PLANT TRANSIENT ANALYSIS FOR THE KEWAUNEE

NUCLEAR POWER PLANT

By

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION AND SUMMARY

The XN1 reload for Core 5 of the Kewaunee Nuclear Power Plant results in core parameter values only slightly different from previous cycle values. 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 plant. The input data for the analysis have been chosen in such a way that the analysis is expected to cover all future Exxon Nuclear reloads for the Kewaunee Plant.

Event	Incident Class*
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 large steamline break and for the primary pump seizure is that basic core geometry is preserved and fuel damage is within allowable limits. The criterion for the small steamline break is that the shutdown margin at the end of the fuel cycle is sufficient to prevent the core from becoming critical following a break at hot standby conditions.

The analyses are 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 1.86, 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 below 1.3 for about 6 percent of the fuel, compared to about 20 percent in the Reference Analysis (see Reference 3, Section 14). 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 used 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	193,874	2220	2.24
Fast Control Rod Withdrawal (II)(0.6×10^{-3}/sec)	124	206,100	2220	2.03
Slow Control Rod Withdrawal (II)(11×10^{-6}/sec)	113	214,100	2366	1.94
Loss of Flow - (III) Coastdown of both pumps	102	193,900	2246	1.86
Loss of Flow - (IV) Primary Pump Seizure	102	193,900	2265	1.3***
Loss of Load (II)	102	194,500	2526	2.19
Large Steam Line Break (IV)	48	62,400	*	1.51
Small Steam Line Break (IV) **	**	**	*	-

* Pressure decreases from initial value.

** The core does not go critical.

*** Except for less than 6.1 percent of the fuel, which is calculated to experience a thermal margin of less than 1.3.

2.0 CALCULATION METHODS AND INPUT PARAMETERS

The transient analysis for the Kewaunee 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 stopvalves, 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	= 534.0 + 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 Kewaunee 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 Core 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 shutdown 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 zero moderator temperature feedback has been used. For the doppler feedback coefficient, either the smallest or the largest value given in Table 2.4 has been used, depending on which input 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 pump seizure 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]

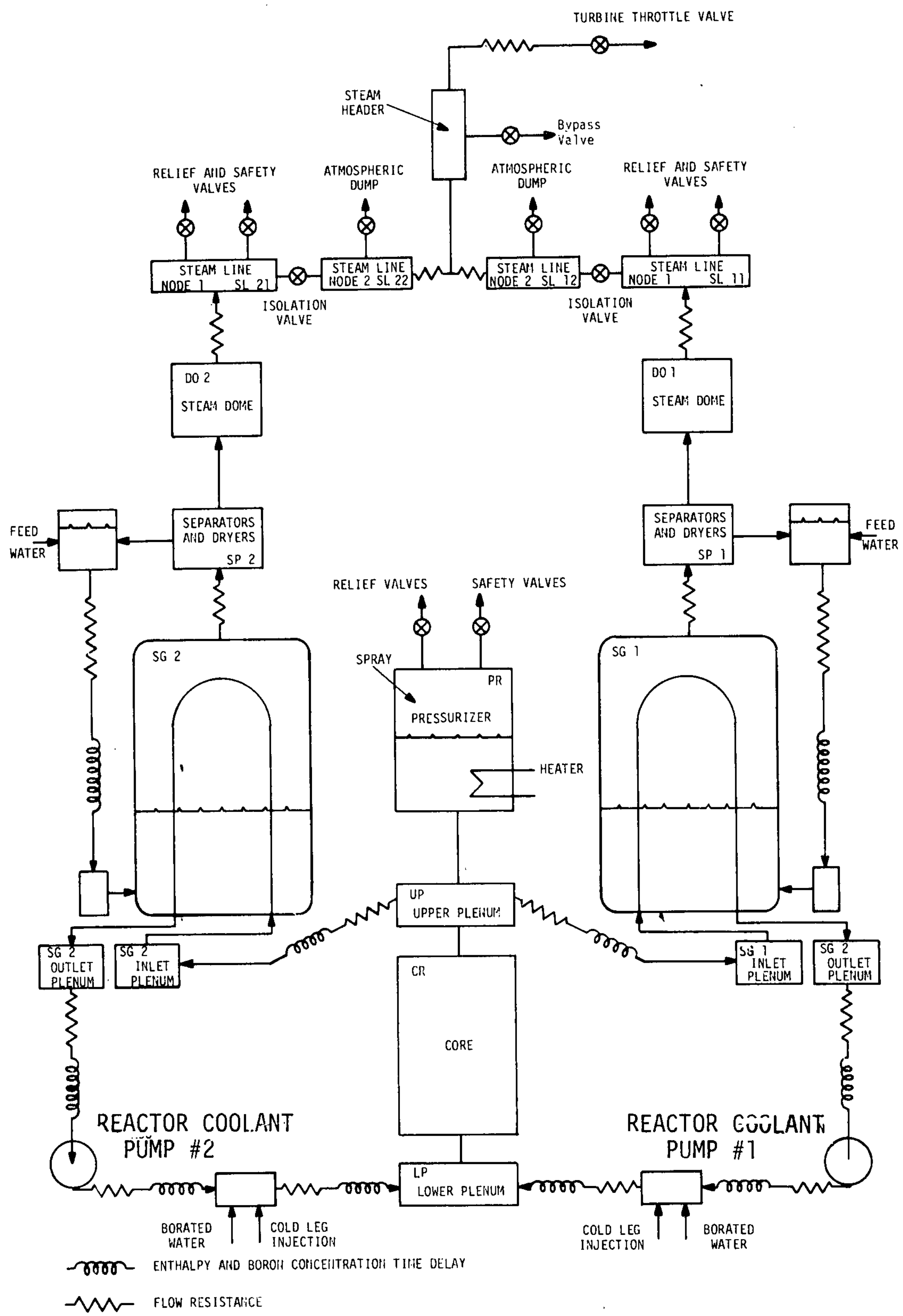

FIGURE 2.1 PTSPWR2 SYSTEM MODEL

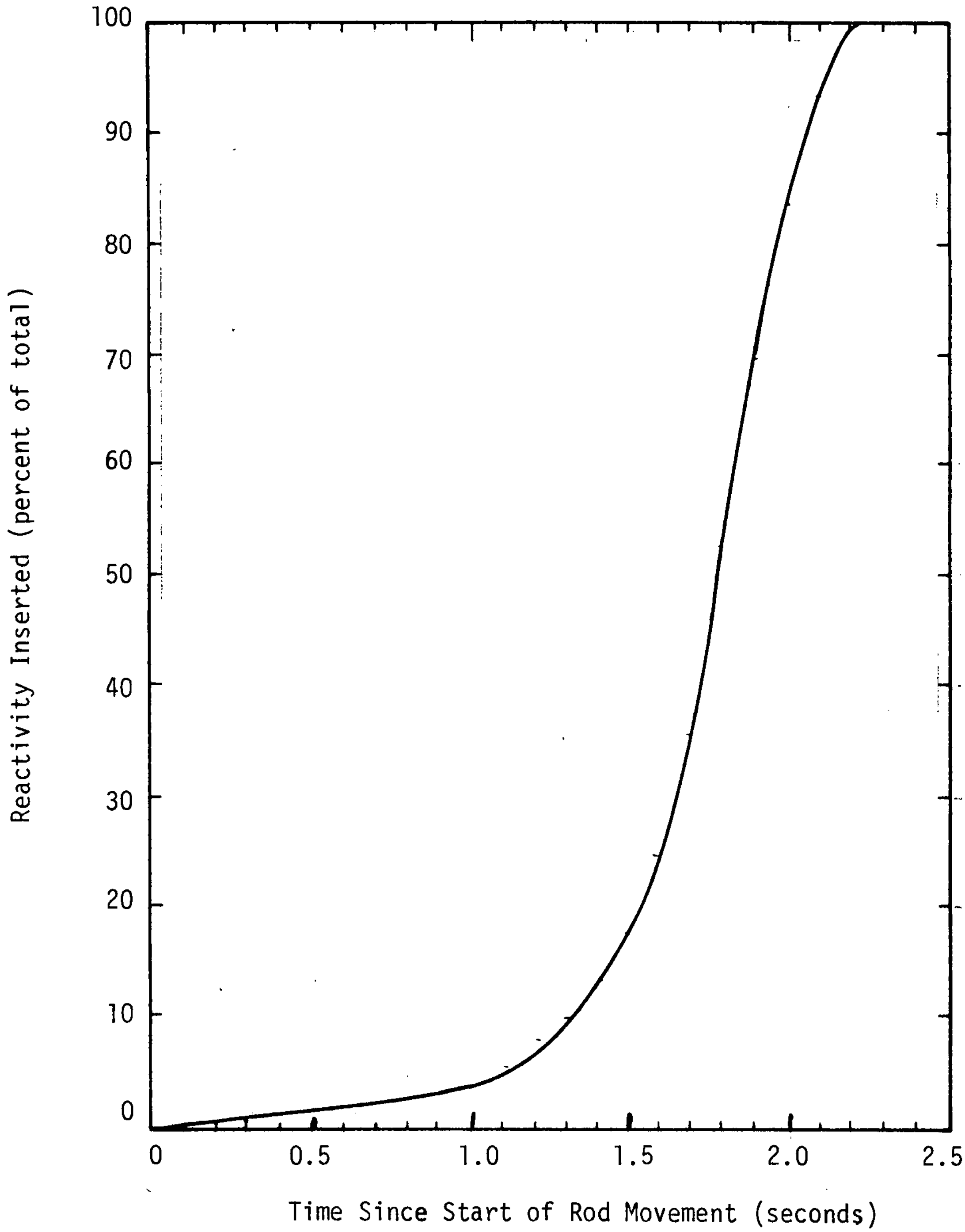

FIGURE 2.2 SCRAM REACTIVITY CURVE USED IN TRANSIENT ANALYSIS OF THE KEWAUNEE NUCLEAR PLANT

TABLE 2.1

PARAMETER VALUES USED IN PTSPWR2

ANALYSIS OF THE KEWAUNEE PLANT

	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	64.64×10^6
Reactor Inlet Temperature, F	538.0
Heat Transfer	
Calculated Average Heat Flux* Btu/hr-ft^2	193,874
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

KEWAUNEE 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	2425 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	1.5 sec
Overtemperature ΔT*	T_{AVE_0} = 570.1°F	T_{AVE_0} = 570.1°F	6.0 sec
	P_0 = 2250 psia	P_0 = 2250 psia	
High Pressure Safety Injection	a) 1830 psia coincident with 5% level in pressurizer	1800 psia coincident with 0% level in pressurizer	10 sec

* The overtemperature ΔT trip is a function of pressurizer pressure, coolant average temperature, and axial offset. The T_{AVE_0} and P_0 setpoints are contained within the functional relationship.

TABLE 2.3

KEWAUNEE FUEL DESIGN PARAMETER VALUES

FOR EXXON NUCLEAR FUEL

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

KEWAUNEE REACTIVITY DATA

Symbol	Parameter	Value	
		Beginning of Cycle	End of Cycle
α_M	Moderator Coefficient (pcm/F)	0.0	-35.0
α_D	Doppler Coefficient (pcm/F)	-1.0 to -2.32	-1.0 to -2.32
α_P	Pressure Coefficient (pcm/psi)	0.0	+0.412
α_V	Moderator Density Coefficient pcm/(g/cm^3)	0.0	+31,010.
α_B	Boron Worth Coefficient (pcm/ppm)	-8.00*	-8.00*
β_{eff}	Delayed Neutron Fraction (pcm)	700	485
α_{CRC}	Total Rod Worth (pcm)	-4,000.*	-5,000.*

* These are conservative values, for analysis purposes only. The actual plant values are 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 power control 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 the weakest doppler feedback coefficient in the range given (-1.0 pcm/F, see Table 2.4).

Figures 3.1 to 3.6 show plant responses for a fast rod withdrawal with $\Delta\dot{K} = 0.6 \times 10^{-3}$ sec^{-1} at full power. A nuclear overpower trip (118% setpoint) occurs at 2.14 seconds. The DNB flux ratio drops from an initial value of 2.24 to 2.03. Pressure increases to a maximum of 2225 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 $\Delta\dot{K} = 11.0 \times 10^{-6}$ sec^{-1} 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 104 sec, and the shutdown rod insertion starts after a 6.0 sec delay. The minimum DNB flux ratio is 1.94 at about 103 sec.

3.3 COASTDOWN OF BOTH PRIMARY COOLANT PUMPS

Flow coastdown incidents resulting from a loss of electric power to the primary coolant pumps result in an 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 with the reactor system operating in 1683.0 MWT. Beginning of cycle values for kinetic coefficients are assumed. For the doppler feedback coefficient, the least negative value in the given range (-1.0 pcm/F, see Table 2.4) was used in order to minimize the initial power decrease during fuel heatup caused by the coolant flow decrease. 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 2.8 sec. A minimum DNB flux ratio of 1.86 is reached at 3.4 sec after beginning of coastdown. At about 5 sec, a pressure peak of 2246 psia is reached.

3.4 SEIZURE OF ONE PRIMARY COOLANT PUMP

In the unlikely event of a seizure of a primary coolant pump, flow through the core is reduced. The reactor is tripped on 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 at 102% of rated power.

The effect of the pressurizer spray and pressurizer relief valves on reducing system pressure was conservatively neglected 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. The weakest doppler feedback coefficient in the range given in Table 2.4 (-1.0 pcm/F) has been used which minimizes the initial power decrease during fuel heatup caused by the flow decrease. The locked pump rotor reduces the core flow to about 47 percent of its nominal value within 2.5 sec, and flow reversal in the accident loop occurs within the first 1.1 sec of the transient. Since this transient is a Category IV event, evaluation of the amount of potential fuel damage is the figure of merit. Fuel failure is assumed coincident with a predicted DNB ratio of less than 1.3. The amount of fuel reaching a thermal margin to DNB below that value has been calculated to be about 6 percent, compared to about 20 percent in the Reference Analysis (see Reference 3, Section 14). Figures 3.19 through 3.24 illustrate the plant response for this transient.

3.5 LOSS OF EXTERNAL ELECTRIC LOAD

The Kewaunee nuclear plant is designed to accept a 100 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, and unavailability of the automatic reactor control is assumed. For doppler feedback, the weakest coefficient (-1.0 pcm /F, see Table 2.4) has been used.

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 1089 psia at 10.5 sec, when the first set of steamline safety valves opens (see Figure 3.28). At 11.5 sec, the second safety valve setpoint of 1105 psia is reached. At 14 sec, the third setpoint of 1120 psia is reached; finally, at 17.5 sec, the fourth setpoint of 1135 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 2526 psia. The pressurizer safety valve is open from about 14.5 sec to 18.0 sec. The average primary coolant temperature increases by about 22°F. The lowest value for the minimum DNB heat-flux ratio is 2.19, at about 12 sec.

3.6 LARGE STEAMLINE BREAK

The large break of a steam pipe 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 fuel temperature used in this analysis are shown in Figures 3.31 and 3.32. Moderator temperature feedback data from the Kewaunee FSAR were used. To ensure that the data used are on the conservative side, a factor of 1.1 was applied to the FSAR values.

Minimum capability of the boron injection system was assumed, which implies that only one of the two high-pressure safety injection pumps (HPSI) is 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 effect of the resident low concentration borated water being swept into the reactor 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 460 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 48 percent of nominal full power at 22 sec with a corresponding peak in core average heat flux of 62,400 Btu/(hr x ft^2). At the same time, the borated water from the high pressure safety injection system reaches the core, initiating a power decrease. A conservatively high local hot spot peaking factor of F_Q^N = 14.0 was used. A shutdown margin of 2,000 pcm was used. The heatflux margin to departure from nucleate boiling goes to a minimum value of 1.51 at about 23 sec.

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, one-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 202 sec, and the core does not become critical. A shutdown margin of 2,000 pcm has been used.

