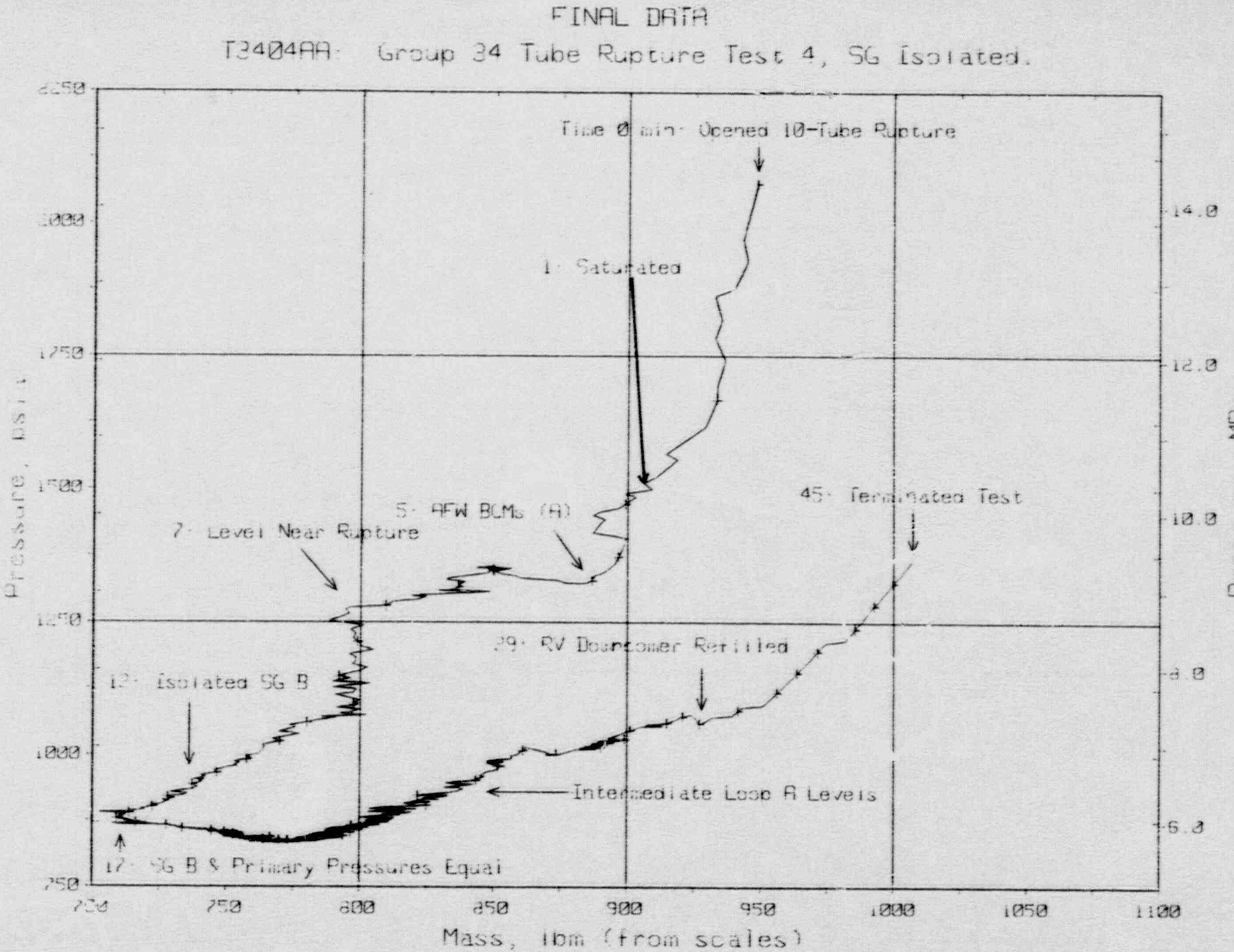

Figure 1.4 Primary System Pressure Vs Primary System Total Fluid Mass

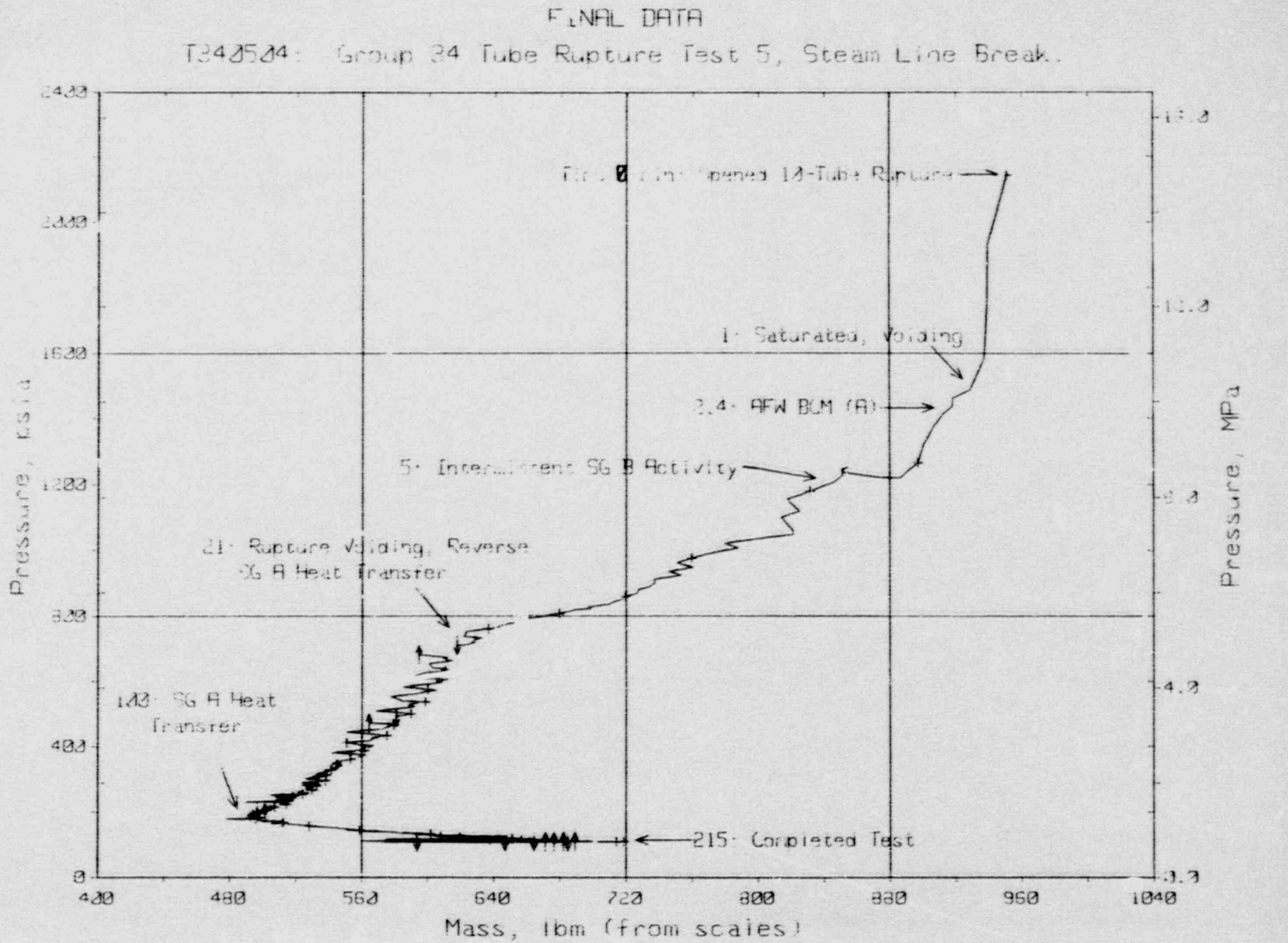

Figure 1.5 Primary System Pressure Vs Primary System Total Fluid Mass

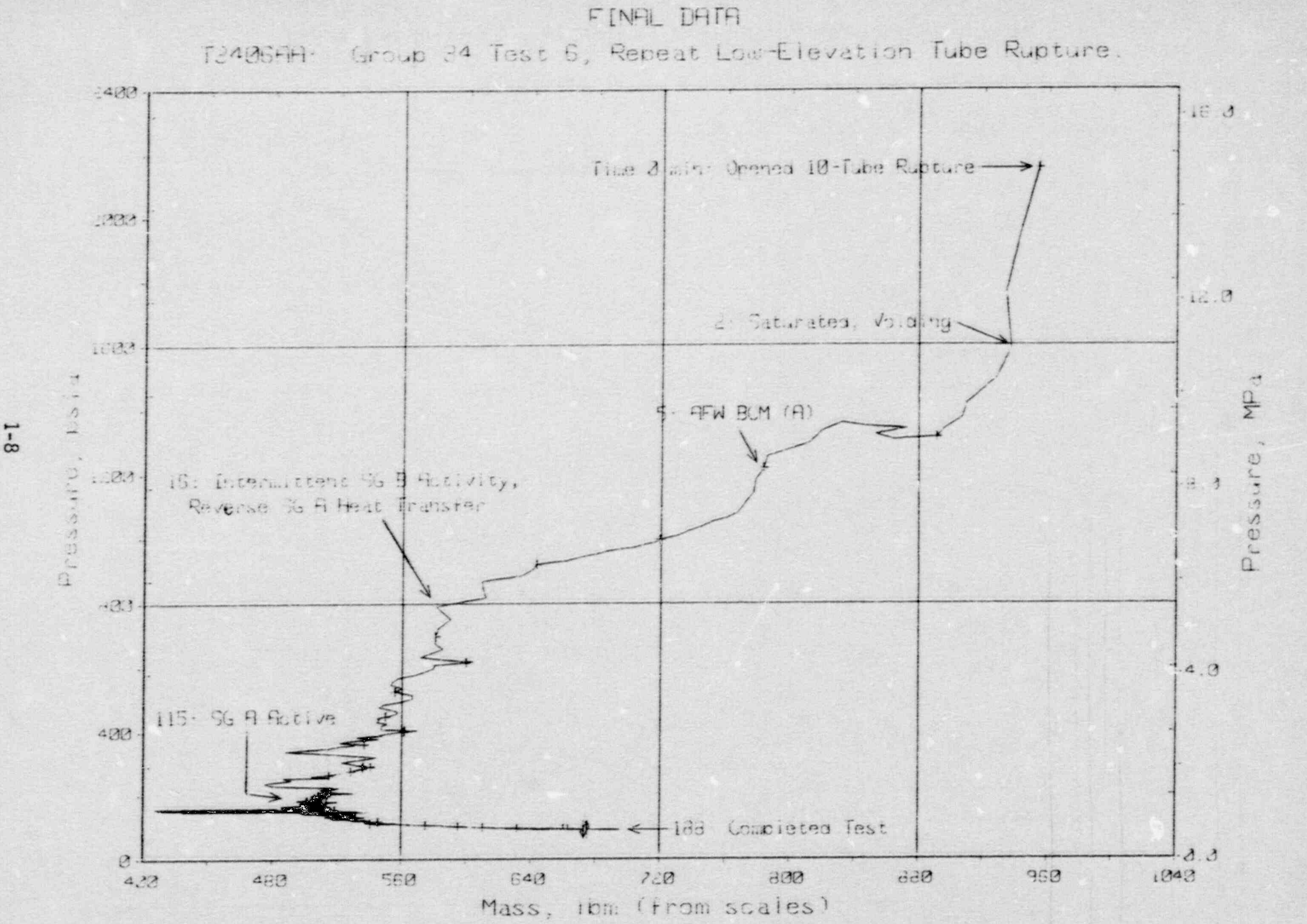

Figure 1.6 Primary System Pressure Vs Primary System Total Fluid Mass

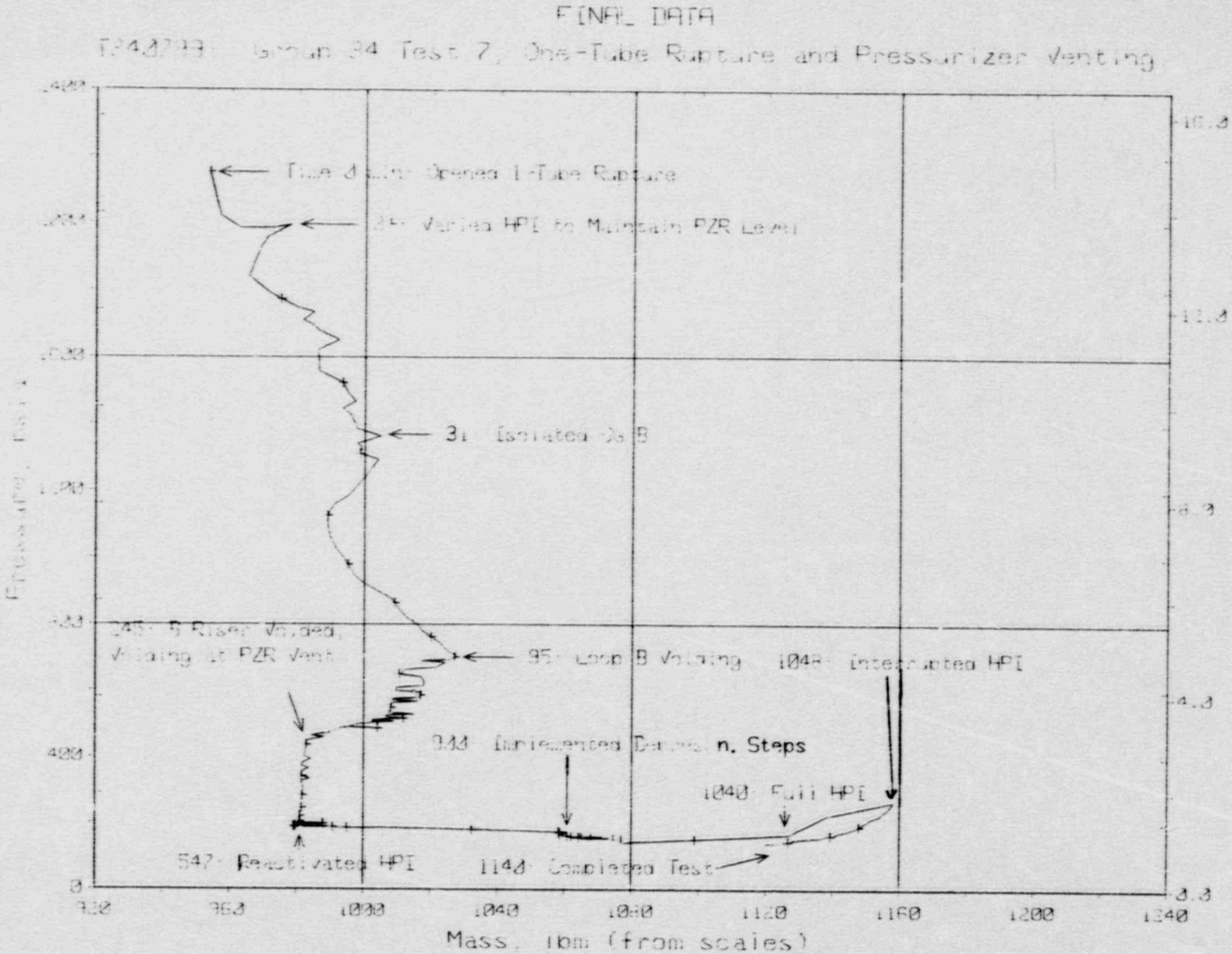

Figure 1.7 Primary System Pressure Vs Primary System Total Fluid Mass

2. FACILITY DESCRIPTION

2.1. Introduction

MIST was a scaled, 2-by-4 (2 hot legs and 4 cold legs) model of a B&W, lowered-loop, nuclear steam supply system (NSSS). MIST was designed to operate at typical plant pressures and temperatures. Experimental data obtained from this facility during post-SBLOCA testing are used for computer code benchmarking.

The reactor coolant system of MIST was scaled according to the following criteria, listed in order of decreasing priority: elevation, post-SBLOCA flow phenomena, component volume, and irrecoverable pressure drop. MIST consisted of two 19-tube, once-through steam generators; reactor; pressurizer; 2 hot legs; and 4 cold legs, each with a scaled reactor coolant pump.

Other loop components in MIST included a closed secondary system, 4 simulated reactor vessel vent valves (RVVVs), a pressurizer power-operated relief valve (PORV), hot leg vents and reactor vessel upper-head vents, high-pressure injection (HPI), core flood system, and critical flow orifices for scaled leak simulation. Guard heaters, used in conjunction with passive insulation to reduce model heat loss, were included on the steam generator secondaries and on all primary coolant components. The system was also capable of noncondensible gas addition at selected loop sites.

The approximately 850 MIST instruments were interfaced to a computer-controlled, high-speed data acquisition system. MIST instrumentation consisted of measurements of temperature, pressure, and differential pressure. Fluid level and phase indications were provided by optical viewports, gamma densitometers, conductivity probes, and differential pressures. Mass flow rates in the circulation loop were measured using venturis and a cooled thermocouple, and at the system boundaries using Coriolis flowmeters and weigh scales.

2.2. MIST Design

MIST was a scaled, full-pressure, experimental facility arranged to represent the B&W lowered-loop plant design. Like the plant, MIST was a 2-by-4 arrangement with 2 hot legs and 4 cold legs, as shown in Figure 2.1. MIST was designed for prototypical fluid conditions, with emphasis on being leak-tight and minimizing heat loss.

The scaling of MIST followed the approach and priorities used for OTIS[1]: that is, elevation, post-SBLOCA phenomenon, component and piping volumes, and irrecoverable pressure losses. MIST retained full plant elevations throughout the primary system and the steam generator secondaries. Only the elevations of several non-flow regions were compromised, primarily to optimize power-to-volume scaling. Key interfaces were maintained -- these included the hot leg U-bend spillover, upper and lower tubesheets of the steam generator (secondary faces), cold leg low point, pump discharge, cold and hot leg nozzles, core (throughout), and points of emergency core cooling system (ECCS) injection.

Two-phase behavior during voiding of the hot leg U-bend and flow interruption was sufficiently prototypical; that is, both the plant and the model were expected to encounter phase separation early in the post-SBLOCA transient. The MIST hot leg pipes were large enough to admit bubbly flow.

Fluid volume was 40% larger than power-to-volume scaling would dictate; the hot legs, cold legs, and upper downcomer were oversized. This atypicality was imposed by the previously described two-phase requirements and by considering component irrecoverable pressure losses. The excess volume of the hot leg slowed the rate of level decrease for power-scaled draining and similarly retarded the rate of level increase for power-scaled injection. Although the excess volume of loop fluid delayed system heatup and cooldown, this effect was usually minor compared to the long-term impact on system energy of leak-HPI cooling. The concentration of excess volume in the piping runs decreased fluid velocities in the hot legs and cold legs and therefore lengthened the transit time of loop fluid. Irrecoverable pressure drops were well preserved.

The MIST core and steam generators were full-length subsections of their plant counterparts. As shown on Figure 2.2, the core consisted of a 7-by-7 array of 45 full-length, 0.430-inch-diameter heater rods and four simulated incore guide tubes. Plant-typical fuel pin pitch and grid geometry were used. The simulated rods were capable of full-scale power output but were limited to approximately 10% of scaled power for the planned MIST testing. (The ratio of plant power to MIST power was 817:1.) A fixed, axial heat flux profile and a flat, radial heat flux profile were used. The axial peak-to-average flux ratio was 1.25:1.

The steam generators, shown in Figure 2.3, each contained 19 full-length tubes. The tubing diameter (5/8-inch OD), material, and triangular pitch of the tube bundle (7/8 inch, tube centerline to centerline) were prototypical. The geometry of the tube support plates (TSPs) was similar to that of the plant TSPs and provided equivalent flow areas and irrecoverable pressure losses. The MIST steam generators contained 16 TSPs, versus 15 in the plant. The flow holes of the MIST TSPs were drilled rather than broached. Also, the thicknesses of the MIST and plant steam generator tubesheets were unequal.

The hot legs used 2.5-inch, schedule-80 piping (2.32-inch ID). This diameter admitted bubbly flow and approximated the irrecoverable pressure loss of a plant hot leg. With the schedule-80 piping, the metal-to-fluid volume ratio in MIST was only 20% greater than that of the plant. The horizontal runs in the hot leg were approximately 1 foot long to accommodate the gamma densitometers. The pipe diameters of the hot leg U-bends maintained the pipe diameters of the hot leg risers and stubs. The radii of curvature of the hot leg U-bends were 1.61 ft. This curvature was chosen to match the horizontal displacement between the riser and stub while preserving the elevation of the U-bend spillover and approximating a power-scaled U-bend volume. Phase separation at the U-bend was predicted to occur at and below approximately 18% of scaled full power in MIST, versus 8% in the plant.[2] Beyond the U-bend, the hot leg piping in the model extended 12 feet, versus 1.5 feet in the plant, to span the height of the inlet plenum for the plant steam generator.

The four cold legs preserved elevation throughout. Two-inch, schedule-80 piping (1.939-inch ID) was used primarily to match irrecoverable pressure drop. This piping size also preserved the cold-leg Froude number, which influenced the mixing of the HPI and RVVV fluid streams. The cold leg horizontal piping runs were shortened, but the slope of the plant cold leg discharge piping was approximately maintained. HPI was injected into the sloping pipes at the appropriate elevation; the diameter of the model HPI nozzle was selected to preserve the ratio of fluid momentum between the cold leg and HPI.

A model reactor coolant pump was mounted in each cold leg. Suction and discharge orientations were prototypical. The pumps delivered single-phase scaled flows at plant-typical heads, allowed for simulated pump bumps by matching the plant pump spinup and coastdown times, and permitted operation under single- and two-phase conditions. The specific speeds of the model pumps were only one-tenth of those of the plant pumps. Therefore, the two-phase characteristics of the model pumps did not simulate those of the plant pumps.

The MIST reactor vessel employed an external annular downcomer. Inter-cold leg coupling was restricted toward that of a plant by using fins in the downcomer annulus to form quadrants, as shown in Figure 2.4. The annular gap was 1.4 inches and the gap at each fin was 0.4 inches. Each downcomer quadrant was connected to a separate RVVV simulation and a cold leg. The two core flood tank nozzles were each located at an interface between two downcomer quadrants.

The geometry of the model downcomer was annular down to the elevation of the top of the core. Just above the top of the core, the downcomer was gradually reconfigured to form a single pipe for the remaining elevation. The lower downcomer region obtained approximately the power-scaled fluid volume over the elevation of the core. Four model RVVVs were used to simulate eight plant valves. The MIST RVVVs could be controlled individually or in unison. Individual controllers provided automatic actuation of the valves on the upper plenum to downcomer-quadrant pressure differences. The MIST RVVVs thus approximated the head-flow response of the plant valves.[3] However, partially open operation was not possible in MIST; therefore, the detailed valve dynamics of the plant swing check valves were absent.

The MIST pressurizer was power-to-volume scaled. It contained heaters, spray, and a PORV. The lower pressurizer elevations were prototypical, as were those of the surge line. The model pressurizer height was reduced from that of the plant to increase its diameter, thus lessening atypical fluid stratification and the likelihood of spray impinging on the vessel wall.

One core flood tank was used in MIST. This tank was power-to-volume scaled to represent the two plant tanks. The model tank was installed vertically, with the bottom of the tank at a prototypical elevation. The injection line from the tank to the nozzles on the downcomer was sized to preserve plant-typical irrecoverable losses, and the nozzles were sized to maintain the plant ratio of (core-flood) injected fluid momentum to the downcomer fluid momentum.

2.3. Boundary Systems

The MIST boundary systems were sized to power-scale the plant boundary conditions. HPI and auxiliary feedwater (AFW) head-flow characteristics were based on composite plant characteristics. Model vents were included in both hot legs and in the reactor vessel upper head. Controlled leaks were located in the cold leg suction and discharge piping and at the upper and lower elevations of steam generator B (for tube rupture simulation). The desired vent and leak flow rates were obtained using critical flow orifices of power-scaled areas.

A steam generator tube rupture was simulated by opening a flow circuit across either the upper or the lower tubesheet of steam generator B. This circuit is shown in Figure 2.5. It consisted of a flow control orifice, isolation valves, and measurements of fluid temperature and differential pressure. The tube rupture simulation flow circuit did not preserve the complex flow path geometry of an actual tube rupture.

2.4. Heat Losses and Guard Heaters

MIST was designed to minimize heat losses from the reactor coolant system. Fin effects (instrument penetrations through the insulation) were minimized by using 1/4-inch penetrations for most of the instrumentation. Heat losses due to conduction through component supports were minimized by designing the supports to reduce the cross-sectional area and by placing ceramic blocks between load-bearing surfaces. The reactor coolant system piping and components were covered with passive insulation, guard heaters, and a sealed outer jacket (to prevent chimney effects). The insulation arrangement is illustrated in Figure 2.6. The guard heaters were divided into 42 zones, each controlled by a zonal temperature difference and a pipe metal temperature. This system provided differential temperature control as a function of temperature. A detailed finite-difference analysis of the insulation system indicated that heat loss was strongly dependent on metal temperature and weakly related to fluid state. The control temperature difference required to minimize heat losses was determined experimentally at several loop temperatures.

However, the guard heaters did not compensate for all the loop heat losses. For example, large local losses at the gamma densitometers and viewports were not compensated. Had these local losses been compensated, the requisite increased metal temperatures would have generated atypically large metal stored energies as well as undesirable local effects. The total MIST primary system heat loss at 650F was approximately 18 kW or 0.55% of scaled full power. The post-trip core power commonly simulated in MIST ranged from 3.5 down to 1% of scaled full power; the uncompensated heat losses of 0.55% of scaled full power thus represented from 16 to 55% of these post-trip power levels. Core power was increased to offset these uncompensated heat losses.

2.5. Instrumentation

The MIST instrumentation was selected and distributed based on the input from experimenters and code analysts. This instrument selection process considered the needs of code benchmarking, indications of thermal-hydraulic phenomena, and system closure.

The approximately 850 MIST instruments were interfaced to a computer-controlled, high-speed, data acquisition system. MIST instrumentation consisted of measurements of temperature, pressure, and differential pressure. Fluid level and phase indications were provided by optical viewports, conductivity probes. differential pressures, and gamma densitometers. Mass flow measurements at the system boundaries were made using Coriolis flowmeters and weigh scales. Mass flow rate measurements in the loop were performed with venturis or turbines. Tables 2.1 and 2.2 provide a summary of the MIST instrumentation by component and instrument type.

The largest grouping of instrumentation was in the two steam generators. About 250, or 30%, of the instruments were located in these two components. The steam generator instrumentation provided for the measurement of fluid temperature, metal and differential temperature, total guard heater power, differential pressure, gauge pressure, and conductivity (for void determination). The allocation of instruments to the steam generators resulted from the judgement that observations of AFW wetting effects and steam generator heat transfer were of major importance. Several other local and multidimensional phenomena were also of considerable interest: noncondensible gas blanketing of primary tubes, intermittent radial advancement of condensation fronts in the region of the AFW nozzle, and boiler-condenser heat transfer in the region of the secondary pool.

The core and RVVV instrumentation measured fluid temperature, metal and differential temperature, total guard heater and core power, conductivity (for void determination), and gauge and differential pressures. The core instrument distribution concentrated on the axially varying parameters. A flat, radial heat flux profile was used in the core, and radial maldistribution of inlet flow was expected to result in only minor variations of enthalpy. Therefore, the majority of the incore temperature instrumentation was located in a single, interior flow channel. Radial temperature variations at the core outlet were recorded, but with a limited number of instruments. The core instrument allocation provided core heat input, inlet and exit fluid properties, and fluid gradients within the reactor vessel. In addition, collapsed levels and regional void fractions were available. The vent valve mass flow rates were obtained by synthesizing RVVV differential pressures, valve positions, and indications of fluid state.

Downcomer instruments measured fluid temperature, metal and differential temperature, total guard heater power, and differential pressures. Forty fluid thermocouples were concentrated in the upper downcomer, detailing mixing information for the RVVV, core flood, and cold leg fluid streams. Six additional fluid thermocouples were spaced uniformly in the lower downcomer to indicate the extent of mixing as the fluid left the upper downcomer. The downcomer flow rate was measured using a venturi and a cooled thermocouple probe.

Table 2.1 MIST Instrumentation by Component

Component	Number of Instruments
Cold legs	164
Core flood	7
Hot legs	121
Pressurizer	25
Primary boundary systems	72
Reactor vessel and core	169
Steam generators	249
Steam generator feedwater and steam circuit	44
TOTAL	851

Table 2.2 MIST Instrumentation by Measurement Type

Measurement Type	Number of Instruments
Conductivity probes	36
Cooled thermocouple	12
Differential pressure	133
Differential temperature	42
Fluid temperature	381
Gamma densitometer	12
Limit switches	79
Mass flow	9
Metal temperature	69
Miscellaneous	17
Power	48
Pressure	9
Volumetric flow	4
TOTAL	851

Cold leg instrumentation provided fluid and metal temperatures, differential temperatures, total guard heater power, and differential pressures. Gamma densitometers indicated pump suction fluid density. Cold leg flow rates were measured using venturis located in the suction piping of each cold leg. For tests requiring full forced flow, turbine meters were used in place of the venturis. In addition, the reactor coolant pump power, speed, and head rise were measured. Thermocouple rakes were installed in the cold legs, upstream and downstream of the HPI injection points, to indicate thermal stratification and counterflow near the junctions of the cold legs and downcomer.

Hot leg instrumentation measured fluid and metal temperatures, differential temperatures, total guard heater power, and differential pressures. Void measurements using gamma densitometers and conductivity probes were also made. In addition, viewports provided visual data to assess the local flow regimes. The placement of the hot leg instruments provided detailed fluid temperature gradients, local void fractions, and overall collapsed level. A conductivity probe, combined with local differential pressures in the hot leg U-bend region, provided additional information regarding loop refill and spillover. Gamma densitometers in the hot leg horizontals, downstream of the reactor vessel outlet nozzles, and viewports at the 29-foot elevation and at the U-bend high points, provided information regarding fluid state and flow conditions. Viewports in the hot leg horizontals near the densitometers probed the developing flow regimes upstream of the hot leg risers.

The boundary systems, which included HPI, leaks, vents, and gas addition, were provided with fluid thermocouples, absolute and differential pressure transmitters, mass flowmeters, and weigh scales. These instruments provided mass and energy closure for the facility. Additional information regarding the design and instrumentation of MIST may be found in the Facility Specification[2] and in the Instrument Report.[4]

2.6. Conversion Factors

The key MIST conversion factors are listed below.

Power: 1% of scaled full power (2700. mW)

= 33. kW = 31.3 Btu/s

Primary Flow Rate (Total Primary System):

1% of scaled full flow (135. x 10^6 lbm/h)

= 0.46 lbm/s = 1660. lbm/h = 0.21 kg/s

Secondary Flow Rate (Total Secondary System, i.e., 2 steam generators)

1% of scaled full flow (11.3 x 10^6 lbm/h)

= 0.0384 lbm/s = 138. lbm/h = 0.0174 kg/s

MIST piping was larger than power-to-volume scaled in consideration of two-phase phenomena and hydraulic losses. Whereas the plant-to-MIST power scaling factor was 817, the corresponding volume scaling factor was 620 for the total primary system volume (CFT excluded), and 600 for the primary system excluding the pressurizer.

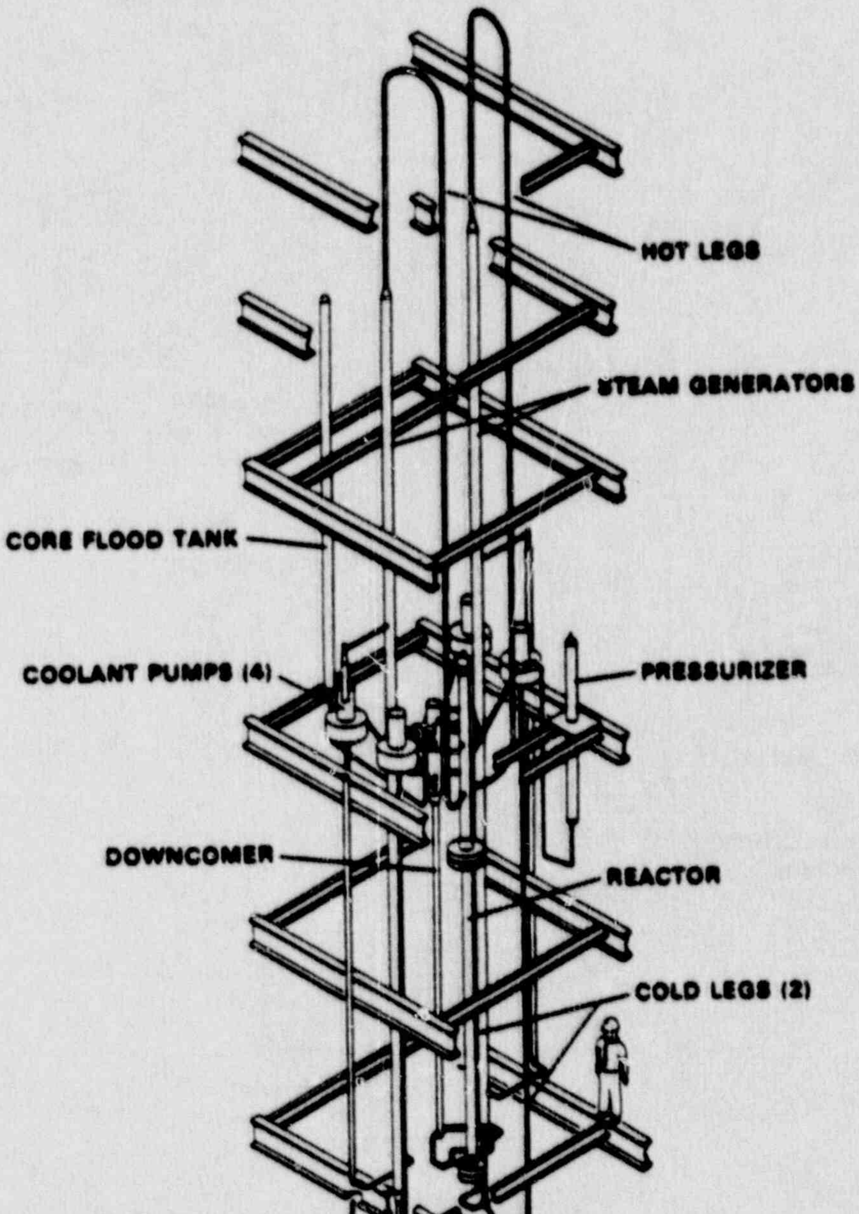

Figure 2.1. Reactor Coolant System -- MIST

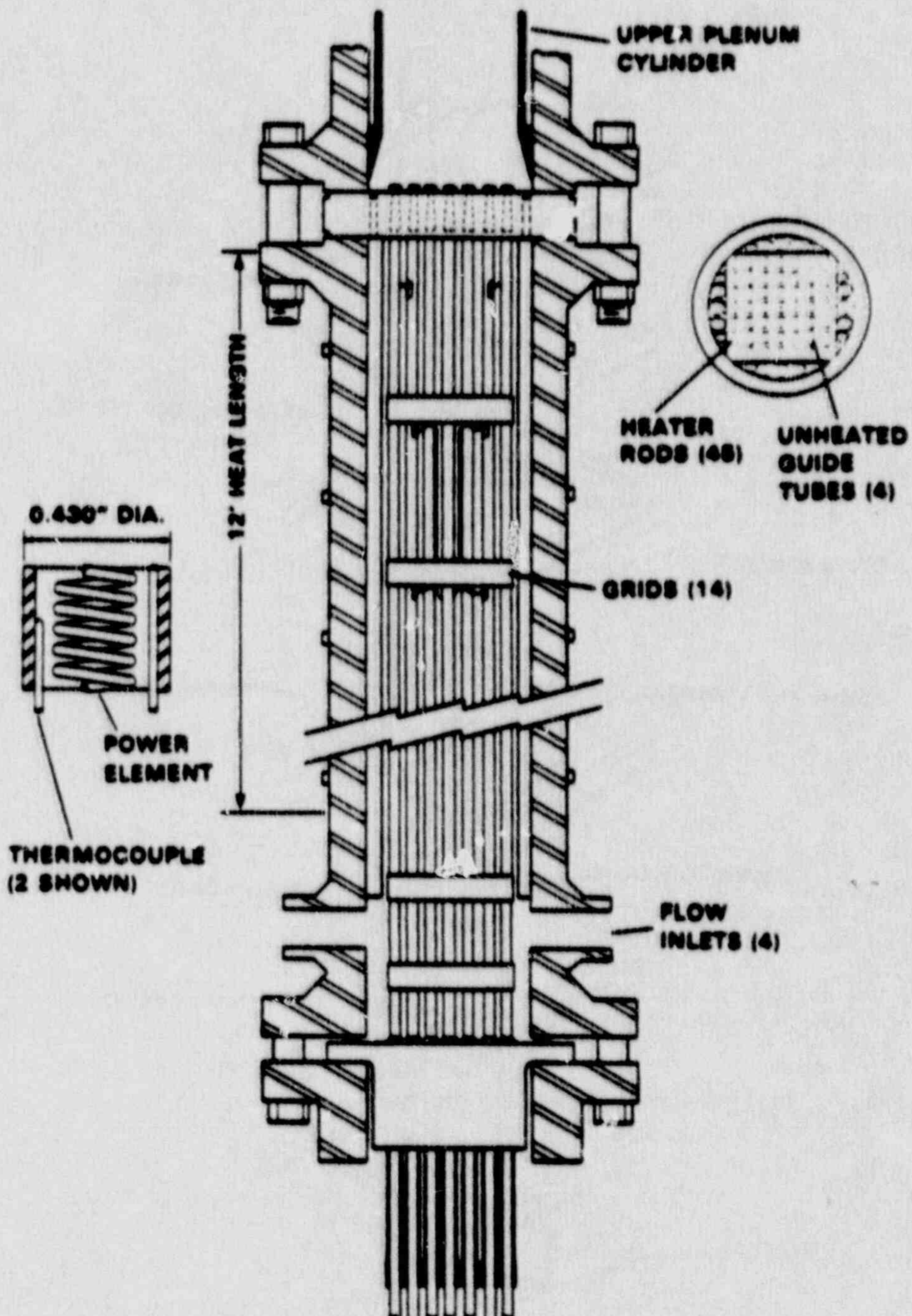

Figure 2.2. MIST Core Arrangement

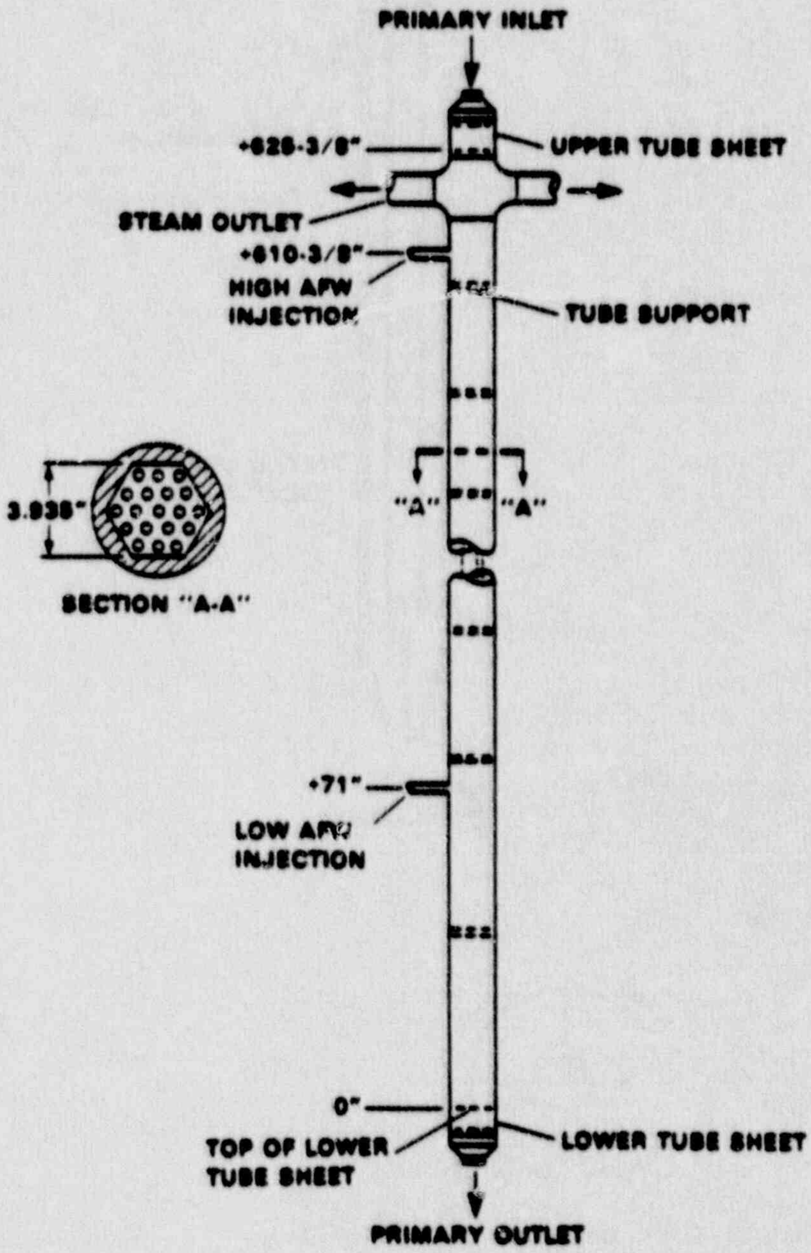

Figure 2.3. Nineteen-Tube, Once-Through Steam Generator

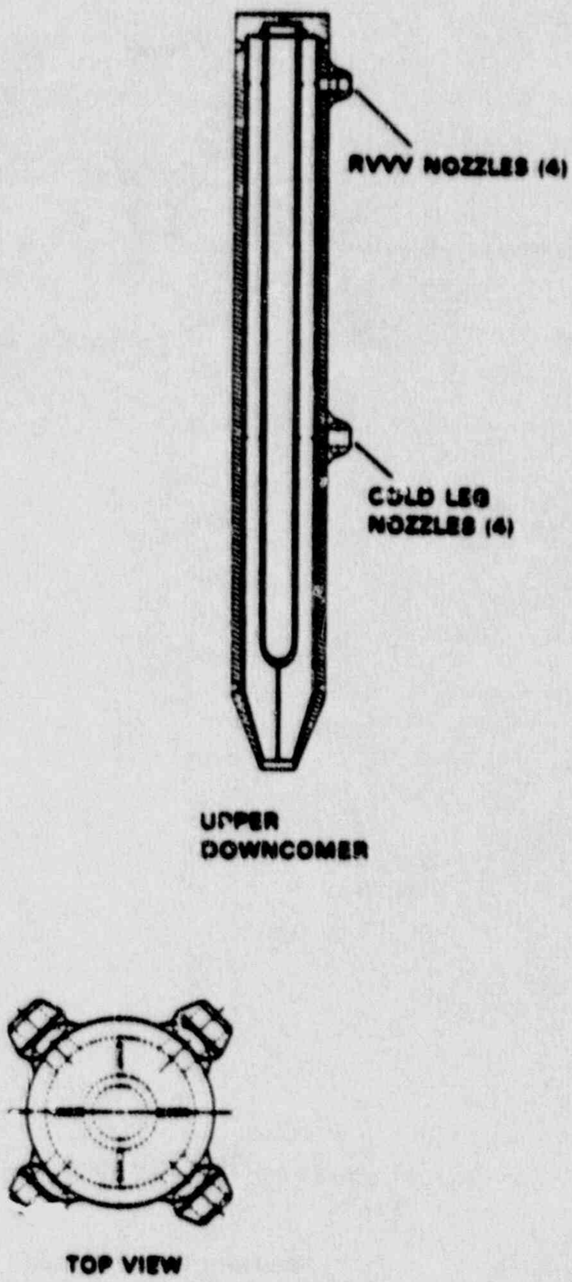

Figure 2.4. Upper Downcomer Arrangement

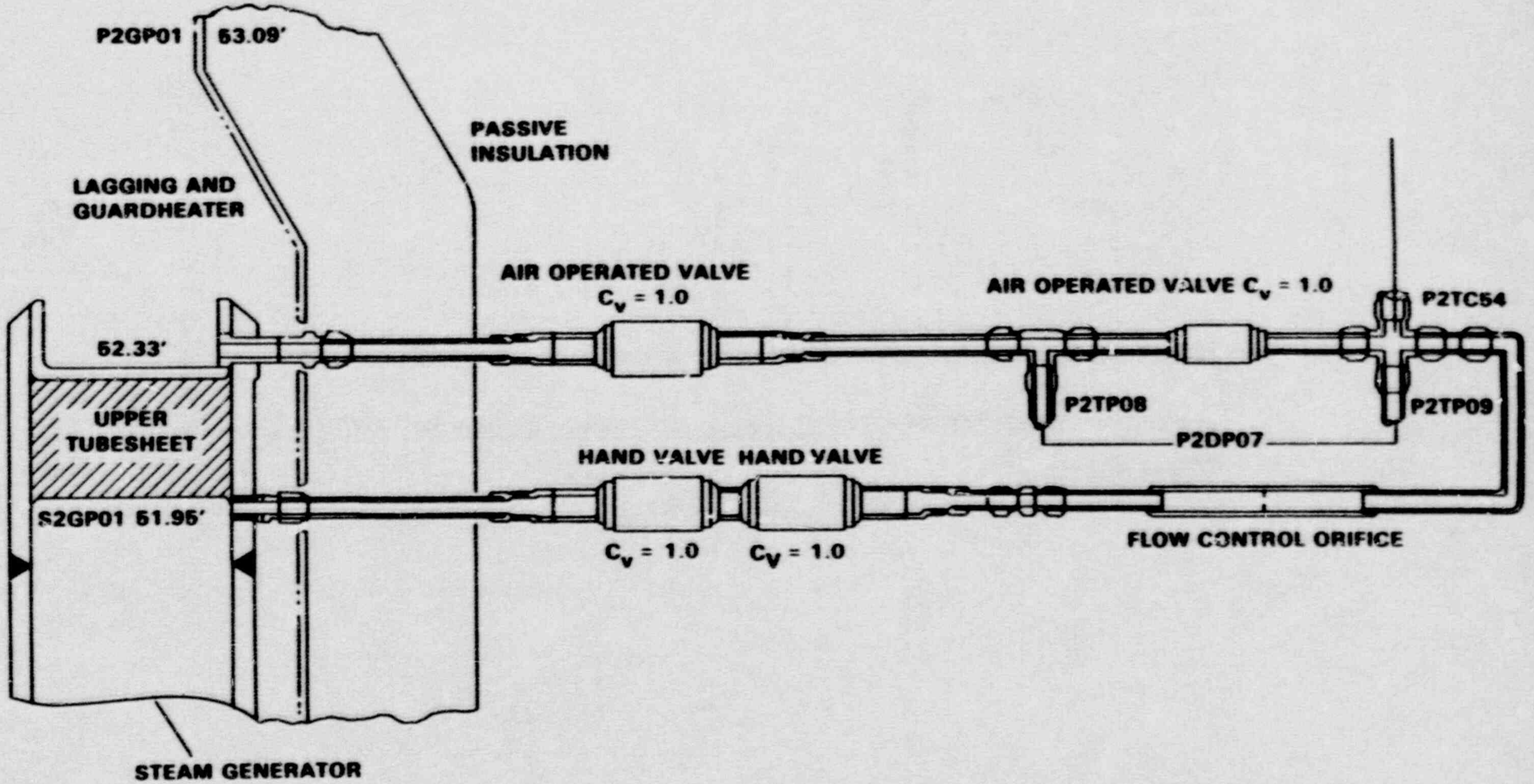

Figure 2.5 Primary-to-Secondary Tube Leak at Upper Tubesheet (Similar Arrangement at Lower Tubesheet)

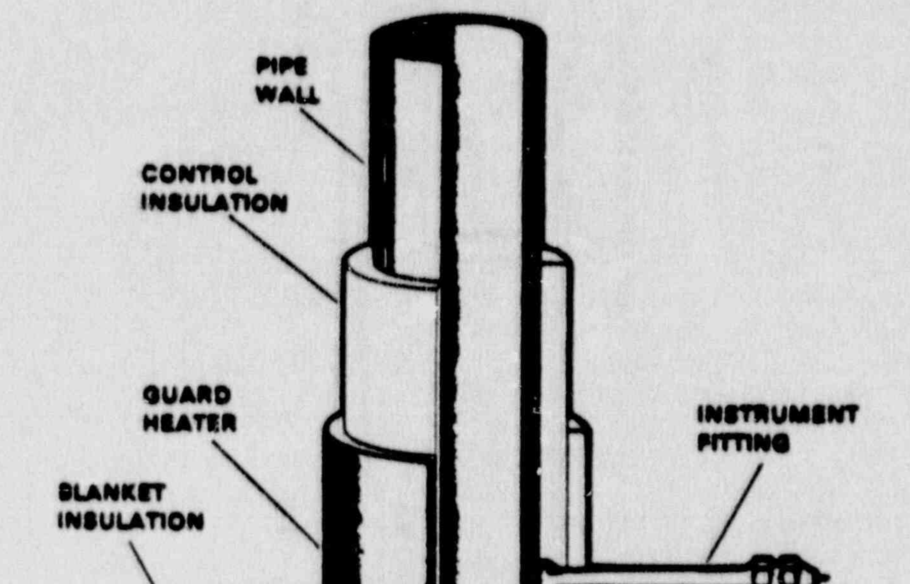

Figure 2.6 MIST Insulation Arrangement

3. TEST SPECIFICATIONS

The specifications herein have been excerpted from the MIST Test Specifications.[5] These specifications were formulated before the tests were conducted, as is reflected by their tense.

3.1. Introduction

Group 34 imposes a series of simulated tube ruptures. The 7 tests of this group are listed in Table 3.1. These tests are more rapid than the usual MIST transients and require a higher degree of test loop operator control and interaction. The operator procedures of these tests are based on, and are simplifications of, the steam generator tube rupture (SGTR) procedures.[6]

The Nominal SGTR Test simulates the (double-ended) rupture of 10 tubes at the top of steam generator B. Both generators are to be kept operational throughout this test; the "affected" (ruptured) steam generator is not to be isolated. The rate of change of system conditions decreases when the primary system voids to the rupture elevation.

A single tube (single-ended) rupture is simulated in Test 2. The primary will not rapidly depressurize through this 1.5-cm^2 break. The operator is to perform a controlled single-loop cooldown and depressurization.

Test 3 simulates a low-elevation SGTR. The primary will void extensively before the rupture site is uncovered. Level control in steam generator B will be correspondingly more difficult than with an upper-elevation rupture.

The affected steam generator is to be isolated in Test 4. The initial portion of the transient should replicate the Nominal SGTR transient. Steam generator fill and idle-loop performance will be observed in this test.

A steam line break and a large SGTR are superimposed in Test 5. Cooldown and depressurization will be largely controlled only by system interactions. The unaffected steam generator (A) may become a heat source. Interloop conditions may differ greatly.

Test 6 is a repeat of Test 3, the low-elevation, double-ended rupture of 10 tubes. Test 7 is a single-tube rupture, as is Test 2. The pressurizer vent is to be used for depressurization in Test 7; also, several methods of primary system depressurization are to be tested late in the transient.

These SGTR Tests pose special challenges to the facility limits. If a test must be aborted, the following steps are suggested: Close the primary-to-secondary leak, deenergize the core, and open the PORV and all vents. The suggested order of testing, based on increasingly challenging conditions should be (test number) 2, 4, 1, 3, and 5.

3.2. Nominal SGTR, Test 1 (340100)

The Nominal SGTR Test simulates the (double-ended) rupture of 10 tubes high in the steam generator. The affected steam generator is not to be isolated.

3.2.1. Purpose

The Nominal SGTR Test simulates 10 tubes ruptured at the top of a steam generator. Both steam generators are to be kept active during the ensuing cooldown. As described below, the transient will be rapid and will provide system interactions that have not otherwise been encountered. These may include early break uncovery but preclude the Boiler-Condenser Mode (BCM) because of the high elevation of the break, coupling between the secondary level and the primary cooldown rate, and steam generator steaming to maintain level.

3.2.2. Description

The SGTR transient is dominated by two concerns: The primary system must be depressurized as quickly as possible to prevent the actuation of the plant main steam safety valve (MSSV), especially with a full or nearly full steam generator, and core cooling must be maintained. The first goal may dictate the use of the emergency cooldown rate (240F/h) until the primary depressurization has reached 950 psia. It also precipitates careful attention to primary pressure control when HPI-PORV cooling is invoked, i.e., the use of additional depressurization mechanisms such as primary vents and steam generator flow paths.

Core exit subcooling is to be watched closely to ensure core cooling. A minimal subcooling margin is maintained to limit the primary-to-secondary pressure difference, but an adequate margin must be maintained using the standard abnormal transient operating guideline (ATOG) techniques. The SGTR transient thus requires close operator control. In the case of a double-ended rupture of 10 steam generator tubes, the transient also develops rapidly -- the primary will initially saturate rapidly with the larger SGTRs.

Upon leak actuation, the primary-to-secondary flow rate in MIST will be about 2000 lbm/h, or the equivalent of 15% of the scaled full secondary mass flow rate. Within approximately 1 minute, the primary loop fluid will saturate and, had the steaming rate of steam generator B not been increased to maintain steam generator pressure, the steam generator B level will exceed 15 feet (3 times that of steam generator A). Full HPI is being injected, but within 2 to 3 minutes the primary level will descend to the rupture site; the break flow will abruptly decrease, from approximately 900 to 300 lbm/h, as steam begins to govern the break flow rate. The HPI flow rate of 500 lbm/h is sufficient to recover the break intermittently; thus, the break conditions are expected to alternate between liquid and vapor and the average break flow rate is governed by the HPI flow rate.

Just after break site uncovery, the unaffected steam generator (A) may obtain a BCM. The occurrence and persistence of this event depends on the interloop fluid density and condition differences and on the elevation to which the primary level of steam generator B descends. Because AFW is being injected into steam generator A to raise its level, the BCM may be relatively strong; it may bring the primary system pressure into the desired range in just a few minutes. If the BCM of steam generator A is not effective, however, primary depressurization will be primarily dependent on the cooling caused by HPI as it flows to the break location. Fluid (density and phase) segregation may render HPI cooling of the hot leg U-bends (HLUBs) relatively ineffective. Primary pressure would thus decrease only slowly below 1500 psia, which is the saturation pressure corresponding to the initial hot leg temperature and which is well above the target pressure of 950 psia.

At 950 ± 25 psia, the affected steam generator could be isolated without the likelihood of actuating an MSSV. The affected steam generator is not to be isolated in the Nominal SGTR Test, rather a cooldown with two steam generators is to be continued. Steam generator B may tend to overfill. Although the primary-to-secondary pressure difference has been mitigated and the remaining difference is only that required to maintain the (50F) primary subcooling, the decreased cooldown rate below ~500F primary system temperature provides less opportunity for steaming the affected steam generator. The plant primary system cooldown rate could have been 240F/h above 500F, but below this temperature it must be limited to 100F/h.

Procedurally, HPI may not be throttled until a subcooling of 50F is achieved, regardless of the cooldown rate. With an SGTR, PORV actuations are used to control excessive subcooling. The cooldown with two steam generators will be continued until the facility low-pressure limits are encountered.

3.2.3. Conduct

The initialization, start, control, terminations, measurements, and acceptance criteria of the Nominal SGTR Test are specified in the subsections that follow. Tables 3.2 through 3.5 summarize these specifications.

Initialization

The Nominal SGTR Test is initialized much like the MIST Nominal Test. These pre-test, steady-state conditions are listed in Table 3.2. The only change involves primary pressure and subcooling, and pressurizer level. Rather than specifying the initial subcooling (which obtained a primary pressure of approximately 1750 psia) the primary pressure is set to 2150 psia. This obtains an initial core exit subcooling of about 50 versus 22F in the preceding tests. With such a large subcooling and a small primary-to-secondary leak, there will be sufficient time to activate HPI and to control subcooling. With a larger break, such as that of the Nominal SGTR Test, the increased initial subcooling allows for the test-initiation sequence and may afford sufficient system interaction time to prevent the secondary level from overfilling.

The initial hot leg fluid temperature of ~595F governs the primary saturation pressure upon leak initiation, viz. 1500 psia. This pressure is well above the target for rapid primary depressurization, 950 psia, which is set below the lowest plant MSSV lift pressure. The primary depressurization from 1500 to 950 psia may prove difficult. Had the primary been initialized with forced flow, the initial hot leg temperature would have been close to the steam generator secondary saturation temperature, and the primary depressurization would have been accomplished much more readily. The somewhat elevated saturation pressure of MIST thus resembles that of a plant in stable natural circulation, and not in forced flow. A pumps-on SGTR Test is included in Group 36.

The 5-ft initial pressurizer level, about half full, obtains a pressurizer inventory similar to that of the plant and expands the time from test initiation to the loss of SCM.

The initial steam generator secondary level of the preceding tests, 5 ft, is retained. The steam generator level in the plant may have been raised to 50% on the Operate Range (20.7 ft) for natural circulation, or it may have been depleted during the post-trip transient. The 5-ft level does facilitate level control during initialization while providing a large level range for use after test initiation. Level control in the affected steam generator is the key to the SGTR. The level should be prevented from reaching the overfill elevation, which is simulated to be 33 ft in MIST.

<u>Initiation</u>

The (Nominal) SGTR Test initiation steps are listed in Table 5.3 and are discussed below. The test loop operator actions are grouped into three steps.

<u>Step 1</u>

Step 1 is started after at least 10 minutes of steady-state data have been recorded at the specified initial conditions. The timing of step 1 is otherwise arbitrary. The subsequent steps will follow in rapid succession, however, as discussed below. The five actions of step 1 are as follows: (1) open the scaled 30.8-cm^2 primary-to-secondary leak at the top of steam generator B; (2) enter the test initiation time into the log; (3) transfer

RVVV control to automatic/independent; (4) actuate the core power decay ramp; and (5) verify that the pressurizer heaters trip on low pressurizer level. The 30.8-cm^2 leak flow area represents the double-ended flow path of 10 steam generator tubes. The tube rupture flow path is simulated by piping (containing a scaled flow orifice) that leads from the steam generator B primary inlet plenum to the tube region just below the upper tubesheet.

Step 2

Step 2 is to be started as the subcooling margin is lost, i.e., as the core exit subcooling decreases below 50F. Because the initial subcooling was only slightly higher than 50F, step 2 is to be performed almost immediately after step 1. The first two actions of step 2 simulate the response to a loss of subcooling margin, viz. (1) actuate HPI and (2) raise the steam generator A secondary level control point to 31.6 ft (which is equivalent to 95% on the Operate Range).

HPI is not to be throttled automatically based on core exit subcooling. The steam generator A control level increase simulates that the ruptured ("affected") steam generator has been identified, i.e., the steam generator refill to promote coupling is not applied to the affected steam generator. (Had the steam generator B control level been reset to 95% and then reduced shortly thereafter, there would have been little net effect.)

The final action of step 2 institutes a cooldown of 100F/h for both steam generator secondaries. This is a continuous rather than a T-based cooldown. An emergency primary system cooldown rate of 240F/h may be used in the plant when there is the likelihood of exceeding the governing criteria. These criteria are the Tube Rupture Alternate Control Criteria (TRACC), which are given in the SGTR guidelines.[6] The plant emergency cooldown rate of 240F/h may be used at primary average temperatures above 500F, but below 500F the plant cooldown rate must be limited to 100F/h. The MIST secondary cooldown rate of 100F/h will obtain MIST primary system cooldown rates that approximate those specified for the plant. The initial MIST primary depressurization will be governed by the relatively large primary-to-secondary break flow; the initial MIST cooldown may thus exceed 240F/h but will subside as the primary pressure approaches that of the steam generator secondaries. The cooldown of 100F/h for the MIST steam generator secondaries will thus limit the long-term primary cooldown rate.

Step 3

Initiation step 3 may be triggered early in the transient (its timing depends on the excess of the flow rate through the ruptured tubes to the steam generator B steaming rate). A steam generator level of 33 feet is used to simulate the level at which filling of the steam lines will begin. (At this elevation in the plant, the clearance around the pressure tap for level permits flow from the tube region to the steam outlet region.) The MIST steam generator level thus should not reach 33 ft before the primary is depressurized below 950 psia. Therefore, a level of 20 feet (and rising) is used to signal that extra measures must be taken to prevent MIST steam generator overfill; at this level in steam generator B, manually open the MIST steam generator B (low flow rate) steam pressure control valve. The capacity of the MIST low steam flow rate system approximates the scaled capacity of the plant turbine bypass valves.

Caution: If the steam generator B secondary pressure exceeds 1300 psia, actuate the MIST steam generator B high flow rate steam pressure control valve to regain pressure control (at ~1200 psia) while simulating plant safety valve actuation. If the secondary pressure exceeds 1375 psia, abort the test and take the appropriate actions to preclude lifting the MIST secondary code safeties.

Control During Testing

Control during testing is specified to approximate the plant SGTR Guidelines. The MIST actions are listed in Table 3.4.

Steam Generator B Level Control (Item 1 of Table 3.4)

If the level of steam generator B is at or above 20 ft, then open the steam generator B (low flow rate) steam pressure control valve to reduce the level to 10 ft. When the steam generator B level reaches 10 ft, revert to automatic (cooldown) control of the B steam valve.

Minimize the SCM (Item 2 of Table 3.4)

If the SCM reaches 75F, then depressurize the primary system to reduce the SCM to 50F. Manually open the PORV. If the SCM does not begin to decrease, also open all the primary vents. When the SCM has been reduced to 50F, close all these flow paths.

Isolate the CFT (Item 3 of Table 3.4)

If the SCM is being maintained and the primary has been depressurized below 715 psia, then isolate the CFT. (The MIST CFT isolation valve automatically closes when the primary pressure exceeds 650 psia and vice versa, but this has no impact on the CFT isolation step being addressed.) The SCM is "being maintained" when at least 50F subcooling has been held for one-half hour or more.

Control Pressurizer Bubble (Item 4 of Table 3.4)

Pressurizer bubble control, like CFT isolation, requires steam generator (versus HPI-PORV) cooling and an SCM of 50F for one-half hour. With these conditions, and when the pressurizer level exceeds 2 ft, manually throttle HPI to maintain the pressurizer level between 2 to 3 ft. In this instance, also energize the pressurizer (main) heaters as required to maintain an SCM of 60 to 70F.

Termination (Item 5 of Table 3.4)

Continue testing for at least 6 hours but not more than 10 hours. Within these time limits, the test may be terminated at the discretion of the Test Engineer when either of two criteria is met: (1) the primary system has been depressurized below 400 psia and at least 50F subcooling has been maintained for 2 hours or more, or (2) the primary system pressure has remained below 200 psia for 2 hours. If neither criteria is met, terminate testing after 10 hours, again at the discretion of the Test Engineer. (The maximum duration of all other SGTR Tests except Test 2 is 8 hours; for these tests, continue testing for at least 4 hours, but not more than 8 hours.) Enter the test termination time in the log and refill the loop expeditiously while continuing to record data.

Measurements and Acceptance Criteria

The measurements and data acquisition frequencies specified in Appendix F of the MIST Test Specifications[5] apply. Also obtain secondary system mass closure, within the limits of the available instruments and methods, by determining the secondary system fluid mass at the start and end of testing. The standard acceptance criteria apply, except for control during testing. Because of the complexity of the SGTR actions, a test is acceptable provided that the general control specifications are adhered to and the actions performed are sufficiently annotated to permit code modeling.

3.3. One-Tube Rupture, Test 2 (340213)

Test 2 simulates the rupture of 1 tube at the top of steam generator B. The affected steam generator is isolated.

3.3.1. Purpose

The primary-to-secondary flow area of Test 2 is too small to rapidly depressurize the primary system. A single-loop cooldown is to be performed. Periodic PORV actuations are used to minimize the primary-to-secondary pressure difference while maintaining a sufficient subcooling margin.

3.3.2. Description

The relatively small primary-to-secondary leakage path of Test 2 will not cause a rapid primary depressurization and will thus require a controlled primary cooldown and depressurization. The normal system cooldown rate of 100F/h is to be used. Both steam generators are to be used to affect the initial cooldown; intermittent PORV actuations are to be used to control primary subcooling.

Leak actuation will obtain an initial primary-to-secondary flow rate of ~100 lbm/h. The concurrent actuation of full HPI should quickly overcome the leak flow rate as the primary depressurizes. HPI is then to be throttled to maintain pressurizer level. The refill of the steam generator secondaries and the secondary depressurizations will cool the primary and allow the core exit subcooling to be maintained. When the subcooling reaches 75F, then the PORV is actuated to obtain a subcooling of 50F.

The affected steam generator (B) is to be isolated when the primary system has been depressurized below 950 psia. The ensuing single-loop cooldown will exacerbate inter-loop asymmetries. Steam generator B will become a heat source and impede the primary system cooldown. The loop B primary flow will stagnate but may reinitiate periodically. The HLUB fluid of B may retain sufficient energy to saturate as the primary system cooldown and depressurization progresses; the HLUB region of generator B may then void and resist further primary depressurization.

3.3.3. Conduct

The 1-tube rupture, Test 2, is to be initialized at the same conditions as the Nominal SGTR Test and as listed in Table 3.2. The initiation and conduct of Test 2 differ from that of the Nominal Tests and are described below. Test termination, measurements, and acceptance criteria parallel those of the Nominal SGTR Test. The maximum test duration is 10 hours.

The test initiation steps are listed in Table 3.6. These actions are to be performed after at least 10 minutes of steady-state data have been recorded at the specified initial conditions. The initial actions of step 1 parallel those of the Nominal Test except that a scaled 1.54-cm^2 leak rather than a 30 8-cm^2 leak is to be opened (at the top of steam generator B) and HPI is to be activated concurrent with the leak opening. The flow area of a steam generator tube is 1.54 cm^2; thus, the imposed leak size simulates the (single-ended) rupture of 1 steam generator tube. A single-ended rather than a double-ended rupture is simulated to ensure that the primary pressure must be reduced by suitable control steps, rather than being decreased solely by fluid discharge.

Full HPI is to be actuated and then throttled to maintain a pressurizer level of 2 to 3 ft. The steam generator level control setpoints are to be reset from 5 to 20.7 ft in both steam generators, which corresponds to 50% on the Operate Range; this is the plant steam generator control level for natural circulation.

The primary cooldown of 100F/h is enacted using the ATOG steam generator pressure control (for both steam generators).

Control during testing involves mainly HPI and the PORV. Continue to throttle HPI to maintain pressurizer level, and periodically actuate the PORV to reduce the core exit subcooling from 75 to 50F. Also, energize the pressurizer heaters when the SCM is less than 60F.

Should core cooling not be maintained, then control of the transient is to change drastically. The change is to be based on the core exit SCM: If the SCM decreases to 25F or less, then raise the steam generator level control setpoint to 31.6 feet and use full (unthrottled) HPI. When the SCM is returned to 50F or more, revert to throttled HPI and hold the current steam generator levels (for example, if an SCM of 50F is regained when the (average) steam generator level is 25 ft, reset the steam generator level controls to 25 ft). (For ease of test conduct, this may be accomplished by using constant level control at the lower setpoint, initially 20.7 ft, and transferring to band control with setpoints of 31.6 and 31.0 ft when appropriate.)

When the primary system has been depressurized below 950 psia, transfer from a two-loop cooldown to a single-loop cooldown. Isolate steam generator B by deactivating its cooldown control and closing its steam and feed isolation valves. Continue steaming and feeding the intact steam generator (A). Retain this asymmetric cooling mode for the duration of the test.

3.4. Low-Elevation SGTR, Test 3 (340302)

A 10-tube rupture is simulated at the bottom of steam generator B.

3.4.1. Purpose

Test 3 will provide data with a low-elevation tube rupture. As discussed previously, this transient will encounter extensive primary voiding, a sustained and asymmetric PCM, and subcritical leak flow.

3.4.2. Description

A 10-tube rupture is simulated at the bottom of steam generator B in Test 3, rather than at the top as in Test 1, the Nominal SGTR transient. This displacement of the break location will cause more extensive primary system voiding and will render steam generator B level control more difficult. Rather than uncovering the break location in a few minutes as in the Nominal Test, perhaps 10 minutes would be required in this test if critical flow were maintained. However, a sustained BCM will occur in steam generator A; thus, the primary may be depressurized to nearly the secondary pressure before break uncovery. Although the break fluid temperature may decrease with HPI cooling, the primary-to-steam generator B pressure difference should be insufficient to support critical flow. By this combined blowdown and BCM, the primary may achieve the target pressure of 950 psia faster than in the Nominal Test. Although the low elevation of the simulated rupture will maintain the steam generator primary levels near the lower tubesheet, HPI will continue to cool the core.

3.4.3. Conduct

The conduct of Test 3 is identical to that of Test 1 (Nominal SGTR) except that the tube rupture simulation at the bottom rather than the top of the steam generator B is to be opened during test initiation. The maximum test duration is 8 hours.

3.5. Isolated Steam Generator, Test 4 (340403)

The affected steam generator is to be isolated in Test 4.

3.5.1. Purpose

Both steam generators were kept active in the Nominal SGTR transient, Test 1. In Test 4, the affected steam generator is to be isolated at 950 psia primary system pressure. The effects of this isolation on tube-to-shell temperature difference and on the ensuing primary cooldown depressurization will be observed. The initial transient will also provide valuable insight into test reproducibility.

3.5.2. Description

The Isolated Steam Generator Test will duplicate the Nominal SGTR Test until steam generator isolation at 950 psia. At steam generator isolation, a primary subcooling margin of 50F, should it be maintained, will obtain a primary-to-secondary pressure difference that is sufficiently large to sustain critical flow through the break. The break flow rate will thus be limited only by the available HPI flow rate, which is approximately 600 lbm/h. That is, if the break flow rate exceeds the HPI flow rate, the break will uncover and the consequent change of state at the break will abruptly reduce the break flow rate. The level in the isolated steam generator will thus reach the overfill elevation of 33 ft in about 5 minutes and will fill completely in an additional 5 minutes. The pressure of steam generator B will then equalize to that of the primary; steam generator B may become a heat (and inventory) source upon subsequent primary depressurization. The primary will fill rapidly, although the inter-loop conditions may differ and a void may persist in the generator B HLUB.

The isolated steam generator shell temperatures may lag behind the system cooldown as the loop B flow stagnates. Further, loop B with its isolated steam generator may flow intermittently, thus perturbing the conditions in the rest of the system.

3.5.3. Conduct

Test 4 is to be performed exactly as the Nominal SGTR Test until the primary system has been depressurized to 950 $\pm$ 25 psia. Then, the affected steam generator (B) is to be isolated and allowed to fill. (Close the steam generator B steam pressure control valves and manually close its isolation valves.) Following the isolation of steam generator B, the cooldown of 100F/h is to be obtained using steam generator A. The usual PORV operation to control subcooling applies.

Test termination criteria, measurements, and acceptance criteria are the same as those of the Nominal Test except that the maximum test duration is 8 hours.

3.6. Steam Line Break with SGTR, Test 5 (340504)

An unisolatable steam line break and simultaneous 10-tube SGTR are simulated in Test 5.

3.6.1. Purpose

Test 5 superimposes an unisolatable steam line break (SLB) and an SGTR in steam generator B. Primary depressurization and system cooldown are then governed by break flow rates rather than being controlled by the operator. The unaffected steam generator may become a heat source in this relatively rapid transient.

3.6.2. Description

The simulated SLB will quickly depressurize steam generator B, with residual steam generator B secondary pressure sustained only by metal stored heat, the discharge from the primary through the leak, and heat transfer from the steam generator B primary. The primary system will rapidly depressurize to the saturation pressure (~1500 psia) corresponding to the initial hot leg temperature.

The subsequent events may resemble those of the Nominal SGTR transient, viz. diminished break flow limited by the HPI flow rate as the primary voids down to the break elevation. The mass flow rate in critical flow is sensitive to the fluid state at the break site. When there is critical flow through the 10-tube rupture, the critical flow in liquid exceeds the HPI flow rate (tending to uncover the break), but the critical flow in vapor is less than the HPI flow rate (tending to cover the break site).

The SLB flow may augment the primary system cooldown to the extent that the unaffected steam generator (A) becomes a heat source. Cooldown control below 500F, which is normally performed to limit the cooldown rate to 100F/h, will not be available. To the degree that HPI flowing to the break selectively cools the B cold legs and steam generator B, loop conditions will be unusually asymmetric.

3.6.3. Conduct

The steam line break with the SGTR Test is to be initialized at the same conditions as the other SGTR tests and as listed in Table 3.2. Test initiation is similar to that of the Nominal SGTR Test (Table 3.3) except that a steam generator B steam line break is to be simulated just before the primary-to-secondary leak is opened; the steam generator B secondary is to be depressurized at the facility limit, which is approximately 100 psi/min. This secondary depressurization affects some of the later actions, as indicated in Table 3.7.

Because the level of steam generator B is controlled by tube leakage and the flow rate from the simulated SLB, the usual initiation step 3 regarding steam generator B level control is obviated. The actions of step 1 are to be performed in rapid succession; step 2 is to be performed as the core exit fluid subcooling decreases below 50F and will follow step 1 closely. The remaining test conduct specifications parallel those of the Nominal SGTR Test, cf. Tables 3.4 and 3.5. The steam line break is simulated to be unisolatable, therefore steam generator B is not to be isolated in MIST. The maximum test duration is 8 hours.

3.7. Repeat Low-Elevation Tube Rupture, Test 6 (3406AA)

Test 6 is to be an exact repeat of the Low-Elevation Tube Rupture Test (Test 3). The description and conduct of Test 6 is identical to that of Test 3. Test 6 was scheduled to provide information regarding repeatability and to supplement Test 3 in which the tube rupture flow rate measurement was not available.

3.8. One Tube Rupture and Single-Loop Cooldown Without a PORV, Test 7 (340799)

3.8.1. Purpose

SGTR Test 7 repeats the 1 tube (single-ended) rupture of SGTR Test 2. In Test 7, the PORV is simulated to be unavailable and the pressurizer vent is used for primary system depressurization. The key feature of this test is the examination of the primary depressurization to the decay heat removal system setpoints (265 psia and 280F) with one loop inactive. Several primary depressurization techniques are to be tested.

3.8.2. Description

The test consists of the following three major phases:

1. SGTR, two-loop cooldown, and isolation of the affected steam generator.
2. Single-loop cooldown and depressurization.
3. Implementation of specific steps to augment the depressurization of the primary system.

The first phase, the two-loop cooldown, parallels that of SGTR Test 2. The major difference is that the pressurizer vent, rather than the PORV, is to be used for primary system depressurization. HPI is to be controlled to maintain pressurizer level. Should the SCM approach 20 or 100F however, then HPI control is to be directed toward stabilization of the SCM, rather than pressurizer level control. The affected steam generator (B) is to be isolated when both hot leg temperatures reach 525F or less; this is somewhat earlier in the test than was specified in Test 2. Also unlike Test 2, feed throttling is to be imposed in Test 7; the feed control of the emergency feedwater initiation and control (EFIC) is to be approximated by using about one-third capacity AFW, if required, and further by manually throttling AFW to maintain the actual steam generator secondary pressures no more than 100 psi below the desired pressures.

The single-loop cooldown is simply a continuation of the two-loop cooldown. Pressurizer venting and HPI control are to be maintained. The affected loop is expected to stagnate and void as the primary system continues to depressurize. The HPI flow rate adjustments are to be made smoothly, rather than abruptly, to avoid affecting such voiding.

The single-loop cooldown is to be continued until primary pressure control is lost, if such a situation occurs. In this circumstance, the gradual primary system cooldown continues but the primary system depressurization rate dwindles so that the core exit fluid SCM gradually increases (with an approximately constant HPI flow rate). At this point, several methods of primary system depressurization are to be tested. If MIST depressurizes more rapidly than expected, then the depressurization steps are to be instituted as MIST approaches the decay heat removal system setpoints.

The following primary system depressurization techniques are to be tested separately:

1. Raise the level in the unaffected steam generator.
2. Vent the idle loop.
3. Cycle HPI.
4. Reactivate the affected steam generator.

These steps are intended to augment the primary system depressurization down to the decay heat removal system setpoints. The cyclic HPI step may accomplish this by refilling the voided loop.

3.8.3. Conduct

Initiation and Cooldown

Initialize Test 7 at the same conditions as for the Nominal SGTR test, as listed in Table 3.2. There are two additional considerations in Test 7. The hot leg riser and U-bend uncompensated heat losses are to be minimized by passively insulating the 4 hot leg (riser and U-bend) viewports. Also, a simulated pressurizer vent (scaled 1-cm^2 area) is to be installed and ready for use. Initiate the test after recording at least 10 minutes of steady-state data at the specified initial conditions. The test-initiation actions are listed in Part 1 of Table 3.8. Control HPI as listed in Part 2 of Table 3.8. Use gradual HPI flow rate adjustments. Isolate the affected steam generator when the core exit fluid temperature reaches 525F. Isolate the CFT when the primary system is depressurized to 715 psia, provided that the SCM is at least 20F. Close the pressurizer vent when the SCM decreases to 25F or less, and reopen it when the SCM increases to 50F or more -- pressurizer vent actuations are expected to be required only once or twice throughout the test. (These control actions are listed in Part 2 of Table 3.8).

Depressurization Steps

Begin the depressurization steps (Part 4 of Table 3.8) after 5 hours of testing and using the criteria given in Part 3 of Table 3.8. The primary criteria involves the loss of primary system pressure control -- the cooldown continues but the primary depressurization virtually ceases so that the SCM slowly rises. The depressurization steps are listed in Part 4 of Table 3.8 and are self-explanatory. Each of the four steps is to be continued for 1 hour.

Termination

The test is to be terminated after the completion of the depressurization steps, and at the discretion of the Test Engineer. Upon termination, log the termination time and refill the loop expeditiously while continuing to record data.

Measurements and Acceptance Criteria

The measurements and data acquisition frequencies specified in Appendix F of the MIST Test Specifications[5] apply. Gradually decrease the data acquisition frequency as the test progresses. Revert to a relatively high frequency, about 15 seconds per scan, during the depressurization steps that occur near the end of the test. Because of its complexity, this test would be acceptable provided that the general control specifications are adhered to, and the actions taken are sufficiently annotated to permit code modelling of the system interactions.

Table 3.1 SGTR Tests (Group 34)

An RCP-running SGTR test is included in Test Group 36. In all tests, RCPs are not available, full HPI capacity is available, RVVV control is nominal, and no NCG is present. The nominal leak size (30.8 cm^2) simulates the double-ended flow path from 10 tubes; the reduced leak size (1.54 cm^2) represents a single flow path from 1 tube.

Test Number	Identifier	Description	Variable	Setting	Nominal Setting
1	340100	Nominal SGTR (cooldown using 2 steam generators)	(Nominal)	--	--
2	340213	One tube rupture (1-loop cooldown)	Leak size (cm^2) Status of affected steam generator	1.54 Isolated	 Uniso-lated
3	340302	Low-elevation SGTR	Leak location	Bottom	Top
4	340403	Steam generator isolated	Status of affected steam generator	Isolated	Uniso-lated
5	340504	Steam line break w/SGTR	Steam line condition	Broken	Intact
6	3406AA	Repeat low-elevation SGTR	Leak location	Bottom	Top
7	340799	One-tube rupture, pressurizer venting, depressurization methods	------------------See Text-----------------		

Table 3.2 Initial Conditions, SGTR Tests (Group 34)

Quantity	Specification	Maximum Deviation	Derived	Note
Primary				
The primary system is in subcooled natural circulation. The primary boundary systems are inactive, the guard heaters are in automatic control, the RVVVs are manually closed, the PORV is in automatic control on overpressure, and the CFT is pressurized to 600 psia and has been recirculated. (The MIST CFT is automatically isolated when primary pressure exceeds 650 psia).				
Core power, % of scaled full power (1% = 33 kW)	3.[illegible]0	0.05		*
Pressure, psia	2150	25		
T_{hot}, F			595	
T_{cold}, F			550	
Flow rate, % of scaled full flow (1% = 1660 lbm/h)		3.5		
Pressurizer level, ft (within pressurizer)	5.0	0.2		
Subcooling, F			51	
Secondaries				
The steam generators are balanced. AFW at 100 ± 20F is being injected at the high-elevation site using the minimum-wetting nozzle.				
Pressure, psia	1010	10		
Level, ft	5	1		
Flow rate, % of scaled full secondary flow (1% = 138 lbm/h)		2.5		

*Increase core power by 0.4% to offset losses to ambient.

Table 3.3 Initiation (Nominal SGTR)

Initiation steps 1 through 3 will follow in rapid succession. See text for a discussion of these steps.

Step 1

Perform step 1 after recording at least 10 minutes of steady-state data at the specified initial conditions.

1. Open the upper 30.8-cm^2 primary-to-secondary leak.

2. Enter the test initiation time into the log.

3. Transfer RVVV control to automatic/independent.

4. Actuate the core power decay ramp simulating post-trip decay (from 1 minute 40 seconds after reactor trip).

5. Verify that the pressurizer heaters trip on low pressurizer level.

Step 2

Perform step 2 as the core exit subcooling decreases to 50F. (The initial subcooling was only slightly greater than 50F.)

1. Actuate HPI (use full head-flow characteristics).

2. Reset the steam generator A secondary level control to 31.6 feet (use full AFW head-flow characteristics), leave the steam generator B level at 5 ft, and do not feed steam generator B.

3. Depressurize the steam generators to obtain a steam generator secondary cooldown rate of 100F/h. The primary system will initially cool near the emergency rate of 240F/h, as described in the test.

Step 3

Perform this step when the steam generator B secondary exceeds 20 ft as a result of the primary-to-secondary leakage. Manually open the MIST steam generator B (low flow rate) steam pressure control valve to halt the level increase. Do not feed steam generator B.

Table 3.4 Control During Testing -- Nominal SGTR

The table entries are as follows:

1. Steam generator B level control
2. SCM control
3. Core flood tank (CFT) isolation
4. Pressurizer bubble
5. Termination criteria

1. Steam Generator B Level Control

- Examine the steam generator B secondary level. If this level is less than 20 ft, no additional level control is needed.

- If the level is greater than or equal to 20 ft, then open the steam generator B (low flow rate) steam pressure control valve to (attempt to) regain a level of 10 ft.

- At 10 ft, revert to automatic steam valve control (to maintain the cooldown of 100F/h).

2. SCM Control

- Examine the SCM of the core exit fluid. For SCMs less than 75F, reduction is not needed.

- If the SCM is greater than 75F, depressurize the primary system to reduce the SCM. Manually open the PORV. If the SCM does not decrease, also open all the primary vents. Reduce the SCM to 50F, then close the PORV and vents.

3. CFT Isolation

- Examine the SCM. If an SCM of 50F or more has been maintained for at least one-half hour, and if the primary system pressure is less than 715 psia, then isolate the CFT.

4. Pressurizer Bubble

- Examine the SCM (as in Item 3). If an SCM of 50F or more has been maintained for at least one-half hour, then pressurizer bubble control is appropriate. In this case, when the pressurizer level reaches or exceeds 2 ft within the pressurizer, manually throttle HPI to maintain 2 to 3 ft. Also energize the pressurizer heaters when the SCM is less than 60F, then deenergize them when an SCM of 70F is attained.

Table 3.4 Control During Testing -- Nominal SGTR (Cont'd)

5. Termination Criteria

- Continue testing for at least 6 hours.

- Between 5 and 10 hours, terminate testing at the discretion of the Test Engineer when either of two criteria is met: (1) 50F or more subcooling has been maintained for at least 2 hours and the primary system pressure is less than 400 psia, or (2) the primary system pressure has remained below 200 psia for at least 2 hours.

- Terminate testing (at the discretion of the Test Engineer) at 10 hours.

- For tests other than Test 1 (Nominal SGTR), Test 2 (One-Tube Rupture), and Test 7, use 4 and 8 hours rather than 6 and 10 hours, i.e., limit the tests to 8 hours.

Table 3.5 Control During Testing -- Functions

Control Function	Control Mode	Reference Table 3.4 (Item No.)
Pressurizer main heaters	Generally deenergized. May be used to maintain the pressurizer bubble.	4
RVVV	Automatic/independent, with actuation setpoints of 0.125 and 0.04 psi.	-
Core power	Automatic decay simulation.	-
HPI	Full head-flow characteristics. Manual throttling for pressurizer level when subcooling maintained.	4
Steam generator level	Steam generator A: automatic control at 31.6 ft (95% operate range). Steam generator B (affected): see text.	1
Steam generator pressure	See text. Note: If the steam generator B secondary pressure exceeds 1300 psia, actuate the MIST steam generator high flow rate steam pressure control valve to regain pressure control while simulating plant safety valve actuation. If the steam generator B pressure exceeds 1375 psia, abort the test and take the appropriate actions to preclude lifting the code steam generator safeties.	-
PORV	Manually open for excess subcooling, see text.	2
Vents (HLHPVs and RVUHV)	Open for excess subcooling, see text.	2
CFT	Automatic isolation on low level. Manual isolation when the primary system depressurizes below 715 psia provided that at least 50F subcooling has been maintained for one-half hour.	3

Table 3.6 Initiation of Test 2, One Tube Ruptured

Perform these test-initiation actions after recording at least 10 minutes of steady-state data at the specified initial conditions:

1. Open the 1.5-cm^2 primary-to-secondary leak at the top of steam generator B.

2. Enter the test initiation time in the log.

3. Actuate full HPI; throttle HPI to maintain pressurizer level.

4. Transfer RVVV control to automatic/independent.

5. Actuate the core power decay ramp (from 1 minute 40 seconds)

6. Reset the steam generator secondary level control setpoints to 20.7 ft. Use full AFW head-flow characteristics during the steam generator refill.

7. Activate the ATOG secondary pressure control (cooldown of 100F/h with a temperature difference of 50F between core exit and the saturation temperature corresponding to the steam generator control pressure).

Table 3.7 Initiation of the Steam Line Break with SGTR, Test 5 (340504)

Step 1

1. Depressurize the steam generator B at the facility limit (~100 psi/min) using the steam pressure control valve(s).

2. Open the 30.8-cm^2 primary-to-secondary leak at the top of steam generator B.

3. Enter the test-initiation time into the log.

4. Transfer RVVV control to automatic/independent.

5. Actuate the core power decay ramp (from 1 minute 40 seconds).

6. Verify that the pressurizer heaters trip on low pressurizer level.

Step 2

1. Actuate HPI using full head-flow characteristics.

2. Reset the steam generator A level to 31.6 feet. (Do not feed steam generator B.)

3. Depressurize the steam generators to obtain a steam generator cooldown rate of 100F/h (only steam generator A will be affected).

Table 3.8 Control Actions for Test 7 (340799)

1. Initiation

a. Open the scaled 1.5-cm^2 primary-to-secondary leak (simulating the single-ended rupture of 1 tube) at the top of steam generator B.

b. Log the time of test initiation.

c. Activate HPI. Control the HPI flow rate to maintain a level of 5 ± 2 ft in the pressurizer.

d. Transfer RVVV control to automatic/independent.

e. Activate the core power reduction ramp (simulating post-trip decay from 1 minute 40 seconds).

f. Refill the steam generator secondaries to 20.7 ft. Use full AFW capacity, but manually throttle AFW as required to limit the depressurization of the steam generator secondaries to not more than approximately 100 psi below the desired secondary pressure.

g. Activate the ATOG pressure control of both steam generator secondaries.

h. Deenergize the pressurizer fluid ("main") heaters.

i. Open the pressurizer vent when the SCM exceeds 50F.

2. Control During the Cooldown

2.1. HPI Control

a. Control HPI to maintain a level of 5 ± 2 ft in the pressurizer, provided that the SCM remains between 25 and 95F. Adjust the HPI flow rate smoothly rather than abruptly.

b. If the SCM decreases to 25F, gradually increase the HPI flow rate to stabilize the SCM. Use full HPI capacity for SCMs of 20F or less. Should the SCM subsequently increase above 25F, gradually revert to HPI throttling for pressurizer level.

c. If the SCM increases above 95F, gradually reduce the HPI flow rate to stabilize the SCM. Interrupt HPI if the SCM reaches 100F or more. Should the SCM subsequently decrease below 95F, gradually revert to HPI throttling for pressurizer level control.

2.2. Steam Generator Isolation at 525F

When the core exit fluid temperature has decreased to 525 ± 5F (or less), isolate the affected steam generator (steam generator B). Deactivate its cooldown control, and close its steam and feed isolation valves.

Table 3.8 Control Actions for Test 7 (340799) (Cont'd)

2.3. CFT Isolation

Isolate the CFT when the primary system has been depressurized to 715 psia or less, provided that the SCM is at least 20F.

2.4. Control of the Pressurizer Vent

Close the pressurizer vent when the SCM decreases to 25F or less. Reopen the vent when the SCM increases to 50F or more.

3. Transition to Depressurization Sequences

Begin the depressurization sequences (listed in step 4) after at least 5 hours of testing, when any one of the following criteria is satisfied:

a. The primary system has been depressurized to 300 psia or less and the core exit fluid has been cooled to 300F or less.

b. The SCM gradually increases by 5F or more, while the HPI flow rate has remained approximately constant.

c. After 15 hours of testing.

4. Depressurization Sequence

Each of the following four primary depressurization steps are to be exercised for 1 hour of testing.

a. Steam Generator A Level

Raise the secondary level in the unaffected steam generator (steam generator A) to 31.6 ft using full, unthrottled AFW.

b. Venting

Open the hot leg B high-point vent (retain the 31.6 ft level in steam generator A). Close the hot leg B vent after 1 hour of venting.

c. Cyclic HPI

Stop HPI. When the SCM has decreased to 20F or after 1/2 hour, whichever occurs first, activate full HPI. Raise the SCM to 100F, then interrupt HPI and repeat the sequence. (Leave the pressurizer vent open during this cyclic-HPI evolution.) Continue this cyclic HPI control for 1 hour, then revert to the previous methods of HPI control (for pressurizer level and SCM).

Table 3.8 Control Actions for Test 7 (340799) (Cont'd)

d. Steam the Affected Steam Generator

Reactivate the isolated steam generator (steam generator B). Attempt to reduce its pressure to obtain a cooldown of 100F/h using the (MIST) low flow rate steaming circuit. As the steam generator B secondary level decreases by steaming, activate AFW to steam generator B to maintain a level of 31.6 ft. Continue the steaming of steam generator B for 1 hour, then terminate the test.

4. PERFORMANCE

The acceptability of each test was determined by examining both the conduct of the test and the performance of the measurement systems. The acceptance criteria for each test were defined in the corresponding test procedure, which was based on the MIST Test Specifications.[5] Any condition, action, or measurement that did not meet the acceptance criteria was evaluated for its impact on test acceptability. The tests reported herein are only those that were determined to be acceptable. Any specific deviations of these tests from the acceptance criteria are described in this section.

The review of test conduct included the following checks for each test:

- System conditions and stability just prior to test initiation
- Sequence and timing of the test initiation actions
- Performance of the manual and automatic control functions
- Test termination criteria and the sequence of actions

The impact of out-of-specification conditions or actions was assessed. The deviations of those tests that were determined to be acceptable are described in section 4.1.

The following pretest and post-test data qualification checks were performed for each test:

- The acquisition of the critical measurements
- The operation of the measurement systems within their calibrated range
- The acquisition of instrument readings within their expected range of operation
- Self-consistent measurements, considering both comparable measurements and derived quantities

The appropriate measurement uncertainties were used to assess the individual measurements. The impact of the individual out-of-specification conditions was assessed. The deviations of the critical measurements of those tests that were determined to be acceptable are noted in section 4.2.

4.1. Conduct

The anomalies in the automatic operation of the control systems or in the manual interaction during the tests are described below.

4.1.1. Initial Conditions

Initial conditions for the tests were defined by the governing test procedure, ARC-TP-734, and are repeated with the actual values from each test in Table 4.1. All initial conditions were met except for the pressurizer surge line fluid temperature in Test 340100, core flood tank pressure in Test 340302, and pressurizer collapsed liquid level in Test 340504. These values are underlined in Table 4.1 to aid in their identification. These deviations did not impact test acceptability.

4.1.2. Test Initiation

The test initiation actions were to be performed as follows in less than 20 seconds:

1. Initiate the depressurization of 100 psi/min in steam generator B (Test 340504 only).
2. Actuate the simulated tube rupture.
3. Activate full capacity HPI (Tests 340213 and 340799 only).
4. Switch the control of the reactor vessel vent valves from manual to the automatic "independent" mode.
5. Activate the controlled core power decay ramp.
6. Raise the level in both steam generators to 20.7 ft using full auxiliary feedwater head/flow capacity (Tests 340213 and 340799 only).
7. Activate the ATOG secondary pressure control (Tests 340213 and 340799 only).
8. De-energize the pressurizer "main" heaters (Test 340799 only).

9. Open the pressurizer vent if core exit subcooling exceeds 50F (Test 340799 only).

The following actions were to be performed in rapid sequence for Tests 340100, 340302, 3404AA, 340504, and 3406AA if the core exit subcooling decreases to 50F:

1. Activate full capacity HPI.

2. Initiate refill of steam generator A to 31.6 ft.

3. Terminate feedwater to steam generator B.

4. Activate the 100F/h depressurization on both steam generators (in Test 340504 activate the depressurization on steam generator A only).

The initiation actions were performed acceptably for all the tests. Core exit subcooling decreased to 50F during test initiation in Tests 340100, 340302, 3404AA, 340504, and 3406AA indicating the requirement for the second phase of the initiation actions. The anomalies encountered during test initiation are summarized in the following paragraphs.

HPI flow in Tests 340100, 340213, 340302, and 340799 was not observed within the specified time limit. The delay in HPI flow initiation, about 30 seconds beyond the target, was expected due to the time required to pressurize the accumulator in the HPI supply circuit.

The initiation time for the 100F/h depressurization of both steam generators was not discernible either due to the overcooling by the auxiliary feedwater (activated for level increase) and/or in steam generator B the influence of the tube rupture flow. In all tests, the performance of the 100F/h depressurization control was acceptable.

During Test 3404AA, the second phase of the initiation actions was erroneously delayed about 2 minutes after the core exit subcooling decreased to 50F. This delay did not affect test initiation acceptability since the timing of these events was known and can be modelled by the code.

4.1.3. Control During Testing

The performance of automatic control systems and manual interactions during the test transients is described in this section. The controls for HPI, core flood tank (CFT), AFW flow to generators A and B, core power, steam pressure, PORV, pressurizer main heaters, and level control for steam generators A and B performed acceptably for all the tests in this group except as noted in the following discussion.

Steam Generator Secondary Level Control

For Tests 340100, 340302, 3404AA, and 3406AA, steam generator constant level control was used in steam generator A to maintain the level at 31.6 ± 1 ft. Steam generator B level was to be maintained between 10 and 20 ft by manually opening and closing the steam pressure control valve when the level reached 20 and 10 ft, respectively. In Tests 340213 and 340799, steam generator level control was set at test initiation to maintain the levels at 20.7 ± 1 ft. In Test 340504, steam generator constant level control was used in steam generator A to maintain the level at 31.6 ± 1 ft, and generator B level control was disabled by closing the feedwater valve at test initiation. Performance of the steam generator level control was acceptable for all the tests. The control anomalies are described in the following paragraphs.

Constant level control performed well during all the tests, and with the exception of two occasions in Tests 340302 and 3406AA, only isolated deviations of less than 1 ft occurred. In Test 340302, secondary level in generator A remained 0.7 ft outside of the desired range (31.6 ± 1 ft) for about 9 minutes, and during Test 3406AA, secondary level in generator A remained below the desired range (31.6 ± 1 ft) for about 8 minutes.

When the level in generator B achieved 20 ft during Tests 340100, 340302, and 3406AA, the low circuit steam valve was fully opened as required. In Test 3404AA, the steam valve was opened about 65 seconds late, when the secondary level was 27 ft. In these instances with the steam valve fully opened and the flow choked, control of the secondary level was lost. The level stayed above 20 ft for the remainder of the tests.

During Test 340213, the initial fill of the steam generators exceeded the control limit of 20.7 ft, achieving 21.7 ft in loop A and [illegible].0 ft in loop B. Once the switch to constant level control was made, the level control performed properly. At 70 minutes, the generator B level began rising due to the continuing tube rupture flow while the steam generator was isolated.

Steam Pressure Control

Tests 340100, 340302, 3404AA, and 3406AA were initiated with a cooldown depressurization ramp of 100F/h in each generator. The depressurization ramp was to be interrupted in generator B when the secondary level increased to 20 ft, and restarted when the level reached 10 ft. For Test 3404AA, the interruption of the generator B steam pressure control was also required if the primary pressure decreased to 950 psia. Tests 340213 and 340799 were initiated using the ATOG steam pressure control in both generators. The ATOG steam pressure control in generator B was to be interrupted when the generator pressure reached 950 psi in Test 340213, and when the hot leg temperature decreased to 525F in Test 340799. Test 340504 was initiated with depressurization ramps of 100 psi/min and 100F/h in steam generators A and B, respectively.

Steam pressure control was performed as required for all the tests. The initiation time of the 100F/h depressurization control in the generators was not apparent in any of the tests, as discussed in section 4.1.2. AFW induced secondary-side depressurization (overcooling) in generator A during the generator fill and the tube rupture leak flow in generator B masked the initiation time of the cooldown ramp. After the initial AFW overcooling in generator A, the cooldown ramp of 100F/h was maintained.

The steam generator B secondary pressure increased in all tests upon actuation of the tube rupture leak. In Tests 340100, 340302, and 3406AA, the secondary steam pressure control valve was manually opened shortly after test initiation to control the generator B secondary level. The steam valve was left fully opened for the remainder of the test, thus overriding the automatic steam pressure control in generator B. During Tests 340213, 3404AA, and 340799, the steam pressure control valve for generator B was manually closed, as required, and stayed closed for the remainder of the test. Steam pressure control in generator B was halted during this period.

In Test 340504, the generator B secondary pressure did not achieve the required depressurization of 100 psi/min, limited by the resistance of the steam circuit and the low pressure difference from the generator to the condenser. The depressurization rate varied from 5 to 90 psi/min during the test.

The ATOG steam pressure control, at 96 minutes in Test 340799, should have automatically switched from a cooldown of 100F/h to constant pressure control as the control differential temperature exceeded 50F. However, due to an error in the pressure control table setup, the 100F/h cooldown was continued. Due to the low secondary pressure at the time of the deviation, the impact of the pressure control error on test acceptability was not significant.

Constant pressure control was in error for Test 340213 at secondary pressures below 170 psia. A control system error allowed the control setpoint to drift upward to 170 psia. The secondary pressure followed this setpoint drift as observed at about 240, 300, and 400 minutes. Since this event can be modelled by the code, it did not warrant repeating the test.

The high-capacity steam circuit for steam generator B was not activated in Test 3404AA for steam pressure control to simulate the plant sec overcooling in generator A, the cooldown ramp of 100F/h was maintained.

The steam generator B secondary pressure increased in all tests upon actuation of the tube rupture leak. In Tests 340100, 340302, and 3406AA, the secondary steam pressure control valve was manually opened shortly after test initiation to control the generator B secondary level. The steam valve was left fully opened for the remainder of the test, thus overriding the automatic steam pressure control in generator B. During Tests 340213, 3404AA, and 340799, the steam pressure control valve for generator B was manually closed, as required, and stayed closed for the remainder of the test. Steam pressure control in generator B was halted during this period.

In Test 340504, the generator B secondary pressure did not achieve the required depressurization of 100 psi/min, limited by the resistance of the steam circuit and the low pressure difference from the generator to the condenser. The depressurization rate varied from 5 to 90 psi/min during the test.

The ATOG steam pressure control, at 96 minutes in Test 340799, should have automatically switched from a cooldown of 100F/h to constant pressure control as the control differential temperature exceeded 50F. However, due to an error in the pressure control table setup, the 100F/h cooldown was continued.

Due to the low secondary pressure at the time of the deviation, the impact of the pressure control error on test acceptability was not significant.

Constant pressure control was in error for Test 340213 at secondary pressures below 170 psia. A control system error allowed the control setpoint to drift upward to 170 psia. The secondary pressure followed this setpoint drift as observed at about 240, 300, and 400 minutes. Since this event can be modelled by the code, it did not warrant repeating the test.

The high-capacity steam circuit for steam generator B was not activated in Test 3404AA for steam pressure control to simulate the plant secondary safeties. As a result, the secondary pressure increased to the maximum allowable and the test was aborted. The discussion in Section 5.4.1 indicates that even with this early termination, the important transient observations were already made so the test did not need to be repeated.

Steam Generator Auxiliary Feedwater

The AFW control performed as intended for both generators in all the tests, maintaining the feedwater flow rate at the head/flow characteristic. For all of the tests, the AFW to generator A remained within 10 to 14 lbm/h (1.4 to 1.8%) of the required head/flow characteristic during the secondary fill transient at test initiation. As required, the AFW to generator B was isolated at test initiation (after 50F core outlet subcooling achieved) during Tests 340100, 340302, 3404AA, 340504, and 3406AA. AFW to generator B remained within 10 lbm/h (1.4%) of the required head/flow characteristic during the secondary fill transient in Tests 340213 and 340799.

Power-Operated Relief Valve

The PORV remained closed during all the tests with the exception of Test 340213, since pressurizer pressure did not approach the actuation setpoint of 2350 psia.

In Test 340213, the PORV was to be opened manually when the core exit subcooling margin reached 75F and closed when it returned to 50F. The PORV was actuated 17 times as needed. The opening and closing of the PORV was performed properly with one exception. The PORV was closed during the first cycle at a subcooling margin of 31.8F instead of 50F; this did not affect test acceptability.

High-Pressure Injection

Control of HPI was performed as required. During Tests 340100, 340302, 3404AA, 340504, and 3406AA, automatic control of HPI was to be performed according to the "full" HPI head/flow characteristic. HPI flow during these tests was maintained within 10 lbm/h (1.5%) of the head/flow characteristic for the entire test. In Test 340213 and during Phases 1 and 2 of Test 340799, HPI flow was manu 34.6F. This error did not impact test acceptability.

Core Power

Core power control performed satisfactorily during each test. Core power was maintained within 2.85 kW (4%) of the intended core power decay curve throughout each test.

One of the 45 core heater rods was not energized during Test 340302, resulting from a blown fuse that was not identified until after test performance. The position of this heater rod (#67) was adjacent to one of the guide tubes and one rod removed from the bundle periphery. Total core power was still achieved. This failure did not impact test acceptability, as discussed in Section 5.3.2.

Tube Rupture Flow Rate

In Tests 340100, 340302, 3404AA, and 340504, the direct measurement of the tube rupture mass flow rate was complicated by the presence of two-phase flow conditions and/or the absence of a differential pressure measurement across the flow element in this circuit. In each instance, the tube rupture flow rate was derived from the primary mass balance.

After completing Test 340213, the tube rupture orifice was determined to have been partially blocked (estimated at about 50% blocked) beginning at about 5 minutes and extending through the end of the test. This blockage is discussed in Section 5.5.

4.1.4. Termination

Test termination activities were acceptable. Tests 340100, 340302, 340504, and 3406AA were terminated when primary pressure remained below 200 psia for

2 hours. Test 340213 was terminated based on a maximum elapsed time of 9 hours after leak initiation. Test 3404AA was terminated when the steam generator B secondary pressure approached the actuation pressure of the safety relief valve. Test 340799 was terminated 19 hours following leak initiation after the completion of the final step in Phase 3 of the test. In all cases, the tube rupture leak was closed, the loop was refilled, and the reactor vessel upper-head void was removed prior to the termination of saving data.

4.2. Instruments

The critical instruments for the 7 tests in the Group 34 series are defined in Table 4.2. The measurements obtained from the instrumentation were checked to assure acceptable operation during the tests. Checks on instrument measurements were performed by computer-automated data qualification activities and manual examination of the analysis plots. Data qualification activities for each test in Group 34 were performed at steady-state, pre-test initial conditions, during the test transient, and after test termination as summarized below:

		Time of Performance		
Check	Purpose	Before Test	During Test	After Test
NOREAD	Definition of instruments not acquiring data	x	x	x
ANDCHK	Calibration check of the Analogic data acquisition system	x		x
ZEROS	Zero check of instrument transmitters	x		x
RANGE	Validity of instrument measurement as compared to expected range	x	x	x
CONSIS	Instrument and derived quantity consistency check	x	x	x

As a result of these manual and automatic data qualification checks applied to the measurements and derived quantities in the test data base, the critical instruments identified in Table 4.3 were determined to be invalid in the test. In most instances, there was sufficient redundancy in the group of critical instruments so that the individual failure did not violate the requirements of the Critical Instrument List. In other cases, the existence of the failed critical instrument did not warrant repeating the test. For the 18 conductivity probes identified by Note 2, the measurement system error was not identified until after the test series was completed. In this instance, the void fraction obtained from neighboring differential pressure measurements provided sufficient backup except for the reactor vessel probes (RVCP01-04). The absence of these measurements did not warrant repeating the test.

In Test 340100, the cold leg gamma densitometers were included as critical instruments without backup. Three gamma densitometers (C1GD04, C3GD04, and C4GD03), failed prior to this test. The deletion of these three instruments was approved by the Project Management Group (PMG Transmittal No. 534). In Test 340302, the differential pressure transmitter for the lower primary/secondary tube rupture flow rate measurement (P2DP08) was not available. This instrument was critical and without backup. The test was repeated as 3406AA (with Project Management Group approval, PMG Transmittal No. 568, with P2DP08 spanned to measure the flow rate during the test without overranging. RVTC17 and DCTC03, two thermocouples in the RVVV line circuit, failed prior to Test 3406AA. These thermocouples were critical without backup. Approval was obtained from the Project Management Group for test performance without these instruments (PMG Transmittal No. 566).

During Tests 340100, 340302, 3404AA, 340504, and 3406AA, the primary flow rate measurements (using the cold leg and downcomer venturis) and/or the generator B steam flow rate measurement were not available or were intermittently available during most of the transient. The primary cold leg and downcomer flowmeters and the generator B secondary steam flowmeter differential pressure transmitters overranged high and low throughout most of these tests. This was caused in many instances by the presence of saturated conditions (potentially two-phase) in the cold legs, downcomer, and steam circuit of generator B. Also, the differential pressure measurements throughout the primary loop displayed erratic behavior during these tests. The calculations based on these measurements (collapsed level and void fraction) will be available, but the user must be aware that the large fluctuations in the calculated values may reflect phenomena other than actual changes in level or void fraction.

In Test 340799, cold leg flow rate measurements were not available following pump coastdown. The turbine flowmeters were installed in the cold legs and the natural circulation flows encountered in this test were not sufficiently large in magnitude to be sensed by the turbine meters.

Prior to and after completion of each test, a "zero" reading was obtained for all differential pressure and pressure transmitters, mass flowmeters, weigh tank load cells, and reactor core voltage and current measurements. For those critical instruments that failed the zero check (defined in the Immediate Report for each test), the magnitude of the failure was small enough that measurement performance was not degraded to a condition that warranted repeating the test. The instrumentation performance during these tests was fully acceptable based upon this check.

RVVV Performance

The behavior of the RVVVs was reviewed using limit switch data. The RVVVs performed symmetrically in each of the 5 Group 34 tests simulating the rupture of 10 tubes, but asymmetrically in the 2 tests involving the rupture of 1 tube. RVVV A1 was closed more often than the other RVVVs in Test 2, whereas RVVV B2 was open more often than the other RVVVs in Test 7. The characteristics of the RVVV differential pressure (DP) transmitters have been examined for each of the Group 34 tests. The responses to the initial DP change were compared among the transmitters using high-speed data. The DP transmitters were determined to have performed satisfactorily in each of the tests with the exception of RVDP08 (associated with RVVV A2) in Tests 6 and 7. In these instances, RVDP08 apparently amplified an abrupt increase of the actual DP, and responded relatively slowly to DP changes. These anomalies are expected to have caused RVVV A2 to open more frequently than the other RVVVs and to have stayed open longer. Comparing this anomaly with the performance of the RVVVs, the defective transmitter had little impact on the Group 34 tests.

Table 4.1 Test Initial Conditions

"Underlined entries indicate out-of-specification conditions. These discrepancies are discussed in the text."

System	Parameter	VTAB	Units	Desired	Tolerance	Actual Values 340100	340213	340302	3404AA	3405CC	3406AA	340799
Primary												
	Pressure	RVGP01	psia	2150	±25	2143	2154	2152	2146	2142	2153	2153
	Core power	RVWM20	kW	128.7	±1.65	128.5	128.3	128.5	128.6	128.9	128.4	129.1
	Pressurizer level	PZLV20	ft	23.0 and varying less than ±0.6 ft/h.	±0.2	23.00 and steady	23.15 and steady	23.04 and steady	22.96 and steady	23.25 and steady	23.01 and steady	22.80 and steady
	Pressurizer surge line fluid temperature	PZTC01	F	Match H1TC11	±5	H1TC11 -5.3	H1TC11 +1.4	H1TC11 -4.2	H1TC11 -2.6	H1TC11 -3.1	H1TC11 -4.3	H1TC11 -1.8
	Fluid/metal temperatures	*	F	Varying less than 3F/h for fluid and 10F/h for metal during a 30-minute interval.		accept-able	accept-able	accept-able	accept-able	accept-able	accept-able	accept-able
Secondary												
	Pressure	S1GP01 S2GP01	psia	1010	±10	1013 1015	1011 1012	1013 1014	1011 1013	1013 1014	1012 1013	1016 1015
	Level	S1LV20 S2LV20	ft	5.0	±1.0	6.0 5.6	5.1 5.5	4.2 4.9	4.0 4.7	4.6 4.8	5.2 5.1	4.4 4.7
	Feedwater temperature	SFRT01 SFRT02	F	110	±20	118.3 119.1	116.3 117.1	123.6 124.6	120.3 121.4	124.2 125.1	110.9 111.4	102.3 103.5
Core Flood Tank												
	Pressure	CFGP01	psia	600	±10	601.2	607.6	610.2	606.9	603.4	605.5	592.4
	Level	CFLV20	ft	42.8	±0.3	43.0	43.1	42.7	42.8	42.6	43.0	42.9

*The following fluid and metal temperature measurements were used to define steady state (minimum time interval of 30 minutes without test operator manual control adjustments):

Fluid: H1RT01, H2RT01, P1RT02, P2RT02.
Metal: P1MT01, P2MT01, C1MT04, C2MT04, C3MT04, C4MT04, RVMT24, RVMT25.

Table 4.2 Critical Instruments for the Group 34 Test Series

Component	Instrument Type	Critical Instruments	Additional for 340100
Reactor vessel	Ammeter	RVAM01	
	Conductivity probe	RVCP01-04	
	Differential pressure transmitter	RVDP01,RVDP03-09	RVDP02
	Differential temperature	RVDT01-04,-23	
	Pressure transmitter	RVGP01	
	Limit switch	RVLS01-04	
	Metal thermocouple	RVMT01-04,-23	RVMT24,-25
		RVMT05-22 (12 of 18)	
	Fluid thermocouple	RVTC01,-02 (1 of 2)	
		RVTC16-20	
		RVTC03-15 (9 of 13)	
		RVTC21-23 (2 of 3)	
	Voltmeter	RVVM01	
	Power controller		RVWM01-04,-23
Hot legs	Conductivity probe	H1CP01-10 (5 of 10)	
		H2CP01-10 (5 of 10)	
	Differential pressure transmitter	H1DP01,-04,-09-12,-14	H1DP02,-03,-05-08,-13,-15
		H2DP01,-04,-09-12,-14,-16	H2DP02,-03,-05-08,-13,-15
	Differential temperature	H1DT01-04	
		H2DT01-04	
	Limit switch	H1LS01,H2LS01	
	Metal thermocouple	H1MT01-04	
		H2MT01-04	
	Resistance temperature detector	H1RT01 or H1TC01,	
		H2RT01 or H2TC01	
	Fluid thermocouple	H1TC02-09 (5 of 8)	
		H2TC02-09 (5 of 8)	
		H1TC10-12 (1 of 3)	
		H2TC10-12 (1 of 3)	
		H1TC13-19 (5 of 7)	
		H2TC13-19 (5 of 7)	
	Power controller		H1WM01-04,H2WM01-04
	Gamma densitometer		H1GD01,-02,H2GD01,-02

Table 4.2 Critical Instruments for the Group 34 Test Series (Cont'd)

Component	Instrument Type	Critical Instruments	Additional for 340100
Steam generator A	Differential pressure transmitter	P1DP04,S1DP01,S1DP03	S1DP02
	Differential temperature	S1DT01-05	
	Pressure transmitter	P1GP01,S1GP01	
	Metal thermocouple	S1MT01-05	P1MT01
	Resistance temperature detector	P1RT01,-02	
	Fluid thermocouple	P1TC13-16,-23-26,-33-36 (10 of 12) P1TC18,-27,-28,-37,-38 (3 of 5) P1TC09-12,-19-22,-29-32 (8 of 12) S1TC01,-02,-26 (2 of 3) S1TC03-12 (7 of 10) S1TC13-23,-25 (8 of 12) S1TC24	
	Power controller		S1WM01-05
Steam generator B	Conductivity probe	S2CP01-12 (6 of 12)	
	Differential pressure transmitter	P2DP06,S2DP01,S2DP12 P2DP07* P2DP08** S2DP02-11 (5 of 10)	
	Differential temperature	S2DT01-05	
	Pressure transmitter	P2GP01,S2GP01	
	Metal thermocouple	S2MT01-05	P2MT01
	Resistance temperature detector	P2RT01,-02	

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

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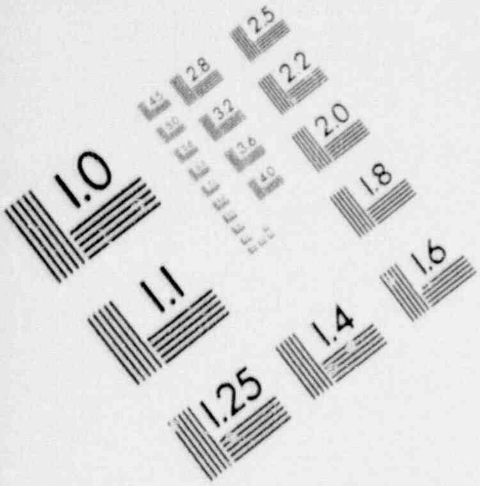
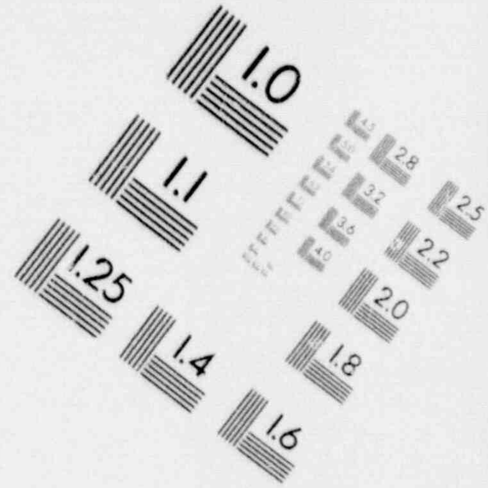
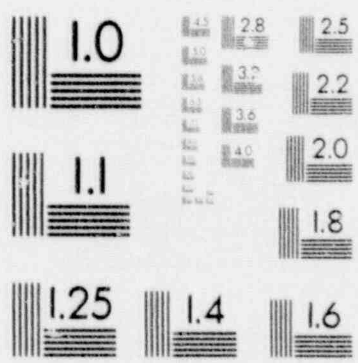
IMAGE EVALUATION TEST TARGET (MT-3)







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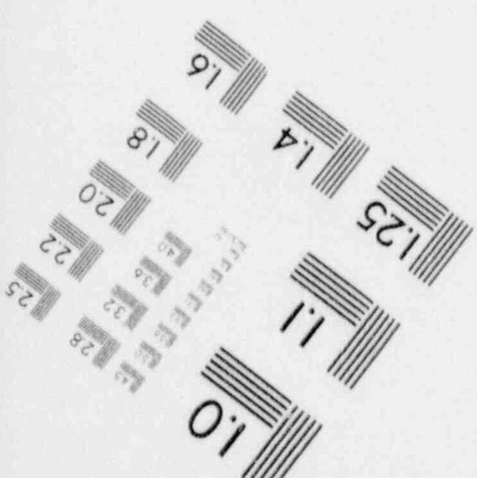

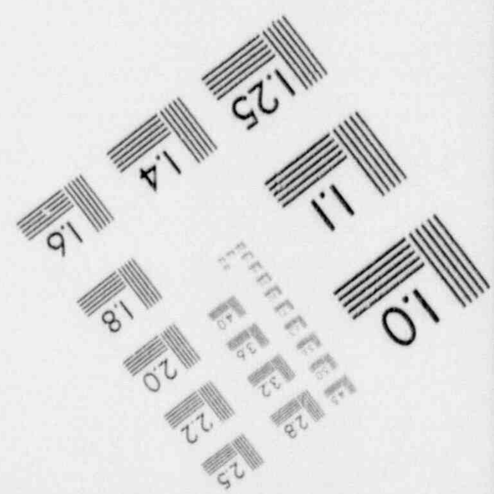

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IMAGE EVALUATION TEST TARGET (MT-3)

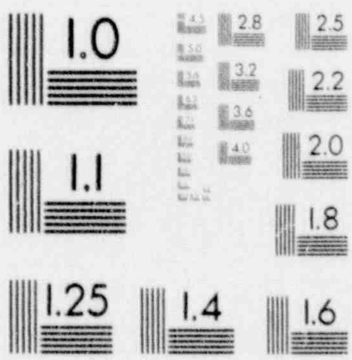

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IMAGE EVALUATION TEST TARGET (MT-3)

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Table 4.2 Critical Instruments for the Group 34 Test Series (Cont'd)

Component	Instrument Type	Critical Instruments	Additional for 340100
	Fluid thermocouple	P2TC01-13 (9 of 13)	
		P2TC14-28 (10 of 15)	
		P2TC29-43 (10 of 15)	
		P2TC44-53 (7 of 10)	
		P2TC54*	
		P2TC55**	
		S2TC01-08,-55 (6 of 9)	
		S2TC09-19 (7 of 11)	
		S2TC20-33,-54 (10 of 15)	
		S2TC34-53 (13 of 20)	
	Power controller		S2WM01-05
	Limit switch	S2LS03*	
		S2LS04**	
Cold legs (n=1,2,3,4)	Differential pressure transmitter	C1DP01,C2DP01,C2DP09	
		CnDP02-04,-06-08***	
	Differential temperature	CnDT01-03	
	Gamma densitometer		CnGD03,-04
	Metal thermocouple	CnMT01-03	CnMT04
	Resistance temperature detector	CnRT01,-02	
	Fluid thermocouple	CnTC02	
		CnTC03-06 (3 of 4)	
		CnTC07-10 (3 of 4)	
		CnTC11-14 (3 of 4)	
	Power controller		CnWM01-03
Reactor vessel downcomer	Differential pressure transmitter	DCDP01, 02,-05-08	
	Cooled thermocouple		DCCT02-04
	Differential temperature	DCDT01-03	
	Metal thermocouple	DCMT01-03	DCMT04
	Resistance temperature detector	DCRT01	

Table 4.2 Critical Instruments for the Group 34 Test Series (Cont'd)

Component	Instrument Type	Critical Instruments	Additional for 340100
	Fluid thermocouple	DCTC01-04 DCTC05-12 (5 of 8) DCTC13-40 (19 of 28) DCTC41-46 (4 of 6)	
	Power controller		DCWM01-03
Pressurizer	Differential pressure transmitter	PZDP01,-02	
	Differential temperature	PZDT01,-02	
	Pressure transmitter	PZGP01	
	Metal thermocouple	PZMT01,-02	PZMT03
	Resistance temperature detector	PZRT01 or PZTC09	
	Fluid thermocouple	PZTC01,-02 PZTC04-08 (4 of 5)	
	Power controller	PZWM04	PZWM01-03
	Limit switch		
HPI	Differential pressure transmitter	HPDP01	
	Flowmeter	HPMM01-05	
	Fluid thermocouple	HPTC01	
Two-phase vent system	Load cell	V2LC01-04	
	Limit switch	V2LS03-06	
	Flowmeter	V2MM01-03	
	Fluid thermocouple	V2TC01-04	
Core flood tank	Differential pressure transmitter	CFDP01	
	Pressure transmitter	CFGP01	
	Limit switch	CFLS01,-02 (1 of 2)	
	Fluid thermocouple	CFTC01	CFTC02,-03

Table 4.2 Critical Instruments for the Group 34 Test Series (Cont'd)

Component	Instrument Type	Critical Instruments	Additional for 340100
Gas addition	Fluid thermocouple	GATC02-04 (1 of 3)	
Feedwater circuit	Differential pressure transmitter	SFDP01-06	
	Resistance temperature detector	SFRT01,-02	
Steam circuit	Differential pressure transmitter	SSDP01-06	
	Resistance temperature detector	SSRT01,-02	
	Fluid temperature	SSTC01,-03 (1 of 2) SSTC02,-04 (1 of 2)	
Miscel-laneous	Resistance temperature detector shunt	MSRF01	
	Reference oven temperature	MSTC01-07	

*The upper steam generator primary-to-secondary tube rupture circuit instrumentation (P2DP07, P2TC54, and S2LS03) was critical for all the tests in Group 34 except 340302 and 3406AA.

**The lower steam generator primary-to-secondary tube rupture circuit instrumentation (P2DP08, P2TC55, and S2LS04) was critical for Tests 340302 and 3406AA only.

***The cold leg venturi differential pressure measurements, CnDP03, -04, -05, and -06 (n=1,2,3,4), were not installed for Test 340799.

Table 4.3 Critical Instruments Not Available for the Group 34 Test Series

Instrument	Description	340100	340213	340302	3404AA	340504	3406AA	340799	Backup Available
CFLS01	Limit switch on loop A header isolation valve	x	x	x					yes
C1DT01	Loop A1 guard heater zone 28 control at 2.60 or 2.57 ft							*	yes
C1GD04	Beam 2 loop A1 cold leg density at 21.25 ft	x							no
C1TC04	Pump suction fluid temperature at 2.36 ft	x	x	x	x	x	x		yes
C3GD04	Beam 2 loop A2 cold leg density at 21.25 ft	x							no
C4DT01	Loop B2 guard heater zone 37 control at 2.59 ft							*	yes
C4DT03	Loop B2 guard heater zone 39 control at 23.47 ft							*	yes
C4GD03	Beam 1 loop B2 cold leg density at 21.25 ft	x							no
DCTC03	Fluid tempe[illegible]tream of vent valve 3 at 24.16 ft						x		no
H1CP01	Hot leg fluid conductivity probe at 23.16 ft	**	**	**	**	**	**		yes
H1CP02	Hot leg fluid conductivity probe at 28.54 ft	**	**	**	**	**	**	x	yes
H1TC04	Hot leg fluid temperature at 29.68 ft							x	yes
H1TC13	Hot leg fluid temperature at 64.68 ft							x	yes
H2CP04	Hot leg fluid conductivity probe at 43.41 ft							x	yes
H2CP05	Hot leg fluid conductivity probe at 50.68 ft							x	yes
H2CP06	Hot leg fluid conductivity probe at 59.69 ft							x	yes
H2CP07	Hot leg fluid conductivity probe at 63.56 ft							x	yes
H2CP09	Hot leg fluid conductivity probe at 66.78 ft	x	x	x	x	x	x		yes

Table 4.3 Critical Instruments Not Available for the Group 34 Test Series (Cont'd)

Instrument	Description	340100	340213	340302	3404AA	340504	3406AA	340799	Backup Available
P2WM03	Spray line guard heater total power	x							no
P1TC30	Generator A primary fluid temperature at 50.58 ft	x	x	x	x	x	x	x	yes
P1TC35	Generator A primary fluid temperature at 39.08 ft	x	x	x	x	x	x	x	yes
P2DP08	Lower tube rupture differential pressure at -2.00 ft			x					no
P2TC01	Generator B primary fluid temperature at 50.50 ft						x	x	yes
P2TC12	Generator B primary fluid temperature at 49.50 ft	x	x	x	x	x	x	x	yes
P2TC30	Generator B primary fluid temperature at 29.25 ft						x	x	yes
P2TC40	Generator B primary fluid temperature at 14.25 ft							x	yes
RVCP01-04	Reactor vessel fluid conductivity probes	**	**	**	**	**	**		no
RVTC07	Core fluid temperature (mid-bundle) at 13.15 ft	x	x	x	x	x	x	x	yes
RVTC17	Fluid temperature upstream of vent valve 1 at 24.18 ft							x	no
S1TC04	Generator A secondary fluid temperature at 11.07 ft	x	x	x	x	x	x	x	yes
S1TC16	Generator A secondary fluid temperature at 38.19 ft	x	x	x	x	x	x	x	yes
S1TC19	Generator A secondary fluid temperature at 41.28 ft		x						yes
S2CP01-12	Generator B secondary conductivity probes	**	**	**	**	**	**		no
S2TC12	Generator B secondary fluid temperature at 14.27 ft		x						yes
S2TC13	Generator B secondary fluid temperature at 20.19 ft		x						yes
S2TC14	Generator B secondary fluid temperature at 26.27 ft		x						yes
S2TC16	Generator B secondary fluid temperature at 26.27 ft		x						yes
S2TC24	Generator B secondary fluid temperature at 38.19 ft		x						yes

*Metal temperature used for guard heater control in place of the differential temperature.

**Raw data obtained with these conductivity probes can not be processed due to a measurement problem observed after completing these tests.

5. OBSERVATIONS

The 7 tests of Group 34 consisted of 5 large-rupture tests and 2 single-tube rupture tests, as summarized in Table 3.1. The pre-test, steady-state conditions and the initiation of these tests are addressed in section 5.1. Subsequent sections provide test observations and inter-test comparisons. These observations are summarized in the final section, section 5.6.

5.1. Pre-Test Steady State

Six of the seven Group 34 tests were initialized almost identically. The exception was Test 7, which was performed with the loop reconfigured for forced flow. The replacement of the cold leg venturi flowmeters with turbine meters reduced the loop hydraulic losses sufficiently to increase the loop flow rate in Test 7, and thus to decrease the fluid temperature rise across the core.

The average cold leg flow rate in Tests 1 through 6 was 4.1% of the scaled full cold leg flow rate, cold leg B1 registered higher than the others at 4.2% and cold leg A2 indicated below the average at 4.0%. The downcomer flow rate indicated just below 3.9% of scaled full flow.

The core exit subcooling ranged from 52 to 54F in Tests 1 through 6, and the core exit fluid temperature was 593 or 594F. The primary fluid temperature changes across the core and steam generators were approximately 48 and 42F. The primary and secondary system pressures were approximately 2150 and 1015 psia, the pressurizer level was 23 ft, and the pretest core power (control electrical input) was 128.6 kW ($\pm$0.4) or 3.9% of scaled full power.

The steam generator secondary levels were initialized between 4 and 6 feet. The steam generator secondary feed and steam flow rates were approximately 2.3% of scaled full secondary flow, the steam generator A flow rates generally indicated 5% higher than those of steam generator B. The feedwater temperatures were quite variable among tests, ranging from 103 to 125F. The steam outlet temperatures were more uniform at 580F.

Test Initiation

Test initiation was similar among the 5 tests simulating the double-ended rupture of 10 tubes, and between Tests 2 and 7 simulating the single-ended rupture of 1 tube. The rupture simulation was opened near time zero and within 0.3 minutes, the RVVVs actuated, the core power was reduced simulating decay, and HPI was activated. High-pressure injection flow was achieved within approximately one-half minute, between 0.6 and 0.9 minutes after rupture activation. Steam generator A feeding was increased between 0.1 and 0.4 minutes and, with the 10-tube rupture simulations, steam generator B feeding was reduced between 0.3 and 0.5 minutes.

Test 4, Steam Generator Isolation, represented the exception. Although the core power reduction and RVVV actuation occurred near time 0, the HPI and steam generator feed rate changes were delayed for approximately 2 minutes. (Test 4 is compared to Nominal Test 1 in section 5.4.2.)

Whereas the system conditions were little perturbed upon the simulated, single-ended rupture of 1 tube (in Tests 2 and 7), the imposition of a double-ended rupture of 10 tubes had pronounced effects. The primary system fluid inventory initially dropped at 2200 lbm/h, approximately 4% of the total inventory per minute. The primary system depressurized at some 500 psi/min, saturating in about 1 minute.

5.2. High-Elevation Rupture of 10 Tubes, Tests 1 and 5

The double-ended, high-elevation rupture of 10 tubes was simulated in Nominal Test 1. Additionally, a steam line break was simulated in Test 5. These tests are described separately in sections 5.2.1 and 5.2.2, and are compared in section 5.2.3.

5.2.1. Test 1, Nominal

Test 1 (340100) was the Nominal SGTR Test. The SGTR simulated the double-ended break of 10 tubes, high in steam generator B. The test transient was rapid and relatively complex. The primary system voided extensively. The unaffected steam generator A became a heat source, and cyclic perturbations of system conditions developed. The tube rupture mass flow rate was obtained directly from the differential pressure measured across the rupture simulation, but only when the rupture fluid state was single phase. Similarly, the steam generator steam flow meter indications were valid only with single-phase effluent. Both flow streams evidenced two-phase conditions during much of this test (and the other larger rupture tests), therefore the direct flow rate measurements were not available. Both mass flow rates have been estimated using indirect methods. The tube rupture mass flow rate was inferred from the primary system mass imbalance; it was taken to be the mass flow rate necessary to balance the primary system total fluid mass after all other flow streams had been considered. This estimated tube rupture mass flow rate was, in turn, applied to the steam generator B secondary mass balance. The steam generator B steaming rate was then determined in a manner analogous to that used for the tube rupture mass flow rate; the steaming rate was inferred from the rate of change of the steam generator B secondary total fluid mass, after all other boundary streams had been considered. These indirect measurements are discussed further in Appendix A of Volume 9.

Initiation, Saturation, and Voiding

The test was initiated by opening the upper-elevation, 30.8-cm^2 flow path to simulate the double-ended rupture of 10 tubes, transferring the RVVV controls to automatic/independent, and activating the core power decay ramp. The core exit SCM almost immediately decreased below 50F, triggering the second set of test-initiating actions. These actions were to actuate HPI, reset the steam generator control level to 31.6 ft, and depressurize the steam generator to obtain a cooldown rate of 100F/h. The primary system depressurized from 2150 to 1500 psia at a rate of more than 500 psi/min (Figure 5.2.1). The secondary pressure in steam generator B increased some 50 psi in the first half-minute until the increased steaming rate took effect. At 1 minute, the (intact) loop A hot leg levels began to indicate voiding. Also, the loop flow rates peaked at 1 minute, then the loop A flow rates stagnated whereas the loop B flow rates remained active. The steam generator B secondary level rose at 6 ft/min (Figure 5.2.2), about twice the rate of increase of the steam generator A secondary. (Only steam generator A was being fed.) The steam generator B secondary level approached 20 ft at 2.7 minutes, triggering the final test-initiating action. The low-flow steam valve was manually opened fully. The steam generator B steam flow rate increased markedly (Figure 5.2.3). The steam generator B secondary level increase was temporarily halted, and the secondary began to depressurize from 1050 psia.

Beyond 4 minutes, the primary system depressurization halted at 1250 psia as the primary system voided extensively. The core-region collapsed liquid level, which had begun to recede at 2.6 minutes, reached the RVVVs at 5 minutes (Figure 5.2.4). The core exit fluid saturated at 4.2 minutes and remained saturated thereafter. The loop B U-bend fluid voided briefly at 3.5 minutes and again beyond 7 minutes. The loop B flow rate subsided as the U-bend voided. Even with the valve fully opened, the steam mass flow rate through the steam generator B low-flow steam circuit gradually became insufficient to hold level as the secondary pressure and the pressure drop across the steam valve continued to drop. The steam generator B secondary level thus descended to 20 feet at 5 minutes and then began to rise (Figure 5.2.2). The steam generator B secondary pressure was then 730 psia and falling (Figure 5.2.1).

The core region and downcomer voided down to the nozzle elevation between 5 and 7 minutes, then the core-region level continued to decline. As the nozzles became increasingly uncovered, the loop voiding increased and the cold leg suction and discharge levels (Figure 5.2.5) began to evidence voiding beyond 12 minutes. At 7 minutes, the loop B hot leg levels receded from the U-bend while the loop A steam generator primary level descended into the steam generator (Figure 5.2.6). The steam generator A secondary side refill was just being completed; thus, the primary system briefly depressurized through an AFW-BCM, which is the condensation of primary system vapor within the steam generator. Steam generator A feed was reactivated between 8.5 and 15 minutes (Figure 5.2.3) to maintain secondary level; the primary system depressurized from 1220 to 850 psia during the resulting BCM.

By 11 minutes, the steam generator B steaming mass flow rate had decreased sufficiently to come back on scale; it continued to decrease from 9% (of scaled full steam generator secondary) flow. The tube rupture mass flow rate was also diminishing as the primary-to-secondary pressure difference declined and as the rupture-site fluid approached saturation. Hence, the rate of refill for steam generator B slowed beyond 30 ft to less than 1 ft/min (Figure 5.2.2).

The loop A hot leg levels continued to recede while the loop B levels remained between the upper tubesheet and the U-bend spillover elevations. The loop A steam generator primary level descended to the secondary pool at 17 minutes (Figure 5.2.2). The temperature difference between the primary system and the steam generator A secondary was so slight, however, that the pool BCM had virtually no effect.

The steam generator B primary level remained near the tube rupture elevation (Figure 5.2.6). The rupture-site fluid apparently voided intermittently so that the time-averaged rupture flow rate gradually decreased from 1350 lbm/h at 18 minutes (Figure 5.2.7). During this period, the primary system pressure also declined at about 12 psi/min. (The rupture mass flow rate has been estimated using the change of the indicated primary system total fluid mass and the HPI flow rate.)

Periodic Loop B Activity

The hot leg riser level in loop B remained near the spillover elevation (Figure 5.2.6); hence, the tube rupture flow from the steam generator B primary was replenished by intermittent flow over the hot leg B U-bend. The primary system depressurized to the steam generator A secondary pressure at 23 minutes (Figure 5.2.1). Therefore, steam generator A became a heat source. The steam generator A primary level began to decrease more rapidly as vapor was generated within the steam generator (Figure 5.2.8).

The void generation within the steam generator A primary apparently caused the hot leg riser level in loop B to ascend to the U-bend. The following events then occurred in rapid succession beginning at 23.8 minutes: The loop B riser level increased, followed by the level beyond the U-bend (Figure 5.2.8). The loop B cold leg flow rates briefly indicated strong forward fluid motion. The loop B cold leg suction fluid temperatures subcooled, almost to the cold leg spillover elevation (Figure 5.2.9). The loop B cold leg suction fluid levels briefly indicated full (Figure 5.2.10). The primary system depressurization rate increased slightly. The loop A hot leg riser and stub (steam generator primary) levels (Figure 5.2.8) abruptly decreased (due to the increased voiding caused by the reduced primary system pressure). The loop B hot leg riser level (Figure 5.2.8) receded some 10 ft from the U-bend (due to the fluid displacement to the cold leg suction region).

The preceding interactions persisted for about 1 minute. The system conditions then gradually realigned. The cold leg suction levels in loop B receded to their pre-perturbation levels and the hot leg B riser level gradually rose toward the spillover elevation (Figure 5.2.8). A second perturbation thus occurred at 26.9 minutes. However, the hot leg A riser had already emptied. Thus, the second perturbation was stronger than the first. During this event, the loop B hot leg riser and stub levels dropped far below the upper tubesheet (and tube rupture) elevations (Figure 5.2.8). All the loop B cold leg suction fluid temperatures subcooled (Figure 5.2.9) and the loop B cold leg discharge and suction levels responded to the fluid displacement (Figure 5.2.10). The primary system depressurization was sufficient to produce a noticeable difference between the pressures in the steam generator A secondary system and the primary system (Figure 5.2.11). Also, the duration of rupture uncovery was sufficient to affect the time-averaged tube rupture flow rate, which decreased to less than 700 lbm/h (Figure 5.2.12).

These periodic perturbations of system conditions persisted for almost one-half hour. The steam generator A primary level gradually approached the lower tubesheet (Figure 5.2.6). The pressure difference across the tube rupture gradually diminished (Figure 5.2.12) and the primary system total fluid mass gradually stabilized (as the tube rupture flow rate decreased and as the increasing HPI flow rate exceeded the rate of loss). Also, the core flood tank became active beyond 35 minutes. Steam generator A began to be steamed at 50 minutes (Figure 5.2.13), thus lessening the magnitude of secondary-to-primary heat transfer within steam generator A.

Depressurization Through Test Termination

The controlled depressurization of the steam generator A secondary became increasingly effective as the transient progressed. The steam generator A steaming and feeding rates were increased and the steam generator A secondary was depressurized below the primary system pressure (Figure 5.2.14). The steam generator B tube rupture flow rate was thereby continually reduced (Figure 5.2.15). The loop A hot leg began to be refilled between 100 and 120 minutes (Figure 5.2.16). The refill of the steam generator A primary was via the loop A cold legs. The flow rates in both loop A cold legs thus were negative between 110 and 123 minutes, then the loop A hot leg riser level reached the U-bend spillover elevation. The remainder of the transient was relatively uneventful. The primary system pressure remained below 200 psia beyond 128 minutes. The test was terminated at 222 minutes based on the low-pressure termination criteria.

5.2.2. Test 5, Steam Line Break

A simulated steam line break was superimposed on the nominal tube rupture in Test 5 (340504). The initial transient was relatively rapid. The rate of discharge from the broken steam generator approached the tube rupture flow rate early in the transient. The intact steam generator became a heat source, whereas the broken steam generator depressurized relatively rapidly. The primary system was depressurized through intermittent activity in the broken loop. The hot leg and steam generator primary of the intact loop A voided almost completely, whereas those components in the broken loop were periodically refilled to the spillover elevation. The analysis of the data was somewhat encumbered by the absence of direct measurements of the rupture and the steam generator B discharge flow rates.

The test was initiated by actuating a simulated tube rupture and steam line break. Also, the RVVV controls were transferred to automatic/independent and the core power decay ramp was activated. The steam line break was simulated by depressurizing the steam generator B secondary at 100 psi/min using the steam flow control valve for high flow rates. The high-elevation, double-ended rupture of 10 tubes was simulated using a scaled 30.8-cm^2 primary-to-secondary flow path at the top of steam generator B. The response of the systems to these initiation actions was almost instantaneous, as indicated by the following interactions: At time zero the upper-elevation tube rupture flow path opened by limit-switch indications. The primary system began to depressurize from 2140 psia at 540 psi/min. The initial tube rupture flow rate exceeded the steam generator B steaming rate, therefore, the steam generator B secondary began to repressurize slightly (Figure 5.2.17). At 0.1 minutes the RVVVs began to open by limit-switch indications. The temperature difference across the RVVVs began to decrease abruptly.

The tube rupture flow rate exceeded 2000 lbm/h (Figure 5.2.18) and the core exit SCM decreased rapidly, thus triggering the second and final set of test-initiating actions. HPI was activated, the control point of the steam generator A secondary level was increased to 31.6 ft, feeding in steam generator B was interrupted, and steam generator A was depressurized to obtain a cooldown of 100F/h. The system conditions continued to change rapidly, as indicated by the following sequence: The HPI isolation valves opened at 0.2 minutes by limit-switch indications. The secondary pressure in steam generator B peaked near 1050 psia and then began to decrease at 100 psi/min (Figure 5.2.17). At 0.3 minutes the feeding rate in steam generator A increased toward the full head-flow capacity, while the rate in generator B decreased toward zero. (The steaming rate in generator A began to vary, and the indicated rate in generator B continued near zero flow.) The RVVVs began to close at 0.7 minutes by limit-switch indications -- valves A1 and A2 closed first. The temperature differences across the RVVVs reached approximately 5F and then began to increase. The riser level in hot leg A began to recede from the U-bend at 0.8 minutes (Figure 5.2.19). The differential pressures across the RVVVs began to decrease more rapidly than before, from 0.02 psi, at 0.9 minutes.

The secondary level and fluid mass of steam generator B briefly stabilized beyond 1 minute (Figure 5.2.20) as both the tube rupture and secondary discharge rates peaked near 2200 lbm/h (Figure 5.2.18). Primary system saturation dominated the interactions, as indicated in the following chronology: The primary system depressurization slowed abruptly at 1575 psia (indicating saturation of the primary loop fluid, Figure 5.2.17) at 1.0 minutes. The steam generator B secondary level and indicated primary fluid mass approximately stabilized. The hot leg A U-bend fluid temperature reached saturation at 1.1 minutes. The steam generator A secondary level began to increase at 3.8 ft/min. The hot leg A stub level began to recede from the U-bend at 1.2 minutes; the loop A cold leg flow rates abruptly decreased from 10% of scaled full cold leg flow (Figure 5.2.21). The differential pressures across the RVVVs began to rapidly increase from -0.1 psi. At 1.3 minutes the RVVVs began to open by limit-switch indications; the temperature differences across the RVVVs decreased from a maximum of approximately 15F. The steaming rate in generator A began to decrease from 3.5% of the scaled full steam generator steaming rate. The steam generator A primary level decreased below the elevation of the upper tubesheet at 2.4 minutes (Figure 5.2.20). The reactor vessel collapsed liquid level began to decrease from full at 2.5 minutes (Figure 5.2.22). The reactor vessel volume above the RVVVs began to indicate voiding (Figure 5.2.23). At 3.3 minutes the loop B stub level began to recede from the U-bend (Figure 5.2.19). The loop B cold leg flow rates began to decrease from 7% of scaled full cold leg flow (Figure 5.2.21). The loop B U-bend fluid temperature reached saturation. The loop B riser level began to recede from the U-bend (Figure 5.2.19). The core exit fluid saturated at 4.0 minutes; the reactor vessel collapsed liquid level began to decrease more rapidly, and the core exit volume began to indicate voiding (Figure 5.2.23). The loop B cold leg flow rates stabilized in reverse flow (Figure 5.2.21). The steam generator A primary level exceeded the elevation of the upper tubesheet by 4.3 minutes.

The interactions beyond 5 minutes were predominated by intermittent flow in loop B. The interactions were as follows: The loop B hot leg riser level approached the U-bend spillover elevation (Figure 5.2.19). The loop B cold leg flow rates began to increase (and reached 11% of scaled full cold leg flow at 5.8 minutes, Figure 5.2.21). The primary system pressure stabilized at 1250 psia and then began to decrease (Figure 5.2.17). The rate of depressurization of the steam generator B secondary slowed. The RVVVs began to close intermittently. The core exit fluid subcooled from 5.7 to 6.3 minutes (Figure 5.2.24). The core exit void fraction diminished whereas the volume between the nozzles and the RVVVs began to void (Figure 5.2.23). The steam generator A primary level again descended below the elevation of the upper tubesheet at 6.5 minutes. The reactor vessel and downcomer levels approached the elevation of the nozzles (Figure 5.2.22). The cold leg B1 discharge piping briefly voided.

The steam generator A secondary level achieved the control point at 8 minutes, thus slowing the primary system depressurization rate. The steam generator A primary level descended to the secondary pool at 12 minutes (Figure 5.2.25), but the remaining primary-to-secondary (A) temperature difference was insufficient to promote much heat transfer (Figure 5.2.26).

Beyond 12 minutes, the hot leg A riser level continued to descend toward its stub level (Figure 5.2.27), the hot leg B riser and stub levels rose gradually toward the U-bend, and the reactor vessel downcomer levels decreased toward the top of the core. The hot leg B riser level achieved the U-bend spillover elevation just before 18 minutes, initiating a prolonged series of periodic interactions. The displacement of fluid within loop B caused the steam generator B secondary to repressurize and the primary to depressurize (Figure 5.2.26). The displacement was recorded on the cold leg B flowmeters and on the cold leg B suction and discharge level measurements (Figure 5.2.28). The core and downcomer levels, now near the top of the core, rose as the loop B hot leg (riser and stub) levels fell.

After the second perturbation at 21 minutes, the steam generator A secondary had become a heat source (Figure 5.2.26). The hot leg A riser voided completely. The steam generator A primary level decreased below the secondary pool level (Figure 5.2.27) as the primary fluid flashed through both primary system depressurization and secondary-to-primary heat transfer. The steam generator B primary level had resided near the rupture elevation prior to these perturbations. Now the rupture site was periodically uncovered (Figure 5.2.27). This change of state of the rupture-site fluid plus the continuing primary system depressurization caused the rupture flow rate to decrease to 880 lbm/h beyond 20 minutes (Figure 5.2.29). The rate of discharge from the steam generator B secondary then became sufficient to approximately stabilize the secondary mass and level. The core flood tank began to discharge beyond 30 minutes. The addition of the core flood tank discharge, as well as the tube rupture and HPI flow rate variations with decreasing primary system pressure, stabilized the primary system mass (Figure 5.2.30).

Beyond approximately 50 minutes, the steam generator A secondary was reactivated to maintain its cooldown rate of 100F/h. Although the steam generator A secondary pressure remained higher than that of the primary system (Figure 5.2.31), the steam generator A steaming facilitated the primary system depressurization. The periodic loop B perturbations gradually diminished as the primary system depressurized.

The remainder of the transient was quite uneventful. The steam generator A secondary pressure was reduced below the primary pressure near 105 minutes (Figure 5.2.31). Both the primary system and the steam generator B secondary approached mass equilibrium. The primary system total fluid mass began to increase beyond 2 hours as the tube rupture flow rate decreased below approximately 500 lbm/h (Figure 5.2.29). The test was terminated at 215 minutes, based on the low-pressure termination criteria (primary system pressure less than 200 psia for 2 hours).

5.2.3. Comparisons

Nominal Test 1 and Steam Line Break Test 5 behaved similarly. Steam generator B was depressurized at 100 psi/min at the initiation of Test 5 (using the high-capacity steam control valve) to simulate the effects of a steam line break (Figure 5.2.32). The low-capacity steam flow control valve was opened fully at 2.7 minutes to control level in Nominal Test 1, as the steam generator B secondary level exceeded 20 feet (Figure 5.2.33). As a result, the steam generator B secondary pressure, which had remained approximately constant in Test 1, was reduced at rates exceeding 100 psi/min. The steam generator B depressurization rate in Test 5 began to slow beyond approximately 5.5 minutes and 550 psia, apparently due to the decreasing differential pressure across the steam flow circuit. The steam generator B secondary pressure in Test 1 thus approached that of Test 5 to within 100 psi by 15 minutes (Figure 5.2.32).

The earlier increased steam generator B steaming in Test 5 was reflected in its secondary levels. The generator B level remained near 10 feet until 8 minutes in Test 5, whereas it was stabilized near 20 feet in Test 1 (Figure 5.2.33). The primary system initially depressurized slightly more rapidly in Test 5 (Figure 5.2.32), apparently due to the greater heat removal rate obtained with the enhanced generator B steam flow rate. The upper reactor vessel and downcomer voided more rapidly in Test 5 due to the earlier primary system depressurization (Figure 5.2.34). The conditions of the rupture-site fluid obtained larger primary-to-secondary flow rates in Test 5 than in Test 1, therefore the primary system fluid mass in the Nominal Test gradually began to exceed that in Test 5 (Figure 5.2.35). (The tube rupture flow rates, Figure 5.2.36, were obtained from the primary system mass balance, as described in Appendix A of Volume 9, and are therefore of qualitative rather than quantitative value.)

Intermittent loop B activity, as well as reverse heat transfer in steam generator A, were experienced beyond approximately 20 minutes (Figure 5.2.32) in both Tests 1 and 5. The loop A hot leg riser voided completely (Figure 5.2.37), the loop A steam generator primary level depleted through vapor generation within the generator (Figure 5.2.38), and the loop B hot leg riser levels alternately dropped below the rupture site and reachieved the spill-over elevation (Figure 5.2.37) in both tests. Also in both tests, the reactor vessel and downcomer levels remained near the elevation of the top of the core (Figure 5.2.34), and the primary as well as the secondary pressures became nearly equal between the two tests (Figure 5.2.32).

The steam generator A secondary was depressurized below the primary system pressure near 105 minutes in both tests (Figure 5.2.39), and the loop A levels and primary system total fluid mass began to increase (Figure 5.2.40 and 5.2.41). Refill in Test 5 was delayed beyond that in Test 1, and the total primary system fluid mass in Test 5 remained some 50 lbm below that in Test 1. This lingering inter-test difference apparently reflected the different methods of steaming the generator B secondary. The low-capacity steam flow circuit was used in Nominal Test 1 whereas the high-capacity circuit was used in Test 5. The increased steam flow area of Test 5 supported a larger steam flow rate as well as a slightly larger primary-to-steam generator B differential pressure. Because of the relatively small primary-to-secondary differential pressures later in the tests (Figure 5.2.39), the rupture flow rates became loss limited and responsive to differential pressure, rather than being choked in critical flow. The rupture flow rate thus apparently remained slightly larger in Test 5, obtaining the inter-test difference of total primary system fluid mass (Figure 5.2.41).

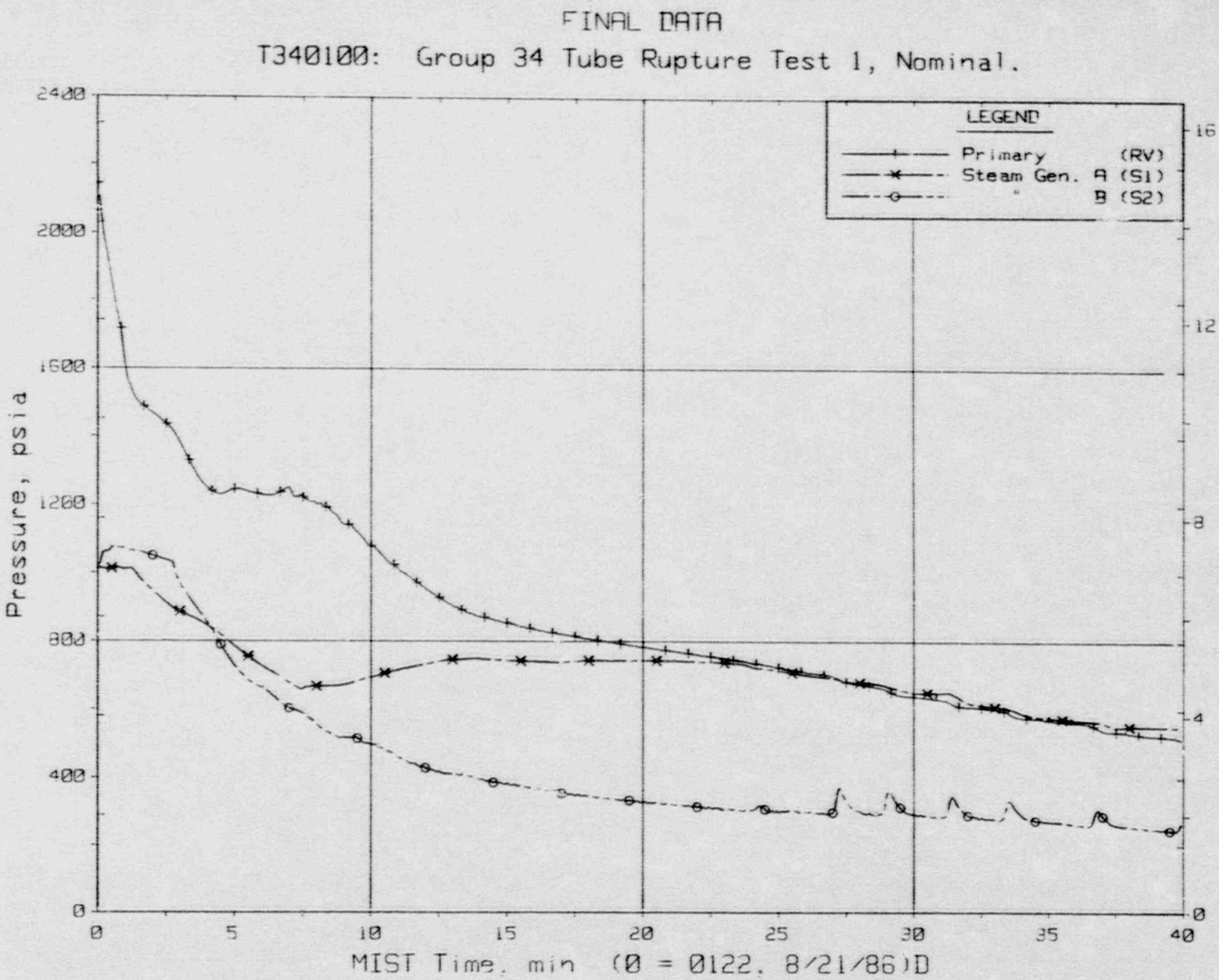

Figure 5.2.1. Primary and Secondary System Pressures (GP01s)

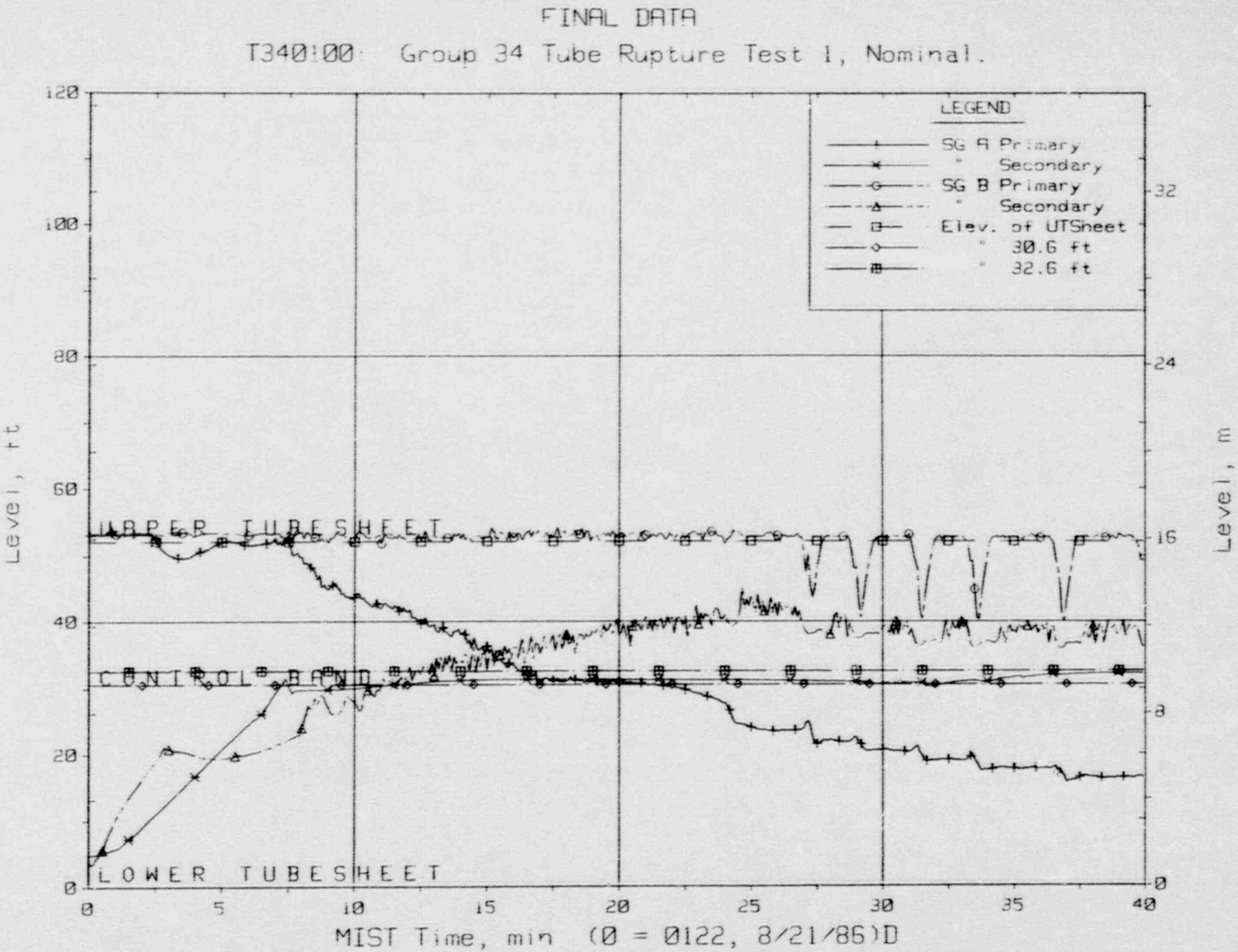

Figure 5.2.2. Steam Generator Collapsed Liquid Levels

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SGLV2

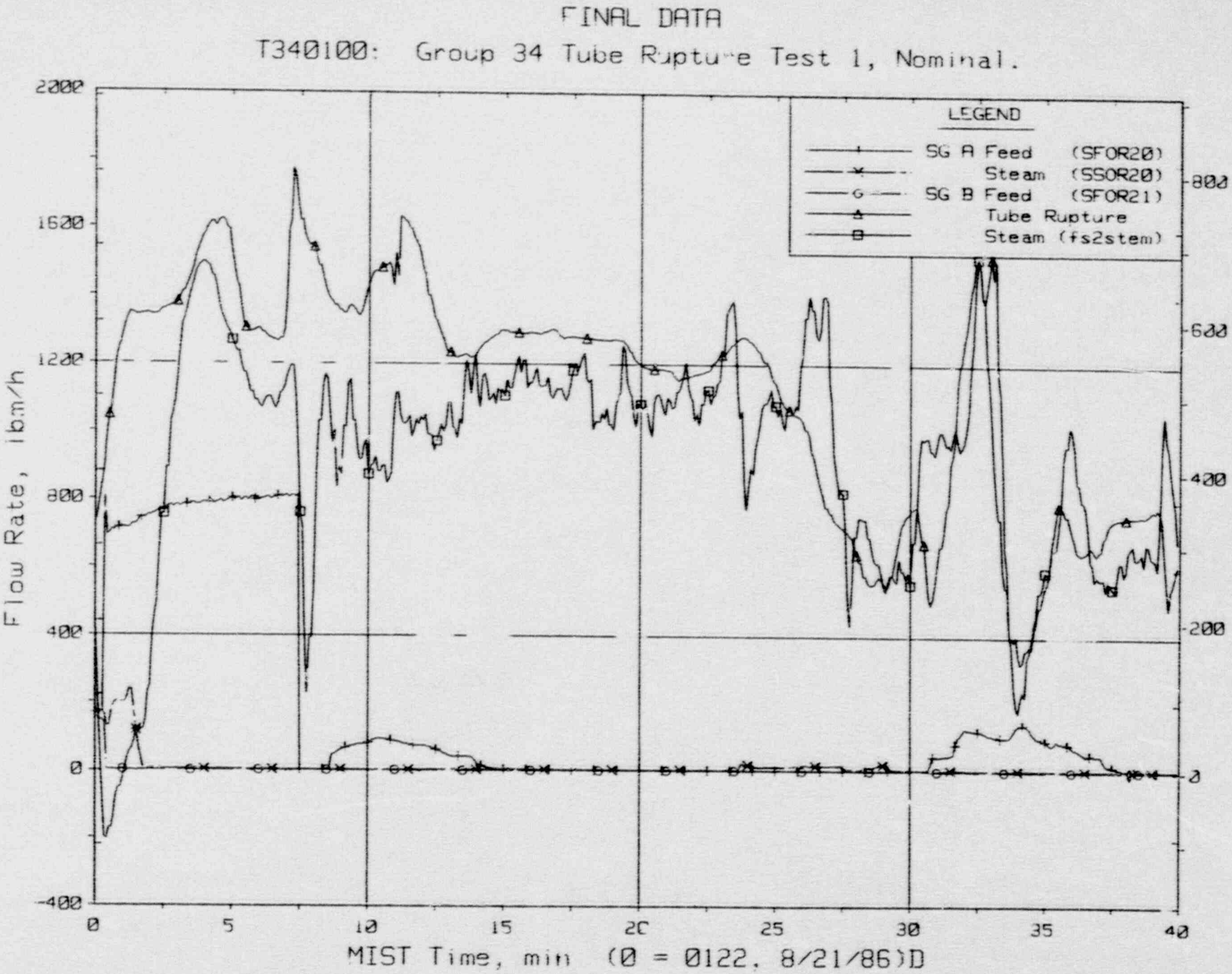

Figure 5.2.3. Secondary System Flow Rates

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SGFL1

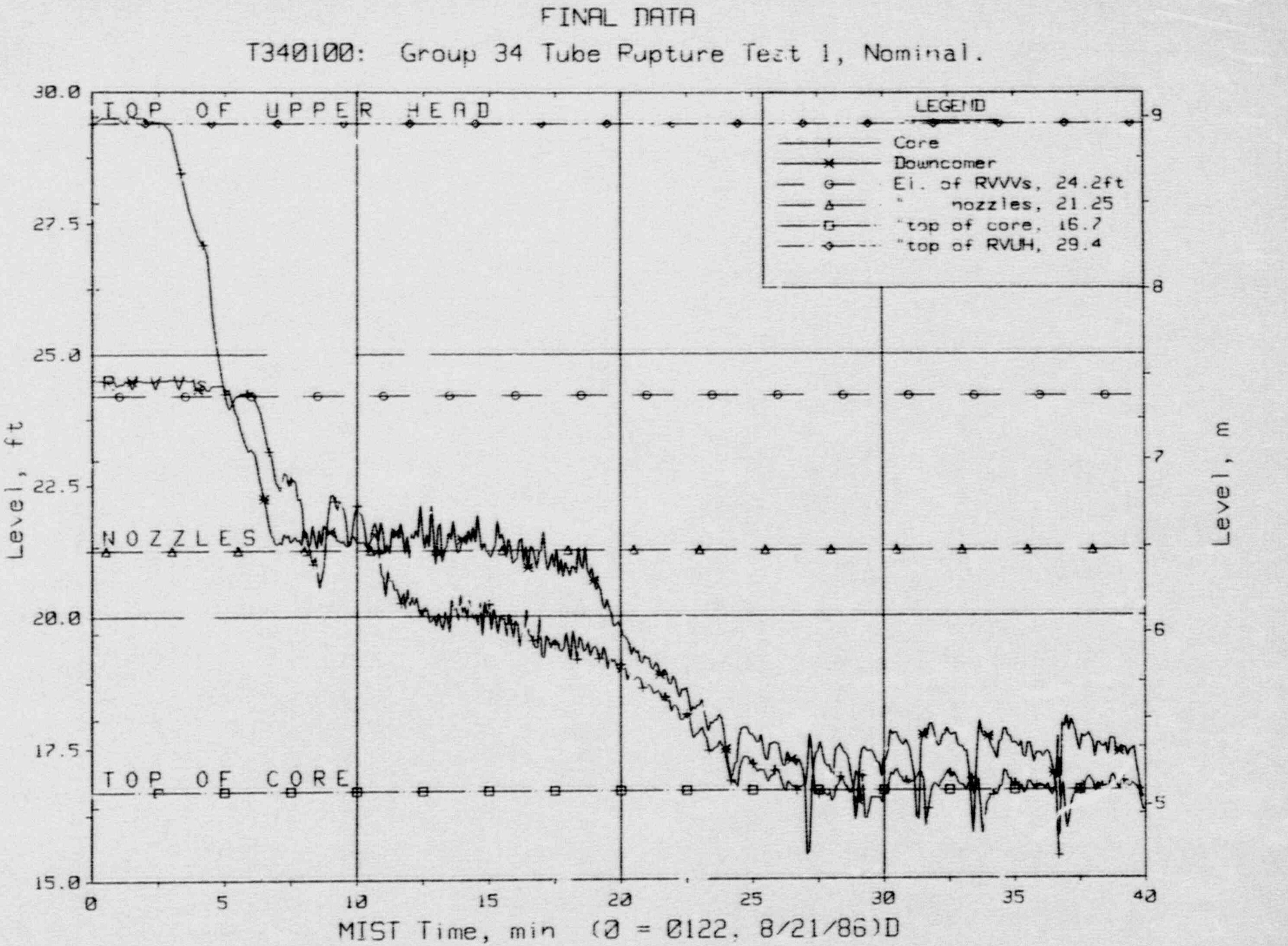

Figure 5.2.4. Core Region Collapsed Liquid Levels

COLV1

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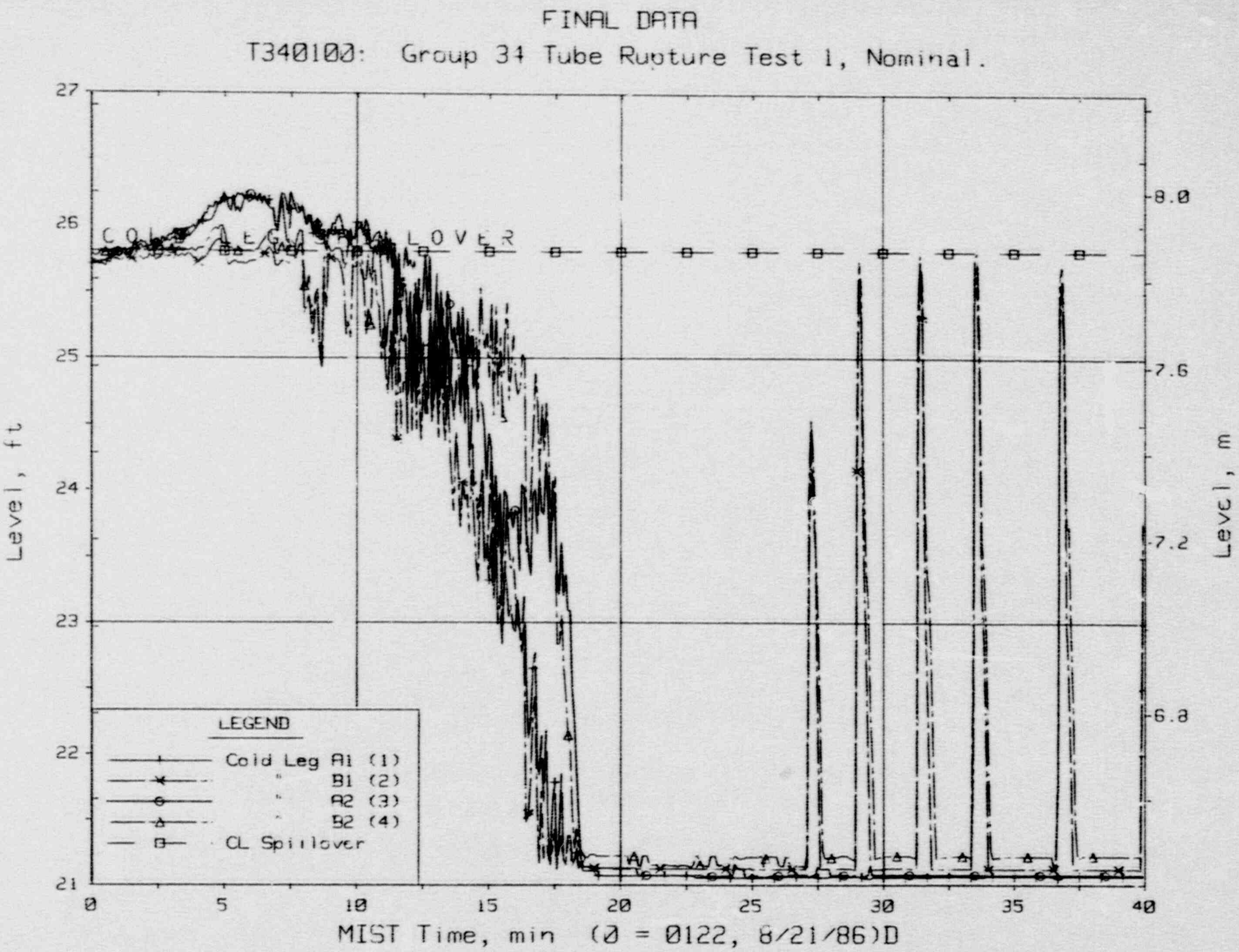

Figure 5.2.5. Cold Leg Discharge Collapsed Liquid Levels (CnLV23s)

Mon Jul 25 09:41:09 1988

CLLV5

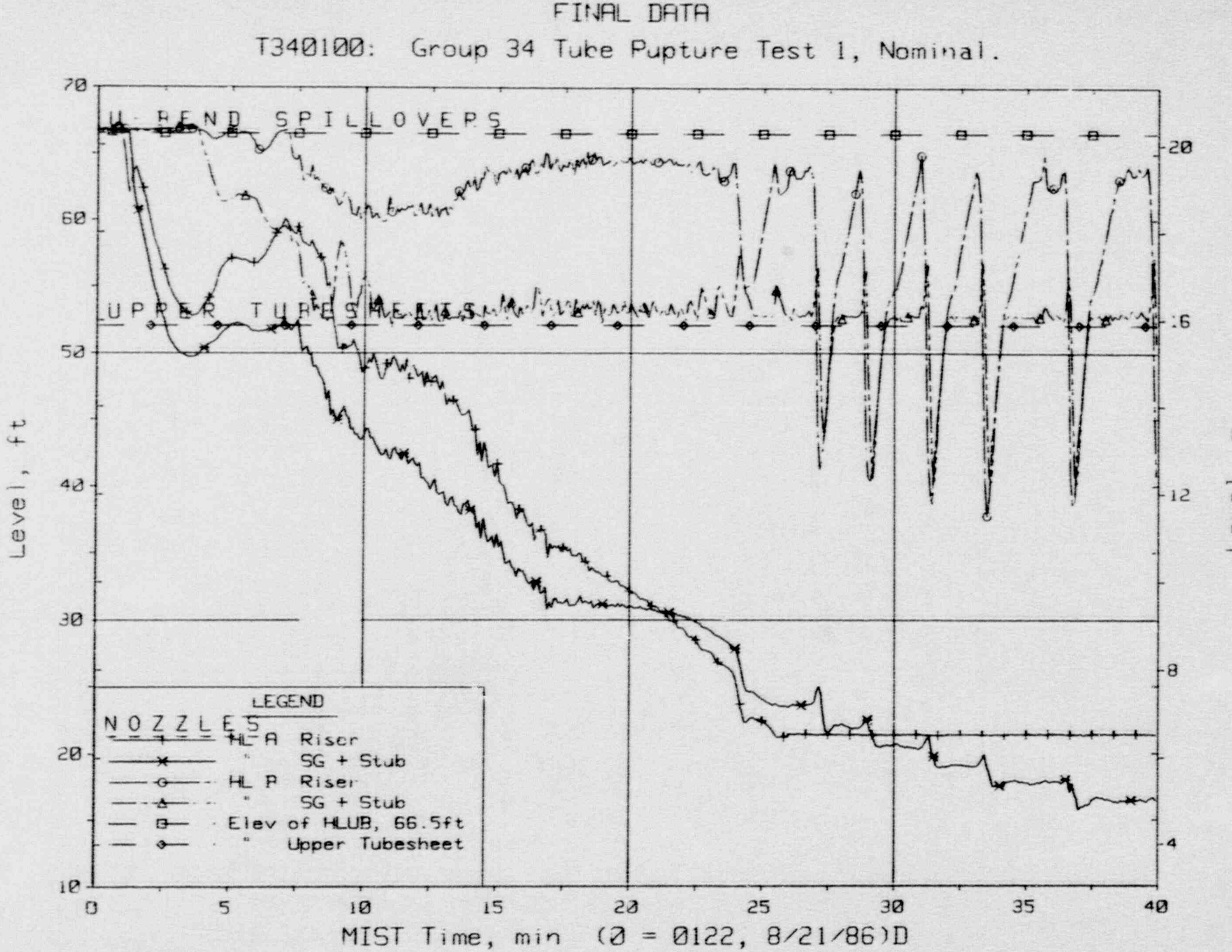

Figure 5.2.6. Hot Leg Riser and Stub Collapsed Liquid Levels

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HLLV1

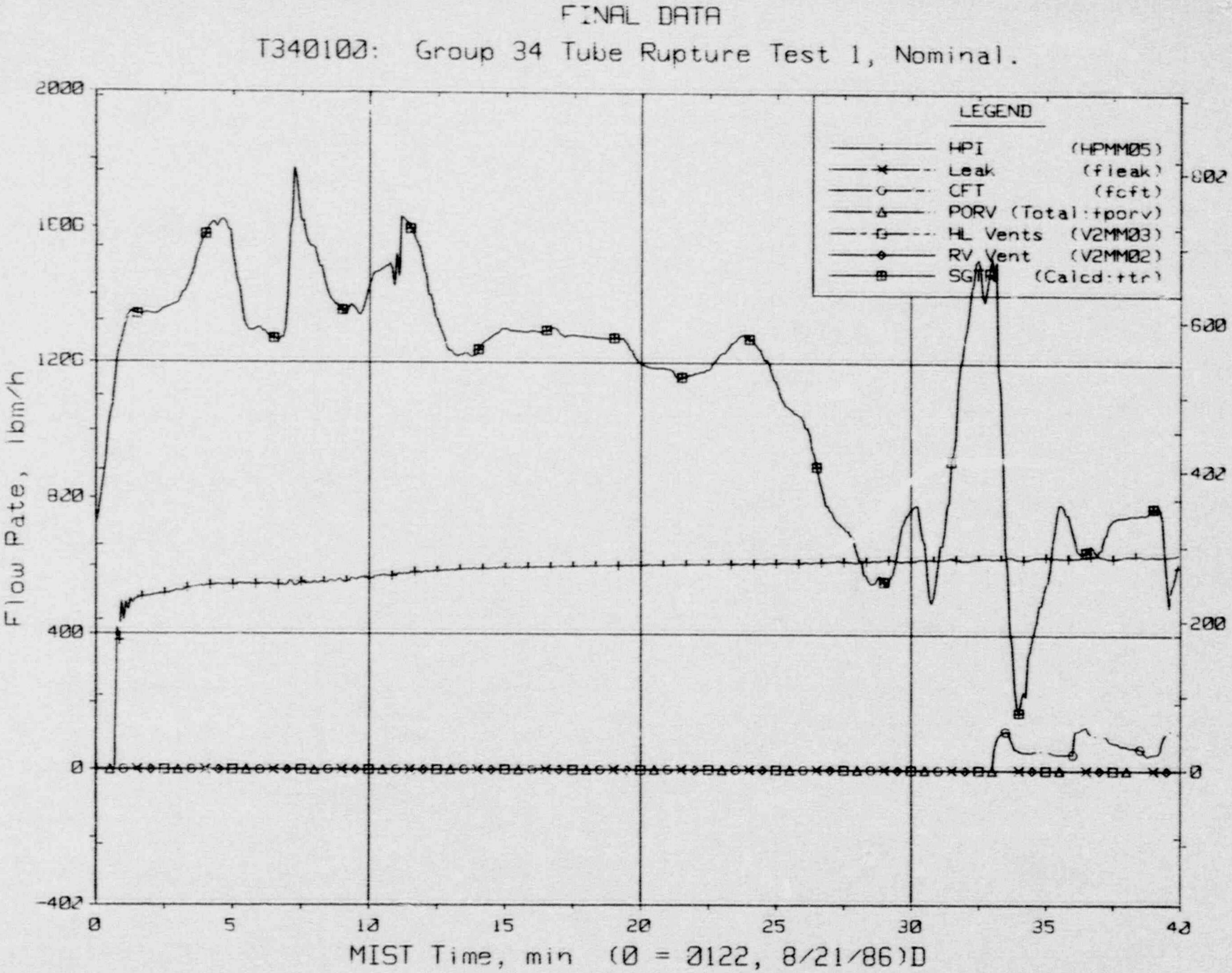

Figure 5.2.7. Primary System Boundary Flow Rates

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PRFL3

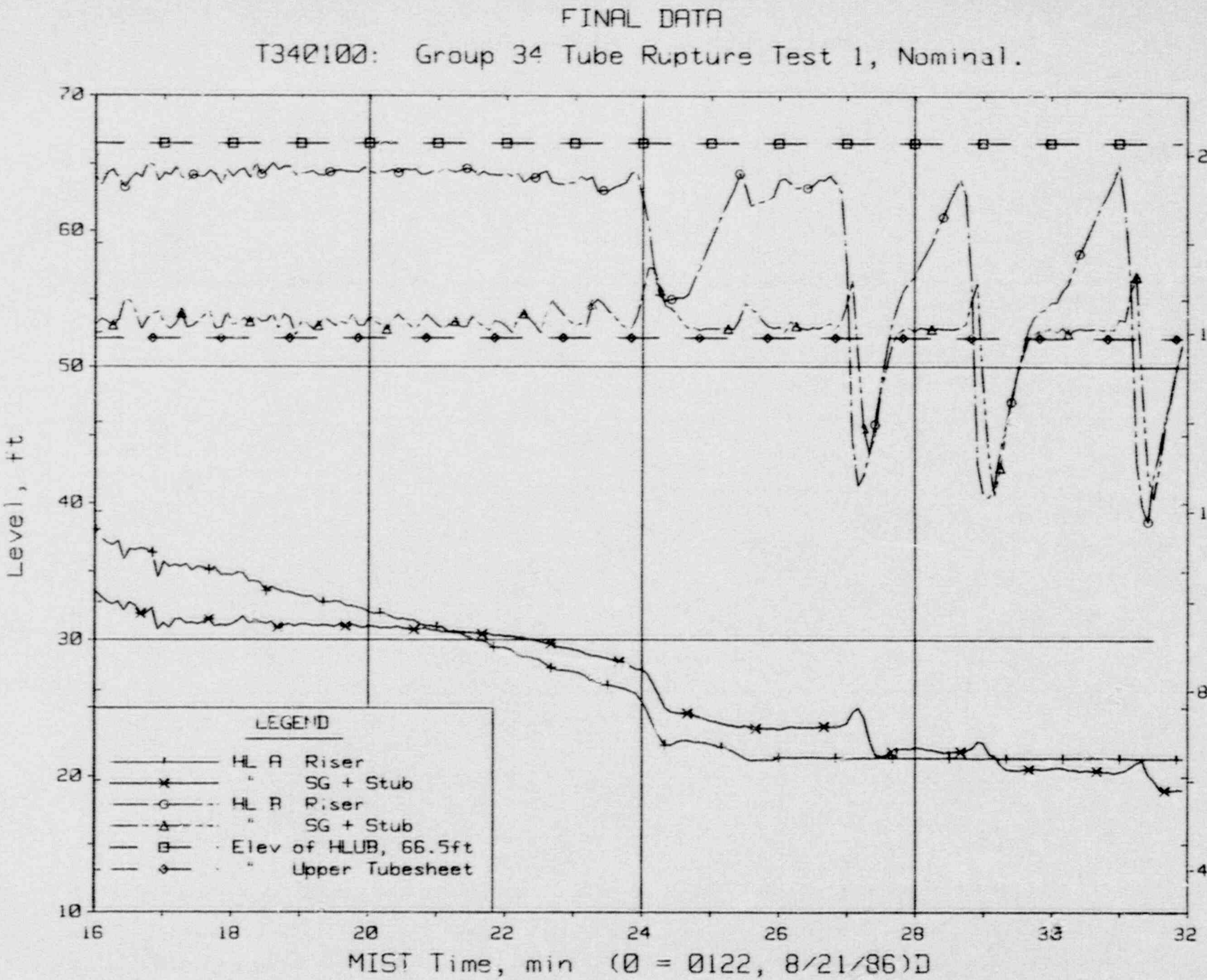

Figure 5.2.8. Hot Leg Riser and Stub Collapsed Liquid Levels

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HLLV1

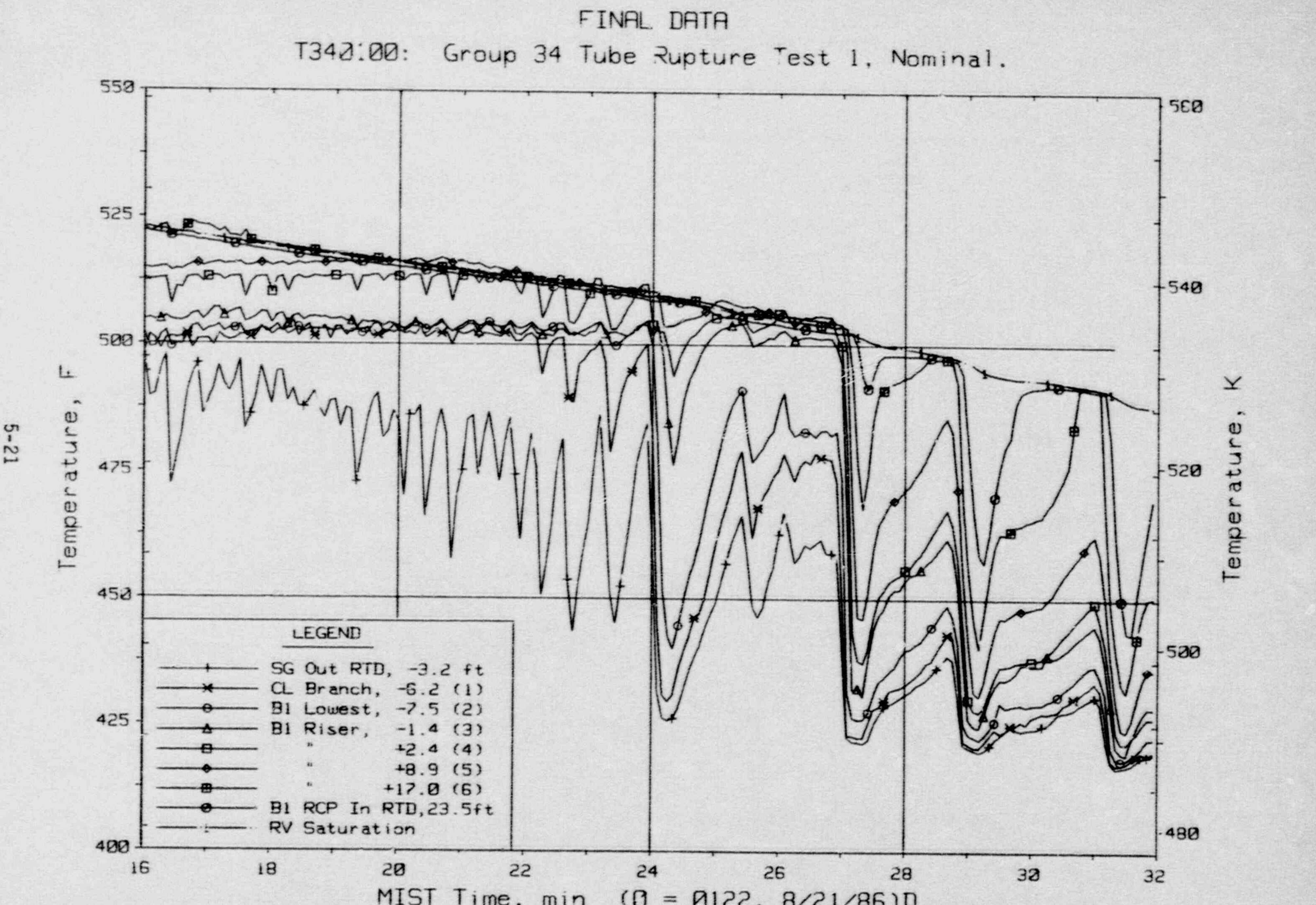

Figure 5.2.9. Cold Leg B1 Suction Fluid Temperatures (C2TCs)

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C2TC1

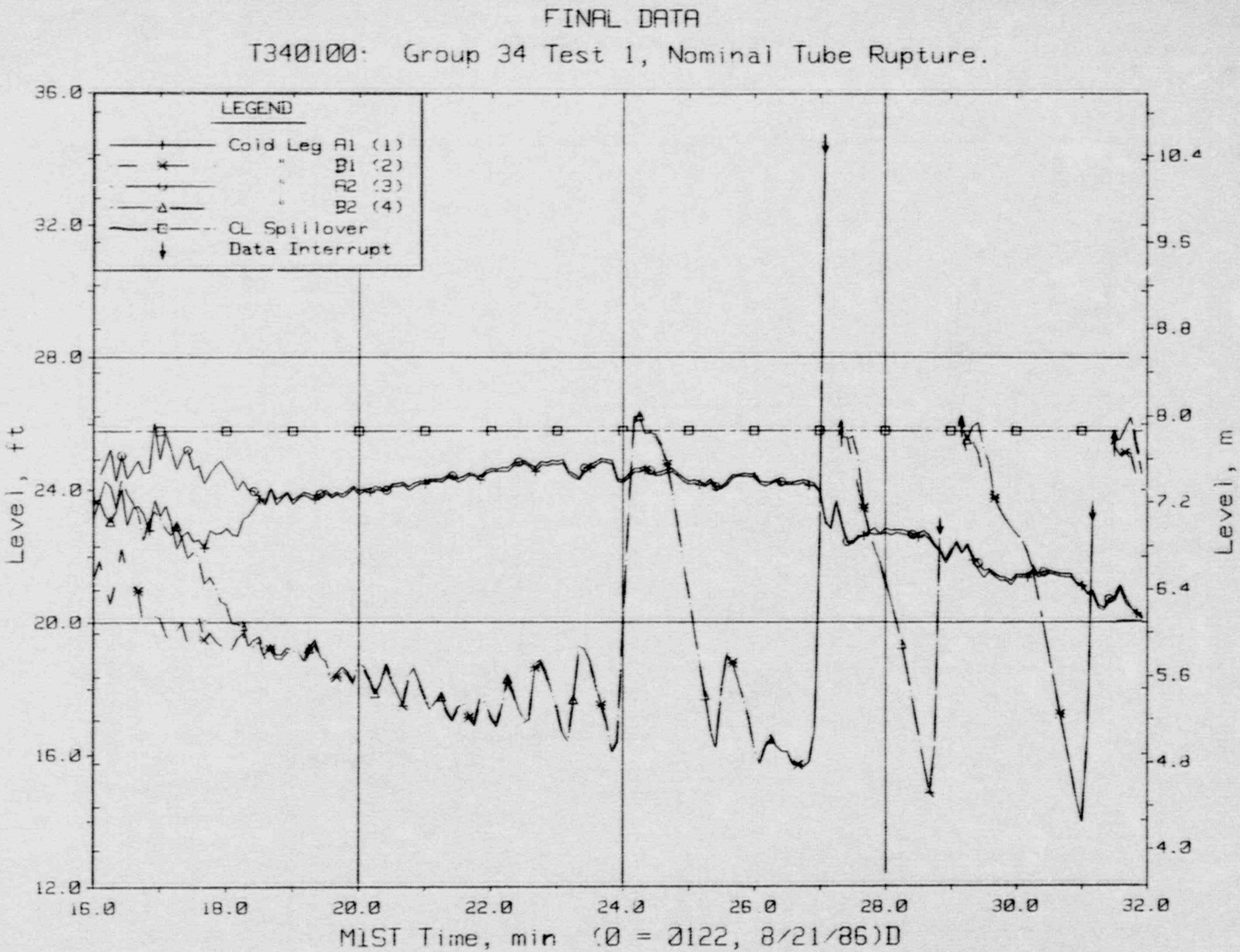

Figure 5.2.10. Cold Leg Suction Collapsed Liquid Levels (CNLV22s)

Wed Jan 4 10:26:20 1989

CLLV4m

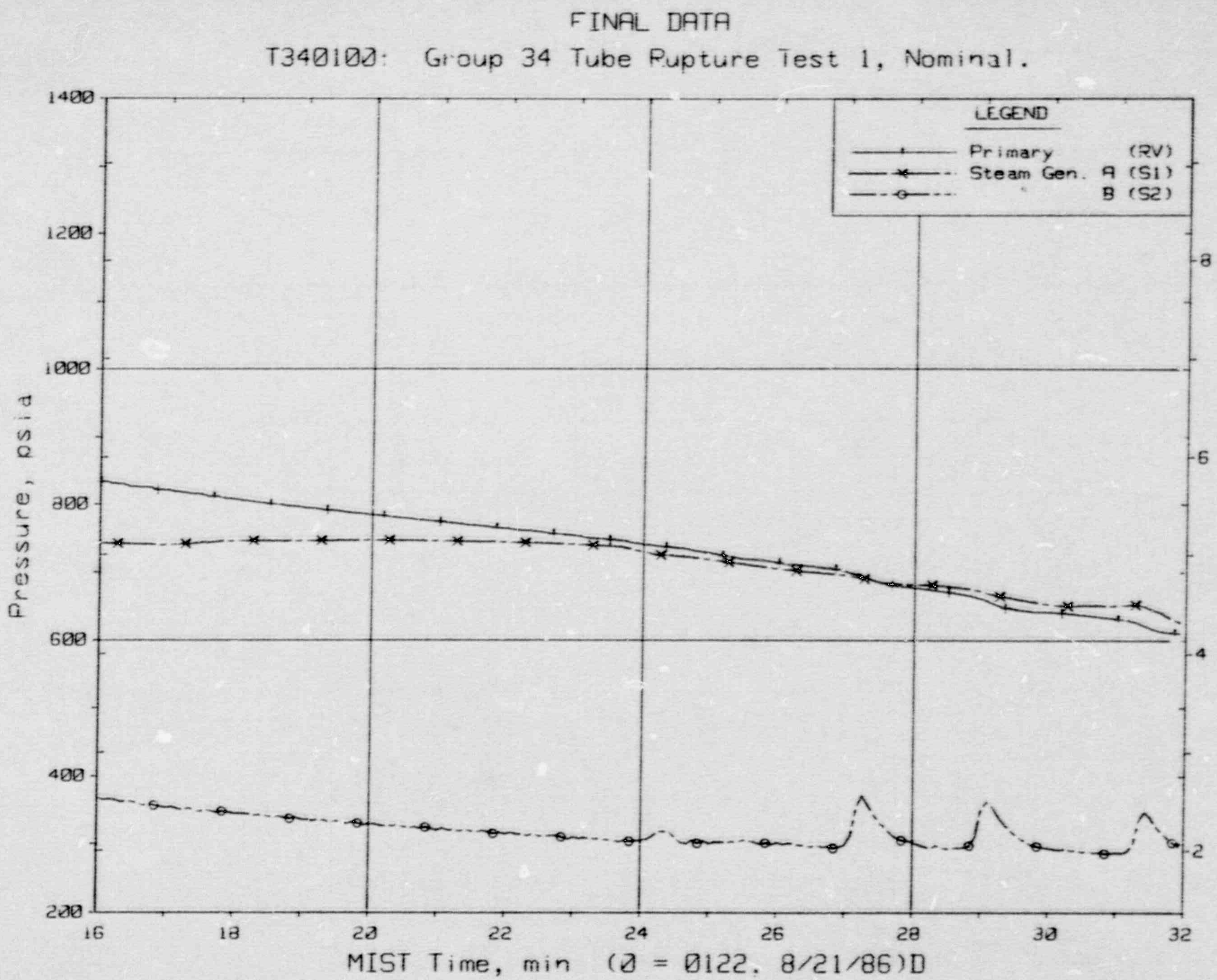

Figure 5.2.11. Primary and Secondary System Pressures (GP01s)

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PSGP1

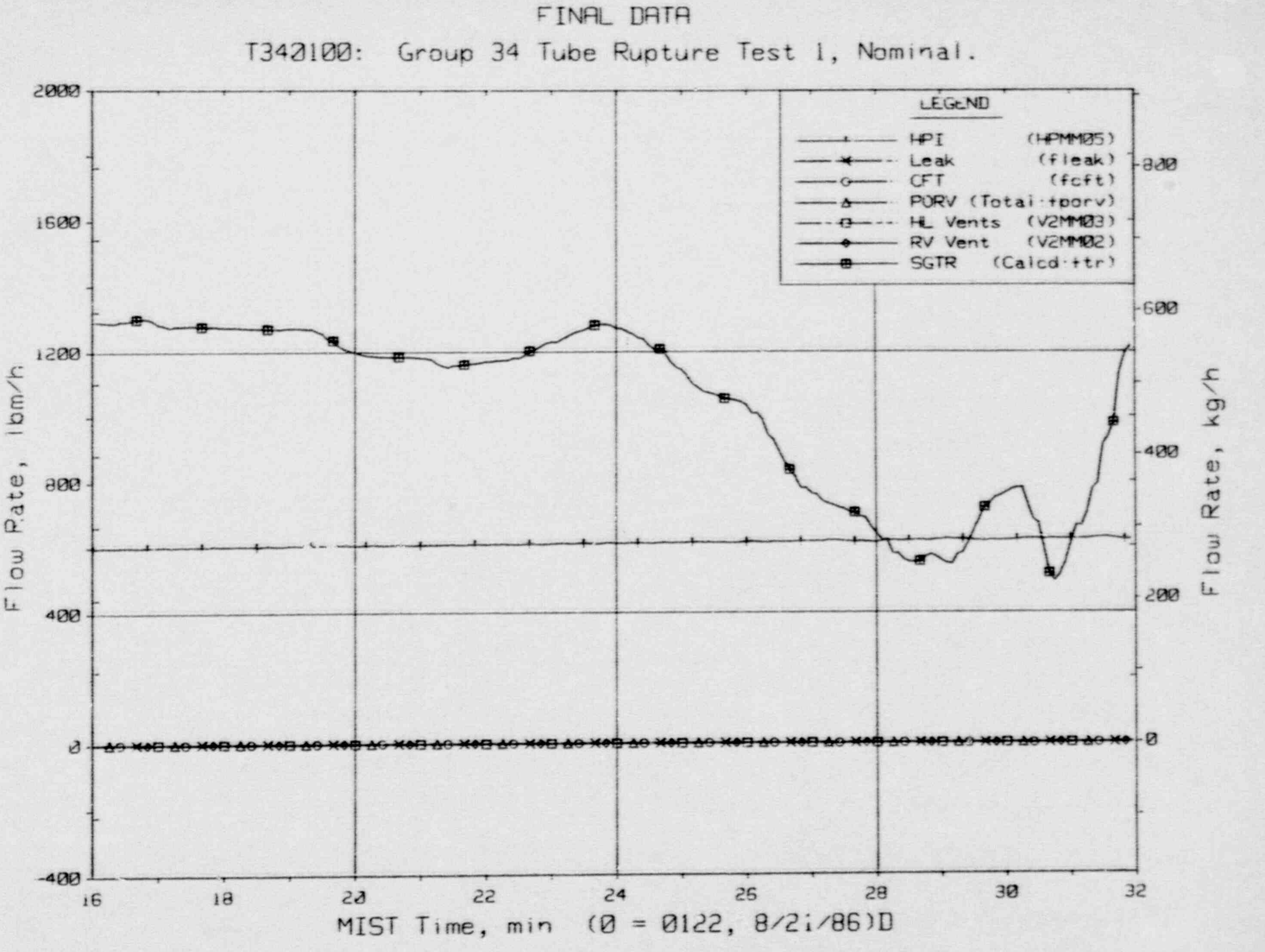

Figure 5.2.12. Primary System Boundary Flow Rates

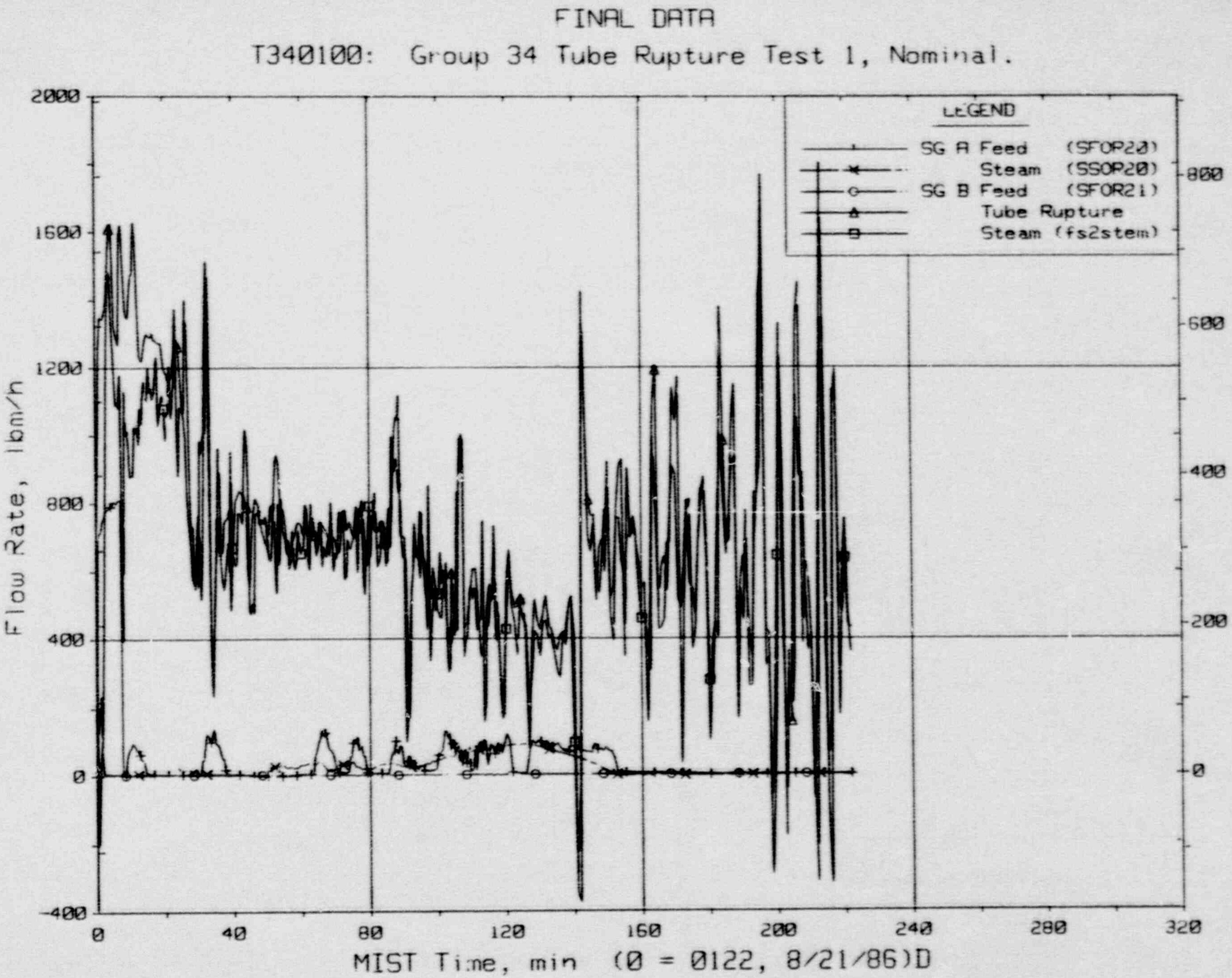

Figure 5.2.13. Secondary System Flow Rates

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SGFL1

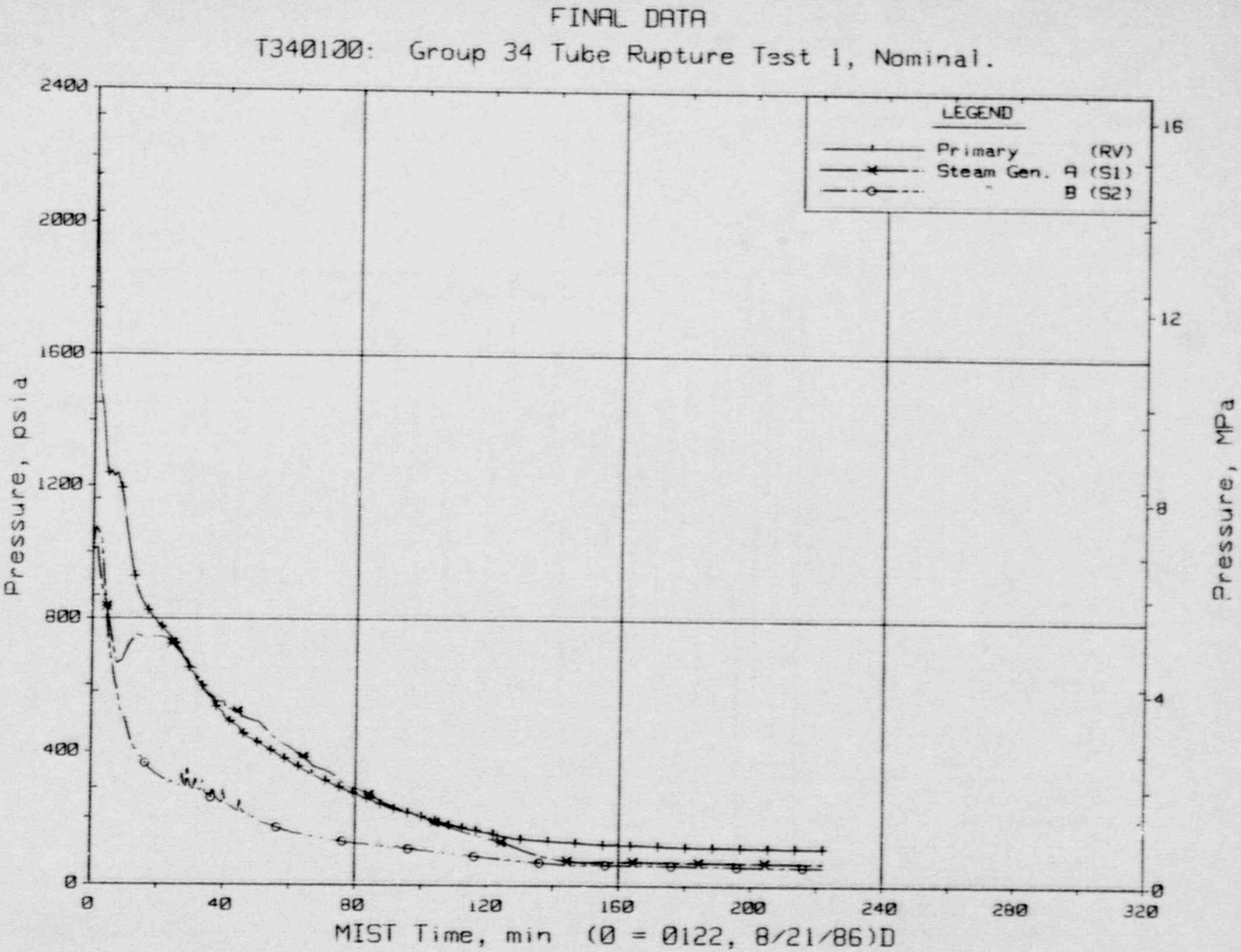

Figure 5.2.14. Primary and Secondary System Pressures (GP01s)

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PSGP1

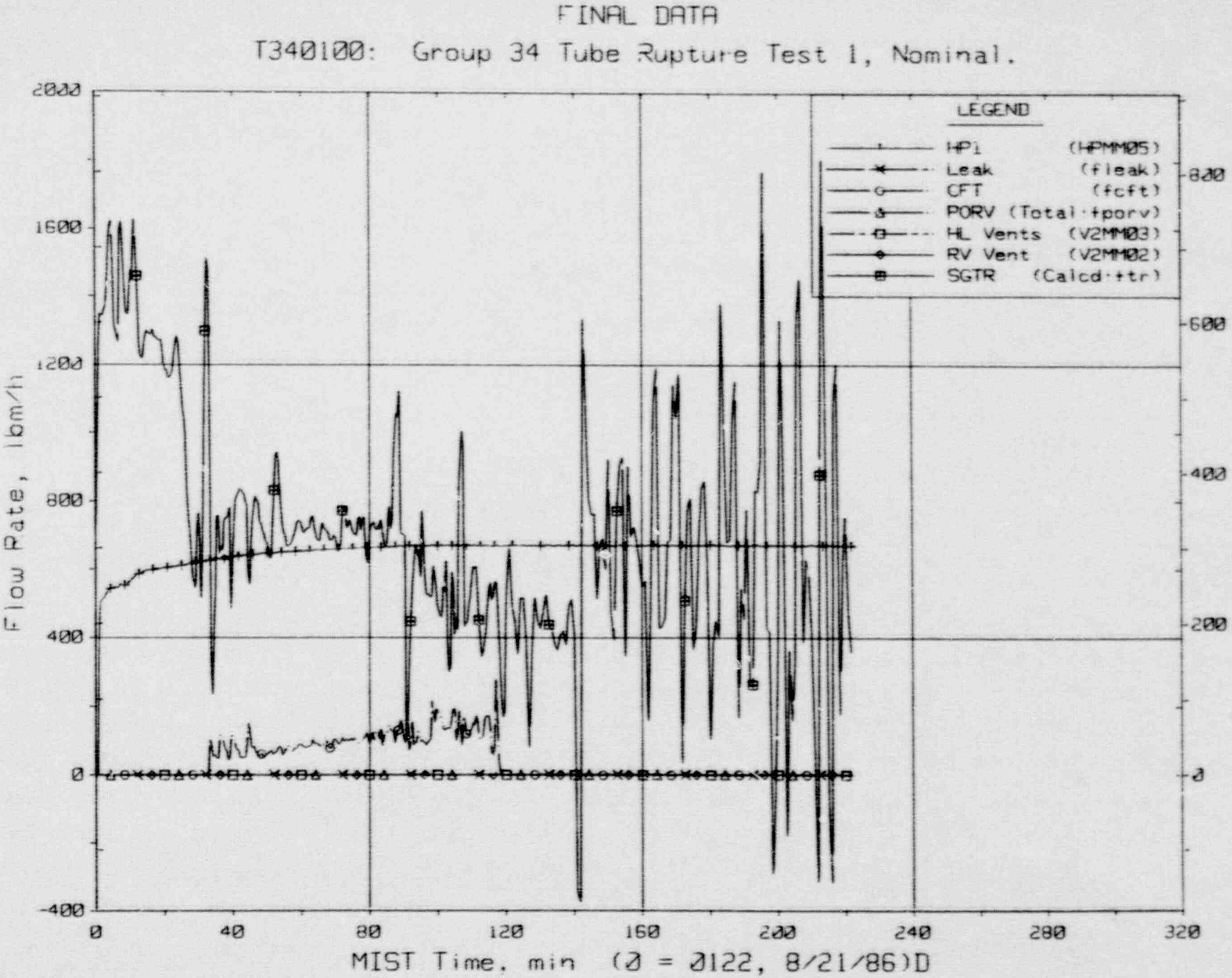

Figure 5.2.15. Primary System Boundary Flow Rates

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PRFL3

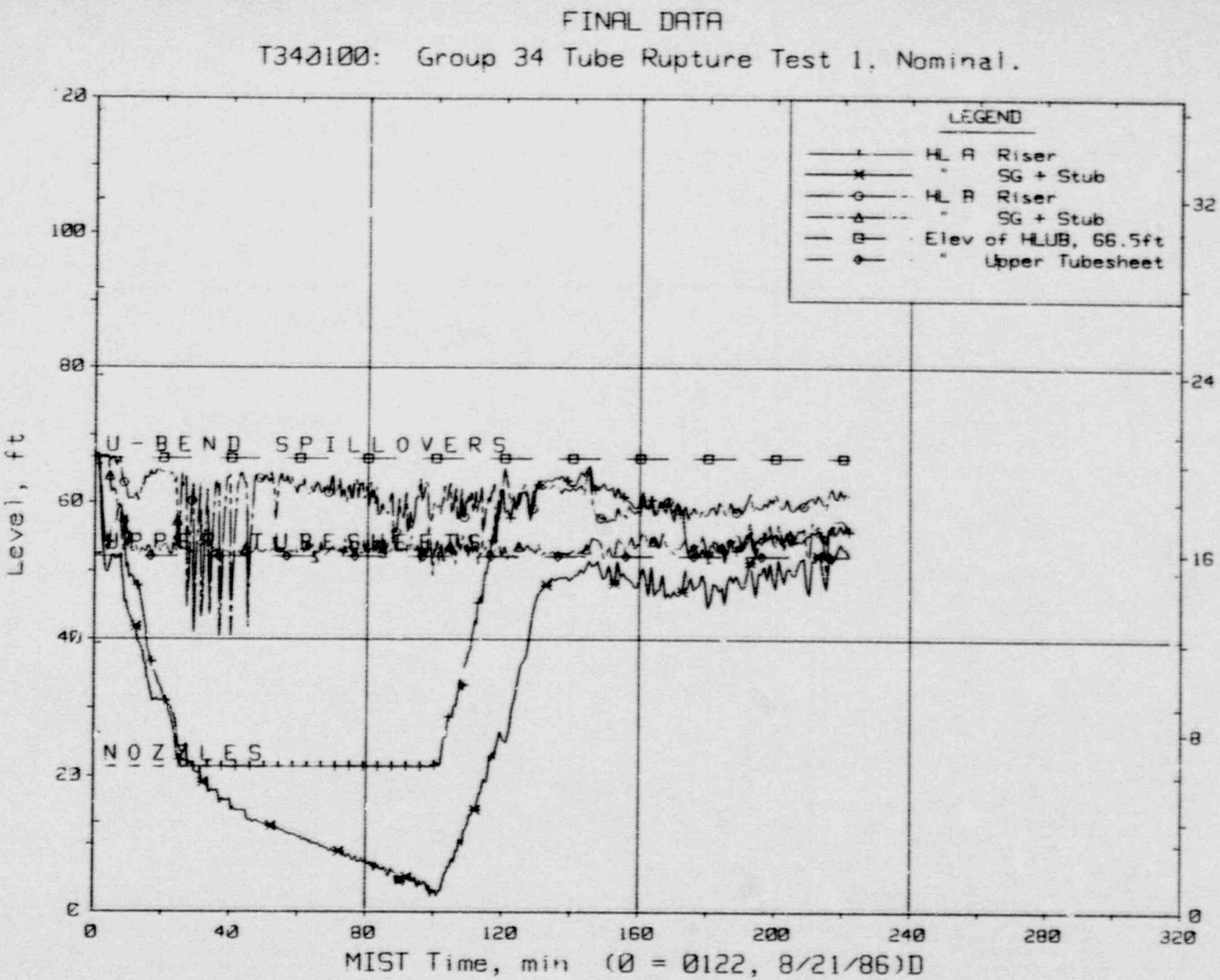

Figure 5.2.16. Hot Leg Riser and Stub Collapsed Liquid Levels

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HLLV1

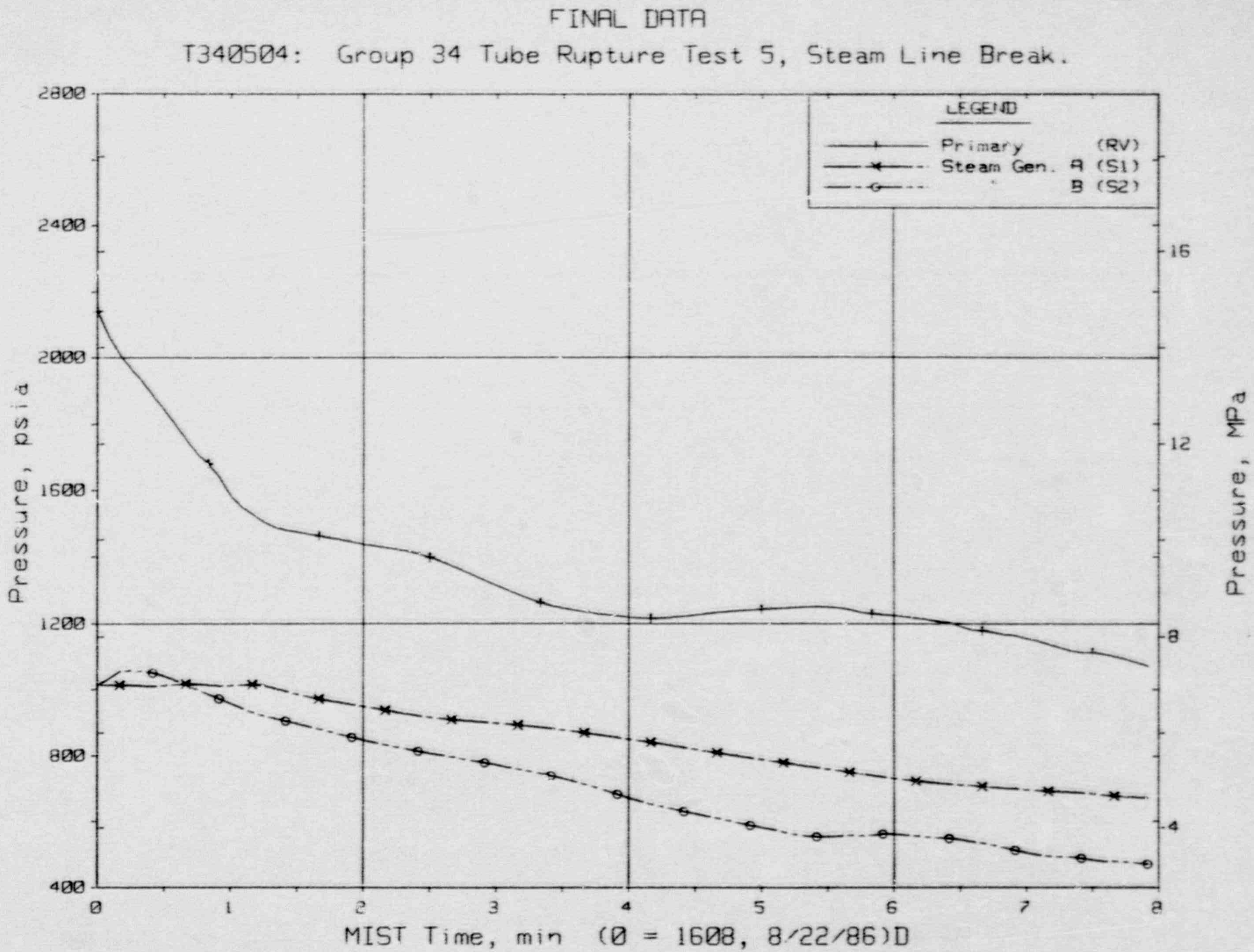

Figure 5.2.17. Primary and Secondary System Pressures (GP01s)

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PSGP1

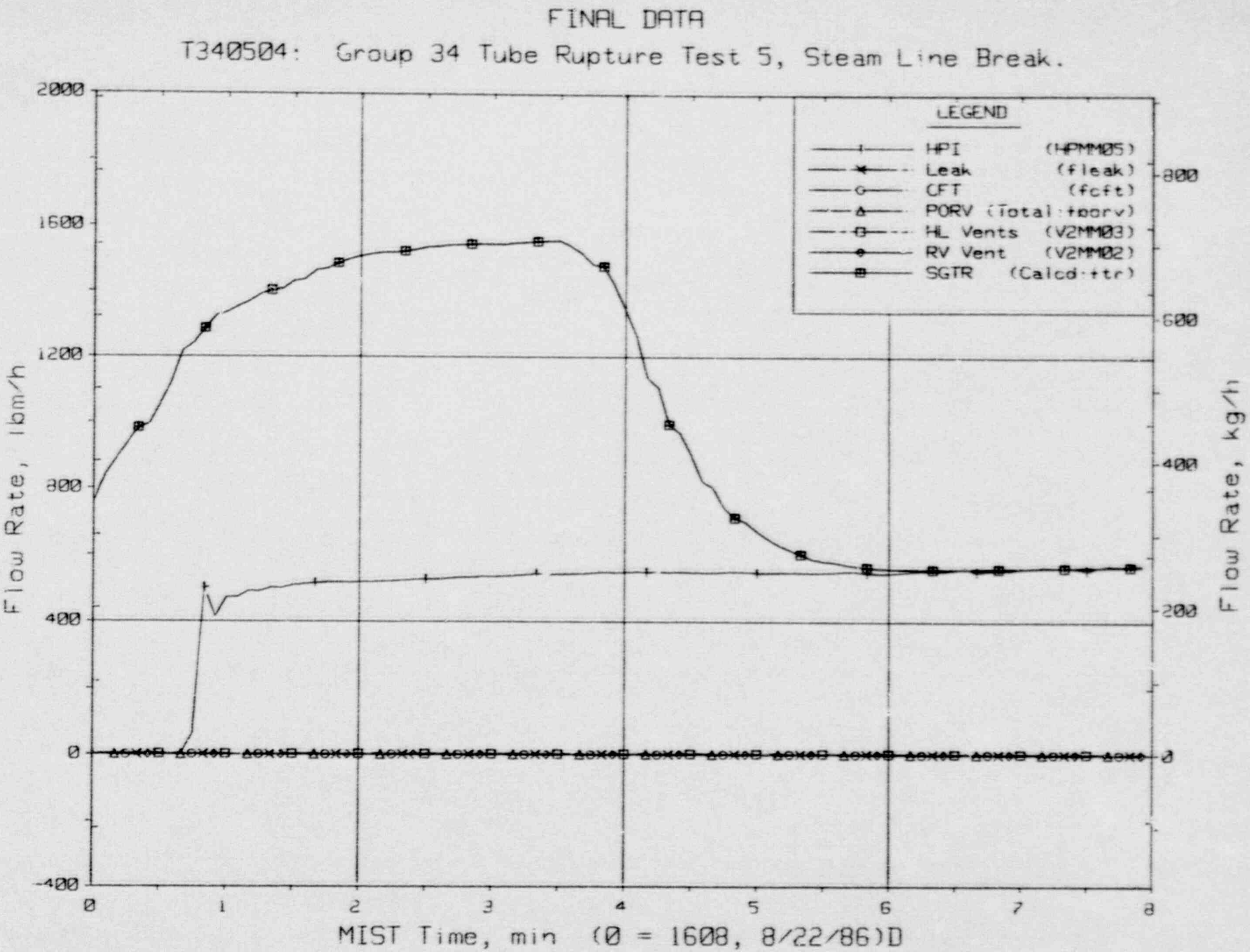

Figure 5.2.18. Primary System Boundary Flow Rates

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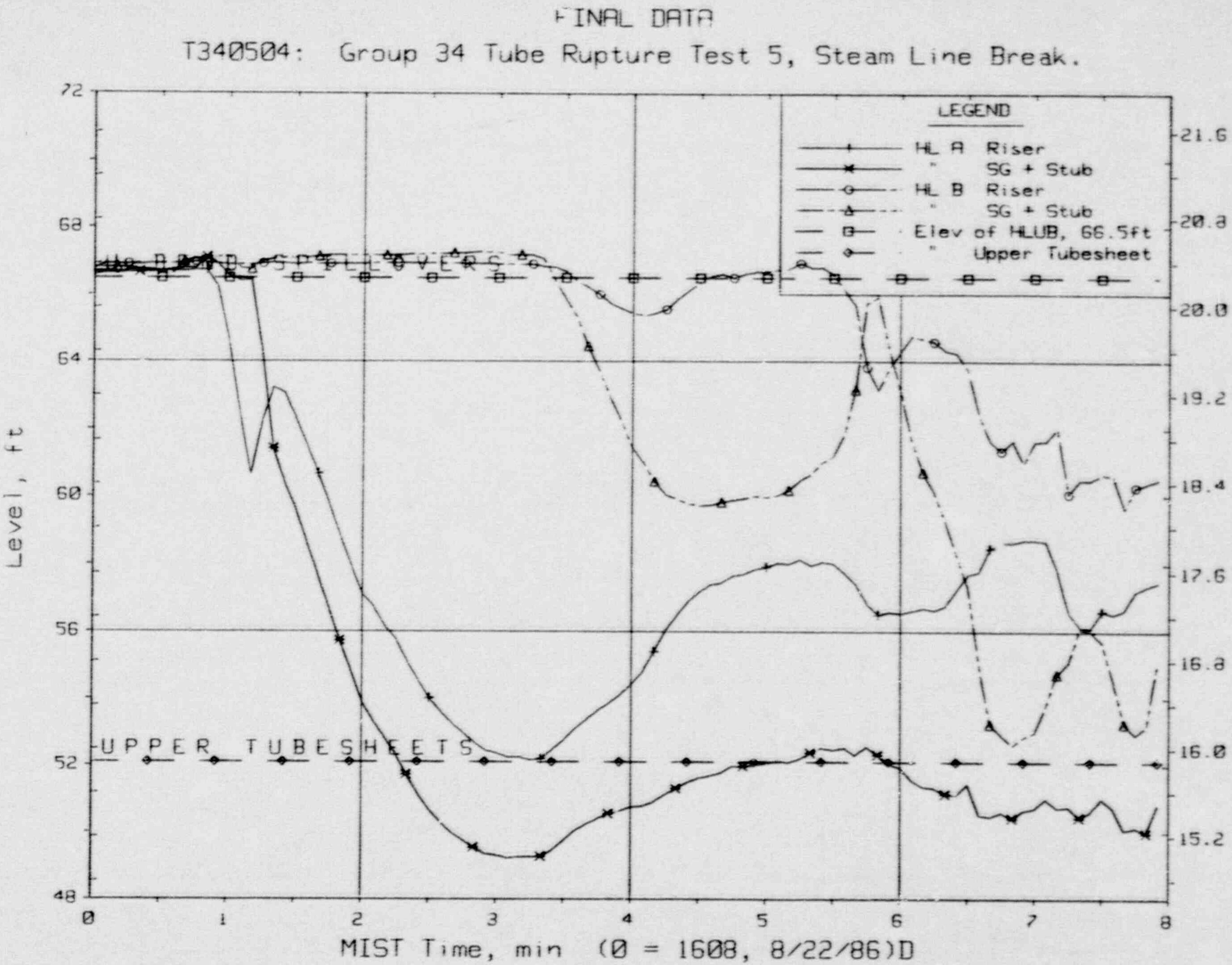

Figure 5.2.19. Hot Leg Riser and Stub Collapsed Liquid Levels

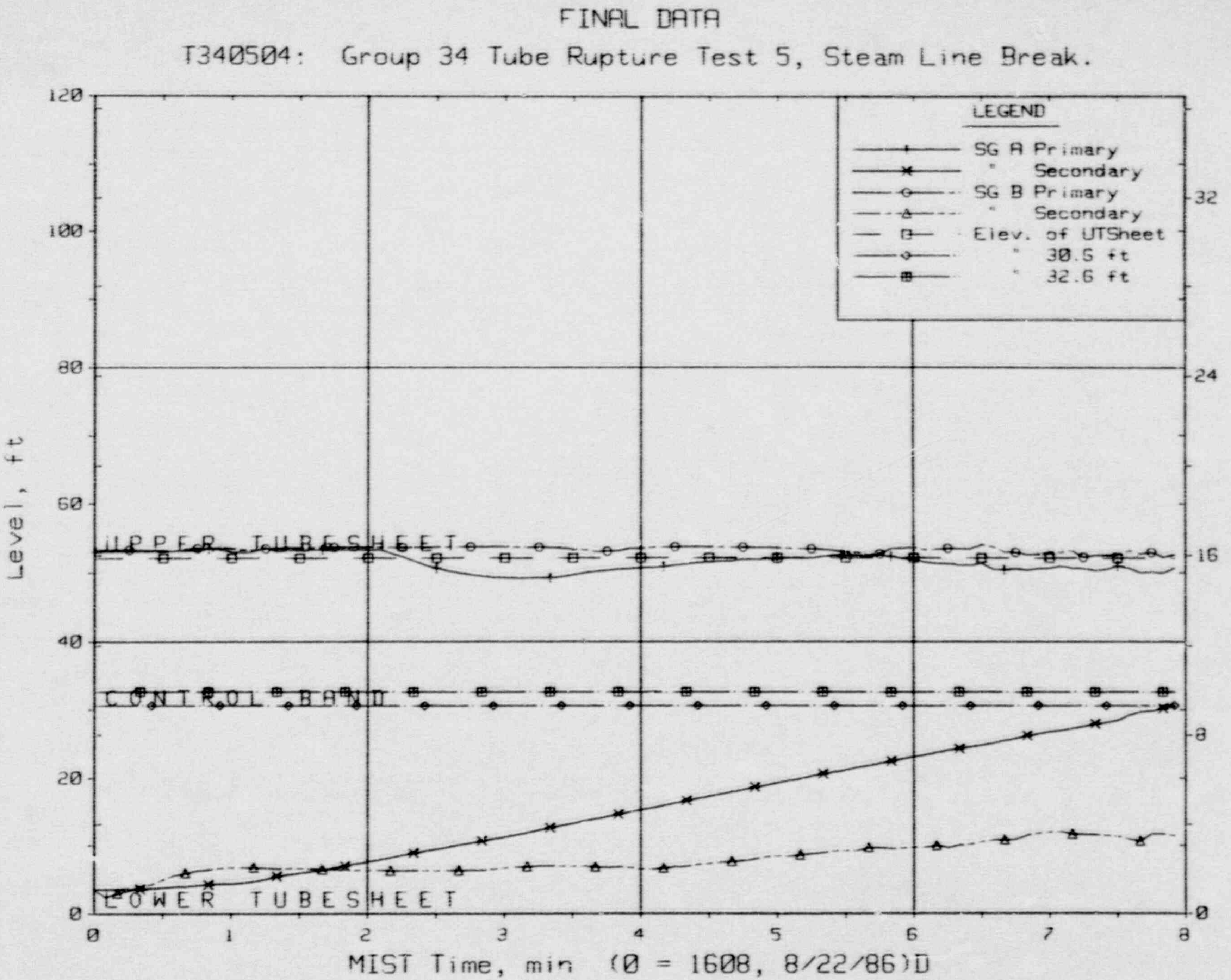

Figure 5.2.20. Steam Generator Collapsed Liquid Levels

Tue Nov 1 15:16:03 1988

SGLV2

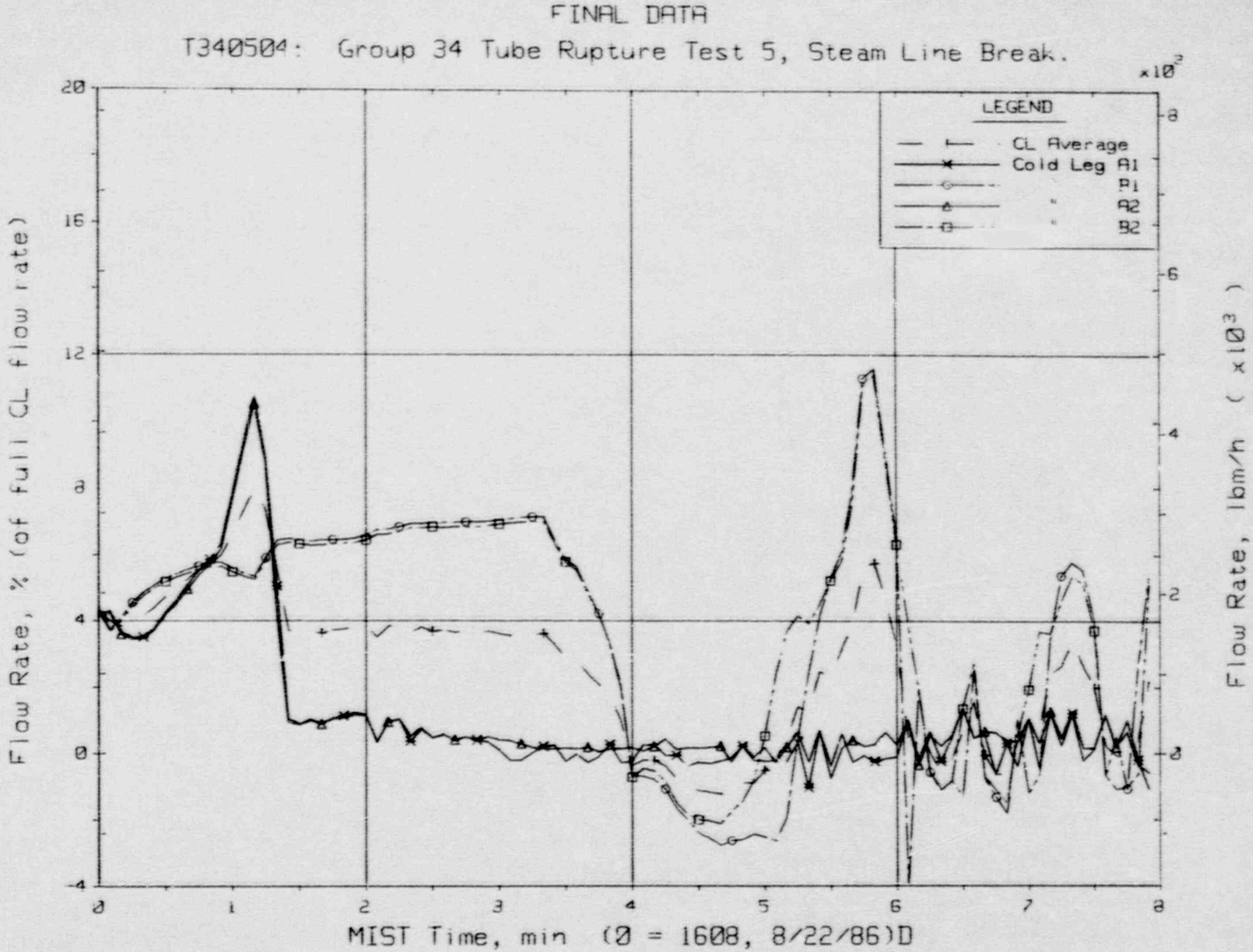

Figure 5.2.21. Cold Leg (Venturi) Flow Rates

Mon Jul 25 12:52:13 1998

CLFL1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.

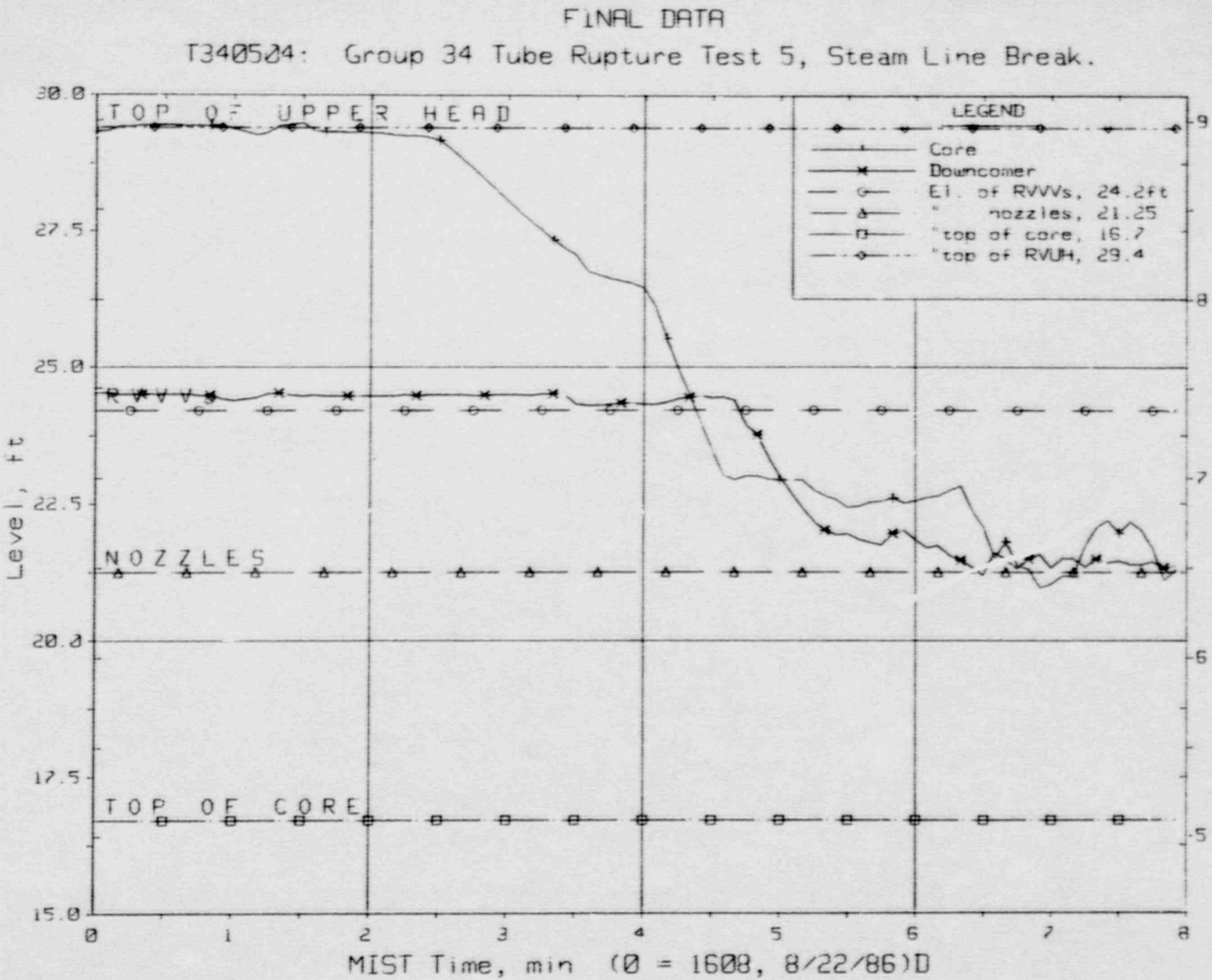

Figure 5.2.22. Core Region Collapsed Liquid Levels

Mon Jul 25 12:54:54 1988

COLV1

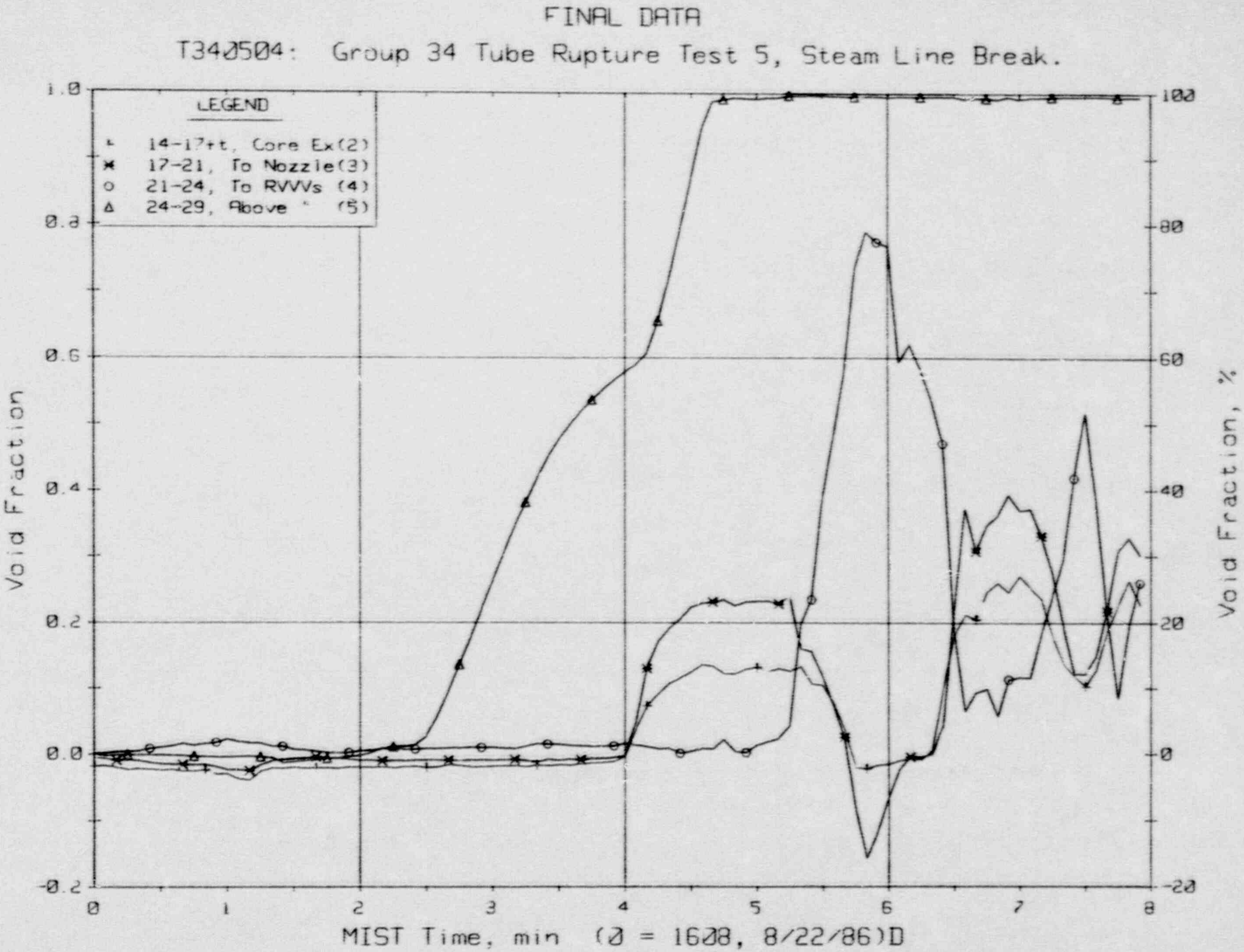

Figure 5.2.23. Reactor Vessel Void Fractions from Differential Pressures (RVVFs)

RVVF1

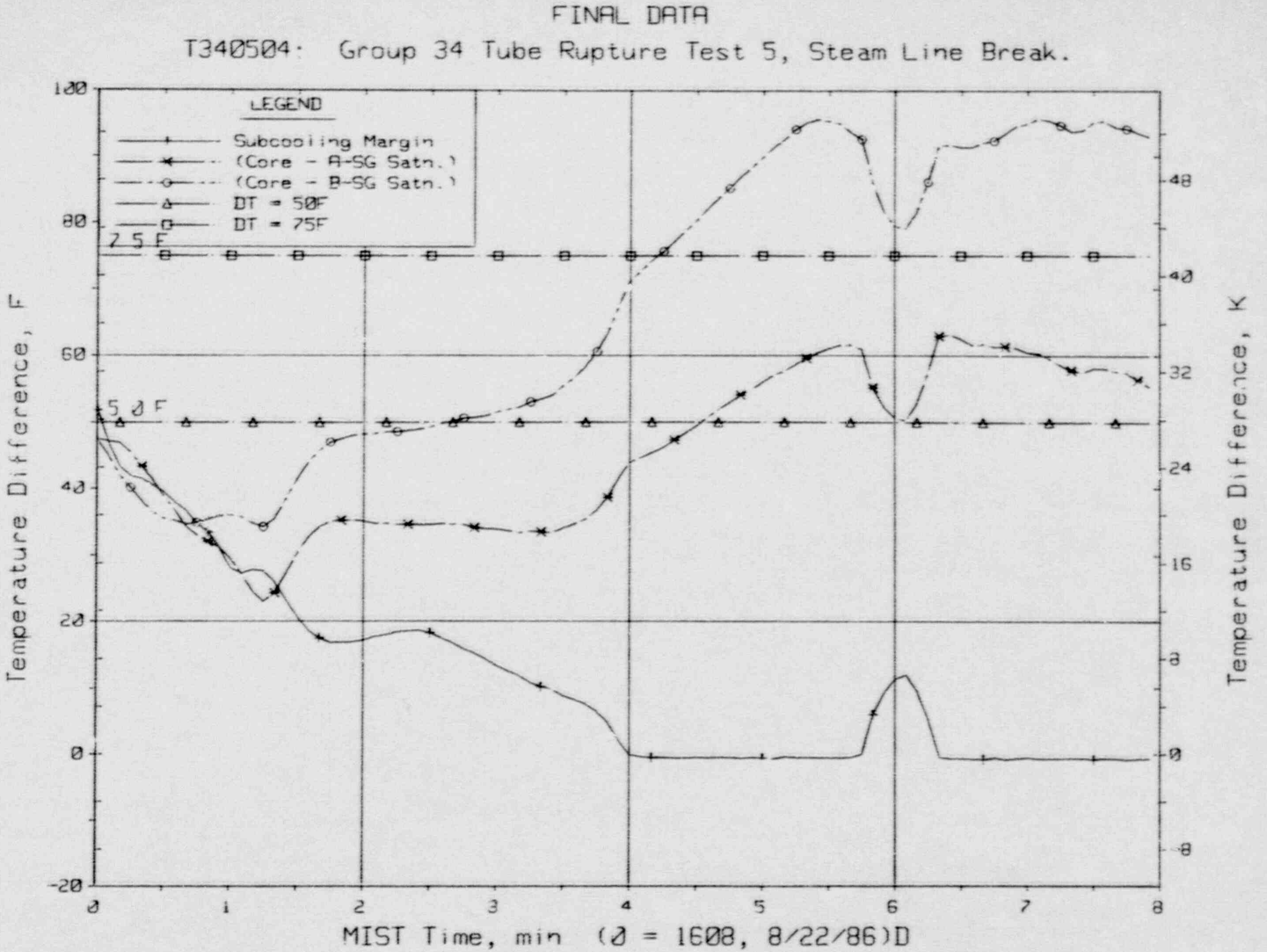

Figure 5.2.24. Control Temperature Differences

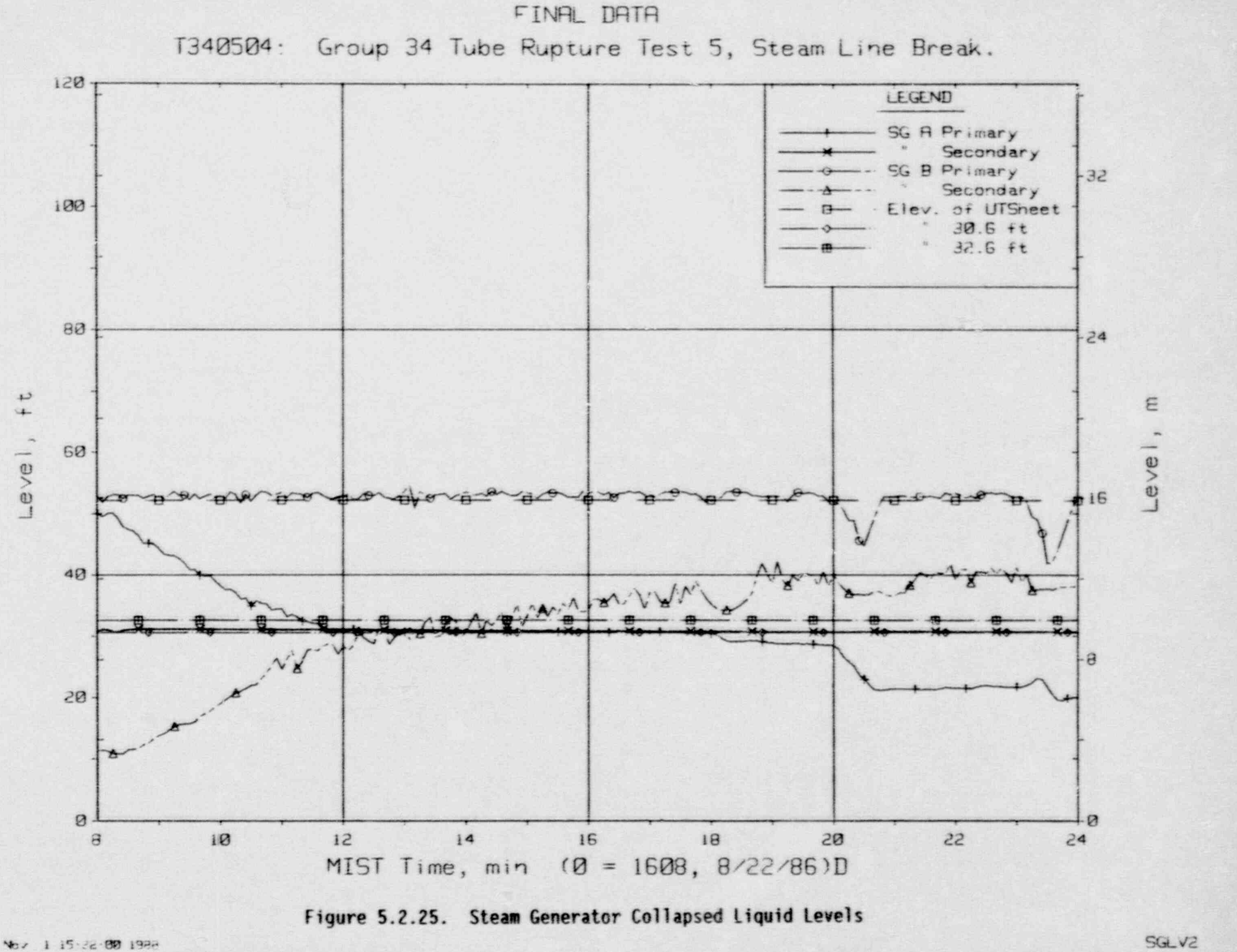

Figure 5.2.25. Steam Generator Collapsed Liquid Levels

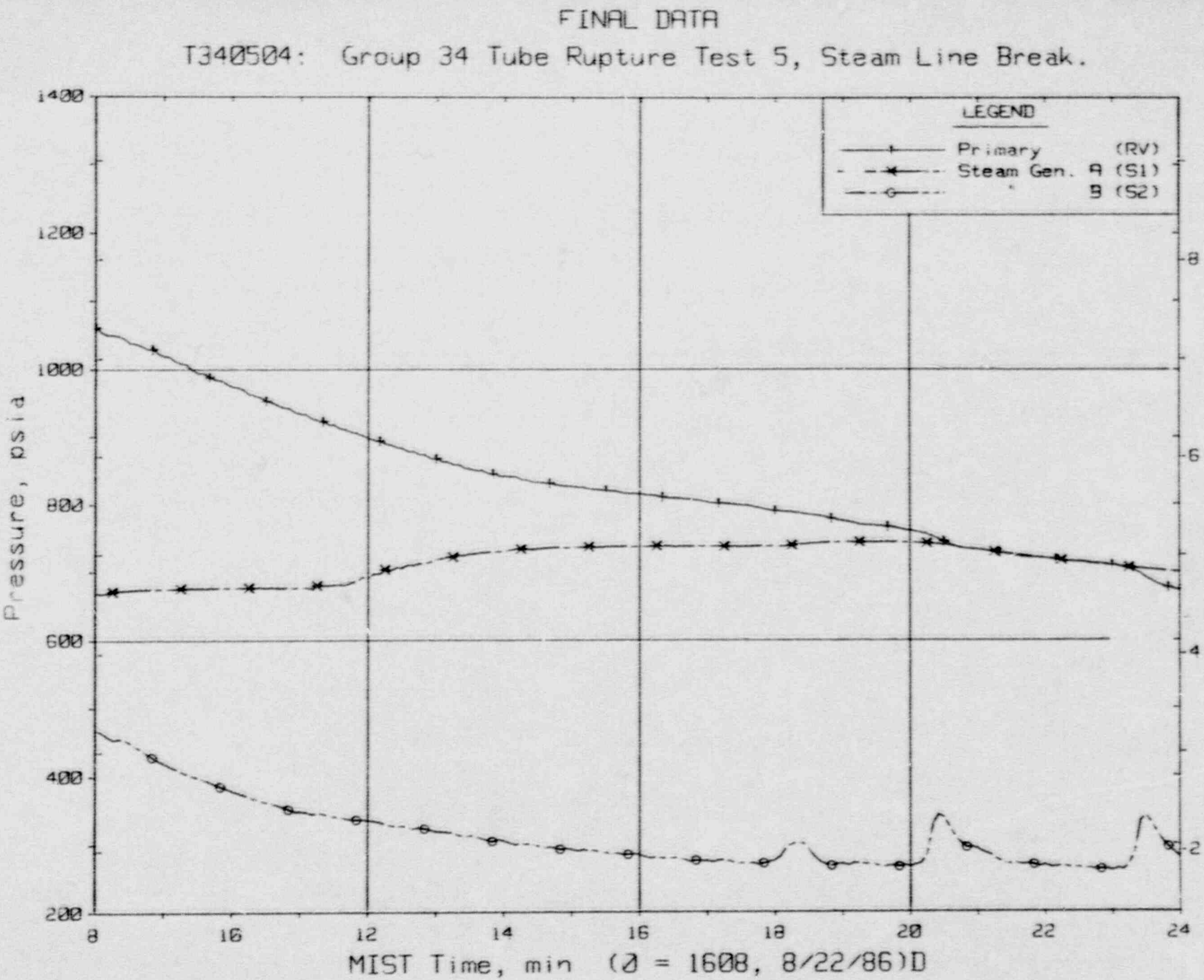

Figure 5.2.26. Primary and Secondary System Pressures (GPOIs)

Mon Jul 25 12:06:02 1988

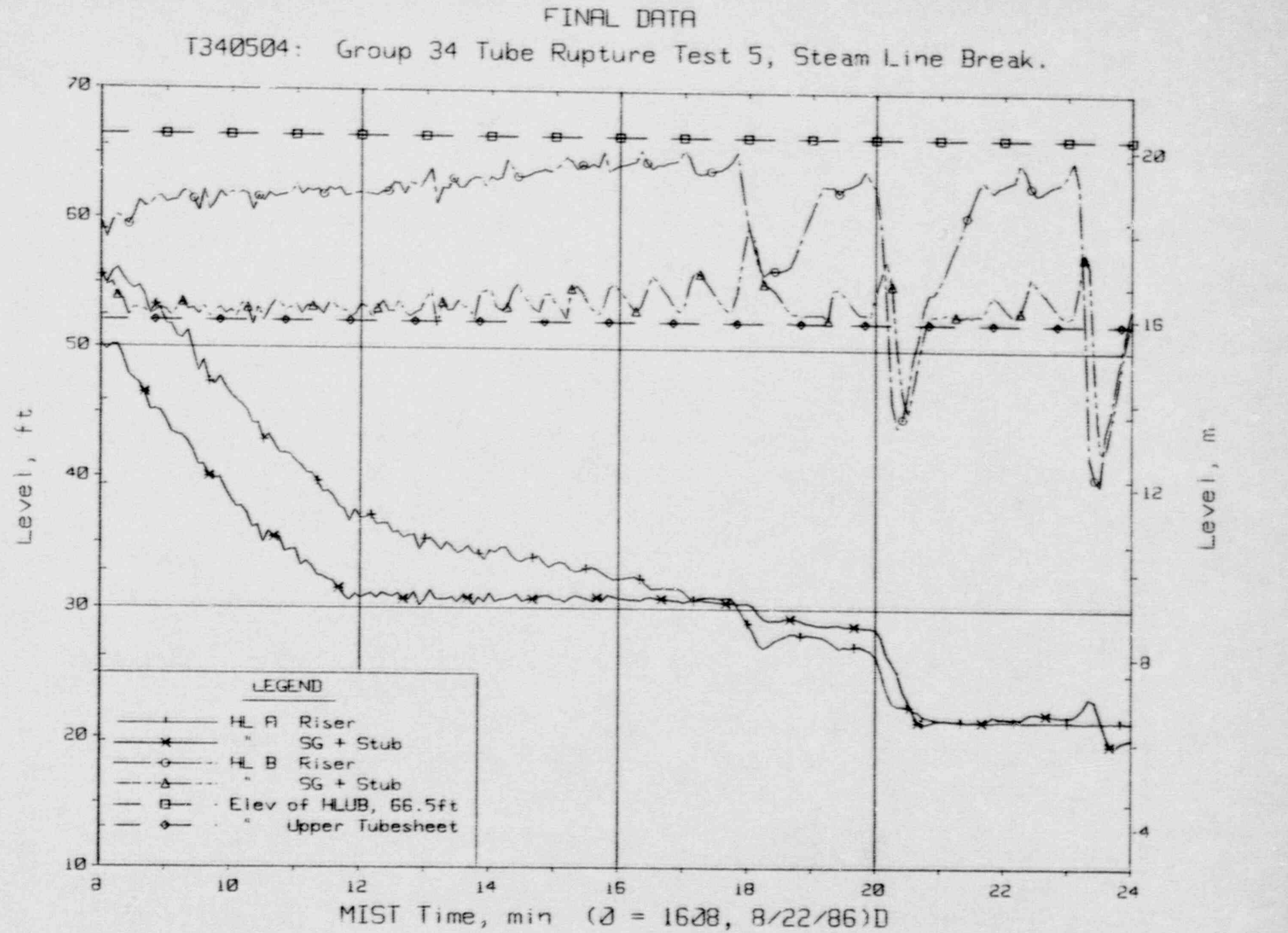

Figure 5.2.27. Hot Leg Riser and Stub Collapsed Liquid Levels

Mon Jul 25 13:08:52 1988

HLLV1

FINAL DATA

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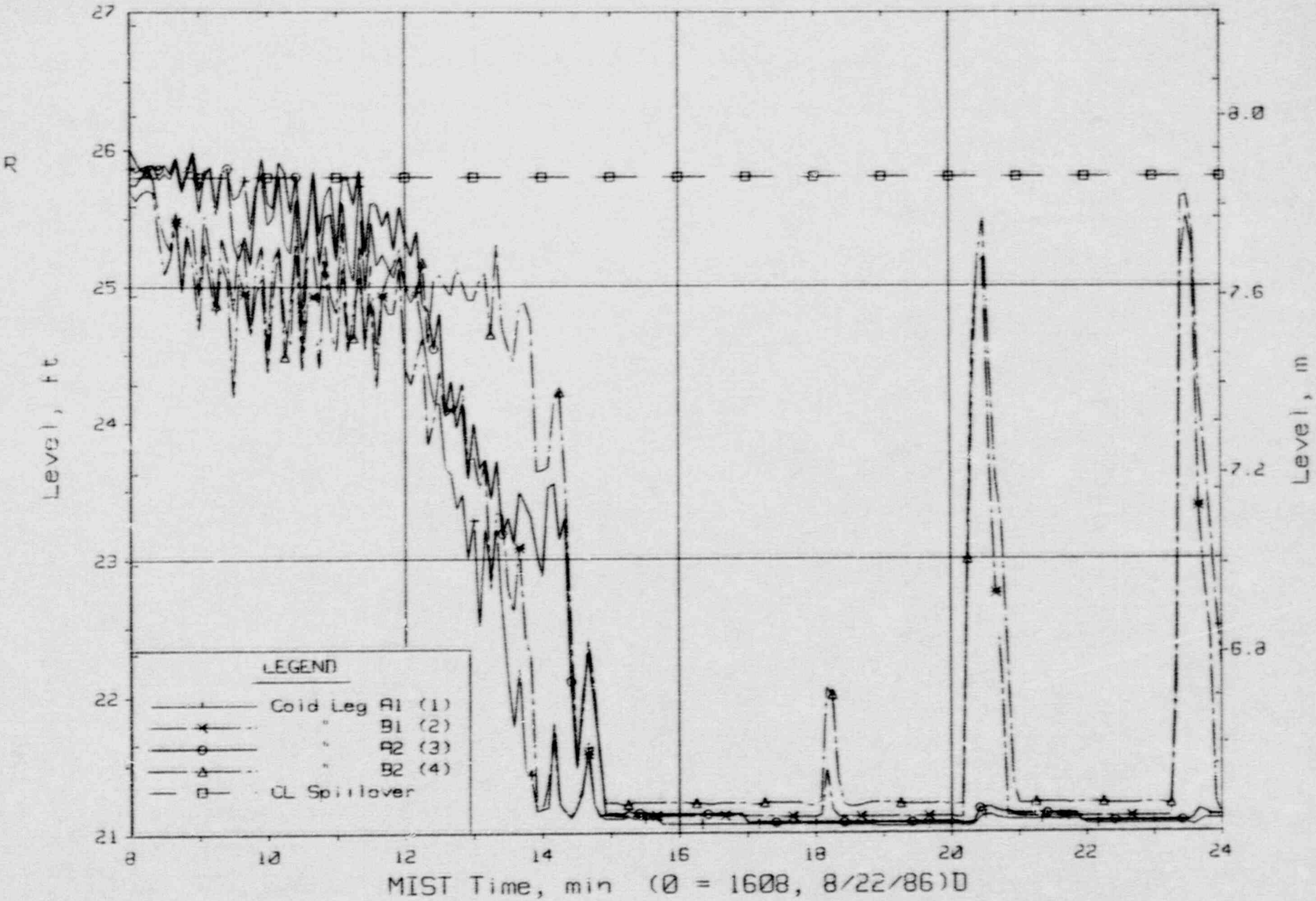


Figure 5.2.28. Cold Leg Discharge Collapsed Liquid Levels (CnLV23s)

CLLV5

Mon Jul 25 13:11:52 1988

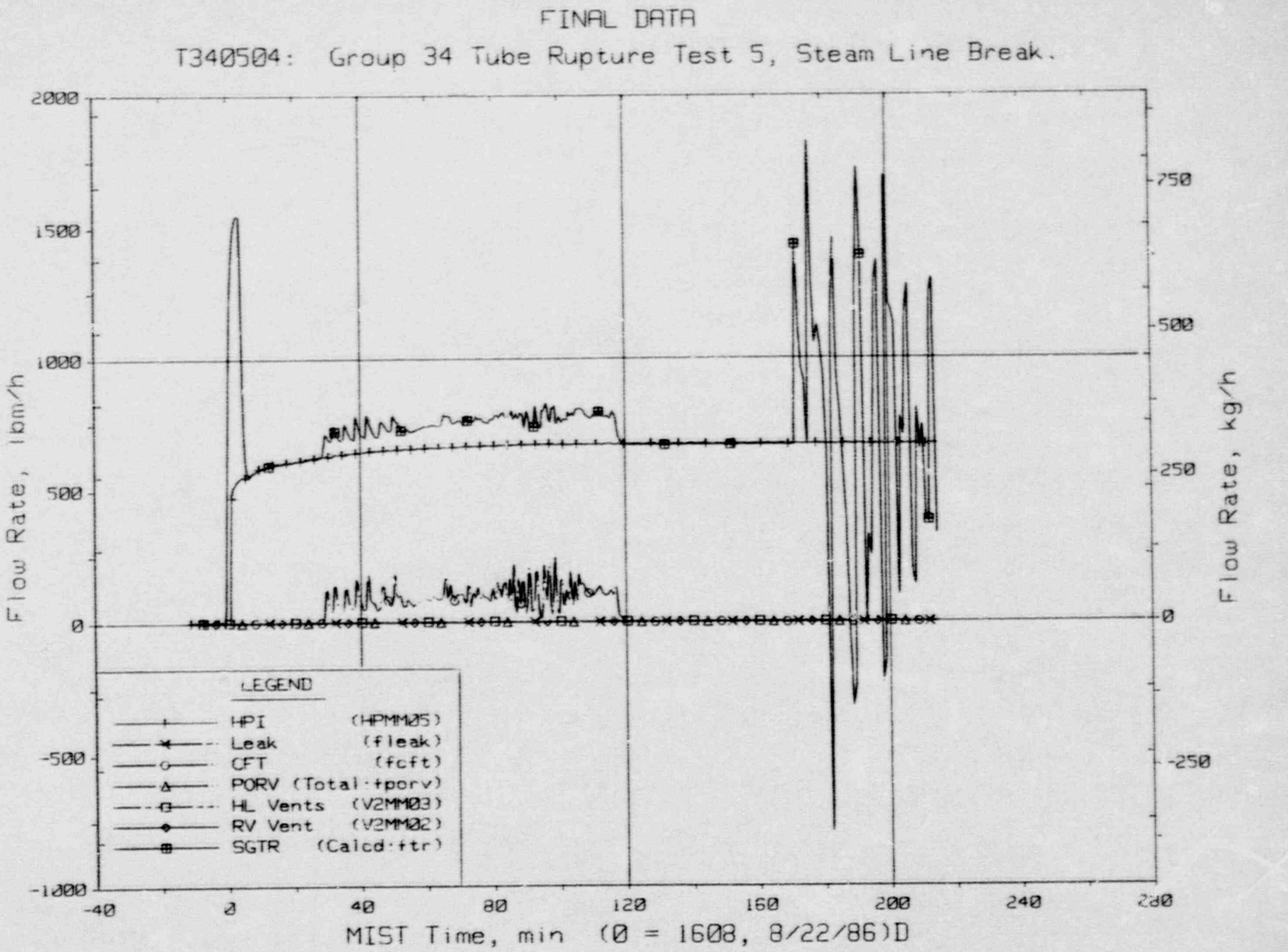


Figure 5.2.29. Primary System Boundary Flow Rates

Mon Jul 25 13:19:19 1988

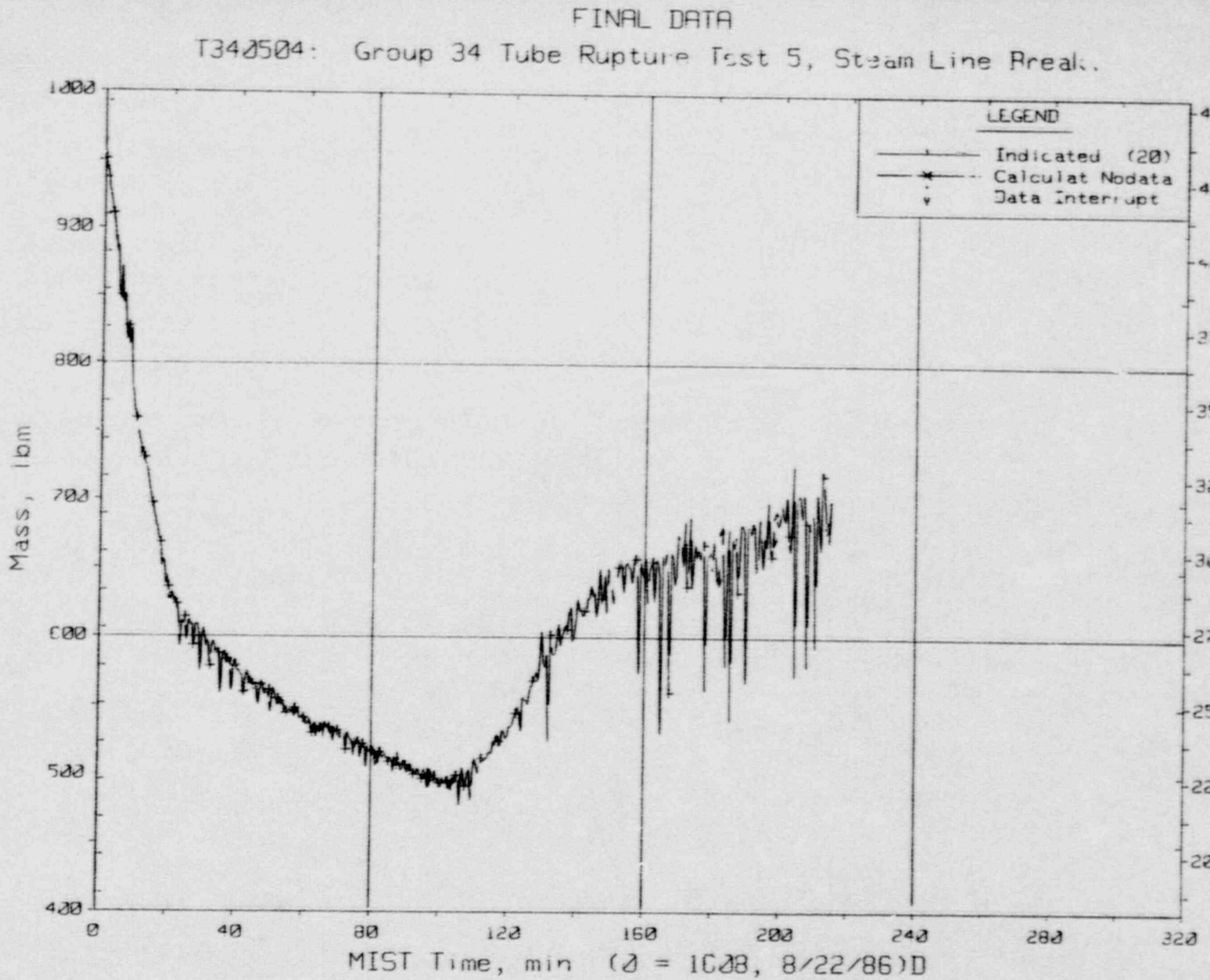


Figure 5.2.30. Primary System Total (Indicated, Trended) Fluid Mass

Tue Nov 1 12:42:23 1988

PPML1

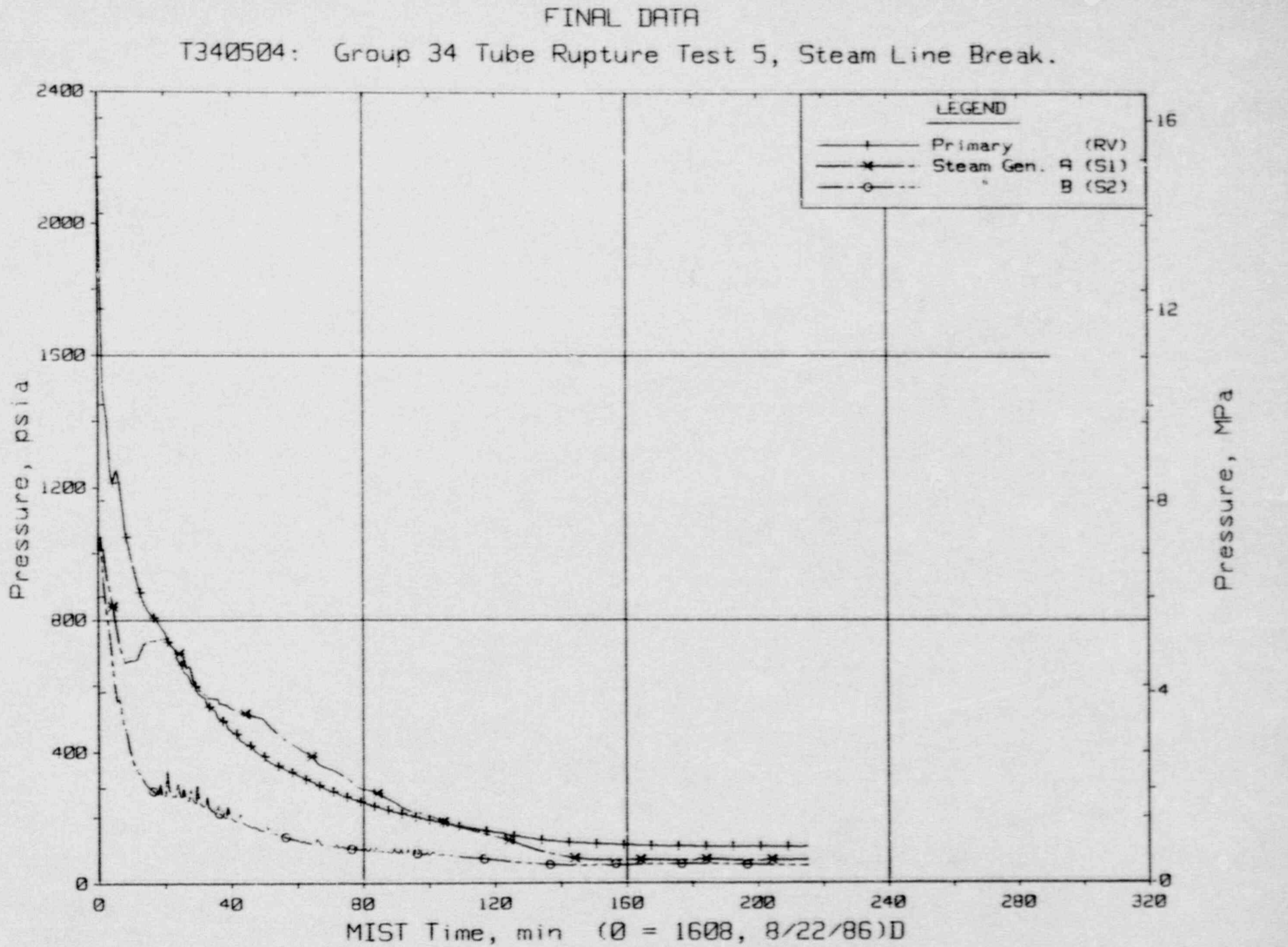
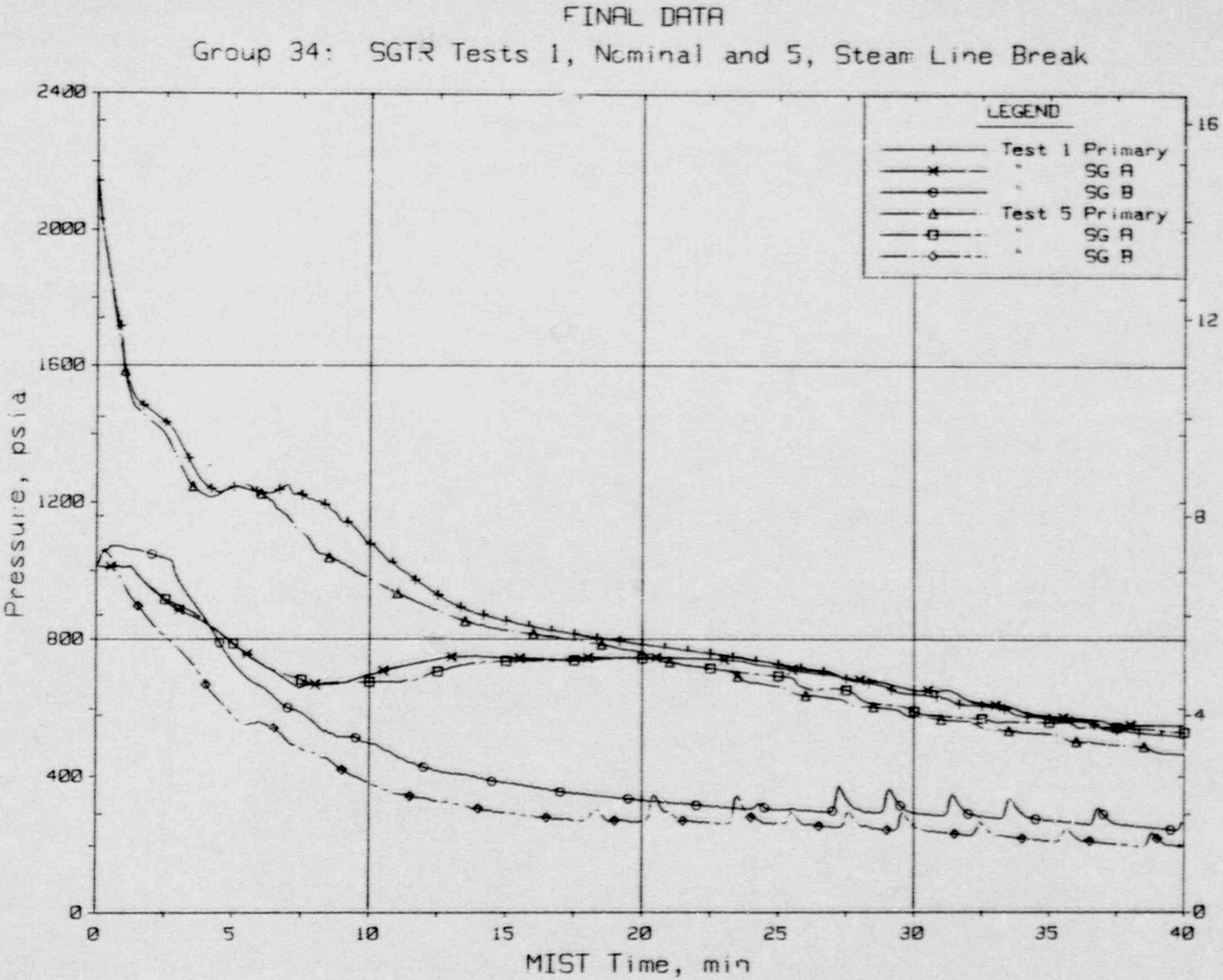


Figure 5.2.31. Primary and Secondary System Pressures (GP01s)

Mon Jul 25 13:25:13 1988

PSGP1



5-44

Figure 5.2.32. Primary and Secondary System Pressures (GPOls)

Mon Jul 25 11:26:20 1988

15PSGPi

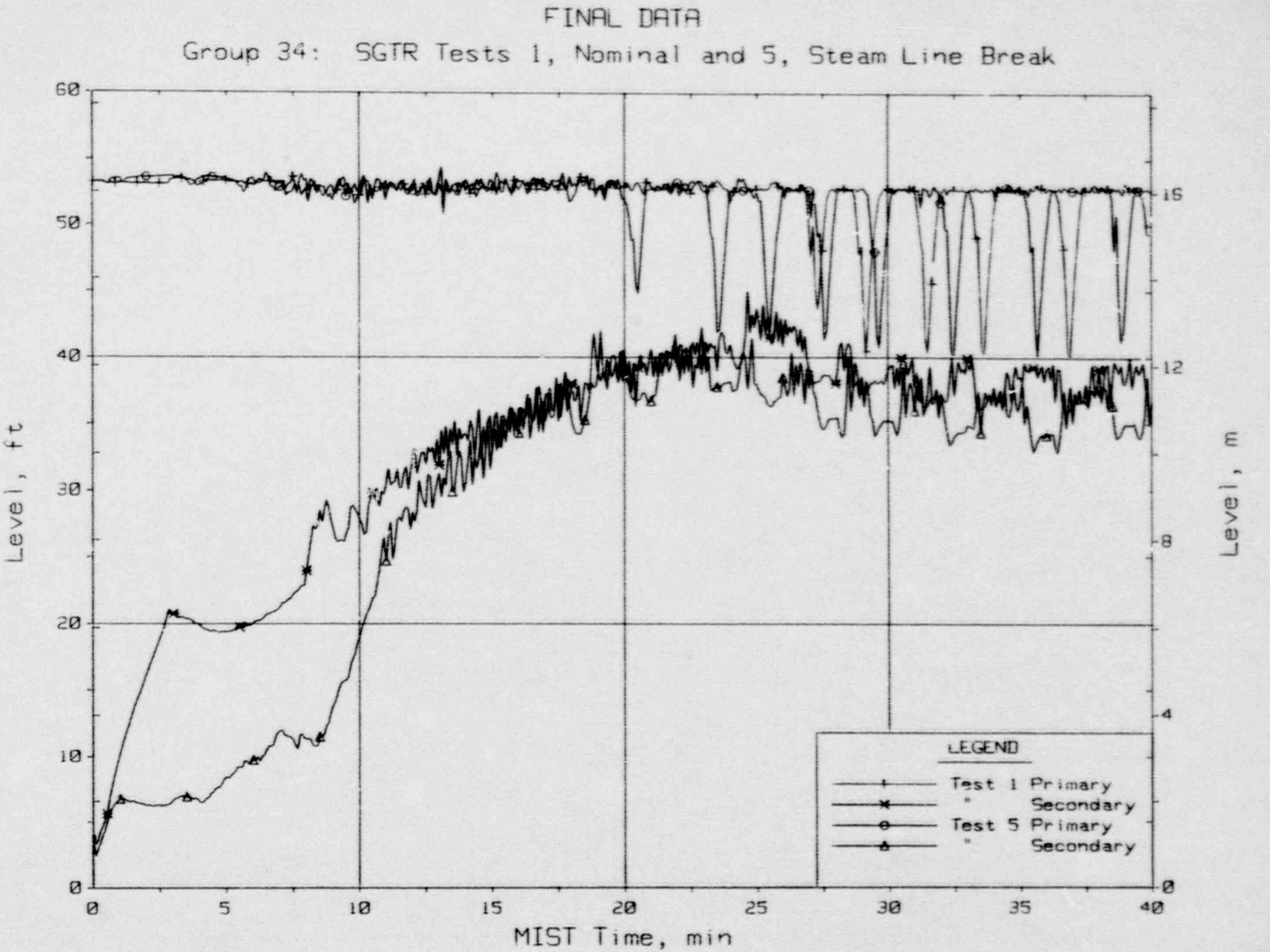


Figure 5.2.33. Steam Generator B Collapsed Liquid Levels

Mon Jul 25 13:38:18 1988

15SGLV2

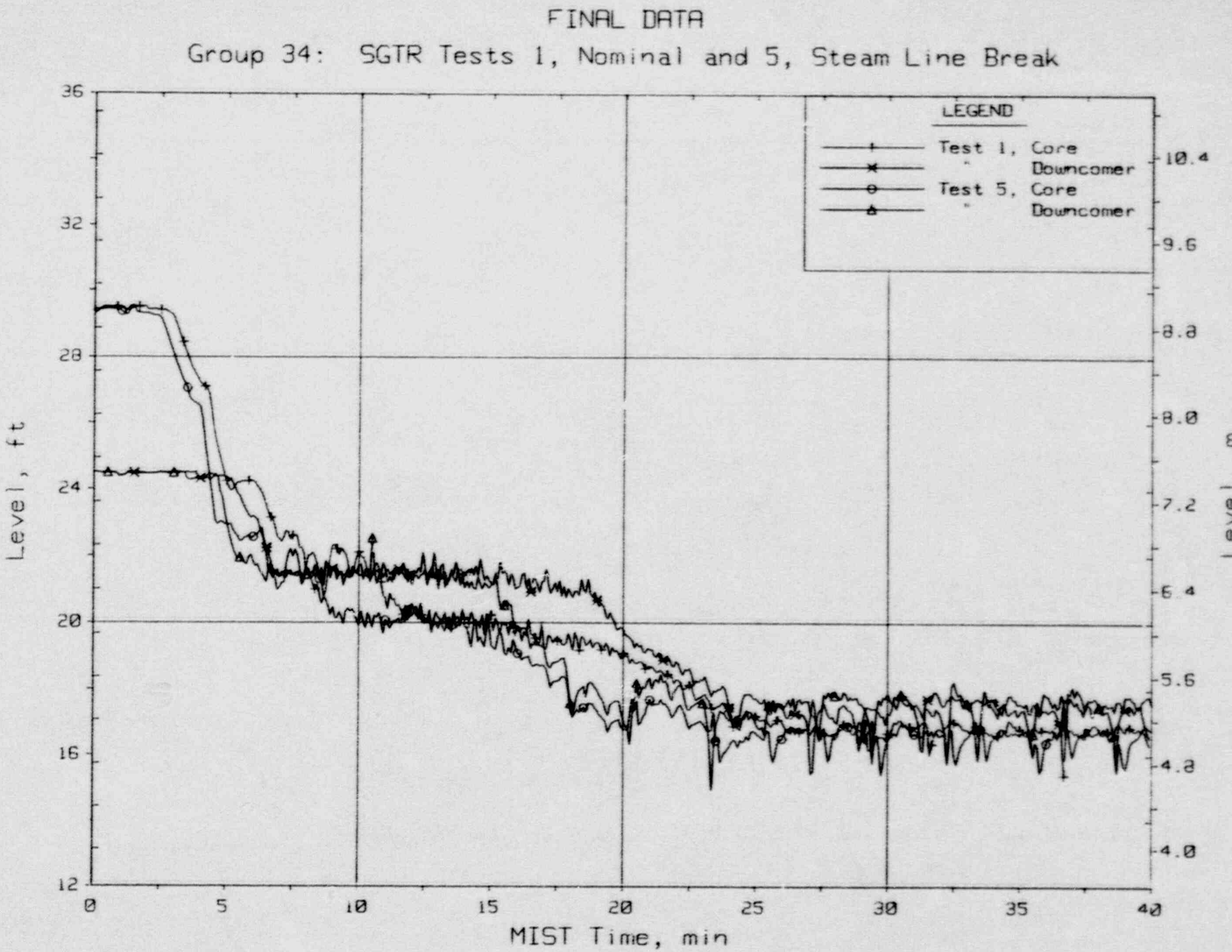


Figure 5.2.34. Core Region Collapsed Liquid Levels

Mon Jul 25 11:38:59 1988

15COLV1

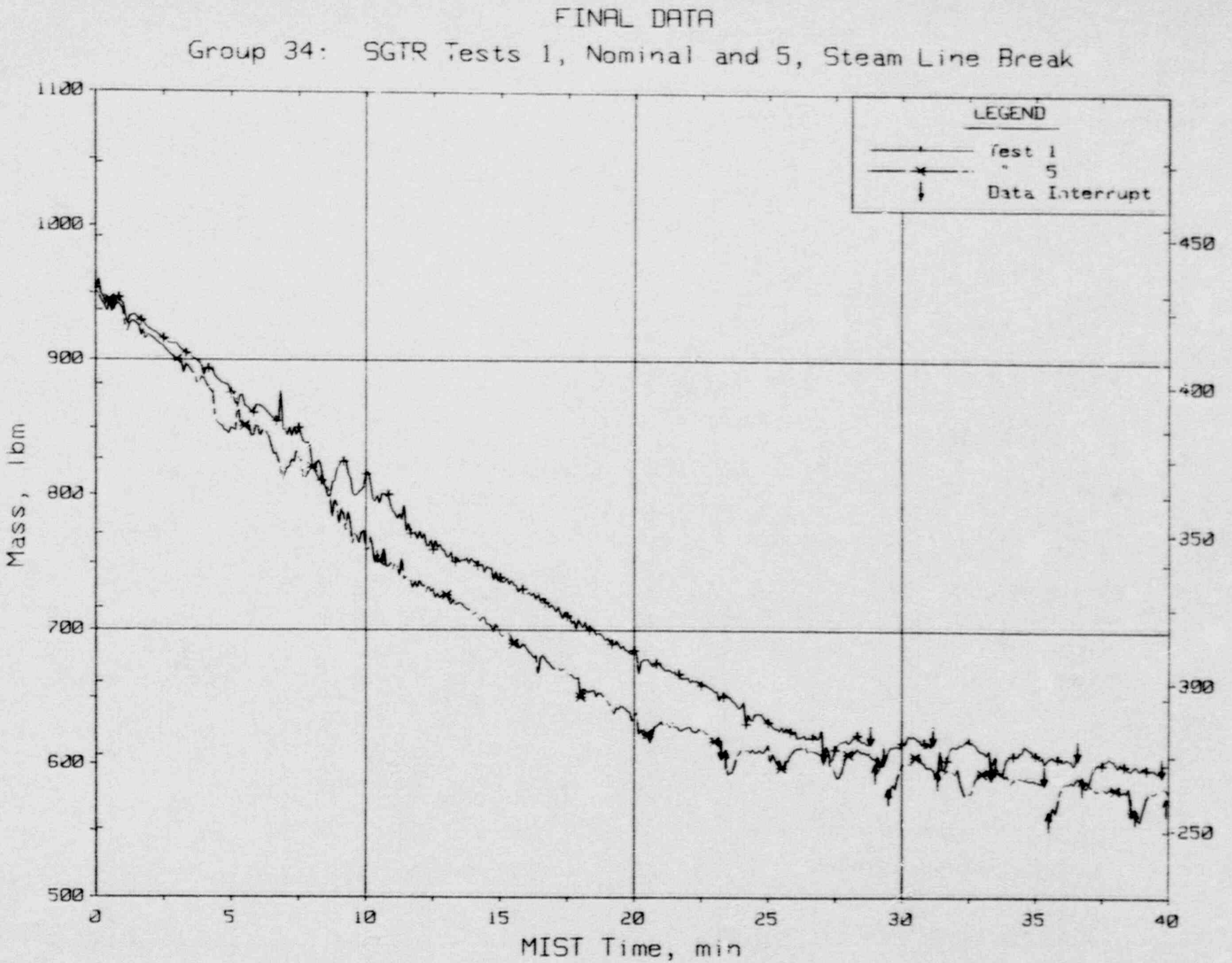


Figure 5.2.35. Primary System Total (Indicated) Fluid Mass

Mon Jul 25 11:41:08 1988 15PRML1

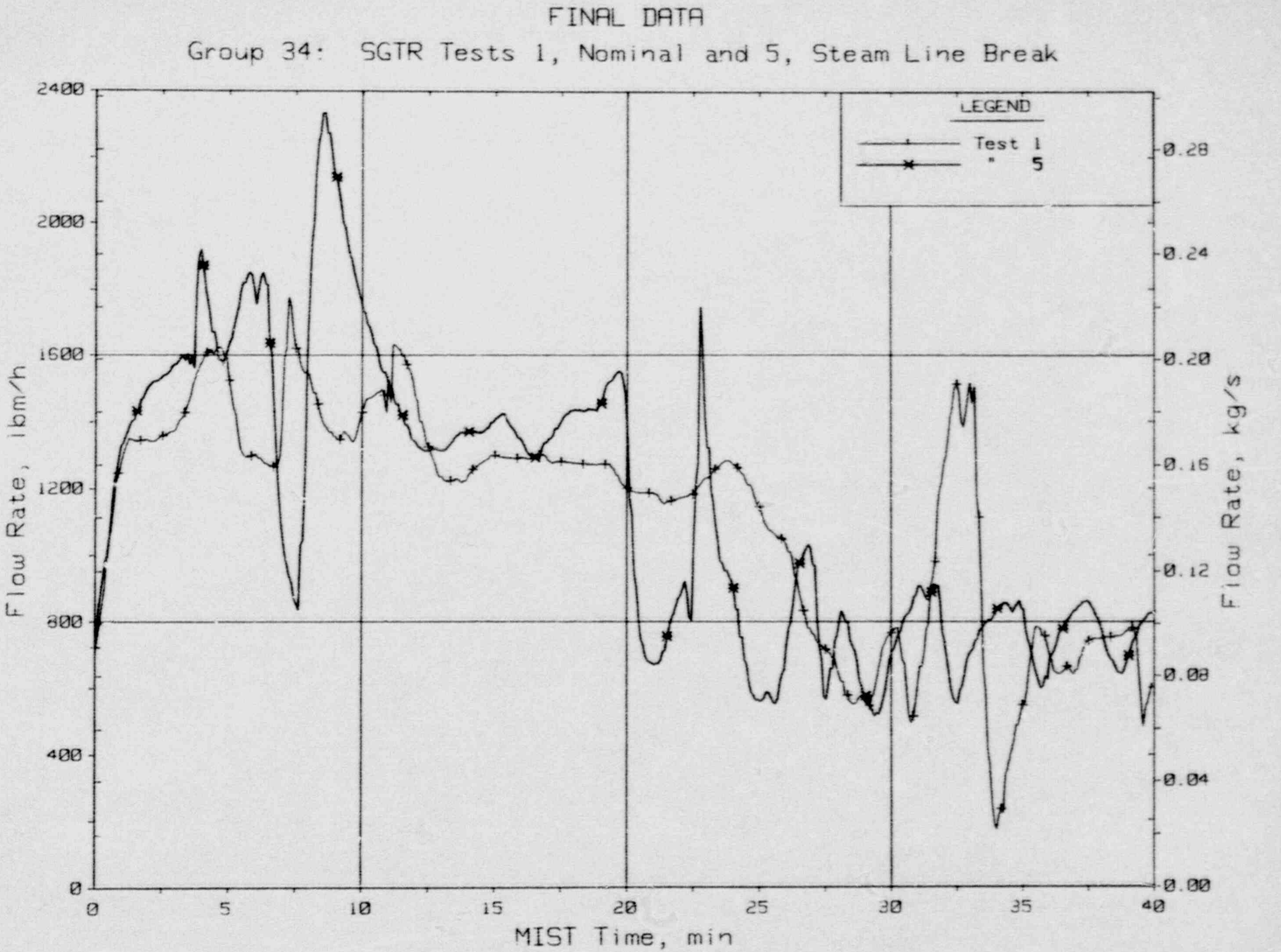


Figure 5.2.36. Tube Rupture Flow Rate

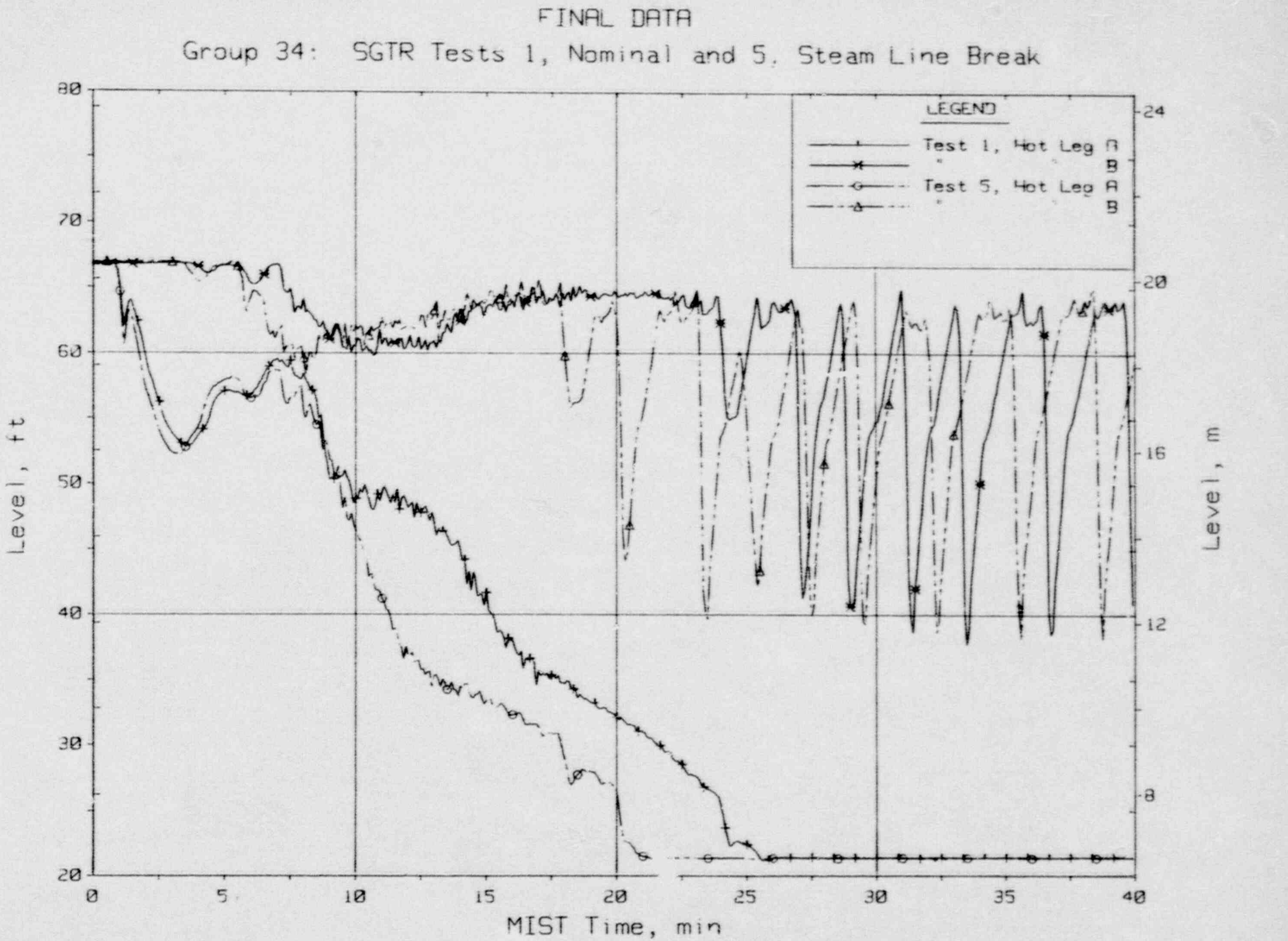


Figure 5.2.37. Hot Leg Riser Collapsed Liquid Levels

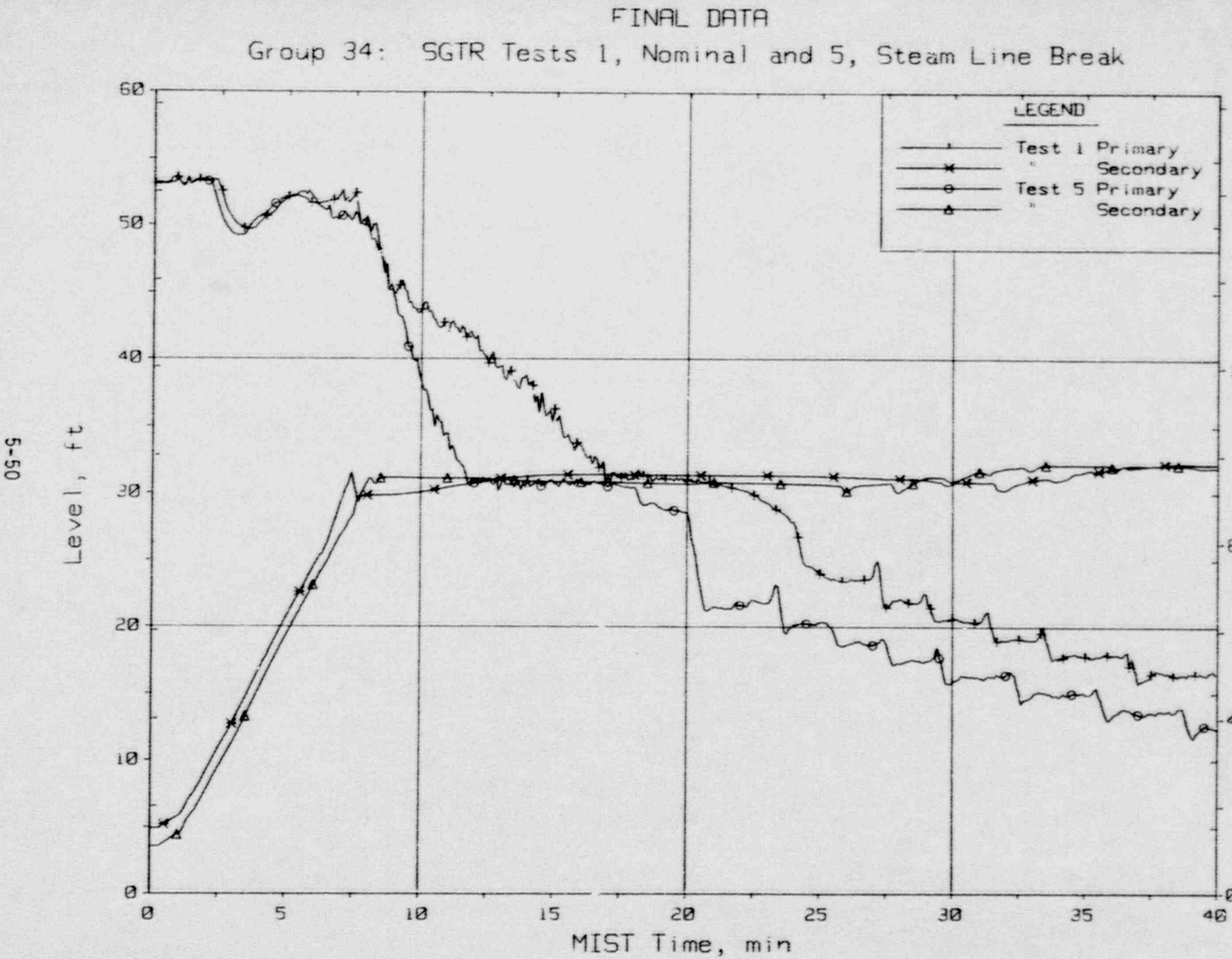


Figure 5.2.38. Steam Generator A Collapsed Liquid Levels

15SGLV1

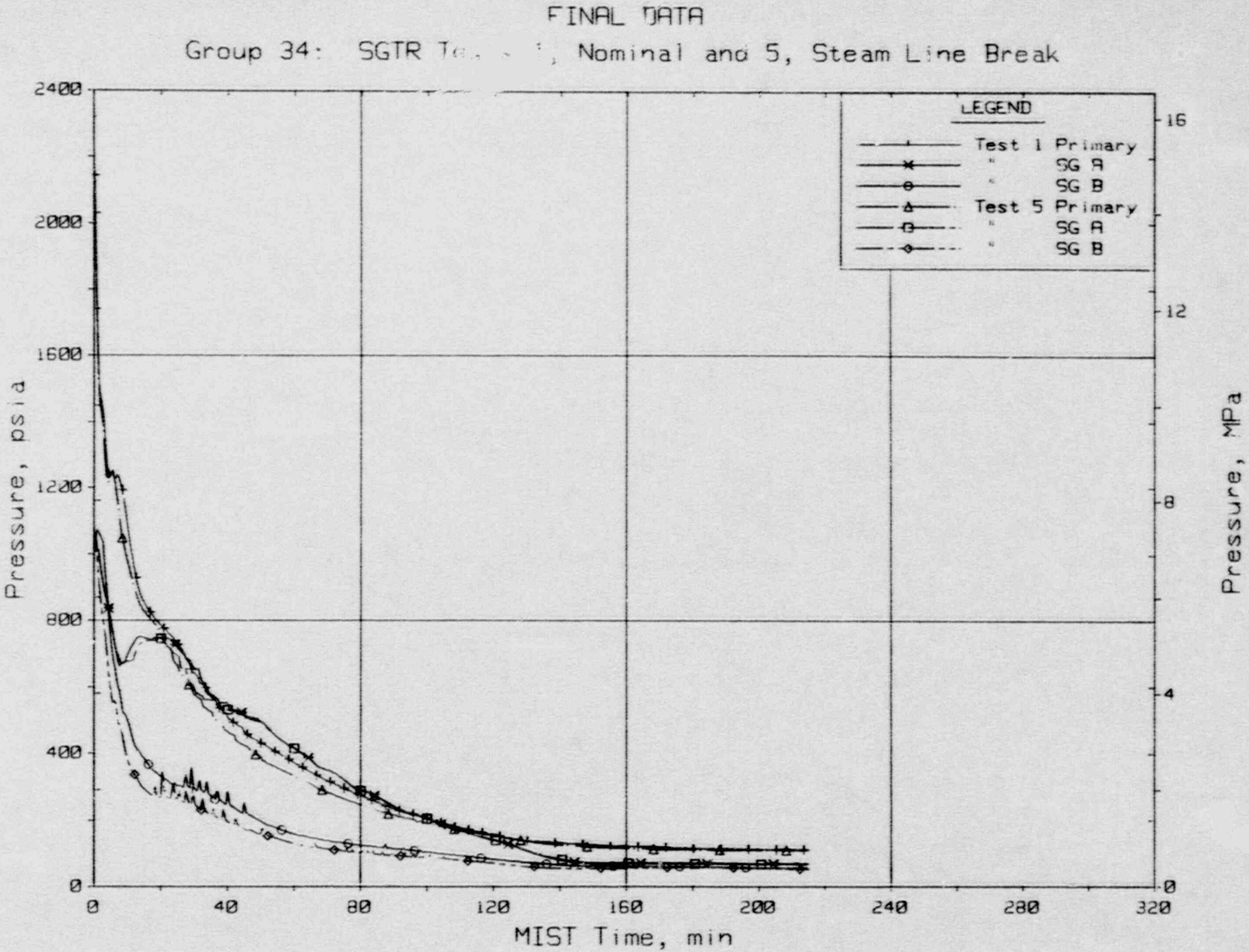


Figure 5.2.39. Primary and Secondary System Pressures (GP01s)

Mon Jul 25 11:53:20 1988

15PSGP1

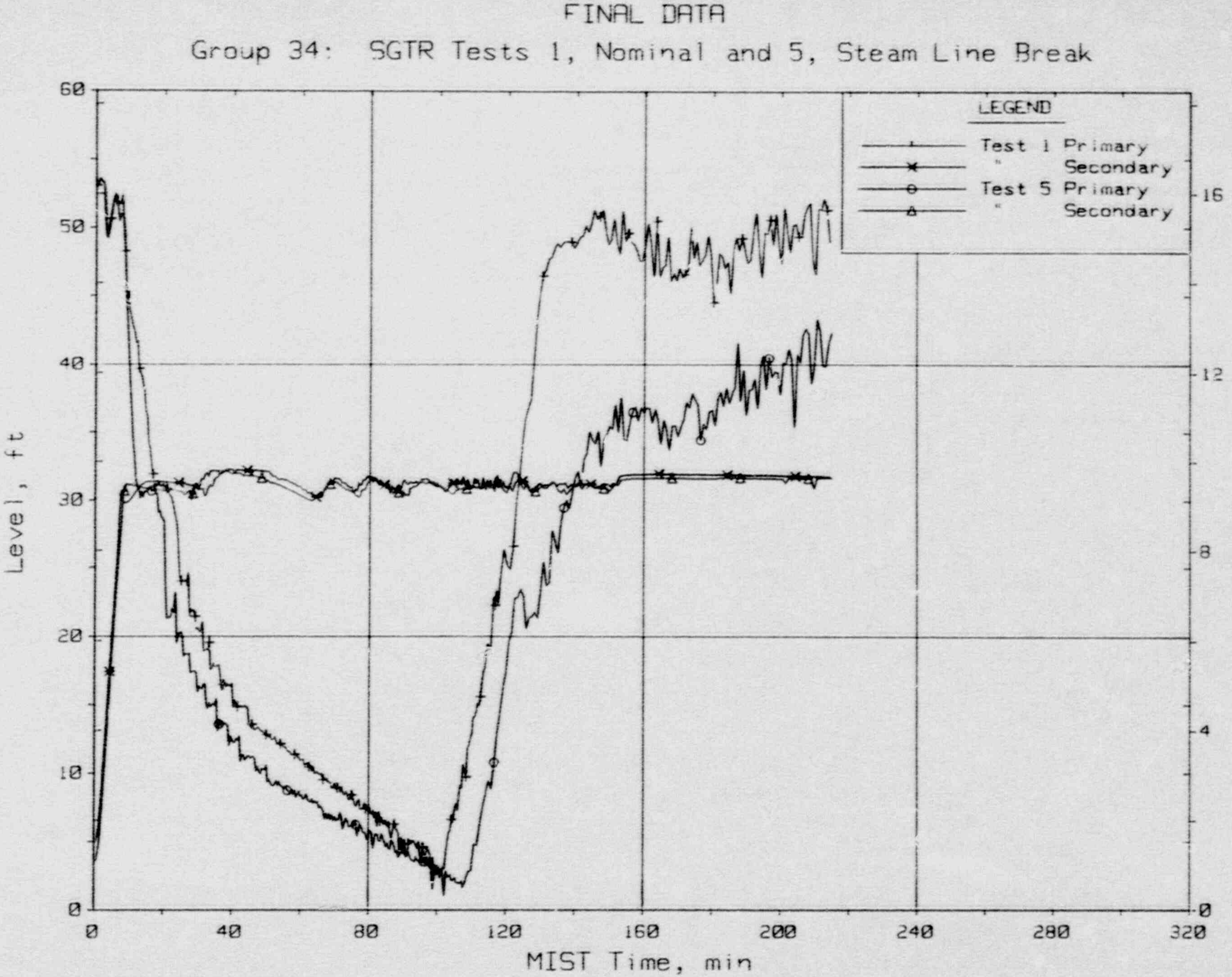


Figure 5.2.40. Steam Generator A Collapsed Liquid Levels

Mon Jul 25 11:56:09 1988

15SGLV1

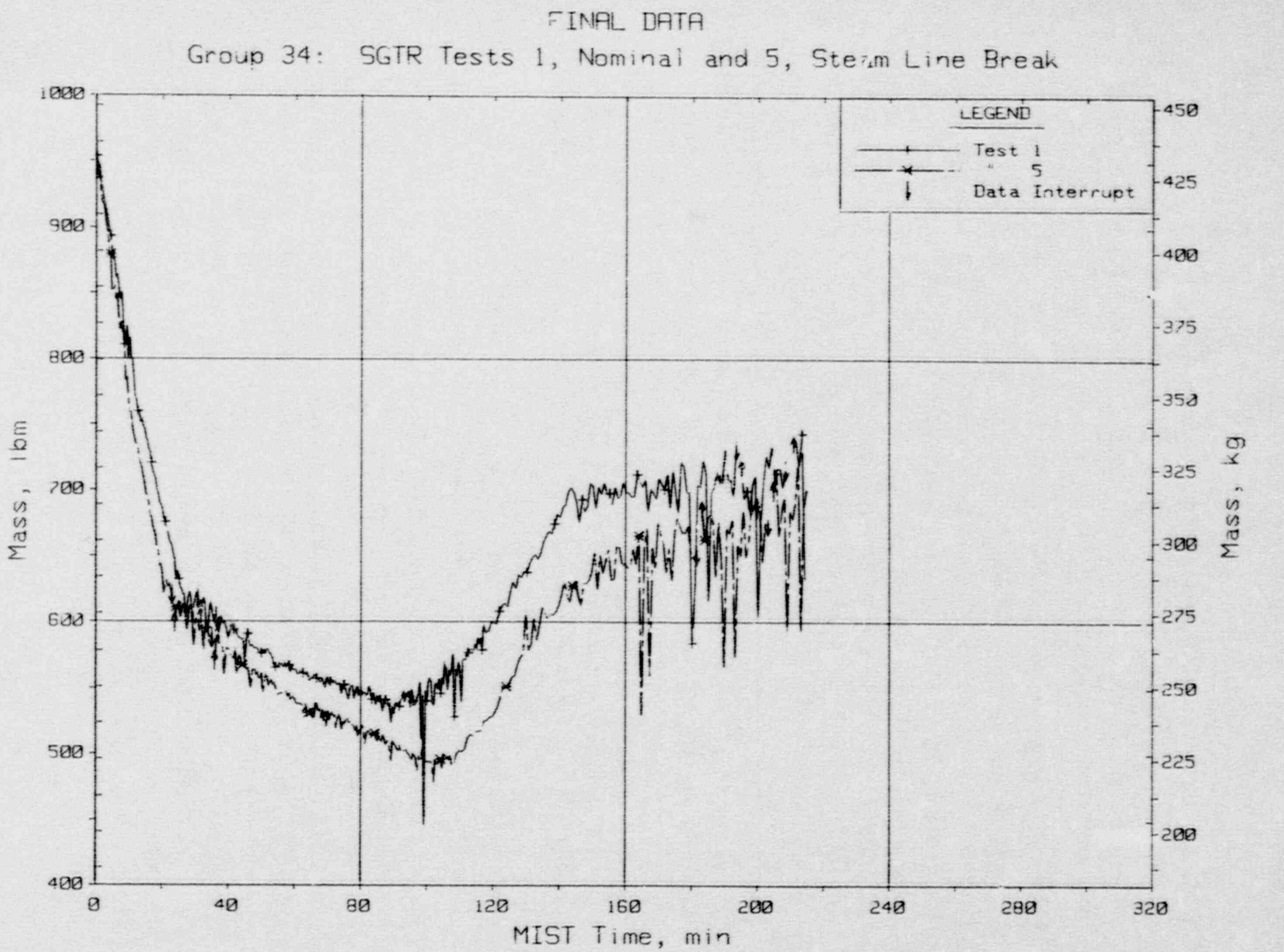


Figure 5.2.41. Primary System Total (Indicated) Fluid Mass

Mon Jul 25 11:58:15 1988

15PRML1

5.3. Low-Elevation Rupture of 10 Tubes, Tests 3 and 6

The low-elevation, double-ended rupture of 10 tubes was simulated in Test 3 and in its repeat, Test 6. Test 3 is described in section 5.3.1, and Tests 3 and 6 are compared in section 5.3.2.

5.3.1. Low-Elevation Rupture, Test 3

A low-elevation (double-ended) rupture of 10 tubes was imposed in Test 3 (340302). The initial primary system depressurization and mass depletion were extremely rapid. With the low-elevation break, the high rate of primary system mass loss continued even with extensive primary system voiding. The tube rupture flow rate gradually diminished as the primary system depressurized through heat transfer to (broken) steam generator B. Steam generator A became a heat source and remained so for more than 2 hours.

SGTR Test 3 was initiated by opening the scaled 30.8-cm^2 primary-to-secondary leak at the bottom of steam generator B. Also at time zero, the core power decay ramp was actuated and the RVVV controls were transferred to automatic/independent. The following interactions were observed at time zero: The lower SGTR limit switch actuated. The primary system began to depressurize from 2160 psia at approximately 550 psi/min (Figure 5.3.1). The primary system fluid mass began to decrease (from 978 lbm) and the fluid mass in steam generator B began to increase, both at approximately 1900 lbm/h. The steam generator B secondary pressure increased slightly (Figure 5.3.1), causing the steam generator B steaming rate to increase.

The initial SGTR flow rate approached 1600 and then 2400 lbm/h (Figure 5.3.2). (The SGTR flow rate was not measured directly; estimates of the SGTR flow rate were obtained from the primary system fluid mass imbalance.) The core exit fluid SCM decreased below 50F at virtually the instant the leak was opened (Figure 5.3.3), therefore the second set of test-initiating actions was performed within approximately the first 30 seconds of the transient. HPI was actuated, the steam generator A secondary level was reset to 31.6 ft, feed to steam generator B was terminated (Figure 5.3.4), and a cooldown of 100F/h was initiated in both steam generators. The HPI isolation valve limit switches actuated at 0.25 minutes.

The RVVVs opened simultaneously at 0.4 minutes (Figure 5.3.5). The steam generator A feed flow rate increased (toward the full head/flow capacity, Figure 5.3.4). The steam generator B feeding rate decreased toward zero beyond 0.5 minutes. At 0.85 minutes the loop A hot leg riser indicated voiding (Figure 5.3.6), and the HPI flow rate increased toward the full head/flow capacity. The hot leg A stub indicated level receded from the U-bend at 1.0 minutes (Figure 5.3.6), and the RVVVs briefly closed (Figure 5.3.5). The primary system depressurization slowed at 1580 psia beyond 1.1 minutes, signalling the saturation of the primary loop fluid (Figure 5.3.1). The loop A U-bend fluid temperature reached saturation (Figure 5.3.3). At 1.2 minutes, the loop A cold leg flow rates peaked above 10% of the scaled full cold leg flow, then abruptly stagnated (Figure 5.3.7). (The loop B flow rates remained mildly active.) The RVVVs reopened at 1.25 minutes (Figure 5.3.5). The steaming rate in steam generator A reached zero at 1.8 minutes.

