Attachment A to AEP:NRC:0745C
Proposed Technical Specifications

DESIGN THERMAL POWER

1.27 DESIGN THERMAL POWER shall be a design total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

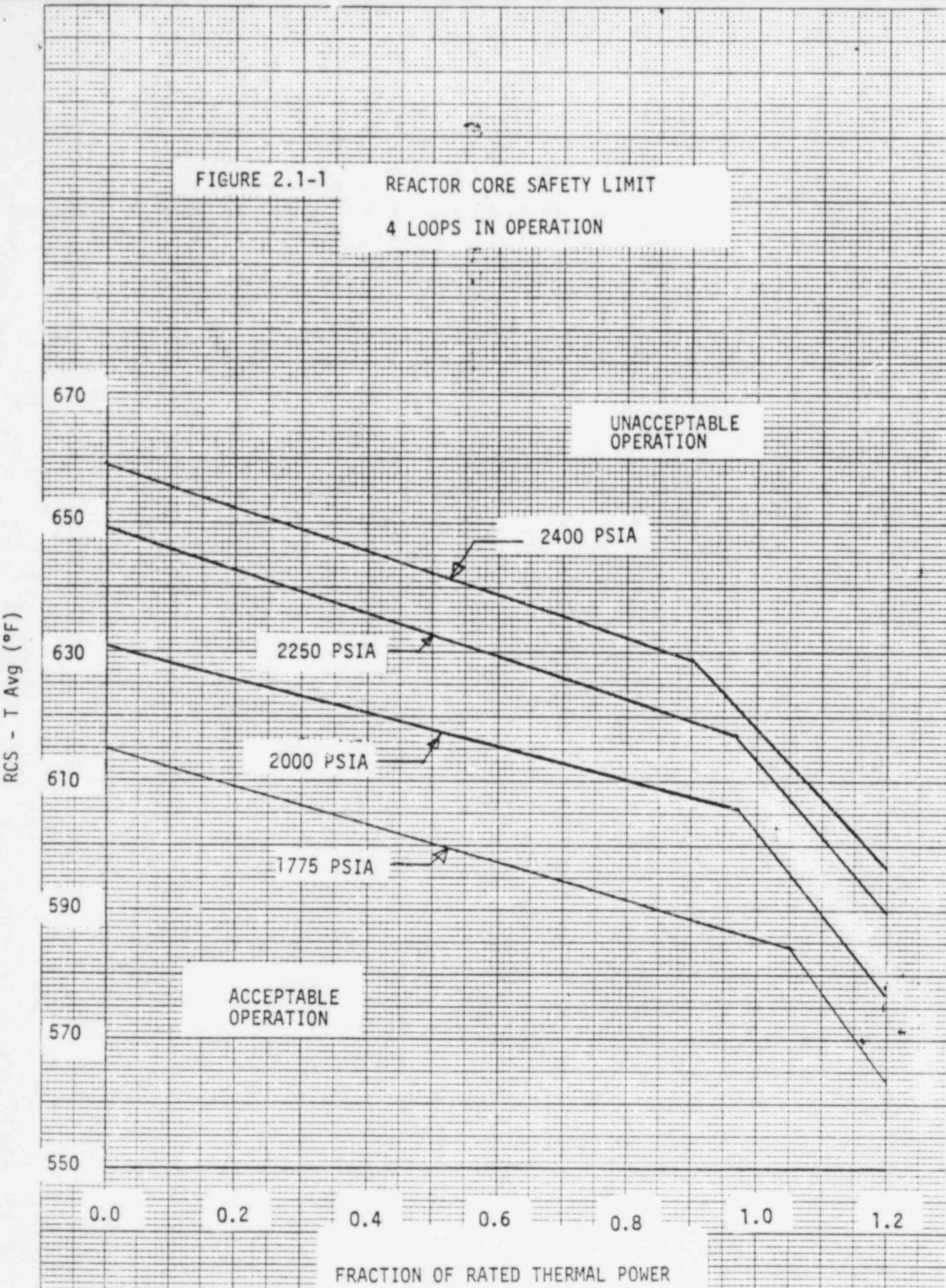


FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT
4 LOOPS IN OPERATION
UNACCEPTABLE OPERATION
2400 PSIA
2250 PSIA
2000 PSIA
1775 PSIA
ACCEPTABLE OPERATION
RCS - T Avg (°F)
670
650
630
610
590
570
550
0.0
0.2
0.4
0.6
0.8
1.0
1.2
FRACTION OF RATED THERMAL POWER
46 1320
K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

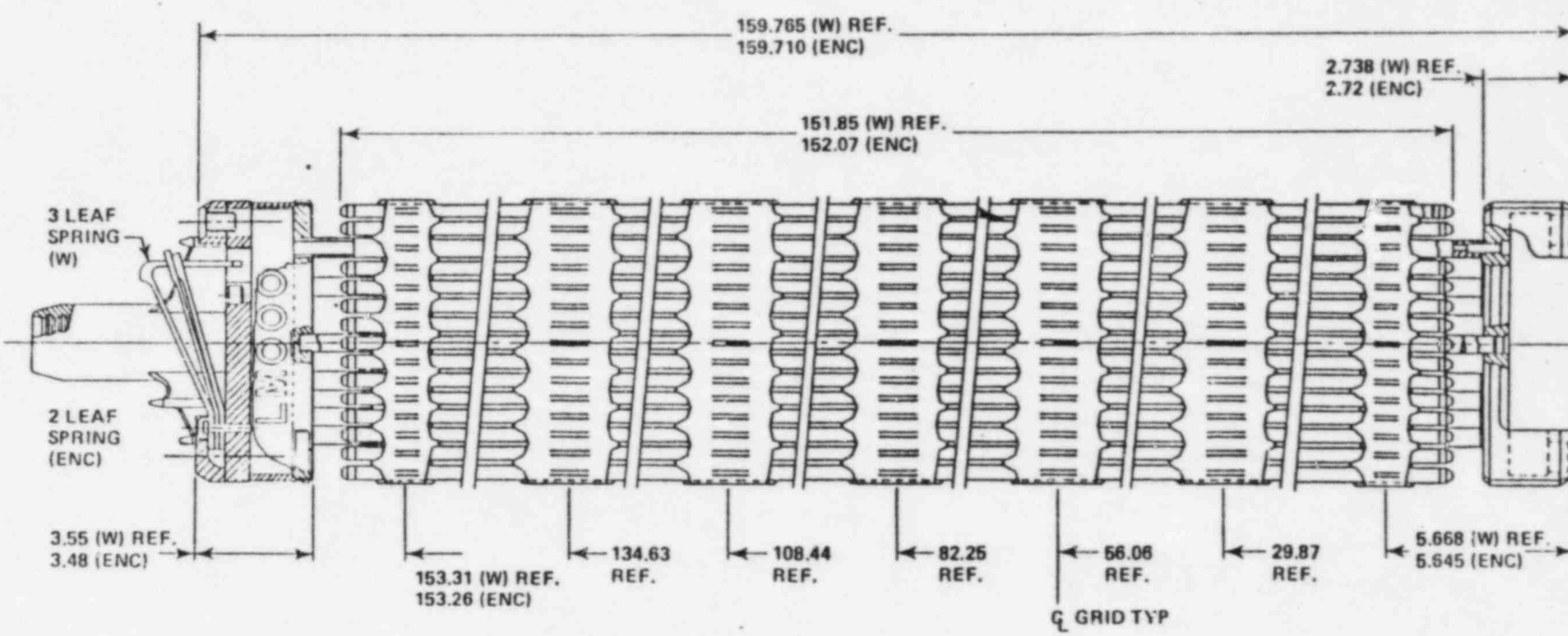
FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 1 (Continued)

Parameter	15x15 W Optimized Fuel Assembly Design	15x15 ENC Fuel Assembly Design
Structural Mat'l - Two End Grids	Inconel	Zircaloy-4 Straps, Inconel Springs
Grid Height, in., Outer Straps, Valley-to Valley	2.25	2.25
Bottom Nozzle	Reconstitutable	
Top Nozzle Holddown Springs	3-leaf	2-leaf

SCHEMATIC OF WESTINGHOUSE 15X15 OFA



W - WESTINGHOUSE 15X15 OPTIMIZED FUEL ASSEMBLY (OFA) DIMENSION
ENC - EXXON NUCLEAR COMPANY (ENC) 15X15 FUEL ASSEMBLY DIMENSION
NOTE: OFA AND ENC ASSEMBLY MID-GRIDS HAVE IDENTICAL AXIAL SPACINGS

ENC GRID HEIGHT - 2.25
WESTINGHOUSE TOP & BOTTOM GRID HEIGHTS - 1.5
WESTINGHOUSE MID GRID HEIGHT - 2.25

Figure 1 Comparison of ENC Fuel Assembly Dimensions With Westinghouse 15X15 OFA Schematic

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2-25

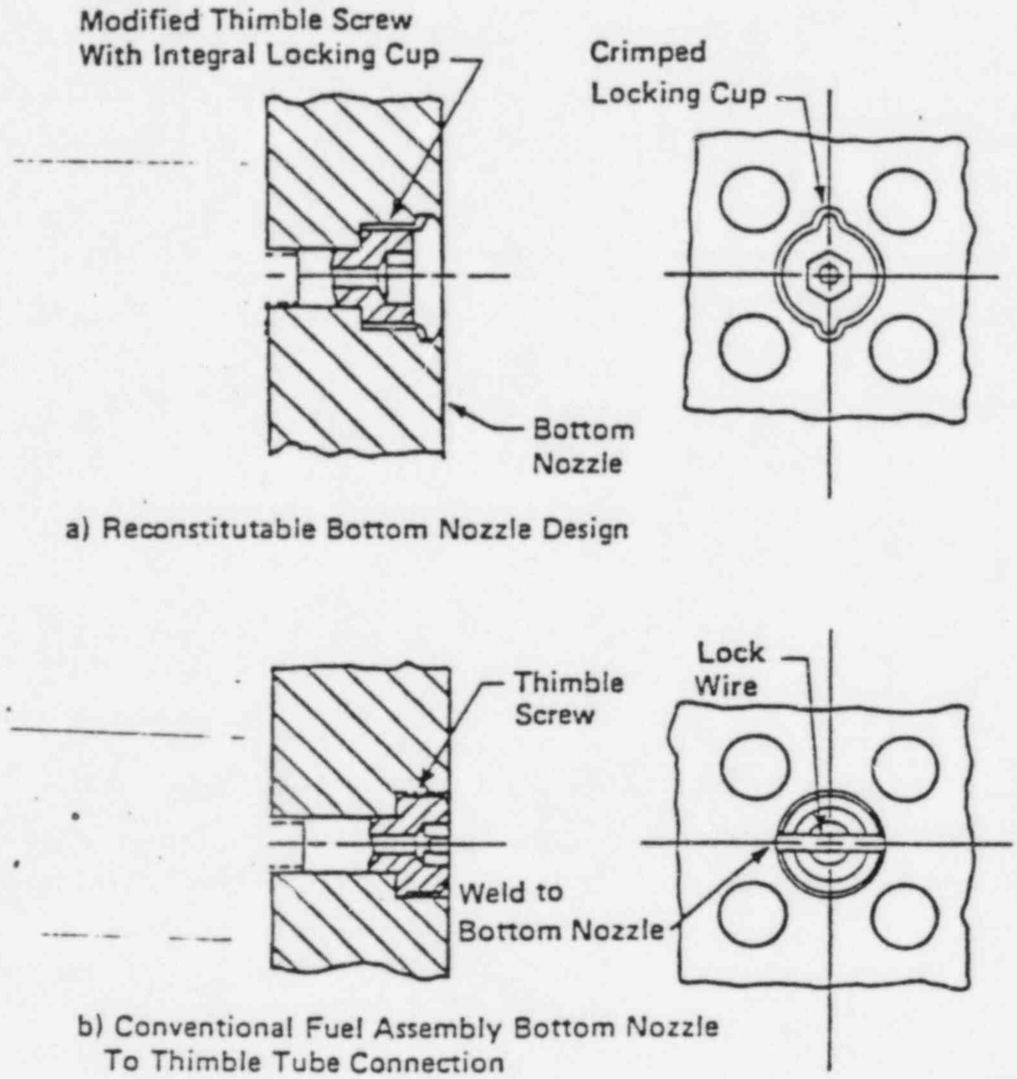
TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_0 \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where:	ΔT_0	=	Extrapolated ΔT at DESIGN THERMAL POWER
	T	=	Average temperature, °F
	T'	=	577.1°F (indicated T_{avg} at DESIGN THERMAL POWER)
	P	=	Pressurizer pressure, psig
	P'	=	2235 psig (indicated RCS nominal operating pressure)
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 33 secs, τ_2 = 4 secs.
	S	=	Laplace transform operator



a) Reconstitutable Bottom Nozzle Design

b) Conventional Fuel Assembly Bottom Nozzle To Thimble Tube Connection

BOTTOM NOZZLE TO THIMBLE TUBE CONNECTION

FIGURE 2

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 4 Loops	Operation with 3 Loops
$K_1 = 1.135$	$K_1 = 0.99$
$K_2 = 0.0130$	$K_2 = 0.01026$
$K_3 = 0.000659$	$K_3 = 0.000617$

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -37 percent and +2 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent DESIGN THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of DESIGN THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 2.3 percent of its value at DESIGN THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +2 percent, the ΔT trip setpoint shall be automatically reduced by 1.8 percent of its value at DESIGN THERMAL POWER.

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 1: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 \left(T - T''\right) - f_2(\Delta I)\right]$

where:

ΔT_o	=	Extrapolated ΔT at DESIGN THERMAL POWER
T	=	Average temperature, °F
T''	=	Indicated T_{avg} at DESIGN THERMAL POWER 577.1°F
K_4	=	1.089
K_5	=	0.0177/°F for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.0011 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	$f_1(\Delta I)$ as defined in Note 1 above.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 4 percent.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures, because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in the heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore, THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent confidence that the minimum DNBR for the limiting rod is greater than or equal to the applicable design DNBR limit for each fuel type (as defined below). For 4 loop operation, the improved thermal design procedure is used. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit (as defined below), establishes a design DNBR limit value, which must be met in plant safety analyses, using values of input parameters without uncertainties. For 3 loop operation, a conservative set of uncertainties are used in the safety analyses.

The table below indicates the relationship between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design.

2.1 SAFETY LIMITS

BASES

	4 Loop Operation				3 Loop Operation	
	(WRB-1 Correlation)		(W-3 Correlation)		(W-3 Correlation)	
	Westinghouse Fuel (15x15 OFA)		Exxon Nuclear Co. Fuel (15x15)		W and ENC Fuels	
	Typical	Thimble	Typical	Thimble	Typical	Thimble
Correlation Limit	1.17	1.17	1.30	1.30	1.30	1.30
Design Limit DNBR	1.32	1.31	1.58	1.50	1.30	1.30
Safety Analysis Limit DNBR	1.69	1.69	1.58	1.50	1.30	1.30

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor $F^N_{\Delta H}$, of 1.49 for Westinghouse fuel and an $F_{\Delta H}$ of 1.45 for Exxon Nuclear Co. fuel and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power, based on the expressions:

$F^N_{\Delta H} = 1.49 [1 + 0.3 (1-P)]$ (for Westinghouse fuel)

and $F^N_{\Delta H} = 1.45 [1 + 0.2 (1-P)]$ (for Exxon Nuclear Co. fuel)

where P is the fraction of RATED THERMAL POWER

Note, do not include a 4% uncertainty value, since this measurement uncertainty has been included in the design DNBR limit values, which are listed in the bases for Section 2.1.1.

Although the N-loop operation curves are calculated for operation at DESIGN THERMAL POWER, $F^N_{\Delta H}$ values for RATED THERMAL POWER are reported here in order to be consistent with Section 3.2.3. The $F^N_{\Delta H}$ values of Section 3.2.3 are limited by the LOCA analyses which were performed at RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion, assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with the core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant, which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless, manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically re according to the notations in Table 2.2-1.

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drops below 90% of nominal full loop flow. Above 51% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the applicable safety analysis design limit DNBR value for each fuel type, (as listed in the bases for Section 2.1.1) during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature ΔT trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip setpoint adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the applicable safety analysis design limit DNBR value for each fuel type, (as listed in the bases for Section 2.1.1) during normal operational transients and anticipated transients when 3 loops are in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip, to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses, but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} > 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1.60% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN < 1.60% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1.60% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2##, at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

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*See Special Test Exception 3.10.1

#With $K_{eff} \geq 1.0$

##With $K_{eff} < 1.0$

Attachment C to AEP:NRC:0745C
Non-LOCA Transients Safety Analyses

6.0 ACCIDENT ANALYSIS AND EVALUATION

6.1 NON-LOCA INTRODUCTION AND SUMMARY

This section evaluates the effects of the transition from the resident ENC fuel to W 15x15 OFA on the D.C. Cook Unit 1 licensing basis with respect to FSAR Chapter 14. Standard Westinghouse reload methodology, as described in reference (1), was used. The results of the analysis, which are presented in the following section show that the transition to W 15x OFA can be accommodated with margin to the applicable FSAR safety limits.

All of the non-LOCA transients were reanalyzed.* There are three major new design parameters for D.C. Cook 1 that affect the transition and use of OFA.

1. The analysis is conservatively performed at 3425 MWt NSSS power with a 577.1°F vessel average temperature, even though the cycle 8 core will continue to be limited to its current rated parameters of 3250 MWt NSSS and 567.8°F vessel average temperature. This affects all of the transients that are limiting at full power.

2. As discussed in Section 6.1 of Attachment B, the analyses employed the Improved Thermal Design Procedure (ITDP)[2] for DNB limiting transients. Both the W-3 and WRB-1 correlations were used (see Section 5.0 of Attachment B). A conservative set of core thermal safety limits, overtemperature ΔT and overpower ΔT setpoints were generated that are applicable for the transitions and complete OFA core.

* Except startup of an Inactive Loop. This transient can not occur above the P-7 setpoint (10% power) and thus was not analyzed.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 541°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: Mode 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to $\leq$ 76 percent of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

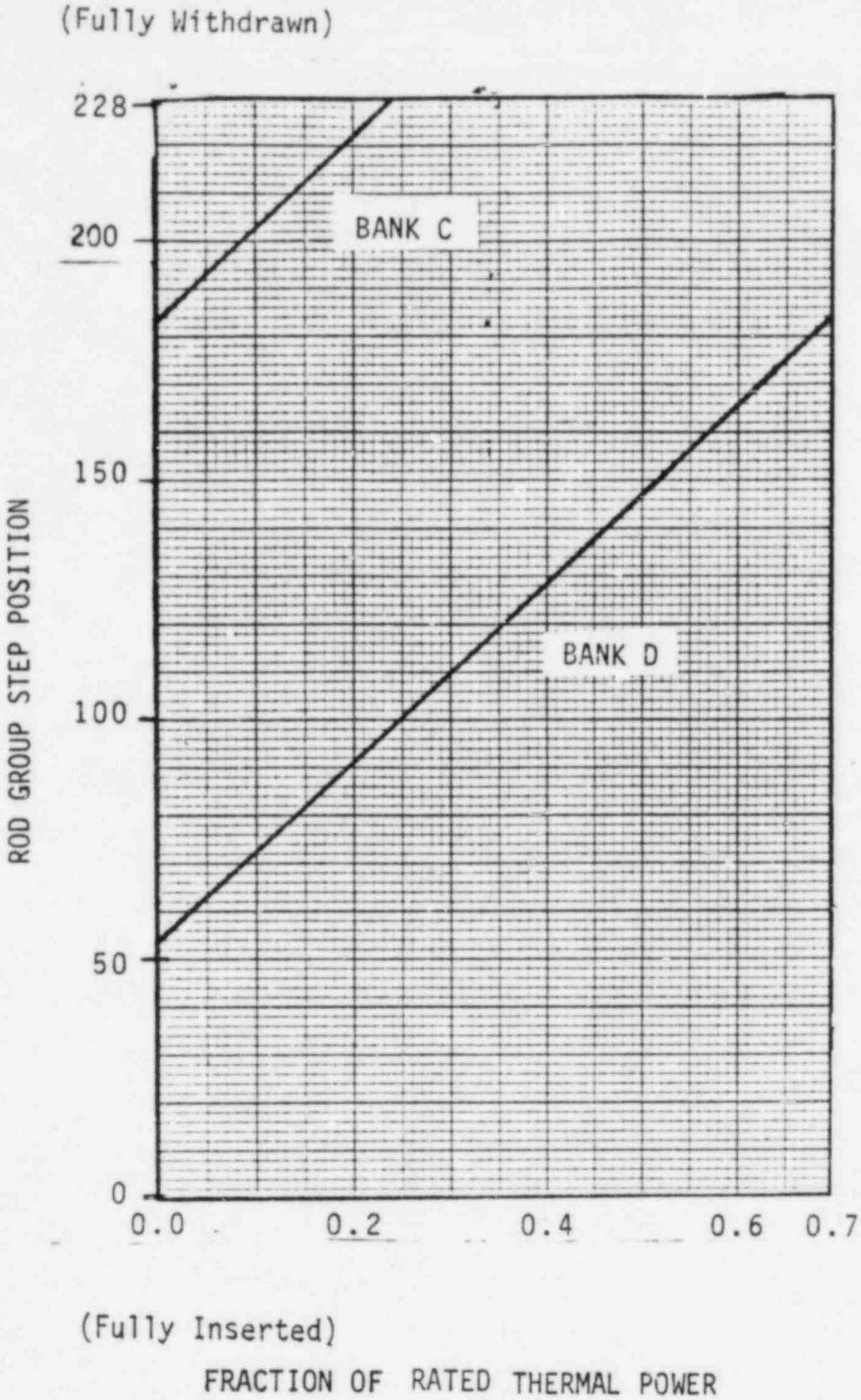

FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER
3 LOOP OPERATION

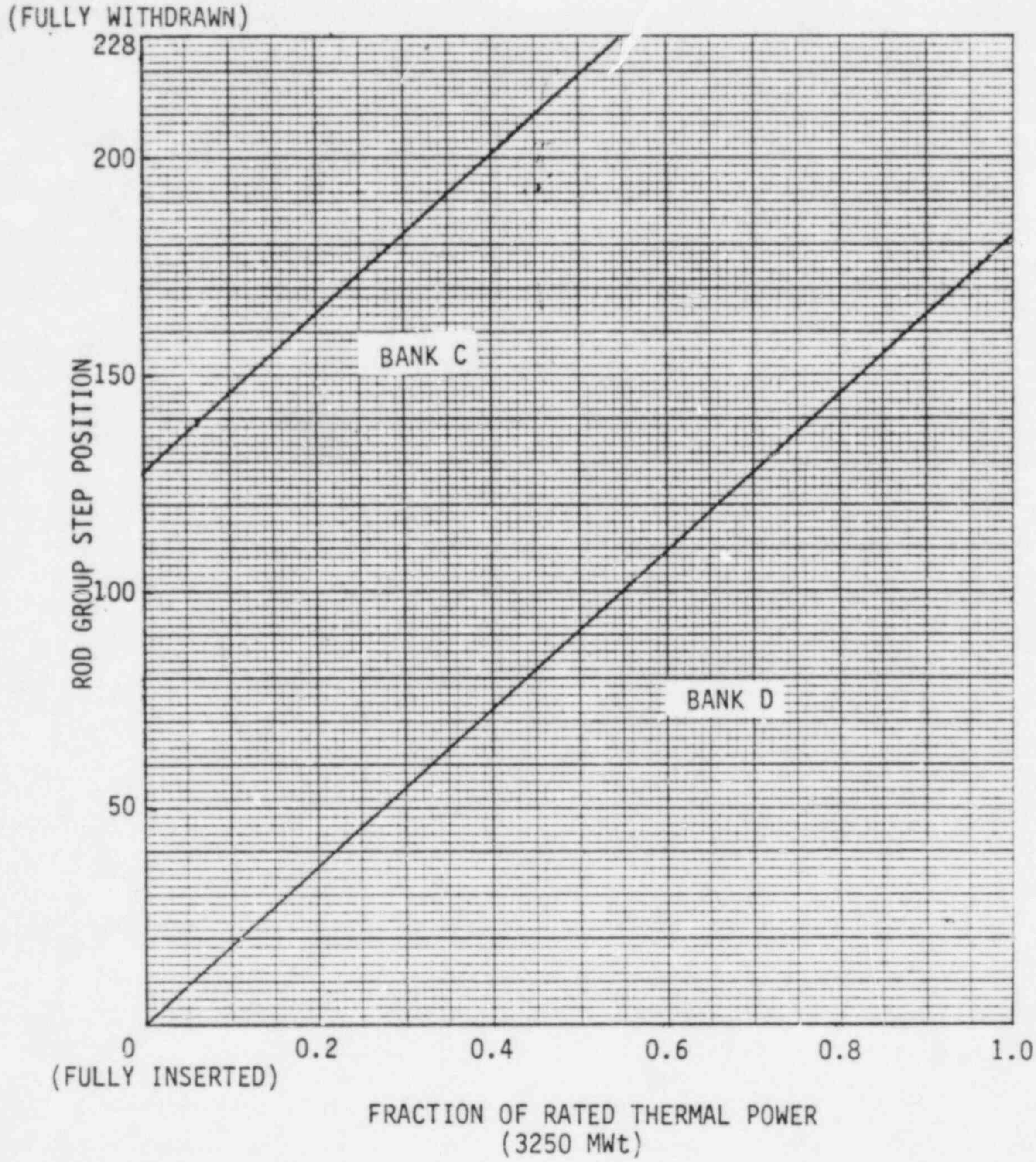

FIGURE 3.1-2 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER
4 LOOP OPERATION

3. The control rod scram time to the dashpot is increased (as discussed in Section 3.0 of Attachment B) from 1.8 seconds to 2.4 seconds. This increased drop time primarily affects the fast reactivity transients but was used in all of the analysis requiring this parameter.

   Also included in the analysis were fuel temperatures based on the revised PAD code and +5 pcm/°F moderator coefficient.

Table 1 lists the accidents that were necessary to reanalyze due to the above design parameters. Table 1 also lists transients that were analyzed in accordance with reference (1) for reasons other than those discussed above (i.e., change in key safety parameters). In addition, steamline break statepoints were generated to use in the key safety parameter evaluation for the steamline break event and are reported in Section 5.3.11.

6.2 ACCIDENT REANALYZED

6.2.0 General

The reanalyzed accidents were performed using current Westinghouse methodology and computer codes. Table 2 summarizes the initial conditions and computer codes used in the analysis. For most accidents which are DNB limited, nominal values of initial conditions and the minimum measured flow (364,900 gpm) are assumed. The allowances on power, temperature and pressure determined on a statistical basis are included in the limit DNBR as described in Section 5.0 of Attachment B.

For accidents that are not DNB limited or in which ITDP is not employed, the initial conditions are obtained by adding the maximum steady state errors to rated values. The following steady state errors are considered:

A. Core Power — + 2°F calorimetric error allowance

B. Average RCS Temperature — + 4°F controller deadband and measurement error allowance

C. Pressurizer Pressure $\pm$ 30 psi - steady state fluctuations and measurement error allowance

D. Reactor Flow Thermal Design Flow (354,000 gpm)

Reactor Protection Setpoints and response times used are listed in FSAR Chapter 14.1 and the Technical Specifications with the exception of the overtemperature T (OTΔT) and overpower T (OPΔT) setpoints. New OTΔT and OPΔT setpoints were calculated for the design basis based on the core thermal limits using the methodology described in reference 3. The results are included in the proposed Technical Specification changes of Section 7.0 of Attachment B.

6.2.1 Computer Codes Utilized

Summaries of some of the principal computer codes used in the transient analyses are given below.

FACTRAN

FACTRAN calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the clad using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the following features:

A. A sufficiently large number of radial space increments to handle fast transients such as rod ejection accidents.

B. Material properties which are functions of temperature and a sophisticated fuel-to-clad gap heat transfer calculation.

C. The necessary calculations to handle post-departure from nucleate boiling (DNB) transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the materials.

FACTRAN is further discussed in Reference 4.

LOFTRAN

The LOFTRAN program is used for transient response studies of a pressurized water reactor (PWR) system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing the reactor vessel, hot and cold leg piping, steam generators (tube and shell sides), and the pressurizer. The pressurizer heaters, spray, relief, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on high neutron flux, overtemperature ΔT, overpower ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The ECCS, including the accumulators, is also modeled.

LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits. The core limits represent the minimum value of DNBR as calculated for typical or thimble cell.

LOFTRAN is further discussed in Reference 5.

LEOPARD

The LEOPARD computer program determines fast and thermal neutron spectra, using only basic geometry and temperature data. The code optionally computes fuel depletion effects for a dimensionless reactor and recomputes the spectra before each discrete burnup step.

LEOPARD is further described in Reference 6.

TURTLE

TURTLE is a two-group, two-dimensional neutron diffusion code, featuring a direct treatment of the nonlinear effects of xenon, enthalpy, and Doppler feedback. Fuel depletion is allowed.

TURTLE was written for the study of azimuthal xenon oscillations. However, the code is useful for general analysis. The input is simple, fuel management is handled directly, and a boron criticality search is allowed.

TURTLE is further described in Reference 7.

TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady-state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided; e.g., channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 8.

THINC

The THINC -IV computer program, as approved by the NRC is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions along parallel flow channels within a reactor core under all expected operating conditions. The THINC-IV Code is described in detail in References 13 and 14, including models and correlations used.

6.3.0 Reanalyzed Accident Descriptions

The following sections contain the detailed descriptions of the reanalyzed accidents. In all cases the applicable FSAR acceptance criteria are satisfied.

6.3.1 Uncontrolled RCCS Withdrawal From A Subcritical Condition

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator and fuel.

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three states: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the departure from nucleate boiling ratio (DNBR) calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods (TWINKLE) to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN. The average heat flux is next used in THINC for transient DNBR calculation.

The accident is analyzed using the Improved Thermal Design Procedure with the initial conditions listed in Table 2. The analysis was performed for a reactivity insertion rate of 75pcm*/sec. This reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). A constant moderator temperature coefficient of +5 pcm/°F was used in the analysis.

Results and Conclusions

The nuclear power, coolant temperature, heat flux, fuel average temperature, and clad temperature versus time for a 75 pcm/sec insertion rate are shown in Figures 1 and 2. This insertion rate, coupled with a positive moderator temperature coefficient of +5 pcm/°F, yields a peak heat flux which does not exceed the nominal value. Therefore the conclusions presented in the FSAR are still applicable.

6.3.2 Uncontrolled Control Rod Assembly Withdrawal At Power

Introduction

An uncontrolled control rod assembly withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature. A discussion of this incident is presented in Section 14.1.2 of the FSAR.

Method of Analysis

This transient is analyzed by the LOFTRAN code. The core limits as illustrated in Figure 3 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

* 1 pcm = 10^{-5} $\Delta k/k$

This accident is analyzed with the improved thermal design procedure described in Reference 2. Plant characteristics and initial conditions are listed in Table 2. For an uncontrolled rod withdrawal at power accident, the following conservative assumptions are made:

A. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Section 5.0 of Attachment B.

B. Reactivity coefficients - two cases are analyzed:

    1. Minimum Reactivity Feedback. A +5 pcm/°F moderator temperature coefficient of reactivity and at least negative Doppler only power coefficient (see Table 2) are assumed.

    2. Maximum Reactivity Feedback. A conservatively large negative moderator temperature coefficient and a most negative Doppler only power coefficients (See Table 2) are assumed.

C. The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

D. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

E. The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks having the maximum combined worth at maximum speed.

Results

Figures 4 through 6 show the transient response for a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and margin to DNB is maintained.

The transient response for a slow RCCA withdrawal from full power is shown in Figures 7 through 9. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 10 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for minimum and maximum reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT channels. The minimum DNBR is always greater than the limit value.

Figures 11 and 12 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10 percent power respectively. The results are similar to the 100 percent power case, except as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the limit value.

The shape of the curves of minimum DNBR versus reactivity insertion rate in the referenced figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates, i.e., the minimum value of DNBR is always larger than the limit value for all fuel types.

6.3.3 Rod Cluster Control Assembly Misalignment

Rod cluster control assembly misalignment accidents include:

A. A dropped RCCA

B. A dropped RCCA bank

C. Statically misaligned RCCA

Each RCCA has a position indicator channel which displays position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by rod bottom light. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups of four mechanisms each, except control bank D which is divided into two groups of two. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the secondary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA or RCCA bank is detected by:

A. Sudden drop in the core power level as seen by the nuclear instrumentation system;

B. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

C. Rod at bottom signal;

D. Rod deviation alarm;

E. Rod position indication.

Misaligned RCCA are detected by:

A. Asymmetric power distribution as seen on out-of-core neutron detectors or core exit thermocouples;

B. Rod deviation alarm;

C. Rod position indicators.

The resolution of the rod position indicator channel is ±5 percent (±7.2 inches of span). Deviation of any assembly from its group by twice this distance will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation before it can exceed ten (10) percent of span. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications.

Analysis of Effects and Consequences

Dropped RCCA's dropped RCCA bank, and statically misaligned RCCA.

Method of Analysis

a. One or more dropped RCCAs from the same group.

For evaluation of the dropped RCCA event, the transient system response is calculated using the LOFTRAN code. The code simulates the neutron kinetics, Reactor Coolant System, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 9.

b. Statically Misaligned RCCA

Steady state power distributions are analyzed using the methodology described in Reference 9. The peaking factors are then used as input to the THINC code to calculate the DNBR.

<u>Results</u>

a. One or more Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion which may be detected by the power range negative neutron flux rate trip circuitry. If detected, the reactor is tripped within approximately 2.5 seconds following the drop of the RCCAs. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

For those dropped RCCAs which do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the Rod Control System detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figure 13 shows a typical transient response to a dropped RCCA (or RCCAs) in automatic control. Uncertainties in the initial condition are included in the DNB evaluation as described in Reference 9. In all cases, the minimum DNBR remains above the limit value.

b. Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.5 seconds following the drop of a RCCA Bank. The core is not adversely affected during this period, since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following the incident.

c. Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with bank D inserted to its full insertion limit and one RCCA fully withdrawn, DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values including uncertainties (as given in Table 2) but with the increased radial peaking factor associated with the misaligned RCCA.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for RCCA ejection analysis. Inspection of the power shapes shows that the DNB and peak kw/ft situation is less severe than the bank D case discussed above assuming insertion limits on the other banks equivalent to a bank D full-in insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, including uncertainties (as given in Table 2) but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of a RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

6.3.4 Chemical and Volume Control System Malfunction

Reactivity can be added to the core by feeding primary grade water into the Reactor Coolant System via the reactor makeup portion of the chemical and volume control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the Reactor Coolant System. The Chemical and Volume Control System (CVCS) is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the Primary Water Makeup Control Valve provides the only supply of makeup water to the Reactor Coolant System which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this valve. In order for makeup water to be added to the Reactor Coolant System, at least one charging pump must also be running in addition to the primary water pumps.

The rate of addition of unborated water makeup to the Reactor Coolant system is limited by the capacity of the primary water pumps. The maximum addition rate in this case is 225 gpm with both primary water pumps running. The 225 gpm reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is the maximum delivery based on the unit piping layout. Normally, only one primary water supply pump is operating while the other is on standby.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the preset flow rates of boric acid and primary grade water on the control board.

In order to dilute, two separate operations are required. First, the operator must switch from the automatic makeup mode to the dilute mode; second, the start button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very remote.

Information on the status of reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of the plant operation, boron dilution during refueling, startup, and power operation were examined. In all cases, the conclusion presented in FSAR Section 14.1.5 remains applicable. That is, sufficient time is available for the operator to take corrective action.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.60%$\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. With T_{avg} <350°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 $\pm$ 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due

6.3.5 Loss Of Reactor Coolant Flow (Including Locked Rotor Analysis)

A loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature which is magnified by the positive MTC. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly. The trip systems available to mitigate the consequence of this accident are discussed in the FSAR.

Simultaneous loss of electrical power to all reactor coolant pumps at full power is the most severe credible loss of flow condition. For this condition reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent RCS overpressurization and the DNB ratio from exceeding the limit values.

Method of Analysis

The following loss of flow cases are analyzed:

1. Loss of four pumps from nominal full power conditions with four loops operating.

2. Loss of one pump from nominal full power conditions with four loops operating.

The normal power supplies for the pumps are four buses connected to the generator. Each bus supplies power to one pump. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps will continue to supply coolant flow the the core. The simultaneous loss of power to all reactor coolant pumps is a highly unlikely event. Since each pump is on a separate bus, a single bus fault would not result in the loss of more than one pump.

A full plant simulation is used in the analysis to compute the core average and hot spot heat flux transient responses, including flow coastdown, temperature, reactivity and control rod insertion effects.

These data are then used in a detailed thermal-hydraulic computation to compute the margin to DNB using ITDP. This computation solves the continuity, momentum and energy equations of fluid flow together with the appropriate DNB correlation, W-3 or WRB-1.

Uncertainties in initial conditions are included in the limit DNBR as described in Section 5.0 of Attachment B. The initial conditions used are listed in Table 2.

This transient is analyzed by three digital computer codes. First the LOFTRAN code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN.

Finally, the THINC code is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell for each type of fuel.

Results

Figures 15 through 18 show the transient response for the loss of power to all RCPs with four loops in operation. The reactor is assumed to be tripped on undervoltage signal. Figure 18 shows the DNBR to be always greater than the limit value for the most limiting fuel assembly cell.

Figures 19 through 22 show the transient response for the loss of one RCP with four loop operation. The reactor is assumed to be tripped on low flow signal. Figure 22 shows the DNBR to be always greater than the limit value for the most limiting fuel assembly cell.

Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperature do not increase significantly above their respective initial values.

Conclusions

The analysis shows that the DNBR will not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

Locked Rotor Accident

A transient analysis has been performed for the instantaneous seizure of a reactor coolant pump rotor. A description of this accident is found in the FSAR Section 14.1.6. Only one locked rotor occurring with four loops operating was analyzed. The initial conditions are listed in Table 2.

Method of Analysis

Two digital-computer codes are used to analyze this transient. The LOFTRAN code is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins 1 second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

3.4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band ($\pm 5\%$ or $\pm 3\%$ flux difference units) about a target flux difference.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

A. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. Above 90% or 0.9 x APL** (whichever is less) of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2
**APL is the Allowable Power Level defined in Specification 3.2.6.

Although these operation are expected to occur and would result in a lower peak RCS pressure, and additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are full open at 2575 psia.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core, and therefore an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium water reaction.

In the evaluation, the rod power at the hot spot is assumed to be 3.0 times the average rod power (i.e., $F_Q = 3.0$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperatures (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 BTU/hr-ft^2-°F at the initiation of the transient. Thus the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

The zirconium-steam reaction is calculated using the methodology described in Section 14.1.6 of the FSAR.

Results

The transient results for the locked rotor accident are shown in Figures 23 through 26. The peak RCS pressure (2587 psia) reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature (1983°F) is considerably less than 2700°F.

For the most limiting fuel assembly, less than three percent (3%) of the rods reach a DNBR value less than the appropriate minimum value.

Conclusions

A. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted conditions stress limits, the integrity of the primary coolant system is not endangered.

LIMITING CONDITION FOR OPERATION (Continued)

c) Surveillance testing of the APDMS may be performed pursuant to Specification 4.3.3.6.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 6 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

d. During power reductions using control rods, the reporting requirements of Specifications 6.9.1.9 shall not apply provided the action items above are satisfied.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

B. Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F, the core will remain in place and intact with no loss of core cooling capability.

6.3.6 Loss of External Electrical Load

The loss of external electrical load may result from an abnormal variation in network frequency or other adverse network operating conditions. It may also result from a trip of the turbine generator or in an unlikely opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large NSSS load reduction by the action of the turbine control. A further discussion is found in FSAR Section 14.1.8.

Method of Analysis

The loss of load transients are analyzed by employing the detailed digital computer program LOFTRAN. The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the improved thermal design procedures. Plant characteristics and initial conditions are listed in Table 2.

Major assumptions are summarized below:

A. Initial Operating Conditions - initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit departure from nucleate boiling ratio (DNBR) as described in Section 5.0 of Attachment B.

B. Moderator and Doppler Coefficients of Reactivity - the loss of load is analyzed with both maximum and minimum reactivity feedback. The maximum feedback case assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback cases assume a +5 pcm/°F MTC and the least negative Doppler coefficients.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. A penalty deviation of one-half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% ofRATED THERMAL POWER.

4.2.1.3 The target axial flux difference of each OPERABLE excore channel shall be determined in conjunction with the measurement of $F_Q^M(Z)$ as defined in Specification 4.2.2.2.c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q^M(Z)$ as defined in Specification 4.2.2.2.c. The allowable values of the target band are $\pm5\%$ or $\pm3\%$. Redefinition of the target band from $\pm3\%$ to $\pm5\%$ between determinations of the target axial flux difference is allowed when appropriate rediefinitions of APL are made. Redefinition of the target band from $\pm5\%$ to $\pm3\%$ is allowed only in conjunction with the determination of a new target axial flux difference. The provisions of Specification 4.0.4 are not applicable.

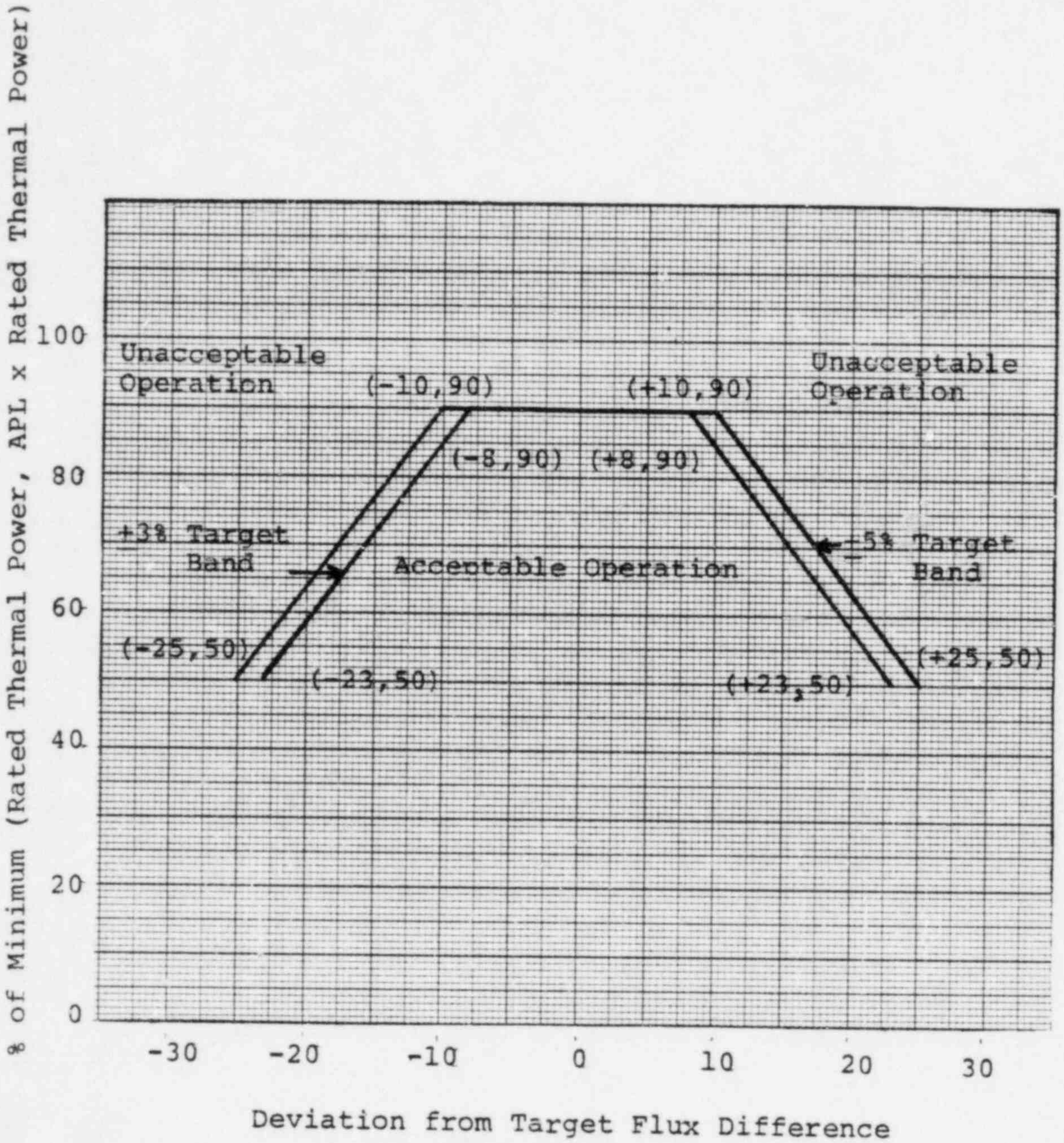

Figure 3.2-1 Allowable Deviation from Target Flux Difference

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z,\ell)$ shall be limited by the following relationships:

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q(Z,\ell) \le [\frac{2.0}{P}]\ [K(Z)]$	$F_Q(Z,\ell) \le [\frac{F_Q^L(E_\ell)}{P}]\ [K(Z)]$	$P > 0.5$
$F_Q(Z,\ell) \le [4.0]\ [K(Z)]$	$F_Q(Z,\ell) \le 2\ [F_Q^L(E_\ell)\ K(Z)]$	$P \le 0.5$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

$F_Q^L(E_\ell)$ is the exposure dependent F_Q limit for rod ℓ and is defined in Figure 3.2-4 for Exxon Nuclear Co. fuel and in Figure 3.2-5 for Westinghouse fuel. E_ℓ is the maximum pellet exposure in rod ℓ. K(Z) is the function obtained from Figure 3.2-3 for Westinghouse fuel and Figure 3.2-2 for Exxon Nuclear Co. fuel. F_Q is defined as the $F_Q(Z,\ell)$ with the smallest margin or the greatest excess of the limit.

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% F_Q exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

LIMITING CONDITION FOR OPERATION (Continued)

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided F_Q is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z,\ell)$ shall be determined to be within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z,\ell)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. This product is defined as $F_Q^M(Z)$.

c. Satisfying the following relationships at the time of the target flux determination.

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q^M(Z) \leq \left[\frac{2.0}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$F_Q^M(Z) \leq \left[\frac{F_Q^L(Z)}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$P > 0.5$
$F_Q^M(Z) \leq \left[\frac{4.0}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$F_Q^M(Z) \leq \left[\frac{2\ F_Q^L(Z)}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$P \leq 0.5$

SURVEILLANCE REQUIREMENTS (Continued)

where

$F_Q^M(Z) = F_Q(Z,\ell)$ at ℓ for which

$\frac{F_Q(Z,\ell)}{T(E_\ell)}$ is a maximum

$F_Q^L(Z) = F_Q^L(E_\ell)$ at ℓ for which

$\frac{F_Q(Z,\ell)}{T(E_\ell)}$ is a maximum

$F_Q^M(Z)$ and $F_Q^L(Z)$ are functions of core height, Z, and correspond at each Z to the rod ℓ for which $\frac{F_Q(Z,\ell)}{T(E_\ell)}$ is a maximum at that Z

V(Z) is a cycle dependent function and is provided in the Peaking Factor Limit Report. K(Z) is defined in Figure 3.2-2 for Exxon Nuclear Company fuel and in Figure 3.2-3 for Westinghouse fuel. $T(E_\ell)$ is defined in Figures 3.2-4 and 3.2-5. $E_p(Z)$ is an uncertainty factor to account for the reduction in the $F_Q^L(E_\ell)$ curve due to accumulation of exposure prior to the next flux map.

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$E_p(Z) = 1.0$	$E_p(Z) = 1.0$	$0 \leq E_\ell \leq 17.62$
$E_p(Z) = 1.0$	$E_p(Z) = 1.0 + [.0040 \times F_Q^M(Z)]$	$17.62 < E_\ell \leq 34.5$
$E_p(Z) = 1.0$	$E_p(Z) = 1.0 + [.0093 \times F_Q^M(Z)]$	$34.5 < E_\ell \leq 42.2$

SURVEILLANCE REQUIREMENTS (Continued)

d. Measuring $F_Q(Z,\ell)$ in conjunction with a target flux difference and target band determination, according to the following schedule:

   1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined*, or

   2. At least once per 31 effective full power days, whichever occurs first.

   *During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

e. With successsive measurements indicating an increase in max over Z of $[\frac{F_Q^M(Z)}{K(Z)}]$ with exposure, either of the following additional actions shall be taken:

   1. $F_Q^M(Z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

   2. $F_Q^M(Z)$ shall be measured and a target axial flux difference reestablished at least once per 7 effective full power days until 2 successive maps indicate that max over Z of $[\frac{F_Q^M(Z)}{K(Z)}]$ is not increasing.

f. With the relationship specified in 4.2.2.2.c not being satisfied, either of the following actions shall be taken:

   1. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied and remeasure the target axial flux difference.

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z,\ell)$ exceeding its limit by the maximum percent calculated with the following expressions with V(Z) corresponding to the target band and $P \geq 0.5$:

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z) \times E_p(Z)}{\frac{F_L^Q(E_\ell)}{P} \times [K(Z)]}\right] - 1\right] \times 100 \quad \text{Exxon Nuclear Co Fuel}$$

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z) \times E_p(Z)}{\frac{2.0}{P} \times [K(Z)]}\right] - 1\right] \times 100 \quad \text{WESTINGHOUSE FUEL}$$

g. The limits specified in 4.2.2.2.c and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 10% inclusive.

2. Upper core region 90% to 100% inclusive.

4.2.2.3 When $F_Q(Z,\ell)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(Z,\ell)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

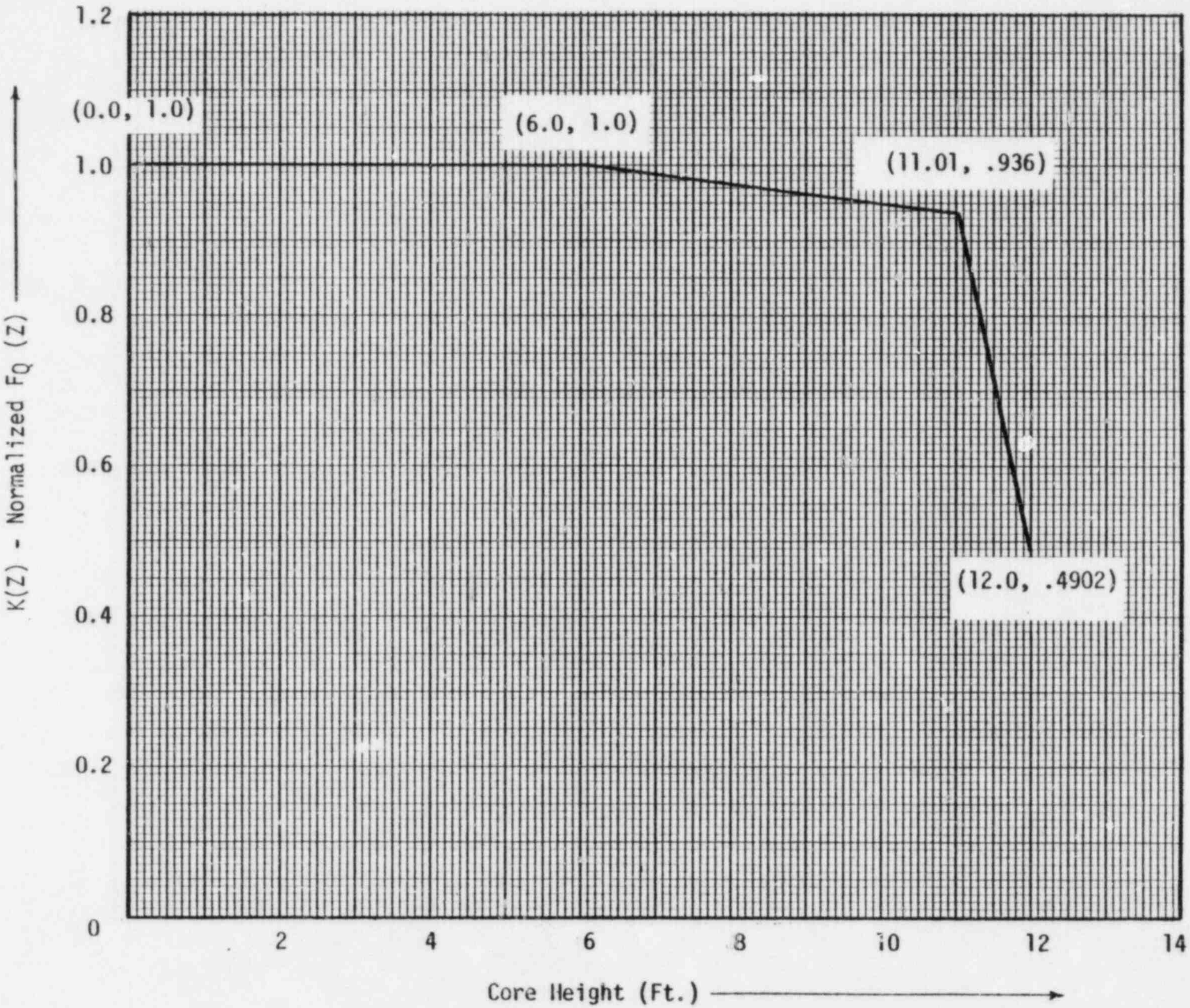

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ As A Function of Core Height for Exxon Nuclear Company Fuel

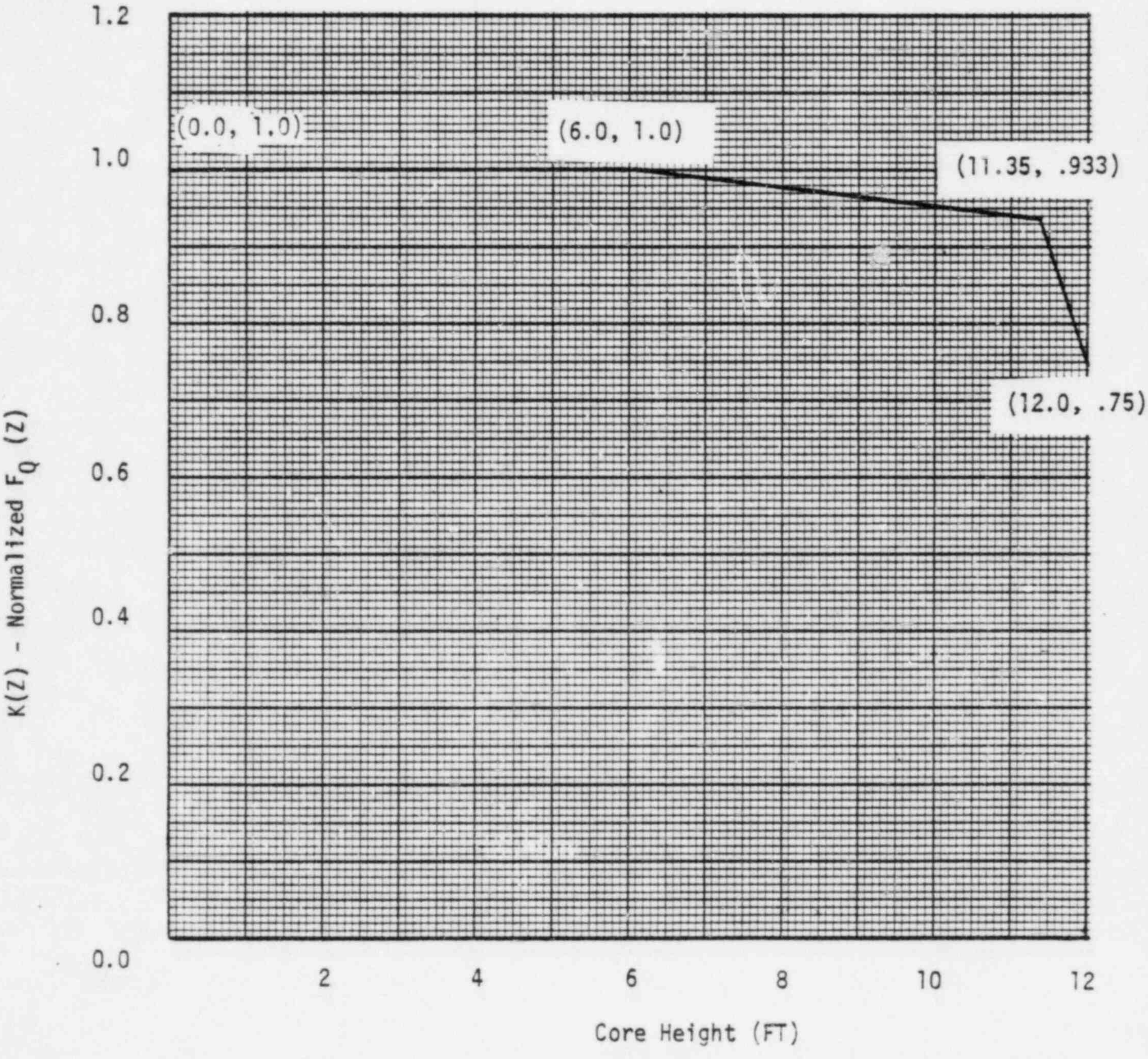

Figure 3.2-3 K(Z) - Normalized F_Q (Z) As A Function of Core Height For Westinghouse Fuel

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationships:

$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$ (for Westinghouse fuel)

and $F^N_{\Delta H} = 1.45\ [1 + 0.2\ (1-P)]$ (for Exxon Nuclear Co. fuel)

where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but $\leq$ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified at 95% or greater RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per month.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 4 Loops In Operation at RATED THERMAL POWER	LIMITS: 4 Loops In Operation at DESIGN THERMAL POWER	LIMITS: 3 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System T_{avg}	$\leq$ 570.5°F	$\leq$ 579.8°F	$\leq$ 570.5°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq 1.386 \times 10^8$ lbs/hr	$\geq 1.386 \times 10^8$ lbs/hr	$\geq 0.9917 \times 10^8$ lbs/hr

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

Westinghouse Fuel

$$[F_j(Z)]_s = \frac{[2.0]\ [K(Z)]}{(\overline{R_j})(P_L)(1.03)(1 + \sigma_j)(1.07)F_p}$$

Exxon Nuclear Co. Fuel

$$[F_j(Z)]_s = \frac{[2.04]\ [K(Z)]}{(\overline{R_j})(P_L)(1.03)(1 + \sigma_j)(1.07)F_p}$$

where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained for a given core height location from Figure 3.2-2 for Exxon Nuclear Company fuel and from Figure 3.2-3 for Westinghouse fuel.

d. R_j, for thimble j, is determined from at least n=6 in-core flux maps covering the full configuration of permissible rod patterns at 100% or APL (whichever is less) of RATED THERMAL POWER in accordance with:

$$\overline{R_j} = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

where:

$$R_{ij} = \frac{F_{Qi\ell}^{Meas}/T(E\ell)}{[F_{ij}(Z)]_{Max}}$$

R_{ij} and its associated σ_i may be calculated on a full core or a limiting fuel batch basis as defined on page B 3/4 3-3 of basis.

LIMITING CONDITION FOR OPERATION (Continued)

e. $F_{Qi\ell}^{Meas}$ is the limiting total peaking factor in flux map i. The limiting total peaking factor is that factor with least margin to the $F_Q^L(E\ell)$ curve defined in Figure 3.2-4 for Exxon Nuclear Company fuel and in Figure 3.2-5 for Westinghouse fuel.

For Exxon Nuclear Company fuel, $T(E\ell)$ is the ratio of the exposure dependent $F_Q^L(E)$ to 2.04 and is defined in Figure 3.2-4. $T(E\ell)$ is equal to 1.0 for fuel supplied by Westinghouse Electric Corporation as given in Figure 3.2-5.

f. $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a limiting total measured peaking factor without uncertainties or densification allowance of $F_{Qi\ell}^{Meas}$.

σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} \left[\overline{R}_j - R_{ij}\right]^2\right]^{1/2}}{\overline{R}_j}$$

The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system respectively.

The factor 1.03 is the engineering uncertainty factor.

g. F_p is an uncertainty factor for Exxon fuel to account for the reduction in the $F_Q^L(E_\ell)$ curve due to an accumulation of exposure prior to the next flux map. The following F_p factor shall apply:

C. Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

D. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the minimum and maximum moderator feedback cases are analyzed:

   1. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

   2. No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

<u>Results</u>

The transient reponses for a loss of load from full power operation are shown for four cases: two cases for minimum reactivity feedback and two cases for maximum reactivity feedback (Figures 27 through 34).

Figures 27 and 28 show the transient responses for the loss of load with minimum reactivity feedback assuming full credit for the pressuirzer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip signal.

The minimum DNBR remains well above the limit value. The pressurizer safety valves are not actuated for this case since primary system pressure remains well below the design value. The steam generator safety valves prevent overpressurization of the secondary system, maintaining pressure below 110 percent of design value.

Figures 29 and 30 show the responses for the total loss of steam load with maximum reactivity feedback. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. Pressurizer relief valves and steam generator safety valves prevent overpressurization in primary and secondary systems, respectively. The pressurizer safety valves are not actuated for this case.

The loss of load accident was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief values, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 31 and 32 show the transients with minimum reactivity feedback. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR never goes below its initial value throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system pressure below 110 percent of the design value.

Figures 33 and 34 are the transients with maximum reactivity feedback with the other assumptions being the same as in the preceding case. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit primary pressure.

Conclusions

Results of the analyses show that the plant design is such that a loss of load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the limit value.

6.3.7 Loss of Normal Feedwater Flow

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The worst postulated loss of normal feedwater event is one initiated by a loss of offsite AC power which is described in FSAR subsection 14.2.12. This is due to the decreased capability of the reactor coolant to remove residual core heat as a result of the RCP coastdown.

The reactor trip on low-low water level in any steam generator provides the necessary protection against a loss of normal feedwater.

The auxiliary feedwater system is started automatically. The turbine driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor driven auxiliary feedwater pumps are supplied by power from the diesel generators if a loss of offsite power occurs. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

An analysis of the system transient is presented below to show that following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or uncovering the core, and returning the plant to a safe condition due to the increased design power level.

Method of Analysis

A detailed analysis using the LOFTRAN code is performed in order to obtain the plant transient following loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

A. The plant is initially operating at 102 percent of the design power rating (3425 MWt NSSS).

B. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

C. Reactor trip occurs on steam generator low-low level.

D. The worst single failure in the auxiliary feedwater system occurs (e.g., failure of turbine drive auxiliary feedwater pump).

E. Auxiliary feedwater is delivered to two steam generators at a rate of 450 gpm.

F. Secondary system steam relief is achieved through the steam generator safety valves.

G. The initial reactor coolant average temperature is 4°F higher than the nominal value, and initial pressurizer pressure is 30 psi higher than nominal.

An additional assumption made for the loss of normal feedwater evaluation is that only the pressurizer safety valves are assumed to function normally. Operation of the valves maintains peak RCS pressure close to or below the actuation setpoint (2500 psia) throughout the transient.

LIMITING CONDITION FOR OPERATION (Continued)

Westinghouse Fuel	ENC Fuel	
$F_p = 1.0$	$F_p = 1.0$	$0 \leq E_\ell \leq 17.62$
$F_p = 1.0$	$F_p = 1.0 + [.0015 \times W]$	$17.62 < E_\ell \leq 34.5$
$F_p = 1.0$	$F_p = 1.0 + [0.0033 \times W]$	$34.5 < E_\ell \leq 42.2$

where W is the number of effective full power weeks (rounded up to the next highest integer) since the last full core flux map.

APPLICABILITY: Mode 1 above the minimum percent of RATED THERMAL POWER indicated by the relationships.*

$$APL = \text{min over } Z \text{ of } \frac{2.0 \times K(Z)}{F_Q(Z,\ell) \times V(Z)} \times 100\ \% \qquad \text{Westinghouse Fuel}$$

$$APL = \text{min over } Z \text{ of } \frac{F_Q^L(E_\ell) \times K(Z)}{F_Q(Z,\ell) \times V(Z) \times E_p(Z)} \times 100\ \% \qquad \text{Exxon Nuclear Co. Fuel}$$

where $F_Q(Z,\ell)$ is the measured $F_Q(Z,\ell)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty, at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the function given in the Peaking Factor Limit Report. The above limit is not applicable in the following core plane regions.

1. Lower core region 0% to 10% inclusive.
2. Upper core region 90% to 100% inclusive.

*The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER 1 percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next 2 hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to APL or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.6 at the following frequencies.

   1. At least once per 8 hours, and

   2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

      a) Increasing the THERMAL POWER above APL of RATED THERMAL POWER, or

      b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

   1. At least once per 8 hours, and

   2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

      a) Increasing the THERMAL POWER above APL of RATED THERMAL POWER, or

      b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

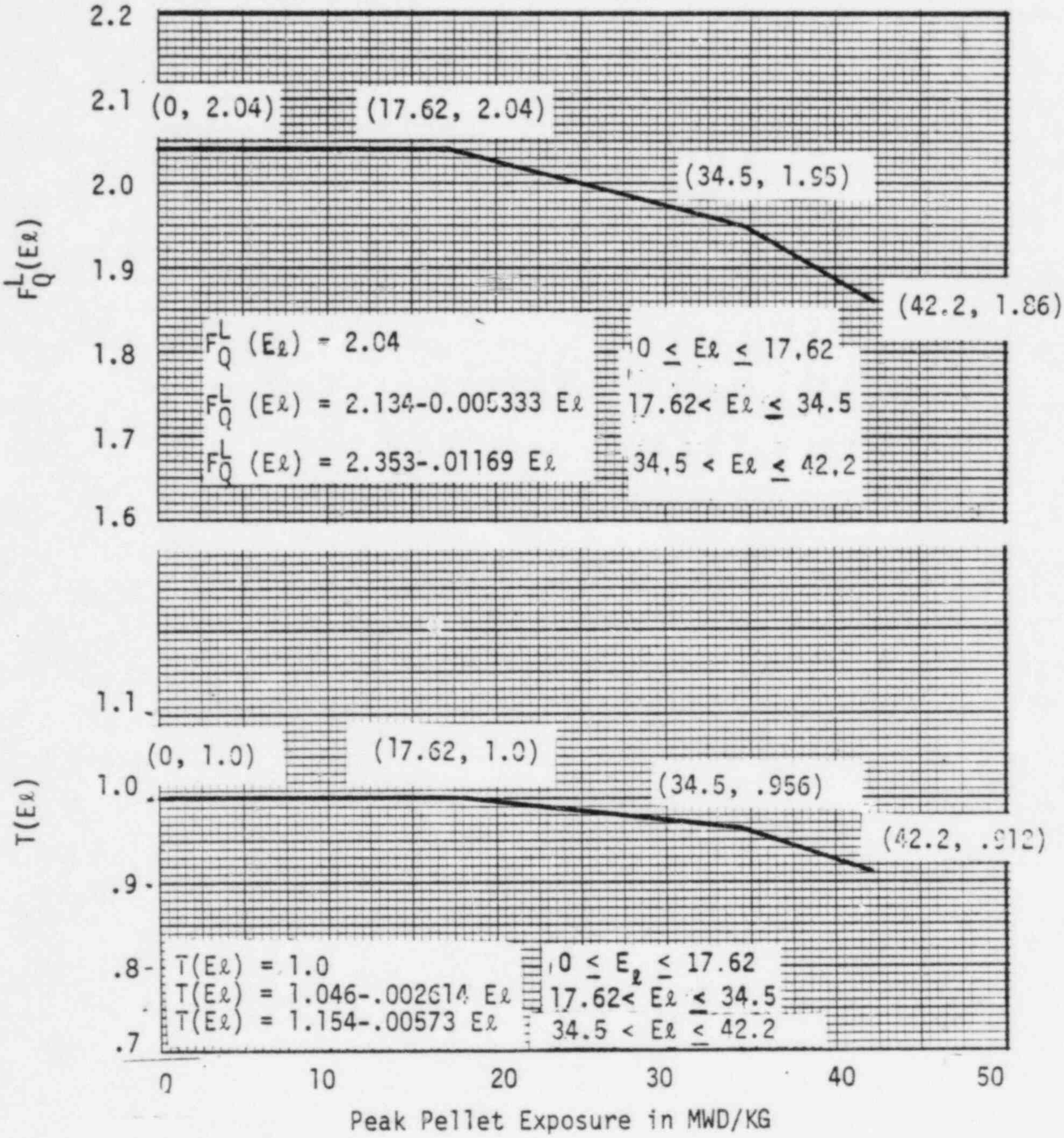

FIGURE 3.2-4

Exposure Dependent F_Q Limit, F_Q^L $(E\ell)$, and Normalized Limit $T(E\ell)$ as a function of Peak Pellet Burnup for Exxon Nuclear Company Fuel

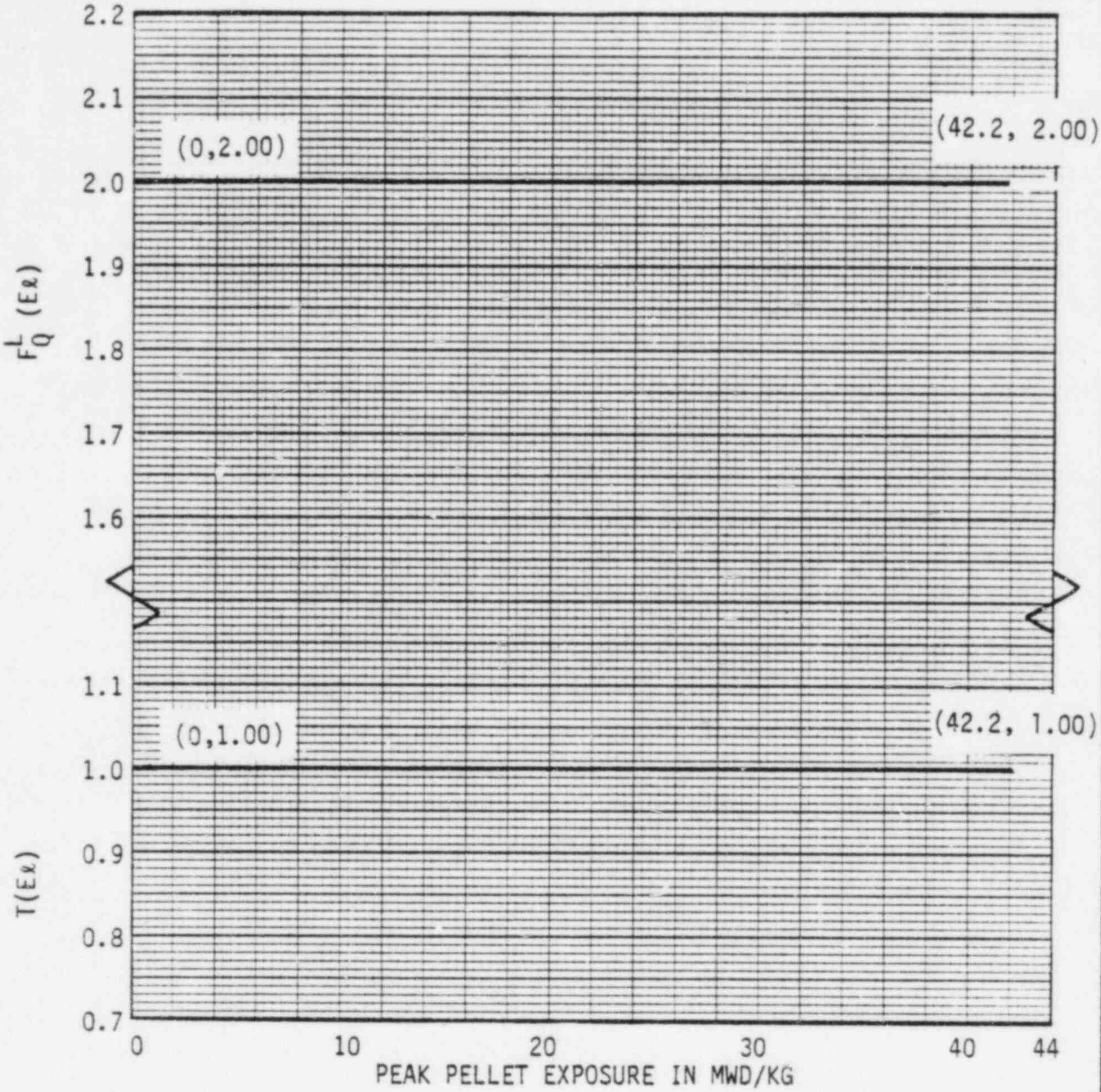

FIGURE 3.2-5

Exposure Dependent F_Q Limit, F_Q^L (E_ℓ), and Normalized Limit $T(E_\ell)$ as a Function of Peak Pellet Burnup for Westinghouse Fuel

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA, the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable, but will normally only be determined periodically, as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits, provided conditions (a) through (d) above are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system, and 3% is the appropriate allowance for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for, and 4% is the appropriate allowance for a full core map taken with the incore detection system. This 4% measurement uncertainty has been included in the design DNBR limit value. The specified limit for $F^N_{\Delta H}$ also contains an additional 4% allowance for uncertainties. The total allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, affect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q, by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

A burnup dependent F_Q is specified as a result of the ECCS evaluation, in accordance with 10 CFR Part 50 Appendix K and to meet the acceptance criteria of 10 CFR 50.46. The basis for this dependence is given in document XN-76-51, Supplements 1, 2, 3, and 4 for Exxon fuels and the exemption granted by the Commission on May 18, 1978 for Westinghouse fuel.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02, but less than 1.09, is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

Results

Figures 35 and 36 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generators void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, at least one auxiliary feedwater pump is automatically started, reducing the rate of water level decrease.

Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves.

6.3.8 Excessive Heat Removal due to Feedwater System Malfunctions

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the Reactor Coolant System. The Overpower-Overtemperature Protection (high neutron flux, overpower ΔT, and overtemperature ΔT trips) prevents any power increase which could lead to DNBR less than minimum allowable value by the Steam Generator Hi-Hi Level Protection.

Excessive feedwater flow may be caused by full opening of a feedwater control valve due to a Feedwater Control system malfunction or an operator error. At power conditions this excess flow causes a greater load demand on the Reactor Coolant System due to increased subcooling in the steam generator. With the plant at no load conditions, the addition of cold feedwater may cause a decrease in Reactor Coolant System temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady s+ate envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated to be adequate to maintain the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The morthly periodic RCS elbow tap flow measurement is adequate to detect flow degradation and to ensure the correlation of the flow indication channels with measured flow, as determined at the beginning of each cycle using a power balance around the steam generators, such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis. Measurement uncertainties have been accounted for in determining the DNB parameters limit values.

3/4.2.6 AXIAL POWER DISTRIBUTION

The limit on axial power distribution ensures that F_Q will be controlled and monitored on a more exact basis through use of the APDMS when operating above APL of RATED THERMAL POWER. This additional limitation on F_Q is necessary, in order to provide assurance that peak clad temperatures will remain below the ECCS acceptance criteria limit of 2200°F in the event of a LOCA.

The unit may operate with fuel assemblies supplied by the Exxon Nuclear Company and by Westinghouse Electric Corporation. An F_Q limit has been specified for each of these two fuel types.

3/4.3.3.6 AXIAL POWER DISTRIBUTION MONITORING SYSTEM (APDMS)

The OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to 1) monitor the core flux patterns that are representative of the power peaking factor in the limiting fuel rod. The limiting fuel rod is the fuel rod that has the least margin to the exposure dependent F_Q limit curve, and 2) limit the core average axial power profile such that the total power peaking factor F_Q in the limiting fuel rod is maintained within acceptable limits.

