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December 1, 1987

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Engineered Safety Features Single Failure Analysis
San Onofre Nuclear Generating Station
Unit 1

On October 7, 1987 NRC notification was made pursuant to 10 CFR 50.72 b(ii)B. to inform the NRC of the existence of newly identified single failure susceptibilities of the Engineered Safety Features (ESF). A conference call with NRC Region V staff was held following the notification to provide a more complete discussion of the affected equipment and postulated scenarios.

On October 9, 1987 a meeting was held with the NRC staff in Bethesda, Maryland to describe the single failure scenarios and to describe actions being taken to ensure continued safe plant operation with the identified scenarios. During this meeting the NRC staff requested that SCE prepare a report to address the information presented during the meeting.

SCE submitted the requested report by letter dated October 16, 1987. The report provided descriptions of each identified single failure scenario and a specific justification for continued operation. The justification for each scenario provided a description of the mitigating operator actions and applicable better estimate analyses performed. The better estimate analysis provided for main steam line break core response (cases 5 and 6) indicated that the moderator temperature coefficient (MTC) utilized is applicable for core burn-up until December 14, 1987. SCE committed to provide additional justification for continued operation for these cases prior to that date. Additionally, SCE committed to submit the ESF Single Failure Analysis and information on the design modifications for corrective actions.

The ESF Single Failure Analysis Report was submitted by letter dated November 6, 1987. The design descriptions for the integration of the third auxiliary feedwater pump, the steam/feedwater flow mismatch trip and main

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feedwater isolation modifications were provided by letter dated November 20, 1987. SCE committed to provide the design descriptions for the additional modifications to resolve the remaining ESF single failure concerns identified in the October 16, 1987 submittal by March 31, 1988. Transient analysis to support the integration of the third auxiliary feedwater pump was also provided.

The purpose of this letter is to submit the revised better estimate analysis for cases 5 and 6 to support justification for continued operation for the remainder of this cycle. Accordingly, provided as an enclosure to this letter is a better estimate analysis for the core response during a main steam line break (cases 5 and 6). Core physics parameters for end of cycle were assumed. In order to model the worst case core response the transient was assumed to result from a steamline break concurrent with the single failure of either HV852A or B. This scenario postulates the diversion of all safety injection flow to the steam generators until the time of manual isolation at 10 minutes. This emphasizes both continued feedwater addition and diversion of safety injection. The results of this transient analysis indicate that charging flow by itself is sufficient to control the return to power condition caused by the cooldown. The acceptance criteria for this transient are met.

The effect of the cooldown for this transient on the reactor vessel was also evaluated by comparison with previous Pressurized Thermal Shock (PTS) analyses. In order to assure the previously established PTS criteria are met, one change was made to the interim operating instructions which have been provided as compensatory measures pending implementation of the modifications to eliminate single failure concerns. Item V, "Loss of SI Flow Due to Feedwater Discharge Valve Failure," of Special Order 87-91 which was provided as an enclosure to our letter dated October 16, 1987 has been revised to direct the operators to terminate safety injection at the time of feedwater isolation. This action will prevent the repressurization of the primary system while providing an acceptable core response by controlling reactivity by the use of the charging pump. The single failures previously identified which result in less severe cooldown transients but which could have higher terminal pressures are being evaluated at this time to confirm that , as expected, they are also bounded by previous PTS analyses.

If you have any questions regarding the above, please contact me.

Very truly yours,

M. O. Medford

Enclosure

cc: J. O. Bradfute, NRR Project Manager, San Onofre Unit 1
J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

MAIN STEAM LINE BREAK
BEST ESTIMATE
TRANSIENT ANALYSIS
SINGLE FAILURE OF
MAIN FEEDWATER ISOLATION VALVE
SAN ONOFRE UNIT 1

BACKGROUND

As a result of a single failure evaluation of the Engineered Safety Features (ESF), SCE identified a new single failure susceptibility for a main steam line break. The failure scenario would result in diversion of safety injection flow from the reactor coolant system (RCS) to the steam generators during a main steam line break event. The scenario results from a main steam line break outside containment with a single failure of a main feedwater isolation valve (HV-852A or B). The redundant feedwater isolation valves (MOV, FCV, and CV) are assumed to fail open due to the harsh environment outside containment. Safety injection flow would be diverted to the steam generators since the RCS pressure would be greater than the steam generator pressure.

SCE has provided the justification for continued operation until December 14, 1987 in the November 6, 1987 submittal. The analysis documented in this report will be used in justification for operation after December 14, 1987 until the next refueling outage.

ANALYSIS METHODOLOGY

To simulate and analyze the proposed accident the LOFTRAN computer code was used. The assumptions used for calculating the input to LOFTRAN are listed in Table 1. Hot Zero Power initial conditions are assumed since they are more limiting than full power conditions due to the reduced decay heat. For the case analyzed, no decay heat is assumed. The reactivity parameters input to LOFTRAN were the same as used for the previous San Onofre Unit 1 steam line break. These parameters were calculated assuming end of life conditions and the worst stuck rod. The shutdown margin was calculated assuming Hot Full Power conditions since the shutdown margin is less at full power.

The resulting RCS cooldown for this transient was outside the applicability range of the W-3 DNB correlation, normally used for steam line break evaluations. Therefore, the Macbeth DNB correlation was used. The Macbeth correlation is applicable at the low pressures resulting from this transient.

Transient Description

The transient is initiated by a double ended rupture of a steam line outside containment. This causes all three steam generators to blow down to atmosphere and rapidly cool the RCS. The rapid RCS cooldown causes the water in the RCS to shrink which reduces the pressurizer level. The reduction in pressurizer level causes pressurizer pressure to decrease which results in a safety injection signal. A main feedwater isolation valve is assumed to fail to close upon receipt of the safety injection signal and results in the

diversion of the safety injection flow to the steam generators instead of the RCS. The redundant main feedwater isolation valves are assumed to fail open due to adverse conditions from the steam line break. The flow of cold SI water to the steam generators enhances the cooldown and shrink of the RCS. Additionally, upon receipt of the SI signal, the charging pump realigns normally to the RWST and delivers borated water to the RCS. The pressurizer empties and the upper head voids and starts to empty. The cooldown of the RCS causes reactivity insertion due to the moderator temperature and doppler temperature parameters. This reactivity insertion is somewhat offset by the boron introduced by the charging flow. The reactivity insertion eventually causes a return to power.

Ten minutes after transient initiation, the operator is assumed to terminate the safety injection flow to the steam generators. No credit is taken for the delivery of safety injection flow to the core. The reactivity slowly increases and is eventually turned around due to doppler power feedback and the boron injected into the RCS from the charging pump.

Results and Conclusions

The results of the transient are shown in figures 1 through 12. Peak power of approximately 6% of nominal power is reached at about 680 seconds. DNBR remains above the limit for the entire transient.