The secondary level of steam generator B exceeded 20 ft at 1.6 minutes (Figure 5.3.8). This action triggered the final test-initiating procedure, the manual (full) opening of the low-flow steam valve in steam generator B. Steam generator B began to depressurize from 1025 psia beyond 1.9 minutes (Figure 5.3.1). The steaming rate in steam generator B became inaccurate and indicated zero (Figure 5.3.4). The steam generator B secondary rate of mass increase and level rise slowed distinctly at 2.1 minutes (Figure 5.3.8). The hot leg B stub level began to recede from the U-bend at 2.3 minutes. The loop A steam generator primary level reached the steam generator elevations (Figure 5.3.6). At 2.4 minutes, the reactor vessel volume above the RVVVs began to indicate voiding; the reactor vessel collapsed liquid level began to decrease (Figure 5.3.9). The hot leg B riser level began to decrease at 2.7 minutes. At 2.9 minutes, the loop B cold leg flow rates slowed abruptly from more than 4% of scaled full flow (Figure 5.3.7).

With the loop B hot leg U-bend voided, the SGTR flow rate was replenished by backflow in the loop B cold legs. The loop B cold leg flow rates stabilized in reverse flow at 3.3 minutes (approximately -3%, Figure 5.3.7). The core exit fluid saturated (Figure 5.3.3) at 3.4 minutes. The upper core volume began to indicate voiding. The reactor vessel level decreased more rapidly than before (Figure 5.3.9). The primary system began to repressurize somewhat (Figure 5.3.1). The upper reactor vessel void fraction reached 100% at 43 minutes. The downcomer began to void.

The reactor vessel and downcomer levels stabilized near the nozzle elevation at 5 minutes (Figure 5.3.9). The cold leg discharge piping began to void (least in cold leg A1). At 5.2 minutes, the hot leg B riser level briefly refilled to the U-bend whereas the loop A steam generator primary level descended below the feed injection site (Figure 5.3.6). The primary system depressurization resumed (from 1370 psia). Backflow within both loop B cold legs heated the rupture site fluid (Figure 5.3.10), causing the tube rupture flow rate to decrease noticeably (Figure 5.3.11). Loop A inter-cold leg circulation began at 5.5 minutes, with backflow in cold leg A2 (Figure 5.3.7).

The hot leg A riser and stub (steam generator primary) levels began to descend (Figure 5.3.12) as the primary system depressurized toward the steam generator A secondary pressure (Figure 5.3.13). The hot leg B riser level increased as the loop A levels dropped. The loop B hot leg riser level attained the U-bend spillover elevation at 12 minutes (Figure 5.3.12) and peaked at intervals of 1-1/2 minutes thereafter. The effects of the resulting intermittent loop B flow propagated throughout the system. The steam generator B secondary briefly repressurized (Figure 5.3.13). The loop B cold leg flow rates peaked near 15% of scaled flow (Figure 5.3.14) and the loop B cold leg suction and discharge levels briefly increased. The accompanying liquid displacement caused the loop B hot leg riser and steam generator primary levels to drop some 20 feet following each flow event (Figure 5.3.12).

Following the event just before 16 minutes, the loop B steam generator primary level dropped below the secondary level. The steam generator B secondary repressurized more than 100 psi, and the primary system depressurized below the steam generator A secondary pressure (Figure 5.3.13), causing steam generator A to become a heat source. At this time, too, the hot leg A riser emptied (Figure 5.3.12), the hot leg B riser remained well below the U-bend, and the intermittent loop B reactivations gradually diminished. The steam generator A primary level continued to decline, apparently due to voiding caused by the primary system depressurization and secondary-to-primary heat transfer. The SGTR flow rate gradually diminished from approximately 1800 lbm/h at 12 minutes to 800 lbm/h at 30 minutes (Figure 5.3.15). The steaming rate in steam generator B became sufficient to handle the tube rupture flow rate. The steam generator B secondary level stabilized near 45 feet and then began to recede.

The core flood tank began to discharge at the approximate rate of 100 lbm/h beyond 20 minutes (Figure 5.3.15) as the primary continued to depressurize. The SGTR flow rate approximately equalled the combined HPI and CFT input until 2 hours. By this time, the primary system fluid mass was less than one-half of its initial value. The primary system pressure was less than 150 psia (Figure 5.3.16), steam generator A had just been depressurized below the primary, and the tube rupture flow rate had decreased below 750 lbm/h. Both the hot leg A riser and steam generator primary had voided completely (Figure 5.3.17) whereas the corresponding loop B levels remained near the upper steam generator elevations. The steam generator A steam and feed systems were generally active from about 60 to 160 minutes. The loop A levels increased relatively rapidly near 120 minutes (Figure 5.3.17) as the total primary system injection rates exceeded the tube rupture flow rate (Figure 5.3.15). The test was terminated at 198 minutes based on the low-pressure termination criteria (primary system pressure below 200 psia for 2 hours).

5.3.2. Repeat Low-Elevation Rupture, Test 6, and Comparisons

Test 6 (3406AA) was a repetition of the low-elevation SGTR Test (340302). Although the major transient events were identical, the interactions and conditions of the two tests gradually diverged as the transients progressed.

The major events of Test 6 (3406AA) were identical to those of its companion Test 3. Both tests were characterized by the following phases:

- A rapid initial primary system blowdown to saturation.
- Primary depressurization through intermittent coupling to the broken steam generator, with reverse heat transfer from steam generator A to the primary.
- A prolonged low-pressure phase during which the tube rupture flow rate gradually decreased, the increasing HPI flow rate finally exceeded the rupture flow rate, and the intact steam generator A was depressurized by the imposed cooldown.

Although the two tests exhibited the same trends and virtually identical interactions during the initial, primary depressurization phase, the tests were by no means identical. At test termination, their intact-loop A hot leg riser levels differed by almost 20 ft (Figure 5.3.18). The inter-test differences first became apparent as the primary and steam generator B intermittently coupled. Rather than appearing suddenly, the inter-test differences increased gradually during and after the depressurization phase. Although virtually every system condition was ultimately affected, the predominant variables were apparently the energy stored in the metal, boundary system flow rates, and primary system total fluid mass. As discussed below, the initial pressurizer surge line fluid and metal temperatures as well as the initial reactor vessel flange metal temperatures were discernibly higher in Test 3 than in Test 6. The primary system and the steam generator B secondary depressurized slightly more readily in Test 6, leading to a decreased tube rupture flow rate and an increased HPI flow rate, and thus to a net increase of primary system fluid mass in Test 6 versus that in Test 3.

Initial Conditions

The primary and secondary system pressures were within 1.5 psi between Tests 3 and 6. The steam generator A secondary level in Test 3, 4.2 ft, was 1 ft below that in Test 6; the steam generator B initial secondary levels were maintained within 0.25 ft between tests (Figure 5.3.19). The steady-state steam and feed flow rates were maintained within 1% between tests, although the steam generator A flow rates were approximately 5% higher than those of steam generator B in both tests. The primary loop flow rates were matched between tests to within 0.02% of scaled full flow; in both tests, the indicated loop A flow rates were approximately 5% less than those of loop B.

The initial core power in Test 6 was 0.4 kW (or 0.3% of current power) less than that of Test 3. However, the current to the heater rods was 1% lower in Test 3 (Figure 5.3.20), and the voltage across the rods was higher by approximately the same fractional amount (Figure 5.3.21).

The loop fluid temperatures were matched between tests, generally to within 1/2F, as indicated by the following comparison of resistance temperature detector (RTD) readings:

Location	Test Temperature (F)	Inter-Test Difference (Test 3 - Test 6), (F)
Bottom of downcomer	546.0	+0.6
Hot leg inlet		
A	593.0	-0.5
B	592.8	-0.9
Steam generator inlet		
A	592.5	-0.5
B	592.5	-0.4
Pump suction		
A1	550.7	+0.5
A2	550.2	+0.4
B1	550.0	-0.3
B2	550.7	-0.5
Pump discharge		
A1	546.5	+0.8
A2	546.2	+0.9
B1	547.0	0
B2	547.0	0

The change of sign of the inter-test temperature difference, from the downcomer to the hot leg inlets, suggests a lower core power in Test 3. The fluid temperature rise across the core region was approximately 593F minus 546F, or 47F. The sign change from +0.6F between tests at the lower downcomer to -0.7F (average) at the hot leg inlets thus represents a change of energy deposition of 1.3/47 or nearly 3% (lower in Test 3). (The indicated core power in Test 3 was slightly higher than that in Test 6.) The core-region inter-test differences were apparently offset by decreased steam generator energy extraction and decreased pump-region heat losses in Test 3 versus Test 6.

Core Fluid Temperatures

The inter-test temperature difference (Test 3 minus Test 6) was approximately +0.6F at the lower downcomer and lower reactor vessel and approximately 0F at the core exit. But core-region inter-test differences evidenced a distinct trend, gradually decreasing from +1.7F at the start of the heated length to

+0.5 at the end of the heated length. These inter-test differences within the core do not appear to be consistent with the bracketing temperature differences (which were taken from the more accurate RTDs rather than from thermocouples). The explanation for this seemingly contradictory behavior involves the core heater rods. One rod was inadvertently deenergized in Test 3 (due to a blown fuse), while all 45 rods were active in Test 6. The individual rod output was increased in Test 3 to obtain the specified total core power using one less rod. The deenergized rod in Test 3 was apparently located near the periphery of the core, or at least sufficiently remote from mid-bundle, so that the core central thermocouples indicated elevated fluid temperatures in Test 3. This inter-test change in radial fluid temperatures was dissipated beyond the heated length through the mixing of the core exit fluid.

Core Power

One of the core heaters was apparently deenergized in Test 3, whereas all 45 heaters were energized in Test 6. If each of the heaters is assumed to have the same electrical resistance (r), then the bundle resistance in Test 3 (R_3) was r/44 and that in Test 6 (R_6) was r/45. If the reactive components of power are ignored and stray currents and line losses are neglected, the inter-test ratios of bundle current and voltage vary as the square root of the resistance ratio. The total bundle power (VI) is

$$VI = I^2R = V^2/R$$

For a constant total power between Tests 3 and 6,

$$I_3{}^2R_3 = I_6{}^2R_6$$

and $$V_3{}^2/R_3 = V_6{}^2/R_6$$

where I and V are total bundle current and voltage drop, and subscripts 3 and 6 denote Tests 3 and 6. Solving for the inter-test ratios of current and voltage,

$$I_3/I_6 = [R_6/R_3]^{1/2} = [44/45]^{1/2} = 0.989$$

and $$V_3/V_6 = [R_3/R_6]^{1/2} = [45/44]^{1/2} = 1.011$$

The observed inter-test current and voltage ratios were in almost exact agreement with the calculated values. The observed values (Figures 5.3.20 and 5.3.21) were as follows:

Quantity	Test 3	Test 6	Ratio, Test 3/Test 6
Power, kW	128.9	128.5	1.003
Current, amps	1538	1554	0.990
Voltage, V	83.6	82.6	1.012
Product of observed current and observed voltage, kW	128.6	128.4	1.002

The core power decay ramps varied between Tests 3 and 6 due to a zero shift of the core power voltage controller. Several minutes after the power decay simulation was activated, the inter-test difference in current increased from 16 amps to approximately 20 amps and the voltage offset decreased from 1 volt to approximately 0.3 volts. These altered offsets persisted throughout the test. As a result, the indicated core power in the Test 3 transient was approximately 1 kW (0.8% of the initial core power) less than that in the Test 6 transient (Figure 5.3.22). The significance of this power offset increased as the core power decreased; it was approximately 1% of the current core power at 2 minutes, 2% at one-half hour, and nearly 3% at 3 hours. Because the primary system depressurized more readily in Test 6 than in Test 3, this offset of core power is concluded to be of little net consequence.

Metal Temperatures

There were inter-test differences in metal and surge line temperatures. The surge line and lower-pressurizer temperatures just before test initiation were as follows:

Location	Test 3 Temperature, (F)	Inter-Test Difference, (Test 3-Test 6), (F)
Surge line fluid at surge line and hot leg connection	590	+ 1
Fluid at surge line low point	496	+ 6
Fluid in pressurizer heater bundle	646	+ 8
Surge line metal at 20 ft	582	+22

In Test 6, the surge line metal temperature (Figure 5.3.23) was rising discernibly before test initiation. The reactor vessel flange metal temperatures were also significantly different between tests. During pre-test steady state, the upper flange read 572F in Test 3, 19F higher than in Test 6, and the lower flange indicated 278F, 40F higher than in Test 6 (Figure 5.3.24). These differences in initial metal temperatures apparently caused an inter-test difference in primary system depressurization.

Comparison of Transient Interactions

The initial interactions of Tests 3 and 6 were virtually identical (Figure 5.3.25). The earliest inter-test differences involved the steam generator A secondary. The level in Test 3 lagged behind that in Test 6 by 1 ft initially, and by 1.5 ft during the refill of the secondary side (Figure 5.3.19). Between 0.5 and 1.2 minutes, the steam generator A secondary pressure increased slightly in Test 3, by approximately 20 psi, but remained almost constant in Test 6 (Figure 5.3.26). The next notable inter-test difference was observed at 1 minute. As the primary loop fluid saturated and the primary system depressurization rates slowed near 1500 psia, the primary pressure in Test 3 exceeded that in Test 6 by 25 psi (Figure 5.3.27). As the HPI flow approached the full head-flow capacity at 1.5 minutes (in both tests), the inter-test difference in primary system pressure obtained a small (~1%) HPI flow rate excess in Test 6. This difference between HPI flow rates was more discernible in the HPI flow to the loop B and A2 cold legs compared with the flow to cold leg A1 (Figures 5.3.28 through 5.3.31).

The low-flow rate steaming valve of steam generator B was manually opened as its level exceeded 20 ft at 1.5 minutes. The rate of level rise for steam generator B was thus reduced near 2 minutes. At this time, the steam generator B secondary inventory in Test 6 exceeded that in Test 3 by about 1%. This inter-test mass difference grew to 7% (8 lbm) by 4 minutes, but subsequently subsided. The steam generator B secondary pressures began to differ between tests beyond 3 minutes (Figure 5.3.27); by 10 minutes, the generator B secondary pressure in Test 3 exceeded that in Test 6 by 20 psi.

An obvious but apparently unimportant inter-test difference involved inter-cold leg flow. Cold legs A1 and A2 began to interact near 6 minutes in both tests, but the direction of flow was reversed between the two tests (Figures

5.3.32 and 5.3.33). The backflowing cold leg was A1 in Test 6 but A2 in Test 3; note the previously described inter-test increase in HPI flow rate to cold leg A2 (higher in Test 6) but not to A1.

The inter-test differences in primary and secondary pressures, and hence in boundary system flow rates, persisted and gradually increased as the transients progressed. The inter-test differences increased noticeably and began to affect virtually every system condition as the primary depressurized beyond 15 minutes. This portion of the transient was characterized by sporadic but relatively pronounced periods of primary-to-steam generator B heat transfer. Steam generator A became a heat source as the primary depressurized. This reverse heat transfer occurred at 15.8 minutes in Test 6 but not until 16.5 minutes in Test 3 (Figure 5.3.34), illustrating the inter-test timing offset that gradually developed. Also, the primary system total fluid mass in Test 6 gradually exceeded that in Test 3, due to the slight differences in primary boundary flow rates (Figure 5.3.25); for example

	Primary System Total Fluid Mass (lbm)		
Time (min)	Test 3	Test 6	Excess in Test 6
40	525	545	20
80	470	500	30
120	460	515	55
180	605	660	55

A chronological inter-test comparison through the first 18 minutes is provided in Table 5.3.1.

The primary system continued to depressurize beyond 15 minutes through periodic coupling to the steam generator B secondary. The inter-test differences gradually increased as the transients continued. The primary and steam generator B secondary pressures were generally lower in Test 6 than in Test 3; the inter-test difference in secondary pressures sometimes reached 20 psi (Figure 5.3.36). The inter-test differences in HPI, CFT, and SGTR flow rates were responsive to these inter-test differences in primary system pressure. The inter-test difference of total primary system fluid mass, reflecting the integral of these boundary flow rates, gradually increased. This mass difference (Test 6 larger) increased to 50 lbm at 180 minutes

(Figure 5.3.35), just prior to the termination of the tests. At 180 minutes, the steam generator A primary level had refilled to approximately 42 feet in Test 6, but to only 24 feet in Test 3; the hot leg A riser levels were also dissimilar (Figure 5.3.18). Both tests were terminated on low primary system pressure. Test 3 was terminated at 198 minutes, Test 6 at 188 minutes.

Summary

Test 6 (3406AA) was a repeat of Test 3 (340302), the low-elevation, double-ended rupture of 10 tubes. Although the tests exhibited the same major phases and interactions, they differed in detail. The inter-test differences gradually increased as the transient progressed. The source of these differences appears to be the unequal initial metal stored energy. The surge line and reactor vessel metal were cooler, by as much as 40F, in Test 6 than in Test 3. The primary system depressurized slightly more readily in Test 6, causing small differences in the primary system boundary flow rates. The inter-test difference of primary system total fluid mass, reflecting the integral of the boundary system flow rate differences, gradually increased as the test progressed. Beyond 15 minutes, inter-test differences became apparent in virtually every system condition. The results of Tests 3 and 6 thus highlighted the sensitivity of integral system transients to apparently subtle changes in initial conditions. Inter-test differences notwithstanding, Tests 3 and 6 underwent the same major events and timings. These global similarities are illustrated in Figures 1.3 and 1.6 which present primary system pressure versus primary system total fluid mass. These results indicate the variation that can be expected between two nominally identical transients, and therefore appear to be a useful addition to the data base for code benchmarking.

5.3.3. Comparison of Tests 3, 6, and 1

The double-ended rupture of 10 steam generator B tubes was simulated in Tests 3, 6, and 1. The rupture was imposed at the bottom of the steam generator in Tests 3 and 6, but at the top of the steam generator in Nominal Test 1. In each of the 3 tests, the initial tube rupture flow rate was about 2000 lbm/h (Figure 5.3.37). (The tube rupture flow rates were inferred from the primary system mass balance, as described in the Appendix, and are of limited accuracy.) The initial rates of primary system total fluid mass loss were virtually equal among the 3 tests (Figure 5.3.38), as were the primary system depressurization rates (Figure 5.3.39). In each of the 3 tests, hot leg A voiding became apparent just before 1 minute (Figure 5.3.40).

The effects of rupture elevation became apparent by 3 minutes. Apparently due to the greater rupture-site fluid subcooling with a lower-elevation break, the rate of primary system mass loss in Tests 3 and 6 began to exceed that in Test 1 (Figure 5.3.38). The hot leg A level decreased more rapidly in the low-elevation rupture tests. Moreover, in Test 1, hot leg B voiding occurred later and to a lesser degree than in Tests 3 and 6 (Figure 5.3.40). By 15 minutes in the lower-elevation rupture tests, steam generator B had become intermittently active whereas the steam generator A primary-to-secondary temperature difference had become negligible. The steam generator B repressurizations (Figure 5.3.41) attested to the magnitude of the primary-to-steam generator B heat transfer. The steam generator B primary levels in Tests 3 and 6 (Figure 5.3.42) alternately dropped into the steam generator, exposing primary system vapor to tube lengths cooled by the steam generator B secondary discharge to the steam lines, and refilled toward the loop B U-bend spillover elevation. This intermittent steam generator B activity was delayed until approximately 25 minutes in Nominal Test 1, apparently due to its lower rate of primary fluid mass depletion. Whereas with the low-elevation rupture the primary system mass began to stabilize as primary system pressure was reduced, in Test 1 the intermittent uncovery of the rupture site allowed mass stabilization beyond 27 minutes (Figure 5.3.43), at a higher primary system pressure (Figure 5.3.44).

The primary and secondary system pressures of the 3 tests gradually converged as the tests progressed (Figure 5.3.45). The primary system was refilled more readily with the upper-elevation rupture than with a lower-elevation rupture (Figure 5.3.46). The 2 lower-elevation rupture tests themselves evidenced noticeable inter-test difference, however, as has been discussed in the previous subsection.

Table 5.3.1 System Interactions in Tests 3 and 6 Through the First 18 Minutes

Events that occurred simultaneously in the two tests are noted by a single time entry; double entries give the times for Tests 3 and 6, respectively. Note that the 5-second data acquisition rate can cause an apparent inter-test timing offset of 0.1 minute.

Time, (min)	Interactions
0	The SGTR flow path opened. The primary system began to depressurize from 2150 psia at 550 psi/min. The steam generator B secondary level began to increase at approximately 10 ft/min.
0.1/0	The RVVVs opened by limit switch indications.
0.2	The HPI isolation valves opened.
0.4/0.3	The feed flow rate to steam generator A increased to the full-scaled head/flow capacity.
0.4	The decreasing uppermost reactor vessel upper head fluid temperatures began to diverge slightly, becoming lower in Test 6 that in Test 3.
0.5	The feed flow rate in steam generator B decreased to zero.
0.5/0.3	The steam generator A secondary level began to increase from approximately 5 ft. The level in Test 3 lagged behind that in Test 6, initially by 1 ft and by 1.5 ft as the secondary refill proceeded.
0.5	The steam generator A secondary pressure increased slightly in Test 3, but remained approximately constant in Test 6.
0.9/0.8	The HPI flow rates began to register.
1.0	The RVVVs closed briefly. The primary system depressurization rate began to slow near 1500 psia. The primary system pressure in Test 6 began to differ from that in Test 3, becoming 25 psi lower. The hot leg A riser level receded from the U-bend.
1.2	The hot leg A level beyond the U-bend began to decrease. The loop A cold leg flow rates peaked at 11% of scaled full cold leg flow and began to decrease rapidly. The steam generator A secondary pressures began to decrease (from a slightly higher pressure in Test 3).
1.3	The RVVVs reopened. The steam flow rates in steam generator A decreased toward zero flow.

Table 5.3.1 System Interactions in Tests 3 and 6 Through the First 18 Minutes (Cont'd)

Time, (min)	Interactions
1.5	The secondary level of steam generator B exceeded 20 ft (slightly earlier in Test 6). The HPI flow rate approached scaled full head-flow capacity. Although the HPI flow rate to cold leg A1 was approximately the same in Tests 3 and 6, the HPI flow rates to cold legs A2, B1, and B2 were approximately 1.5 lbm/h (1%) higher in Test 6 than in Test 3).
1.9	The secondary pressure of steam generator B began to decrease.
2	The rate of increase of the steam generator B secondary fluid mass and level abruptly slowed. The steam generator B mass in Test 6 exceeded that in Test 3 by approximately 2 lbm or 1%; the corresponding level excess in Test 6 was 0.8 ft.
2.3	The reactor vessel upper head (above the RVVVs) began to void (based on local differential pressure indications).
2.7	The hot leg B riser level and stub levels (beyond the U-bend) began to decrease.
3	The steam generator B secondary pressures began to differ between tests, becoming lower in Test 6. (This inter-test difference gradually increased to 20 psi, a saturation temperature difference of 5F, by 10 minutes.) The uppermost reactor vessel upper head fluid temperature again began to evidence slight inter-test differences ~1F lower in Test 6 than in Test 3. The loop B cold leg flow rates decreased from nearly 6% of scaled full flow to reverse flow.
3.3	The core-region fluid began to void (based on local differential pressure measurements).
3.4	The core exit subcooling margin (SCM) decreased to zero.
4	The secondary level in steam generator B abruptly stabilized near 30 ft; the steam generator B level in Test 3 then began to exceed that in Test 6. The indicated steam generator B secondary fluid mass in Test 3 exceeded that in Test 6 by as much as 8 lbm (7%).
4.3	The reactor vessel volume above the RVVVs voided completely.
5	The primary began to depressurize from 1375 psia; the pressure in Test 3 continued to exceed that in Test 6 by about 20 psi.

Table 5.3.1 System Interactions in Tests 3 and 6 Through the First 18 Minutes (Cont'd)

Time, (min)	Interactions
5.2	The reactor vessel volume between the nozzles and the RVVVs began to void (based on measurements of local differential pressure).
~6	The loop A cold legs began to interact, one flowing forward and the other backward. The forward-flowing leg was A1 in Test 3 and A2 in Test 6. The cold leg discharge fluid in the backflowing leg (leg A2 in Test 3 and A1 in Test 6) abruptly decreased more than 150F, then returned to its previous temperature. The inter-test difference between upper head fluid temperatures increased to as much as 5F (warmer in Test 6).
6.3	The hot leg riser level of loop B briefly refilled to the U-bend elevation.
6.5	The steam generator primary level of loop B stabilized near the elevation of the steam generator upper tubesheet.
8.2/7.8	The steam generator A secondary level approached the control band and stabilized. Steam generator A feeding was terminated.
8.2/8.0	The hot leg A riser level began to decrease rapidly and unevenly. (The inter-test difference in the hot leg A riser levels increased to as much as 3 ft).
8.3/8.0	The steam generator A secondary pressure stabilized (at 680 psia in Test 6 and 690 psia in Test 3). The loop A steam generator primary level, which had stabilized near the upper tubesheet, began to decrease; the level in Test 3 exceeded that in Test 6 by as much as 3 ft.
10.8	The loop B cold leg flow rates, which had become oscillatory, evidenced a distinct peak at 4% of scaled full cold leg flow.
11.8/11.7	The loop B hot leg riser level briefly approached the U-bend spillover elevation.
12/11.8	The loop B cold leg flow rates peaked near 6% of scaled full flow (higher in Test 3).
12.7	The steam generator B secondary level abruptly dropped several feet (earlier and lower in Test 6).

Table 5.3.1 System Interactions in Tests 3 and 6 Through the First 18 Minutes (Cont'd)

Time, (min)	Interactions
13.0	The inter-test difference in primary pressures increased discernibly.
13.3/12.8	The loop B hot leg riser level again approached the U-bend spillover elevation.
13.3/13.0	The steam generator B secondary repressurized approximately 40 psi; the inter-test difference remained about 20 psi (higher in Test 3).
13.5/13.0	The RVVVs began to close intermittently.
13.7/13.2	The cold leg flow rates of loop B peaked near 14% (higher in Test 3). The steam generator primary level of loop A dropped 5 ft.
13.7/13.0	The steam generator primary level in loop B briefly dropped almost 7 ft below the upper tubesheet.
14.2/13.7	The steam generator A secondary began to repressurize.
14.3/13.7	The loop B hot leg riser level approached the spillover elevation.
14.3/13.8	The steam generator B secondary level peaked near 44 ft.
15.0/14.5	The steam generator B secondary level began to rise.
15.0/14.7	The loop B cold leg flow rates peaked near 16% of scaled full cold leg flow. The loop A steam generator primary level dropped 3 ft. The loop B steam generator primary level decreased to 7 ft below the tubesheet elevation (lower in Test 6) and generally remained below the tubesheet.
15.1/14.7	The steam generator B secondary pressure peaked near 500 psia (higher in Test 3).
~15	The HPI flow rates became noticeably different between Tests 3 and 6. The Test 6 total HPI flow rate exceeded that in Test 3 by 18 lbm/h, or 3%. The inter-test difference was apparently least in loop A1. The cold leg discharge cooled more in Test 6 than in Test 3. The downcomer level rose in Test 6, and the Test 6 RVVV differential pressures increased correspondingly. Inter-test differences developed generally.
16.0/15.6	The loop A hot leg riser voided completely.

Table 5.3.1 System Interactions in Tests 3 and 6 Through the First 18 Minutes (Cont'd)

Time, (min)	Interactions
~16	The loop A cold leg flow rates stabilized in backflow. The steam generator B secondary level peaked at 47 ft and then began a relatively gradual decline.
16.5/15.8	The primary system pressure decreased below that in the steam generator A secondary. The inter-test difference of primary pressures increased, reaching 70 psi at 18 minutes (lower in Test 6).
16.5/16.0	The steam generator B secondary pressure peaked near 600 psia (higher in Test 3) and then decreased somewhat gradually.
17.8/16.5	Steam generator A feed was reactivated.
~18/16	The RVVVs generally remained open.
18.4/17.5	The steam generator A secondary began to be depressurized.

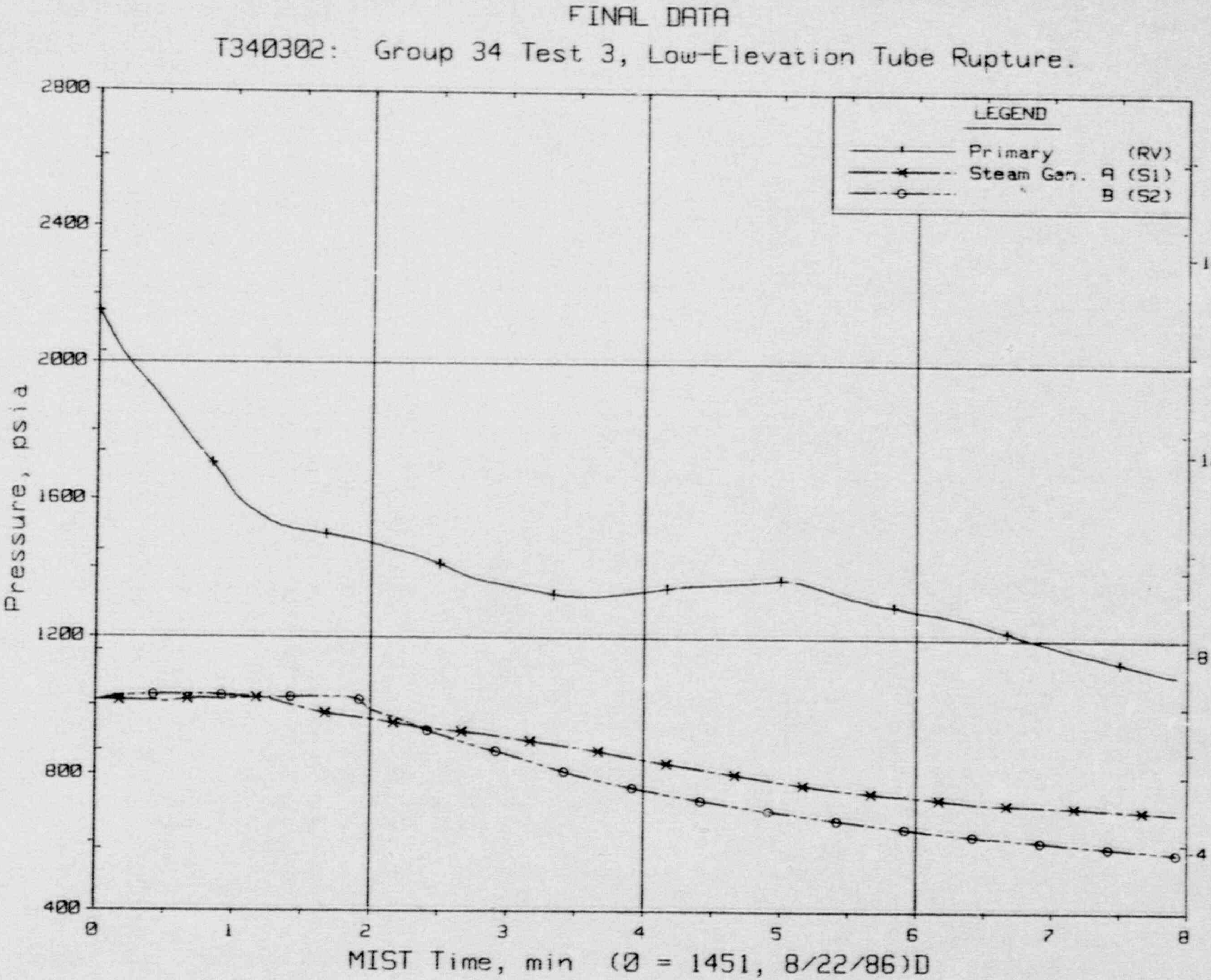


Figure 5.3.1. Primary and Secondary System Pressures (GP01s)

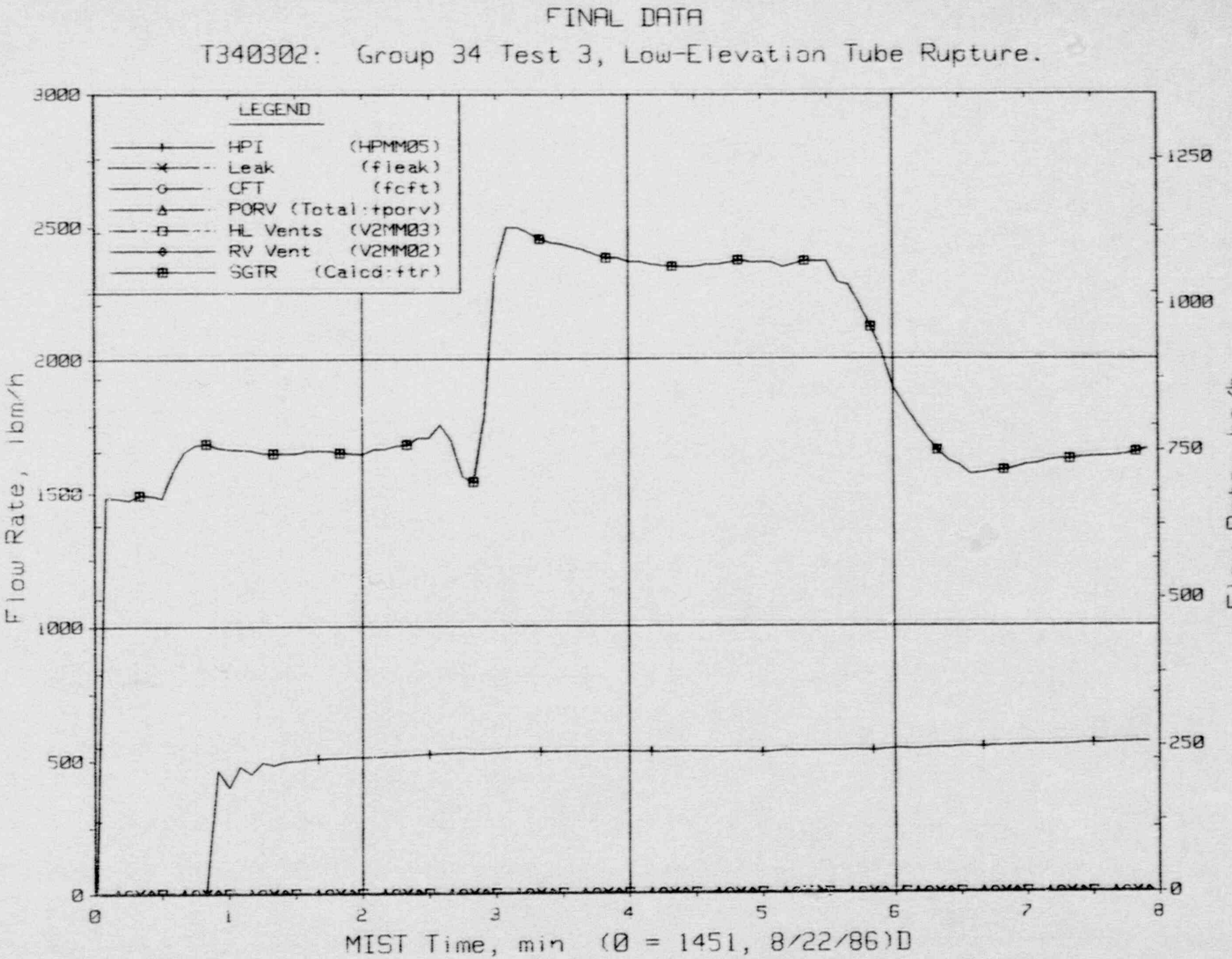


Figure 5.3.2. Primary System Boundary Flow Rates

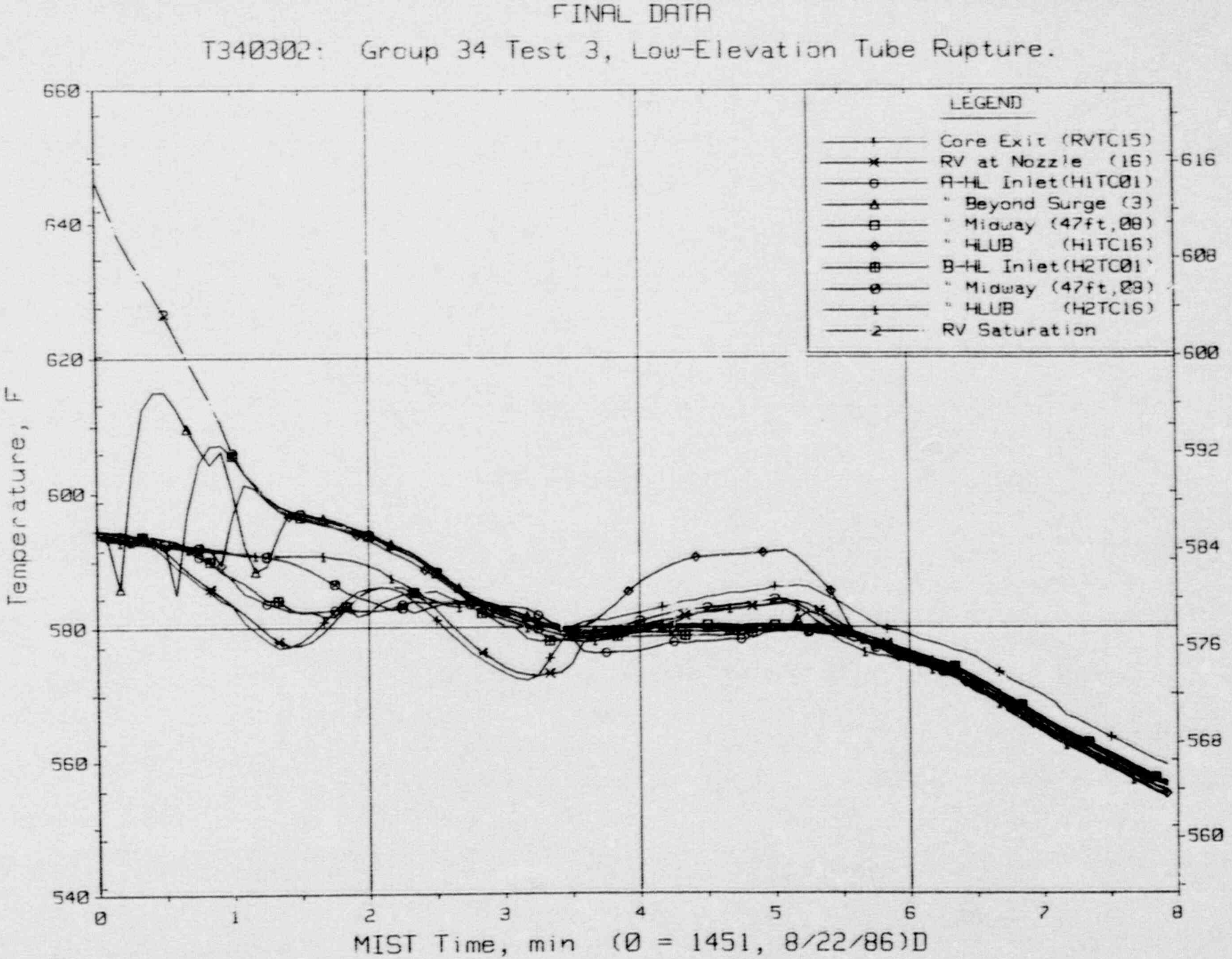


Figure 5.3.3. Composite Core Exit and Hot Leg Fluid Temperatures

Mon Jul 25 13:49:41 1988 COTC1

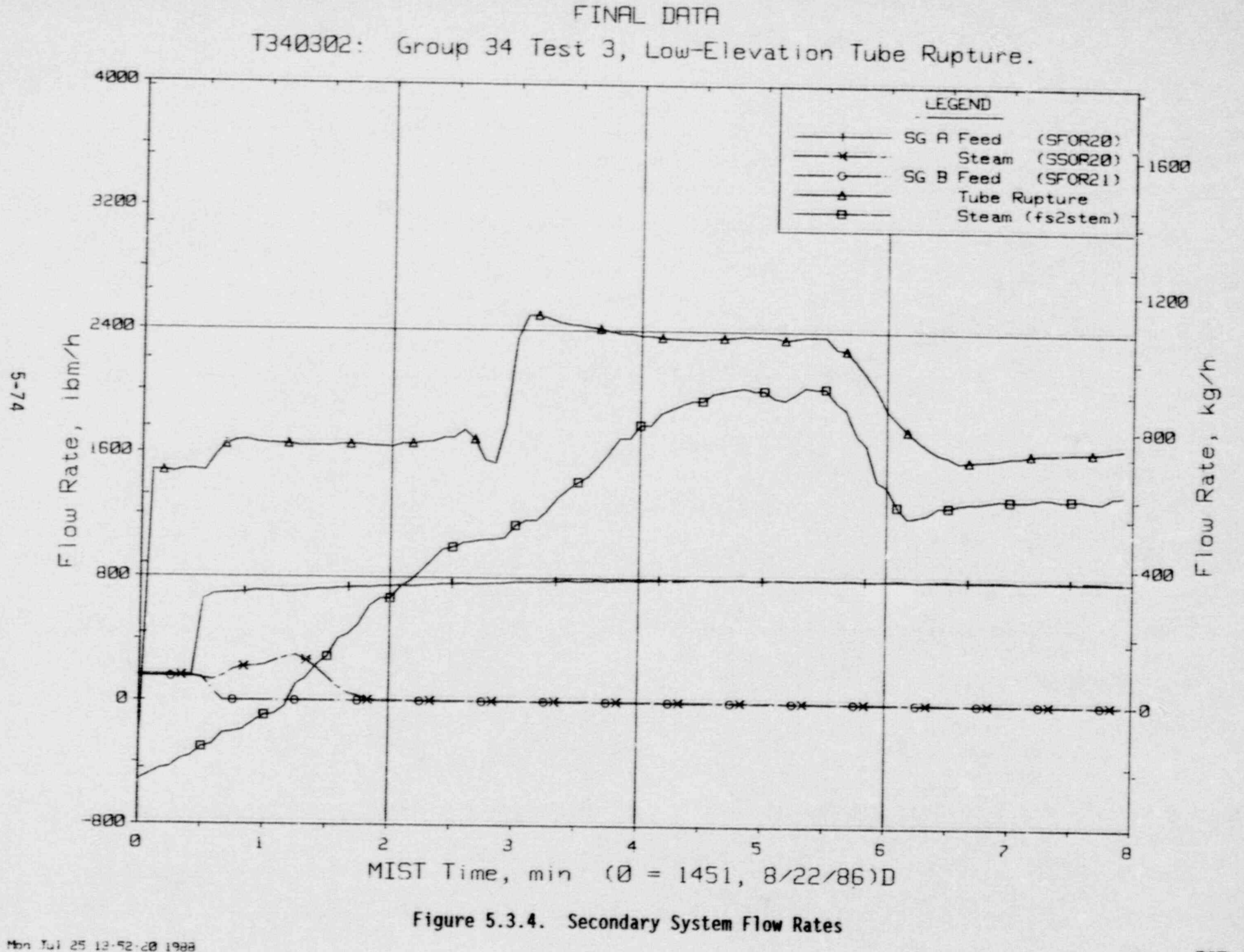


Figure 5.3.4. Secondary System Flow Rates

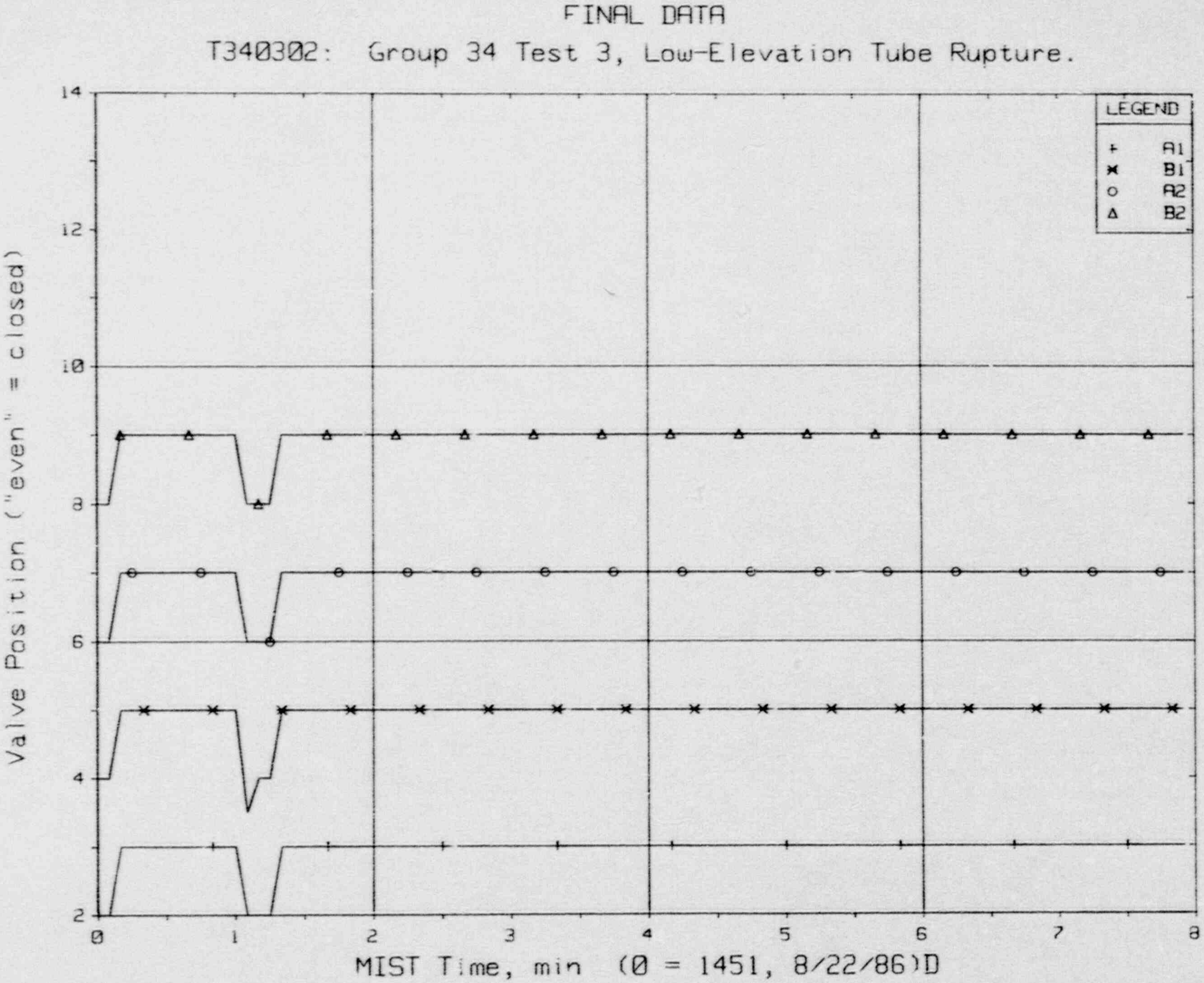


Figure 5.3.5. Reactor Vessel Vent Valve Positions

Mon Jul 25 13:54:29 1988

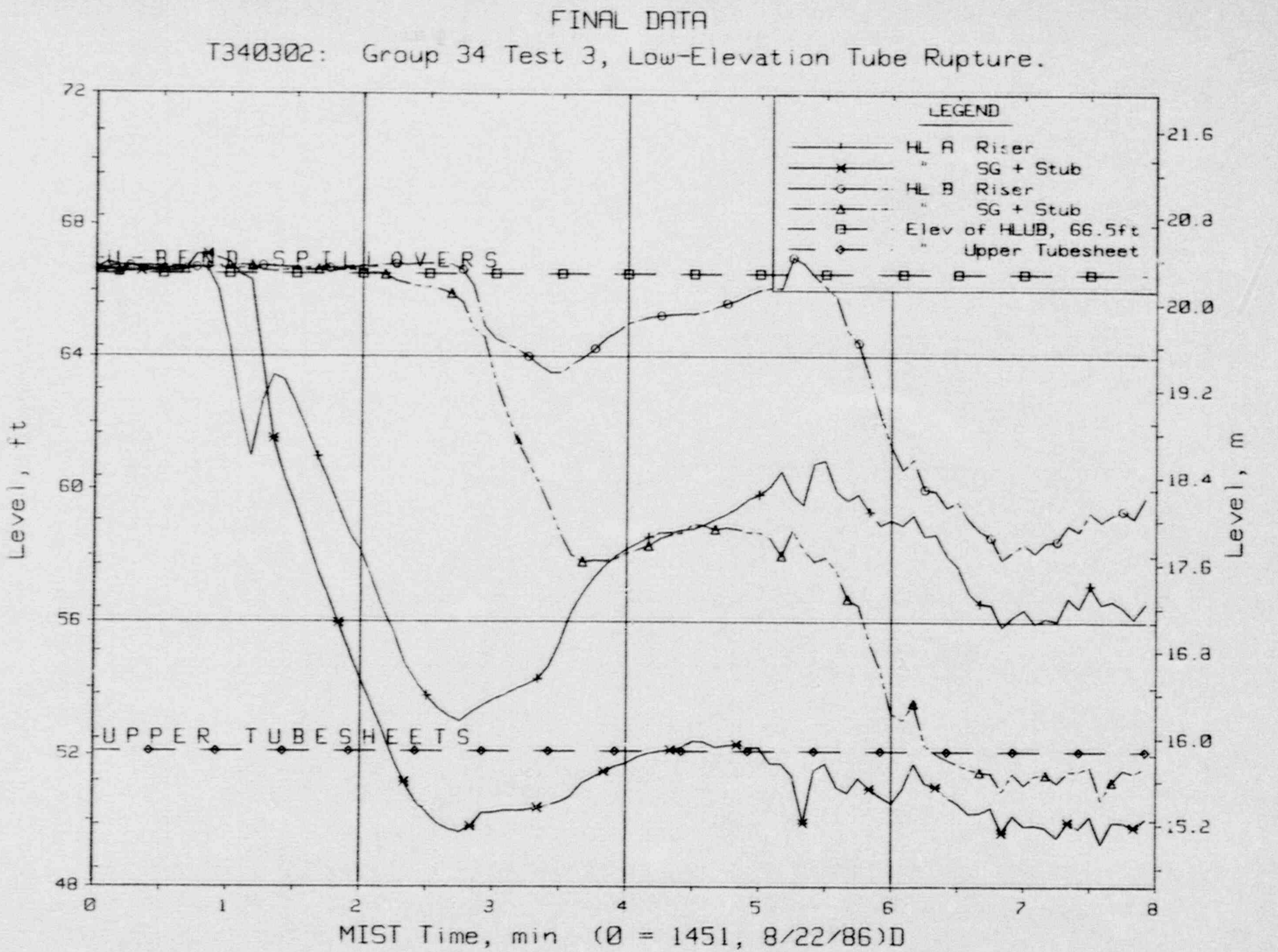


Figure 5.3.6. Hot Leg Riser and Stub Collapsed Liquid Levels

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HLLVI

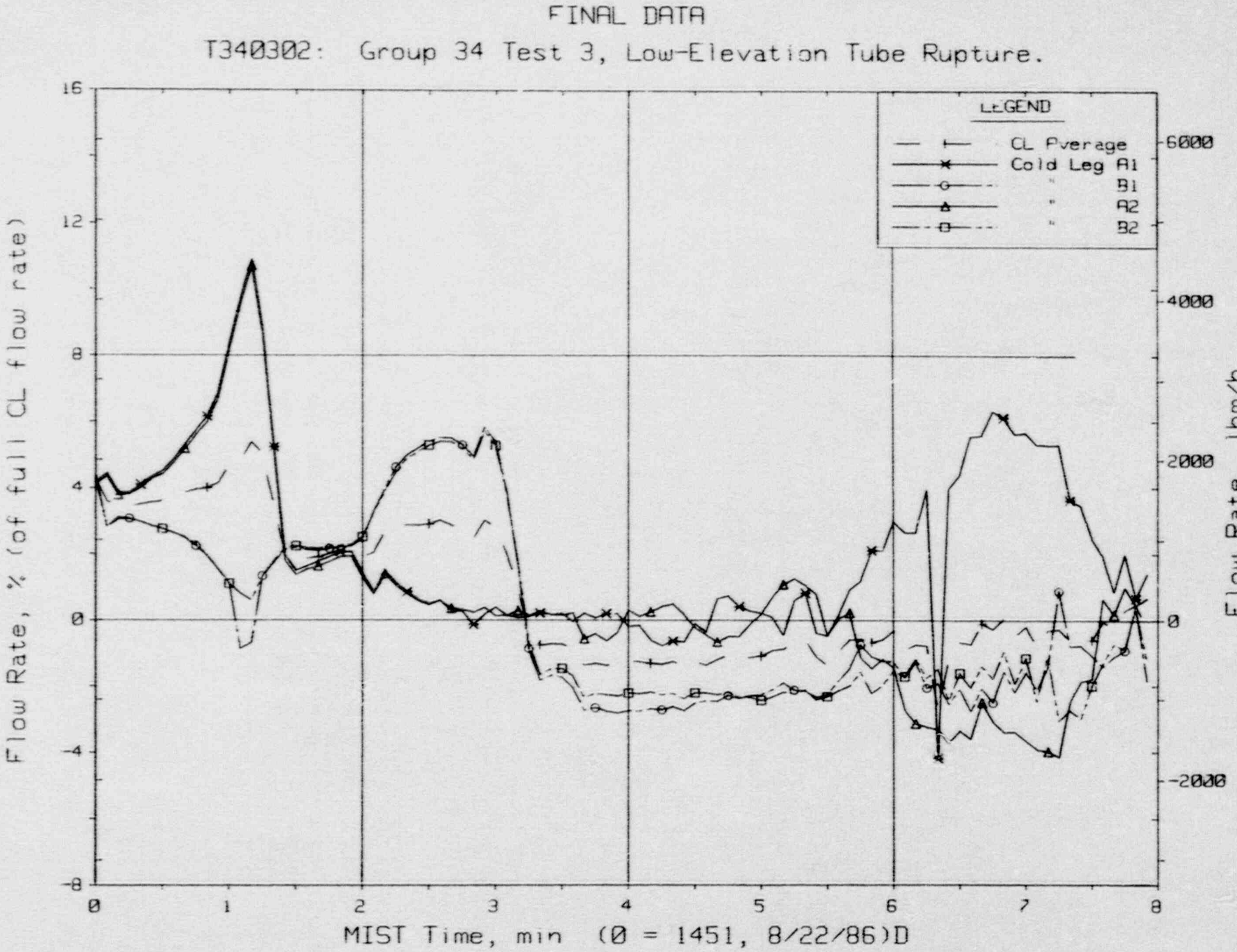

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Figure 5.3.7. Cold Leg (Venturi) Flow Rates

Mon Jul 25 13:59:22 1988

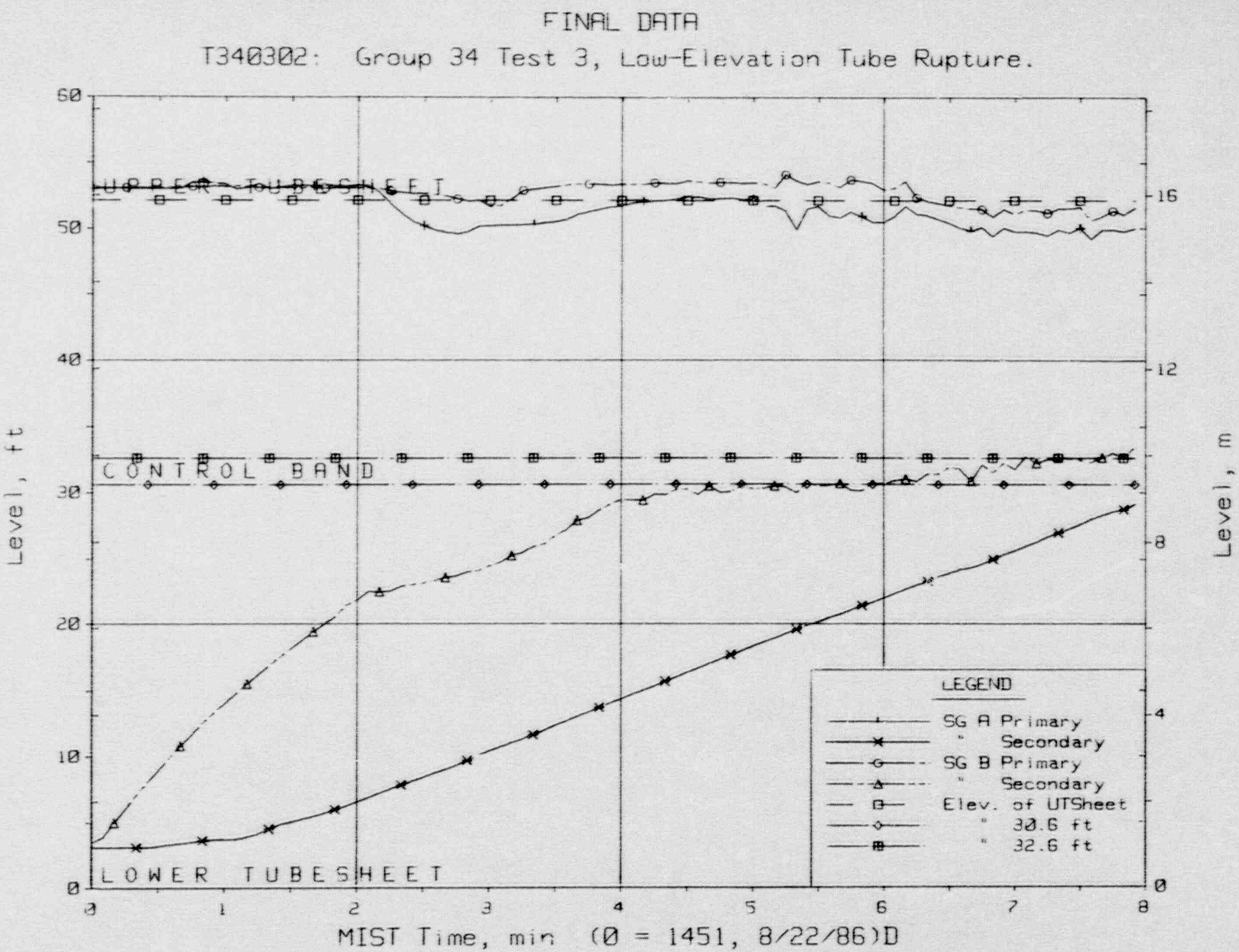

Figure 5.3.8. Steam Generator Collapsed Liquid Levels

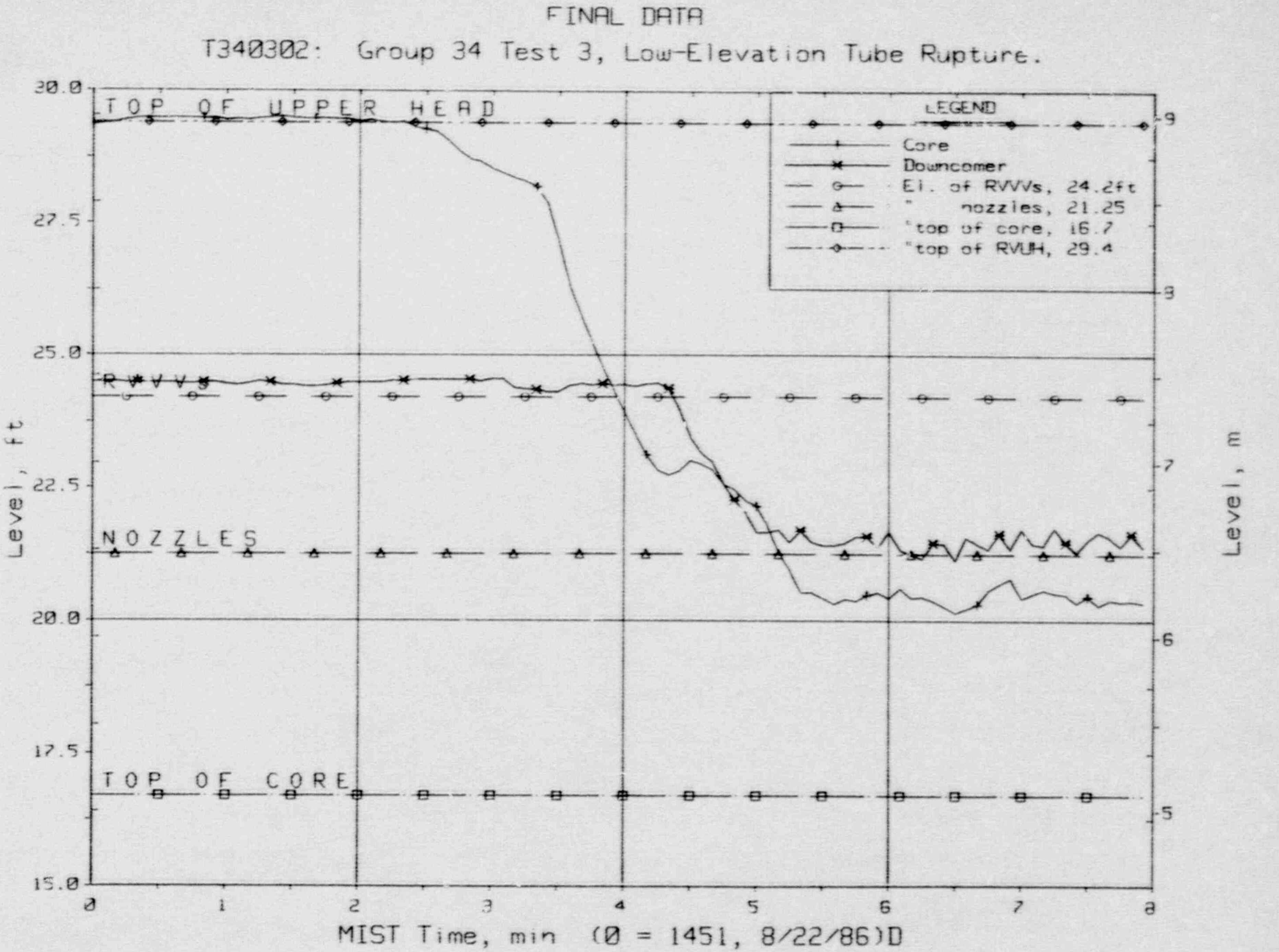

Figure 5.3.9. Core Region Collapsed Liquid Levels

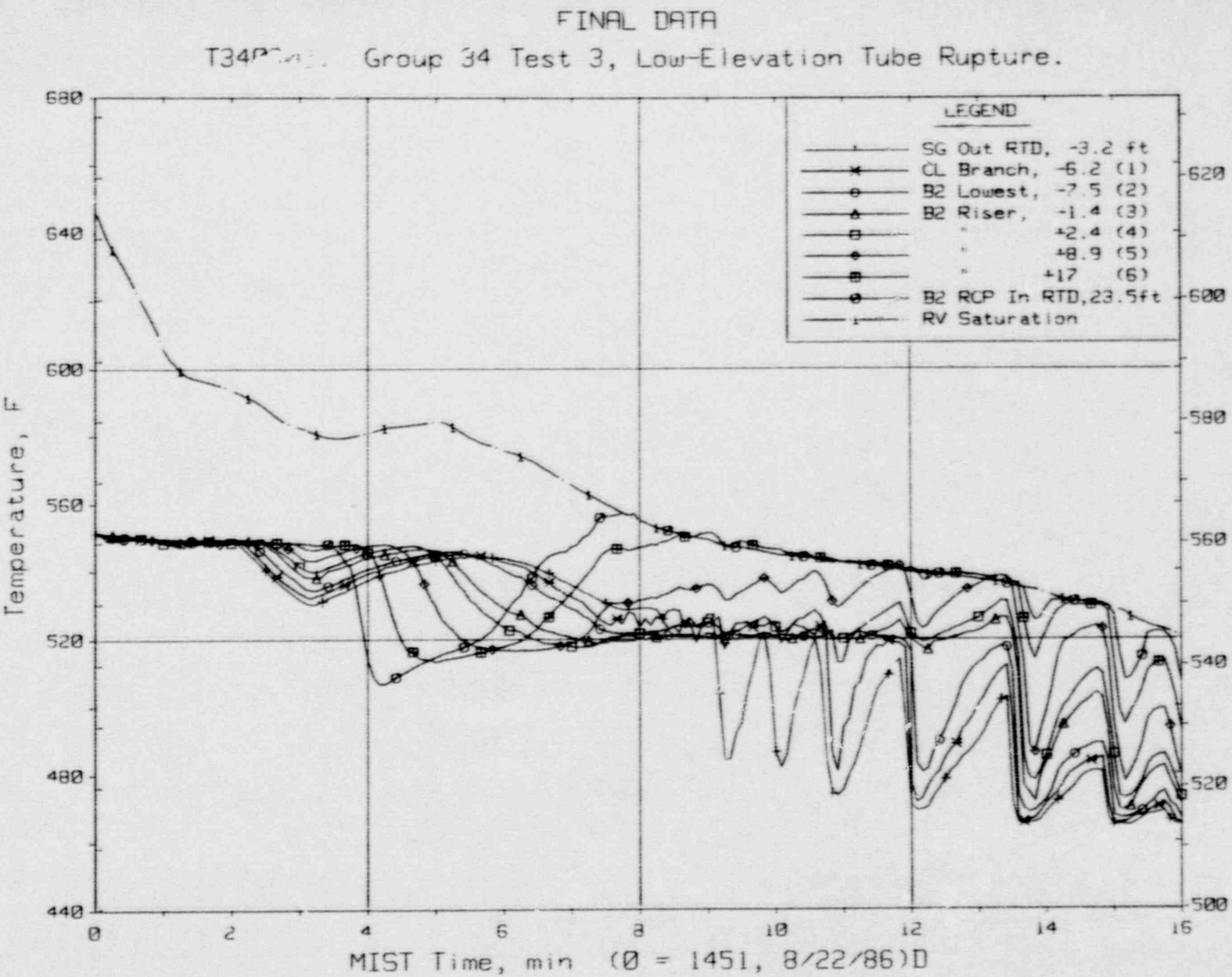

Figure 5.3.10. Cold Leg B2 Suction Fluid Temperatures (C4TCs)

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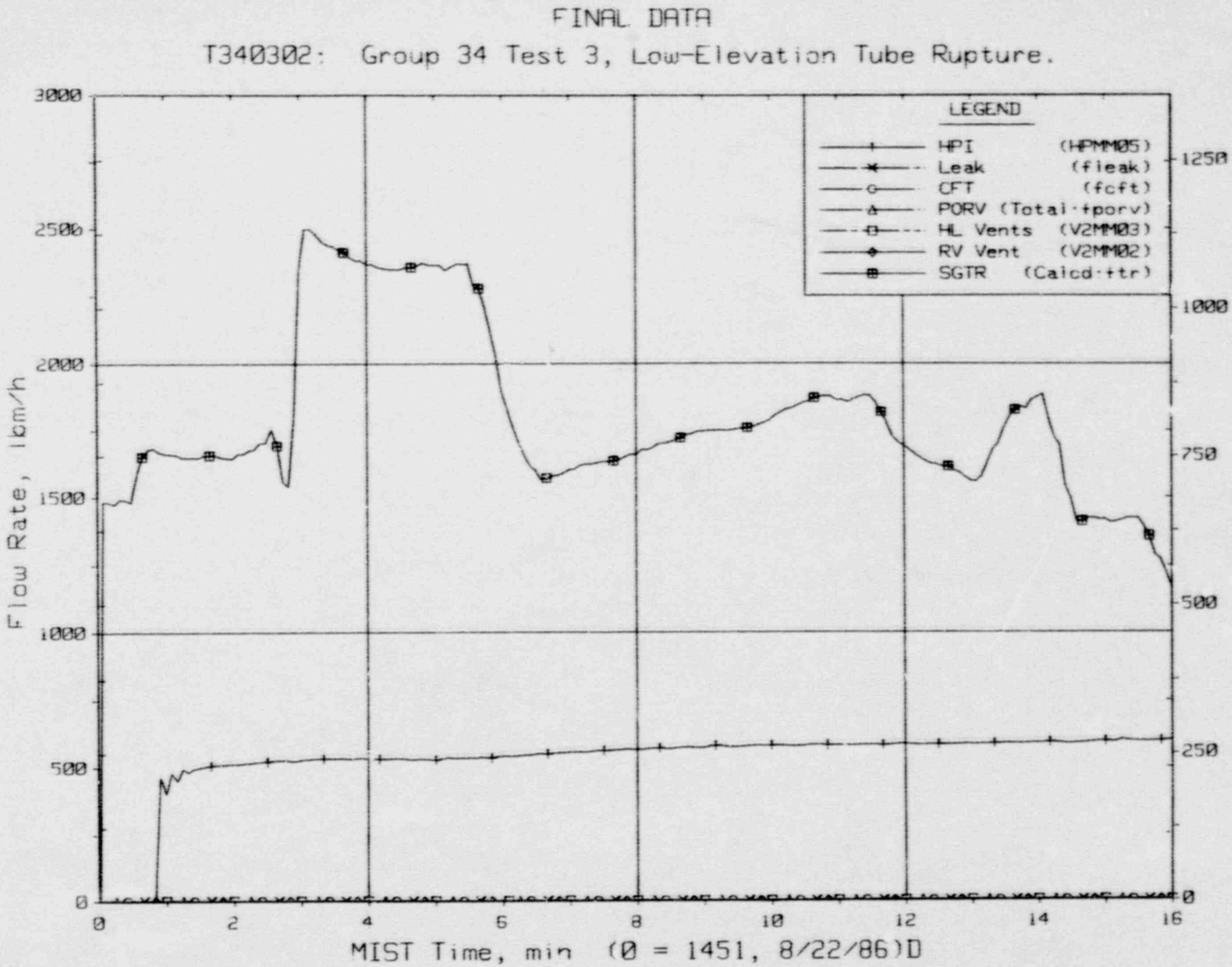
C4TC1



5-81

Figure 5.3.11. Primary System Boundary Flow Rates

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PRFL3

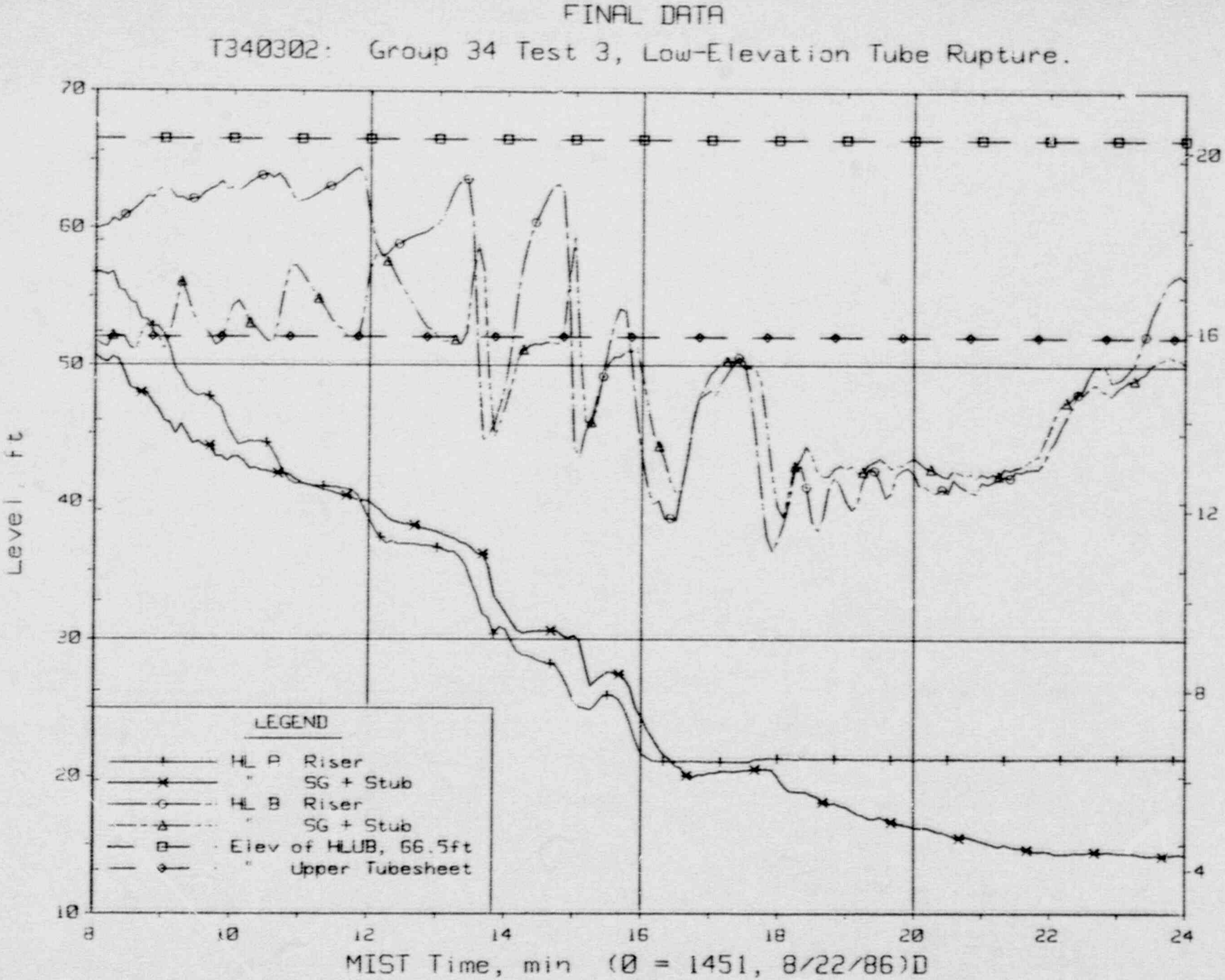

Figure 5.3.12. Hot Leg Riser and Stub Collapsed Liquid Levels

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HLLV1

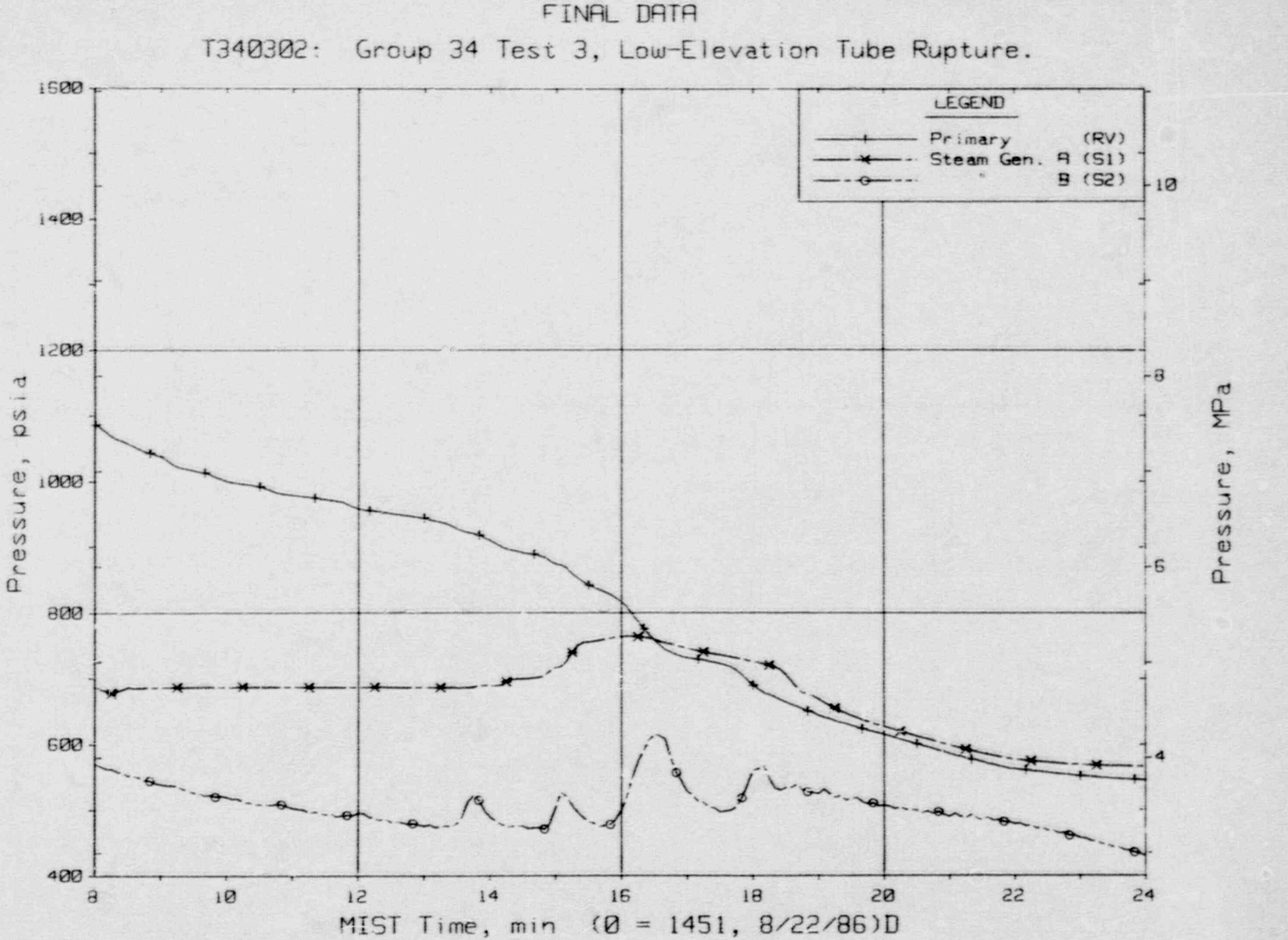

Figure 5.3.13. Primary and Secondary System Pressures (GP01s)

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PSGP1

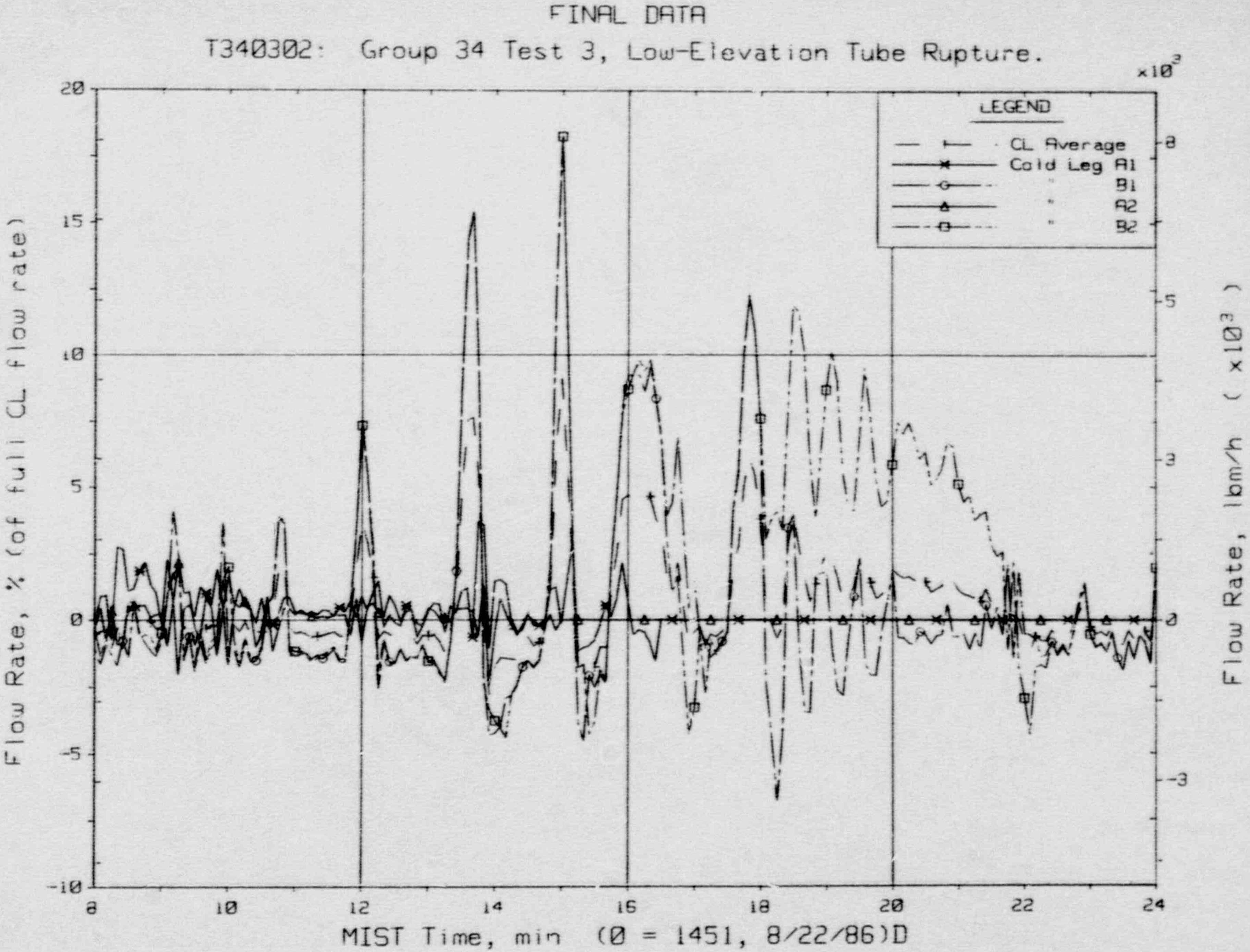

Figure 5.3.14. Cold Leg (Venturi) Flow Rates

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CLFL1

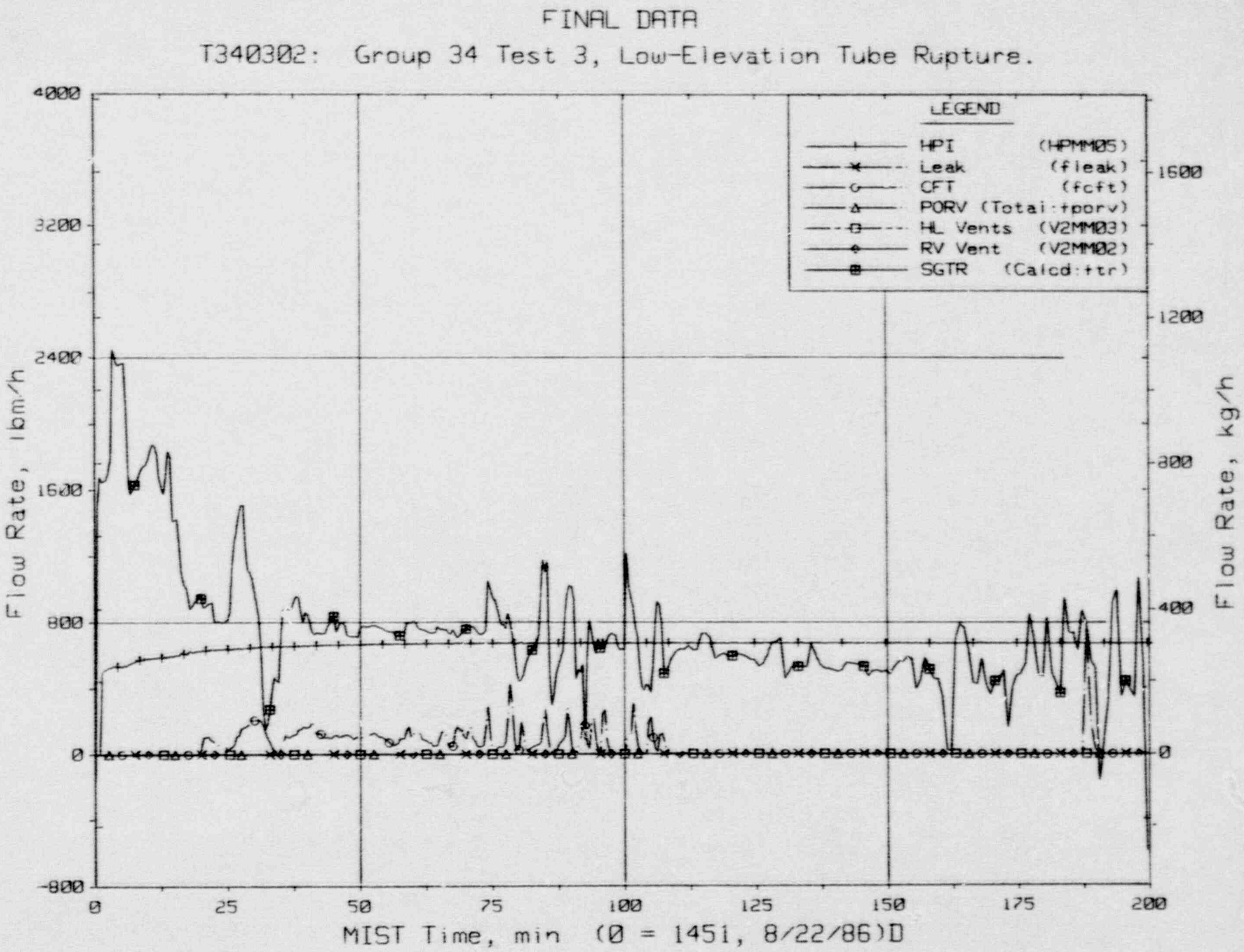

Figure 5.3.15. Primary System Boundary Flow Rates

Mon Jul 25 14:28:11 1988

PRFL3

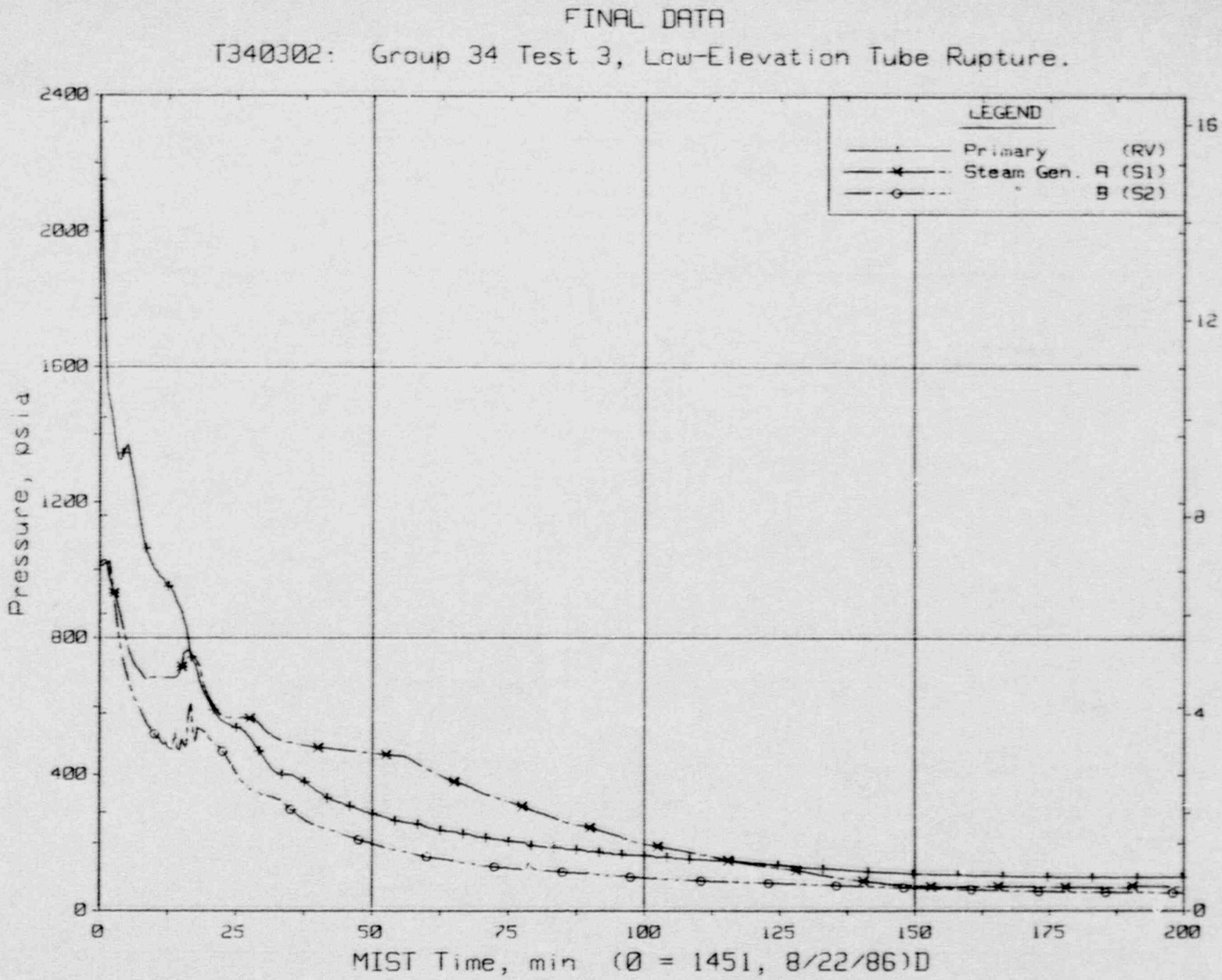

Figure 5.3.16. Primary and Secondary System Pressures (GP01s)

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PSGP1

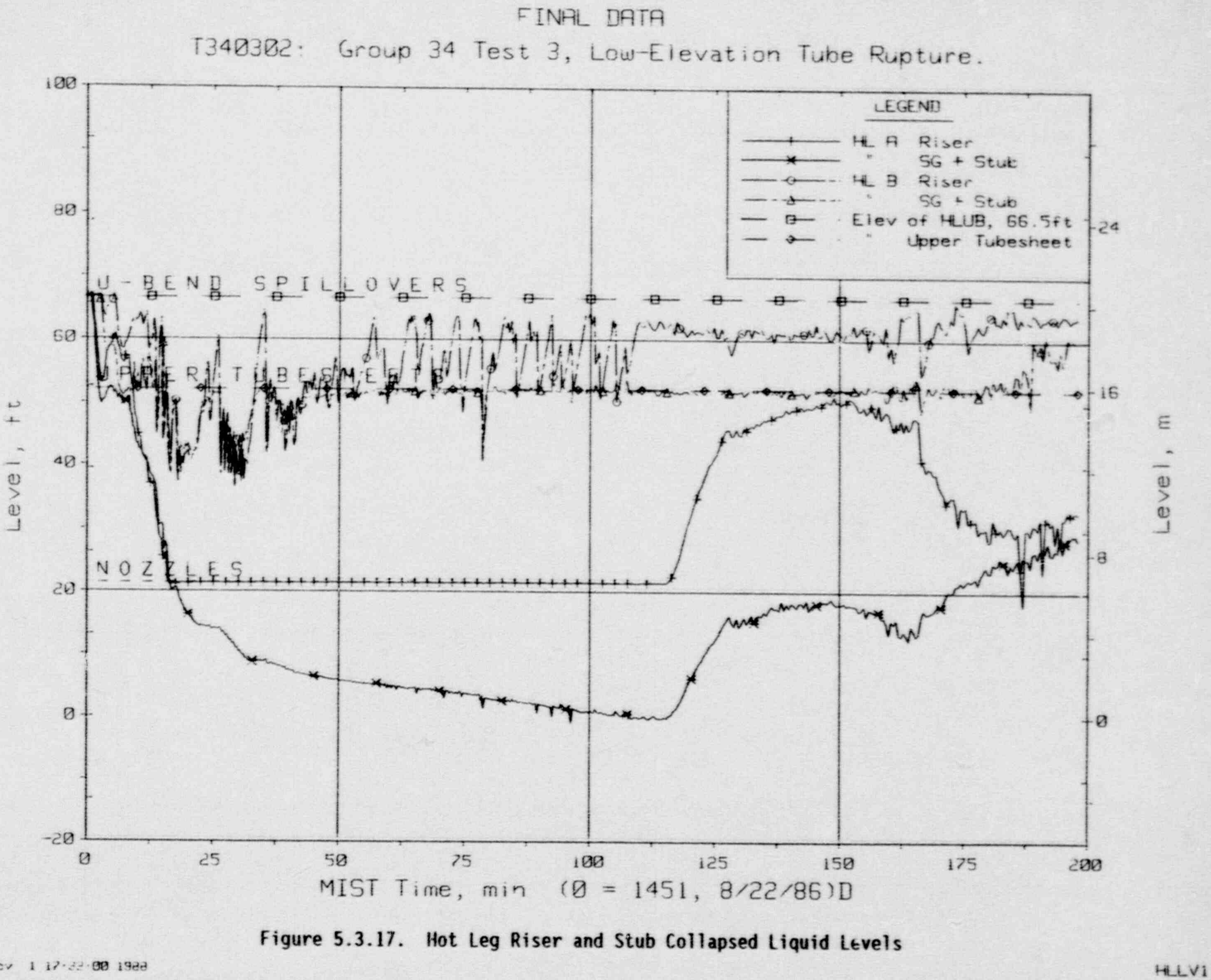

Figure 5.3.17. Hot Leg Riser and Stub Collapsed Liquid Levels

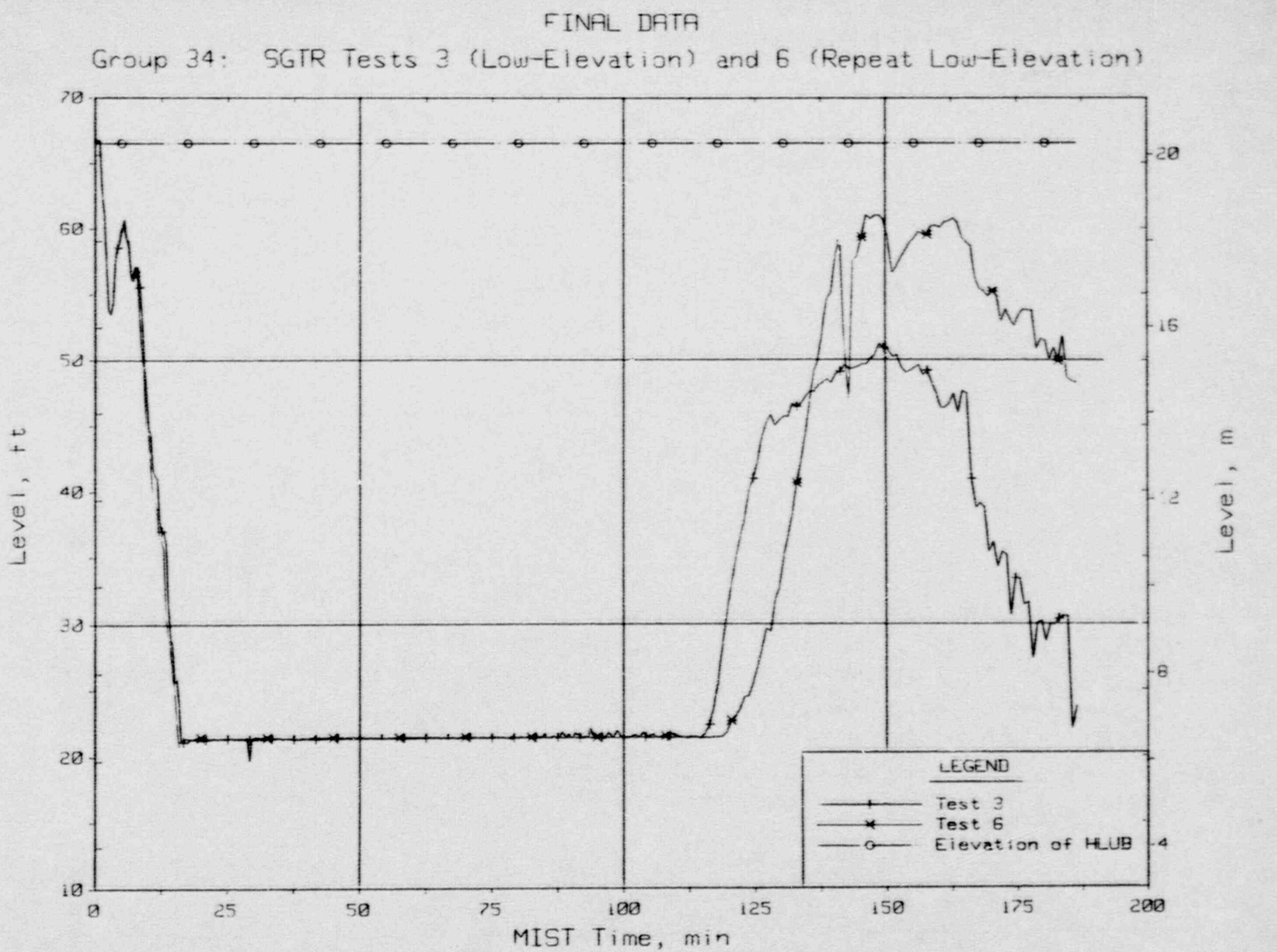

Figure 5.3.18. Hot Leg A Riser Collapsed Liquid Levels

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3CHLLV1

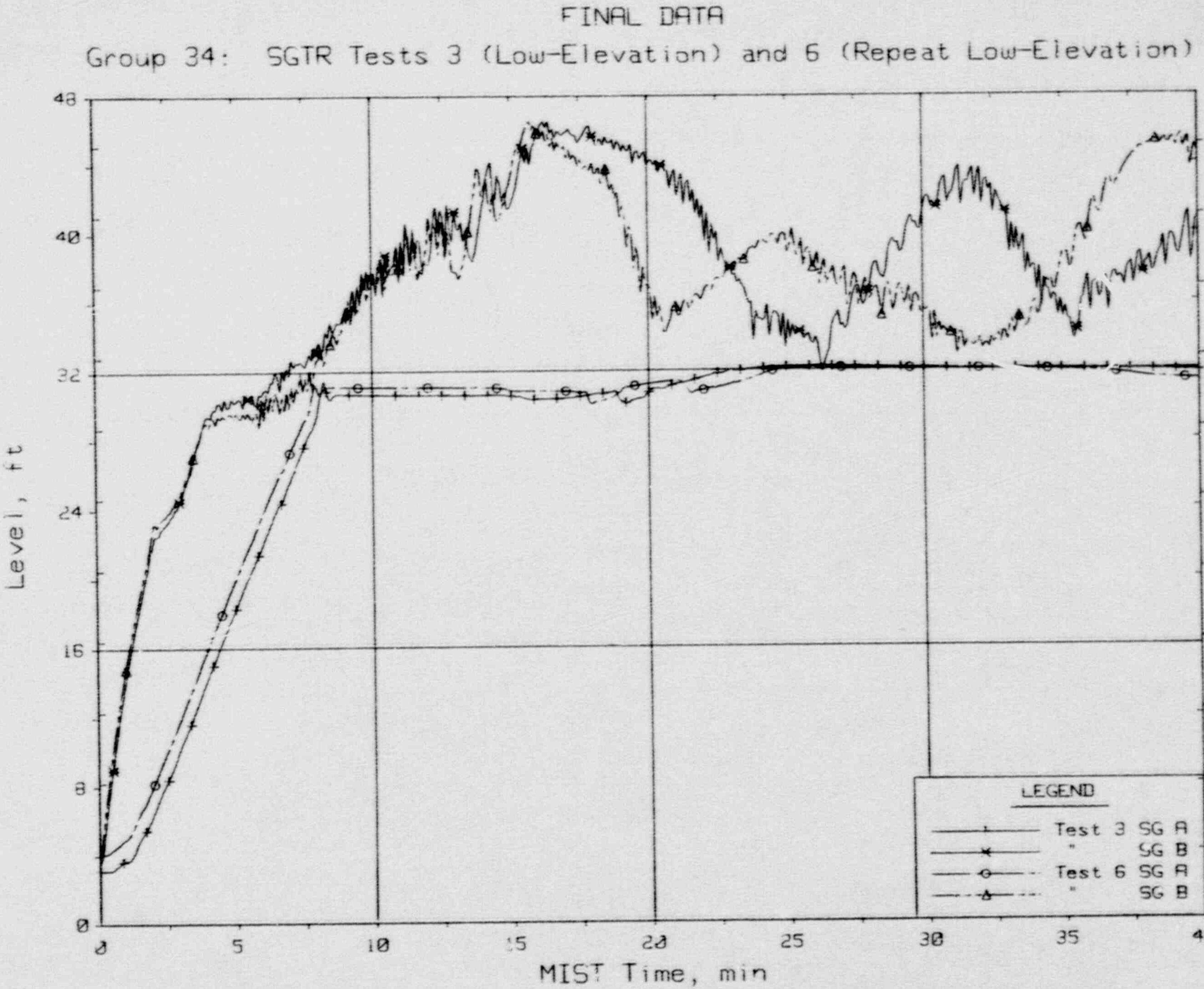

Figure 5.3.19. Steam Generator Secondary Levels

36SGLV1

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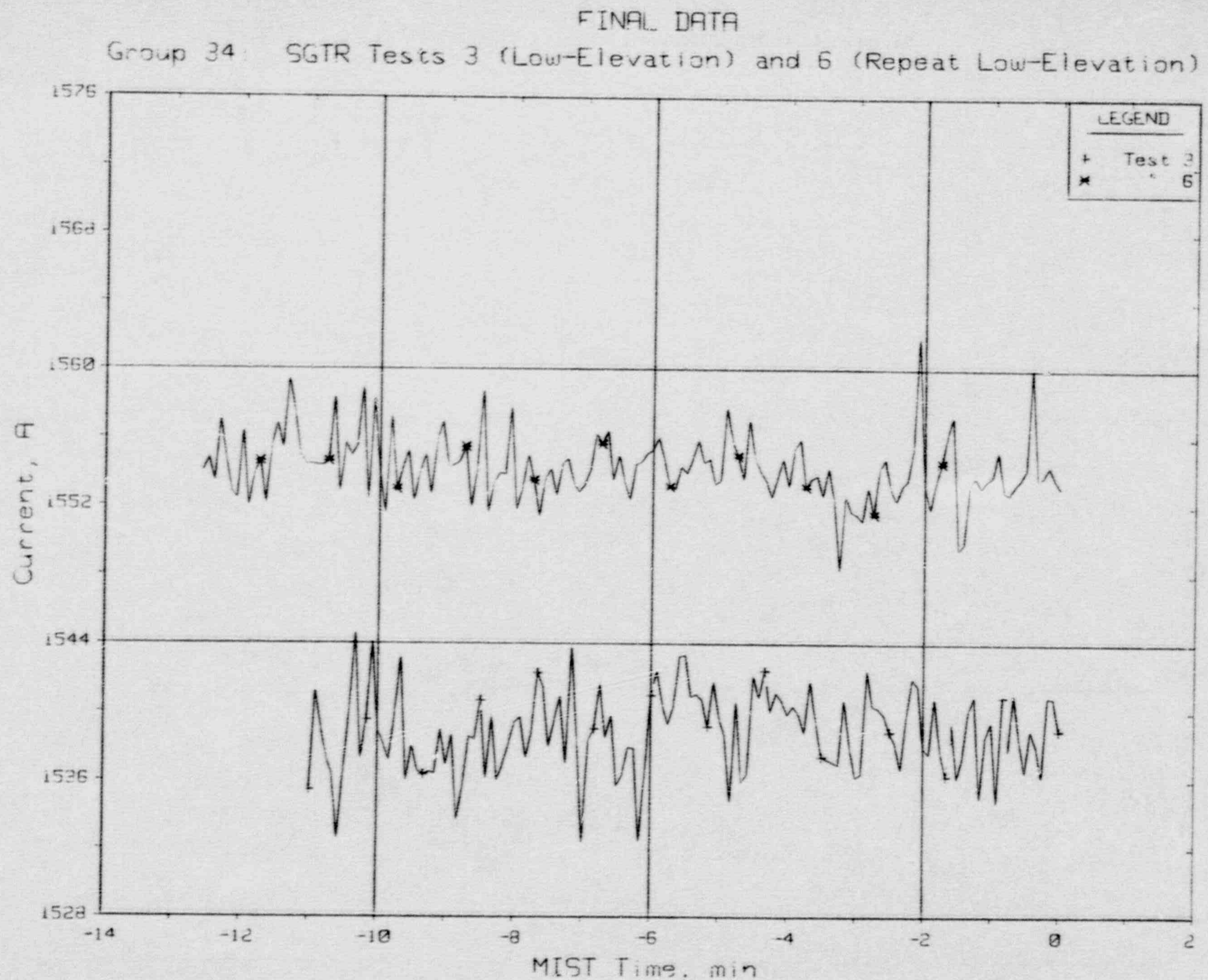

Figure 5.3.20. Current to Core Heaters

36COAM1

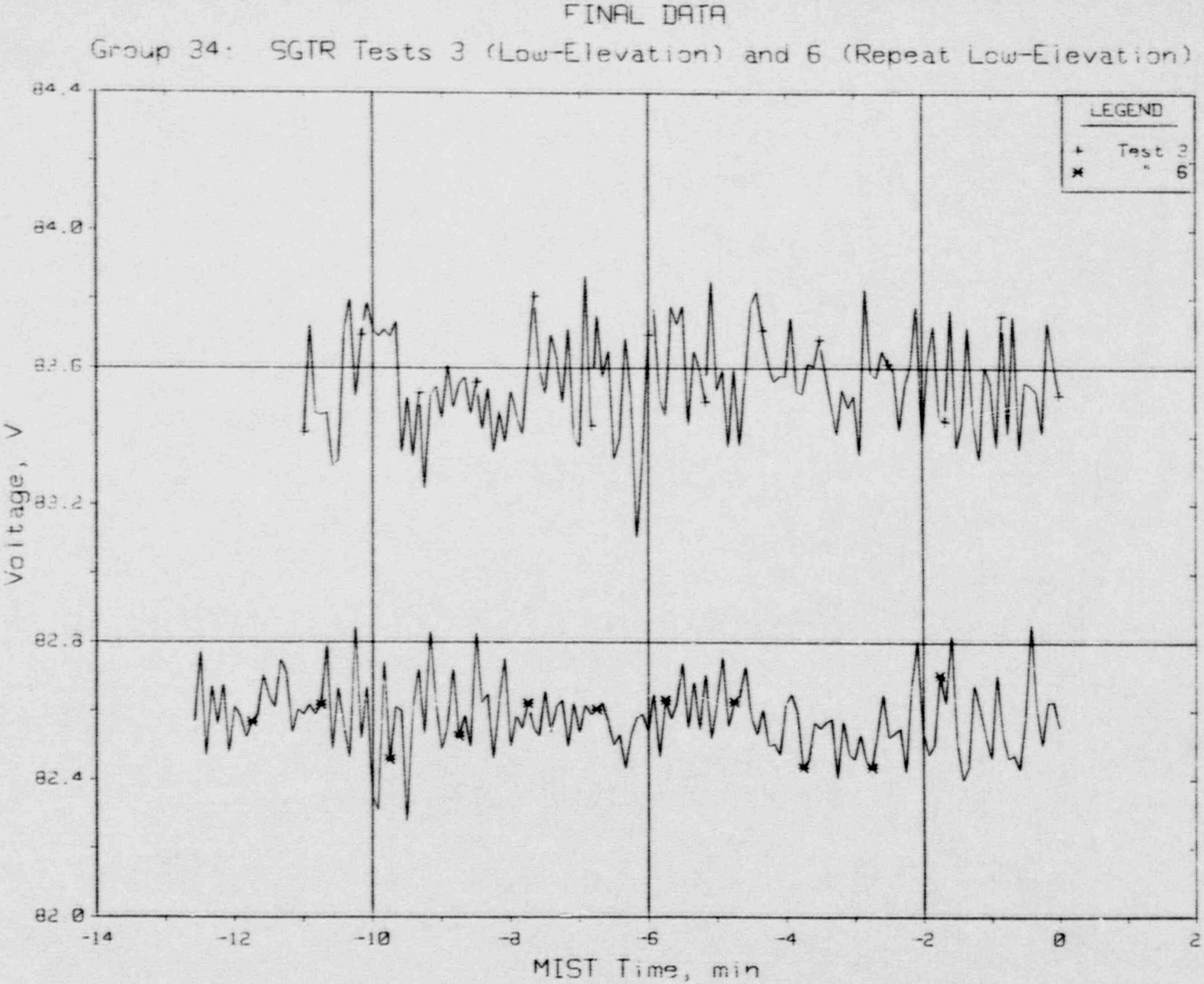

Figure 5.3.21. Electrical Potential Across Core Heaters

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3GCOV01

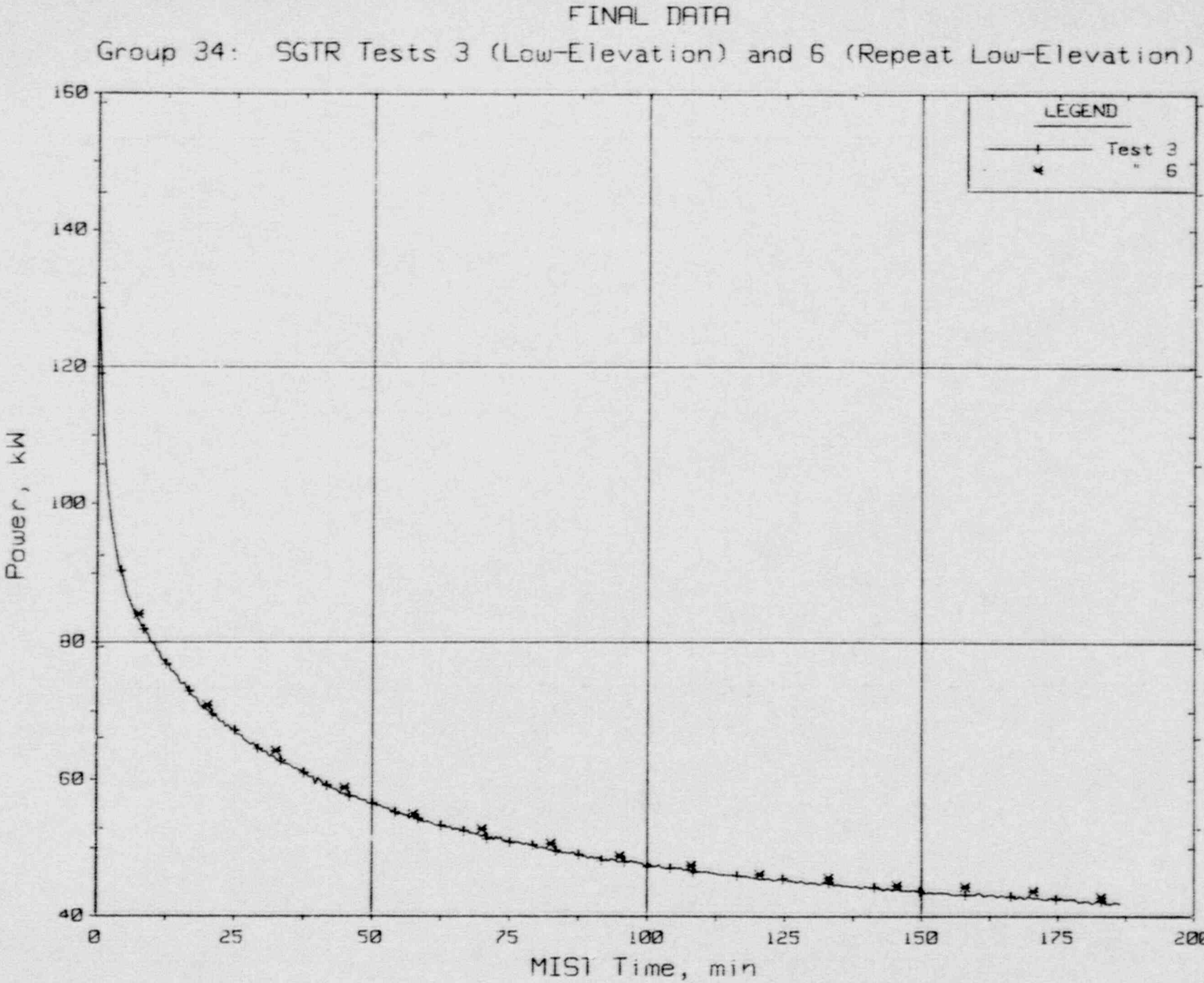

Figure 5.3.22. Core Power

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36RVKW1

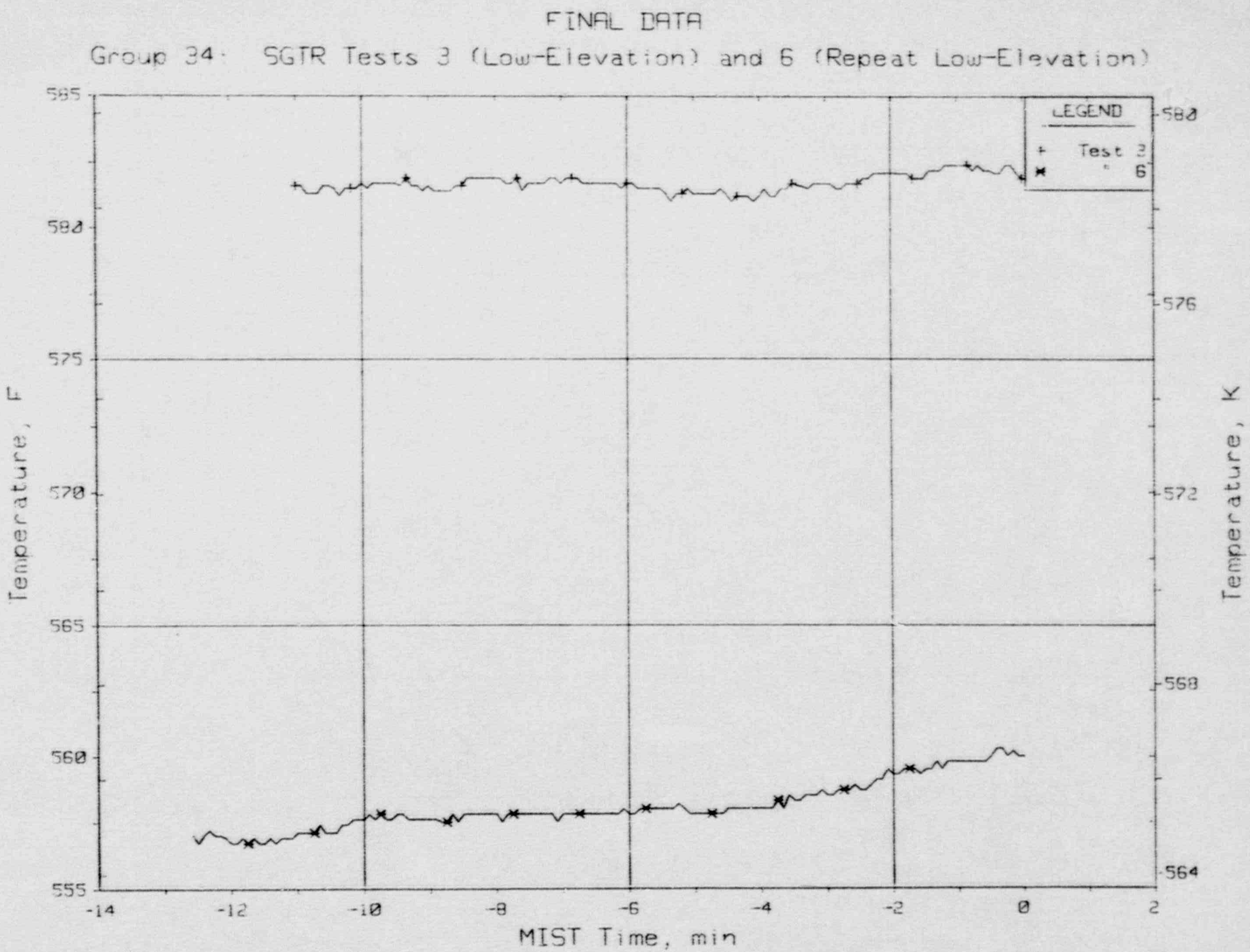

Figure 5.3.23. Pressurizer Surge Line Metal Temperature (PZMT02)

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36PZMT1

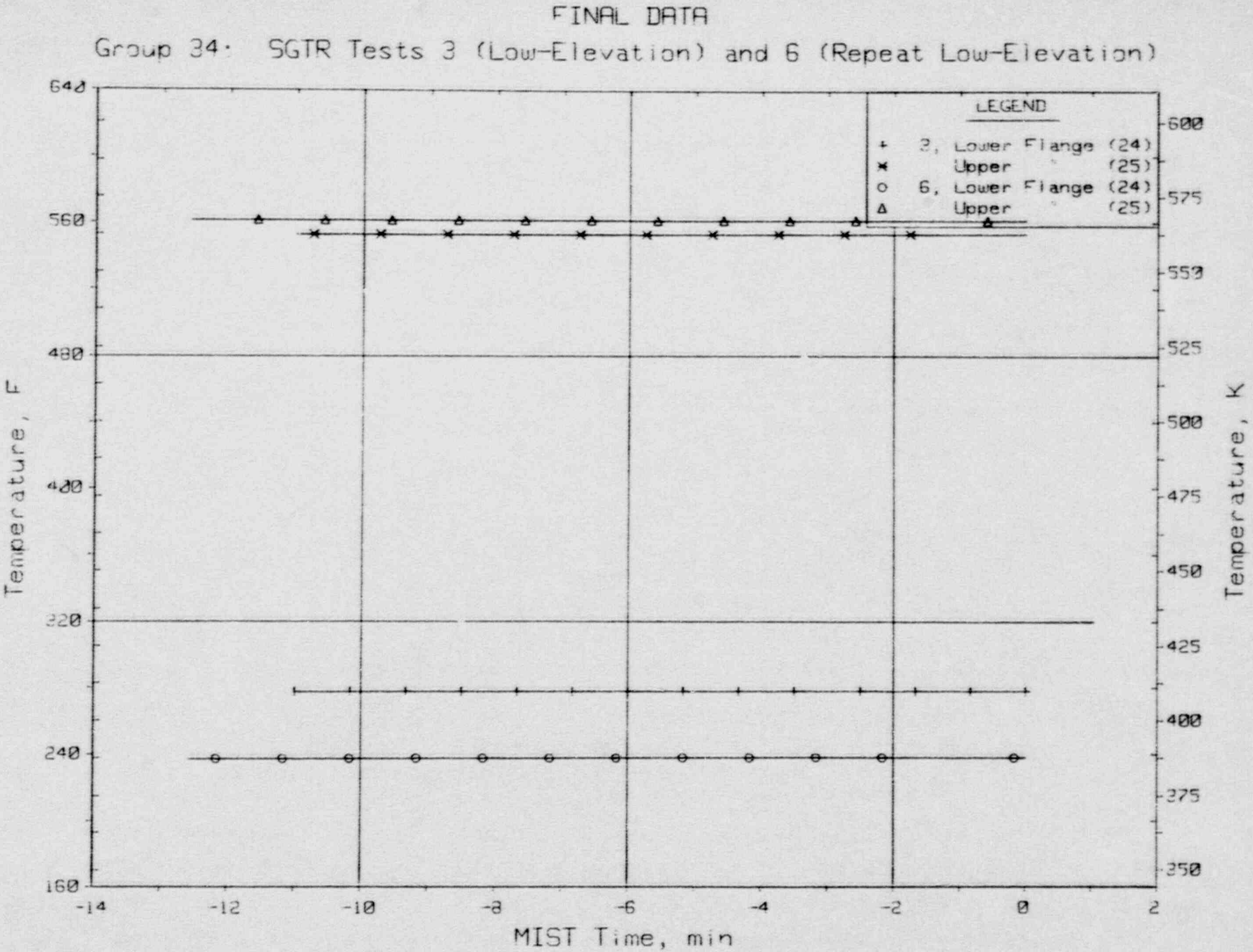

Figure 5.3.24. Reactor Vessel Flange Metal Temperatures (RVMTs)

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36RVMT1

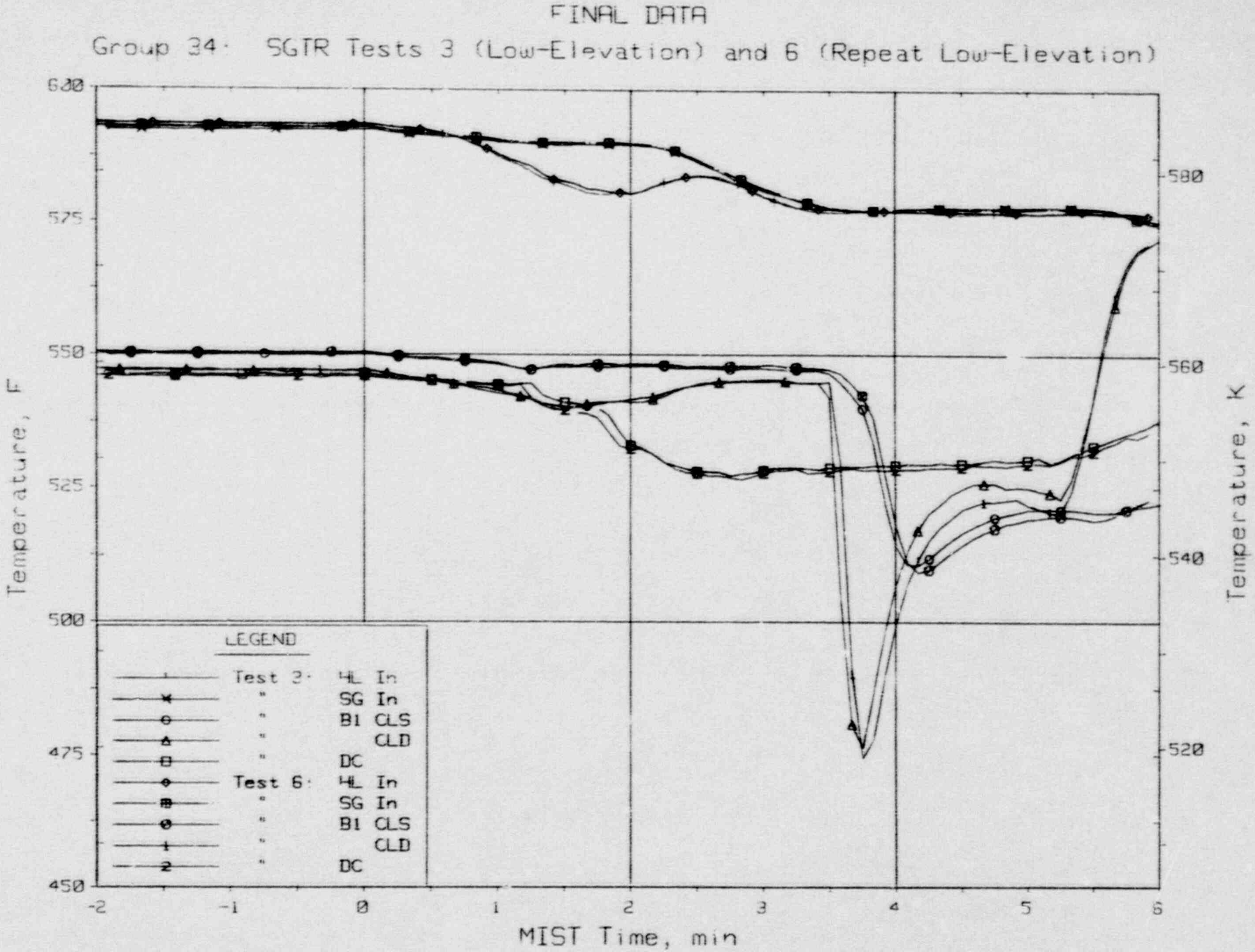

Figure 5.3.25. Loop B Primary Fluid Temperatures (RTDs)

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36PRRT2

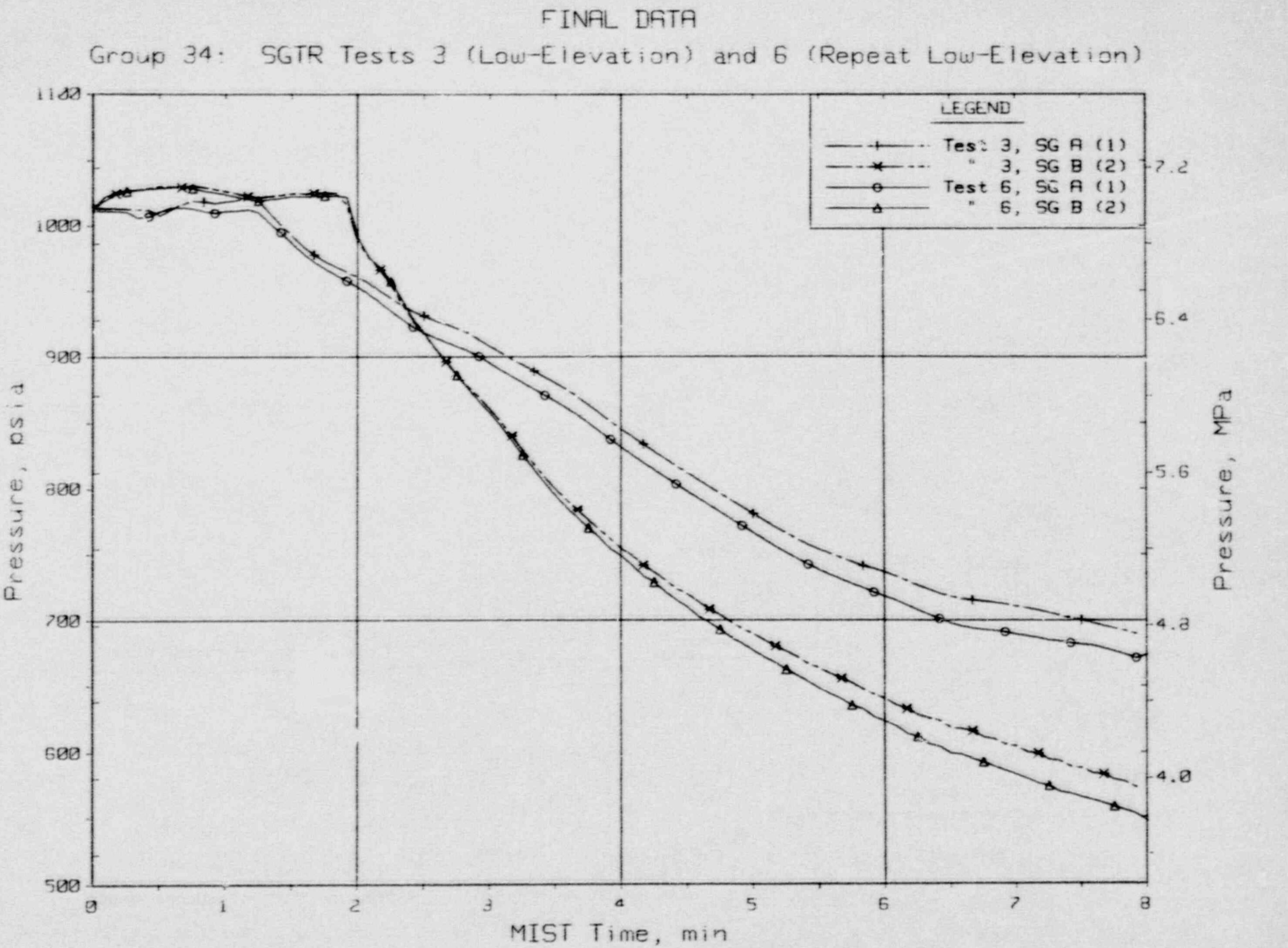

Figure 5.3.26. Steam Generator Secondary Pressures (SnGP01s)

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36SCPC1

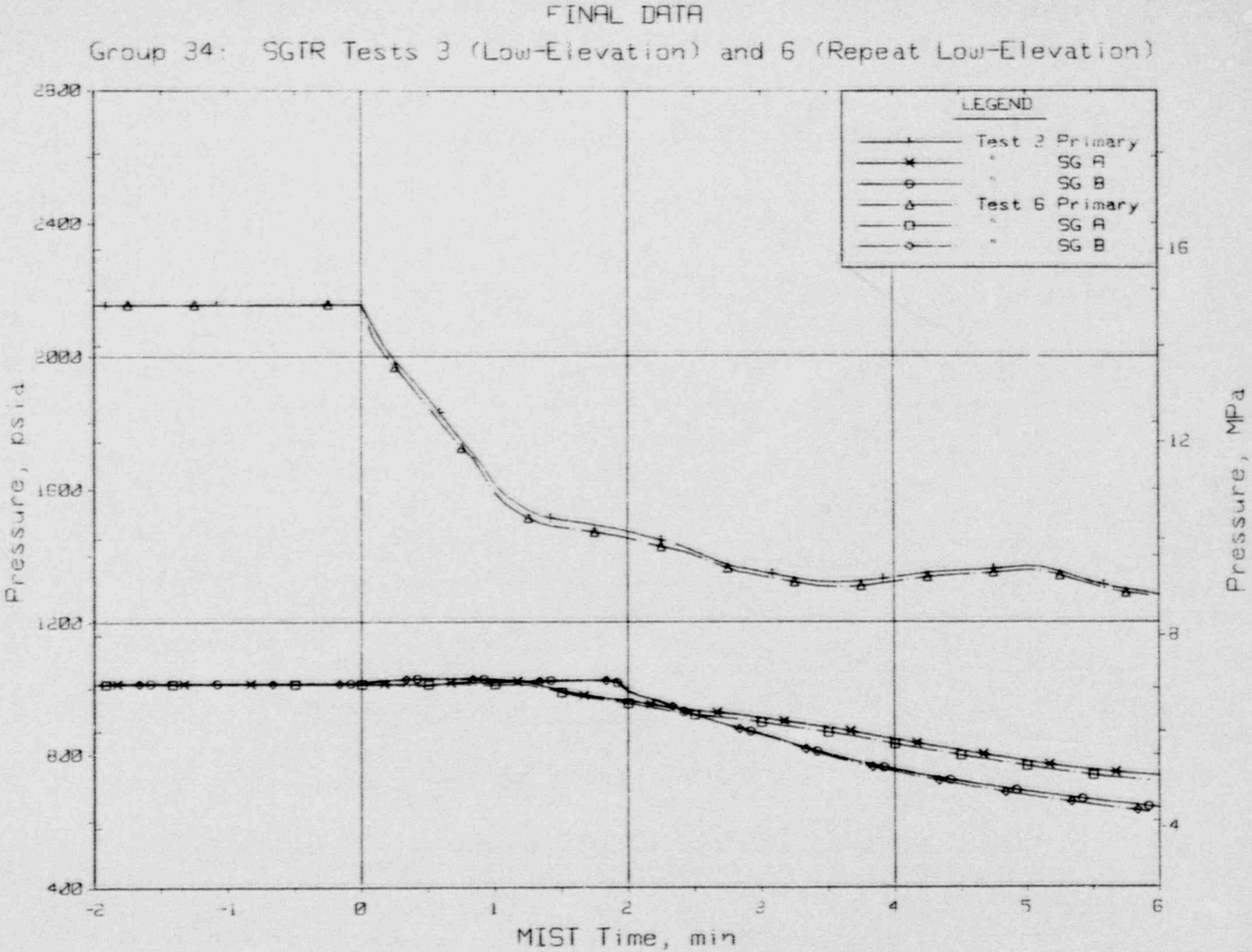

Figure 5.3.27. Primary and Secondary System Pressures (GP01s)

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36PSGP1

FINAL DATA

Group 34: SGTR Tests 3 (Low-Elevation) and 6 (Repeat Low-Elevation)

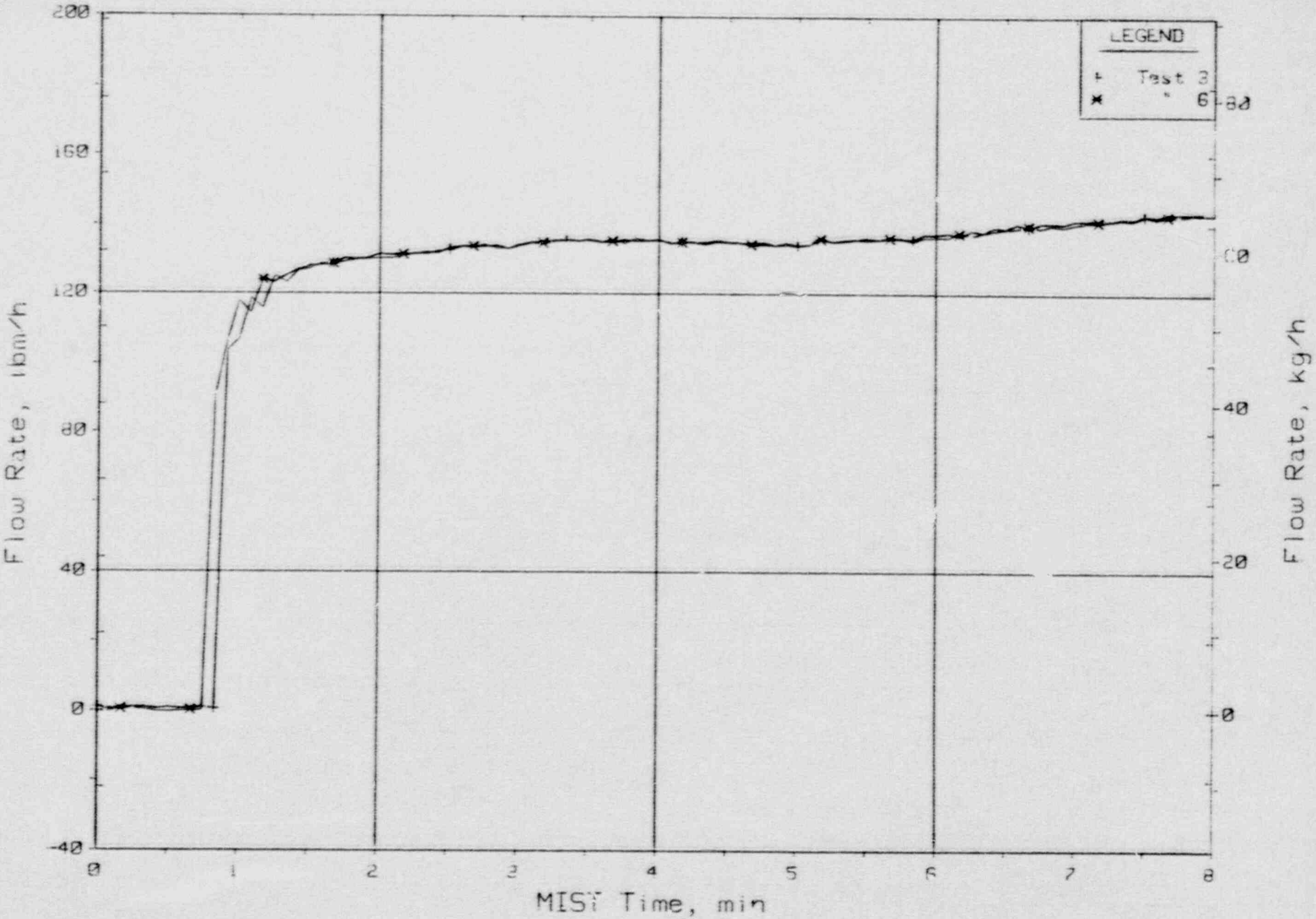

Figure 5.3.28. HPI Flow Rate to CL A1

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3GHPFL1

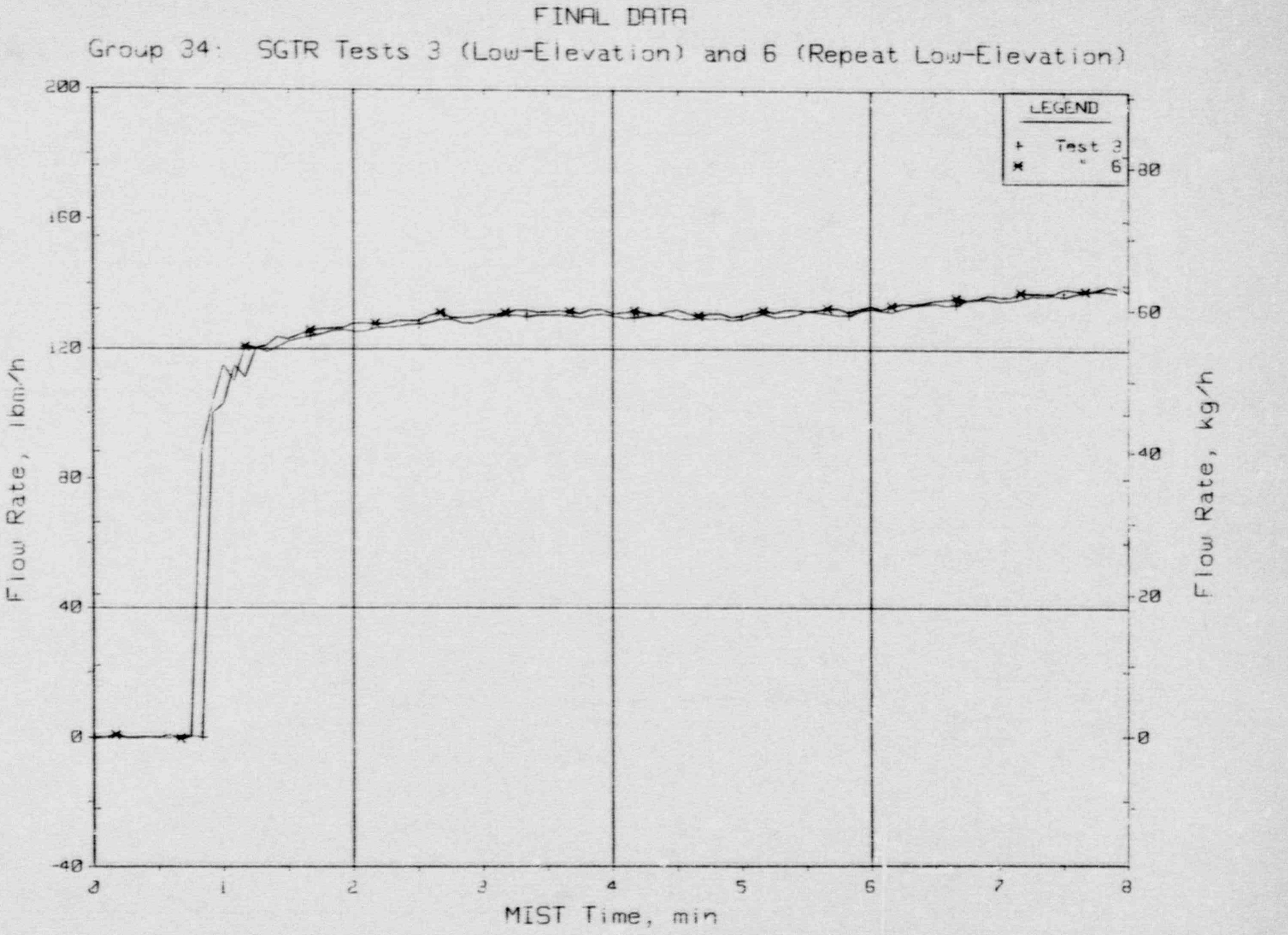

Figure 5.3.29. HPI Flow Rate to CL B1

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3GHPFL2

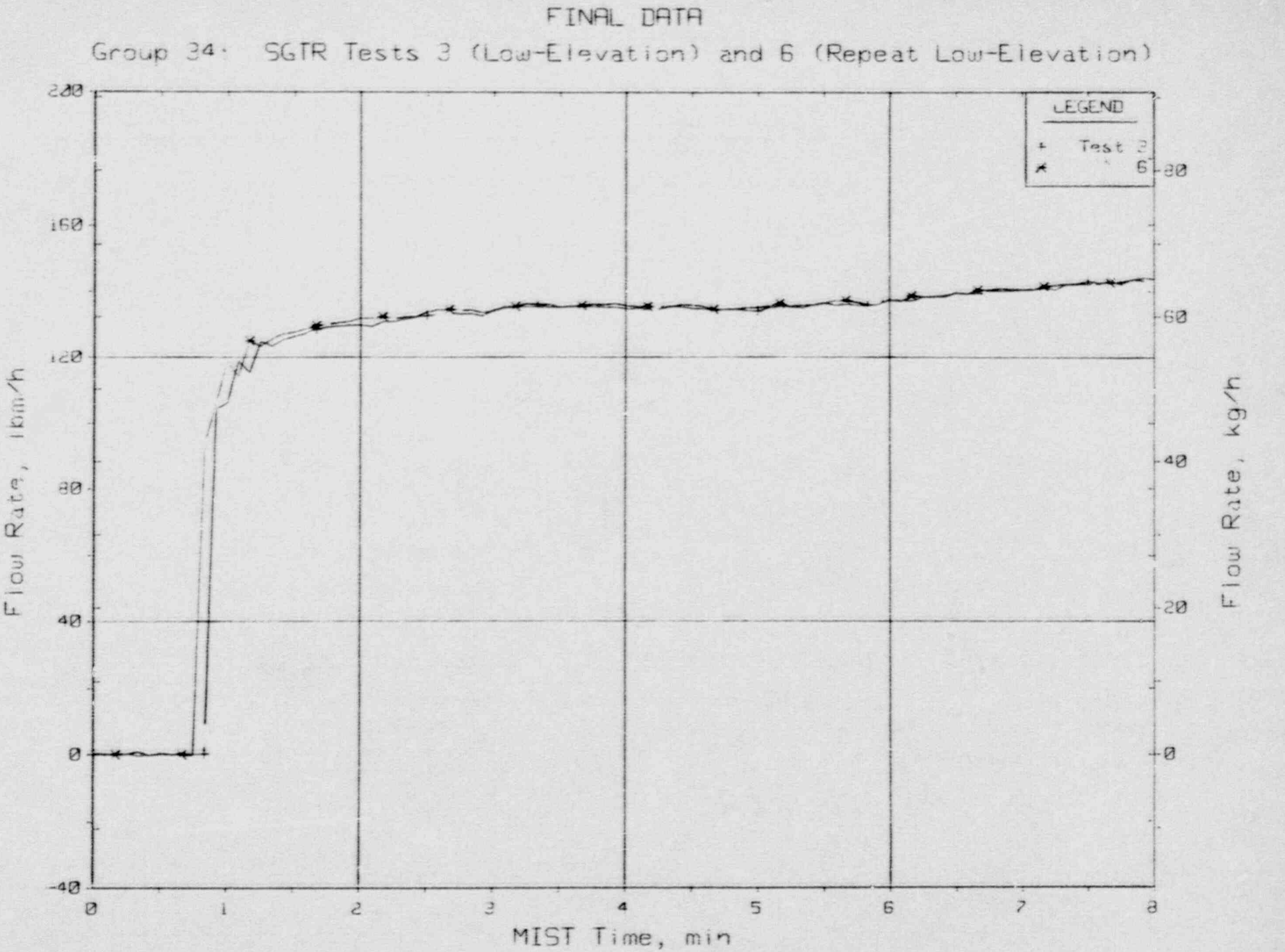

Figure 5.3.30. HPI Flow Rate to CL A2

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36HPFL2

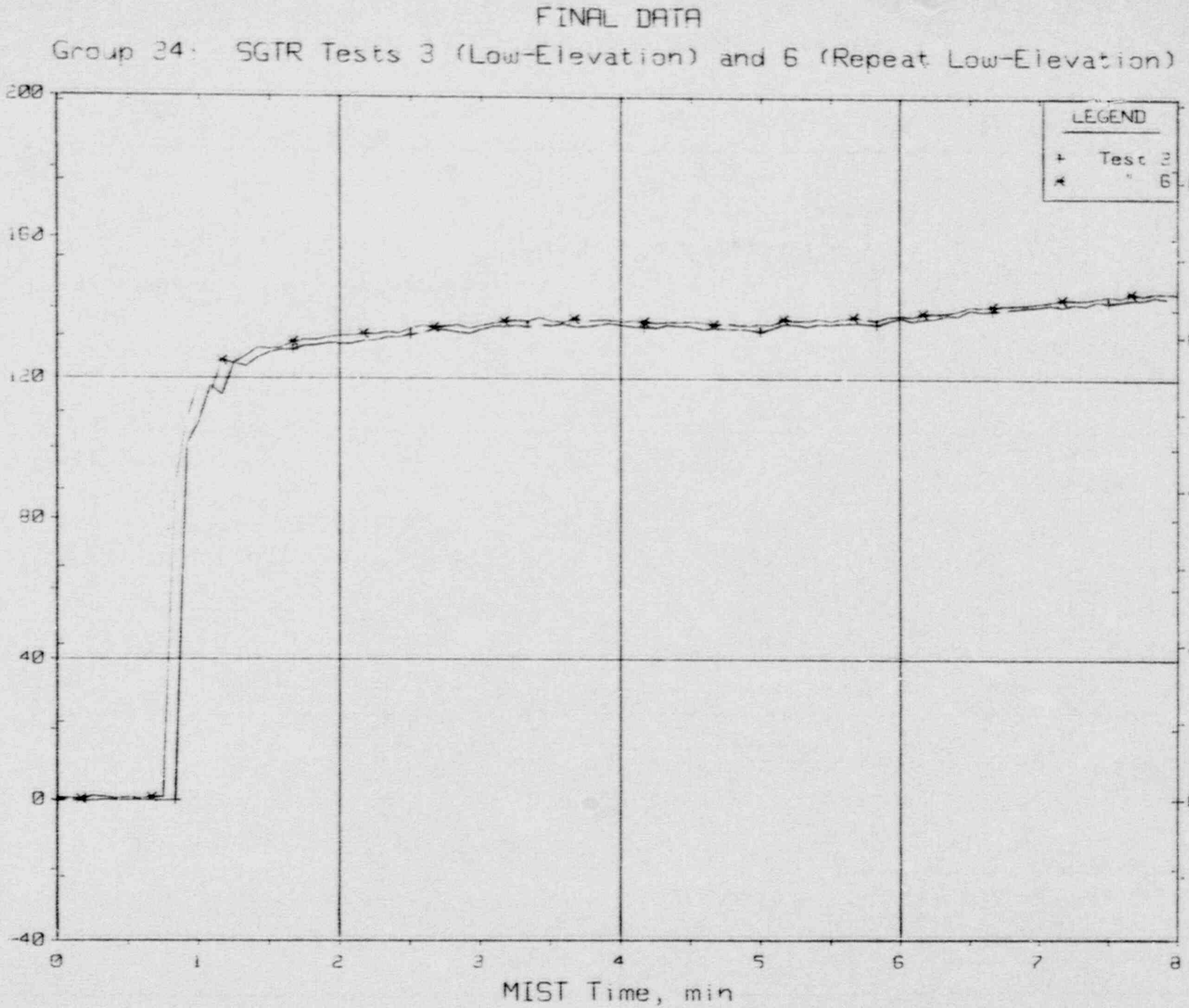

Figure 5.3.31. HPI Flow Rate to CL B2

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36HPFL4

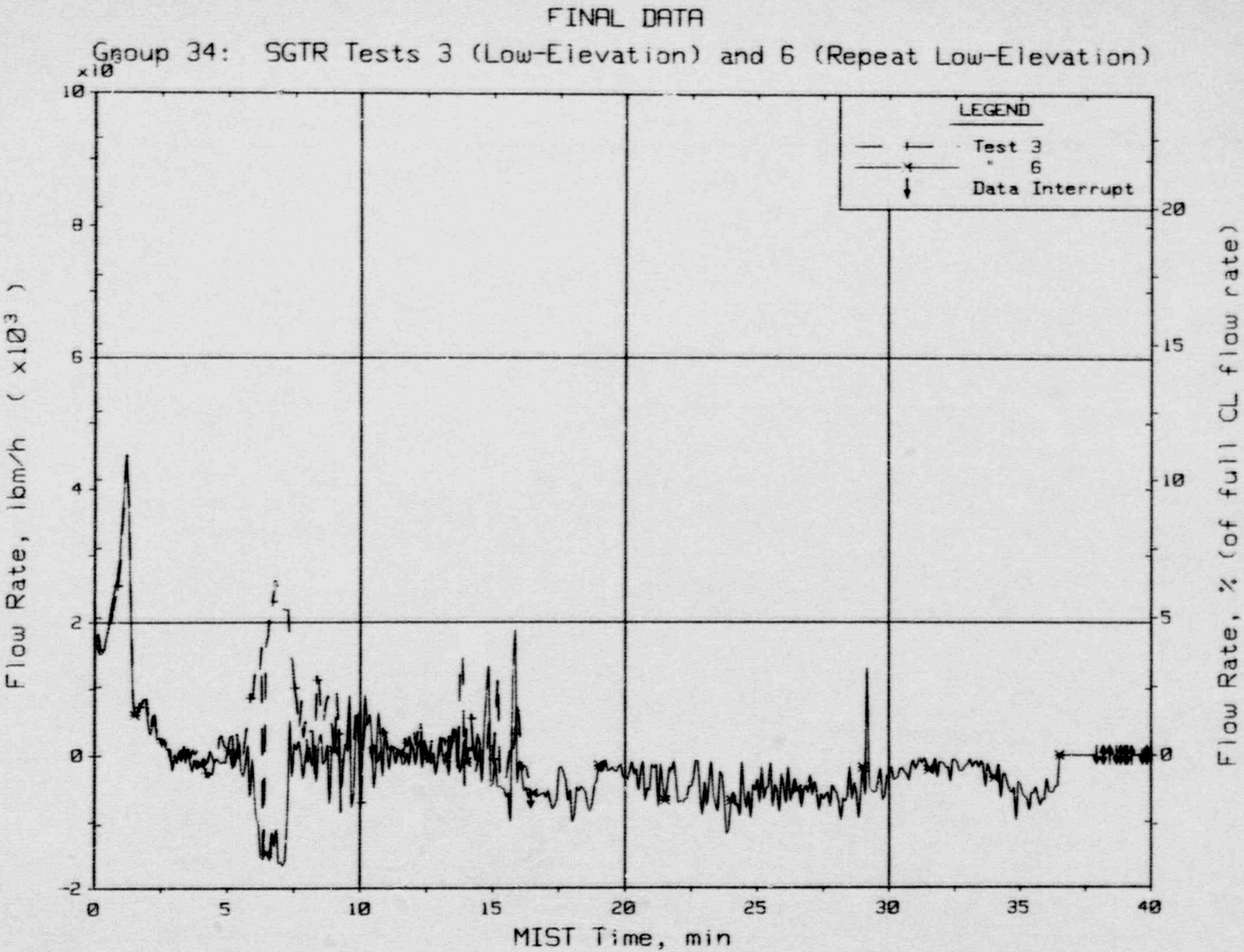

Figure 5.3.32. Cold Leg A1 (Venturi) Flow Rates

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36CLFL1

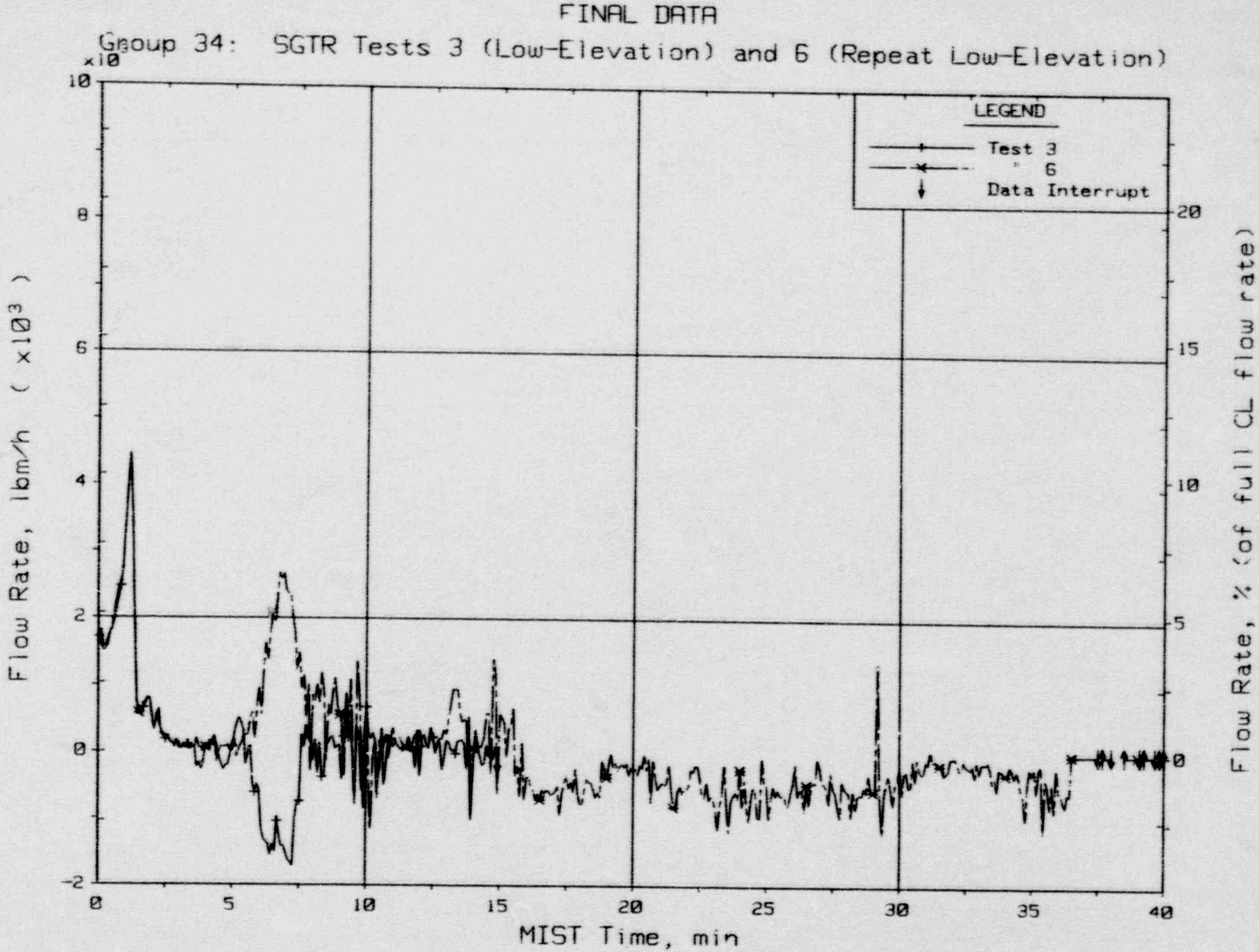

Figure 5.3.33. Cold Leg A2 (Venturi) Flow Rates

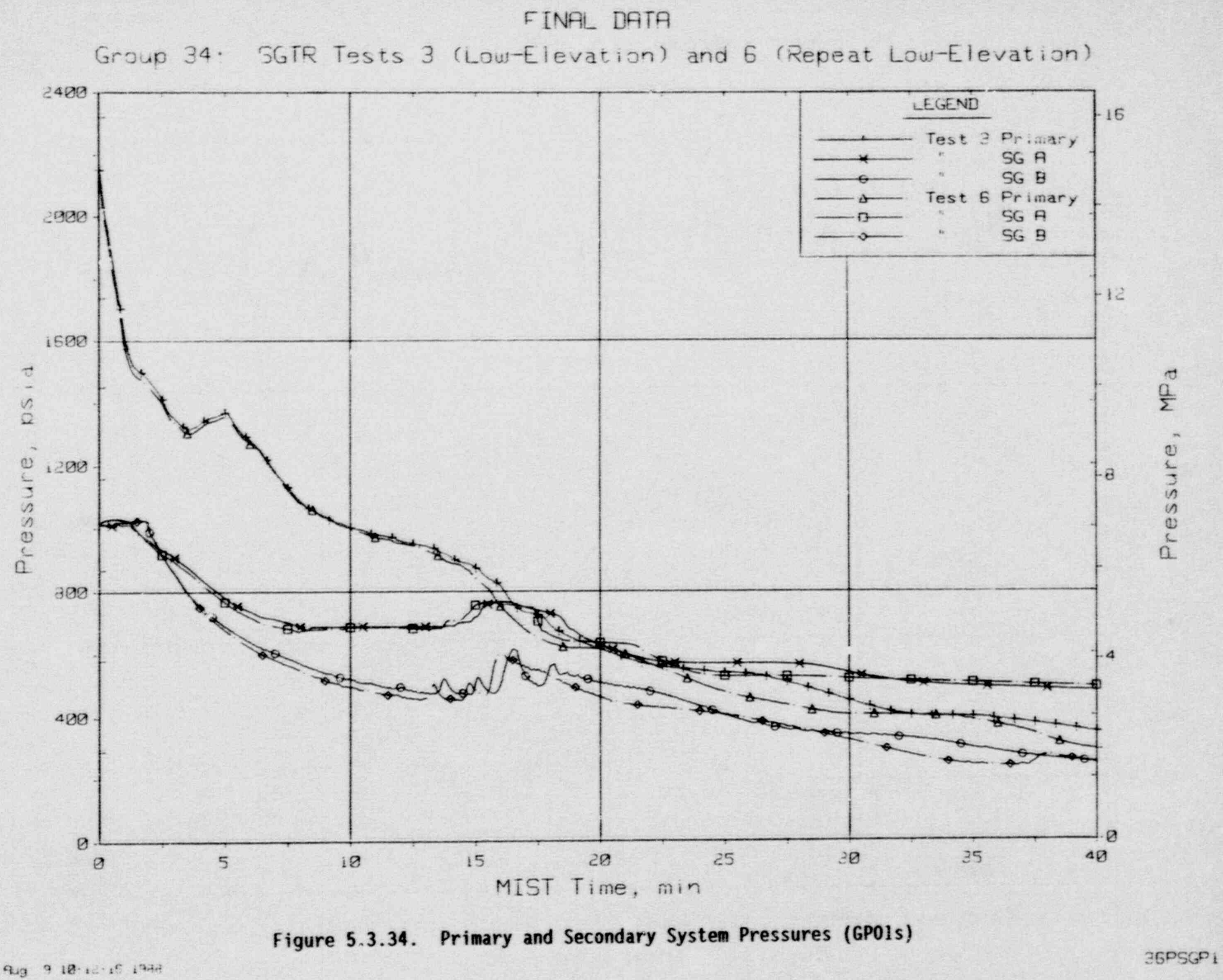

Figure 5.3.34. Primary and Secondary System Pressures (GPOIs)

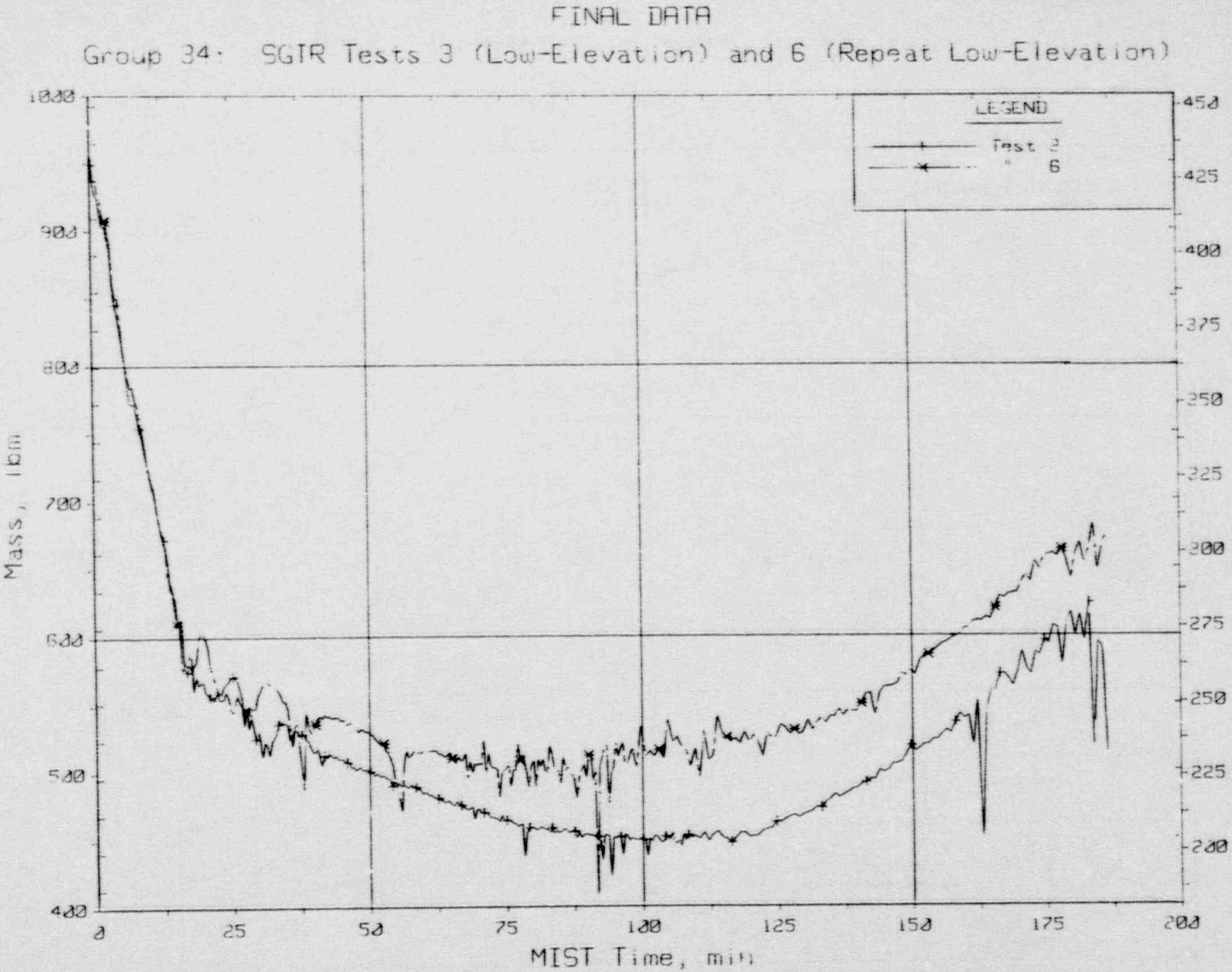

Figure 5.3.35. Primary System Total (Indicated) Fluid Mass

36PRML1

Tue Aug 9 13:41:04 1988

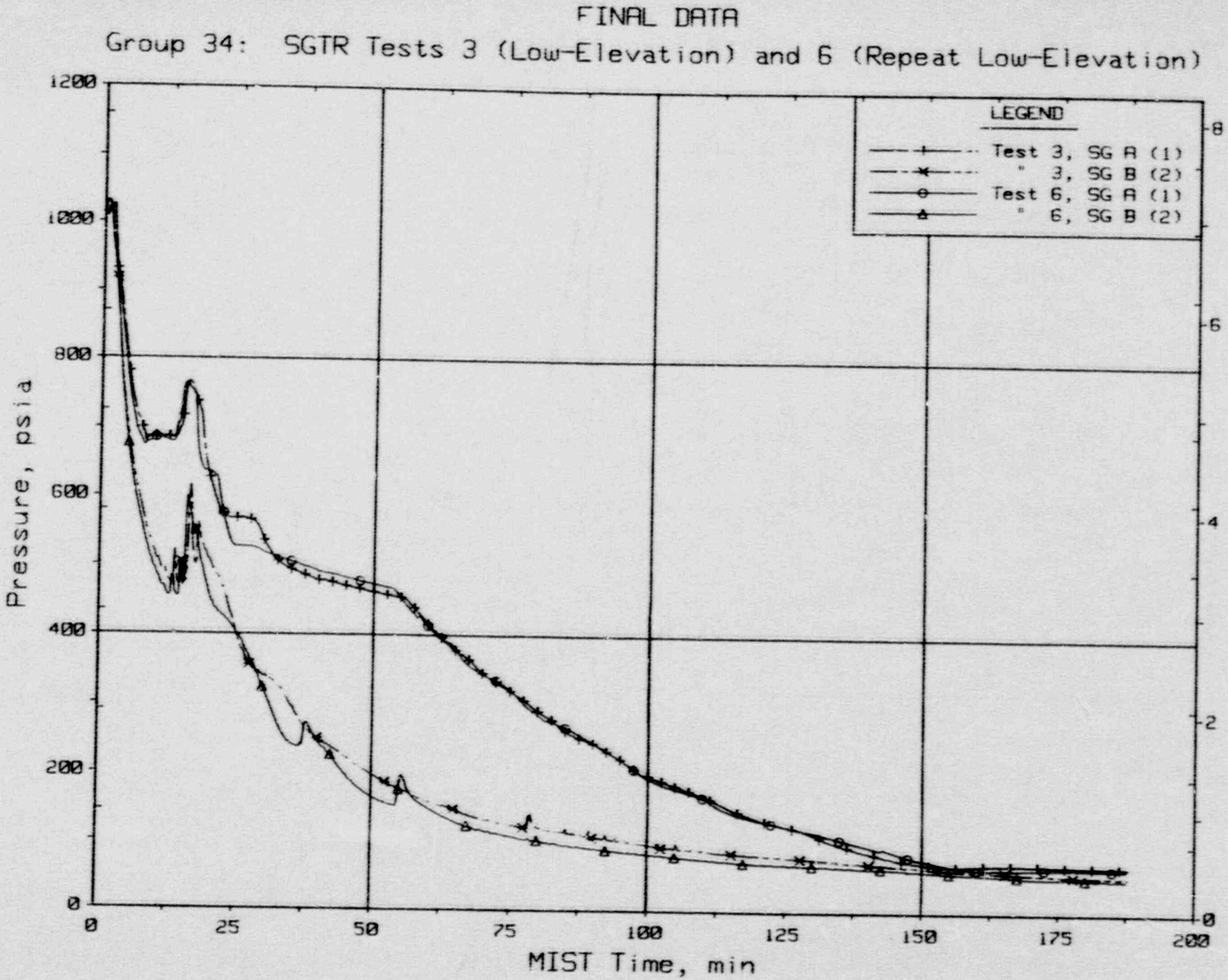

Figure 5.3.36. Steam Generator Secondary Pressures (SnGP01s)

Tue Nov 1 16:45:38 1988

36SCPC1

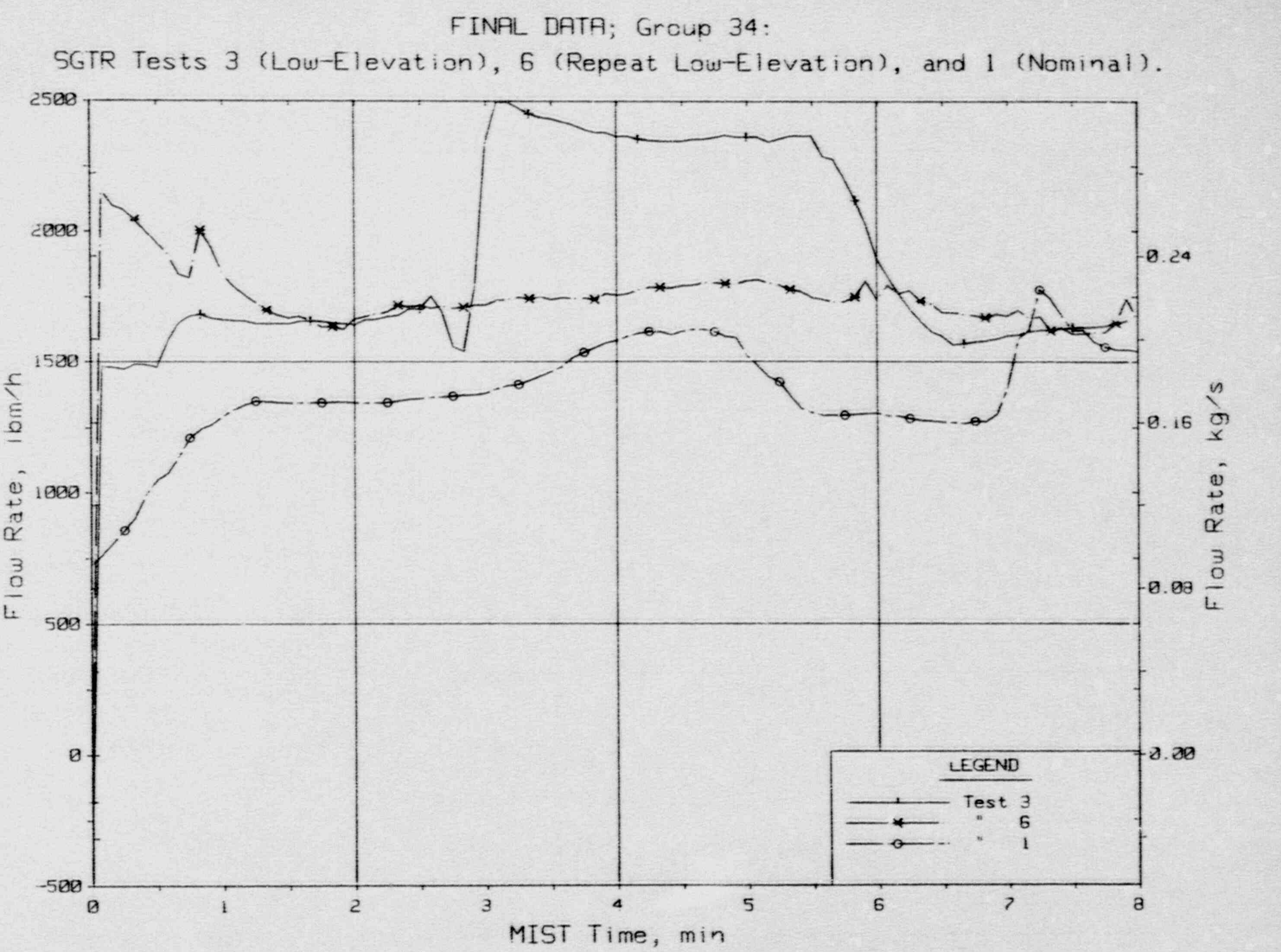

Figure 5.3.37. Tube Rupture Flow Rate

Wed Aug 10 10:57:28 1988

361PRFL3

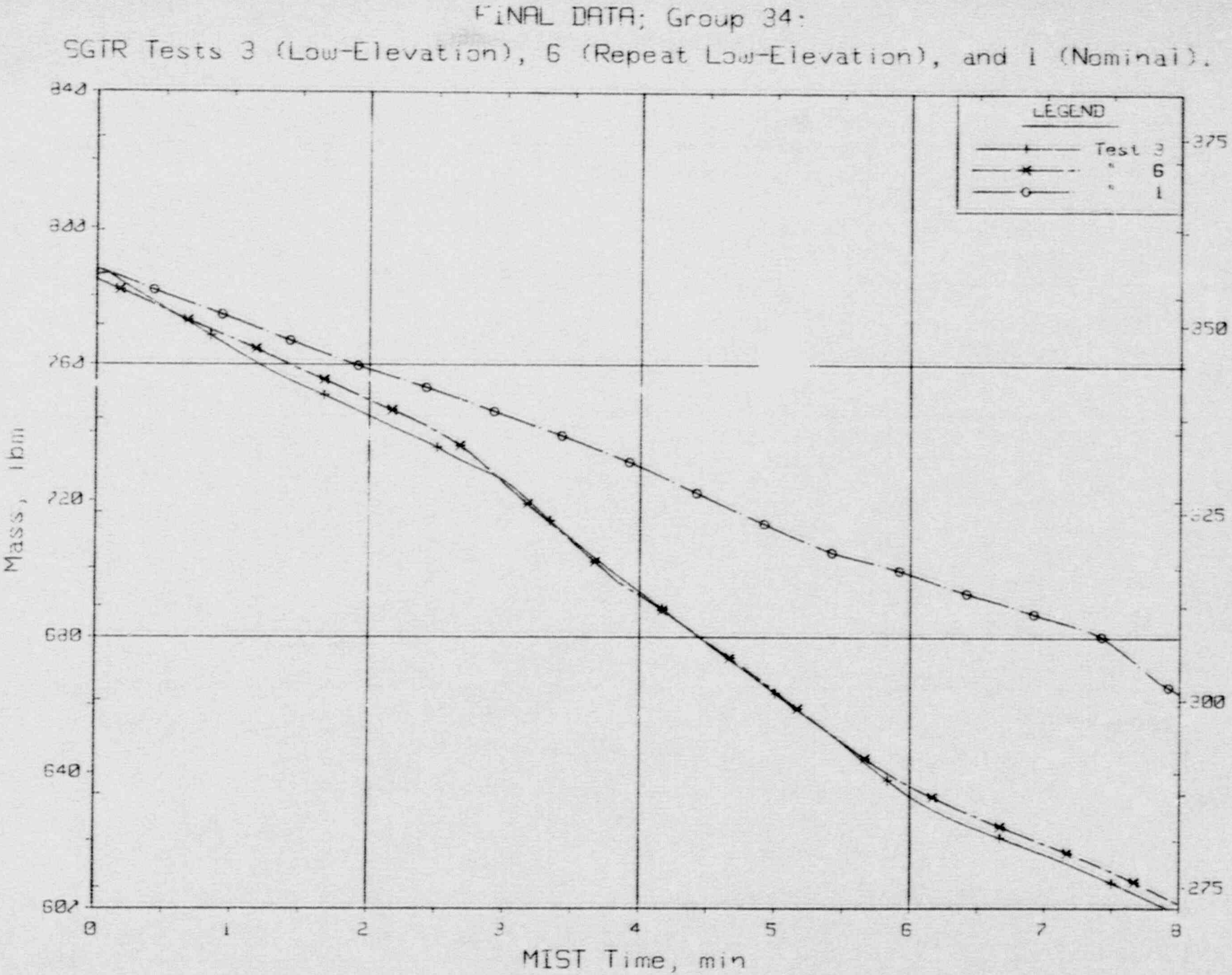

Figure 5.3.38. Primary System Total (Indicated, Trended) Fluid Mass

Tue Aug 9 15:08:42 1988

361PRML1

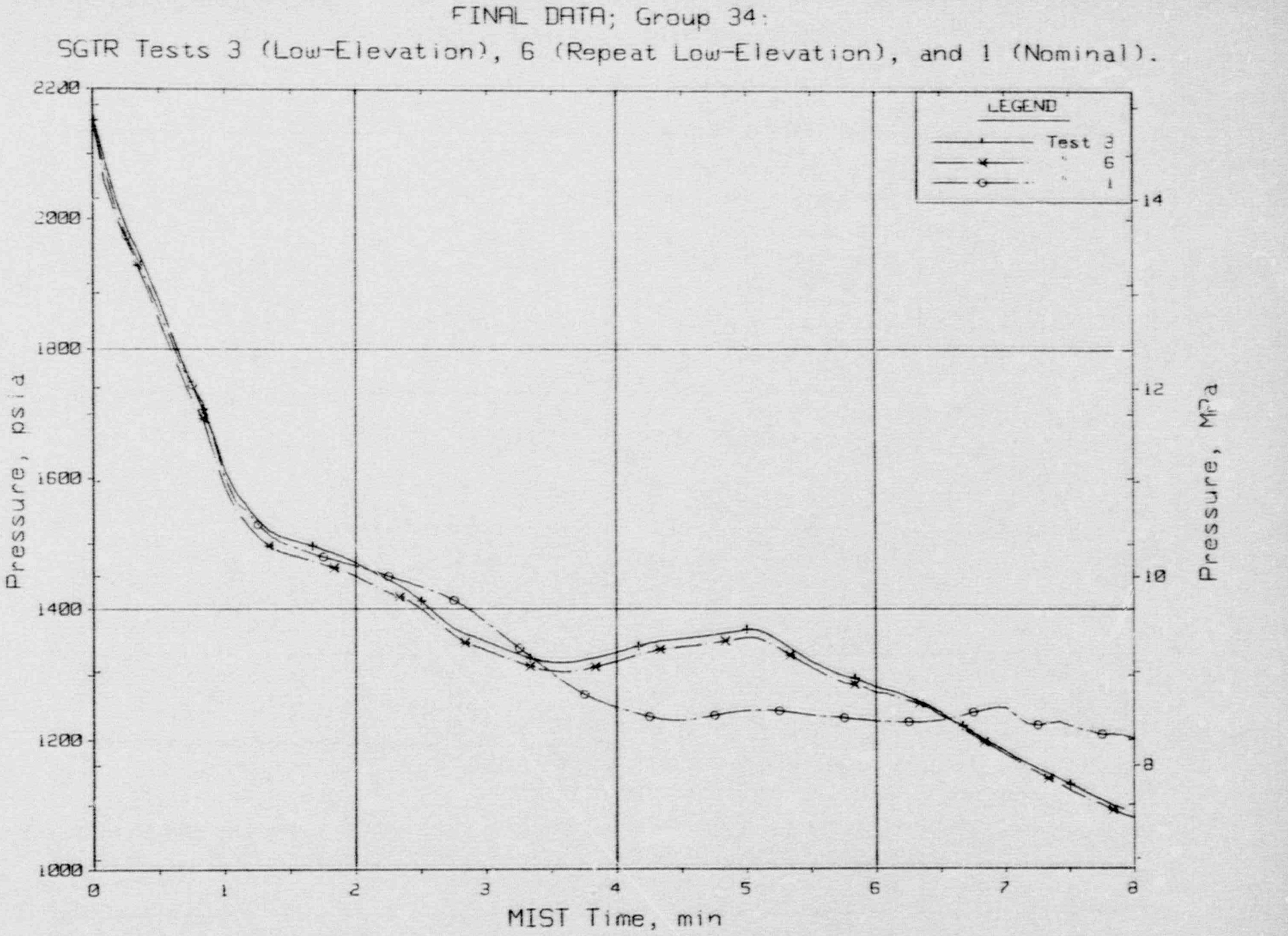

Figure 5.3.39. Primary System Pressure

Tue Aug 9 14:57:23 1988

361RVGP1

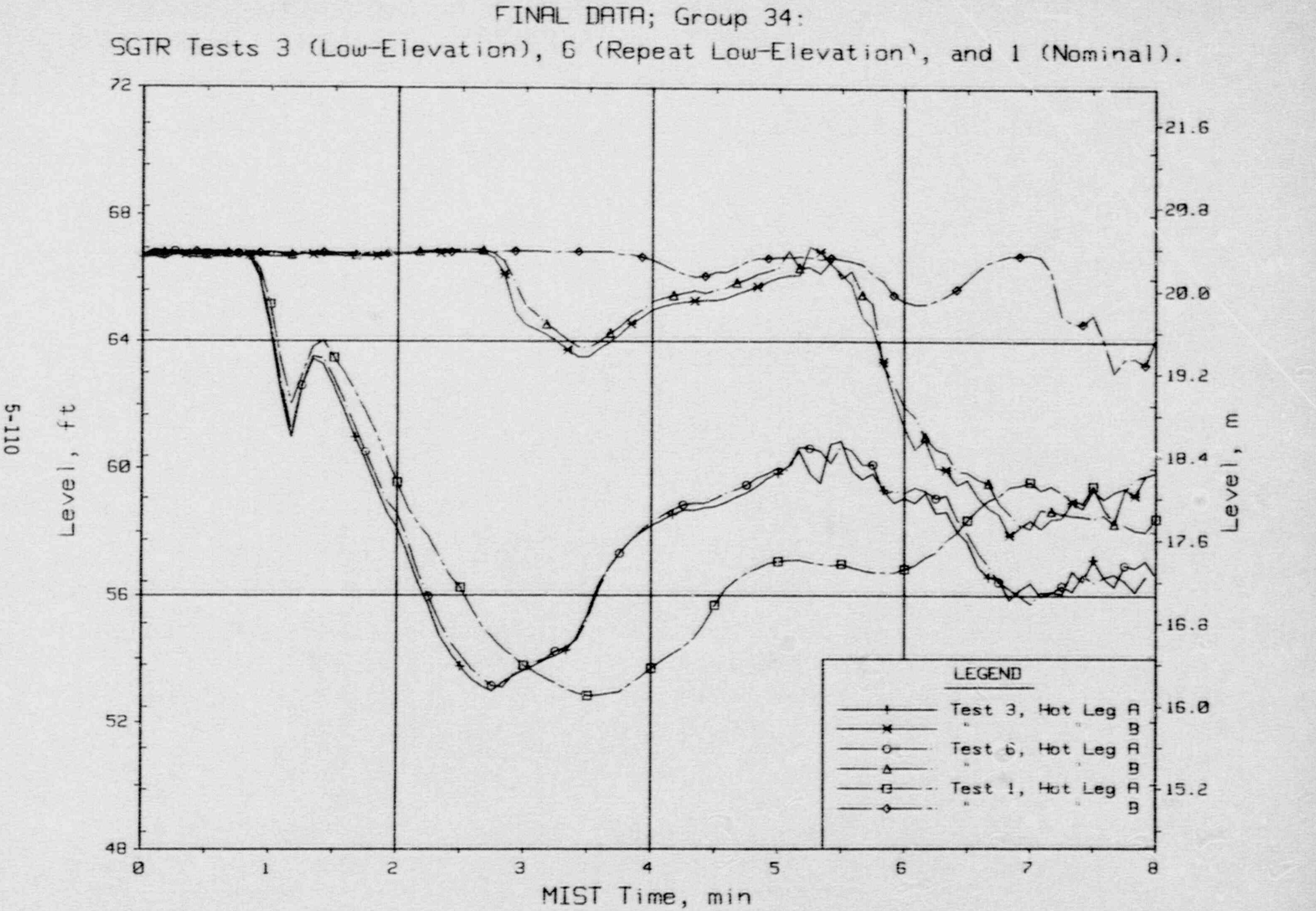

Figure 5.3.40. Hot Leg Riser Collapsed Liquid Levels

Tue Nov 1 16:30:56 1988

361HLLV1

FINAL DATA; Group 34:

SGTR Tests 3 (Low-Elevation), 6 (Repeat Low-Elevation), and 1 (Nom ial)

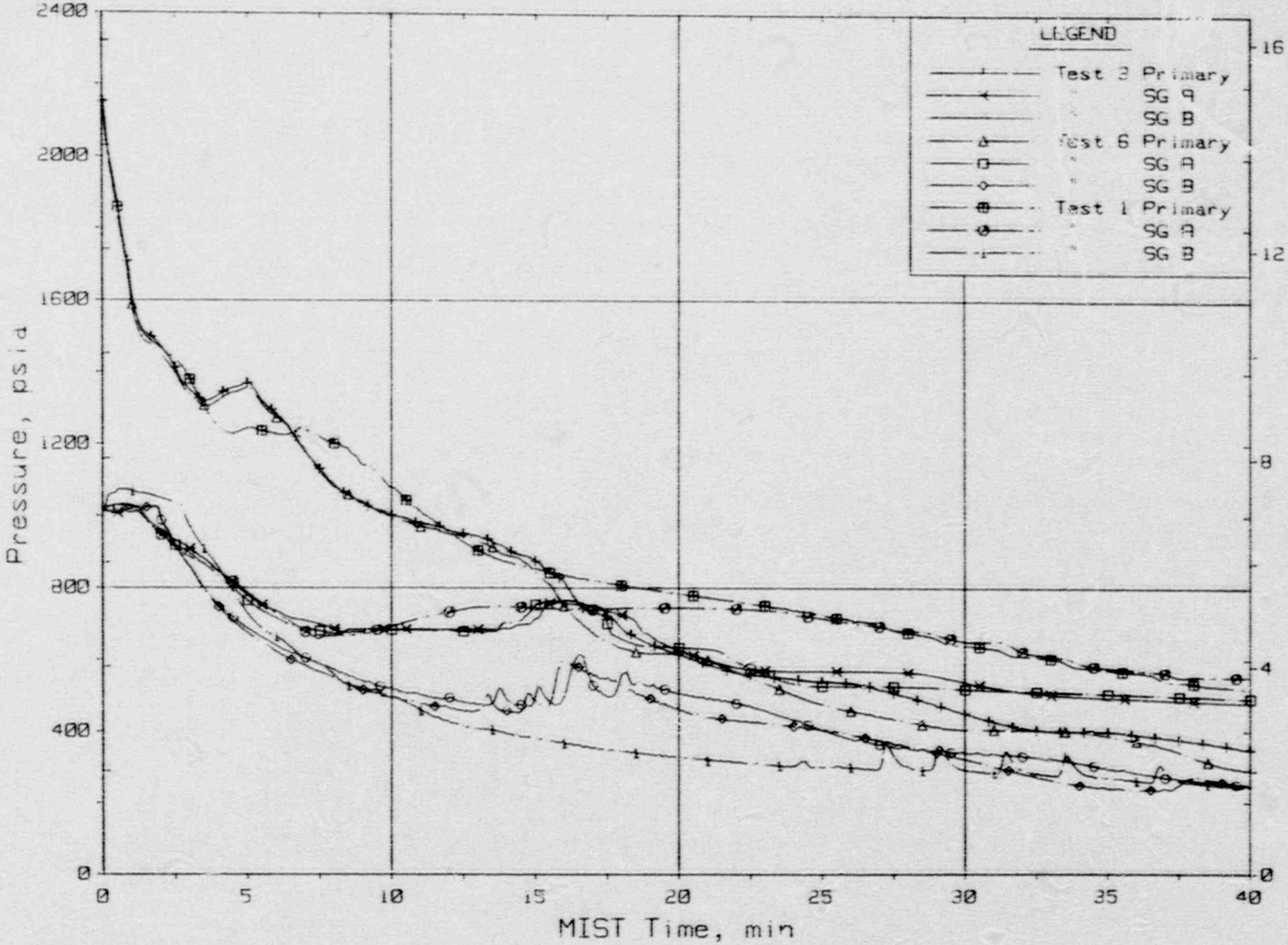

Figure 5.3.4i. Primary and Secondary System Pressures (GP01s)

Tue Aug 9 15:04:36 1988

361PSGP1

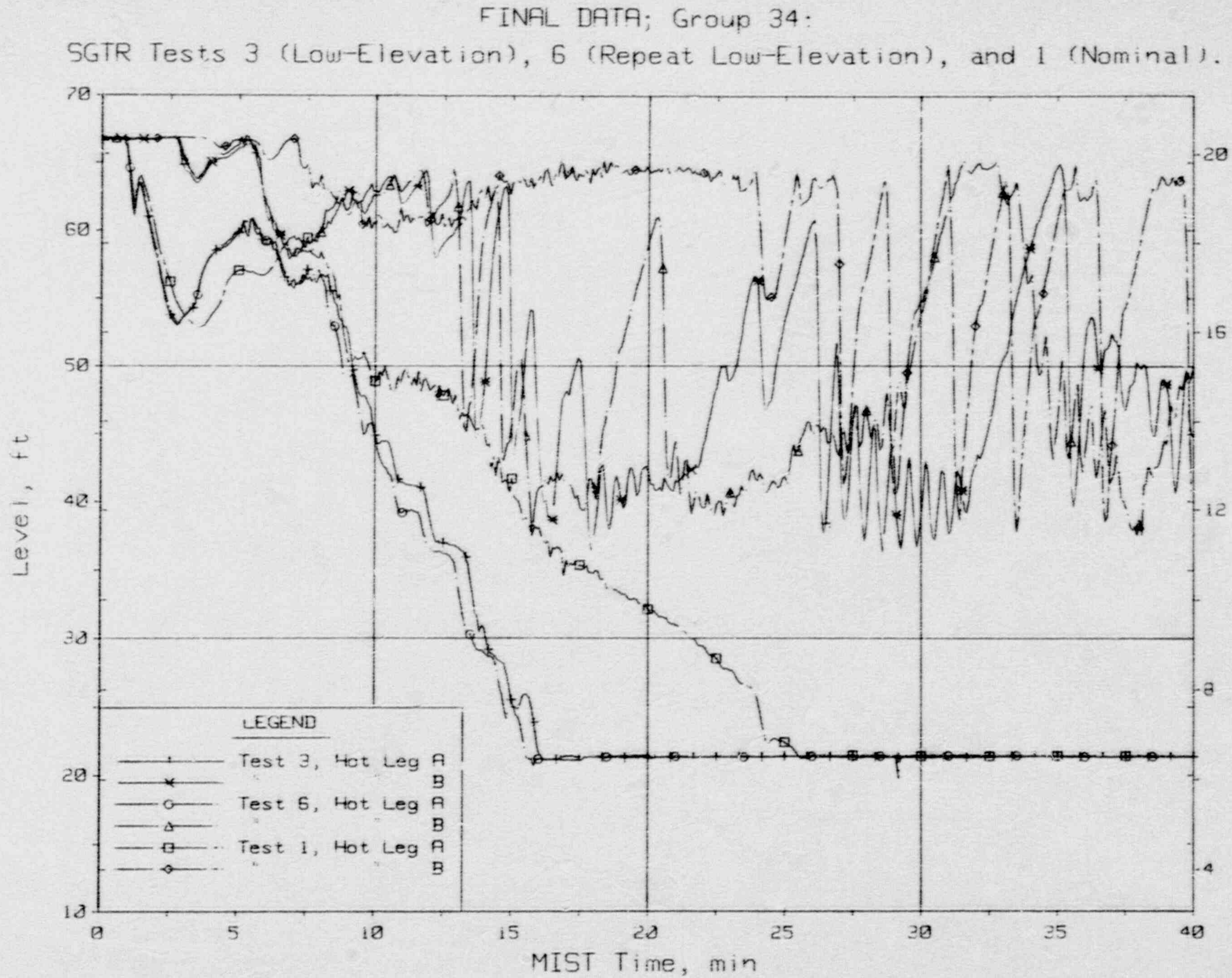

Figure 5.3.42. Hot Leg Riser Collapsed Liquid Levels

Tue Aug 9 15:09:25 1988

361HLLV1

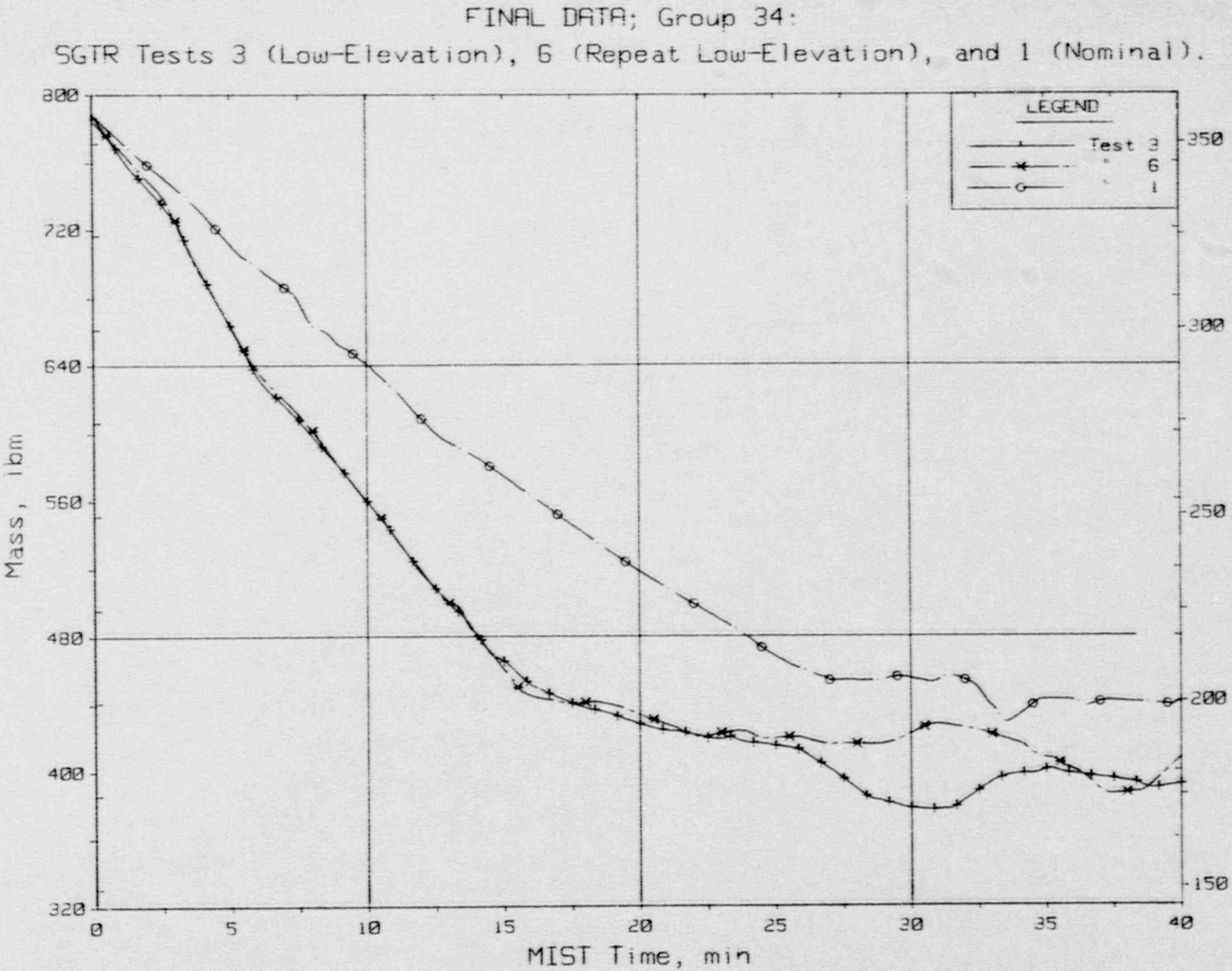

Figure 5.3.43. Primary System Total (Indicated, Trended) Fluid Mass

Tue Aug 9 15:12:20 1988

361PRML.1

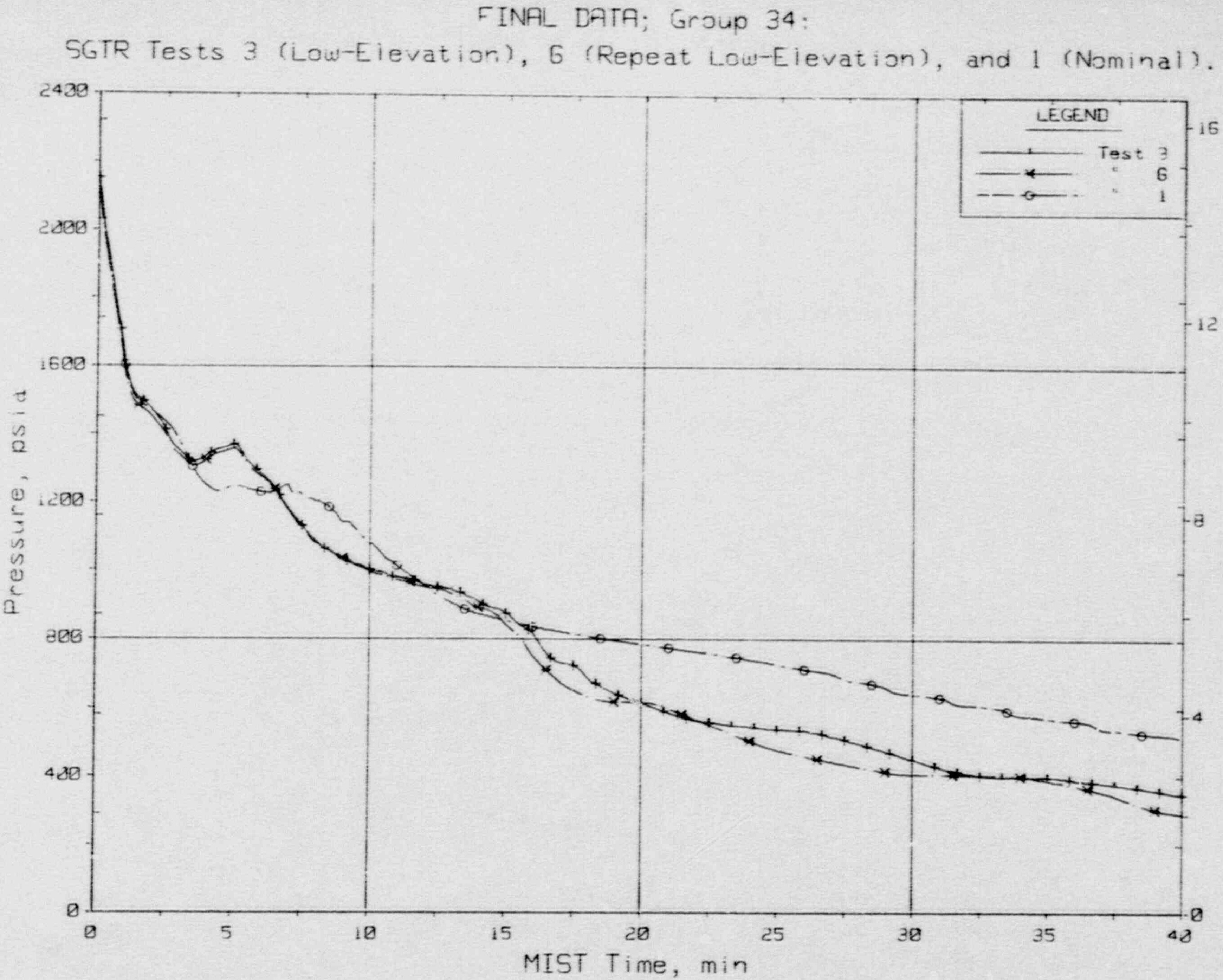

Figure 5.3.44. Primary System Pressure

Tue Aug 9 15:14:54 1988

36IRVGP1

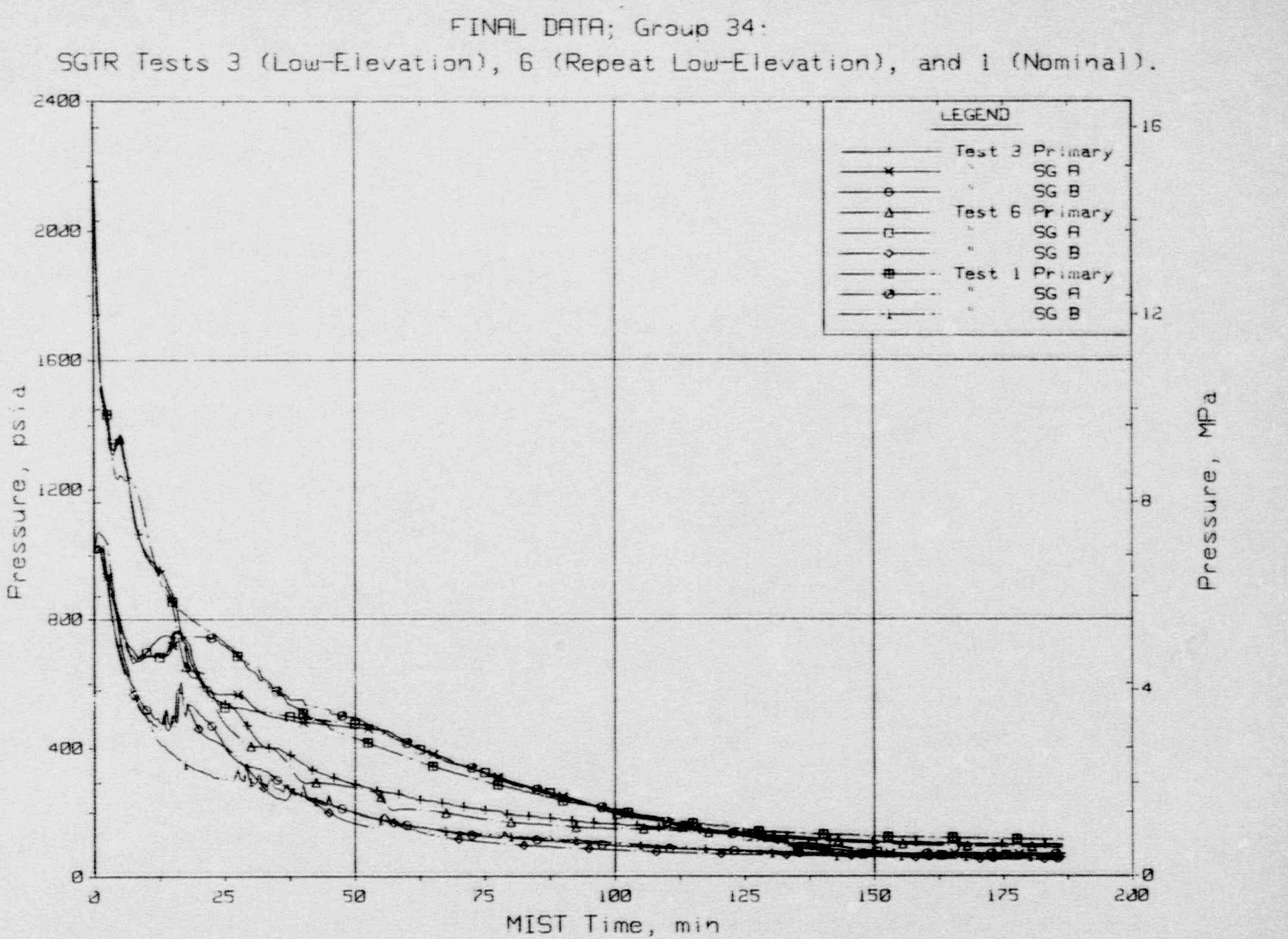

Figure 5.3.45. Primary and Secondary System Pressures (GPOls)

Tue Aug 9 15:22:18 1988

361PSGP1

FINAL DATA; Group 34:
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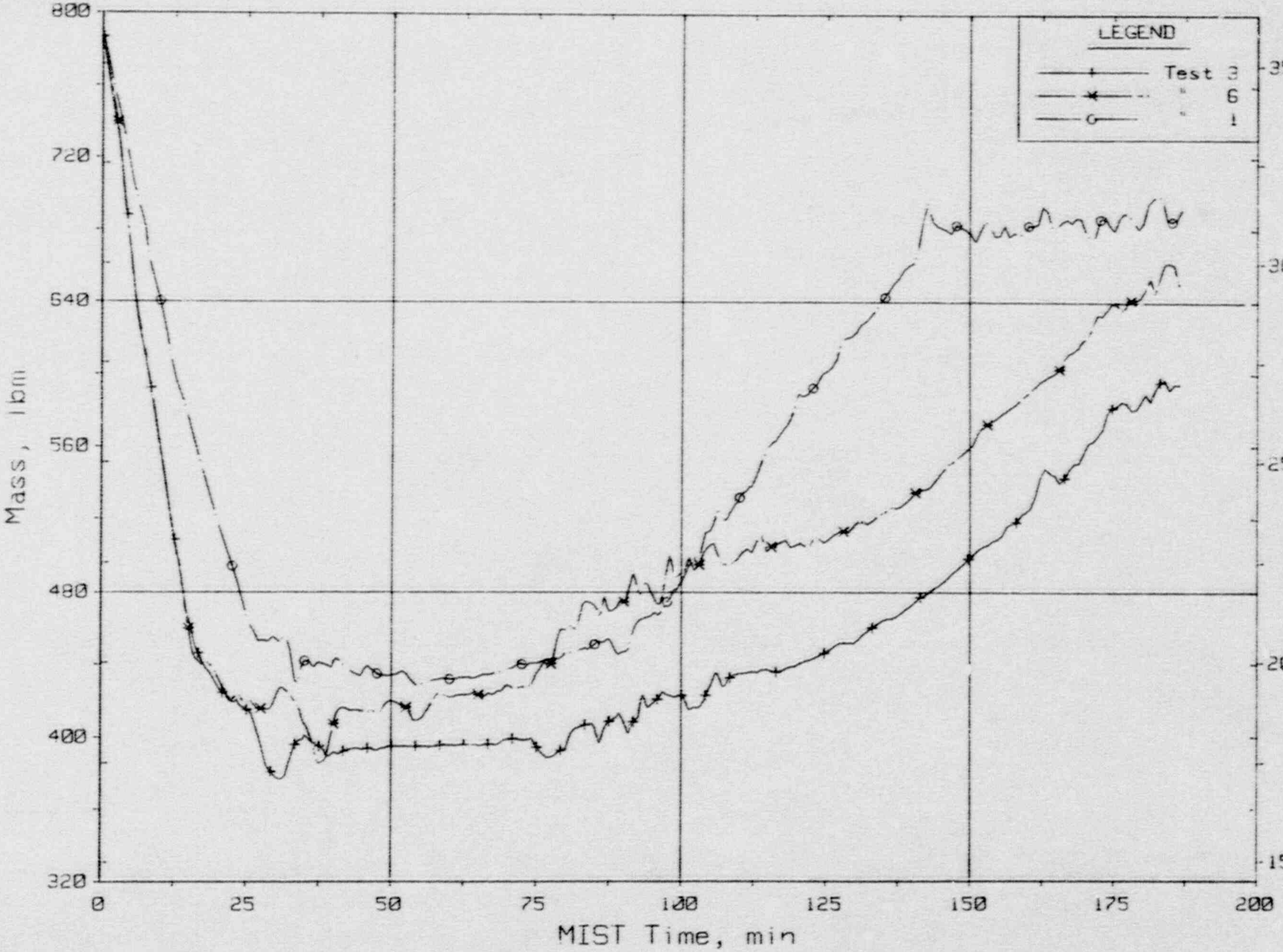


Figure 5.3.46. Primary System Total (Indicated, Trended) Fluid Mass

Tue Aug 9 15:25:04 1988

361PRML1

5.4. Steam Generator Isolation, Test 4

Test 4 of the SGTR group (3404AA) imposed isolation on the affected steam generator (B). Steam generator B was isolated by procedure with a primary system pressure of 950 psia and decreasing. Upon isolation, the affected steam generator secondary side filled completely and pressurized to the primary system pressure. Shortly afterward, the primary system (and isolated steam generator) repressurized and precipitated the termination of the test. The unaffected steam generator (A) was available for heat transfer, but remained largely inactive because of voiding in the loop A U-bend. The core exit subcooling margin increased during the repressurization but did not quite reach the specified subcooling margin, 75F, at which the operator was to actuate the PORV. Test 4 is described below, and is compared to the Nominal Test in section 5.4.2.

5.4.1. Observations

Immediately upon break actuation (at time zero), the primary system began to depressurize at 400 psi/min, from 2000 psia, and the steam generator B secondary level began to rise at approximately 6 ft/min. After approximately 1 minute, the rate of primary system depressurization slowed as the upper reactor vessel and hot leg A U-bend fluid saturated. The hot leg A levels then receded from the U-bend elevation. Also in the period from 1 to 2 minutes, the steam generator A feed rate increased (as its control level was reset from 5 to 31.6 ft) and HPI became active, both in response to test-initiating actions.

The steam generator B secondary level surpassed 20 feet at 2.3 minutes (Figure 5.4.1), triggering manual actuation of the high-flow rate steam flow control valves in an attempt to control level. This action was evidenced by a stabilization of the steam generator B level near 25 feet beyond 3 minutes, and by a gradually decreasing steam generator B secondary pressure, from 1000 psia at 3.4 minutes (Figure 5.4.2).

The hot leg B levels briefly receded from the U-bend at 3 minutes and remained below the U-bend after approximately 5 minutes (Figure 5.4.3). The intermittent steam generator A BCM heat transfer during its secondary-side refill suppressed primary system repressurization (near 1300 psia). The loop

B steam generator primary level descended to the break elevation and stabilized there beyond 7 minutes (Figure 5.4.3). The consequent reduction of break mass flow rate (Figure 5.4.4) caused a continuing reduction of the steam generator B secondary pressure, whereas the increased break volumetric flow rate (Figure 5.4.5) allowed the primary system to gradually depressurize. At 13.3 minutes, the primary system pressure had been reduced to 950 psia (Figure 5.4.2) and the operator isolated steam generator B (the affected steam generator) according to the test procedure.

Upon isolation of steam generator B at 13 minutes, its secondary pressure rose from 405 psia to 855 psia, about 15 psi less than the primary pressure, by 17.4 minutes (Figure 5.4.2). Its secondary level increased concurrently from 31 feet to full of liquid (Figure 5.4.1). Primary pressure continued to slowly decrease, apparently in response to heat transfer to (unaffected) steam generator A. The loop A primary level was well below the elevation of the steam generator upper tubesheet, and steam generator A feeding was recorded until 20 minutes. However, the primary system levels, including the loop A hot leg and steam generator primary levels, began to generally increase after the isolation of the affected steam generator. (The exceptions occurred in the cold leg suction and discharged piping, which voided about 5 feet from 14 to 19 minutes.) The primary system thus began to repressurize after approximately 25 minutes, but with saturated core exit conditions (Figure 5.4.5). The increasing reactor vessel collapsed liquid level exceeded the nozzle elevation at 29 minutes, and the downcomer level exceeded this elevation at 32.5 minutes. The core exit fluid began to evidence subcooling after the later event. Also at this time, the inter-cold leg circulation in loop B was extinguished, although the loop A circulation continued until 42 minutes. The primary system (and steam generator B secondary) pressure reattained 950 psia at 31 minutes, when the (unaffected) steam generator A primary level exceeded the elevation of the steam generator upper tubesheet (Figure 5.4.3), thus inhibiting heat transfer to steam generator A.

The reactor vessel downcomer was completely refilled at 39 minutes (Figure 5.4.6), leaving only the upper loop A hot leg, pressurizer, and upper reactor vessel volumes for subsequent compression. The primary system (and steam generator B secondary) then began to repressurize more rapidly, increasing from 1050 psia at 38.5 minutes to 1350 psia at 45 minutes.

The MIST high-capacity steam circuit had been specified to be opened at 1300 psia to simulate the plant secondary safeties and to control secondary pressure. However, the MIST high-steam flow rate circuit was not opened, therefore the secondary pressure in the isolated steam generator continued to rise toward the lift pressure of the test facility operational safeties. Thus, the test was aborted out of necessity. Had the plant safety simulation been actuated, the ensuing transient would likely have been altered markedly. Actuation notwithstanding, the important observation had already been made: The primary system (and isolated steam generator) repressurized although the unaffected steam generator was available for heat transfer.

The system conditions were apparently approaching a threshold near the time at which the test was aborted. The loop A riser level, which had generally been increasing since 23 minutes, came within 1 foot of the U-bend spillover elevation at 45 minutes (Figure 5.4.3). Feed and steam flow had been (nominally) active to steam generator A since 37 minutes, when the cooldown of 100F/h (in steam generator A only) had become active. Thus, a loop A spillover and its attendant cooldown and depressurization appear to have been imminent when the test was aborted.

The MIST operator was to open the PORV at 75F (core exit) subcooling and reduce the subcooling to 50F. This procedure was intended to track a natural circulation cooldown while minimizing primary pressure (and thus the likelihood of repressurizing the secondary). The core exit subcooling at test abortion was 73F (Figure 5.4.5), that is, just 2 degrees below the PORV actuation value. Based on this observation and the preceding description of the imminent spillover in loop A, the repressurization that gave rise to test abortion appears to have been unlikely. However, the subcooling response, the intermediate loop A level, and the primary system repressurization were each linked to the isolation of the affected steam generator. Their responses thus appear to be linked and therefore to be expected, rather than to be unlikely.

5.4.2. Comparison

Test 4 repeated the initial and boundary conditions of Nominal Test 1 until steam generator B was isolated at 13.3 minutes. The transients progressed similarly until the time of steam generator isolation (Figures 5.4.7 through 5.4.12) but were not identical. The inter-test differences were due primarily to the 2-minute delay in activating HPI and AFW to steam generator A, in Test 4. The steam generator A secondary level increase of Test 4 lagged behind that of Test 1 (Figure 5.4.7), as did the depressurization of steam generator A (Figure 5.4.8). The primary system total fluid mass decreased more rapidly in Test 4 than in Test 1 (Figure 5.4.9), and hot leg and core-region voiding occurred earlier in Test 4 (Figures 5.4.10 and 5.4.11). Notwithstanding these inter-test differences, similar inter-test conditions existed at the time of steam generator isolation in Test 4.

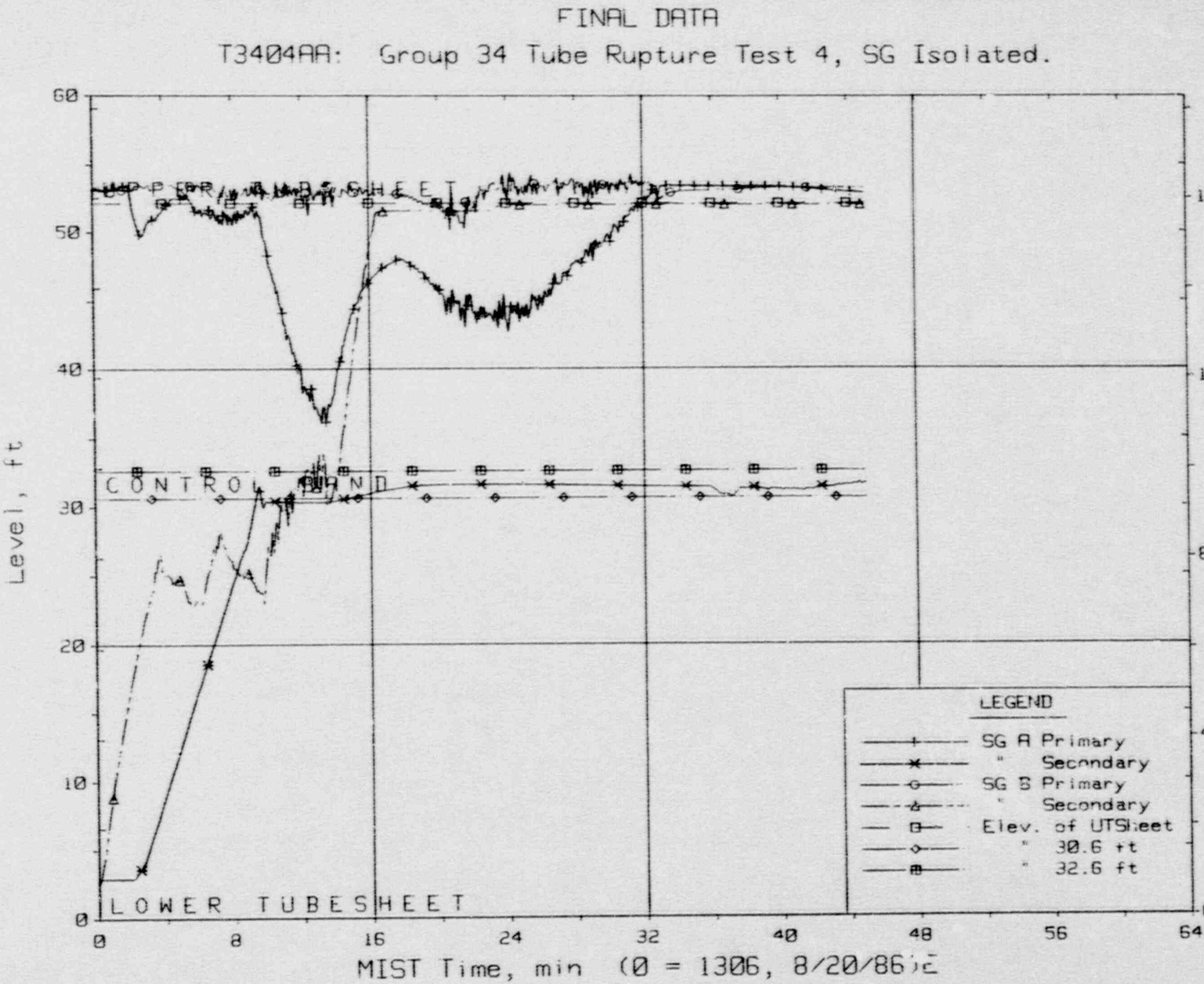
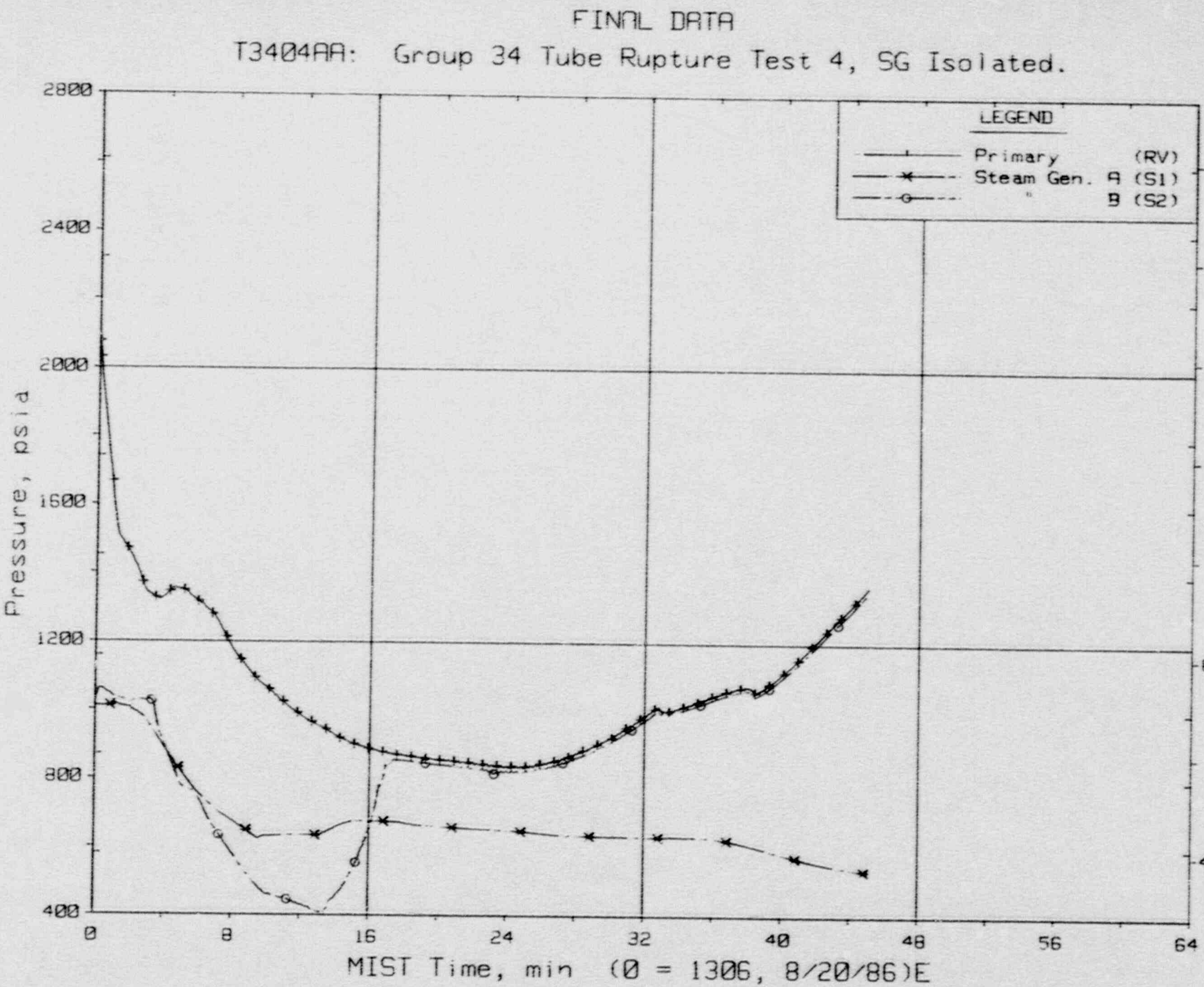


Figure 5.4.1. Steam Generator Collapsed Liquid Levels

SGLV2

Mon Jul 25 14:58:54 1988



5-122

Figure 5.4.2. Primary and Secondary System Pressures (GPOls)

Mon Jul 25 15:01:33 1988

PSGP1

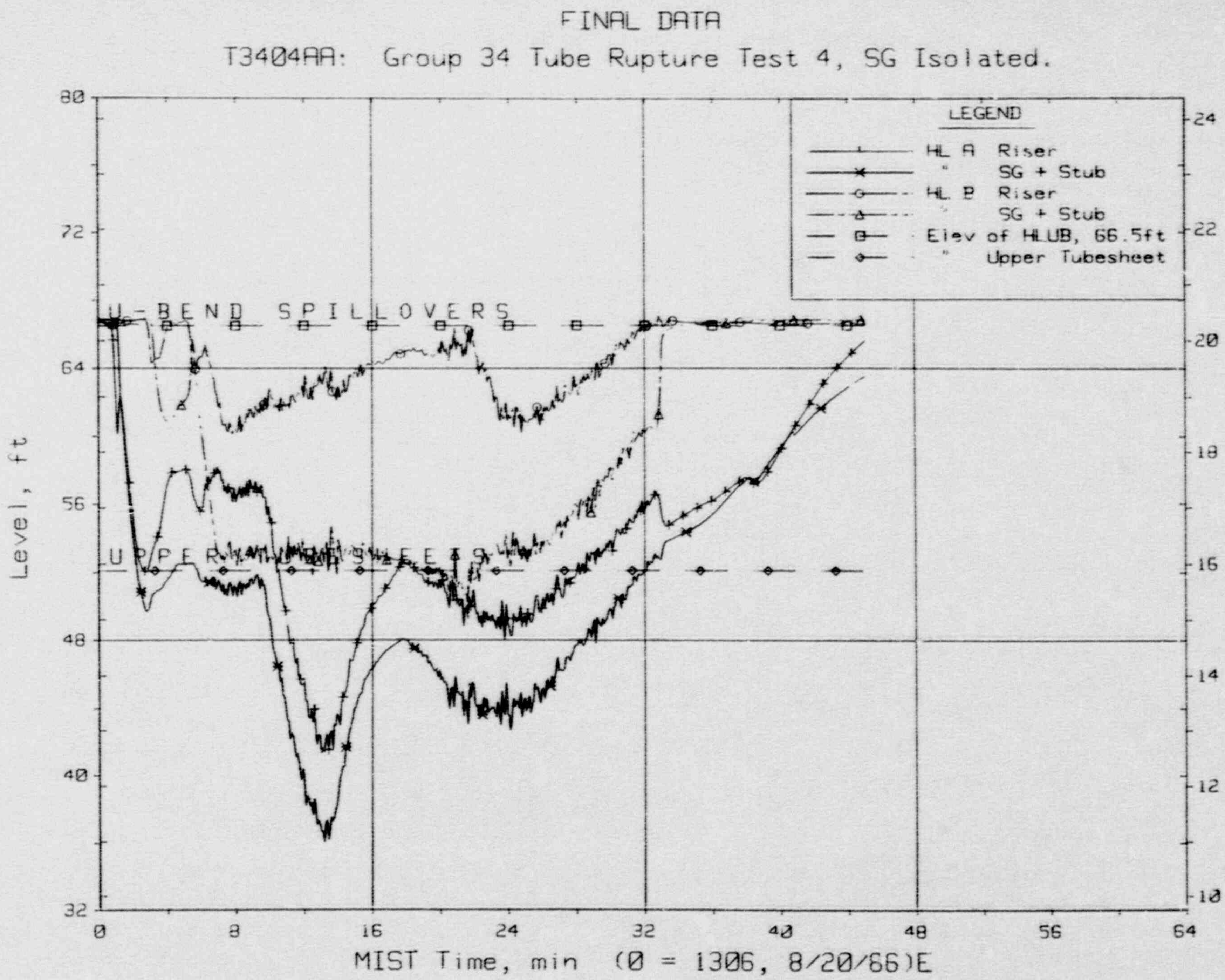


Figure 5.4.3. Hot Leg Riser and Stub Collapsed Liquid Levels

Mon Jul 25 15:05:01 1988

HLLV1

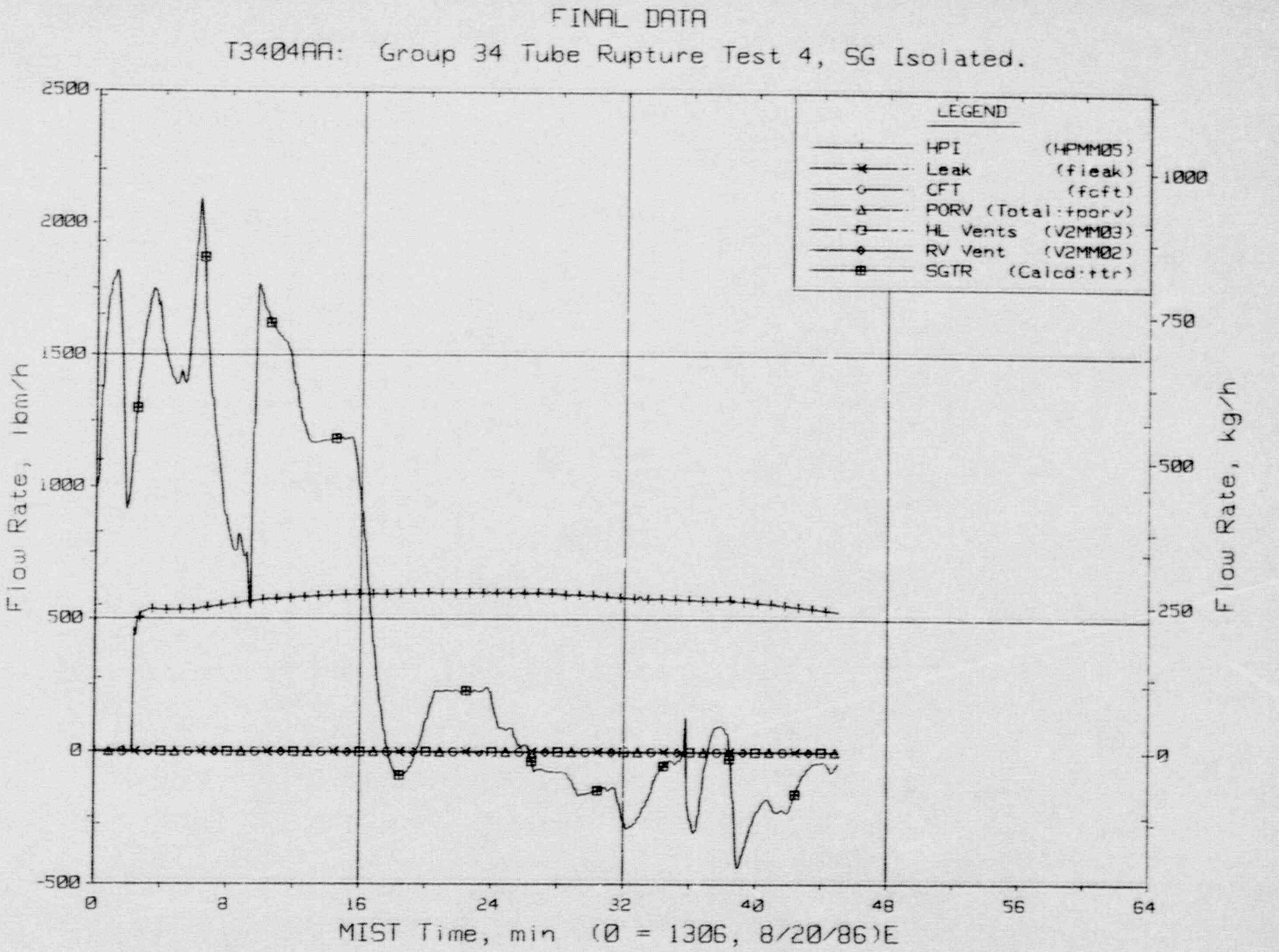


Figure 5.4.4. Primary System Boundary Flow Rates

Mon Jul 25 15:09:48 1988

PRFL3

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.

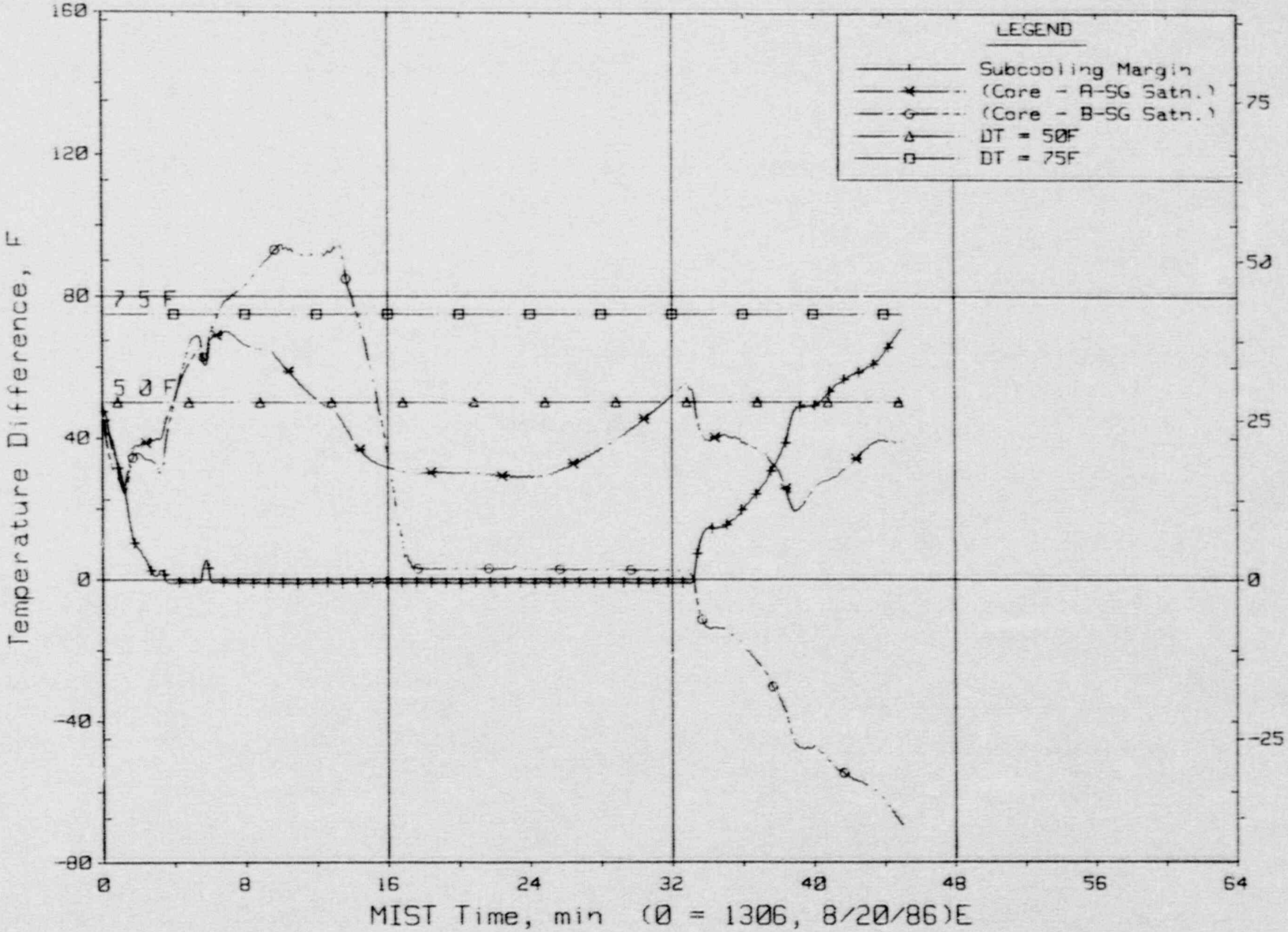


Figure 5.4.5. Control Temperature Differences

Mon Jul 25 15:12:59 1988

PRTD1

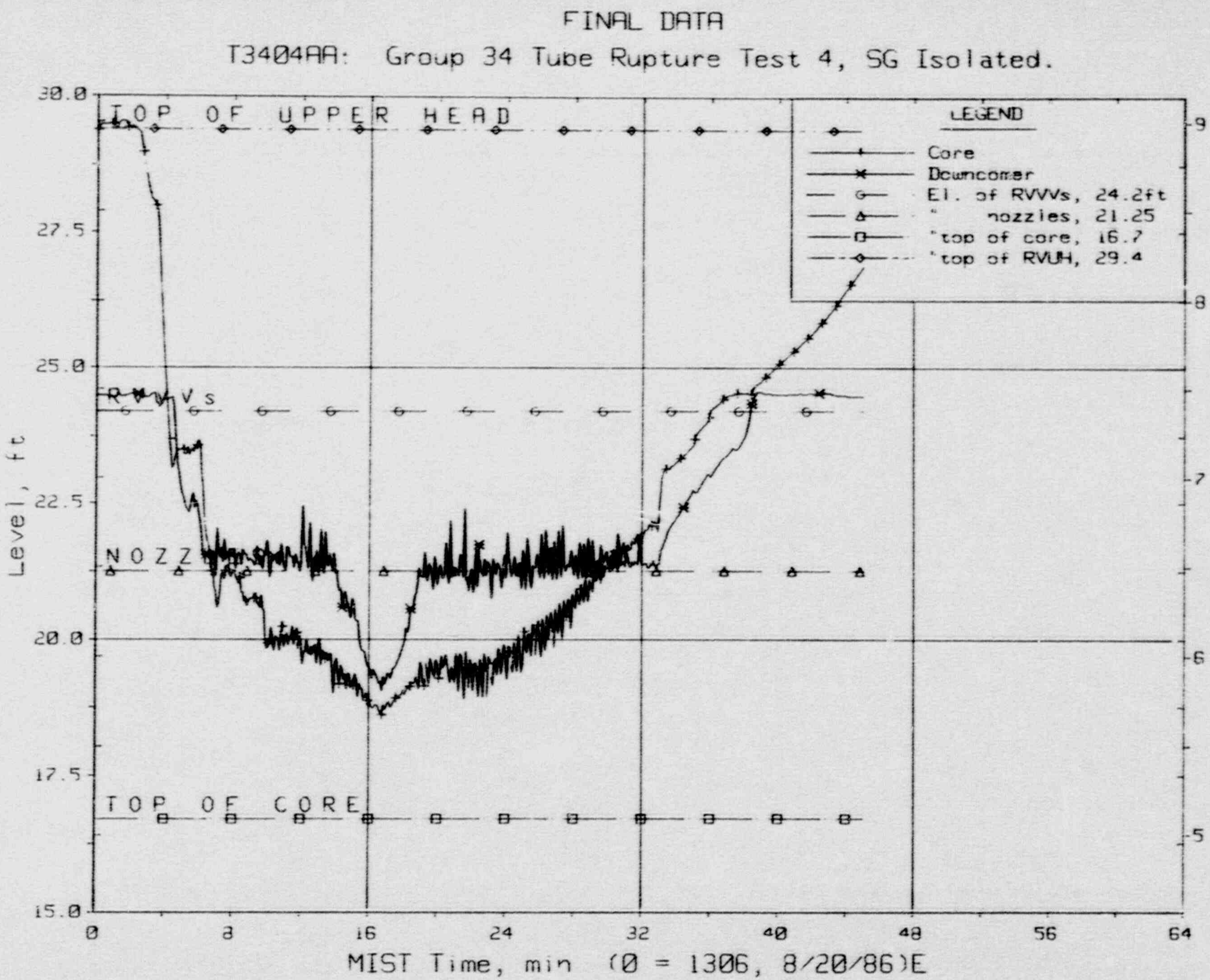


Figure 5.4.6. Core Region Collapsed Liquid Levels

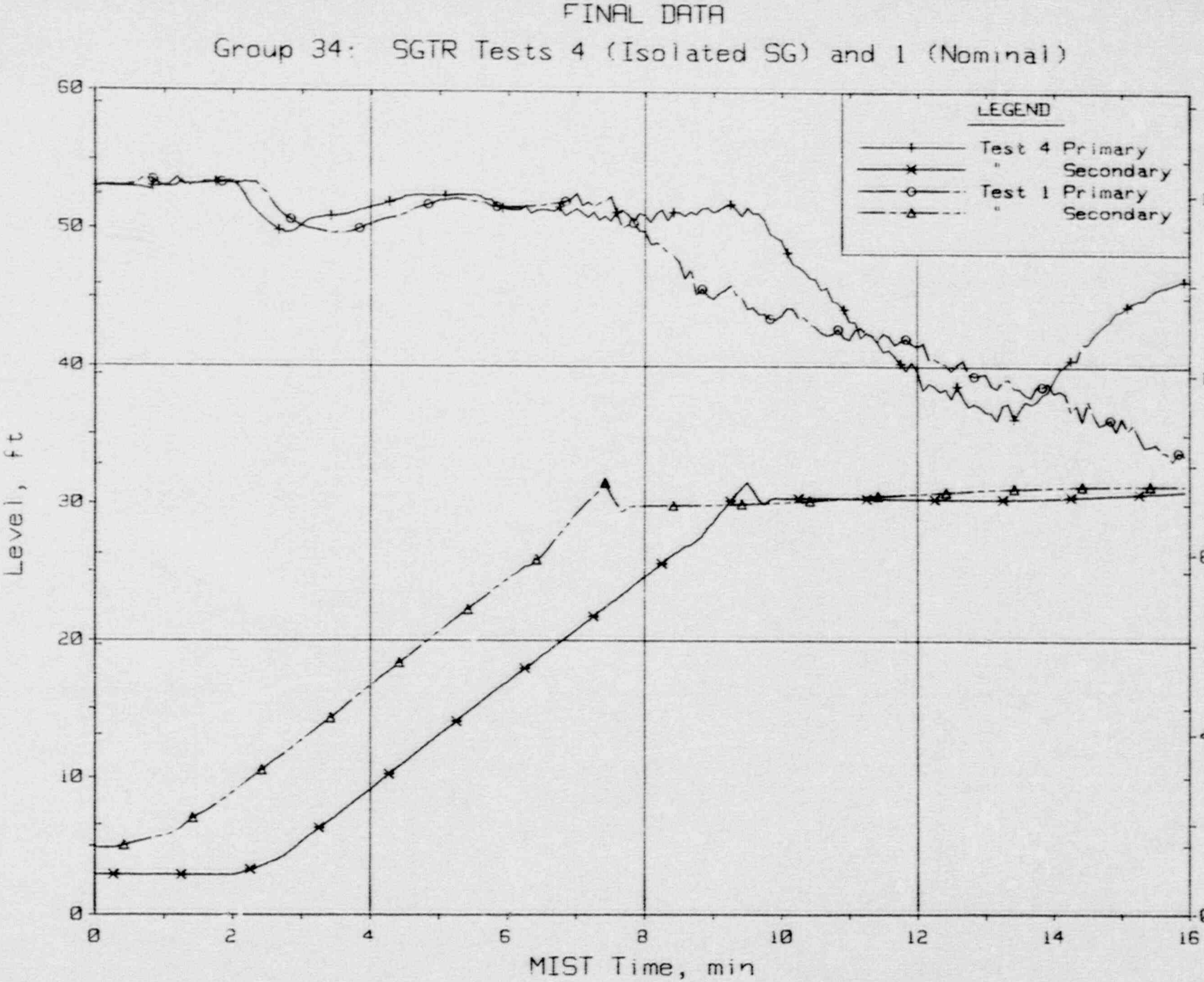


Figure 5.4.7. Steam Generator A Collapsed Liquid Levels

Mon Jul 25 15:24:20 1988

41SGLV1

5-128

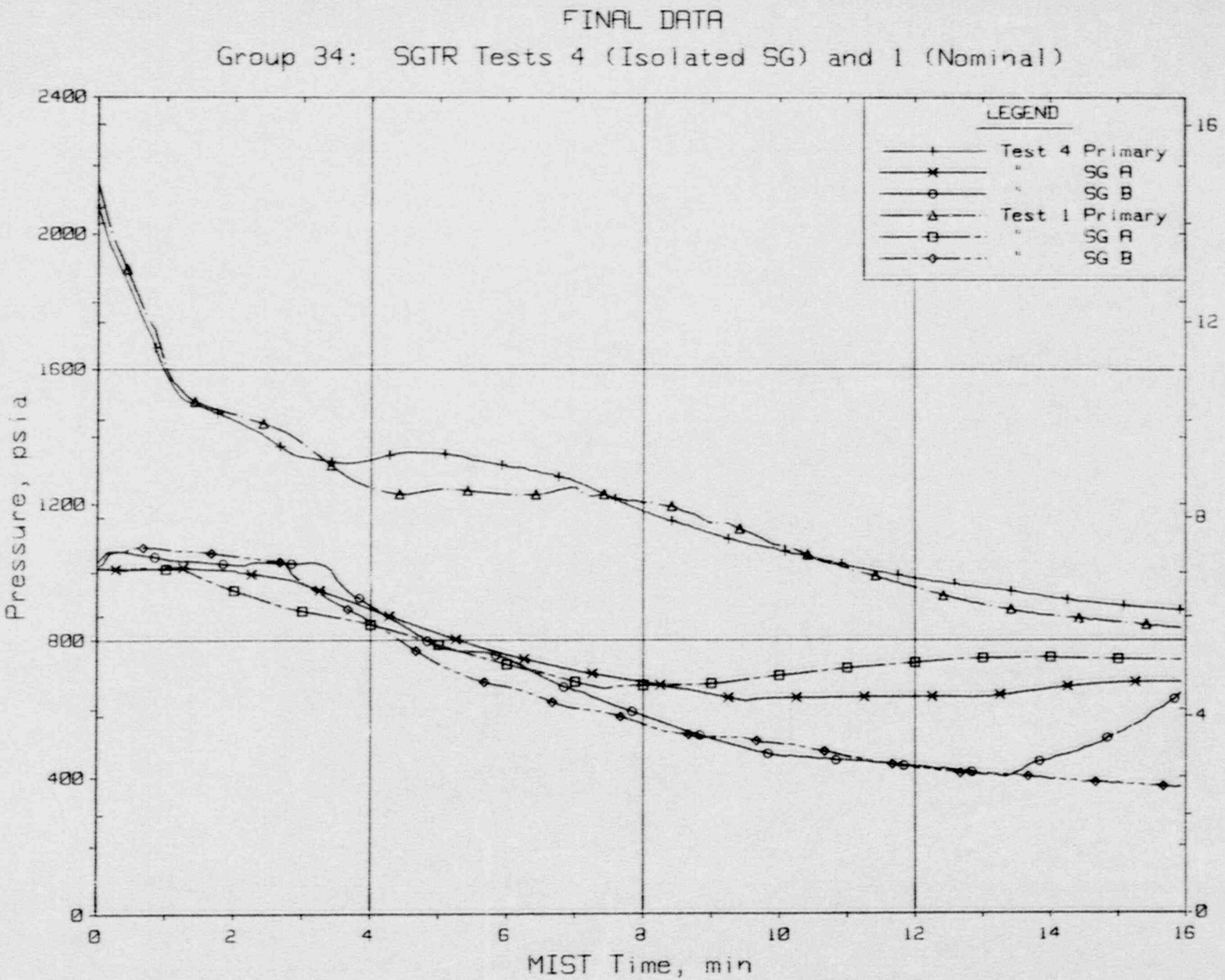


Figure 5.4.8. Primary and Secondary System Pressures (GP01s)

Mon Jul 25 15:26:53 1988

41PSGP1

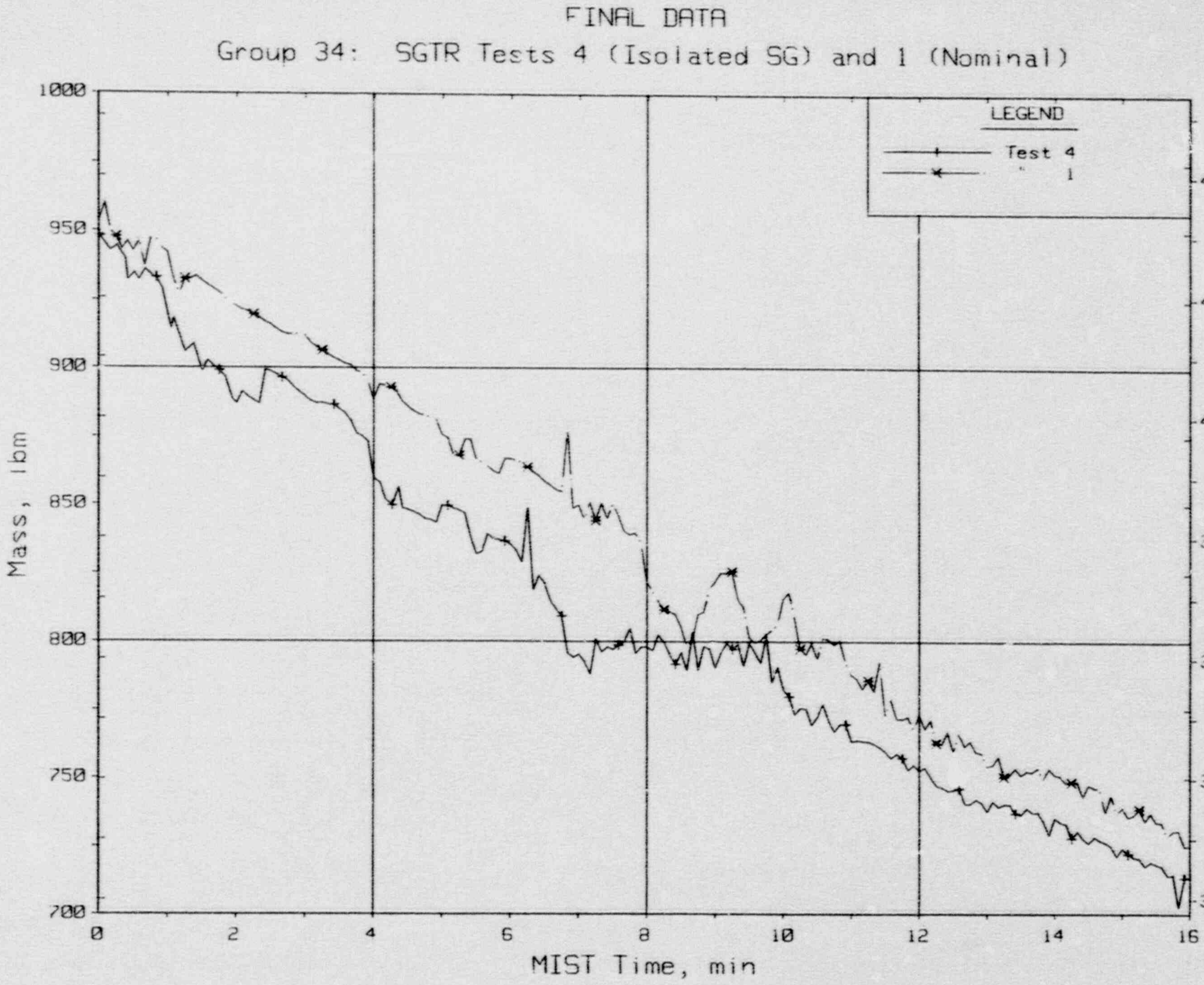


Figure 5.4.9. Primary System Total (Indicated) Fluid Mass

Tue Nov 1 16:36:14 1988

41PRML1

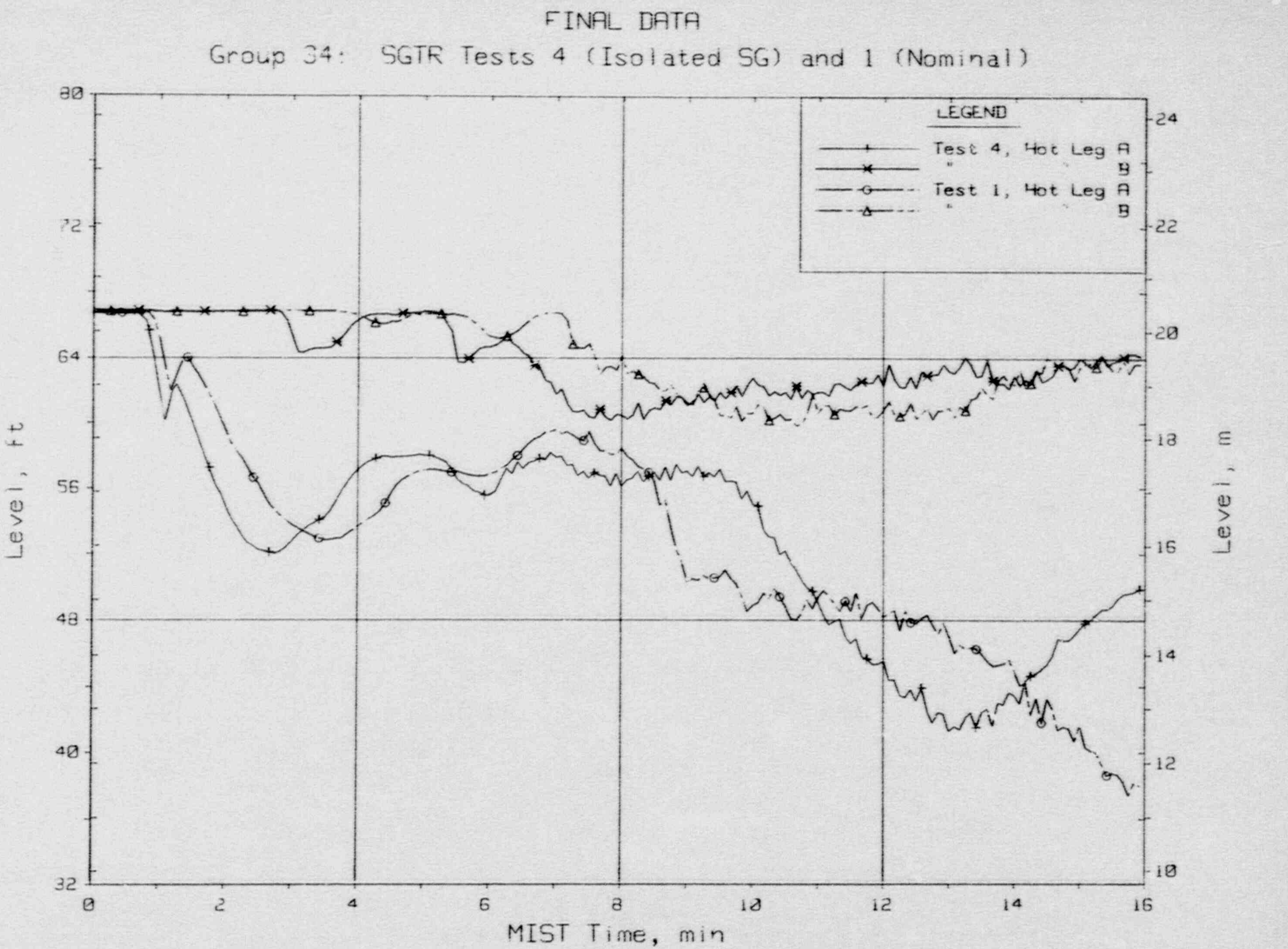


Figure 5.4.10. Hot Leg Riser Collapsed Liquid Levels

Mon Jul 25 15:31:00 1988

41HLLV1

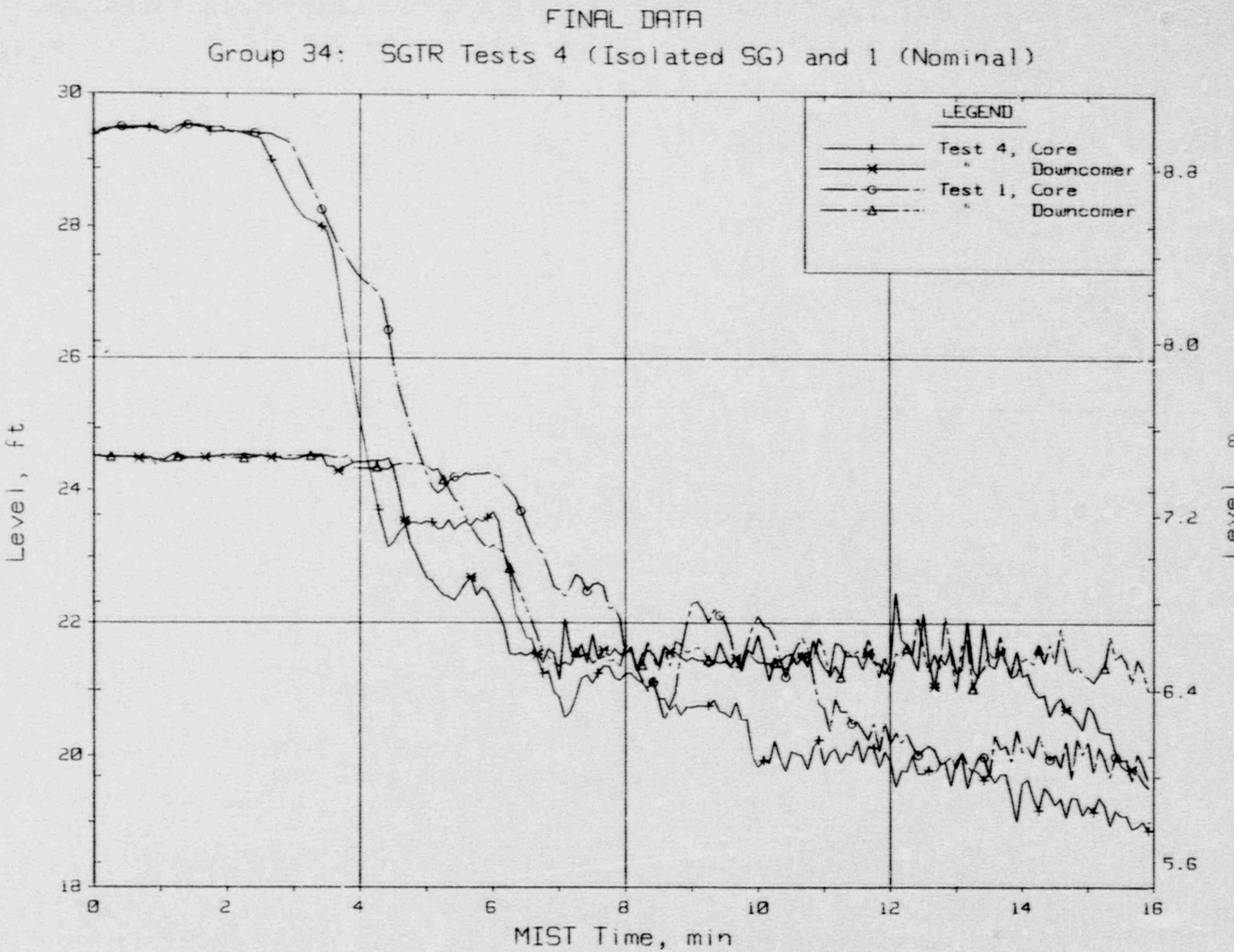


Figure 5.4.11. Core Region Collapsed Liquid Levels

Mon Jul 25 15:39:21 1988

41COLV1

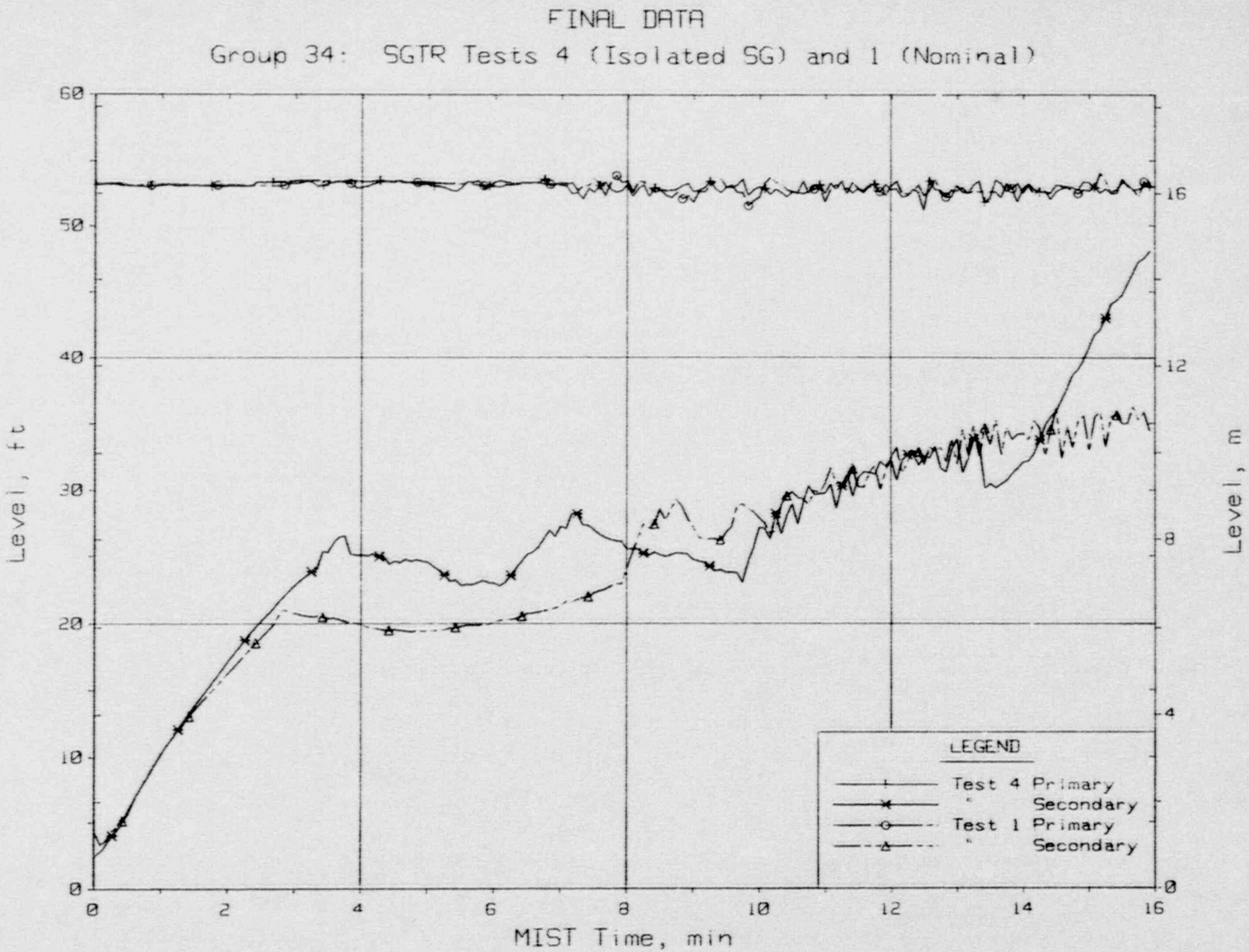


Figure 5.4.12. Steam Generator B Collapsed Liquid Levels

Mon Jul 25 16:20:06 1988

41SGLV2

5.5. One Tube Rupture

The high-elevation, single-ended rupture of 1 steam generator tube was simulated in Tests 2 and 7. The PORV was used to depressurize the primary system in Test 2 but pressurizer venting was simulated in Test 7, as described in sections 5.5.1 and 5.5.2. After primary system depressurization, the affected steam generator was isolated in both tests. Alternate methods of cooldown and depressurization were implemented late in Test 7. The rupture simulation became partially blocked, early in Test 2. Tests 2 and 7 are compared in section 5.5.3.

5.5.1. Rupture of One Tube, Test 2

The (single-ended) rupture of a single tube near the top of steam generator B was simulated in Test 2 (340213). Also, the affected steam generator was isolated as the primary system was depressurized below 950 psia. The initial phase of the transient, with both steam generators active, was largely uneventful. The rupture simulation flow area apparently decreased significantly, shortly after initiation. The tube rupture was evidenced only by slight inter-loop inequalities in the initial steam generator refill rate and in the feed rates. The operator manually operated the PORV and pressurizer heaters, as specified, to control the core exit subcooling margin. The operator also manually throttled HPI to maintain pressurizer level. Upon saturation and voiding in the reactor vessel upper head (as the primary system saturation temperature was reduced), the pressurizer level indicated corresponding increases.

Upon isolation of the affected steam generator at 80 minutes, the loop B hot leg flow stagnated and inter-cold leg flow intermittently occurred. The inactive-loop hot leg fluid temperatures remained nearly constant until the hot leg began to alternately void and refill with the continuing PORV actuations. The displacement of nearly saturated upper-elevation hot leg fluid to the lower elevations, and the subsequent displacement of subcooled core-region fluid into hot leg B upon hot leg refill, effectively cooled and ultimately subcooled the inactive loop. The hot leg refill characteristics were shaped by the repeated use of the pressurizer heaters, and perhaps by the uncompensated heat losses in the MIST U-bend region. Several anomalies regarding ATOG steam generator pressure control (of steam generator A) occurred during the test.

Initiation, Two-Loop Cooldown, SCM Control

The test-initiating actions were performed in a single group, in Test 2, and were completed within 20 seconds. These actions included

- Actuation of the primary-to-secondary flow path simulating the single-ended rupture of 1 tube at the top of the B steam generator.
- Activation of HPI with full head-flow characteristics.
- Transfer of RVVV control to automatic/independent.
- Initiation of the core power ramp simulating post-trip decay.
- Reset of the steam generator secondary level to 20.7 ft (representing 50% on the Operate Range).
- Activation of the ATOG steam generator secondary pressure control.

The opening of the primary-to-secondary leak had a relatively small impact on system conditions compared to the effects of the other test-initiating actions.

The primary system pressure decreased slightly from 2150 to 2080 psia at 1 minute. Also, the steam generator B secondary refilled slightly faster than A. The steam generator B secondary level reached 25 ft (more than 4 feet above the intended control level) at 5 minutes whereas the steam generator A level reached 21.5 ft at 5.2 minutes. The pressurizer level varied only a few feet during test initiation. Within 1 minute after test initiation, the operator began manually throttling the HPI flow rate to control pressurizer level.

The steam generator secondaries depressurized at almost equal rates and slightly faster than the rate required to obtain a cooldown of 100F/h during secondary refill. The enhanced steam generator heat transfer and the augmented AFW flow rates increased primary flow. Thus, both the steam generator primary outlet temperature and the temperature rise through the core decreased, causing a corresponding increase in the core exit subcooling margin.

The RVVVs opened upon test initiation, when their control was transferred to automatic/independent. The transient natural circulation driving force imposed by the increased steam generator heat transfer, as well as the valve openings, caused the differential pressure across the RVVVs to drop abruptly. The differential pressure reached the closing setpoint, 0.04 psi, at approximately 1 minute and the valves closed. The temperature differences across the RVVVs had approached zero with the valves open. As the valves closed, the temperature differences immediately began to increase toward the difference between the hot leg and cold leg temperatures.

Between 5 and 6 minutes after test initiation, the steam generator secondary levels achieved their control values and the core exit SCM exceeded 75F. The former event terminated AFW to both steam generators; both steam generators were steamed only a small amount because their secondary pressures were slightly below the control value for a cooldown of 100F/h. The resulting reduction of steam generator heat transfer caused the loop flow rates to decrease abruptly, from 5% to about 2% of scaled full flow, and caused the differential pressure across the RVVVs to abruptly increase. The actuation setpoint, 0.125 psi, was achieved at 6 minutes and all 4 vent valves opened.

The operator was to maintain the SCM between 50 and 75F using the PORV. Also, the pressurizer heaters were to be energized at 60F and deenergized at 70F SCM. The operator accordingly opened the PORV at 5.3 minutes. The primary system depressurized from 2050 psia at 5.3 minutes to 1600 psia at 7.3 minutes, obtaining a depressurization rate of 225 psi/min. The SCM dropped below 50F at 6.6 minutes. The operator closed the PORV (and energized the pressurizer heaters) at 7.3 minutes, by which time the SCM was less than 35F. (The succeeding manual actions were performed within a few degrees of the specified SCMs; the sequence of these operator actions throughout the test is listed in Table 5.5.1.)

The tube rupture flow rate decreased abruptly at 5 minutes (Figure 5.5.1). Although this decrease corresponded to a primary depressurization through PORV actuation (Figure 5.5.2), the differential pressure across a valve in the rupture simulation circuit (Figure 5.5.3) did not recover upon subsequent primary system repressurizations (Figure 5.5.2). The rupture-site fluid remained subcooled (Figure 5.5.4), and the hot leg levels remained full, thus the rupture simulation flow area is concluded to have been approximately 50% blocked after 5 minutes.

The continuing sequence of PORV manipulations and continuing cooldown were repetitive and uneventful. The temperature difference across the heat transfer components (core and steam generators) gradually rose, and the time-averaged loop flow rates gradually declined in response to the decreasing natural circulation driving head. This driving-pressure decrease was caused by the decreasing core power (simulating decay) and by the decreasing thermal expansion coefficient of water (with the decreasing primary-system average fluid temperatures). At 10.7 minutes AFW was restored to steam generator A to maintain level, and at 13 minutes it was restored to steam generator B. The steam generator steaming and feed rates finally stabilized near 2% of the scaled full steam generator steam and feed flow rates (Figure 5.5.5). Due to the primary-to-secondary tube leak flow, however, the steam generator B feed flow rate remained discernibly lower than the other (steam and feed) rates. The differential pressure across the open RVVVs gradually decreased as steam generator heat transfer was reactivated. The valves closed singly between 14.5 and 19.8 minutes.

The primary system depressurization upon PORV actuation at 30 minutes lowered the saturation temperature to 578F. The initial core outlet temperature, 593F, set the current reactor vessel upper-elevation fluid temperature; thus, the reactor vessel began to void. This voided volume successively expanded toward the elevation of the RVVVs during the PORV actuations at 41, 53, and 69 minutes. The depressurization at 69 minutes lowered primary system pressure from 1020 to 800 psia (Figure 5.5.6), and saturation temperature from 547 to 518F. The stagnant upper downcomer fluid saturated and began to void, whereas the reactor vessel level stayed near the elevation of the top of the downcomer.

Isolation of the Affected Steam Generator, Single-Loop Cooldown

The affected steam generator (B) was isolated, as specified, as the primary was depressurized below 950 psia at 70 minutes (Figures 5.5.5 and 5.5.6). As the steam generator B AFW and steaming were terminated, the loop B flow rates dropped abruptly from 3% to nearly 1% of scaled full flow. The steam generator B secondary began to gradually repressurize from 350 psia at approximately 10 psi/min. The steam generator B secondary level began to increase at approximately 10 ft/h, from 21 ft (Figure 5.5.7). The ongoing changes in reactor vessel voiding as well as loop fluid density profiles caused the RVVV differential pressure to decrease drastically between 70 and 80 minutes (to less than -0.15 psi).

The loop B cold leg flow rates stabilized near 1-1/2% (of scaled full cold leg flow rate, each) beyond 70 minutes, then began to diminish further beyond 80 minutes. At 87 minutes, relatively cold fluid was displaced backward over the cold leg B1 spillover (at the pump) and the loop B cold legs began to interact (Figure 5.5.8). The flow directions were backward in cold leg B1 and forward in cold leg B2, the net loop B flow was virtually zero. The loop B hot leg fluid temperature trends confirmed this stagnation of the total loop B flow -- they remained virtually constant after 80 minutes. The loop B hot leg inlet fluid temperature evidenced the periodic passage of colder fluid and, hence, a generally decreasing trend, however. The steam generator B secondary pressure increase halted at 470 psia at 85 minutes (Figure 5.5.6) as the secondary fluid equilibrated with the nearly stagnant primary fluid. Steam generator B then began to slowly depressurize. With inter-cold leg flow in loop B, the steam generator primary outlet fluid tracked with the cold leg suction fluid temperatures. The flow reversal in cold leg B1 began to cool the steam generator outlet at 90.5 minutes, at approximately the same time as the interruption of the steam generator B secondary repressurization. At 90 minutes, the primary fluid temperature difference across steam generator B, which had been decreasing, abruptly increased from 18 to 40F, signalling the arrival of the HPI-cooled cold leg fluid at the steam generator primary outlet. The lower-elevation steam generator B primary and secondary fluid cooled with the primary outlet fluid, both at the onset of inter-cold leg flow and subsequently. However, this secondary-to-primary heat transfer was in evidence only at the lowest steam generator elevations.

The periodic PORV actuations maintained the core exit SCM within the specified band, gradually lowered the time-averaged primary pressure, and hence decreased the pressure difference across the ruptured tube (Figure 5.5.6). The PORV actuation from 153 to 155 minutes lowered this pressure difference across the break from 280 to 125 psi. The resulting reduction of break flow rate momentarily reduced the rate of refill of the steam generator B secondary (Figure 5.5.7).

The PORV actuation from 218 to 222 minutes saturated the generally stagnant loop B hot leg fluid. The pressurizer level, which was being controlled near 21 feet by manual HPI throttling, quickly rose to 24 ft as the loop B U-bend region voided. The loop B voiding persisted from 220 to 240 minutes and achieved loop B hot leg riser and stub levels as low as 59 ft, some 7 feet below the U-bend spillover elevation (Figure 5.5.9). The middle elevation and upper elevation B hot leg fluid temperatures followed saturation through the depressurization period -- the B U-bend fluid temperature had been some 15F less than the B hot leg mid-elevation fluid temperature, apparently reflecting uncompensated heat losses in the U-bend region (Figure 5.5.10). During the repressurization and hot leg refill period following PORV closure, the U-bend temperature stabilized near its value preceding the event and the mid-elevation temperature increased in parallel with the saturation temperature. The hot leg B inlet fluid temperature cooled most noticeably, reflecting the displacement of cooler core exit fluid into the hot leg as hot leg B was refilled.

The PORV was again opened to control SCM at 258 minutes, only some 18 minutes after the loop B U-bend had been refilled. Again the middle- and upper-elevation hot leg B temperatures decreased with saturation temperature, and the hot leg B inlet temperature increased as the warmer upper elevation hot leg fluid was displaced downward. Unlike the earlier hot leg voiding event, the U-bend fluid superheated slightly during the depressurization and followed saturation during the subsequent repressurization. Also, the hot leg B mid-elevation fluid temperatures before and after the event indicated a net reduction of some 15F, apparently achieved by the inflow of the cooler, lower elevation hot leg fluid.

At 298 minutes, the steam generator A steam and feed flow rates were reduced toward zero. The control temperature difference (core exit temperature minus steam generator A secondary saturation temperature) at this time was 48F, 2F below the control value and some 5F below its preceding maximum difference. The ATOG pressure control system thus apparently malfunctioned, as described in section 4. A less extensive malfunction of the temperature-difference control occurred near 240 minutes, and another pronounced malfunction occurred between 400 and 420 minutes.

The reductions of the steam generator A feeding and steaming rates decreased steam generation heat transfer, loop flow, and the primary system cooldown. The attenuated cooldown rate caused the core exit SCM to stabilize so that the PORV actuation sequence was altered. The operator was able to maintain the SCM between 60 and 70F by deenergizing the pressurizer heaters at 287 minutes and reenergizing them at 310 minutes.

The rate of refill of the isolated steam generator B secondary had gradually diminished with primary depressurizations. As the primary-to-secondary pressure difference was reduced with the PORV closed from 264 to 347 minutes, the steam generator B secondary finally filled completely at 318 minutes. Then, the steam generator B secondary quickly pressurized to the primary system pressure, 450 psia at 322 minutes (Figure 5.5.6). The loop B hot leg riser and stub were refilled to the U-bend spillover elevation at 320 minutes, just after the refill of the isolated steam generator secondary. The hot leg B temperature trends indicate that the loop B fluid was momentarily displaced backward following the refill of the U-bend: The U-bend fluid temperature dropped approximately 25F (to 25F subcooled) while the lower elevation hot leg B riser fluid temperatures abruptly increased by a similar amount.

The primary system was again depressurized, from 347 to 350 minutes, by PORV actuation. The loop B cold leg flow rates briefly became active, reaching 2% (of the scaled full cold leg flow rate), and the downcomer flow rate spiked at nearly 3% (of scaled full flow rate); the loop A cold leg flow rates remained near 3%. This loop B flow activity was apparently of insufficient duration to much affect the loop B hot leg fluid temperatures, which were more responsive to the ongoing depressurization. The steam generator B primary level descended slightly below the steam generator upper tubesheet at 351 minutes, then the usual repressurization, pressurizer outsurge, and hot leg B refill began. The net effect of this depressurization and repressurization cycle, like that between 257 and 322 minutes, was to cool the lower loop B hot leg fluid temperatures apparently by fluid displacement.

At 400 minutes, the feeding and steaming of steam generator A were reduced due to a control system malfunction, as has previously been discussed. As in the preceding event, the steam generator A secondary repressurized, the primary system cooldown was interrupted, and the operator was able to control the SCM using pressurizer heaters. The steaming of the steam generator A was restored at 413 minutes and the PORV was opened at 417 minutes to control the SCM. The hot leg B U-bend had just been refilled as the PORV was opened, thus some movement of the loop B fluid was obtained. Hot leg B was refilled at 440 minutes, following PORV closure at 422 minutes. Beyond 440 minutes, the loop B hot leg fluid subcooled throughout by about 40F (Figure 5.5.11). The order of subcooling versus elevation suggests that the loop B hot leg fluid movement was in the reverse-flow direction (and at a slow rate, perhaps 2 ft/min).

The test was terminated at 540 minutes based on the maximum duration termination criteria. At termination, the hot leg B fluid remained approximately 25F subcooled. Steam generator A remained active, and the loop A cold leg flow rates were each 2.7% (of scaled full cold leg flow rate). The primary and isolated steam generator secondary pressures were 350 psia and the active steam generator secondary pressure was 75 psia (i.e., near the facility minimum secondary pressure).

5.5.2. Rupture of 1 Tube, Pressurizer Venting, and Depressurization Methods, Test 7

Test 7 (340799) employed a simulated single-ended, high-elevation rupture of 1 steam generator tube, isolation of the affected steam generator, and a prolonged single-loop cooldown. After 15 hours, several methods of primary system cooling and depressurization were introduced. These depressurization and cooldown steps had little lasting effect on the core outlet fluid temperature. After 19 hours, the core outlet fluid temperature remained at 330F, the core fluid temperature rise was 34F, and the SCM was 21F. The core inlet fluid temperature, 296F, equalled the steam generator secondary saturation temperature. Test 7 (340799) consisted of the following three major phases:

1. Rupture initiation, two-loop cooldown, and depressurization.

2. Isolation of the affected steam generator, single-loop cooldown.

3. Primary system depressurization steps. These steps were to (1) raise the steam generator A secondary level, (2) open the hot leg B high-point vent, (3) perturb HPI, and (4) reactivate the isolated steam generator.

These phases are described in the sections below.

5.5.2.1. Initiation, Two-Loop Cooldown, and Depressurization

Test 7 was initialized in natural circulation at 3.9% of scaled full power and a downcomer flow rate of 4.5% of scaled full flow. The core fluid temperature rise was 43.2F (Figure 5.5.12), the primary fluid temperature drop across both steam generators was 37.5F. The primary system pressure was 2153 psia, and the SCM was approximately 50F.

Initiation

Test 7 was initiated by opening the scaled 1.5-cm^2 leak (simulating the single-ended rupture of 1 tube) at the top of steam generator B, and by activating several boundary system simulations. These systems and their initial responses were as follows:

- Tube Rupture: The tube system limit switch activated and the primary-to-secondary flow rate began to register, both at time 0. The rupture flow rate stabilized near 95 lbm/h (Figure 5.5.13). Primary system pressure decreased slightly, stabilizing at 1980 psia beyond 2 minutes.

- Steam Generator Secondary Refill: The feed flow rate increased and the steam generator secondary levels began to rise, more rapidly in steam generator B than A, at 0.1 minutes (Figure 5.5.14).

- Core Power: At 0.1 minutes, core power began to decrease in response to the activation of the core power decay ramp.

- Reactor Vessel Vent Valves: At 0.1 minutes, the RVVVs opened and the differential temperatures across the valves began to decrease. The valves remained open only briefly (Figure 5.5.15). RVVV A1 closed first, near 0.5 minutes, by both limit switch and temperature difference indications.

- Abnormal Transient Operating Guideline Steam Generator Pressure Control: The steam flow rates increased at 0.4 minutes.

The scaled 1-cm^2 pressurizer vent was opened approximately 20 seconds after test initiation. This vent simulated the plant pressurizer vent. The MIST vent was an ad hoc fixture, without remote indications. The PORV total flow rate, which also metered the pressurizer vent flow rate, began to indicate flow shortly thereafter (Figure 5.5.16).

Steam Generator Refill

The steam generator steaming rates were gradually reduced (Figure 5.5.17) as the refill feeding lowered the secondary pressures (Figure 5.5.18). Because primary-to-secondary heat transfer remained active, however, the secondary pressures were only slightly affected, reaching 900 psia at the completion of secondary refill. Steam generator B was filled to the control level, 20.6 ft, slightly before steam generator A due to the primary-to-secondary mass transfer in steam generator B (Figure 5.5.19). Steam generator B feed was interrupted at 4.5 minutes and feed was continued to steam generator A until 5.3 minutes (Figure 5.5.17).

RVVV Actuation

HPI was adjusted to establish and maintain pressurizer level. The HPI flow rate peaked above 800 lbm/h at 2.9 minutes and was briefly interrupted beyond 4.8 minutes (Figure 5.5.16) as the steam generator feed rates and, hence, the primary system cooldown rate were reduced (Figure 5.5.20). The HPI flow rate and the steam generator secondary control variations affected both the primary system flow rate (Figure 5.5.21) and the RVVV differential pressures (Figure 5.5.22). The differential pressures approached the 0.125-psi actuation value at 6.7 minutes, and the 0.04-psi closing value beyond 11.5 minutes. Although the differential pressures across the four individual valves were virtually identical, only RVVVs B1 and B2 responded, opening at 6.6 minutes and closing at 12.5 minutes (Figure 5.5.23). The downcomer flow rate, too, was quite variable during this time, decreasing to as low as 3% of scaled full flow (Figure 5.5.21).

Two-Loop Cooldown

The system cooldown gradually became more stable (Figure 5.5.20). The downcomer flow rate remained near 4% of scaled full flow. The steam generator secondaries were depressurized at 22 psi/min (Figure 5.5.18) to obtain a secondary cooldown rate of 100F/h. The primary loop fluid temperatures declined similarly, the core exit-to-secondary saturation temperature difference varied between 60 and 75F. HPI at about 200 lbm/h was used to offset the contraction of the primary system liquid (Figure 5.5.16).

5.5.2.2. Isolation of the Affected Steam Generator, Single-Loop Cooldown

Steam Generator Isolation

The hot leg fluid temperatures achieved 525F at 31 minutes, satisfying the criterion for the isolation of the affected steam generator. Steam generator B feed and steaming were interrupted (Figure 5.5.17), the steam generator B secondary began to repressurize (Figure 5.5.18), and its level began to increase (Figure 5.5.19), as did its primary outlet fluid temperature. The steam generator B secondary level rose at 0.4 ft/min, but increased less rapidly as the primary-to-secondary pressure difference for steam generator B dwindled.

Upon isolation, the steam generator B secondary pressure increased toward saturation at the current hot leg inlet fluid temperature, rising from 650 psia upon isolation to almost 800 psia at 45 minutes (Figure 5.5.24). The steam generator B primary outlet and loop B cold leg fluid temperatures increased correspondingly, then decreased irregularly (Figure 5.5.25), apparently reflecting the stagnation of loop B primary flow. The feed and steam rates of steam generator A gradually increased to 5% (of scaled full steam generator secondary flow), approximately twice their value before the isolation of steam generator B (Figure 5.5.26). The downcomer flow rate gradually decreased toward 2-1/2% of the scaled primary flow (Figure 5.5.27).

Reactor Vessel Voiding

The reactor vessel upper head (RVUH) fluid temperature was virtually unaffected by the continuing two-loop cooldown (Figure 5.5.28). The RVUH fluid temperature decreased only 5F in the first half-hour of cooldown. At 34 minutes, the primary saturation temperature was reduced to the upper head fluid temperature, and reactor vessel voiding ensued (Figure 5.5.29). The accompanying displacement of reactor vessel fluid to the pressurizer prevented pressurizer level control using throttled HPI; thus, the HPI flow rate was gradually reduced toward zero (Figure 5.5.30).

Actuation of RVVV B2

The differential pressures across the RVVVs rose as HPI was throttled and as the primary loop flow rate declined. The valve differential pressures approached the actuation level, 0.125 psi, at 47 minutes (Figure 5.5.31). Curiously, only RVVV B2 opened (Figure 5.5.32) and the valve differential pressures, which had been virtually identical, now evidenced a distinct difference: The differential pressure across the open RVVV (B2) remained about 0.005 psi greater than the other three differential pressures. At 48 minutes (Figure 5.5.30), the operator increased the HPI flow rate to maintain pressurizer level. The vent valve differential pressures thus decreased slightly, with RVVV B2 remaining open.

The reactor vessel (collapsed liquid) level descended to the vicinity of the RVVVs, and stabilized there at 63 minutes (Figure 5.5.29). The operator adjusted the HPI throttling to maintain pressurizer level.

The steam generator B secondary level stabilized briefly, between 70 and 75 minutes, as the horizontal piping to the steam generator downcomer was being filled (Figure 5.5.33).

Loop B Stagnation and Voiding

The loop B hot leg fluid had cooled only gradually after the isolation of steam generator B, decreasing from 520F at 42 minutes to 505F at 95 minutes (Figure 5.5.34). The loop B primary fluid saturated. The hot leg B riser and stub levels descended at nearly 1/2 foot/min beyond 95 minutes (Figure 5.5.25). The pressurizer level rose correspondingly (Figure 5.5.36), therefore the operator interrupted HPI. Loop B inter-cold leg flow became active, based on the cold leg fluid temperatures -- the steam generator B primary outlet fluid temperature stabilized whereas the loop B cold leg temperatures continued to decline (Figure 5.5.25). Cold leg B1 led B2, indicating reverse flow in cold leg B1 and forward flow in cold leg B2 (in the circuit formed by the two adjacent cold legs and linking the steam generator primary outlet and the downcomer). The rate of primary system depressurization abruptly decreased (Figure 5.5.24). At this time, too, the appropriate conditions were met and the CFT was isolated without having discharged.

Level Descends to the Rupture Elevation

The loop B hot leg riser and stub levels approached the elevation of the steam generator upper tubesheet, and of the tube rupture, between 120 and 125 minutes (Figure 5.5.35). In response to the change of state of the fluid at the rupture site, the primary-to-secondary mass flow rate dropped abruptly at 124 minutes (Figure 5.5.30). The hot leg B levels thereafter decreased more gradually than before the tube rupture was uncovered and the pressurizer level increased more gradually. The steam generator B secondary level stabilized at 41 feet, 20 feet above its level at isolation (Figure 5.5.33).

Reduction of Steam Generator B Secondary Pressure

The steam generator B secondary pressure had peaked at a saturation corresponding to the hot leg inlet fluid temperature, 520F and 800 psia, at 45 minutes. Thereafter, the loop B flow stagnated, and the hot leg B fluid temperatures decreased only gradually, thus the steam generator B secondary pressure slowly diminished. Whereas the loop B hot leg fluid saturated before 100 minutes, the primary and secondary pressures of steam generator B equalized only after 135 minutes. Thereafter, the steam generator B secondary pressure declined with the primary system pressure (Figure 5.5.24) and the steam generator B secondary fluid mass approximately stabilized.

Liquid at the Pressurizer Vent

Although HPI remained off, the pressurizer level continued to rise as the loop B levels declined. The pressurizer indicated full at 190 minutes (Figure 5.5.36) and, at 192 minutes, the pressurizer vent flow rate (shown as "PORV" flow rate, Figure 5.5.30) abruptly increased to 25 lbm/h. The primary system depressurization rate decreased slightly, as did the rate of decline of the loop B levels (Figure 5.5.35), both in response to the decreased volumetric flow rate discharged by the vent.

Secondary-to-Primary Flow

The loop B hot leg riser emptied at 245 minutes (Figure 5.5.35). The pressurizer vent discharge mass flow rate abruptly decreased (Figure 5.5.30), indicating vapor at the vent, although the pressurizer level continued to indicate full (Figure 5.5.36), and the rate of primary system depressurization increased from 450 psia. The steam generator B secondary began to lose fluid mass to the primary as the secondary-to-primary pressure difference stabilized at 25 psi (Figure 5.5.24). The mid-height hot leg B fluid temperature, which had remained about 10F superheated, increased to 490F and stabilized.

The steam generator B primary outlet fluid temperature, which had remained near 420F since loop B saturated at 95 minutes, cooled toward the cold leg B1 fluid temperature (340F) between 275 and 325 minutes. This change may have been caused by enhanced inter-cold leg flow within loop B. The indicated pressurizer level began to recede from full at 315 minutes (Figure 5.5.36) with no change in loop interactions.

Minimum SCM

The primary system cooldown rate gradually diminished. The steam generator A secondary pressure reached 100 psia at 300 minutes (Figure 5.5.37) and decreased only 40 psi thereafter. The SCM decreased as the primary system depressurization overtook the primary cooldown. At 430 minutes, the SCM approached 25F (Figure 5.5.38) and the operator closed the pressurizer vent as specified. The pressurizer then transferred inventory to the loop B riser and steam generator primary (Figures 5.5 39 and 5.5.40). Also, the primary system pressure stabilized sufficiently so that the steam generator B secondary depressurized to the primary pressure, 200 psia (Figure 5.5.37). The steam generator B secondary depressurized as the primary side refilled. HPI was reactivated at 547 minutes (Figure 5.5.41) to stabilize the pressurizer level at 24 feet.

Rupture Recovered

The hot leg B levels achieved the upper tubesheet and tube rupture elevations at 710 minutes (Figure 5.5.40). The steam generator B secondary had gradually depressurized below the primary, beginning at 600 minutes, therefore tube rupture flow was reactivated (Figure 5.5.41). The rupture flow rate increased to 40 lbm/h, the loop B primary levels stabilized, and the steam generator B secondary level began to rise from 35 feet (Figure 5.5.42). These conditions persisted until 15 hours, at which time the primary depressurization phase was begun. At 15 hours, the primary and secondary system pressures (psia) were 157, 66 (A), and 132 (B). The core inlet and outlet fluid temperatures (Figure 5.5.43) were 302 and 338F. The hot leg B riser and stub levels remained at the rupture elevation (Figure 5.5.40), and the steam generator B secondary had been nearly refilled. The reactor vessel level remained 1 foot above the RVVVs (Figure 5.5.44), and the pressurizer level continued to be controlled near 24 feet. The tube rupture flow rate approximated the throttled HPI flow rate at about 17 lbm/h (Figure 5.5.41).

5.5.2.3. Primary System Depressurization Steps

The four primary system depressurization steps were implemented between 15 and 19 hours. The conditions during this period are summarized in Table 5.5.2.

Increased Steam Generator A Secondary Level (15 hours)

The operator raised the steam generator A secondary level to 31.6 ft at 900 minutes (Figure 5.5.45), as specified. The feed flow rate to steam generator A was increased to 12.5% of scaled full steam generator secondary flow at 900 minutes (Figure 5.5.46). The steam generator A secondary level rose from 20.5 to 32.6 ft at 904 minutes. The steam generator A secondary pressure dropped from 66 to 59 psia, then returned to its former value by 910 minutes (Figure 5.5.47). The downcomer flow rate jumped from 1.8 to 2.7% of full scaled flow during the steam generator A secondary level increase, then subsided (Figure 5.5.48). The core region and active loop fluid temperatures dropped during the flow surge, then returned to within about 6F of their former values (Figure 5.5.49). The net effect on core outlet temperature was a reduction from 338 to 332F.

The subcooling margin increased by a similar amount, but the primary system continued to depressurize slowly. The RVVV differential pressures dropped sharply during the flow surge (Figure 5.5.50). RVVV B2, which had remained open singly for most of the test, finally closed (Figure 5.5.51). The RVVV B2 differential pressure remained slightly greater than the others.

Actuation of the Hot Leg B High-Point Vent (16 hours)

The operator opened the loop B hot leg high-point vent at 16 hours, as specified. The loop B hot leg and stub levels began to rise (Figure 5.5.52), covering the rupture site. The steam generator B secondary refilled at 970 minutes (Figure 5.5.53), thus the rupture flow stopped (Figure 5.5.54). The operator increased the HPI flow rate to maintain pressurizer level, but reduced it as the rupture flow stopped. The steam generator B secondary repressurized from 126 to 143 psia (Figure 5.5.55), which was the primary system (reactor vessel) pressure plus elevation head. Between 16 and 17 hours, the loop B hot leg levels increased from 52 feet to only 55 feet (Figure 5.5.52). The core region fluid temperatures were largely unaffected by the hot leg high-point vent opening (Figure 5.5.56). At 17 hours, the core inlet and outlet fluid temperatures were 297 and 331F. The SCM was 27F.

HPI Perturbations (17 hours)

The operator perturbed HPI between 17 and 18 hours, as specified. The hot leg B high-point vent was inadvertently left open. At 1020 minutes (17 hours), the operator opened the pressurizer vent and deactivated HPI. HPI had been almost fully throttled prior to its deactivation. The uppermost pressurizer fluid was apparently vapor, so the pressurizer vent discharge mass flow rate was negligible (Figure 5.5.57). However, the vent opening did have several subtle effects. The hot leg B levels, which had been rising, stabilized near 55 feet (Figure 5.5.58). Primary system pressure began to slowly decline (Figure 5.5.59), causing a similar reduction of the SCM.

The SCM approached 25F at 1040 minutes (Figure 5.5.60), causing the operator to actuate full HPI as specified. Full HPI flow rate was used, 670 lbm/h (Figure 5.5.57).

The primary system and steam generator B secondary pressures rose rapidly from 140 to 260 psia (Figure 5.5.59). The SCM increased directly (Figure 5.5.60) from 23F at 1041 minutes. The hot leg B levels began a gradual increase (Figure 5.5.58), as did the reactor vessel level, but the pressurizer level increased rapidly from 24 ft (Figure 5.5.61). The pressurizer indicated full at 1047 minutes, the pressurizer vent mass flow rate began to register (indicating liquid at the vent, Figure 5.5.57), the primary system repressurization rate increased (Figure 5.5.59), and the SCM approached 100F at 1048 minutes, causing the operator to terminate HPI as specified.

The HPI actuation at 1040 minutes generally perturbed loop conditions. The lower downcomer, core, and hot leg A fluid cooled sequentially as the relatively cold HPI fluid traversed the loop (Figures 5.5.62 and 5.5.63).

The downcomer flow rate first increased from 1.8 to 2.6% of scaled full flow as the downcomer cooled, then decreased at 1043 minutes, to less than its initial value, as the hot leg cooled (Figure 5.5.64). The steam generator A steam and feed system controls became active. The RVVV differentials responded inversely with the loop flow rate (Figure 5.5.65), causing RVVV B2 (only) to open at 1043 minutes (Figure 5.5.66).

The reactor vessel collapsed liquid level rose from 25.2 feet upon HPI actuation and increased quite rapidly from 1046 to 1048 minutes (Figure 5.5.67) after the pressurizer filled (Figure 5.5.61). The hot leg B riser and stub levels momentarily dropped some 2 feet (Figure 5.5.58). Apparently due to this diversion of hot leg fluid to the reactor vessel, the downcomer flow rate dropped and the RVVV differential pressures increased dramatically (Figure 5.5.65), causing RVVV A2 to open at 1046 minutes (Figure 5.5.66). The hot leg B riser fluid abruptly cooled 5 to 10F due to this fluid displacement (Figure 5.5.63). The loop B cold legs began to interact. HPI fluid was displaced backwards in cold leg B2, causing the pump suction fluid temperature to plummet from 225F at 1047 minutes to 155F at 1049 minutes (Figure 5.5.68). The cold leg B suction fluid temperatures cooled beyond 1050 minutes (Figure 5.5.69).

HPI was interrupted at 1048 minutes (Figure 5.5.57) as specified, as the SCM approached 100F (Figure 5.5.60). The system conditions quickly returned toward their pre-actuation values. The RVVVs closed at 1051 minutes (Figure 5.5.66), the primary and steam generator B secondary pressures declined (Figure 5.5.59), and the downcomer flow rate stabilized near 1.8% of scaled full flow after 1070 minutes (Figure 5.5.64). The core region level gradually decreased from 27.5 feet (Figure 5.5.67). The post-actuation core inlet and outlet fluid temperatures were 292 and 325F, 4F less than their pre-actuation values. The hot leg B riser and stub levels stabilized near 58 feet (Figure 5.5.58). The pressurizer apparently remained liquid full until 1080 minutes as indicated by the continuing pressurizer vent flow rate (Figure 5.5.57). At 1080 minutes, the primary and secondary pressures (psia) were 170, 60 (A), and 162 (B). The steam generator B secondary remained liquid full. The SCM was 41F and stabilizing.

Reactivation of Steam Generator B (18 hours)

The operator reactivated the isolated steam generator B at 1080 minutes, as specified. Also at 1080 minutes, the pressurizer vent was closed. As steam generator B began to be steamed (Figure 5.5.70), a primary-to-secondary pressure difference developed in steam generator B (Figure 5.5.71), and the rupture flow rate reinitiated and increased to 20 lbm/h (Figure 5.5.72). The steam generator B secondary level receded from full at 1095 minutes, attaining 39 ft by 1140 minutes (Figure 5.5.73).

The primary system continued to gradually depressurize, reaching 136 psia by 1140 minutes. The steam generator B secondary was depressurized to the steam generator A secondary value, approximately 64 psia (Figure 5.5.71). But the loop B hot leg riser and stub levels remained at intermediate elevations, above the upper tubesheet but below the hot leg U-bend spillover (Figure 5.5.74). Thus, the steam generator B activity had a negligible impact on the core fluid temperatures. The core inlet and outlet fluid temperatures increased slightly during this final hour of testing (Figure 5.5.75) due to a corresponding increase of the (minimum) steam generator secondary pressure. The core inlet and outlet fluid temperatures at 1140 minutes were 296 and 330F, the SCM was 21F and decreasing gradually.

Test Termination

The test was terminated at 1140 minutes, having completed the specified primary depressurization steps. The primary-to-secondary (outlet-to-saturation) temperature difference across steam generator A was virtually zero. The core fluid temperature rise was approximately 34F. The primary and secondary pressures were 136 and 64 psia. The input core power was 33 kW or 1% of scaled full power. Of this input power, 40% was the MIST power augmentation to offset (uncompensated) heat losses to ambient. The decay power at 19 hours was approximately 20 kW or 0.6% of scaled full power. Because the MIST system temperatures were generally low at test termination, little heat was being lost to ambient. Thus, the core fluid temperature rise and loop conditions reflected nearly the full power input rather than the decay portion. Speculations are that the observed core fluid temperature rise, 32F, would have been approximately 20F at the decay power level. Also, the baseline temperature, the core inlet fluid temperature, would have remained coupled to the saturation temperature at the (minimum) steam generator secondary pressure. In MIST, this baseline temperature was approximately 295F, corresponding to a saturation pressure of 60 psia.

5.5.3. Comparison of Tests 2 and 7

Tests 2 and 7 were initiated similarly by opening a simulated single-ended tube rupture flow path at the top of steam generator B. In both tests, the rupture mass flow rate initially stabilized near 100 lbm/h (Figure 5.5.76). The tube rupture simulation became partially blocked at 5 minutes in Test 2, reducing the rupture flow rate to approximately one-half of its former value. The inter-test similarities were little affected by this blockage, however. The available HPI capacity far exceeded the rupture flow rate, and the rupture discharge was insufficient to cause rapid primary system depressurization, even with the unblocked rupture area. Thus, the general system conditions were little affected by the rupture blockage in Test 2.

Two-loop circulation continued in both tests. The operator controlled the SCM in Test 2 by intermittently actuating the PORV, whereas continuous pressurizer venting was used in Test 7. Although the primary system depressurization was incremental in Test 2 and continuous in 7, the general pressure trends remained similar through the first hour of the transient (Figure 5.5.77).

The affected steam generator was isolated (as specified) at 950 psia primary system pressure in Test 2, and at 525F hot leg fluid temperature in Test 7. Thus, isolation occurred at 31 minutes in Test 7 but not until 70 minutes in Test 2. Upon isolation, the corresponding loop flow gradually stagnated and the isolated steam generator (B) secondary pressurized to saturation at the steam generator B primary fluid temperature (Figure 5.5.77). The inter-test difference of primary system depressurization techniques now affected the loop voiding characteristics.

In Test 7, the loop B hot leg and steam generator primary levels gradually descended, completely voiding the hot leg B riser at 6 hours (Figure 5.5.78). In Test 2, on the other hand, the hot leg B intermittently voided and refilled. The periodic PORV actuations and consequent HPI adjustments obtained sufficient displacement of the loop B fluid to prevent continued voiding in Test 2. Due to the disparity of the loop B levels in Tests 2 and 7, their primary system total fluid masses diverged after 1-1/2 hours (Figure 5.5.79). As the steam generator B primary level descended into the steam generator, the now-isolated steam generator B secondary pressure merged with the primary system pressure (Figure 5.5.77). This event occurred near 140 minutes in Test 7, but not until 330 minutes in Test 2. Whereas the steam generator B secondary pressure generally remained just below the primary system pressure in Test 2, the ongoing primary system depressurization in Test 7 caused the steam generator B secondary pressure to remain slightly greater than the primary system pressure from 250 to 460 minutes (Figure 5.5.77). The tube rupture flow rates responded accordingly. In Test 7, the rupture flow rate became negligible beyond 2 hours, but a small (approximately 15 lbm/h) rupture flow rate persisted in Test 2 at 8 hours (Figure 5.5.80).

The depressurization of the intact steam generator, steam generator A, was completed by 5 hours in both tests (Figure 5.5.77). Whereas the steam generator A secondary pressure remained below 100 psia in Test 7, it periodically increased to 150 psia in Test 2 due to a control system anomaly.

Table 5.5.1 Operator Control of Pressurizer Heaters and PORV, Test 340203

Table entries give the time in minutes (after test initiation) of the associated operator action. The PORV was activated to reduce the subcooling of the core exit fluid from 75 to 50F, the pressurizer heaters were energized to raise the subcooling from 60 to 70F. The pressurizer heaters supplied 2.1 kW upon activation.

Heaters Off	PORV Opened	Heaters On	PORV Closed
3.1	5.3	7.6	7.6
16.2	18.5	19.1	19.7
26.0	29.6	30.3	31.3
37.7	40.9	42.1	42.9
52.0	52.8	54.1	55.0
63.8	69.0	70.5	71.5
82.8	95.3	96.3	97.2
104.5	107.9	108.0	109.5
117.6	122.8	123.7	124.5
132.5	137.3	138.1	138.9
145.2	153.2	154.2	155.3
163.0	170.7	171.6	172.5
178.6	218.6	218.8	221.8
241.5	257.8	260.1	263.7
287	--	310.2	--
315.3	346.7	349.9	352.6
380.7	--	404	--
417	418	419	422
445.7	487.3	489.8	493.3
505	(test terminated at 540 minutes)		

Table 5.5.2 Fluid Conditions During the Depressurization Phase

	Time				
Hours	15	16	17	18	19
Minutes	900	960	1020	1080	1140
Pressures, psia					
Reactor Vessel (RVGP01)	157	153	149	170	136
Secondary Steam Generator A (S1GP01)	66	66	65	60	64
Secondary Steam Generator B (S2GP01)	132	126	140	162	64
Temperatures, F					
Core Inlet (DCRT01)	302	298	297	293	296
Core Outlet (RVTC16)	338	332	331	327	330
Saturation					
@ RV Pressure	362	360	358	368	351
@ Steam Generator A Secondary Pressure	299	299	298	293	297
Temperature Differences, F					
SCM	24	28	27	41	21
Across Core (Out-In)	36	34	34	34	34
Across Steam Generator A (Primary Outlet - Secondary Saturation)	3	-1	-1	0	-1
Downcomer Flow Rate					
lbm/h	2900	3000	3100	3000	3000
% Full Flow	1.8	1.8	1.8	1.8	1.8

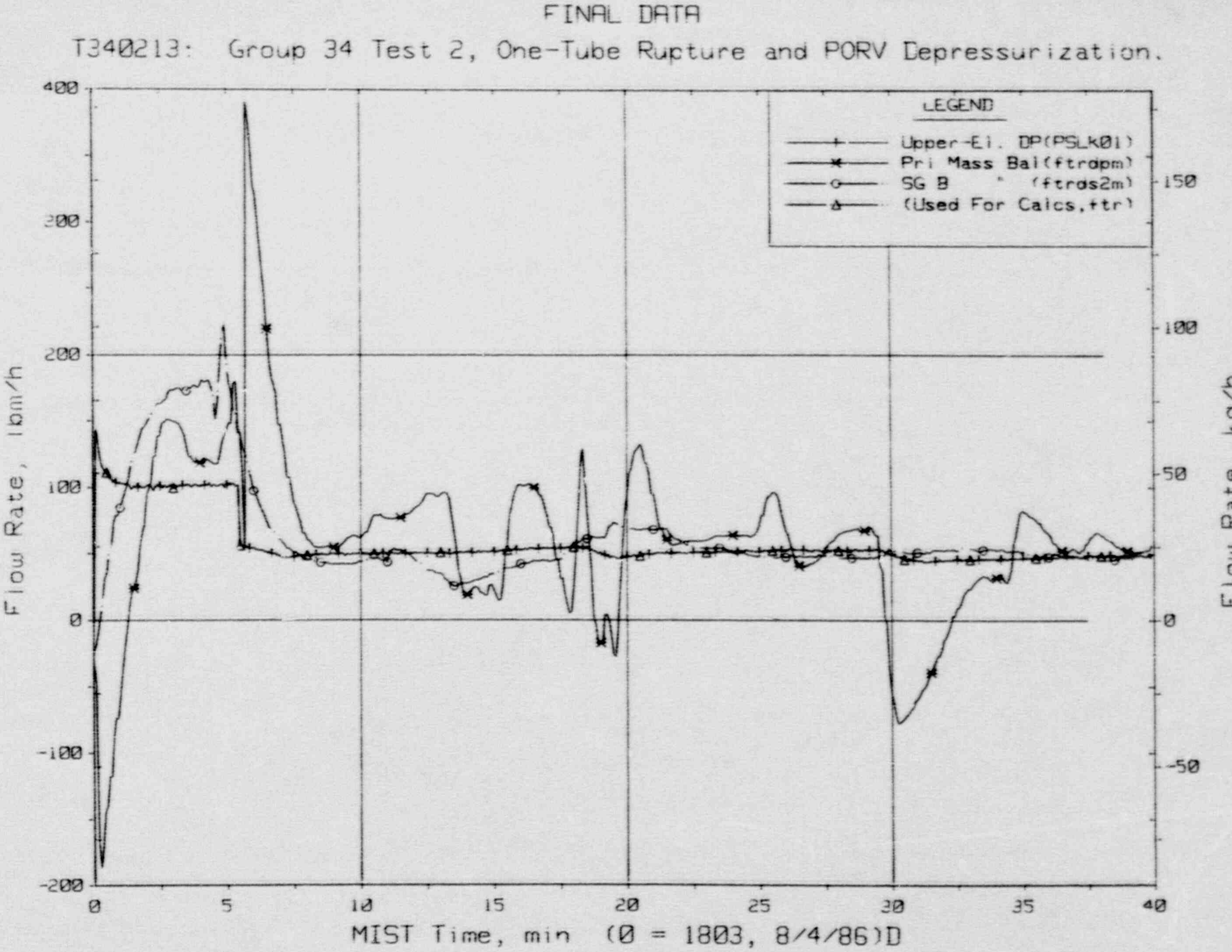


Figure 5.5.1. Tube Rupture Flow Rates

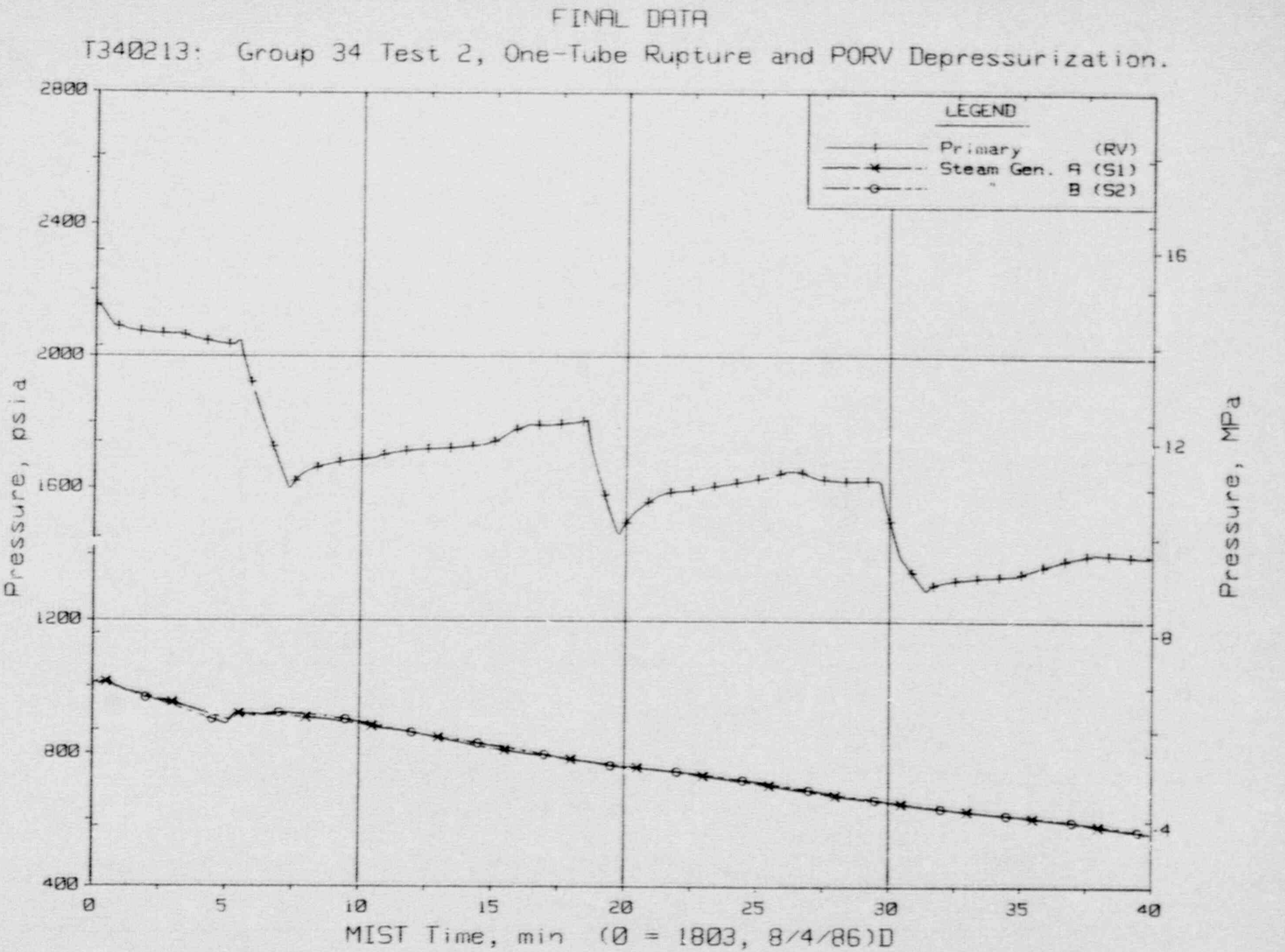


Figure 5.5.2. Primary and Secondary System Pressures (GPOls)

Tue Jul 26 14:17:29 1988

PSGP1

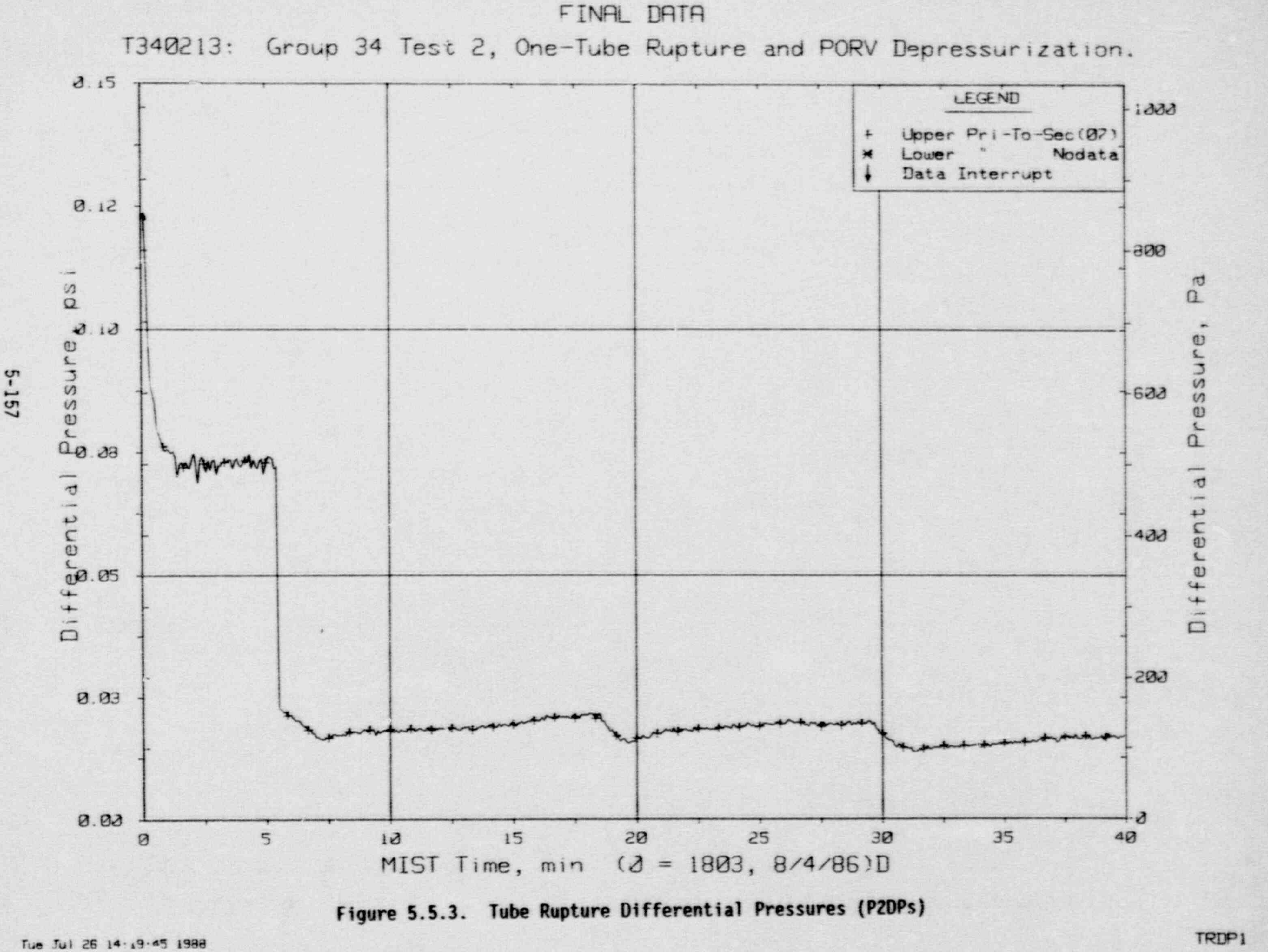


Figure 5.5.3. Tube Rupture Differential Pressures (P2DPs)

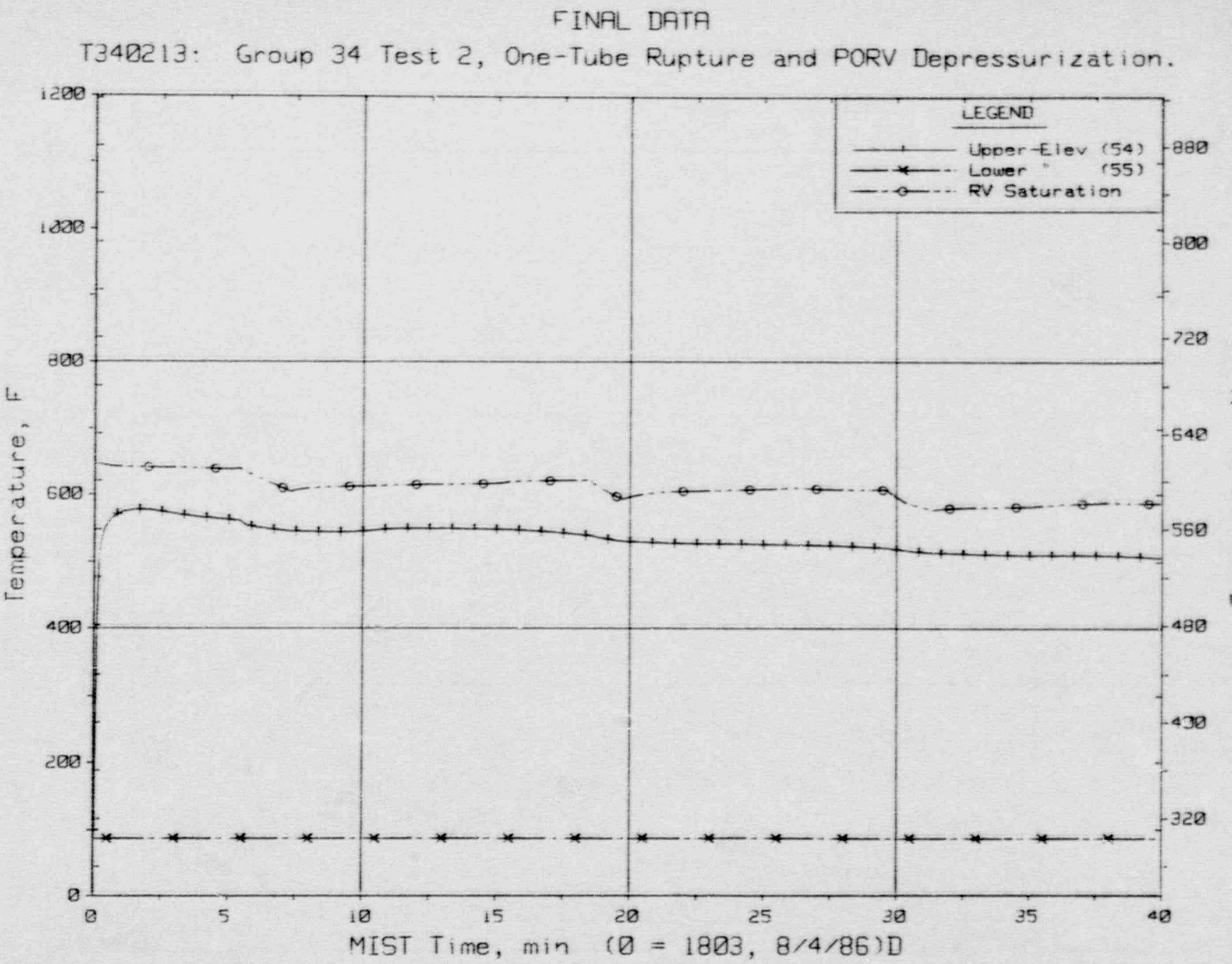
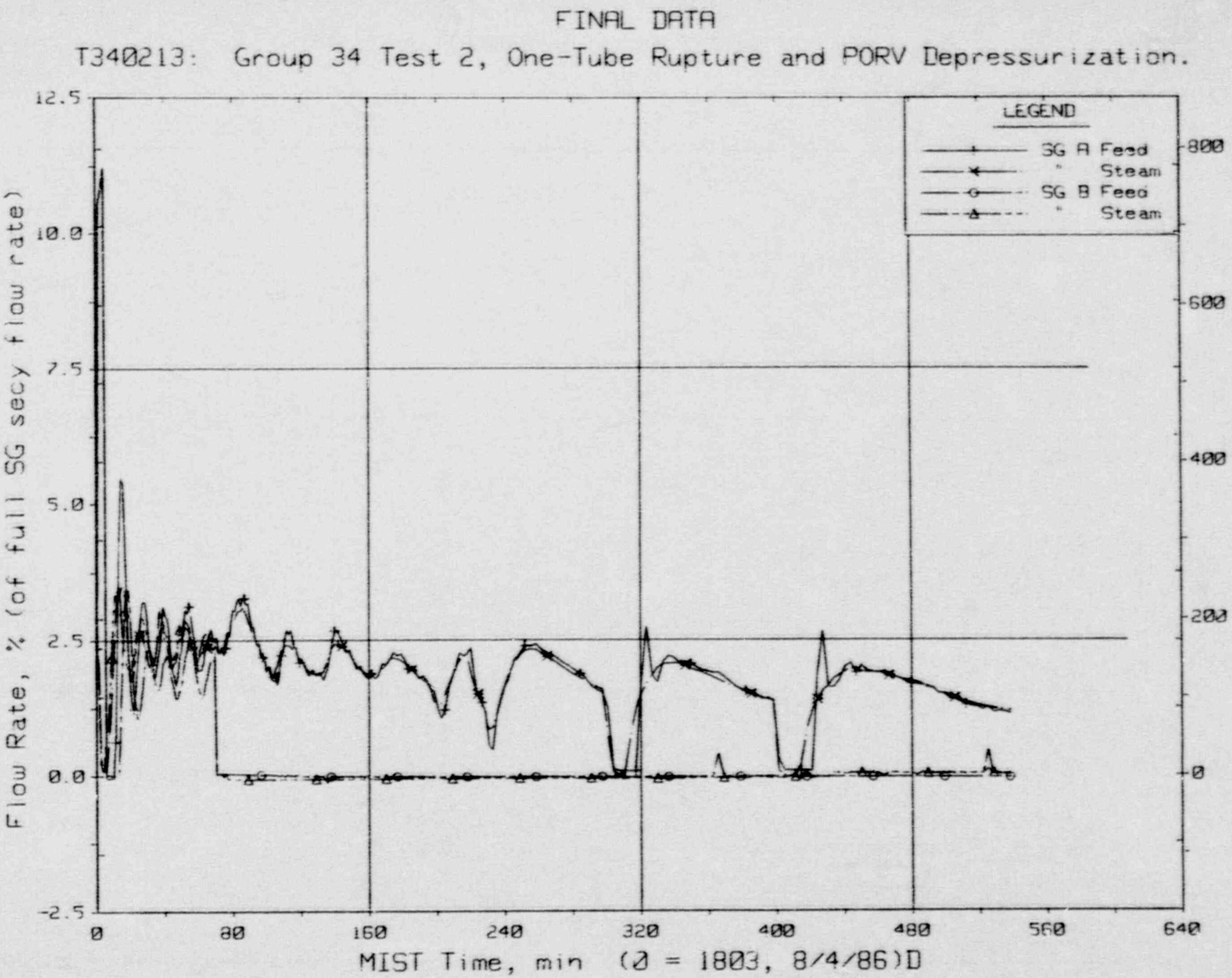


Figure 5.5.4. SGTR Fluid Temperatures (P2TCs)

Tue Jul 26 15:20:12 1988

TRTC1



5-159

Figure 5.5.5. Secondary System Flow Rates

Tue Jul 26 14:25:01 1988

SGFL2

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.

