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Revision 1

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PLANT TRANSIENT ANALYSIS FOR THE R. E. GINNA UNIT 1 NUCLEAR POWER PLANT

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Updates for Revision 1

Revision 1 was issued in June 1979. The revision was issued because the pressure feedback coefficient and the boron worth coefficient for Cycle 9 were slightly changed from their respective Cycle 8 values. Using the same dimensions as in Table 2.4, the changes are:

	Previous Value	Updated Value
EOC pressure coefficient	+4.0	+3.5
EOC boron worth coefficient	-0.950	-0.872

The transients affected by these changes were the large and the small steamline break. Since the steamline break is a depressurization transient, the positive moderator feedback contributes some negative reactivity. The boron worth coefficient is used for the borated water coming from the high pressure injection line. This injection terminates the power generation in case of the large steamline break, and it preserves the shutdown margin in case of the small steamline break.

The analysis showed that the thermal margin values previously calculated were still protected for the updated reactivity data. All changes have been incorporated into the report, all applicable pages have been marked "Revision 1."

The revised analysis is expected to cover the future reload cycles for the R. E. Ginna Plant.

Table 1.1

Summary of Results

Transient Class	Maximum Power Level (MWt)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (W-3)
Initial conditions for transients	1550.4	177,560	2220.	2.00
Uncontrolled rod withdrawal (II)				
@ 6.0×10^{-4} 1/sec	1906	195,120	2235	1.77
Uncontrolled rod withdrawal (II)				
@ 5.0×10^{-5} 1/sec	1748	197,850	2283	1.73
Loss of flow (III) 2-pump coastdown	1550.4	181,170	2250	1.61
Loss of flow (IV) Locked pump rotor	1550.4	181,160	2273	1.23
Loss of load (II)	1648	184,320	2511	1.83
Large steam line break (IV)	532	61,670	*	1.58***
Small steam line break (IV)	**	**	*	

* Pressure decreases from initial value.

** The core does not go critical.

*** MDNBR calculated using Macbeth correlation.

Table 2.4

R. E. Ginna Unit 1 ENC Kinetic Parameters

Symbol	Parameter	Value: Beginning-of-Cycle	Value: End-of-Cycle
α_M	Moderator Coefficient ($\Delta\rho$/F x 10^4)	0.0	-3.5
α_D	Doppler Coefficient ($\Delta\rho$/F x 10^5)	-1.25	-2.00
α_P	Pressure Coefficient ($\Delta\rho$/psia x 10^6)	+2.5	+3.5
α_V	Moderator Density Coefficient (%$\Delta\rho$)/(g/cm^3)	0.0	+29.635
α_B	Boron Worth Coefficient ($\Delta\rho$/ppm x 10^4)	-0.875	-0.872
β_{eff}	Delayed Neutron Fraction (%)	0.610	0.510
α_{CRC}	Total Rod Worth (%$\Delta\rho$)	-1.89**	-2.83**

** Minimum required (these are conservative values for analysis purposes only; the actual values are significantly higher).

TABLE 2.5

MODERATOR AND DOPPLER COEFFICIENTS

Transient	Moderator: Desired Moderator Feedback Effect	Moderator: Multiplier	Moderator: Resulting Coefficient $\Delta\rho/°F \times 10^4$	Doppler: Desired Doppler Feedback Effect	Doppler: Multiplier	Doppler: Resulting Coefficient $\Delta\rho/°F \times 10^5$
Fast Rod Withdrawal	Minimum	*	0.0	Minimum	0.8	-1.0
Slow Rod Withdrawal	Minimum	*	0.0	Minimum	0.8	-1.0
Pump Coastdown	Minimum	*	0.0	Maximum	1.2	-1.5
Locked Pump Rotor	Minimum	*	0.0	Maximum	1.2	-1.5
Loss of Load	Minimum	*	0.0	Minimum	0.8	-1.0
Steam Line Break	Maximum	---	***	Minimum	---	**

* For minimal effect no moderator feedback allowed.

** See Figure 3.31

*** See Figure 3.32

which could, under pessimistic circumstances, lead to criticality and core damage if unchecked.

As a worst case, the steam line break is assumed to occur at hot zero power conditions. At this time, the steam generator secondary side water inventory is at a maximum, prolonging the duration and increasing the magnitude of the primary loop cooldown. For conservatism, the most reactive control rod is assumed to be stuck out of the core when evaluating the shutdown capability of the control rods. The reactivity as a function of core average temperature and the variation of reactivity as a function of core power used in this analysis are shown in Figures 3.31 and 3.32. The moderator and Doppler feedback coefficients are valid for Cycle 8 fuel.

Minimum capability of the boron injection system was assumed, which implies that only two of the three high-pressure safety injection pumps (HPSI) are available. A low pressurizer pressure signal in combination with low pressurizer level initiates the safety injection system. Borated water starts entering the injection lines after the pressurizer pressure has come down to the shutoff head (1400 psia) of the injection pumps. The time required to sweep the lines of low concentration borated water prior to the introduction of 20,000 ppm borated water from the Boric Acid Tanks, has been accounted for in the analysis. No credit was taken for the effects of the resident low concentration borated water being swept into the primary loop from the safety injection lines. The initial pressure was set at 2280 psia (nominal +30 psi) to delay the onset of safety injection.

Two steam line breaks were analyzed. The large break at the exit of the steam generator with outside power available was shown in the reference cycle analysis[6] to give the greatest return to power and the highest core

average heat flux. This case was analyzed for an ENC-fueled core to ensure the core integrity is maintained during the transient. A 1.8 percent shutdown margin has been used in this analysis. A small steam line break assuming one loop operation was analyzed to ensure an adequate shutdown margin exists at the end of the cycle such that the core does not go critical during such an event (safety valve, relief valve, or bypass valve failed open).

The system responses for a large steam line break at the exit of one steam generator (initial flow - 518 percent of rated value) are shown in Figures 3.33 to 3.38. The core returns to criticality at about 20 sec. The power reaches a peak value of about 532 MW (35 percent of rated power) at 90 sec with a corresponding peak core average heat flux of 61,670 Btu/hr-ft^2. At this time, the borated water reaches the core, initiating a power decrease. As the core parameters (pressure, flow, inlet enthalpy) at the time of peak heat flux are outside of the range of the W-3 correlation, the critical heat flux was determined using the modified Macbeth CHF correlation.(5) At the time of maximum core average heat flux, the margin to the critical heat flux is minimized. Using the core conditions for this time (22 percent of rated core averate heat flux, 386 psia, and 387°F inlet temperature) and applying a conservatively large local hot rod peaking (F_Q^T = 14.0), the minimum CHF ratio was calculated to be 1.58.