R_j factors are used to determine the APDMS setpoint limits $[F_j(Z)]_S$. On a full core basis the R_j and σ_j factors are calculated in accordance with the equations on Pages 3/4 2-18 and 3/4 2-19.

However, near BOC, thimbles not in the region of fuel which contains the limiting total peaking factor, $F_{Qi\ell}$, may not follow the axial power distribution of the hot rod. This situation will manifest itself in the form of large σ_j for thimbles not in the same region as the total peak $F_{Qi\ell}$. In this situation, if the rod with the limiting total peaking factor were to move from one fuel region to another, the neutron flux in the thimble with the smallest σ_j would not necessarily follow the axial power distribution of the power in the new limiting rod.

In order to cope with this difficulty, it is permissible to calculate as many σ_j's and $\bar{R}_j$'s for each thimble as there are fuel types or regions in the core. Each $\bar{R}_j$ and σ_j for a thimble j is to be calculated from the equations on Pages 3/4 2-18 and 3/4 2-19 with the following exception. For each $\bar{R}_j$ and σ_j for thimble j, a different $F_{Qi\ell}$ and $\bar{T}(E)$ shall be used. The different σ_j's and R_j's for thimble j shall be calculated substituting for $F_{Qi\ell}^{meas}$ and T(E) the values pertaining to the limiting peak relative power from each fuel region. Obviously for one of these calculations the limiting peak relative power from one region will be the core limiting total peaking factor.

If this option is chosen, the σ_j set to use for APDMS thimble selection and the R_j set to use for the calculation of $[F_j(Z)]_S$ shall be the set obtained using the limiting peak relative power from the same fuel type as the $F_{Qi\ell}$ from the most recent incore flux map.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the applicable design limit DNBR value during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER, until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above the applicable design limit DNBR values for each fuel type. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (51 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing an RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 188°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any over-pressure conditions which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

Method of Analysis

The excessive heat removal due to a feedwater system malfunction transient is analyzed by using the detailed digital computer code LOFTRAN. This code simulates the neutron kinetics of the reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

The system is analyzed to demonstrate acceptable consequences in the event of an excessive feedwater addition, due to a control system malfunction or operator error which allows a feedwater control valve to open fully. The following cases have been analyzed:

1. Accidental full opening of one feedwater control valve with the reactor at power:

    a. Assuming the reactor in manual control and a +5 pcm/°F MTC. This represents a condition where the unit has the least inherent transient capability.

    b. Assuming reactor control and conservatively large negative moderator coefficient of reactivity. This case gives the largest reactivity feedback and results in the greatest power increase.

2. Accidental full opening of a feedwater control valve with the reactor at no load conditions and assuming a conservatively large negative moderator coefficient of reactivity.

This accident is analyzed with the improved thermal design procedure as described in Section 5.0 of Attachment B. Plant characteristics and initial conditions, are listed in Table 2. The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

A. Initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Section 5.0 of Attachment B.

B. For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase to 150% of nominal feedwater flow to one steam generator.

C. For the feedwater control valve accident at zero load condition, a feedwater control valve malfunction occurs which results in an increse in flow to one steam generator from zero to 100 percent of the nominal full load value.

D. For the zero load condition, feedwater temperature is at a conservatively low value of 32°F.

E. No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

F. The feedwater flow resulting from a fully open control valve is terminated by a steam generator high-high level trip signal which closes all feedwater control and isolation valves, trips the main feedwater pumps and trips the turbine.

Normal reactor control system and engineered safety systems are not required to function. The reactor protection system may function to trip the reactor due to overpower or high-high steam generator water level conditions.

Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in subsection 5.3.1, Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition and therefore, the results of the analysis are not presented here. It should be noted that if the incident occurs with the unit just critical at no load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25 percent of nominal full power.

The full power case (maximum reactivity feedback coefficients automatic rod control) gives the largest reactivity feedback and results in the greatest power increase. Assuming the reactor to be in the manual rod control mode results in a slightly less severe transient. The rod control system is not required to function for an excessive feedwater flow event.

For all excessive feedwater cases continuous addition of cold feedwater is prevented by automatic closure of all feedwater isolation valves on steam generator high-high level signal. In addition, a reactor trip and a turbine trip are initiated.

Following reactor trip and feedwater isolation, the plant will approach a stabilized condition at hot standby. Normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

Transient results, Figures 37 through 40, show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor. The DNBR does not drop below the limit value.

Since the power level rises during the excessive feedwater flow incident, the fuel temperatures will also rise until after reactor trip occurs. The core heat flux lags behind the neutron flux response due to the fuel rod thermal time constant; hence the peak value does not exceed 118 percent of its nominal value (i.e., the assumed high neutron flux trip setpoint). The peak fuel temperature will thus remain below the fuel melting temperature.

The transient results have shown that DNB does not occur at any time during the excessive feedwater flow incident; thus, the ability of the primary coolant to remove heat from the fuel rod is not reduced. The fuel cladding temperature therefore does not rise significantly above its initial value during the transient.

Conclusions

The results of the analysis show that the DNB ratios encountered for an excessive feedwater addition at power are above the limit value; hence, no fuel or clad damage is predicted. Additionally, it has been shown that the reactivity insertion rate which occurs at no load conditions following excessive feedwater addition is less than the maximum value considered in the analyis of the rod withdrawal from a subcritical condition analysis.

6.3.9 Excessive Increase in Secondary Steam Flow

An excessive load increase incident is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a ten percent (10%) step load increase and a five percent (5%) per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals, i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided which blocks the opening of the valves unless a large turbine load decrease or turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

- o Overpower ΔT
- o Overtemperature ΔT
- o Power range high neutron flux

Method of Analysis

This accident is analyzed using the LOFTRAN Code. This code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and feedwater system. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10 percent step load increase from rated load. These cases are as follows:

A. Reactor control in manual with minimum moderator reactivity feedback

B. Reactor control in manual with maximum moderator reactivity feedback

C. Reactor control in automatic with minimum moderator reactivity feedback

D. Reactor control in automatic with maximum moderator reactivity feedback

For the minimum moderator feedback cases, the core has the positive moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve, therefore the least inherent transient response capability. For the maximum moderator feedback cases, the moderator temperature coefficient of reactivitiy has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studies without credit being taken for pressurizer heaters.

This accident is analyzed with the improved thermal design procedure as described in Section 5.0 of Attachment B. Initial reactor power, pressure, and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

Plant characteristics and initial conditions are listed in Table 2.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints.

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required.

Results

Figures 41 through 44 illustrate the transient with the reactor in the manual control mode. As expected, for the minimum moderator feedback case there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the maximum moderator feedback, manually controlled case there is a much large increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the limit value.

Figures 45 through 48 illustrate the transient assuming the reactor is in the automatic control mode. Both the minimum and maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the mininum DNBR remains above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for most of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow.

Conclusions

The analysis presented above shows that for a ten percent (10%) step load increase, the DNBR remains above the limit value, thereby precluding fuel or clad damage. The plant reaches a stabilized condition rapidly following the load increase.

6.3.10 Loss of All AC Power to the Plant Auxiliaries

Identification of Causes and Accident Description

A complete loss of all (non-emergency) AC power (e.g. offsite power and gas turbines) may result in the loss of all power to the plant auxiliaries, i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of the onsite AC distribution system.

This transient is more severe than the loss of load event analyzed because in this case the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip due to: (1) turbine trip; (2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant.

Following a loss of power with turbine and reactor trips, the sequence described below will occur:

A. Plant vital instruments are supplied from emergency DC power sources.

B. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for steam dump. If the steam flow rate through the power relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor.

C. As the no load temperature is approached, the steam generator power-operated relief valves (or safety valves, if the power operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition.

D. The standby diesel generators, started on loss of voltage on the plant emergency busses, begin to supply plant vital loads.

The motor driven auxiliary feedwater pumps are supplied power by the diesels and the turbine-driven pump utilizes steam from the main steam system. Both type pumps are designed to supply rated flow within one minute of the initiating signal even if a loss of all non-emergency AC power occurs simultaneously with loss of normal feedwater. The turbine exhausts the used steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank for delivery to the steam generators.

Following the RCP coastdown caused by the loss of AC power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxililary feedwater in the secondary system. An analysis is presented here to show that the natural circulation flow in the RCS following a loss of AC power event is sufficient to remove residual heat from the core.

Method of Analysis

A detailed analysis using the LOFTRAN Code is performed to obtain the plant transient following a station blackout. The simulation describes the plant thermal kinetics, RCS including the natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent varables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

The assumptions used in the analysis are as follows:

A. The plant is initially operating at 102% of the design rating (3426 mwt NSSS).

B. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

C. A heat transfer coefficient in the steam generator associated with RCS natural circulation following the RCP coastdown.

D. Reactor trip occurs on steam generator low-low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.

E. Auxiliary feedwater is delivered by only one motor driven auxiliary feed pump at 450 gpm.

F. Auxiliary feedwater is delivered to two steam generators.

G. Secondary system steam relief is achieved through the steam generator safety valves.

H. The initial reactor coolant average temperature is 4°F higher than the nominal value, and initial pressurizer pressure is 30 psi higher than nominal.

Plant characteristics and initial conditions listed in Table 2.

Results

The transient response of the RCS following a loss of AC power is shown in Figures 49 and 50.

The LOFTRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or clad damage.

6.3.11 Rupture of a Steam Pipe

The worst case steamline break (Case b of FSAR 14.2.5) was reanalyzed to generate limiting statepoints to verify nuclear design calculations in accordance with Reference 1 methodology. These statepoints are listed in Table 3.

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

A. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN Code has been used.

B. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC, has been used to determine if DNB occurs for the core conditions computed in item A above.

Studies have been performed to determine the sensitivity of steamline break results to various assumptions (Reference 10). Based upon this study, the following conditions were assumed to exist at the time of a main steam line break accident:

A. End-of-life shut down margin (1.60%Δk/k) at no load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

B. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position: The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used is shown in Figure 51.

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculation. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the limiting conditions for the cases analyzed. This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and non-uniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism; i.e., underprediction of negative reactivity feedback from power generation.

C. Minimum capability for injection of high concentration boric acid (20,000 ppm) solution corresponding to the most restrictive single failure in the safety injection system. The Emergency Core Cooling System (ECCS) consists of three systems: 1) the passive accumulators, 2) the low head safety injection (residual heat removal) system, and 3) the high head safety injection (charging) system. Only the safety injection system and the passive accumulators are modeled for the steam line break accident analysis.

The modeling of the safety injection system in LOFTRAN is described in Reference 5. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration borated water, which must be swept from the lines downsteam of the boron injection tank isolation valves prior to the delivery of high concentration boric acid to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 13 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept into core before the 20,000 ppm borated water reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 10-second delay is assumed to start the diesel generators and to commence loading the necessary safety injection equipment onto them.

D. Design value of the steam generator heat transfer coefficient including allowance for fouling factor.

E. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location, would have the same effect on the NSSS as the 1.4 square foot break. The following case has been considered in determining the core power and RCS transients:

Complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

F. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

The analyses assumed initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the reactor coolant system contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

In addition, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of RCS cooldown are more severe than steam line breaks occurring at power.

G. In computing the steam flow during a steam line break, the Moody Curve (4) for fl/D = 0 is used.

Results

Table 3 lists the limiting statepoints for the worst case. The results presented are a conservative indication of the events which would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

Figures 52 through 54 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition (Case b of FSAR Section 14.2.5).

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by high containment pressure signals or low steam line pressure. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steam line stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 54, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) before boron solution at 20,000 ppm enters the RCS. A peak core power less than the nominal full power value is attained.

The calculation assumes the boric acid is mixed with, and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

A DNB analysis was performed for this case. It was found that all cases had a minimum DNBR greater than the limit value.

Conclusions

The analysis has shown that the criteria stated earlier are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis, in fact, shows that no DNB occurs for the rupture assuming the most reactive RCCA stuck in its fully withdrawn position.

5.3.12 Rupture of Control Rod Drive Mechanism Housing (RCCA Ejection)

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. This accident is discussed further in FSAR Chapter 14.2.6.

The limiting criteria is described in Reference 11 and summarized below:

A. Average fuel pellet enthalpy at hot spot below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.

B. Average clad temperature at the hot spot below the temperature at which clad embrittlement may be expected (2700°F).

C. Peak reactor coolant pressure less than that which could cause stresses to exceed the faulted condition stress limits.

D. Fuel melting will be limited to less than ten percent 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of criterion A above.

Method of Analysis

The calculation of the RCCA ejection transient is performed in two stages, first an average core channel calculation and then a hot region calculation . The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 11.

Average Core Analysis

The spatial kinetics computer code, TWINKLE, is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equation in one, two or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculation of pointwise Doppler and moderator feedback effects. In this analysis, the code is used as a one dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. Further description of TWINKLE appears in Section 5.2.1.

Hot Spot Analysis

In the hot spot analysis, the initial heat flux is equal to the nominal times the design hot channel factor. During the transient, the heat flux hot channel factor is linearly increased to the transient value in 0.1 second, the time for full ejection of the rod. Therefore, the assumption is made that the hot spot before and after ejection are coincident. This is very conservative since the peak after ejection will occur in or adjacent to the assembly with the ejected rod, and prior to ejection the power in this region will necessarily be depressed.

The hot spot analysis is performed using the detailed fuel and clad transient heat transfer computer code, FACTRAN. This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative radial power distribution is used within the fuel rod.

FACTRAN uses the Dittus-Boelter or Jens-Lottes correlation to determine the film heat transfer before DNB, and the Bishop-Sandburg-Tong correlation (see Reference 12) to determine the film boiling coefficient after DNB. The Bishop-Sanburg-Tong correlation is conservatively used assuming zero bulk fluid quality. The DNB ratio is not calculated, instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady-state temperature distribution to agree with the fuel heat transfer design codes. Further description of FACTRAN appears in Section 5.2.1.

System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore by calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in the LOFTRAN computer code. This code calculates the pressure transient taking into account fluid transport in the RCS and heat transfer to the steam generators. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

Input parameters for the analysis are conservatively selected on the basis of values calculated for this type of core. The more important parameters are discussed below. Table 4 presents the parameters used in this analysis.

Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using either three dimensional static methods or by a synthesis method employing one dimensional and two dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation to provide worst case results.

Appropriate margins are added to the ejected rod worth and hot channel factors to account for any calculational uncertainties, including an allowance for nuclear power peaking due to densification.

Power distribution before and after ejection for a "worst case" can be found in Reference 11. During plant startup physics testing, ejected rod worths and power distributions are measured in the zero and full power configurations and compared to values used in the analysis. Experience has shown that the ejected rod worth and power peaking factors are consistently overpredicted in the analysis.

Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks occur in channels where the power is higher than average. Since the weight of a region is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one dimensional (axial) spatial kinetics method is employed, axial weighting is not necessary if the initial condition is made to match the ejected rod configuration. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors have also been shown to be conservative compared to three dimensional analysis.

Moderator and Doppler Coefficient

The critical boron concentrations at the beginning of life and end of life are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using a one dimensional steady-state computer code with a Doppler weighting factor of 1.0. The Doppler weighting factor will increase under accident conditions, as discussed above.

Delayed Neutron Fraction, β_{eff}

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.70% at beginning of life and 0.50% at end of life. The accident is sensitive to β_{eff} if the ejected rod worth is equal to or greater than β_{eff} as in zero power transients. In order to allow for future cycles, pessimistic estimates of β_{eff} of 0.55% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 4 and includes the effect of one stuck RCCA adjacent to the ejected rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point is reached before significant shutdown reactivity is inserted into the core. This is particularly important conservatism for hot full power accidents.

The minimum design shutdown margin available for this plant at hot zero power (HZP) may be reached only at end of life in the equilibrium cycle. This value includes an allowance for the worst stuck rod, an adverse xenon distribution, conservative Doppler and moderator defects, and an allowance for calculational uncertainties. Physics calculations have shown that the effect of two stuck RCCAs (one of which is the worst ejected rod) is to reduce the shutdown by about an additional 1% $\Delta k/k$. Therefore, following a reactor trip resulting from an RCCA ejection accident, the reactor will be subcritical when the core returns to HZP.

Depressurization calculations have been performed assuming the maximum possible size break (2.75 inch diameter) located in the reactor pressure vessel head. The results show a rapid pressure drop and a decrease in system water mass due to the break. The emergency core cooling system (ECCS) is actuated on the coincidence of low pressurizer pressure and level within one minute after the break. The RCS pressure continues to drop and reaches saturation (1100 to 1300 psi depending on the system temperature) in about two to three minutes. Due to the large thermal inertia of primary and secondary system, there has been no significant decrease in the RCS temperature below noload by this time, and the depressurization itself has caused an increase in shutdown margin by about 0.2% $\Delta k/k$ due to the pressure coefficient. The cooldown transient could not absorb the available shutdown margin until more than 10 minutes after the break. The addition of highly borated (20,000 ppm) safety injection flow starting one minute after the break is much more than sufficient to ensure that the core remains subcritical during the cooldown.

Reactor Protection

Reactor protection for a rod ejection is provided by high neutron flux trip (high and low setting) and high rate of neutron flu.. increase trip. These protection functions are part of the reactor trip system. No single failure of the reactor trip system will negate the protection functions required for the rod ejection accident, or adversely affect the consequences of the accident.

Results

Table 4 summarizes the results. Cases are presented for both beginning and end of life at zero and full power.

A. Beginning of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively calculated to be 0.17% Δk/k and 6.8 respectively. The peak clad average temperature was 2415°F. The peak spot fuel center temperature reached melting, conservatively assumed at 4900°F. However, melting was restricted to less than 10% of the pellet.

B. Beginning of Cycle, Zero Power

For this condition, Control Bank D was assumed to be fully inserted and banks B and C were at their insertion limits. The worst ejected rod is located in Control Bank D and has a worth of 0.75% Δk/k and a hot channel factor of 12.0. The peak clad average temperature reached 2527°F, the fuel center temperature was 4021°F.

C. End of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively calculated to be 0.20% Δk/k and 7.1 respectively. This resulted in a peak clad average temperature of 2316°F. The peak hot spot fuel center temperature reached melting at 4800°F. However, melting was restricted to less than 10% of the pellet.

D. End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming Control Bank D to be fully inserted and banks B and C at their insertion limits. The results were .80% Δk/k and 20.0 respectively. The peak clad average and fuel center temperatures were 2690°F and 4144°F. The Doppler weighting factor for this case is significantly higher than for the other cases due to the very large transient hot channel factor.

The nuclear power and hot spot fuel and clad temperature transients for two cases (end of life full power and end of life zero power) are presented in Figures 55 through 58.

The ejection of an RCCA constitutes a break in the RCS, located in the reactor pressure vessel head. The effects and consequences of loss of coolant accidents (LOCA) are discussed in subsection 6.0 of Attachment B. Following the RCCA ejection, the operator would follow the same emergency instructions as for any other LOCA to recover from the event.

Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at beginning of life, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits(11). Since the severity of the present analysis does not exceed the "worst case" analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a differential expansion tending to bow the midpoint of the rods toward the hotter side of the rod. Calculations have indicated that this bowing would result in a negative reactivity effect at the hot spot since Westinghouse cores are under-moderated, and bowing will tend to increase the under-moderation at the hot spot. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from the fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analysis.

Conclusions

Even on a pessimistic basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the RCS. The analyses have demonstrated the fission product release as a result of fuel rods entering DNB is limited to less than 10% of the fuel rods in the core.

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TABLE 1

NON-LOCA ACCIDENTS REANALYZED FOR TRANSITION TO WESTINGHOUSE OFA

FSAR SECTION	ACCIDENT
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition
14.1.2	Uncontrolled RCCS Withdrawal at Power
14.1.3	RCCA Misalignment
14.1.4	RCCA Drop
14.1.5	Chemical Volume and Control System Malfunction
14.1.6	Loss of Reactor Coolant Flow (including Locked Rotor Analysis)
14.1.8	Loss of External Load
14.1.9	Loss of Normal Feedwater
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunction
14.1.11	Excessive Load Increase Incident
14.1.12	Loss of All A.C. Power to the Station Auxiliaries
14.2.5	Rupture of a Steam Pipe
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

TABLE 2

Faults	Computer Codes Utilized	Improved Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)	Doppler Power Coefficient (pcm/% power)
Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical Condition	TWINKLE, FACTRAN, THINC	Yes	0	547	2250	min(2)
Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power(1)	LOFTRAN	Yes	3425/2124/411	577.1/565.1 550.0	2250	max(3) and min
Rod Cluster Control Assembly Misalignment	LOFTRAN, TURTLE, THINC, LEOPARD	Yes	3425	577.1	2250	N/A
Uncontrolled Boron Dilution	NA*	NA	0 and 3415	NA	NA	N/A
Loss of Forced Reactor Coolant Flow, Locked Rotor	LOFTRAN, THINC, FACTRAN	Yes	3425	577.1	2250	max
Loss of External Electrical Load and/or Turbine Trip	LOFTRAN	Yes	3425	577.1	2250	max and min
Loss of Normal Feedwater	LOFTRAN	NA	3494	581.1	2280	max
Excessive Heat Removal Due to Feedwater System Malfunctions	LOFTRAN	Yes	3425	577.1	2250	min
Excessive Load Increase Incident	LOFTRAN	Yes	3425	577.1	2250	max and min

*NA - Not Applicable

(1) Multiple power levels and corresponding vessel average temperatures were examined. See Section 5.3.2
(2) Minimum Doppler power defect (pcm/%power) = -10.18 + 0.035Q where Q is in % power.
(3) Maximum Doppler power defect (pcm/%power) = -19.40 + 0.068Q.
(4) The integral of the Doppler power coefficient is used: zero% power defect is zero pcm; 10% power, 1000 pcm; 20% power, 1430 pcm; 30% power, 1700 pcm.

TABLE 2 (Con't)

Faults	Computer Codes Utilized	Improved Thermal Design Procedure	Initial NSSS Thermal Power Output (MWt)	Vessel Average Temperature (°F)	Pressurizer Pressure (PSIA)	Doppler Power Coefficient (pcm/% power)
Loss of Offsite Power to the Station Auxiliaries	LOFTRAN	NA	3494	581.1	2280	max
Rupture of a Steam Line	THINC, LOFTRAN	No	0 (Subcritical)	547.0	2250	(4)
Rupture of a Control Rod Drive Mechanism Housing(1)	TWINKLE, FACTRAN, LOFTRAN, THINC	NA *	3425/0	581.1/547.0	2250	min

*NA - Not Applicable

(1) Multiple power levels and corresponding vessel average temperatures were examined. See Section 5.3.2
(2) Minimum Doppler power defect (pcm/%power) = -10.18 + 0.035Q where Q is in % power.
(3) Maximum Doppler power defect (pcm/%power) = -19.40 + 0.068Q.
(4) The integral of the Doppler power coefficient is used: zero% power defect is zero pcm; 10% power, 1000 pcm; 20% power, 1430 pcm; 30% power, 1700 pcm.

TABLE 3

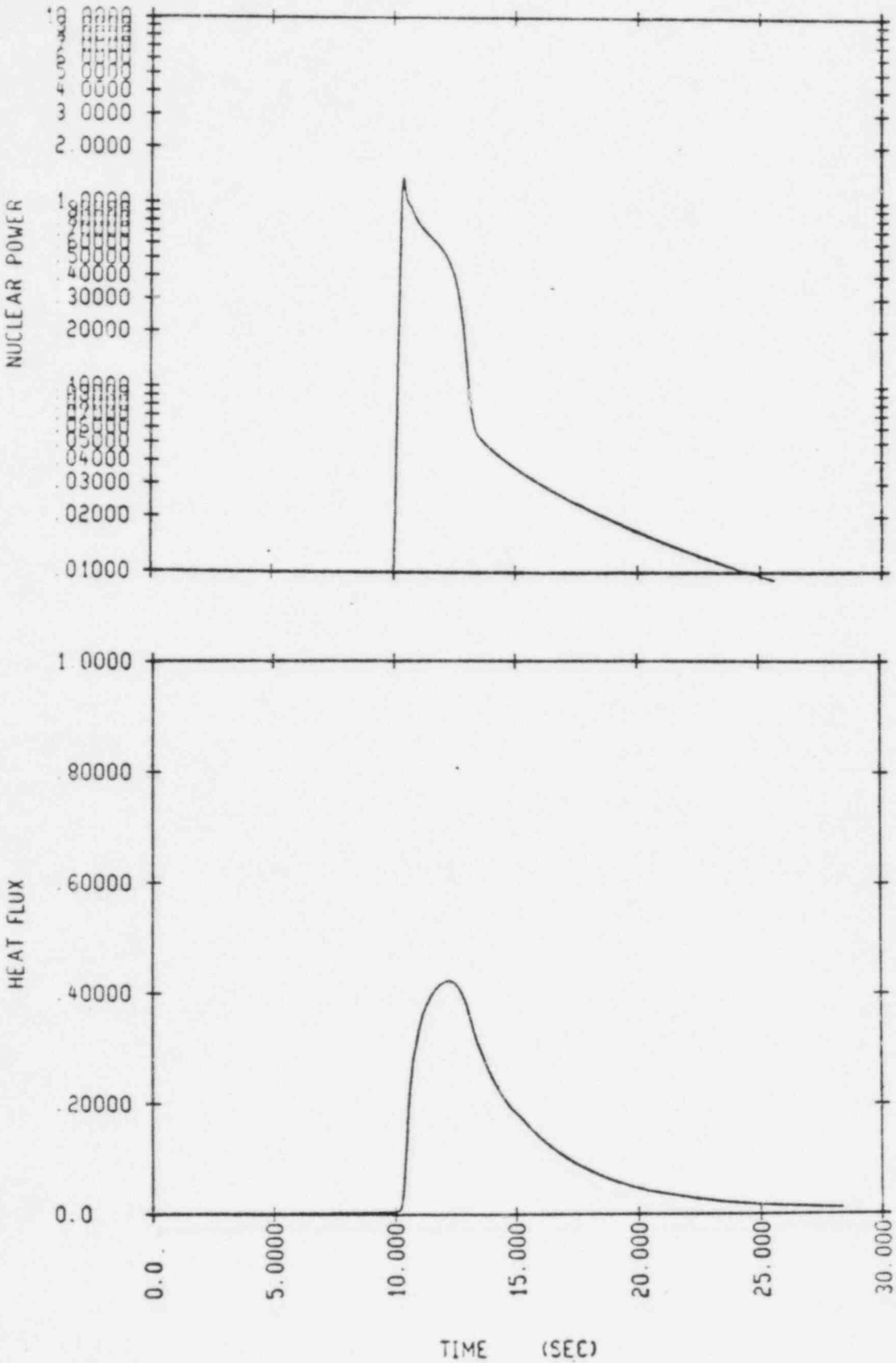
LIMITING STEAMLINE BREAK STATEPOINT
DOUBLE ENDED RUPTURE INSIDE CONTAINMENT
WITH OFFSITE POWER AVAILABLE

Time Sec	Pressure Psia	Heat Flux Fraction	Inlet Cold	Temp Hot	Flow Frac	Boron PPM	Reactivity Per Cent	Density GM/CC
62.20	775.	.053	347.4	469.4	1.00	79.0	.059	.849

TABLE 4

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	HZP Beginning	HFP Beginning	HZP End	HFP End
Power Level (%)	0	102	0	102
Ejected Rod Worth (%Δk)	0.75	0.17	0.8	0.2
Delayed Neutron Fraction (%)	0.0055	0.0055	0.0044	0.0044
Feedback Reactivity Weighting	2.071	1.30	2.755	1.30
Trip Reactivity (%Δk)	2.	4.	2.	4.
F_q Before Rod Ejection	2.50	2.50	2.50	2.50
F_q After Rod Ejection	12.	6.8	20.	7.1
Number of Operational Pumps	2.	4.	2.	4.
Maximum Fuel Pellet Average Temperature (°F)	3472.	4216.	3630.	4092.
Maximum Fuel Center Temperature (°F)	4021.	5016.	4144.	4923.
Maximum Clad Average Temperature (°F)	2527.	2415.	2690.	2316.
Maximum Fuel Stored Energy (cal/gm)	147.2	185.8	152.2	179.2
Fuel Melt in Hot Pellet, %	0	<10	0	<10



FICURE 1

ROD WITHDRAWAL FROM SUBCRITICAL NUCLEAR POWER AND HEAT FLUX VERSUS TIME

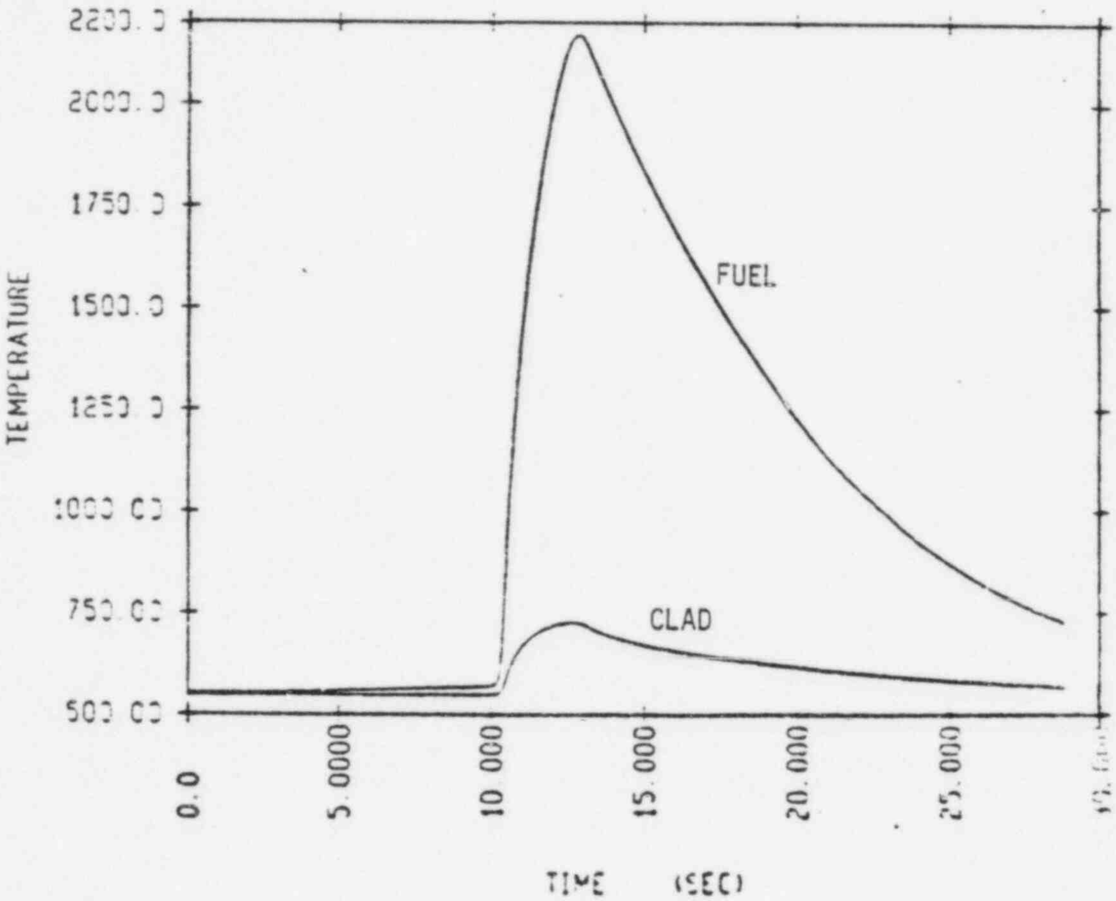
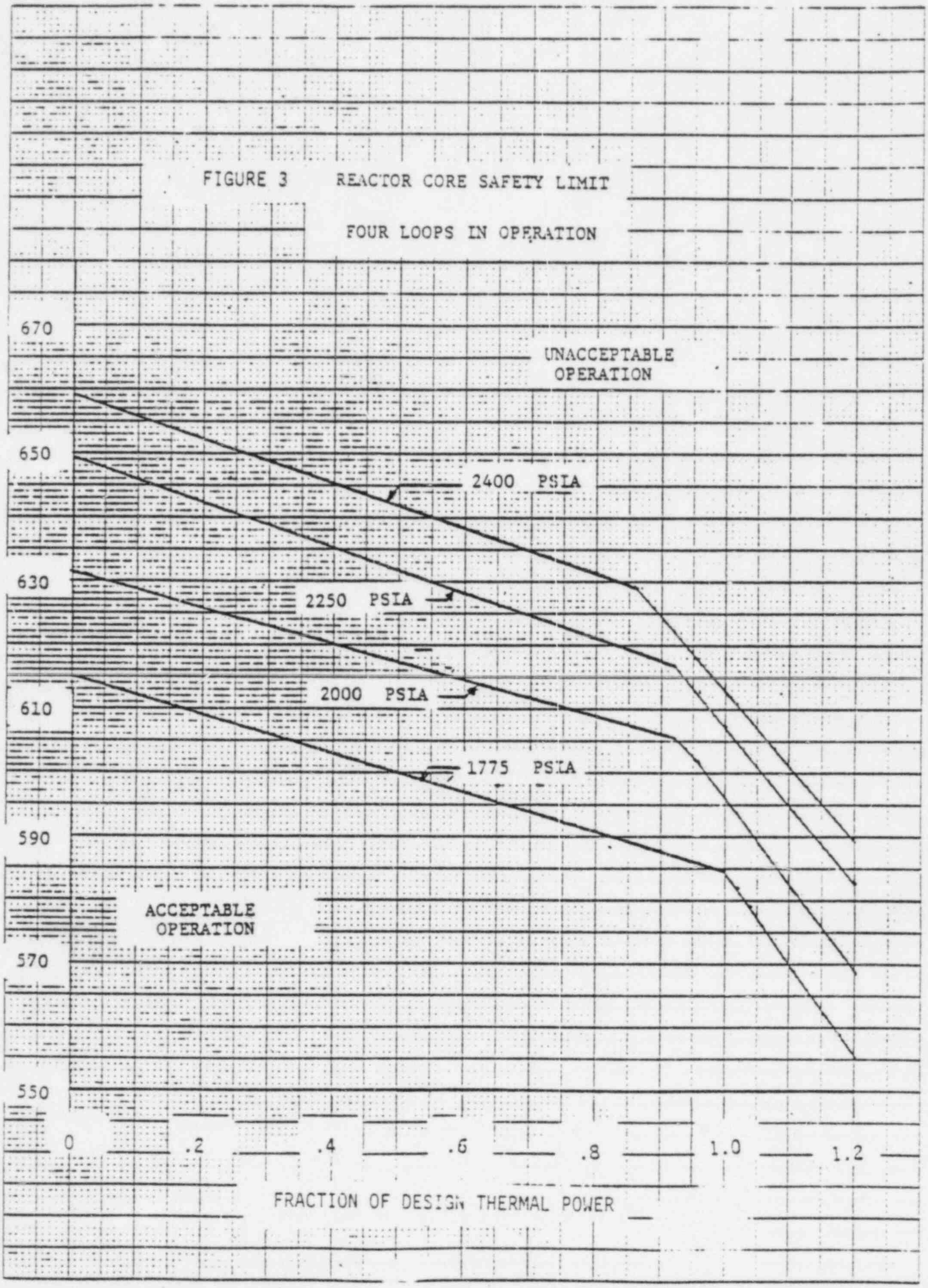


FIGURE 2

RCD WITHDRAWAL FROM SUBCRITICAL FUEL AVERAGE
AND CLAD TEMPERATURE VERSUS TIME



K•E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

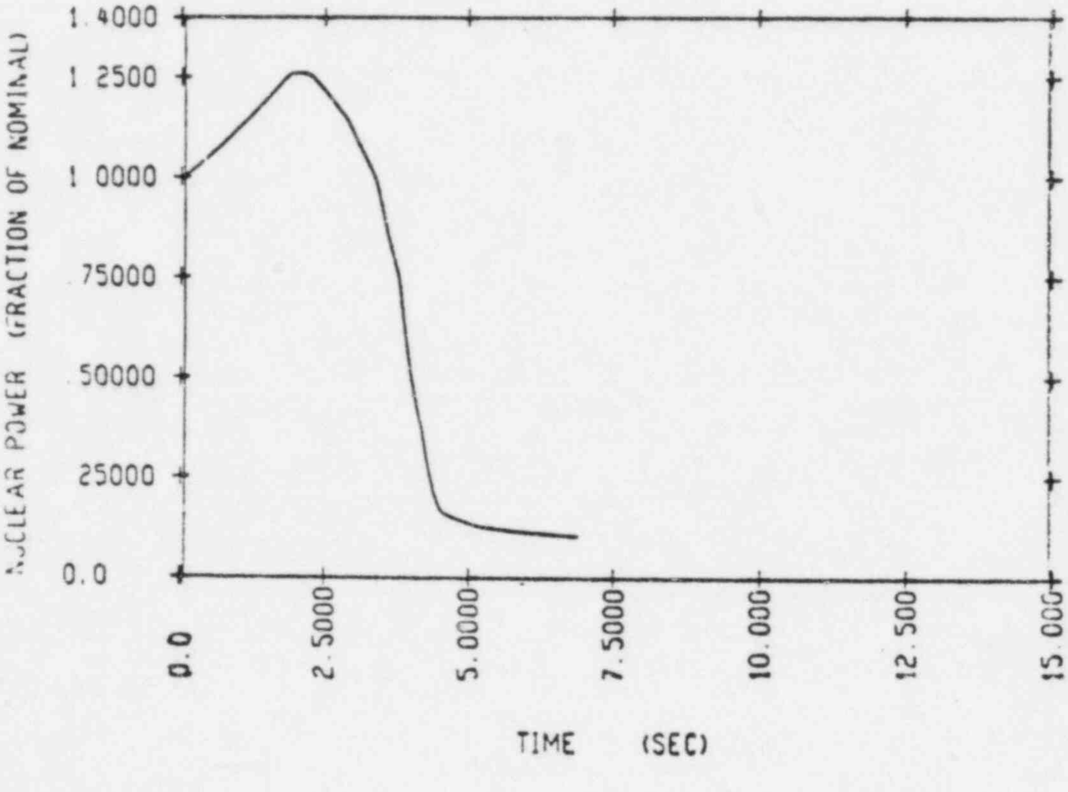


FIGURE 4

ROD WITHDRAWAL AT POWER
FULL POWER, 80 PCM, MINIMUM
REACTIVITY FEEDBACK
NUCLEAR POWER VERSUS TIME

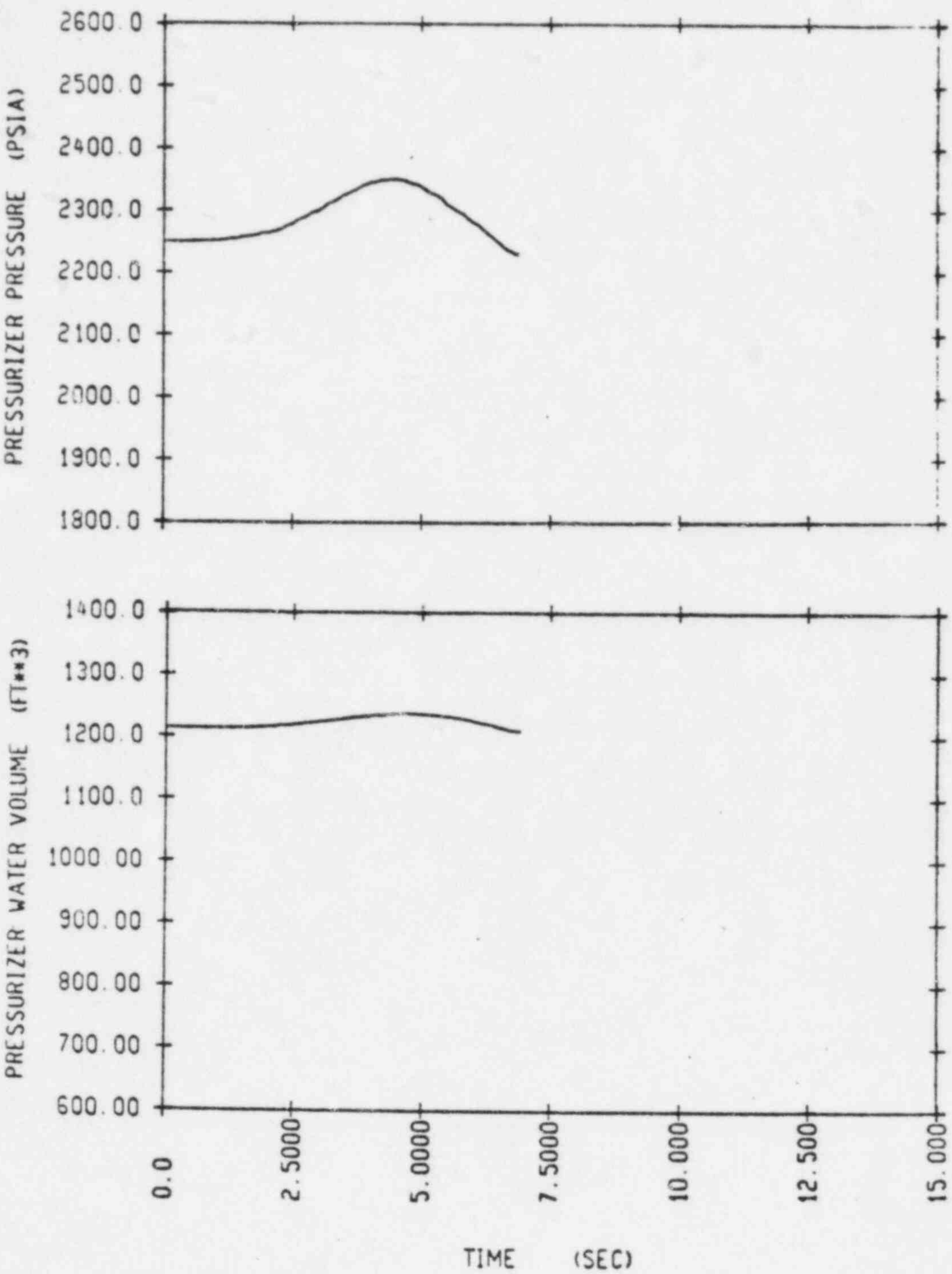


FIGURE 5

ROD WITHDRAWAL AT POWER

FULL POWER, 80 PCM, MINIMUM REACTIVITY FEEDBACK

PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME

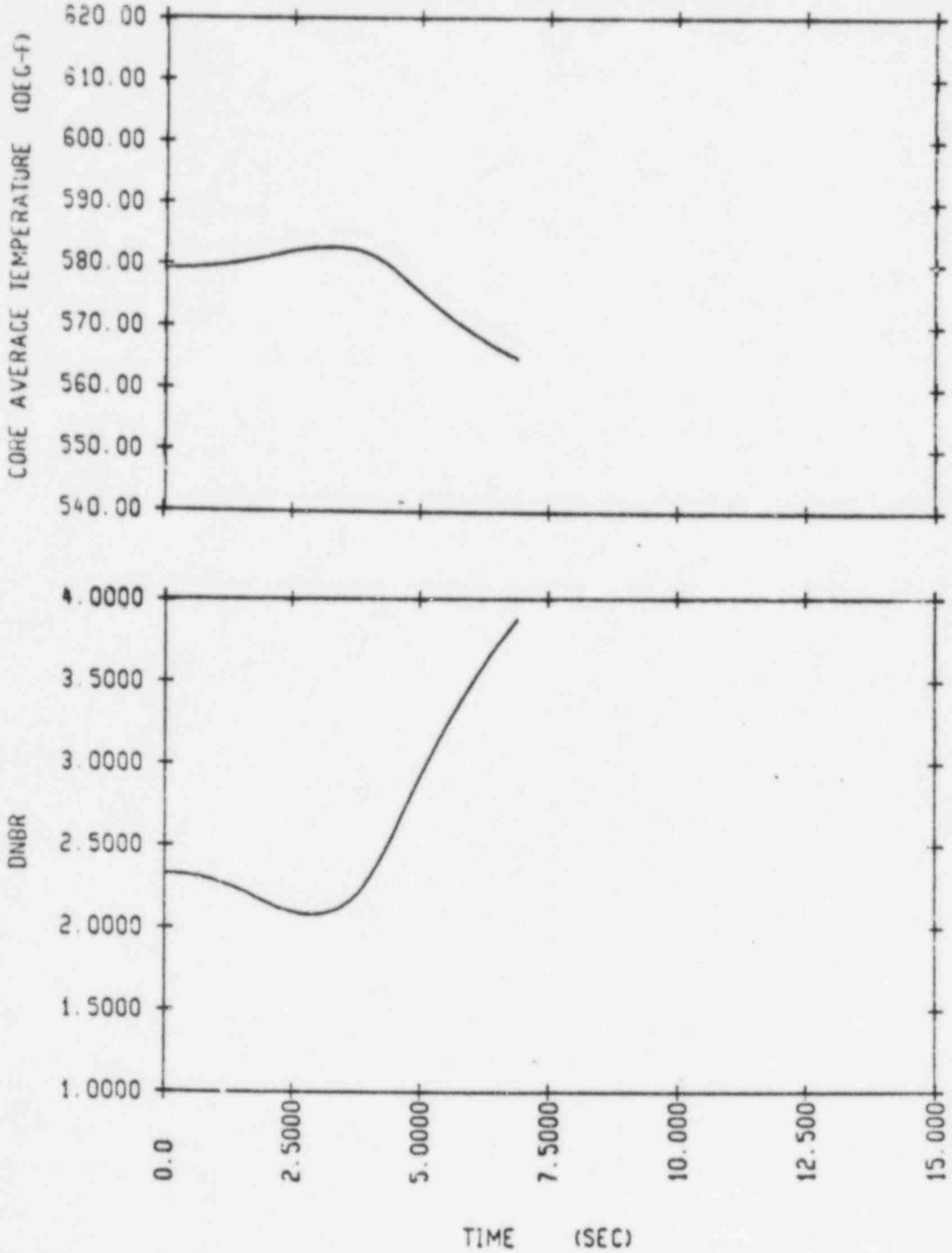


FIGURE 6

ROD WITHDRAWAL AT POWER
FULL POWER, 80 PCM, MINIMUM REACTIVITY FEEDBACK
CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME

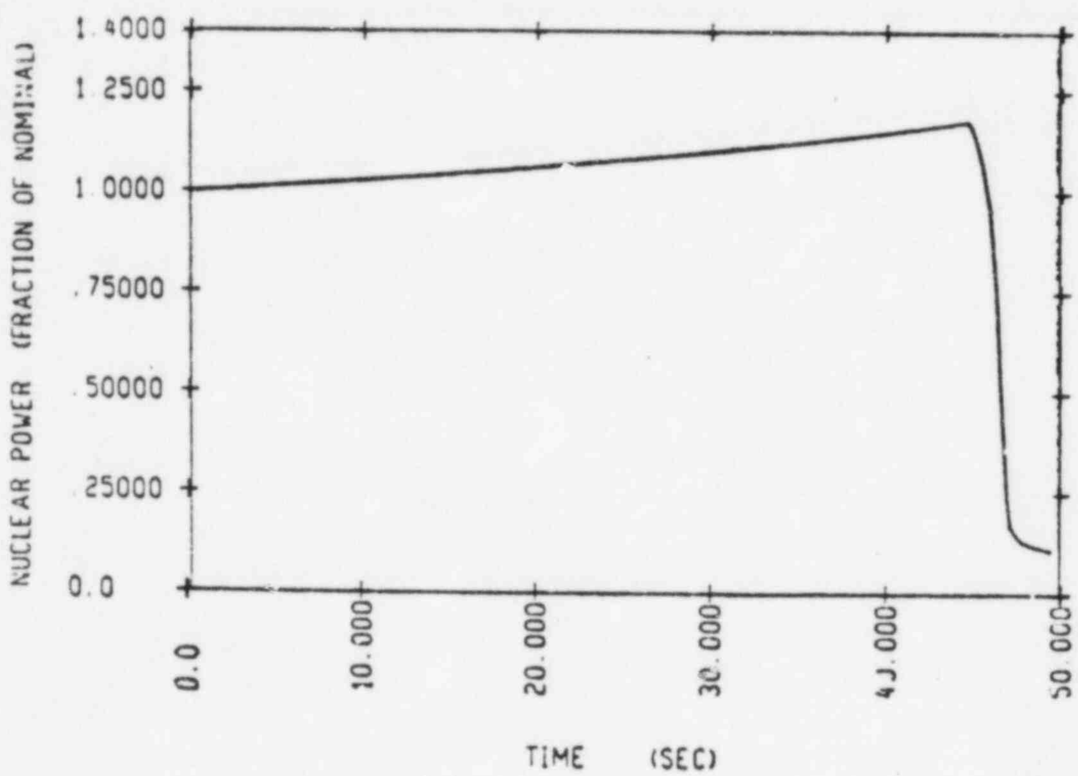


FIGURE 7

ROD WITHDRAWAL AT POWER
FULL POWER, 2 PCM, MINIMUM REACTIVITY FEEDBACK
NUCLEAR POWER VERSUS TIME

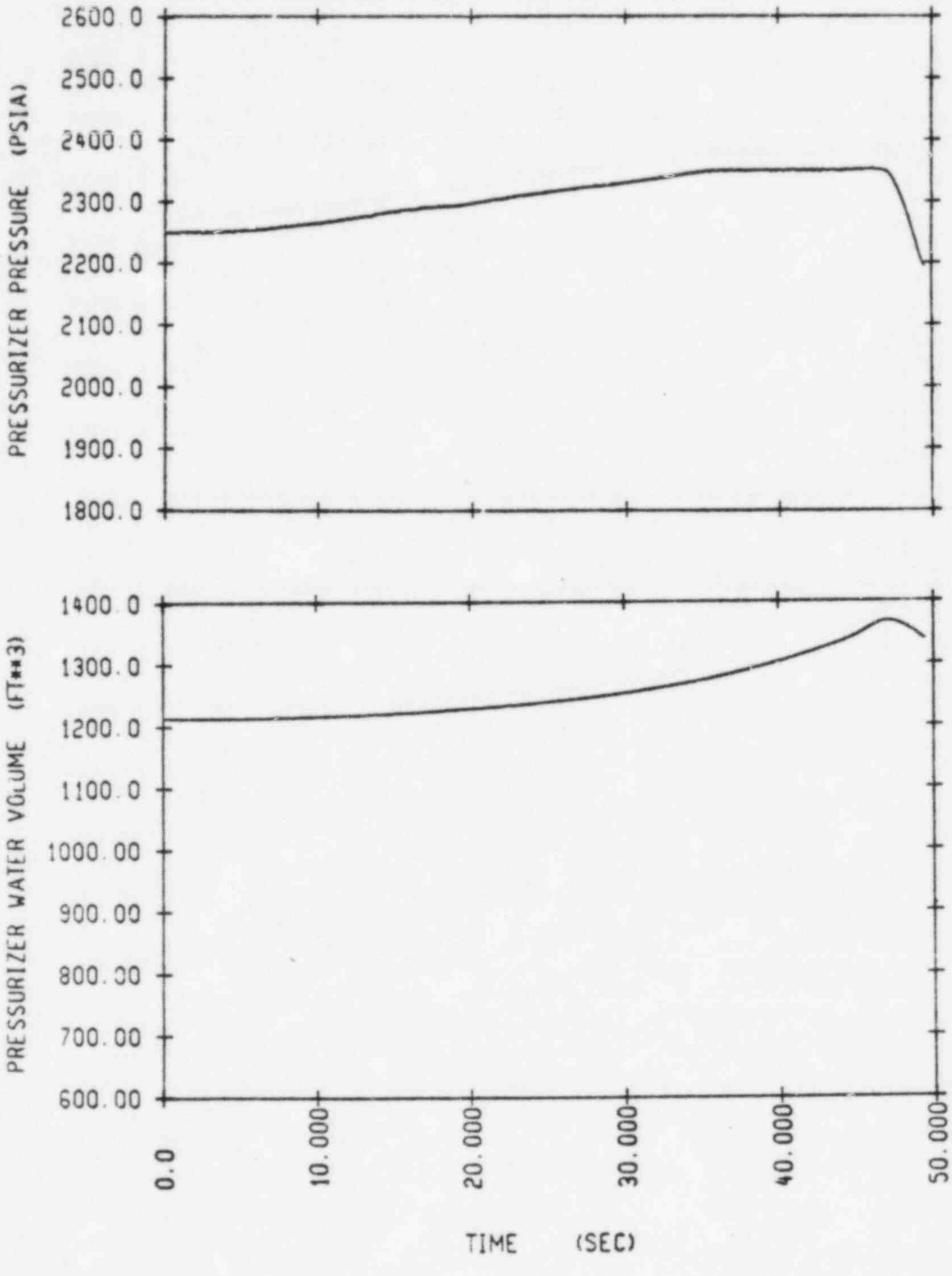


FIGURE 8

ROD WITHDRAWAL AT POWER
FULL POWER, 2 PCM, MINIMUM REACTIVITY FEEDBACK
PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME

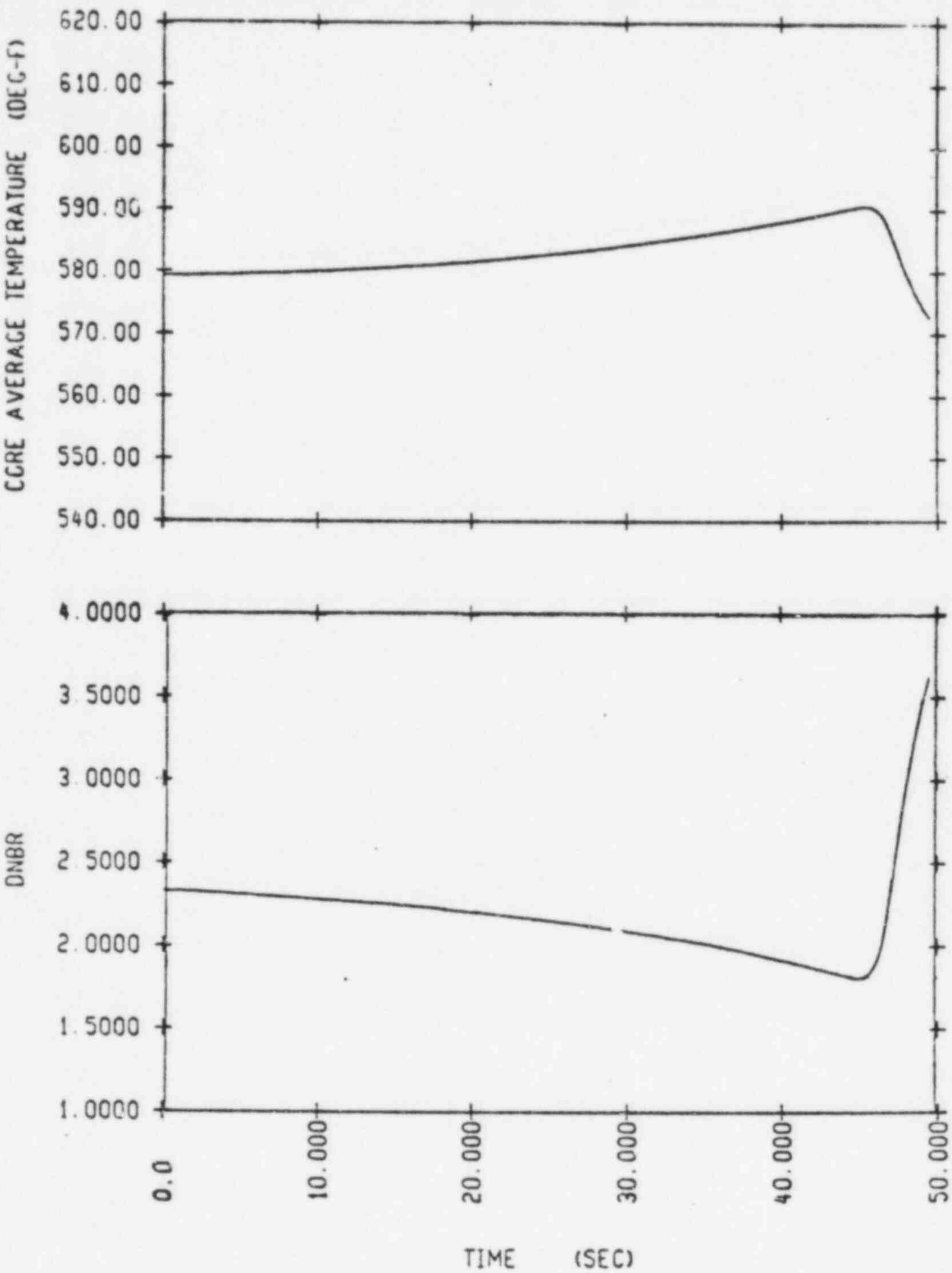
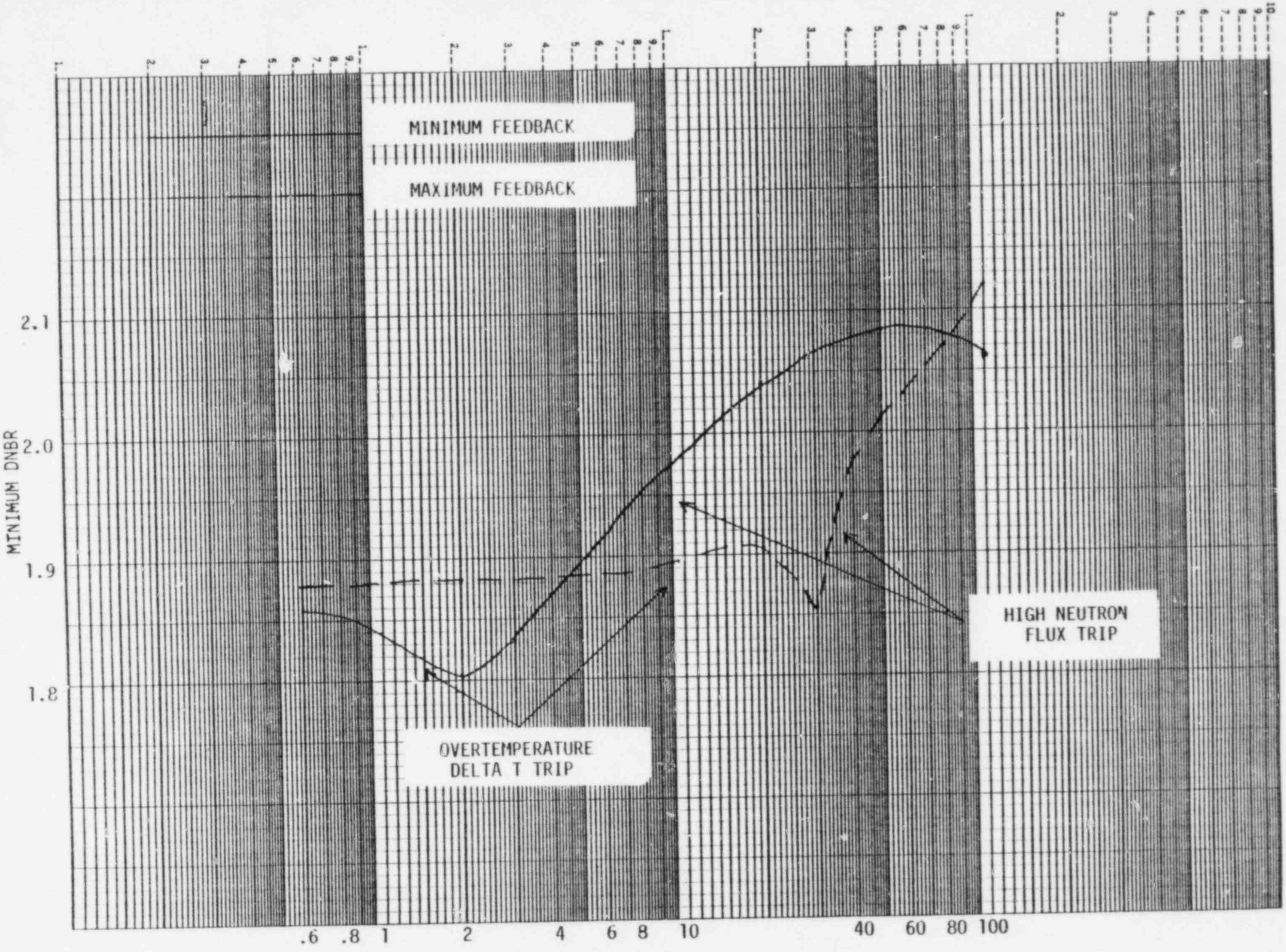


FIGURE 9

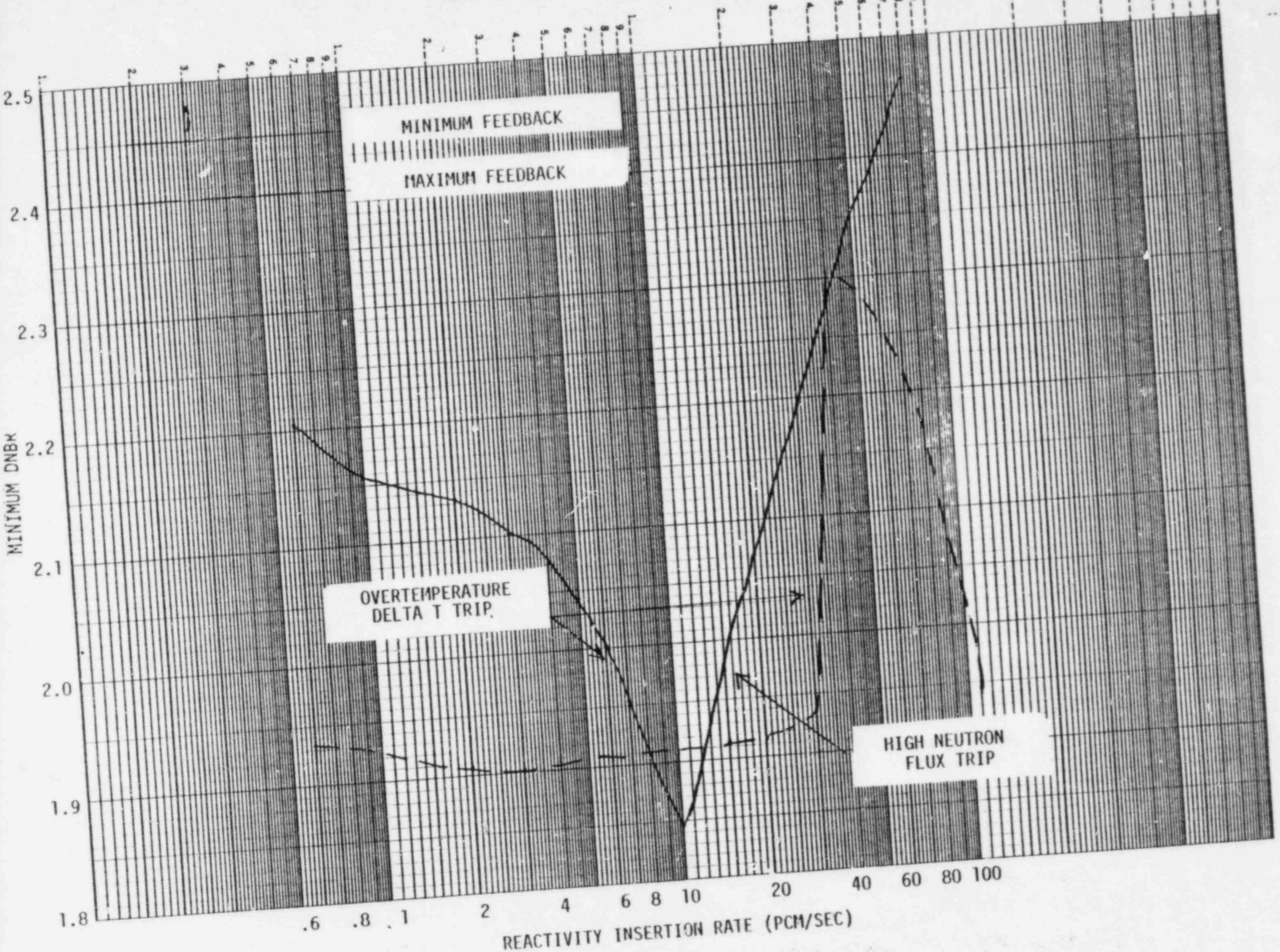
ROD WITHDRAWAL AT POWER
FULL POWER, 2 PCM, MINIMUM REACTIVITY FEEDBACK
CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME

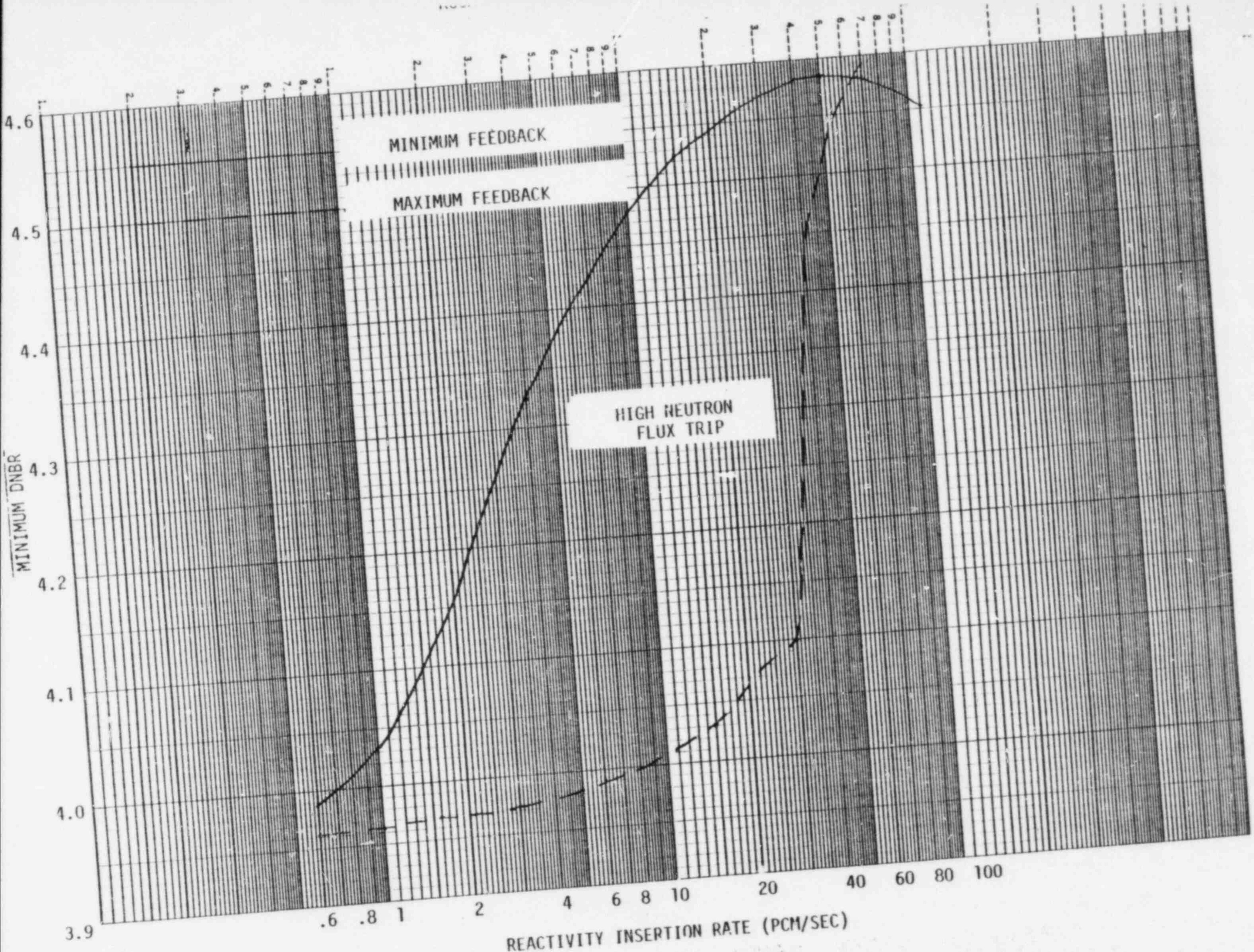
FIGURE 10

RCCA WITHDRAWAL AT POWER 100% POWER



RCCA WITHDRAWAL AT POWER





MINIMUM FEEDBACK
MAXIMUM FEEDBACK
HIGH NEUTRON FLUX TRIP
MINIMUM DNBR
4.6
4.5
4.4
4.3
4.2
4.1
4.0
3.9
.6
.8
1
2
4
6
8
10
20
40
60
80
100
REACTIVITY INSERTION RATE (PCM/SEC)

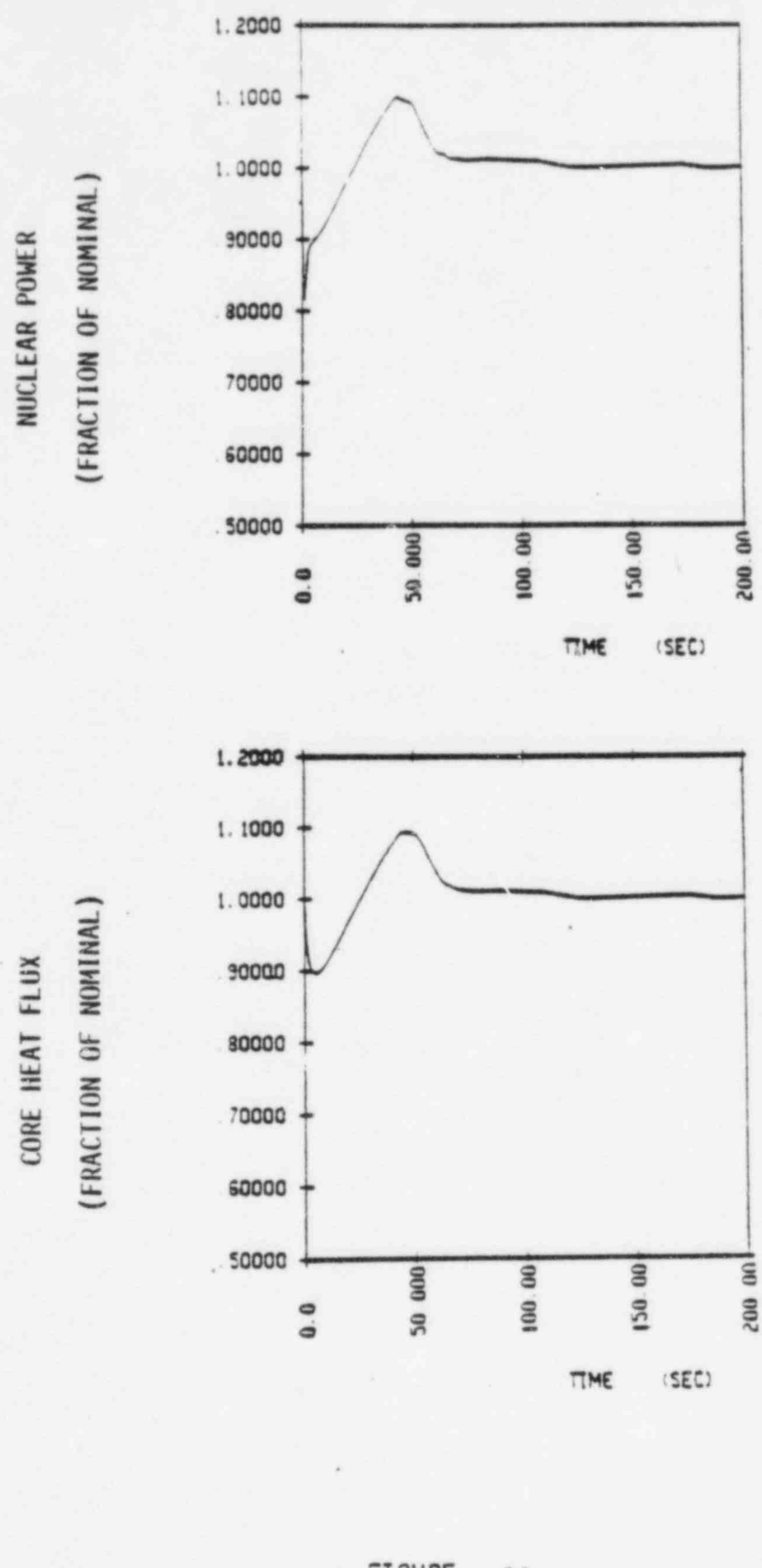


FIGURE 13

NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME FOR A TYPICAL RESPONSE TO A DROPPED RCCA(S) IN AUTOMATIC CONTROL

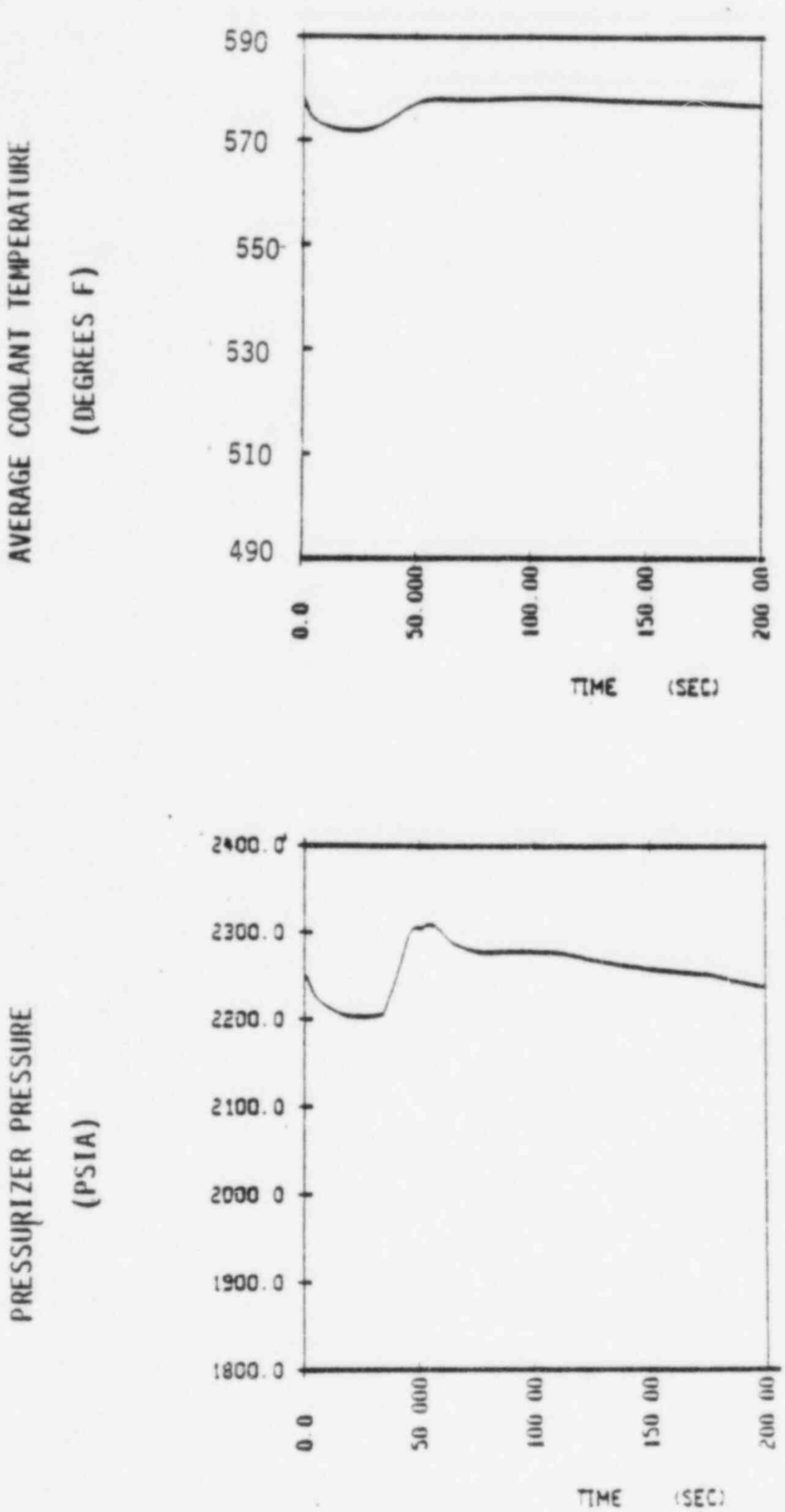


FIGURE 14

AVERAGE COOLANT TEMPERATURE AND PRESSURIZER PRESSURE
VERSUS TIME FOR A TYPICAL RESPONSE TO A DROPPED RCCA(S)
IN AUTOMATIC CONTROL