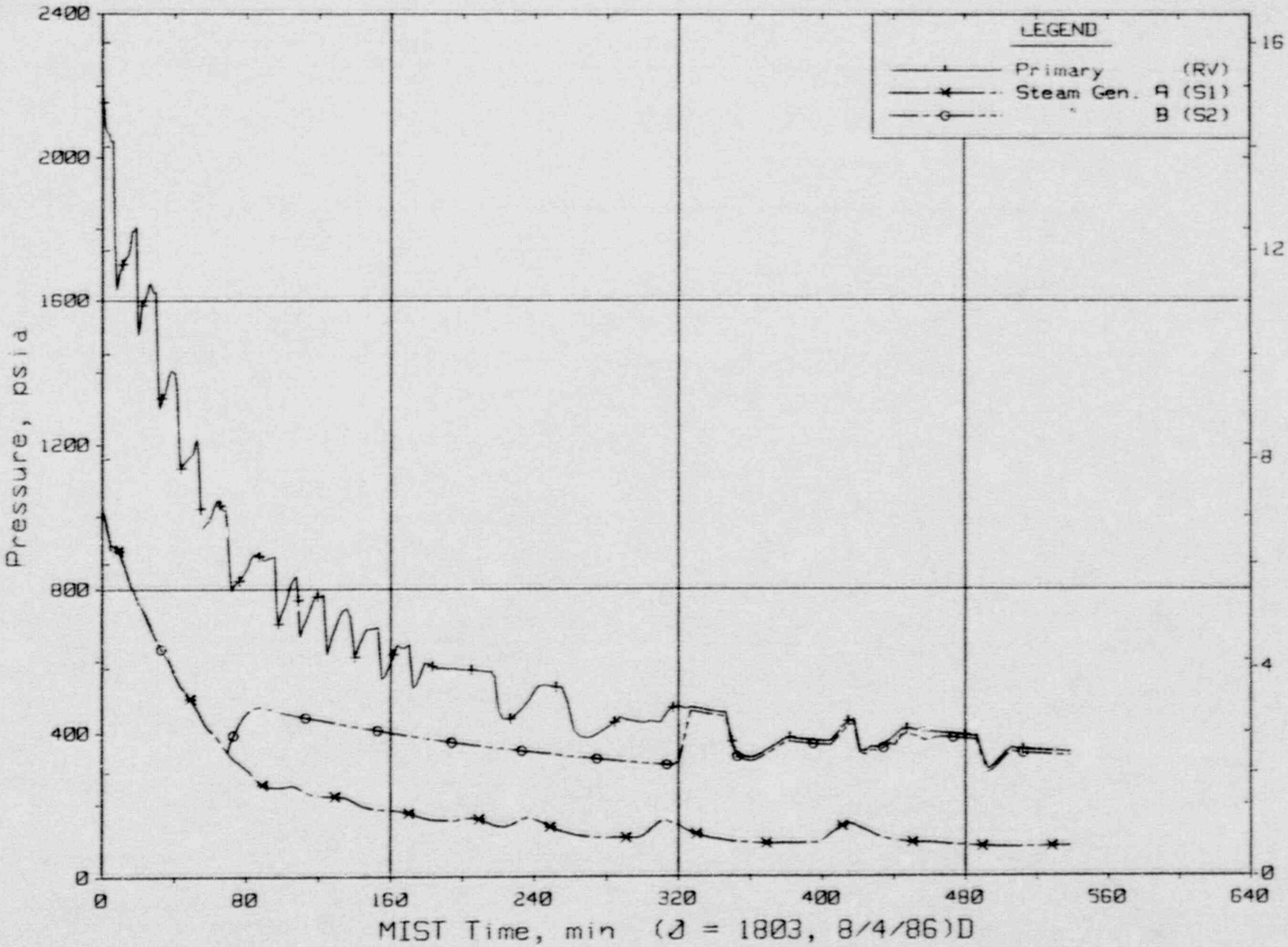


Figure 5.5.6. Primary and Secondary System Pressures (GPOls)

Tue Jul 26 14:27:20 1988

PSGP1

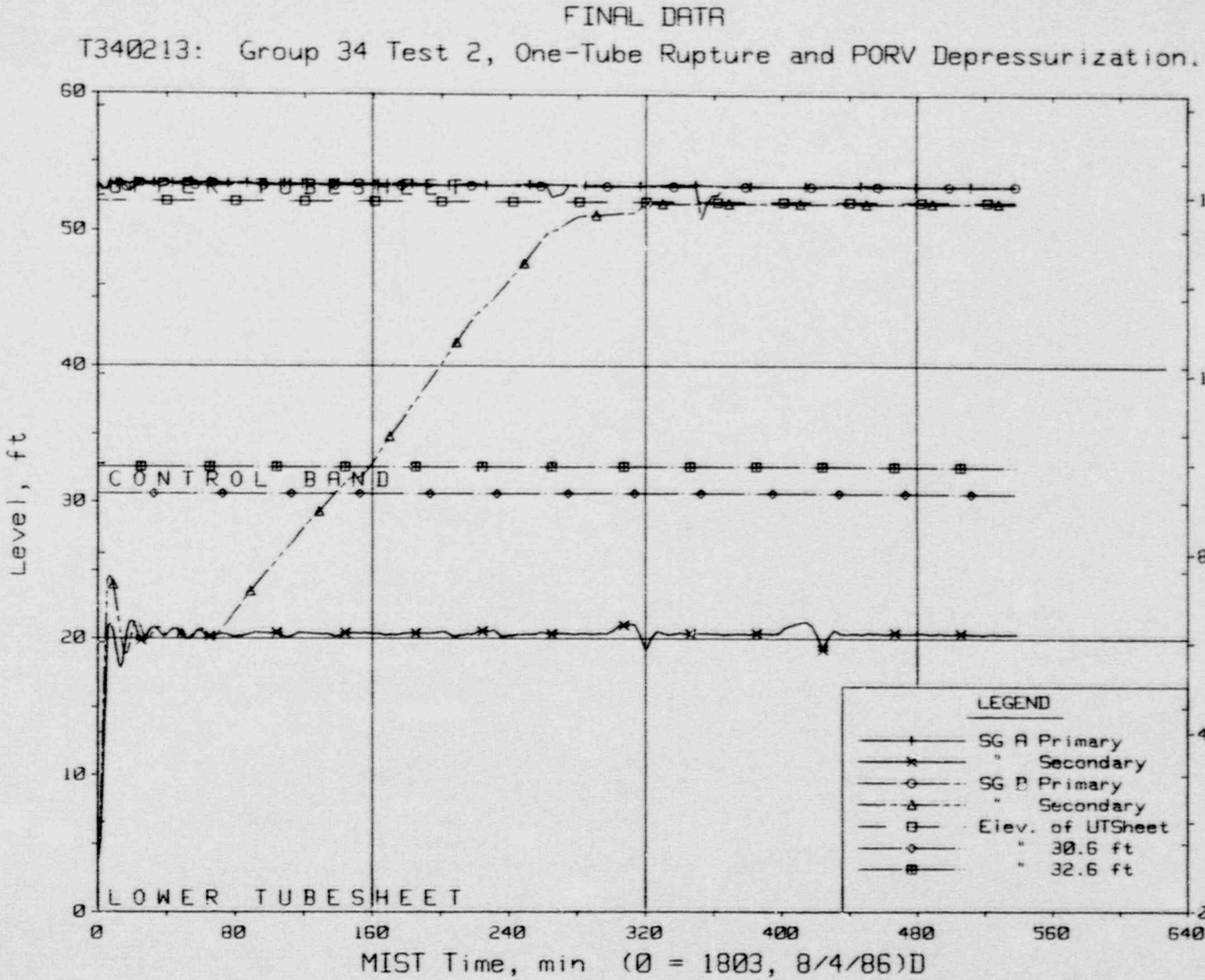


Figure 5.5.7. Steam Generator Collapsed Liquid Levels

Tue Jul 26 14:40:45 1988

SGLV2

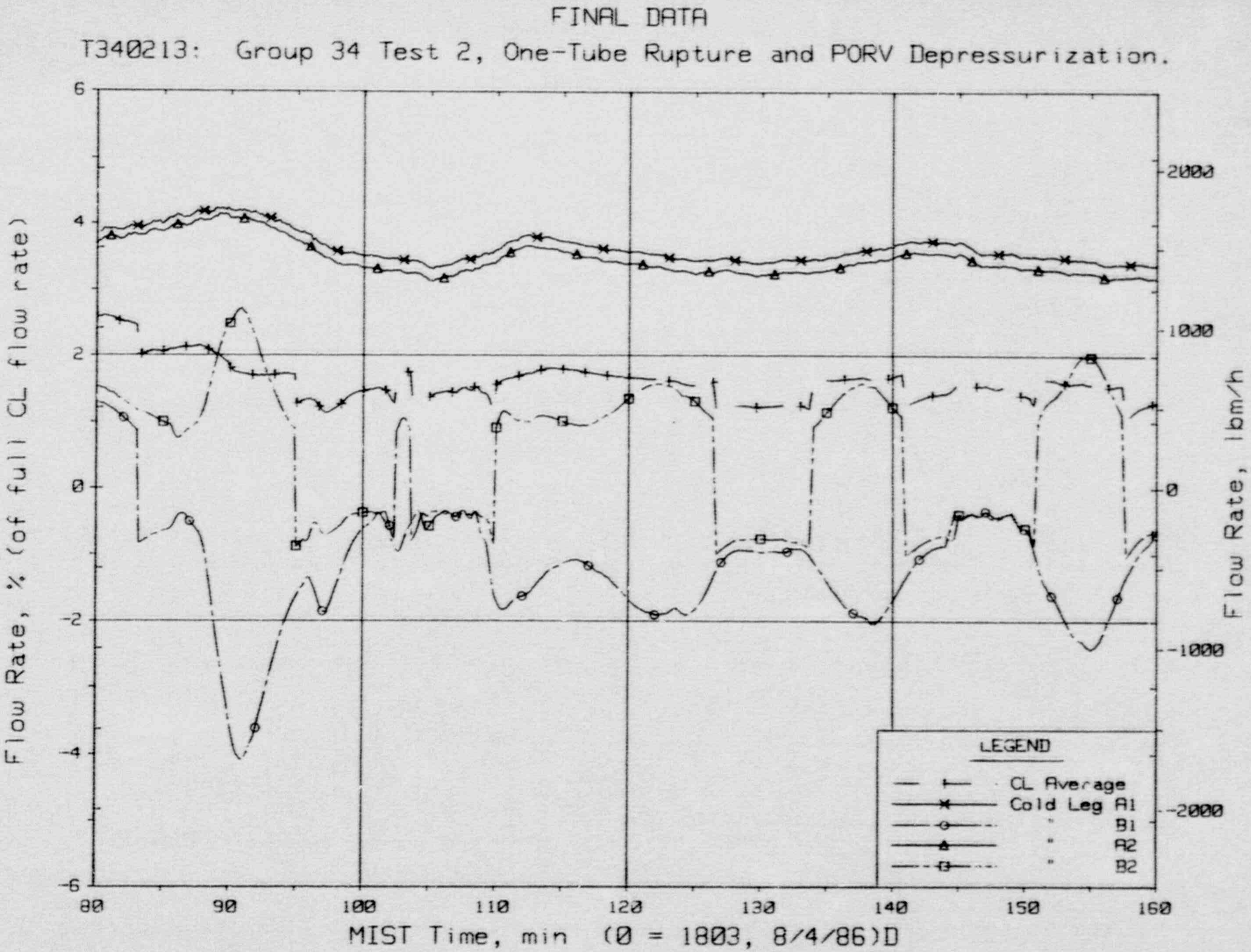
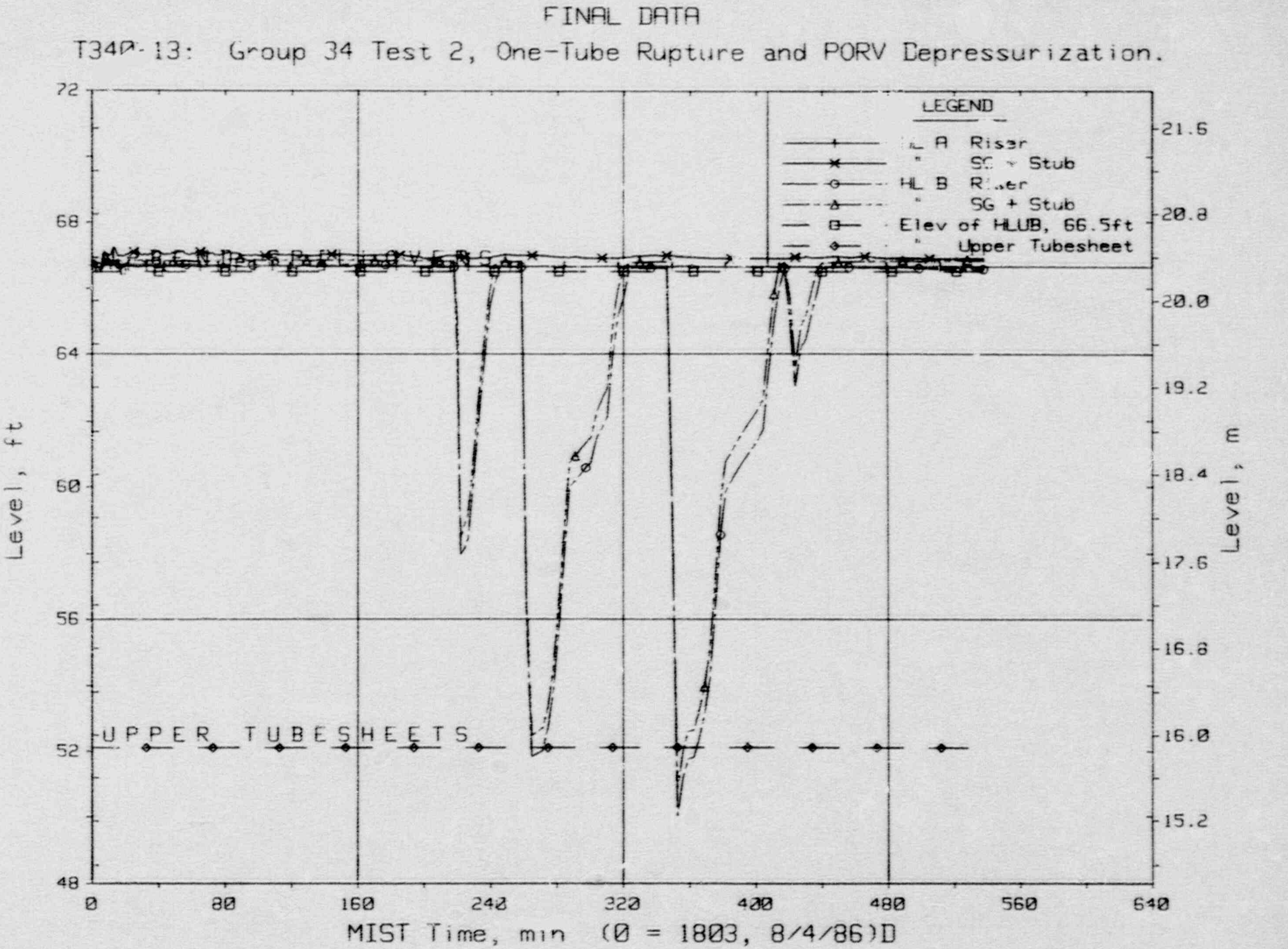


Figure 5.5.8. Cold Leg (Venturi) Flow Rates

Tue Jul 26 14:50:32 1988

CLFL1

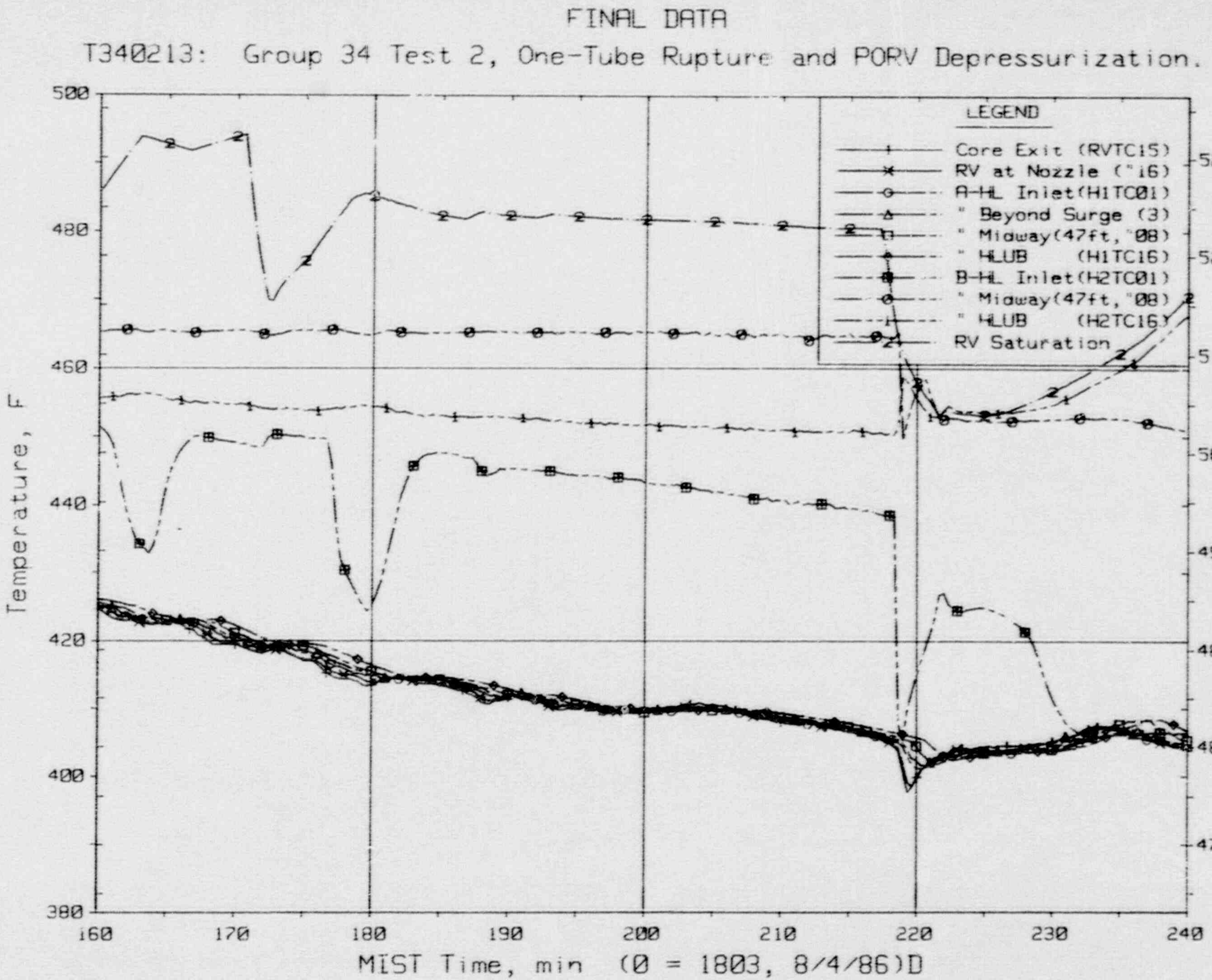


5-163

Figure 5.5.9. Composite Core Exit and Hot Leg Fluid Temperatures

Tue Jul 26 14:44:04 1988

HLLV1



5-164

Figure 5.5.10. Composite Core Exit and Hot Leg Fluid Temperatures

Tue Jul 26 14:58:52 1988

PRKT1

5-165

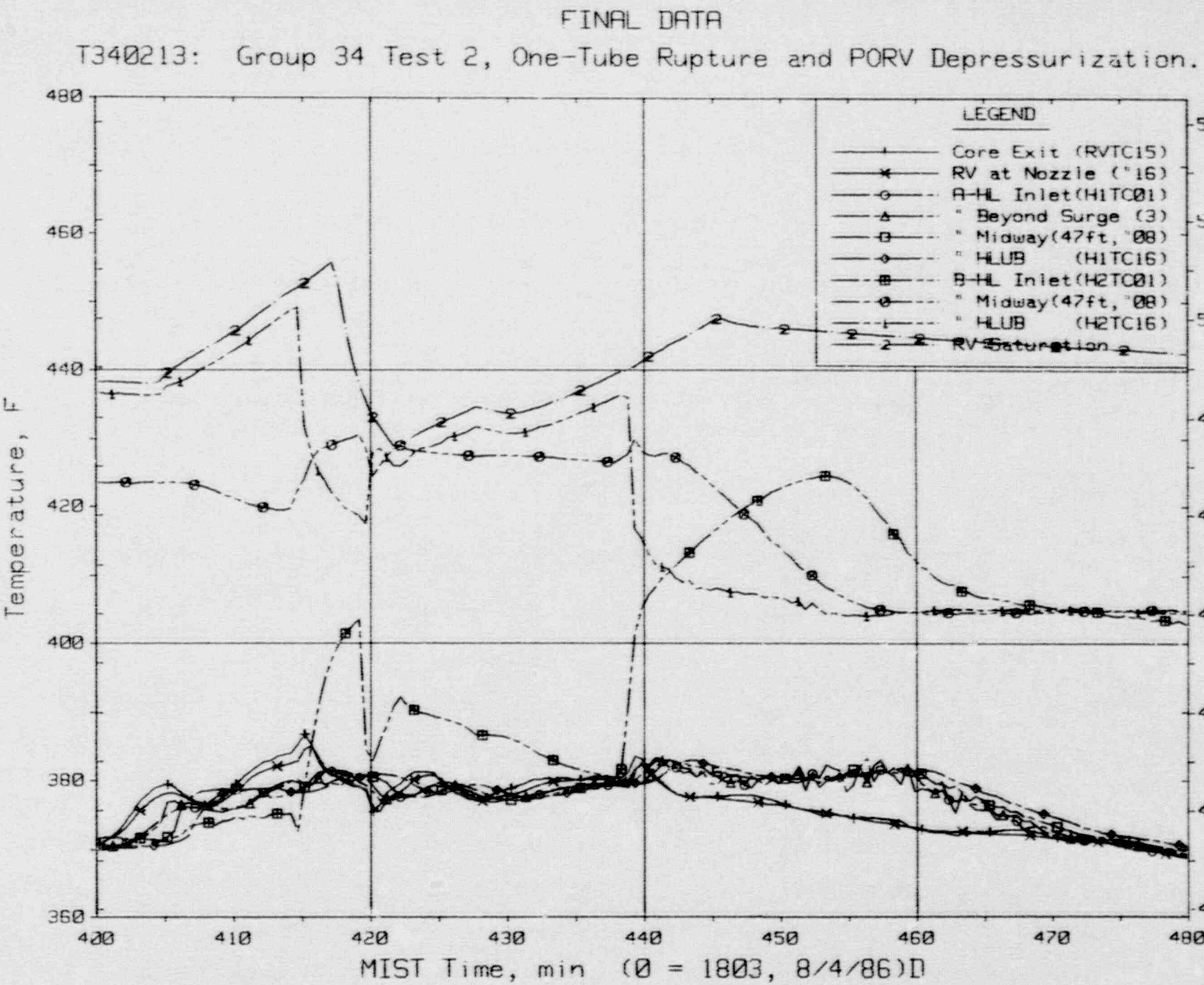


Figure 5.5.11. Composite Core Exit and Hot Leg Fluid Temperatures

Tue Jul 26 15:0?:12 1988 PRKT1

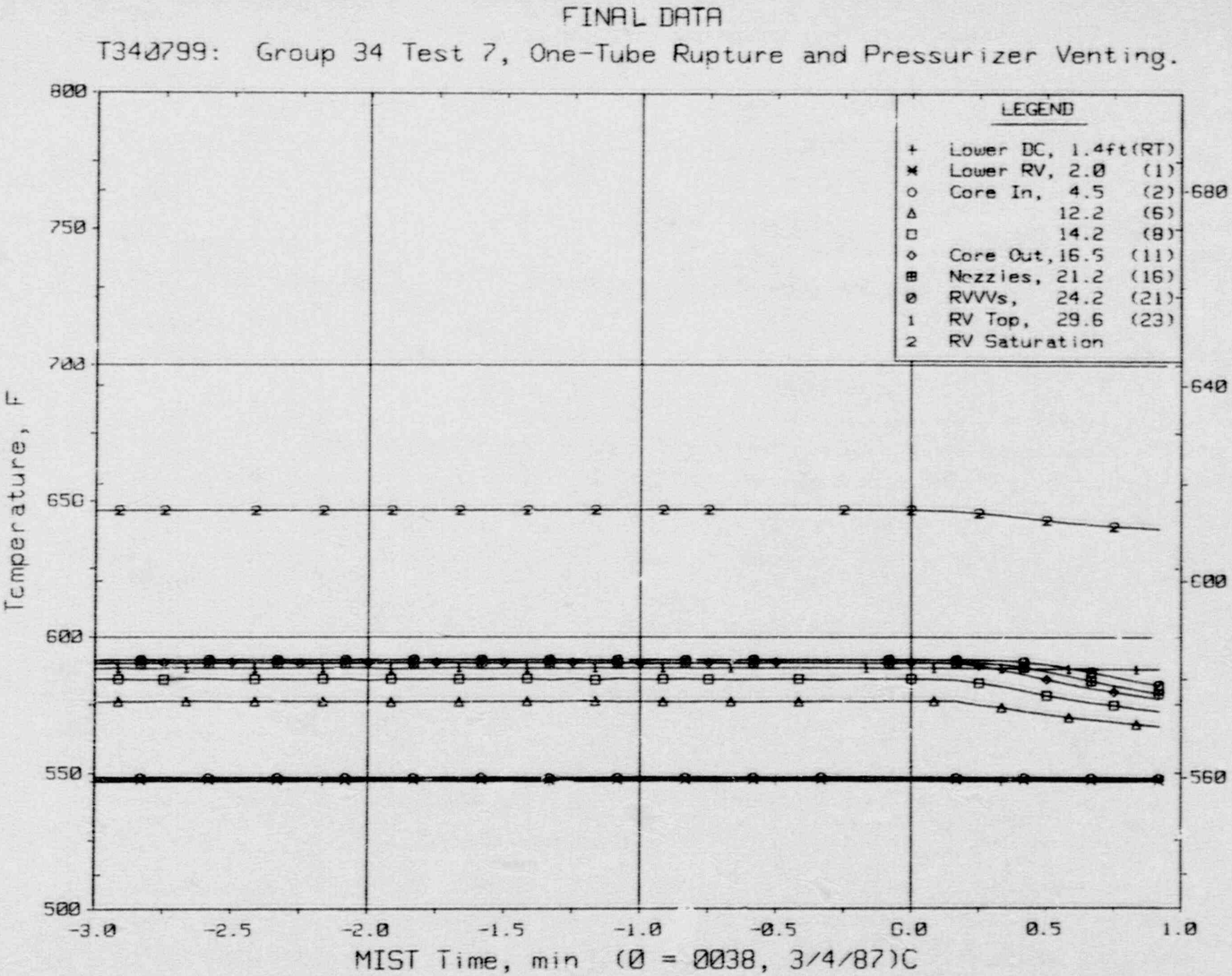


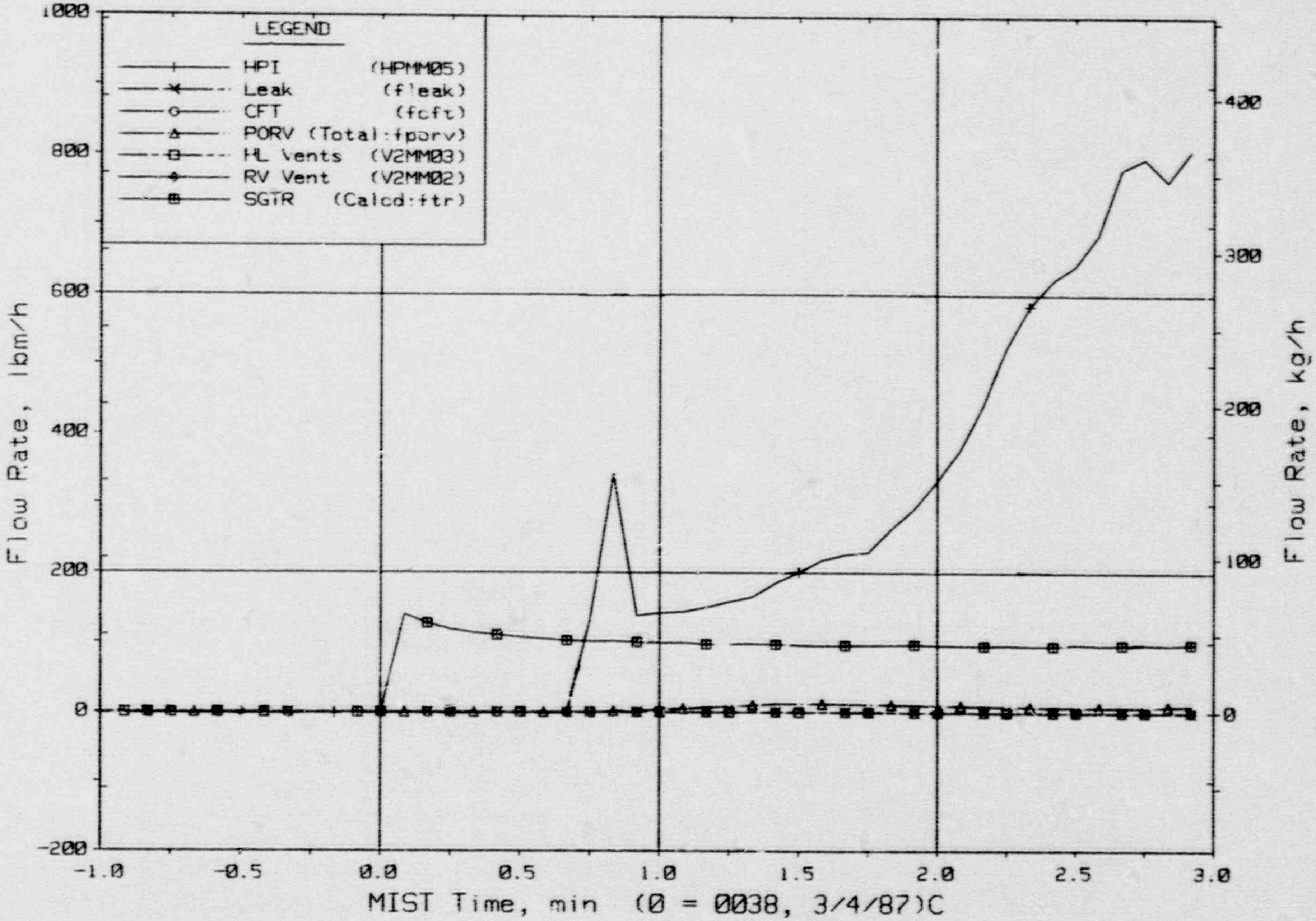
Figure 5.5.12. Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs)

Wed Nov 2 14:51:12 1988

RVTC0

FINAL DATA

T340799: One-Tube Rupture, SG Isolated with Pressurizer Venting



5-167

Figure 5.5.13. Primary System Boundary Flow Rates

Fri Aug 5 12:53:08 1988

PRFL3

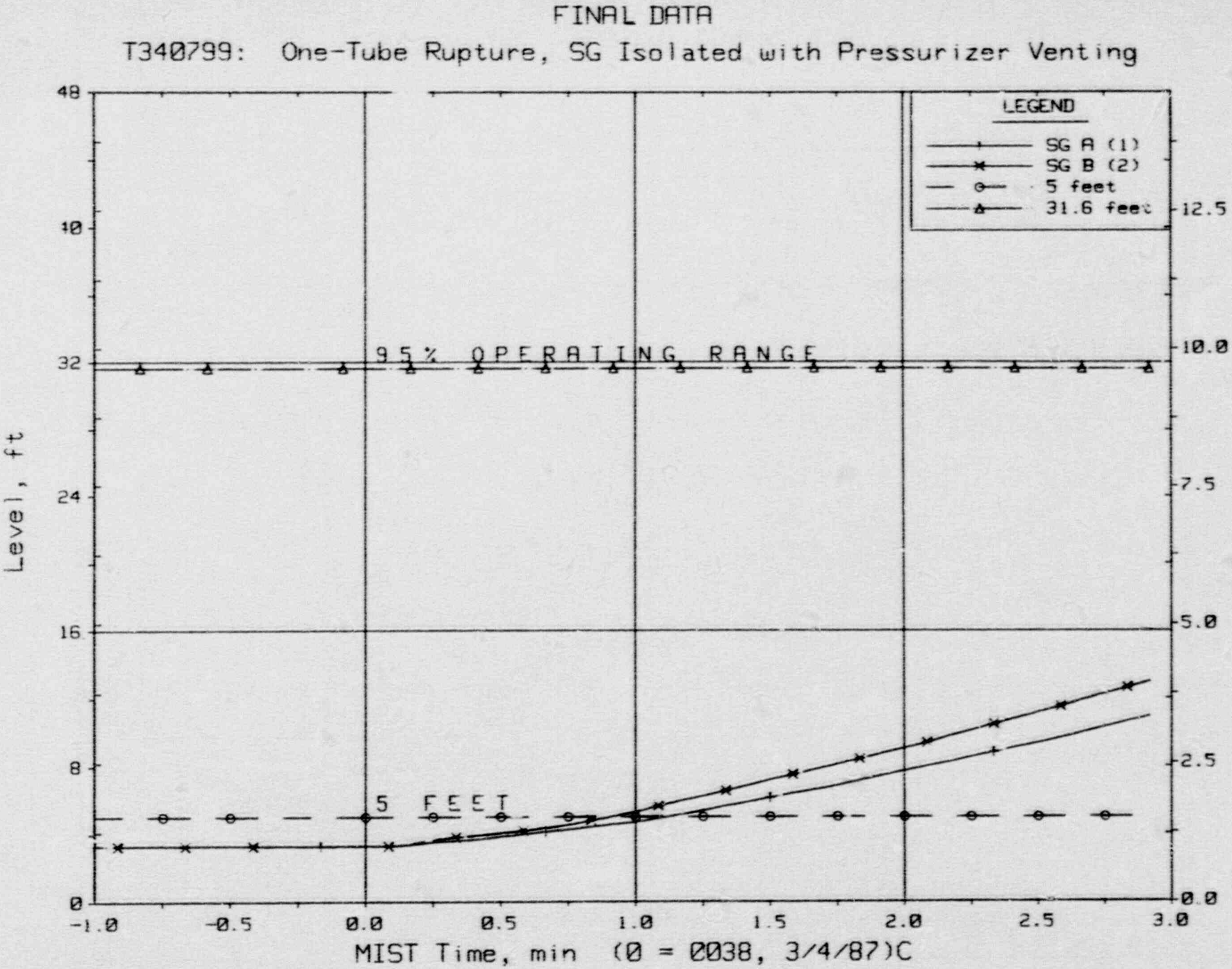


Figure 5.5.14. Steam Generator Secondary Collapsed Liquid Levels (SnLV20s)

Fri Aug 5 12:55:52 1988

SCLV1

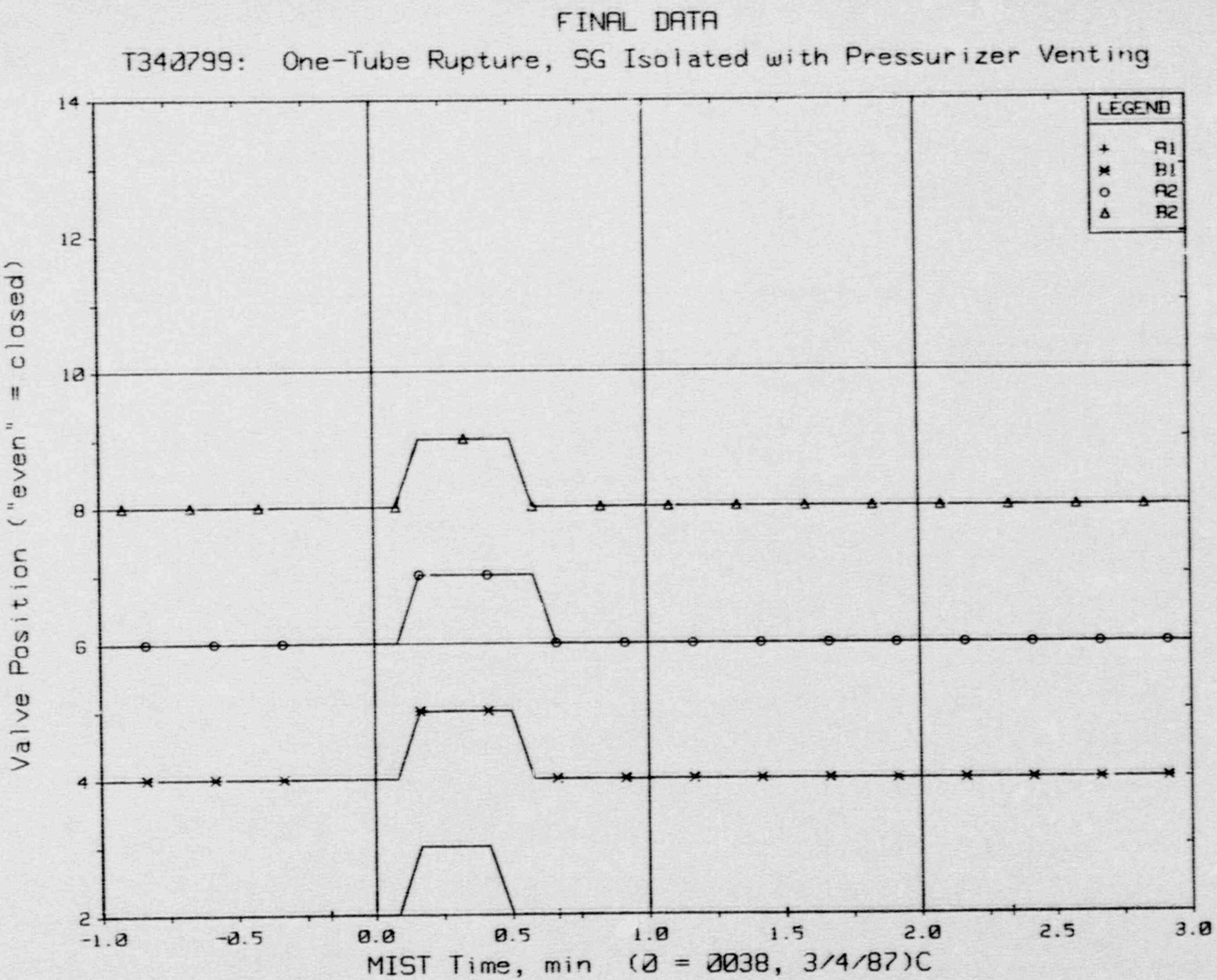


Figure 5.5.15. Reactor Vessel Vent Valve Positions

VVLS1

Fri Aug 5 12:58:02 1988

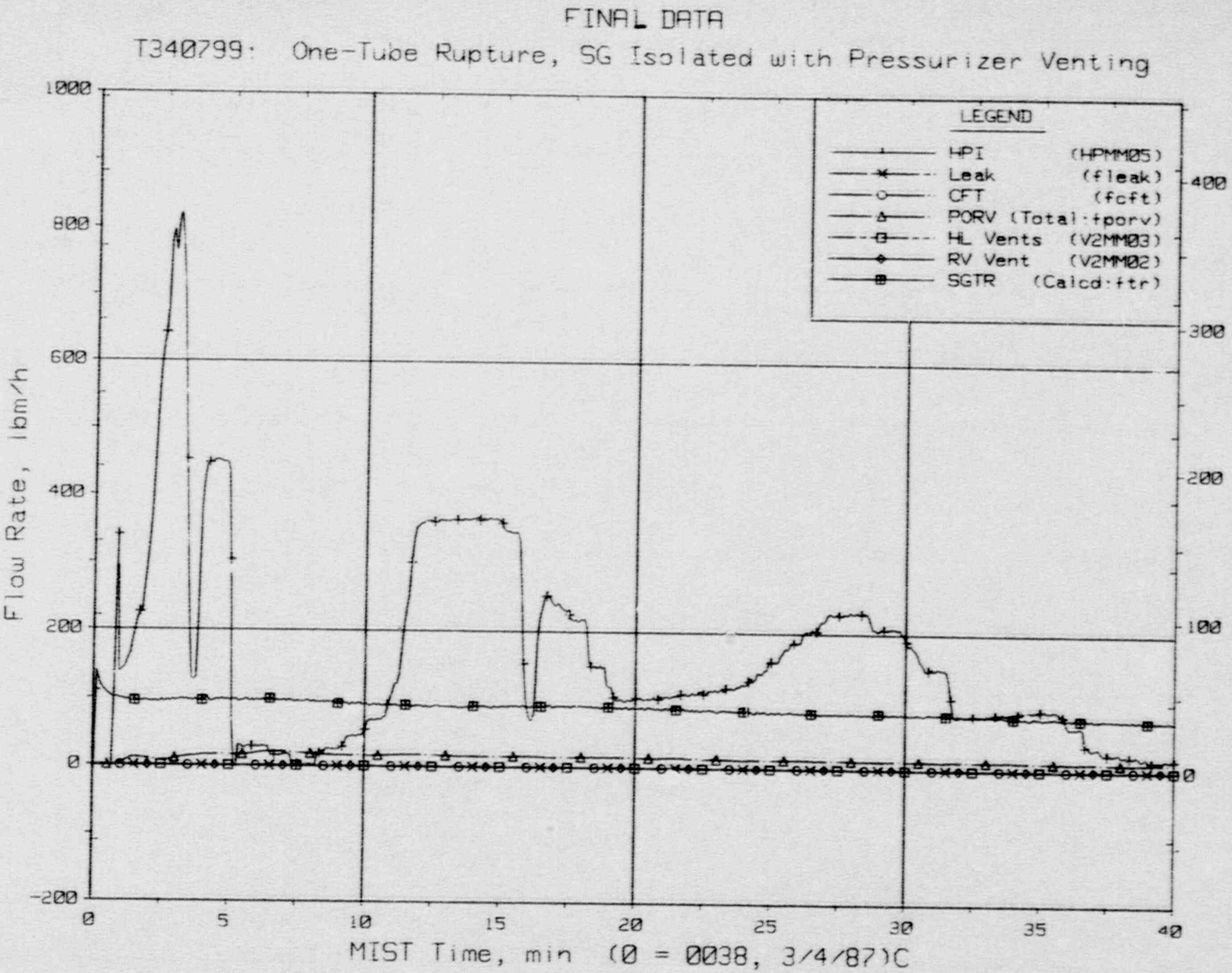


Figure 5.5.16. Primary System Boundary Flow Rates

Fri Aug 5 10:28:09 1988

PRFL3

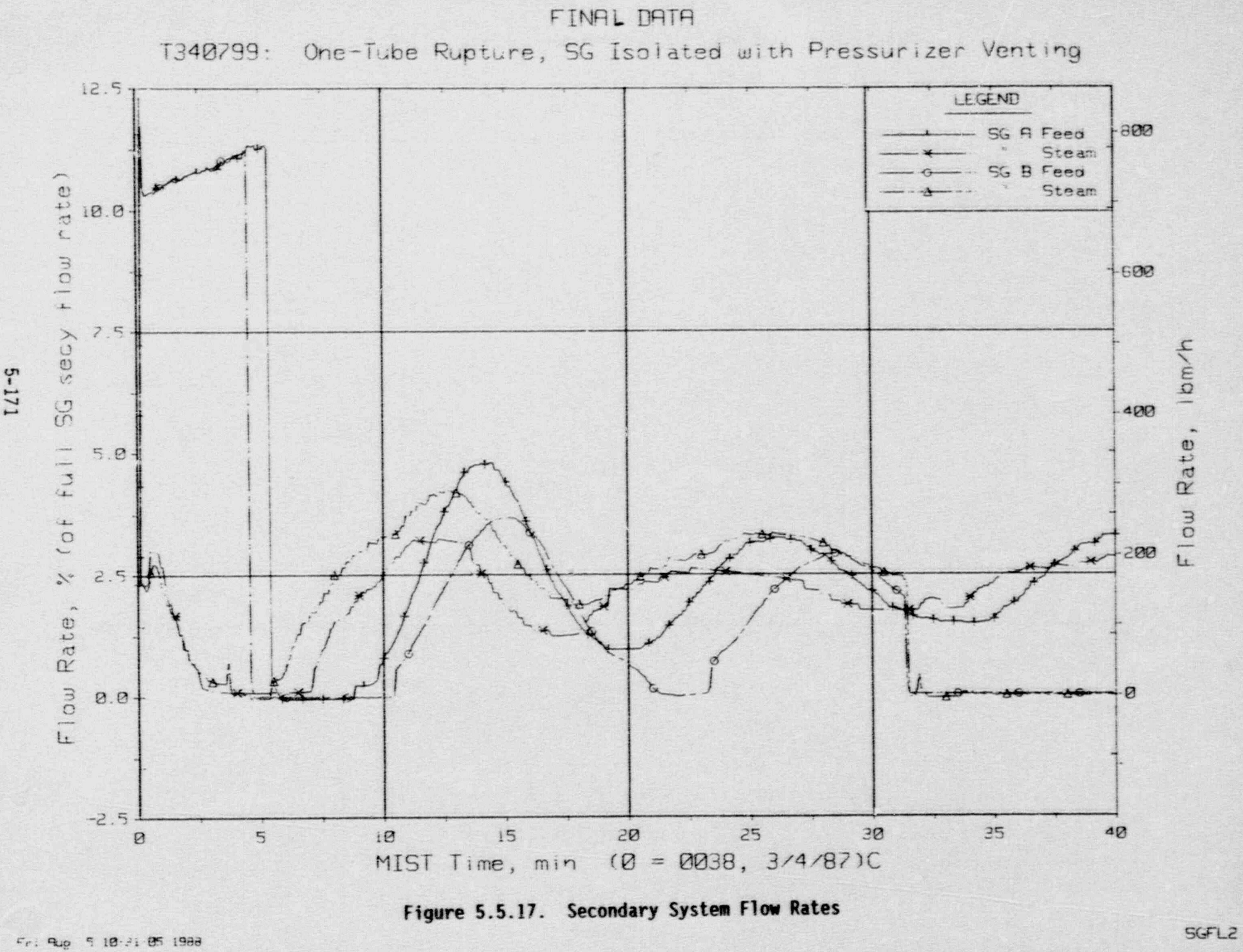


Figure 5.5.17. Secondary System Flow Rates

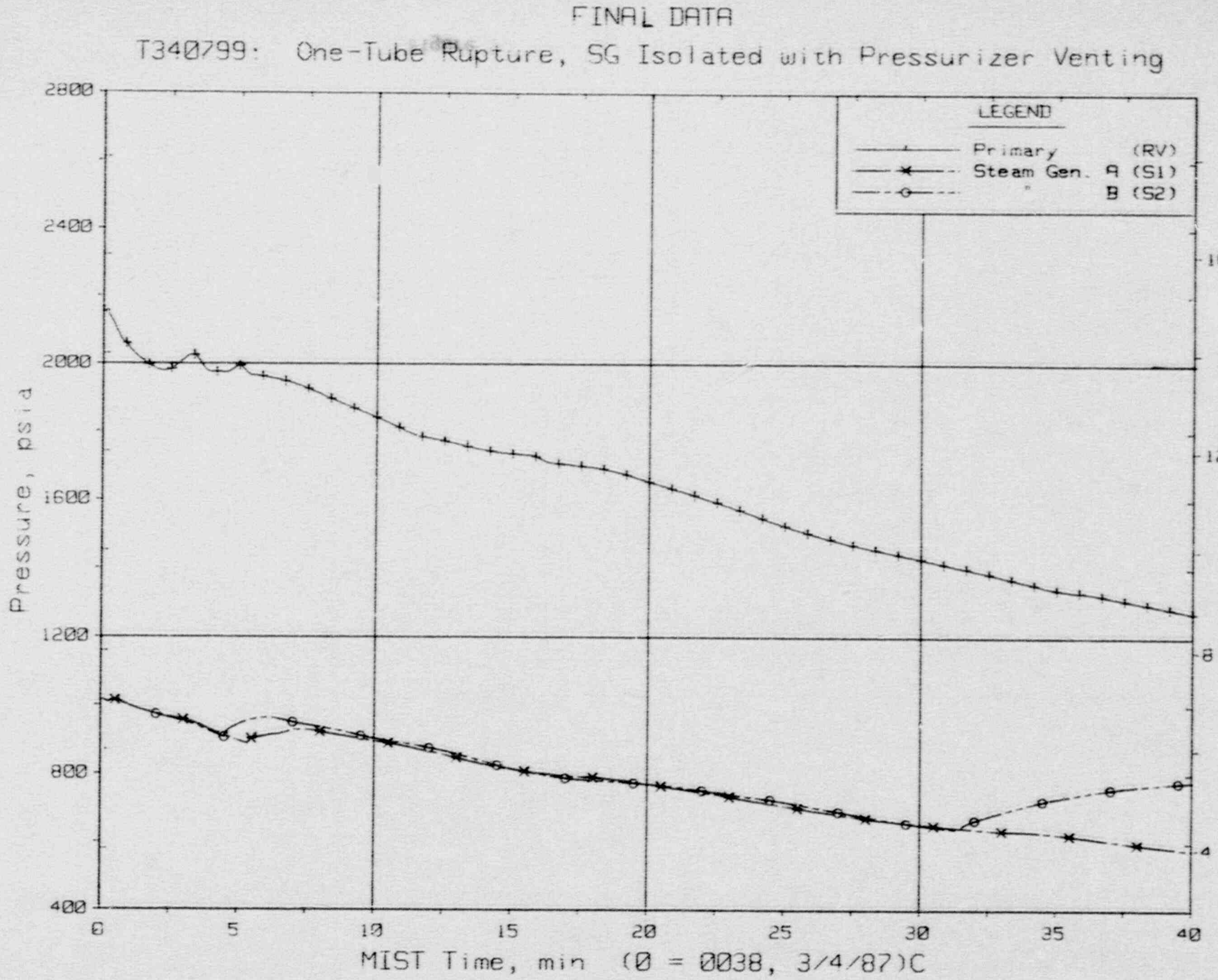


Figure 5.5.18. Primary and Secondary System Pressures (GPO's)

Fri Aug 5 10:33:49 1988

PSGP1

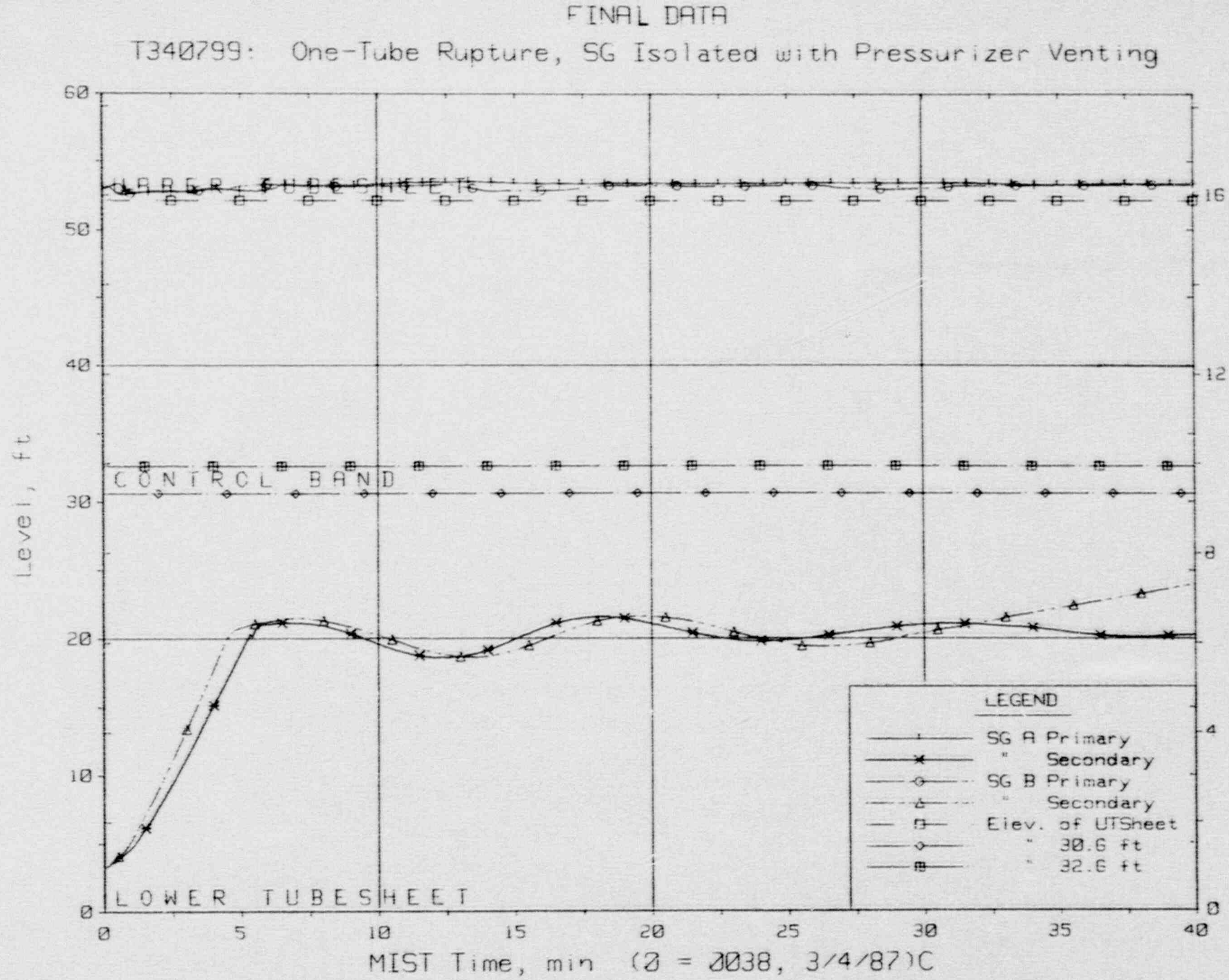


Figure 5.5.19. Steam Generator Collapsed Liquid Levels

Fri Aug 5 10:28:12 1988

SGLV2

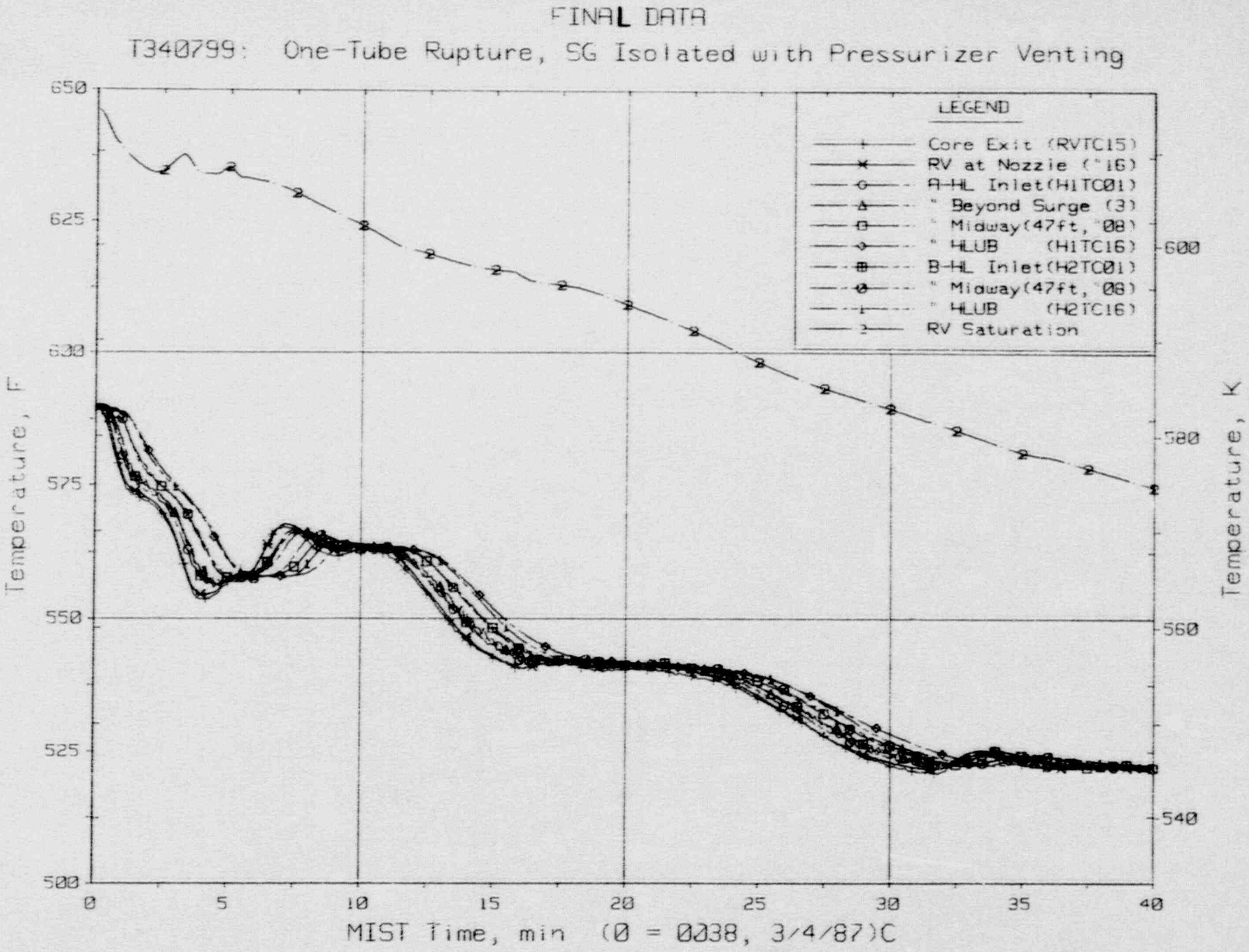


Figure 5.5.20. Composite Core Exit and Hot Leg Fluid Temperatures

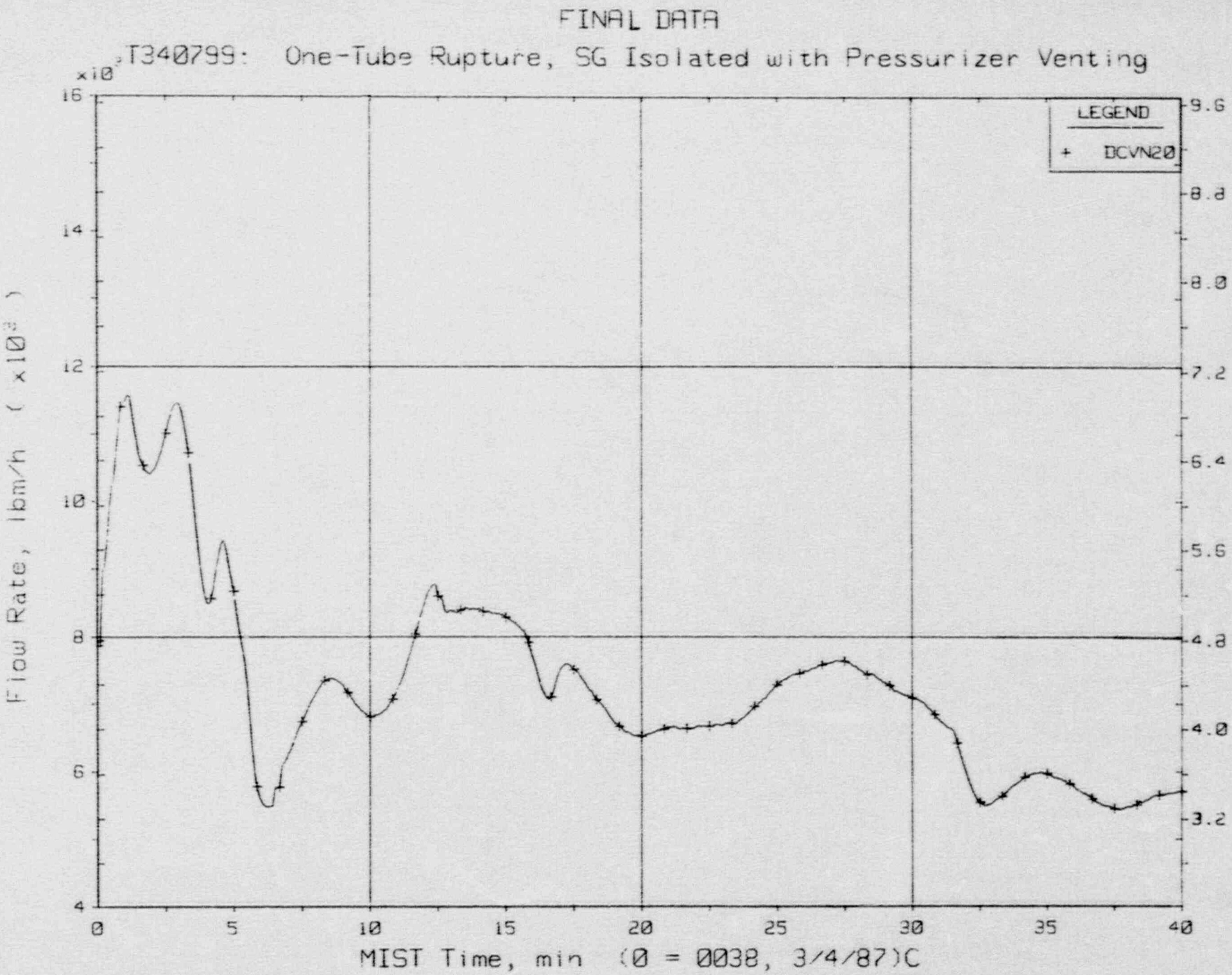


Figure 5.5.21. Primary System Venturi Flow Rates

Fri Aug 5 10:45:02 1988

DCFL1

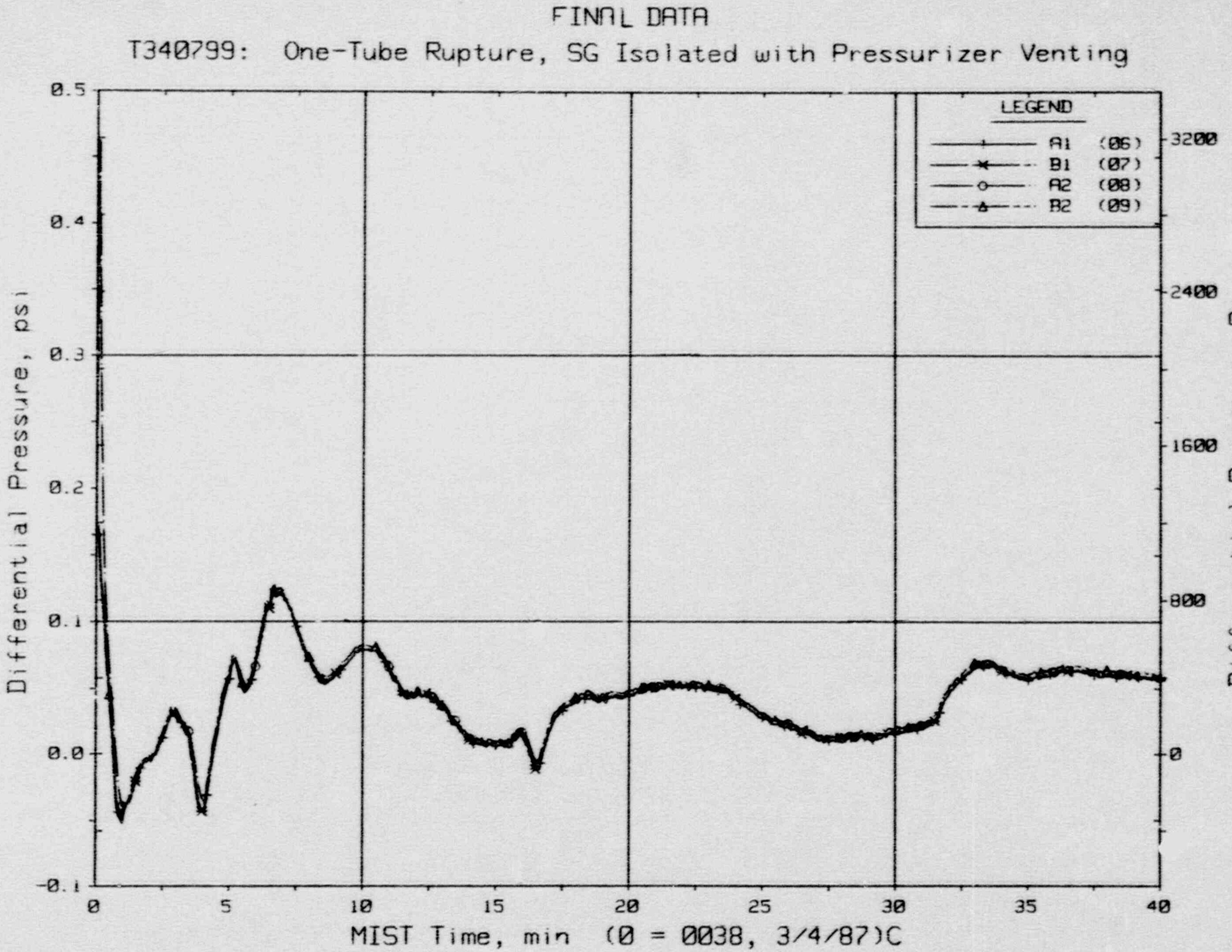


Figure 5.5.22. Reactor Vessel Vent Valve Differential Pressures (RVDPs)

Fri Aug 5 10:48:20 1988

VVDP1

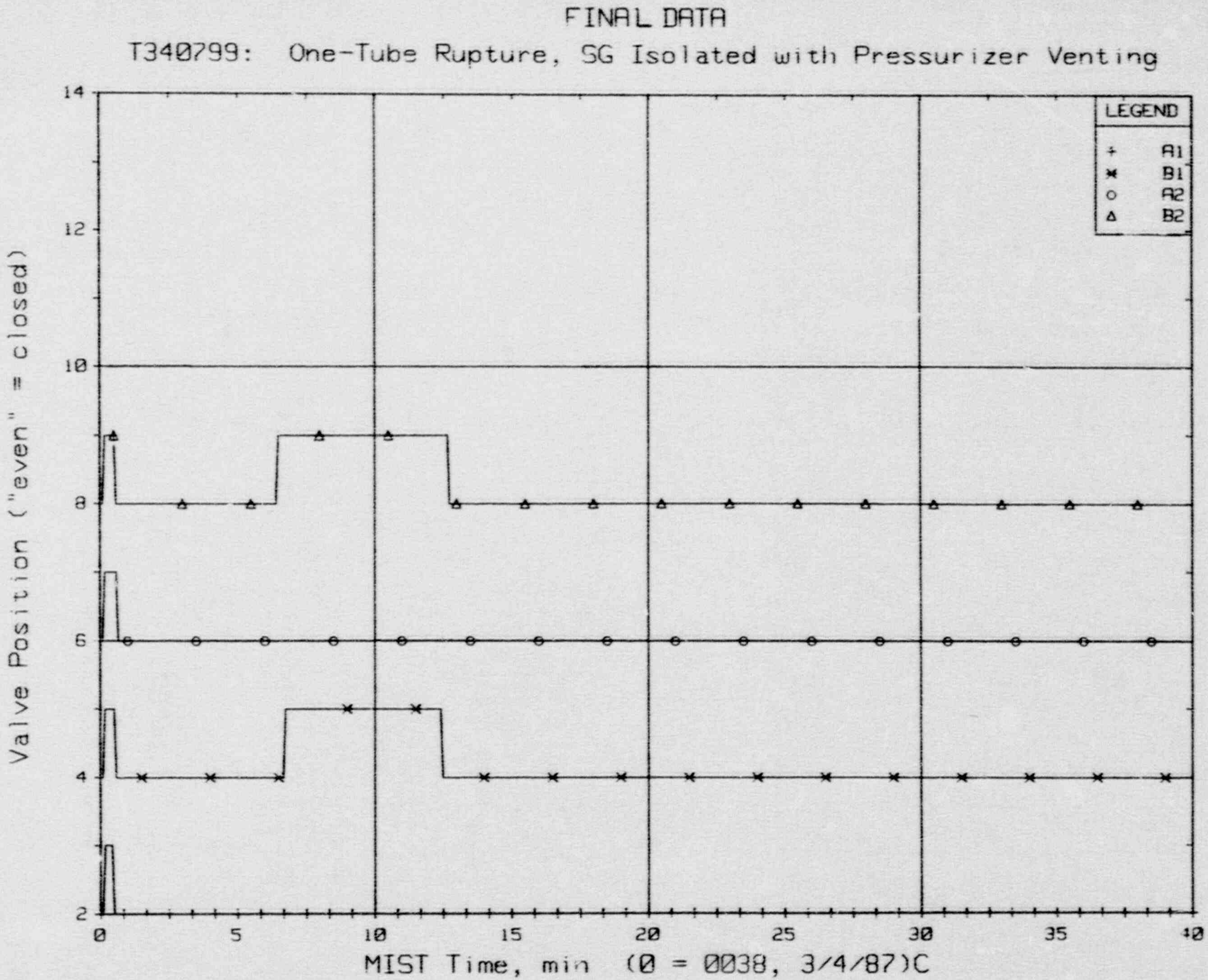


Figure 5.5.23. Reactor Vessel Vent Valve Positions

Fri Aug 5 10:51:04 1988

VVLS1

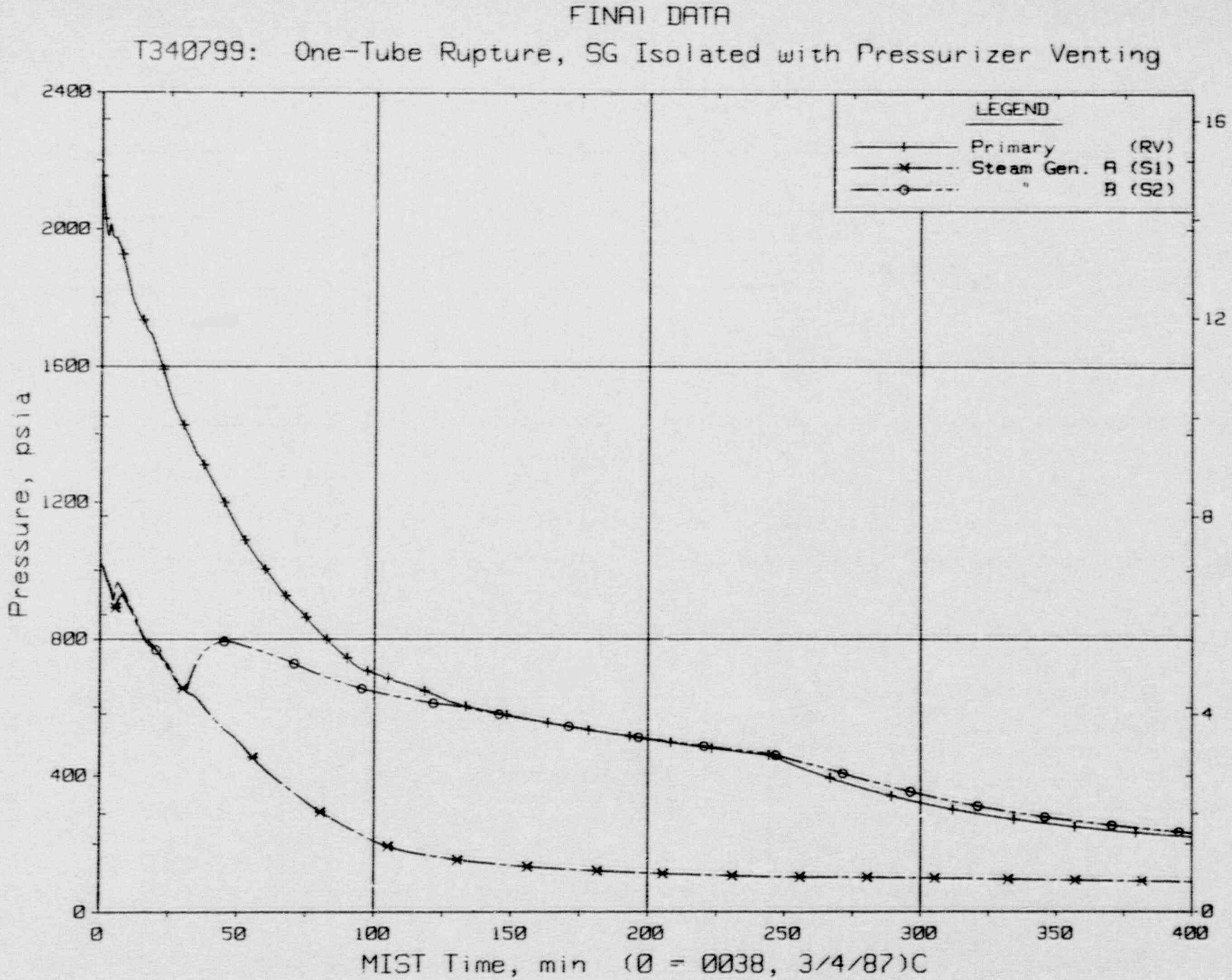


Figure 5.5.24. Primary and Secondary System Pressures (GP01s)

Thu Jul 28 12:38:08 1988

PSGP1

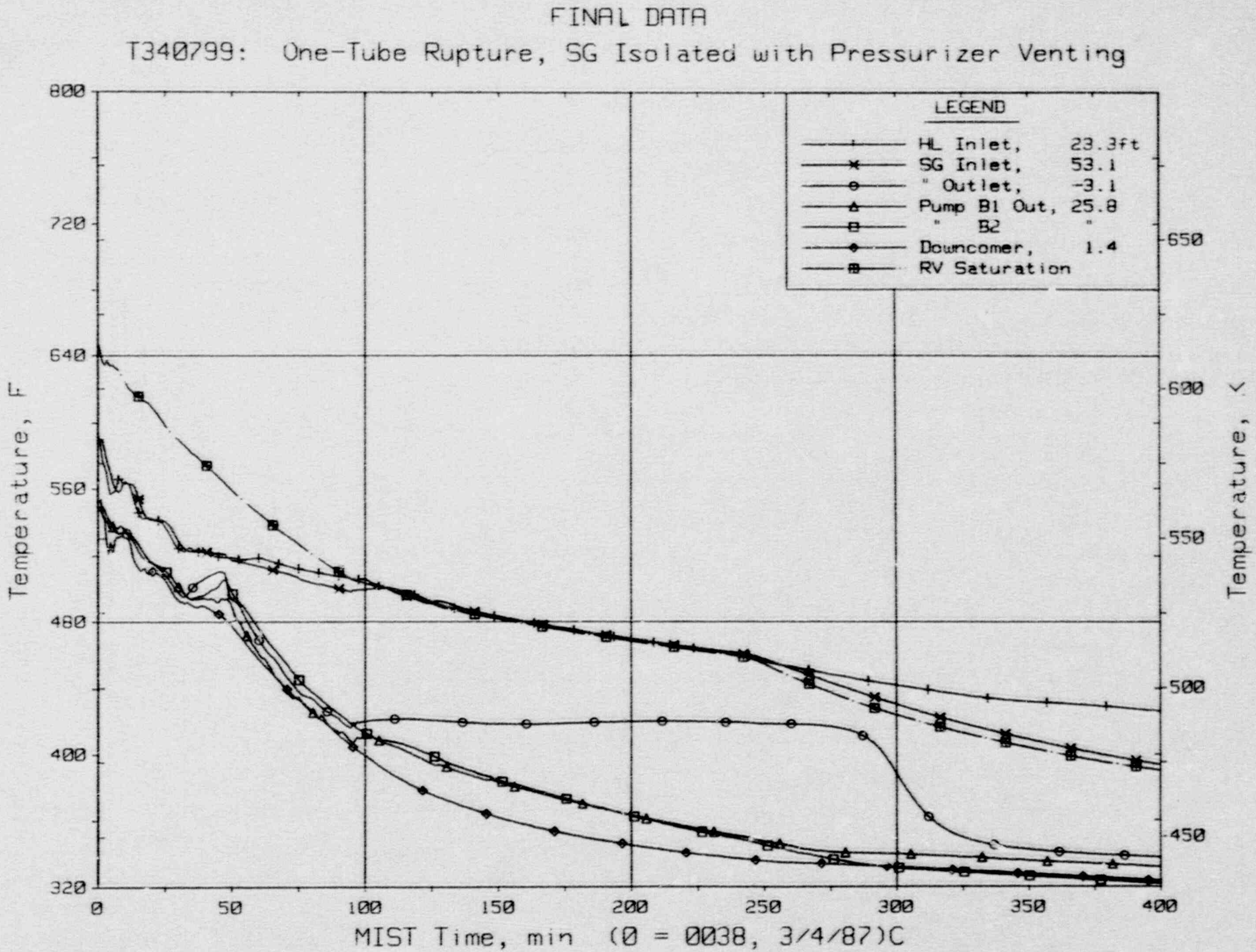


Figure 5.5.25. Loop B Primary Fluid Temperatures (RTDs)

Thu Jul 28 12-42-32 1988

PRRT2

FINAl DATA

T340799: One-Tube Rupture, SG Isolated with Pressurizer Venting

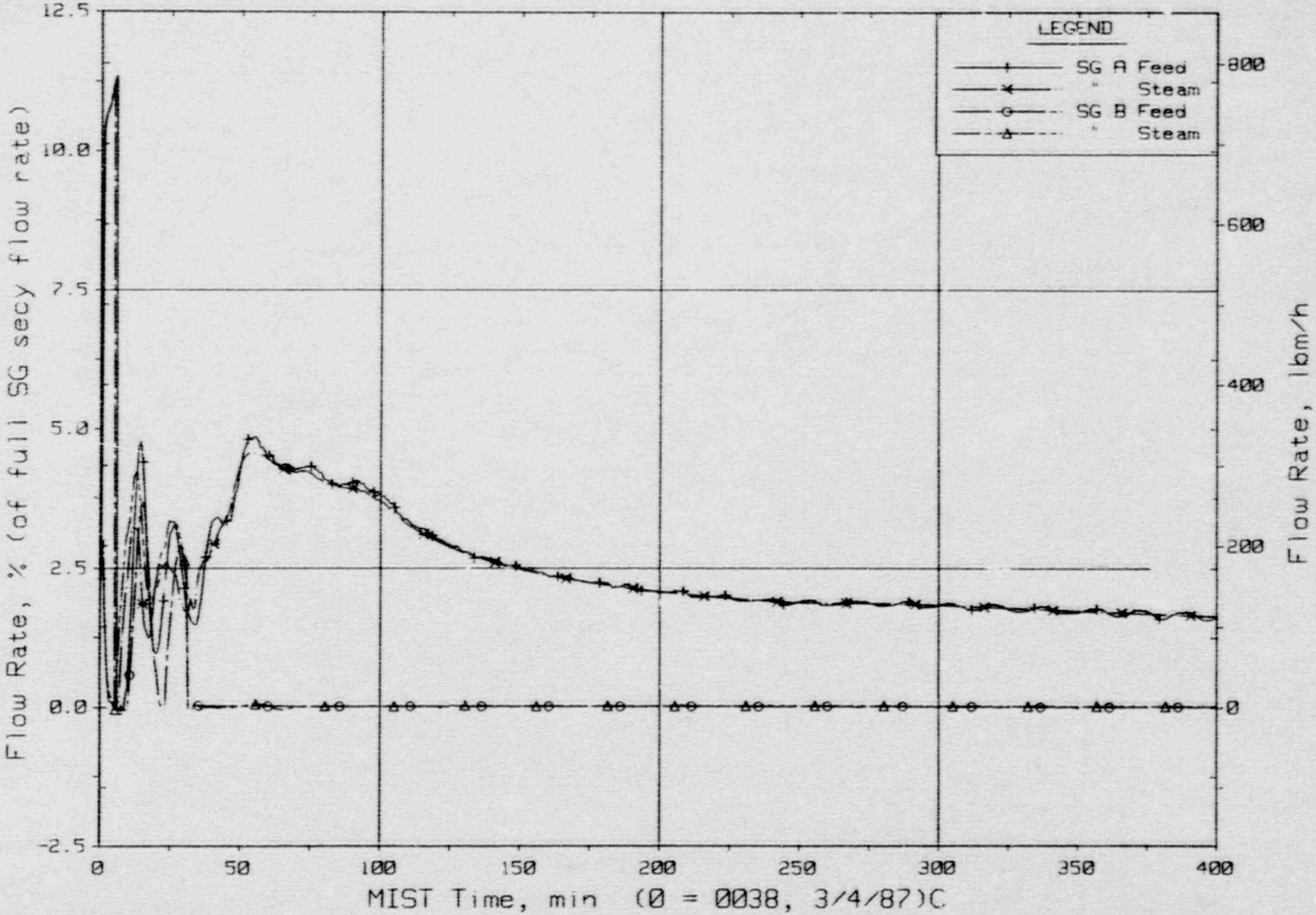


Figure 5.5.26. Secondary System Flow Rates

Thu Jul 28 12:45:20 1988

SGFL2

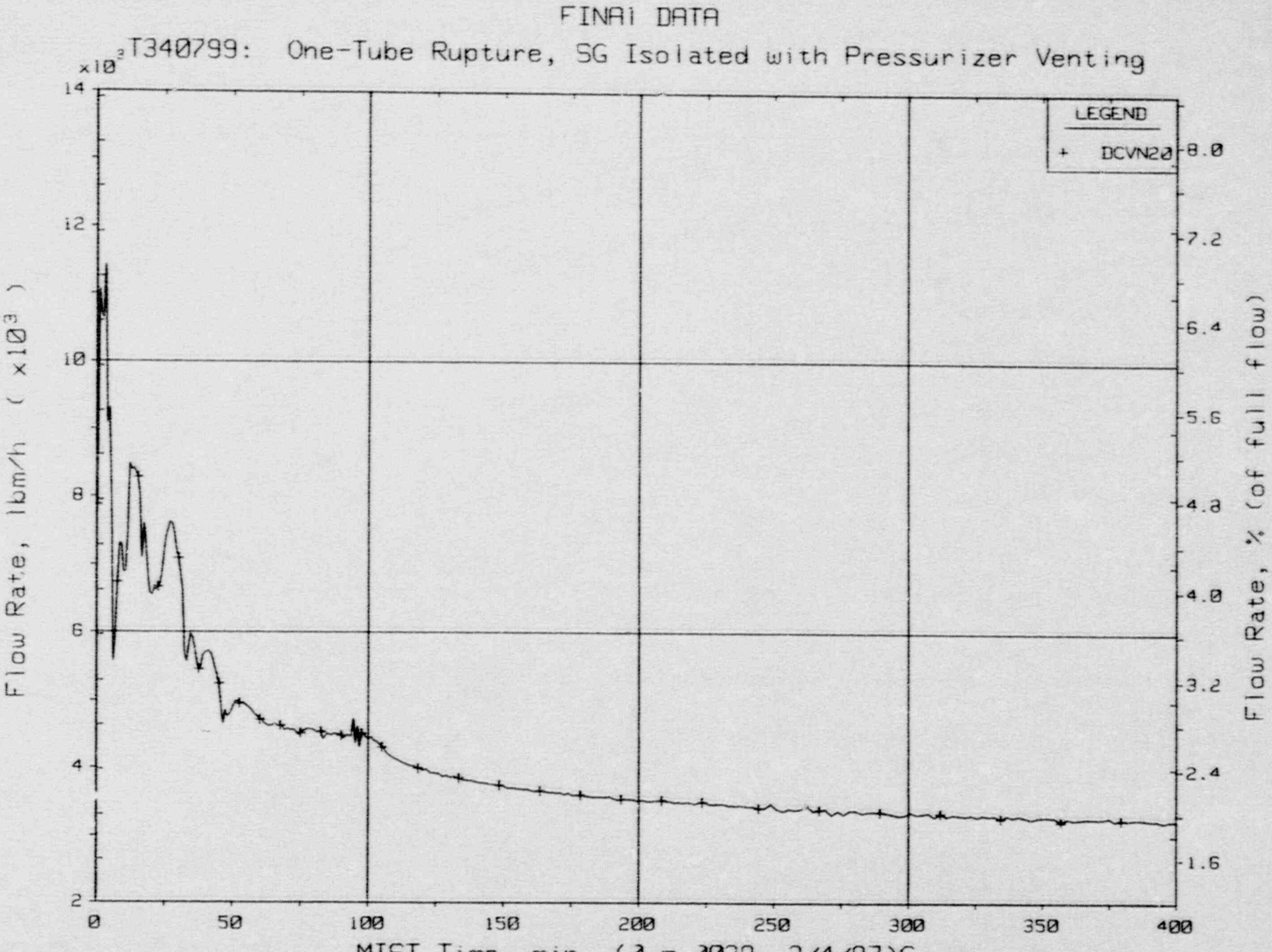


Figure 5.5.27. Primary System Venturi Flow Rates

Thu Jul 28 12:47:10 1988

DCFLi

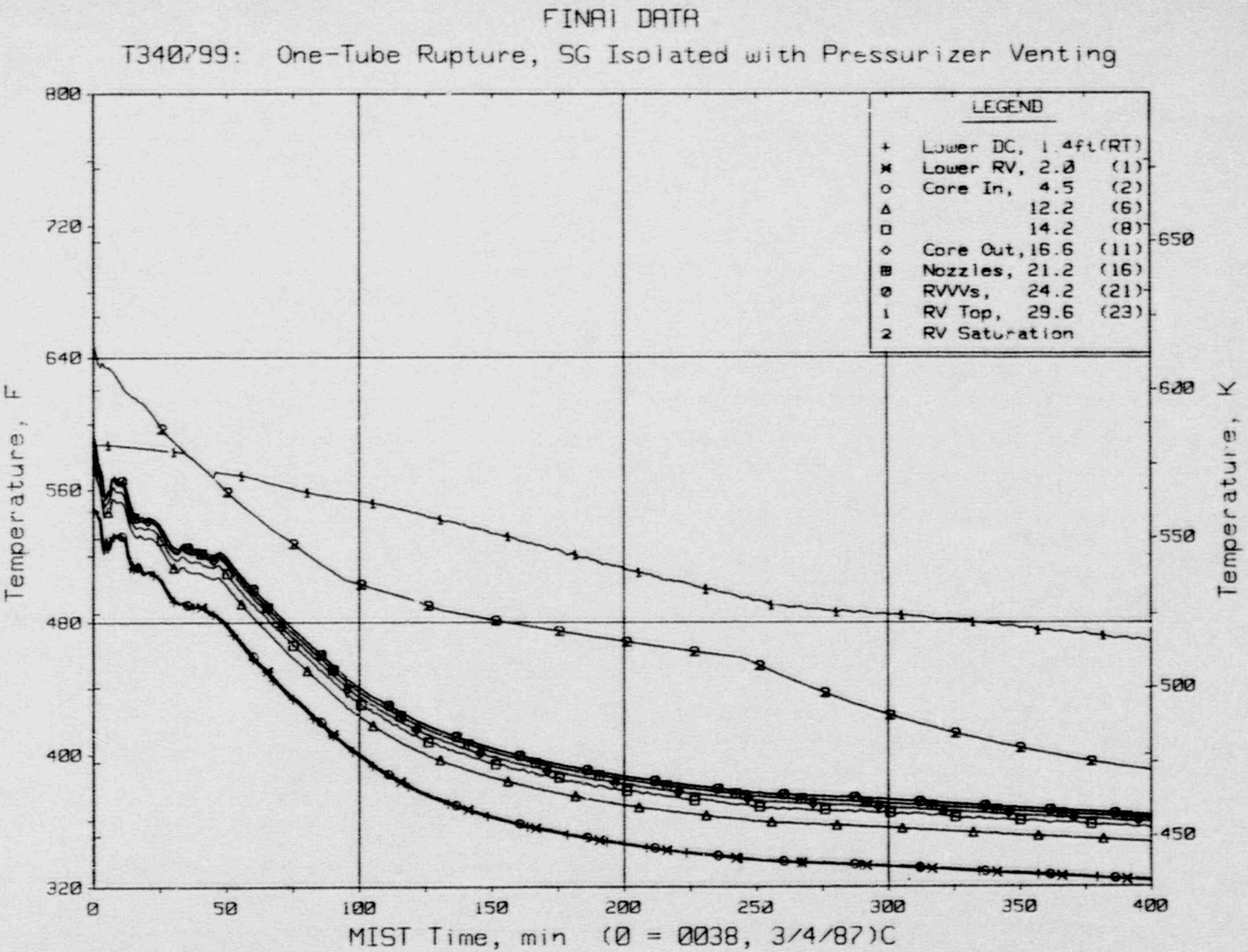


Figure 5.5.28. Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs)

Thu Jul 28 12:50:24 1988

RVTC0

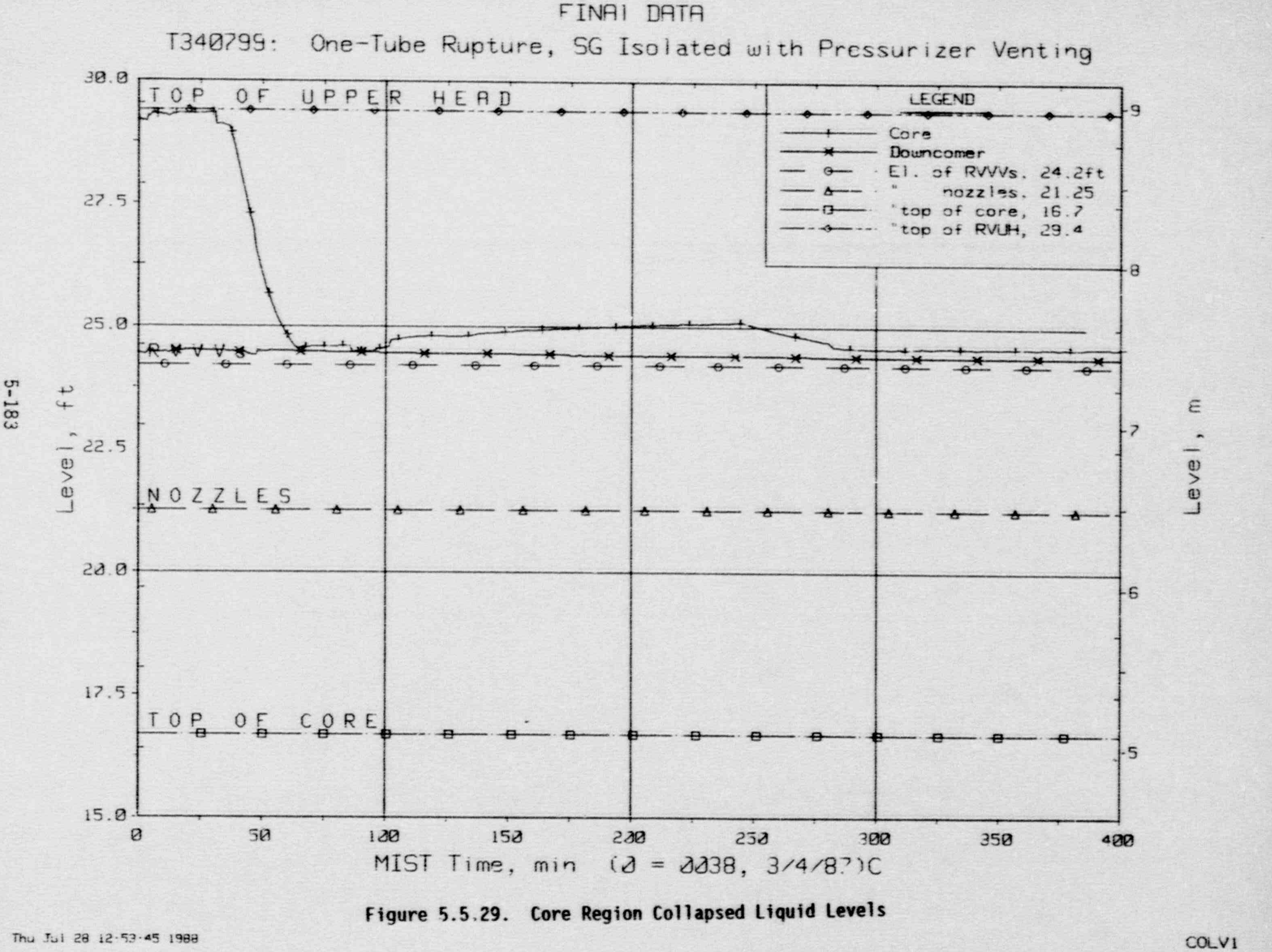


Figure 5.5.29. Core Region Collapsed Liquid Levels

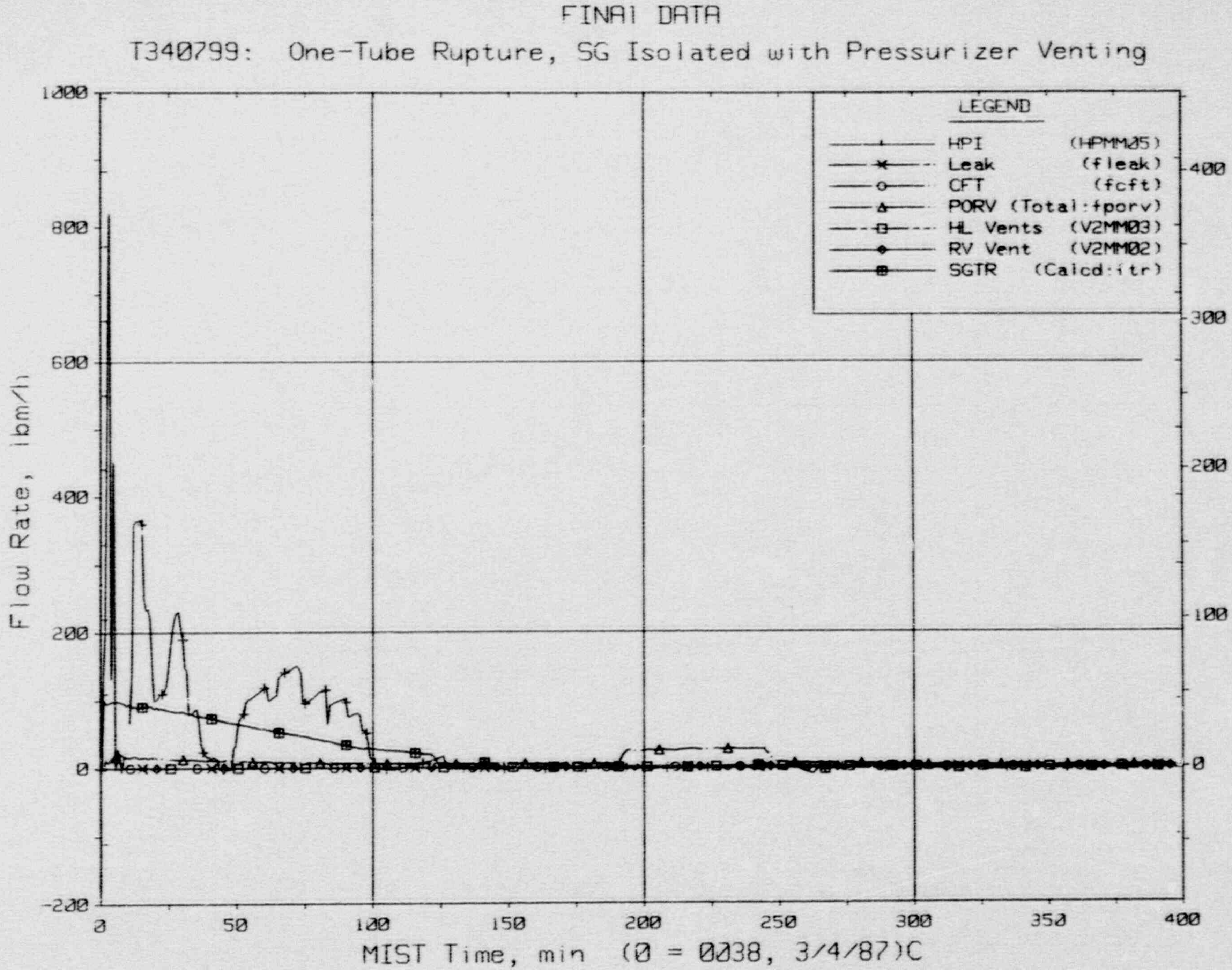
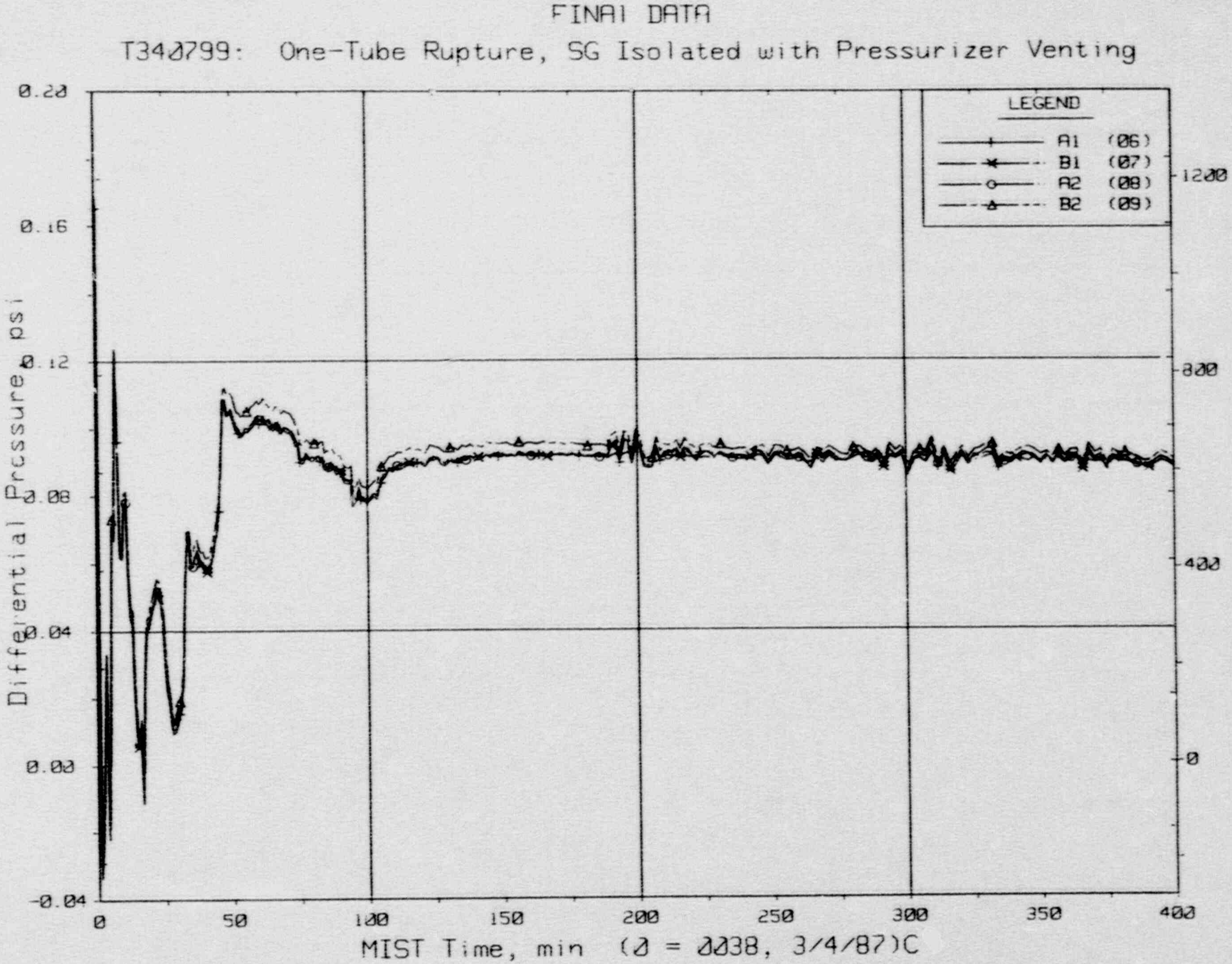


Figure 5.5.30. Primary System Boundary Flow Rates

Thu Jul 28 12:57:05 1988

PRFL3



5-185

Figure 5.5.31. Reactor Vessel Vent Valve Differential Pressures (RVDPs)

Thu Jul 28 13:00:00 1988

VVDP1

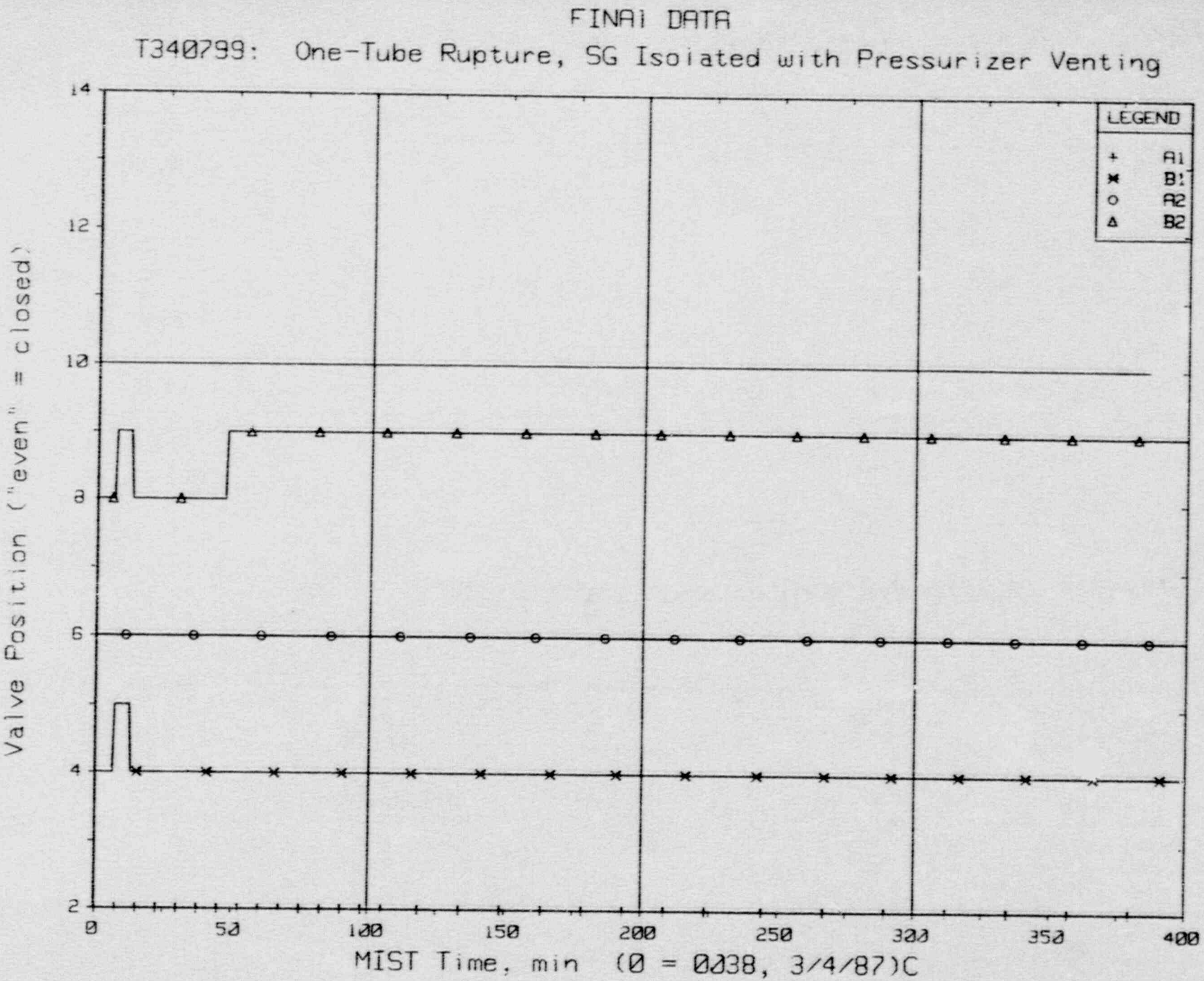


Figure 5.5.32. Reactor Vessel Vent Valve Positions

Thu Jul 28 13:02:46 1988

VVLS1

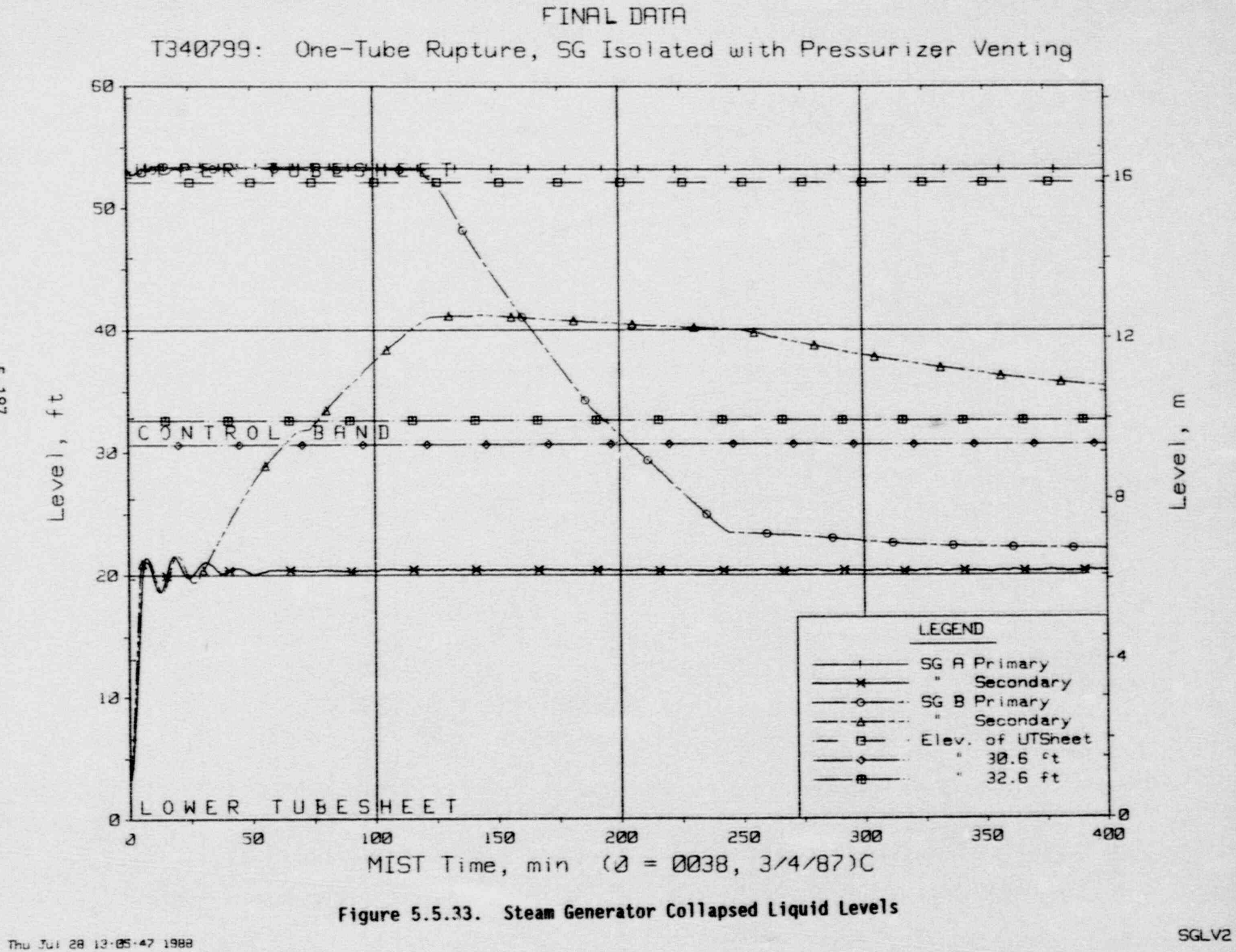


Figure 5.5.33. Steam Generator Collapsed Liquid Levels

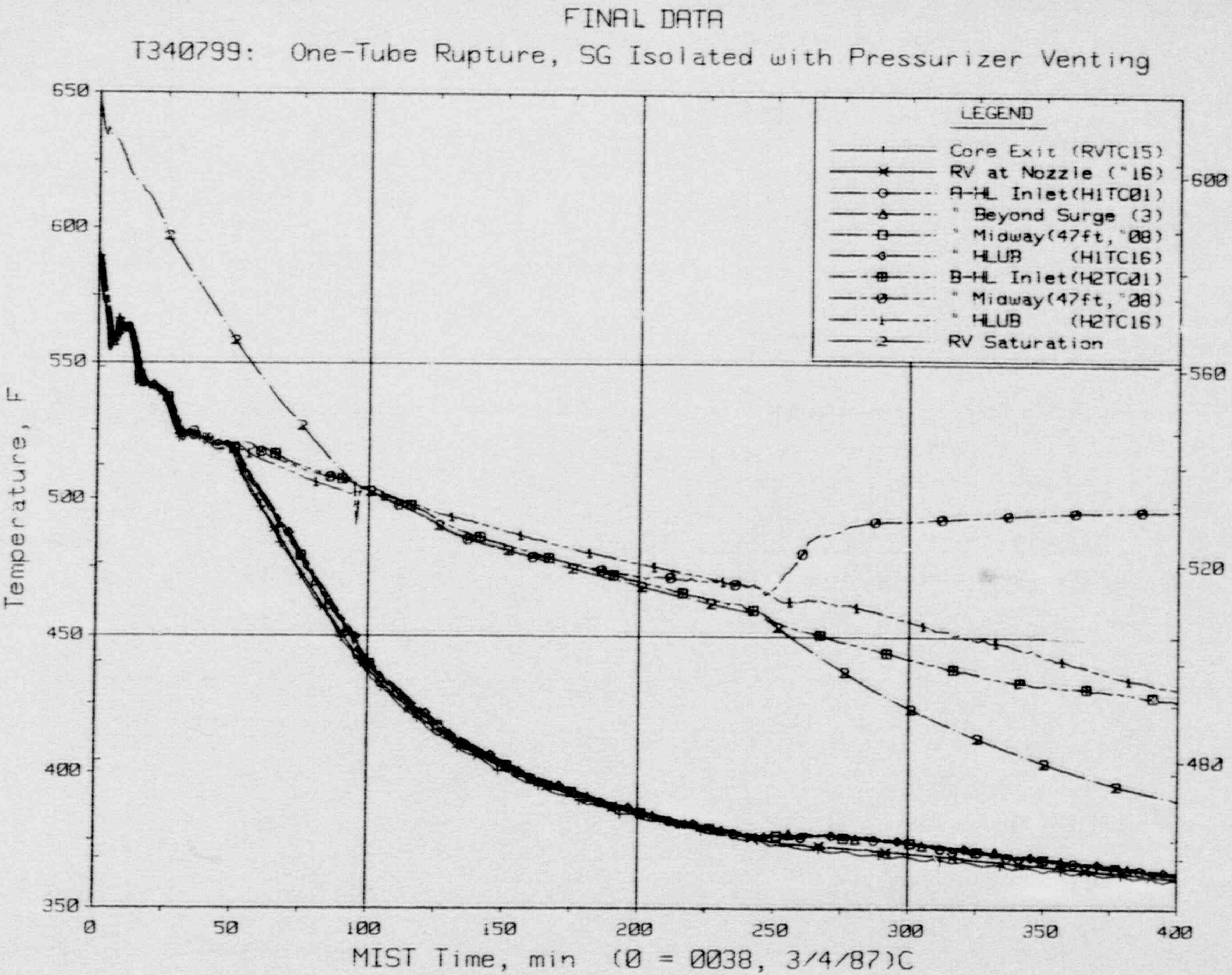


Figure 5.5.34. Composite Core Exit and Hot Leg Fluid Temperatures

Thu Jul 28 13:09:53 1988

PRKT1

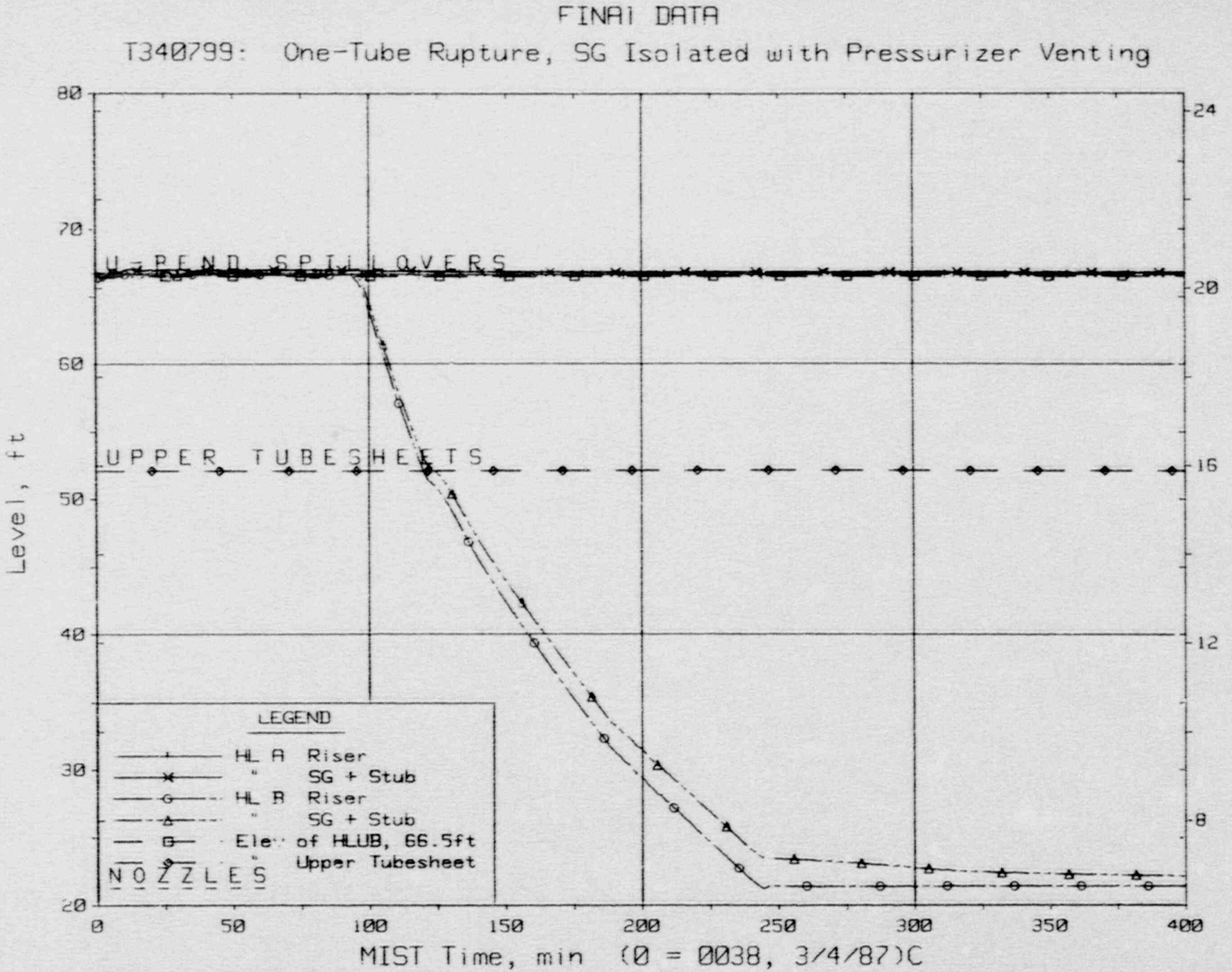


Figure 5.5.35. Hot Leg Riser and Stub Collapsed Liquid Levels

Thu Jul 28 13:13:27 1988

HLLV1

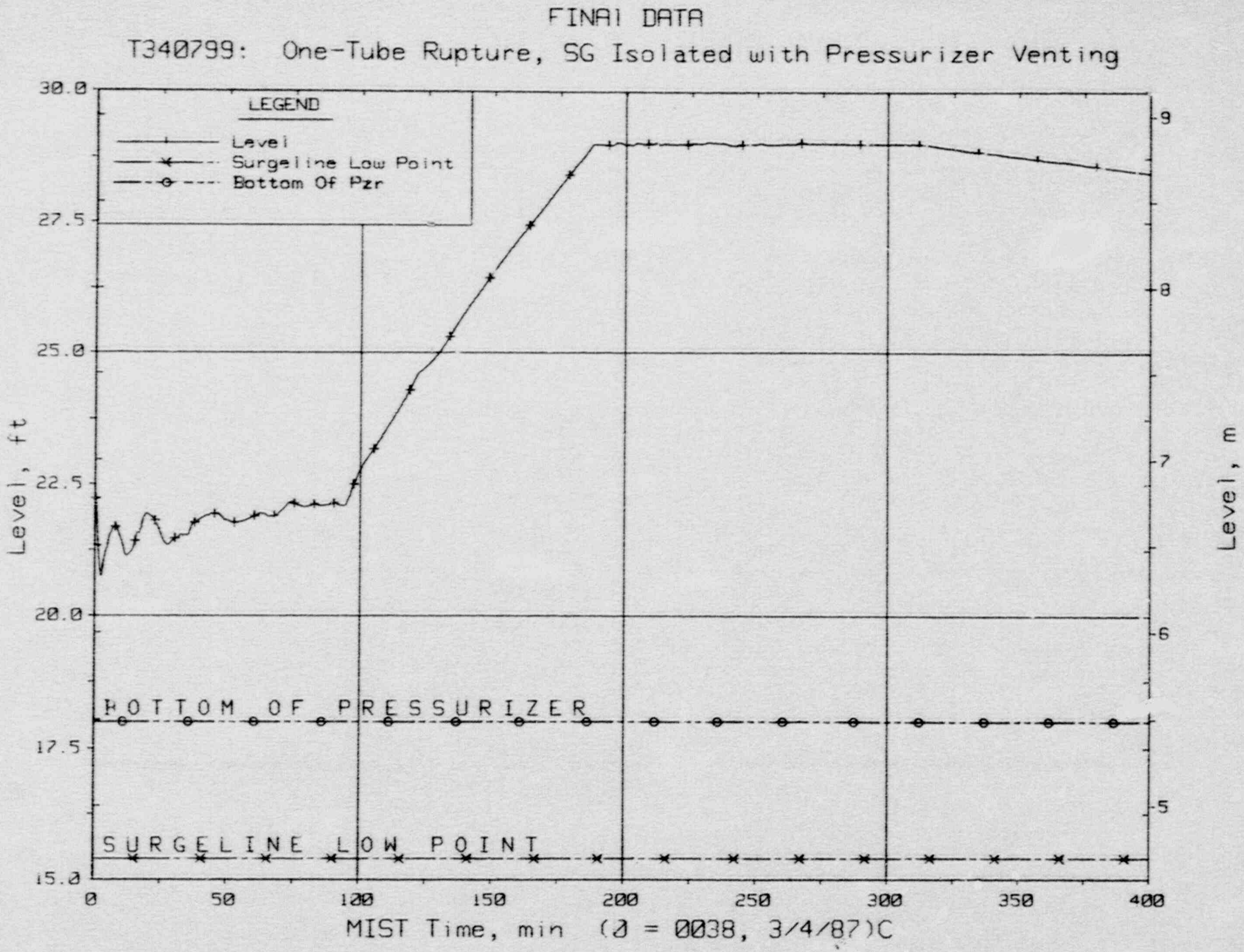


Figure 5.5.36. Pressurizer Collapsed Liquid Level (PZLV20)

Thu Jul 28 13:16:13 1988

PZLV1

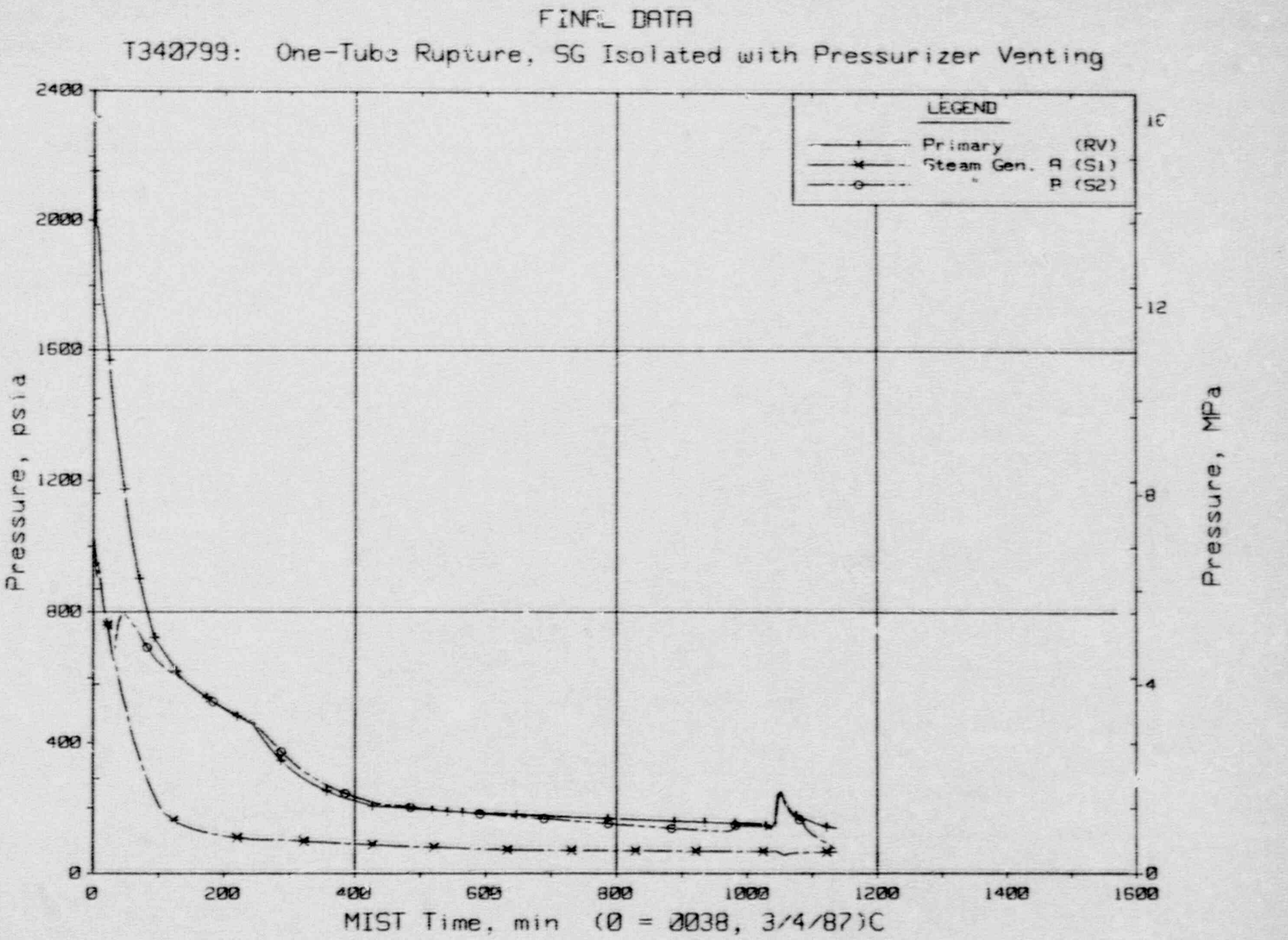


Figure 5.5.37. Primary and Secondary System Pressures (GPOIs)

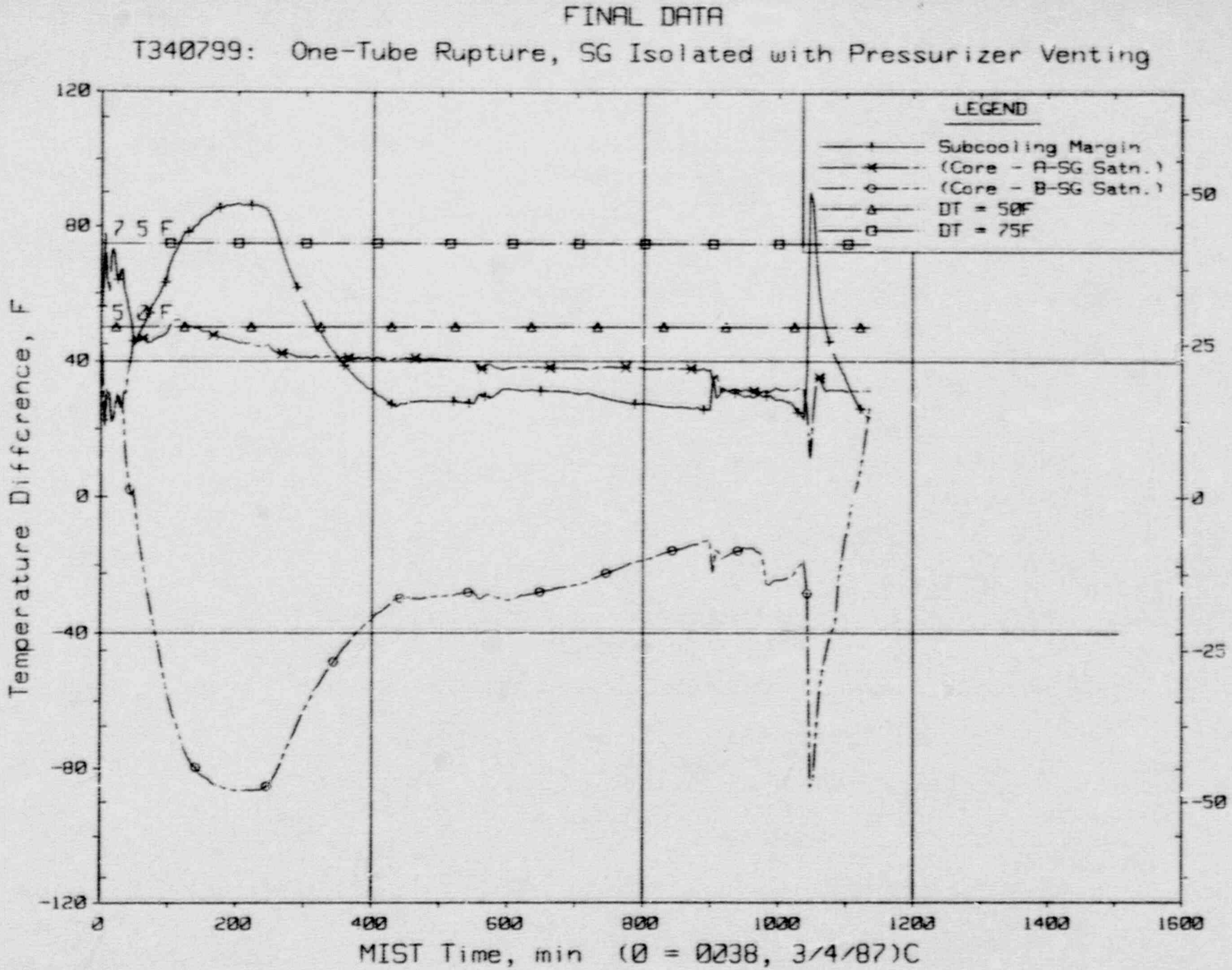


Figure 5.5.38. Control Temperature Differences

Fri Aug 5 14:21:16 1988

PRTD1

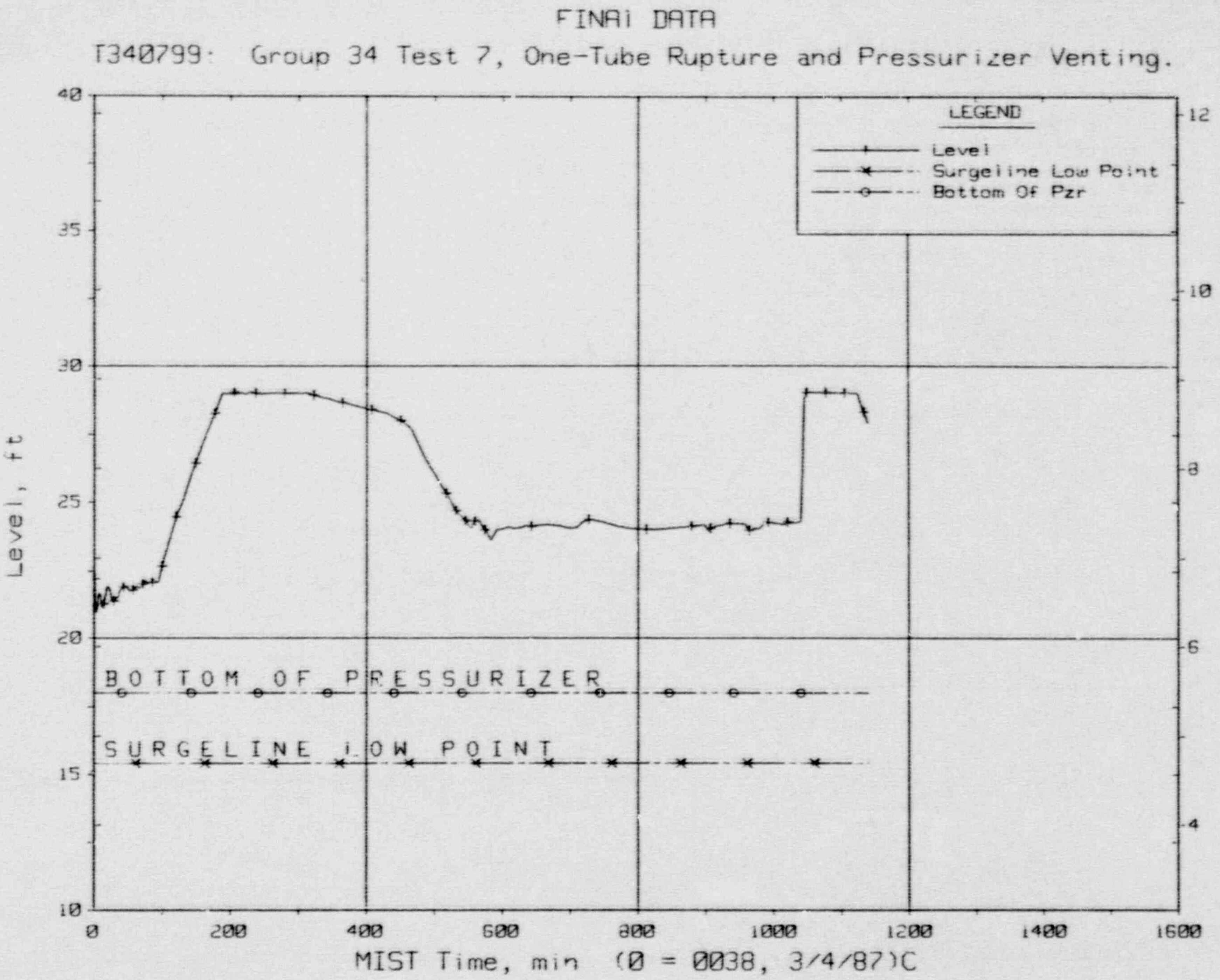


Figure 5.5.39. Pressurizer Collapsed Liquid Level (PZLV20)

Wed Nov 2 14:49:24 1988

PZLV1

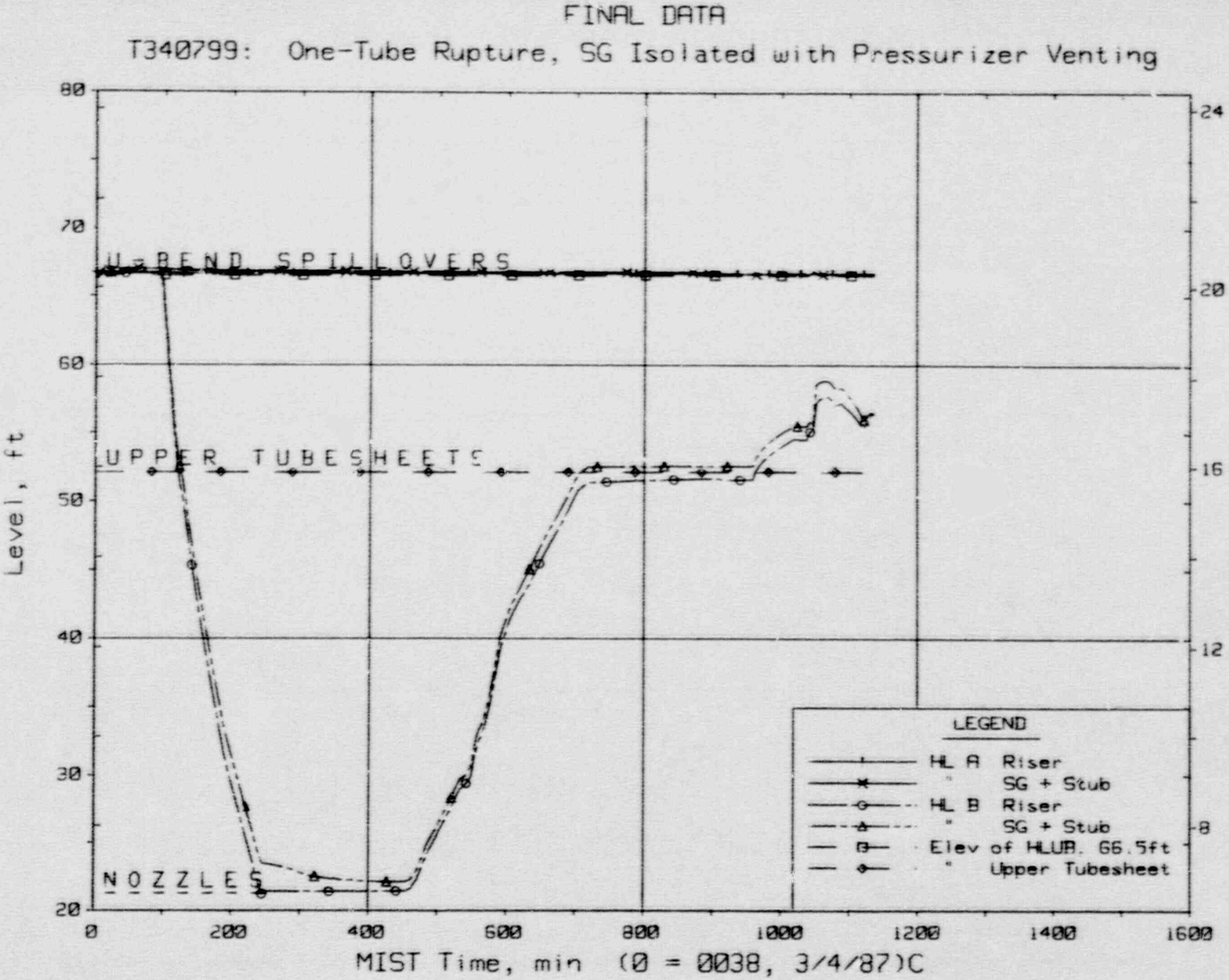


Figure 5.5.40. Hot Leg Riser and Stub Collapsed Liquid Levels

Fri Aug 5 14:26:34 1988

HLLV1

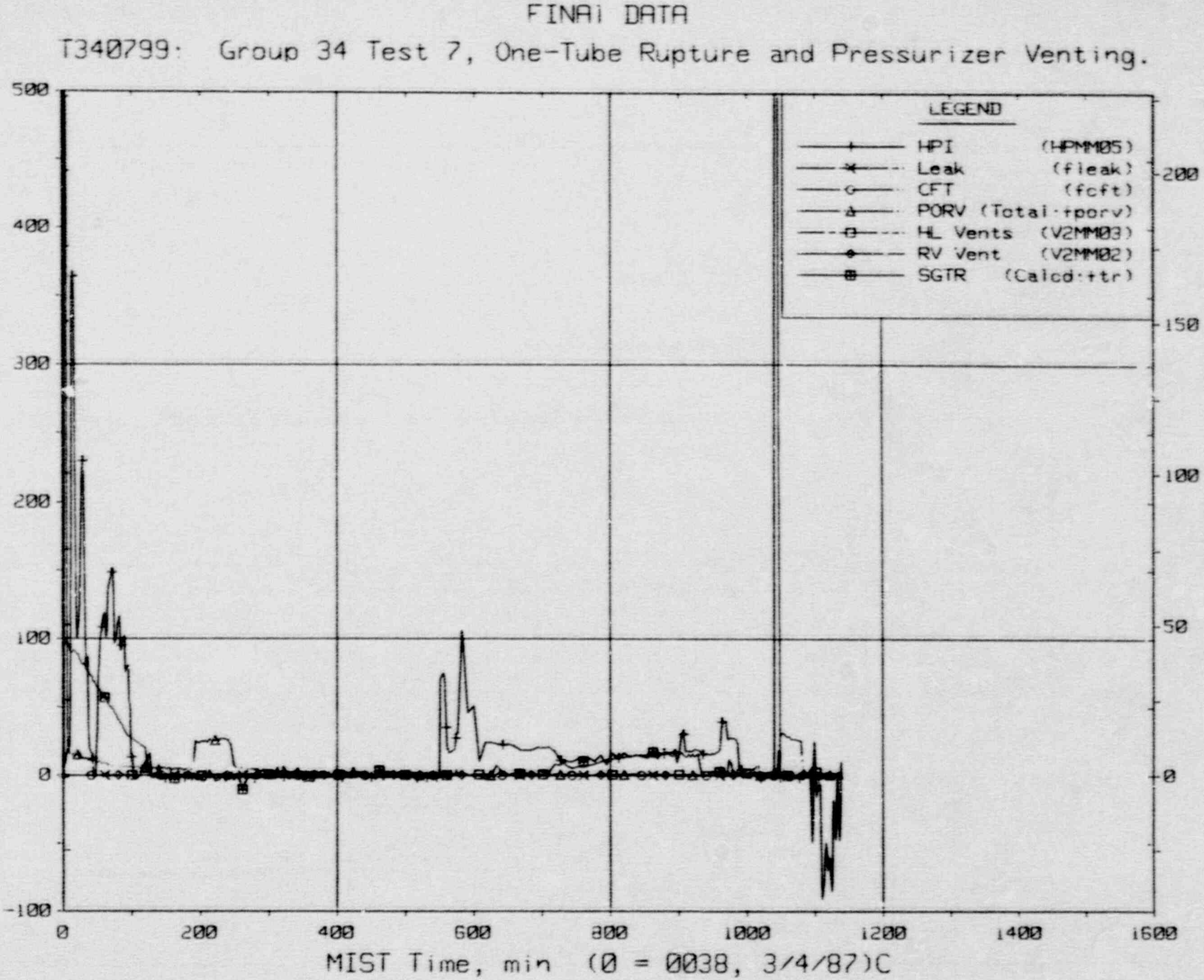


Figure 5.5.41. Primary System Boundary Flow Rates

FINAL DATA

T340799: One-Tube Rupture, SG Isolated with Pressurizer Venting

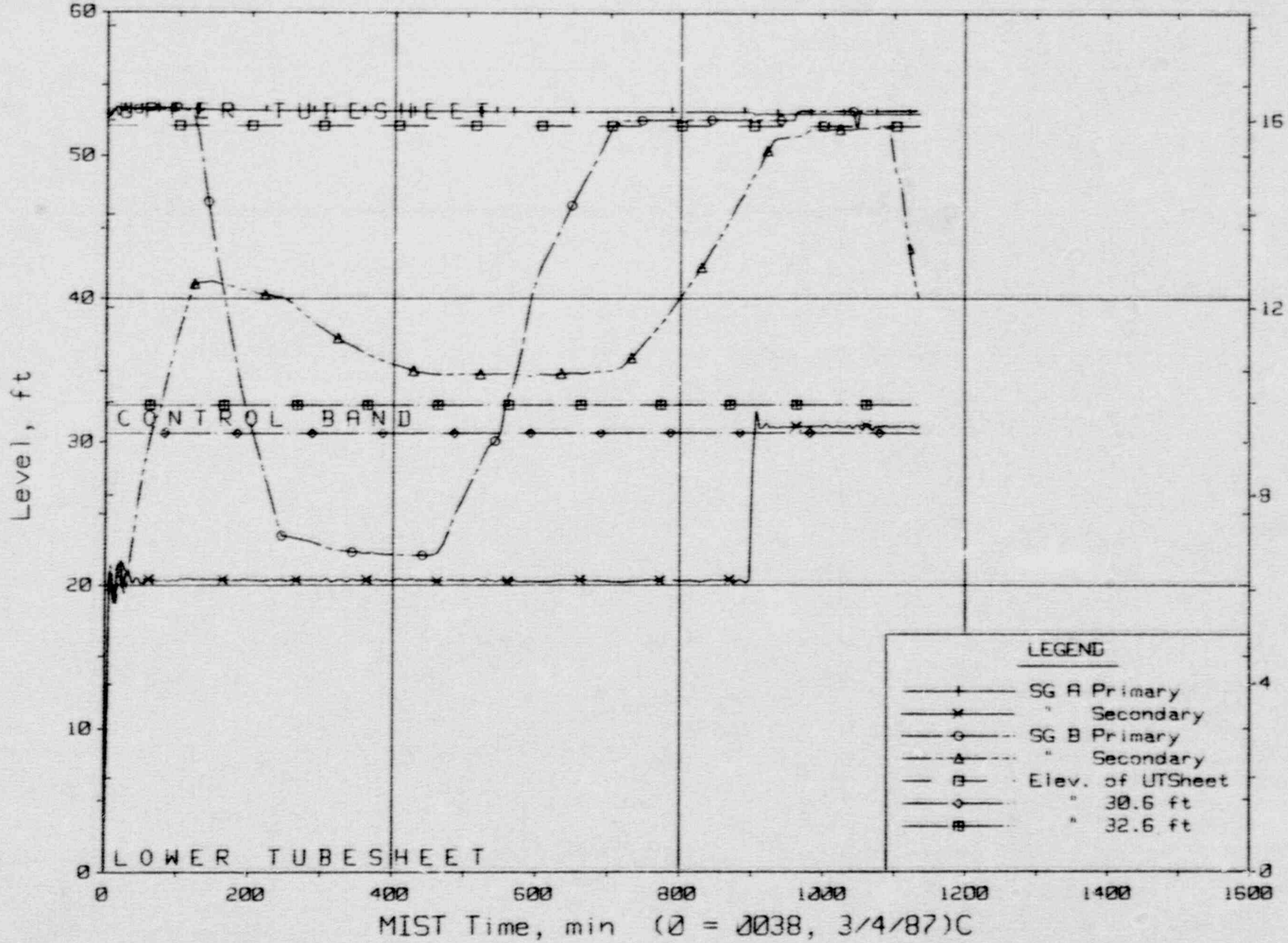


Figure 5.5.42. Steam Generator Collapsed Liquid Levels

Fri Aug 5 14:22:46 1988

SGLV2

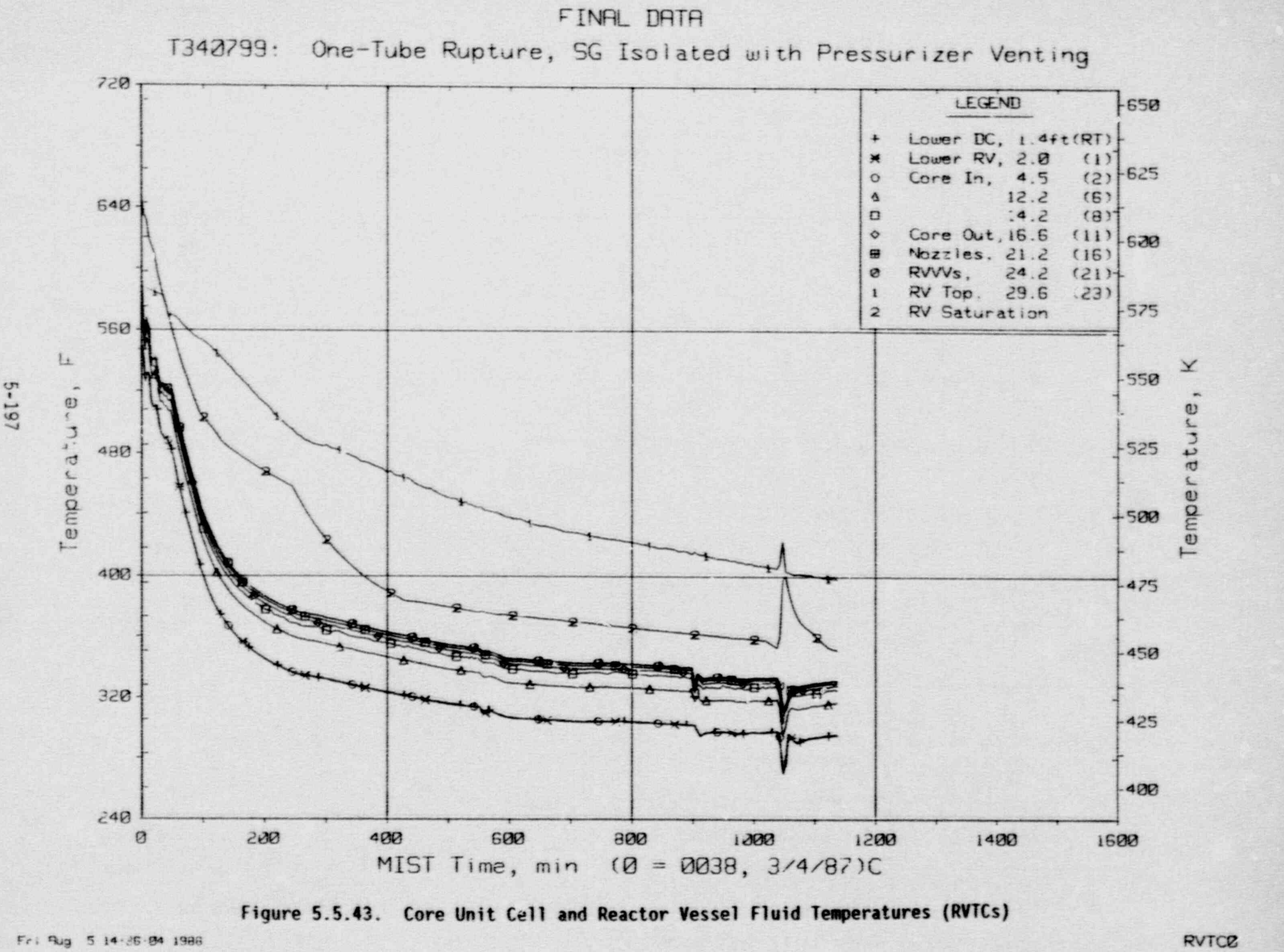


Figure 5.5.43. Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs)

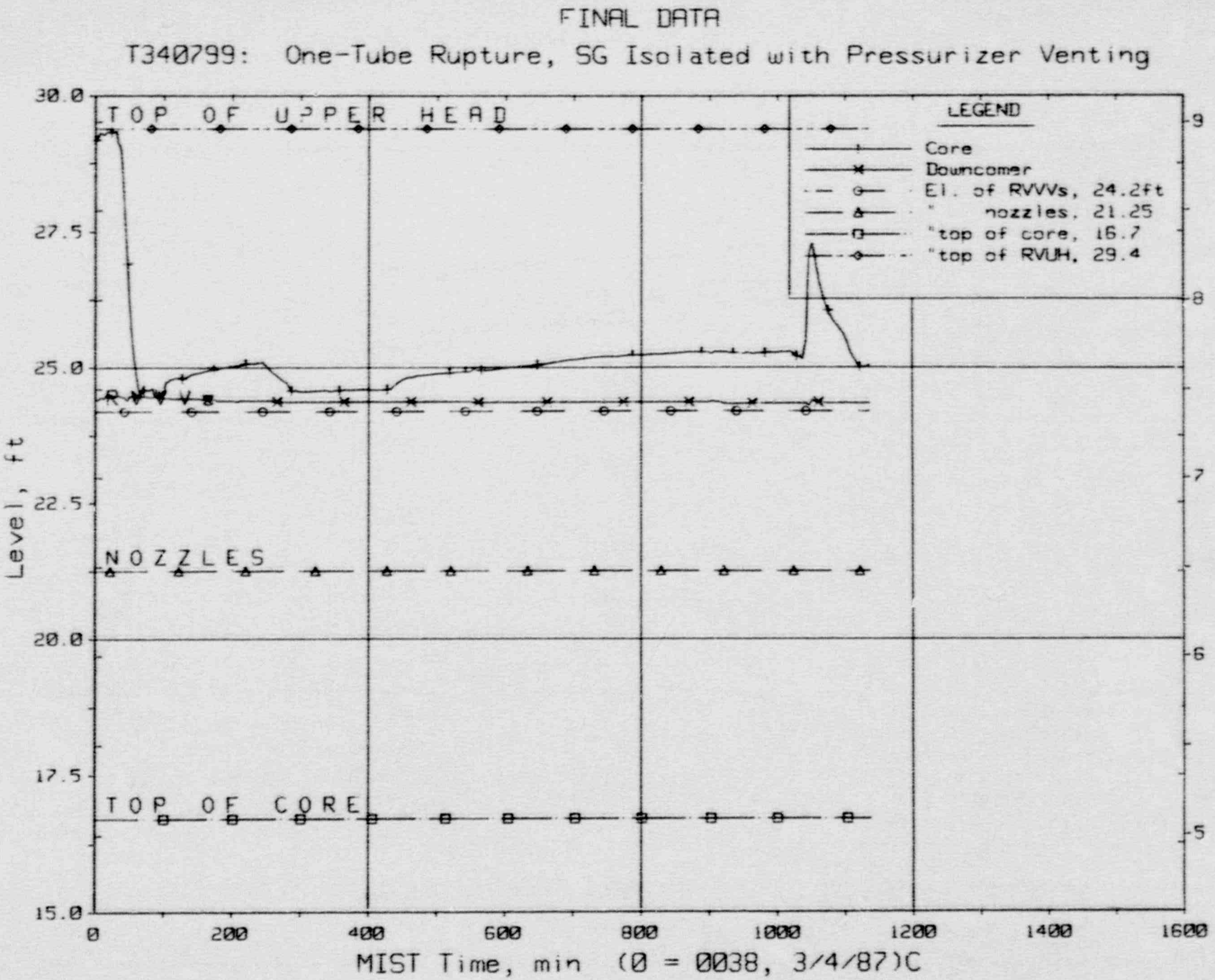


Figure 5.5.44. Core Region Collapsed Liquid Levels

Fri Aug 5 14:39:04 1988

COLV1

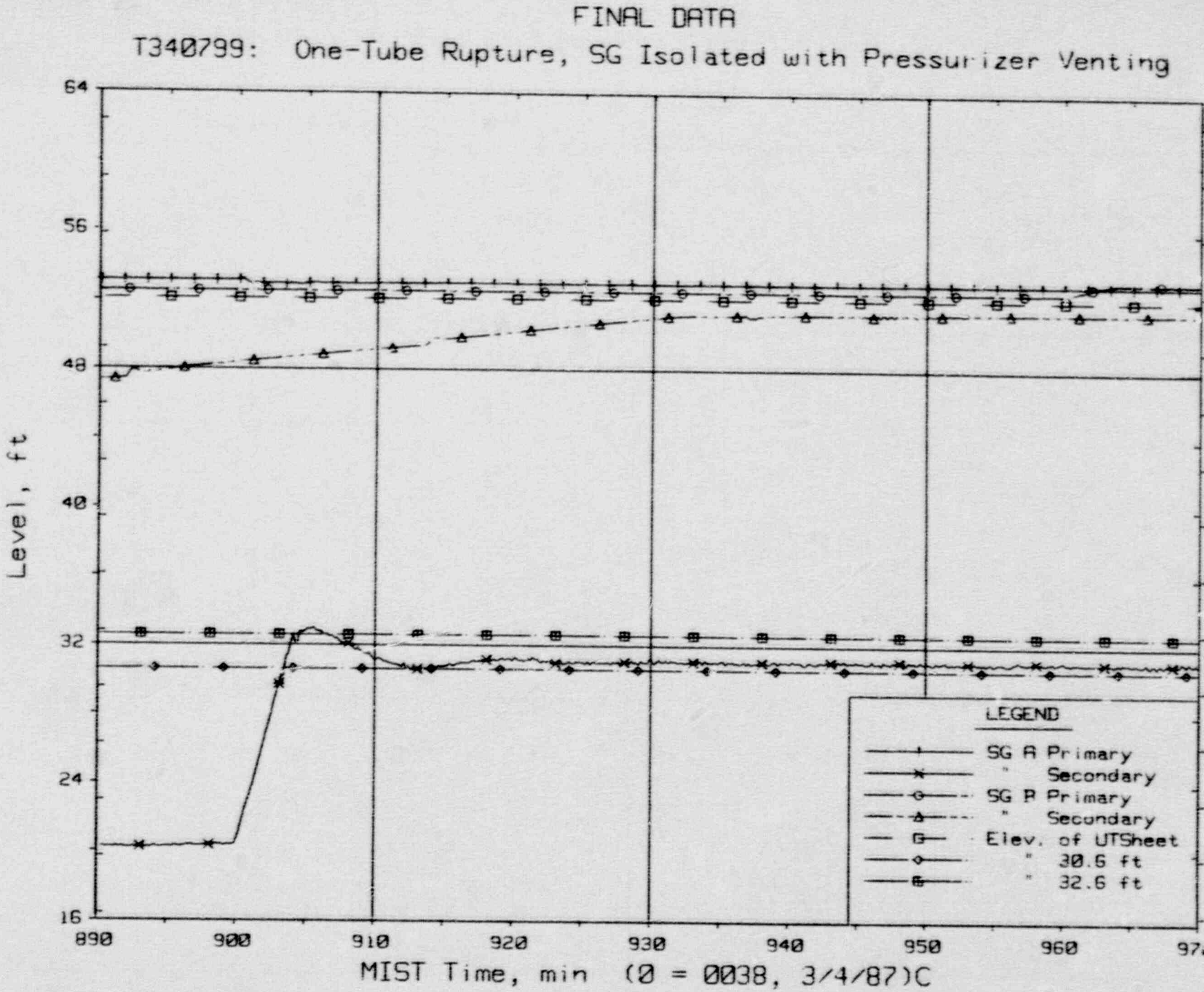


Figure 5.5.45. Steam Generator Collapsed Liquid Levels

Fri Aug 5 13:10:45 1988

SGLV2

FINAL DATA

T340799: One-Tube Rupture, SG Isolated with Pressurizer Venting

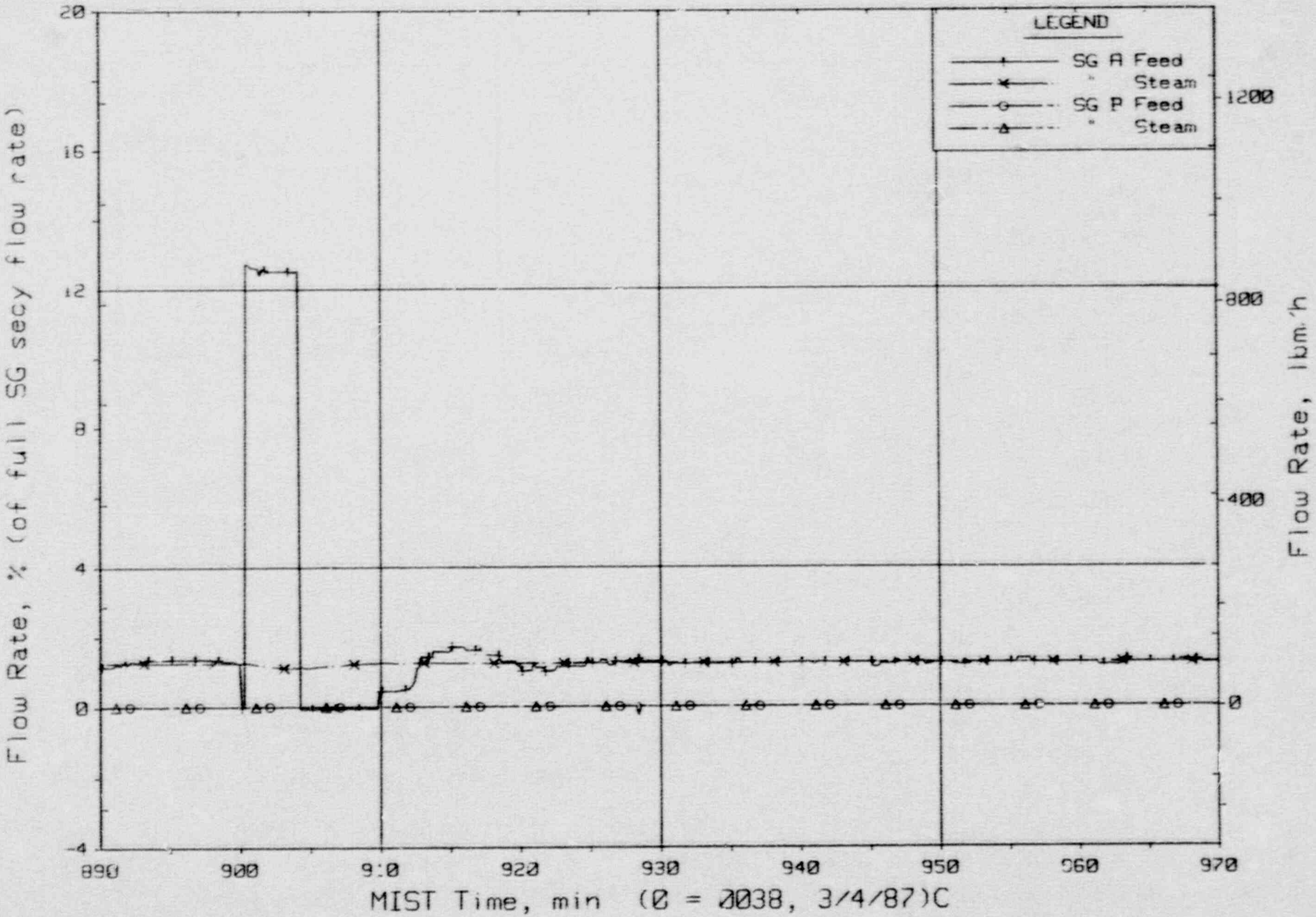


Figure 5.5.46. Secondary System Flow Rates

Fri Aug 5 13:13:41 1988

SGFL2

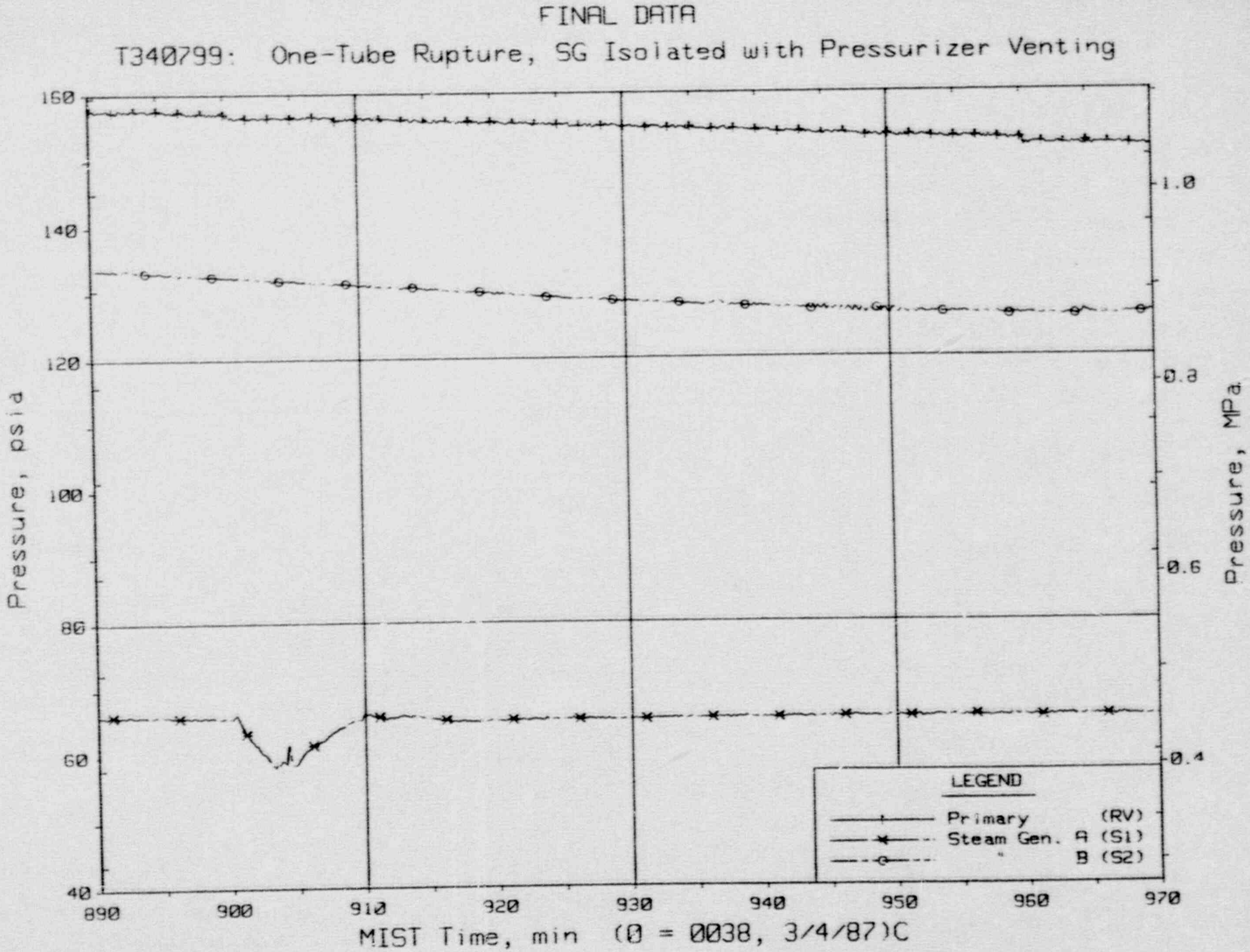


Figure 5.5.47. Primary and Secondary System Pressures (GP01s)

PSGP1

Fri Aug 5 13:19:29 1988

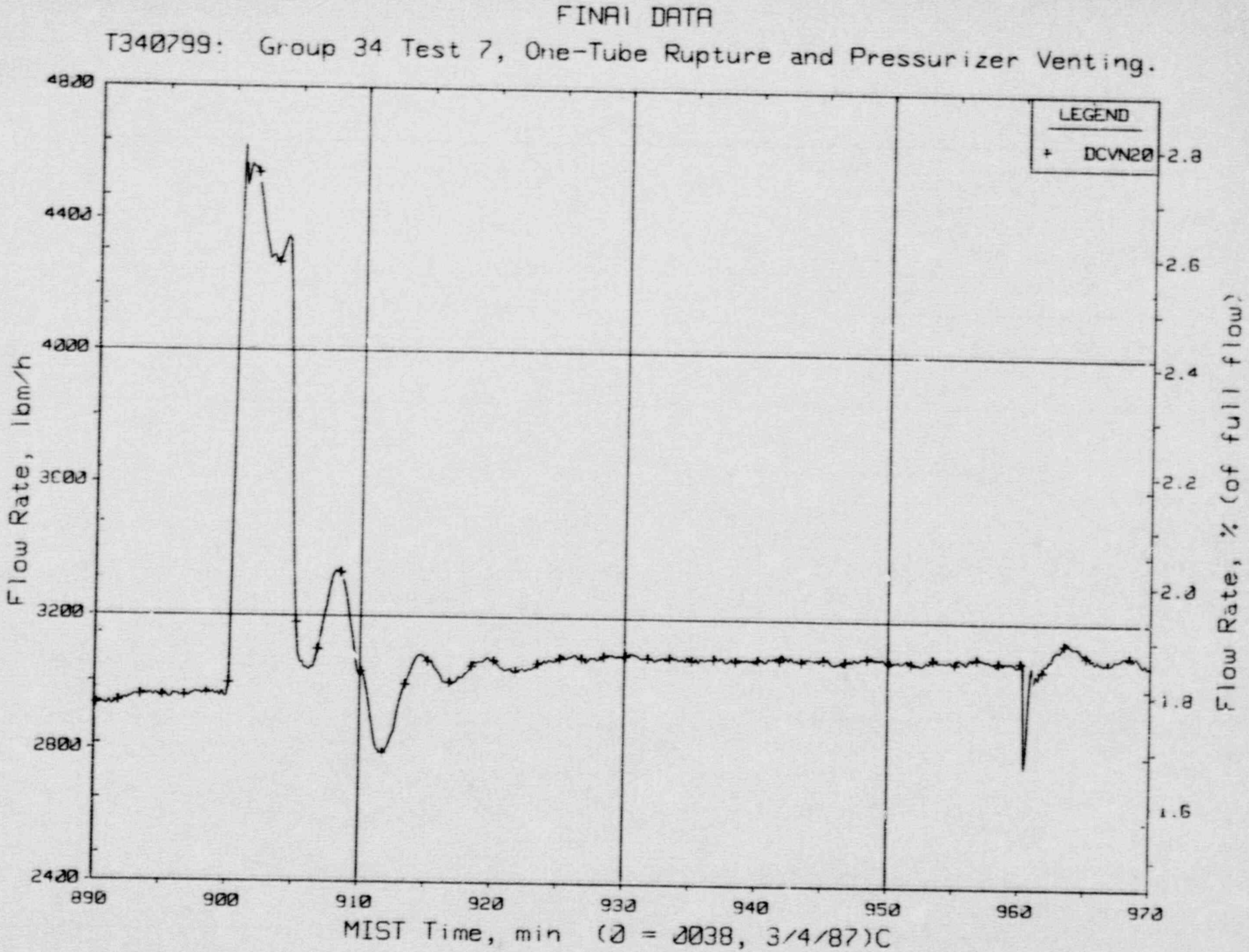


Figure 5.5.48. Primary System Venturi Flow Rates

Wed Nov 2 15:04:57 1988

DCFL1

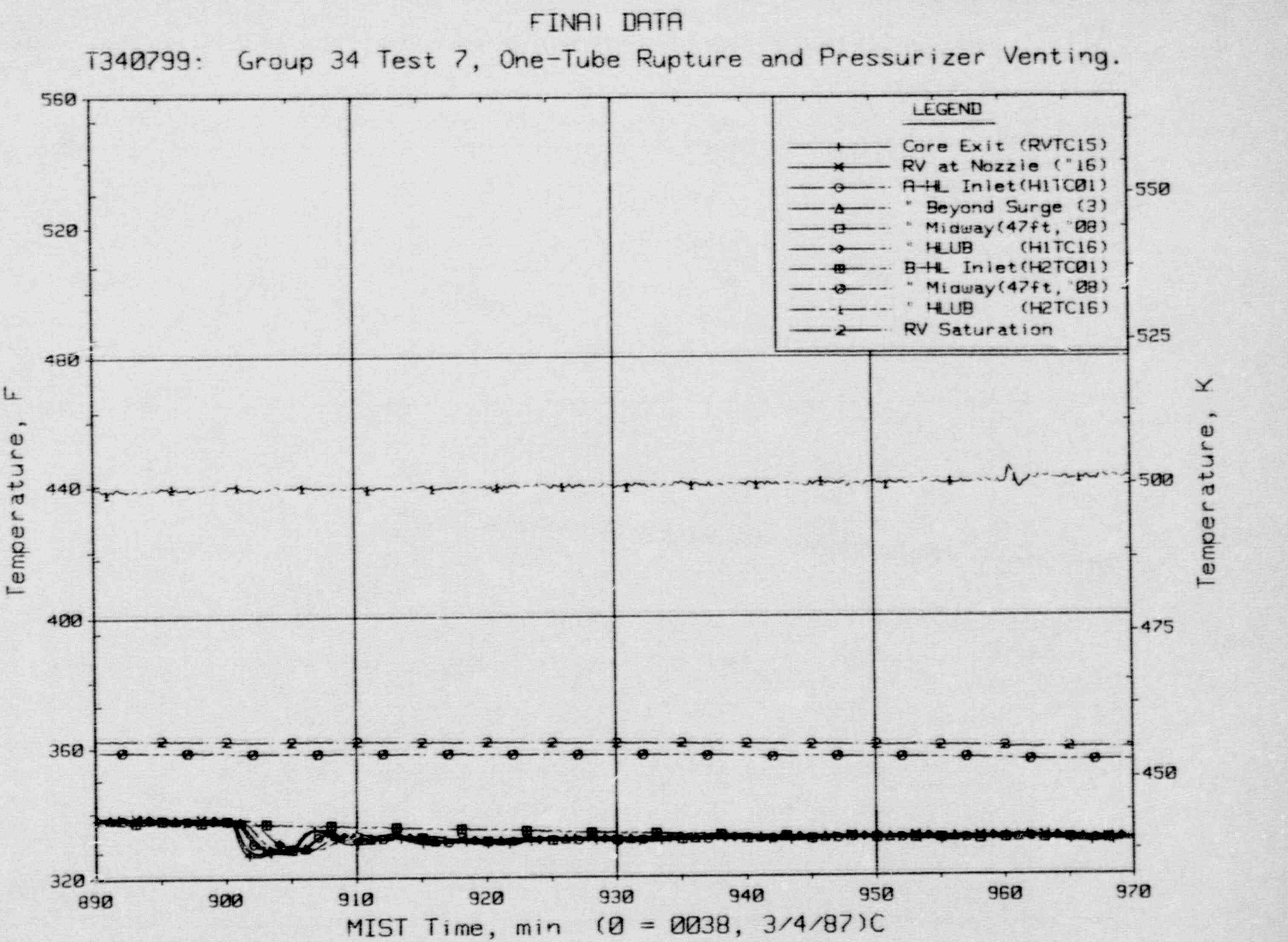


Figure 5.5.49. Composite Core Exit and Hot Leg Fluid Temperatures

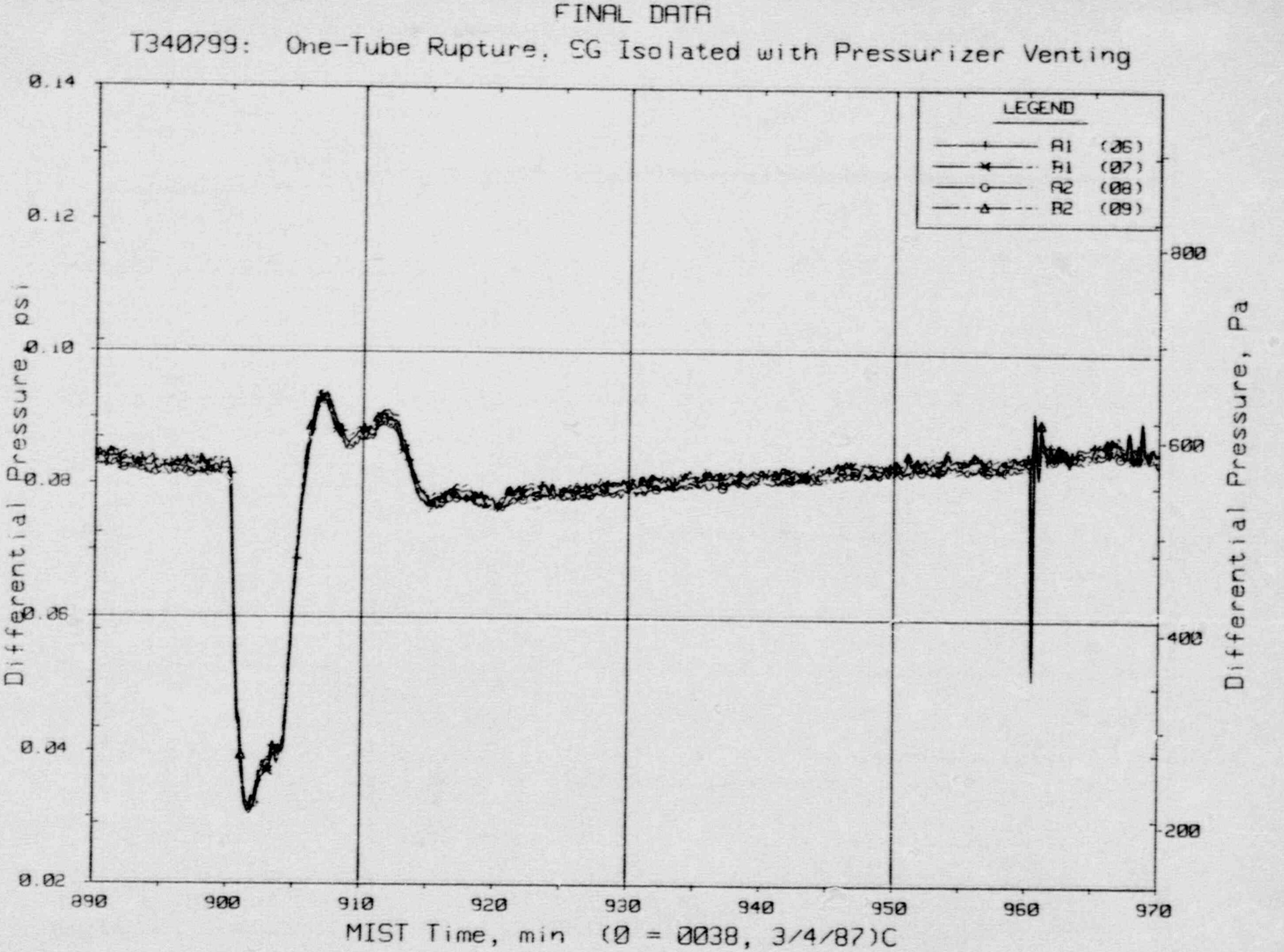


Figure 5.5.50. Reactor Vessel Vent Valve Differential Pressures (RVDPs)

VVDP1

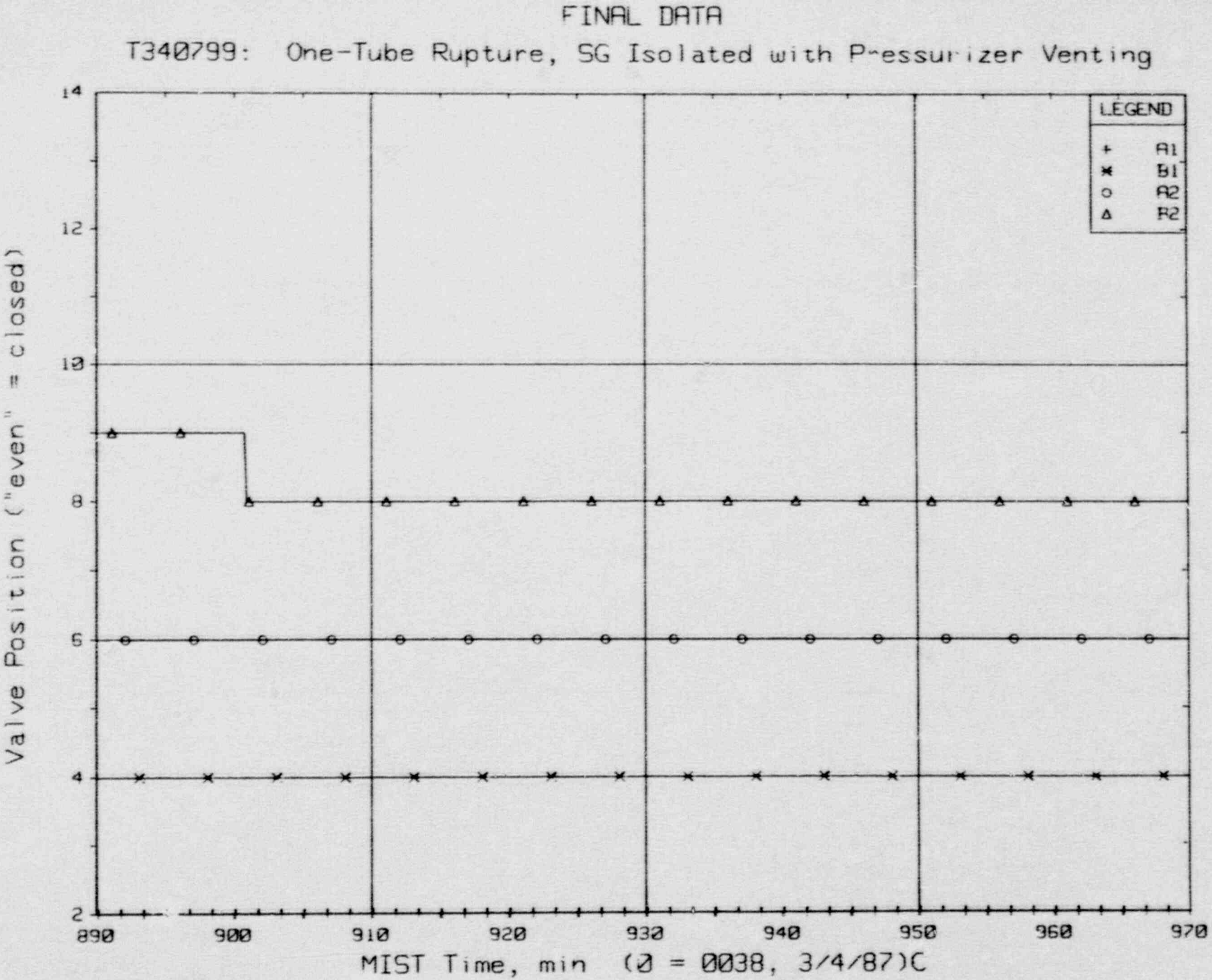


Figure 5.5.51. Reactor Vessel Vent Valve Positions

Fri Aug 5 13:35:12 1988

VVLS1

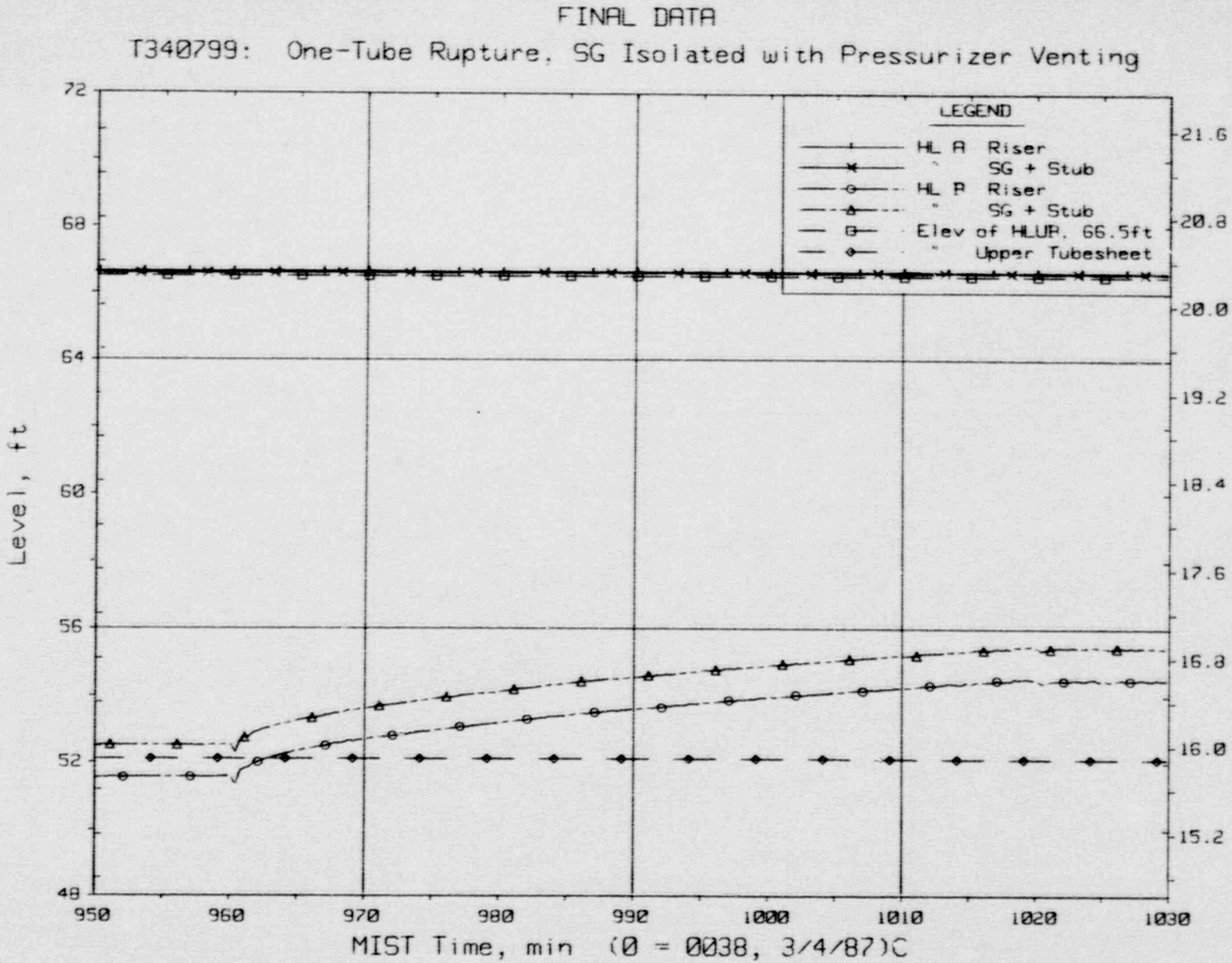


Figure 5.5.52. Hot Leg Riser and Stub Collapsed Liquid Levels

Fri Aug 5 15:21:21 1988

HLLV1

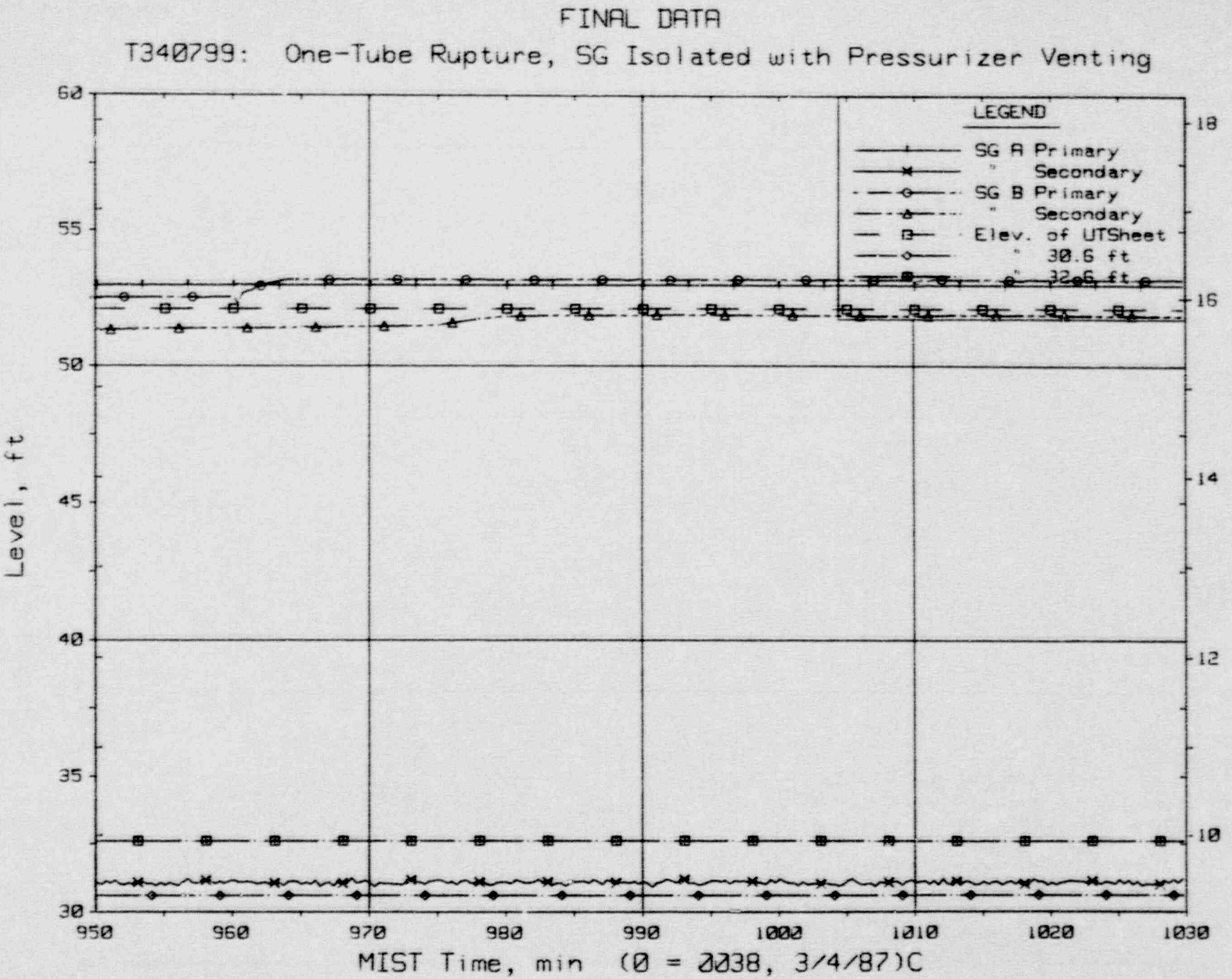


Figure 5.5.53. Steam Generator Collapsed Liquid Levels

Fri Aug 5 15:24:42 1988

SGLV2

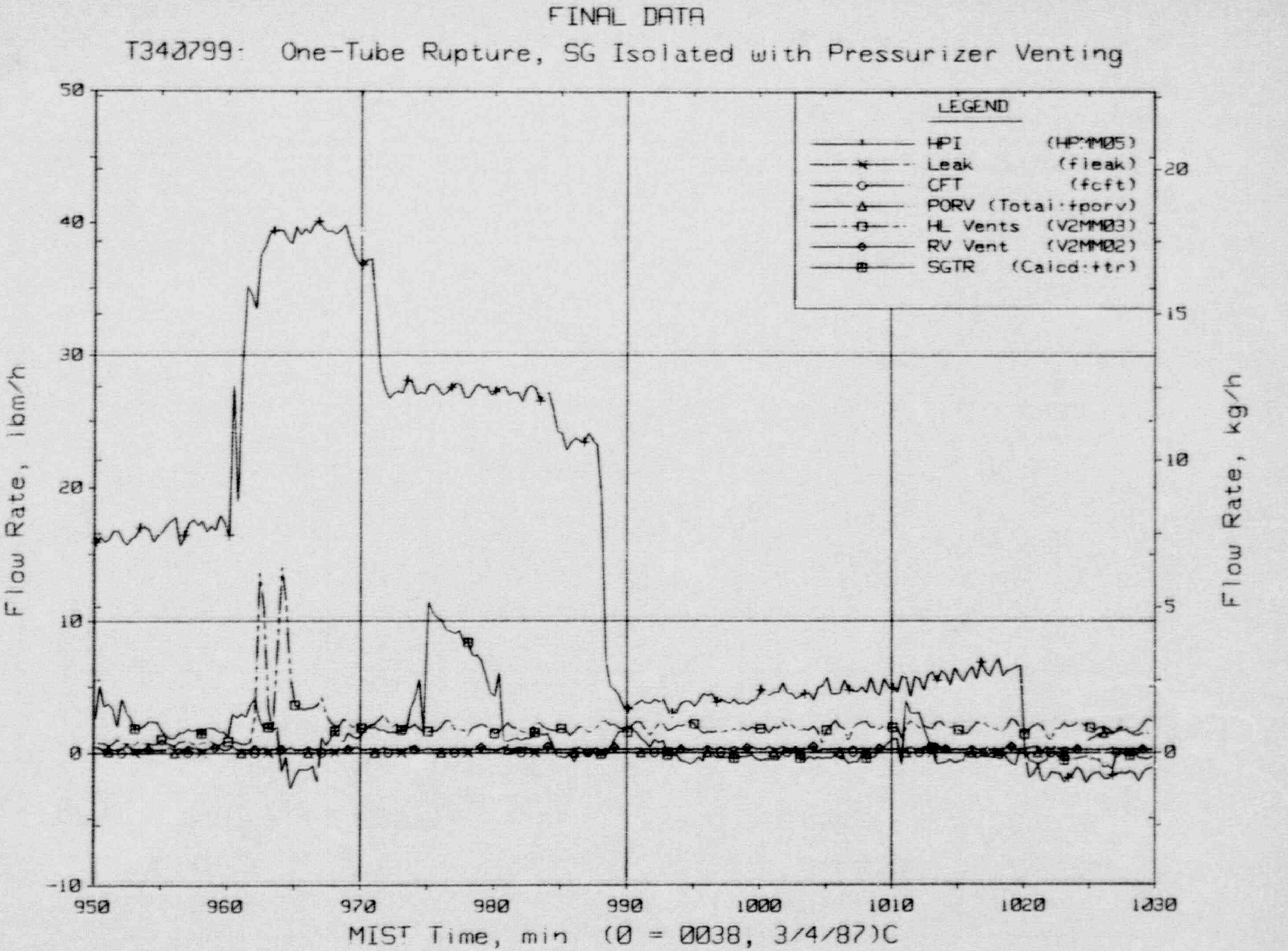
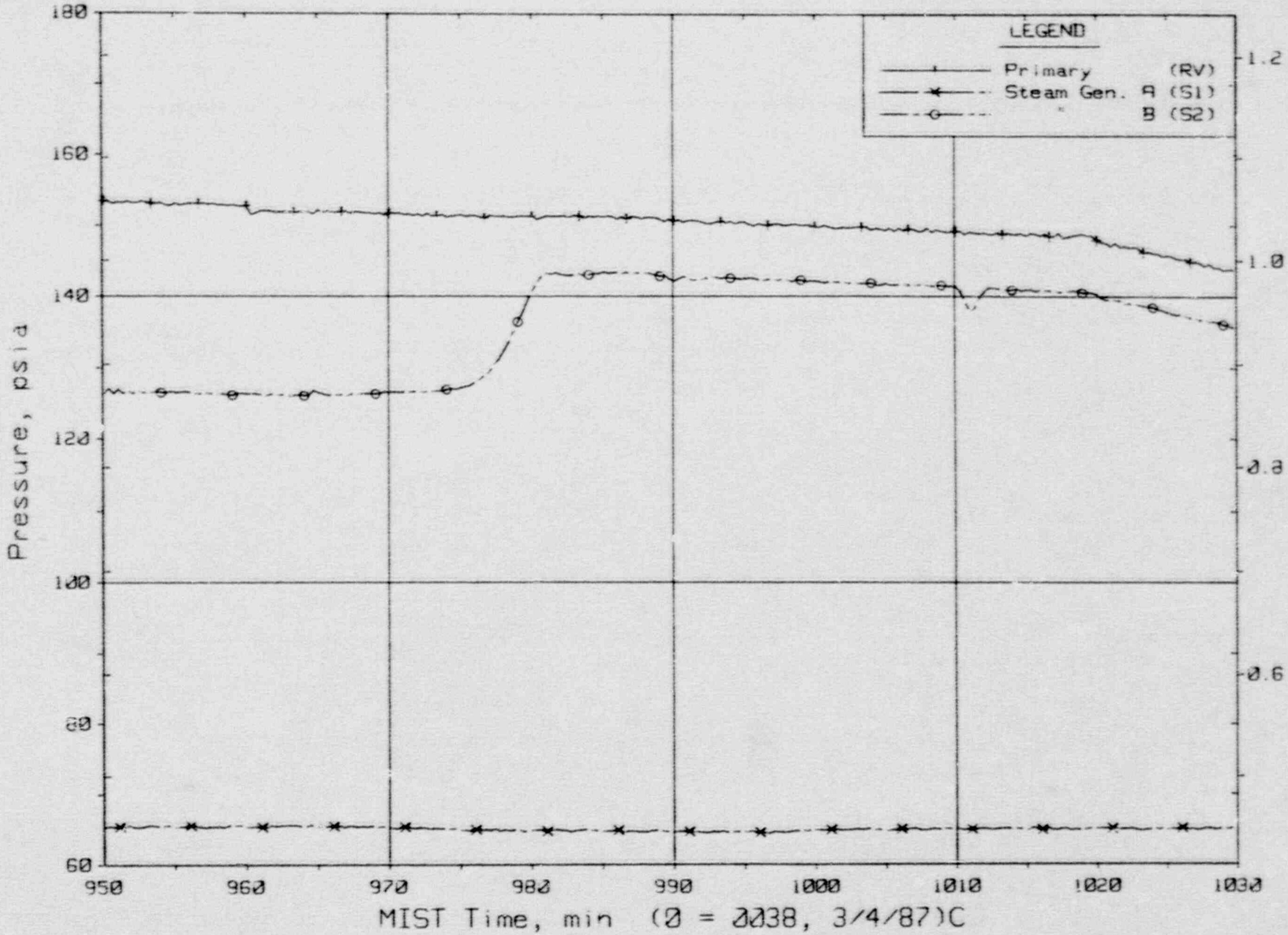


Figure 5.5.54. Primary System Boundary Flow Rates

FINAL DATA

T340799: One-Tube Rupture, SG Isolated with Pressurizer Venting



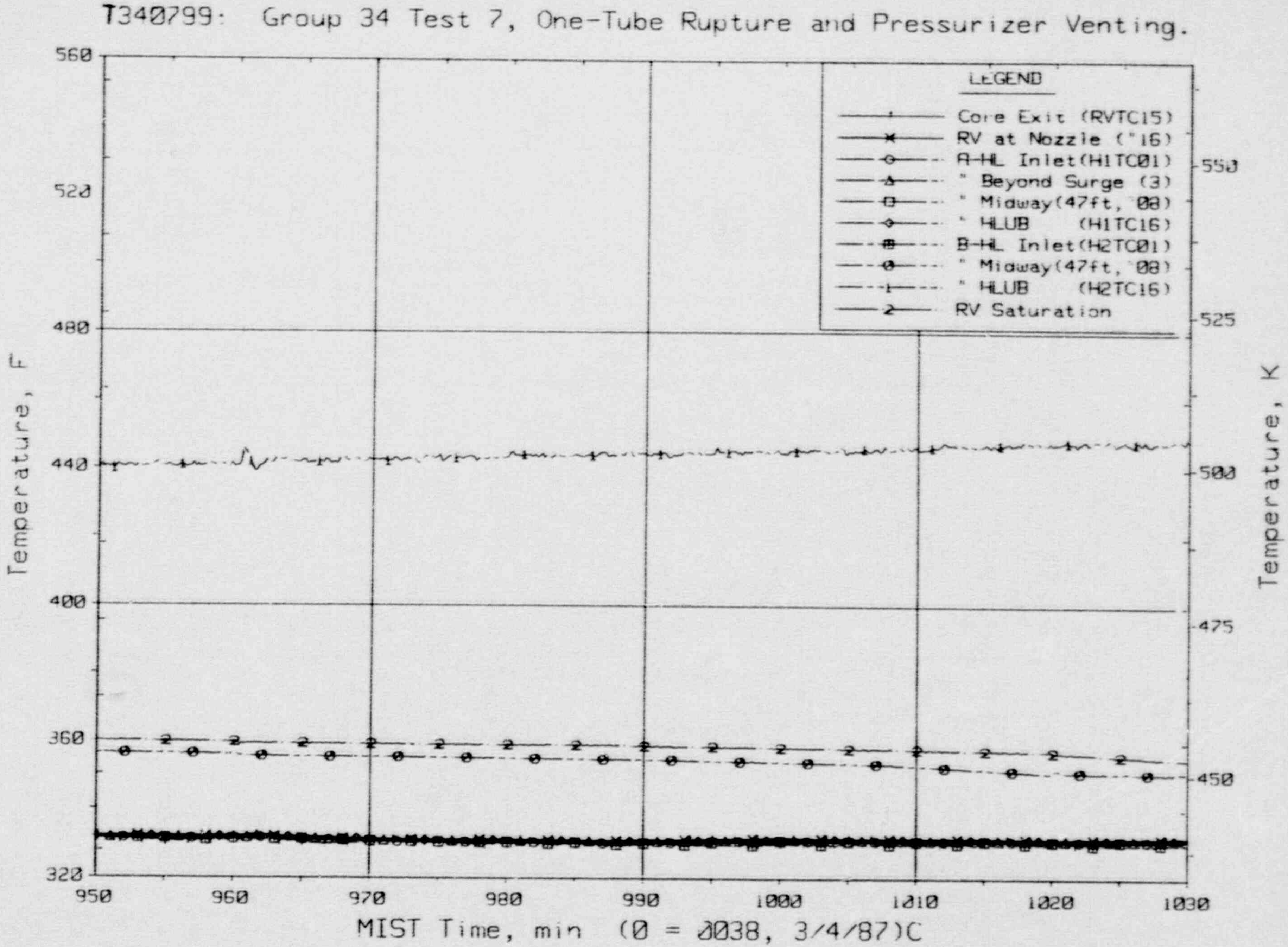
5-209

Figure 5.5.55. Primary System Boundary Flow Rates

Fri Aug 5 15:30:19 1988

PSGP1

FINAL DATA

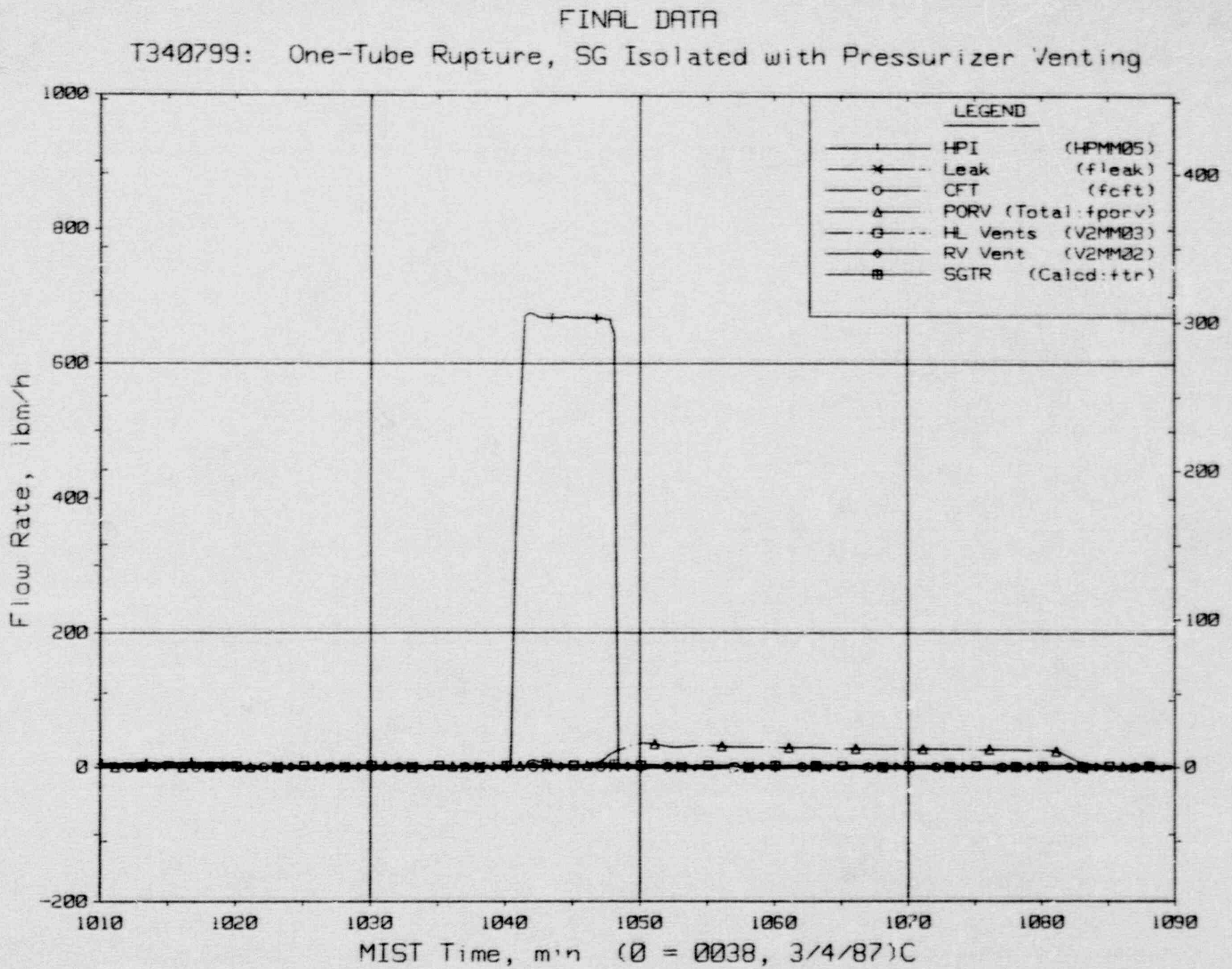


5-210

Figure 5.5.56. Composite Core Exit and Hot Leg Fluid Temperatures

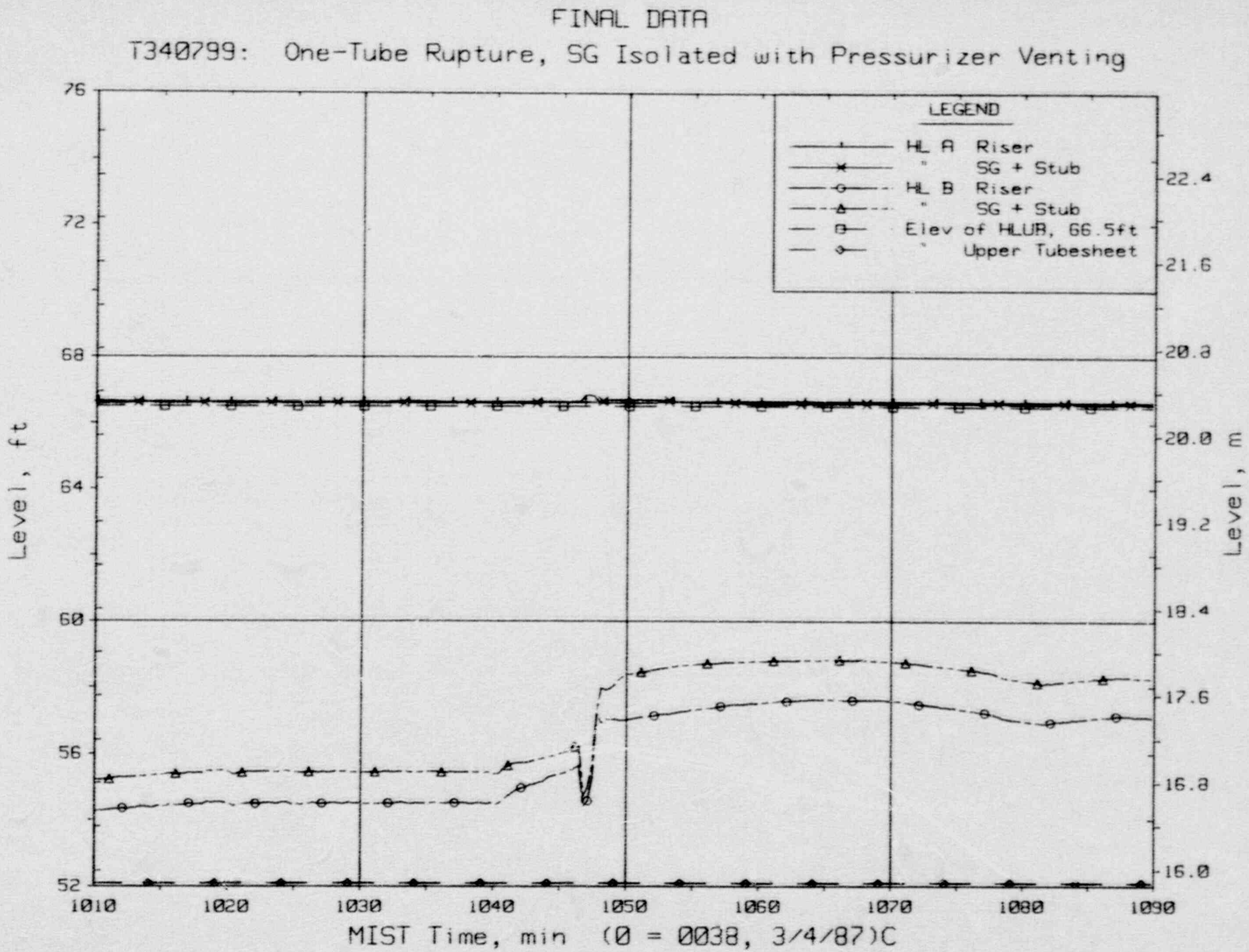
Wed Nov 2 15:00:45 1988

PRKT1



5-211

Figure 5.5.57. Primary System Boundary Flow Rates



5-212

Figure 5.5.58. Primary System Boundary Flow Rates

Mon Aug 8 13:06:21 1988

HLLV1

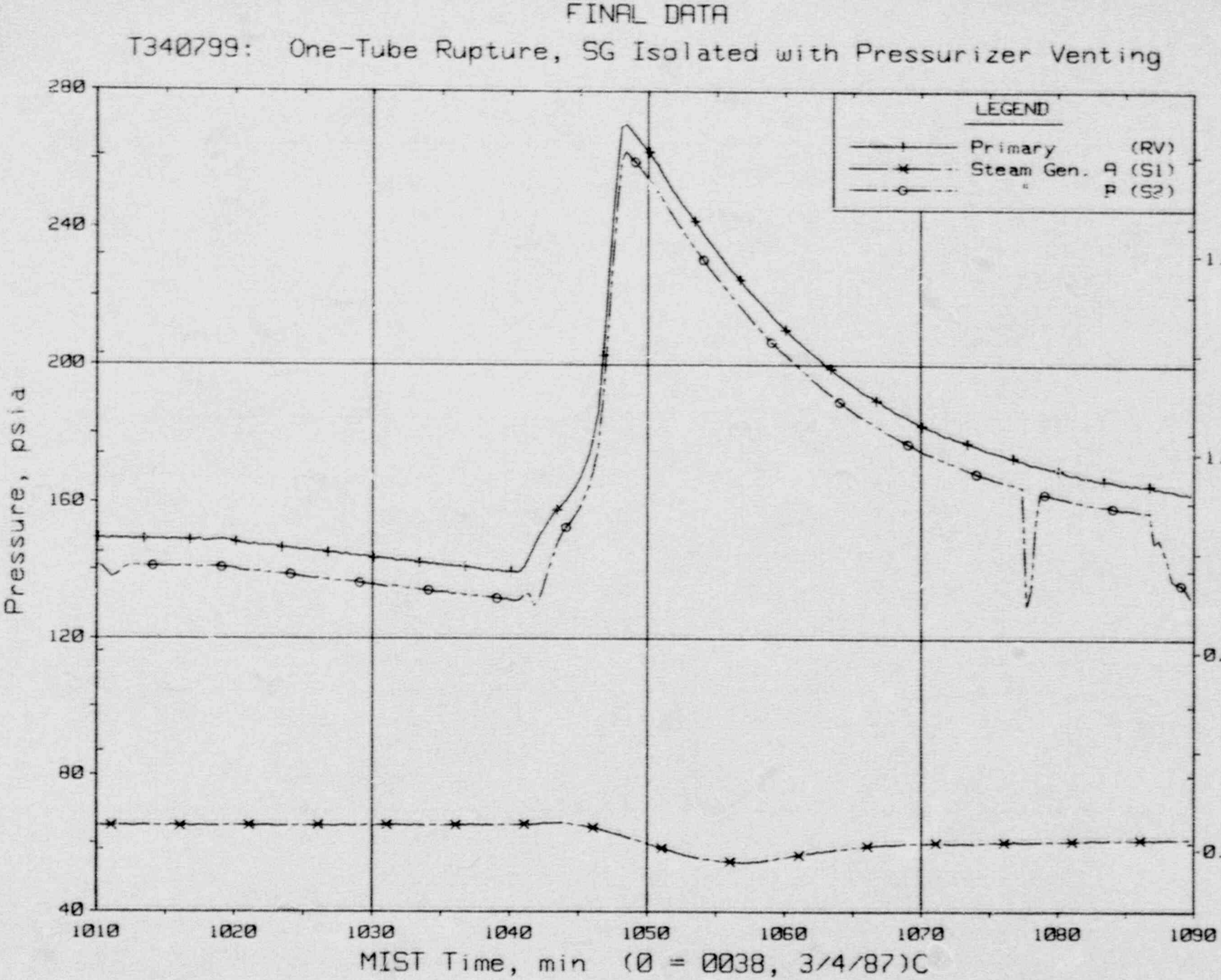


Figure 5.5.59. Primary and Secondary System Pressures (GP01s)

Mon Aug 8 13:08:56 1988

PSGP1

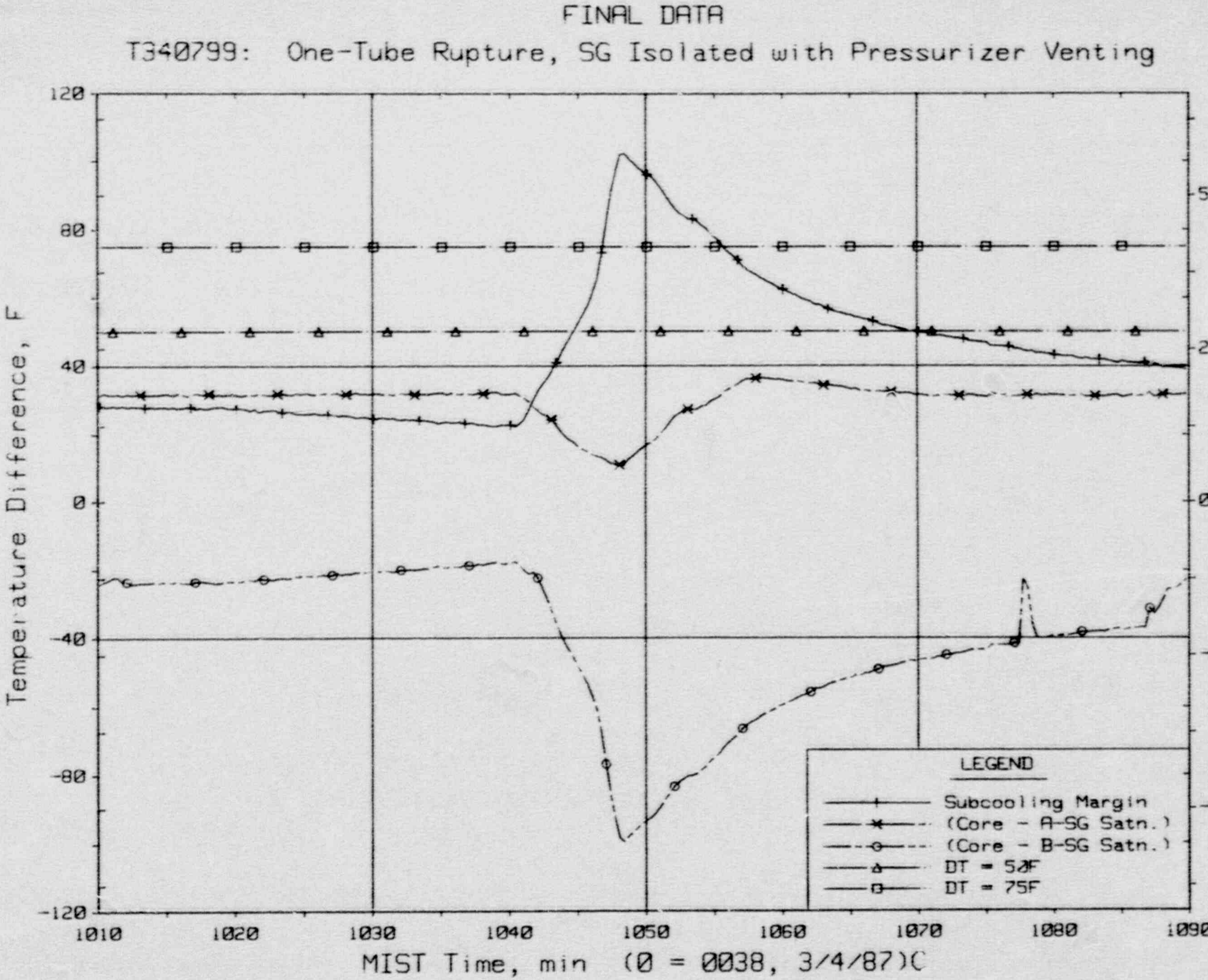


Figure 5.5.60. Control Temperature Differences

Mon Aug 8 13 11 40 1988

PRTD1

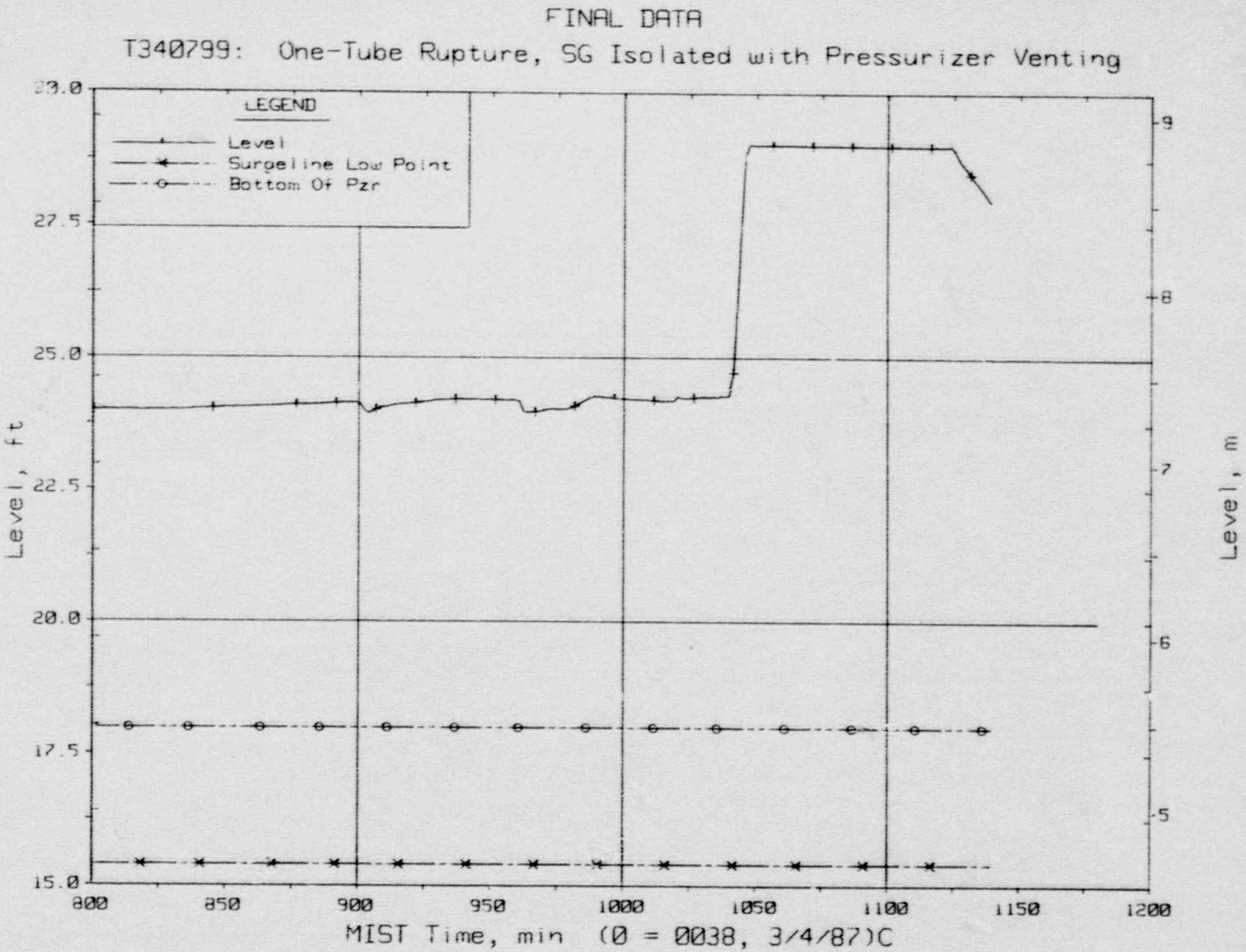


Figure 5.5.61. Pressurizer Collapsed Liquid Level (PZLV20)

Mon Aug 8 14:59:14 1988

PZLV1

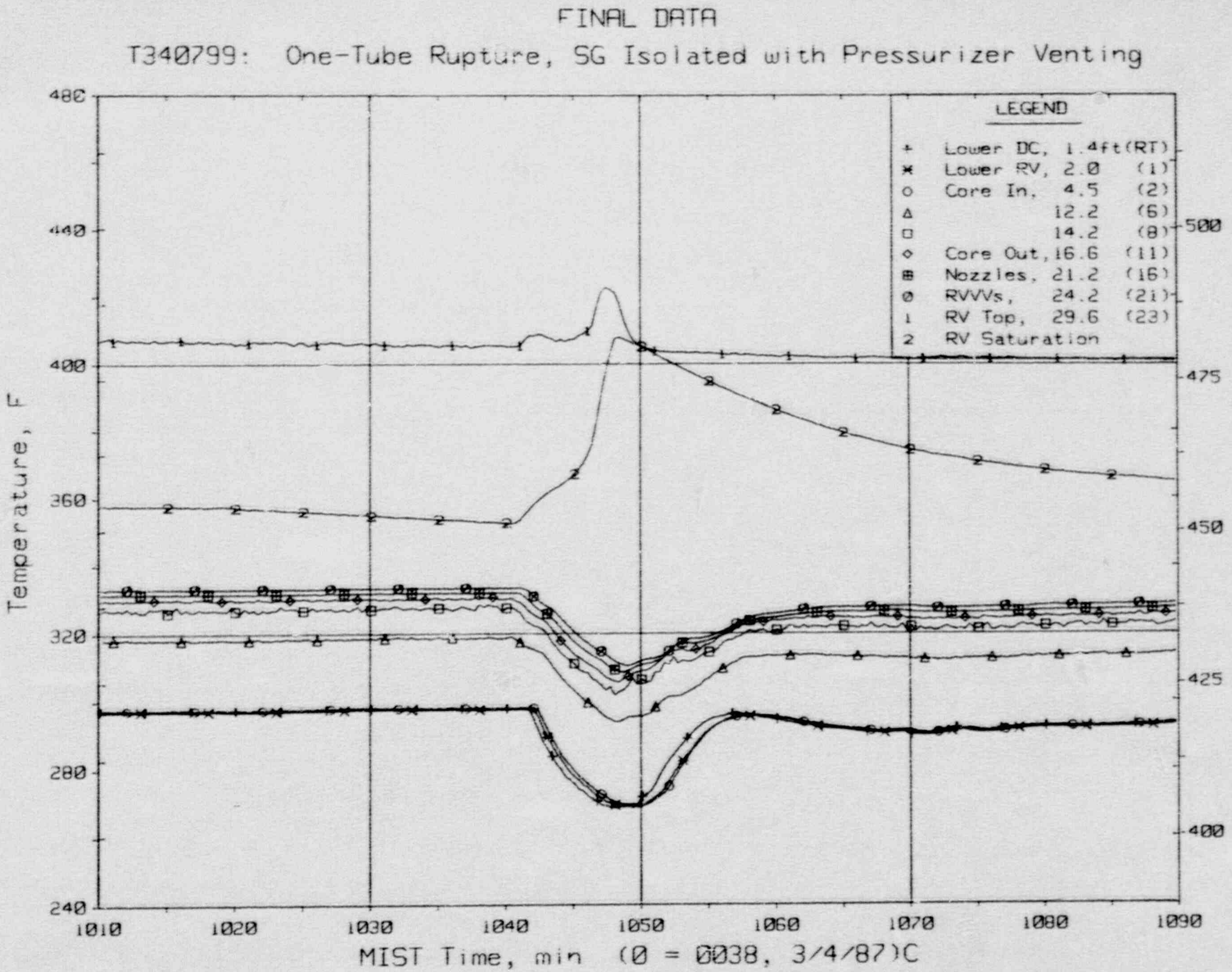


Figure 5.5.62. Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs)

Mon Aug 8 14:50:23 1988

RVTC0

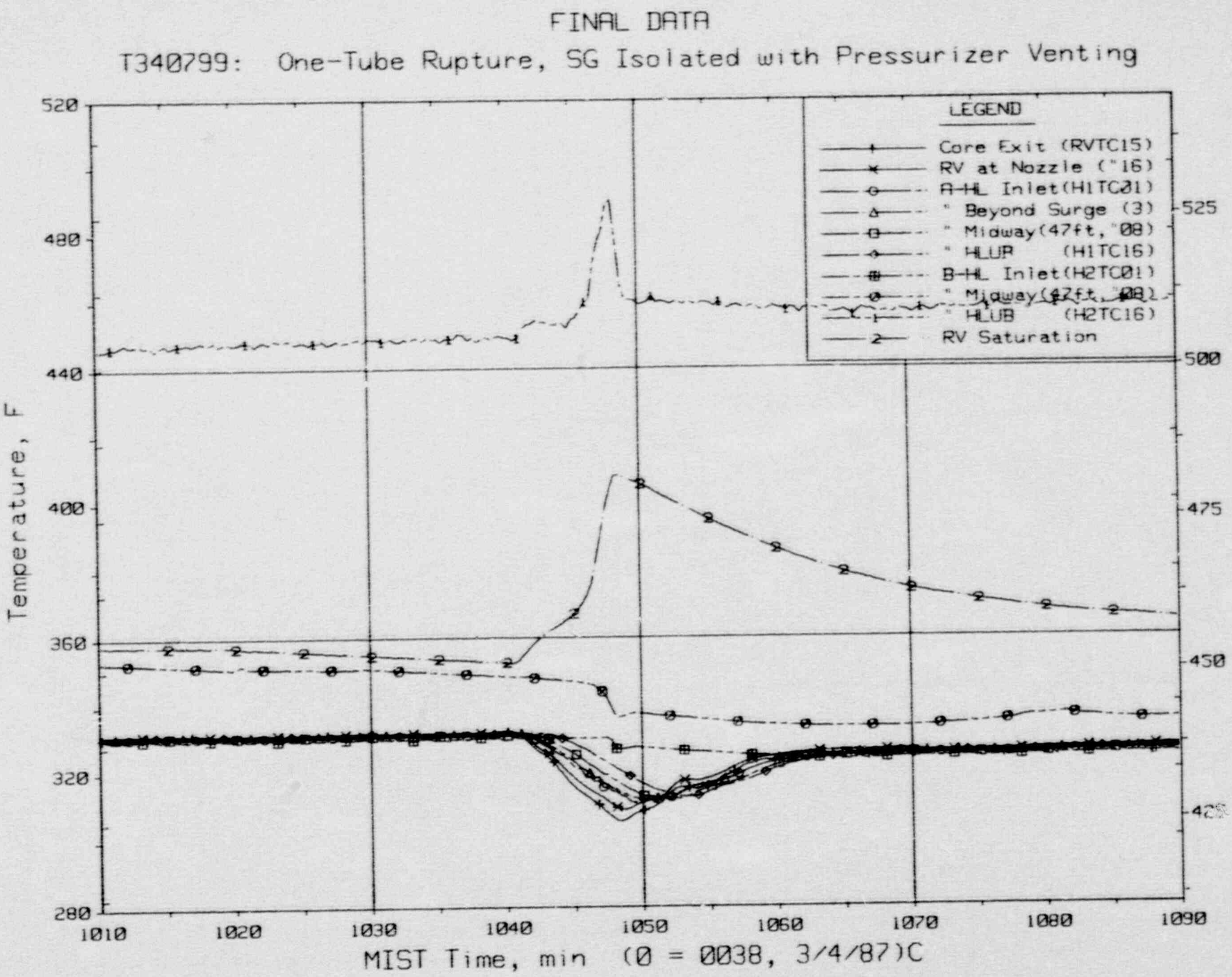


Figure 5.5.63. Composite Core Exit and Hot Leg Fluid Temperatures

PRKT1

Mon Aug 8 13:15:20 1988

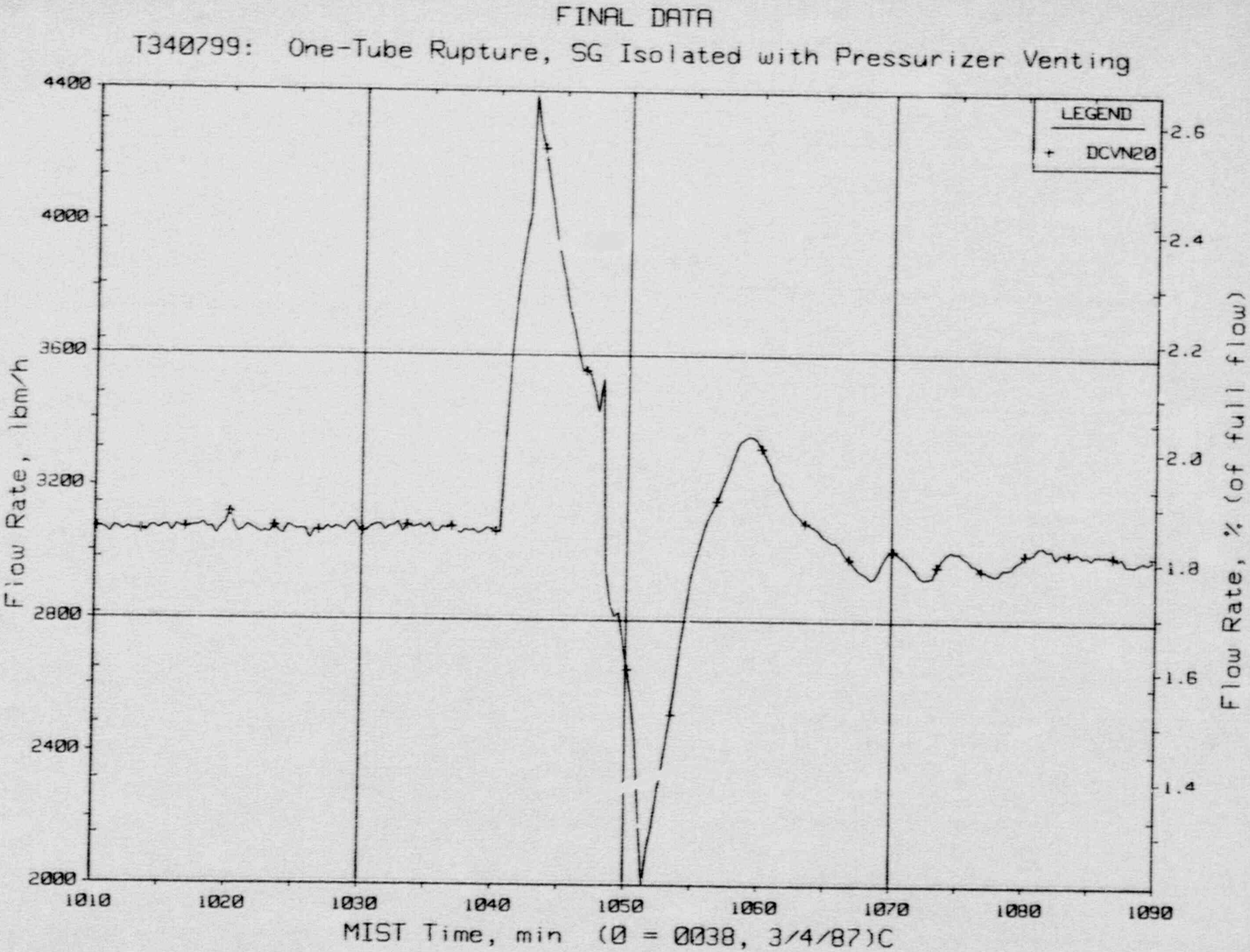


Figure 5.5.64. Primary System Venturi Flow Rates

Mon Aug 8 13:17:54 1988

DCFL1

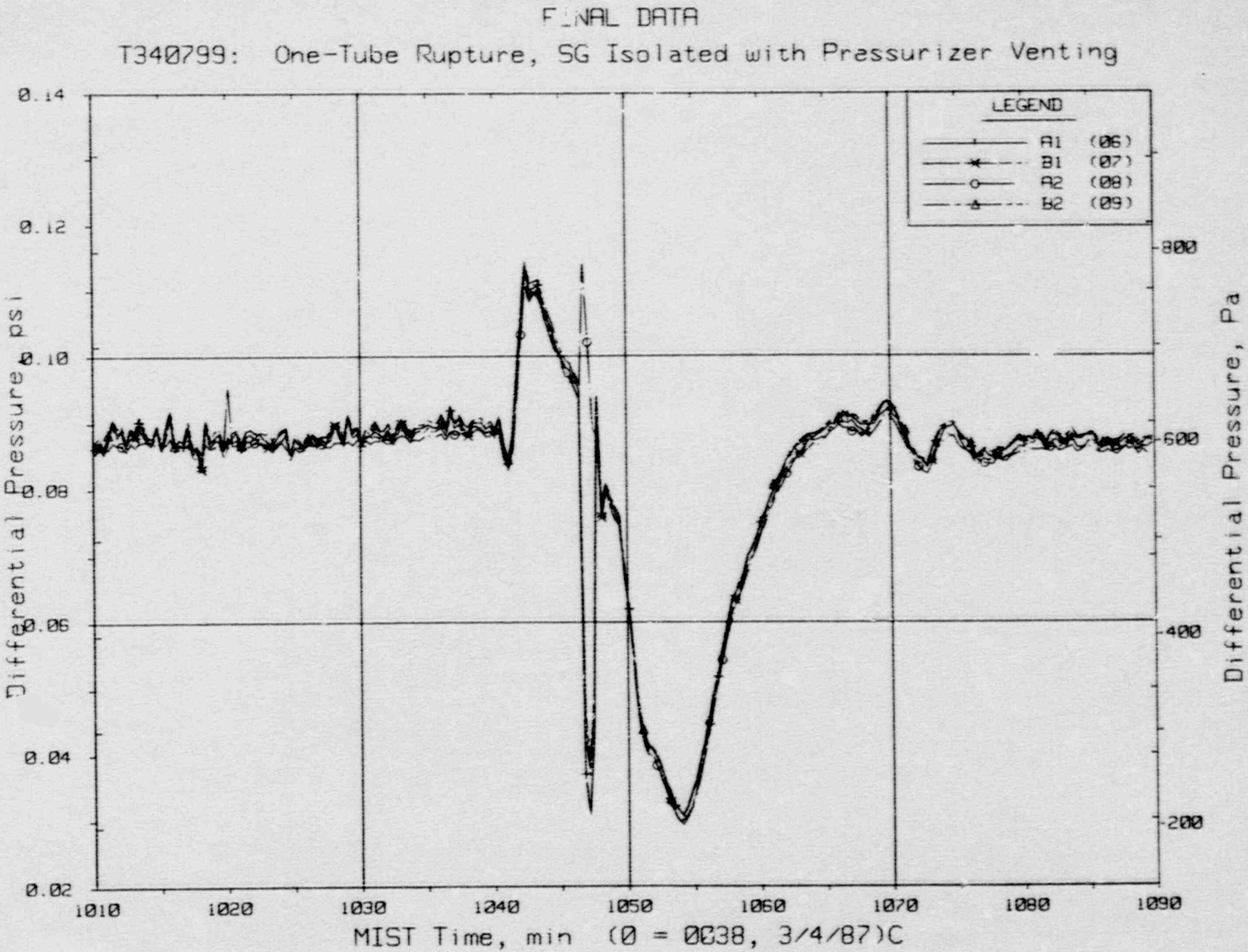


Figure 5.5.65. Reactor Vessel Vent Valve Differential Pressures (RVDPs)

Mon Aug 8 13:20:22 1988

VVDP1

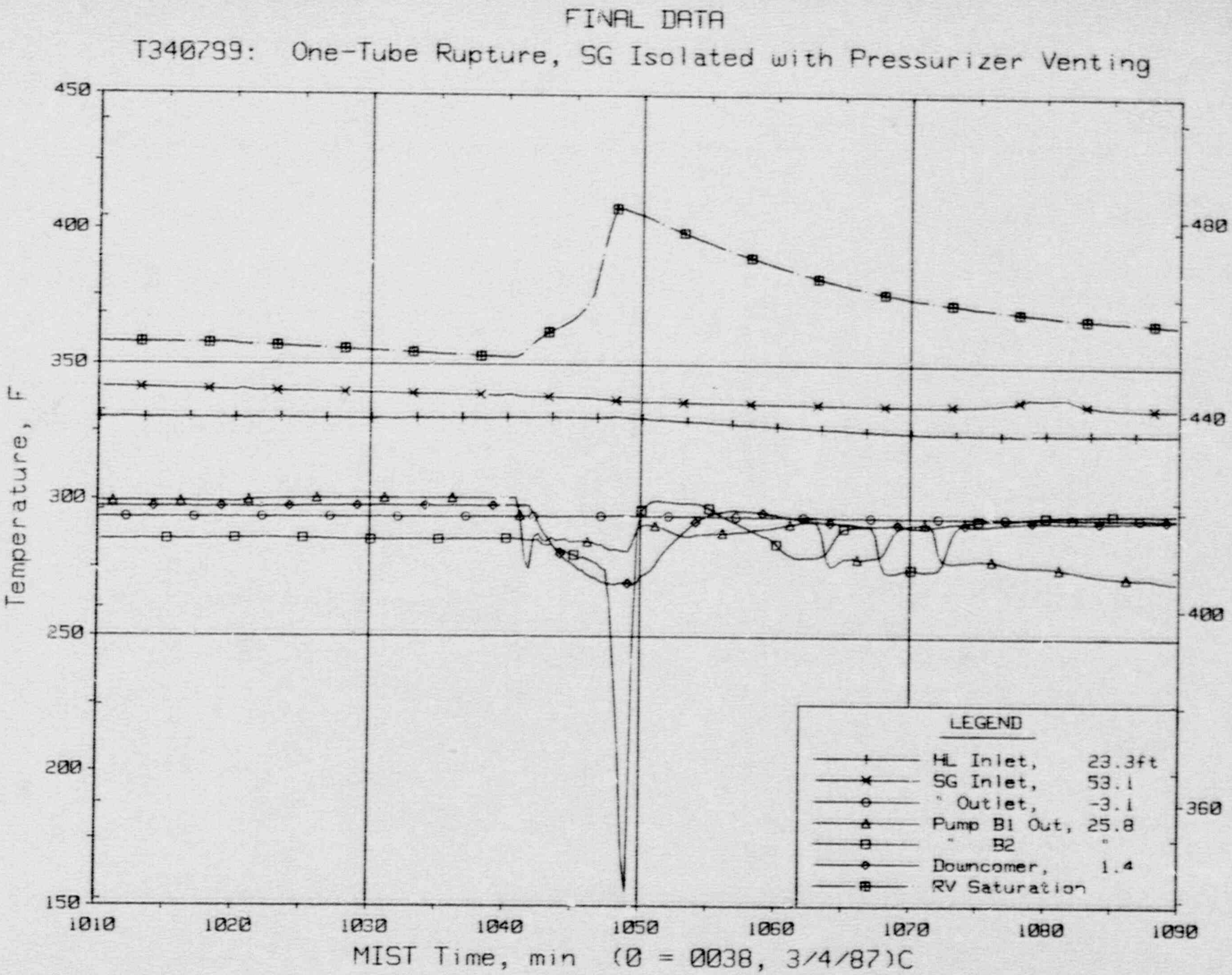


Figure 5.5.66. Loop B Primary Fluid Temperatures (RTDs)

Mon Aug 8 13:23:26 1988

PRRT2

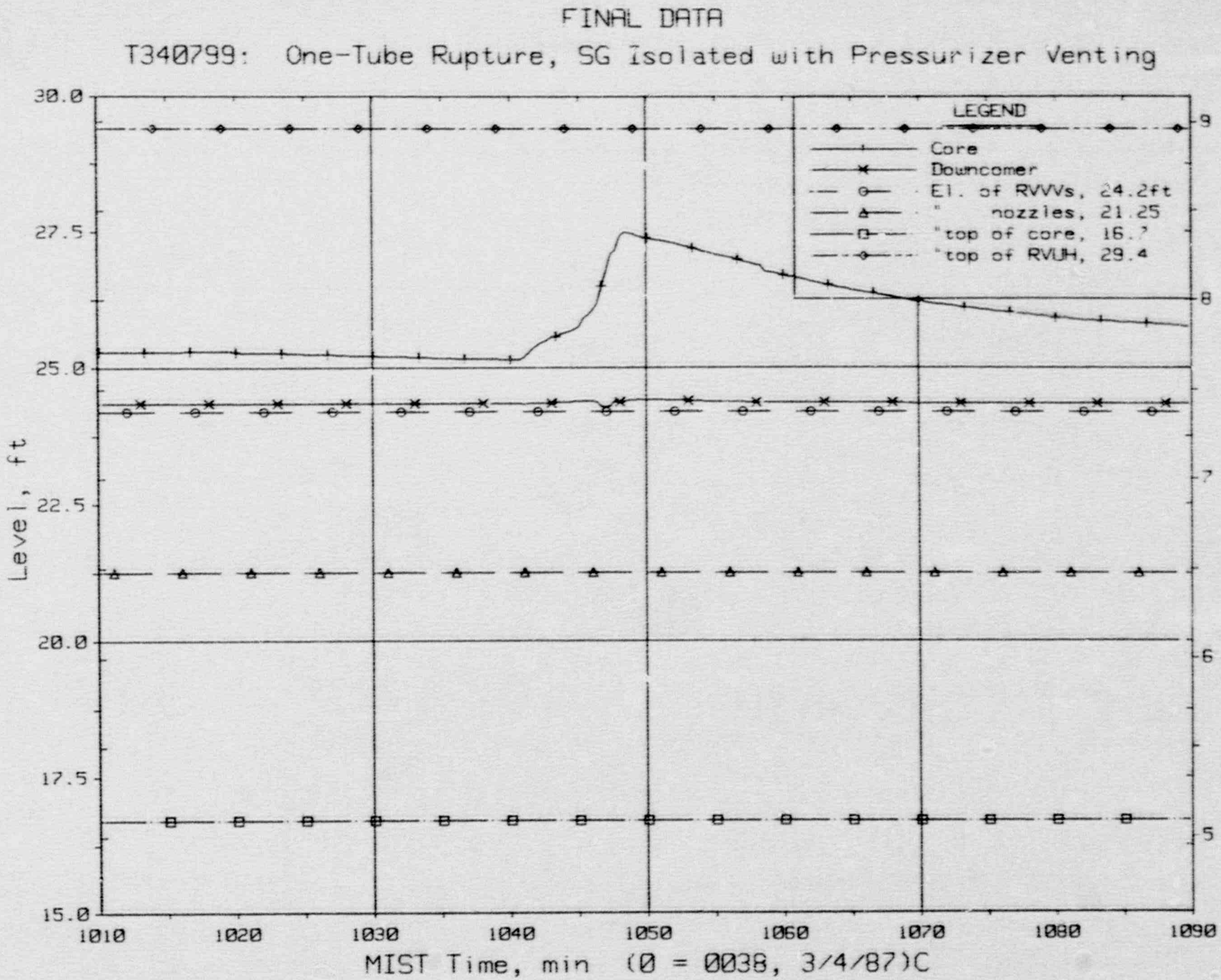


Figure 5.5.67. Core Region Collapsed Liquid Levels

Mon Aug 8 13:26:30 1988

COLV1

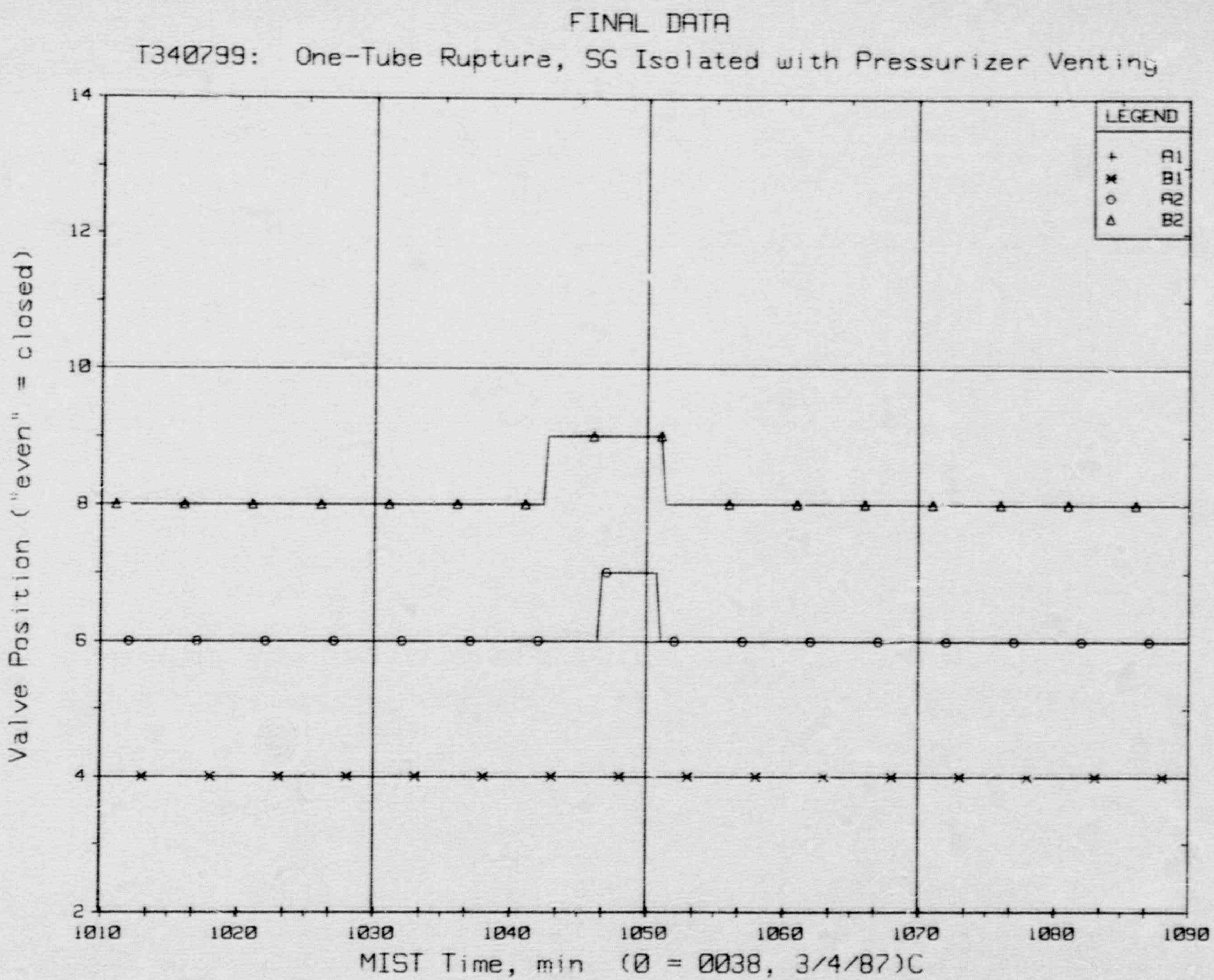


Figure 5.5.68. Reactor Vessel Vent Valve Positions

Mon Aug 8 13:29:05 1988

VVLS1

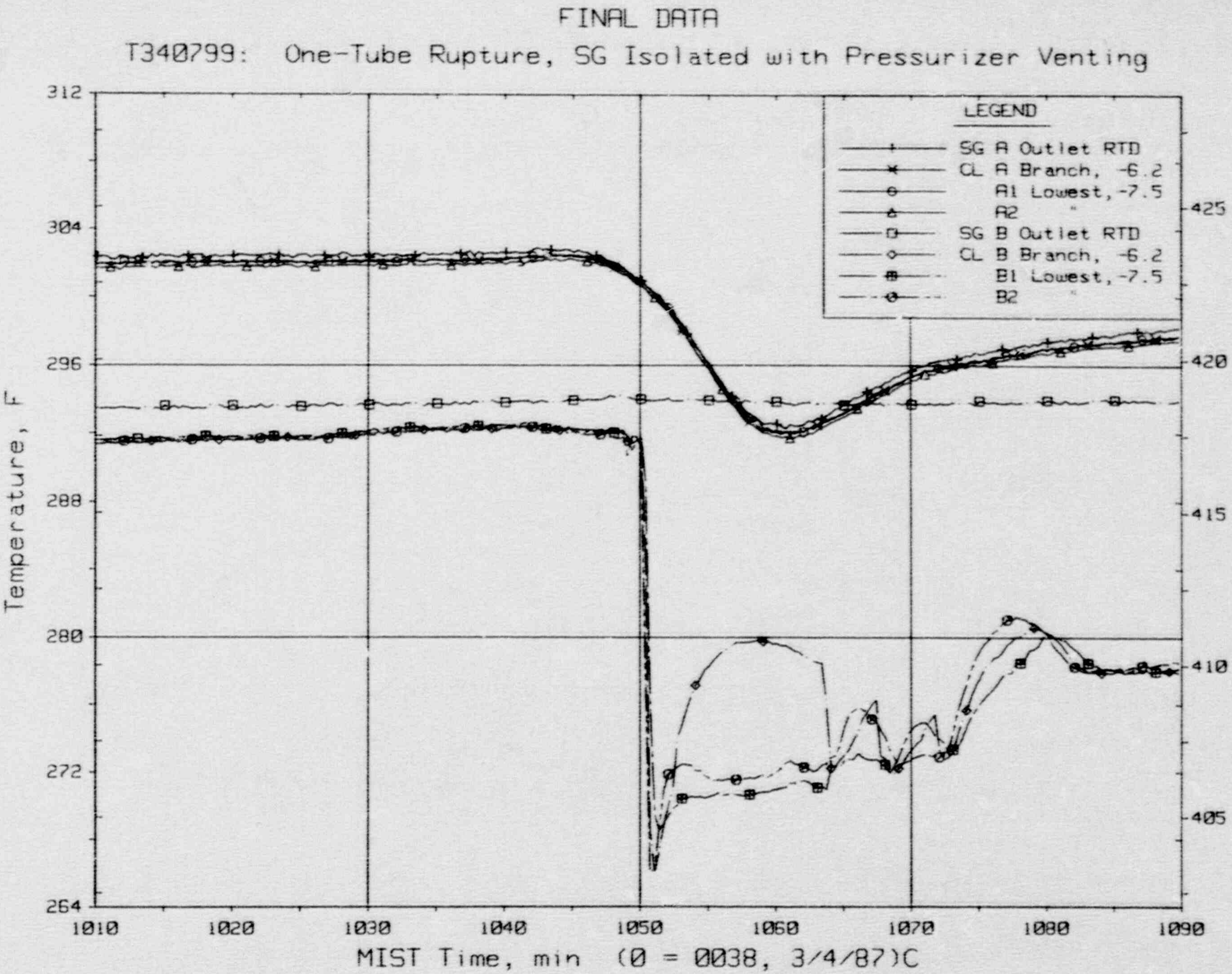


Figure 5.5.69. Cold Leg Suction Fluid Temperatures (CnTCs)

Mon Aug 8 13:32:02 1988

CLTC3

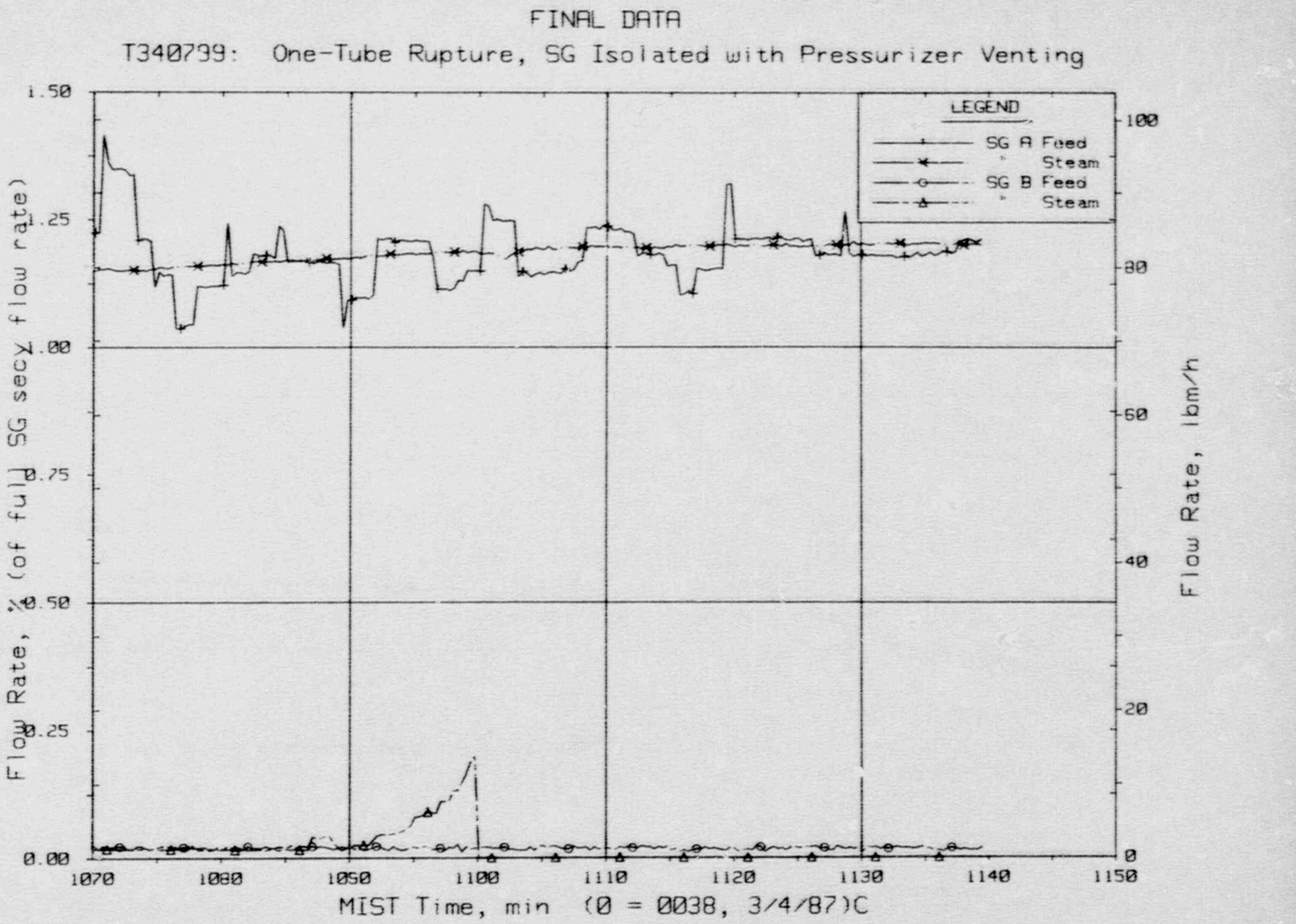


Figure 5.5.70. Secondary System Flow Rates .

Mon Aug 8 13:56:27 1988

SGFL2

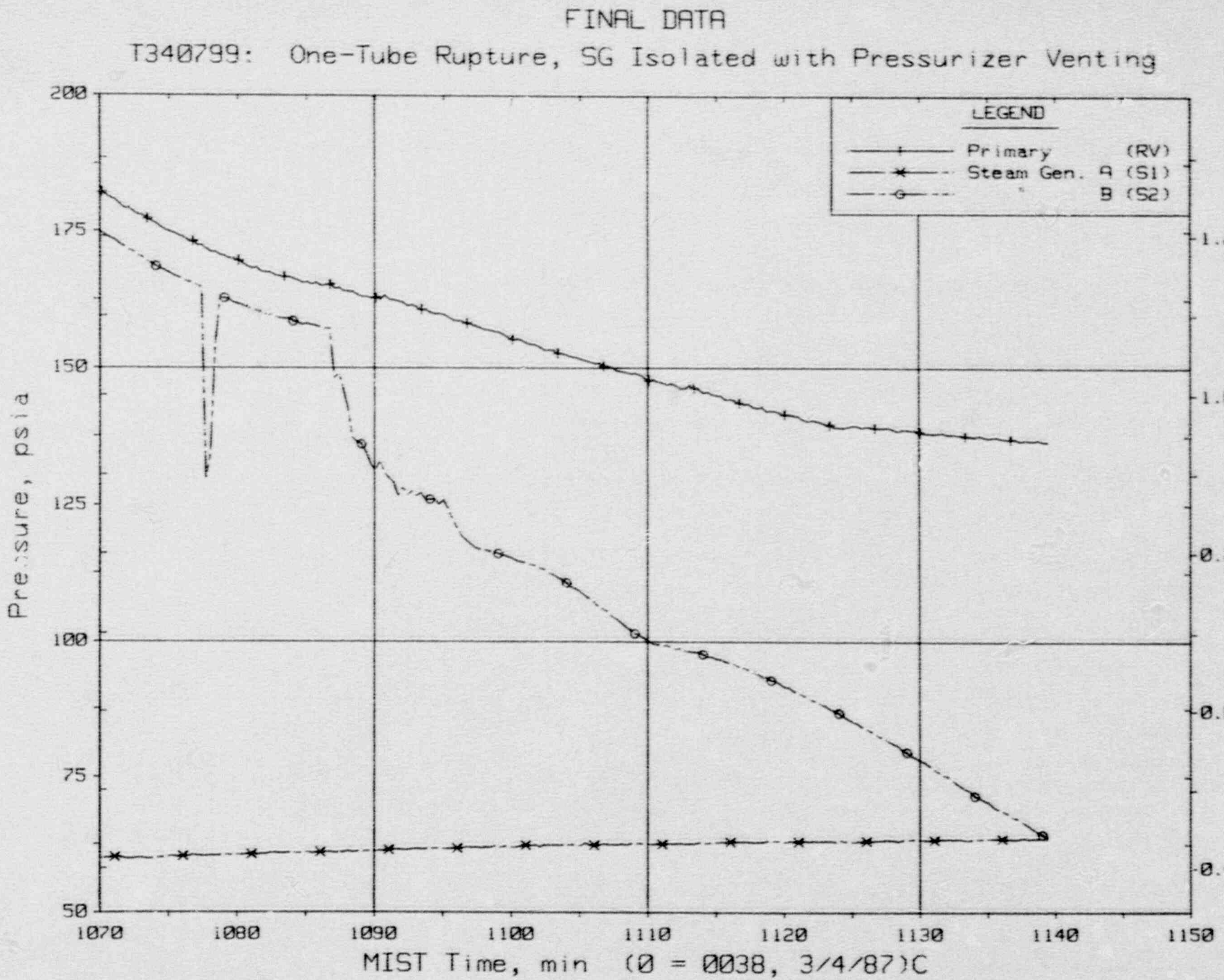


Figure 5.5.71. Primary and Secondary System Pressures (GP01s)

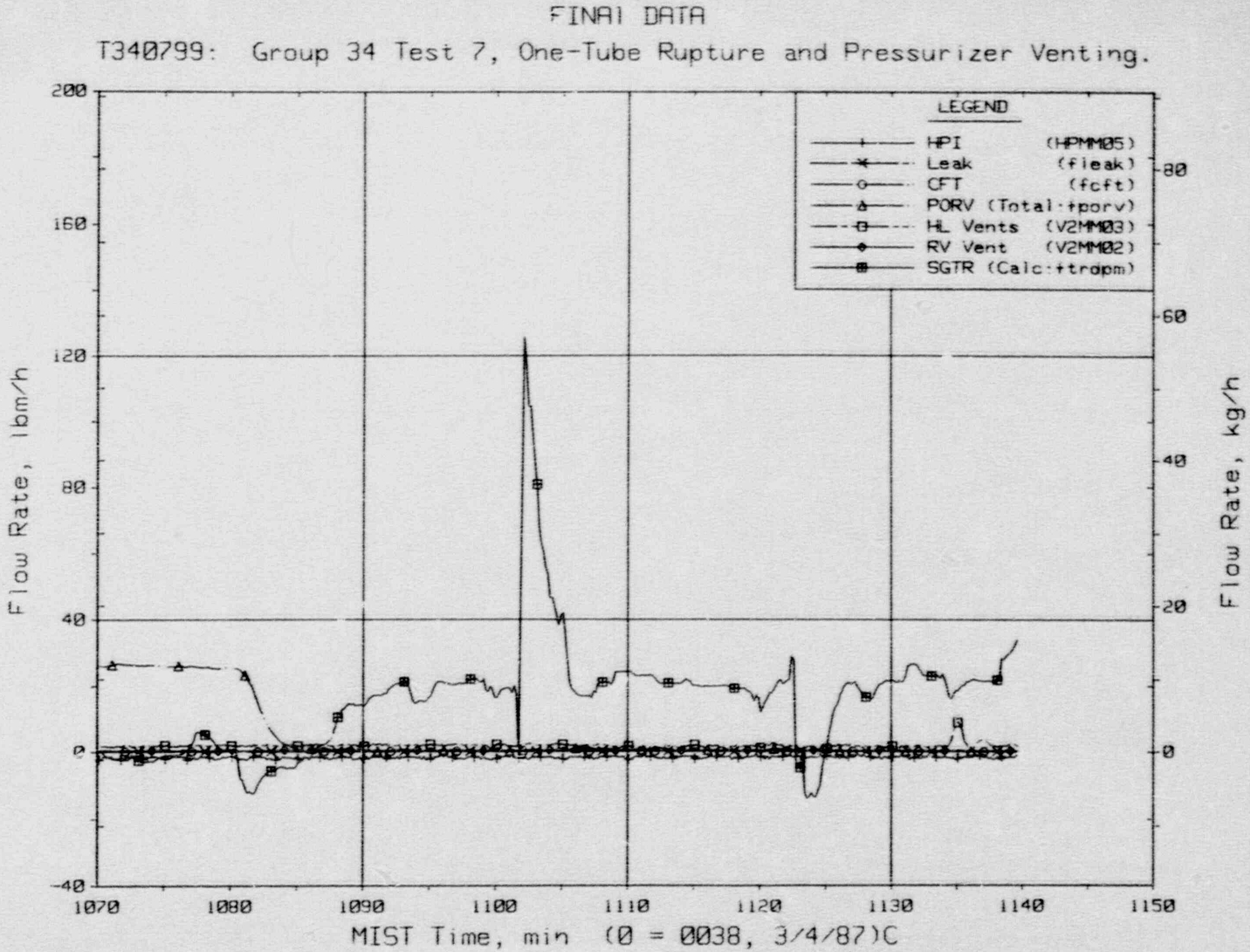


Figure 5.5.72. Primary System Boundary Flow Rates

Fri Nov 4 14:47:34 1988

PBFL4

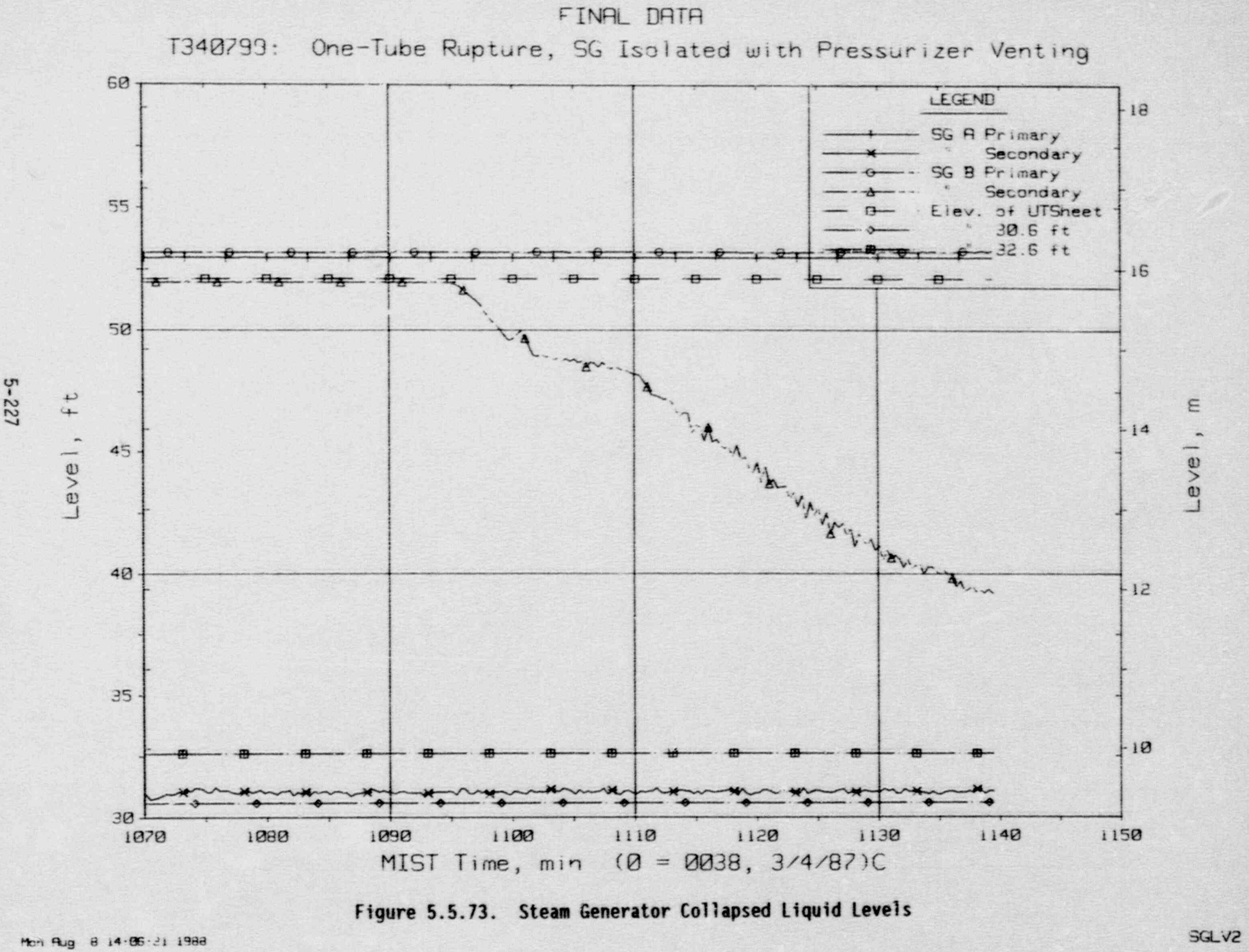


Figure 5.5.73. Steam Generator Collapsed Liquid Levels

5-228

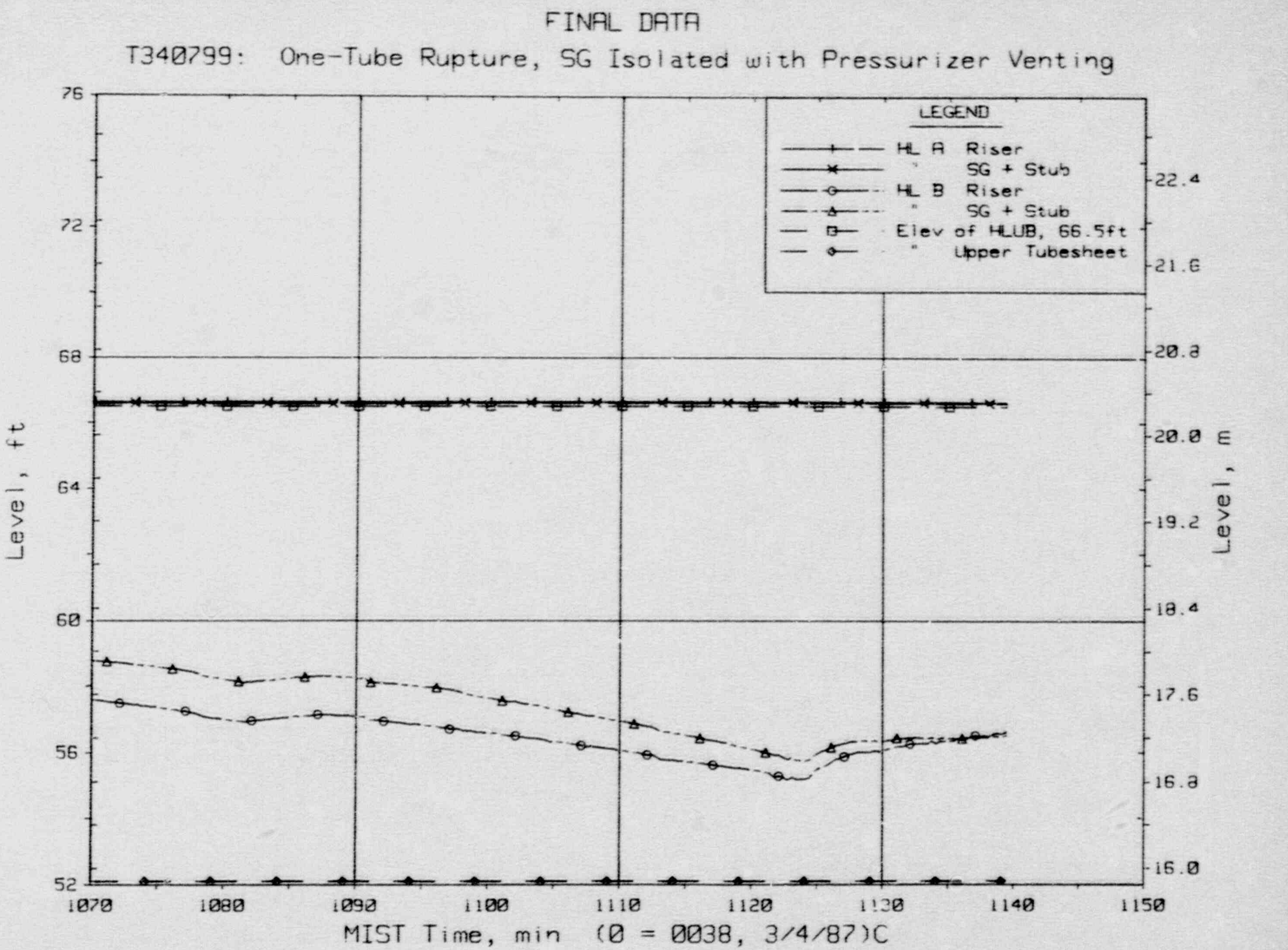


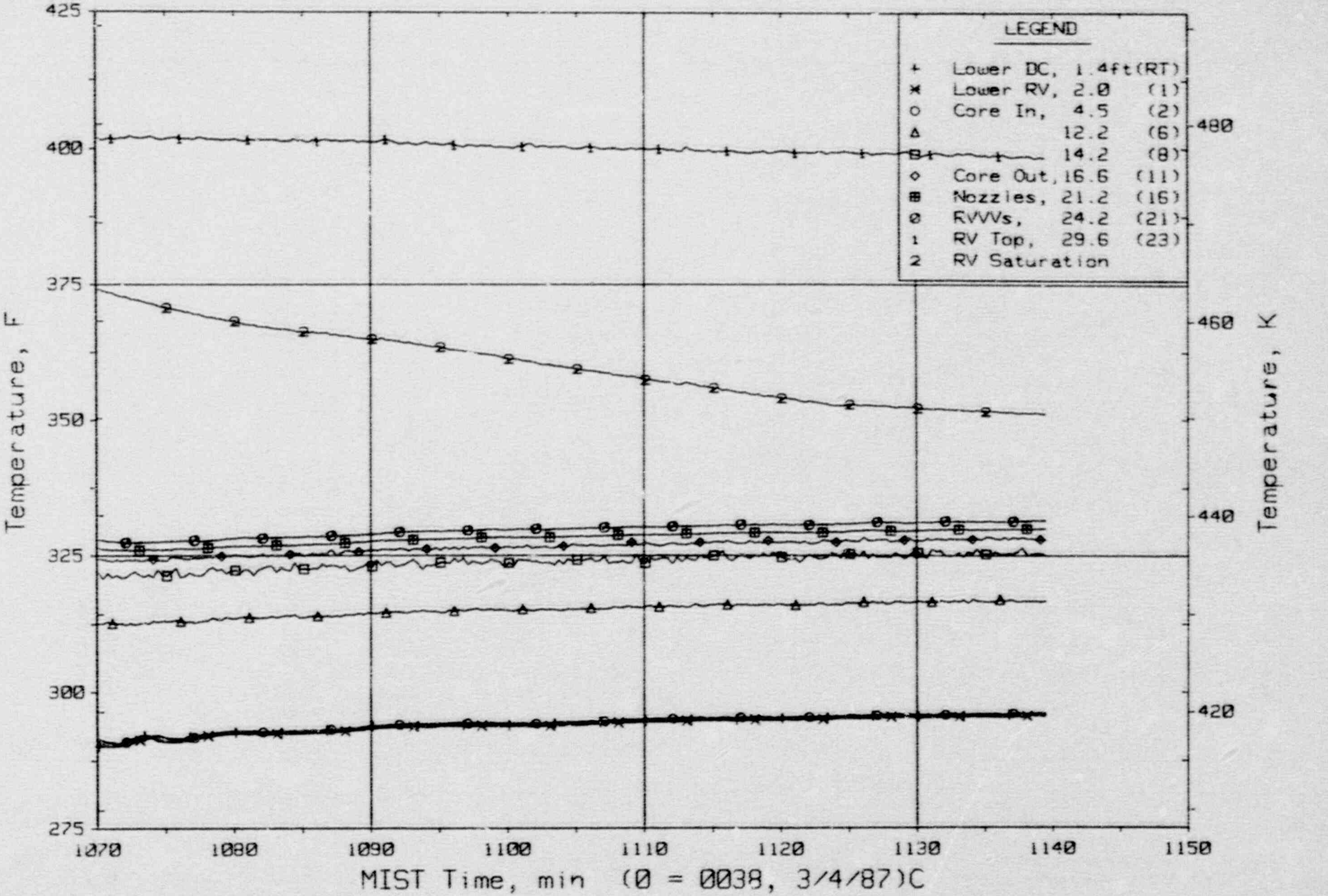
Figure 5.5.74. Hot Leg Riser and Stub Collapsed Liquid Levels

Mon Aug 8 14:09:27 1988

HLLV1

FINAL DATA

T340799: One-Tube Rupture, SG Isolated with Pressurizer Venting



5-229

Figure 5.5.75. Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs)

Mon Aug 8 14:13:33 1988

RVTC0

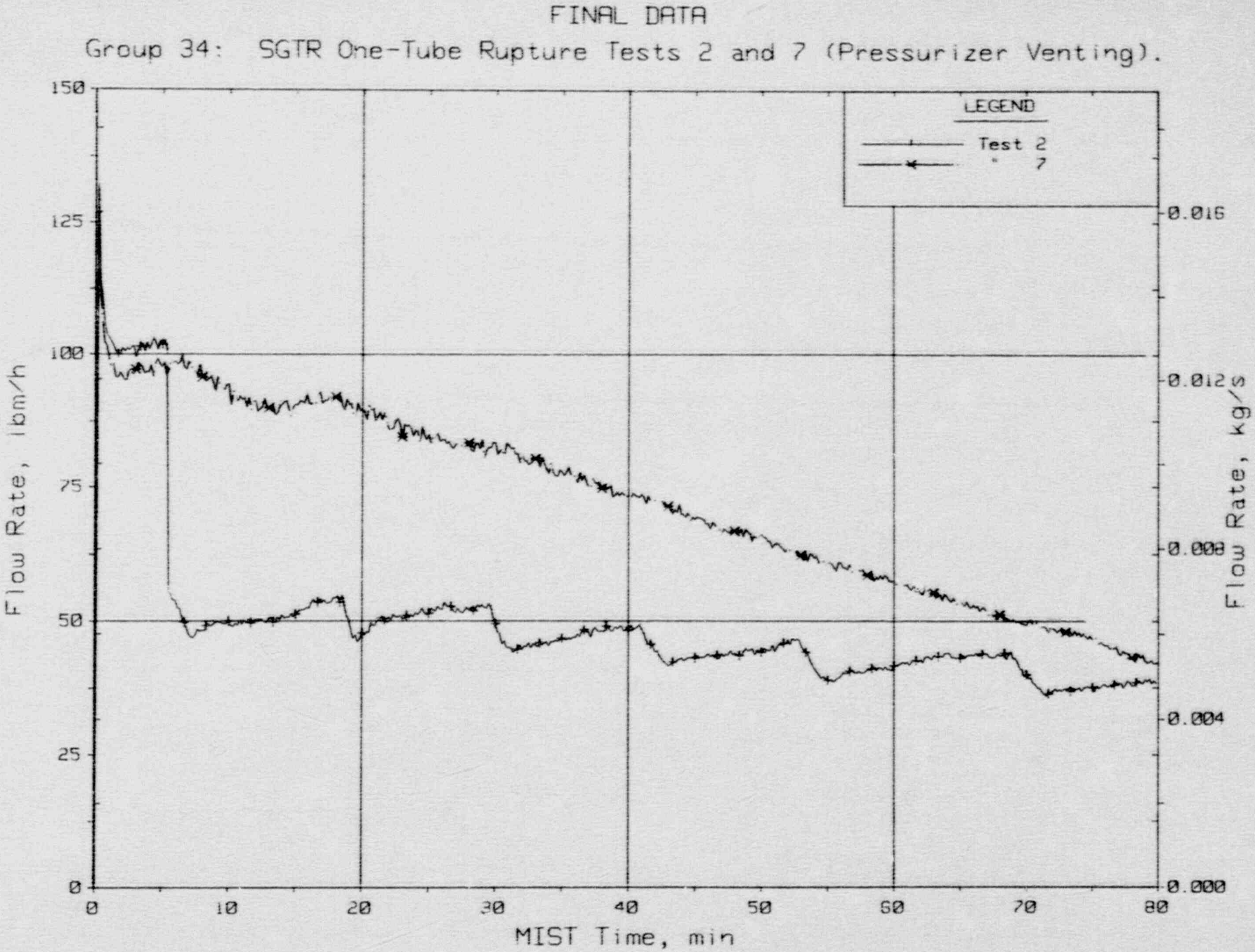


Figure 5.5.76. Tube Rupture Flow Rate

Tue Jul 26 15:35:11 1988

27PRFL3

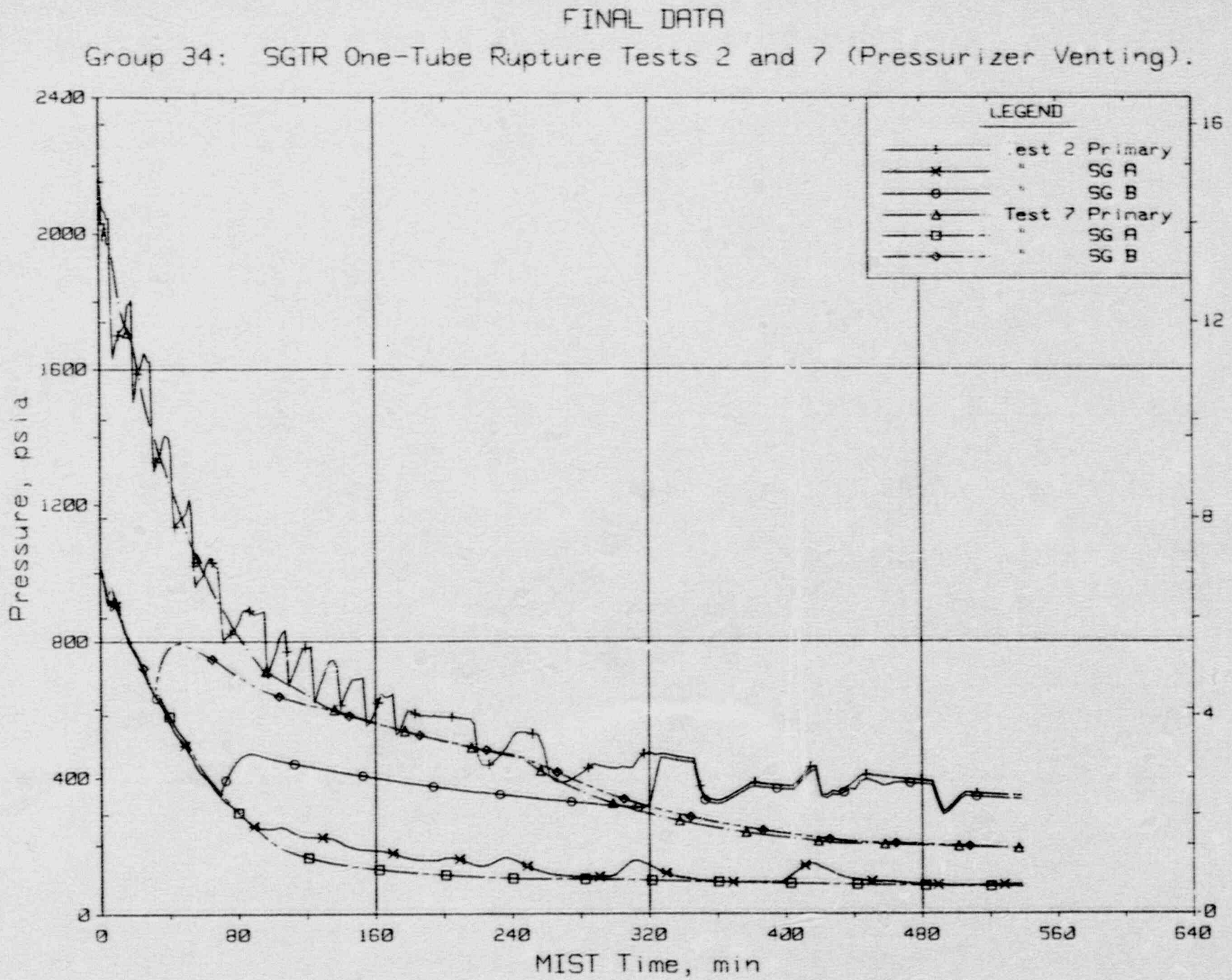


Figure 5.5.77. Primary and Secondary System Pressures (GP01s)

Tue Jul 26 15:38:54 1988

27PSGP1

FINAL DATA

Group 34: SGTR One-Tube Rupture Tests 2 and 7 (Pressurizer Venting).

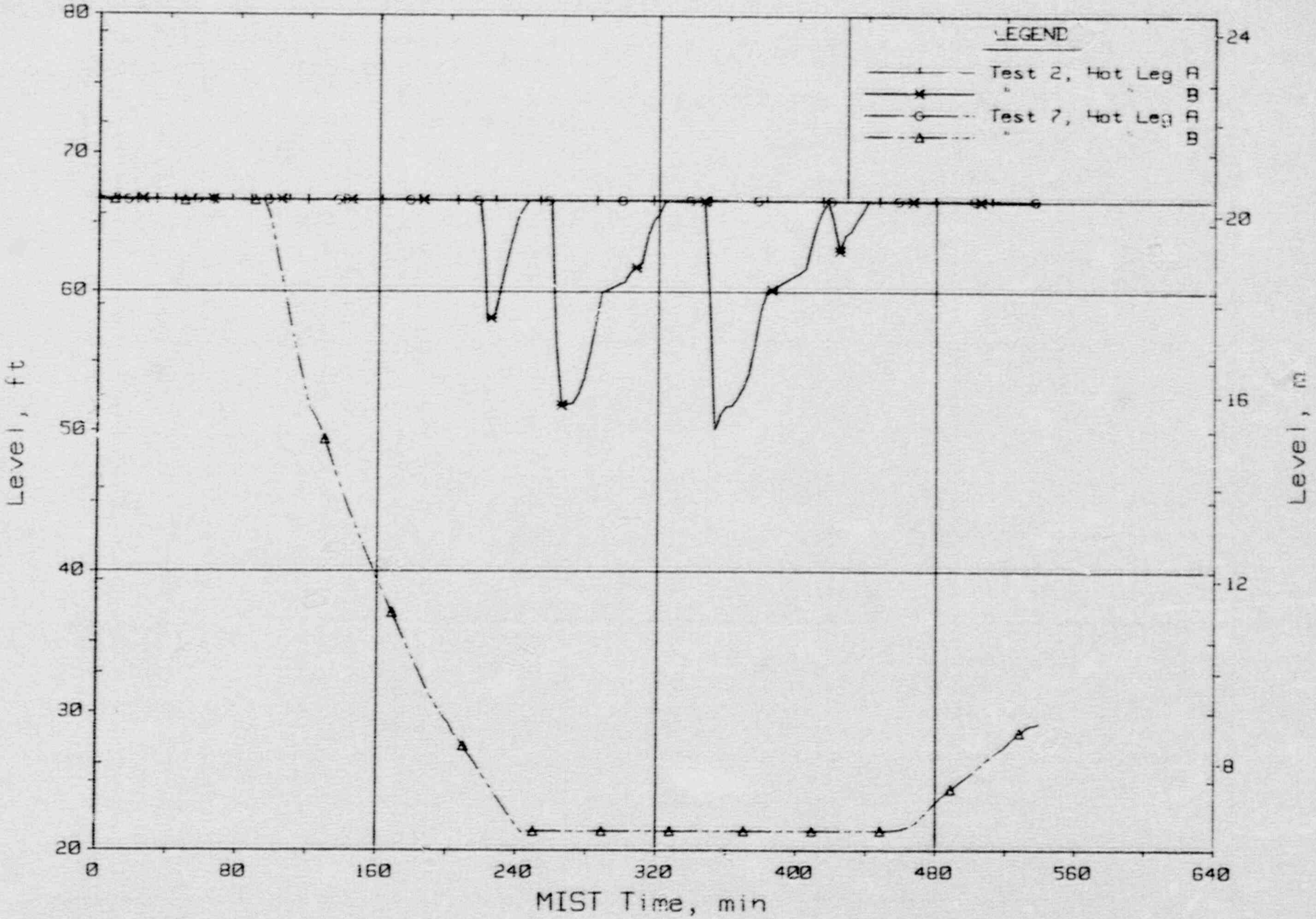


Figure 5.5.78. Hot Leg Collapsed Liquid Levels

Tue Jul 26 15:41:26 1988

27HLLV1

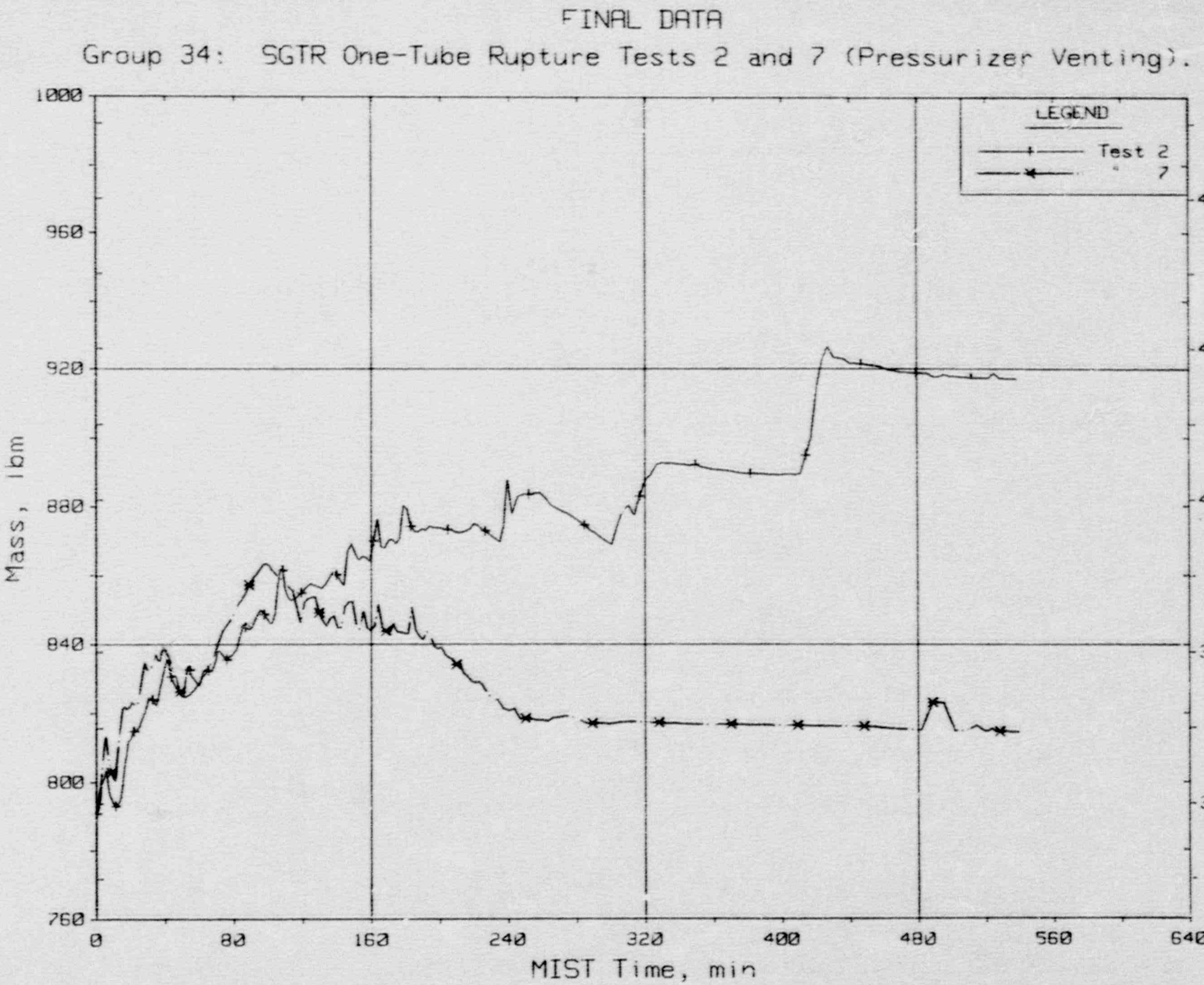


Figure 5.5.79. Primary System Total (Trended) Fluid Mass

Tue Jul 26 15:43:19 1988

27PRML1

FINAL DATA

Group 34: SGTR One-Tube Rupture Tests 2 and 7 (Pressurizer Venting).

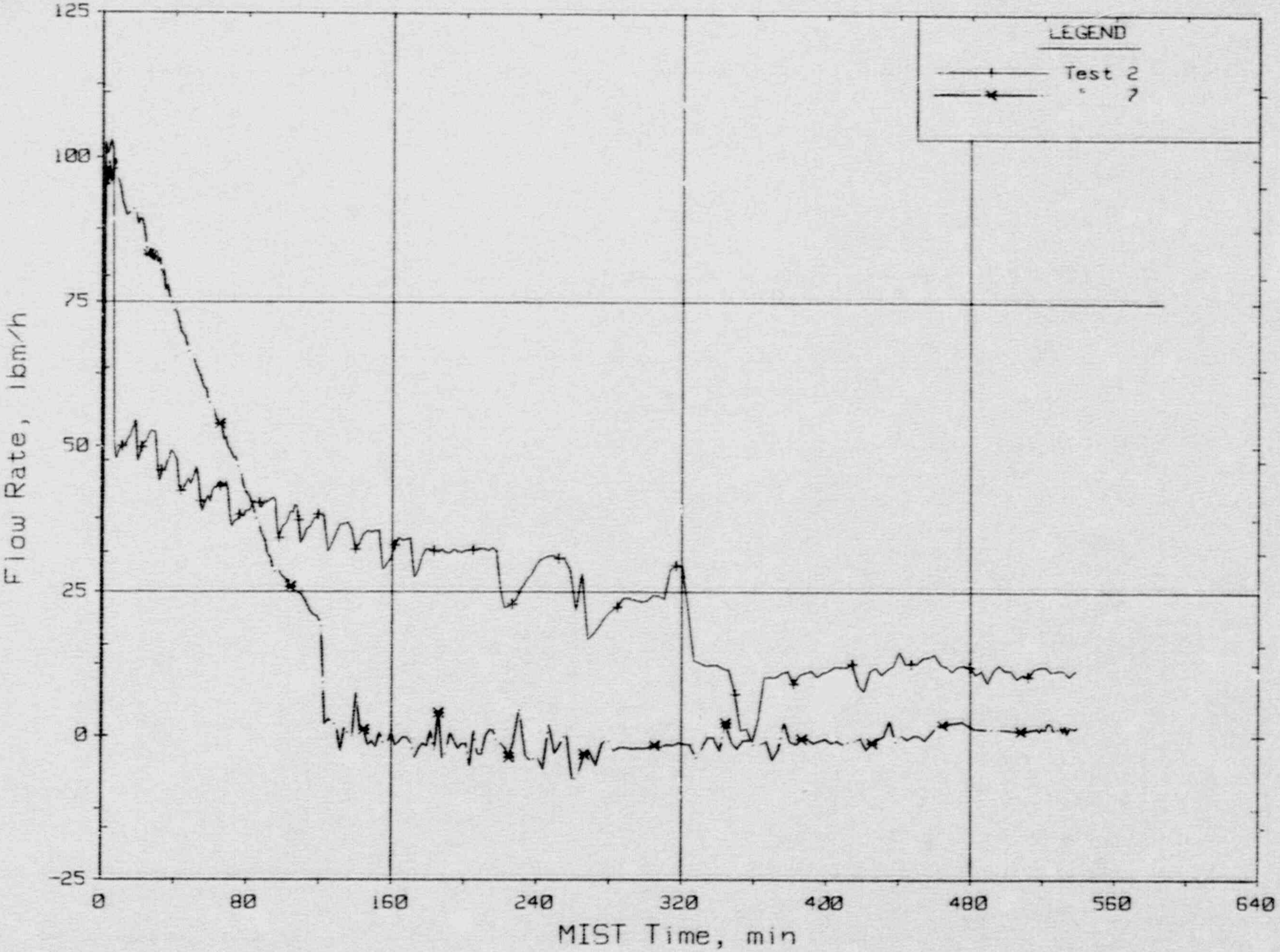


Figure 5.5.80. Tube Rupture Flow Rate

Tue Jul 26 15:45:16 1988

27PRFL3

5.6. Summary of Observations

The observations by test and test type are summarized in section 5.6.1. Noteworthy interactions observed in the 10-tube rupture tests and in the single-tube rupture tests are listed in subsections 5.6.2 and 5.6.3.

5.6.1. General Observations

The test results are summarized by test type.

Nominal Test 1

The double-ended rupture of 10 steam generator tubes at the top of generator B was simulated in Nominal Test 1. The steam generator was not isolated. Full-capacity HPI was available. The steam generator secondaries were depressurized to obtain a cooldown rate of 100F/h, but the low-capacity generator B steam pressure control valve was opened in an attempt to control the generator B secondary level at 20 feet.

The initial events were relatively rapid. The tube rupture flow rate initially exceeded 2000 lbm/h. The primary system depressurized from 2150 psia at rates exceeding 500 psi/min, obtaining saturation of the primary system fluid by approximately 1 minute. The loop A hot leg U-bend voided, loop A stagnated, and the steam generator A primary level descended sufficiently to obtain BCM during the controlled refill of the steam generator A secondary.

The steam generator A secondary was depressurized to obtain a (secondary) cooldown of 100F/h rather than being controlled based on a primary-to-secondary temperature difference as in the preceding test groups. Steam generator A thus became inactive following secondary refill and depressurization, and primary-to-secondary heat transfer shifted to steam generator B. The steam generator B secondary level had initially increased at approximately 6 ft/min due to the rupture. The generator B secondary level reached 20 ft before 3 minutes, triggering the manual full opening of its (low capacity) steam pressure control valve. (The steam pressure control circuit was being used to control level rather than pressure in the ruptured steam generator.) The low-capacity circuit was insufficient to maintain level, however. The generator B secondary level approached 30 feet at 11 minutes and 40 feet at

20 minutes. An elevation of 33 feet simulated the level at which filling of the steam lines would begin.

The tube rupture flow rate decreased markedly as the transient progressed. The initial decrease was due primarily to the depressurization of the primary system. Early in the transient, the rupture flow rate further decreased as the loop B voiding intermittently uncovered the rupture site.

The transient interactions were dominated by intermittent loop B primary-to-secondary heat transfer from 24 to approximately 50 minutes. The continuing primary system depressurization, due to both the rupture discharge and primary-to-steam generator B heat transfer, caused the generator A secondary to become a heat source to the primary system. The hot leg A riser and generator A primary voided quite rapidly, thereby sustaining the loop B hot leg riser level. Loop B hot leg U-bend flow periodically reinitiated, causing enhanced primary-to-steam generator B heat transfer, increases of the generator B secondary pressure, and brief loop B cold leg refill and flow. Whereas the loop B hot leg riser level periodically achieved the U-bend spillover elevation, the loop A hot leg riser voided completely by 25 minutes. This extreme inter-loop asymmetry was also reflected in the steam generator primary levels. The steam generator B primary level generally remained near the rupture elevation while the generator A primary continued to deplete, achieving 5 ft by 100 minutes. Although the generator B secondary briefly repressurized upon the reinitiation of primary-to-secondary heat transfer, the generator B secondary pressure generally depressurized through the manually full open (low capacity) steam pressure control valve. By 1 hour, the generator B secondary had been depressurized to 160 psia. The primary system pressure, following that of generator B, was 370 psia, and the generator A secondary pressure lagged at 420 psia.

The steam generator A secondary began to be steamed at 50 minutes to obtain the specified cooldown of 100F/h. The steam generator A-to-primary pressure difference thereby decreased and the intermittent loop B activity diminished. The continuing controlled depressurization of the generator A secondary finally obtained sustained primary-to-steam generator A secondary heat transfer beyond approximately 90 minutes. The loop A hot leg riser and steam generator primary began to refill beyond 100 minutes.

Steam Line Break, Test 5

Test 5 with a simulated steam line break behaved quite similarly to Nominal Test 1. Both tests experienced a rapid depressurization to saturation, periodic loop B activity, reverse heat transfer in loop A, and refill after the controlled depressurization of the generator A secondary had been resumed. A simulated high-elevation, double-ended rupture of 10 steam generator tubes was used in both tests. Whereas the high-capacity steam pressure control circuit was used to obtain a depressurization of 100 psi/min for generator B (simulating a steam line break) in Test 5, the low-capacity steaming circuit was opened fully early in Test 1 to control the secondary level. Thus, the timing of the earliest events differed, and the generator B secondary level first stabilized near 10 feet in Test 5 versus 20 feet in Test 1, but the system conditions of the 2 tests soon became similar. The larger flow area of the high-capacity circuit used in Test 5 had an increasing effect as the system pressures decreased. This was reflected in an earlier hot leg A refill and a higher final total primary system fluid mass in Nominal Test 1 compared to the Steam Line Break Test (Test 5).

Low-Elevation Rupture, Tests 3 and 6

Tests 3 and 6 were nominally identical. The low-elevation, double-ended rupture of 10 tubes was simulated. Although the initial transient interactions were virtually identical, the tests gradually differed as the transients progressed. The origin of the inter-test inequalities was apparently the difference between the initial metal temperatures. Several initial metal temperatures were lower in Repeat Test 6 than in Test 3. Among the largest differences, the surge line metal temperature at 20 feet was 22F lower in Test 6 than in Test 3, the reactor vessel upper flange was 19F lower, and the reactor vessel lower flange was 40F lower. The primary system thus depressurized slightly more readily in Test 6 than in Test 3. This difference led to a larger HPI flow rate and a smaller rupture flow rate in Test 6. These flow rate differences compounded with time, ultimately leading to sizeable differences between the tests. At test termination, their intact loop hot leg levels differed by almost 20 feet.

The interactions of Test 3, low-elevation rupture, and its repeat (Test 6) were not unlike those of Nominal Test 1. All 3 tests were characterized by a rapid initial primary system depressurization and a high rate of primary system fluid mass loss, approximately 2000 lbm/h. By 3 minutes, the rate of mass loss with the low-elevation ruptures began to exceed that with the high-elevation rupture, apparently in response to the greater subcooling of the low-elevation rupture site fluid. Whereas the primary system fluid mass gradually stabilized as primary system pressure was reduced with the low-elevation ruptures, with a high-elevation rupture the mass stabilized at a higher pressure, when the rupture site was intermittently uncovered. Intermittent loop B activity was encountered in both types of tests, but earlier with a low-elevation rupture. The system conditions of the 3 tests gradually converged as the transients progressed, although the primary system was more readily refilled with a high-elevation rupture than with a low-elevation rupture.

Steam Generator Isolation, Test 4

The affected steam generator was isolated in Test 4. As in the Nominal Test, the high-elevation and double-ended rupture of 10 tubes was simulated. Steam generator isolation occurred at 13 minutes, as the primary system pressure decreased through 950 psia. Until steam generator isolation, the Test 4 transient was similar to that of Nominal Test 1. Differences between the tests are attributable primarily to the 2-minute delay in activating HPI and AFW to steam generator A in Test 4.

Upon isolation, the affected generator quickly filled and repressurized to the primary system pressure. The unaffected generator had its feed available and could be steamed, but was rendered ineffective by intermediate loop A levels. The steam generator A primary level had refilled above the generator but the hot leg A riser level remained below the U-bend spillover elevation. Thus, the primary system and the isolated steam generator repressurized toward the actuation pressure of the secondary system operational safeties, forcing the termination of the test.

Single Tube Rupture, Tests 2 and 7

The high-elevation, single-ended rupture of 1 tube was simulated in Tests 2 and 7. Whereas the PORV was used to depressurize in Test 2, pressurizer venting was used in Test 7. The criteria for isolation of the affected generator were also somewhat different between tests. Specific cooldown and depressurization steps were implemented late in Test 7.

Test 2

The rupture flow area decreased by approximately 50% in Test 2, 5 minutes after rupture initiation and near the time of the first PORV actuation to reduce pressure. The rupture simulation apparently became partially blocked and remained so, judging by the primary-to-secondary pressure differential available to drive flow versus the apparent flow rate. Although this partial blockage encumbers the code simulation of the test, the basic interactions were unaffected. With and without the blockage, the rupture flow was insufficient to rapidly depressurize the primary system (requiring that the primary system be depressurized through operator actions), and the available HPI flow rate was far greater than the rupture flow rate (requiring that HPI be throttled).

The 2-loop cooldown continued while the affected steam generator remained in service. The rupture had only a minor impact, such as the unequal feed rates to the generators. The operator periodically opened the PORV to control the SCM and adjusted HPI to maintain pressurizer level. The pressurizer heaters were also used for SCM control.

Both steam generators were depressurized to obtain a cooldown of 100F/h. The PORV actuation to control the SCM at 30 minutes caused the reactor vessel upper head to saturate and to void. The voided volume expanded with subsequent PORV actuations. The upper downcomer began to void upon the PORV actuation at 69 minutes.

The affected generator was isolated at 70 minutes, as the primary system was depressurized below 950 psia. The steam generator B secondary pressure and level began to increase. The loop B primary flow rate declined, but the cooldown continued using loop A. The steam generator B secondary pressure stabilized at saturation corresponding to the loop B hot leg fluid temperature. The reduced primary-to-steam generator B pressure decreased the tube rupture flow rate as well as the rate of level rise in the isolated secondary. The lower-elevation fluid in the isolated steam generator cooled somewhat through inter-cold leg flow.

The nearly stagnant loop B fluid was largely unresponsive to the loop A cooldown. The loop B hot leg saturated and voided during the PORV actuations beyond 220 minutes. This voiding displaced liquid into the pressurizer, requiring adjustments to the HPI flow rate, which had been throttled to maintain pressurizer level.

The loop B hot leg riser and steam generator primary gradually refilled during the repressurization following PORV closure. This alternate voiding and refill of loop B was repeated during the subsequent PORV actuations. The accompanying fluid displacements served to cool loop B. This cooling was most apparent over the middle riser elevations, which were cooled by the upward displacement of relatively cool fluid from the bottom of the riser.

The isolated steam generator secondary finally filled completely at 318 minutes. The generator B secondary pressure then rose to the current primary system pressure, 450 psia. The single-loop cooldown and periodic PORV actuations were continued through test termination. The loop B hot leg fluid subcooled at 440 minutes, following PORV closure and loop B refill. Upon test termination at 9 hours, the loop A flow rate remained at 2.7% of scaled full flow and generator A remained active. The primary system pressure and the pressure in the isolated generator secondary were 350 psia. The steam generator A secondary pressure was 75 psia and near the facility minimum pressure. The loop B hot leg fluid remained approximately 25F subcooled.

Test 7

The sustained 2-loop natural circulation cooldown early in Test 7 resembled that of Test 2. Although the primary system pressure was reduced incrementally in Test 2 and gradually in Test 7, their general pressure trends remained similar. The affected steam generator was isolated at a hot leg fluid temperature of 525F in Test 7, delaying isolation until 70 minutes.

Whereas loop B alternately voided and refilled through periodic PORV actuations in Test 2, loop B voided gradually and continuously in Test 7, completely emptying the hot leg B riser by 6 hours. The rupture elevation was uncovered near 2 hours, causing the primary and steam generator B secondary pressures to approximately equalize. The attendant decrease of the rupture flow rate caused the steam generator B secondary level to stabilize at 41 feet.