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Table 1

ANALYSIS ASSUMPTIONS

1. Initial Conditions (Hot Zero Power)

   a. Pressurizer Pressure - 100% of Nominal
   b. RCS Flow Rate-100% of Nominal
   c. RCS Tavg - No Load Tavg
   d. Pressurizer Level - 20%
   e. Core Power - 0. MWt
   f. Boron Concentration - 0. ppm
   g. S/G Pressure - No Load Pressure

2. Feedwater (To S/Gs From the Time of Transient Initiation to SI Signal)

   a. Flow - 100% of Nominal Flow Delivered Equally to Three S/Gs
   b. Temperature - 100°F
   c. Initiation Time - Transient Initiation

3. Safety Injection (To S/Gs After SI Initiation for 10 min.)

   a. Flow - 150% of Nominal Flow Delivered Equally to Three S/Gs
   b. Temperature - 70°F
   c. Initiation Time - At the Time of the SI Signal

4. Safety Injection (To Core)

   Safety injection Flow is assumed terminated at 10 Minutes after Transient Initiation. No credit is taken for safety injection delivered to the core.

5. Auxiliary Feedwater

   AFW Flow is assumed to be unavailable for 15 minutes

6. Reactor Coolant Pumps

   RCPs are assumed to trip as a result of the SI signal

Table 1 (cont.)

7. Reactivity Parameters

   a. Moderator Density Coefficient - End of Life coefficient
   b. Doppler Coefficients - Conservative End of Life coefficients defect
   c. Shutdown Margin - Best Estimate for End of Life Hot Full Power Conditions: 5.58% delta-k/k

8. Charging/Letdown

   a. Immediate Letdown Isolation
   b. After the transient initiation and prior to the SI signal a charging pump will deliver 90 gpm to the RCS from the volume control tank
   c. After the SI signal the charging pump will realign to the RWST and deliver 120 gpm of borated water to the RCS

9. Accident Simulation

   Double Ended Rupture of A Steam Line Outside Containment

10. Decay Heat

   No Decay Heat

Table 2

TIME SEQUENCE OF EVENTS

Event	Time, Seq.
Steam Line outside Containment ruptures (double ended)	0.
Low pressurizer pressure SI setpoint is reached	17.
Feedwater flow to the SGs stops	17.
SI flow to the SGs starts	17.
Pressurizer empties	18.
The charging pump realigns to the RWST and delivers 120gpm of borated water to the RCS	20.
RCPs trip and coastdown	20.
Upper head saturates and starts to void	34.
SGs start to overfill	145.
Shutdown margin is lost (reactor is critical)	314.
SI flow to the S/Gs is stopped	600.
Peak power is reached	680.
Shutdown margin is regained (reactor is subcritical)	718.