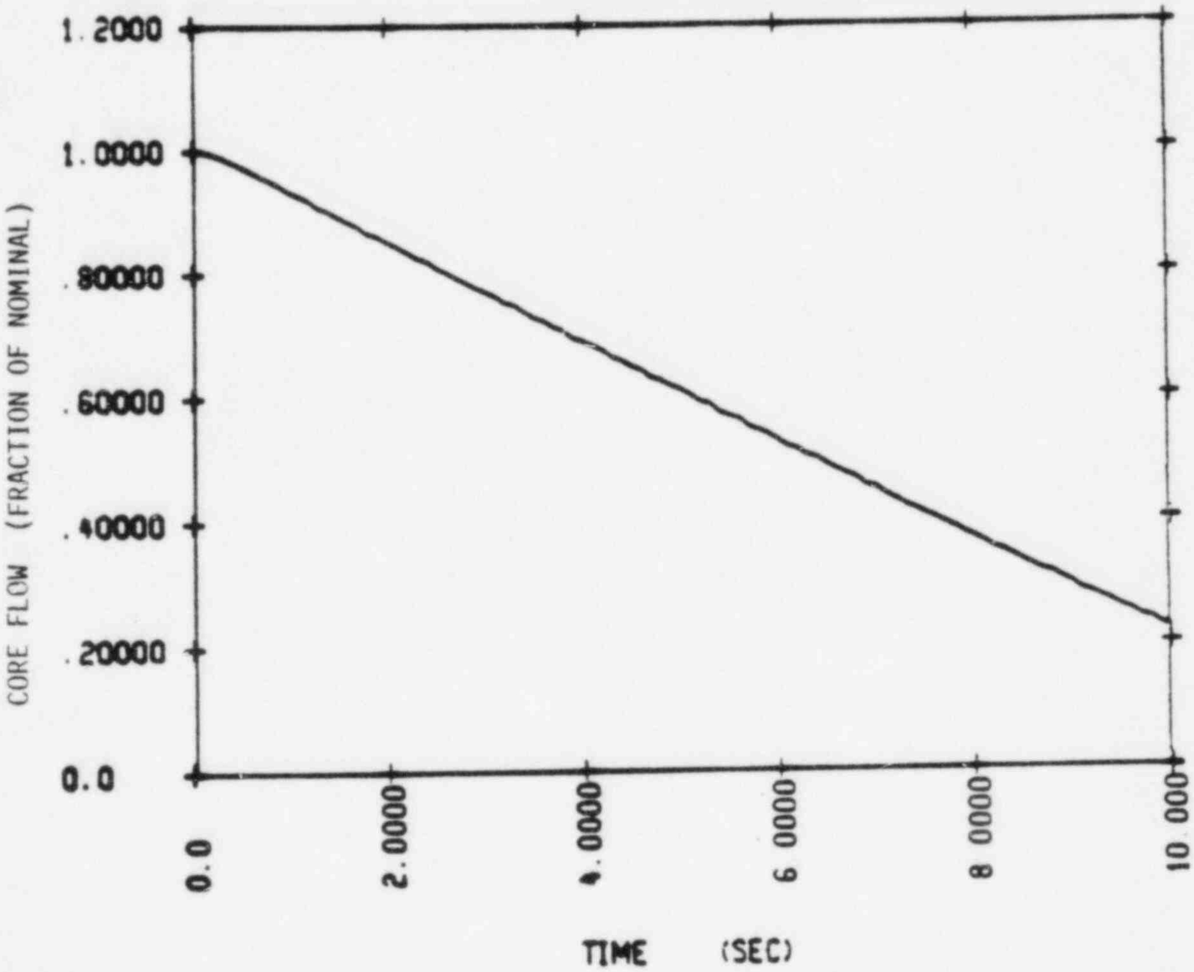


FIGURE 15

CORE FLOW COASTDOWN VERSUS TIME, COMPLETE LOSS OF FLOW

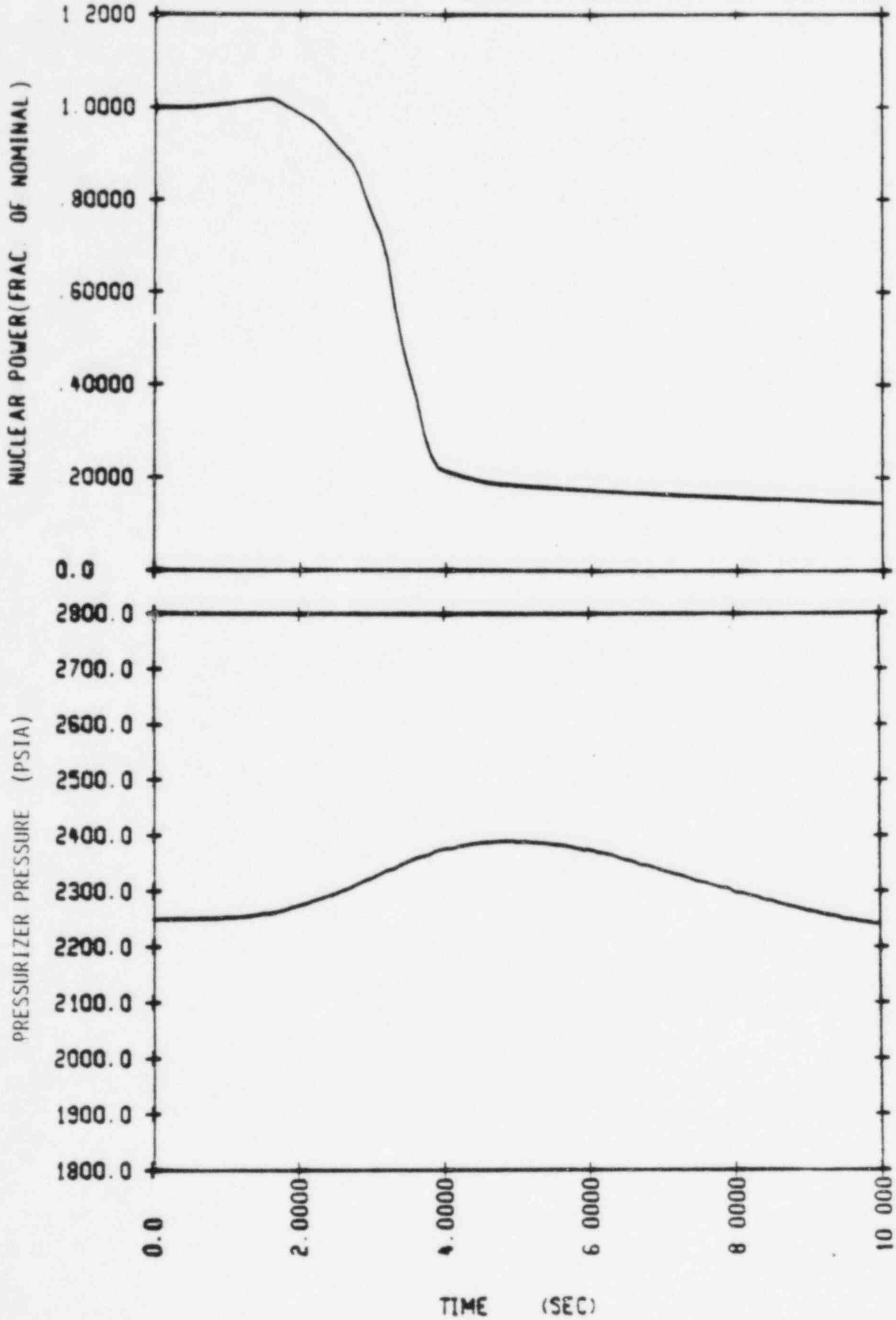


FIGURE 16

NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME, COMPLETE LOSS OF FLOW

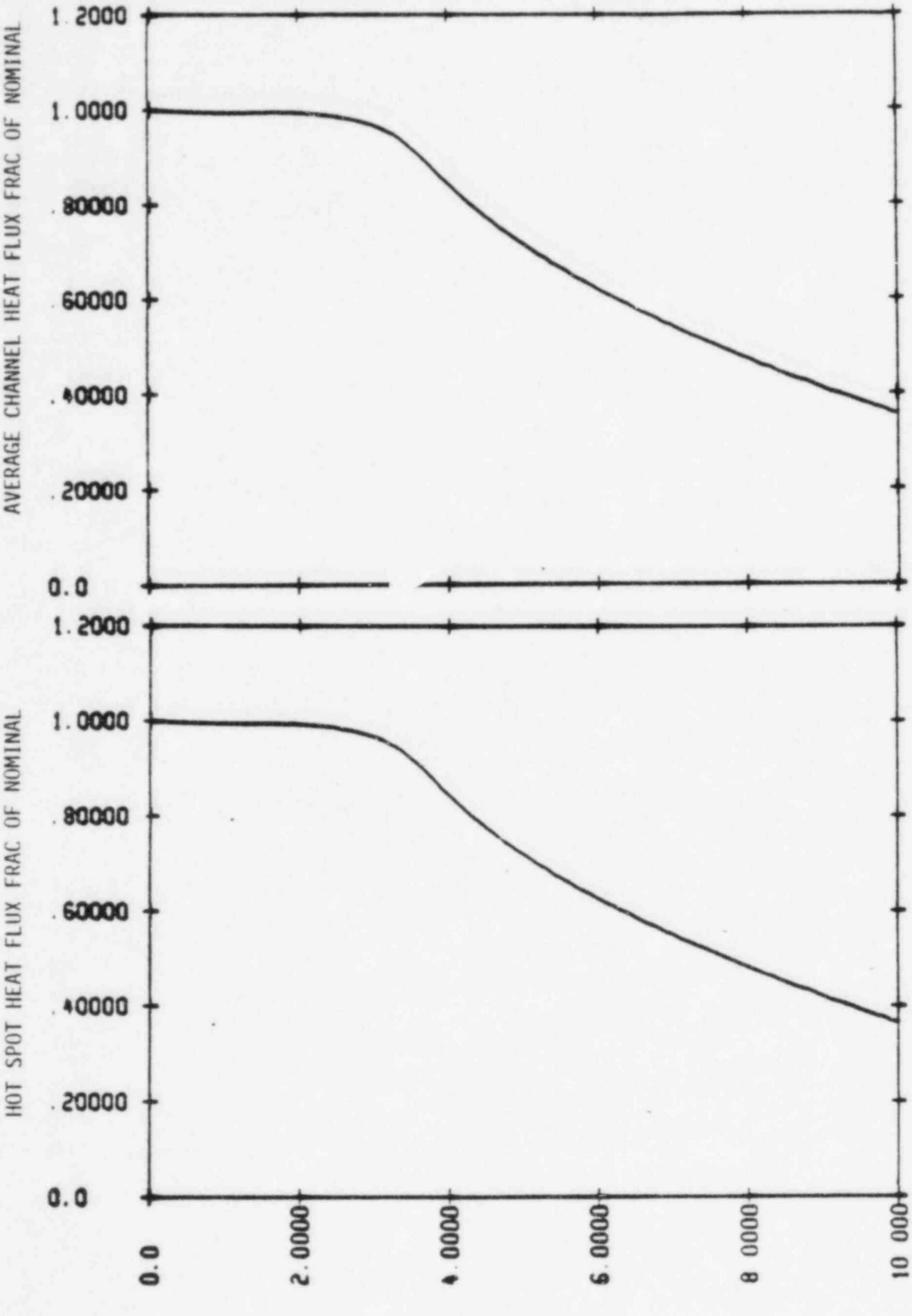


FIGURE 17

HOT SPOT AND AVERAGE CHANNEL HEAT FLUX VERSUS TIME, COMPLETE LOSS OF FLOW

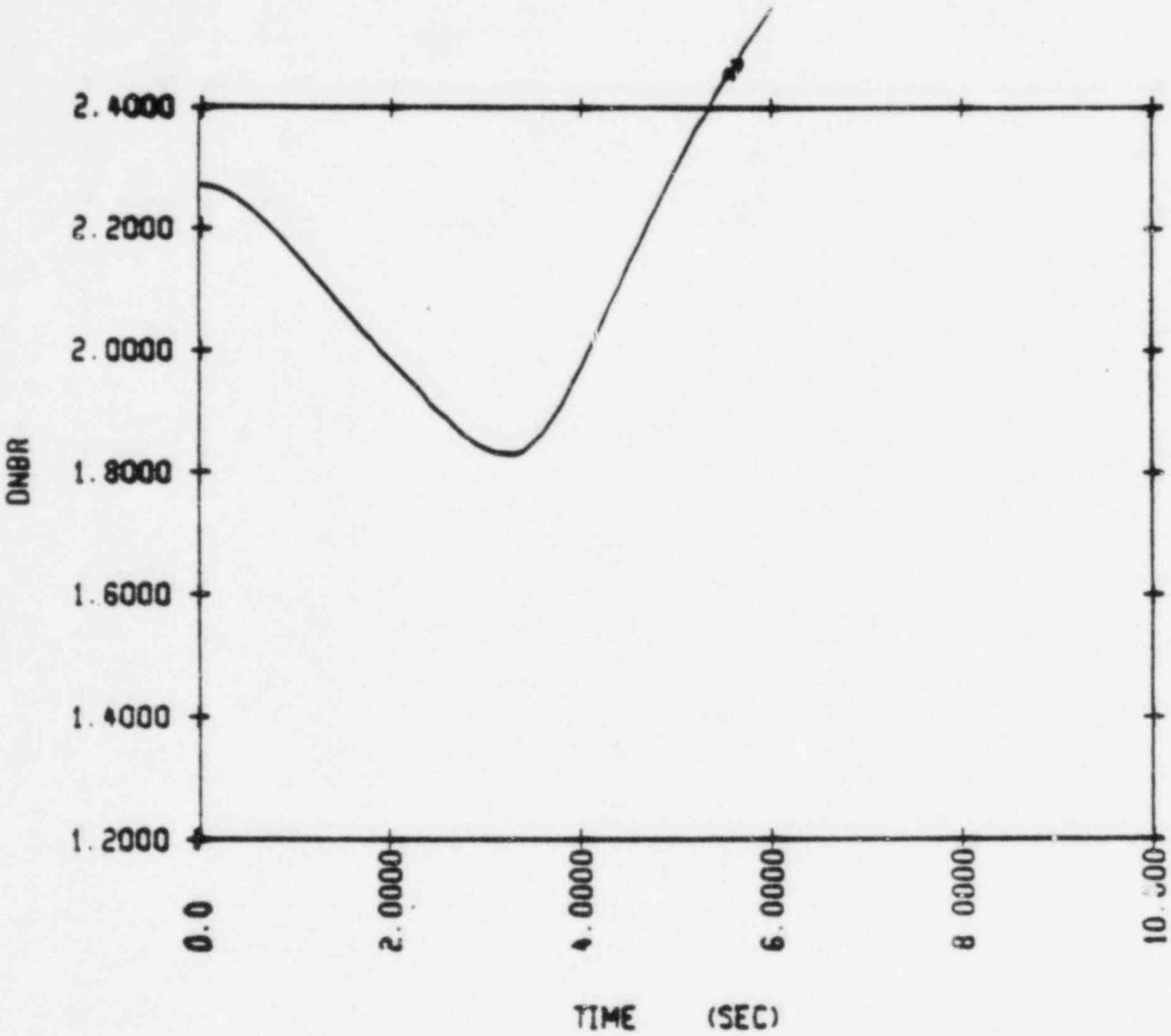


FIGURE 18

DNBR VERSUS TIME,COMPLETE LOSS OF FLOW

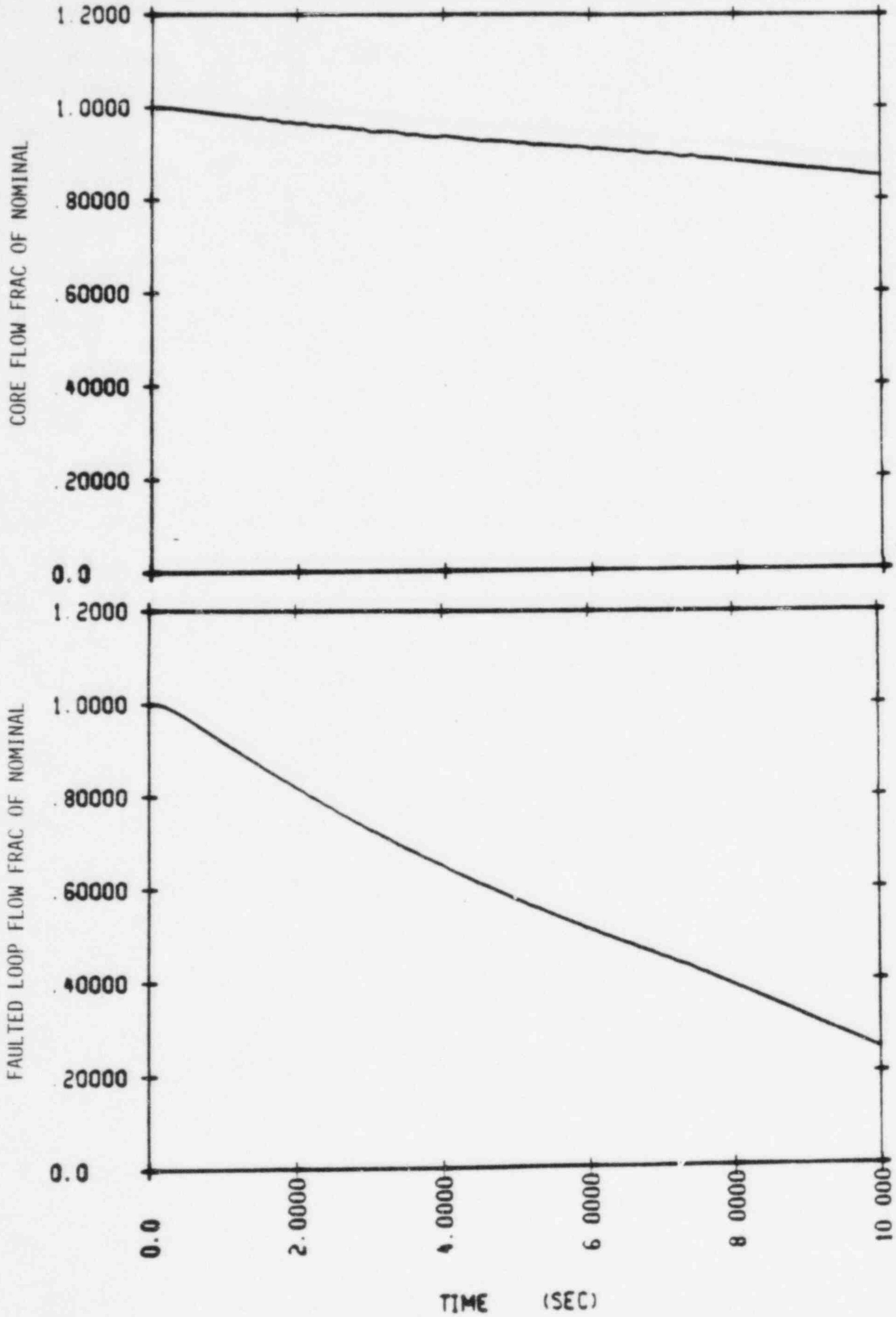


FIGURE 19

FAULTED LOOP AND CORE FLOW VERSUS TIME, PARTIAL LOSS OF FLOW 1/4

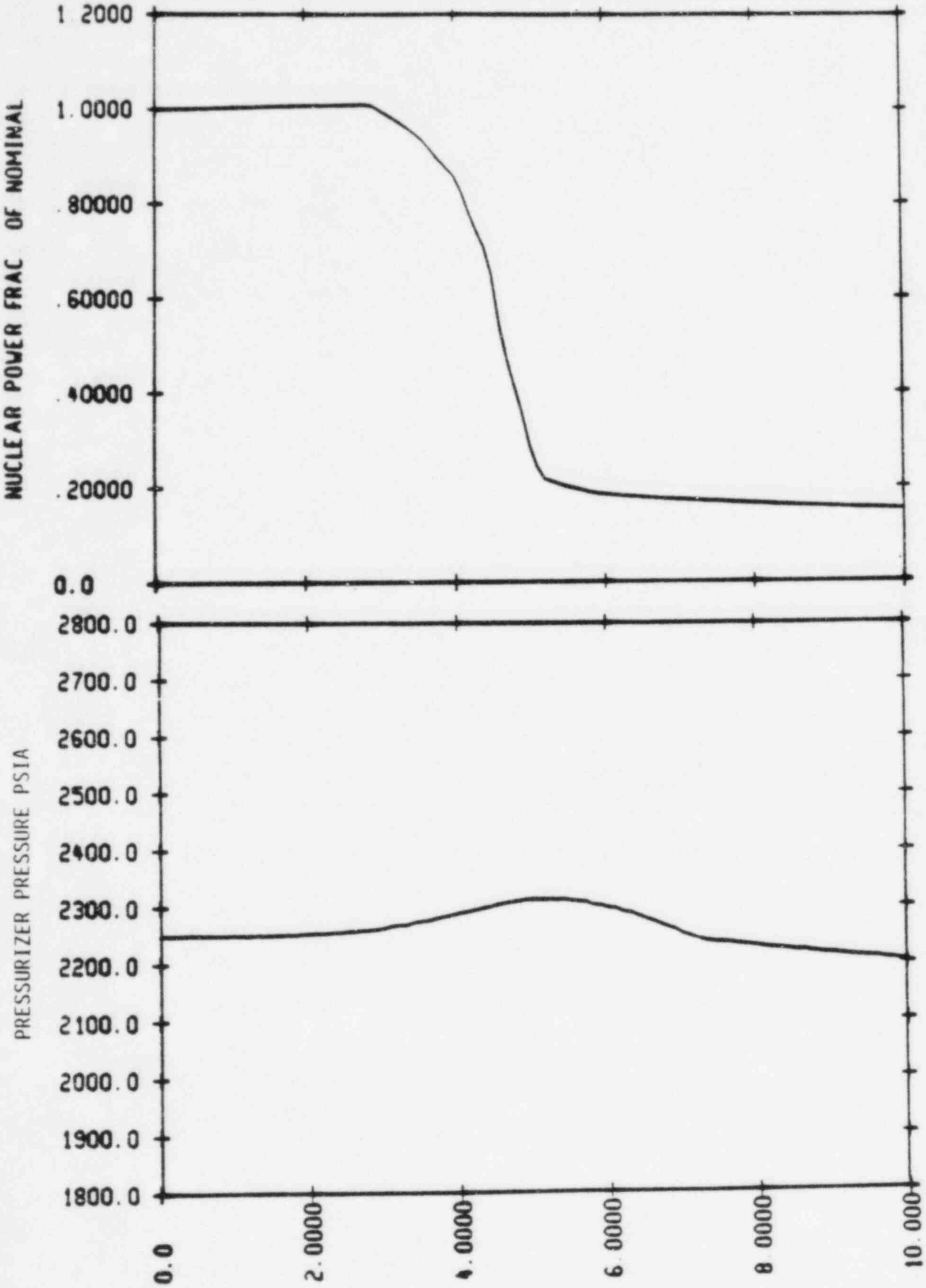


FIGURE 20
NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME, PARTIAL LOSS OF FLOW 1/4

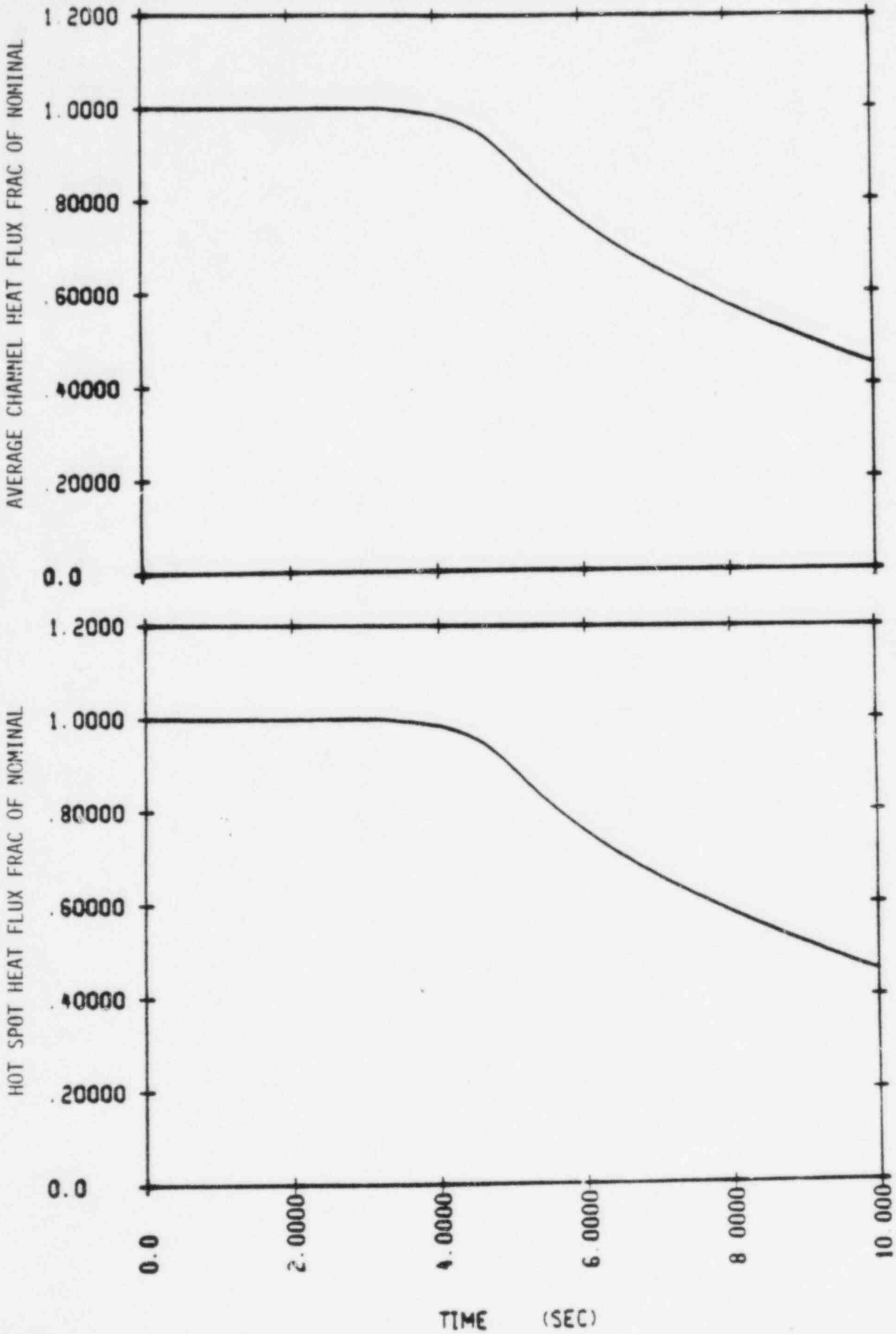


FIGURE 21

AVERAGE CHANNEL AND HOT SPOT HEAT FLUX VERSUS TIME, PARTIAL LOSS OF FLOW 1/4

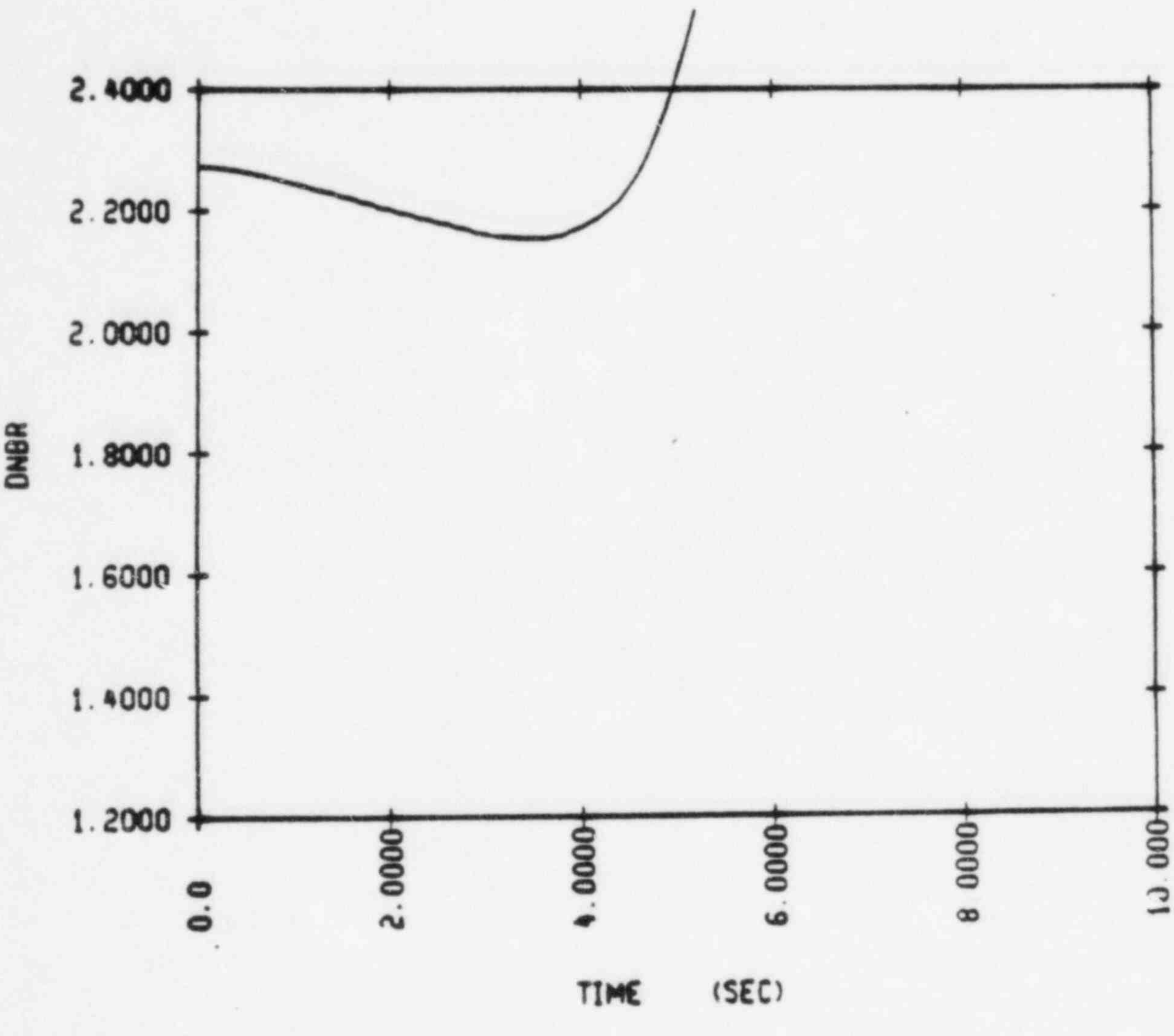


FIGURE 22

DNBR VERSUS TIME, PARTIAL LOSS OF FLOW 1/4

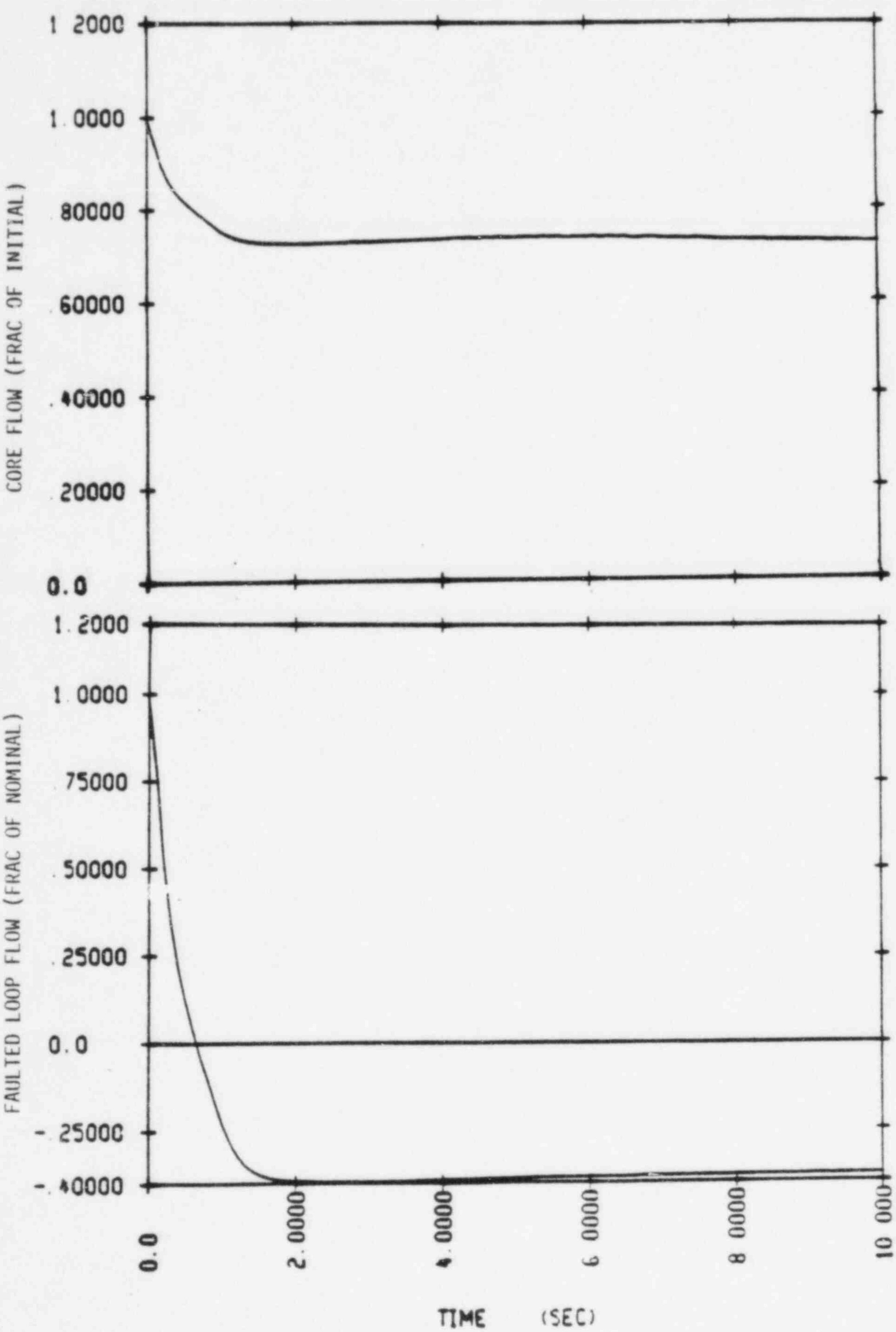


FIGURE 23

CORE AND FAULTED LOOP FLOW VERSUS TIME, 1/4 LOCKED ROTOR

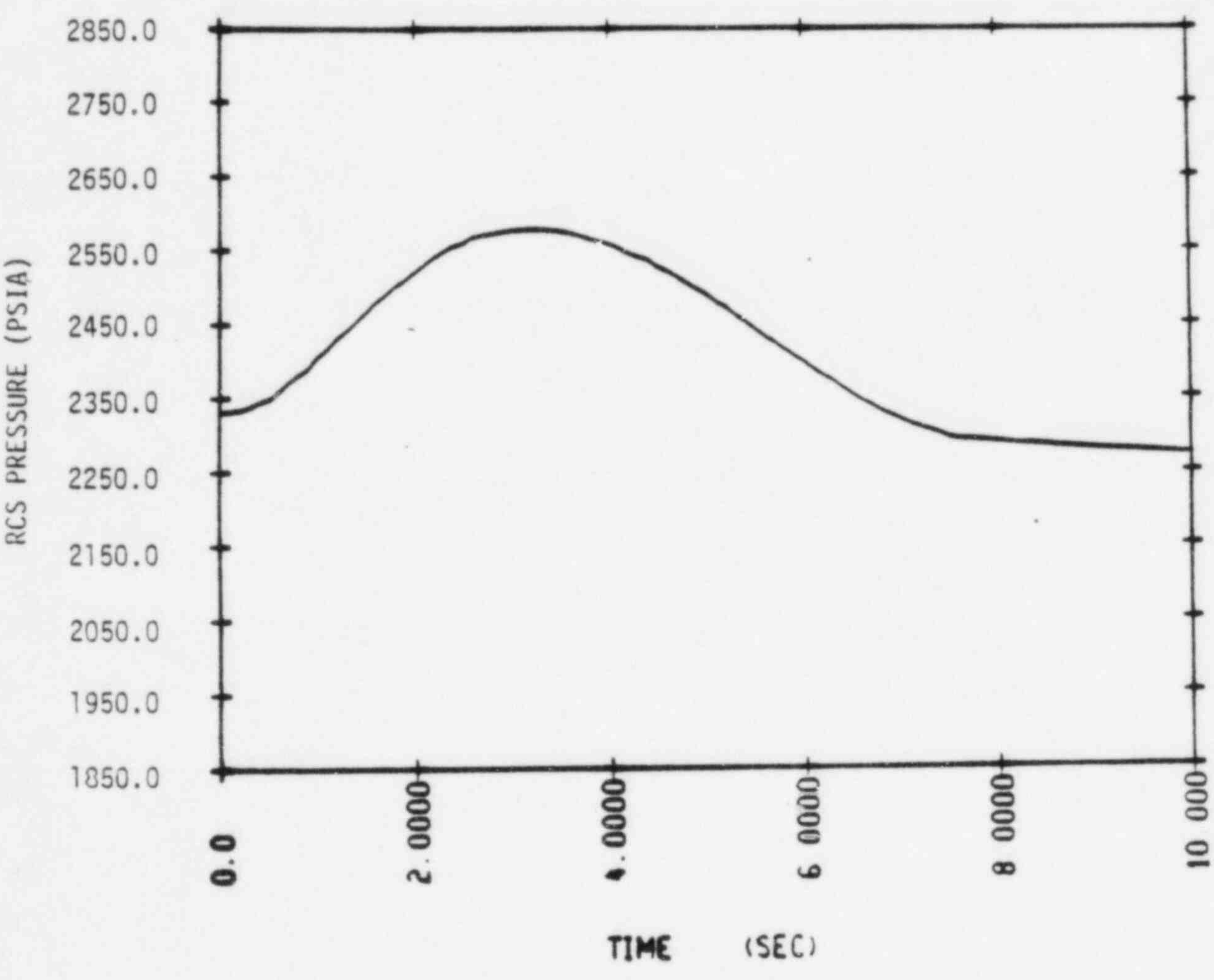


FIGURE 24

REACTOR PRESSURE VERSUS TIME, 1/4 LOCKED ROTOR

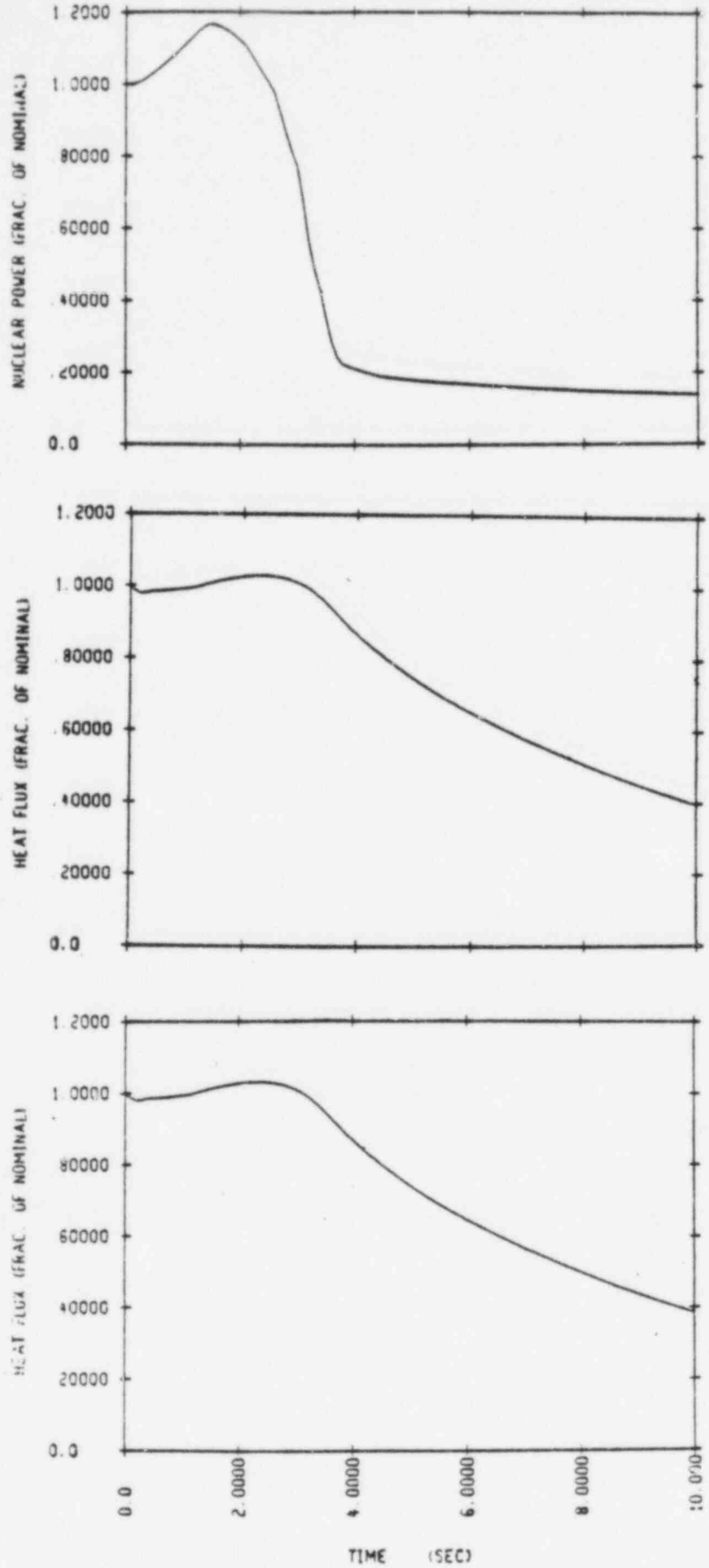


FIGURE 25

NUCLEAR POWER, AVERAGE CHANNEL, AND HOT SPOT HEAT FLUX VERSUS TIME, 1/4 LOCKED ROTOR

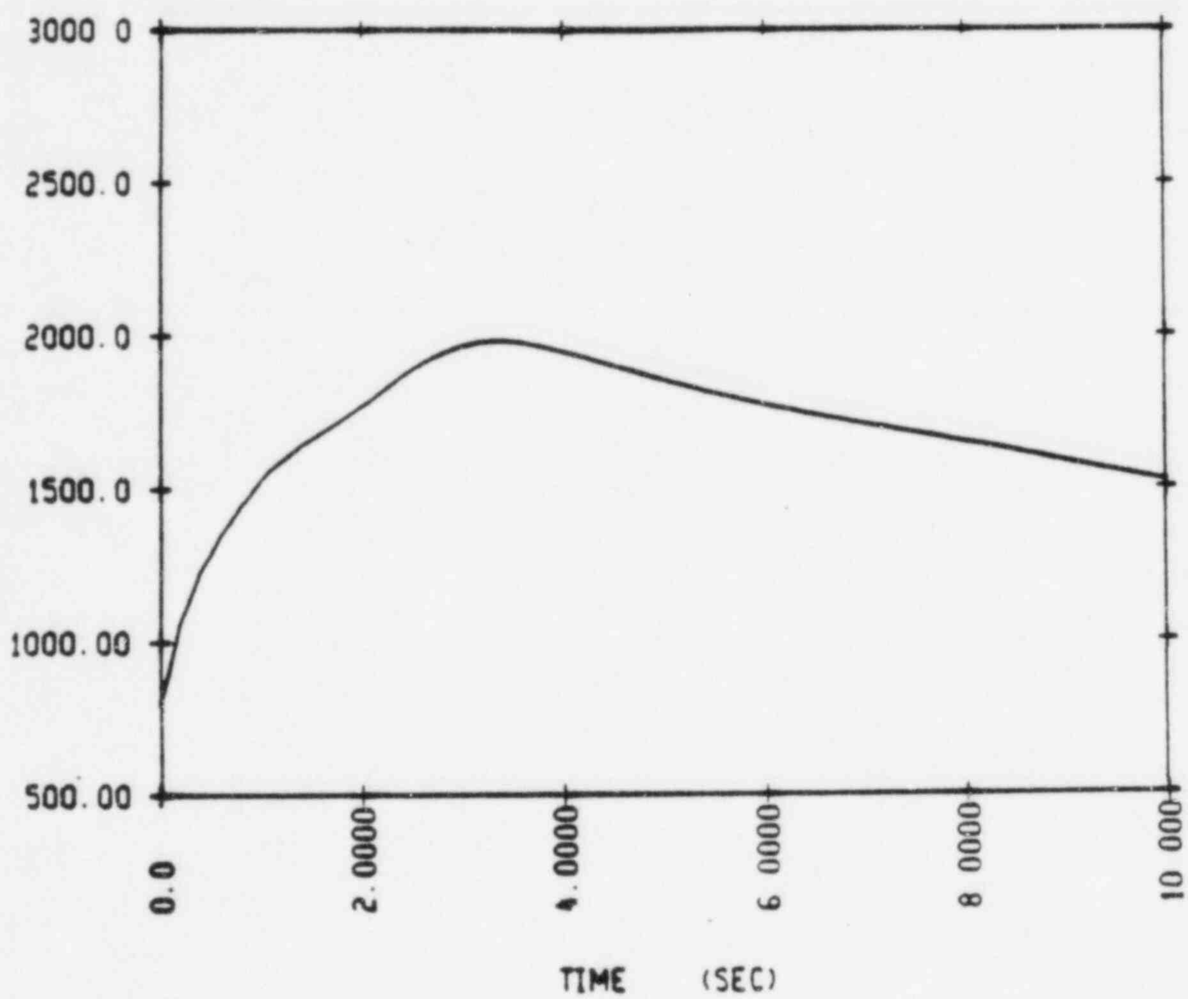


FIGURE 26

CLAD INNER TEMPERATURE VERSUS TIME, 1/4 LOCKED ROTOR

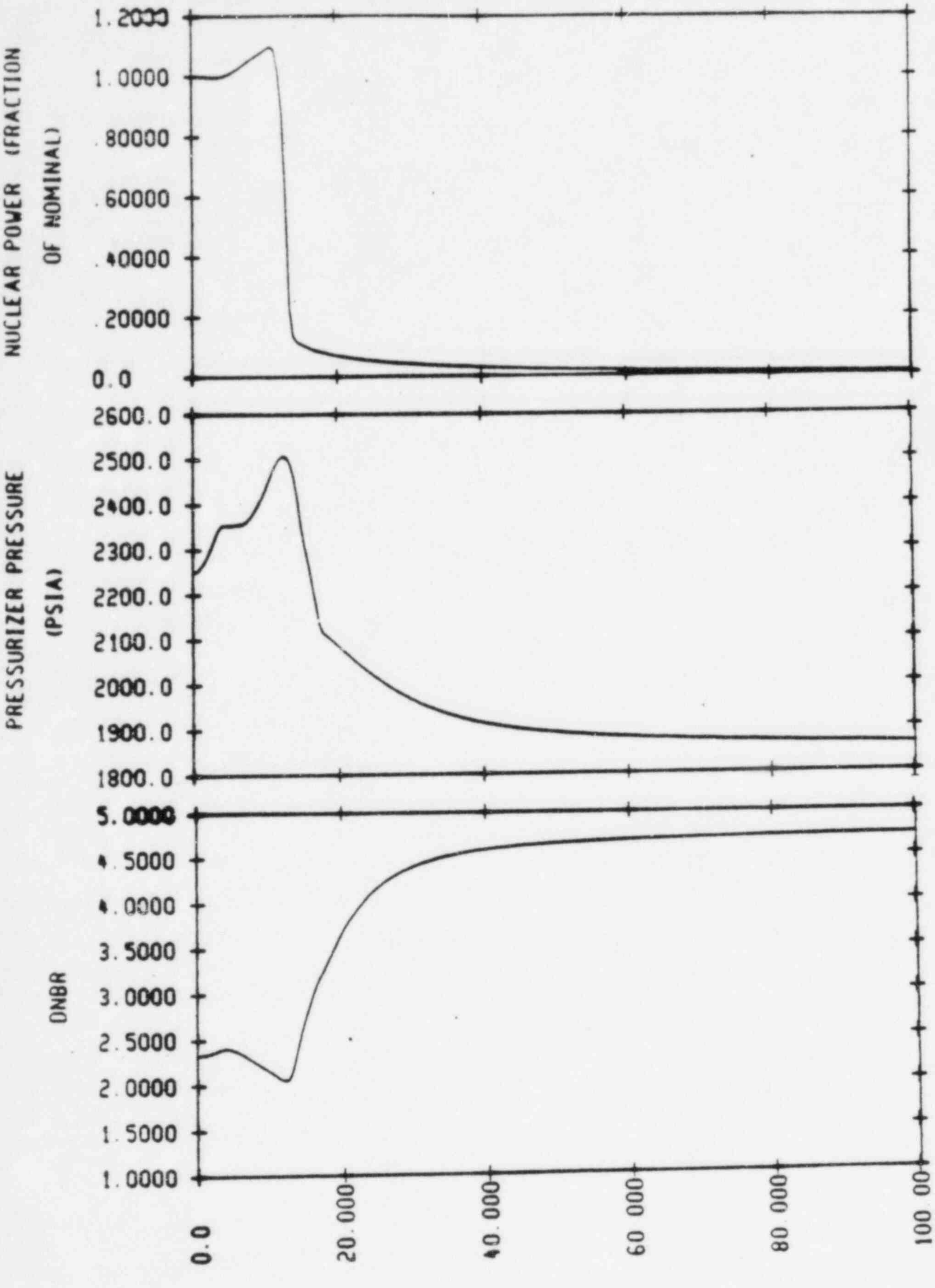


FIGURE 27

NUCLEAR POWER, PRESSURIZER PRESSURE, AND DNBR VERSUS TIME FOR LOSS OF LOAD MINIMUM REACTIVITY FEEDBACK WITH PRESSURIZER SPRAY AND PORV's

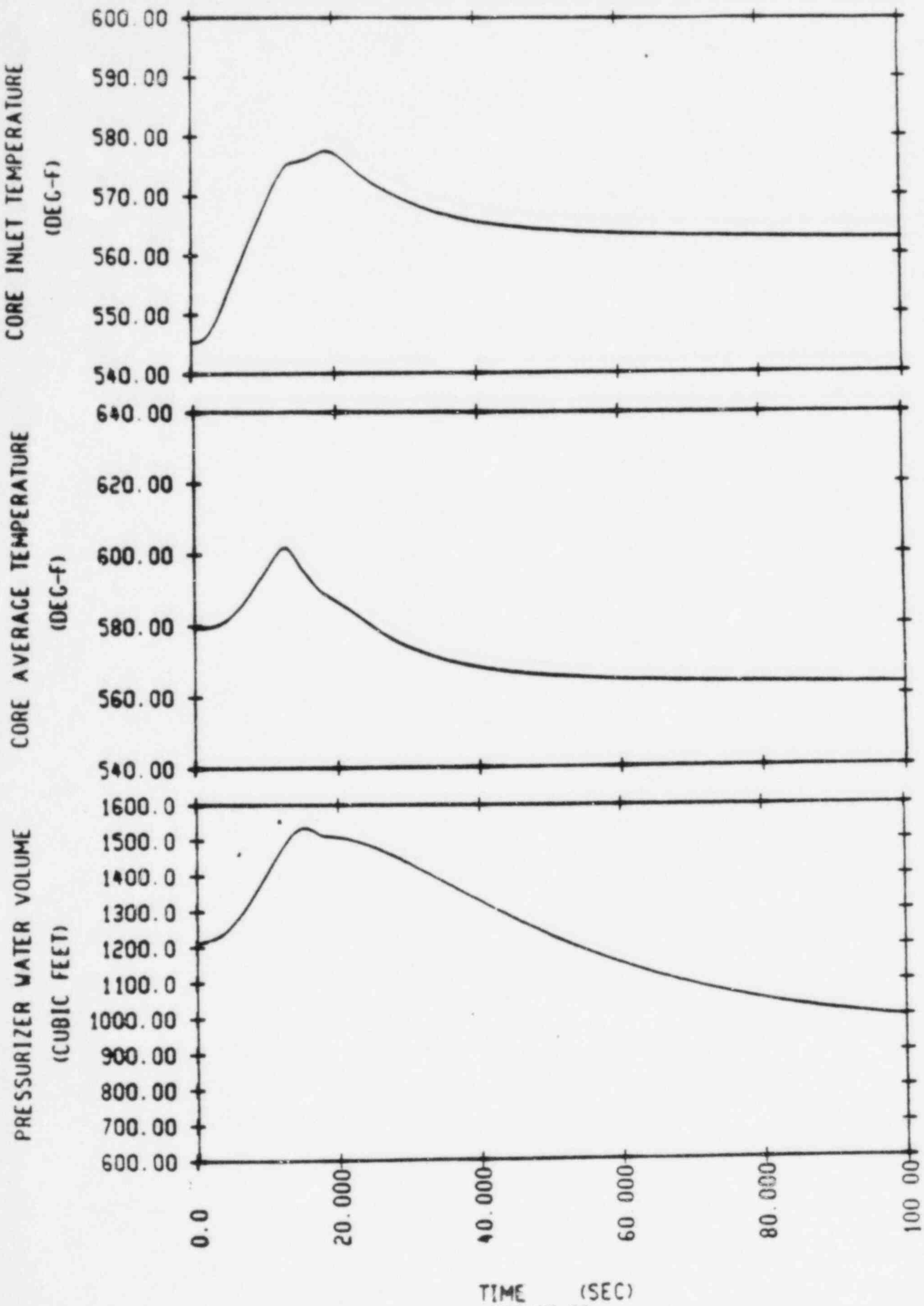


FIGURE 28
CORE INLET TEMPERATURE, CORE AVERAGE TEMPERATURE, AND PRESSURIZER WATER VOLUME VERSUS TIME FOR LOSS OF LOAD, MINIMUM REACTIVITY FEEDBACK WITH PRESSURIZER SPRAY AND PORV's

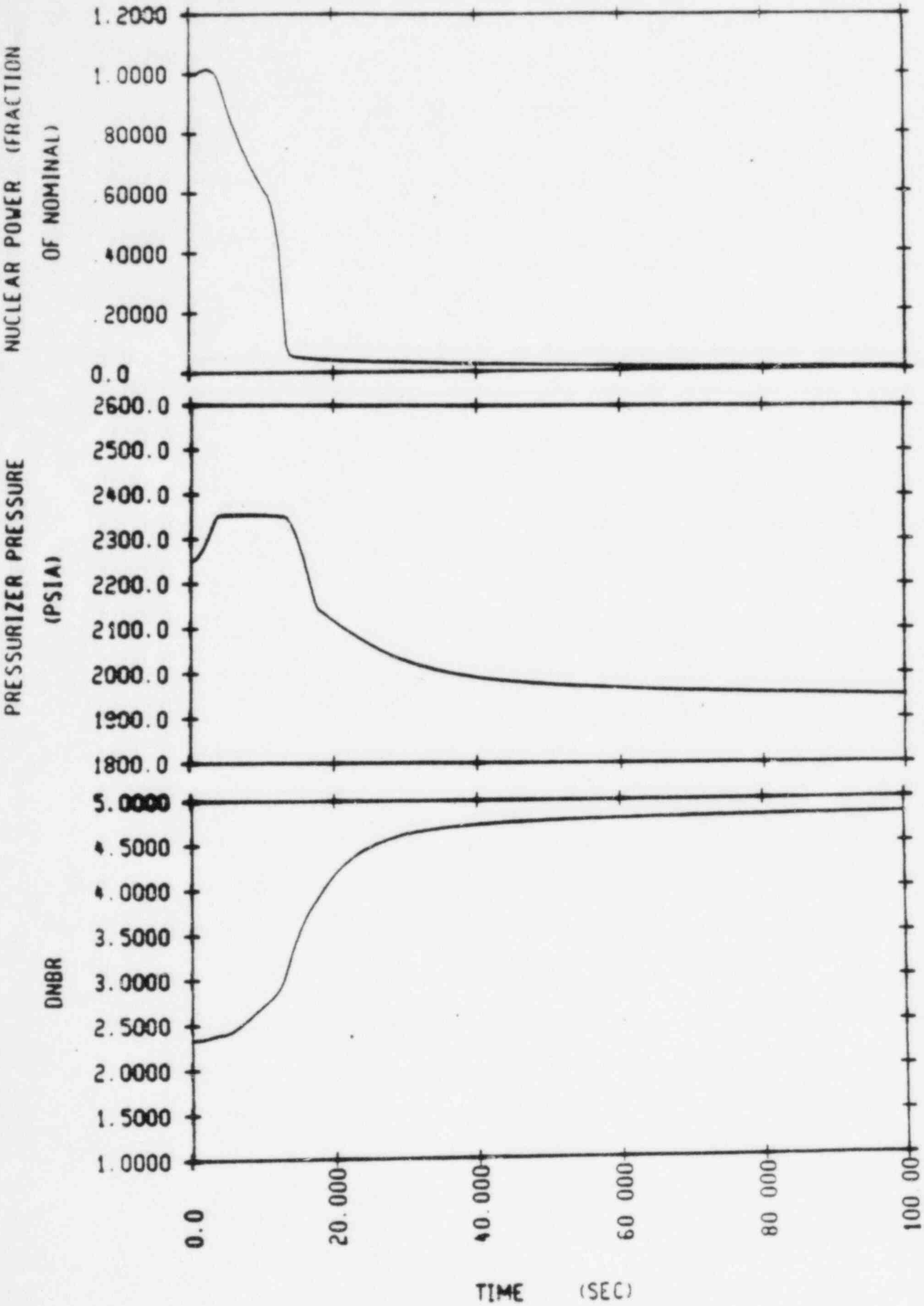


FIGURE 29

NUCLEAR POWER, PRESSURIZER PRESSURE, AND DNBR VERSUS TIME FOR LOSS OF LOAD, MAXIMUM REACTIVITY FEEDBACK WITH PRESSURIZER SPRAY AND PORV's

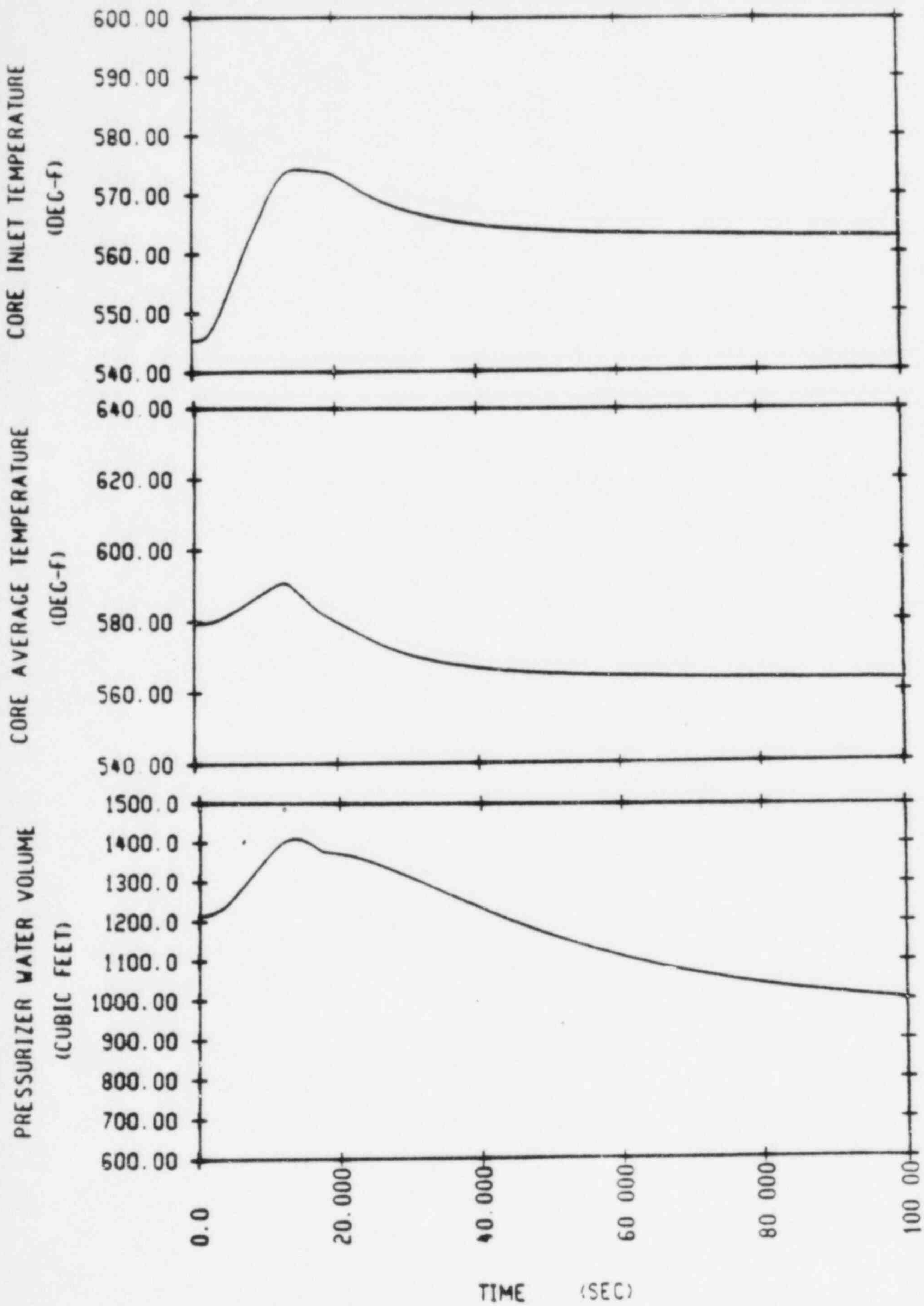


FIGURE 30
CORE INLET TEMPERATURE, CORE AVERAGE TEMPERATURE, AND PRESSURIZER WATER VOLUME FOR LOSS OF LOAD, MAXIMUM REACTIVITY FEEDBACK, WITH PRESSURIZER SPRAY AND PORV's

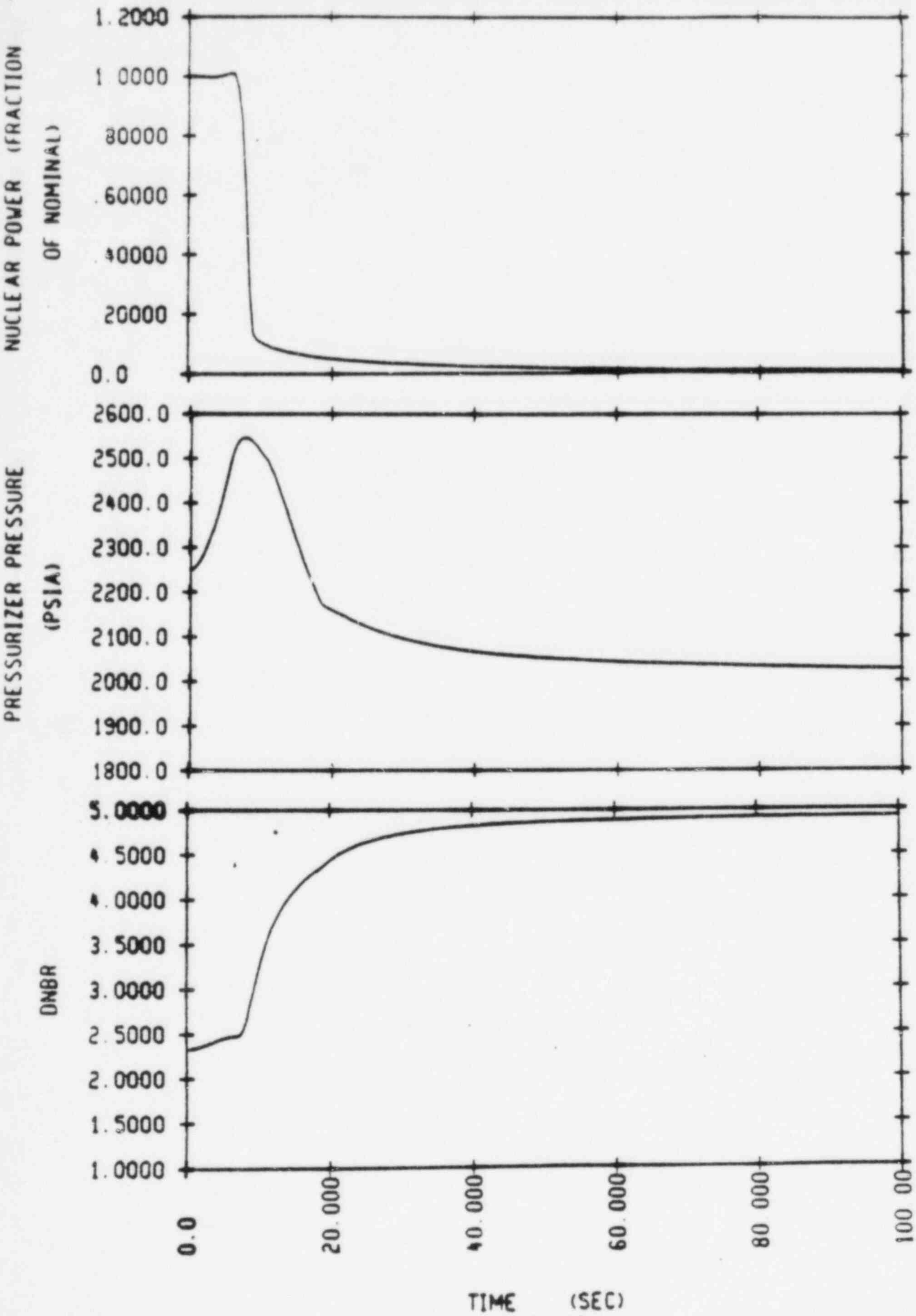


FIGURE 31

NUCLEAR POWER, PRESSURIZER PRESSURE & DNBR VERSUS TIME FOR LOSS OF LOAD MINIMUM REACTIVITY FEEDBACK WITHOUT PRESSURIZER SPRAY & PORV's

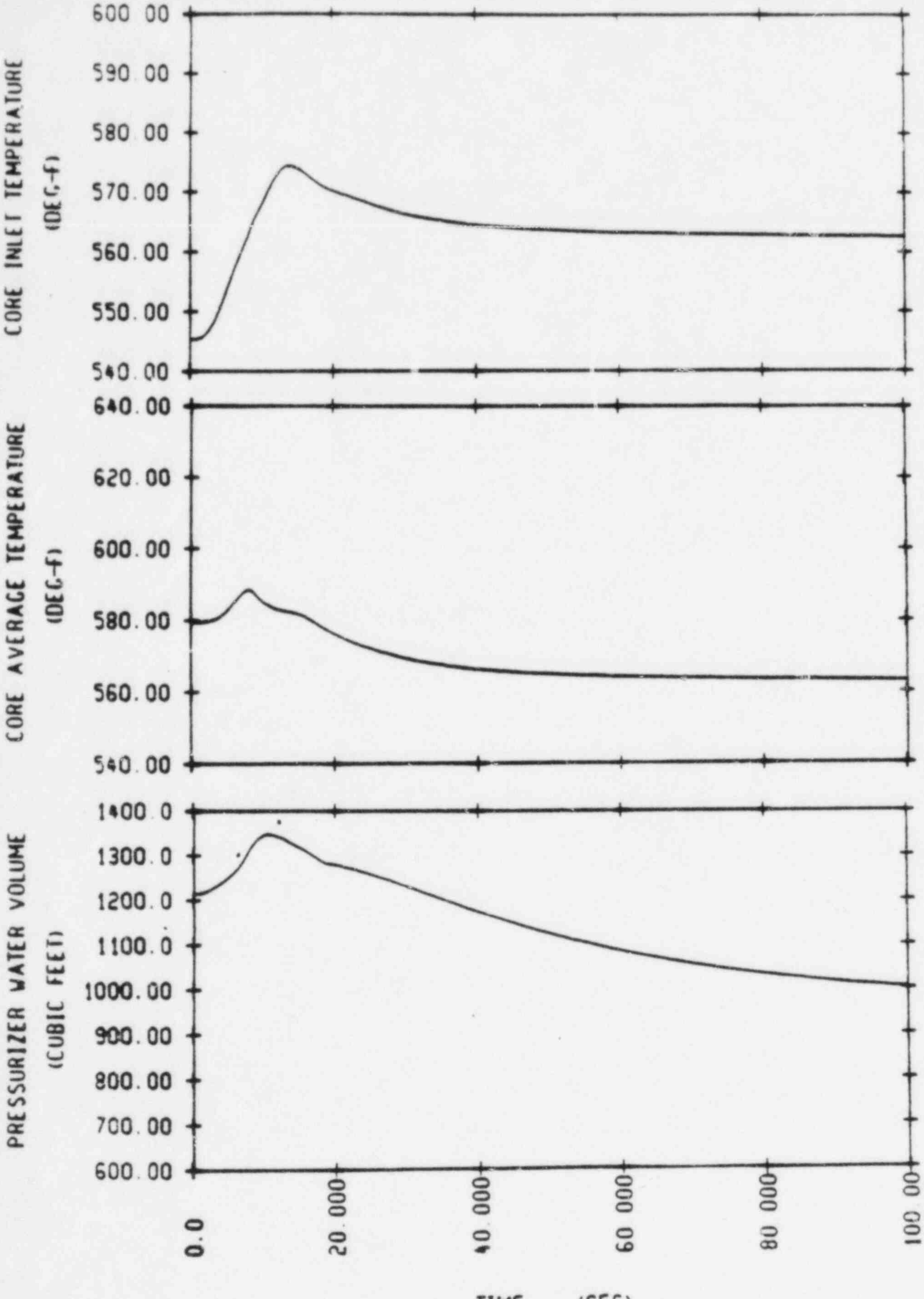


FIGURE 32
CORE INLET AND CORE AVERAGE TEMPERATURE, PRESSURIZER WATER VOLUME VERSUS TIME FOR LOSS OF LOAD MINIMUM REACTIVITY FEEDBACK WITHOUT PRESSURIZER SPRAY AND PORV's

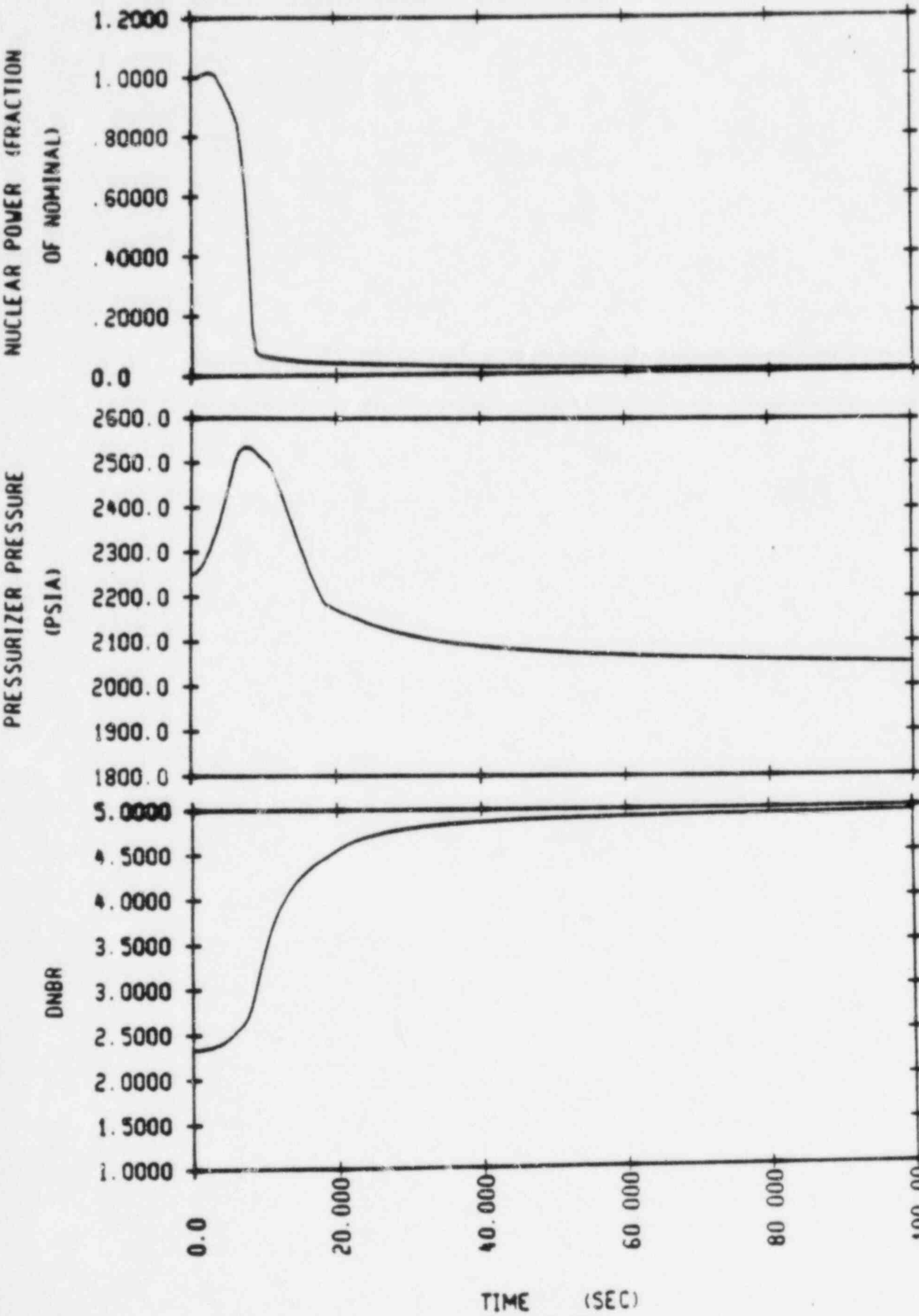


FIGURE 33
NUCLEAR POWER, PRESSURIZER PRESSURE AND DNBR VERSUS TIME FOR LOSS OF LOAD MAXIMUM REACTIVITY FEEDBACK, NO PRESSURIZER SPRAY OR PORV's