The loop B voiding caused the pressurizer to fill near 3 hours. The change of state of the fluid at the pressurizer vent slowed the primary system depressurization rate. When the loop B hot leg riser emptied near 6 hours, the state of the pressurizer vent fluid apparently reverted to vapor, the primary system depressurization rate increased, and a discernible steam generator B secondary-to-primary system pressure difference developed. The steam generator B secondary began to discharge fluid to the primary.

The primary system single-loop cooldown slowed as the steam generator A secondary pressure approached the facility minimum pressure. The SCM decreased as the depressurization due to pressurizer venting overtook the cooldown. Thus, the pressurizer vent was closed at 430 minutes with an SCM of 25F. Upon closure of the pressurizer vent, the primary system pressure stabilized and the pressurizer began to transfer liquid to loop B. HPI was reactivated at 9 hours to stabilize the pressurizer level, and positive (primary-to-secondary) rupture flow rate reinitiated. The generator B primary level returned to the rupture site at 12 hours. The specific depressurization and cooldown steps were instituted at 15 hours. At this time, the primary and secondary pressures of steam generators B (isolated) and A (active), respectively, were 157, 132, and 66 psia. The SCM was 24F. The core inlet fluid temperature was 302F, 3F above the saturation temperature at the generator A secondary pressure. The core fluid temperature rise was 36F at a core power level of approximately 1% of scaled full power. The decay contribution was 0.6% of scaled full power, and the remaining 0.4% was the (constant) power augmentation used to offset uncompensated heat losses to ambient. At 15 hours, the pressurizer vent remained closed and HPI remained throttled to maintain pressurizer level. The steam generator B secondary was nearly liquid full, and the loop B hot leg riser and stub levels remained near the rupture elevation.

The steam generator A secondary level was increased to 31.6 ft at 15 hours using full-capacity AFW flow rate. The imposed perturbation quickly subsided, the net effect was an approximately 5F reduction of the core inlet and outlet fluid temperatures.

The loop B hot leg high-point vent was opened at 16 hours. Although by 17 hours the rupture site was recovered, the steam generator B secondary was filled completely and pressurized toward primary system pressure and the loop B hot leg levels rose 5 feet, there was virtually no net change of the general primary system fluid temperatures and pressure.

The pressurizer vent was reopened and HPI was interrupted at 17 hours. The primary system pressure and the SCM slowly decreased and, at 17 hours and 20 minutes, full-capacity HPI was actuated. The SCM quickly rose to 100F and, at 17 hours and 28 minutes, HPI was again interrupted. Although the system conditions were altered generally by the cyclic operation of HPI, they quickly reverted to their earlier values. At 18 hours, the steam generator A secondary pressure was 5 psi lower than at 17 hours, and the core inlet and outlet fluid temperatures were 4F lower.

The isolated steam generator was reactivated and the pressurizer vent was closed at 18 hours. The steam generator B secondary pressure quickly decreased to that of generator A (64 psia) and the rupture flow rate reactivated. The loop B hot leg levels remained below the U-bend spillover, however, so loop B remained inactive and the steam generator B activity had little effect. The final core inlet and outlet fluid temperatures, at 19 hours, were 3F higher than they had been at 18 hours.

The system conditions throughout the four depressurization and cooldown steps were set chiefly by the (minimum) steam generator secondary pressure and by core power. The core inlet fluid temperature corresponded closely to saturation temperature at the secondary pressure of the active steam generator. The core outlet fluid temperature reflected the inlet temperature plus the rise due to core energy deposition at the natural circulation flow rate. This core fluid temperature rise at test termination was 34F, of which 20F was attributable to the decay power contribution.

5.6.2. Noteworthy Interactions -- 10-Tube Rupture

Rapid Initial Response

The initial events with a double-ended 10-tube rupture were quite rapid. The primary depressurized at rates exceeding 500 psi/min, obtaining saturation and voiding within 1 minute. The affected steam generator secondary level reached the 20-ft control point by 3 minutes. The initial primary system fluid mass loss rate, in excess of 2000 lbm/h, was sufficient to obtain primary vapor condensation within the unaffected generator (BCM) during the initial steam generator secondary refill.

Steam Generator Secondary Overfill

The level rise was halted near 40 feet, i.e., above the simulated level at which the steam lines would begin to fill. The level rise was halted as the generator steaming rate matched the tube rupture flow rate. Nominally, the low-capacity steam pressure control valve was opened fully to control level. The tube rupture flow rate was reduced as the primary system depressurized and, with a high-elevation rupture, as the rupture site intermittently uncovered.

Inter-Loop Asymmetries

Extreme inter-loop asymmetries were experienced midway in the 10-tube rupture transients, from about 20 to 50 minutes. The intact steam generator became a heat source as the primary system depressurized through both rupture discharge and intermittent heat transfer to the affected generator. The intact loop voided extensively, completely draining the hot leg riser and depleting the steam generator primary level toward the lower tubesheet. The levels in the affected loop were sustained by the intact-loop voiding and by the flow toward the high-elevation rupture. The affected-loop hot leg level intermittently achieved the hot leg U-bend spillover elevation, reactivating flow in the loop containing the ruptured generator. The attendant heat transfer from the primary to the ruptured steam generator briefly repressurized the affected steam generator secondary and augmented the primary system depressurization. The inter-loop asymmetries subsided as the intact generator was depressurized to sustain its cooldown of 100F/h.

Stabilization of Primary System Total Fluid Mass

The total primary fluid mass stabilized as HPI plus the CFT discharge equalled the rupture flow rate. This generally occurred after the intact generator had been reactivated and was no longer a heat source, approximately 2 hours after rupture initiation. After the intact generator had become active, the intact hot leg and steam generator primary were readily refilled.

Effects of the Simulated Steam Line Break

The 100-psi/min initial depressurization of the generator B secondary used to simulate a steam line break was nearly replicated in the large-rupture tests when the steam pressure control valve was fully opened to control secondary level. Thus, the behavior of the Steam Line Break Test resembled that of the Nominal Test. The secondary level in the ruptured generator was initially stabilized at a lower level in the Steam Line Break Test and the primary system was more readily refilled in the Nominal Test.

Effects of Rupture Elevation

The system depressurized more rapidly and lost fluid mass more rapidly with a low-elevation rupture than with a high-elevation rupture. Whereas the rupture mass flow rate decreased with rupture uncovery in the high-elevation cases, the flow rate decrease was obtained with the continuing primary system depressurization in the low-elevation case.

Sensitivity of Integral System Transient Trends To Minor Changes of Conditions

Long-term integral system conditions are sensitive to seemingly minor changes of initial conditions. Low-Elevation Rupture Test 3 and its repeat, Test 6, were conducted nominally identically. Several initial metal temperatures were 20 to 40F colder in Test 6 than in Test 4, however. The lower metal temperatures of Test 6 obtained a slightly more rapid primary system depressurization, and therefore increased HPI flow rates and a decreased rupture mass flow rate. These slight flow rate differences accumulated with time, obtaining discernible inter-test differences in total fluid mass and hot leg levels.

Repressurization of Isolated Steam Generator

Following the isolation of the generator having a high-elevation rupture of 10 tubes, the primary and isolated steam generator secondary repressurized through 1350 psia, precipitating test termination. The ruptured generator was isolated below 950 psia primary system pressure. The isolated generator secondary repressurized toward the primary system pressure, then both depressurized through heat transfer to the intact generator. This heat transfer was interrupted, however, when the intact loop levels rose above the generator while the riser level remained below the U-bend spillover elevation. This interruption of heat transfer precipitated the aformentioned repressurization.

5.6.3. Noteworthy Interactions -- Rupture of 1 Tube

Minor Effects of the Rupture

The single-ended rupture of 1 tube obtained minor inequalities of the feed and steaming rates, but had no major impact on the initial two-loop natural circulation cooldown. The SCM was readily controlled by depressurization using either the pressurizer vent or intermittent PORV actuation.

Reactor Vessel Voiding

The reactor vessel upper head temperatures were little affected by the two-loop natural circulation cooldown. The reactor vessel upper head fluid saturated and voided as the primary system was depressurized to maintain the SCM during the cooldown The upper head void expanded with the continuing reductions of pressure. The two-loop cooldown was unaffected by the head voiding.

Effects of Steam Generator Isolation

Upon isolation of the affected generator, the affected loop flow stagnated, the cooling of the affected loop fluid largely ceased, the generator secondary pressurized to saturation at the adjacent primary fluid temperature, and the secondary level slowly rose. The intact generator assumed the total heat transfer load and the cooldown continued in single-loop natural circulation. In the isolated steam generator, the lower-elevation fluid was cooled somewhat through inter-cold leg flow.

Voiding of the Affected Loop

The affected loop hot leg fluid saturated and voided upon continued depressurization to control SCM. This voiding was intermittent with PORV pressure control, but continuous and pronounced with pressurizer venting. The fluid displacement accompanying voiding elevated the pressurizer level, requiring additional throttling of HPI. The single-loop cooldown continued to effectively cool the core. Primary depressurization continued to control the SCM.

Equilibration of Pressures

The isolated steam generator pressure increased to the primary system pressure when the isolated secondary filled completely and also after the rupture site was uncovered. The equalization of pressures stopped the rupture flow. Reverse rupture flow, from the secondary to the primary, occurred when the primary system was depressurized, after the primary and isolated secondary system pressures had equilibrated.

Long-Term Primary System Conditions

The long-term primary system conditions were dictated largely by the steam generator secondary pressure and decay power. The core inlet temperature was very closely equal to the saturation temperature at the (active) generator secondary pressure. The core temperature rise corresponded to the core decay heat deposition and the core flow rate. Abrupt alterations of boundary system conditions did alter the primary system conditions, but these condition changes were temporary.

6. SUMMARY

MIST Test Group 34 consisted of 7 tube rupture transients. The single-ended rupture of 1 tube was simulated in Tests 2 and 7, and the remaining 5 tests addressed the double-ended rupture of 10 tubes. The ruptures were nominally located at the top of steam generator B; this location was shifted to the bottom of steam generator B in Test 3 and its repeat, Test 6. The affected steam generator was isolated in both single-tube rupture tests and in Test 4. Finally, a steam line break and a 10-tube rupture were simulated in Test 5. The tests were generally conducted as specified. The MIST interactions are of intrinsic interest because they may provide insight into expected plant behavior. MIST was necessarily atypical of a plant in certain important respects, however. The MIST interactions therefore are not to be applied directly to a plant.

The tests simulating 10 ruptured tubes experienced relatively rapid initial depressurization and mass-depletion transients. Primary system saturation occurred in approximately 1 minute, and the intact loop voided sufficiently to obtain boiler-condenser mode (BCM) depressurization during the initial refill of the intact steam generator secondary. The level in the ruptured steam generator quickly rose to 20 feet, prompting the full opening of the low-capacity steam pressure control valves to control level. The level generally continued to rise, however, exceeding the simulated overfill elevation of 33 feet and stabilizing near 40 feet in the ruptured steam generator secondary.

The mid-term transients of the 10-tube rupture tests were characterized by extreme inter-loop asymmetries. Because the primary system depressurization caused reverse heat transfer in the intact steam generator, the intact loop voided extensively, helping to sustain the levels in the affected loop containing the ruptured steam generator. The hot leg levels of the affected loop intermittently achieved the U-bend spillover elevation, reinitiating flow in the affected loop and obtaining primary-to-ruptured steam generator heat transfer. These interactions augmented the depressurization of the primary system and thus helped to mitigate the rupture flow rate. The inter-loop asymmetries abated as the controlled depressurization of the intact steam generator took effect at about 1 hour.

The 10-tube tests with altered break location (low versus high) and with the simulated steam line break obtained variations in timing and degree of interactions rather than causing singular interactions. Minor changes in initial conditions caused marked differences of the accumulated conditions between nominally identical tests in the two tests having low-elevation tube ruptures.

The steam generator secondary repressurized beyond the secondary safety lift pressure following isolation of the ruptured generator. The intact steam generator was operable, but primary-to-secondary heat transfer was inhibited by intermediate hot leg levels -- above the steam generator but below the U-bend spillover.

The high elevation, single-ended rupture of 1 steam generator tube was simulated in Tests 2 and 7. Power-operated relief valve (PORV) depressurizations were used in Test 2, and continuous pressurizer venting was used in Test 7. Depressurization differences notwithstanding, the subcooling margin (SCM) was controlled and the two-loop cooldown was maintained in both tests. Upon isolation of the affected steam generator, the cooldowns were continued using 1 loop. Whereas the hot leg in the affected loop voided intermittently in the test using PORV actuations for depressurization, the affected loop voided continuously and extensively in the continuous-venting test. The loop voiding altered the pressurizer level, but the single-loop cooldown continued and the primary system was readily depressurized. The long-term primary fluid conditions were set mainly by the pressure in the active steam generator secondary and by core power. Abrupt changes of the boundary system controls only temporarily altered the primary system fluid conditions.

7. REFERENCES

1. H. R. Carter and J. R. Gloudemans, "An Experimental Study of the Post-Small Break Loss-of-Coolant Accident Phenomena in a Scaled Babcock & Wilcox System," NUREG/CP-0058, Vol. 1, pp. 113-135. Proceedings of the U.S. Nuclear Regulatory Commission, Twelfth Water Reactor Safety Research Information Meeting, October 1984.

2. "Multi-Loop Integral System Test (MIST) Facility Specification," RDD:84:4091-01-01:01 (distributed November 1984, revision pending).

3. J. R. Gloudemans, "Simulation of Reactor Vessel Vent Valves," ASME Paper 85-WA/HT-29, presented at the 106th ASME Winter Annual Meeting, Miami, Florida, November 1985.

4. "Multi-Loop Integral System (MIST) Instrumentation -- Revision 3," RDD:84:4127-30-01:03, March 1987.

5. "MIST Test Specifications," BAW-1894, Rev. 1, March 1986.

6. "SGTR Procedures," Chapter III E of the EOP, B&W Document No. 74-1149184-

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Lynchburg, VA 24506-0935

Babcock & Wilcox
Research & Development Division
Alliance Research Center
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Alliance, OH 44601

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Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Multiloop Integral System Test (MIST) is part of a multiphase program started in 1983 to address small-break loss-of-coolant accidents (SBLOCAs) specific to Babcock and Wilcox designed plants. MIST is sponsored by the U. S. Nuclear Regulatory Commission, the Babcock & Wilcox Owners Group, the Electric Power Research Institute, and Babcock and Wilcox. The unique features of the Babcock and Wilcox design, specifically the hot leg U-bends and steam generators, prevented the use of existing integral system data or existing integral facilities to address the thermal-hydraulic SBLOCA questions. MIST and two other supporting facilities were specifically designed and constructed for this program, and an existing facility--the Once Through Integral System (OTIS)--was also used. Data from MIST and the other facilities will be used to benchmark the adequacy of system codes, such as RELAP5 and TRAC, for predicting abnormal plant transients.

The MIST program is reported in 11 volumes. The program is summarized in Volume 1; Volumes 2 through 8 describes groups of tests by test type; Volume 9 presents inter-group comparisons; Volume 10 provides comparisons between the calculations of RELAP5/MOD2 and MIST observations, and Volume 11 presents the later Phase 4 tests. This Volume 6 pertains to Test Group 34, Steam Generator Tube Rupture. The specifications, conduct, observations, and results of these tests are described.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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T340100

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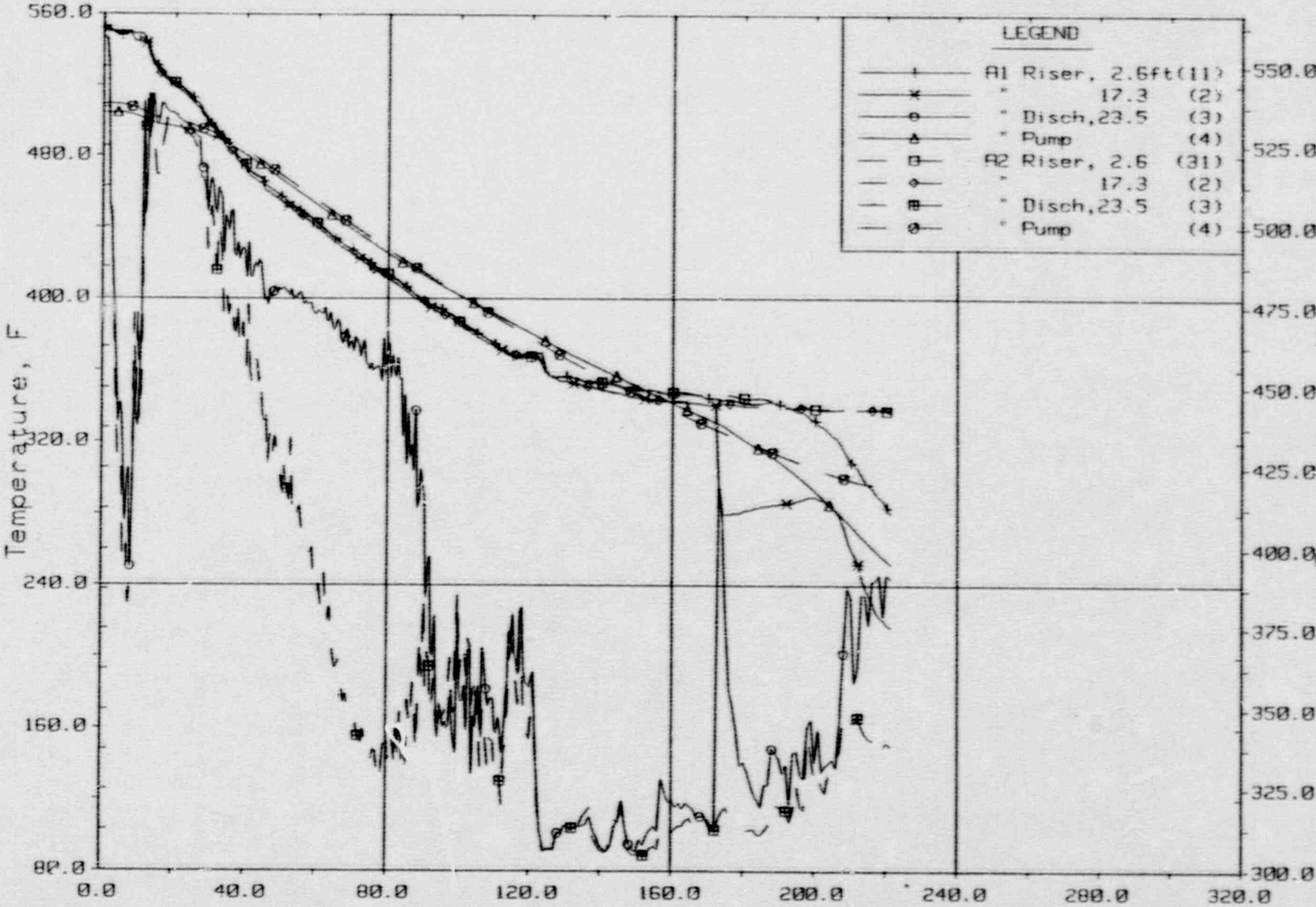

Loop A Cold Leg Metal Temperatures (C1,3MTs).

Tue May 31 10:27:44 1988

C1MT1

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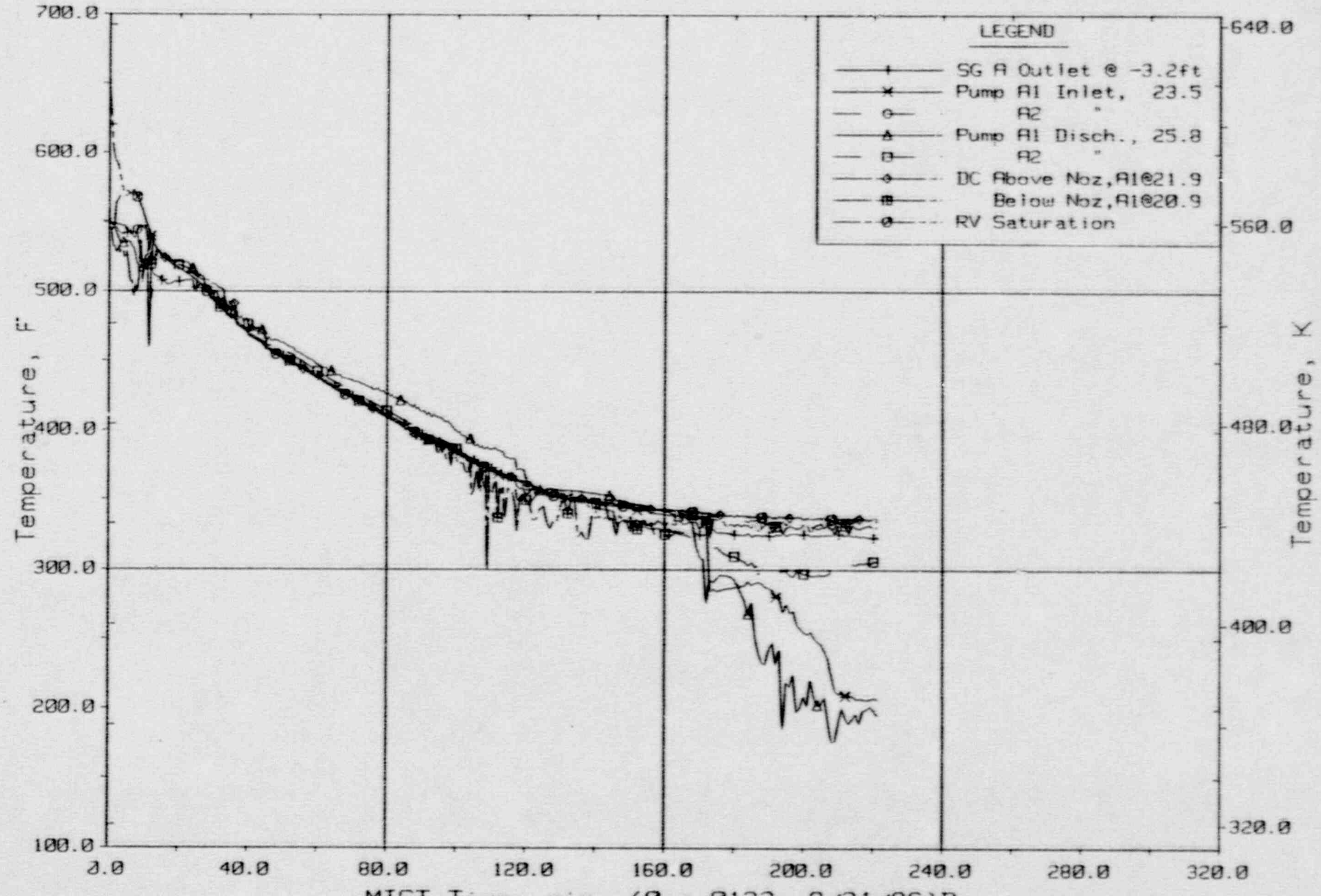

Loop A Cold Leg Fluid Temperatures (RTDs).

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C1RT0

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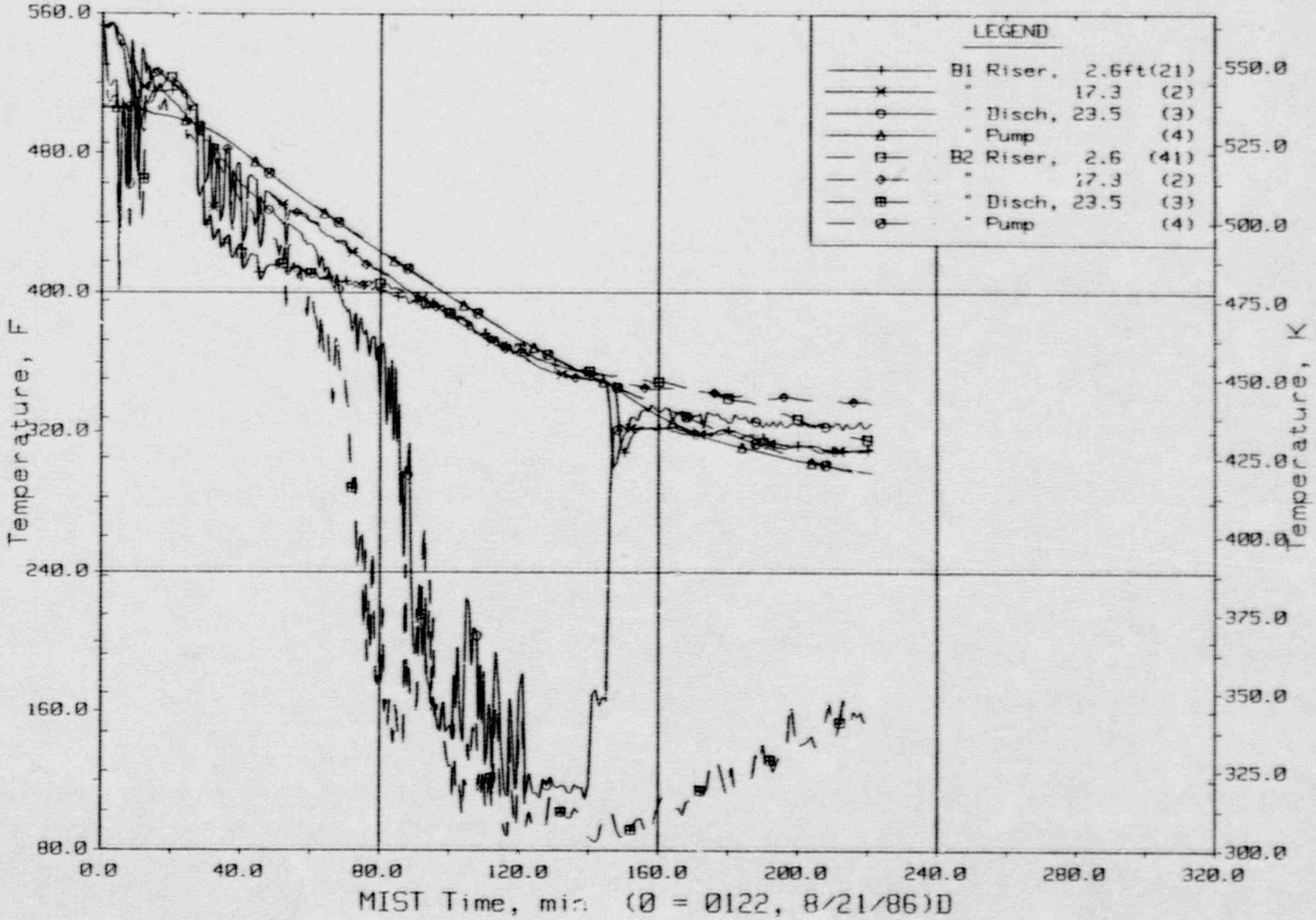

Loop B Cold Leg Metal Temperatures (C2,4MTs).

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C2MT1

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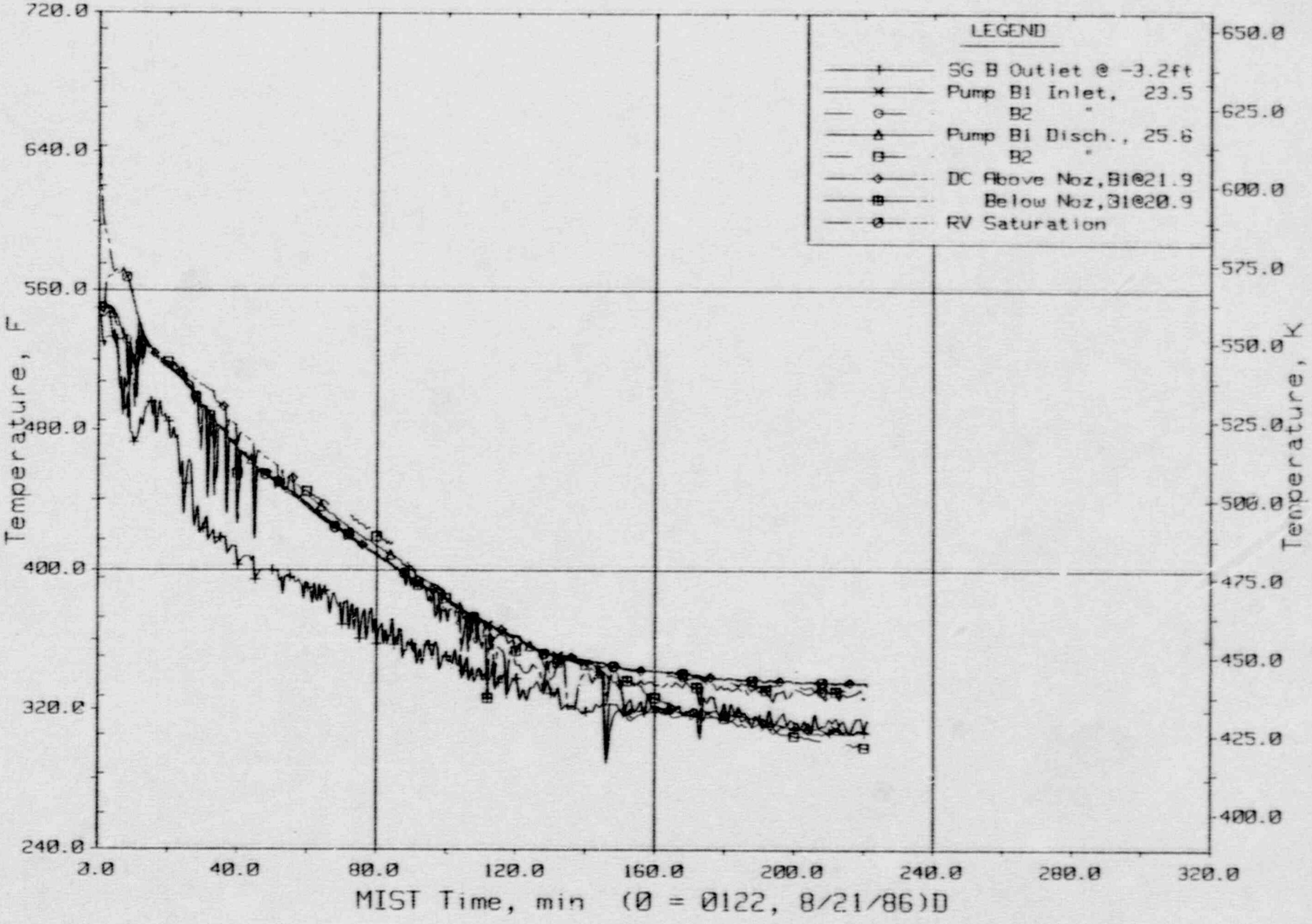

Loop B Cold Leg Fluid Temperatures (RTDs).

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C2RT0

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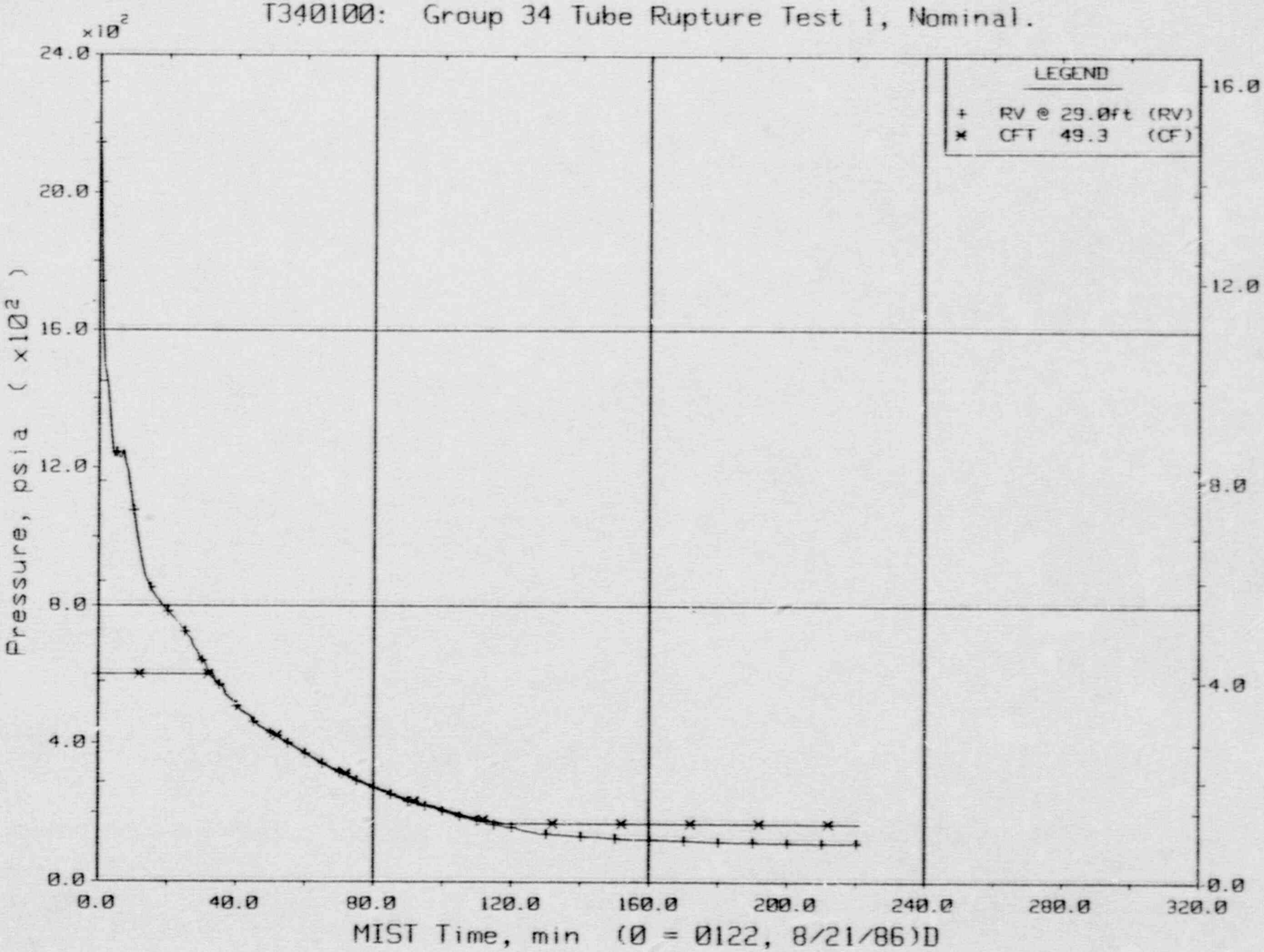

Primary System and Core Flood Tank Pressures (GP01s).

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CFGP1

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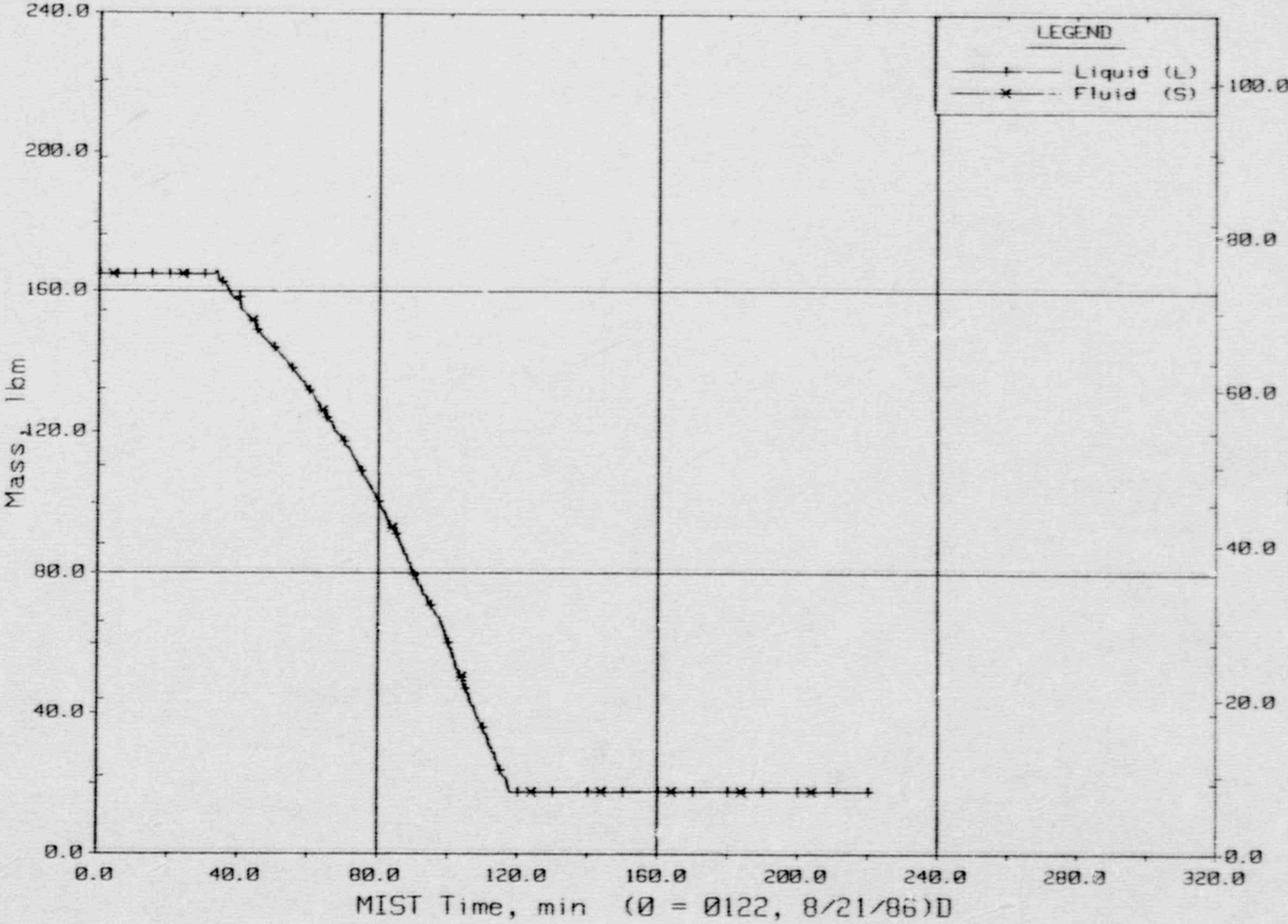

Core Flood Tank Liquid and Fluid Mass (CFMa20s).

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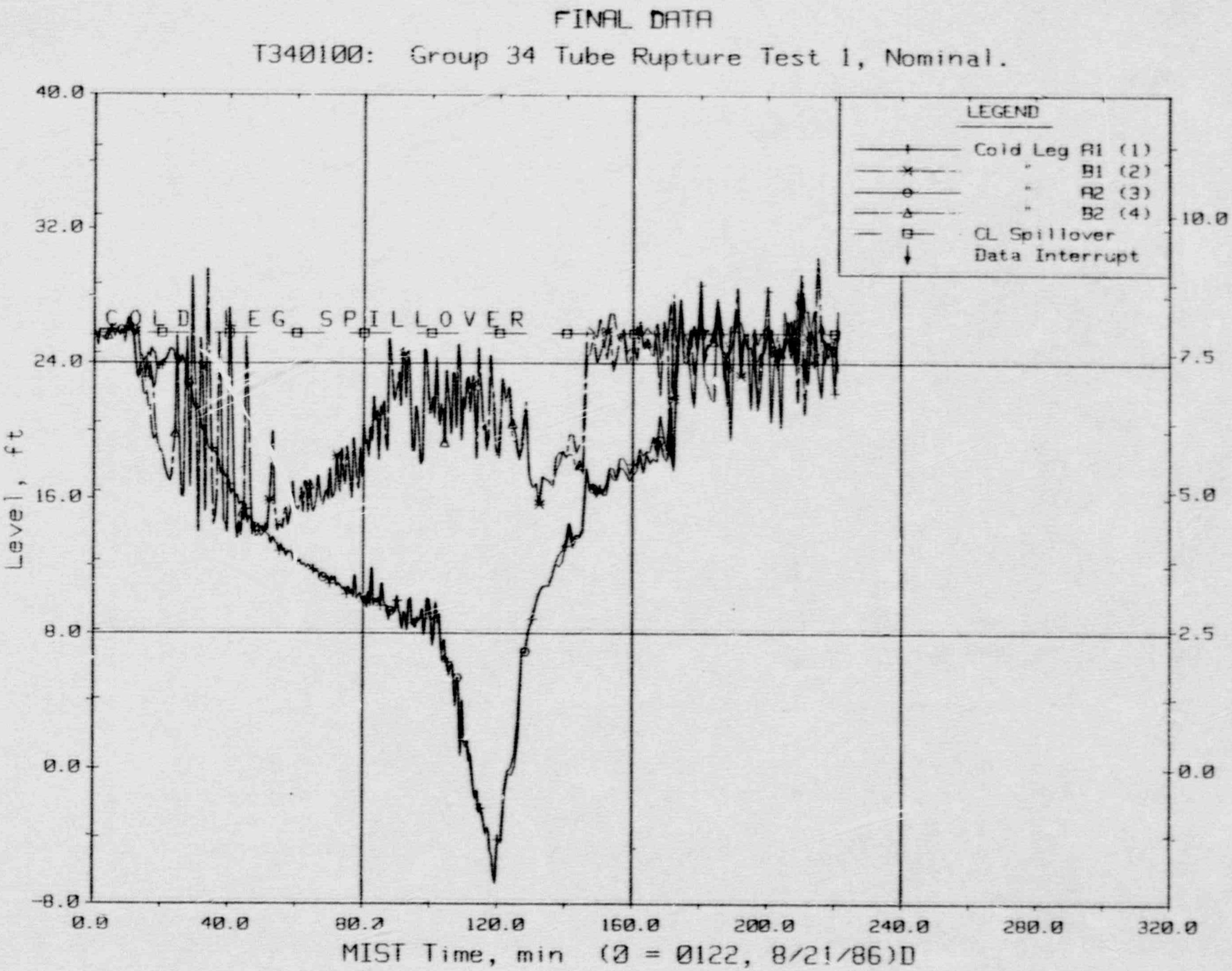
CFML1



Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

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CLLV4

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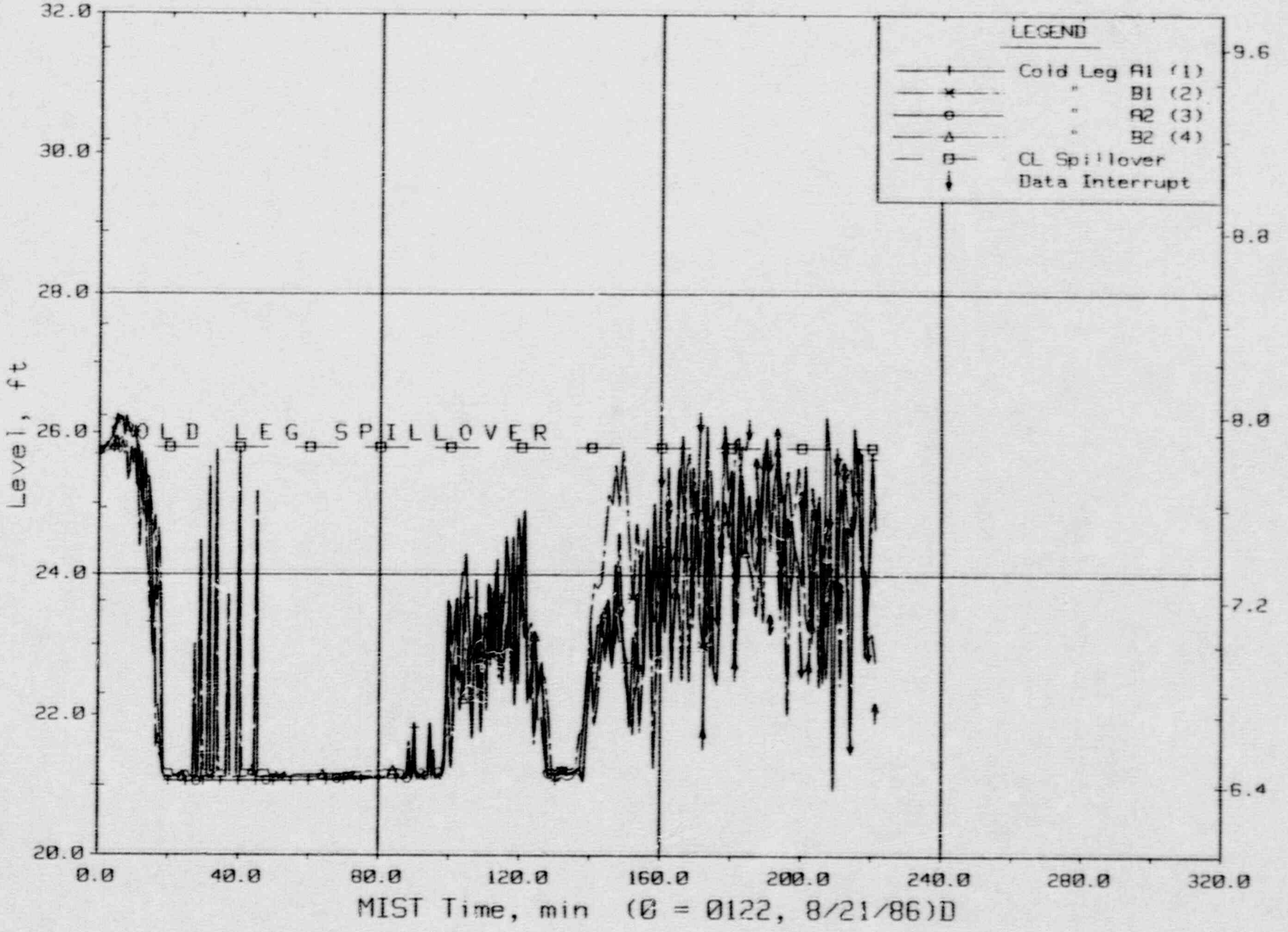
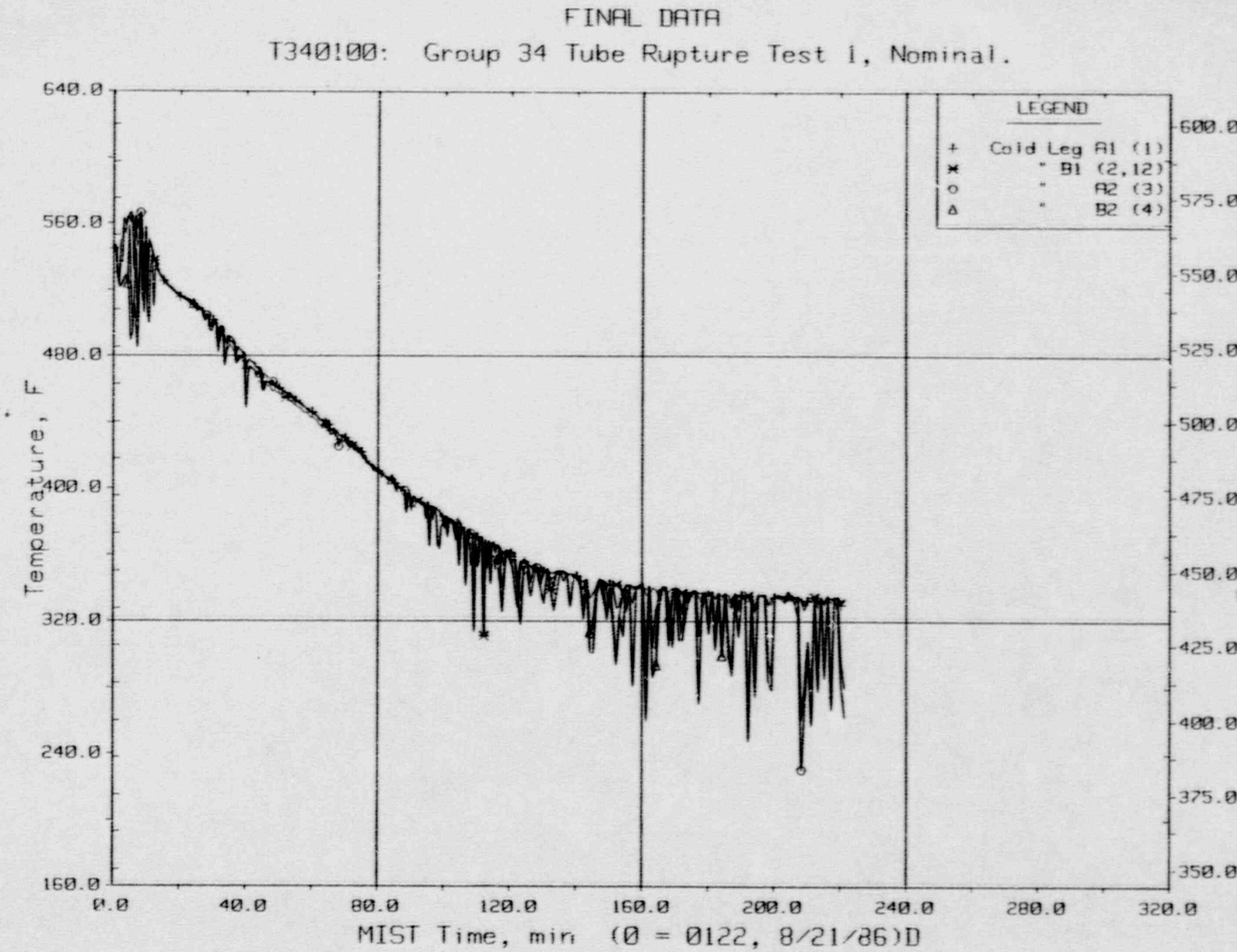


Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Tue May 31 10:54:49 1988

CLLV5



Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC11s).

Tue May 31 11:00:08 1988

CLTC1

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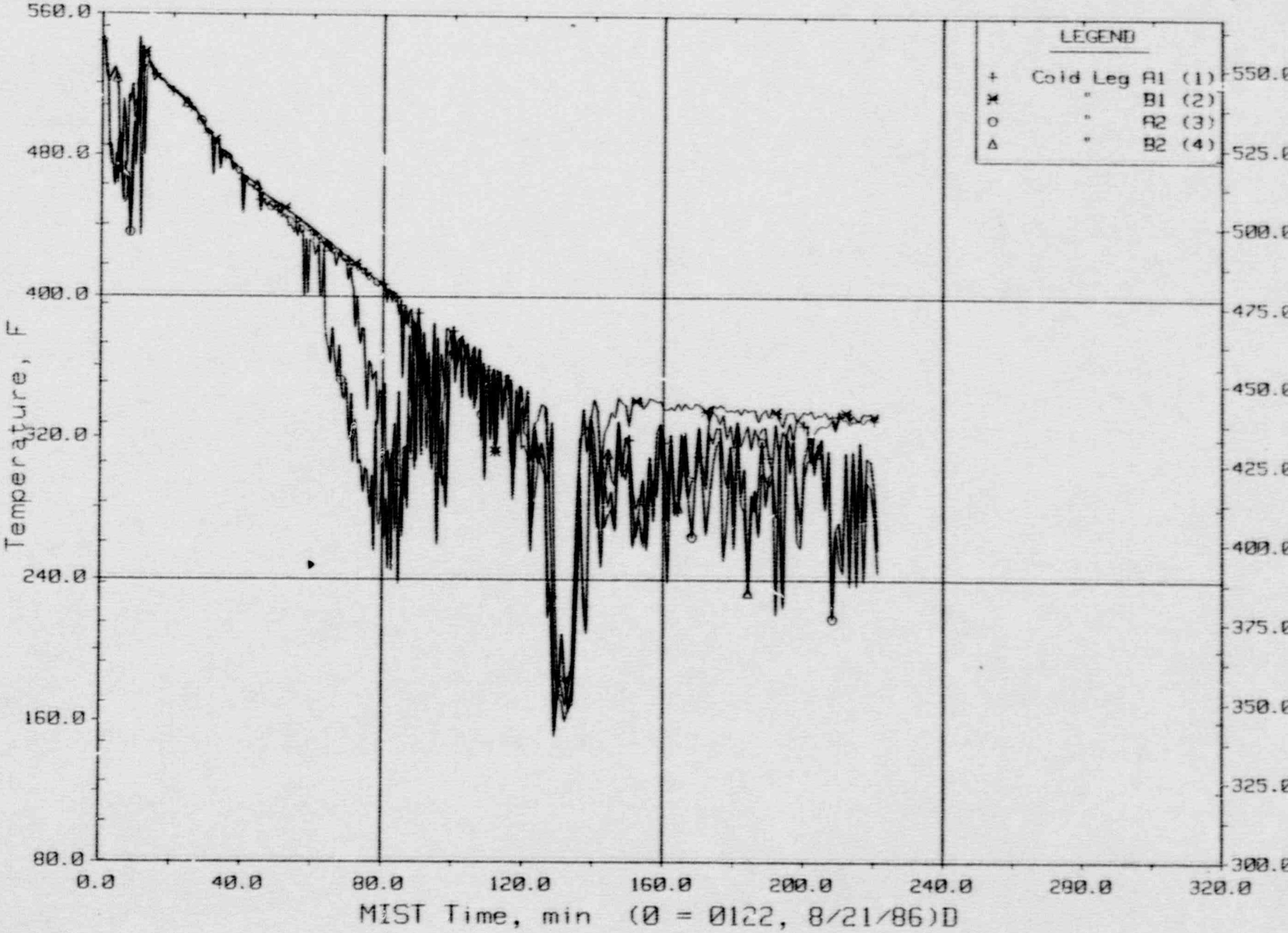

Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

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CLTC2

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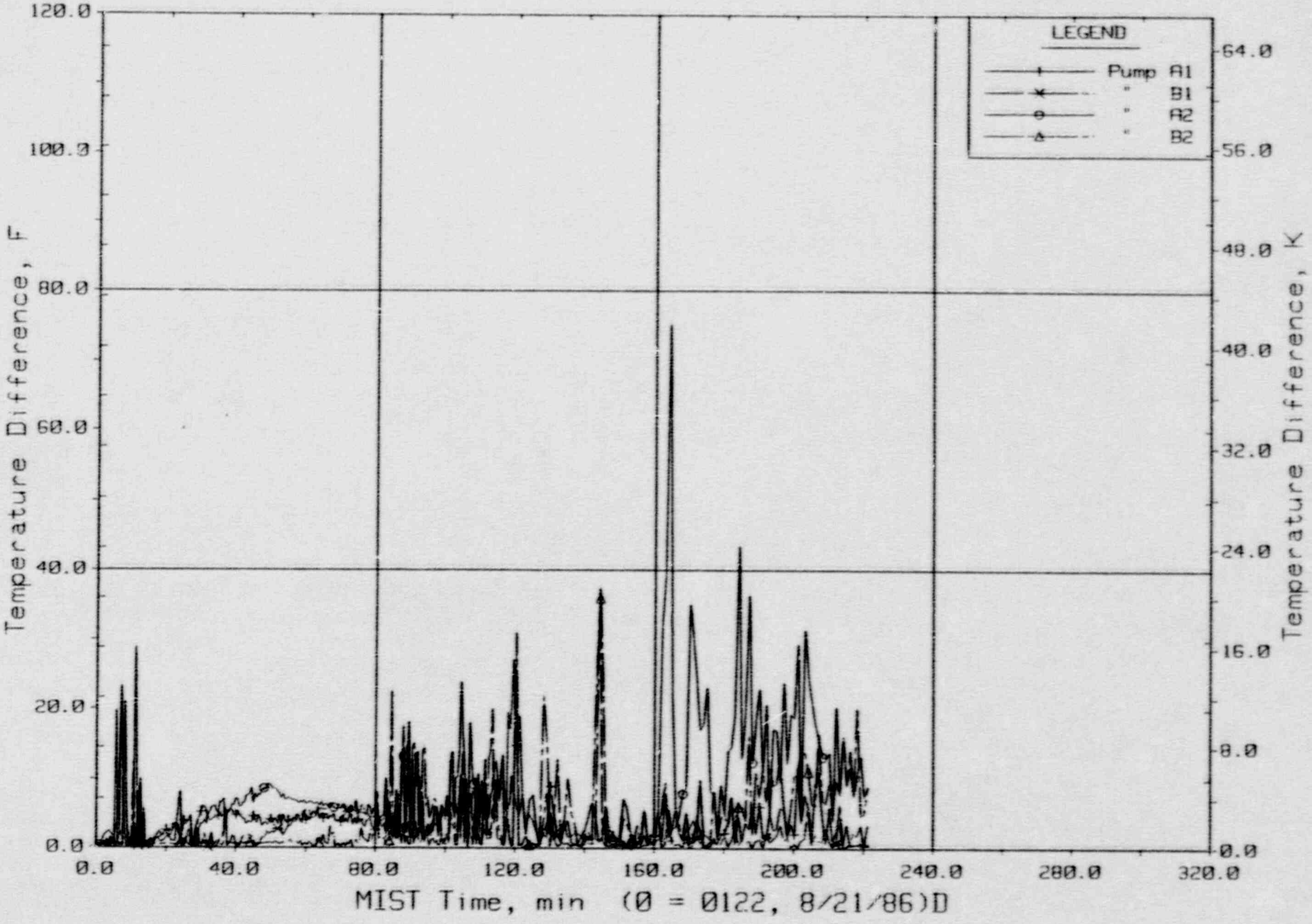

Maximum Differences Among RCP Rake FLuid Temperatures.

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CLTD1

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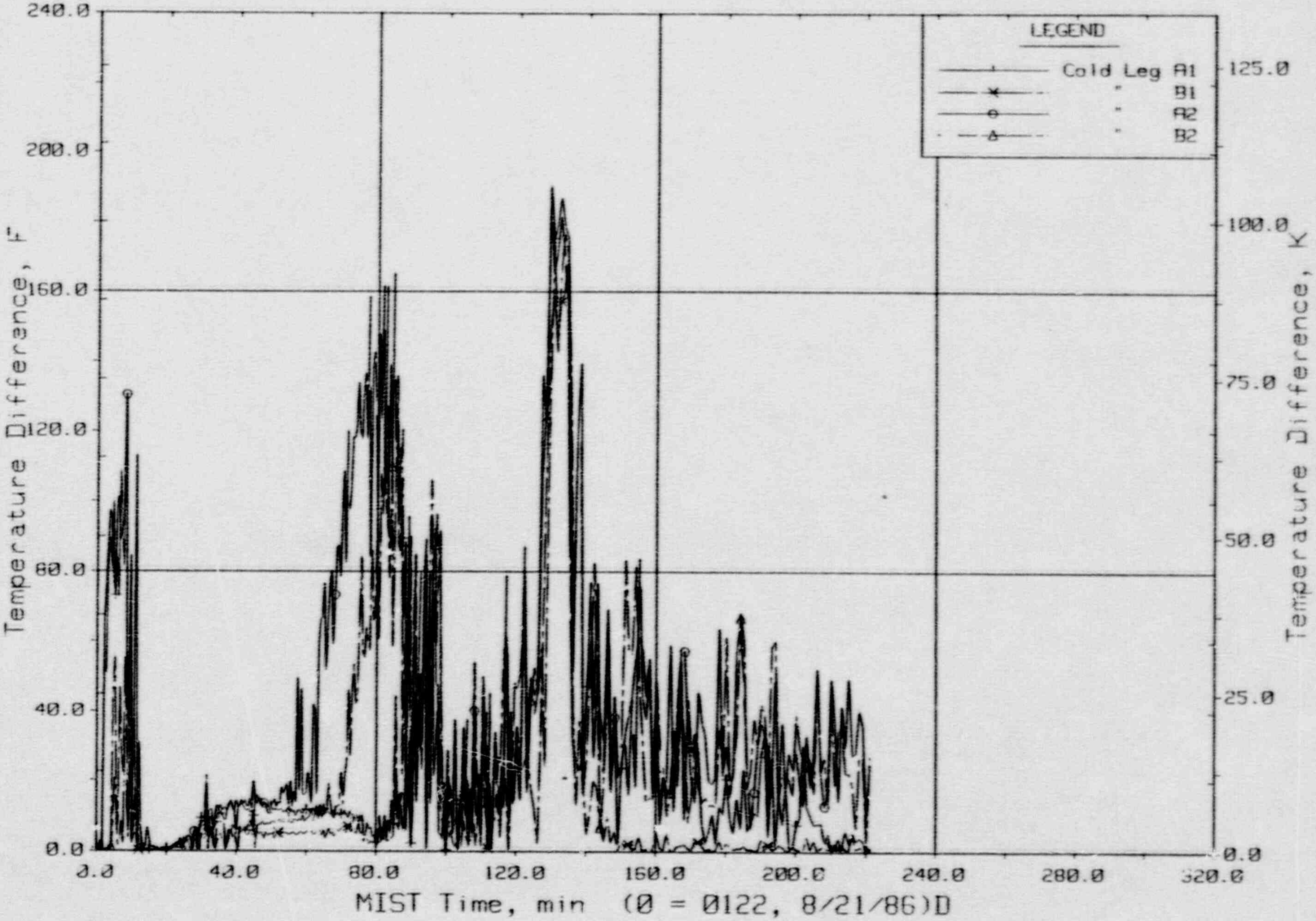

Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

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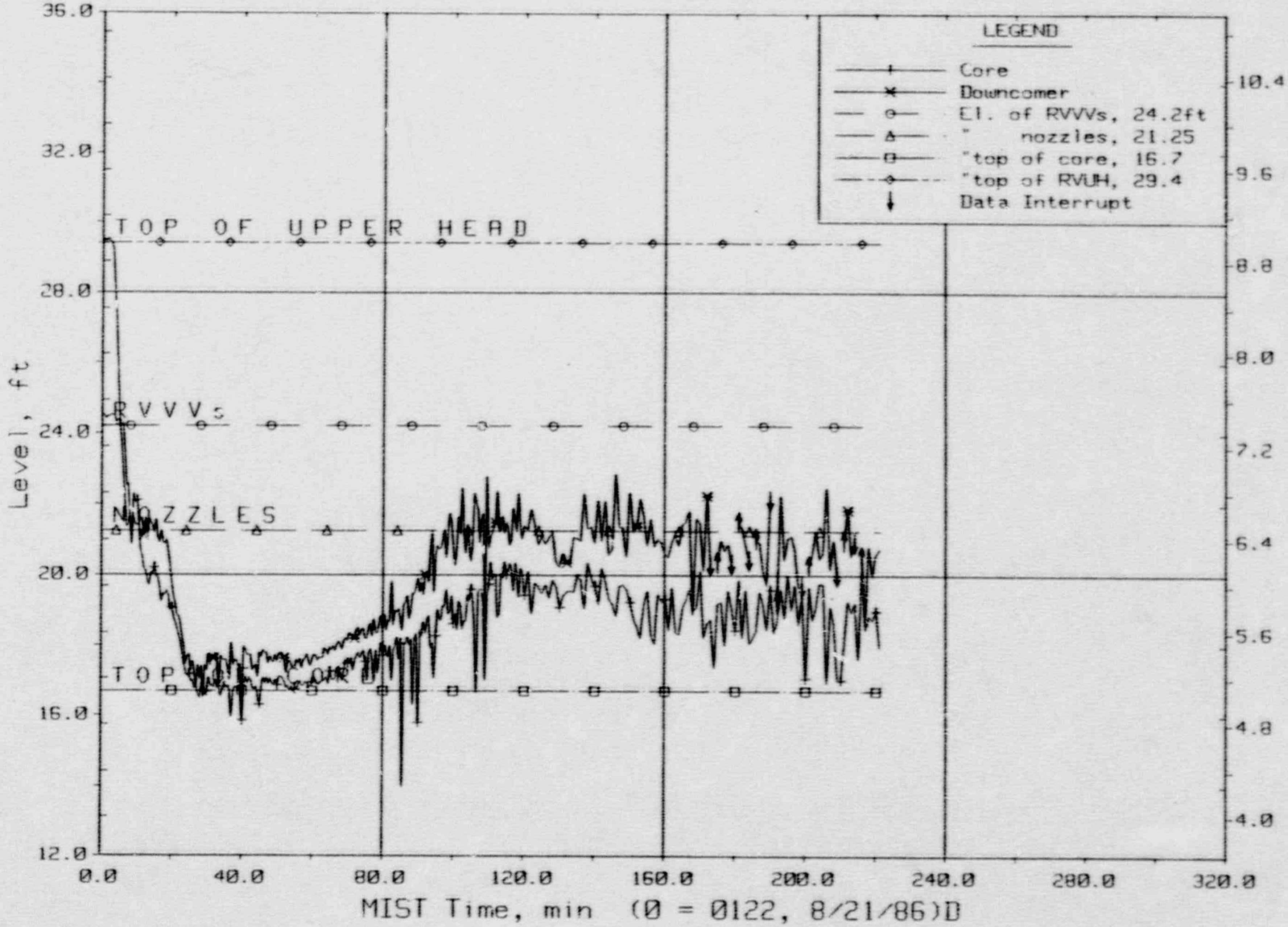

Core Region Collapsed Liquid Levels.

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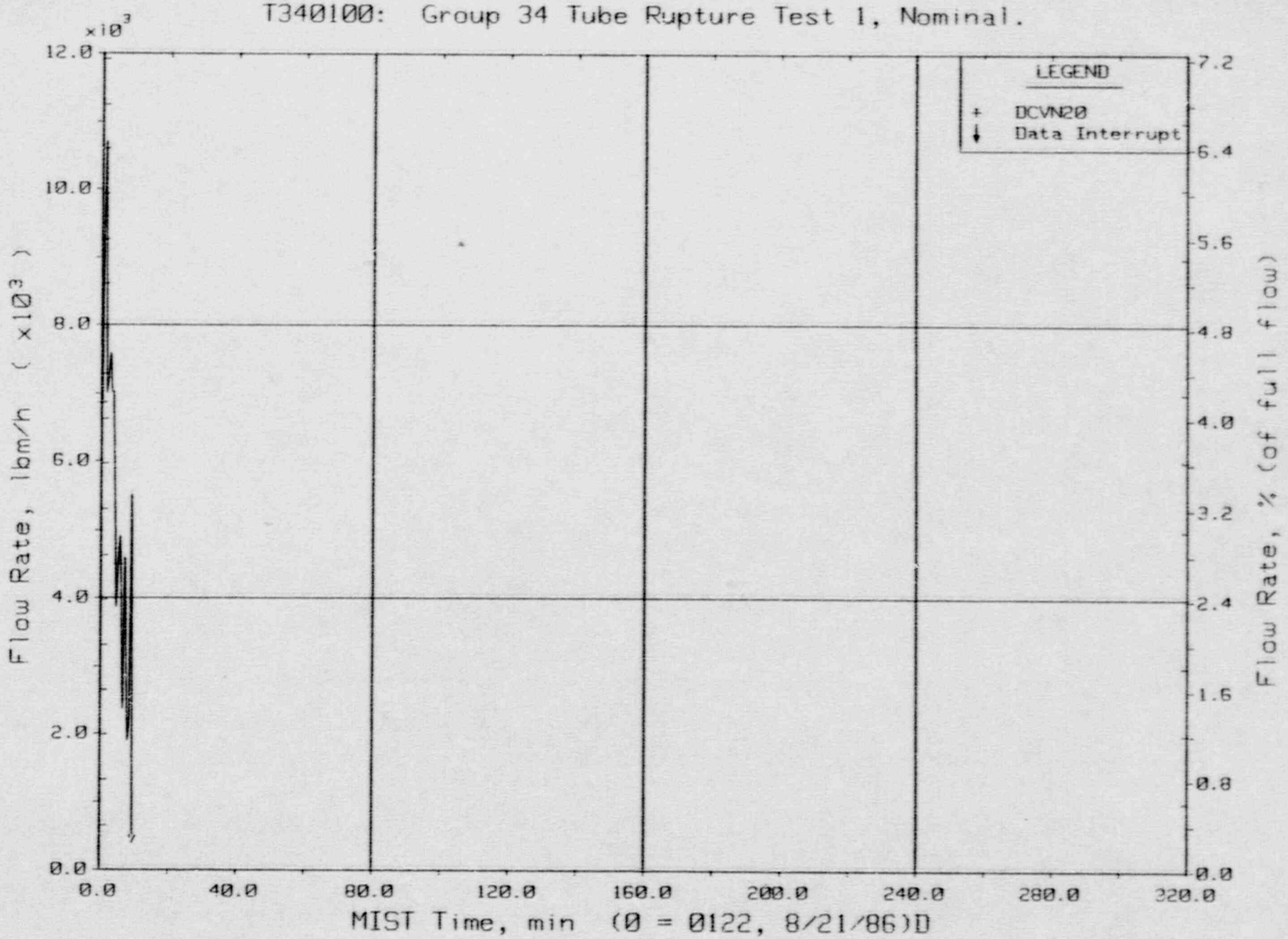

Downcomer (Venturi) Flow Rate.

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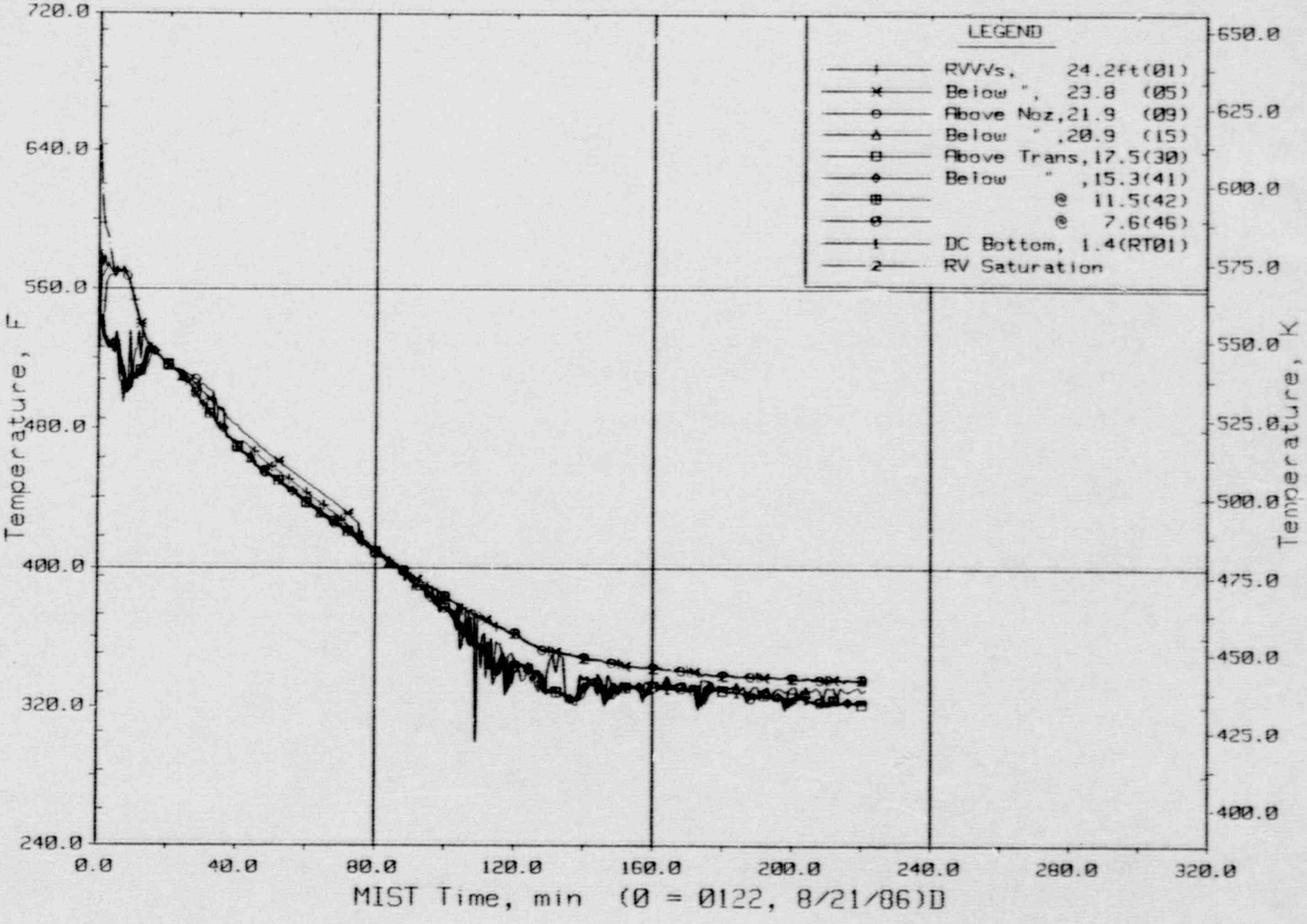

Downcomer Quadrant A1 Flüid Temperatures (DCTCs).

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DCTC6

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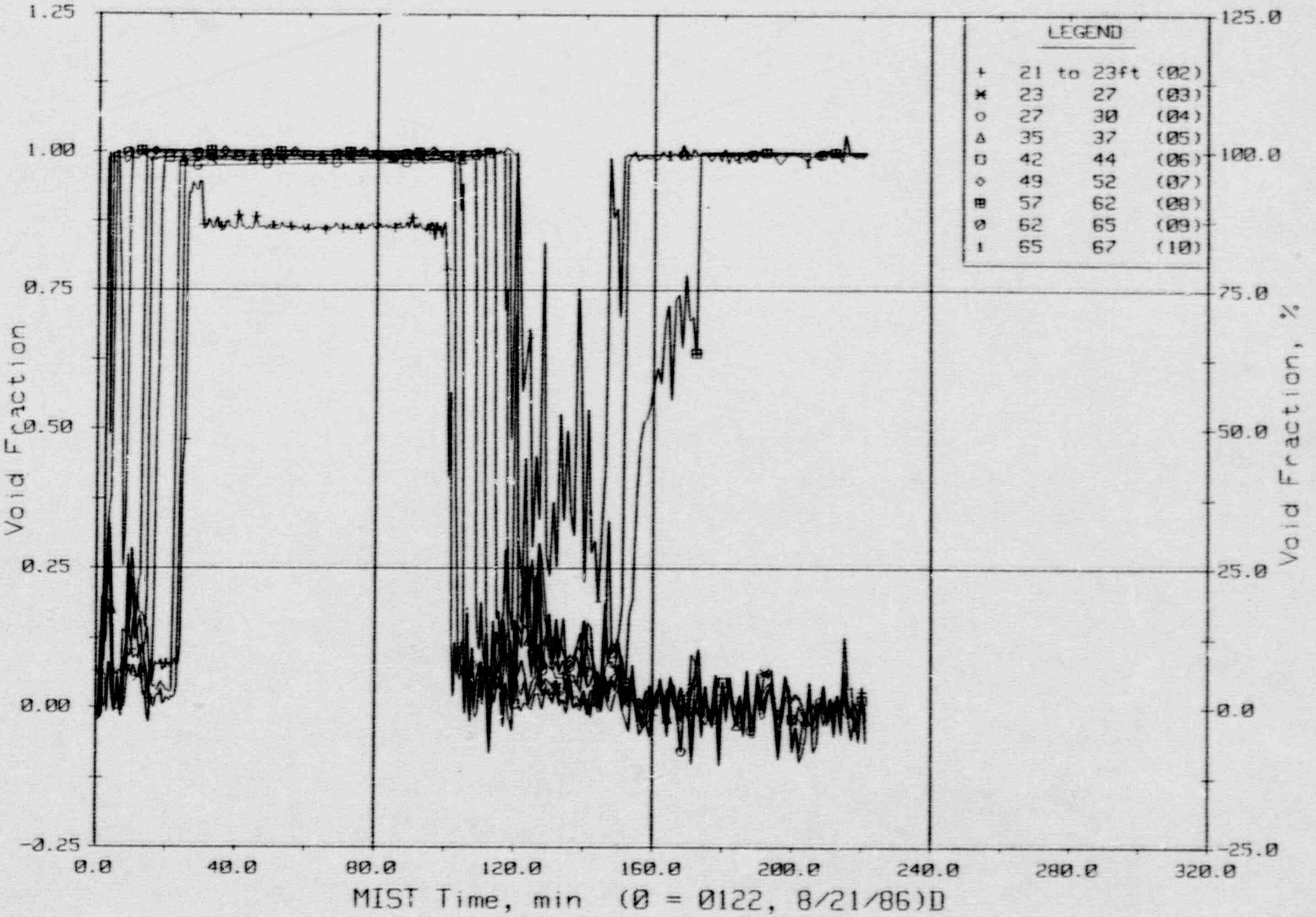

Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

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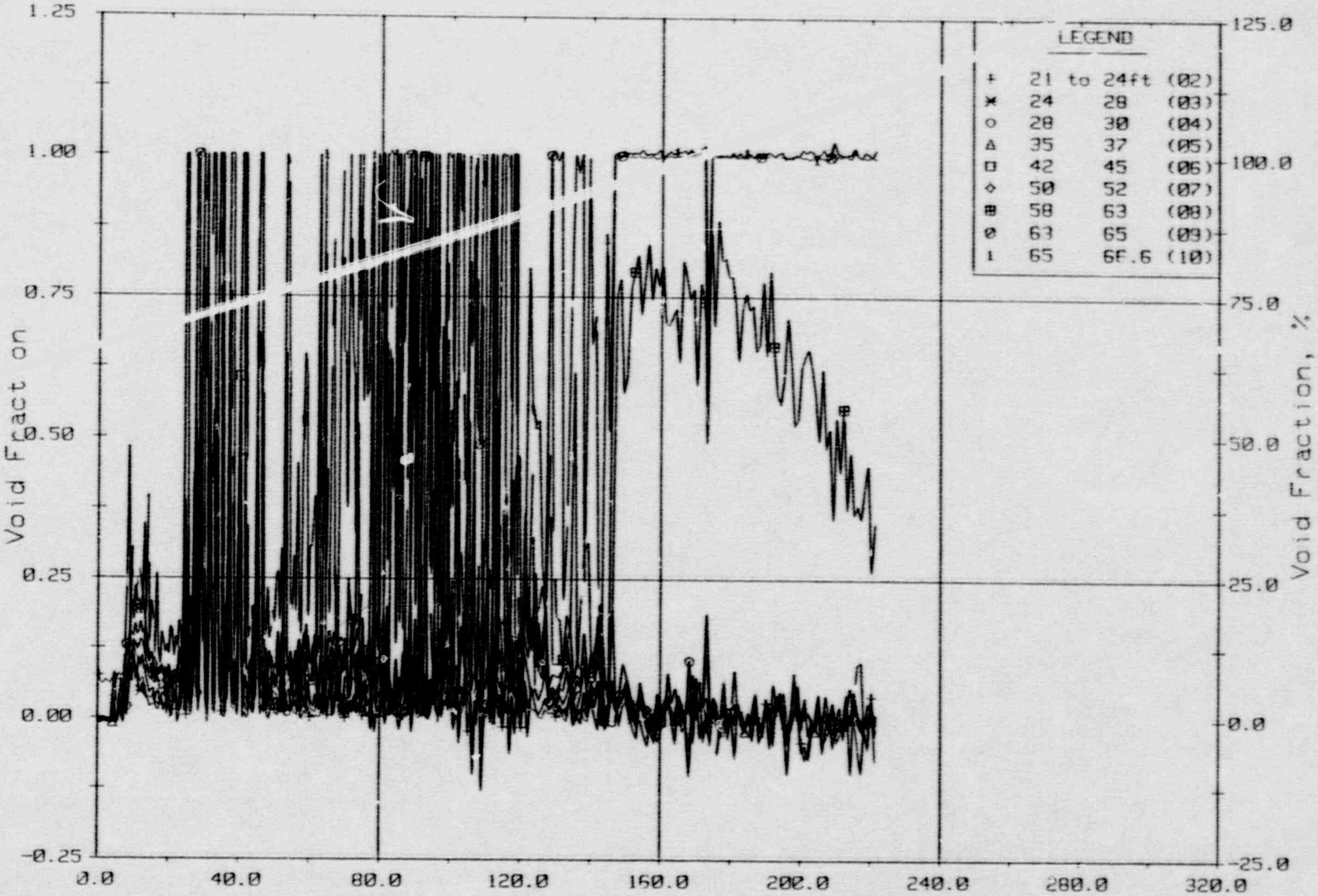

Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

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H2VF1

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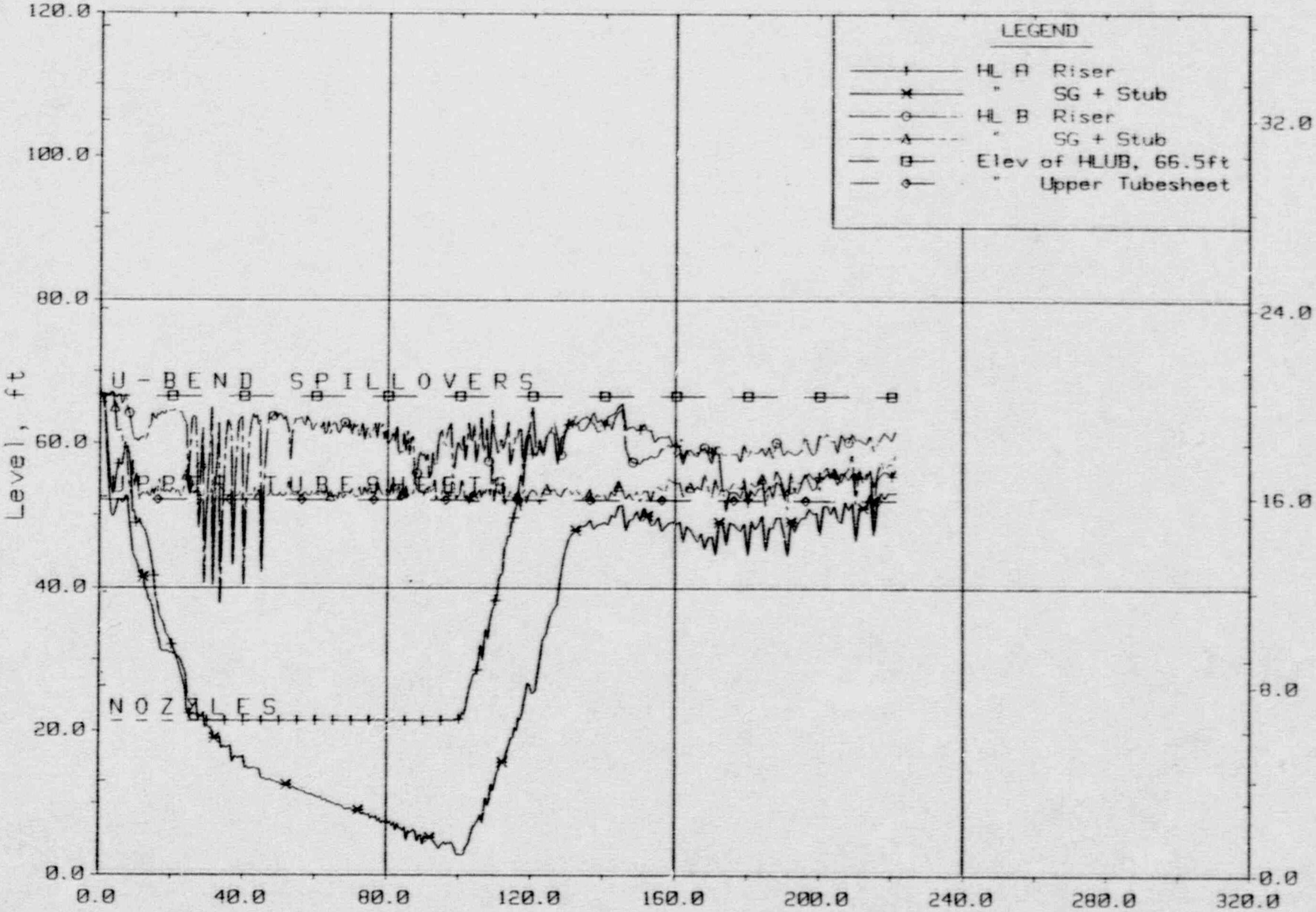

Hot Leg Riser and Stub Collapsed Liquid Levels.

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HLLV1

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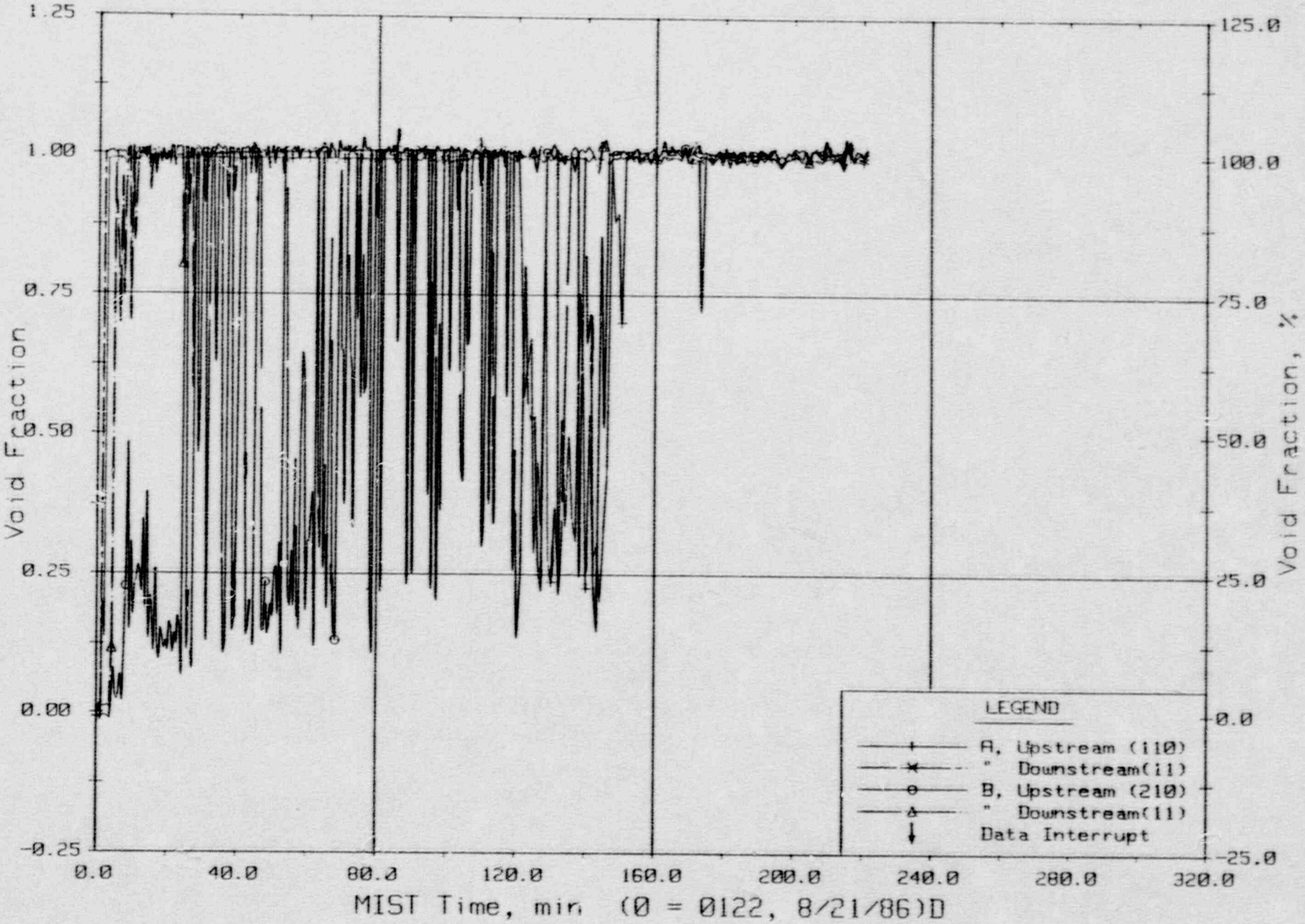

Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Tue May 31 11:36:00 1988

HLVF1

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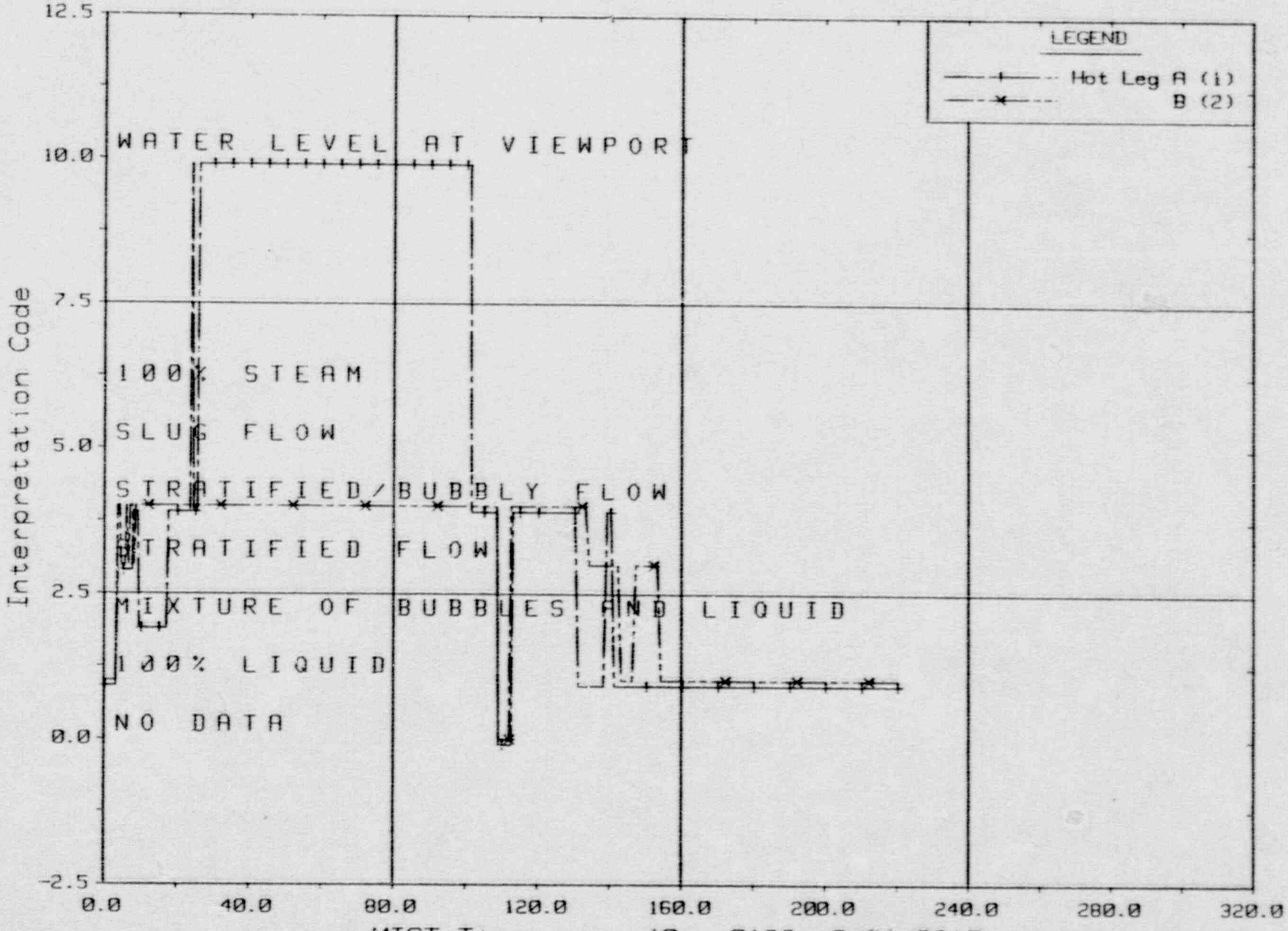

Hot Leg Horizontal Viewport Indications (HnMS01s).

Tue May 31 13:42:00 1988

HLVP2

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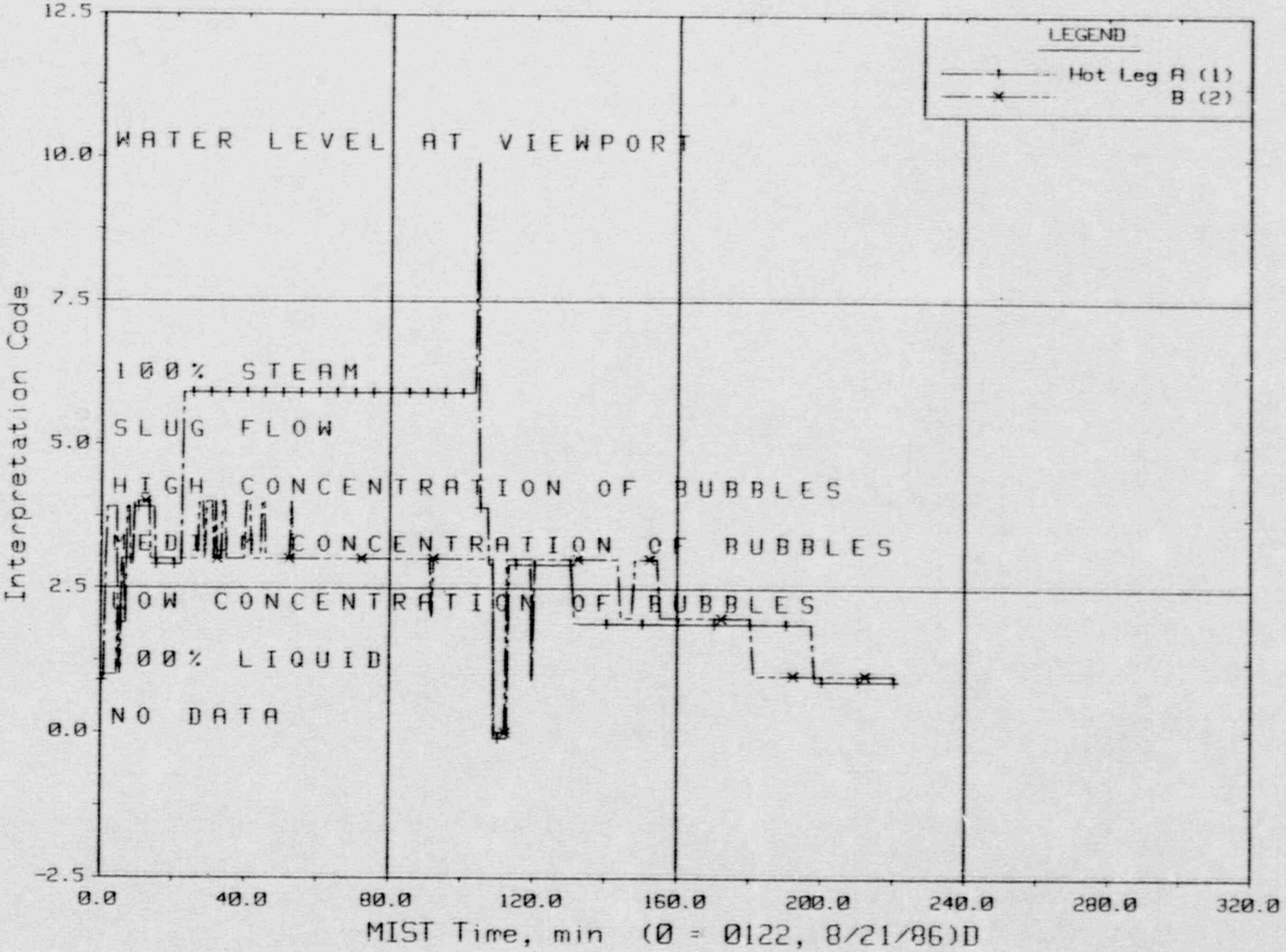

Hot Leg Riser Viewport Indications (HnMS02s).

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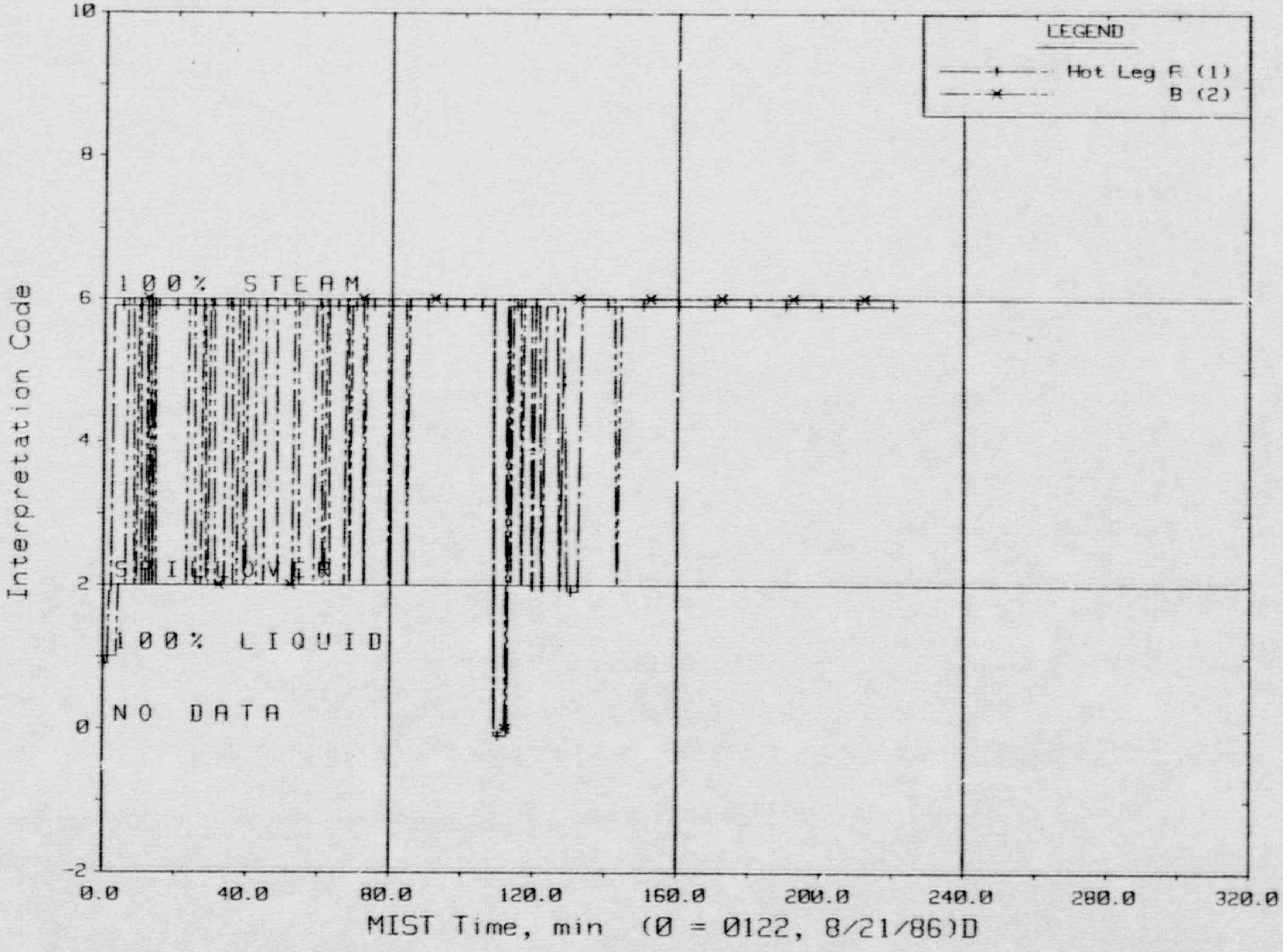

Hot Leg U-Bend Viewport" Indications (HnMS03s).

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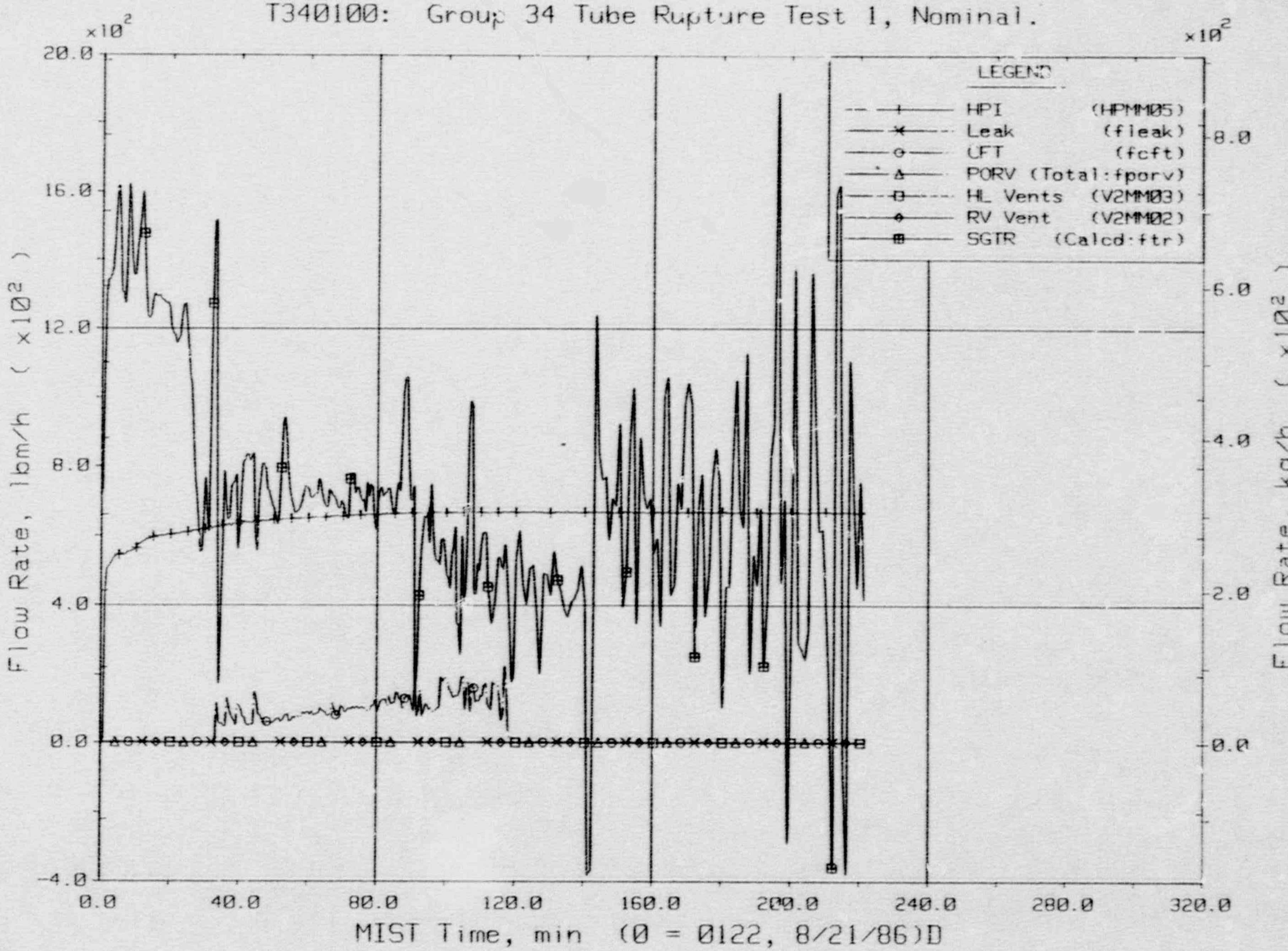
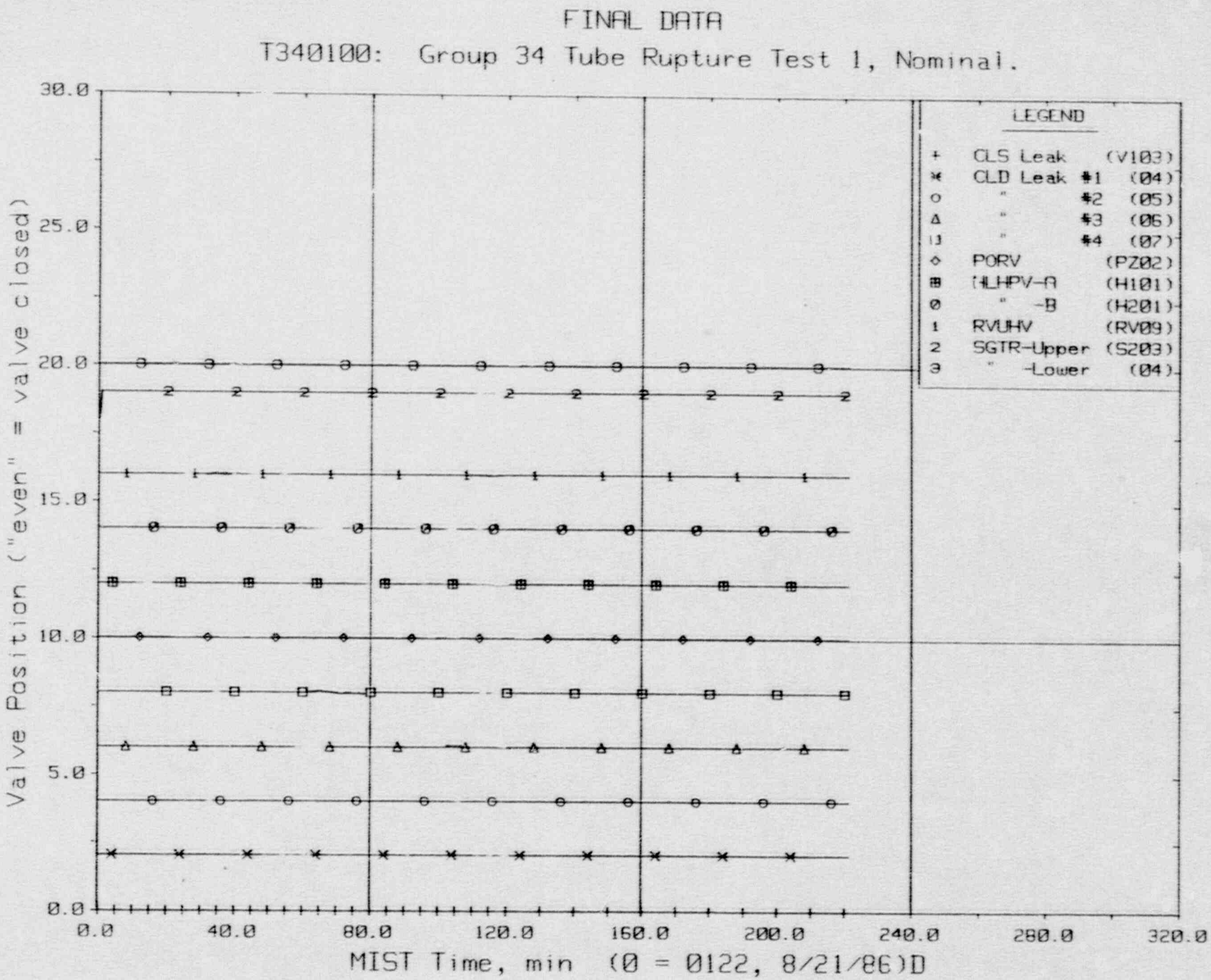


Primary System Boundary Flow Rates.

Tue May 31 11:40:03 1988

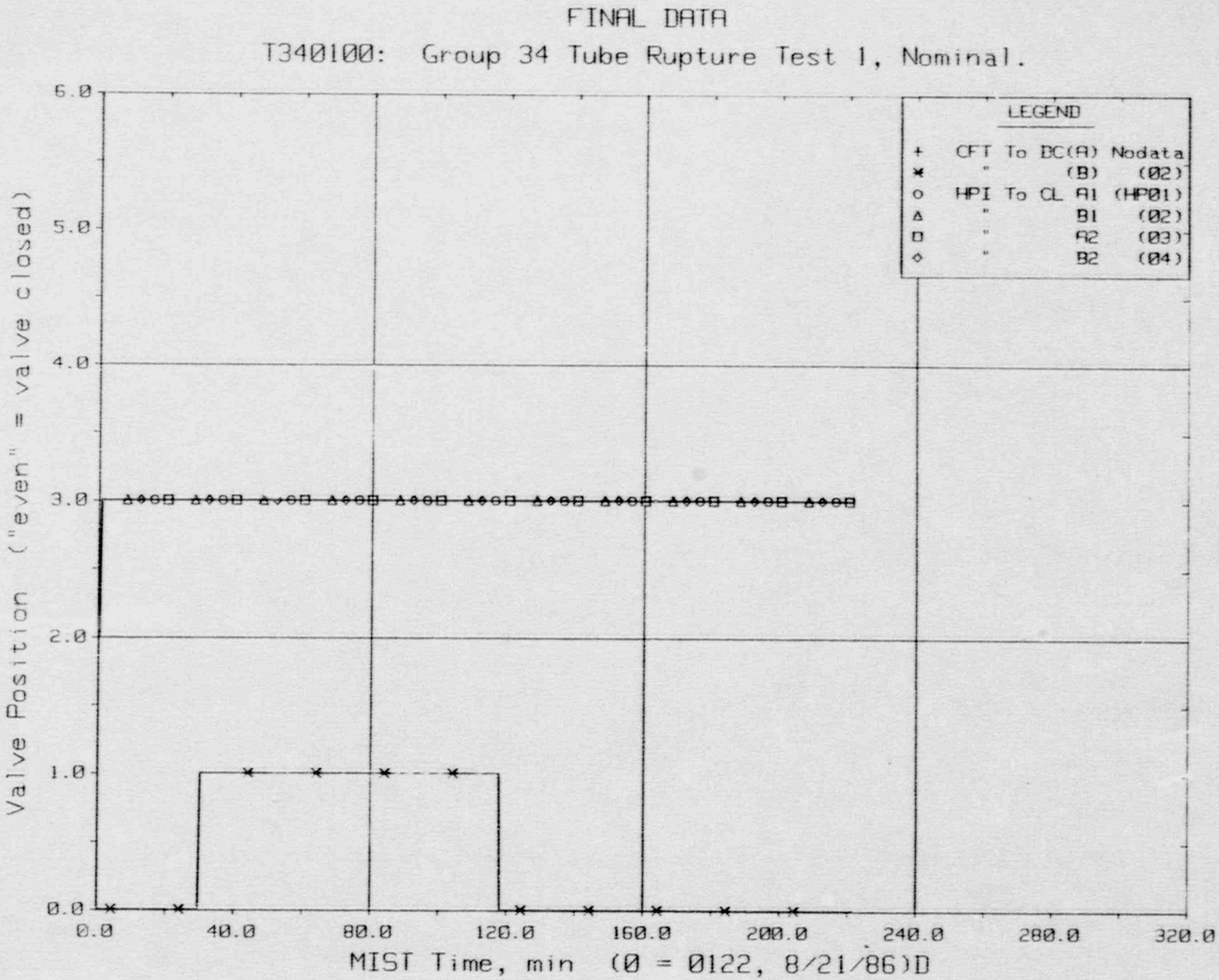
PBFL3



Primary System Discharge Limit Switch Indications (LSs).

Tue May 31 11:44:51 1988

PBLS1



Primary System Injection Limit Switch Indications (LSs).

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PBLS2

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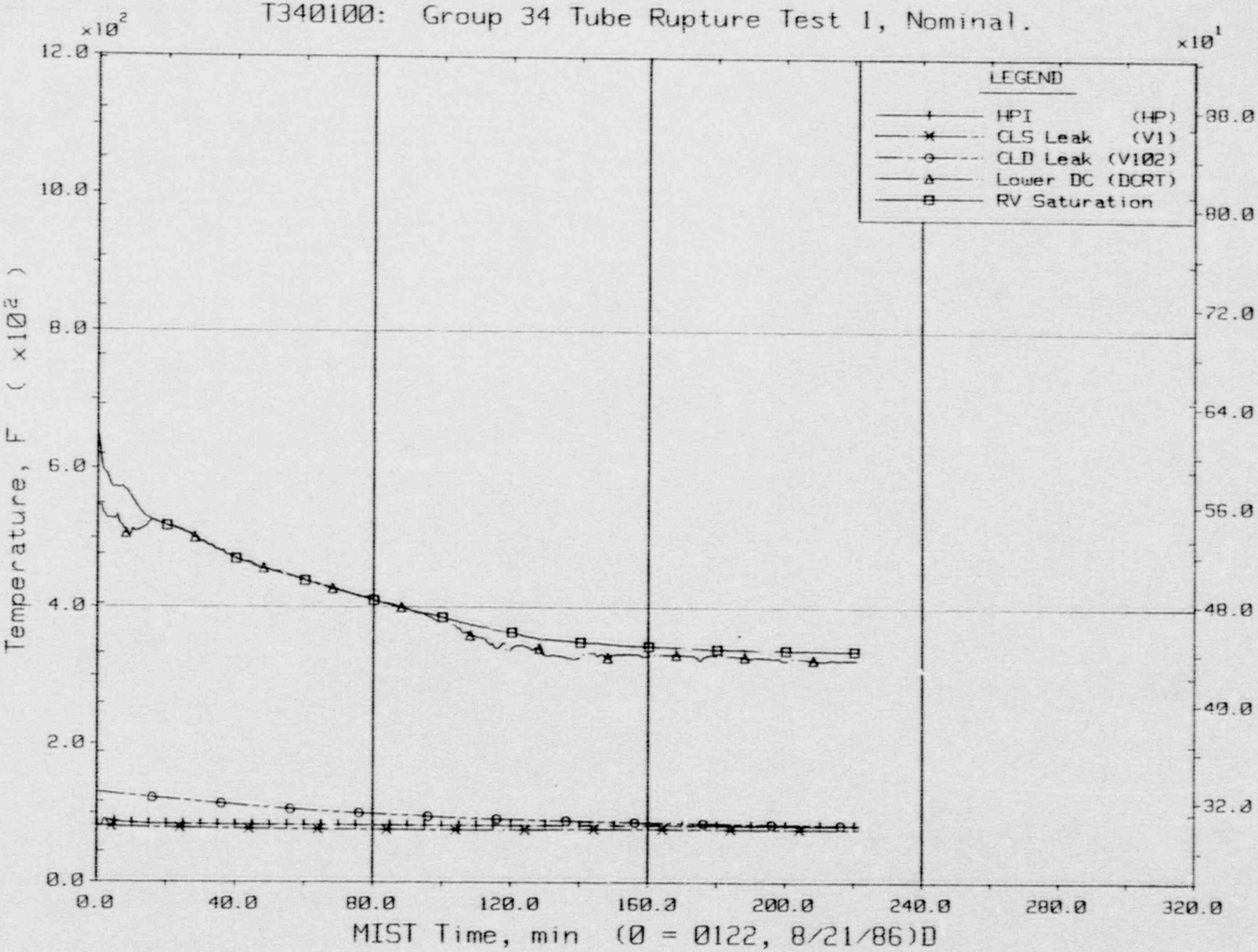

Single-Phase Discharge and HPI Fluid Temperatures (TC0is).

Tue May 31 11:49:53 1988

PBTC1p

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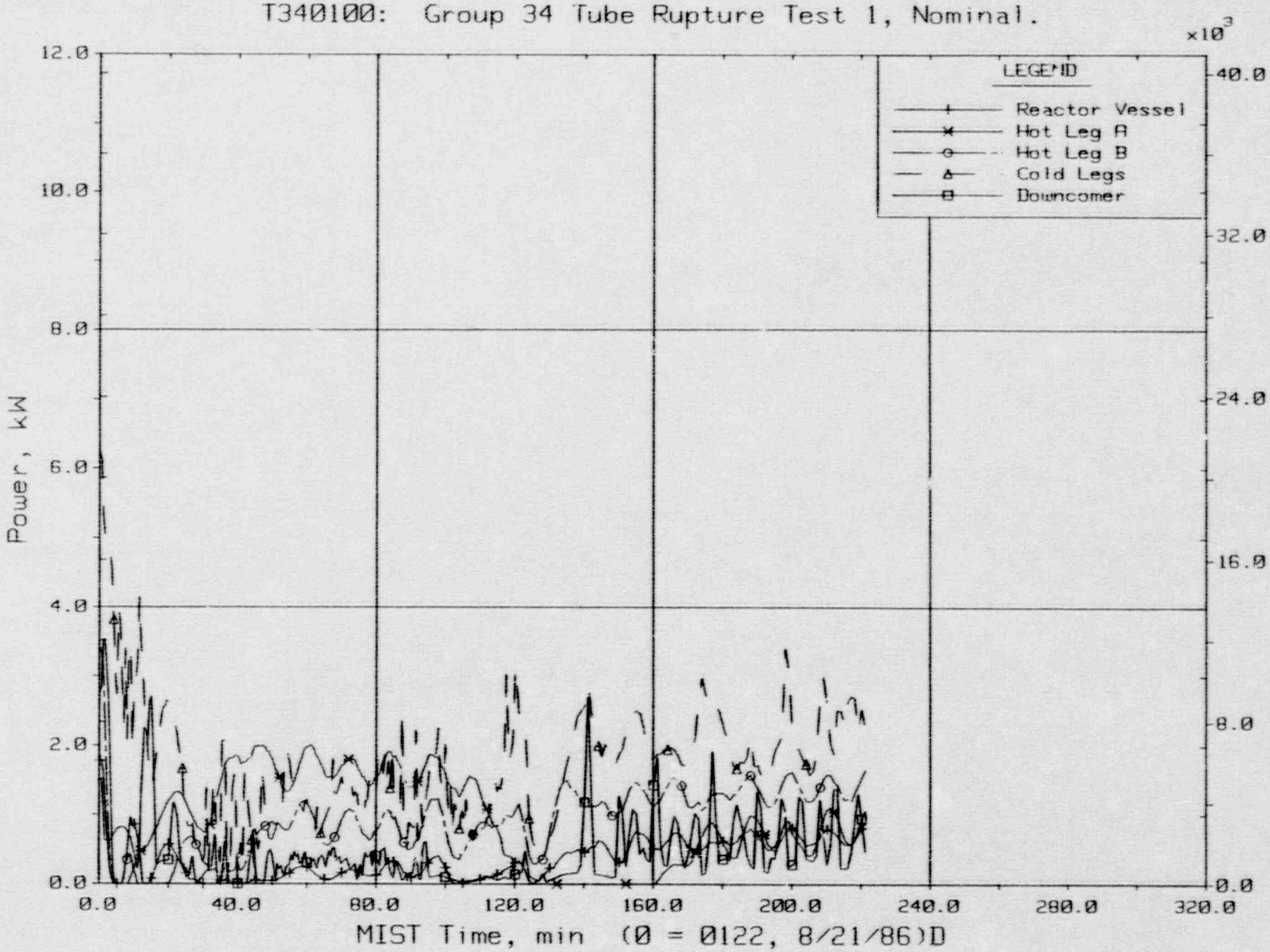

Guard Heater Specified Power Per Primary Component.

Tue May 31 11:55:40 1988

PRGQ1

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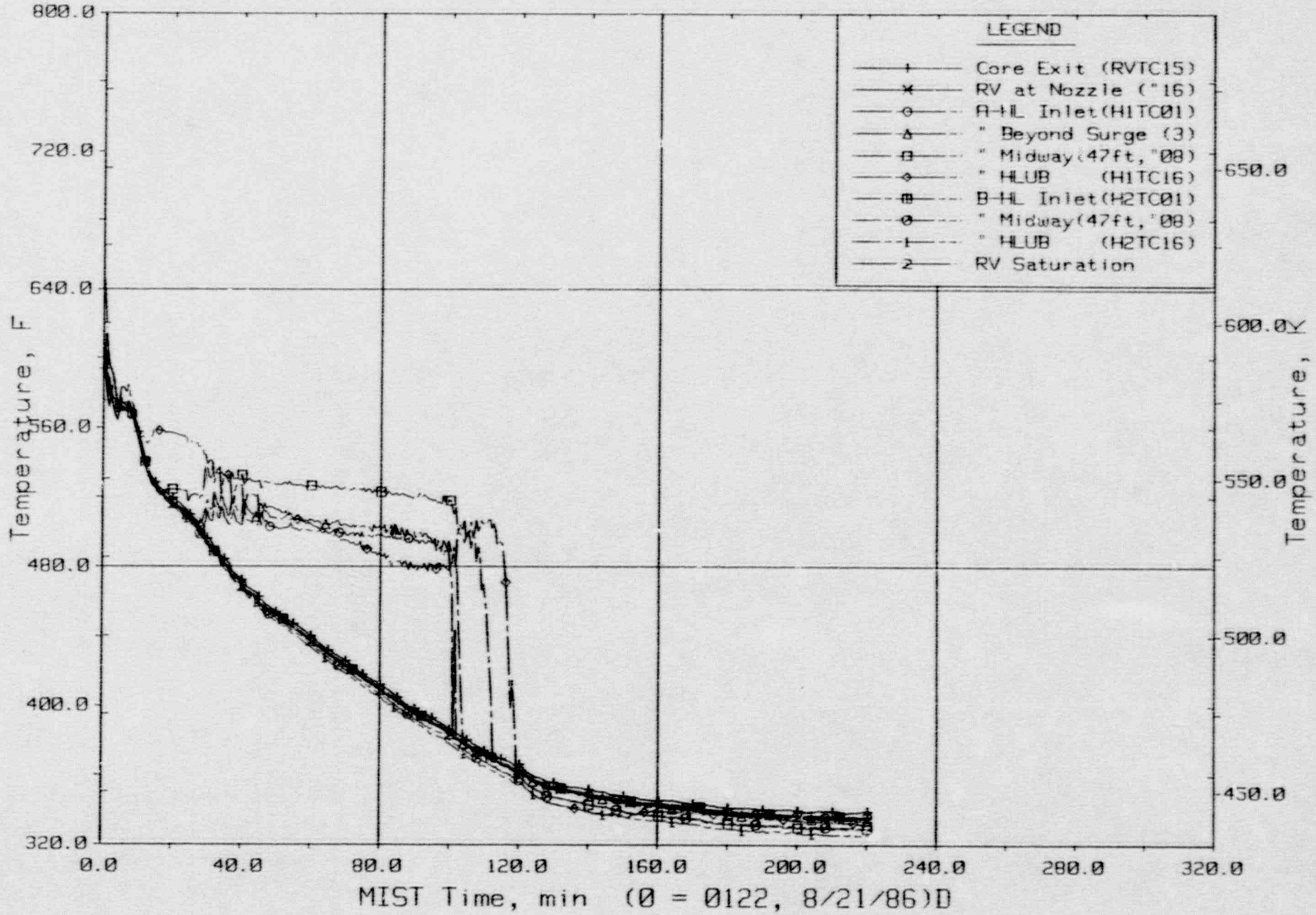

Composite Core Exit and Höt Leg Fluid Temperatures.

Tue May 31 12:00:18 1988

PRKT1

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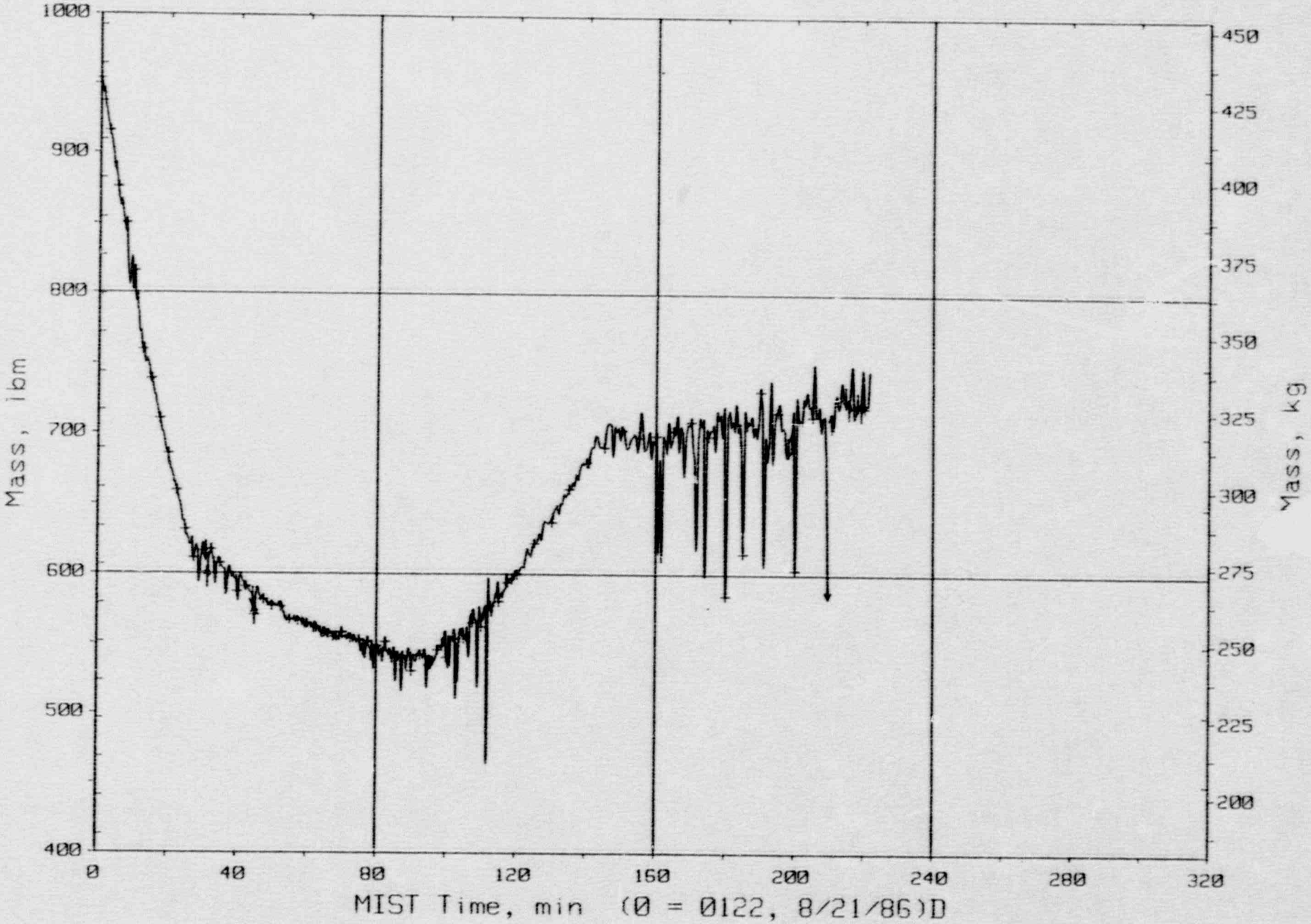

Indicated Primary System Total Fluid Mass (PLML20).

Wed Nov 2 14:04:13 1988

PRML9

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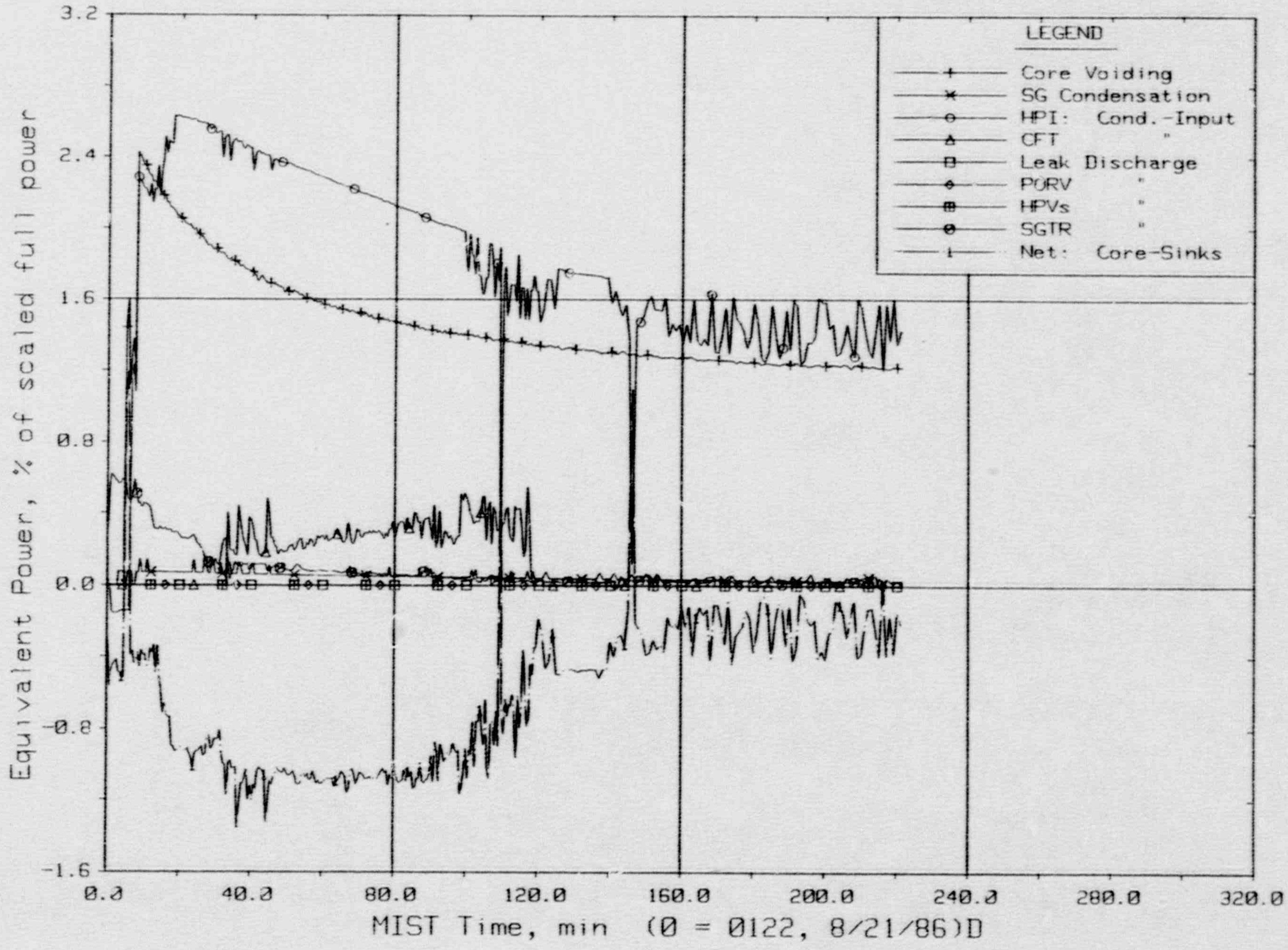

Primary Fluid Volume Changes By Components.

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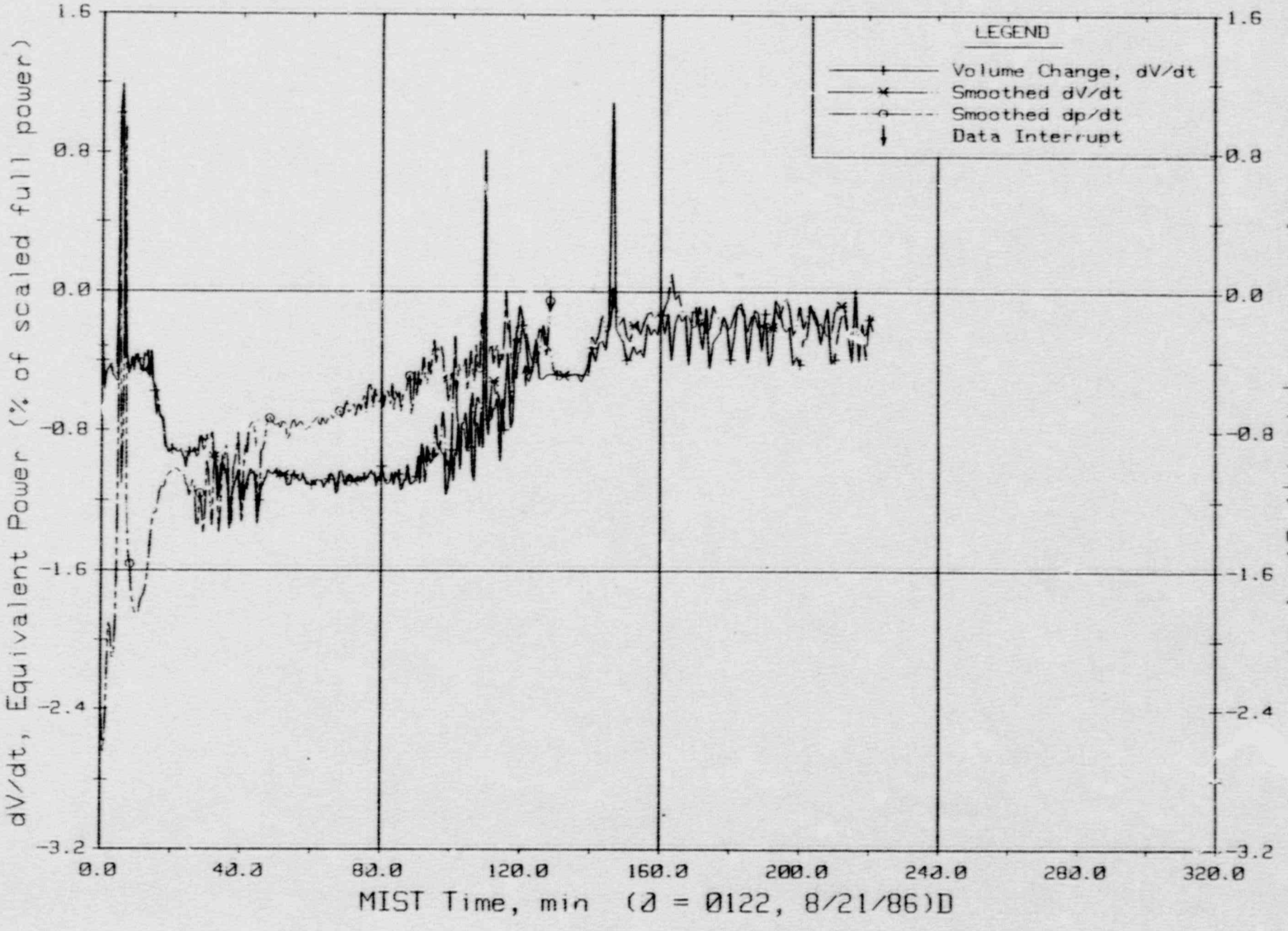

Primary Fluid Volume and Pressure Changes, dV/dt and dp/dt.

Tue May 31 12:09:45 1988

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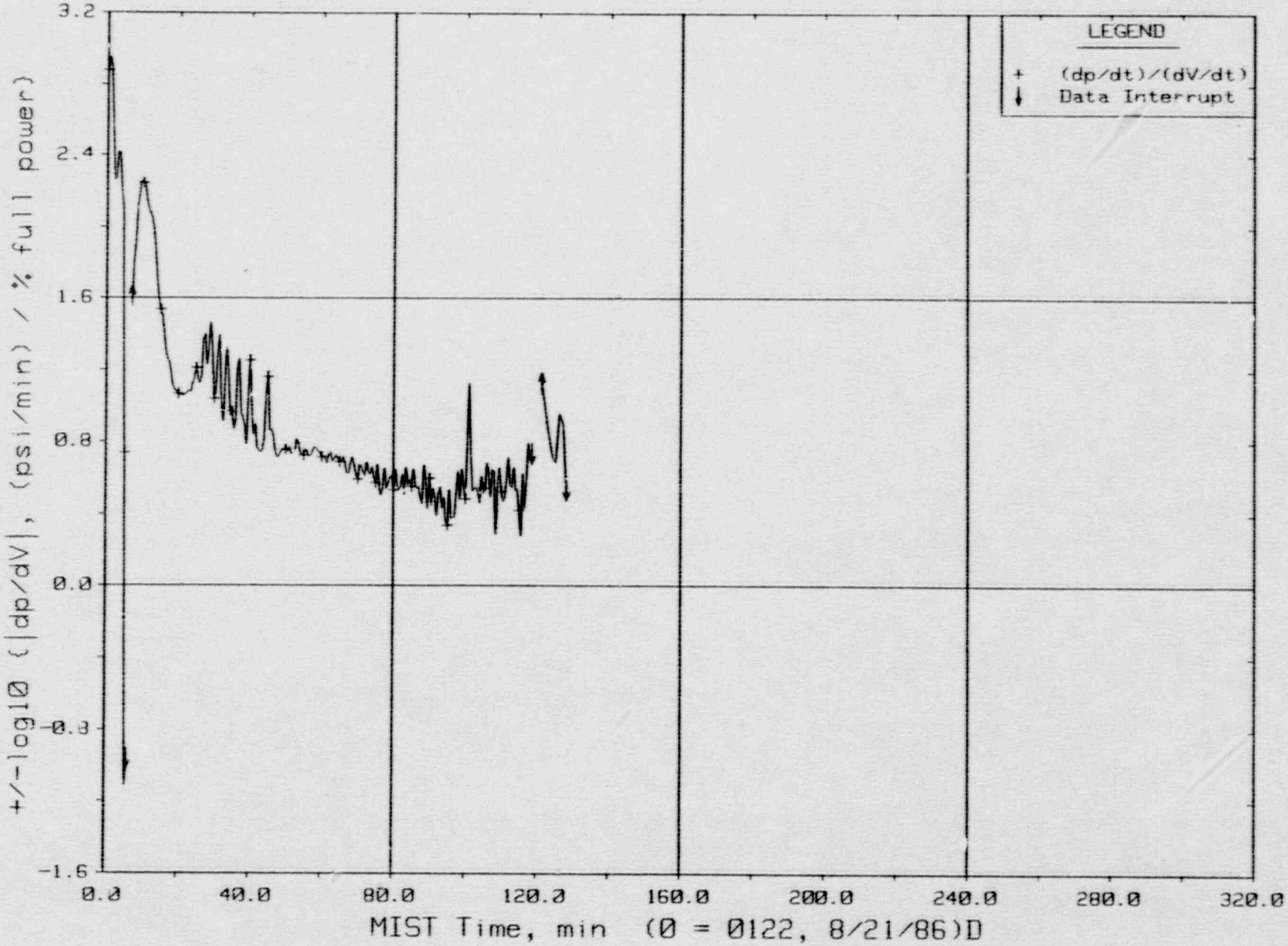

Primary System Response: dp/dt divided by dV/dt.

Tue May 31 12:12:32 1988

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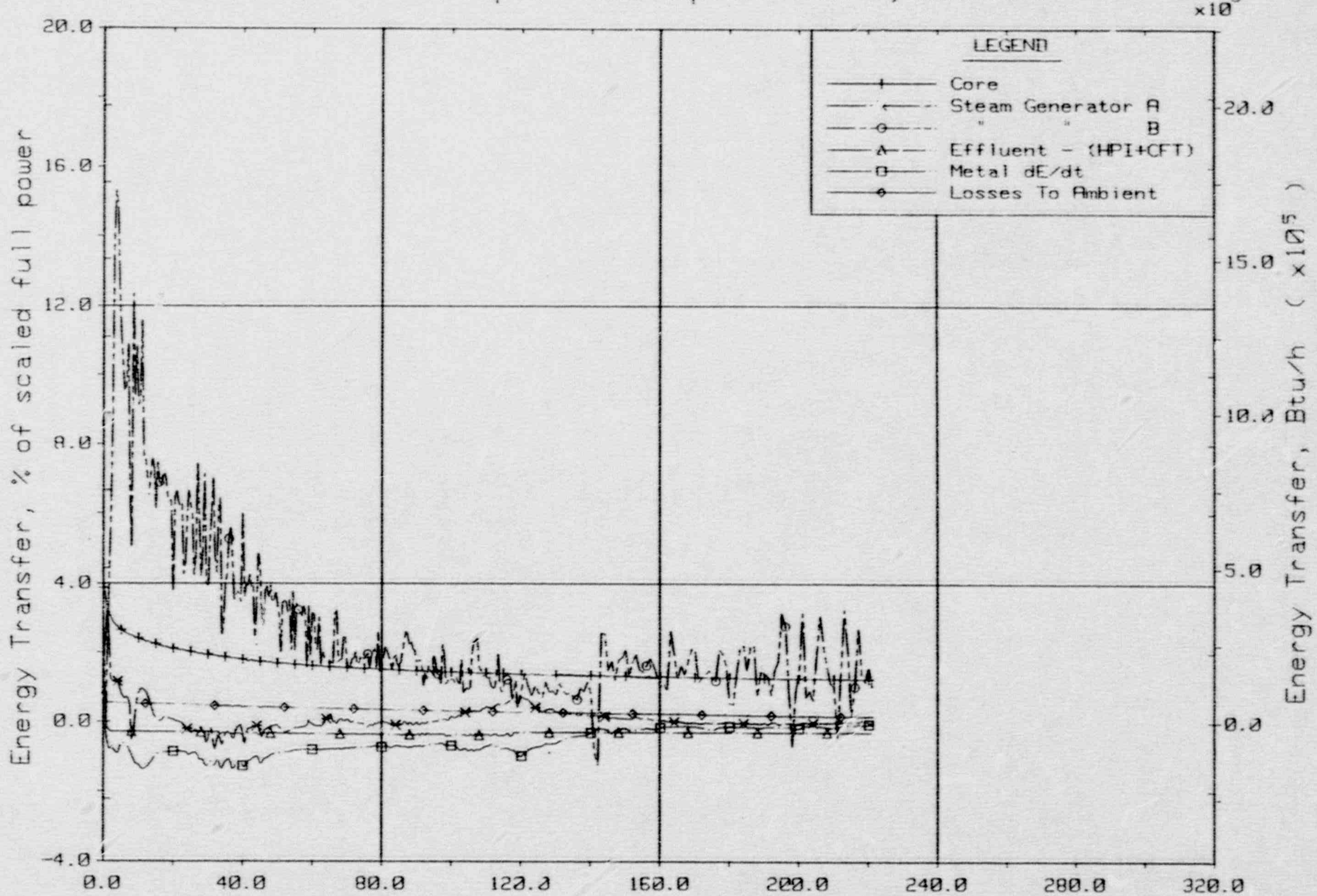

MIST Time, min (0 = 0122, 8/21/86)D

Primary System Energy Transfer.

Tue May 31 12:15:55 1988

PROE2

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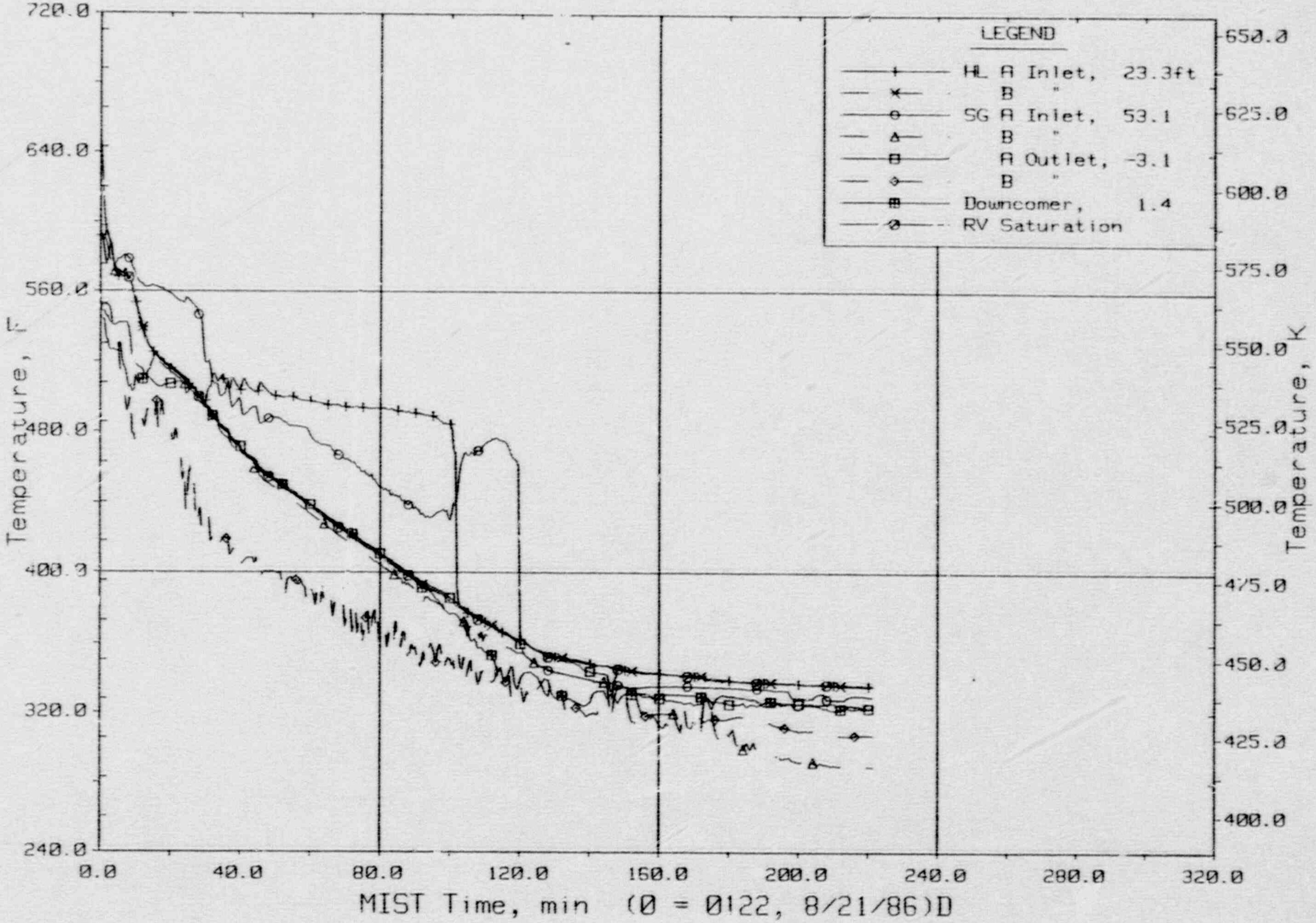

Primary System Fluid Temperatures (RTDs).

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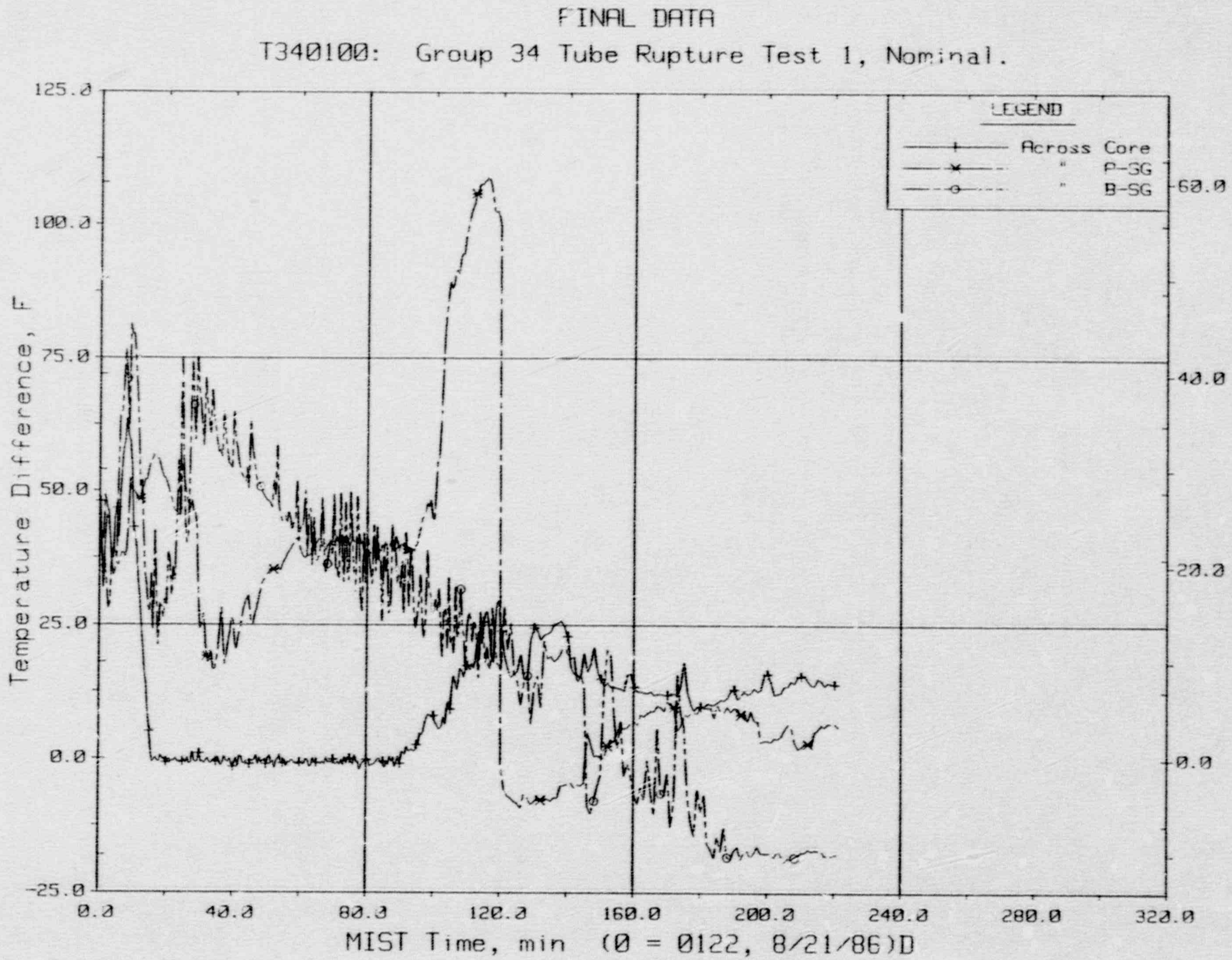
PRRT0



Key Temperature Differences.

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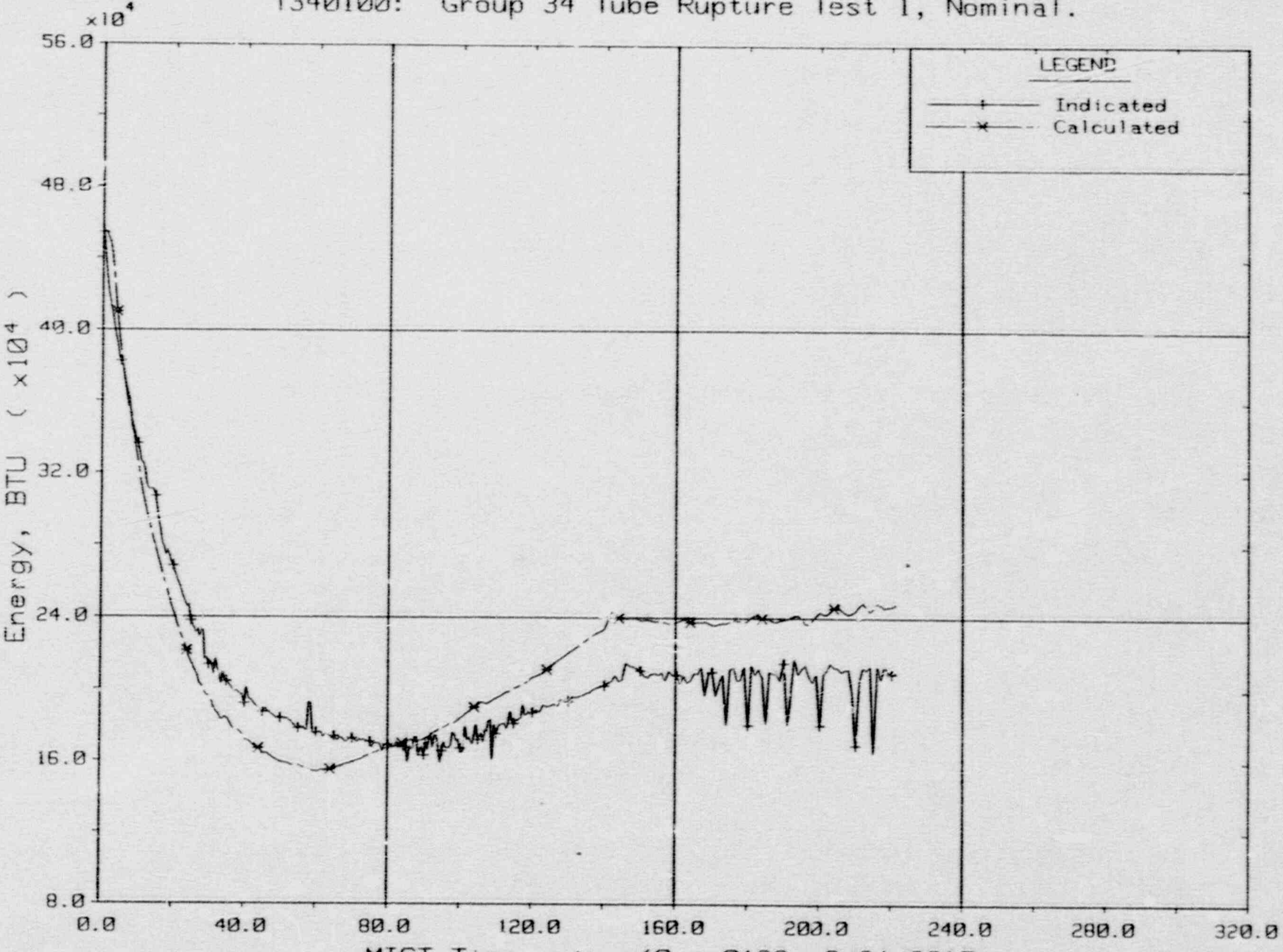
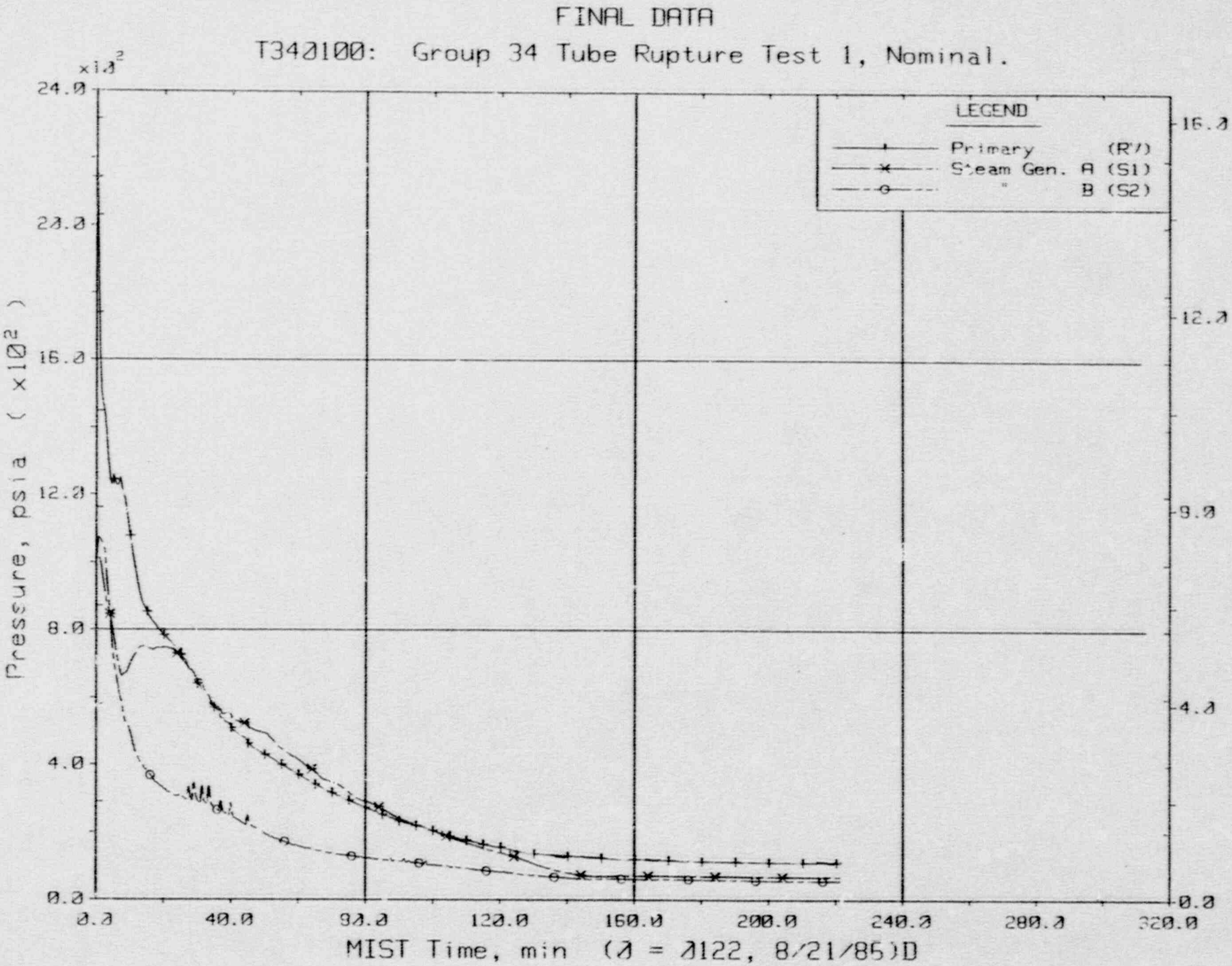


Primary System Total Fluid Energy.

Tue May 31 12:24:15 1988

PRTE2



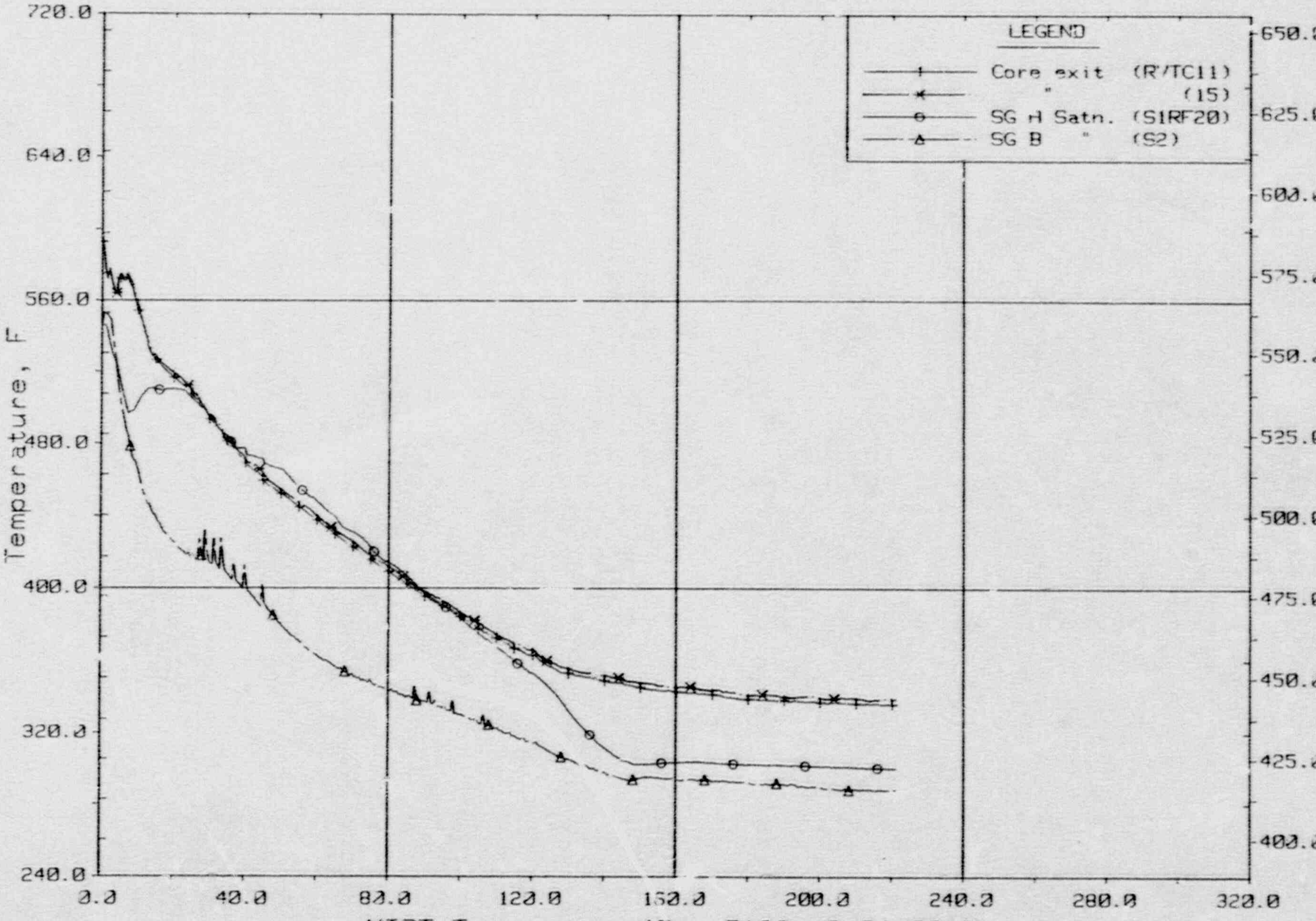
Primary and Secondary System Pressures (GP01s).

Tue May 31 12:26:29 1988

PSGP1

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



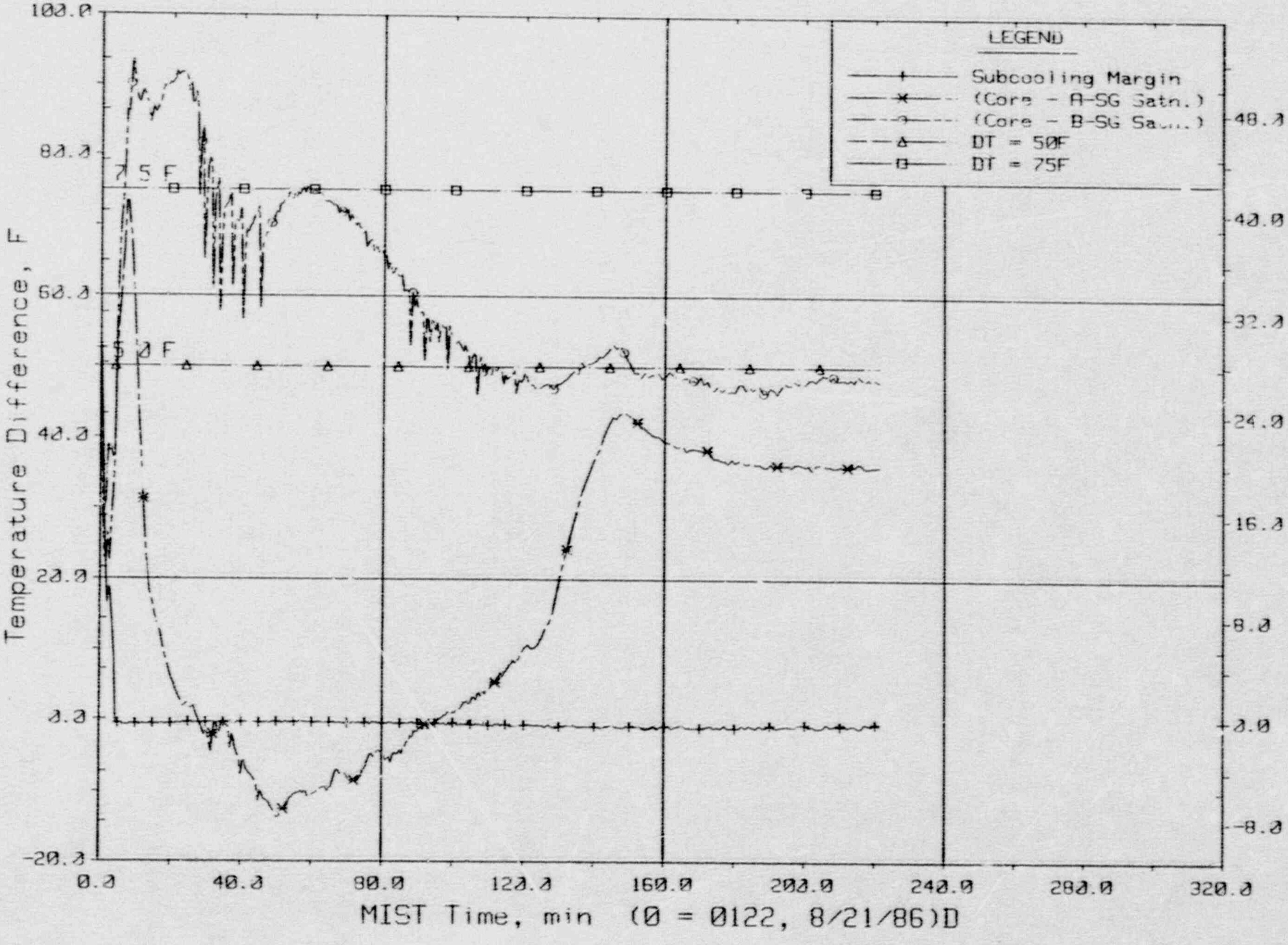
Core Exit and Steam Generator Secondary Saturation Temperatures.

Tue May 31 12:28:54 1988

PSTC3p

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



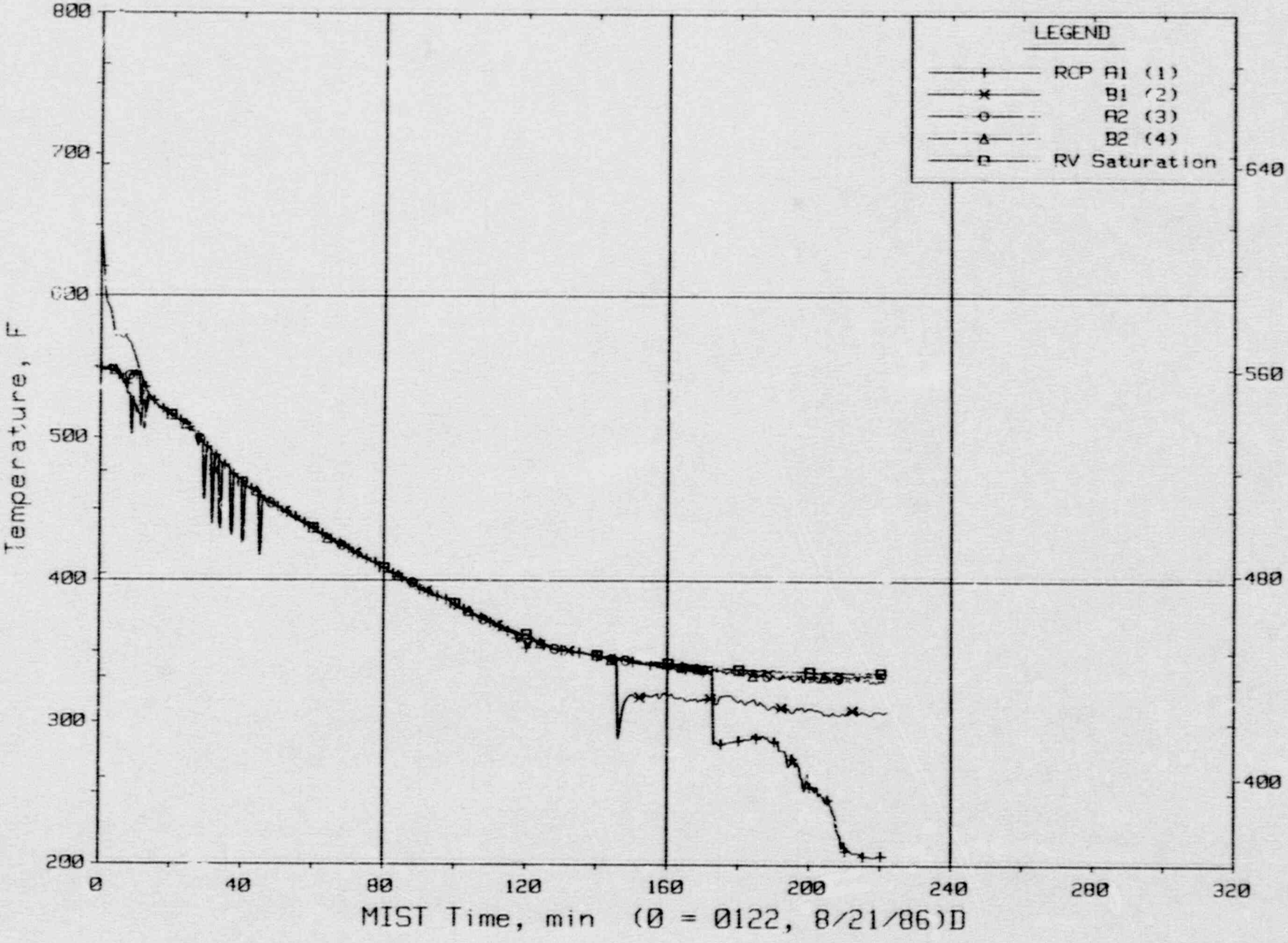
Control Temperature Differences.

Tue May 21 12:32:40 1991

PSTD1

FINAL DATA

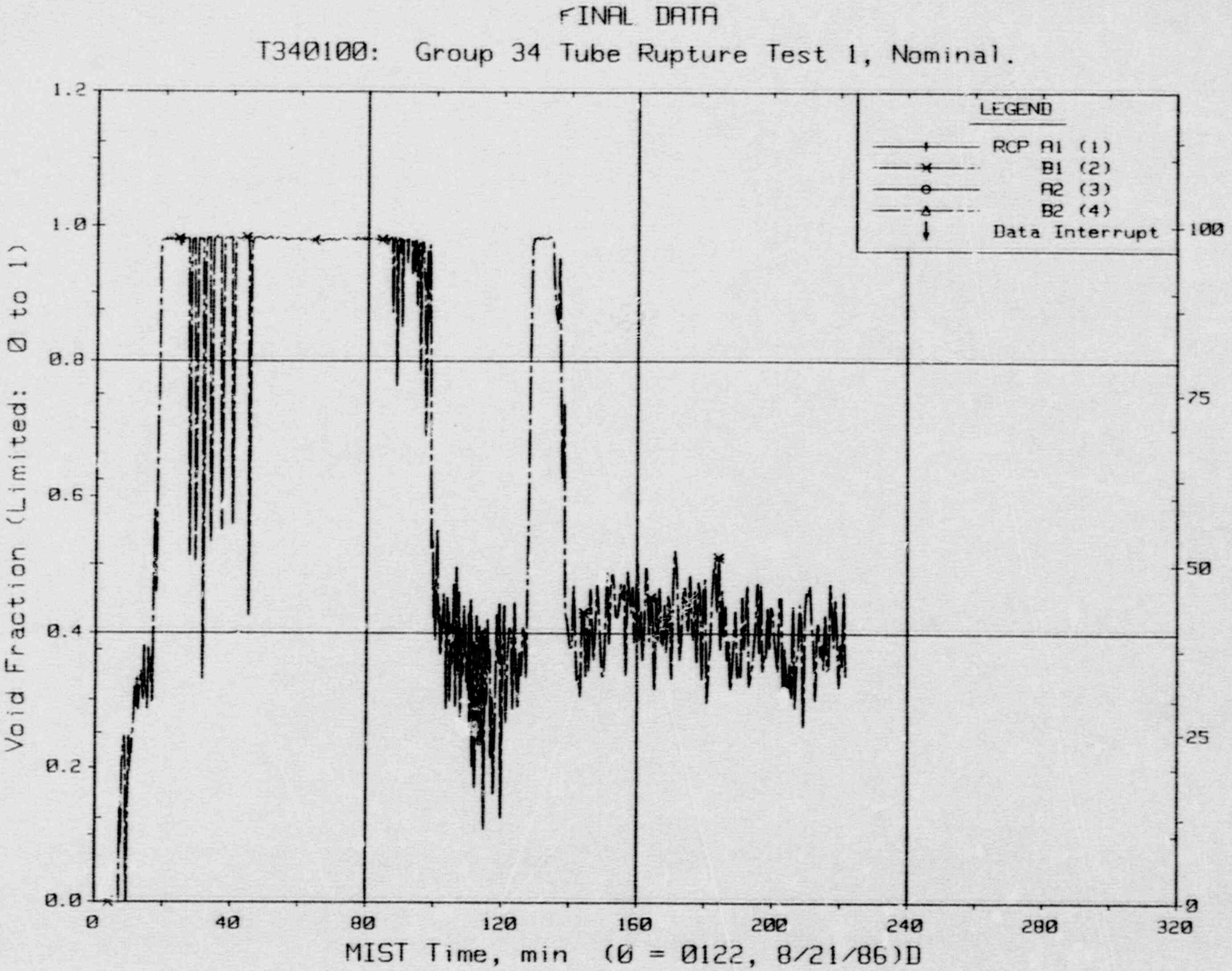
T340100: Group 34 Tube Rupture Test 1, Nominal.



Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 17:03:11 1988

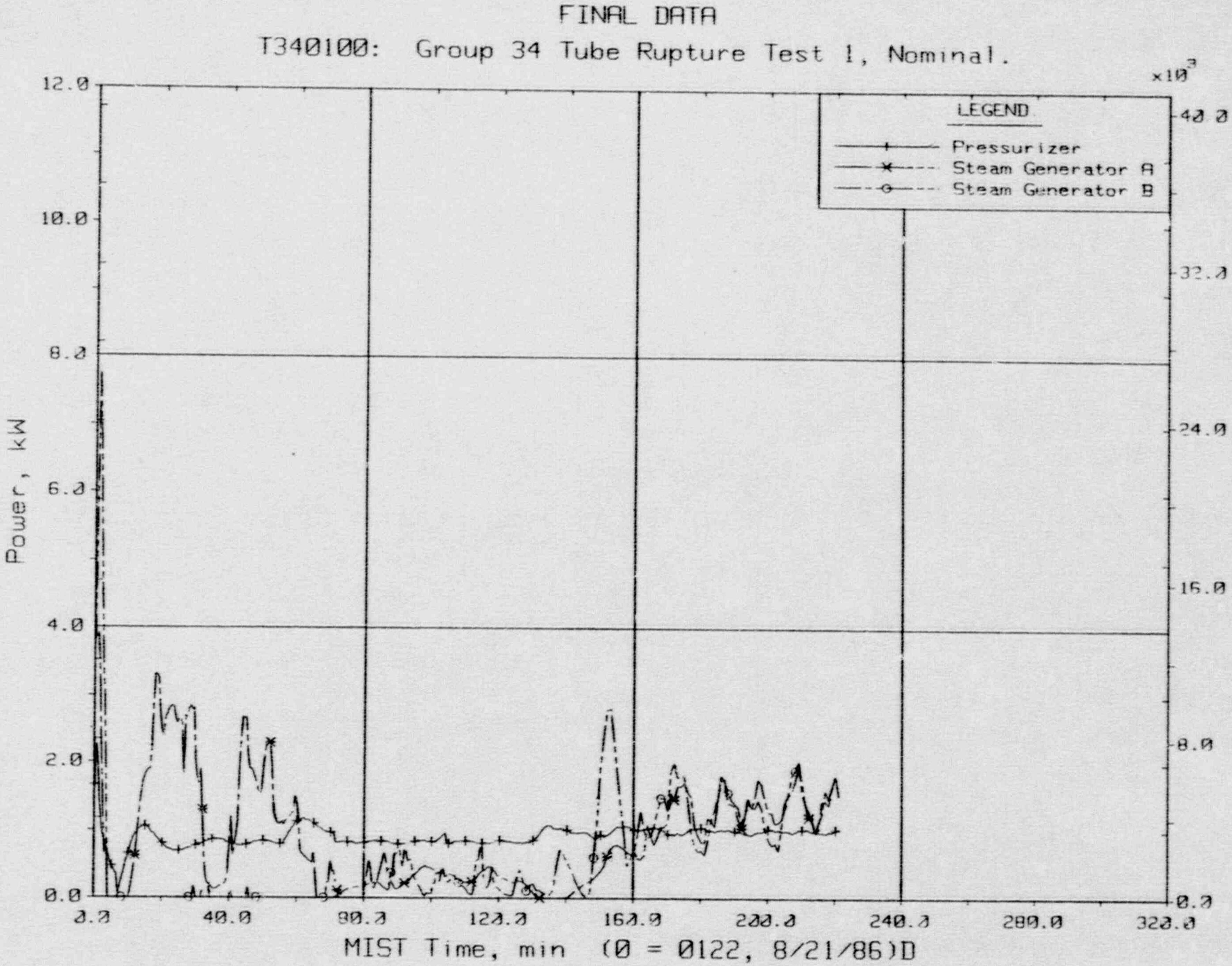
PURT0



Pump Suction Void Fraction From Gamma Densitometers (CnGD21).

Tue Nov 1 17:05:59 1988

PUVF2



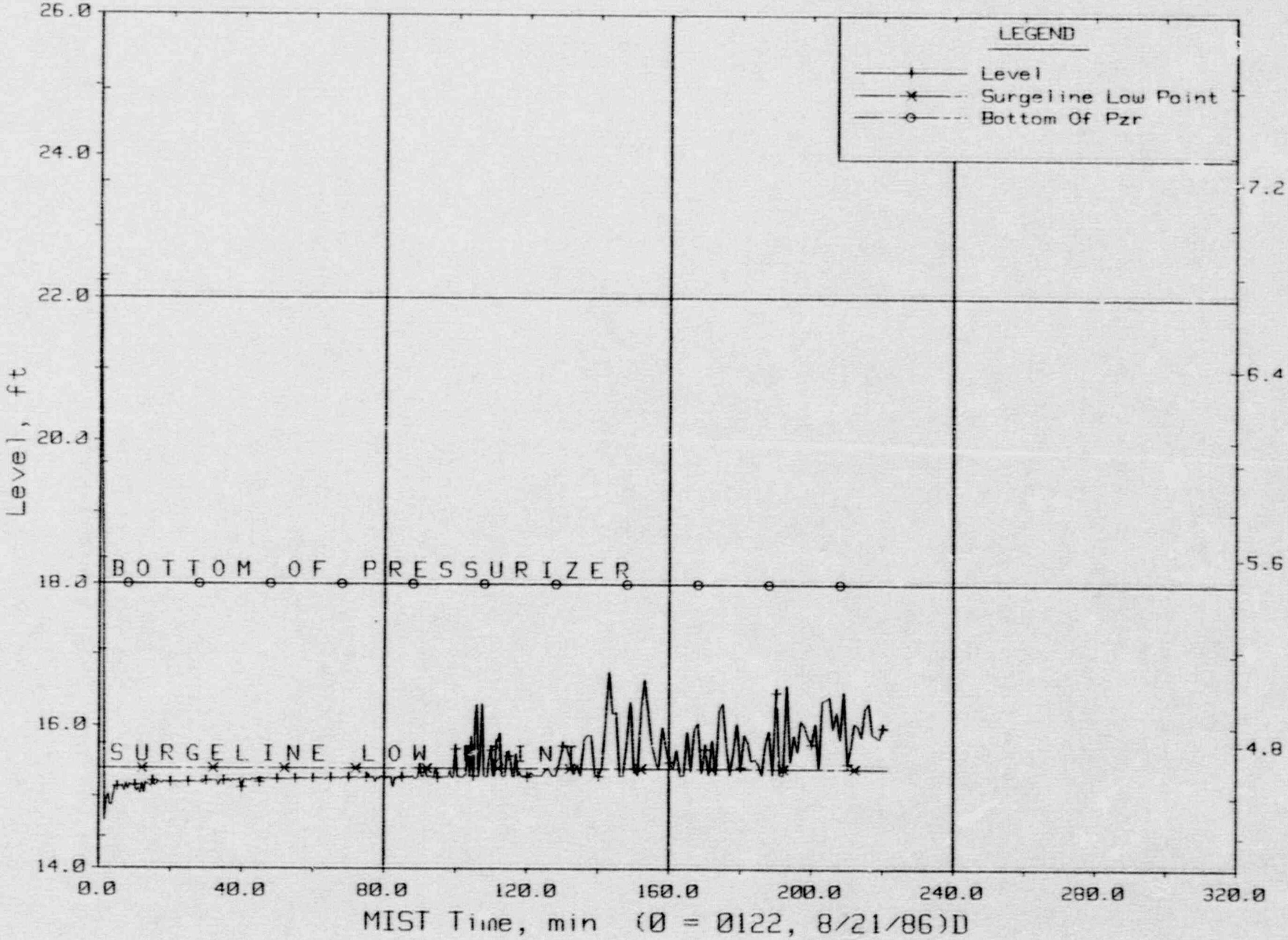
Guard Heater Specified Power, Pressurizer and Steam Generators.

Tue May 31 12:35:50 1999

PZGQ1

FINAL DATA

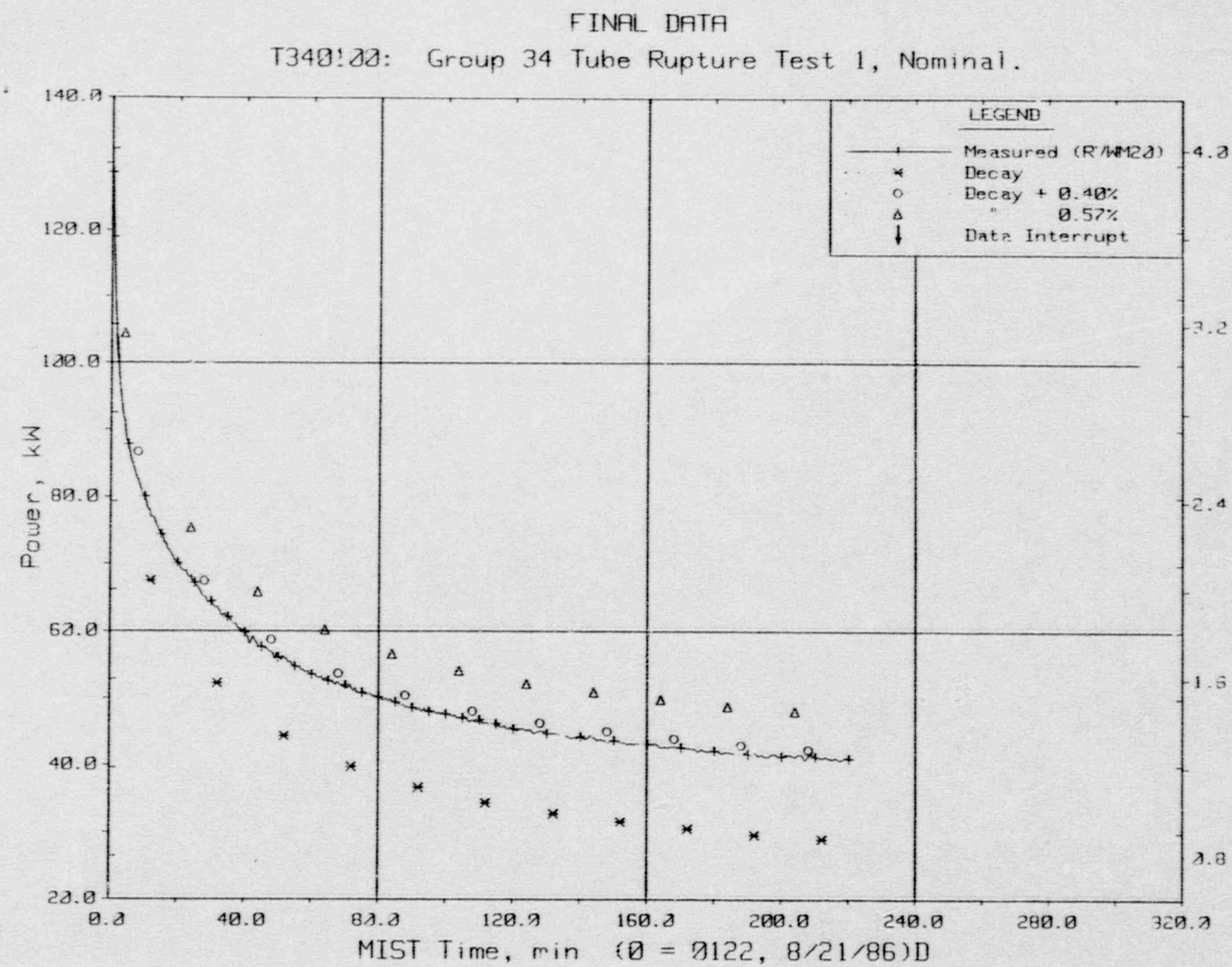
T340100: Group 34 Tube Rupture Test 1, Nominal.



Pressurizer Collapsed Liquid Level (PZLV20).

Tue May 31 12:39:10 1999

PZLV1



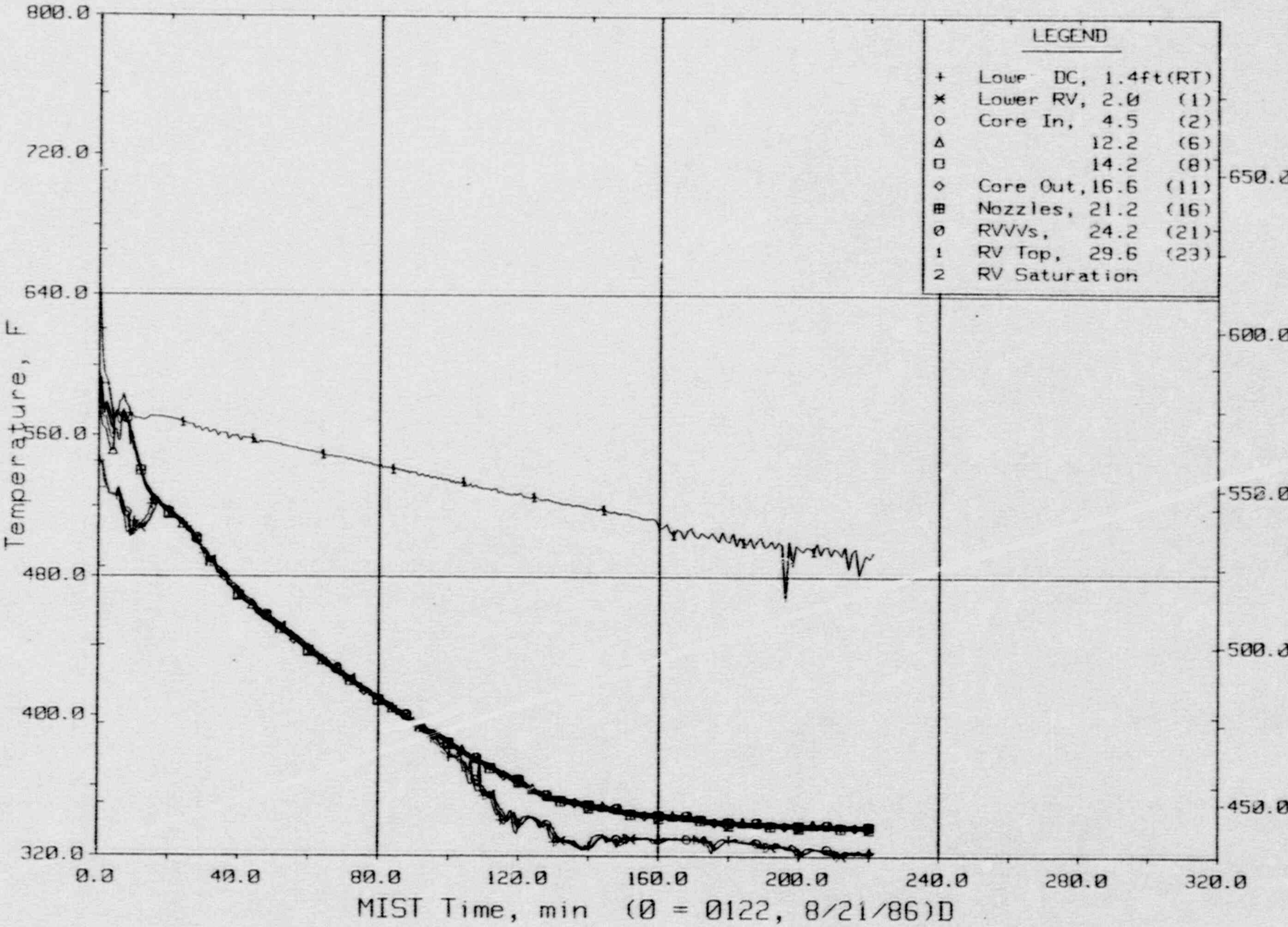
Core Power.

Tue May 31 12:41:37 1988

RVKW0

FINAL DATA

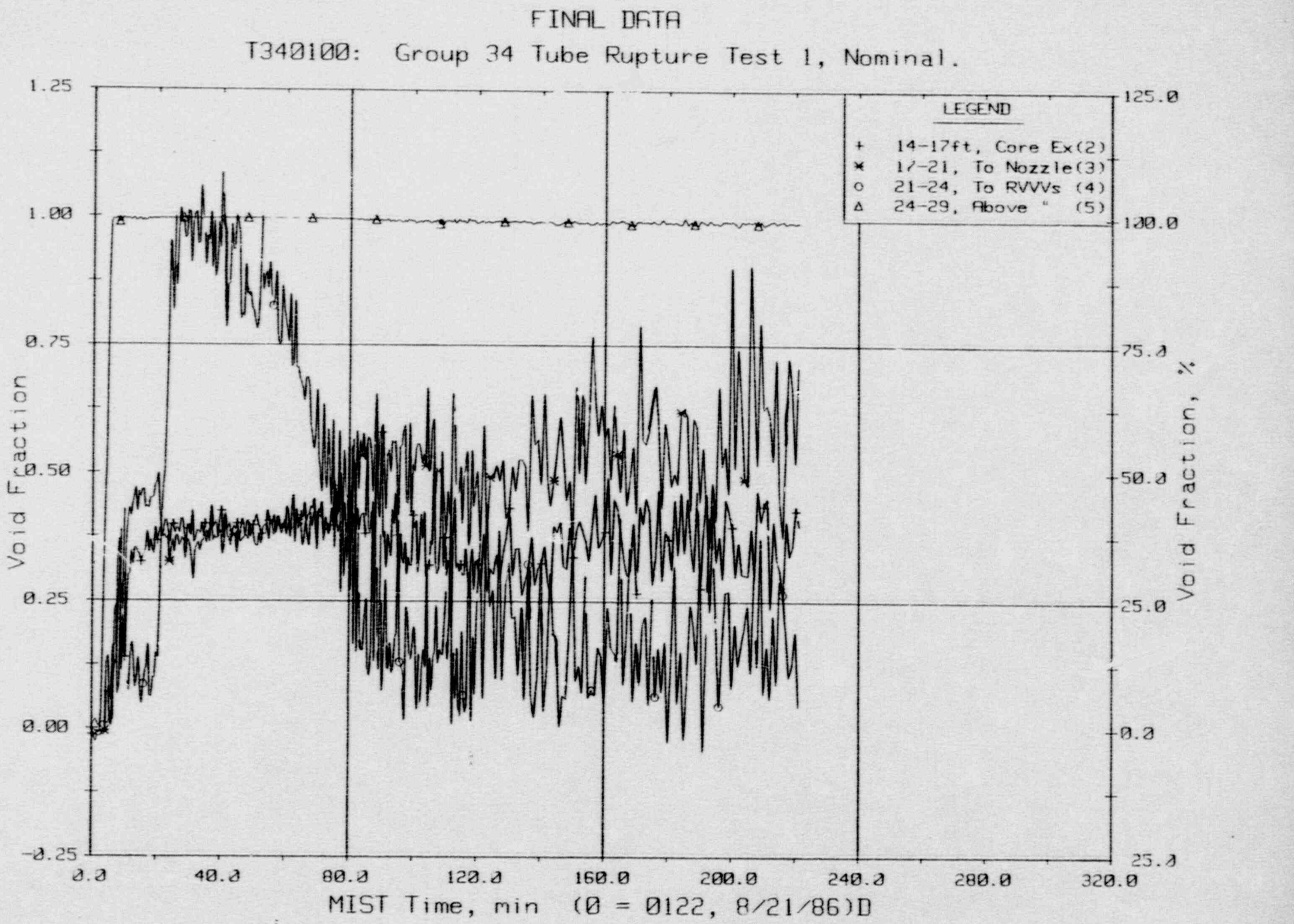
T340100: Group 34 Tube Rupture Test 1, Nominal.



Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Tue May 31 12:44:50 1988

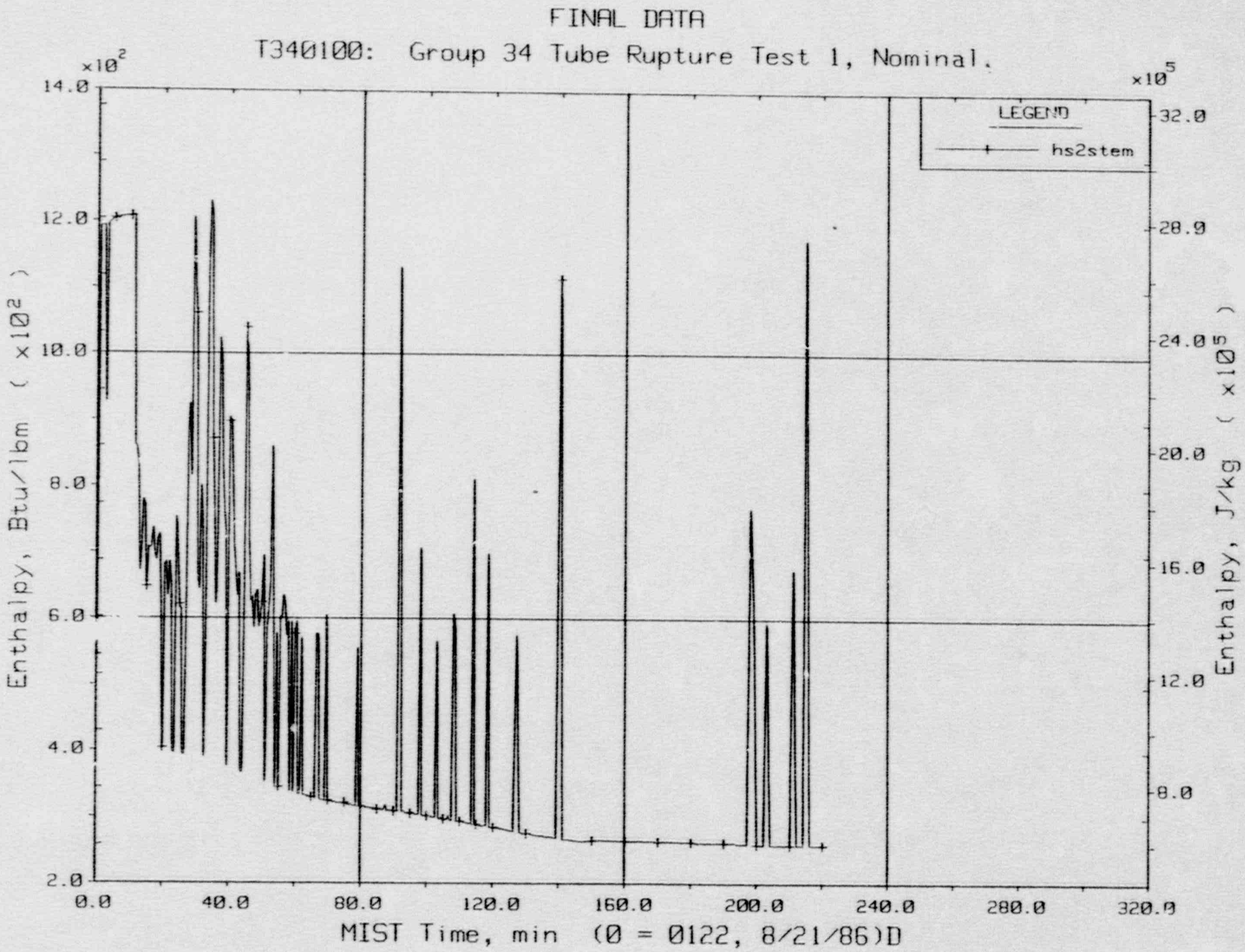
RVTC0



Reactor Vessel Void Fractions From Differential Pressures (RVVFs).

Tue May 31 12:47:39 1988

RVVF1



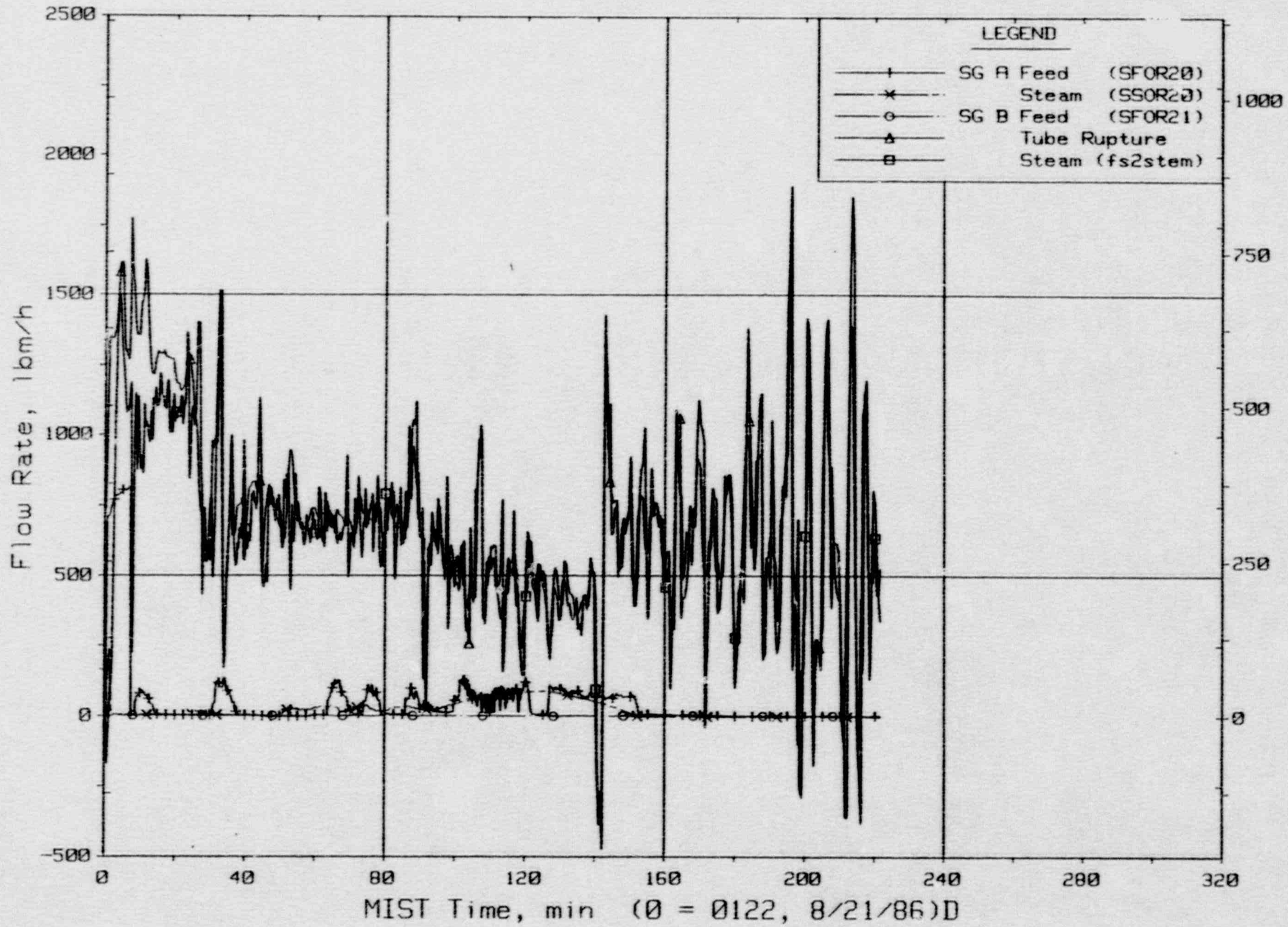
Steam Generator B Steam Exit Enthalpy.

Wed Jun 1 08:34:57 1988

S2EN1

FINAL DATA

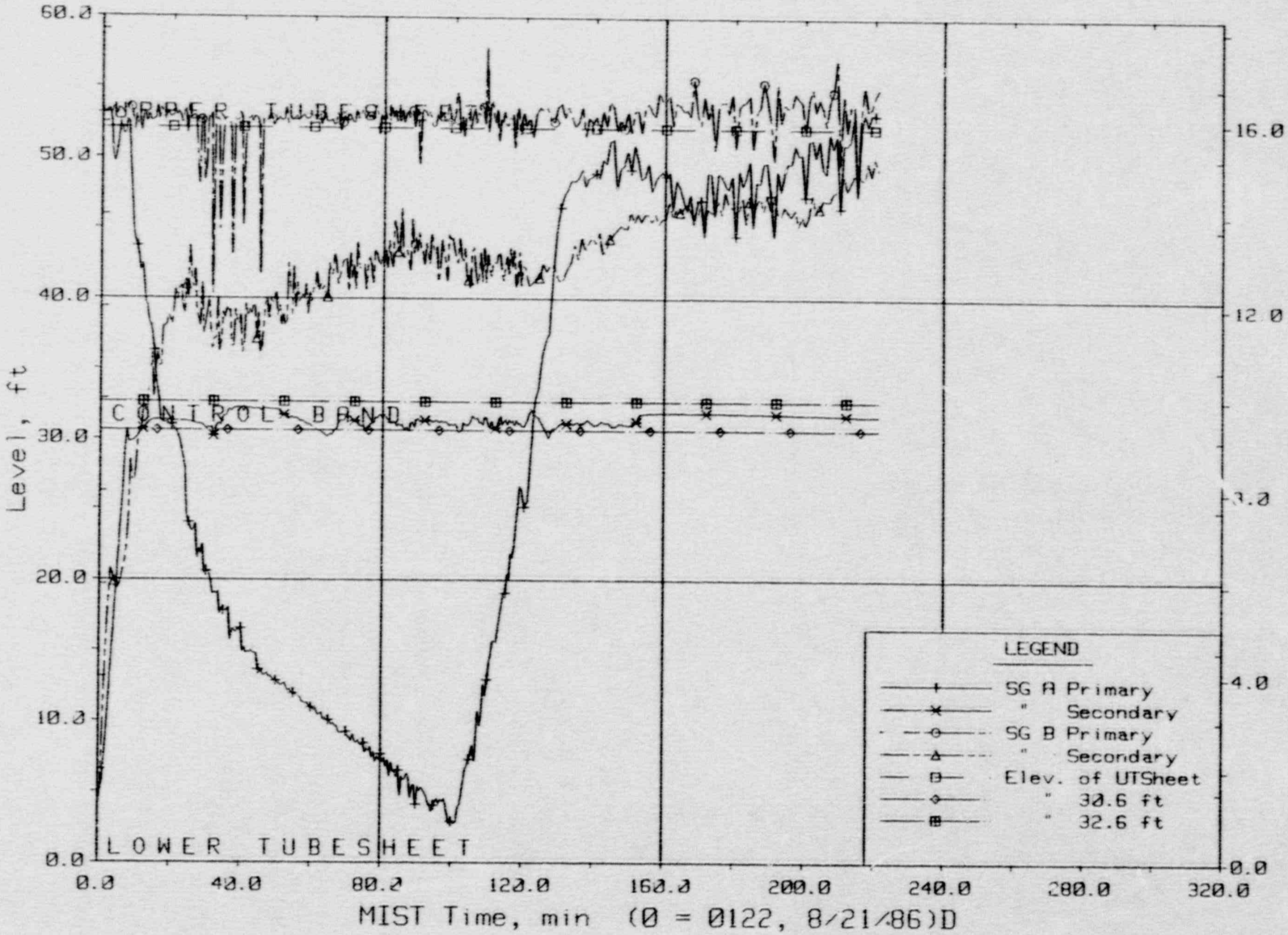
T340100: Group 34 Tube Rupture Test 1, Nominal.



Steam Generator Secondary Flow Rates.

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



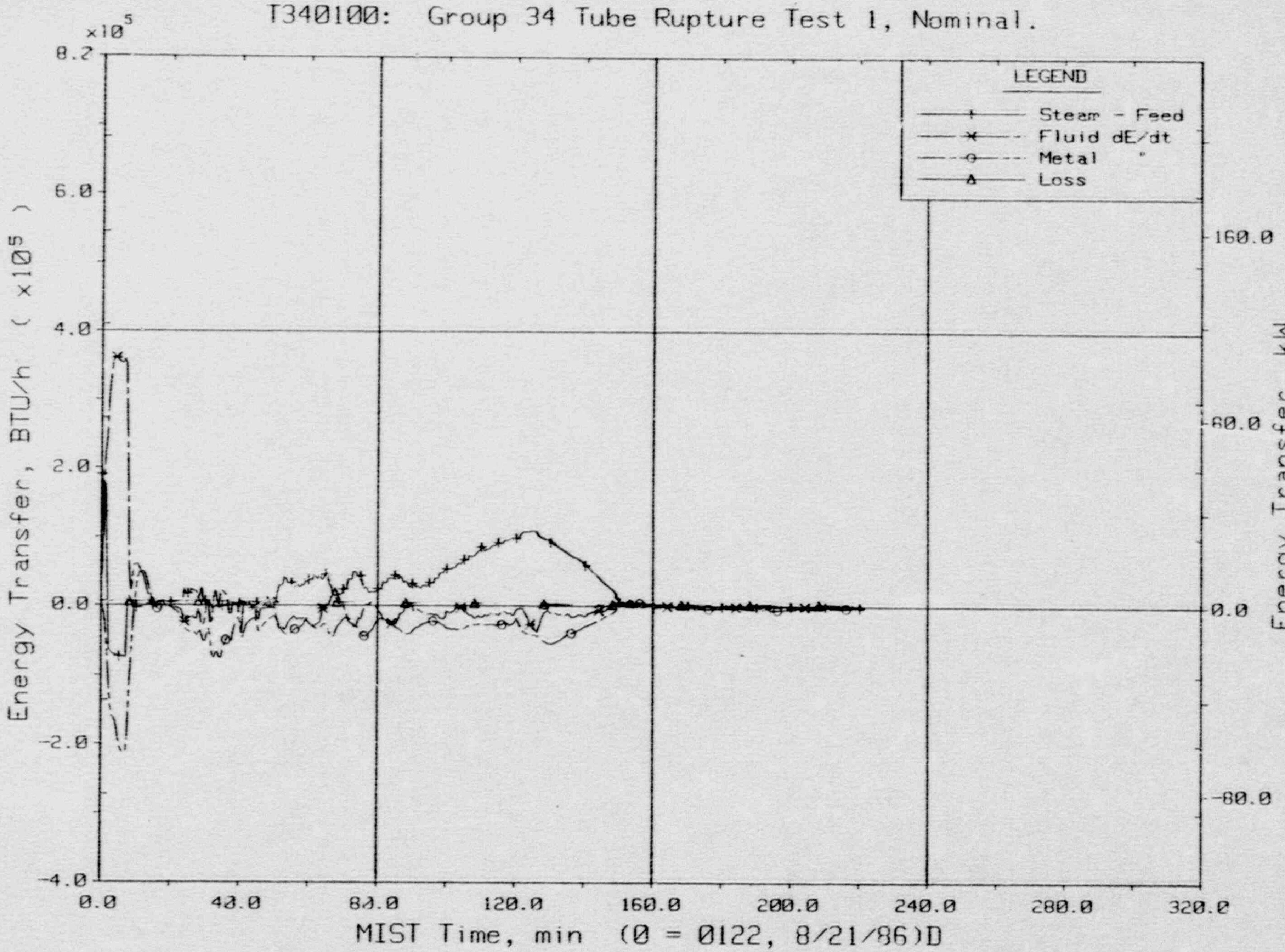
Steam Generator Collapsed Liquid Levels.

Tue May 31 12:53:36 1999

SGLV2

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



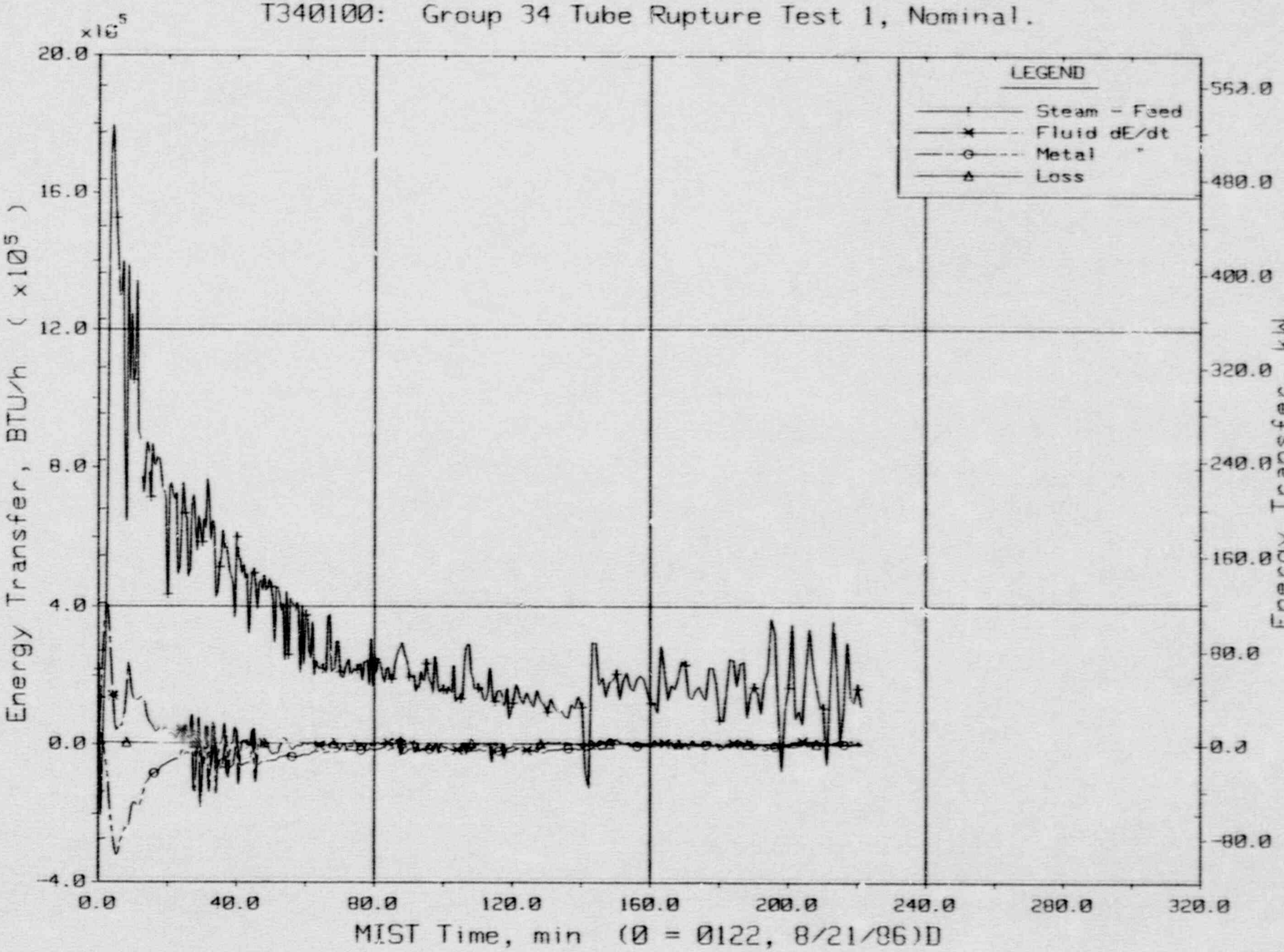
Steam Generator F Energy Transfer.

Tue May 31 12:59:26 1988

SGQE3

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



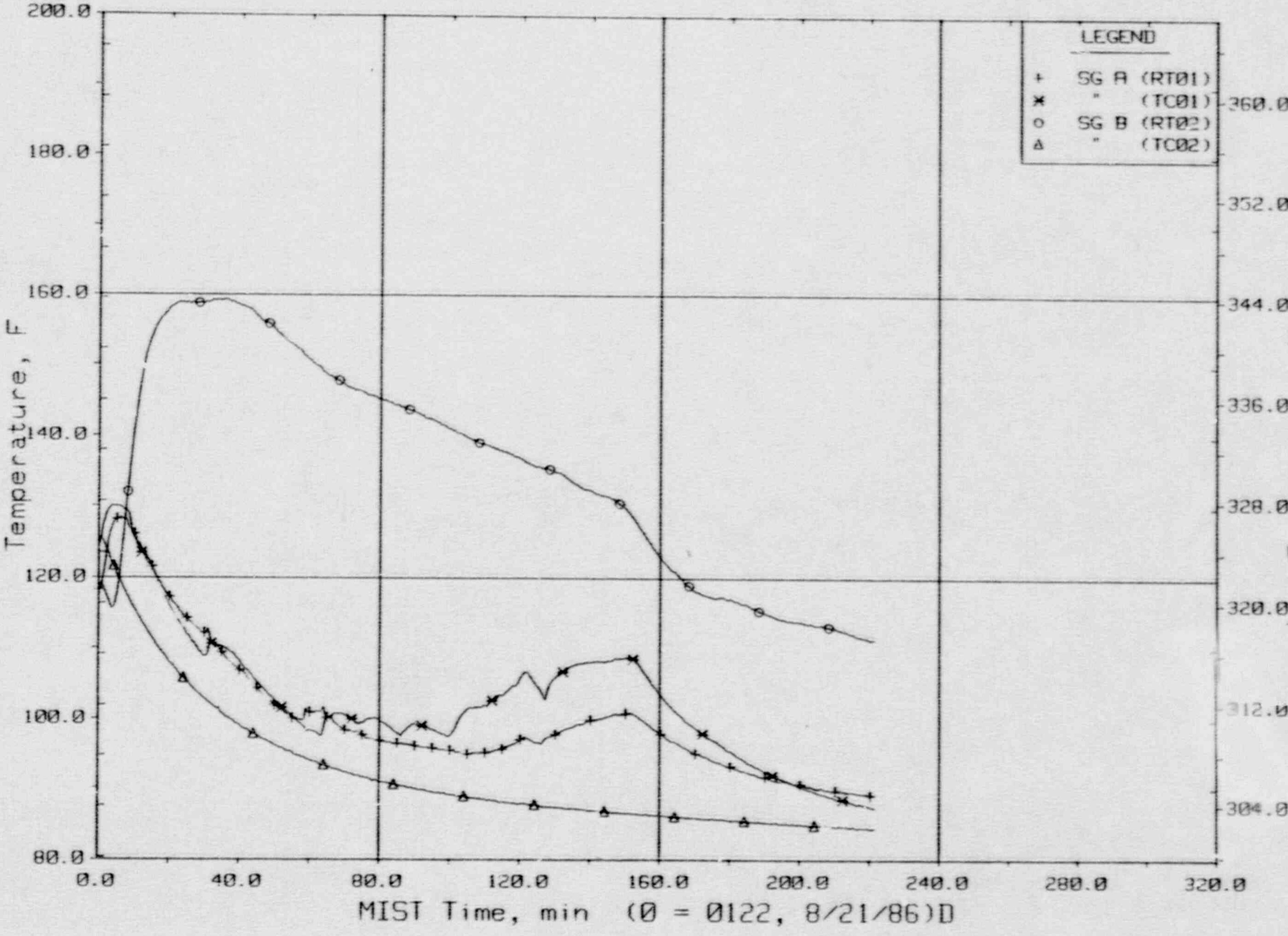
Steam Generator B Energy Transfer.

Tue May 31 13:03:17 1988

SGQE6

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



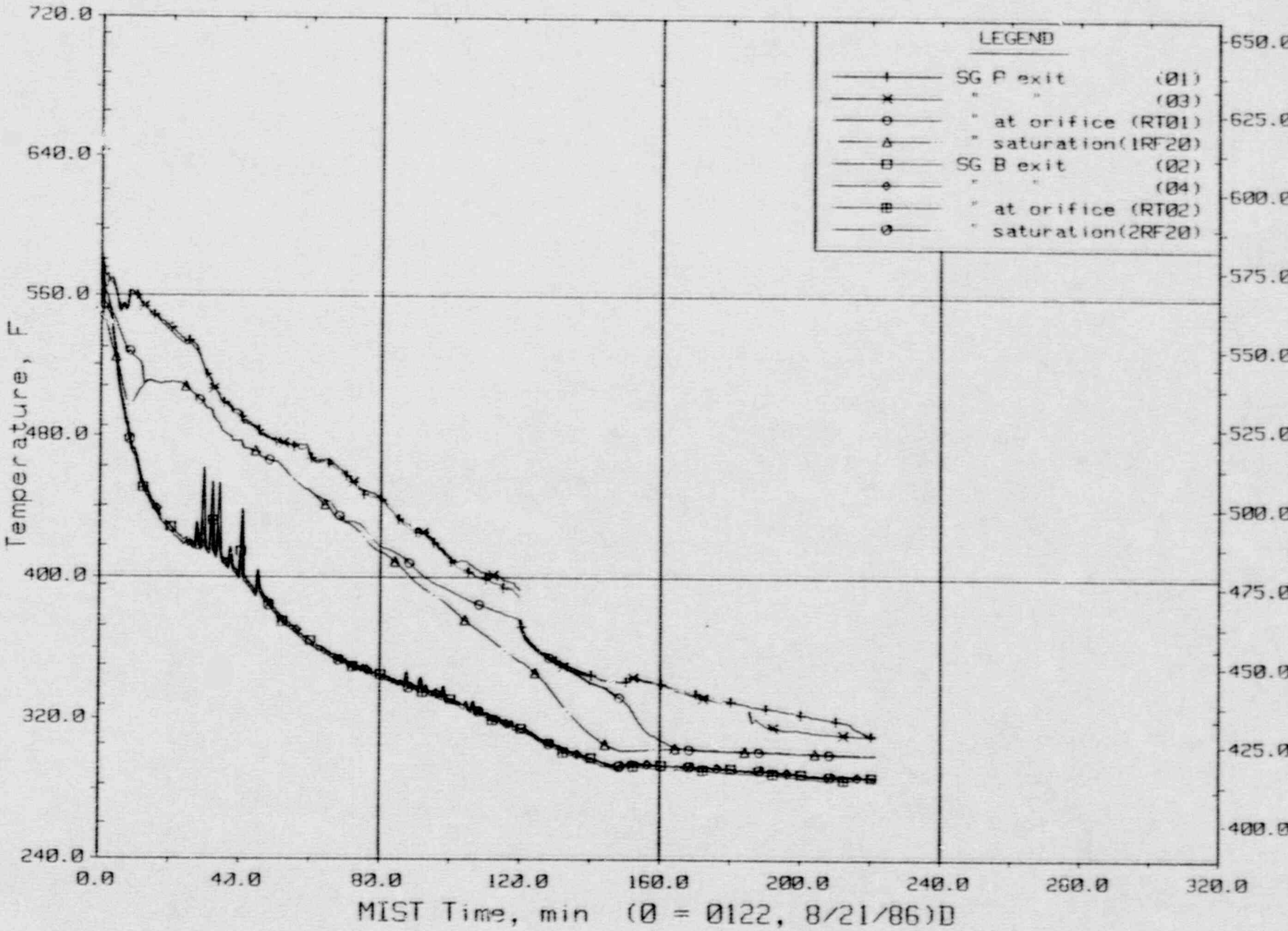
Feedwater Temperatures (SFs).

Tue May 31 13:06:11 1988

SGRT1

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



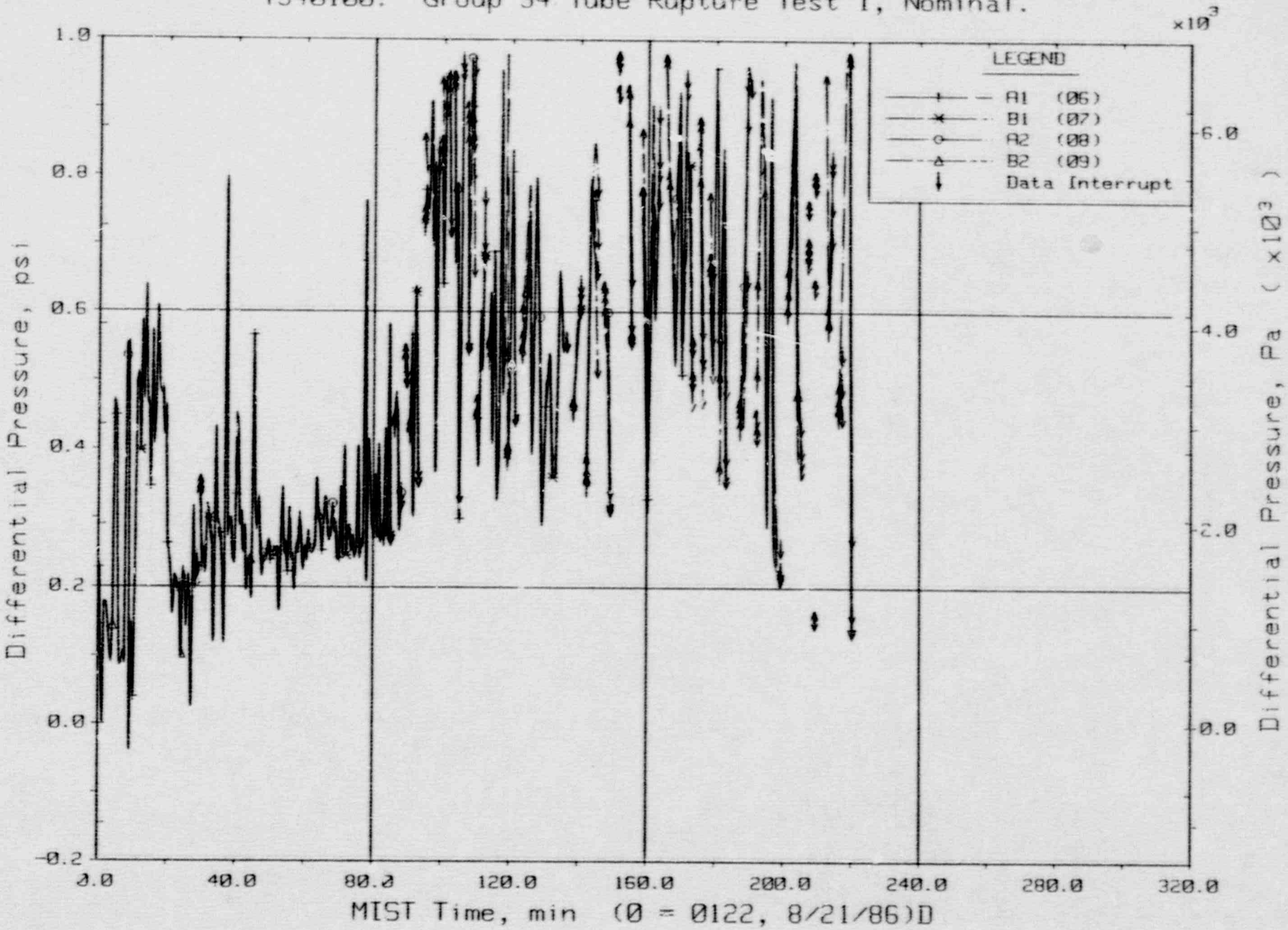
Steam Generator Steam Outlet Temperatures (SSTCs).

Tue May 31 13:16:25 1988

SGRT2

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



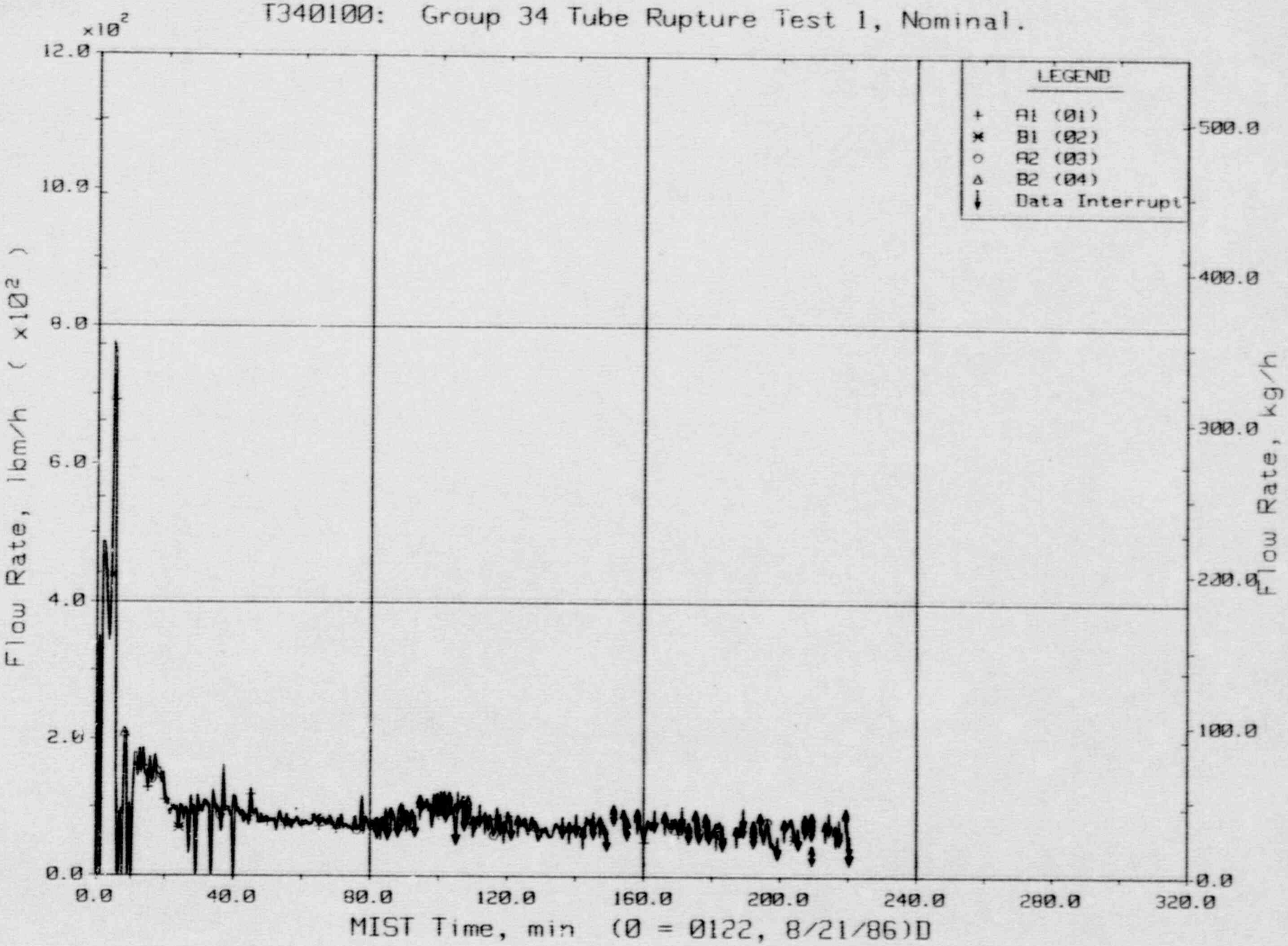
Reactor Vessel Vent Valve Differential Pressures (RVDFs).

Tue May 31 13:23:33 1988

VVDP1

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



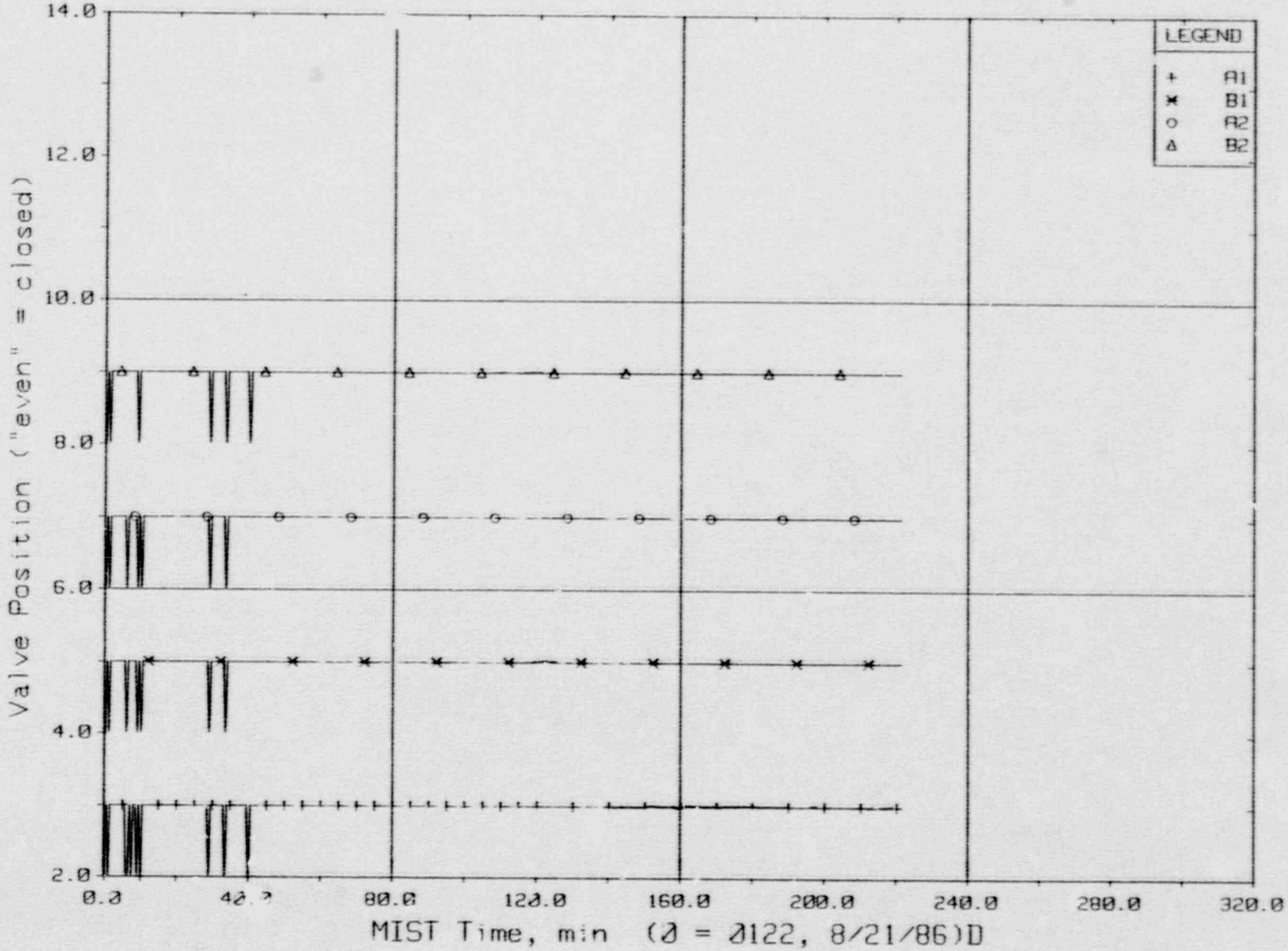
Reactor Vessel Vent Valve Flow Rates (RVORs).

Tue May 31 13:28:22 1988

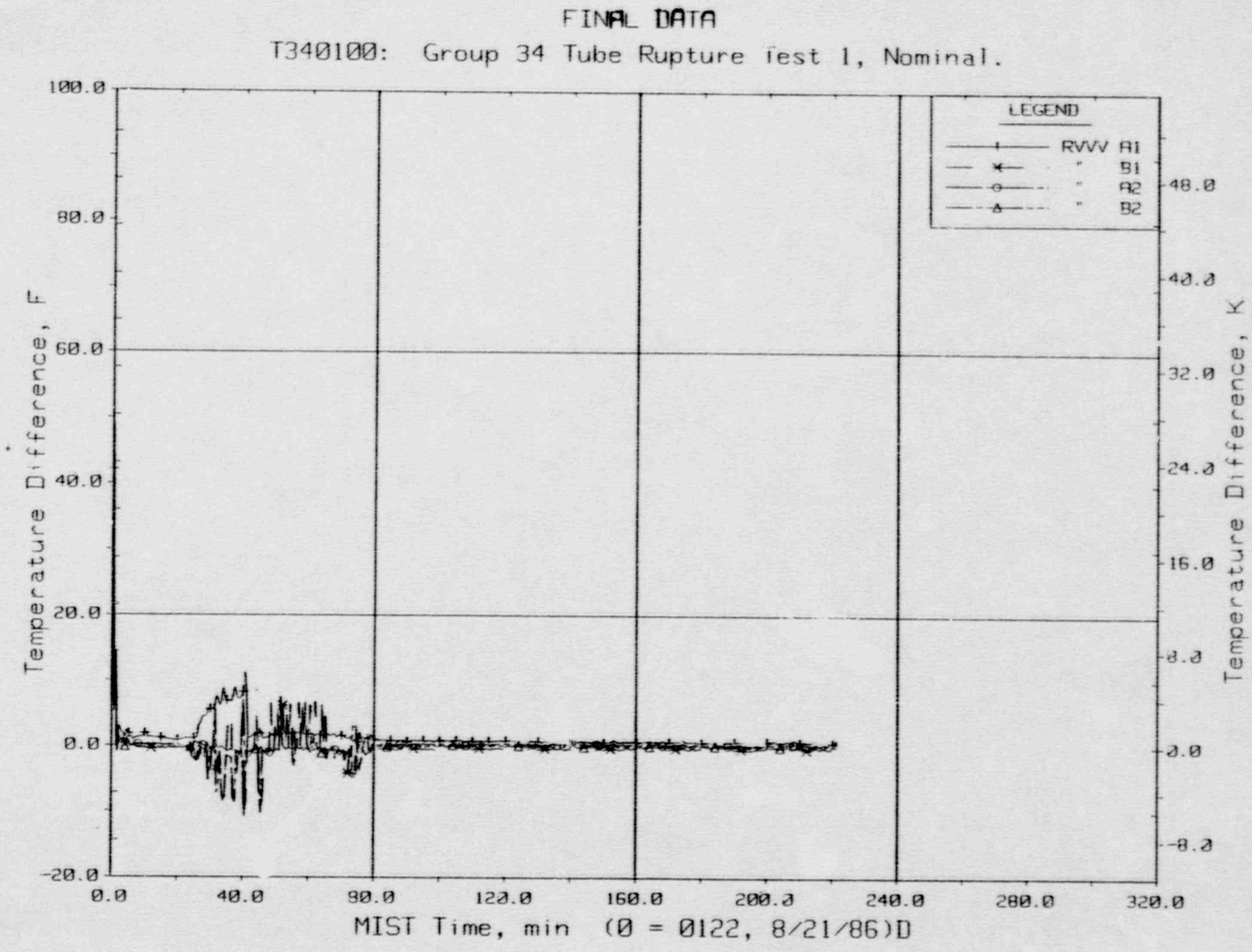
VVFL1

FINAL DATA

T340100: Group 34 Tube Rupture Test 1, Nominal.



Reactor Vessel Vent Valve Positions.



Temperature Differences Across Vent Valves.

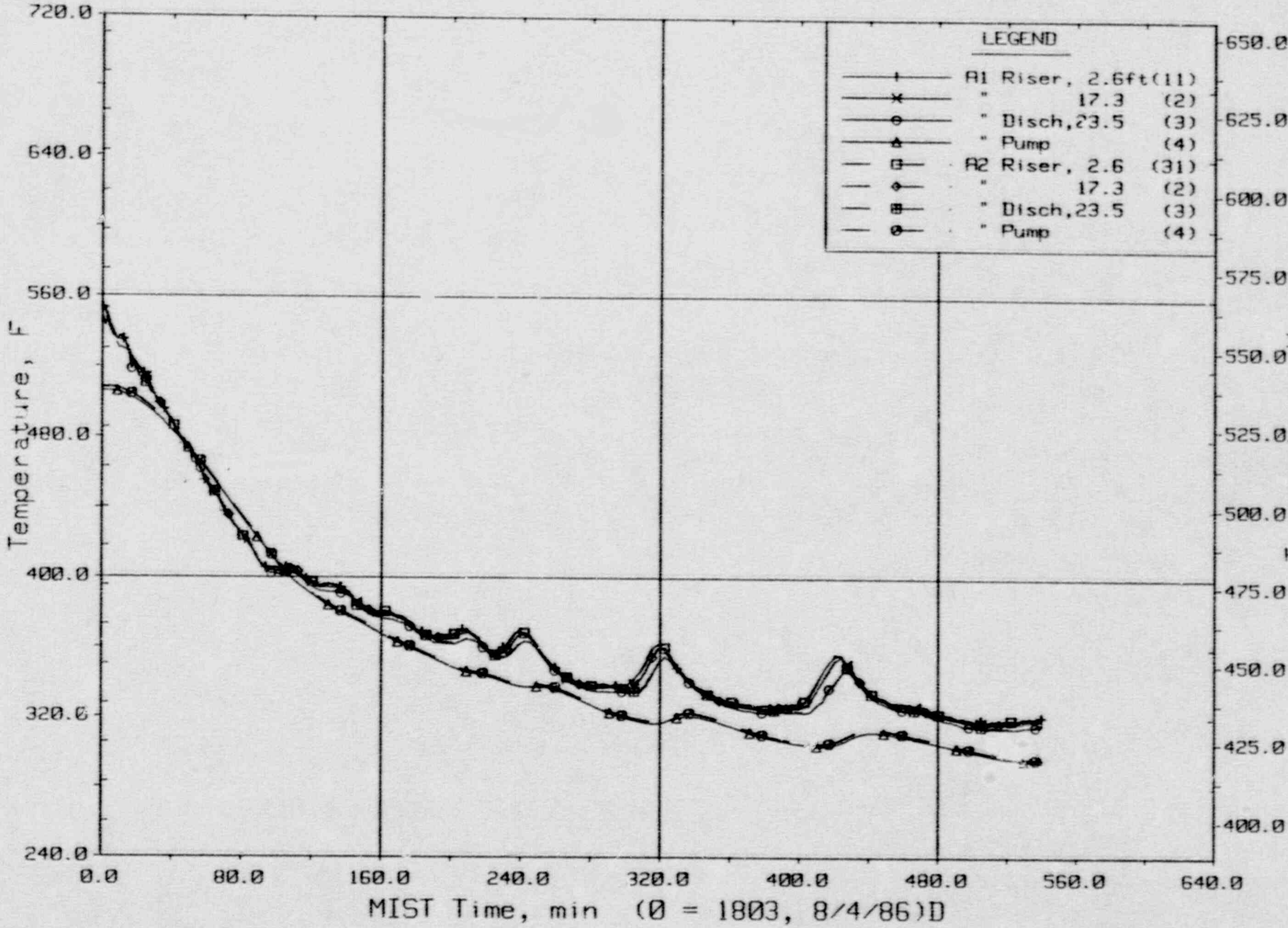
Tue May 31 13:25:50 1988

VVTD1

T340213

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



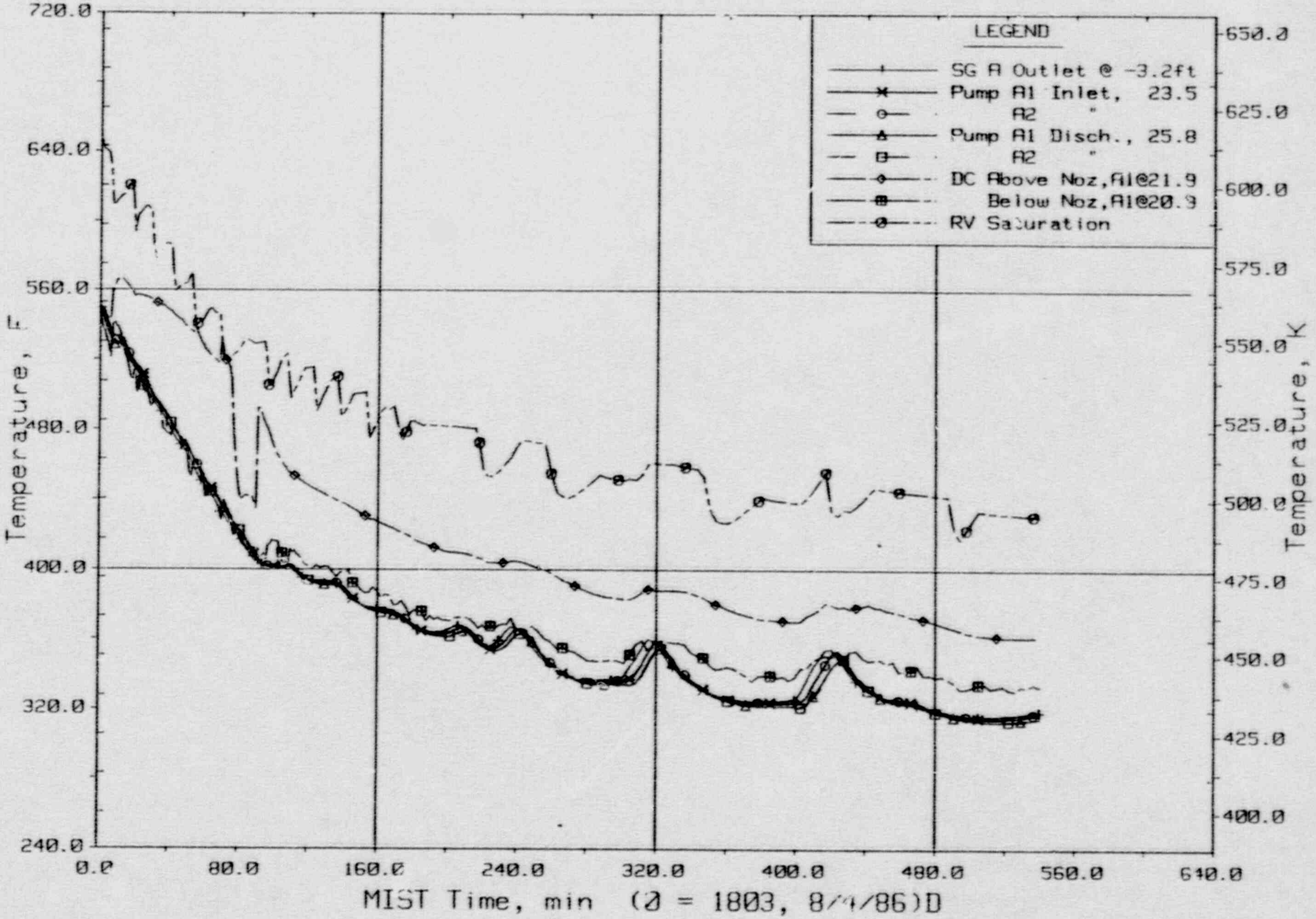
Loop A Cold Leg Metal Temperatures (C1,3MT-).

Tue May 31 14:25:51 1988

C1MT1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



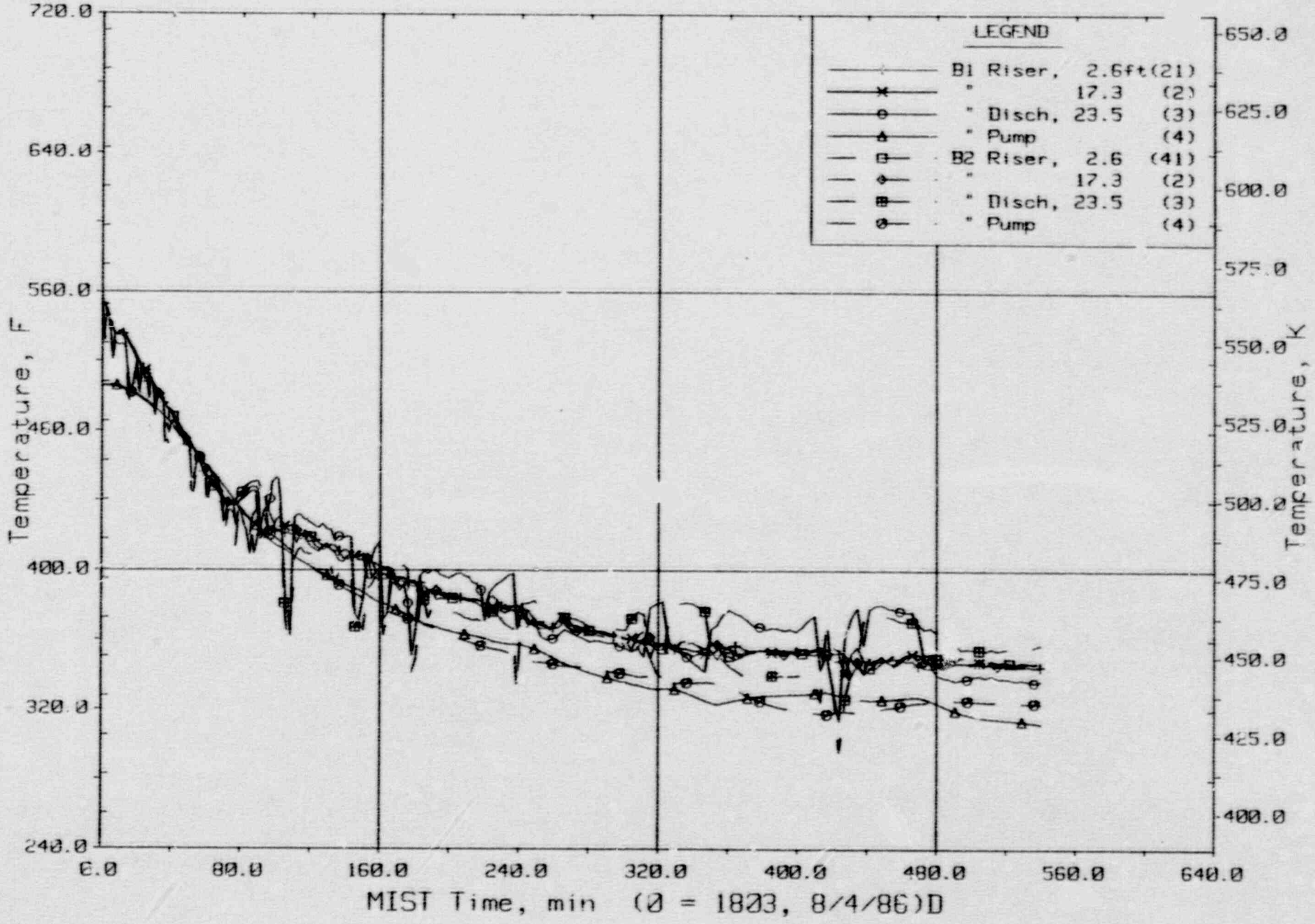
Loop A Cold Leg Fluid Temperatures (RTDs).

Tue May 31 14:29:40 1988

C1RT0

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurizat



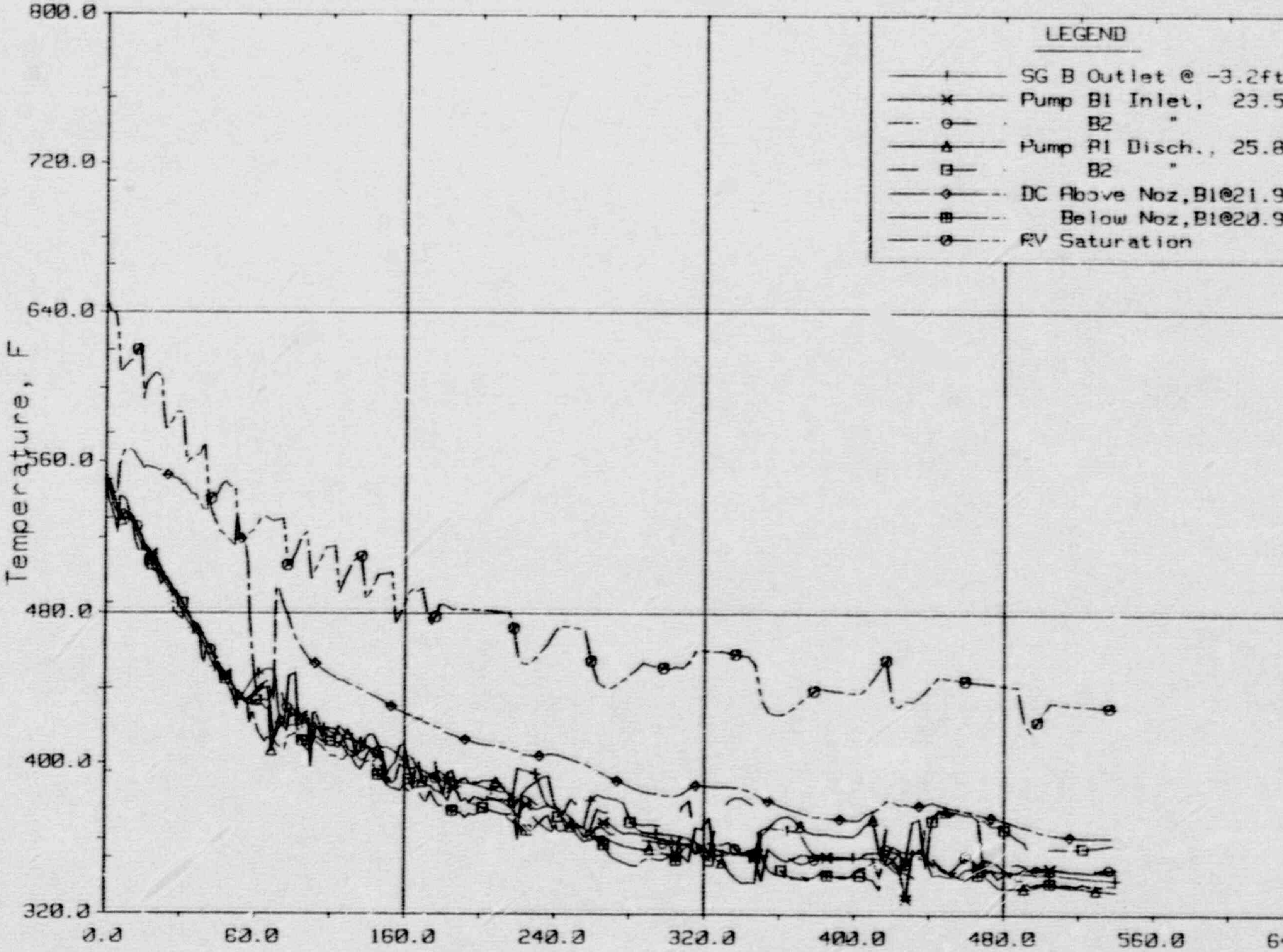
Loop B Cold Leg Metal Temperatures (C2,4MTs).

Tue May 31 14:33:44 1988

C2MT1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



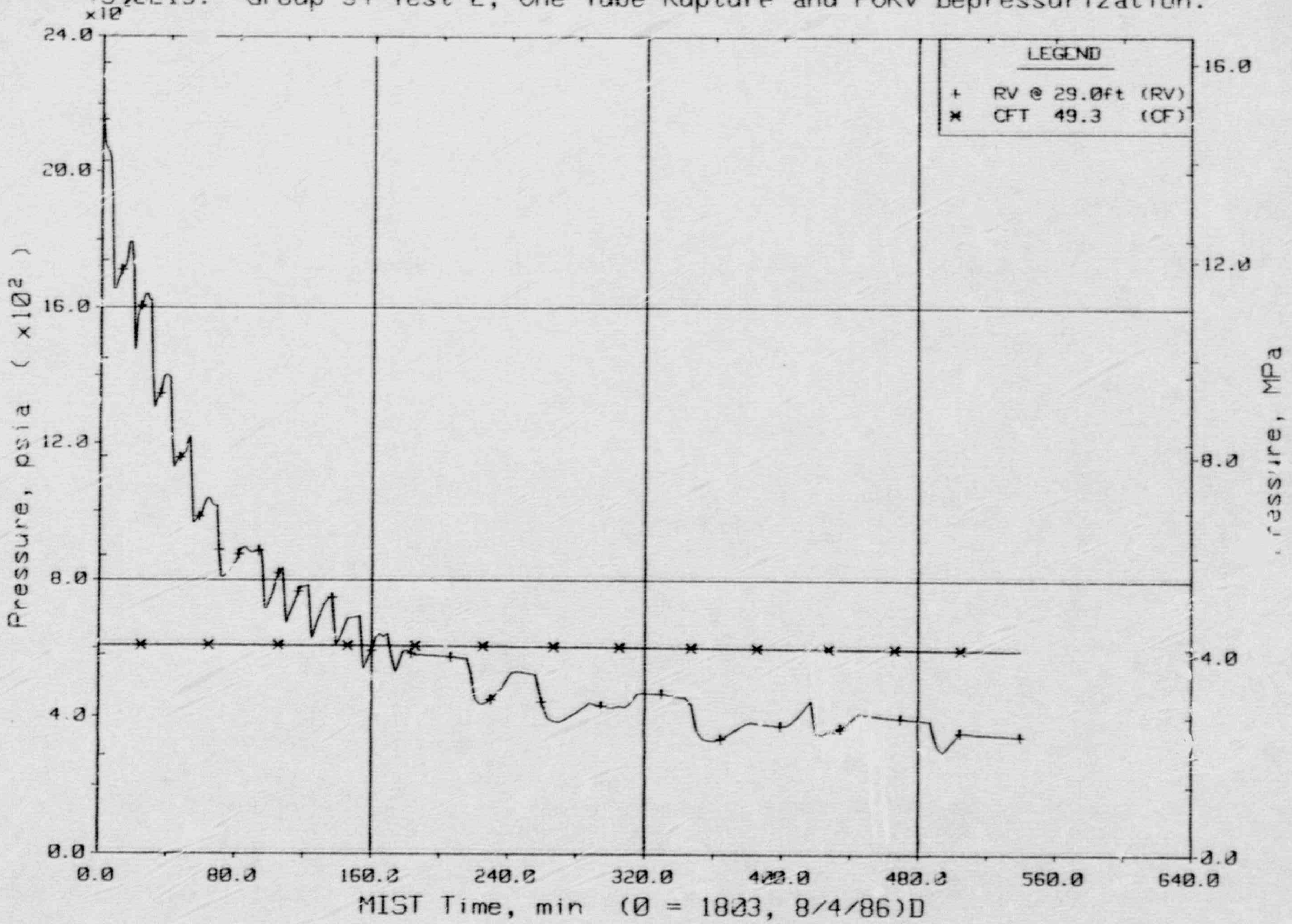
Loop B Cold Leg Fluid Temperatures (RTDs).

Tue May 31 14:38:03 1988

C2RT0

FINAL DATA

T342213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



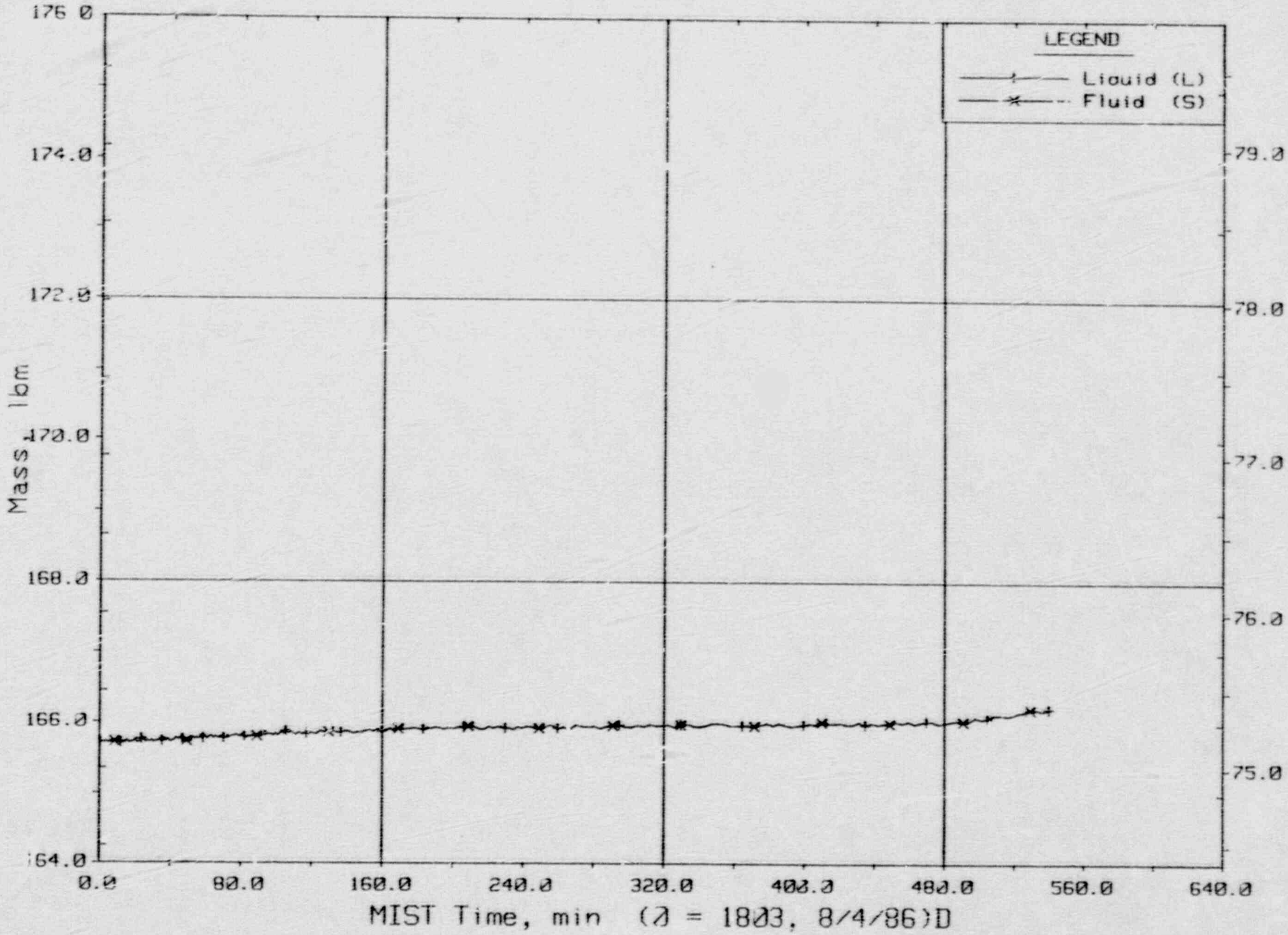
Primary System and Core Flood Tank Pressures (GP01s).

Tue May 31 14:40:49 1988

CFGP1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



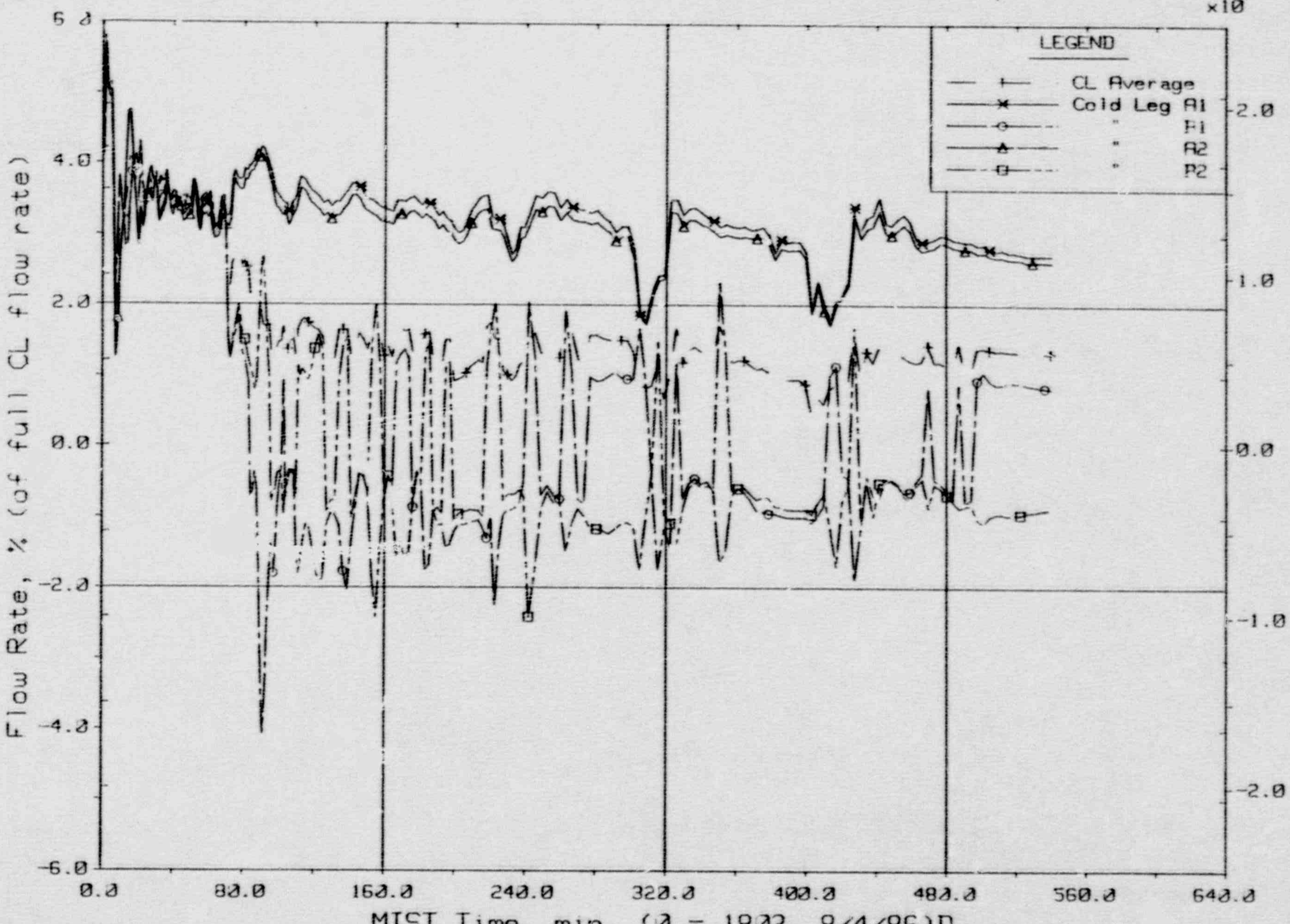
Core Flood Tank Liquid and Fluid Mass (CFMa20s).

Tue May 31 14:43:02 1988

CFML1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



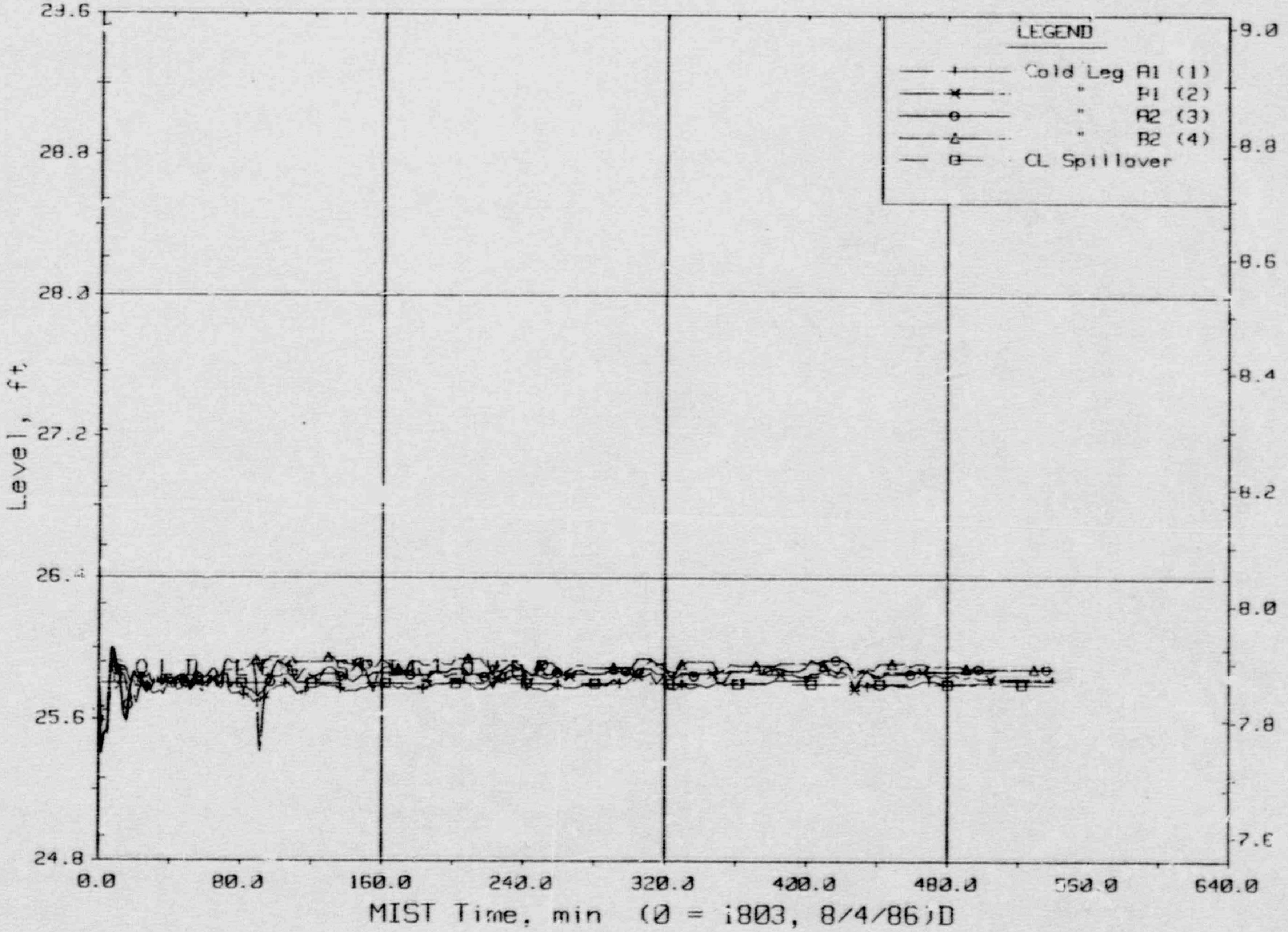
Cold Leg (Venturi) Flow Rates.

Tue May 31 14:46:37 1988

CLFL1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



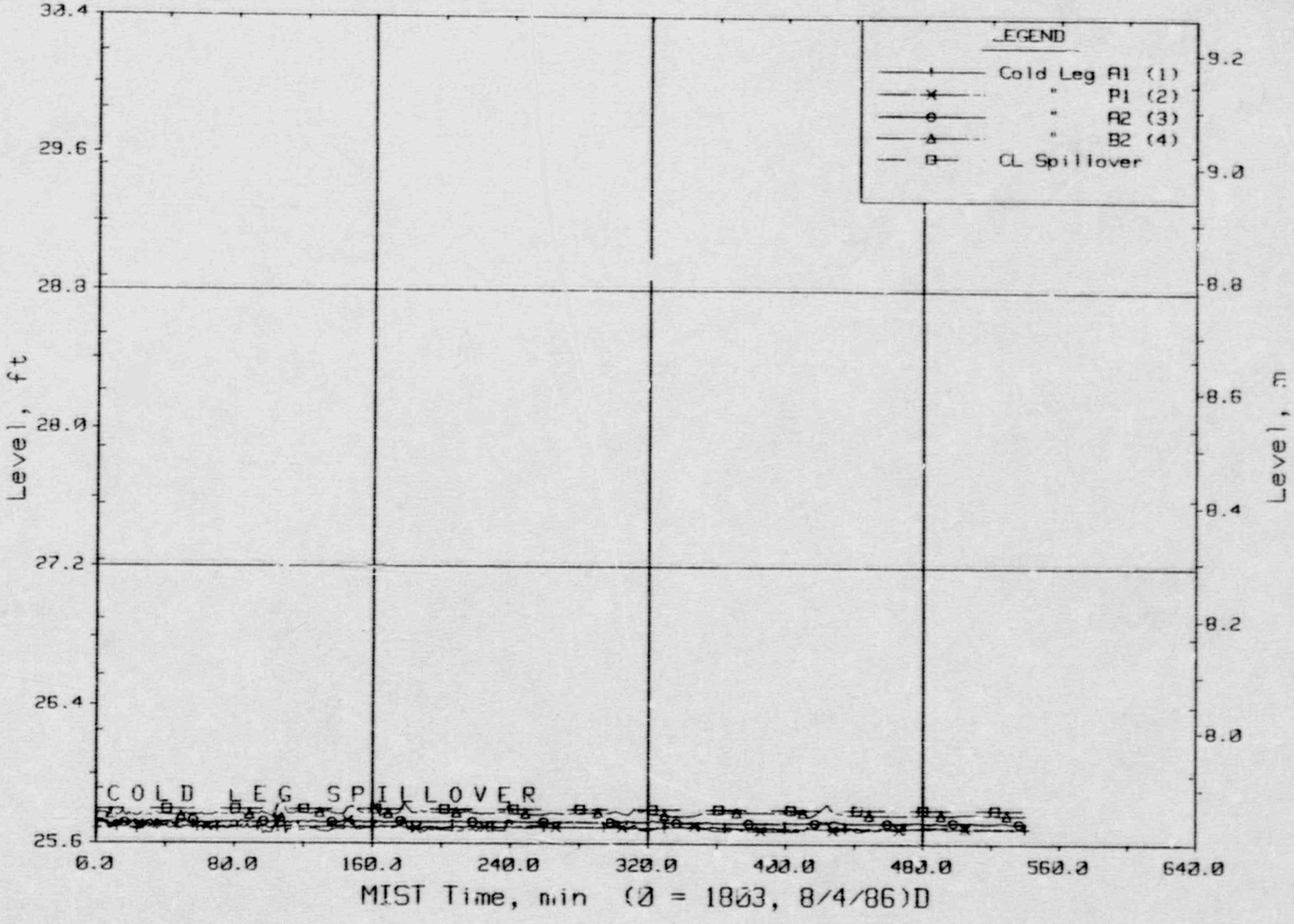
Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

Tue May 31 14:50:19 1988

CLLV4

FINAL DATA

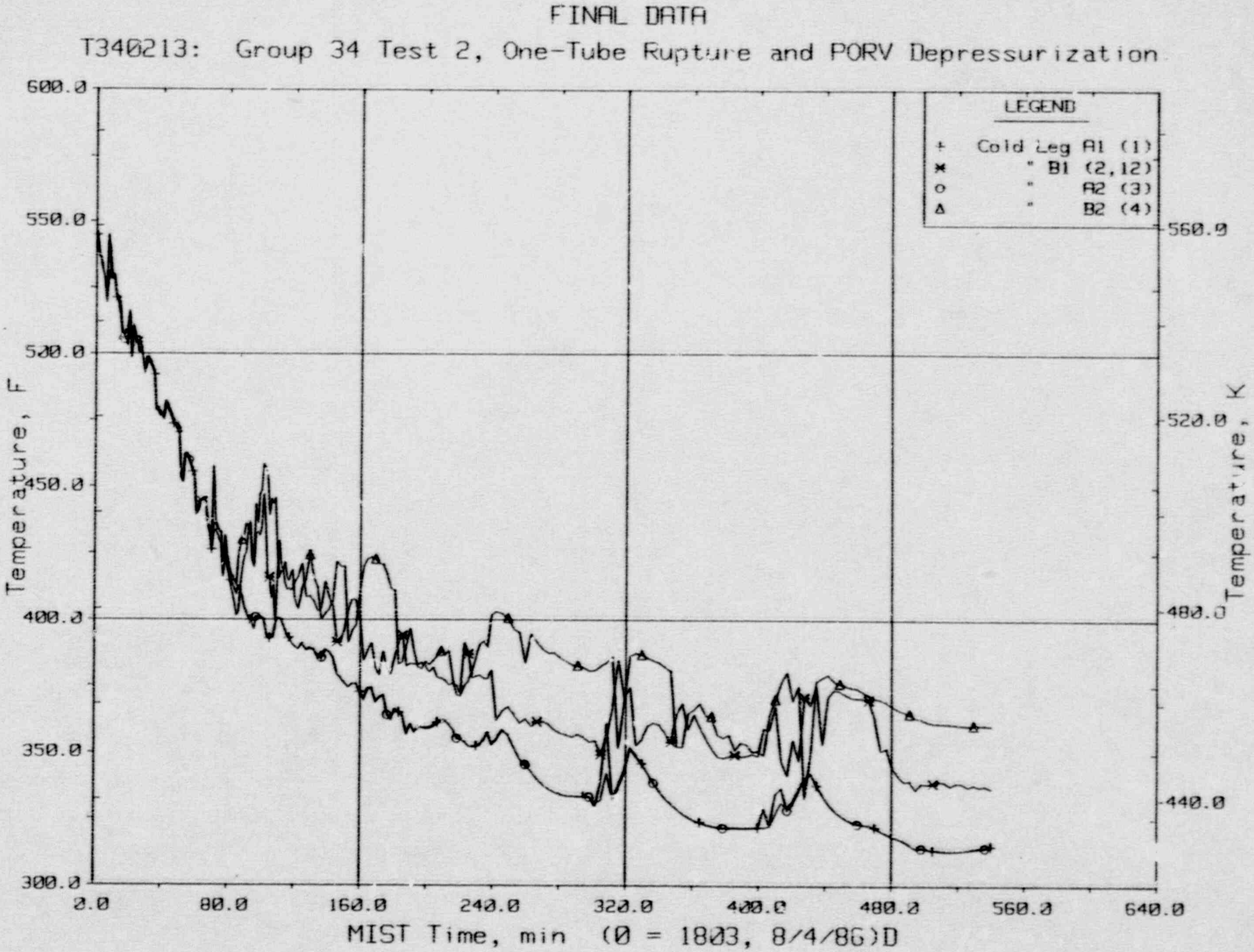
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Tue May 31 14:53:18 1988

CLLV5



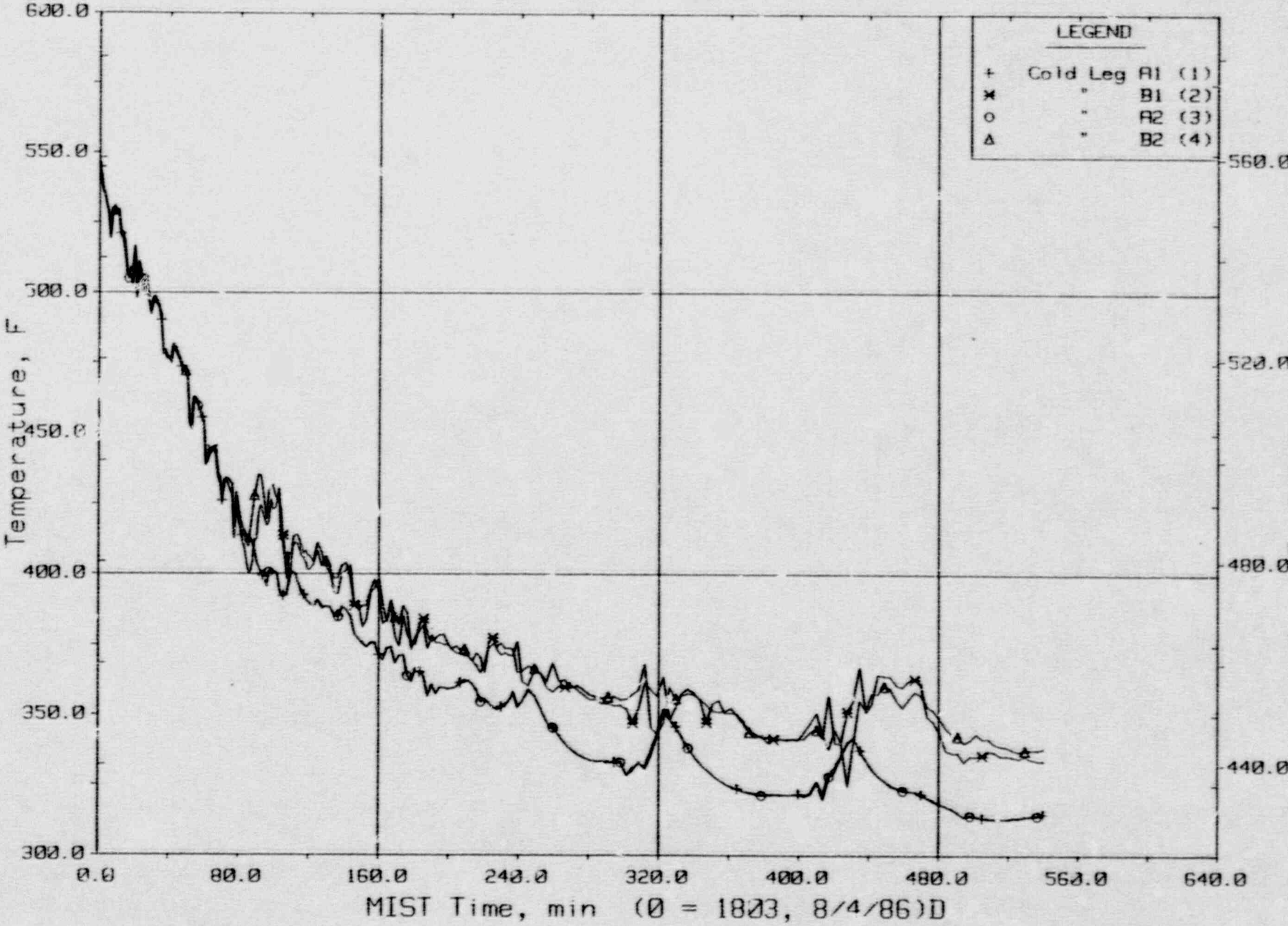
Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC1ls).

Tue May 31 14:59:33 1988

CLTC1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



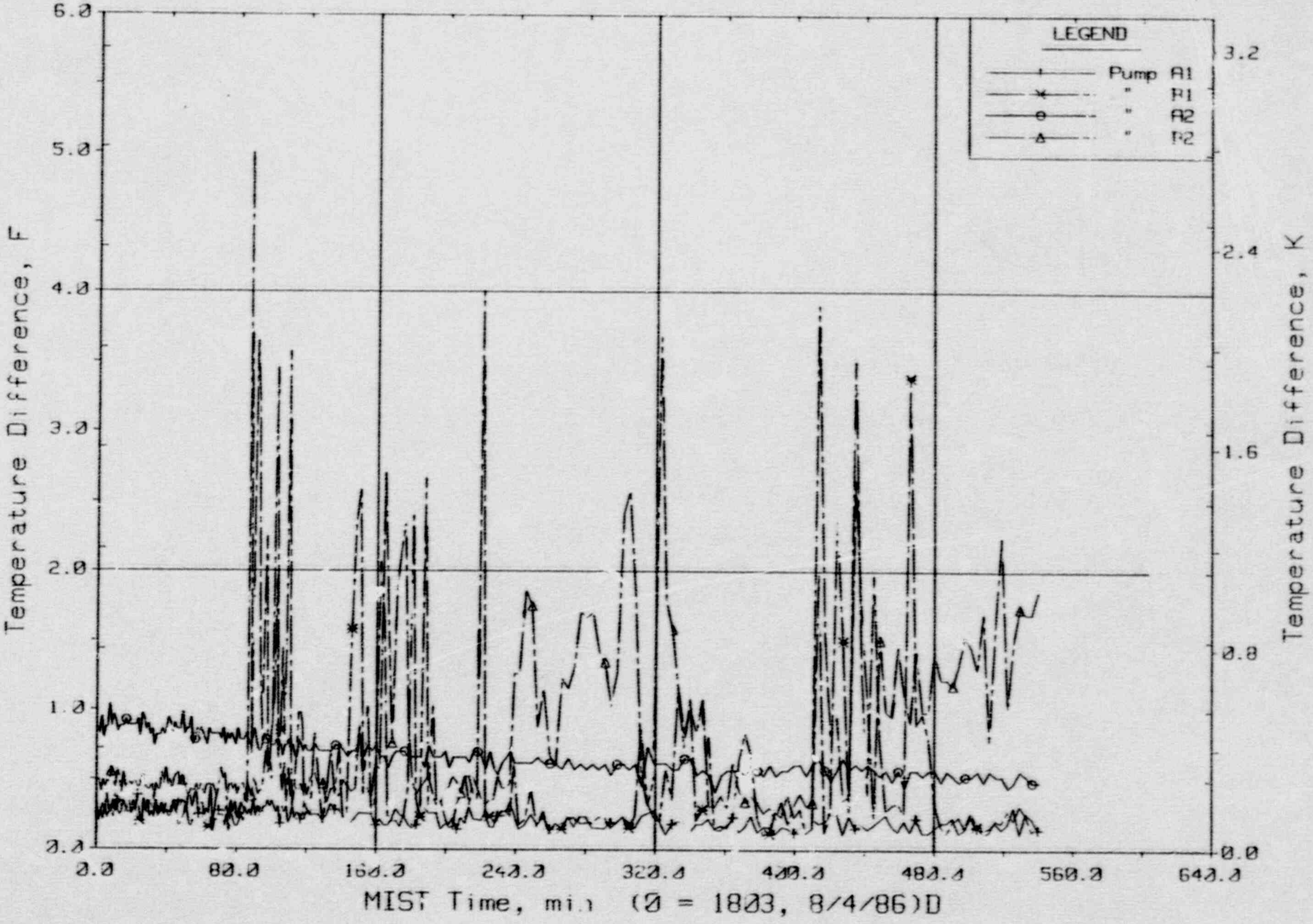
Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

Tue May 31 15:01:55 1988

CLTC2

FINAL DATA

1340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



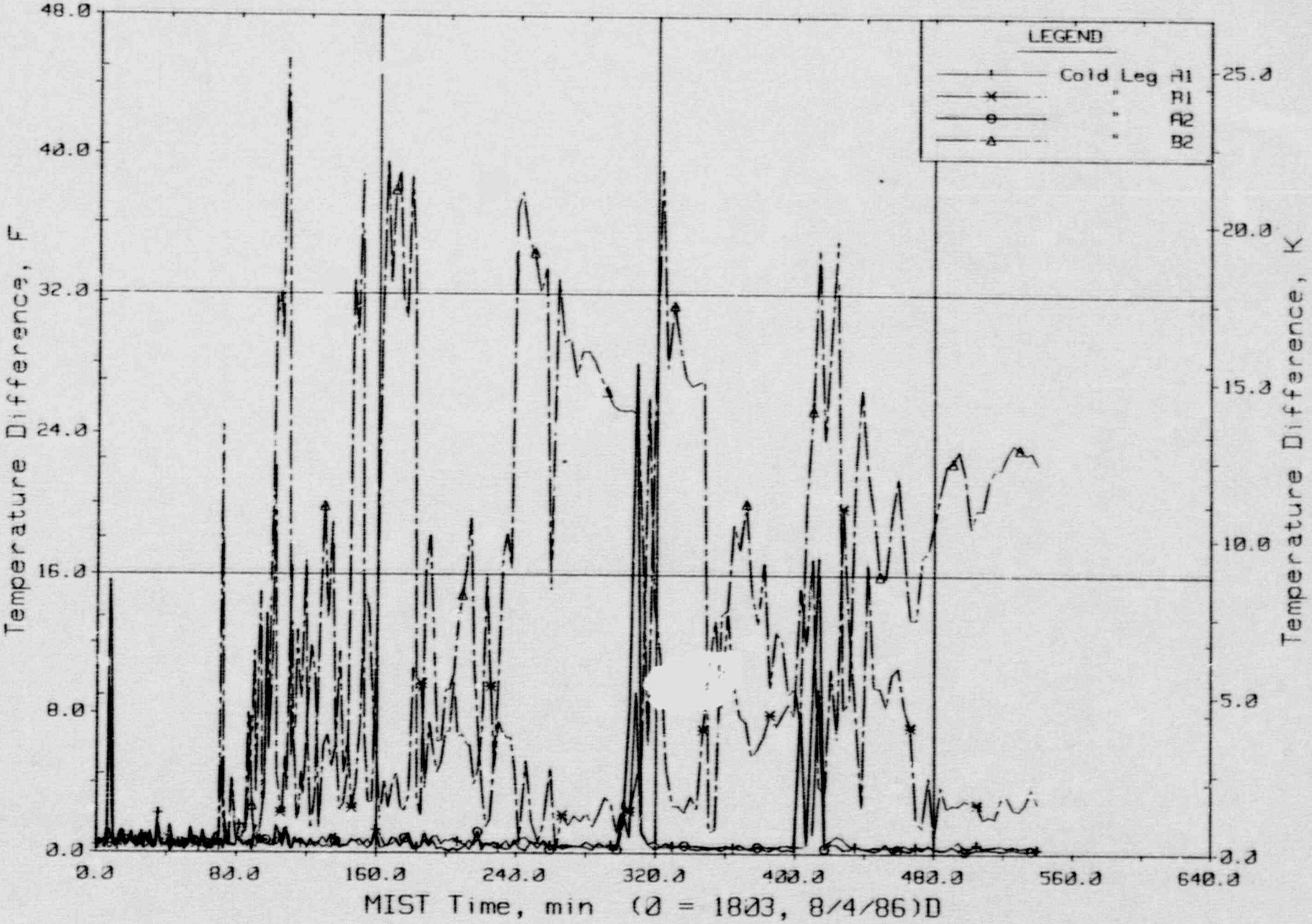
Maximum Differences Among RCP Rake Fluid Temperatures.

Tue May 31 15:05:30 1988

CLTD1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



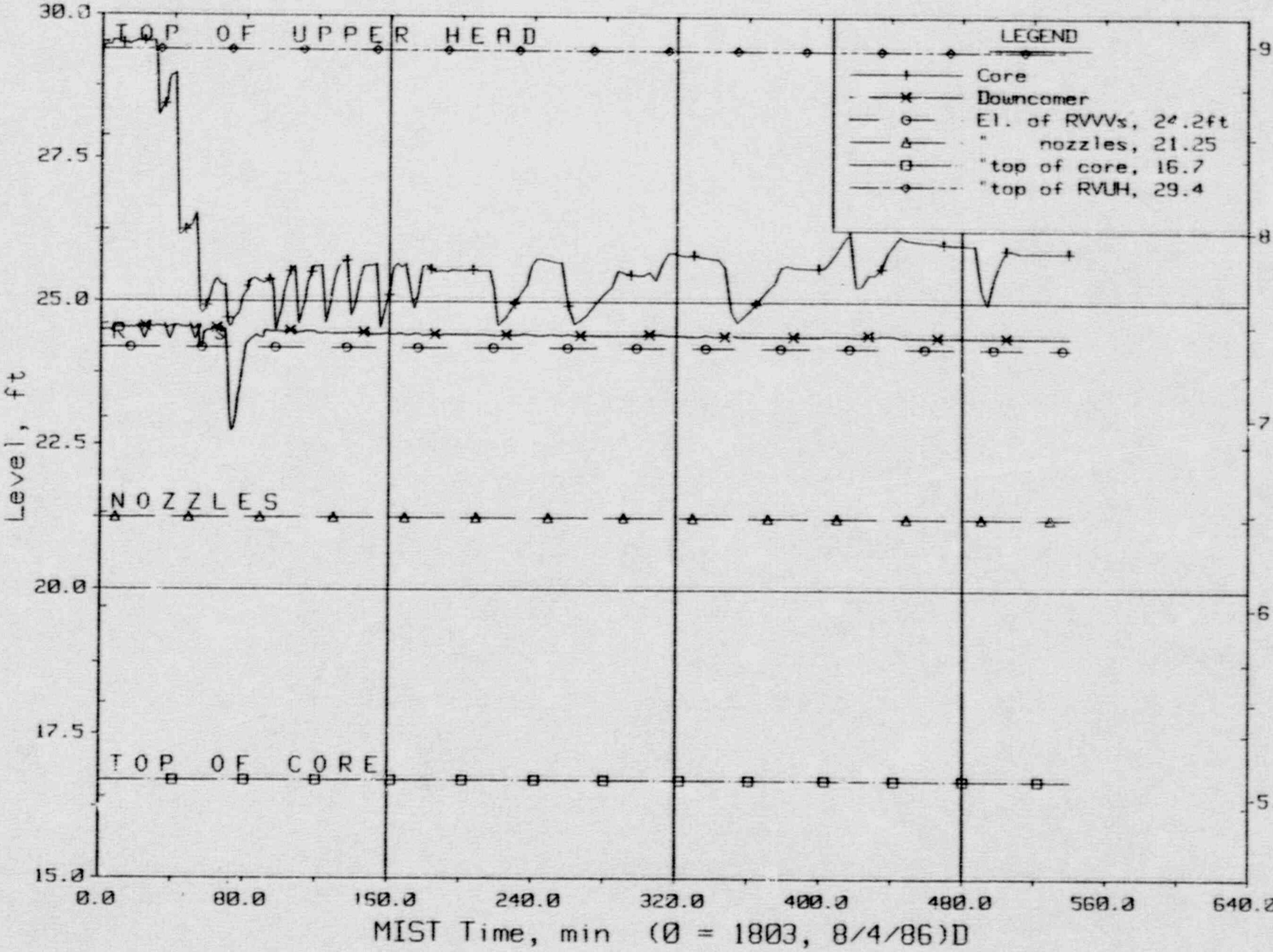
Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

Tue May 31 15:11:32 1988

CLTD2

FINAL DATA

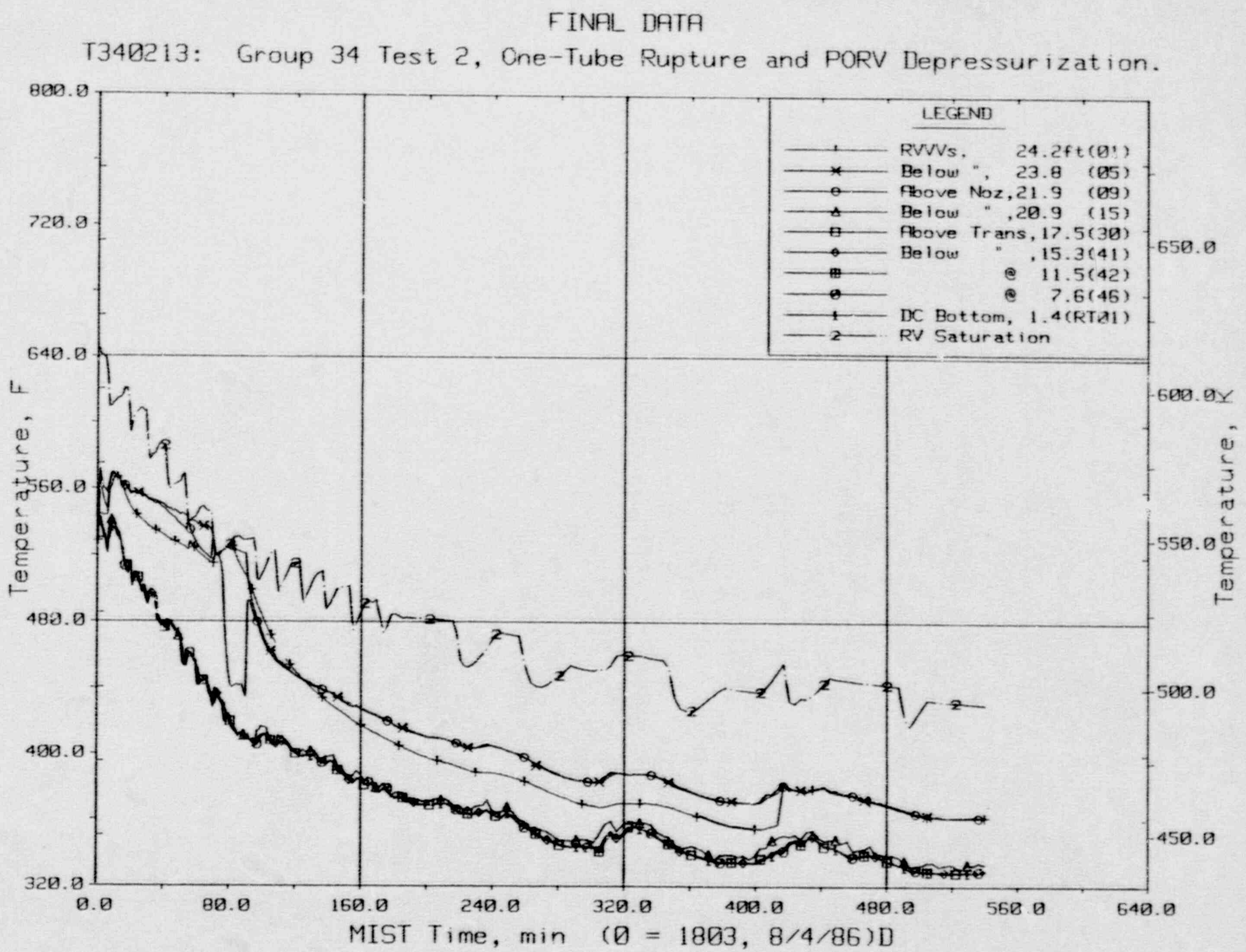
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Core Region Collapsed Liquid Levels.

Tue May 31 15:16:09 1988

COLV1



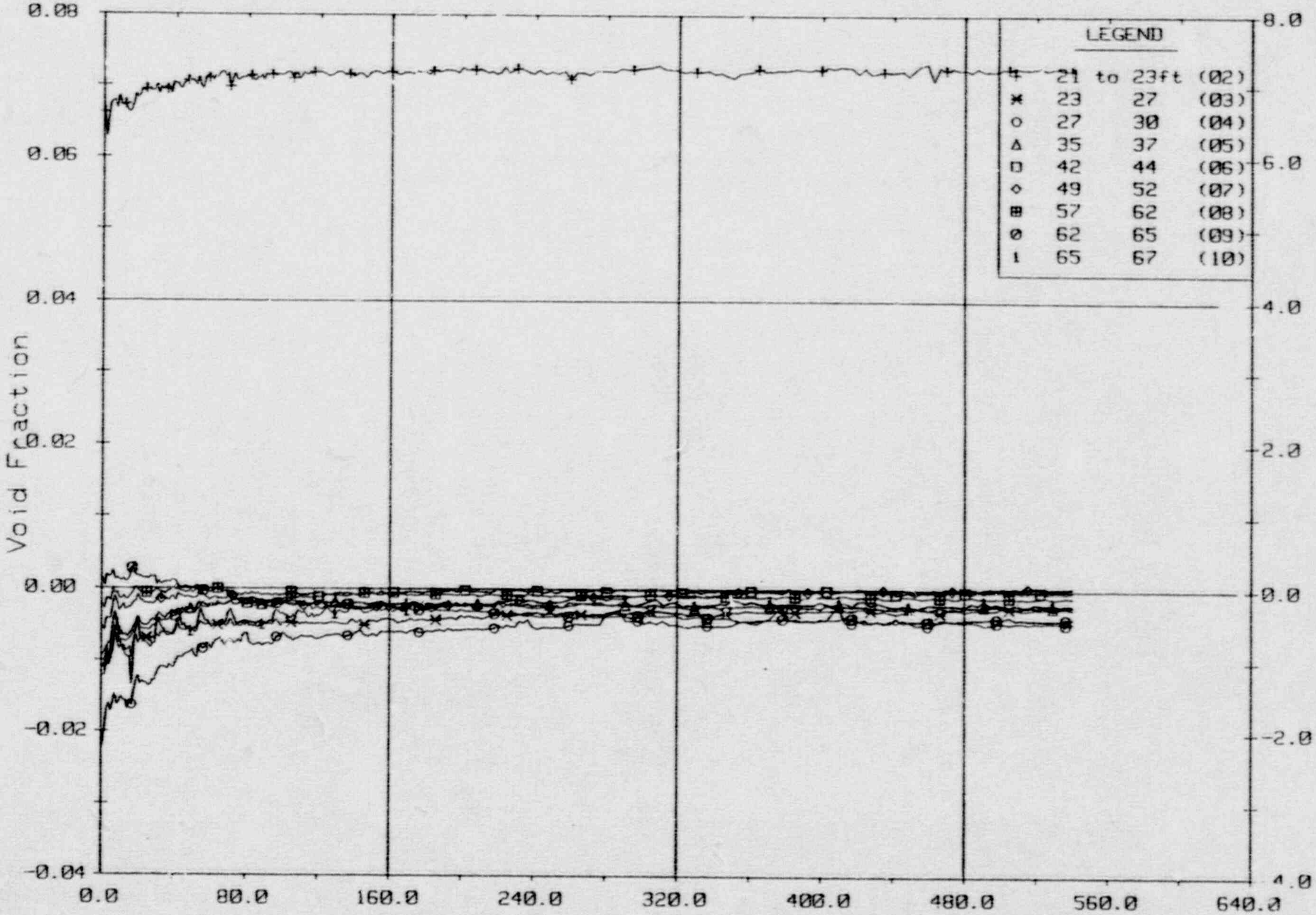
Downcomer Quadrant A1 Flüid Temperatures (DCTCs).

Tue May 31 15:30:43 1988

DCTC6

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



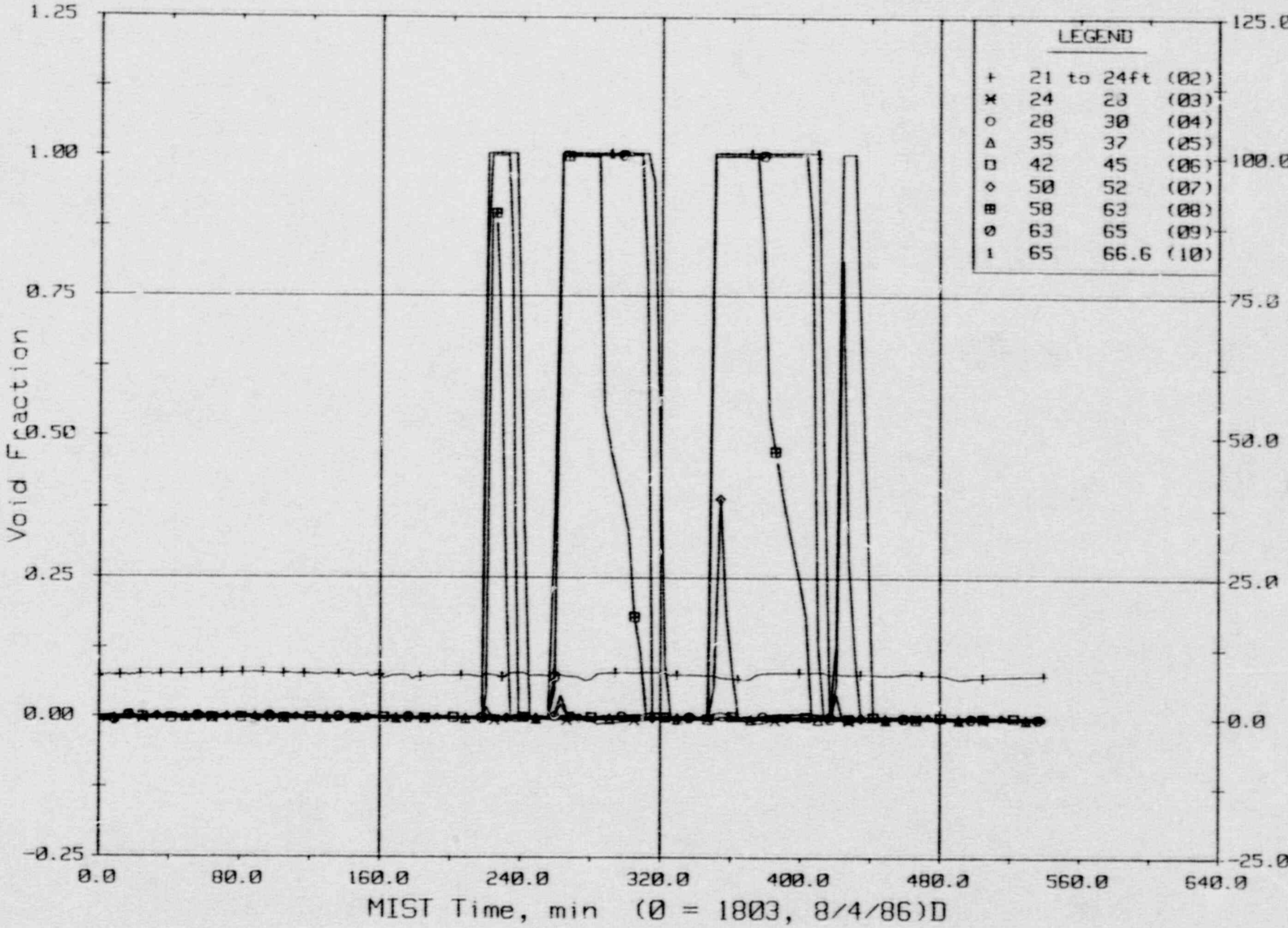
Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

Tue May 31 15:34:34 1988

H1VF1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Ruptu and PORV Depressurization.



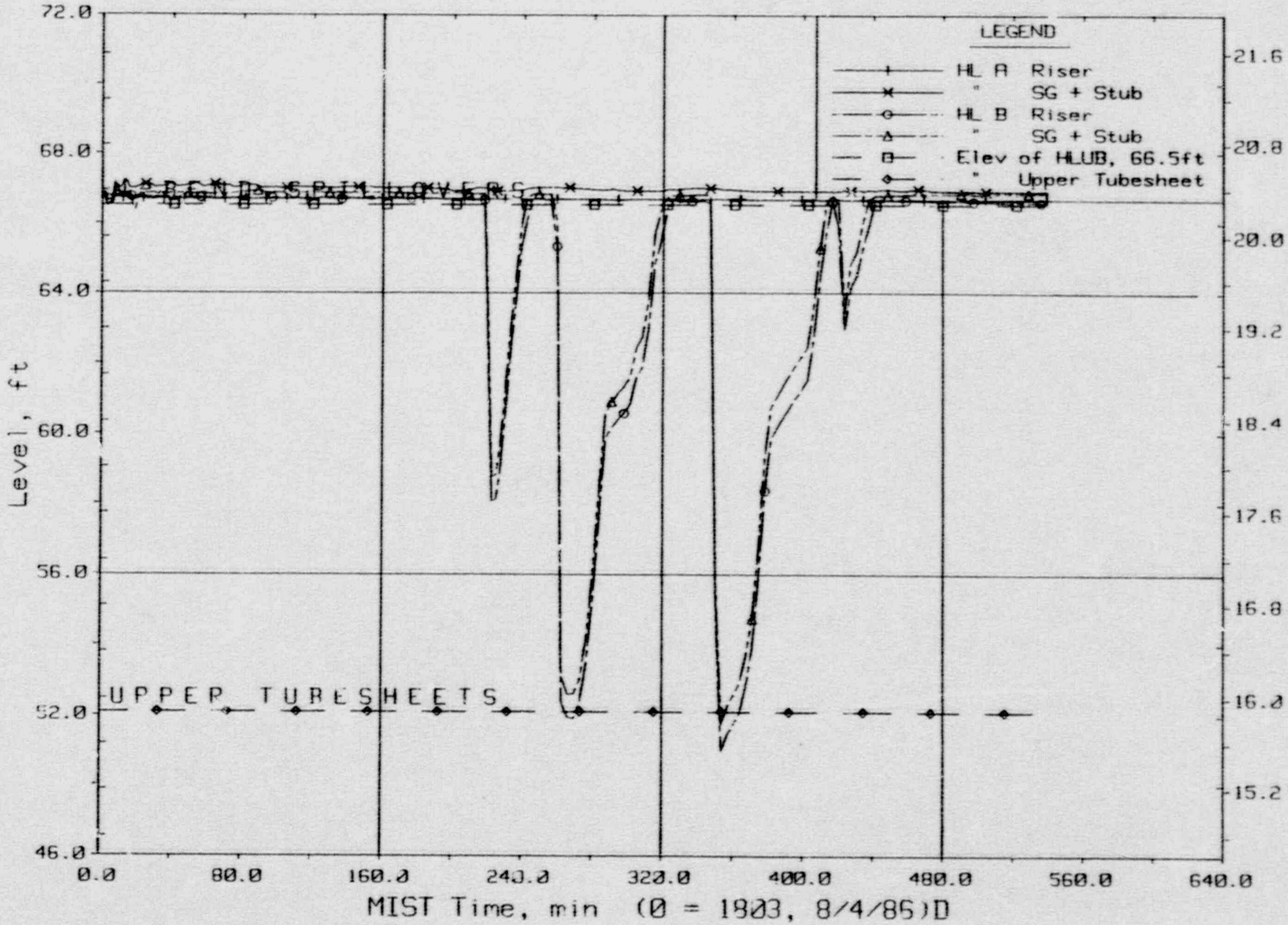
Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

Tue May 31 15:37:56 1988

H2VF1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



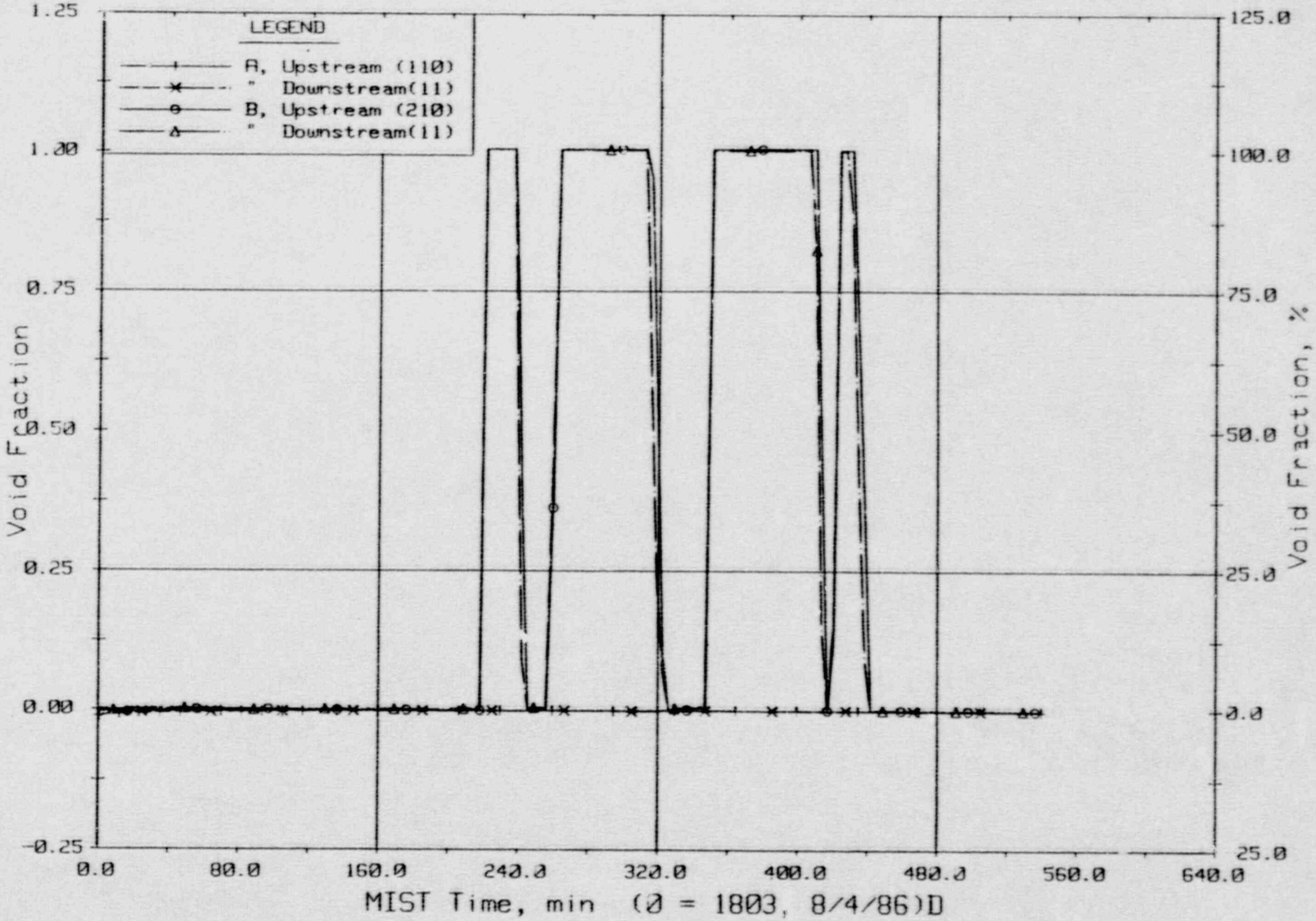
Hot Leg Riser and Stub Collapsed Liquid Levels.