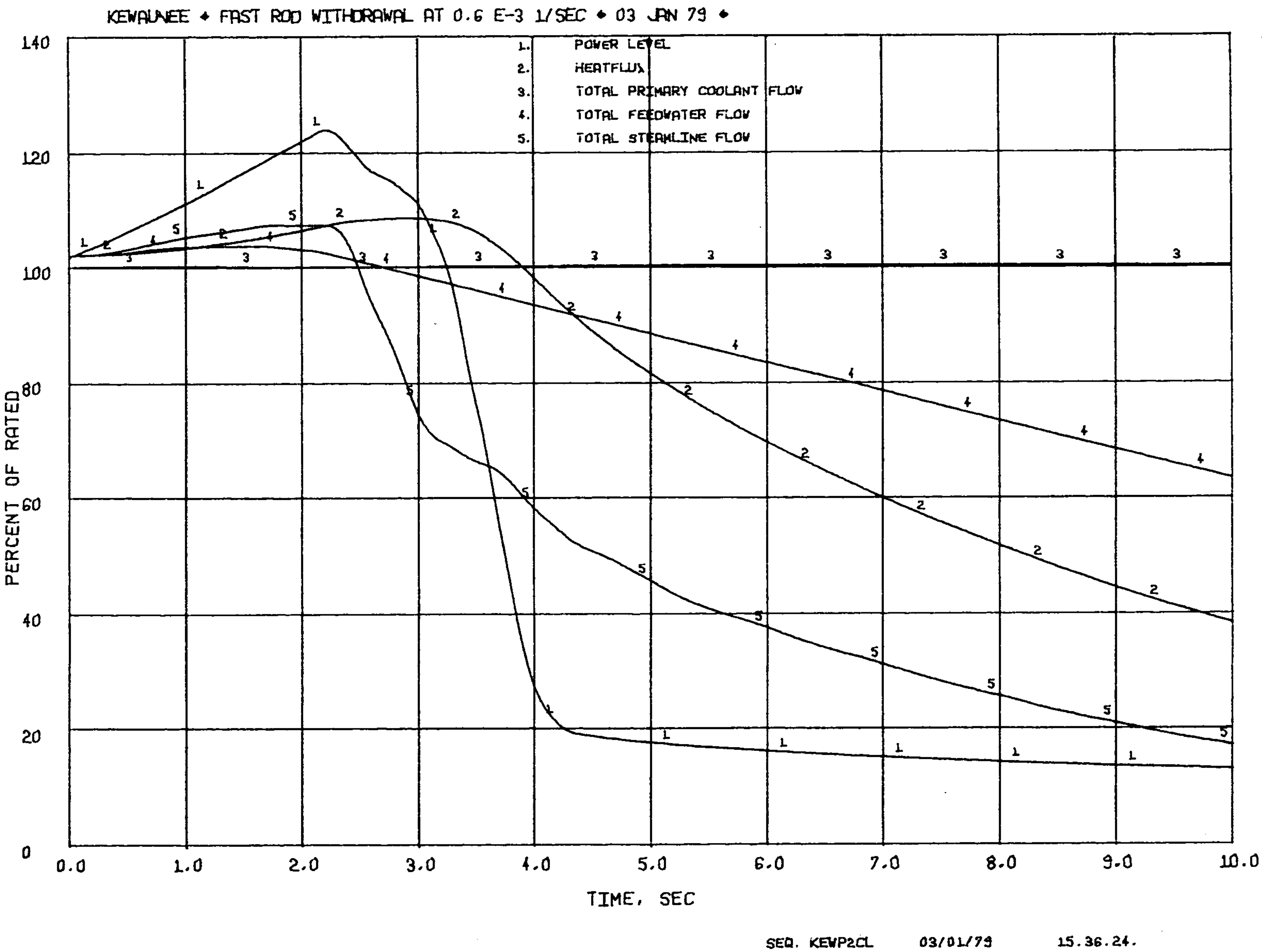

FIGURE 3.1 Power, Heatflux and System Flows for Fast Control Rod Withdrawal

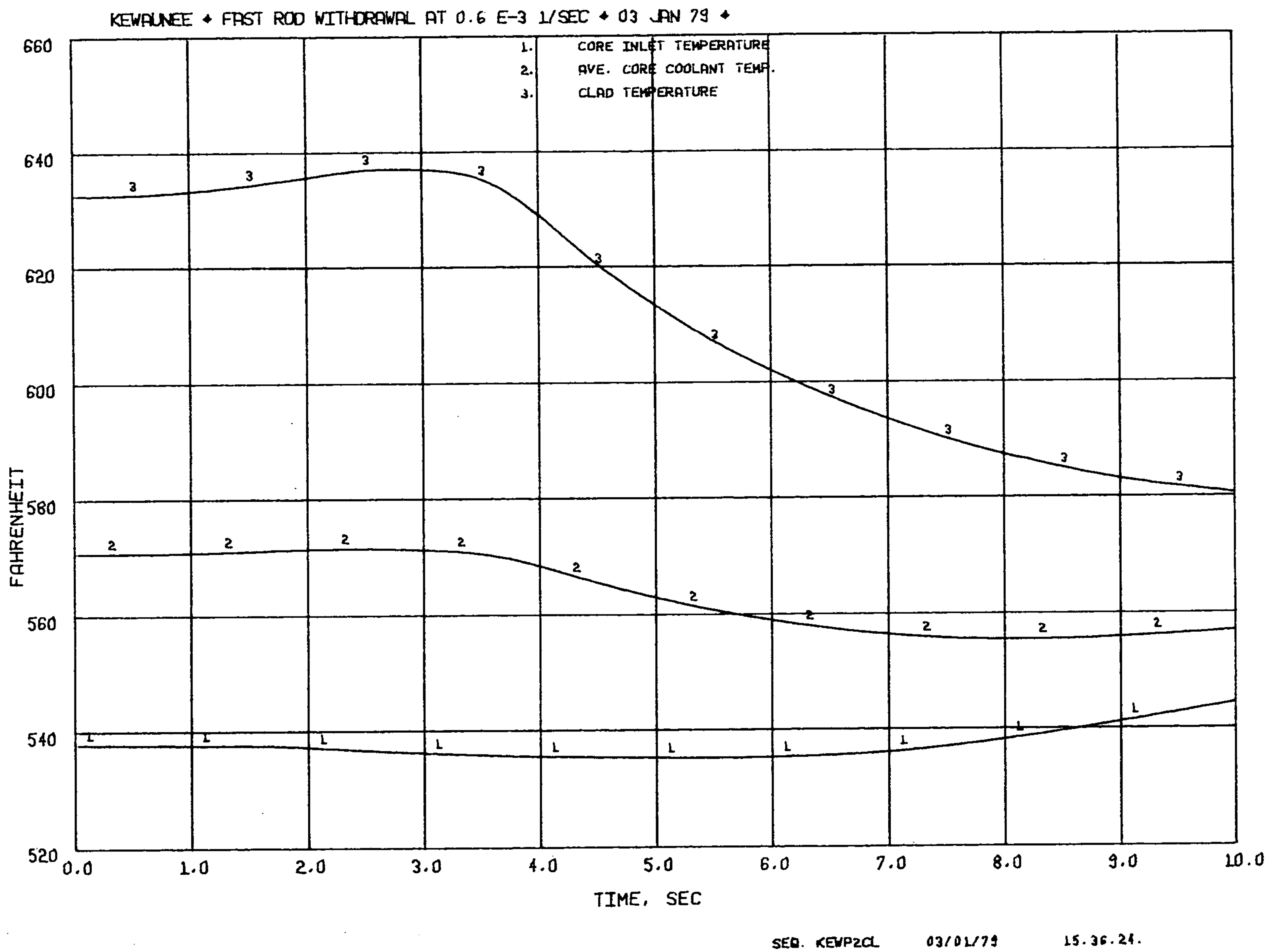

FIGURE 3.2 Core Temperature Response for Fast Control Rod Withdrawal

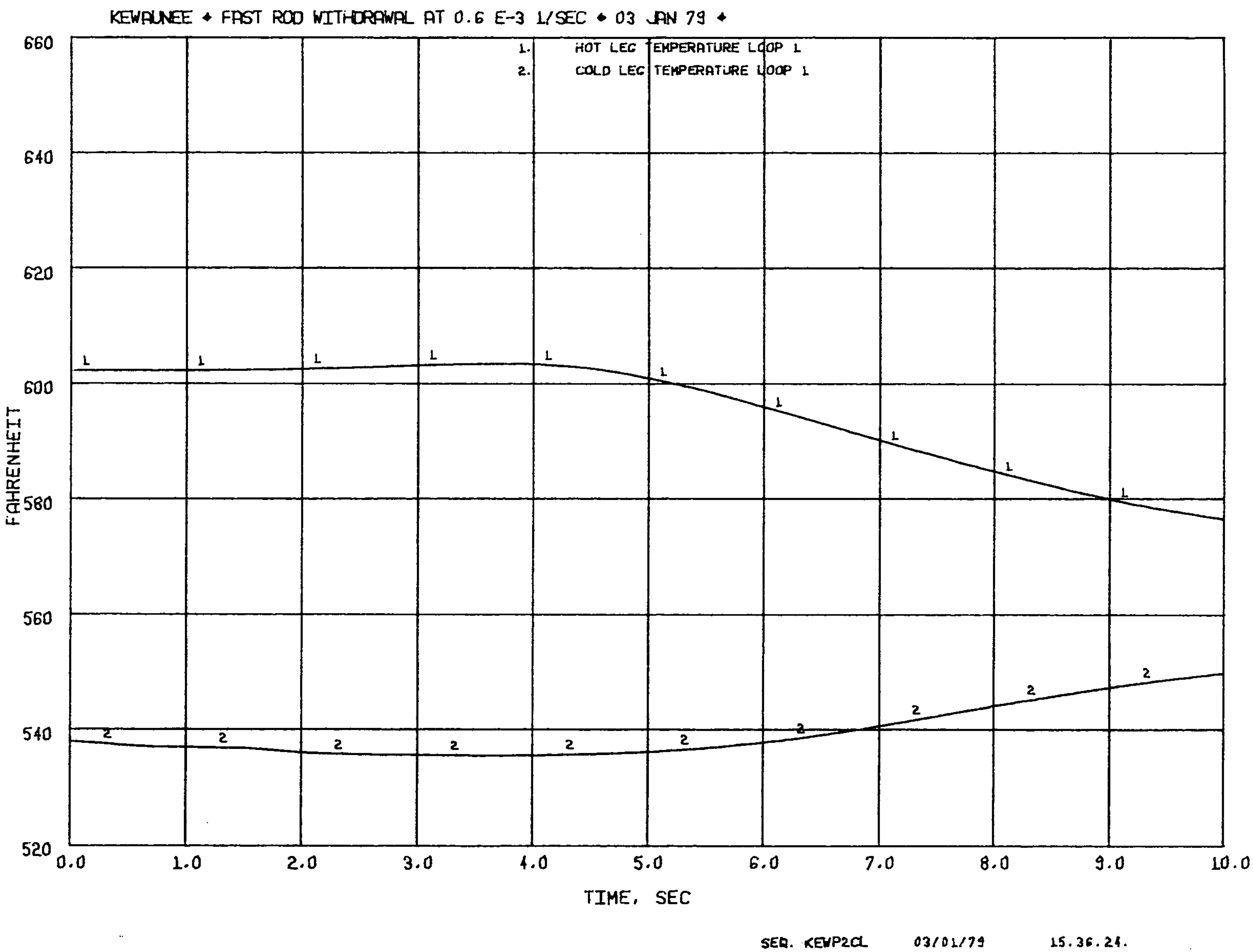

FIGURE 3.3 Primary Loop Temperature Response for Fast Control Rod Withdrawal

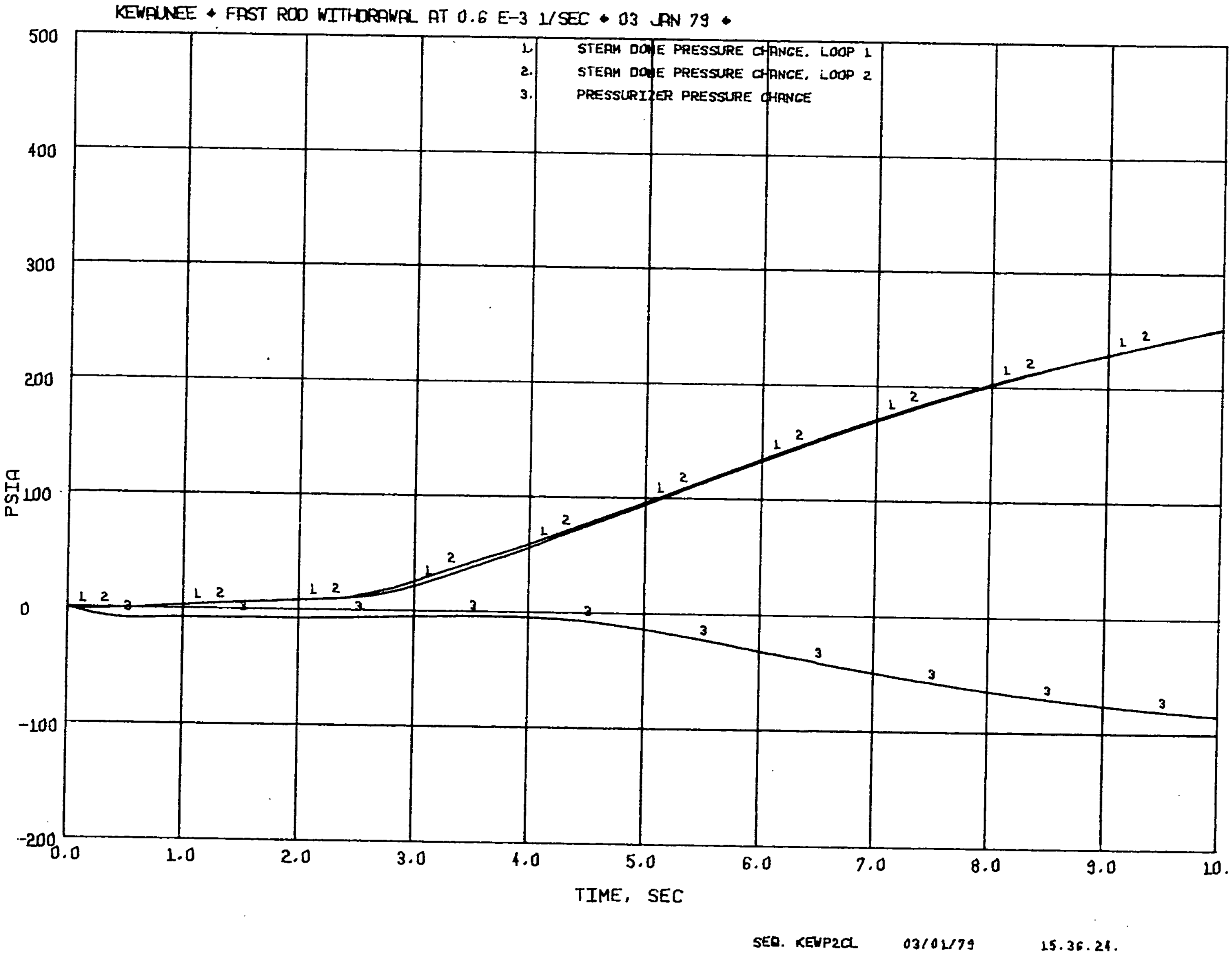

FIGURE 3.4 Pressure Changes in Pressurizer and Steam Generators for Fast Control Rod Withdrawal

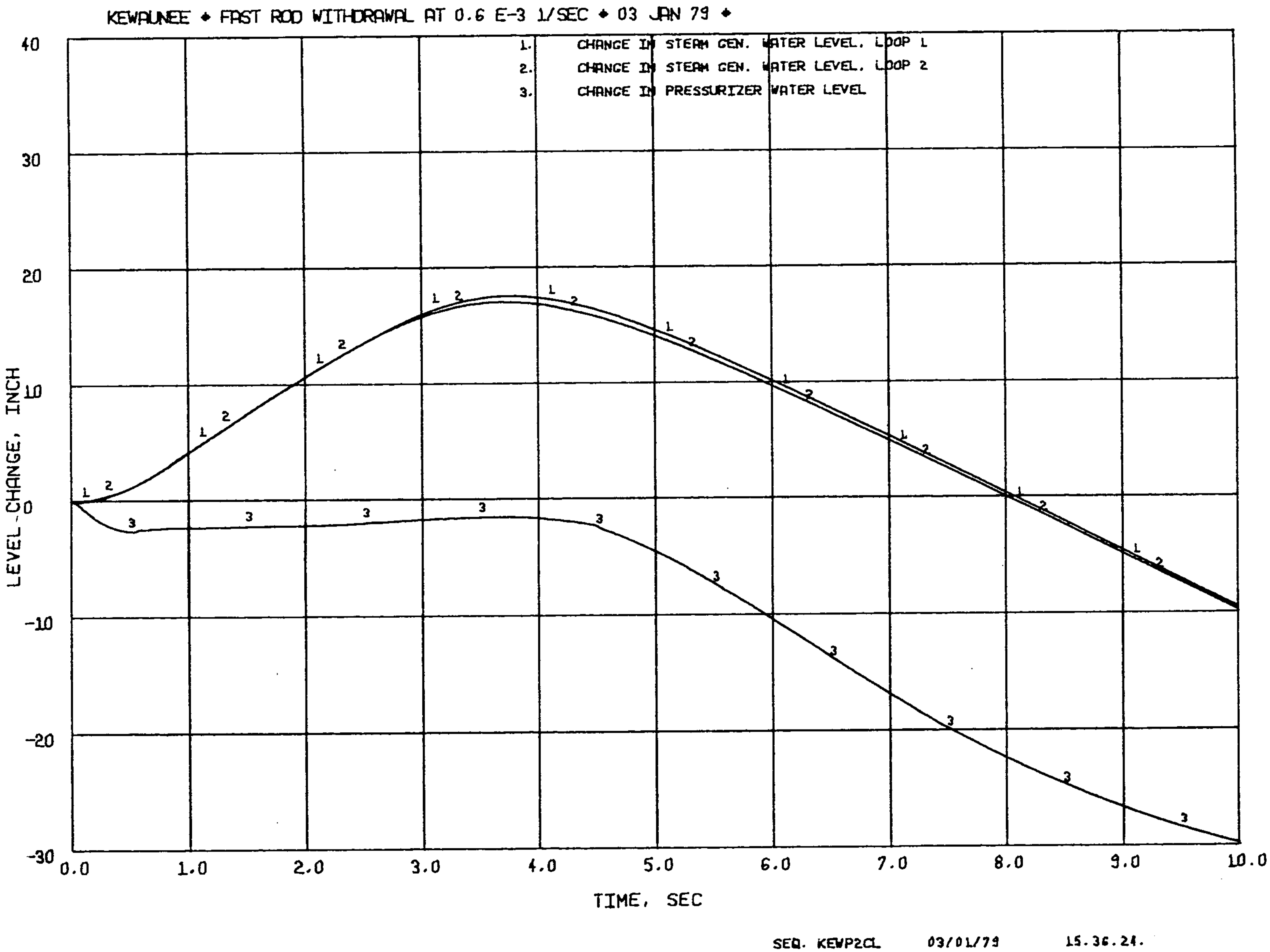

FIGURE 3.5 Level Changes in Pressurizer and Steam Generators for Fast Control Rod Withdrawal