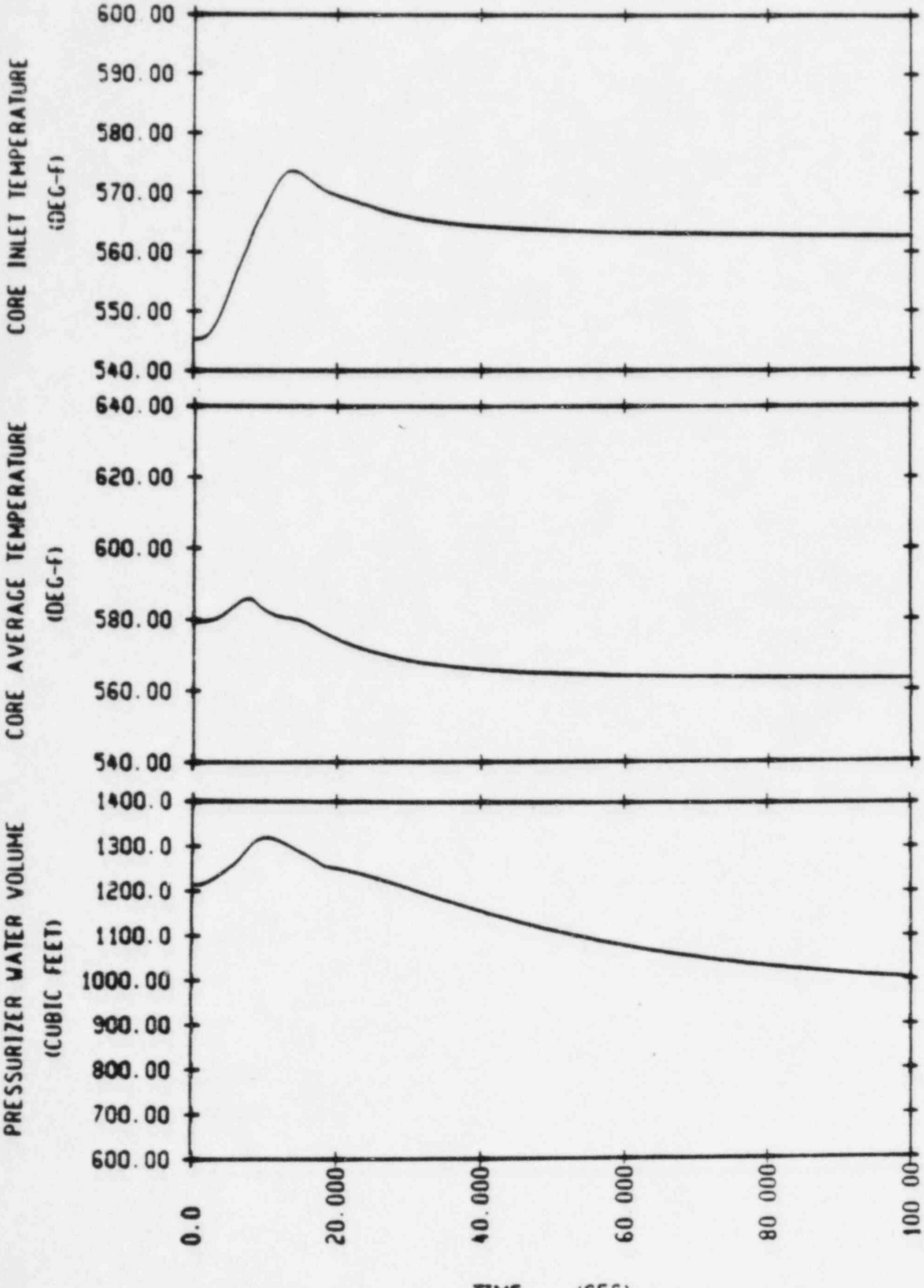


FIGURE 34
CORE INLET, AND CORE AVERAGE TEMPERATURE, PRESSURIZER WATER VOLUME VERSUS TIME FOR LOSS OF LOAD MAXIMUM REACTIVITY FEEDBACK, NO PRESSURIZER SPRAY OR PORV's

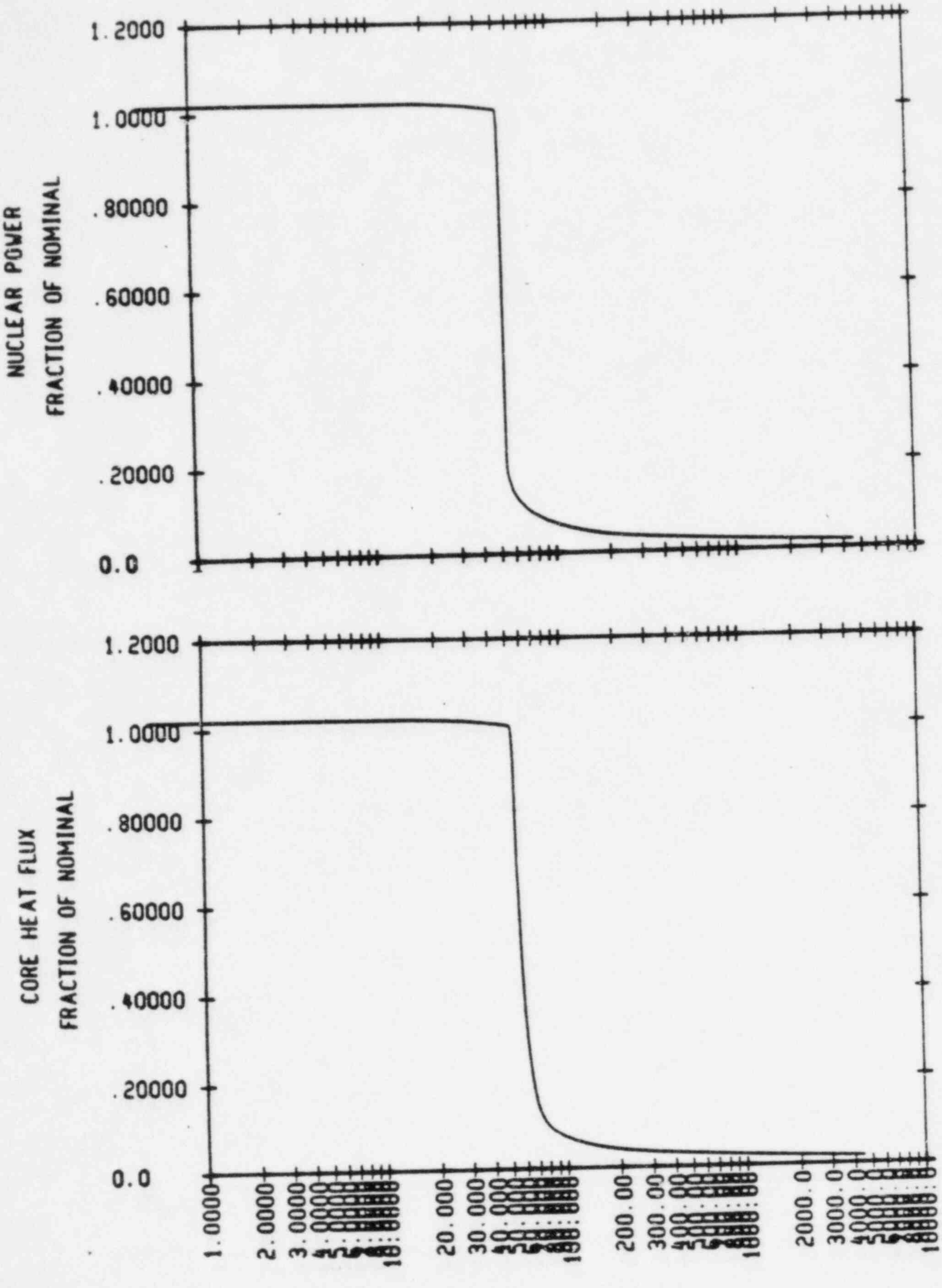


FIGURE 35

NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME (LOSS OF NOMINAL FEEDWATER)

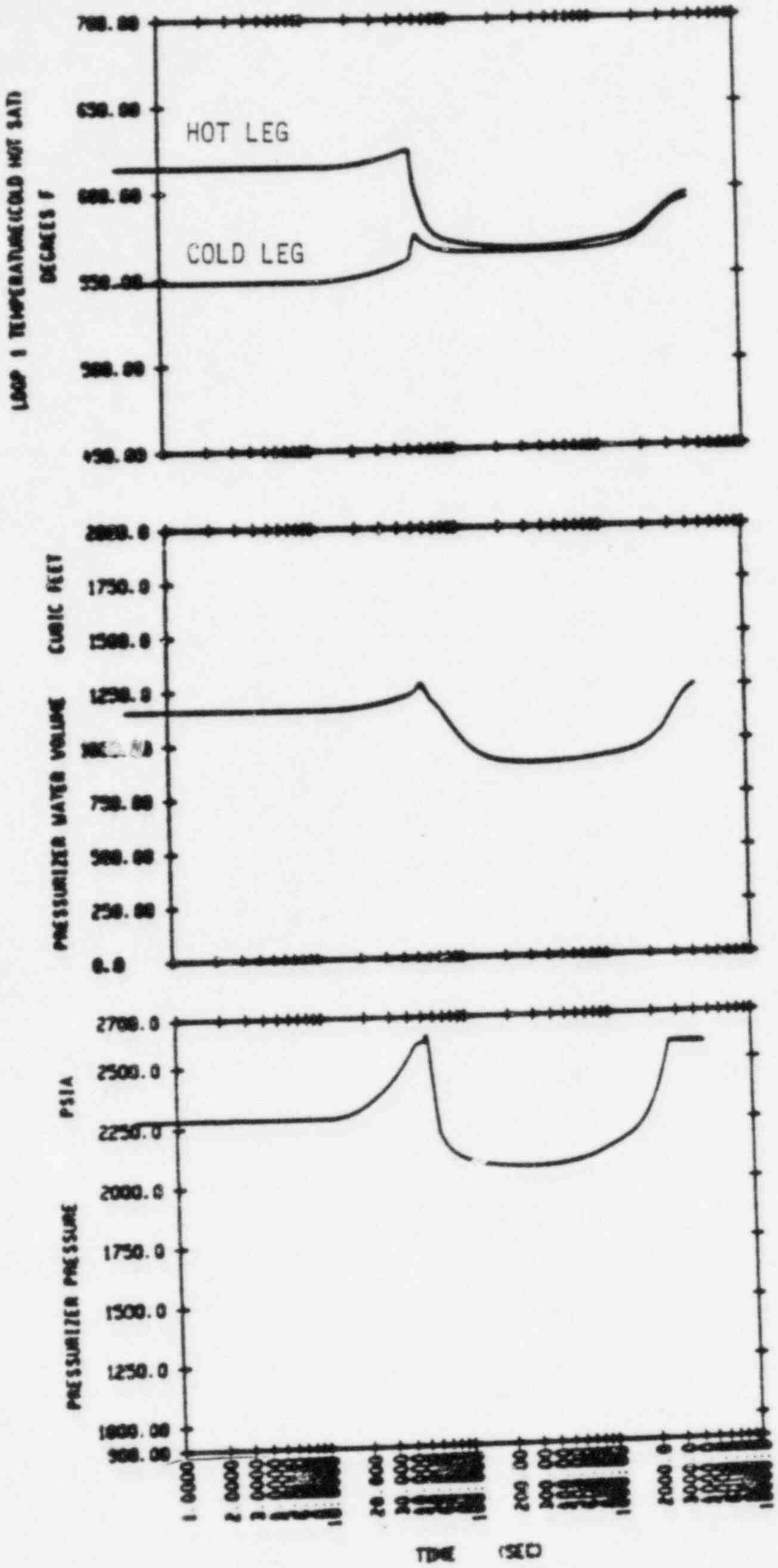


FIGURE 36

LOOP TEMPERATURE, PRESSURIZER WATER VOLUME AND PRESSURIZER PRESSURE VERSUS TIME (LOSS OF NOMINAL FEEDWATER)

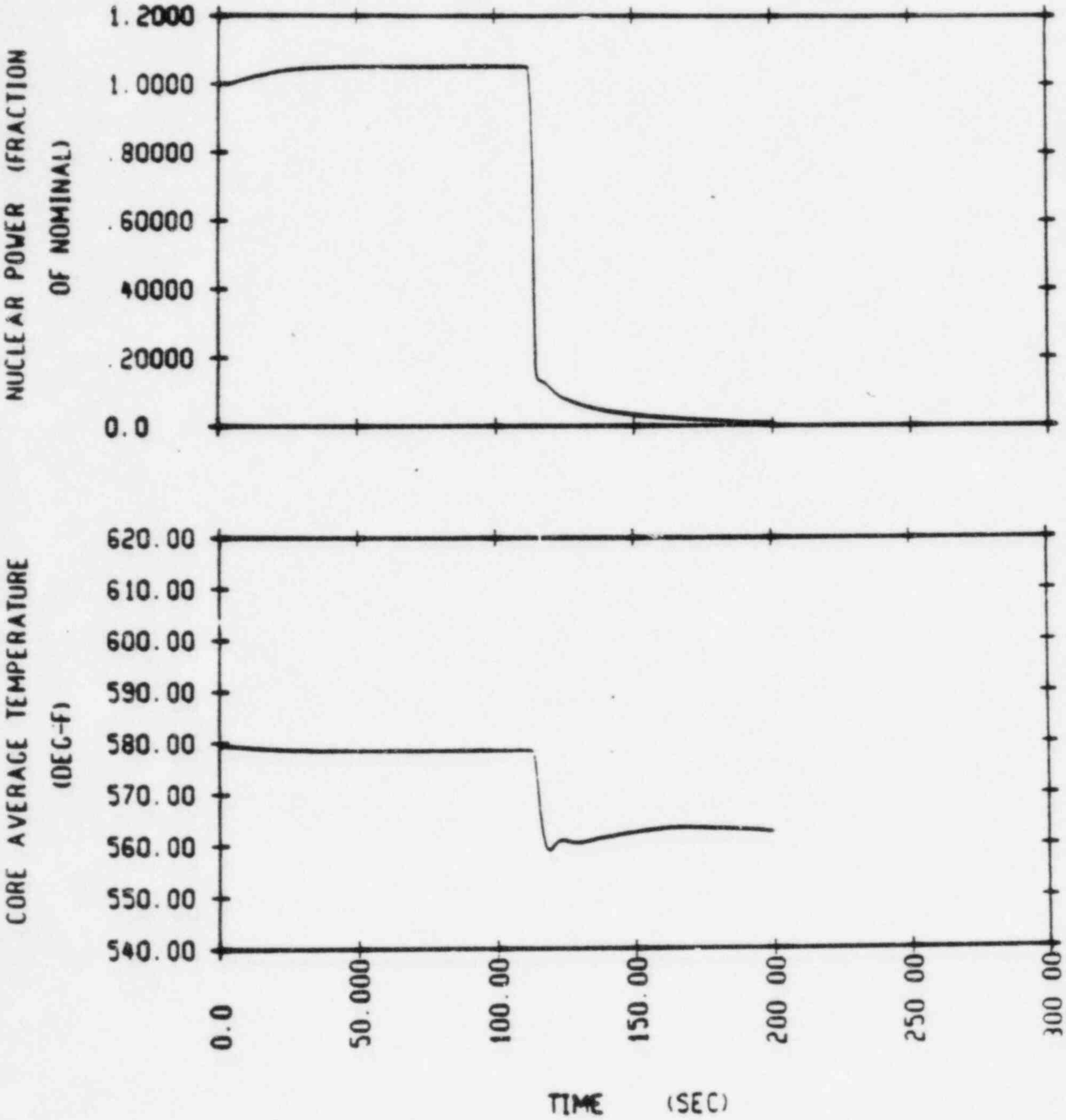


FIGURE 37

NUCLEAR POWER AND CORE AVERAGE TEMPERATURE VERSUS TIME
FEEDWATER MALFUNCTION WITH AUTOMATIC ROD CONTROL

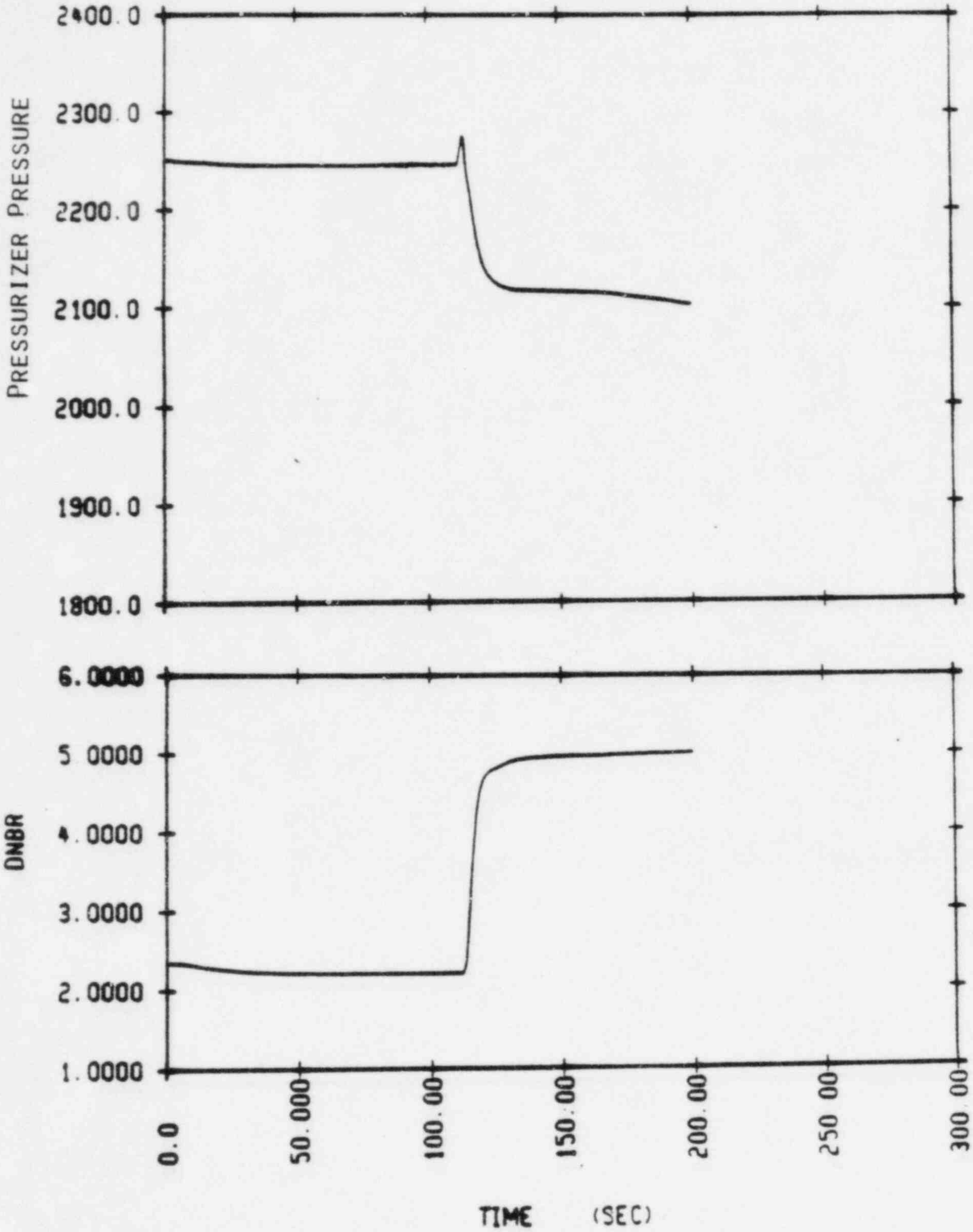


FIGURE 38

PRESSURIZER PRESSURE AND DNBR VERSUS TIME
FEEDWATER MALFUNCTION WITH AUTOMATIC ROD CONTROL

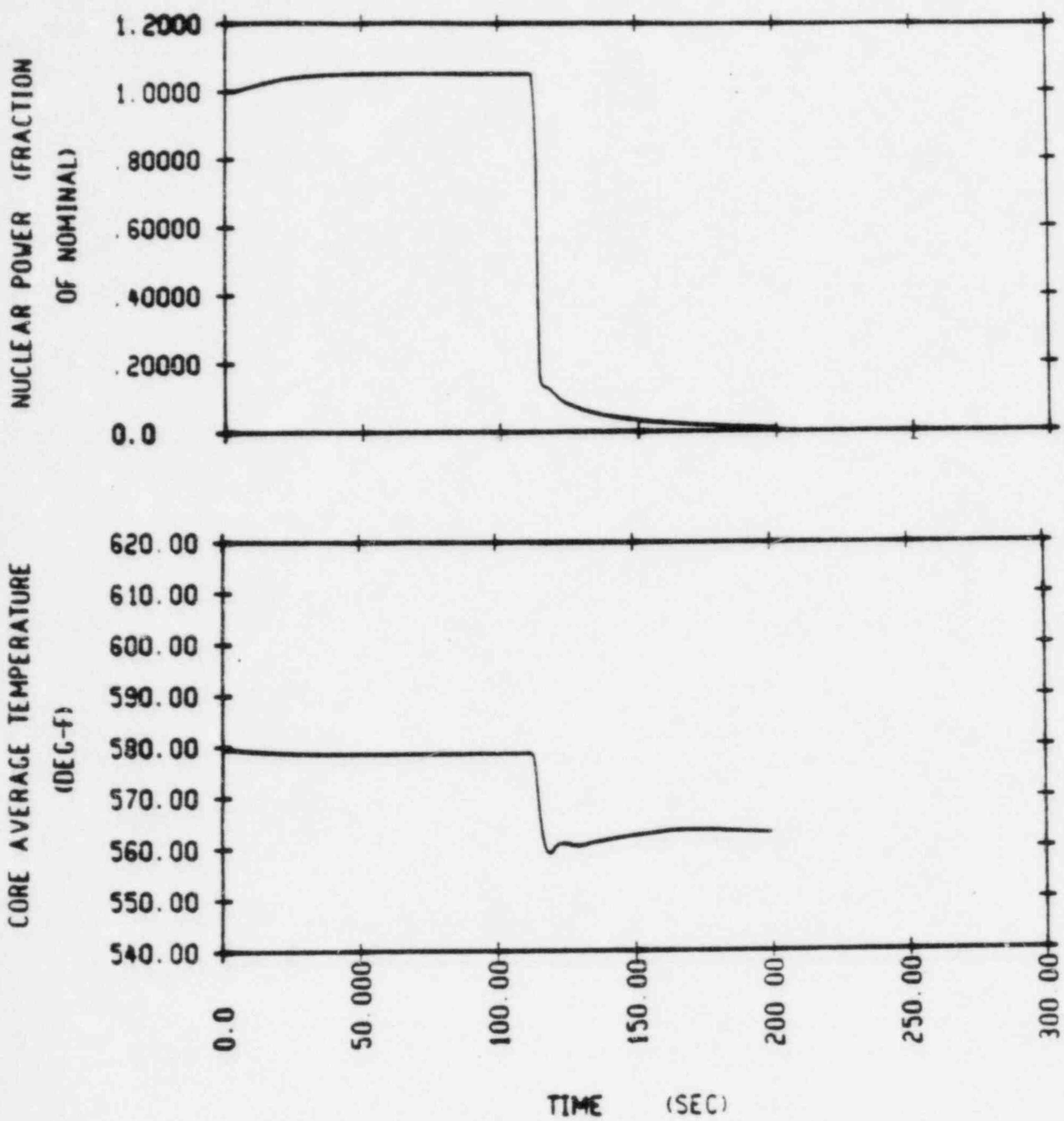


FIGURE 39

NUCLEAR POWER AND CORE AVERAGE TEMPERATURE VERSUS TIME
FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL

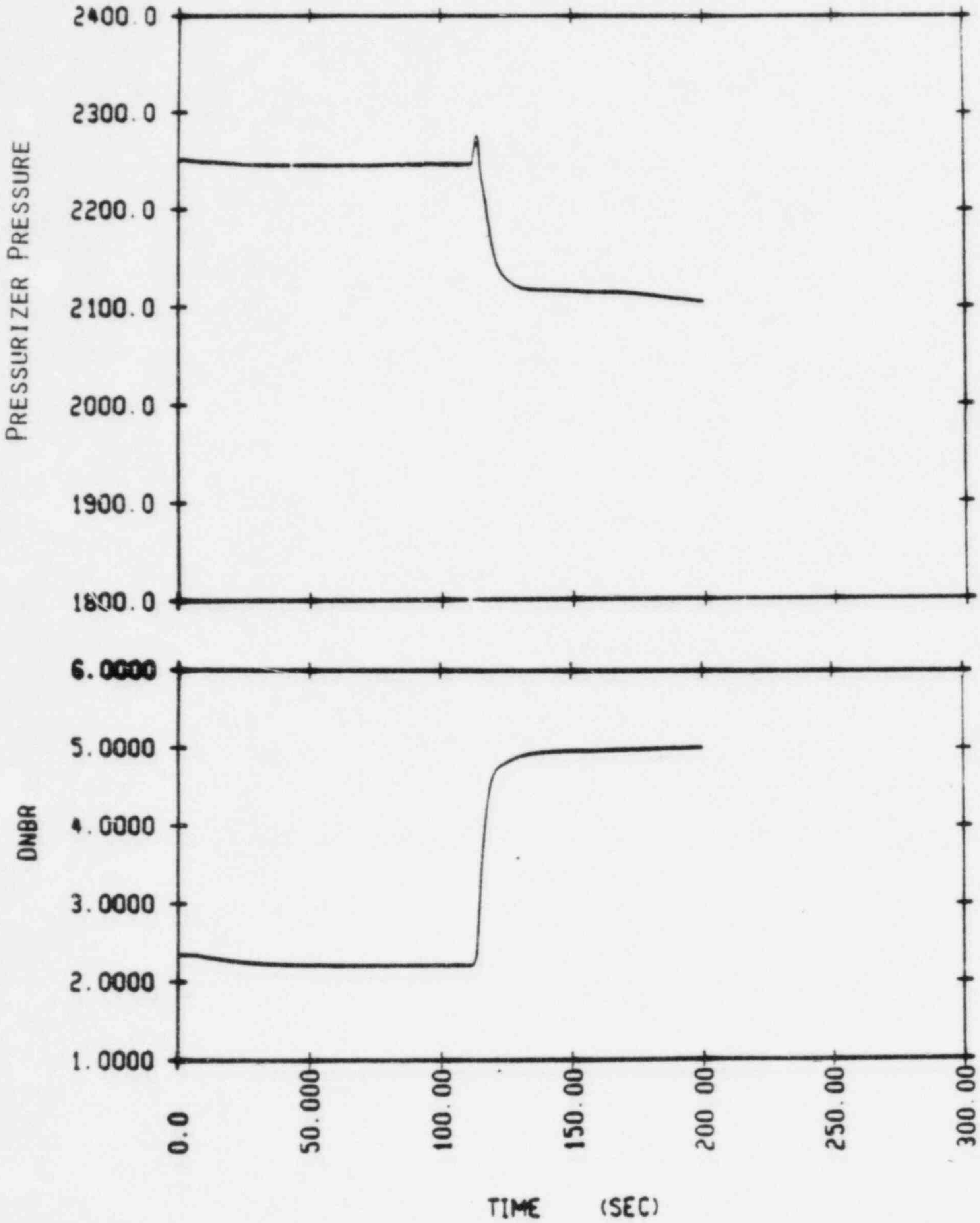


FIGURE 40

PRESSURIZER PRESSURE AND DNBR VERSUS TIME
FEEDWATER MALFUNCTION WITH MANUAL ROD CONTROL

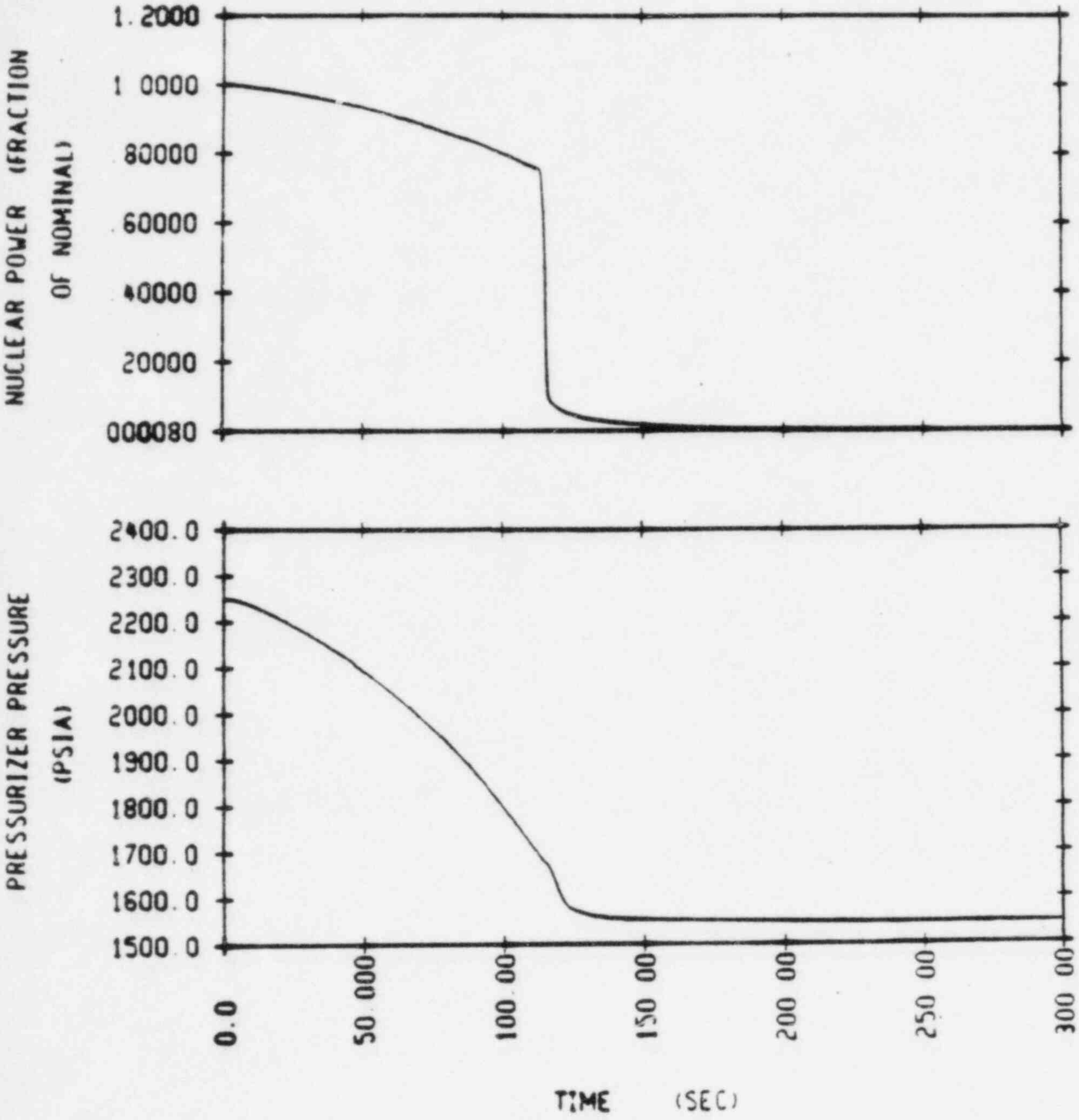


FIGURE 41

NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME FOR EXCESSIVE LOAD INCREASE MINIMUM REACTIVITY FEEDBACK WITH MANUAL ROD CONTROL

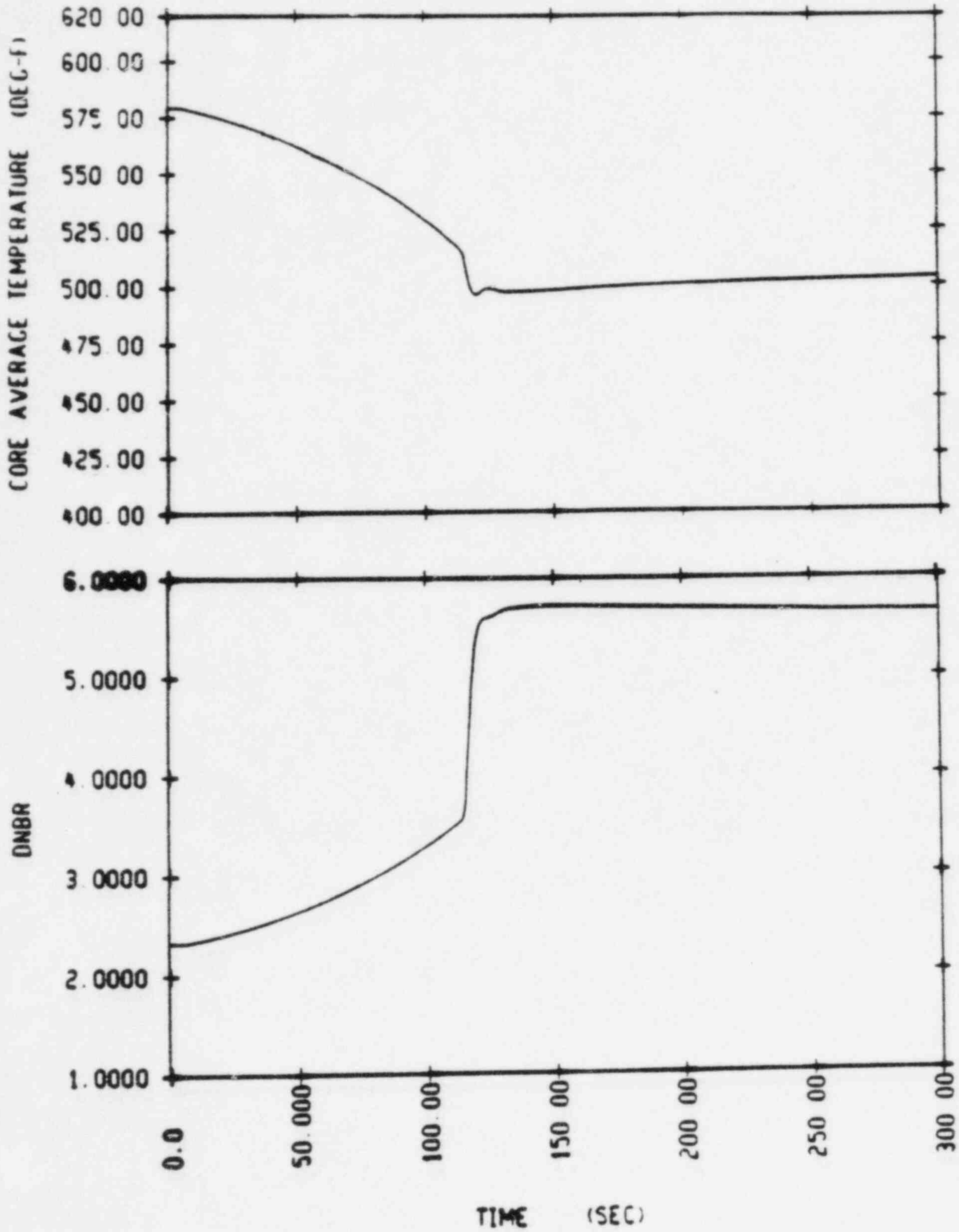


FIGURE 42

CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME FOR EXCESSIVE LOAD INCREASE MINIMUM REACTIVITY FEEDBACK WITH MANUAL ROD CONTROL

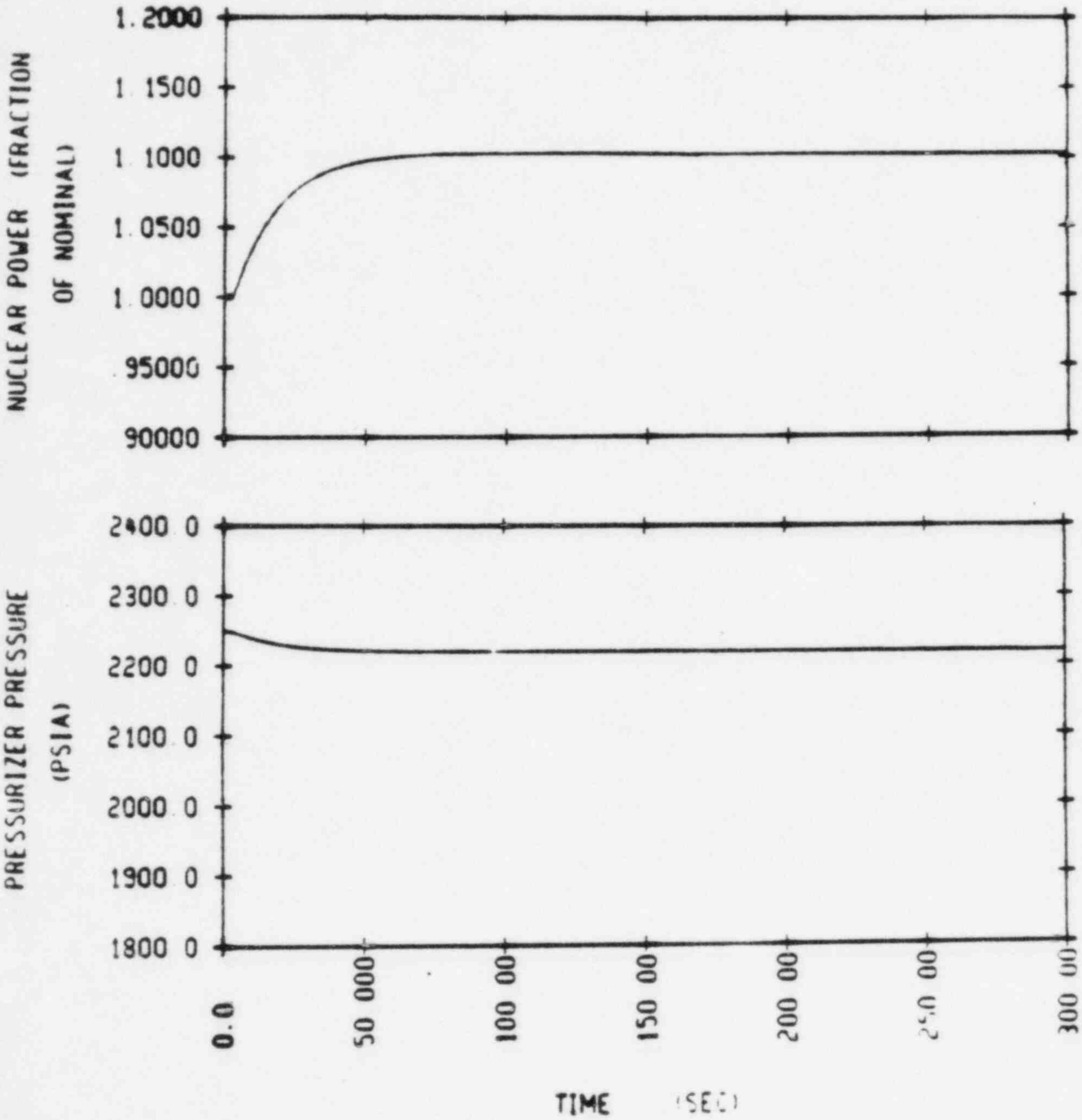


FIGURE 43

NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME FOR EXCESSIVE LOAD INCREASE MAXIMUM REACTIVITY FEEDBACK WITH MANUAL CONTROL

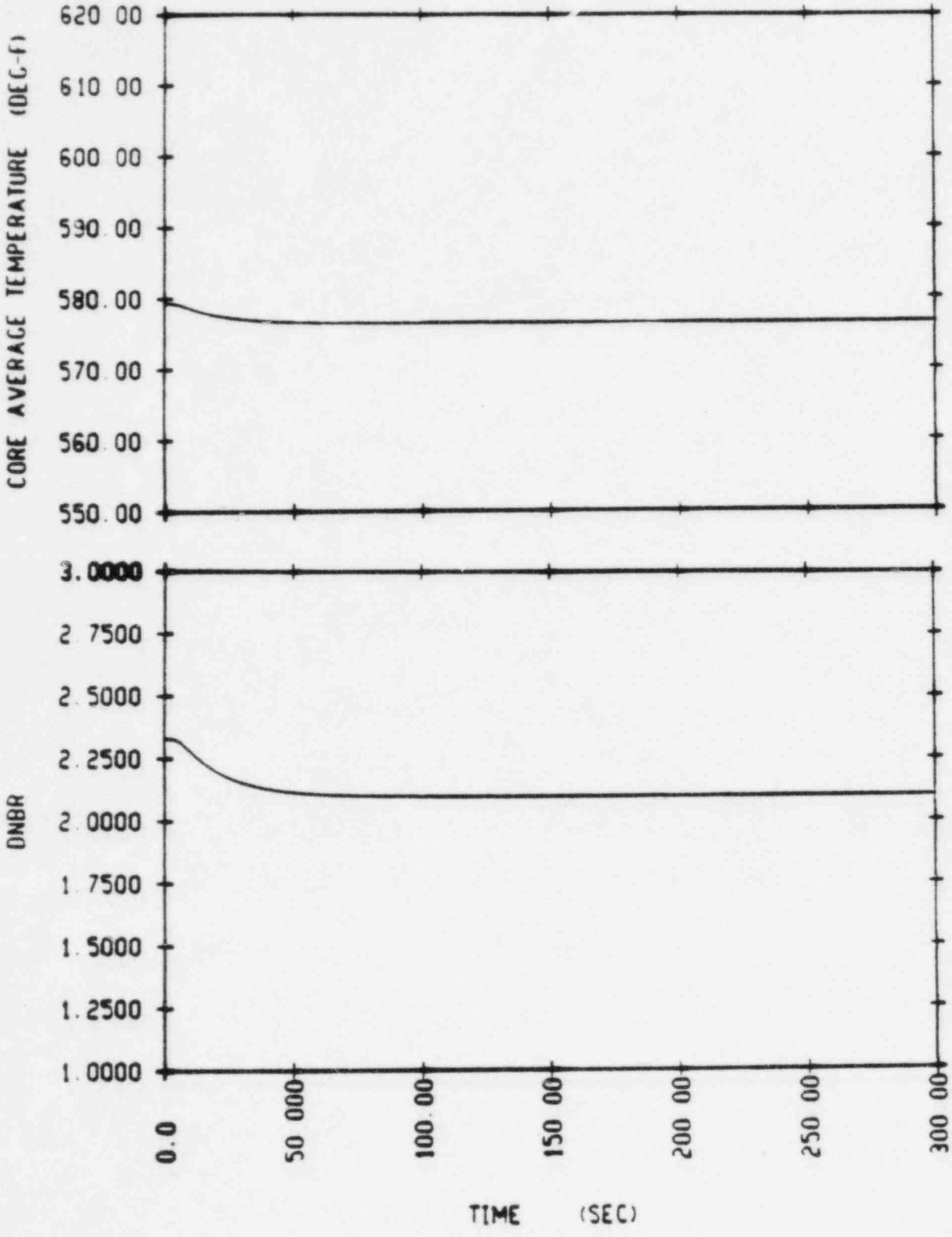


FIGURE 44

CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME FOR EXCESSIVE LOAD INCREASE
MAXIMUM REACTIVITY FEEDBACK WITH MANUAL CONTROL

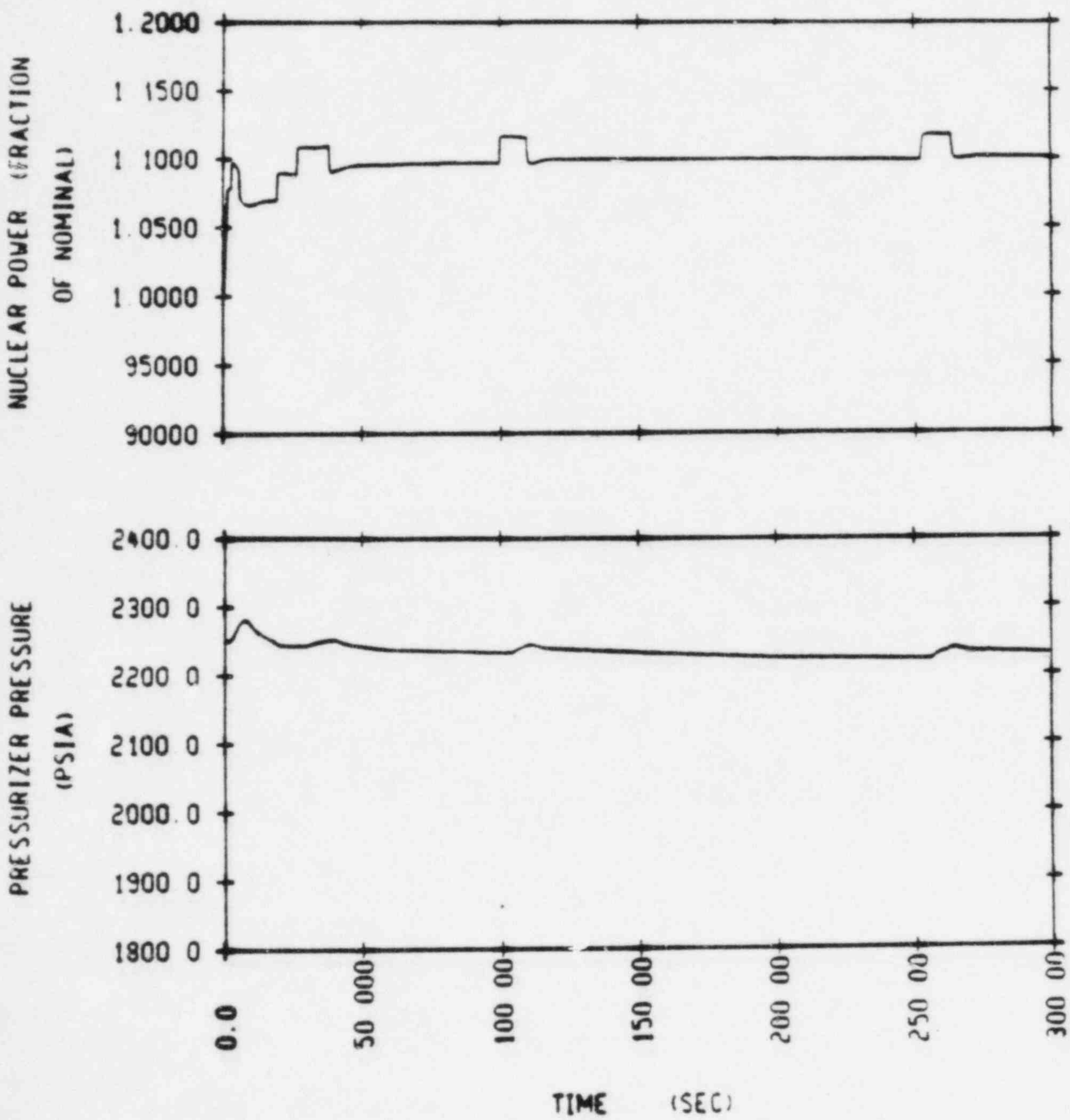


FIGURE 45

NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME FOR EXCESSIVE LOAD INCREASE
MINIMUM REACTIVITY FEEDBACK WITH AUTOMATIC ROD CONTROL

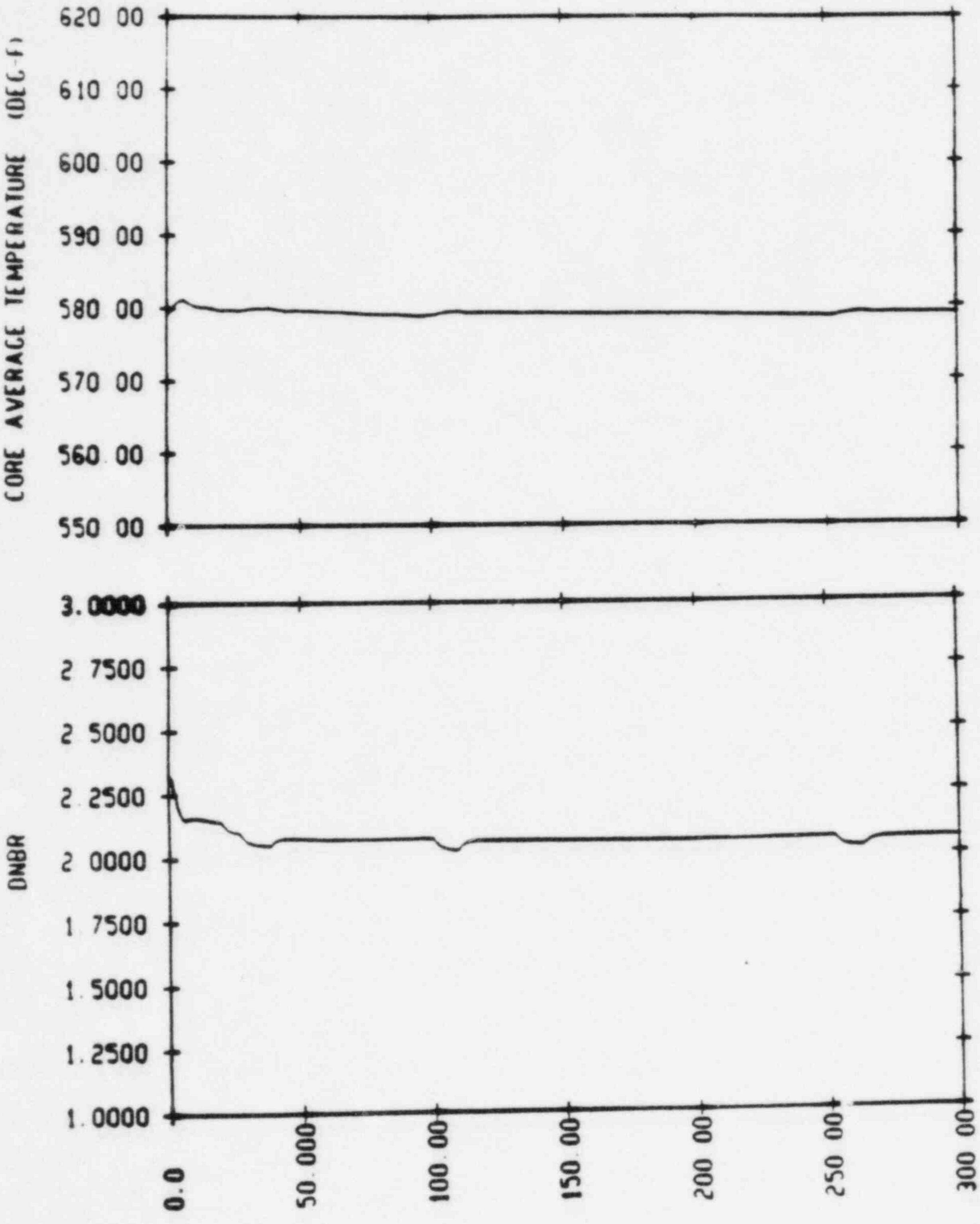


FIGURE 46

CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME FOR EXCESSIVE LOAD INCREASE
MINIMUM REACTIVITY FEEDBACK WITH AUTOMATIC ROD CONTROL

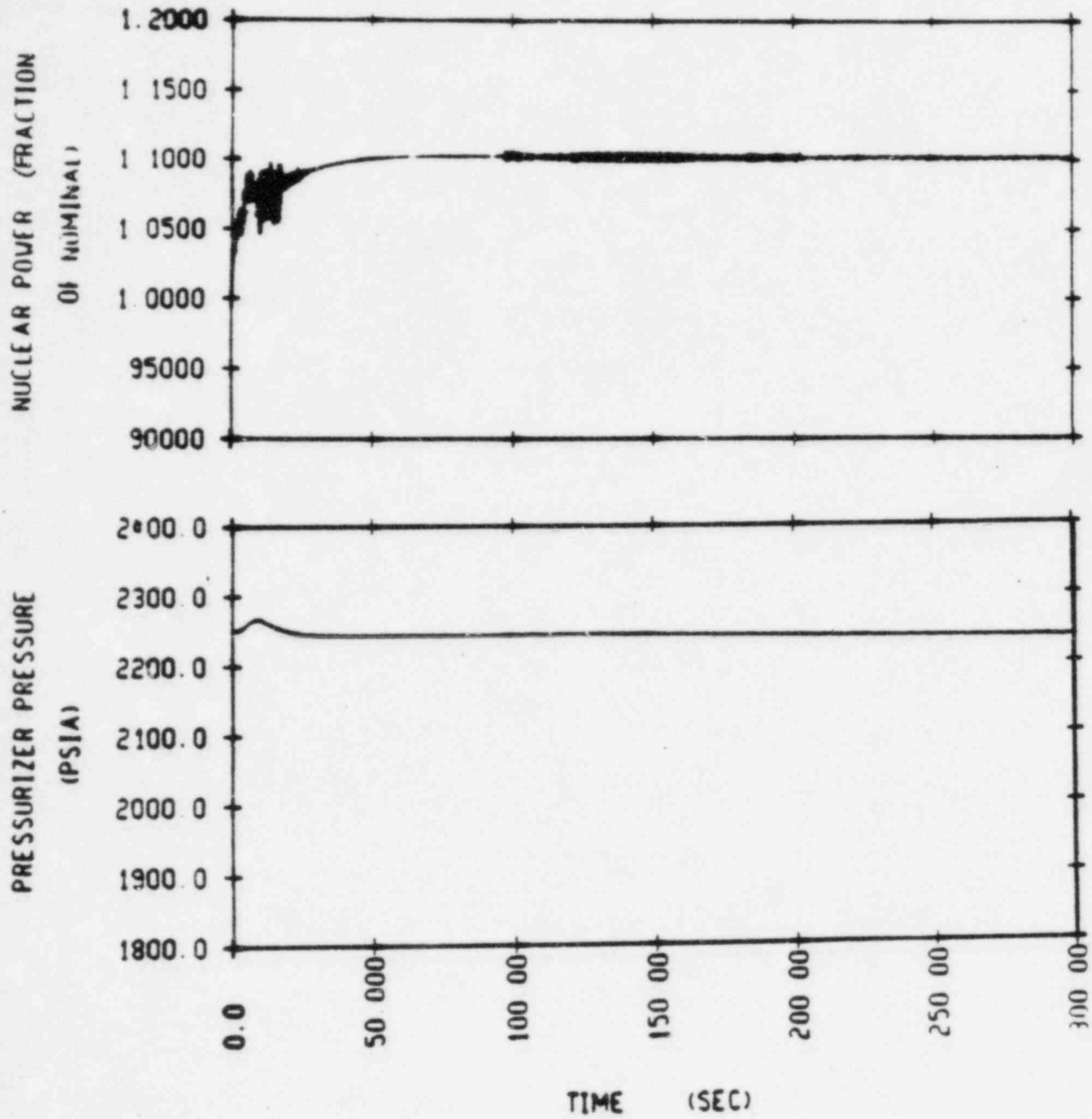


FIGURE 47

NUCLEAR POWER AND PRESSURIZER PRESSURE VERSUS TIME FOR EXCESSIVE LOAD INCREASE MAXIMUM REACTIVITY FEEDBACK, AUTOMATIC ROD CONTROL

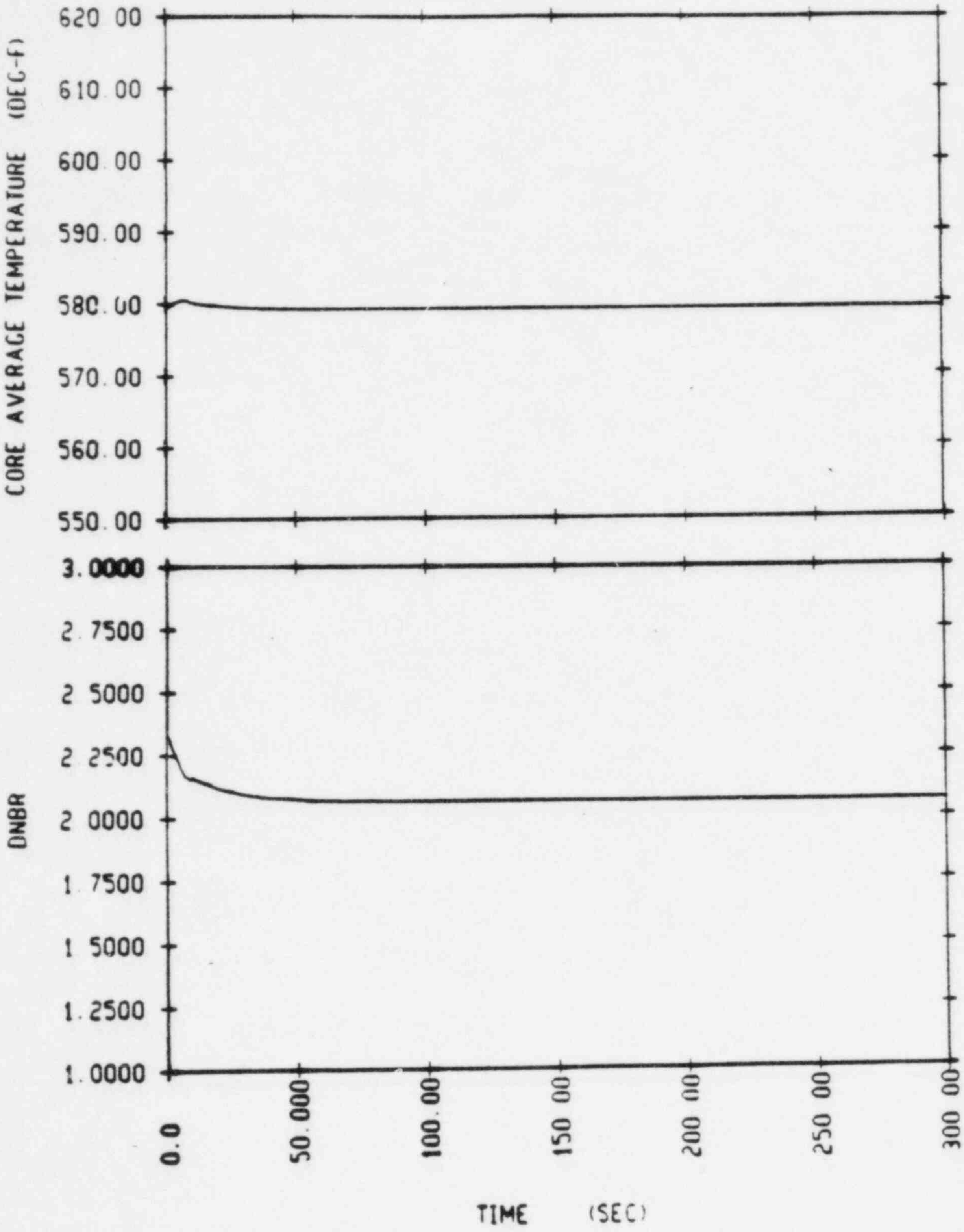


FIGURE 48

CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME FOR EXCESSIVE LOAD INCREASE
MAXIMUM REACTIVITY FEEDBACK, AUTOMATIC ROD CONTROL

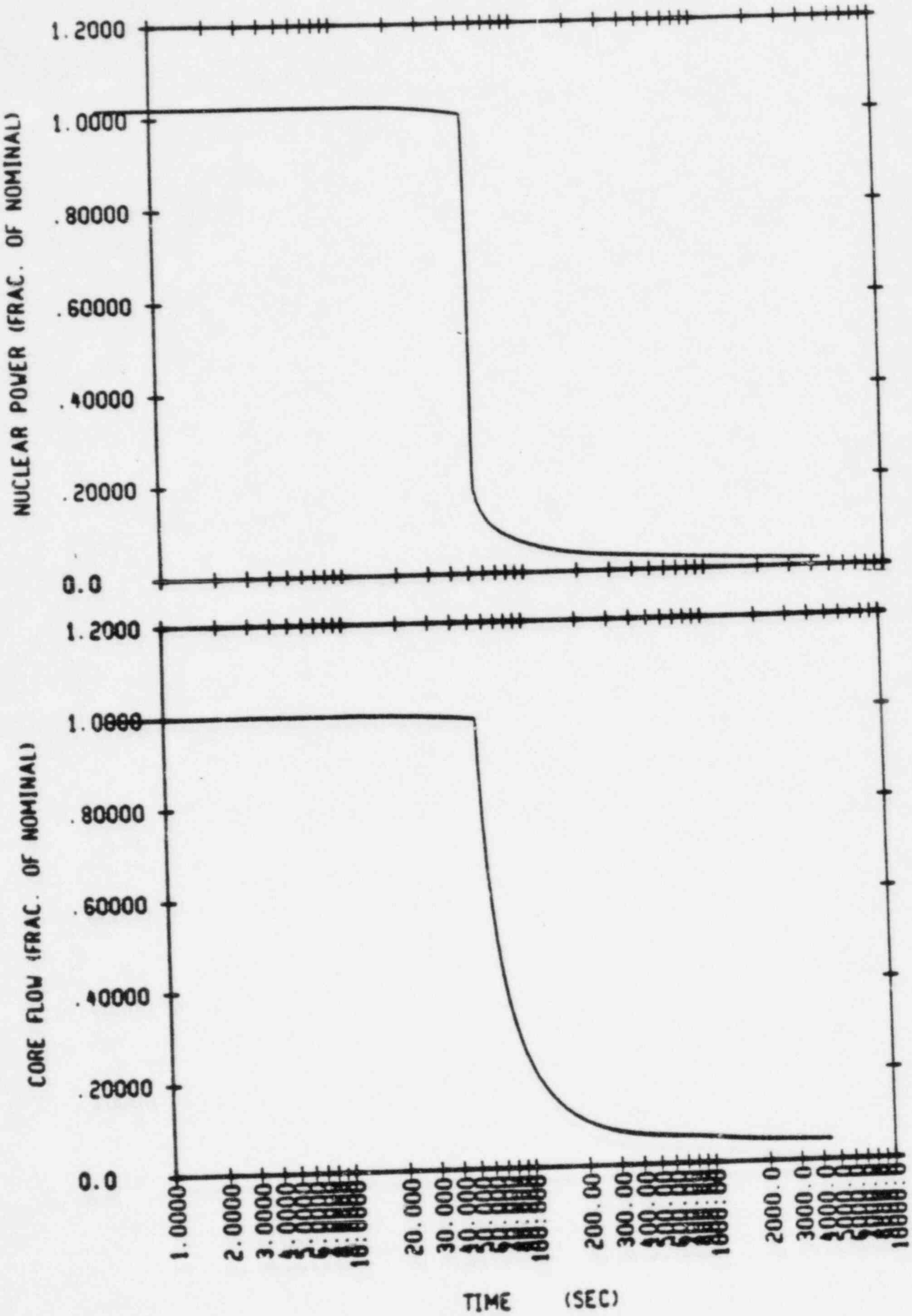


FIGURE 49
NUCLEAR POWER AND CORE FLOW VERSUS TIME (STATION BLACKOUT)

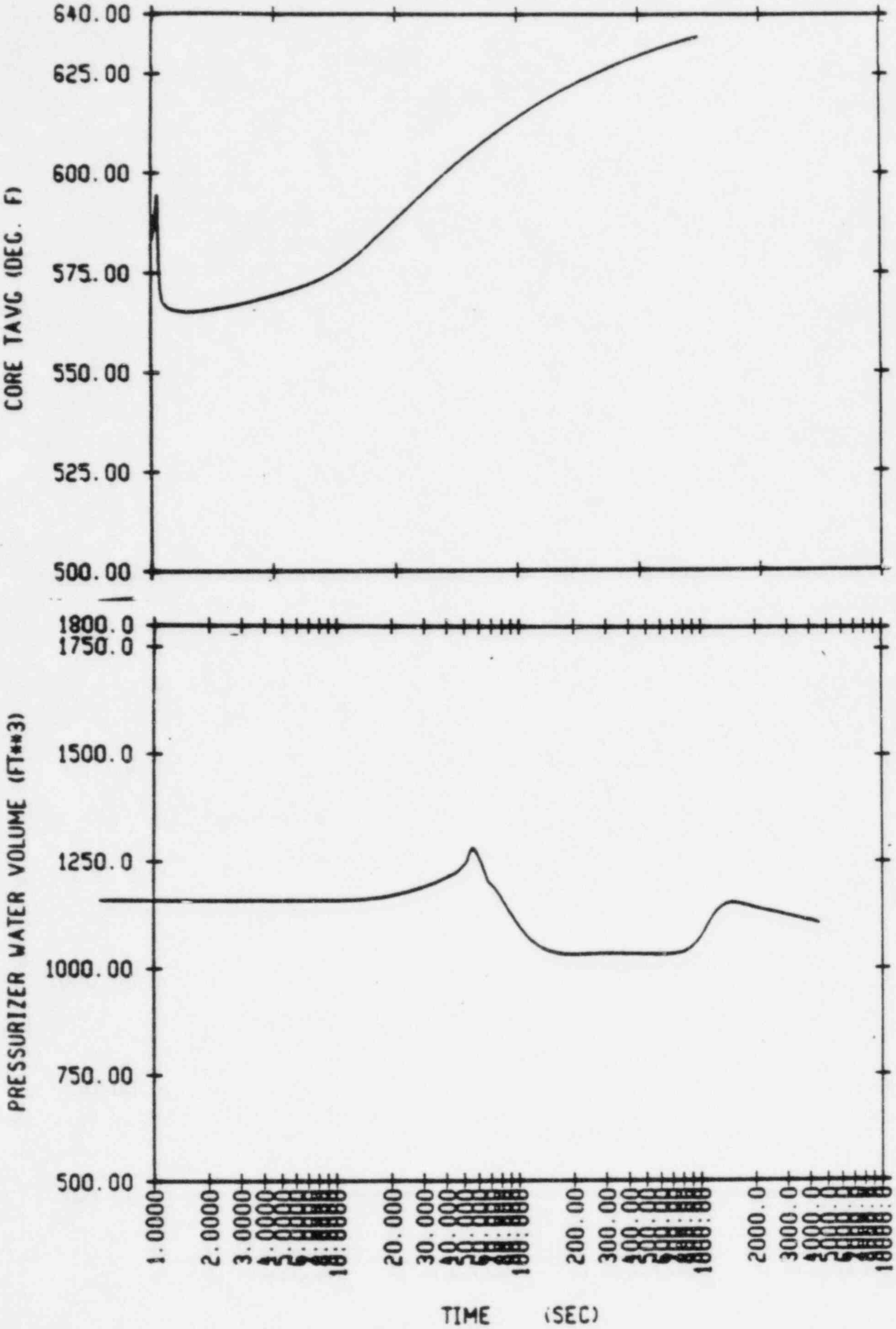


FIGURE 50

CORE AVERAGE TEMPERATURE AND PRESSURIZER WATER VOLUME VERSUS TIME (STATION BLACKOUT)

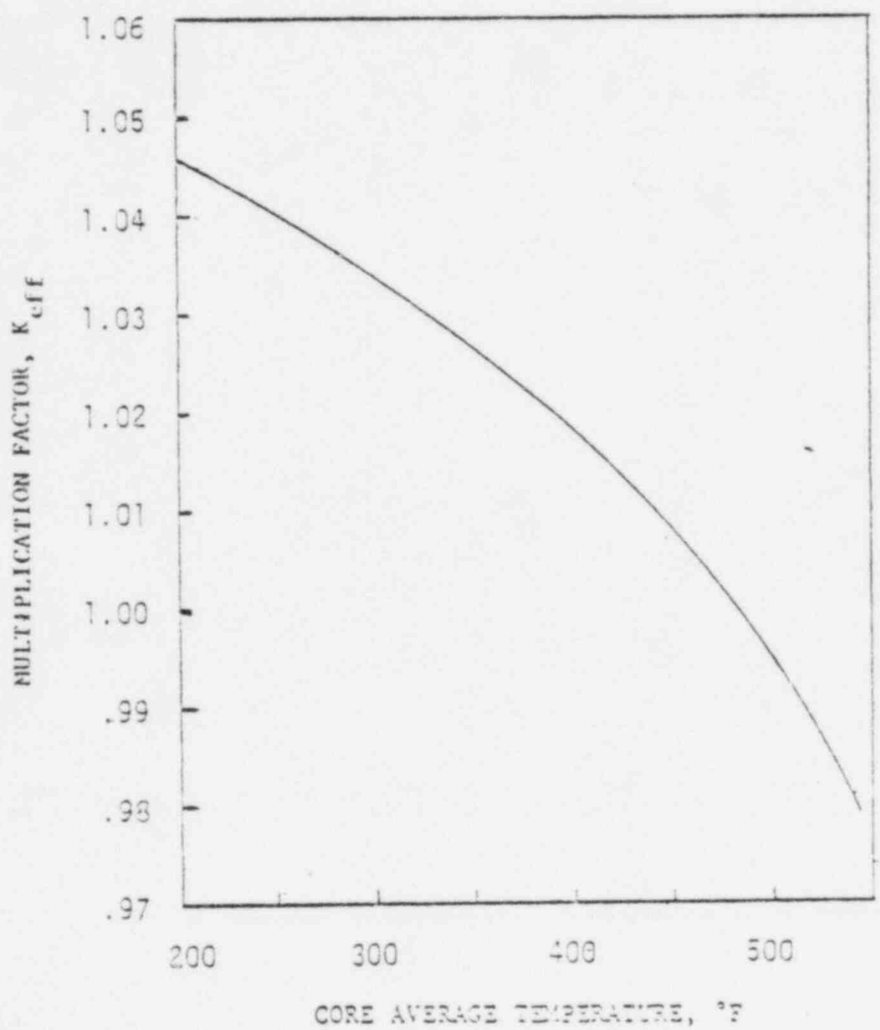


FIGURE 51

VARIATION OF REACTIVITY WITH CORE TEMPERATURE
AT 1000 PSIA FOR THE END OF LIFE RODDED CORE
WITH ONE CONTROL ROD ASSEMBLY STUCK
(ASSUMES ZERO POWER)

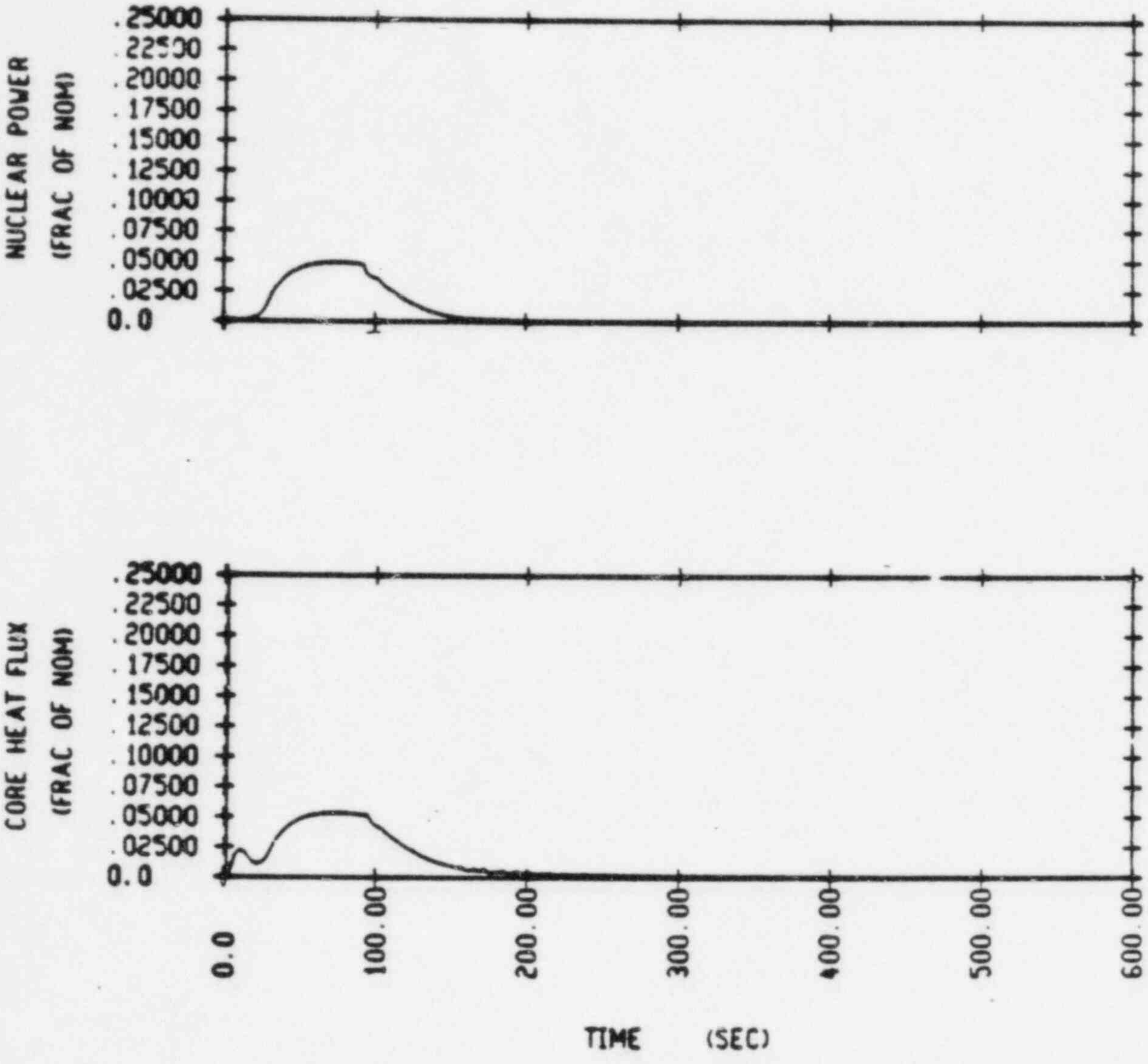


FIGURE 52

NUCLEAR POWER AND CORE HEAT FLUX VERSUS TIME
STEAMLINE BREAK DER INSIDE CONTAINMENT WITH POWER

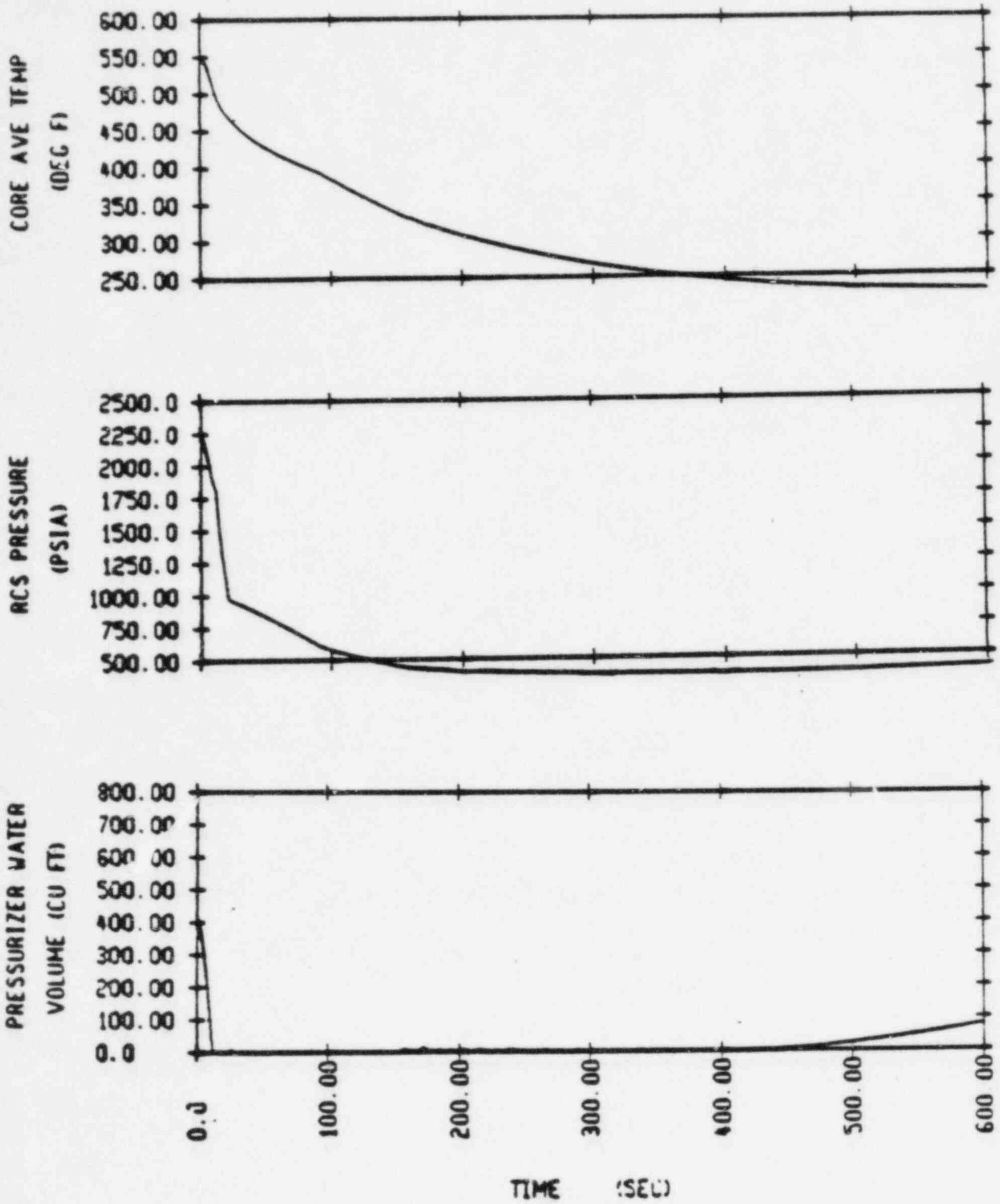


FIGURE 53

CORE AVERAGE TEMPERATURE, RCS PRESSURE AND PRESSURIZER WATER VOLUME VERSUS TIME, STEAMLINE BREAK DER INSIDE CONTAINMENT WITH POWER

