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PLANT TRANSIENT ANALYSIS FOR THE PRAIRIE ISLAND NUCLEAR POWER PLANT UNITS 1 & 2

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By

F. J. MARKOWSKI

SES 11/17/78

Approved: K P Galbraith 11-29-78
K. P. Galbraith, Manager
Nuclear Safety Engineering

Approved: George A Sofer 11-29-78
G. A. Sofer, Manager
Nuclear Fuels Engineering

Approved: G J Busselman 11/30/78
G. J. Busselman, Manager
Contract Performance

Approved: 11/30/78
W. S. Nechodom, Manager
Licensing and Compliance

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1.0 INTRODUCTION AND SUMMARY

The Cycle 5 extended burnup reload of the Prairie Island Nuclear Power Plant with Exxon Nuclear fuel results in core parameter values only slightly different from previous cycle values. The only significant difference is a slightly positive moderator temperature feedback coefficient at low power operation during the initial part of the cycle. The positive moderator coefficient results from the extended burnup requested for Prairie Island Cycle 5. The coefficient is positive at low power, approaches zero at about 70 percent power and is calculated to be $-36.9 \pm 20 \times 10^{-6}$/F at full power for beginning of Cycle 5. The reload fuel design has been shown to be both neutronically and hydraulically compatible with the existing fuel, and thus, the system response during plant transients would not be expected to be particularly sensitive to the fuel type. To demonstrate that the reload fuel meets plant regulatory requirements during design basis events, the most limiting transients identified for the existing fuel were reanalyzed with Exxon Nuclear fuel using the Exxon Nuclear plant transient simulation code PTSPWR2.[1] This report presents the results of the analysis of the following design basis events, as well as the input parameters used to simulate the reactor system.

Event	Incident Clas *
1. Fast Control Rod Withdrawal	II
2. Slow Control Rod Withdrawal	II
3. Loss of Power to Both Reactor Coolant Pumps	III
4. Locked Rotor in One Reactor Coolant Pump	IV
5. Loss of Electric Load	II
6. Large Steam Line Break	IV
7. Small Steam Line Break	IV

* Consistent with current FSAR incident classification for PWR's.

Events 1 through 5 are initiated from full power, while events 6 and 7 are initiated from hot standby conditions. The criteria to be satisfied in the Class II and III full power events are a peak system pressure of $\leq$ 2750 psia and a Minimum Departure from Nucleate Boiling Ratio (MDNBR) of $\geq$ 1.30 based on the W-3 correlation.[2] The criterion for the steam line break is that the end-of-cycle shutdown margin be adequate to ensure (1) the design thermal margin, MDNBR $\geq$ 1.30 for the large break, and (2) that the core does not become critical from hot standby following a small break.

The analysis is based on an equilibrium ENC fueled core using conservative neutronic parameters calculated for ENC fuel. The results of the analysis are summarized in Table 1.1. The lowest MDNBR for Class II and III events was 1.87, which is above the acceptable minimum of 1.30. The locked rotor incident, a Class IV event, was analyzed and the MDNBR was found to be 1.09. This result is acceptable for this low probability incident. The peak pressure criterion for the reactor coolant system was met in all cases. The small steam line break analysis showed that the smallest expected shutdown margin at the end of Cycle 5 is adequate to prevent return to criticality during such an event.

The analysis is valid for a maximum power peaking factor of $F_Q^T = 2.32$ and an axial power peaking factor of $F_Z^N = 1.45$, with the axial peak located at $X/L \leq 0.60$.

TABLE 1.1

SUMMARY OF RESULTS

Transient (Class)	Maximum Power Level (Percent)	Maximum Core Average Heat Flux (Btu/hr-ft^2)	Maximum Pressurizer Pressure (psia)	MDNBR (W-3)
Initial Conditions For Transients	102	194,790	2220	2.32
Fast Control Rod Withdrawal (II)(10^{-3}/sec)	134	213,870	2229	1.97
Slow Control Rod Withdrawal (II)(25×10^{-6}/sec)	112	210,022	2279	2.03
Loss of Flow - (III) 2 Pump Coastdown	103	194,840	2246	1.87
Loss of Flow - (IV) Locked Pump Rotor	105	194,840	2277	1.09
Loss of Load (II)	105	194,840	2511	2.16
Large Steam Line Break (IV)	52	38,600	*	1.35
Small Steam Line Break (IV) **	**	**	*	-

* Pressure decreases from initial value.
** The core does not go critical.

2.0 CALCULATION METHODS AND INPUT PARAMETERS

The transient analysis for the Prairie Island plant was performed using the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR2).[1] The PTSPWR2 code is an Exxon Nuclear digital computer program developed to model the behavior of pressurized water reactors under normal and abnormal operating conditions. The model is based on the solution of the basic transient conservation equations for the primary and secondary coolant systems. The transient conduction equation is solved for the fuel rods, and the point kinetics equation is used to calculate the core neutronic behavior. The program calculates fluid conditions such as flow, pressure, mass inventory and steam quality, heat flux in the core, reactor power, and reactivity during the transient. Various control and safety system components are included as necessary to analyze postulated events. A hot channel model is included to trace the departure from nucleate boiling (DNB) during transients. The DNB evaluation is based on the hot rod heat flux in the high enthalpy rise subchannel and uses the W-3 correlation[2] to calculate the DNB heat flux. Model features of the PTSPWR2 code are described in detail in Reference 1.

A diagram of the system model used by PTSPWR2 is shown in Figure 2.1. As illustrated, the PTSPWR2 code models the reactor, two independent primary coolant loops including all major components (pressurizer, pumps), two steam generators, and the feedwater lines and steam lines, including all major valves (turbine stop valves, isolation valves, pressure relief valves; etc.).

To ensure conservative predictions of system responses with resulting minimum values for the DNB flux ratios, as well as maximum values for the system peak pressure, conservative assumptions are applied to the input data. These assumptions can be grouped into three general categories:

1. Generic assumptions, applicable to all transients, based on steady-state offsets.
2. Assumptions which conservatively encompass ENC neutronic parameters.
3. Transient specific assumptions yielding the most adverse system responses.

The generic assumptions (Category 1) are applied to all full power transients to account for steady-state and instrumentation errors. The initial core conditions are obtained by adding the maximum steady-state errors to the rated values as follows:

Reactor Power	= 1650 MWt + 2% (33 MWt) for calorimetric error.
Reactor Inlet Temperature	= 530.5 + 4°F for deadband and measurement error.
Primary Coolant System Pressure	= 2250 - 30 psia for steady-state fluctuation and measurement errors.

The combination of the above parameters acts to minimize the initial minimum DNB flux ratio. These values are consistent with those in the Plant Technical Specifications. Table 2.1 shows a list of operating parameters used in the analysis.

The trip setpoints incorporated into the PTSPWR2 model for the Prairie Island Plant are based on the Technical Specification limits and the assumptions used are consistent with those used in the reference cycle

analysis (Ref. 3). These limiting trip setpoints with their associated time delays for each trip function are listed in Table 2.2.

The design parameter values for the Cycle 5 ENC fuel are summarized in Table 2.3. Table 2.4 lists the neutronic parameter values which conservatively bound the ENC fuel for both the beginning and the end of Cycle 5. A symmetric axial power profile with a peaking factor $F_z^N = 1.45$ was used. The scram reactivity curve used in the analysis is shown in Figure 2.2.

The assumptions in category 2 refer to the reactivity feedback effects from moderator temperature changes and Doppler broadening. For all BOC transients, a positive moderator temperature feedback has been used. To the Doppler feedback coefficient, an attenuation factor of 0.8 or a magnification factor of 1.2 has been applied, depending on which factor results in the worst case.

The assumptions in category 3 apply to plant control and protection systems. As an example, pressurizer spray and pressurizer relief valve action are ignored in the locked pump rotor transient. Since these assumptions are considered separately for each transient, they are detailed in Section 3 where each transient is described. The conservatisms applied to each transient analyzed are usually identical to those used in the reference cycle analysis.[3] The assumptions are quite standard, as given by any PWR FSAR or other ENC safety analysis reports.[5]

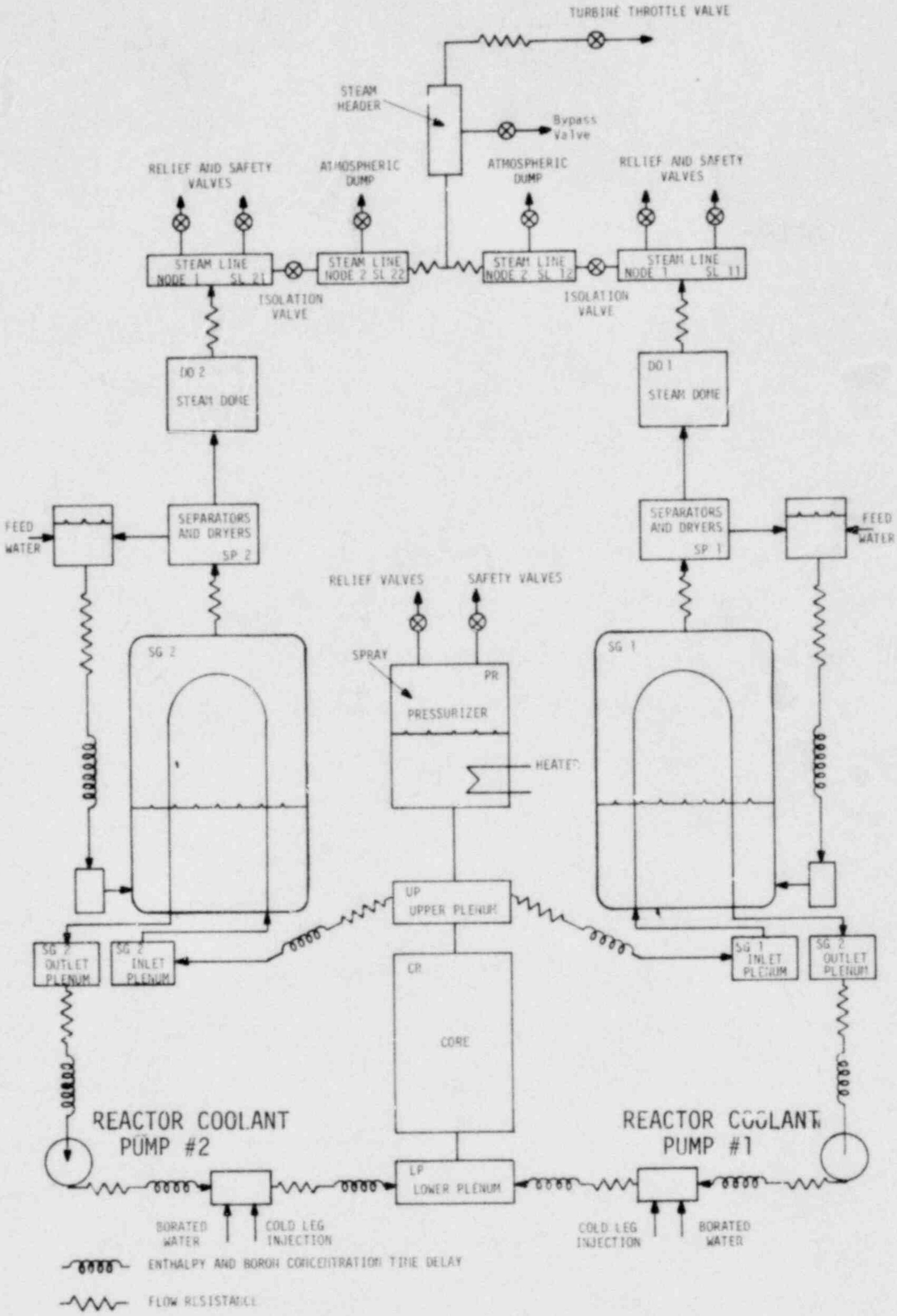

FIGURE 2.1 PTSPWR2 SYSTEM MODEL

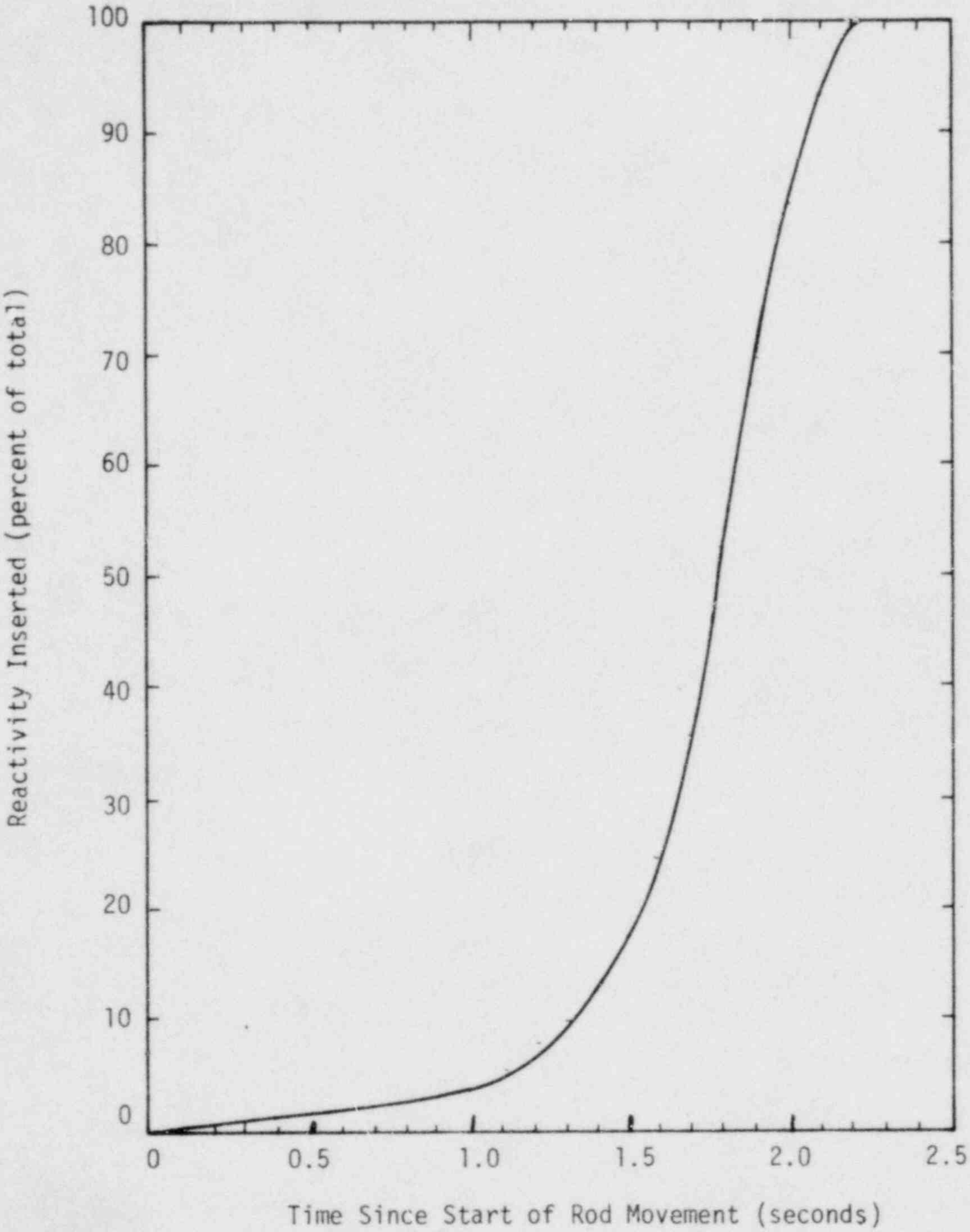

FIGURE 2.2 SCRAM REACTIVITY CURVE USED IN TRANSIENT ANALYSIS OF THE PRAIRIE ISLAND NUCLEAR PLANT

TABLE 2.1

PARAMETER VALUES USED IN PTSPWR2

ANALYSIS OF PRAIRIE ISLAND UNIT 1 AND 2

	Analysis Input Value
Core	
Total Core Heat Output, MW	1,683.0
Heat Generated in Fuel, %	97.4
System Pressure, psia	2,220.
Hot Channel Factors	
Total Peaking Factor, F_Q^T	2.32
Enthalpy Rise Factor, $F_{\Delta H}^N$	1.55
Total Coolant Flow, lb/hr	68.20×10^6
Effective Core Flow, lb/hr	64.54×10^6
Reactor Inlet Temperature, °F	534.5
Heat Transfer	
Calculated Average Heat Flux* Btu/hr-ft^2	190,973
Steam Generators	
Calculated Total Steam Flow** lb/hr	7.233×10^6
Steam Temperature, °F	510.9
Feedwater Temperature, °F	427.3

* Calculated from total thermal power and total cladding surface.
** Calculated from thermal power, feedwater and steam conditions.

TABLE 2.2

PRAIRIE ISLAND UNIT 1 AND 2 TRIP SETPOINTS

	Setpoint	Used in Analysis	Delay Time
High Neutron Flux	108%	118%	0.5 sec
Low Reactor Coolant Flow	90%	87%	0.6 sec
High Pressurizer Pressure	2400 psia	2400 psia	1.0 sec
Low Pressurizer Pressure	1830 psia	1700 psia	1.0 sec
High Pressurizer Water Level	90% of Span	100% of Span	1.5 sec
Low-Low Steam Generator Water Level	5% of Span	0% of Span	1.0 sec
Overtemperature ΔT*	T_{AVE_o} = 567.3°F	T_{AVE_o} = 567.3°F	6.0 sec
	P_o = 2250 psia	P_o = 2250 psia	
High Pressure Safety Injection	a) 1830 psia coincident with 5% level in pressurizer	1800 psia coincident with 0% level in pressurizer	25 sec
	b) 3.32 ft steam generator level	3.32 ft steam generator level	25 sec

* The overtemperature ΔT trip is a function of pressurizer pressure, coolant average temperature, and axial offset. The T_{AVE_o} and P_o setpoints are contained within the functional relationship.

TABLE 2.3

PRAIRIE ISLAND UNIT 1 AND 2 FUEL DESIGN PARAMETERS

FOR EXXON NUCLEAR FUEL, CYCLE 5

Fuel Pellet Diameter	0.3565	Inch
Inner Cladding Diameter	0.3640	Inch
Outer Cladding Diameter	0.4240	Inch
Active Length	144.0	Inch
Number of Fuel Rods in Core	21,659	

TABLE 2.4

PRAIRIE ISLAND UNIT 1 AND 2

ENC KINETIC PARAMETERS

Symbol	Parameter	Value Beginning-of-Cycle	Value End-of-Cycle
α_M	Moderator Coefficient (pcm/F)	+2.0*	-35.0
α_D	Doppler Coefficient (pcm/F)	-1.25	-1.60
α_P	Pressure Coefficient (pcm/psi)	-0.02	+0.40
α_V	Moderator Density Coefficient pcm/(g/cm^3)	-1,800.	+31,500.
α_B	Boron Worth Coefficient (pcm/ppm)	-7.70	-8.60
β_{eff}	Delayed Neutron Fraction (pcm)	610	510
α_{CRC}	Total Rod Worth (pcm)	-2,120.**	-2,830.**

* This value applies to hot standby conditions. The coefficient is calculated to approach zero at about 70 percent power, and -3.69 ± 2 pcm at full power (6).

** These are conservative values, for analysis purposes only. The actual plant values are significantly higher.

3.0 TRANSIENT ANALYSIS

3.1 FAST CONTROL ROD WITHDRAWAL

The withdrawal of control rods adds reactivity to the reactor core causing both the power level and the core heat flux to increase. Since the heat extraction from the steam generator remains relatively constant, there is an increase in primary coolant temperature. Unless terminated by manual or automatic action, this power mismatch and the resultant coolant temperature rise could eventually result in a DNB flux ratio of less than 1.3. While the inadvertent withdrawal of control rods is unlikely, the reactor protection system is designed to terminate such a transient while maintaining an adequate margin to DNB. Two potential causes for such an incident are: 1) operator error, and 2) a malfunction in the reactor regulating system or rod drive control system resulting in continuous withdrawal of a control rod group.

In this incident, the reactor is tripped by the nuclear overpower function. The rod withdrawal rate was chosen to give the most severe thermal response based on established core limit curves.[3] The analysis is presented here to provide a check on those limits. The fast rod withdrawal was analyzed from an initial power level of 1683.0 MWT. The reactivity insertion rate used is consistent with the rates analyzed in the reference cycle analysis.[3] Beginning-of-cycle kinetic coefficients were used with an appropriate multiplier applied to the Doppler coefficient (see Table 2.5).

Figures 3.1 to 3.6 show plant responses for a fast rod withdrawal, $\dot{\Delta K} = 1.0 \times 10^{-3}$ 1/sec, from full power. A nuclear overpower trip (118% setpoint) occurs at 1.36 seconds. The DNB flux ratio drops from an initial value of 2.32 to 1.97. Pressure increases to a maximum

of 2229 psia, with core average temperature increasing by less than 2°F.

3.2 SLOW CONTROL ROD WITHDRAWAL

The slow control rod withdrawal results in a smooth heatup of the primary system, limited by the overtemperature ΔT or the overpower ΔT function long before any significant level of overpower is reached. Based on the reference cycle analysis, a withdrawal value of $\dot{\Delta K} = 25.0 \times 10^{-6}$ 1/sec was chosen.

The plant responses for the slow rod withdrawal are presented in Figures 3.7 to 3.12. The overtemperature ΔT trip setpoint is reached at 35 sec, and the shutdown rod insertion starts after a 6.0 sec delay. The minimum DNB flux ratio is 2.03 at about 35 sec.

3.3 LOSS OF REACTOR COOLANT FLOW

Flow coastdown accidents resulting from a loss of electric power to the primary coolant pumps result in a rapid increase in coolant temperature, which combined with the reduced flow, reduces the heatflux margin to DNB. Only the most severe case is analyzed: Loss of both pumps from the reactor system operating at 1683.0 MWT resulting from simultaneous loss of power to the pumps. Beginning-of-cycle values for kinetic coefficients are assumed. For conservatism a multiplier of 0.8 was applied to the Doppler coefficient. The loss of power to all pumps will result in a reactor trip due to either undervoltage or underfrequency at the bus. For conservatism, however, the trip was taken to be on a low flow signal. This allows a further flow reduction at full power, and a more conservative calculation of heatflux margin to DNB.

Figures 3.13 - 3.18 present plant responses after the loss of both pumps. A reactor trip occurs at 2.7 sec. A minimum DNB flux ratio of 1.87 is reached at 3.7 sec after beginning of coastdown. At about 5 sec, a pressure peak of 2246 psia is reached.

3.4 LOCKED PUMP ROTOR

In the unlikely event of a seizure of a primary coolant pump, flow through the core is drastically reduced. The reactor is tripped by the resulting low flow signal. The coolant enthalpy rises, decreasing the heat flux margin to DNB. The locked rotor transient was analyzed assuming two loop operation with instantaneous seizure of one pump from 102% of rated power.

The effect of the pressurizer spray and pressurizer relief valves on reducing system pressure was ignored in the analysis. Also, steam dump to the condenser was not allowed, and the feedwater pumps were assumed to trip with the reactor. Kinetic parameter values for the beginning of Cycle 5 have been used since they cause the most adverse plant response. A multiplier of 1.2 has been applied to the doppler coefficient. Two cases have been analyzed for this transient, one with $\dot{\Delta}k = +20.0 \times 10^{-6}$/F and one with $\dot{\Delta}k = -16.9 \times 10^{-6}$/F for the moderator temperature feedback coefficient (see Table 3.4). The first case is conservative since it combines the most positive hot standby feedback coefficient with assumed full power operation.

The conservative case (positive moderator coefficient) results in a DNB flux ratio of 1.09, the more realistic case yields a DNB flux ratio of 1.19. Results for the conservative case are reported. The transient responses are shown in Figures 3.19 to 3.24. The reactor is tripped at 0.7 sec by a low flow signal. The core average temperature

increases by 13 F with a system pressure reaching 2,277 psia, well below the power operated relief valve setting of 2,350 psia. The number of fuel rods expected to experience departure of nucleate boiling has been calculated to be less than 1 percent for the conservative case.

3.5 LOSS OF EXTERNAL ELECTRIC LOAD

The Prairie Island plant is designed to accept a 50 percent step decrease of electric load without a reactor trip. For a complete loss of electric load at full power, the reactor is tripped by a signal derived from the turbine stopvalves. In the analysis of this transient, it is conservatively assumed that only the turbine is tripped on the Loss of Electric Load signal, but not the reactor. In addition, the pressurizer spray system and the power-operated relief valves are assumed to be inoperative. On the secondary side, the turbine bypass into the condenser as well as the actuated steam relief valves are assumed to be inoperative. Neutronic data for the beginning of the cycle are used (positive moderater temperature feedback), and unavailability of the automatic reactor control is assumed. In addition, a factor of 0.8 is applied to the doppler coefficient.

The criteria for this transient are 1) the ability of the passive pressurizer safety valves to limit the reactor coolant system pressure to a value below 110 percent of the design pressure (2750 psia) in accordance with Section III of the ASME Boiler and Pressure Vessel Code and 2) a sufficient thermal margin in the hot fuel assembly to assure that no departure of nucleate boiling occurs throughout the transient.

Figures 3.25 through 3.30 show the plant responses for a complete loss of electric load at 102 percent of full power without a direct reactor trip. After closure of the turbine stop valves, the pressure in both steam generators increases at an average rate of 20 psi/sec, reaching 1090 psia at 13 sec, when the first set of steamline safety valves opens (see Figure 3.28). At 16 sec, the second safety valve setpoint of 1105 psia is reached. After that point, the steam pressure continually decreases. In the primary system, the pressure increases at the same average rate as in the secondary system, only delayed by about 5 sec (see Figure 3.28). The reactor is tripped on the overpressure signal at 13 sec, the peak pressurizer pressure is 2537 psia. The pressurizer safety valve is open from about 15.5 sec to 17.5 sec The average primary coolant temperature increases by about 23 F. The lowest value for the minimum DNB heatflux ratio is 2.16, at about 13 sec.

3.6 LARGE STEAMLINE BREAK

The break of a steam pipe (or safety valve failure) results in a sharp reduction in steam inventory in the steam generator. The resulting pressure decrease causes an energy demand from the primary coolant which reduces coolant temperature and pressure. With a negative moderator temperature feedback coefficient (at the end of the cycle), this causes a reactivity insertion into the core 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 5 fuel.