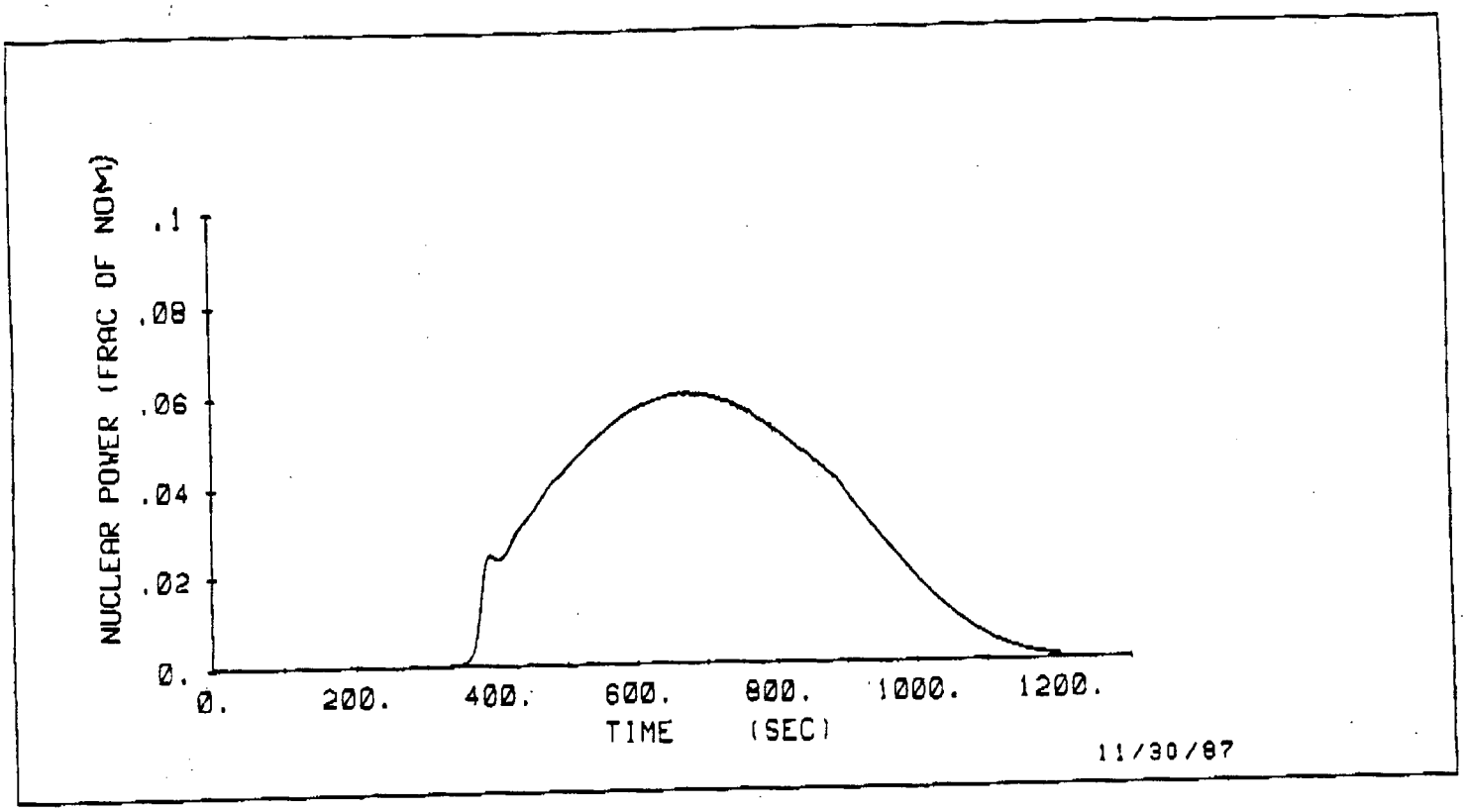

Figure 1. Nuclear Power

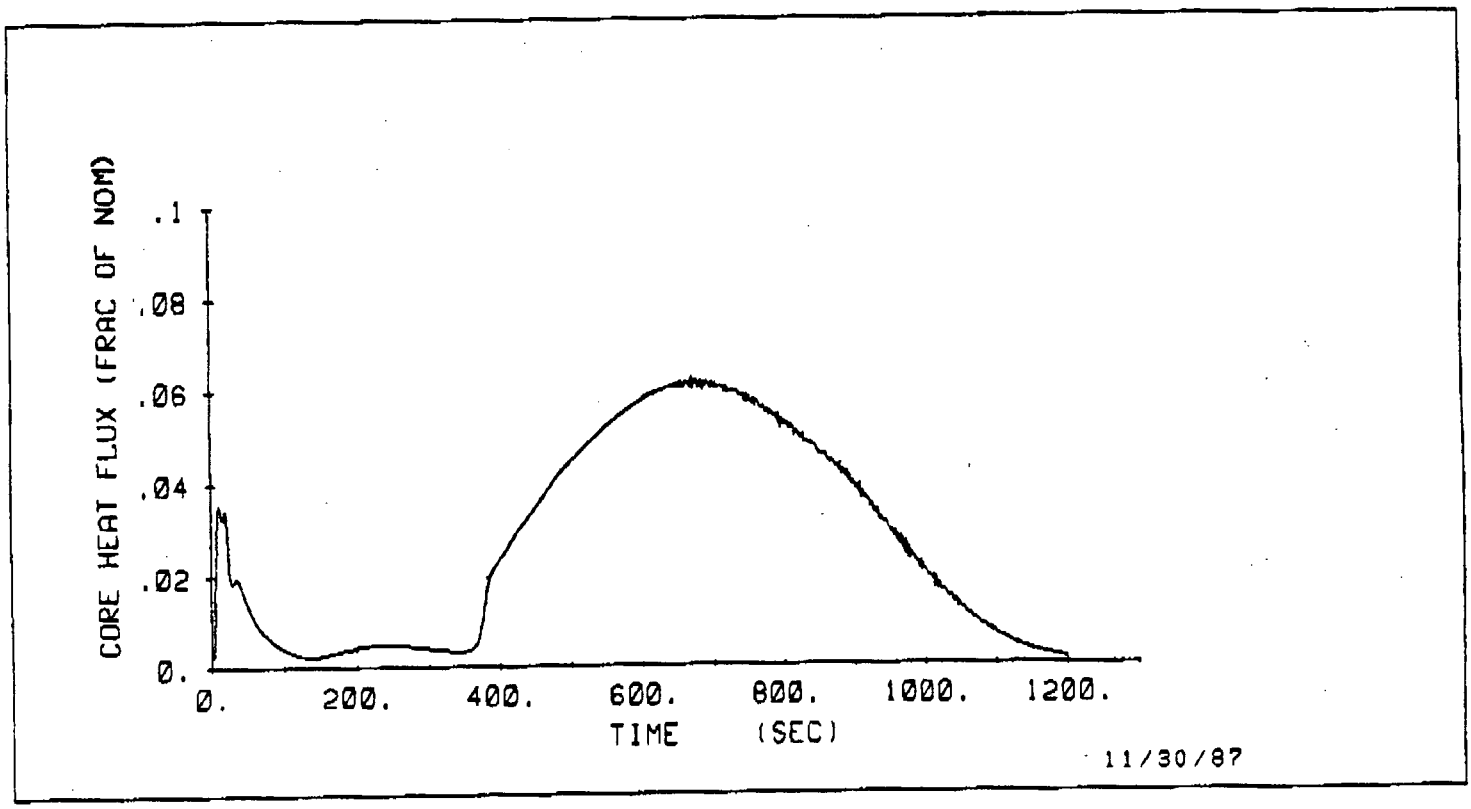

Figure 2. Core Heat Flux