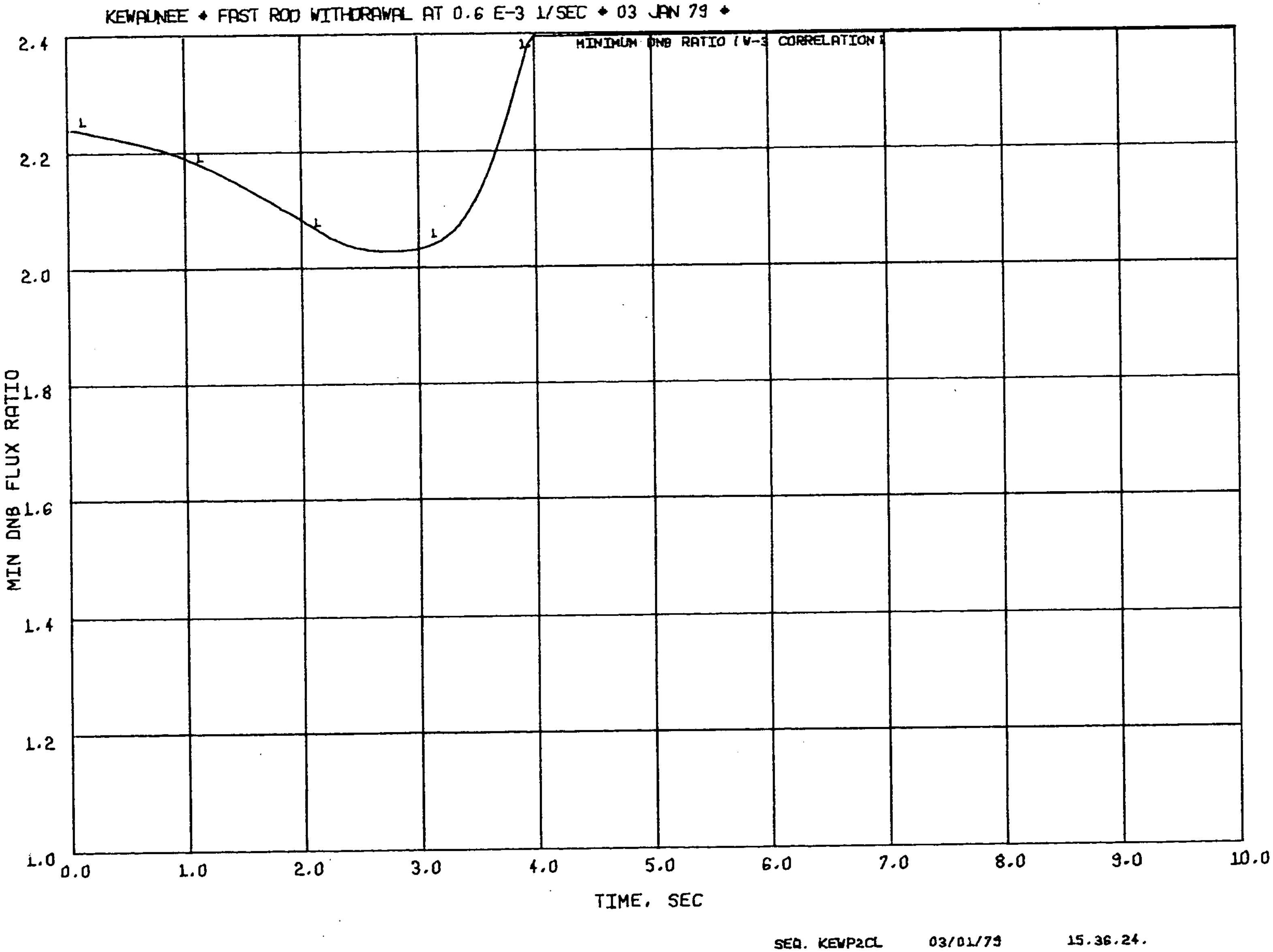

FIGURE 3.6 Minimum DNB Flux Ratio for Fast Control Rod Withdrawal

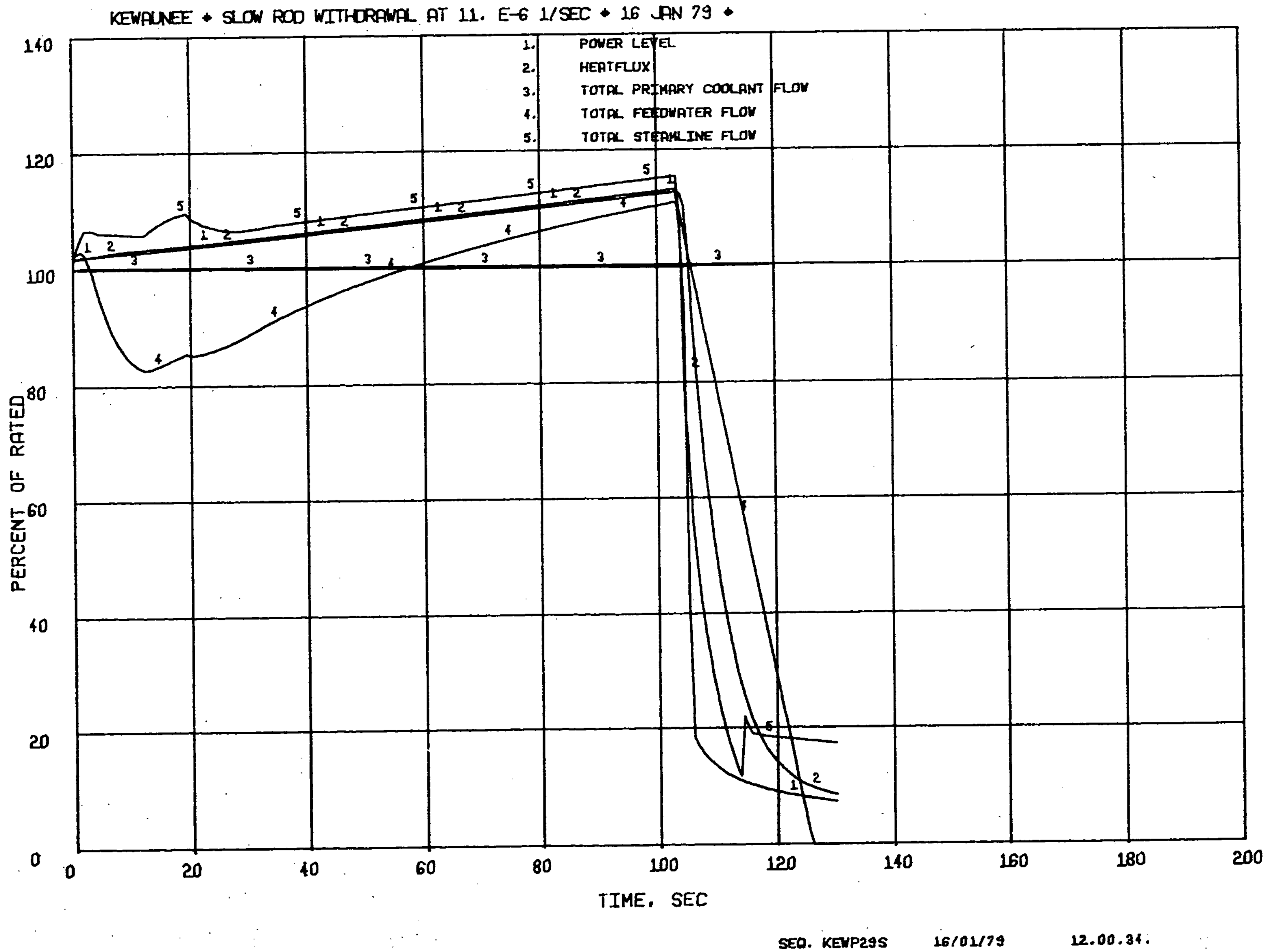

FIGURE 3.7 Power, Heatflux, and System Flows for Slow Control Rod Withdrawal

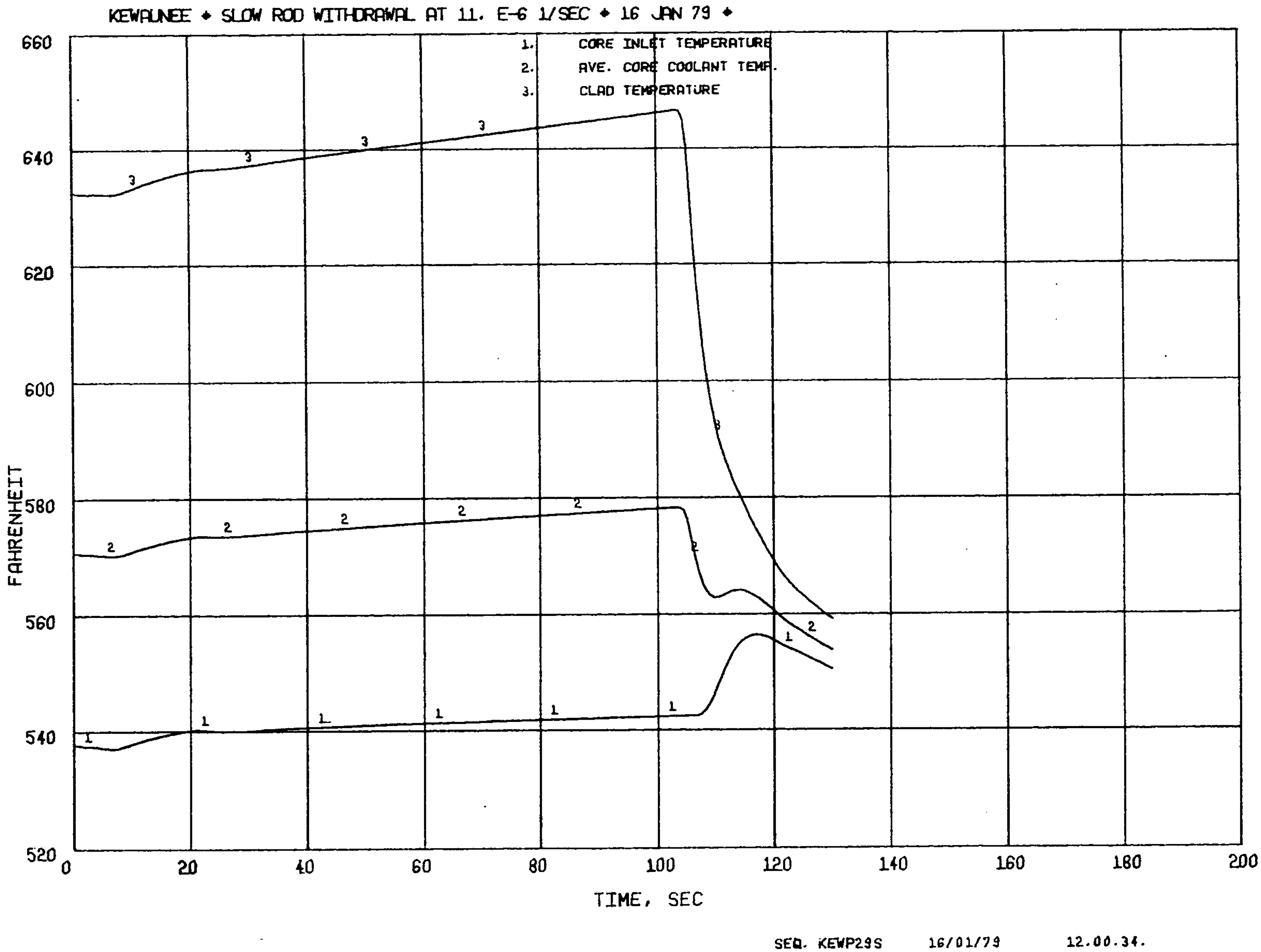

FIGURE 3.8 Core Temperature Response for Slow Control Rod Withdrawal

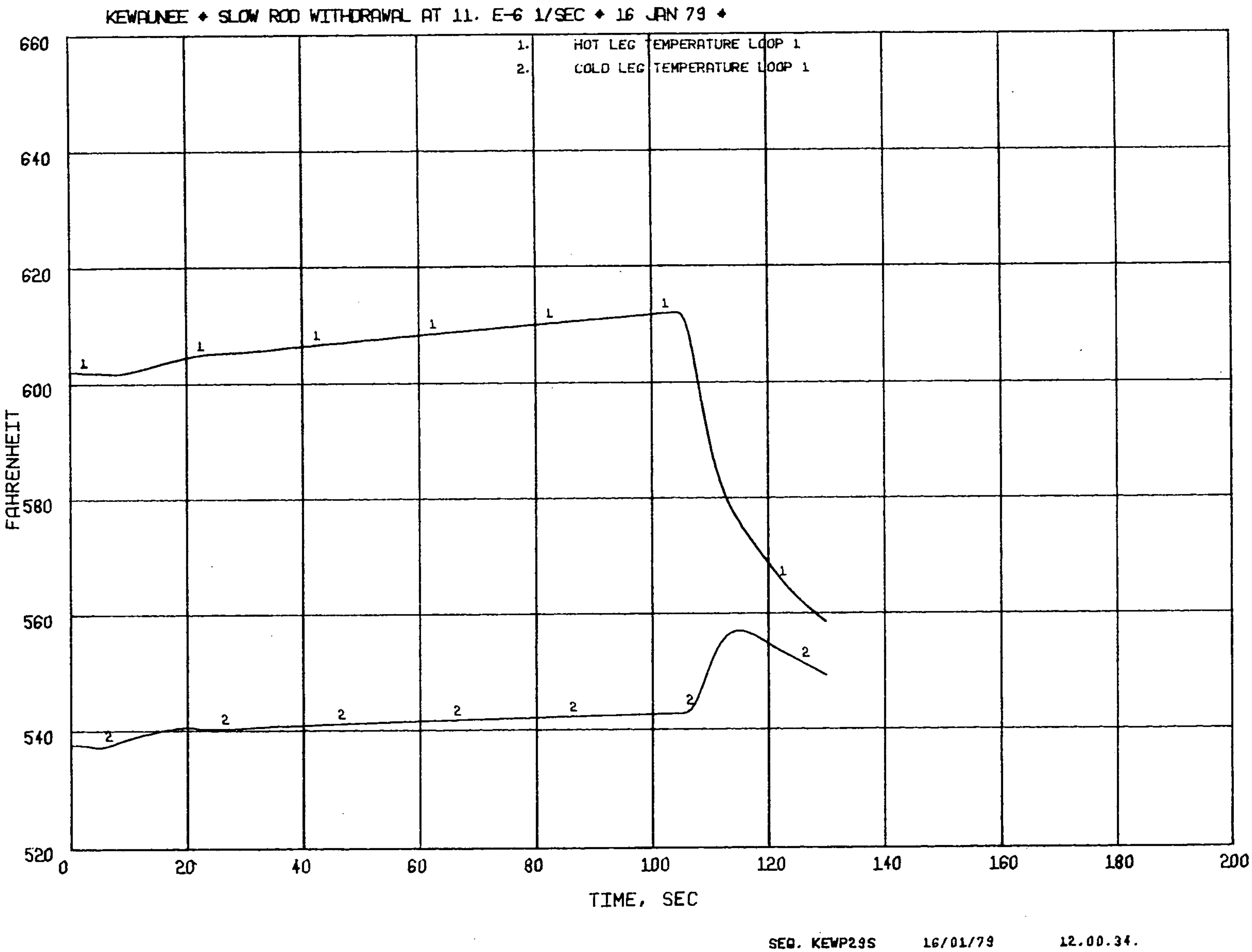

FIGURE 3.9 Primary Loop Temperature Response for Slow Control Rod Withdrawal

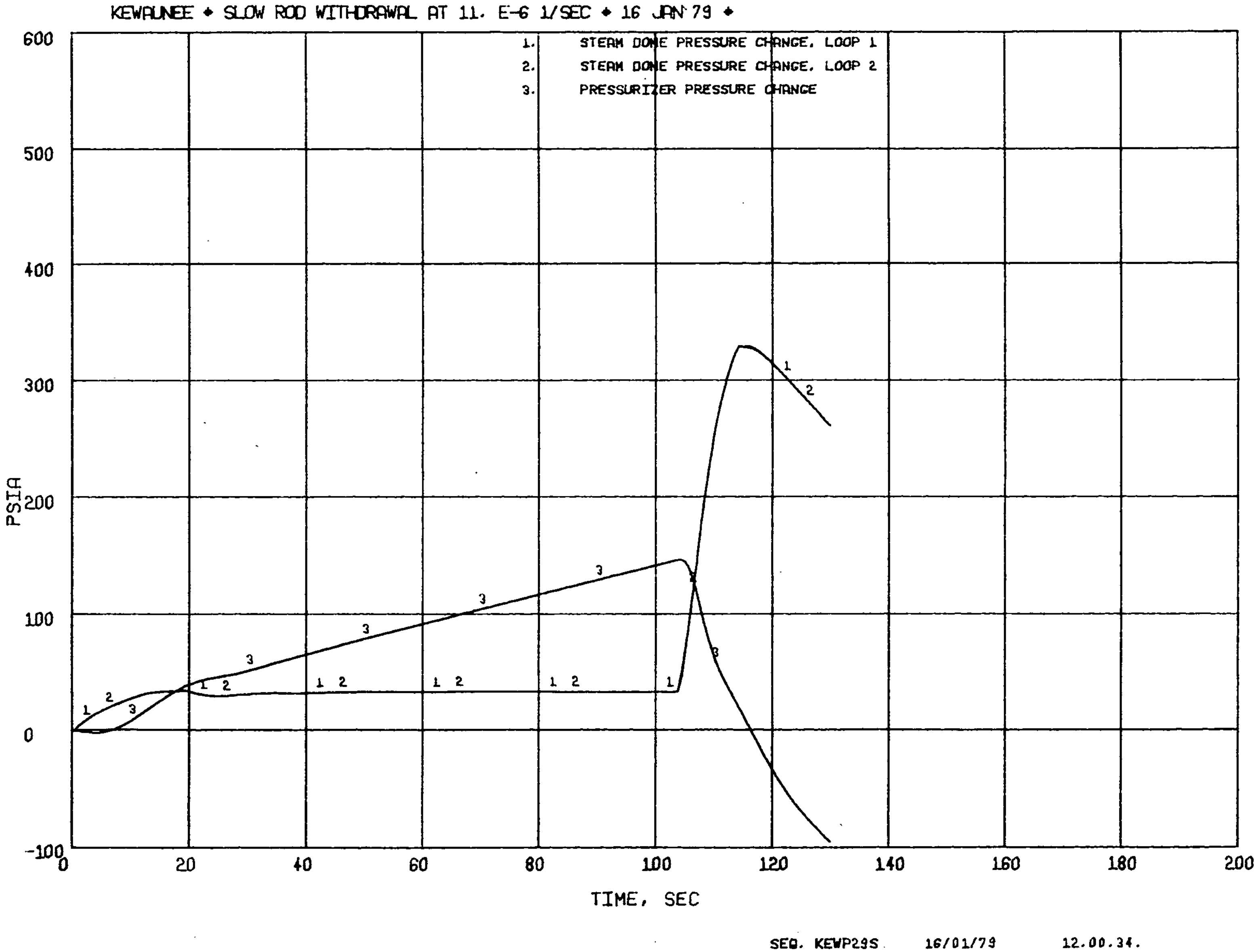

FIGURE 3.10 Pressure Changes in Pressurizer and Steam Generators for Slow Control Rod Withdrawal

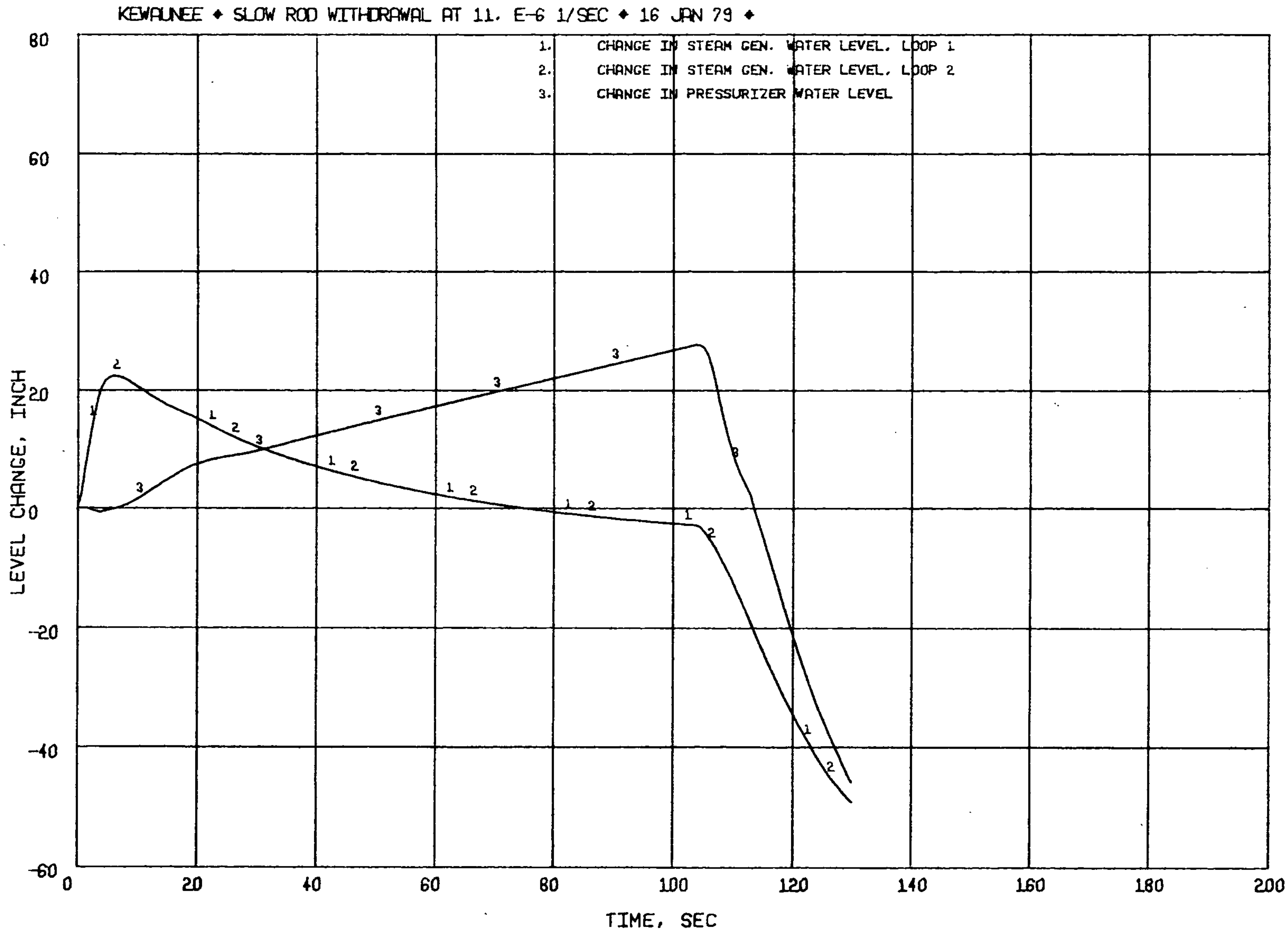

FIGURE 3.11 Level Changes in Pressurizer and Steam Generators for Slow Control Rod Withdrawal