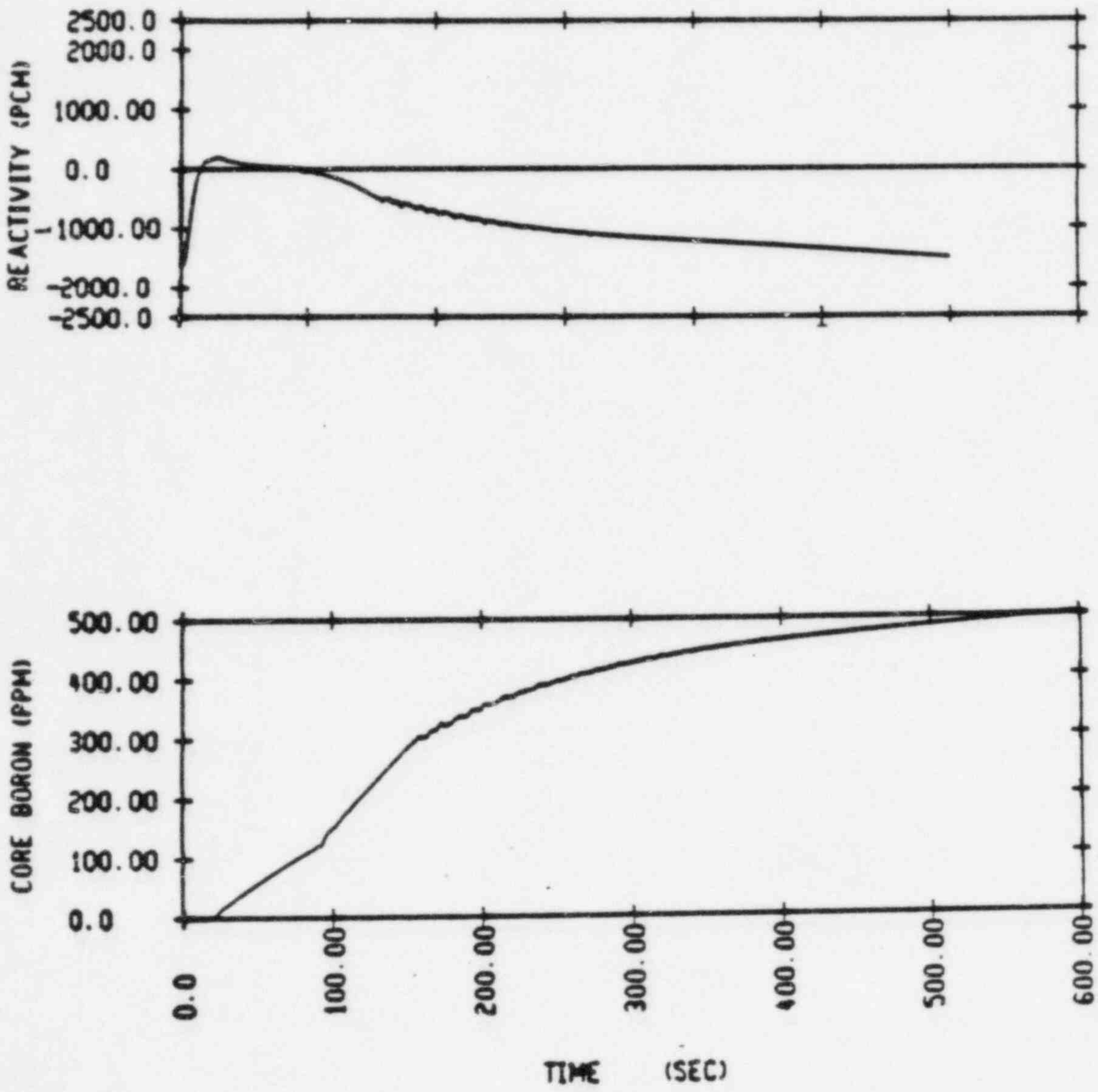


FIGURE 54

REACTIVITY AND CORE BORON CONCENTRATION VERSUS TIME
STEAMLINE BREAK DER INSIDE CONTAINMENT WITH POWER

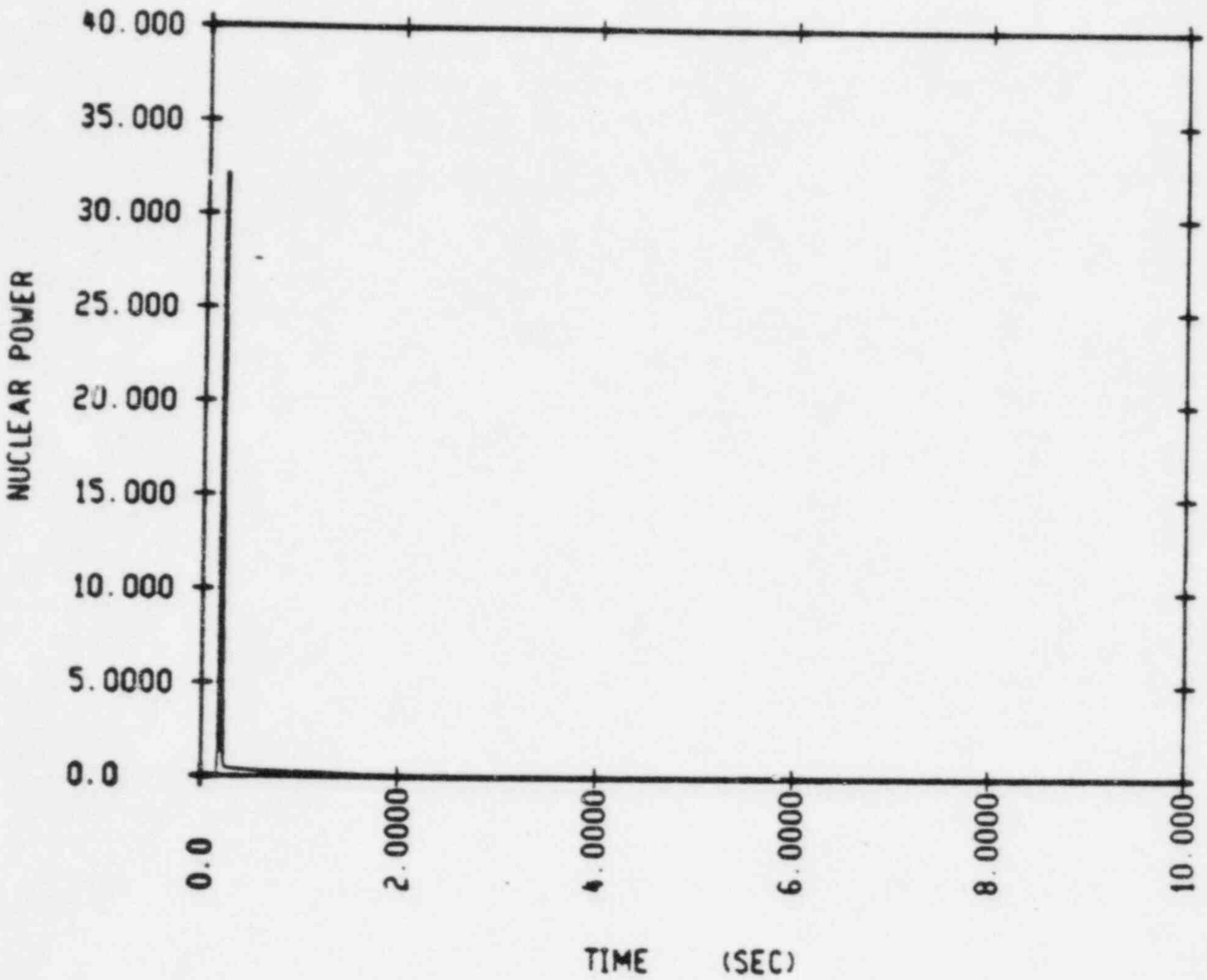


FIGURE 55

ROD EJECTION, HOT ZERO POWER
(END OF LIFE)

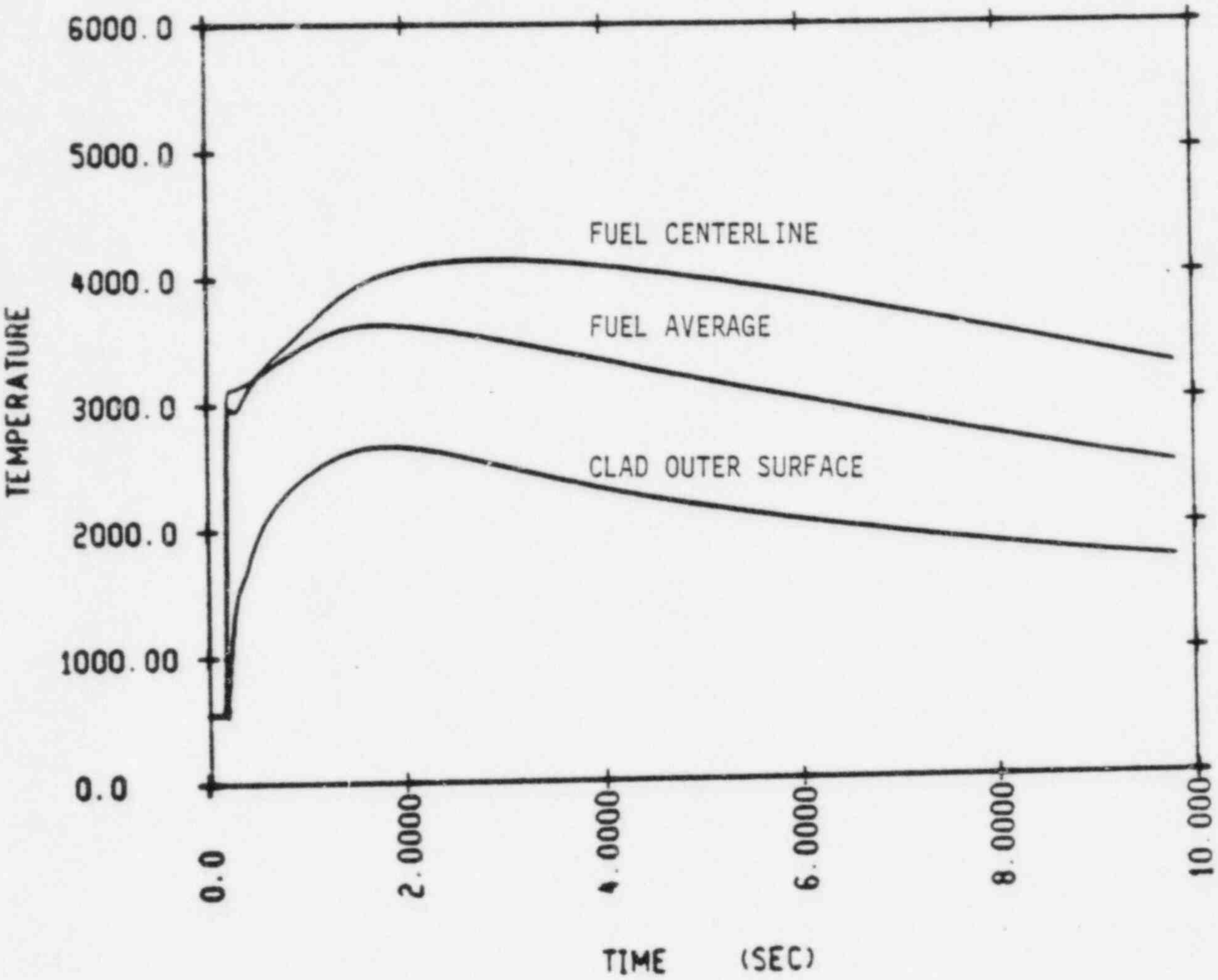


FIGURE 56

ROD EJECTION, HOT ZERO POWER, END OF LIFE ,
FUEL CENTERLINE, FUEL AVERAGE AND CLAD
OUTER SURFACE TEMPERATURE

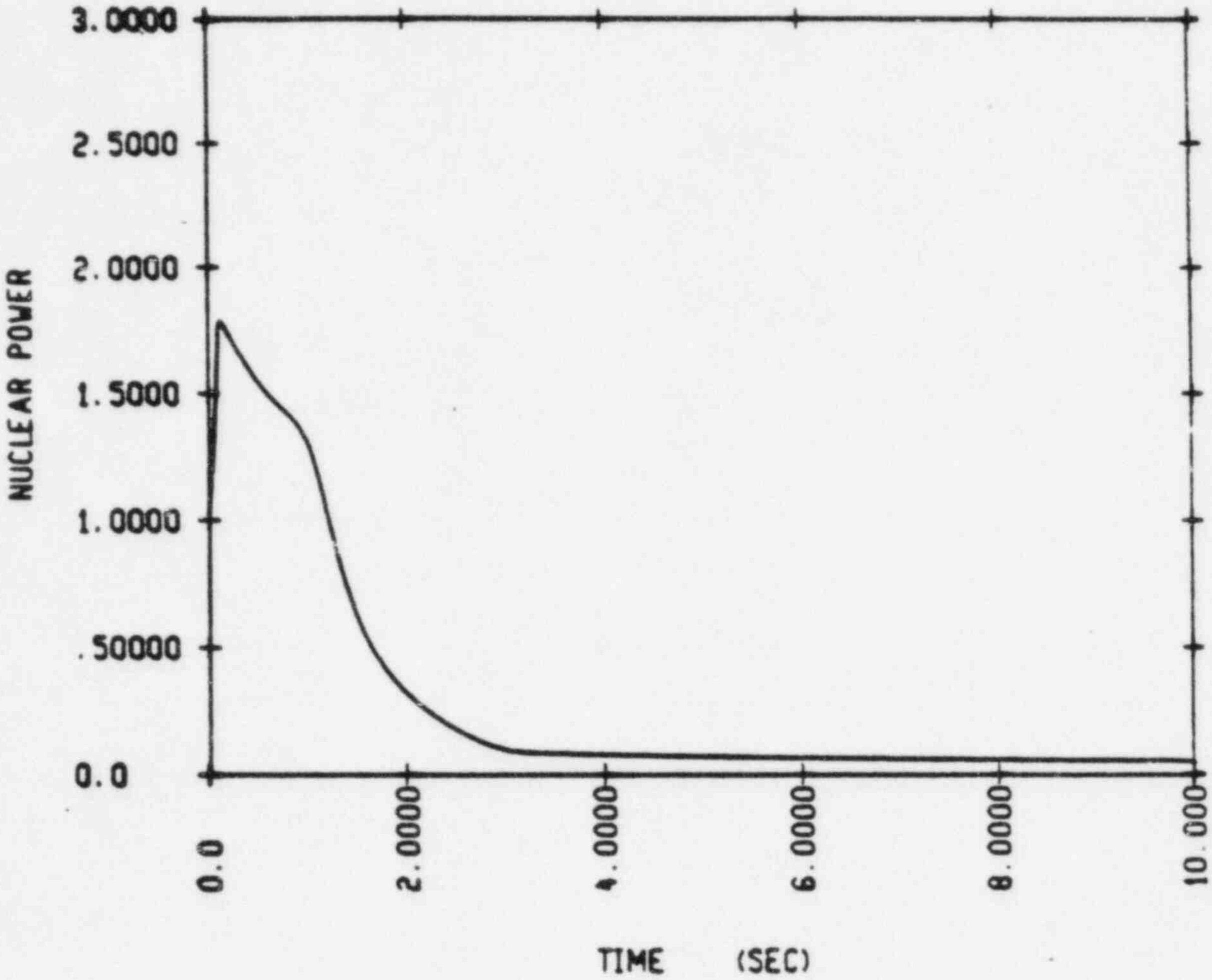


FIGURE 57

ROD EJECTION, HOT FULL POWER
END OF LIFE
NUCLEAR POWER VERSUS TIME

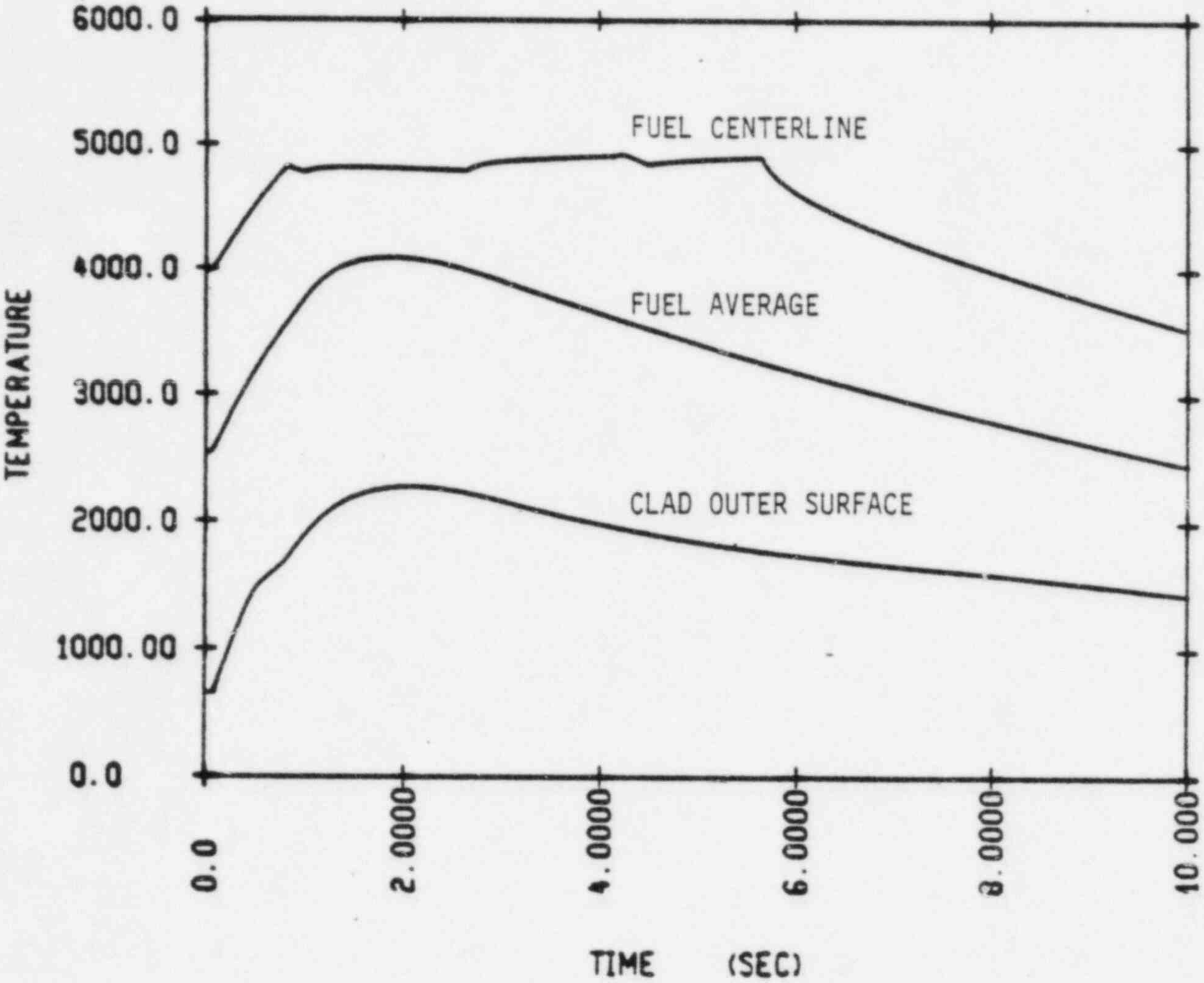


FIGURE 58

ROD EJECTION, HOT FULL POWER,
END OF LIFE , FUEL CENTERLINE, FUEL
AVERAGE AND FUEL OUTER SURFACE TEMPERATURE ROD

Attachment D to AEP:NRC:0745C
Large Break LOCA Safety Analysis

14.3.1.1 Major LOCA Analyses Applicable to Westinghouse Fuel

Identification of Causes and Frequency Classification

A loss-of-coolant accident (LOCA) is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of D. C. Cook Unit 1, but is postulated as a conservative design basis.

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (10 CFR 50.46 and Appendix K of 10 CFR 50 1974)(1) as follows:

1. The calculated peak fuel element clad temperature is below the requirement of 2,200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in emergency core cooling system (ECCS) performance following a LOCA. WASH-1400 (USNRC 1975)[10] presents a recent study in regards to the probability of occurrence of RCS pipe ruptures.

Sequence of Events and Systems Operations

Should a major break occur, depressurizaton of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken in the LOCA analysis for the boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

2. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

Description of Large Break Loss-of-Coolant Accident Transient

The sequence of events following a large break LOCA is presented in Table 14.3.1-6.

Before the break occurs, the unit is in an equilibrium condition; that is, the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel, continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10 CFR 50.[1] Thereafter the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered, both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the emergency feedwater system. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves, and also initiates emergency feedwater flow by starting the emergency feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. Since loss of offsite power (LOOP) is assumed, the RCPs are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the emergency core cooling water injected into the RCS bypassing the core are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is completed when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the latter stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during longterm cooling. Core temperatures have been reduced to longterm steady state levels associated with dissipation of residual heat generation. After the water level of the residual water storage tank (RWST) reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold recirculation phase of operation, in which spilled borated water is drawn from the engineered safety features (ESF) containment sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure.

Approximately 24 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Core and System Performance

Mathematical Model:

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Federal Register 1974).[1]

Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

A description of the various aspects of the LOCA analysis methodology is given by Bordelon, Massie, and Zordan (1974).[6] This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail by Bordelon et al. (1974)[5]; Kelly et al. (1974)[9]; Bordelon and Murphy (1974)[4]; and Bordelon et al. (1974).[6] Code modifications are specified in Reference 13. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code calculates this transient during the refill and reflood phases to the accident. The LOTIC computer code, described by Hsieh and Raymund in

WCAP-8355 (1975) and WCAP-8345 (1974)[3], calculates the containment pressure transient. The containment pressure transient is input to WREFLOOD for the purpose of calculating the reflood transient. The LOCTA-IV computer code calculates the thermal transient of the hottest fuel rod during the three phases. The Revised Pad Fuel Thermal Safety Model, described in Reference 15, generates the initial fuel rod conditions input to LOCTA-IV.

SATAN-VI calculates the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown phase, these data are transferred to the WREFLOOD code. Also, at the end-of-blowdown, the mass and energy release rates during blowdown are input to the LOTIC code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (that is, the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the reflood phase of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the containment through the break. WREFLOOD is also linked to the LOCTA-IV code, in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature. LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core.

The large break analysis was performed with the December 1981 version of the Evaluation Model, which includes modifications delineated by E. P. Rahe (1981)[7] and E. P. Rahe (1982).[2]

Input Parameters and Initial Conditions:

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS hot leg temperature.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from extensive sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions which were made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs, and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

A meeting was held at the Westinghouse Licensing Office in Bethesda on December 17, 1981 between members of the U. S. Nuclear Regulatory Commission and members of the Westinghouse Nuclear Safety Department to discuss the impact of maximum safety injection on the large break ECCS analysis on a generic basis. Further discussion of this issue is provided in a letter from E. P. Rahe, Manager of Westinghouse Nuclear Safety Department, to Robert L. Tedesco of the U. S. Nuclear Regulatory Commission.[14] A brief description of this issue is given below.

Westinghouse ECCS analyses currently assume minimum safeguards for the safety injection flow, which minimizes the amount of flow to the RCS by assuming maximum injection line resistances, degraded ECCS pump performance, and the loss of one residual heat removal (RHR) pump as the most limiting single failure. This is the limiting single failure assumption when offsite power is unavailable for most Westinghouse plants. However, for some Westinghouse four loop, non-UHI, non-burst node limited plants, the current nature of the Appendix K ECCS evaluation models is such that it may be more limiting to assume the maximum possible ECCS flow delivery. In that case, maximum safeguards which assume minimum injection line resistances, enhanced ECCS pump performance, and no single failure, result in the highest amount of flow delivered to the RCS.

Discussions of this phenomena with members of the U. S. Nuclear Regulatory Commission resulted in the following agreement:

> In future analyses, the single failure assumed will be the same as modelled currently. For four loop non-UHI non-burst node limited plants, an additional analysis will be repeated for the worst break size assuming no single failure. All cases which are analyzed will be reported to the NRC.

In accordance with this agreement, the worst break for D. C. Cook (C_D = 0.4) was re-analyzed, assuming maximum safeguards.

Results:

Based on the results of the LOCA sensitivity studies (Westinghouse 1974[12]; Salvatori 1974[11]; Johnson, Massie, and Thompson 1975[8]) the limiting large break was found to be the double ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 14.3.1-5 and 14.3.1-6.

The containment data used to generate the LOTIC backpressure transient are shown in Table 14.3.1-1. The mass and energy release data for the C_D = 0.4 break for the minimum and maximum safeguards cases are shown in Tables 14.3.1-2 and 14.3.1-3 respectively. Nitrogen release rates to the containment are given in Table 14.3.1-4.

Figures 14.3.1-1 through 14.3.1-64 present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

<u>Figures 14.3.1-1 through 14.3.1-12</u> The following quantities are presented at the clad burst location and at the hot spot (location of maximum clad temperature), both on the hottest fuel rod (hot rod):

1. fluid quality;
2. mass velocity;
3. heat transfer coefficient.

The heat transfer coefficient shown is calculated by the LOCTA-IV code.

<u>Figures 14.3.1-13 through 14.3.1-24</u> The system pressure shown is the calculated pressure in the core. The flow rate from the break is plotted as the sum of both ends for the guillotine break cases. The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.

<u>Figures 14.3.1-25 through 14.3.1-36</u> These figures show the hot spot clad temperature transient and the clad temperature transient at the burst location. The fluid temperature shown is also for the hot spot and burst location. The core flow (top and bottom) is also shown.

<u>Figures 14.3.1-37 through 14.3.1-44</u> These figures show the core reflood transient.

Figures 14.3.1-45 through 14.3.1-52 — These figures show the Emergency Core Cooling System flow for all of the cases analyzed. As described earlier, the accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in the refill and the reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.

Figures 14.3.1-53 through 14.3.1-54 — The containment pressure transient used in the analysis is also provided for the $C_D = 0.4$ minimum and maximum SI cases.

Figures 14.3.1-55 and 14.3.1-60 — These figures show the heat removal rates of the heat sinks found in the lower compartment and the heat removal by the lower containment drain, and the heat removal by the sump and LC sprays ($C_D = 0.4$ minimum and maximum SI cases).

Figures 14.3.1-61 through 14.3.1-64 — These figures show the temperature transients in both the upper and lower compartments of the containment and flow from the upper to lower compartments. Total heat removal in the lower compartment is the sum of all the heat removal rates shown (for $C_D = 0.4$ minimum and maximum SI cases).

The maximum clad temperature calculated for a large break is 2170°F, which is less than the Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 6.63 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

References for Section 14.3.1.1

1. "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register 1974, Volume 39, Number 3.

2. Rahe, E. P. (Westinghouse), letter to J. R. Miller (USNRC), Letter No. NS-EPRS-2679, November 1982.

3. Hsieh, T., and Raymund, M., "Long Term Ice Condenser Containment LOTIC Code Supplement 1," WCAP-8355 Supplement 1, May 1975, WCAP-8345 (Proprietary), July 1974.

4. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301 (Proprietary) and WCAP-8305 (Non-proprietary), 1974.

5. Bordelon, F. M. et al., "SATAN-VI Program: Comprehensive Space, Time Dependent Analysis of Loss-of-Coolant," WCAP-8302 (Proprietary) and WCAP-8306 (Non-proprietary), 1974.

6. Bordelon, F. M.; Massie, H. W.; and Zordan, T.A., "Westinghouse ECCS Evaluation Mode - Summary," WCAP-8339, 1974.

7. Rahe, E. P., "Westinghouse ECCS Evaluation Mode, 1981 Version," WCAP-9220-P-A (Proprietary Version), WCAP-9221-P-A (Non-proprietary Version), Revision 1, 1981.

8. Johnson, W. J.; Massie, H. W.; and Thompson, C. M., "Westinghouse ECCS-Four Loop Plant (17x17) Sensitivity Studies," WCAP-8586-P-A (Proprietary) and WCAP-8566-A (Non-proprietary), 1975.

9. Kelly, R. D. et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-proprietary), 1974.

10. U. S. Nuclear Regulatory Commission 1975, "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014.

11. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies," WCAP-8340 (Proprietary) and WCAP-8356 (Non-proprietary), 1974.

12. "Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-proprietary), 1974.

13. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471 (Proprietary) and WCAP-8472 (Non-proprietary), 1975.

14. Rahe, E. P. (Westinghouse). Letter to Robert L. Tedesco (USNRC), Letter No. NS-EPR-2538, December 1981.

15. Rahe, E. P., Westinghouse Letter to C. O. Thomas of NRC, Letter No. NS-EPR-2673, October 27, 1982; subject: "Westinghouse Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2 (Proprietary).

TABLE 14.3.1-1

LARGE BREAK

CONTAINMENT DATA

(ICE CONDENSER CONTAINMENT)

NET FREE VOLUME		
(Includes Distribution between Upper, Lower	UC	746,829 ft^3
and Deadend compartments)	LC	249,446
	DE	116,168
	IC	122,400
Initial Conditions		
Pressure		14.7 psia
Temperature for the Upper, Lower and	UC	100°F
Dead Ended Compartments	LC	120°F
	DE	120°F
RWST Temperature		70°F
Service Water Temperature		40°F
Temperature Outside Containment		-7°F
Initial Spray Temperature		70°F
Spray System		
Burnout Flow for a Spray Pump		3600 gpm
Number of Spray Pumps Operating		2
Post Accident Initiation of Spray System		40 secs
Distribution of the Spray Flow to the	LC	2835 gpm
Upper and Lower Compartments	UC	4365 gpm
Deck Fan		
Post Accident Initiation of Deck Fans		600 secs
Flow Rate Per Fan		39,000 cfm per fan
Hydrogen Skimmer System Flow Rate		2800 cfm per fan
Assumed Spray Efficiency of Water from Ice Condenser Drains		100%

TABLE 14.3.1-1

(continued)

STRUCTURAL HEAT SINKS

Compartment		Area (ft^2)	Thickness (ft)	Material
1.	LC	12,105	0.0469/2.0	steel/concrete
2.	LC	11,700	2.0	concrete
3.	LC	65,980	1.35	concrete
4.	LC	5,481	0.0833	steel
5.	LC	4,735	0.01147	steel
6.	LC	289	0.25	lead
7.	LC	14,690	0.0079	steel
8.	LC	3,439	0.1561	steel
9.	LC	5,775	0.009	steel
10.	LC	4,966	0.0096	steel
11.	LC	7,013	0.037	steel
12.	LC	2,457	0.0334	steel
13.	UC	378	.1667/.0365	steel/concrete
14.	UC	29,772	.0092	steel
15.	UC	8,033	.0209	steel
16.	UC	420	.0052	steel
17.	UC	29,330	1.47	concrete
18.	UC	34,125	0.0469/2.0	steel/concrete
19.	UC	210	.0052	steel

UC: Upper Compartment

LC: Lower Compartment

DE: Dead Ended Compartment

IC: Ice Condenser Compartment

TABLE 14.3.1-2

MASS AND ENERGY RELEASE RATES
MINIMUM SI

TIME (sec)	MASS (lb/sec)	ENERGY (BTU/sec)
0.	.5829E+05	.3082E+08
.2000E+01	.4849E+05	.2495E+08
.4000E+01	.3508E+05	.1831E+08
.6000E+01	.2747E+05	.1458E+08
.8000E+01	.2231E+05	.1207E+08
.1000E+02	.2081E+05	.1139E+08
.1200E+02	.1816E+05	.1032E+08
.1240E+02	.1700E+05	.9828E+07
.1400E+02	.1604E+05	.9348E+07
.1500E+02	.1490E+05	.8782E+07
.1600E+02	.1371E+05	.8166E+07
.1700E+02	.1262E+05	.7580E+07
.1900E+02	.1107E+05	.6643E+07
.2000E+02	.1046E+05	.6277E+07
.2100E+02	.9743E+04	.5884E+07
.2200E+02	.9256E+04	.5534E+07
.2300E+02	.7786E+04	.4880E+07
.2400E+02	.7322E+04	.4503E+07
.2500E+02	.6399E+04	.3951E+07
.2600E+02	.5422E+04	.3398E+07
.2700E+02	.6389E+04	.3418E+07
.2800E+02	.6947E+04	.3298E+07
.2900E+02	.7588E+04	.3176E+07
.3000E+02	.7484E+04	.2906E+07
.3200E+02	.6516E+04	.2162E+07
.3400E+02	.5140E+04	.1430E+07
.3600E+02	.3525E+04	.9890E+06
.3800E+02	.2665E+04	.6749E+06
.3950E+02	.7905E+03	.1672E+06
.4500E+02	.4880E+03	.2909E+05
.5000E+02	.4880E+03	.2909E+05
.5513E+02	.4880E+03	.2909E+05
.5573E+02	.4917E+03	.3386E+05
.5593E+02	.4917E+03	.3384E+05
.5603E+02	.4917E+03	.3383E+05
.5623E+02	.4916E+03	.3374E+05
.5633E+02	.4916E+03	.3373E+05
.6223E+02	.5196E+03	.7003E+05
.7183E+02	.8391E+03	.2273E+06
.8643E+02	.8787E+03	.2353E+06
.1045E+03	.8924E+03	.2328E+06
.1248E+03	.9013E+03	.2282E+06
.1468E+03	.9086E+03	.2226E+06
.1963E+03	.9222E+03	.2103E+06
.2539E+03	.9358E+03	.1972E+06
.3236E+03	.9507E+03	.1837E+06
.3625E+03	.9592E+03	.1775E+06

TABLE 14.3.1-3

MASS AND ENERGY RELEASE RATES
MAXIMUM SI

TIME (sec)	MASS (lb/sec)	ENERGY (BTU/sec)
0.	.5829E+05	.3082E+08
.2000E+01	.4848E+05	.2495E+08
.4000E+01	.3309E+05	.1831E+08
.6000E+01	.2747E+05	.1458E+08
.8000E+01	.2231E+05	.1207E+08
.1000E+02	.2081E+05	.1139E+08
.1200E+02	.1816E+05	.1032E+08
.1240E+02	.1700E+05	.9828E+07
.1400E+02	.1604E+05	.9348E+07
.1500E+02	.1490E+05	.8782E+07
.1600E+02	.1371E+05	.8166E+07
.1700E+02	.1262E+05	.7580E+07
.1900E+02	.1107E+05	.6643E+07
.2000E+02	.1046E+05	.6277E+07
.2100E+02	.9743E+04	.5884E+07
.2200E+02	.9256E+04	.5534E+07
.2300E+02	.7786E+04	.4880E+07
.2400E+02	.7322E+04	.4503E+07
.2500E+02	.6299E+04	.3951E+07
.2600E+02	.5422E+04	.3398E+07
.2700E+02	.6389E+04	.3418E+07
.2800E+02	.6947E+04	.3298E+07
.2900E+02	.7442E+04	.3168E+07
.3000E+02	.7339E+04	.2897E+07
.3200E+02	.6371E+04	.2153E+07
.3400E+02	.4994E+04	.1422E+07
.3600E+02	.3380E+04	.3903E+06
.3800E+02	.2519E+04	.6662E+06
.3950E+02	.6450E+03	.1585E+06
.4500E+02	.3425E+03	.2042E+06
.5000E+02	.3425E+03	.2042E+05
.5352E+02	.3425E+03	.2042E+05
.5412E+02	.3467E+03	.2588E+05
.5432E+02	.3467E+03	.2587E+05
.5442E+02	.3467E+03	.2586E+05
.5452E+02	.3467E+03	.2585E+05
.5462E+02	.3466E+03	.2579E+05
.6004E+02	.3750E+03	.6254E+05
.7004E+02	.1346E+04	.2307E+06
.8624E+02	.1475E+04	.2494E+06
.1065E+03	.1490E+04	.2477E+06
.1294E+03	.1498E+04	.2442E+06
.1543E+03	.1505E+04	.2404E+06
.2100E+03	.1515E+04	.2322E+06
.2747E+03	.1523E+04	.2239E+06
.3538E+03	.1535E+04	.2158E+06
.4494E+03	.1544E+04	.2074E+06

TABLE 14.3.1-4

NITROGEN MASS AND ENERGY

RELEASE RATES

Time (sec)	Flow Rate (lbs/sec)
37.5	71.9
39.5	60.7
45.5	37.2
47.5	31.6
53.5	18.8
55.5	15.6
61.5	8.5
63.5	6.9
70.3	186.0
72.3	158.0
78.5	97.3
80.5	82.4
86.3	48.5
88.3	40.0
94.3	21.9
96.3	18.2
102.2	11.7
104.2	10.5
110.2	7.6
112.2	6.8
126.2	3.3
128.2	2.9
138.2	1.8
140.2	1.6
146.2	1.2
148.2	1.1
174.2	0.25
176.2	0.075

TABLE 14.3.1-5

LARGE BREAK

Results	DECLG C_D=0.8 Min SI	DECLG C_D=0.6 Min SI	DECLG C_D=0.4 Min SI	DECLG C_D=0.4 Max SI
Peak Clad Temp. °F	1971	1977	1999	2170
Peak Clad Location Ft.	7.25	7.25	7.50	7.50
Local Zr/H_2O Reaction (Max)%	3.74	3.78	4.13	6.63
Local Zr/H_2O Location Ft.	7.50	7.25	7.50	7.50
Total Zr/H_2O Reaction %	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time sec	67.8	64.2	69.6	79.2
Hot Rod Burst Location Ft.	6.00	6.00	6.25	6.75

Calculation	
Licensed Core Power (Mwt) 102% of	3250
Peak Linear Power (kw/ft) 102% of	13.426
Peaking Factor (at License Rating)	2.00
Accumulator Water Volume (ft^3) per Accumulator	950

Cycle Analyzed Cycle 8

TABLE 14.3.1-6

LARGE BREAK
TIME SEQUENCE OF EVENTS

	Min SI DECLG C_D=0.8 (sec)	Min SI DECLG C_D=0.6 (sec)	Min SI DECLG C_D=0.4 (sec)	Max SI DECLG C_D=0.4 (sec)
START	0.00	0.00	0.00	0.00
Reactor Trip Signal	0.59	0.59	0.60	0.60
Safety Injection Signal	3.73	3.81	4.05	4.05
Accumulator Injection	12.80	15.20	20.80	20.50
End of Blowdown	30.24	31.05	39.29	38.70
Bottom of Core Recovery	43.22	45.29	54.63	52.78
Accumulator Empty	57.09	59.37	66.42	67.45
Pump Injection	28.73	28.81	29.05	29.05

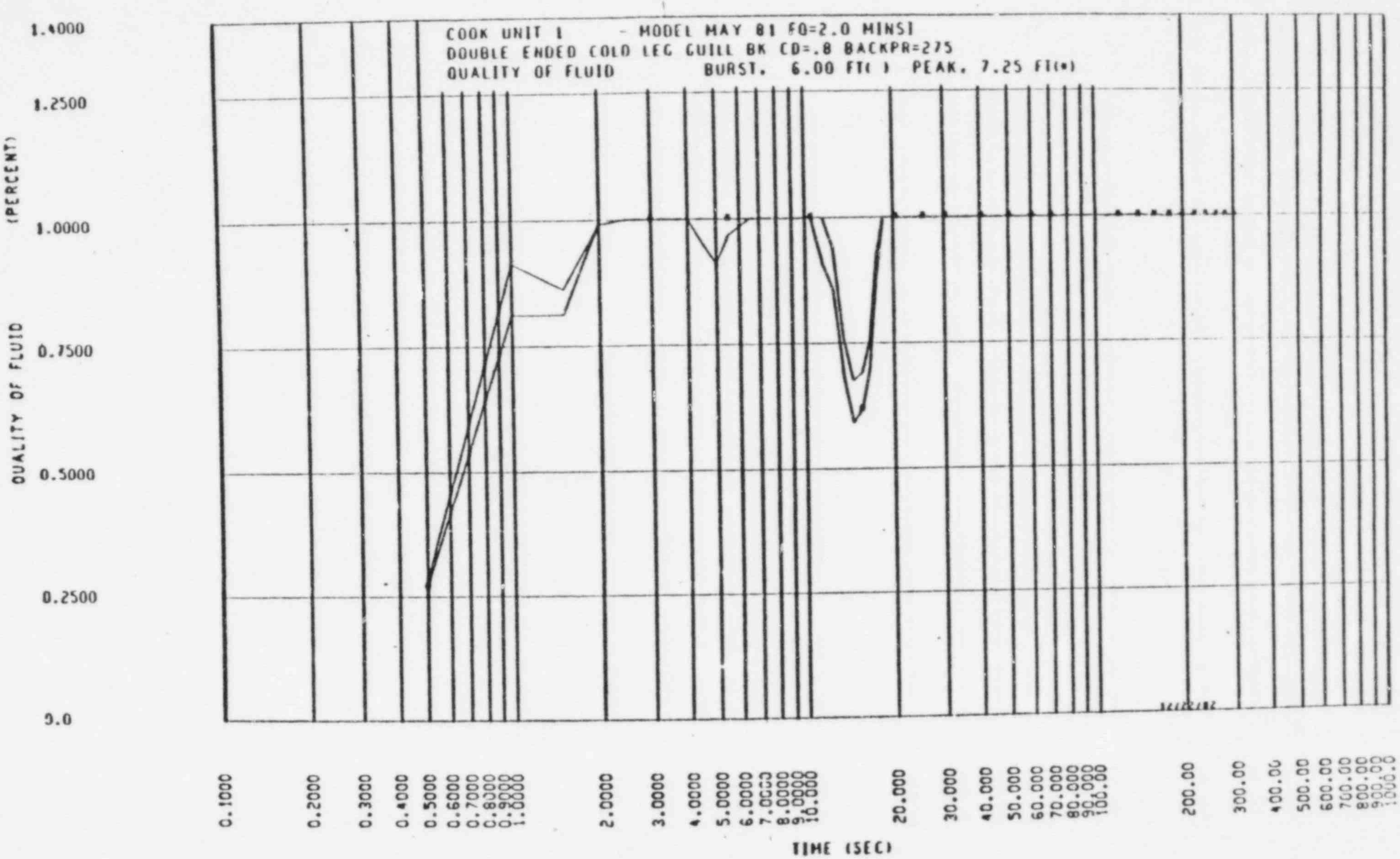


FIGURE 14.3.1-1 FLUID QUALITY DECLG(CD = 0.8) | MIH SI

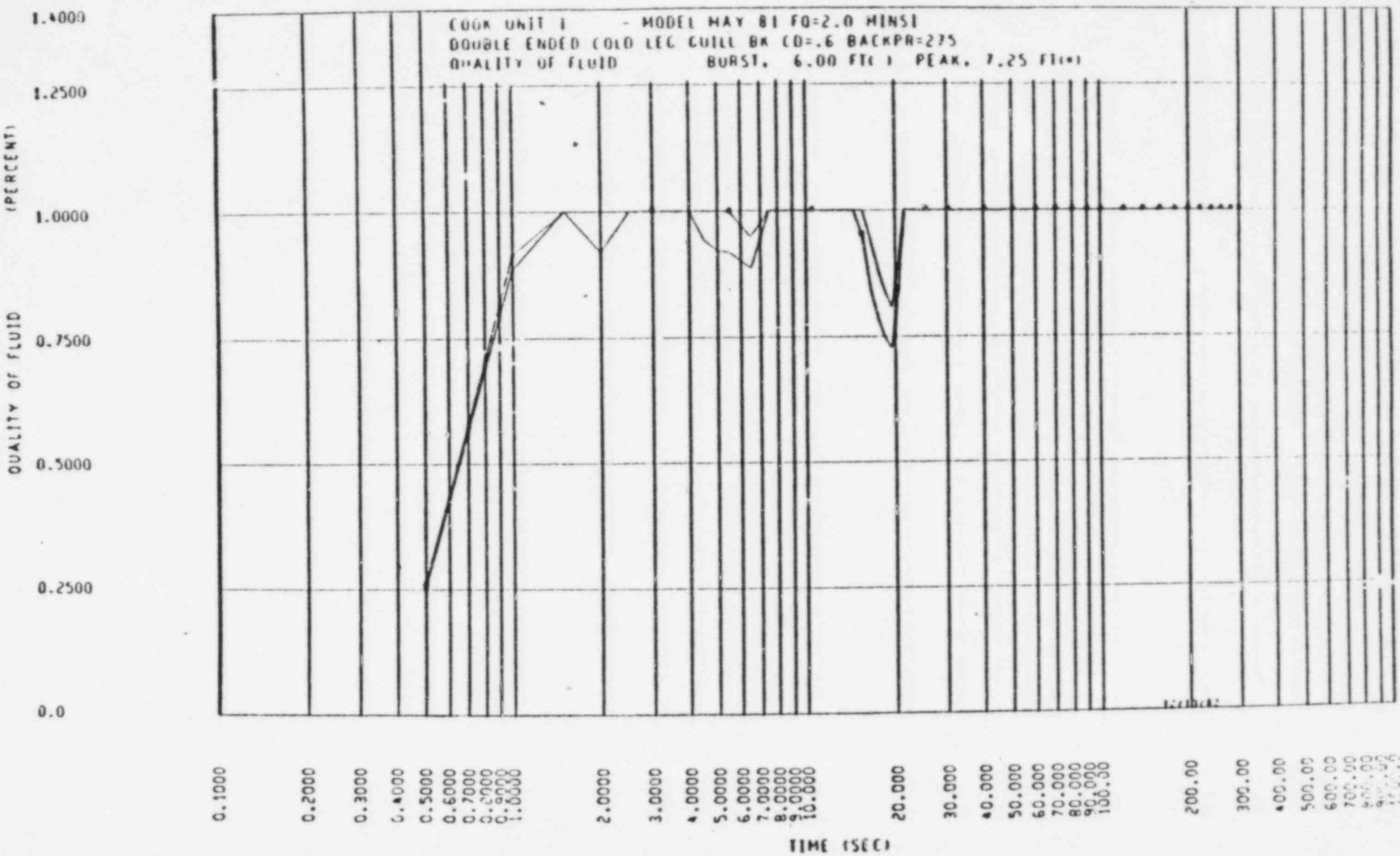


FIGURE 14.3.1-2 FLUID QUALITY DECLG(CD - 0.6)

MIN
SI

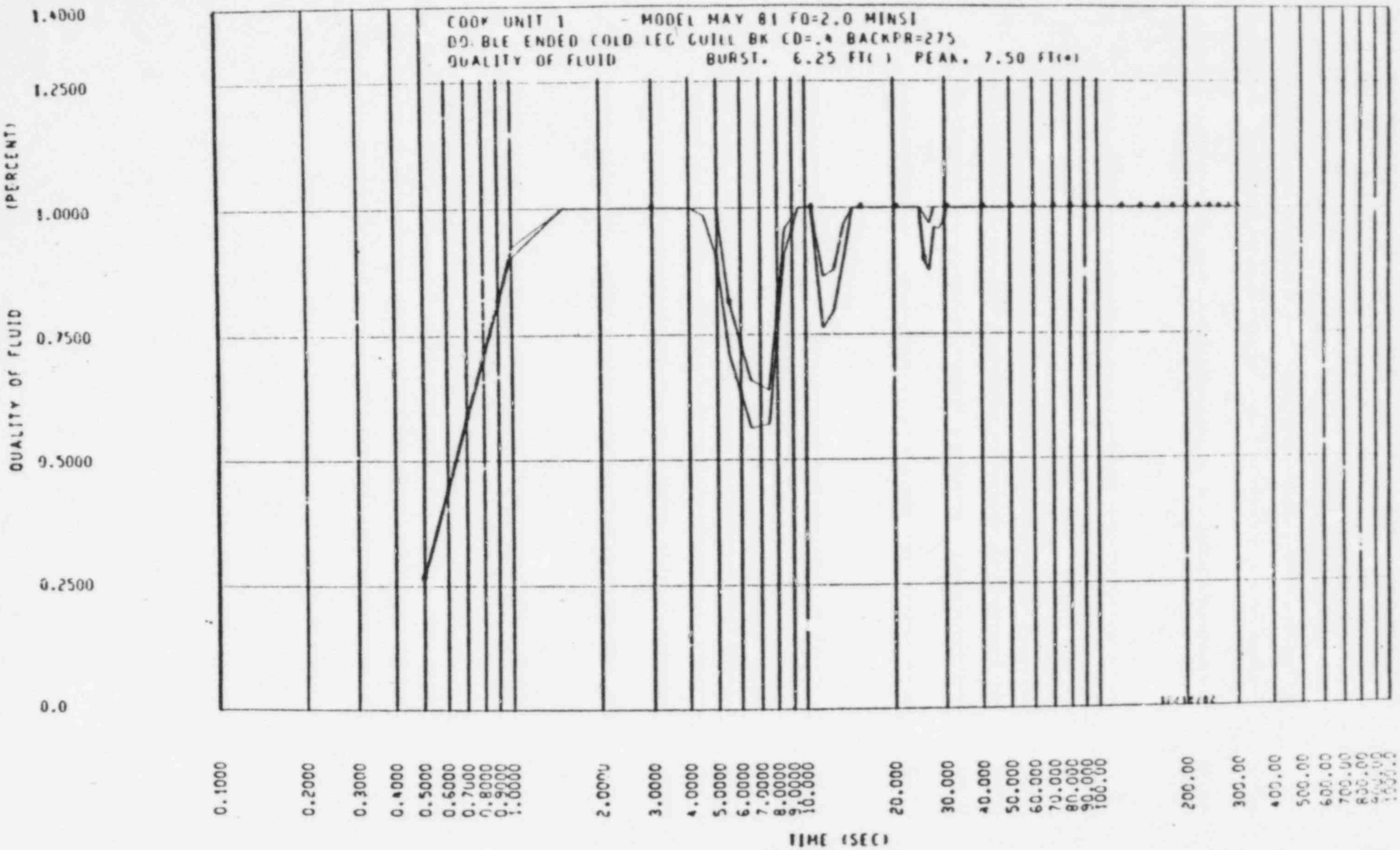


FIGURE 14.3.1-3 FLUID QUALITY MIN
DECLG(CD = 0.4) SI

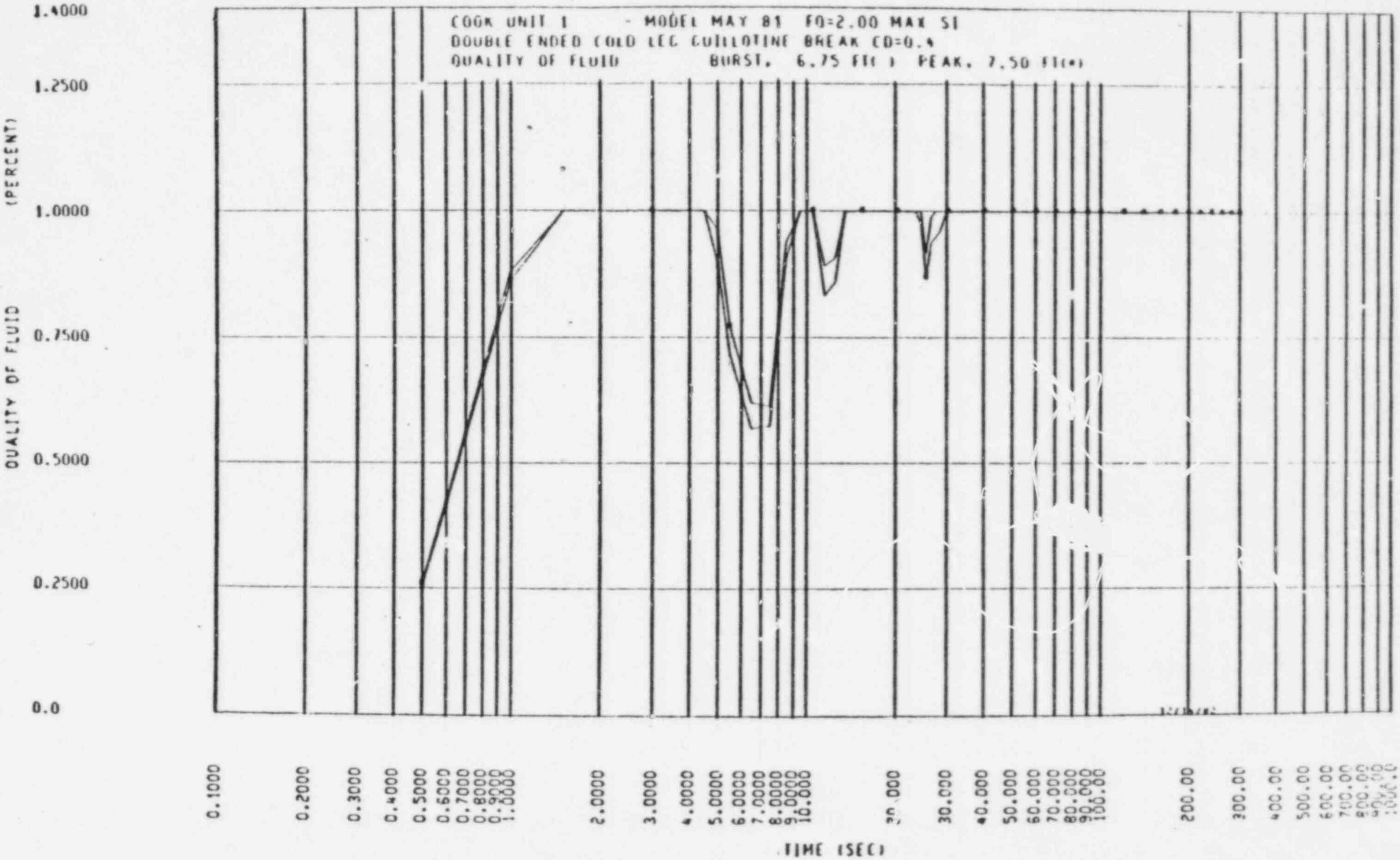


FIGURE 14.3.1-4 FLUID QUALITY MAX
DECLG(CD = 0.4) SI

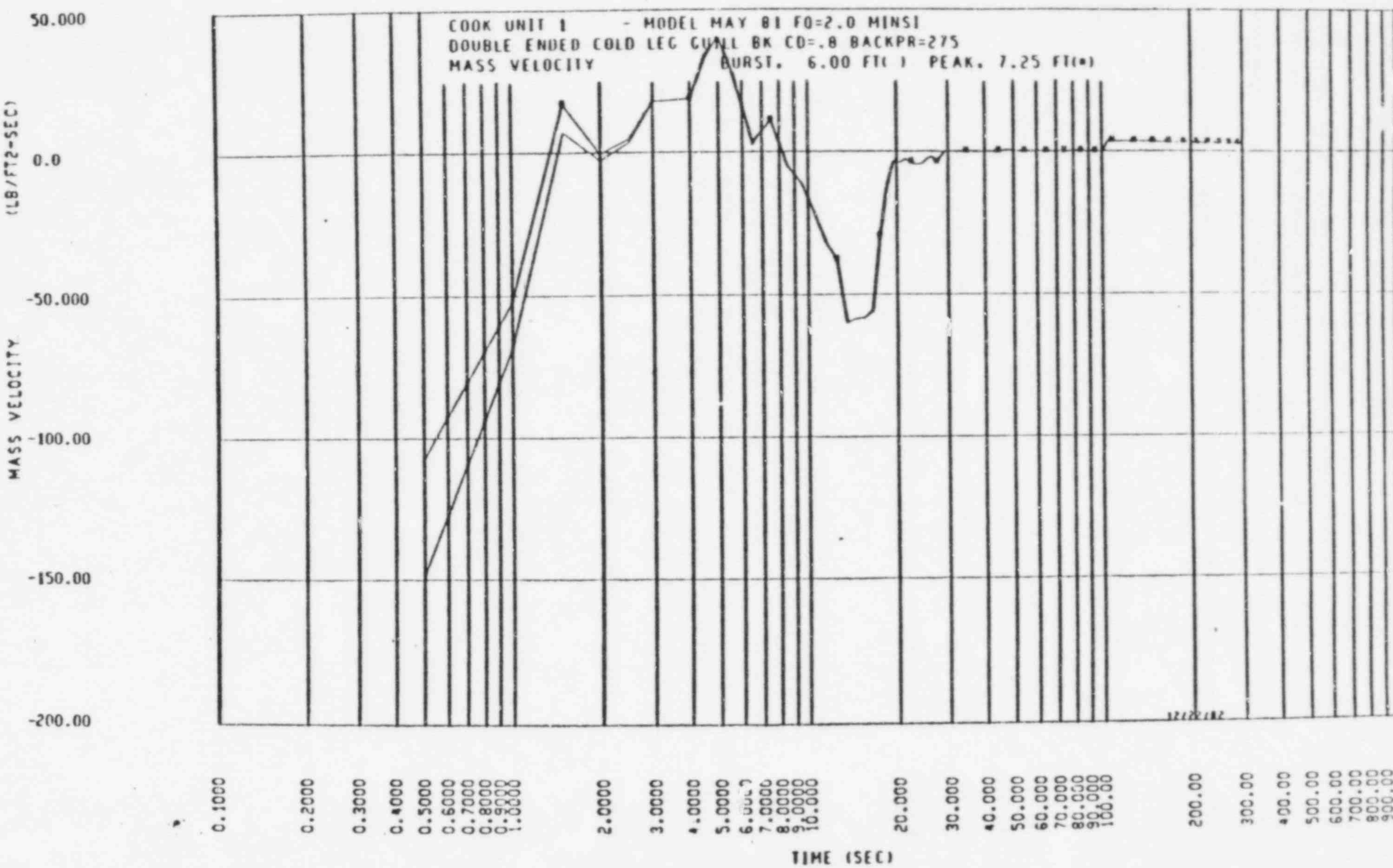


FIGURE 14.3.1-5 MASS VELOCITY
DECLG(CD = 0.8)