Minimum capability of the boron injection system was assumed, which implies that only one of the two 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 trip point (1800 psia). 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.

A large break at the exit of the steam generator with offsite power available was analyzed. A 10 sec delay was used to cover the startup time for the high pressure injection pump. An initial break flow of 600 percent of rated flow was chosen. Figures 3.33 to 3.38 show the plant responses. The core returns to criticality at about 10 sec. The power reaches a peak value of 52 percent of nominal full power at 56 sec with a corresponding peak in core average heat flux of 98,000 Btu/(hr x ft^2). At this time, the borated water from the high pressure safety injection system reaches the core, initiating a power decrease. A conservatively large local hot rod peaking) factor of F_Q^T = 10.0 was used. The lowest value for the heatflux margin to departure of nucleate boiling was 1.35, at about 54 sec (W-3 correlation). A shutdown reactivity of 1,800 pcm has been used.

3.7 SMALL STEAMLINE BREAK

The small steamline break transient is intended to envelope a valve failure. For instance, an actuated steamline relief valve or a turbine bypass valve could fail open and release steam. A small break at hot standby conditions, two-loop operation, with an initial steamflow of 25 percent of nominal full flow with offsite power available has been analyzed. The most significant parameter responses are presented in Figures 3.39 to 3.44. The boron injection is triggered by the same signal as in the large break case. The borated water reaches the core at about 150 sec, and the core does not become critical. A shutdown margin of 1,800 pcm has been used.

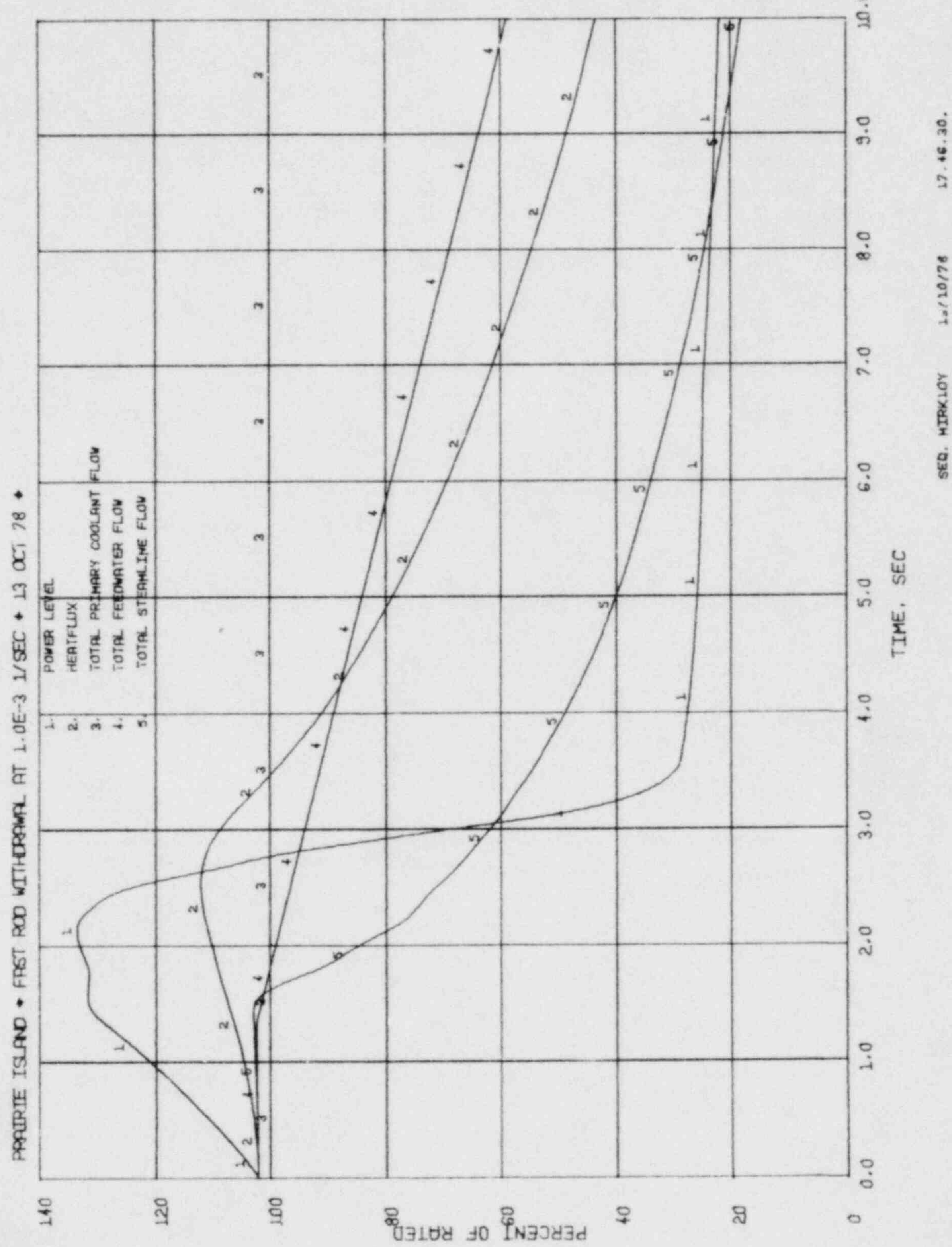

FIGURE 3.1 - Power, Heatflux and System Flows for Fast Control Rod Withdrawal

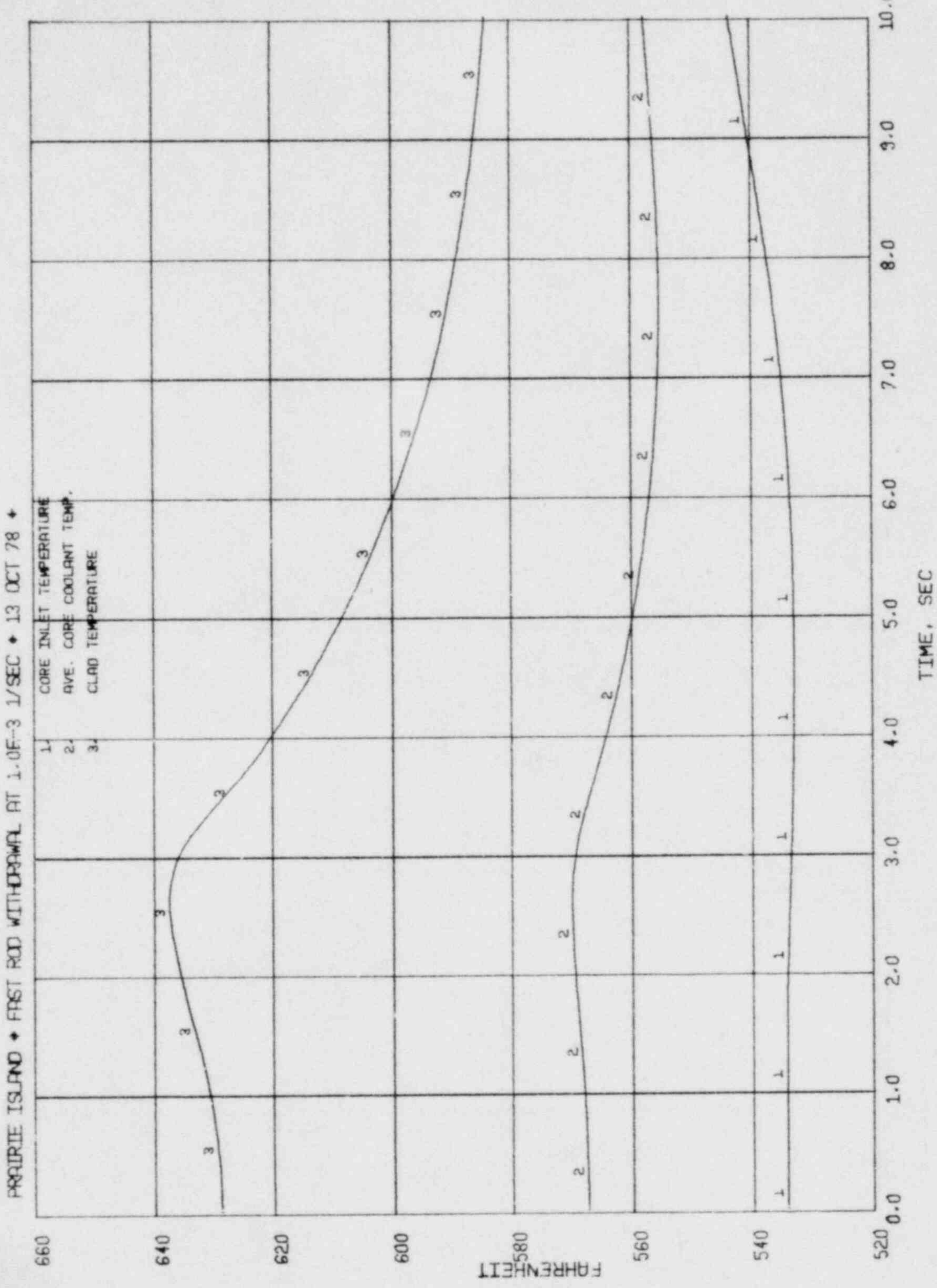

FIGURE 3.2 - Core Temperature Response for Fast Control Rod Withdrawal

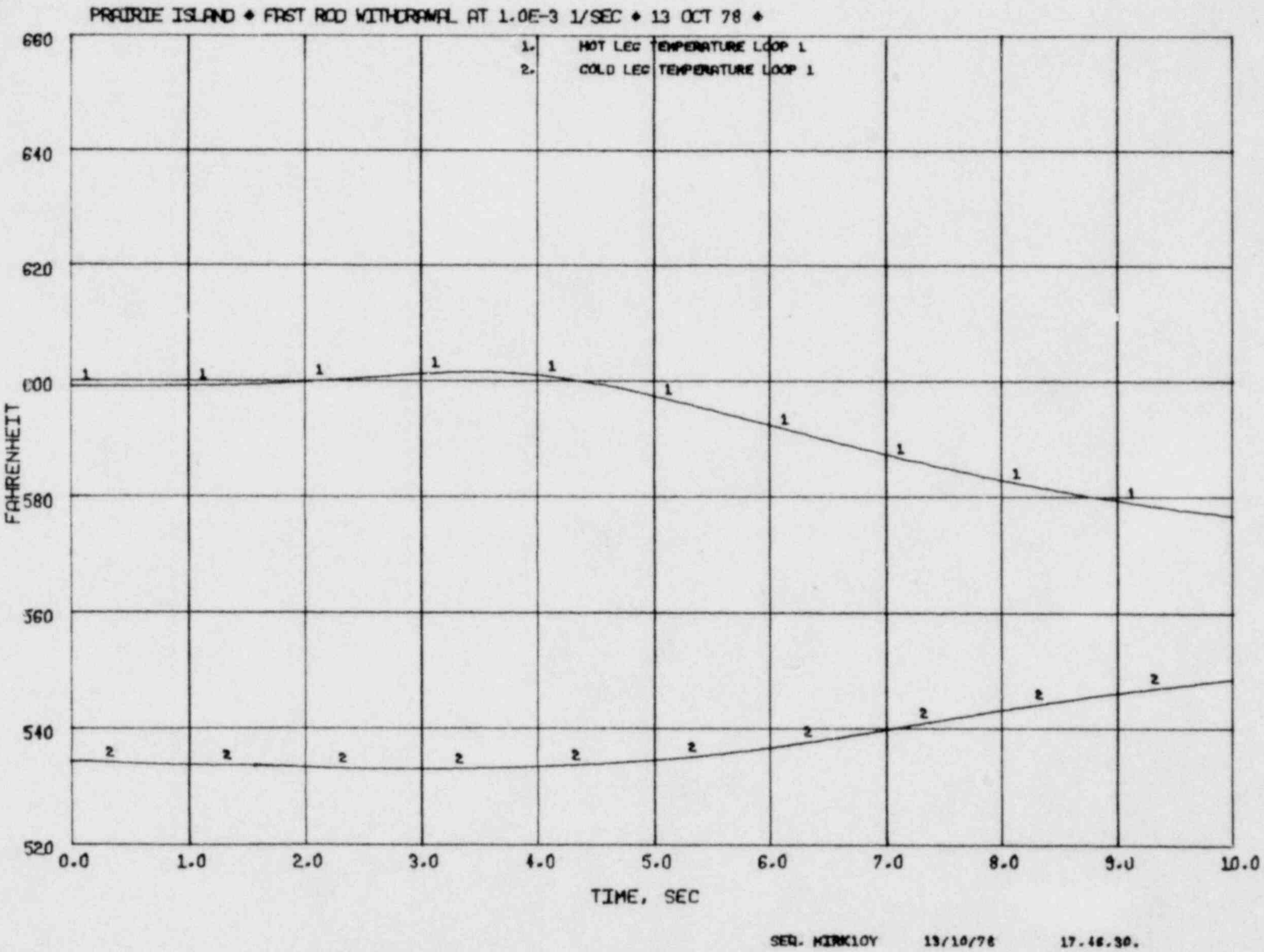

FIGURE 3.3 - Primary Loop Temperature Response for Fast Control Rod Withdrawal

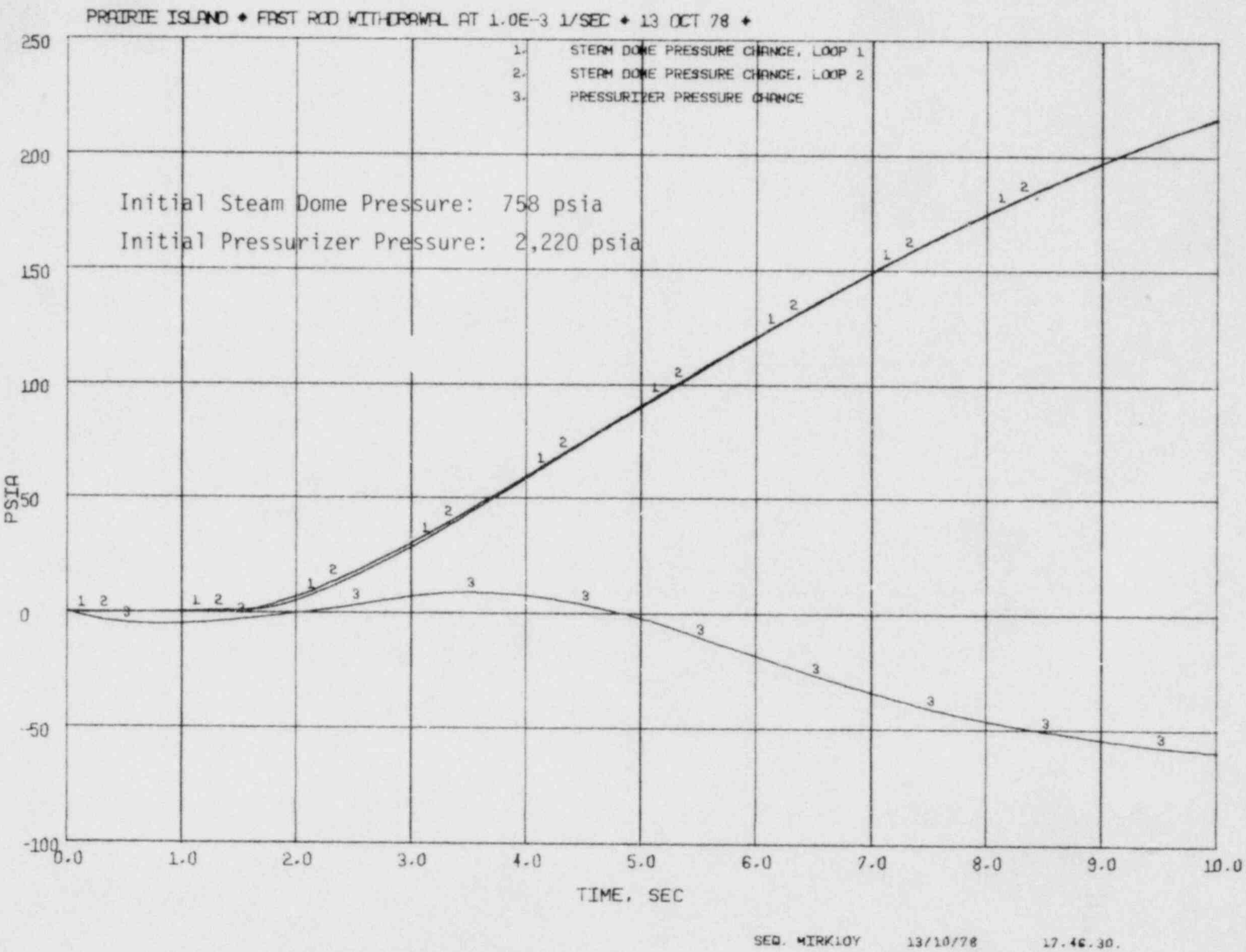

FIGURE 3.4 - Pressure Changes in Pressurizer and Steam Generators for Fast Control Rod Withdrawal

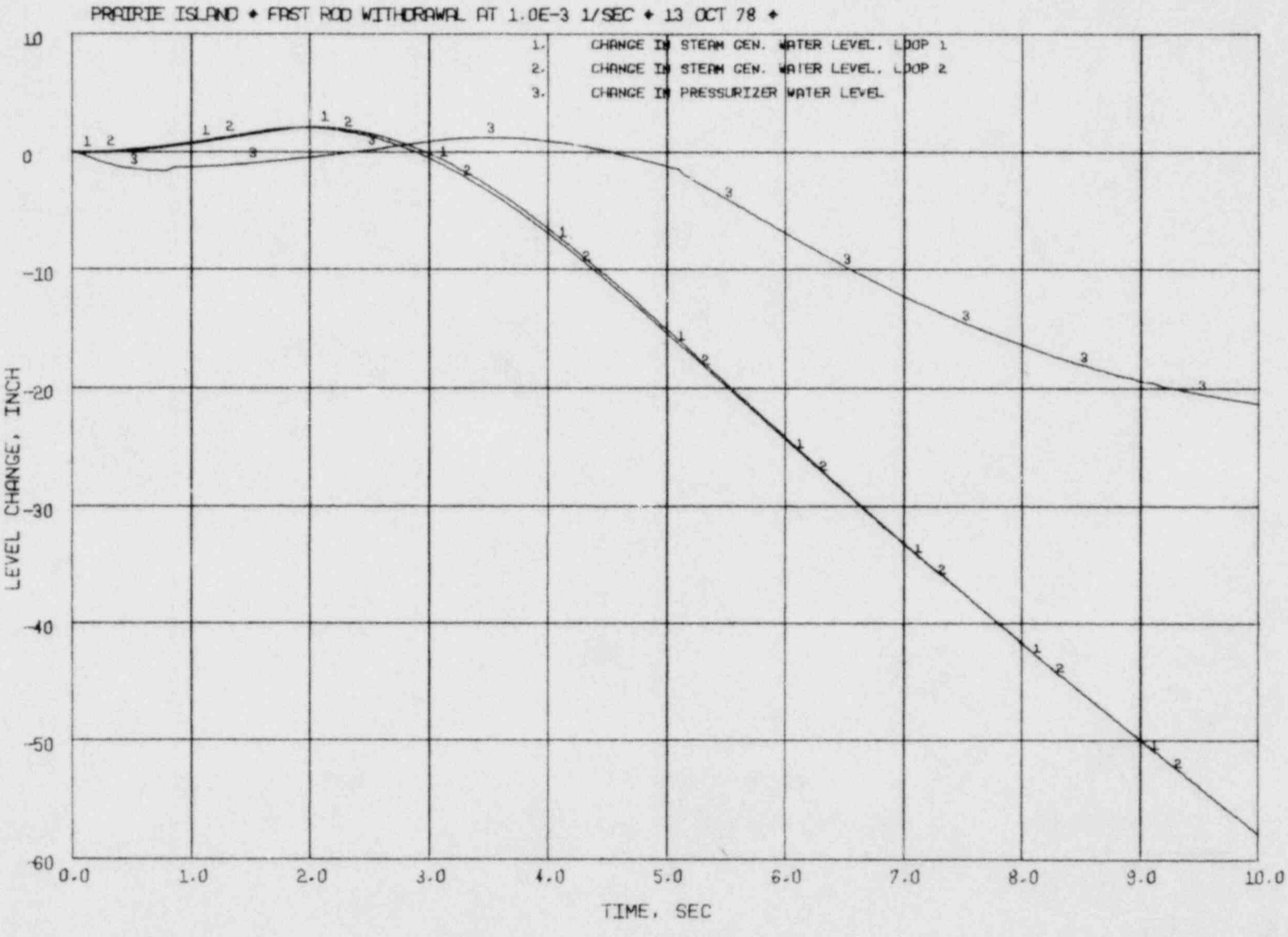

FIGURE 3.5 - Level Changes in Pressurizer and Steam Generators for Fast Control Rod Withdrawal