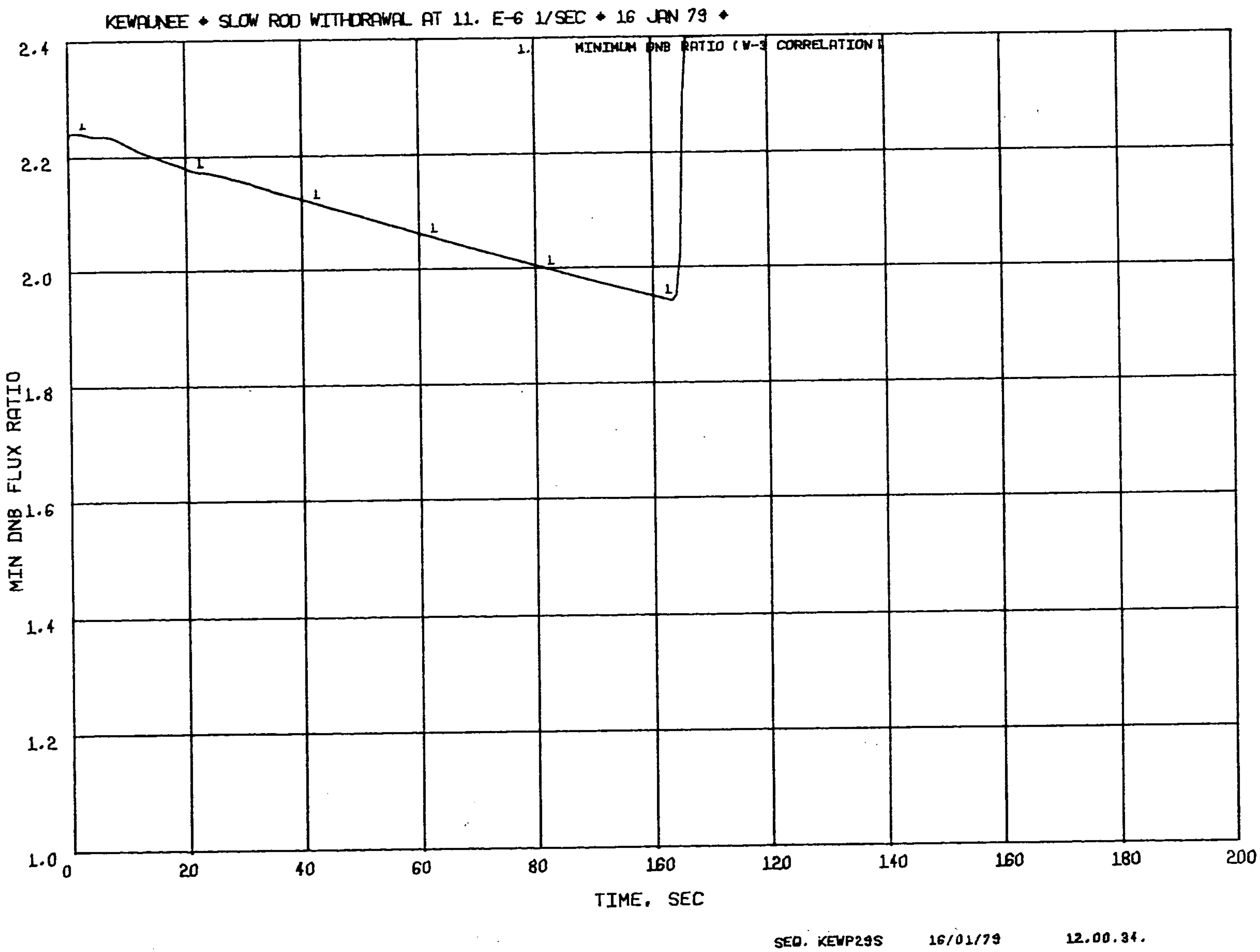

FIGURE 3.12 Minimum DNB Flux Ratio for Slow Control Rod Withdrawal

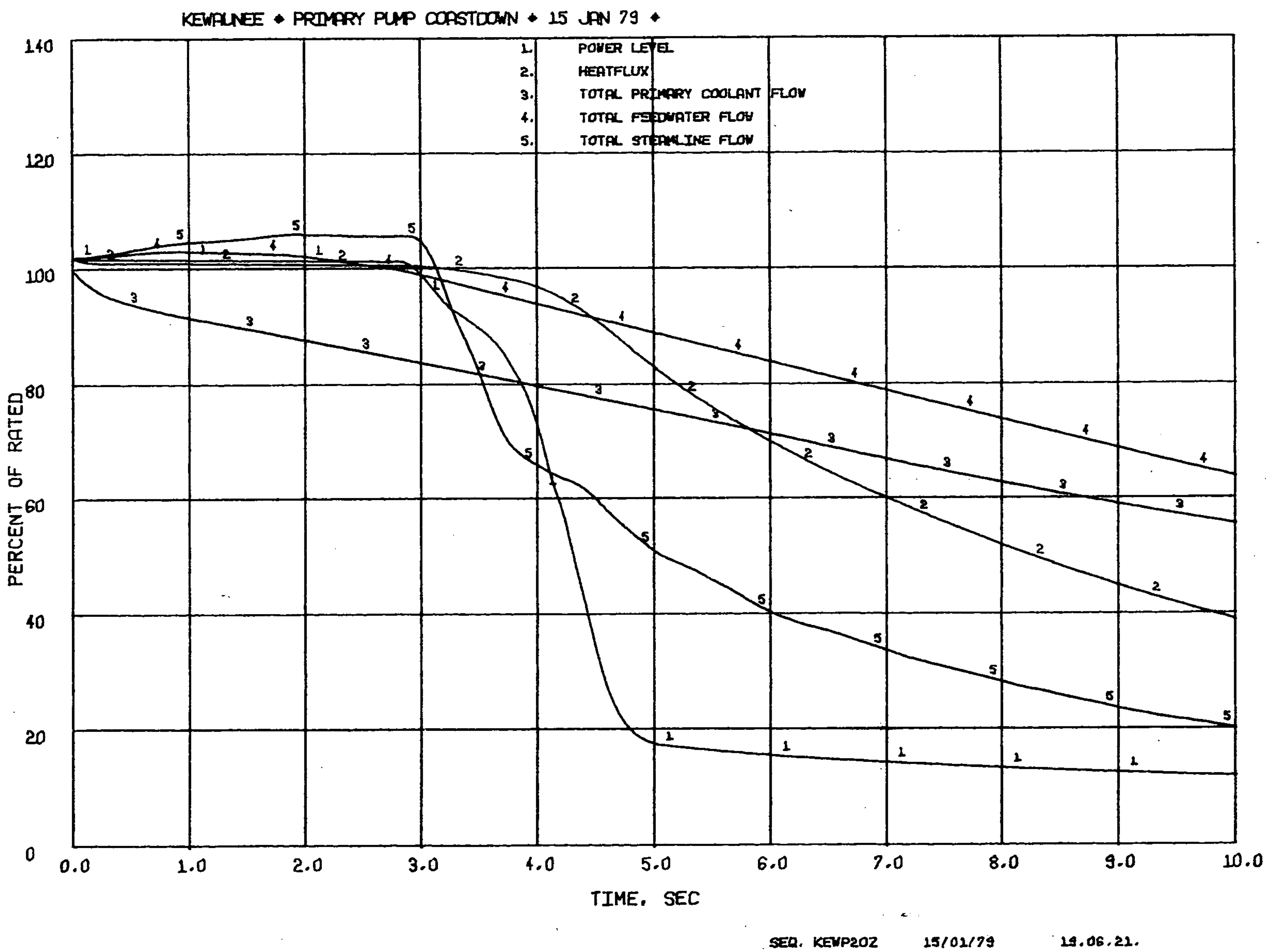

FIGURE 3.13 Power, Heatflux, and System Flows for Coolant Pump Trip

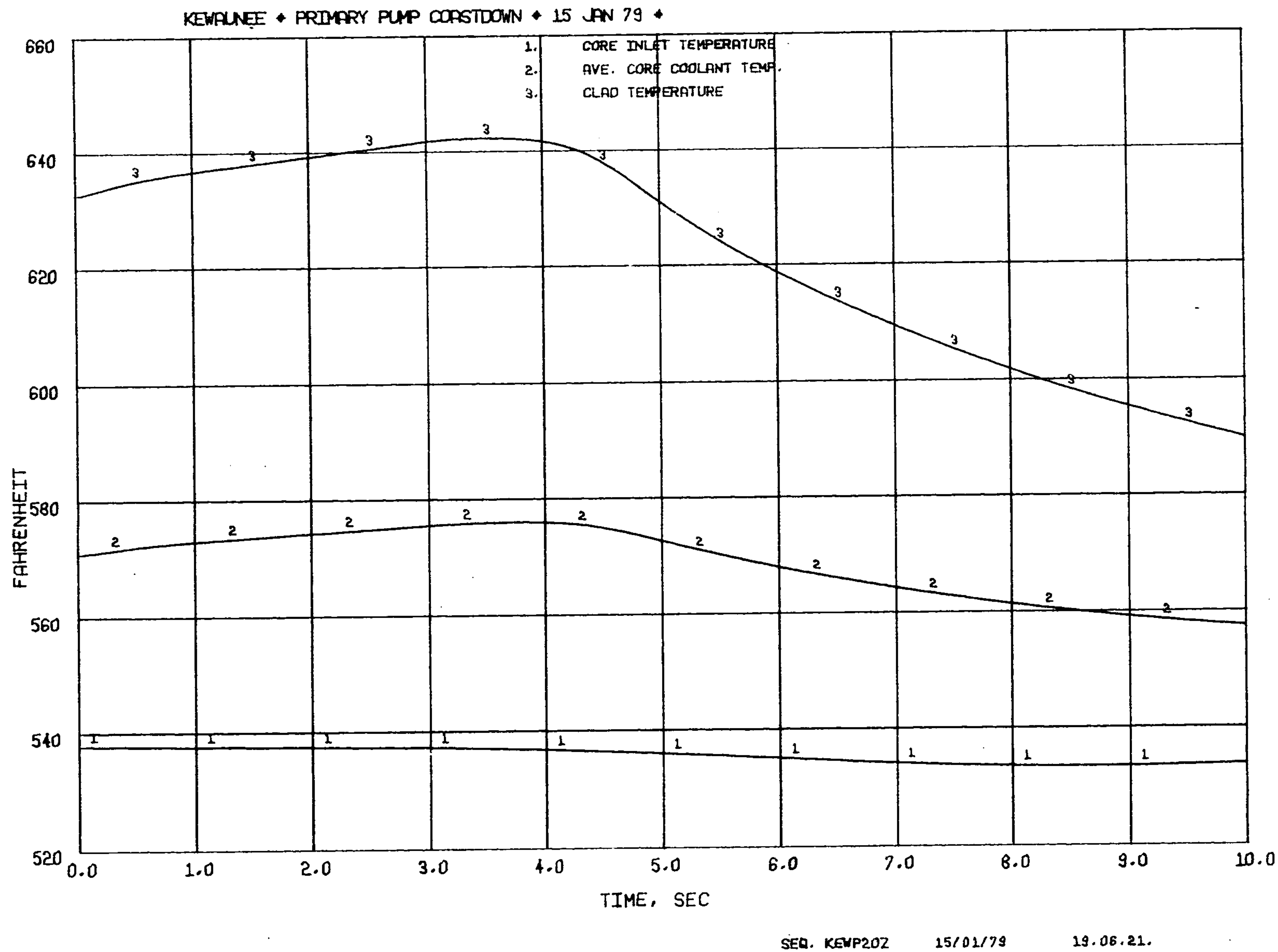

FIGURE 3.14 Core Temperature Response for Coolant Pump Trip

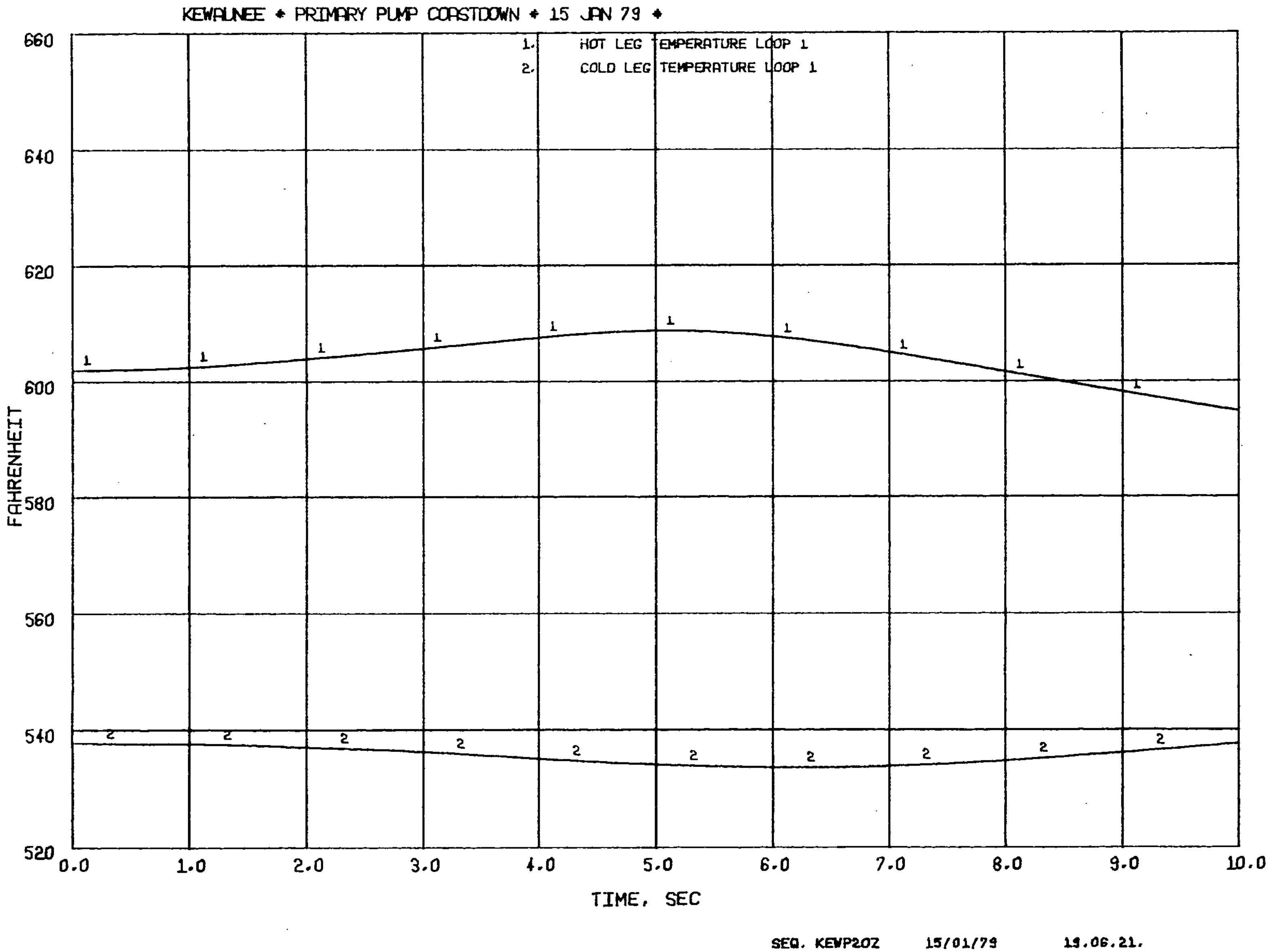

FIGURE 3.15 Primary Loop Temperature Response for Coolant Pump Trip

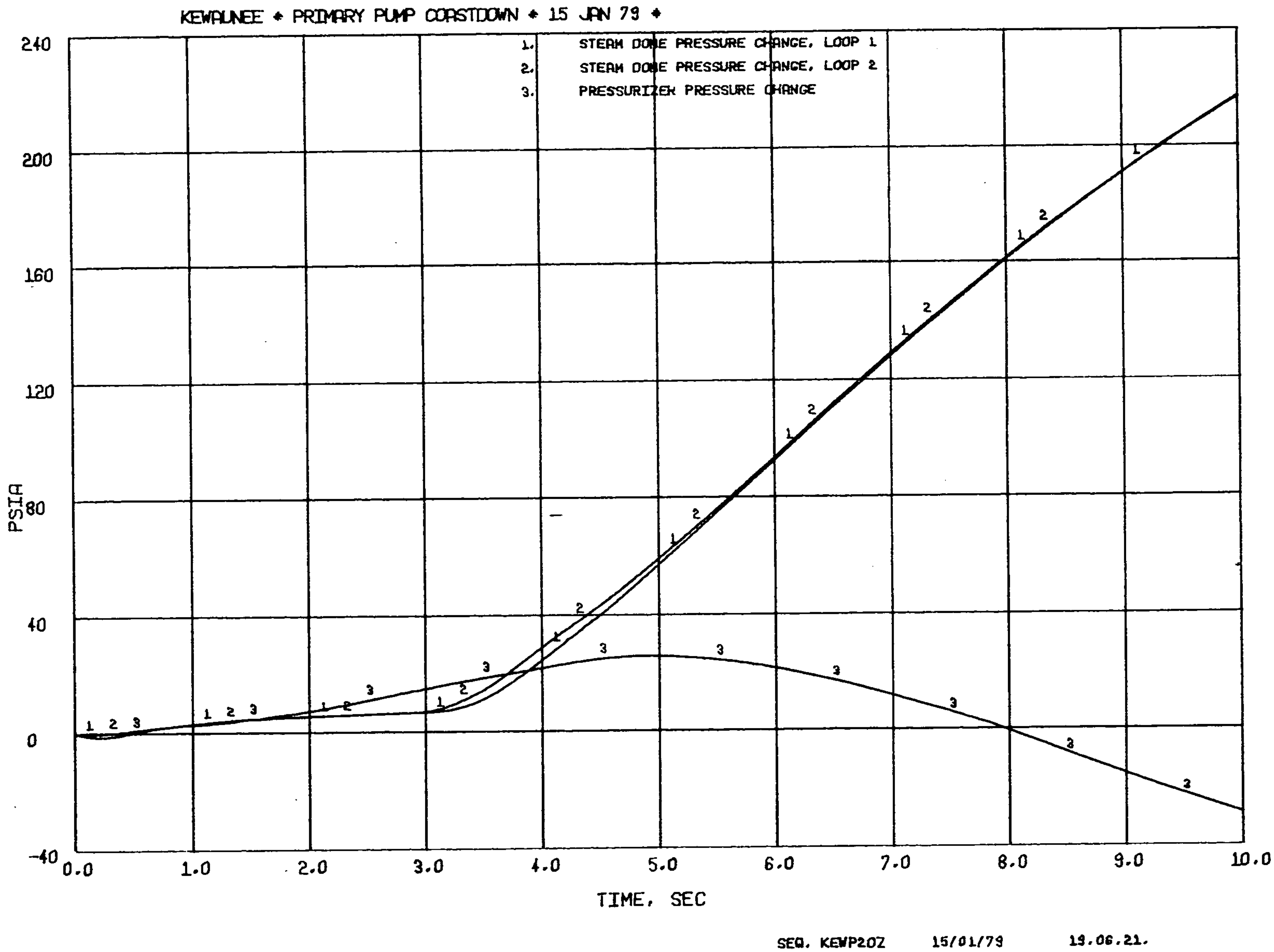

FIGURE 3.16 Pressure Changes in Pressurizer and Steam Generators for Coolant Pump Trip

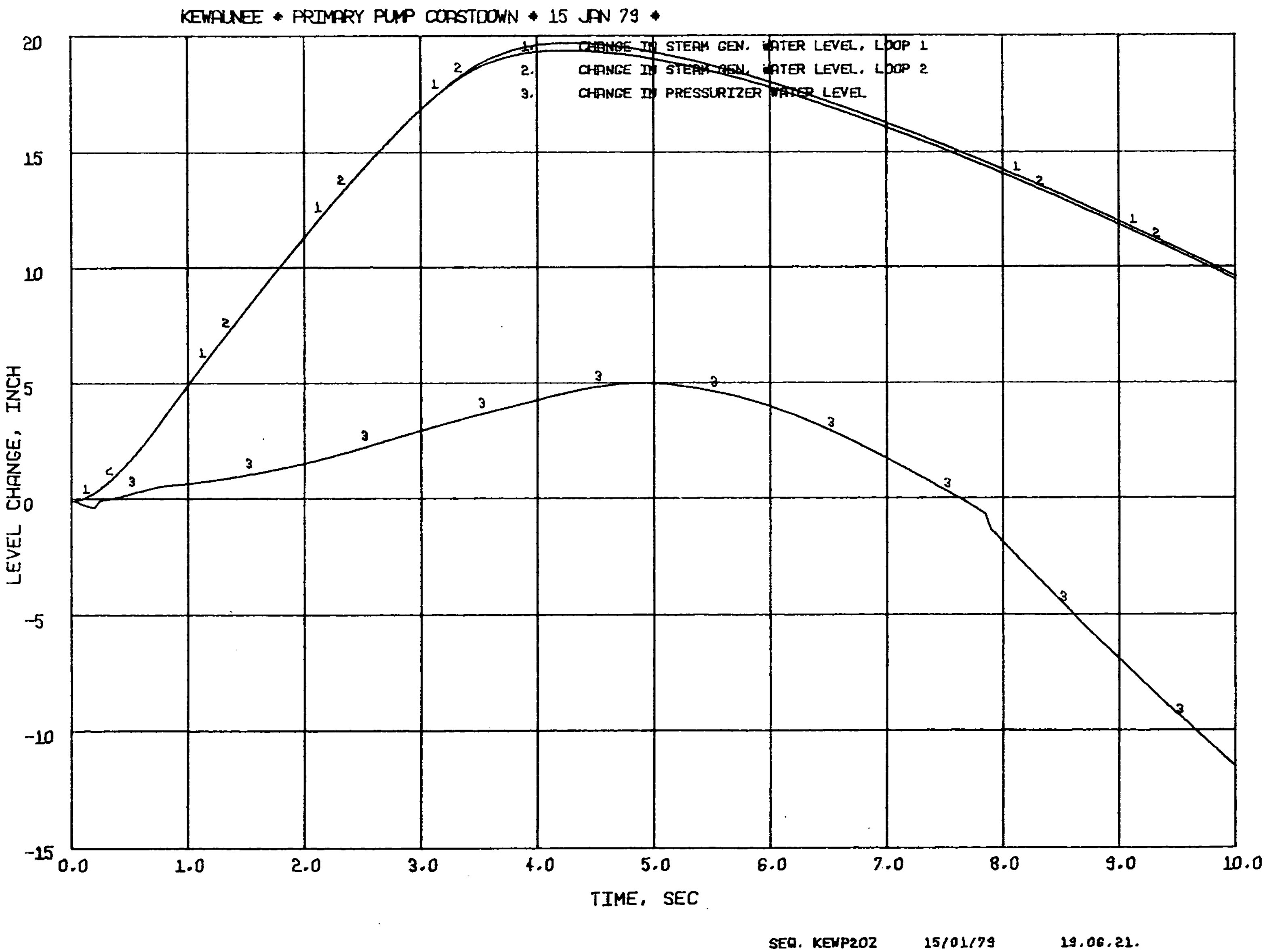

FIGURE 3.17 Level Changes in Pressurizer and Steam Generators for Coolant Pump Trip