Tue May 31 15:45:26 1988

HLLV1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



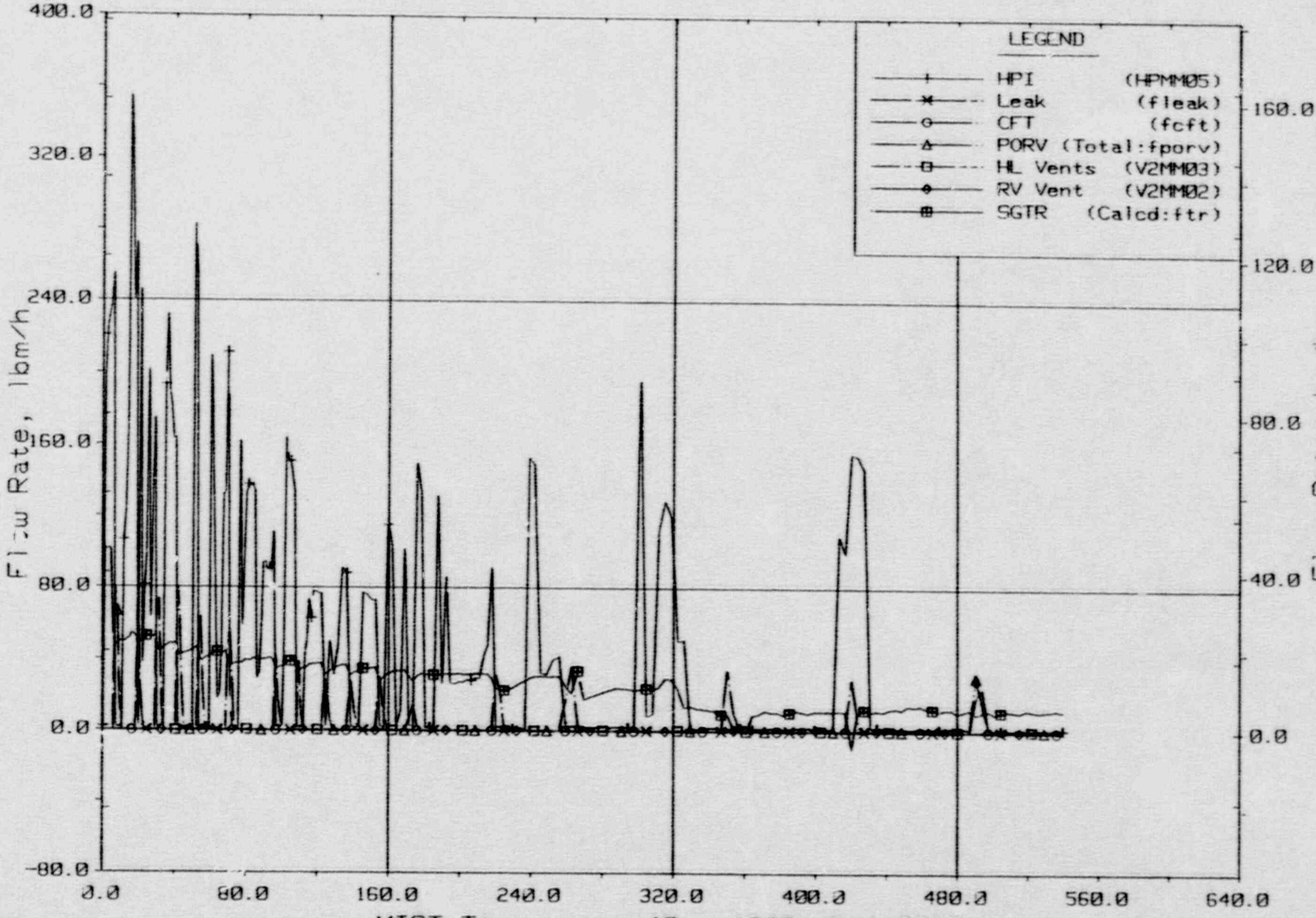
Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Tue May 31 15:50:37 1988

HLVF1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



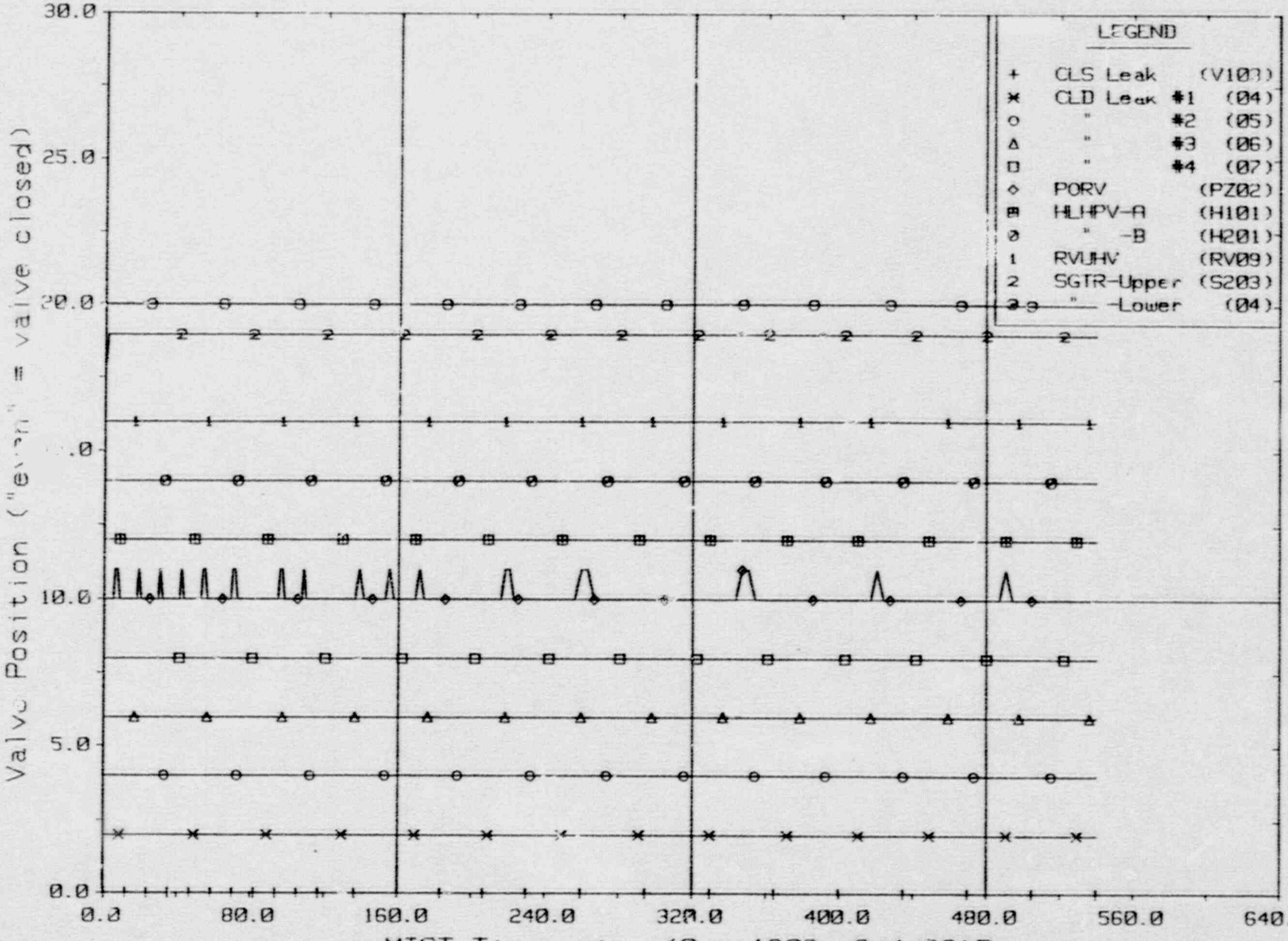
Primary System Boundary Flow Rates.

Tue May 31 15:55:56 1988

PBFL3

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



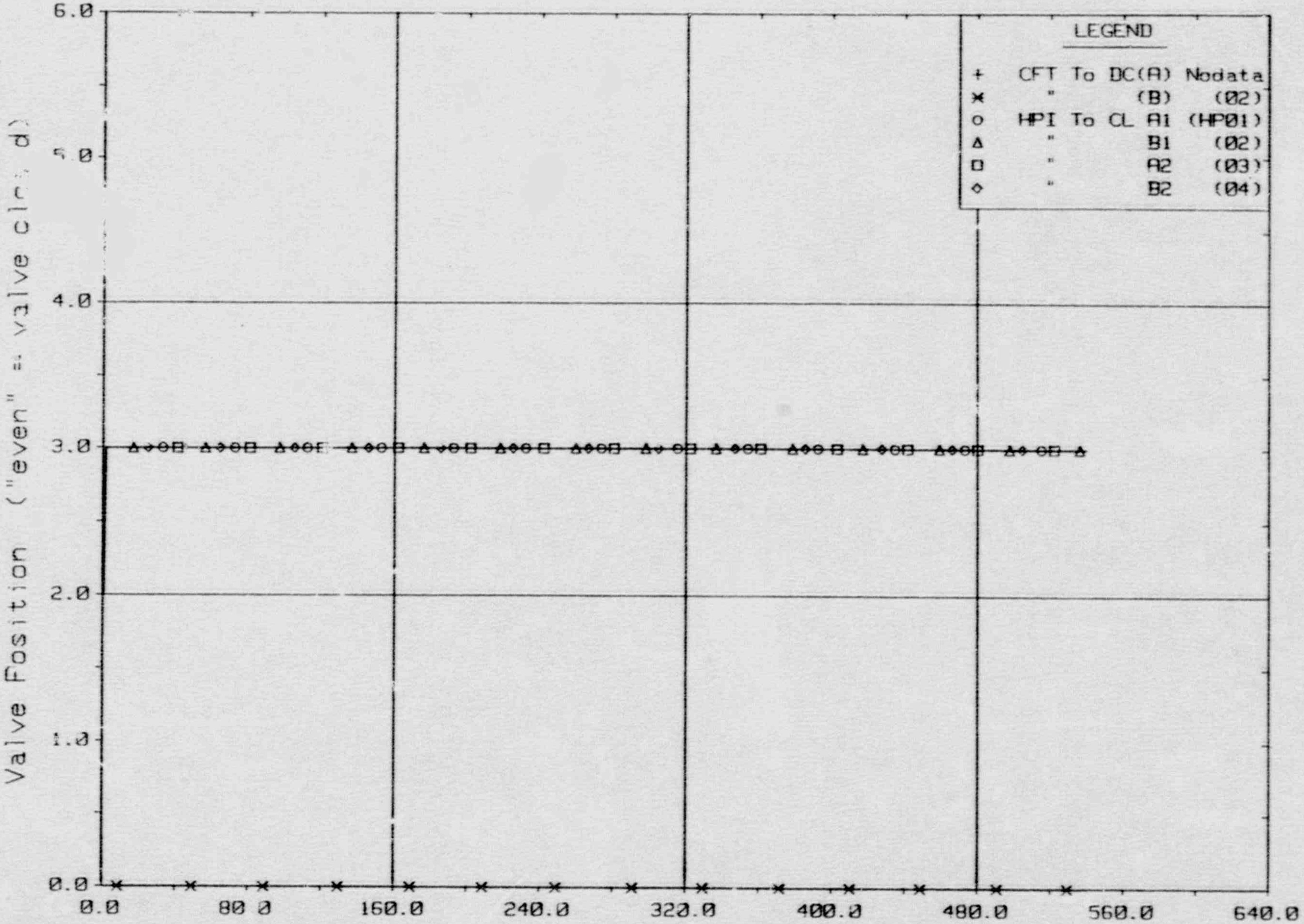
Primary System Discharge Limit Switch Indications

Tue May 31 16:00:04 1988

PBLS1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



MIST Time, min (0 = 1803, 8/4/86)D

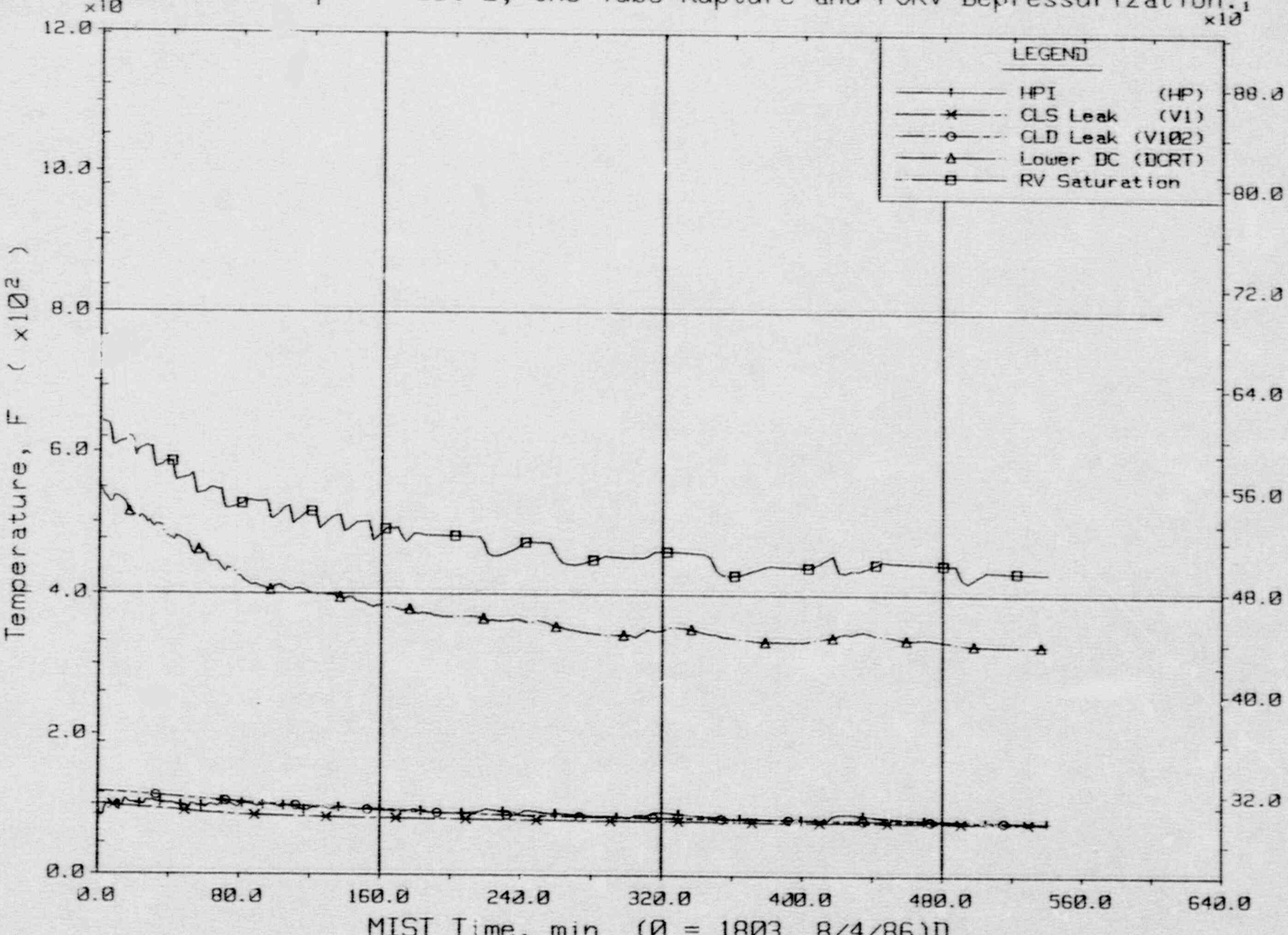
Primary System Injection Limit Switch Indications (LSs).

Tue May 31 16:03:10 1988

PBLS2

FINAL DATA

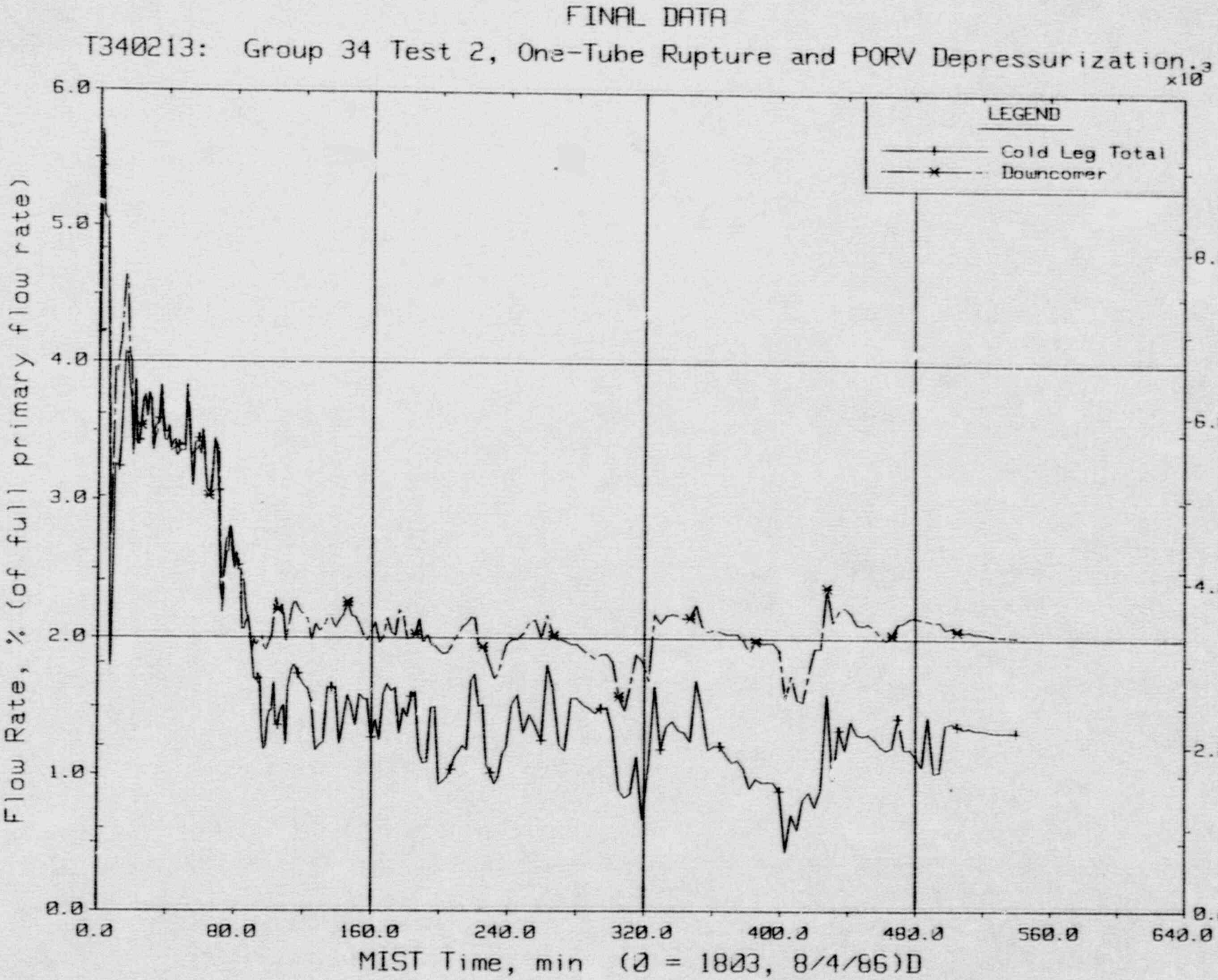
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Single-Phase Discharge and HPI Fluid Temperatures (TC01s).

Tue May 31 16:06:52 1988

PRTC1p



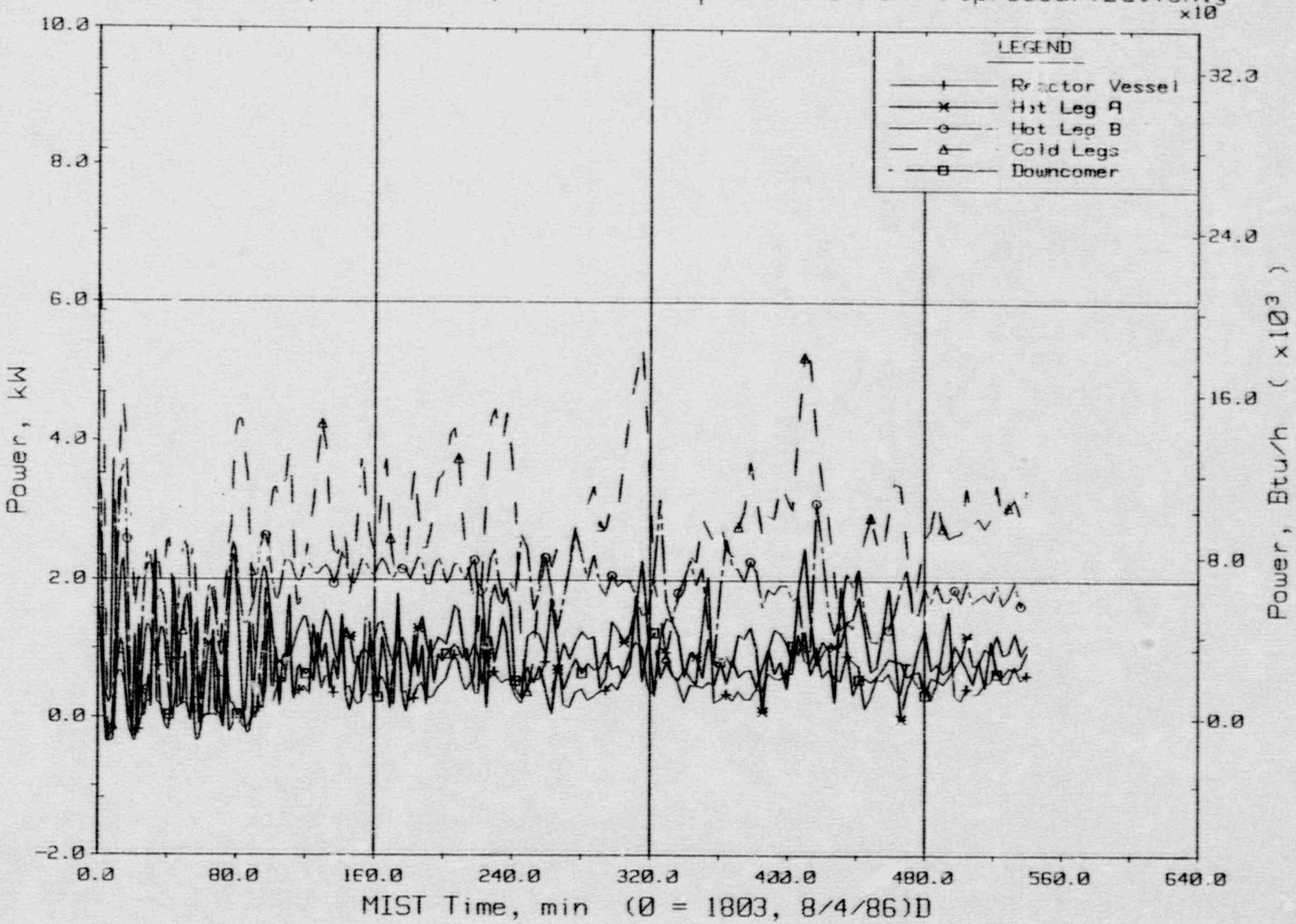
Primary System Venturi Flow Rates.

Tue May 31 16:09:09 1988

PRFL7

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization



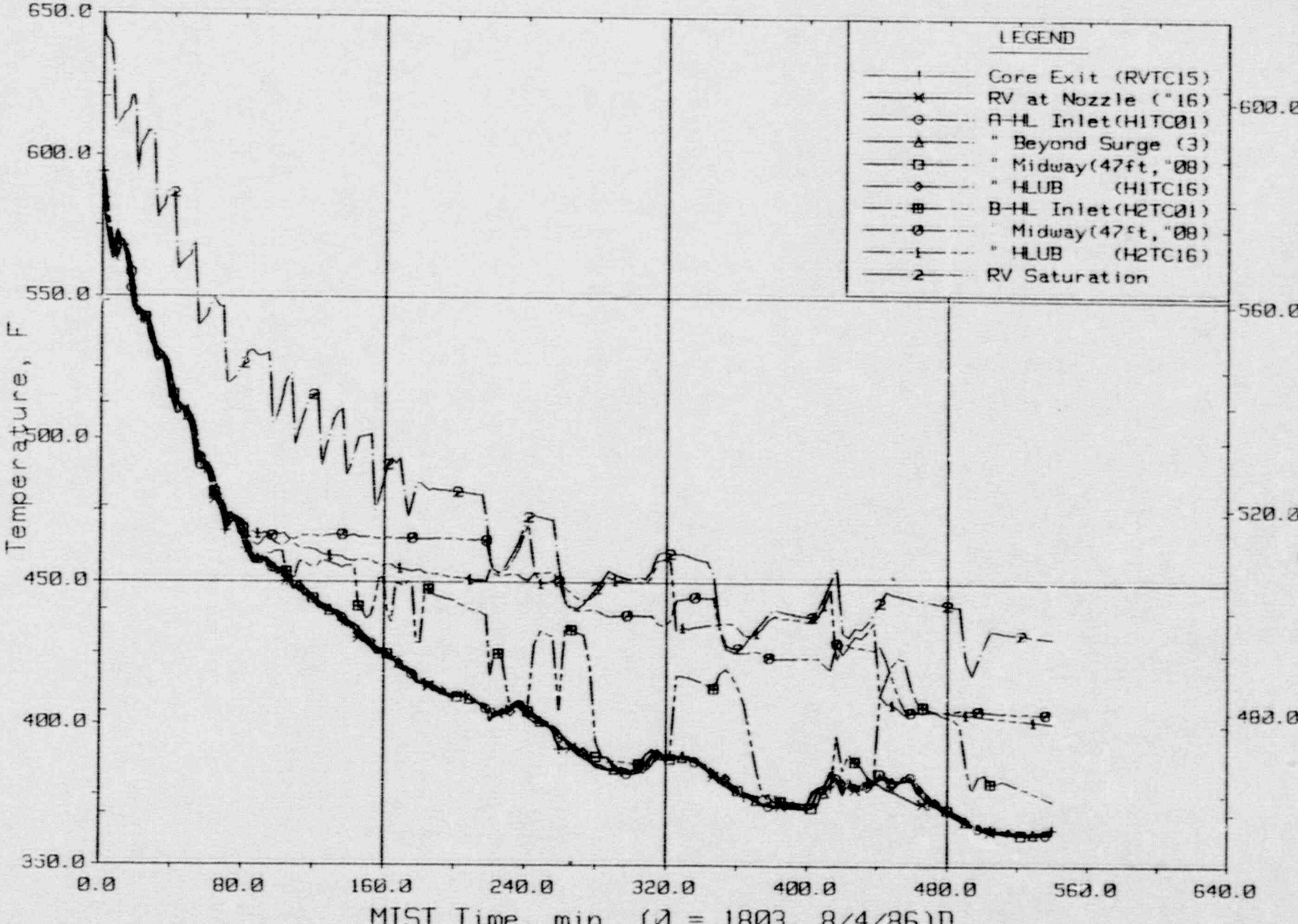
Guard Heater Specified Power Per Primary Component.

Tue May 31 16:12:07 1988

PRGQ1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



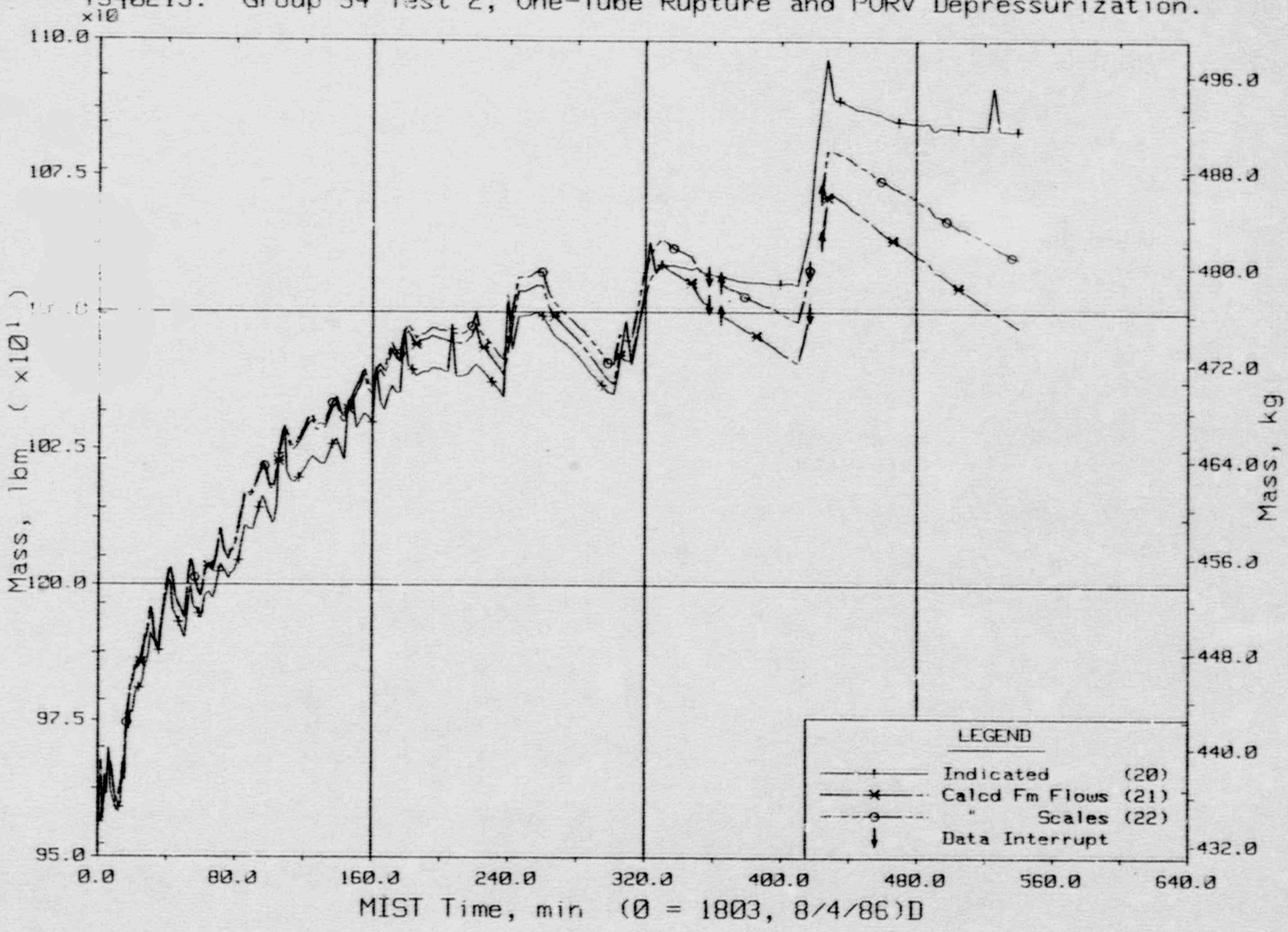
Composite Core Exit and Hot Leg Fluid Temperatures.

Tue May 31 16:16:09 1988

PRKT1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



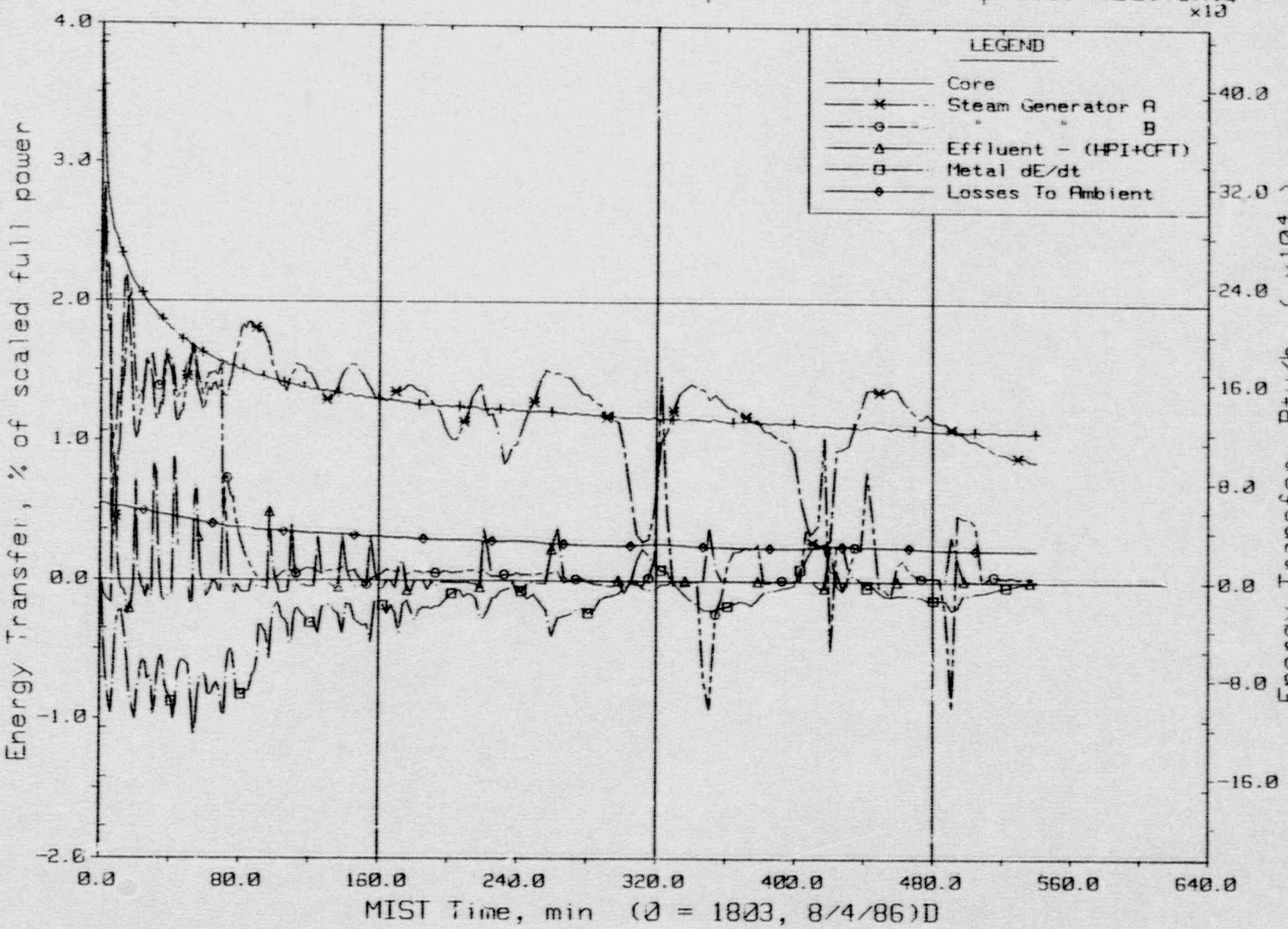
Primary System Total" Fluid Mass (PLMLs).

Tue May 31 16:19:07 1988

PRML2

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization



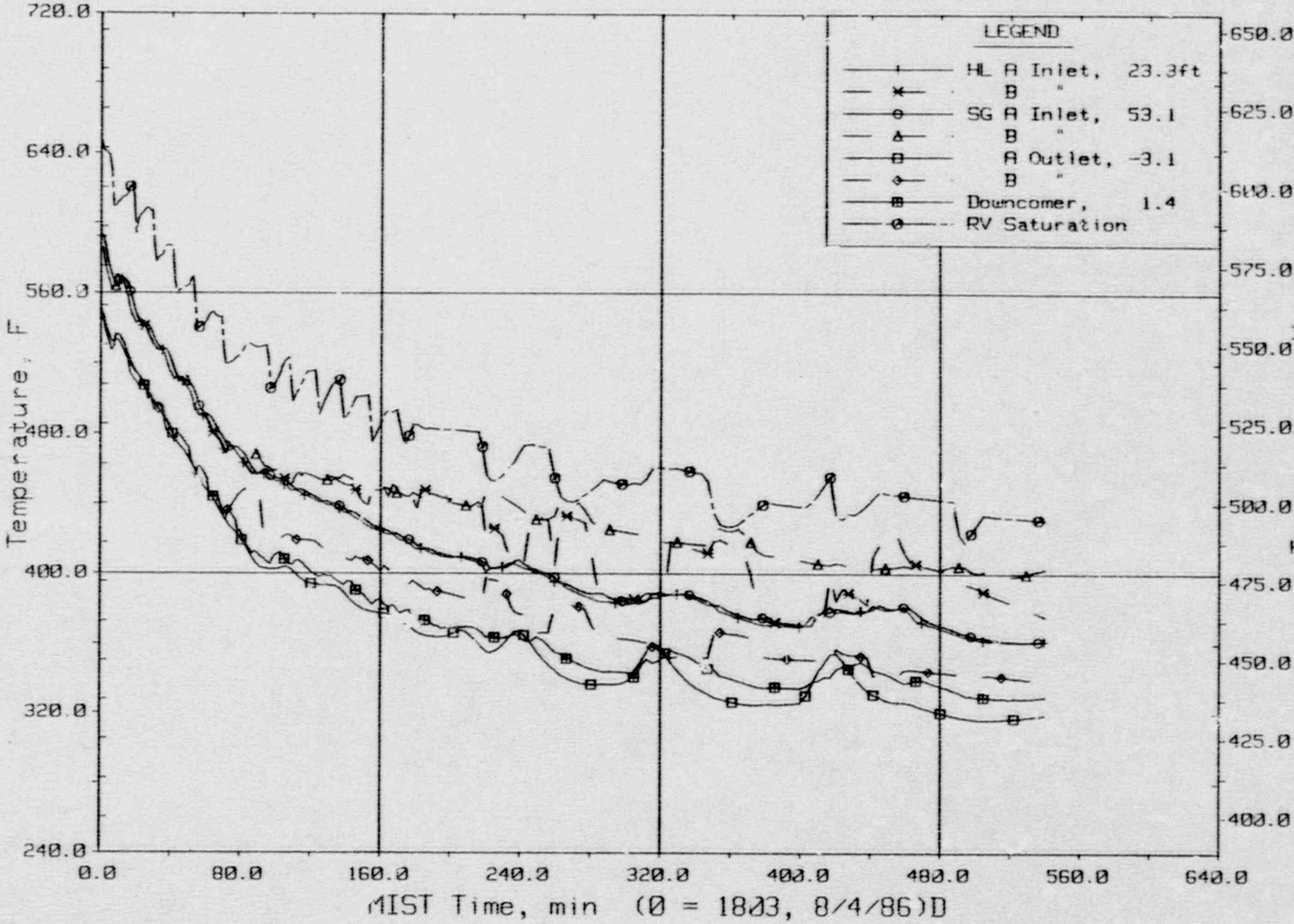
Primary System Energy Transfer.

Tue May 31 16:31:34 1988

PROE2

FINAL DATA

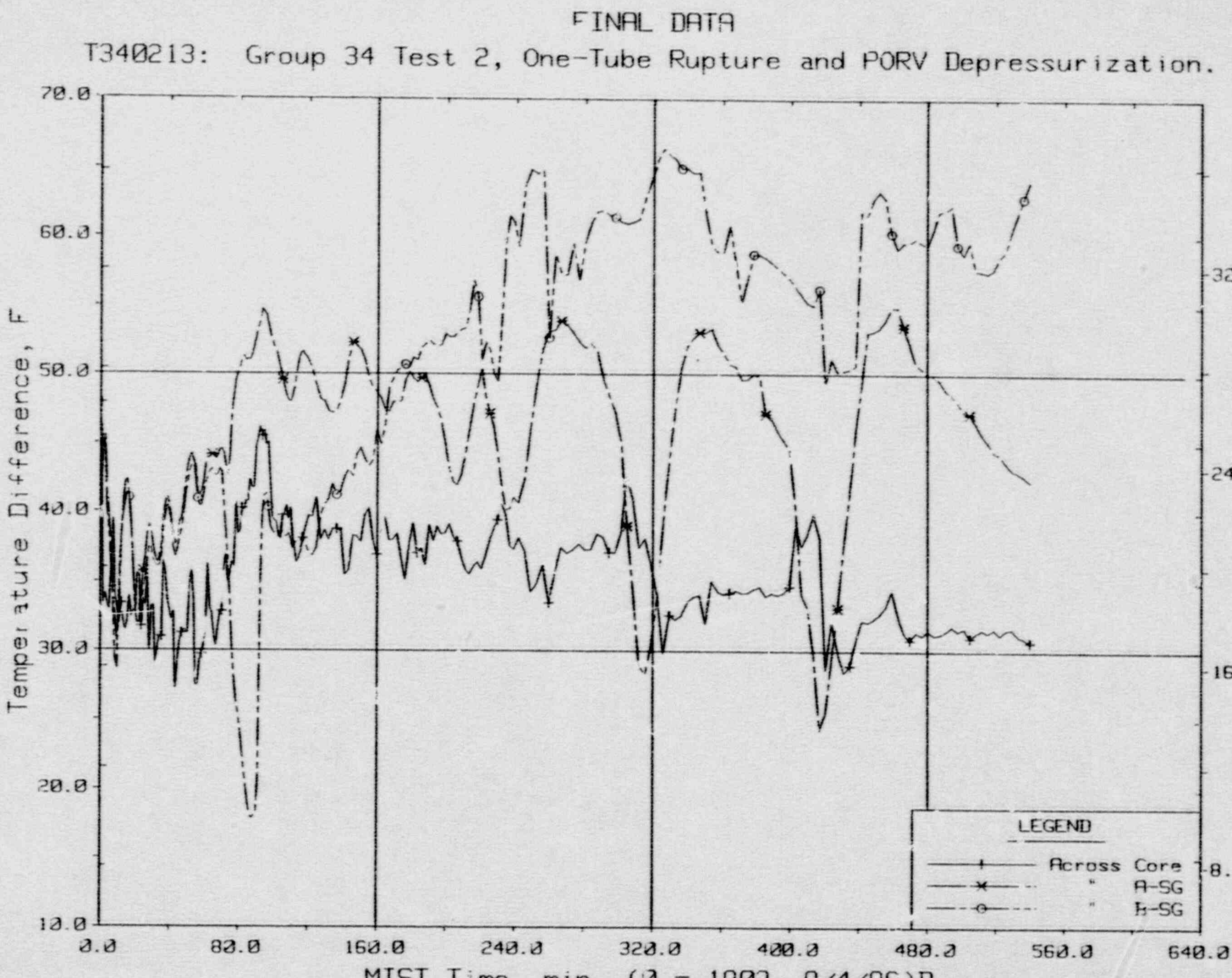
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Primary System Fluid"Temperatures (RTDs).

Tue May 31 16:34:52 1988

PRRT0



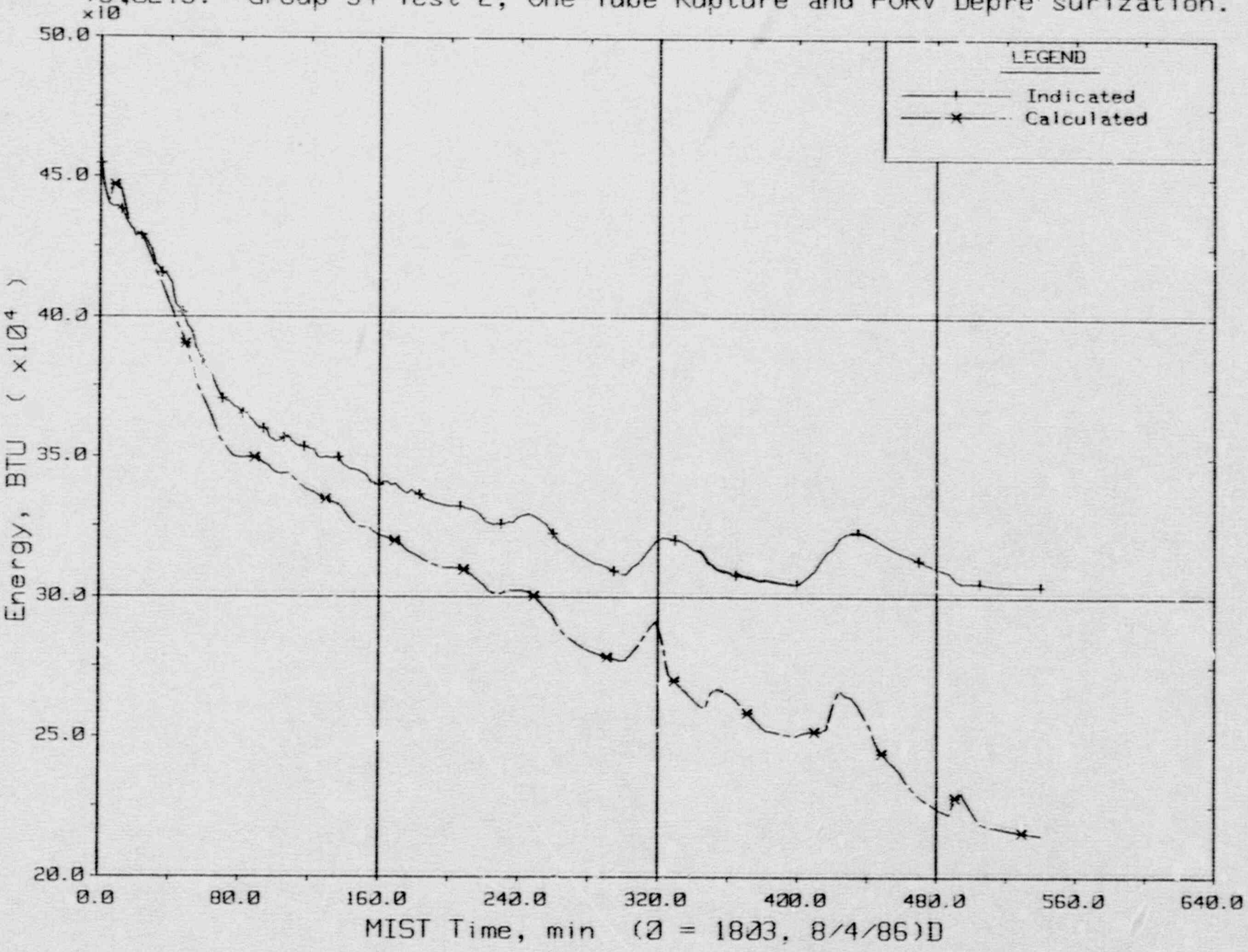
Key Temperature Differences.

Tue May 31 16:37:37 1988

PRTD2

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depre surization.



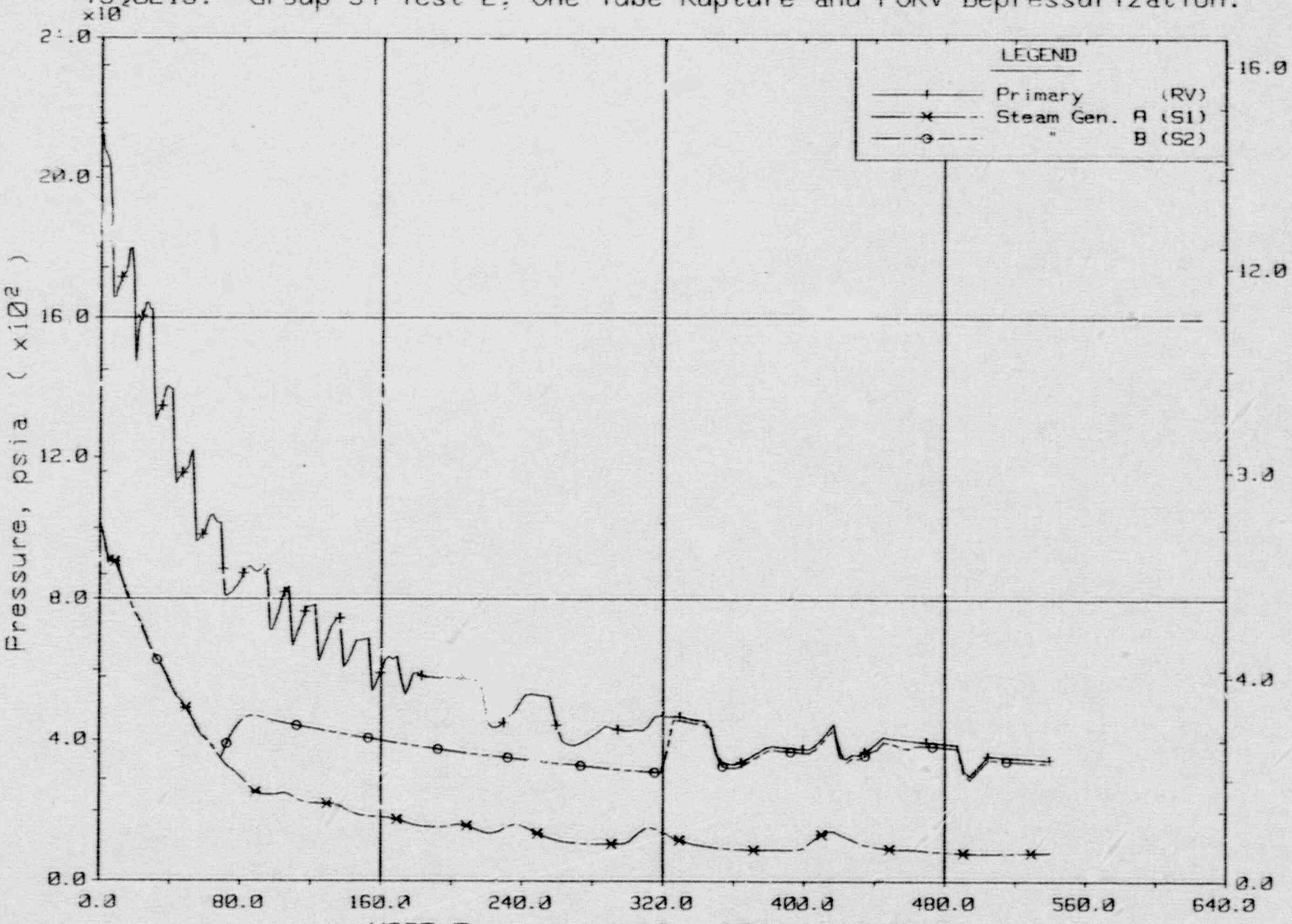
Primary System Total Fluid Energy.

Tue May 31 16:39:46 1988

PRTE2

FINAL DATA

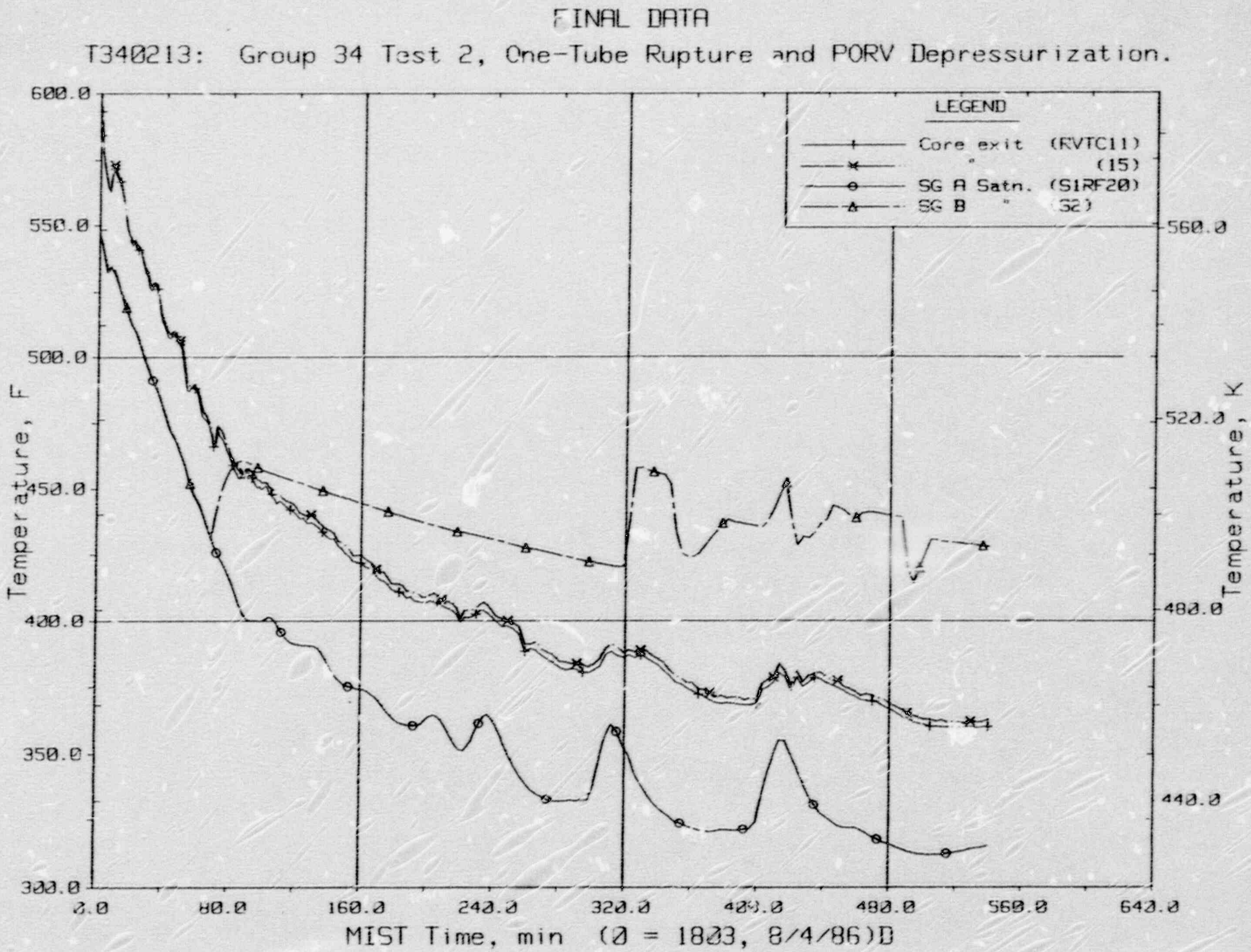
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Primary and Secondary System Pressures (GP01s).

Tue May 31 16:41:58 1988

PSGPi



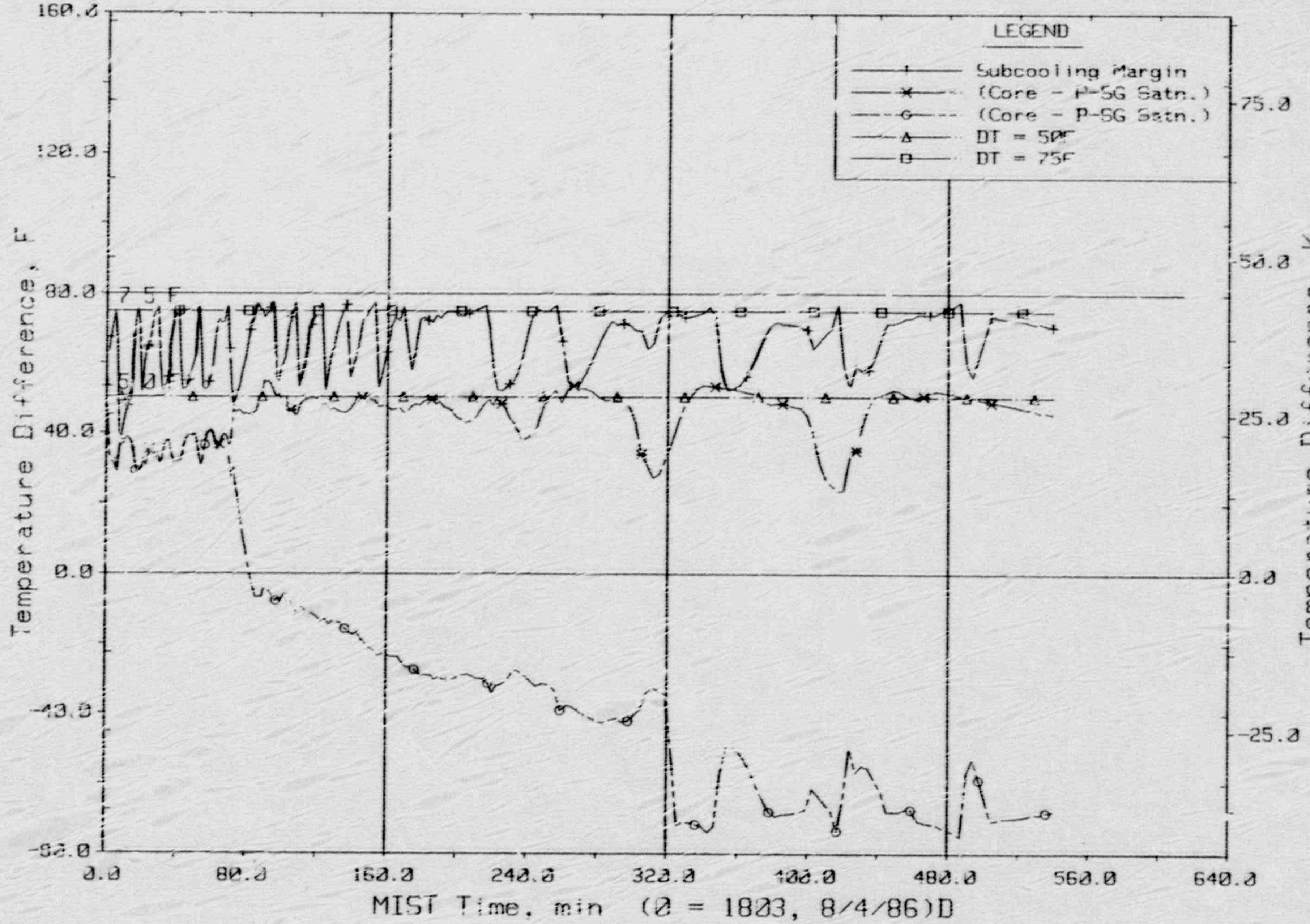
Core Exit and Steam Generator Secondary Saturation Temperatures.

Tue May 31 16:44:27 1988

PSTC3p

FINAL DATA

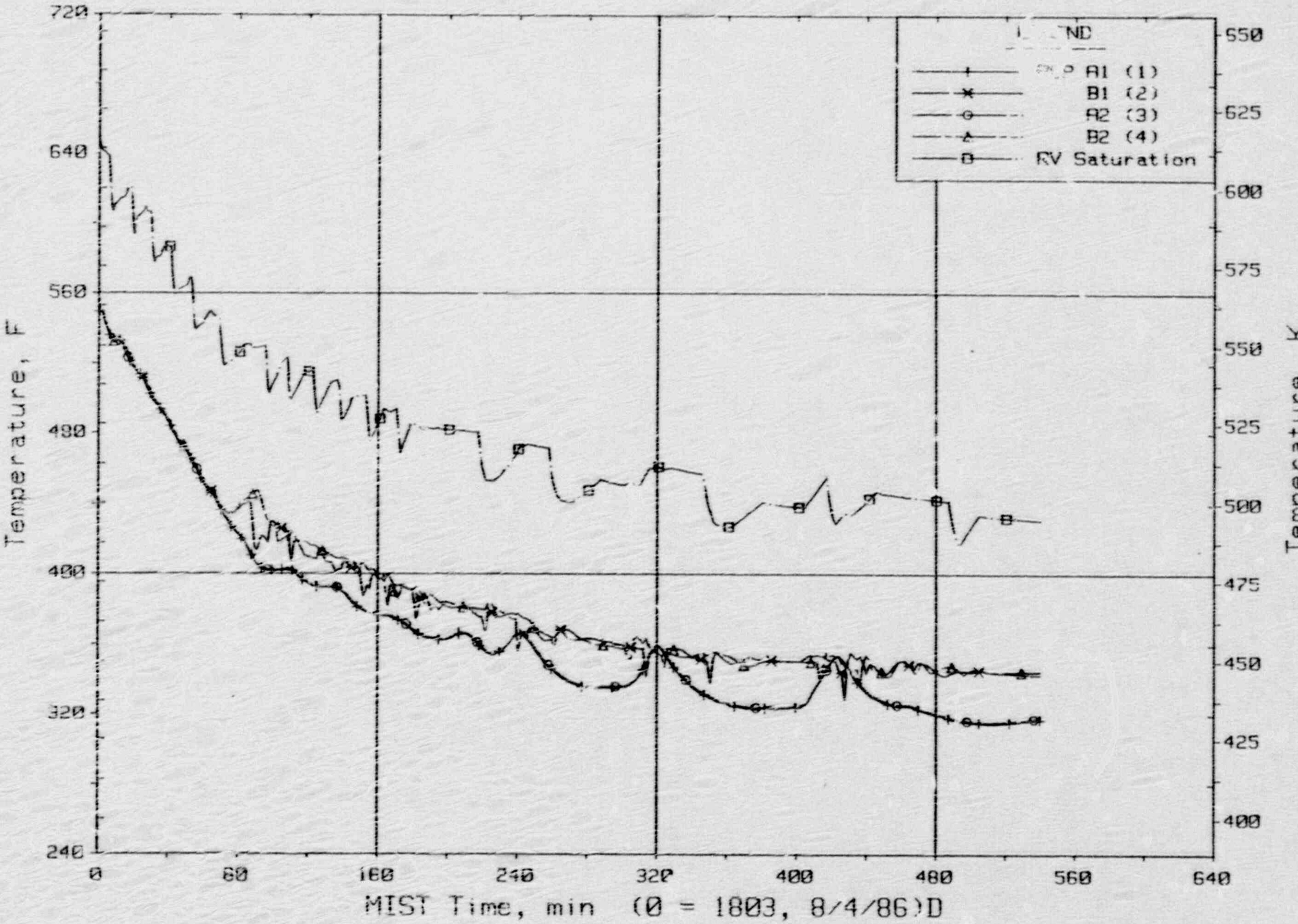
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Control Temperature Differences.

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



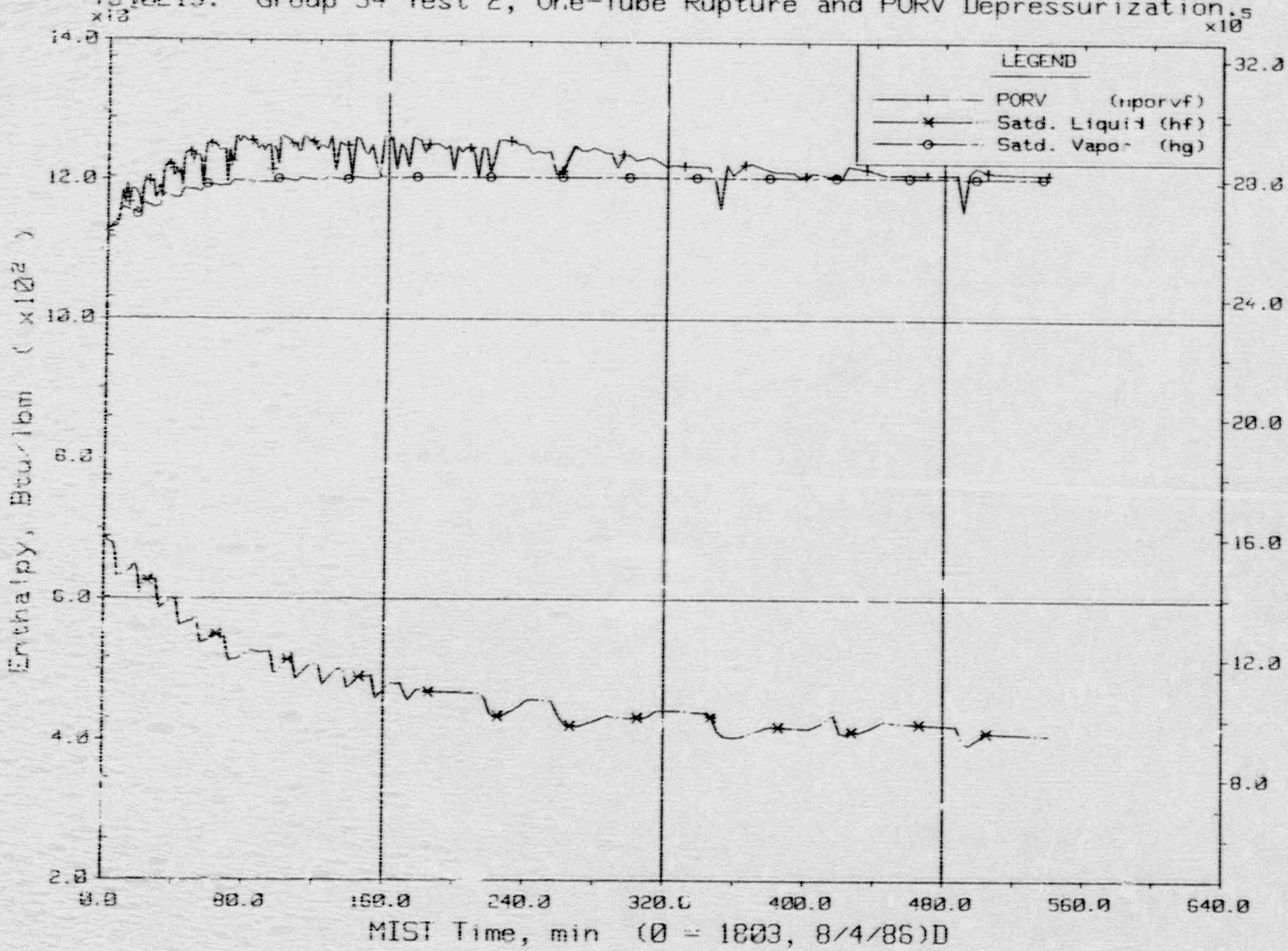
Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 17:22:21 1988

PURT0

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization



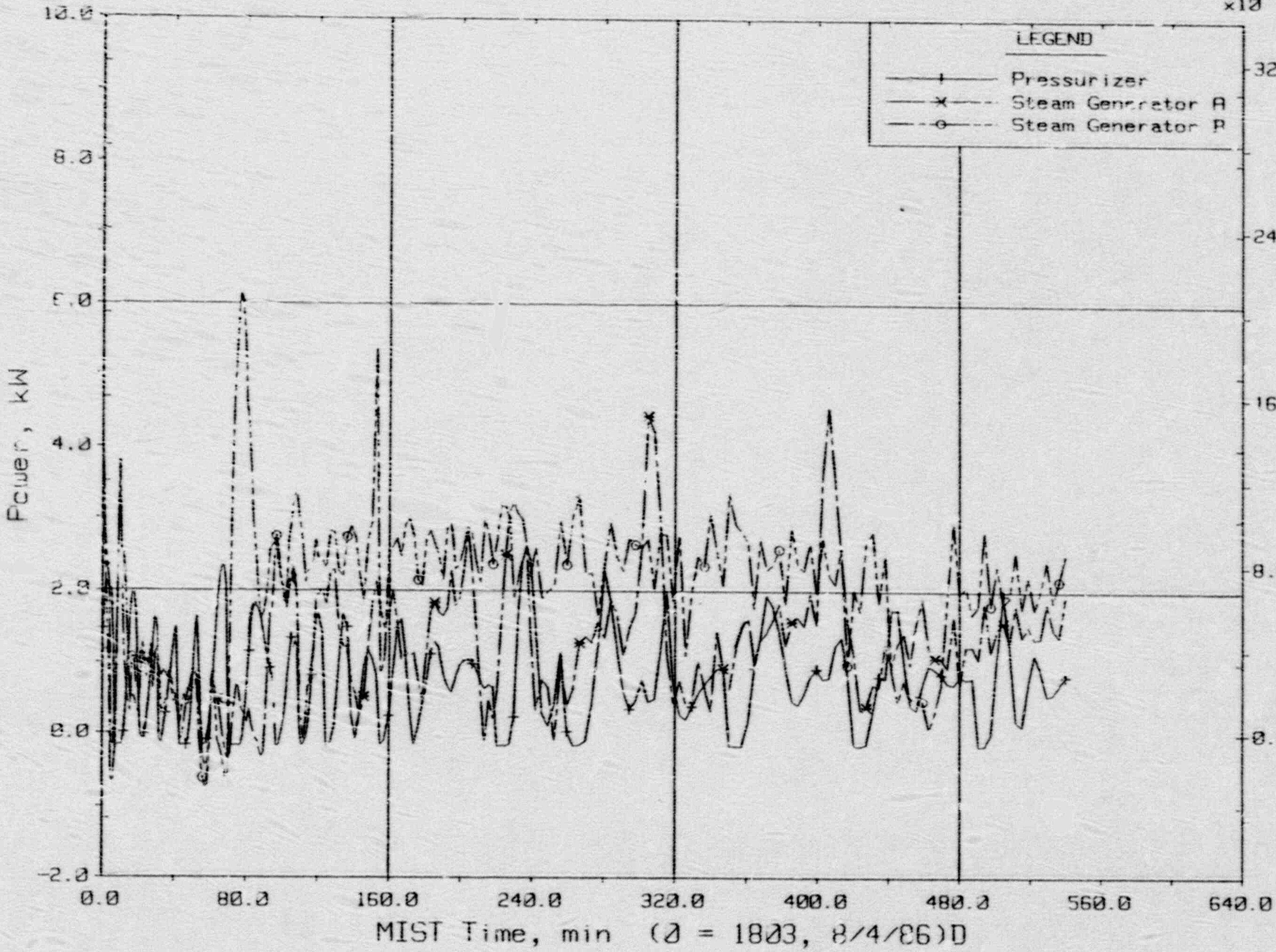
Power-Operated Relief Vclve Enthalpy (Based On Flow Rate).

Tue May 21 15:45:44 1982

PZEN1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization



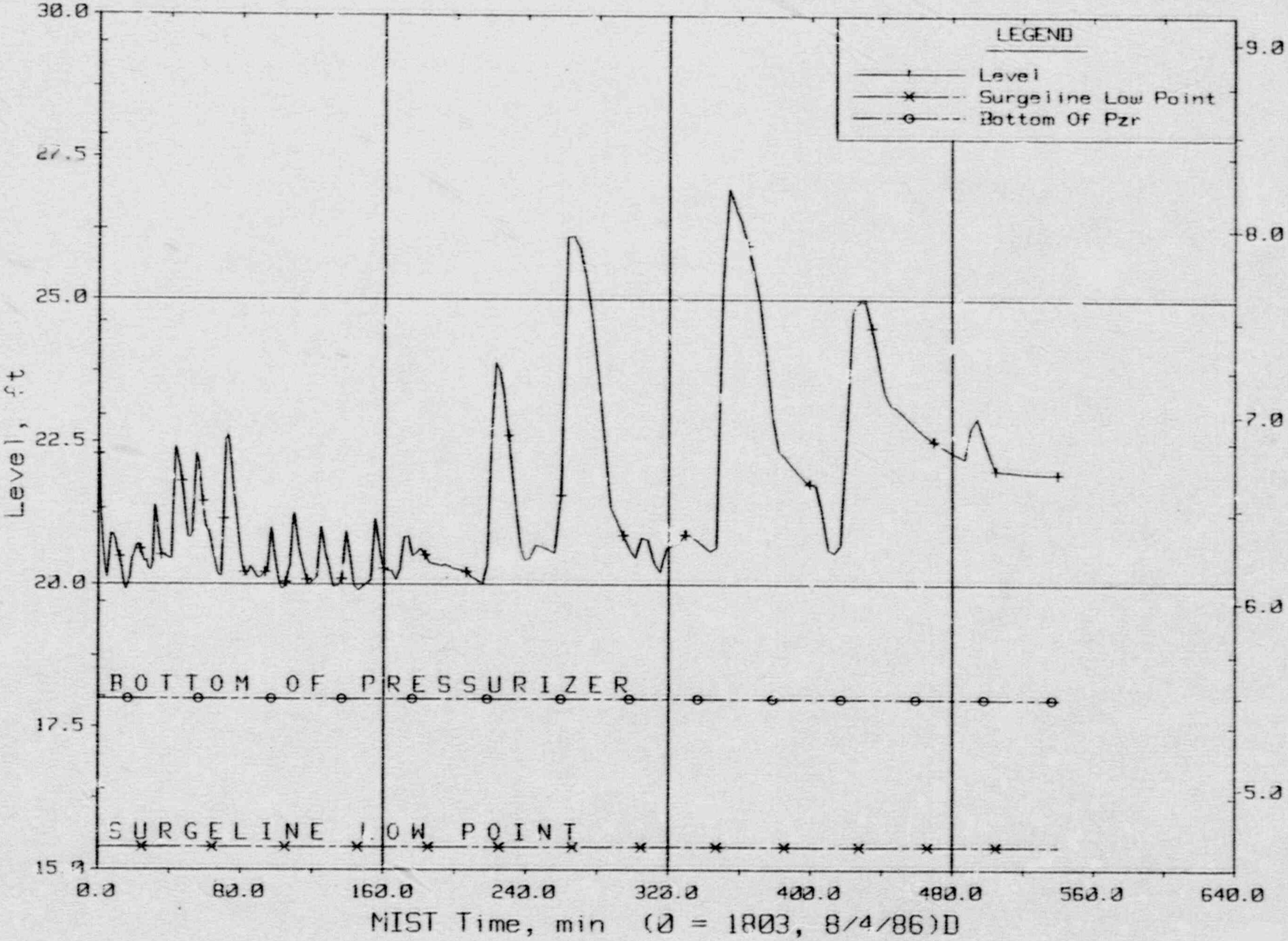
Guard Heater Specified Power, Pressurizer and Steam Generators.

Tue May 31 16:52:23 1988

PZG01

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



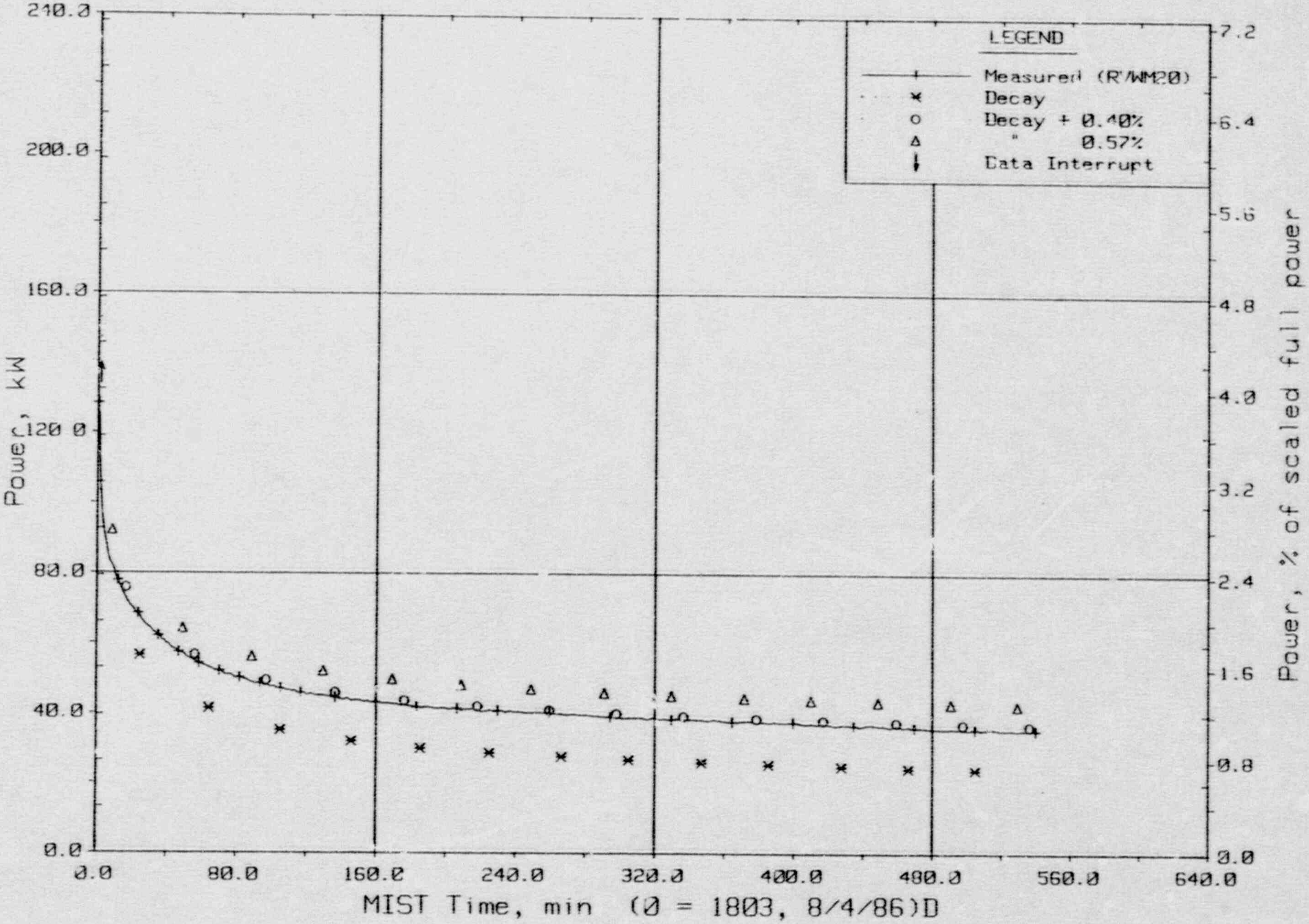
Pressurizer Collapsed Liquid Level (PZLV20).

Tue May 31 16:55:22 1988

PZLV1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



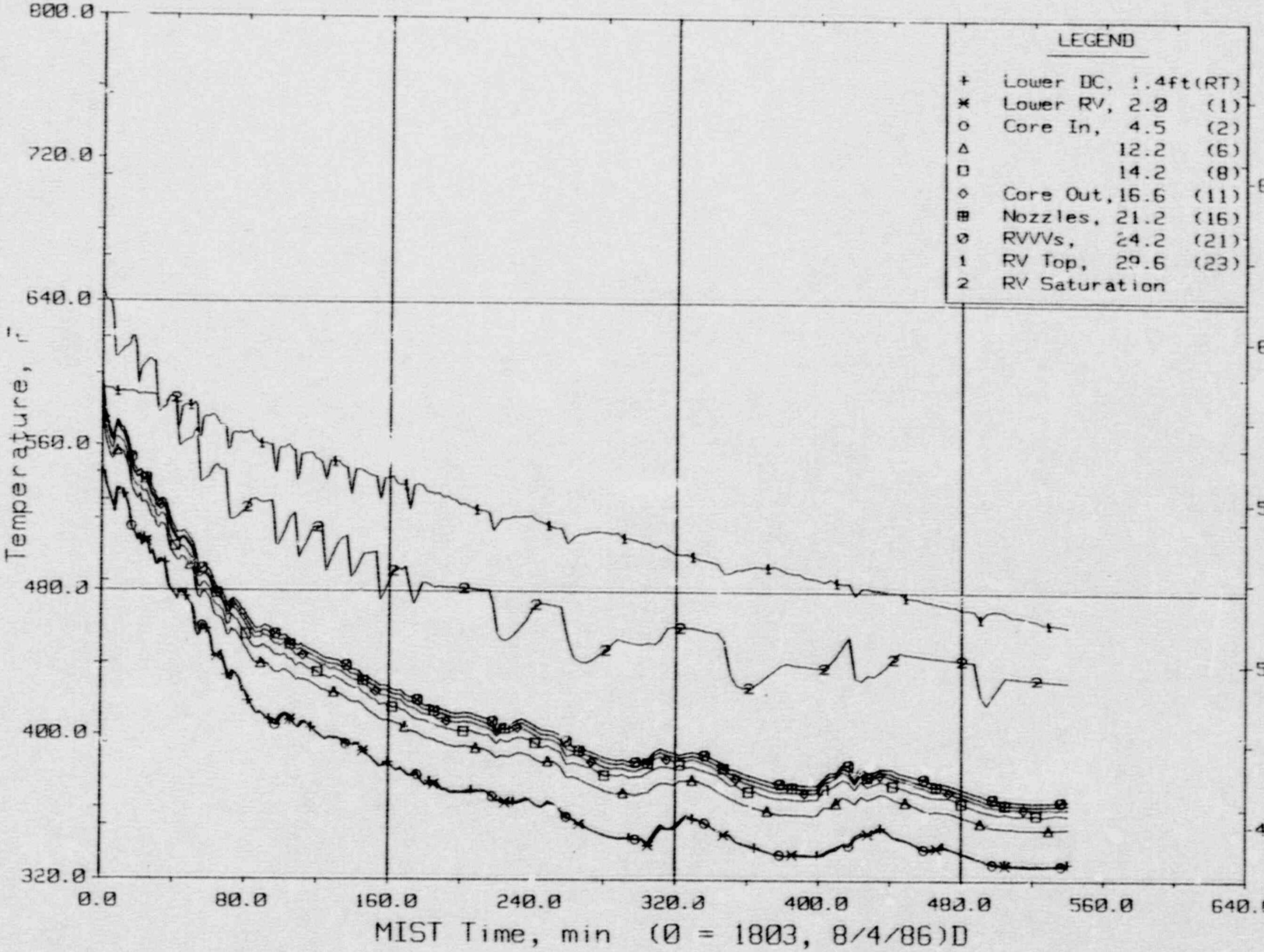
Core Power.

Tue May 31 16:57:37 1988

RVKW0

FINAL DATA

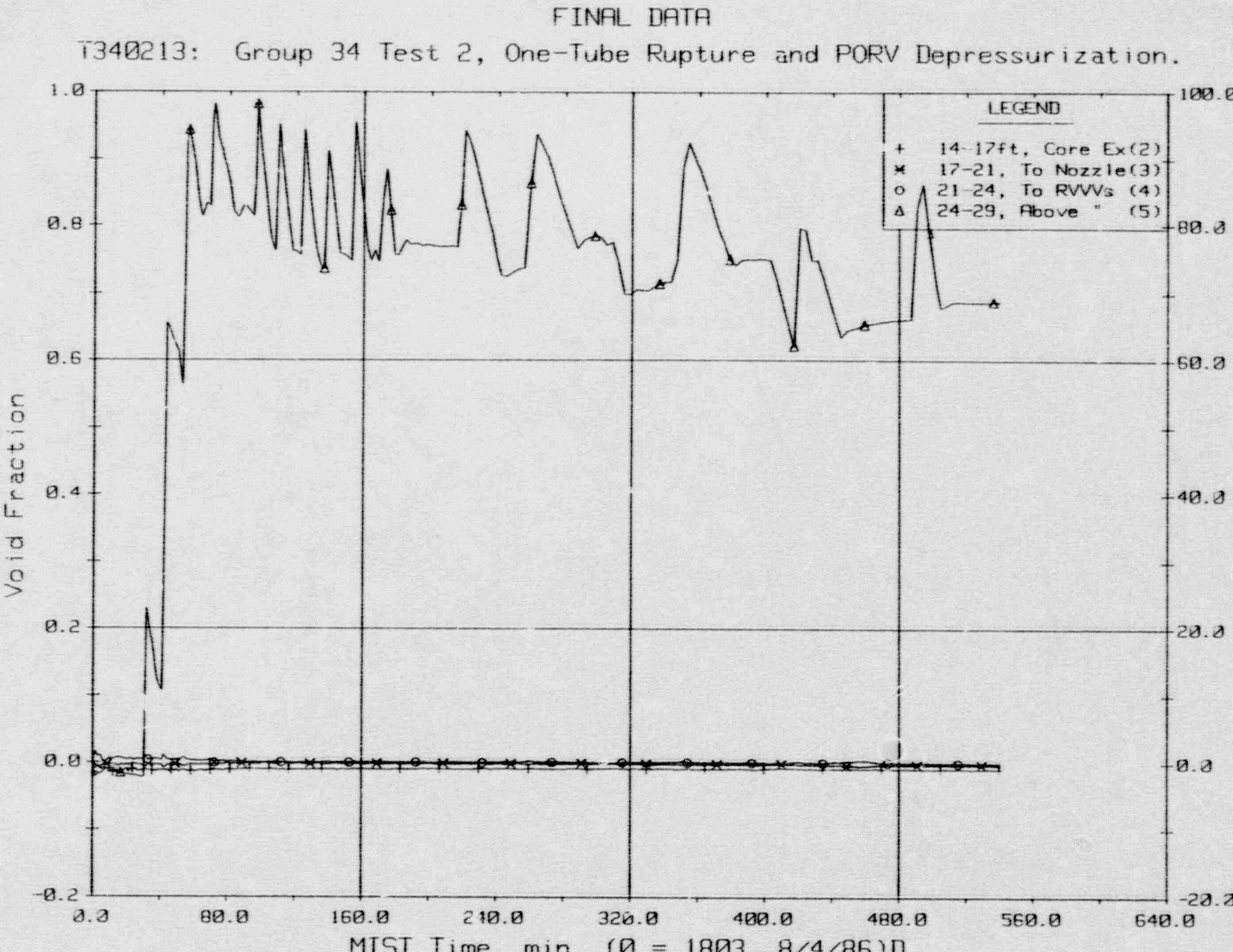
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Tue May 31 17:00:56 1988

RVTC0



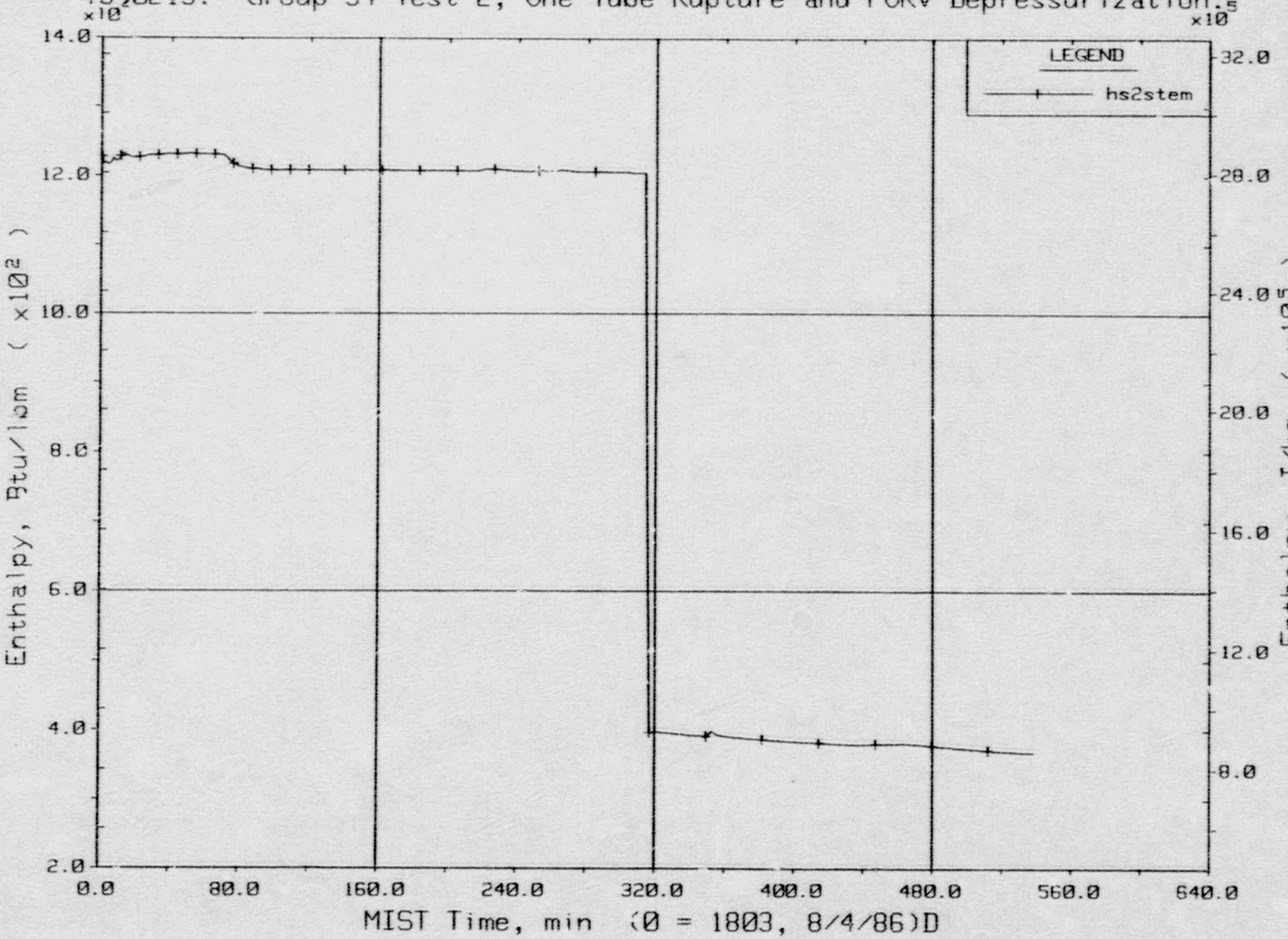
Reactor Vessel Void Fractions From Differential Pressure (RV/Fs).

Tue May 31 17:03:37 1988

RVVF1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization



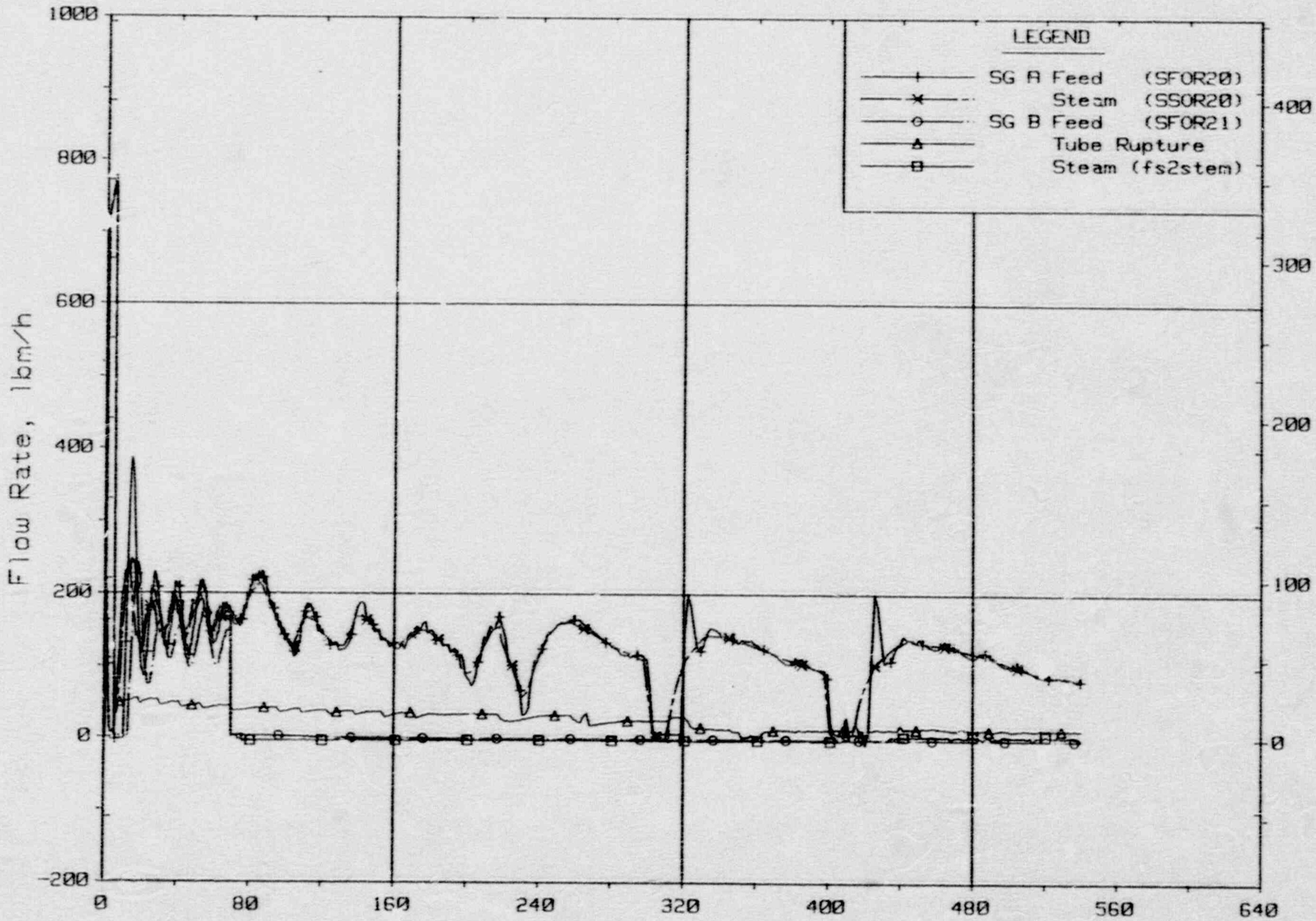
Steam Generator B Steam Exit Enthalpy.

Wed Jun 1 08:29:26 1988

S2EN1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



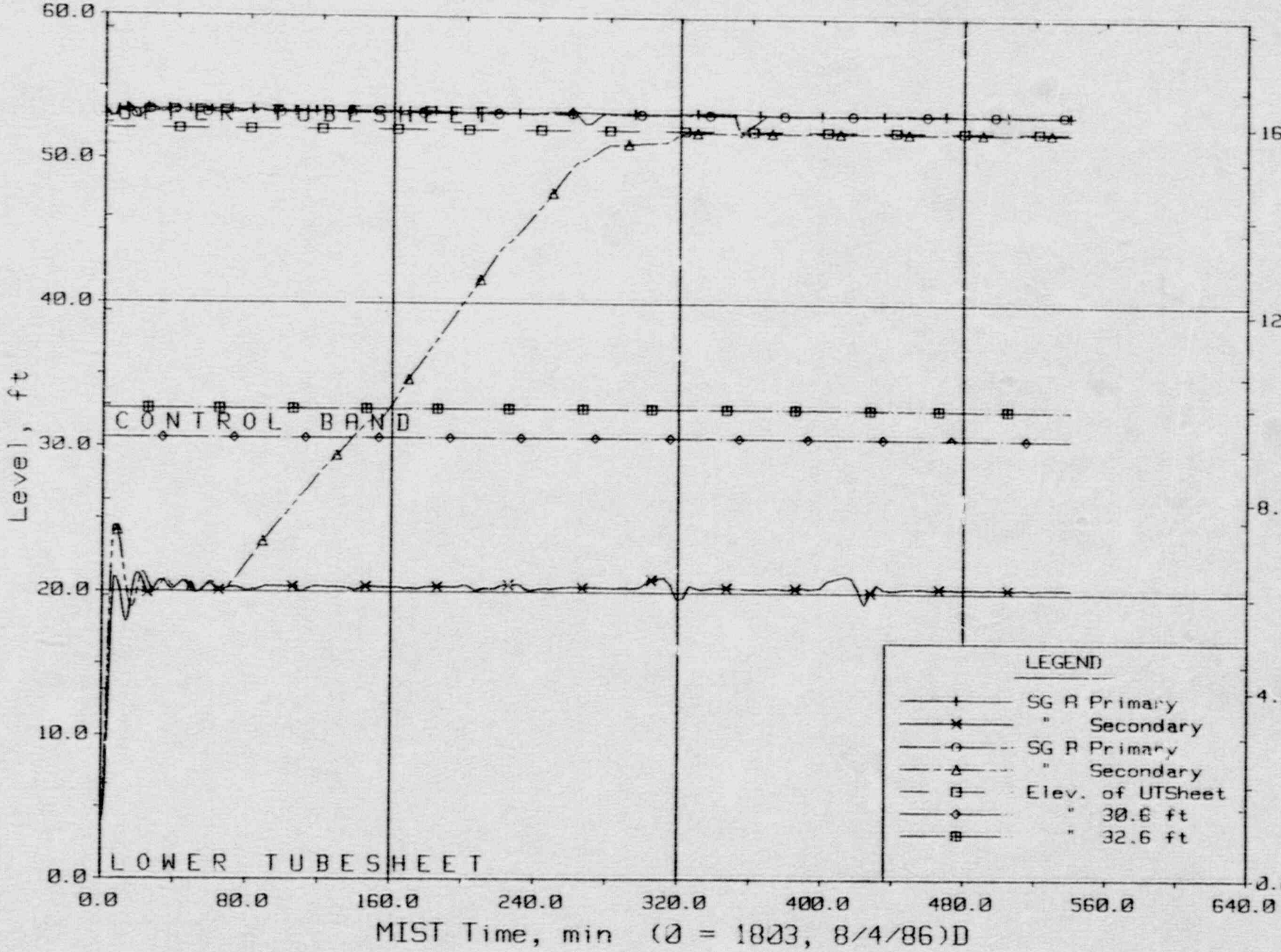
Steam Generator Secondary Flow Rates.

Tue Nov 1 17:19:21 1988

SGFL1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



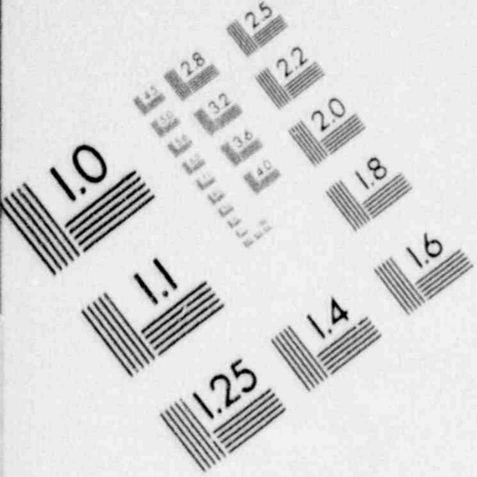
Steam Generator Collapsed Liquid Levels.

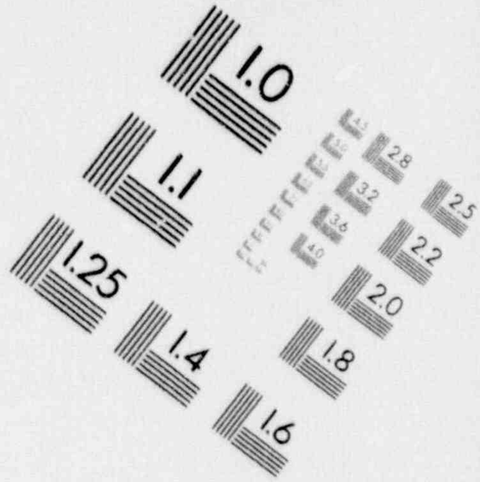
Tue May 31 17:09:23 1988

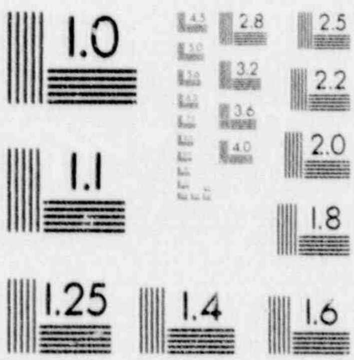
SGLV2

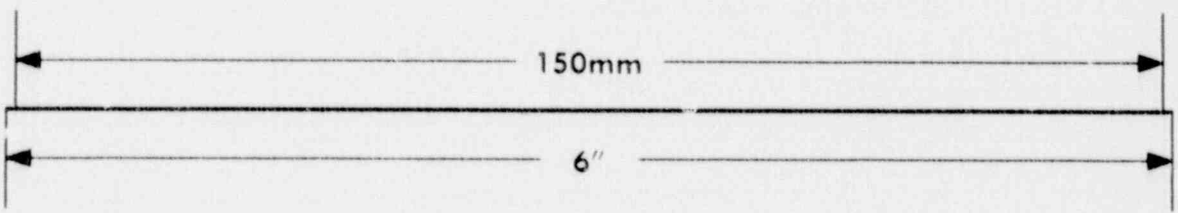
2

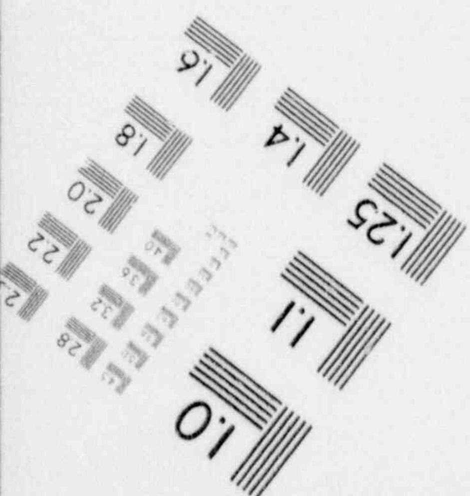
IMAGE EVALUATION TEST TARGET (MT-3)

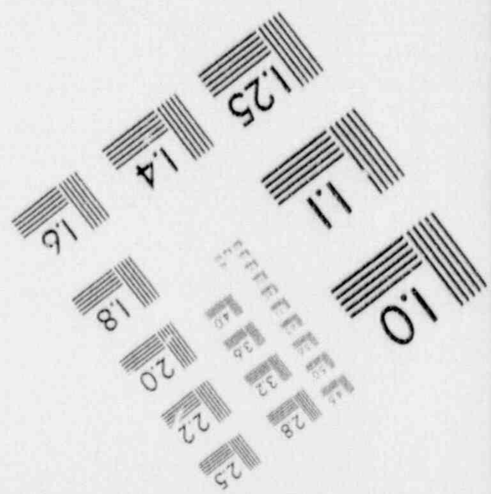












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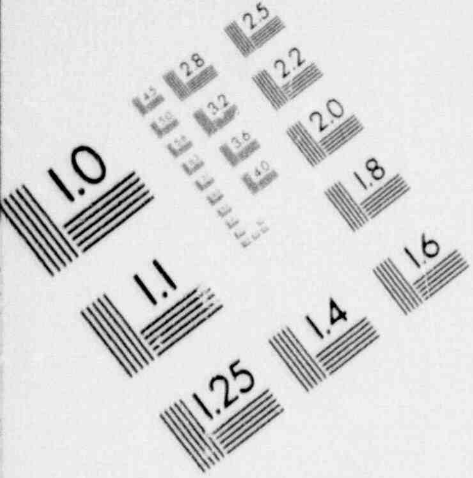
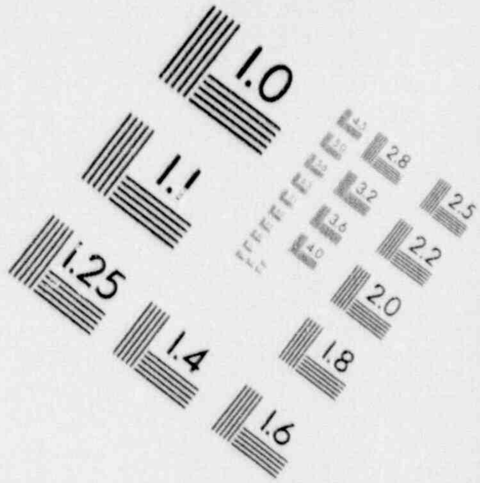
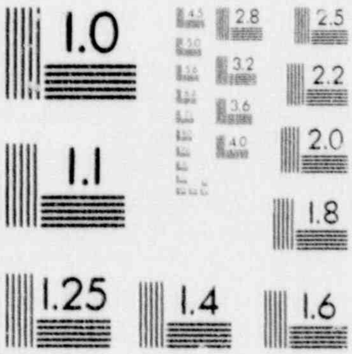
770 BASKET ROAD
P.O. BOX 338
WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

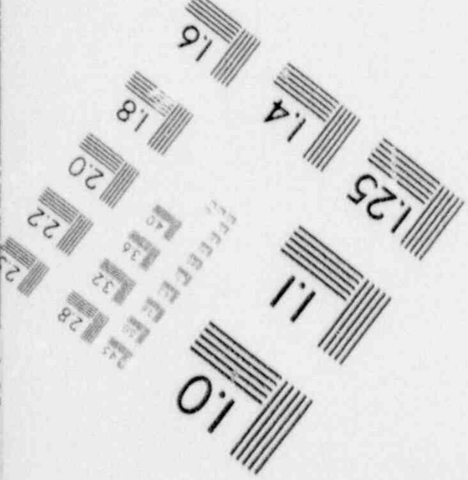


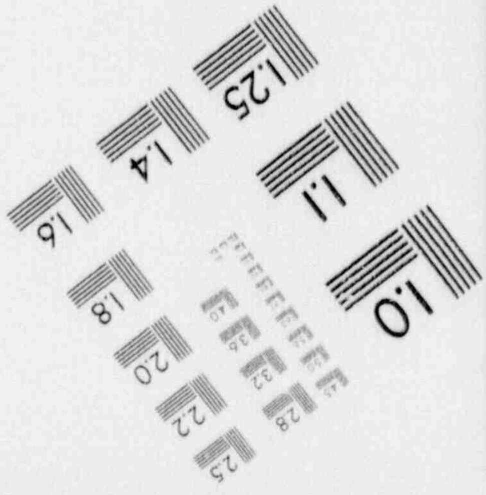




150mm

6″





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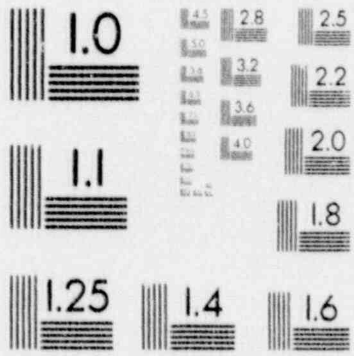
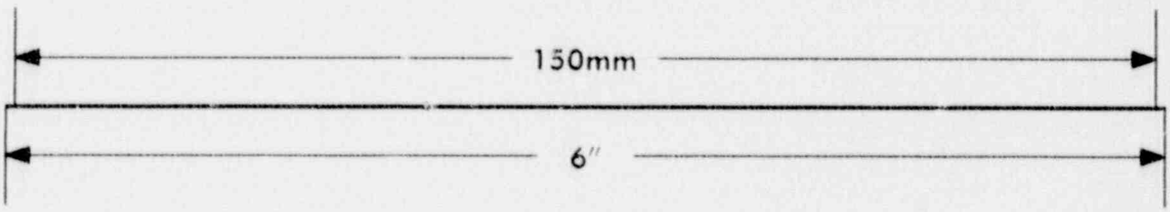
P.O. BOX 338
WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)





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WEBSTER, NEW YORK 14580
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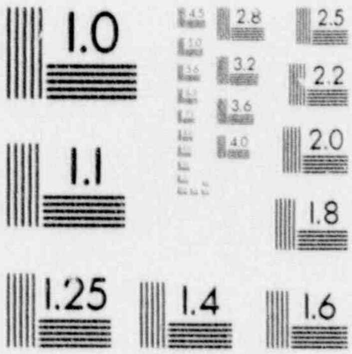
2

IMAGE EVALUATION TEST TARGET (MT-3)



150mm

6"

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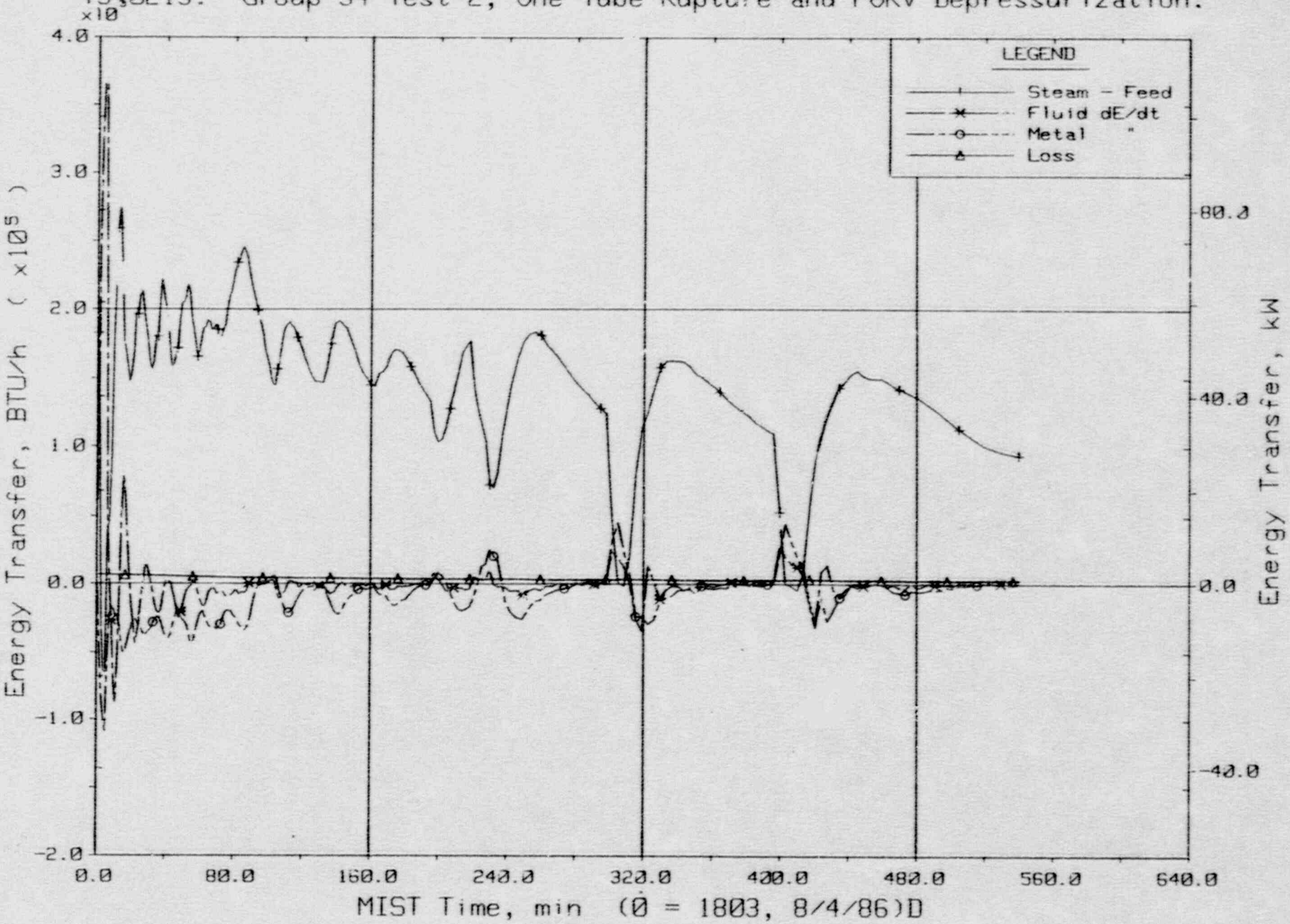
WEBSTER, NEW YORK 14580
(716) 265-1600

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



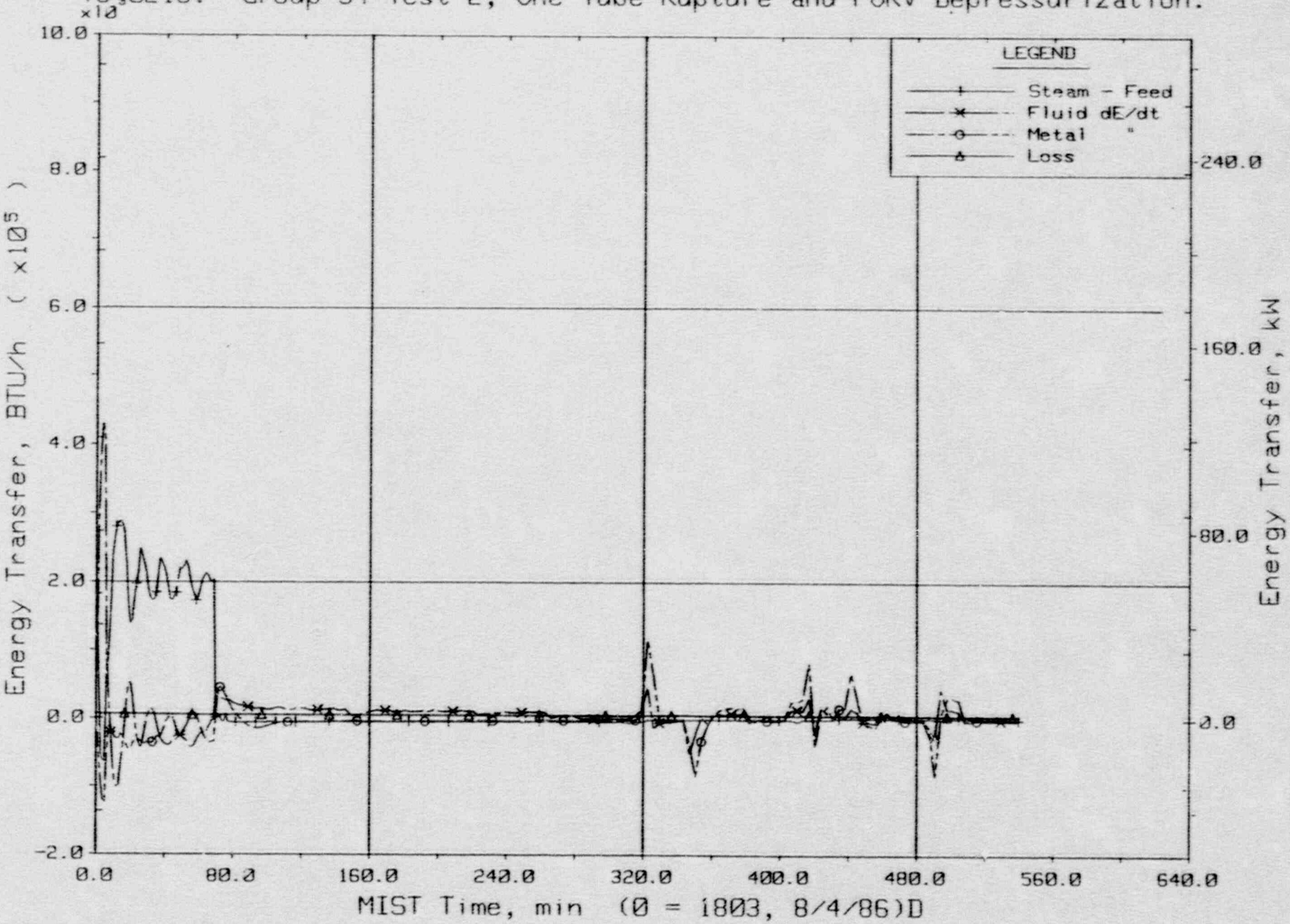
Steam Generator A Energy Transfer.

Tue May 31 :14:56 1988

SGQE3

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



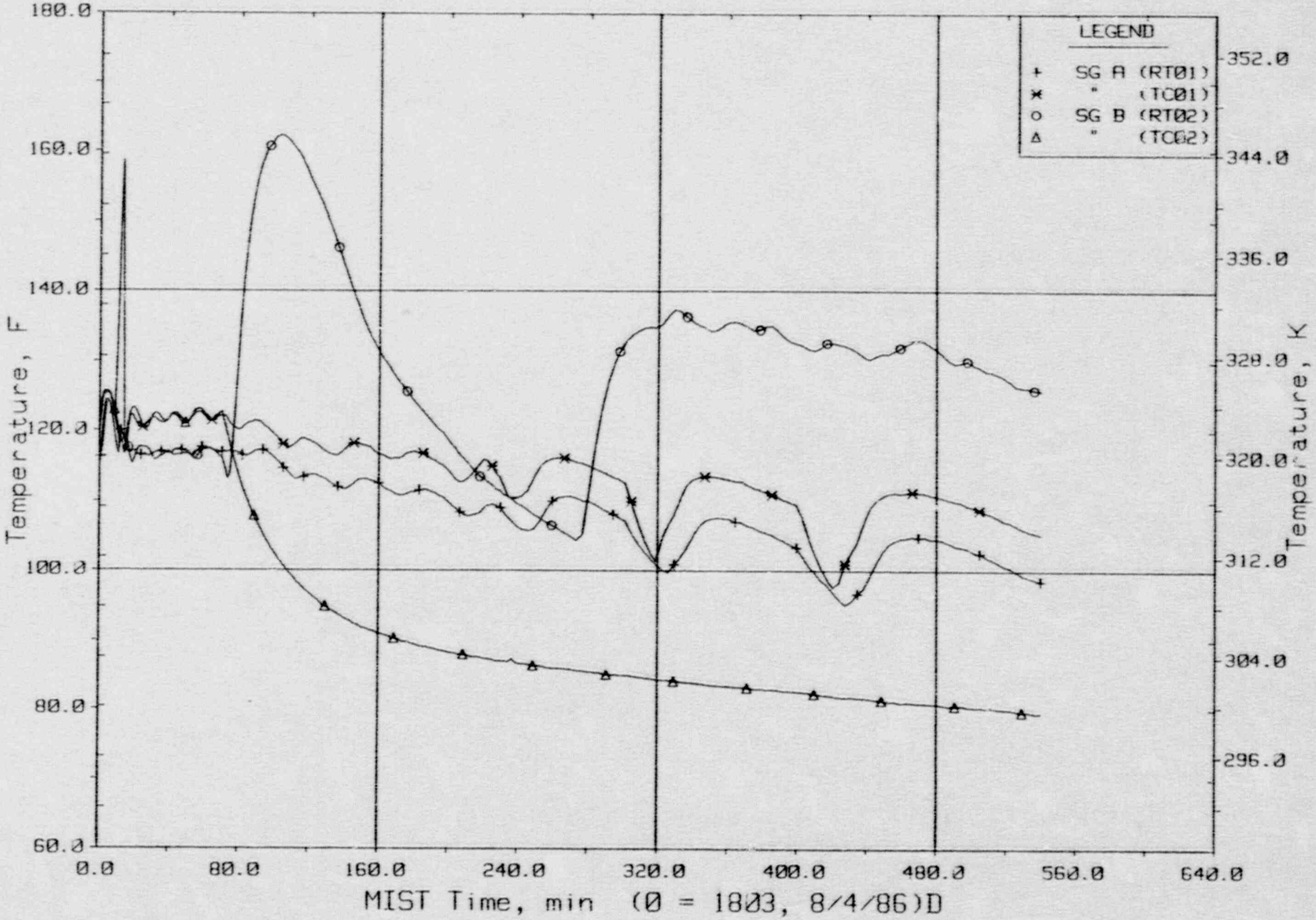
Steam Generator B Energy Transfer.

Tue May 31 17:17:36 1988

SGOE6

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



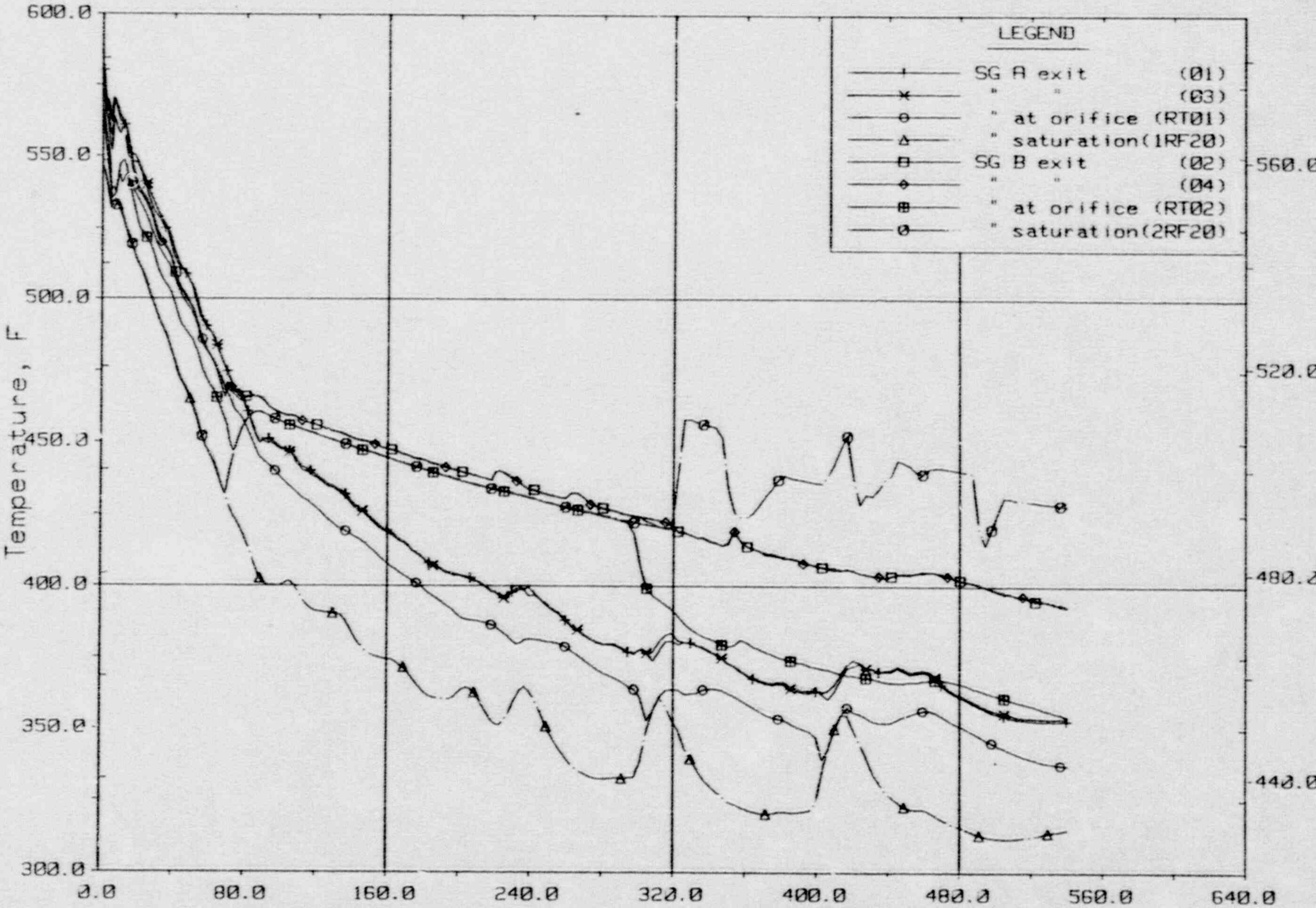
Feedwater Tempëratures (SFs).

Tue May 31 17:20:02 1988

SGRT1

FINAL DATA

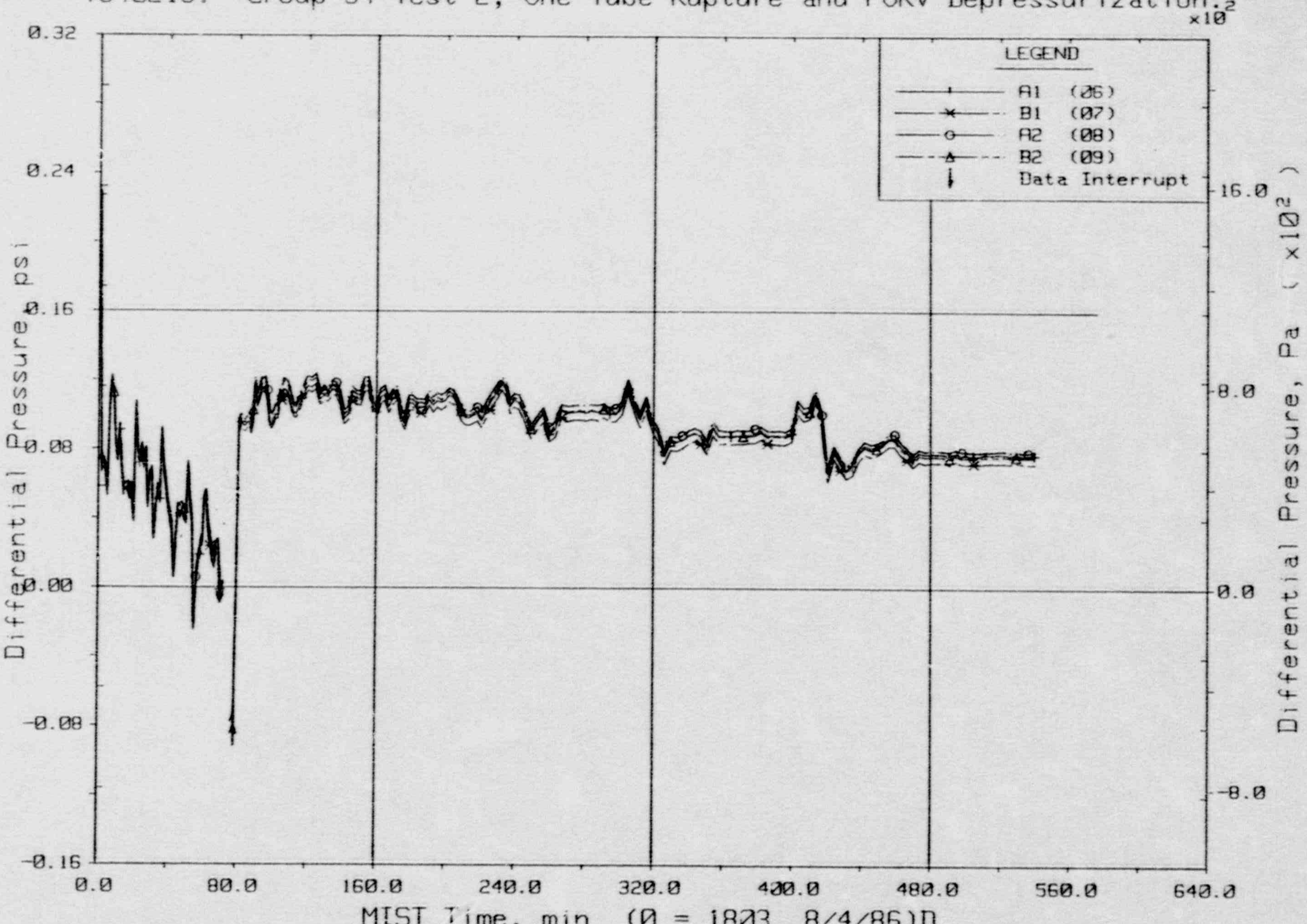
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Steam Generator Steam Outlet Temperatures (SSTCs).

FINAL DATA

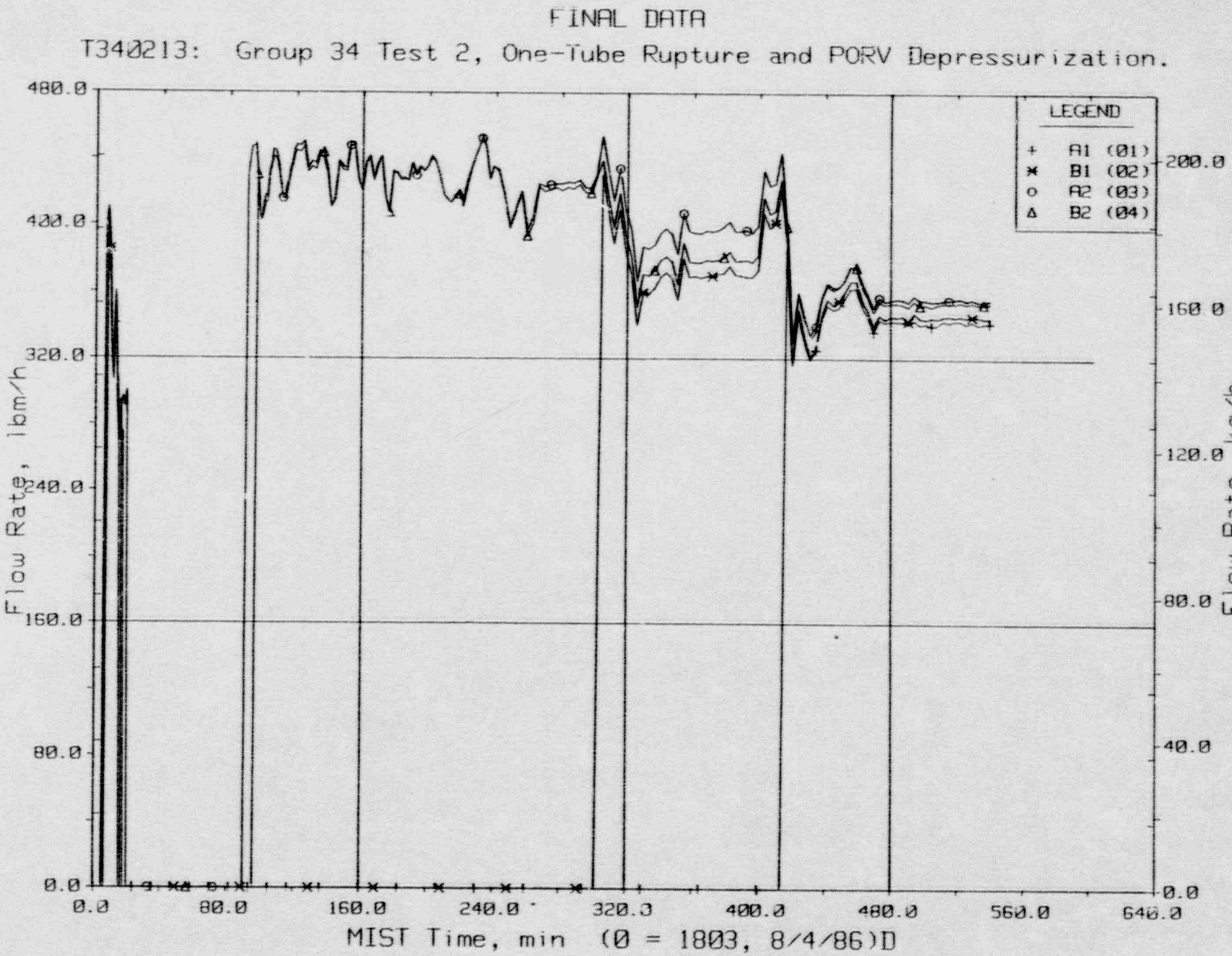
T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization



Reactor Vessel Vent Valve Differential Pressures (RVDPs).

Tue May 31 17:26:05 1988

VVDP1



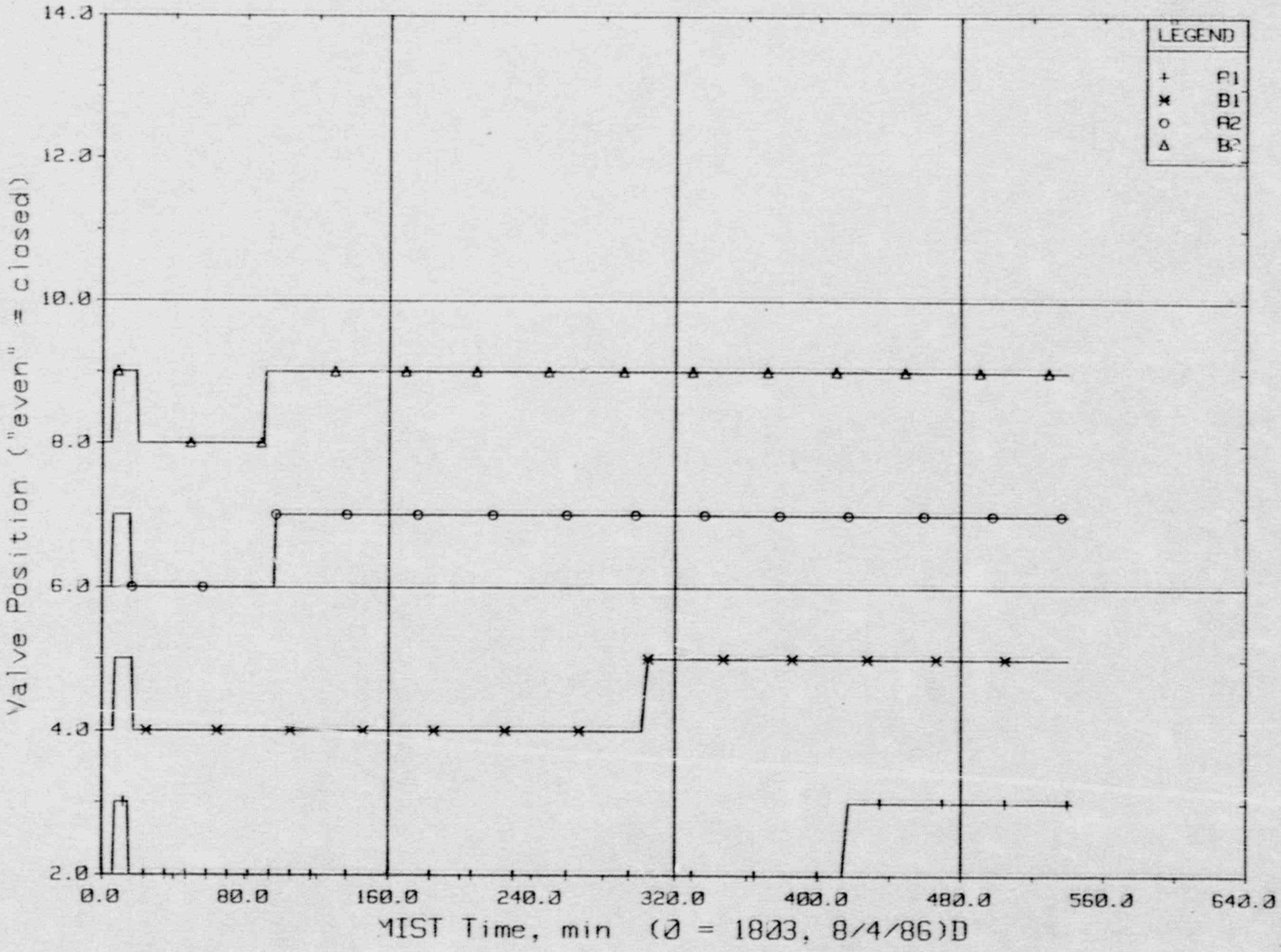
Reactor Vessel Vent Valve Flow Rates (RVORs).

Tue May 31 17:28:58 1988

VVFL1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



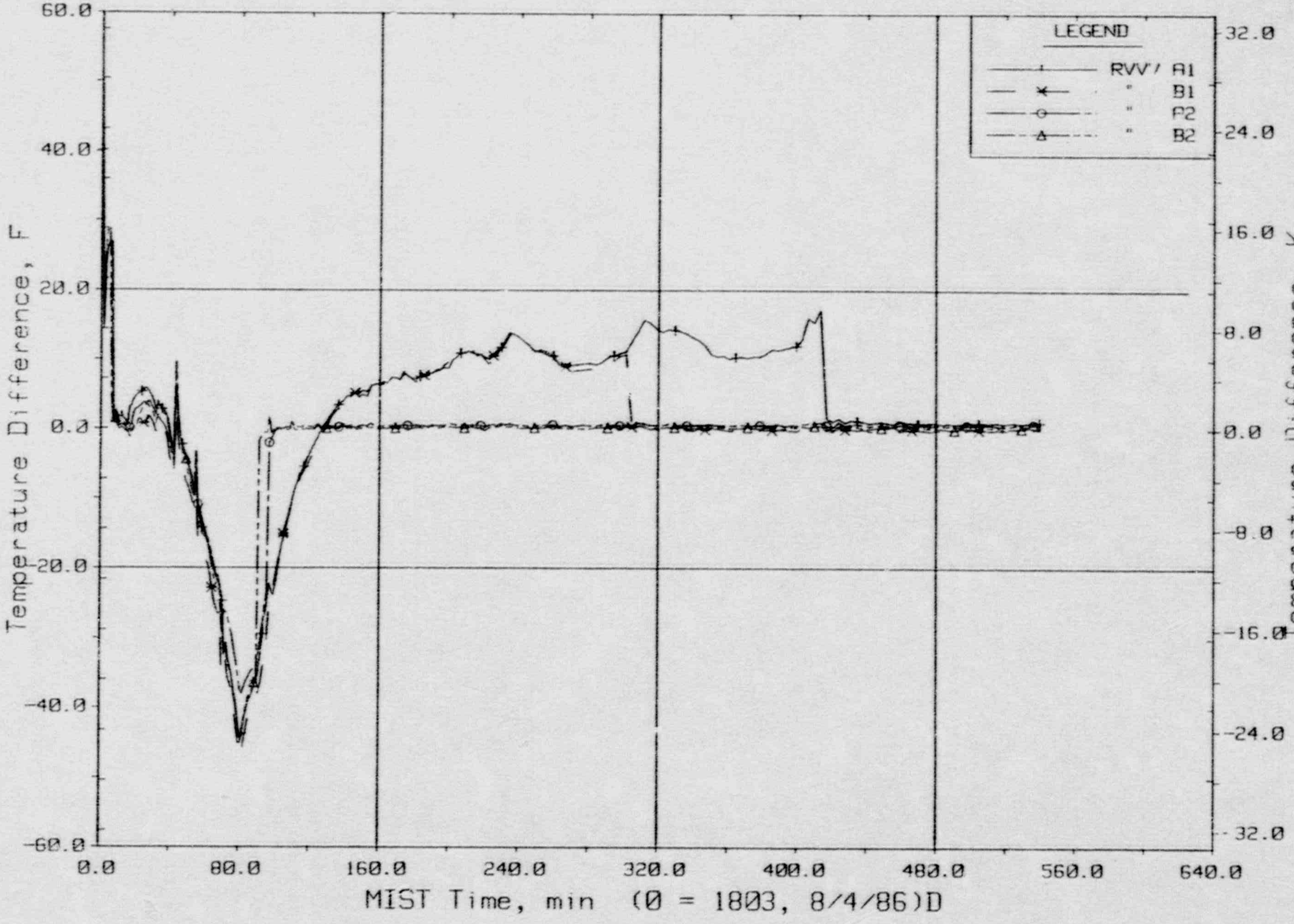
Reactor Vessel Vent Valve Positions.

Tue May 31 17:31:22 1988

VVLS1

FINAL DATA

T340213: Group 34 Test 2, One-Tube Rupture and PORV Depressurization.



Temperature Differences Across Vent Valves.

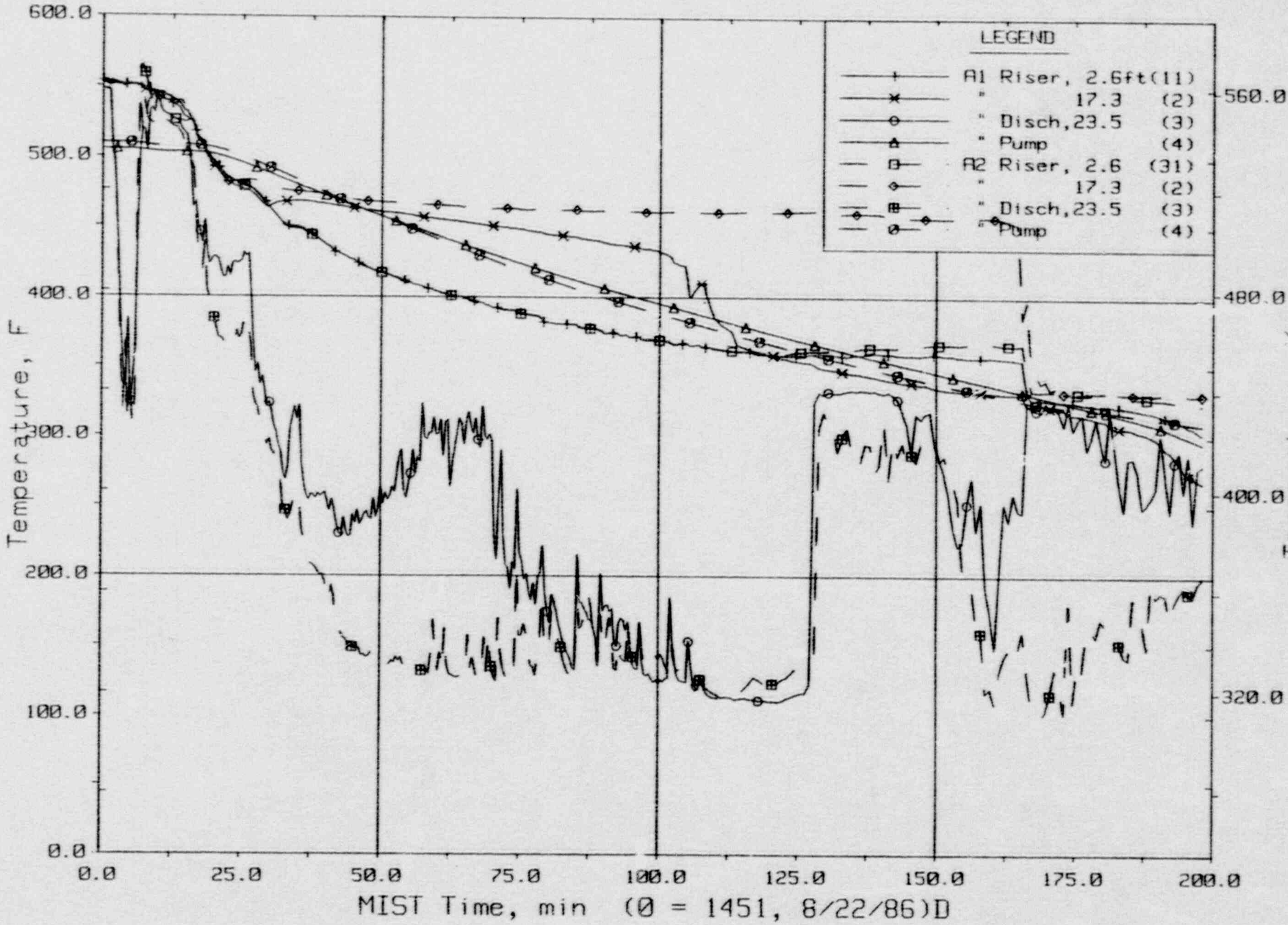
Tue May 31 17:33:35 1988

VVTD1

T340302

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



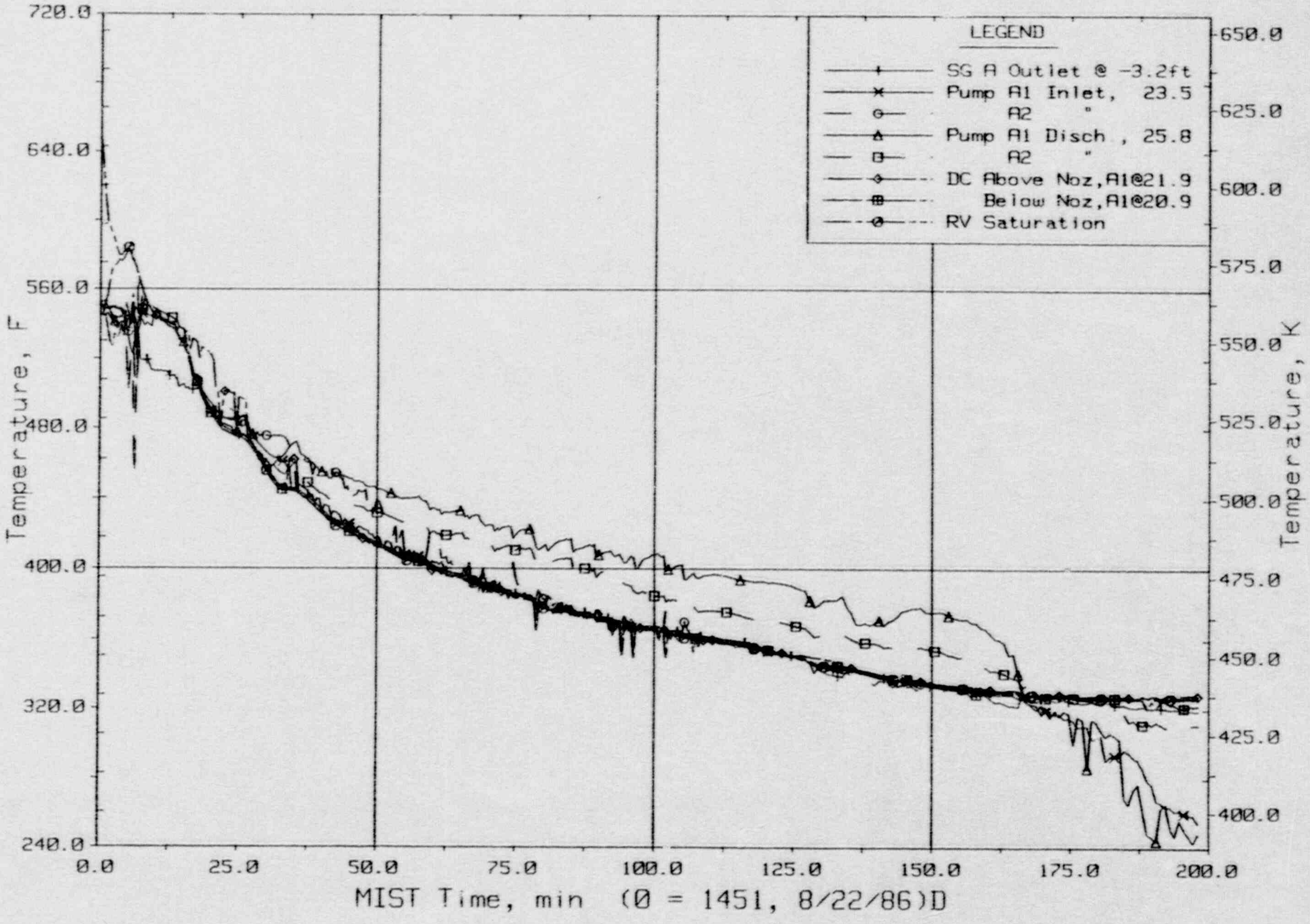
Loop A Cold Leg Metal Temperatures (C1,3MTs).

Wed Jun 1 08:07:20 1988

C1MT1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



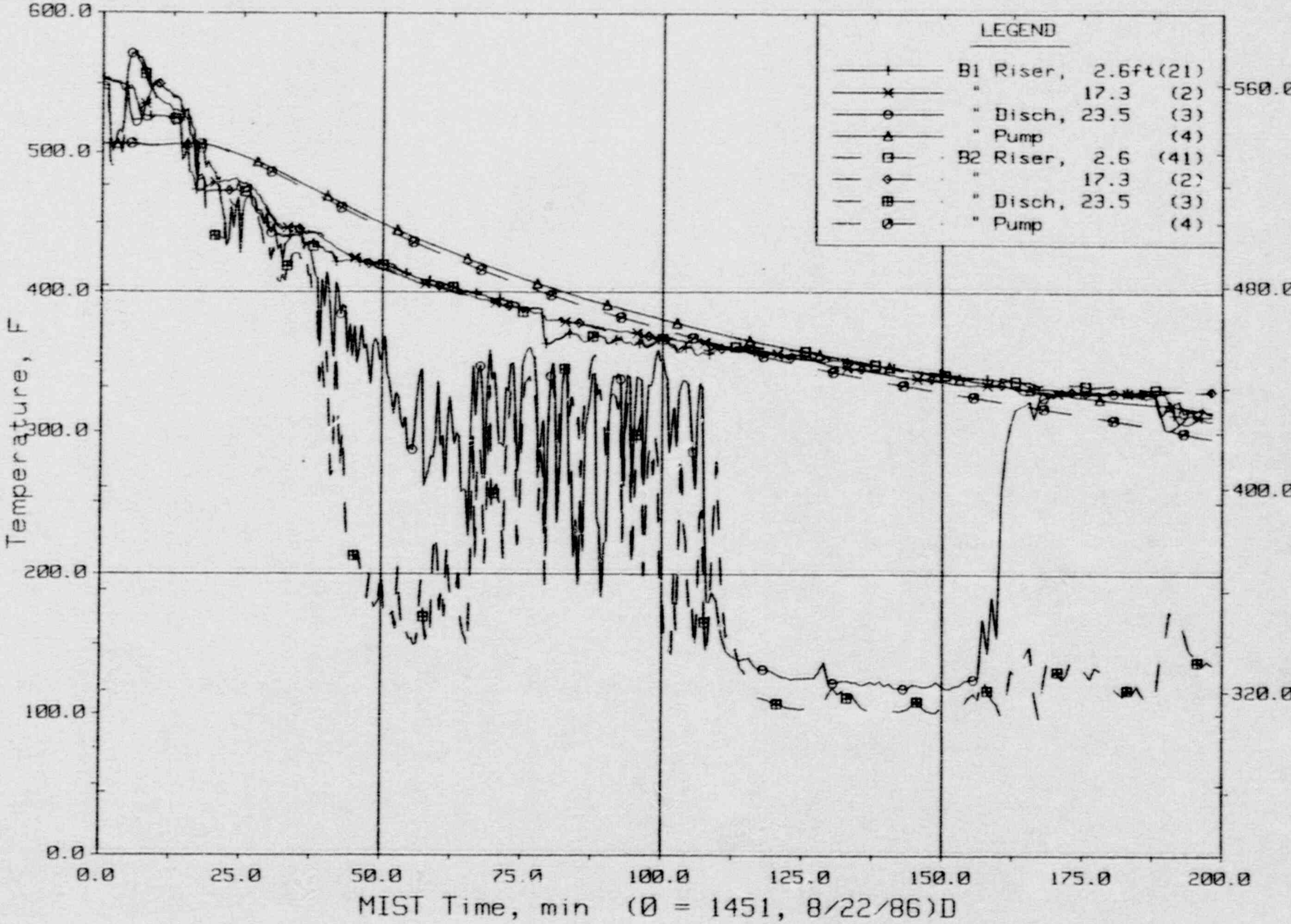
Loop A Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 08:11:59 1988

C1RT0

FINAL DATA

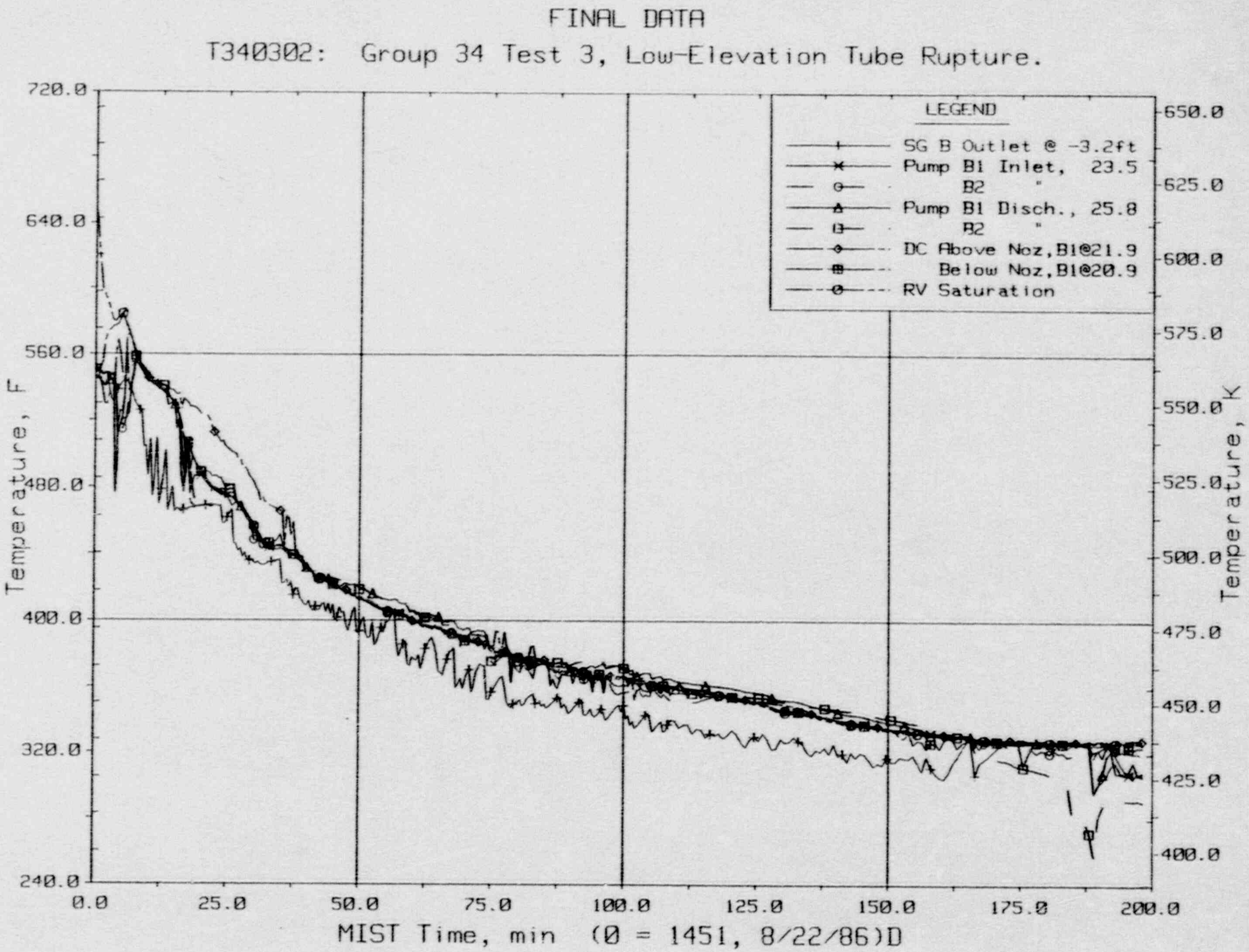
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Loop B Cold Leg Metal Temperatures (C2,4MTs).

Wed Jun 1 08:16:51 1988

C2MT1



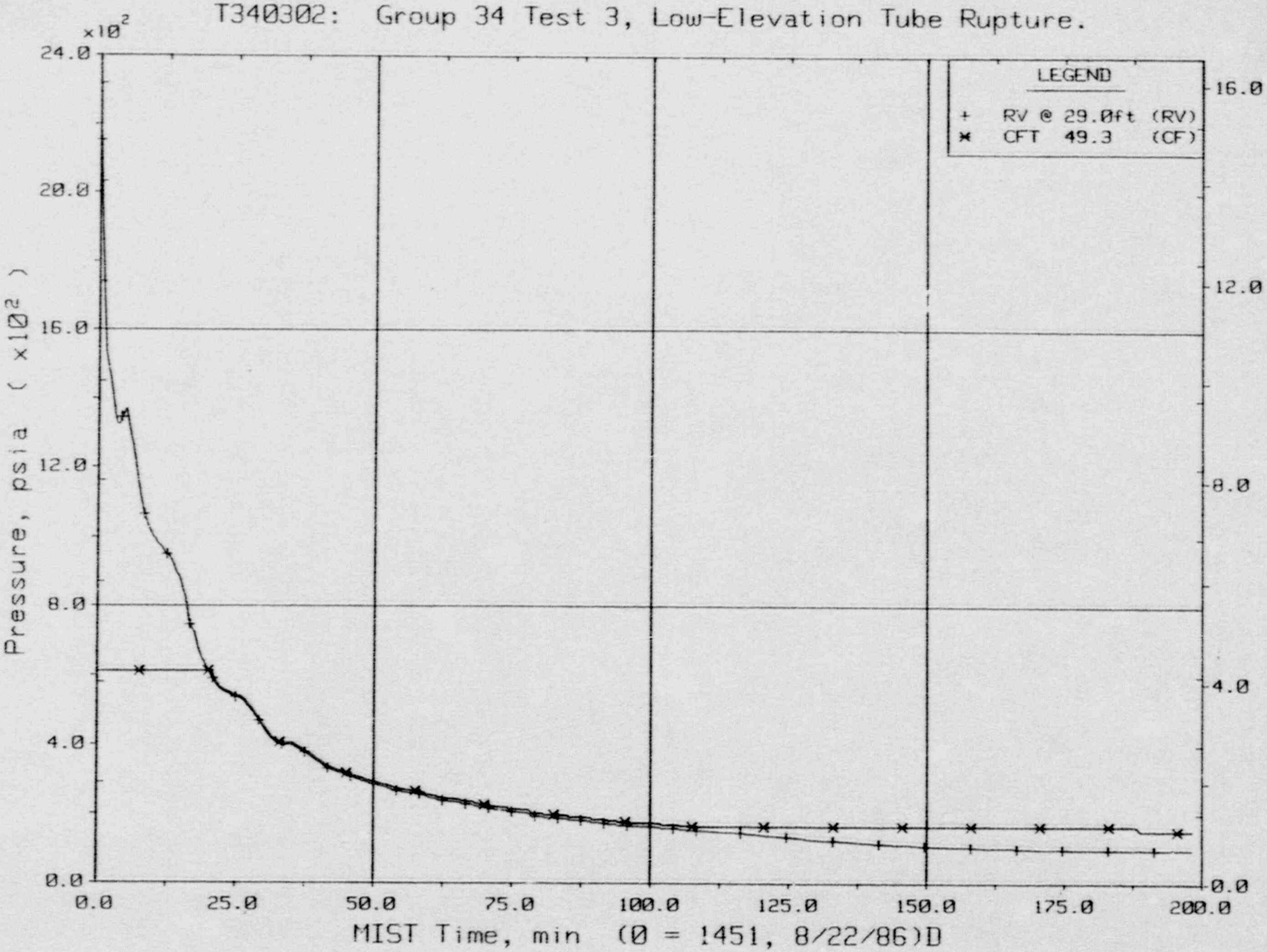
Loop B Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 08:22:00 1988

C2RT0

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



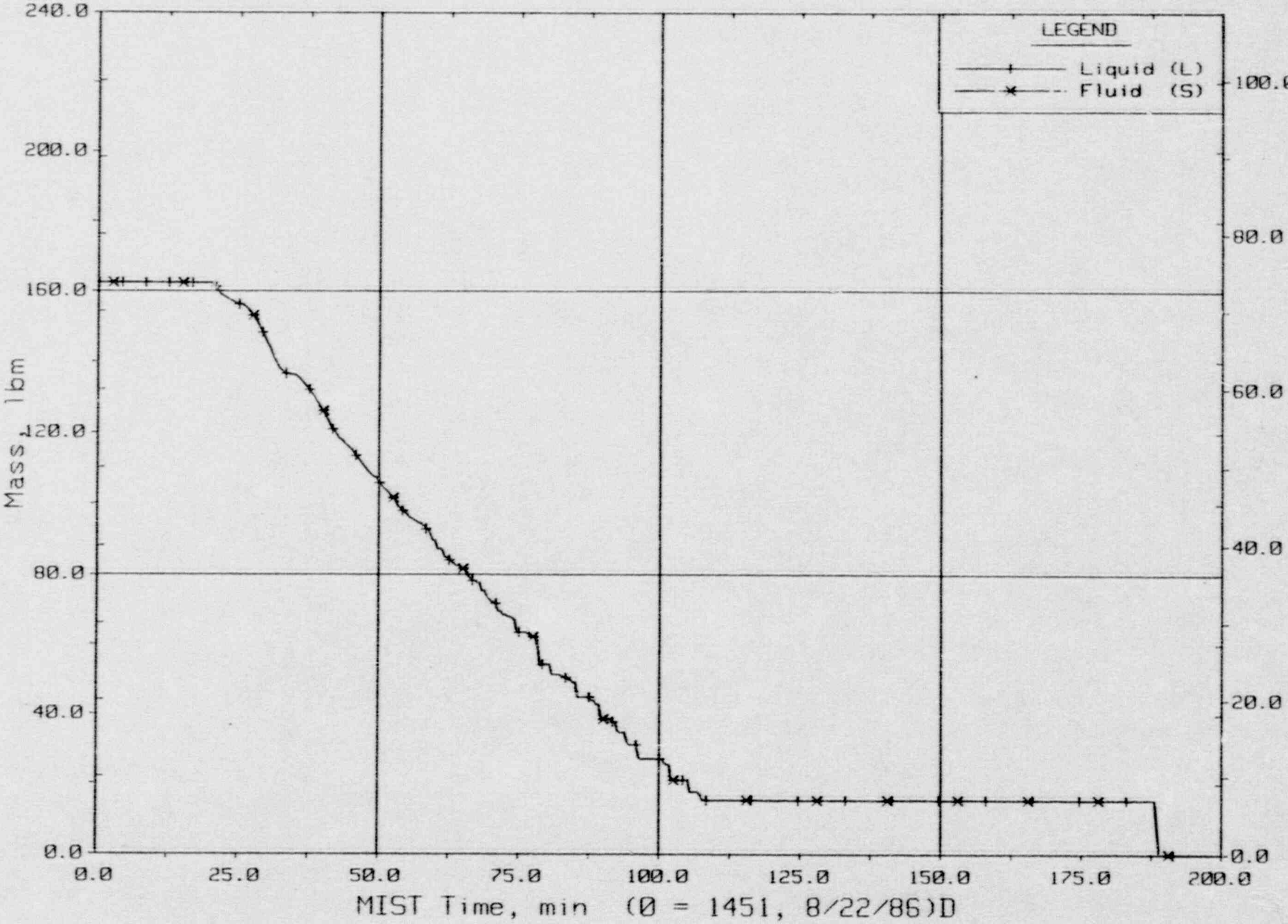
Primary System and Core Flood Tank Pressures (GP01s).

Wed Jun 1 08:25:08 1988

CFGP1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



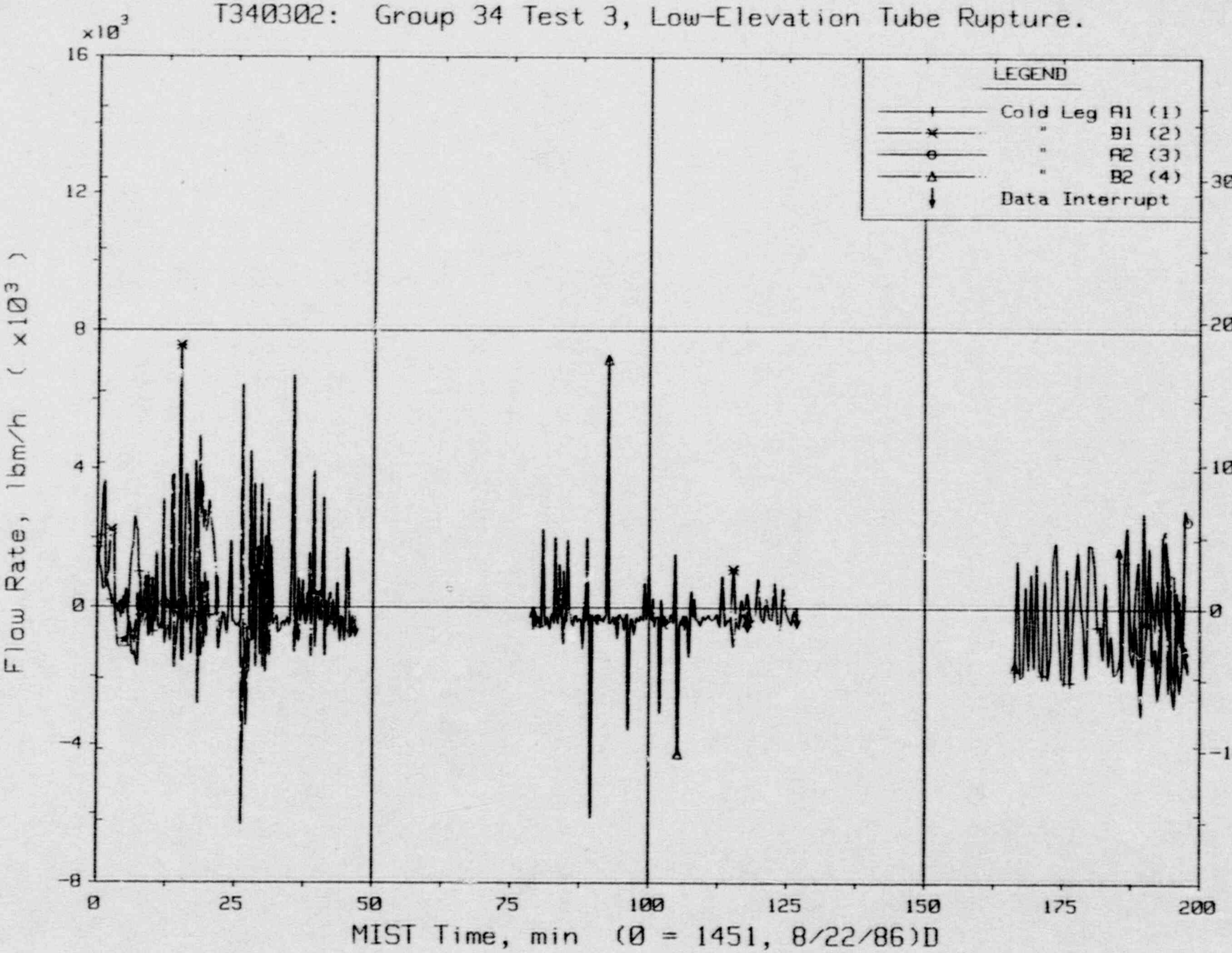
Core Flood Tank Liquid and Fluid Mass (CFMa20s).

Wed Jun 1 08:42:47 1988

CFML1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



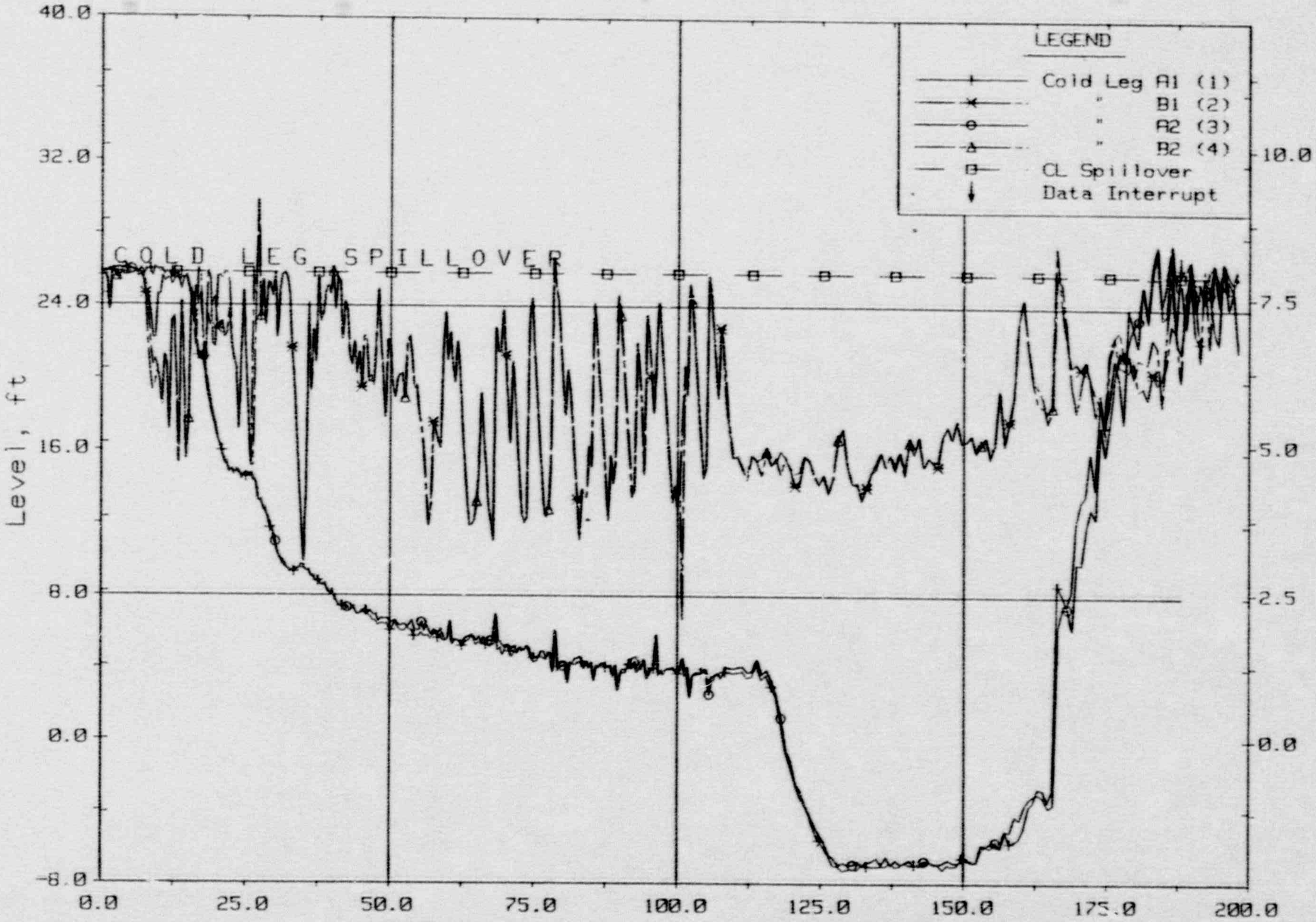
Cold Leg (Venturi) Flow Rates (CnVN20s).

Tue Nov 1 17:36:13 1988

CLFL0

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



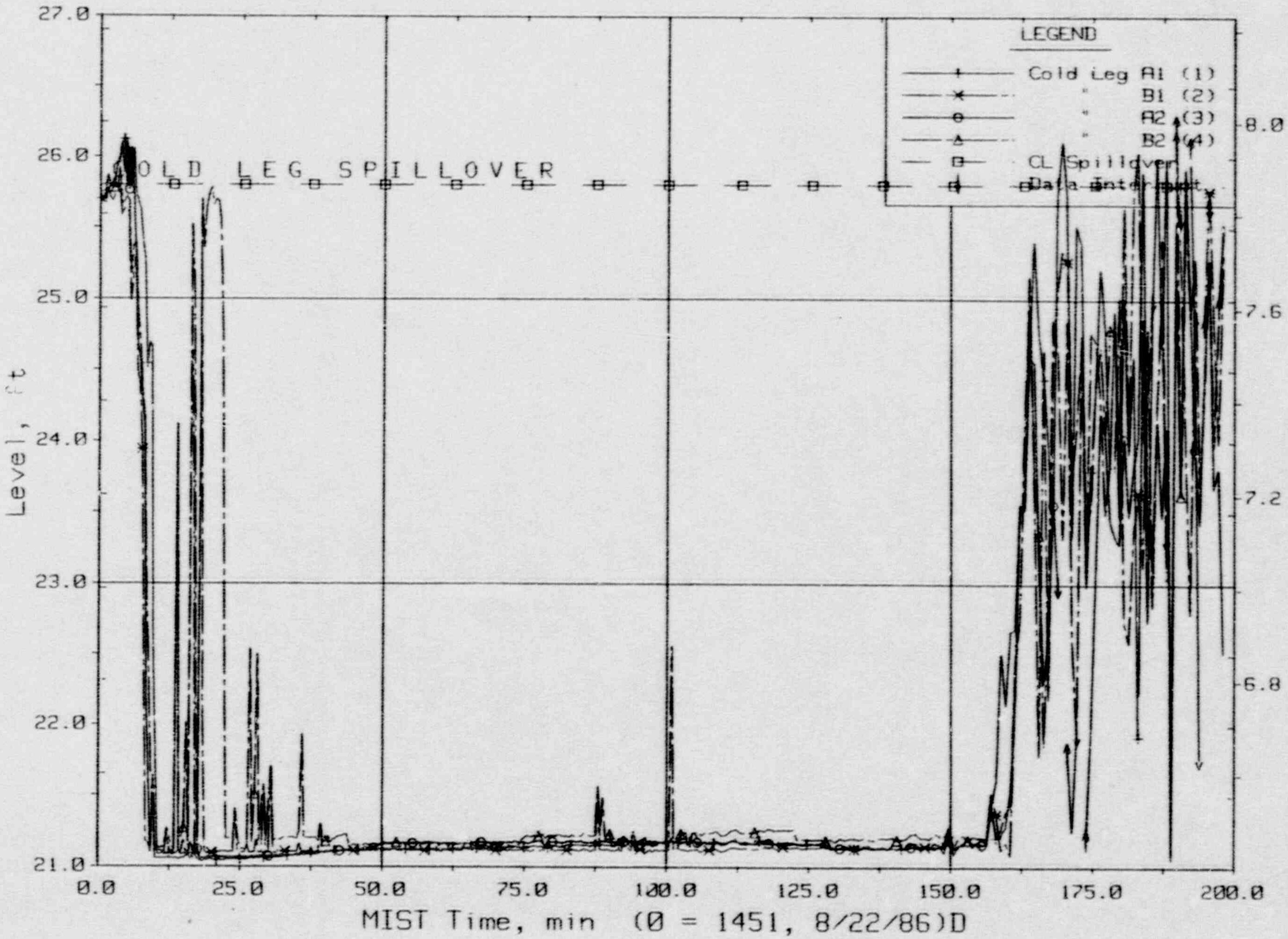
Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

Wed Jun 1 08:52:50 1988

CLLV4

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



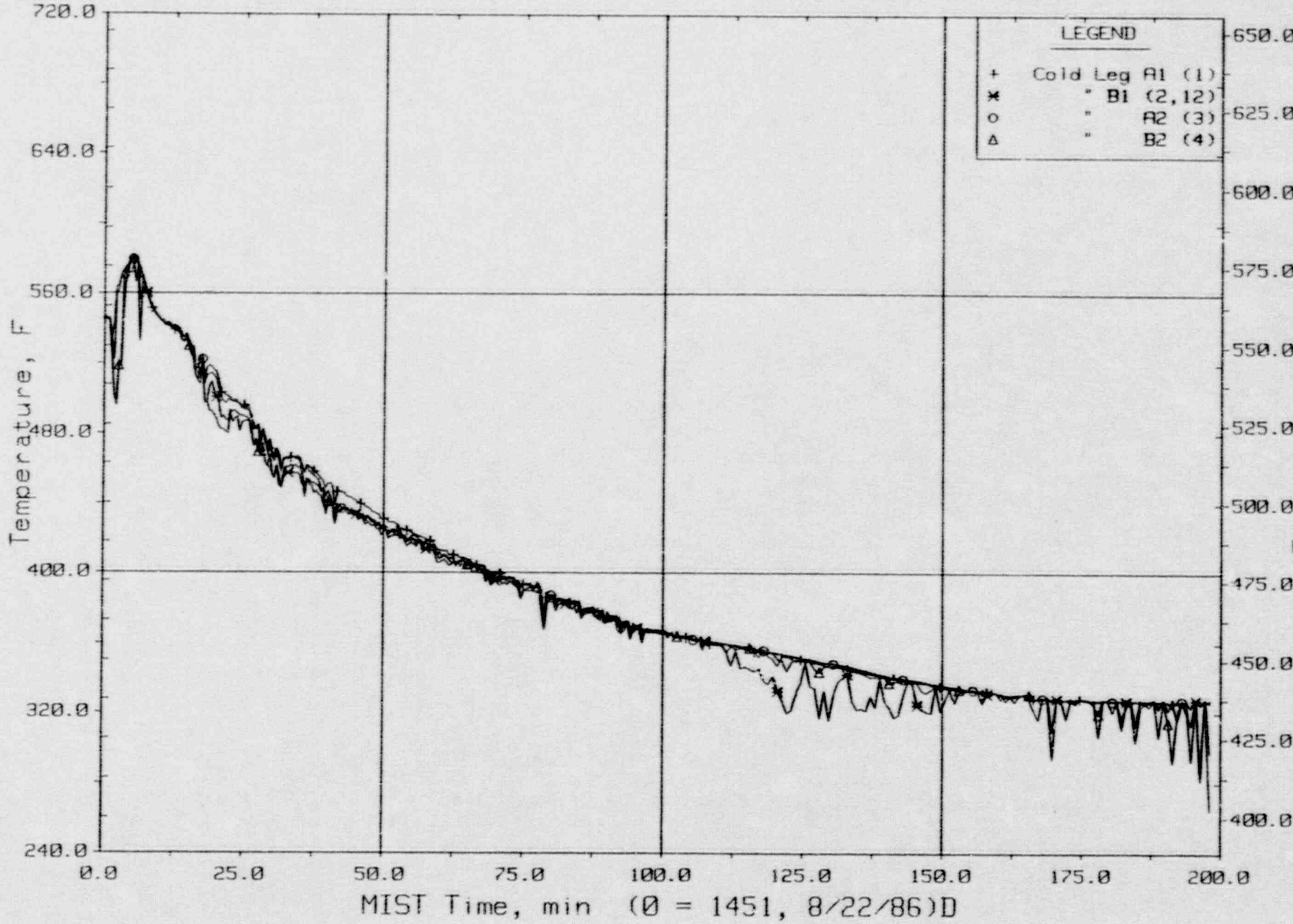
Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Wed Jun 1 08:57:22 1988

CLLV5

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



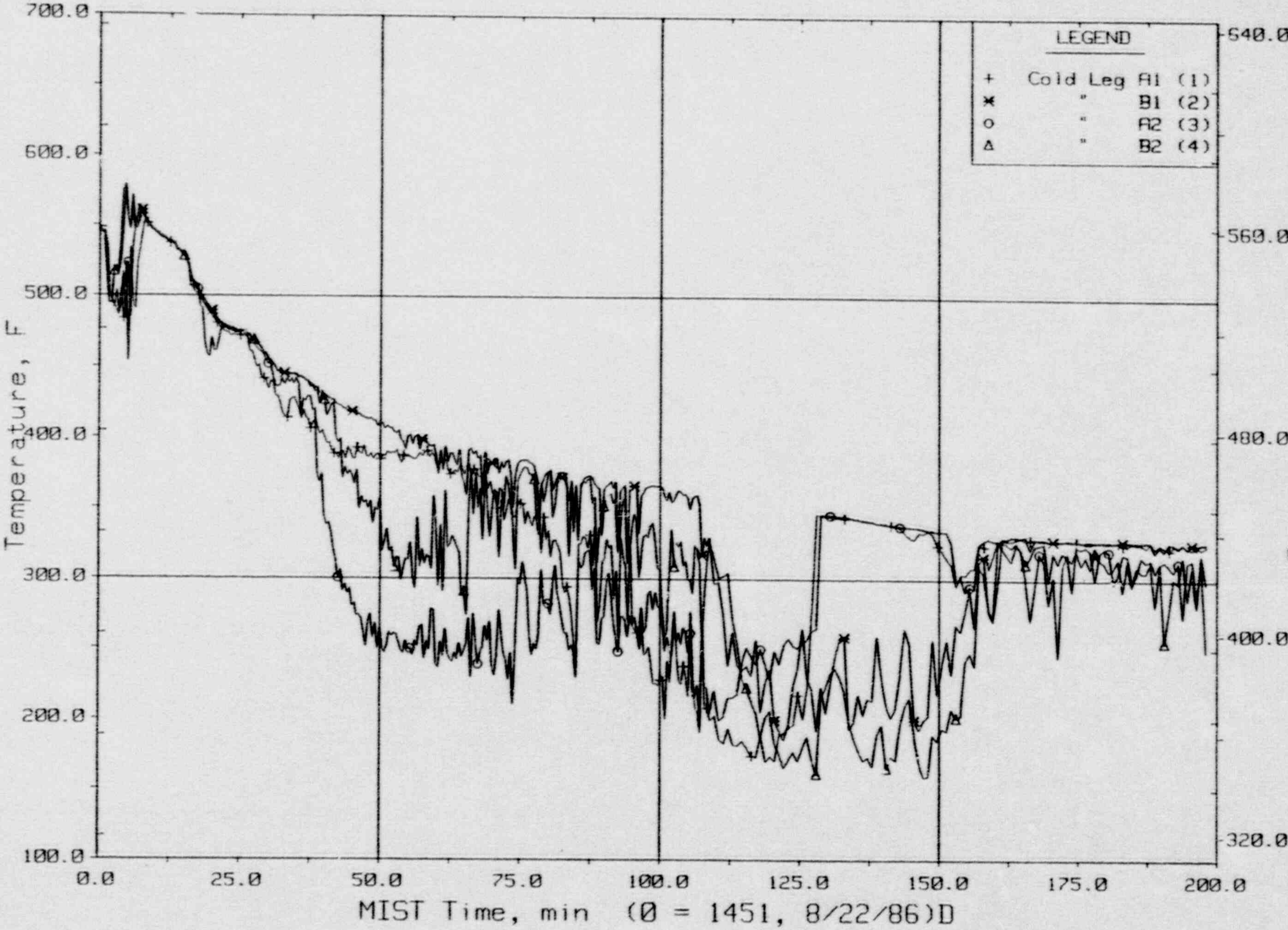
Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC11s).

Wed Jun 1 09:04:17 1988

CLTC1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



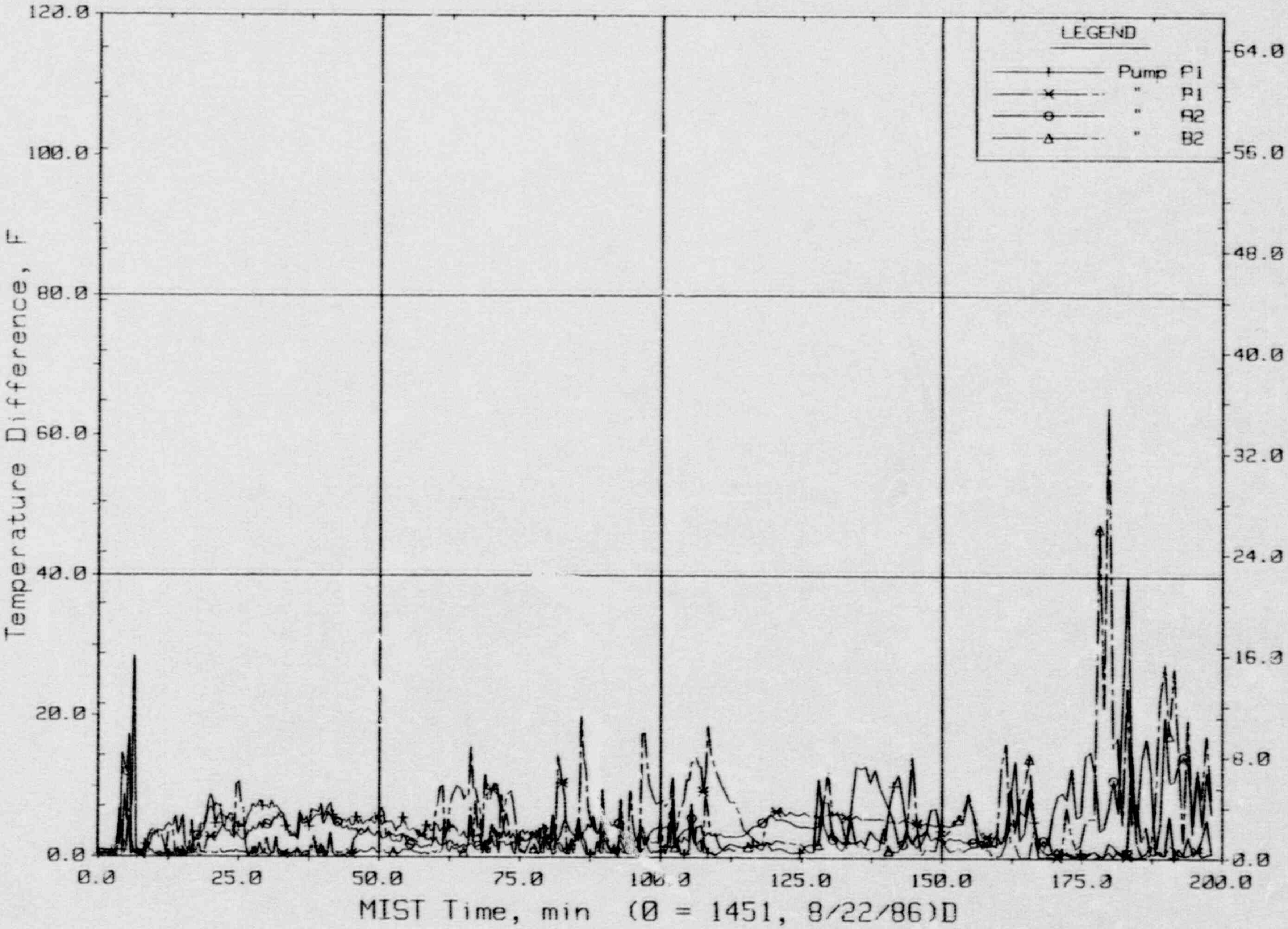
Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

Wed Jun 1 09:08:19 1988

CLTC2

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



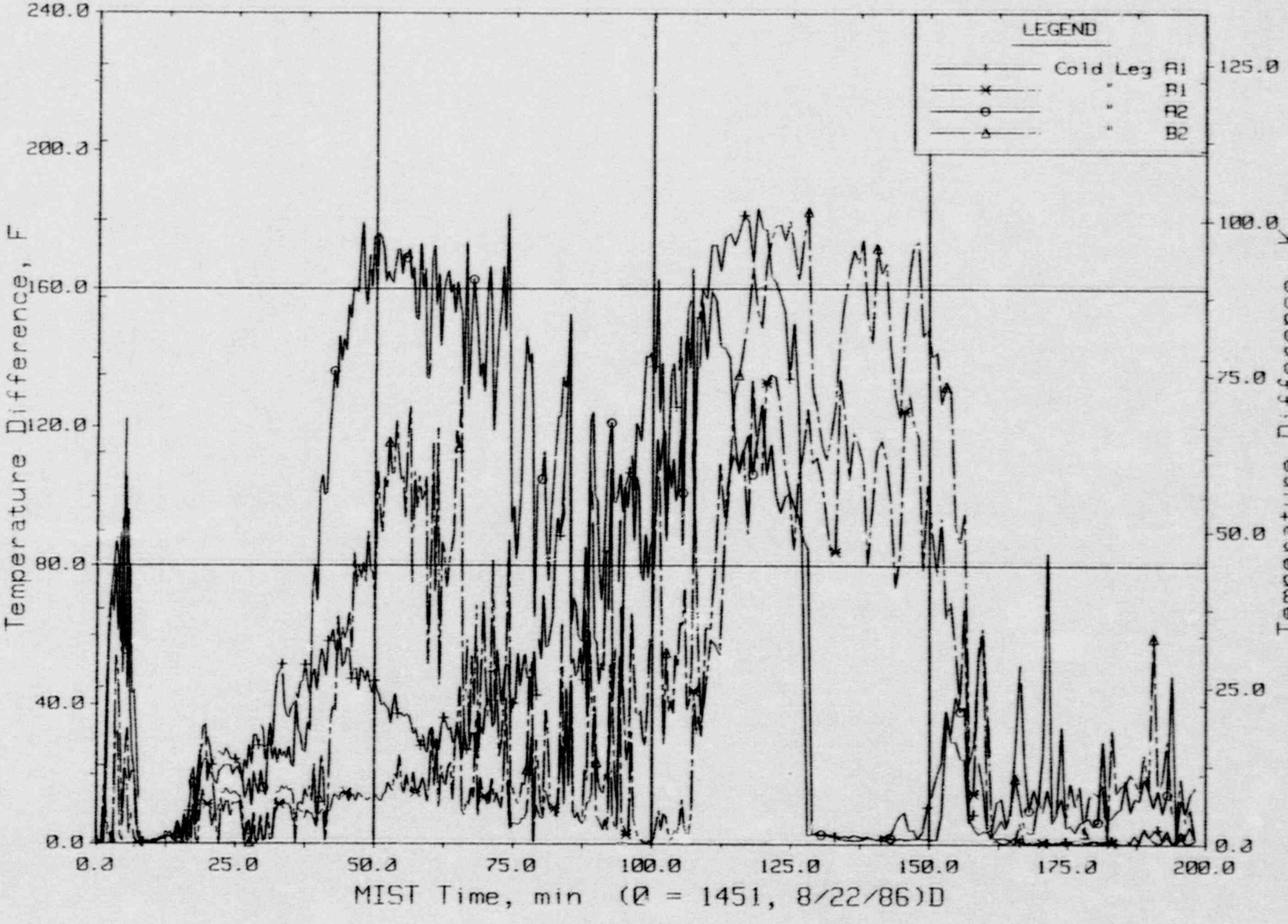
Maximum Differences Among RCP Rake FLuid Temperatures.

Wed Jun 1 09:11:41 1988

CLTD1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



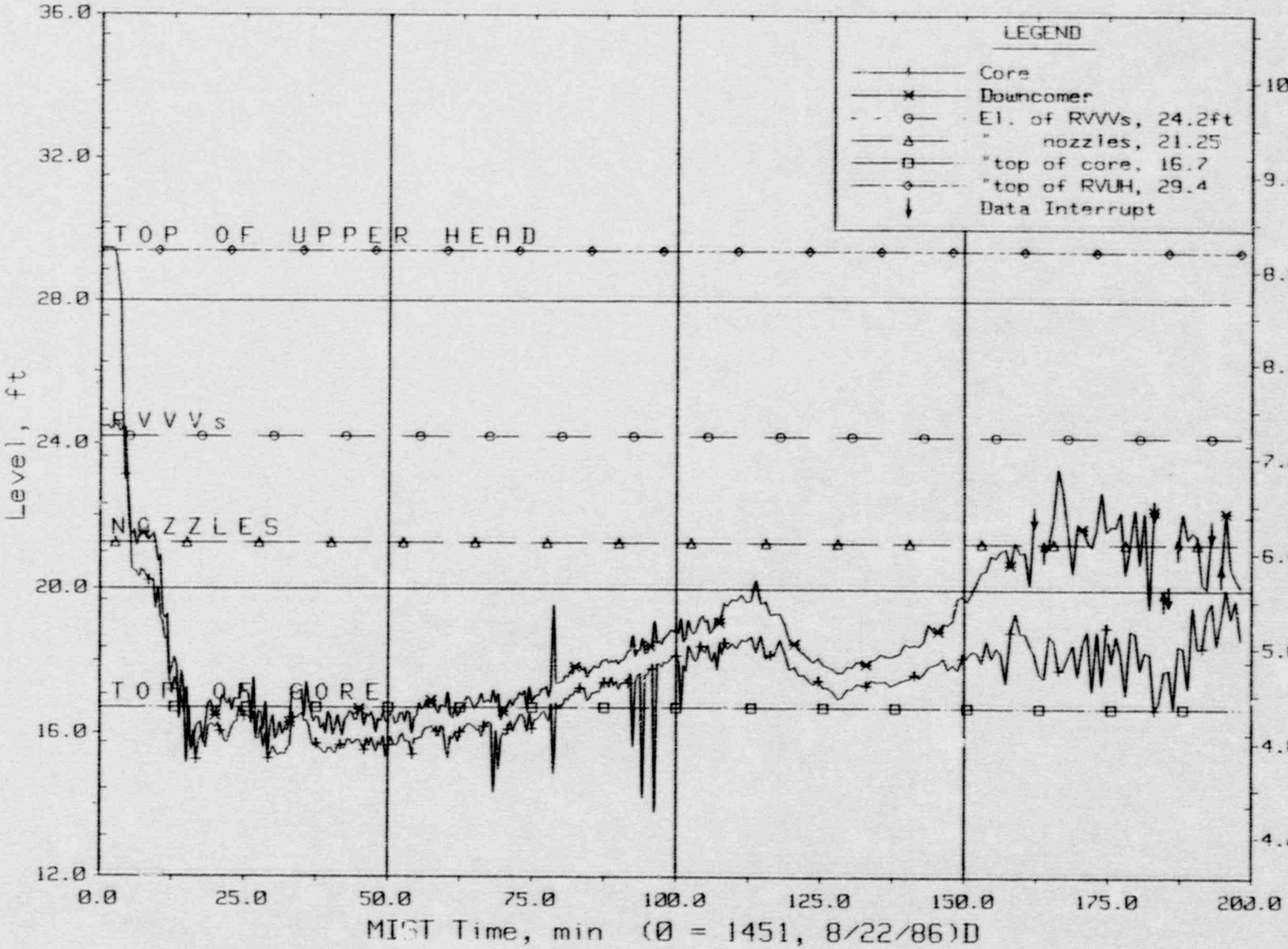
Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

Wed Jun 1 09:15:40 1988

CLTD2

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



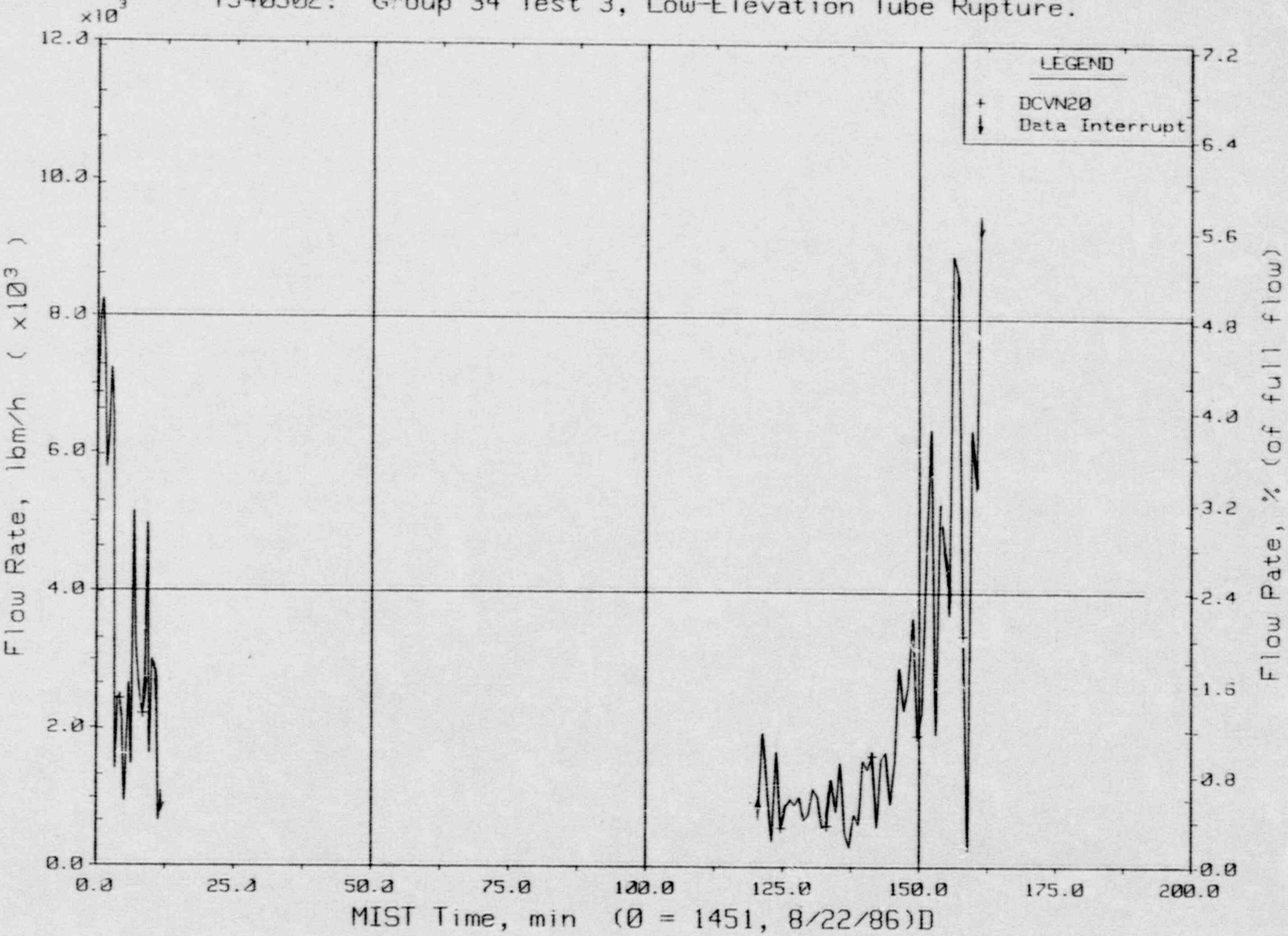
Core Region Collapsed Liquid Levels.

Wed Jun 1 09:19:48 1988

COLV1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



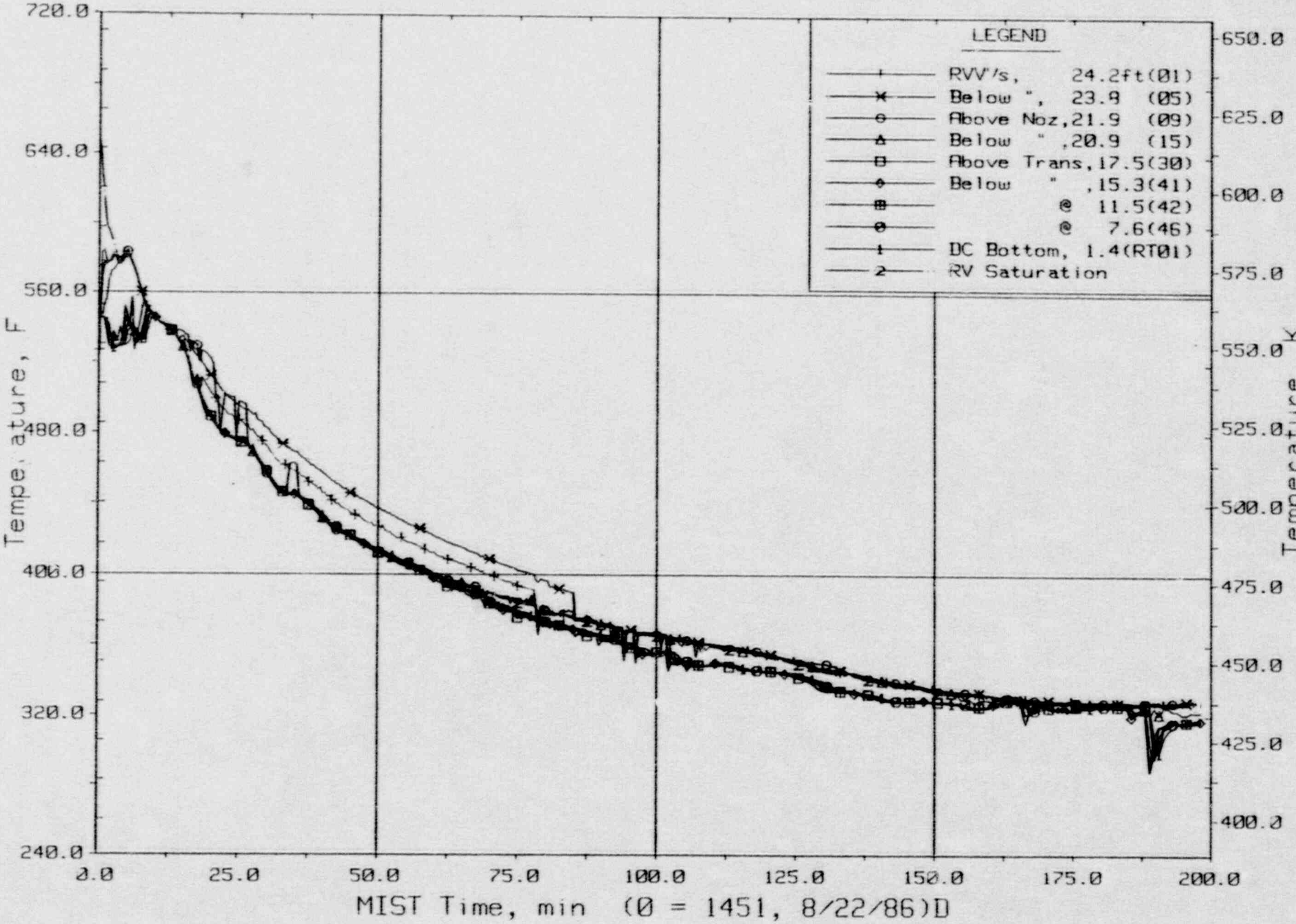
Downcomer (Venturi) Flow Rate.

Wed Jun 1 09:24:14 1988

DCFL1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



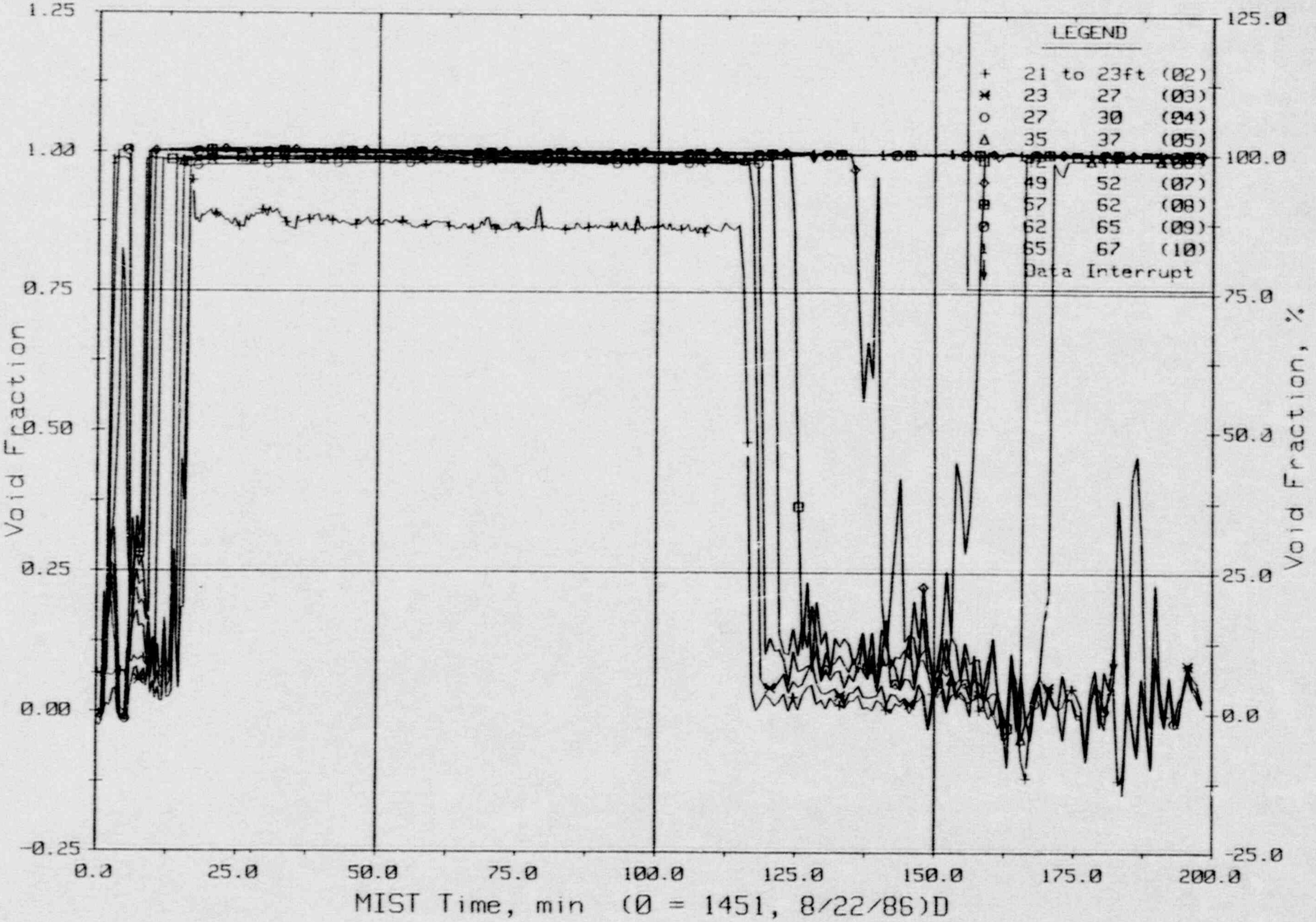
Downcomer Quadrant A1 Fluid Temperatures (DCTCs).

Wed Jun 1 09:28:49 1988

DCTC6

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



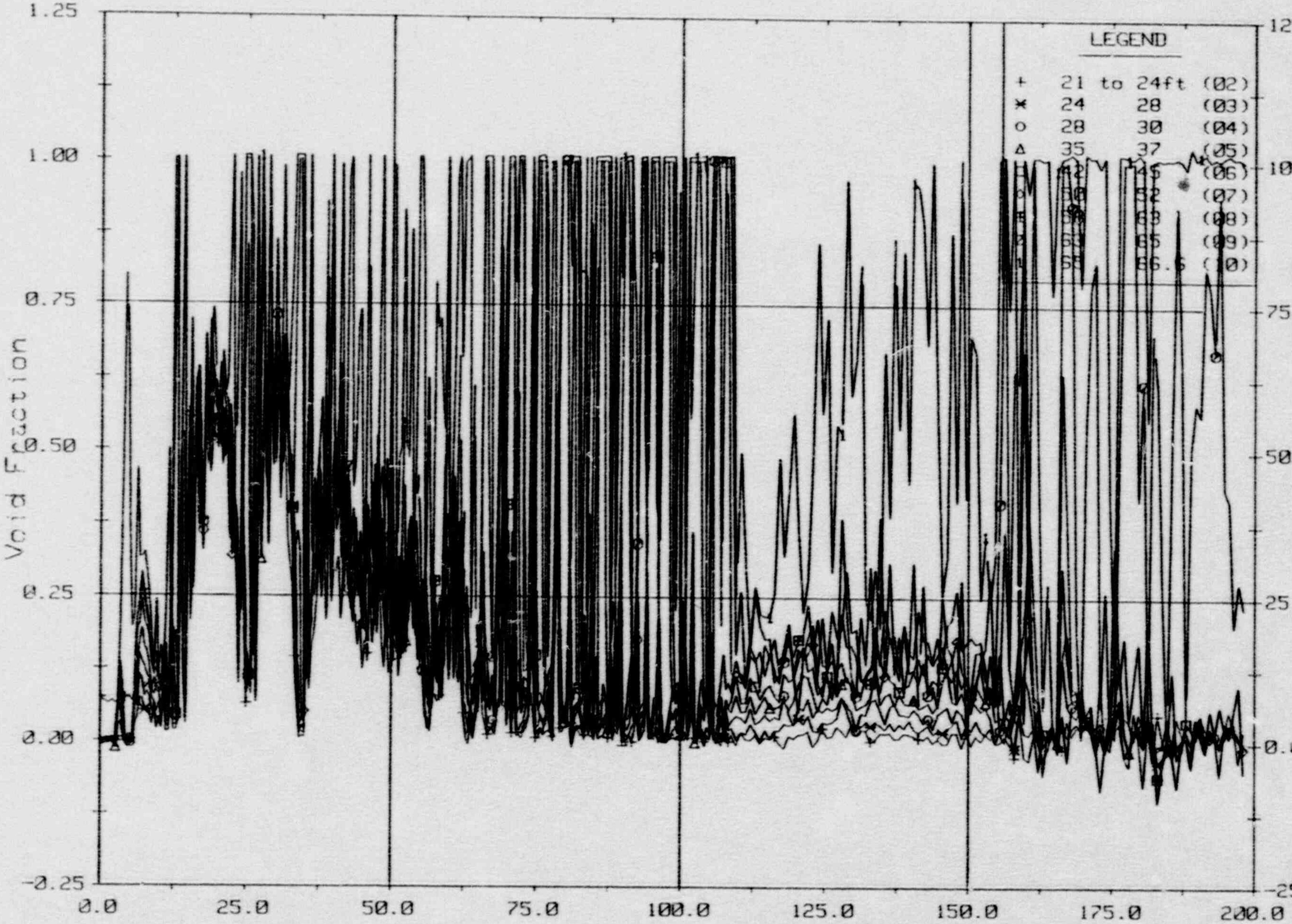
Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

Wed Jun 1 09:33:47 1988

H1VF1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



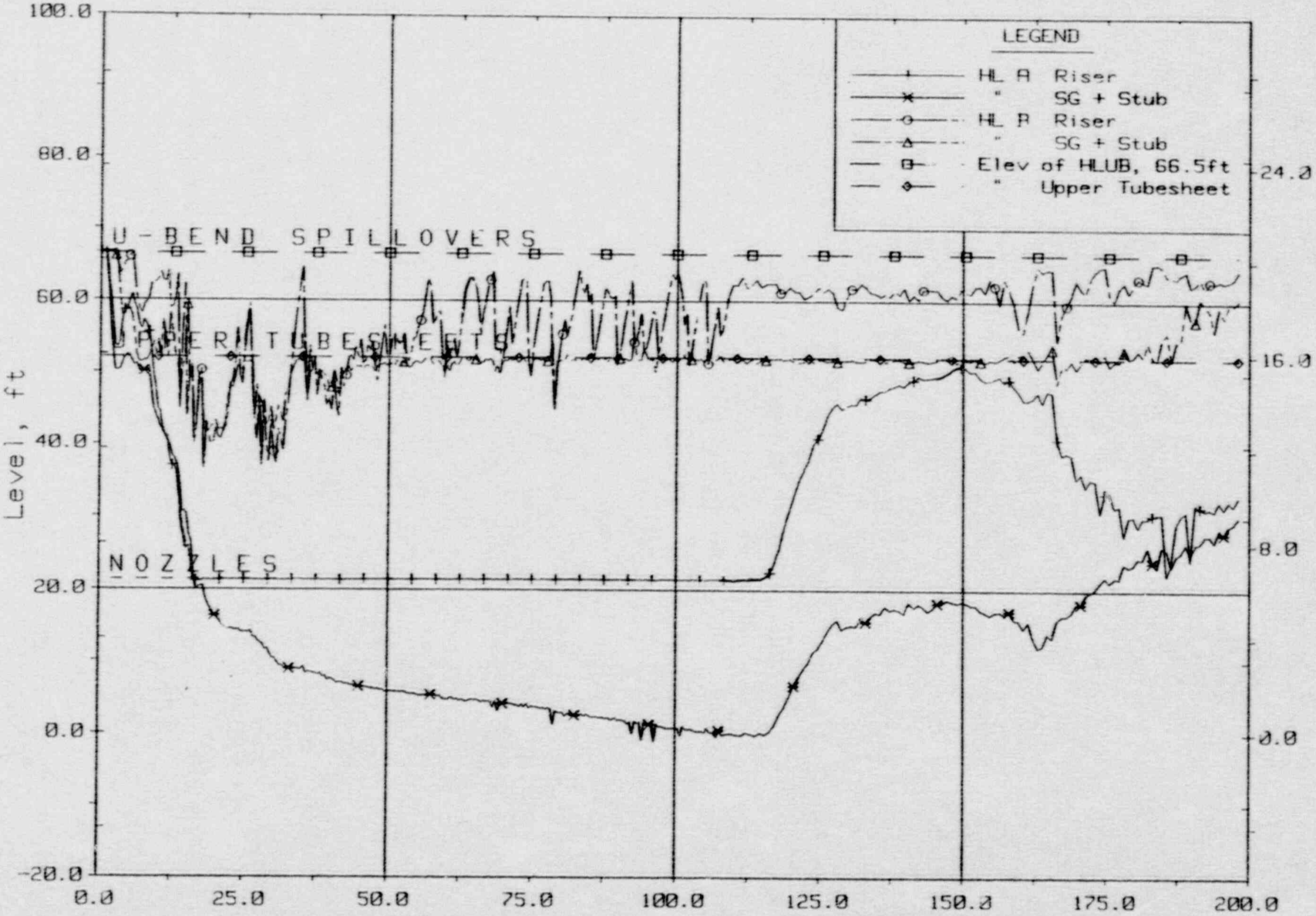
Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

Wed Jun 1 09:39:11 1988

H2VF1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.

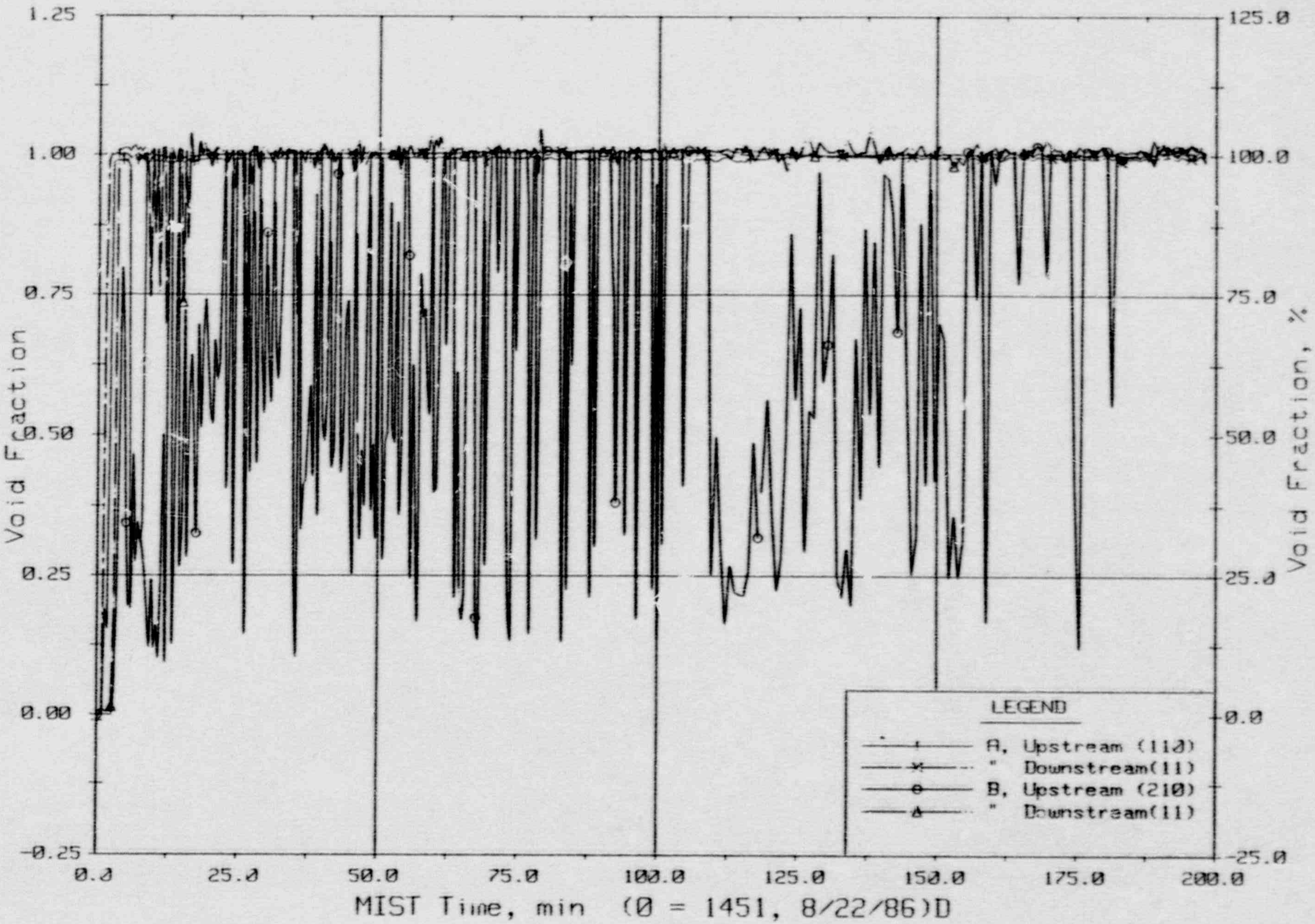


Hot Leg Riser and Stub Collapsed Liquid Levels.

Wed Jun 1 09:44:03 1988 HLLV1

FINAL DATA

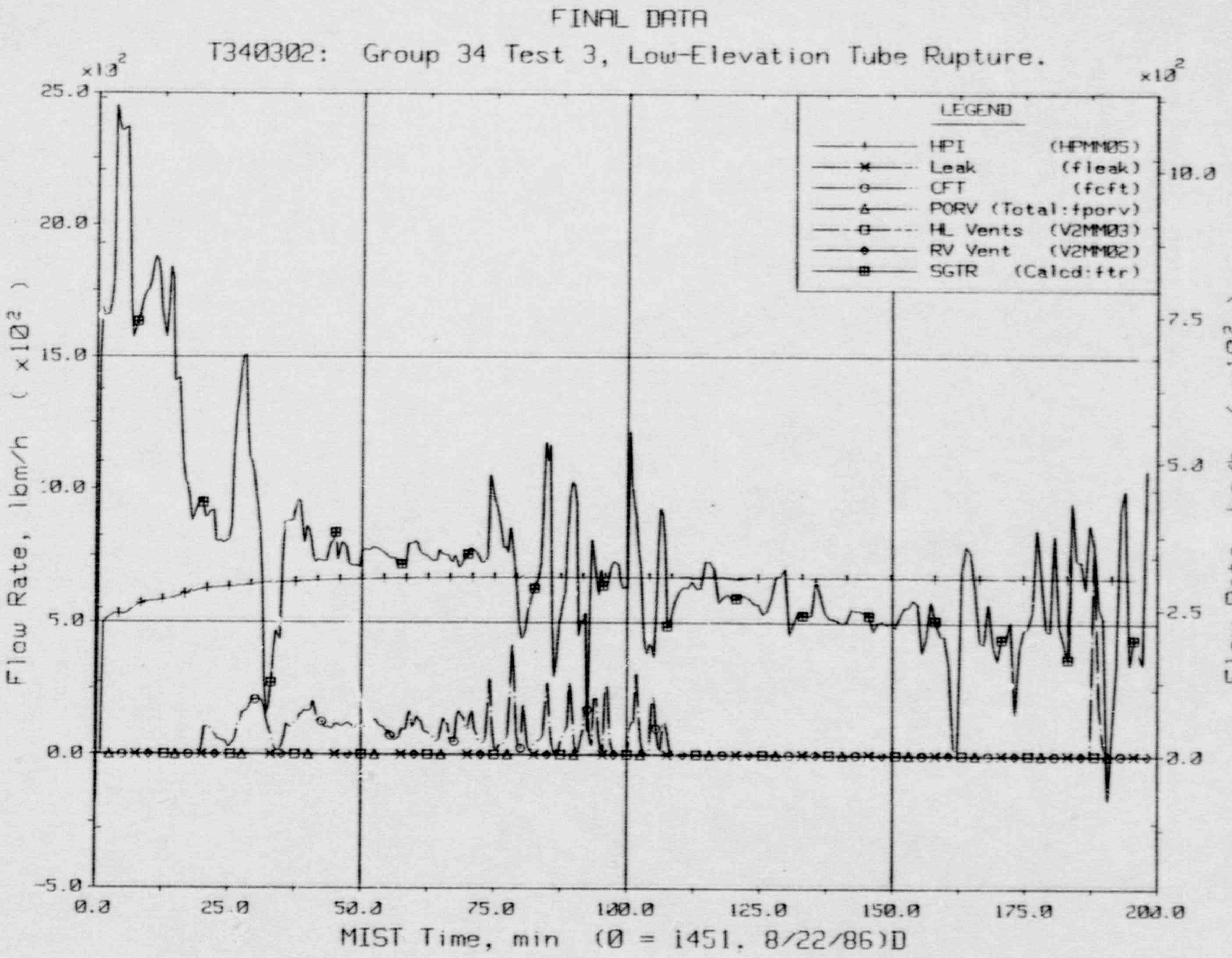
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Wed Jun 1 09:47:58 1988

HLVF1



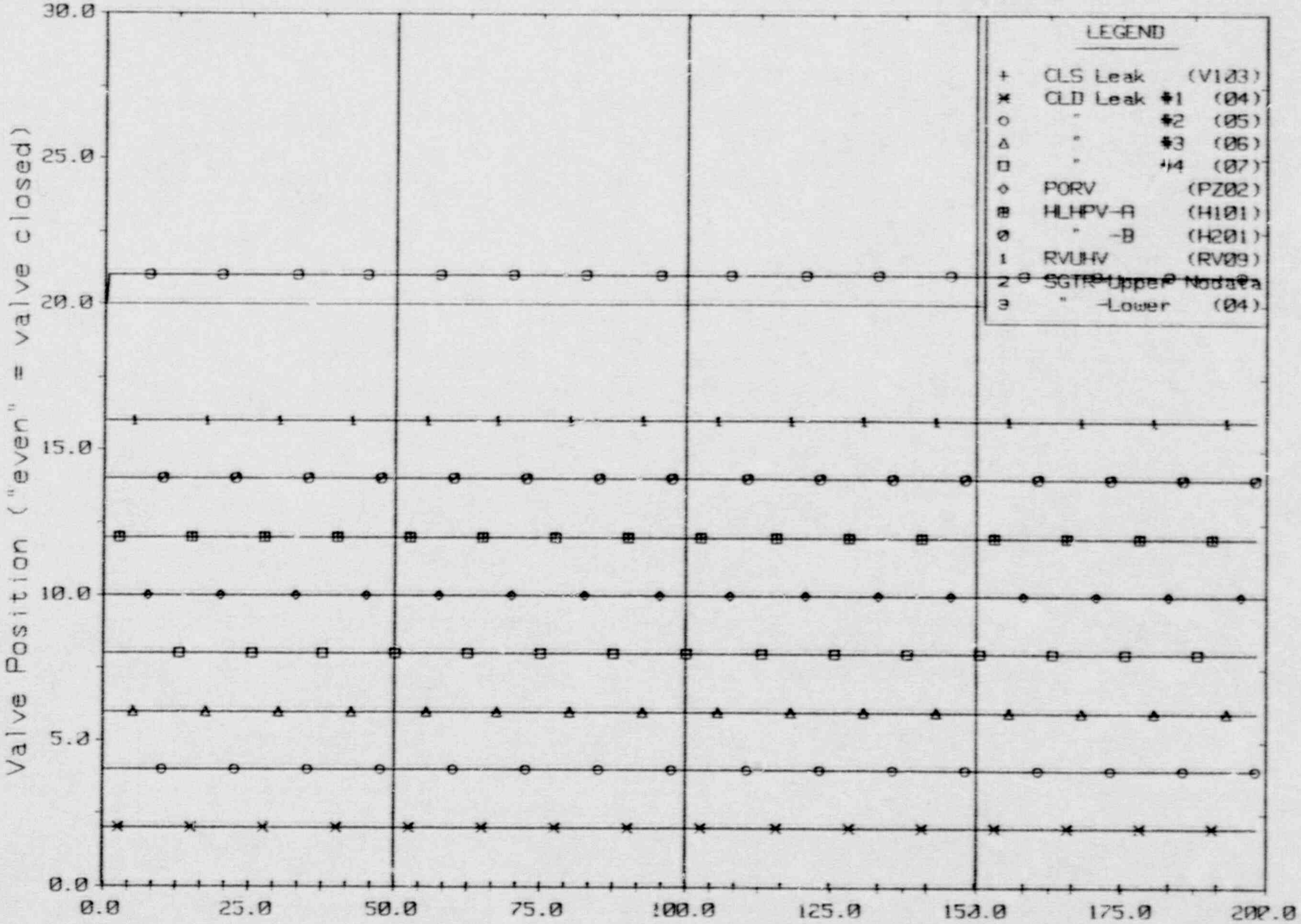
Primary System Boundary Flow Rates.

Wed Jun 1 09:51:24 1988

PHFL3

FINAL DATA

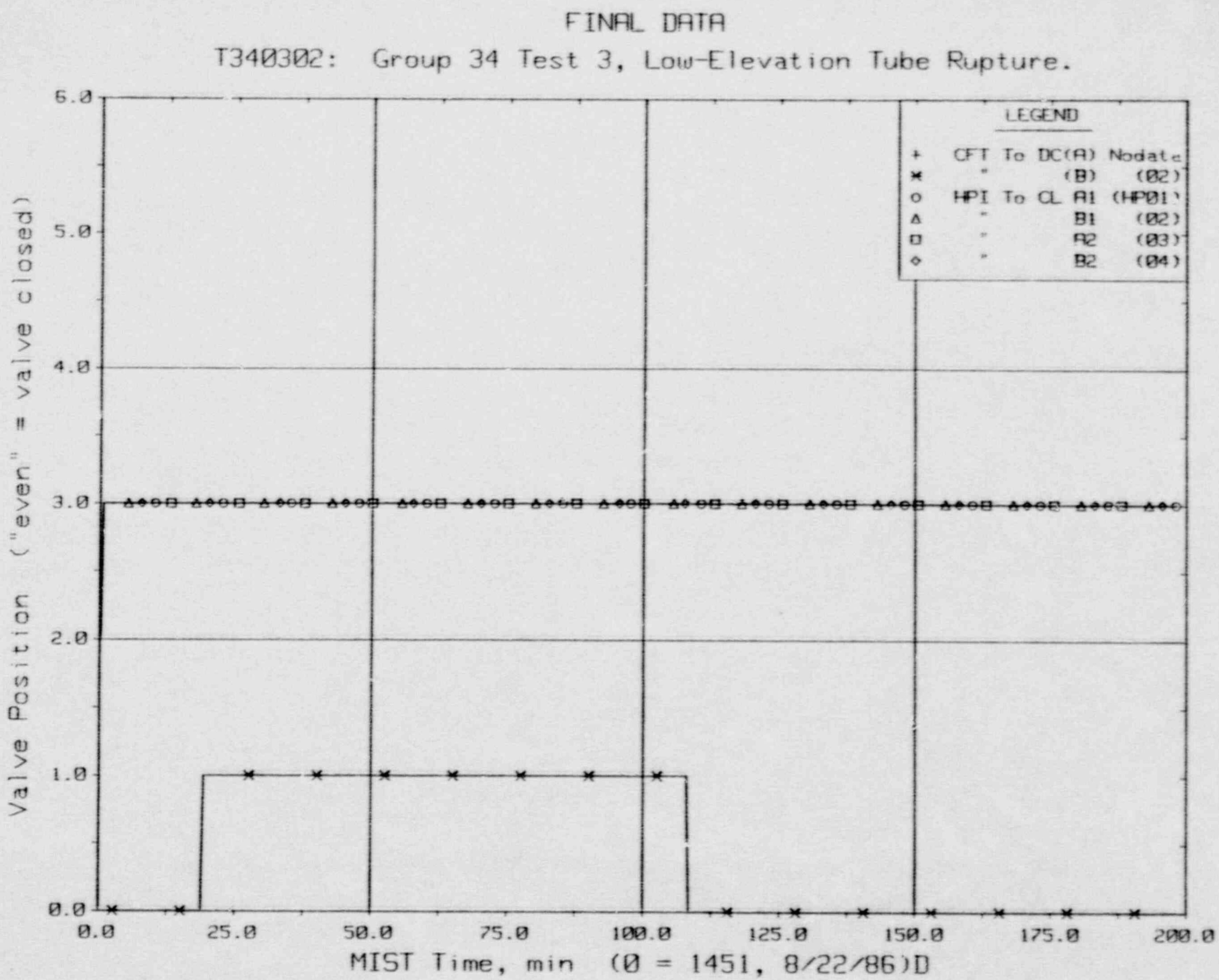
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Primary System Discharge Limit Switch Indications (LSs).

Wed Jun 1 09:56:35 1988

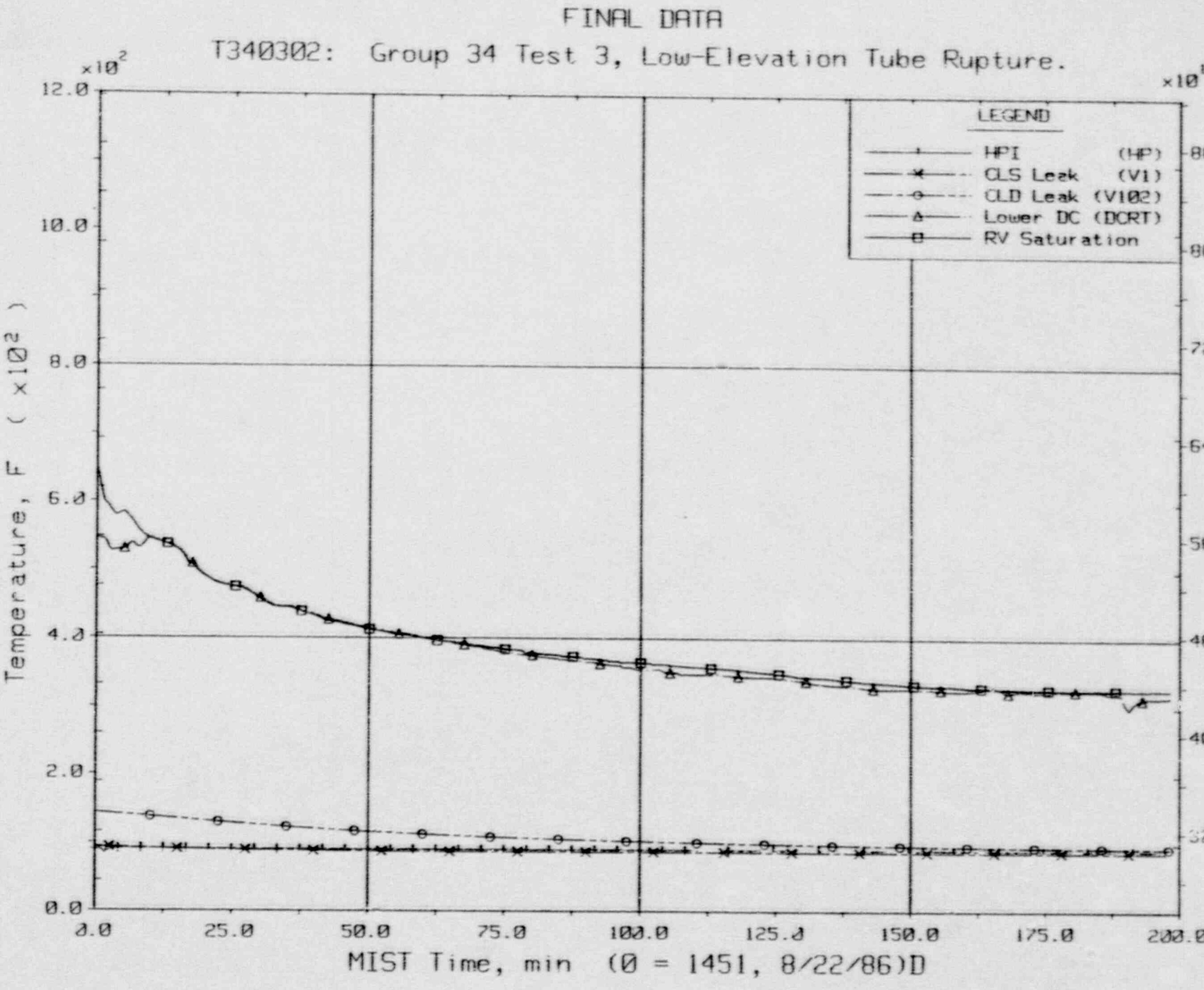
PRLS1



Primary System Injection Limit Switch Indications (LSs).

Wed Jun 1 09:59:49 1988

PBLS2



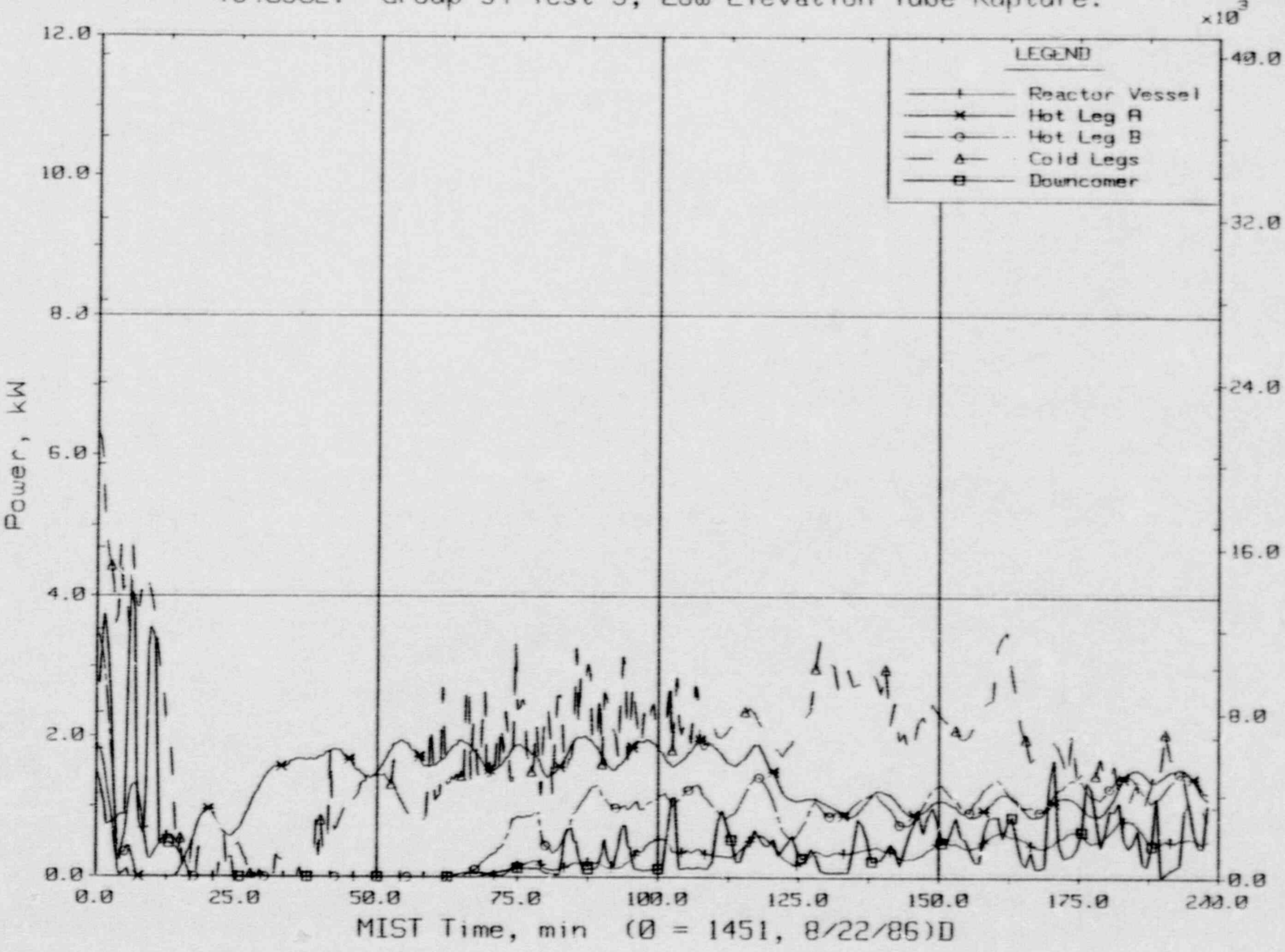
Single-Phase Discharge and HPI Fluid Temperatures (TC01s).

Wed Jun 1 10:03:24 1988

PRTC1p

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



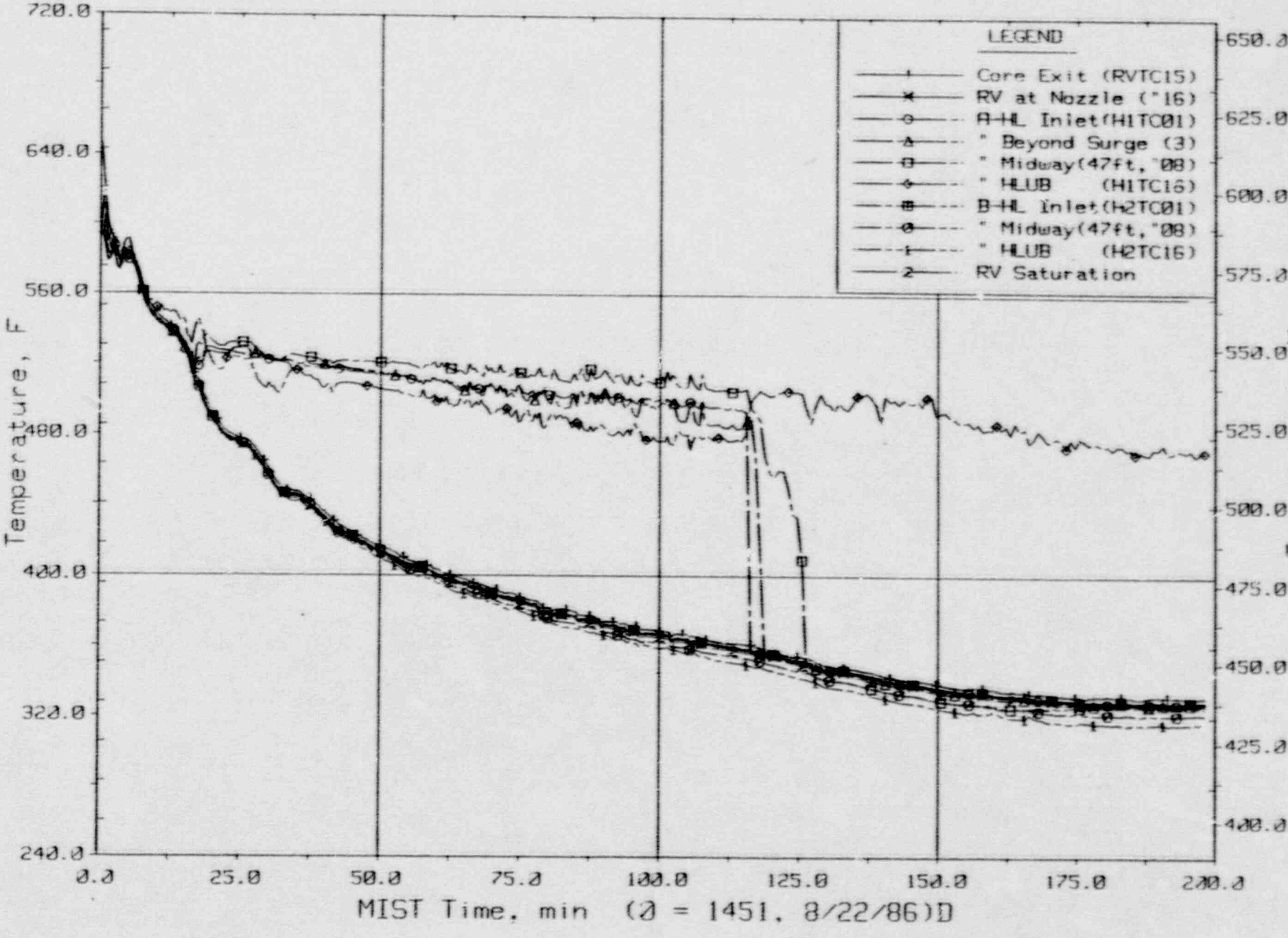
Guard Heater Specified Power Per Primary Component.

Wed Jun 1 10:09:46 1988

PRG01

FINAL DATA

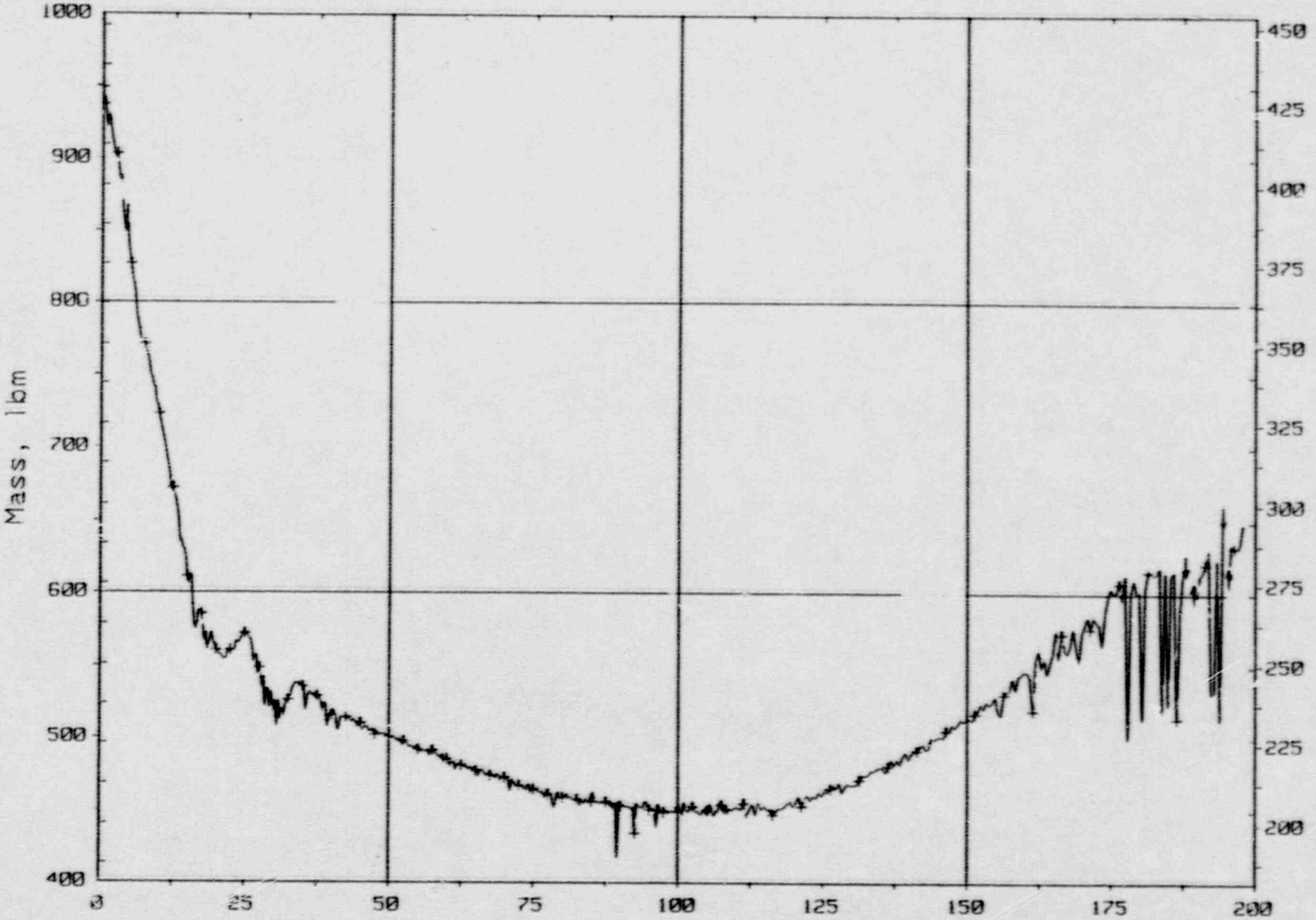
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Composite Core Exit and Hot Leg Fluid Temperatures.

FINAL DATA

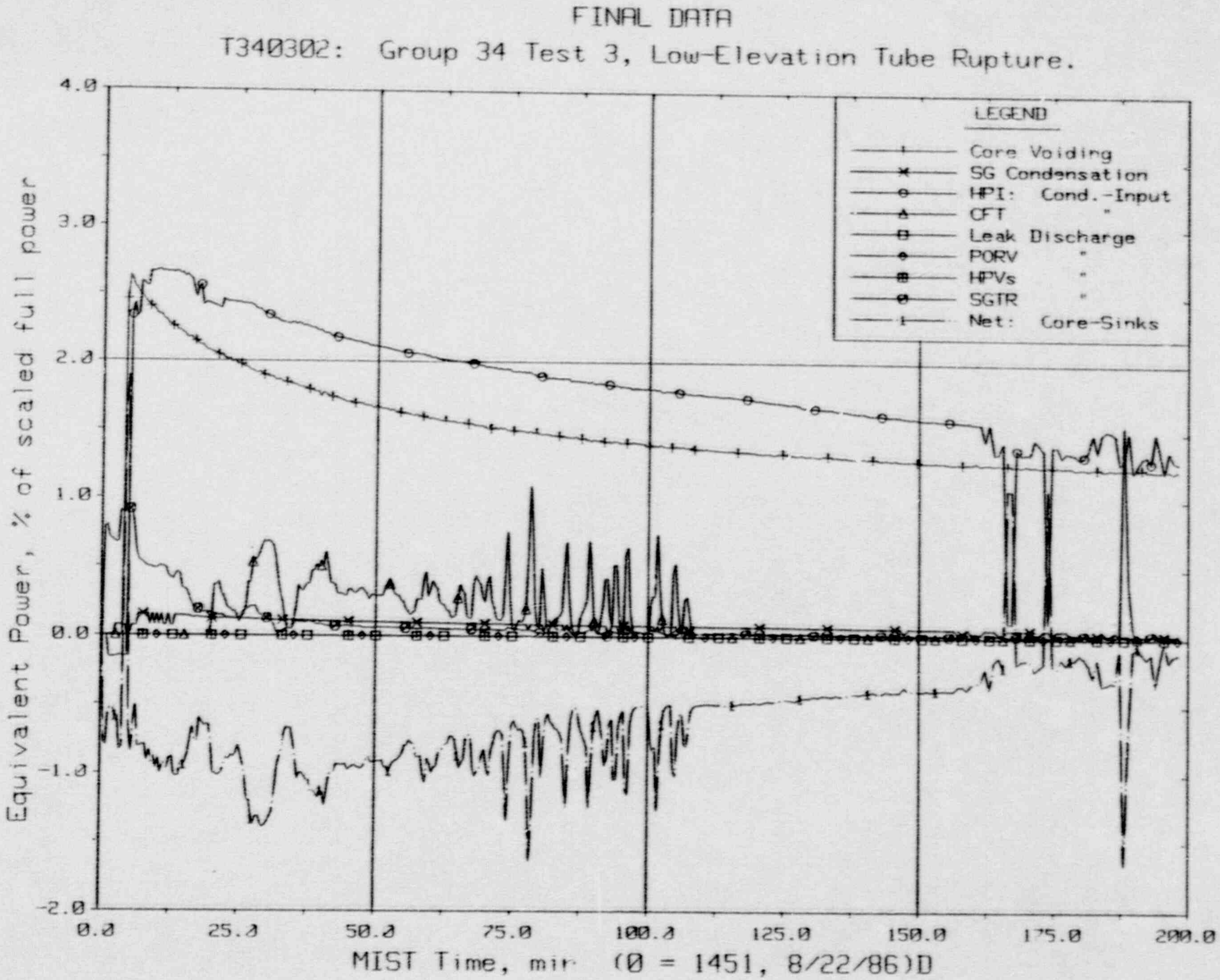
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Indicated Primary System Total Fluid Mass (PLML20).

Wed Nov 2 14:14:34 1988

PRML9



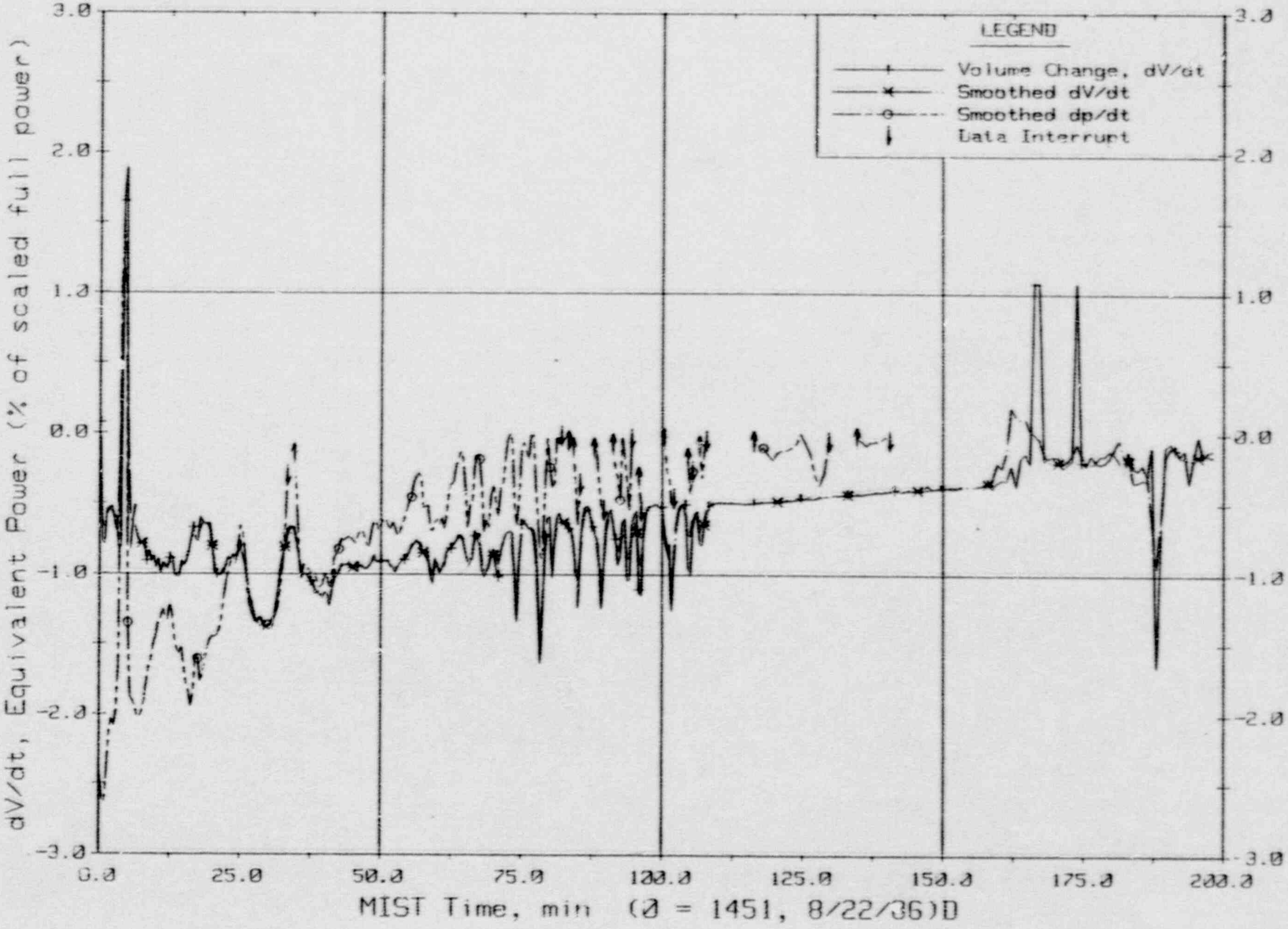
Primary Fluid Volume Changes By Components.

Wed Jun 1 10:21:25 1988

FRPP0

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



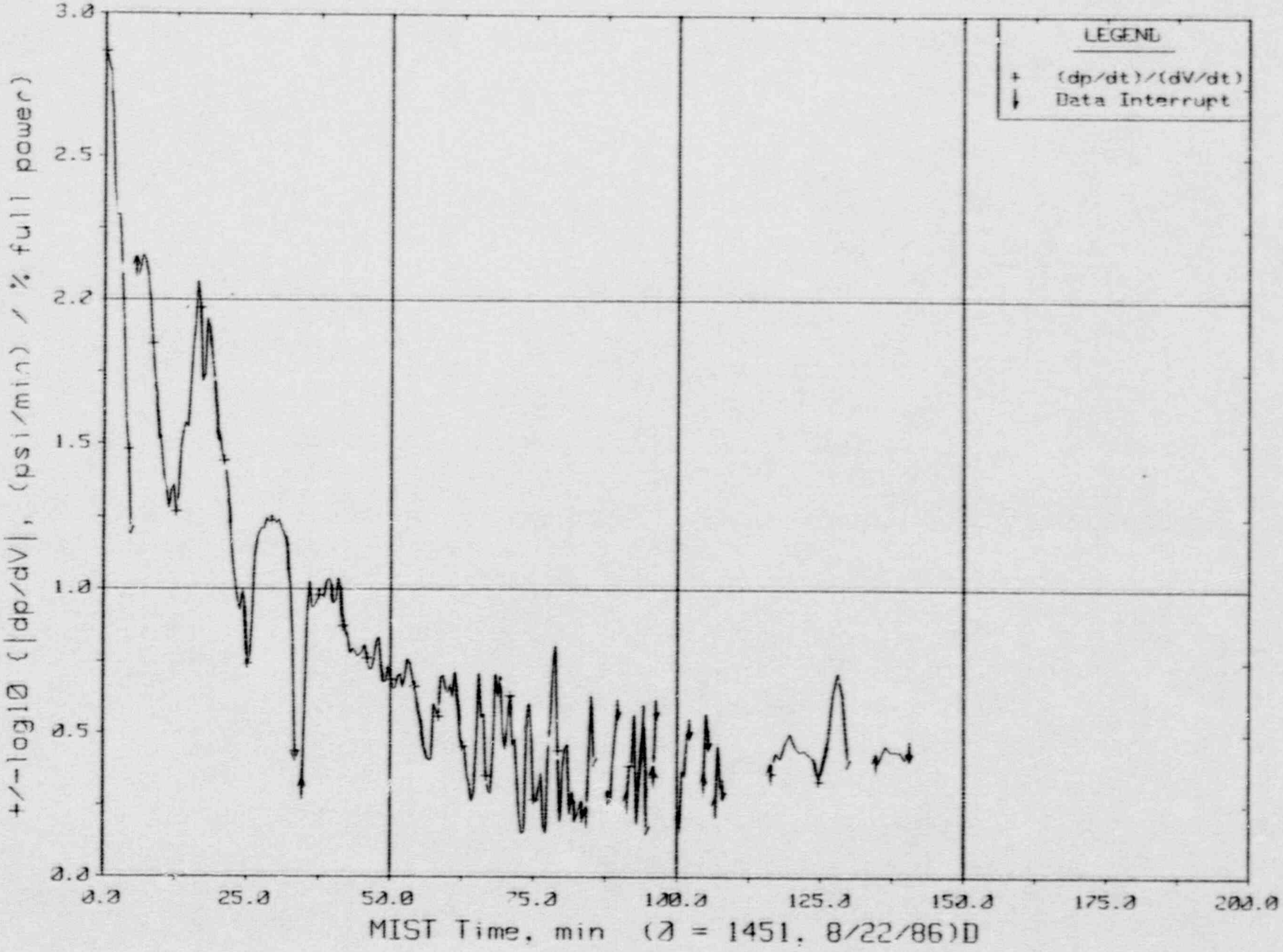
Primary Fluid Volume and Pressure Changes, dV/dt and dp/dt.

Wed Jun 1 10:25:17 1988

PRPA2

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



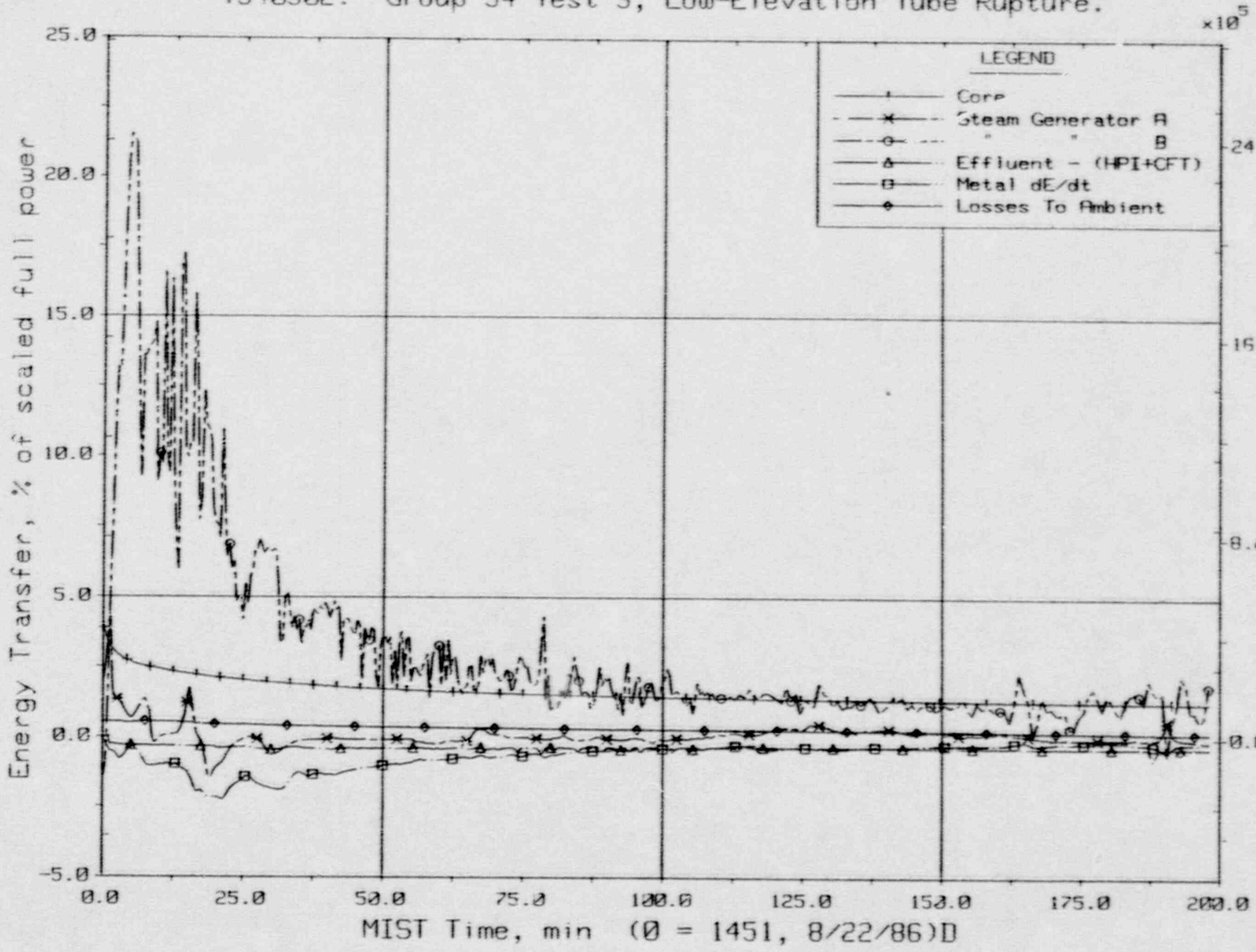
Primary System Response: dp/dt divided by dV/dt.

Wed Jun 1 10:27:48 1988

PRPA3

FINAL DATA

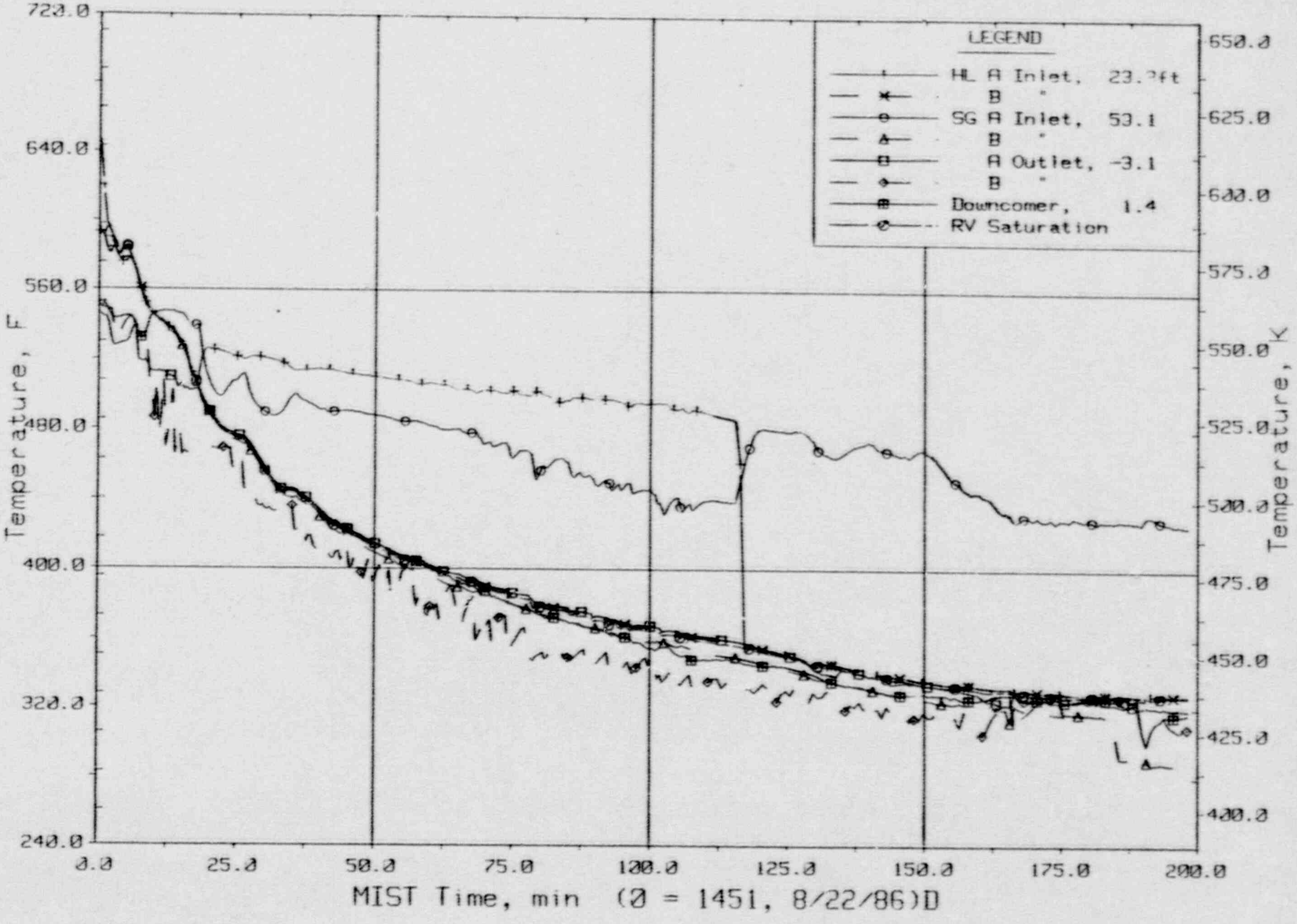
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Primary System Energy Transfer.

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



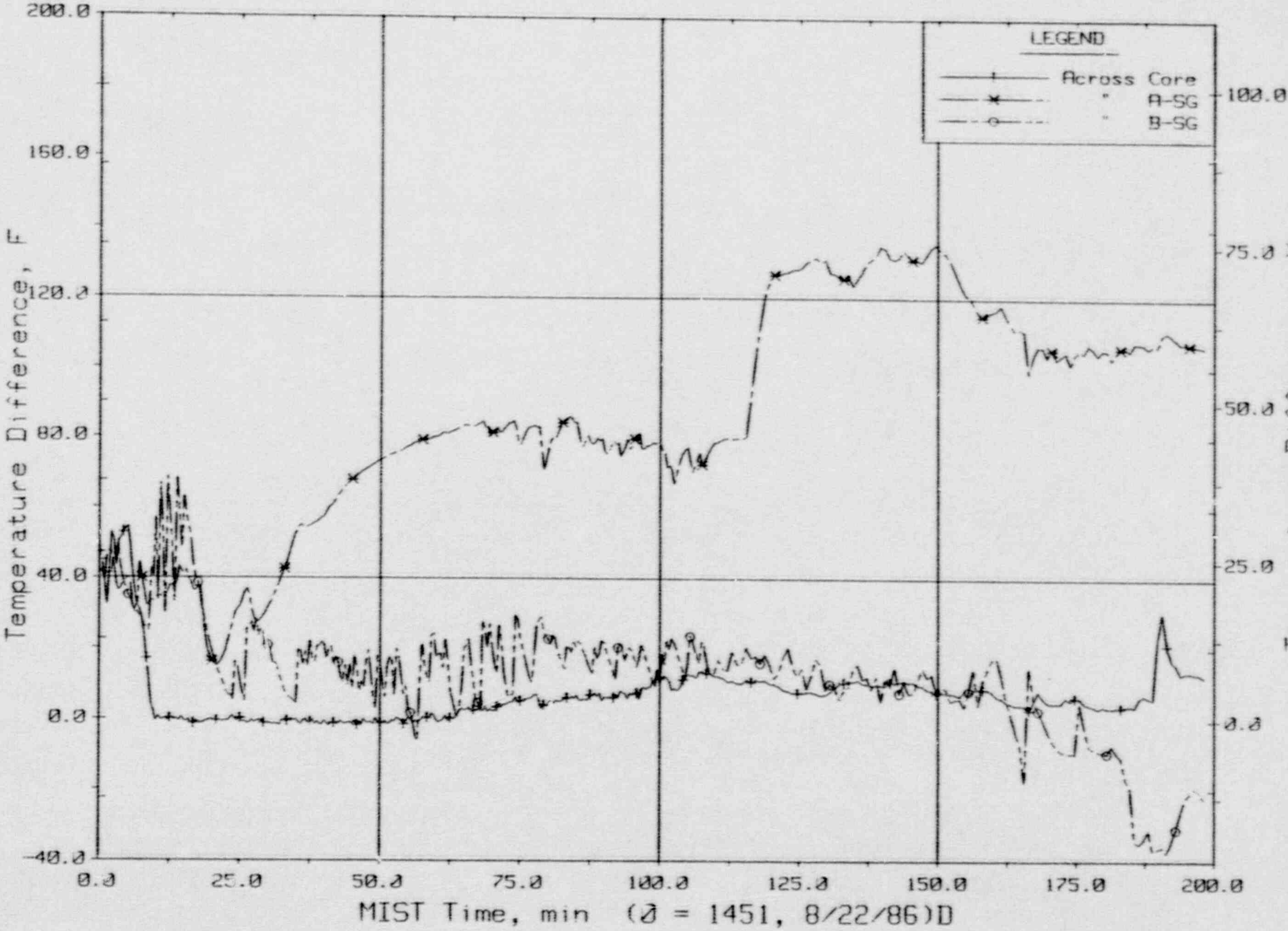
Primary System Fluid Temperatures (RTDs).

Wed Jun 1 10:35:42 1988

PRRT0

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



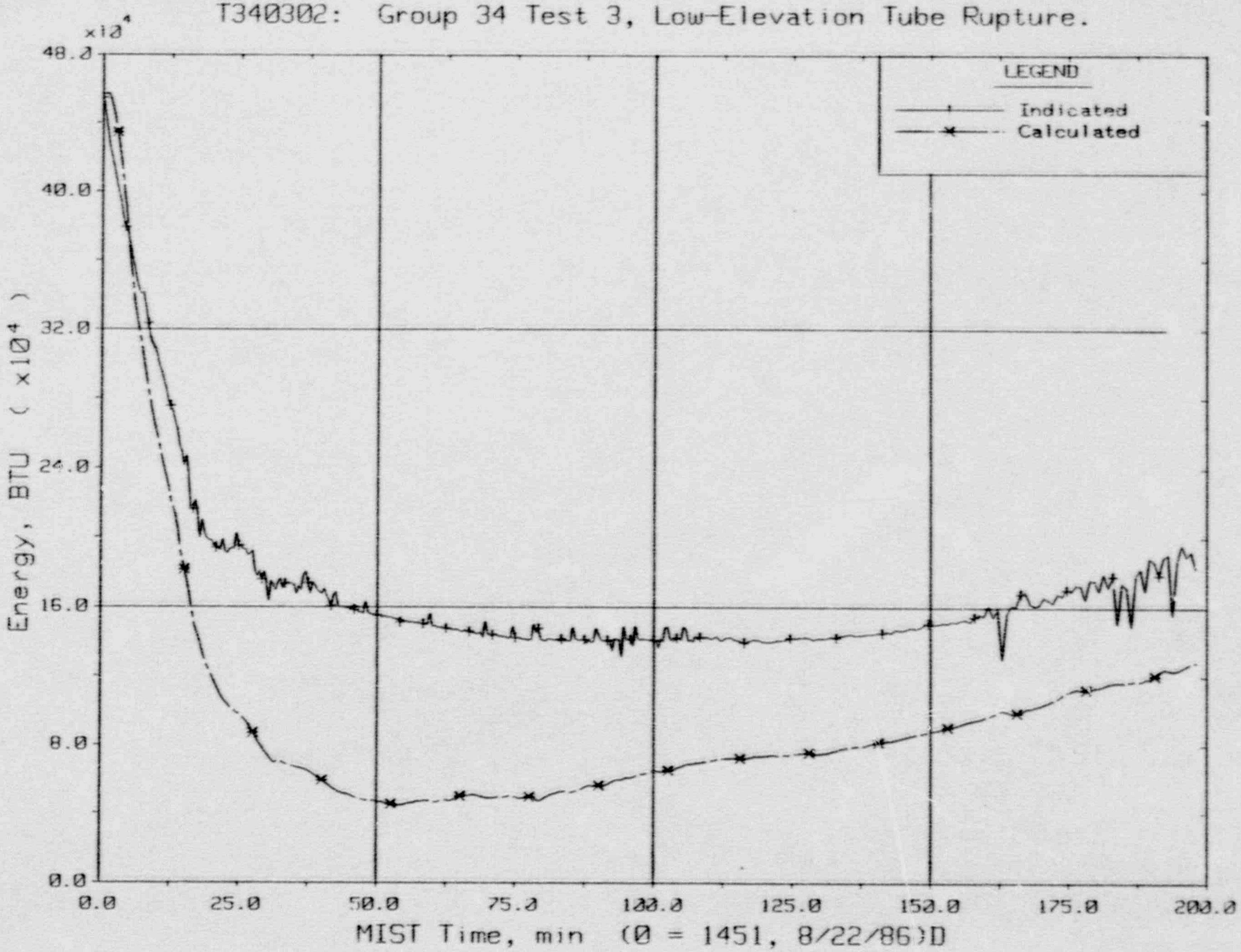
Key Temperature Differences.

Wed Jun 1 10:41:11 1988

PRTD2

FINAL DATA

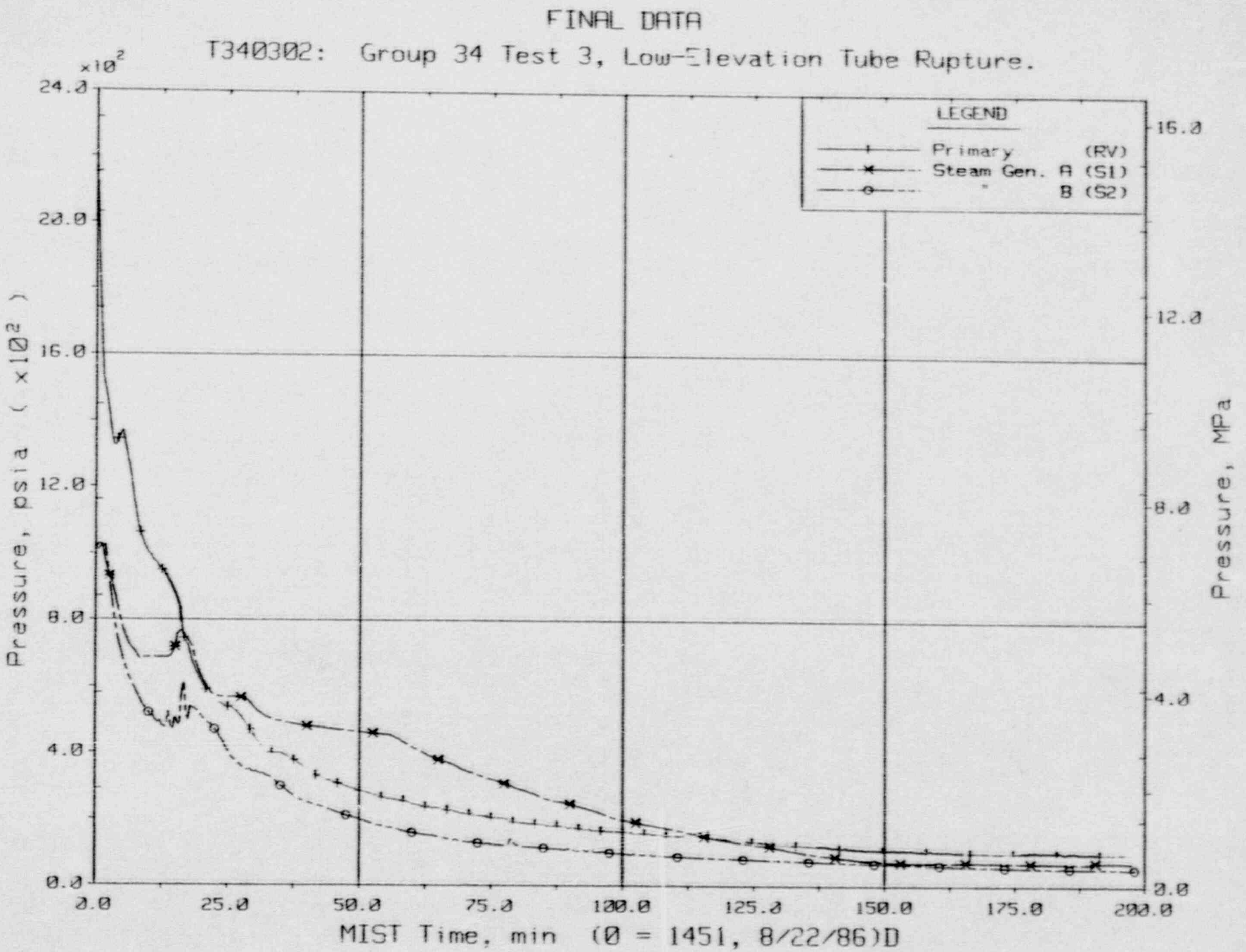
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Primary System Total Fluid Energy.

Wed Jun 1 10:43:58 1988

PRTE2



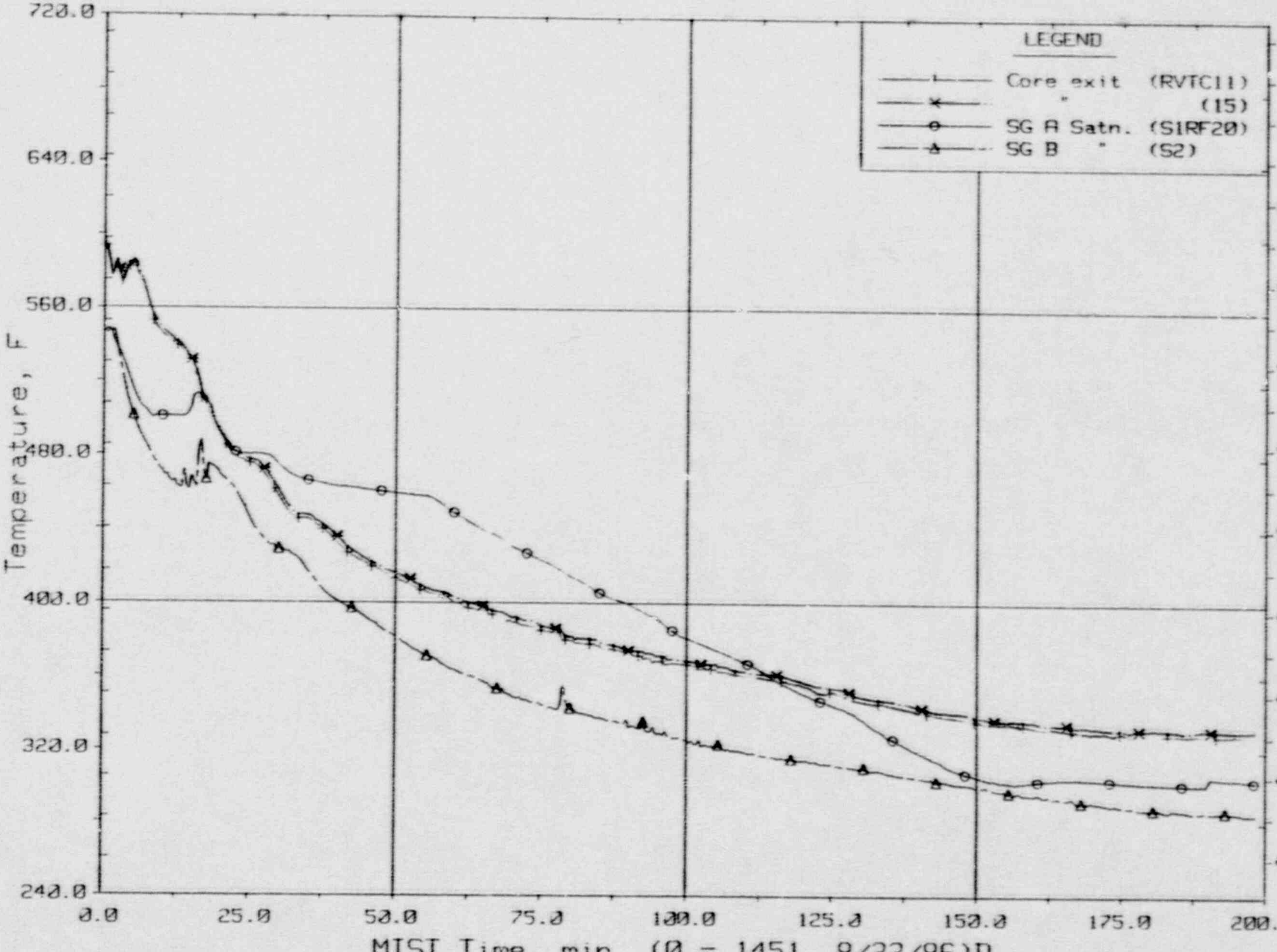
Primary and Secondary System Pressures (GP01s).

Wed Jun 1 10:47:03 1988

PSGP1

FINAL DATA

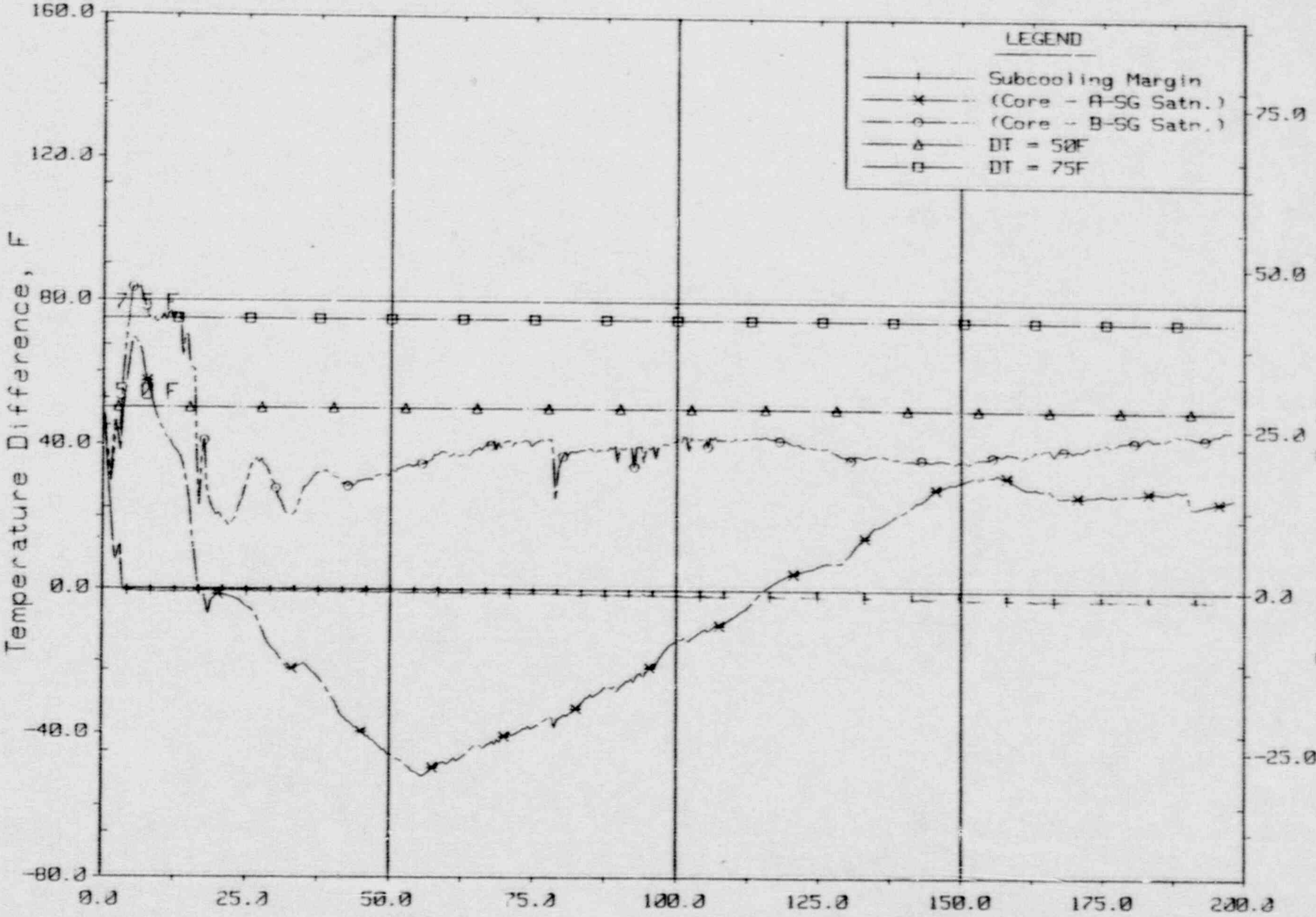
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



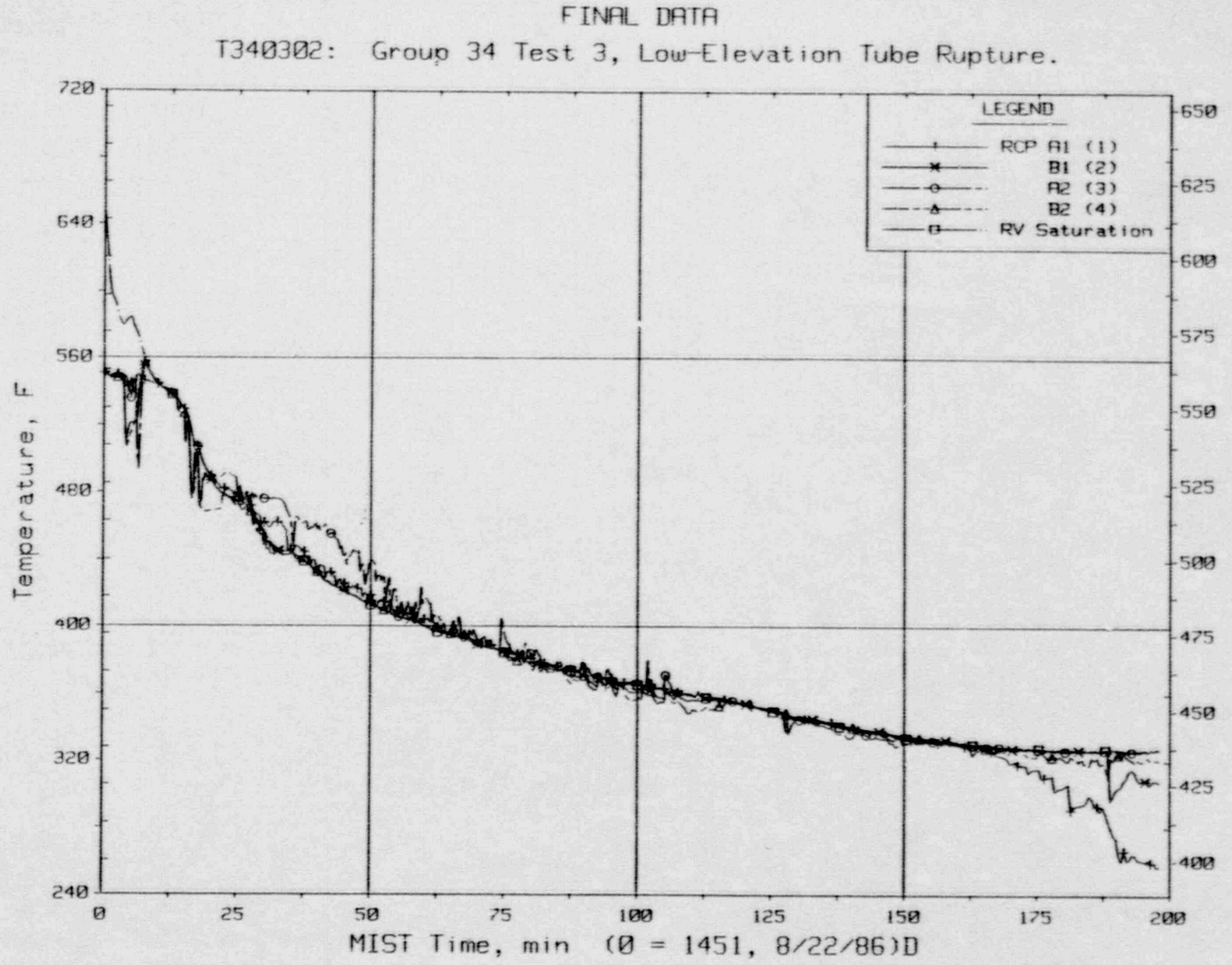
Core Exit and Steam Generator Secondary Saturation Temperatures.

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



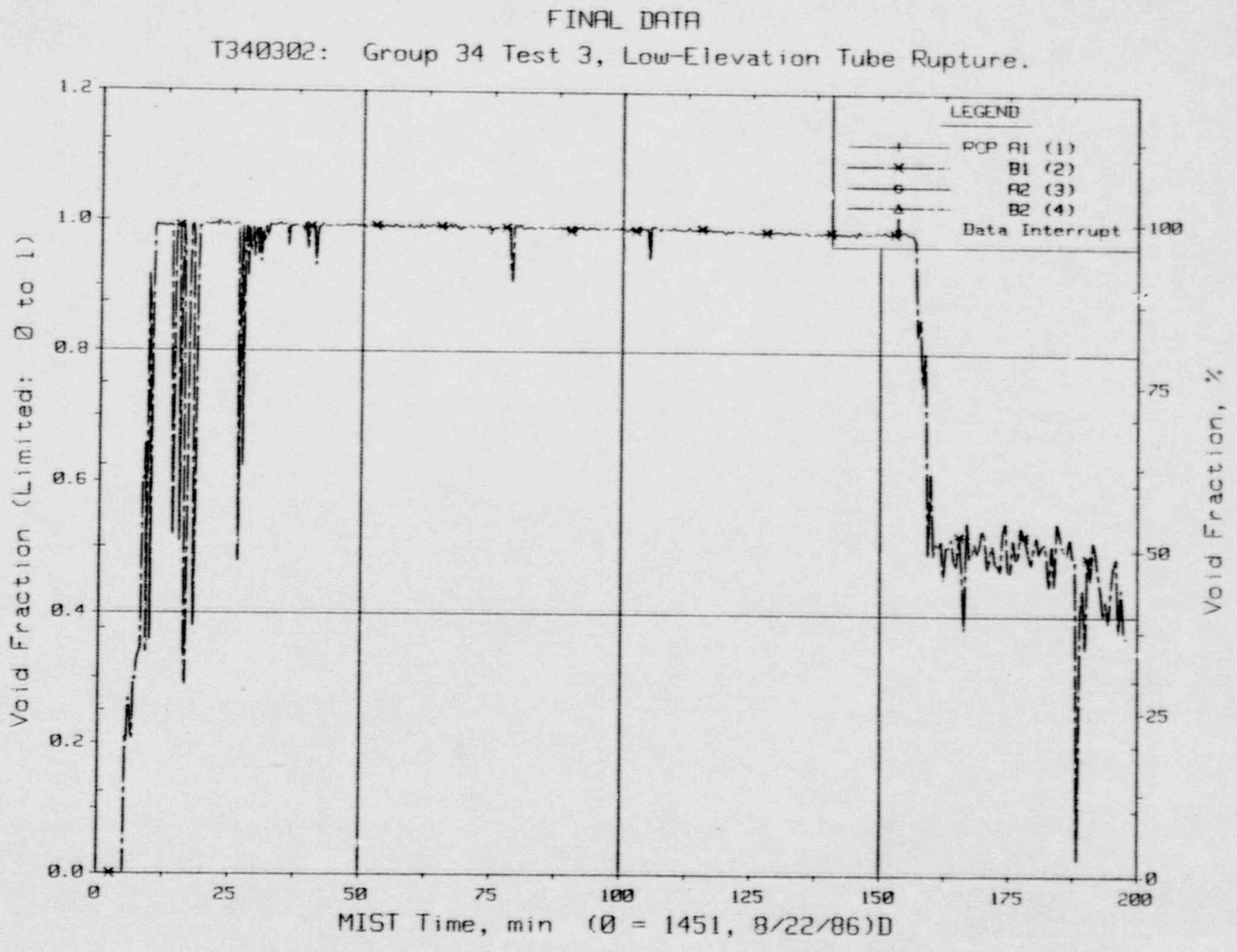
Control Temperature Differences.



Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 17:47:40 1988

PURT0



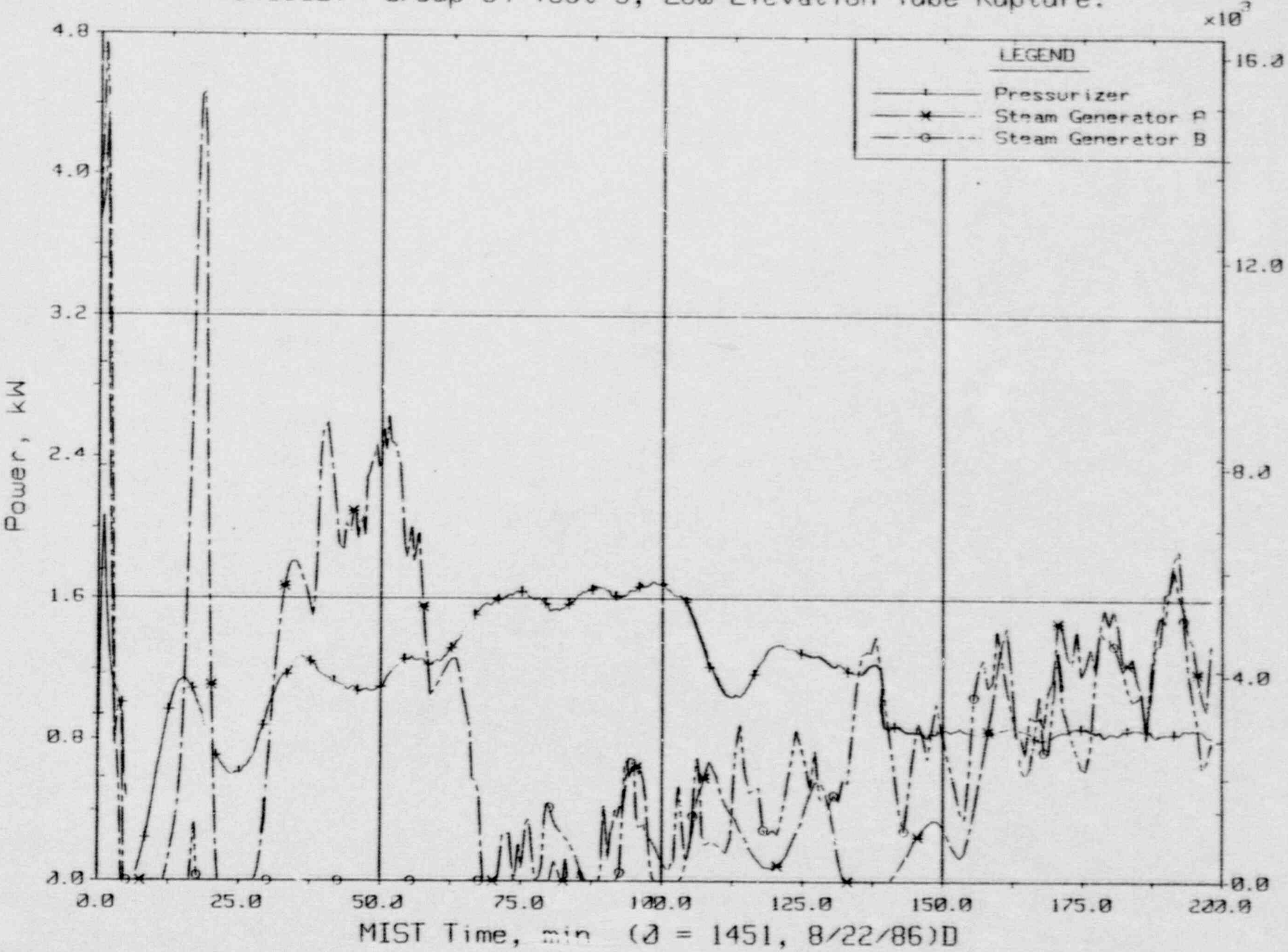
Pump Suction Void Fraction From Gamma Densitometers (CnGD21).

Tue Nov 1 17:50:43 1988

PUVF2

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



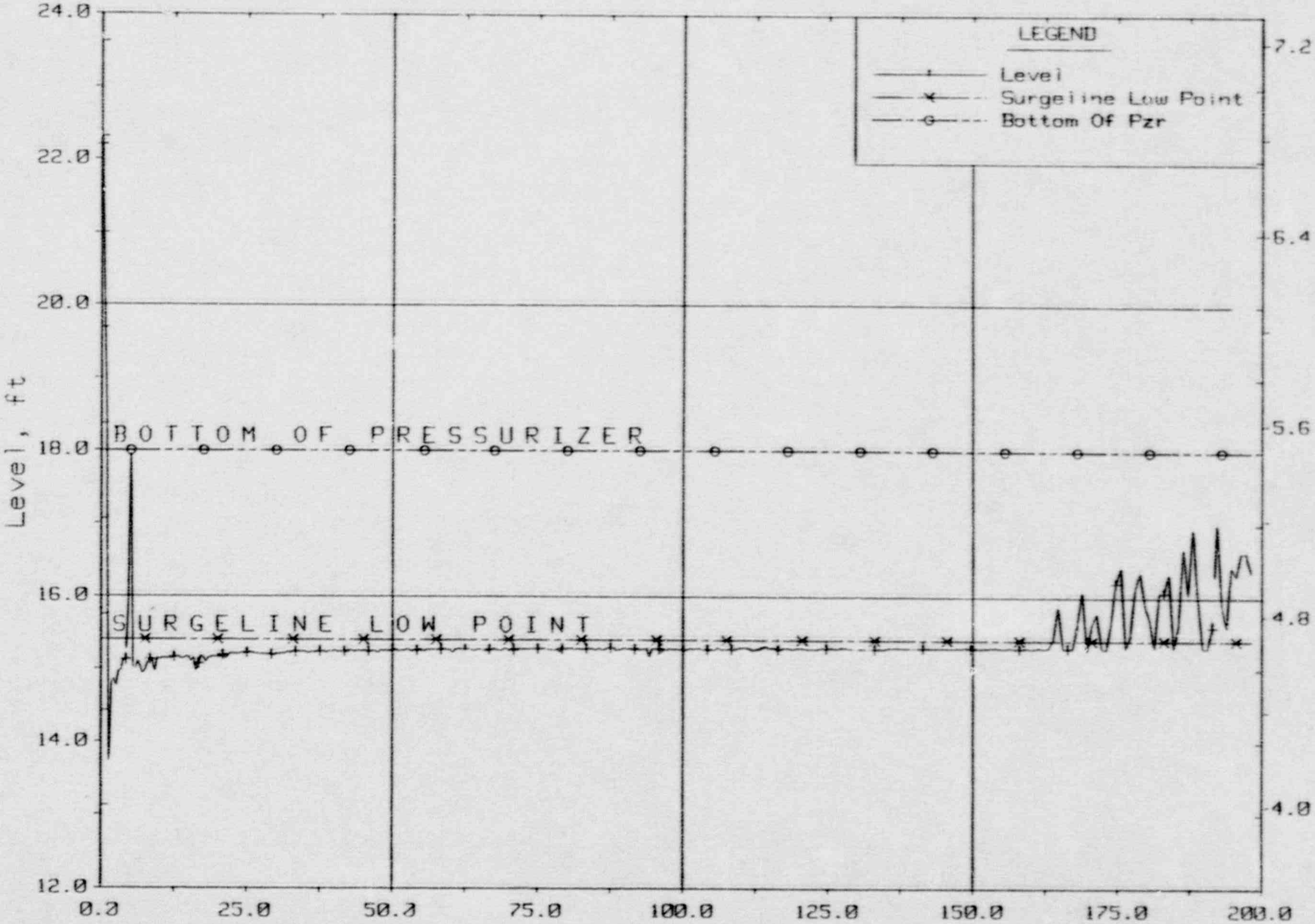
Guard Heater Specified Power, Pressurizer and Steam Generators.

Wed Jun 1 11:00:31 1988

PZG01

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



MIST Time, min (Ø = 1451, 8/22/86)D

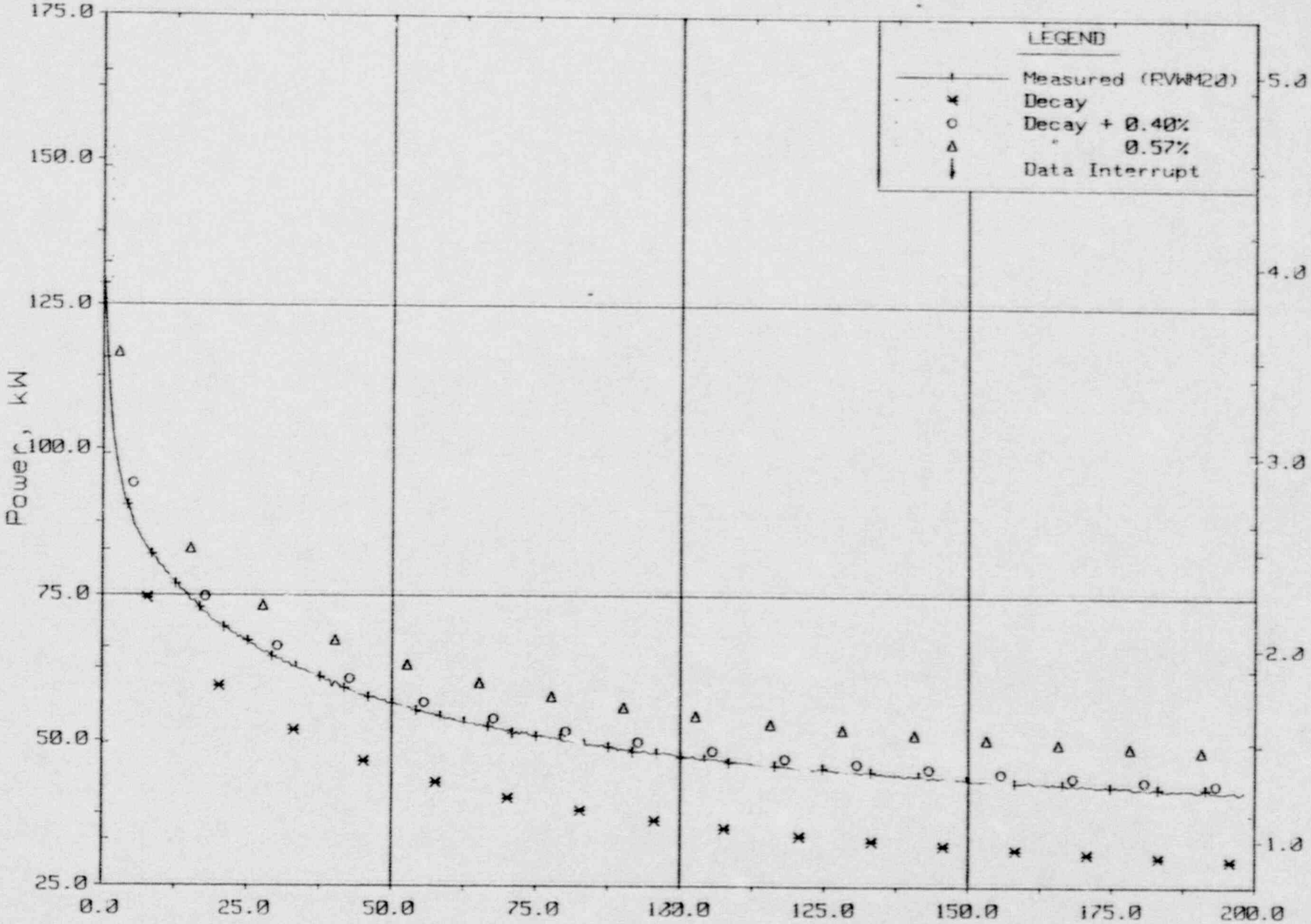
Pressurizer Collapsed Liquid Level (PZLV20).

Wed Jun 1 11:0?:41 1988

PZLV1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



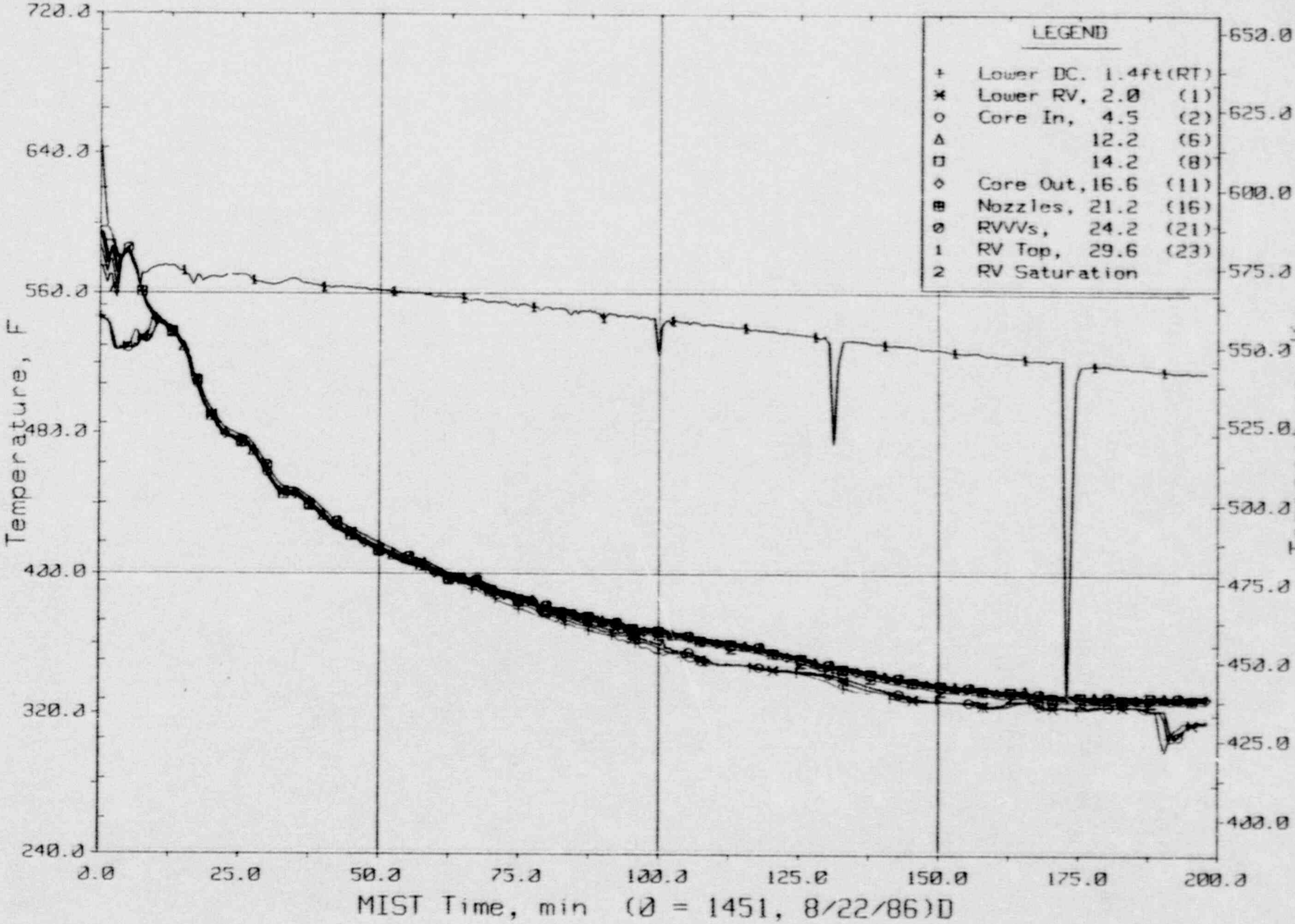
Core Power.

Wed Jun 1 11:07:06 1988

RVKW0

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



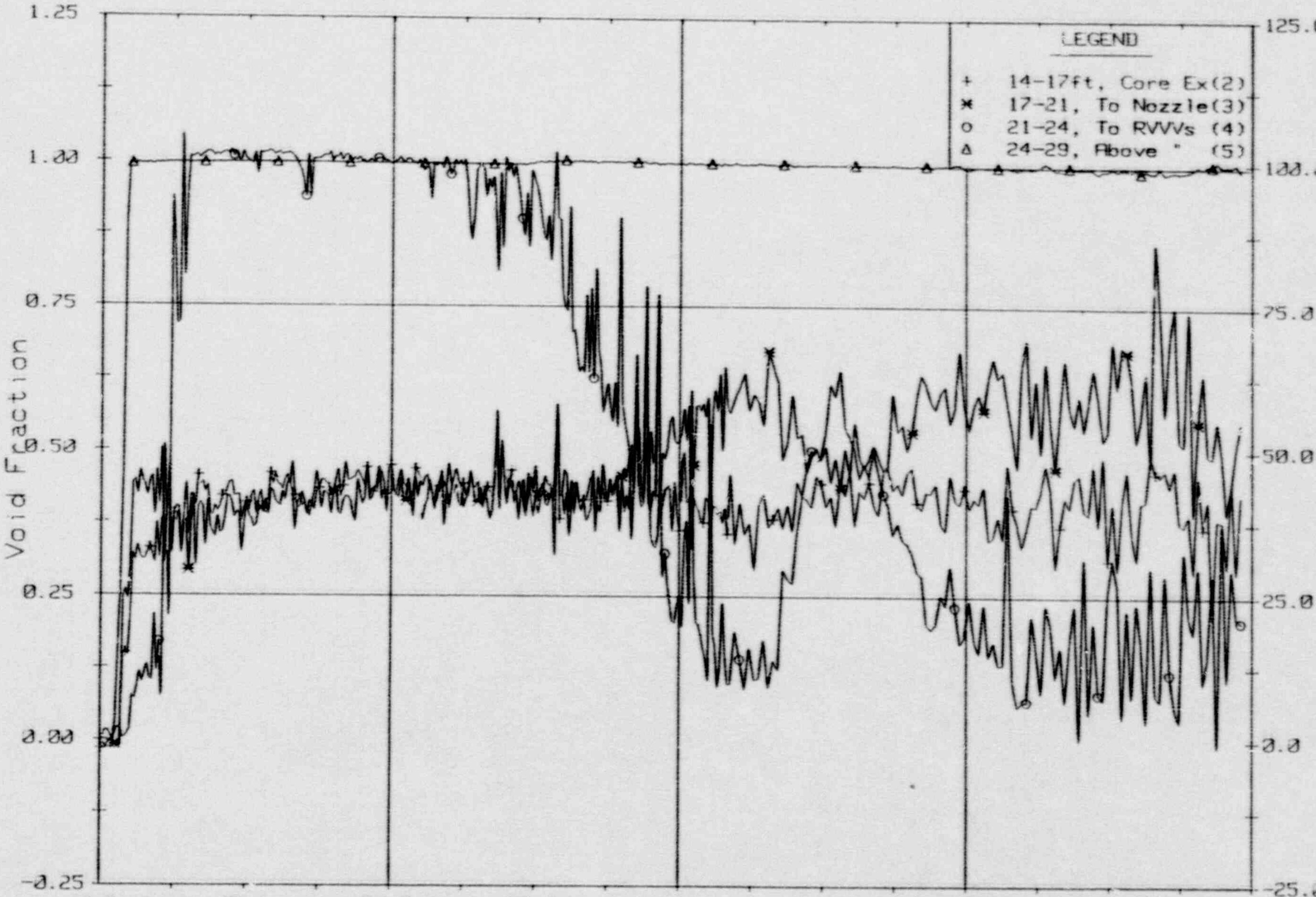
Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Wed Jun 1 11:11:30 1988

RVTC0

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



MIST Time, min (0 = 1451, 8/22/86)D

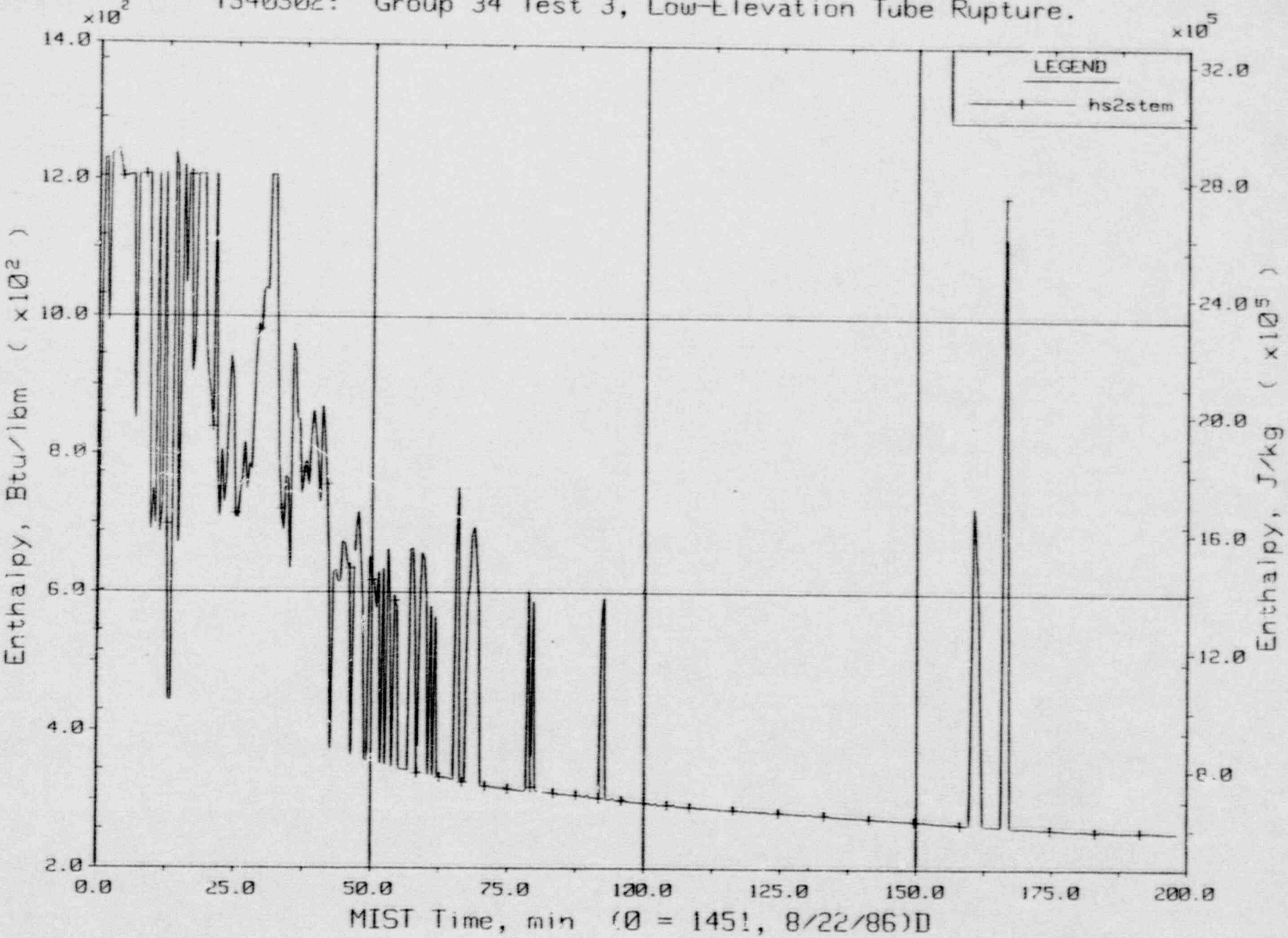
Reactor Vessel Void Fractions From Differential Pressures (RVVFs).