MIN
SI

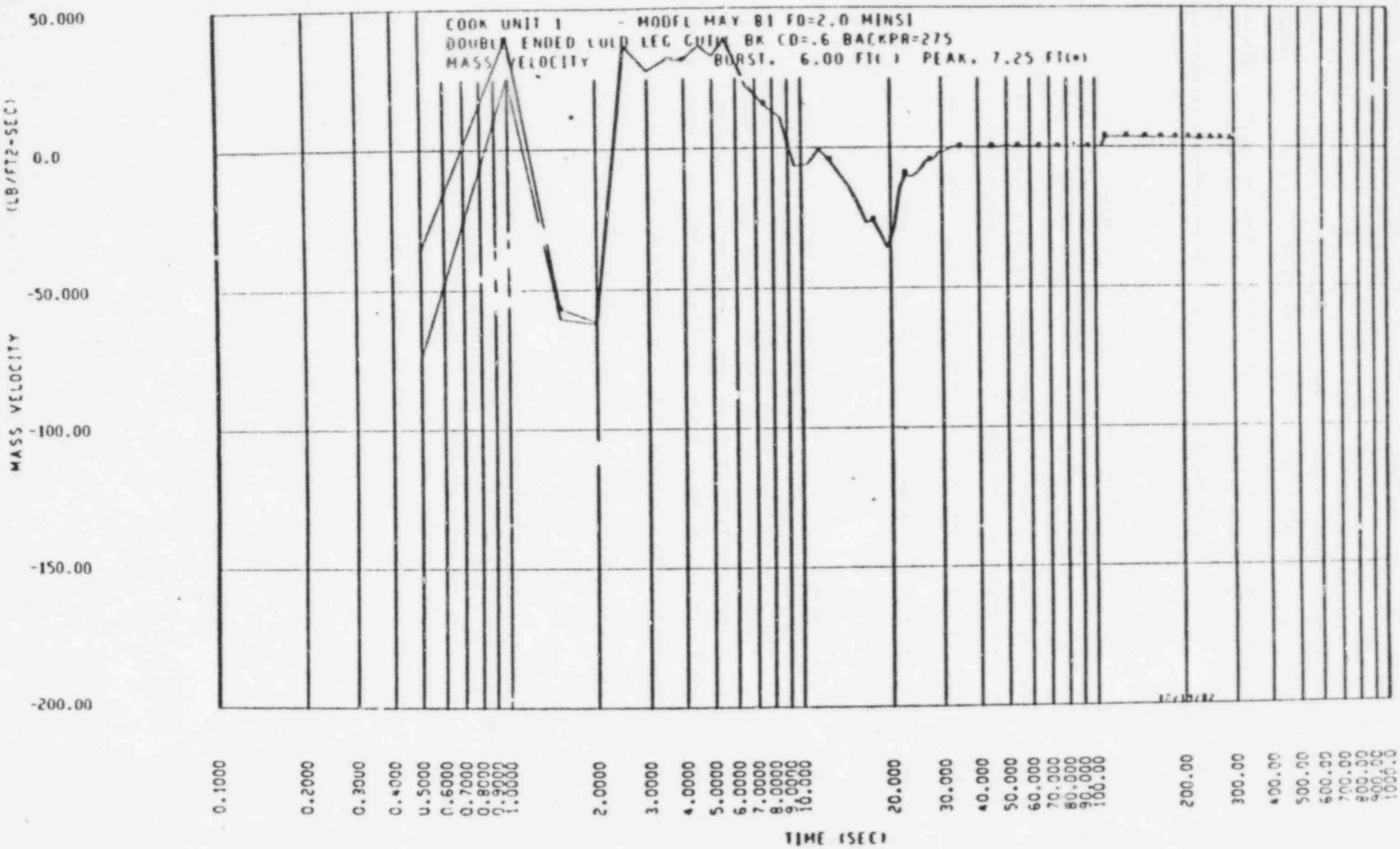


FIGURE 14.3.1-6 MASS VELOCITY DECLG(CD = 0.6) | MIN SI

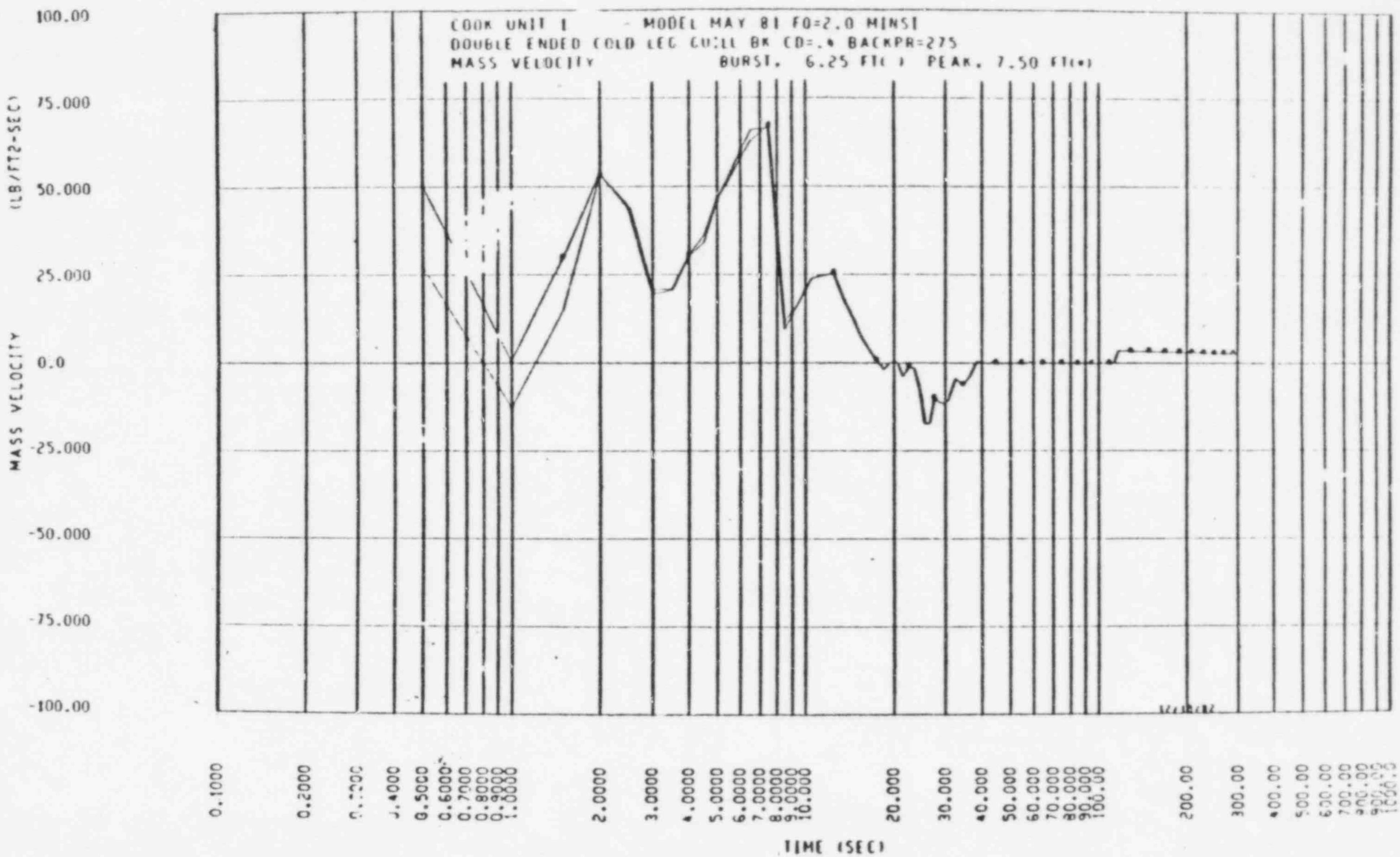


FIGURE 14.3.1-7 MASS VELOCITY MIN
DECLG(CD = 0.4) SI

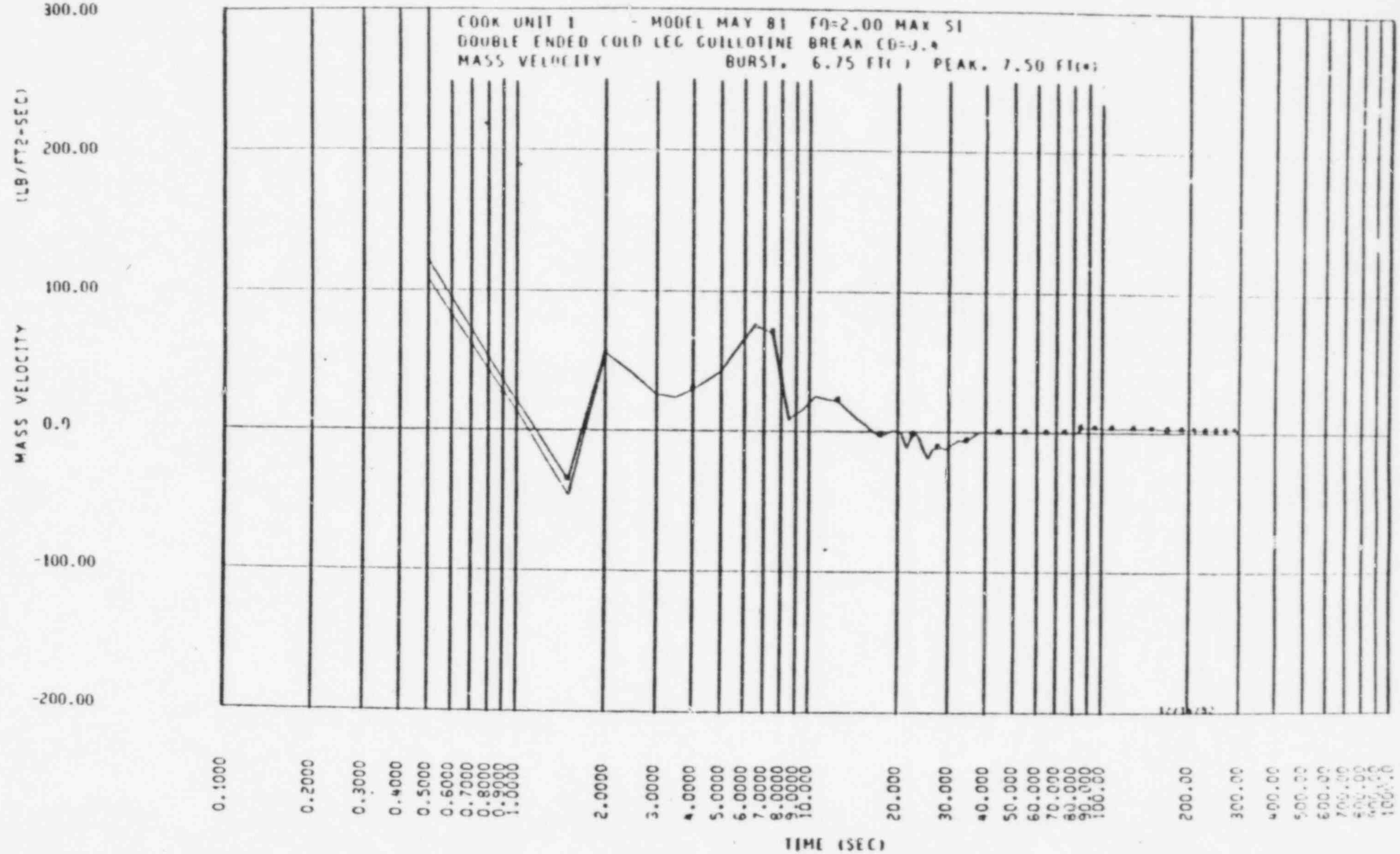


FIGURE 14.3.1-8 MASS VELOCITY MAX
DECLG(CD = 0.4) SI

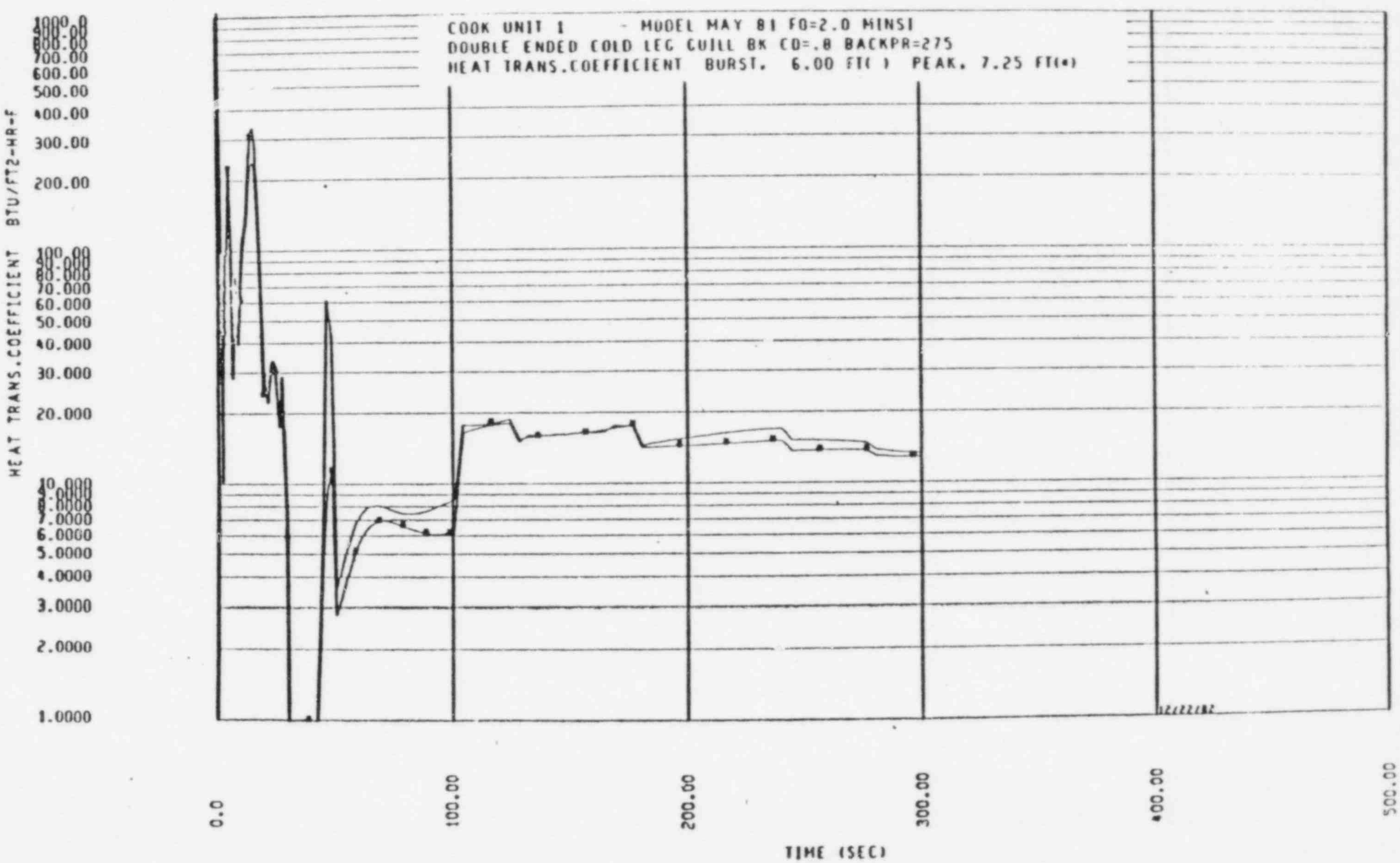


FIGURE 14.3.1-9 HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.8) MIN SI

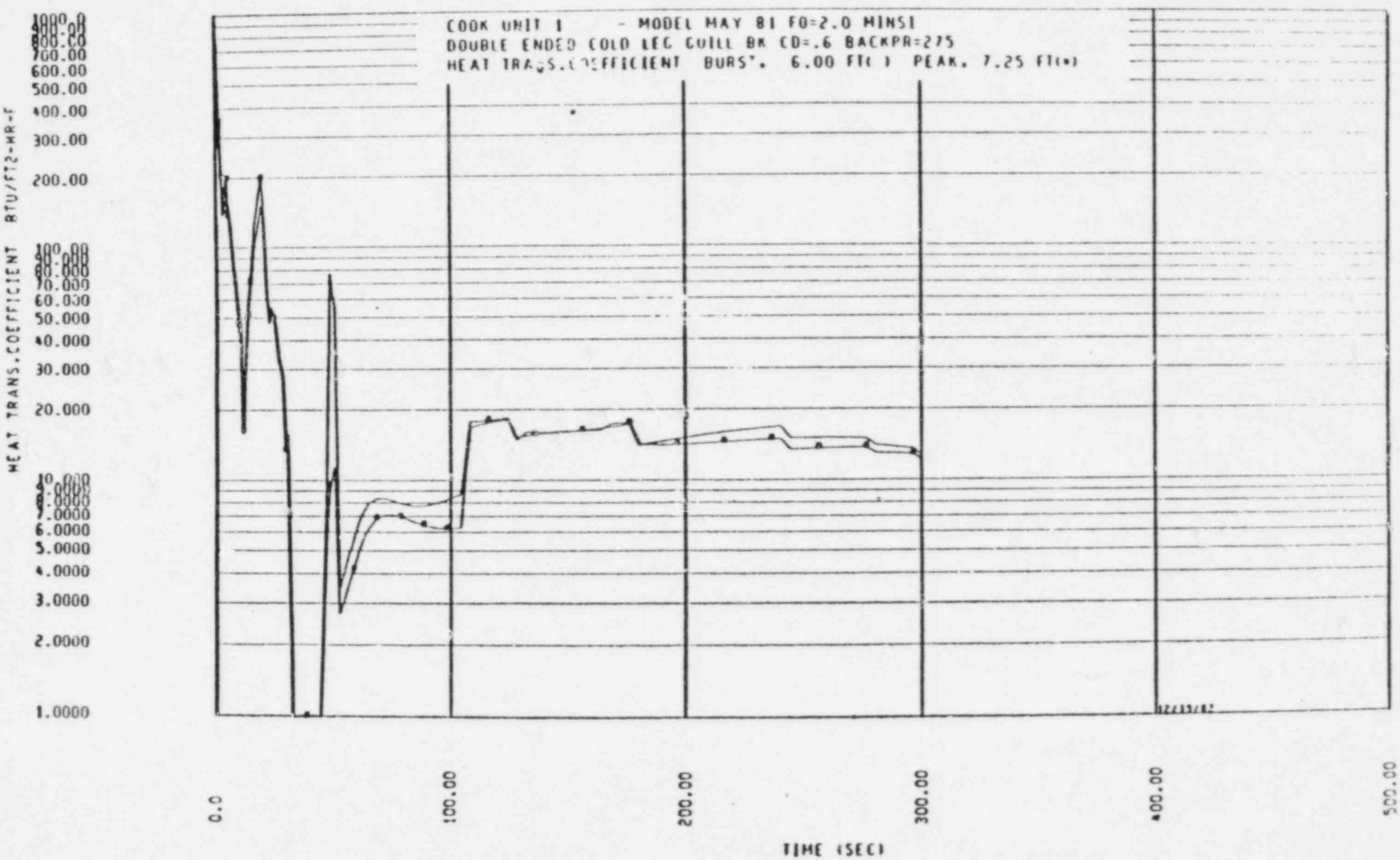


FIGURE 14.3.1-10 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.6) MIN SI

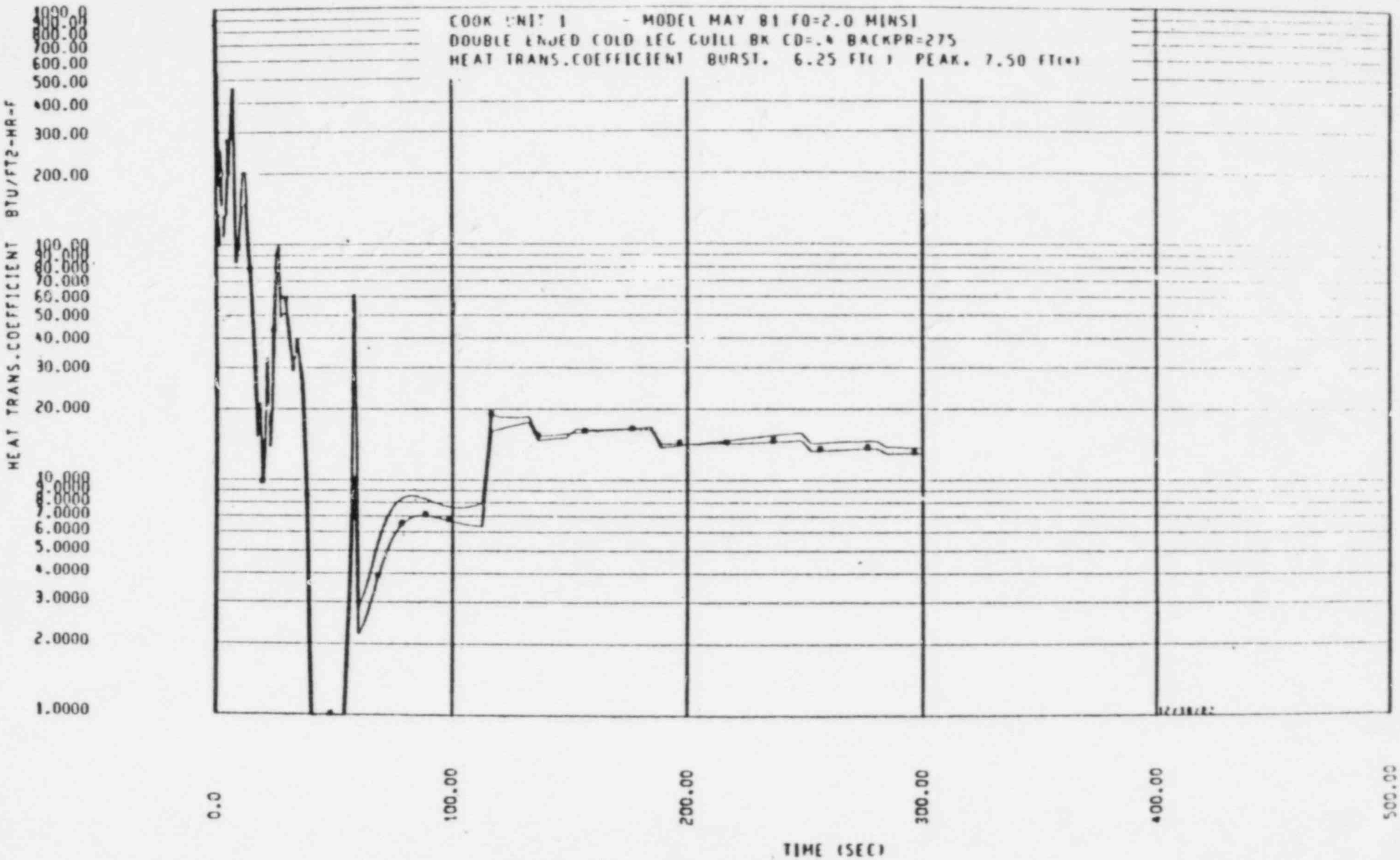


FIGURE 14.3.1-11 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.4) MIN SI

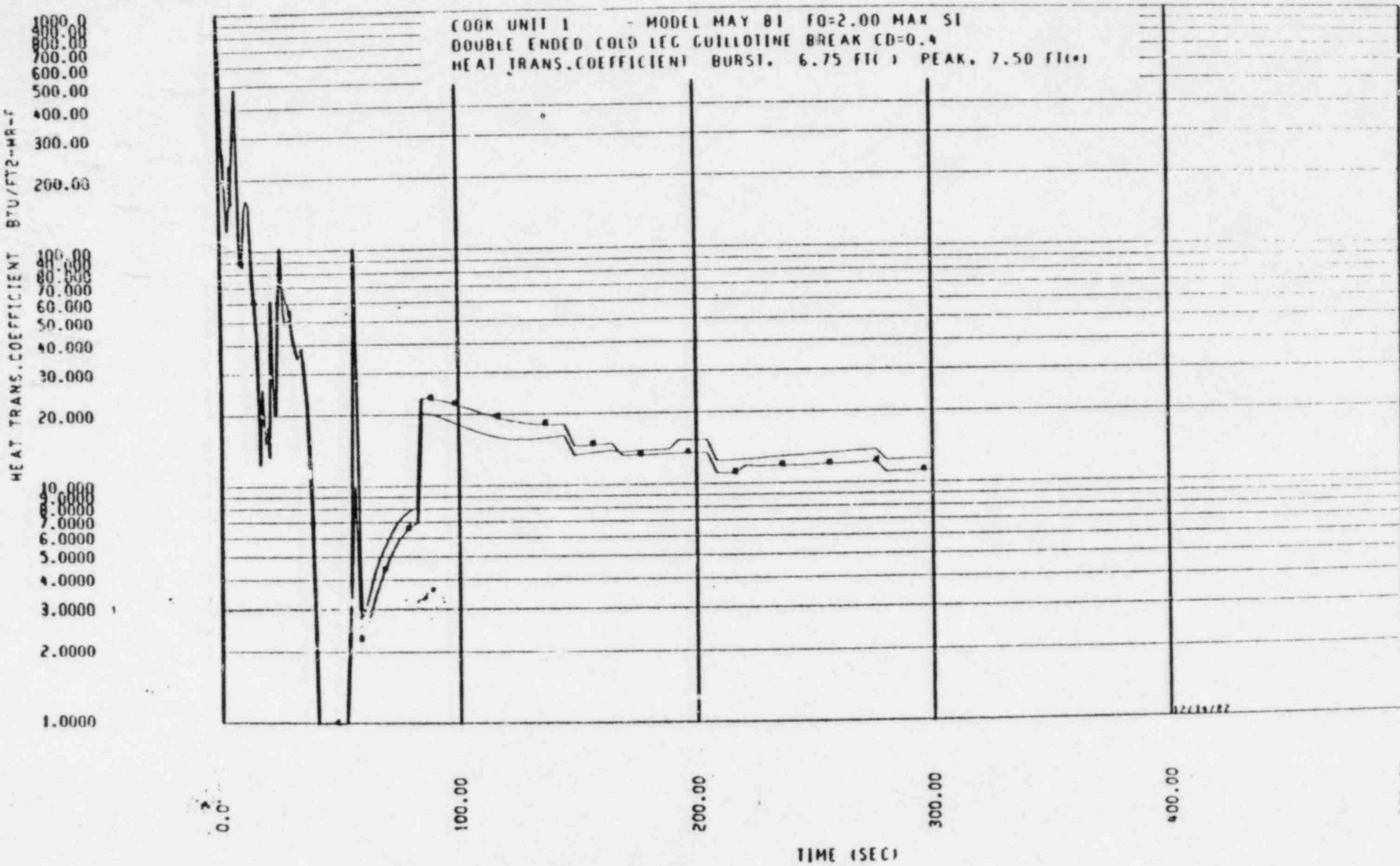


FIGURE 14.3.1-12 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.4) MAX SI

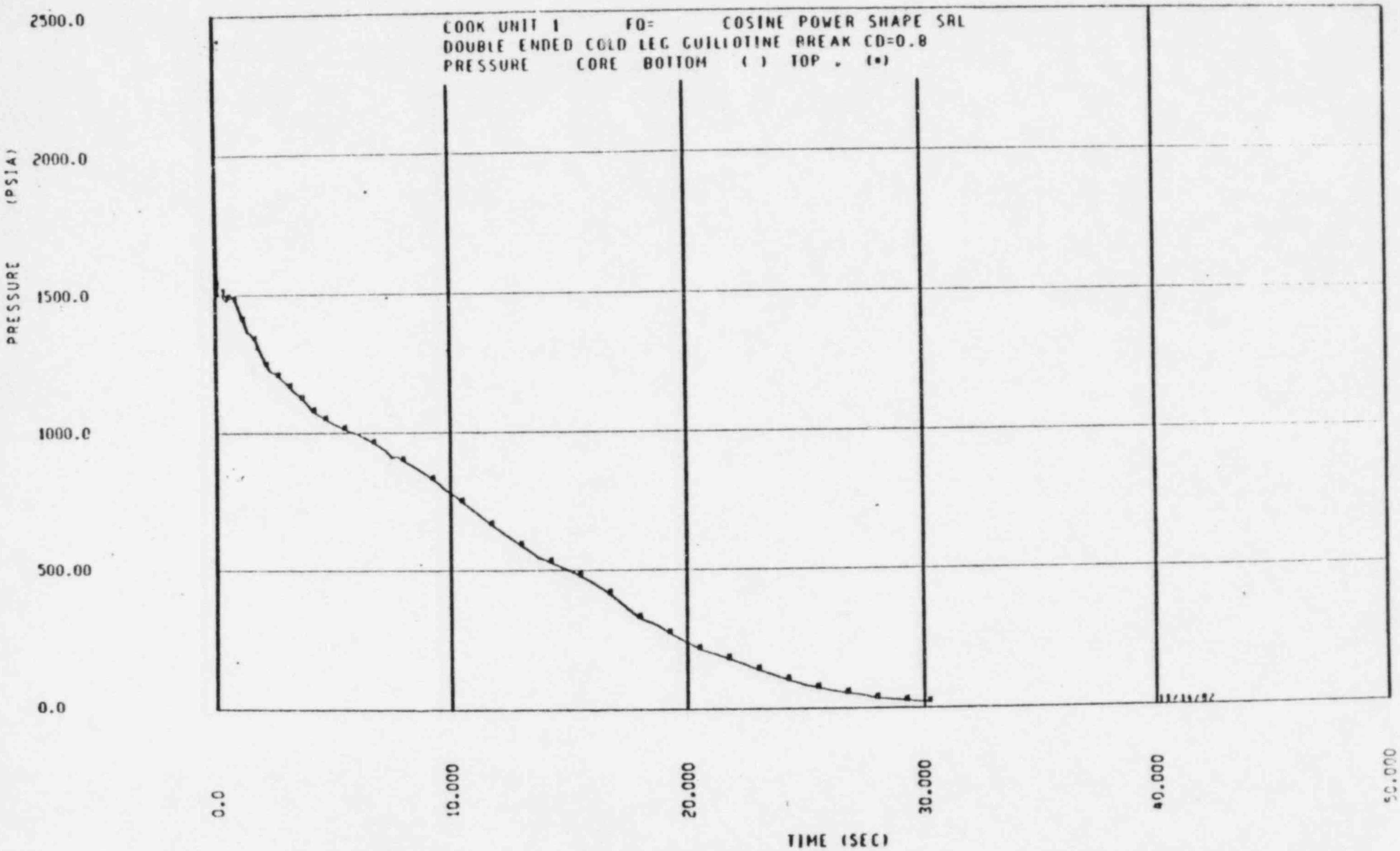


FIGURE 14.3.1-13 CORE PRESSURE DECLG(CD=0.8)

MIN SI

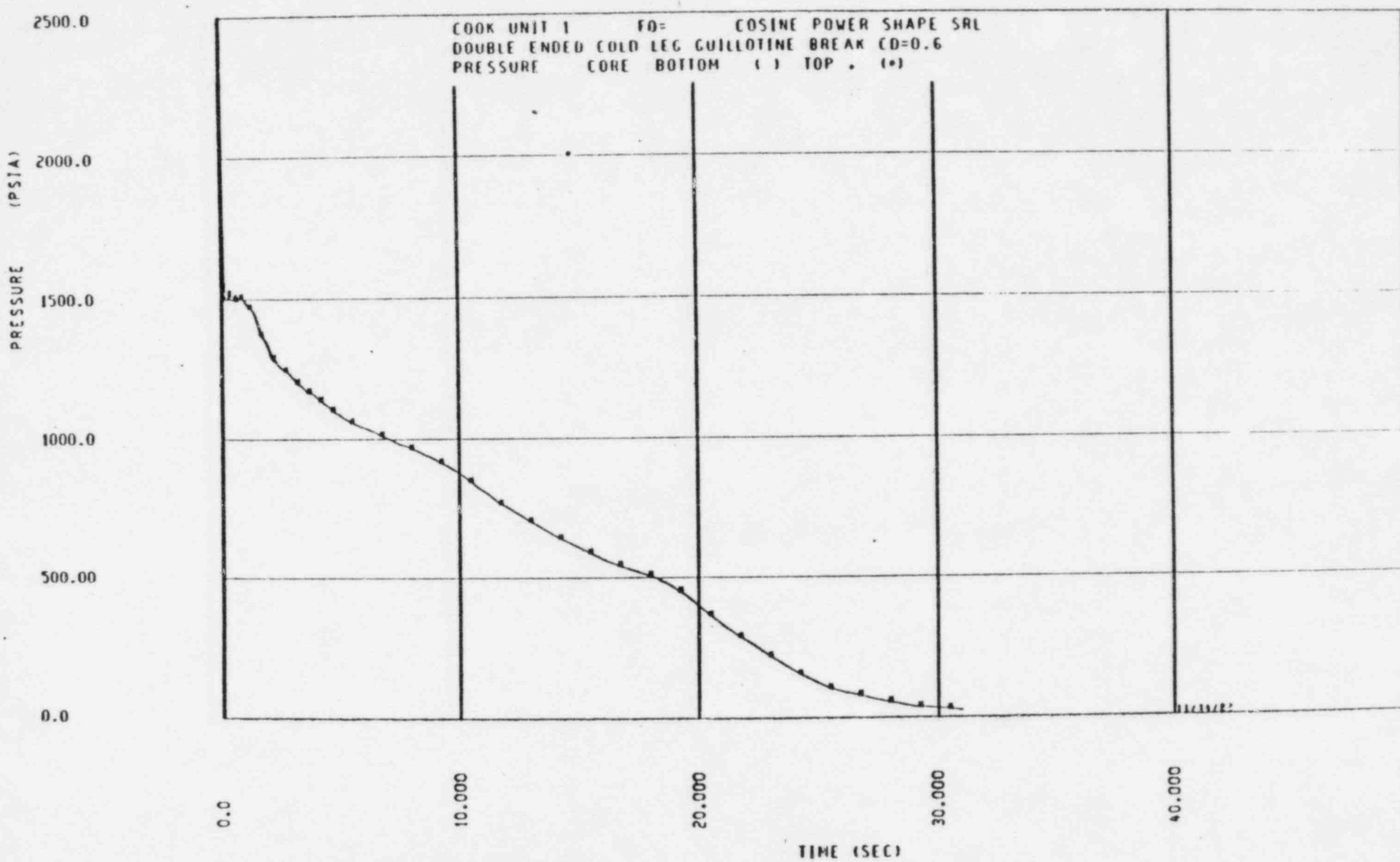


FIGURE 14.3.1-14 CORE PRESSURE DECLG(CD=0.6)

MIN
SI

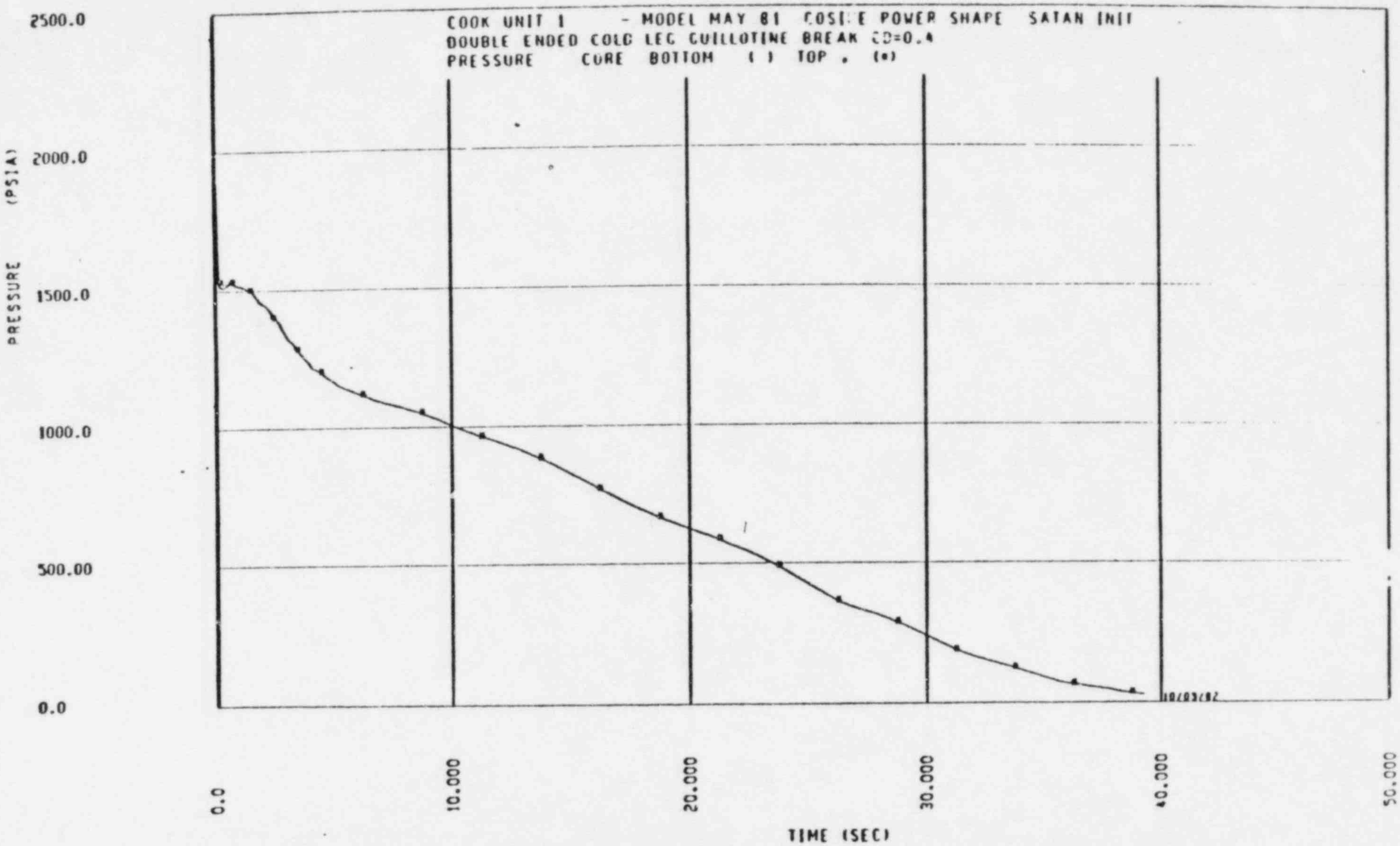


FIGURE 14.3.1-15 CORE PRESSURE DECLG(CD=0.4) MIN SI

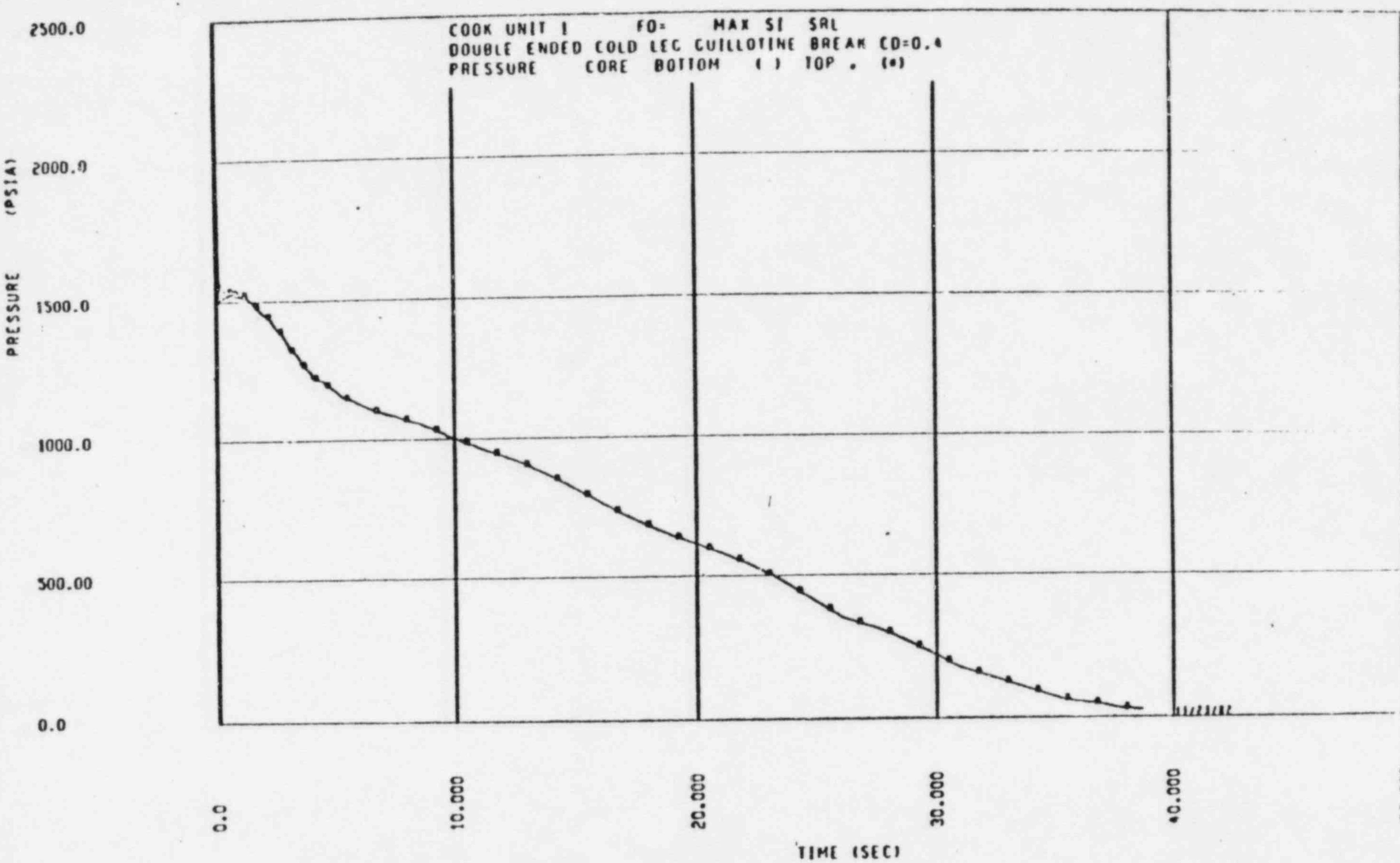


FIGURE 14.3.1-16 CORE PRESSURE DECLG(CD = 0.4) MAX SI

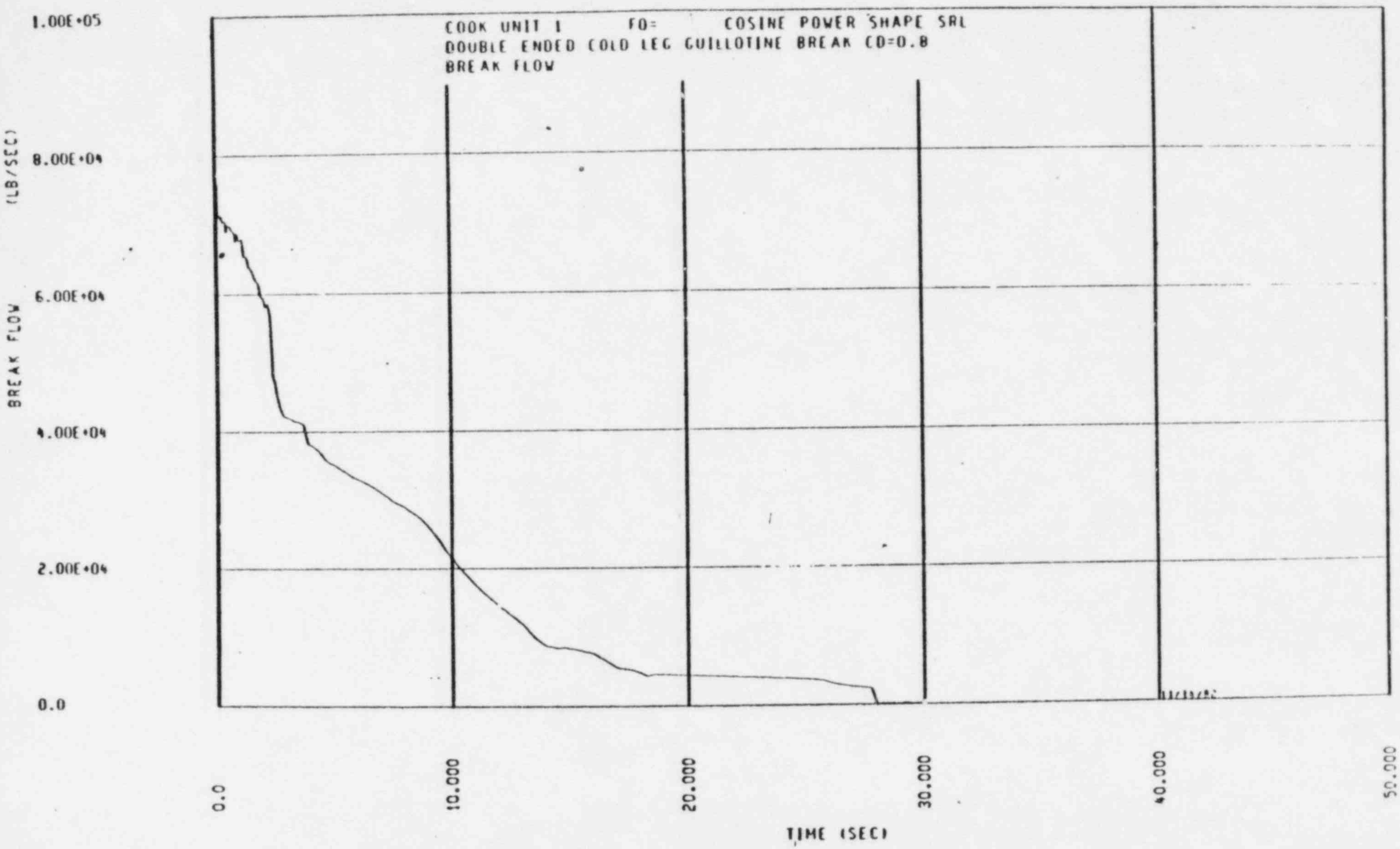


FIGURE 14.3.1-17 BREAK FLOW RATE MIN SI
DECLG (CD=0.8)

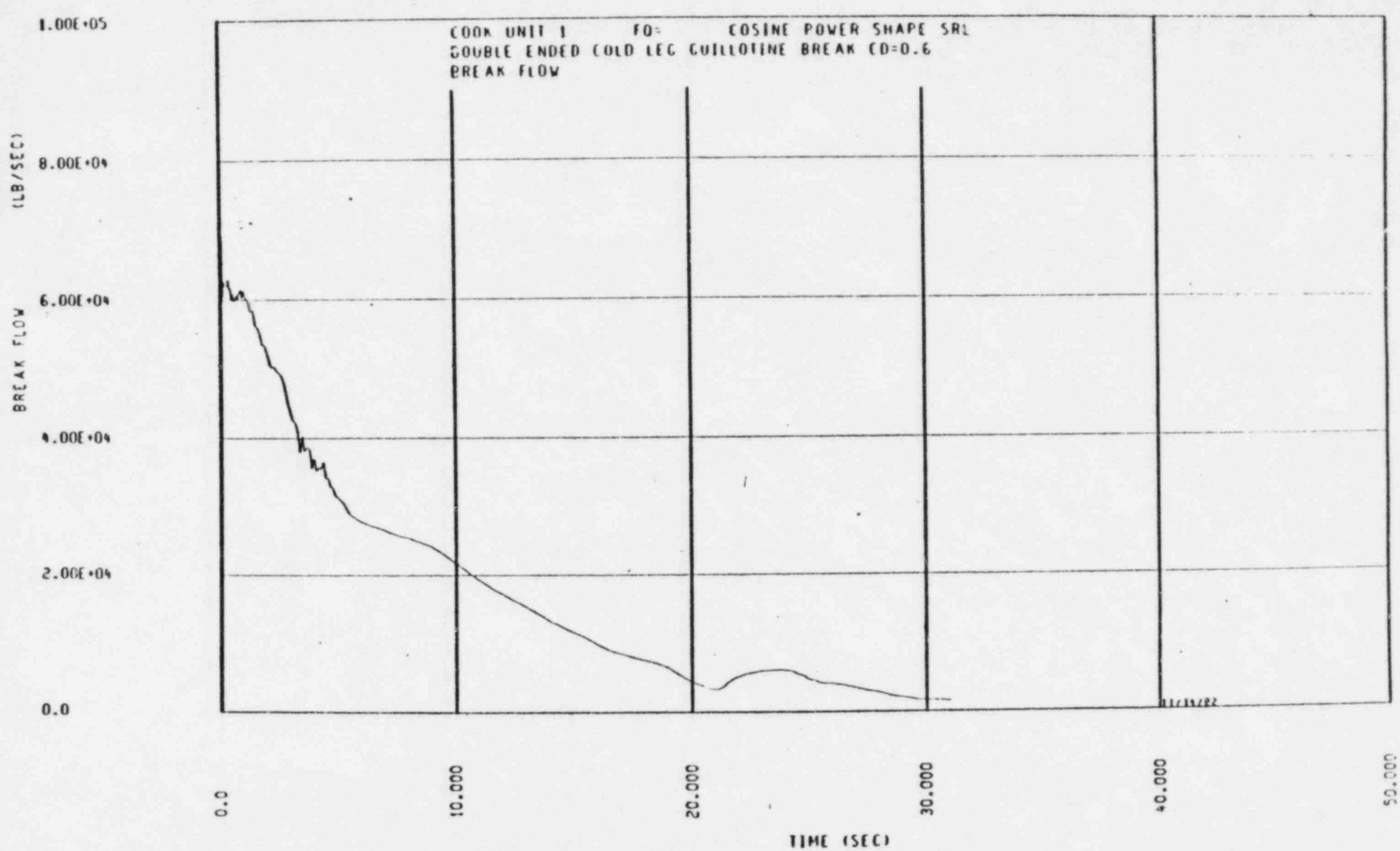


FIGURE 14.3.1-18 BREAK FLOW RATE DECLG(CD=0.6) | MIN SI

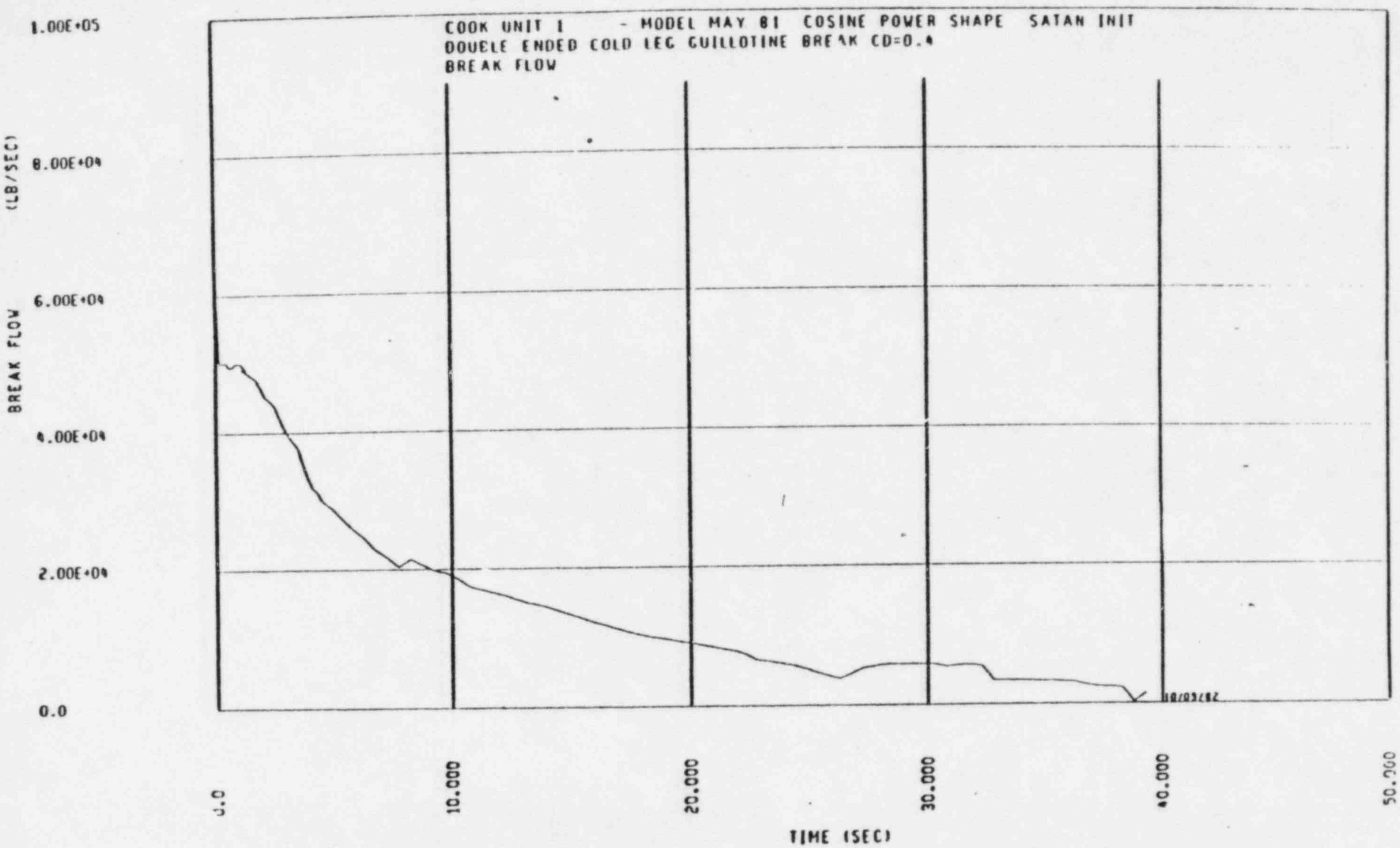


FIGURE 14.3.1-19 BREAK FLOW RATE MIN
DECLG(CD=0.4) SI

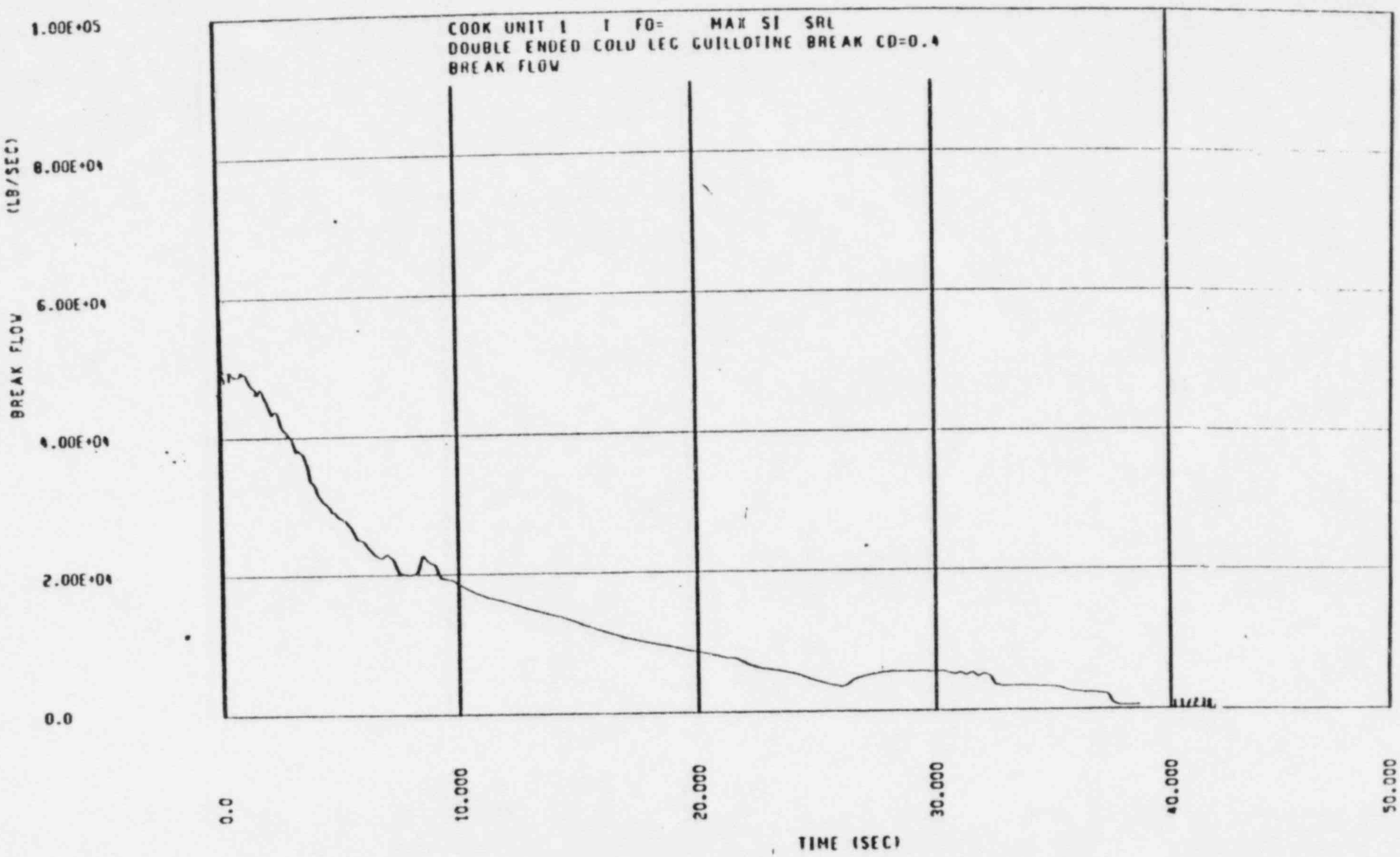


FIGURE 14.3.1-20 BREAK FLOW RATE MAX SI
DECLG(CD=0.4)

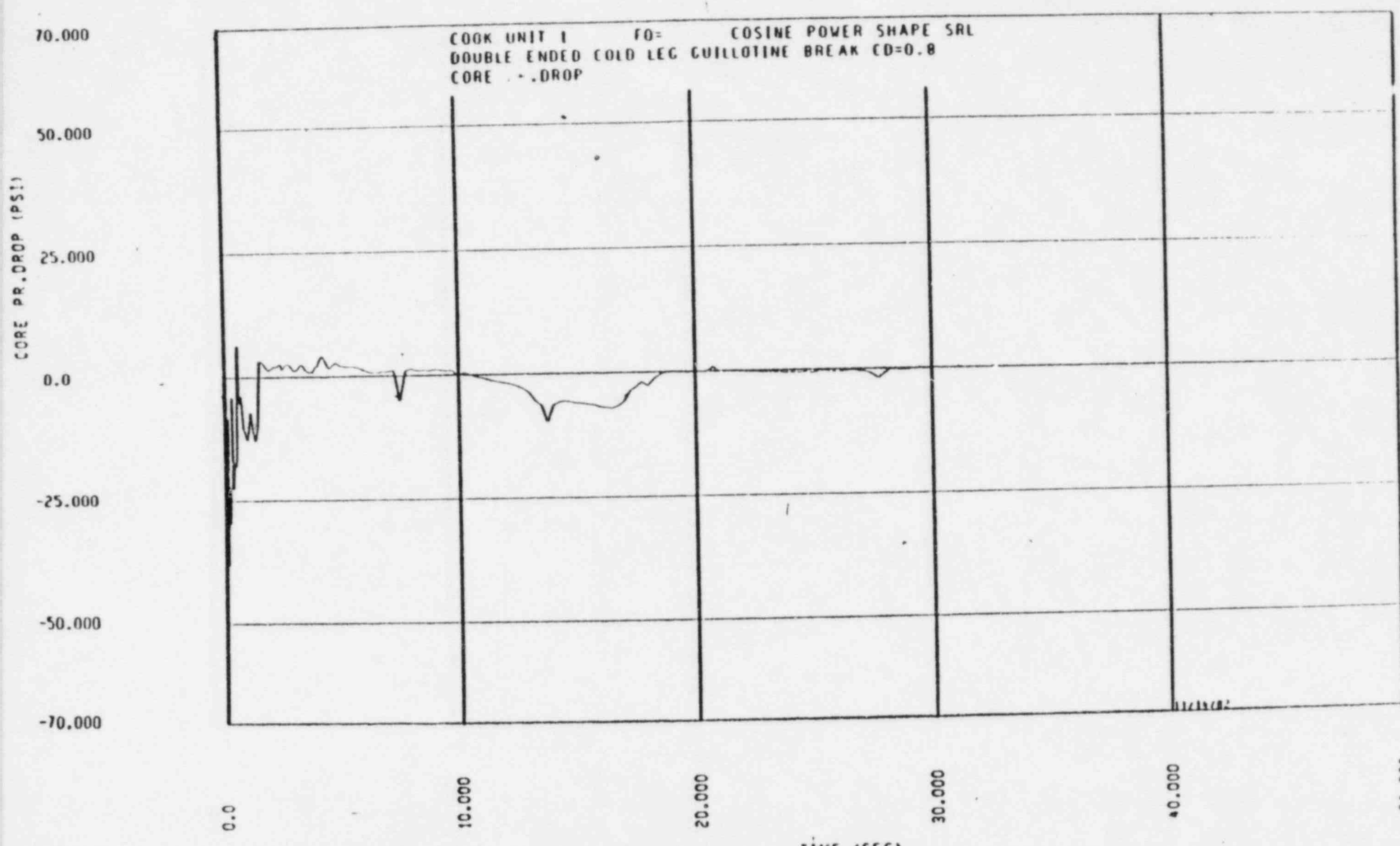


FIGURE 14.3.1-21 CORE PRESSURE DROP DECLG(CD = 0.8) MIN SI

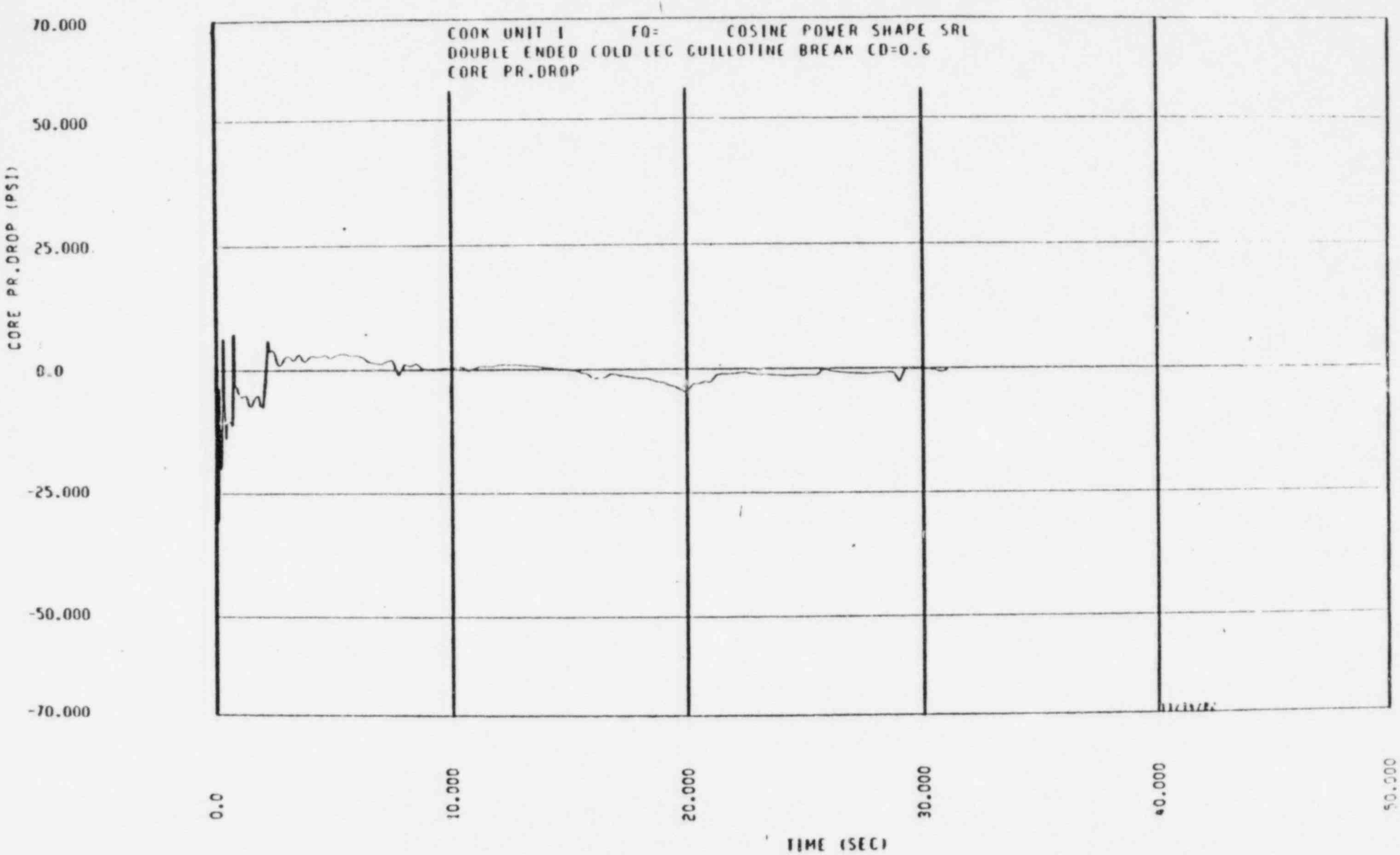


FIGURE 14.3.1-22 CORE PRESSURE DROP DECLG(CD=0.6) MIN SI

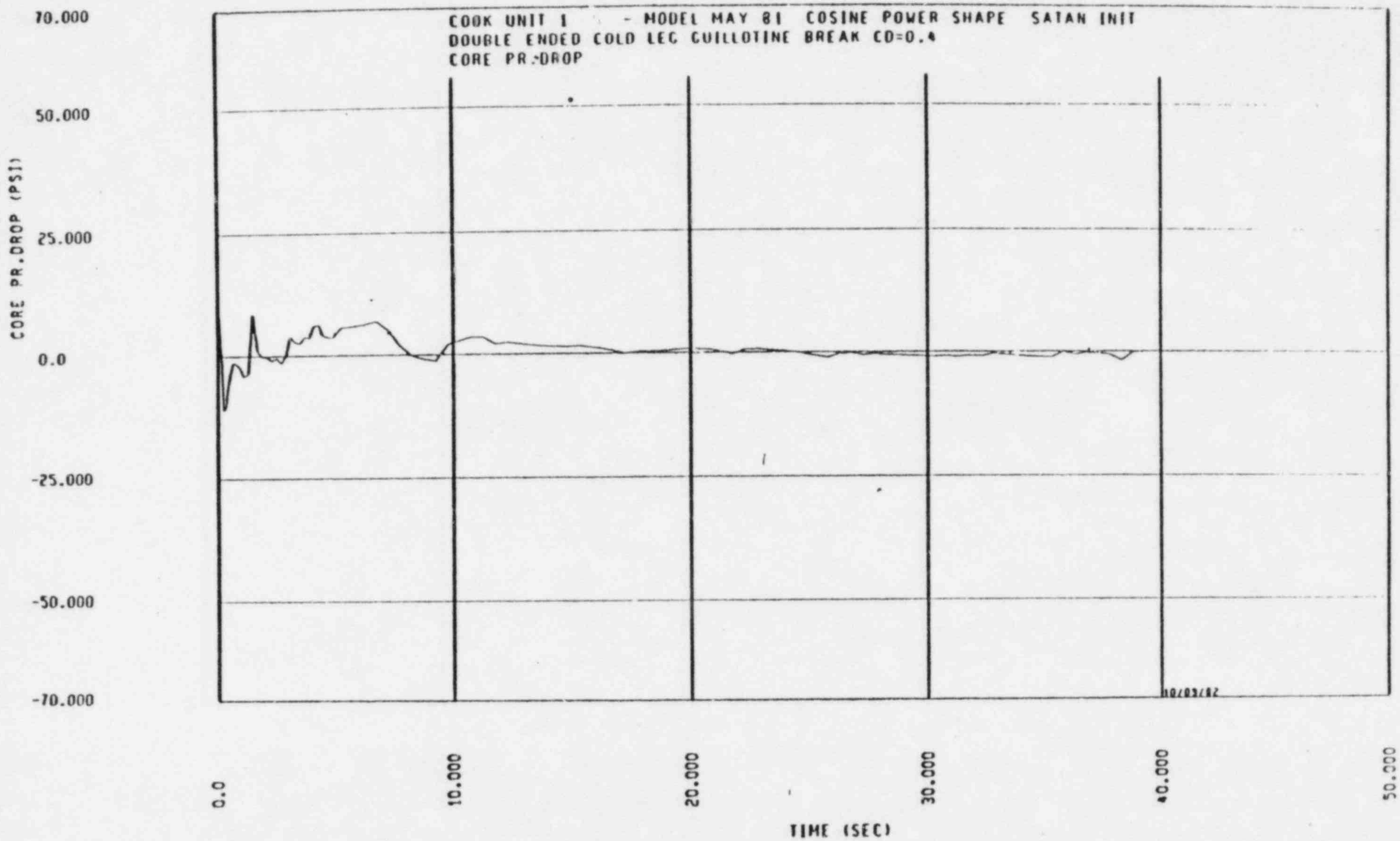


FIGURE 14.3.1-23 CORE PRESSURE DROP DECLG(CD = 0.4)

MIN
SI

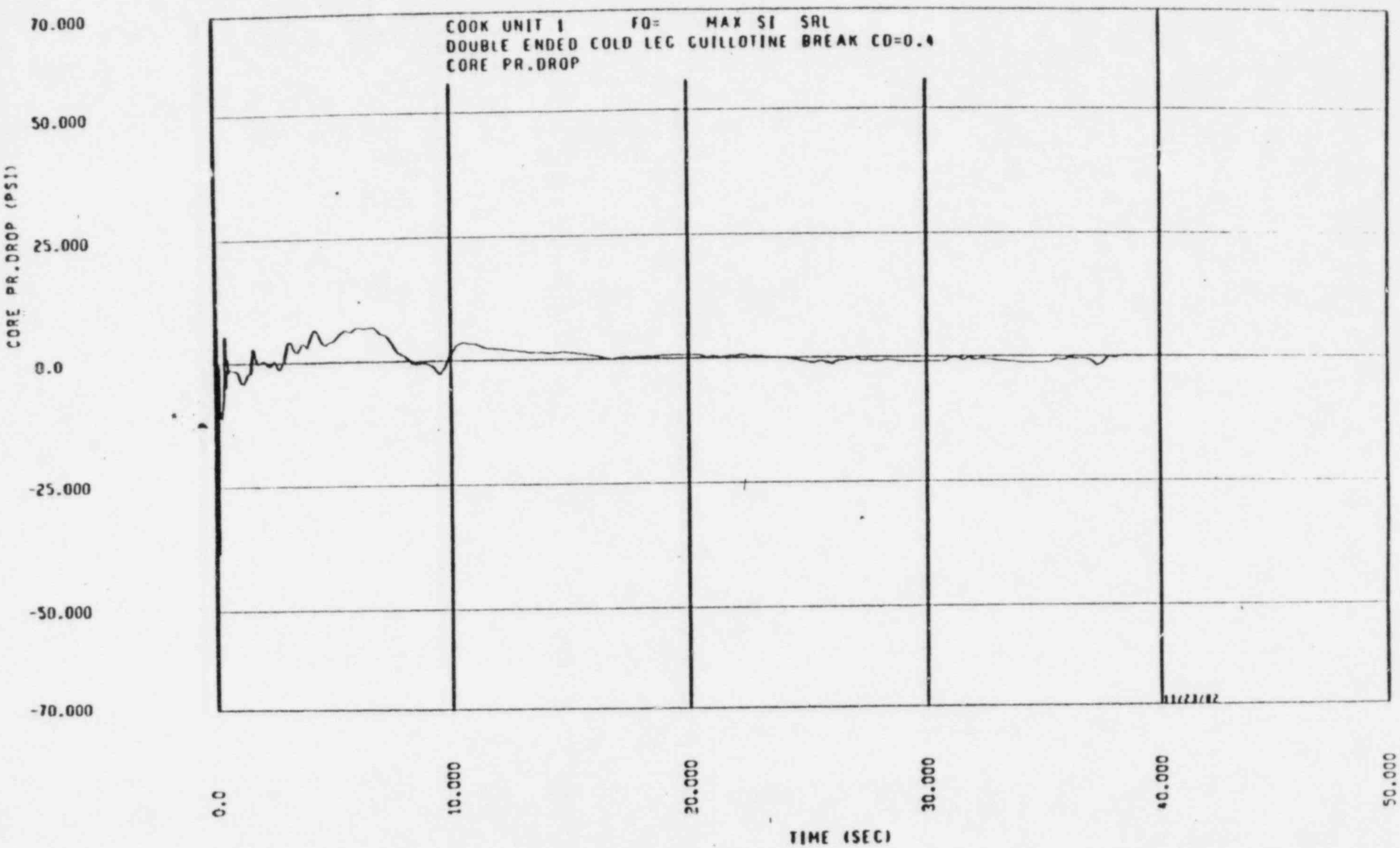


FIGURE 14.3.1-24 CORE PRESSURE DROP
DECLG(CD = 0.4)

MAX
SI

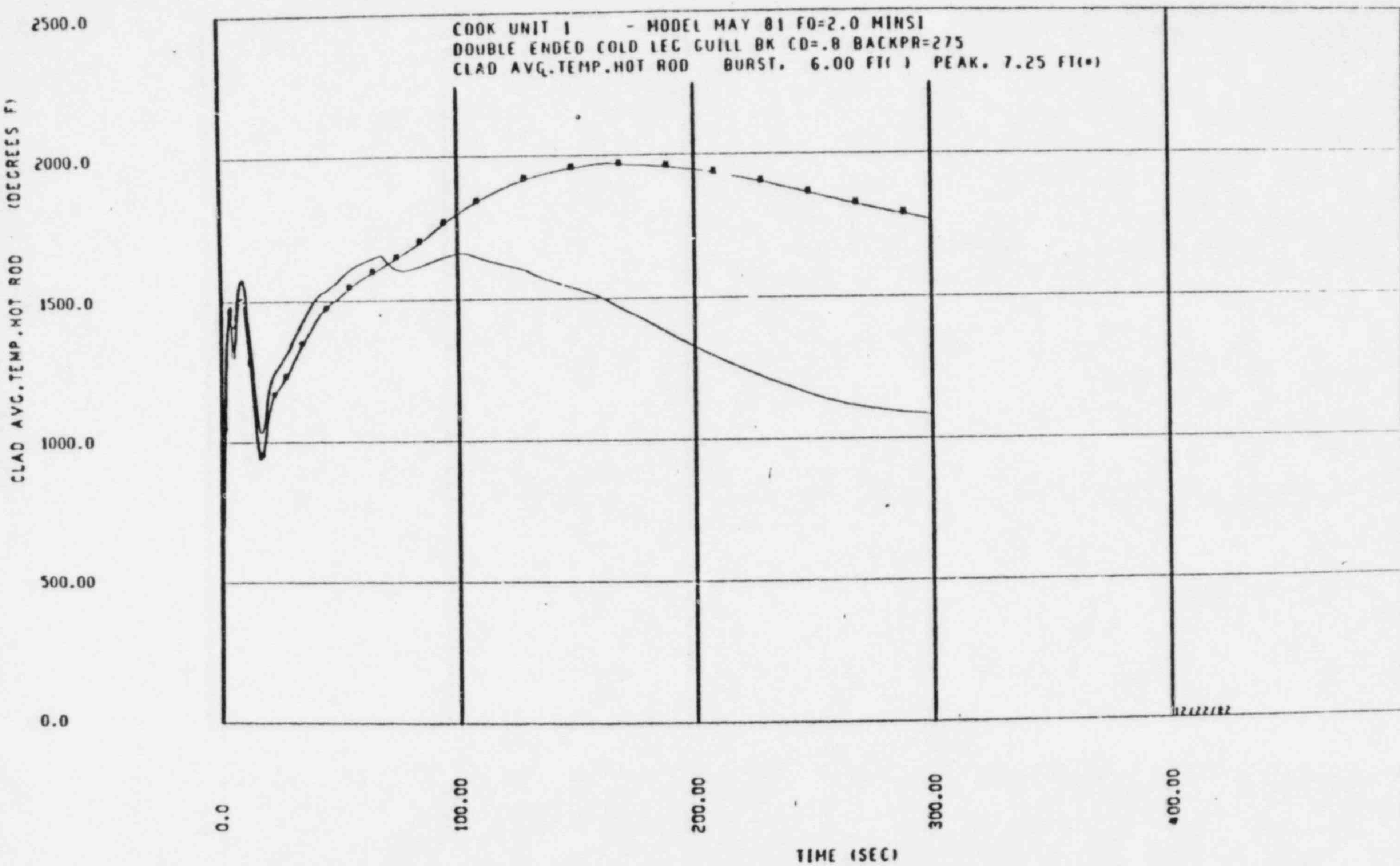


FIGURE 14.3.1-25 PEAK CLAD TEMPERATURE DECLG(CD = 0.8)

MIN SI

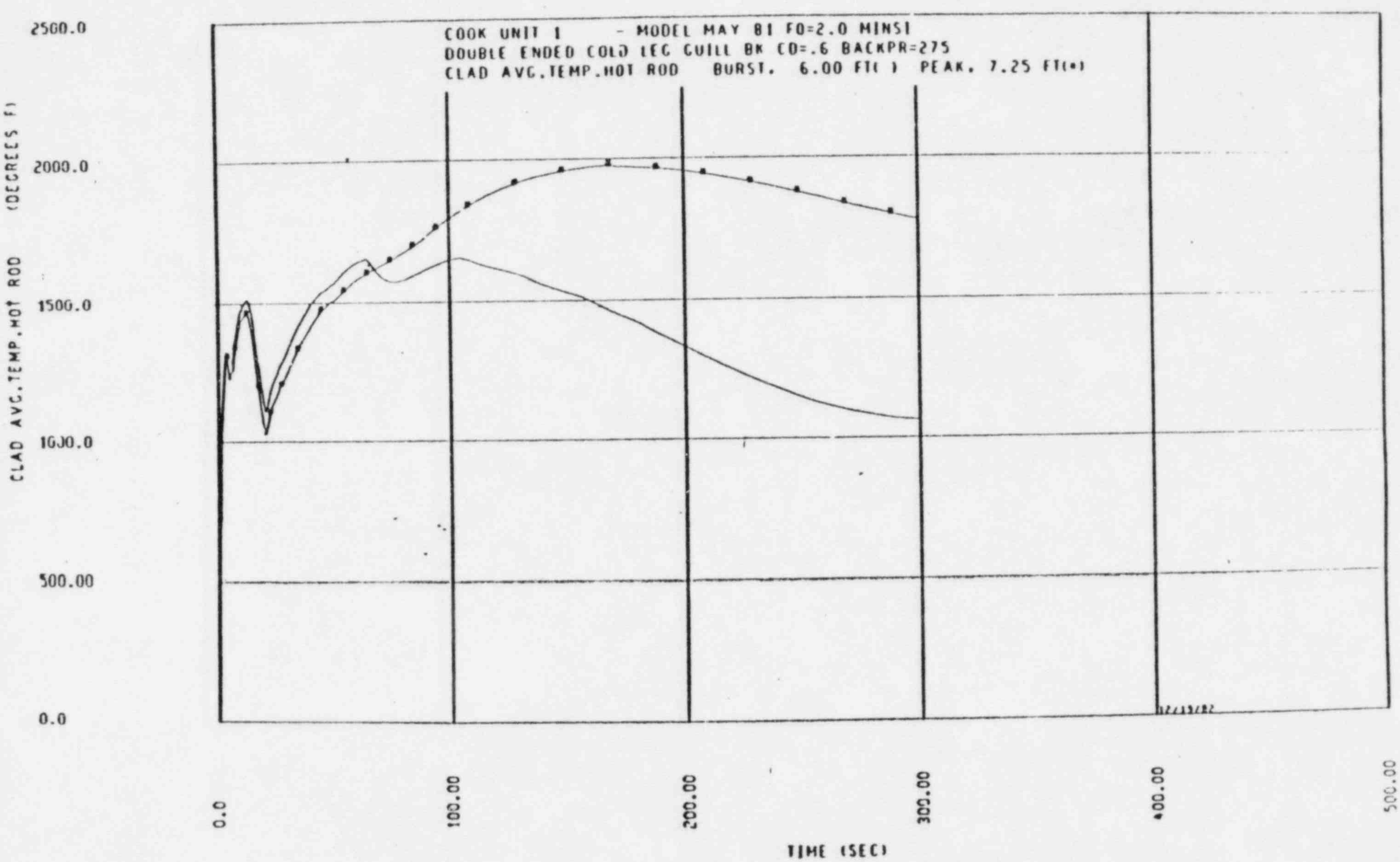


FIGURE 14.3.1-26 PEAK CLAD TEMPERATURE
DECLG(CD = 0.6)

MIN
SI

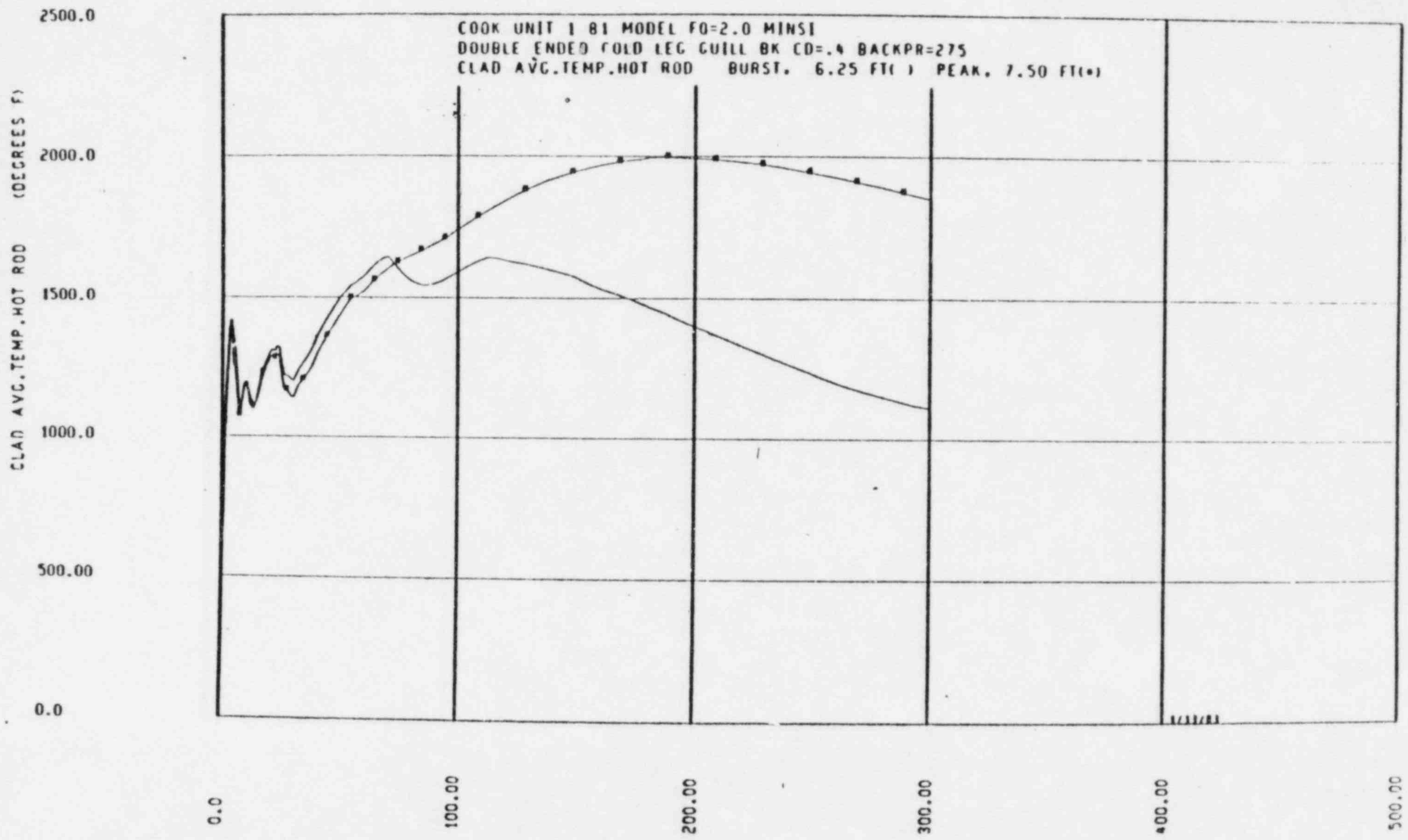


FIGURE 14.3.1-27 PEAK CLAD TEMPERATURE DECLG(CD = 0.4) MIN SI

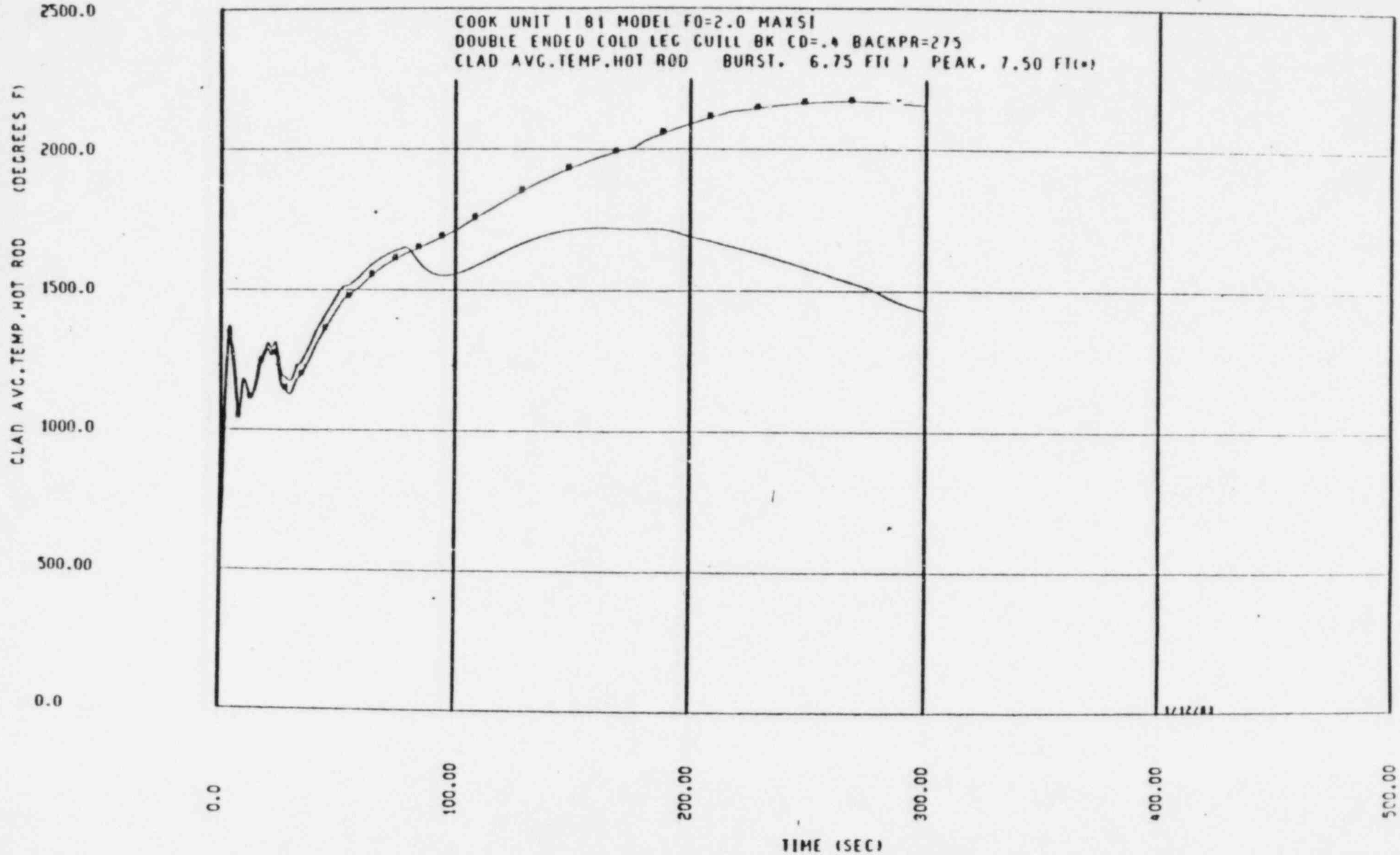


FIGURE 14.3.1-28 PEAK CLAD TEMPERAT URE MAX
DECLG(CD = 0.4) SI

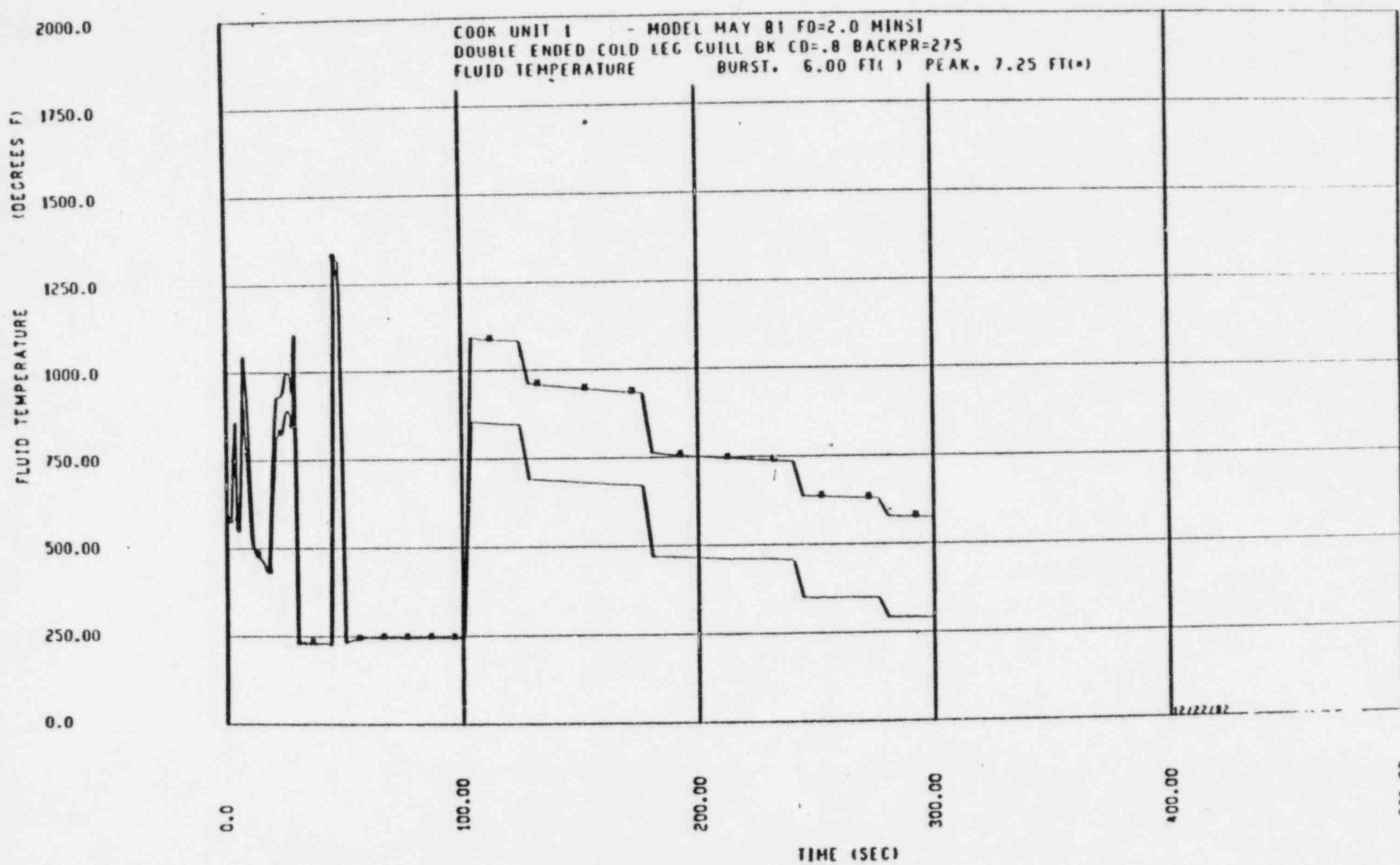


FIGURE 14.3.1-29 FLUID TEMPERATURE
DECLG(CD = 0.8)

