Consumers Power Company

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

COPY

July 14, 1980

Mr James G Keppler
Office of Inspection and Enforcement
Region III
US Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

DOCKET 50-155 - LICENSE DPR-6 -
BIG ROCK POINT PLANT - RESPONSE TO
IE BULLETIN NO 80-17 - FAILURE OF 76 OF 185
CONTROL RODS TO FULLY INSERT DURING SCRAM
AT A BWR

IE Bulletin No 80-17, dated July 3, 1980 required for plants without ATWS related RPT, an analysis of the net safety of derating such that, in the event of an ATWS, calculated peak pressures do not exceed the service Level "C" limit (~/1500 psig) by taking into consideration the heat removal capability of safety valves, isolation condenser, bypass to the main condenser and other available heat removal systems. This analysis was to be completed within 10 days from the date of the bulletin and submitted under the provisions of 10 CFR 50.54(f).

Consumers Power Company has concluded that the Big Rock Point primary system will not exceed design pressure in the event of an ATWS; therefore, changes in plant operating parameters such as power level will show no net increase in the margin of safety with respect to the code pressure limit. The attached analysis provides the basis for Consumers Power Company's conclusion.

David P Hoffman (Signed)

David P Hoffman
Nuclear Licensing Administrator

A001 S/1

CC Director, Office of Nuclear Reactor Regulation
Director, Office of Inspection and Enforcement
NRC Resident Inspector Big Rock Point

Attachment - 13 pages

8007180 424

CONSUMERS POWER COMPANY

Big Rock Point Plant

IE Bulletin 80-17

Docket 50-155
License DPR-6

At the request of the Commission and pursuant to the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, as amended, and the Commission's Rules and Regulations thereunder, Consumers Power Company submits our response to IE Bulletin 80-17, dated July 3, 1980 entitled, "Failure of 76 of 185 Control Rods to Fully Insert During Scram at a BWR". Consumers Power Company's response is dated July 14, 1980.

CONSUMERS POWER COMPANY

By R B DeWitt (Signed)
R B DeWitt, Vice President

Sworn and subscribed to before me this 14th day of July 1980.

Dorothy H Bartkus (Signed)
Dorothy H Bartkus, Notary Public
Jackson County, Michigan
My commission expires March 26, 1983.

(SEAL)

ANALYSIS FOR ITEM 7 OF IE BULLETIN NO 80-17

Big Rock Point is a non-jet pump BWR in which steam separation takes place external to the reactor vessel. The primary coolant system has a design pressure of 1700 psig and a code allowable pressure (10% over design pressure) of 1870 psig. The primary steam drum is equipped with six (6) spring loaded safety valves each of which has a flow capacity at its setpoint pressure (~1500 psia) of approximately 33% of rated steam flow. The safety valves were sized and set based specifically on a turbine trip without bypass event in which the reactor scram system was assumed to fail. No other event could cause such a rapid increase in system pressure and reactor power as this event. The analysis of this event for the original plant design is discussed in Chapter 12 (page 28) of the Big Rock Point FHSR and in Reference 1. This analysis shows that primary coolant system design pressure will not be exceeded even in this extreme case. Analysis of the containment response to such an event is presented in Reference 2.

An analysis has been conducted using the Retran computer code (Reference 3) to verify that the primary coolant system transient results presented previously are still applicable to the plant as presently operated. Although this anlysis has not been completely reviewed per the CPCo QA program, the results are believed to be accurate and are presented to provide further assurance that previous conclusions remain valid. A comparison of important plant and core parameters for the new and previous analyses is presented by the attached Table 1. Significant assumptions made in this new analysis are as follows:

1. The emergency condenser was conservatively not modeled. The emergency condenser has a capacity to remove between 5 and 10% of rated core power.

2. Operator action to manually trip the recirculation pumps or actuate the liquid poison system was not considered. Either of these actions would result in a significant reduction in core power and therefore in steam production rate.

3. End of cycle void reactivity feedback (times a 1.25 factor of conservatism) was assumed.

4. For each safety valve, a constant capacity equivalent to that at 1550 psia (the setting of the first valve - other valves are set higher by consecutive increments of 10 psi) based on the Moody critical flow model (Reference 4) was assumed. The Moody model predicts critical steam flows that are approximately 2% higher than the code rating of the valves. A valve accumulation of 3% of the setpoint was also assumed.

Results of this analysis are presented on the attached Figures (1) thru (9). Presented are core power (Figure 1), reactor pressure (Figure 2), steam drum pressure (Figure 3), steam drum water level (Figure 4), void reactivity (Figure 5), vessel outlet plenum void fraction (Figure 6), recirculation flow one of two loops (Figure 7), vessel lower plenum coolant temperature (Figure 8), and safety valve flow (Figure 9). Because the safety valve closing characteristics were not explicitly modeled, Figure 9 may be used to establish the number of safety valves that may open in such an event, but should not be used to determine the rate of valve cycling. The maximum predicted valve flow rate is 346 lb/sec which is the approximate capacity of four safety valves. Hence, there exists considerable excess safety valve capacity for providing over-pressure protection in the event of an ATWS. The maximum predicted primary coolant system pressure is 1671 psia at the pump discharge at 11.8 seconds after the turbine trip. This pressure is well within code limits, and in fact, is less than primary co lant system design pressure.