The responses to a small steam line break (273 lb/sec) are shown in Figures 3.39 to 3.44. The boron injection again is triggered by a low pressurizer pressure signal in combination with low pressurizer level. Borated water from the safety injection system reaches the core at 175 sec. The shutdown margin of 1.8 percent ensures that the core does not go critical following the small steam line break.

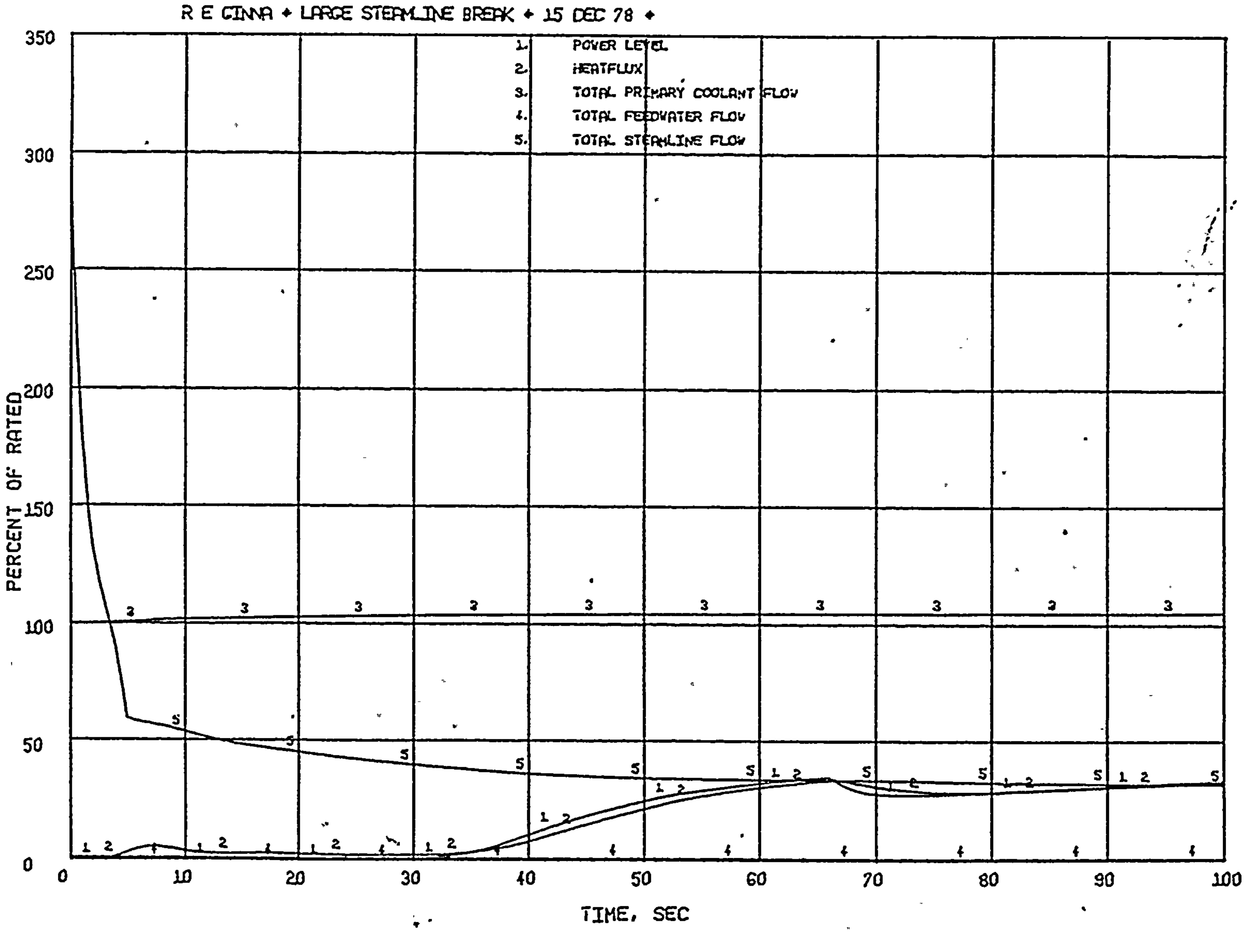

Figure 3.33 Power, heat flux and system flows - large steam line break

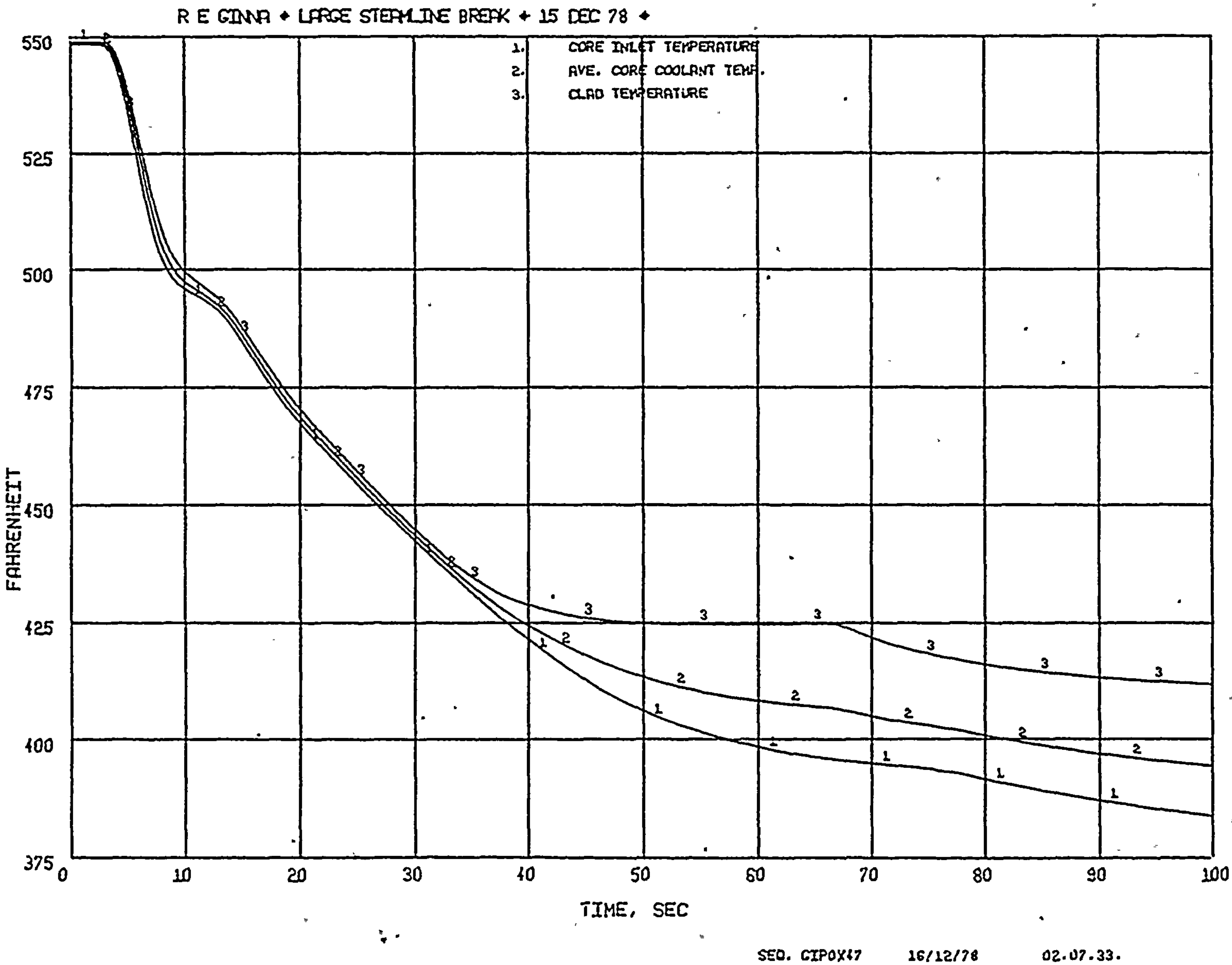

Figure 3.34 Core temperature response - large steam line break.

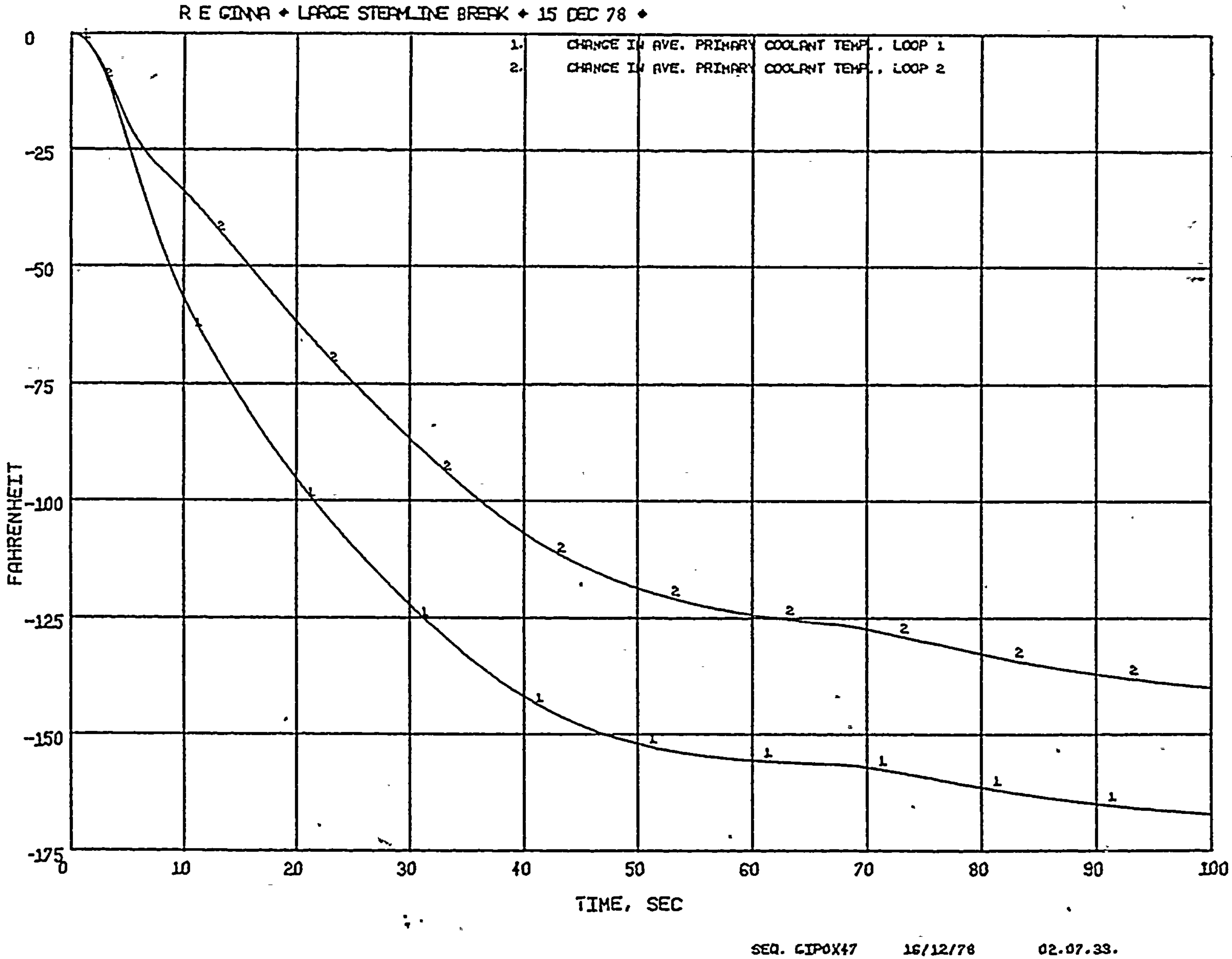

Figure 3.35 Primary loop temperature changes - large steam line break.