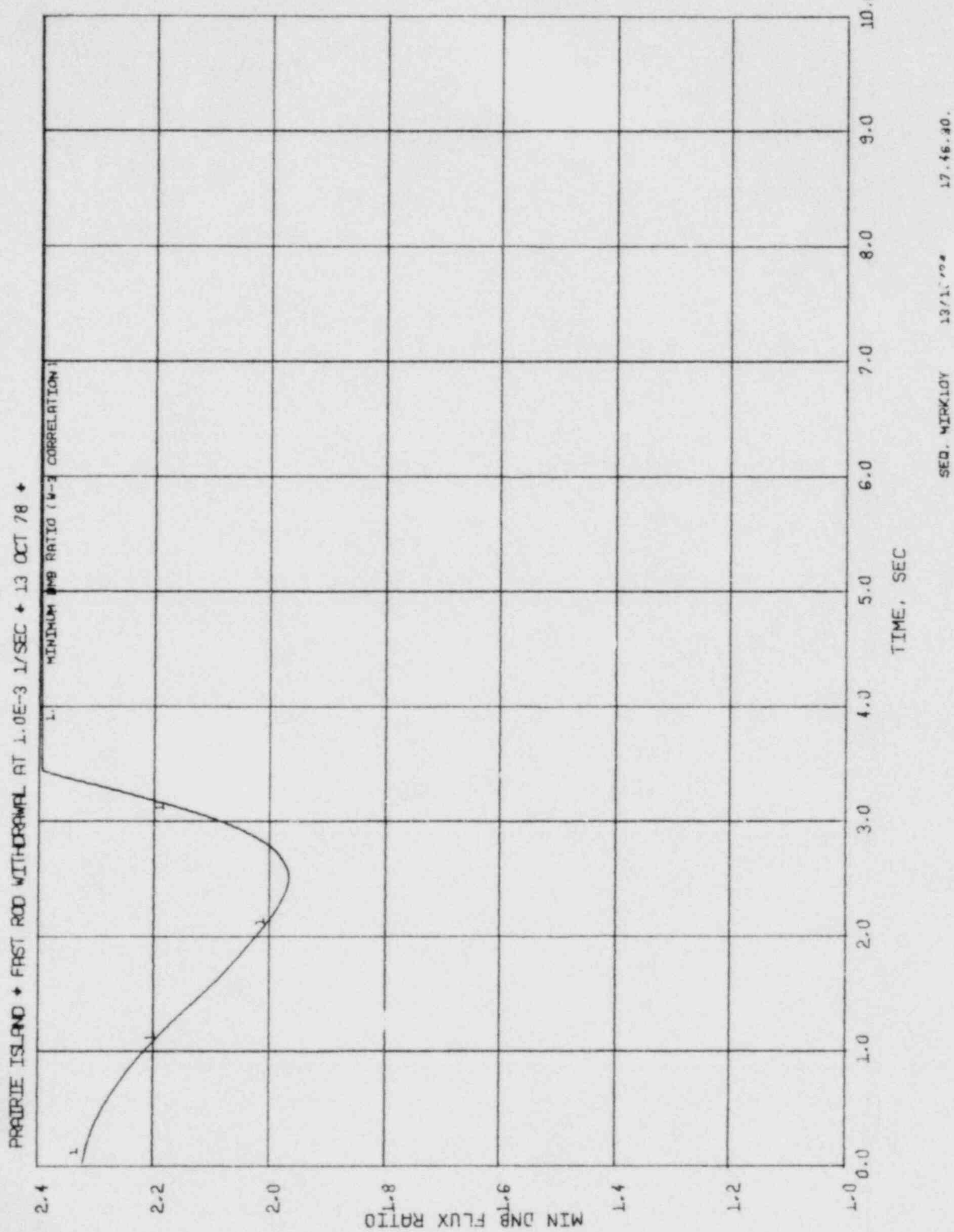

FIGURE 3.6 - Minimum DNB Flux Ratio for Fast Control Rod Withdrawal

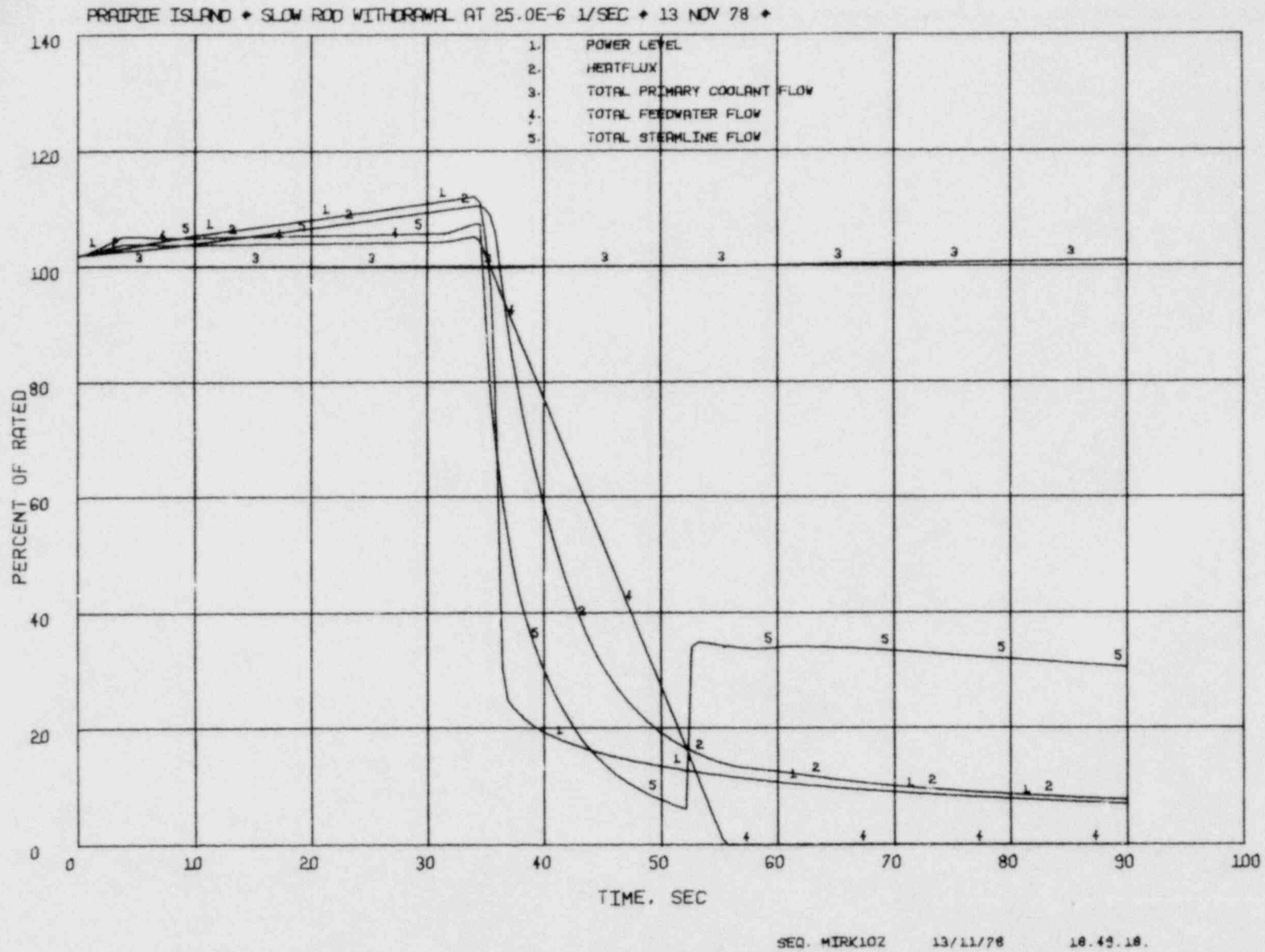

FIGURE 3.7 - Power, Heatflux and System Flows for Slow Control Rod Withdrawal

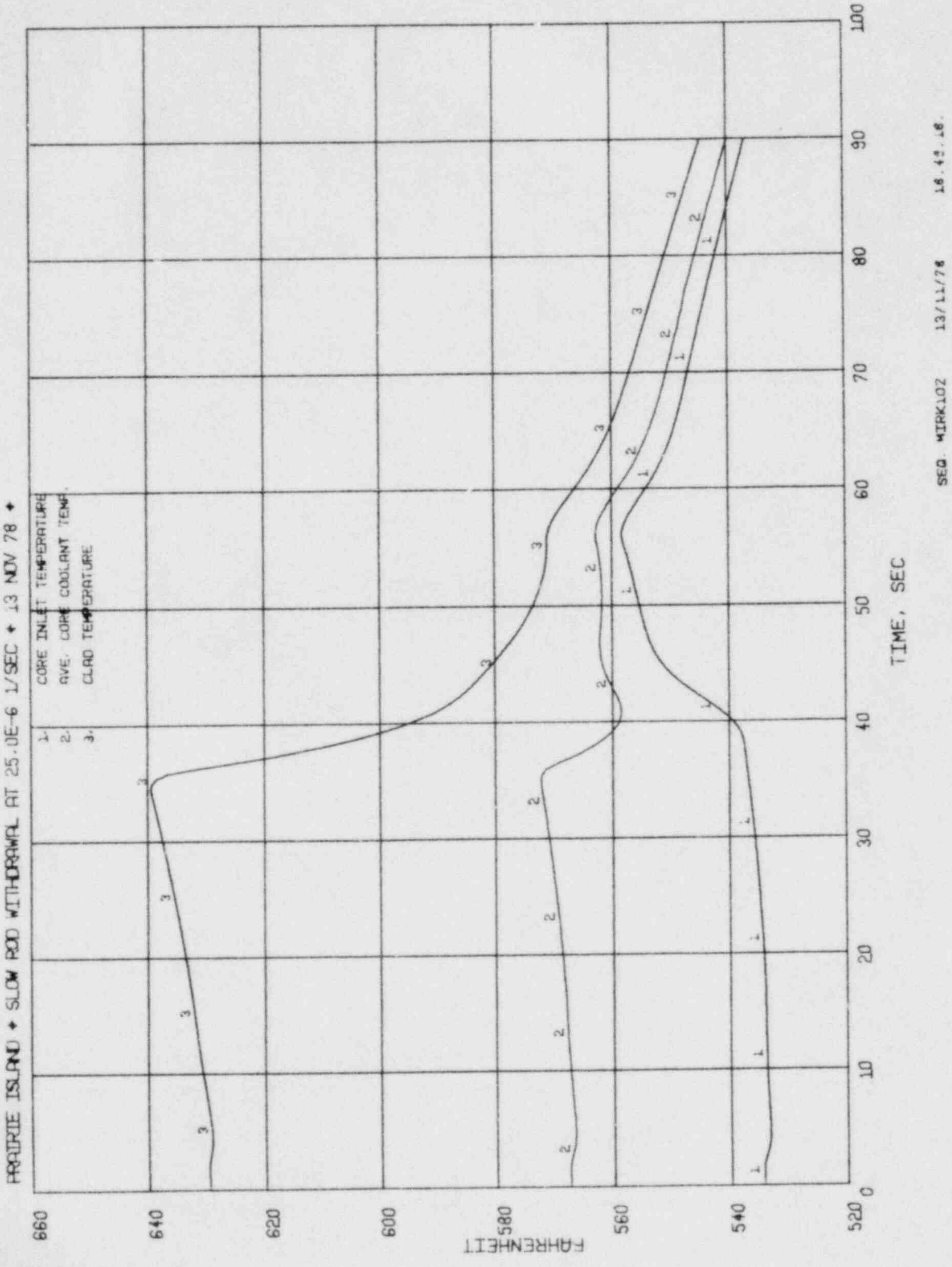

FIGURE 3.8 - Core Temperature Response for Slow Control Rod Withdrawal

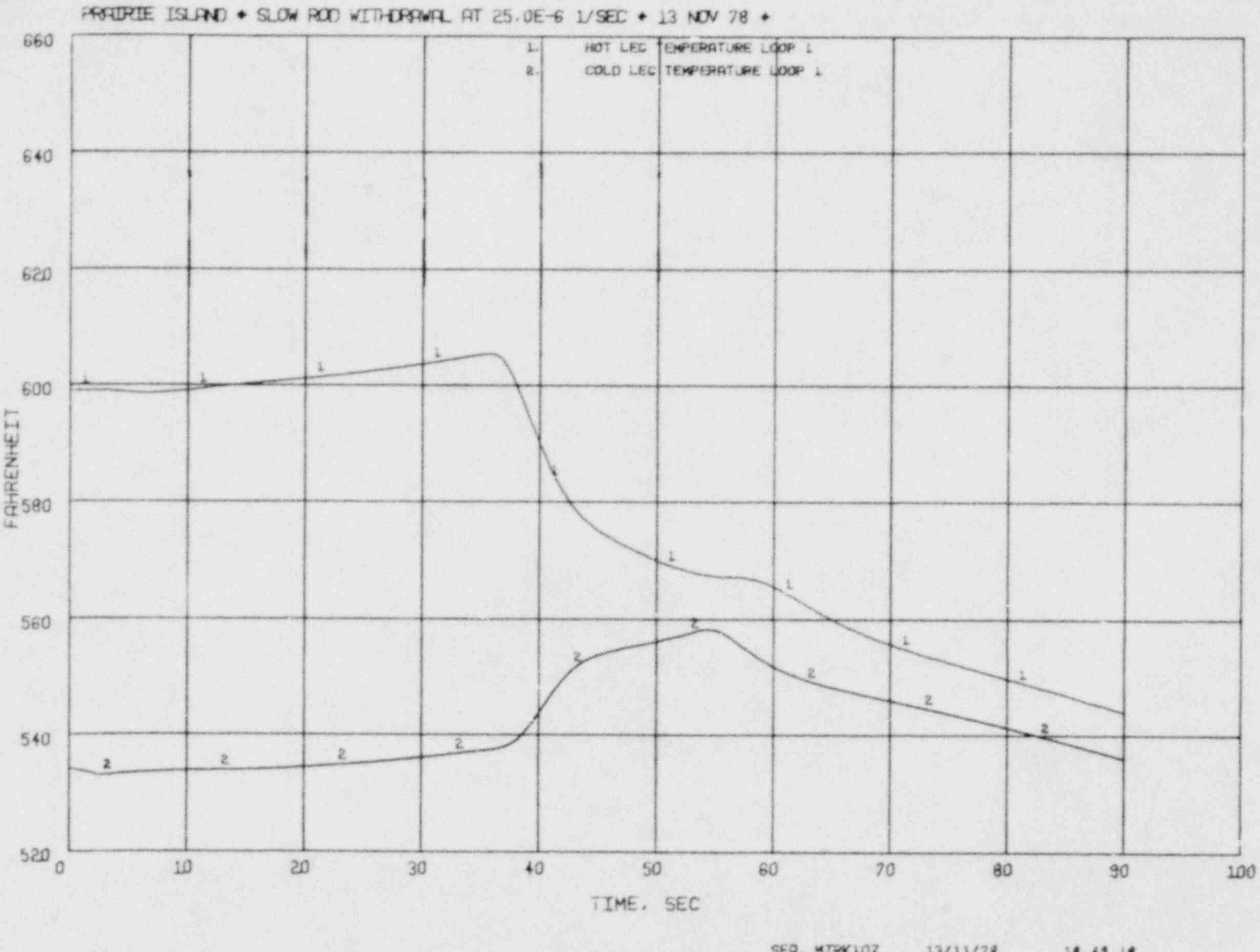

FIGURE 3.9 - Primary Loop Temperature Response for Slow Control Rod Withdrawal

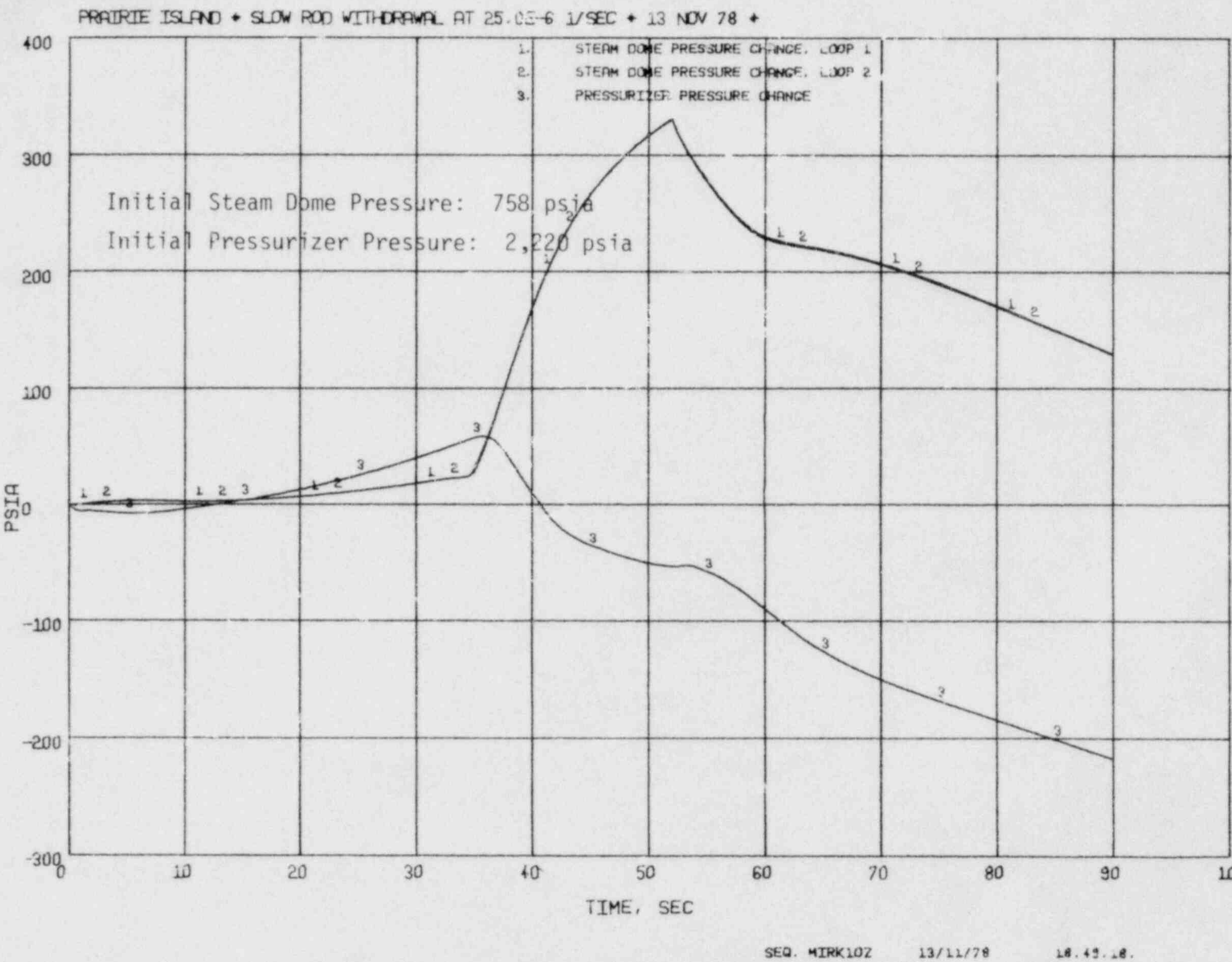

FIGURE 3.10 - Pressure Changes in Pressurizer and Steam Generators for Slow Control Rod Withdrawal

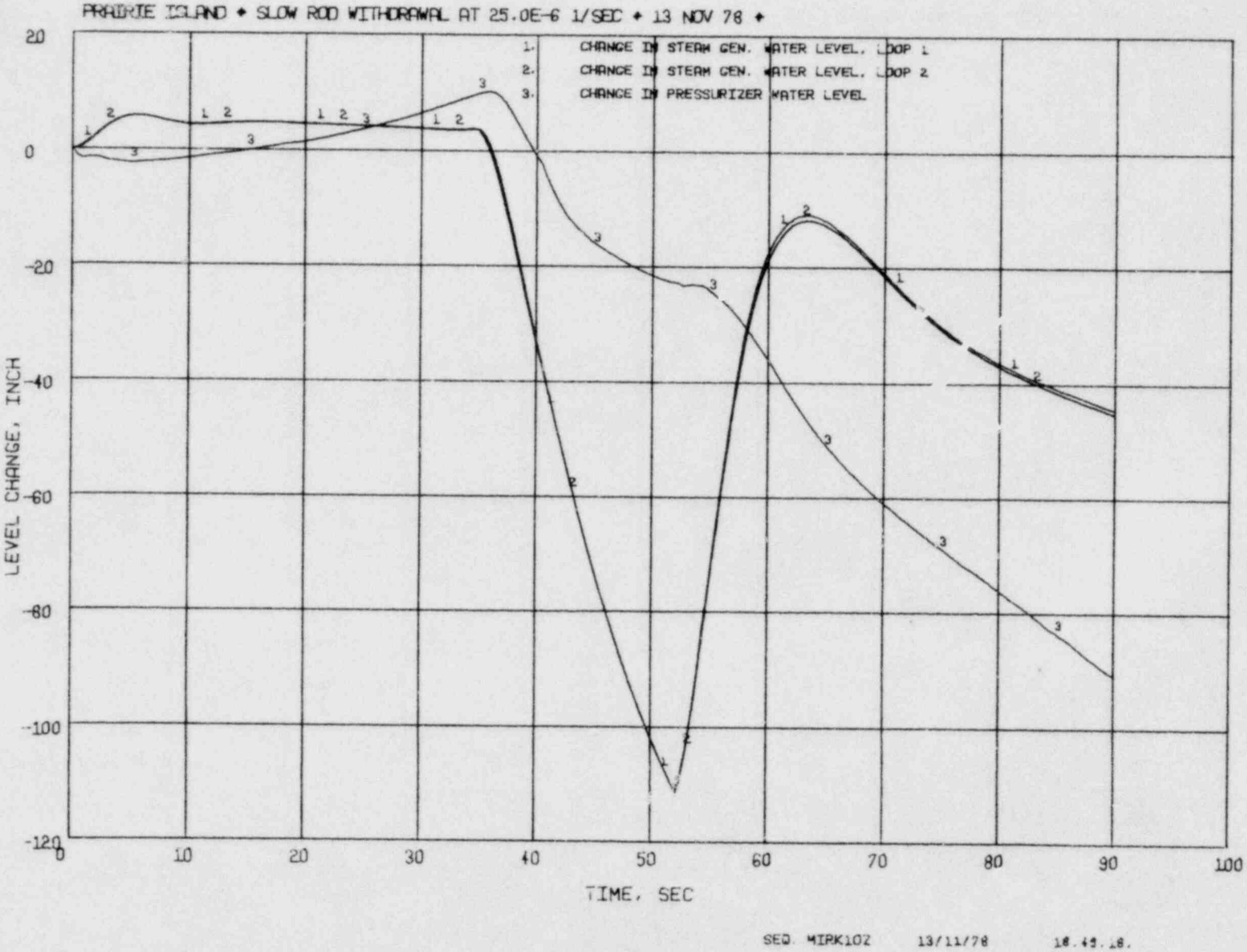

FIGURE 3.11 - Level Changes in Pressurizer and Steam Generators for Slow Control Rod Withdrawal