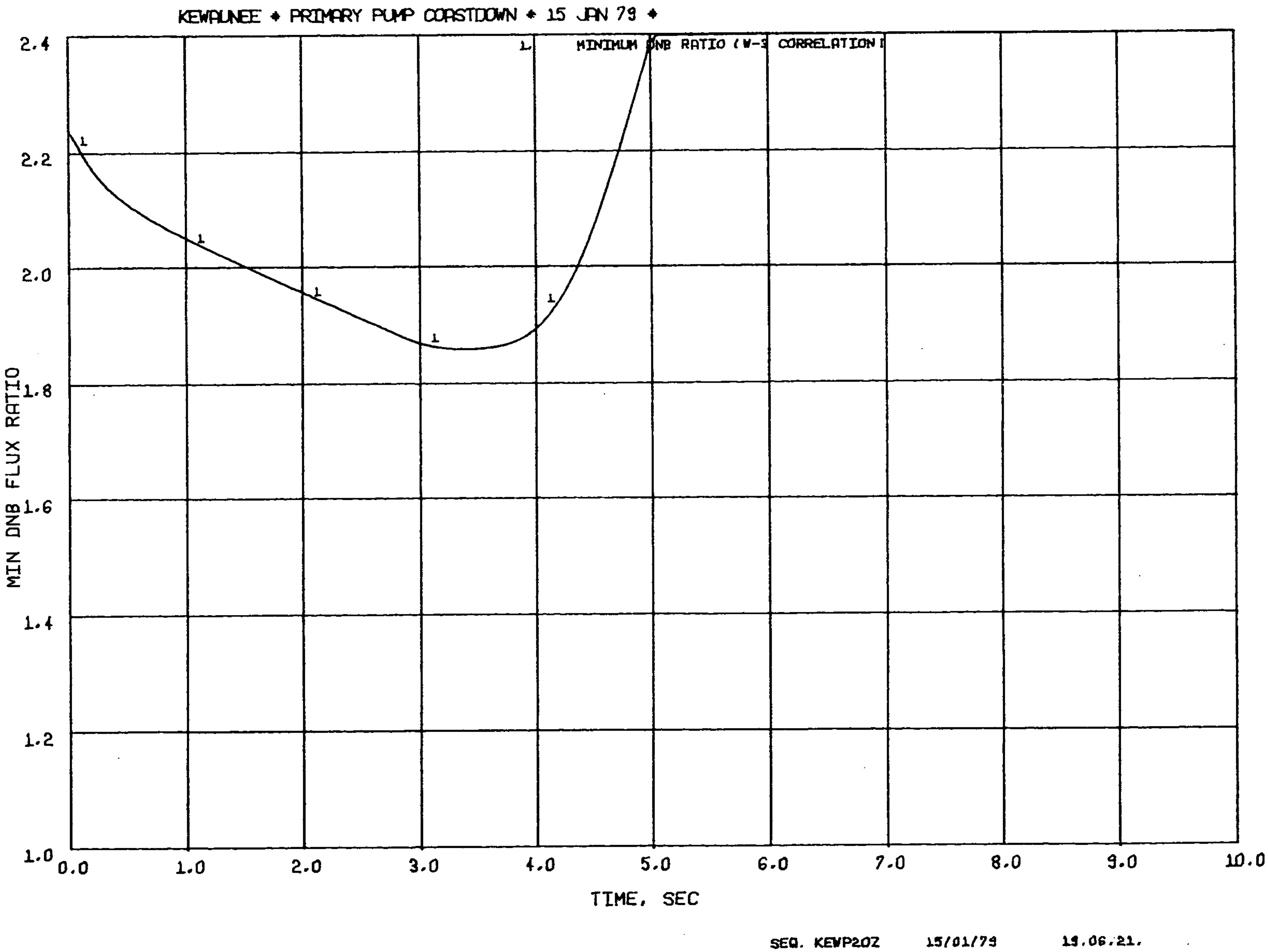

FIGURE 3.18 Minimum DNB Flux Ratio for Coolant Pump Trip

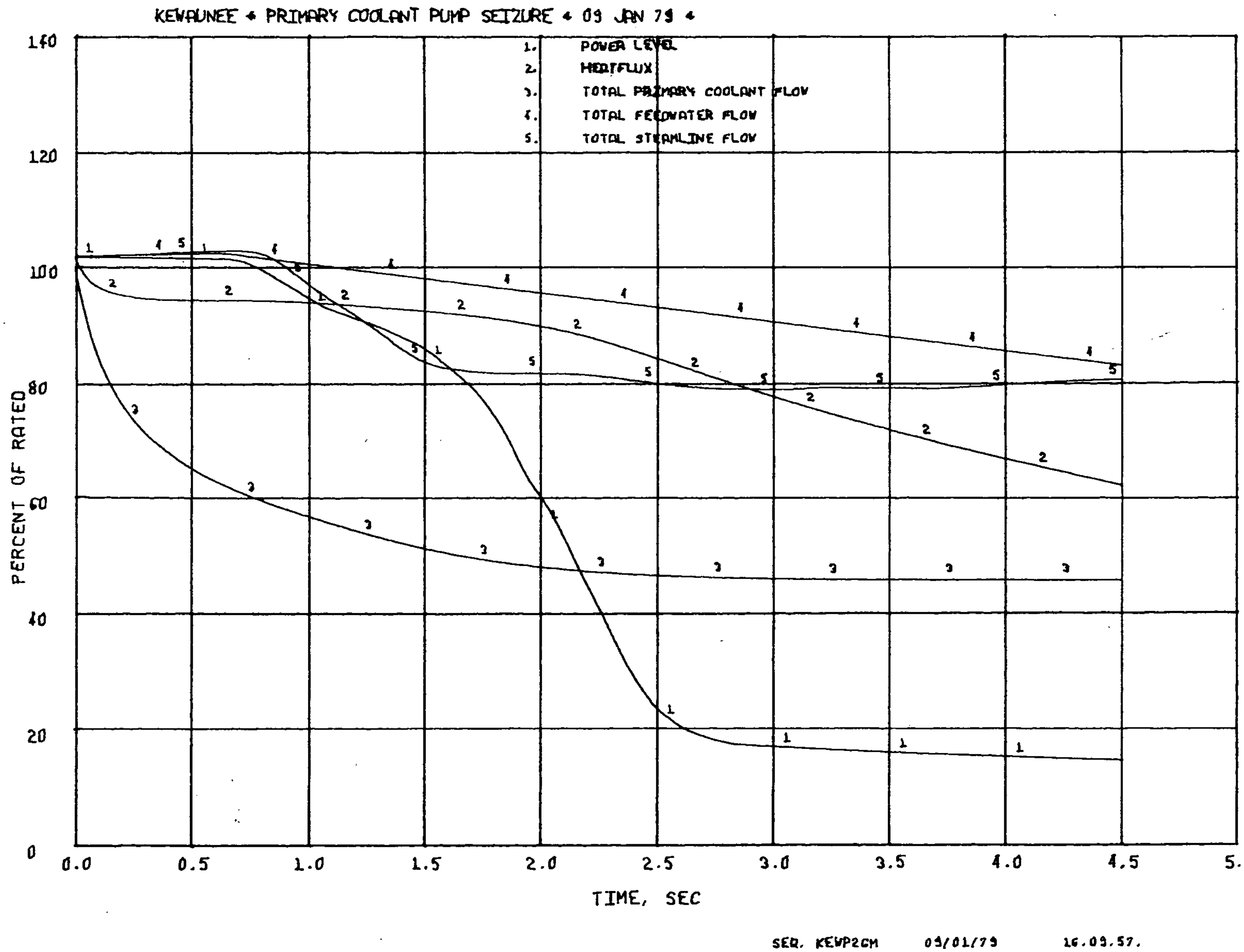

FIGURE 3.19 Power, Heatflux and System Flows for Coolant Pump Seizure

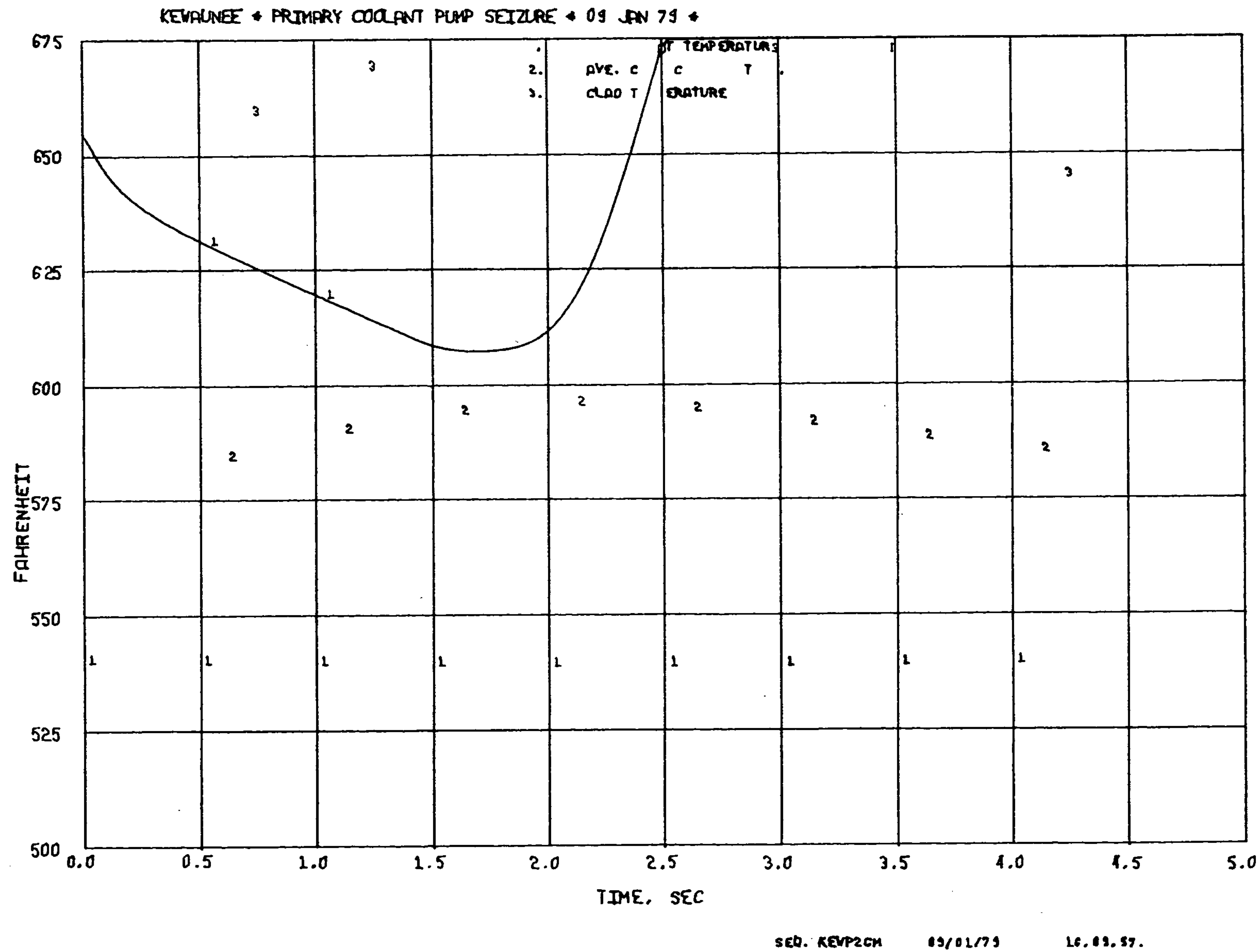

FIGURE 3.20 Core Temperature Response for Coolant Pump Seizure

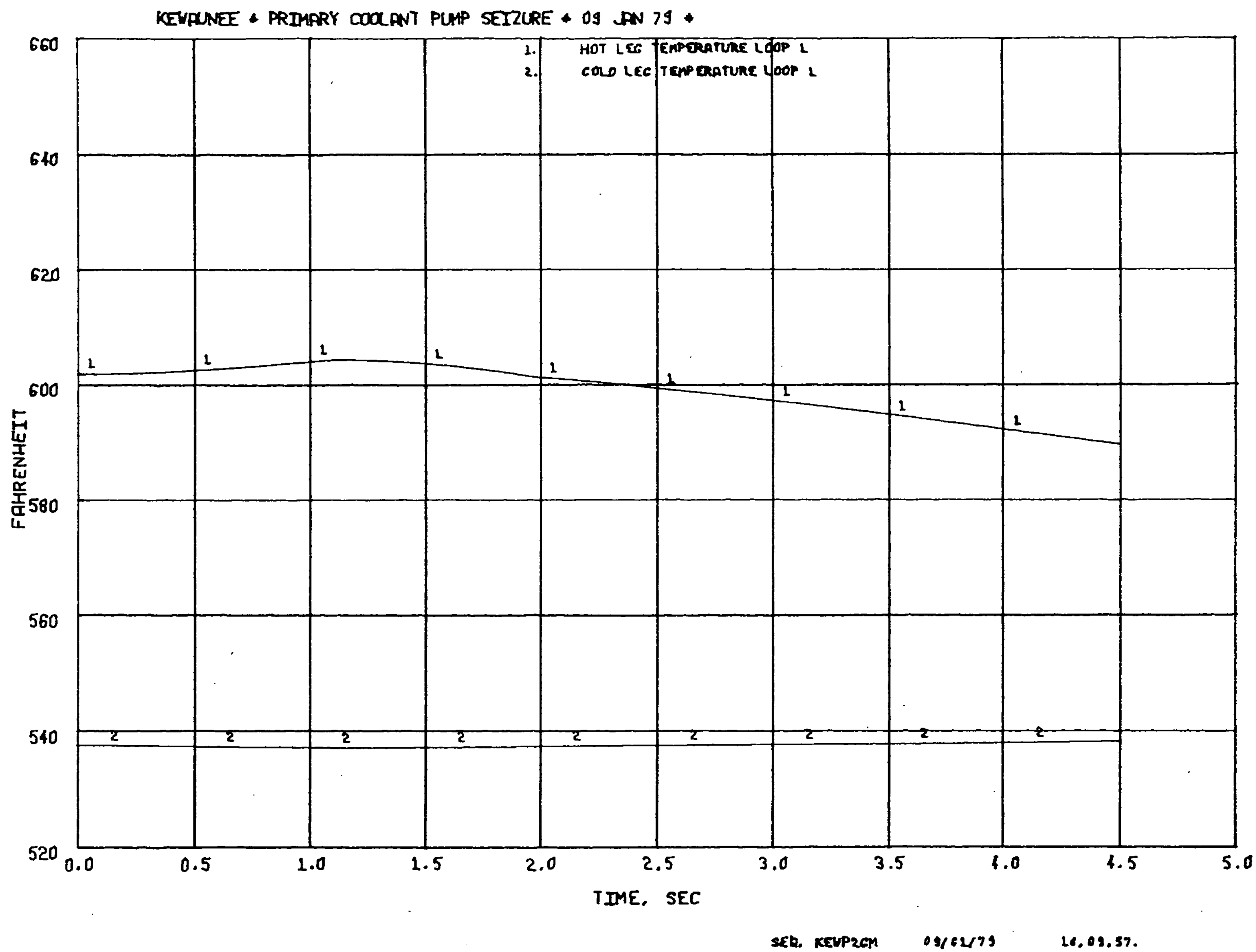

FIGURE 3.21 Primary Loop Temperature Response for Coolant Pump Seizure

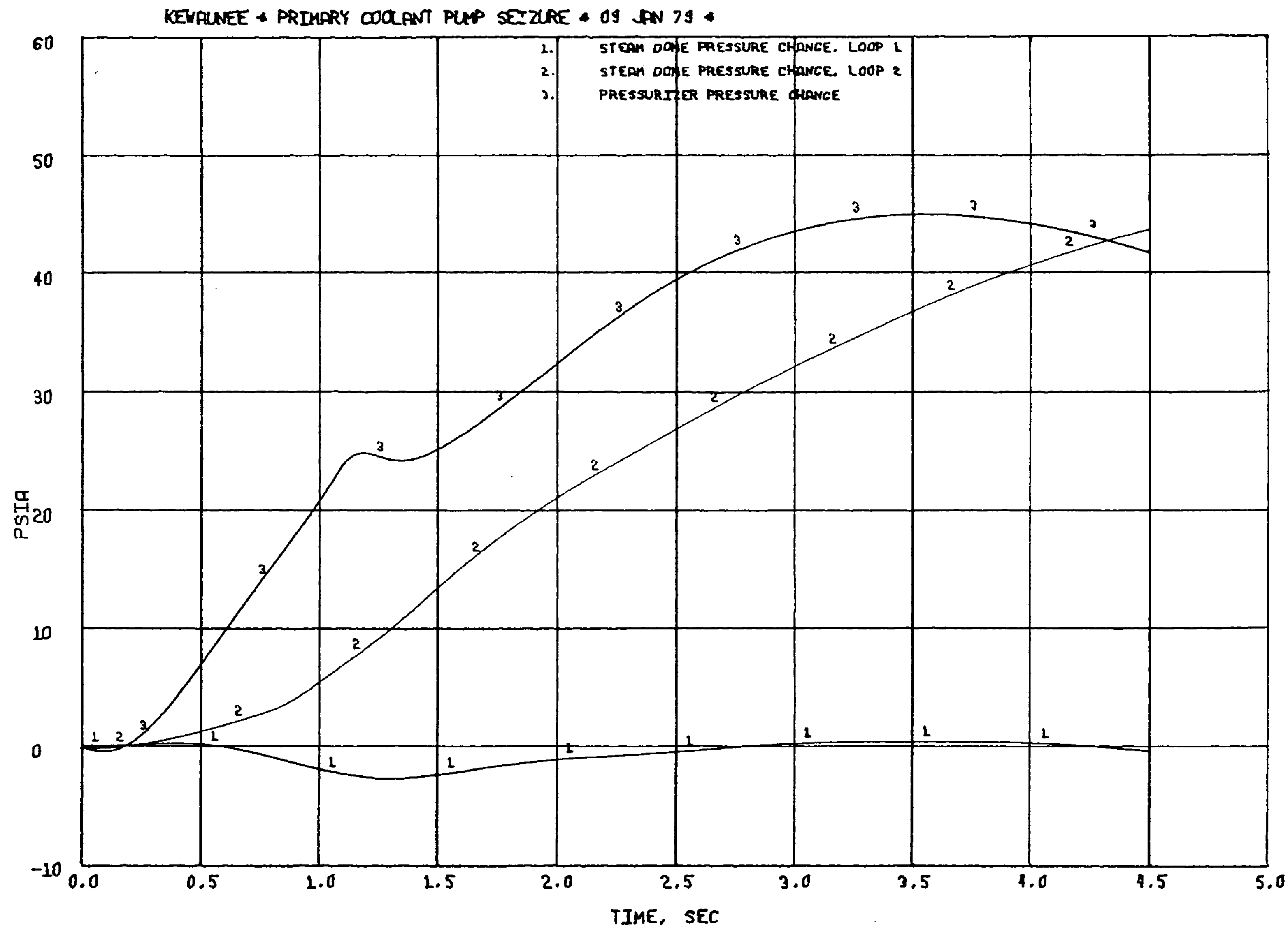

FIGURE 3.22 Pressure Changes in Pressurizer and Steam Generators for Coolant Pump Seizure

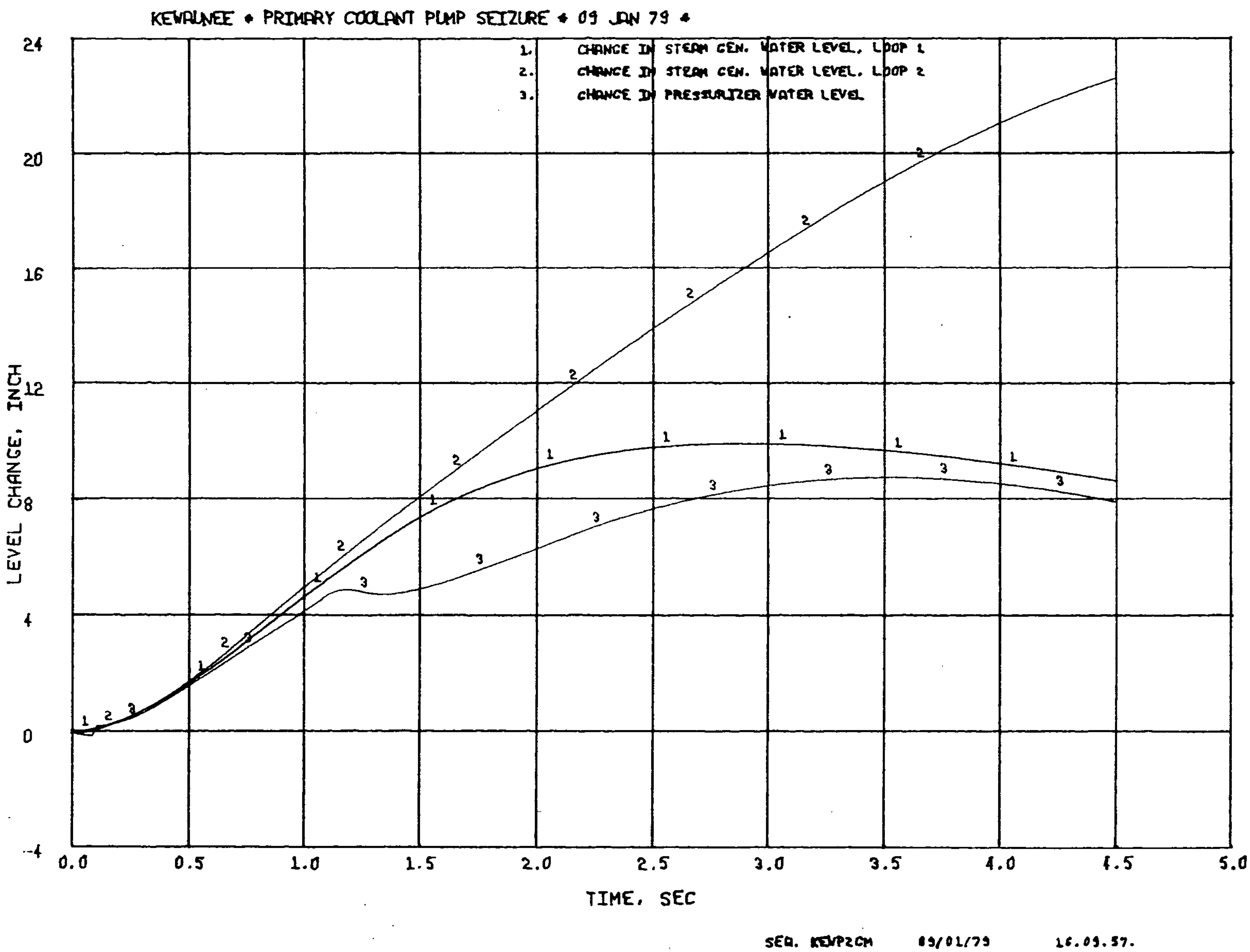

FIGURE 3.23 Level Changes in Pressurizer and Steam Generators for Coolant Pump Seizure