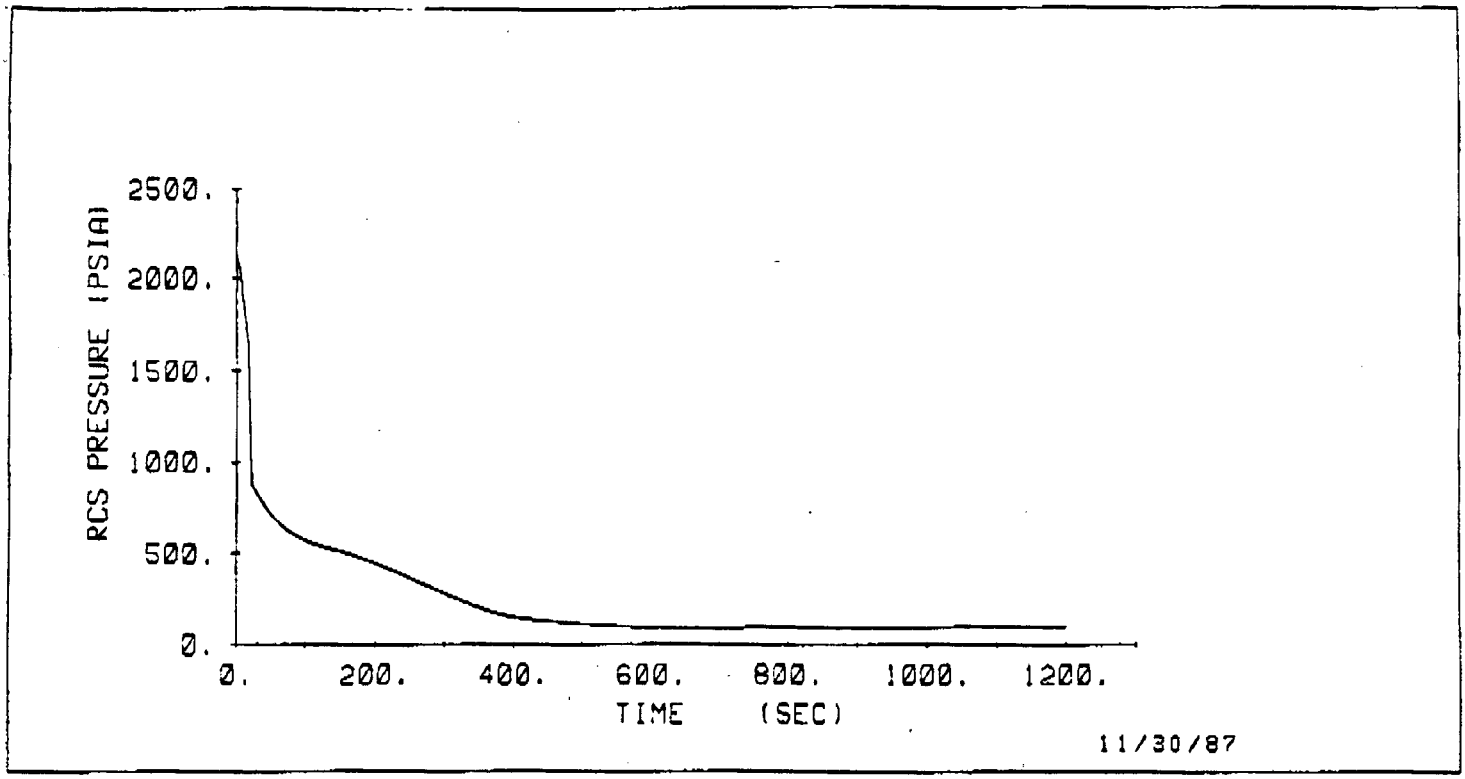

Figure 3. RCS Pressure

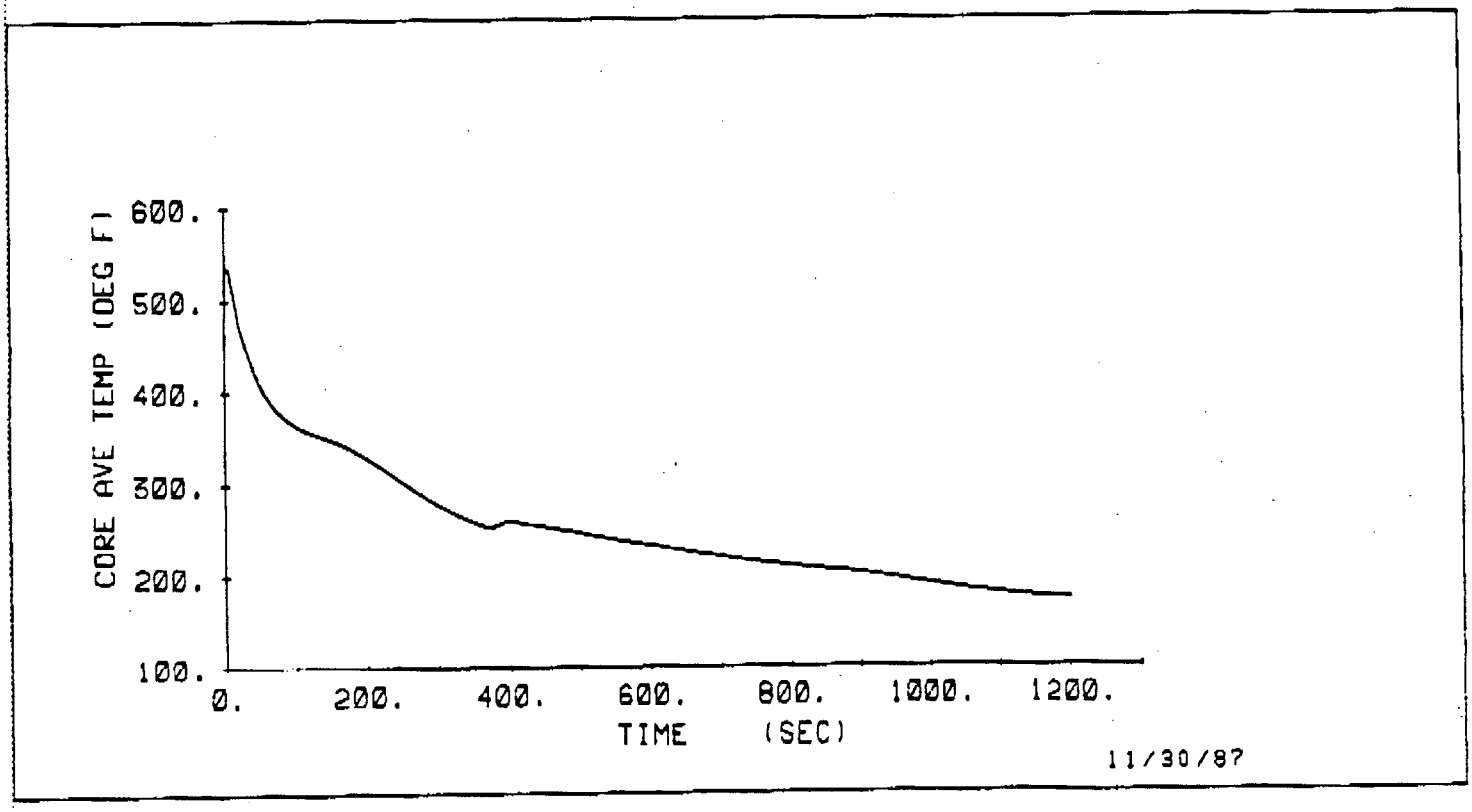

Figure 4. Core Average Temperature