Wed Jun 1 11:15:10 1988

RVVF1

FINAL DATA

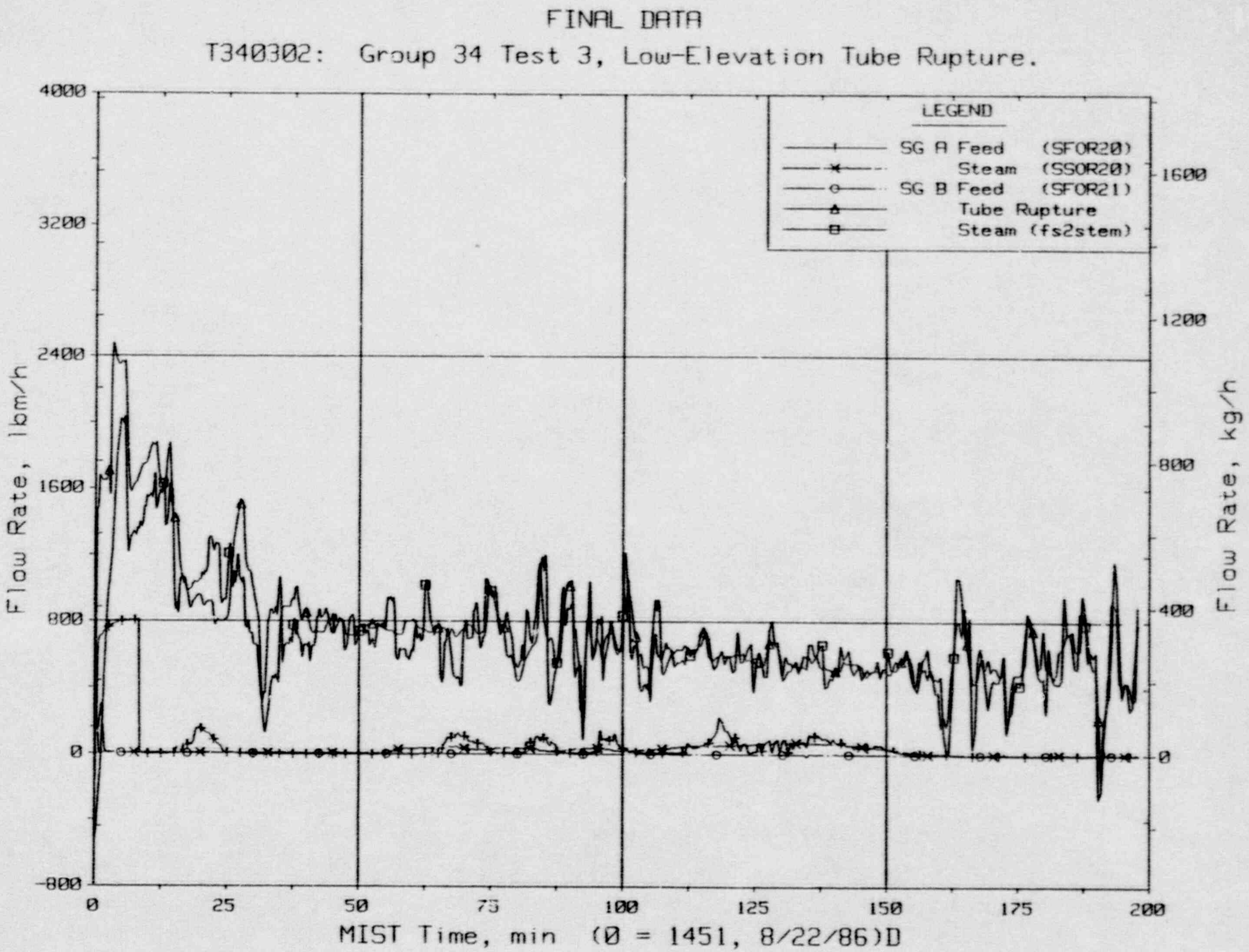
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Steam Generator B Steam Exit Enthalpy.

Wed Jun 1 12:58:05 1988

S2EN1



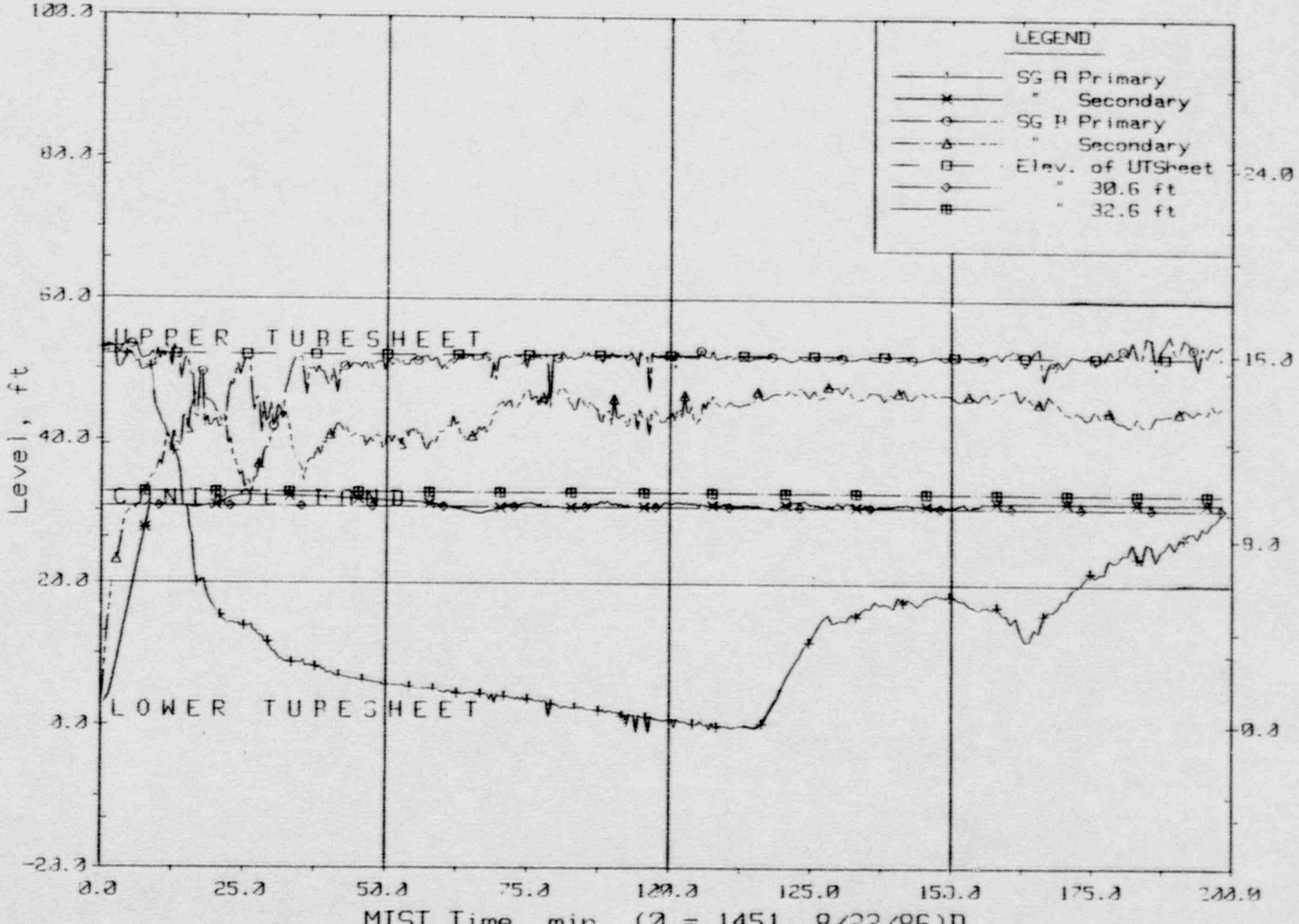
Steam Generator Secondary Flow Rates.

Tue Nov 1 17:43:49 1988

SGFL1

FINAL DATA

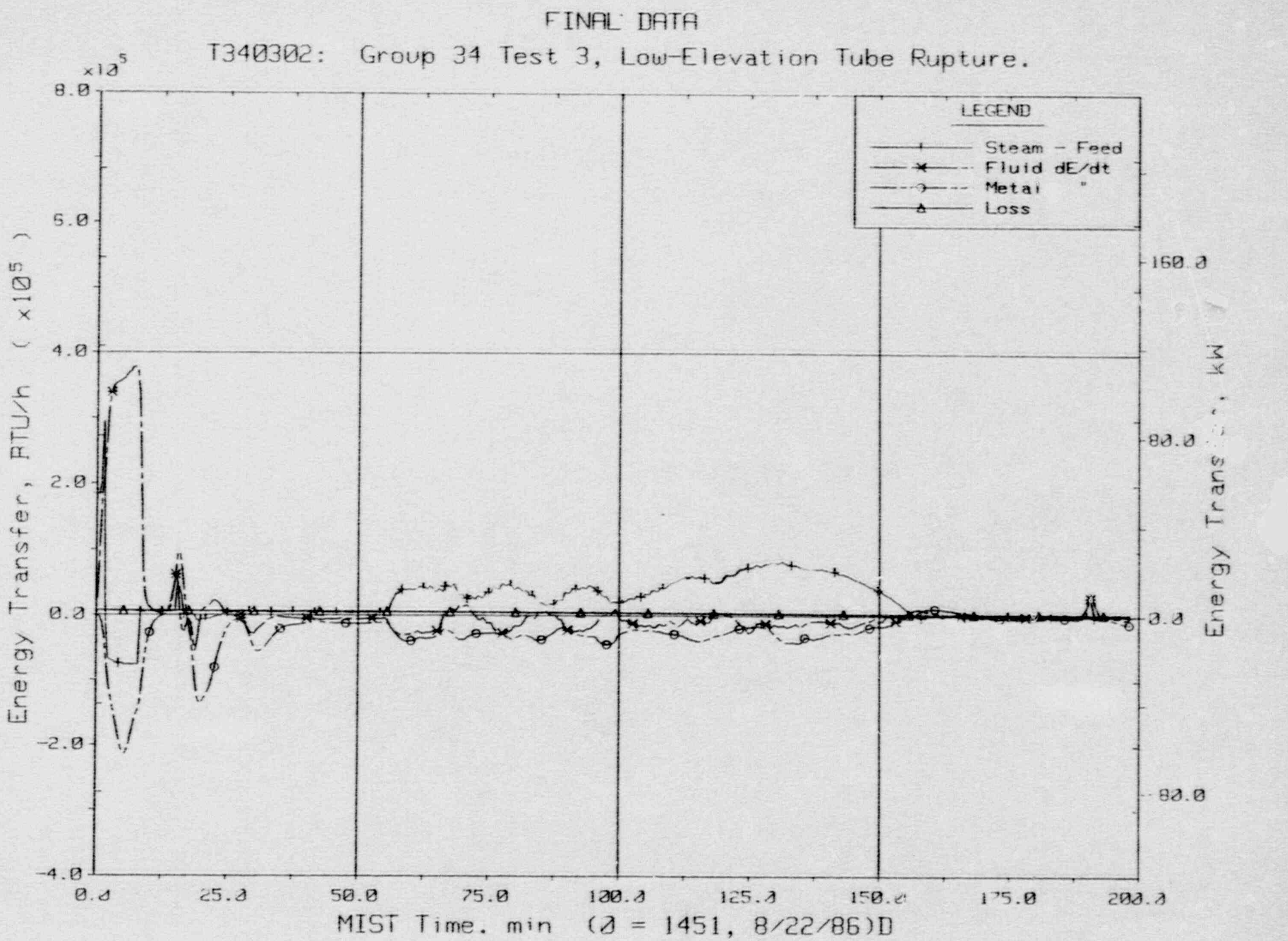
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Steam Generator Collapsed Liquid Levels.

Wed Jun 1 11:24:49 19…

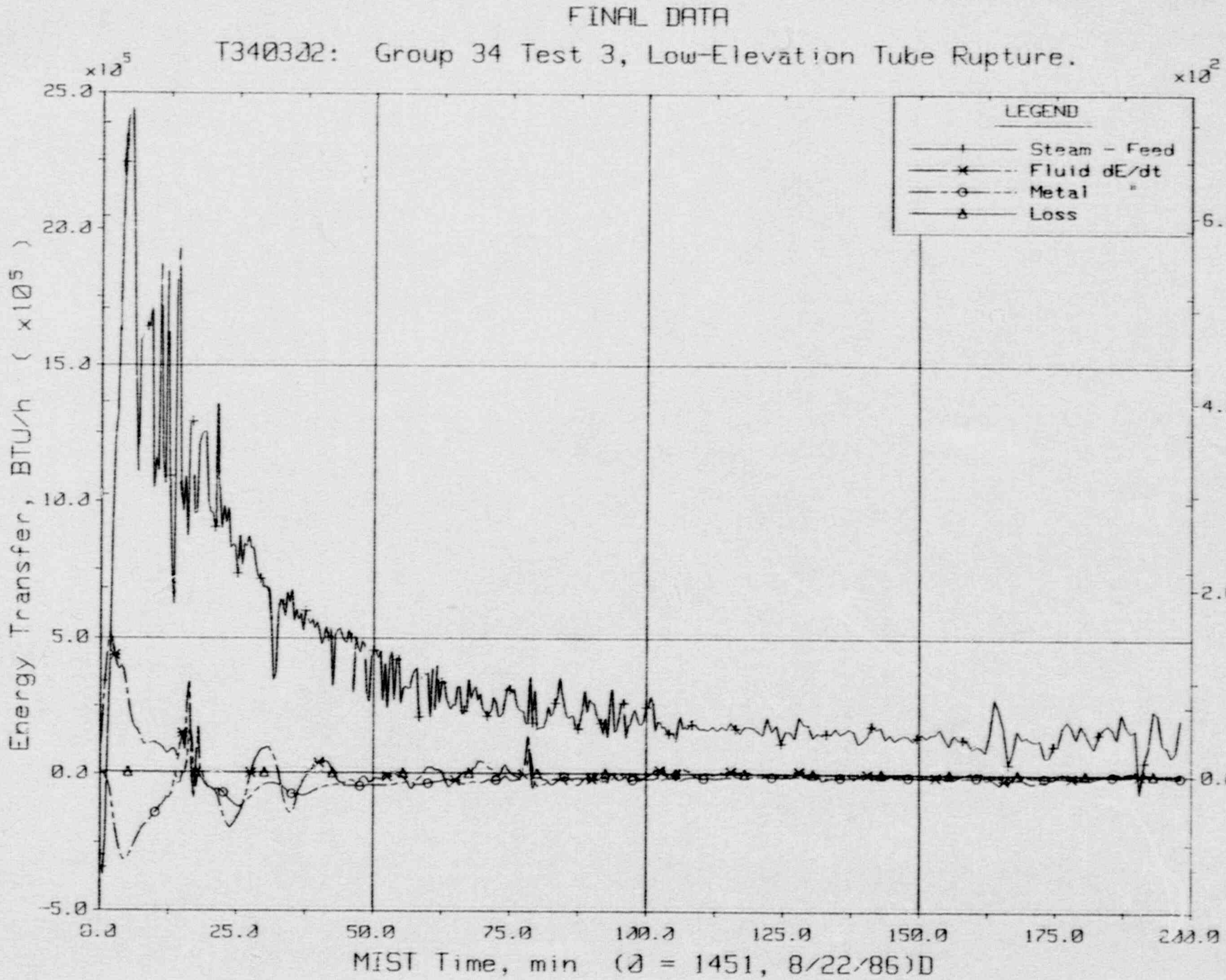
SGLV2



Steam Generator A Energy Transfer.

Wed Jun 1 11:43:23 1988

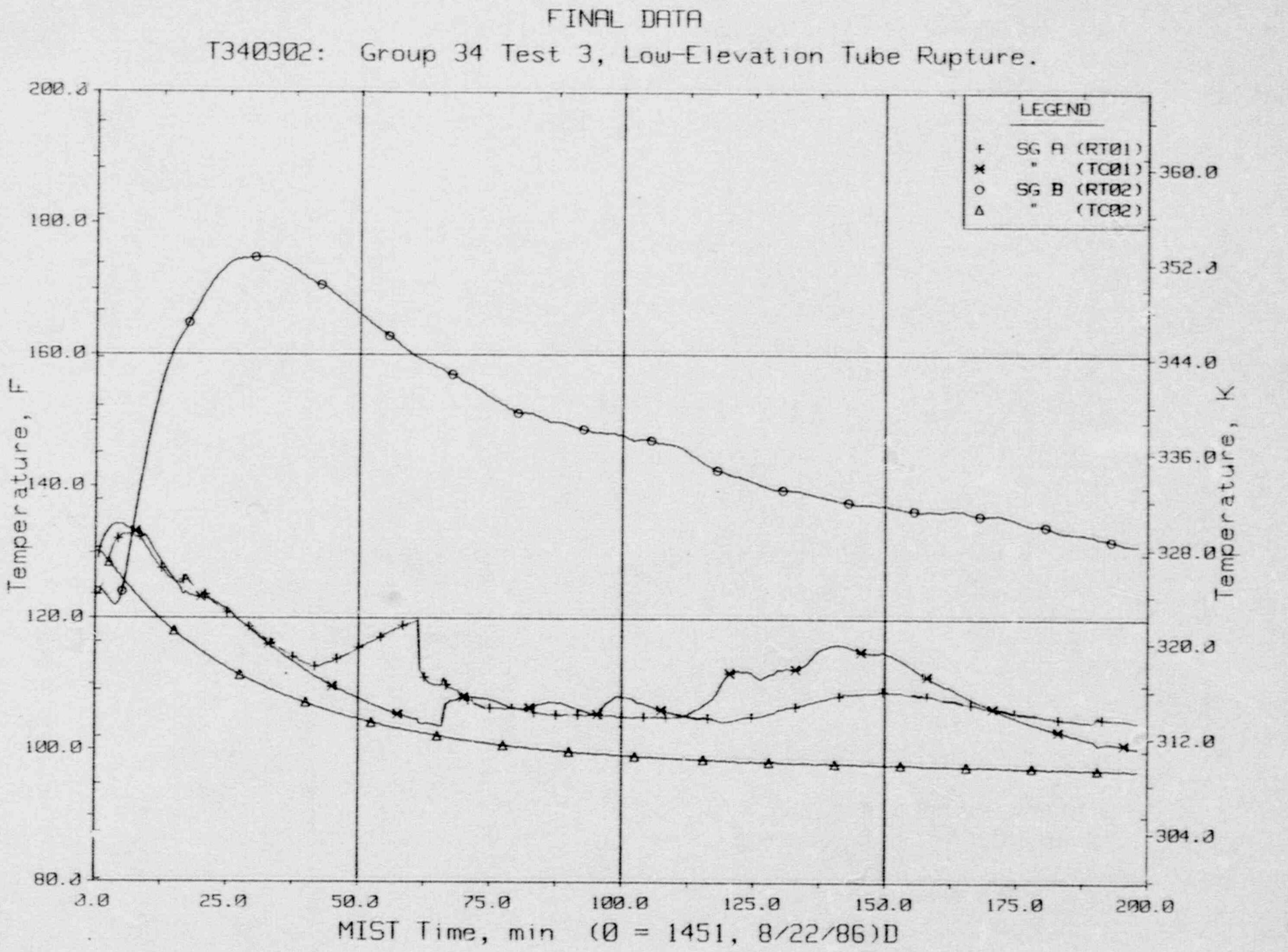
SGOE3



Steam Generator B Energy Transfer.

Wed Jun 1 11:46:57 1988

SGOE6



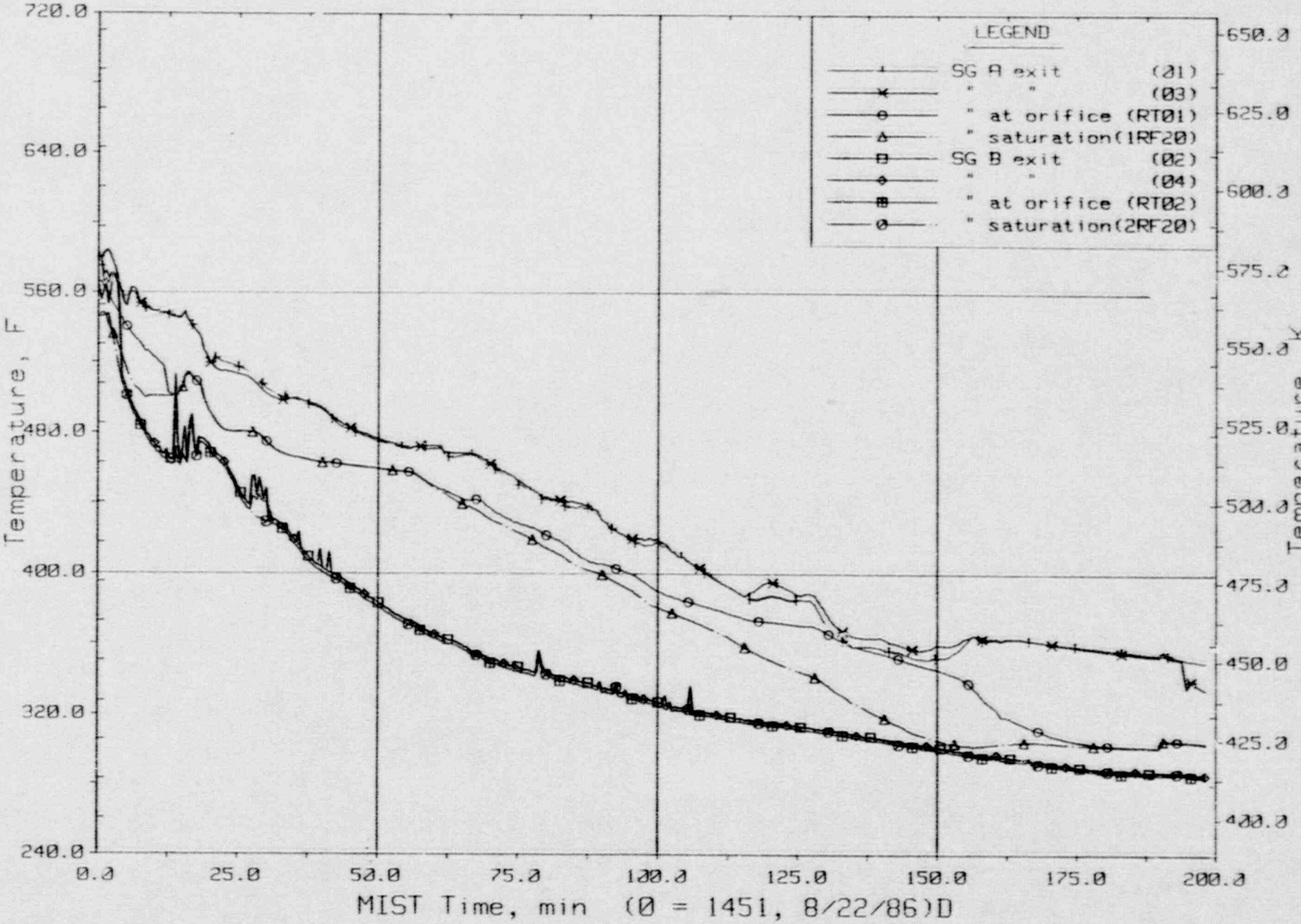
Feedwater Temperatures (SFs).

Wed Jun 1 11:51:10 1988

SGRT1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



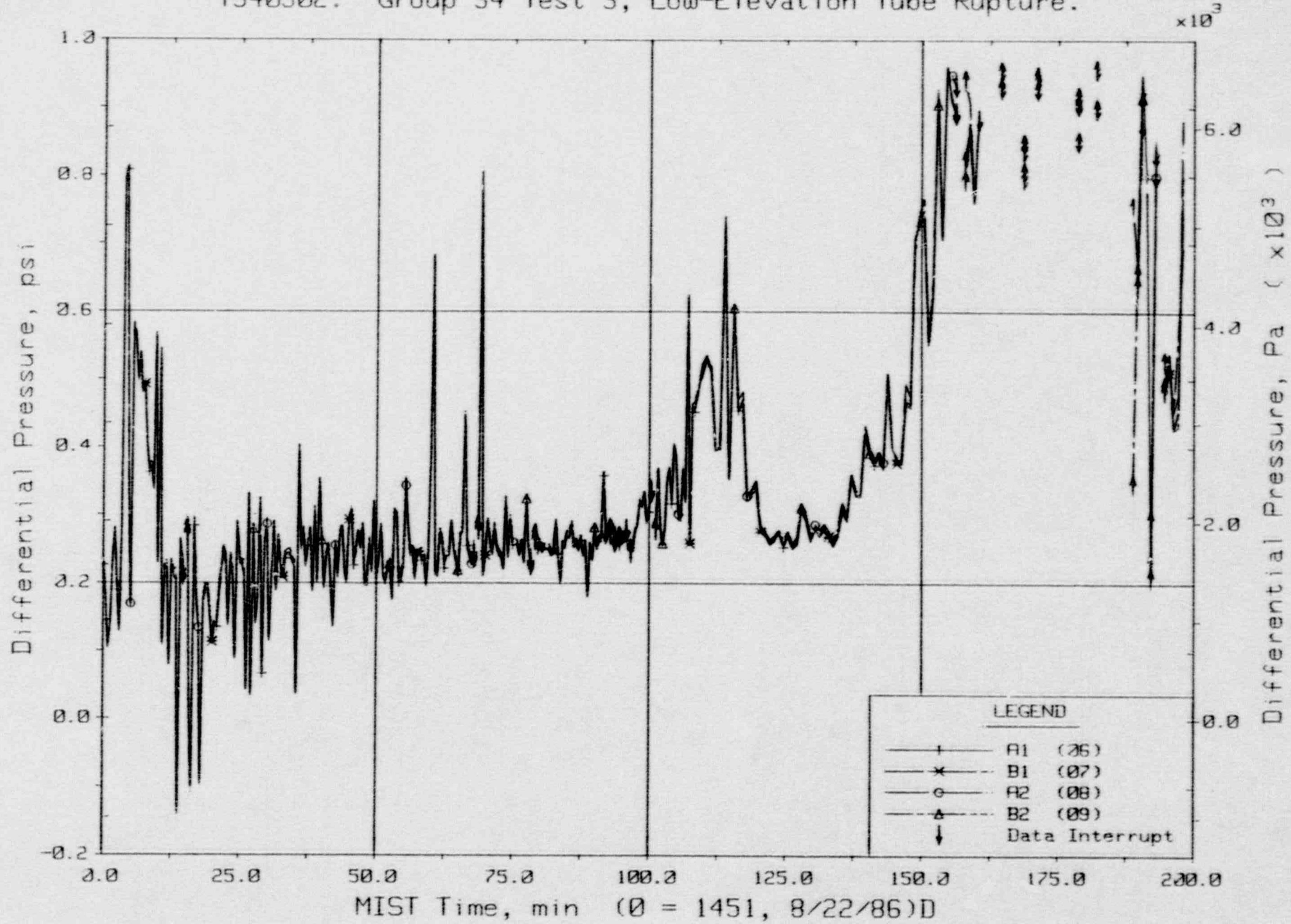
Steam Generator Steam Outlet Temperatures (SSTCs).

Wed Jun 1 11:55:51 1988

SGRT2

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



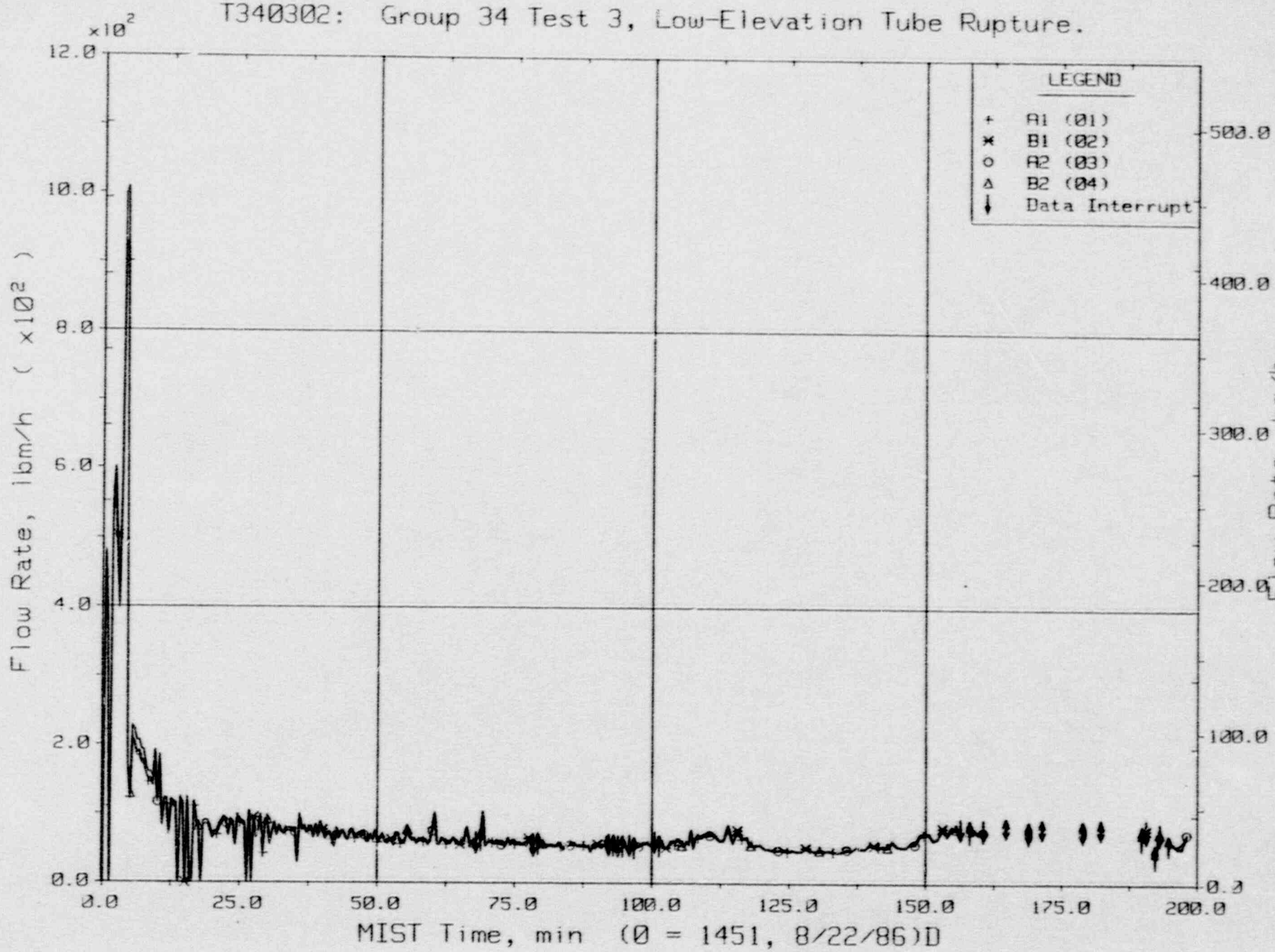
Reactor Vessel Vent Valve Differential Pressures (RVDPs).

Wed Jun 1 12:03:44 1988

VVDP1

FINAL DATA

T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



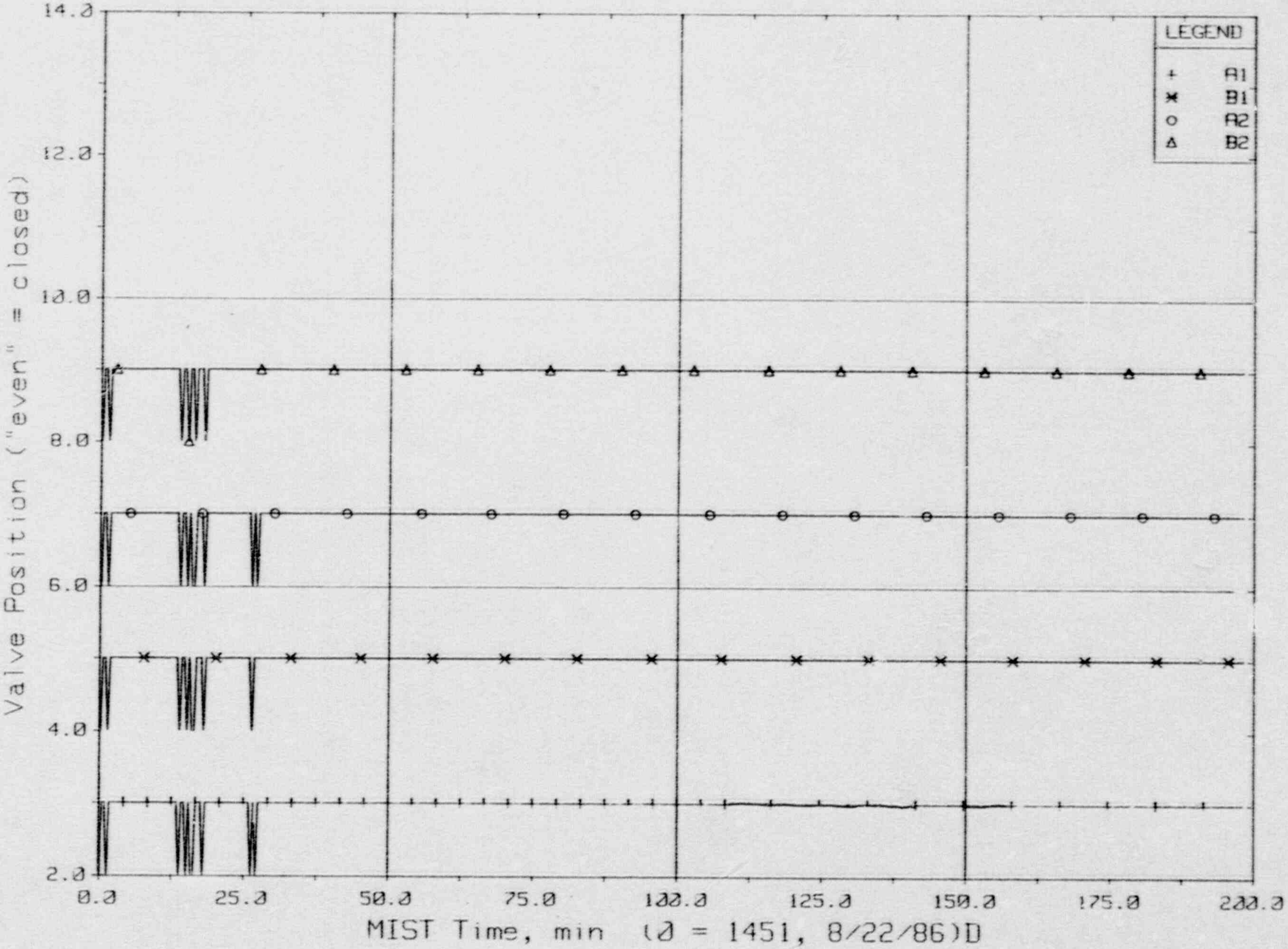
Reactor Vessel Vent Valve Flow Rates (RVORs).

Wed Jun 1 12:07:52 1988

VVFL1

FINAL DATA

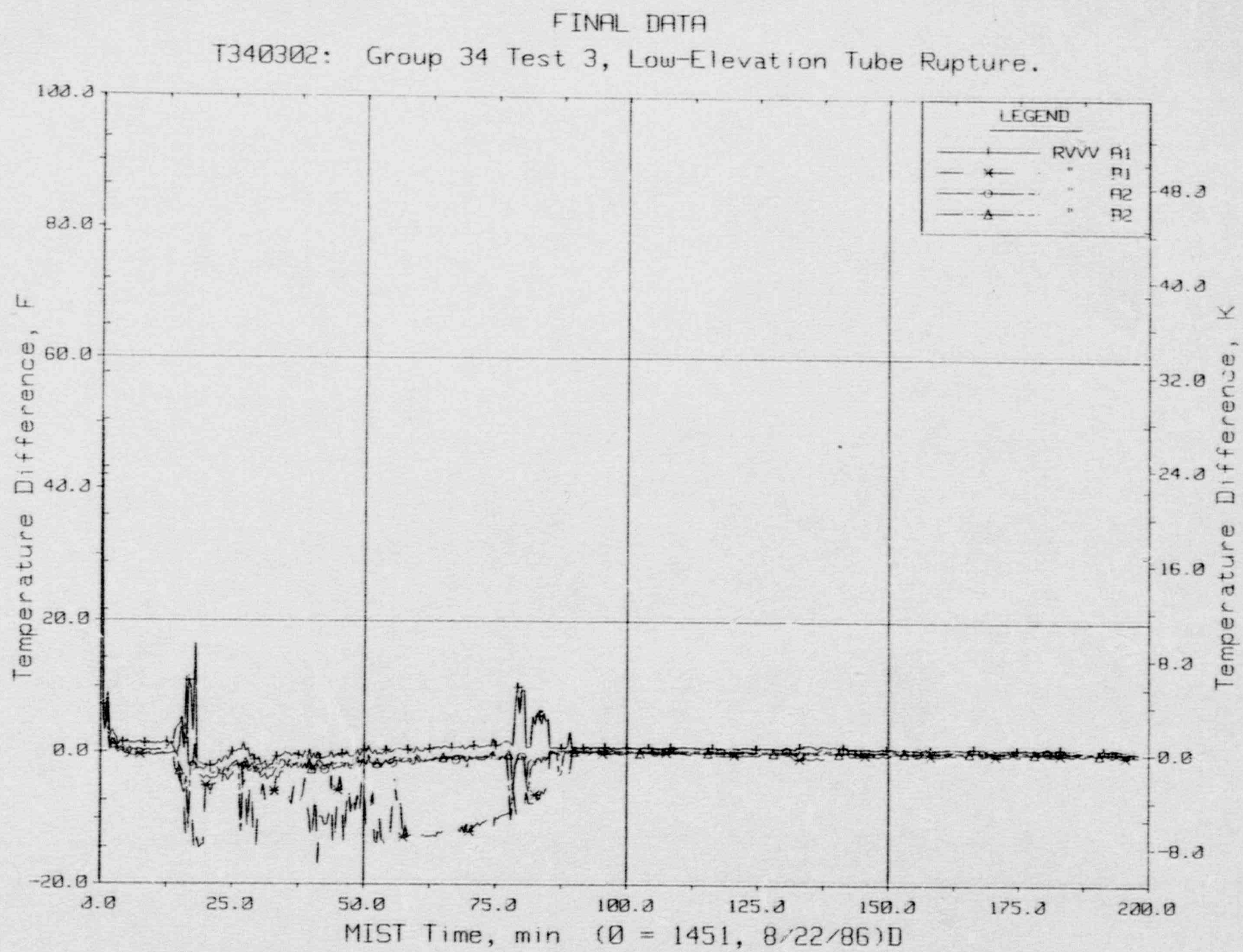
T340302: Group 34 Test 3, Low-Elevation Tube Rupture.



Reactor Vessel Vent Valve Positions.

Wed Jun 1 12:11:17 1988

T/LS1

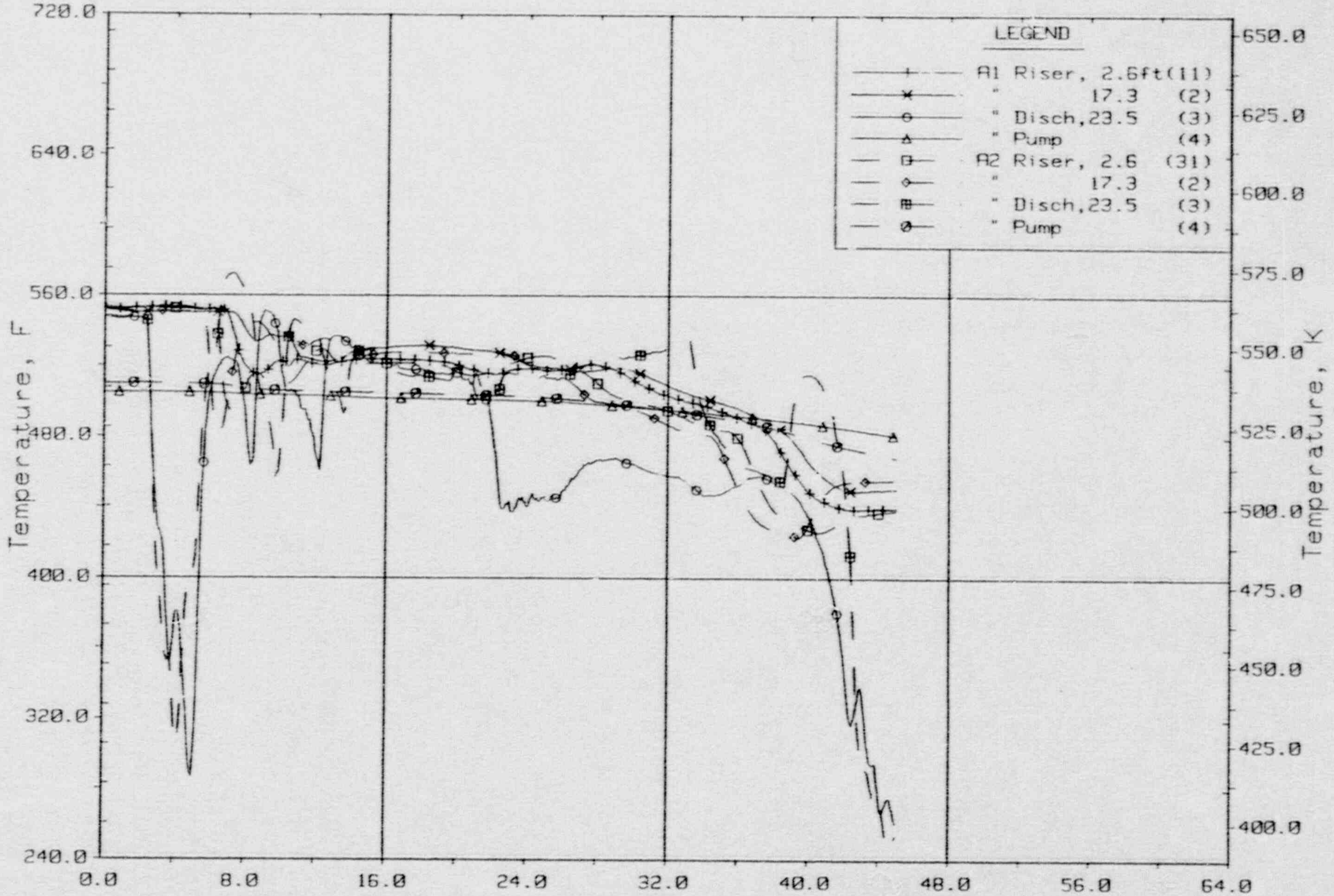


Temperature Differences Across Vent Valves.

T3404AA

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



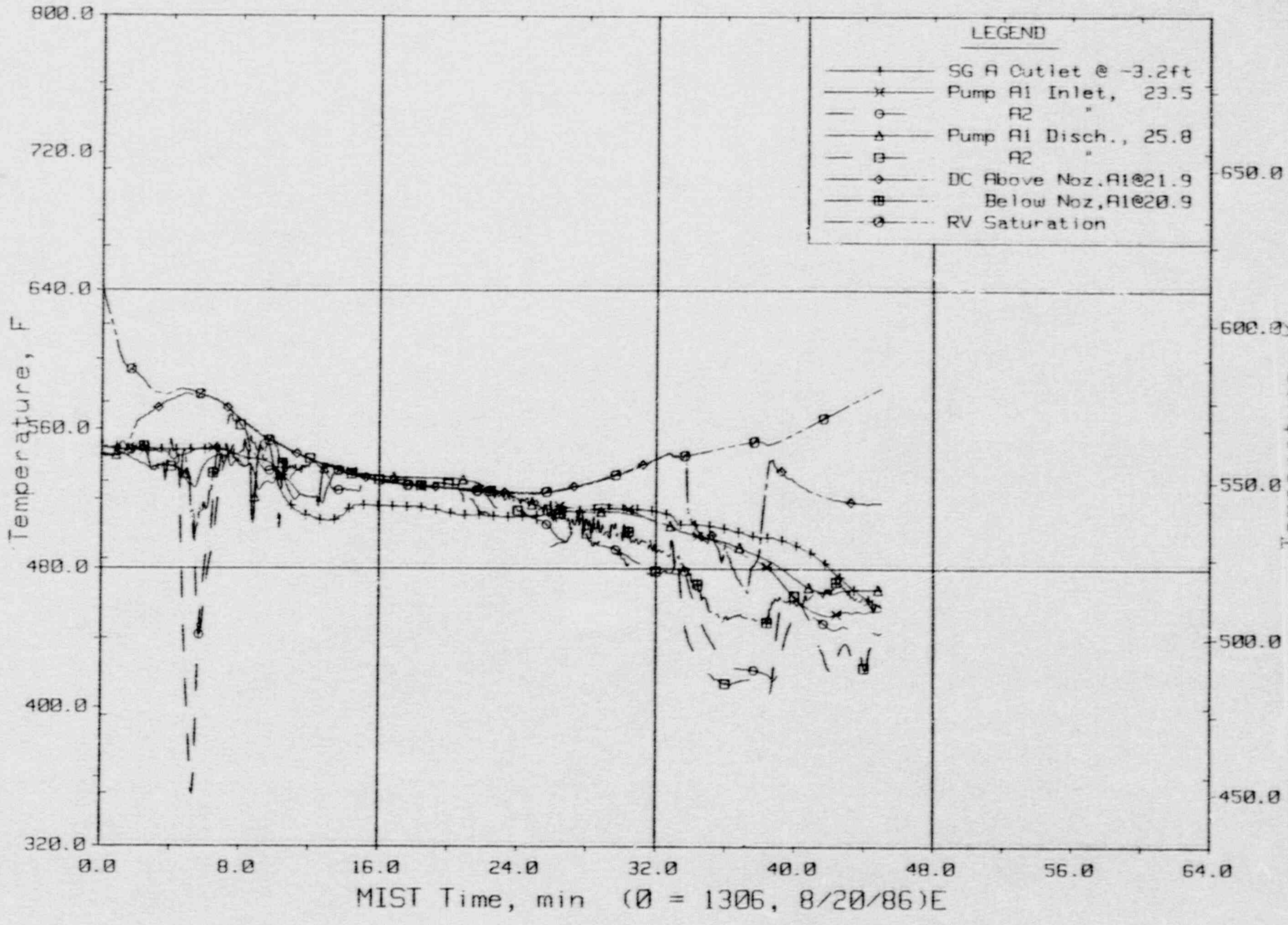
Loop A Cold Leg Metal Temperatures (C1,3MTs).

Wed Jun 1 08:09:59 1988

C1MT1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



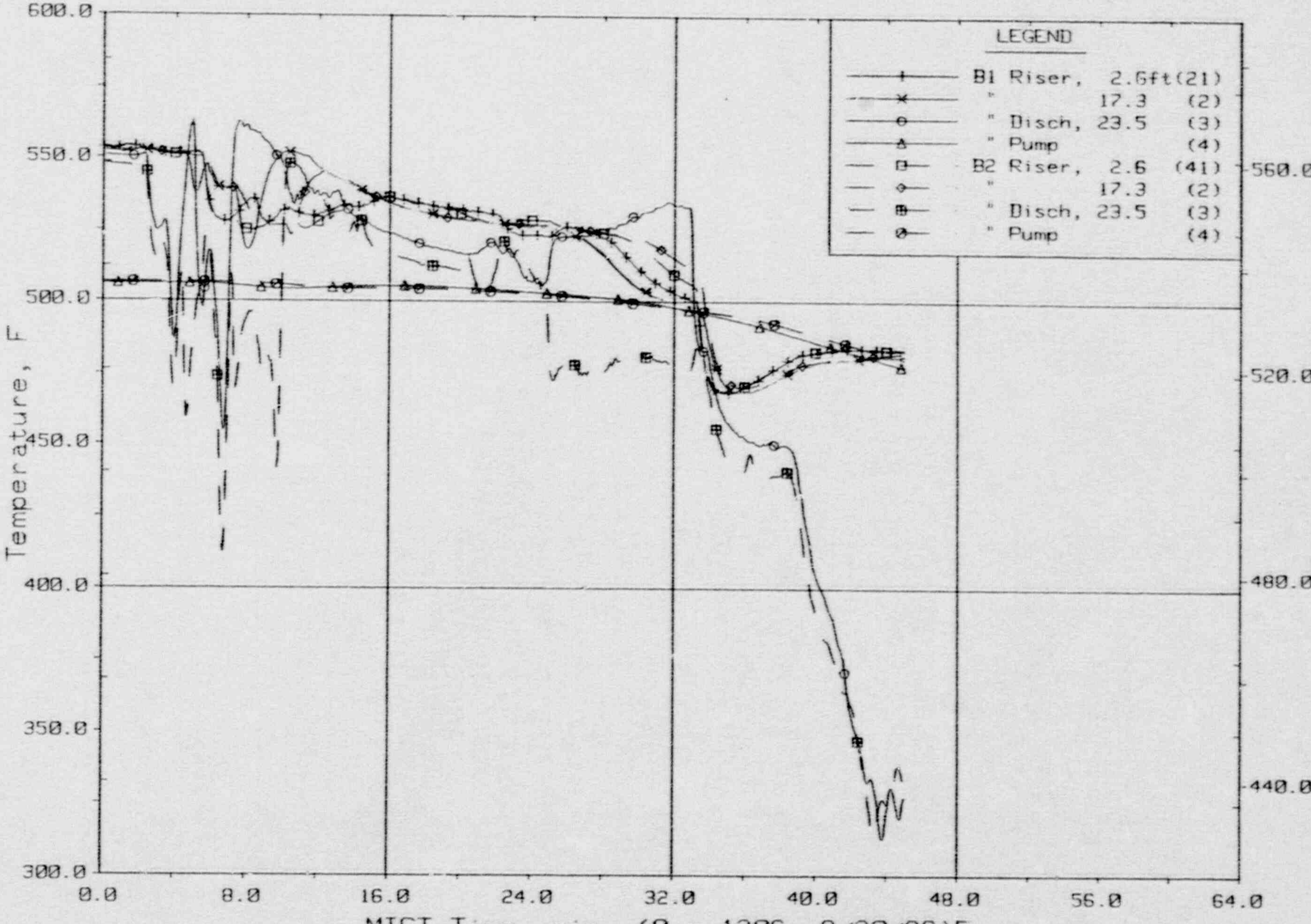
Loop A Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 08:14:56 1988

C1RT0

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



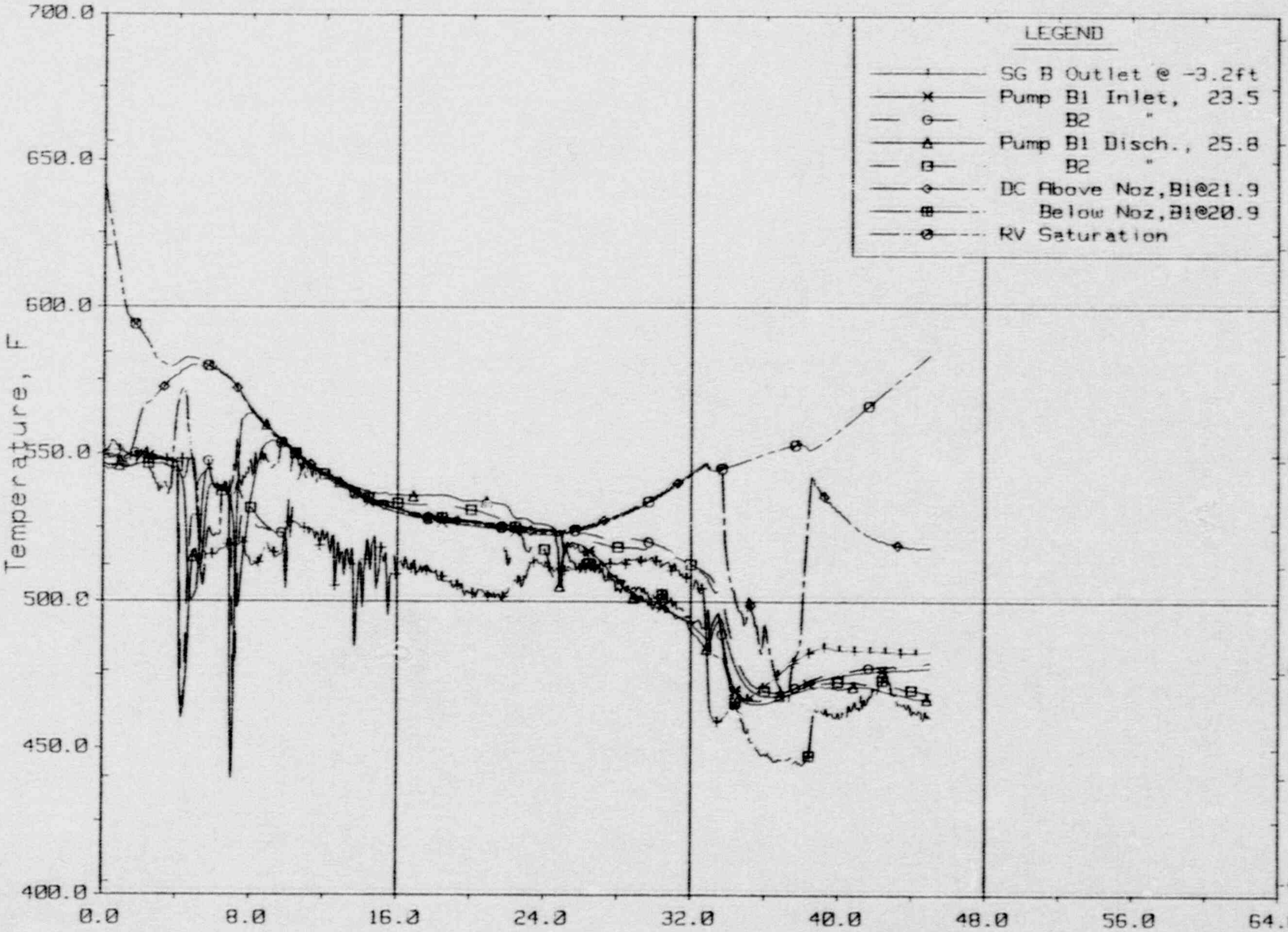
Loop B Cold Leg Metal Temperatures (C2,4MTs).

Wed Jun 1 08:19:39 1988

C2MT1

FINAL DATA

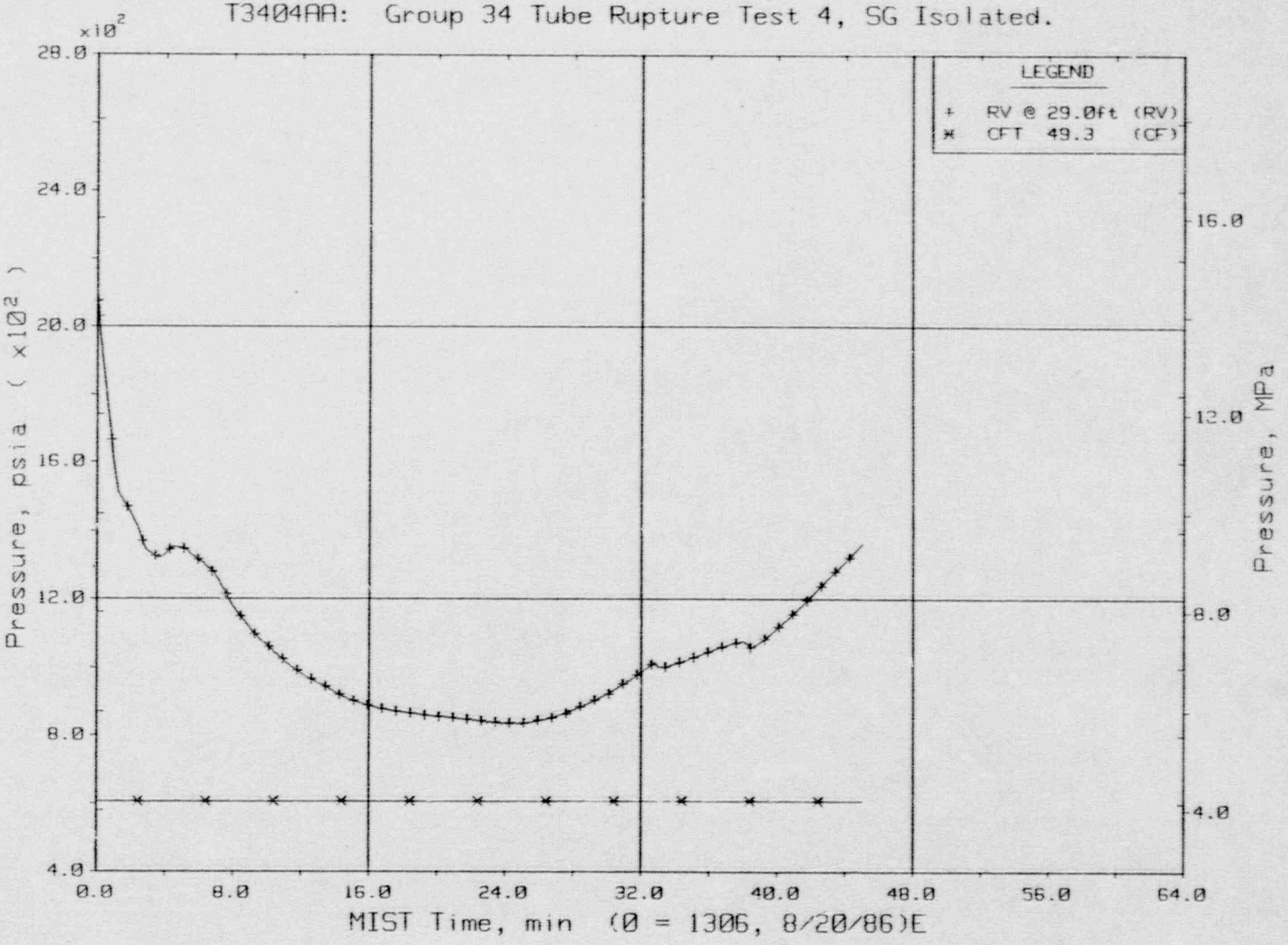
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Loop B Cold Leg Fluid Temperatures (RTDs).

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



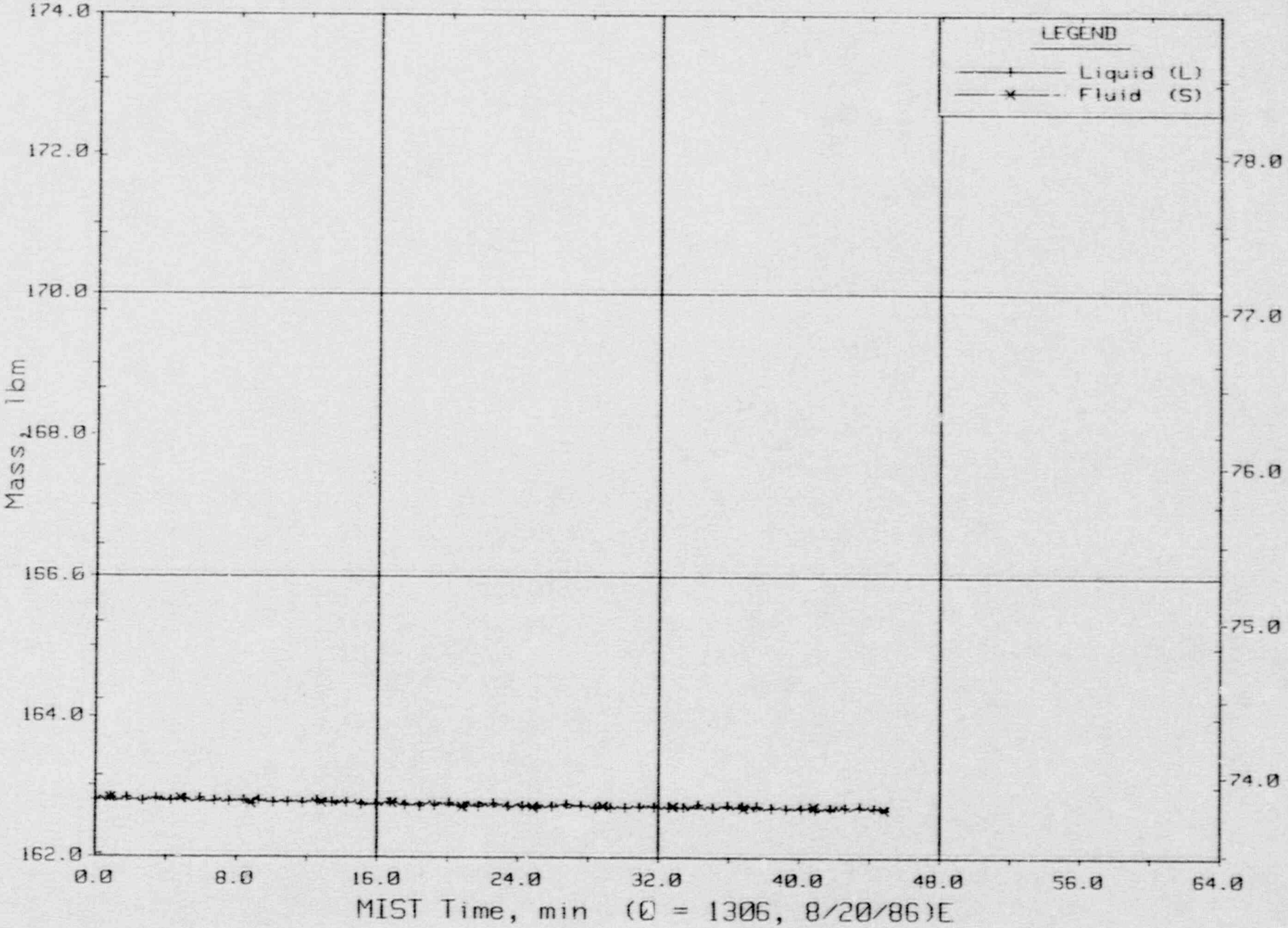
Primary System and Core Flood Tank Pressures (GP01s).

Wed Jun 1 08:27:54 1988

CFGP1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



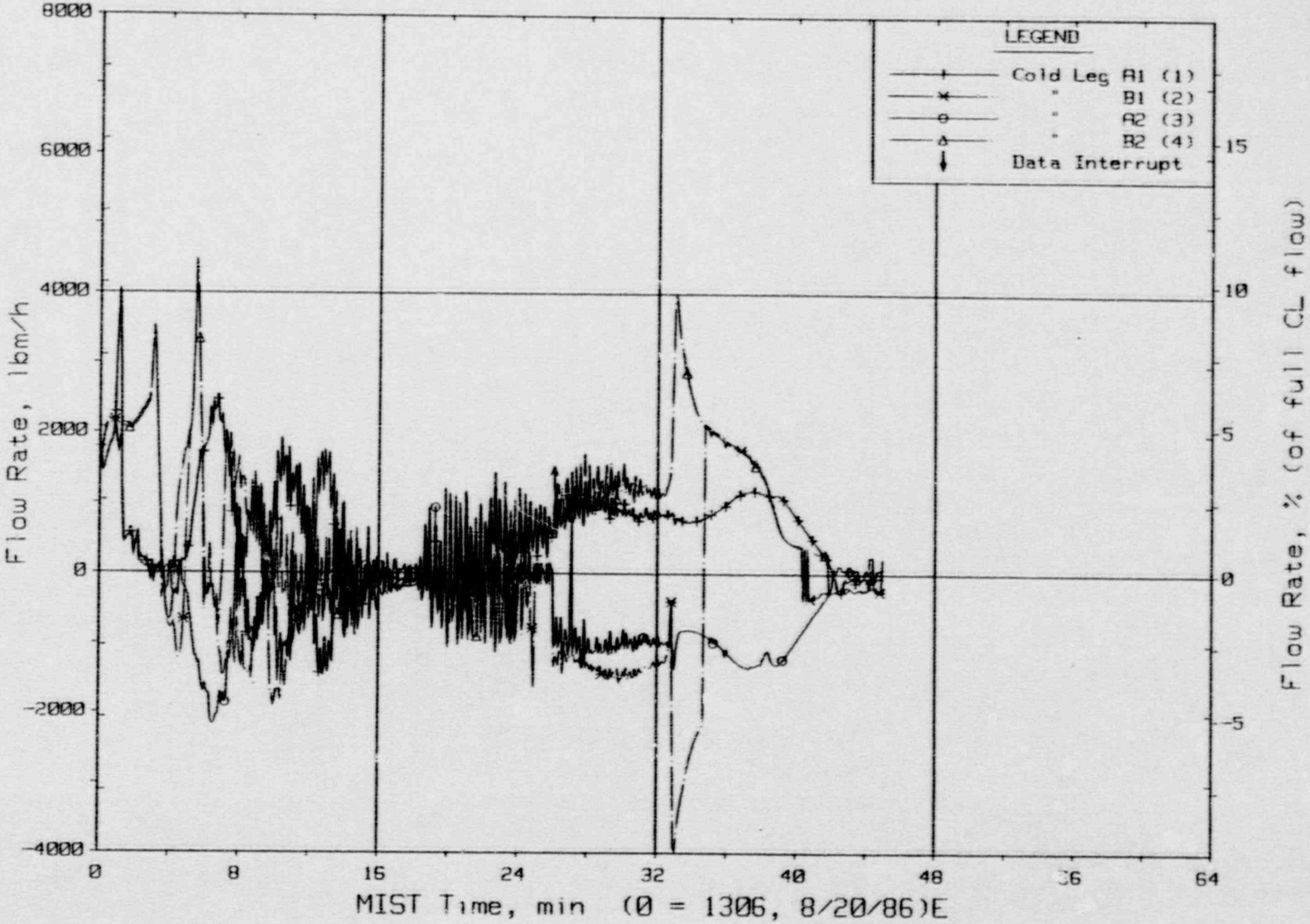
Core Flood Tank Liquid and Fluid Mass (CFMa20s).

Wed Jun 1 08:30:55 1988

CFML1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



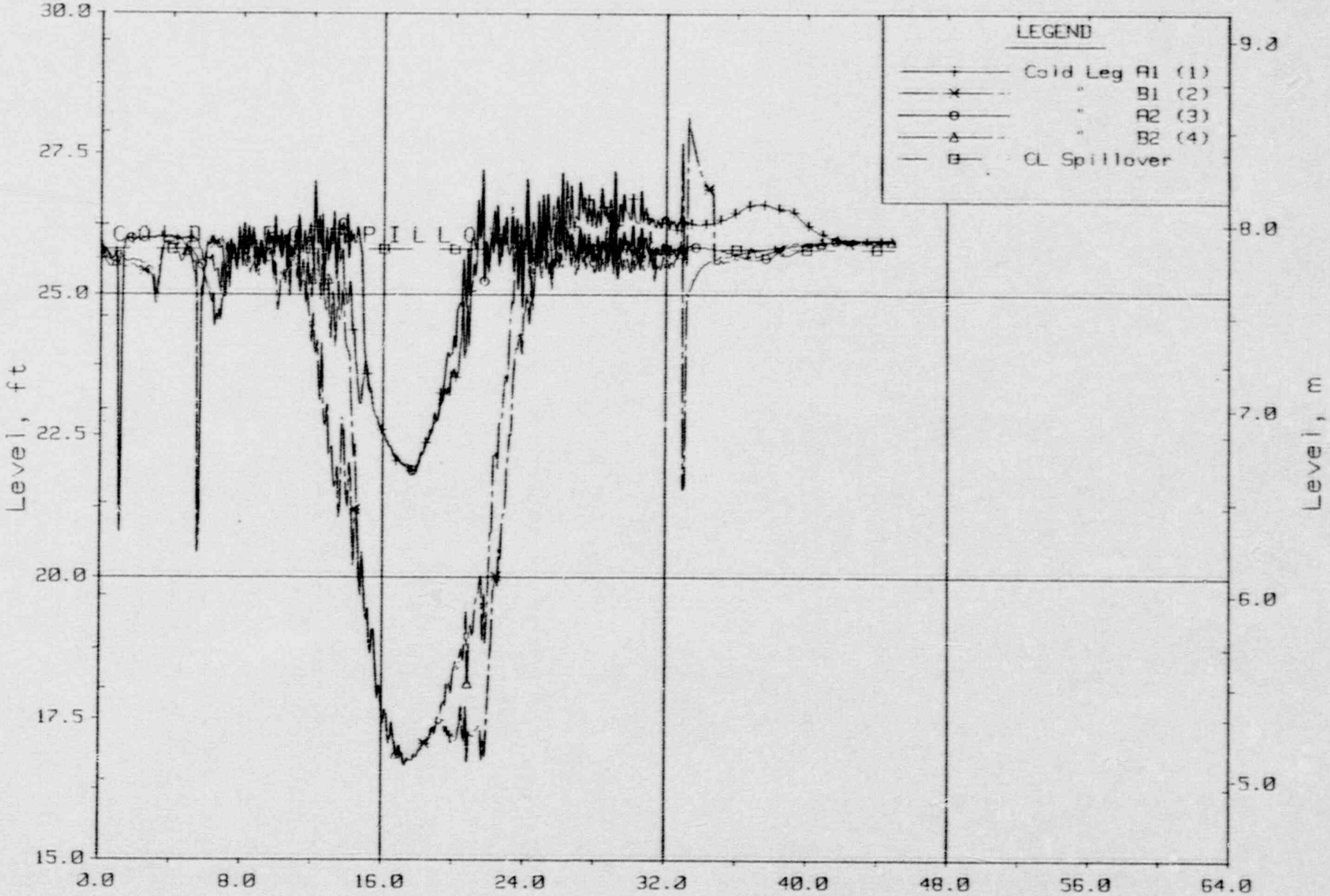
Cold Leg (Venturi) Flow Rates (CnVN20s).

Tue Nov 1 17:50:43 1988

CLFL0

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.

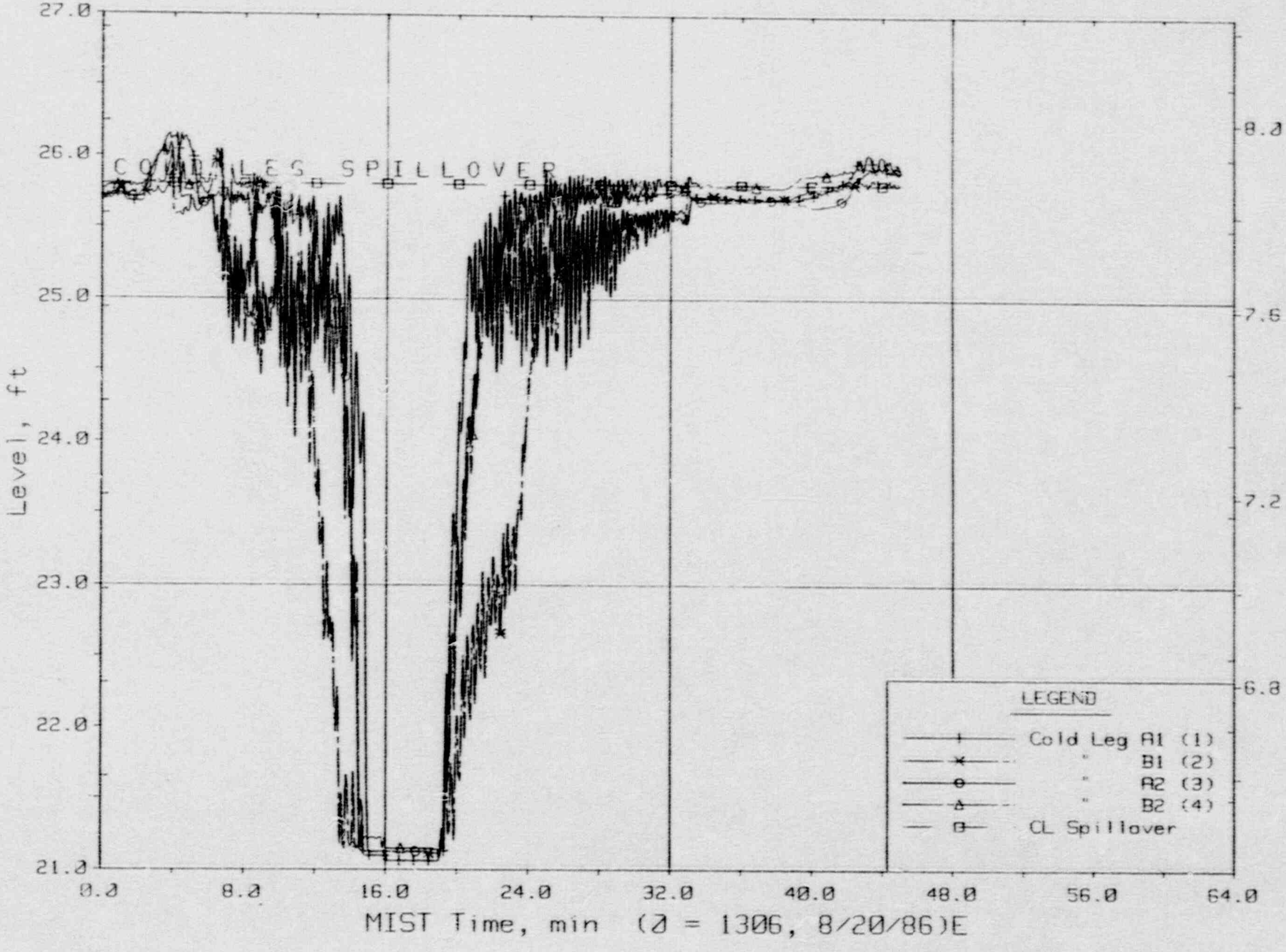


MIST Time, min (0 = 1306, 8/20/86)E

Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

FINAL DATA

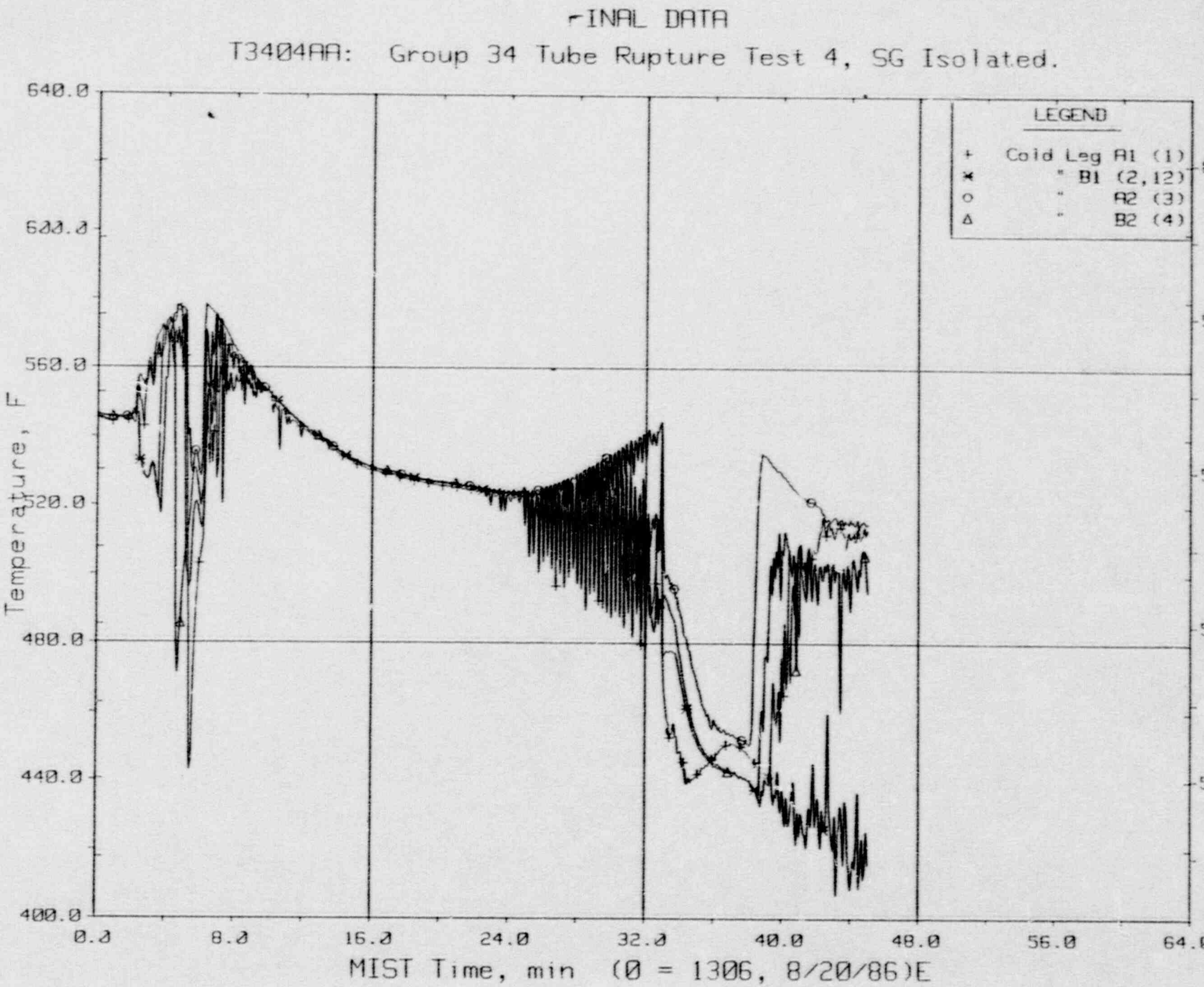
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Wed Jun 1 08:44:44 1988

CLLV5



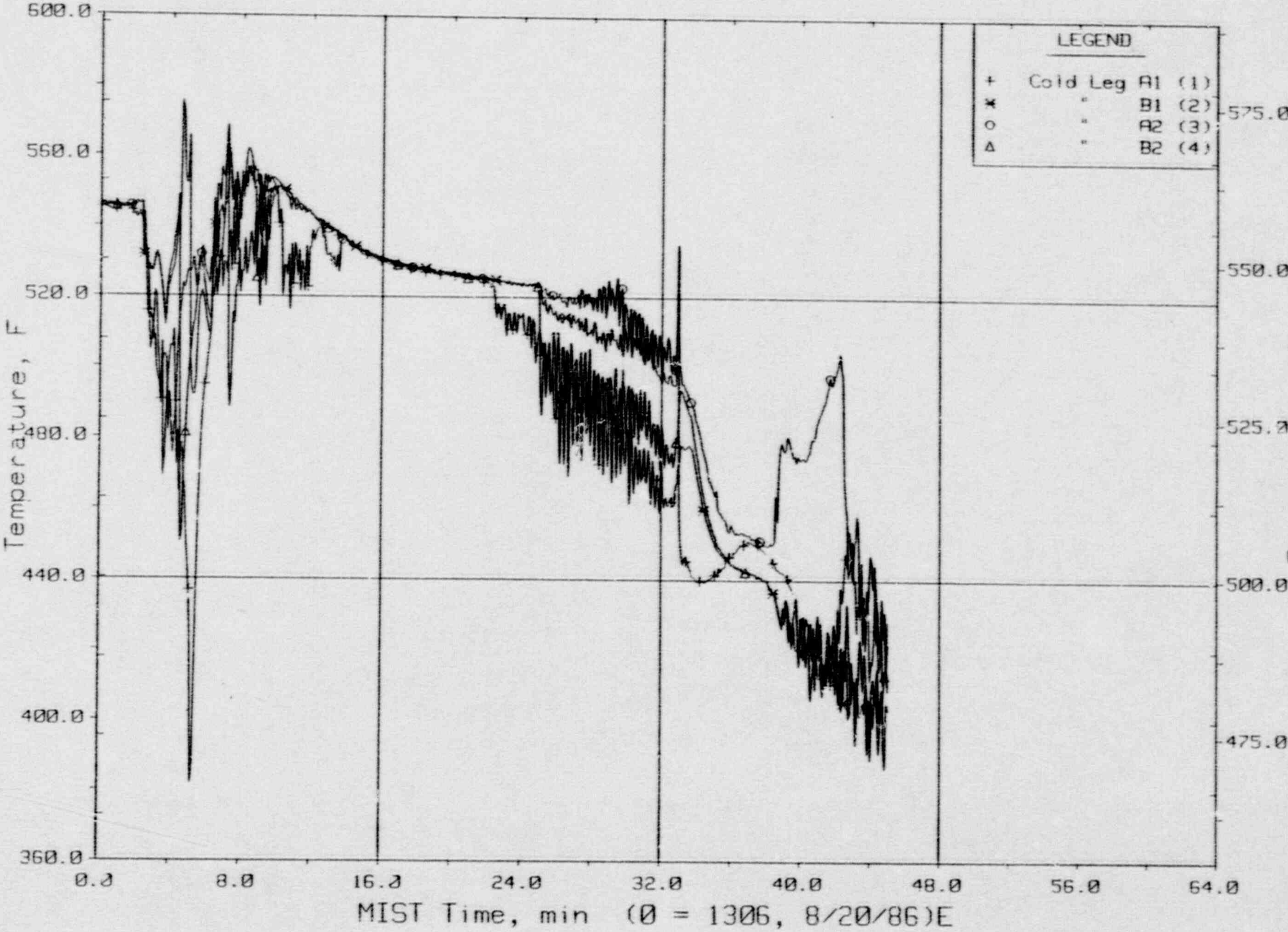
Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC11s).

Wed Jun 1 08:53:02 1988

CLTC1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



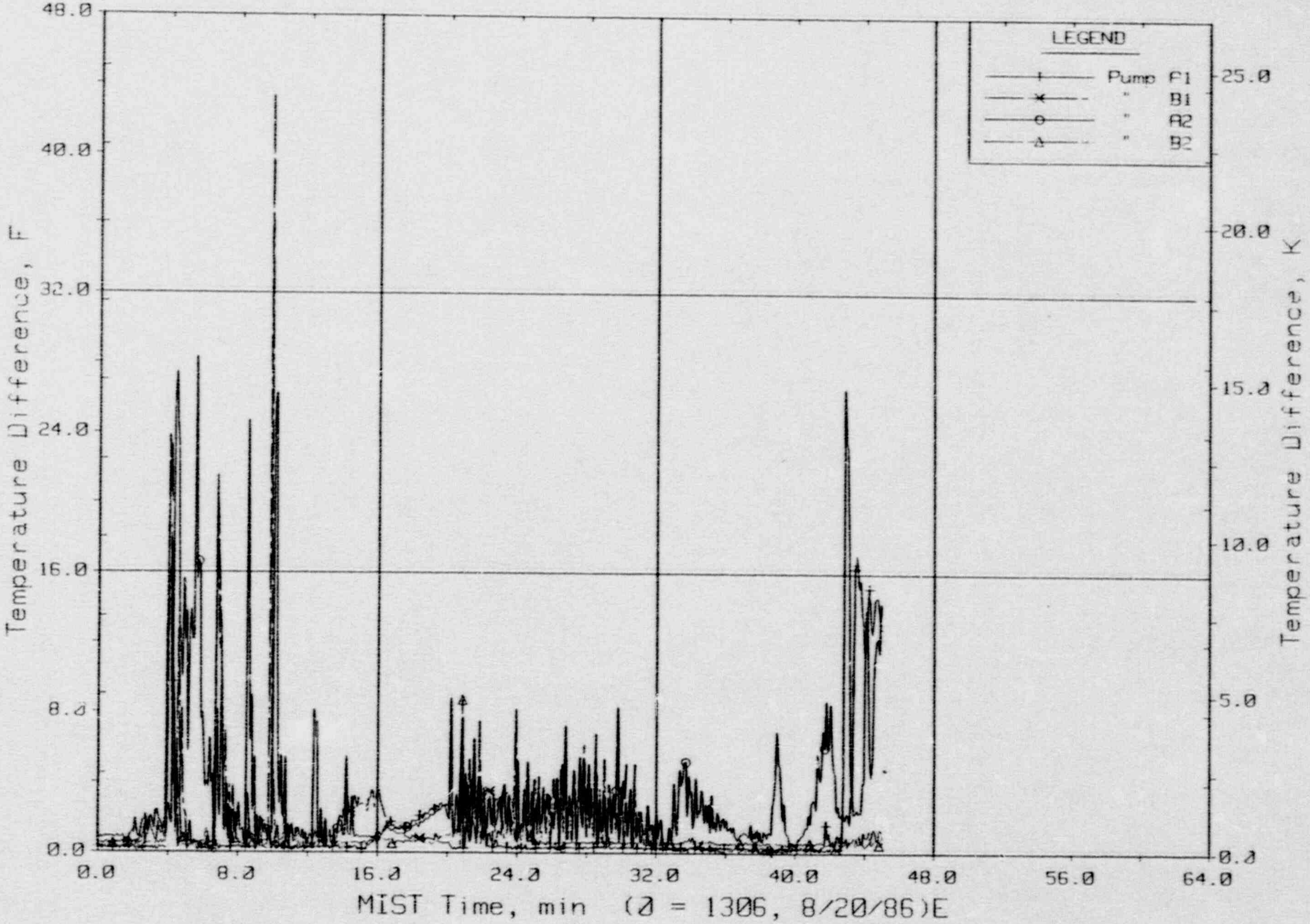
Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

Wed Jun 1 08:58:11 1988

CLTC2

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



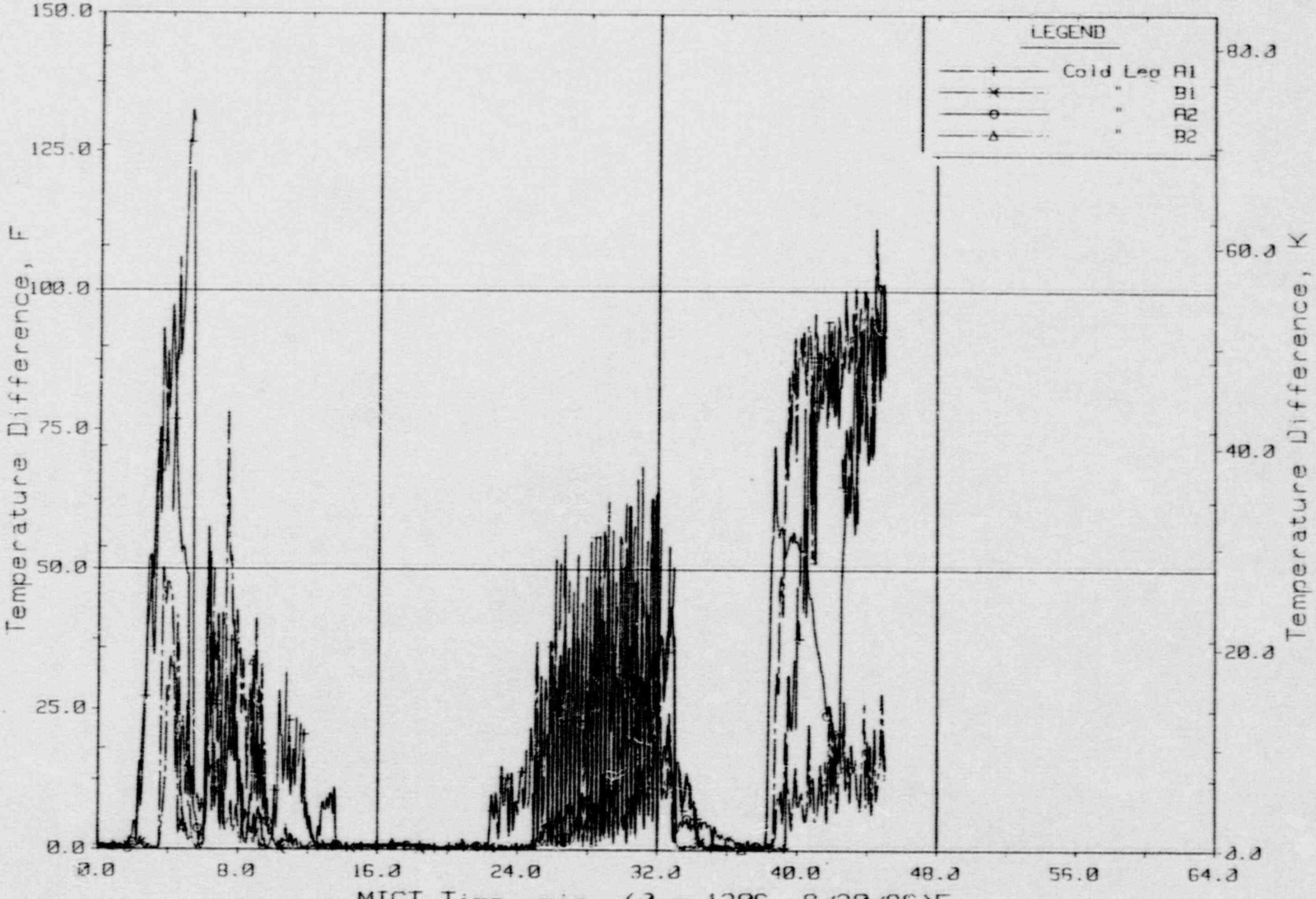
Maximum Differences Among RCP Rake FLuid Temperatures.

Wed Jun 1 09:02:44 1988

CLTD1

FINAL DATA

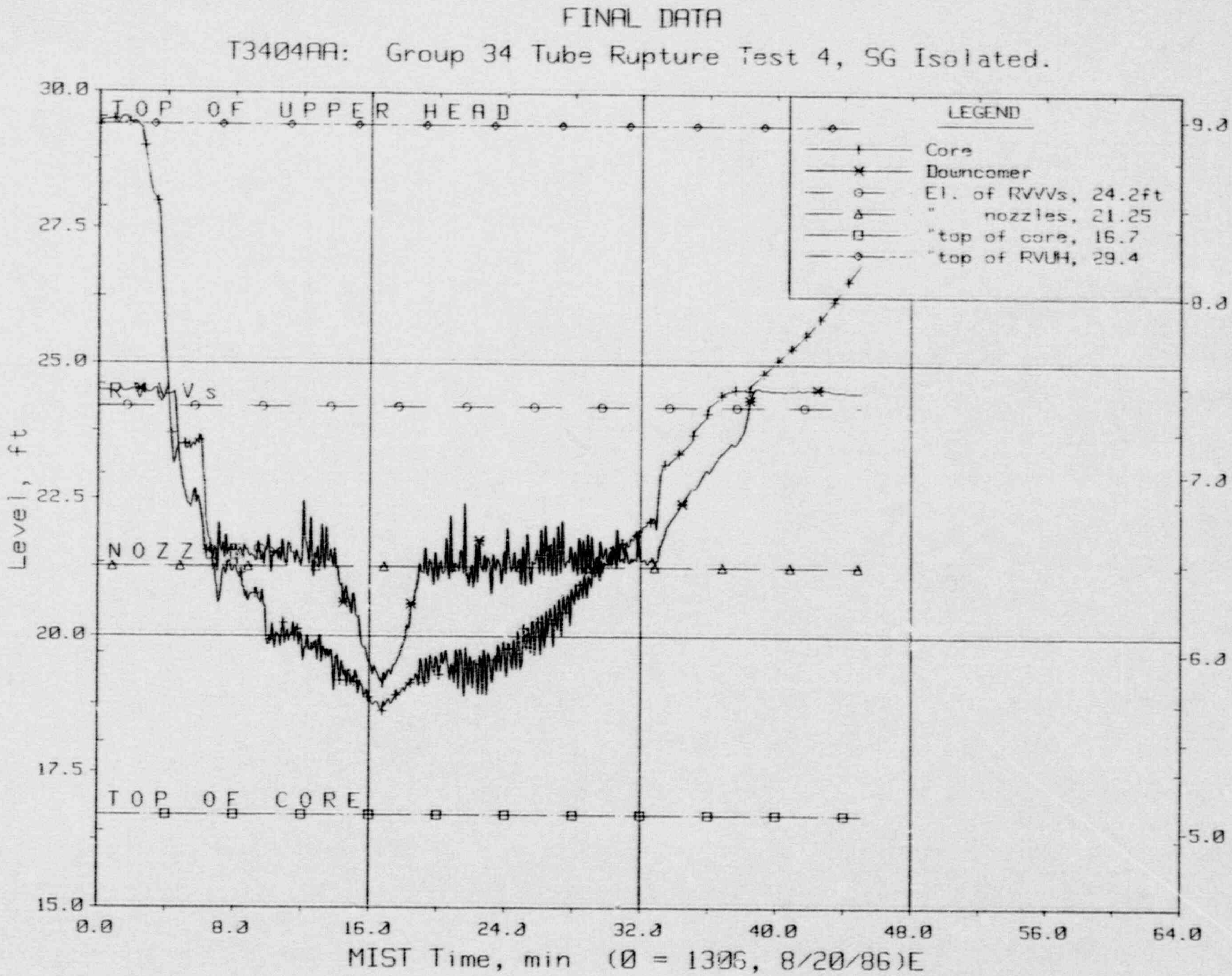
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

Wed Jun 1 09:07:22 1988

CLTD2



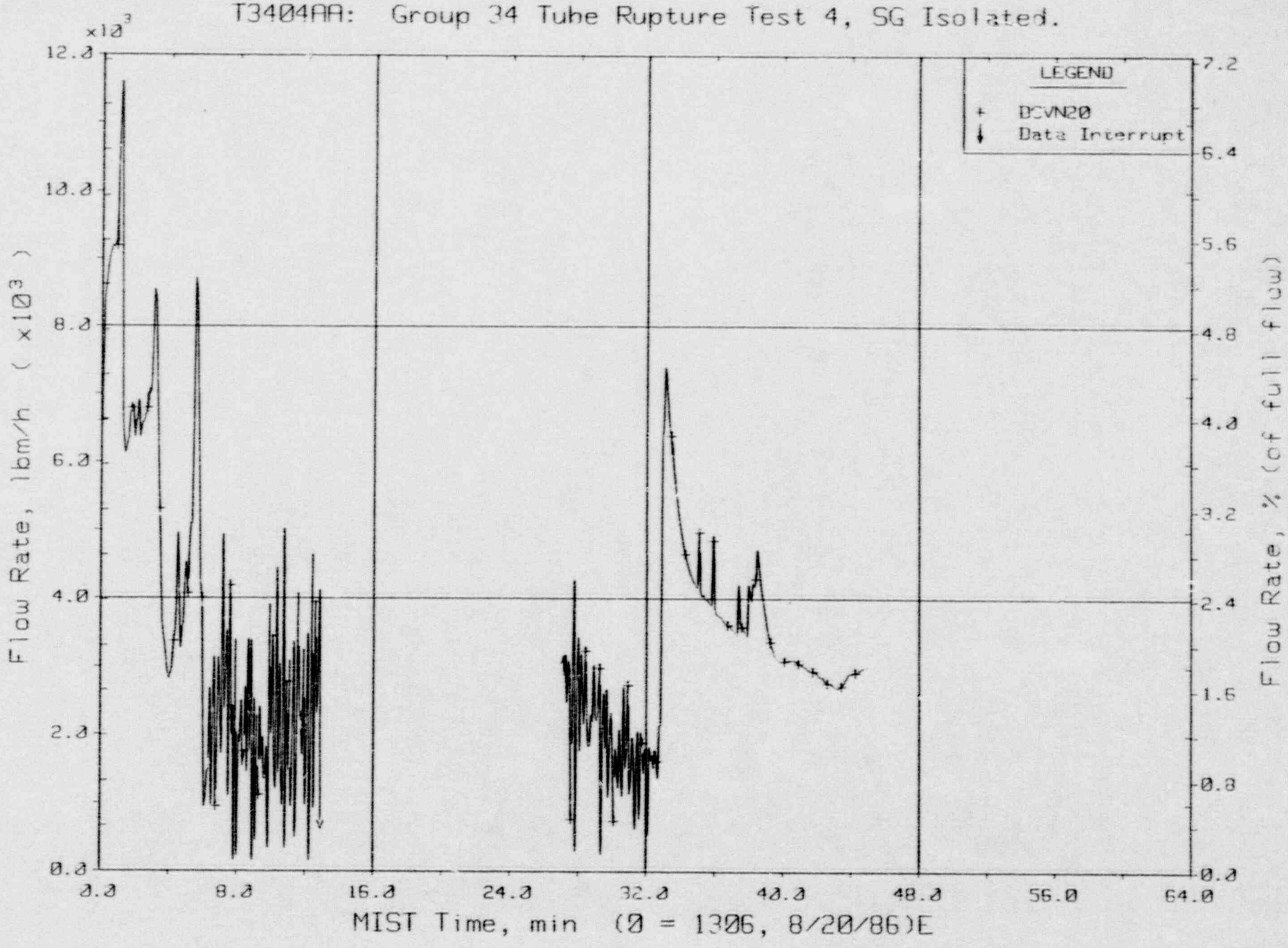
Core Region Collapsed Liquid Levels.

Wed Jun 1 09:11:37 1988

COLV1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



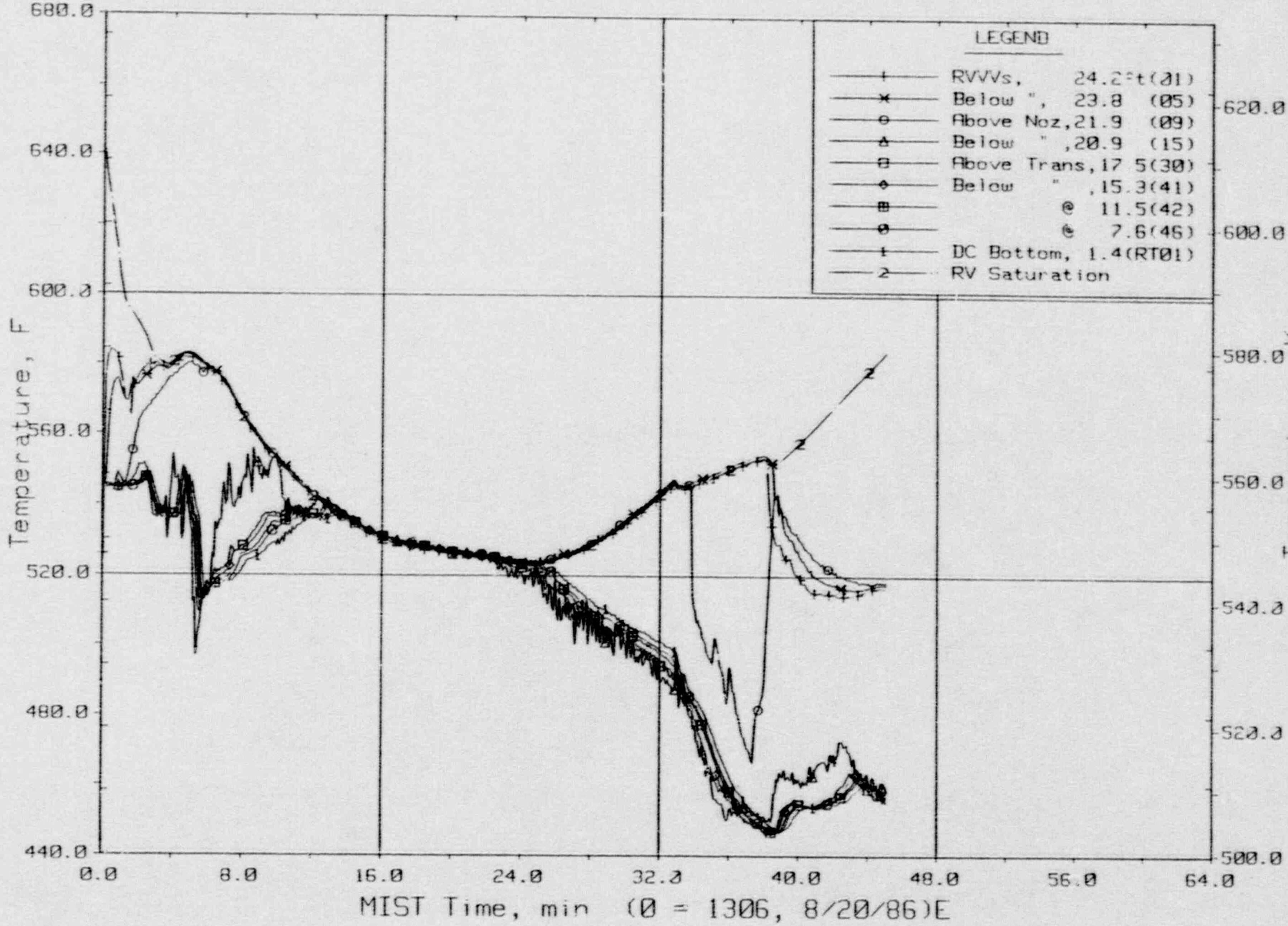
Downcomer (Ventüri) Flow Rate.

Wed Jun 1 09:15:44 1988

DCFL1

FINAL.DATA

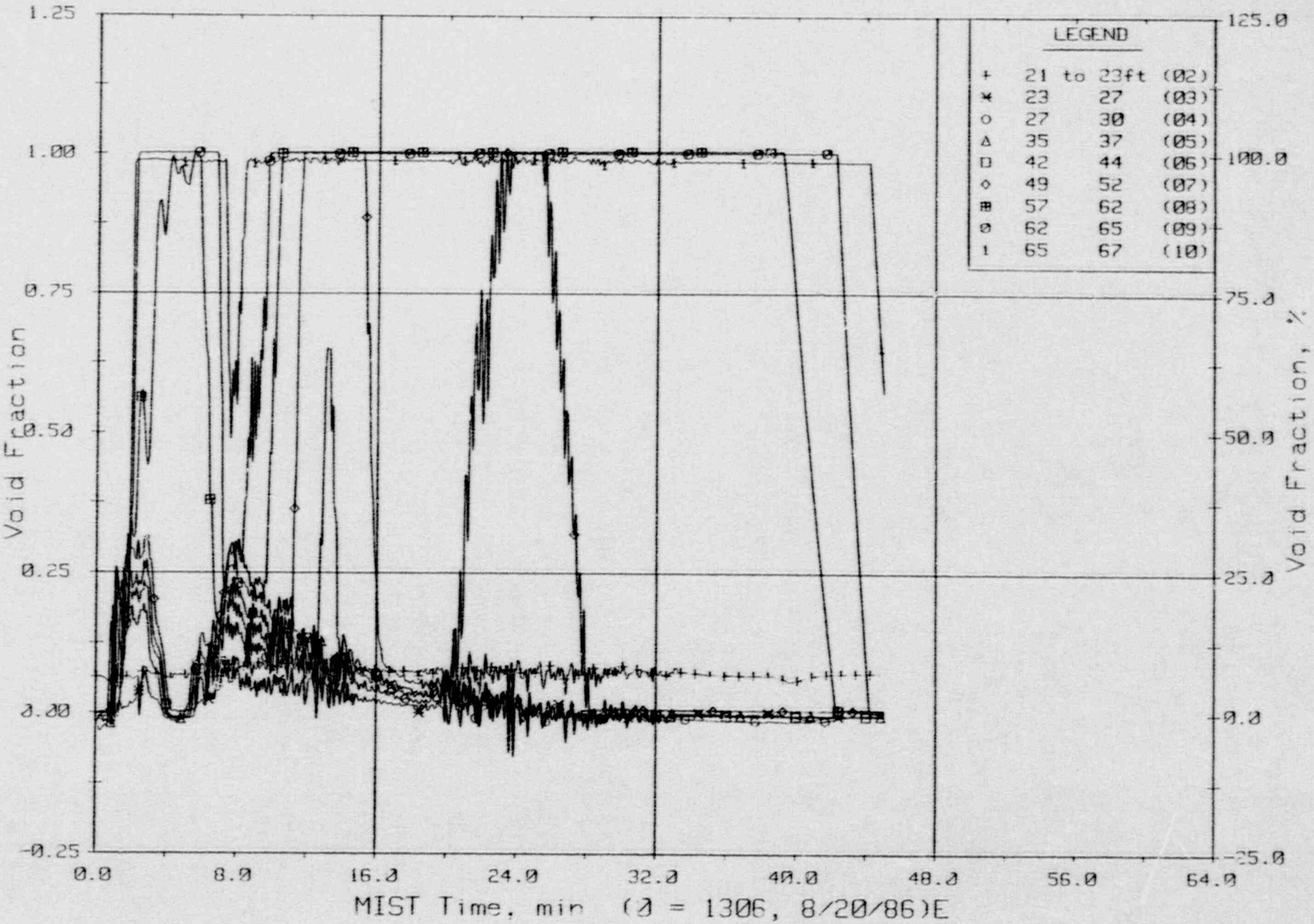
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Downcomer Quadrant A1 Flüid Temperatures (DCTCs).

FINAL DATA

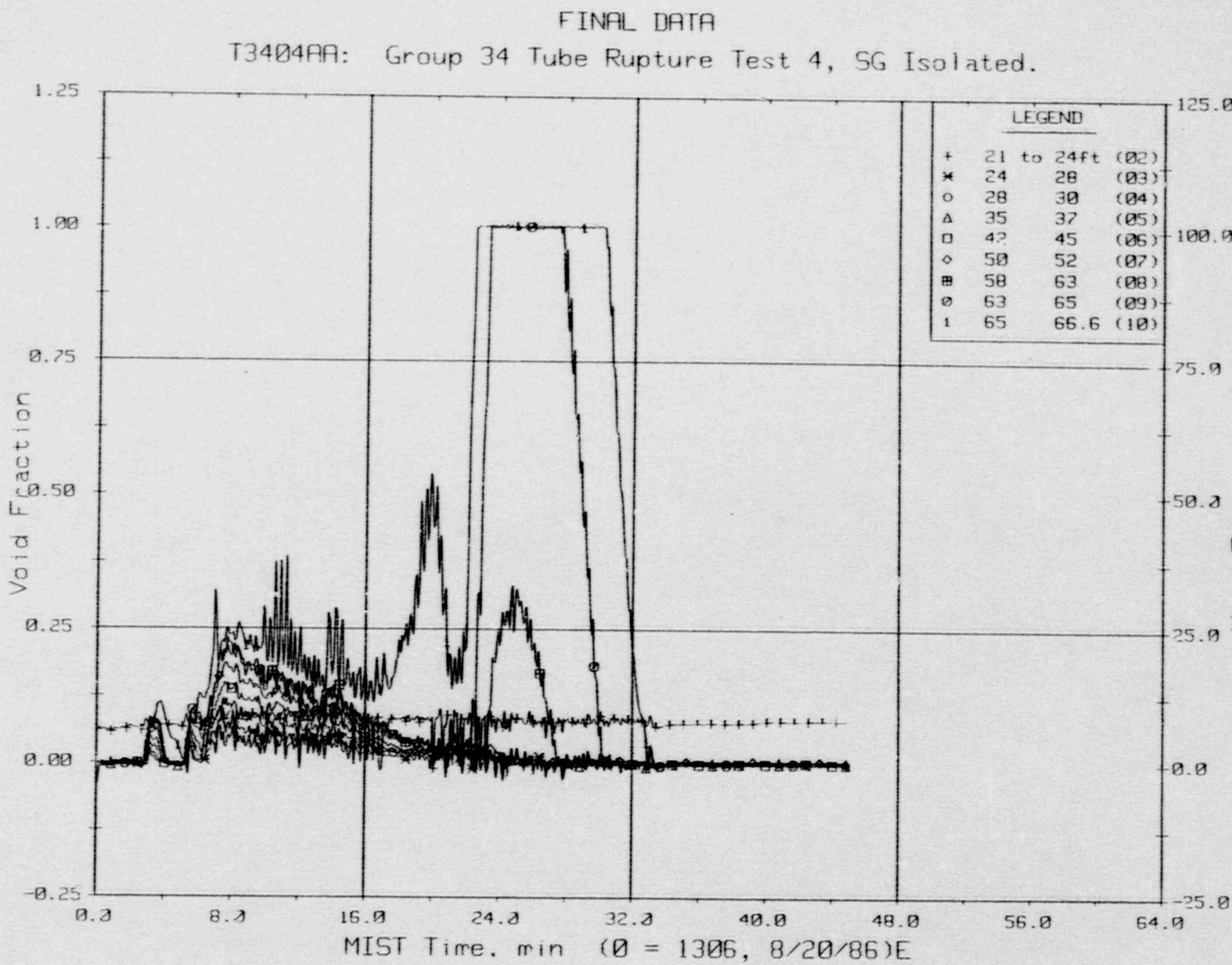
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

Wed Jun 1 09:26:56 1988

H1VF1



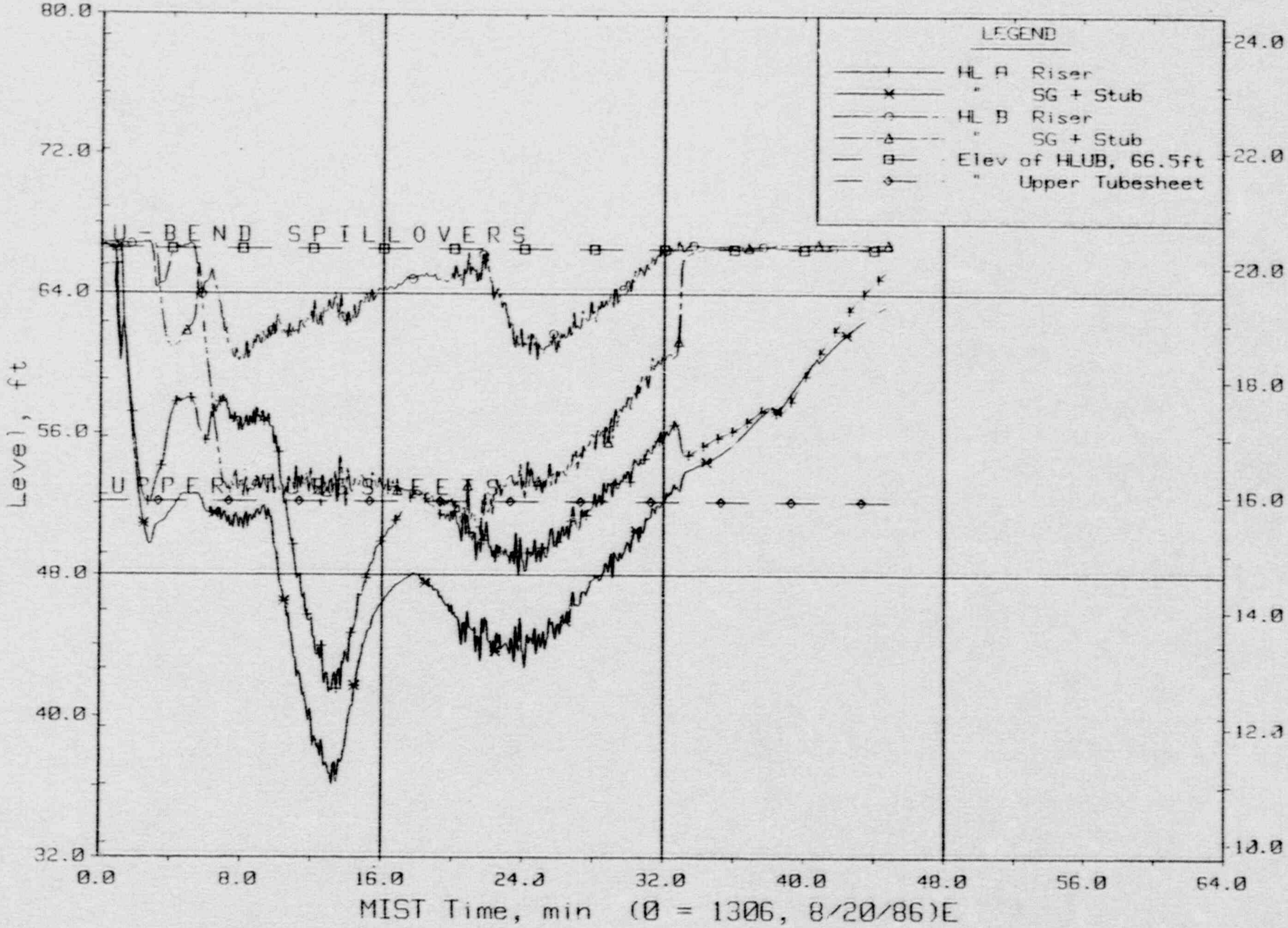
Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

Wed Jun 1 09:21:37 1988

H2VF1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



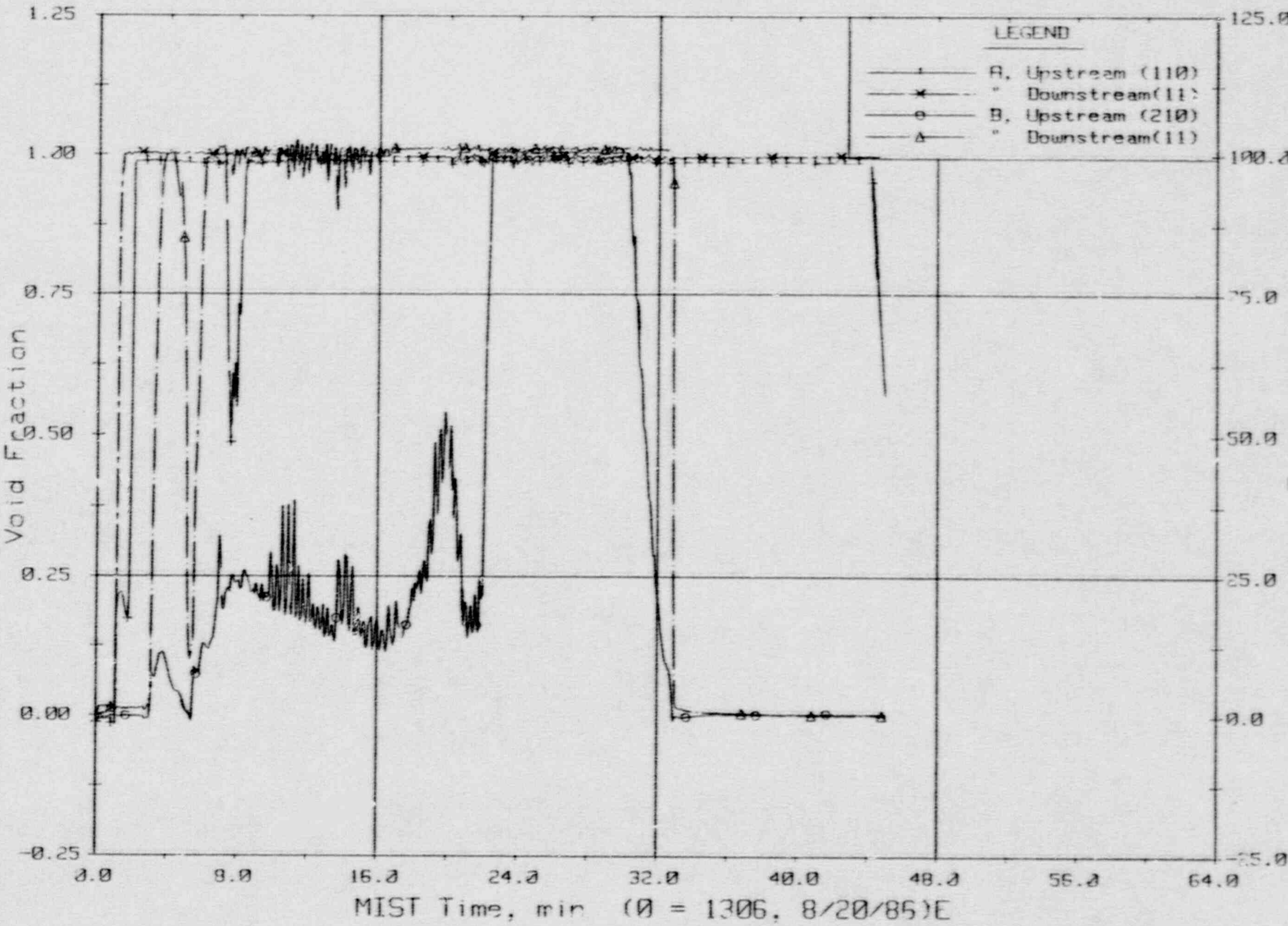
Hot Leg Riser and Stub Collapsed Liquid Levels.

Wed Jun 1 09:35:46 1988

HLLV1

FINAL DATA

T3404AF: Group 34 Tube Rupture Test 4, SG Isolated.



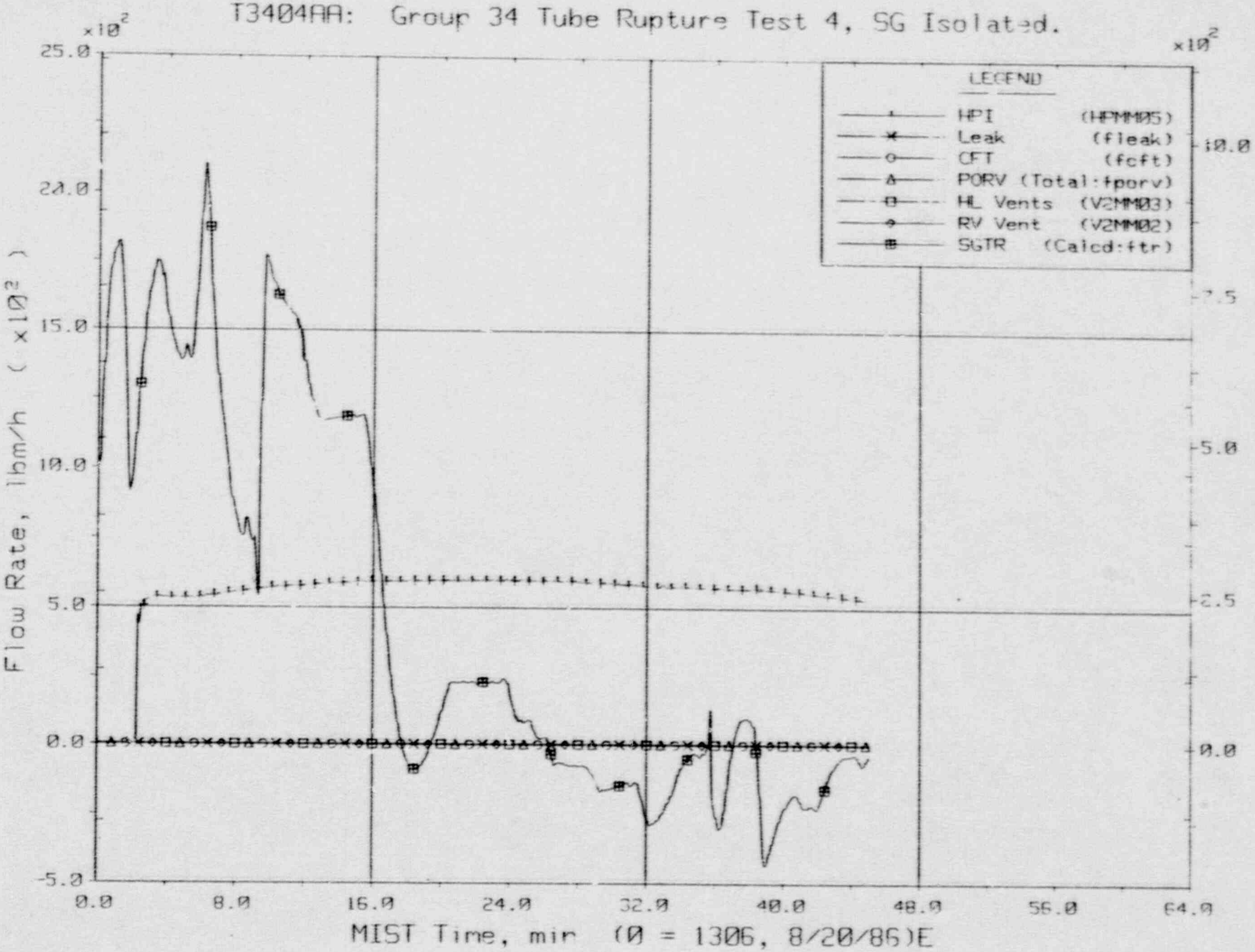
Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Wed Jun 1 09:41:02 1988

HLVF1

'FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



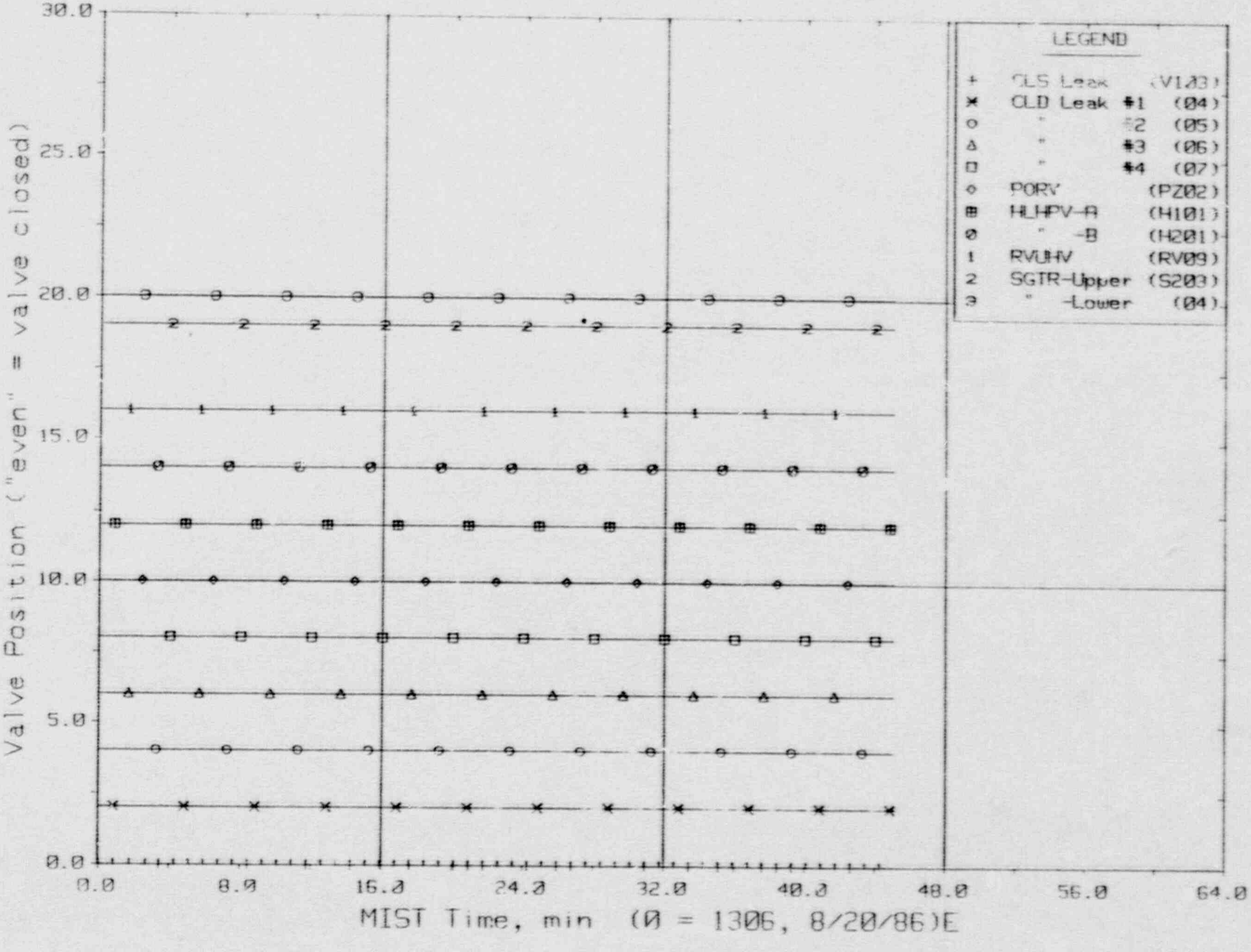
Primary System Boundary Flow Rates.

Wed Jun 1 09:46:11 1988

PBFL3

FINAL DATA

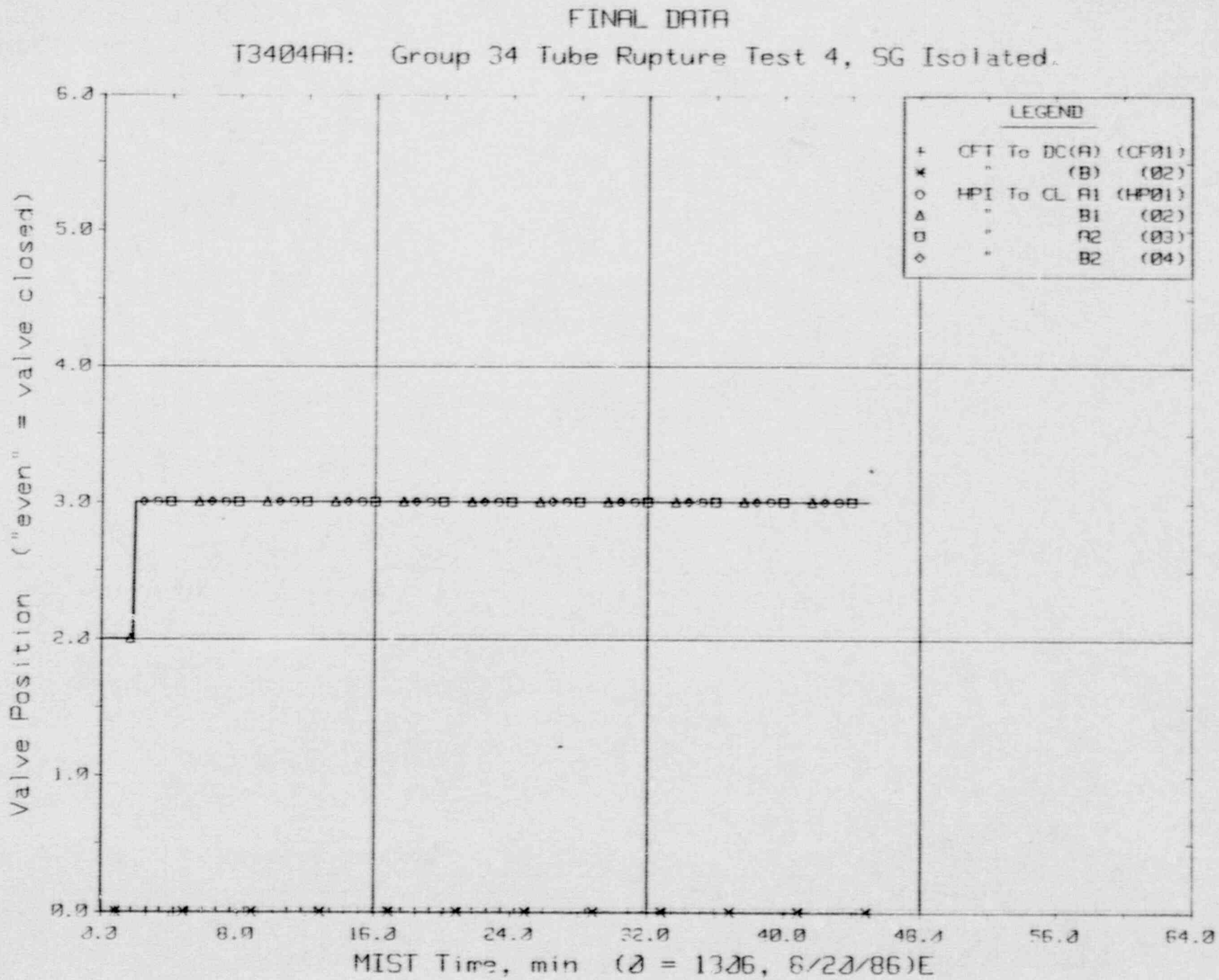
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Primary System Discharge Limit Switch Indications (LSs).

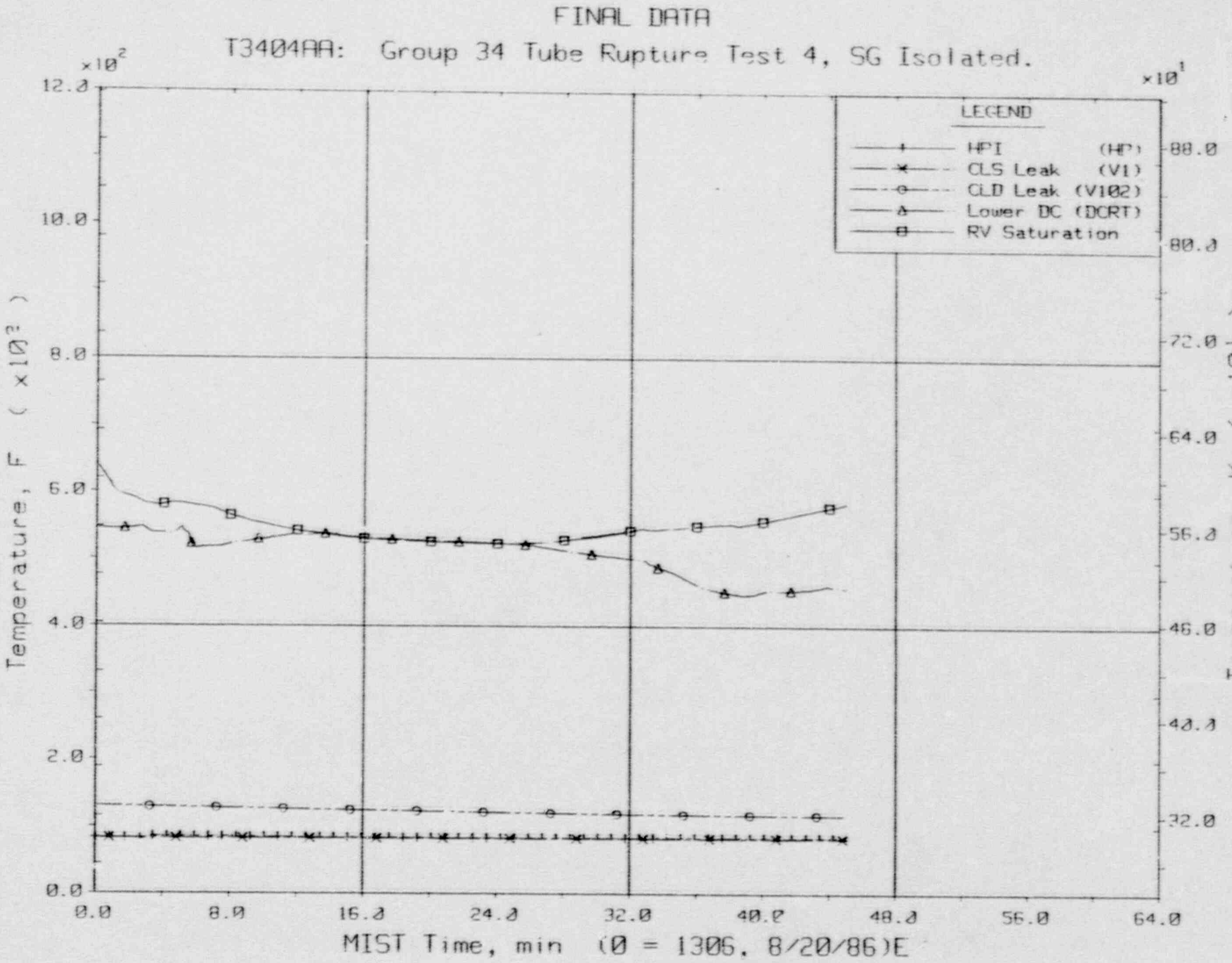
Wed Jun 1 09:51:13 1988

PBLS1



Primary System Injection Limit Switch Indications (LSs).

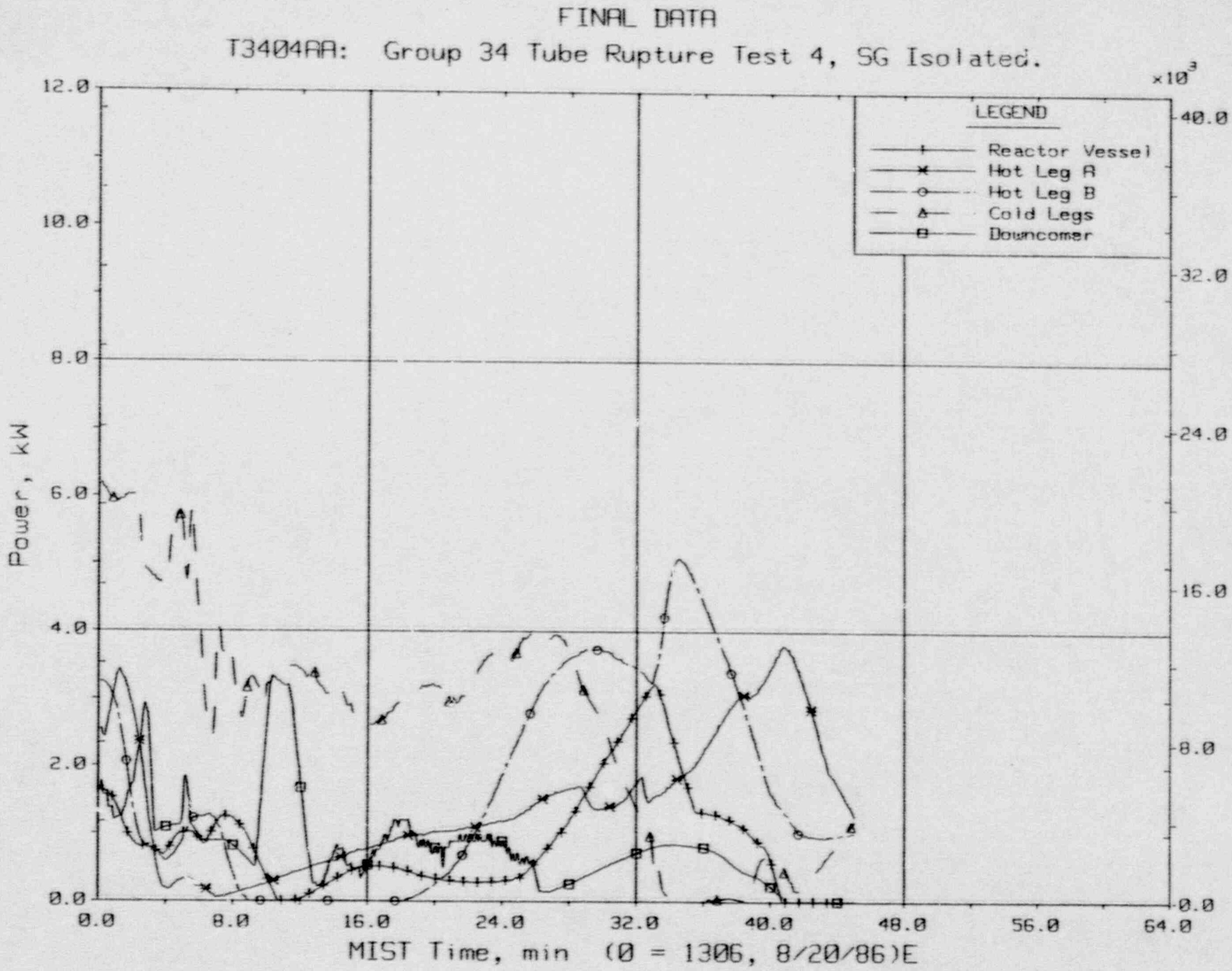
PBLS2



Single-Phase Discharge and HPI Fluid Temperatures (TC01s).

Wed Jun 1 09:57:40 1988

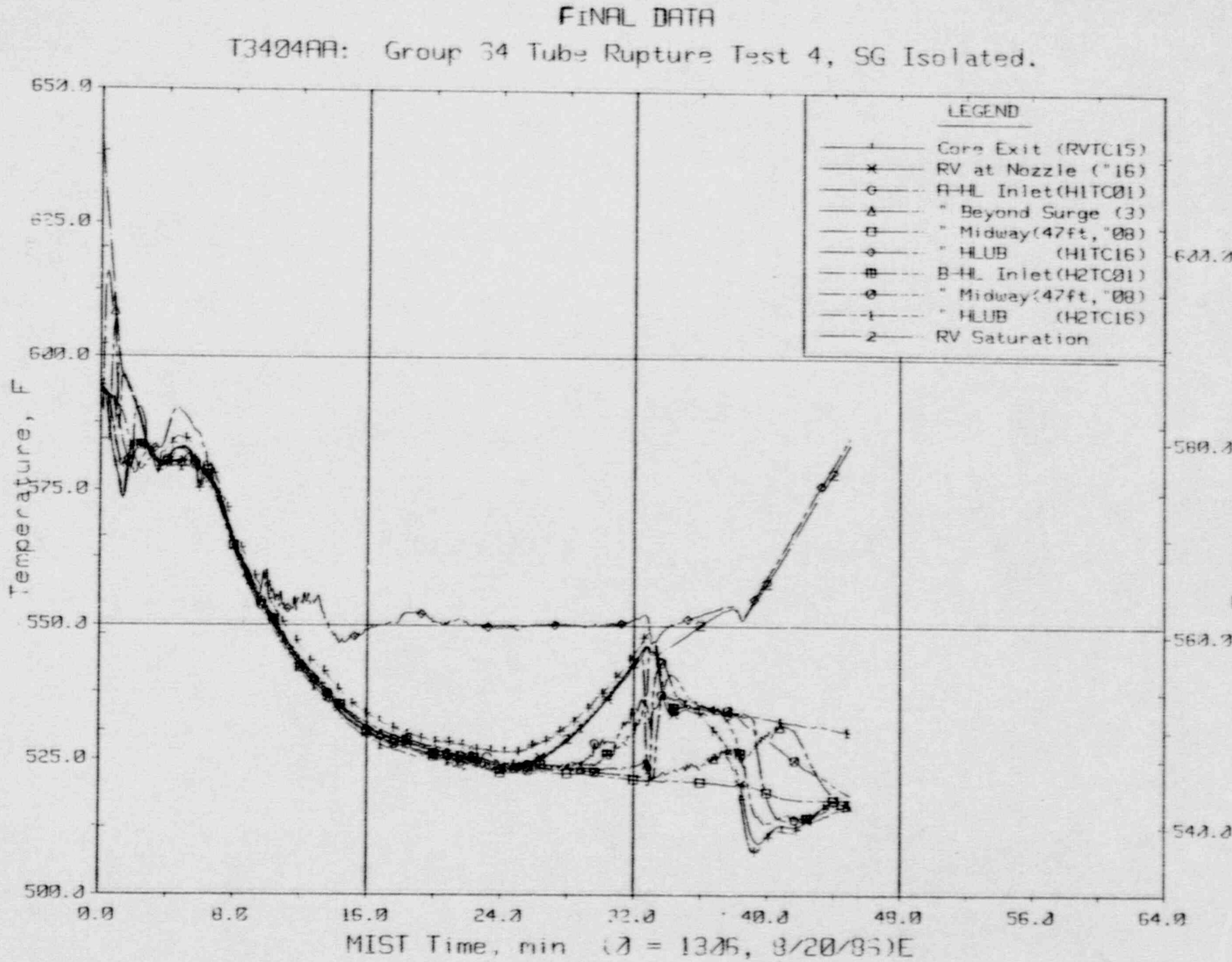
PBTC1p



Guard Heater Specified Power Per Primary Component.

Wed Jun 1 13:41:04 1988

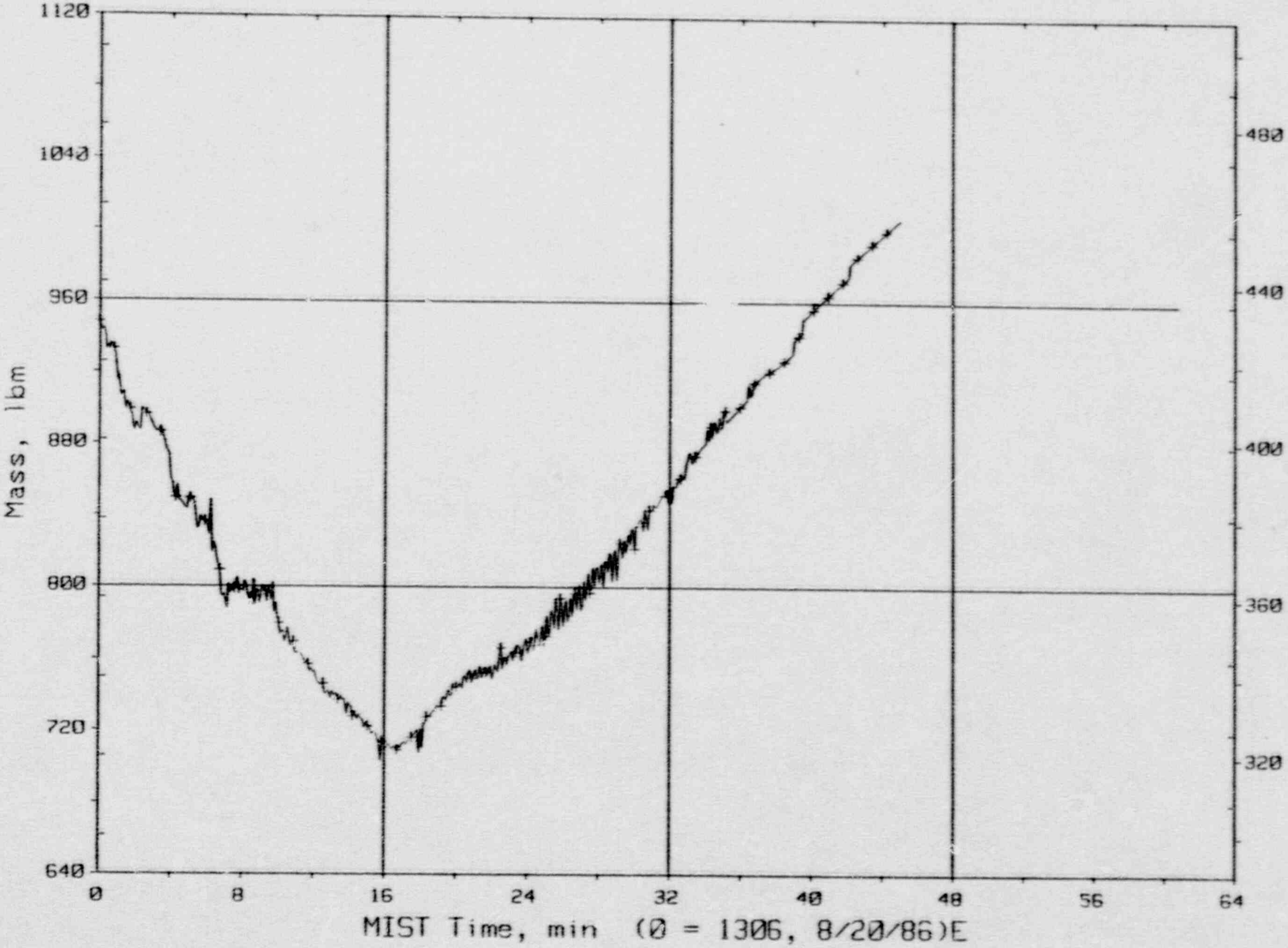
PRGQ1



Composite Core Exit and Hot Leg Fluid Temperatures.

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



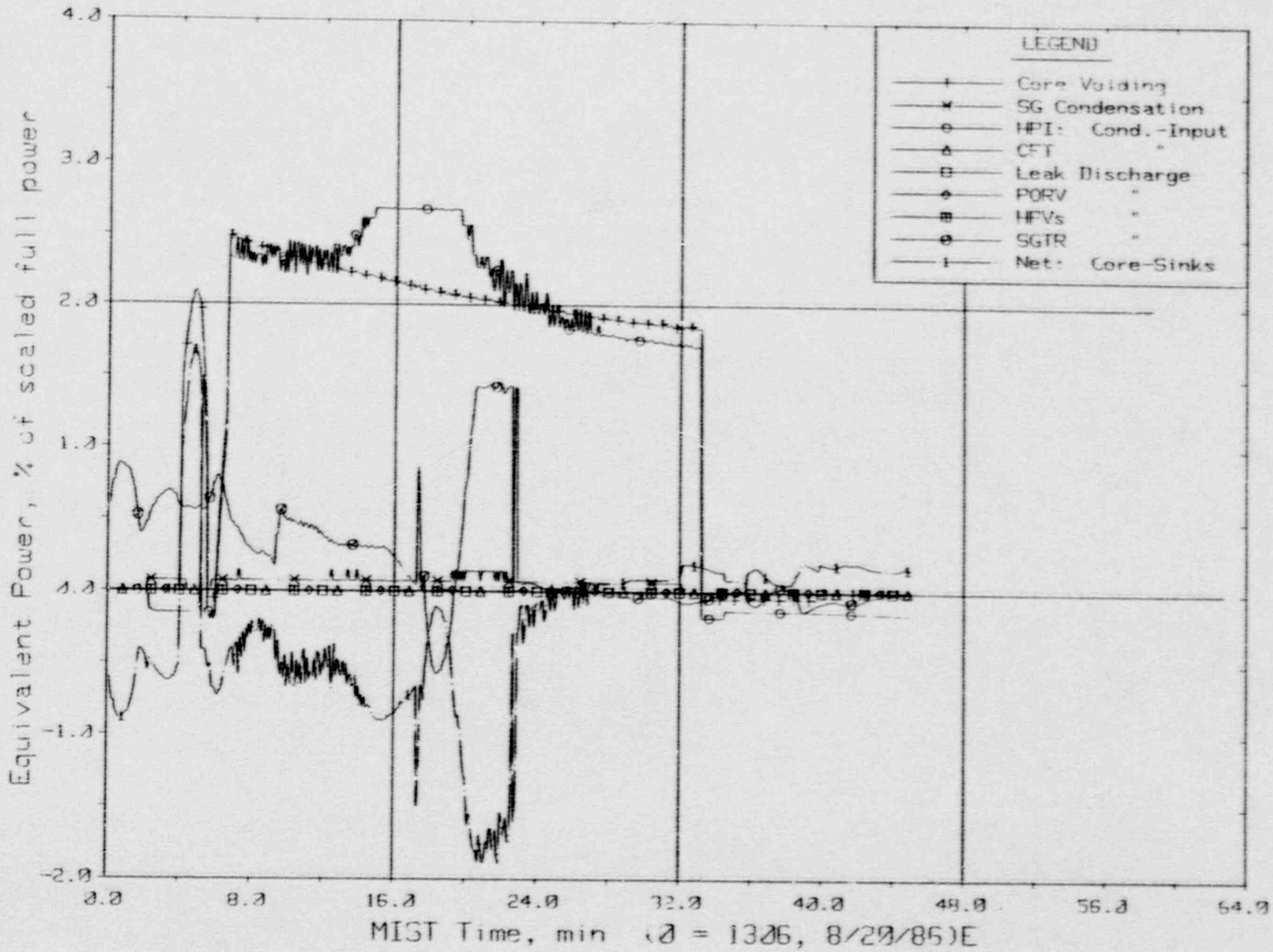
Indicated Primary System Total Fluid Mass (PLML20).

Wed Nov 2 14:06:41 1988

PRML9

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



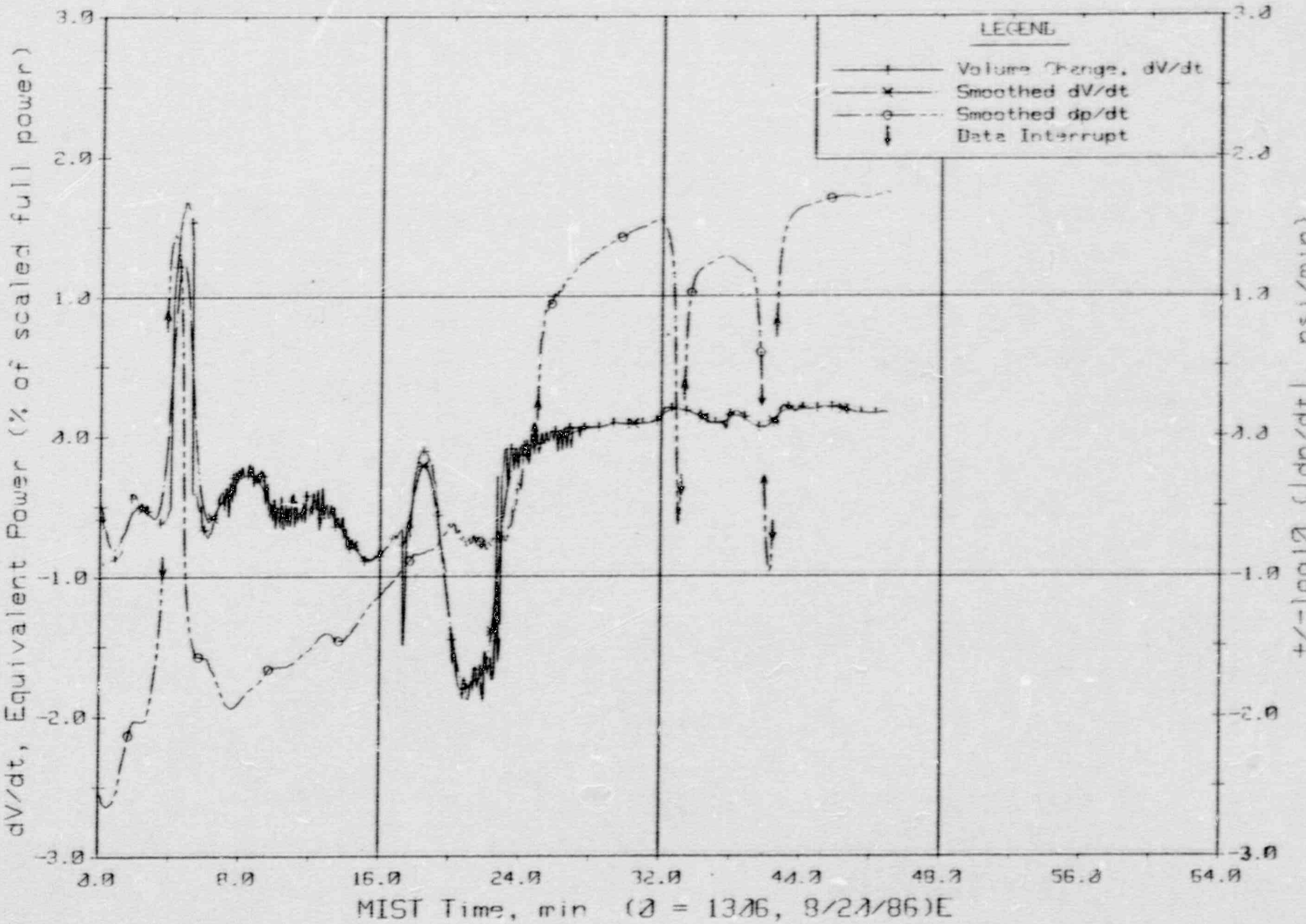
Primary Fluid Volume Changes By Components.

Wed Jun 1 10:17:46 1998

PRPA0

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



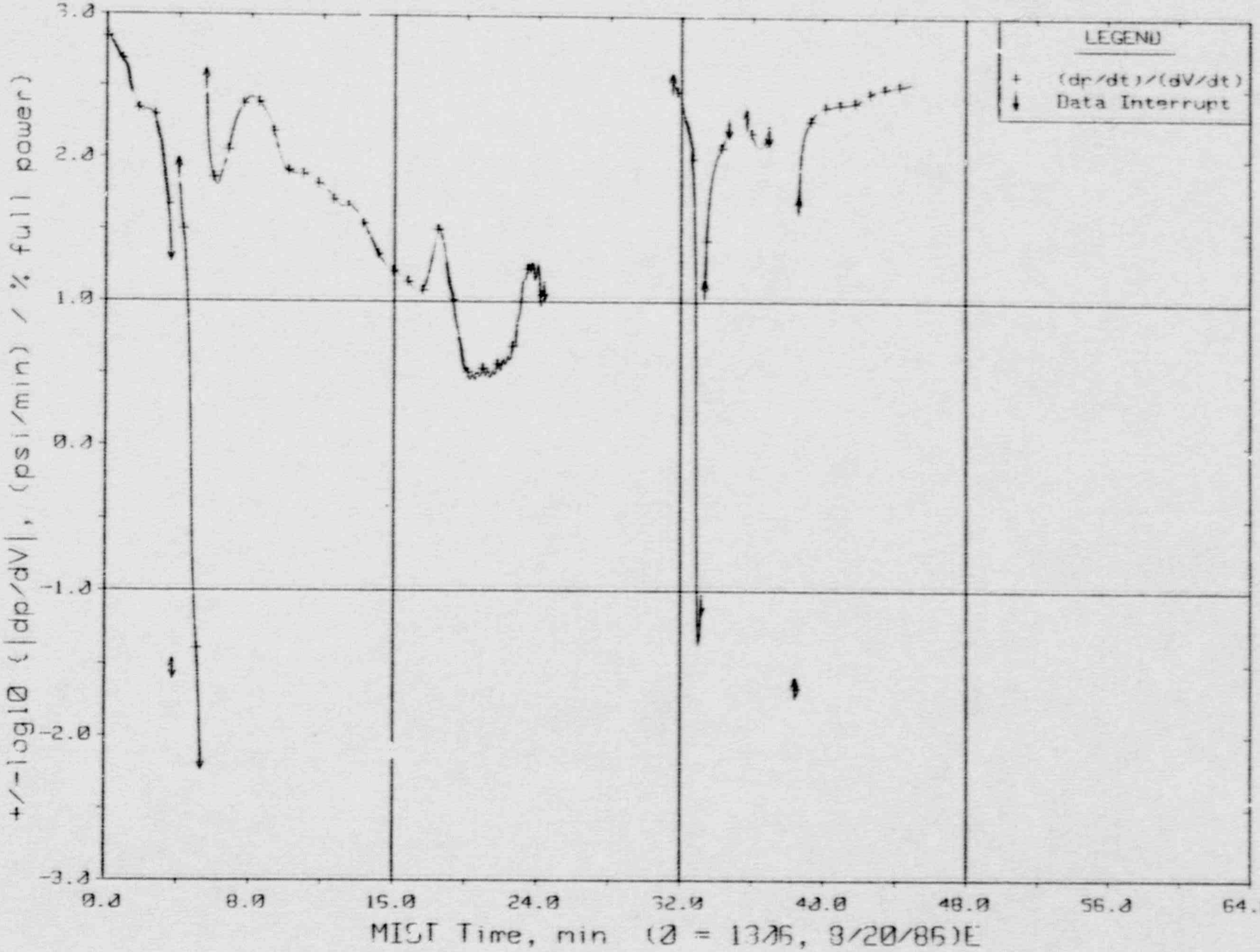
Primary Fluid Volume and Pressure Changes, dV/dt and dp/dt.

Wed Jun 1 10:20:57 1988

PRPA2

FINAL DATA

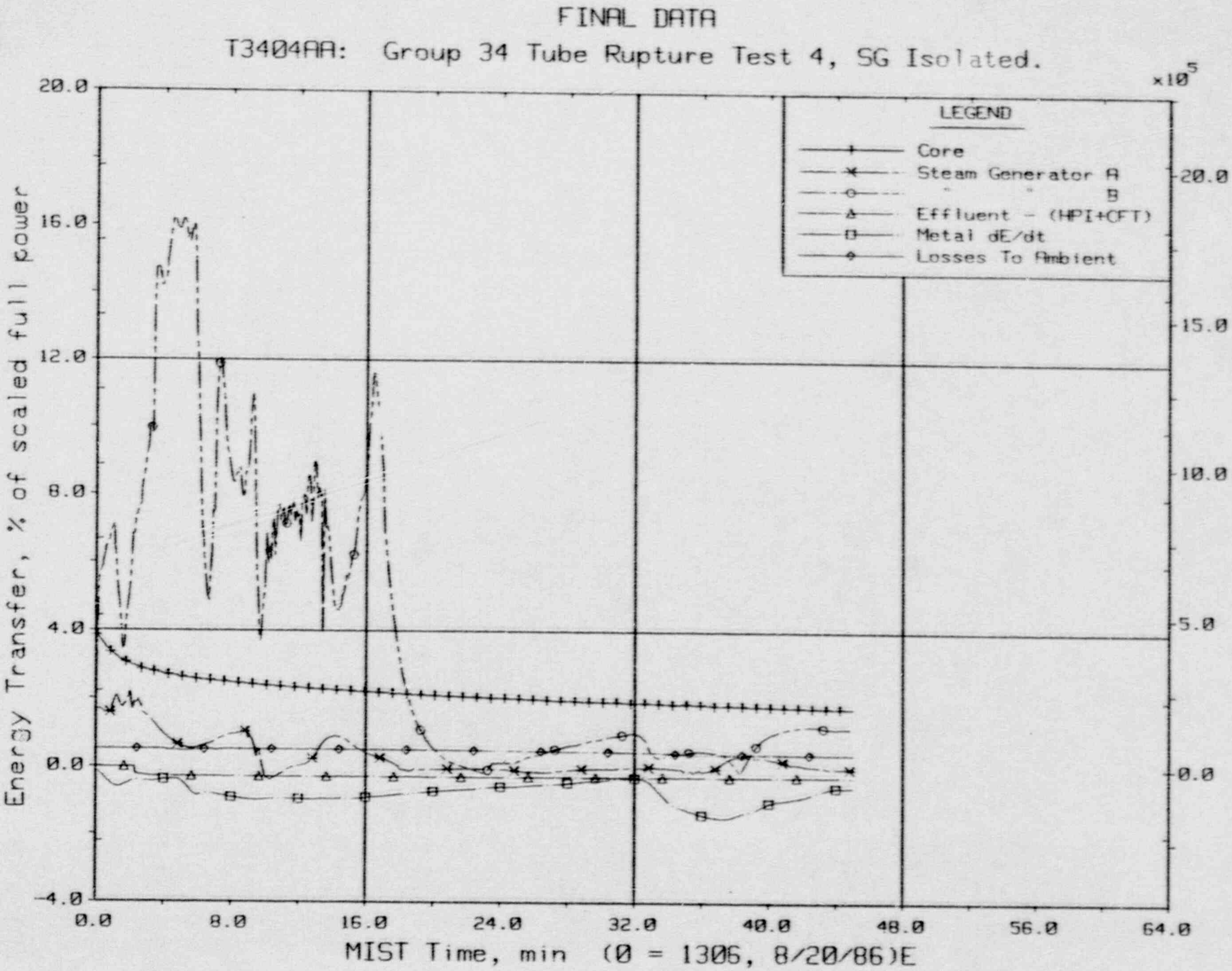
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Primary System Response: dp/dt divided by dV/dt.

Wed Jun 1 18:22:24 1994

PRPA3



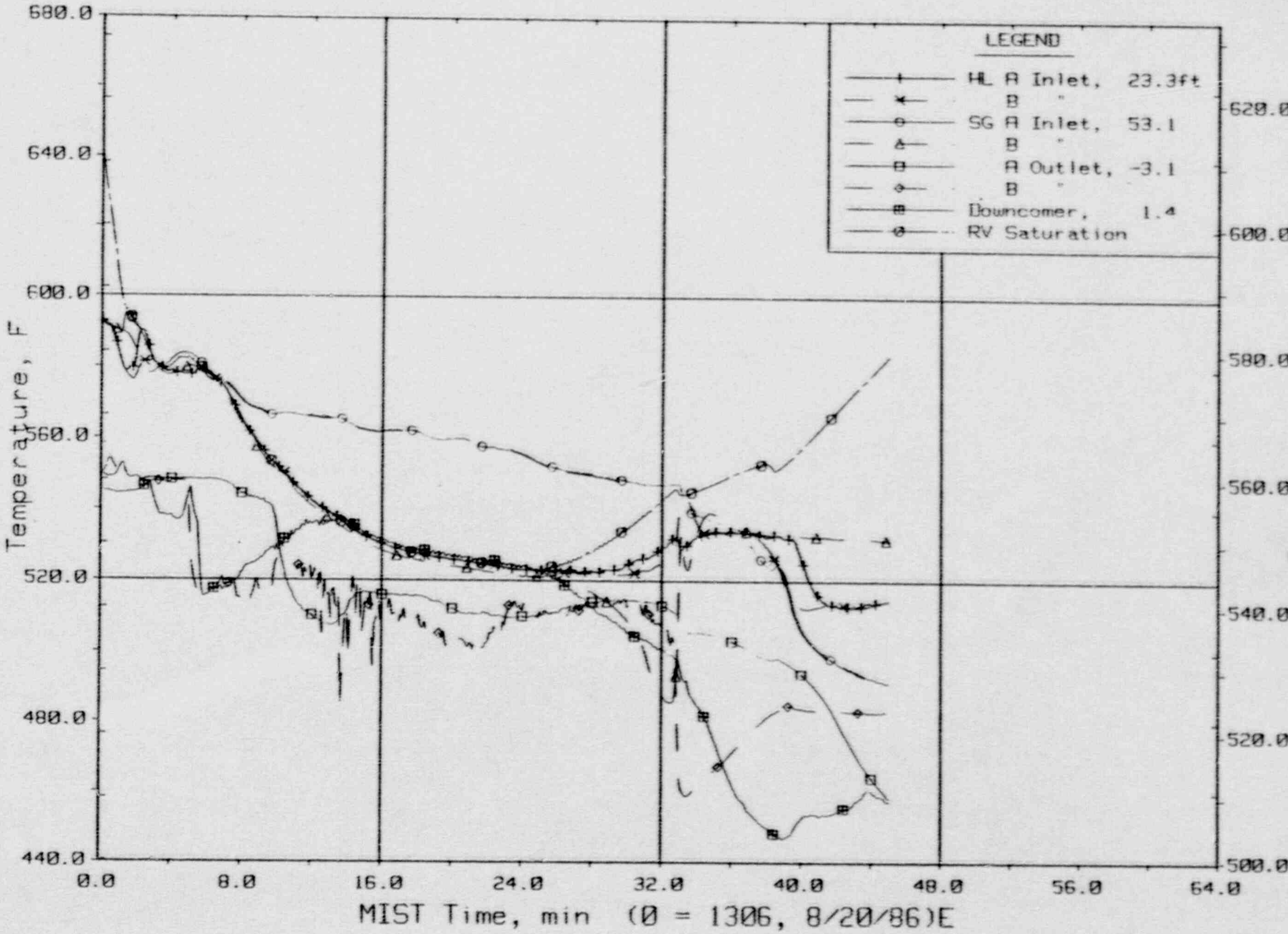
Primary System Energy Transfer.

Wed Jun 1 10:27:48 1988

PRQE2

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



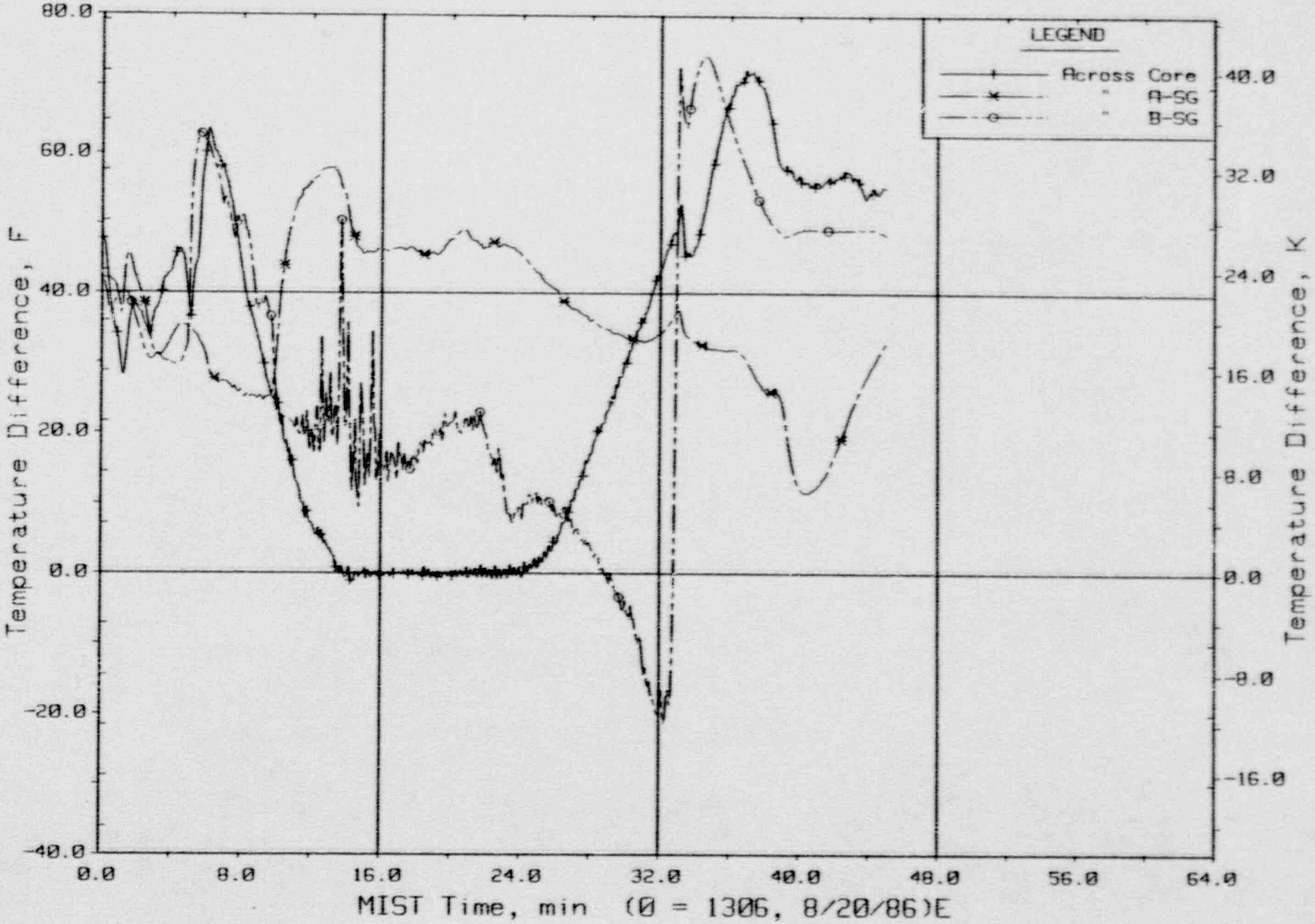
Primary System Fluid"Temperatures (RTDs).

Wed Jun 1 10:33:11 1988

PRRT0

FINAL DATA

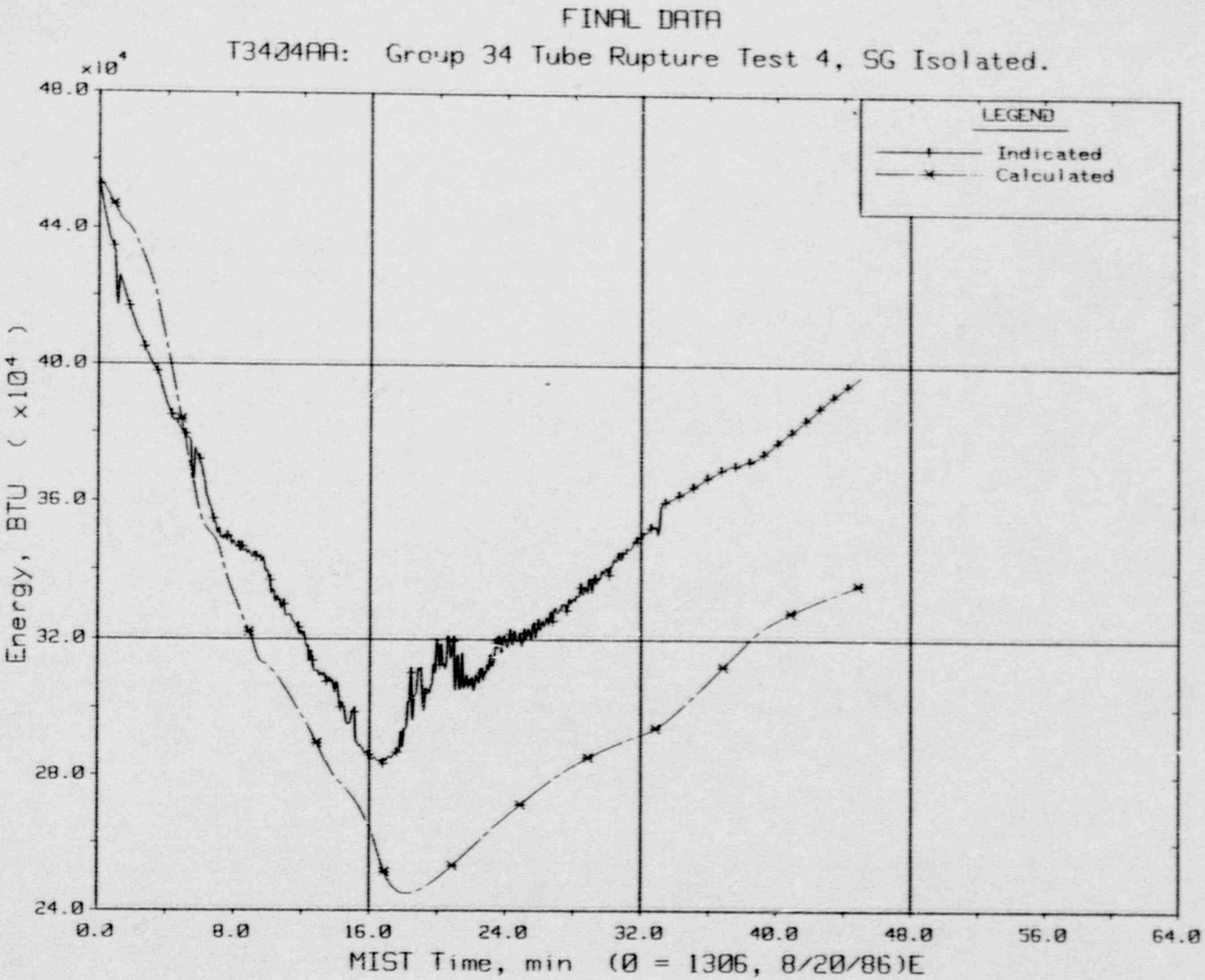
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



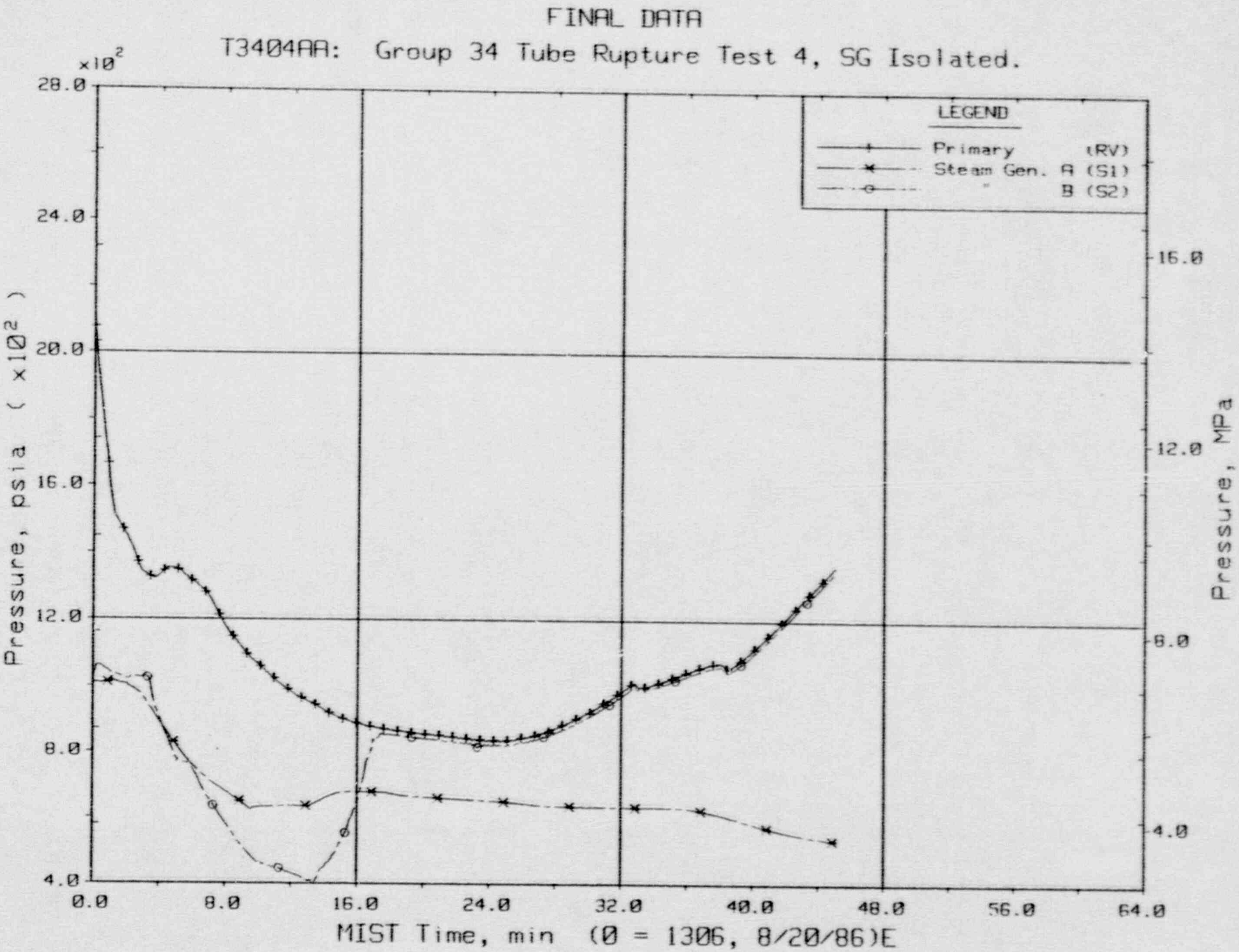
Key Temperature Differences.

Wed Jun 1 10:36:31 1988

PRTD2



Primary System Total Fluid Energy.



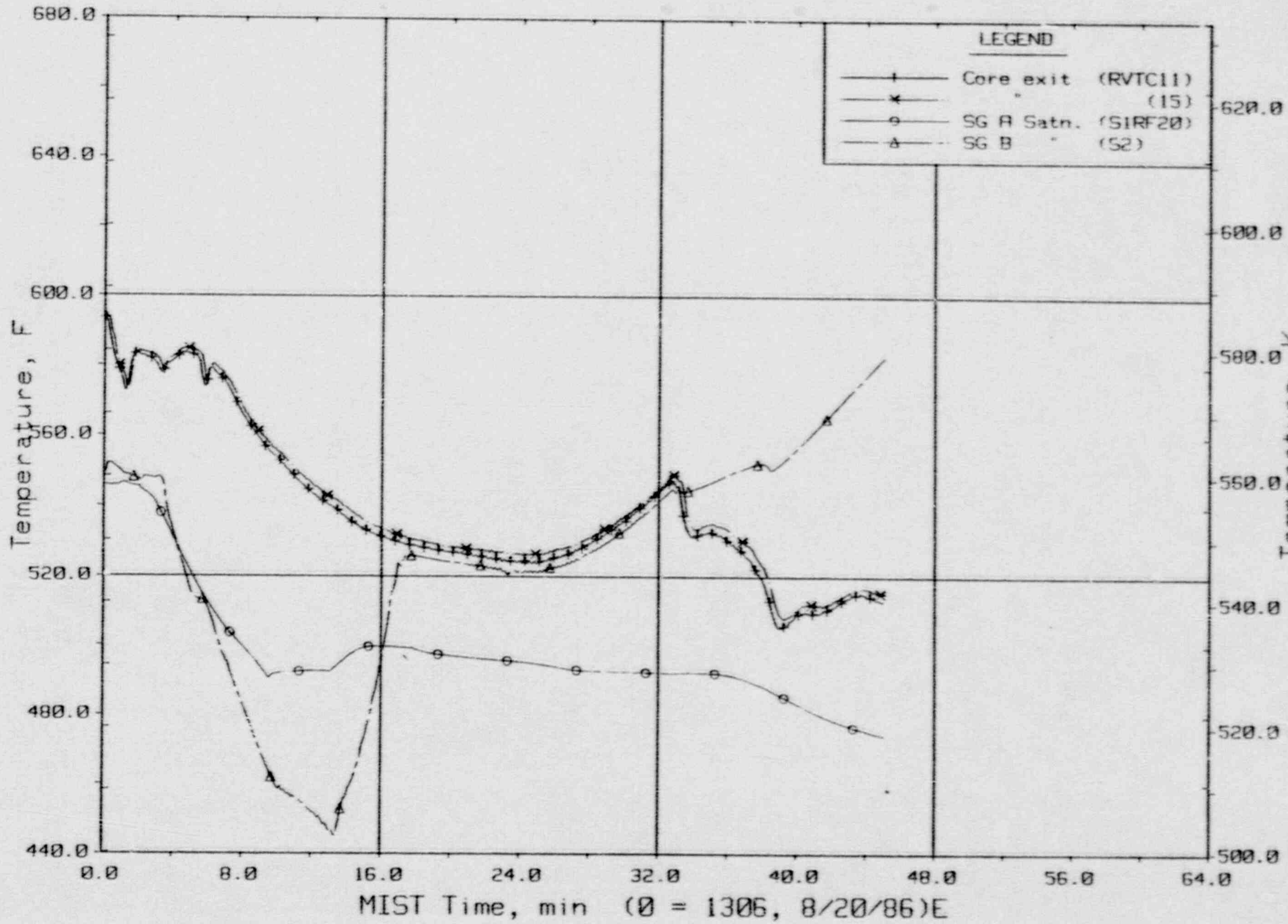
Primary and Secondary System Pressures (GP01s).

Wed Jun 1 10:45:29 1988

PSGP1

FINAL DATA

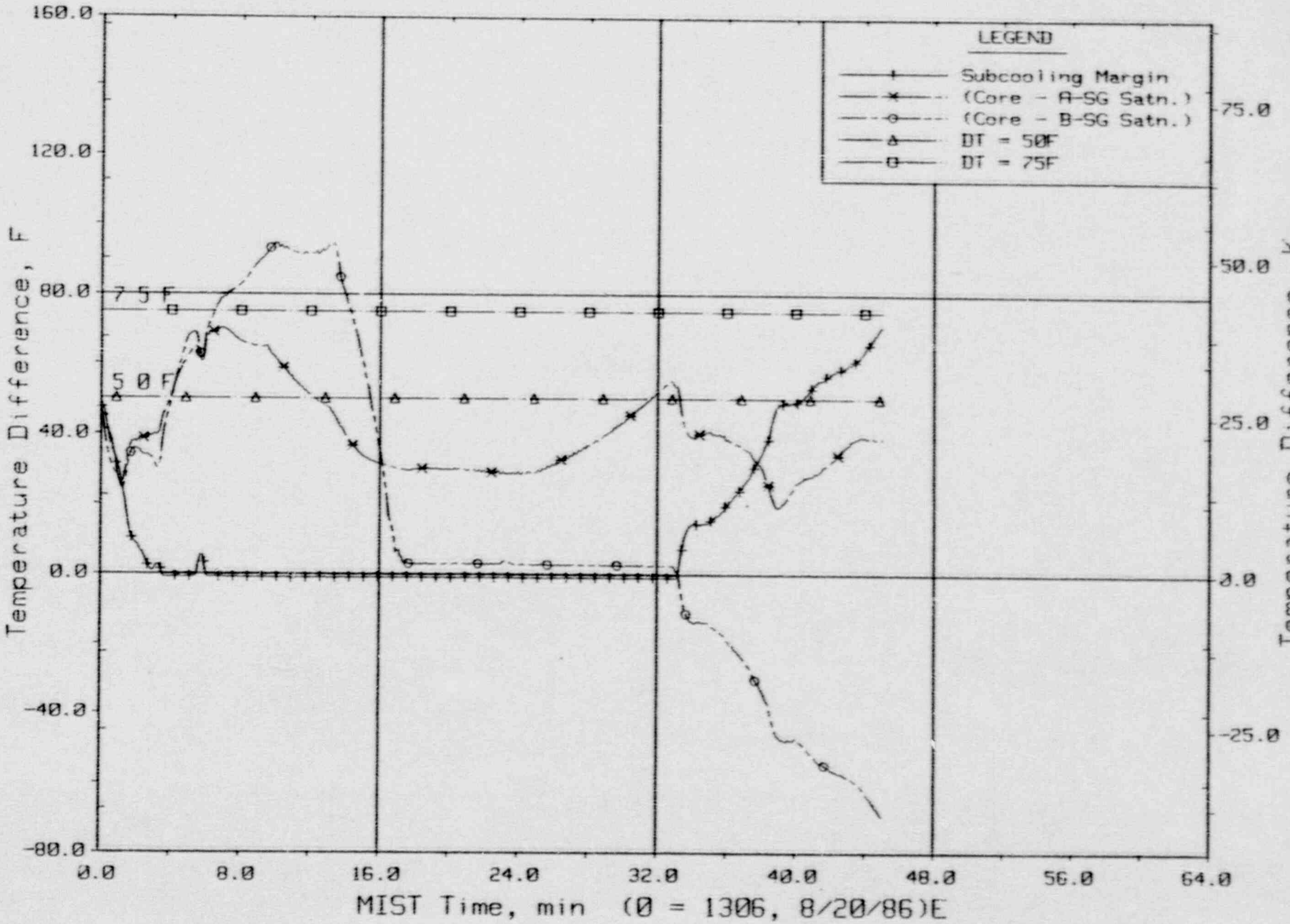
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Core Exit and Steam Generator Secondary Saturation Temperatures.

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



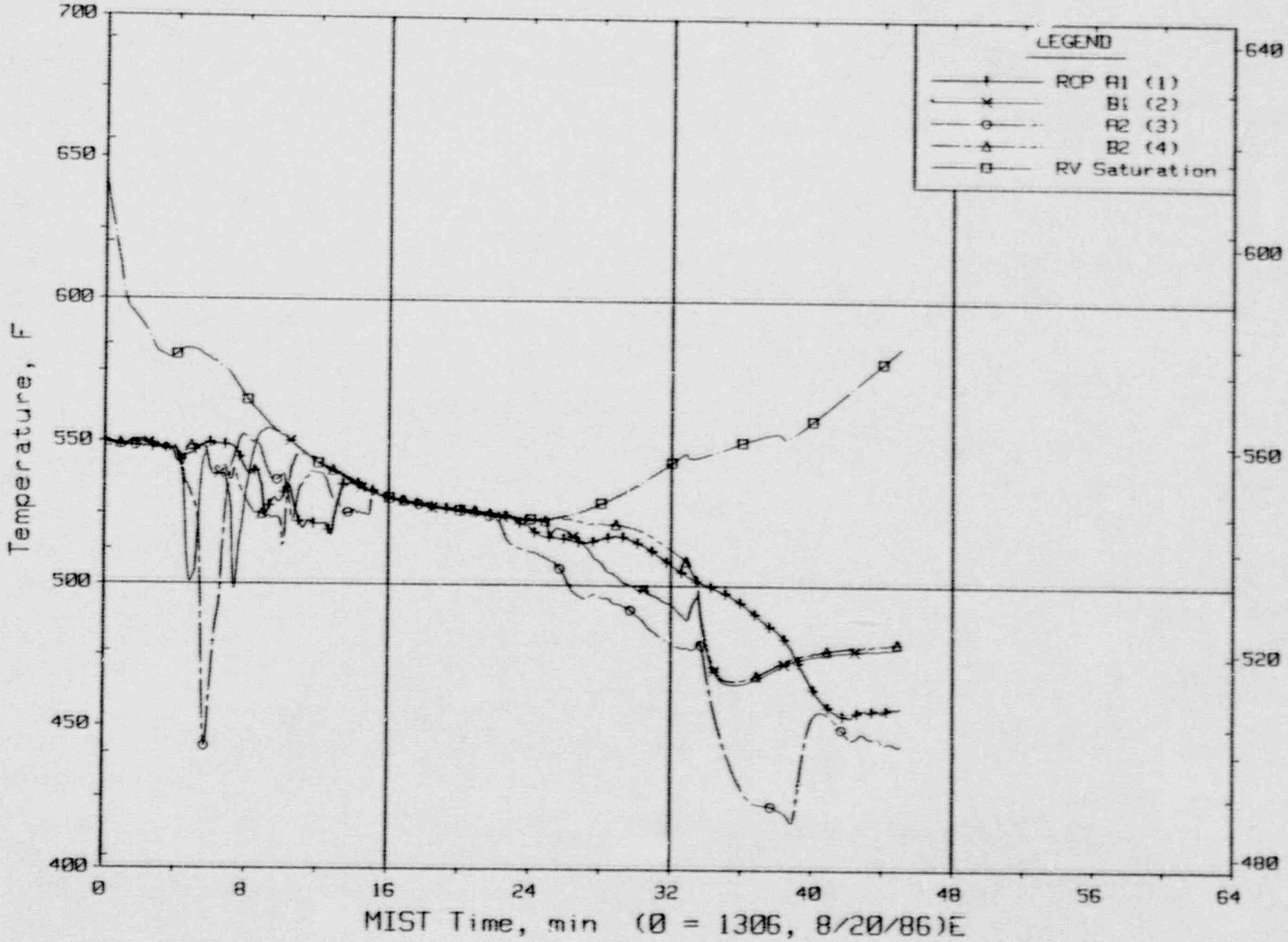
Control Temperature Dif~erences.

Wed Jun 1 10:52:54 198?

PSTD1

FINAL DATA

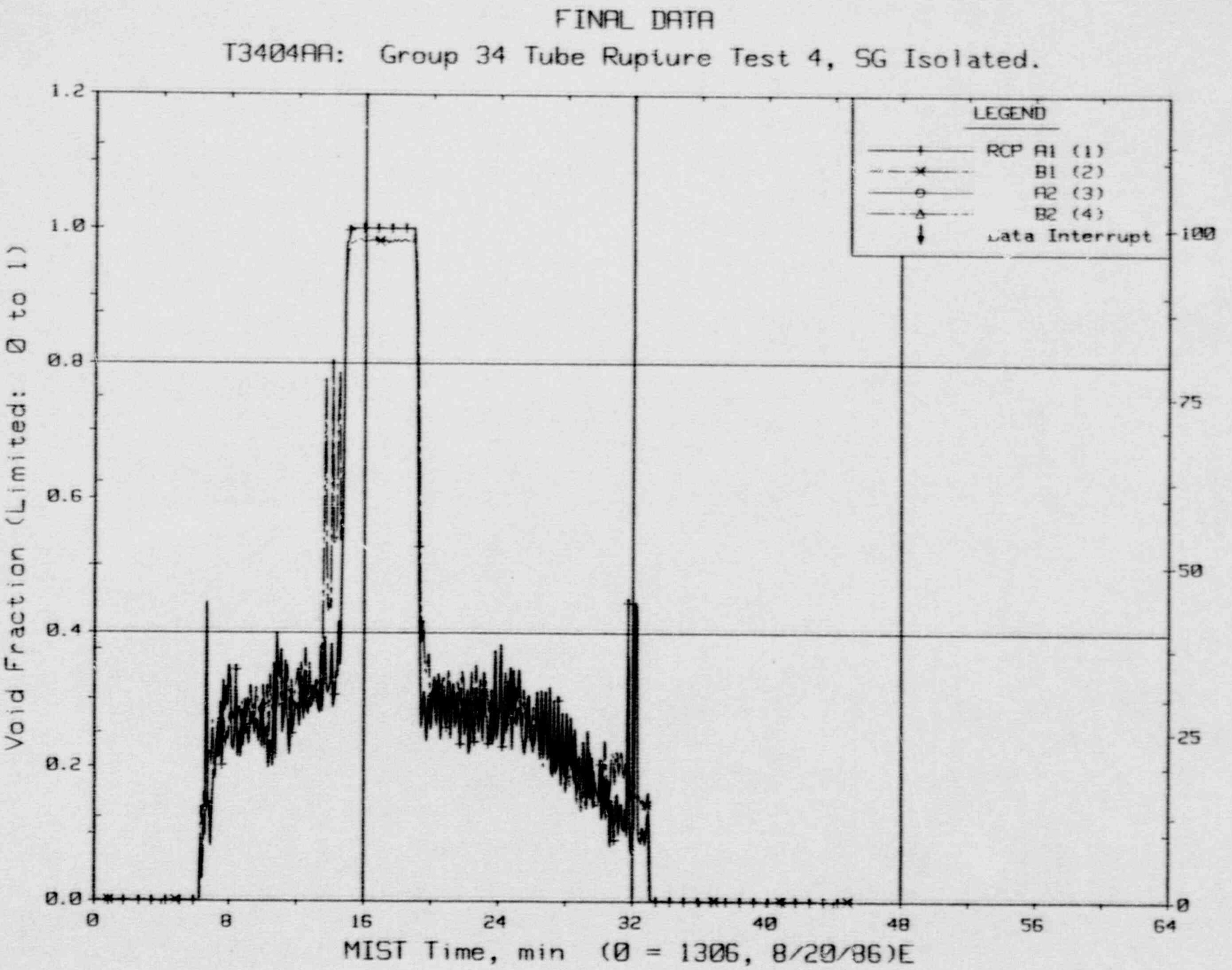
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 18:01:09 1986

PURT0



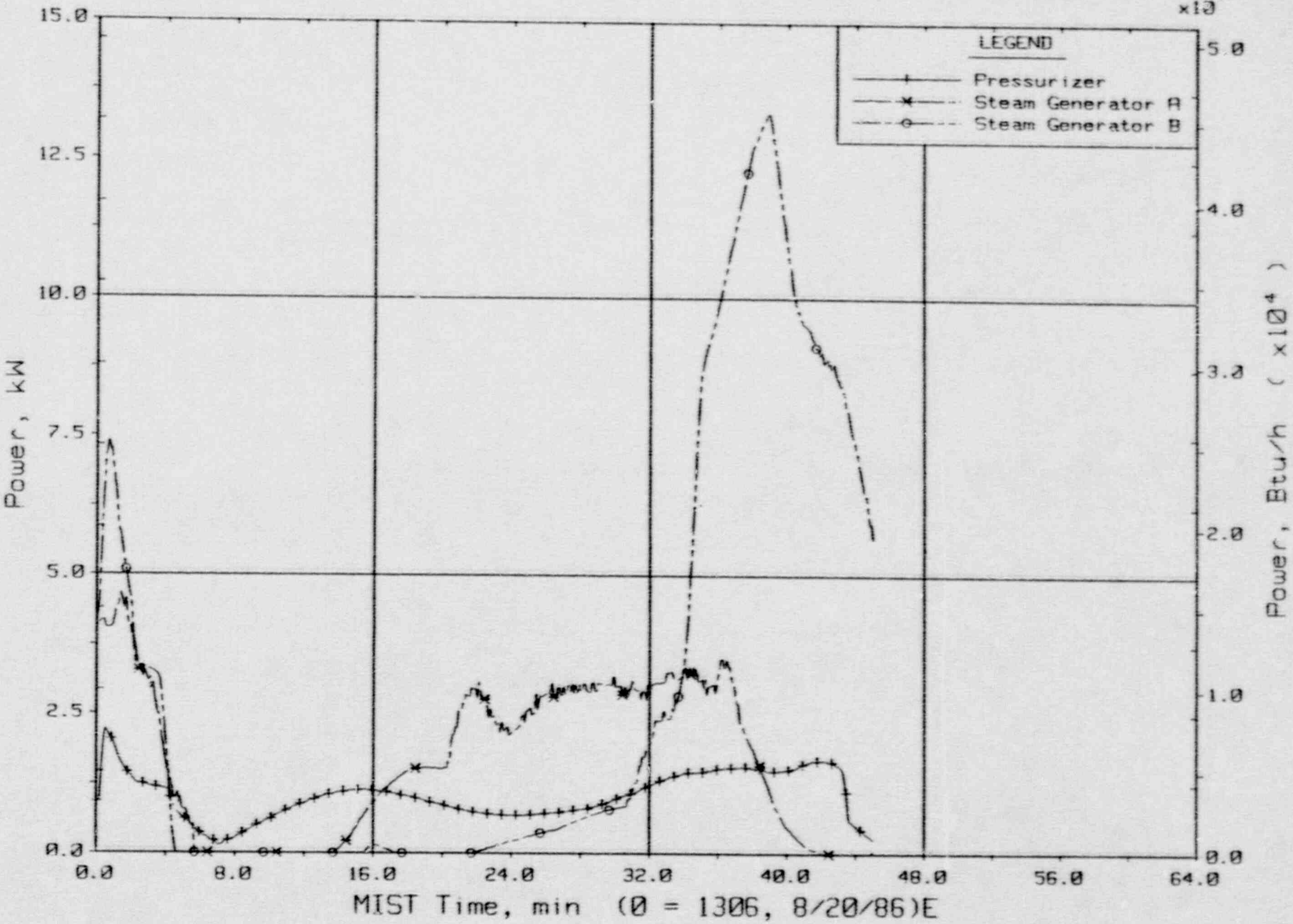
Pump Suction Void Fraction From Gamma Densitometers (CnGD21).

Tue Nov 1 18:03:38 1988

PUVF2

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



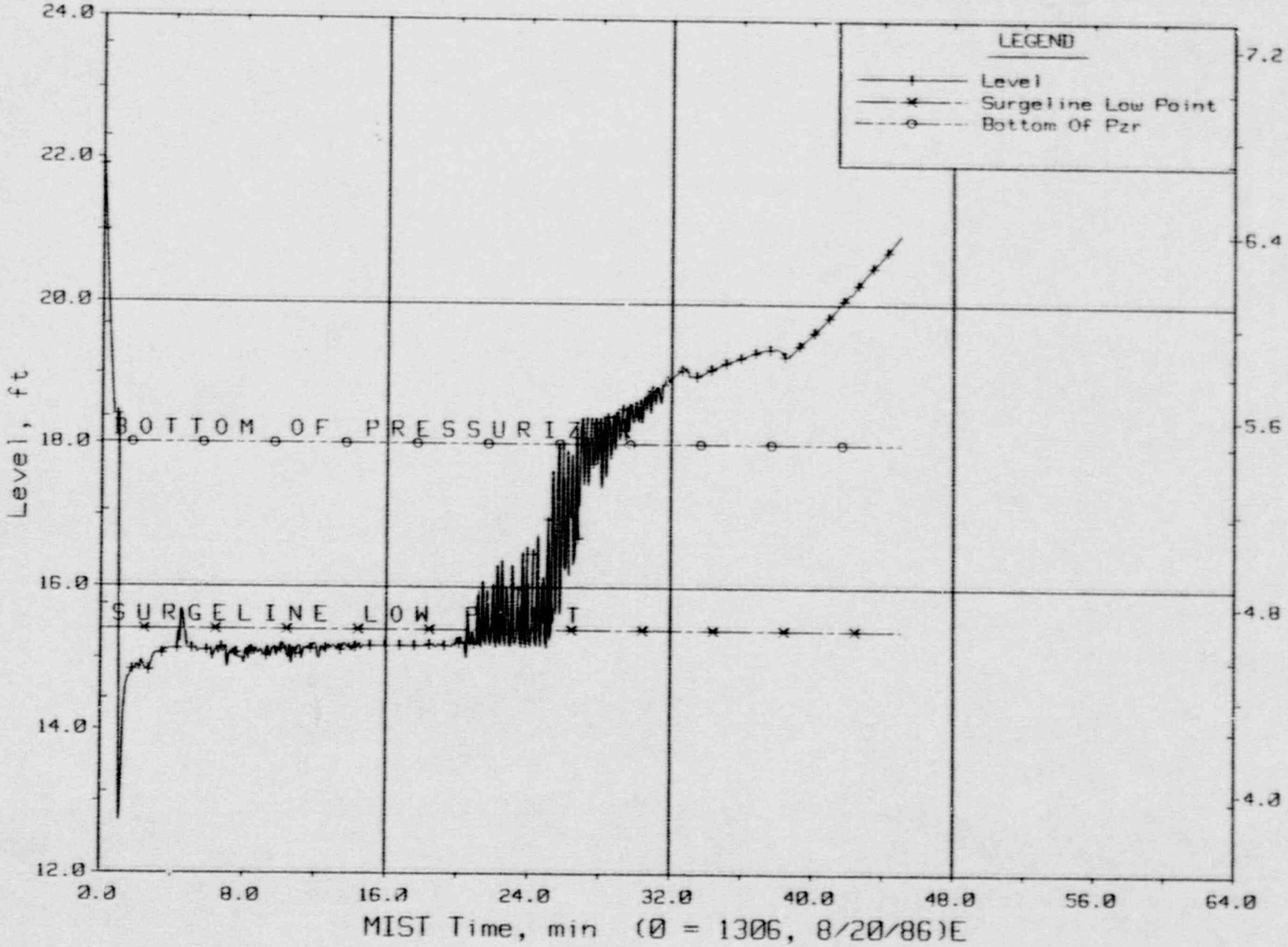
Guard Heater Specified Power, Pressurizer and Steam Generators.

Wed Jun 1 10:59:37 1988

PZG01

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



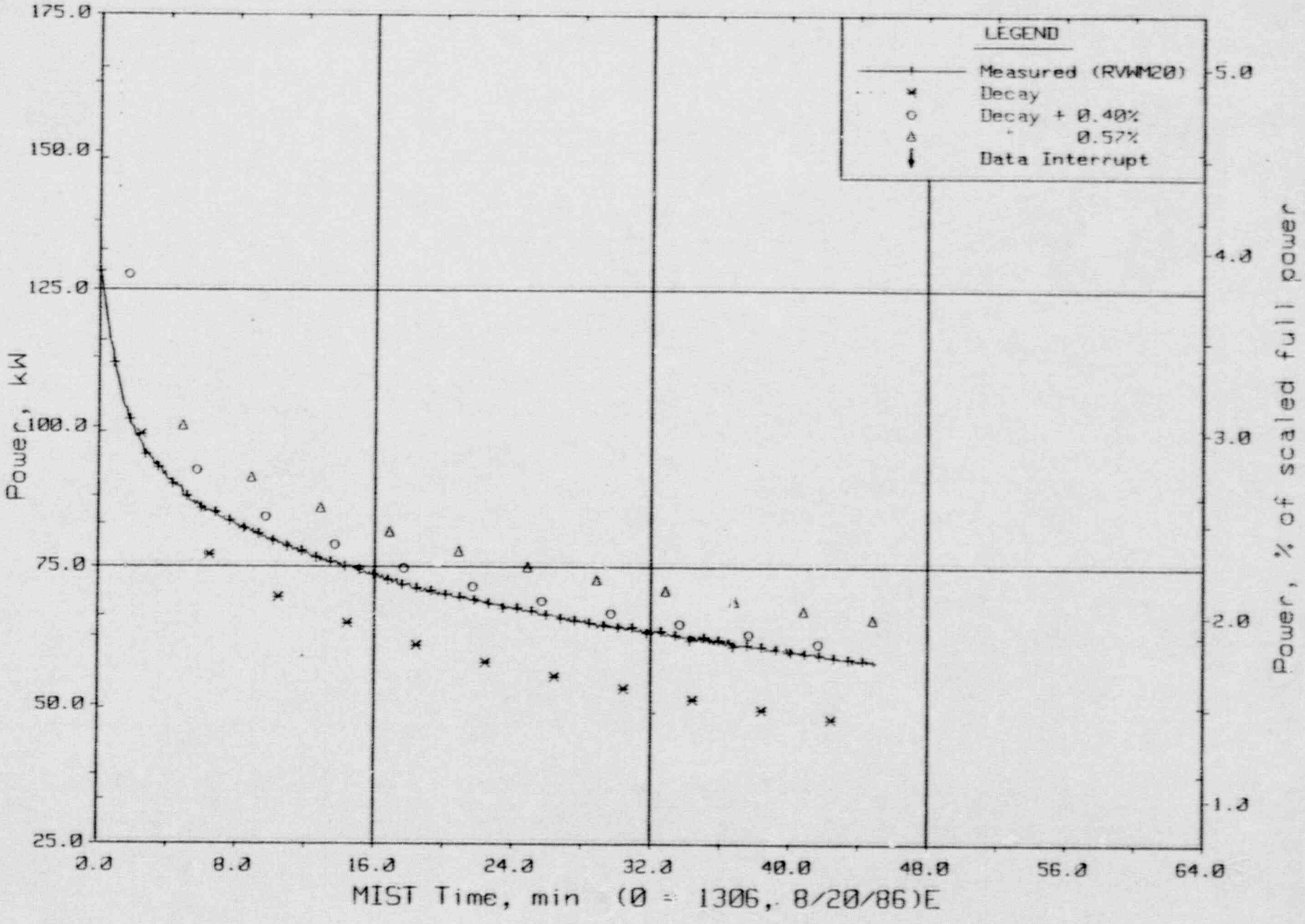
Pressurizer Collapsed Liquid Level (PZLV20).

Wed Jun 1 11:02:48 1988

PZLV1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



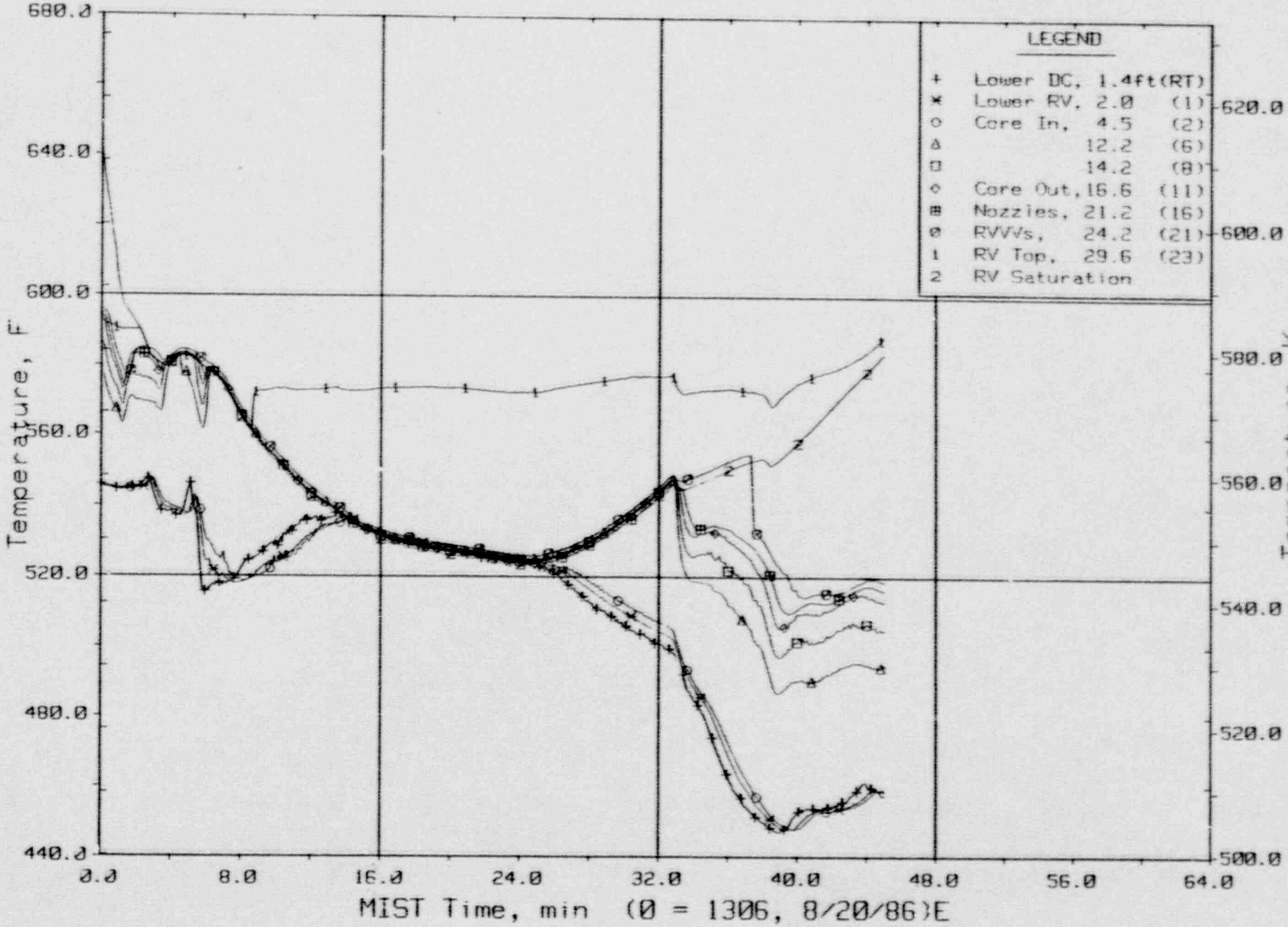
Core Power.

Wed Jun 1 11:06:12 1988

RVKW0

FINAL DATA

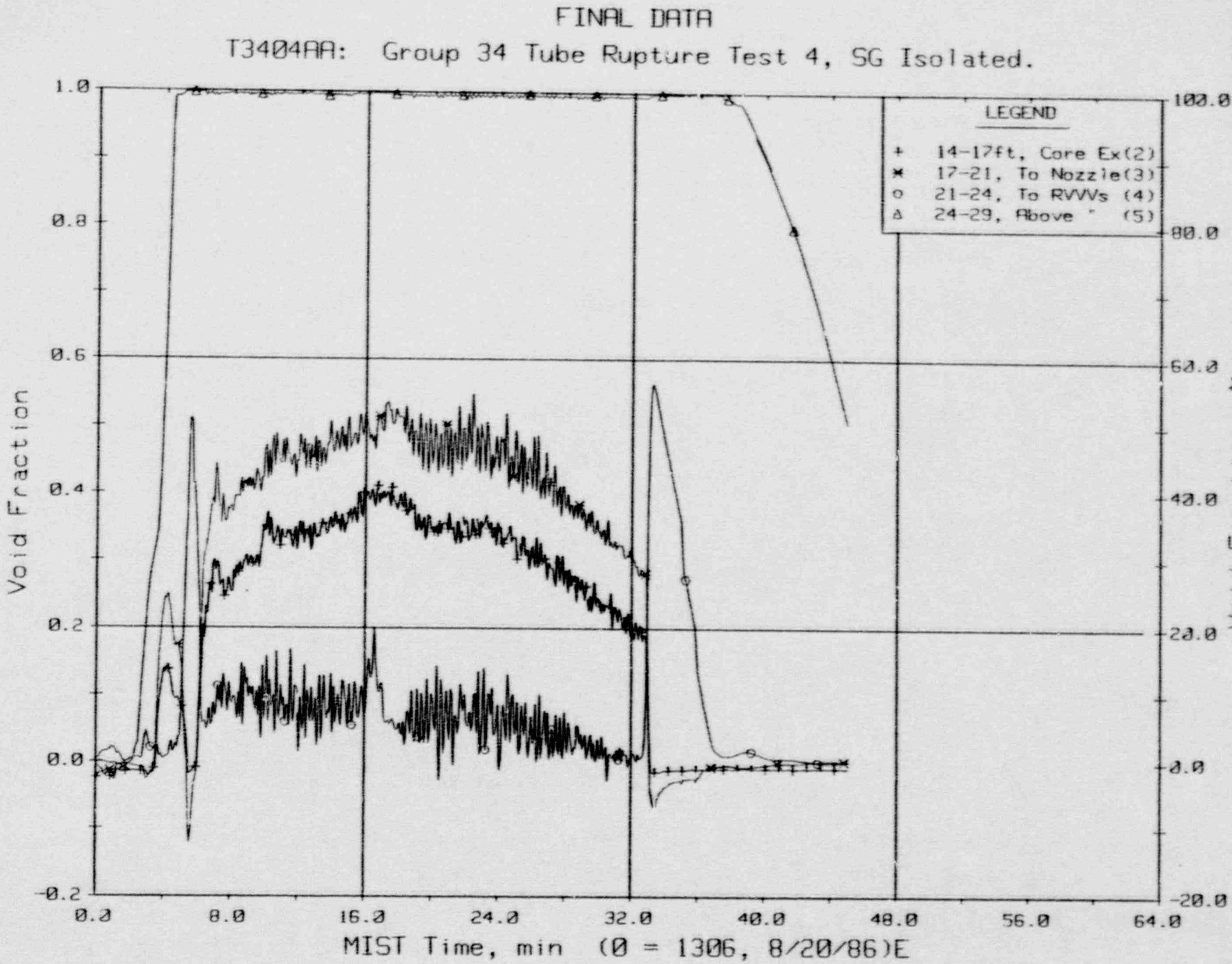
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Wed Jun 1 11:10:26 1988

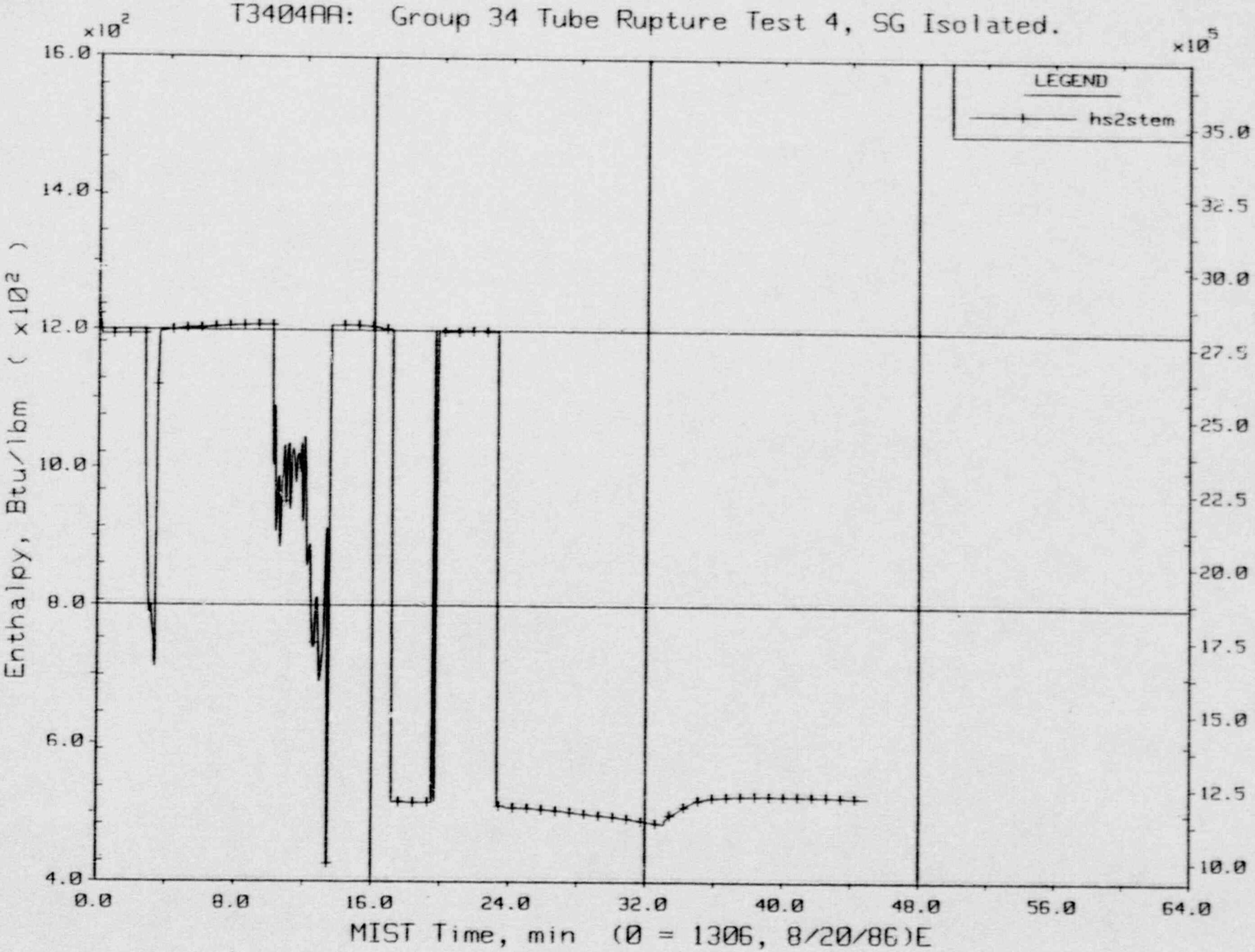
RVTC0



Reactor Vessel Void Fractions From Differential Pressures (RVVFs).

.FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



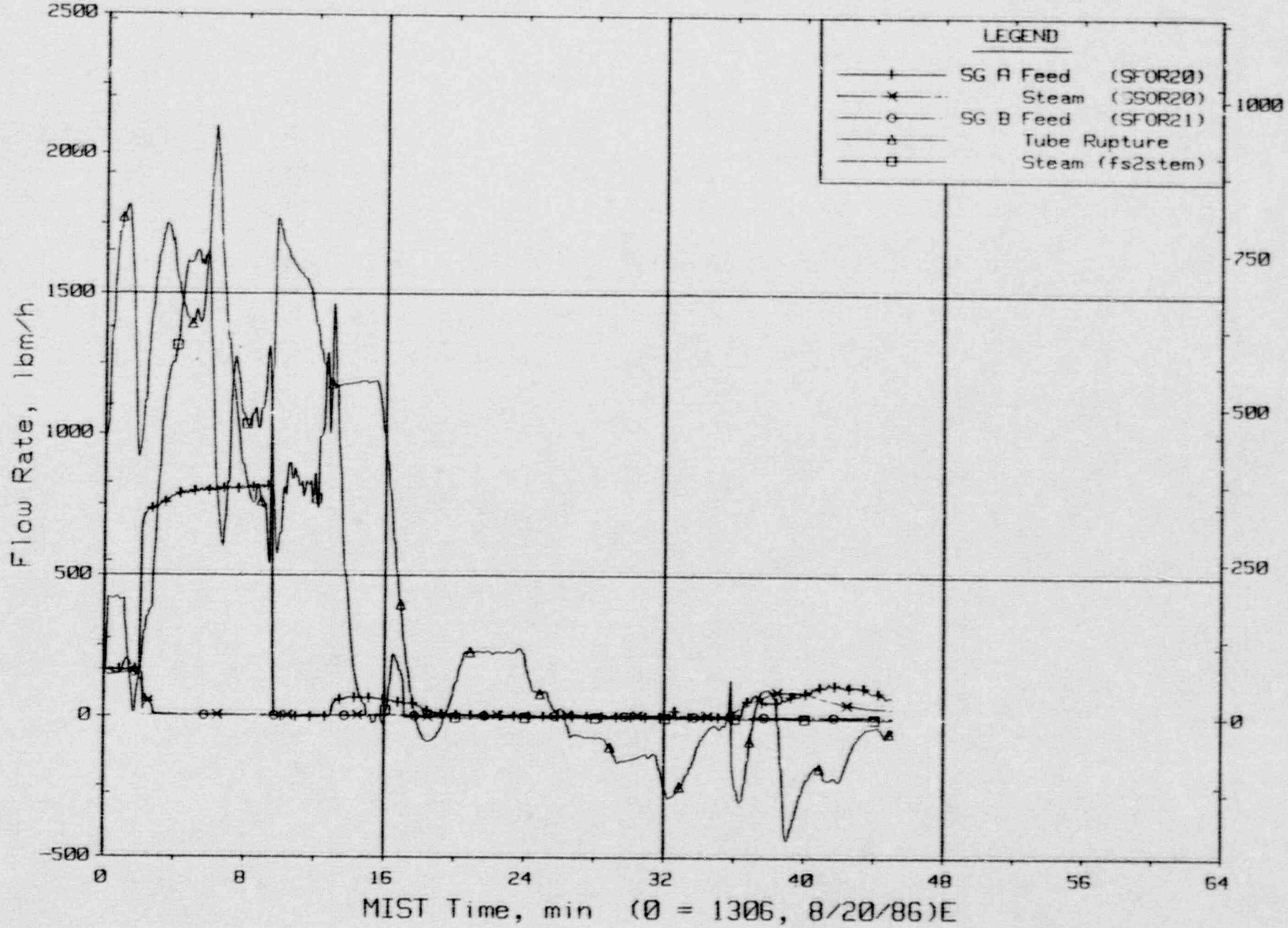
Steam Generator B Steam Exit Enthalpy.

Wed Jun 1 12:57:49 1988

S2EN1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



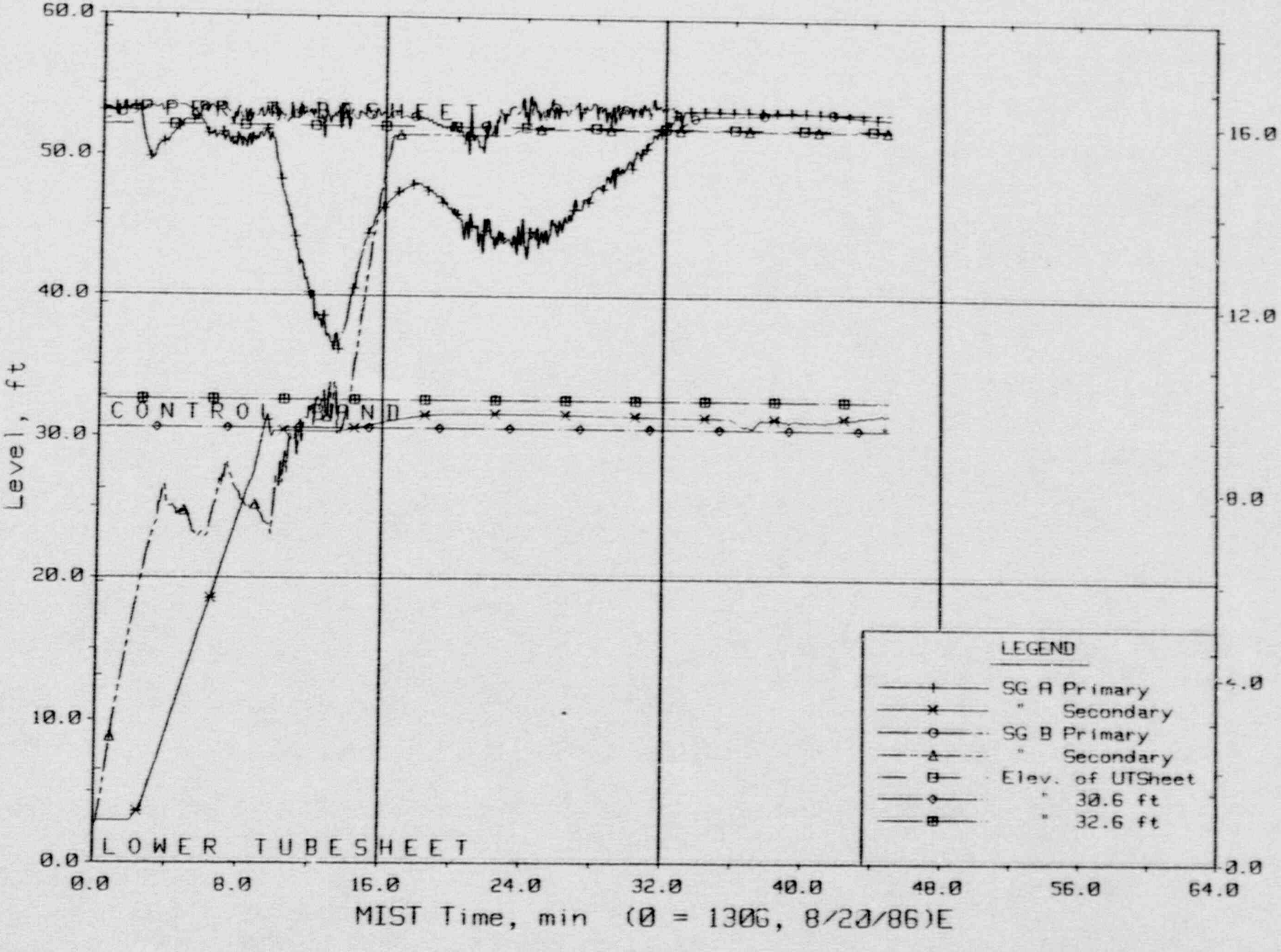
Steam Generator Secondary Flow Rates.

Tue Nov 1 17:58:00 1988

SGFL1

FINAL DATA

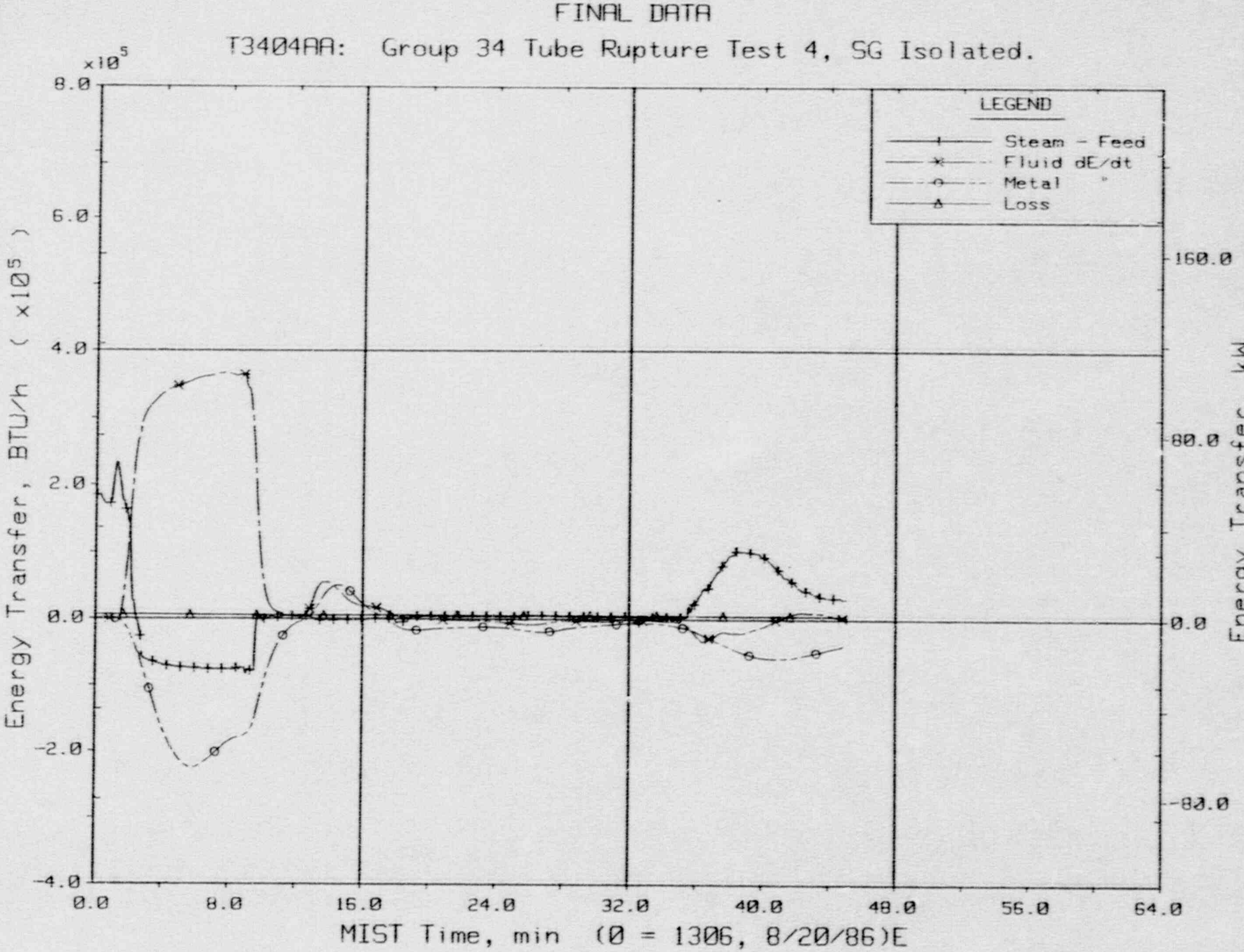
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Steam Generator Collapsed Liquid Levels.

Wed Jun 1 11:21:46 1988

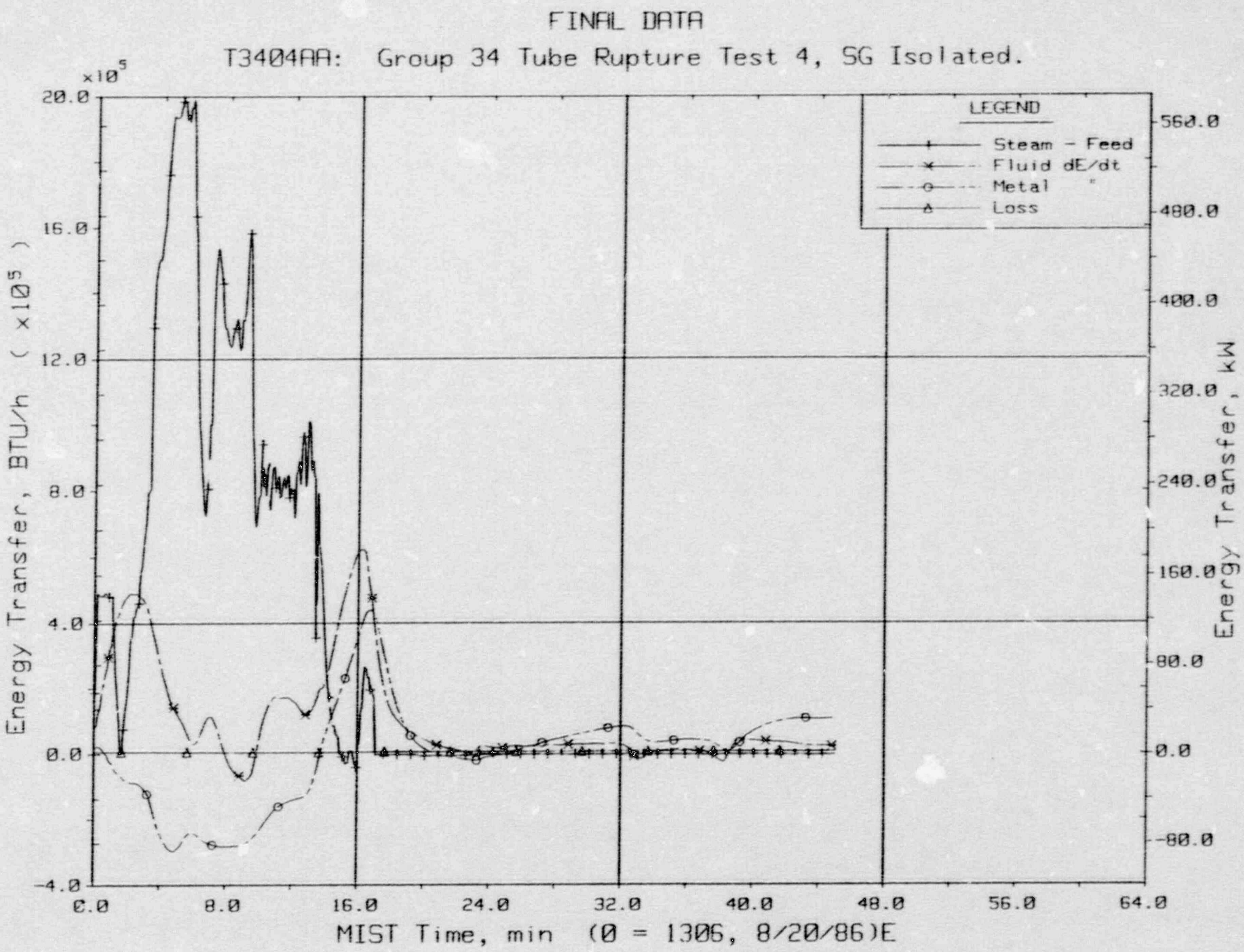
SGLV2



Steam Generator A Energy Transfer.

Wed Jun 1 11:42:29 1988

SGOE3



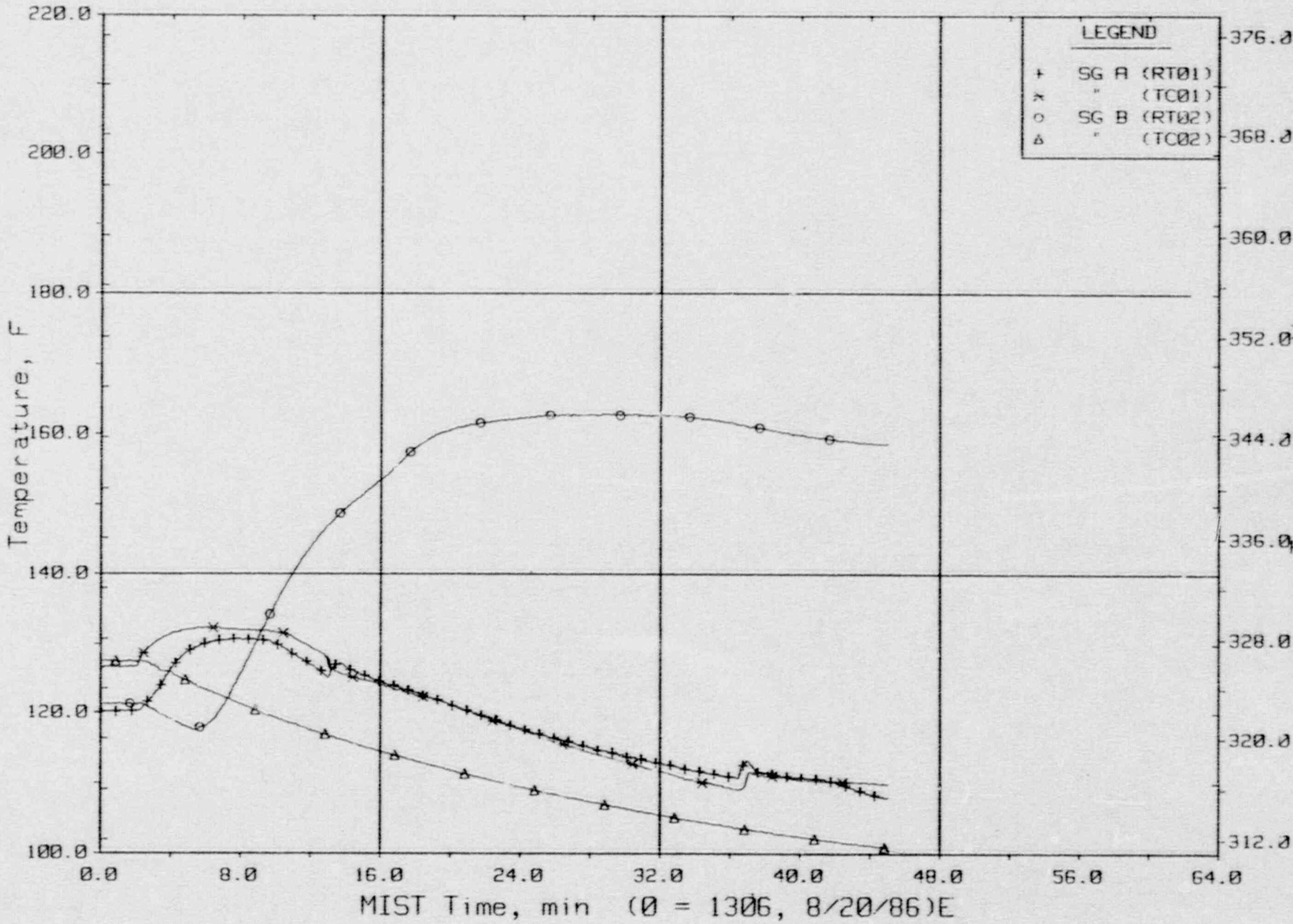
Steam Generator B Energy Transfer.

Wed Jun 1 11:45:59 1988

SGQE6

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



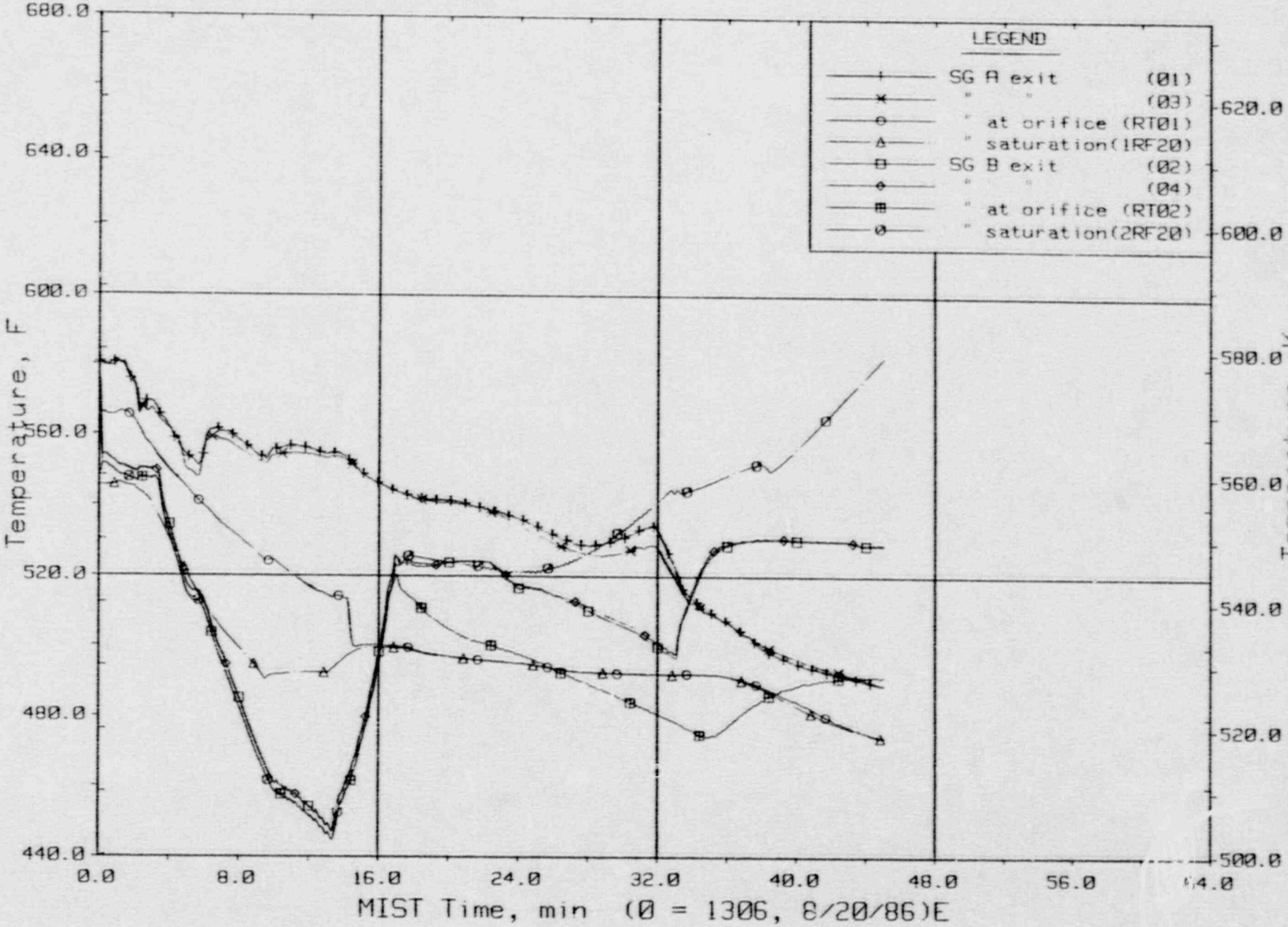
Feedwater Temperatures (SFs).

Wed Jun 1 11:49:29 1988

SGRT1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



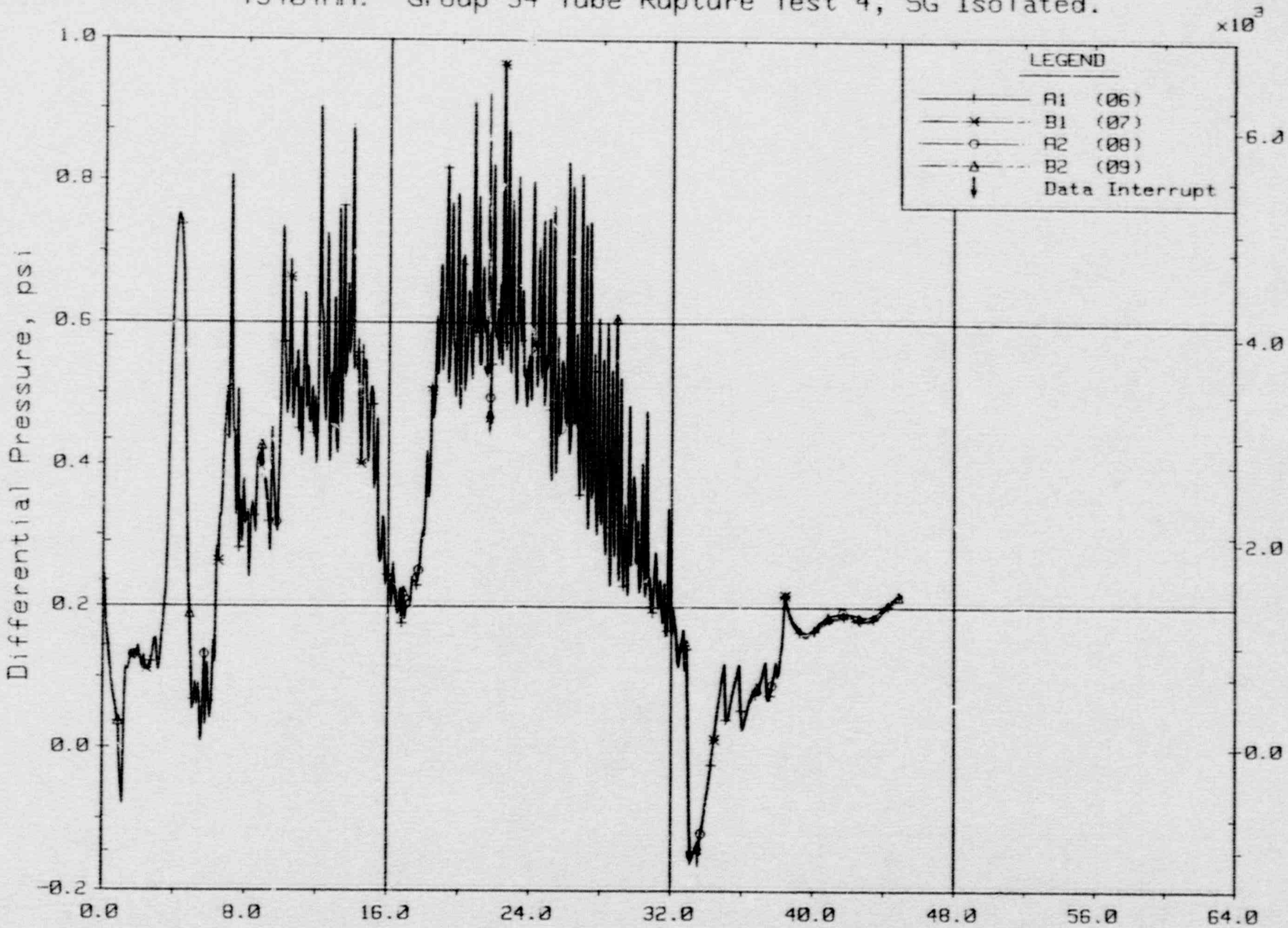
Steam Generator Steam Outlet Temperatures (SSTCs).

Wed Jun 1 11:53:46 1988

SGRT2

FINAL DATA

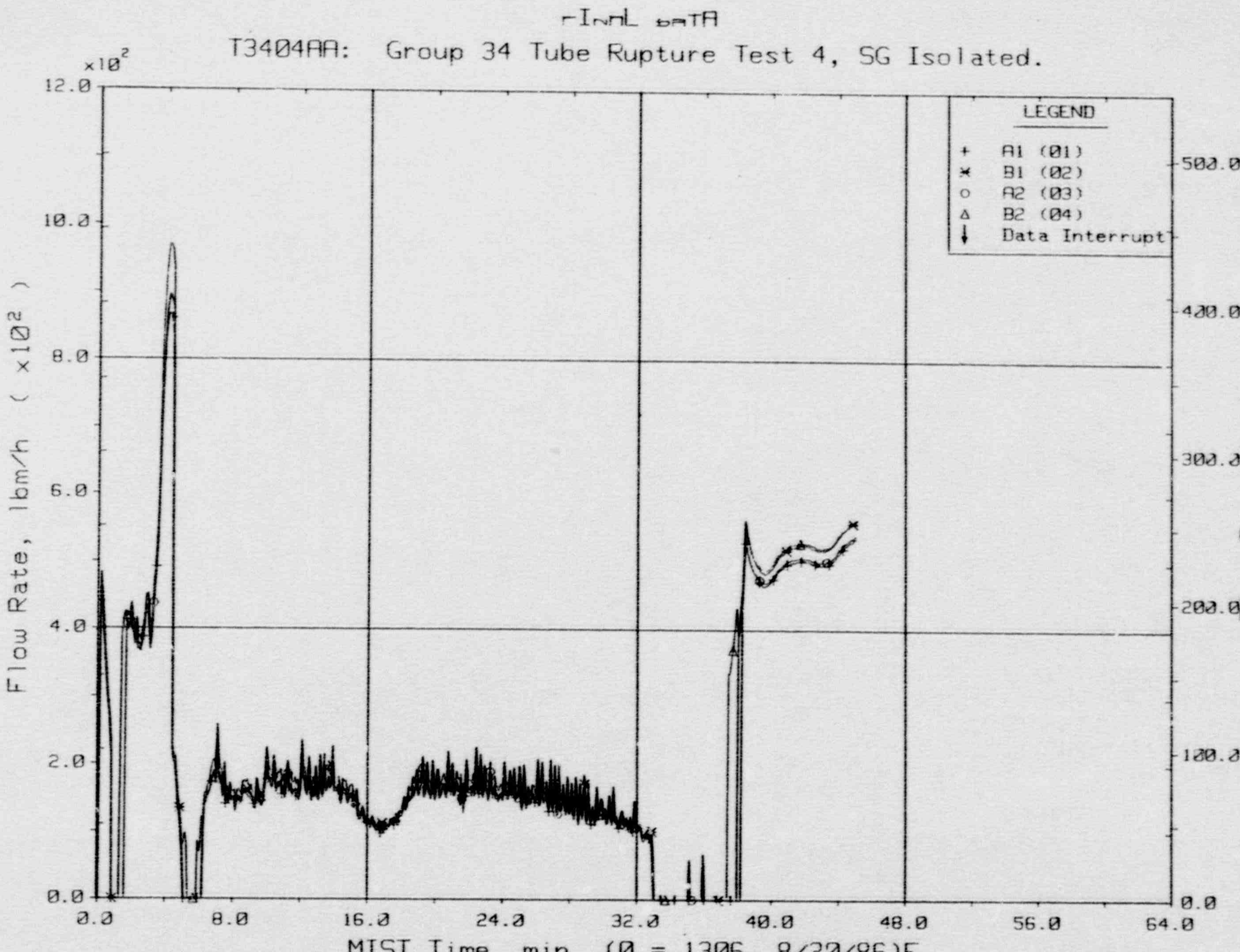
T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Reactor Vessel Vent Valve Differential Pressures (RVDPs).

Wed Jun 1 11:59:37 1988

VVDP1



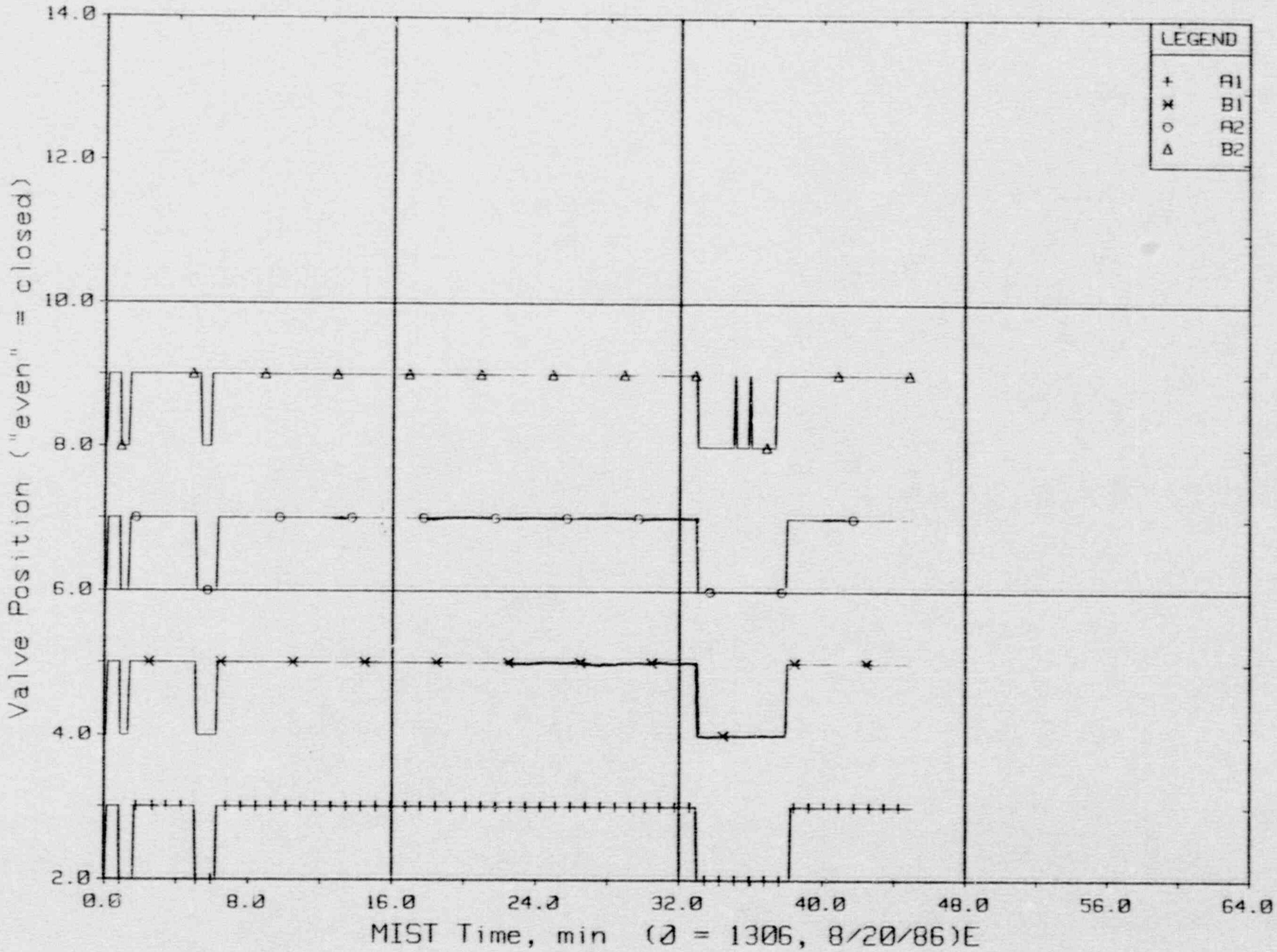
Reactor Vessel Vent Valve Flow Rates (RVORs).

Wed Jun 1 12:05:25 1988

VVFL1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



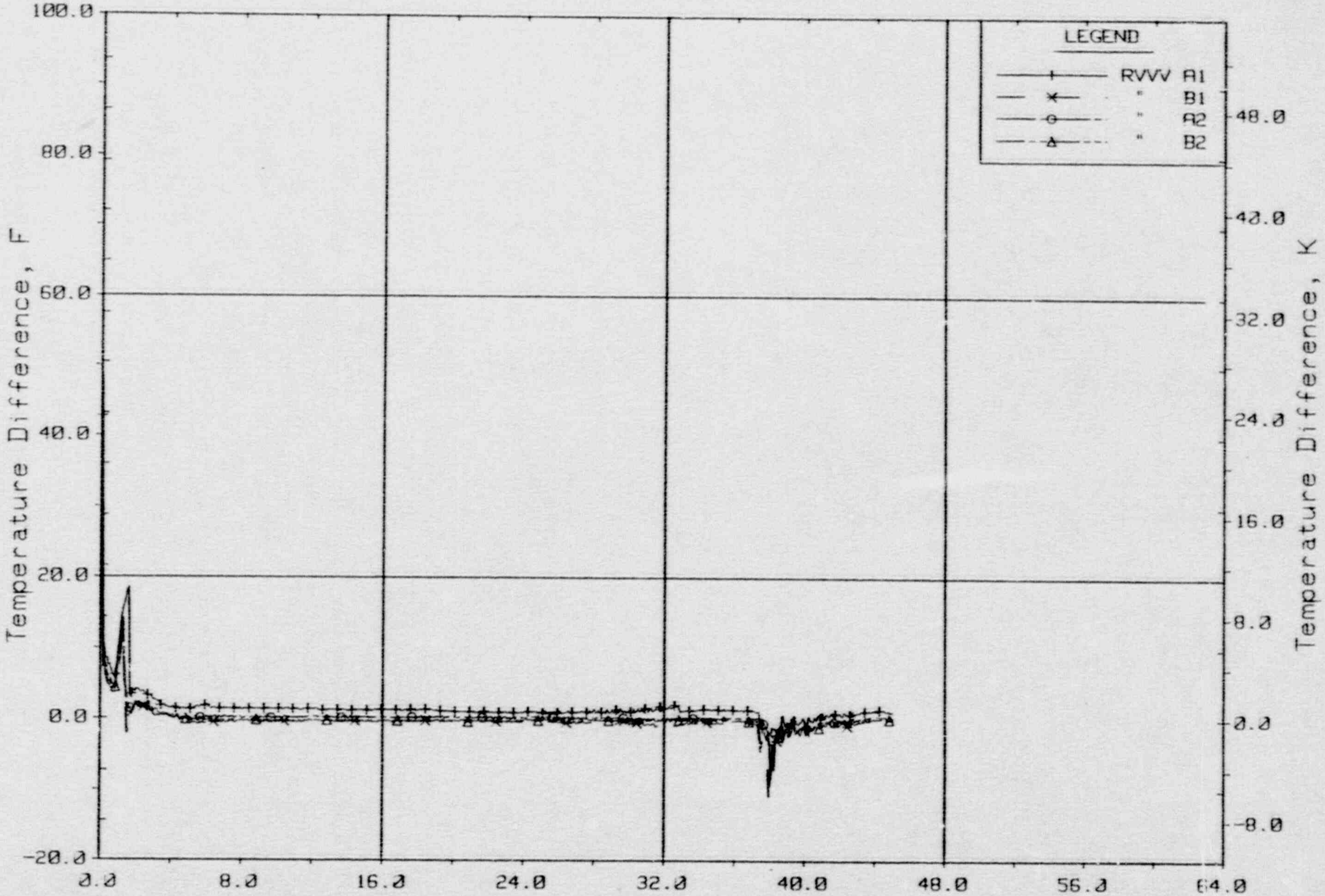
Reactor Vessel Vent Valve Positions.

Wed Jun 1 12:09:37 1988

VVLS1

FINAL DATA

T3404AA: Group 34 Tube Rupture Test 4, SG Isolated.



Temperature Differences Across Vent Valves.

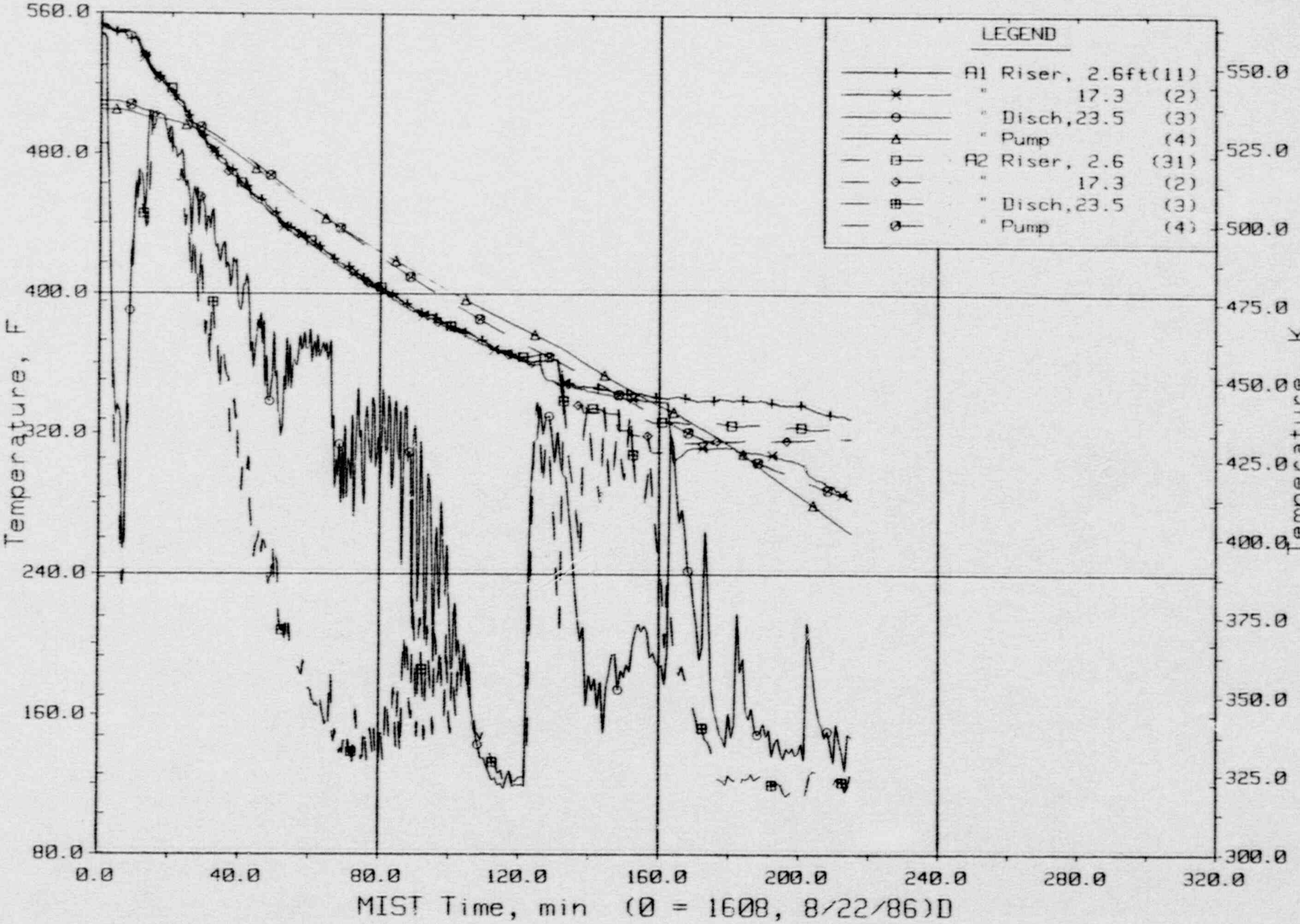
Wed Jun 1 12:12:51 1988

VVTD1

T340504

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



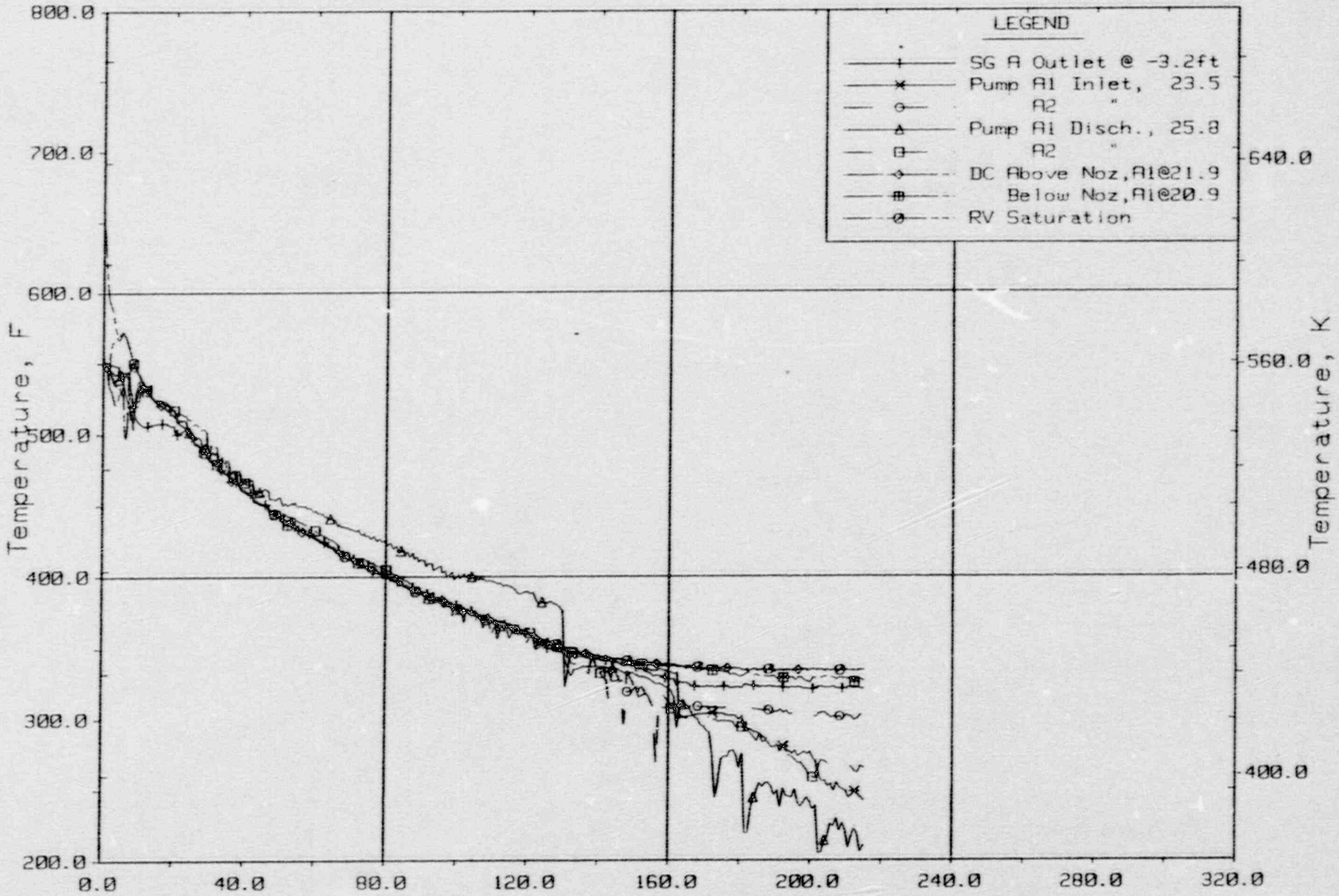
Loop A Cold Leg Metal Temperatures (C1,3MTs).

Wed Jun 1 13:02:22 1988

C1MT1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



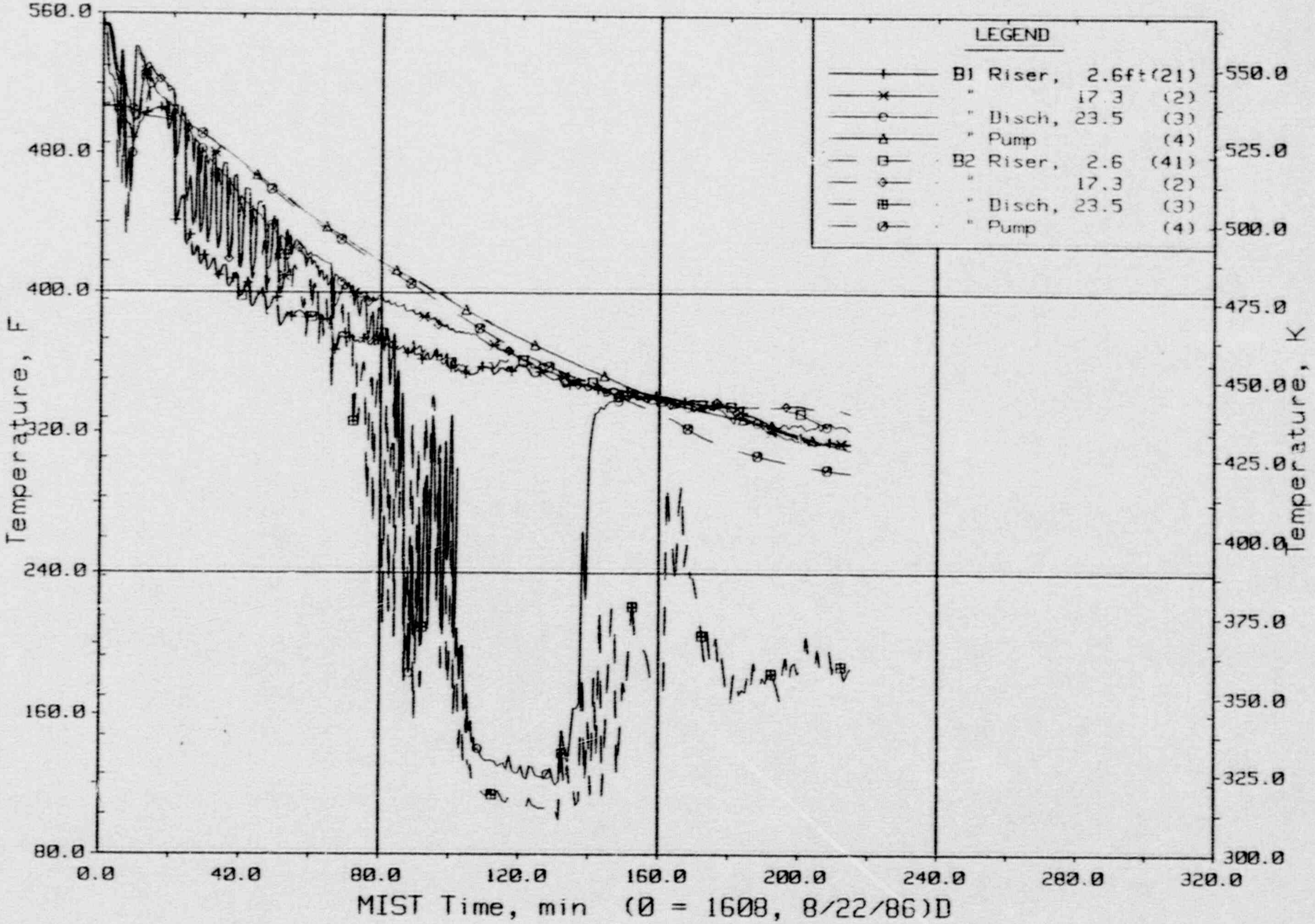
Loop A Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 13:06:54 1988

C1RT0

FINAL DATA

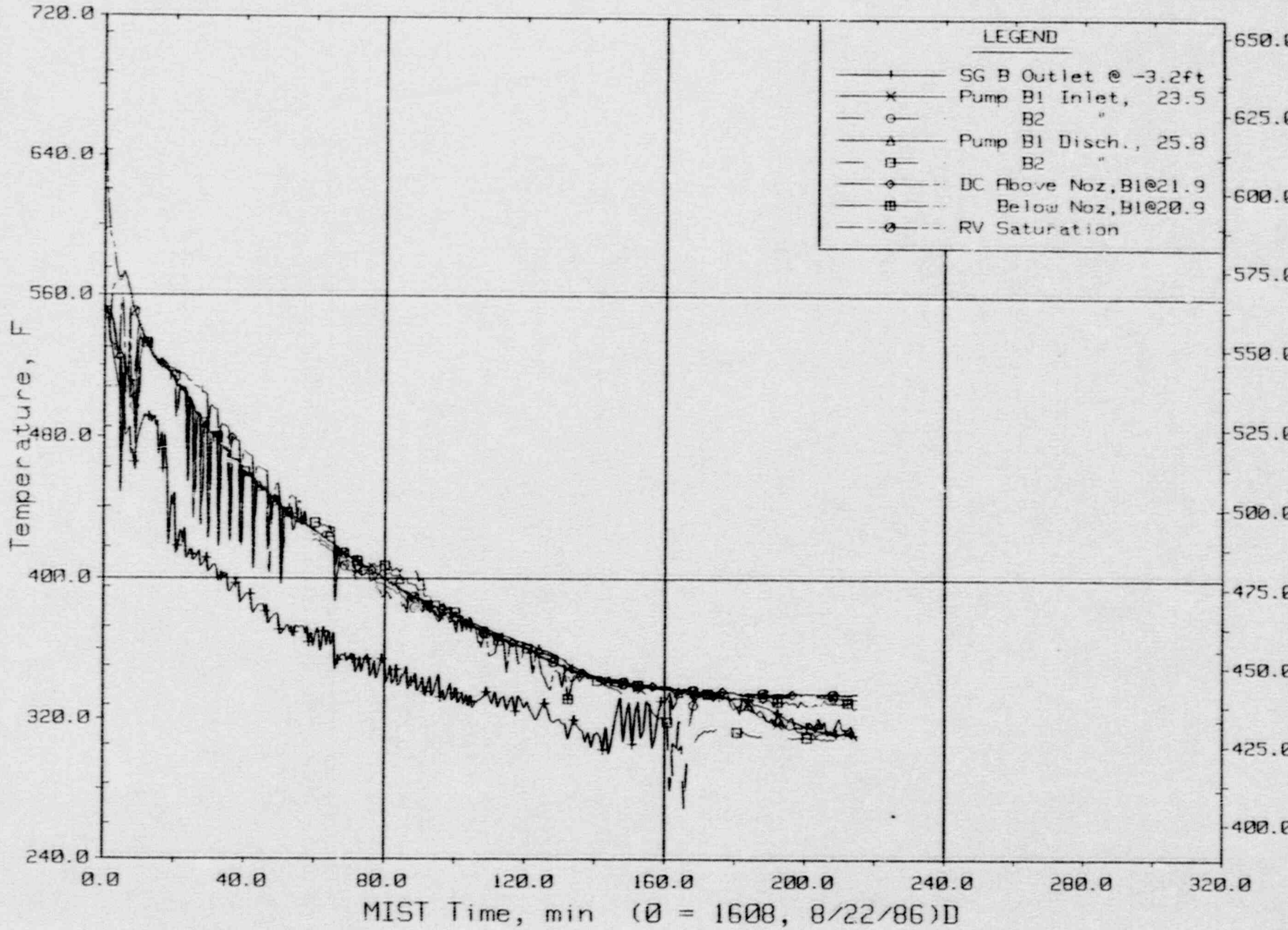
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Loop B Cold Leg Metal Temperatures (C2,4MTs).

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



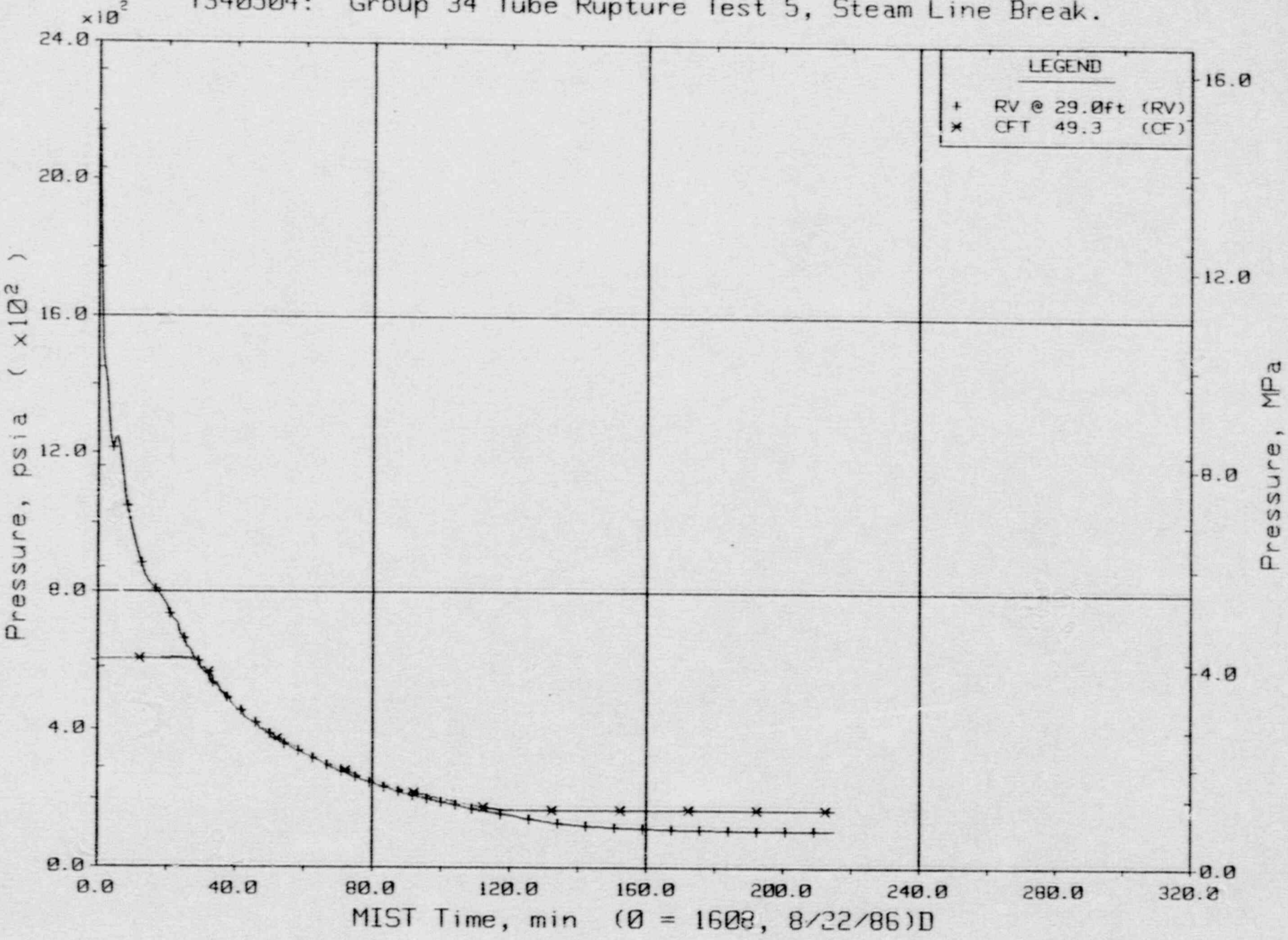
Loop B Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 13:16:27 1988

C2RT0

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



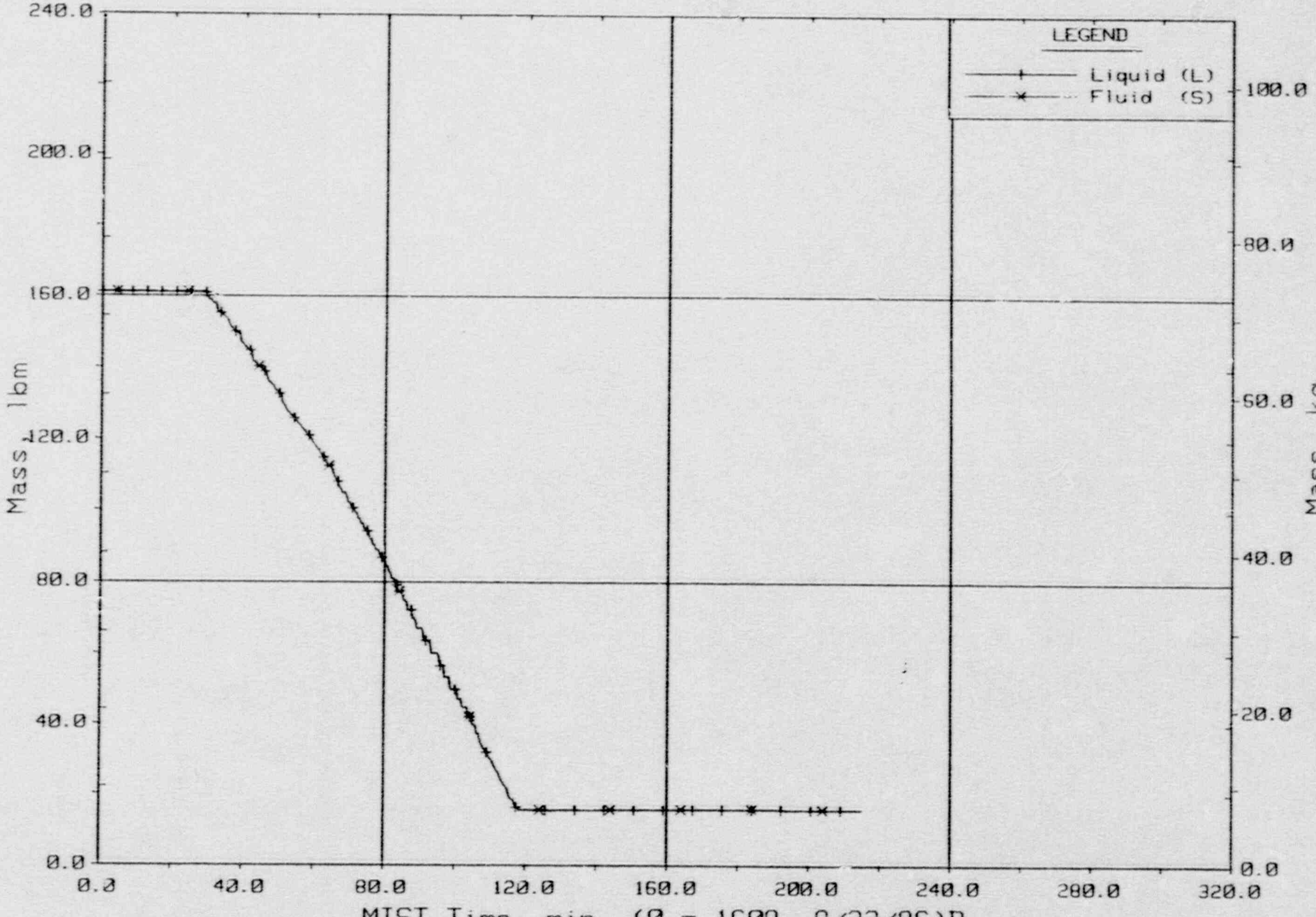
Primary System and Core Flood Tank Pressures (GP01s).

Wed Jun 1 13:19:25 1988

CFGP1

FINAL DATA

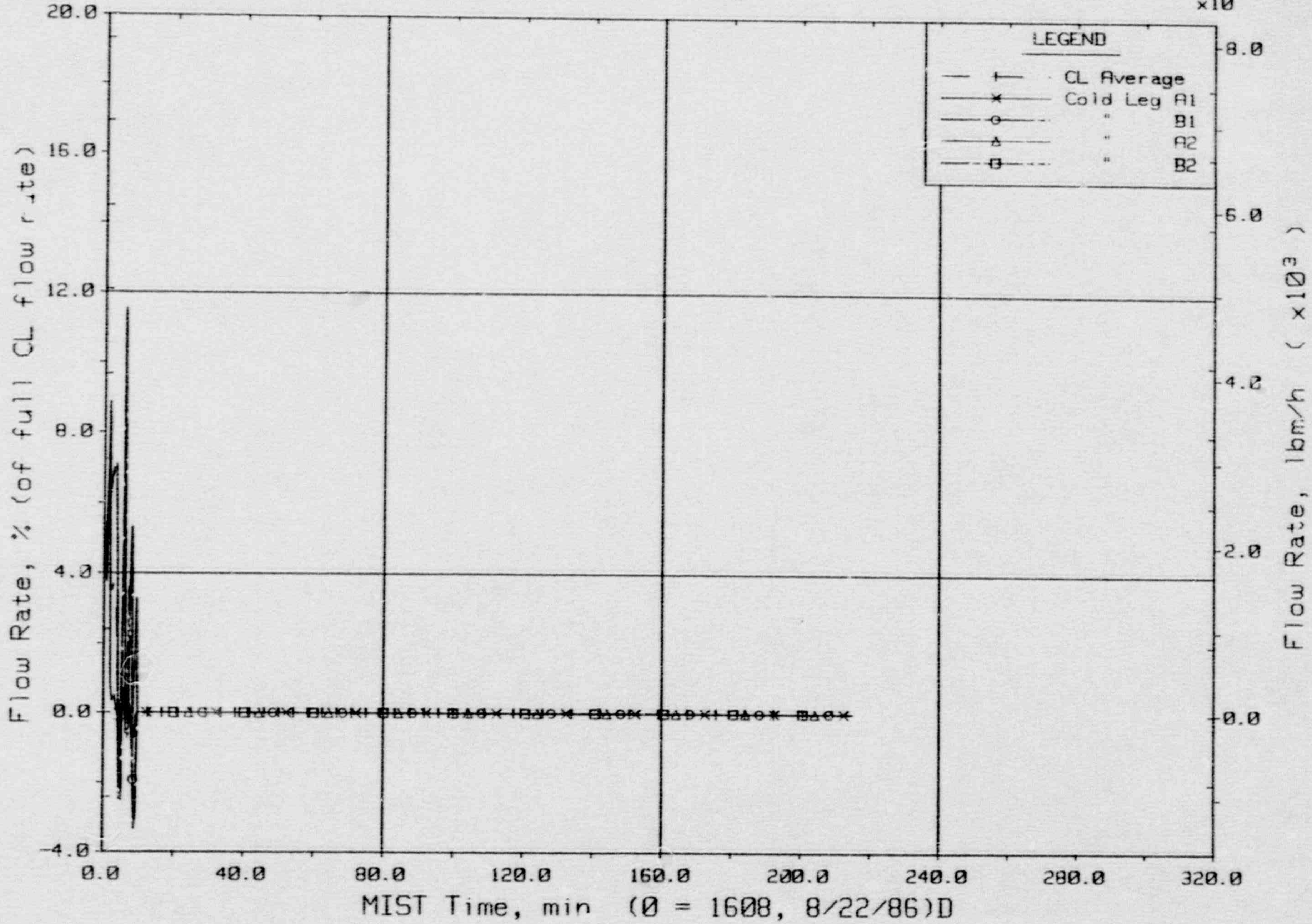
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Core Flood Tank Liquid and Fluid Mass (CFMa20s).

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



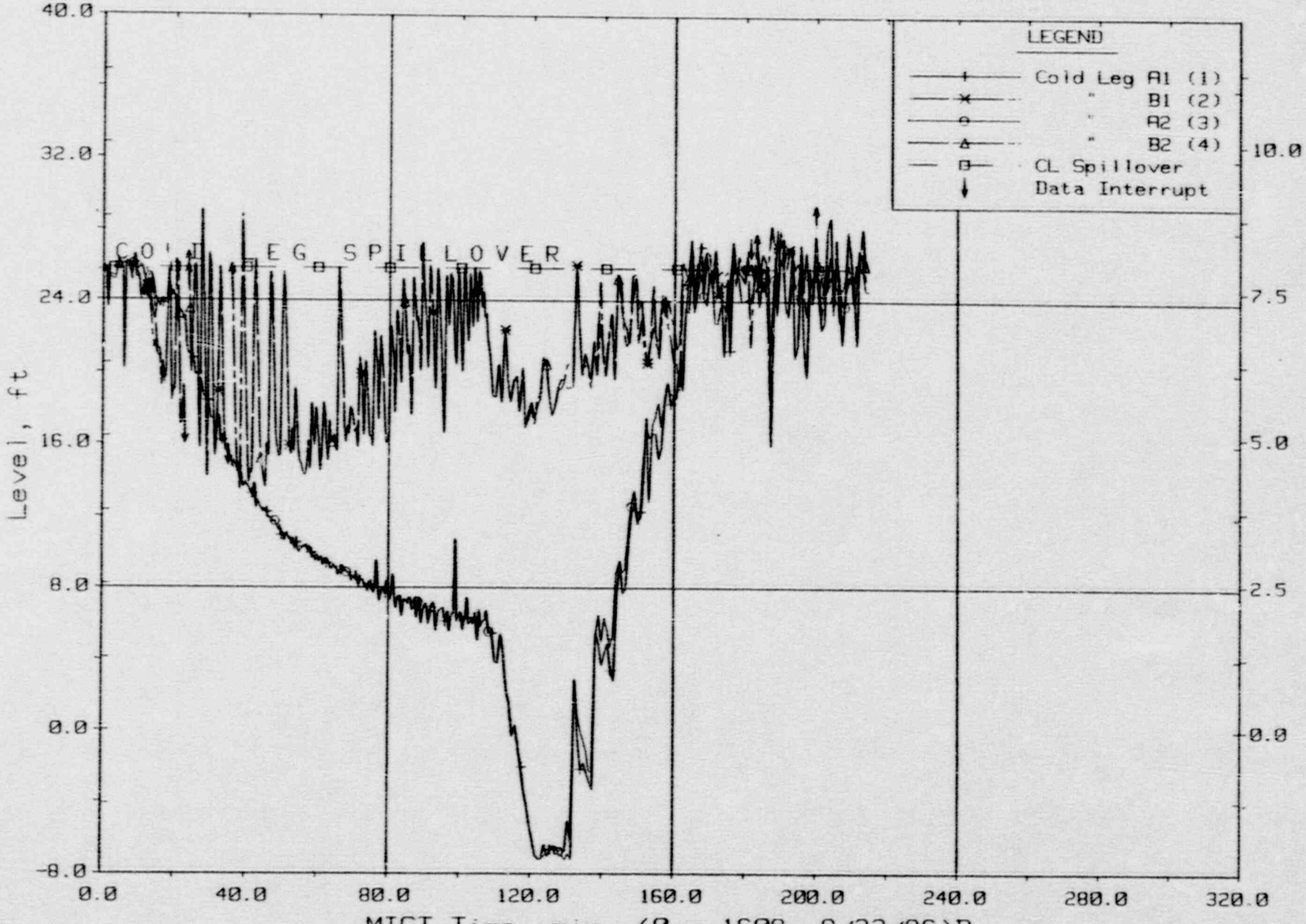
Cold Leg (Venturi) Flow Rates.

Thu Jun 2 09:54:55 1988

CLFL1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



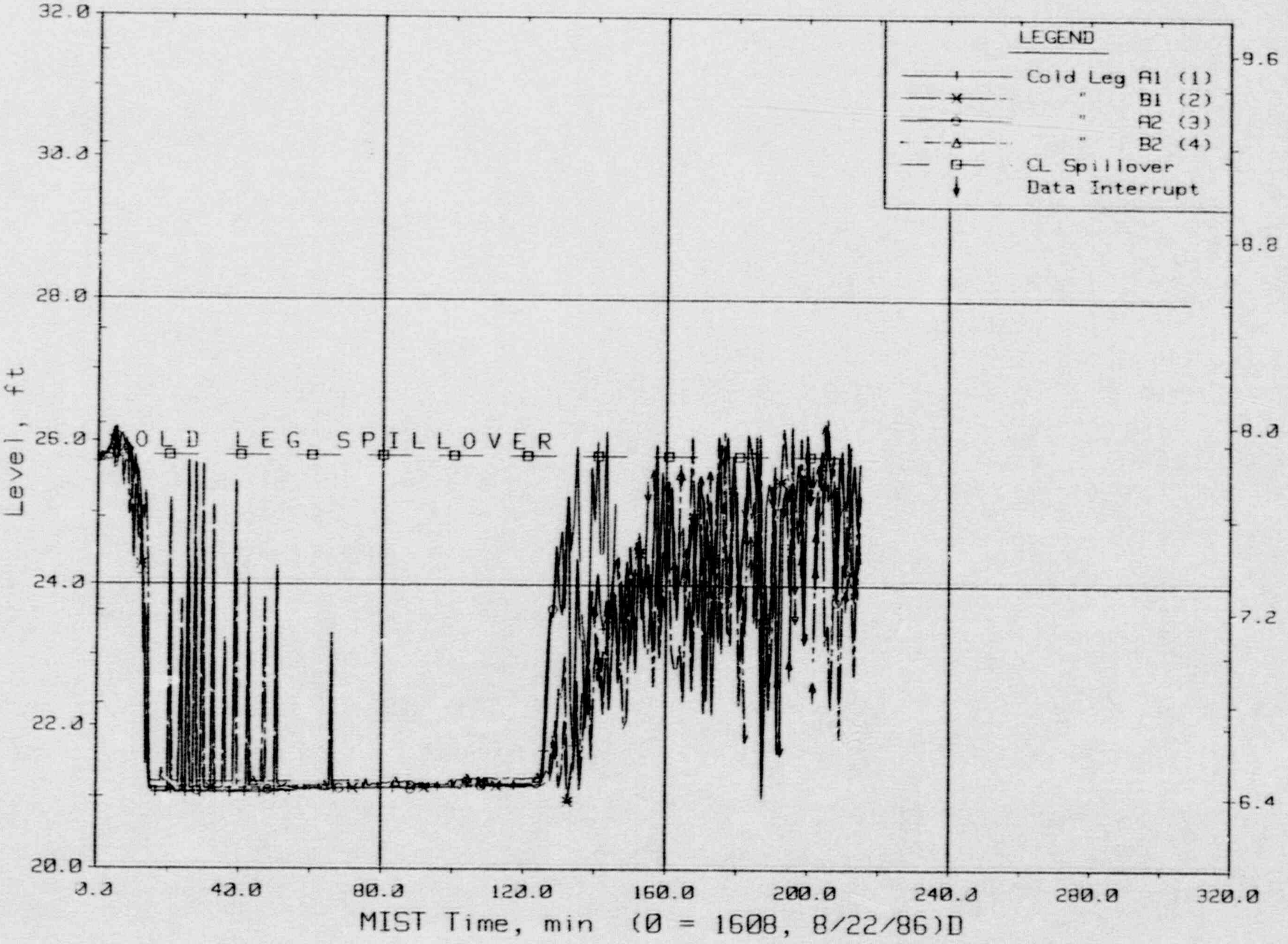
Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

Wed Jun 1 13:30:02 1988

CLLV4

FINAL DATA

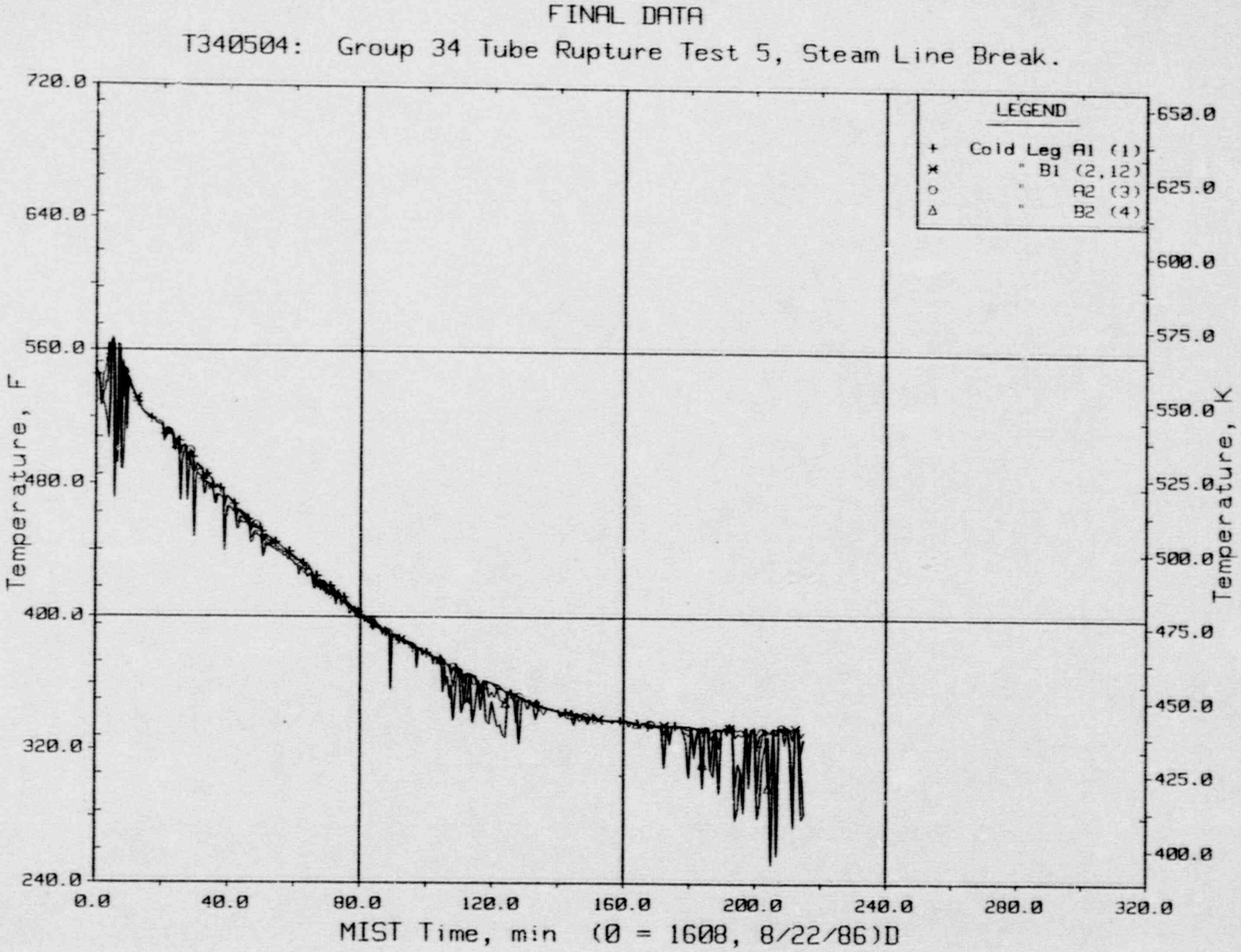
T3405Ø4: Group 34 Tube Rupture Test 5, Steam Line Break.



Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Wed Jun 1 13:34:13 1988

CLLV5



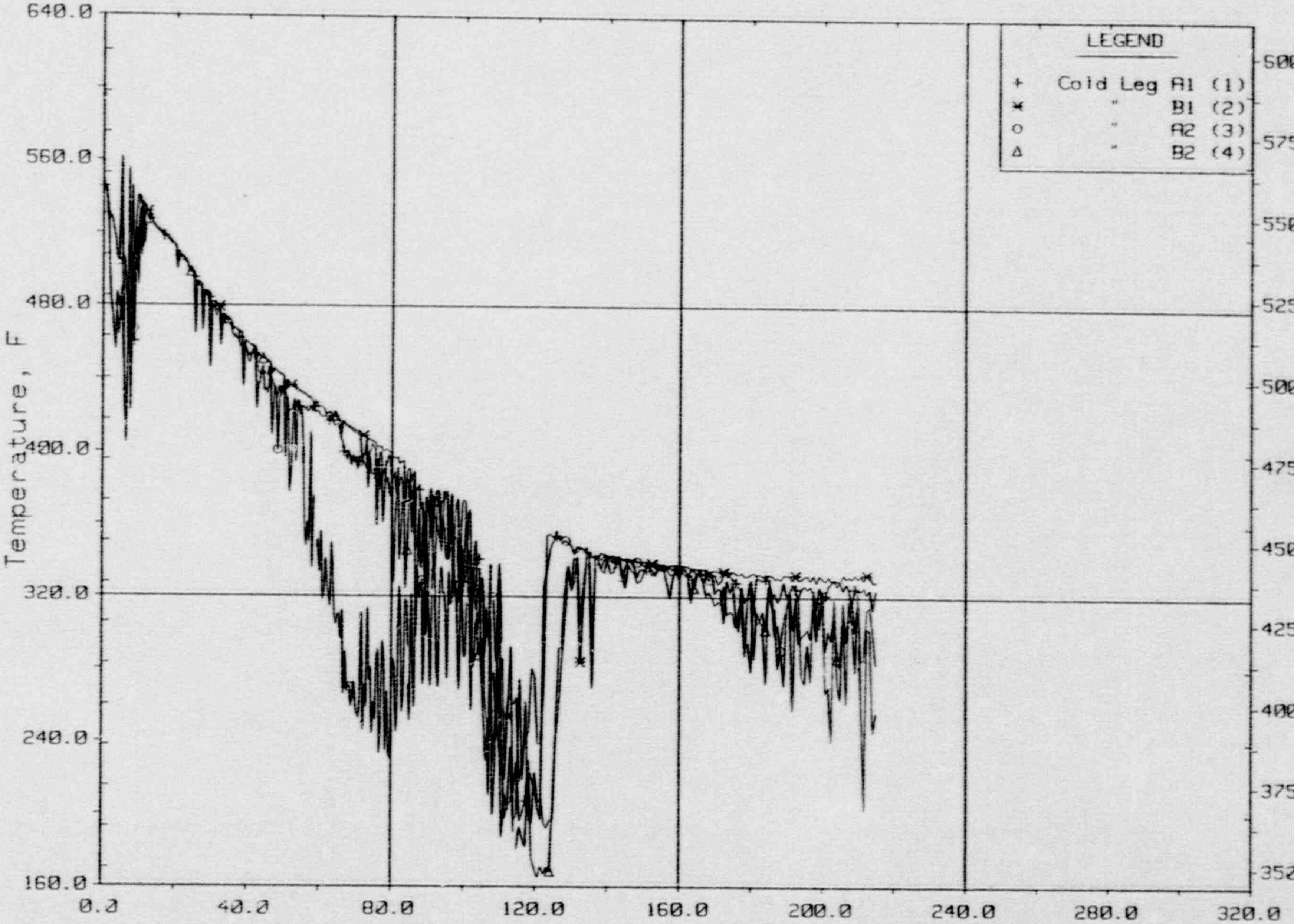
Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC11s).

Thu Jun 2 09:24:53 1988

CLTC1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



MIST Time, min (0 = 1608, 8/22/86)D

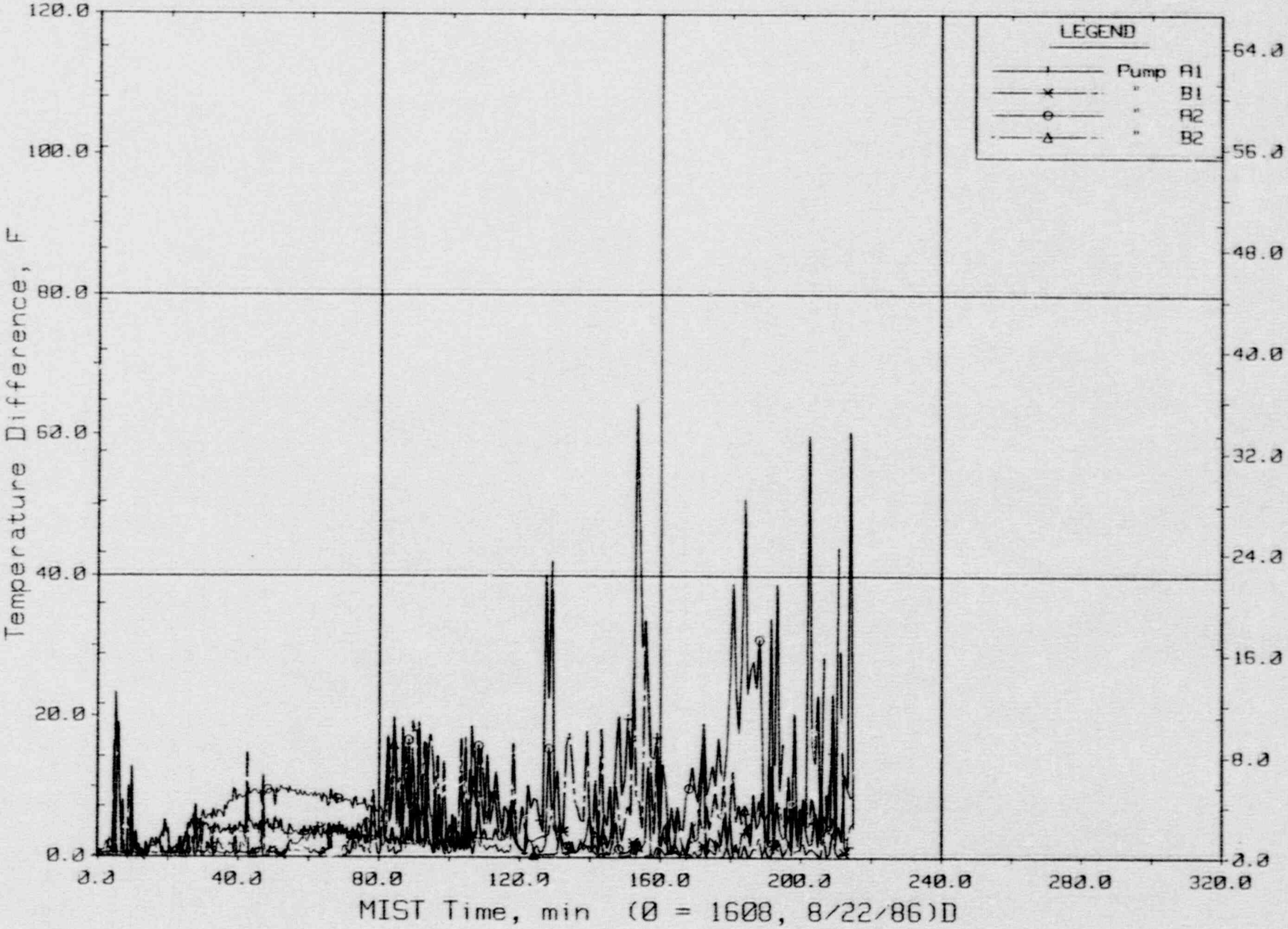
Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

Wed Jun 1 13:46:16 1988

CLTC2

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



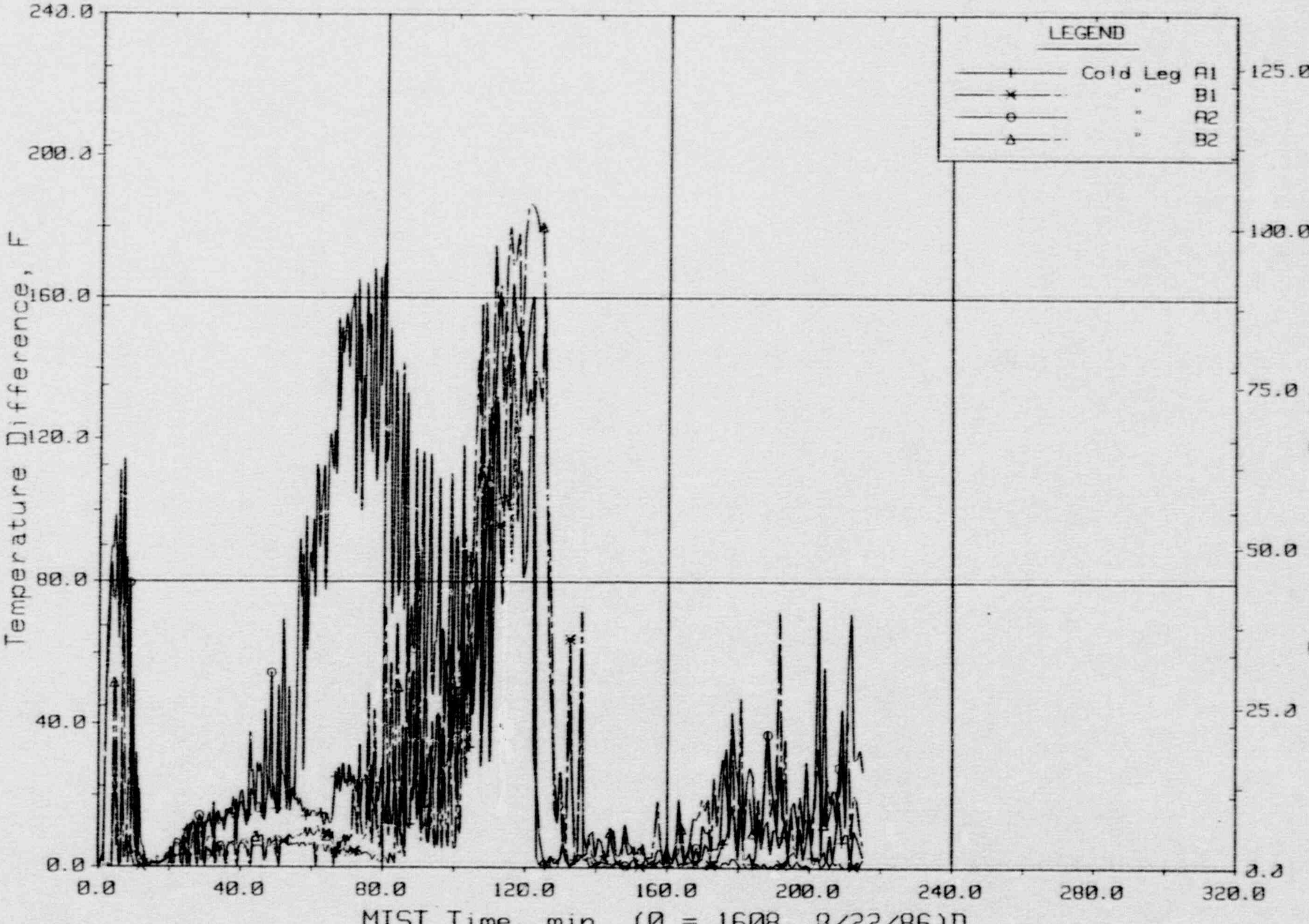
Maximum Differences Among RCP Rake FLuid Temperatures.

Wed Jun 1 13:50:53 1988

CLTD1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



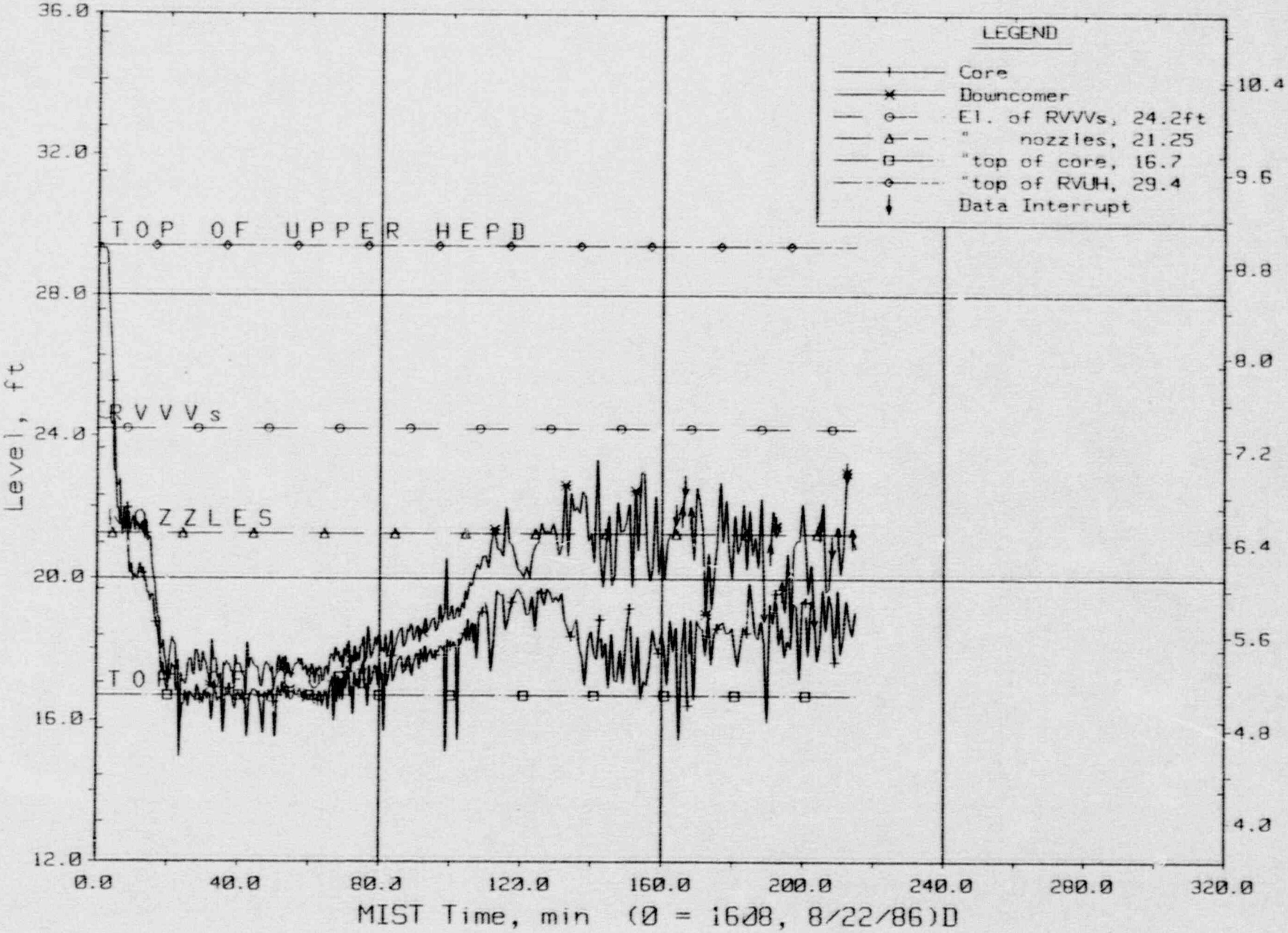
Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

Wed Jun 1 13:54:11 1988

CLTD2

FINAL DATA

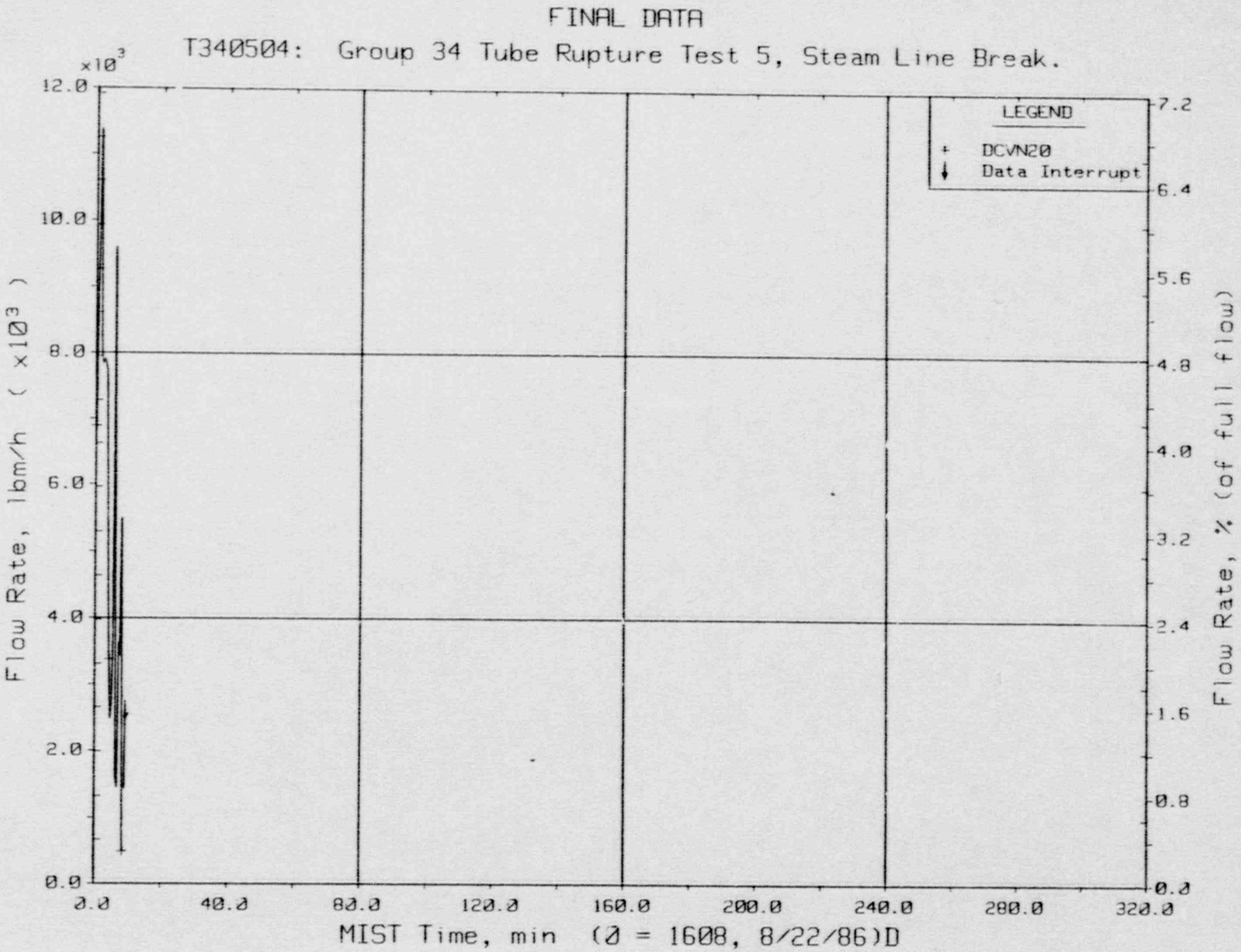
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Core Region Collapsed Liquid Levels.