ANALYSIS FOR ITEM 7 OF IE BULLETIN 80-17

References

1. "Transient Analysis Consumers Power Company Big Rock Point Plant", APED-4093, October, 1962.

2. "Anticipated Transients Without Scram Study for Big Rock Point Plant", NEDE-21065, October, 1975.

3. "RETRAN - A Program for One-Dimensional Transients Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", EPRI CCM-5, December, 1978.

4. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture", J Heat Transfer, 87, 134-142, 1965.

ANALYSIS FOR ITEM 7 OF IE BULLETIN 80-17

Table 1

| Parameter | FHSR | New Analysis |
|---|---|---|
| Core Power, (Mwt) | 240 | 244.8 |
| Reactor Pressure, (psia) | 1500 | 1350 |
| $K_V$*, ($) | 5.7 | 8.5 |
| Total Safety Valve Capacity (% of rated -270 lb/sec) | 200 | 192 |
| Setting of First Valve, (psia) | 1700 | 1550 |

*Void reactivity at Full Power

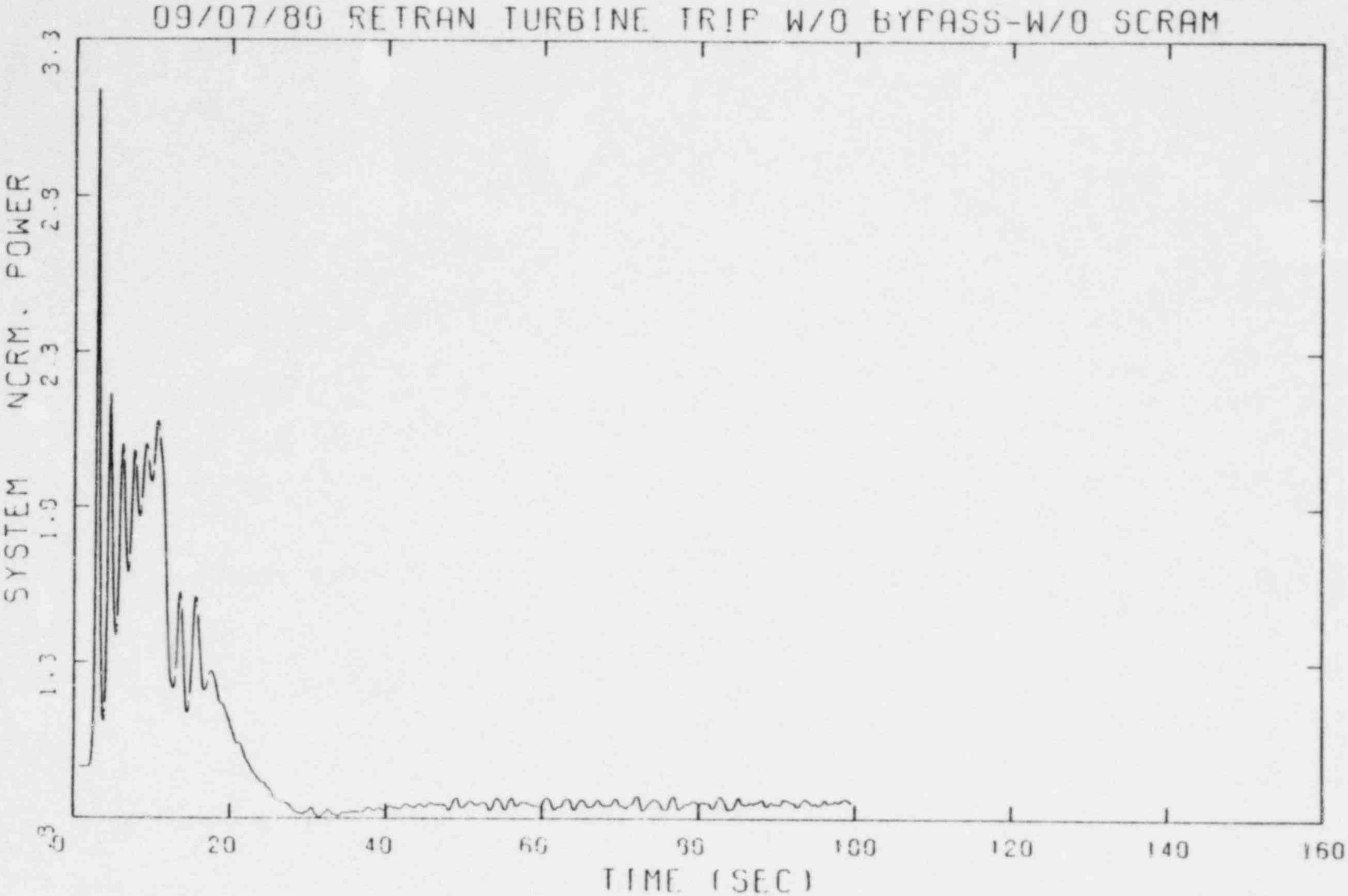