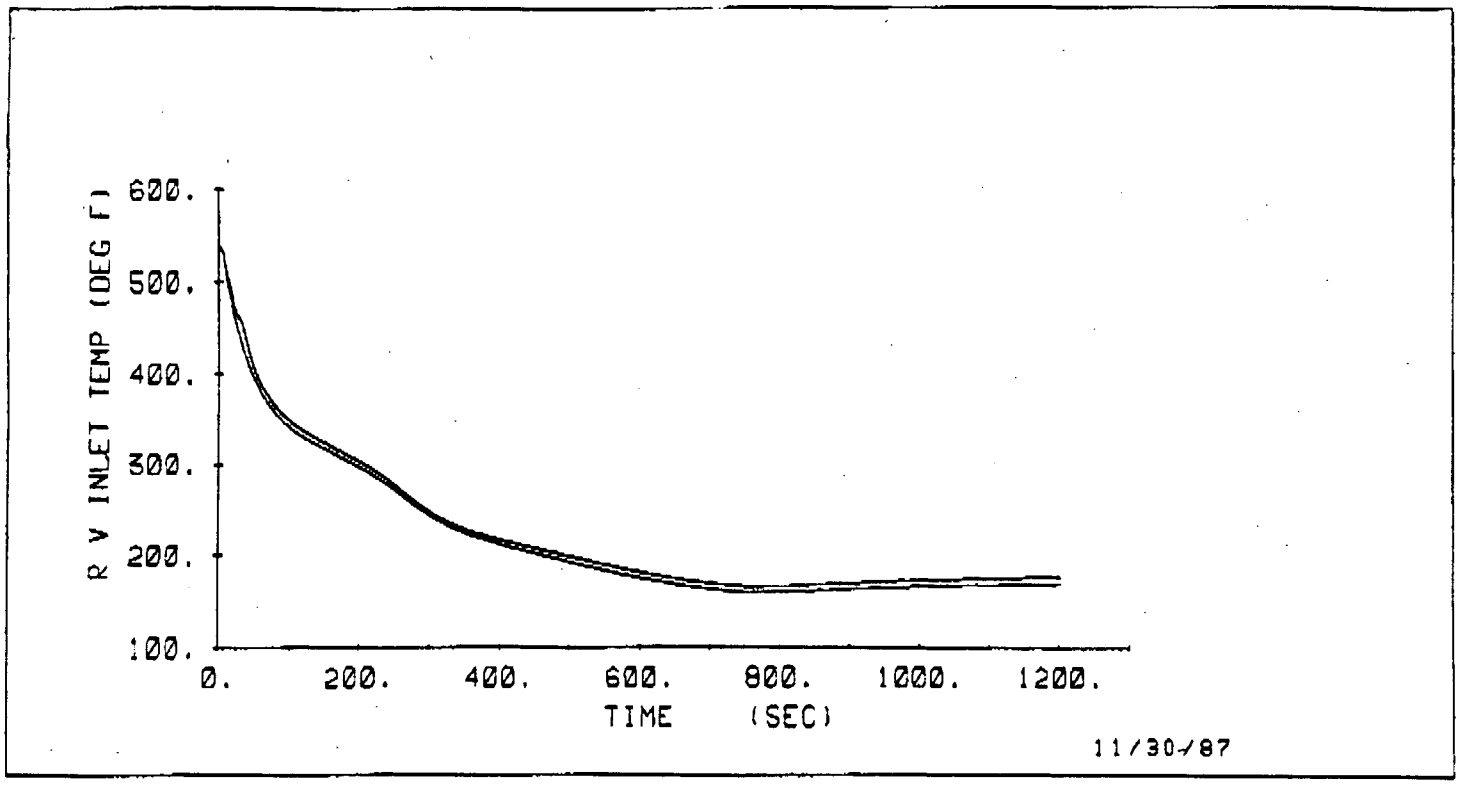

Figure 5. Reactor Vessel Inlet Temperature

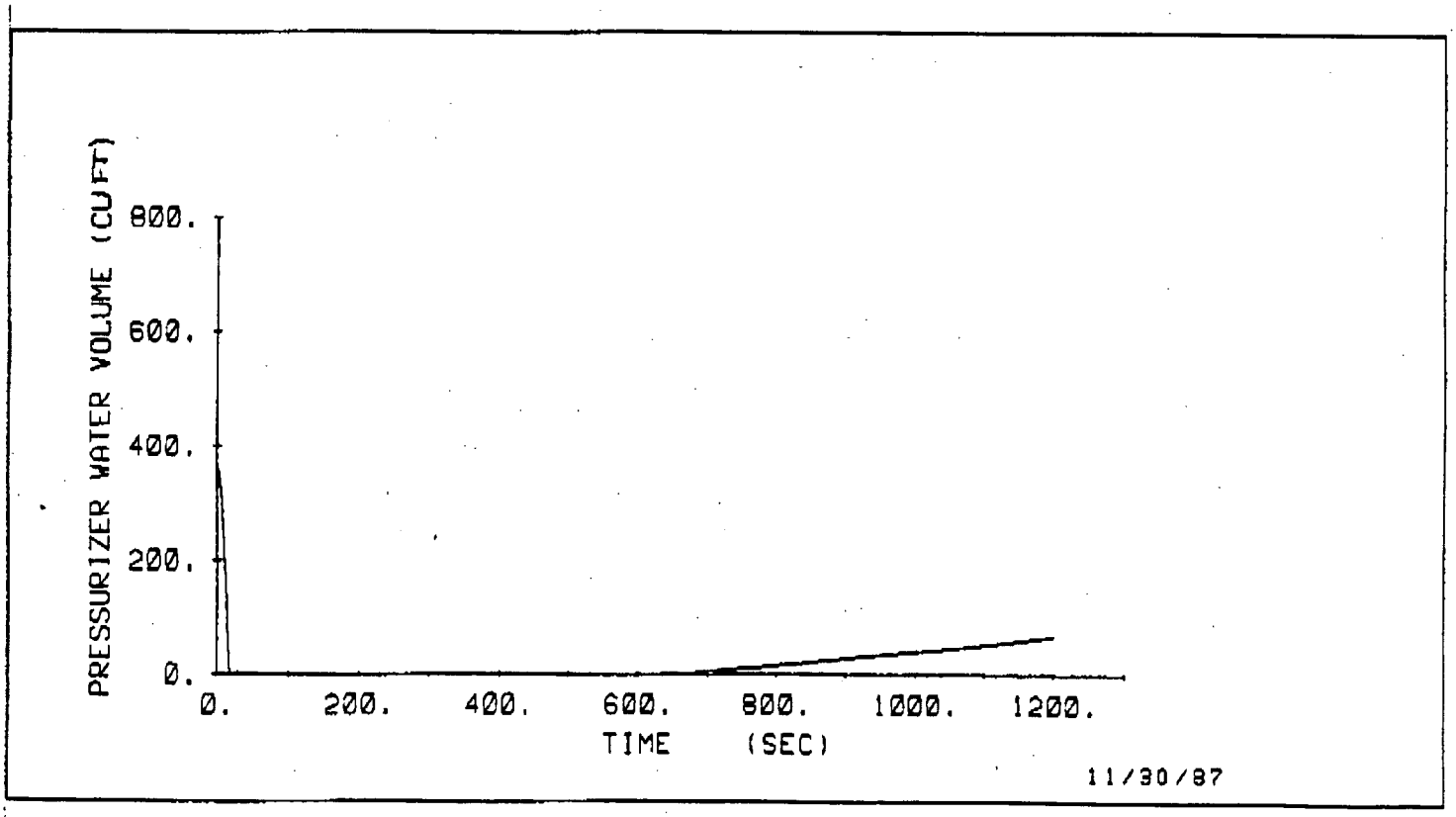