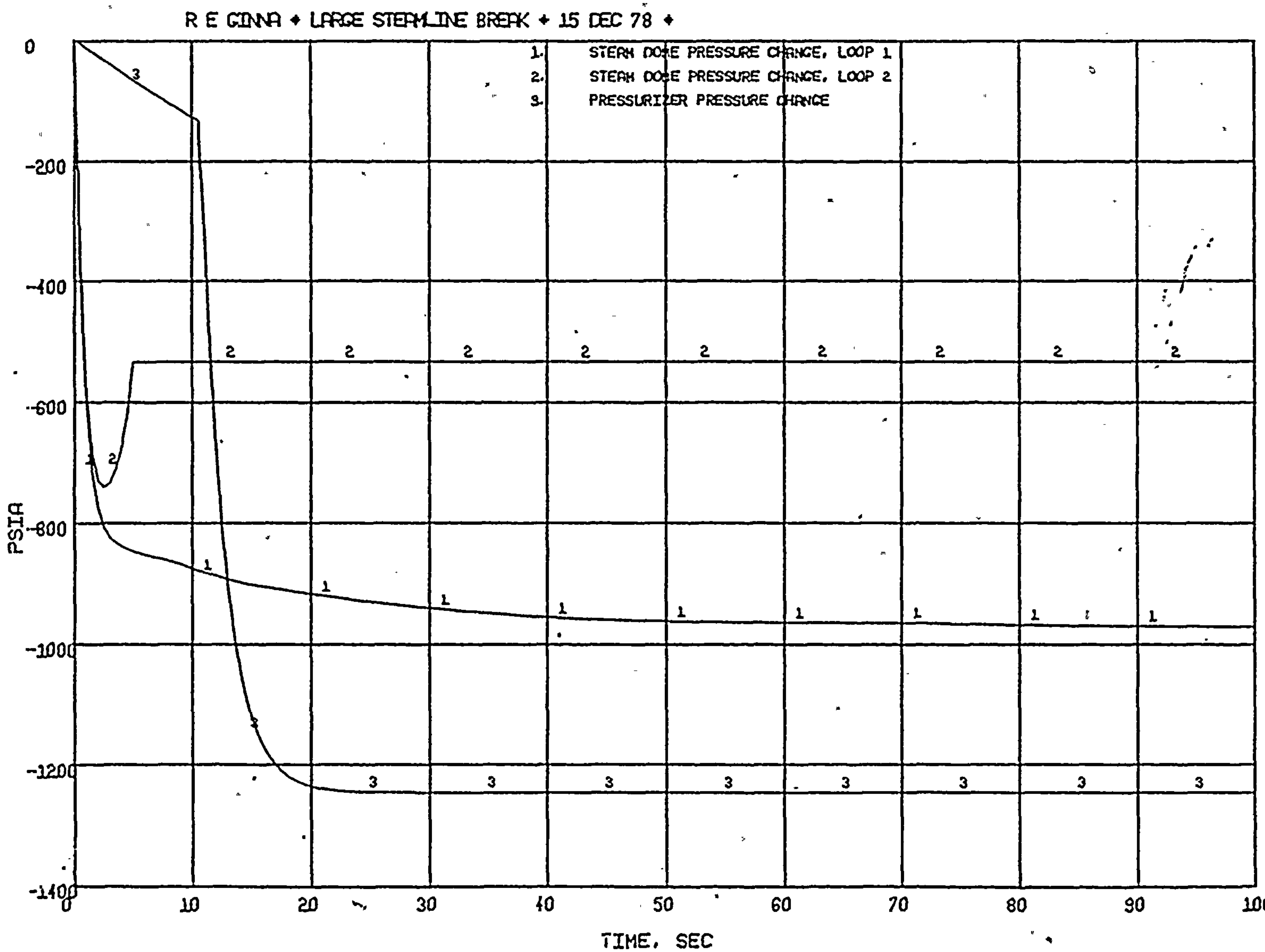

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Figure 3.36 Pressure changes in pressurizer and steam generators - large steam line break

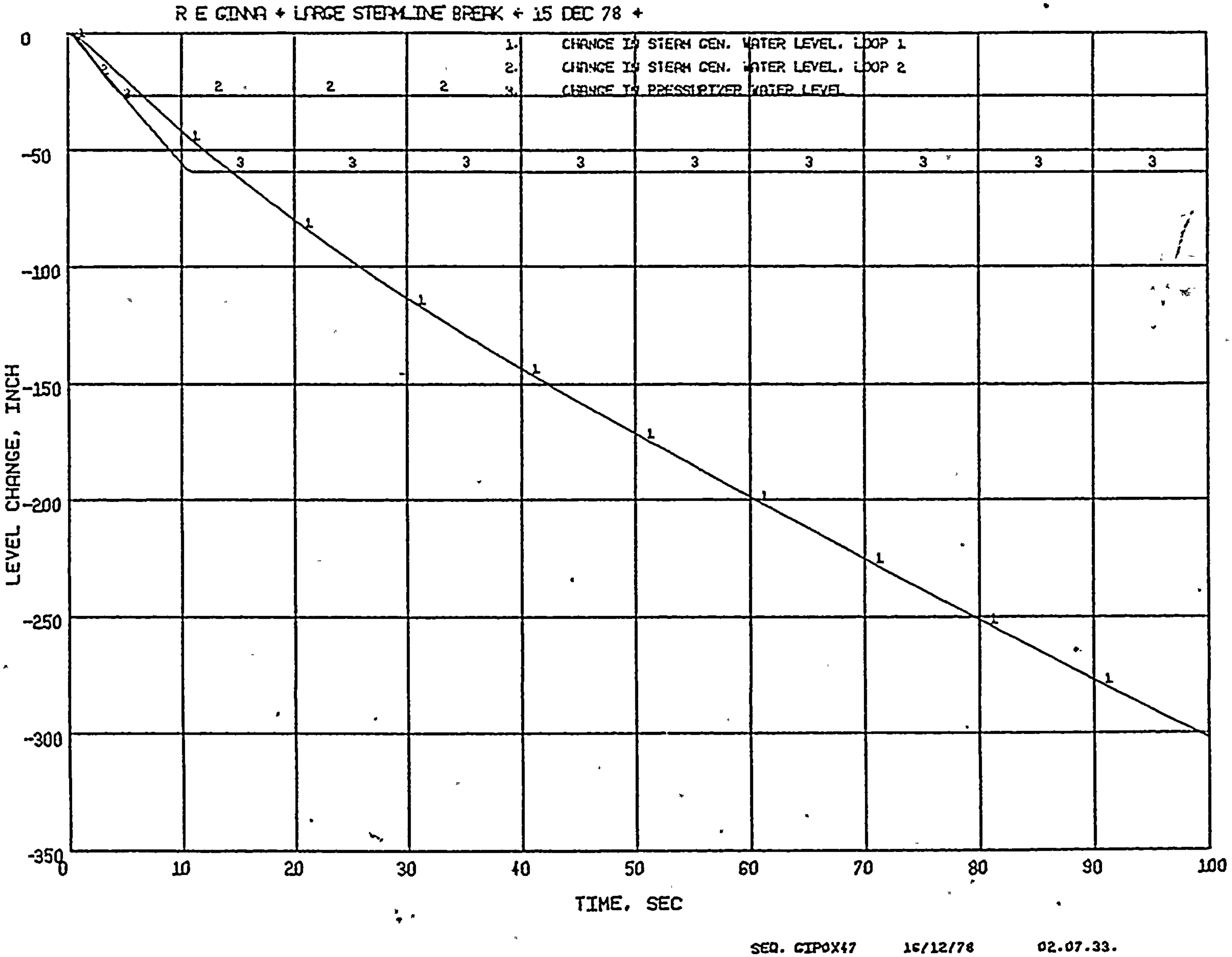

Figure 3.37 Level changes in pressurizer and steam generators - large steam line break.