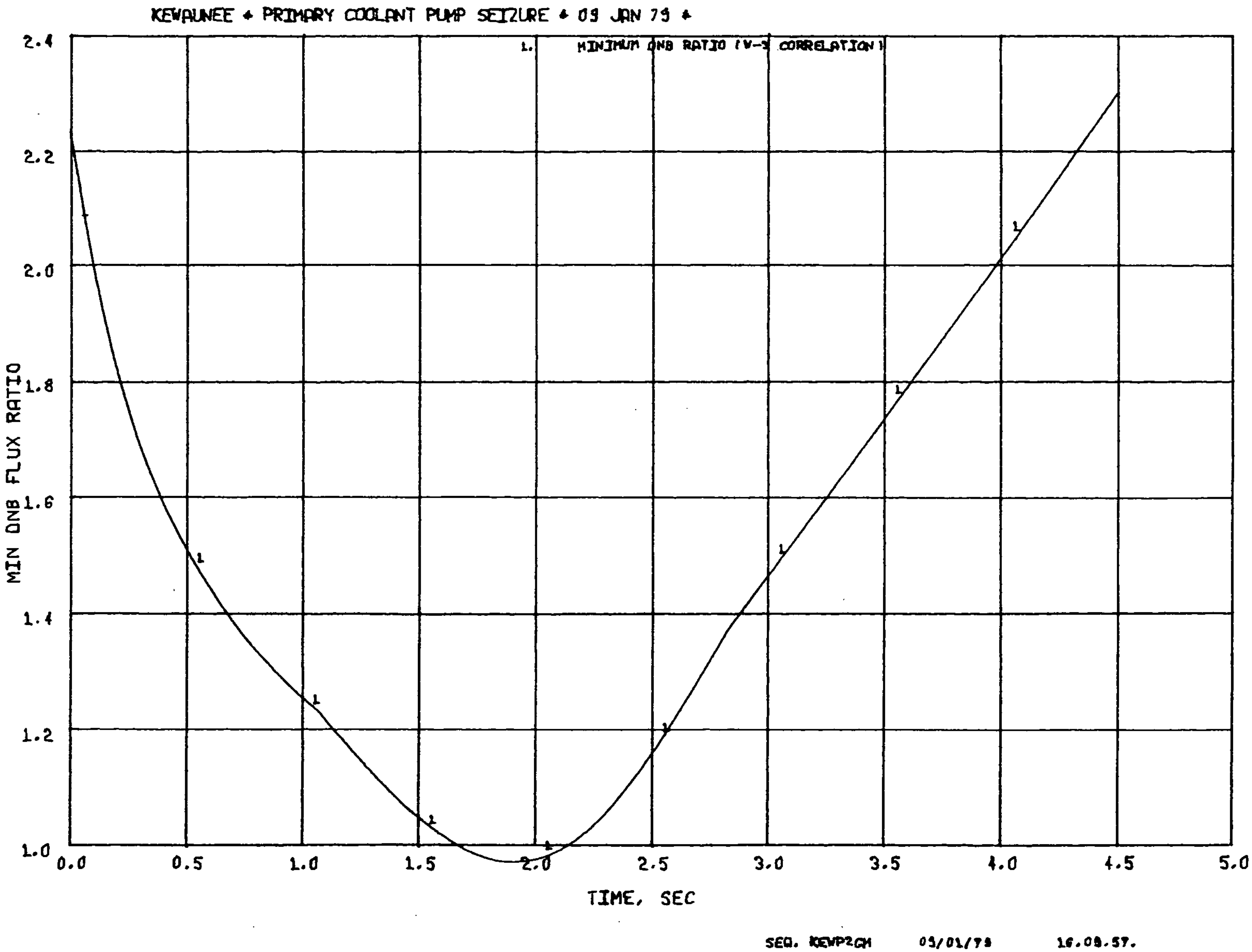

FIGURE 3.24 Minimum DNB Flux Ratio for Coolant Pump Seizure

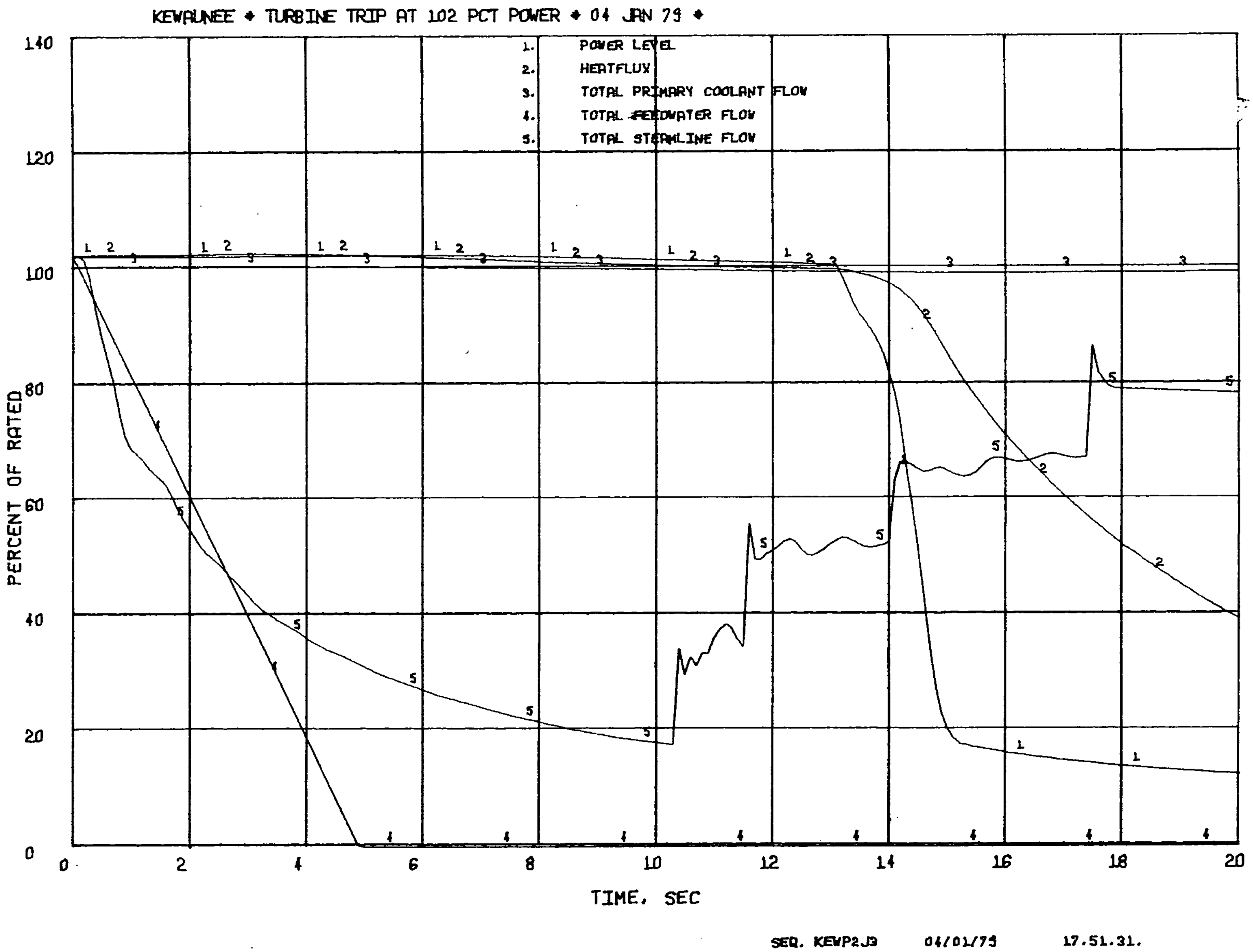

FIGURE 3.25 Power, Heatflux, and System Flows for Turbine Trip

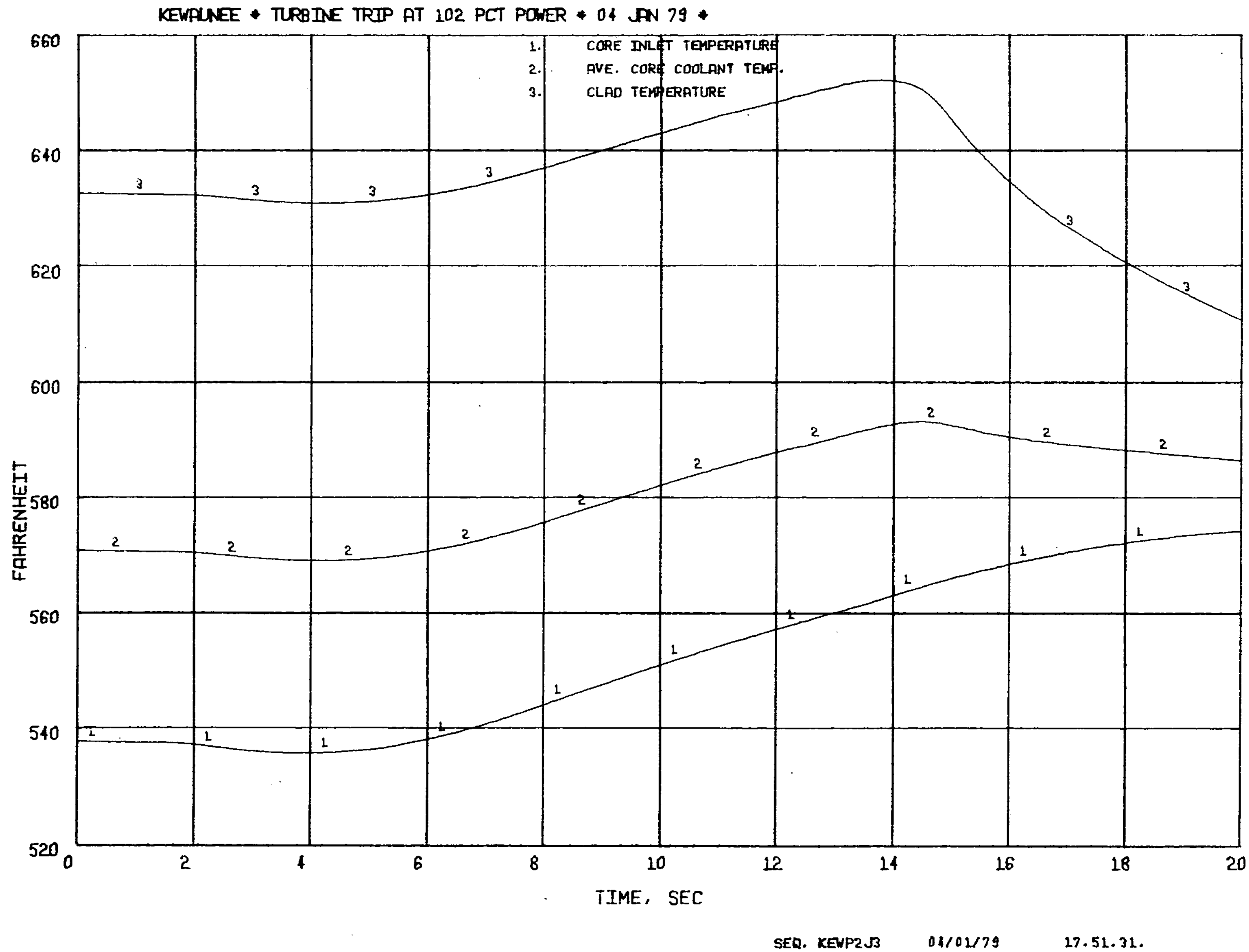

FIGURE 3.26 Core Temperature Response for Turbine Trip

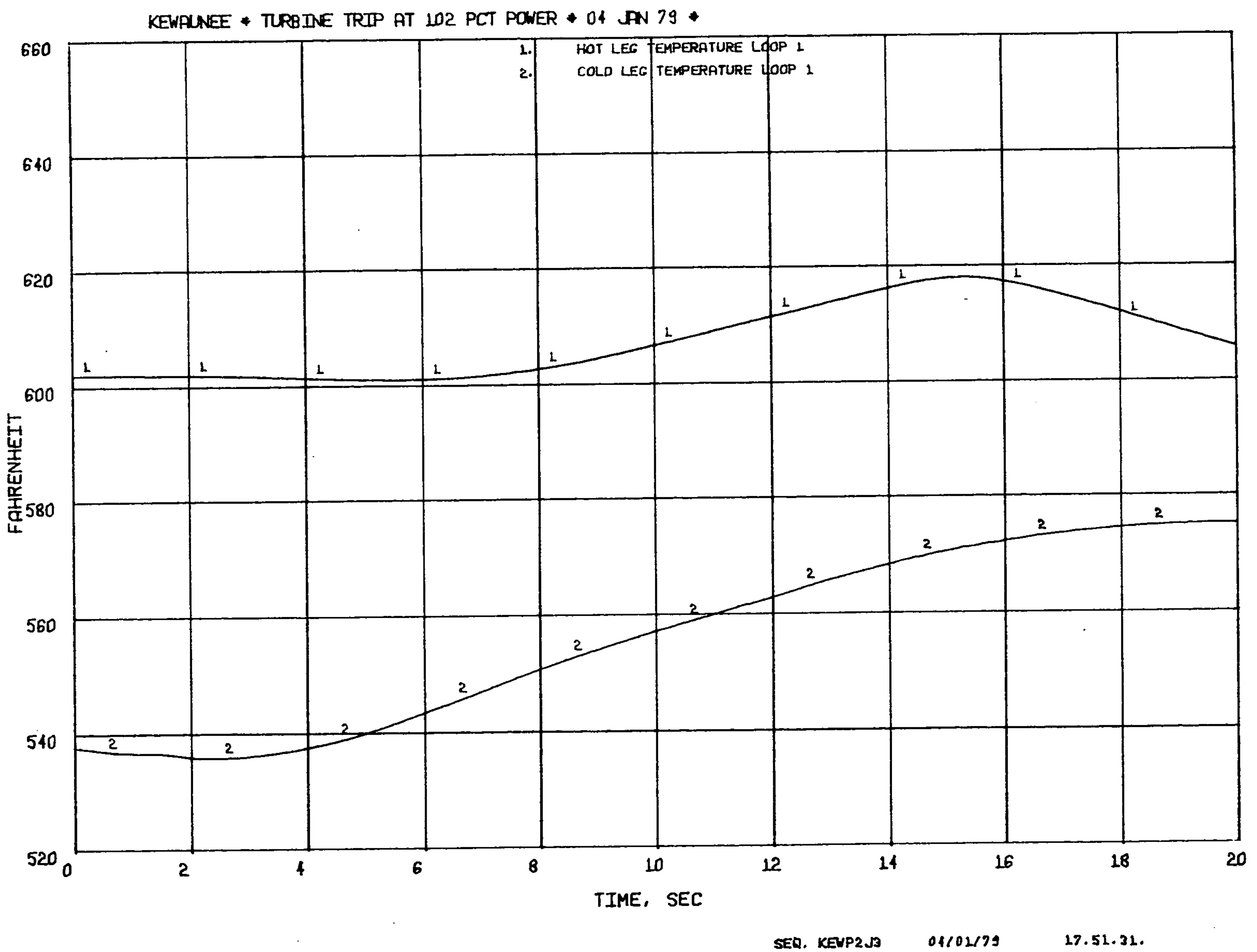

FIGURE 3.27 Primary Loop Temperature Response for Turbine Trip

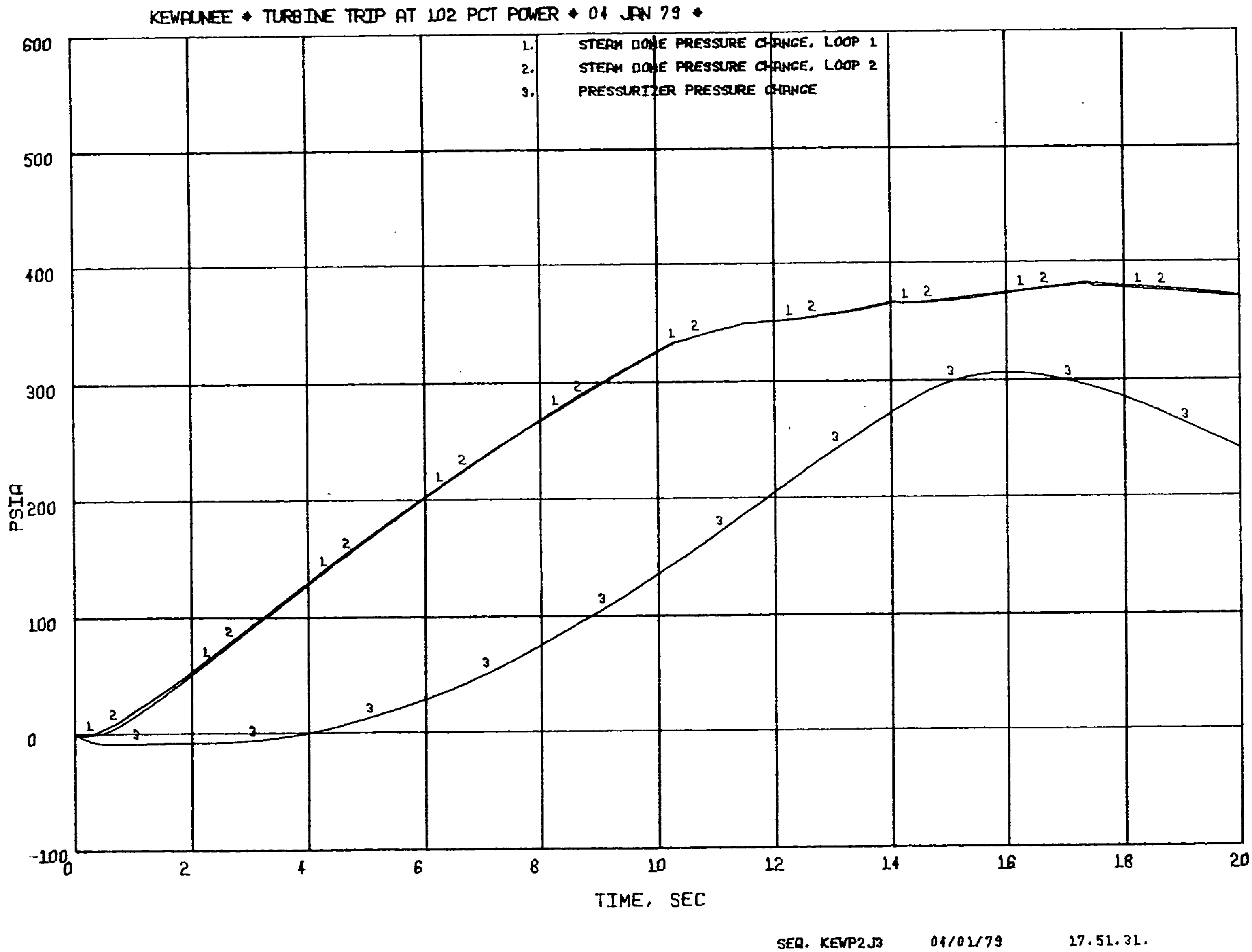

FIGURE 3.28 Pressure Changes in Pressurizer and Steam Generators for Turbine Trip

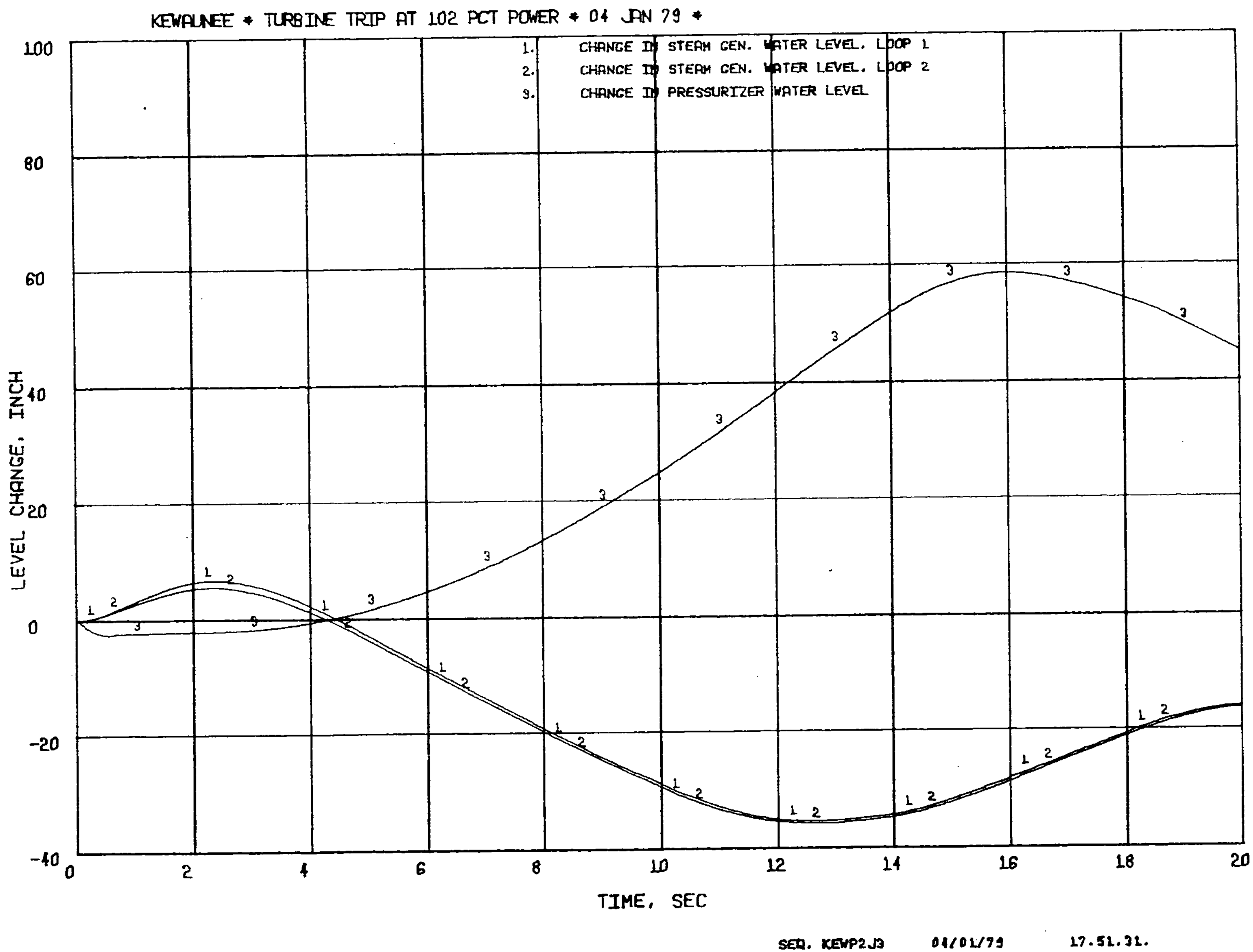

FIGURE 3.29 Level Changes in Pressurizer and Steam Generators for Turbine Trip

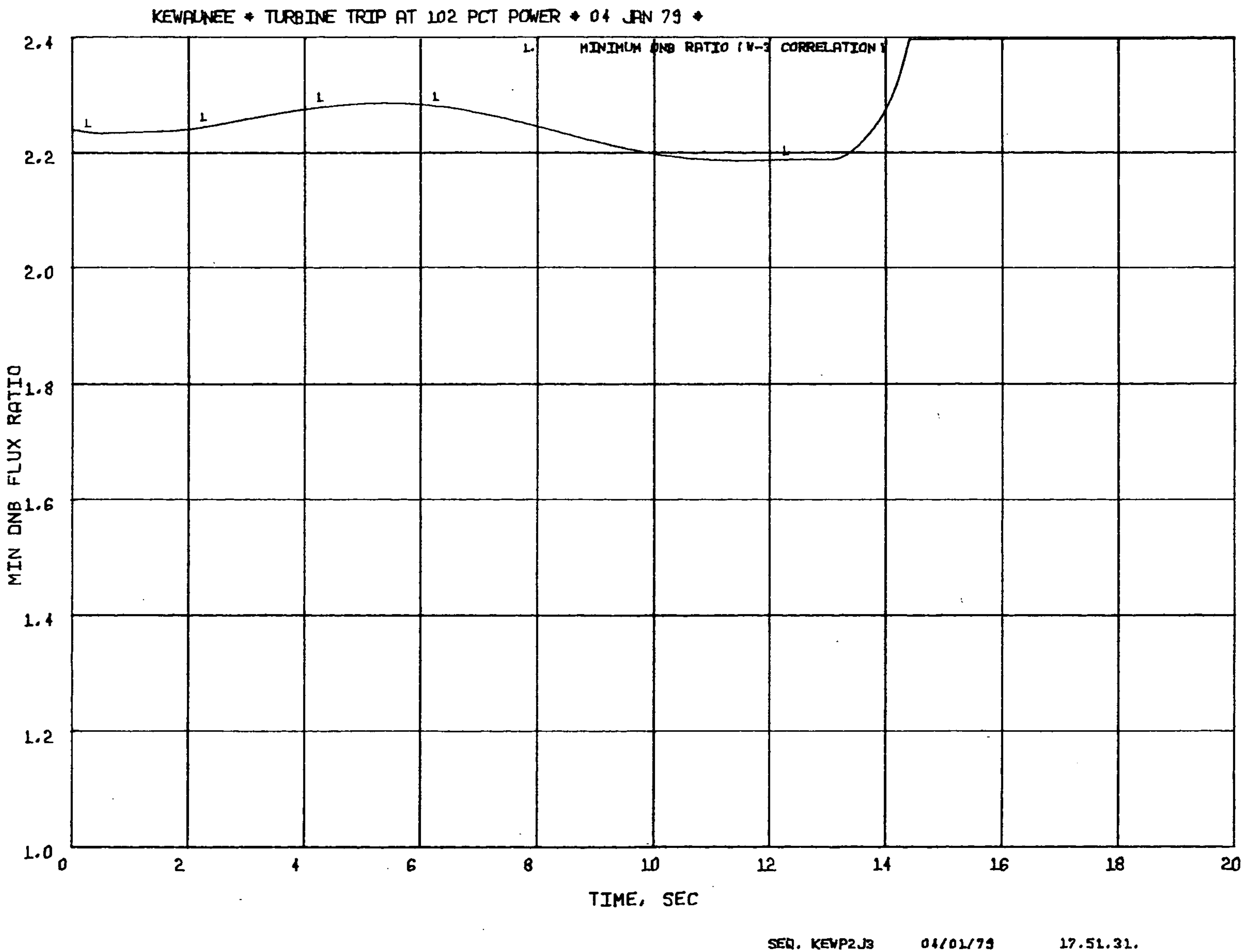

FIGURE 3.30 Minimum DNB Flux Ratio for Turbine Trip

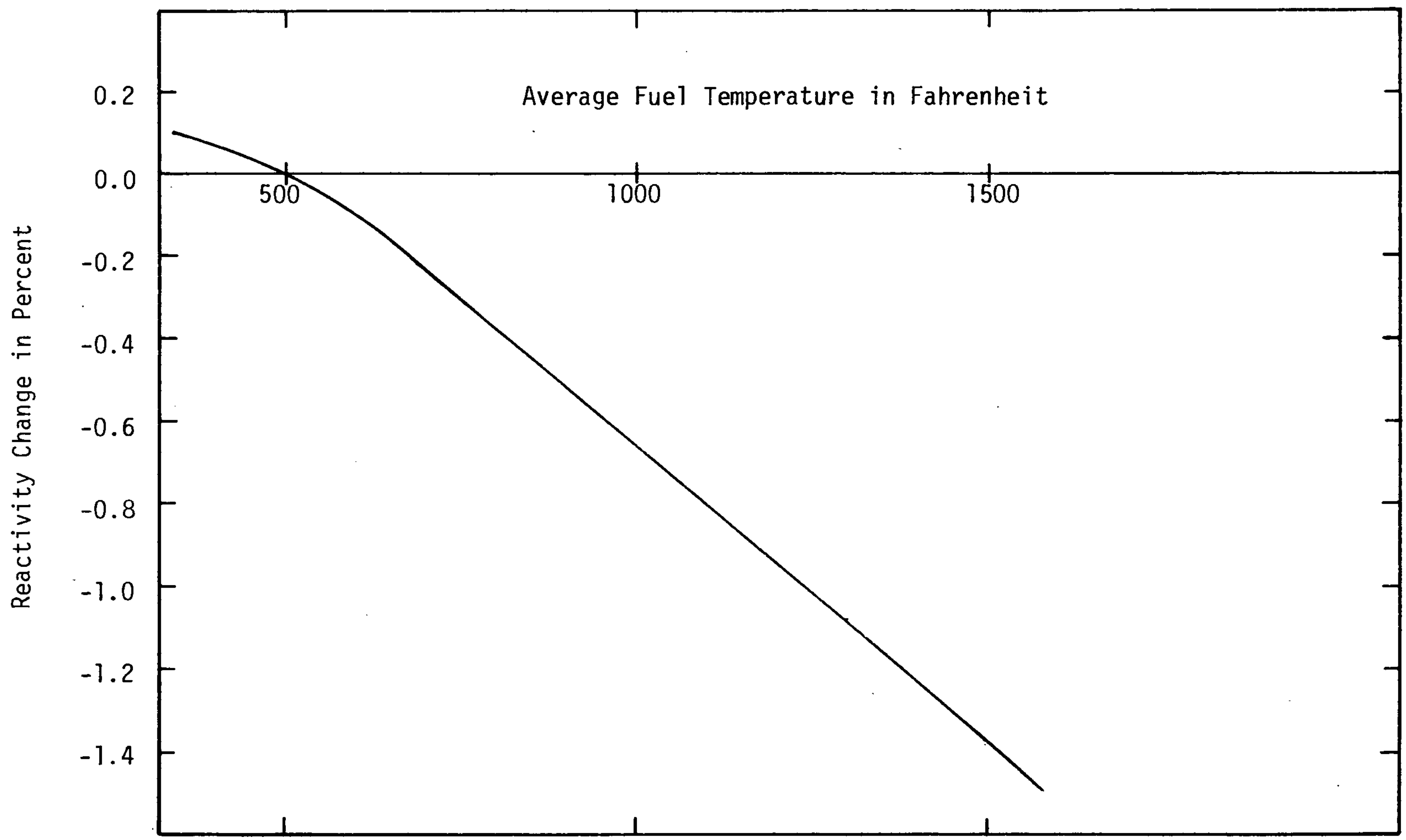

FIGURE 3.31 Doppler Feedback Reactivity for Steamline Break Transients

103.5

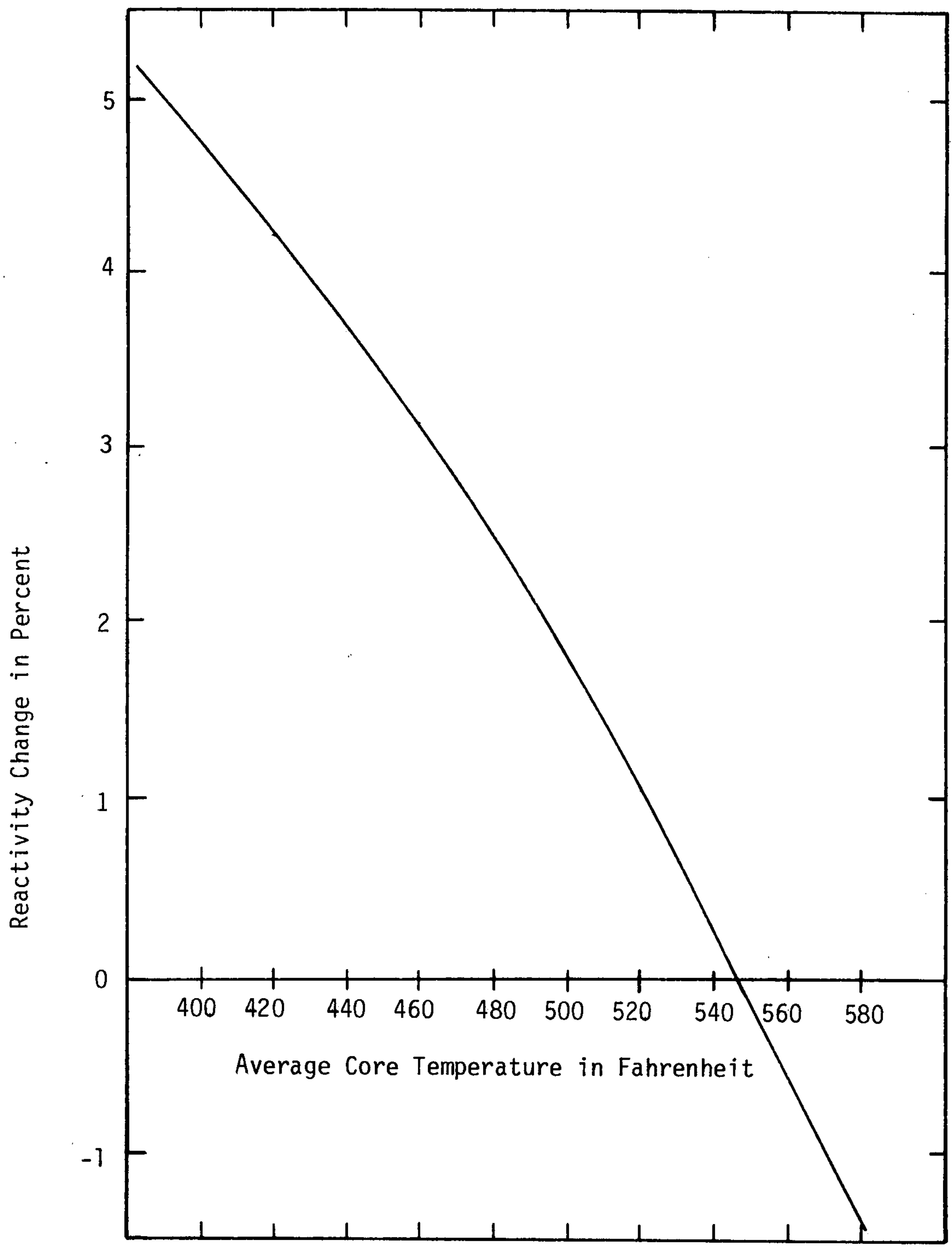

FIGURE 3.32 Moderator Temperature Feedback Reactivity for Steamline Break Transients