Figure 6. Pressurizer Water Volume

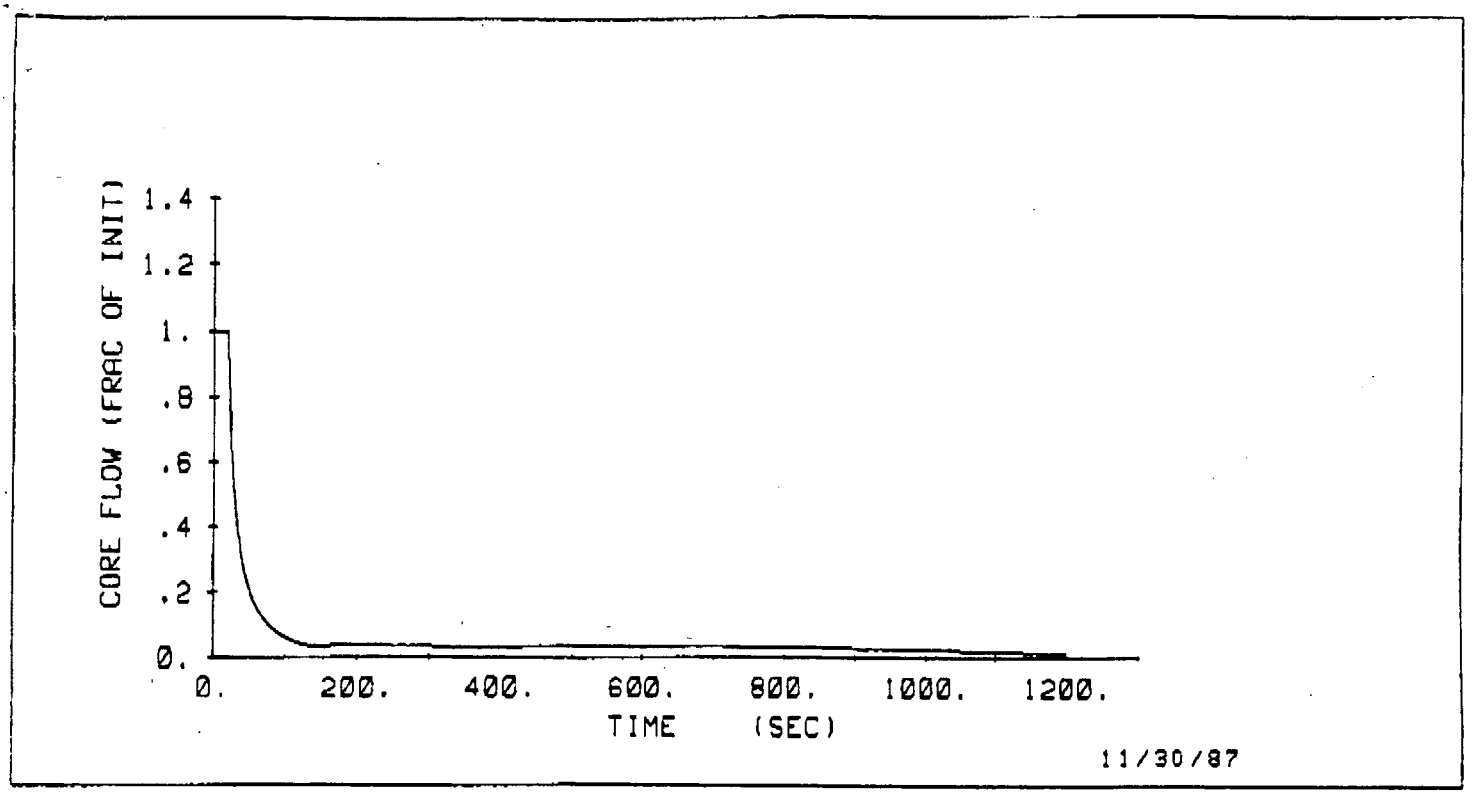

Figure 7. Core Flow

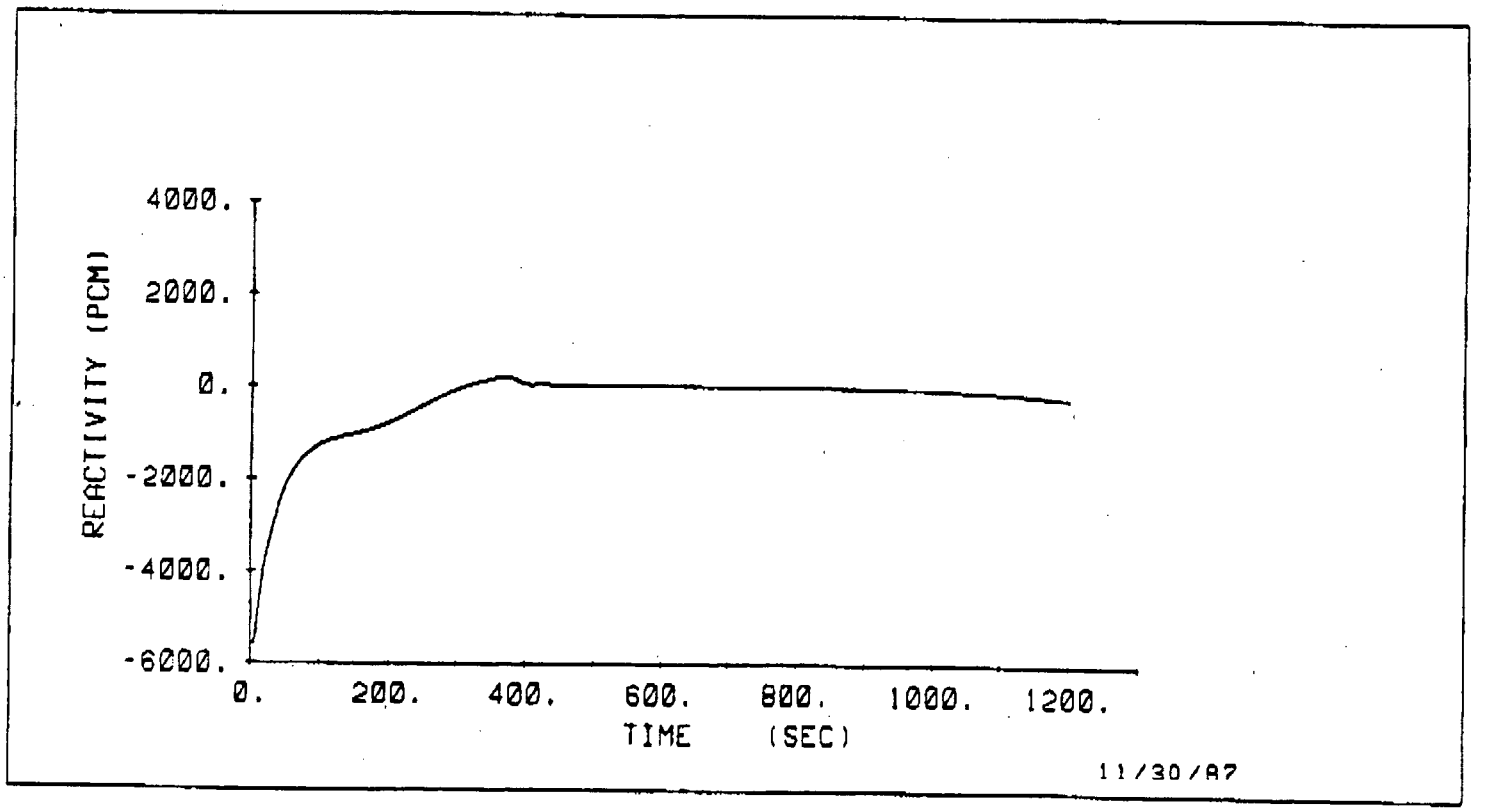