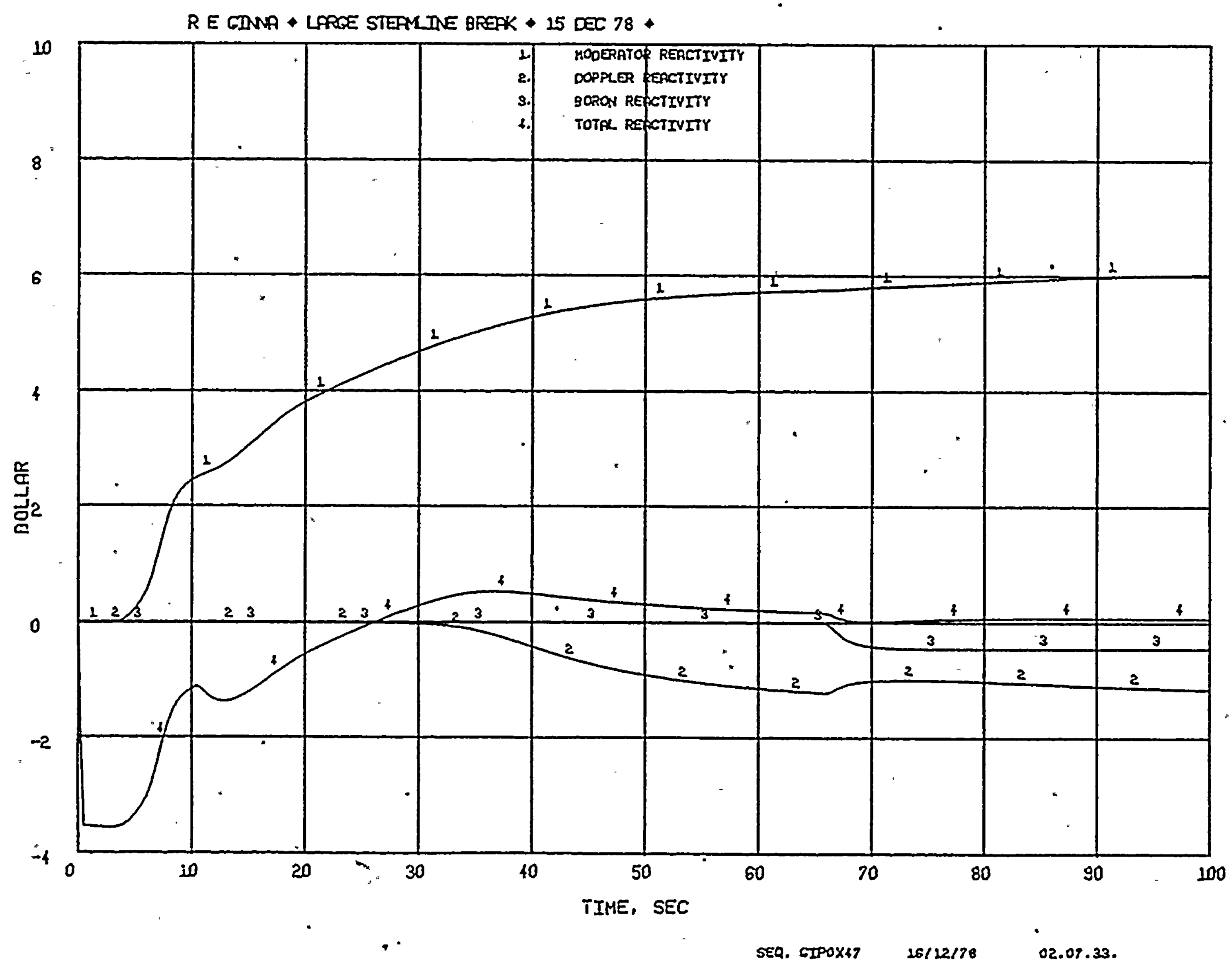

Figure 3.38 Nuclear reactivity feedback effects - large steam line break.

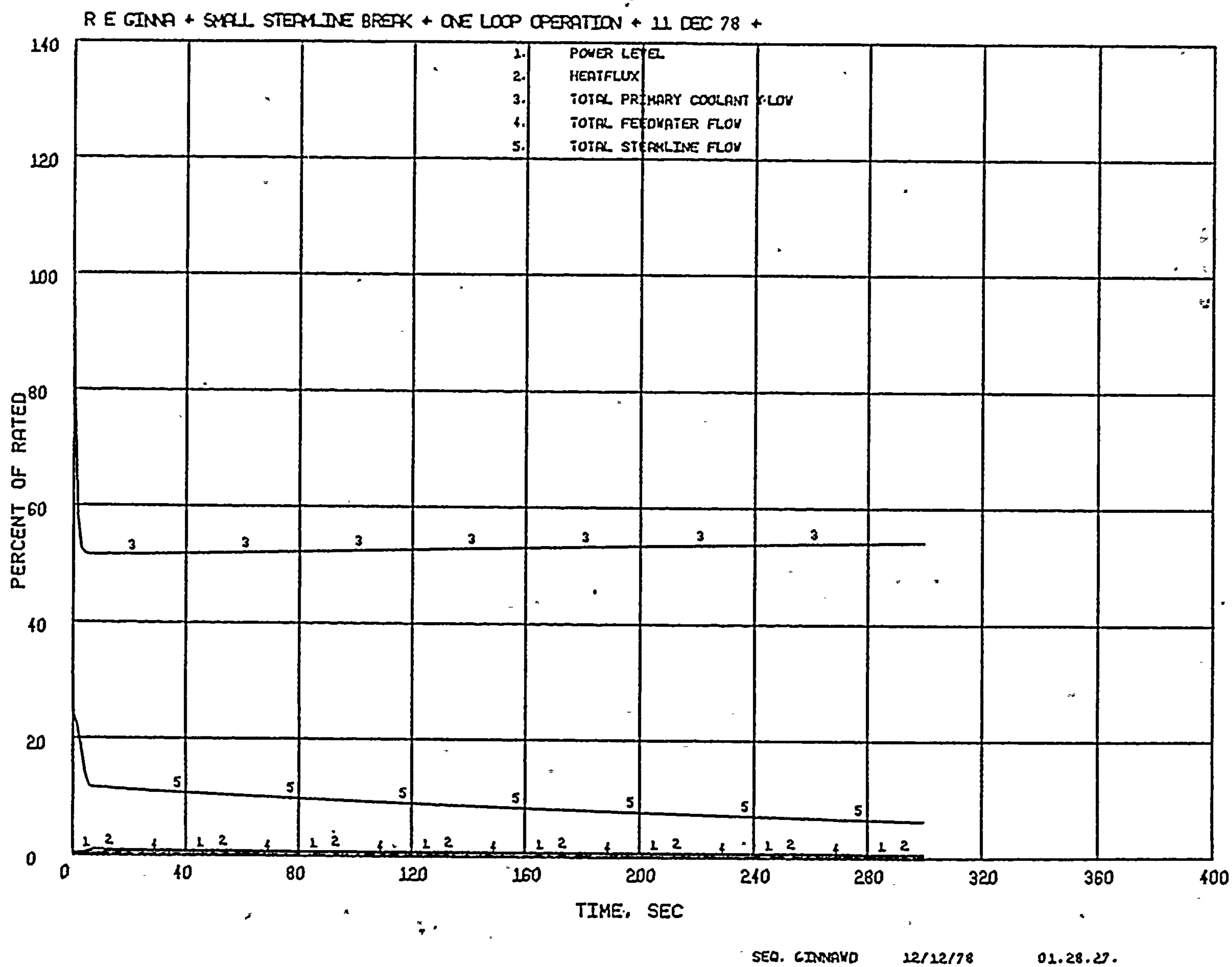

Figure 3.39 Power, heat flux and system flows - small steam line break.