Wed Jun 1 13:57:54 1988

COLV1



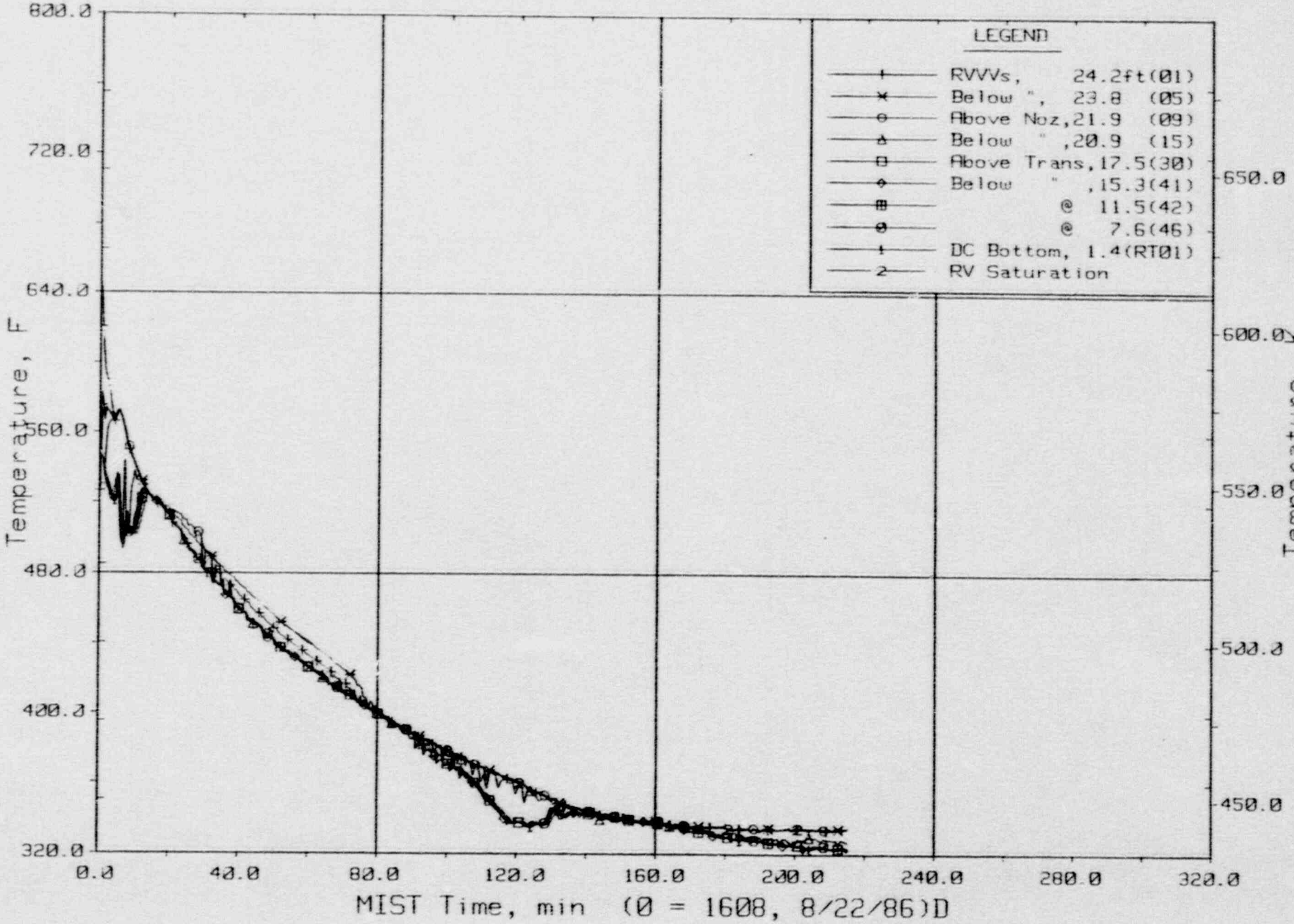
Downcomer (Venturi) Flow Rate.

Wed Jun 1 14:01:44 1988

DCFL1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



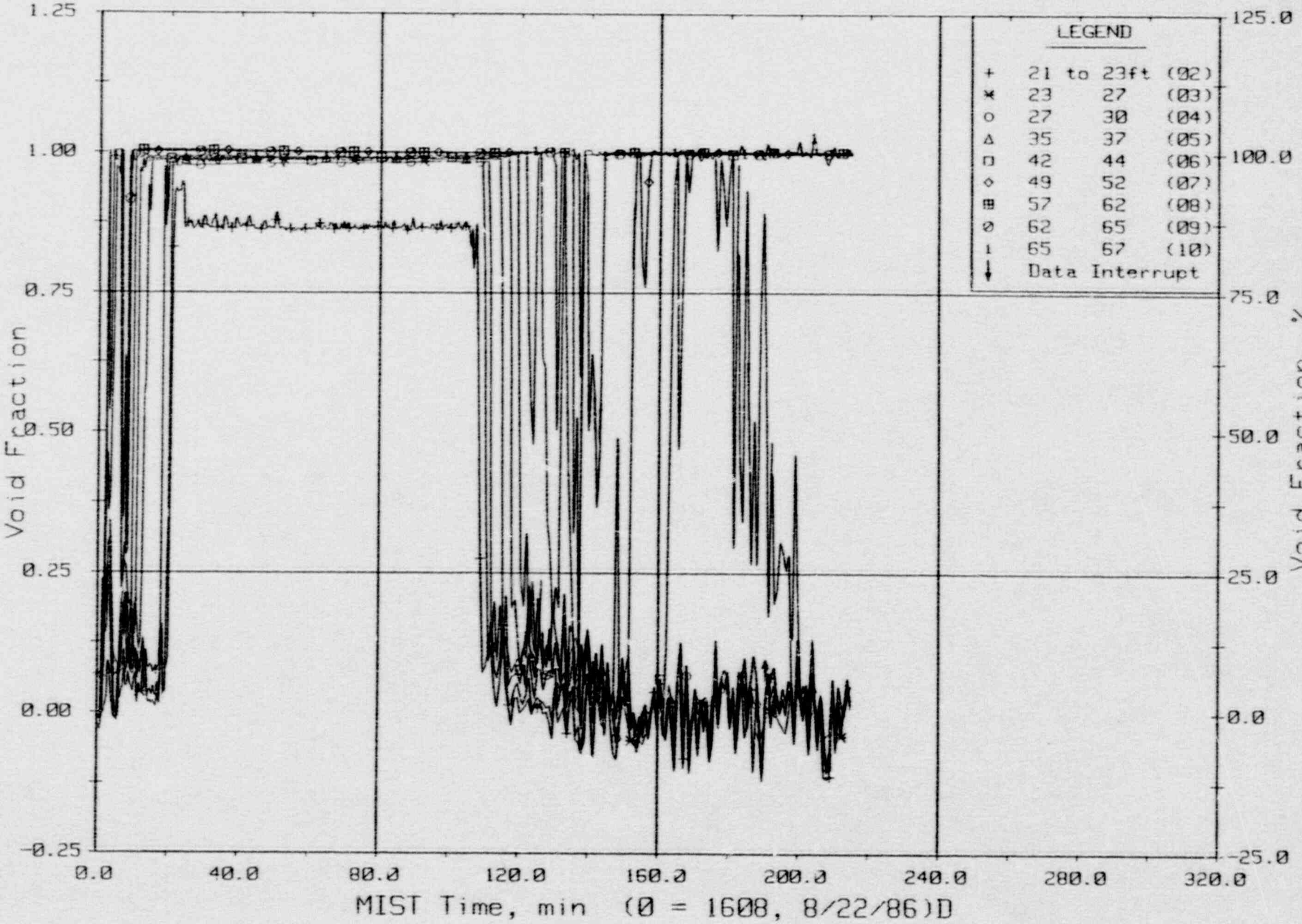
Downcomer Quadrant A1 Fluid Temperatures (DCTCs).

Wed Jun 1 14:05:27 1988

DCTC6

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



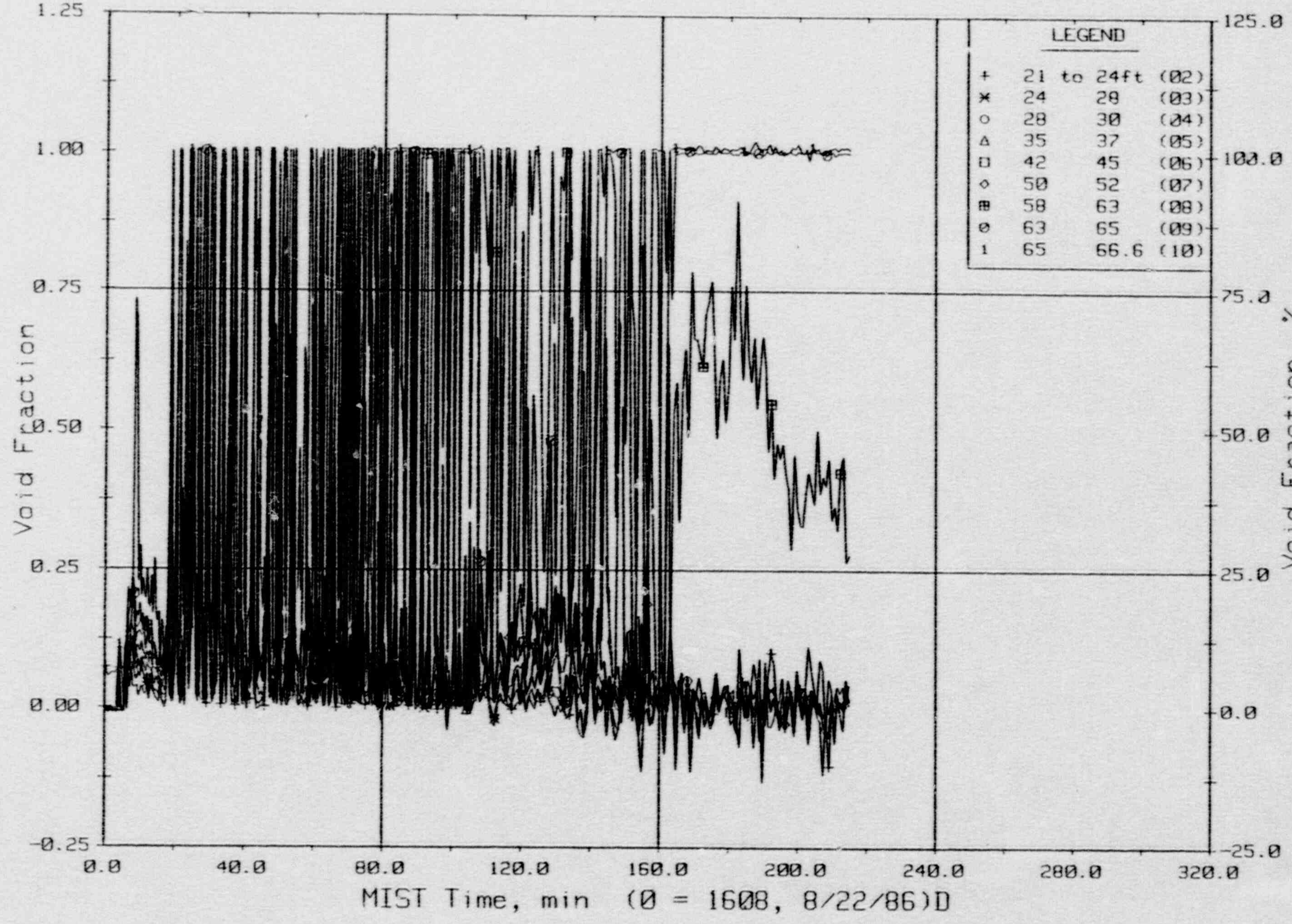
Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

Wed Jun 1 14:10:05 1988

H1VF1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



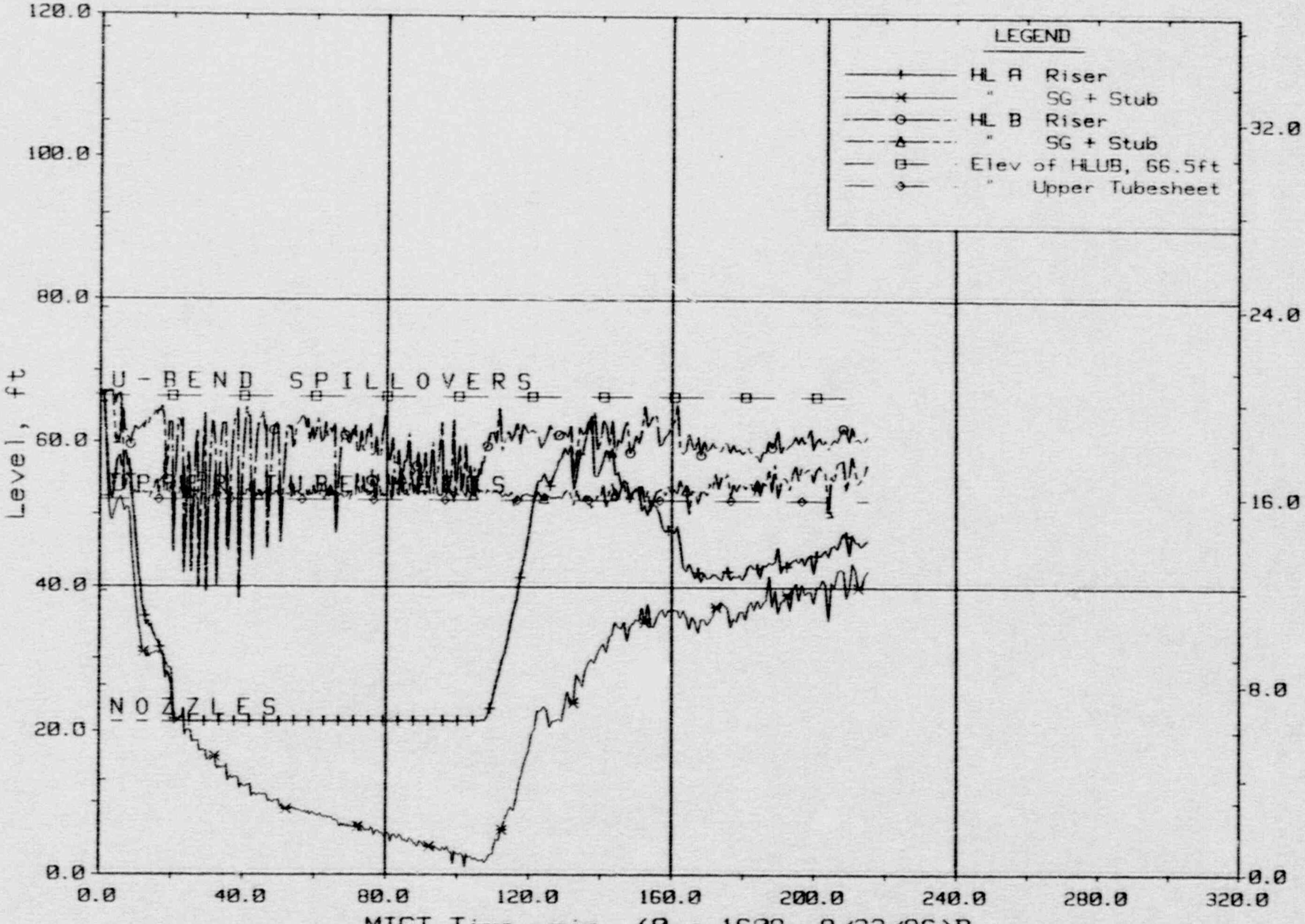
Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

Wed Jun 1 14:16:08 1988

H2VF1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



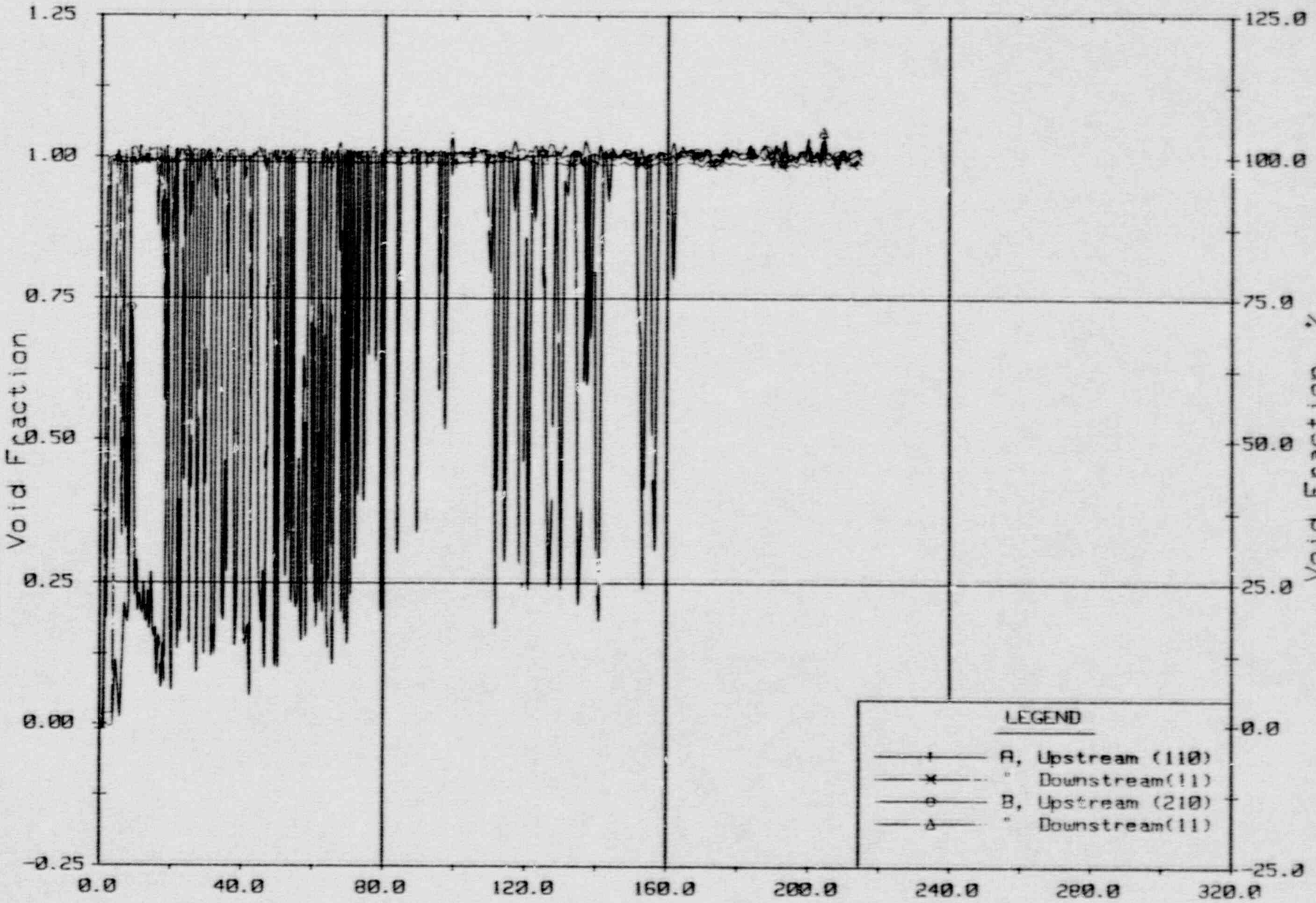
Hot Leg Riser and Stub Collapsed Liquid Levels.

Thu Jun 2 09:38:27 1988

HLLV1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



MIST Time, min (0 = 1608, 8/22/86)D

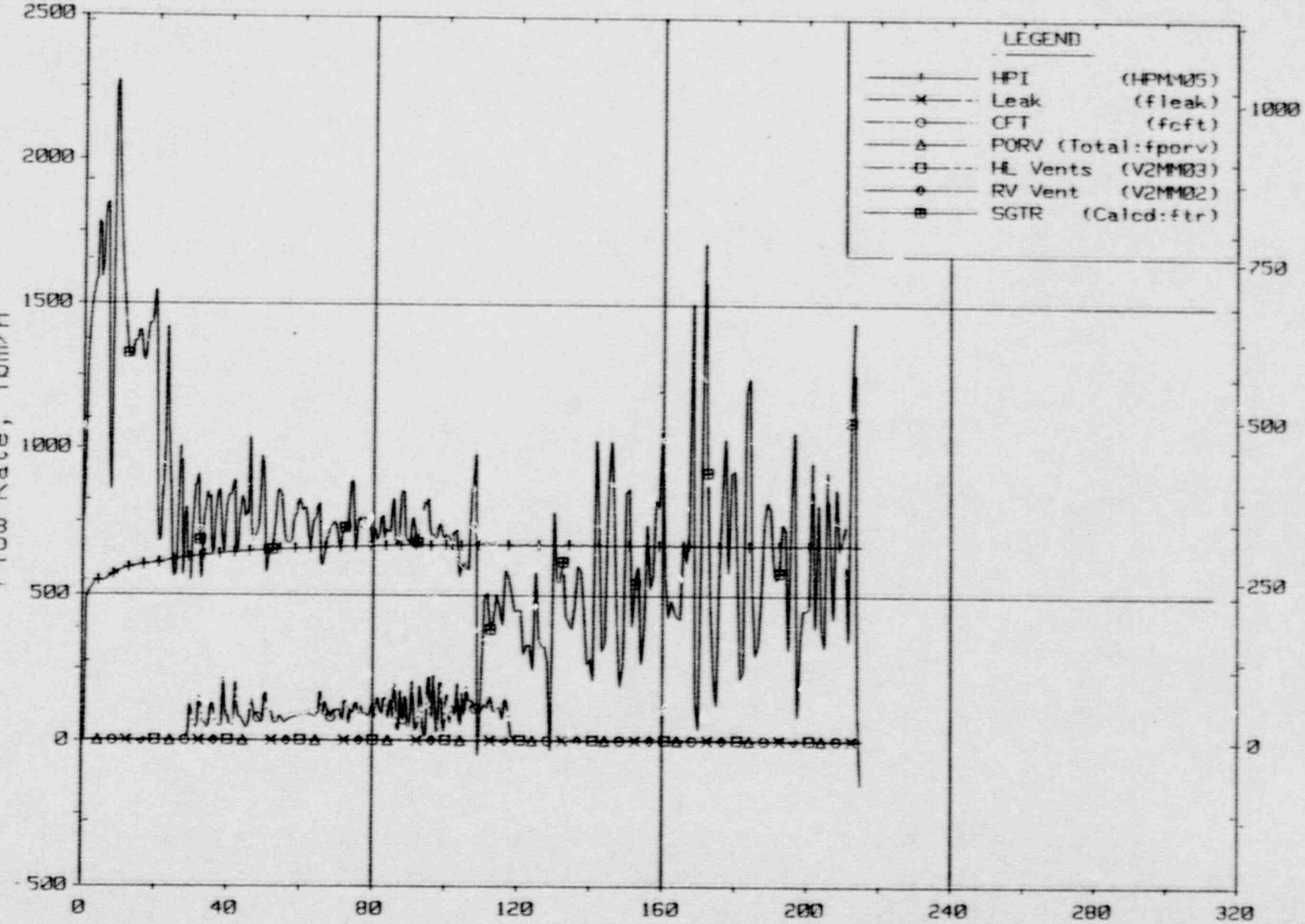
Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Thu Jun 2 09:42:35 1988

HLVF1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Br..k.



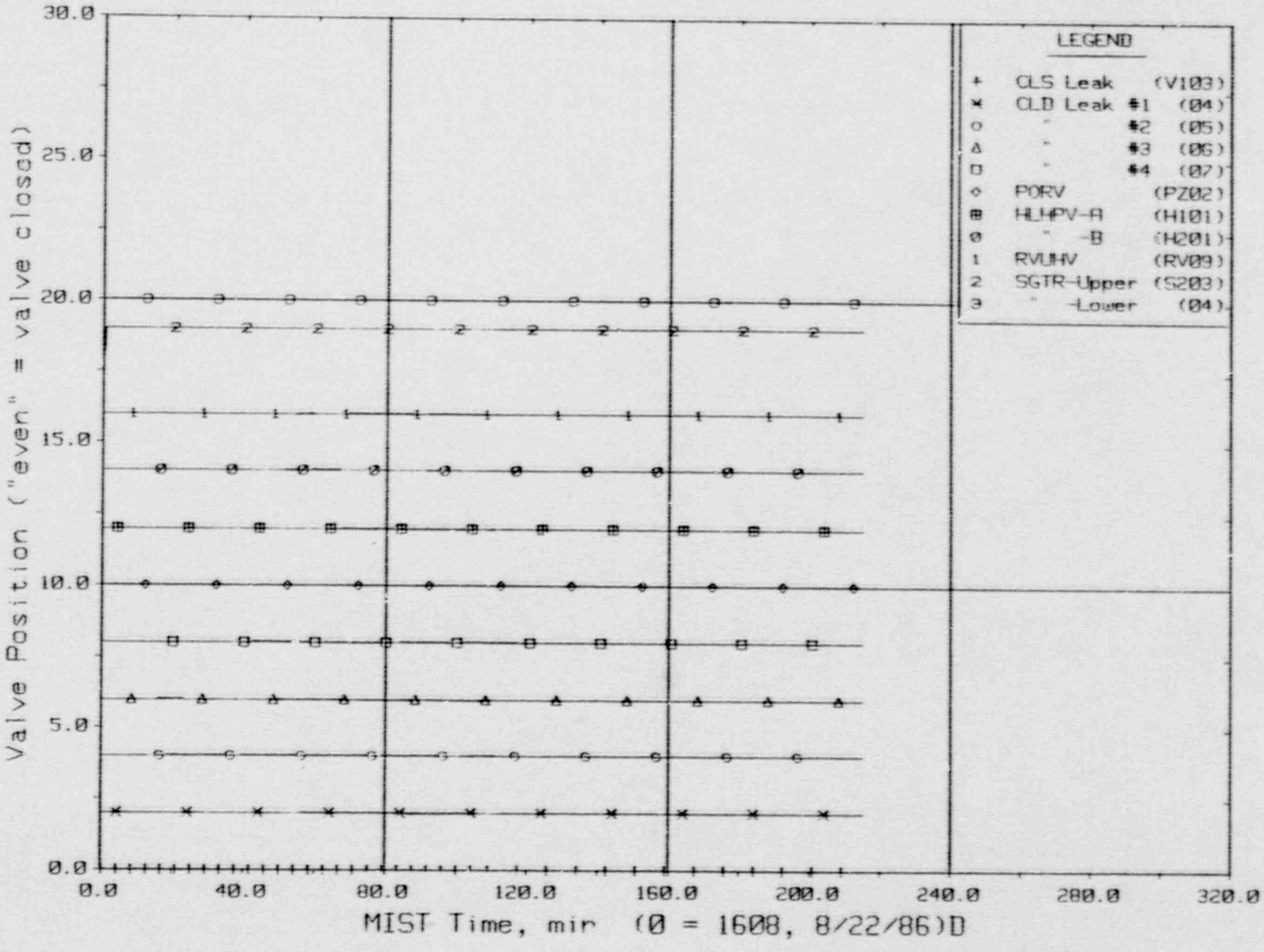
Primary System Boundary Flow Rates.

Thu Nov 3 17:27:06 1988

PRFL3

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



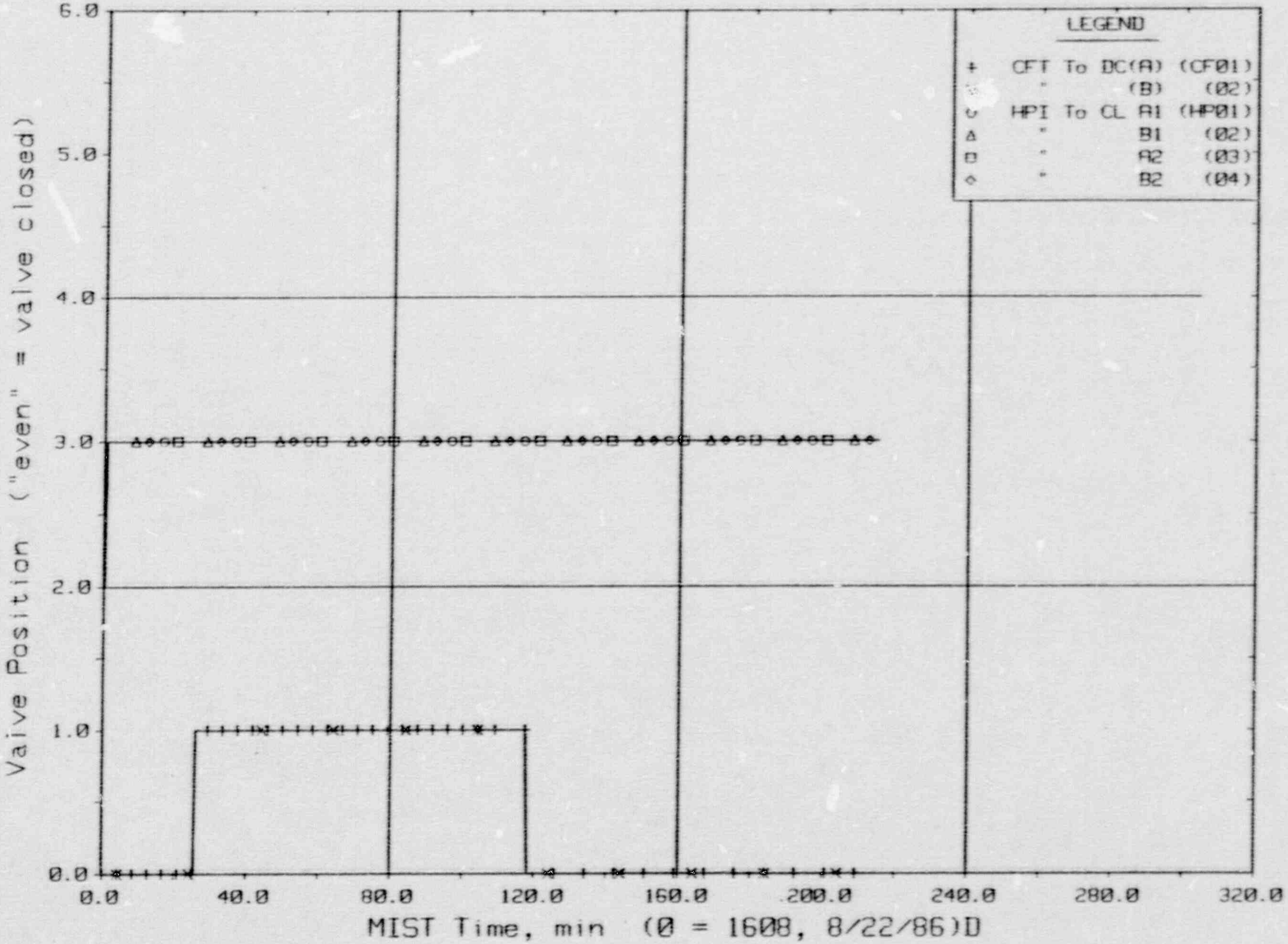
Primary System Discharge Limit Switch Indications (LSs).

Wed Jun 1 14:54:10 1988

PBLSI

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



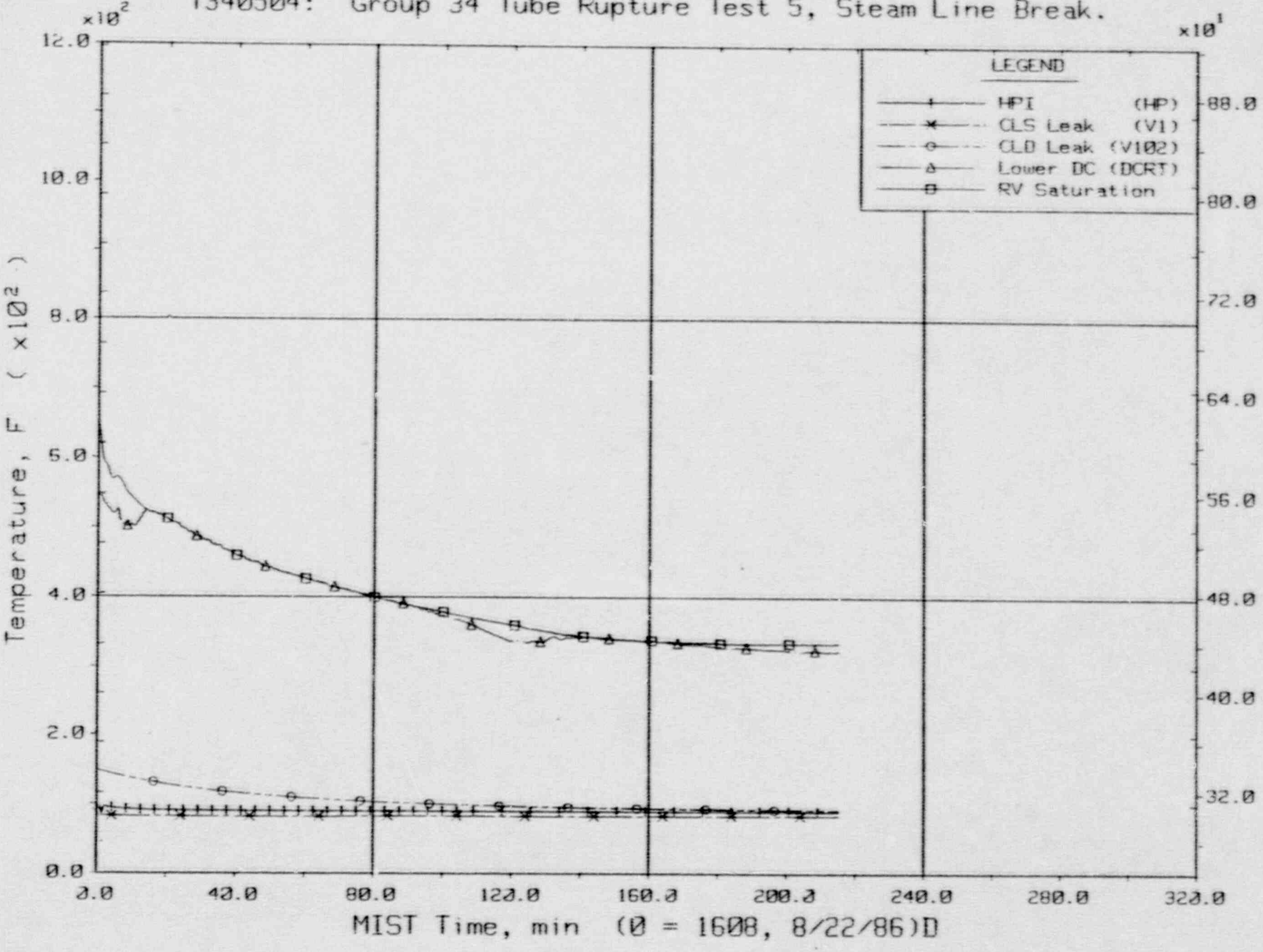
Primary System Injection Limit Switch Indications (LSs).

Wed Jun 1 15:03:32 1988

PBLS2

FINAL DATA

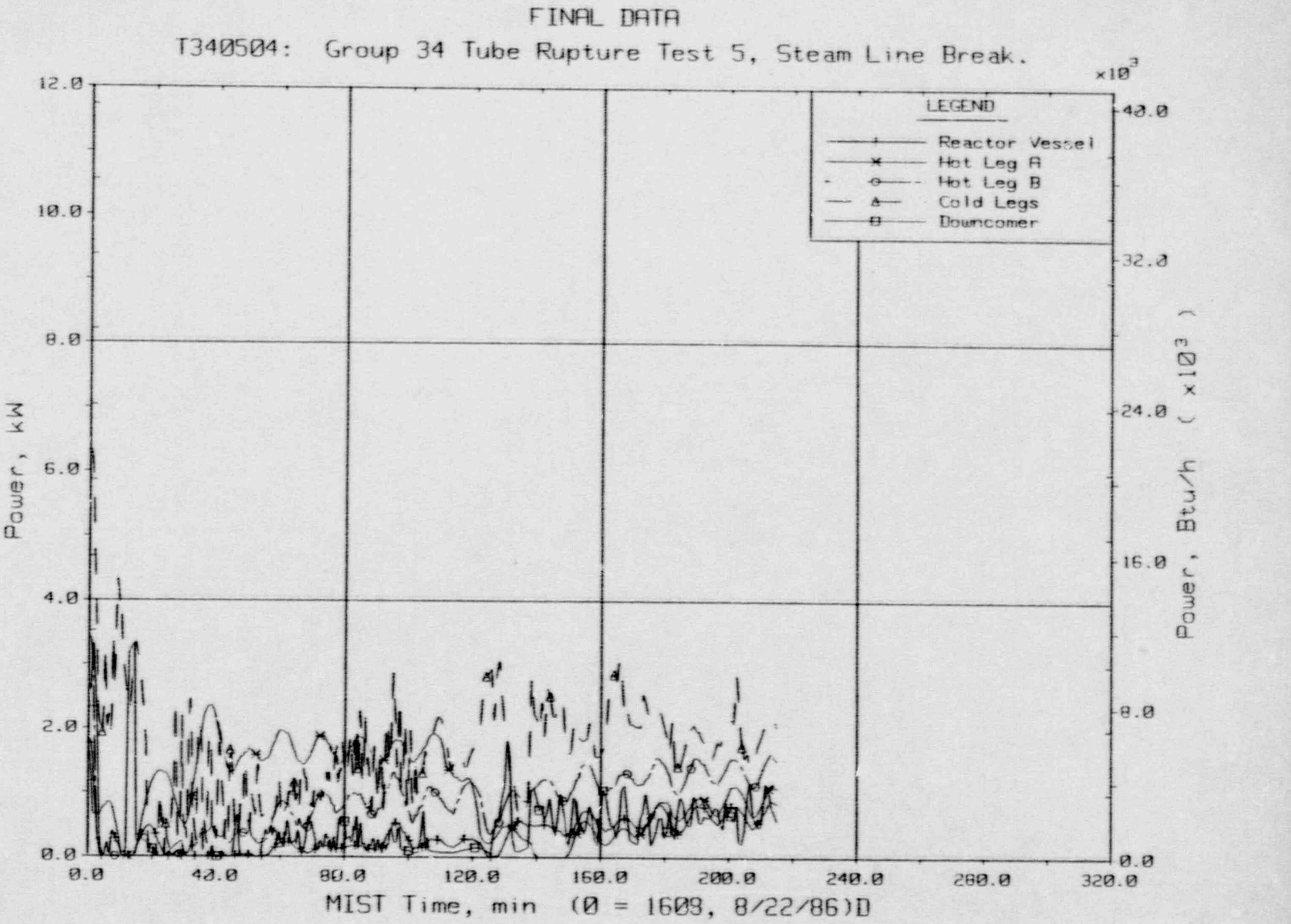
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Single-Phase Discharge and HPI Fluid Temperatures (TC01s).

Wed Jun 1 15:10:29 1988

PBTC1p



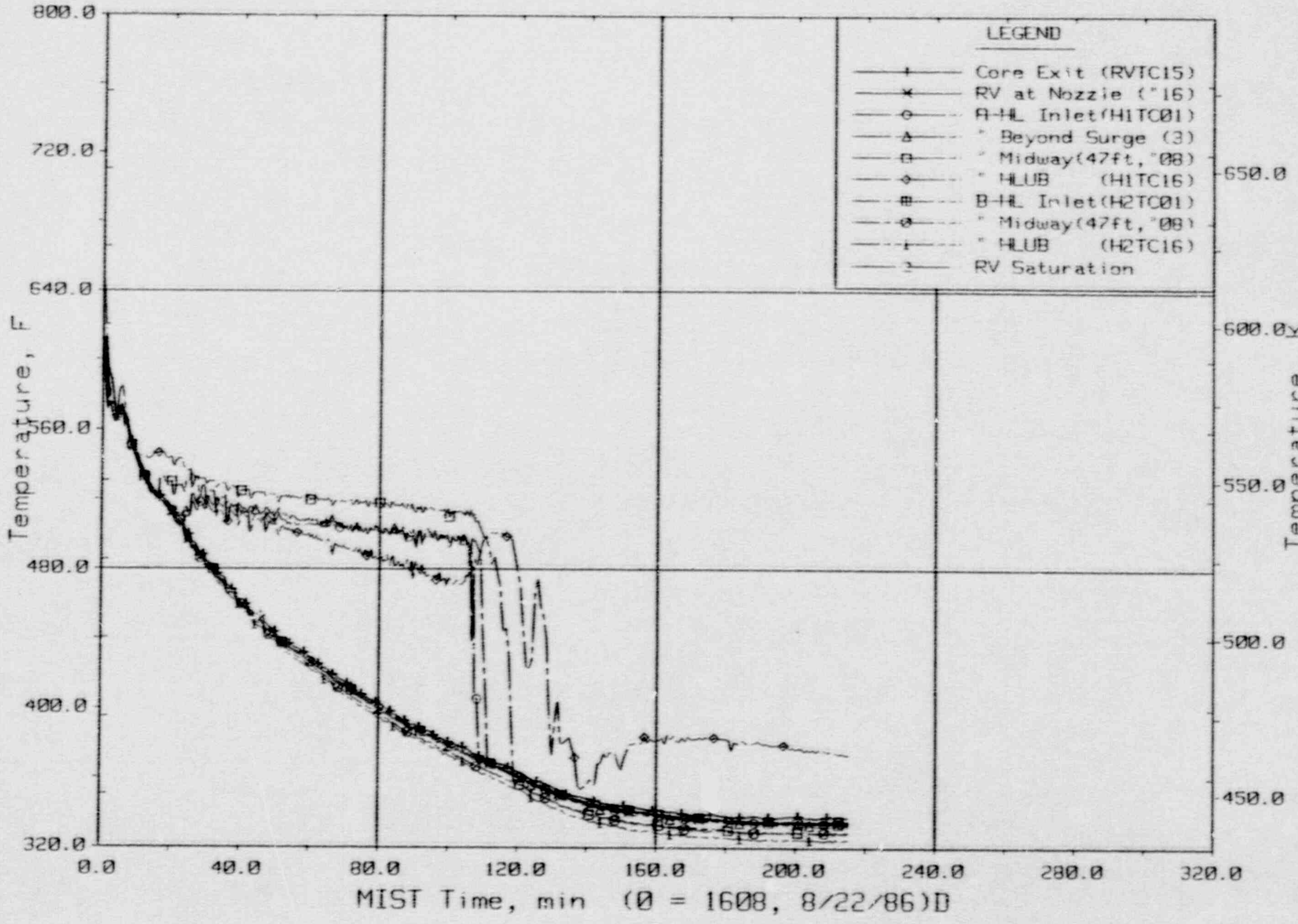
Guard Heater Specified Power Per Primary Component.

Wed Jun 1 15:35:34 1988

PRGQ1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



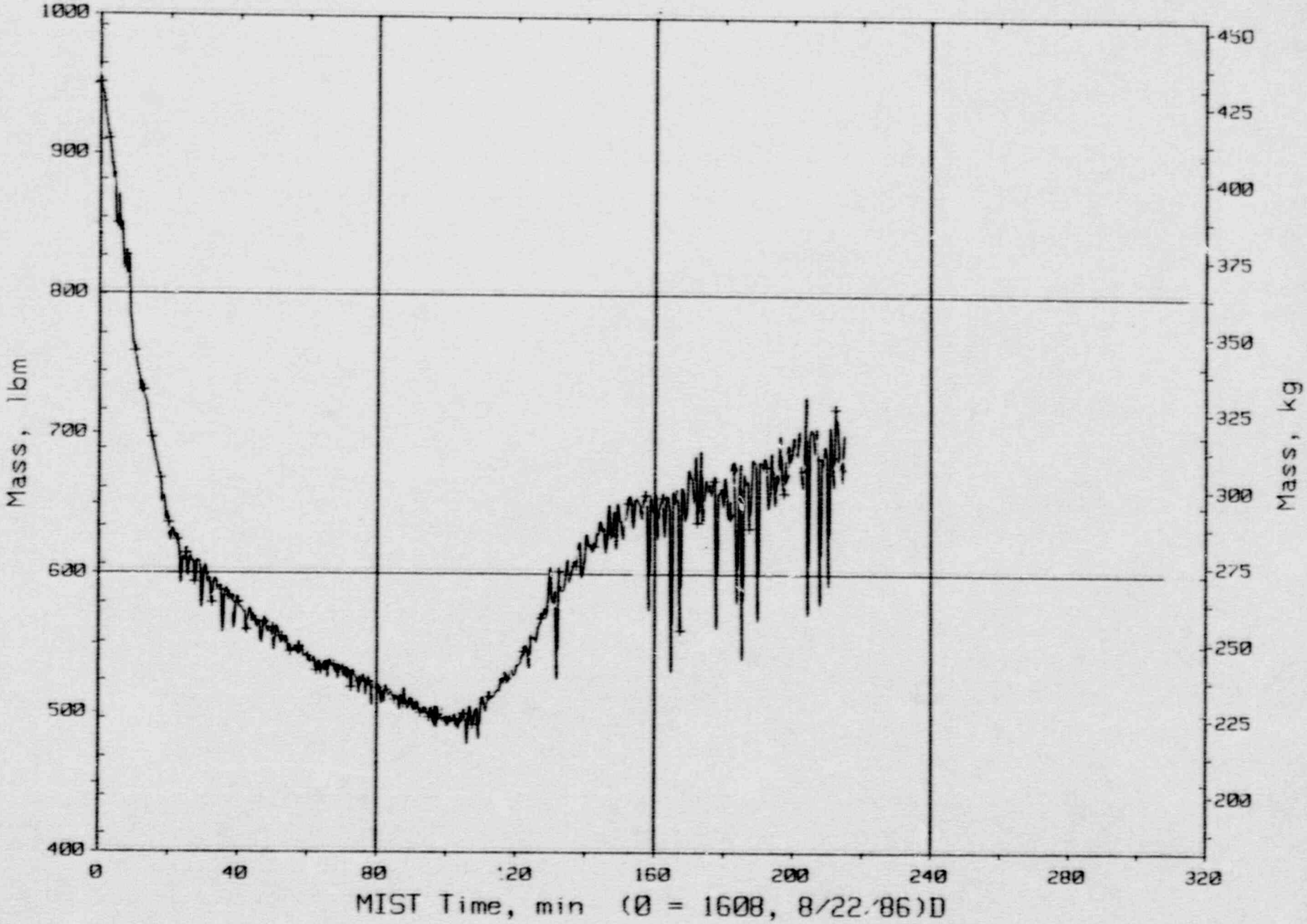
Composite Core Exit and Hot Leg Fluid Temperatures.

Wed Jun 1 15:40:25 1988

PRKT1

FINAL DATA

T340504: Group 34 Tube Pupture Test 5, Steam Line Break.



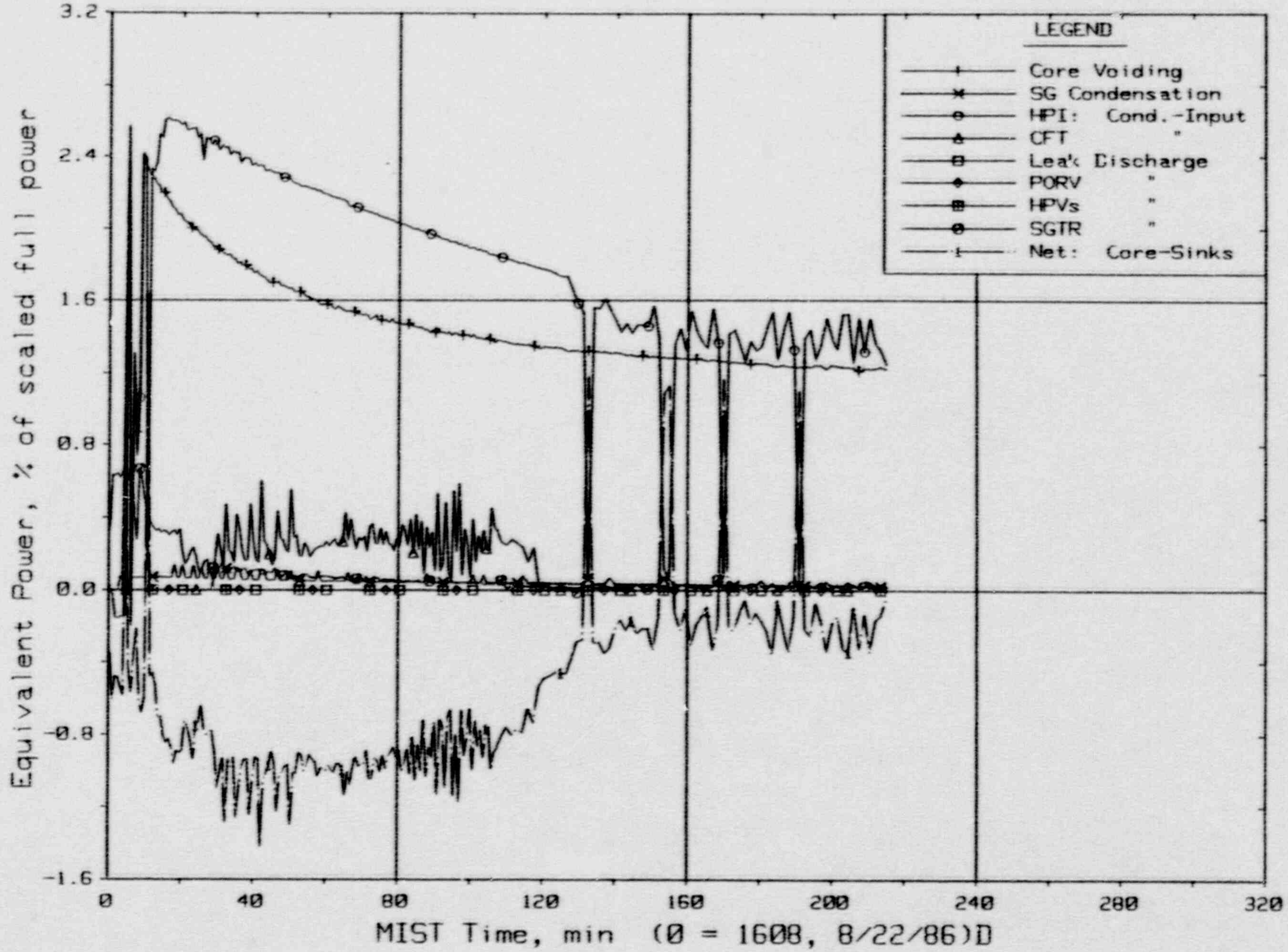
Indicated Primary System Total Fluid Mass (PLML20).

Wed Nov 2 14:08:51 1988

PRML9

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



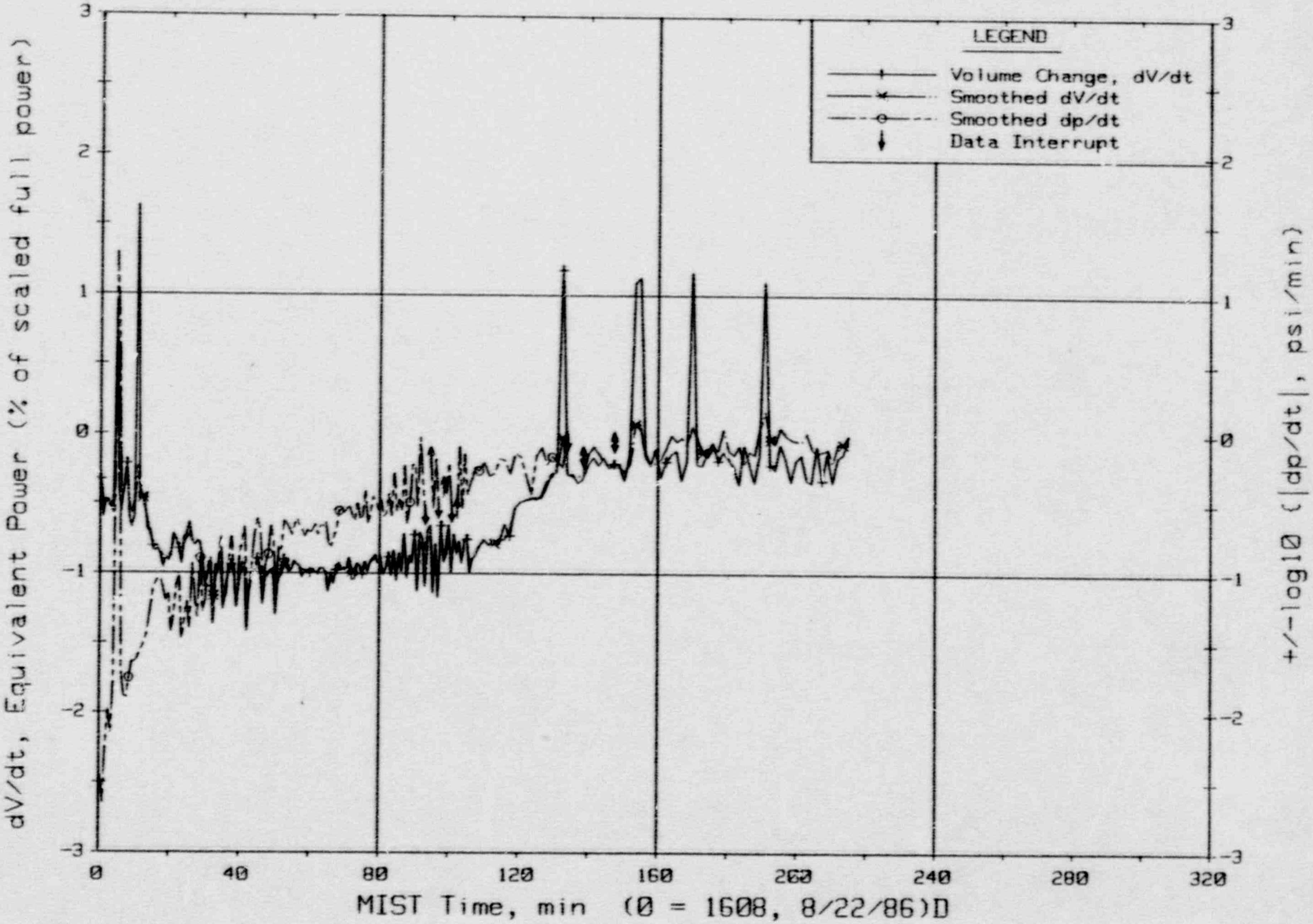
Primary Fluid Volume Changes By Components.

Fri Nov 4 13:54:22 1988

PRPA0

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



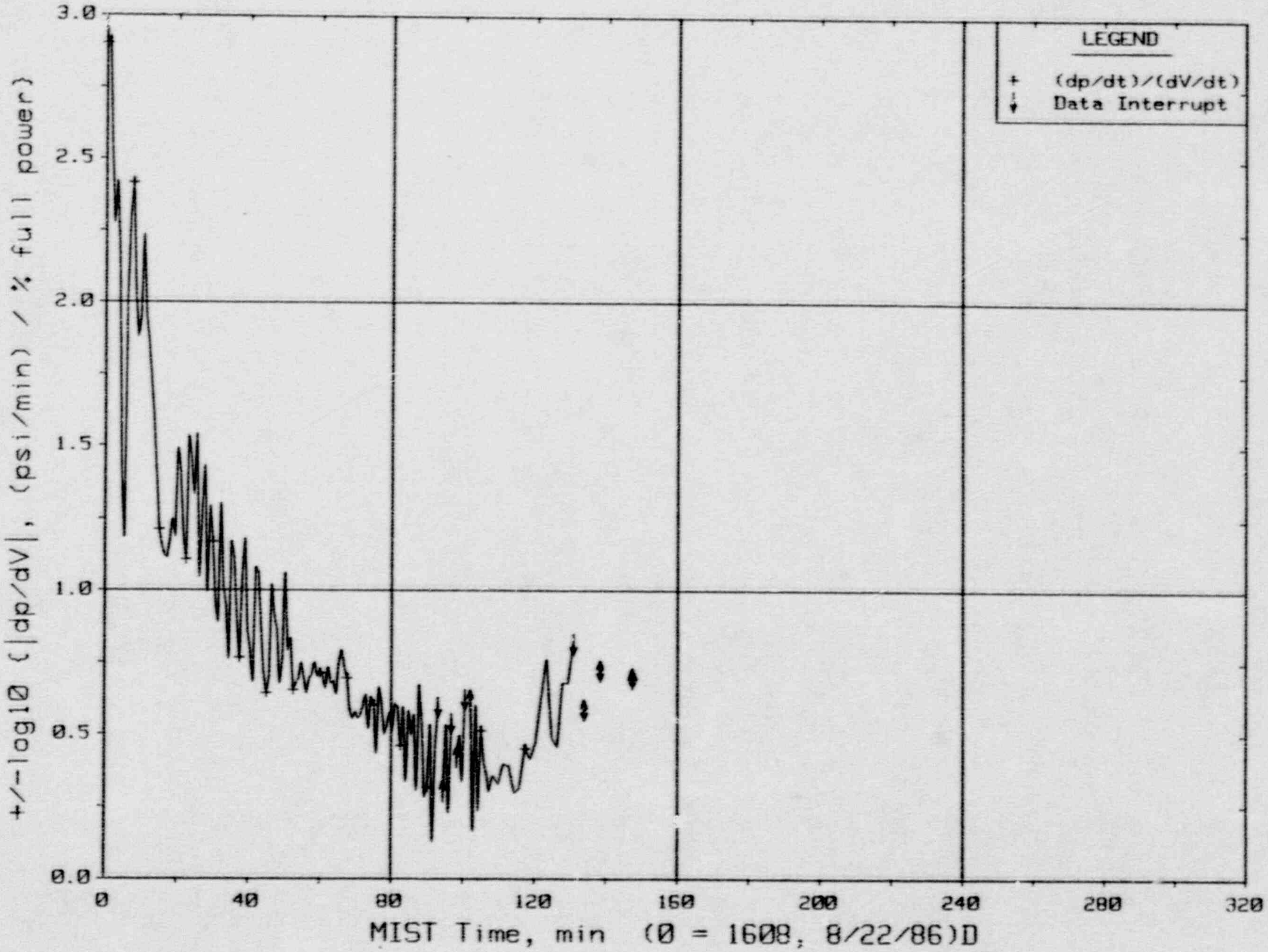
Primary Fluid Volume and Pressure Changes, dV/dt and dp/dt.

Fri Nov 4 13:57:11 1988

PRPR2

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



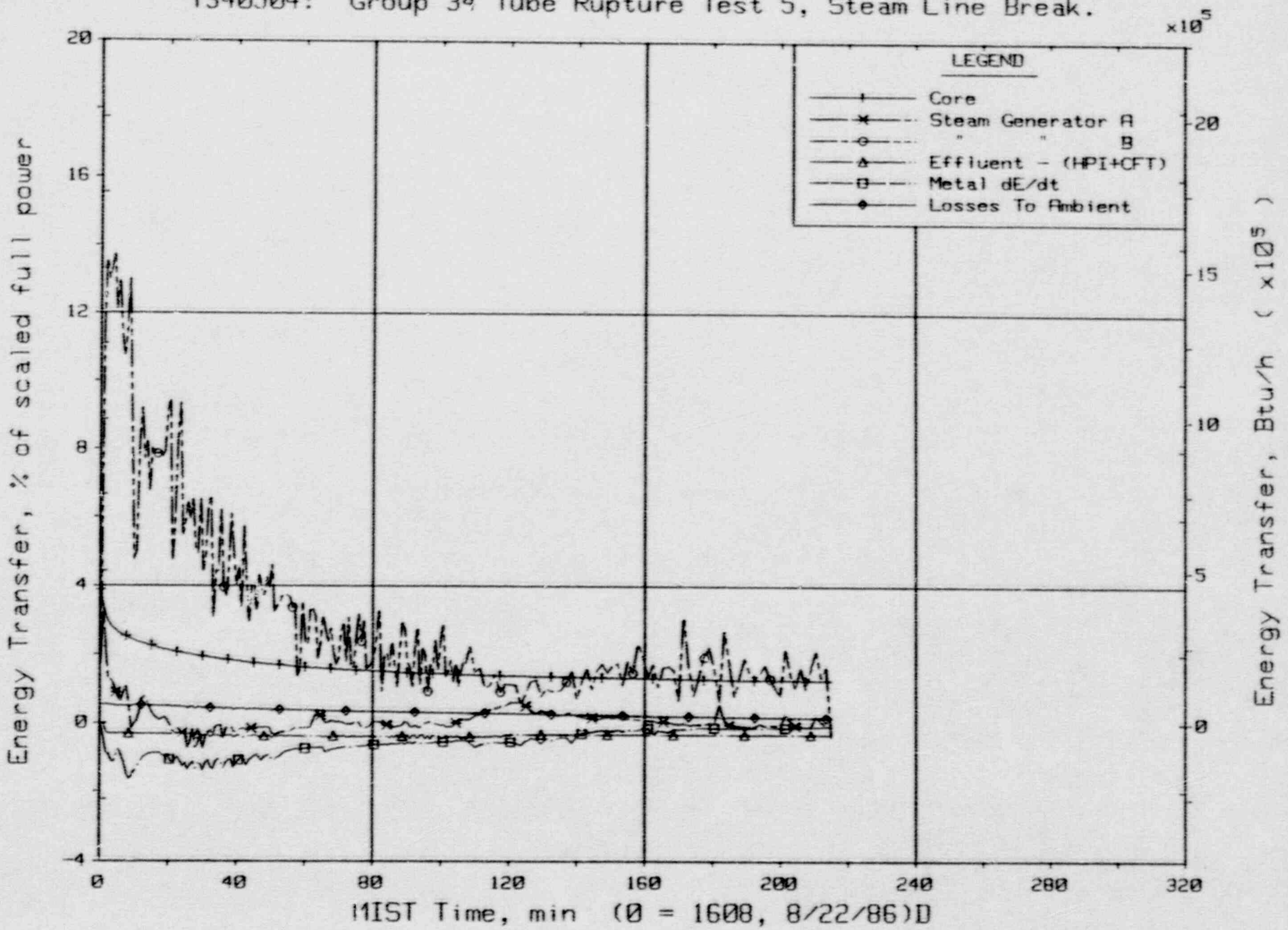
Primary System Response:" dp/dt divided by dV/dt.

Fri Nov 4 13:59:54 1988

PRPA3

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



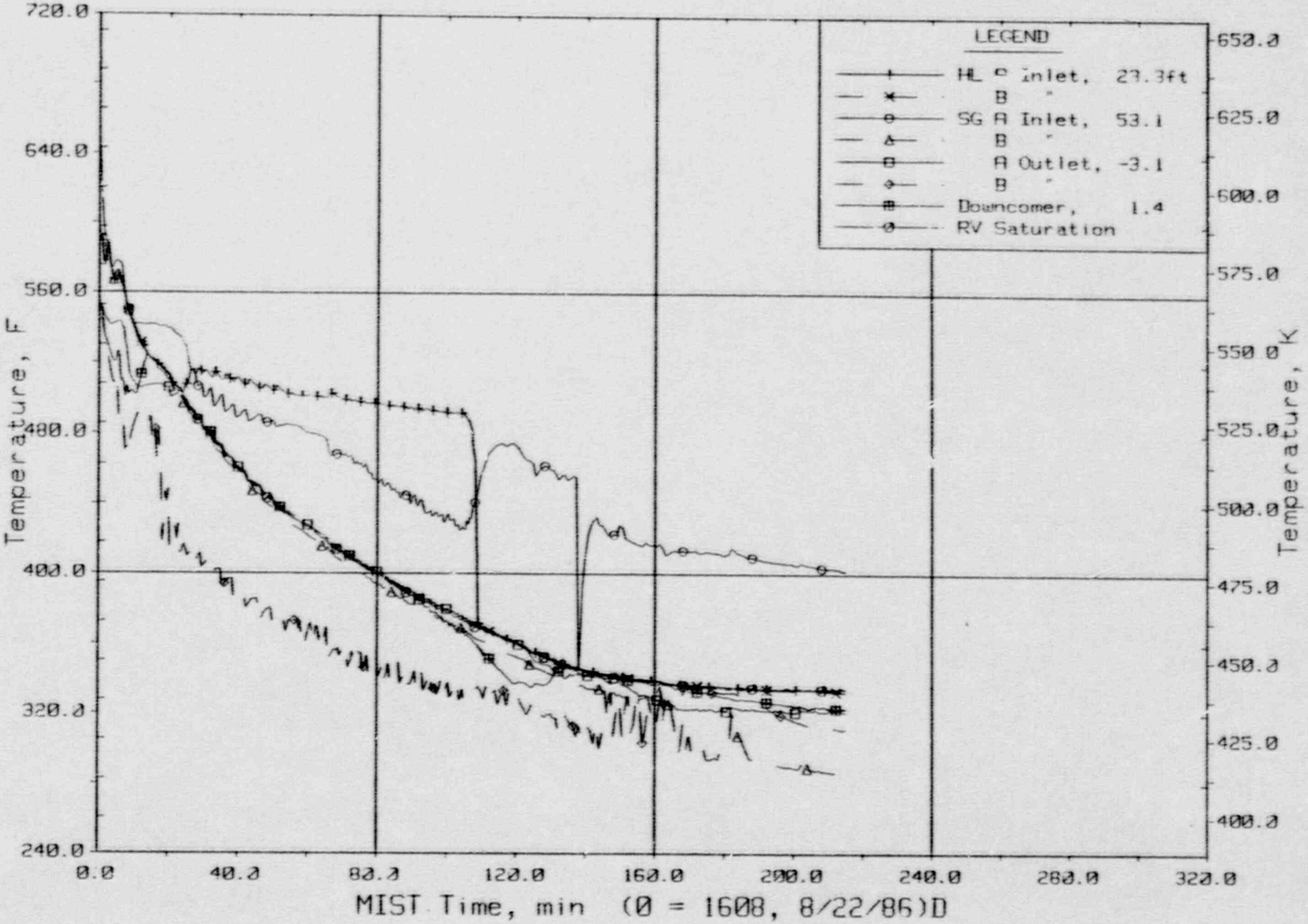
Primary System Energy Transfer.

Fri Nov 4 14:03:18 1988

PROE2

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



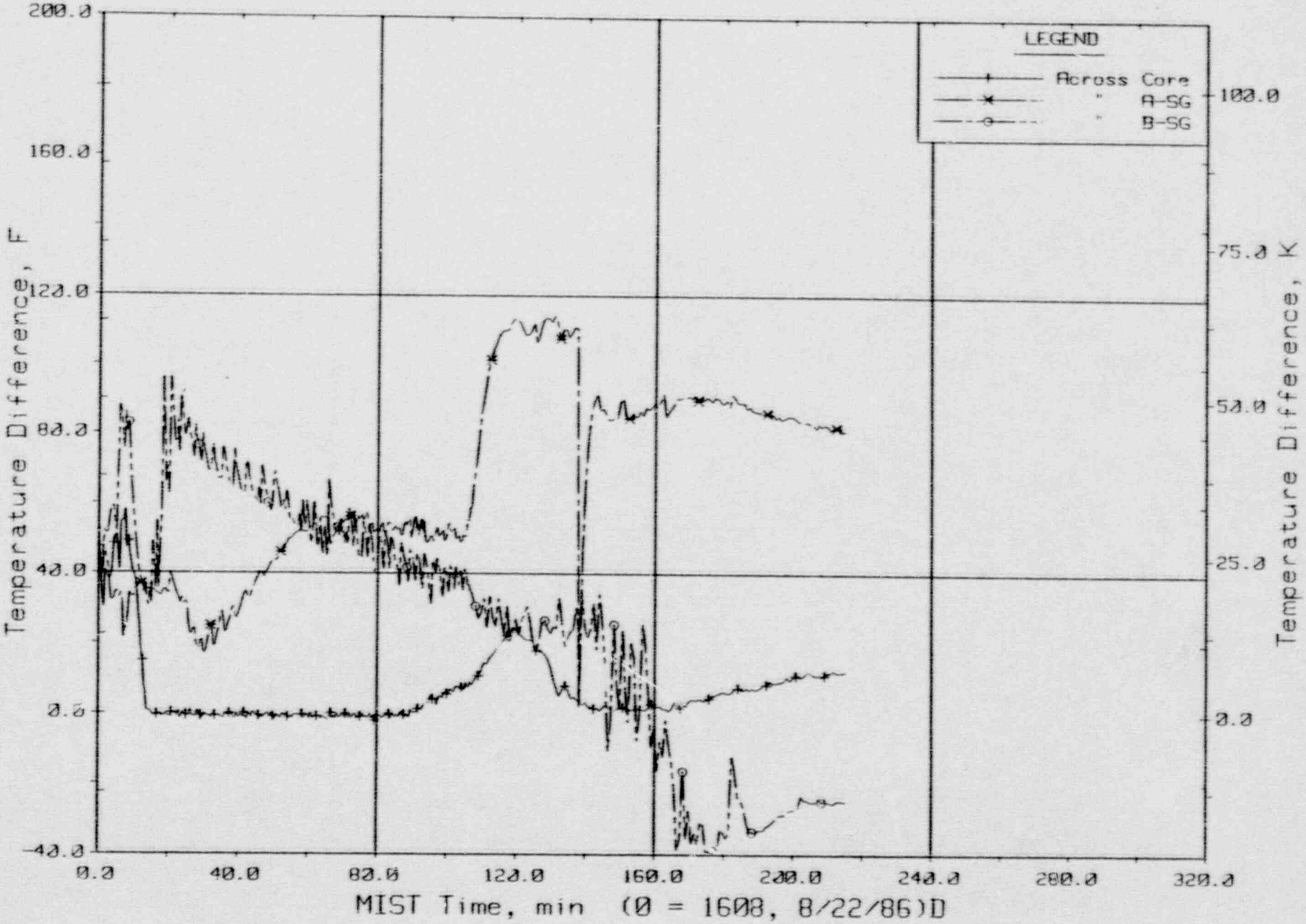
Primary System Fluid Temperatures (RTDs).

Wed Jun 1 15:09:02 1988

PRRT0

FINAL DATA

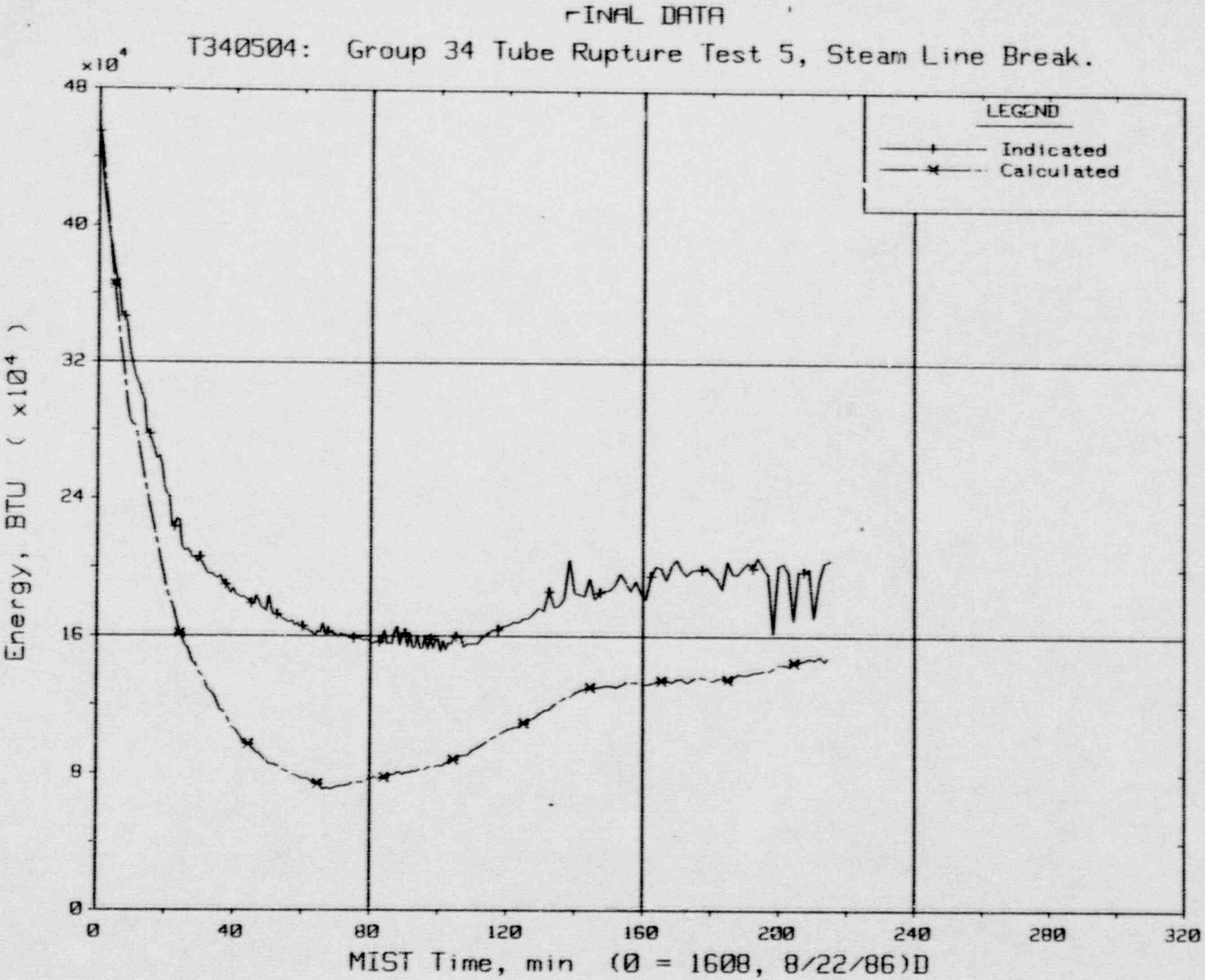
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Key Temperature Differences.

Wed Jun 1 16:13:11 1988

PRTD2



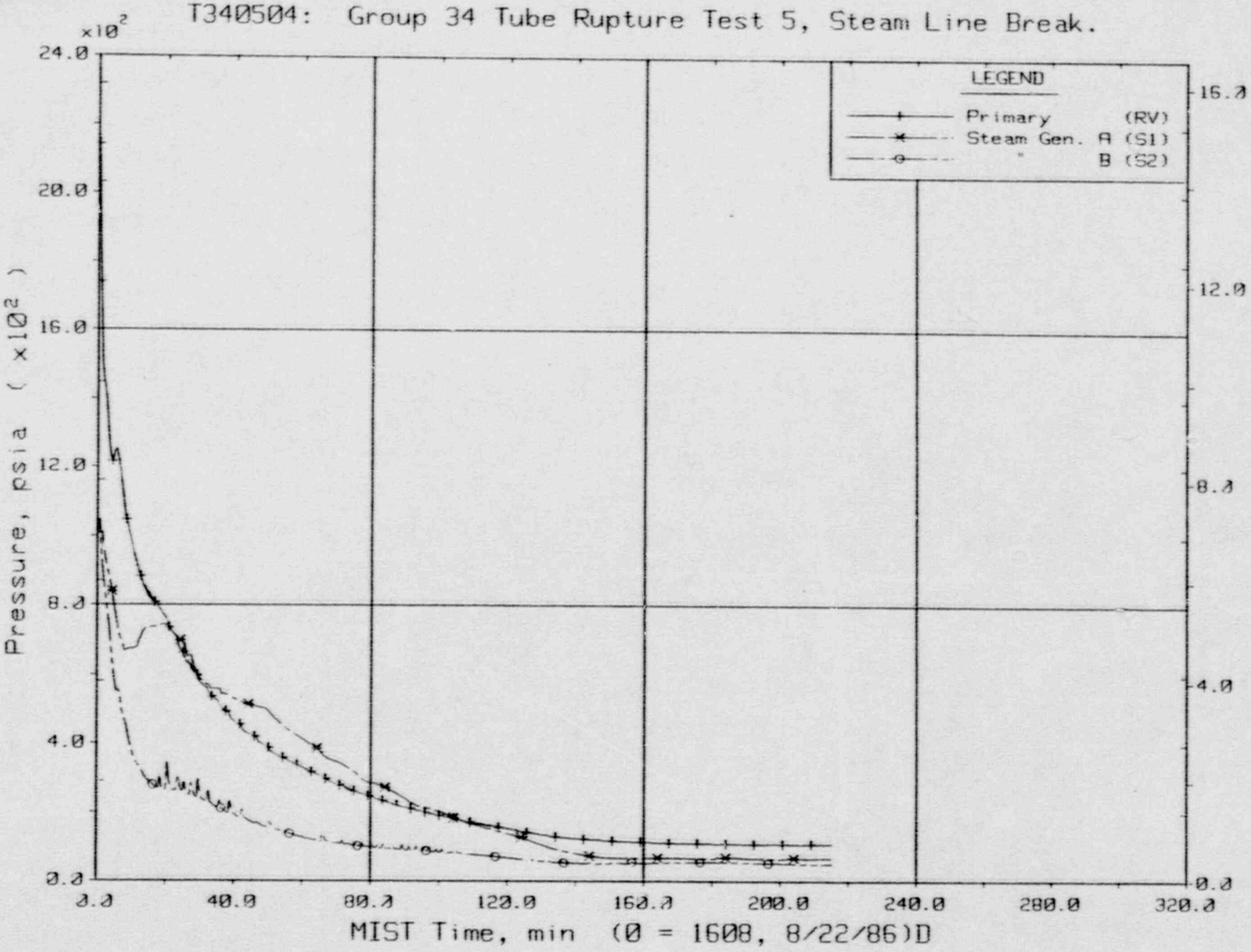
Primary System Total Fluid Energy.

Fri Nov 4 14:05:58 1988

PRTE2

FINAL DATA

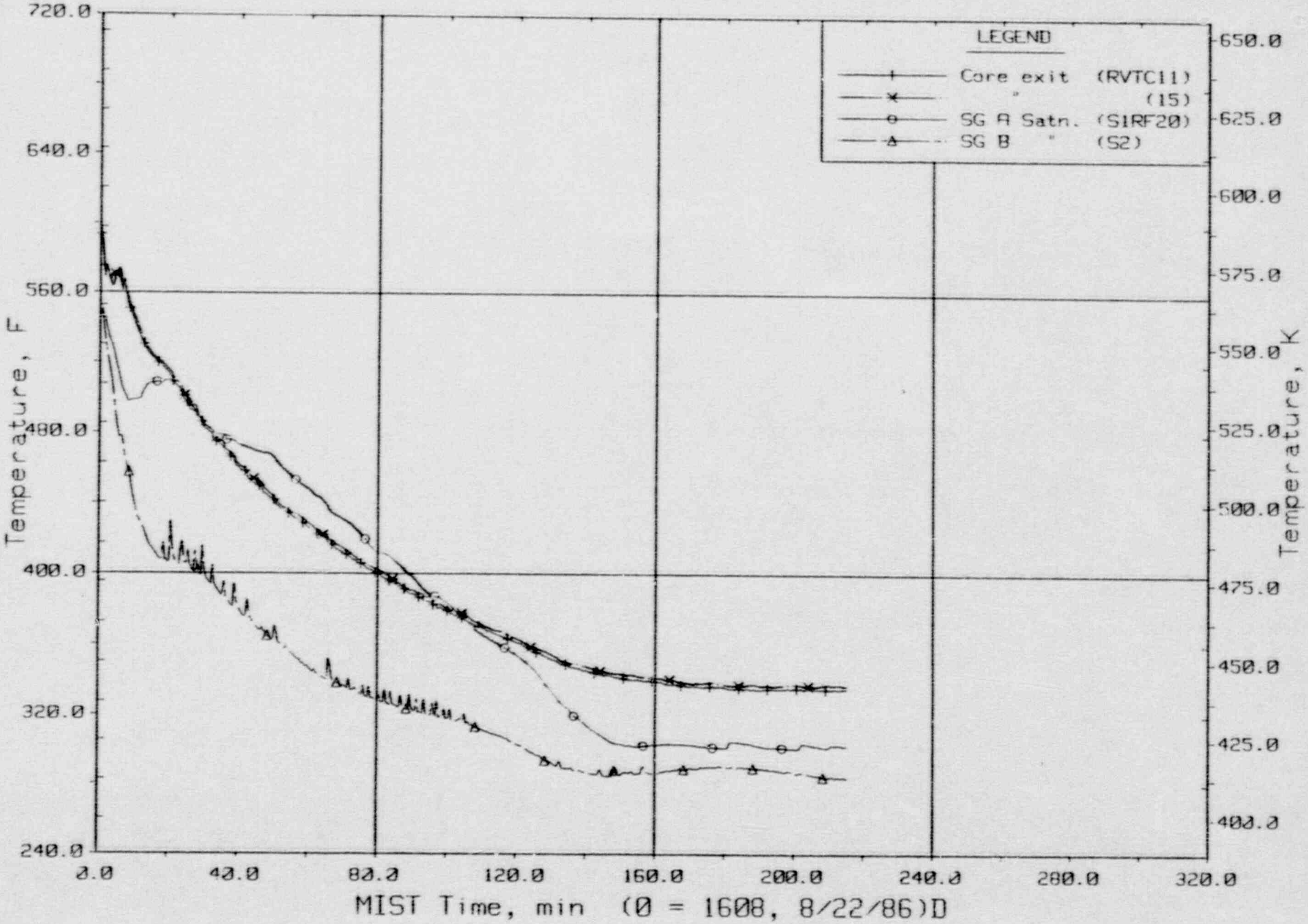
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Primary and Secondary System Pressures (GP01s).

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



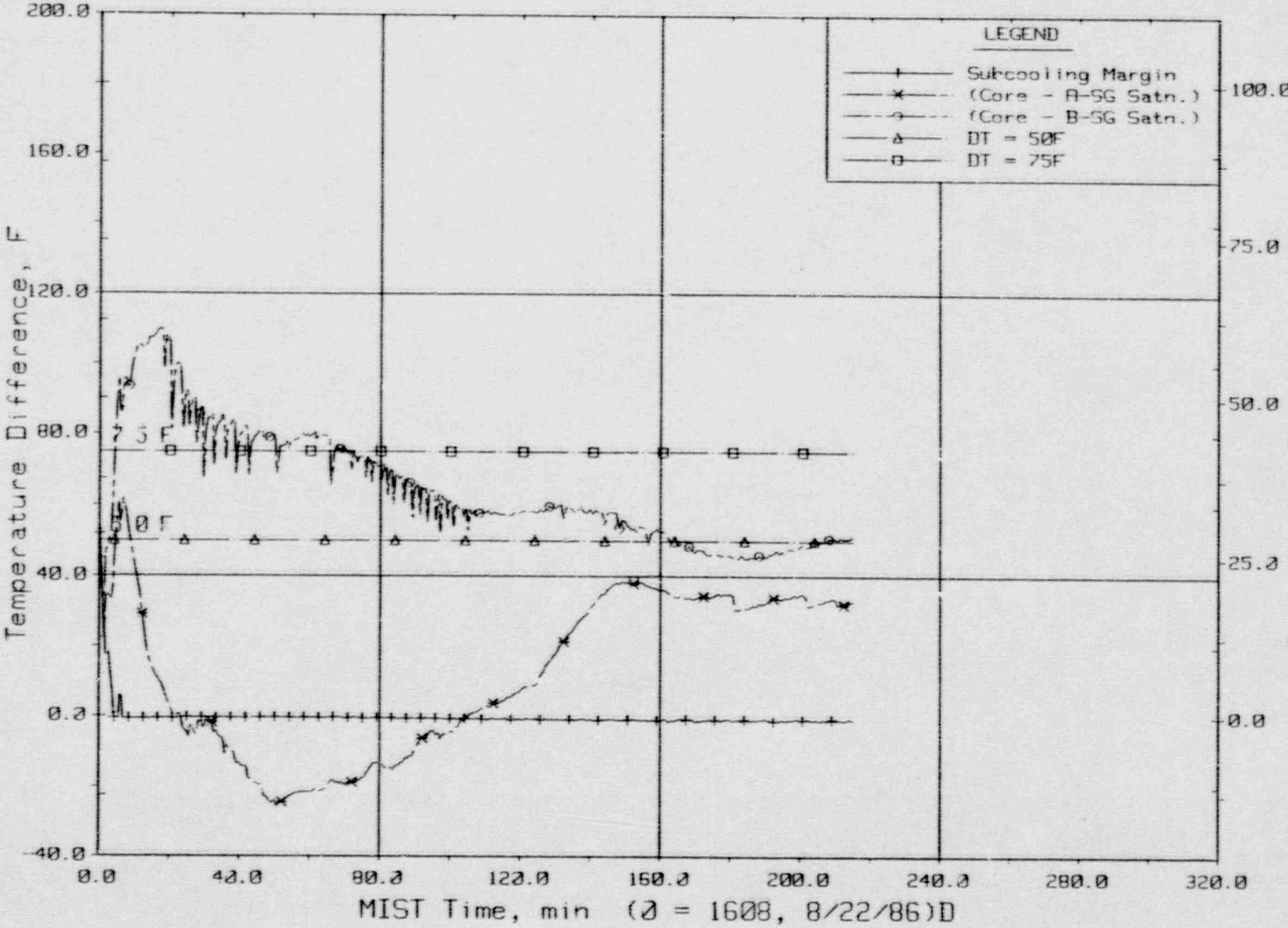
Core Exit and Steam Generator Secondary Saturation Temperatures.

Wed Jun 1 16:29:25 1988

PSTC3p

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



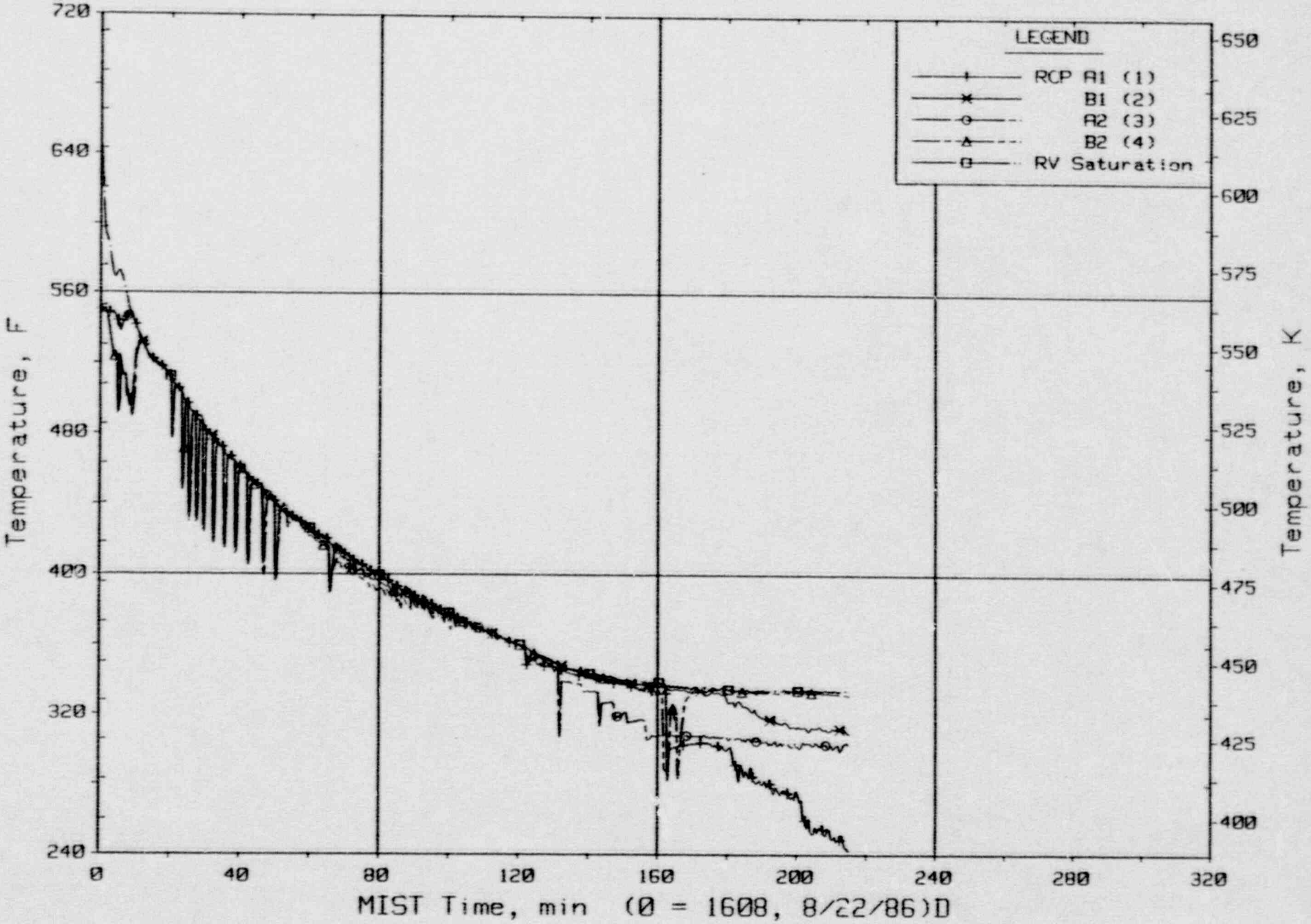
Control Temperature Differences.

Wed Jun 1 16:32:39 1988

PSTD1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



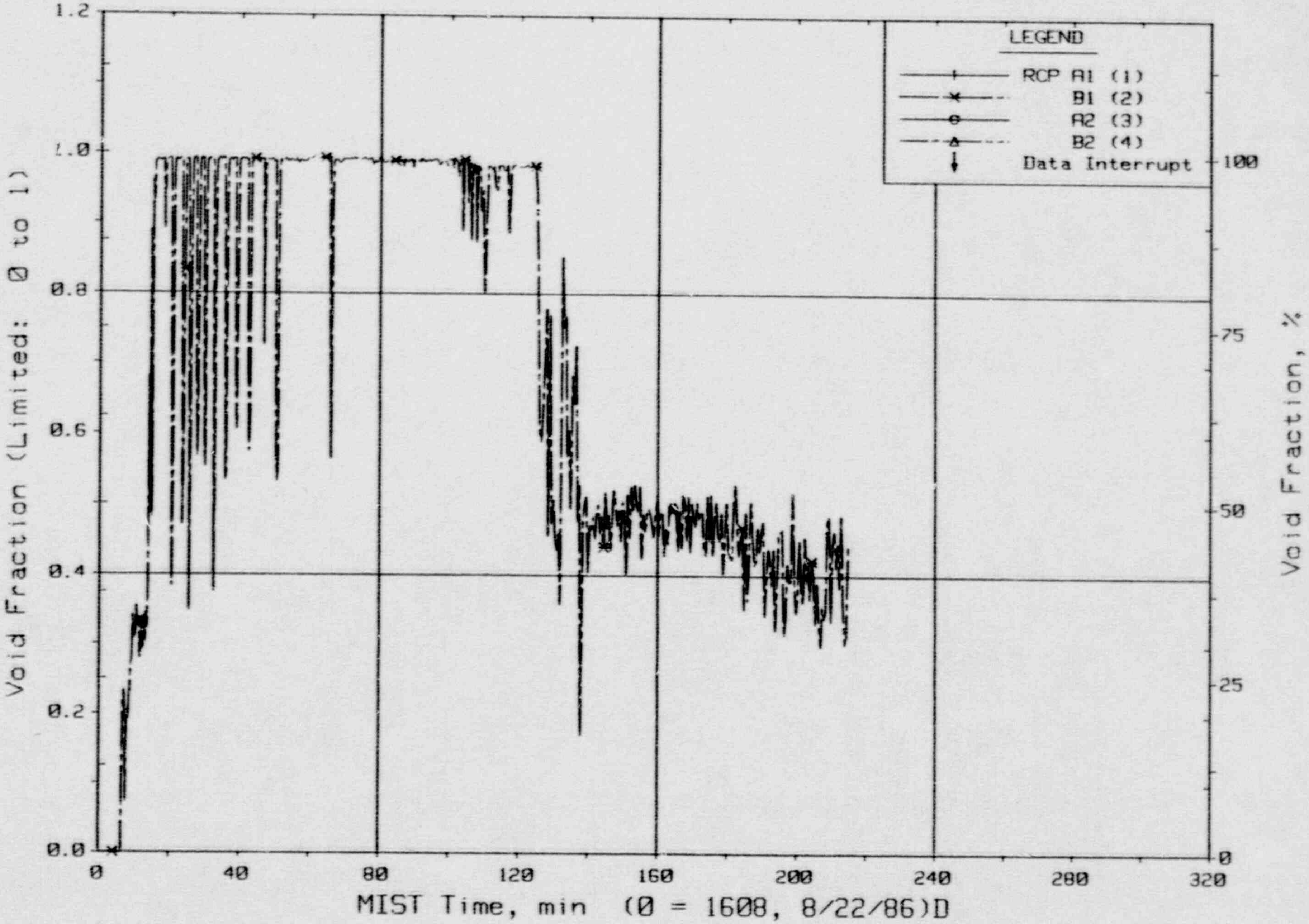
Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 14:48:53 1988

PURT0

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



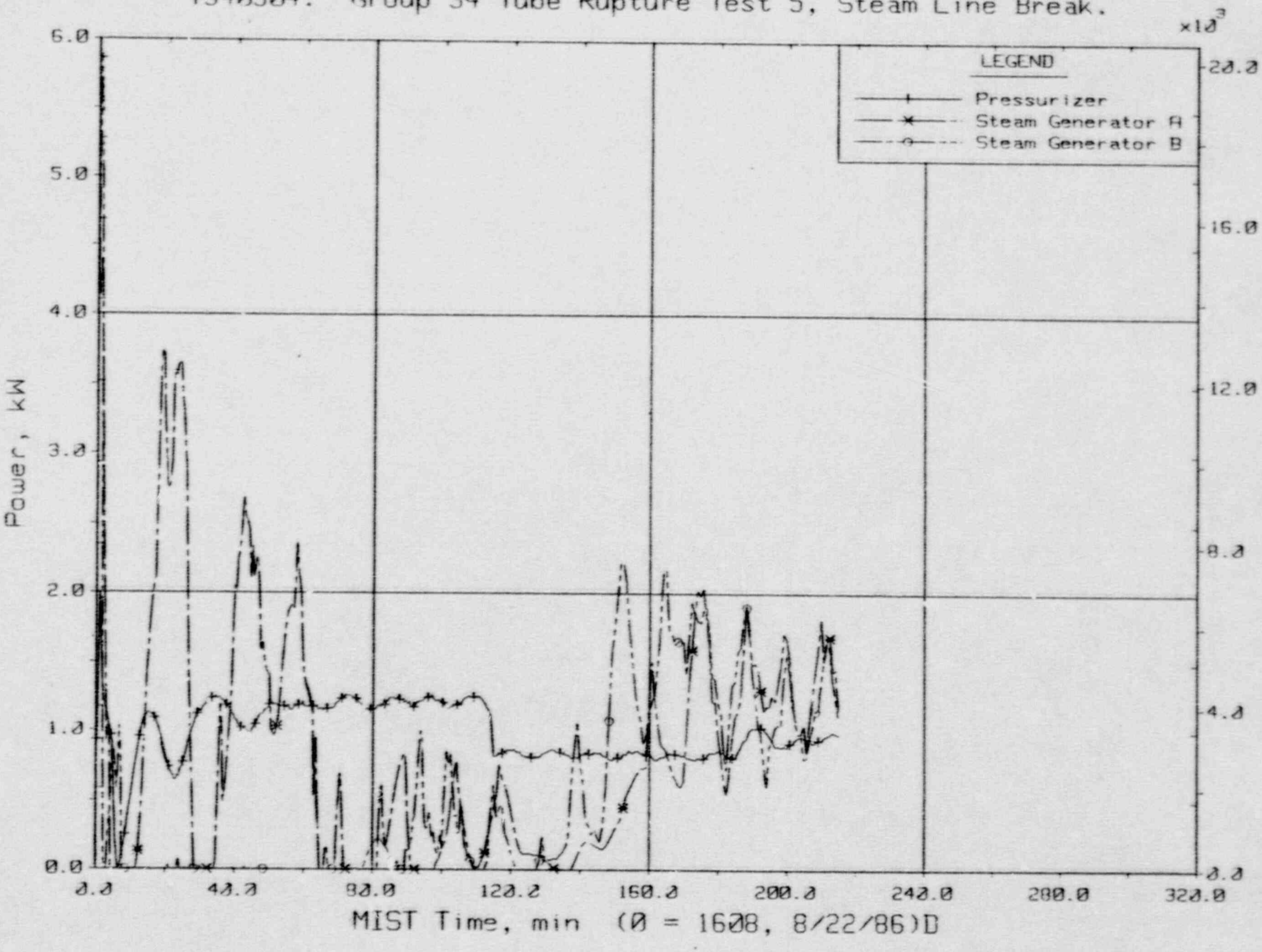
Pump Suction Void Fraction From Gamma Densitometers (CnGD21).

Tue Nov 1 14:52:05 1988

PUVF2

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



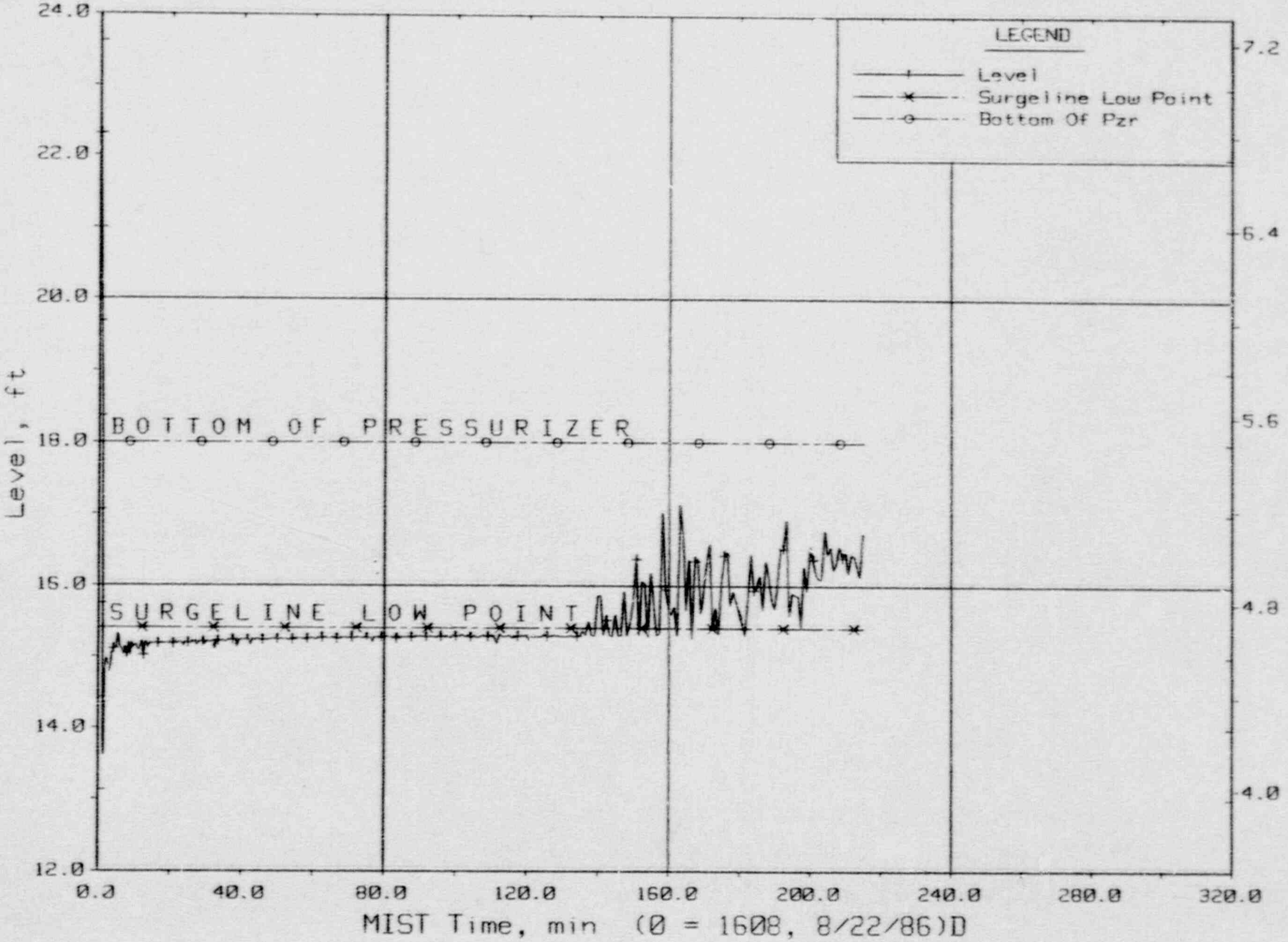
Guard Heater Specified Power. Pressurizer and Steam Generators.

Wed Jun 1 15:37:52 1988

PZG01

FINAL DATA

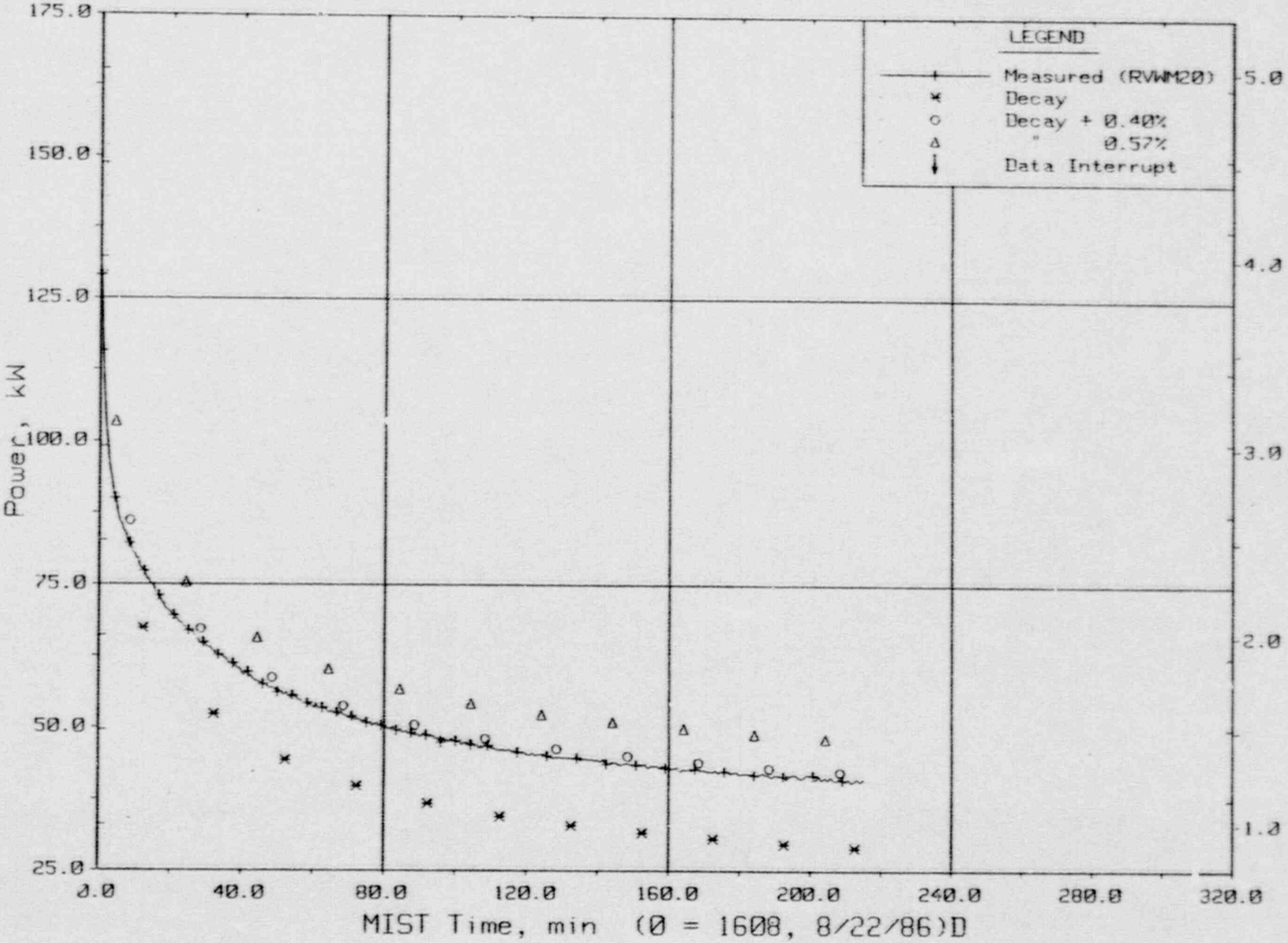
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Pressurizer Collapsed Liquid Level (PZLV20).

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



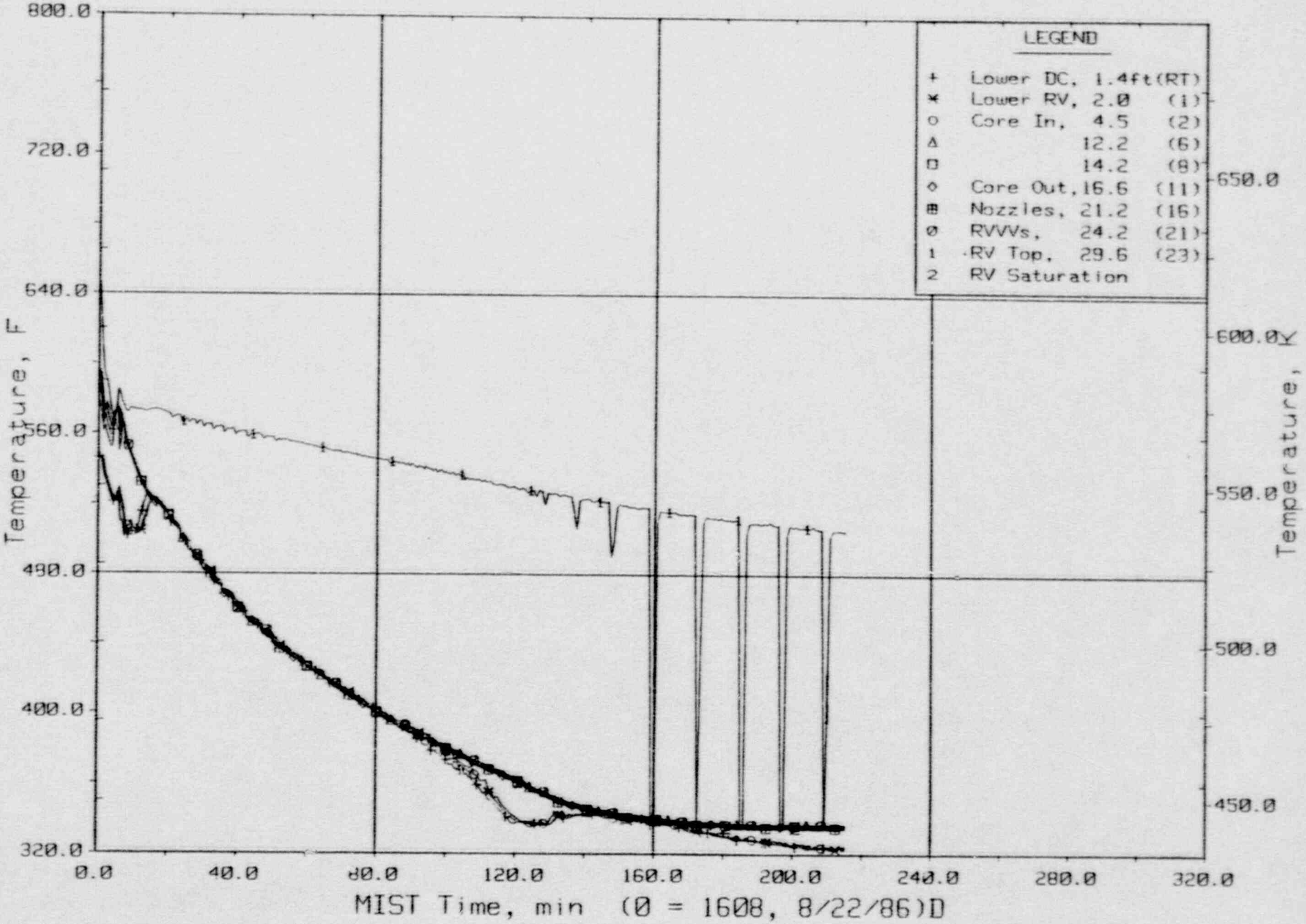
Core Power.

Wed Jun 1 16:43:20 1989

RVKW0

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



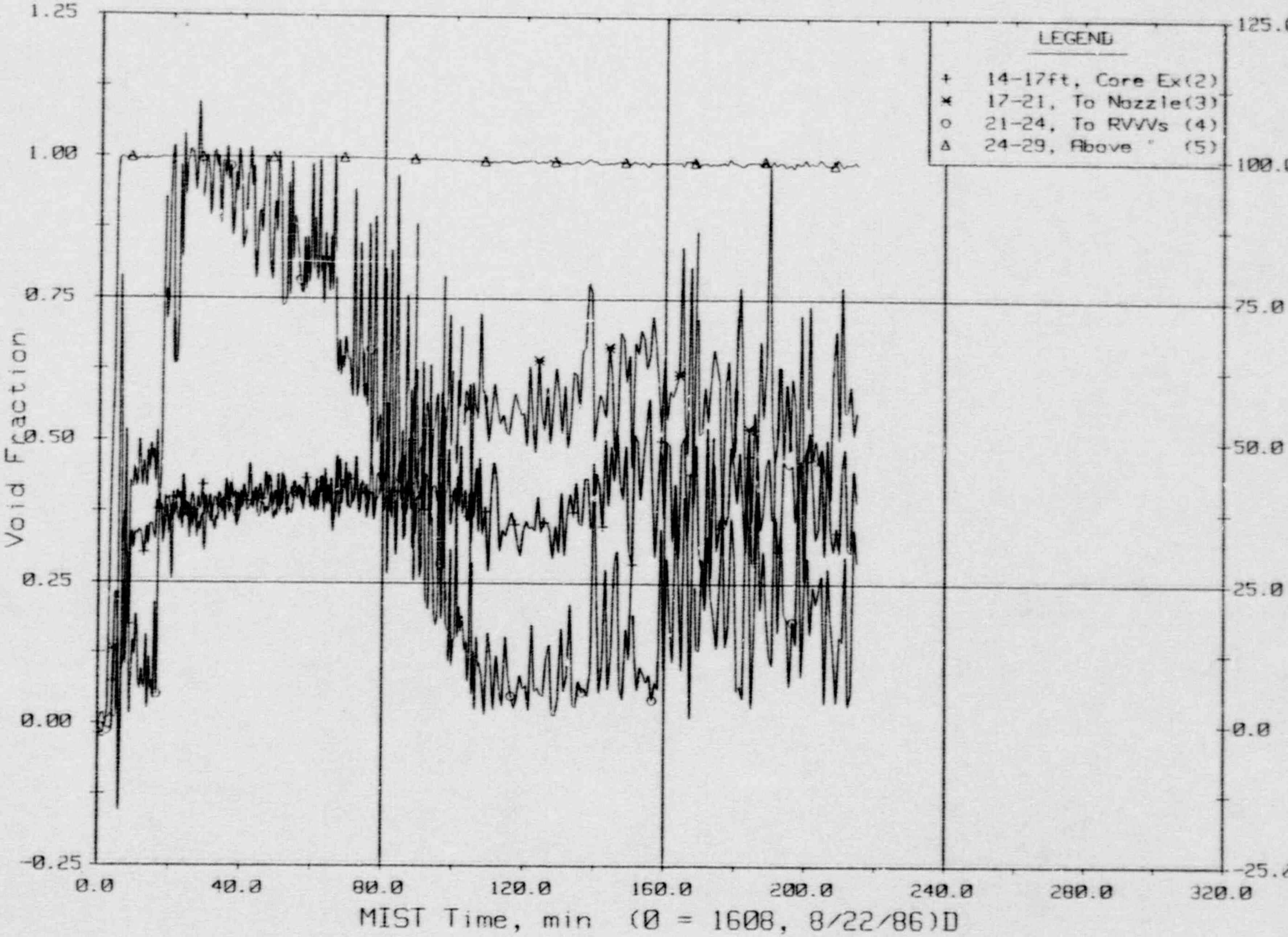
Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Wed Jun 1 16:46:49 1988

RVTC0

FINAL DATA

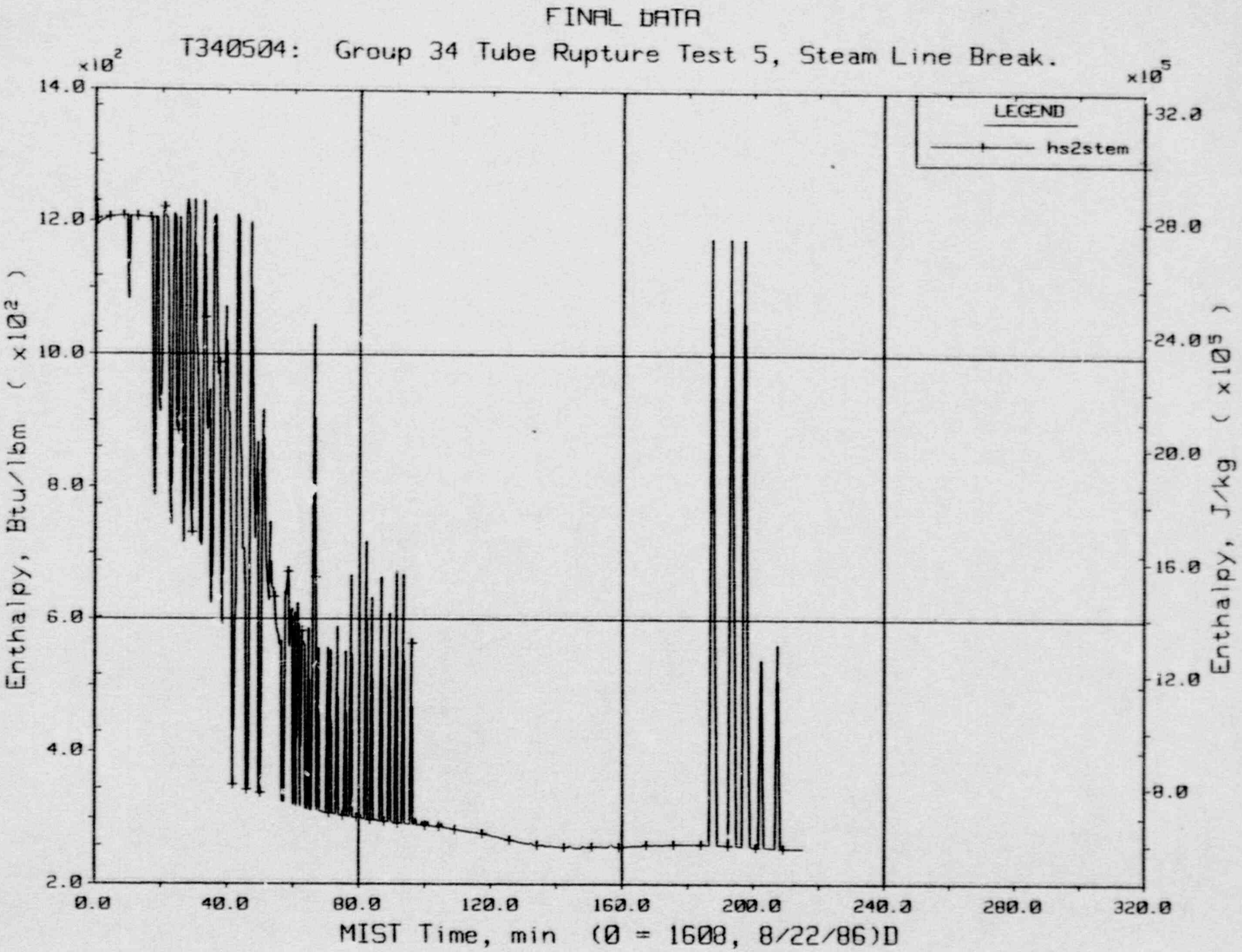
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Reactor Vessel Void Fractions From Differential Pressures (RVVFs).

Wed Jun 1 16:49:50 1988

RVVF1



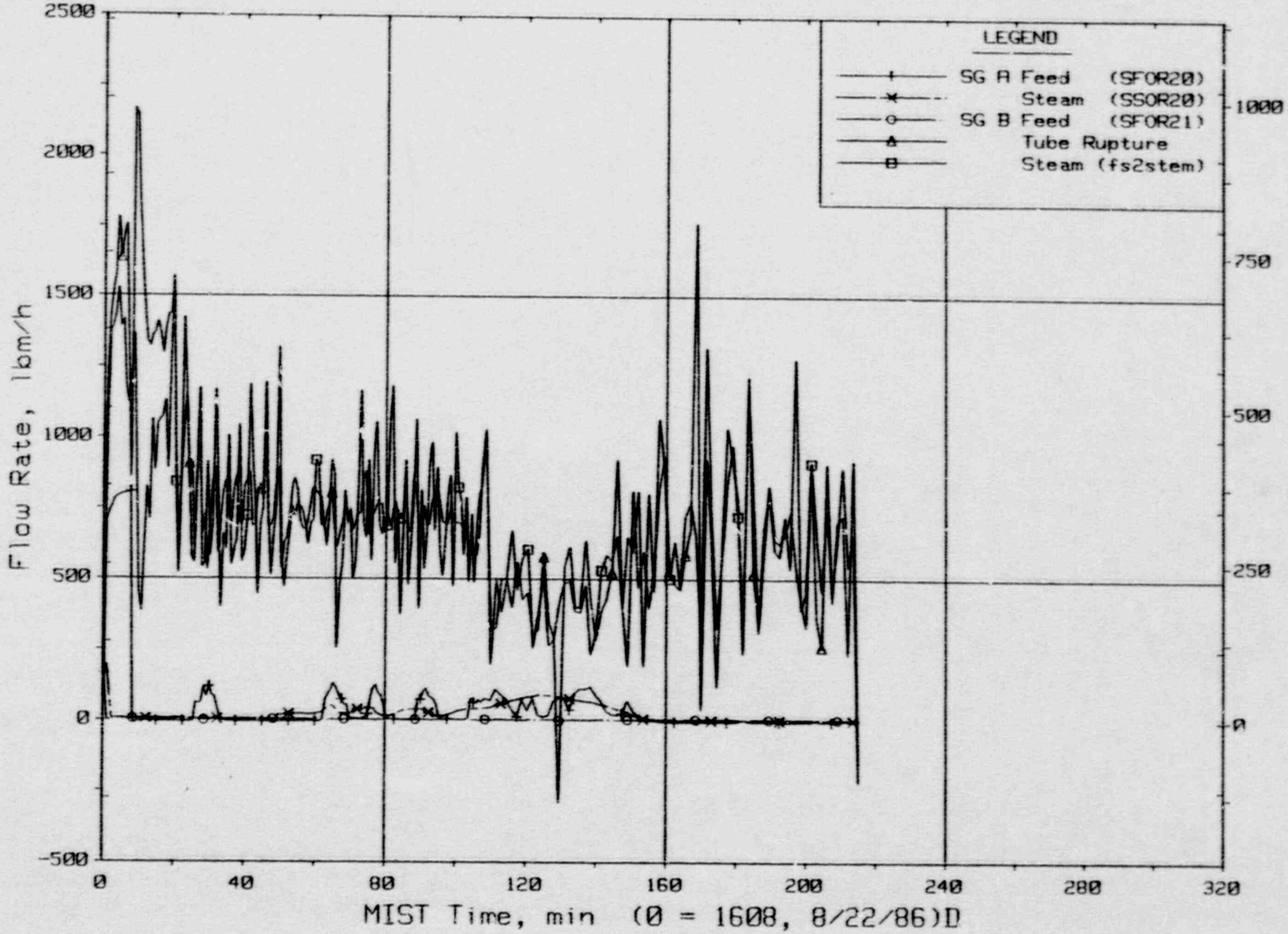
Steam Generator B Steam Exit Enthalpy.

Thu Jun 2 08:32:21 1988

S2EN1

rINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



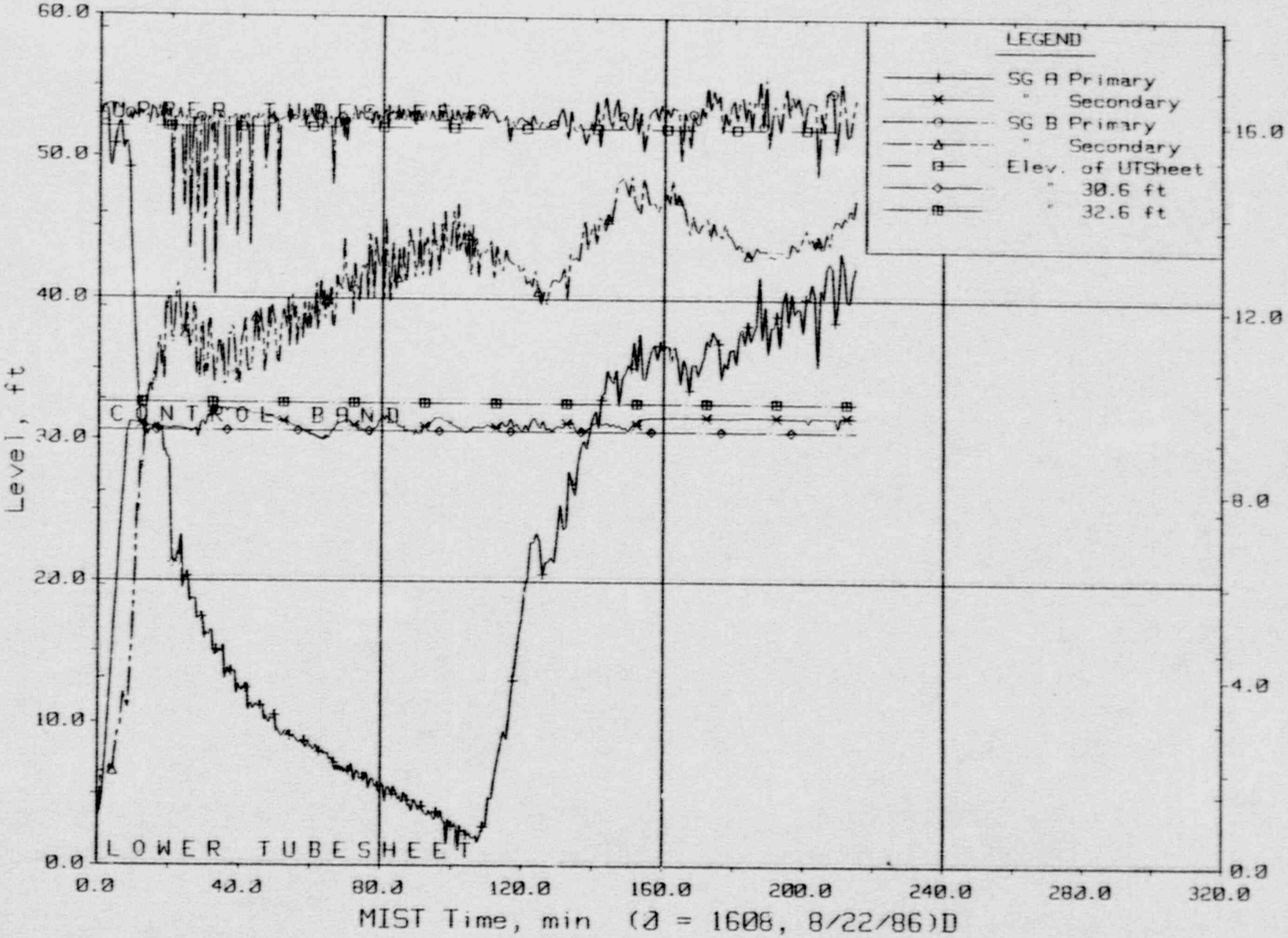
Steam Generator Secondary Flow Rates.

Fri Nov 4 14:08:45 1988

SGFL!

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



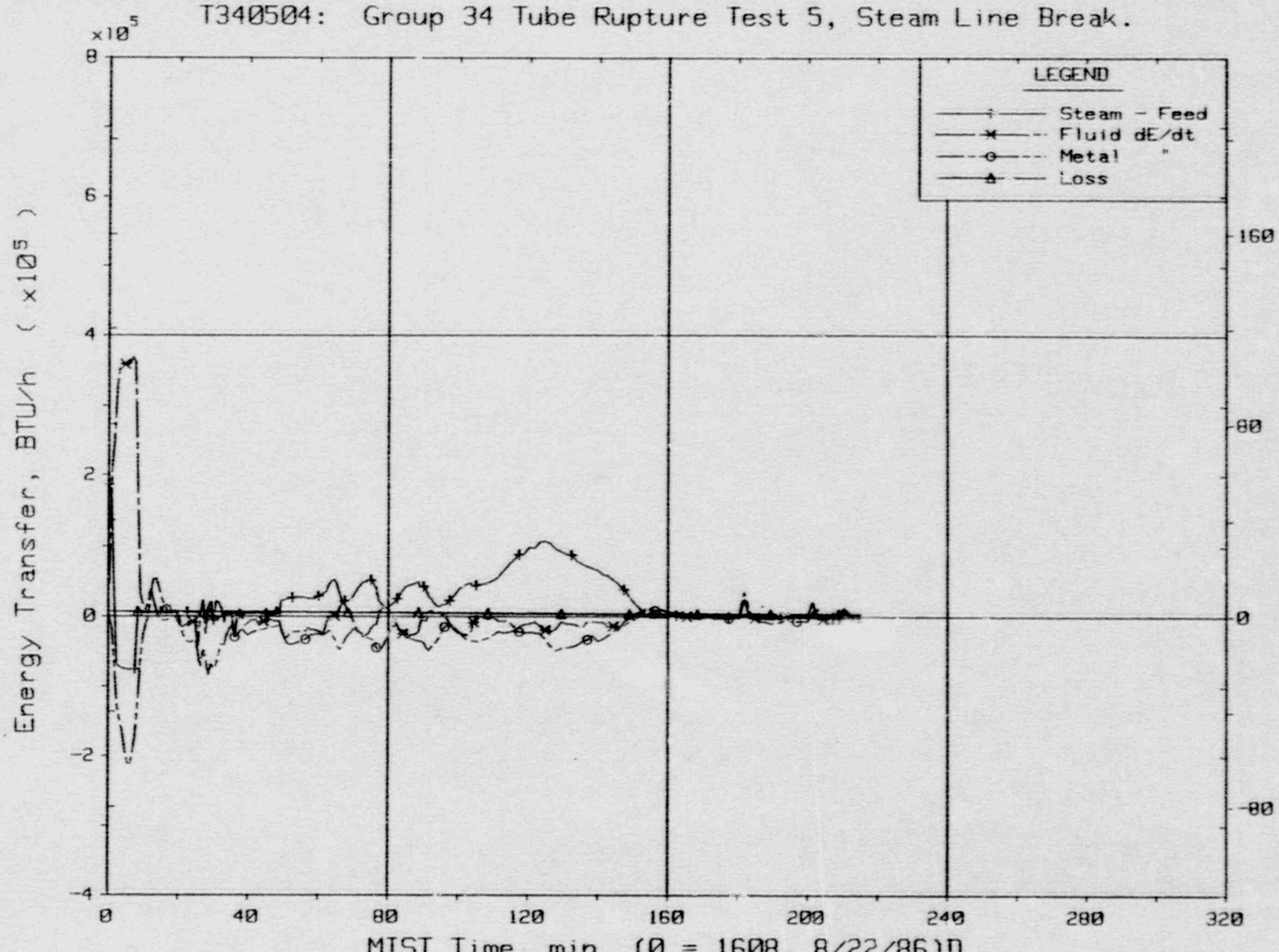
Steam Generator Collapsed Liquid Levels.

Wed Jun 1 16:59:11 1988

SGLV2

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



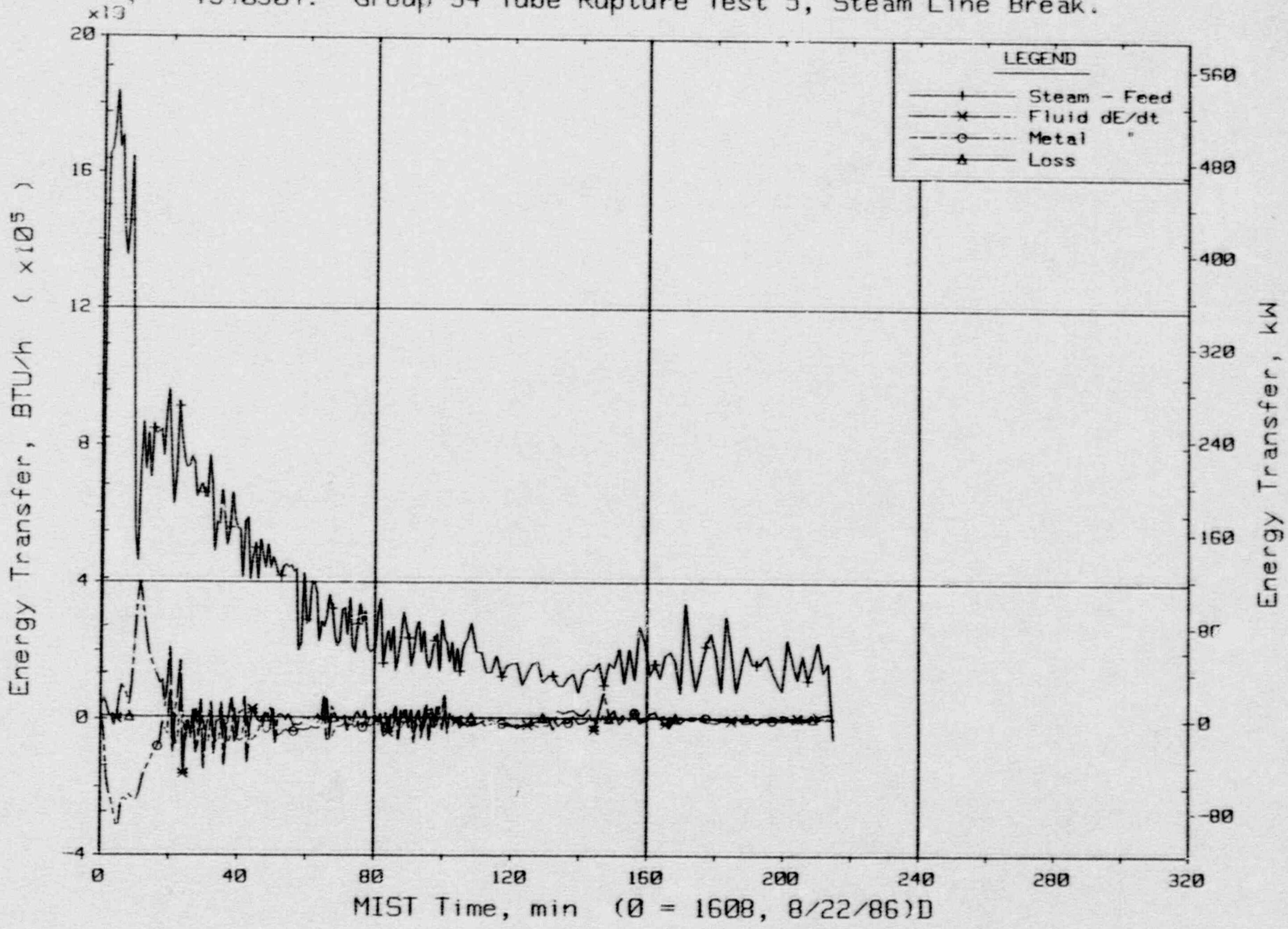
Steam Generator A' Energy Transfer.

Fri Nov 4 14:12:33 1988

SGOE3

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



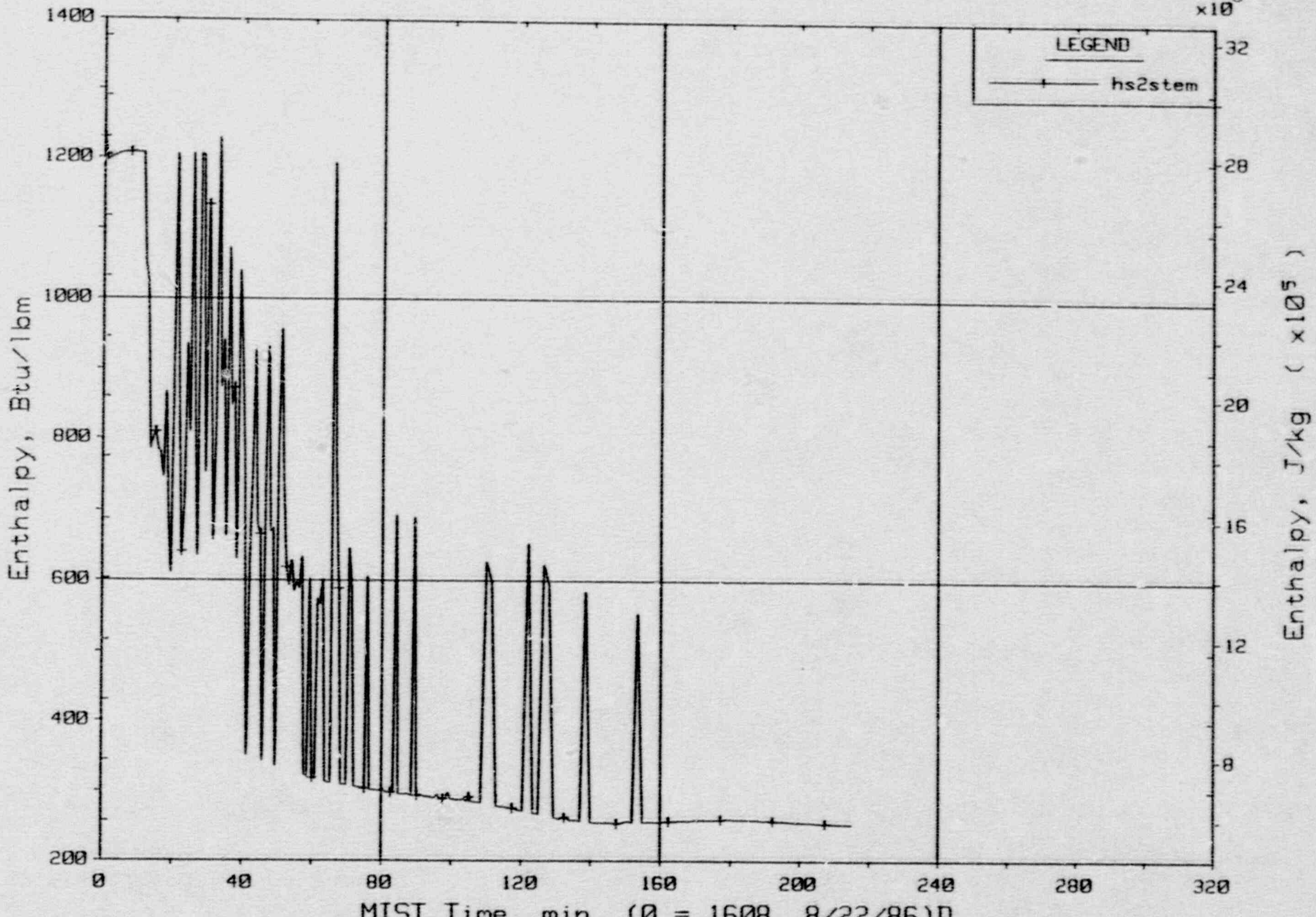
Steam Generator B Energy Transfer.

Fri Nov 4 14:18:31 1988

SGOE6

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



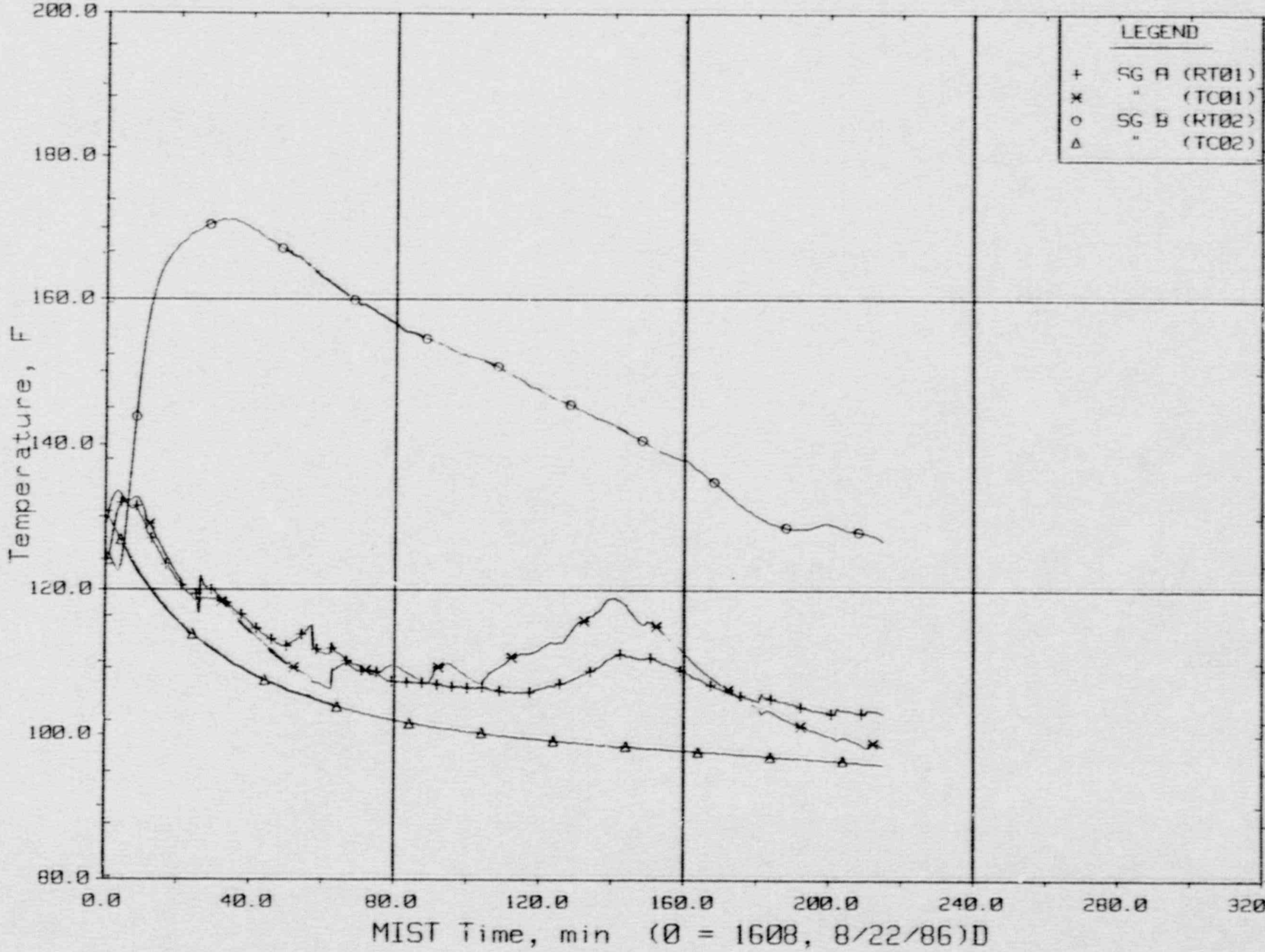
Steam Generator B Steam Exit Enthalpy.

Fri Nov 4 14:32:31 1988

S2EN1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



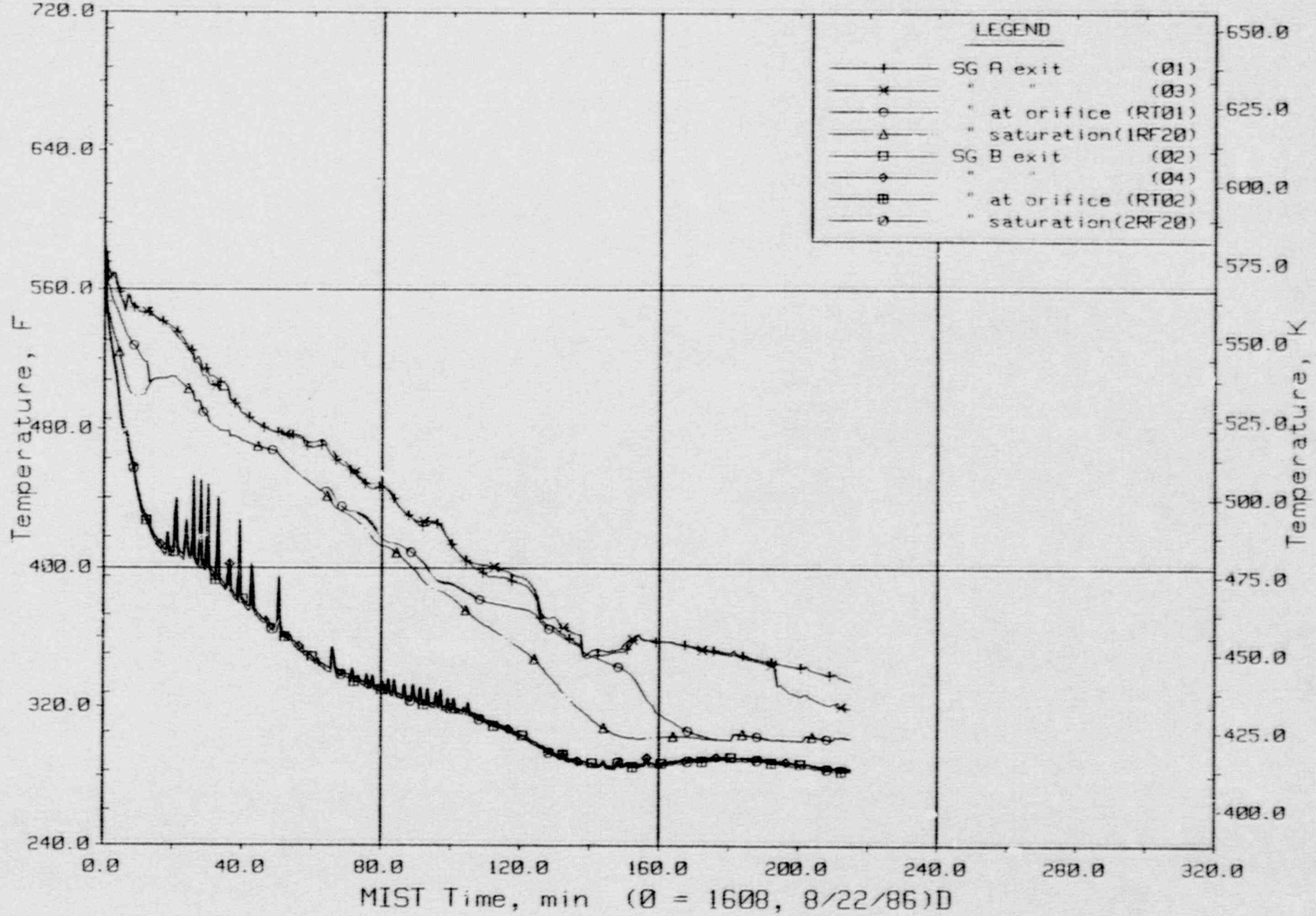
Feedwater Tempëratures (SFs).

Wed Jun 1 17:52:57 1988

SGRT1

FINAL DATA

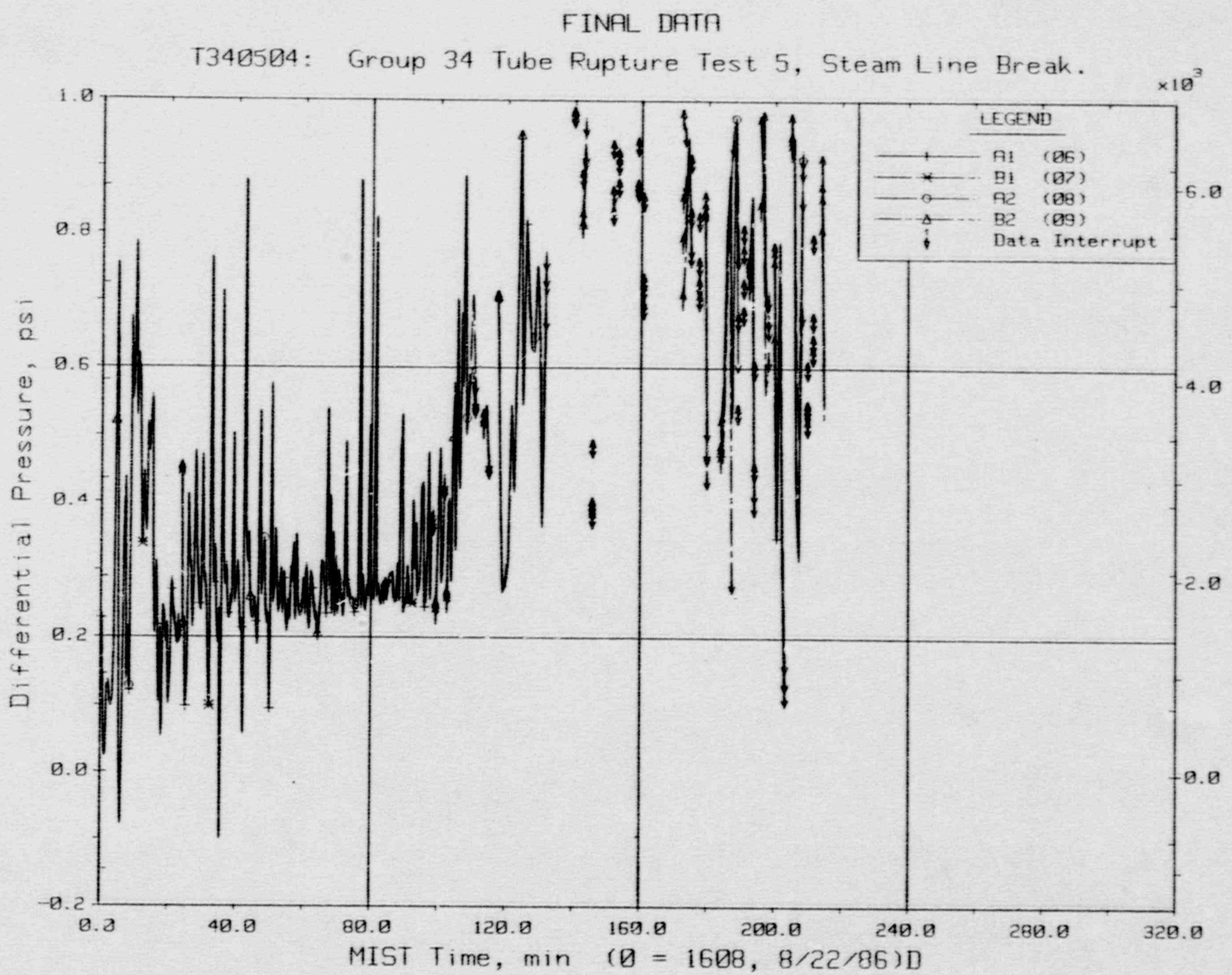
T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Steam Generator Steam Outlet Temperatures (SSTCs).

Wed Jun 1 17:56:17 1988

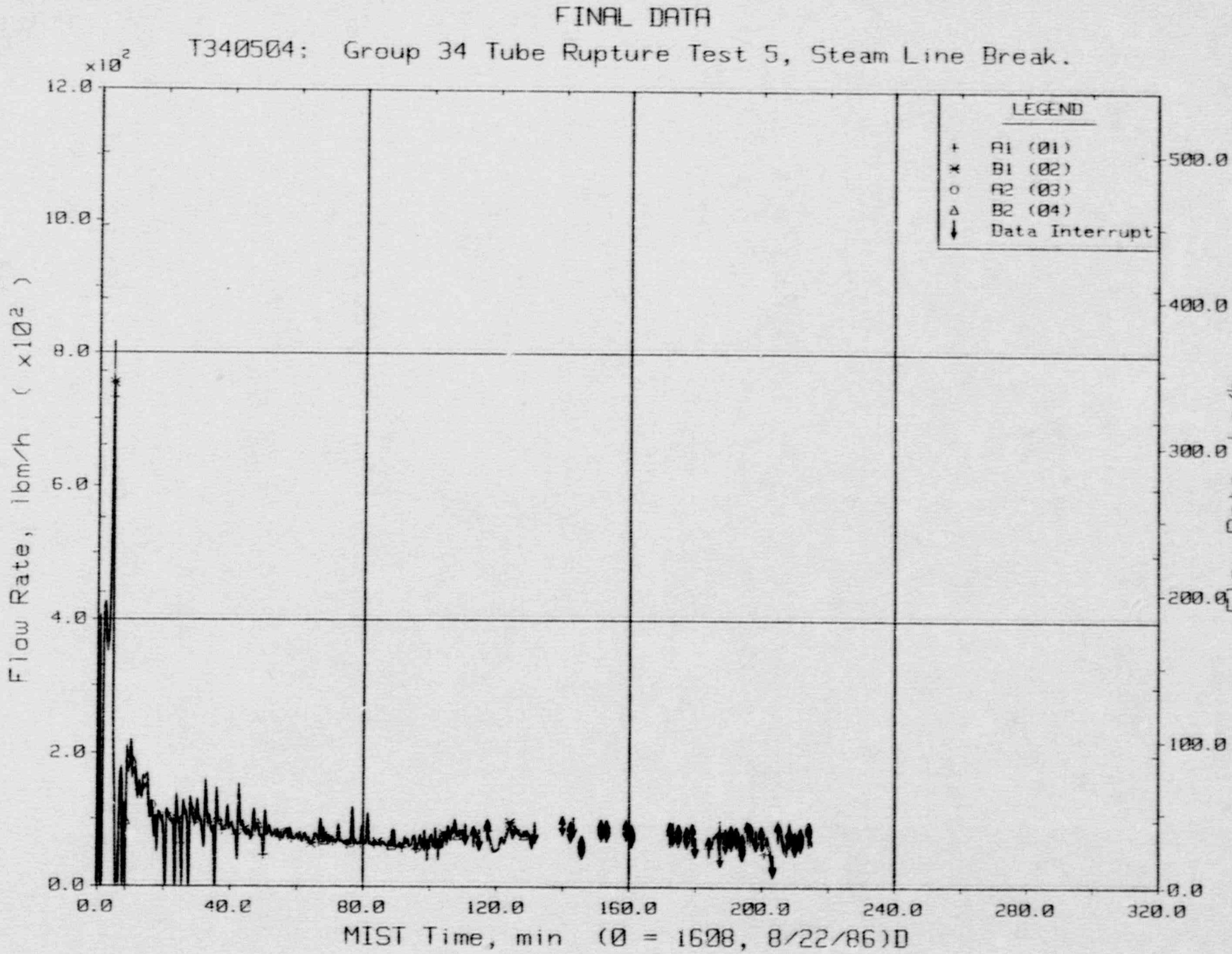
SGRT2



Reactor Vessel Vent Valve Differential Pressures (RVDPs).

Wed Jun 1 18:02:38 1988

VVDP1



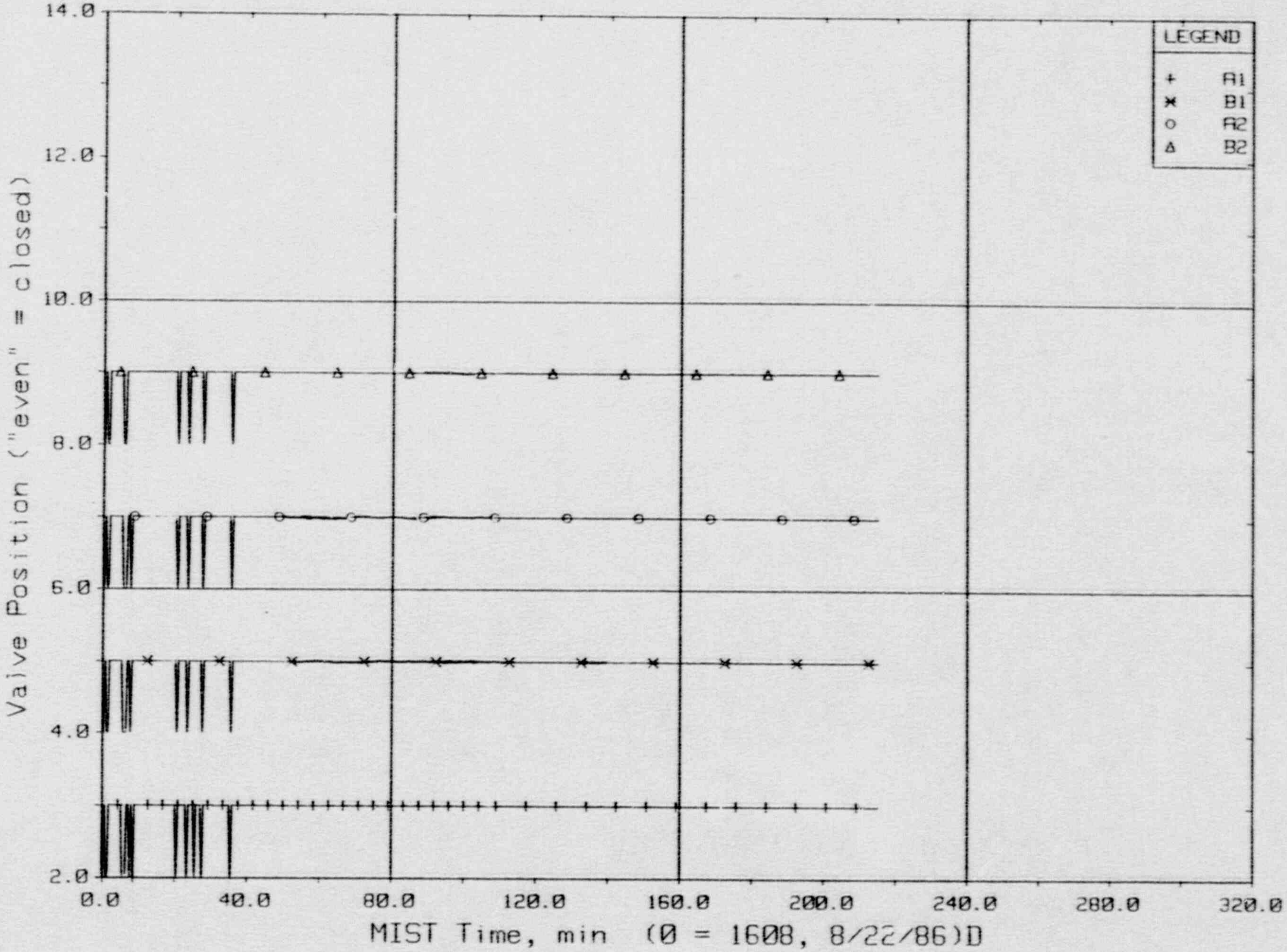
Reactcr Vessel Vent Valve Flow Rates (RVORs).

Wed Jun 1 18:06:47 1988

VVFL1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



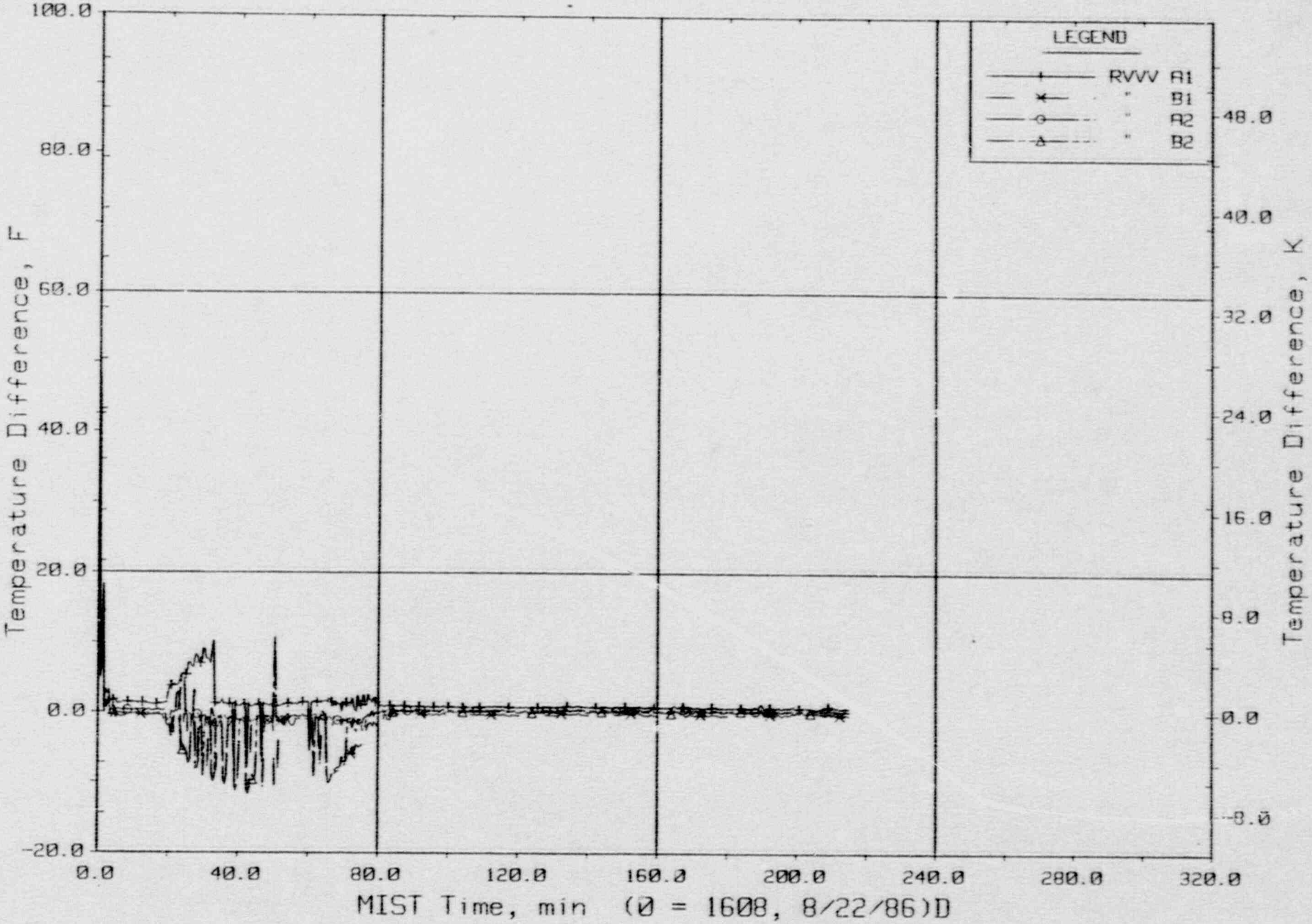
Reactor Vessel Vent Valve Positions.

Wed Jun 1 10:10:38 1988

VVLS1

FINAL DATA

T340504: Group 34 Tube Rupture Test 5, Steam Line Break.



Temperature Differences Across Vent Valves.

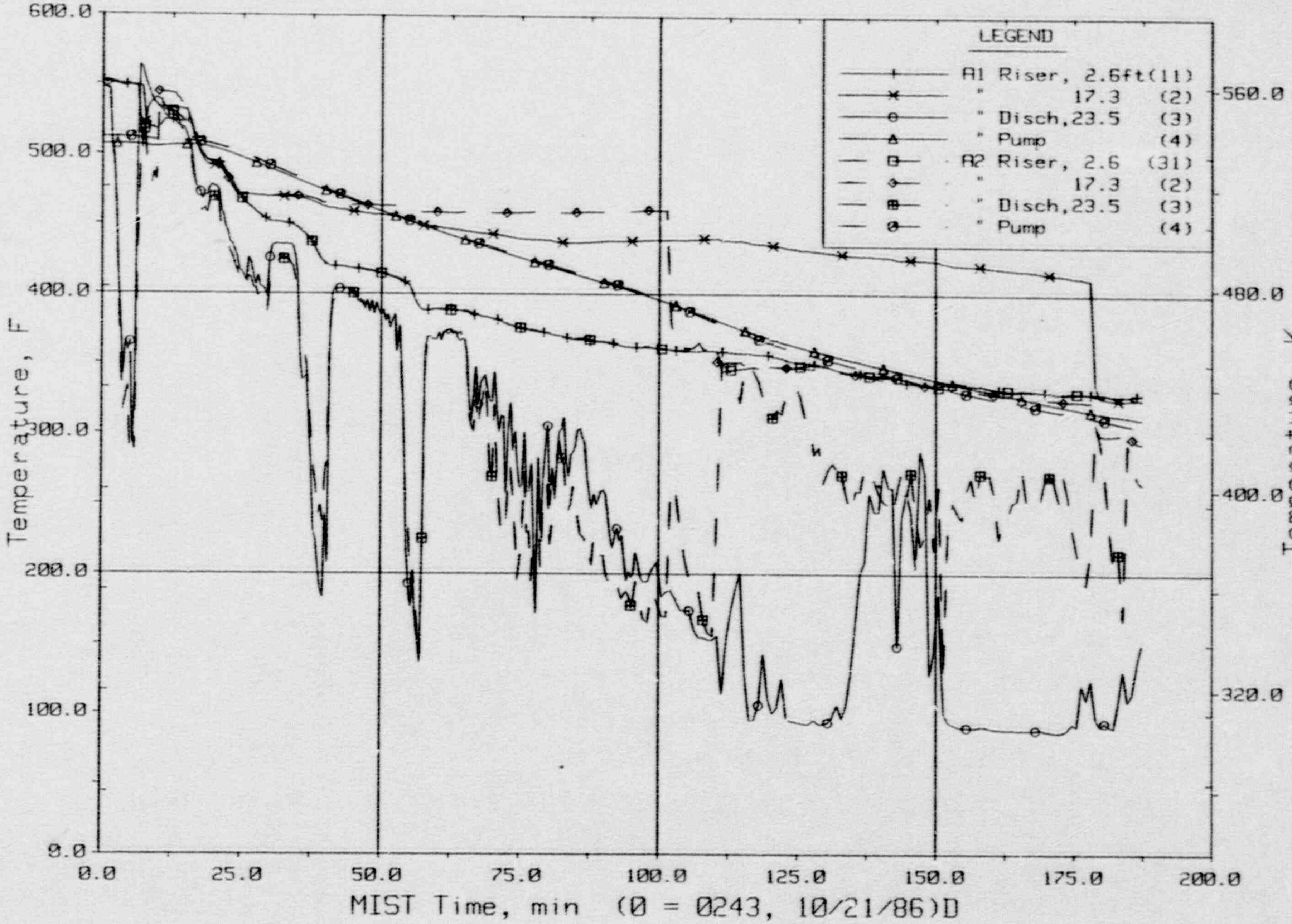
Wed Jun 1 18:13:43 1988

VVTD1

1340699

FINAL DATA

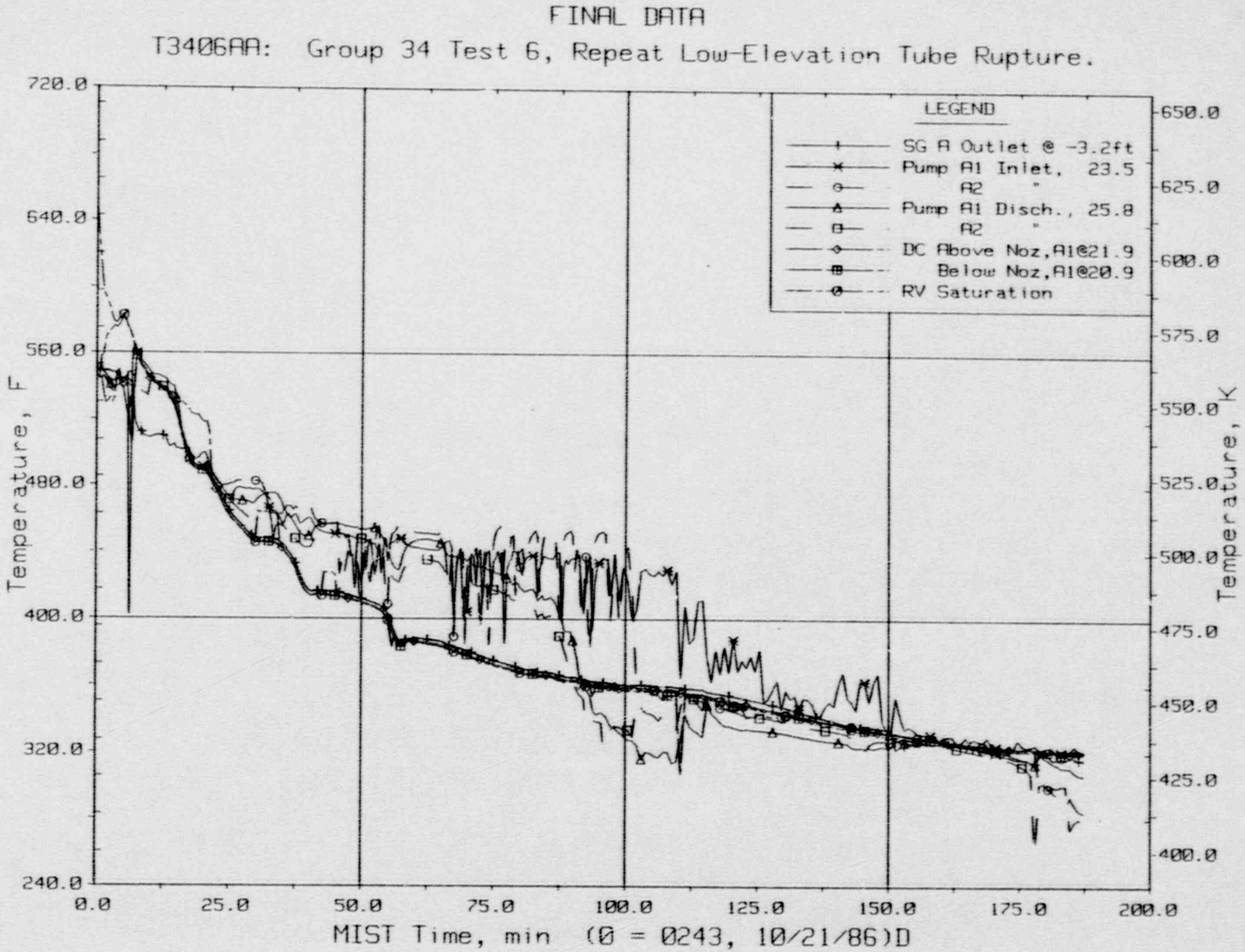
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Loop A Cold Leg Metal Temperatures (C1,3MTs).

Wed Jun 1 14:06:03 1988

C1MT1



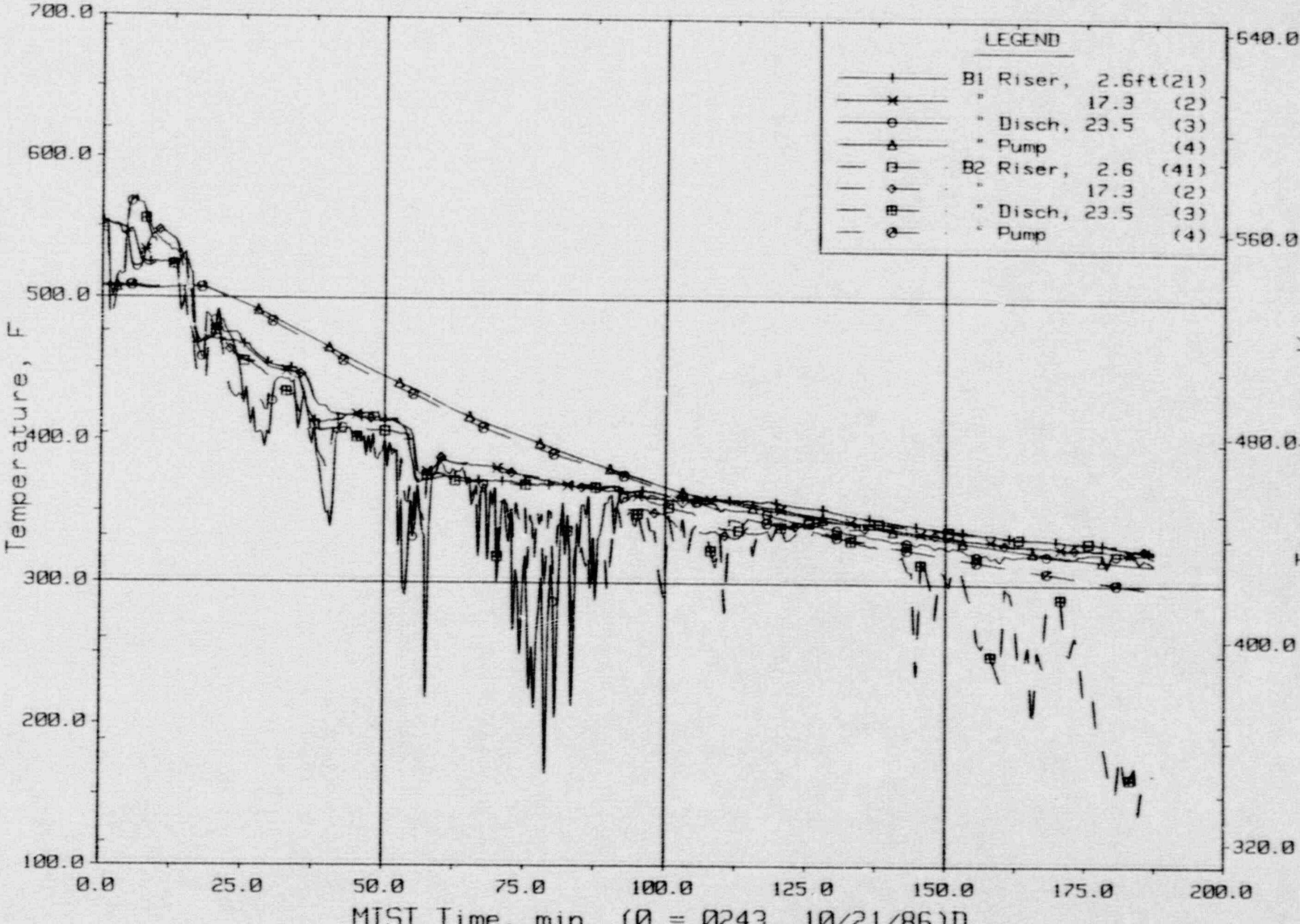
Loop A Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 14:11:16 1988

C1RT0

FINAL DATA

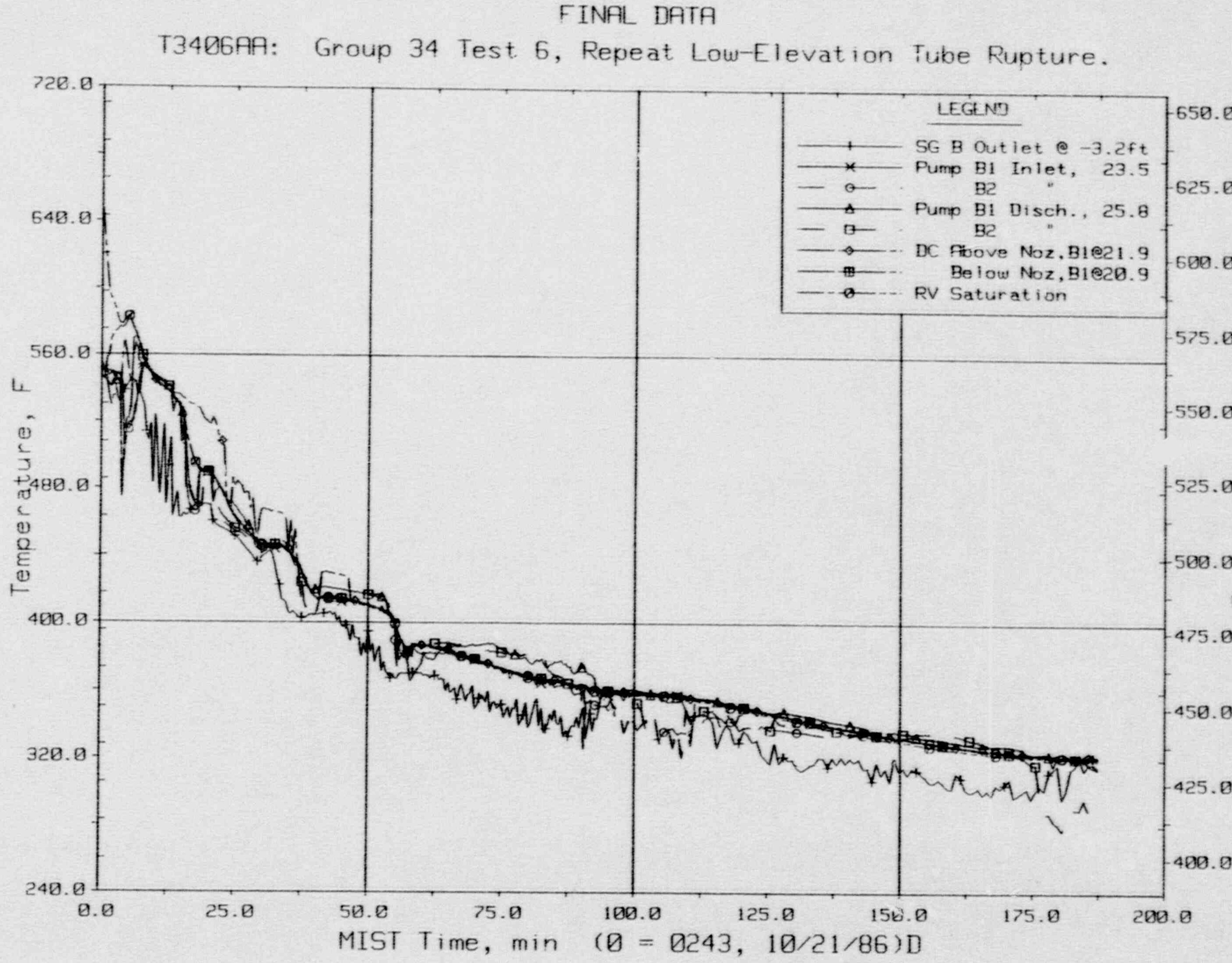
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Loop B Cold Leg Metal Temperatures (C2,4MTs).

Wed Jun 1 14:17:45 1988

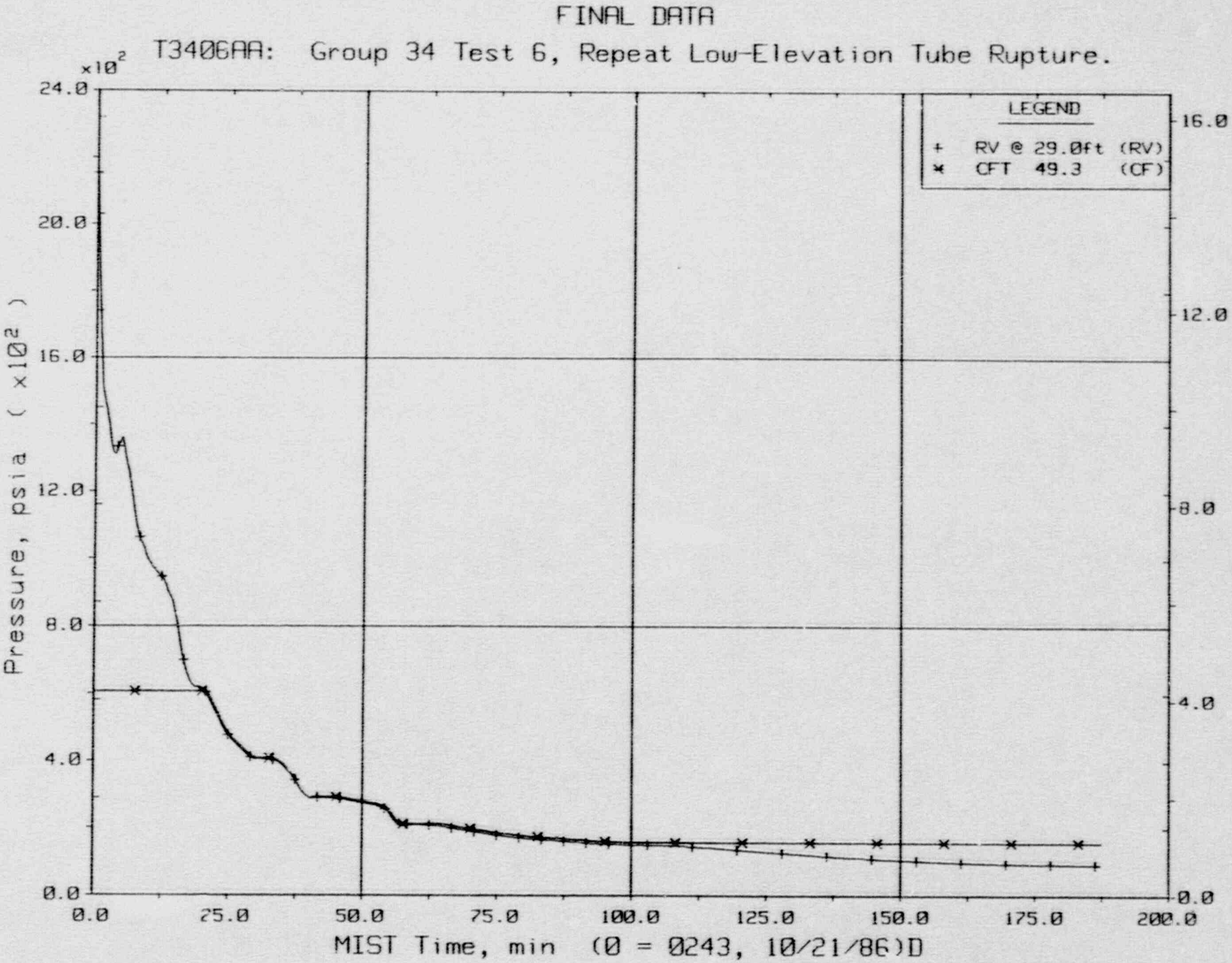
C2MT1



Loop B Cold Leg Fluid Temperatures (RTDs).

Wed Jun 1 14:23:39 1988

C2RT0



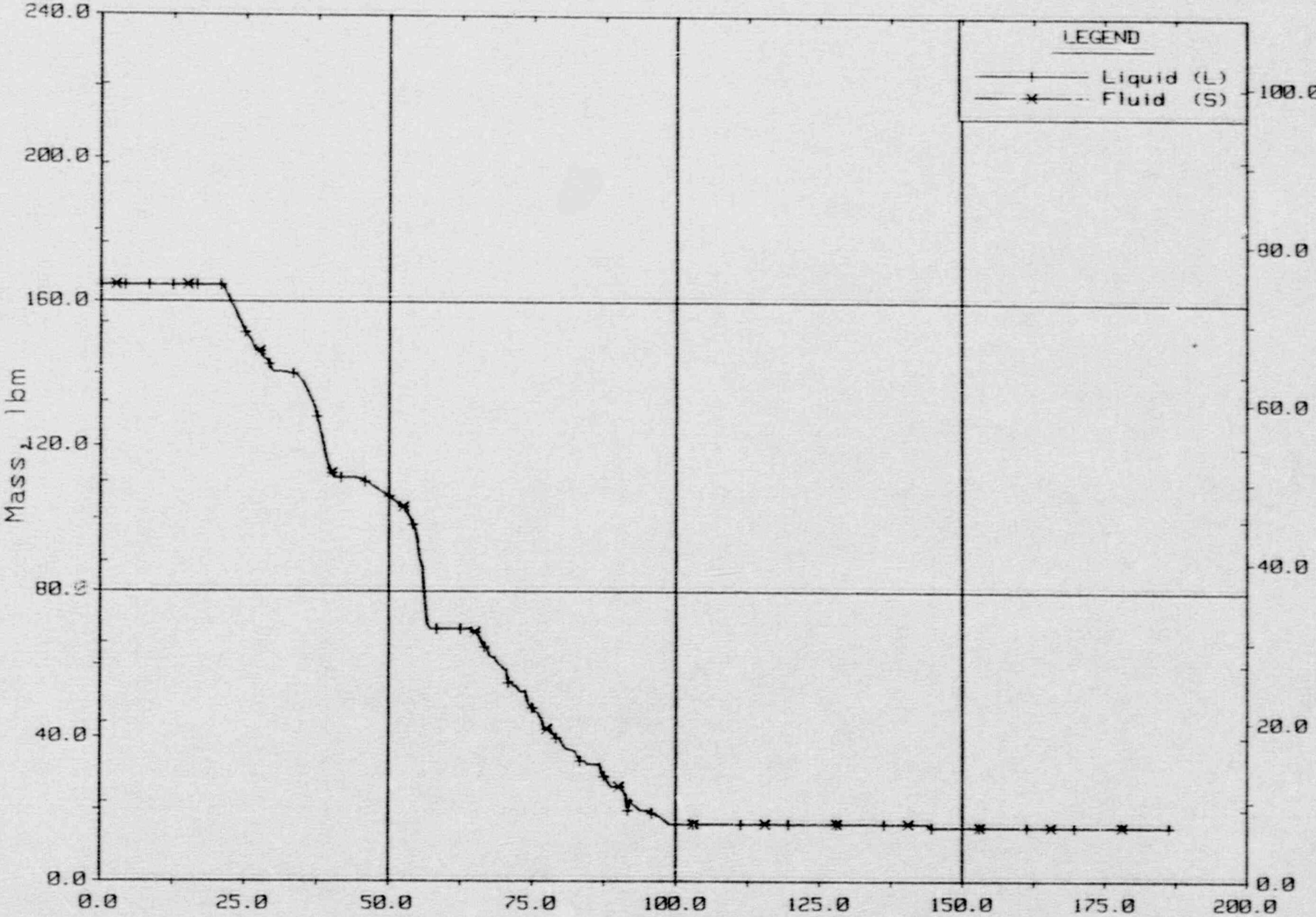
Primary System and Core Flöod Tank Pressures (GP01s).

Wed Jun 1 14:27:12 1988

CFGP1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



MIST Time, min (0 = 0243, 10/21/86)D

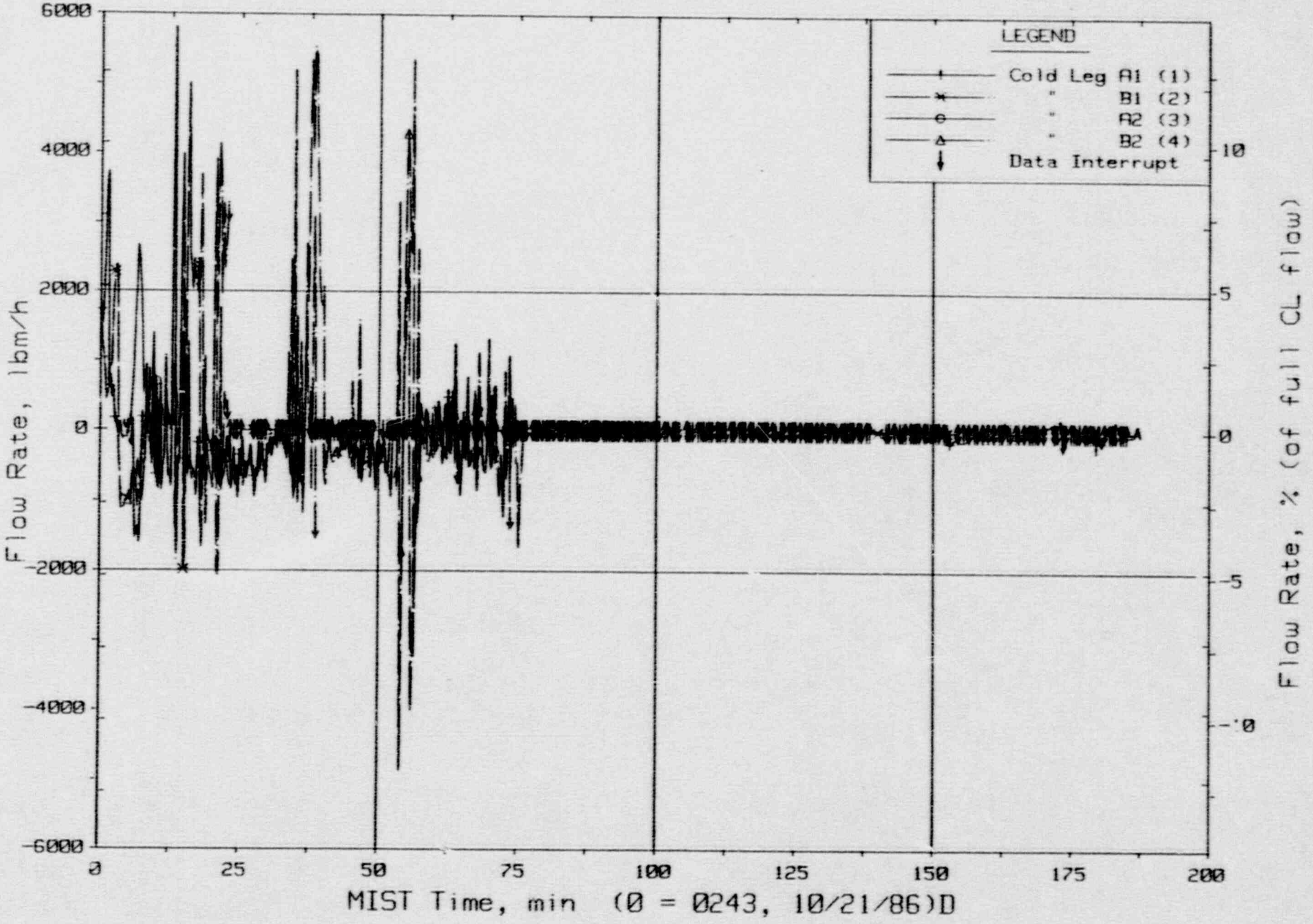
Core Flood Tank Liquid and Fluid Mass (CFMa20s).

Wed Jun 1 14:29:55 1988

CFML1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



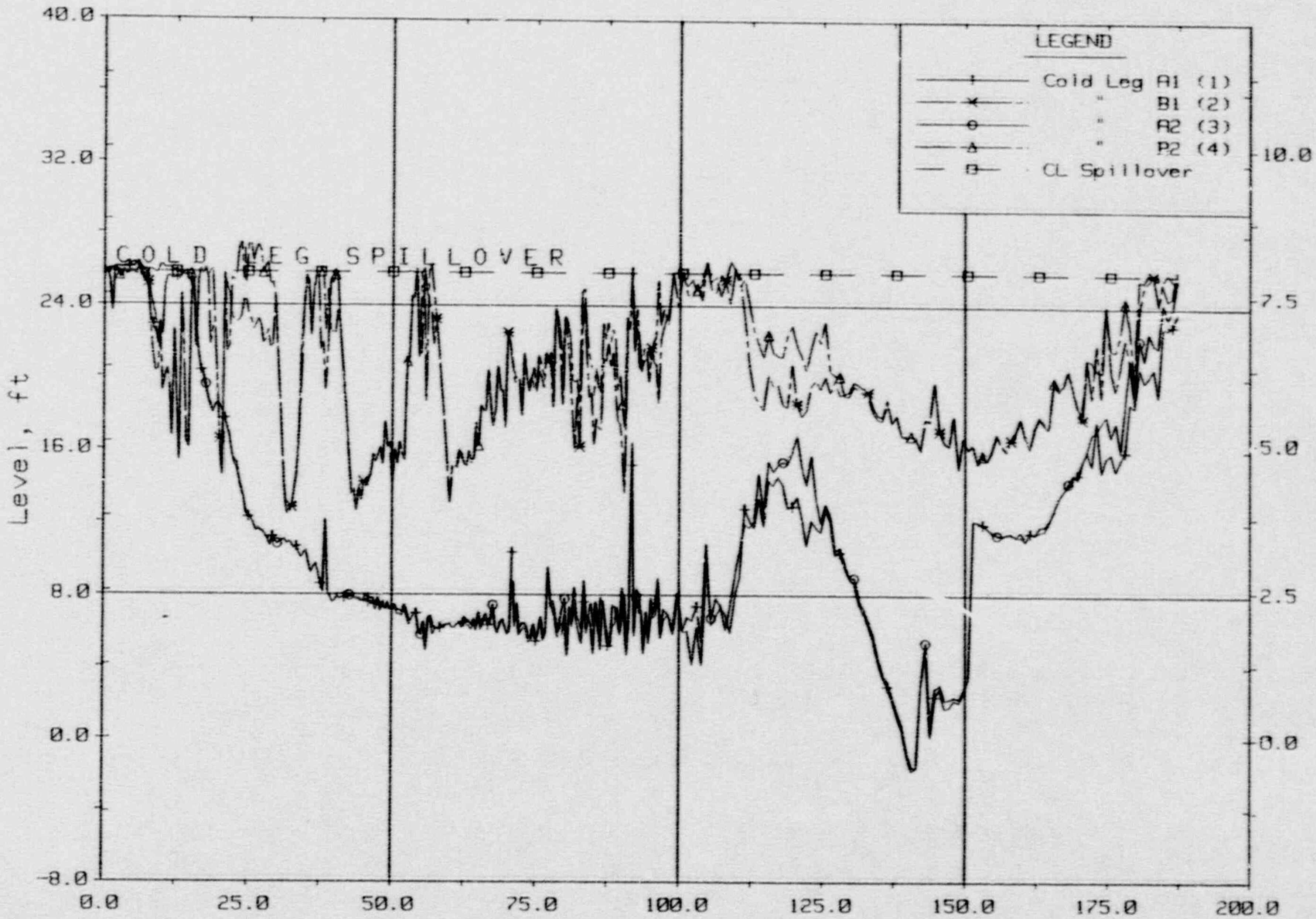
Cold Leg (Venturi) Flow Rates (CnVN20s).

Tue Nov 1 14:49:43 1988

CLFL0

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



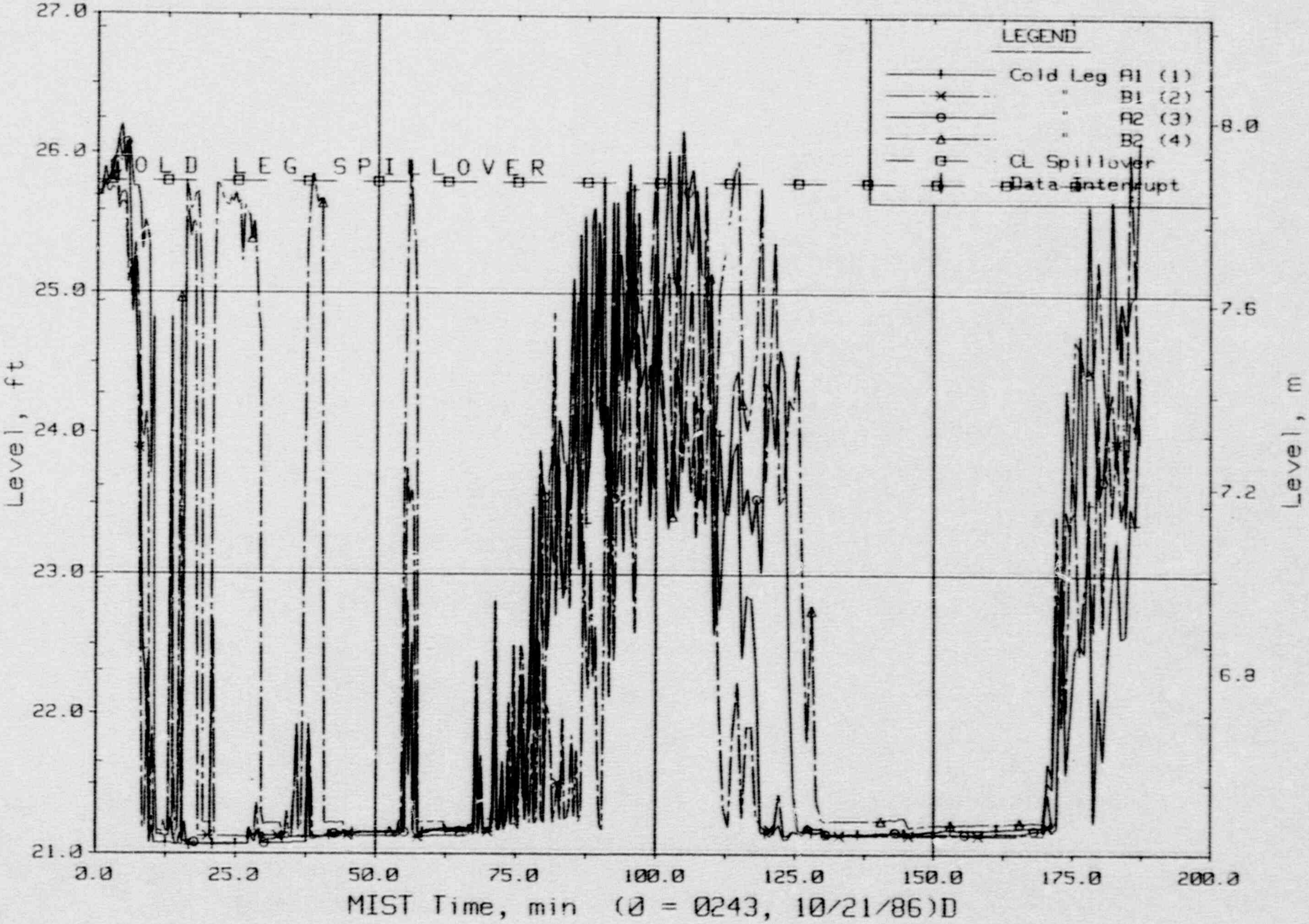
Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

Wed Jun 1 15:02:19 1988

CLLV4

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



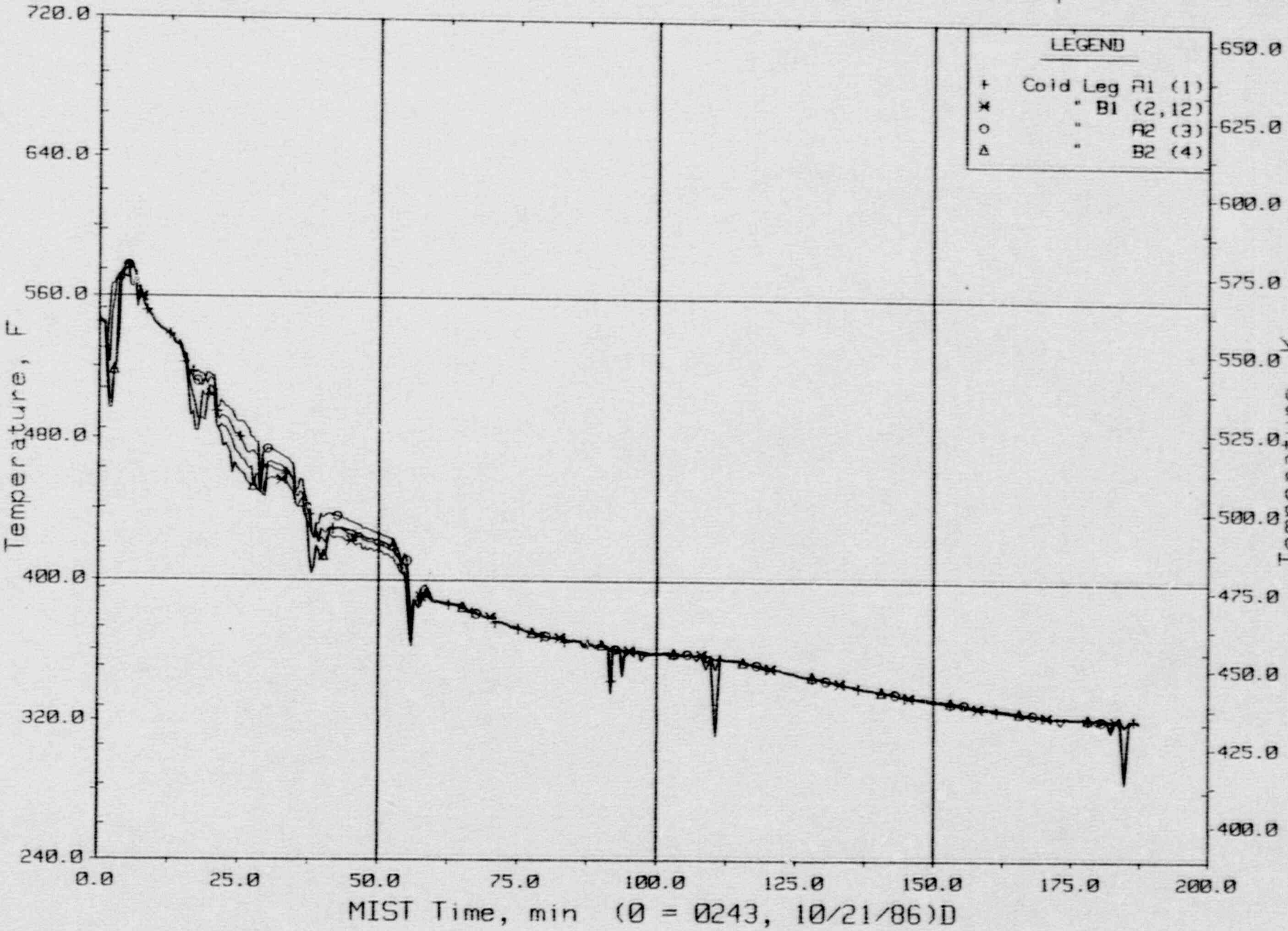
Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Wed Jun 1 15:10:12 1988

CLLV5

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



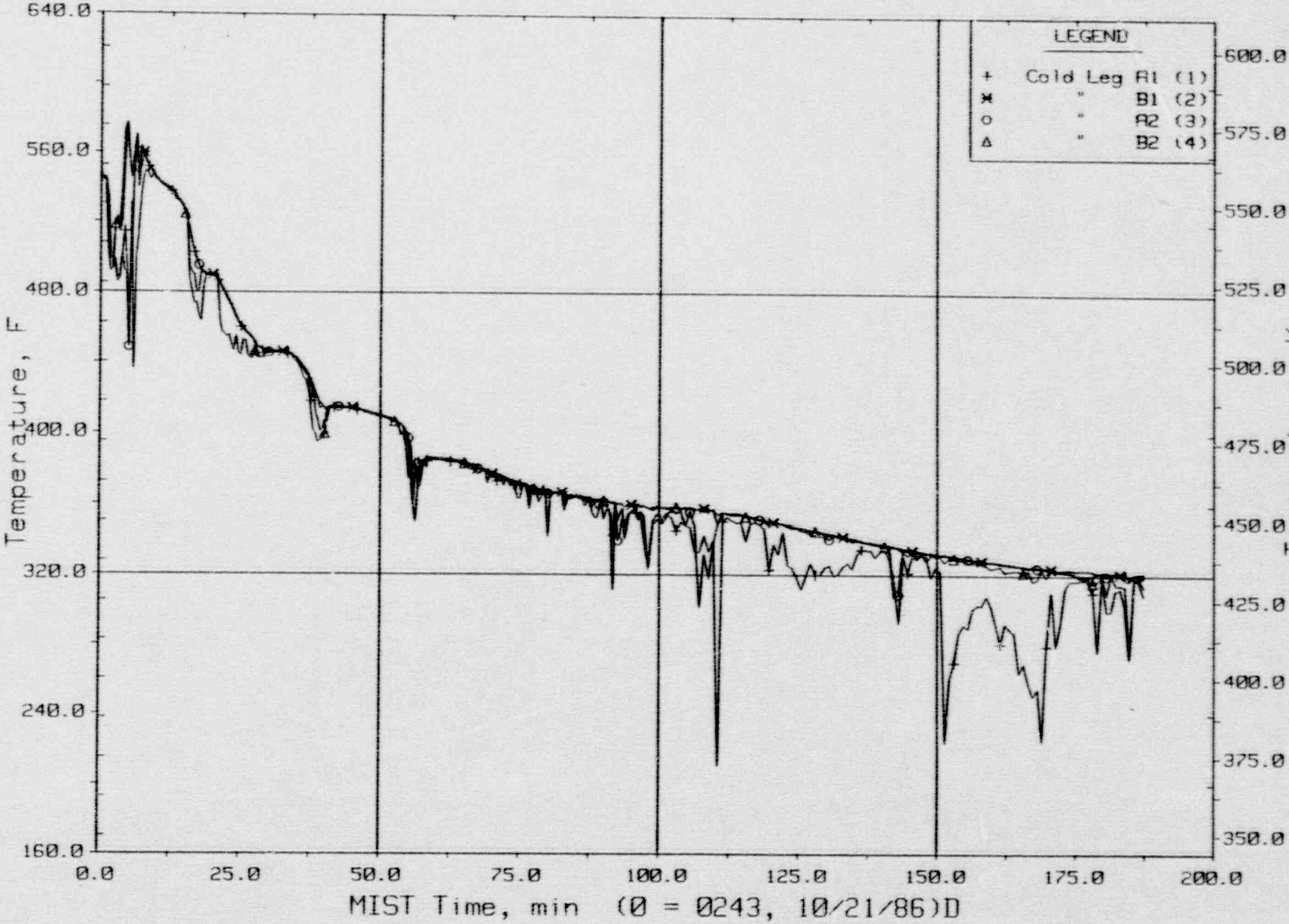
Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC1ls).

Wed Jun 1 15:27:16 1988

CLTC1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



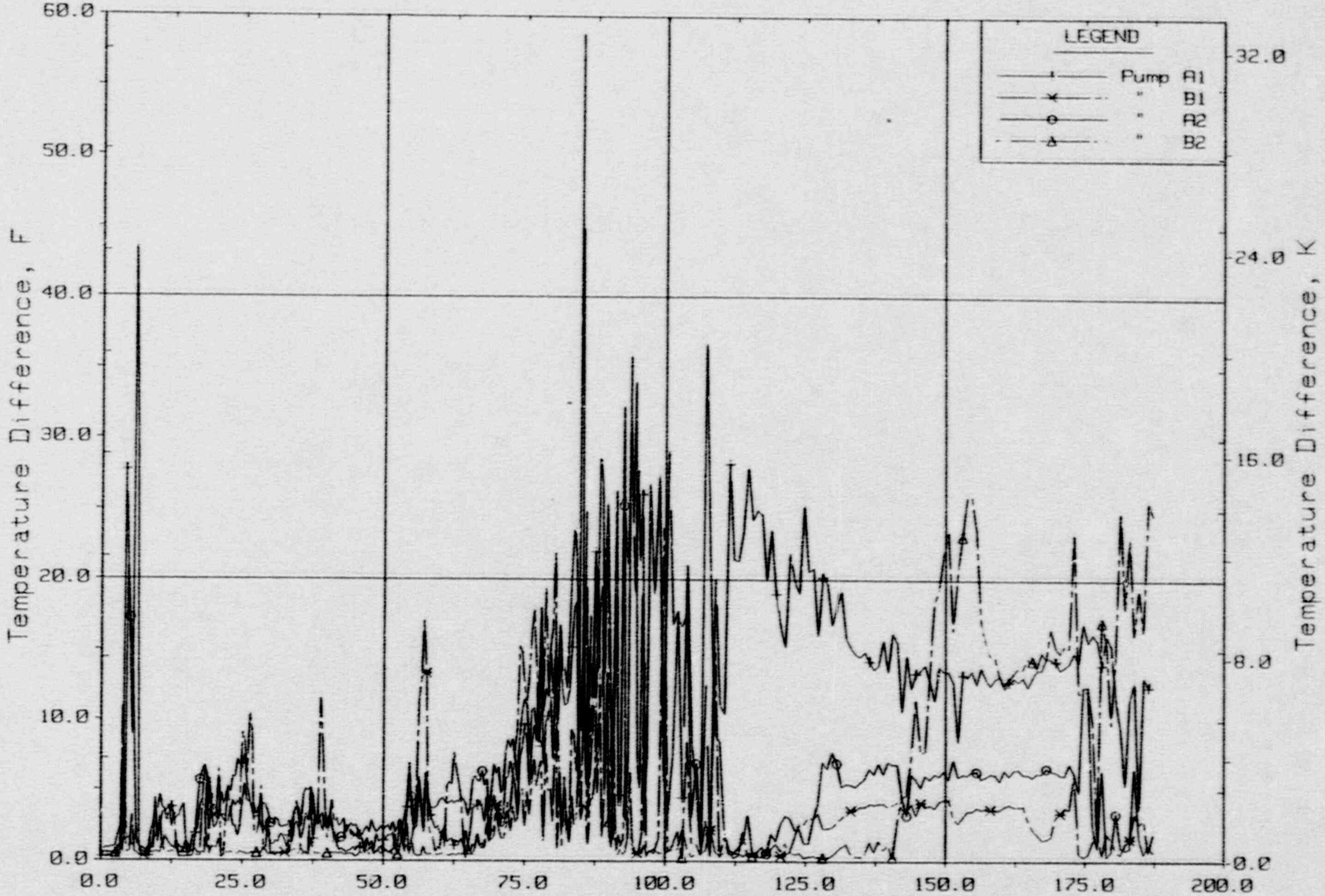
Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

Wed Jun 1 15:36:10 1988

CLTC2

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



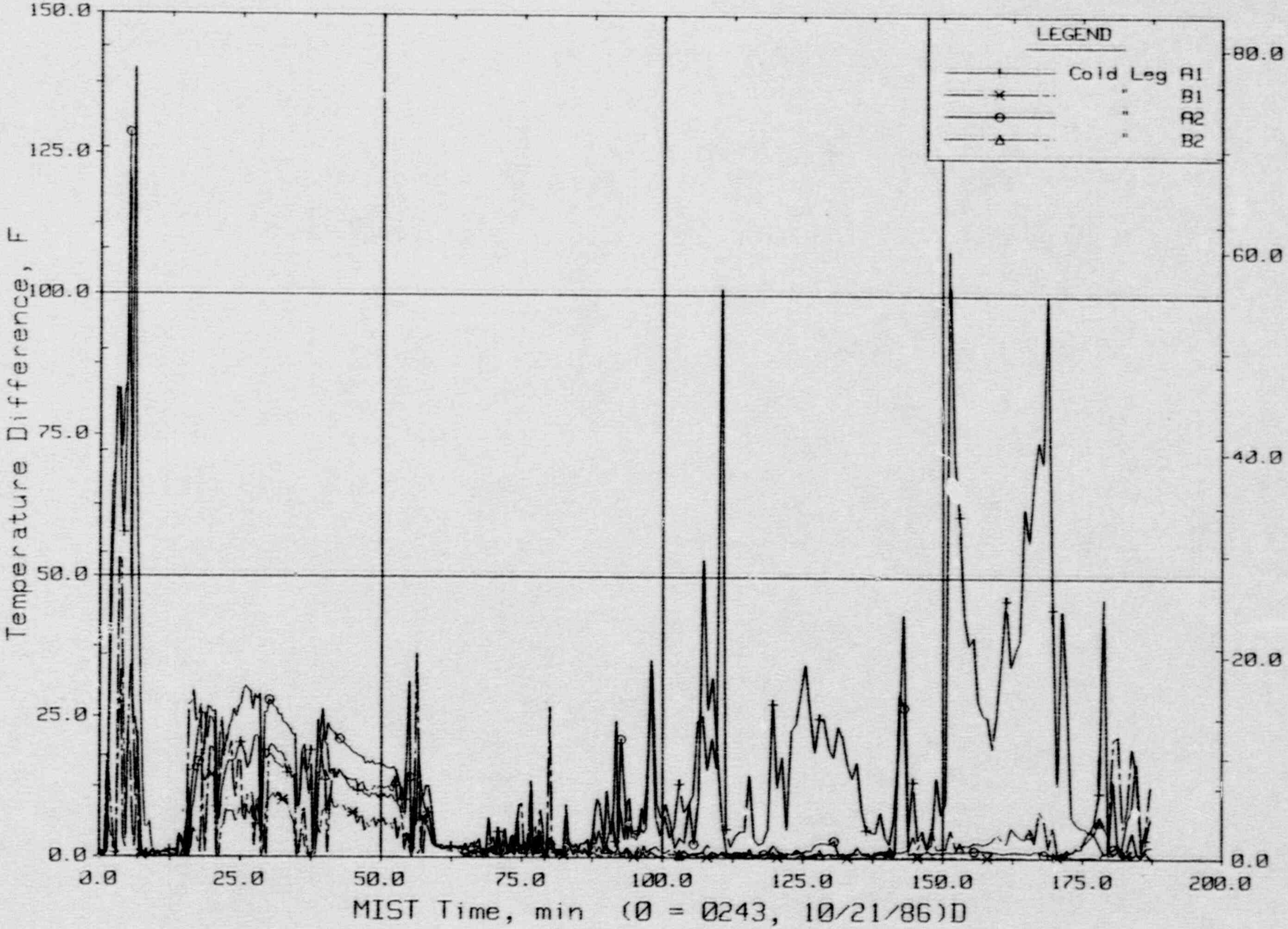
Maximum Differences Among RCP Rake FLuid Temperatures.

Wed Jun 1 15:40:33 1988

CLTD1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



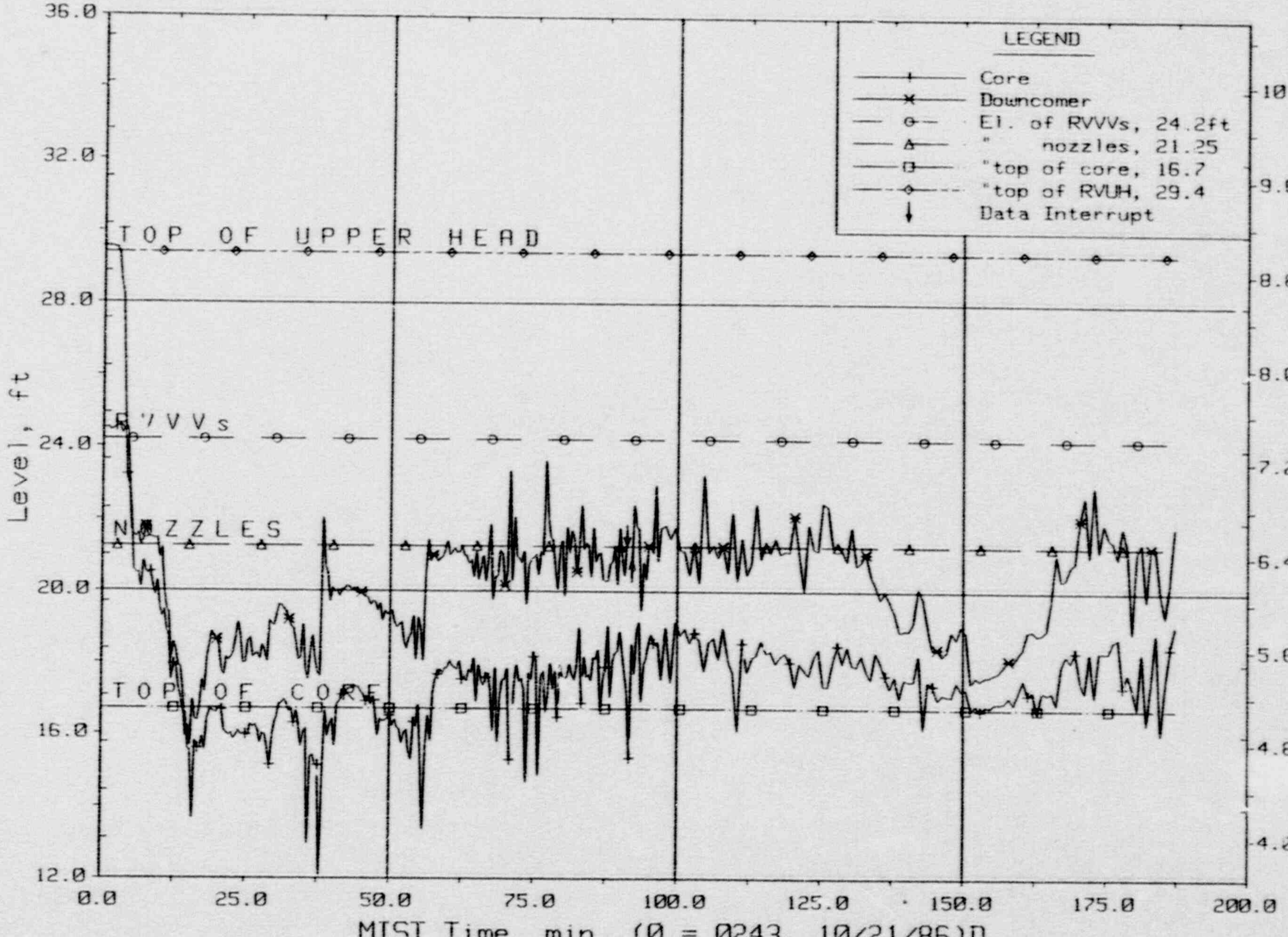
Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

Wed Jun 1 15:47:03 1988

CLTD2

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



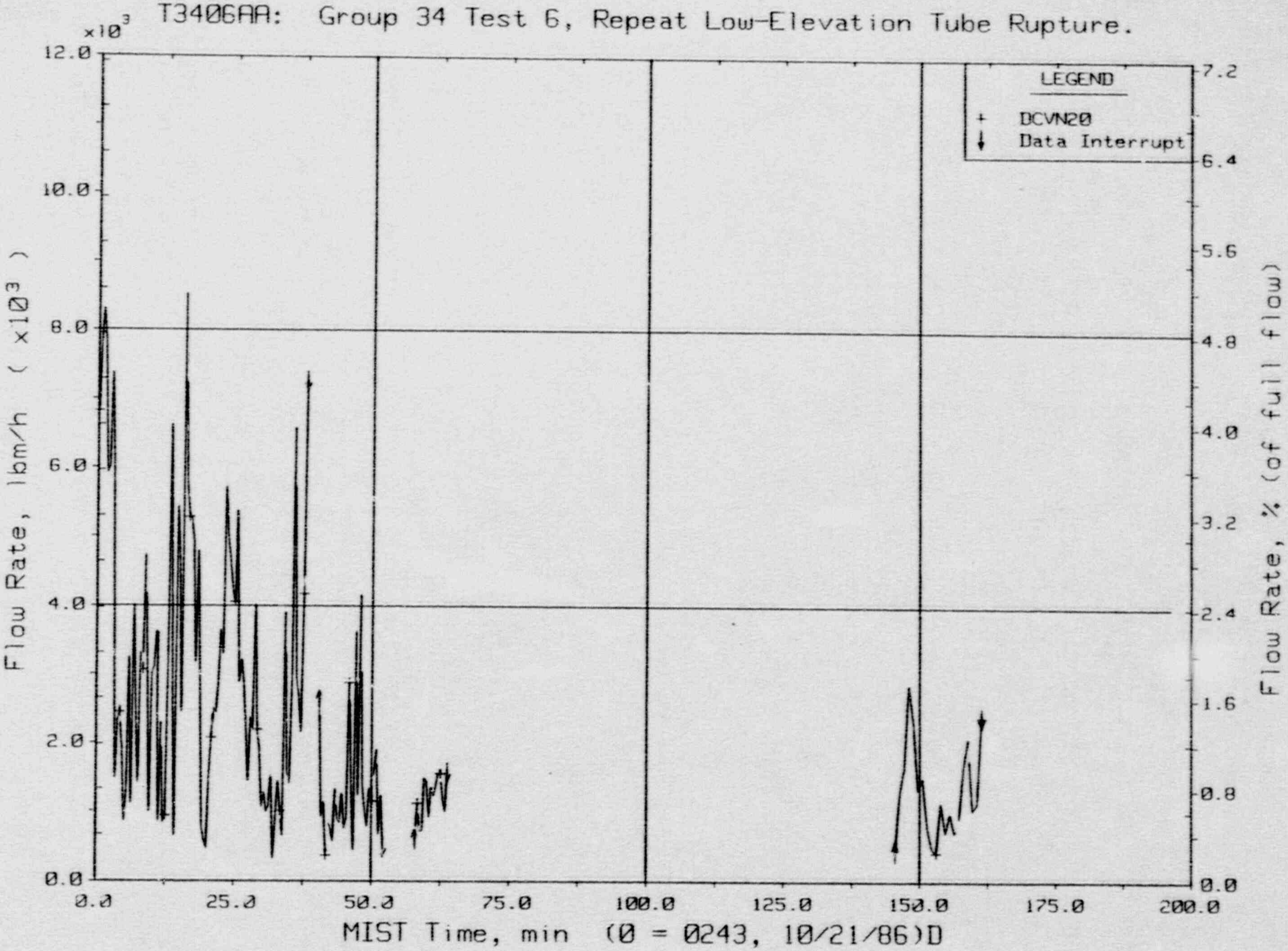
Core Region Collapsed Liquid Levels.

Wed Jun 1 15:51:59 1988

COLV1

FINAL DATA

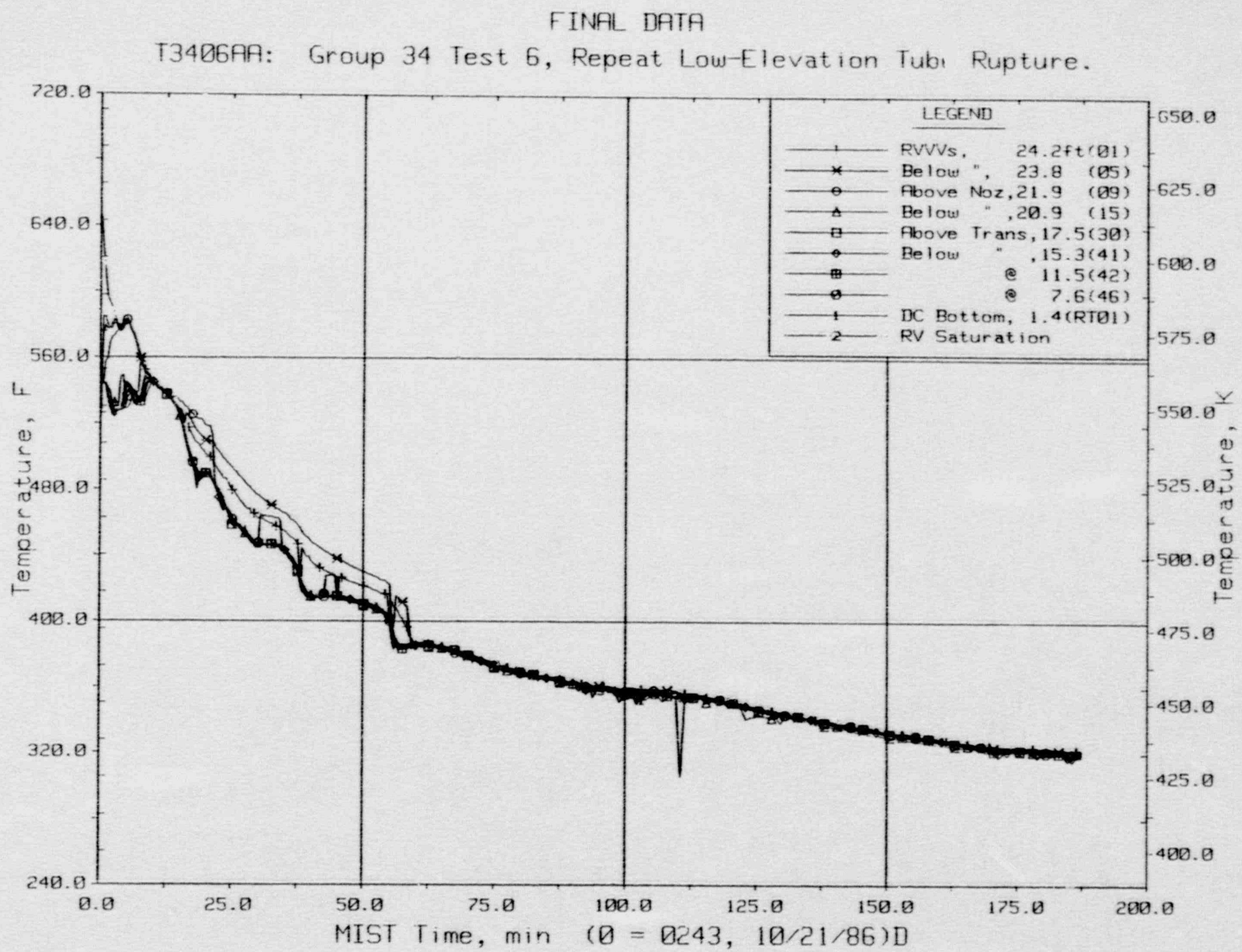
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Downcomer (Venturi) Flow Rate.

Wed Jun 1 15:57:31 1988

DCFL1



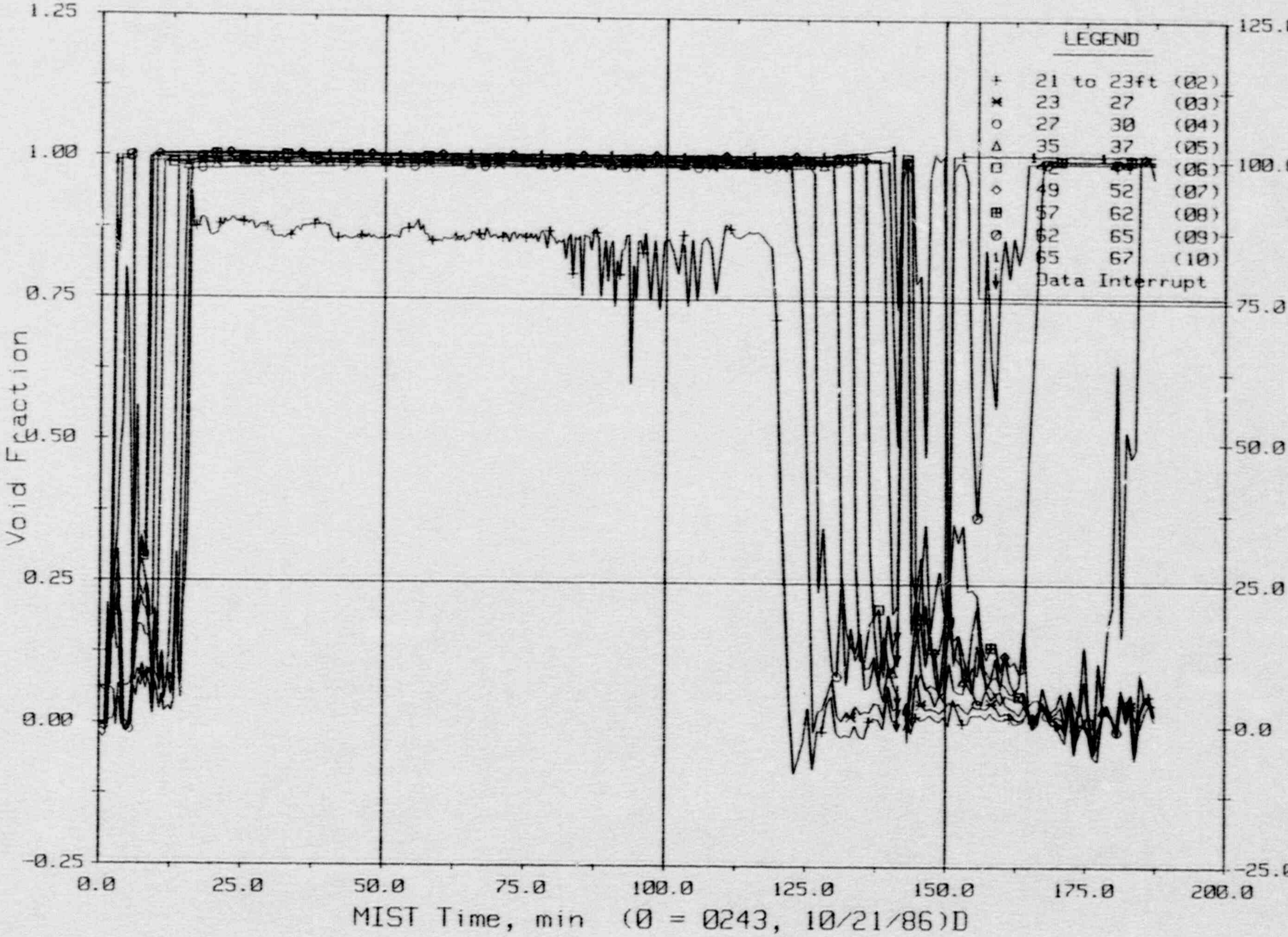
Downcomer Quadrant A1 Flüid Temperatures (DCTCs).

Wed Jun 1 16:02:24 1988

DCTC6

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



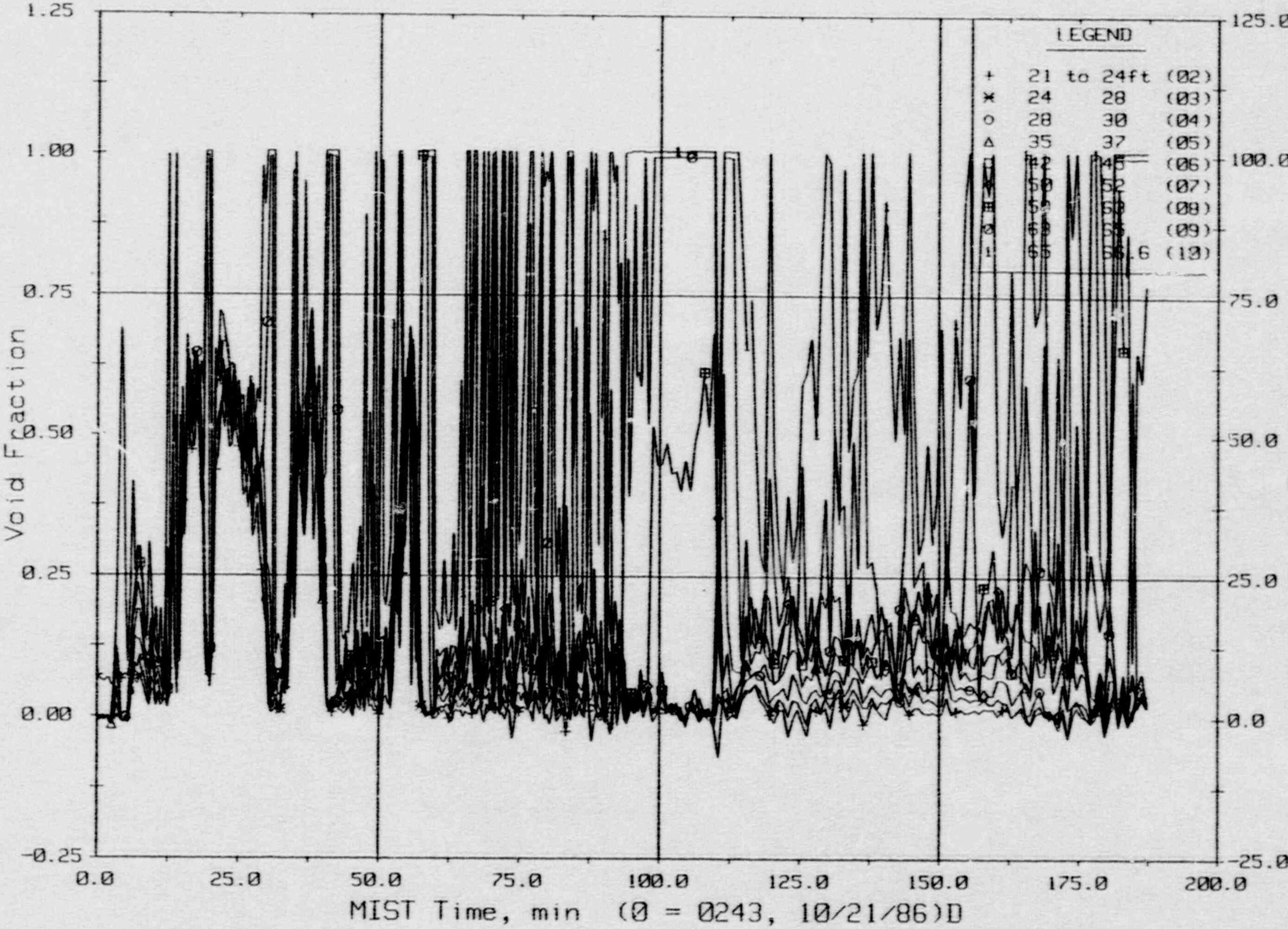
Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

Wed Jun 1 16:07:32 1988

H1VF1

FINAL DATA

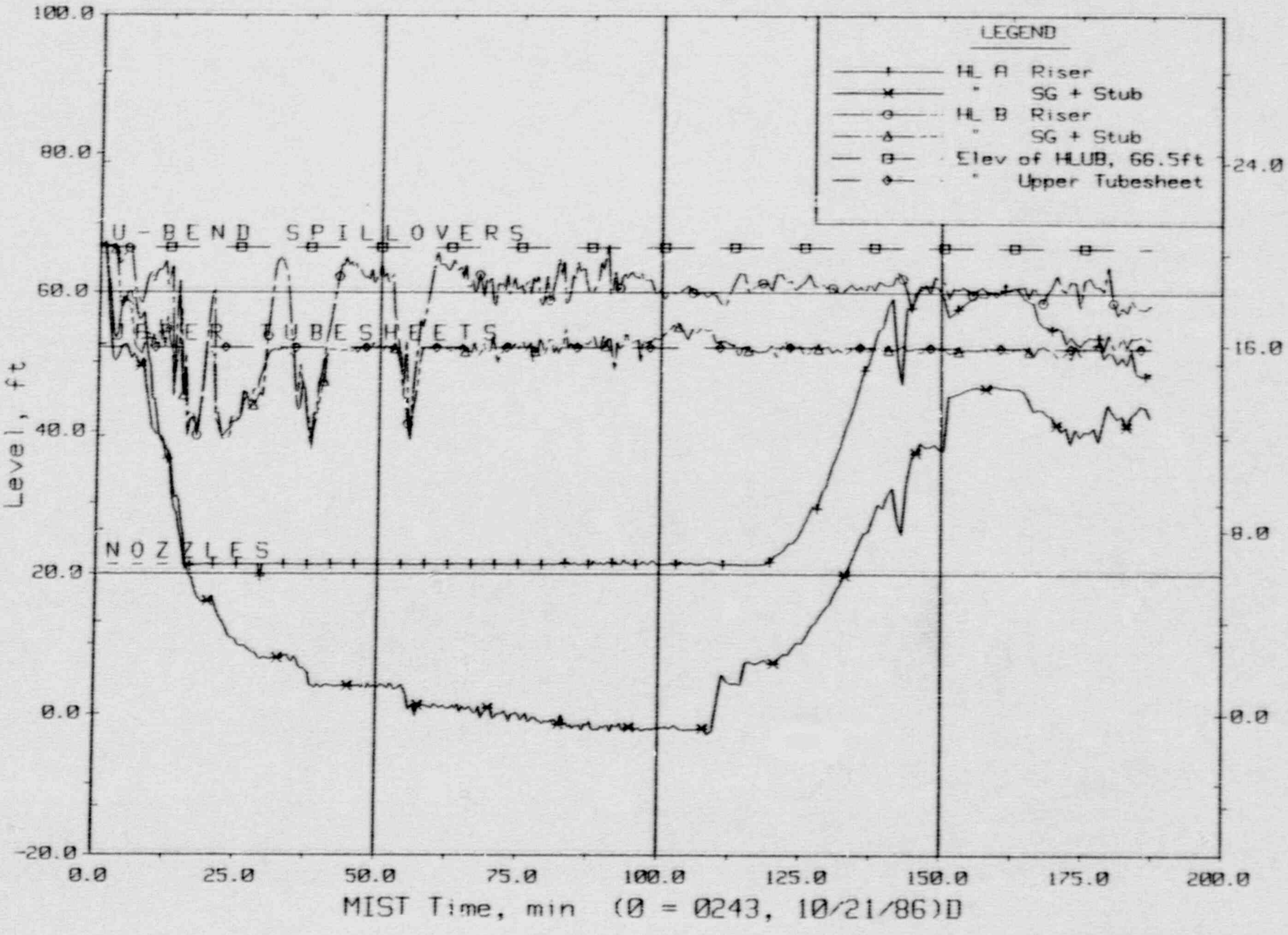
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

FINAL DATA

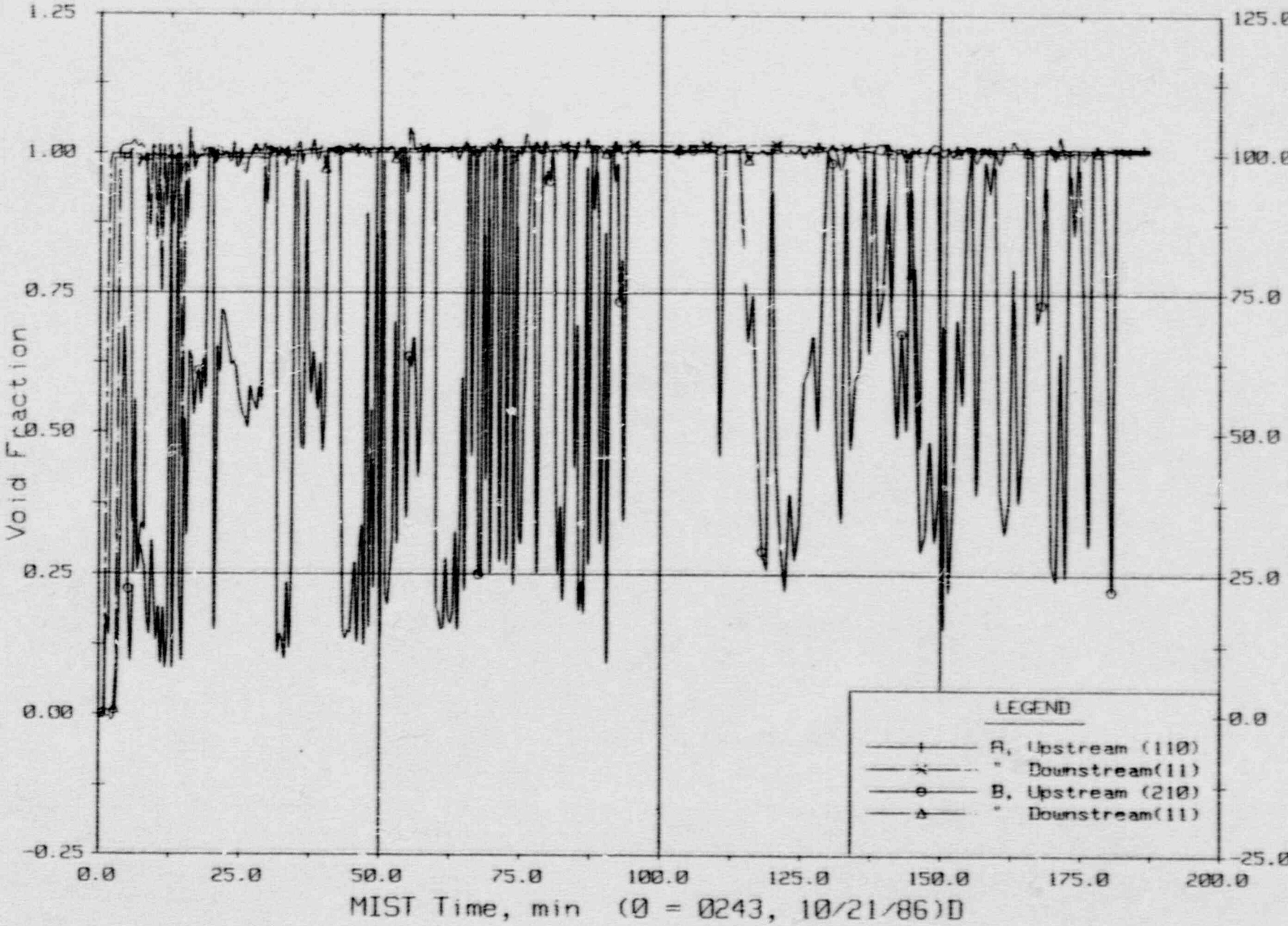
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Hot Leg Riser and Stub Collapsed Liquid Levels.

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



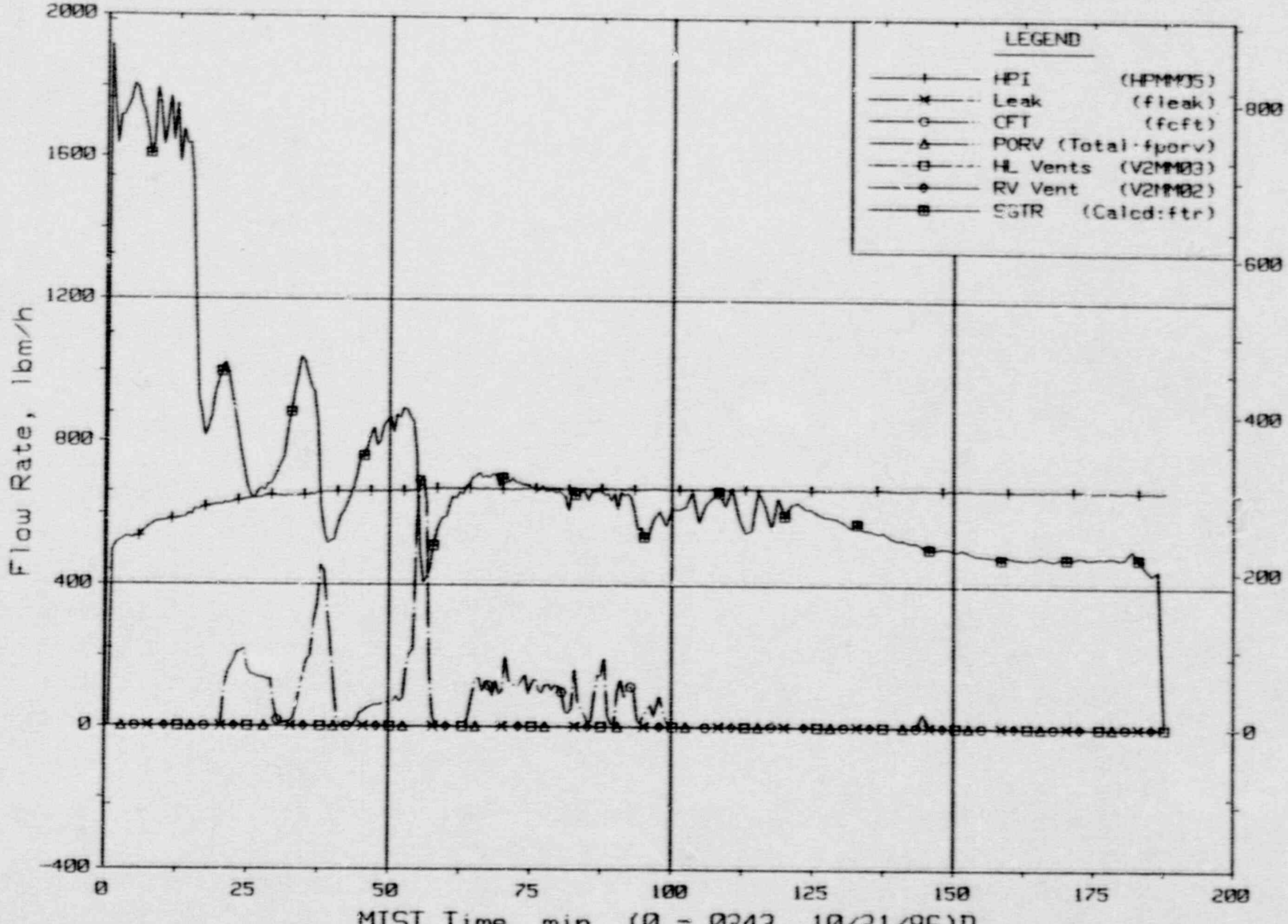
Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Thu Jun 2 08:13:39 1988

HLVF1

FINAL DATA

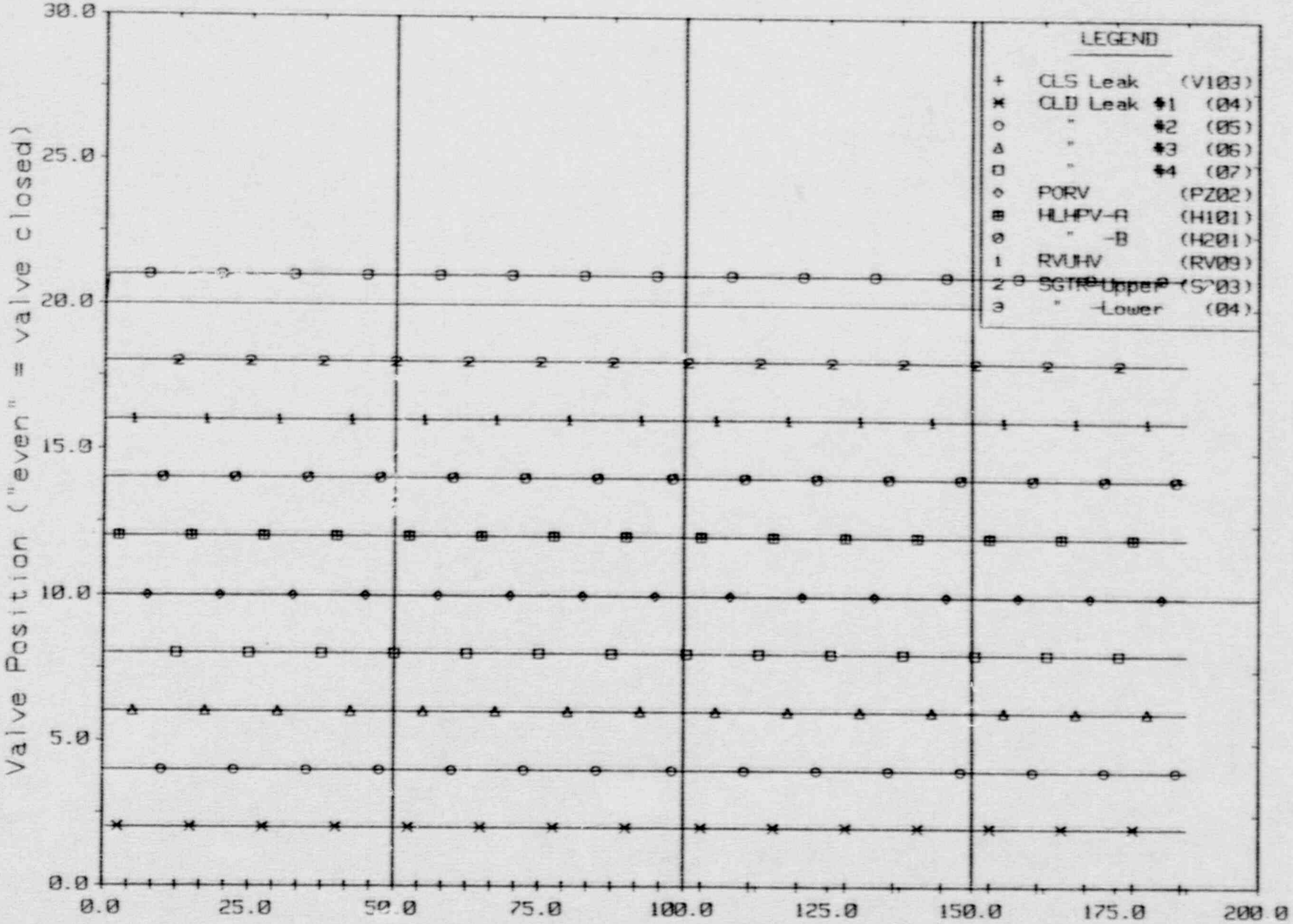
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Primary System Boundary Flow Rates.

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tu... Rupture.



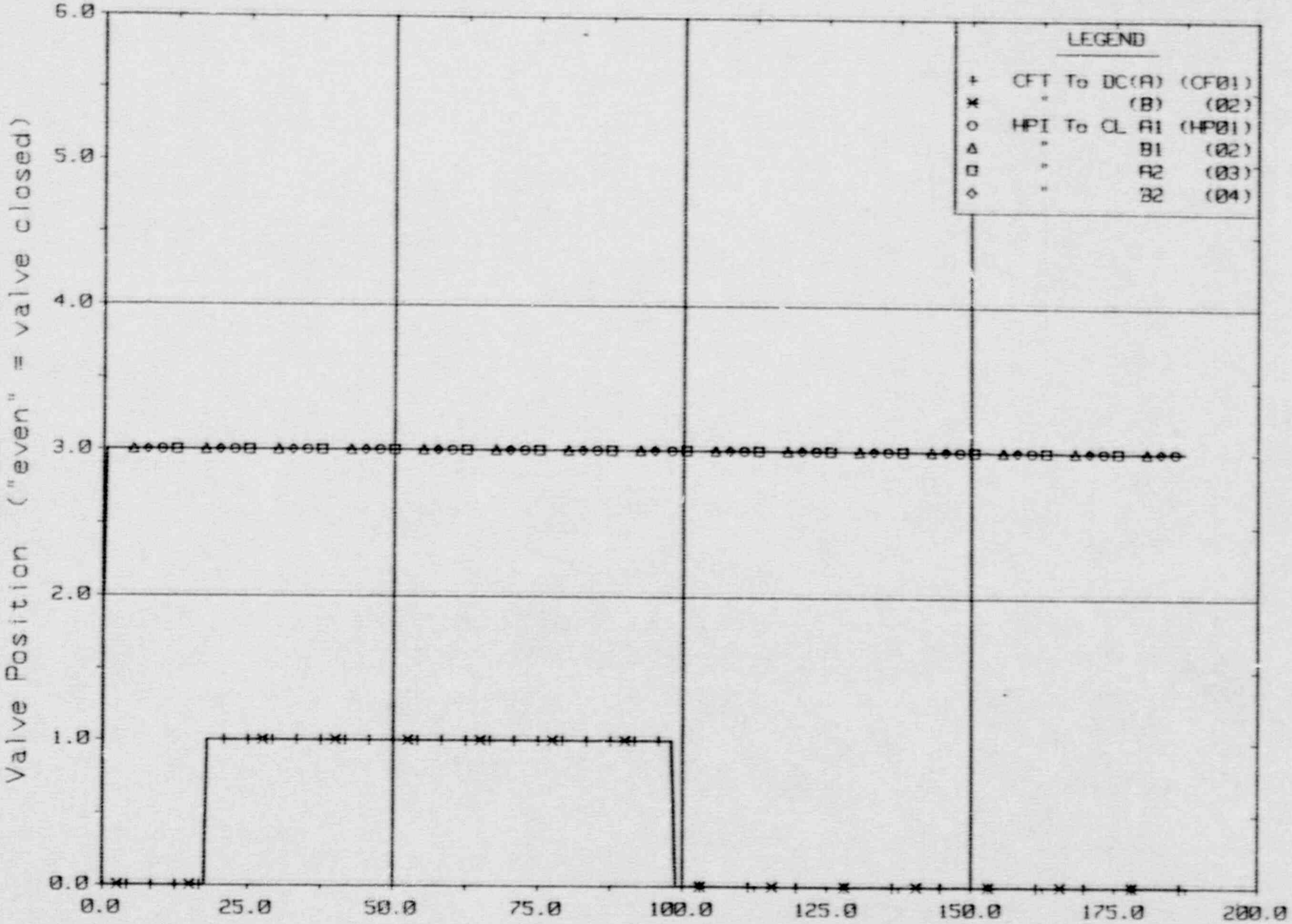
Primary System Discharge Limit Switch Indications (LSs).

Thu Jun 2 08:22:11 1988

PBLS1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



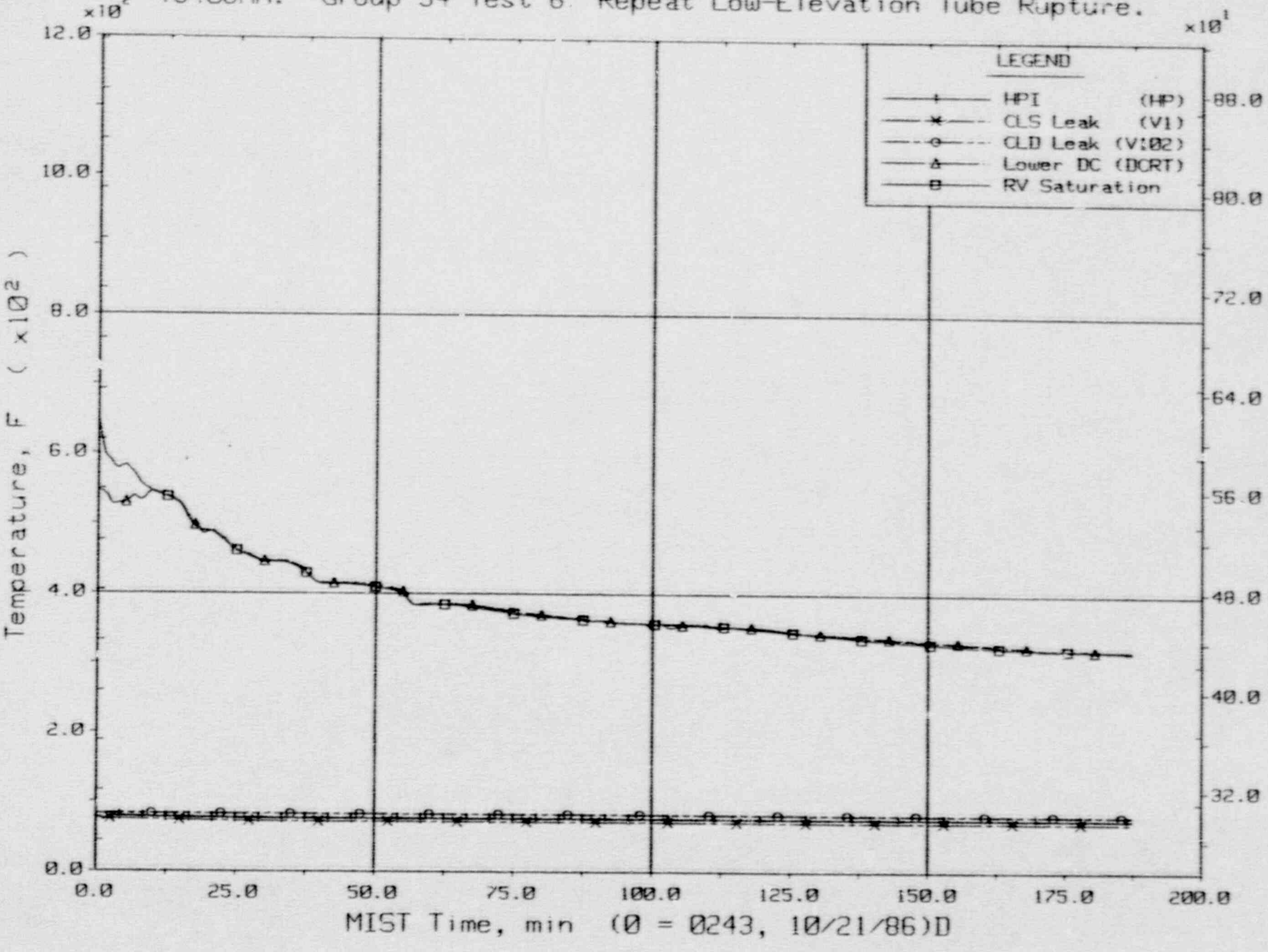
Primary System Injection Limit Switch Indications (LSs).

Thu Jun 2 08:25:07 1988

PBLS2

FINAL DATA

T3406AA: Group 34 Test 6 Repeat Low-Elevation Tube Rupture.



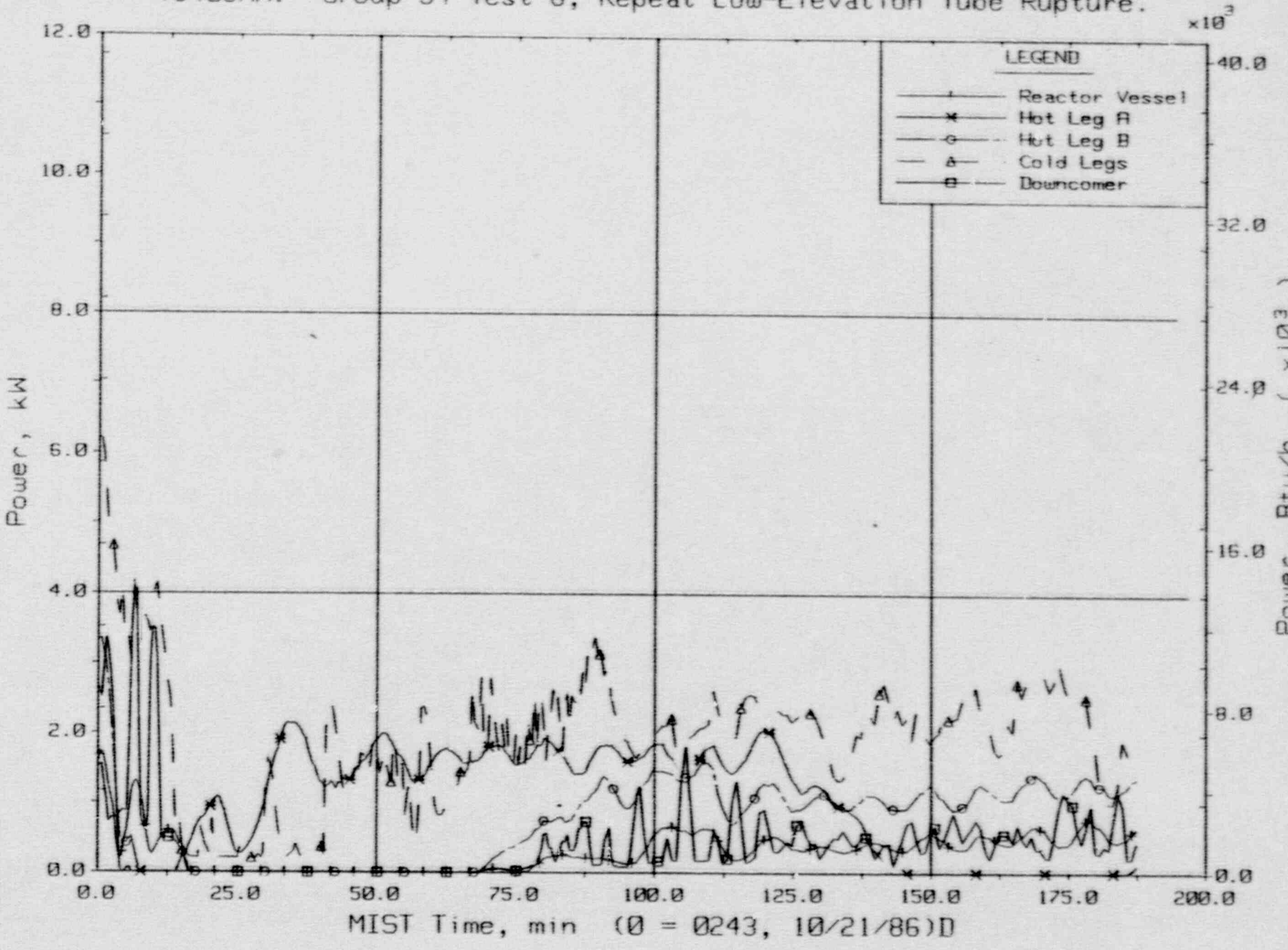
Single-Phase Discharge and HPI Fluid Temperatures (TC01s).

Thu Jun 2 08:27:54 1988

PBTC1p

FINAL DATA

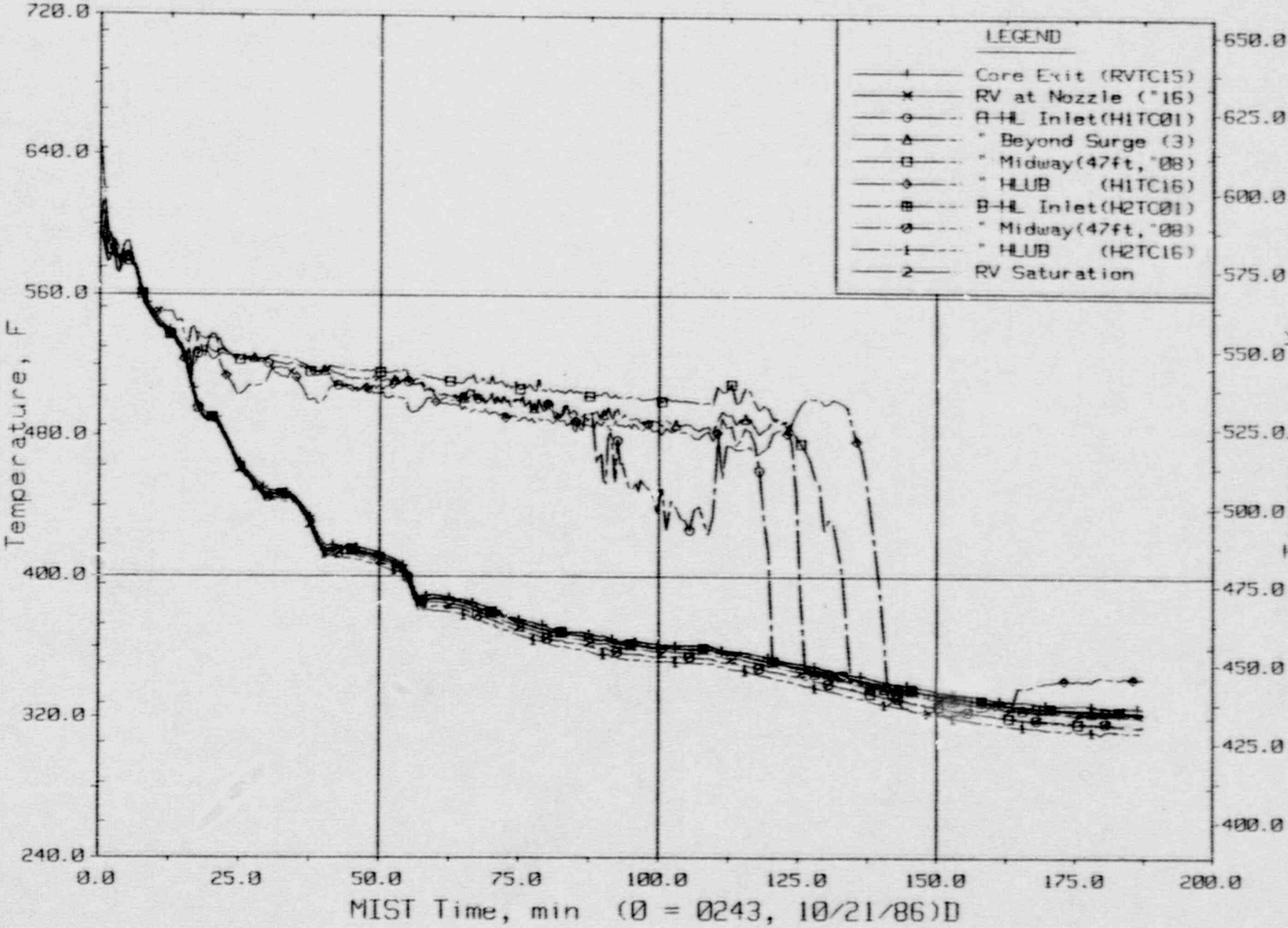
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Guard Heater Specified Power Per Primary Component.

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



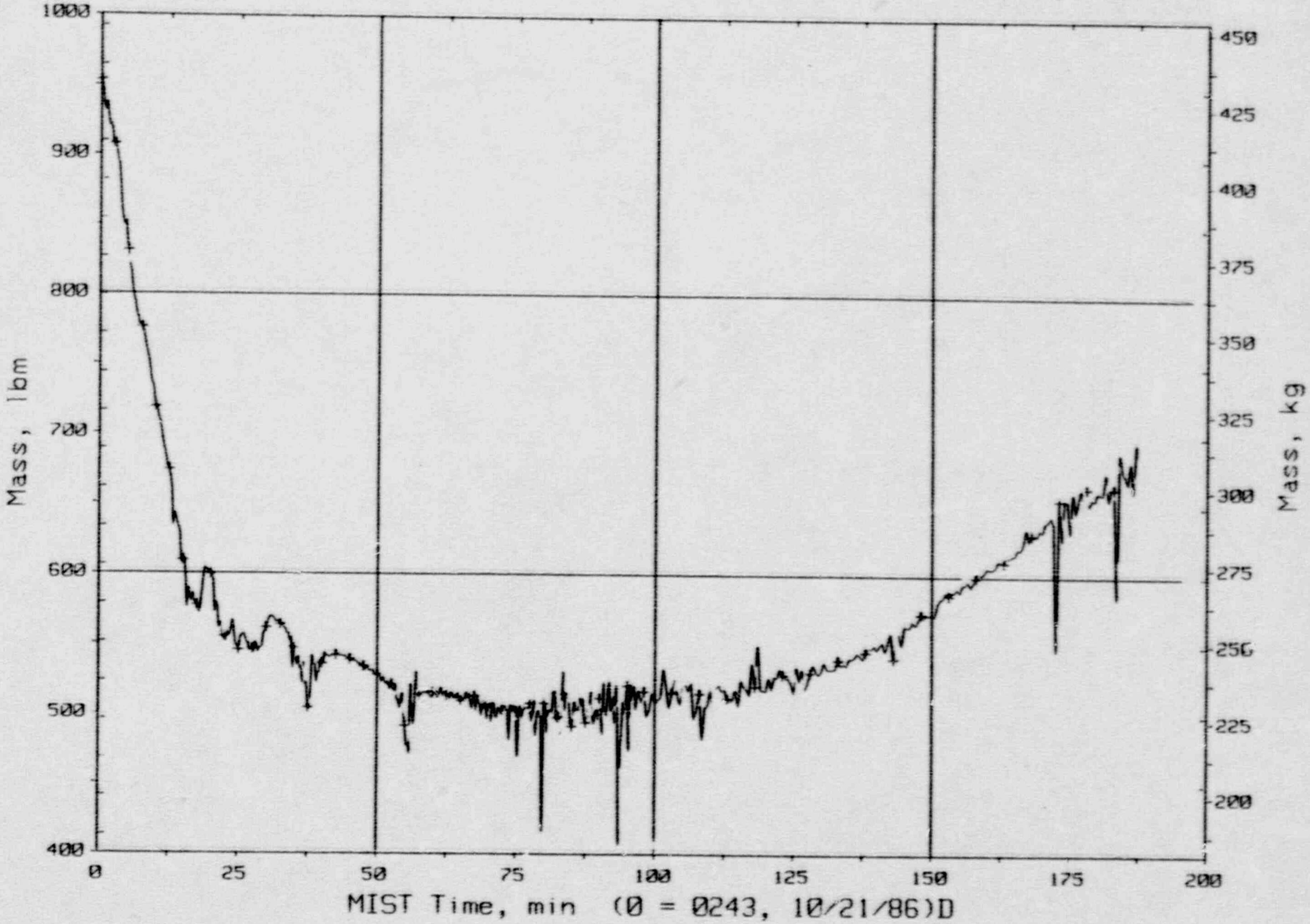
Composite Core Exit and Hot Leg Fluid Temperatures.

Thu Jun 2 08:38:18 1988

PRKT1

FINAL DATA

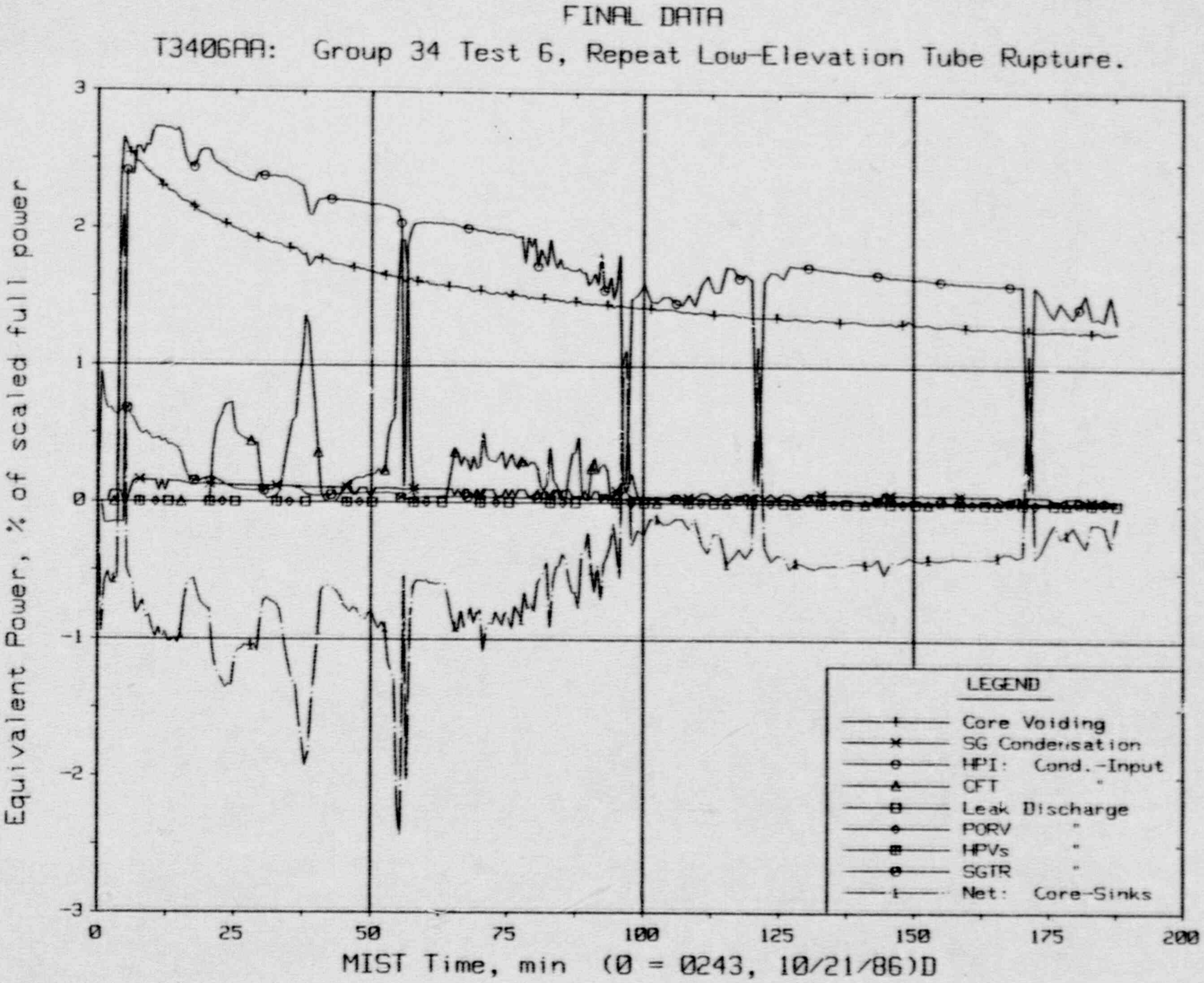
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Indicated Primary System Total Fluid Mass (PLML20).

Wed Nov 2 14:10:42 1988

PRML9



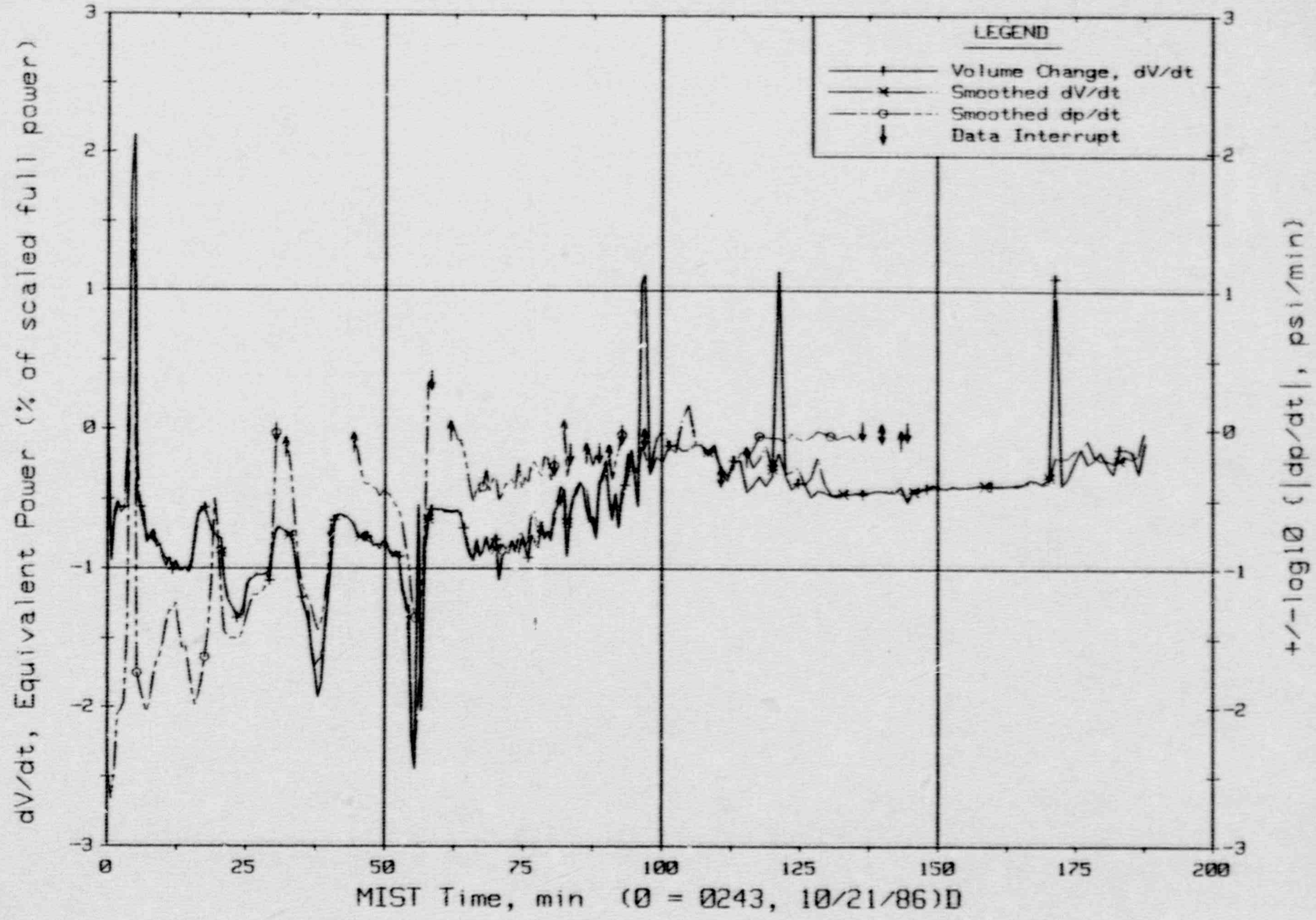
Primary Fluid Volume Changes By Components.

Fri Nov 4 17:03:44 1988

PRPA0

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



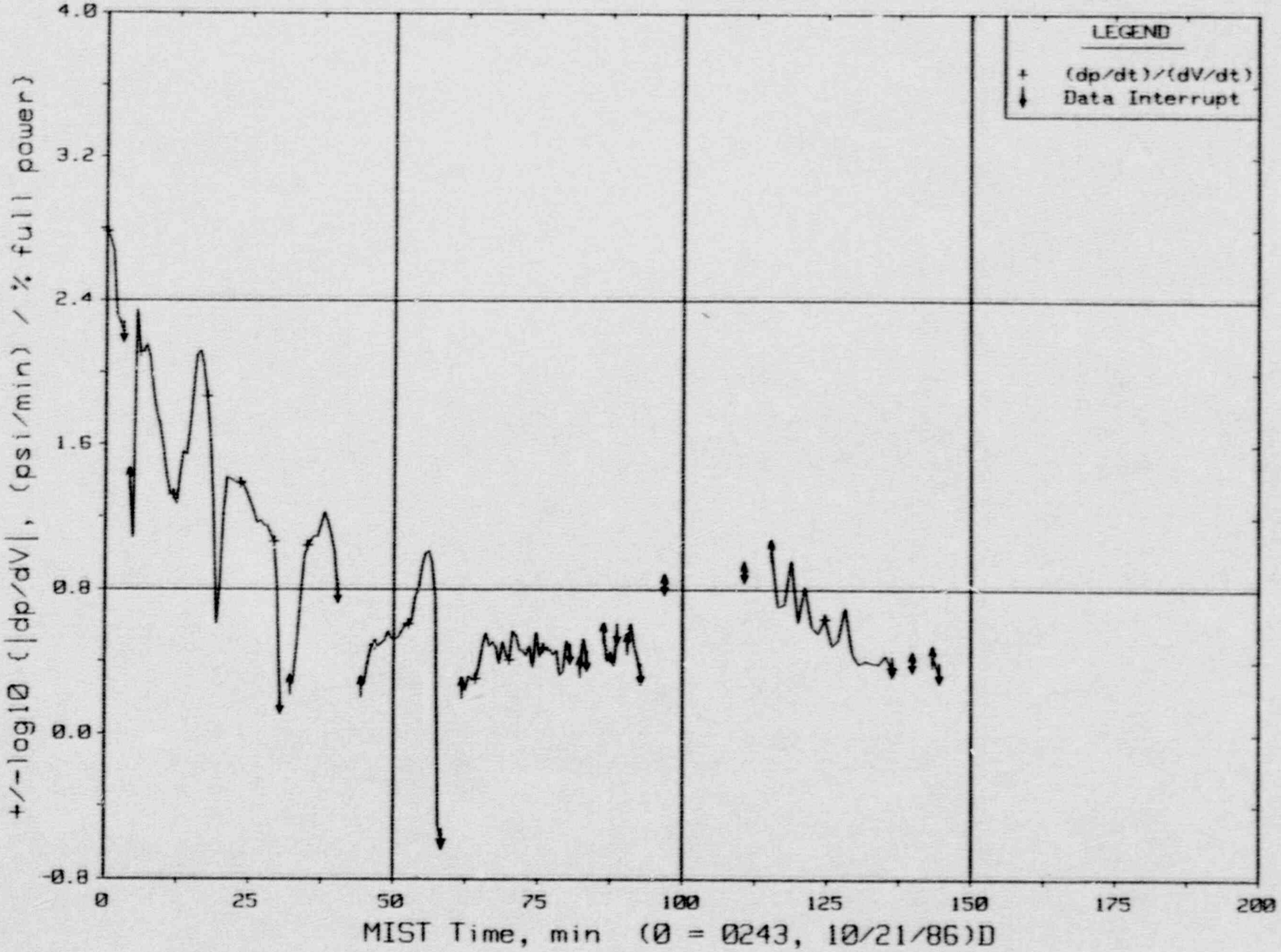
Primary Fluid Volume and Pressure Changes, dV/dt and dp/dt.

Fri Nov 4 17:12:11 1988

PRPA2

FINAL DATA

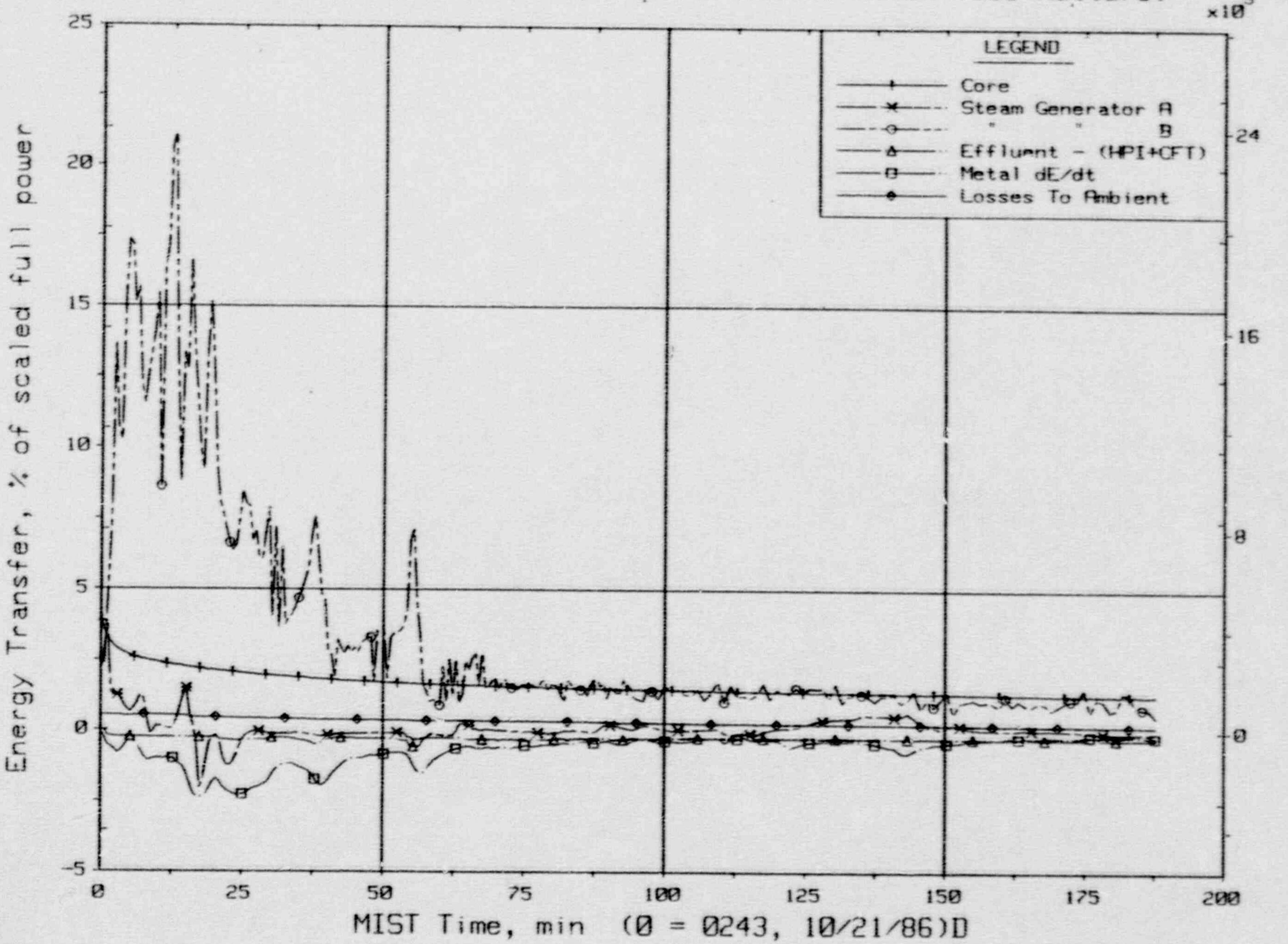
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Primary System Response: dp/dt divided by dV/dt.

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



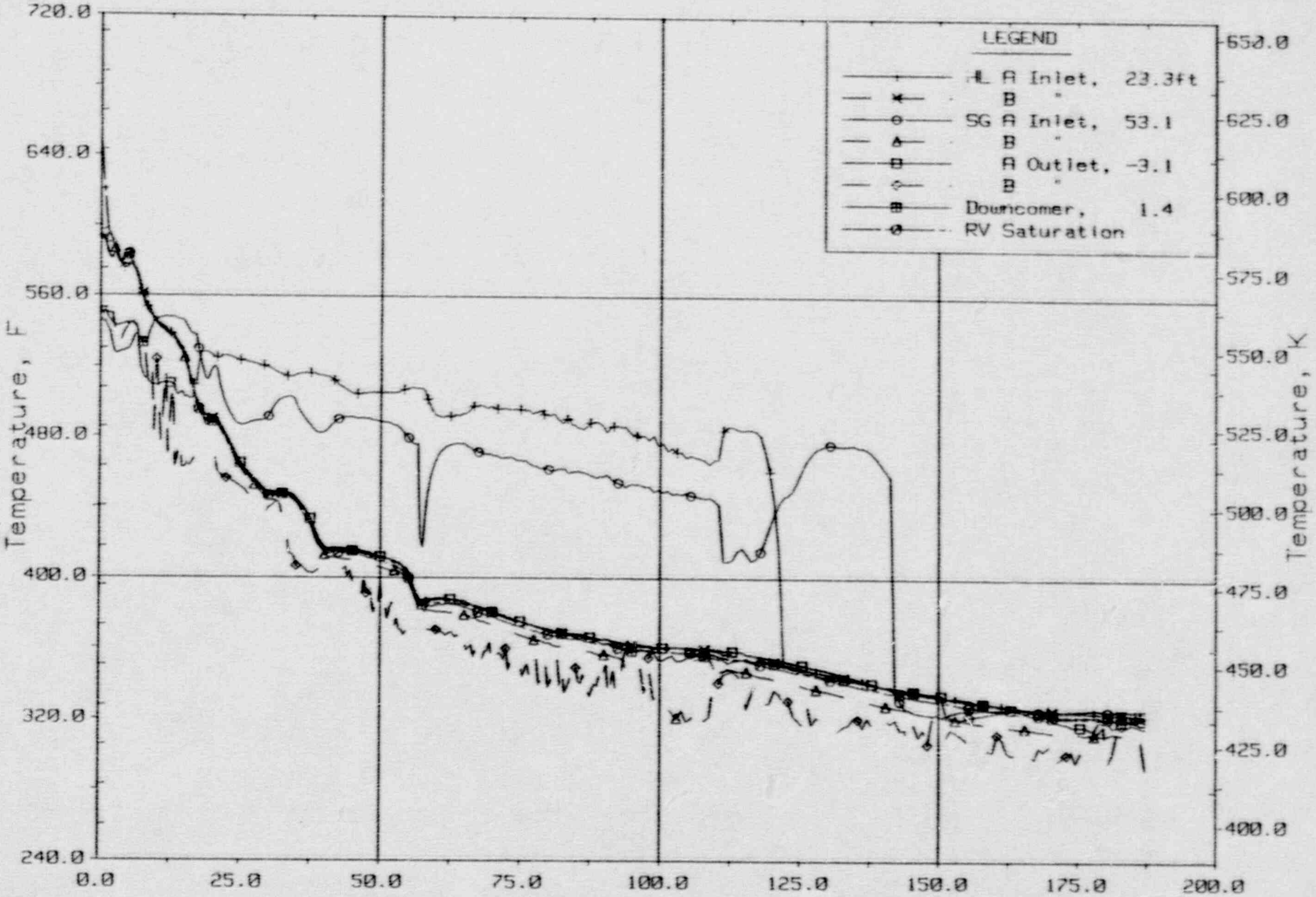
Primary System Energy Transfer.

Fri Nov 4 16:44:41 1988

PRQE2

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



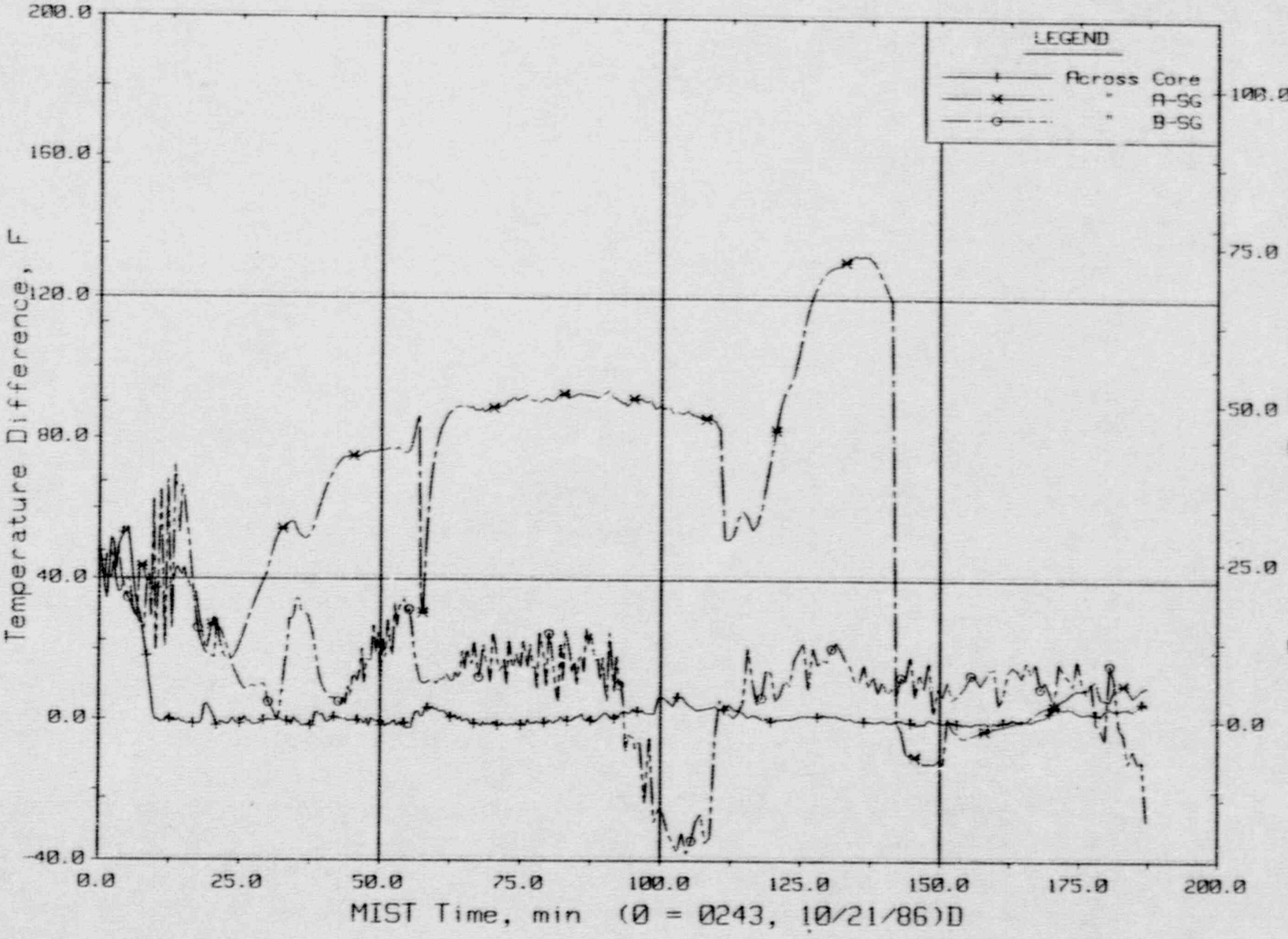
Primary System Fluid Temperatures (RTDs).

Thu Jun 2 08:57:03 1988

PRRT0

FINAL DATA

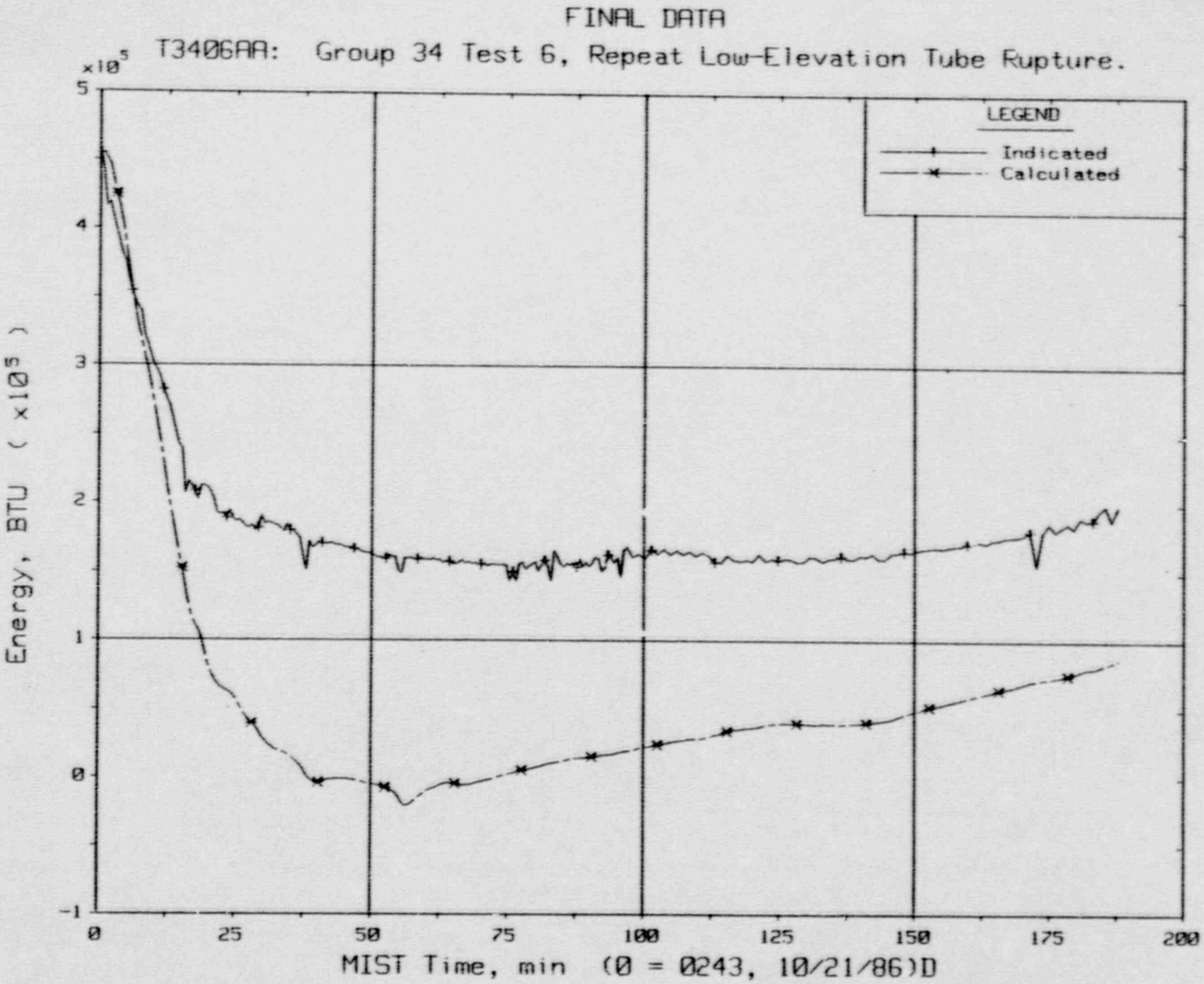
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Key Temperature Differences.

Thu Jun 2 08:59:55 1988

PRTD2



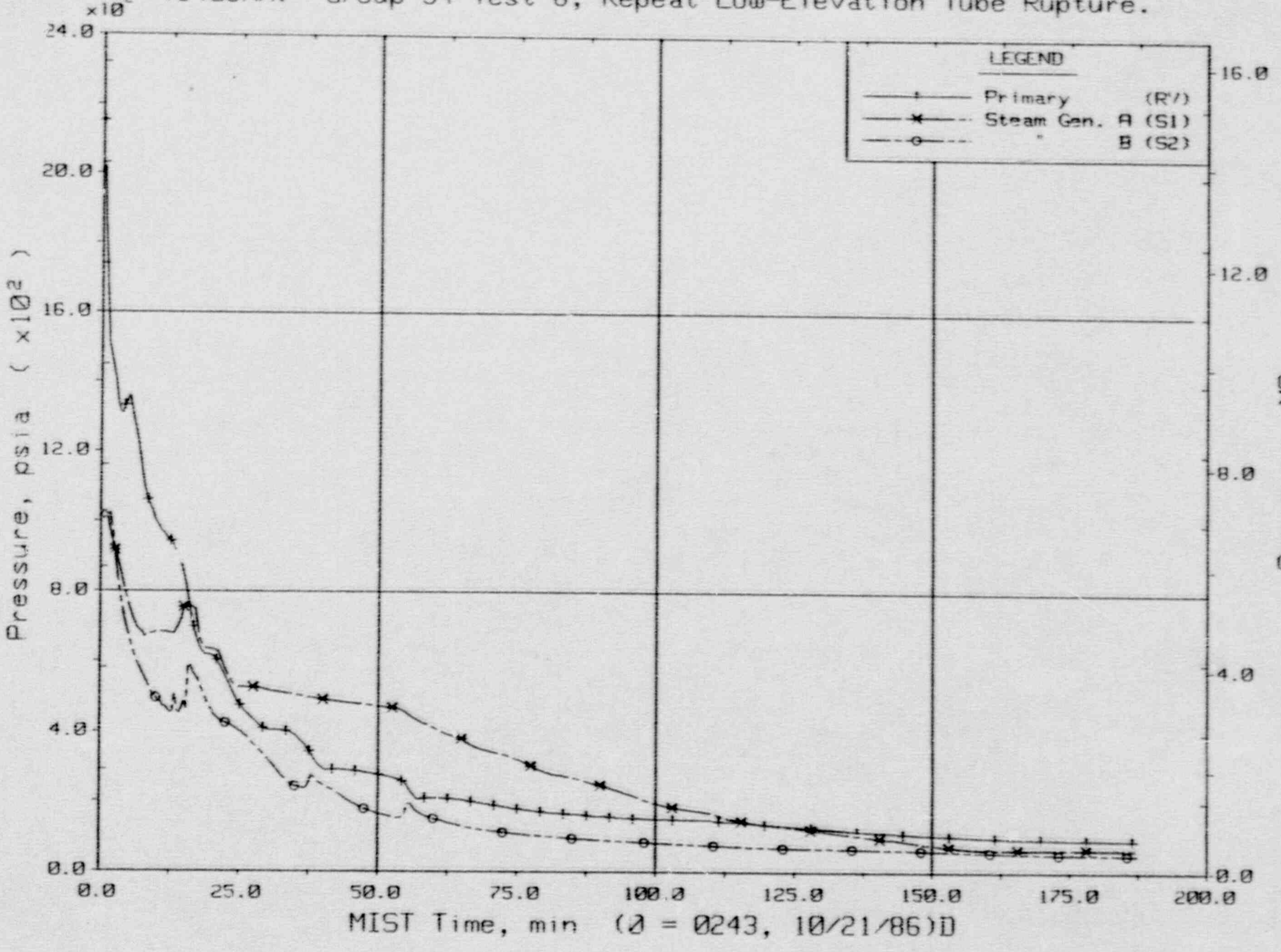
Primary System Total Fluid Energy.

Fri Nov 4 16:46:54 1988

PRTE2

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



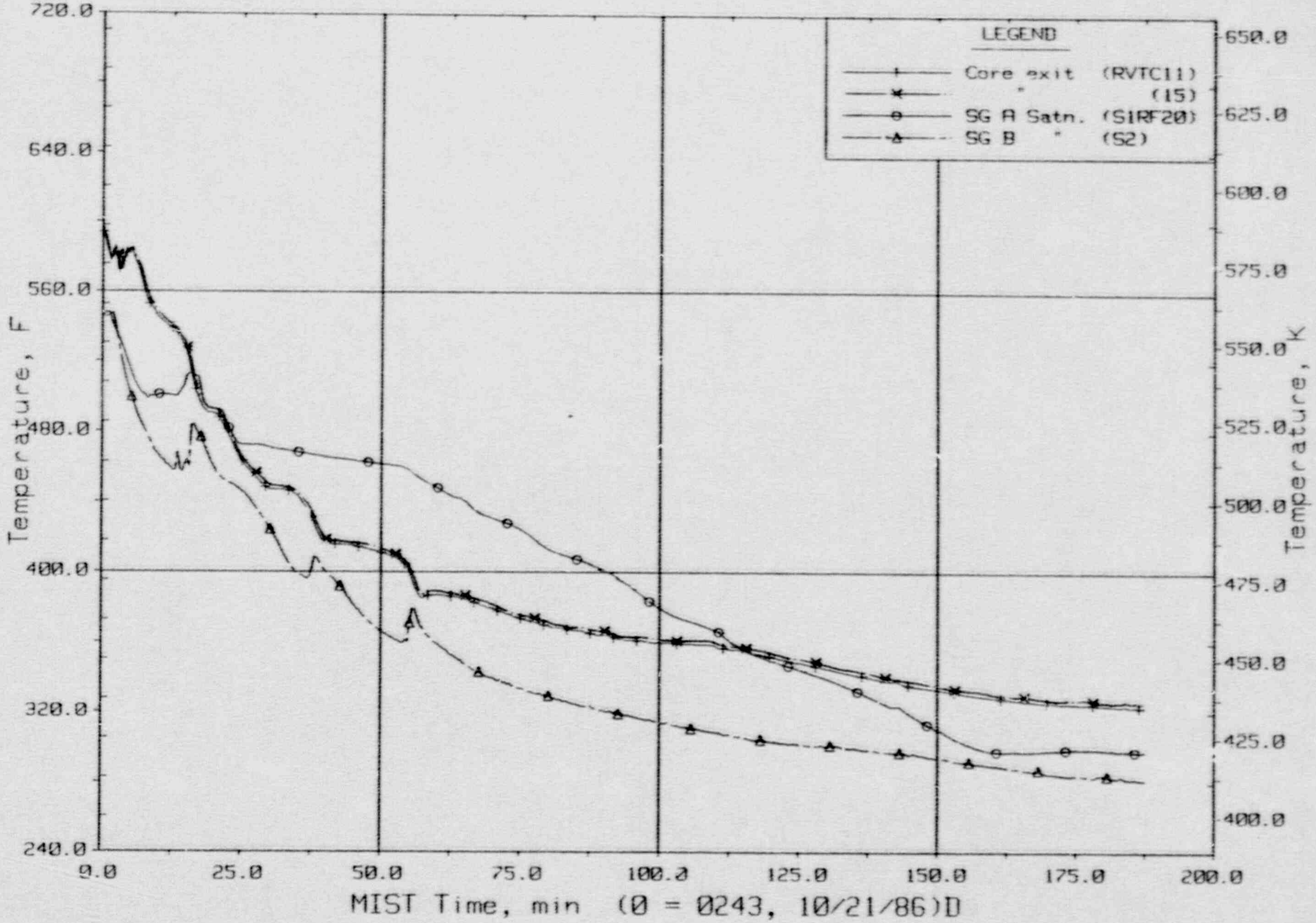
Primary and Secondary System Pressures (GP01s).

Thu Jun 2 09:06:29 1988

PSGP1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



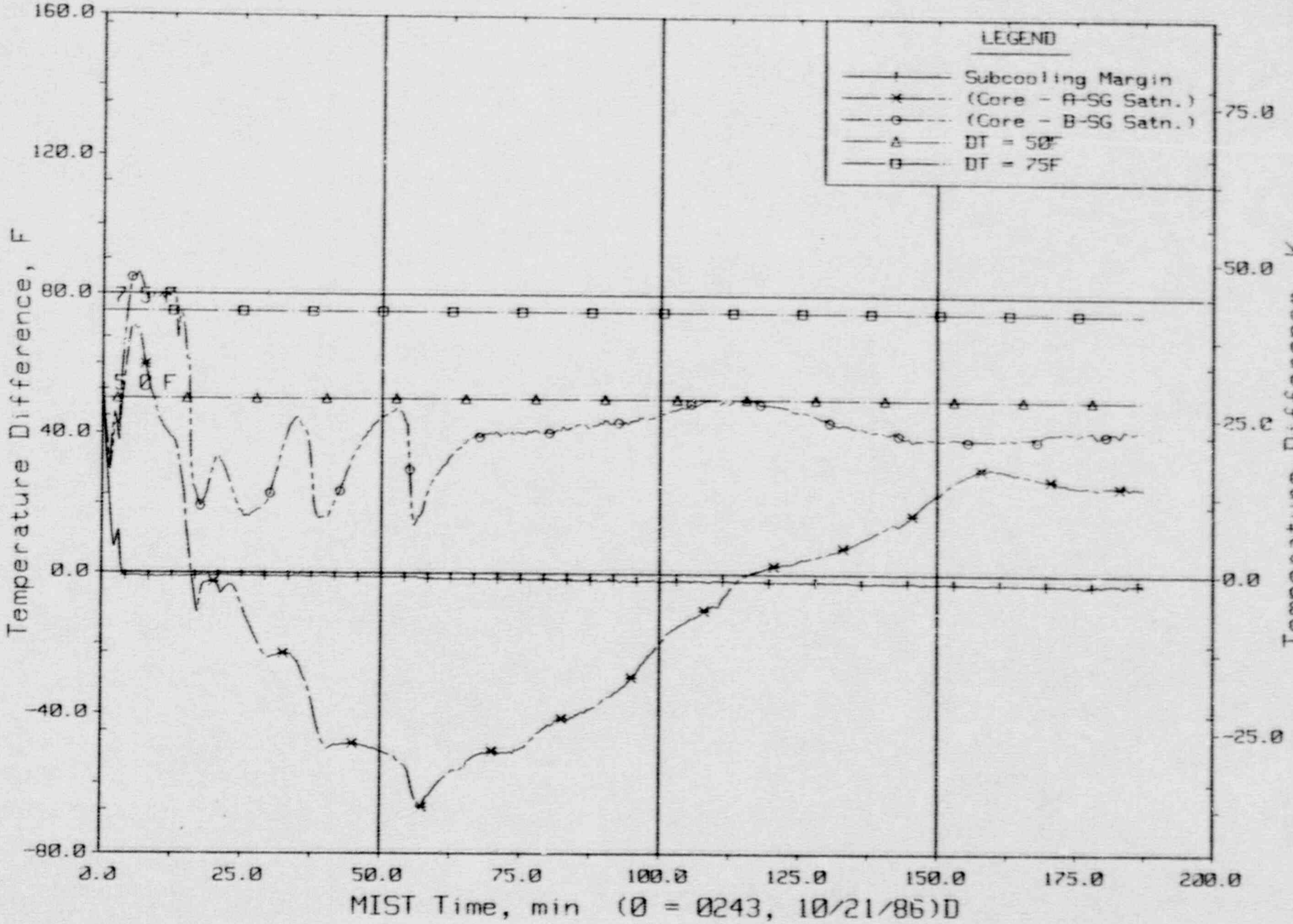
Core Exit and Steam Generator Secondary Saturation Temperatures.

Thu Jun 2 09:09:56 1988

PSTC3p

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



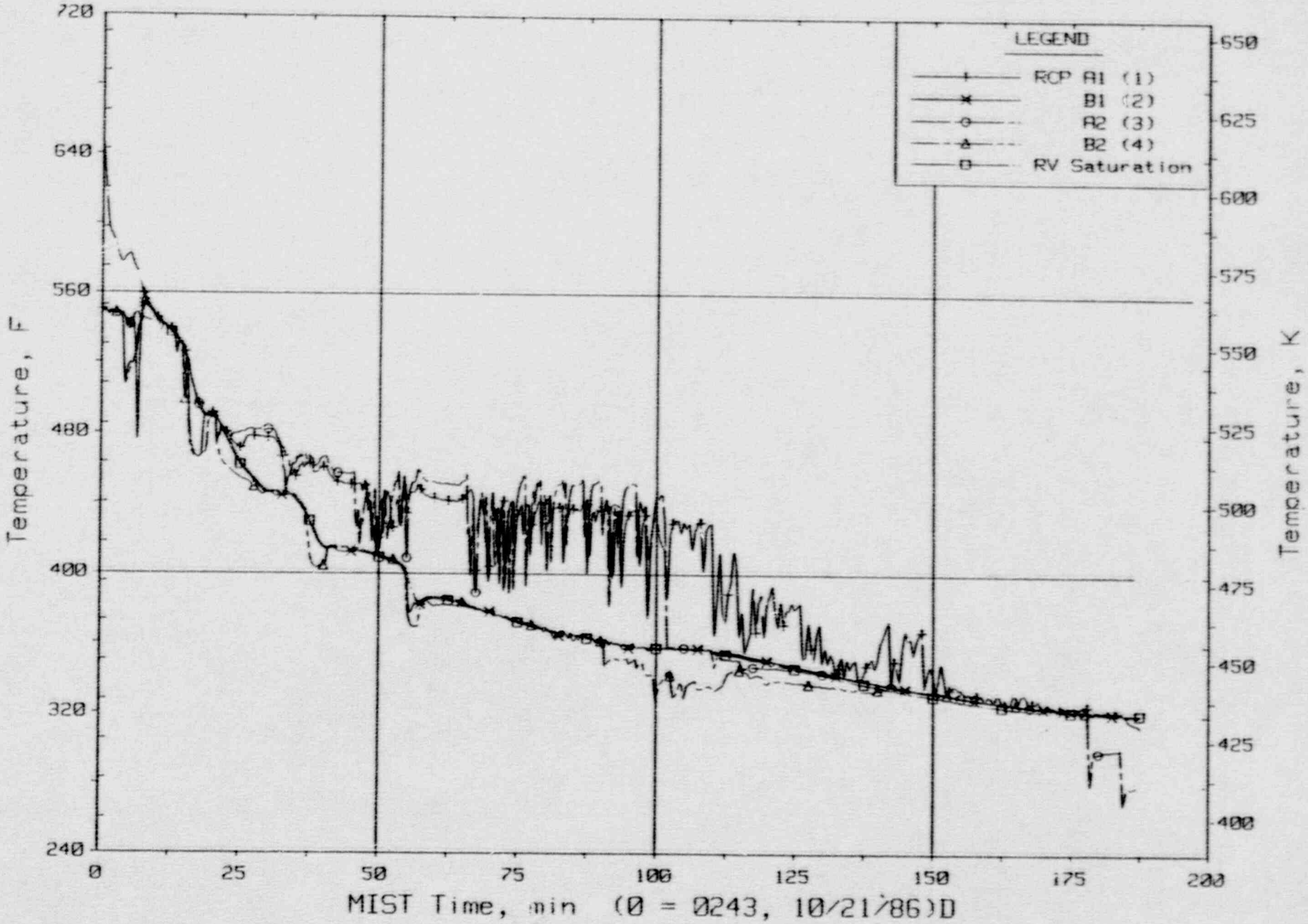
Control Temperature Differences.

Thu Jun 2 09:16:34 1988

PSTD1

FINAL DATA

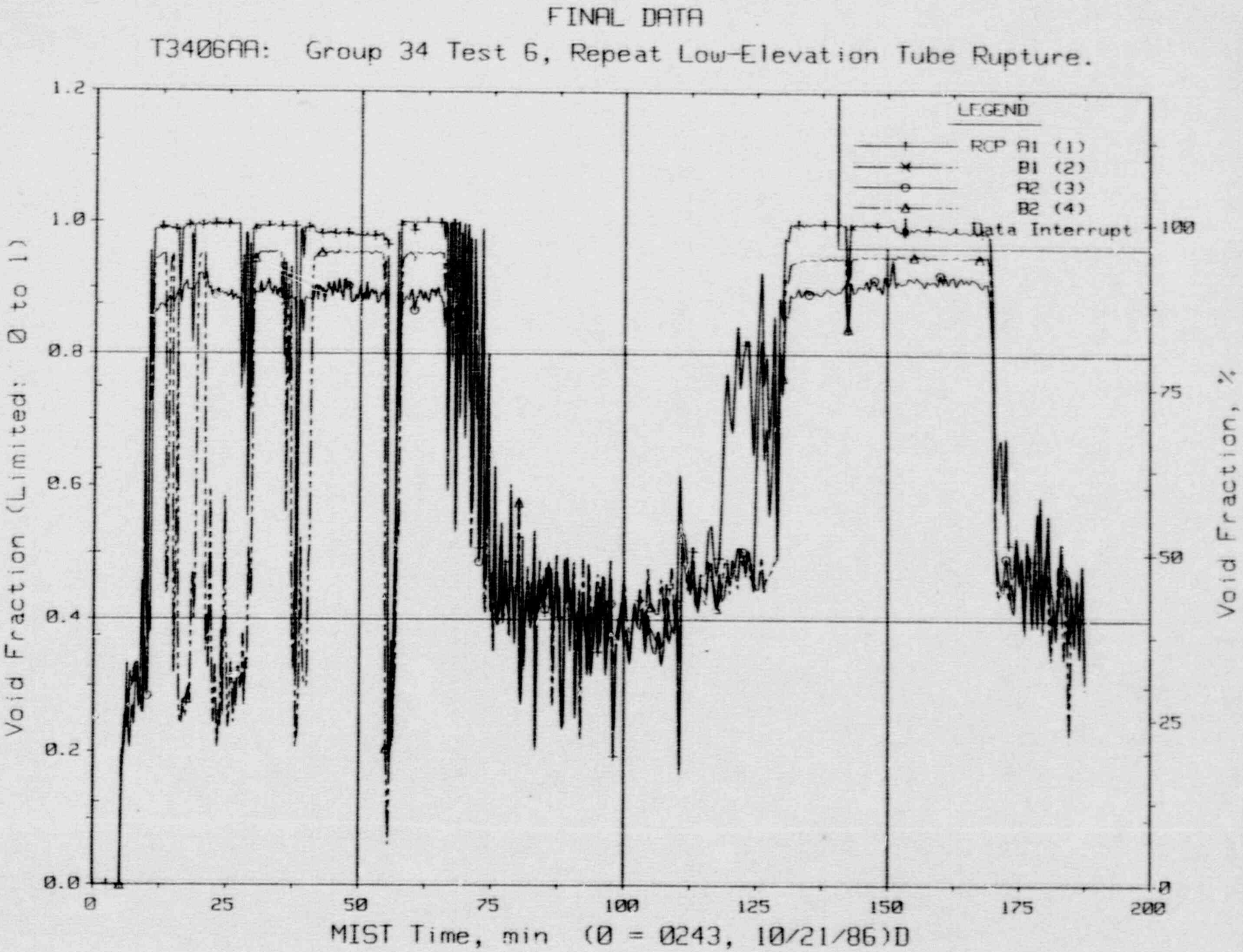
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 15:25:39 1988

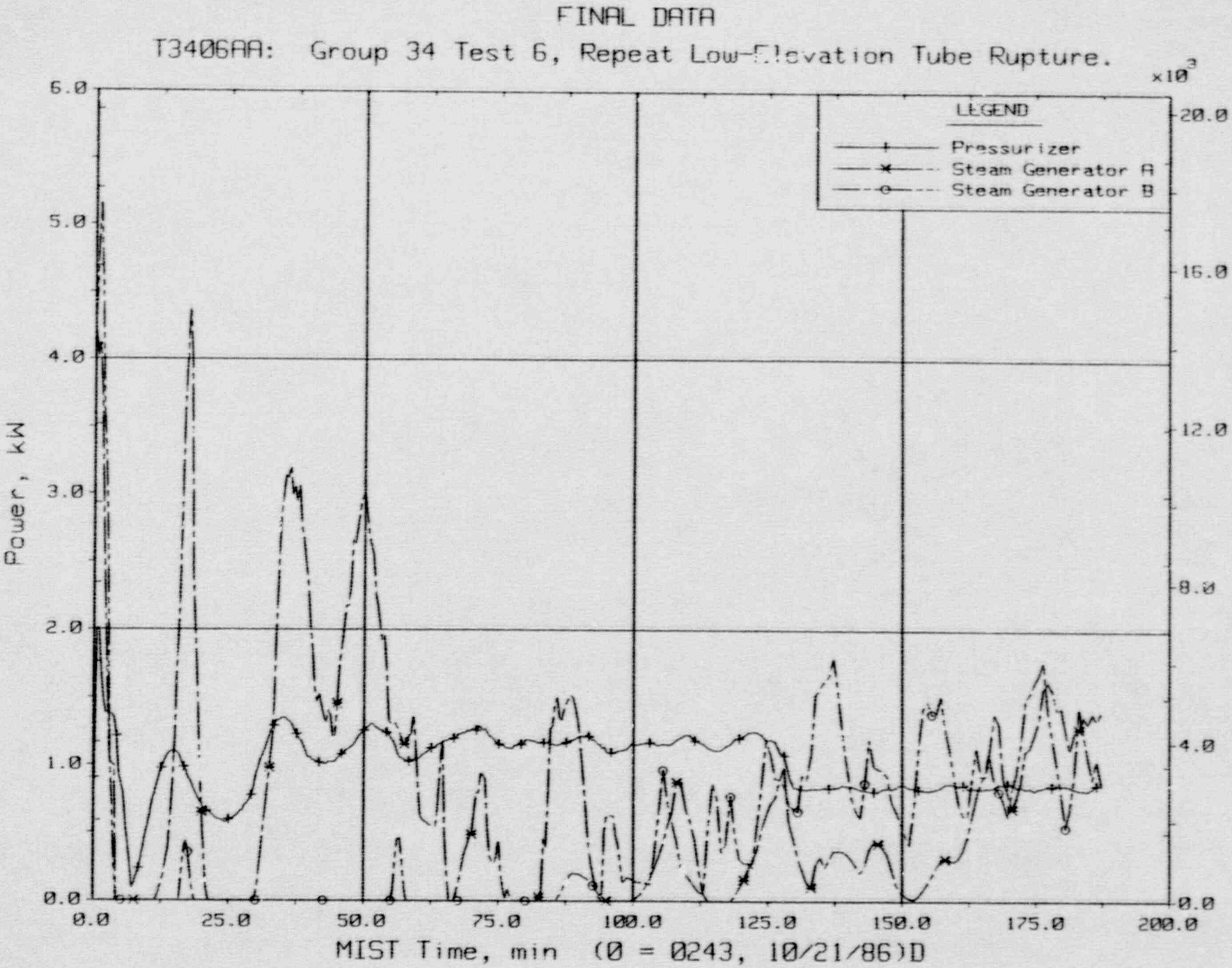
PURT0



Pump Suction Void Fraction From Gamma Densitometers (CnGD21).