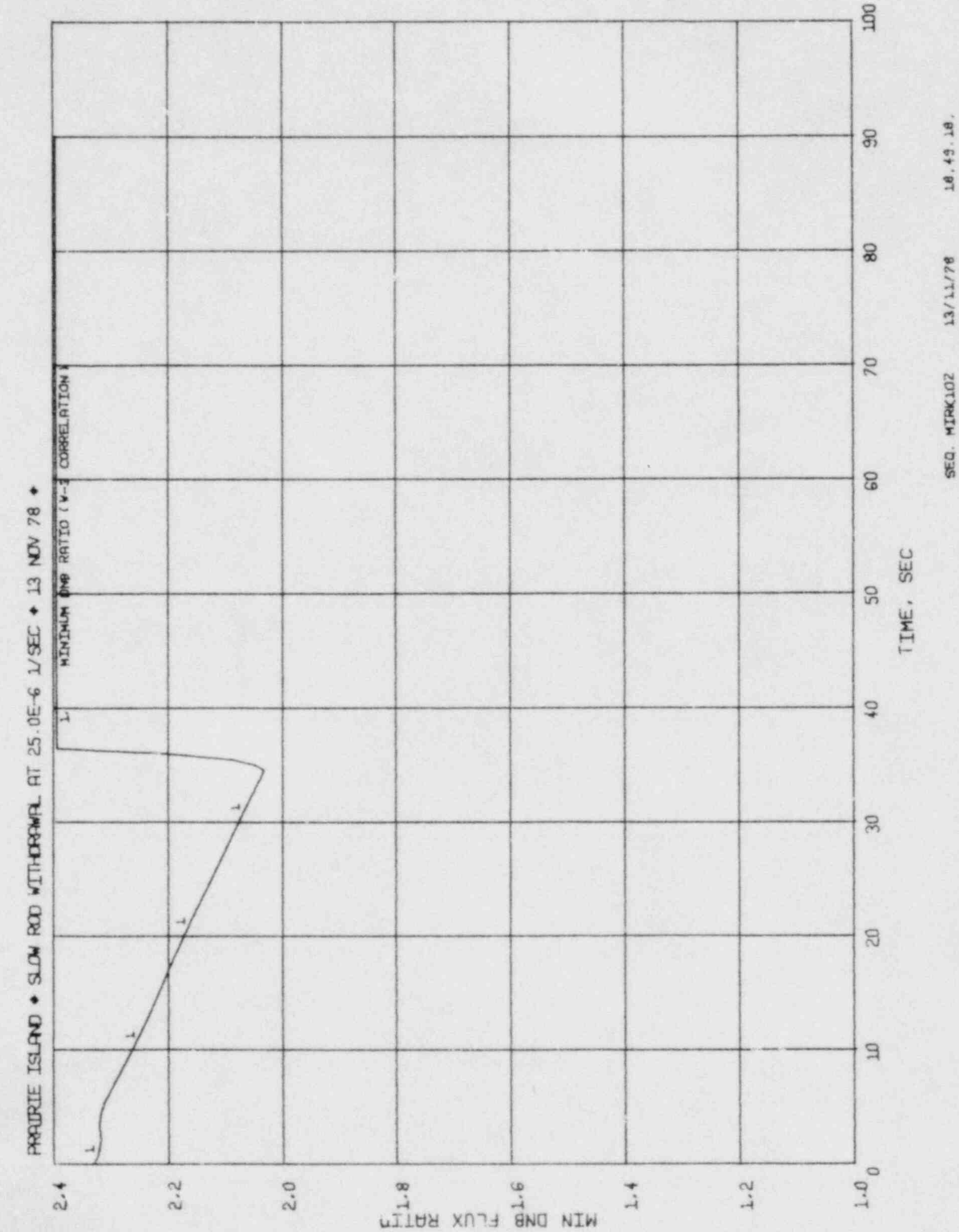

FIGURE 3.12 - Minimum DNB Flux Ratio for Slow Control Rod Withdrawal

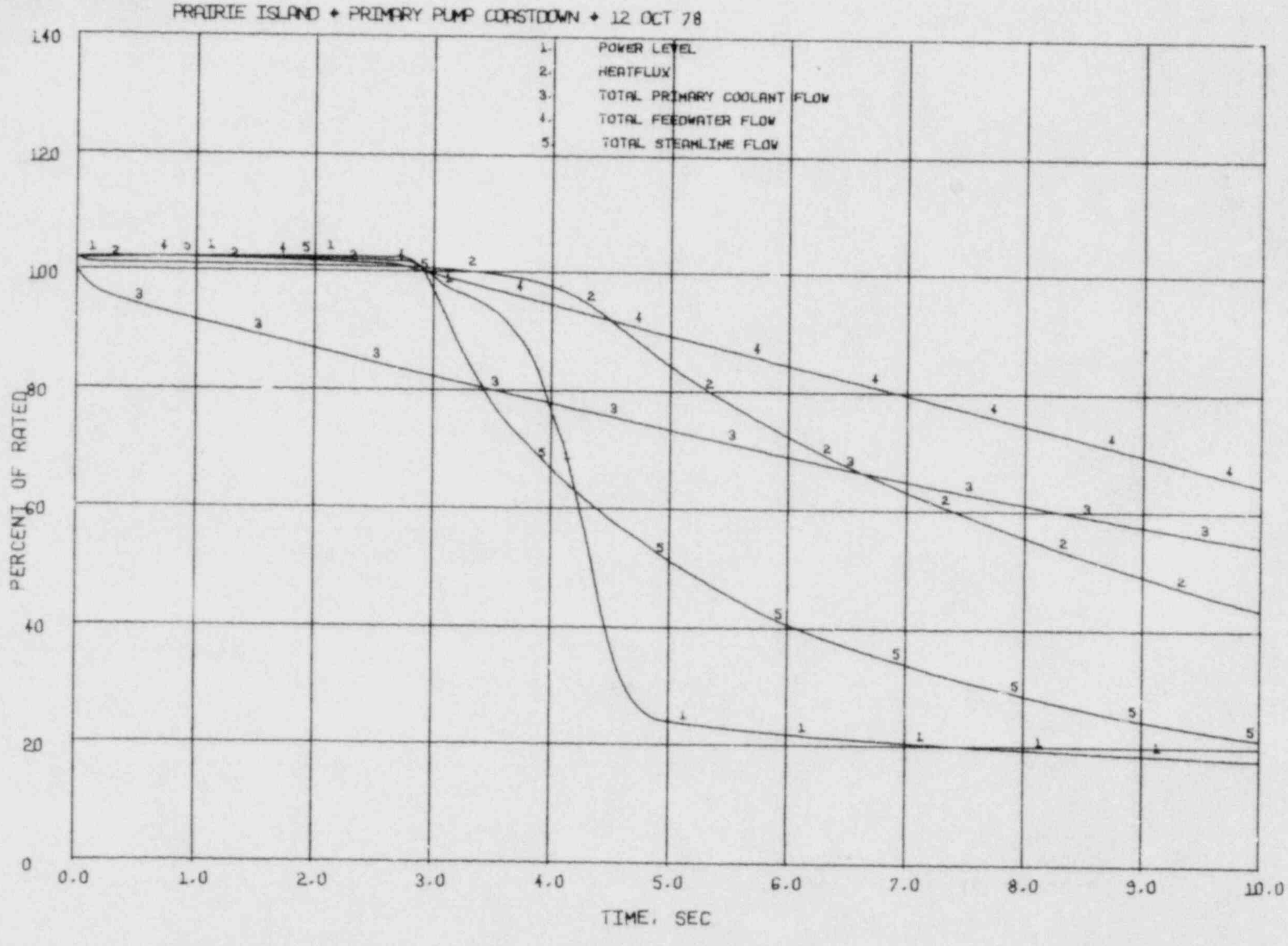

FIGURE 3.13 - Power, Heatflux and System Flows for Coolant Pump Trip

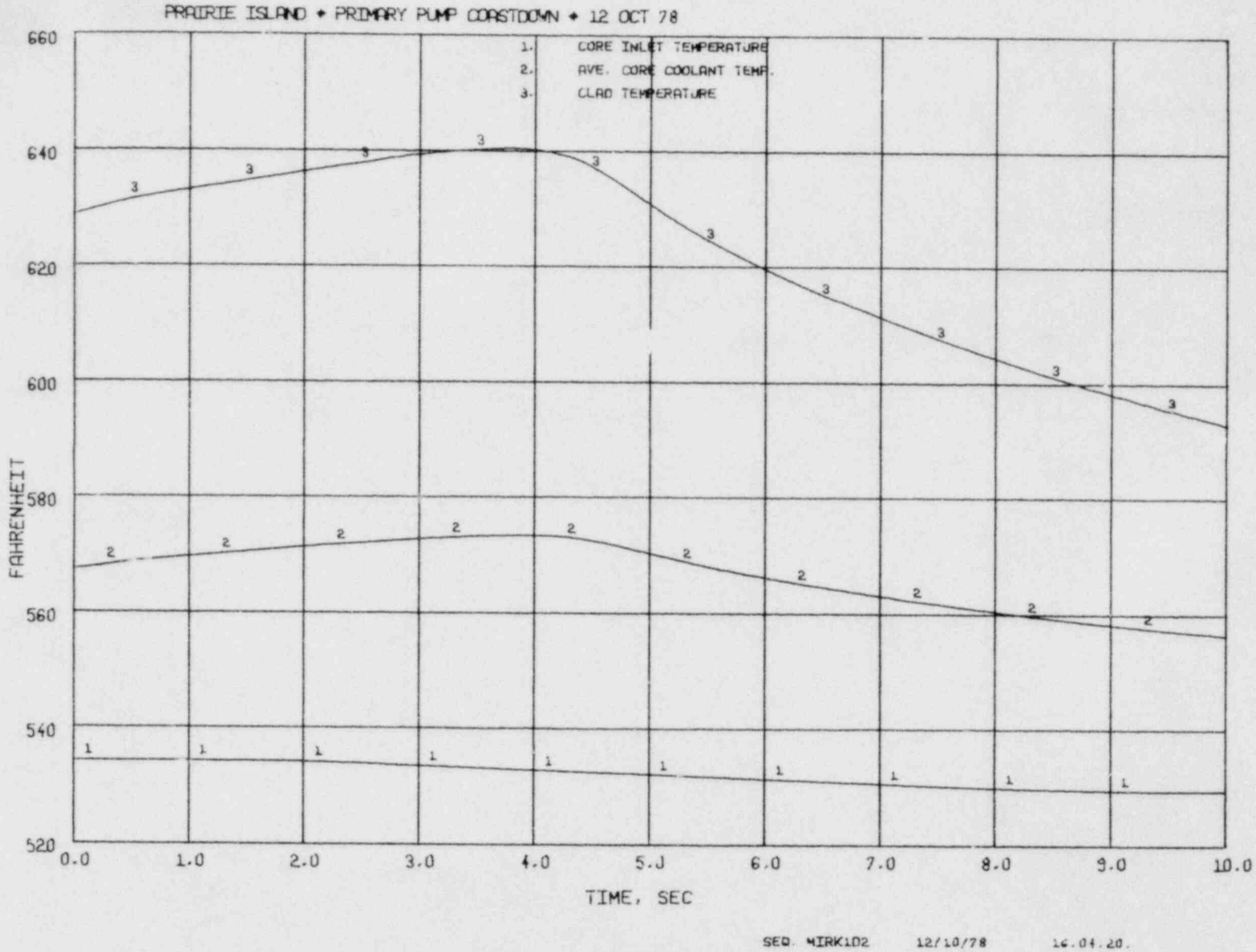

FIGURE 3.14 - Core Temperature Response for Coolant Pump Trip

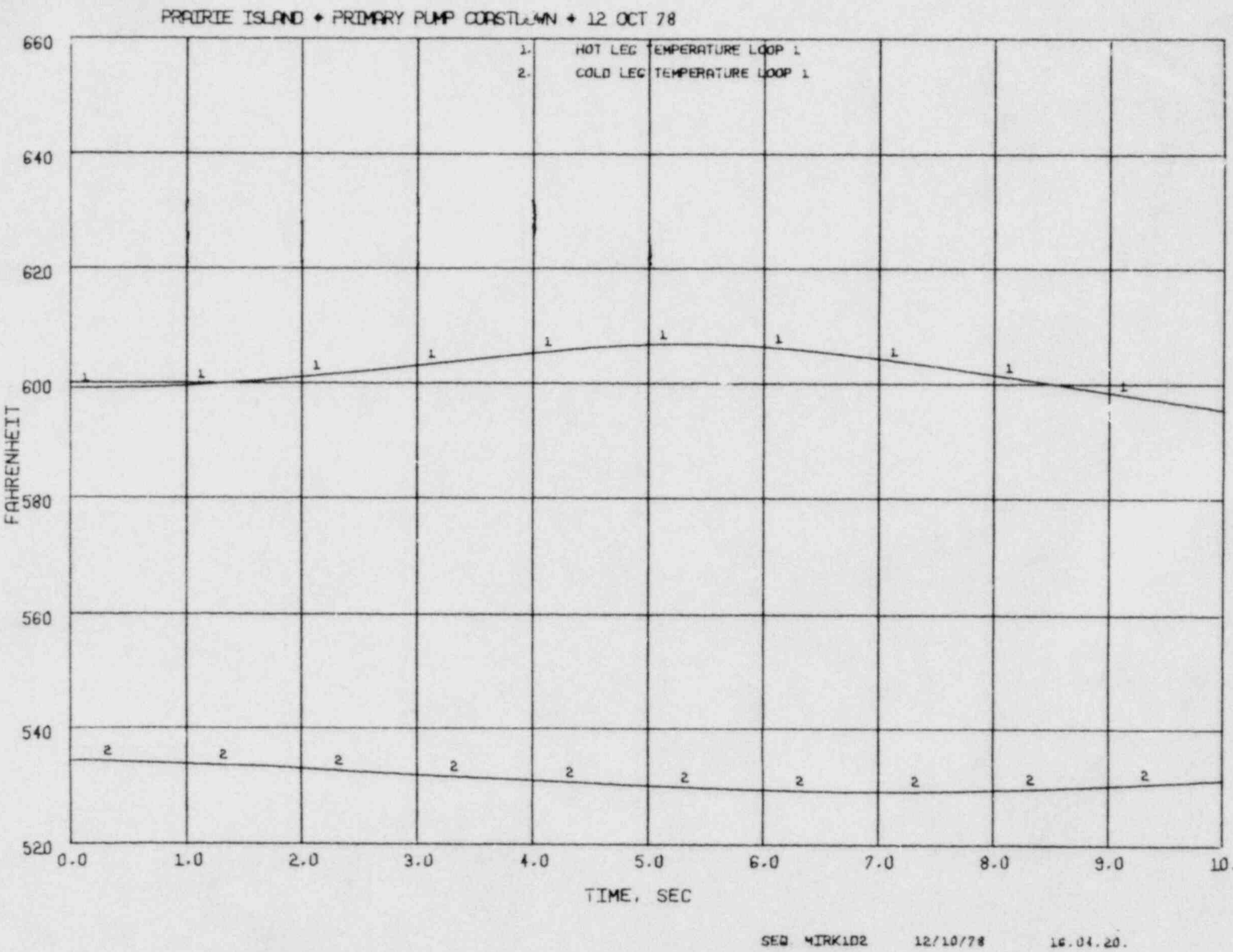

FIGURE 3.15 - Primary Loop Temperature Response for Coolant Pump Trip

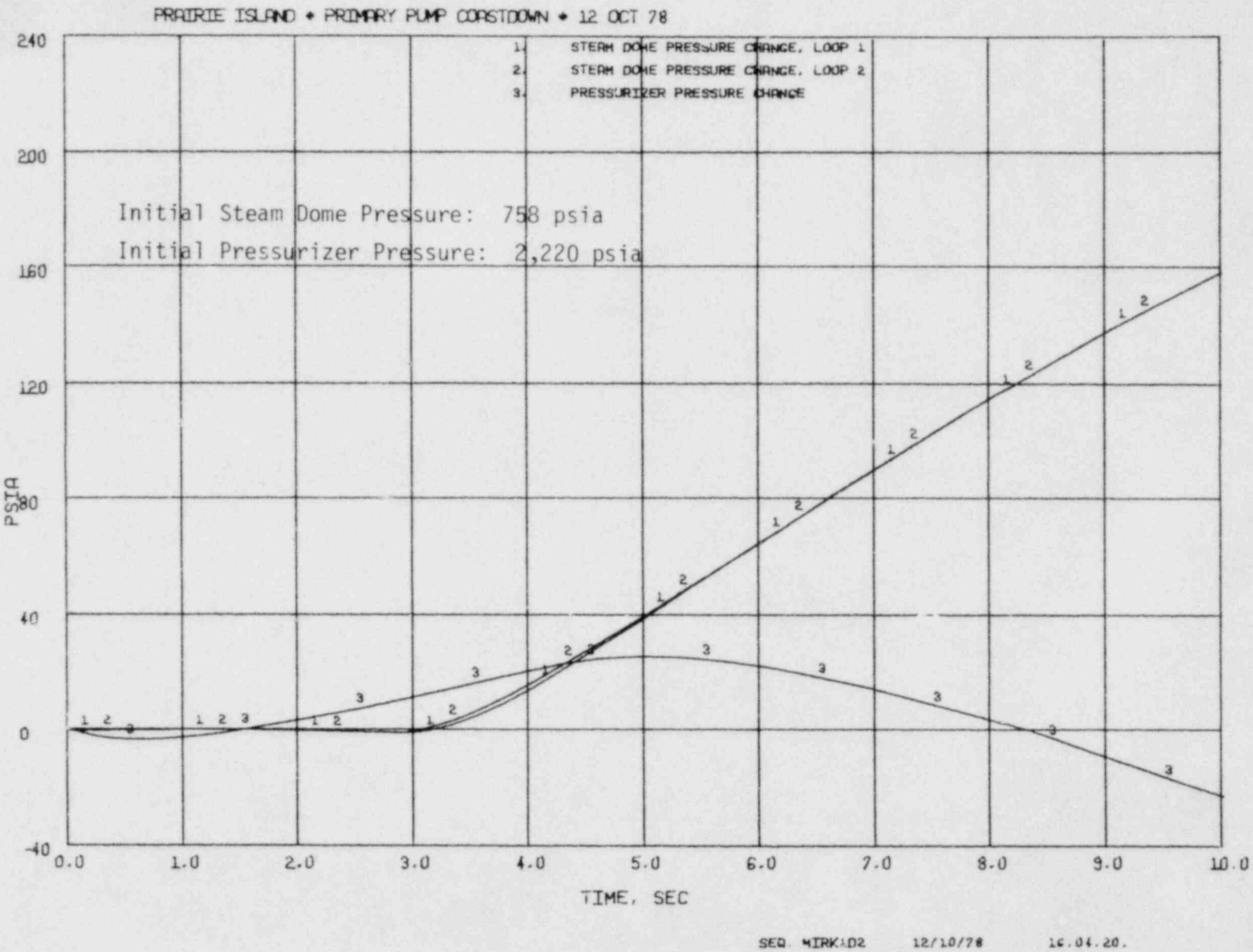

FIGURE 3.16 - Pressure Changes in Pressurizer and Steam Generators for Coolant Pump Trip

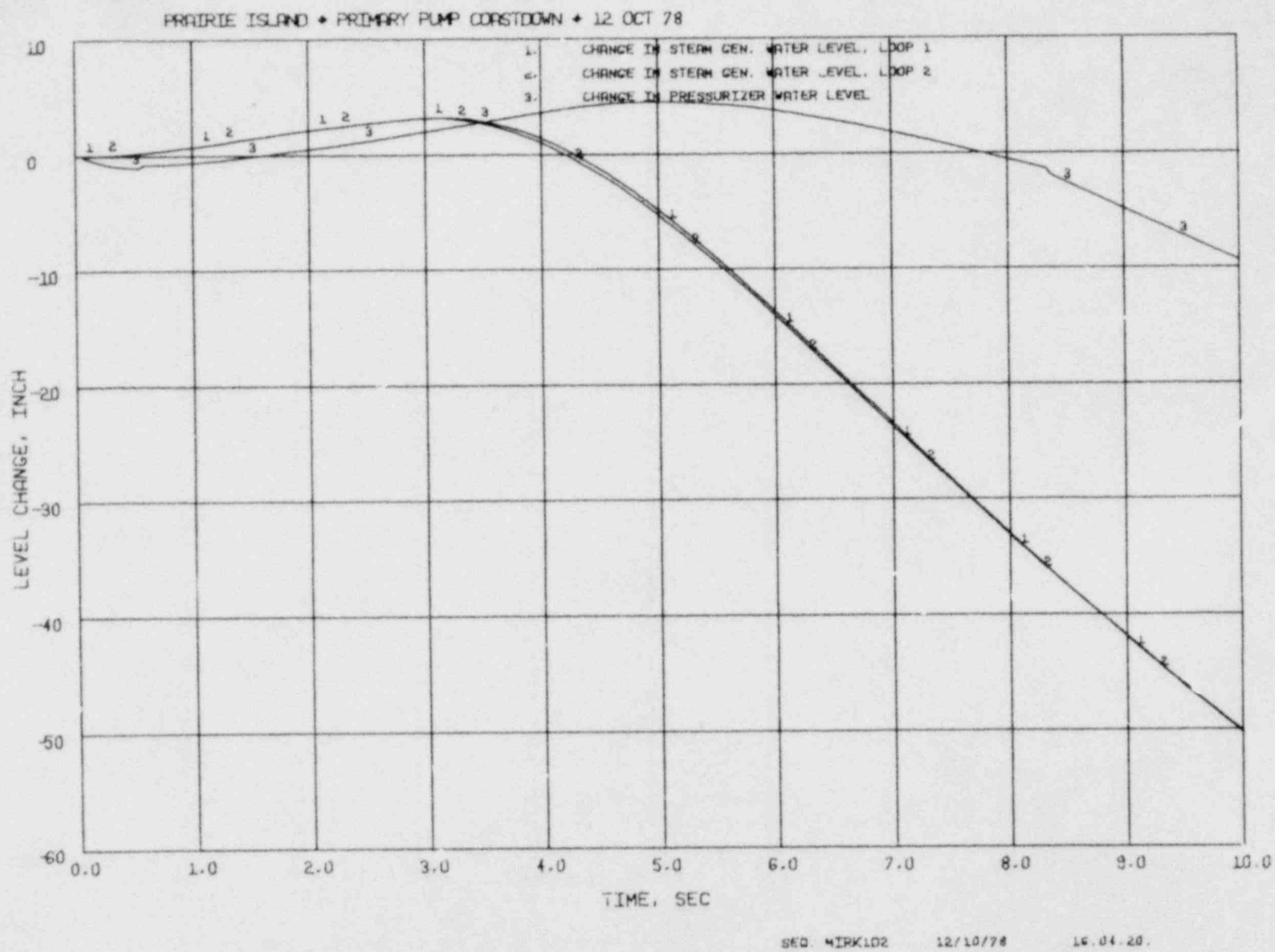

FIGURE 3.17 - Level Changes in Pressurizer and Steam Generators for Coolant Pump Trip