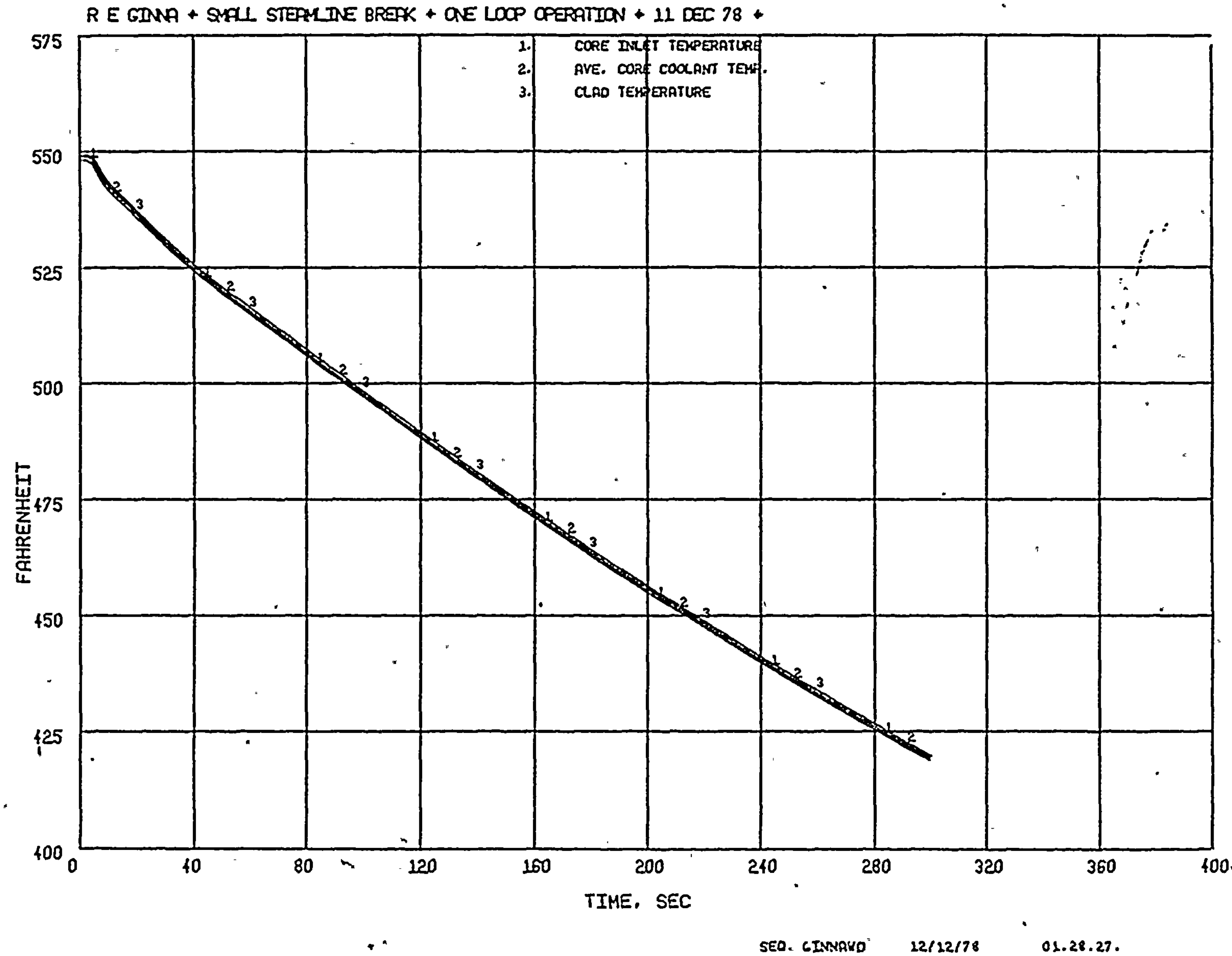

Figure 3.40 Core temperature response - small steam line break.