MIN
SI

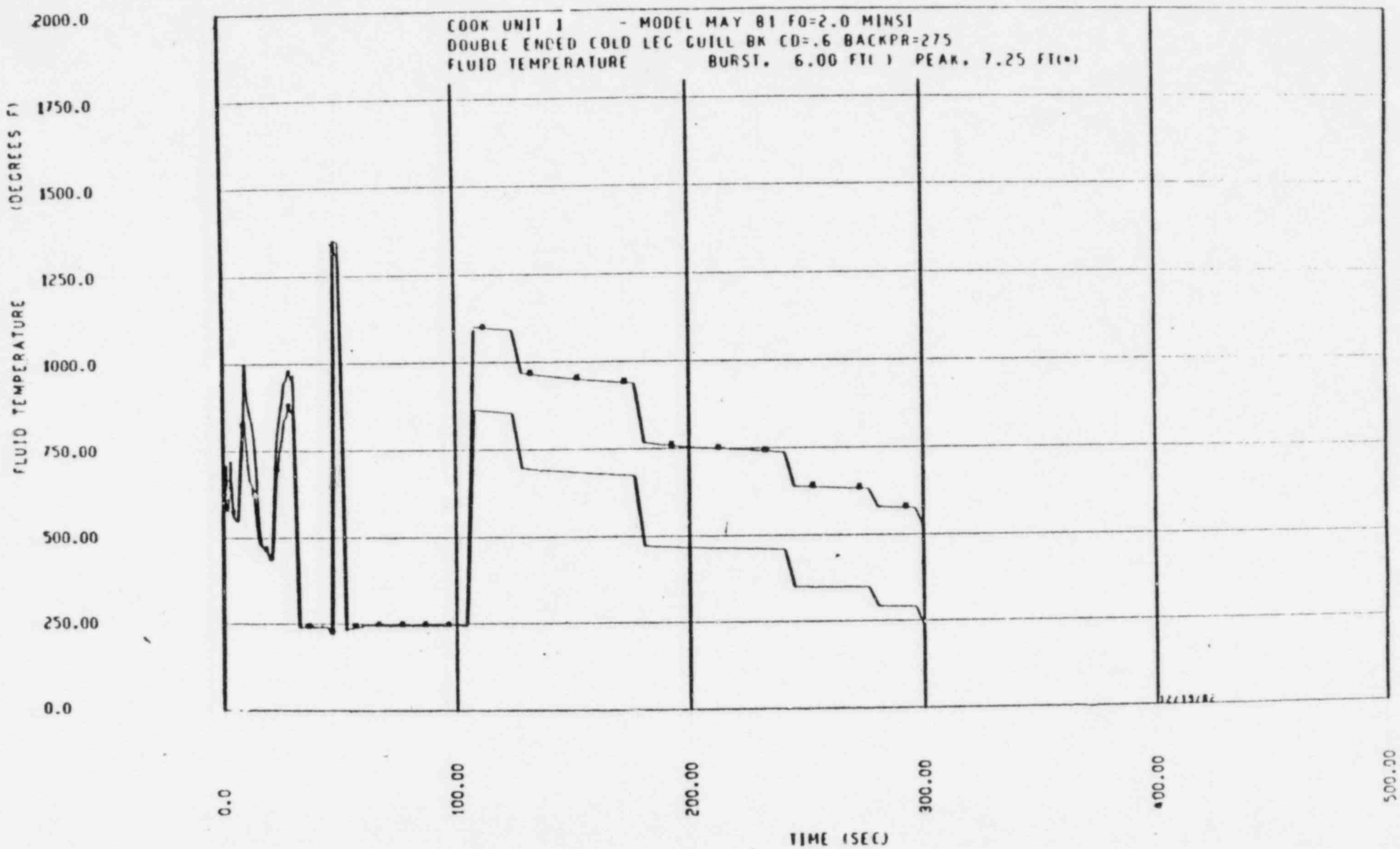


FIGURE 14.3.1-30 FLUID TEMPERATURE DECLG(CD = 0.6)

MIN SI

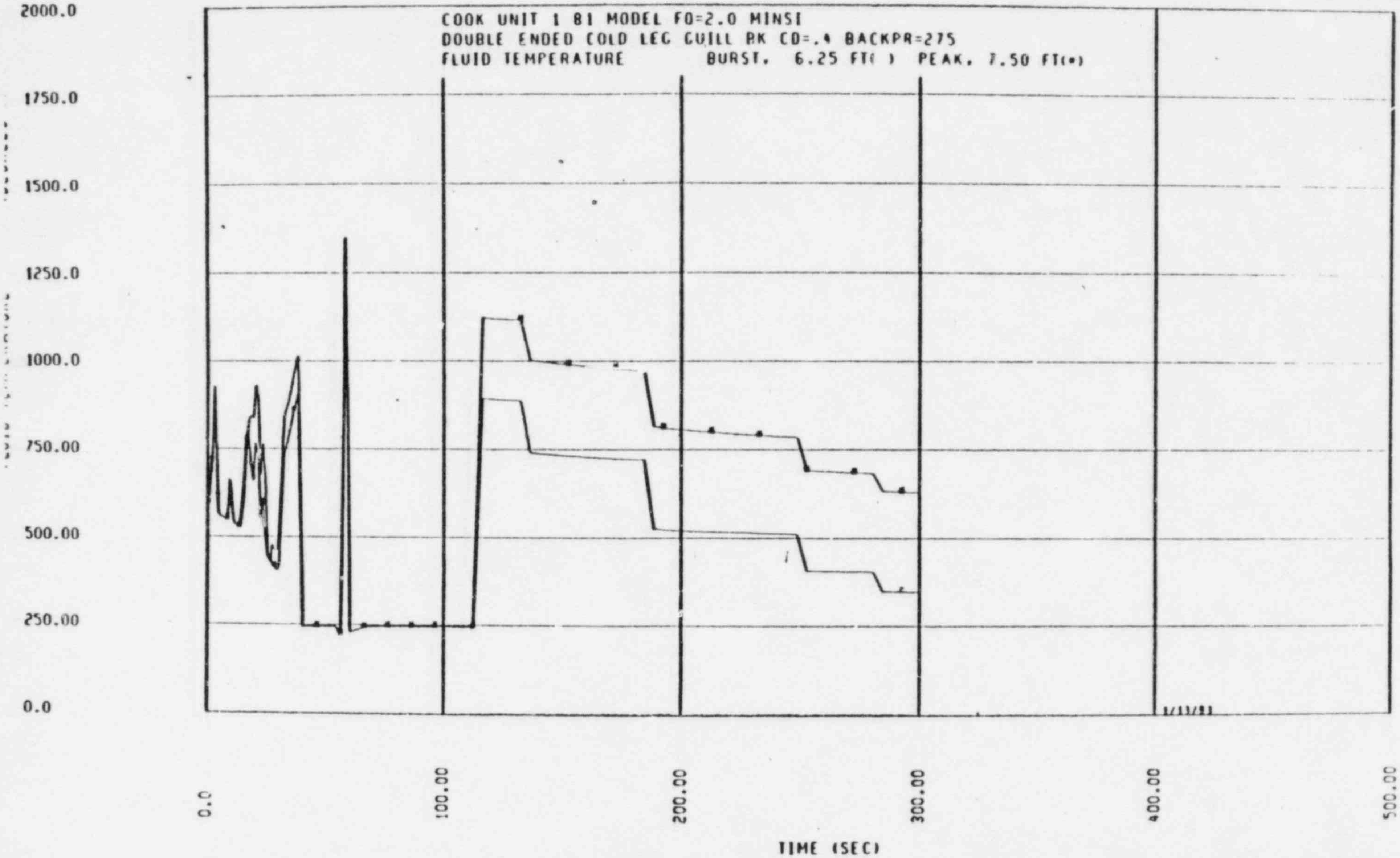


FIGURE 14.3.1-31 FLUID TEMPERATURE DECLG(CD = 0.4)

MIN SI

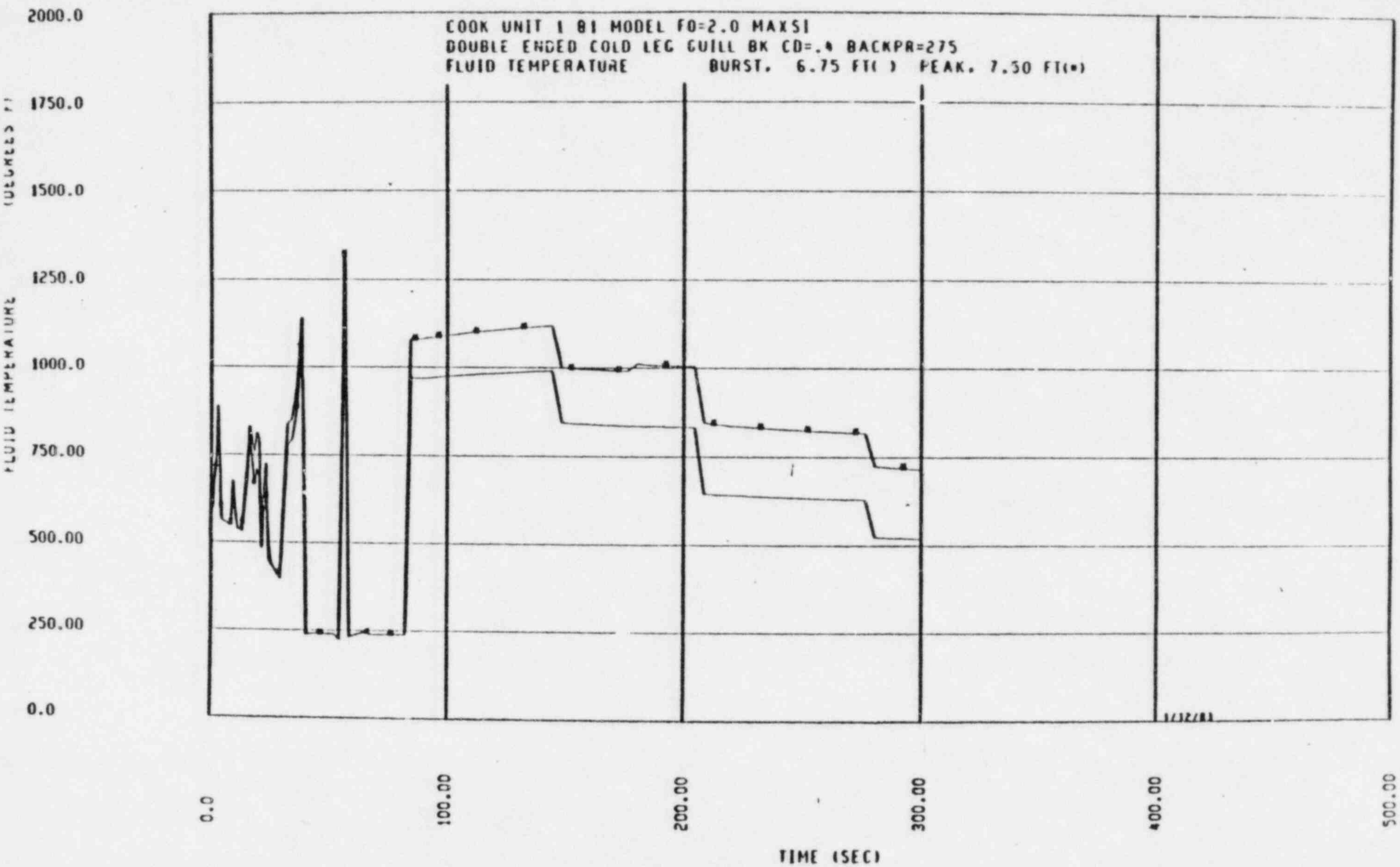


FIGURE 14.3.1-32 FLUID TEMPERATURE
DECLG(CD = 0.4)

MAX
SI

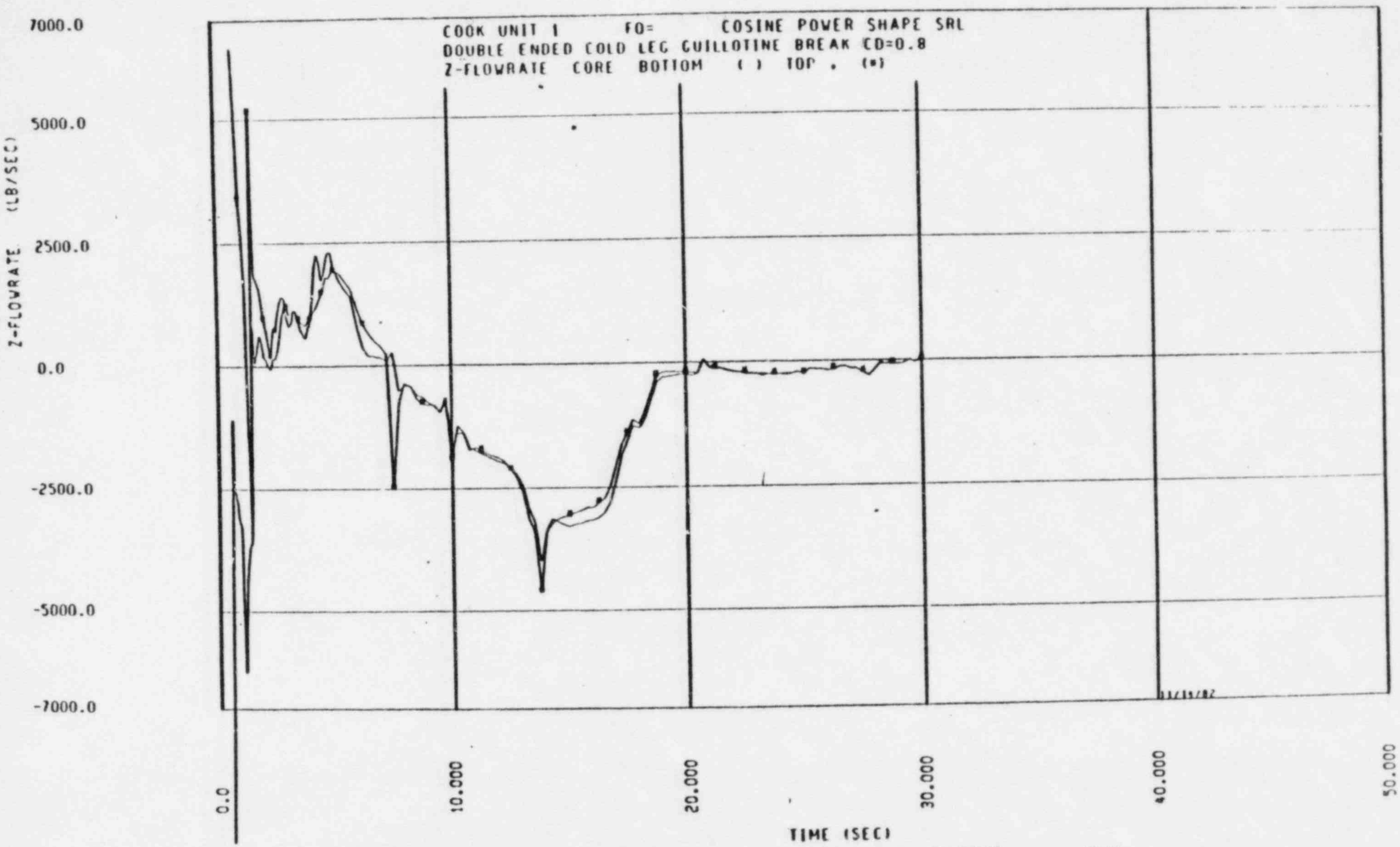


FIGURE 14.3.1-33 CORE FLOW (TOP AND BOTTOM) MIN SI
DECLG(CD = 0.8)

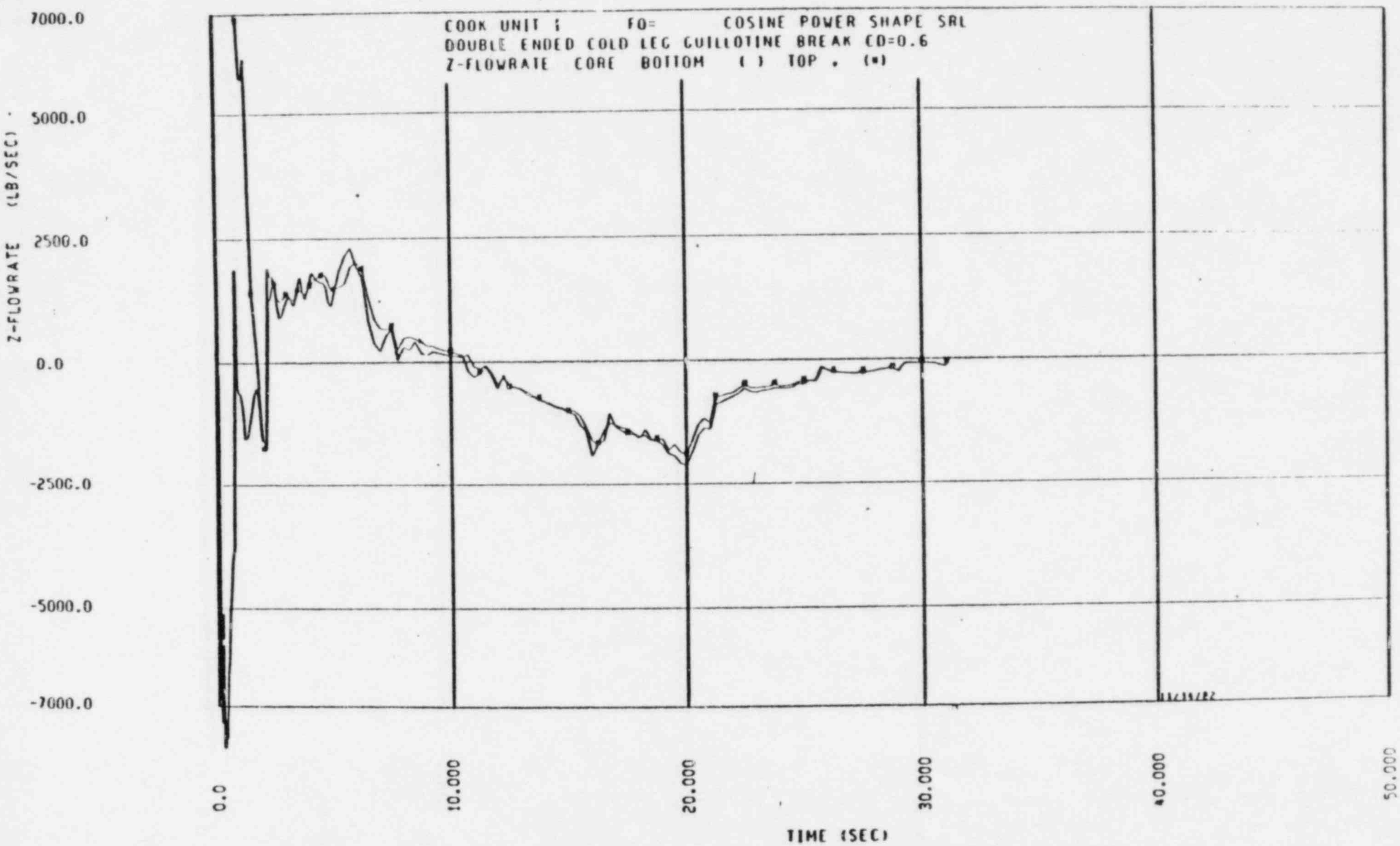


FIGURE 13.3.1-34 CORE FLOW (TOP AND BOTTOM) MIN
DECLG(CD = 0.6) SI

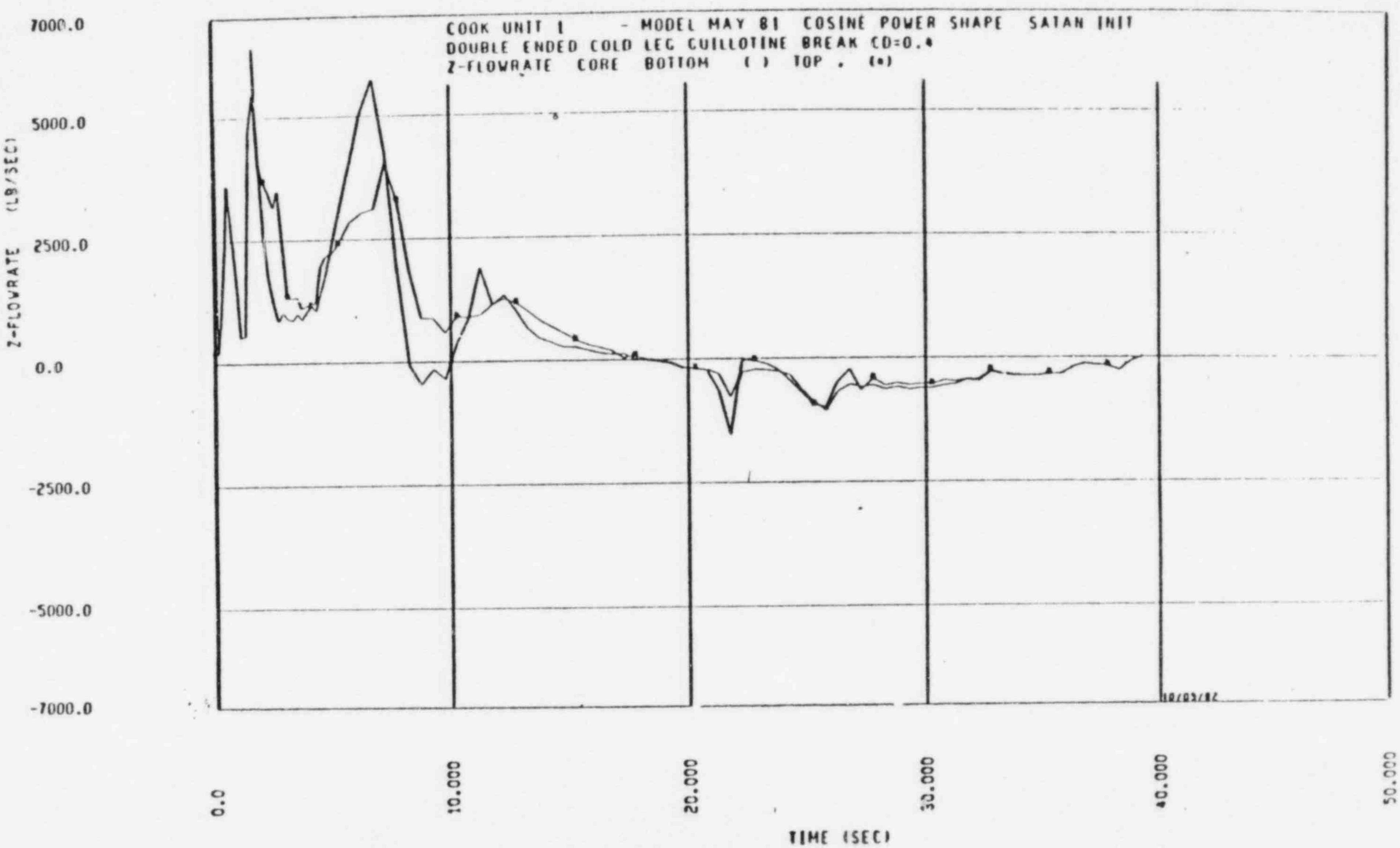


FIGURE 14.3.1-35 CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.4)

MIN
SI

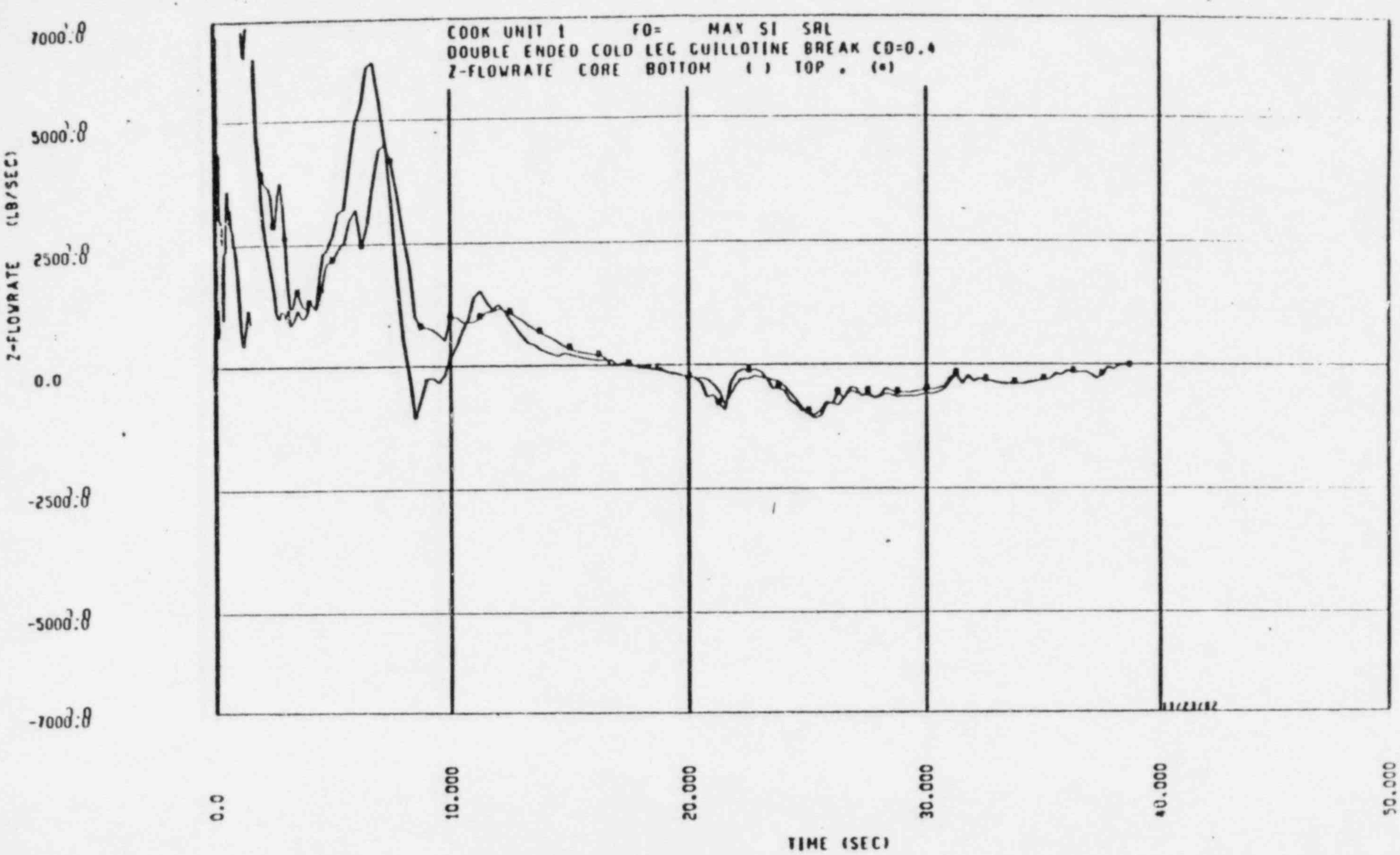


FIGURE 14.3.1-36 CORE FLOW (TOP AND BOTTOM) DECLG(CD = 0.4) MAX SI

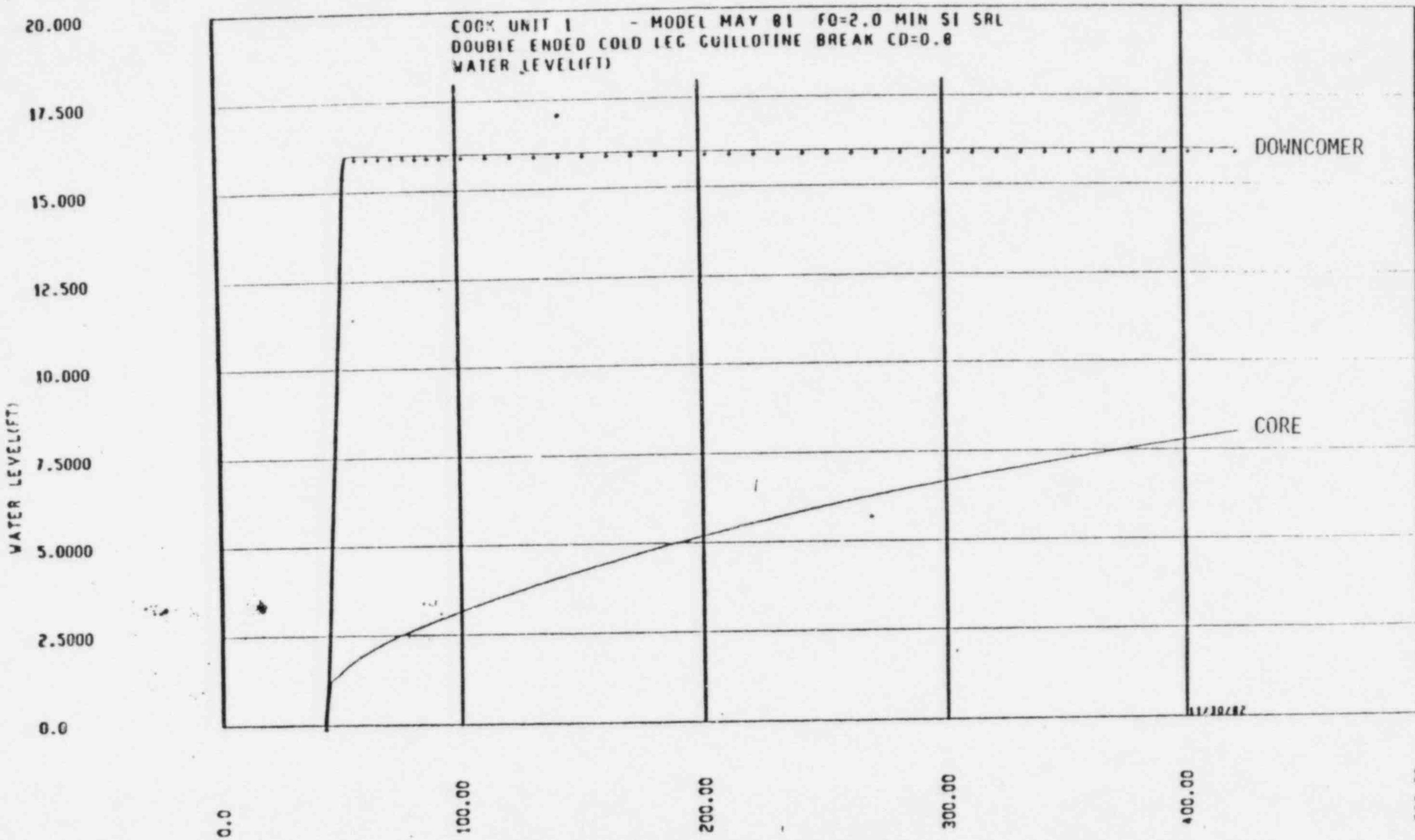


FIGURE 14.3.1-37 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.8) MIN SI

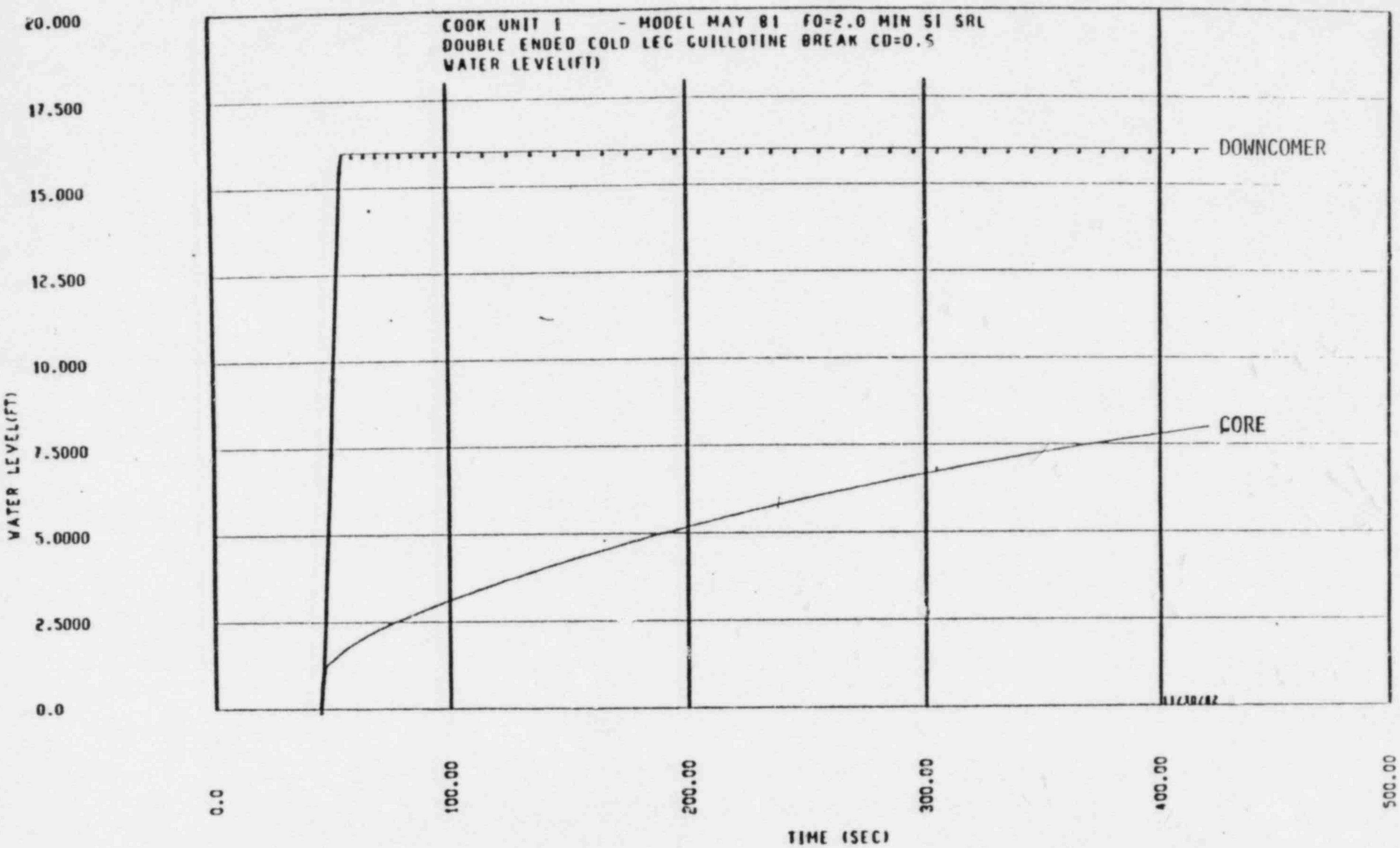


FIGURE 14.3.1-38 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.6) MIN SI

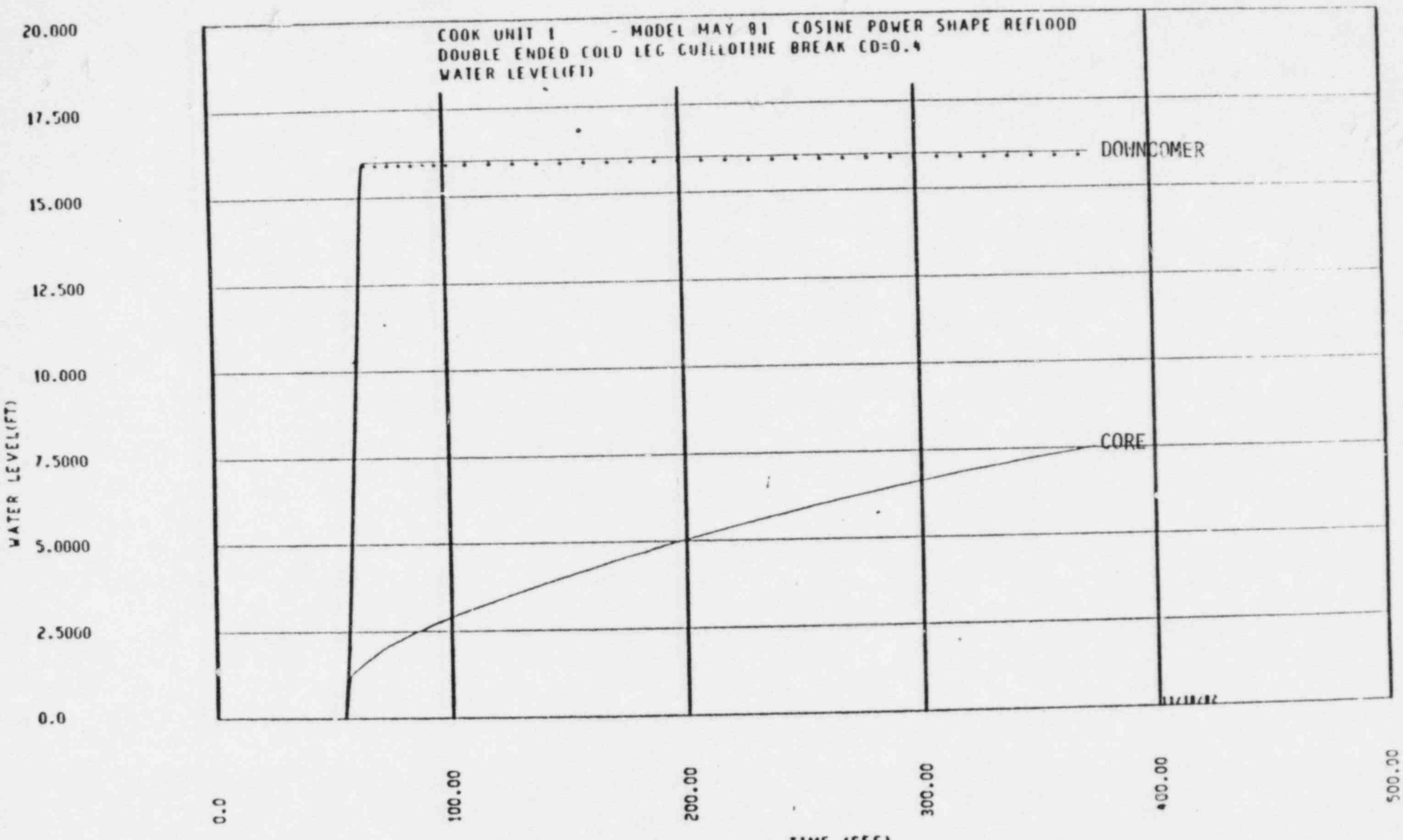


FIGURE 14.3.1-39 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4)

MIN SI

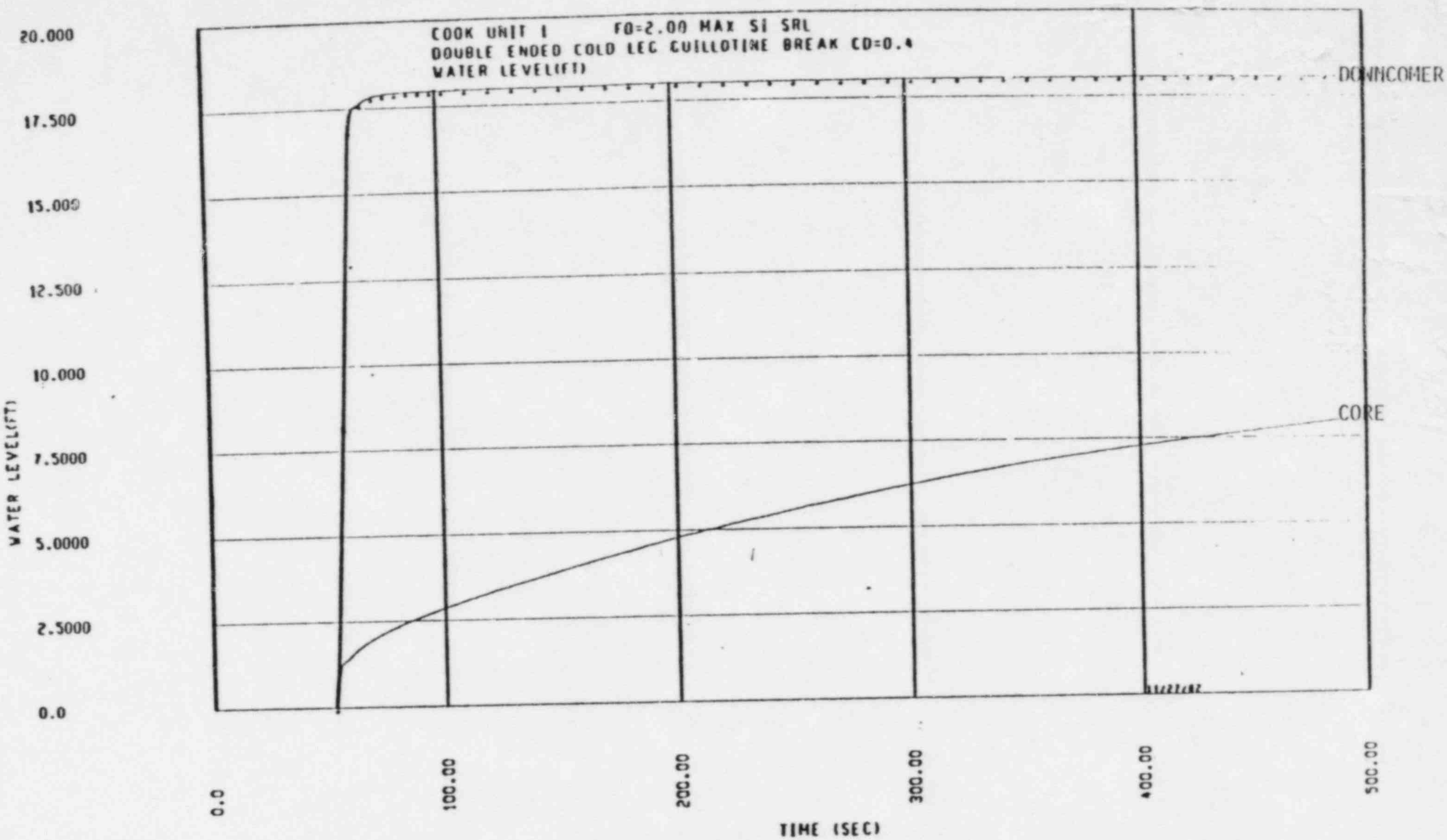


FIGURE 14.3.1-40 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4)

MAX SI

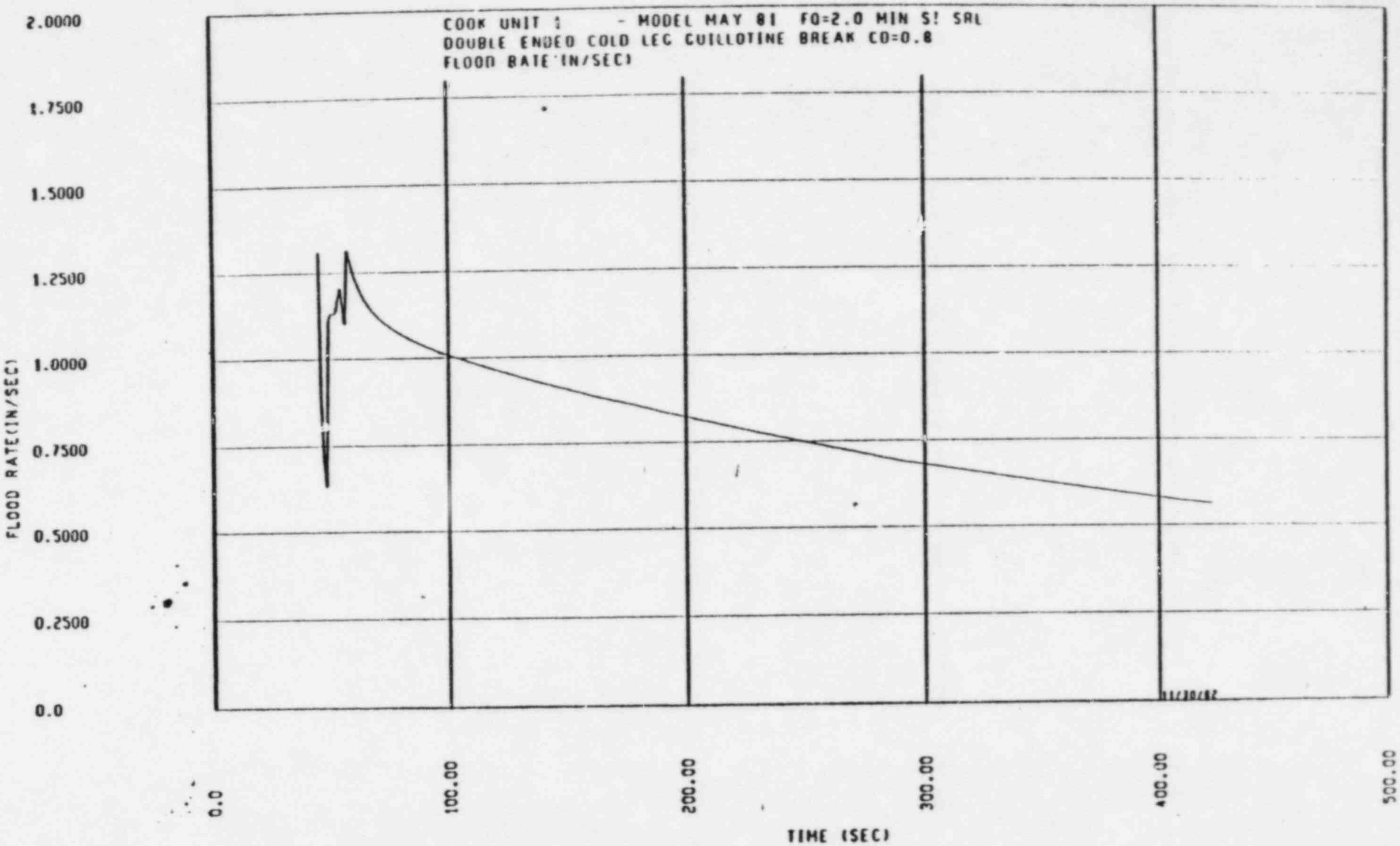


FIGURE 14.3.1-41 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.8)

MIN
SI

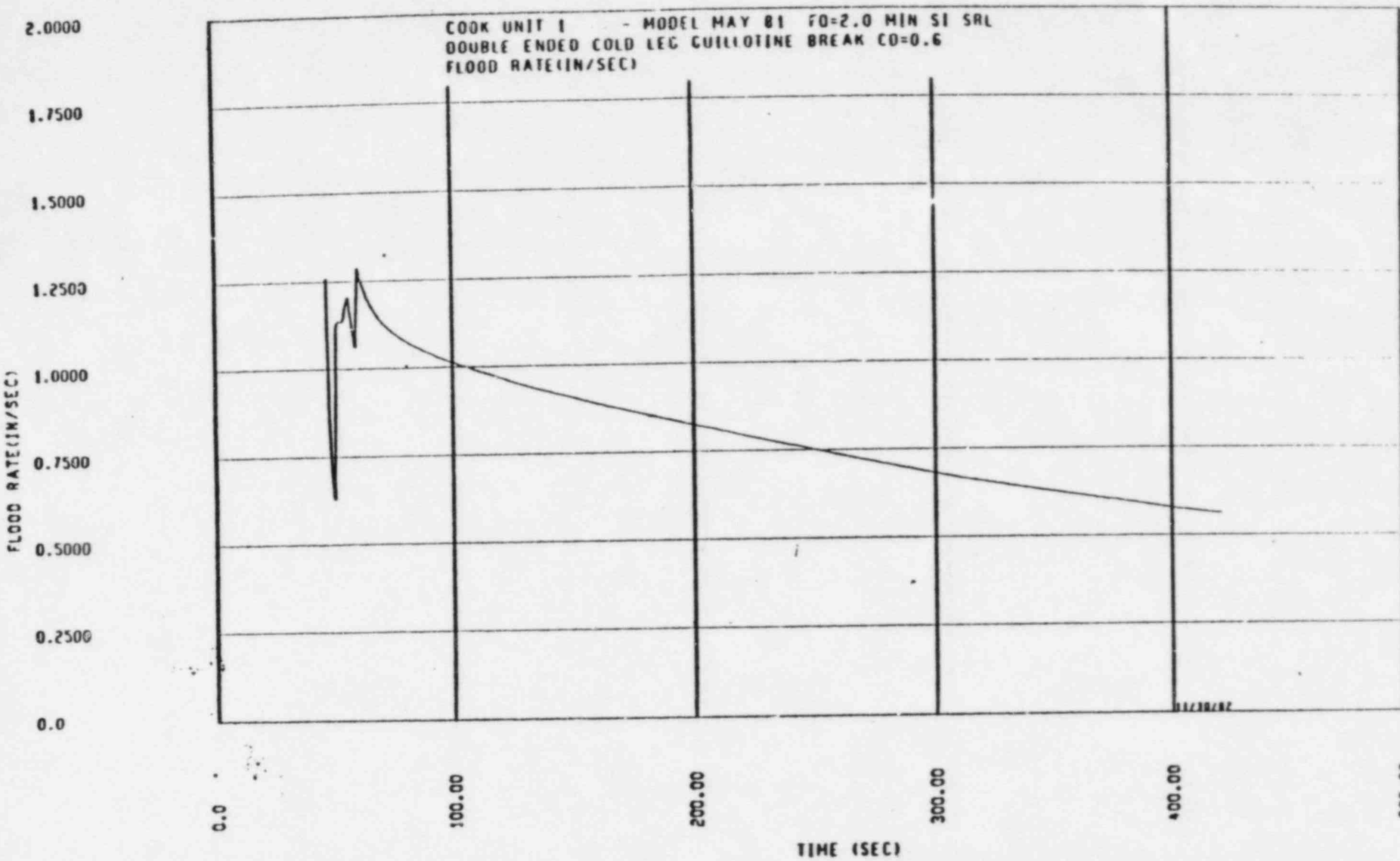


FIGURE 14.3.1-42 REFLOOD TRANSIENT MIN SI
CORE INLET VELOCITY
DECLG(CD = 0.6)

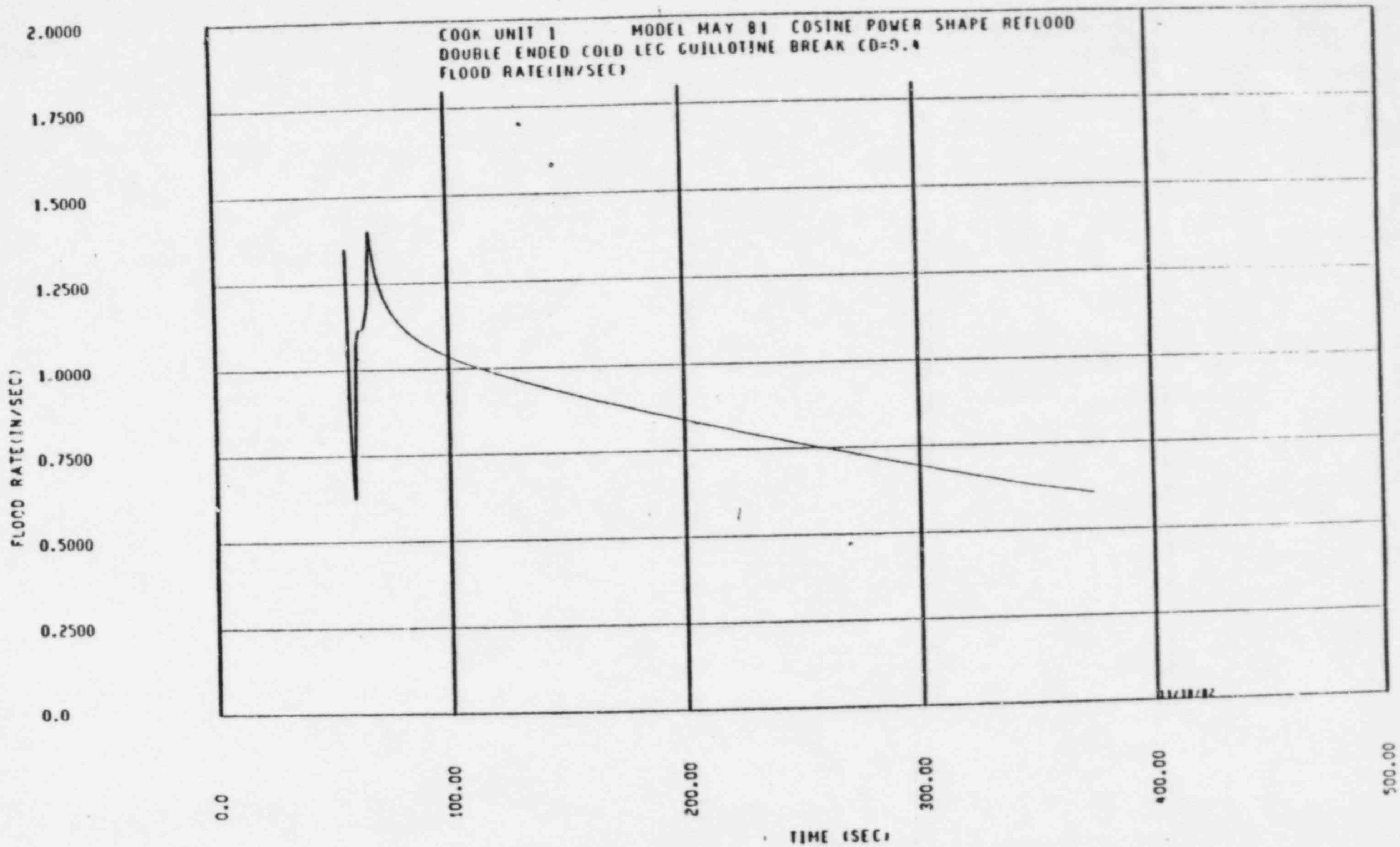


FIGURE 14.3.2-43 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.4)

IIIN
SI

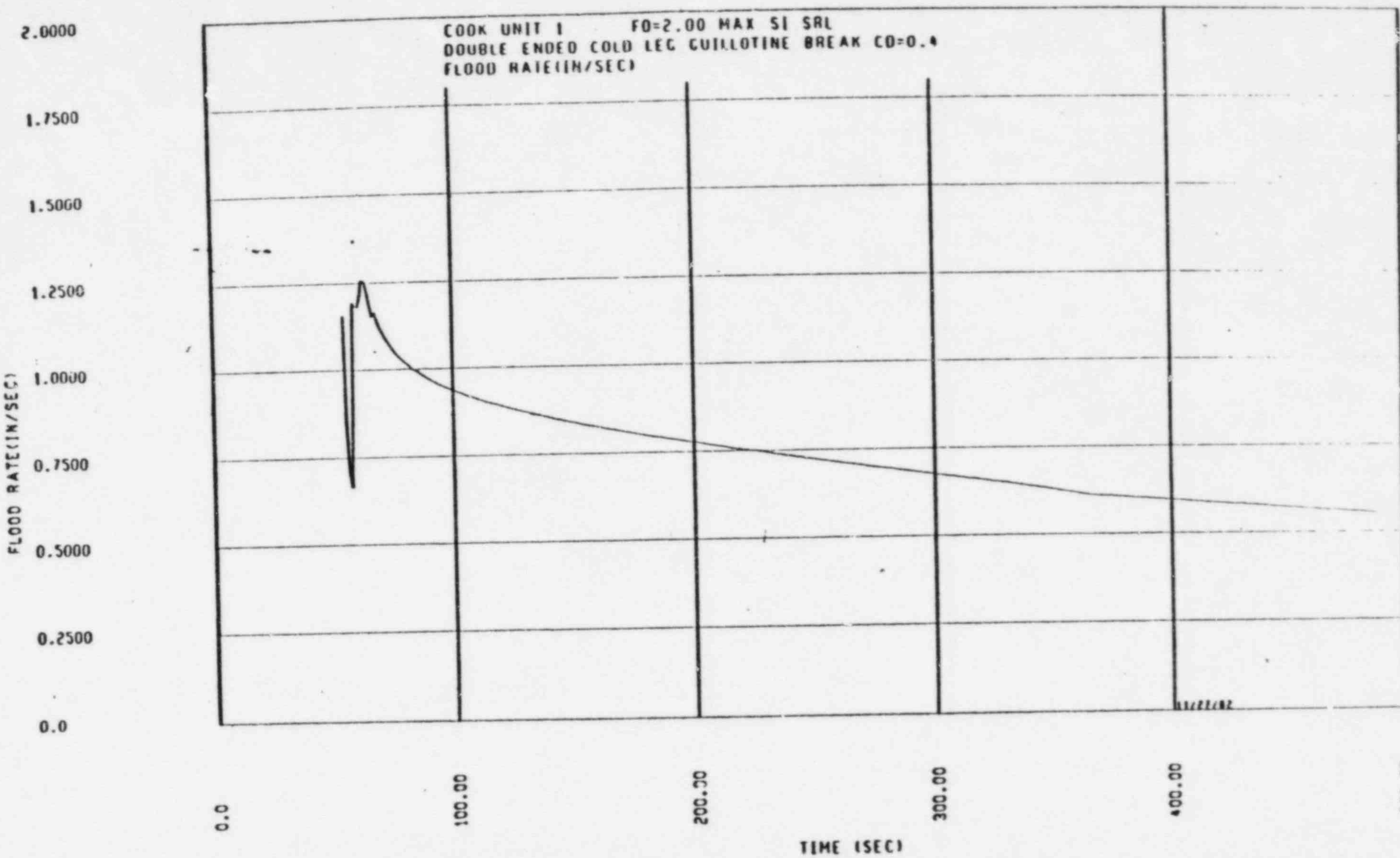


FIGURE 14.3.1-44 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.4)

MAX
SI

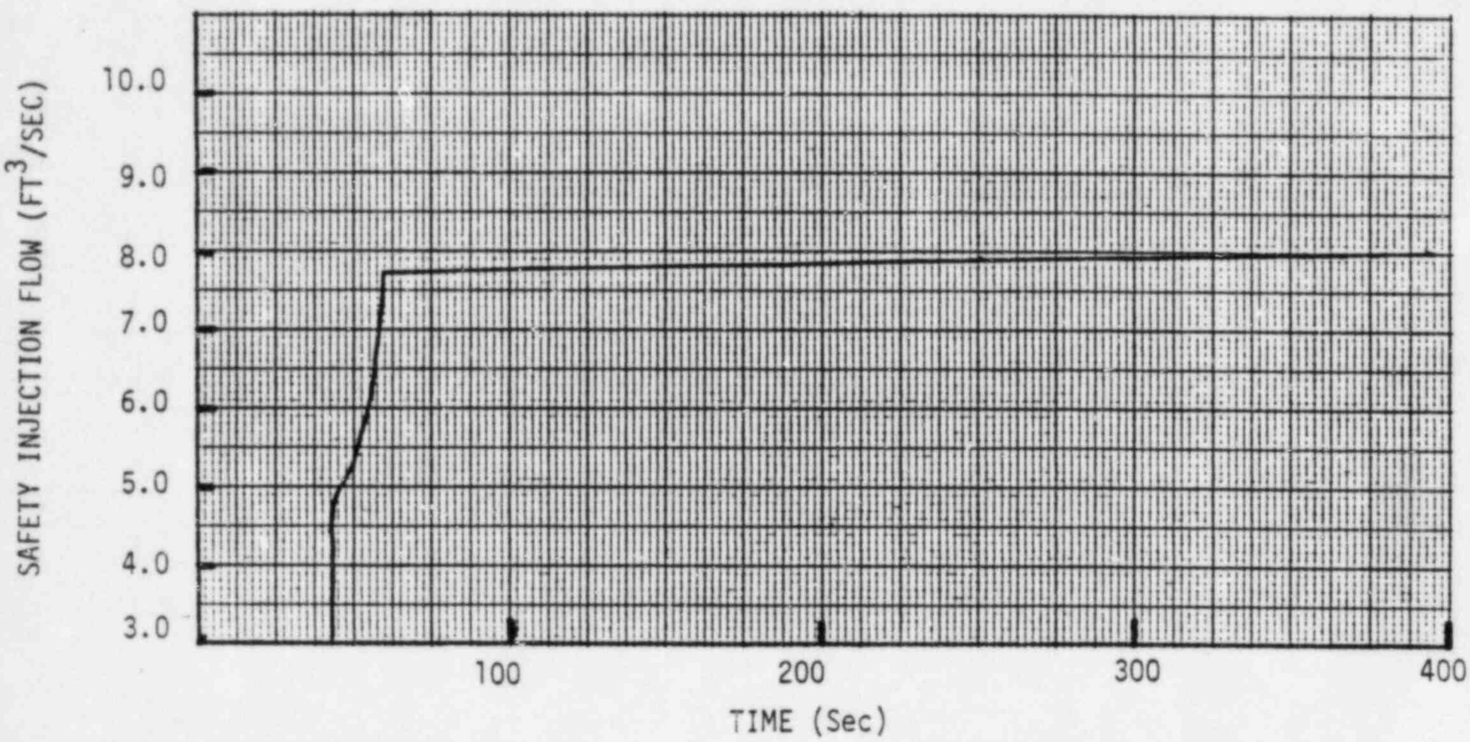


FIGURE 14.3.1-45 PUMPED ECCS FLOW (REFLOOD)
DECLG(CD = 0.8) MIN SI

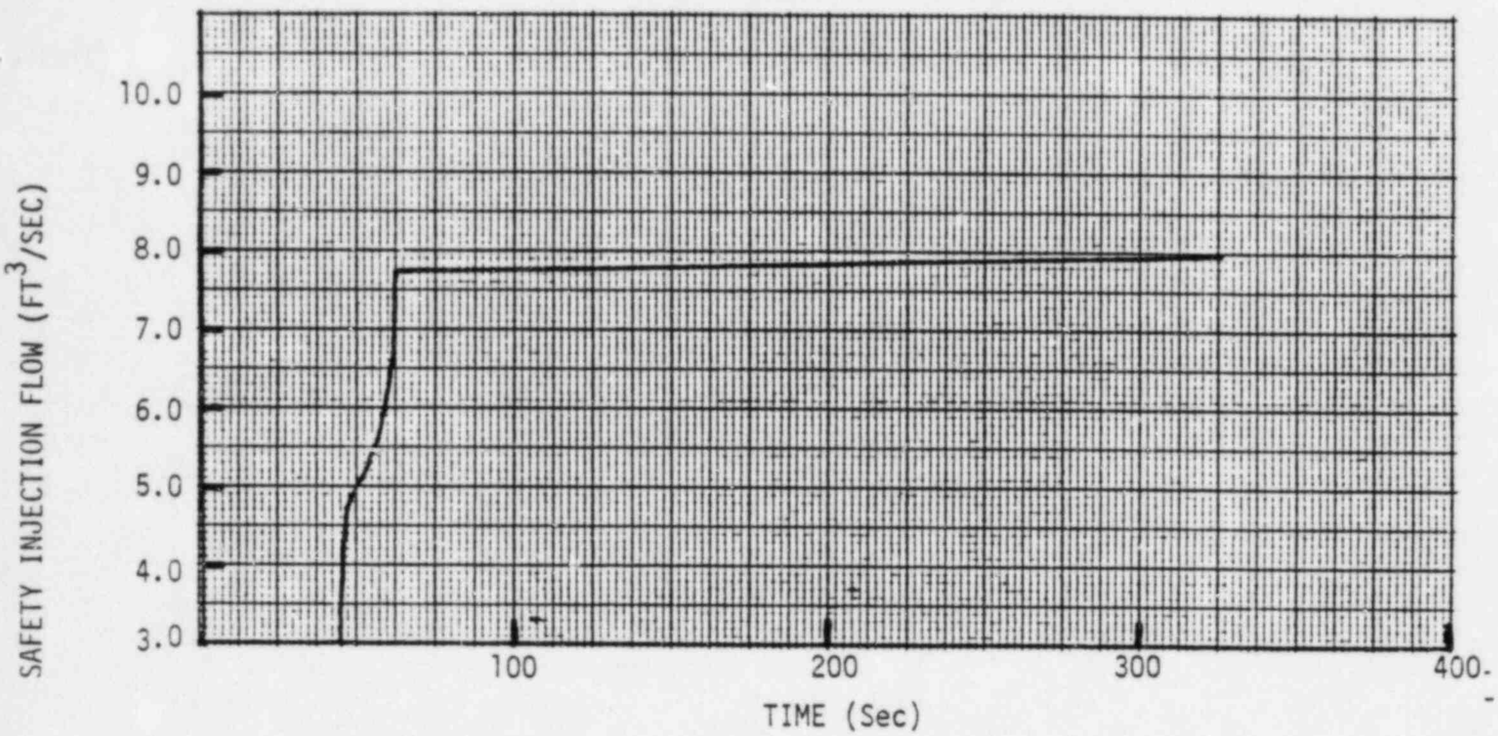


FIGURE 14.3.1-46 PUMPED ECCS FLOW (REFLOOD) DECLG(CD = 0.6) MIN SI

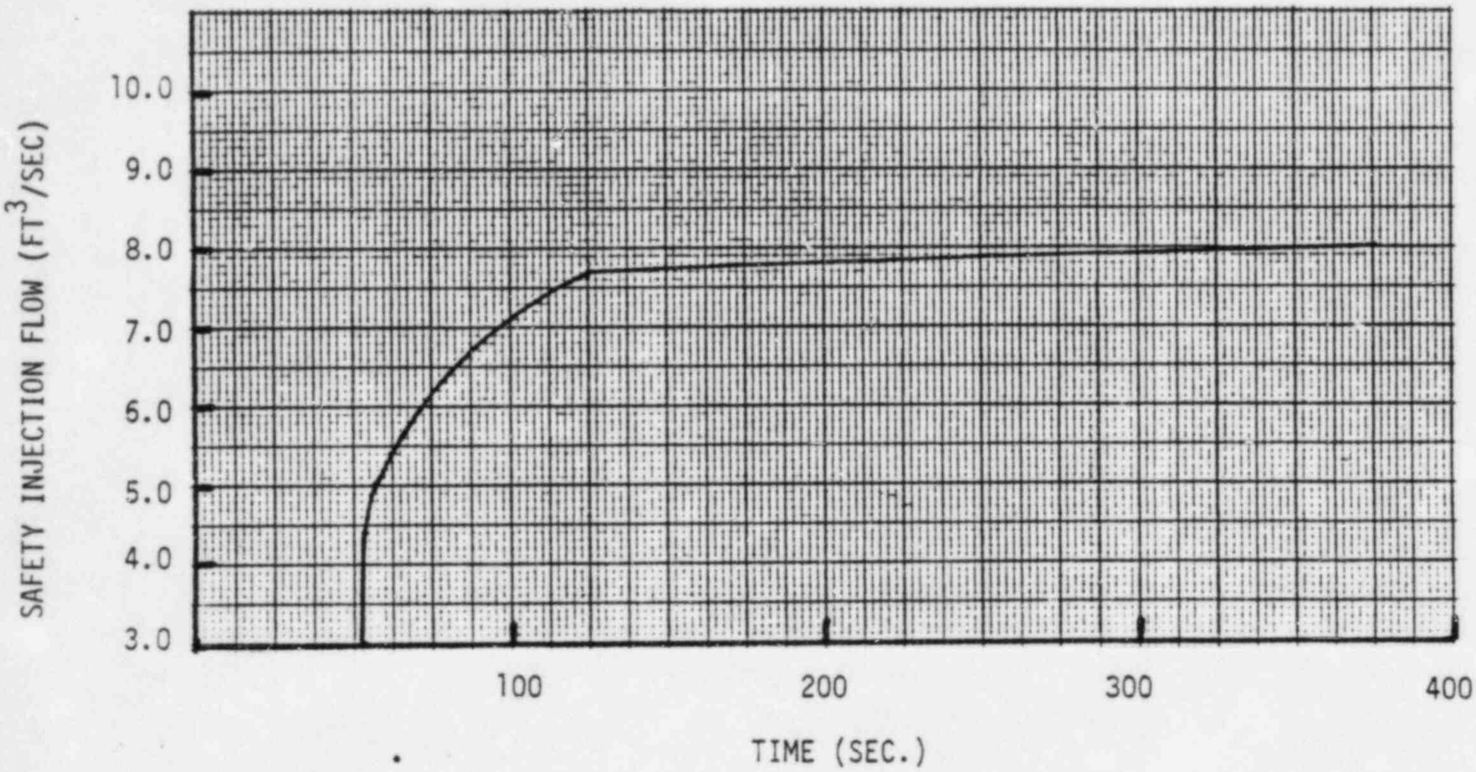


FIGURE 14.3.1-47 PUMPED ECCS FLOW (REFLOOD) DECLG(CD = 0.4)
MIN SI

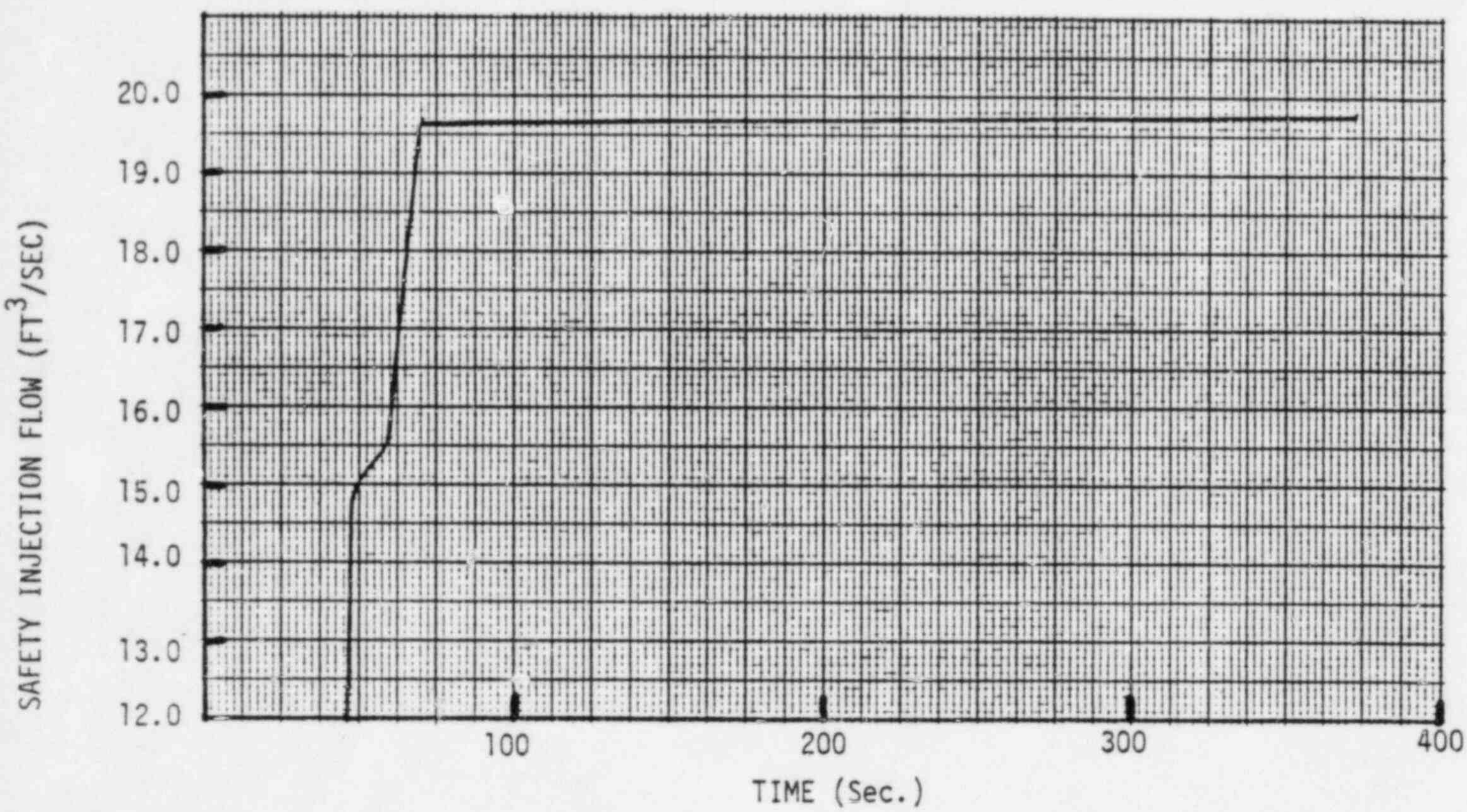


FIGURE 14.3.1-48 PUMPED ECCS FLOW (REFLOOD - DECLG (CD - 0.4) MAX SI

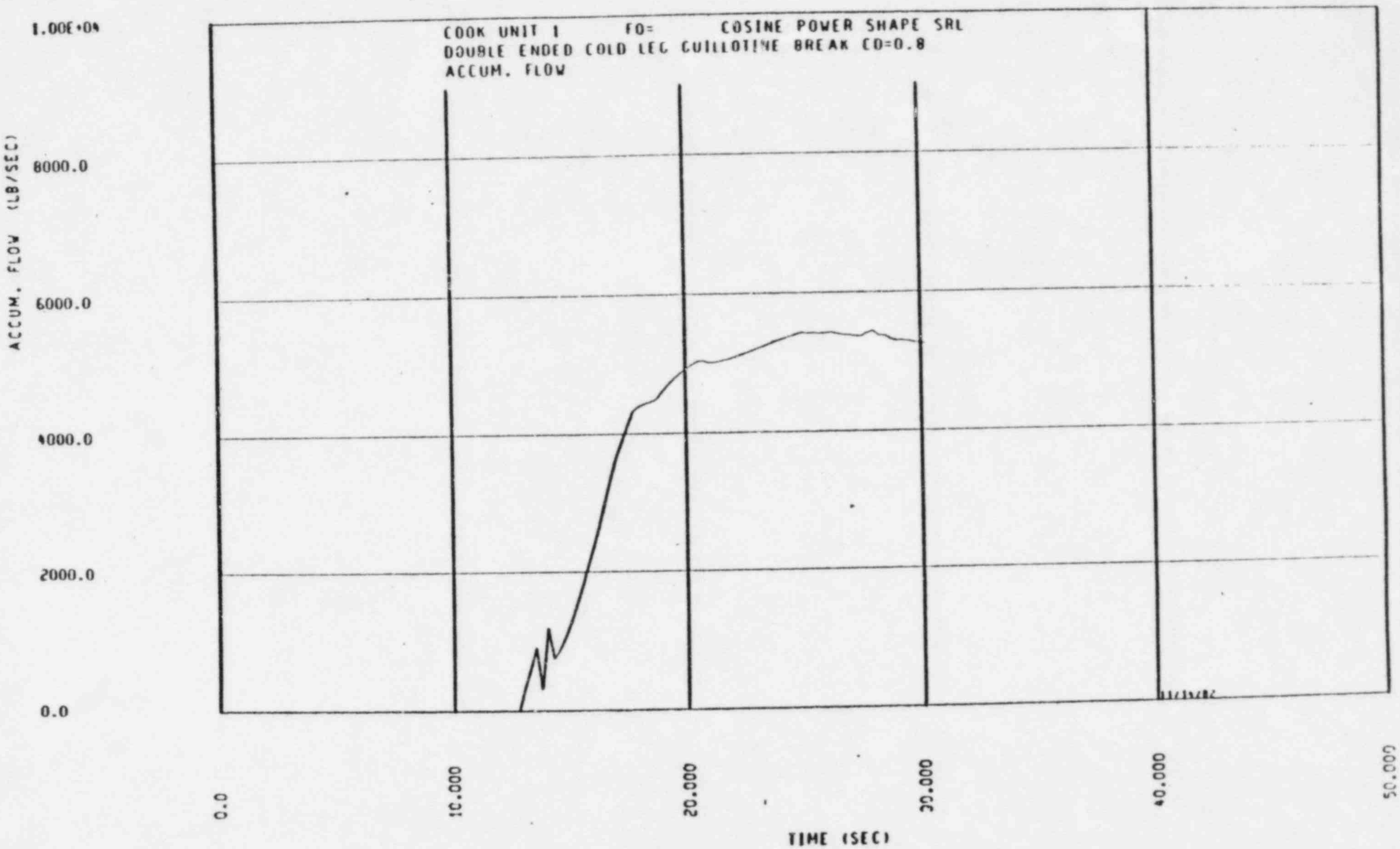


FIGURE 14.3.1-49 ACCUMULATOR FLOW (BLOWDOWN) DECLG(CD = 0.8) MIN SI

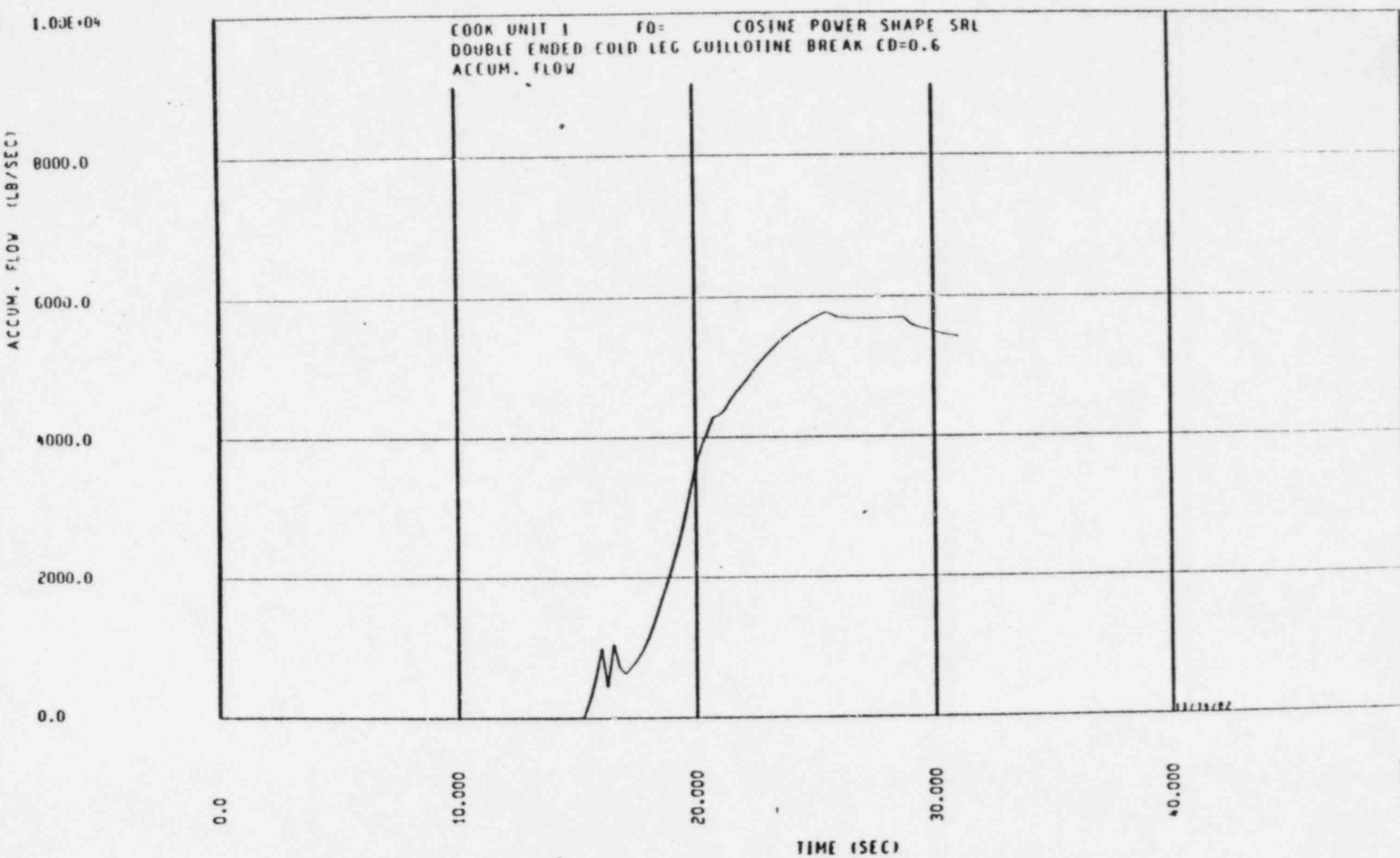


FIGURE 14.3.1-50 ACCUMULATOR FLOW (BLOWDOWN) MIN
DECLG(CD = 0.6) SI

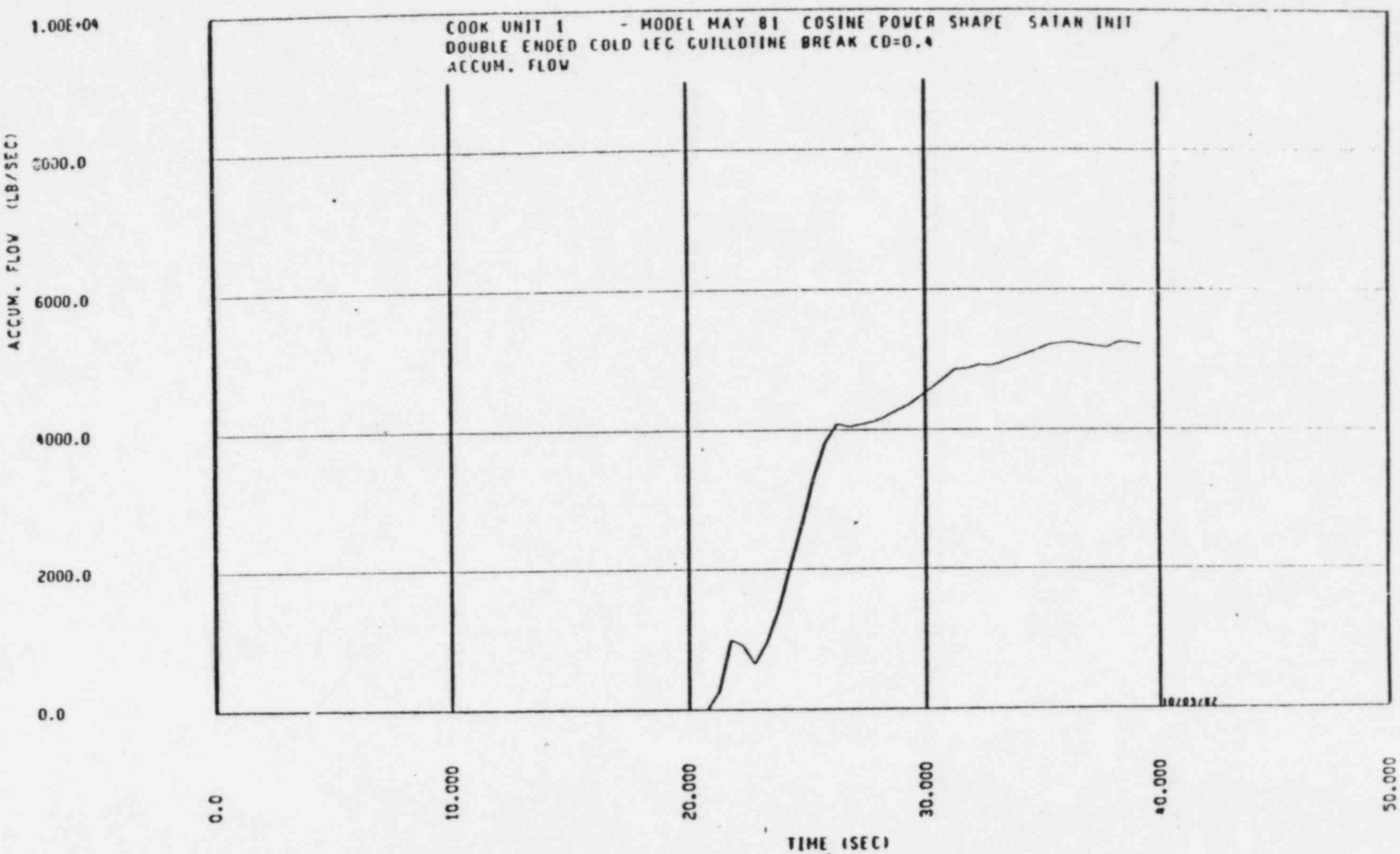


FIGURE 14.3.1-51 ACCUMULATOR FLOW (BLOWDOWN) MIN SI
DECLG(CD = 0.4)