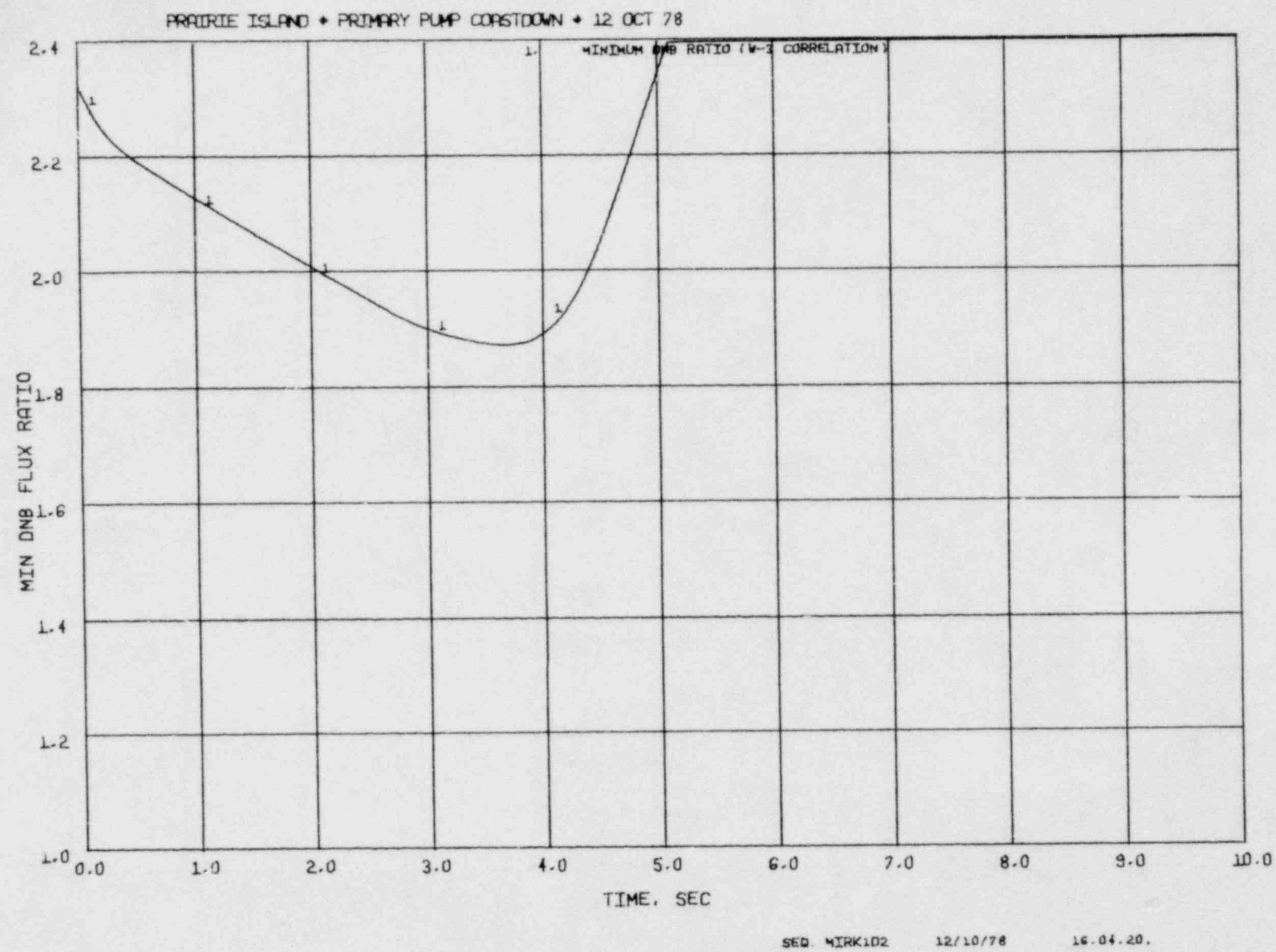

FIGURE 3.18 - Minimum DNB Flux Ratio for Coolant Pump Trip

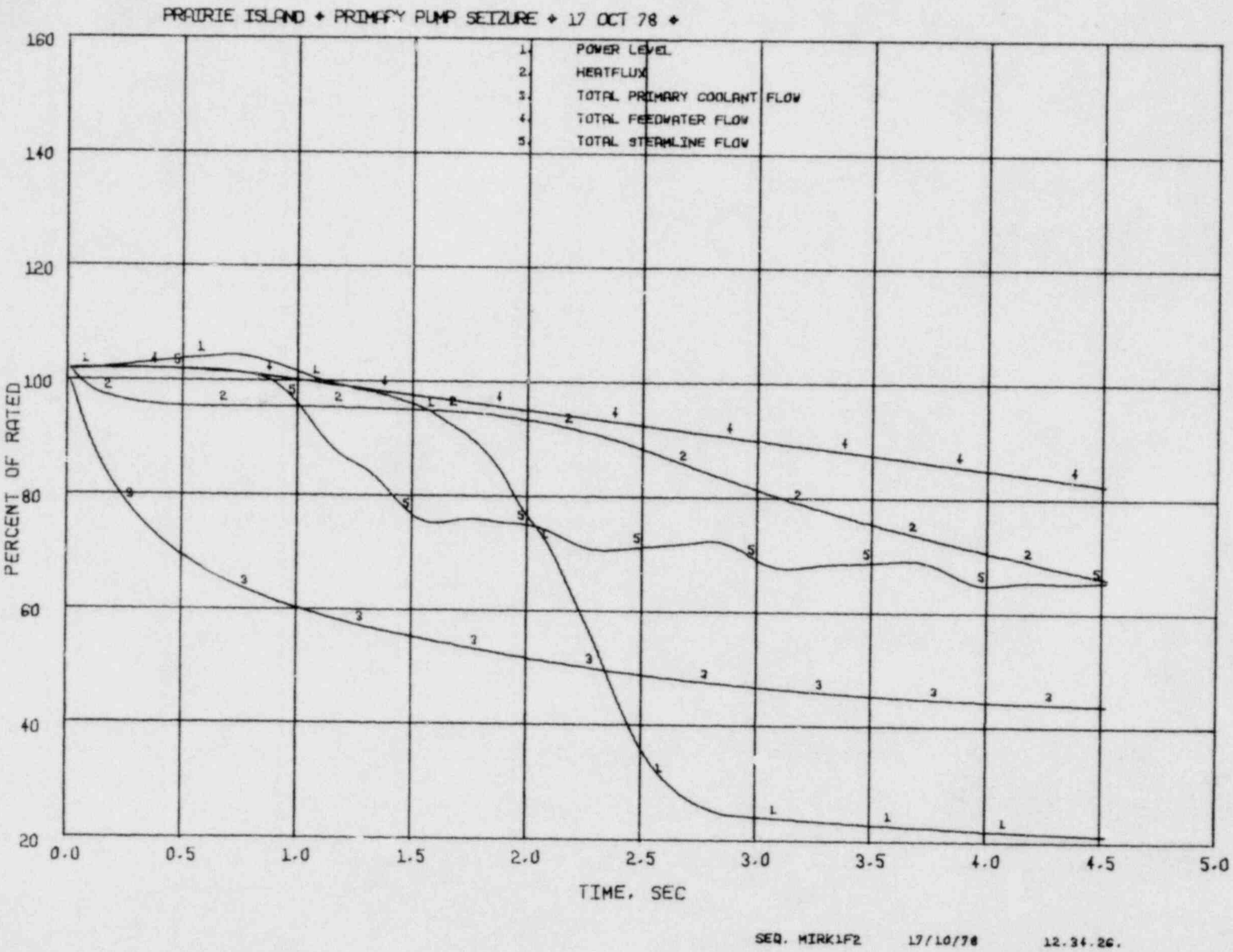

FIGURE 3.19 - Power, Heatflux and System Flows for Coolant Pump Seizure

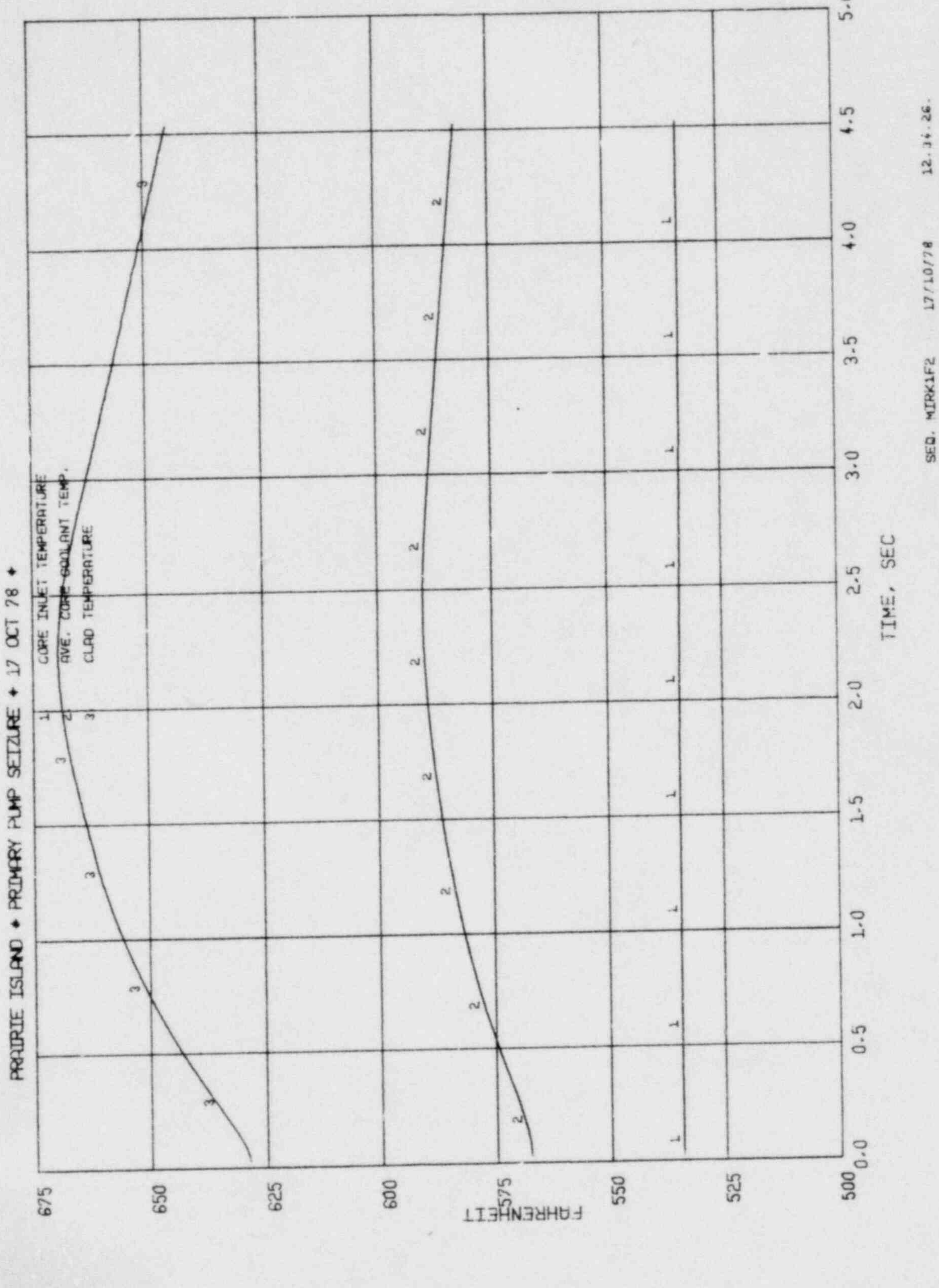

FIGURE 3.20 - Core Temperature Response for Coolant Pump Seizure

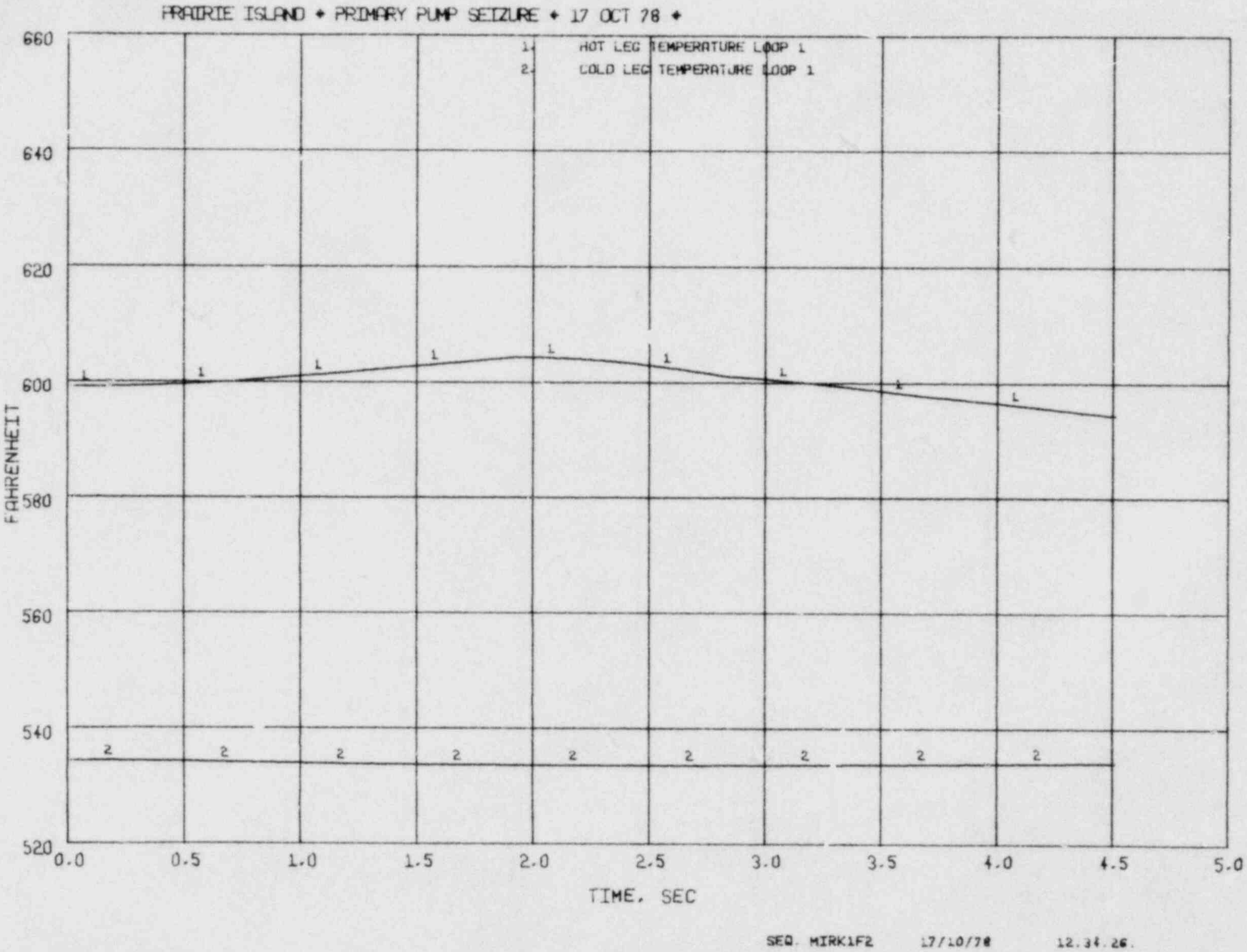

FIGURE 3.21 - Primary Loop Temperature Response for Coolant Pump Seizure

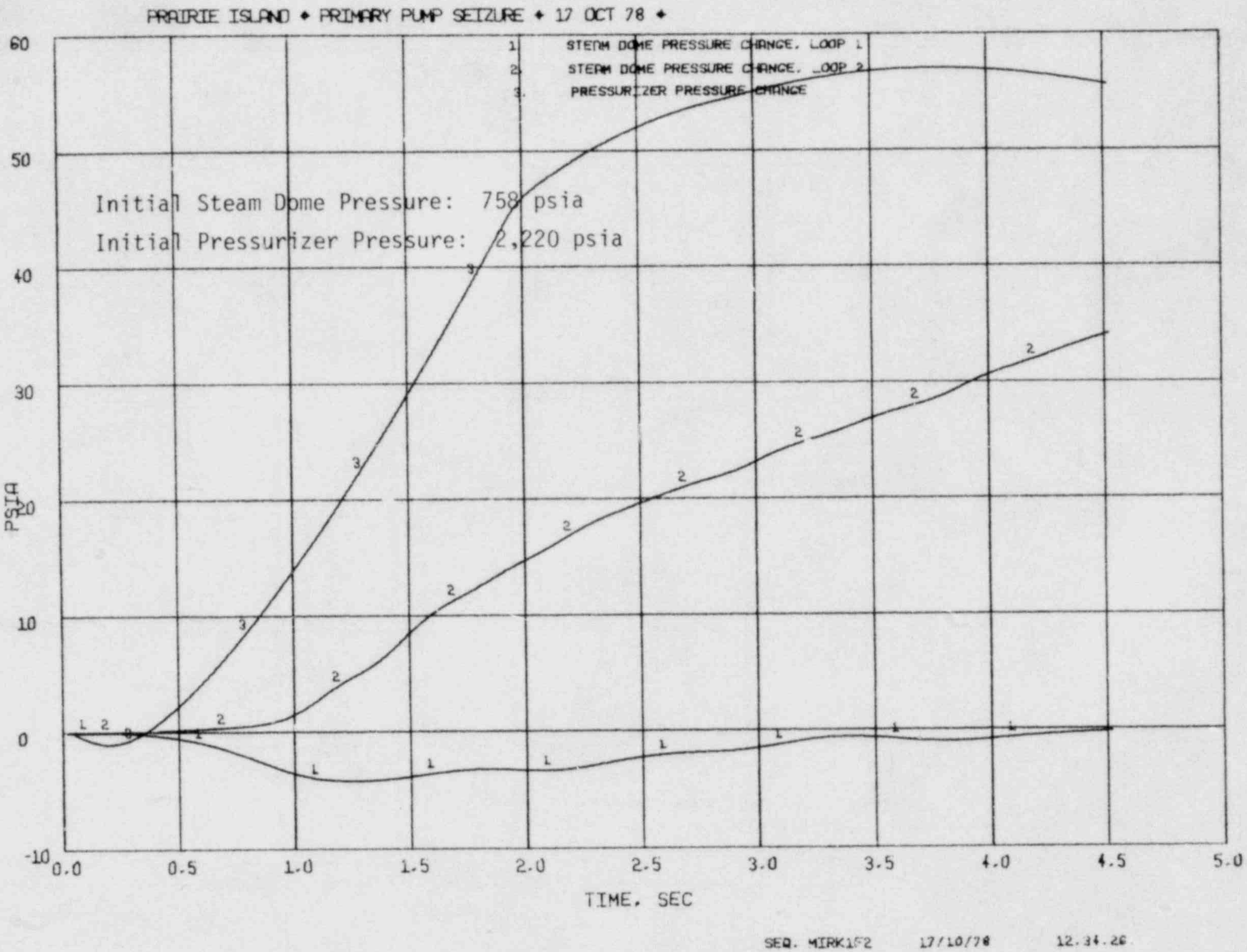

FIGURE 3.22 - Pressure Changes in Pressurizer and Steam Generators for Coolant Pump Seizure

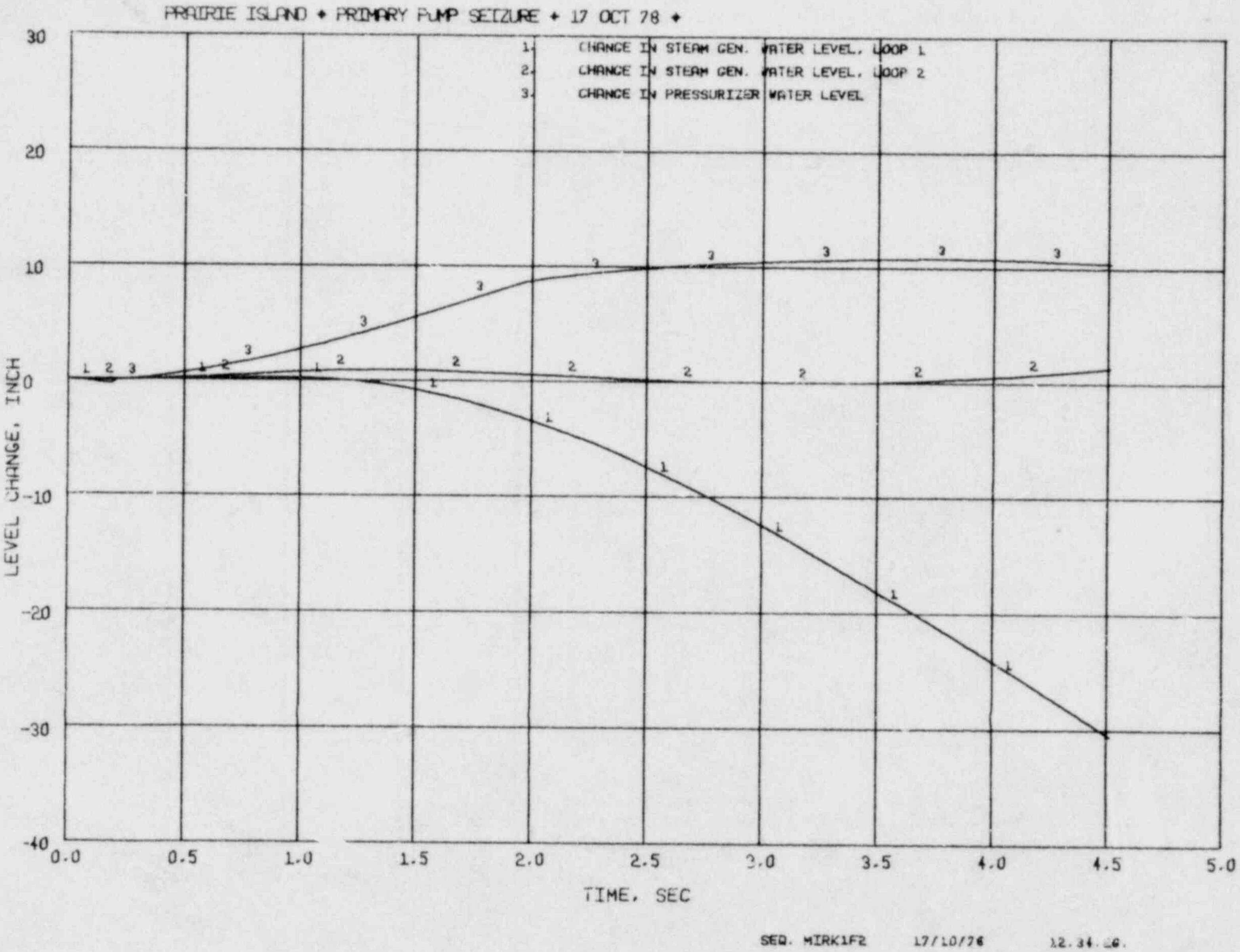

FIGURE 3.23 - Level Changes in Pressurizer and Steam Generators for Coolant Pump Seizure

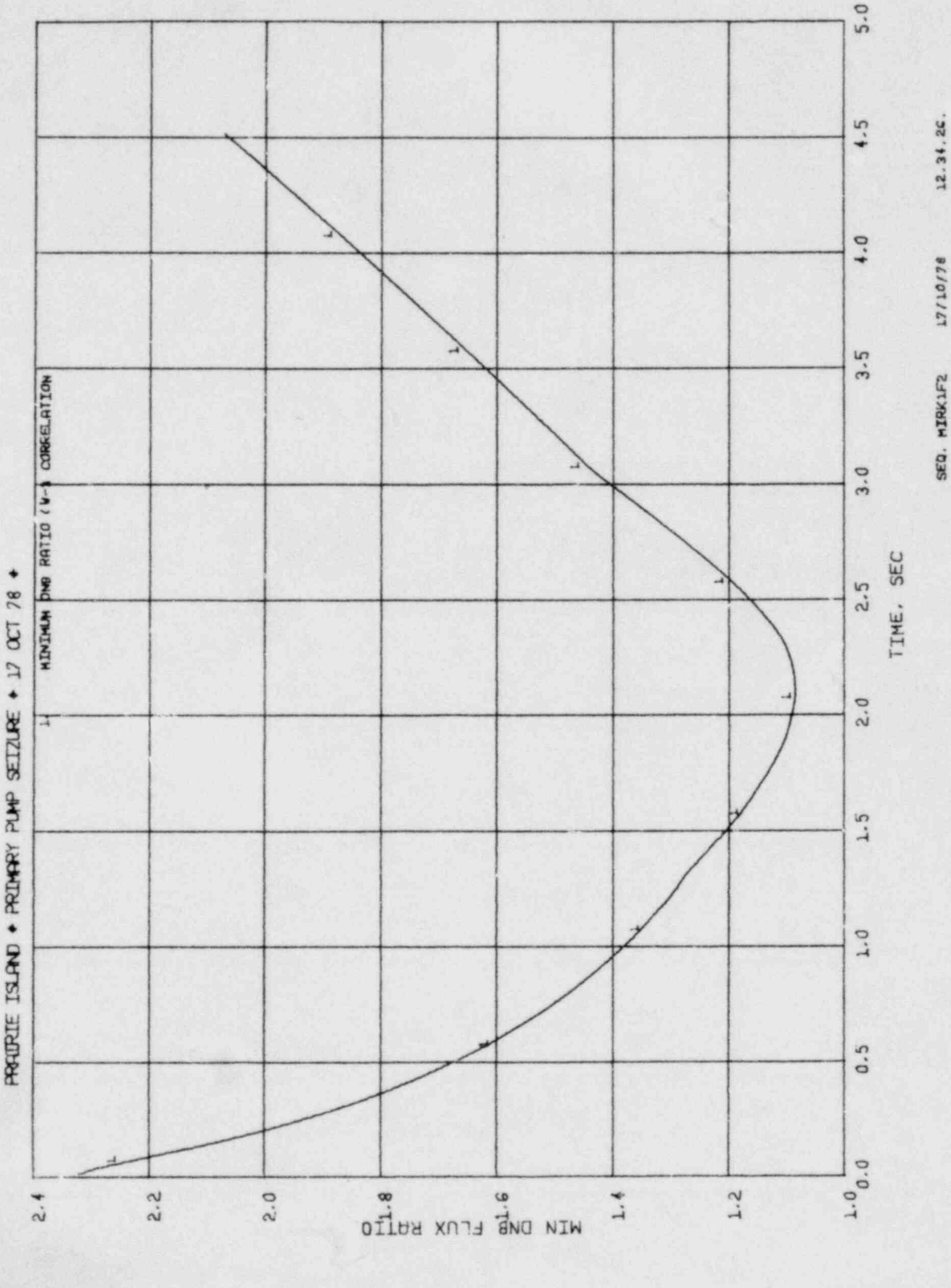

FIGURE 3.24 - Minimum DNB Flux Ratio for Coolant Pump Seizure

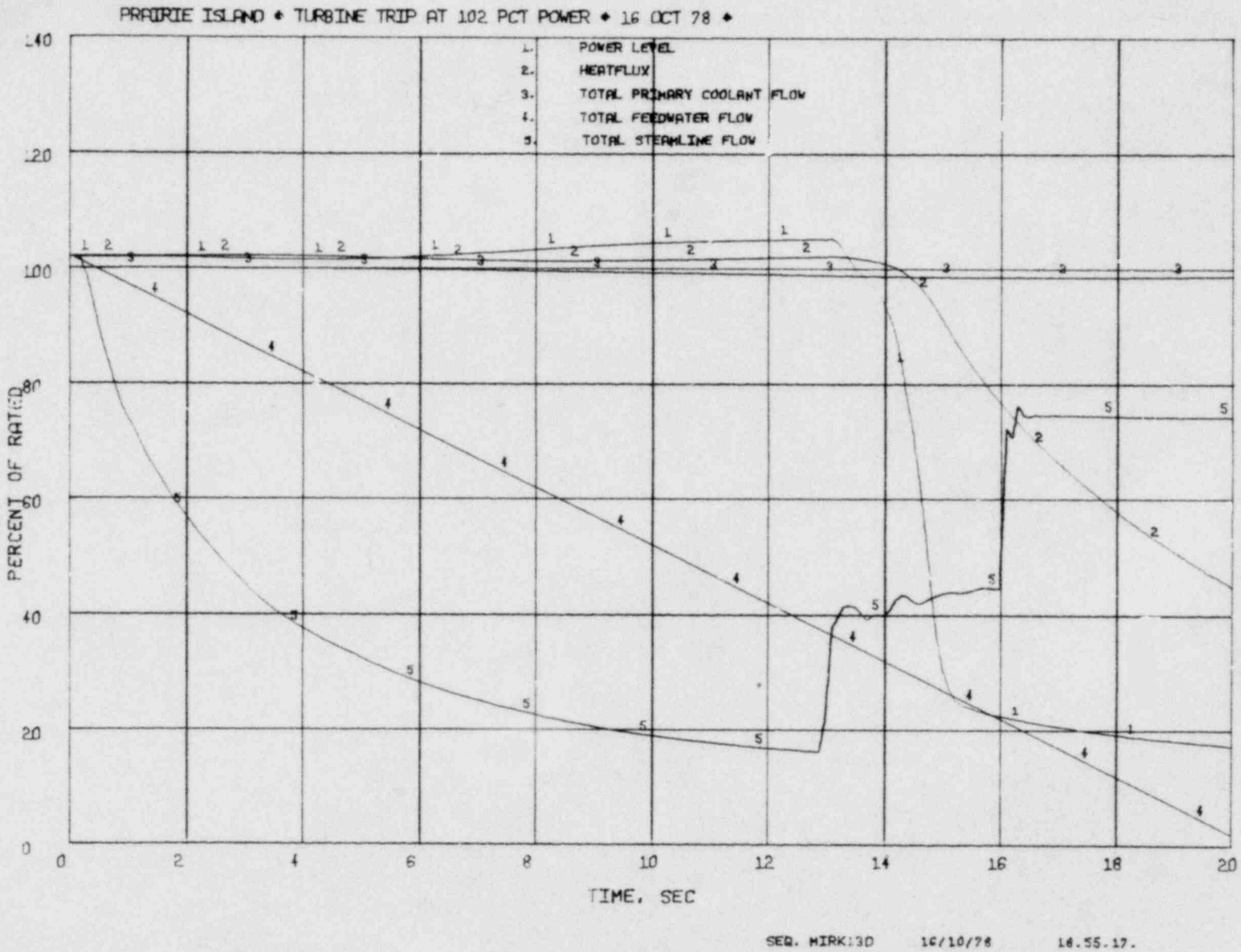

FIGURE 3.25 - Power, Heatflux and System Flows for Turbine Trip

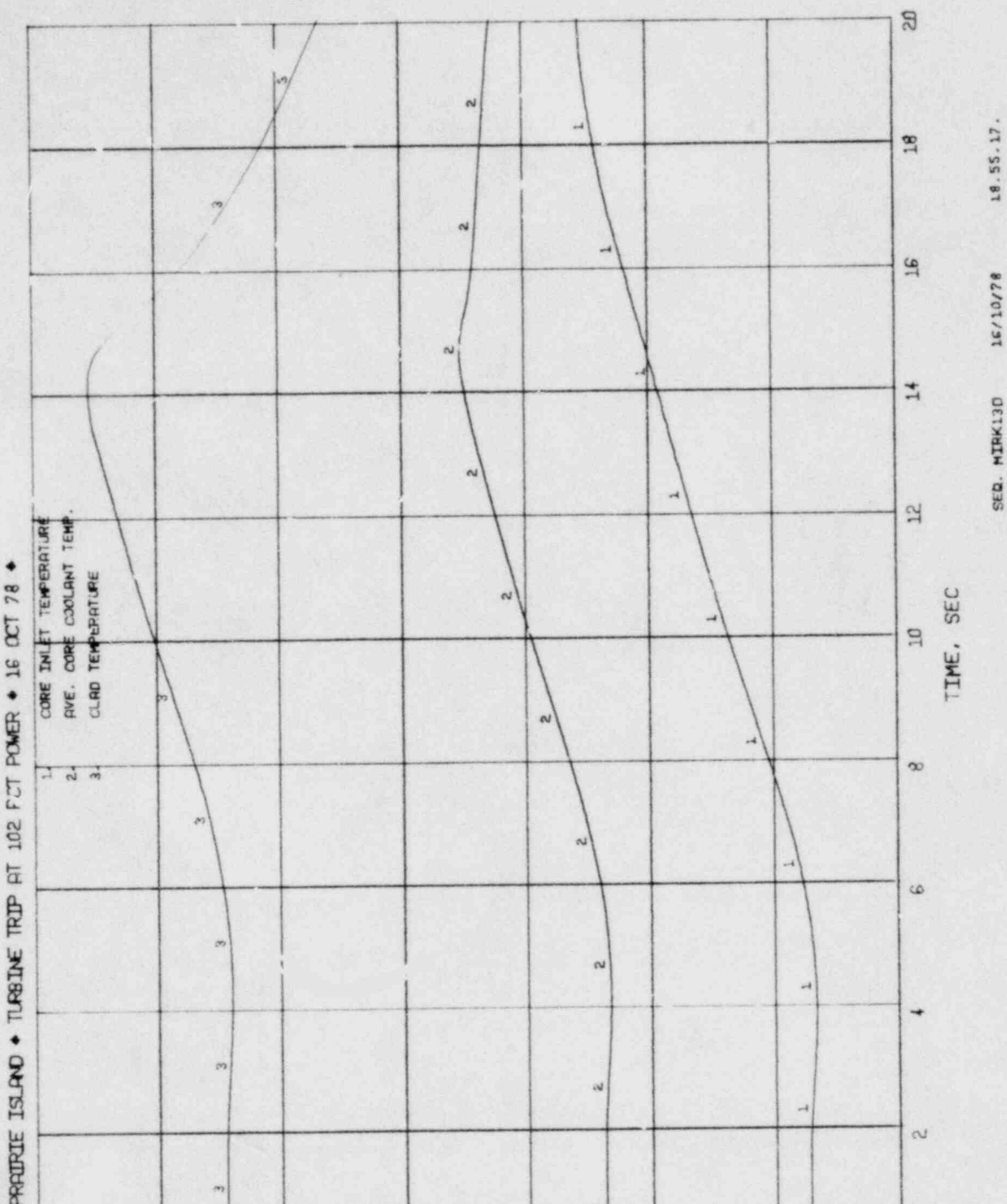

FIGURE 3.26 - Core Temperature Response for Turbine Trip

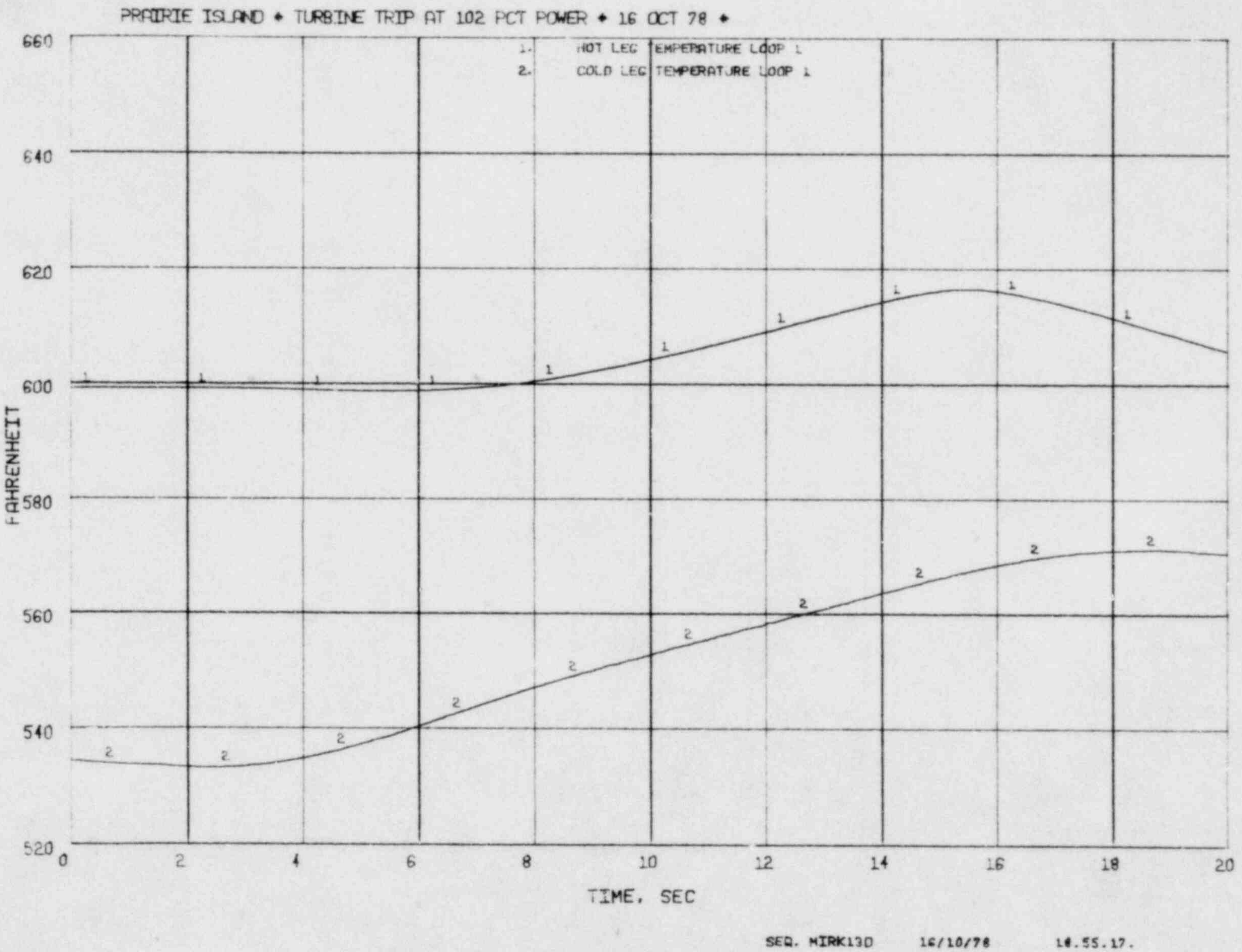

FIGURE 3.27 - Primary Loop Temperature Response for Turbine Trip

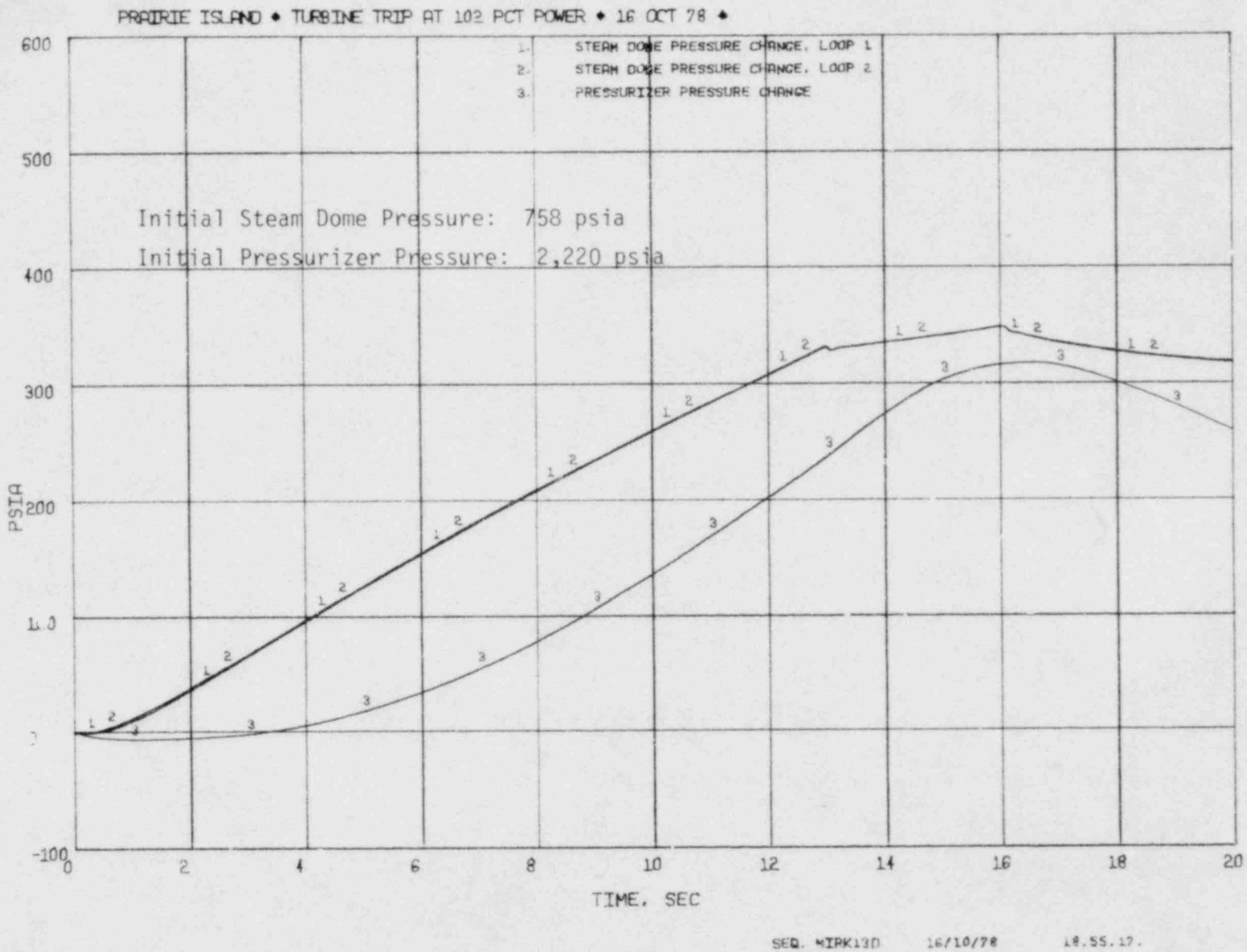

FIGURE 3.28 - Pressure Changes in Pressurizer and Steam Generators for Turbine Trip