FIGURE 1 - CORE POWER

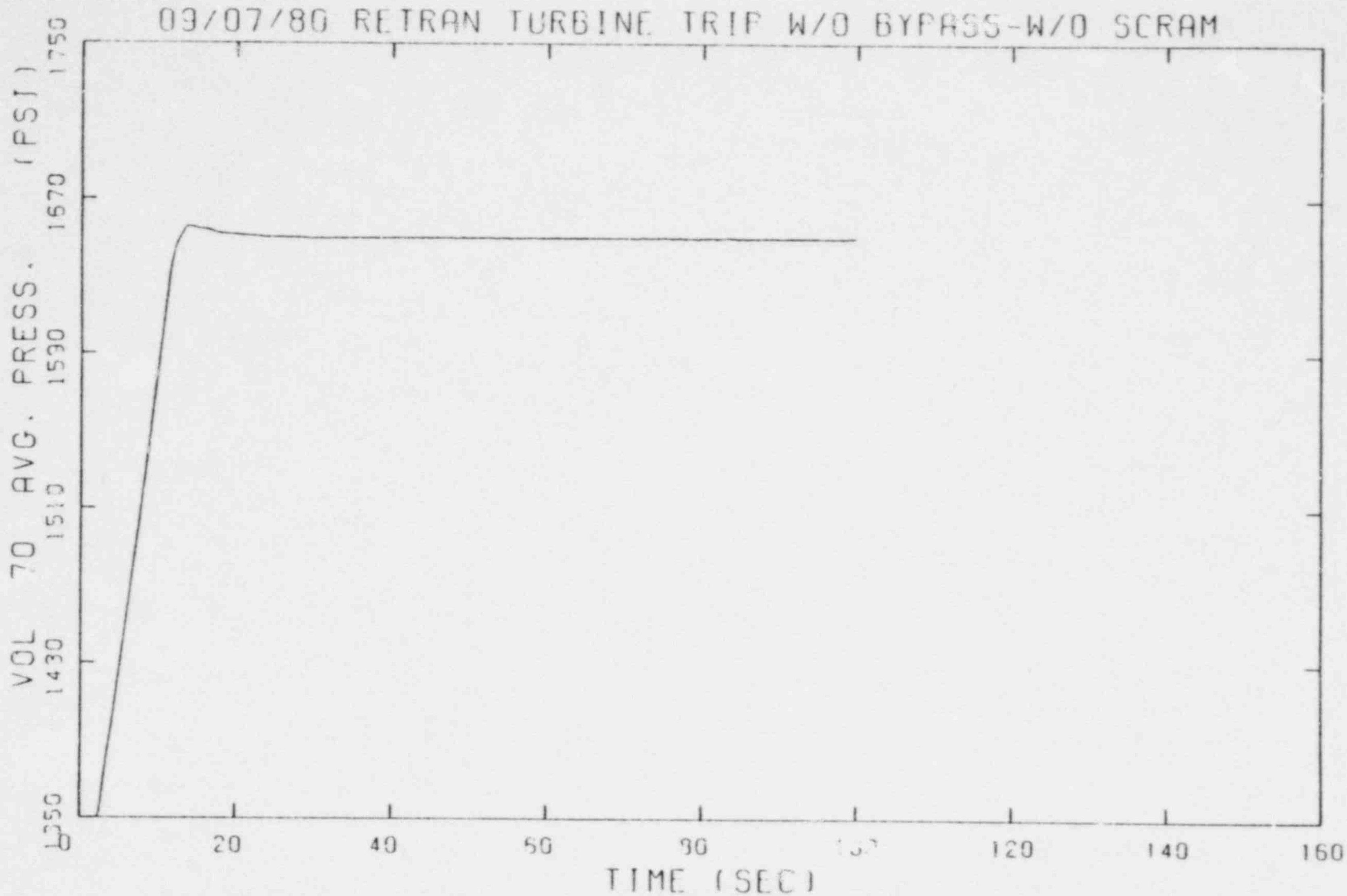


FIGURE 2 - REACTOR PRESSURE

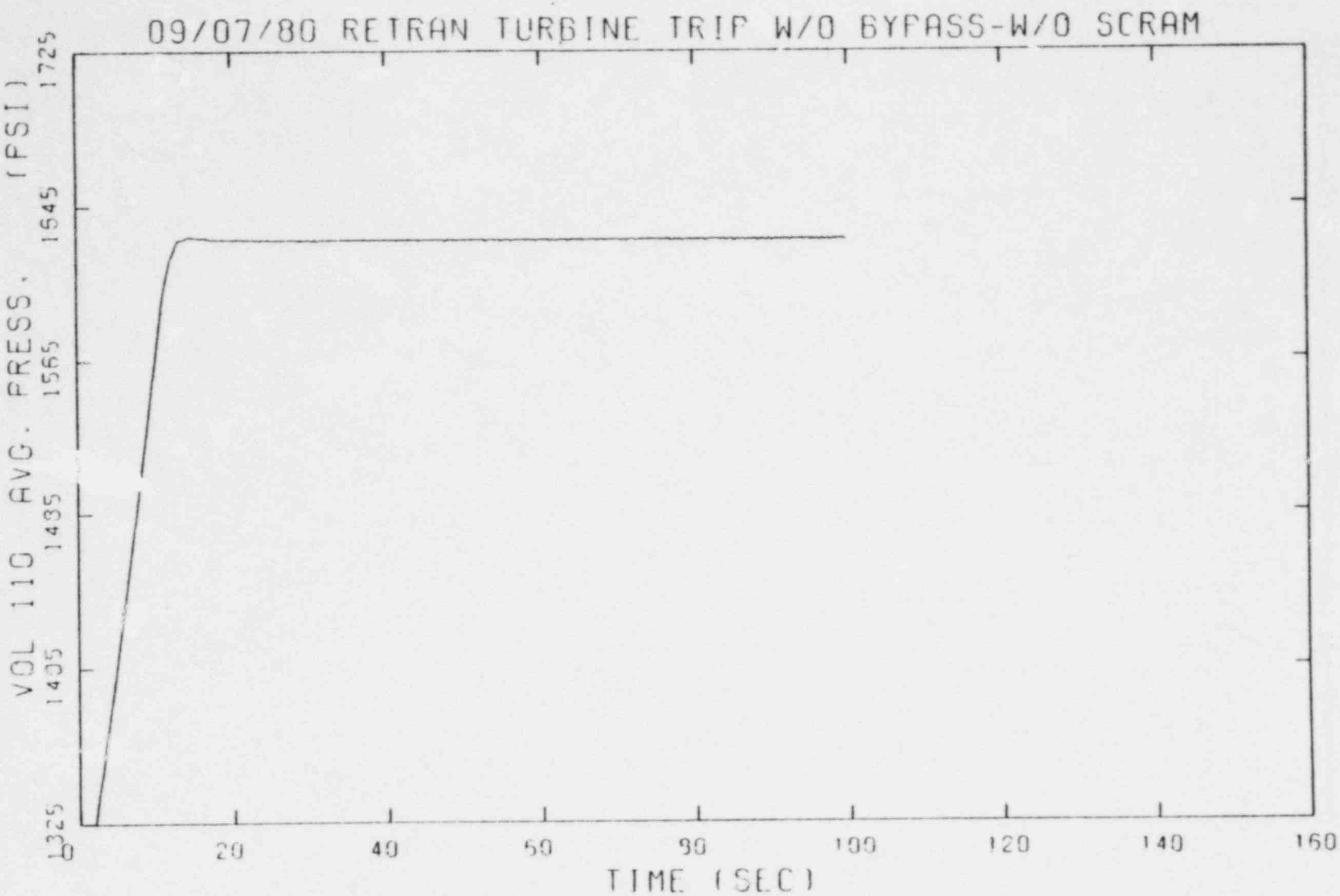


FIGURE 3 - STEAM DRUM PRESSURE

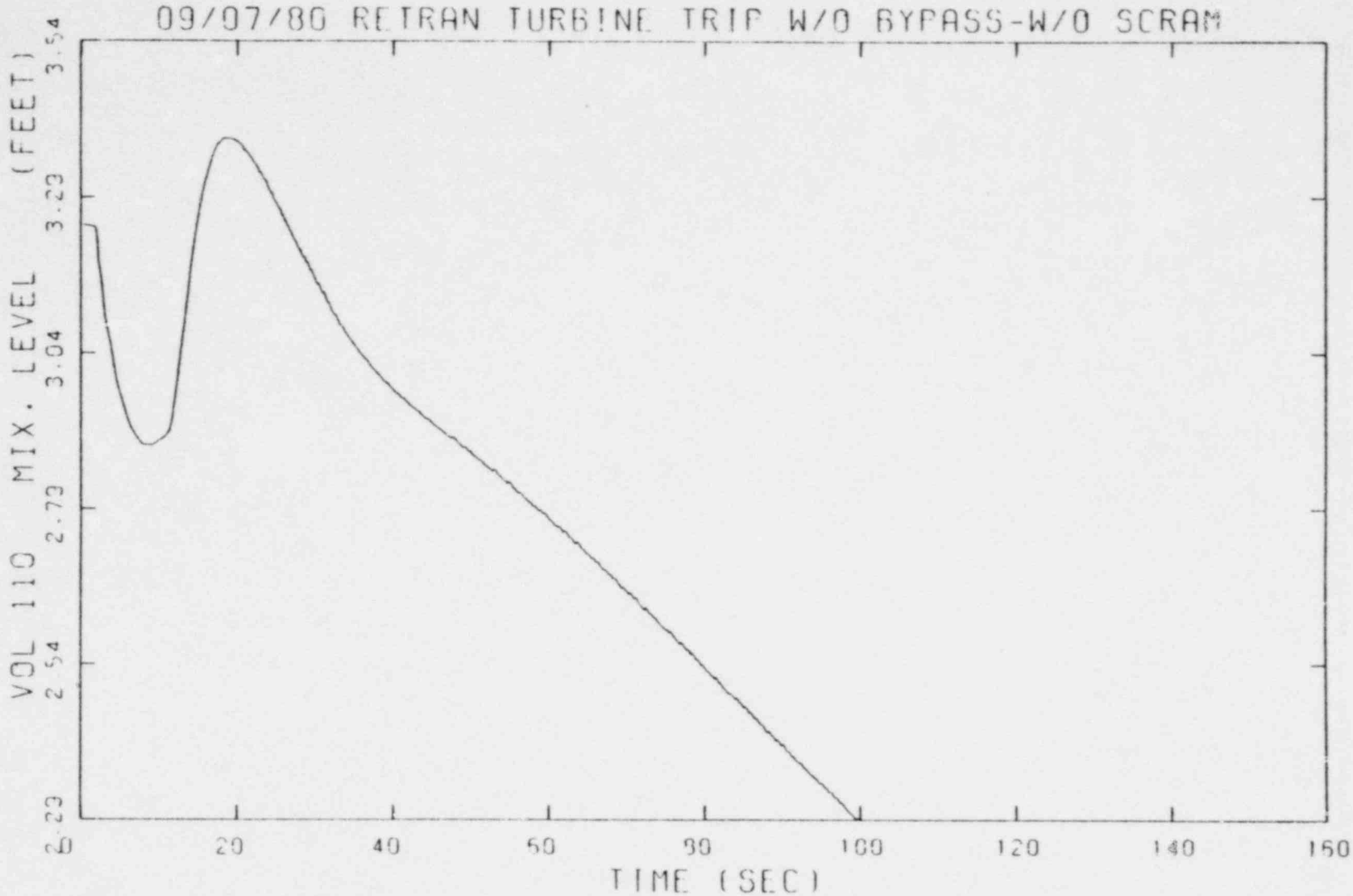