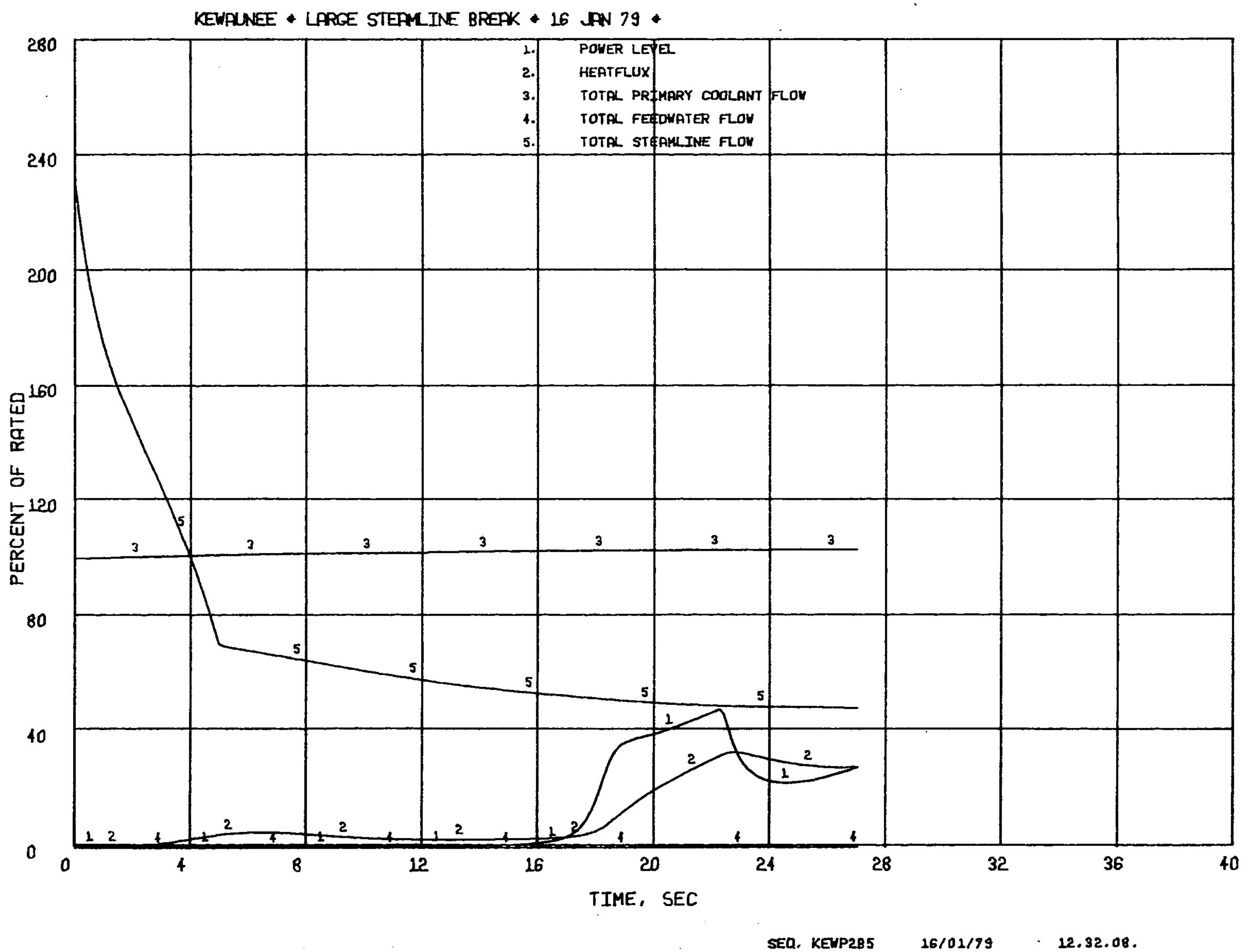

FIGURE 3.33 Power, Heatflux, and System Flow for Large Steamline Break

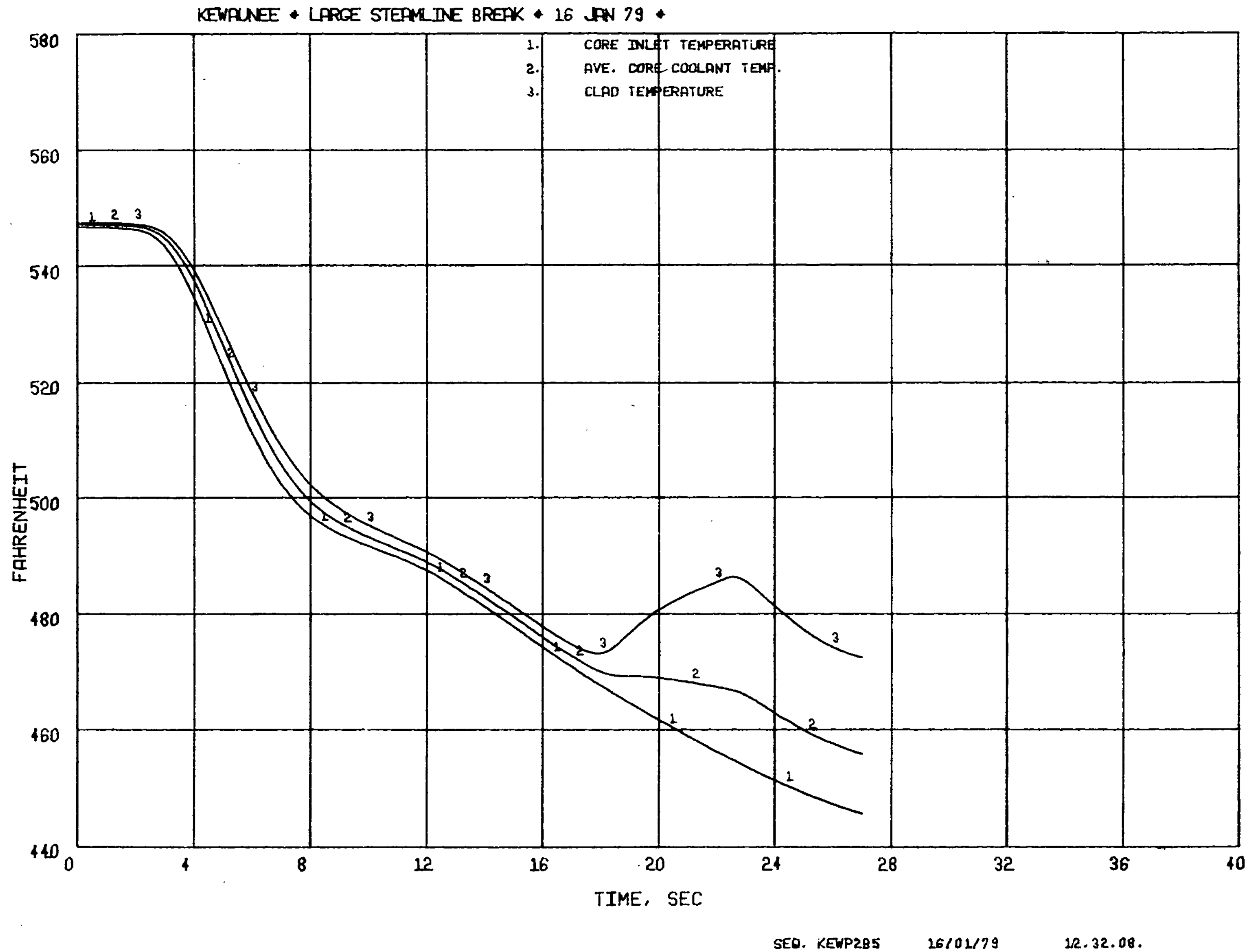

FIGURE 3.34 Core Temperature Response for Large Steamline Break

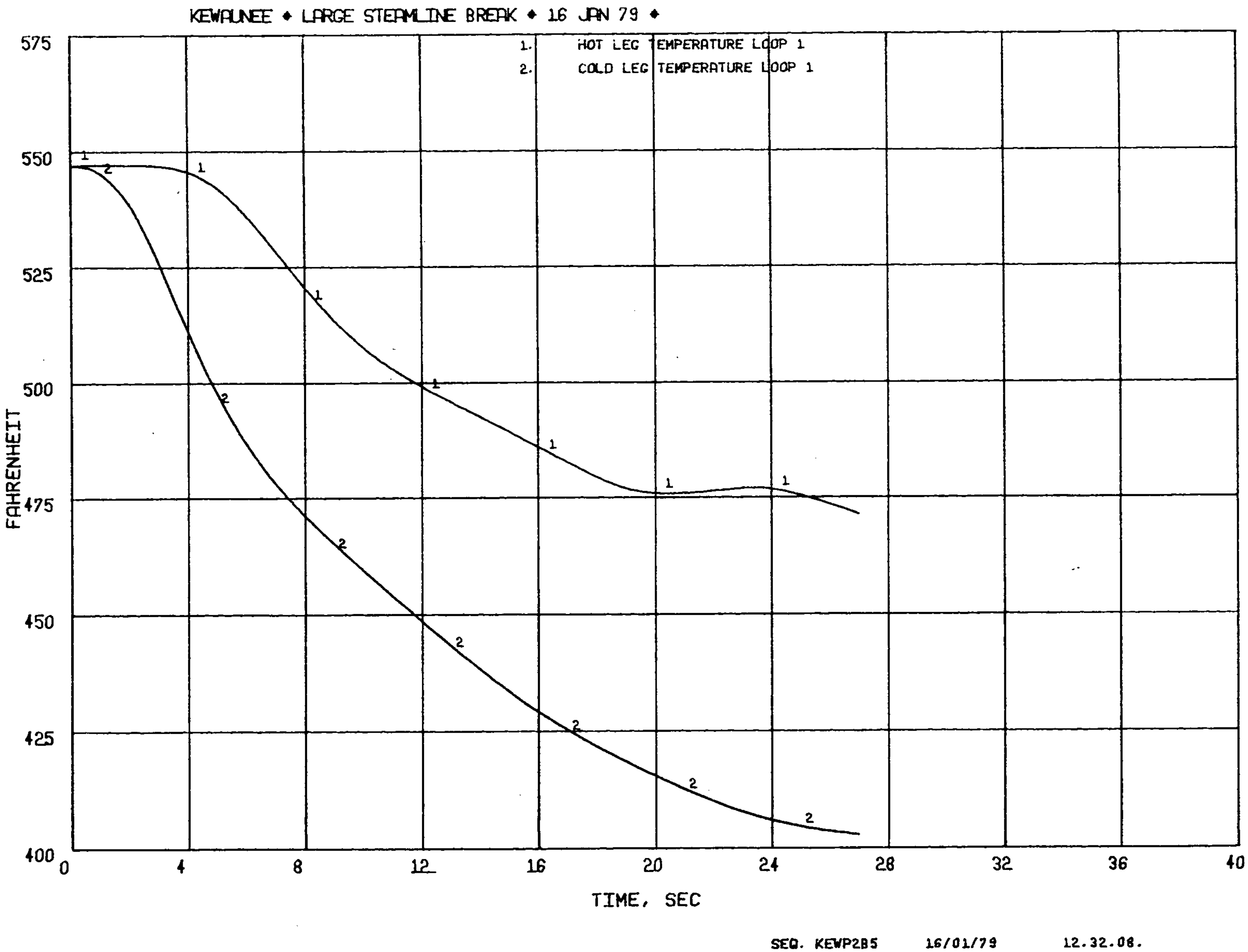

FIGURE 3.35 Primary Loop Temperature Response for Large Steamline Break

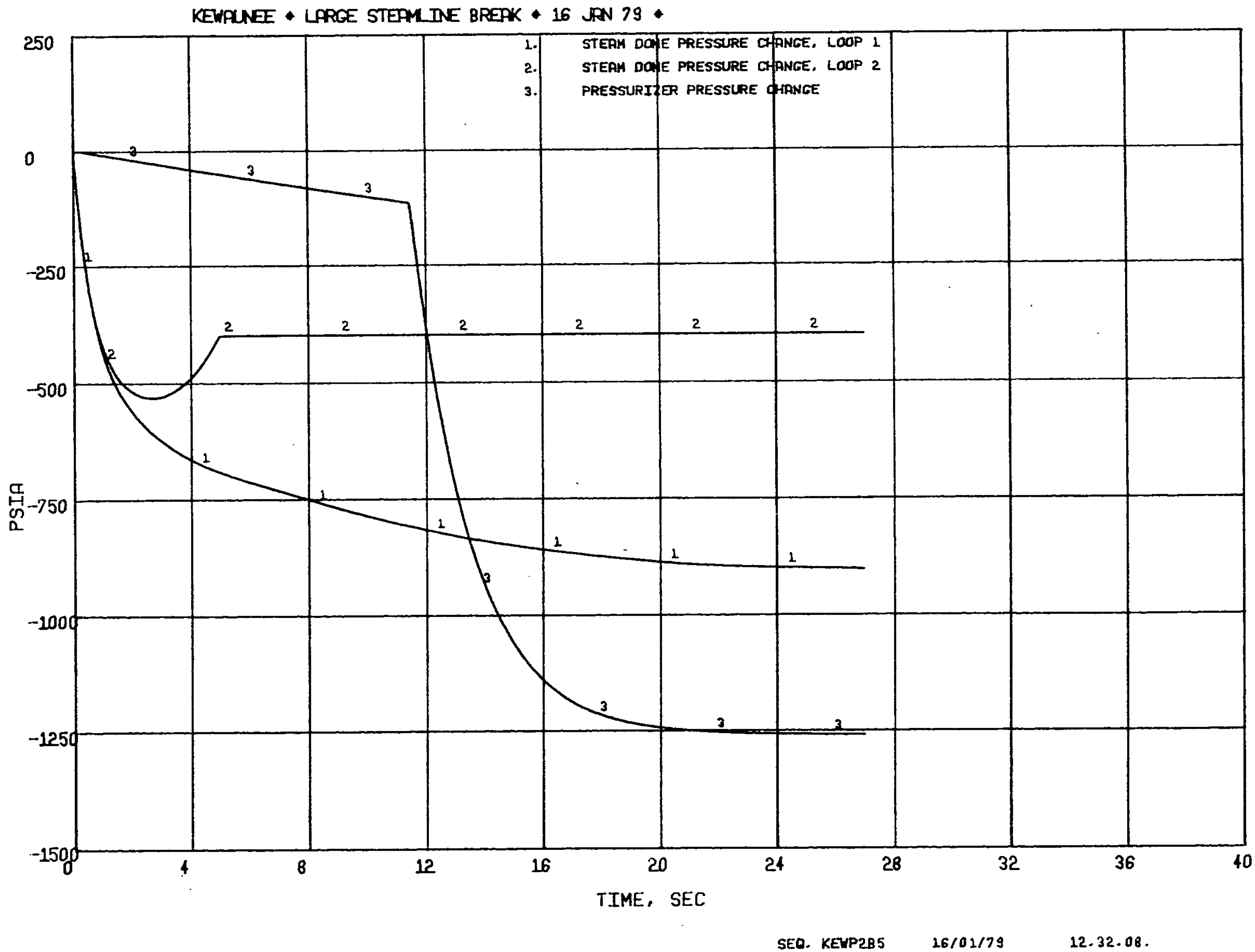

FIGURE 3.36 Pressure Changes in Pressurizer and Steam Generators for Large Steamline Break

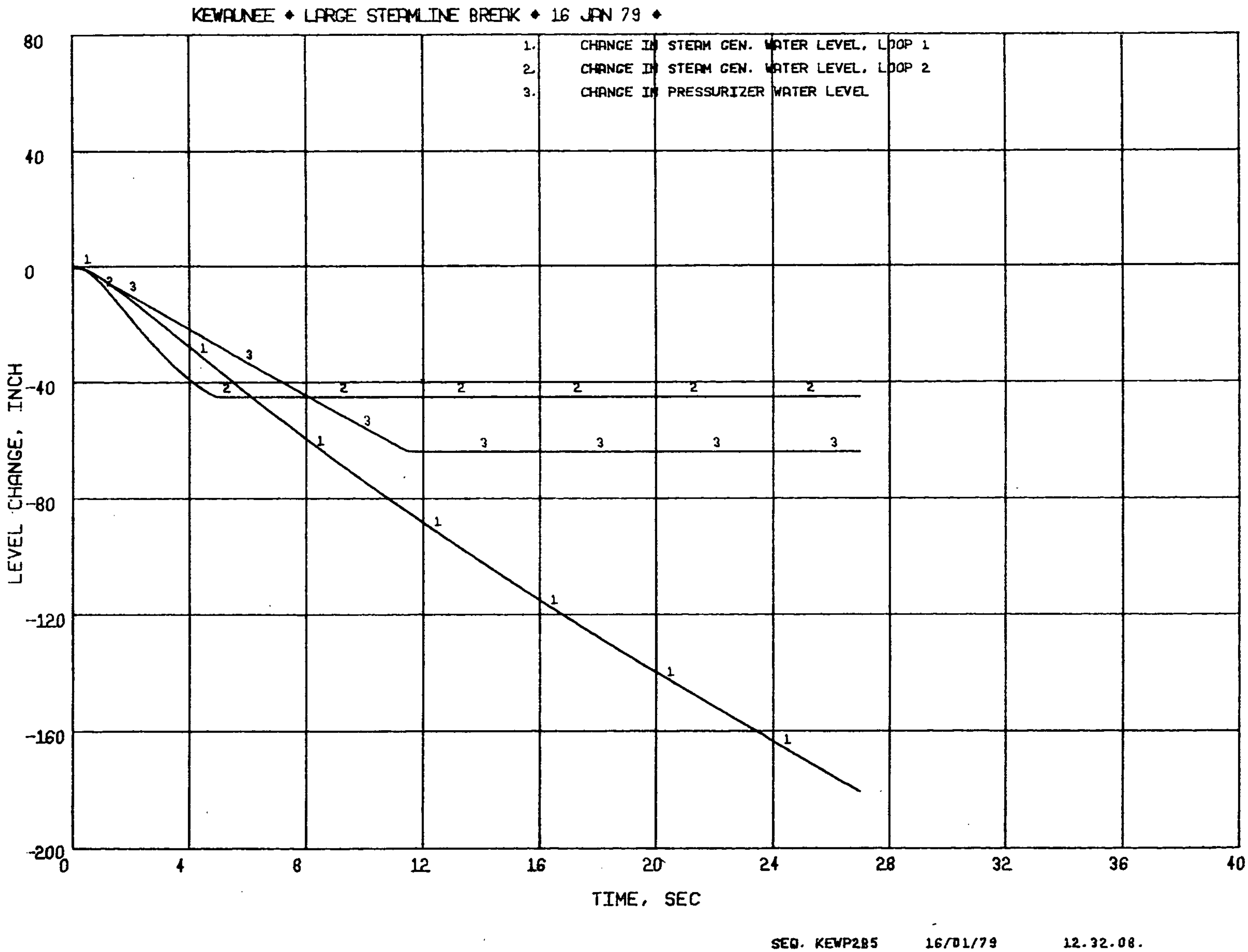

FIGURE 3.37 Level Changes in Pressurizer and Steam Generators for Large Steamline Break