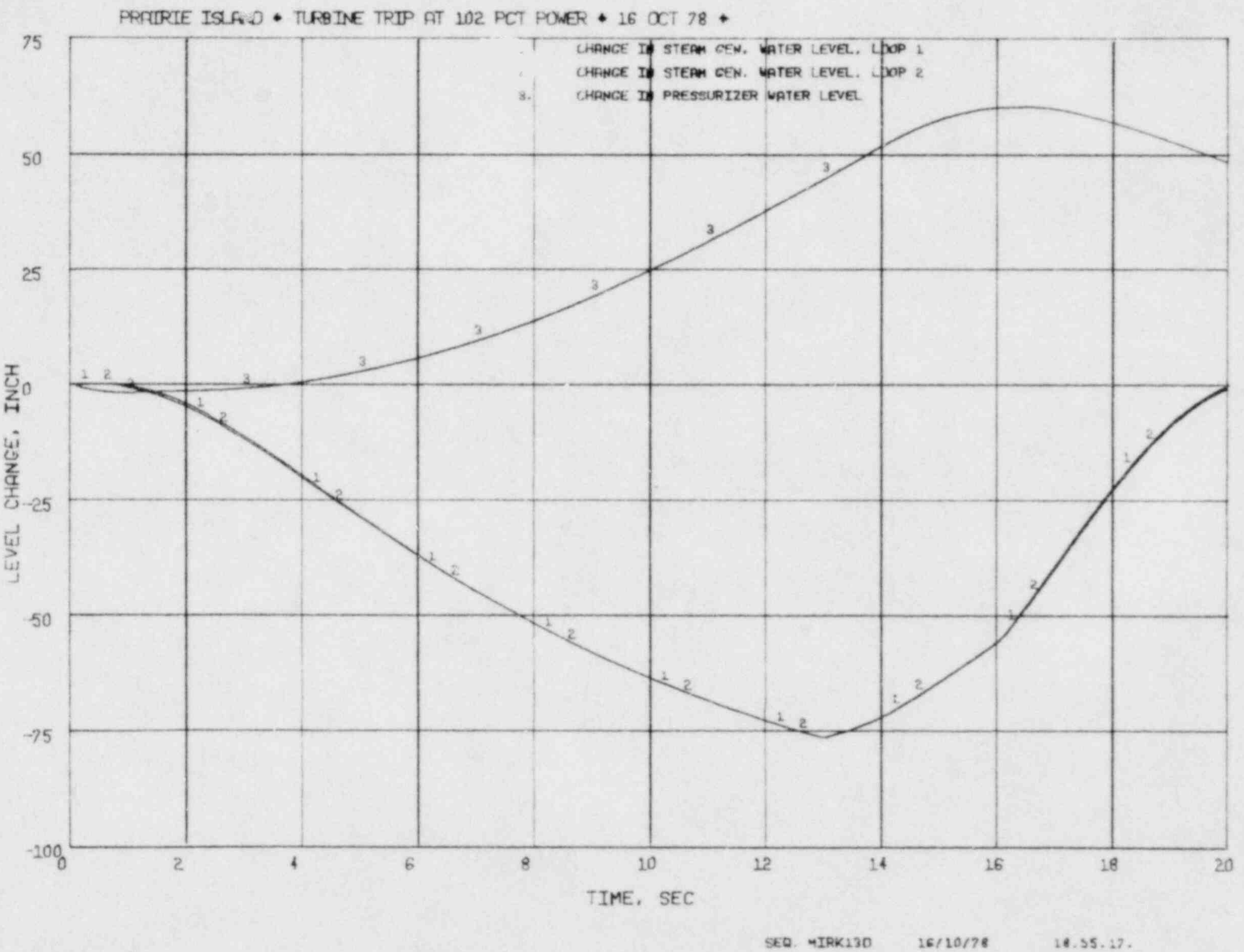

FIGURE 3.29 - Level Changes in Pressurizer and Steam Generators for Turbine Trip

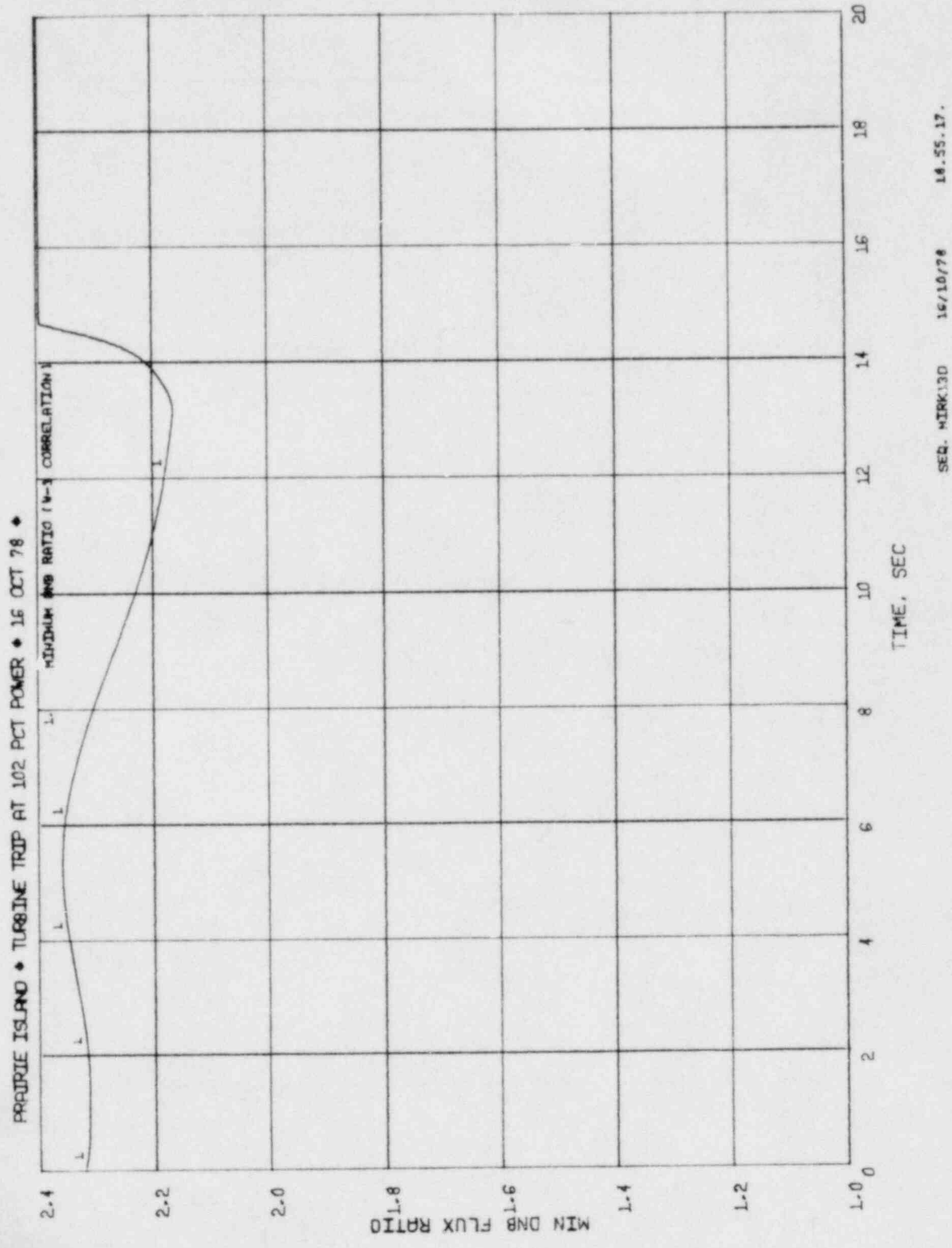

FIGURE 3.30 - Minimum DNB Flux Ratio for Turbine Trip

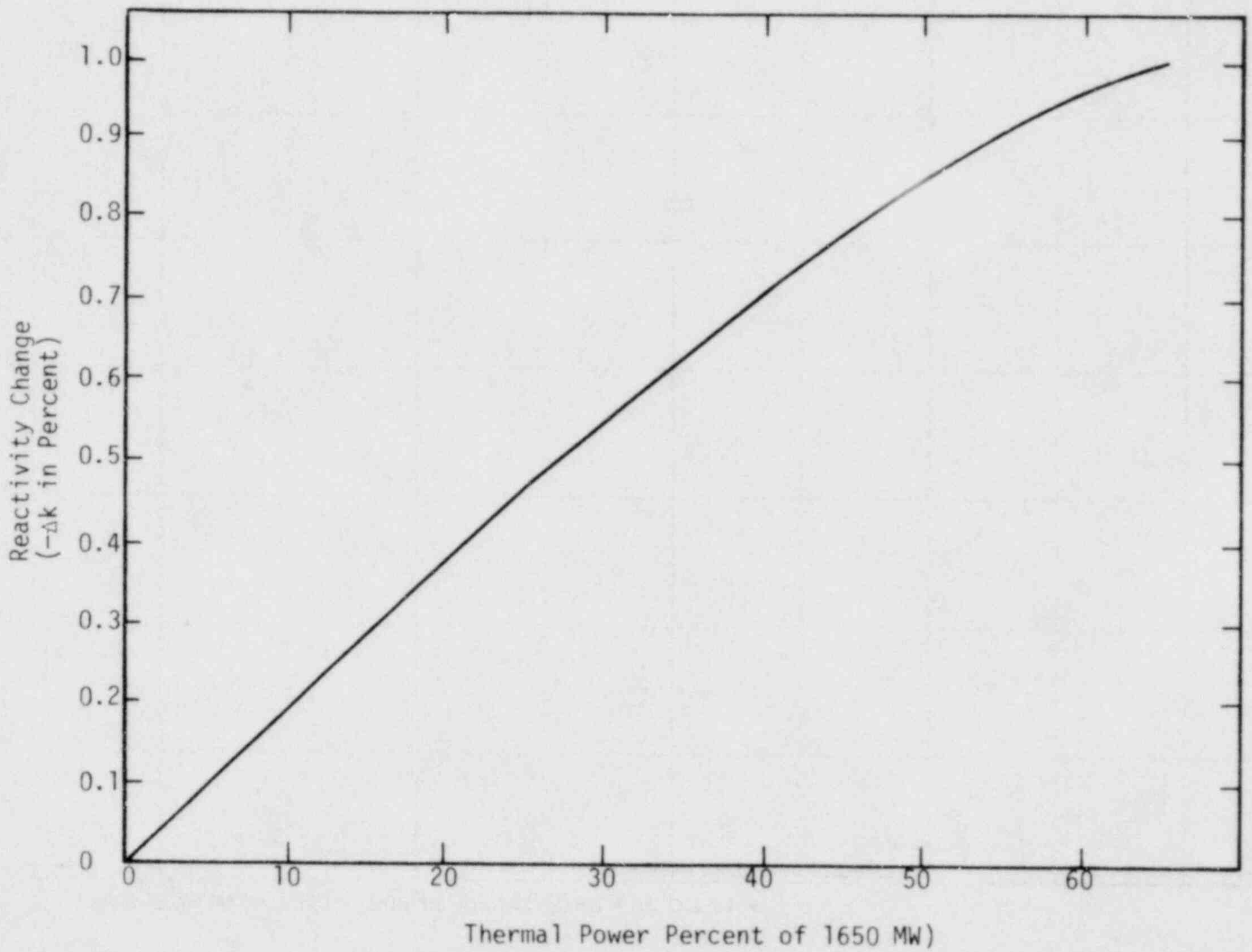

FIGURE 3.31 - Variation of Reactivity with Power at Constant Core Average Temperature

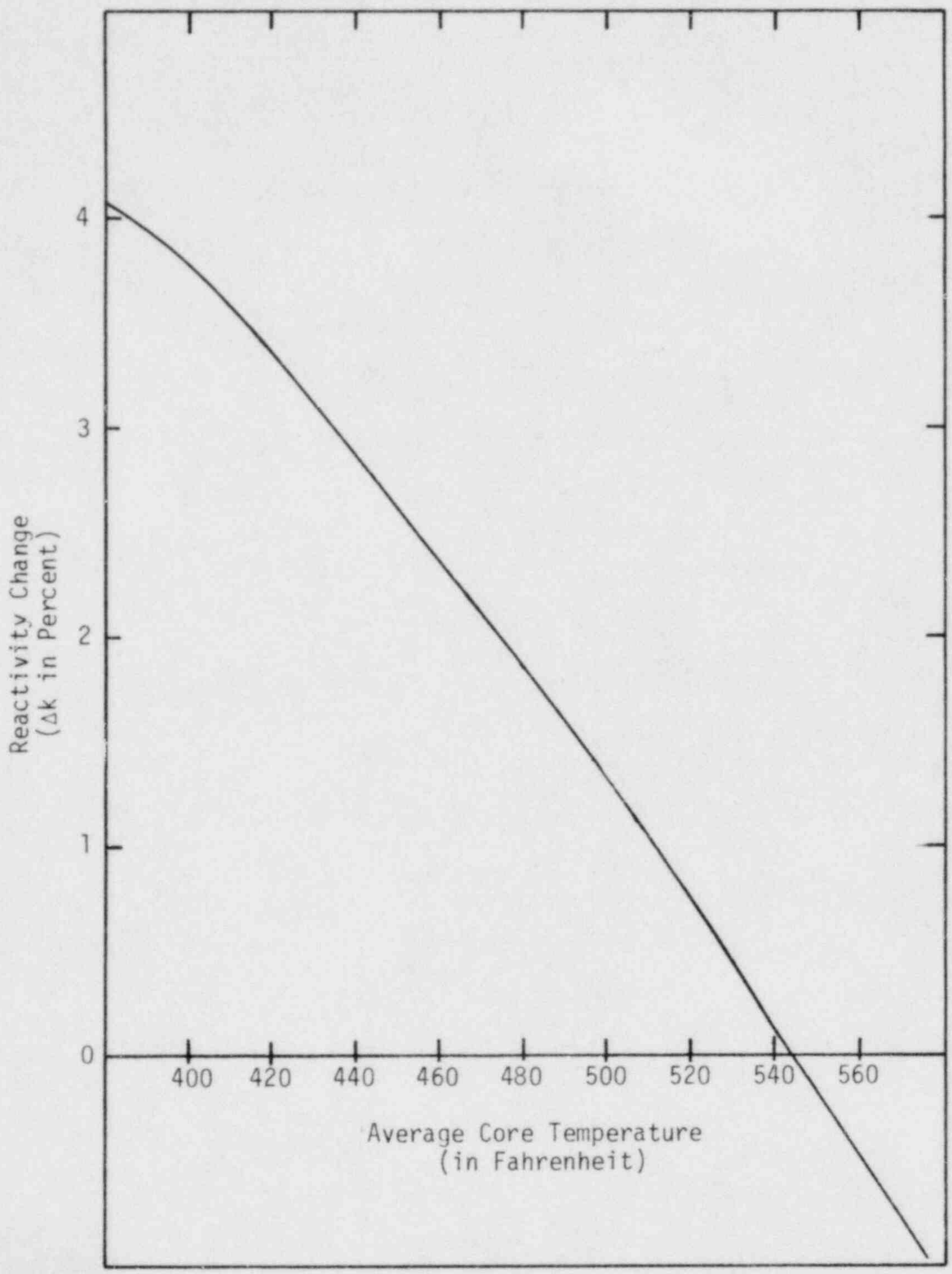

FIGURE 3.32 - Variation of Reactivity with Core Average Temperature at the End of the Cycle