Tue Nov 1 15:29:00 1988

PUVF2



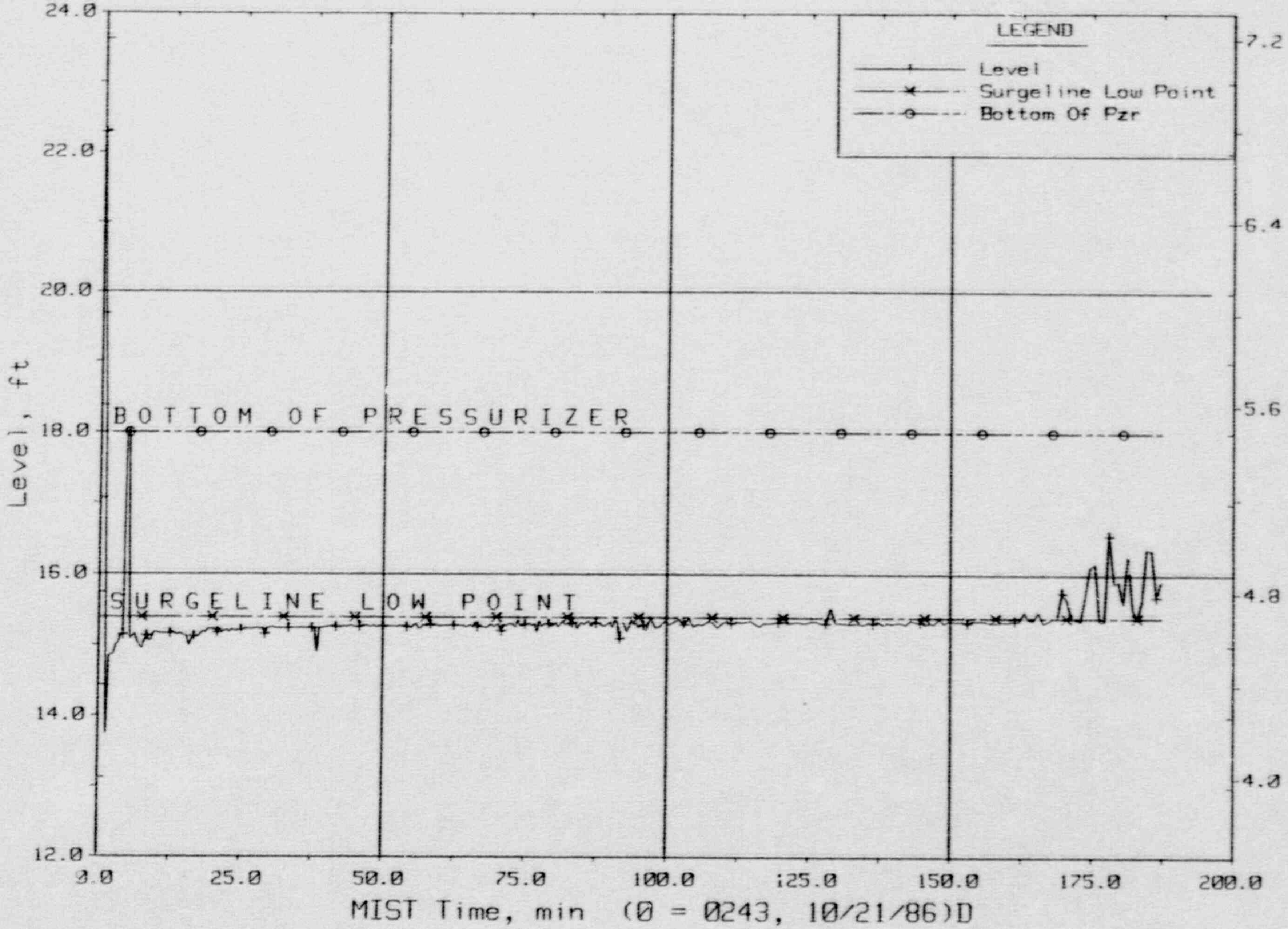
Guard Heater Specified Power, Pressurizer and Steam Generators.

Thu Jun 2 09:22:58 1988

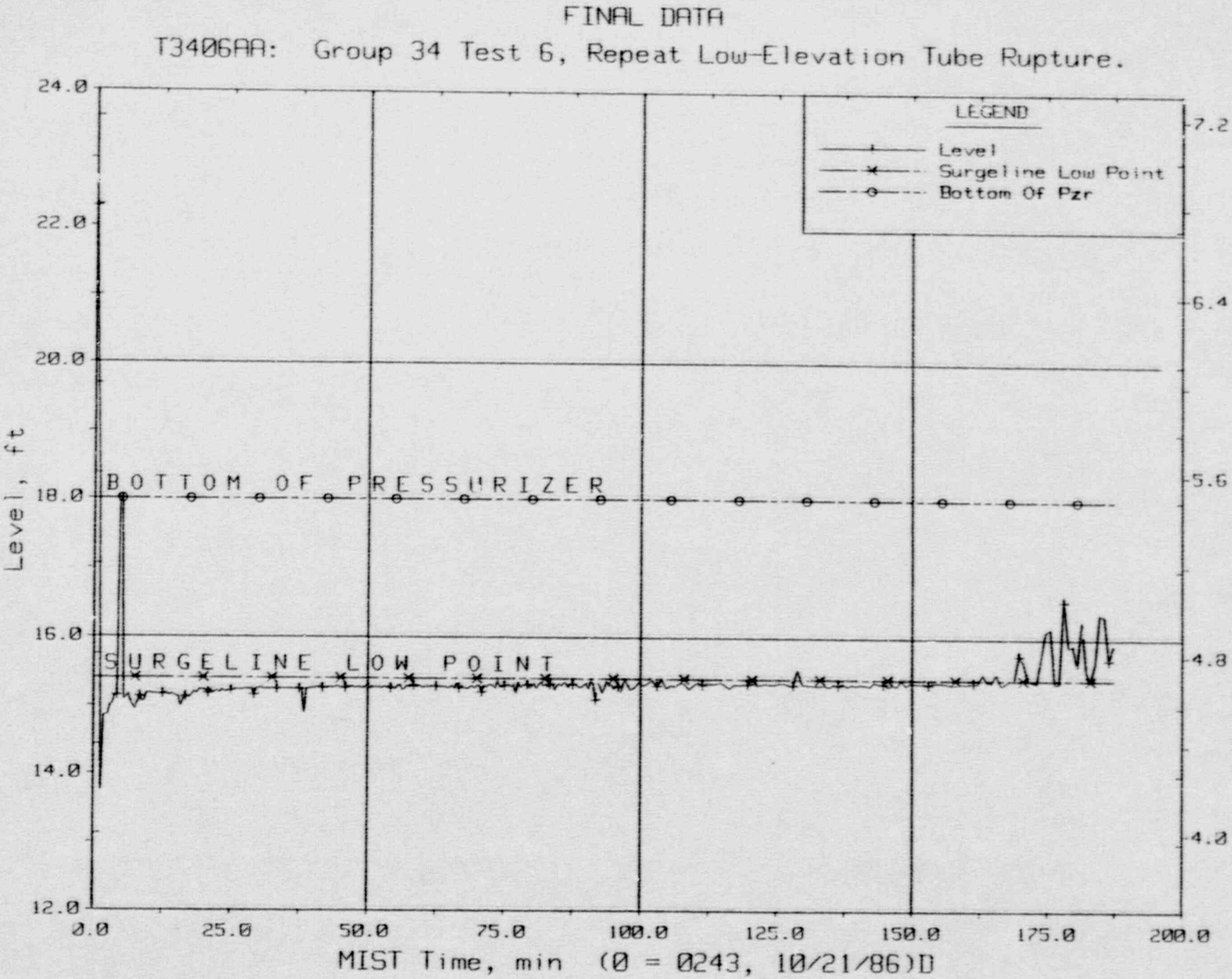
PZG01

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Pressurizer Collapsed Liquid Level (PZLV20).



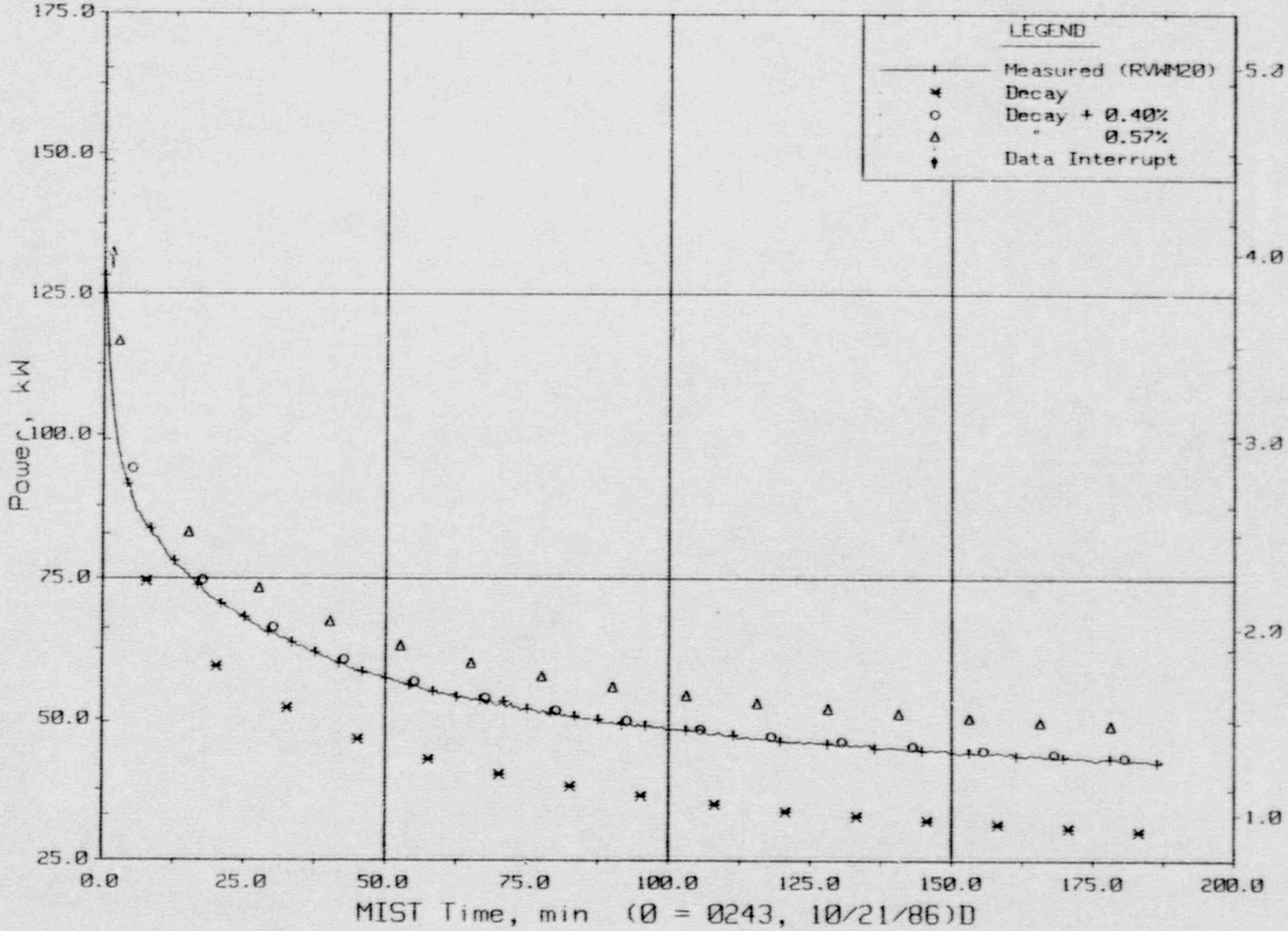
Pressurizer Collapsed Liquid Level (PZLV20).

Thu Jun 2 09:26:01 1988

PZLV1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



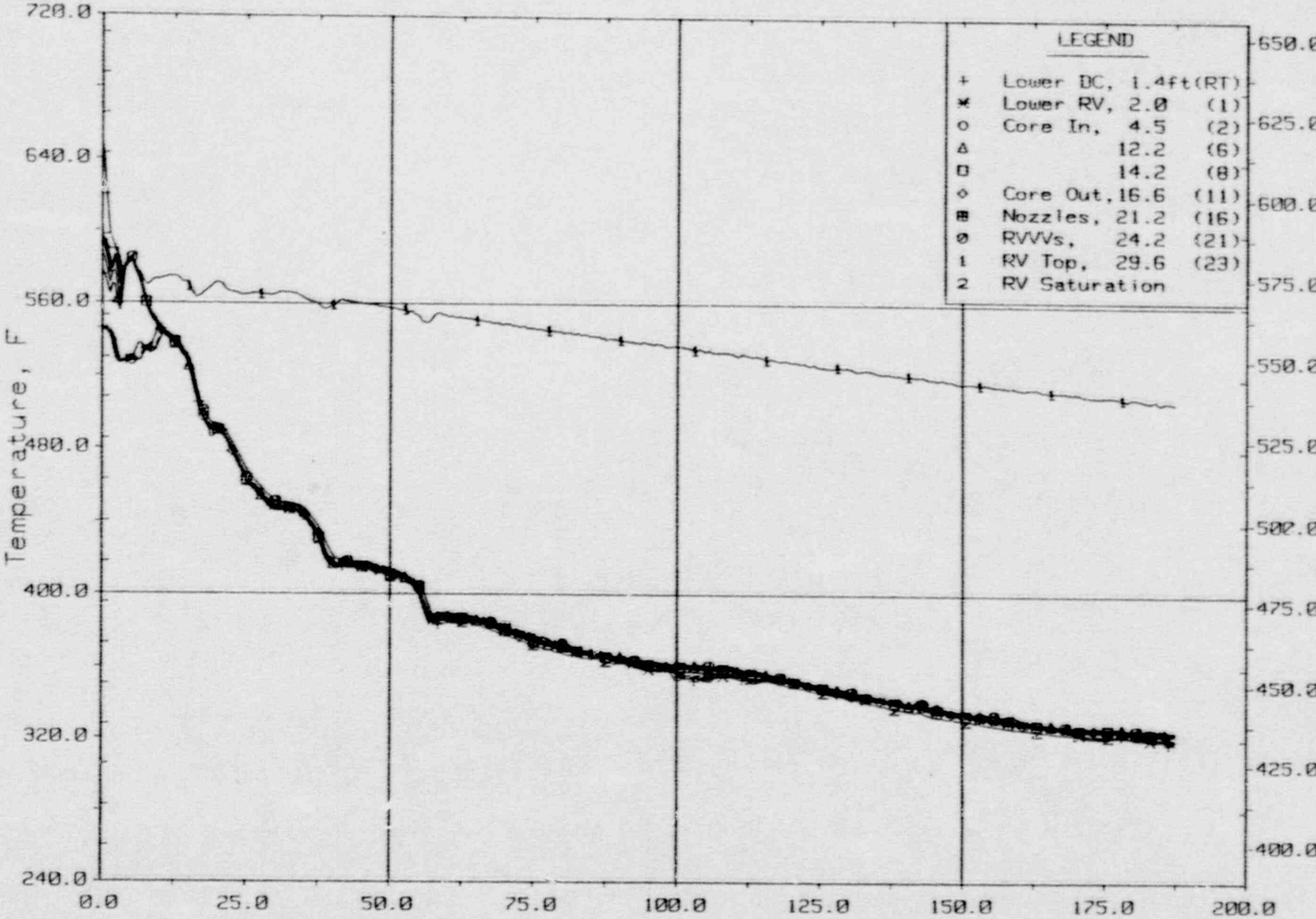
Core Power.

Thu Jun 2 09:29:27 1988

RVKW0

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.

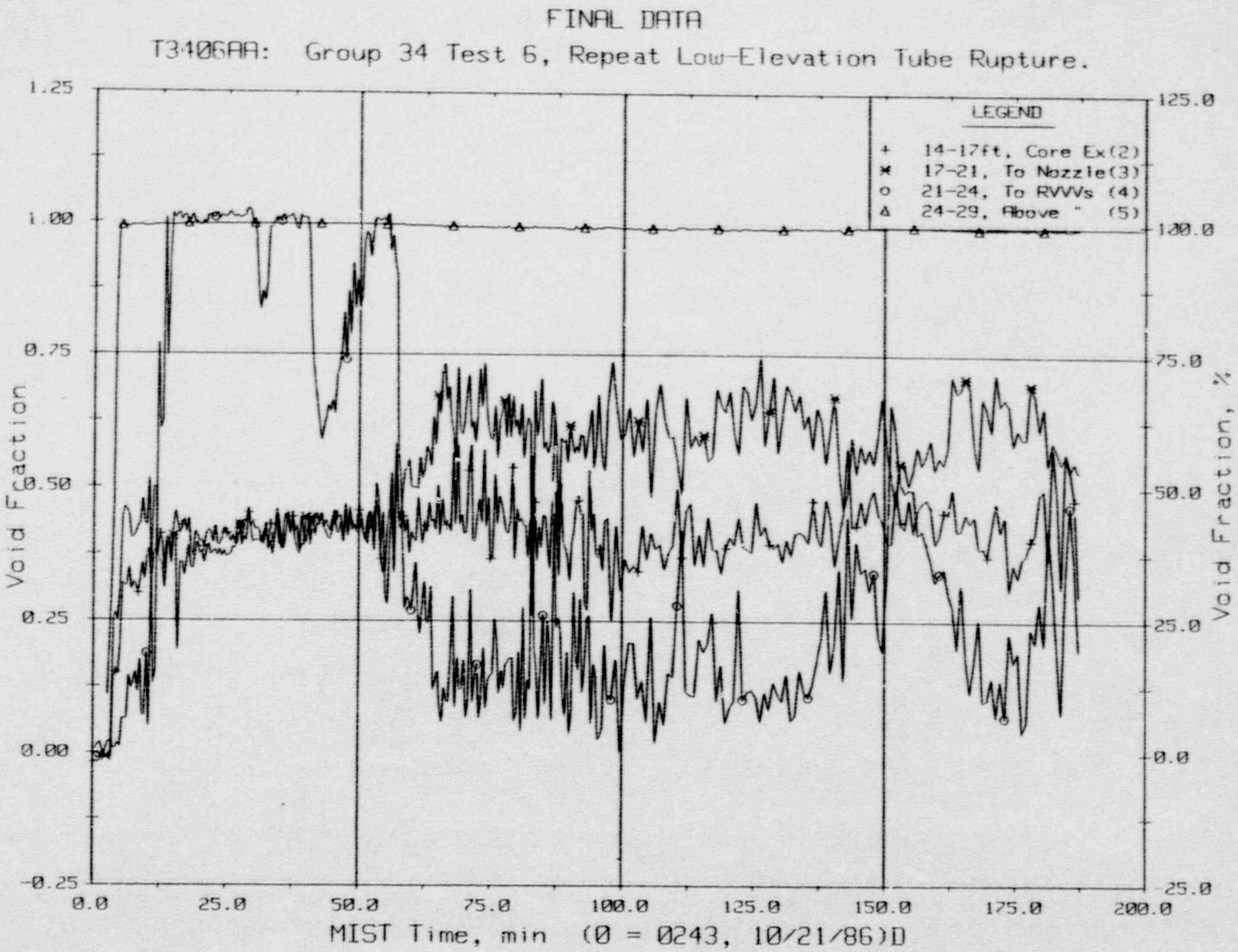


MIST Time, min (0 = 0243, 10/21/86)D

Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Thu Jun 2 09:33:19 1988

RVTC0



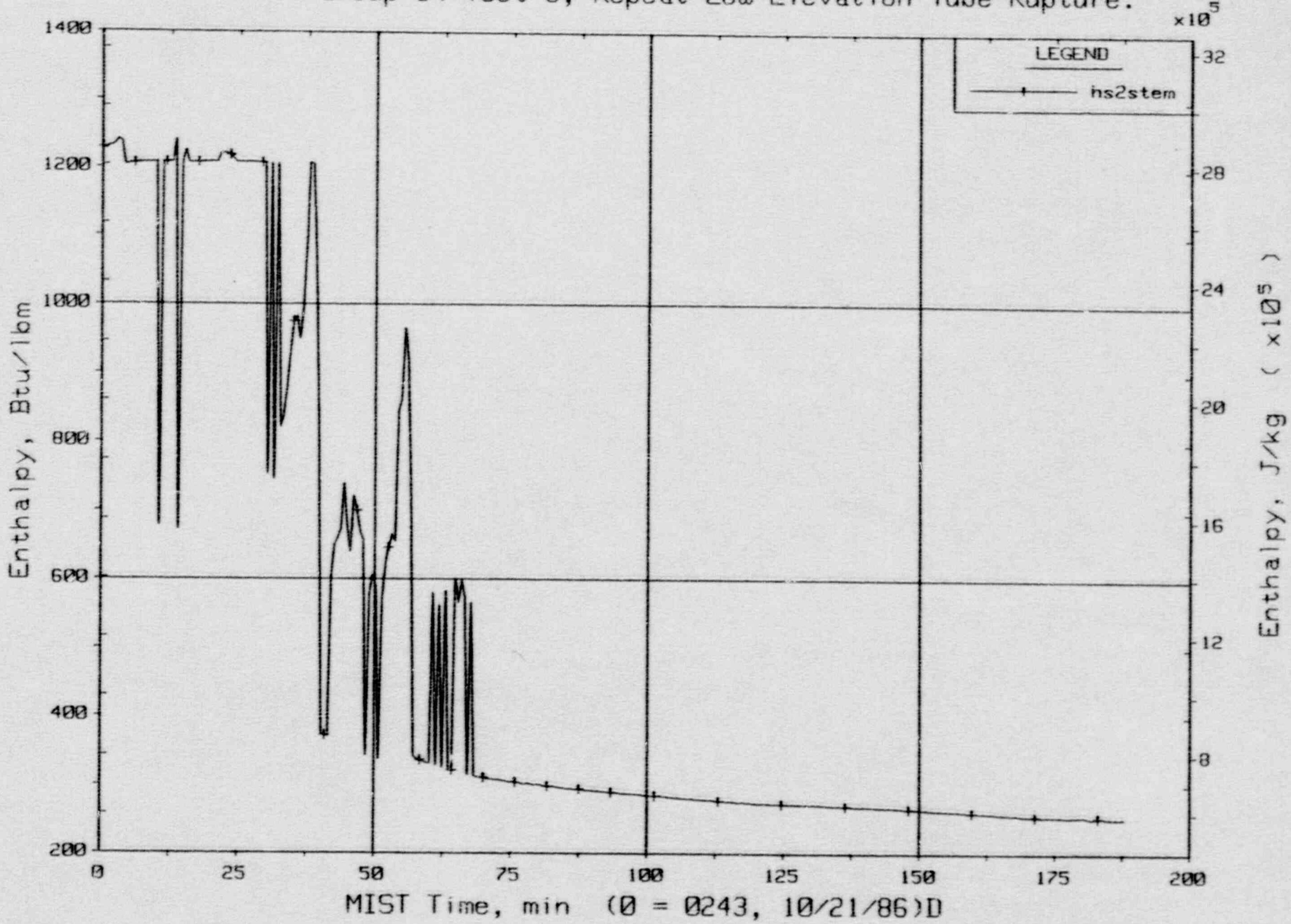
Reactor Vessel Void Fractions From Differential Pressures (RVVFs).

Thu Jun 2 09:37:08 1988

RVVF1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



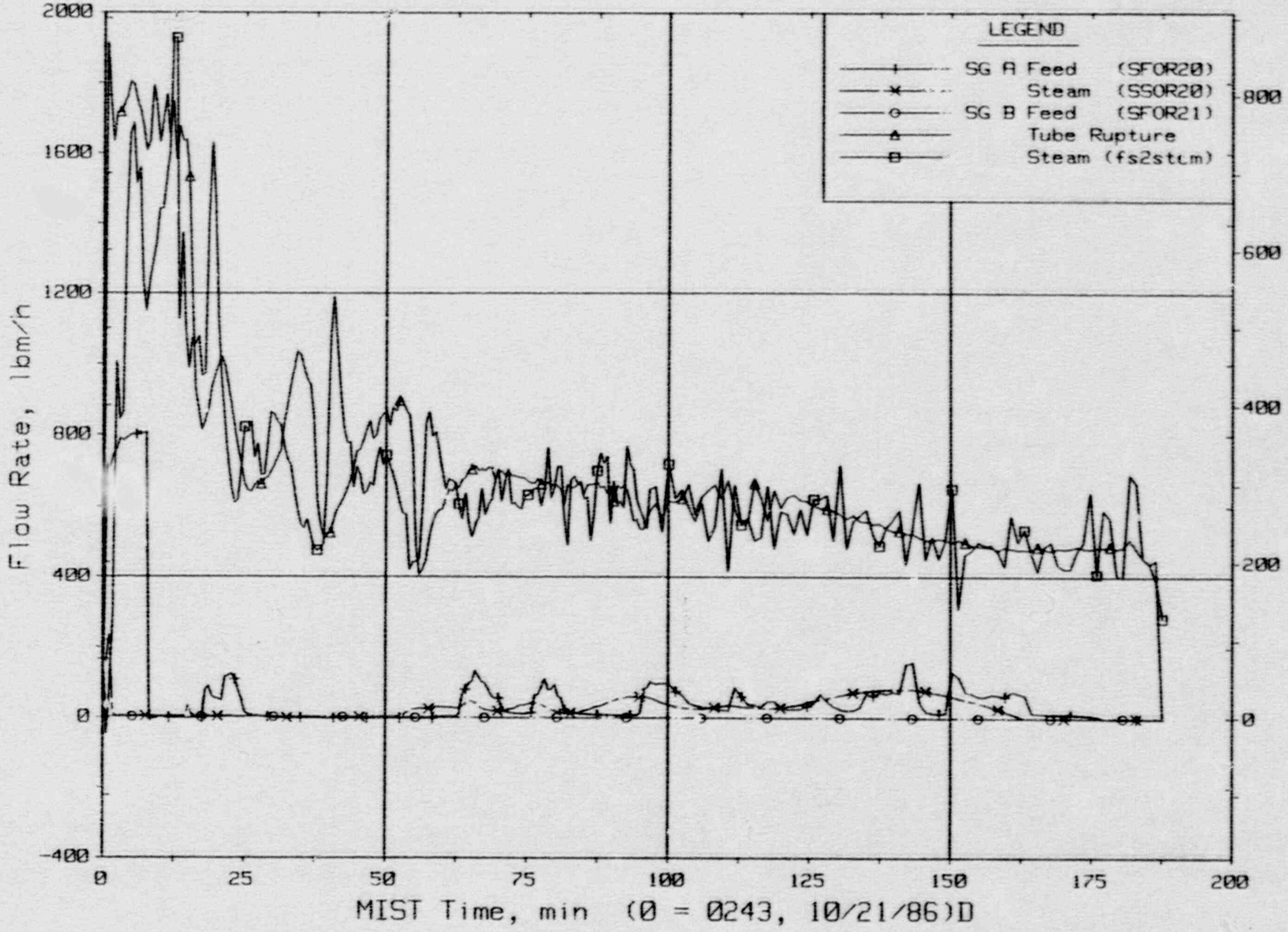
Steam Generator B Steam Exit Enthalpy.

Fri Nov 4 16:59:47 1988

S2EN1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



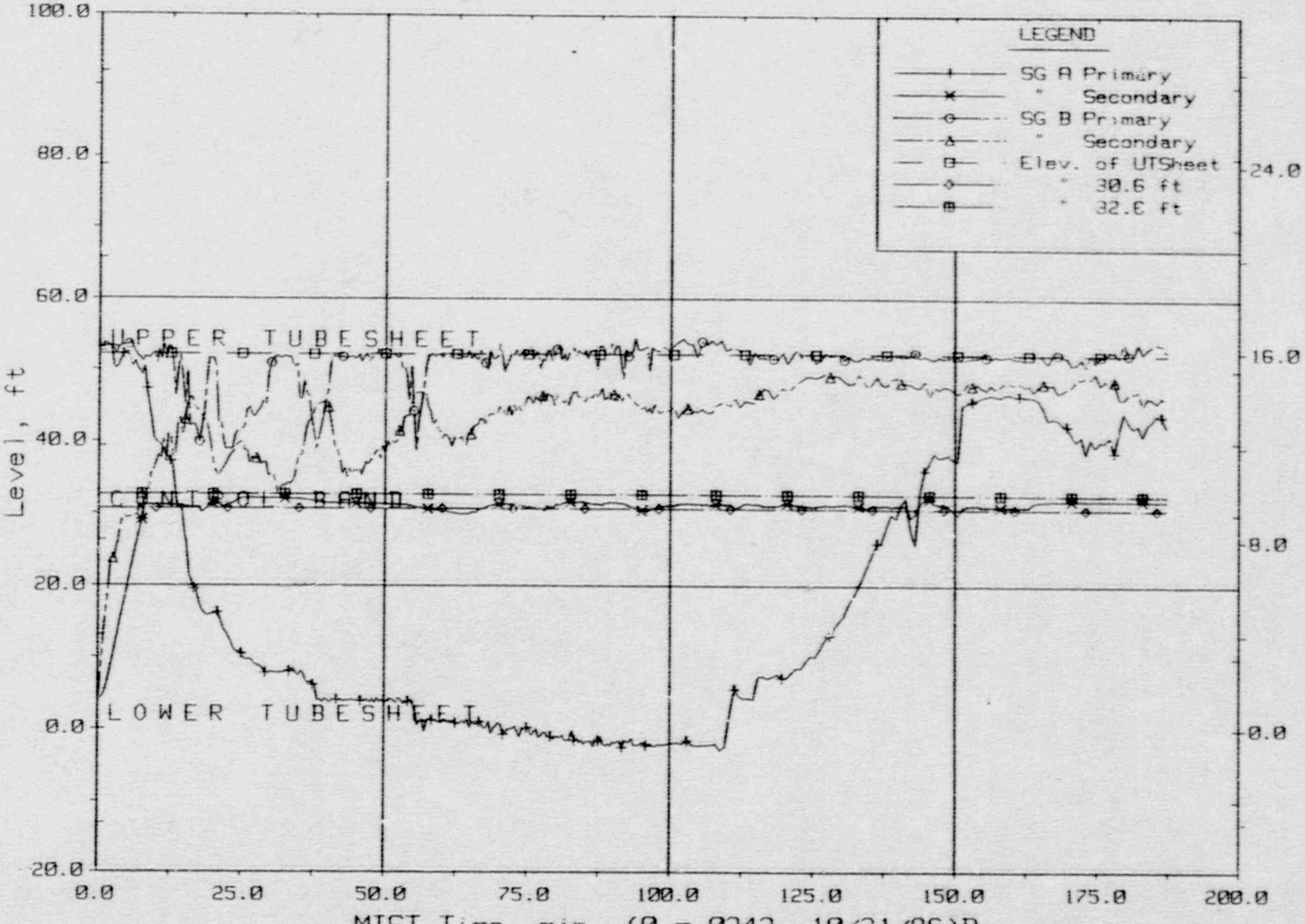
Steam Generator Secondary Flow Rates.

Fri Nov 4 16:49:25 1988

SGFL1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



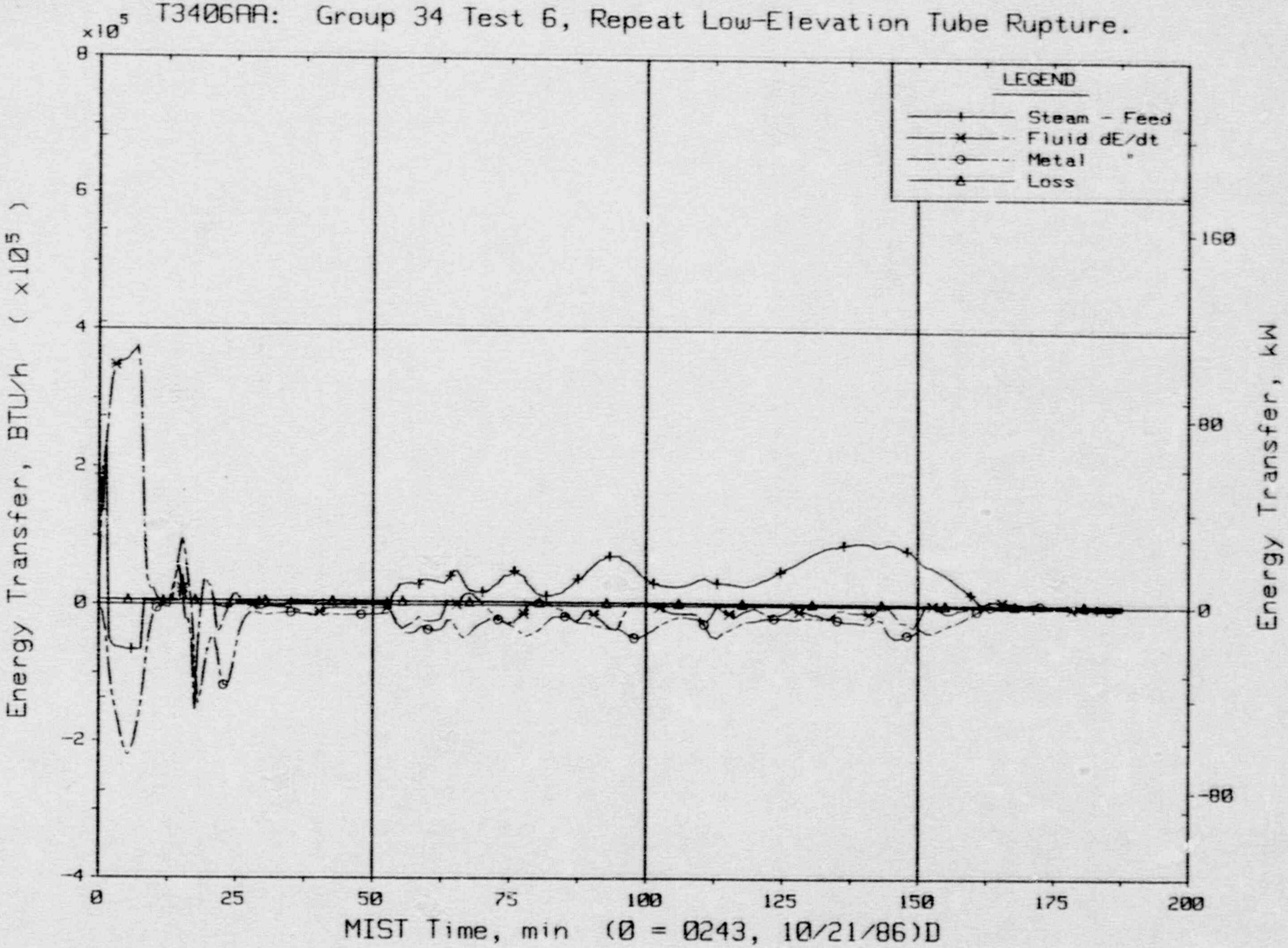
Steam Generator Collapsed Liquid Levels.

Thu Jun 2 09:45:03 1988

SGLV2

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



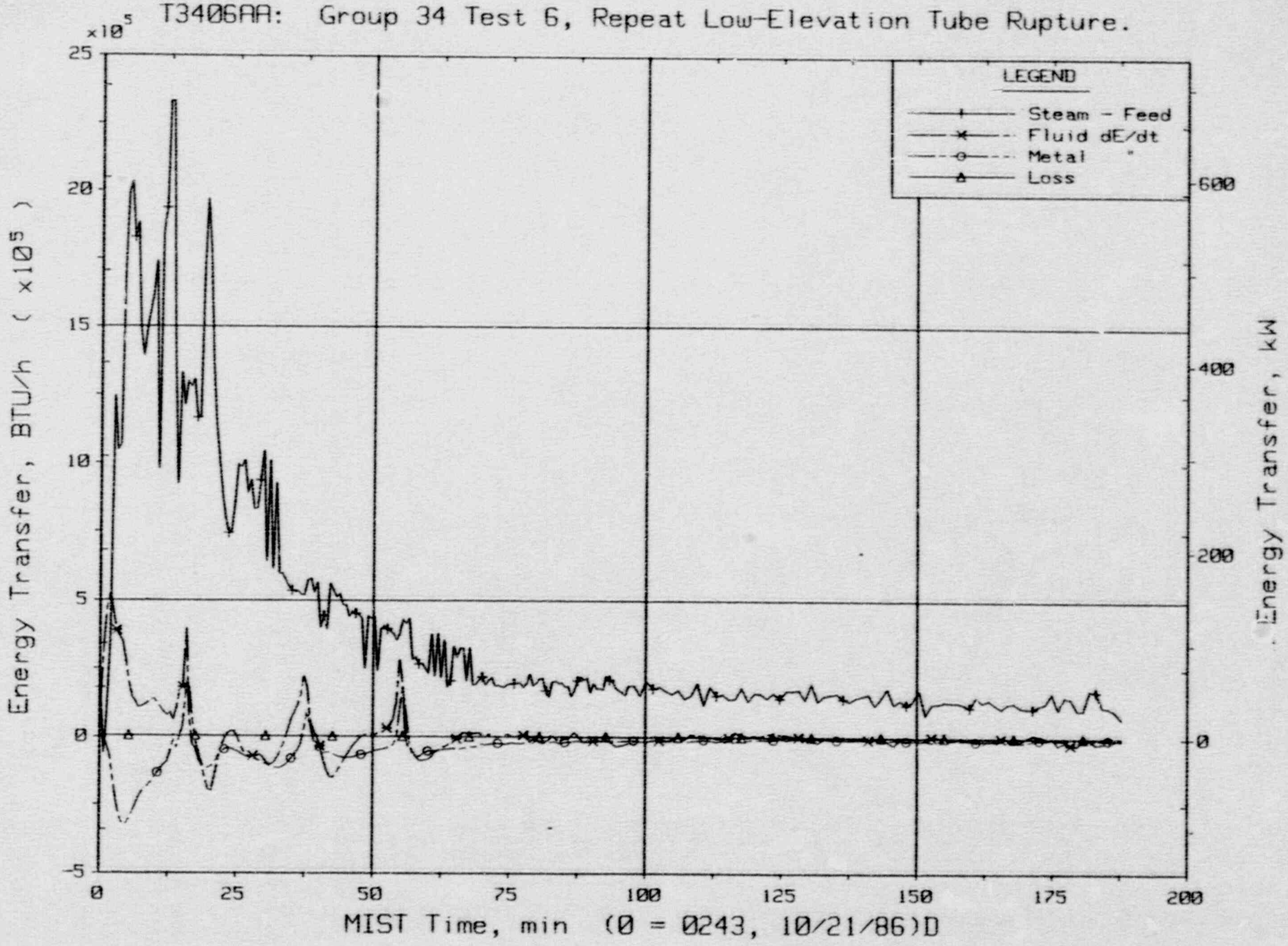
Steam Generator A Energy Transfer.

Fri Nov 4 16:52:44 1988

SGQE3

FINAL DATA

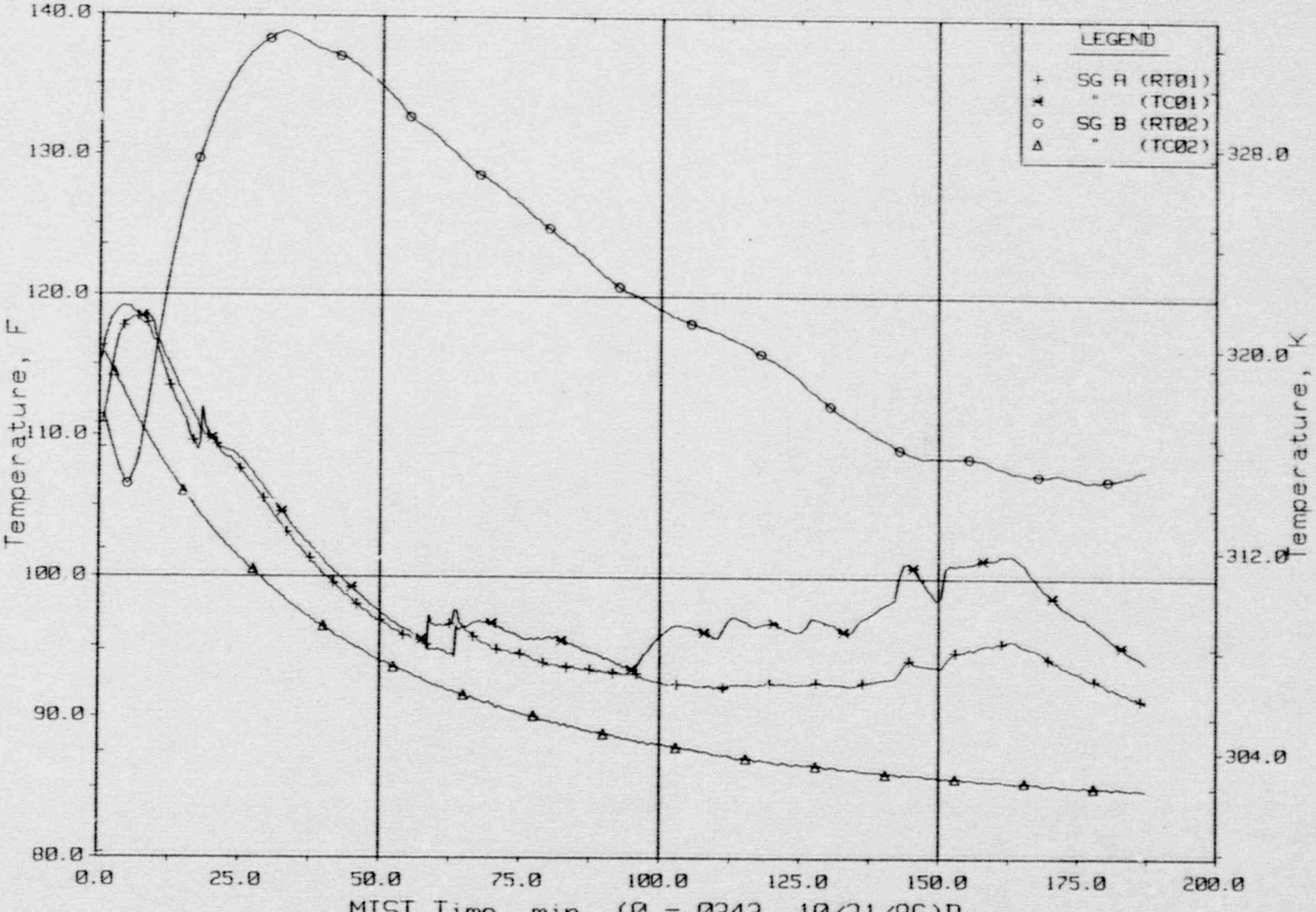
T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



Steam Generator B Energy Transfer.

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



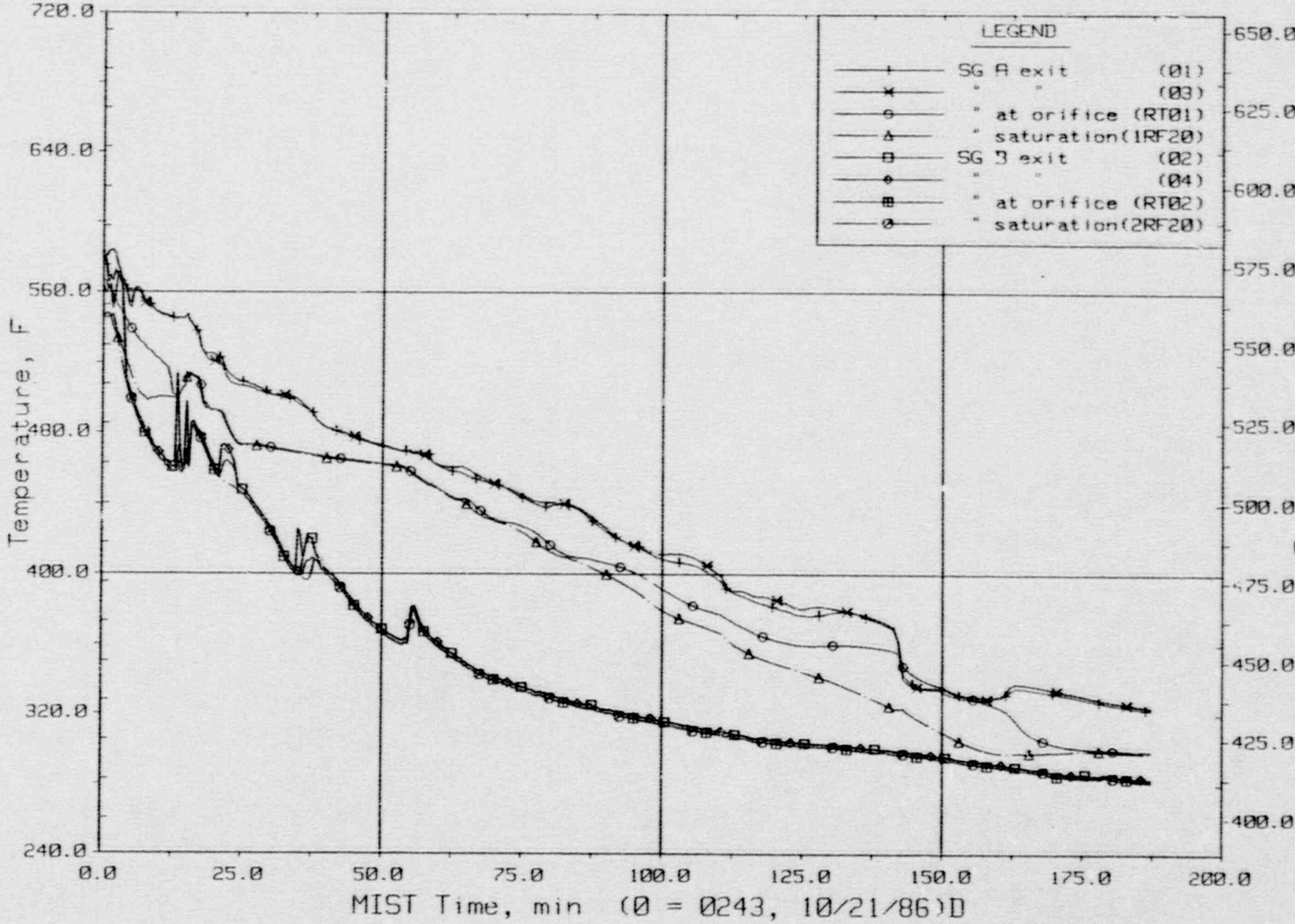
Feedwater Temperatures (SFs).

Thu Jun 2 09:57:09 1988

SGRT1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



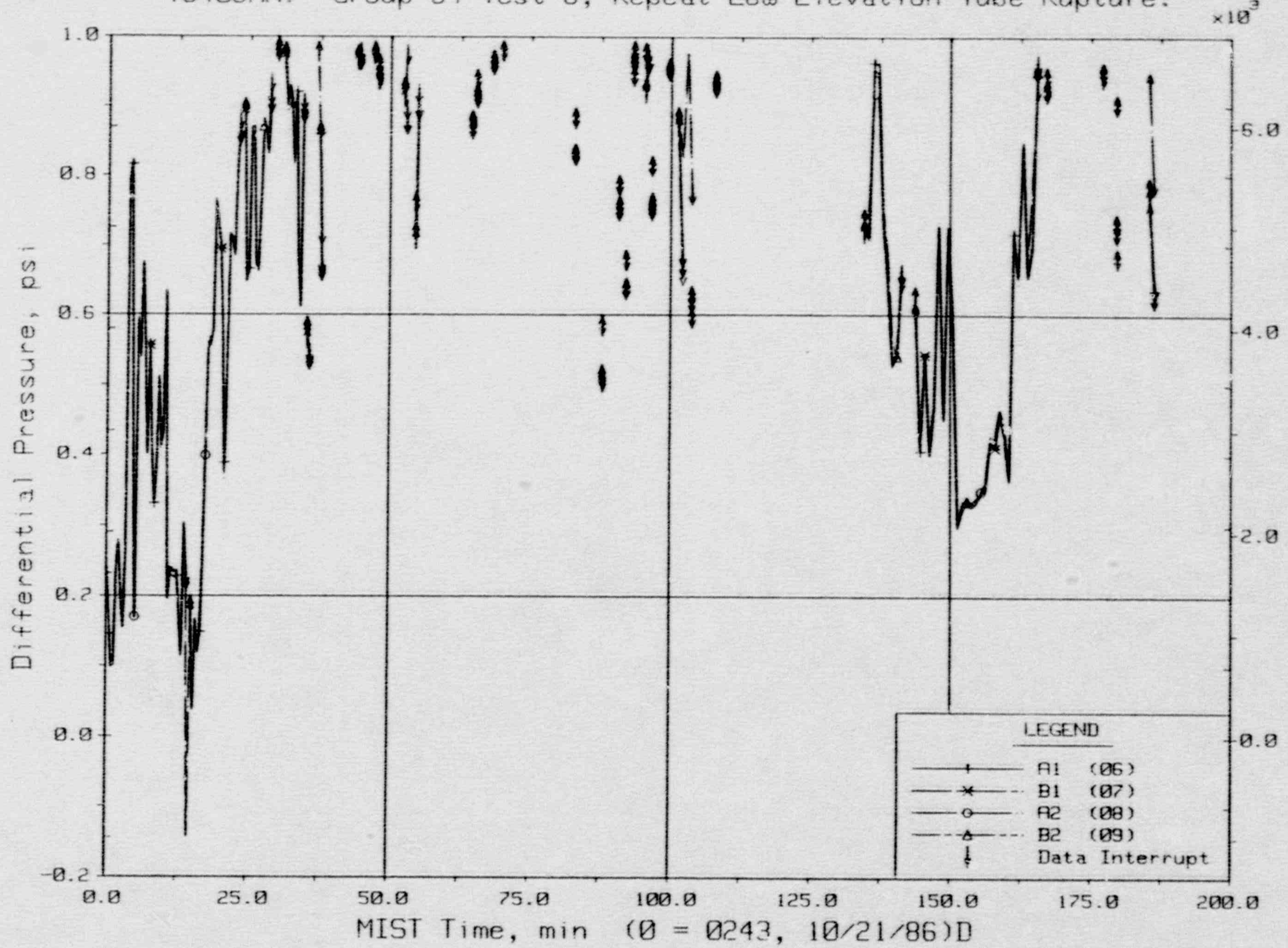
Steam Generator Steam Outlet Temperatures (SSTCs).

Thu Jun 2 10:01:38 1988

SGRT2

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



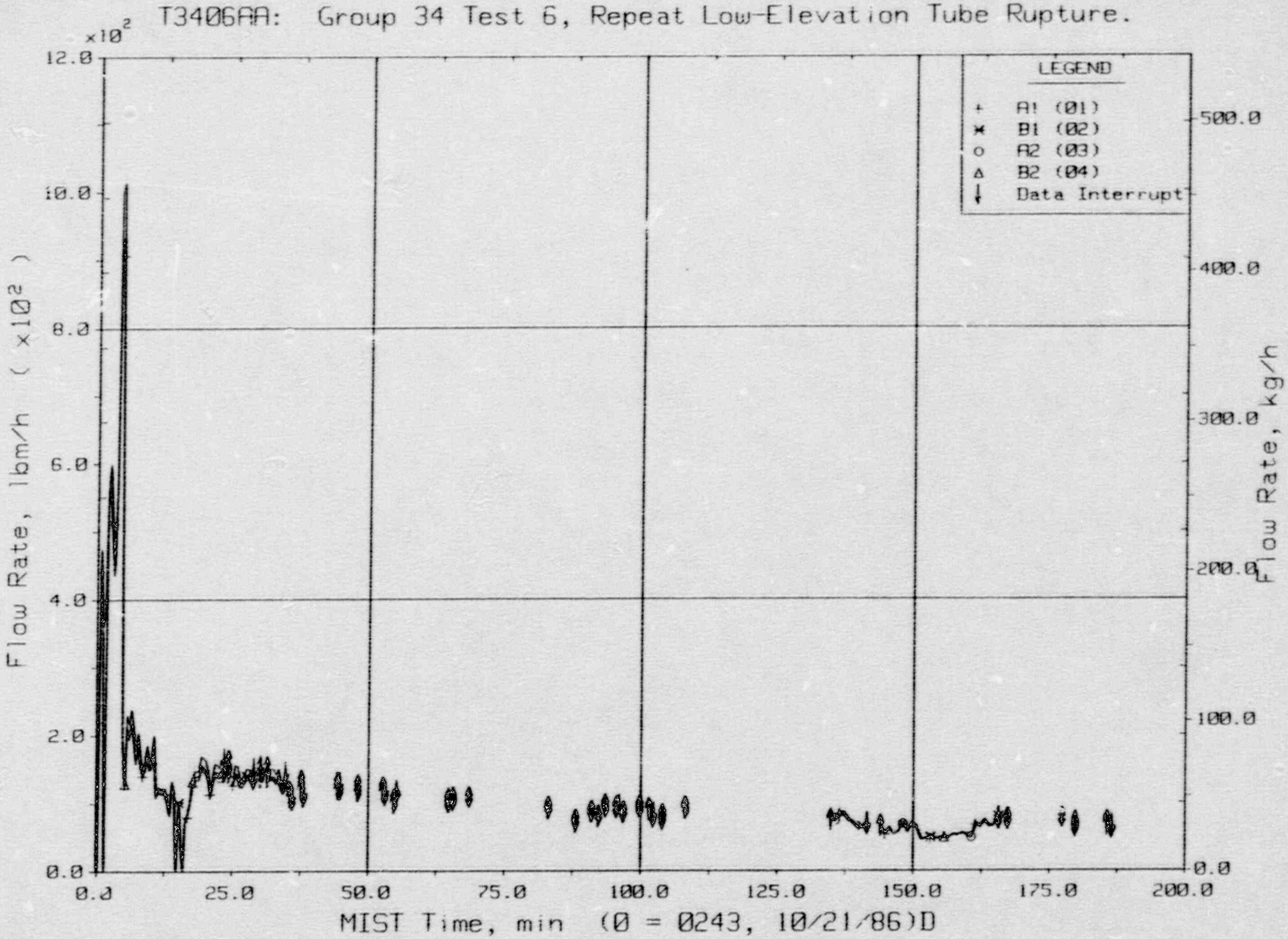
Reactor Vessel Vent Valve Differential Pressures (RVDPs).

Thu Jun 2 10:07:08 1988

VVDP1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



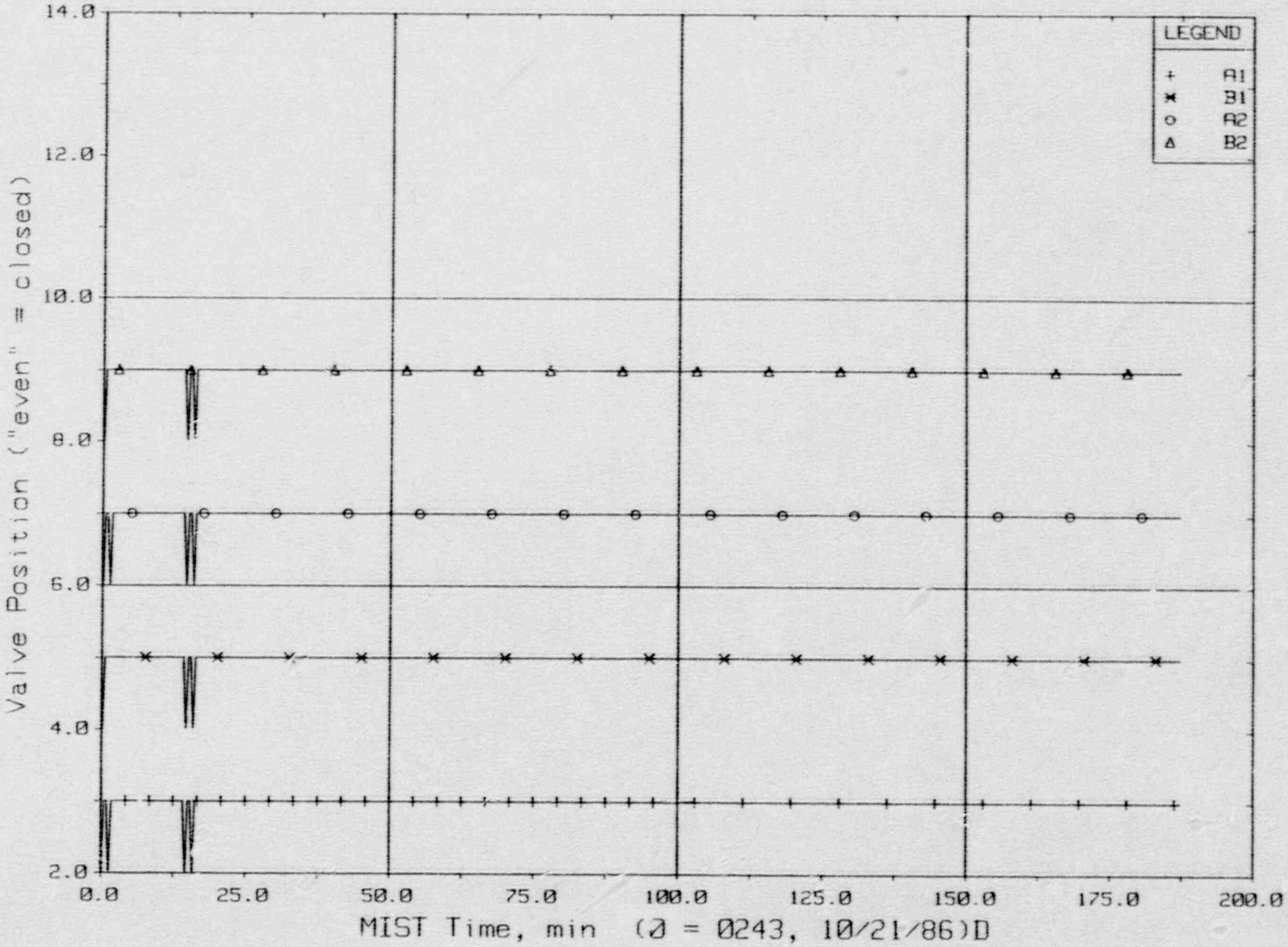
Reactor Vessel Vent Valve Flow Rates (RVVRs).

Thu Jun 2 10:10:34 1988

VVFL1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



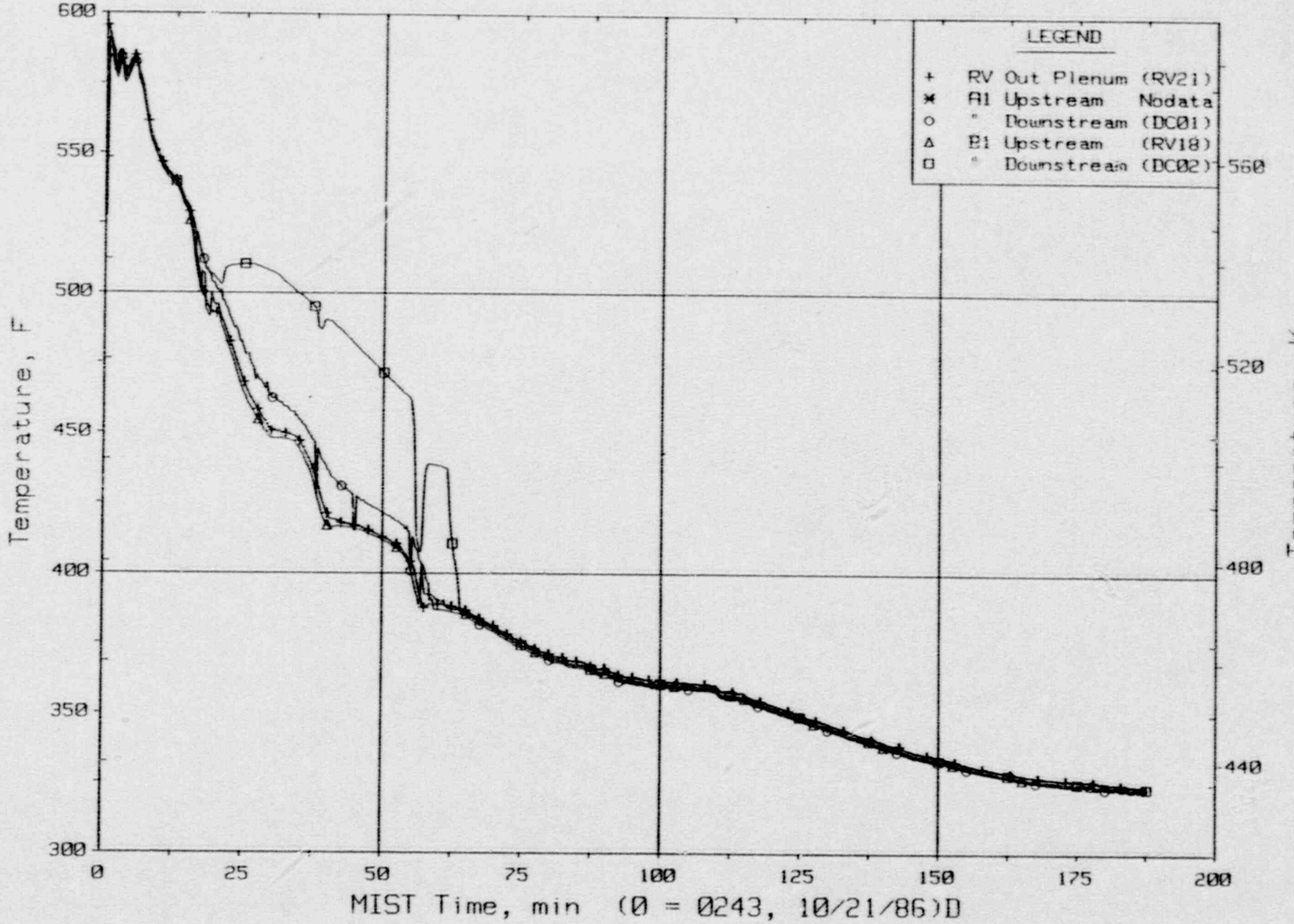
Reactor Vessel Vent Valve Positions.

Thu Jun 2 10:14:26 1988

VVLS1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.



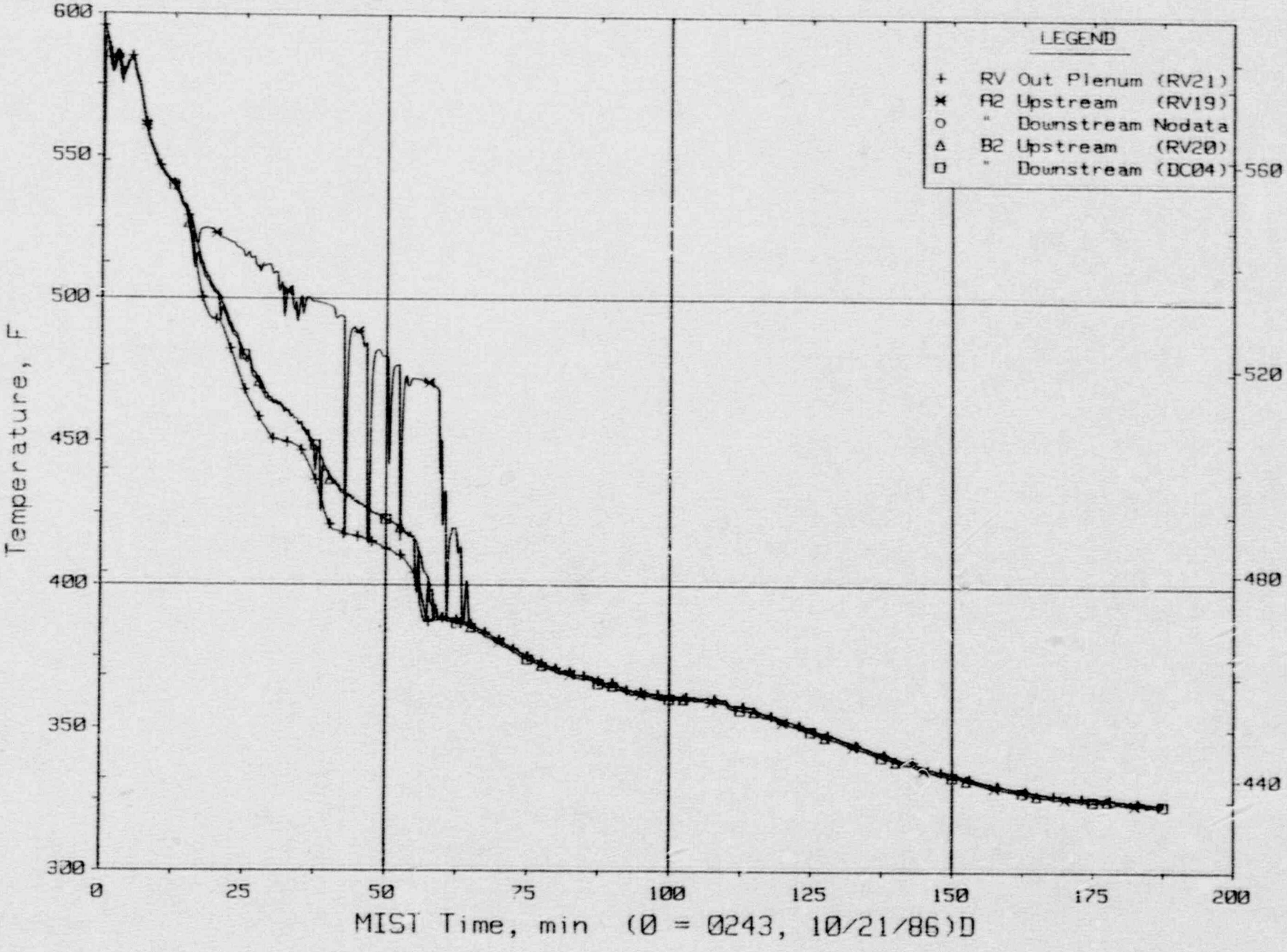
RVVVs A1 and B1 Bracketing Fluid Temperatures (TCs).

Tue Nov 1 15:16:14 1988

VVTC1

FINAL DATA

T3406AA: Group 34 Test 6, Repeat Low-Elevation Tube Rupture.

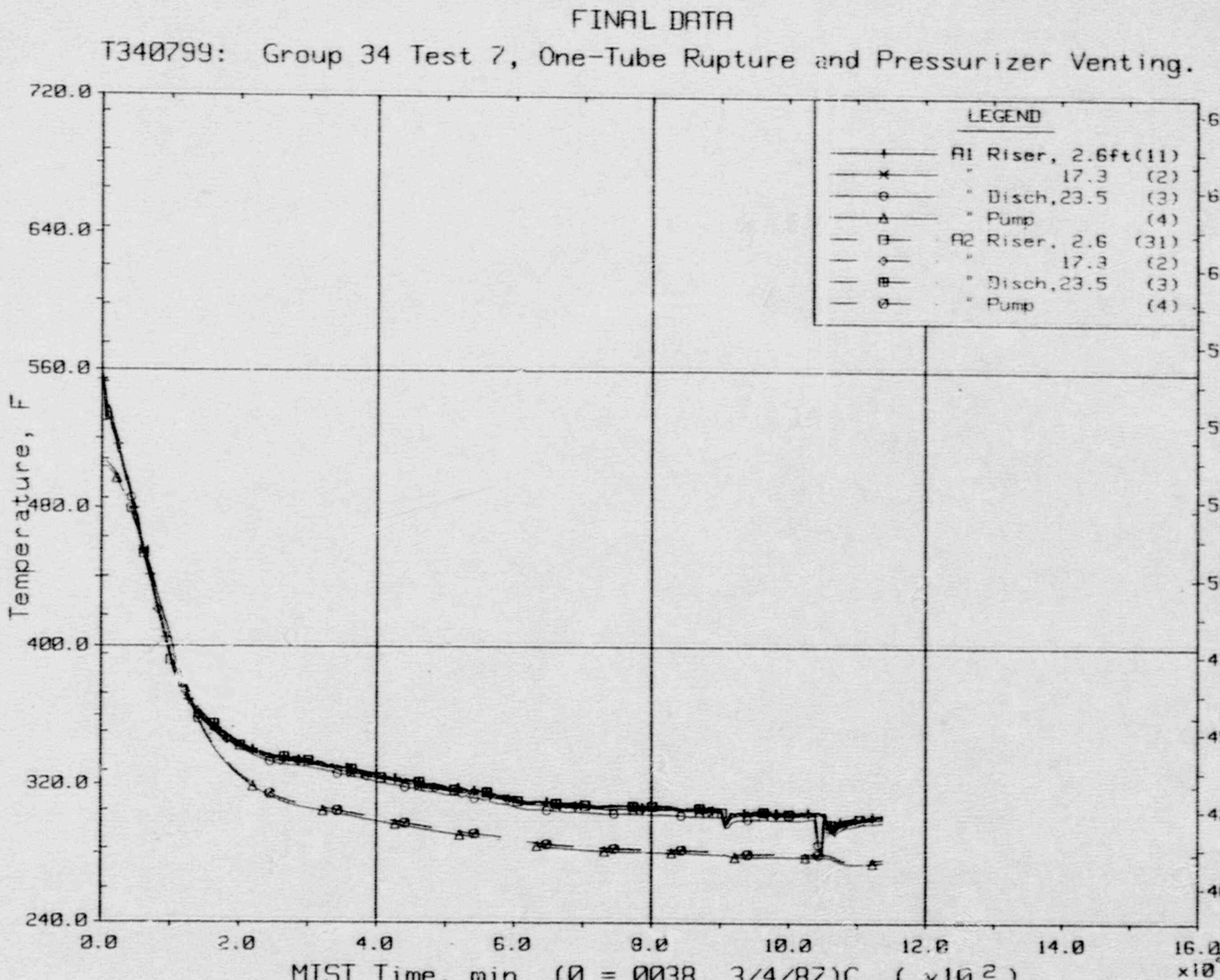


RVVVs A2 and B2 Bracketing Fluid Temperatures (TCs).

Tue Nov 1 15:19:49 1988

VVTC2

T340799



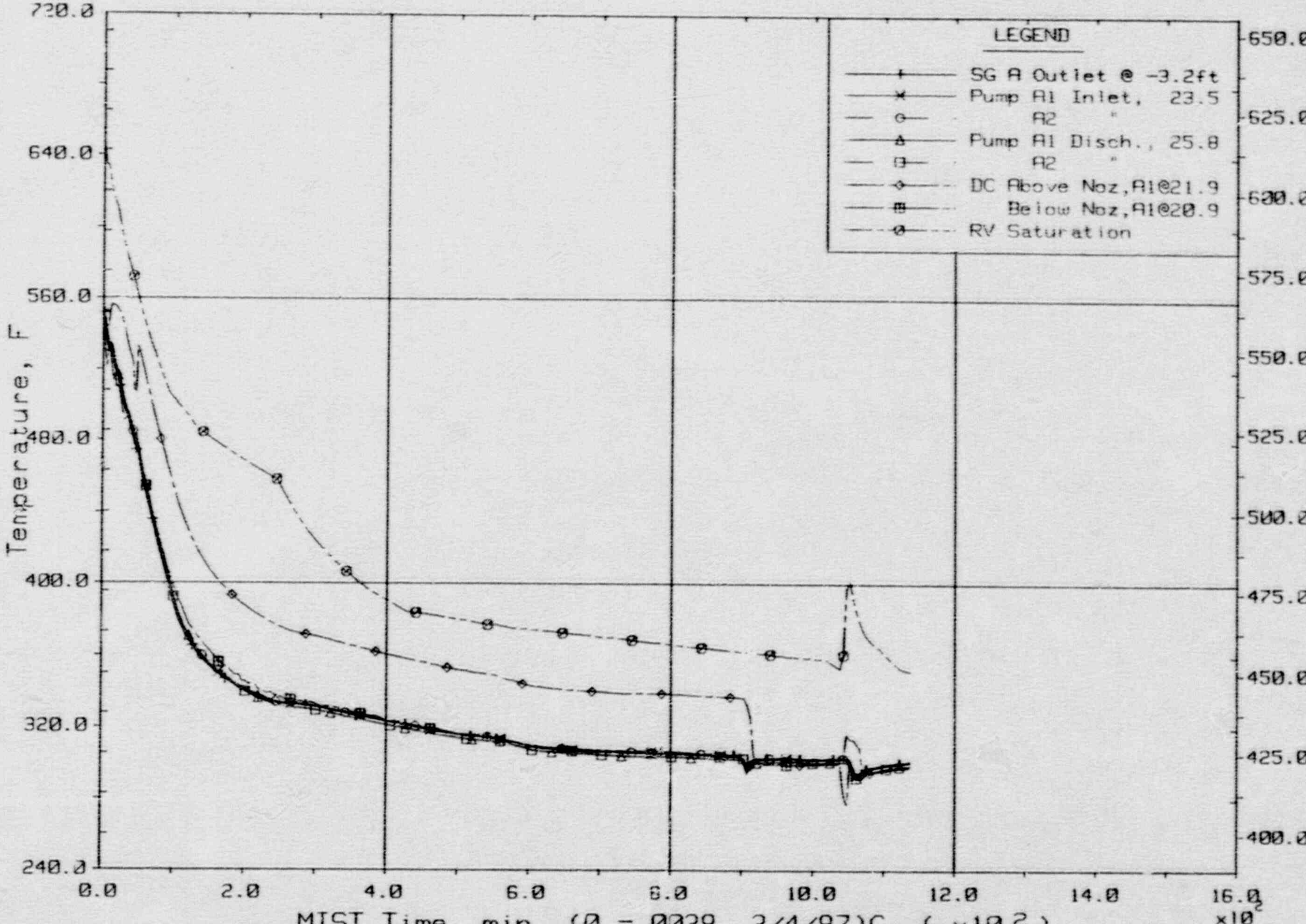
Loop A Cold Leg Metal Temperatures (C1,3MTs).

Thu Jun 2 10:09:16 1988

C1MT1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



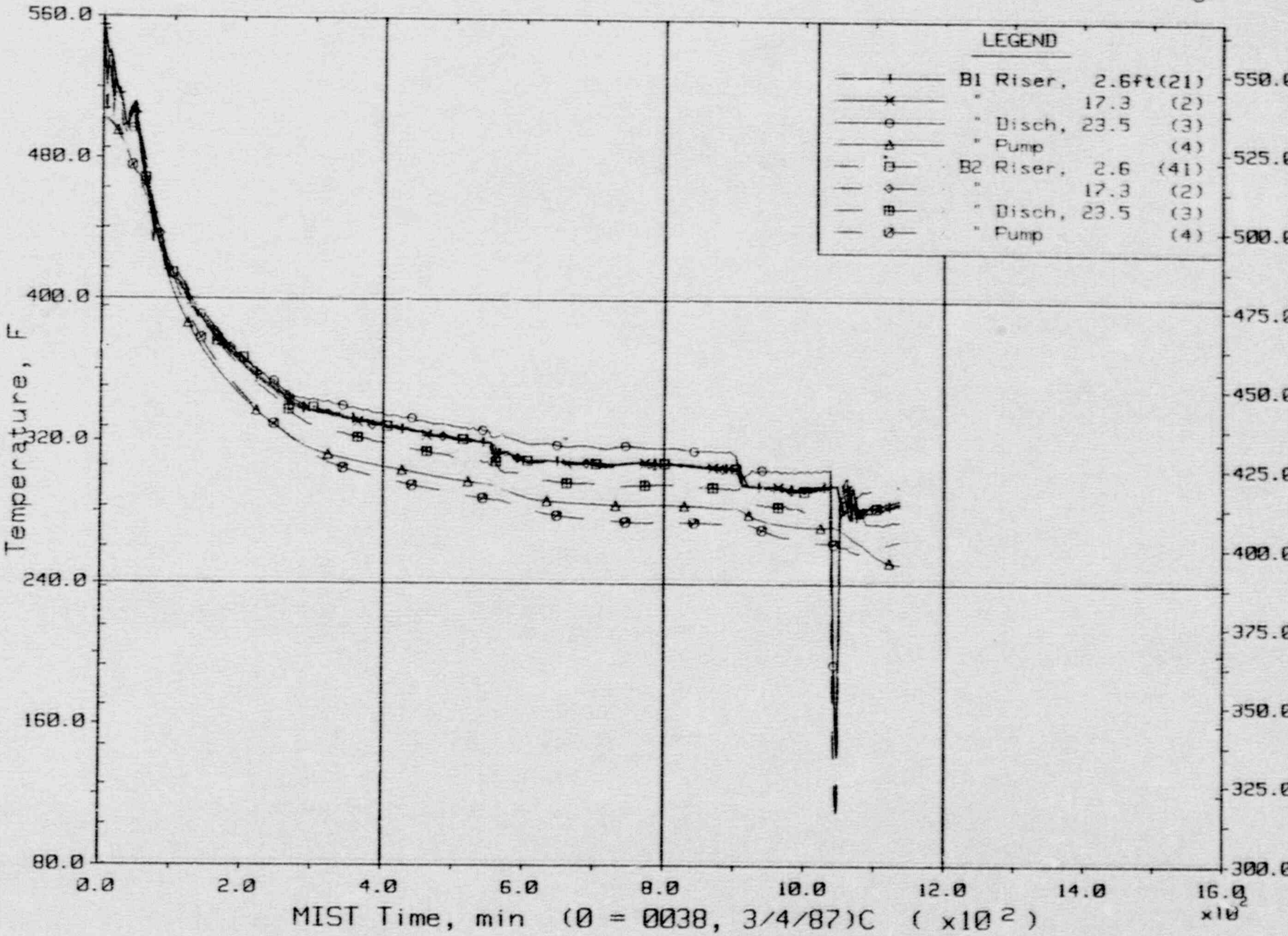
Loop A Cold Leg Fluid Temperatures (RTDs).

Thu Jun 2 10:13:11 1988

C1RT0

FINAL DATA

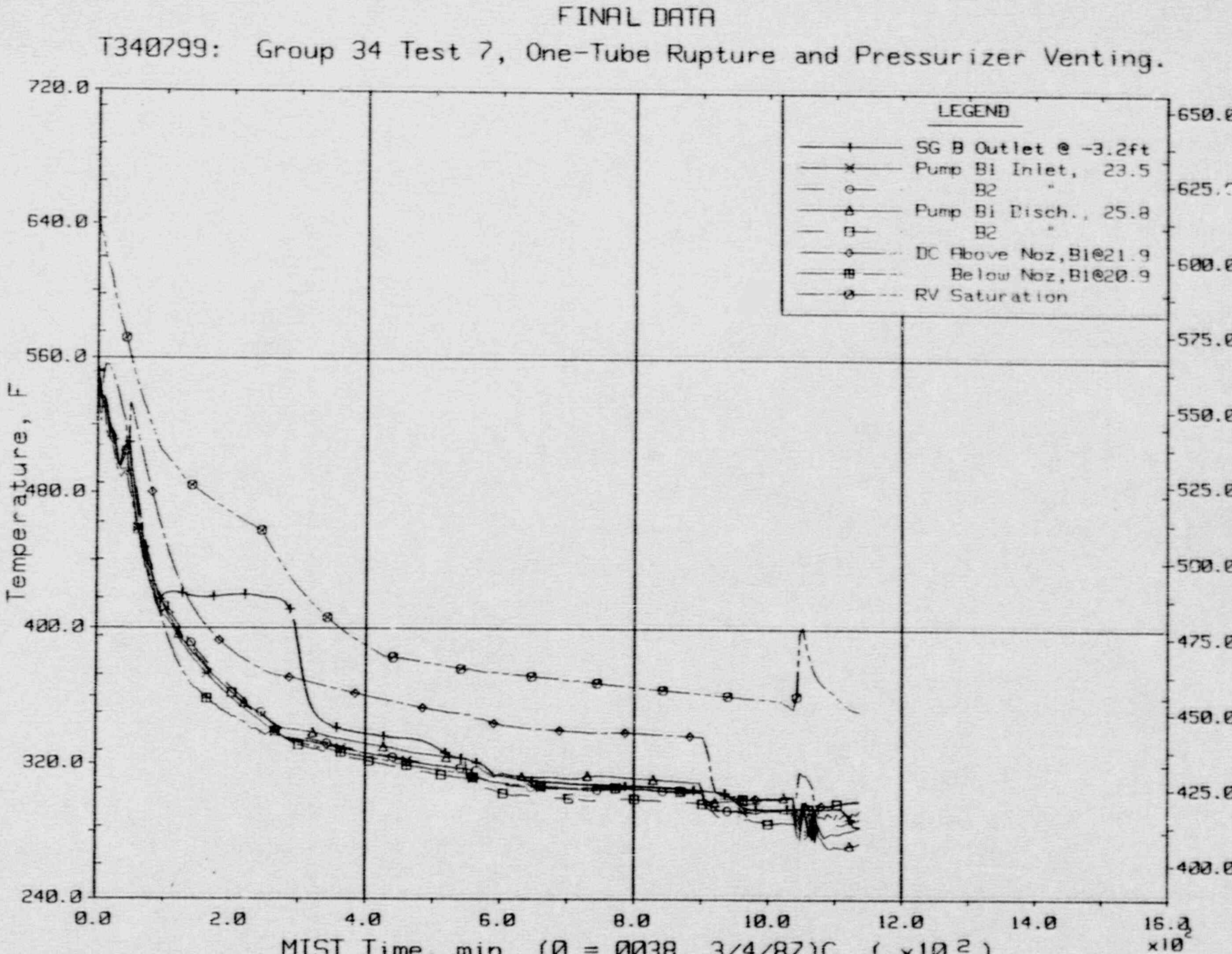
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Loop B Cold Leg Metal Temperatures (C2,4MTs).

Thu Jun 2 10:16:54 1988

C2MT1



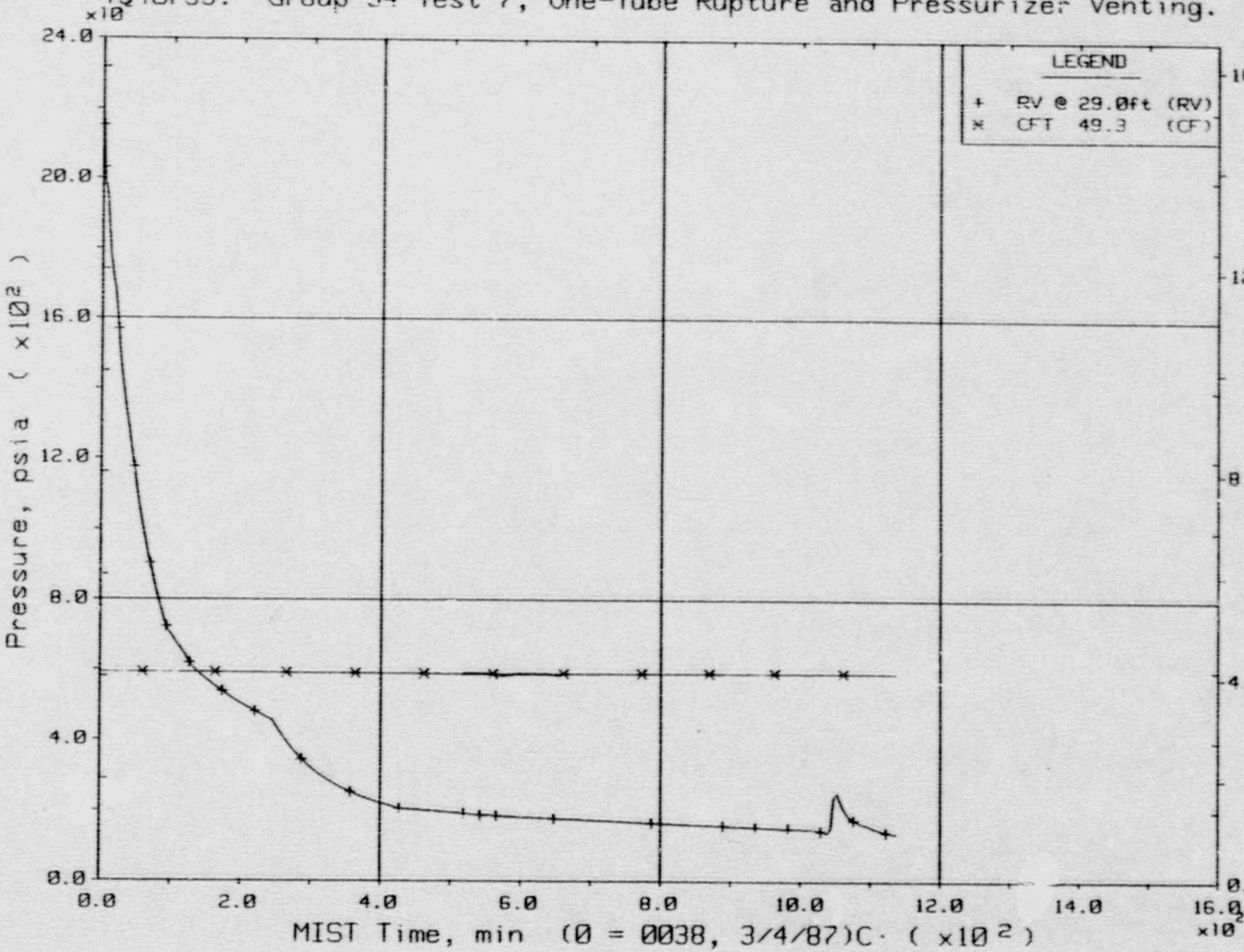
Loop B Cold Leg Fluid Temperatures (RTDs).

Thu Jun 2 10:20:38 1988

C2RT0

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



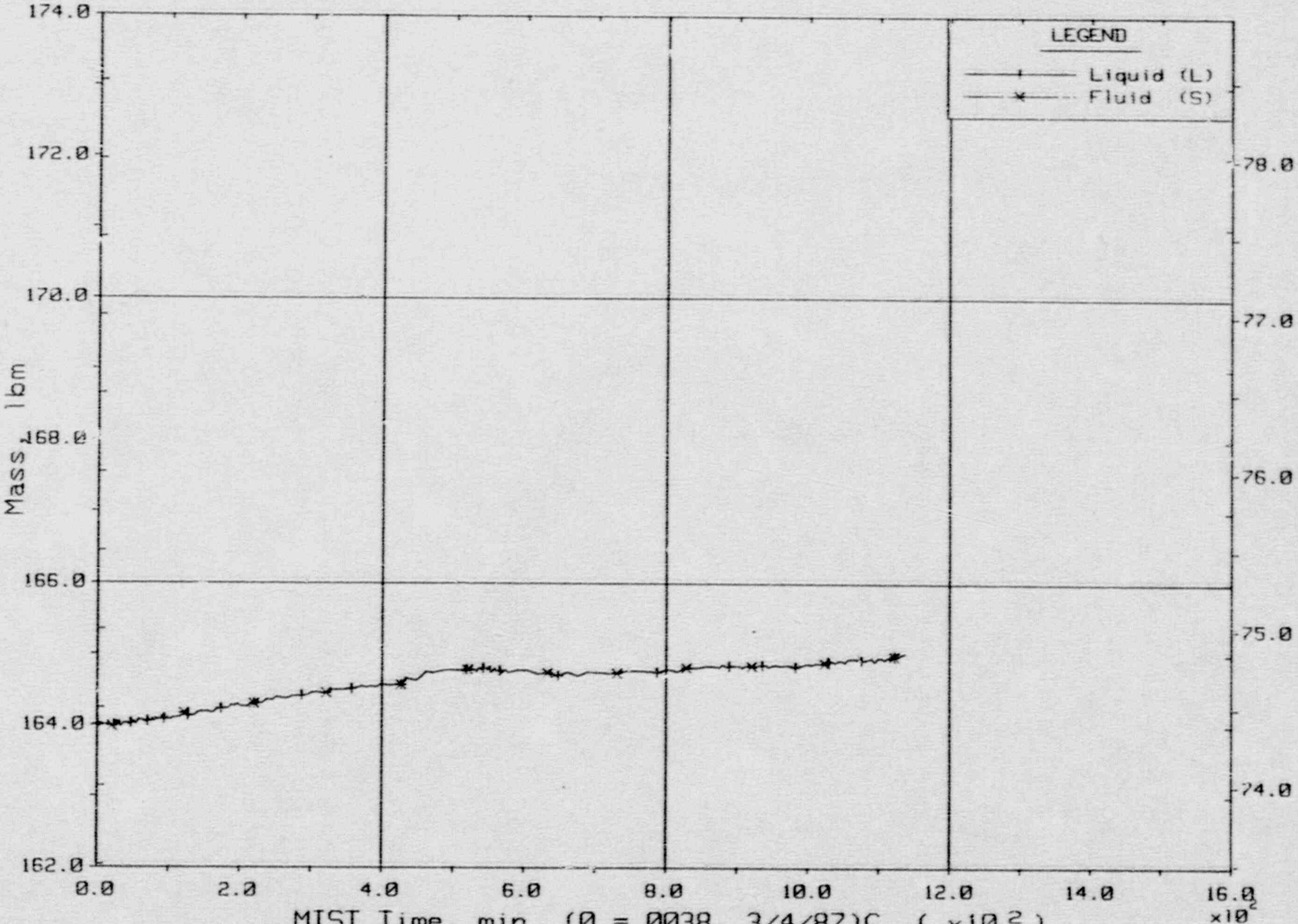
Primary System and Core Flood Tank Pressures (GP01s).

Thu Jun 2 10:23:02 1988

CFGP1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



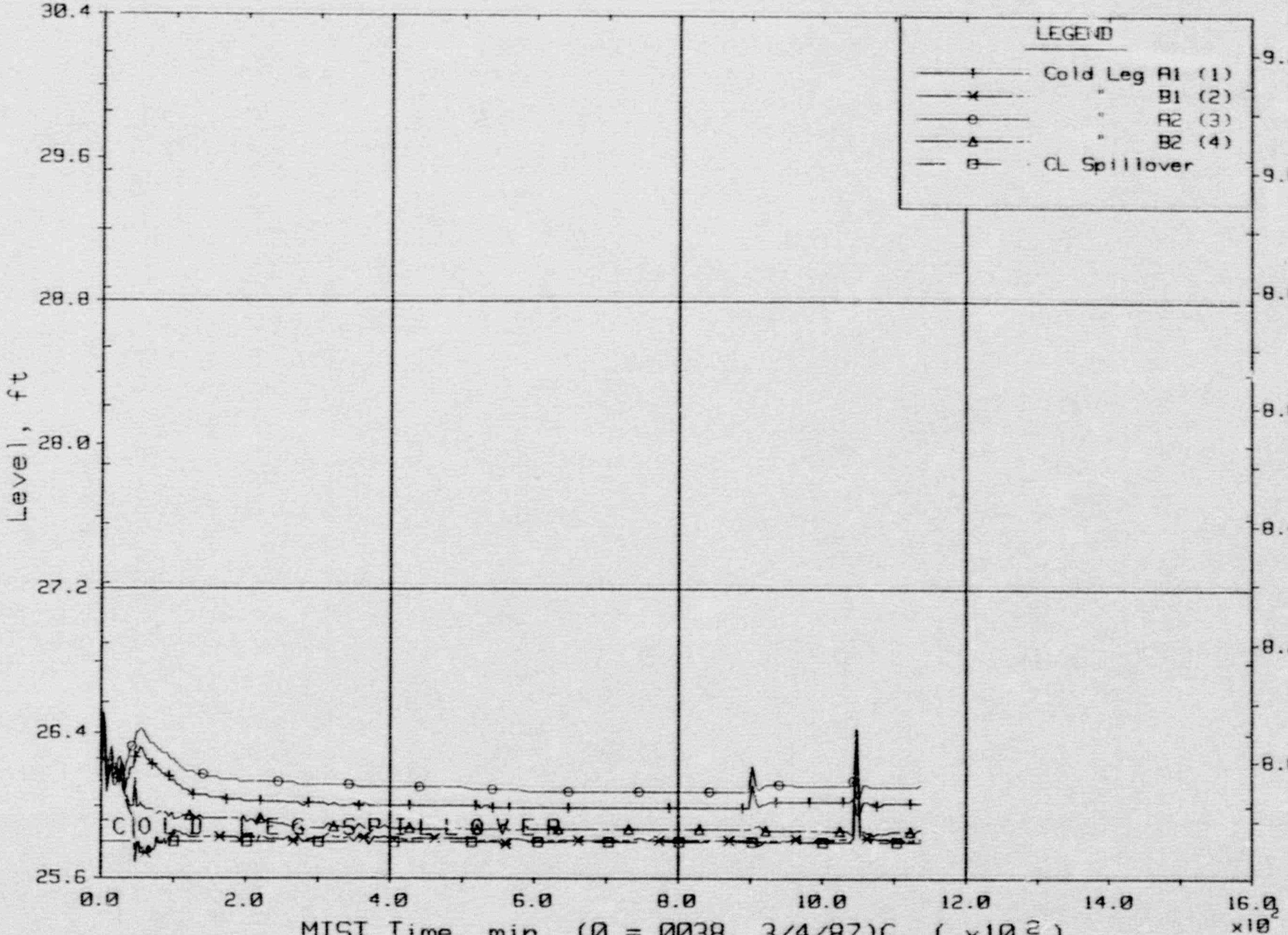
Core Flood Tank Liquid ahd Fluid Mass (CFMa20s).

Thu Jun 2 19:25:01 1988

CFML1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



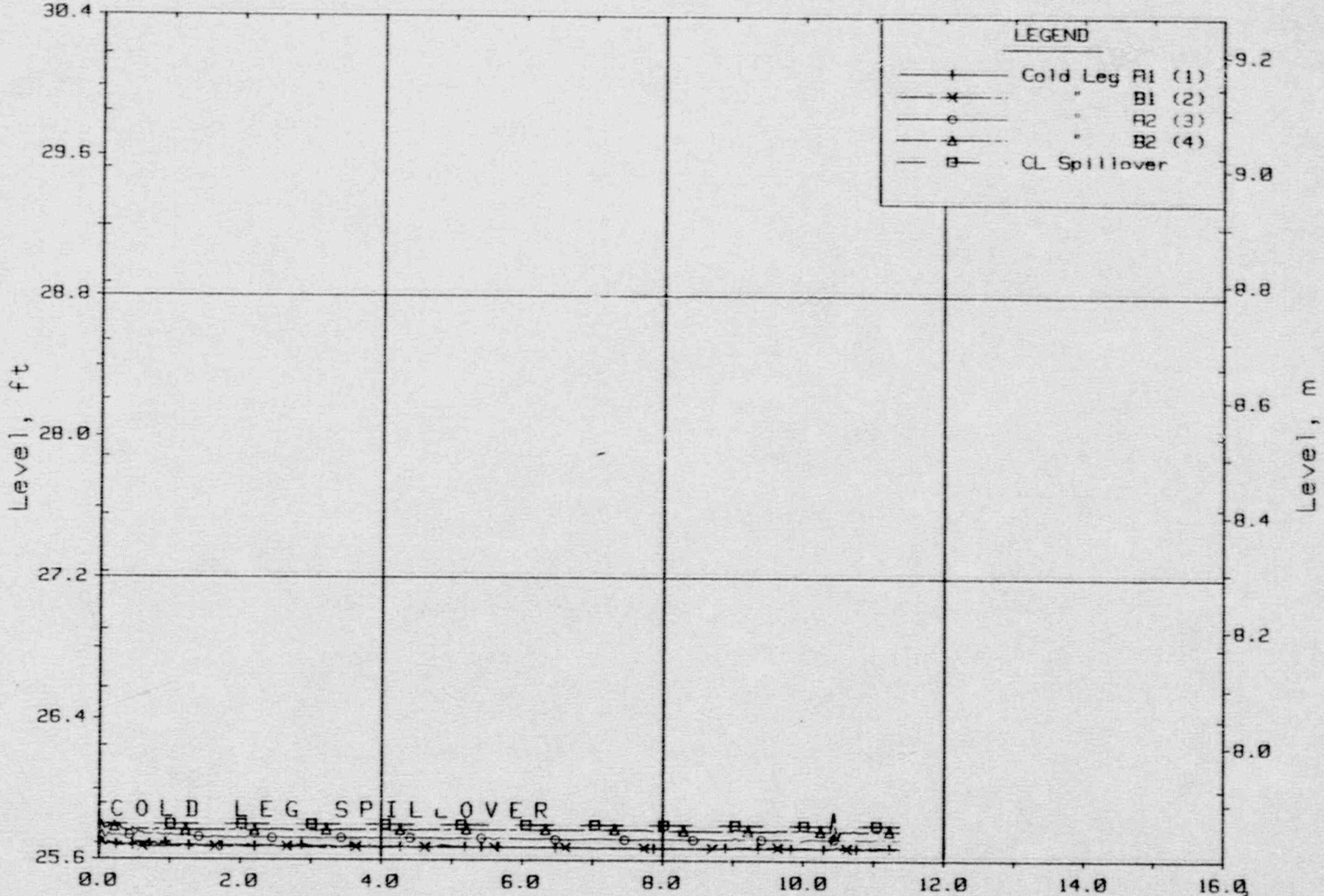
Cold Leg Suction Collapsed Liquid Levels (CnLV22s).

Thu Jun 2 10:30:18 1988

CLLV4

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



MIST Time, min (0 = 0038, 3/4/87)C (x10 ²)

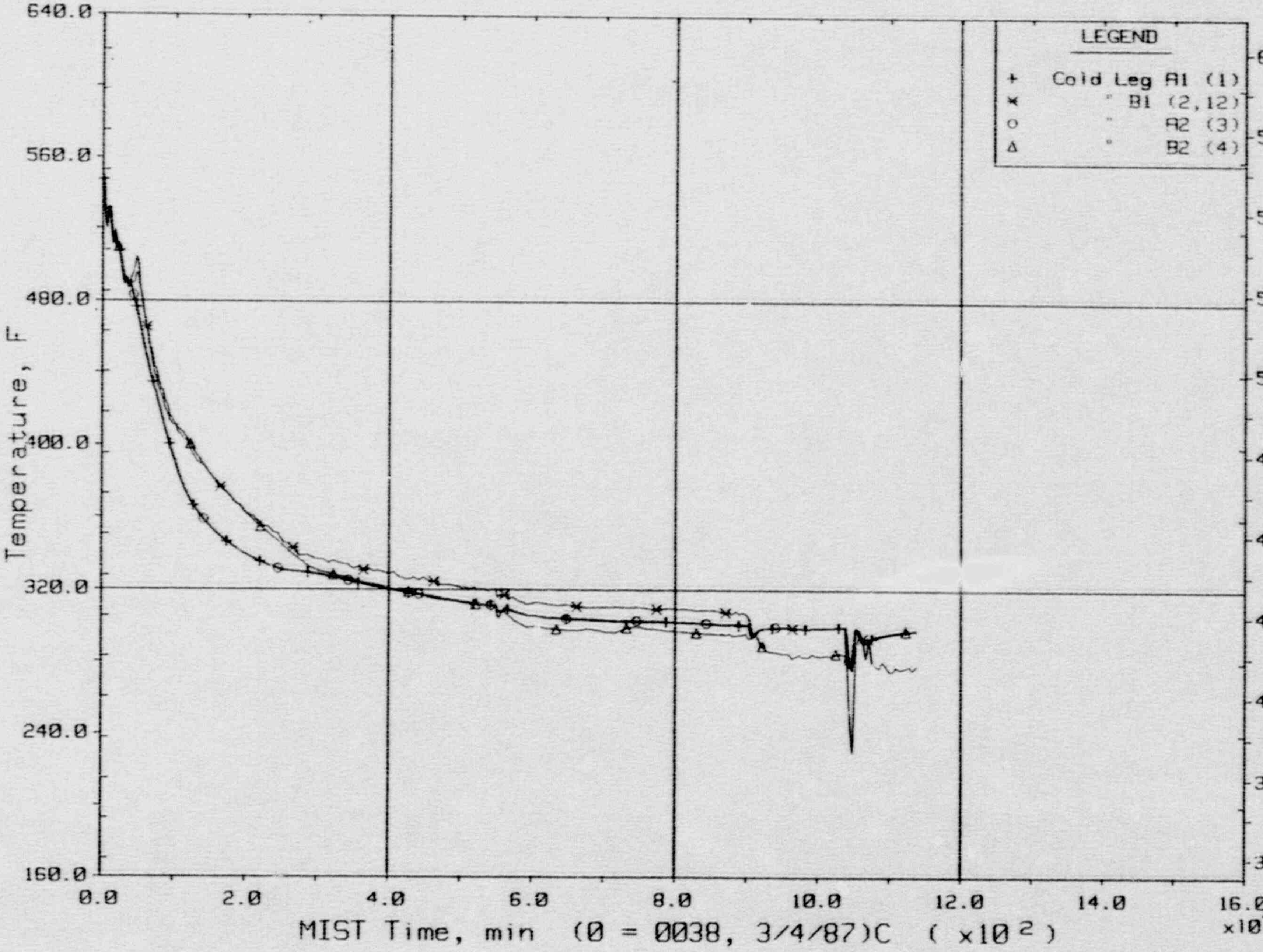
Cold Leg Discharge Collapsed Liquid Levels (CnLV23s).

Thu Jun 2 10:32:51 1988

CLLV5

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



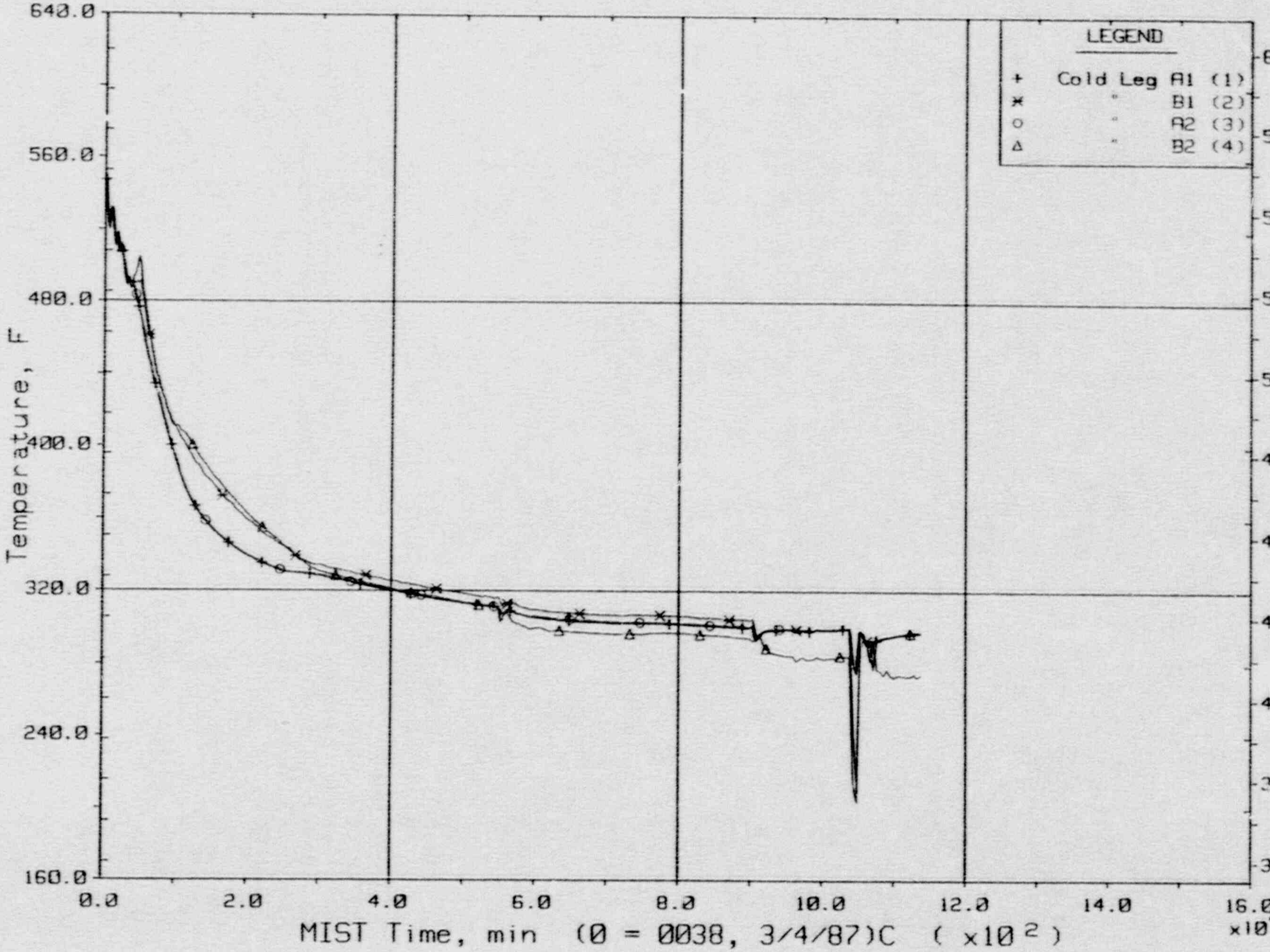
Cold Leg Nozzle Fluid Temperatures, Top of Rake (21.3ft, CnTC11s).

Thu Jun 2 16:37:03 1988

CLTC1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



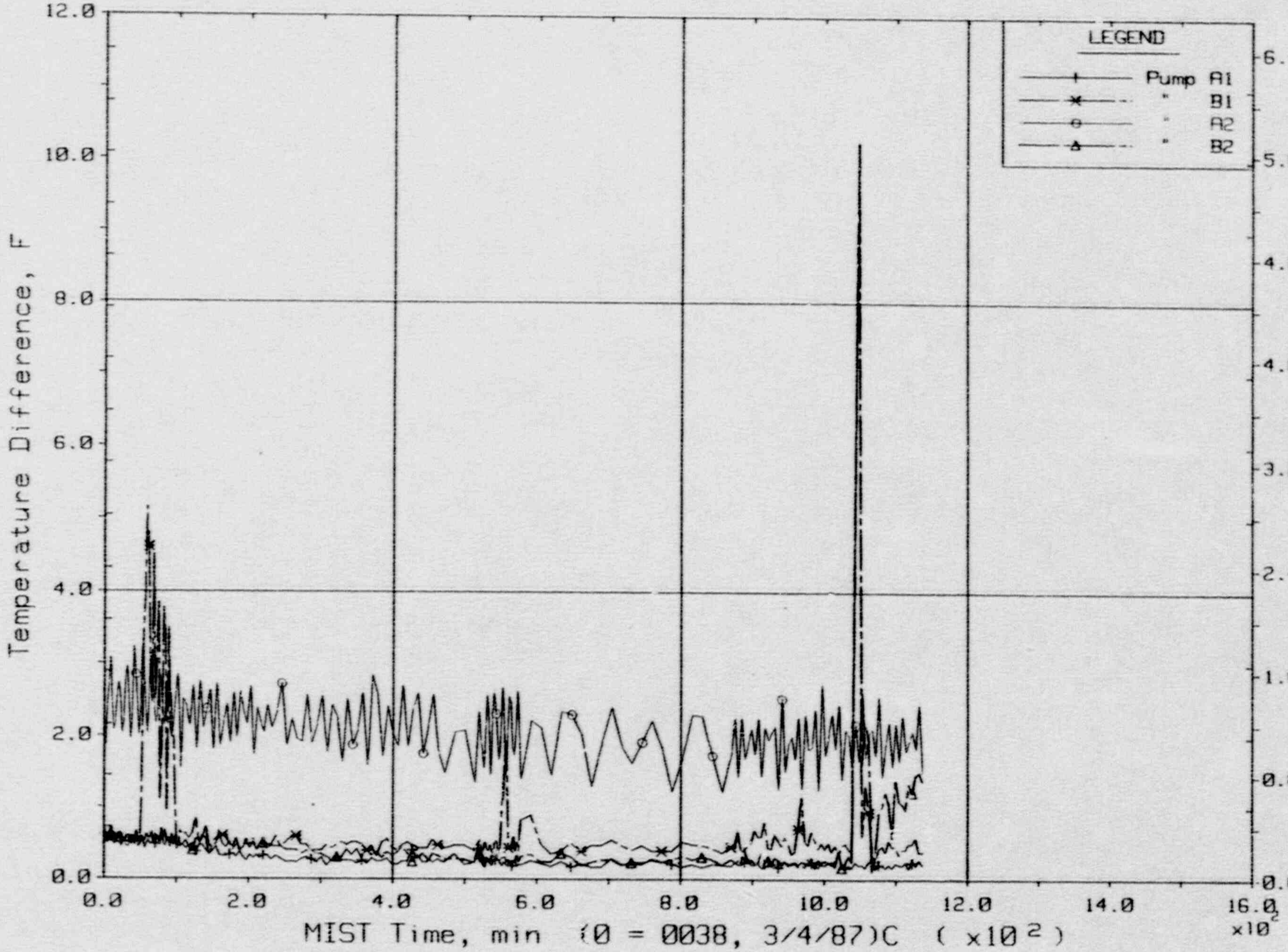
Cold Leg Nozzle Fluid Temperatures, Bottom of Rake (21.2ft, CnTC14s).

Thu Jun 2 10:39:23 1988

CLTC2

FINAL DATA

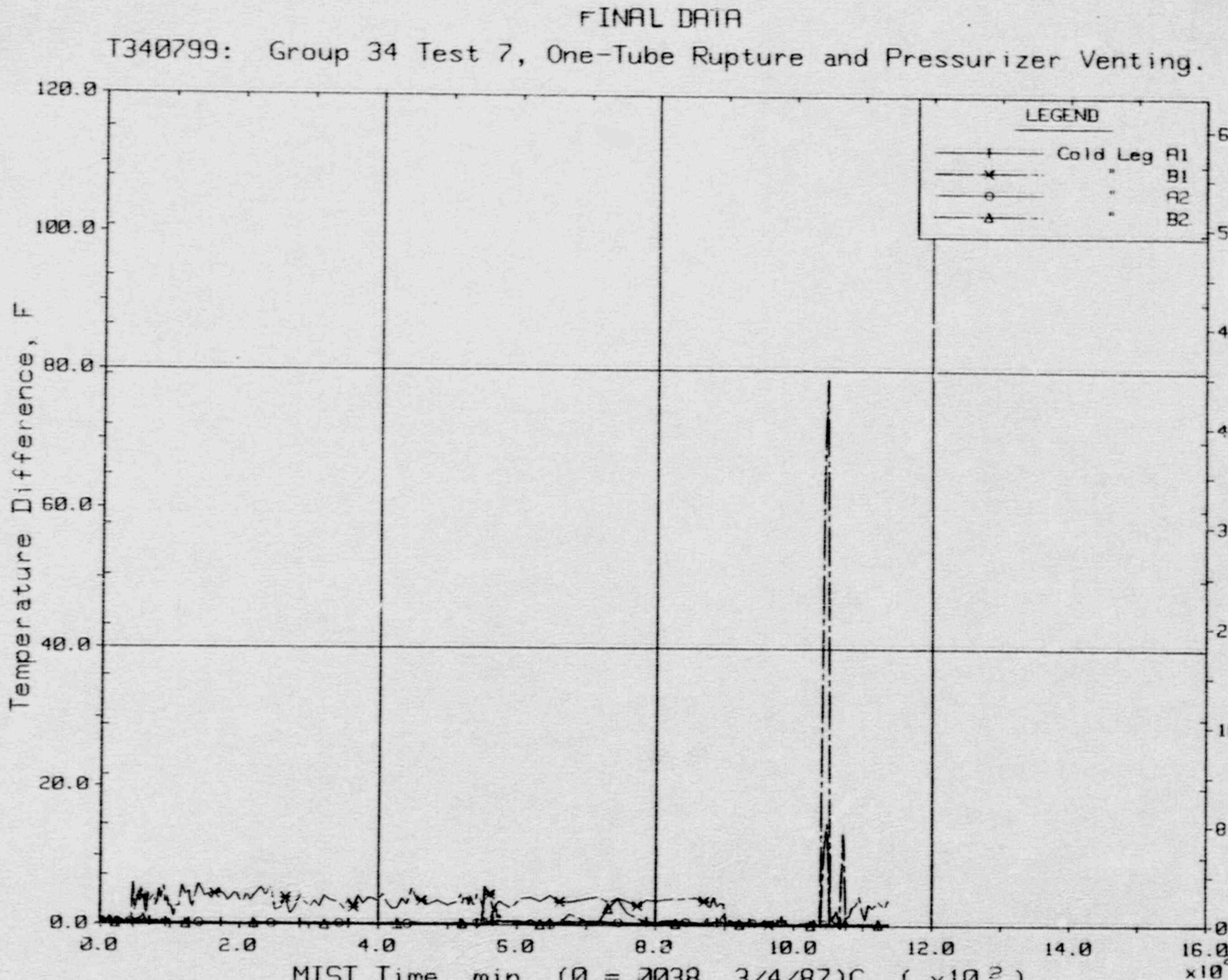
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Maximum Differences Among RCP Rake FLuid Temperatures.

Thu Jun 2 10:42:00 1988

CLTD1



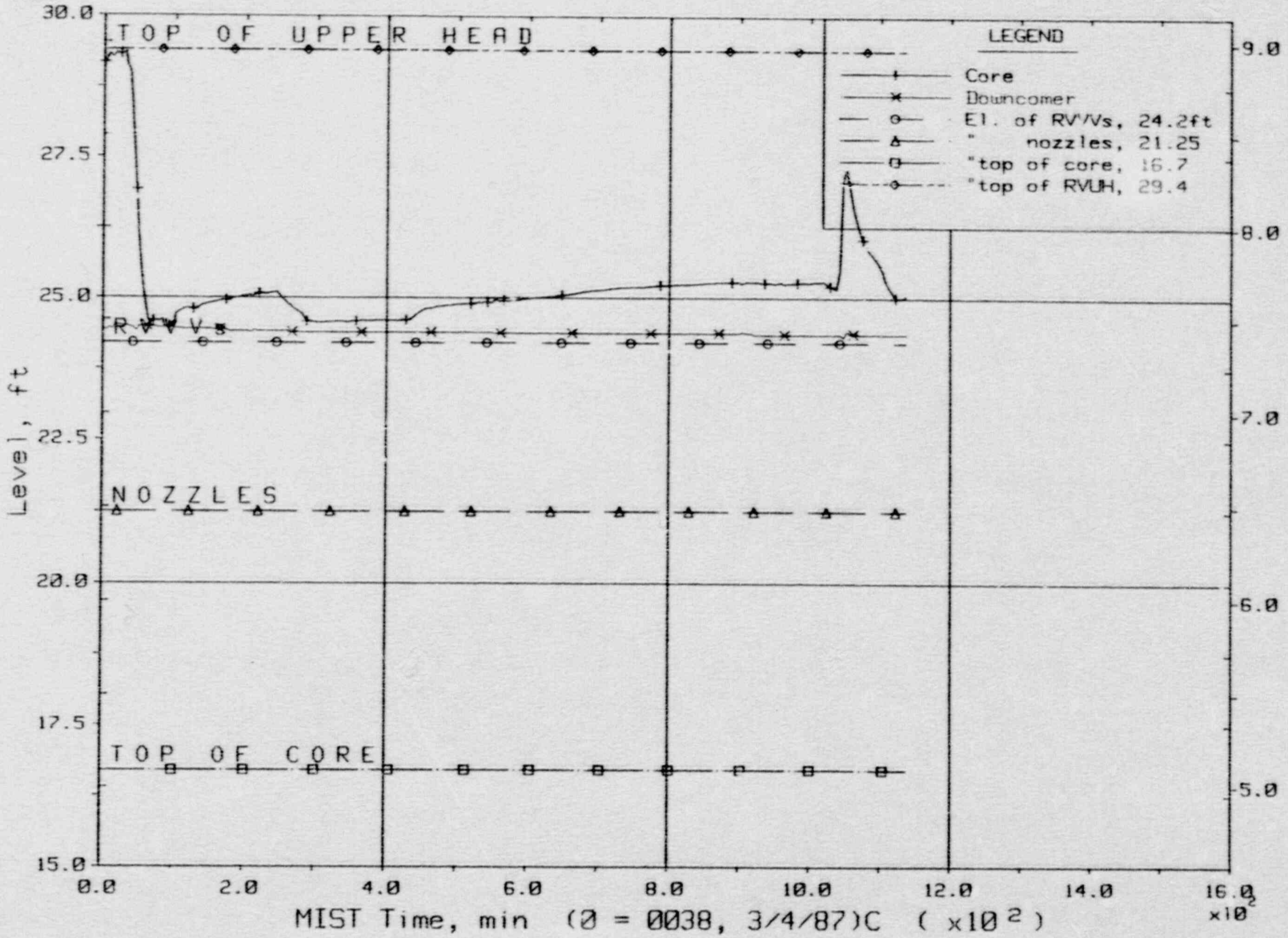
Maximum Differences Among CL Nozzle Rake Fluid Temperatures.

Thu Jun 2 10:44:34 1988

CLTD2

FINAl DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



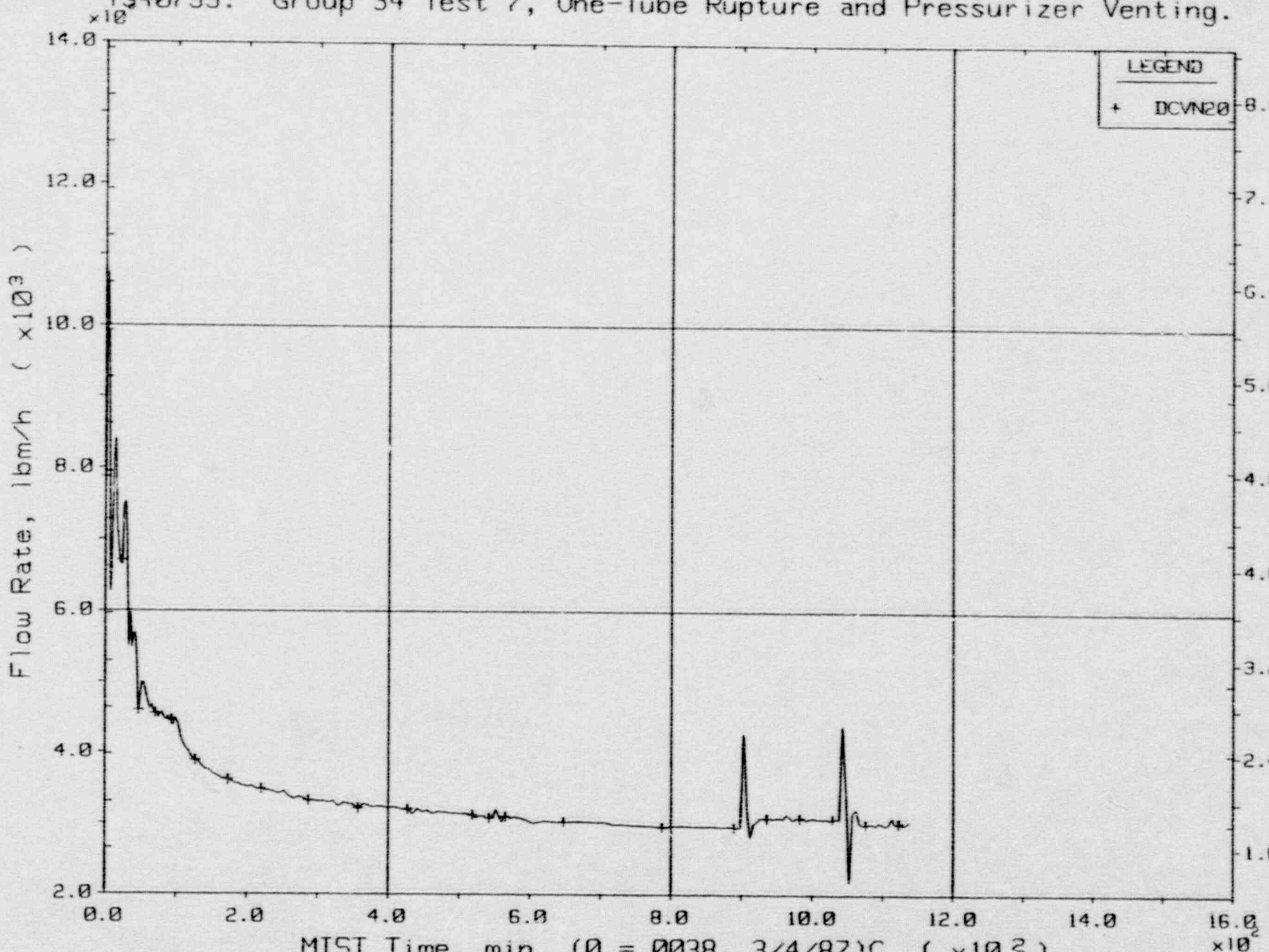
Core Region Collapsed Liquid Levels.

Thu Jun 2 10:47:24 1988

COLV1

FINAL DATA

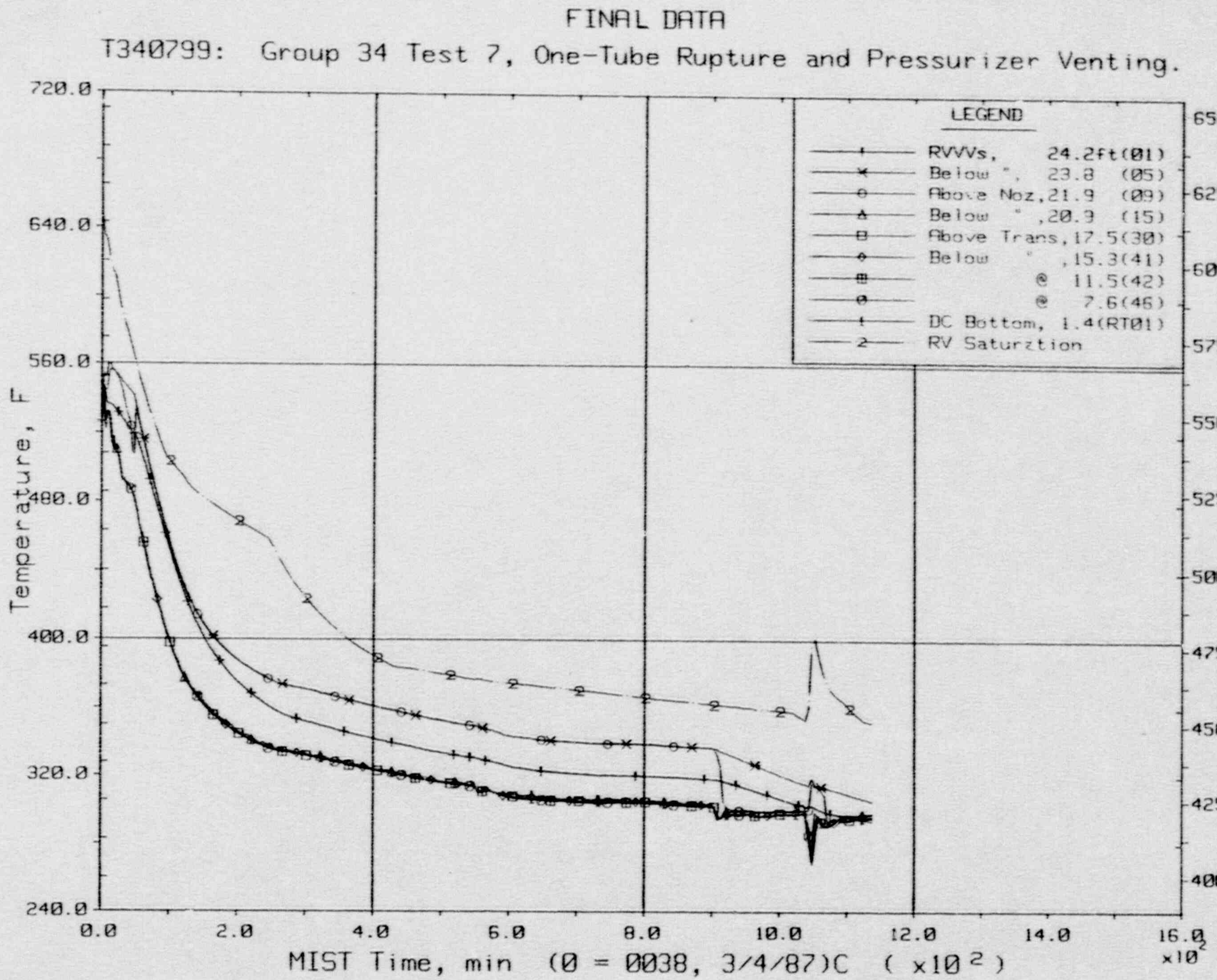
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Downcomer (Ventüri) Flow Rate.

Thu Jun 2 10:51:00 1988

DCFL1



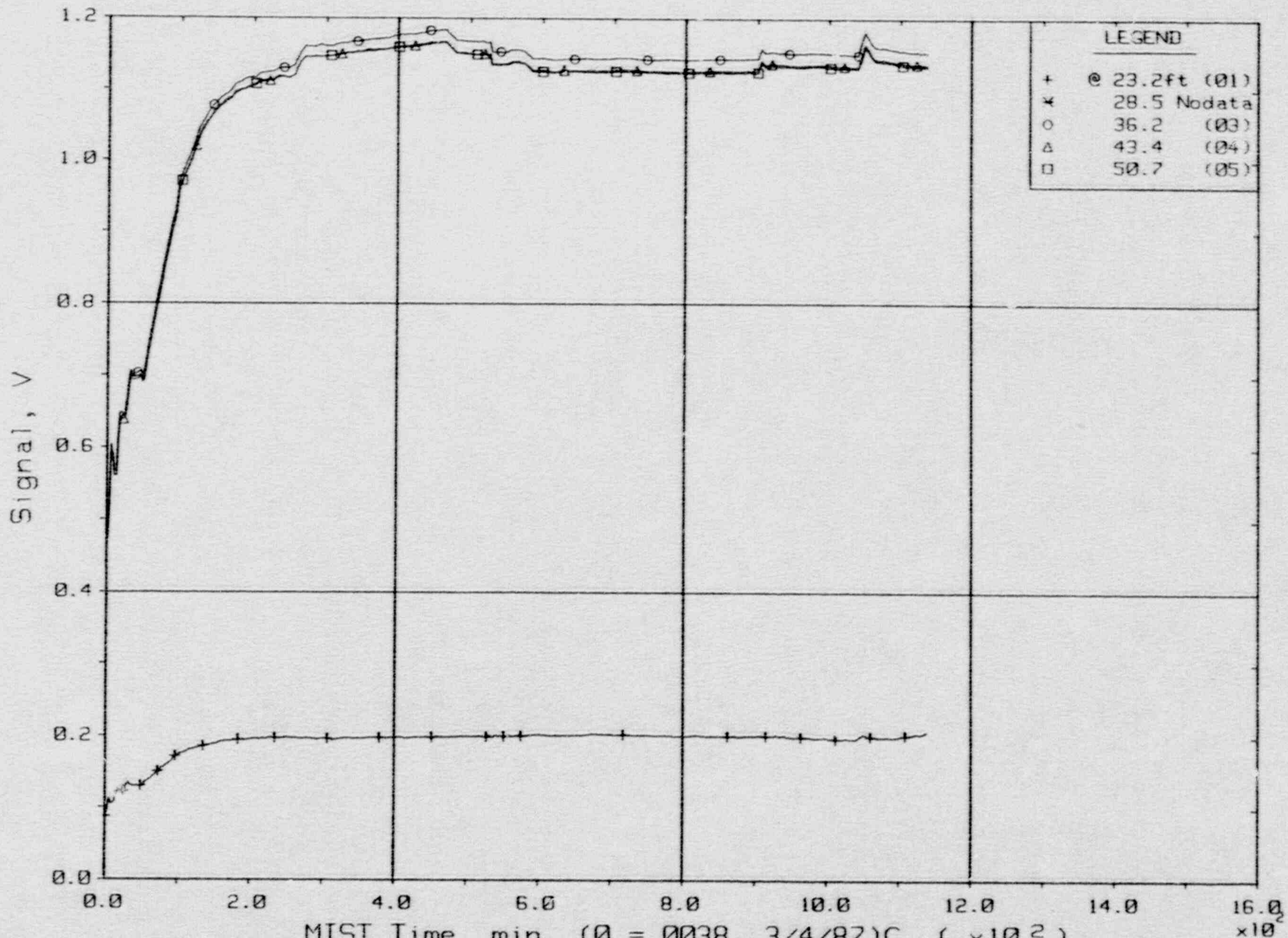
Downcomer Quadrant A1 Flüid Temperatures (DCTCs).

Thu Jun 2 10:53:46 1988

DCTC6

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



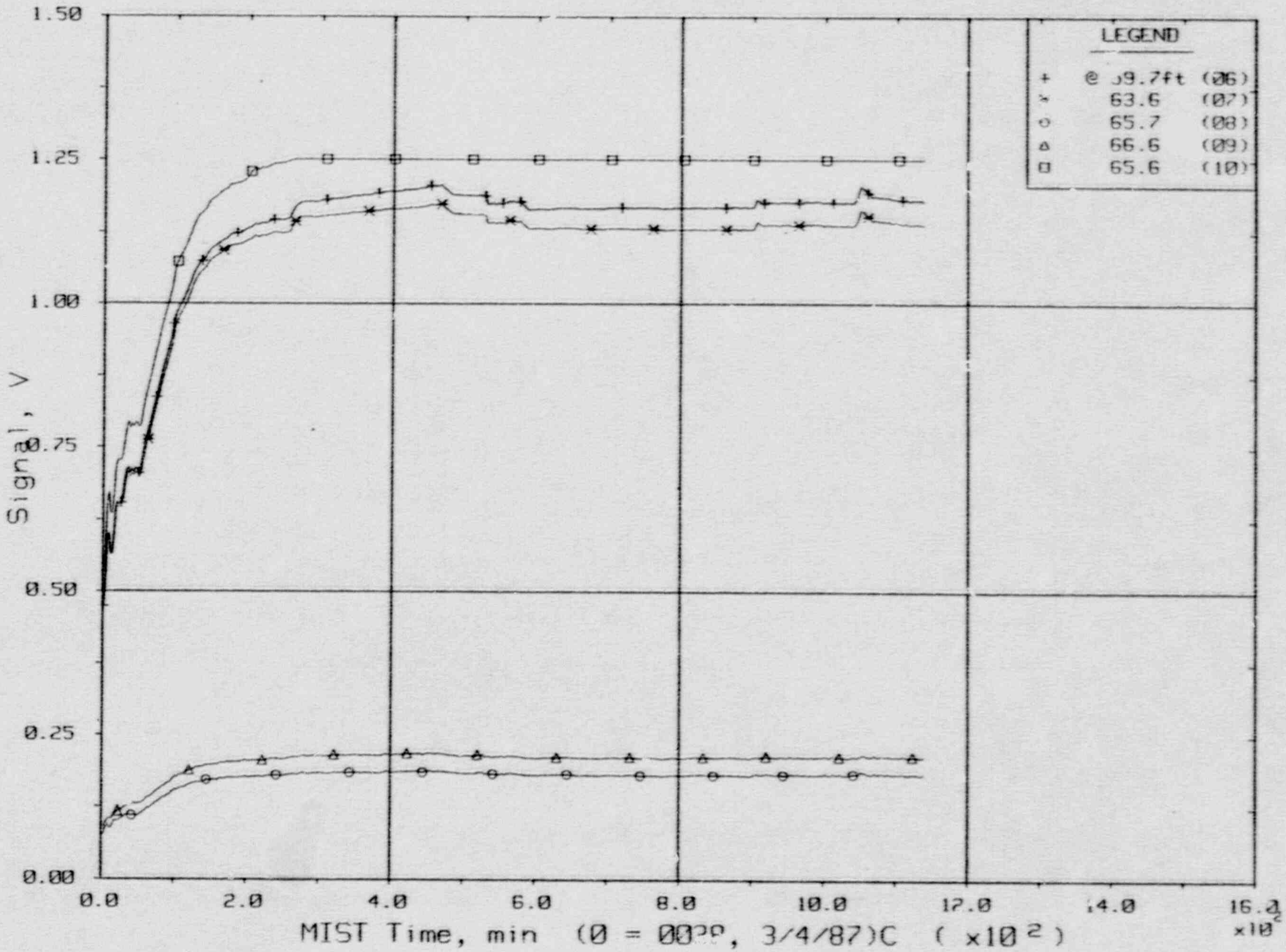
Hot Leg A Riser Conductivity Probe Signals (H1CPs).

Wed May 18 17:10:24 1988

H1CP1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



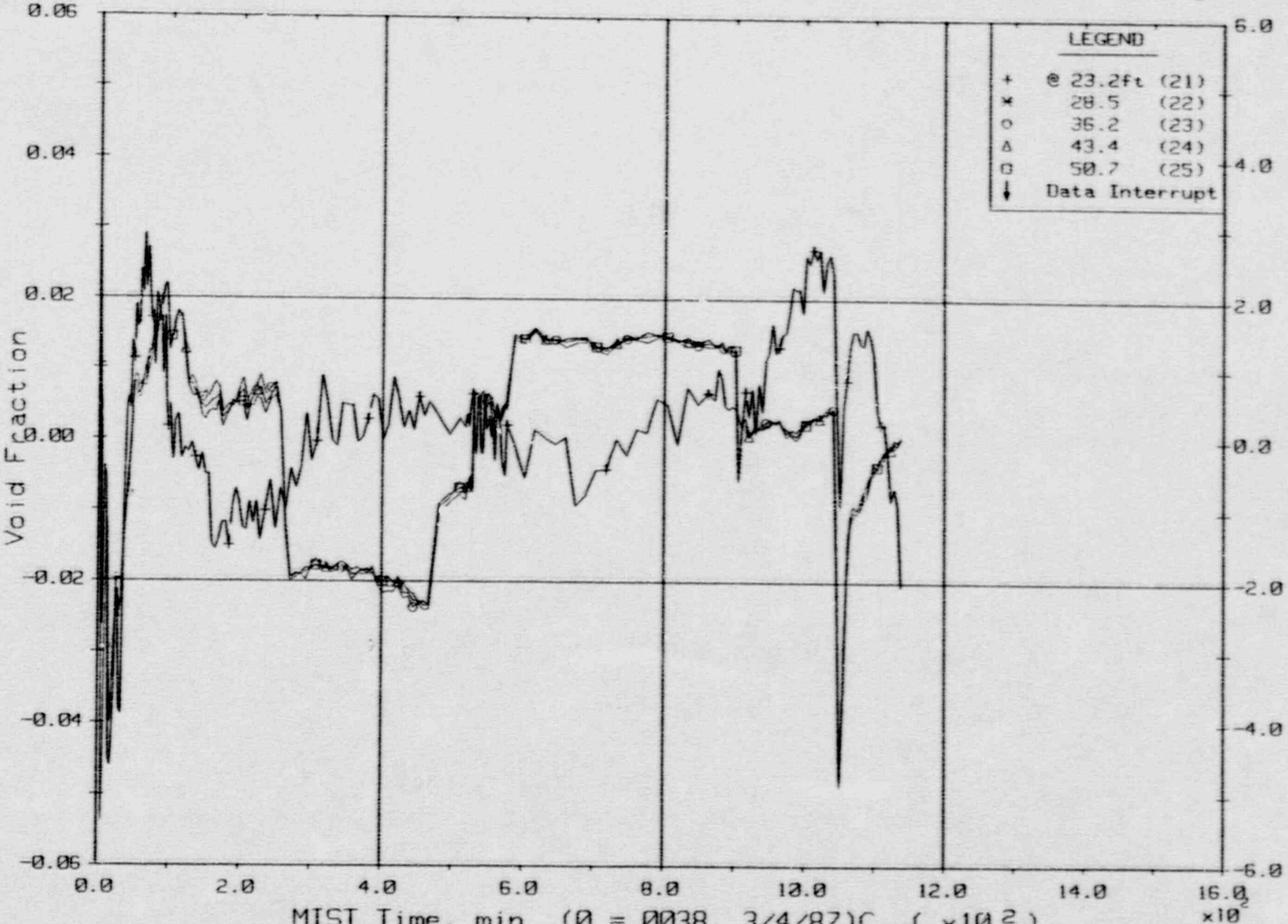
Hot Leg A U-Bend Conductivity Probe Signals (H1CPs).

Wed May 18 17:12:39 1988

H1CP2

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



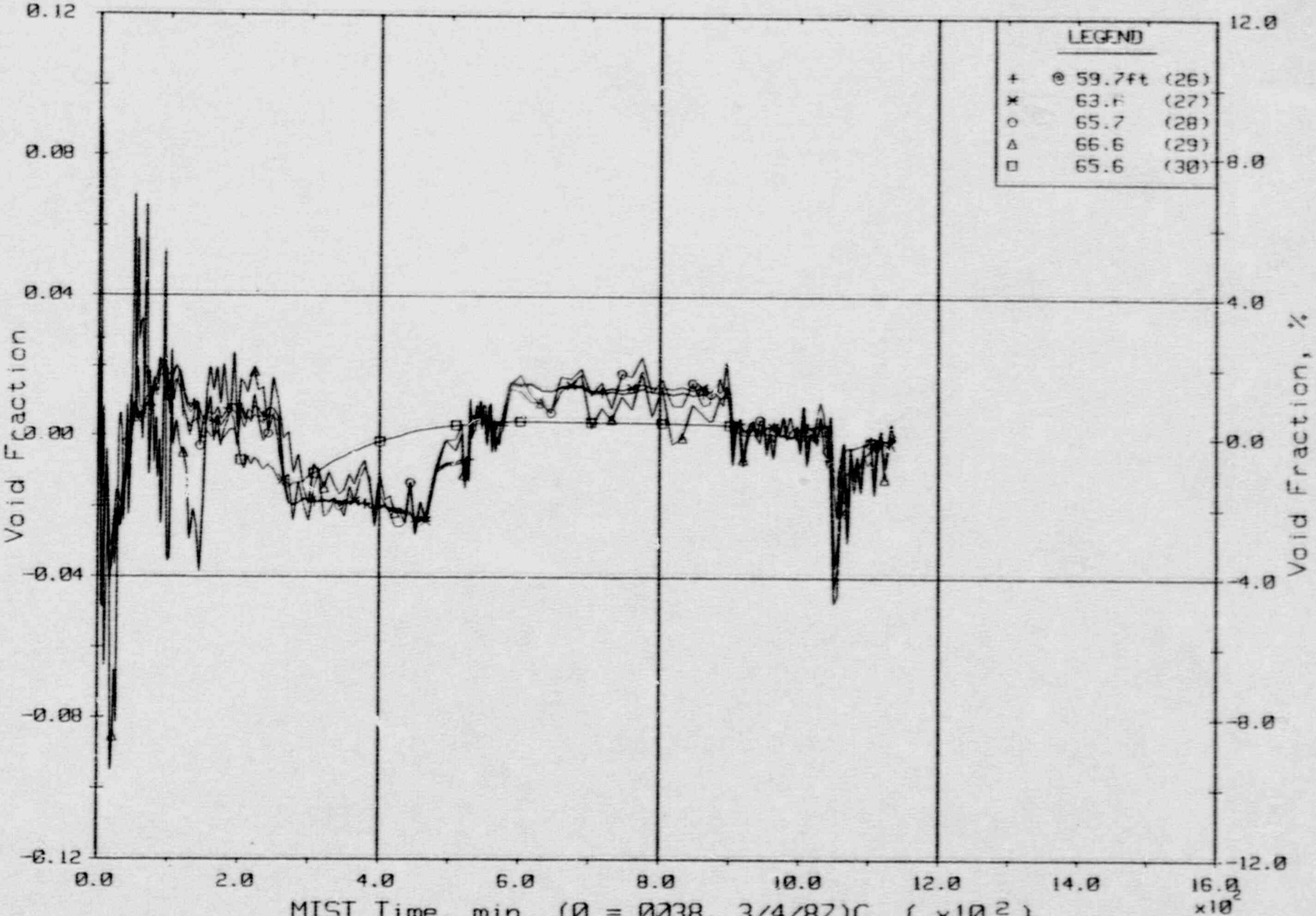
Hot Leg A Riser Void Fractions From Conductivity (H1CFs).

Wed May 18 17:15:52 1988

H1VC1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



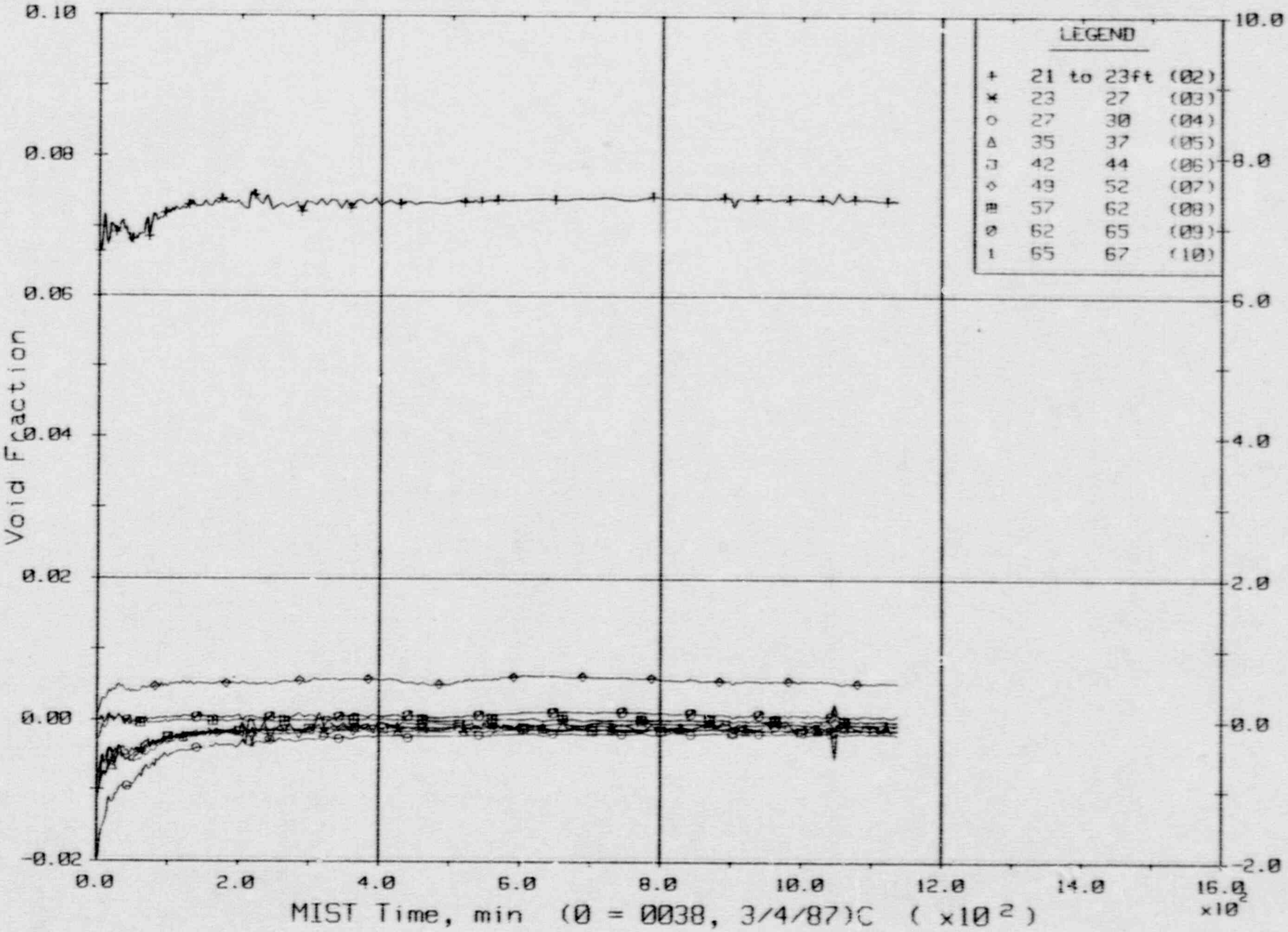
Hot Leg A U-bend Void Fractions From Conductivity (H1CPs).

Wed May 18 17:18:41 1988

H1VC2

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



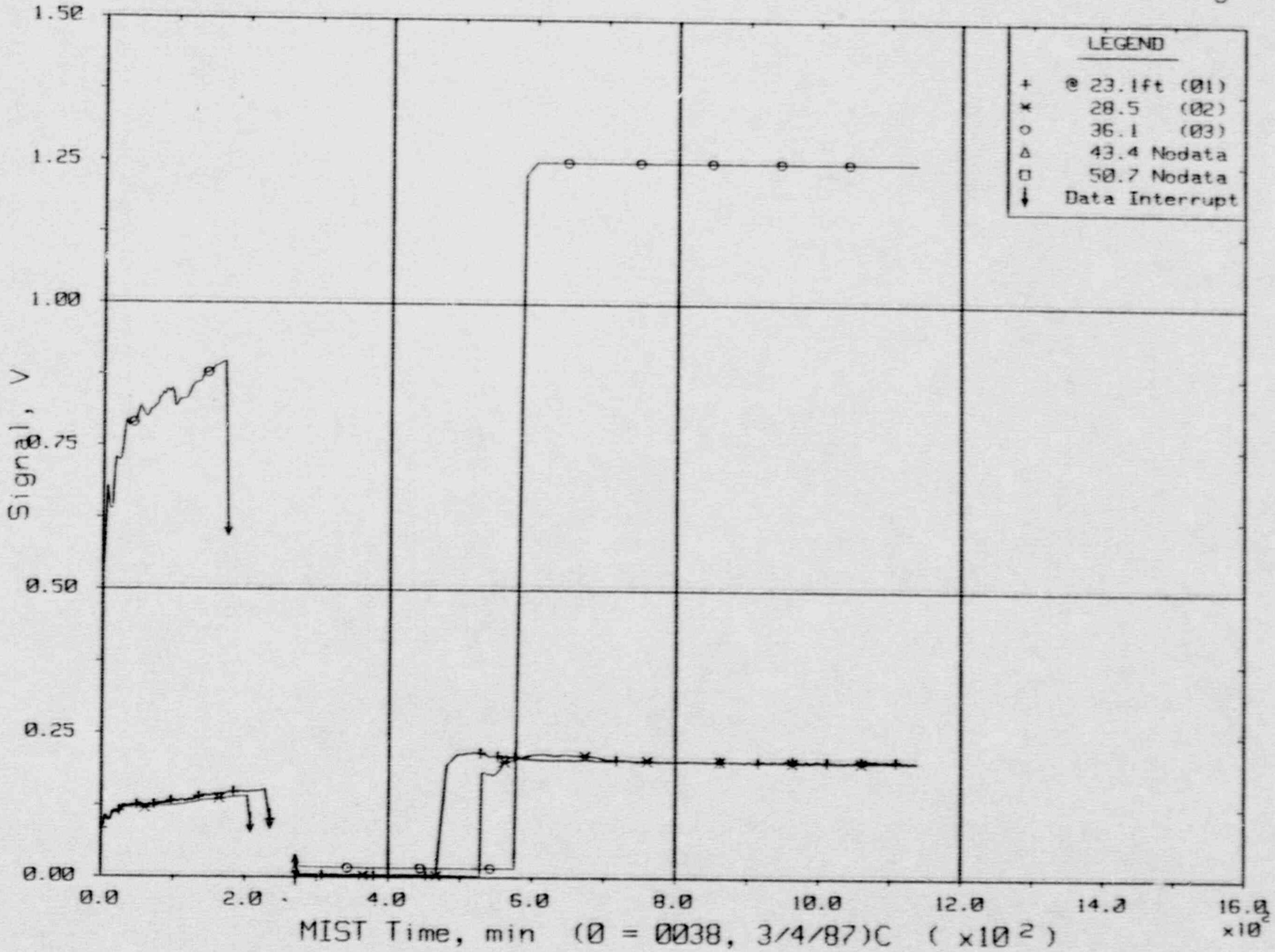
Hot Leg A Riser Void Fractions From Differential Pressures (H1VFs).

Thu Jun 2 10:57:16 1988

H1VF1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



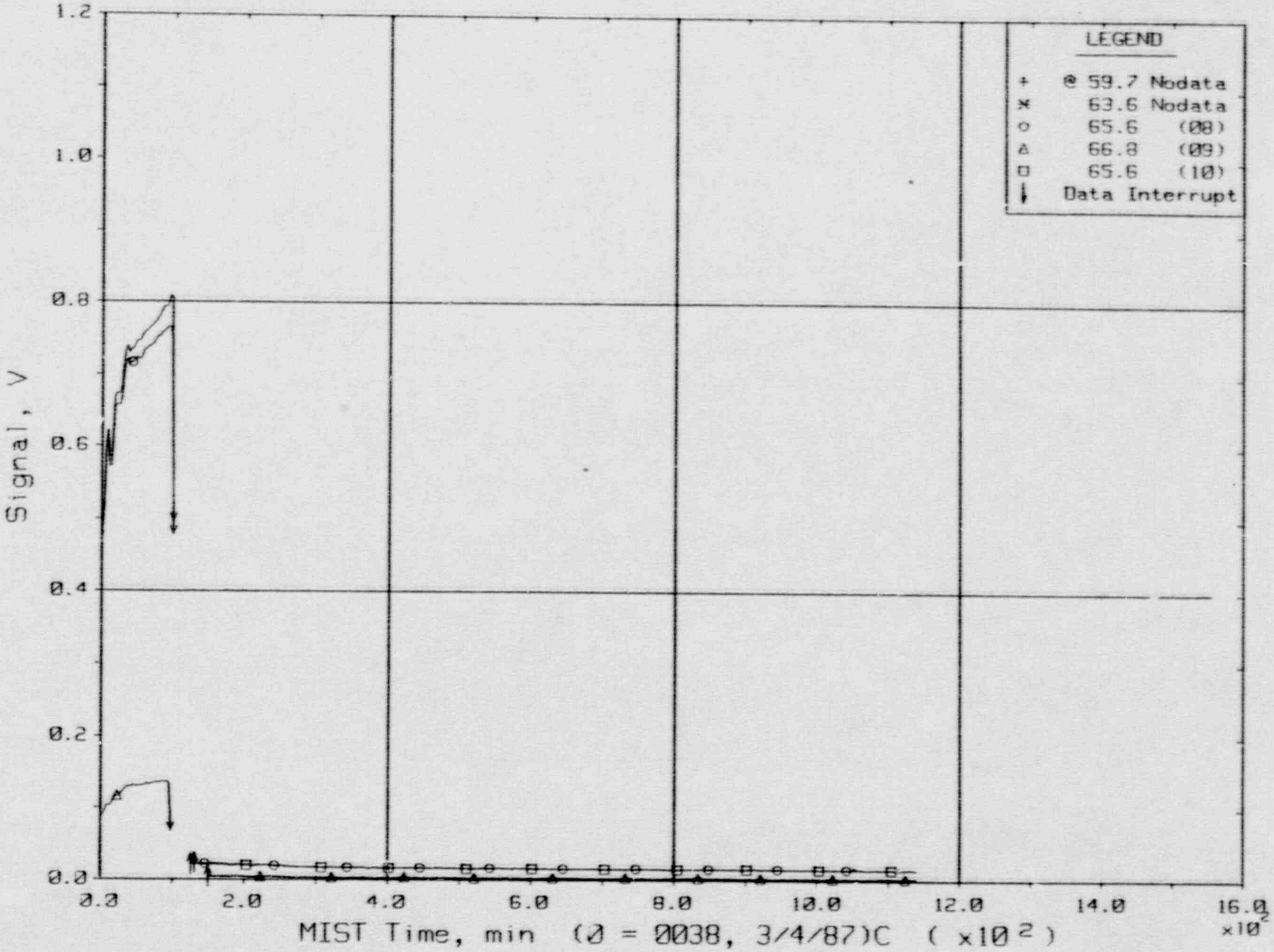
Hot Leg B Riser Conductivity Probe Signals (H2CPs).

Wed May 18 17:21:08 1988

H2CP1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



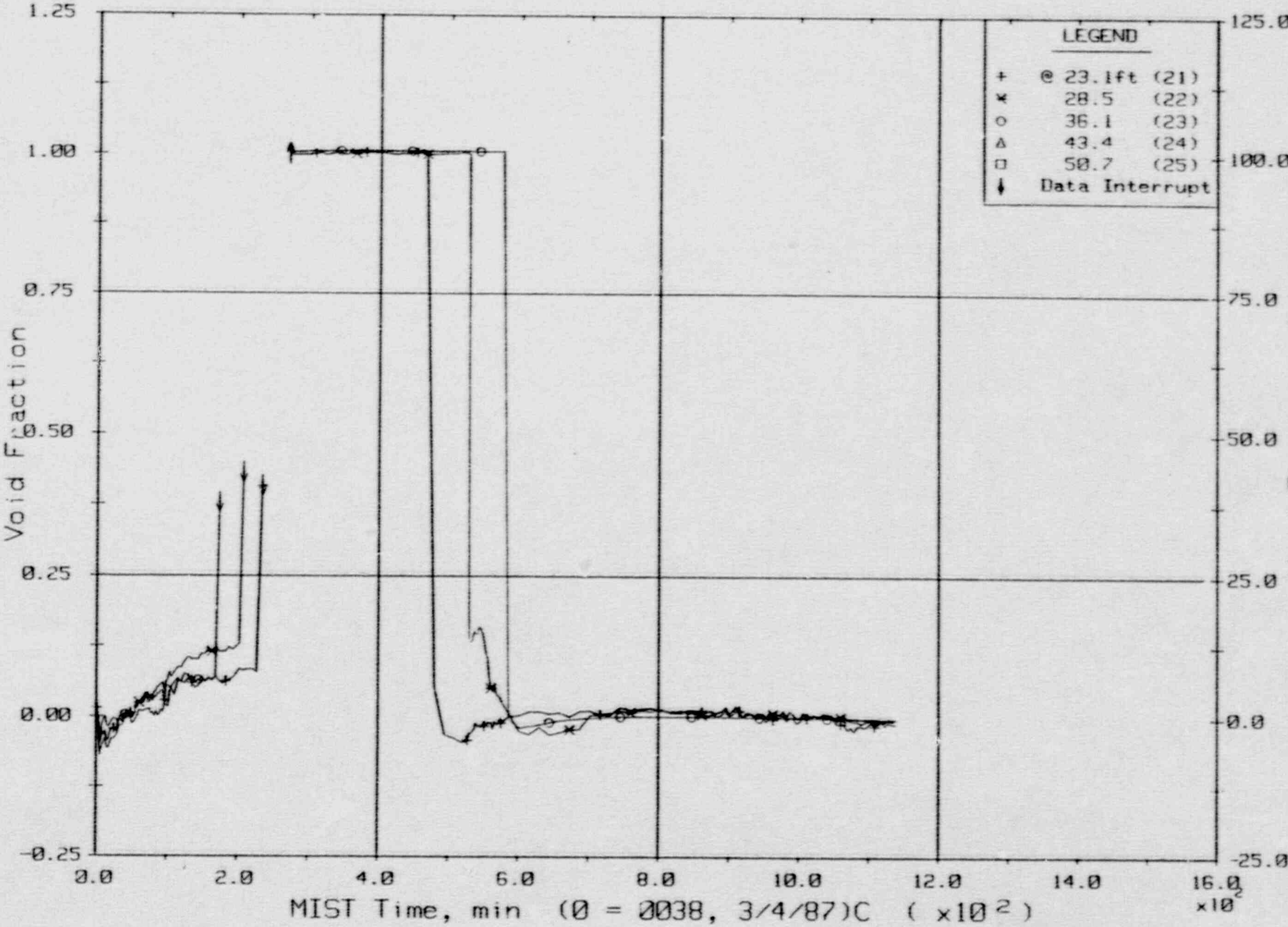
Hot Leg B U-Bend Conductivity Probe Signals (H2CPs).

Wed May 18 17:23:22 1988

H2CP2

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



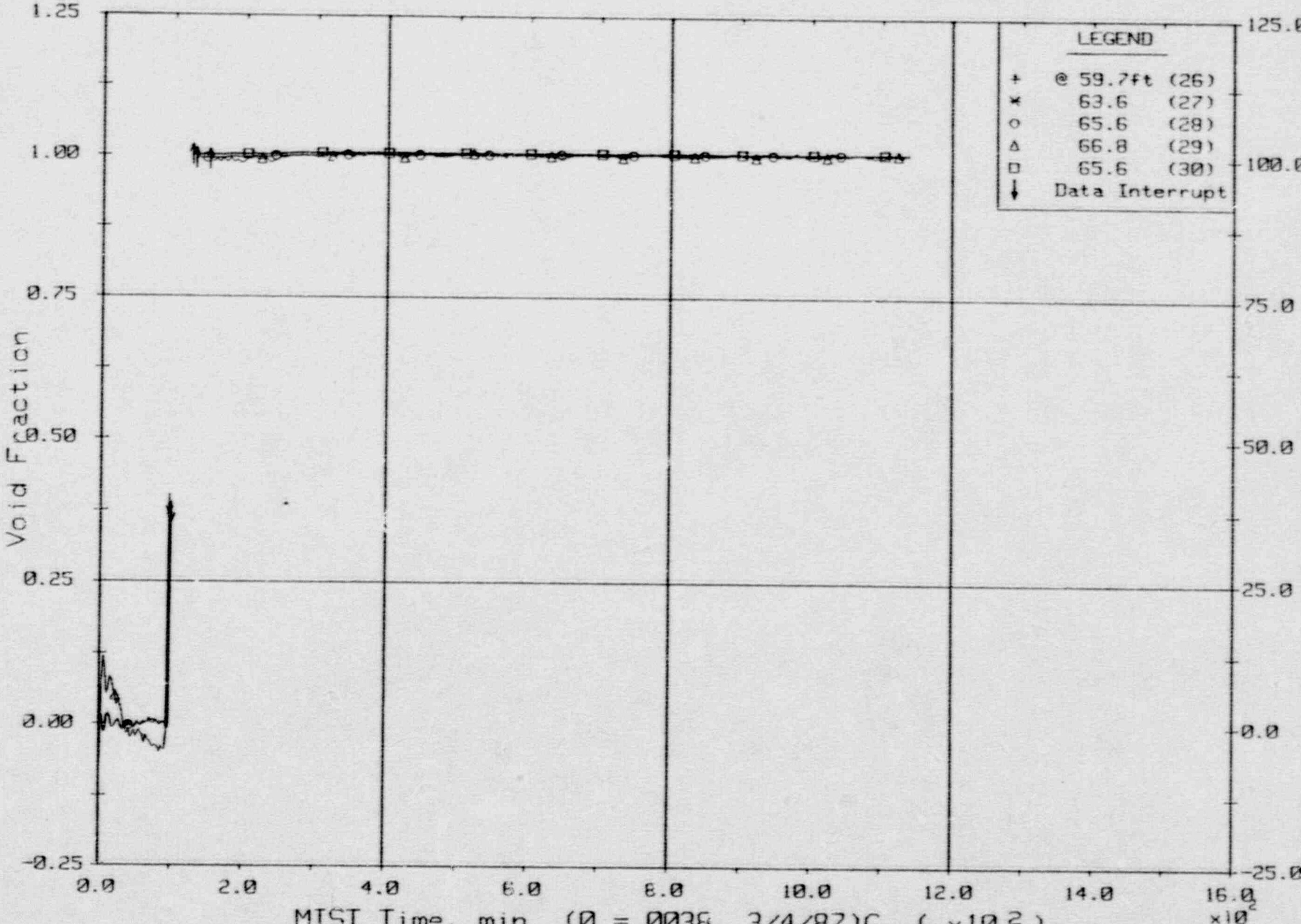
Hot Leg B Riser Void Fractions From Conductivity (H2CPs).

Wed May 18 17:25:40 1988

H2VC1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



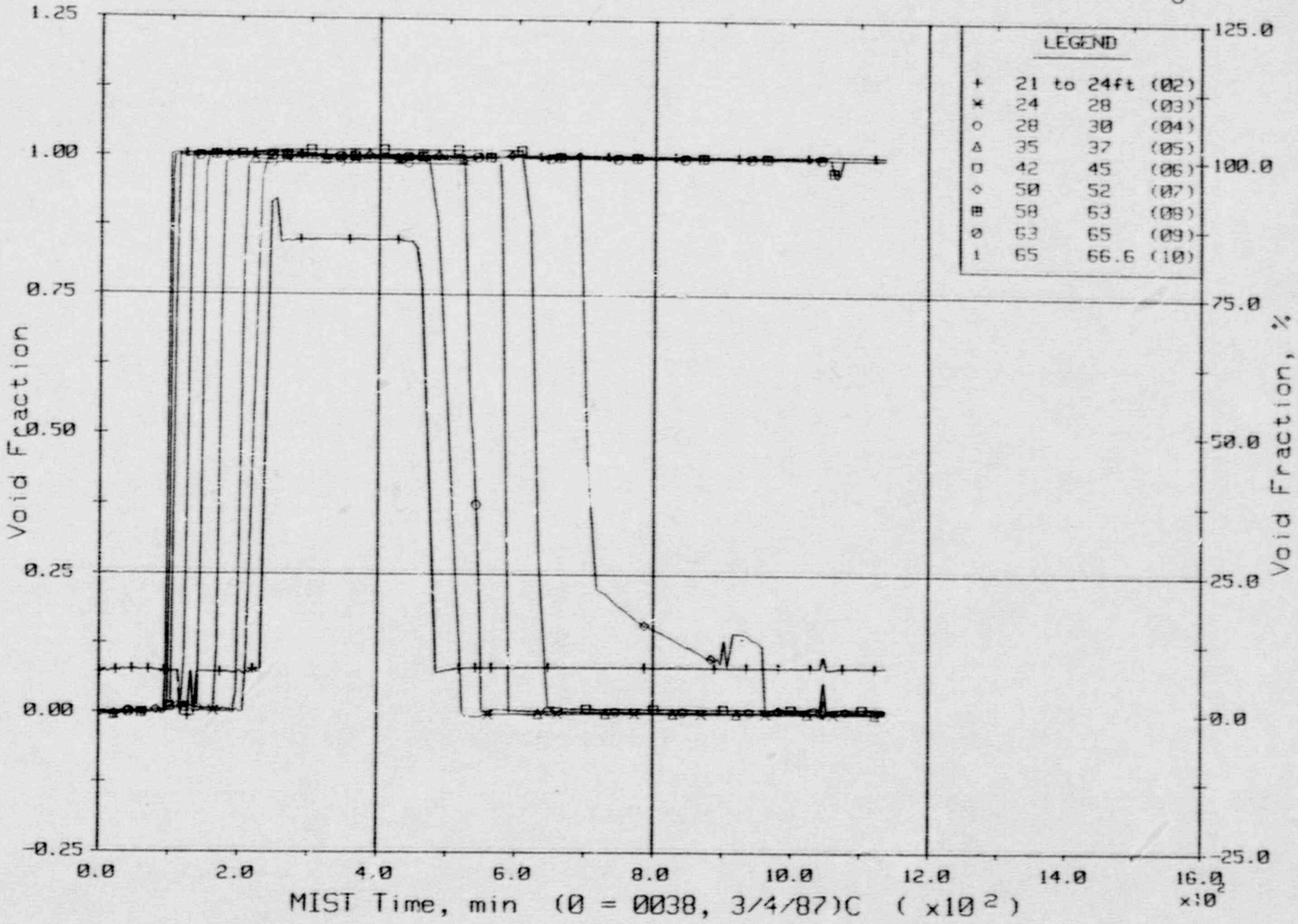
Hot Leg B U-bend Void Fractions From Conductivity (H2CPs).

Wed M / 18 17:42:32 1988

H2VC2

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



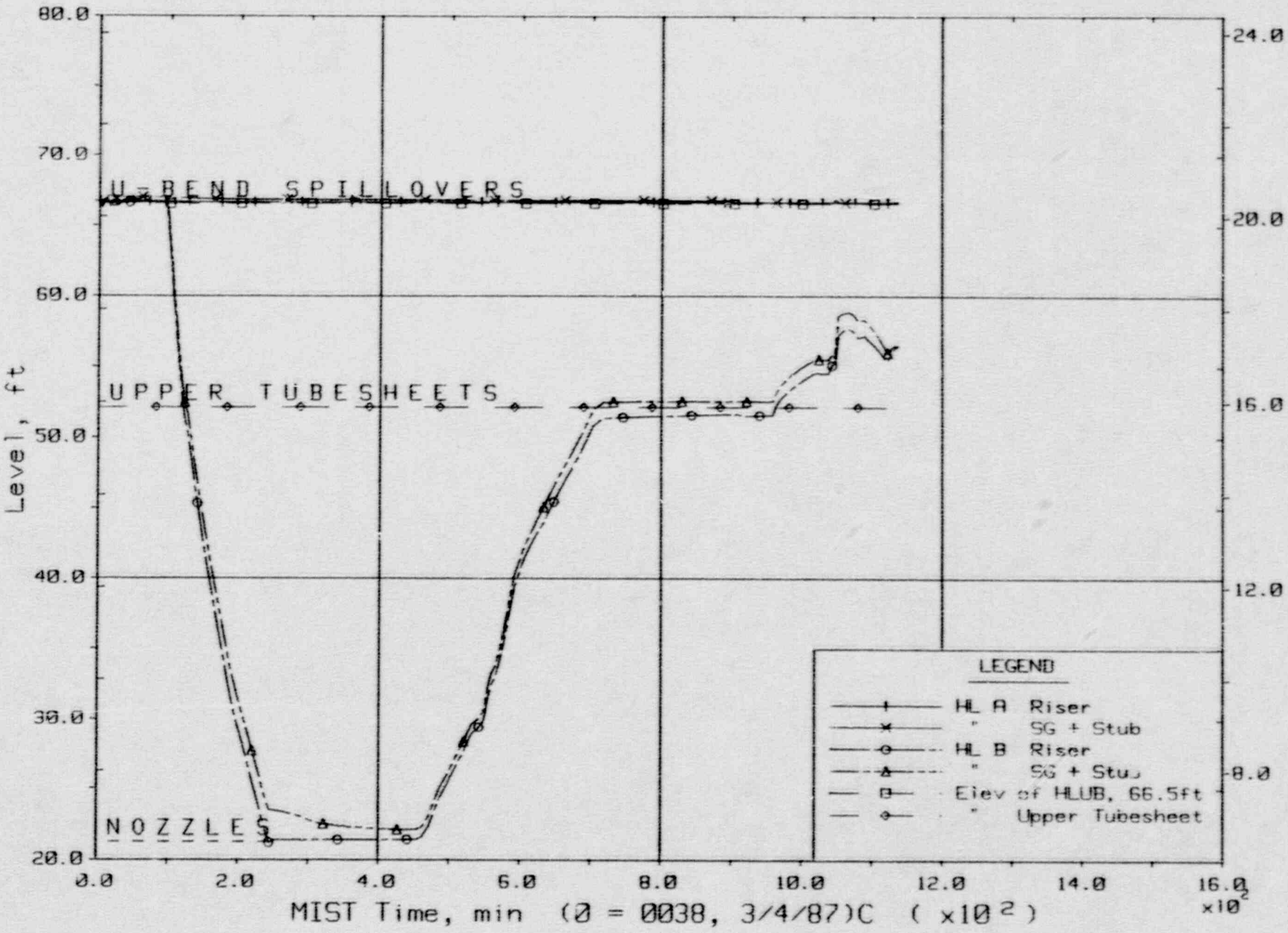
Hot Leg B Riser Void Fraction From Differential Pressures (H2VFs).

Thu Jun 2 11:00:16 1988

H2VF1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



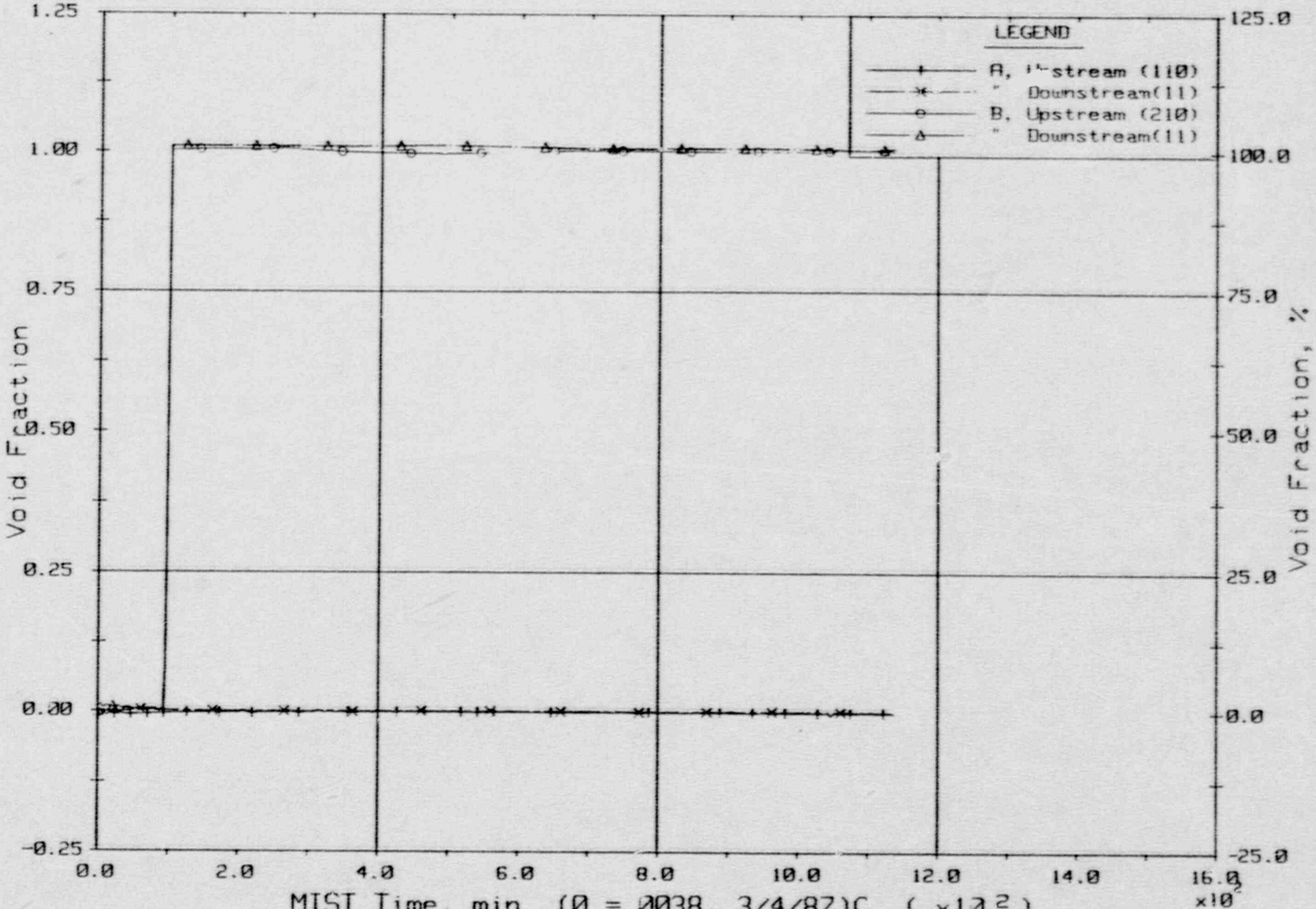
Hot Leg Riser and Stub Collapsed Liquid Levels.

Thu Jun 2 11:03:14 1988

HLLV1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



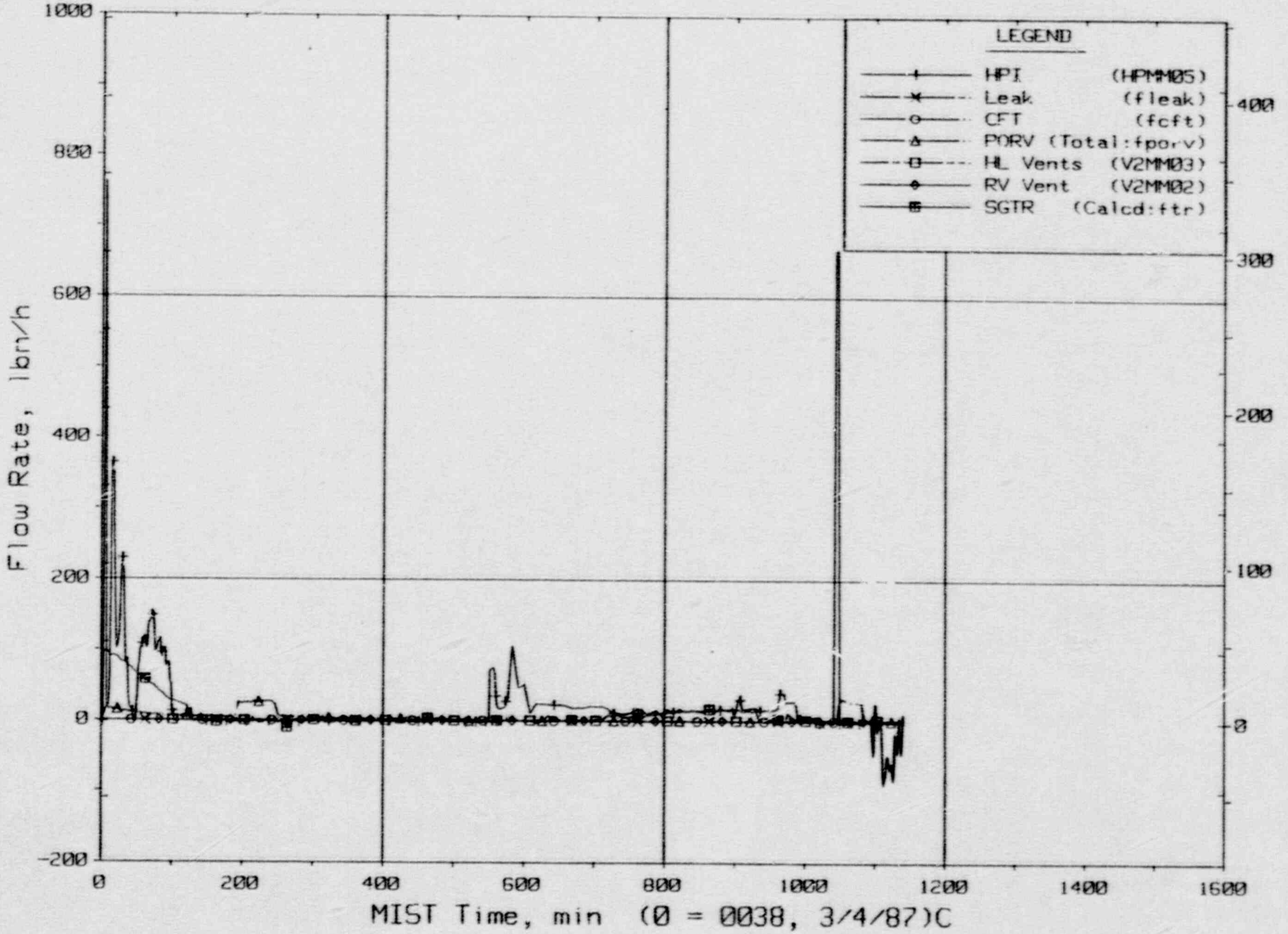
Hot Leg U-Bend Void Fractions From Diffl. Pressures (64.8 to 66.6 ft, HnVFs).

Thu Jun 2 11:05:52 1988

HLVF1

FINAl DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



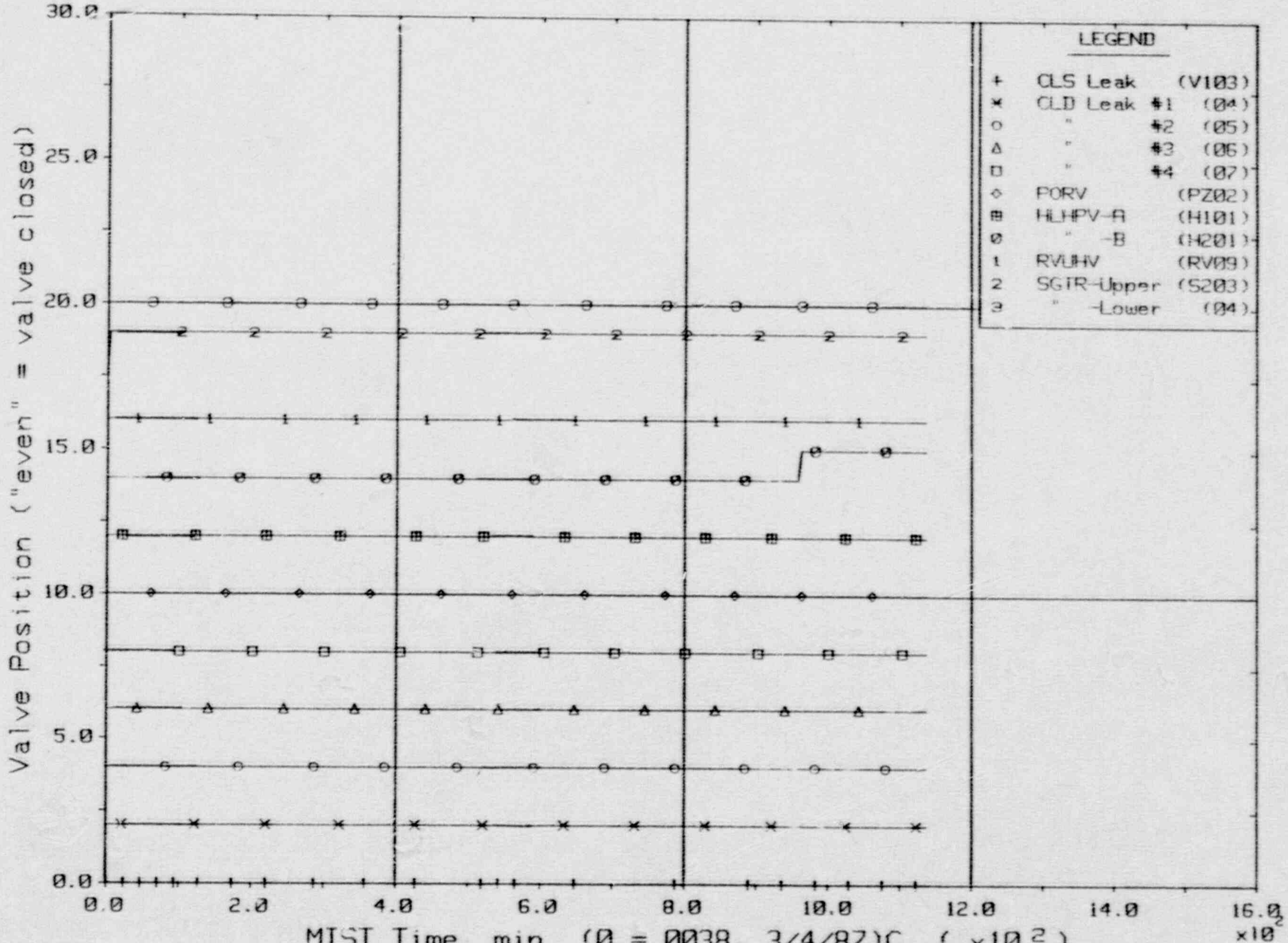
Primary System Boundary Flow Rates.

Fri Nov 4 13:53:02 1988

PBFL3

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



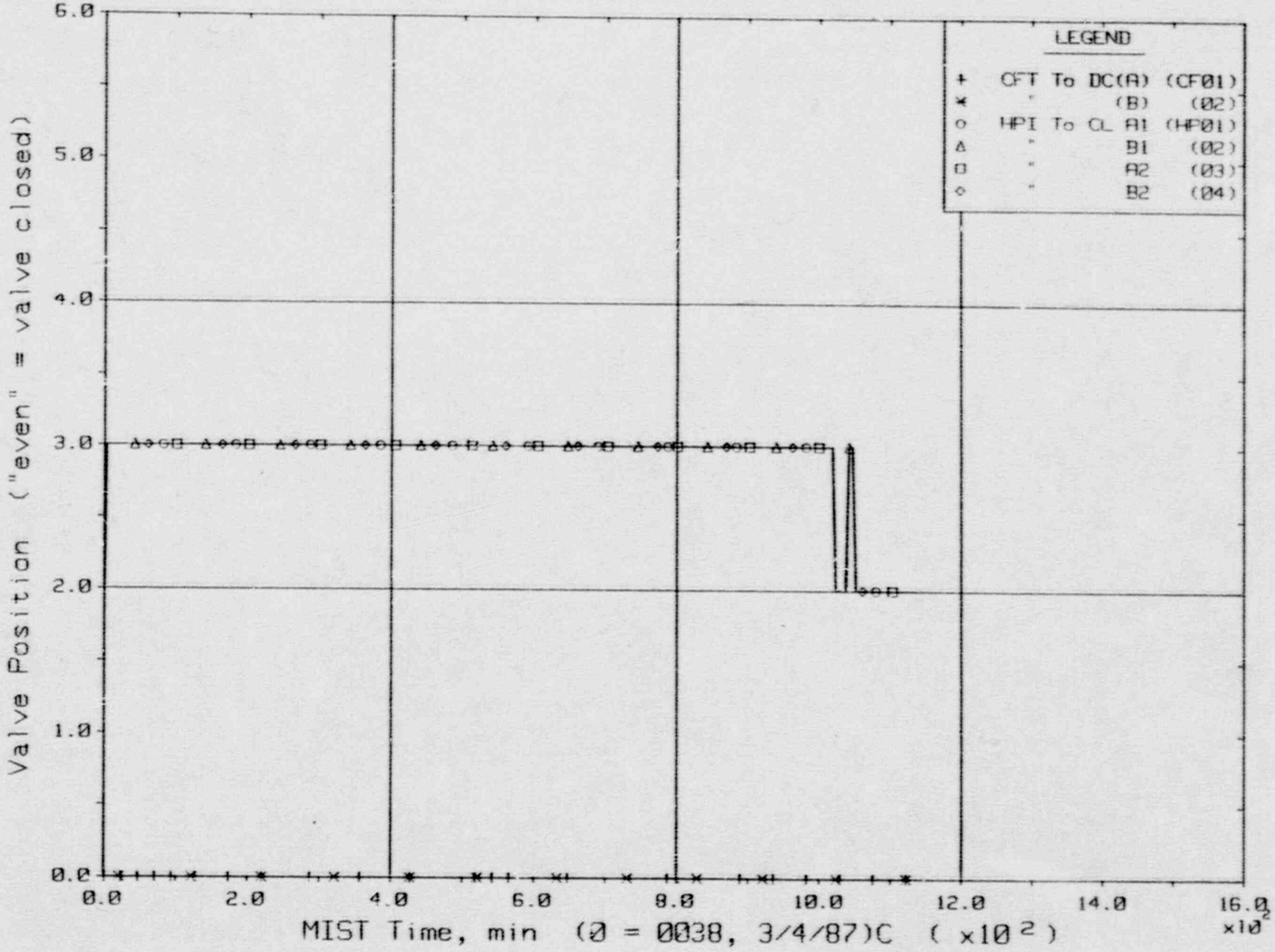
Primary System Discharge Limit Switch Indications (LSs).

Thu Jun 2 11:12:06 1988

PBLS1

FINAL DATA

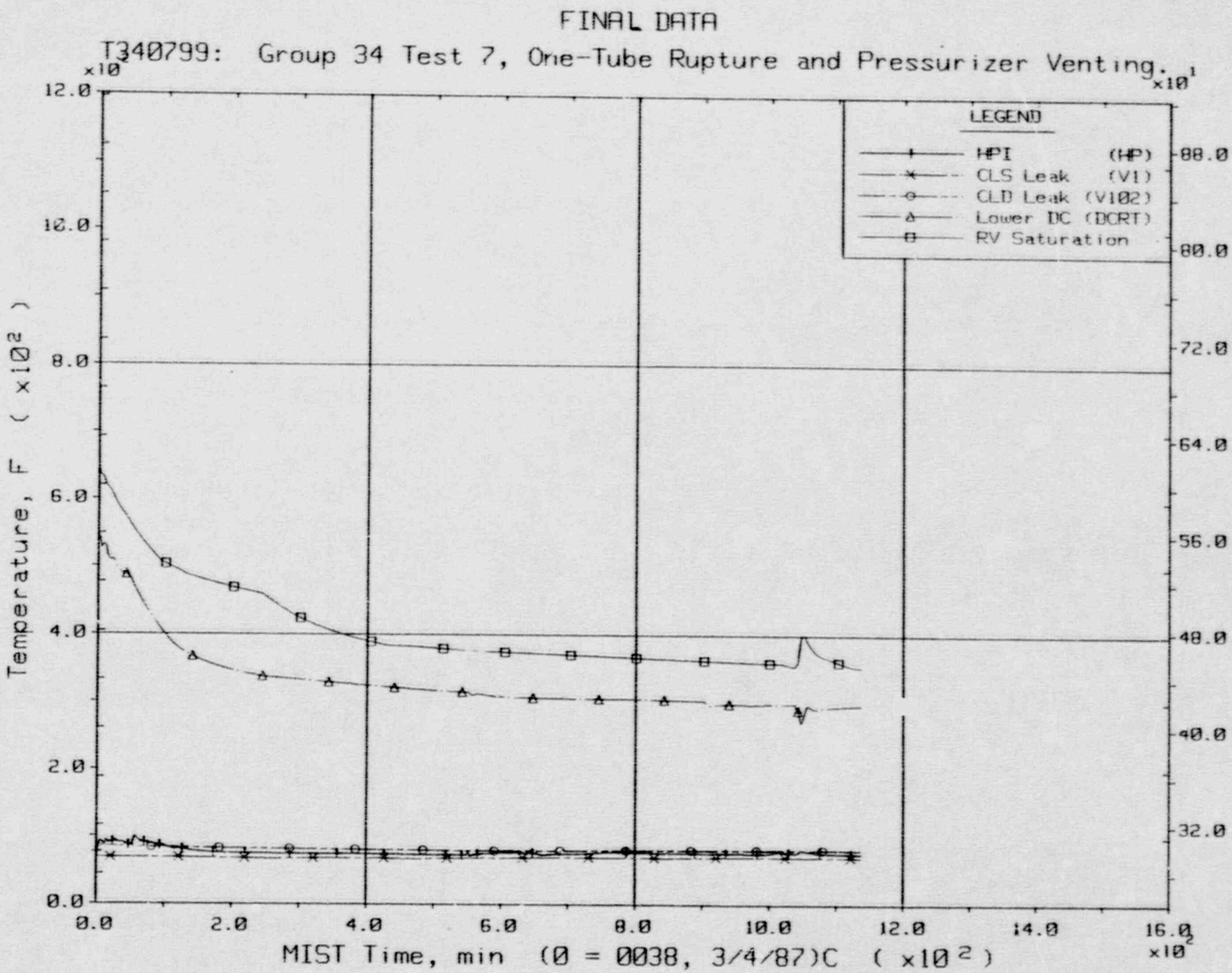
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Primary System Injection Limit Switch Indications (LSs).

Thu Jun 2 11:14:39 1988

PBLS2



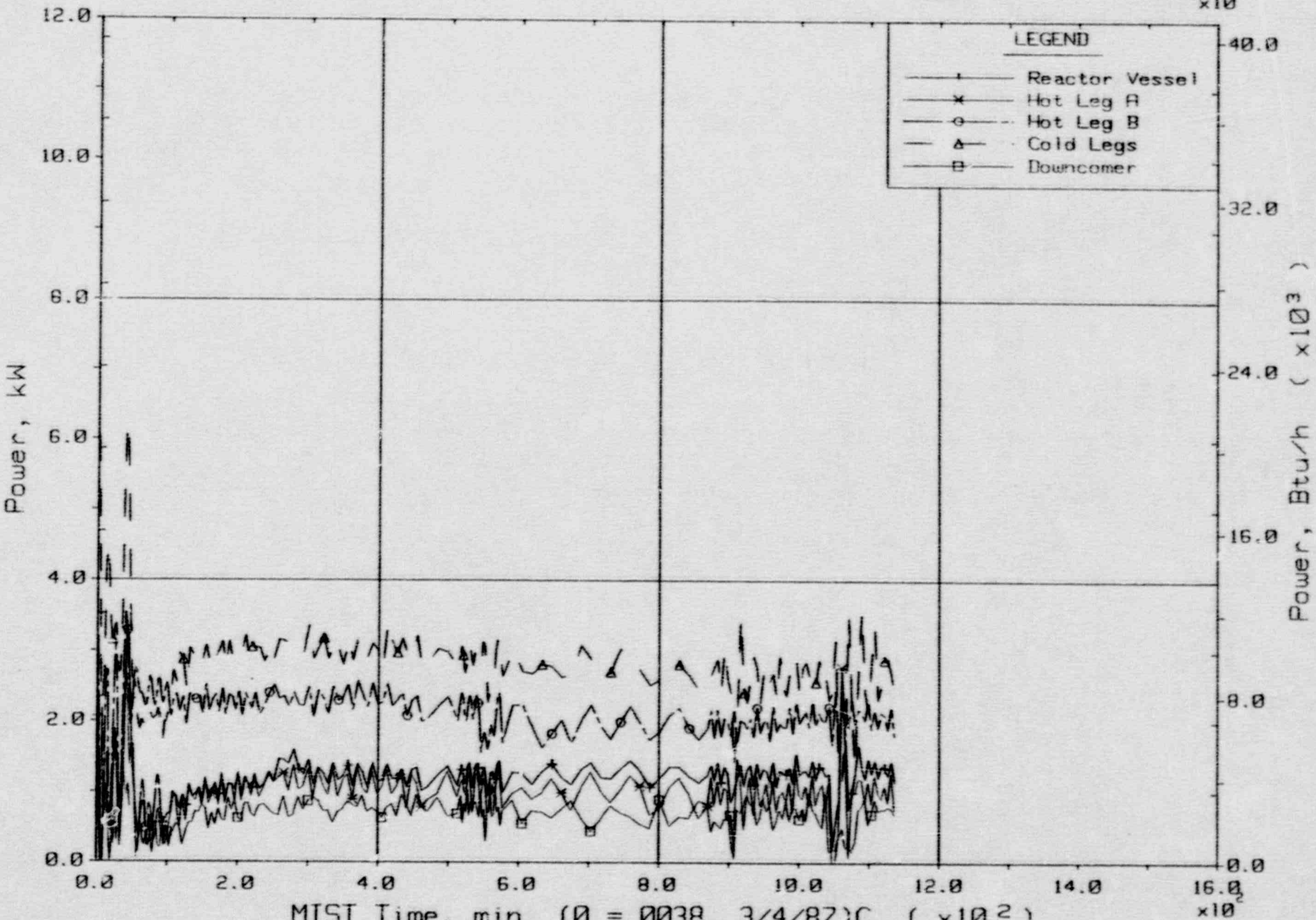
Single-Phase Discharge and HPI Fluid Temperatures (TC01s).

Thu Jun 2 11:17:04 1988

PBTC1p

FINAL DATA

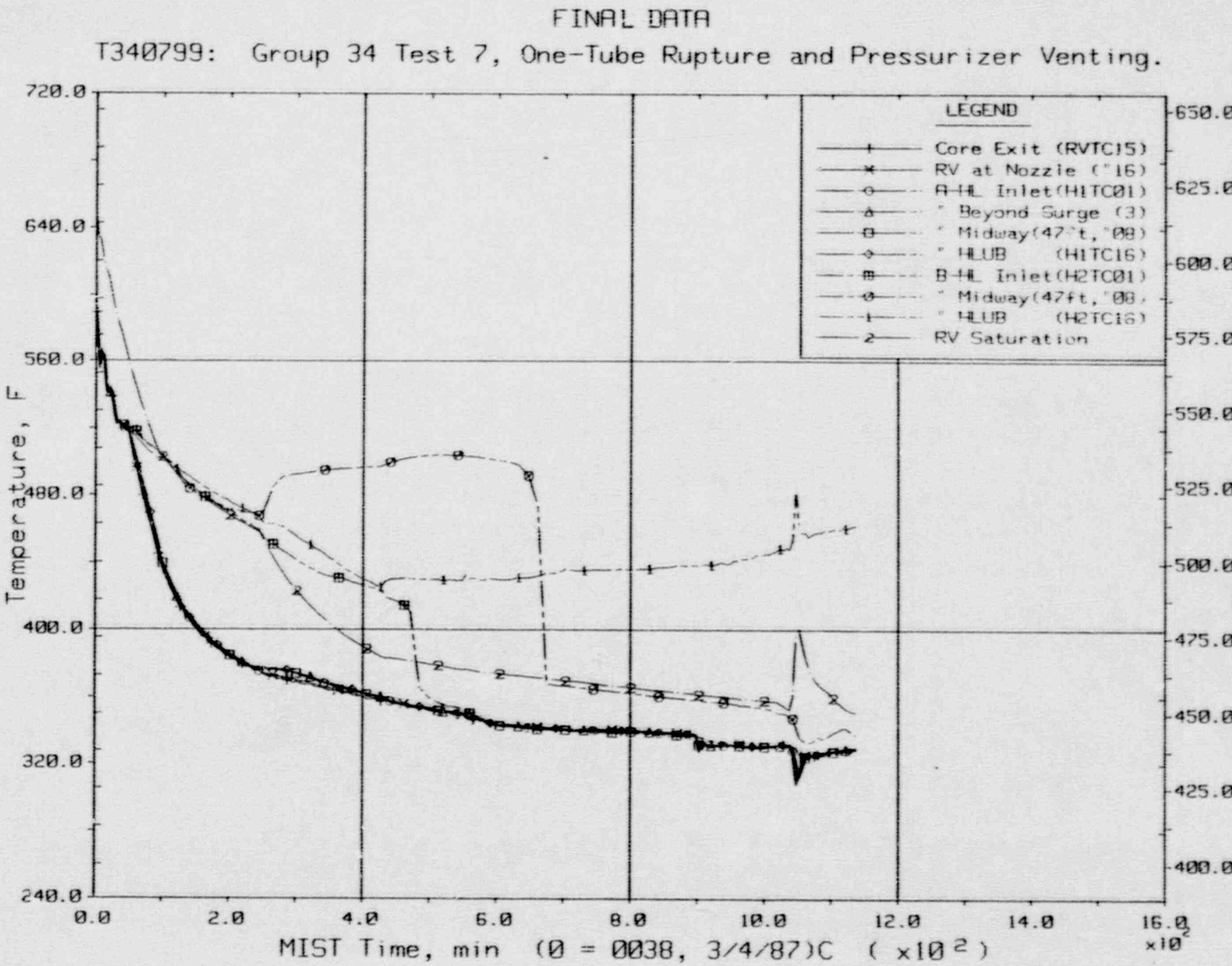
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Guard Heater Specified Power Per Primary Component.

Thu Jun 2 11:21:53 1988

PRGQ1



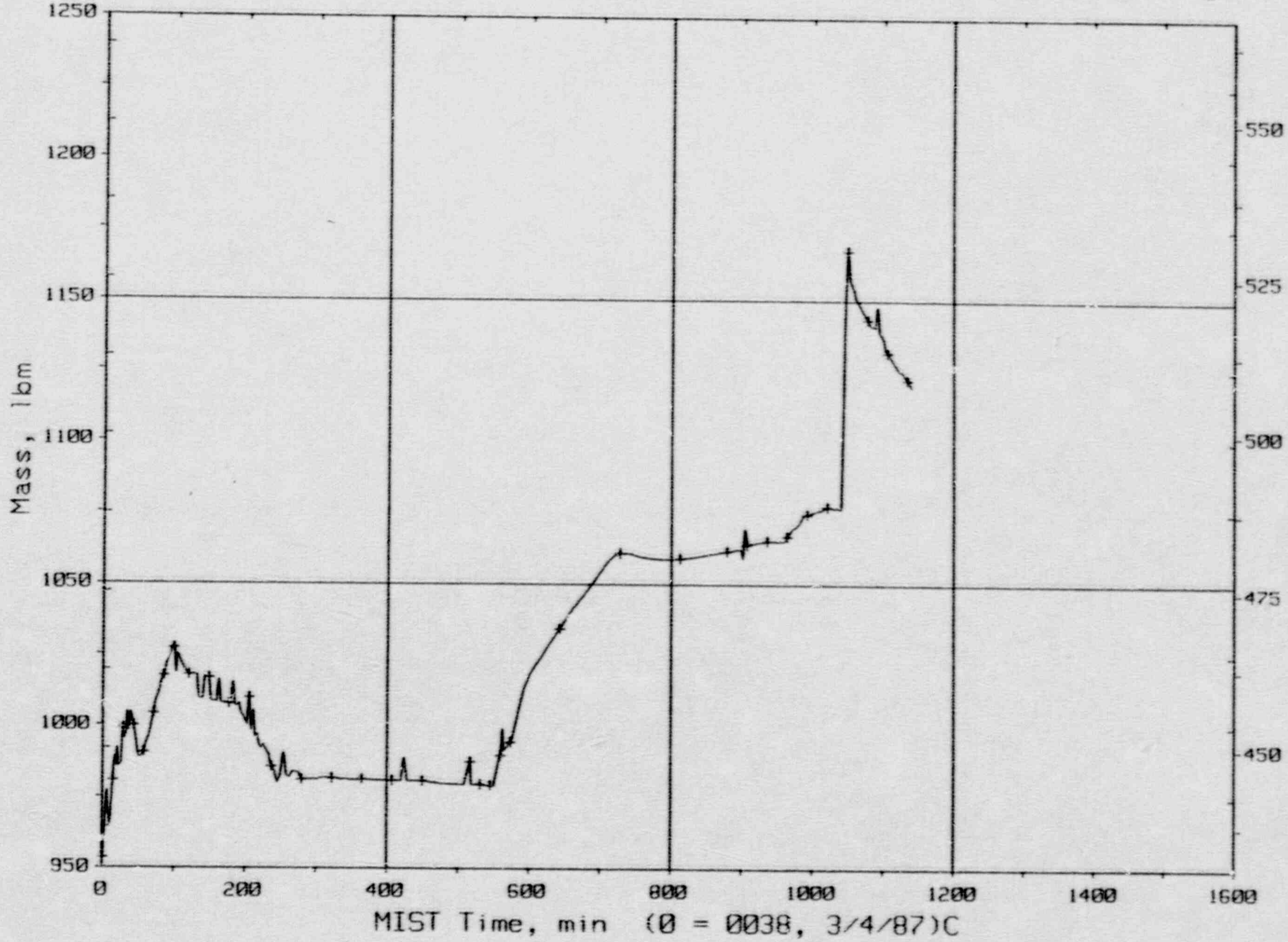
Composite Core Exit and Hot Leg Fluid Temperatures.

Thu Jun 2 11:26:22 1988

PRKT1

FINAl DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



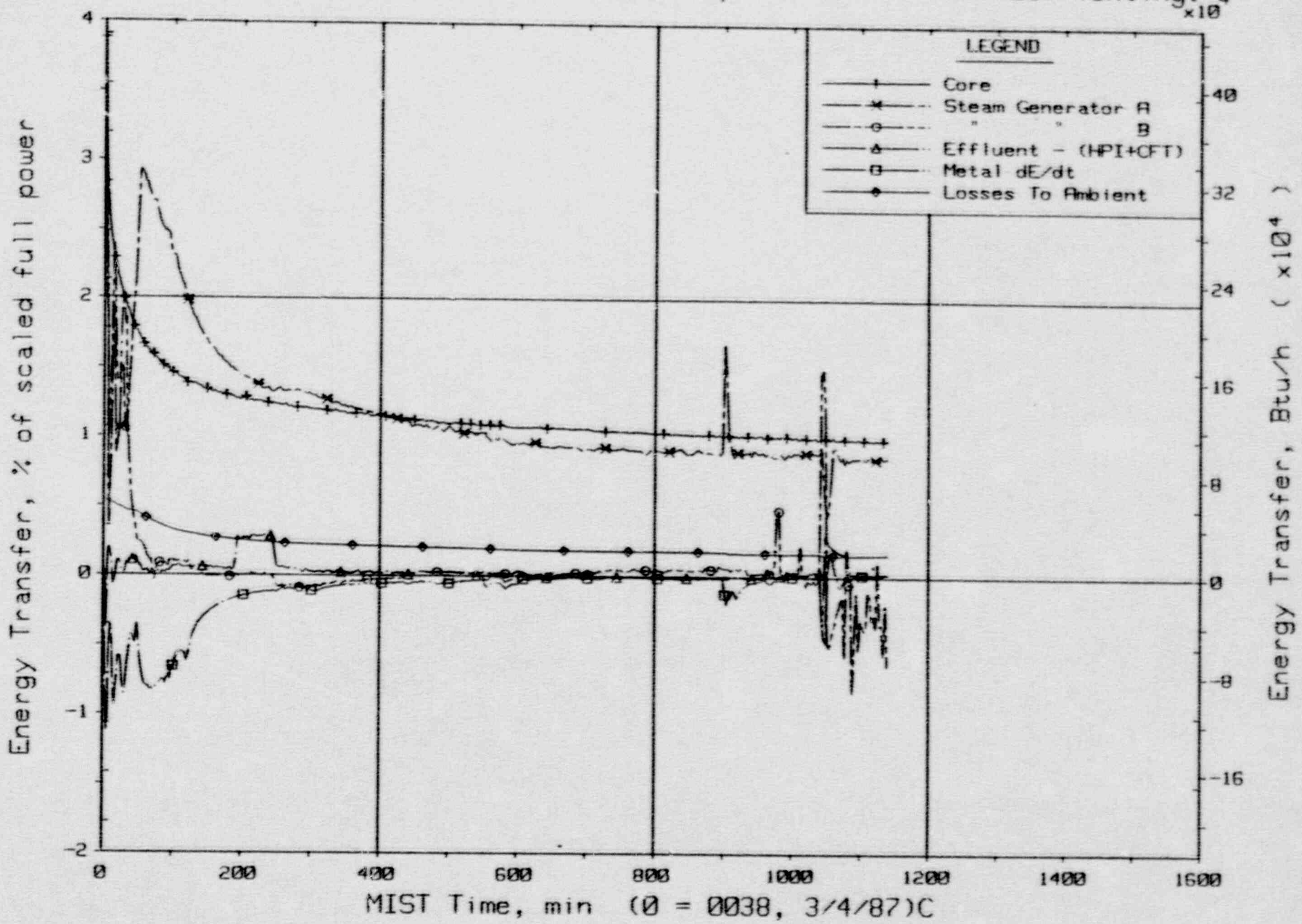
Indicated Primary System Total Fluid Mass (PLML20).

Wed Nov 2 14:12:12 1988

PRML9

FINAl DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



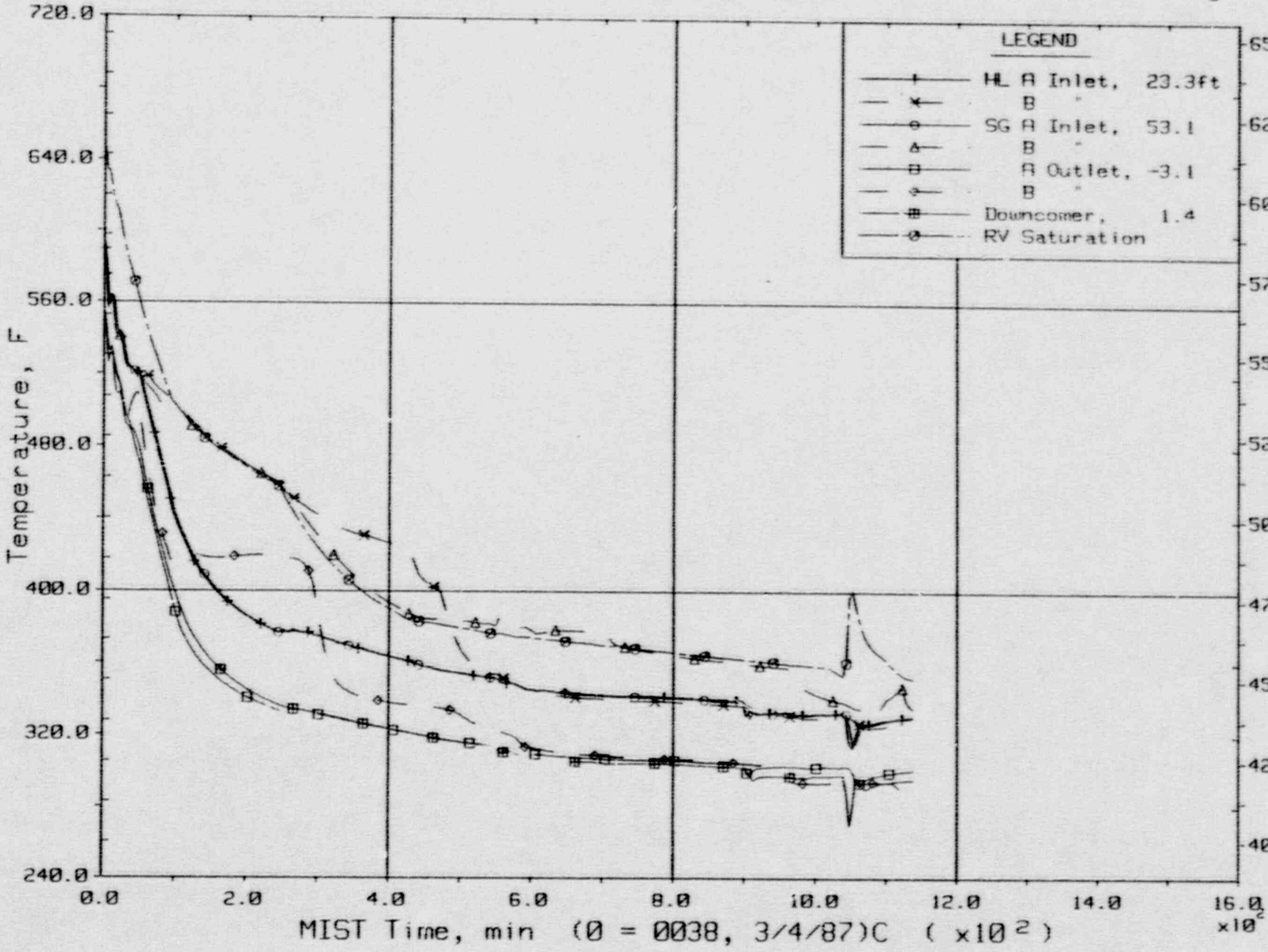
Primary System Energy Transfer.

Fri Nov 4 14:00:00 1988

PROE2

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



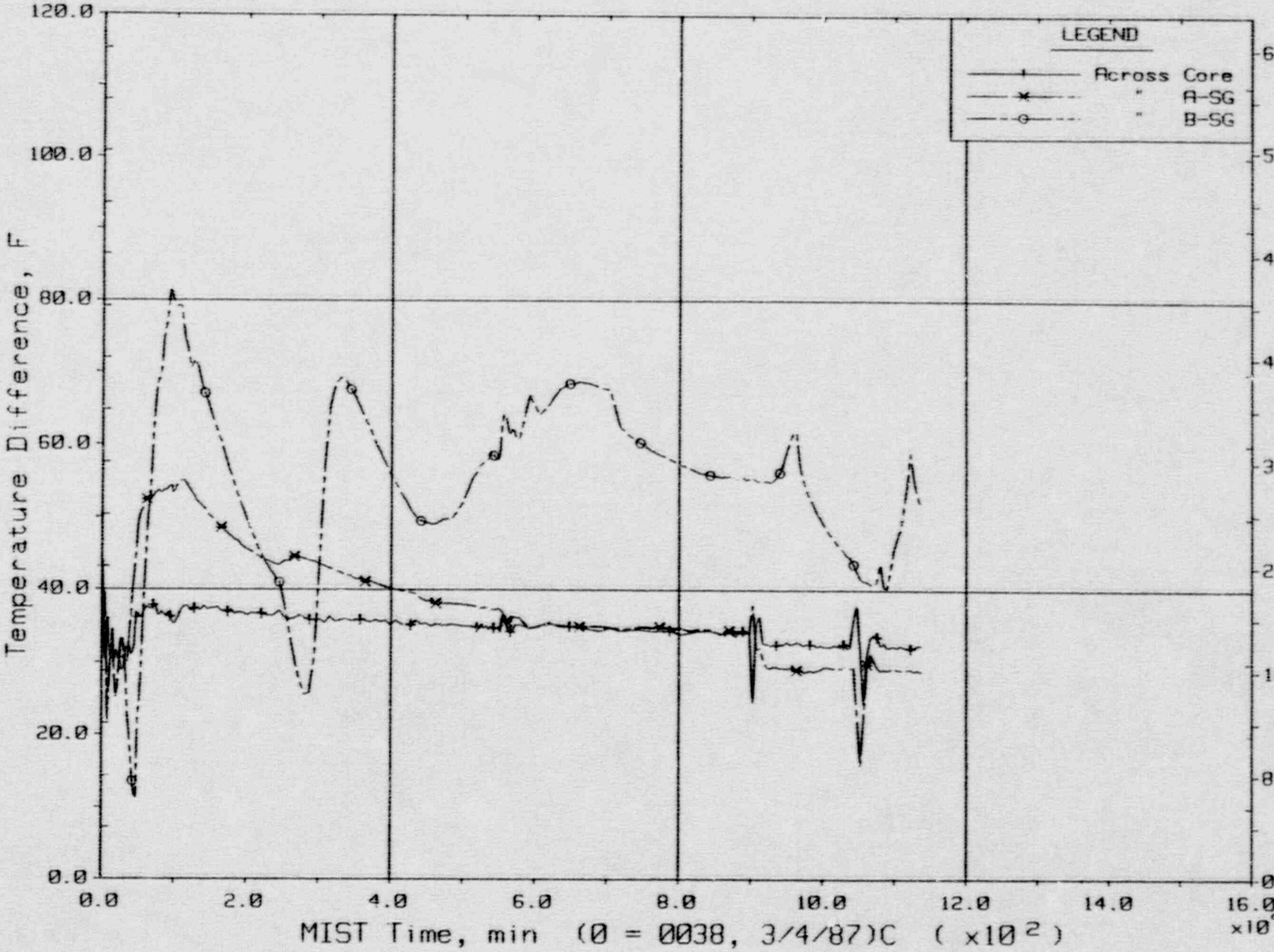
Primary System Fluid"Temperatures (RTDs).

Thu Jun 2 11:44:32 1988

PRRT0

FINAL DATA

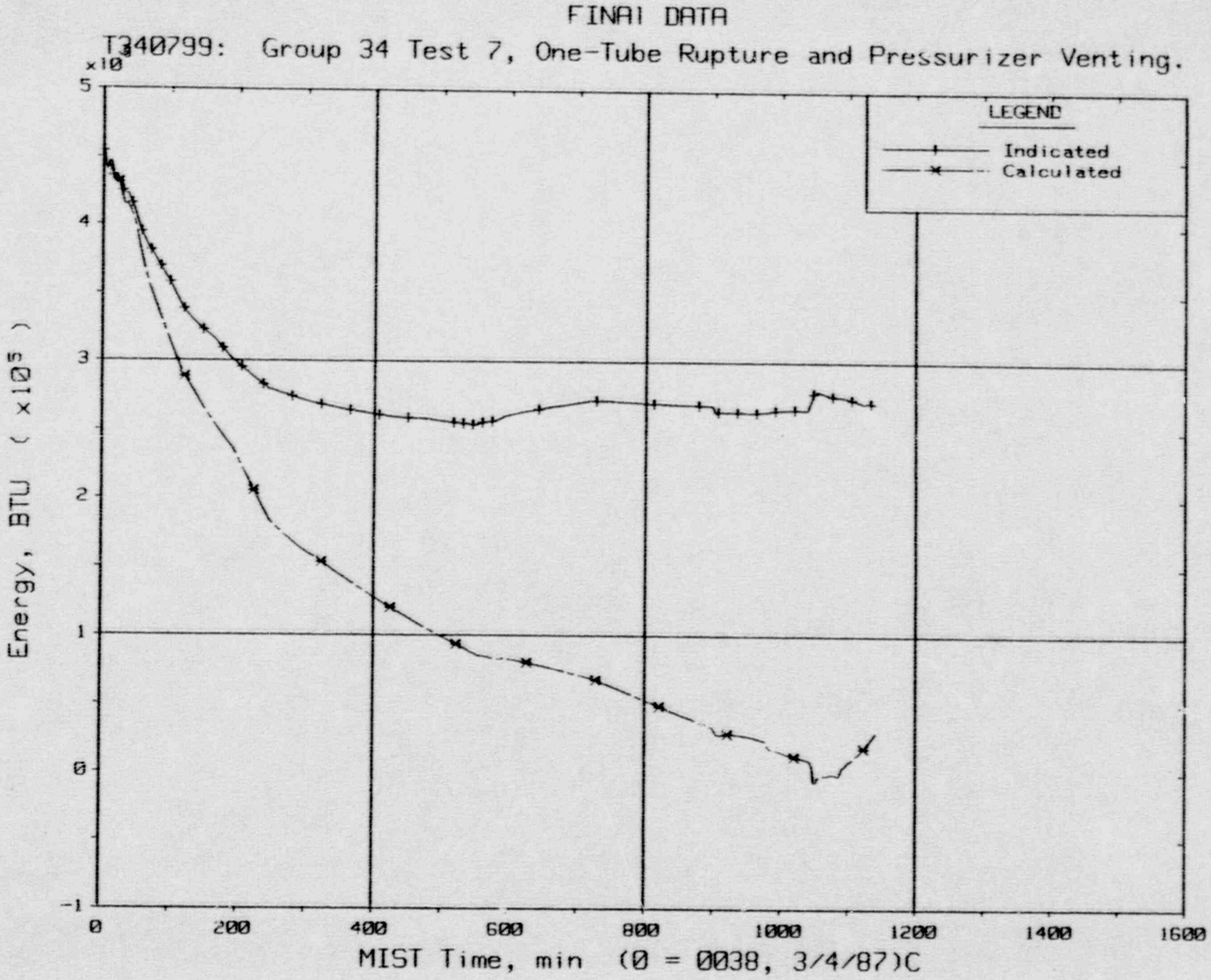
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Key Temperature Differences.

Thu Jun 2 11:48:19 1988

PRTD2



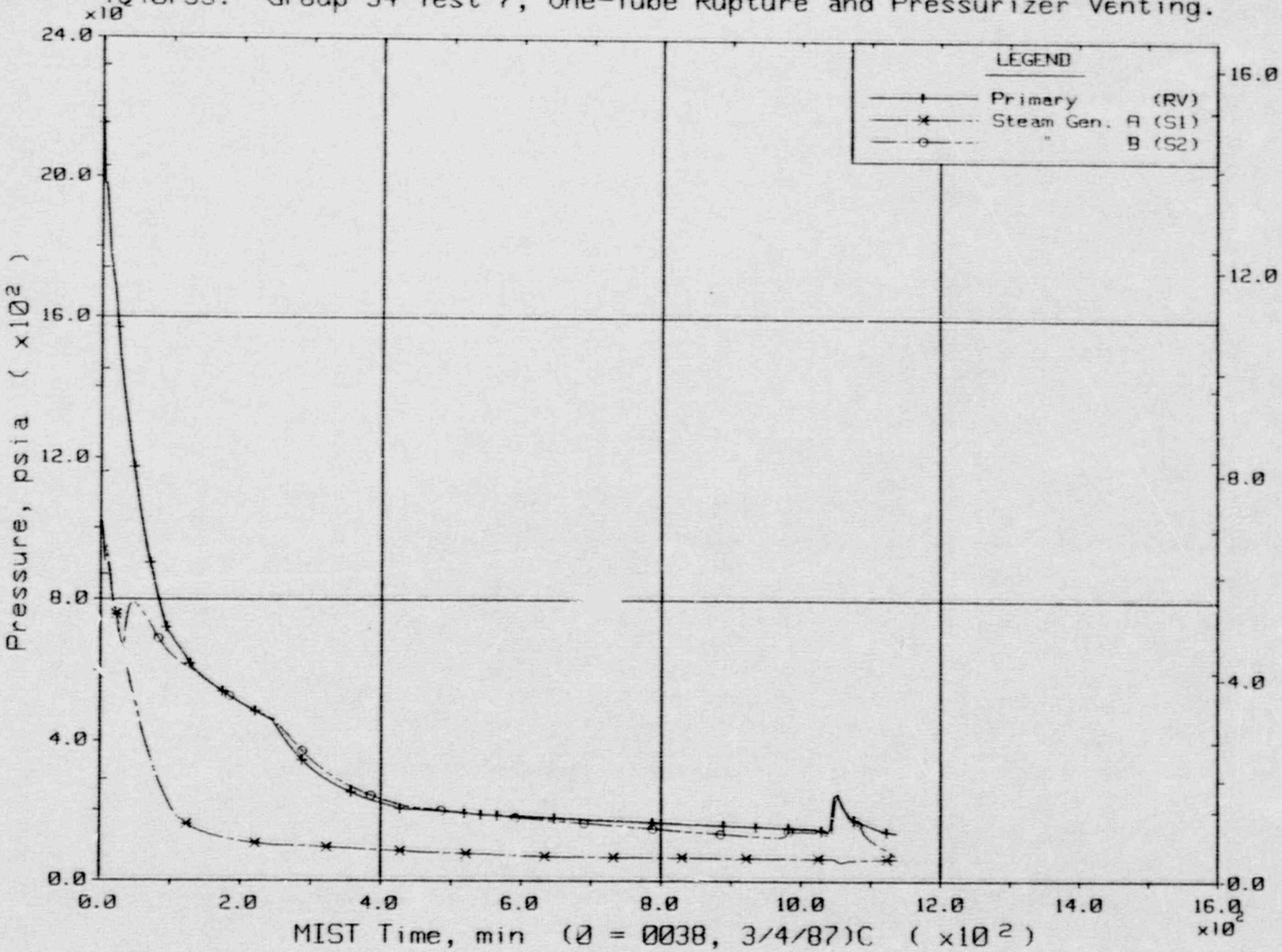
Primary System Total Fluid Energy.

Fri Nov 4 14:11:00 1988

PRTE2

FINAL DATA

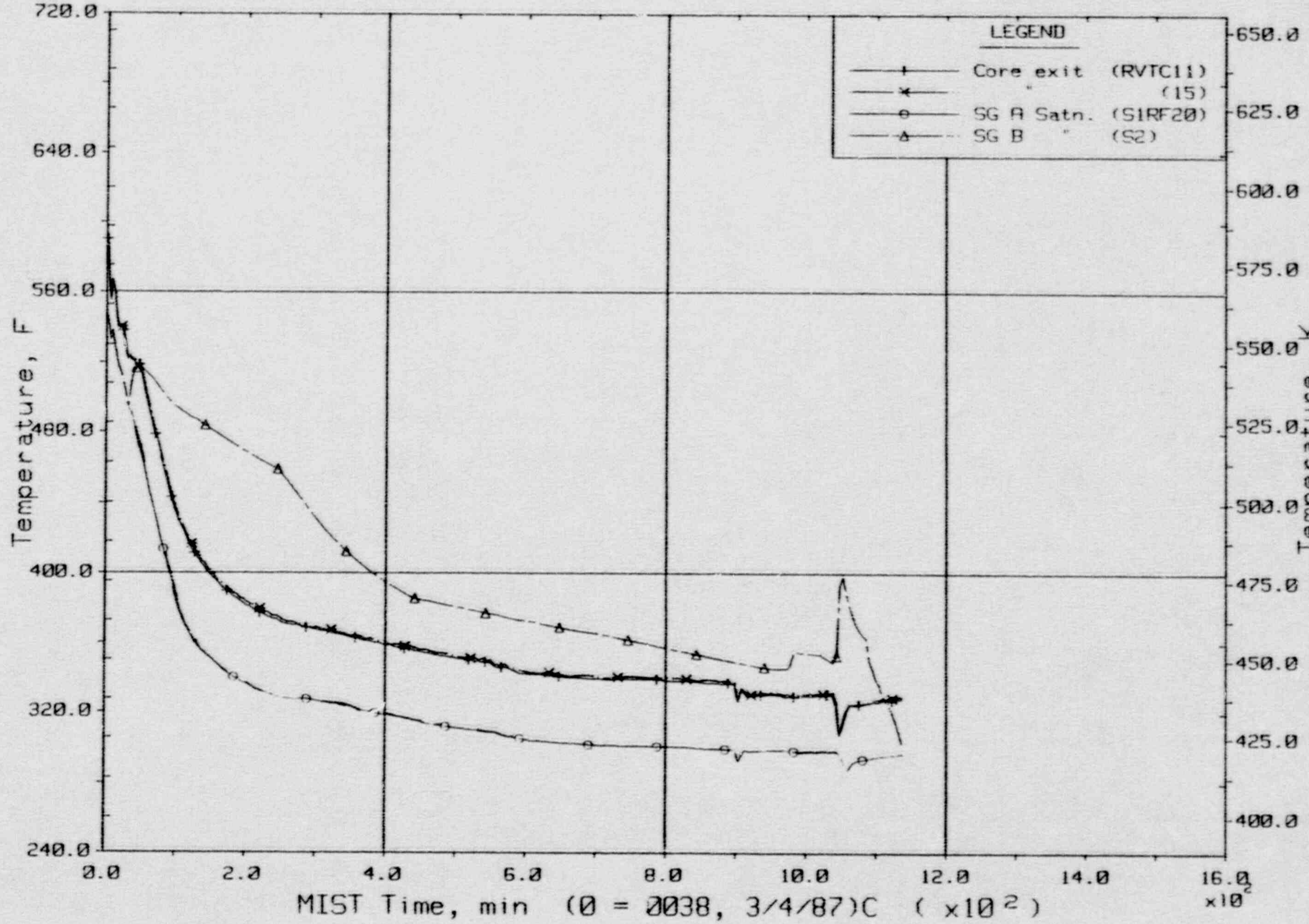
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Primary and Secondary System Pressures (GP01s).

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



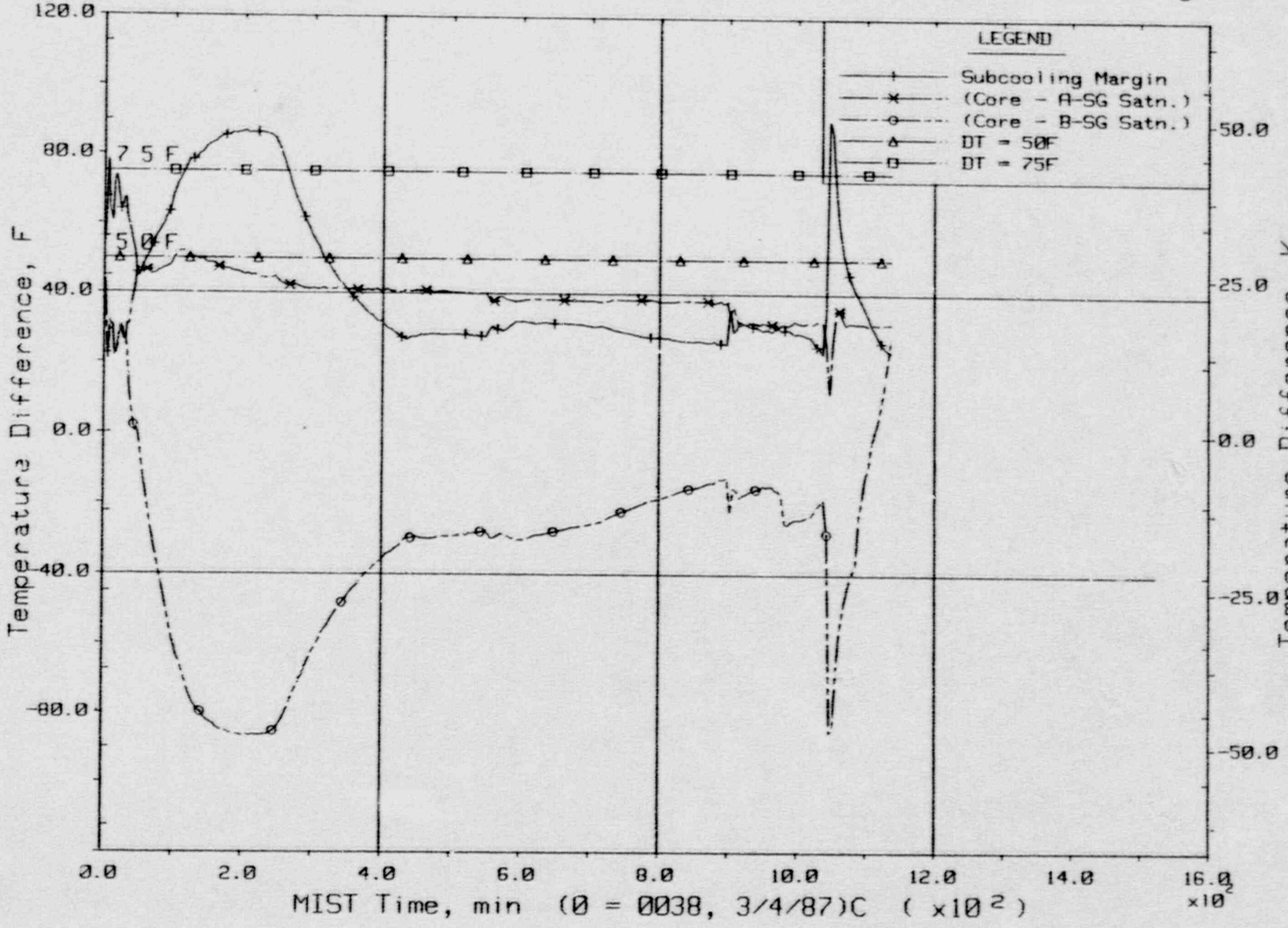
Core Exit and Steam Generator Secondary Saturation Temperatures.

Thu Jun 2 11:56:01 1988

PSTC3p

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



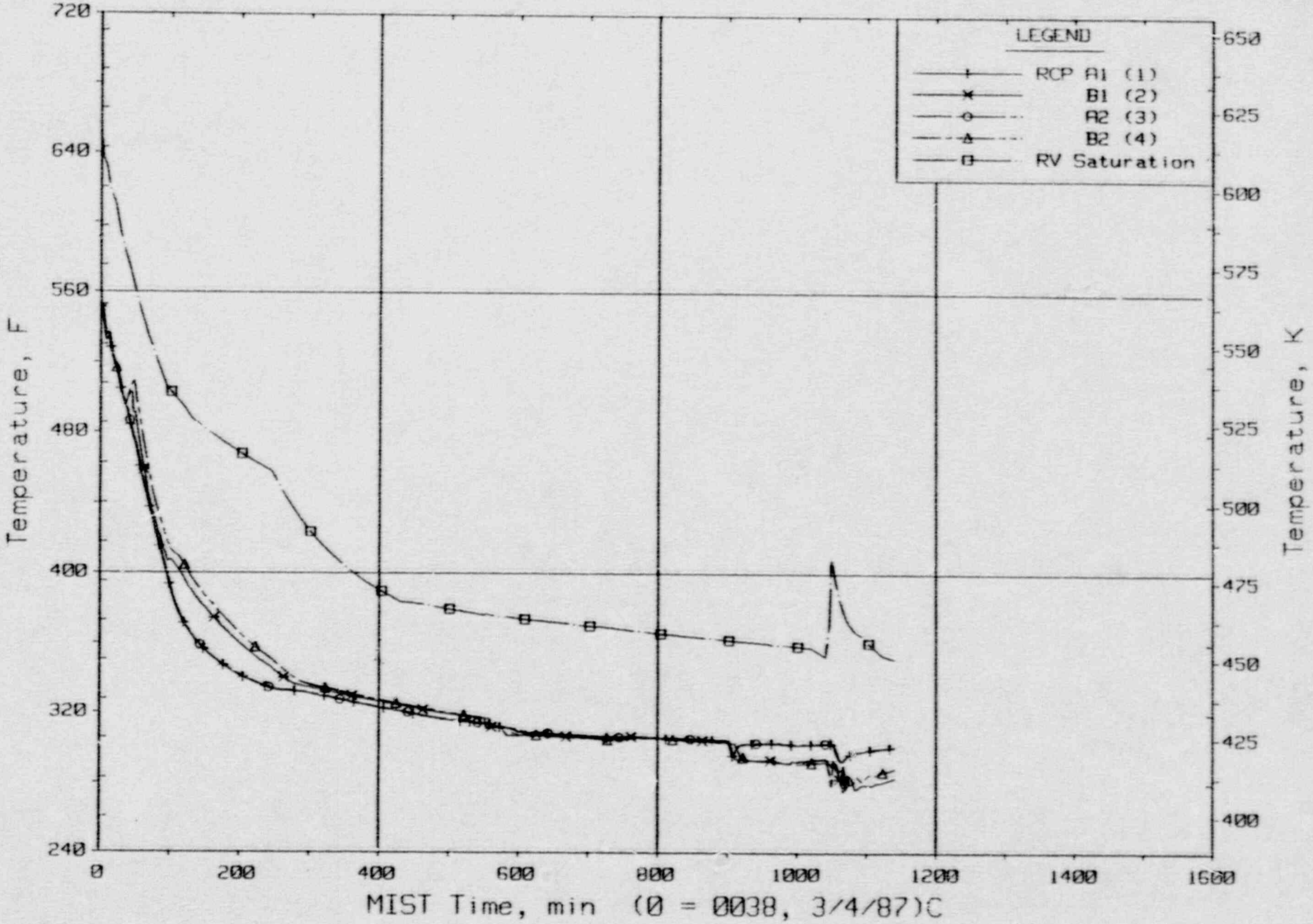
Control Temperature Differences.

Thu Jun 2 11:59:23 1988

PSTD1

FINAi DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



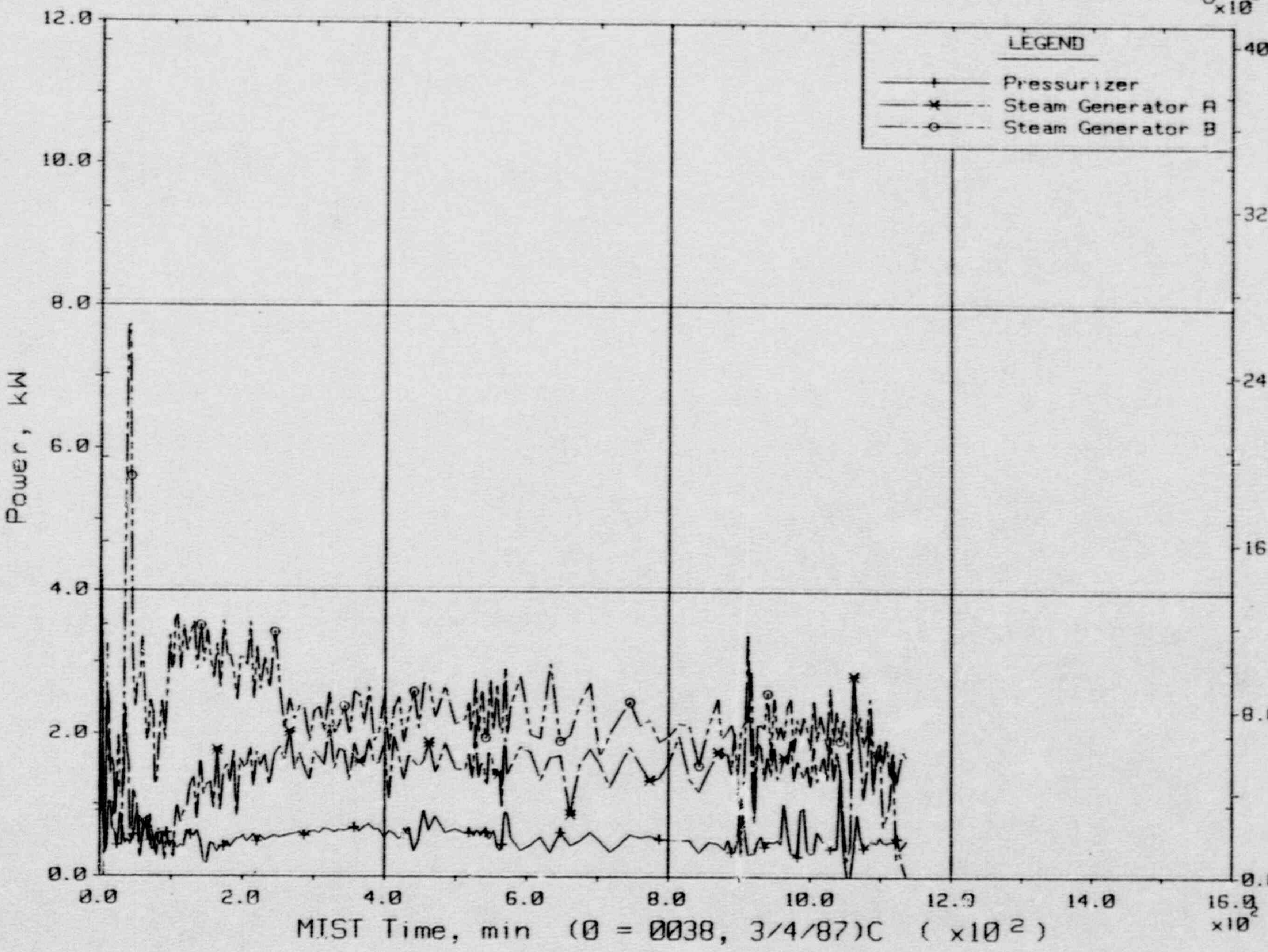
Pump Suction Fluid Temperature (CnRT01s).

Tue Nov 1 15:47:02 1988

PURT0

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



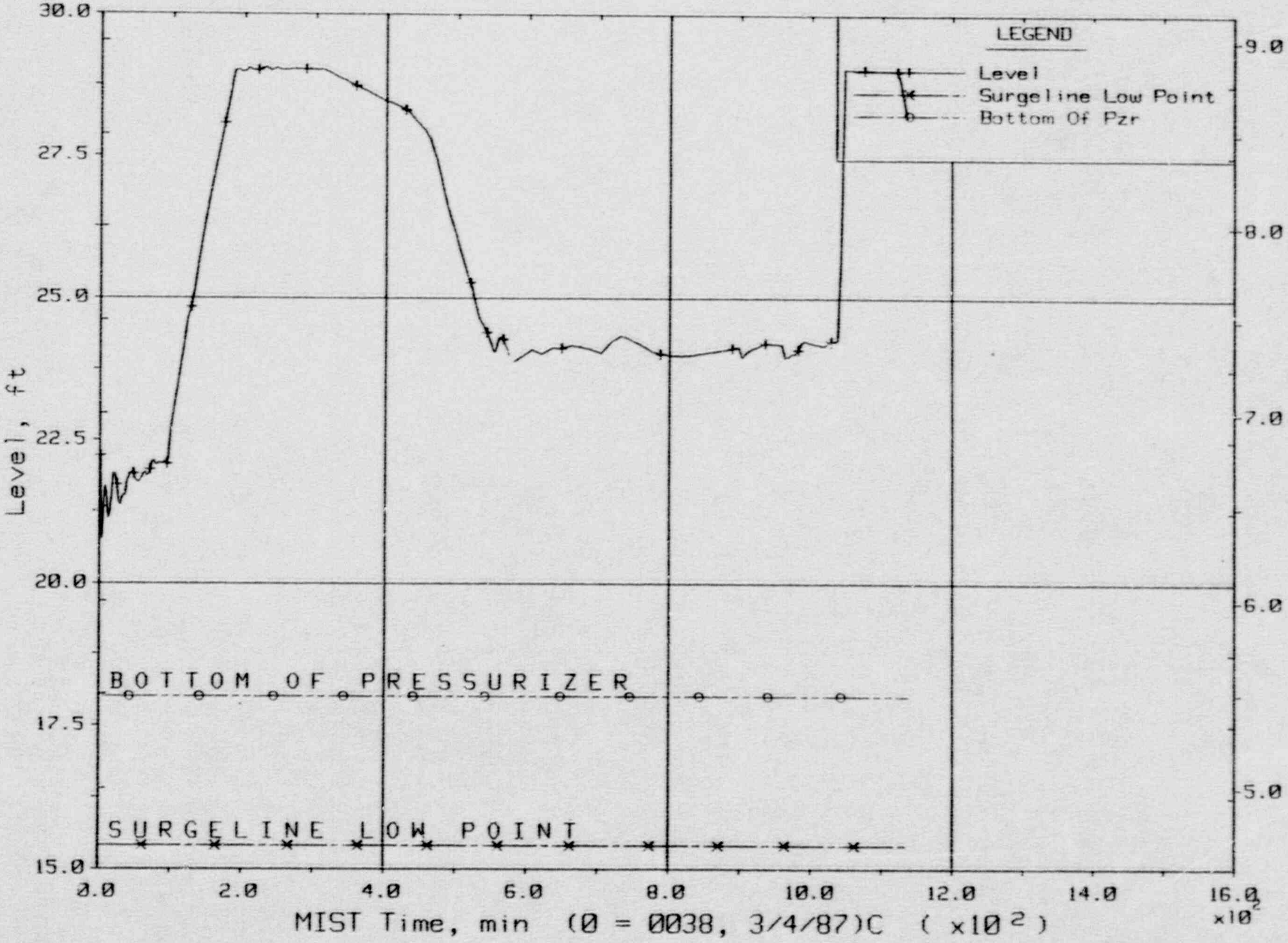
Guard Heater Specified Power, Pressurizer and Steam Generators.

Thu Jun 2 12:06:12 1988

PZG01

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



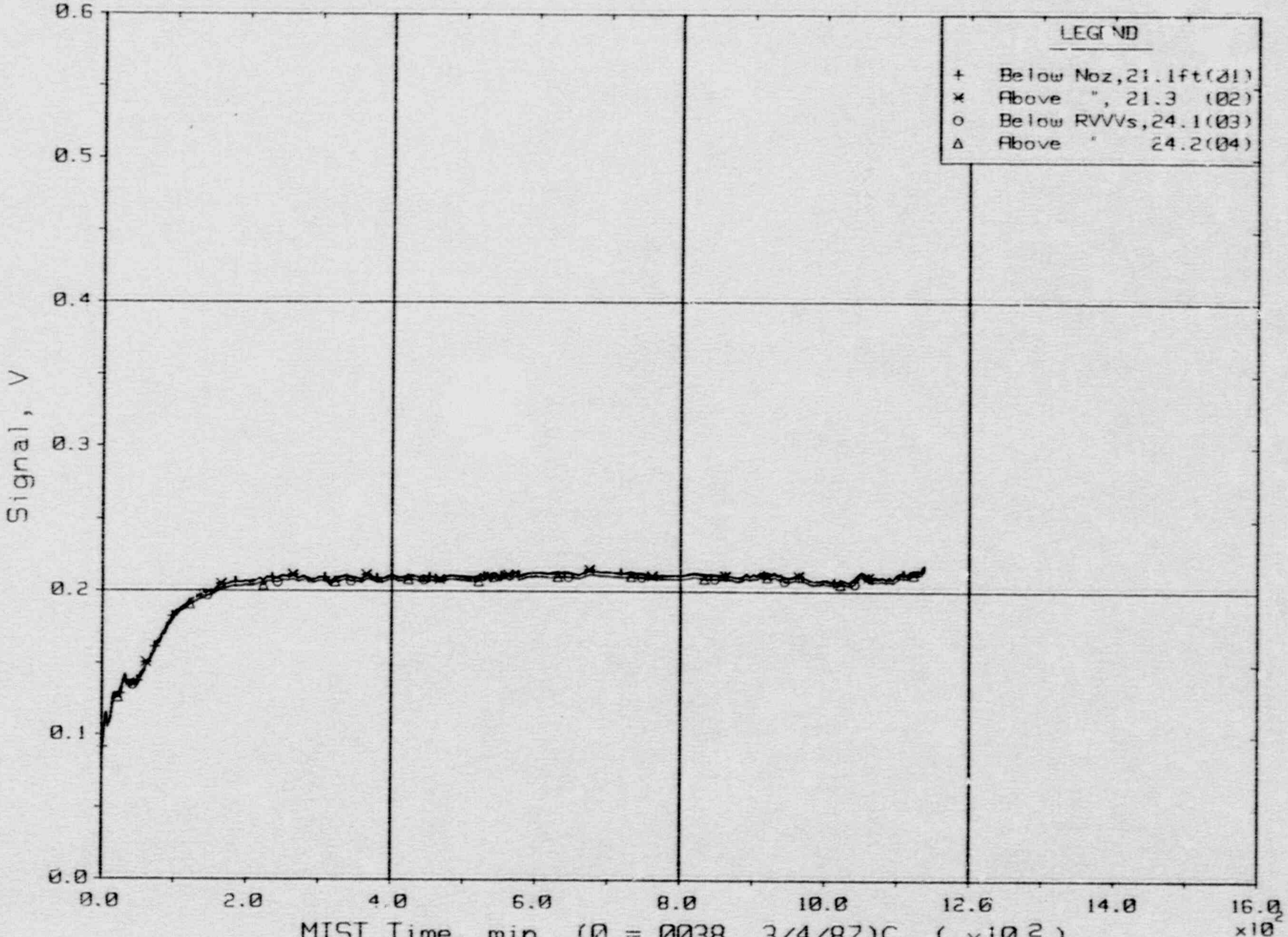
Pressurizer Collapsed Liquid Level (PZLV20).

Thu Jun 2 12:09:19 1988

PZLV1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



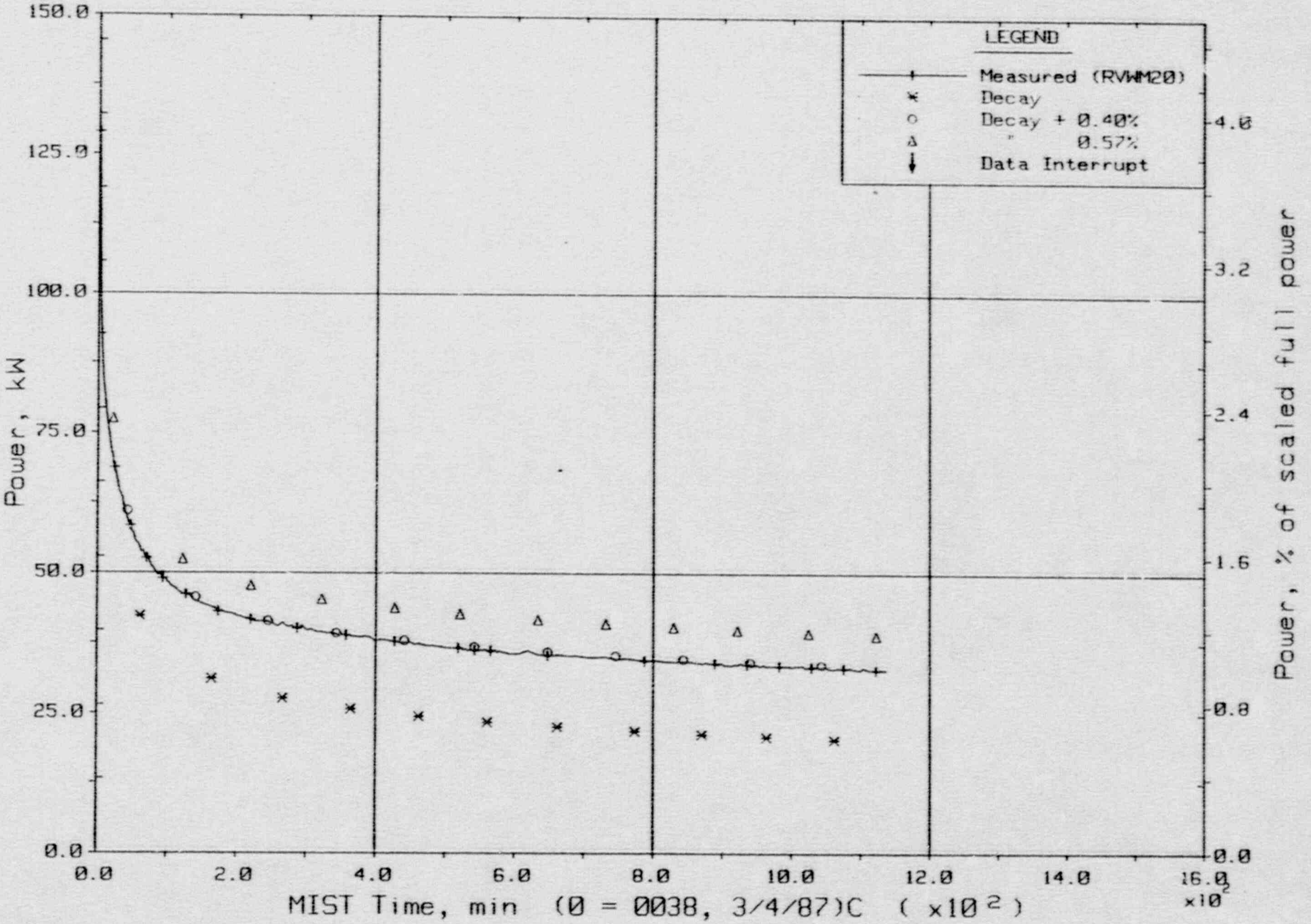
Reactor Vessel Conductivity Probe Signals (RVCPs).

Wed May 18 17:46:10 1988

RVCP1

FINAL DATA

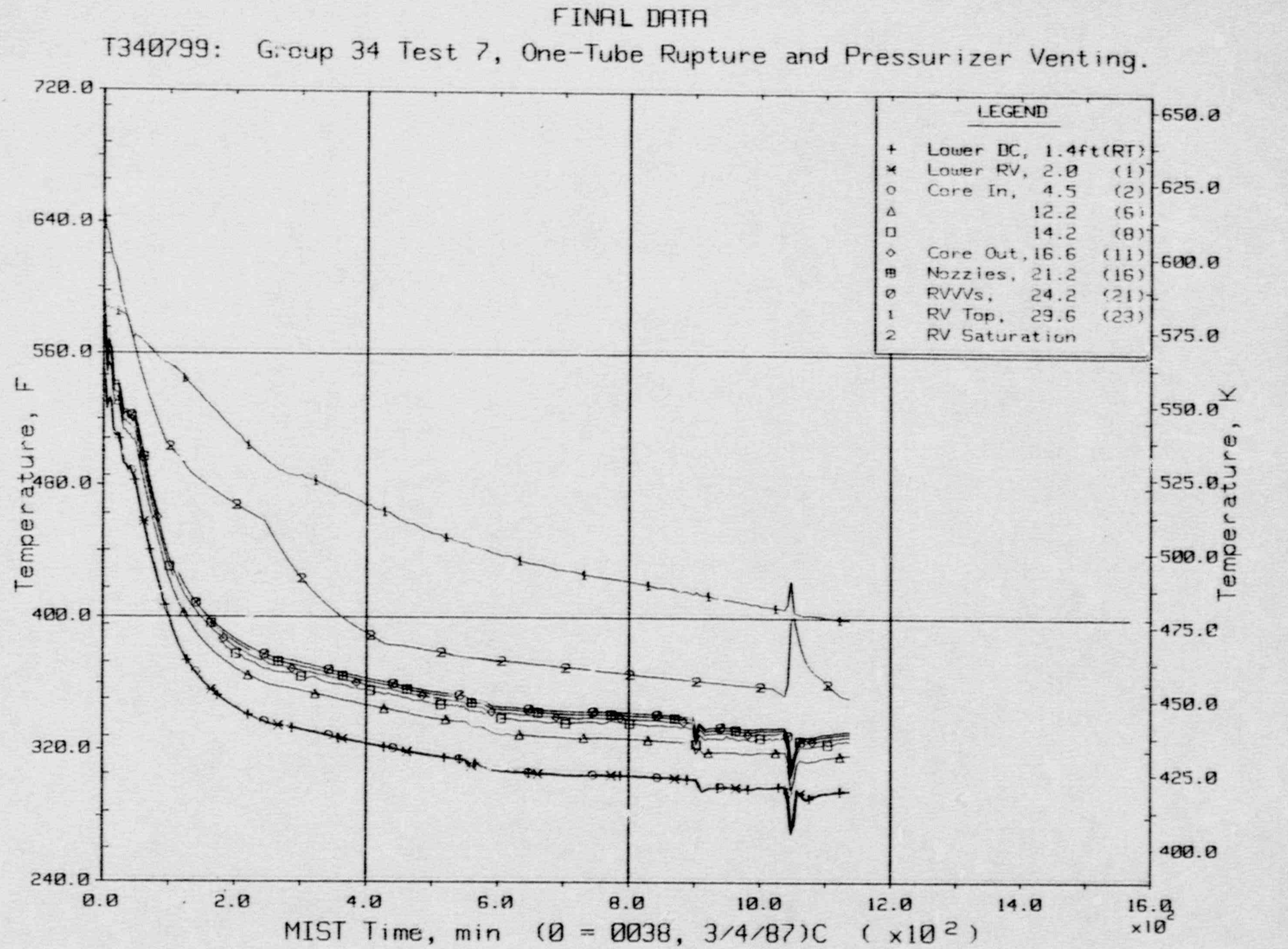
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Core Power.

Thu Jun 2 12:12:47 1988

RVKW0



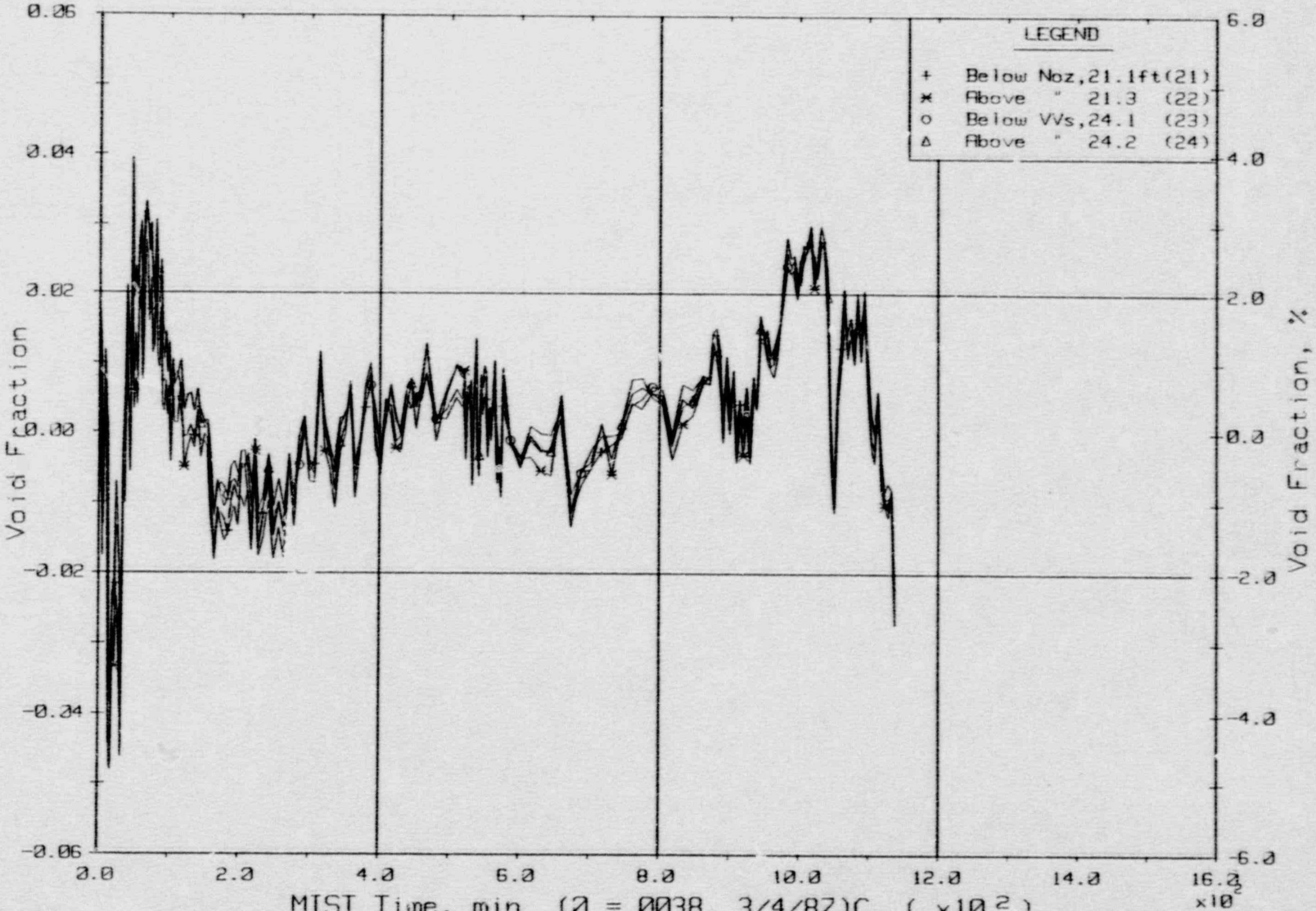
Core Unit Cell and Reactor Vessel Fluid Temperatures (RVTCs).

Thu Jun 2 12:17:11 1988

RVTC0

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



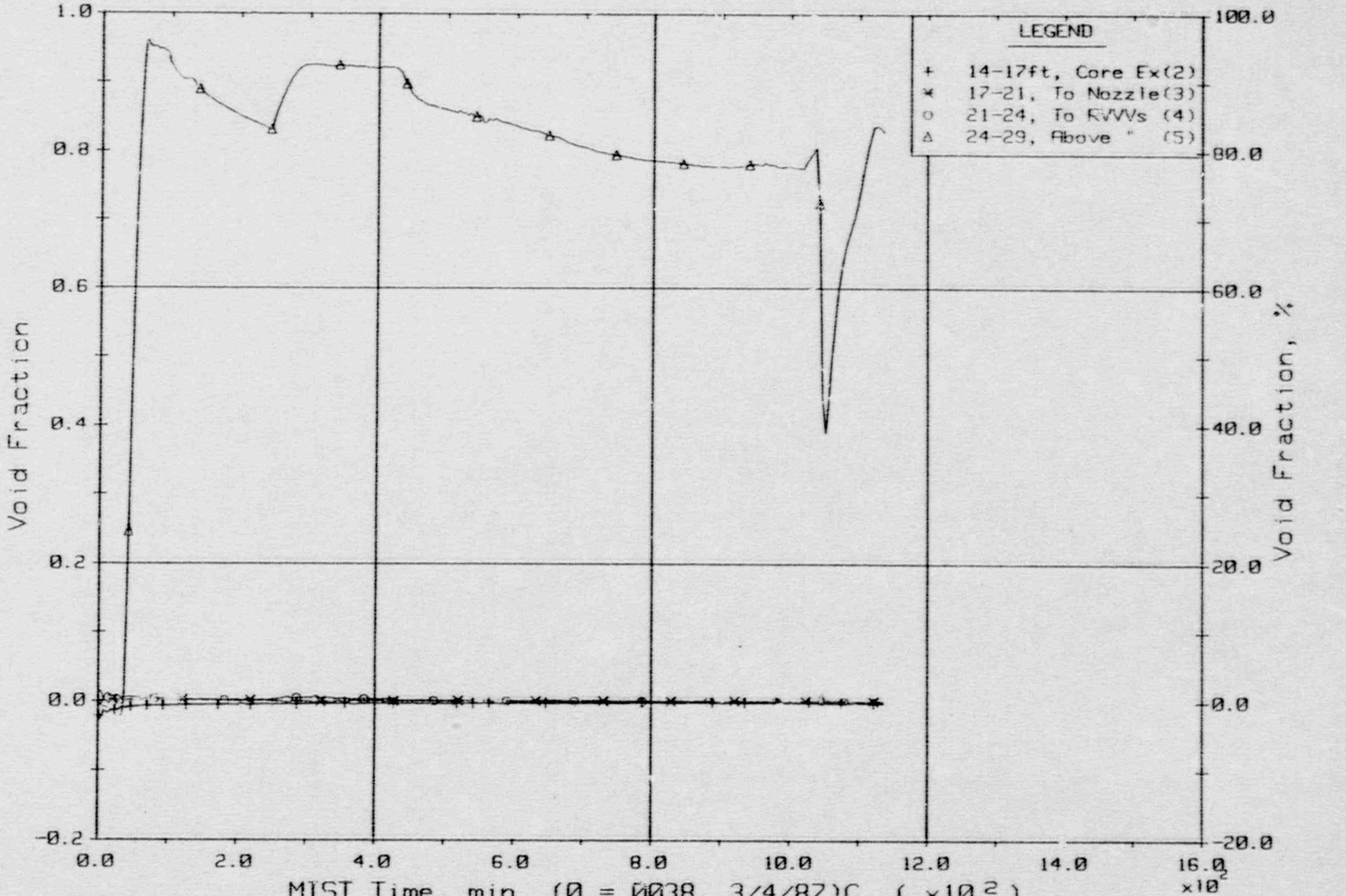
Reactor Vessel Void Fraction's From Conductivity (RVCPs).

Wed May 18 17:48:31 1988

RVVC1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



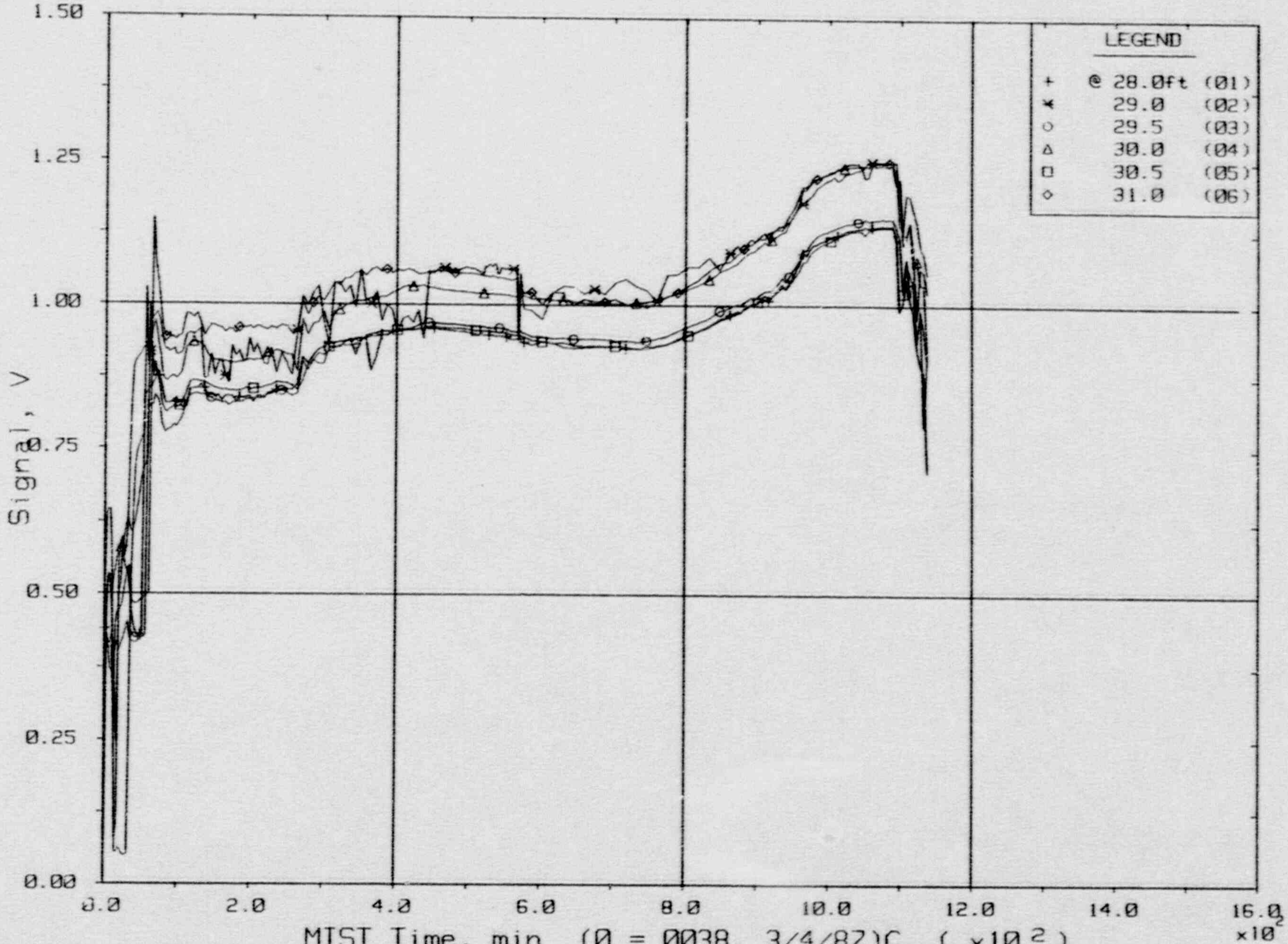
Reactor Vessel Void Fractions From Differential Pressures (RVVFs).

Thu Jun 2 12:20:47 1988

RVVF1

FINAL DATA

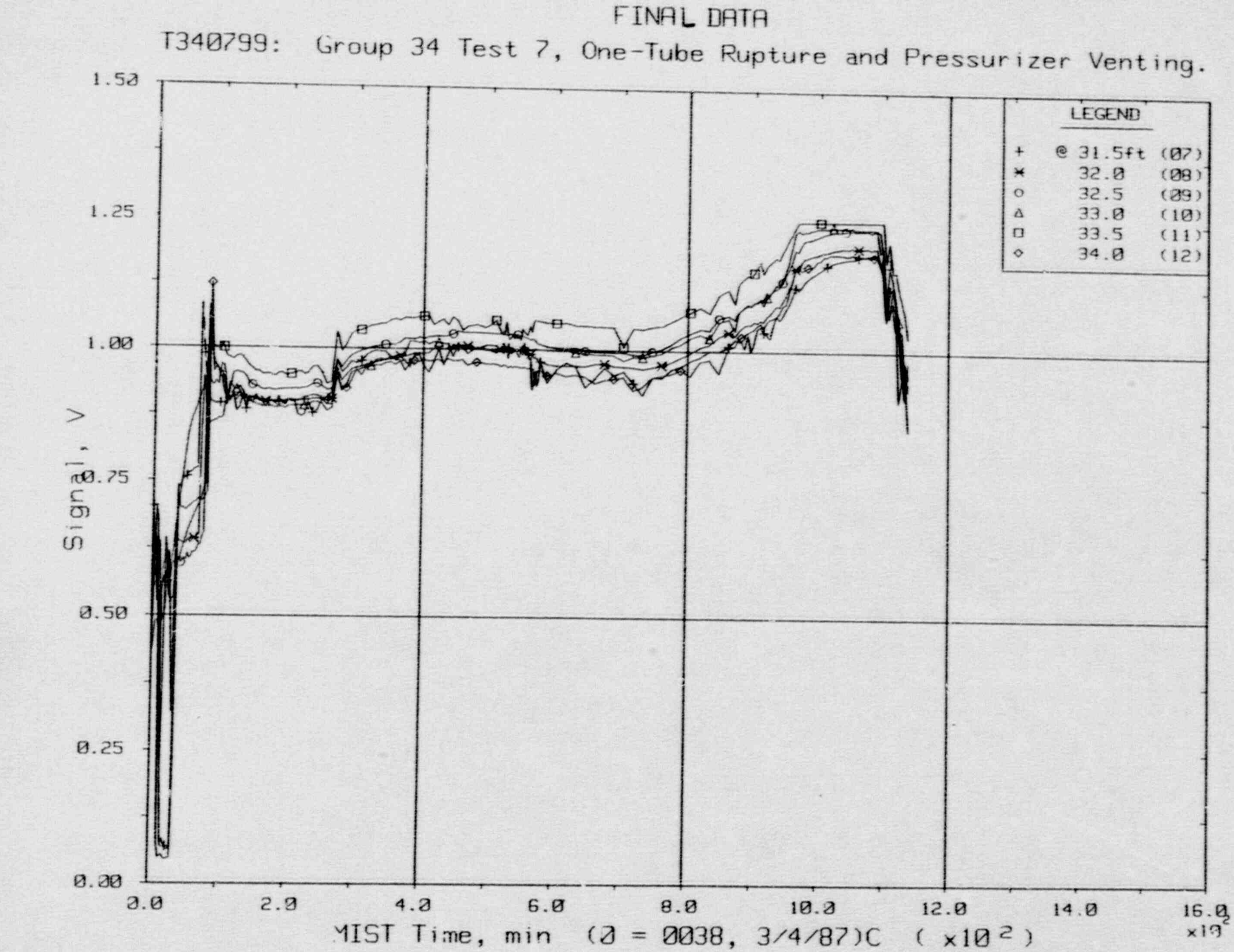
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



SG B Secondary Lower-Elevation Conductivity Probe Signals (S2CPs).

Wed May 18 17:51:11 1988

S2CP1



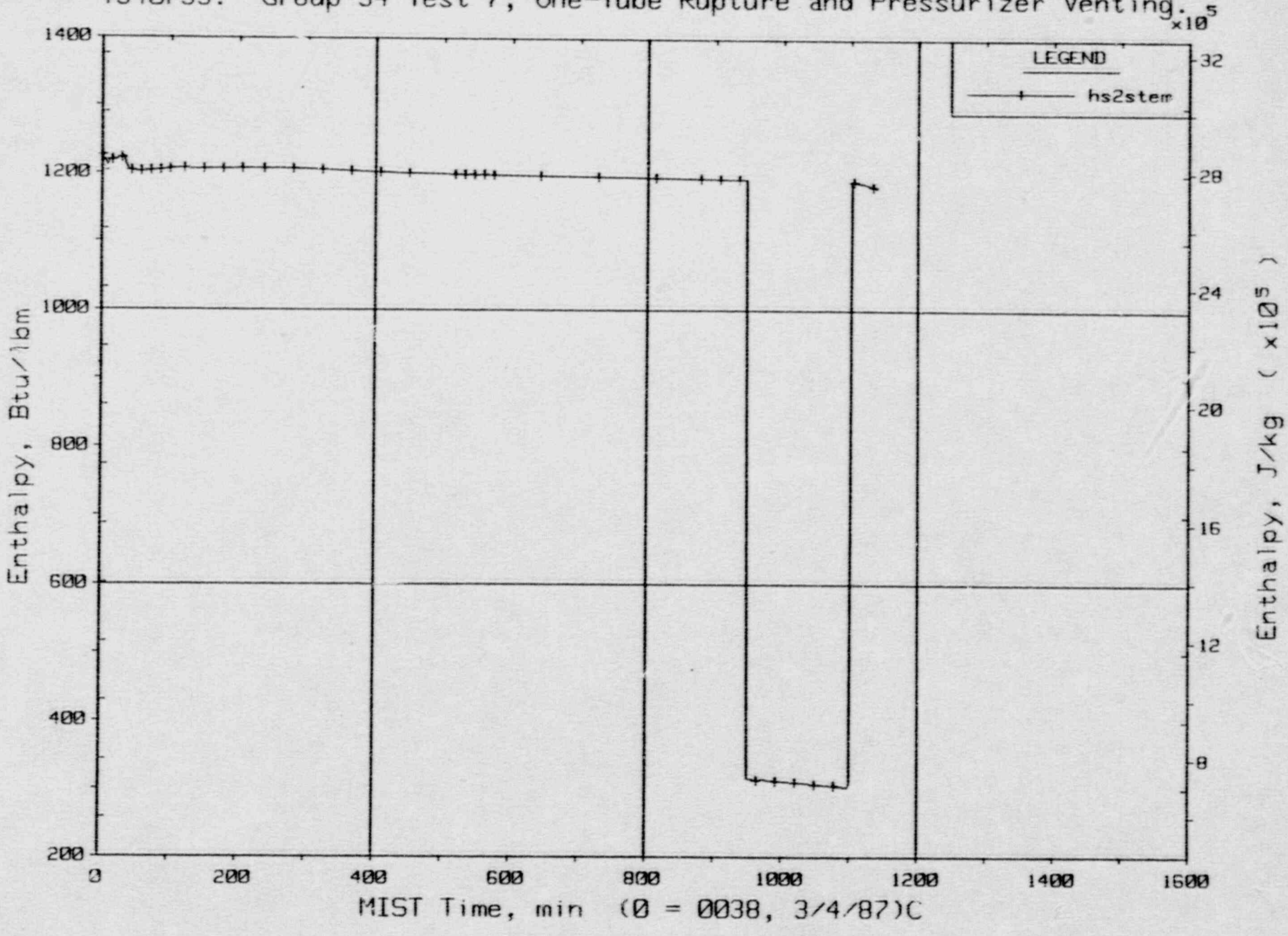
SG B Secondary Upper-Elevation Conductivity Probe Signals (S2CPs).

Wed May 18 17:53:38 1988

S2CP2

FINAl DATA

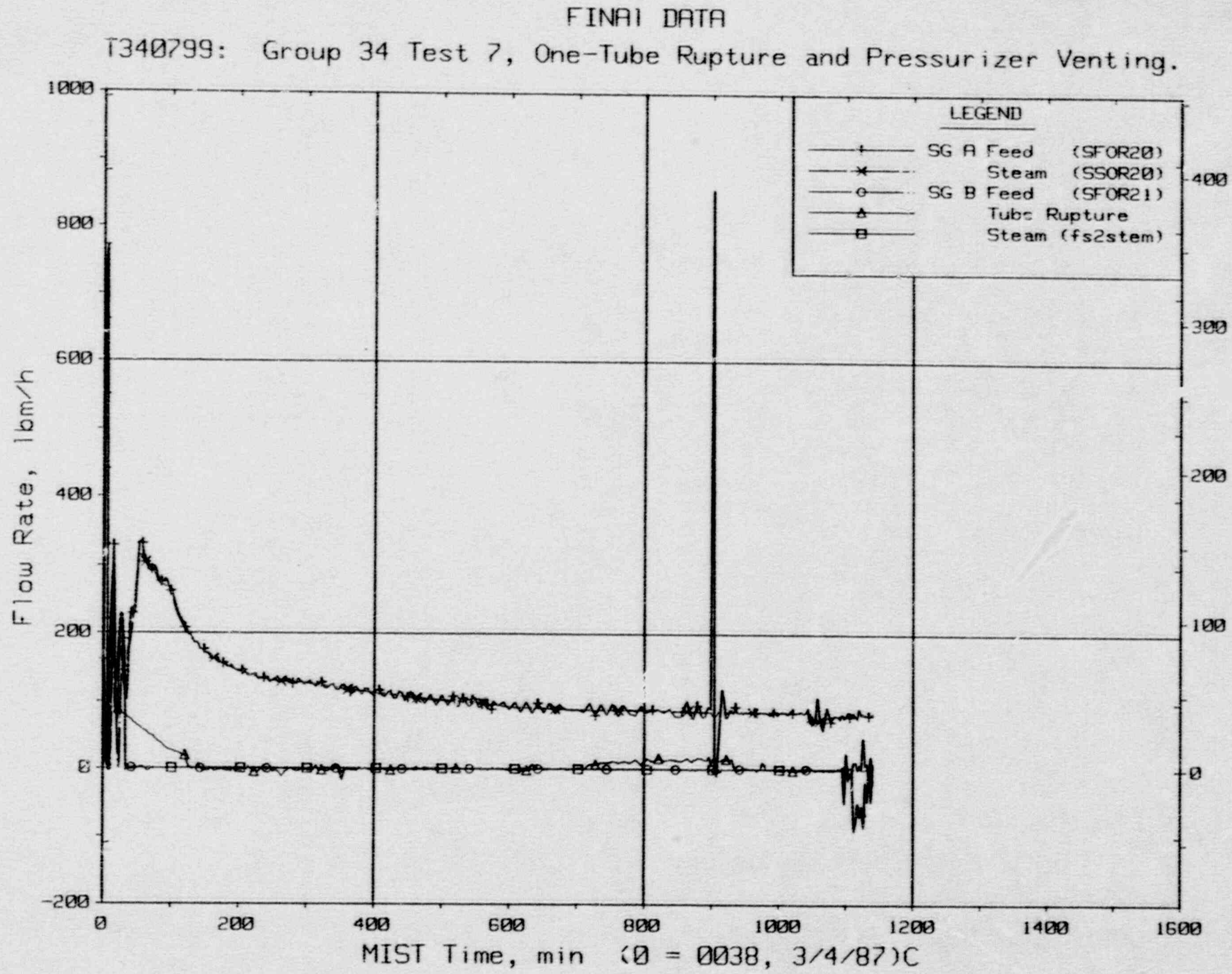
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Steam Generator B Steam Exit Enthalpy.

Fri Nov 4 14:42:49 1988

S2EN1



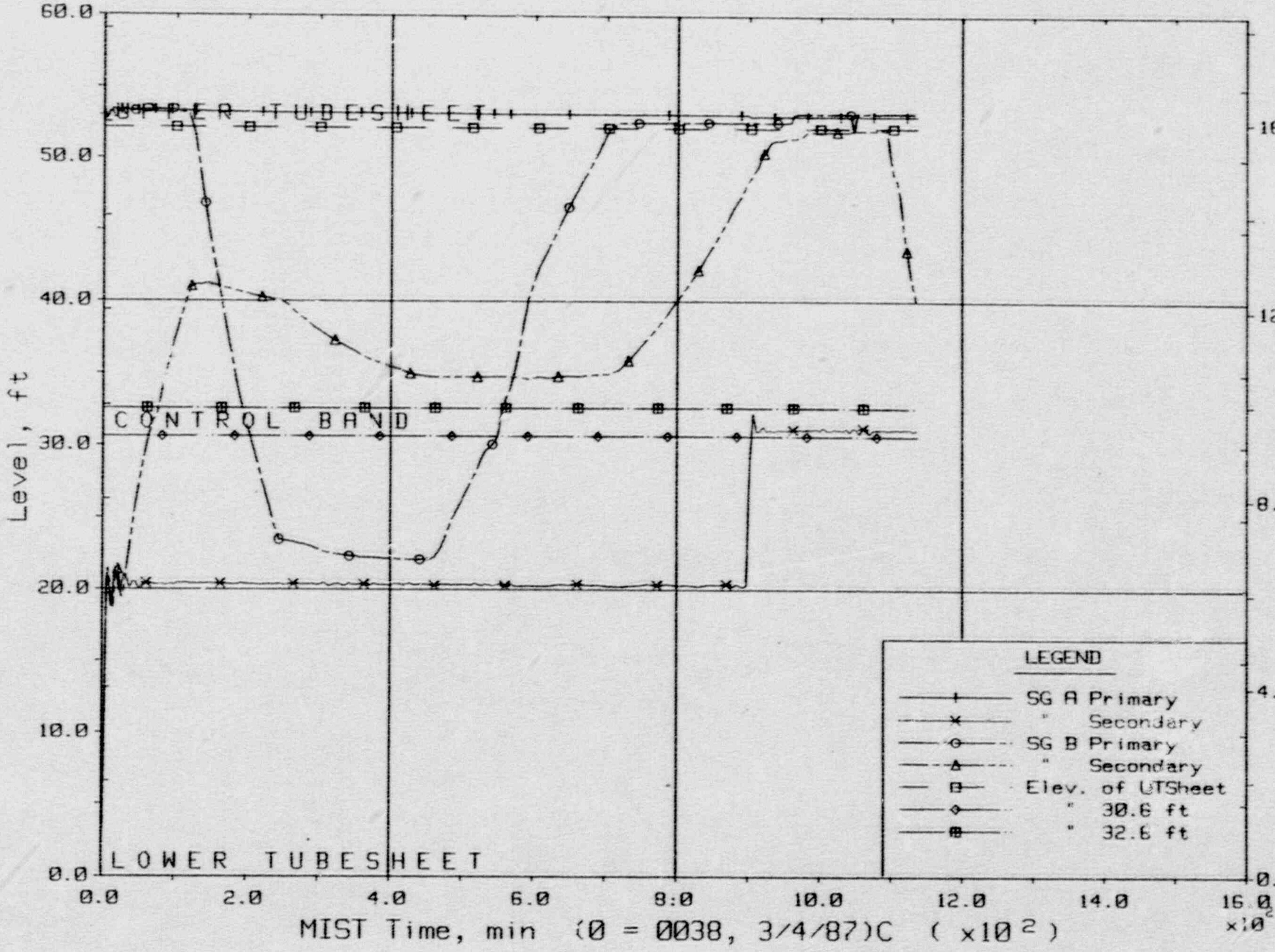
Steam Generator Secondary Flow Rates.

Fri Nov 4 14:49:38 1988

SGFL1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



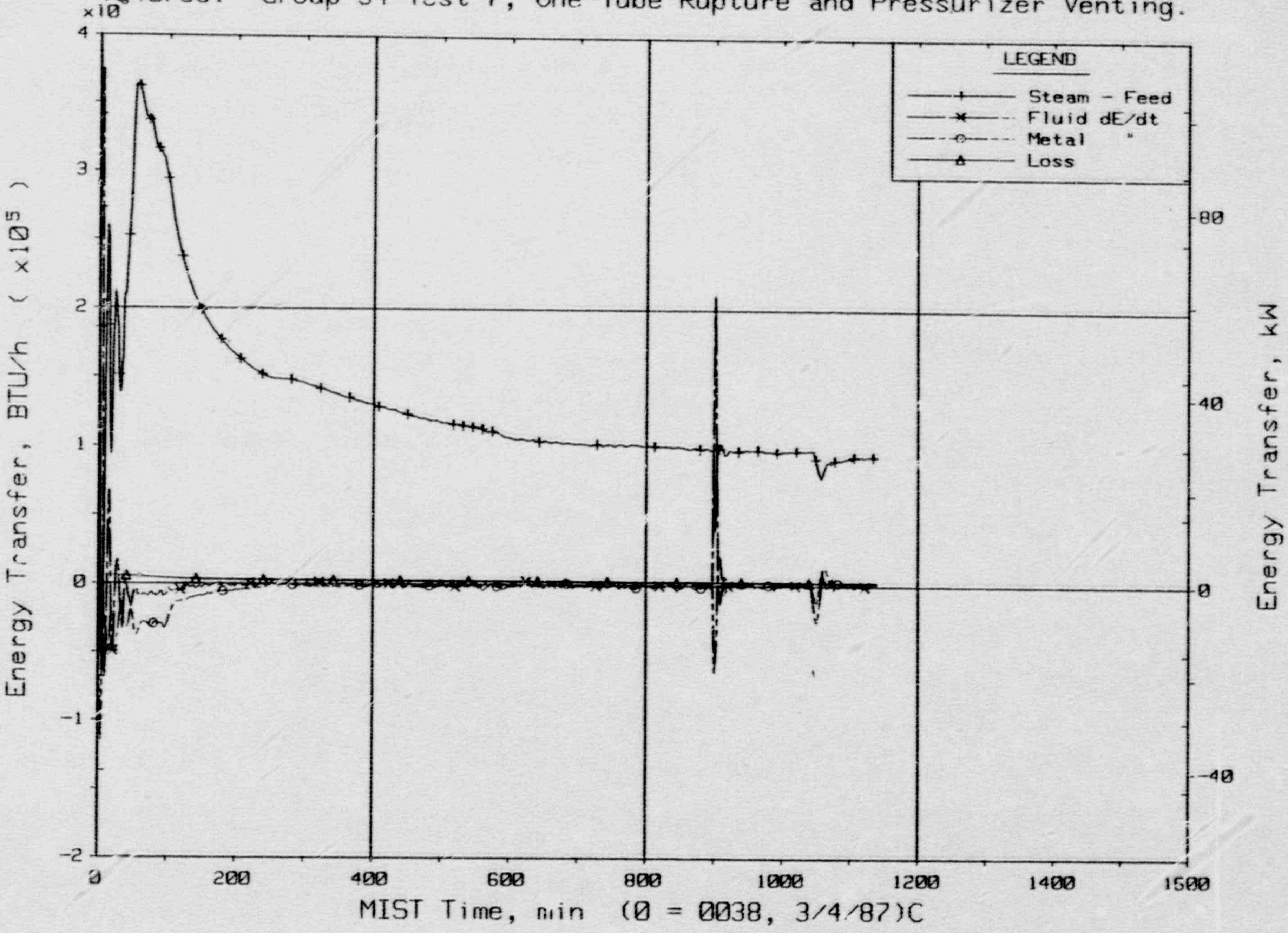
Steam Generator Collapsed Liquid Levels.

Thu Jun 2 12:27:25 1988

SGLV2

FINAl DATA

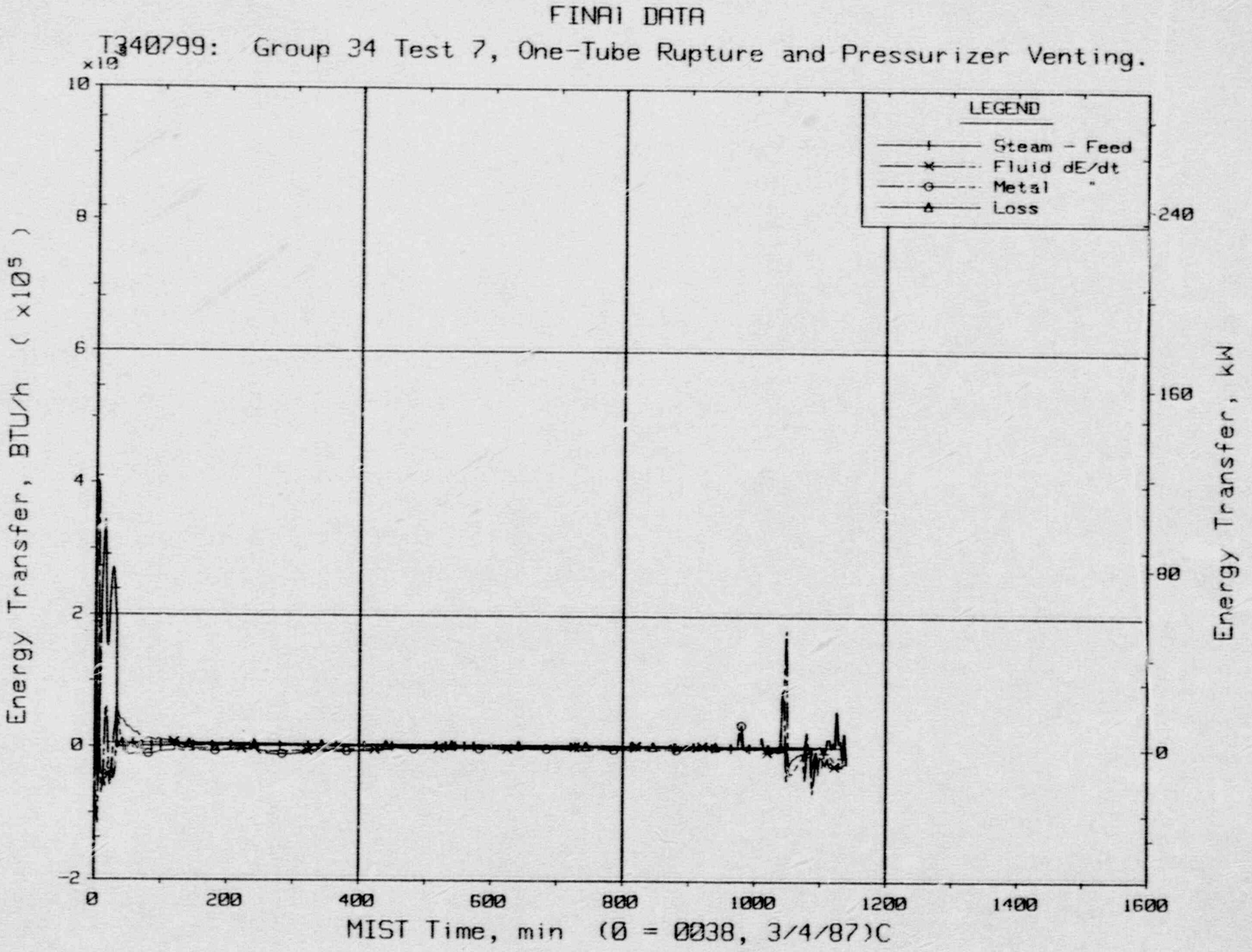
T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Steam Generator A Energy Transfer.

Fri Nov 4 14:25:36 1988

SGQE3



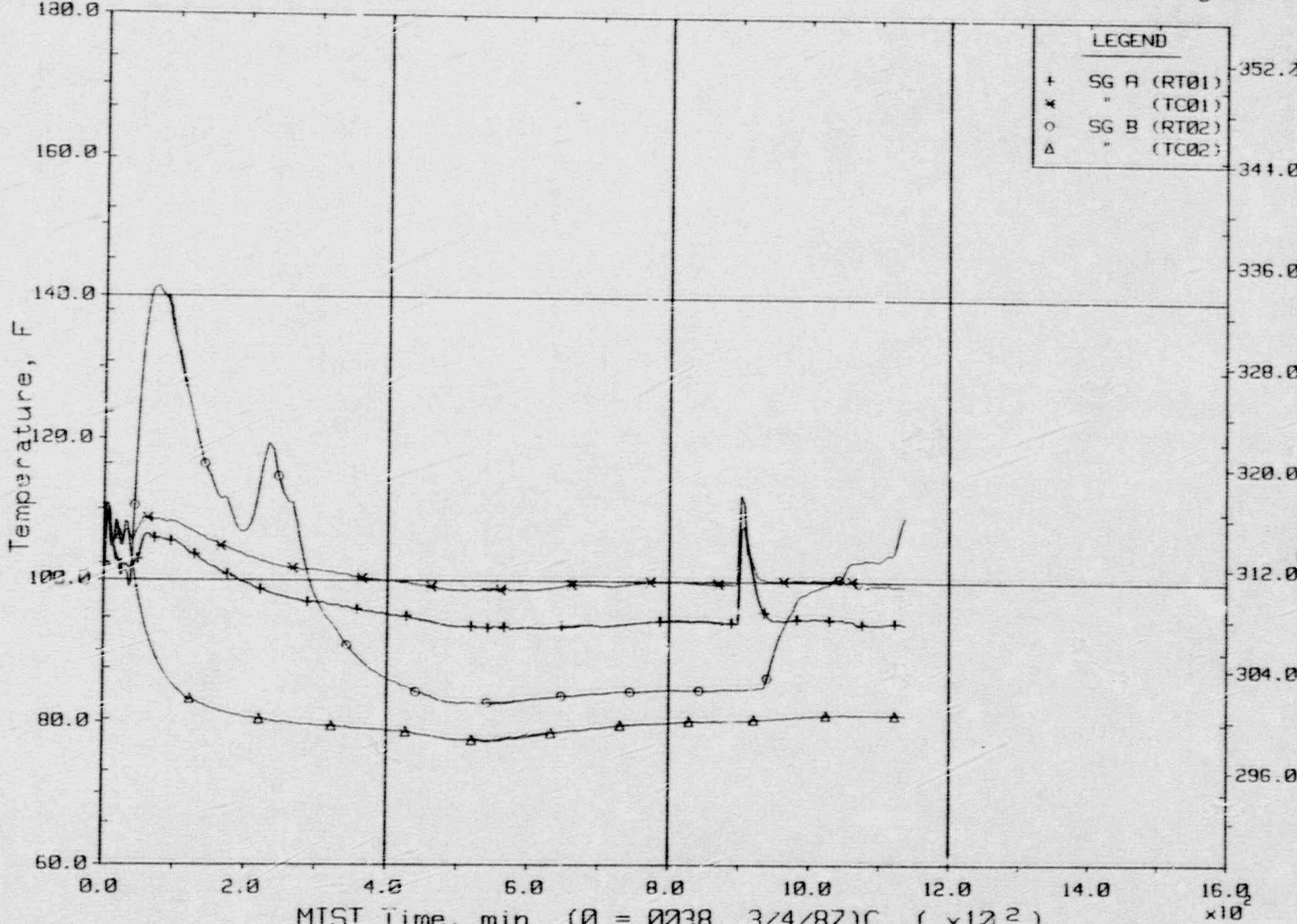
Steam Generator B Energy Transfer.

Fri Nov 4 14:38:38 1988

SGQE6

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



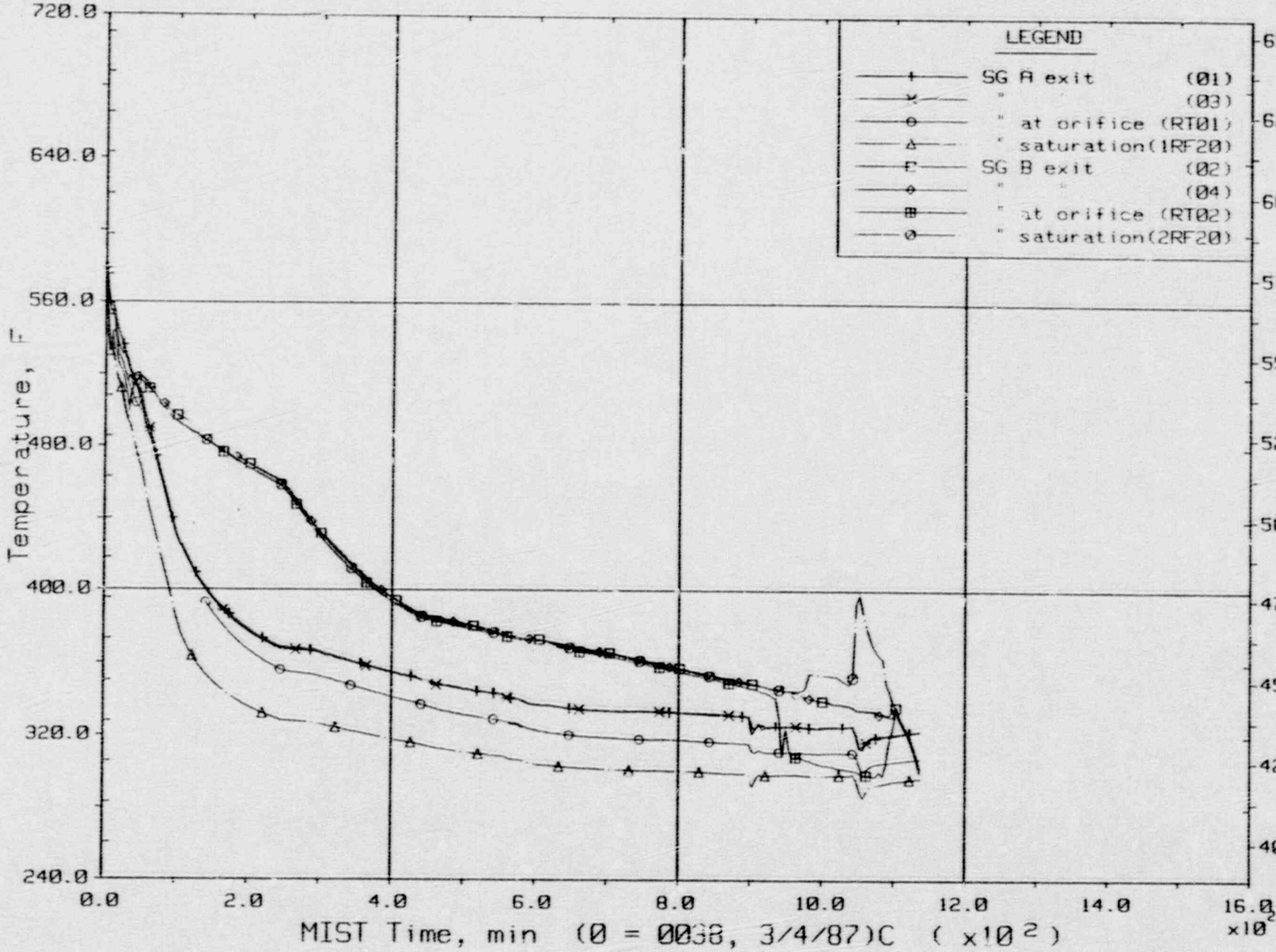
Feedwater Temperatures (SFs).

Thu Jun 2 12:53:48 1988

SGRT1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



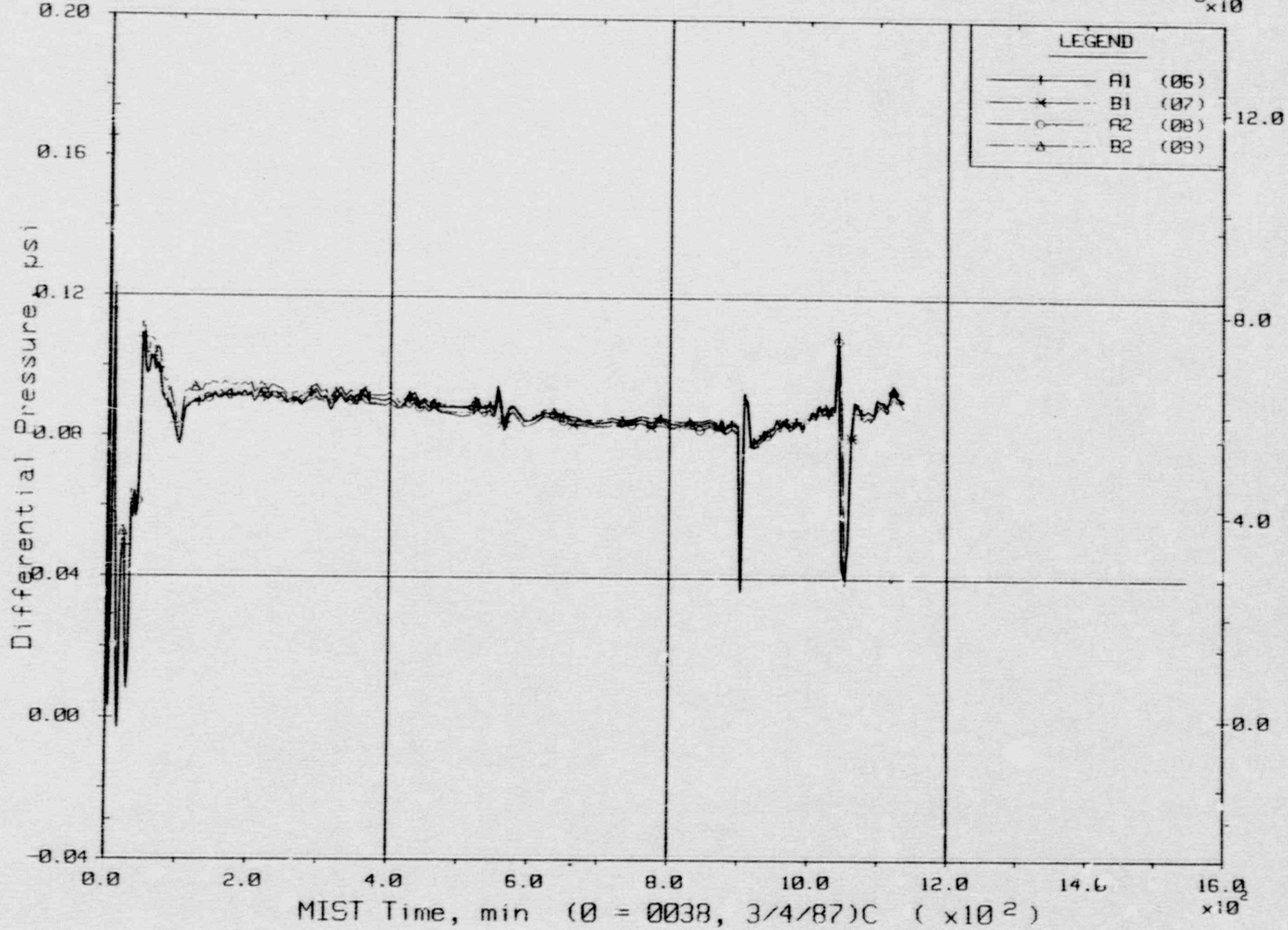
Steam Generator Steam Outlet Temperatures (SSTCs).

Thu Jun 2 13:13:13 1988

SGRT2

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



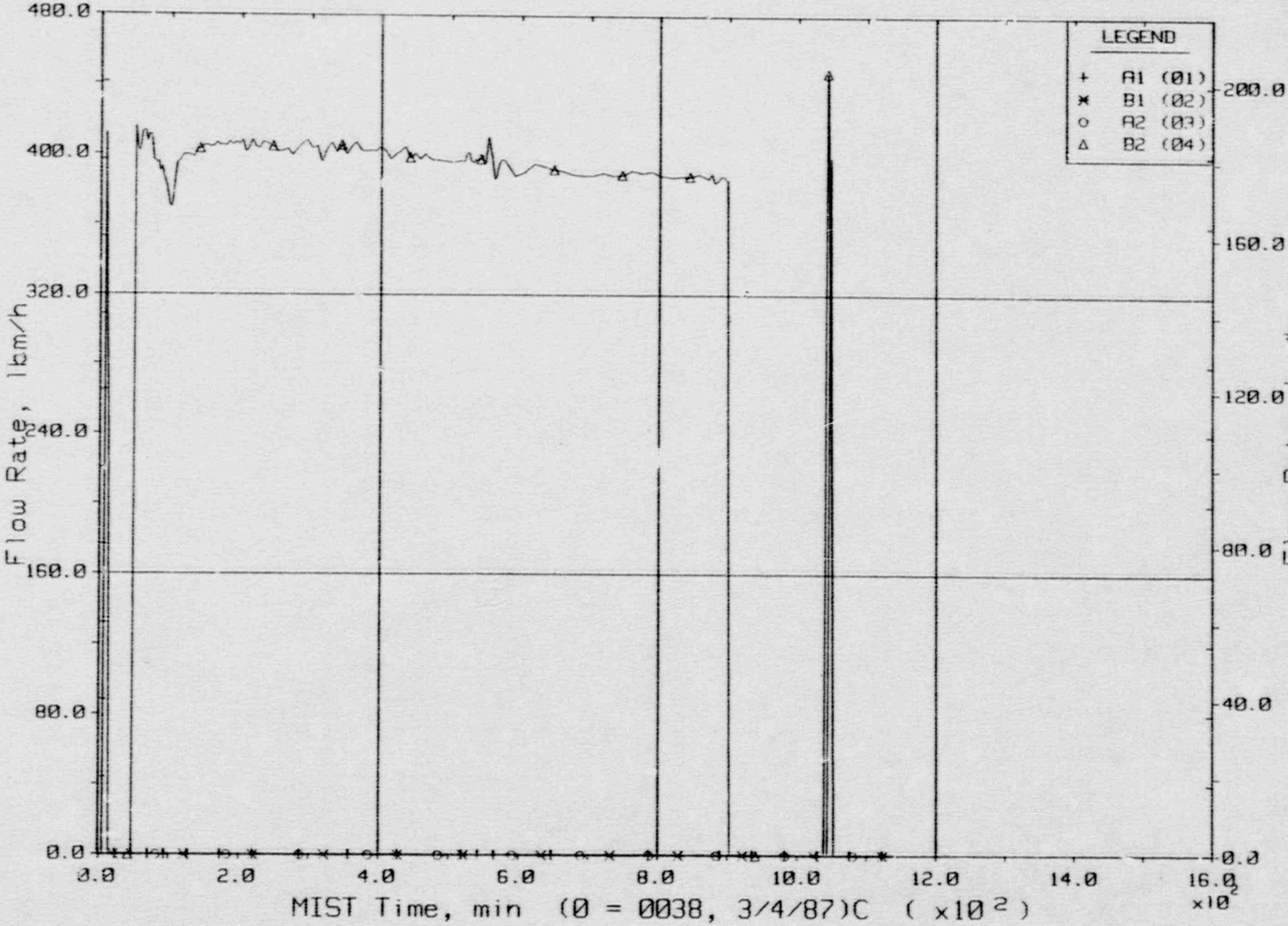
Reactor Vessel Vent Valve Differential Pressures (RVDPs).

Thu Jun 2 13:16:20 1988

VVDP1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



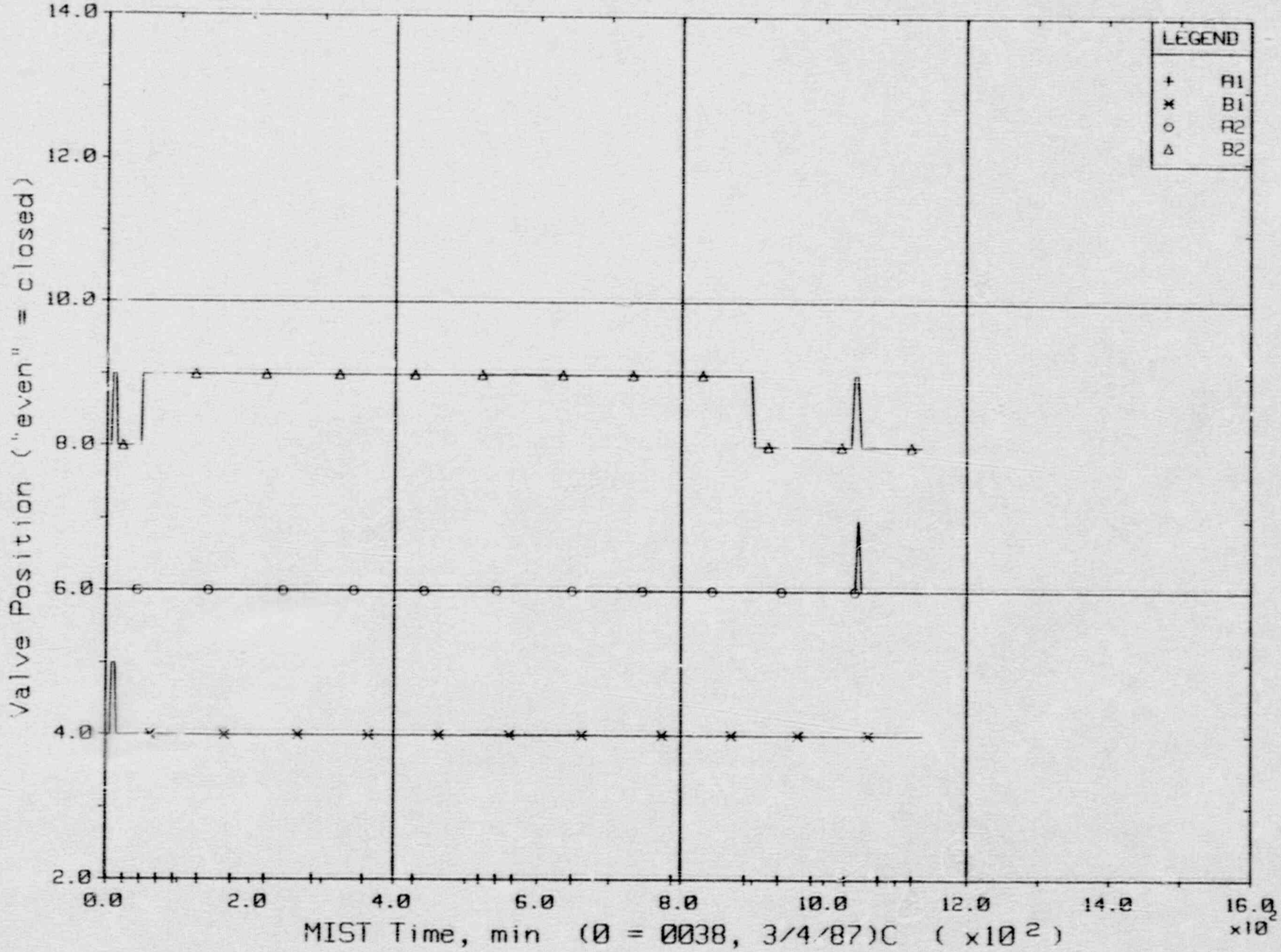
Reactor Vessel Vent Valve Flow Rates (RVGRs).

Thu Jun 2 13:19:01 1988

VVFL1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



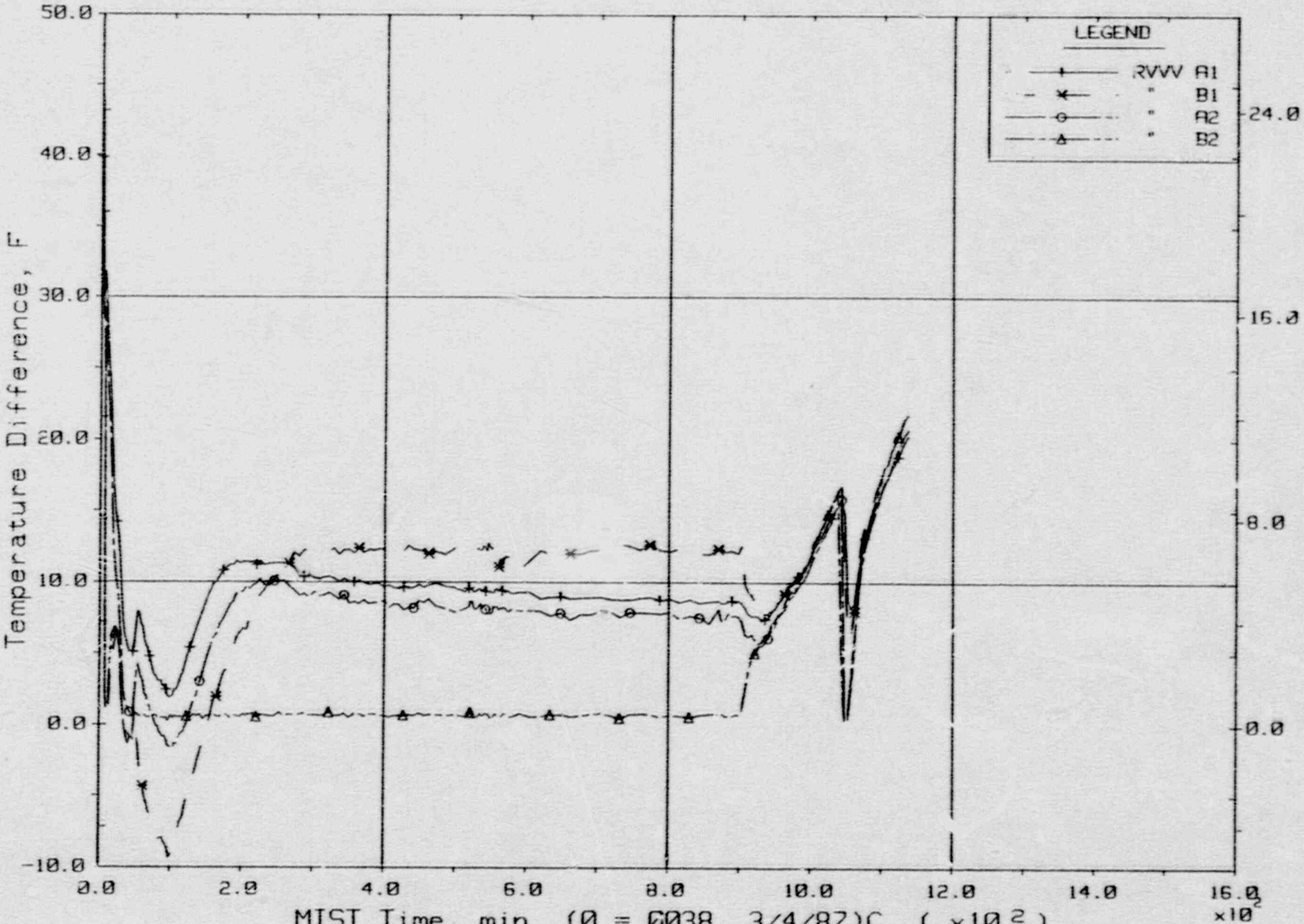
Reactor Vessel Vent Valve Positions.

Thu Jun 2 13:21:40 1988

VVLS1

FINAL DATA

T340799: Group 34 Test 7, One-Tube Rupture and Pressurizer Venting.



Temperature Differences Across Vent Valves.