Figure 8. Reactivity

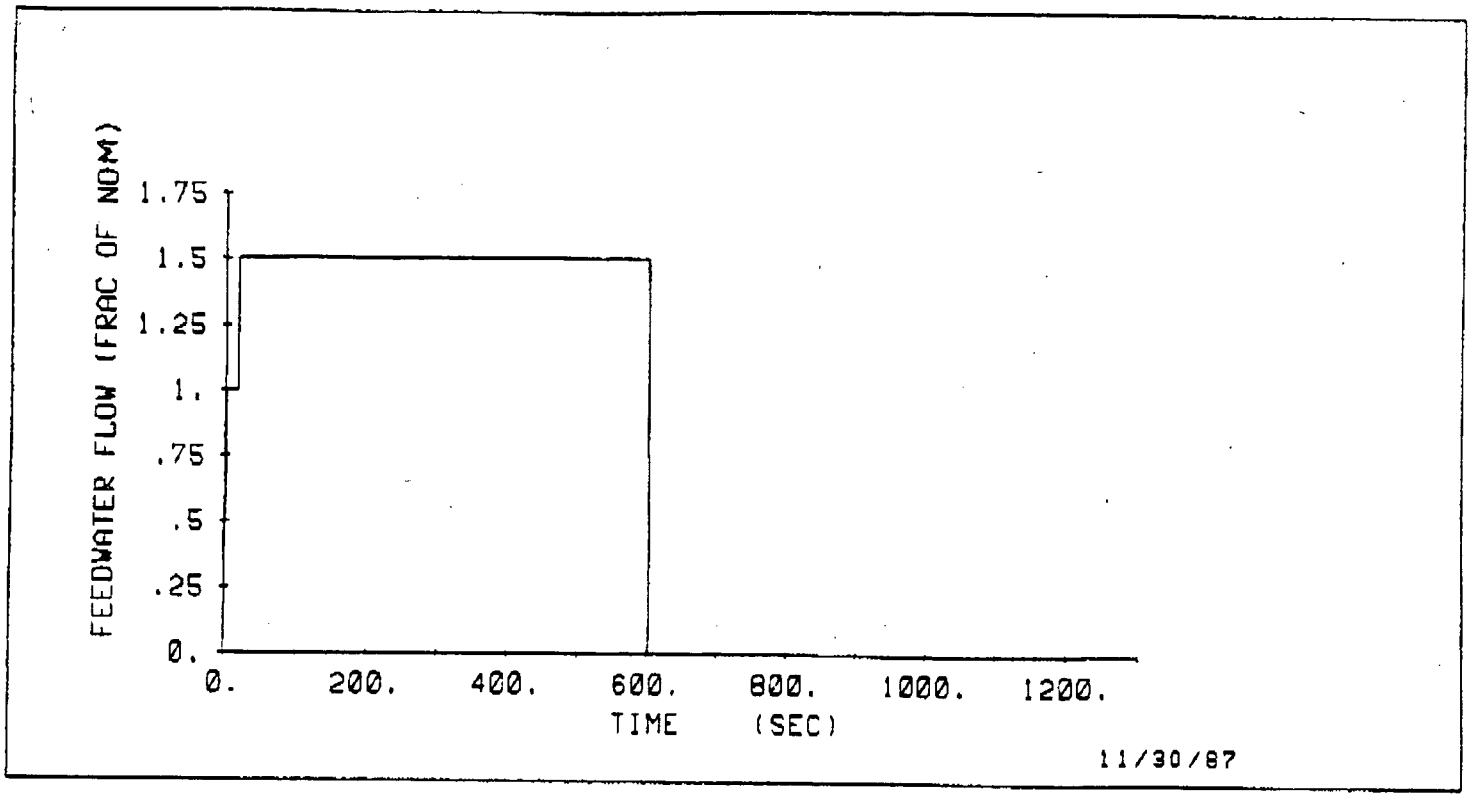

Figure 9. Feedwater Flow

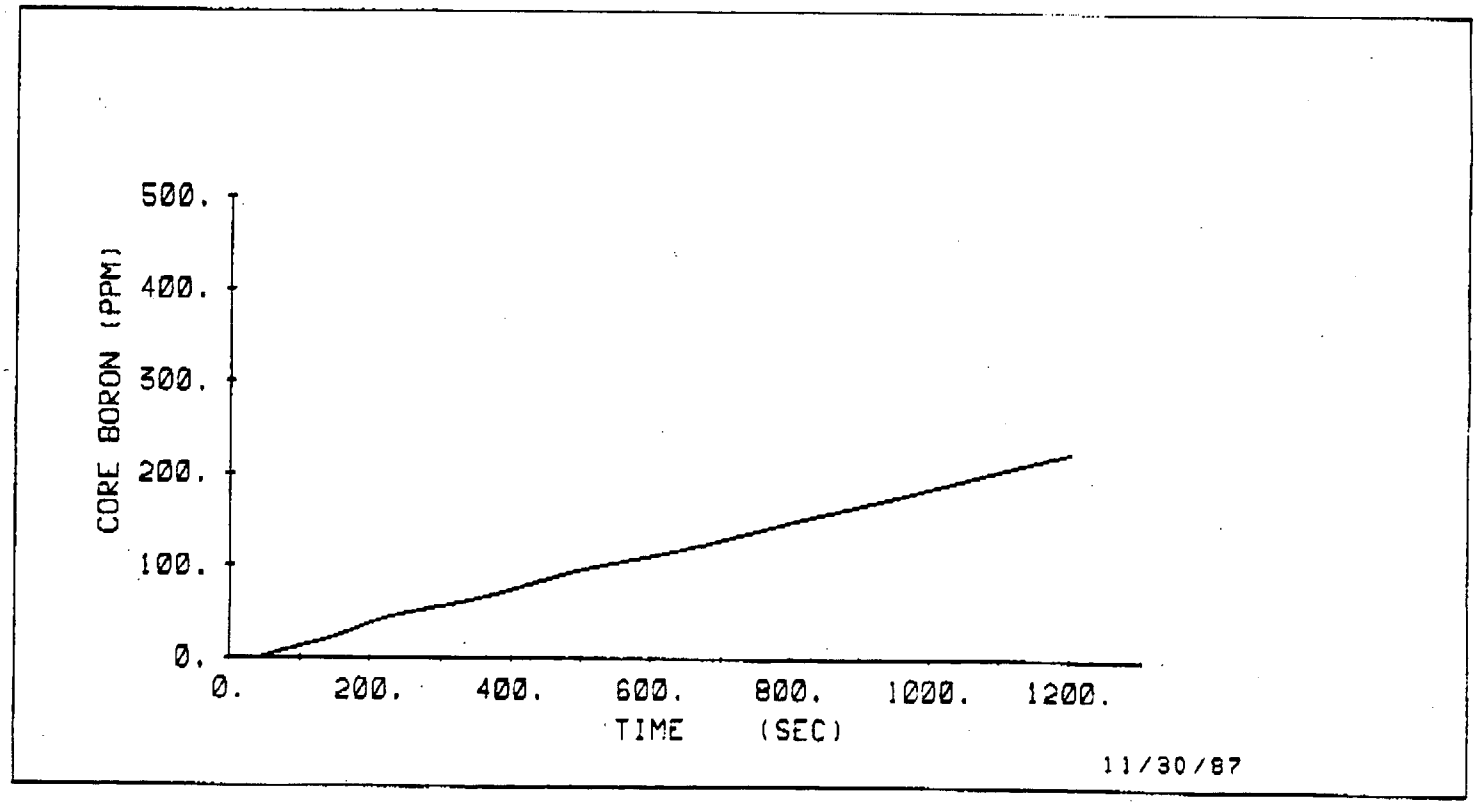

Figure 10. Core Boron