FIGURE 4 - STEAM DRUM WATER LEVEL

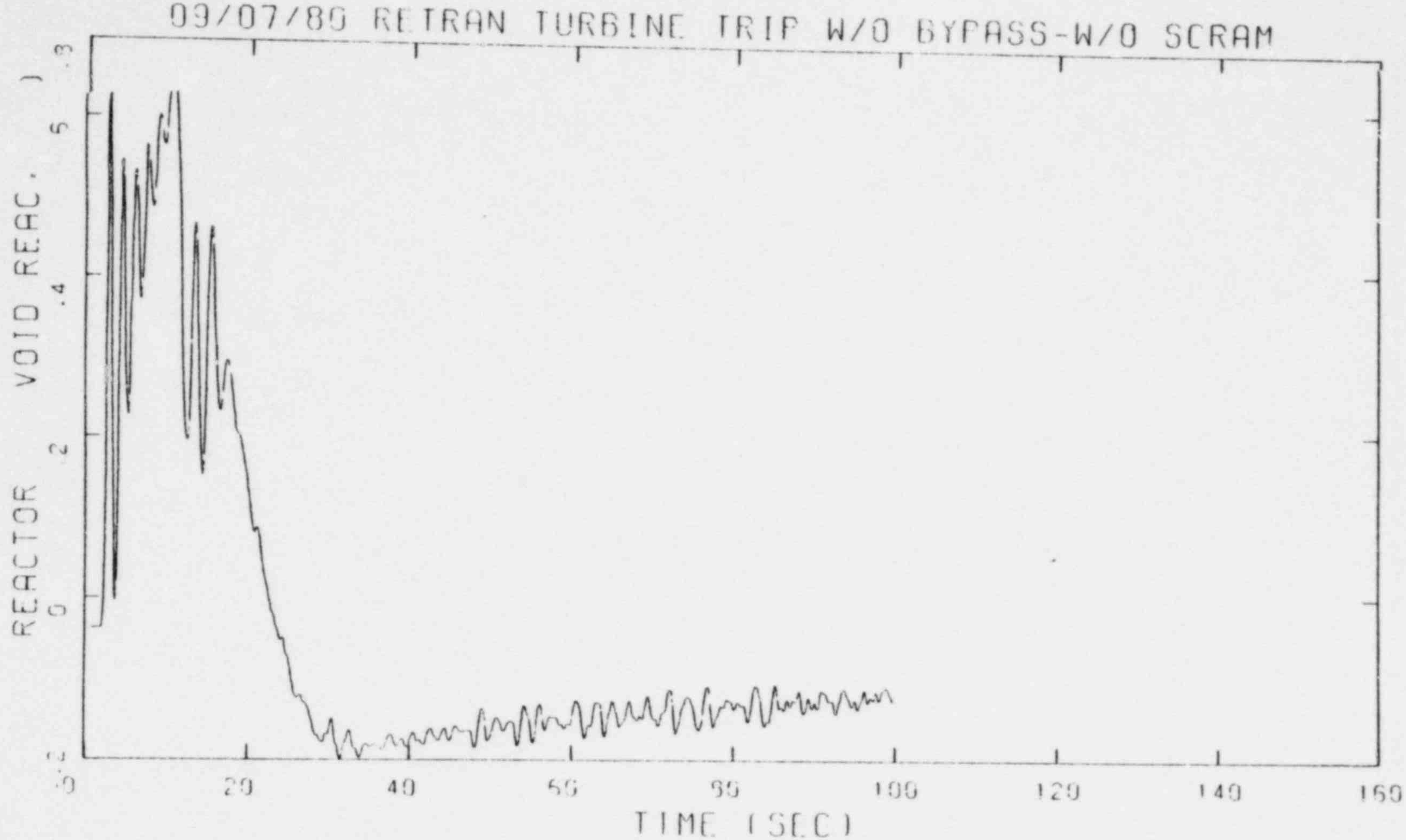


FIGURE 5 - VOID REACTIVITY

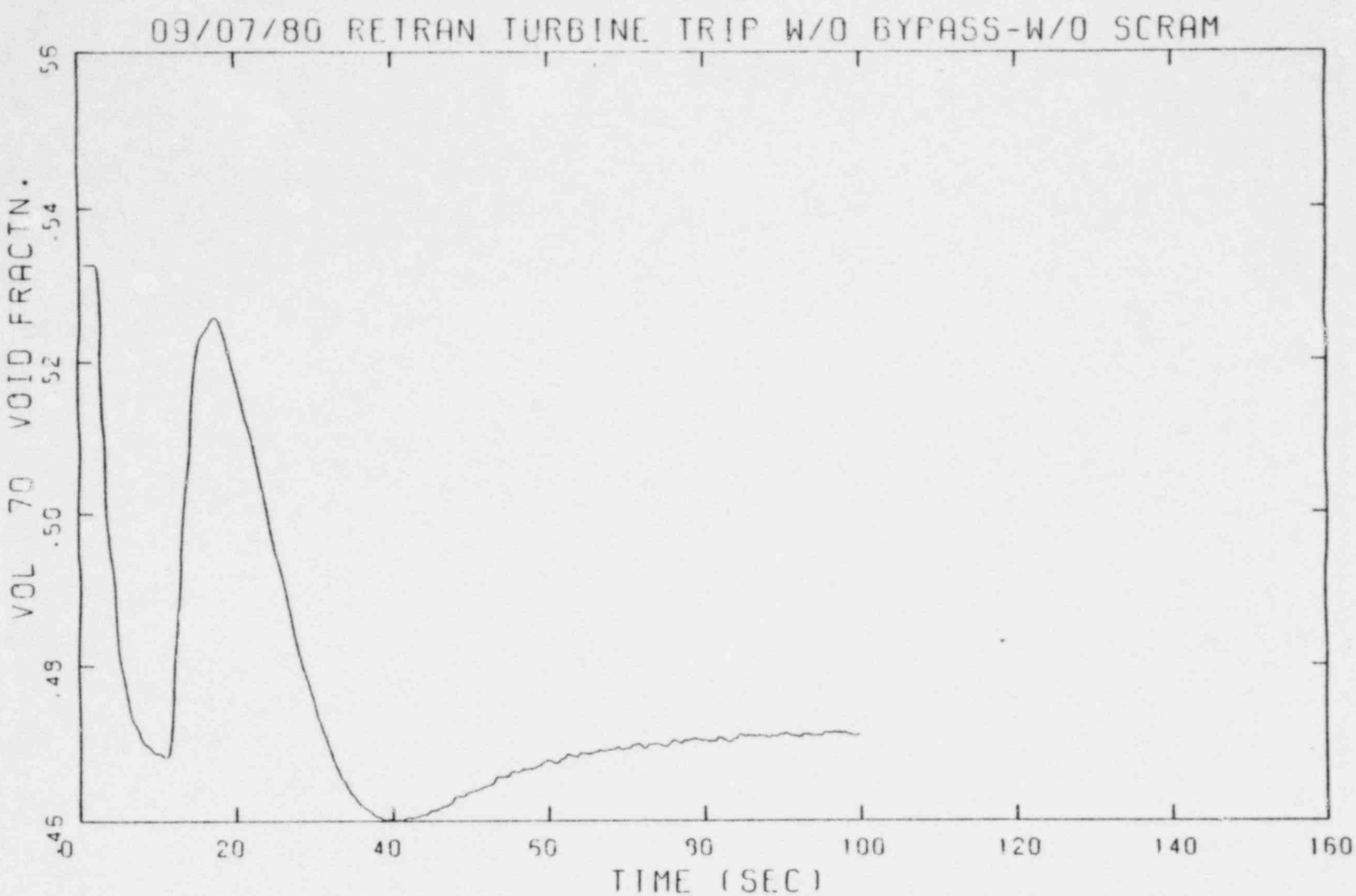


FIGURE 6 - VESSEL OUTLET PLENUM VOID FRACTION

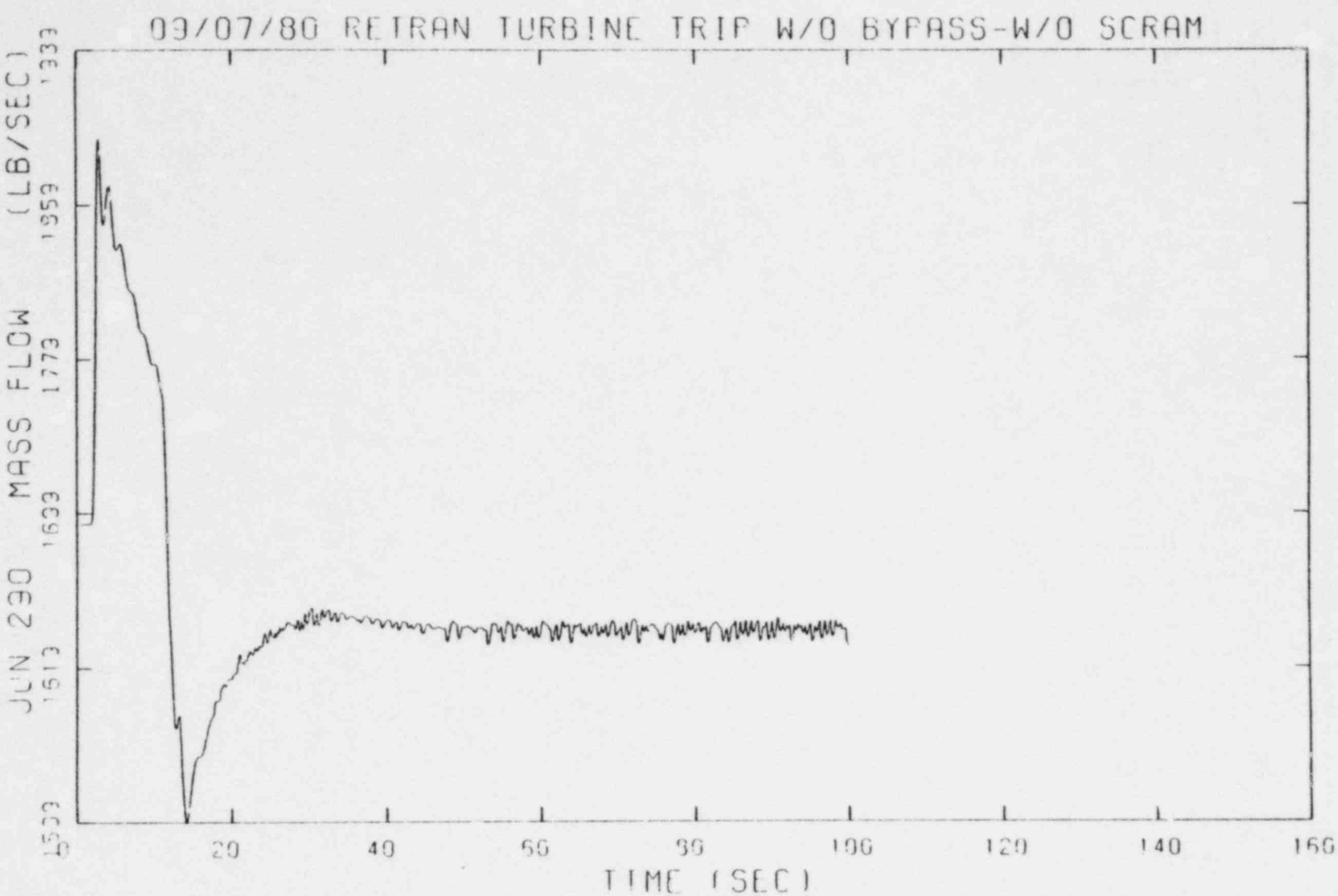