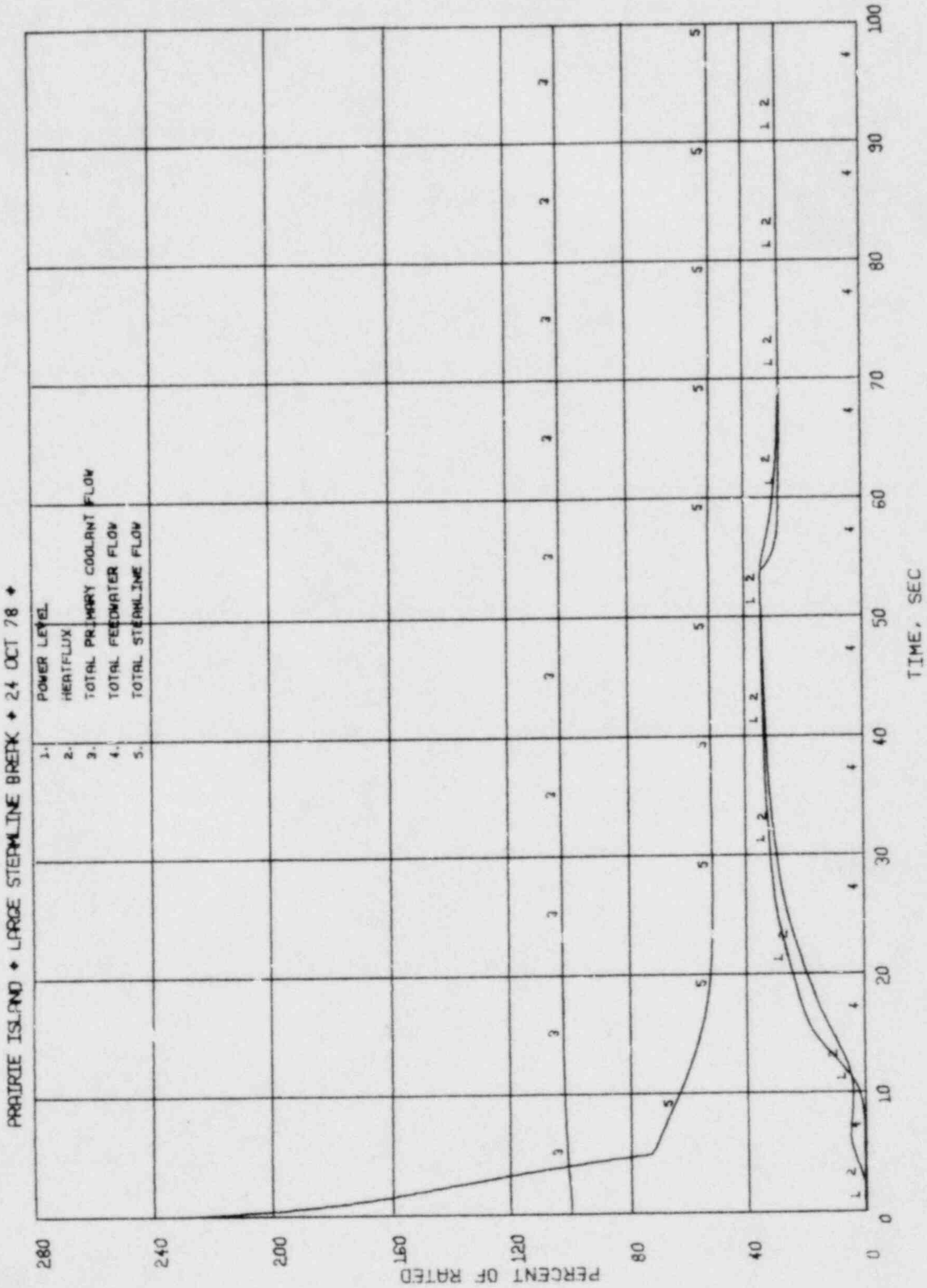

FIGURE 3.33 - Power, Heatflux and System Flow for Large Steamline Break

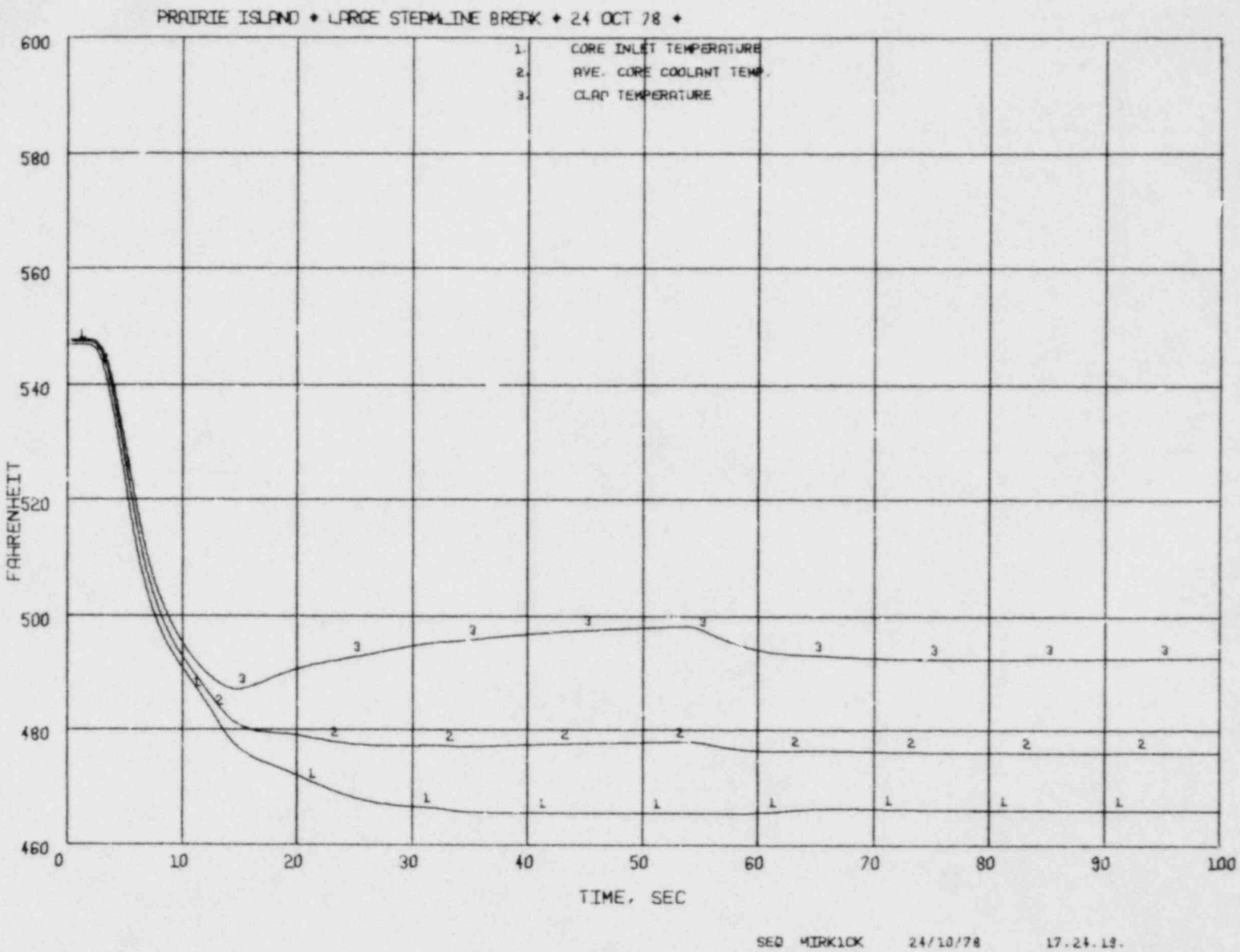

FIGURE 3.34 - Core Temperature Response for Large Steamline Break

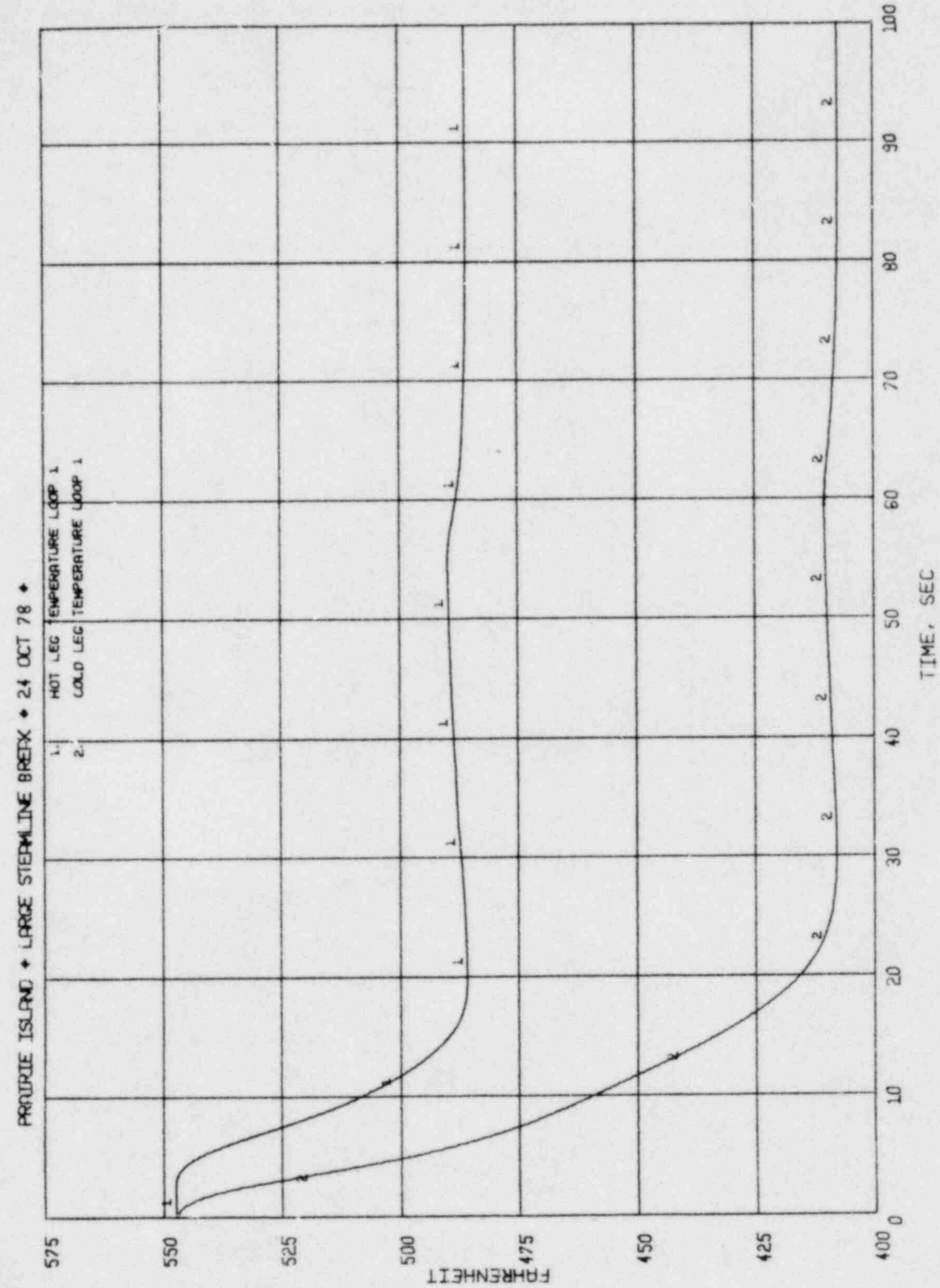

FIGURE 3.35 - Primary Loop Temperature Response for Large Steamline Break

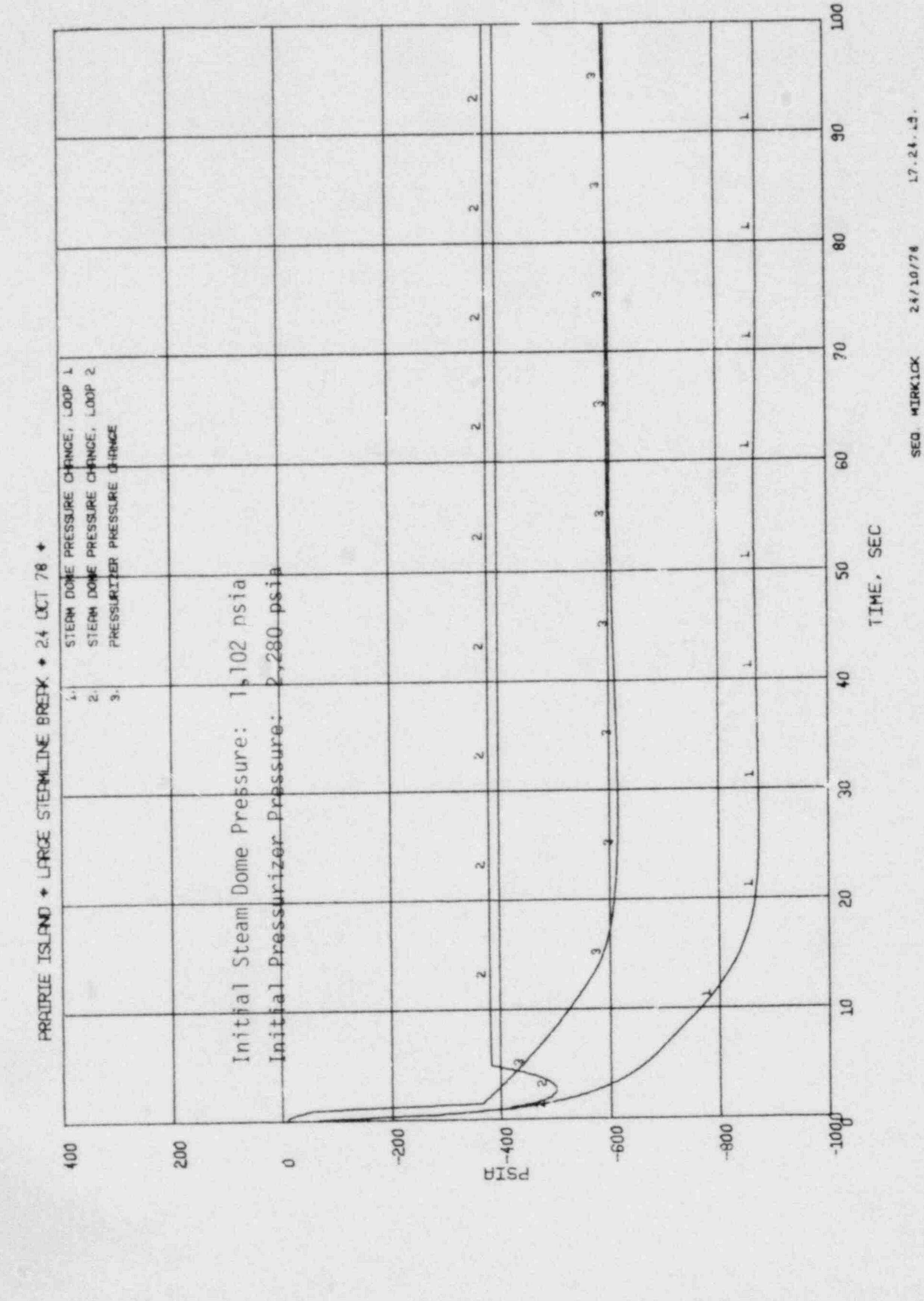

FIGURE 3.36 - Pressure Changes in Pressurizer and Steam Generators for Large Steamline Break

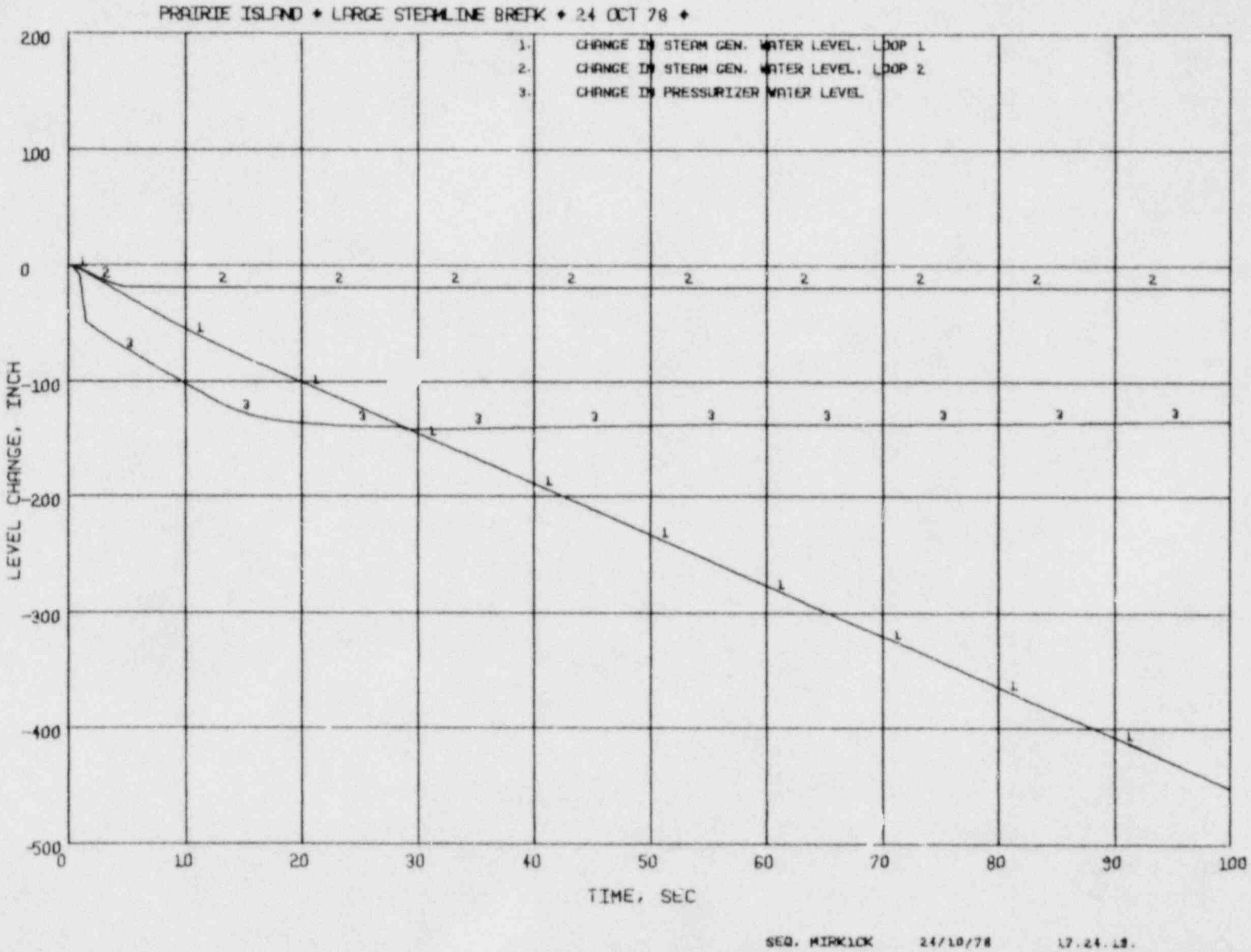

FIGURE 3.37 - Level Changes in Pressurizer and Steam Generators for Large Steamline Break