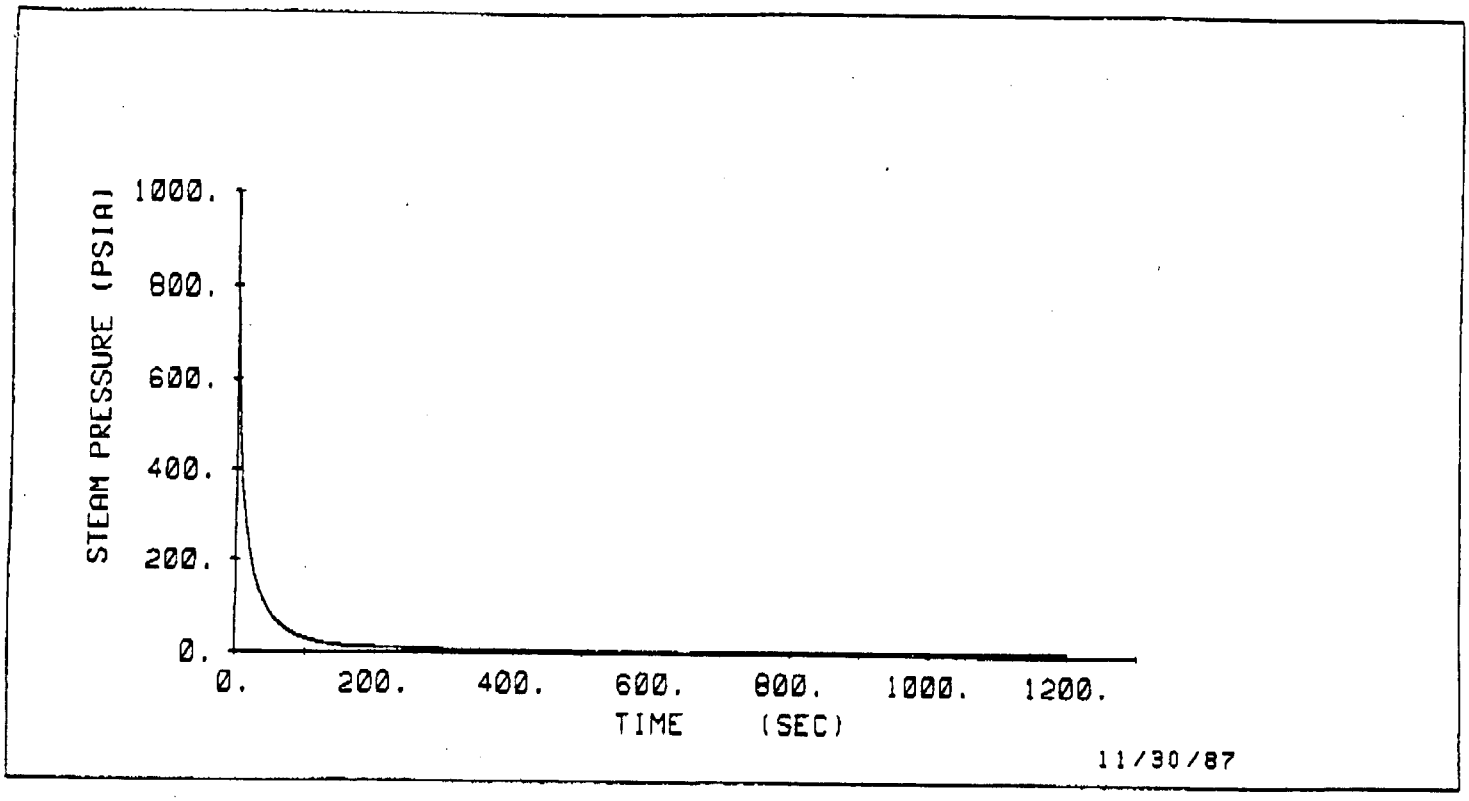

Figure 11. Steam Pressure

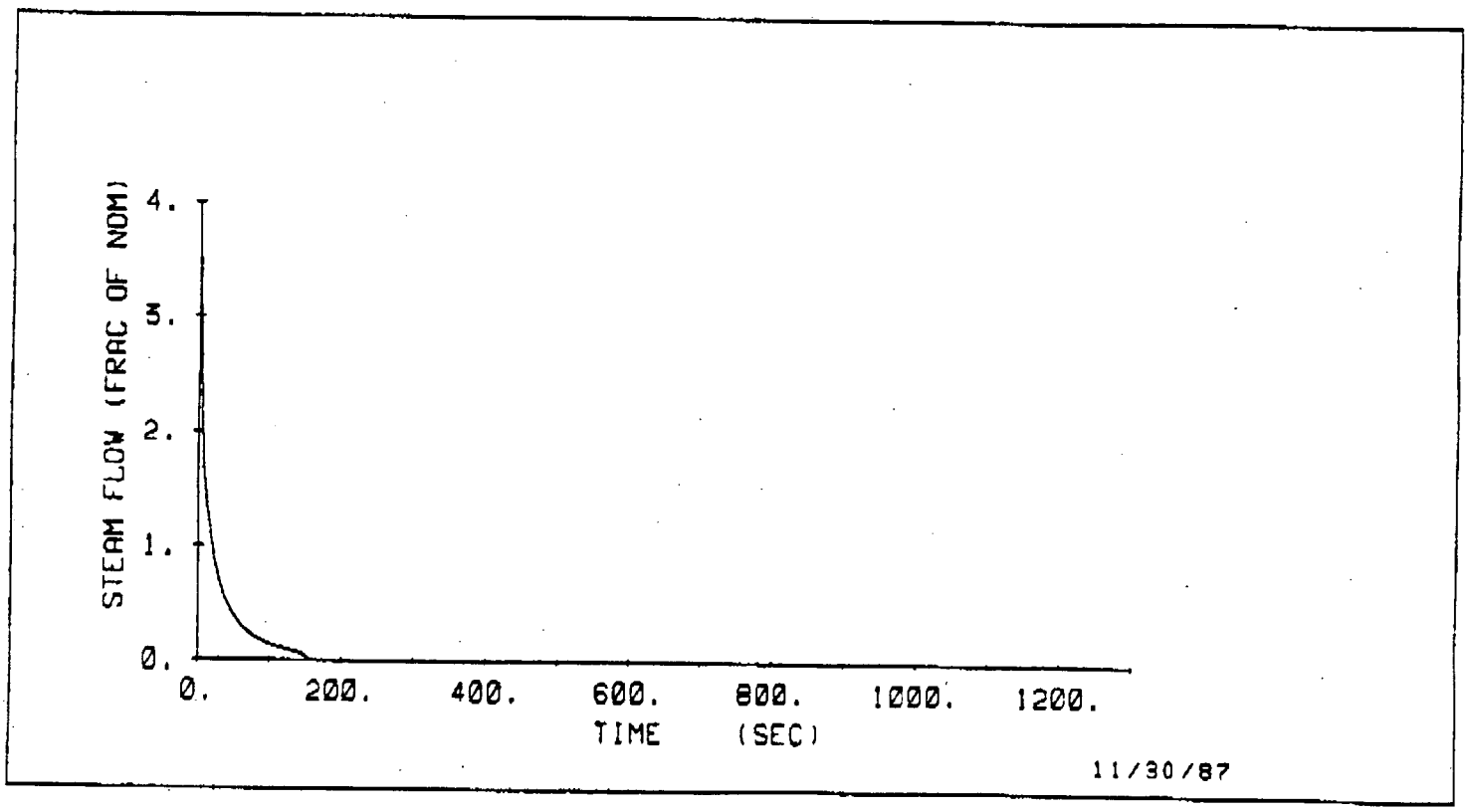

Figure 12. Steam Flow