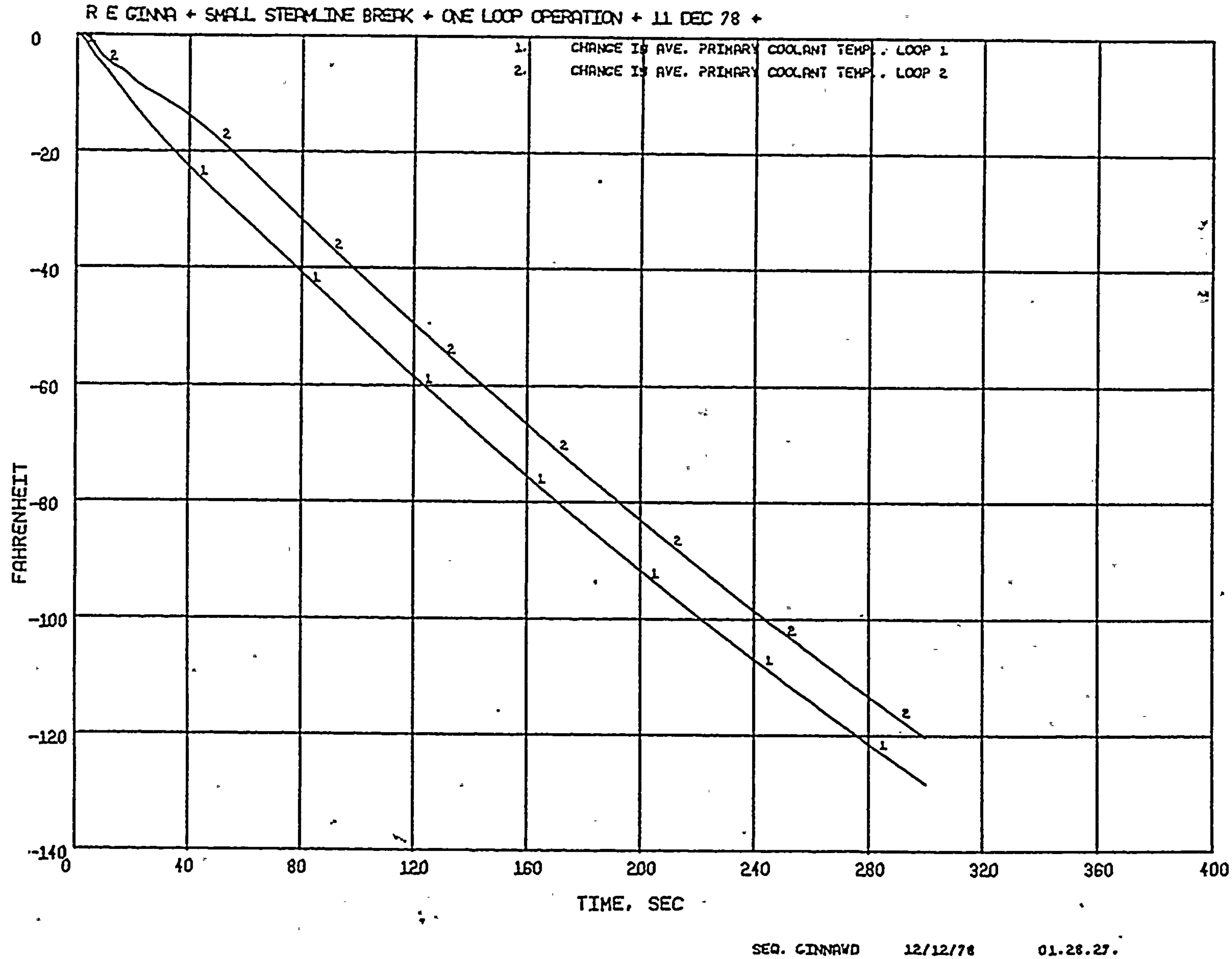

Figure 3.41 Primary loop temperature changes - small steam line break.

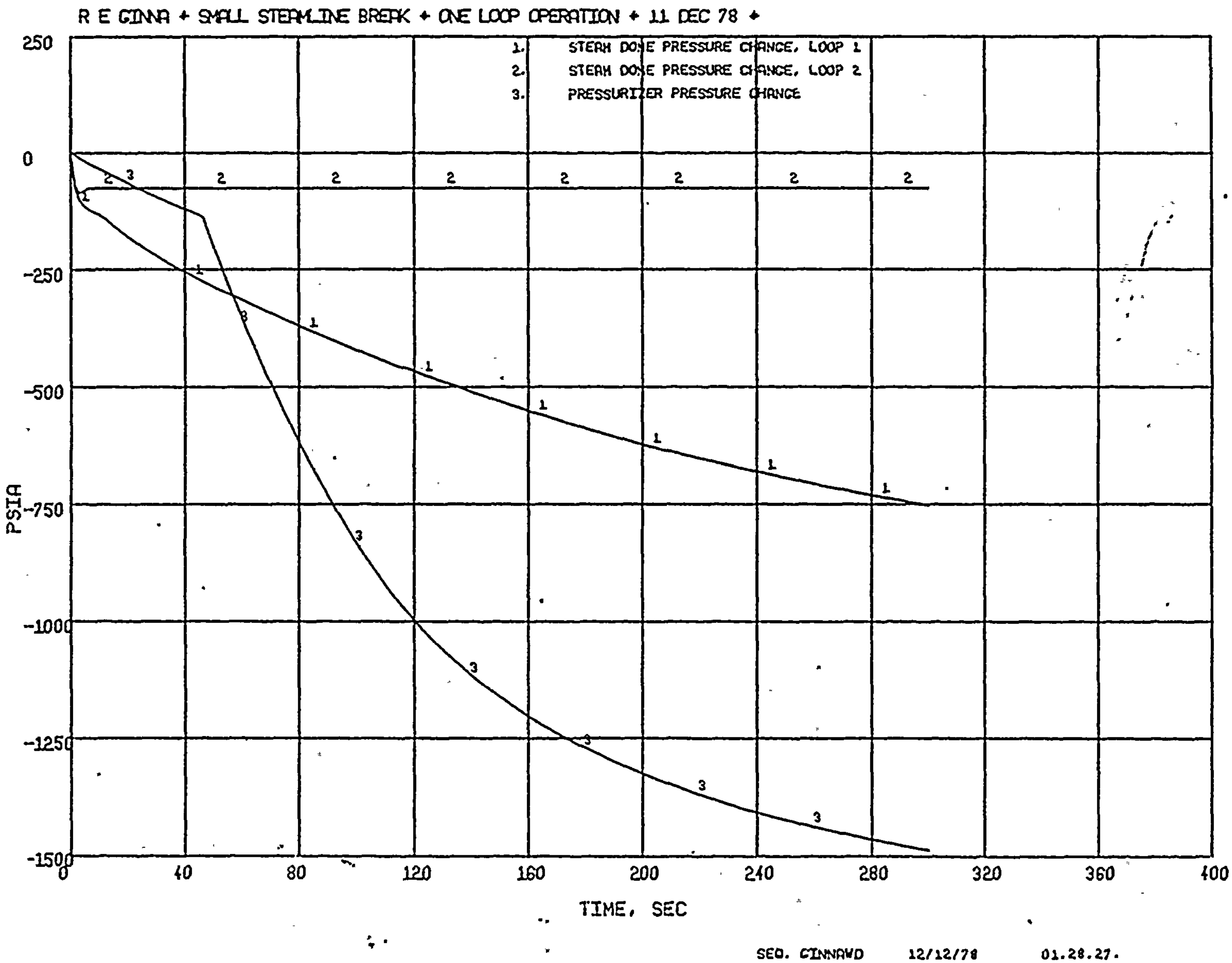

Figure 3.42 Pressure changes in pressurizer and steam generators - small steam line break.