FIGURE 7 - RECIRCULATION FLOW - ONE OF TWO LOOPS

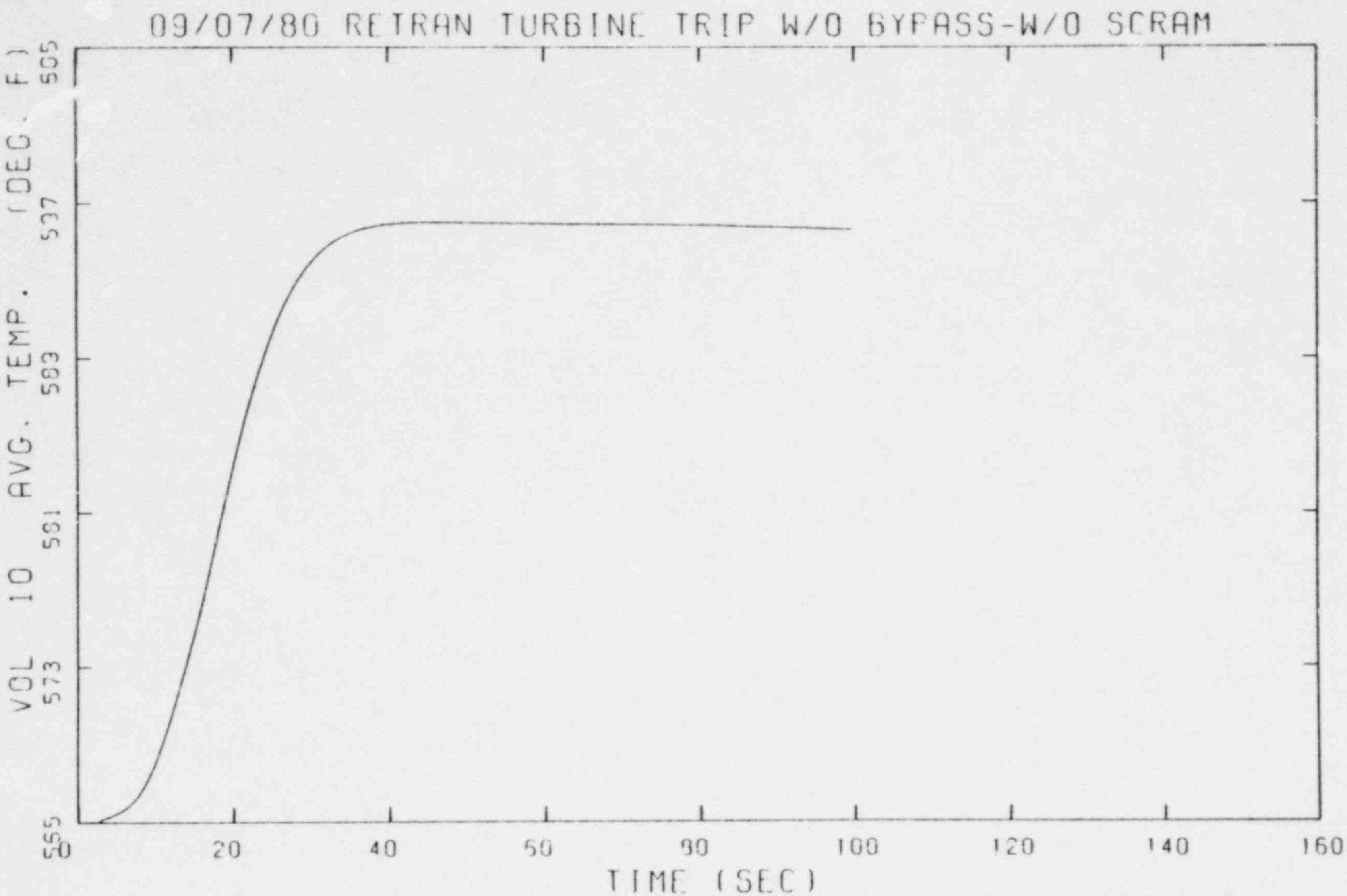


FIGURE 8 - VESSEL LOWER PLENUM COOLANT TEMPERATURE

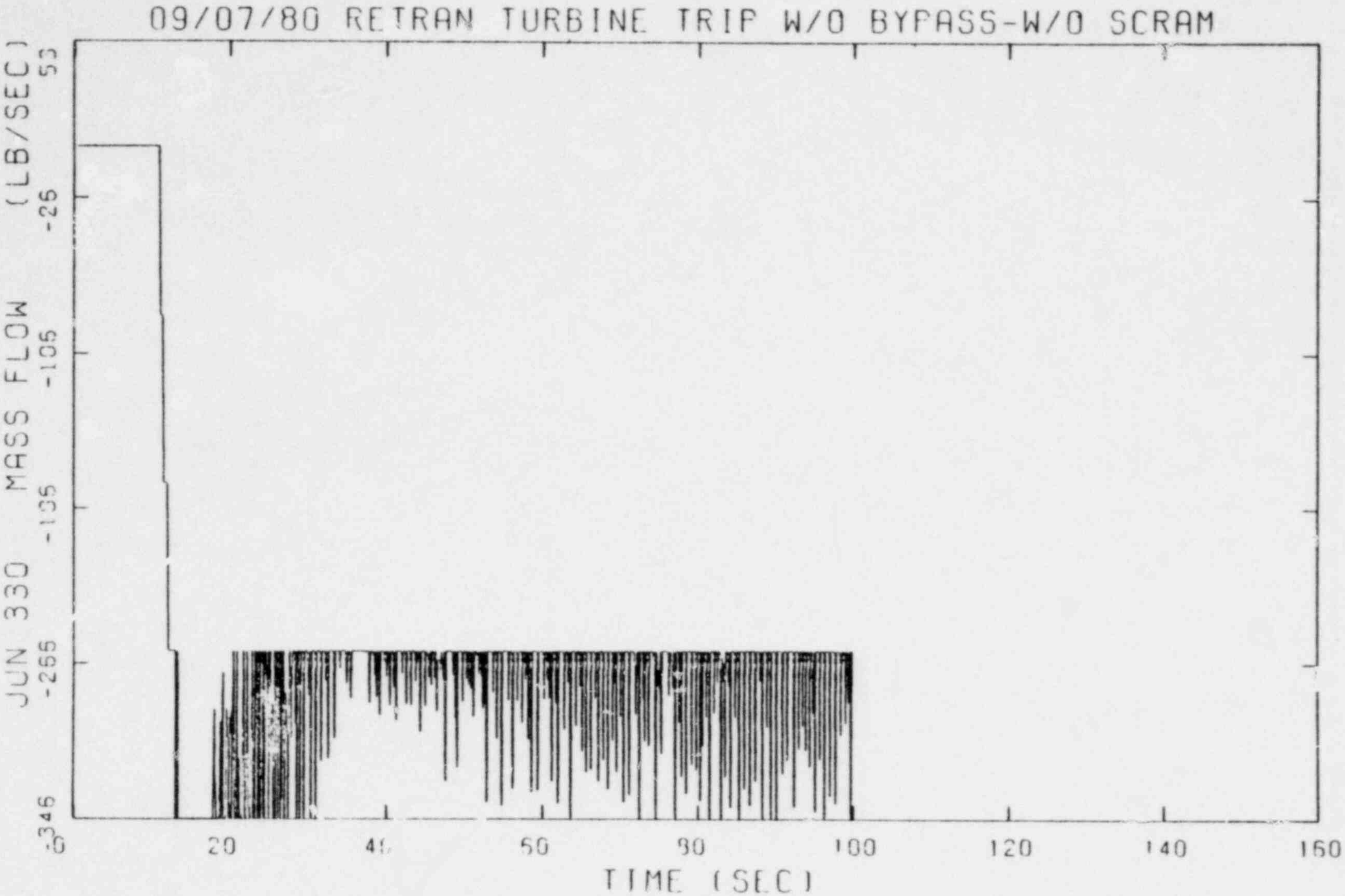


FIGURE 9 - SAFETY VALVE FLOW