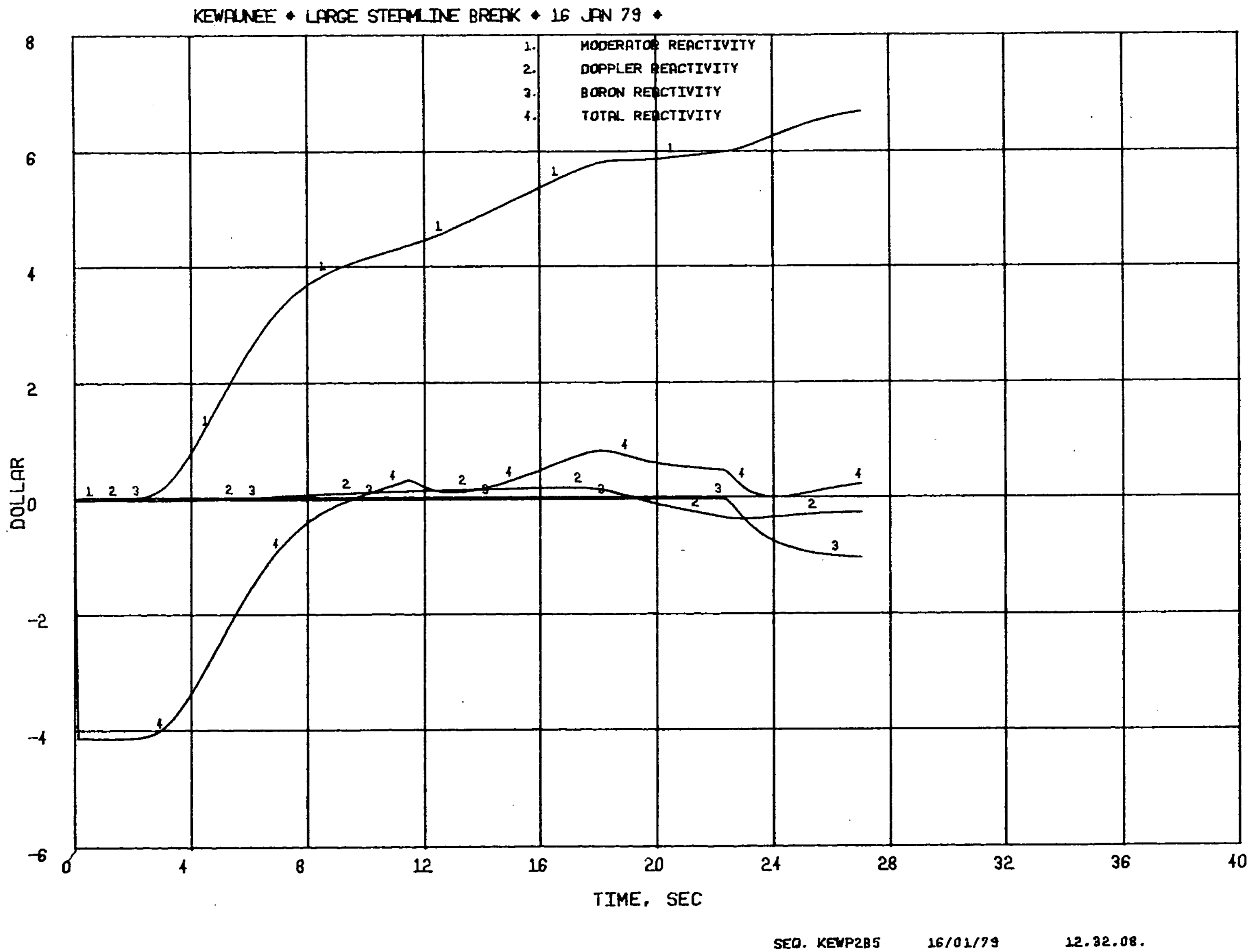

FIGURE 3.38 Nuclear Reactivity Feedback Effects for Large Steamline Break

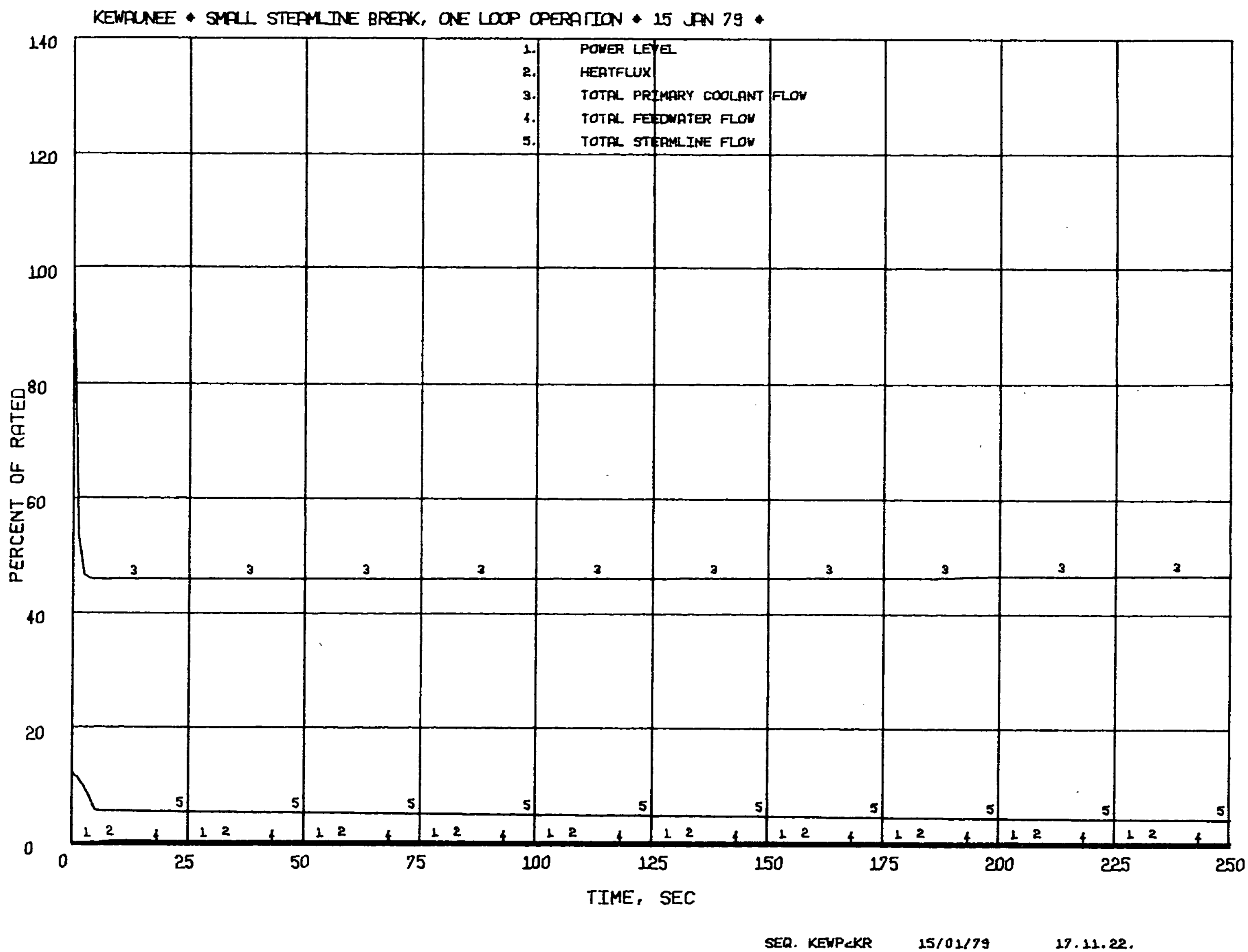

FIGURE 3.39 Power, Heatflux and System Flows for Small Steamline Break

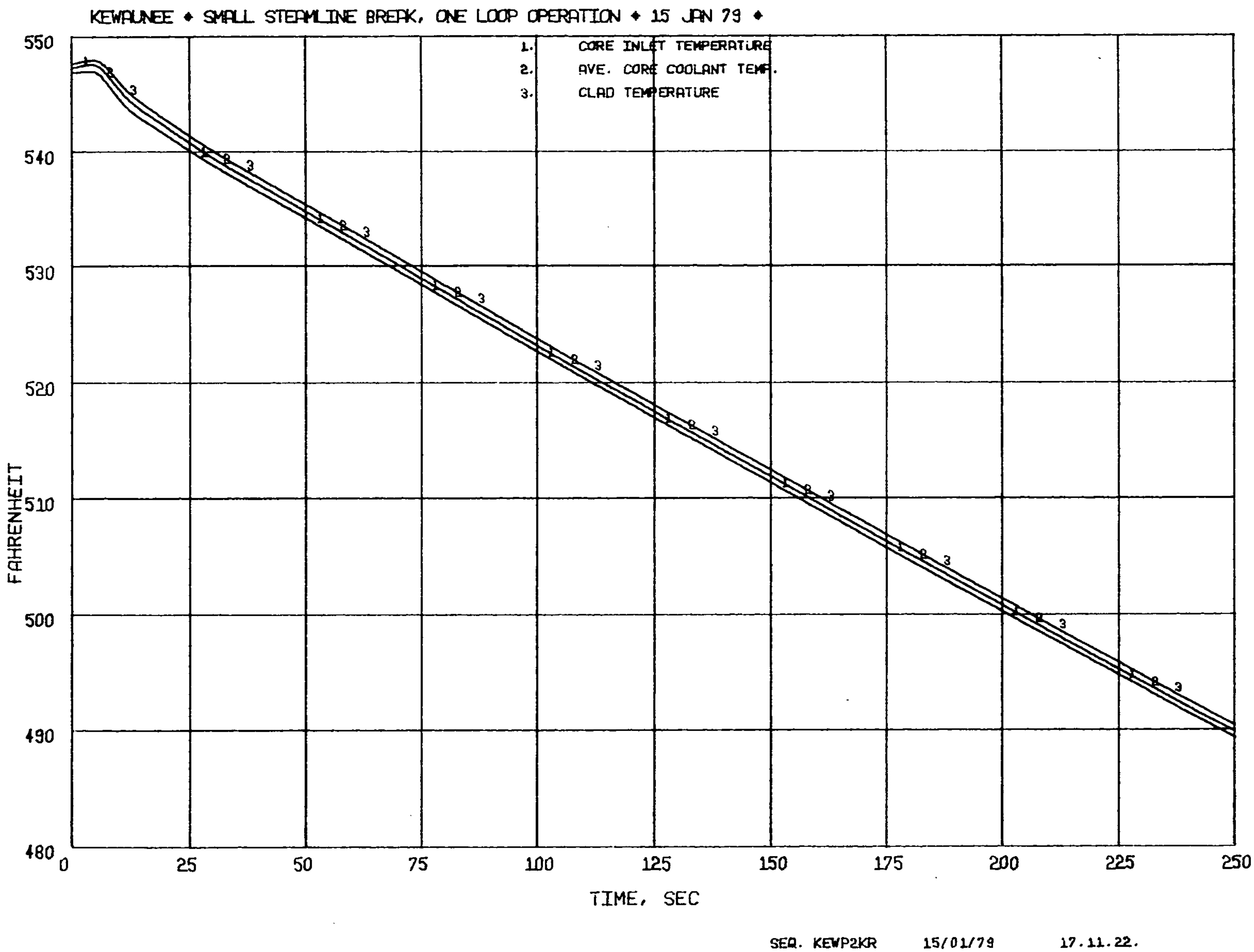

FIGURE 3.40 Core Temperature Response for Small Steamline Break

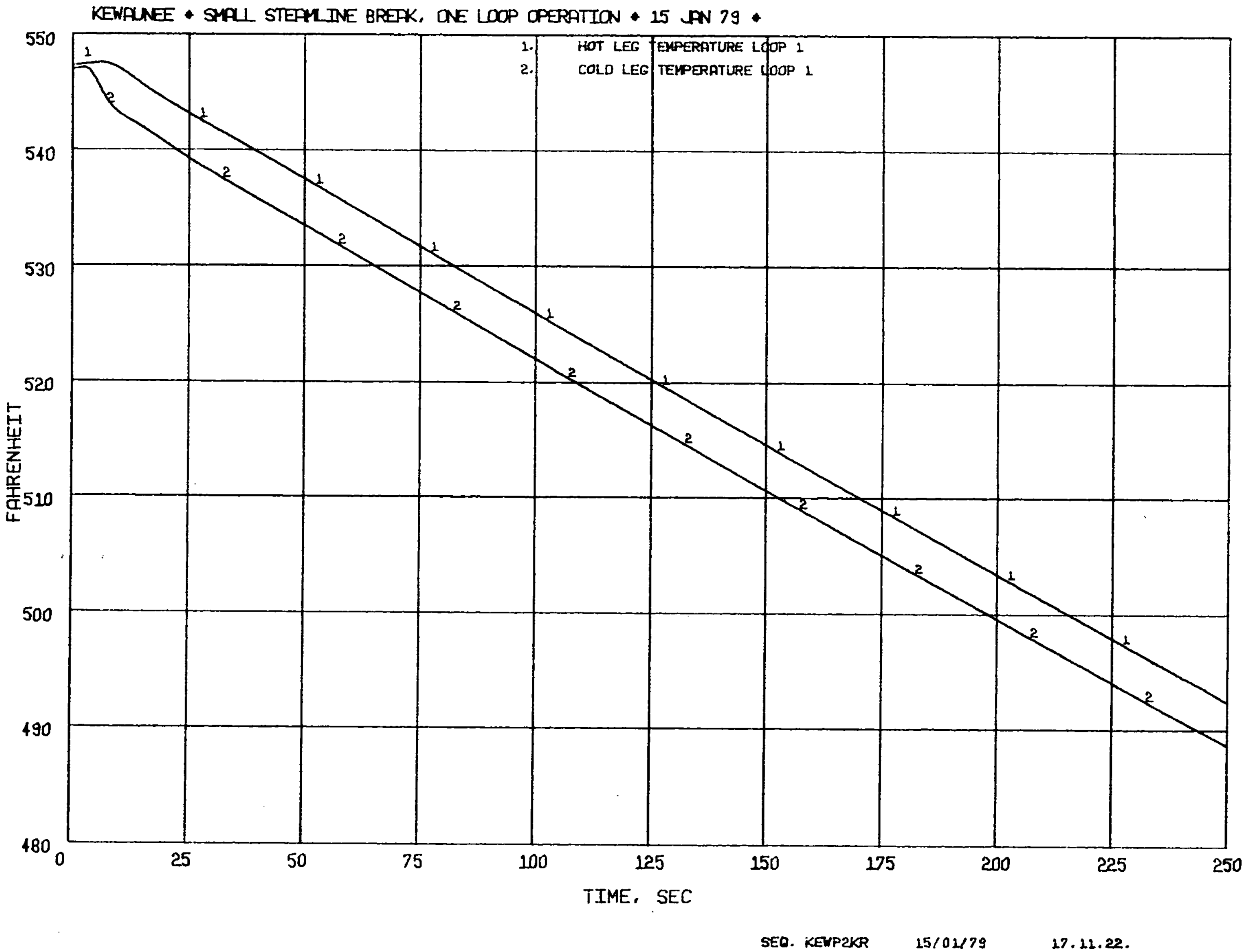

FIGURE 3.41 Primary Loop Temperature Response for Small Steamline Break

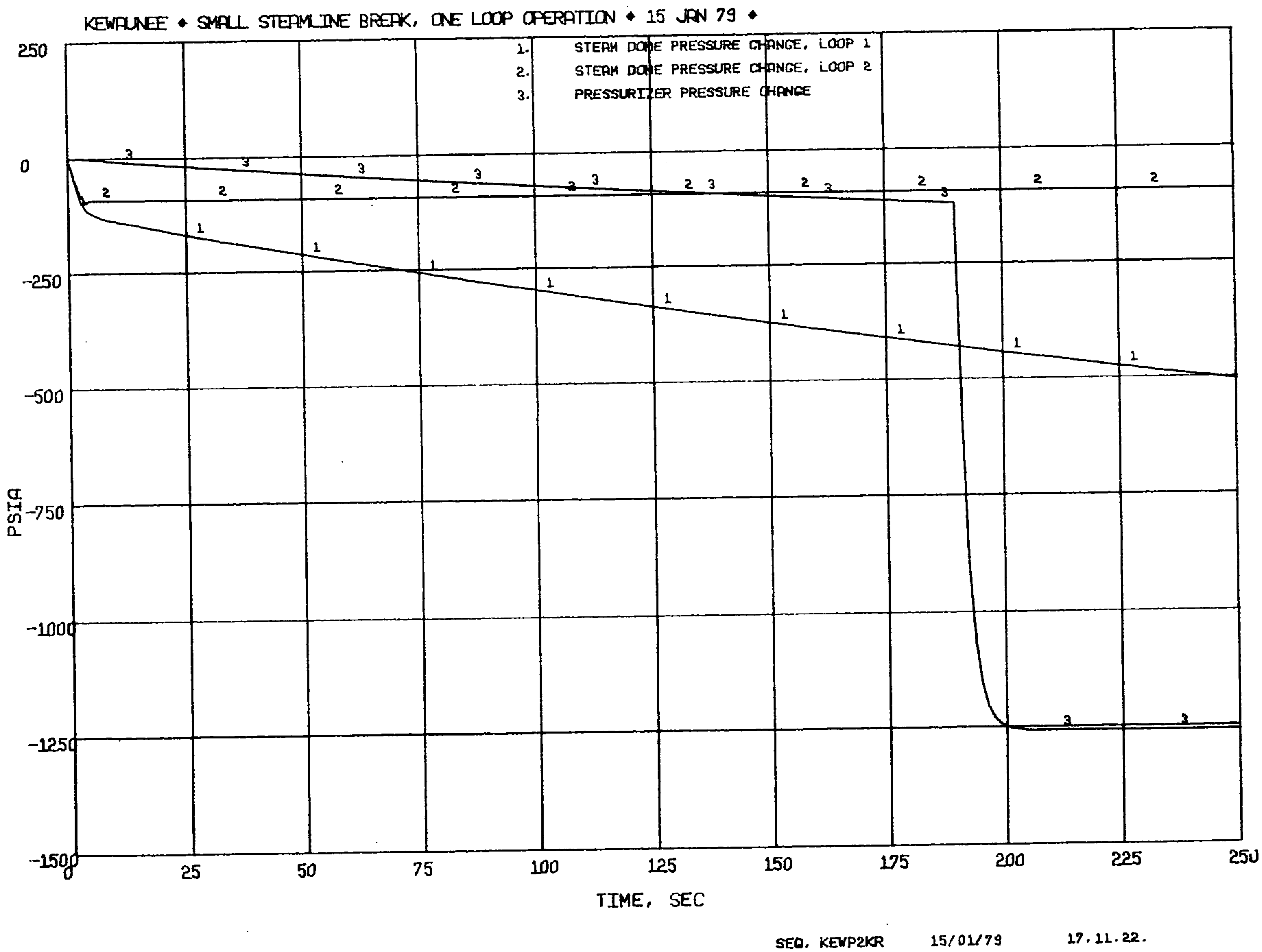

FIGURE 3.42 Pressure Changes in Pressurizer and Steam Generators for Small Steamline Break

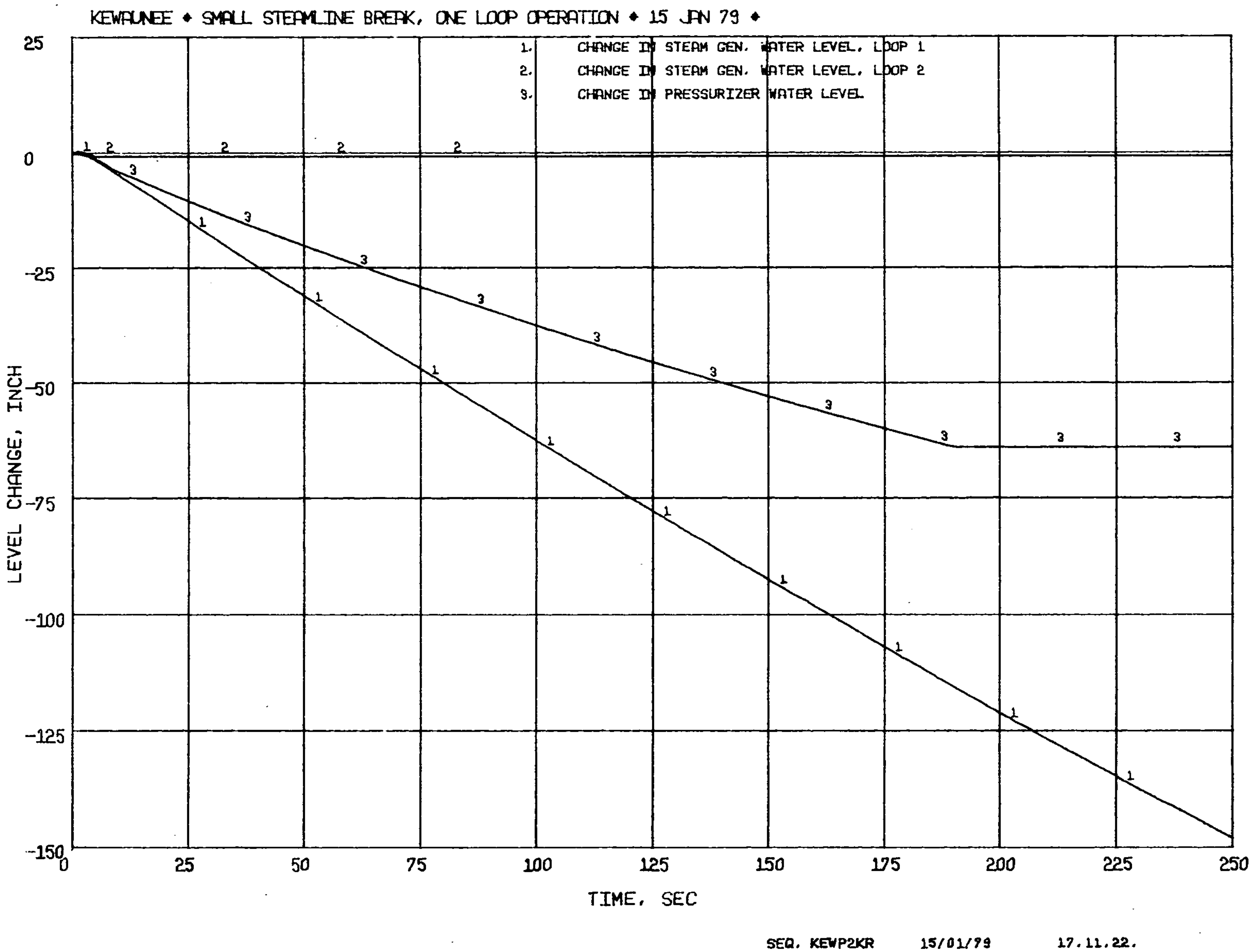

FIGURE 3.43 Level Changes in Pressurizer and Steam Generators for Small Steamline Break

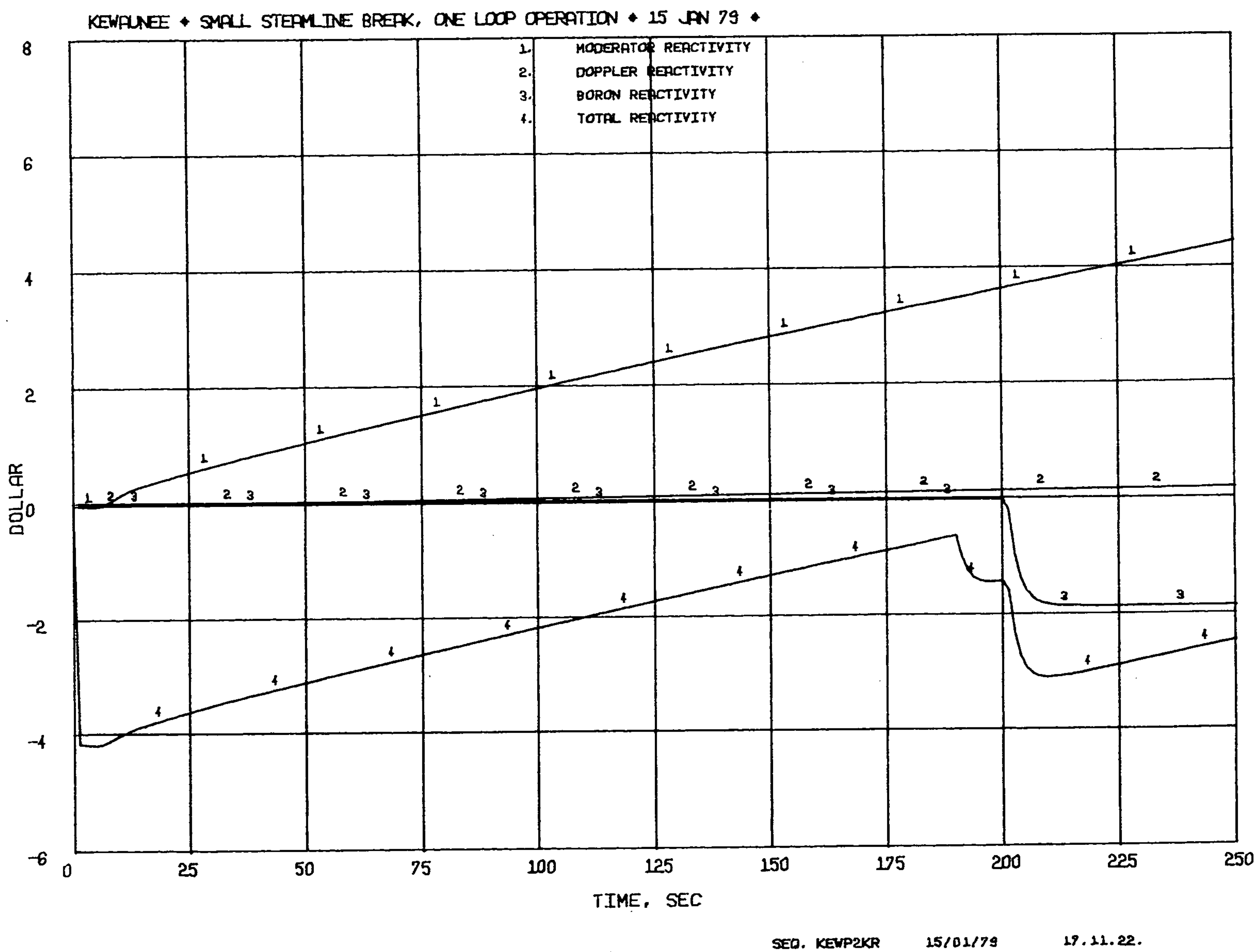

FIGURE 3.44 Nuclear Reactivity Feedback Effects for Small Steamline Break

4.0 DISCUSSION OF RESULTS

The transient analysis as performed by ENC for the Kewaunee 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) Coastdown of both Primary Coolant Pumps
4) Seizure of One Primary Coolant Pump
5) Loss of External 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. The evaluation criteria 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. 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 heatflux margin to DNB is most limiting in the seized pump rotor case, a Category IV event. Likewise, the ENC analysis showed it to be the most limiting event. It is the only transient where the DNB heatflux ratio is calculated to be below 1.3. A statistical analysis shows that a fraction of about 0.06 of fuel experiences a thermal margin to DNB of less than 1.3, and a fraction of less than 0.015 is expected to experience DNB.

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 $\dot{\Delta k} = 10^{-6}$ sec^{-1} to $\dot{\Delta k} = 10^{-3}$ sec^{-1}. 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} sec^{-1} 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 $\dot{\Delta k} = 1.5 \times 10^{-4}$ sec^{-1} for 60 percent power and up to $\dot{\Delta k} = 2.8 \times 10^{-4}$ sec^{-1} 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 Core 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 can be lower. A typical boron dilution event would cause a reactivity insertion at $\dot{\Delta k} = 10^{-5}$ sec^{-1}. 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 $\dot{\Delta k} = 10^{-5}$ sec^{-1}) 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 DATA

	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	0.0	-10.0
From Reduced Power	0.0	small	---	---
Loss of Flow				
Pump Coastdown	0.0	-16.3	0.0	-10.0
Locked Rotor	0.0	NA*	0.0	-10.0
Inactive Loop Startup	-400.0	-10.0	---	---
Loss of Load	0.0	NA*	0.0	-10.0
Loss of Feedwater	NA*	NA*	---	---
Excessive Feedwater	0.0 and -400.0	NA*	---	---
Excessive Load Increase	0.0 and -400.0	NA*	---	---
Steam Line Break	Variable	Variable	Fig. 3.32	Fig. 3.31

* Information not available in reference cycle analysis

TABLE 4.2

COMPARISON OF OPERATING PARAMETERS FOR KEWAUNEE

	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	1.75	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	530.0
Heat Transfer		
Average Heatflux, Btu/hr-ft^2	191,000	190,072

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

** Information not available in reference cycle analysis.

TABLE 4.2 (Continued)

COMPARISON OF OPERATING PARAMETERS

FOR KEWAUNEE

	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 KEWAUNEE REACTIVITY DATA

	Reference Cycle		Cycle 5 with ENC Fuel	
	BOC	EOC	BOC	EOC
Moderator Temperature Coefficient in 1/F	(+30* to 0.0) 10^{-6}	-350×10^{-6}	0.0	-350×10^{-6}
Moderator Pressure Coefficient in 1/psia	-0.3×10^{-6}	$+3.5 \times 10^{-6}$	0.0	$+4.12 \times 10^{-6}$
Doppler Coefficient in 1/F	-10×10^{-6}	-16×10^{-6}	(-10 to -23.2)10^{-6}	(-10 to -23.2)10^{-6}
Delayed Neutron Fraction	7.1×10^{-3}	5.1×10^{-3}	7.0×10^{-3}	4.85×10^{-3}

* Value for hot standby conditions.

5.0 SIMULATION CODE UPDATES

The basic digital plant simulation code as documented in Reference 1 has been used in performing the plant transient analysis for the Kewaunee plant. Starting from the version PTS-PWR2-NOV78, code updates have been implemented resulting in the version PTS-PWR2-JAN79. Most updates were restricted to the initialization modules of the code, so that the dynamic plant model of the PTS code was not affected. The purpose of these updates was (1) to remove a number of redundant variables from the input list and generate them internally in the code and (2) to redefine some input parameters such that hand calculations are eliminated or reduced. The only change affecting a transient code module was done in the pressurizer pressure calculation for cases where the pressurizer runs empty of water. This occurs in cooldown transients (steamline breaks for instance). The new version of the pressure calculation predicts a more rapid pressure decrease after emptying of the pressurizer than the old version did, and thus, follows more accurately the physical behavior during depressurization. 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 JAN 79
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

from Input or Redefined

Removed

CL1TS
CL2TS
HCLADO*
HL1TS
HL2TS

Redefined

Variable	New Definition
PLO	Initial reactor power in percent of nominal value
POWMAX	Overpower trip setpoint in percent of nominal value
WLP1SC, WLP2SC	Loop massflow values for low flow trip in percent of nominal value
LNL, LNMAX	Pressurizer level values in percent of level span
KCL1, KCL2, KHL1, KHL2, KRV, KSG1, KSG2	Delta-P values in psi (rather than pressure drop coefficients)

* Sieder-Tate correlation used

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. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Final Safety Analysis Report.

4. Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Technical Specifications, Docket 50-305.

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.