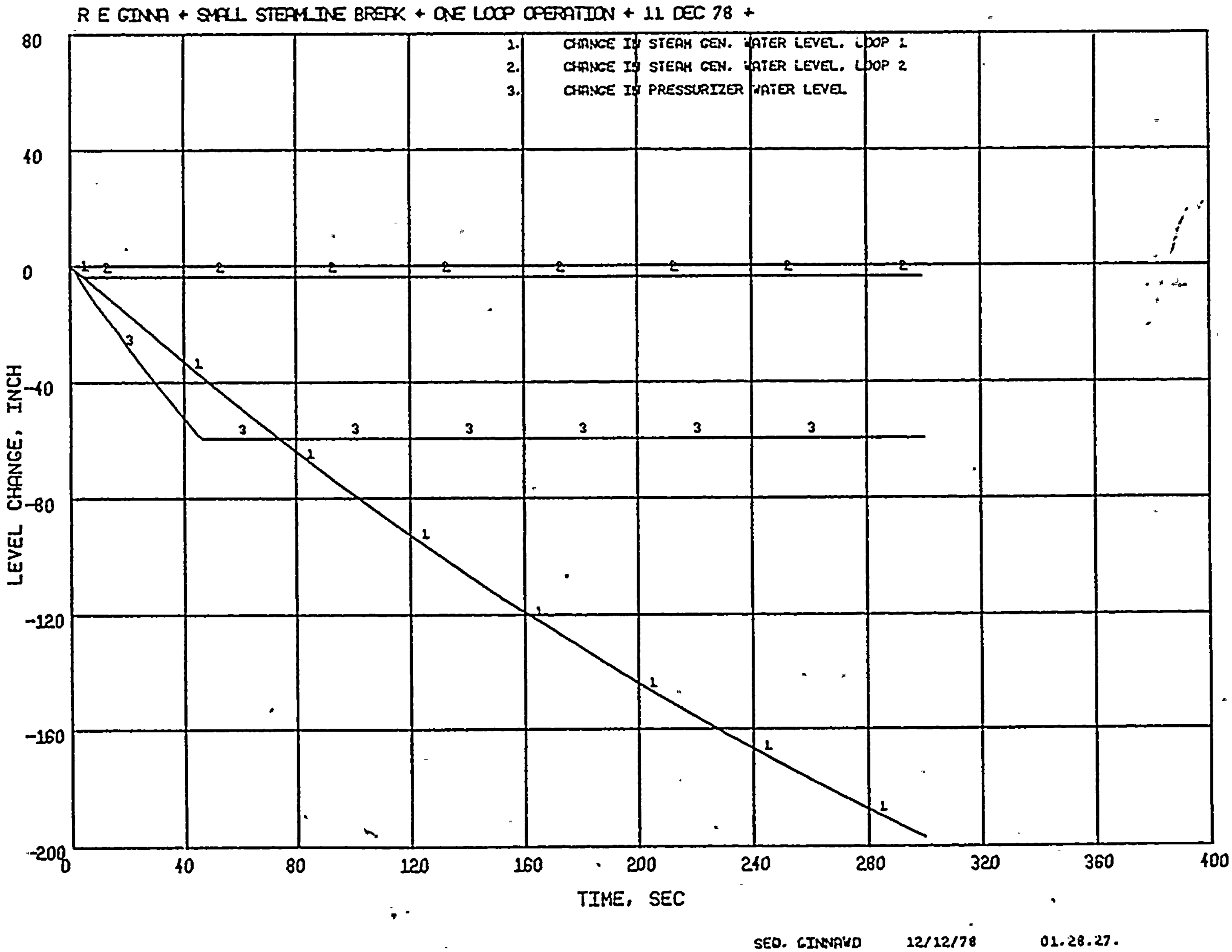

Figure 3.43 Level changes in pressurizer and steam generators - small steam line break.

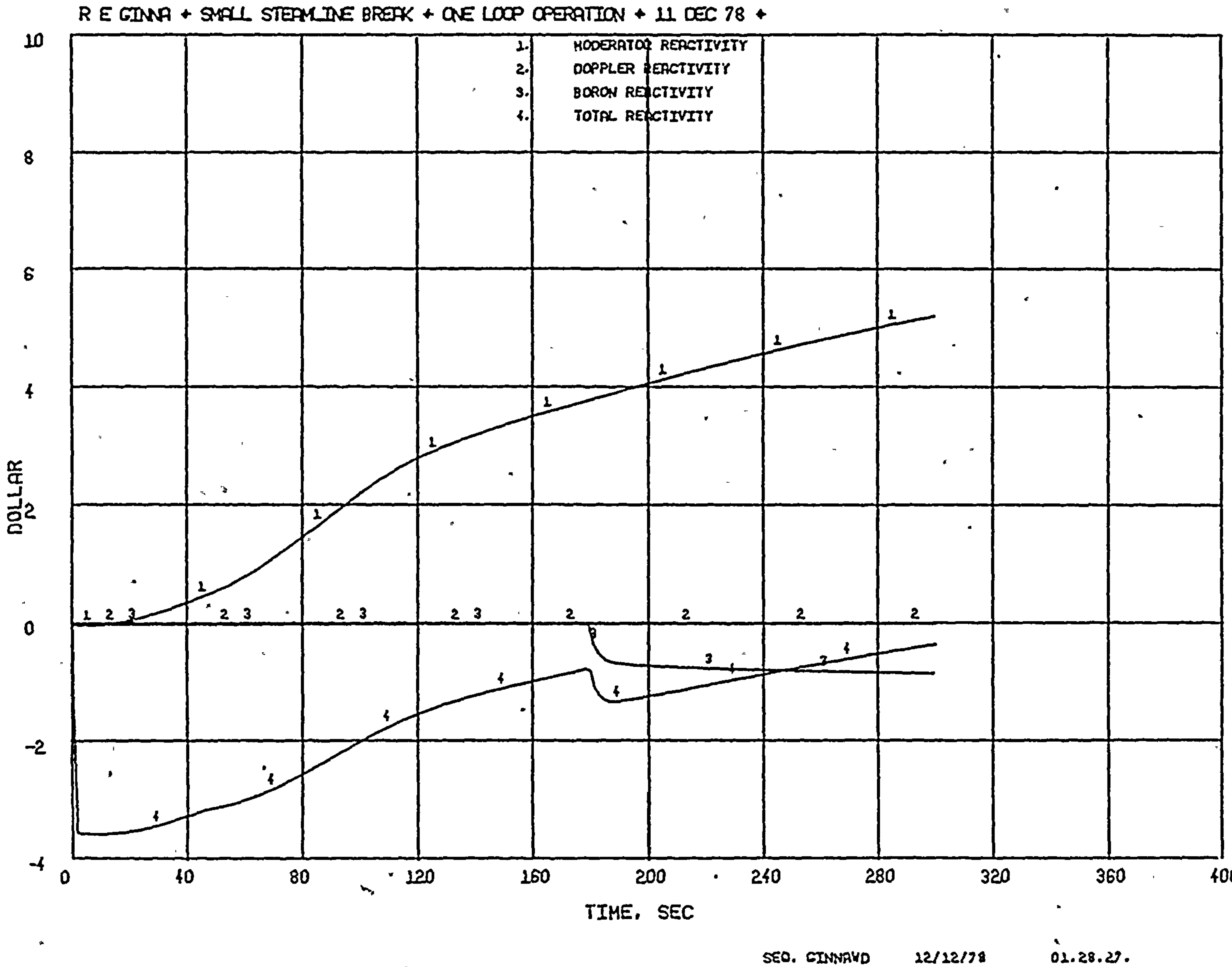

Figure 3.44 Nuclear reactivity feedback effects - small steam line break.

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R. E. Ginna Unit 1 Nuclear Power Plant

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