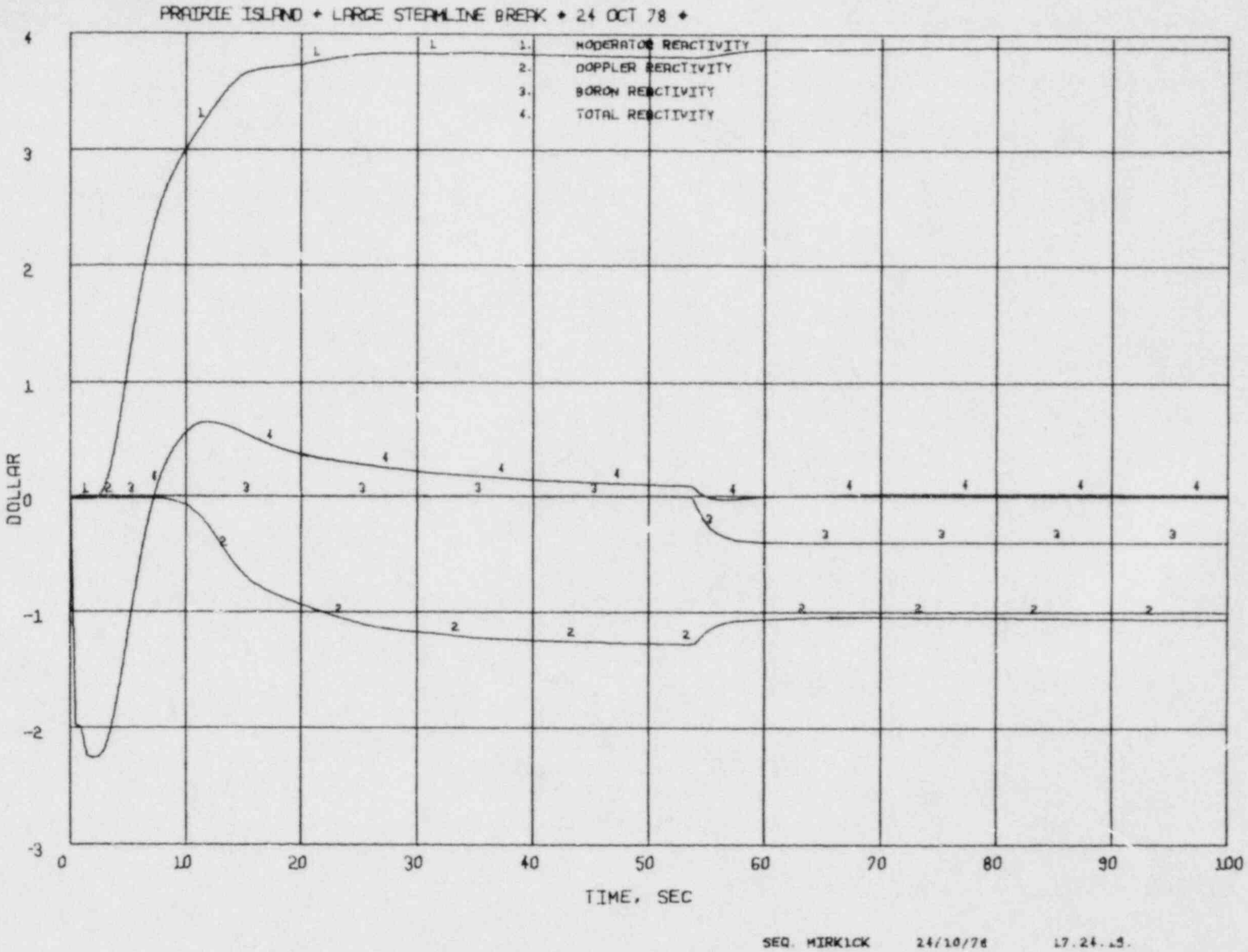

FIGURE 3.38 - Nuclear Reactivity Feedback Effects for Large Steamline Break

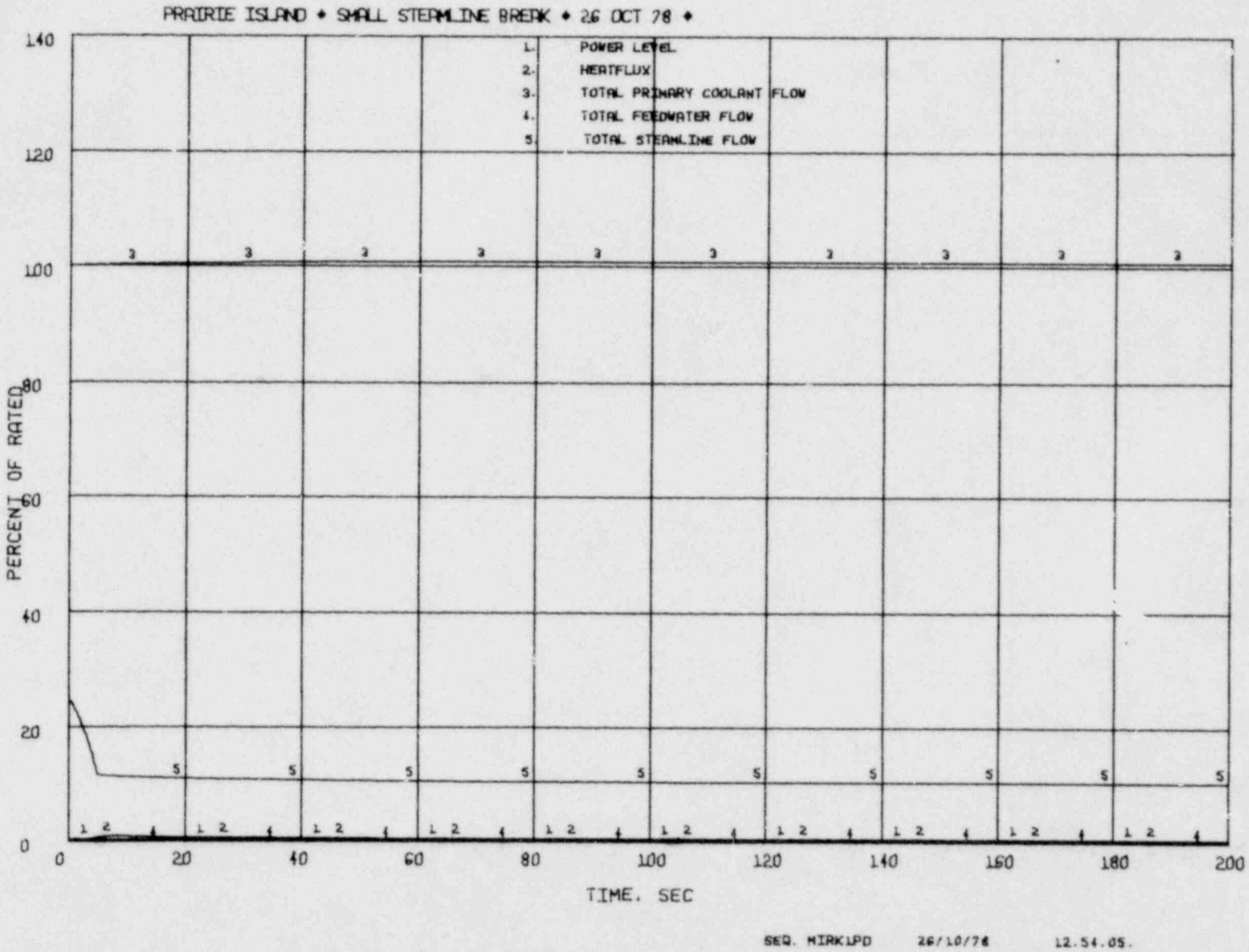

FIGURE 3.39 - Power, Heatflux and System Flows for Small Steamline Break

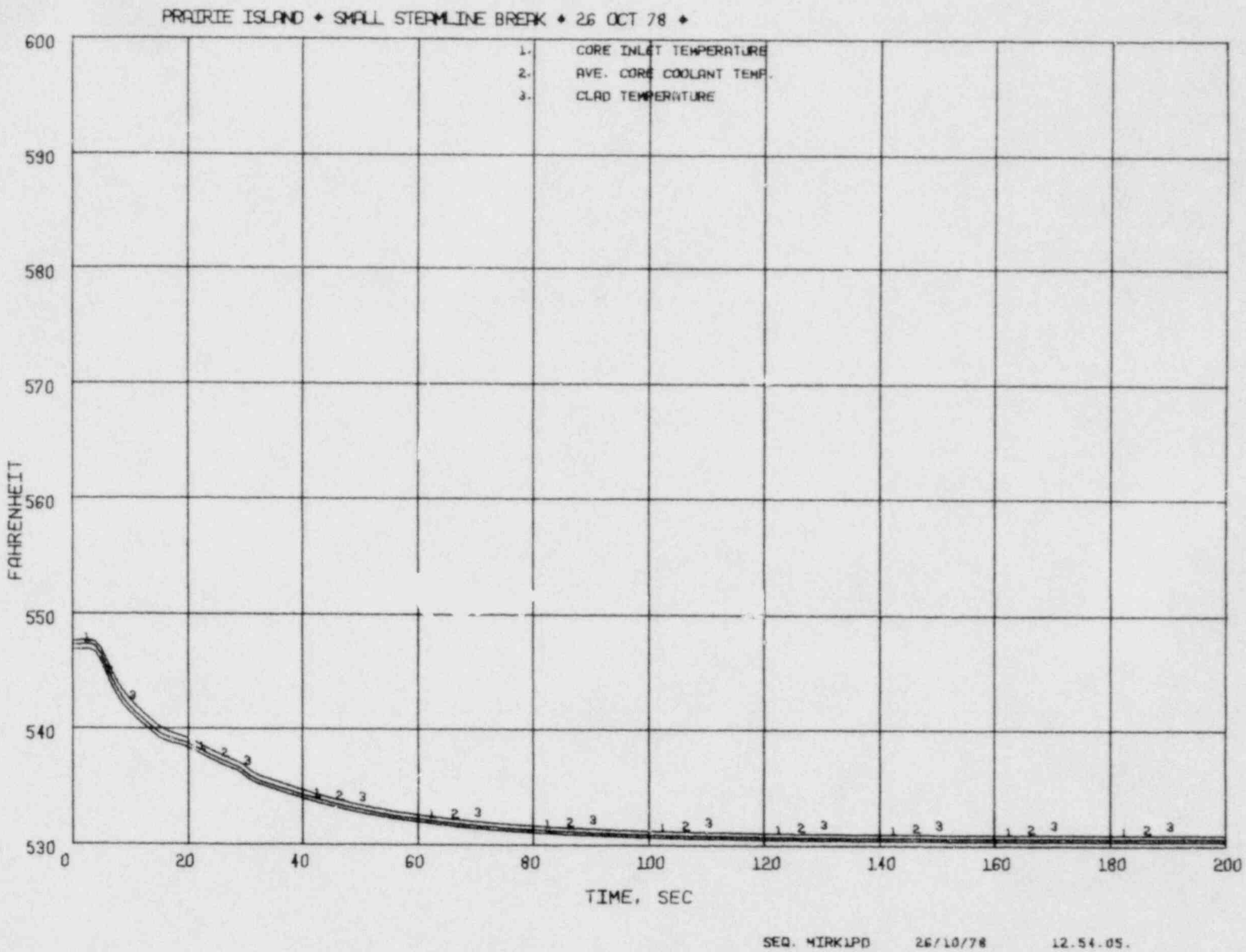

FIGURE 3.40 - Core Temperature Response for Small Steamline break

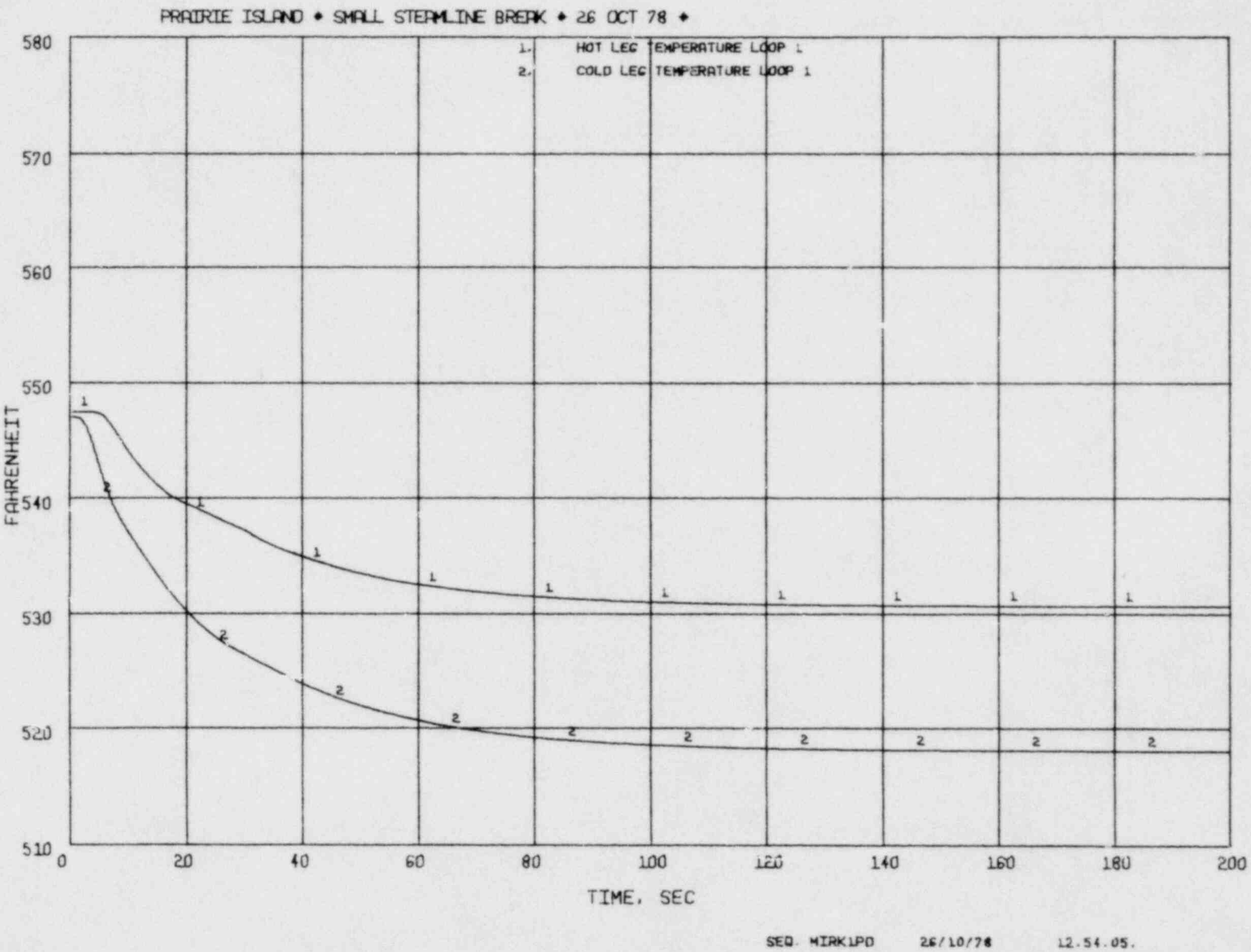

FIGURE 3.41 - Primary Loop Temperature Response for Small Steamline Break

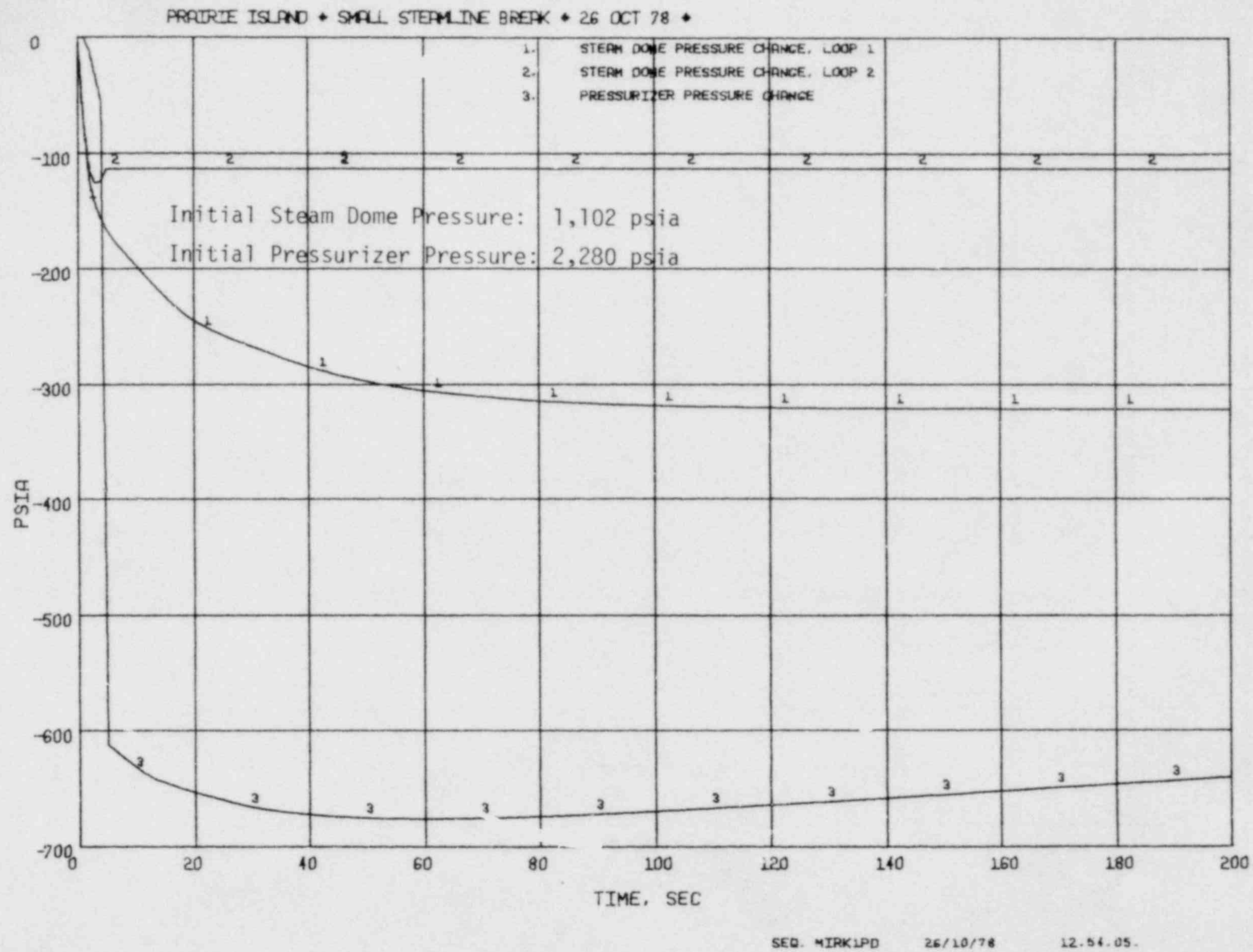

FIGURE 3.42 - Pressure Changes in Pressurizer and Steam Generators for Small Steamline Break

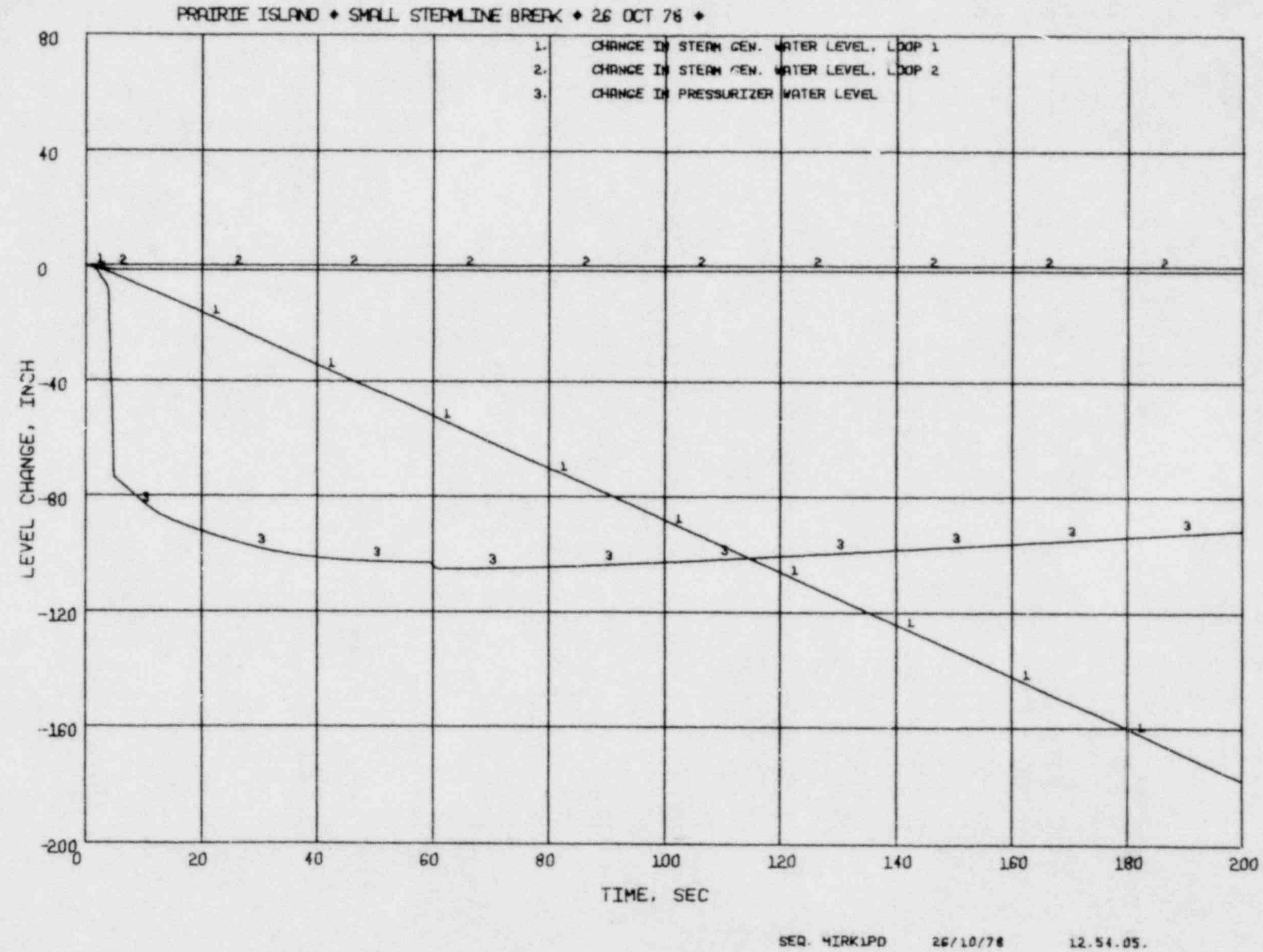

FIGURE 3.43 - Level Changes in Pressurizer and Steam Generators for Small Steamline Break

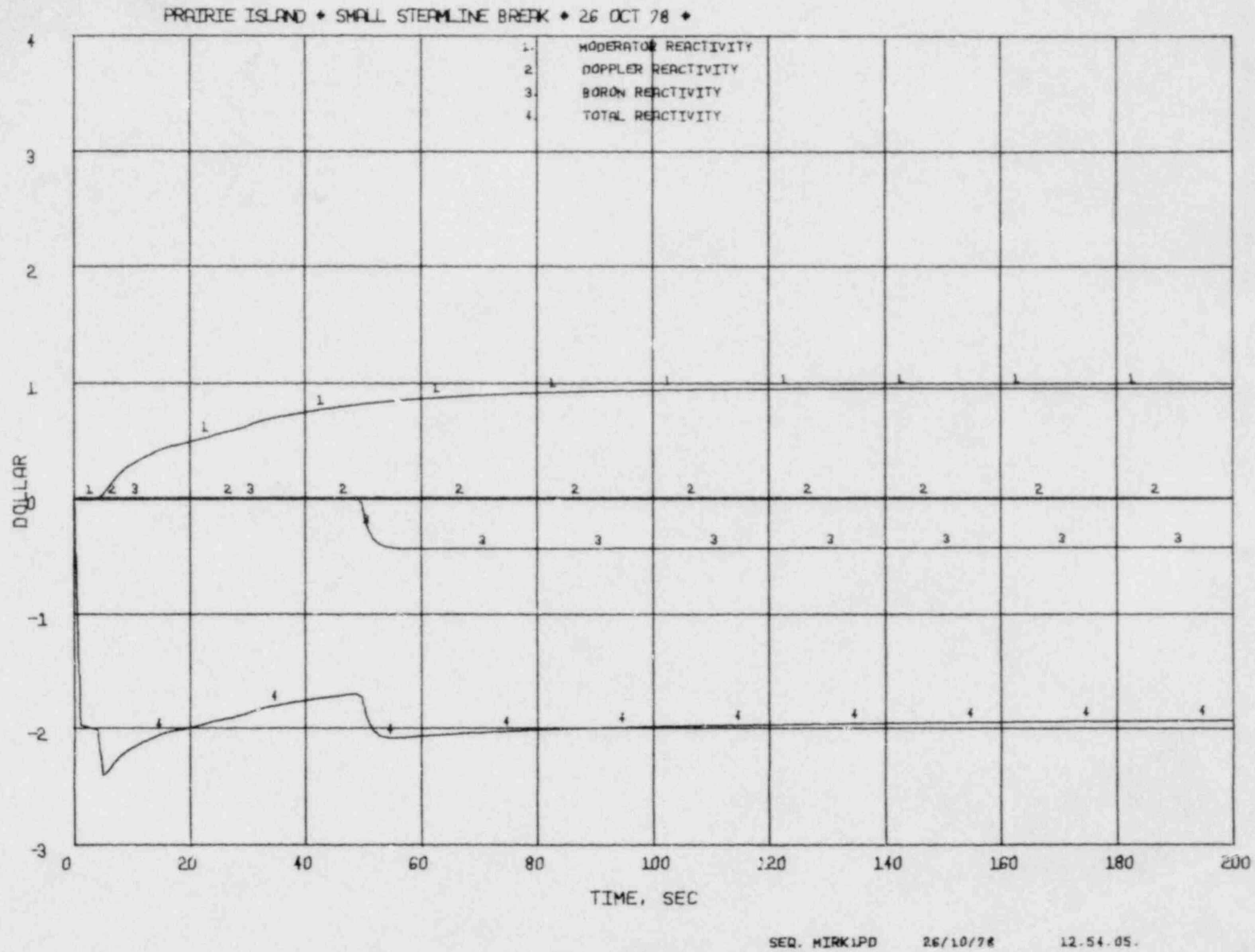

FIGURE 3.44 - Nuclear Reactivity Feedback Effects for Small Steamline Break

4.0 DISCUSSION OF RESULTS

The transient analysis as performed by ENC for the Prairie Island Unit 1 Nuclear Power Plant ensures adequate margin to regulatory limits for the ENC fueled core for anticipated operating conditions. The following transients were analyzed using the ENC PTSPWR2 model.

1) Fast Rod Withdrawal
2) Slow Rod Withdrawal
3) Loss of Power to both Primary Coolant Pumps
4) Locked Rotor in One Primary Coolant Pump
5) Loss of Electric Load
6) Large Steam Line Break
7) Small Steam Line Break

These transients were considered because they were shown in the reference cycle analysis[3] to have the least margin to technical specification limits. Table 4.1 provides a tabulation of the operational transients as analyzed in the FSAR for the reference cycle. The technical specification limits for the transients are a minimum DNB ratio of 1.30 and a peak pressure of 2750 psia. In addition, for the small steam line break, an adequate shutdown margin must be demonstrated such that the reactor does not become critical following the break.

Table 4.2 shows a comparison of general operating parameter values for the reference fuel cycle and for the ENC fuel cycle. The data in Tables 4.1 and 4.2 illustrate that the parameter values used in the Cycle 5 analysis for most cases are either equal to the reference data, or they are enveloped by them. The only exception is the positive moderator coefficient at the beginning of Cycle 5. This means that under most comparable transient conditions, the response of the ENC fuel is either

enveloped by or equivalent to the response of the reference cycle fuel.

The transient analysis of the reference cycle indicated that the heat flux margin to DNB is most limiting in the locked pump rotor case, a Class IV event. Likewise, the ENC analysis showed it to be the most limiting event. It is the only transient where the DNB heat flux ratio is calculated to be below 1.3. The minimum DNB value calculated is 1.09 (see Table 1.1). A statistical analysis shows that fewer than 1 percent of the fuel rods are likely to experience DNB during this event.

If the same transient is analyzed with the calculated full power moderator coefficient of -16.9×10^{-6}/F, the heat flux margin to departure of nucleate boiling is 1.19. The conservative case with a moderator coefficient is $+20 \times 10^{-6}$/F has been reported here, and the following points apply:

1. The design value of the moderator coefficient at the beginning of Cycle 5 is $+16.6 \times 10^{-6}$ 1/F. To cover measurement uncertainties, an analysis value of $+20 \times 10^{-6}$ 1/F has been chosen. The design value, however, applies to hot standby conditions. As reactor power increases, the coefficient decreases, turning negative at about 70 percent power, and reaching a calculated value of $-36.9 \pm 20 \times 10^{-6}$/F at full power.
2. For the Plant Transient Analysis, the design value of 63.2×10^{6} lb/hr has been used for the reactor flow, which is only 92 percent of the actual plant flow as measured and documented in the Technical Specifications[4].

3. The setpoint for the low loop flow trip function is decreased for the analysis in order to envelope instrumentation errors; 87 percent is used rather than the plant value of 90 percent. In addition, the design flow value of 68.2 M lb/hr has been used instead of the measured plant flow which is at least 75.8 M lb/hr.

In the reference cycle analysis, the following transients also showed a reduction in MDNBR from steady state conditions:

- Startup of Inactive Loop
- Feedwater System Malfunctions
- Excessive Load
- Loss of AC Power

For the reference cycle, the lowest MDNBR during a Class II or III incident was 1.61 for the 2 pump trip incident. These transients were not reanalyzed because they did not result in as large MDNBR changes as those analyzed in this report and thus were not limiting in the reference cycle analysis and would not be limiting for an ENC fueled core either. Since the system response for these transients is insensitive to the fuel type, the only variation in results would be the DNB ratio.

Table 4.3 compares the neutronic parameter values of the reference cycle analysis to the ones for the Cycle 5 analysis. As pointed out earlier, conservative values for the moderator and Doppler feedback coefficients have been used in the analysis.

In the reference cycle analysis, the rod withdrawal transient has been analyzed for a spectrum of reactivity insertion rates from $\dot{\Delta k} = 10^{-6}$ 1/sec to $\dot{\Delta k} = 10^{-3}$ 1/sec. This spectrum of insertion rates is covered by two trip functions: the overtemperature ΔT trip for low insertion rates and the high nuclear flux trip for high insertion rates.

At full power, the crossover point between these 2 functions is at 2.7×10^{-5} 1/sec which corresponds to the point of lowest DNB margin for the high nuclear flux trip regime. For insertion rates below the crossover value, the the minimum DNB heat flux ratio stays almost constant, and for insertion rates above it, the MDNBR rapidly increases due to the fast acting high flux function. Going to partload operation changes the plant response somewhat. In the reference cycle analysis, the response spectra for 60 percent power and 10 percent power are shown. The crossover value moves to $\dot{\Delta k} = 1.5 \times 10^{-4}$ 1/sec for 60 percent power and up to $\dot{\Delta k} = 2.8 \times 10^{-4}$ 1/sec for 10 percent power. Going to partload increases the margin to DNB flux in the regime of the high nuclear power function, and it slightly lowers the MDNBR in the overtemperature ΔT regime from a typical value of 1.34 down to 1.30. The ENC analysis has shown that the rod withdrawal transient is not limiting at 102 percent of nominal power. Since the change in plant response caused by lower power levels is mainly dependent on the plant protection system, it can be expected that the response trend in partload cases for Cycle 5 fuel is analogous to the trend for the reference cycle fuel. Therefore, adequate protection is ensured over the complete range of power levels.

A malfunction of the chemical and volume control system is also enveloped by the rod withdrawal transient. During this malfunction, reactivity is added to the core by addition of unborated primary coolant makeup water. The plant response is similar to that for the slow rod withdrawal

transient analyzed in Section 3.1, except that the rate of reactivity insertion is lower. A typical boron dilution event would cause a reactivity insertion at $\dot{\Delta k} = 10^{-5}$ 1/sec. At all power levels, this insertion rate falls into the regime of the overtemperature ΔT function. The plant response for this event would be identical to the slow rod withdrawal case (at $\dot{\Delta k} = 10^{-5}$ 1/sec) analyzed in Section 3.1.

Certain operational incidents are not dependent on fuel type. These include:

- RCCA Misalignment
- Turbine Generator Overspeed
- Fuel Handling Incident
- Accidental Waste Gas Release
- Radioactive Liquid Release
- Steam Generator Tube Rupture

These incidents as discussed in the reference cycle analysis were shown to be protected for any fuel type by administrative controls, redundancy of alarms, and/or integrity of system components. The conclusions drawn for these incidents as given in the reference cycle analysis are valid for Cycle 5 and all future reload cycles with ENC fuel.

TABLE 4.1

COMPARISON OF TRANSIENT-SPECIFIC

INPUT PARAMETERS

	Reference Cycle		PTS Analysis for Cycle 5 ENC Fuel	
	Moderator Coefficient	Doppler Coefficient	Moderator Coefficient**	Doppler Coefficient
	$(\Delta\rho / F \times 10^6)$	$(\Delta\rho / F \times 10^6)$	$(\Delta\rho / F \times 10^6)$	$(\Delta\rho / F \times 10^6)$
Rod Withdrawal				
From Full Power	0.0	small	+20.0	-10.0
From Reduced Power	0.0	small	---	---
Loss of Flow				
Pump Coastdown	0.0	-16.3	+20.0	-15.0
Locked Rotor	0.0	NA*	+20.0	-15.0
Inactive Loop Startup	-40.0	-10.0	---	---
Loss of Load	0.0	NA*	+20.0	-10.0
Loss of Feedwater	NA*	NA*	---	---
Excessive Feedwater	0.0 and -40.0	NA*	---	---
Excessive Load Increase	0.0 and -40.0	NA*	---	---
Steam Line Break	Variable	Variable	Fig. 3.32	Fig. 3.31

* Information not available in reference cycle analysis

** See discussion of moderator coefficient on page 65

TABLE 4.2

COMPARISON OF OPERATING PARAMETERS

PRAIRIE ISLAND UNIT 1 AND UNIT 2

	Reference Cycle	Cycle 5 With ENC Fuel
Core		
Total Core Heat Output, MW	1650	1650
Heat Generated in Fuel, percent	97.4	97.4
System Pressure, psia	2250	2250
Hot Channel Factors*		
Total Peaking Factor, F_Q^T	2.80	2.32
Enthalpy Rise Factor	1.58	1.55
Axial Peaking Factor, F_Z	1.72	1.45
Location of Axial Peak, ft	NA**	6.2
Coolant Massflow, lb/hr	68.20×10^6	68.20×10^6***
Effective Core Massflow, lb/hr	64.64×10^6	64.64×10^6
Reactor Inlet Temperature, F	535.5	530.5
Heat Transfer		
Average Heatflux, Btu/hr-ft^2	190,973	190,973

* Hot channel factors as applied to safety analysis and thermal-hydraulic analysis only.

** Information not available in reference cycle analysis.

*** This is the design value. The actual reactor flow as stated in the Technical Specifications and confirmed by measurements is at least 75.8×10^6 lb/hr.

TABLE 4.2 (Continued)

COMPARISON OF OPERATING PARAMETERS

FOR PRAIRIE ISLAND UNIT 1 AND UNIT 2

	Reference Cycle	Cycle 5 With ENC Fuel
Steam Generators		
Total Steam Flow, lb/hr	7.080×10^6	7.091×10^6
Steam Temperature, F	510.8	510.9
Steam Pressure, psia	750.0	750.0
Feedwater Temperature, F	427.3	427.3

TABLE 4.3

COMPARISON OF KINETIC PARAMETER VALUES FOR PRAIRIE

ISLAND UNIT 1 AND UNIT 2

	Reference Cycle		Cycle 5 with ENC Fuel	
	BOC	EOC	BOC	EOC
Moderator Temperature Coefficient in 1/5	$+30 \times 10^{-6}$*	-350×10^{-6}	$+20 \times 10^{-6}$*	-350×10^{-6}
Moderator Pressure Coefficient in 1/psia	-0.3×10^{-6}	$+3.5 \times 10^{-6}$	-0.2×10^{-6}	$+4 \times 10^{-6}$
Doppler Coefficient in 1/F	-10×10^{-6}	-16×10^{-6}	-12.5×10^{-6}	-16×10^{-6}
Delayed Neutron Fraction	7.1×10^{-3}	5.1×10^{-3}	6.1×10^{-3}	5.1×10^{-3}

* Value for hot standby conditions.

5.0 SIMULATION CODE CHANGES

The basic digital plant simulation code as documented in Reference 1 has been used in performing the plant transient analysis for the Prairie Island plant. Starting from the version PTS-PWR2-NOV76A, several code changes have been inplemented resulting in the version PTS-PWR2-NOV78. All changes were restricted to the initialization modules of the code. Therefore, the dynamic plant model of the PTS code was not affected. The purpose of the changes was (a) to remove a number of redundant variables from the input list and generate them internally in the code and (b) to redefine some input parameters such that hand calculations are eliminated or reduced. All code changes have been checked individually. In addition, the pump seizure transient for the R. E. Ginna plant has been rerun, and the results were found to be very close to previous results. Some key results are shown in Table 5.1. An alphabetic list of the affected variables is shown in Table 5.2.

TABLE 5.1

Comparison of Results for the

RE Ginna Pump Seizure Transient

	Version PTS-PWR2 NOV 76A	Version PTS-PWR2 NOV 78
Minimum DNB Flux Ratio	1.23	1.23
Maximum Reactor Power, %	102.	102.
Peak Value of Average Core Heatflux, btu/(hr x ft^2)	181,163.	181,160.
Reactor Flow at 5 sec., %	49.	49.
Peak Core Average Temperature, F	590.	591.

TABLE 5.2

LIST OF CODE VARIABLES REMOVED REMOVED FROM INPUT OR REDEFINED

BETA	KSHTB	NBETA1	WBDG1I
	KSLSH1	NBETA2	WBDG2I
CFSPR	KSLSH2	NBETA3	WDOSLR
CFWPR	KUPSP	NBETA4	WDOSL1
CFWPRS	K1112	NBETA5	WDOSL2
	K2122	NBETA6	WFWMAX
DOSL1I			WFW1
DOSL2I	LEVG1I		WFW2
	LEVG2I	QOAR	WIV1IC
FWCIC2	LTOPSG		WIV2IC
		SLSH1I	WLPCRR
HFW1IC	MDO1IC	SLSH2I	WLP1IC
HFW2IC	MDO2IC		WLP2IC
HGUP1I	MSG1IC	T1POIC	WTBMAX
HGUP2I	MSG2IC	T2POIC	WSO
HUP1IC	MSHIC	T1P3IC	
HUP2IC	MSL11I	T2P3IC	
HWPRIC	MSL12I		
	MSL21I	UPSP1I	
KBDSG	MSL22I	UPSP2I	
KBDTD	MSPRIC		
KDOSL1	MUP1IC		
KDOSL2	MUP2IC		
KSGUP	MWPRIC		

REDEFINED VARIABLES

Variable	New Definition
CFWPRS	Setpoint for High Pressurizer Level Trip, in Percent of Span
HFW1IC	Feedwater Temperature for Both Coolant Loops, In Fahrenheit
MWPRIC	Initial Pressurizer Level, in Percent of Span
NBETA1	Delayed Neutron Fraction, Group 1
NBETA2	Delayed Neutron Fraction, Group 2
NBETA3	Delayed Neutron Fraction, Group 3
NBETA4	Delayed Neutron Fraction, Group 4
NBETA5	Delayed Neutron Fraction, Group 5
NBETA6	Delayed Neutron Fraction, Group 6

All variables not redefined have been removed.

6.0 REFERENCES

1. Kahn, J. D., Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR), XN-74-5, Revision 1, May 1975.

2. Galbraith, K. P., et al., Definition and Justification of Exxon Nuclear Company DNB Correlation for Pressurized Water Reactors, XN-75-48, October 1975.

3. Northern States Power Company, Prairie Island Nuclear Generating Plant, Units 1 and 2, Final Safety Analysis Report.

4. Northern States Power Company, Prairie Island Nuclear Generating Plant, Units 1 and 2, Technical Specifications, Docket 50-282 and 50-306.

5. Kahn, J. D., Assumptions Used in the Plant Transient Analysis for the Donald C. Cook Unit 1 Nuclear Power Plant, XN-76-35, Supplement 1, November 1976.

6. Lyle, J. M. and Killgore, M. R., Prairie Island Unit 1 Cycle 5 Fuel Management Analysis, Exxon Nuclear Company, XN-NF-78-43, October 1978.