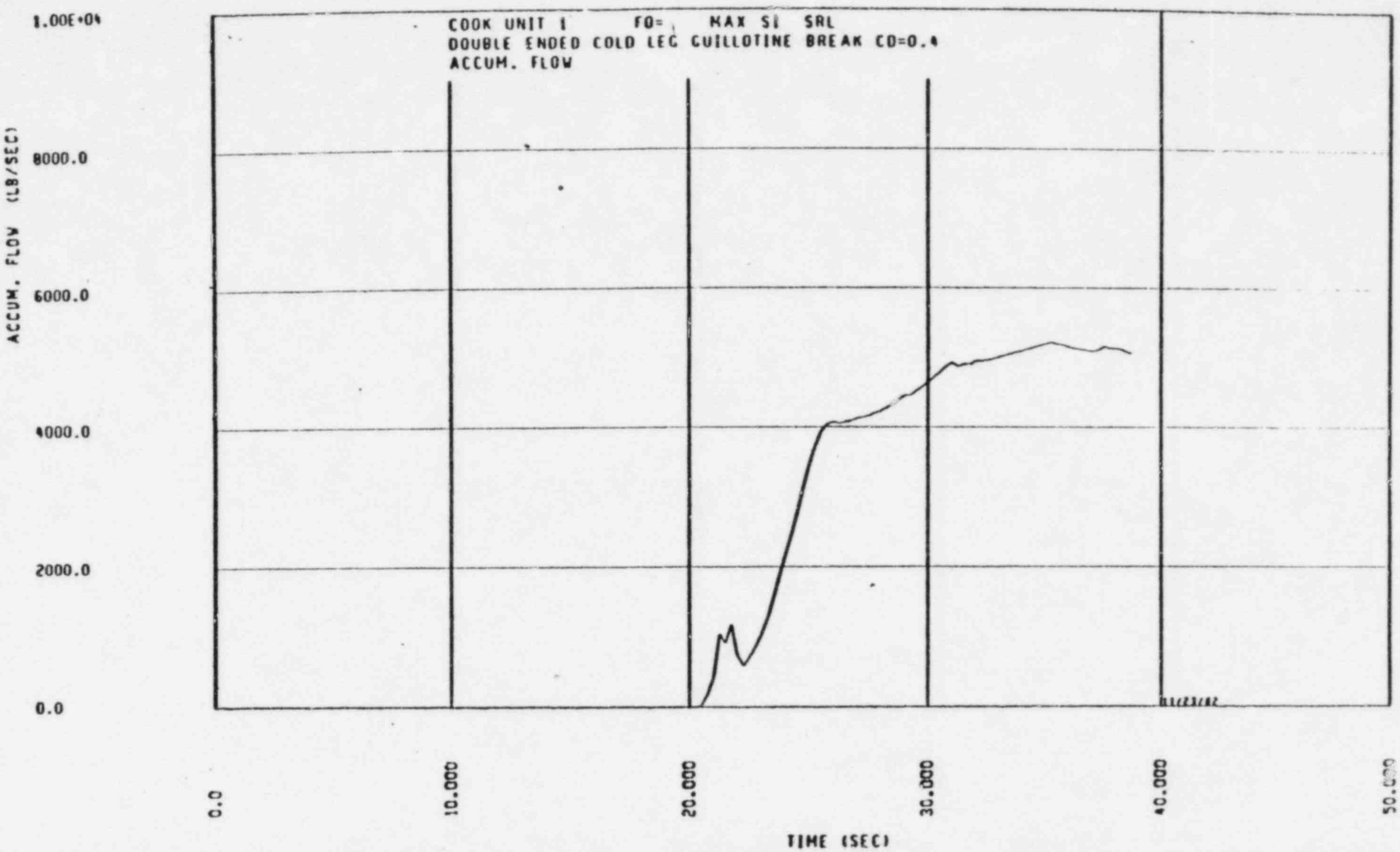
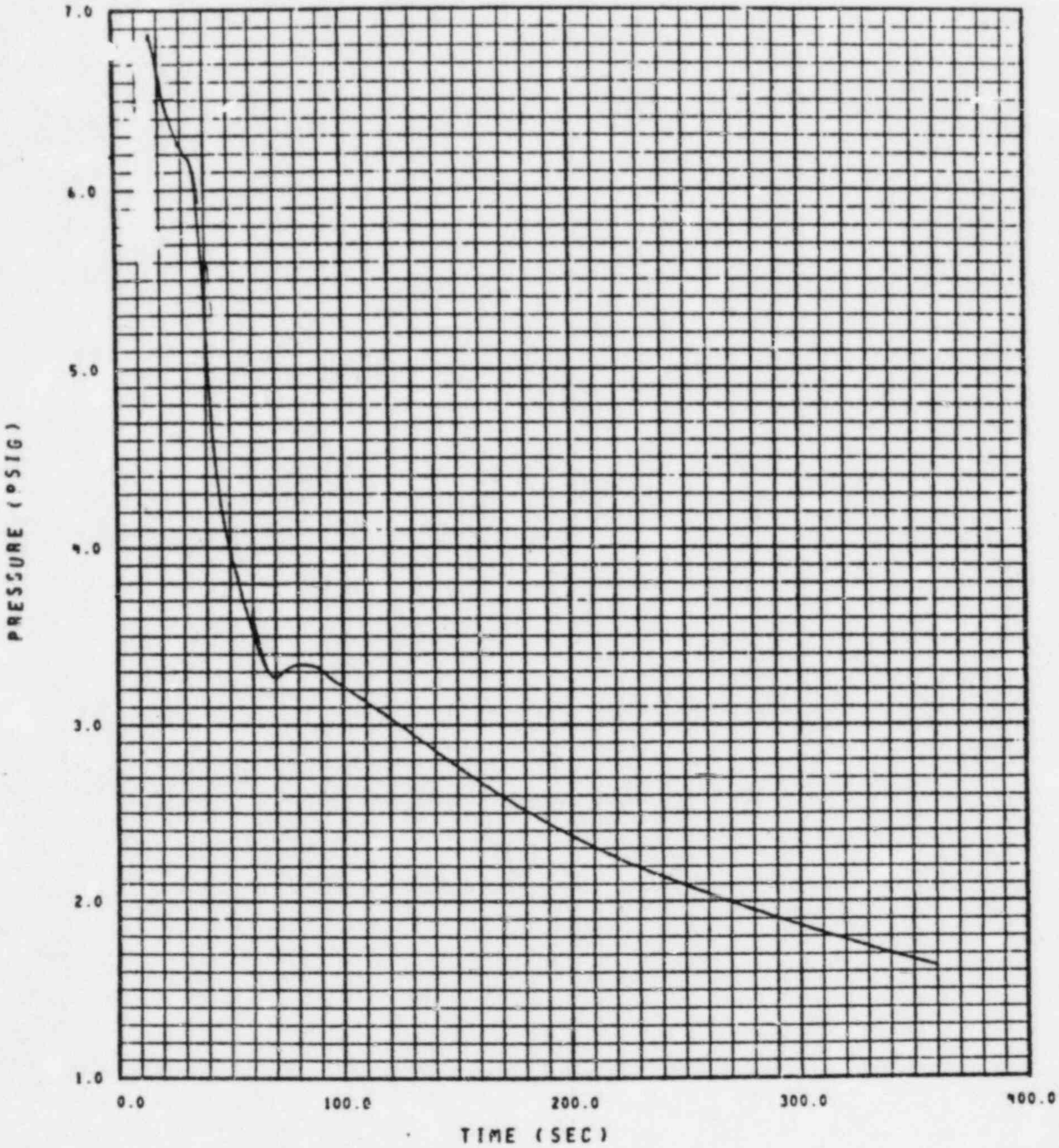


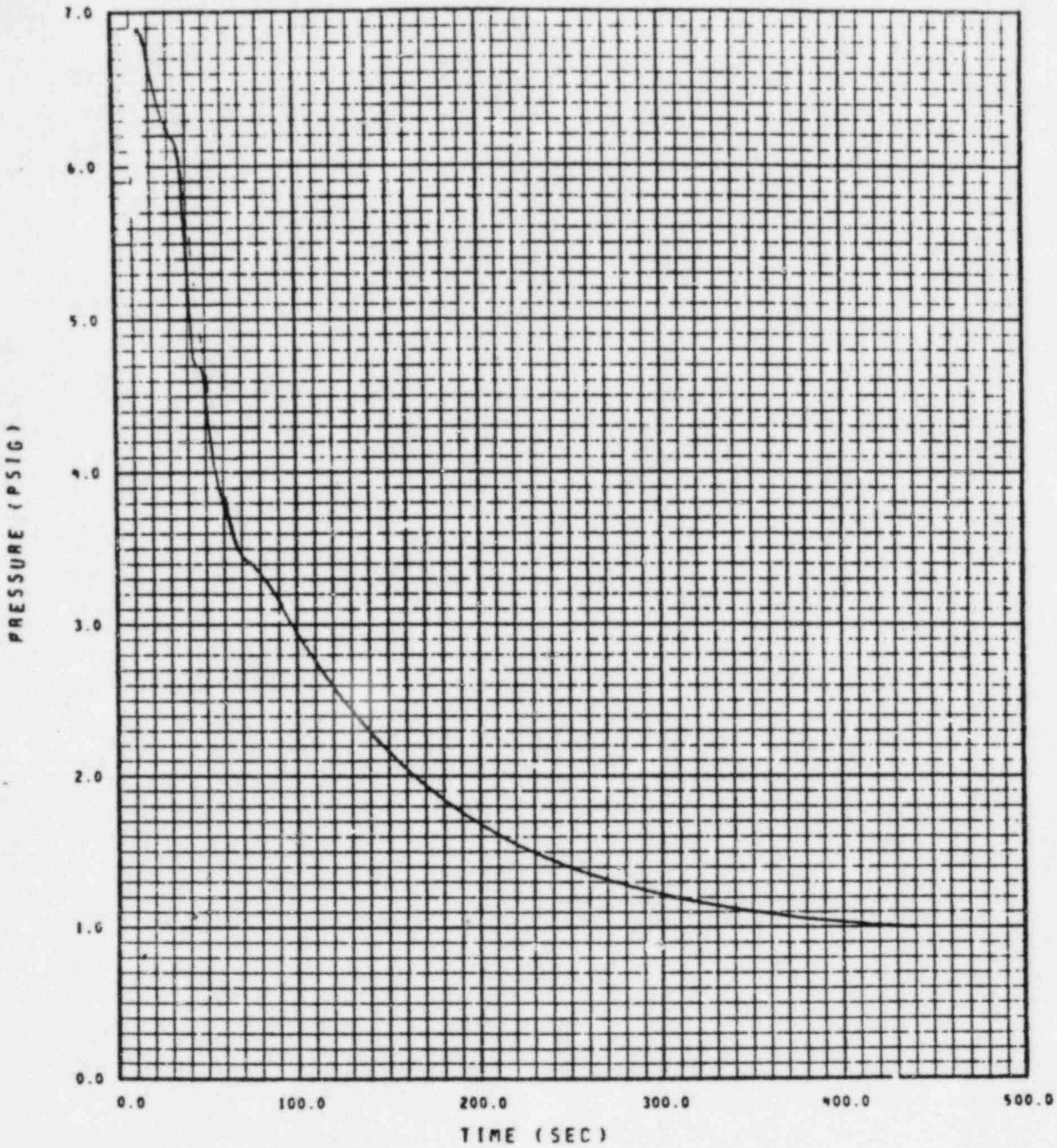
FIGURE 14.3.1-52 ACCUMULATOR FLOW (BLOWDOWN) MAX
DECLG(CD = 0.4) SI



COMPARTMENT PRESSURE

MINIMUM SI

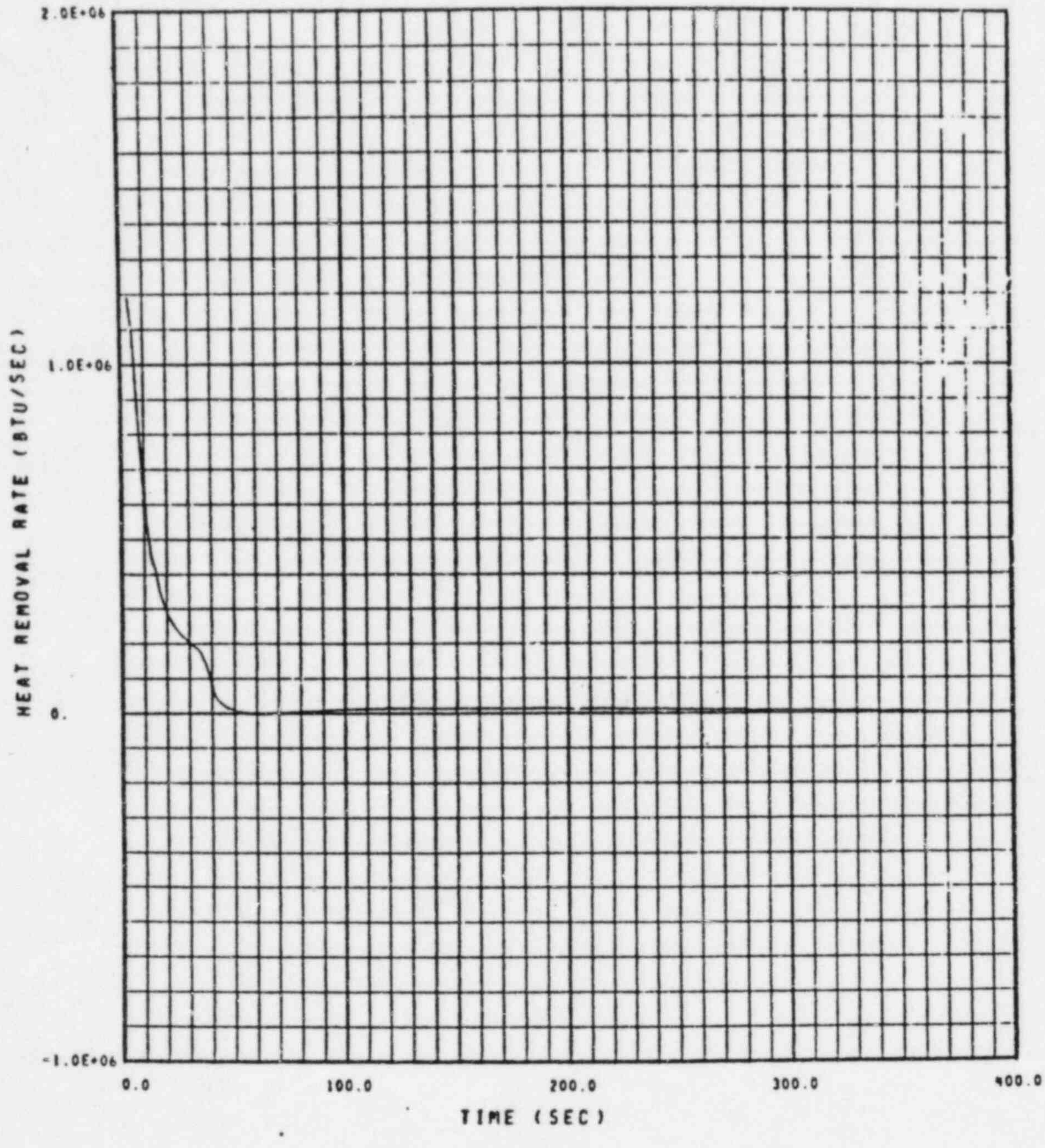
FIGURE 14.3.1-53 CONTAINMENT PRESSURE
DECLG(CD = 0.4)



COMPARTMENT PRESSURE

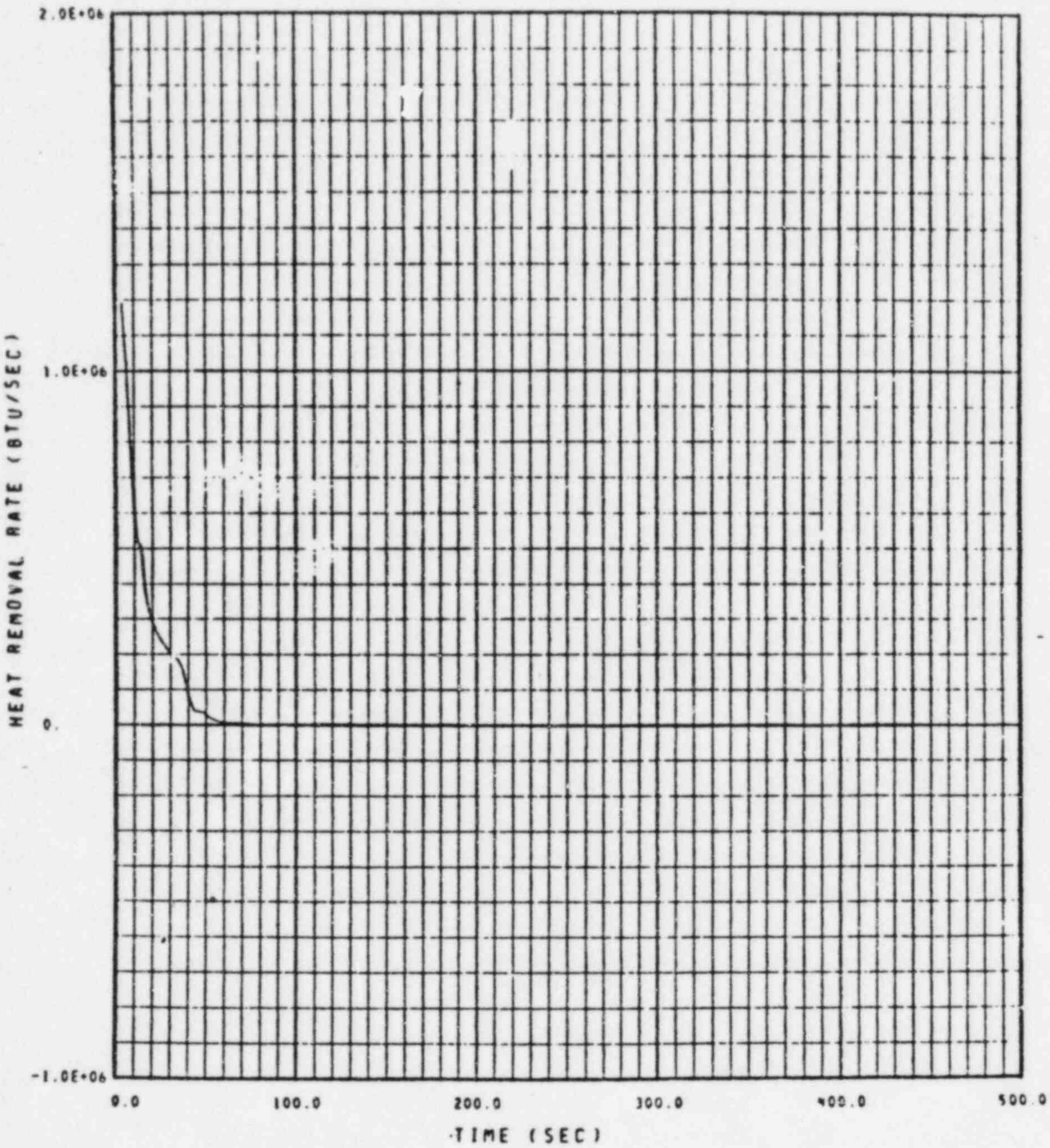
MAXIMUM SI

FIGURE 14.3.1-54 CONTAINMENT PRESSURE
DECLG(CD = 0.4)



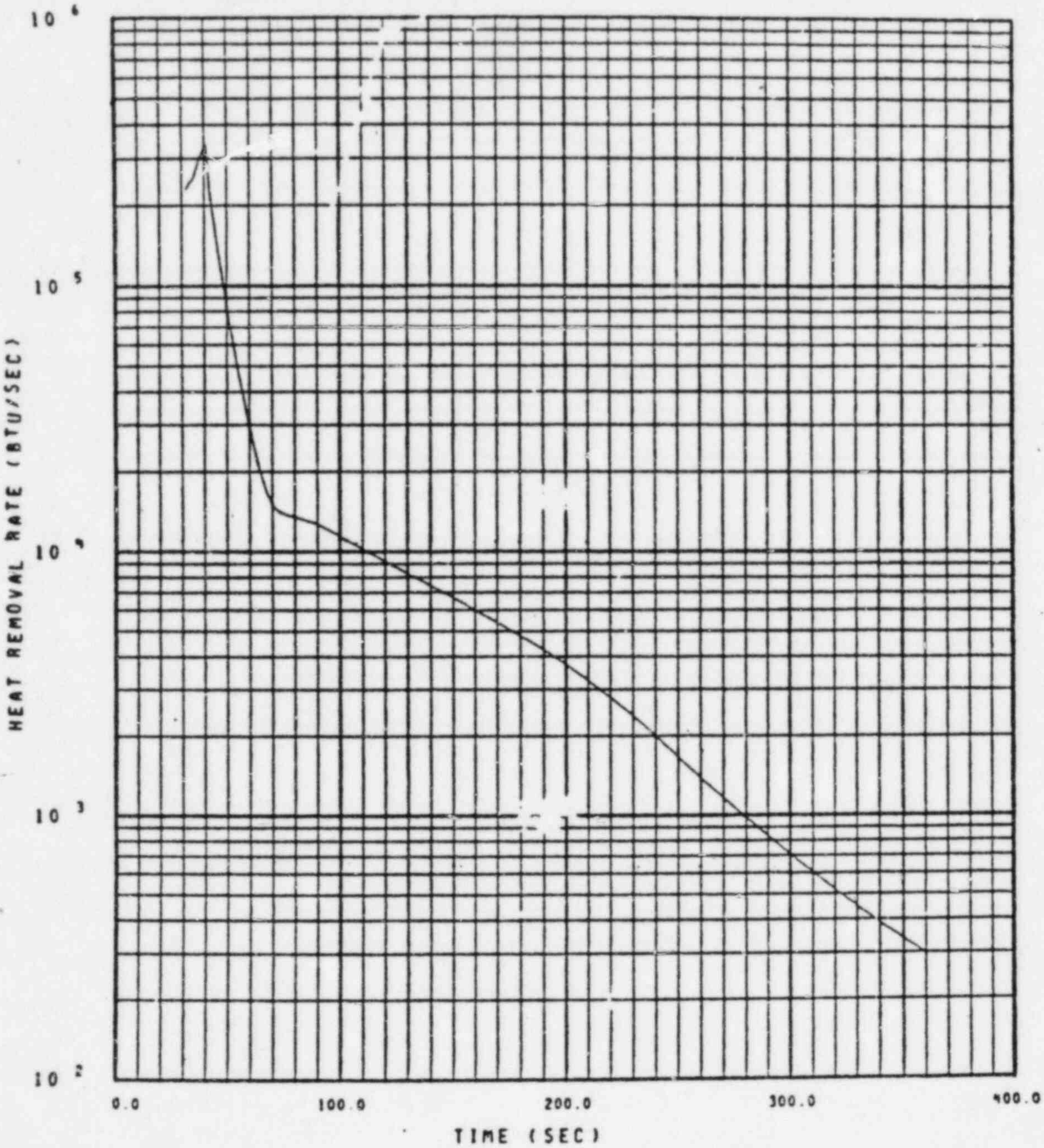
MINIMUM SI

FIGURE 14.3.1-55 LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



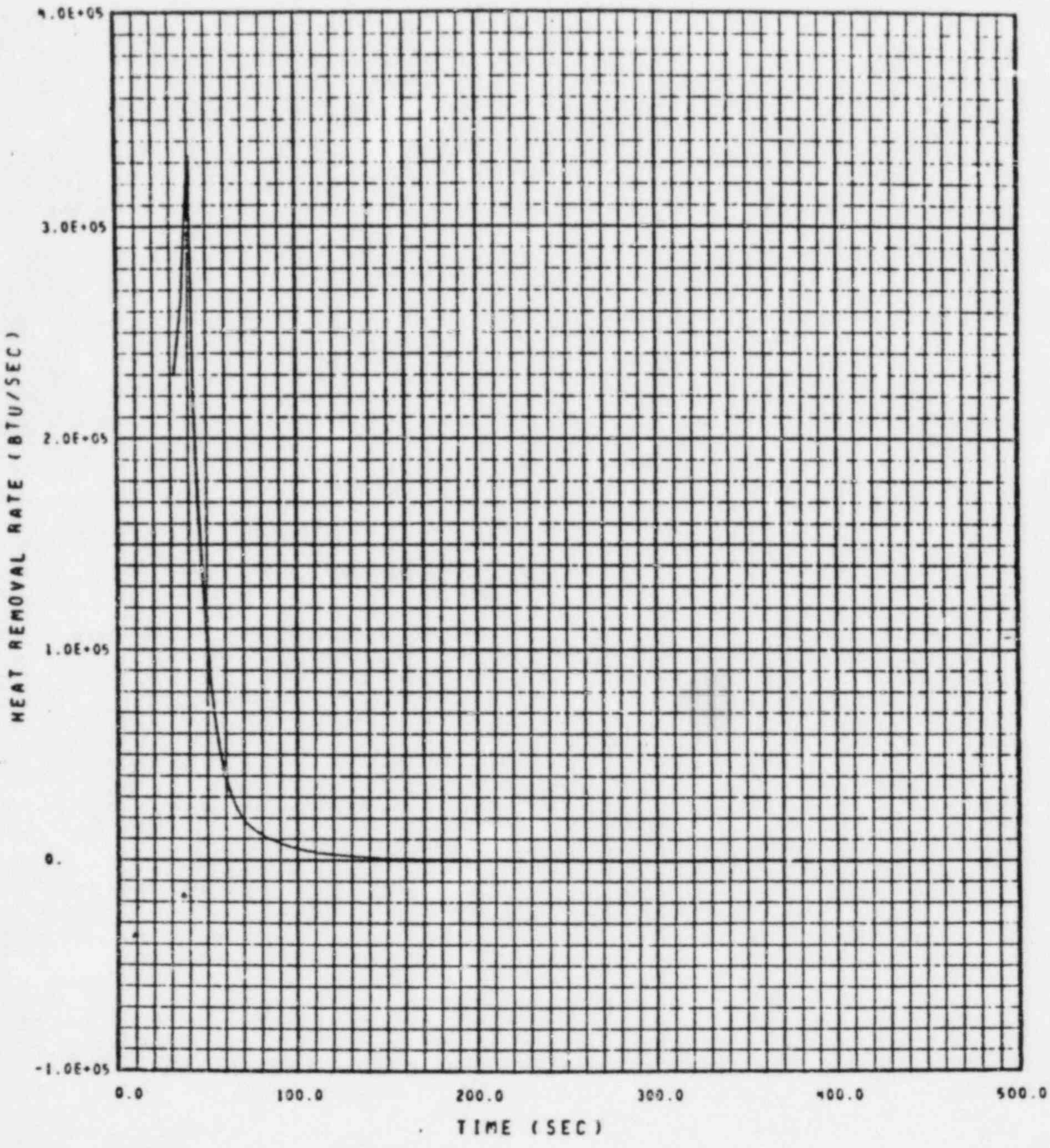
MAXIMUM SI

FIGURE 14.3.1-56 LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



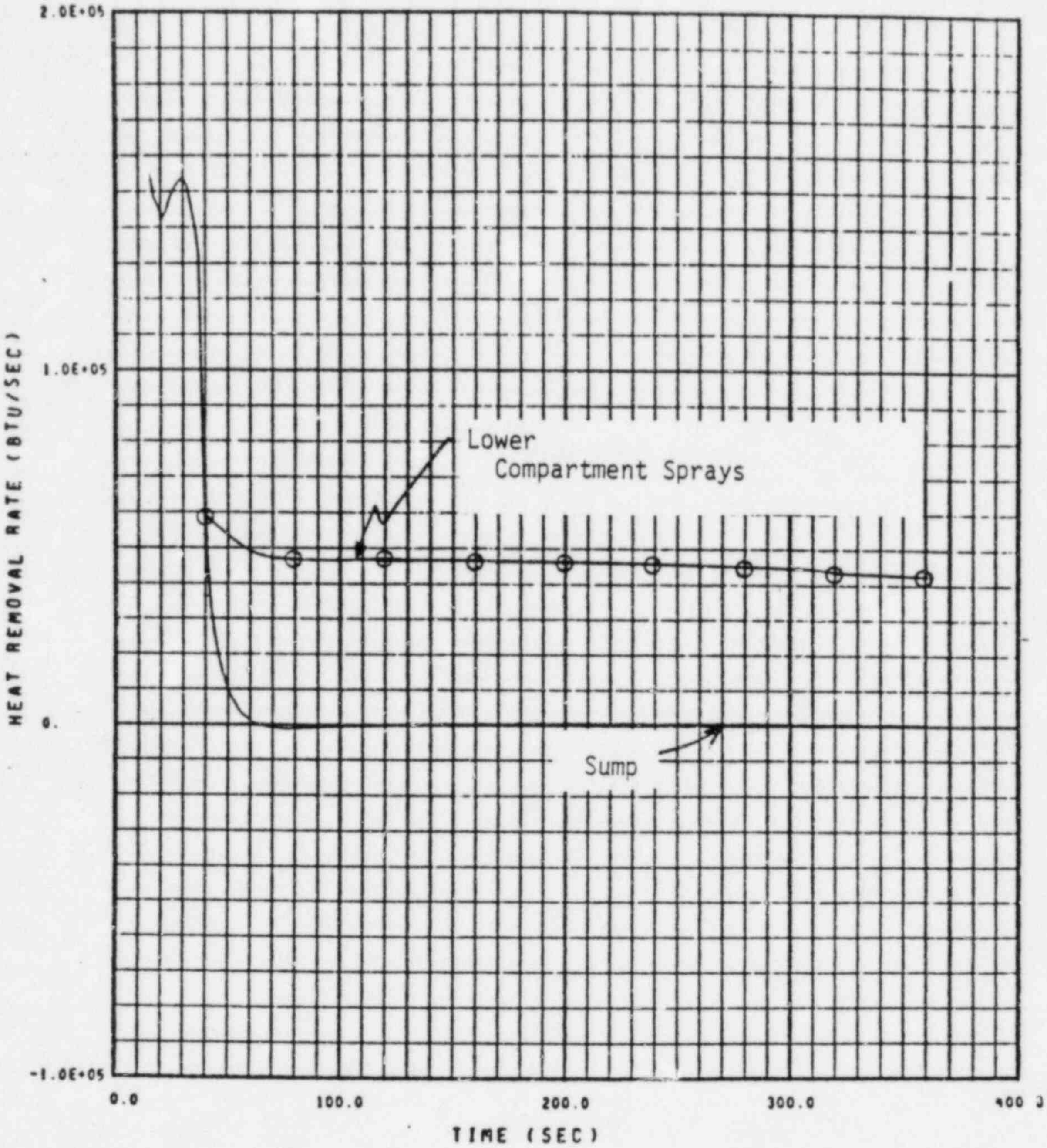
MINIMUM SI

FIGURE 14.3.1-57 HEAT REMOVAL BY LC DRAIN



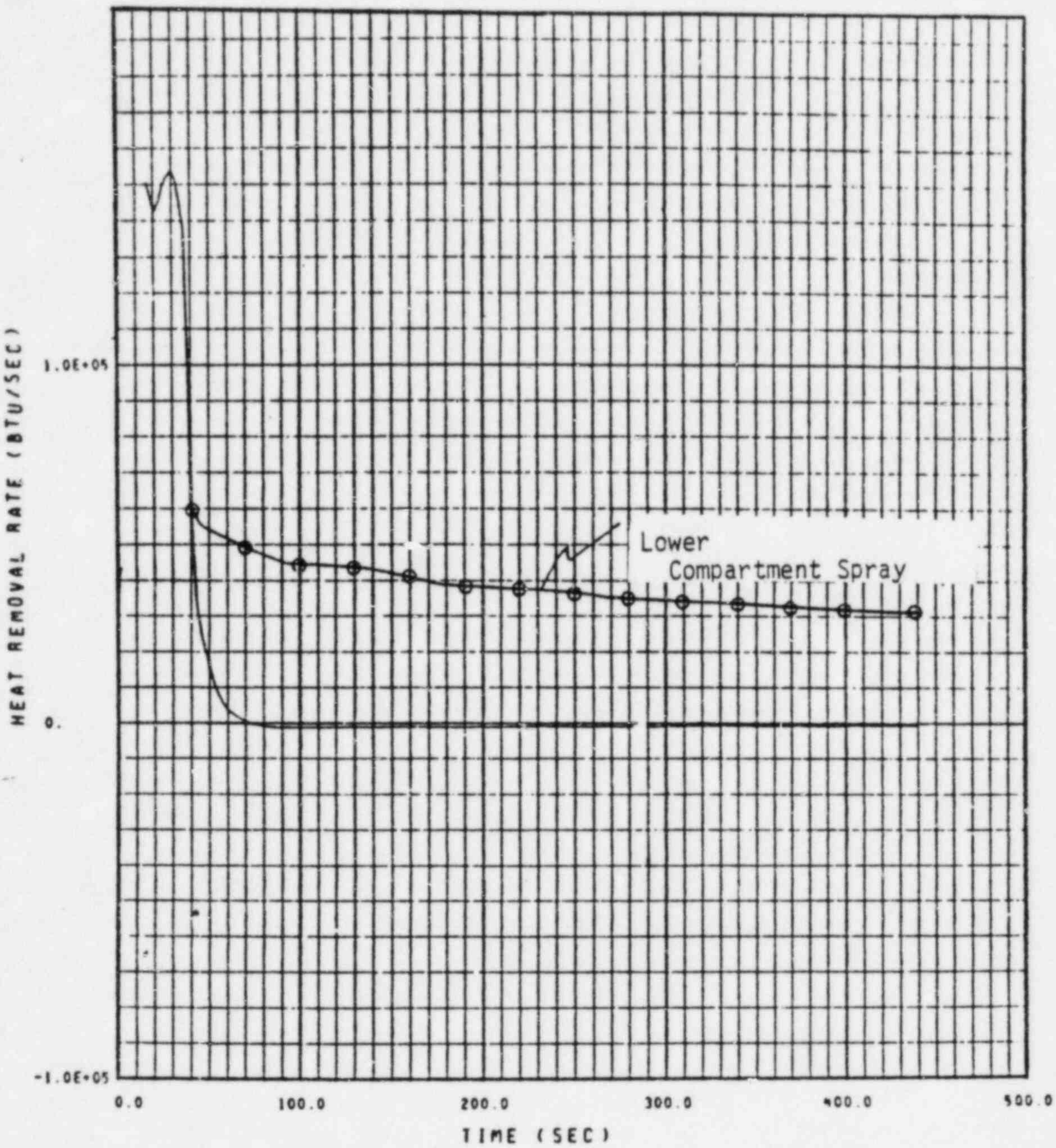
MAXIMUM SI

FIGURE 14.3.1-58 HEAT REMOVAL BY LC DRAIN



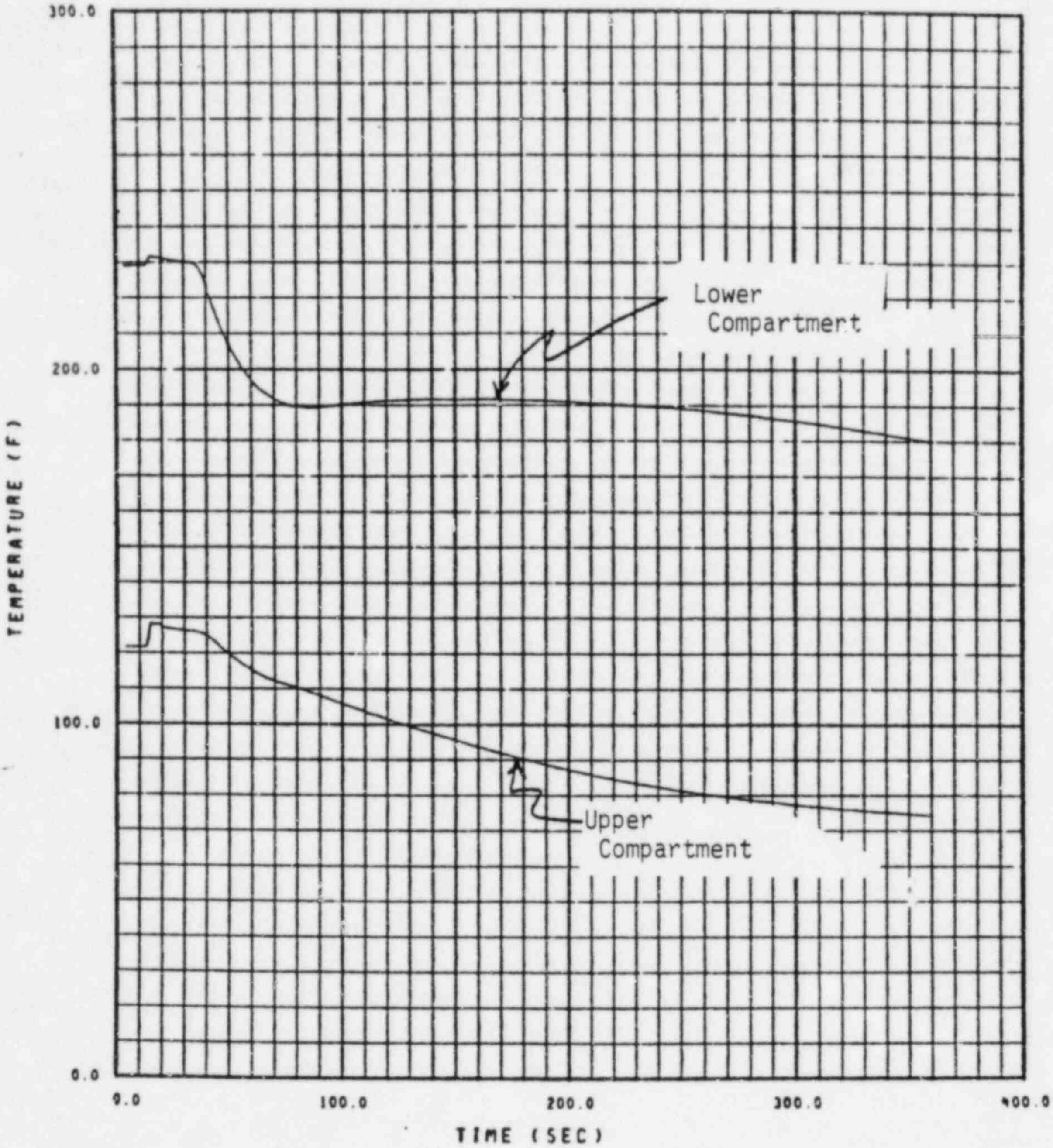
MINIMUM SI

FIGURE 14.3.1-59 HEAT REMOVAL BY SUMP AND LC SPRAYS



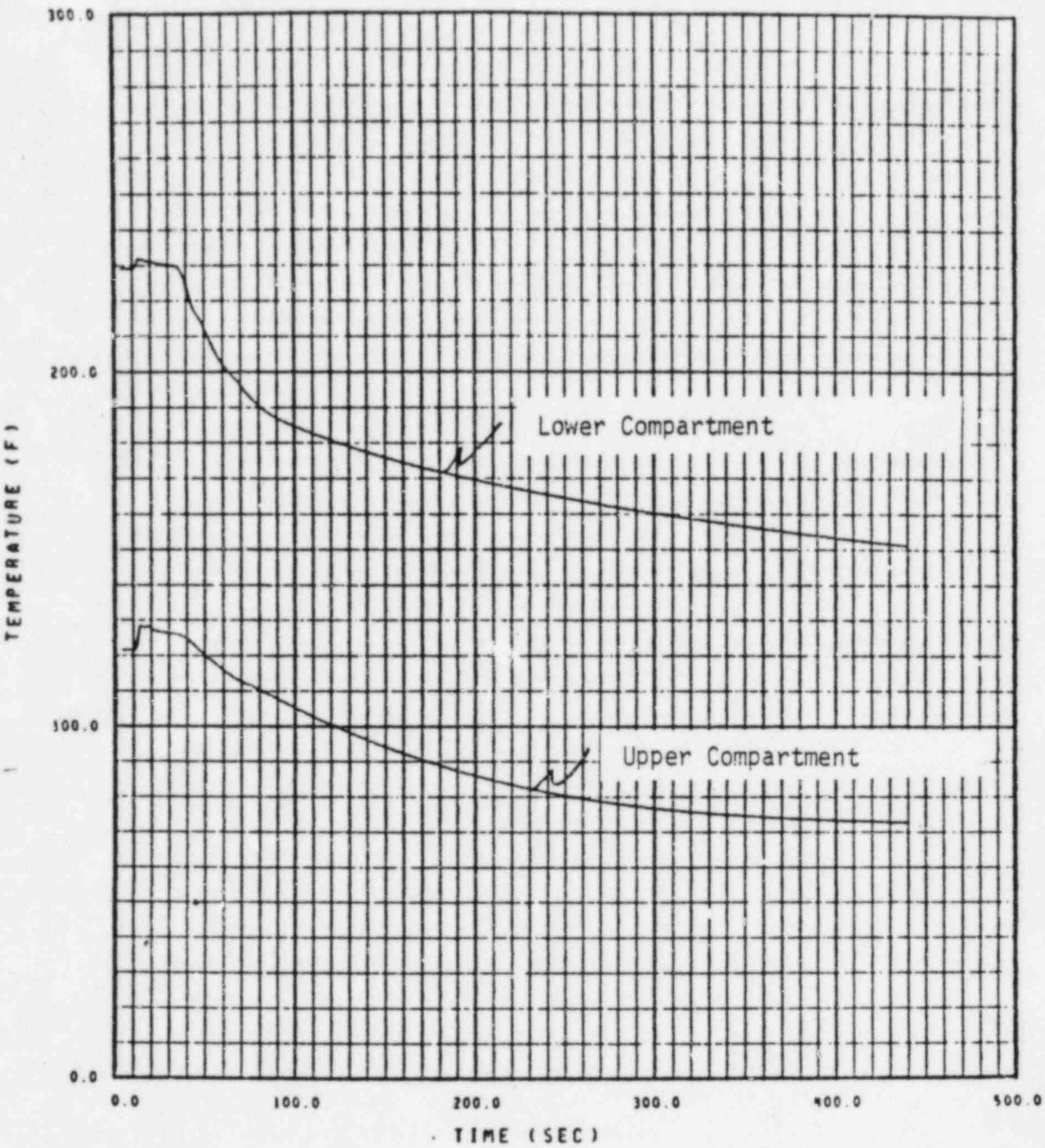
MAXIMUM SI

FIGURE 14.3.1-60 HEAT REMOVAL BY SUMP AND LC SPRAY



MINIMUM SI

FIGURE 14.3.1-61 COMPARTMENT TEMPERATURE



MAXIMUM SI

FIGURE 14.3.1-62 COMPARTMENT TEMPERATURE

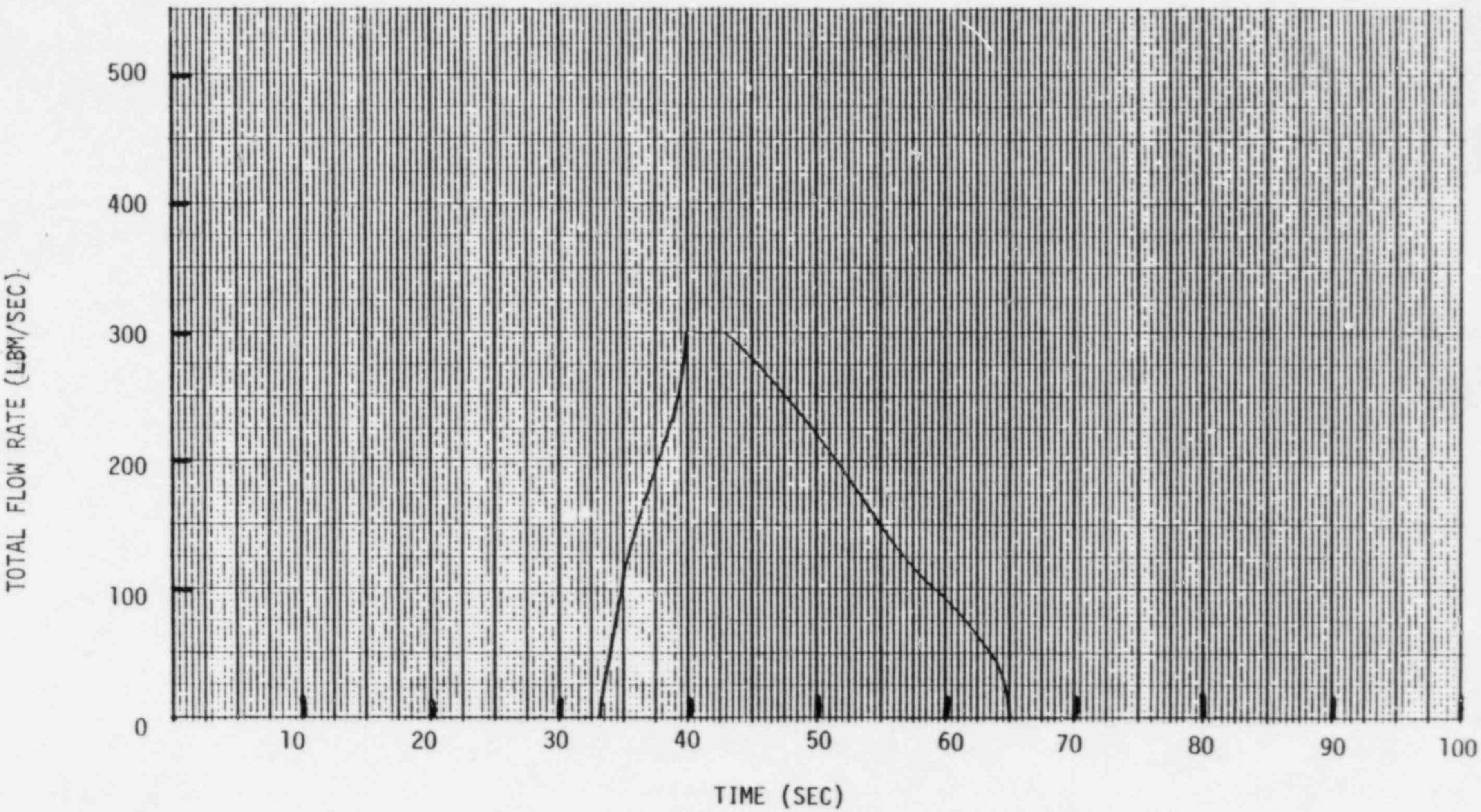


FIGURE 14.3.1-63 FLOW FROM THE UPPER TO LOWER COMPARTMENT
MIN. SI

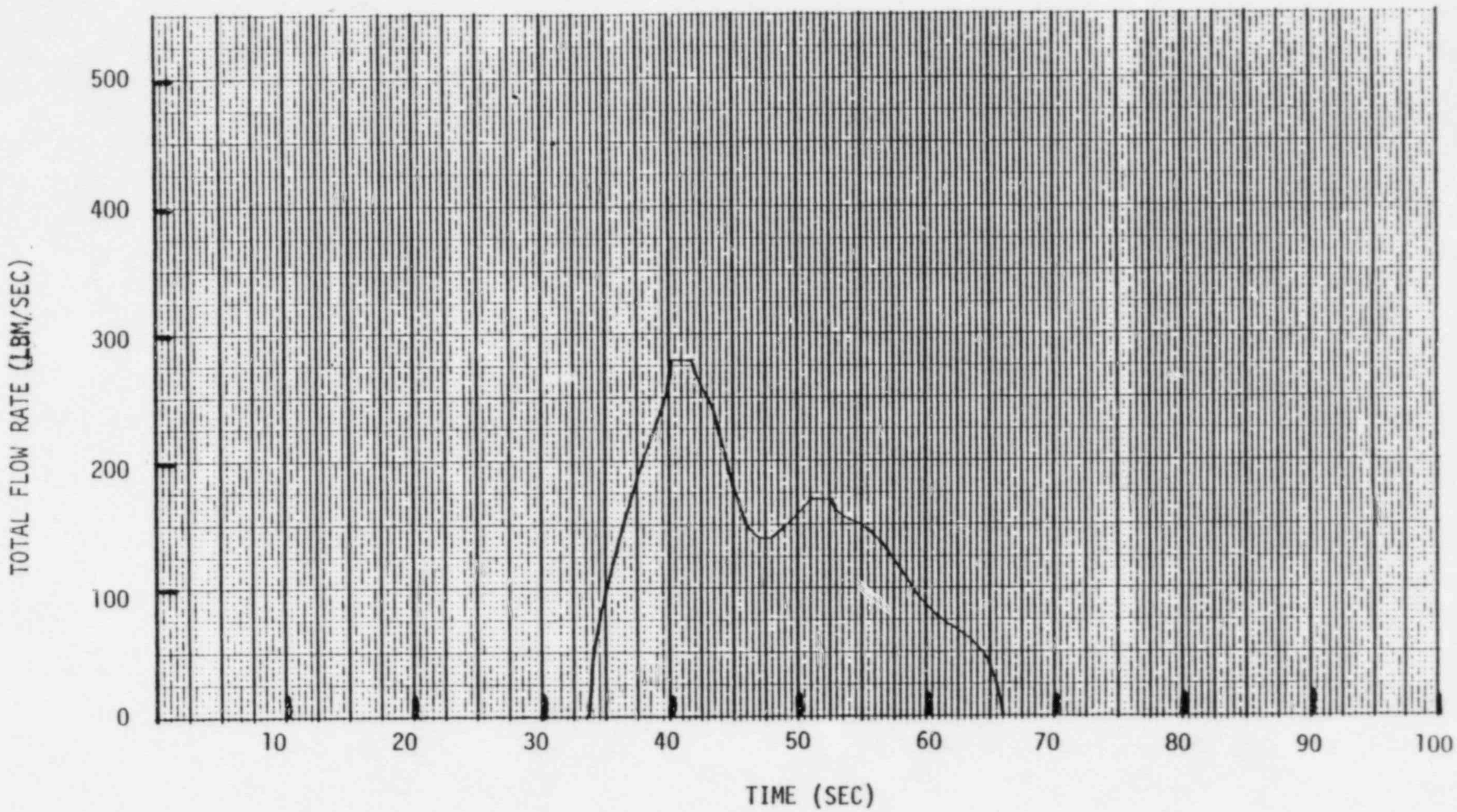


FIGURE 14.3.1-64 FLOW FROM THE UPPER TO LOWER COMPARTMENT

MAX. SI

Attachment E to AEP:NRC:0745C
Small Break LOCA Safety Analysis

14.3.2 Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

The analysis for small break loss of coolant accidents incorporating the criteria specified by 10 CFR 50.46[1] "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors" is presented in this section. The analytical techniques used are all in compliance with Appendix K of 10 CFR 50 and are described in the topical report, "Westinghouse ECCS Evaluation Model - Summary"[2].

14.3.2.1 Identification of Causes and Accident Description

A loss of coolant accident is defined as a rupture of the Reactor Coolant System piping or of any line connected to the system up to the first closed valve. Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the containment contains the fission products existing in it.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the Reactor Coolant System through the postulated break against the charging pump makeup flow at a normal Reactor Coolant System pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain the pressurizer level at 2250 psia for a break through a .375 inch diameter hole. This break results in a loss of approximately 17.5 lb/sec.

Should a larger break occur, depressurization of the Reactor Coolant System causes fluid to flow to the Reactor Coolantr System from the pressurizer, resulting in a pressure and level decrease in the presurizer. Reactor trip occurs when the low pressure trip or over-temperature ΔT setpoint is reached. The Safety Injection System is actuated when the appropriate setpoint is reached. Reactor trip and Safety Injection System actuation can also be initiated by a high containment pressure signal. The consequences of the accident are limited in two ways:

A. Reactor trip and borated water injection supplement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

B. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the Reactor Coolant System. The heat transfer between the Reactor Coolant System and the secondary system may be in either direction depending on the relative temperature. In the case of continued heat addition to the secondary, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting the motordriven auxiliary feedwater pumps. The secondary flow aids in the reduction of Reactor Coolant System pressure. When the RCS depressurizes to 600 psia, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the initiation of the accident and effect of pump coastdown are included in the blowdown analyses.

14.3.2.2 Analysis of Effects and Consequences

Method of Analysis

For small breaks (less than 1.0 ft^2) the WFLASH [3,8] digital computer code is employed to calculate the the transient depressurization of the Reactor Coolant System as well as to describe the mass and enthalpy of the flow through the break.

Small Break LOCA Analysis Using WFLASH

The WFLASH program used in the analysis of the small break loss-of-coolant accident is an extension of the FLASH-4[4] code developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the Reactor Coolant System.

The Reactor Coolant System is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicityly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of WFLASH is given in References 3 and 8.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubvble rise model to permit a transient mixture height calculation. The multi-node capability of the program allowes for an explicit and detailed spatial representation of various system components. In particular it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Safety injection flow rate to the Reactor Coolant System as a function of the system pressure is used as part of the input. The Safety Injection (SI) System was assumed to be delivering to the RCS 25 seconds after the generation of a safety injection signal.

For these analyses, the SI delivery considers pumped injection flow which is depicted in Figure 14.3.2-1 as a function of RCS pressure. This figure represents injection flow from the SI pumps based on performance curves degraded 5 percent from the design head. The 25 seconds delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. The effect of RHR pump flow is not considered here since their shutoff head is lower than RCS pressure during the time portion of the transient considered here. Also, minimum Safeguards Emergency Core Cooling System capability and operability has been assumed in these analyses.

Peak clad temperature analyses are performed with the LOCTA-IV[5,8] code which determines the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history. For this analysis, the Revised Pad Fuel Thermal Safety Model, described in WCAP-8720, addendum 2, generated the initial fuel rod condition input to LOCTA-IV.

14.3.2.3 Results

This section presents results of the limiting break size in terms of highest peak clad temperature. The selection of the three break sizes reported here is based on the extensive sensitivity studies reported in Reference 7. The worst break size (small break) is a 4-in. diameter break. The depressurization transient for this break is shown in figure 14.3.2-2. The extent to which the core is uncovered is shown in figure 14.3.2-3.

During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rod and clad to very near the coolant temperatures as long as the core remains covered by a two phase mixture.

The maximum hot spot clad temperature calculated during the transient is 1630°F including the effects of fuel densification as described in Reference 6. The peak clad temperature transient is shown in Figure 14.3.2-4 for the worst break size, i.e., the break with the highest peak clad temperature. The steam flow rate for the worst break is shown on Figure 14.3.2-5 When the mixture level drops below the top of the core, the steam generated in the lower region flows upward and provides cooling to the upper portion of the core. The hot rod film coefficient for this phase of the transient is given in Figure 14.3.2-6. The hot spot fluid temperature for the worst break is shown in Figure 14.3.2-7.

Figure 14.3.2-8 presents the hot rod power distribution utilized to perform the small break analysis presented here. This power shape was chosen because it provides an appropriate distribution of power versus core height, and also because local power is maximized in the upper regions of the reactor core (10 feet to 12 feet). This power shape is skewed to the top of the core with the peak local power occurring at the 10.0-foot core elevation. This is limiting for the small break analysis because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding tempertures in the lower elevation of the core, below the two-phase mixture height, remains low. The peak clad temperature occurs above 10 feet.

14.3.2.4 Conclusions

Analyses presented in this section show that the high head portion of the Emergency Core Cooling System, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperatures below required limits of 10 CFR 50.46. Hence, adequate protection is afforded by the Emergency Core Cooling System in the event of a small break loss-of-coolant accident.

Additional Break Sizes

Additional break sizes are analyzed. Figures 14.3.2-9 and -10 present the RCS pressure transient for the 3 and 6 inch breaks, respectively, and Figures 14.3.2-11 and -12 present the volume history (mixture height) plots for both breaks. The peak clad temperatures for both cases are less than the peak clad temperature of the 4 inch break. The peak clad temperatures for both cases are given in Figures 14.3.2-13 and -14. The hot spot fluid temperature and steam flow rate for the 3 and 6 inch breaks are shown in Figure 14.3.2-15 through Figure 14.3.2-18.

The time sequence of events for all breaks analyzed is shown in Table 14.3.2-1 and a summary of the results is shown in Table 14.3.2-2.

References, Section 14.3.2

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors" 10 CFR 50.46 and Appendix K of 10 CFR 50. Federal Register, Volume 39, Number 3, January 4, 1974.

2. "Westinghouse ECCS Evaluation Model-Summary, "WCAP-8339, Bordelon, F. M., Massie, H. W., and Zordan, T. A., July 1974, WCAP-8341 (Proprietary), June 1974.

3. Esposito, V. J., Kesaven, K., Maul, B. A., "WFLASH-A FORTRAN IV, Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8261 Rev. 1, July 1974, WCAP-8200, (Proprietary), June 1974.

4. Porsching, T. A., Murphy, J. H., Redfield, J. A., and Davis, V. C., "FLASH-4: A Fully Implicit FORTRAN-IV Program for the Digital Simulation of Transients in a Reactor Plant, "WAPD-TM-84; Bettis Atomic Power Laboratory (March, 1969).

5. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis, "WCAP-8305, June 1974, WCAP-8301, (Proprietary), June 1974.

6. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, June 1974, WCAP-8302, (Proprietary), June 1974.

7. "Report on Small Break Accidents for Westinghouse NSSS System," Vols. I to III, WCAP-9600,.June 1979.

8. Skwarek, R. J., Johnson, W. J., and Meyer, P. E., 1977. "Westinghouse Emergency Core Cooling System Small Break," October 1975, Model. WCAP-8970-P-A (Proprietary) and WCAP-8971-P-A (Non-Proprietary).

TABLE 14.2.2-1

SMALL BREAK

TIME SEQUENCE OF EVENTS (SEC)

EVENT	TIME 3 INCH	 4 INCH	 6 INCH
START	0.0	0.0	0.0
REACTOR TRIP SIGNAL	27.3	17.5	11.25
TOP OF CORE UNCOVERY	809.0	411.0	167.0
ACCUMULATOR INJECTION BEGINS	1830.0	800.0	321.0
PEAK CLAD TEMPERATURE OCCURS	1532.0	824.0	330.0
TOP OF CORE COVERED	1830.0	1250.0	343.0

TABLE 14.3.2-2

SMALL BREAK RESULTS

RESULTS	3 INCH	4 INCH	6 INCH
PEAK CLAD TEMPERATURE (°F)	1237	1630	1598
PEAK CLAD TEMPERATURE LOCATION (FT)	12.0	12.0	11.25
LOCAL Zr/H_2O REACTION, MAXIMUM (%)	0.11	0.66	0.40
LOCAL Zr/H_2O LOCATION (FT)	12.0	11.75	11.0
TOTAL Zr/H_2O REACTION (%)	<0.3	<0.3	<0.3
HOT ROD BURST TIME (SEC)	---	---	---
HOT ROD BURST LOCATION (FT)	---	---	---

CALCULATION

NSSS Power MWt 102% of	3411
Peak Linear Power kw/ft 102% of	15.50
Hot Rod Power Distribution (kw/ft)	See Figure 14.3.2-8
Accumulator Water Volume, cu. ft.	950

Fuel Region + Cycle Analyzed	Cycle	Region
UNIT 1	8	W Fuel

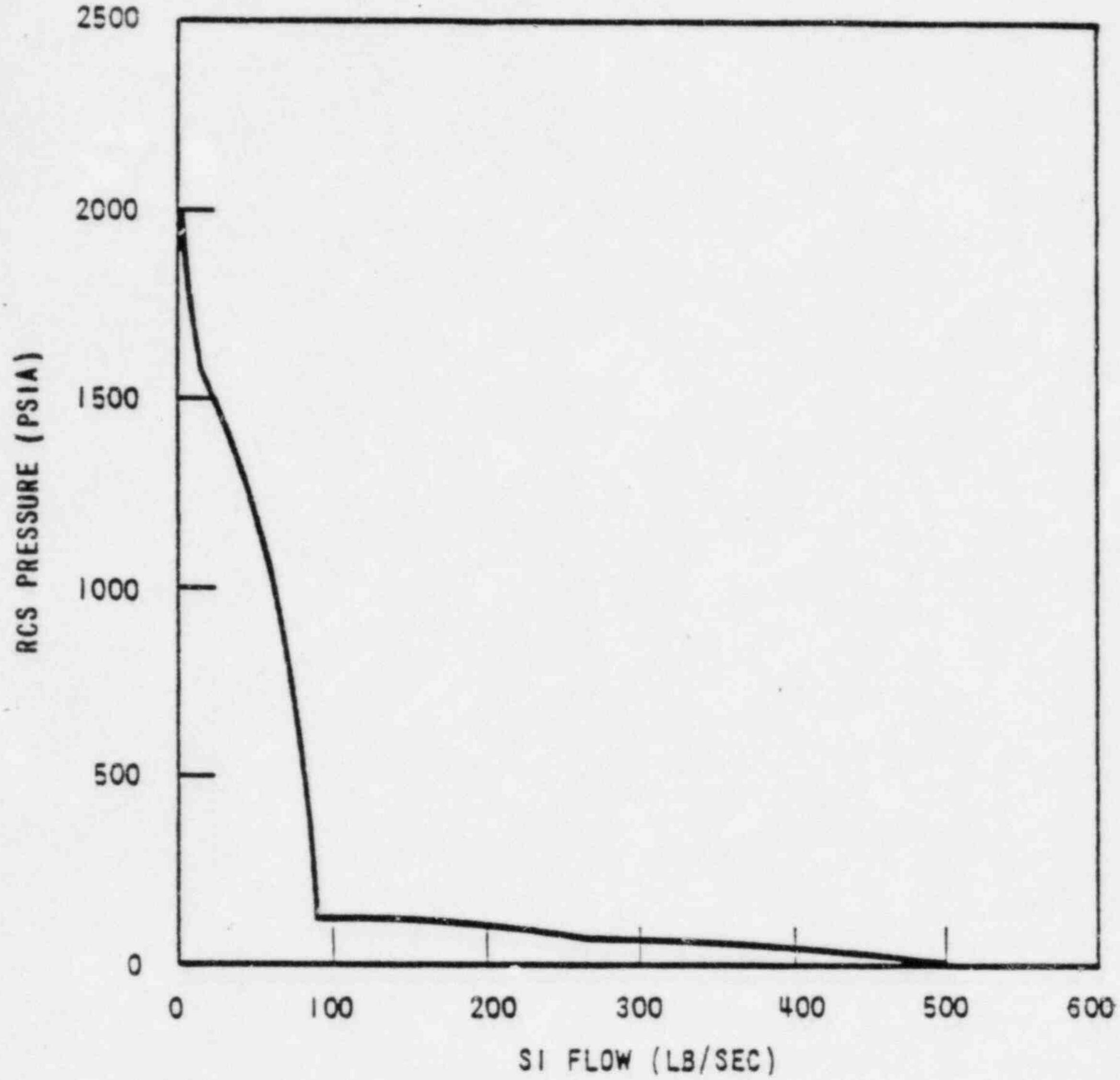


Figure 14.3.2-1 Safety Injection Flowrate

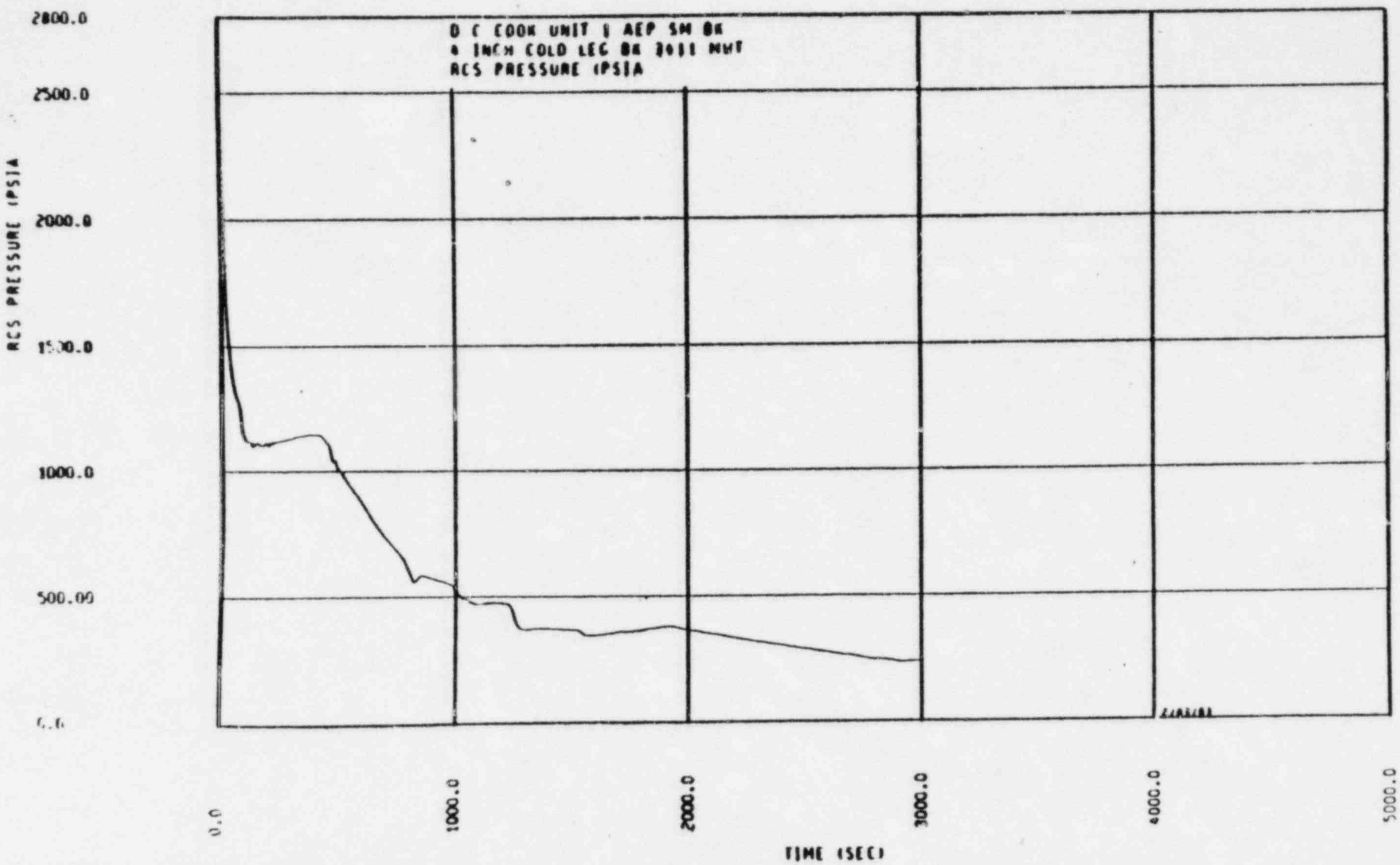


FIGURE 14.3.2-2 RCS PRESSURE - 4 IN. BREAK

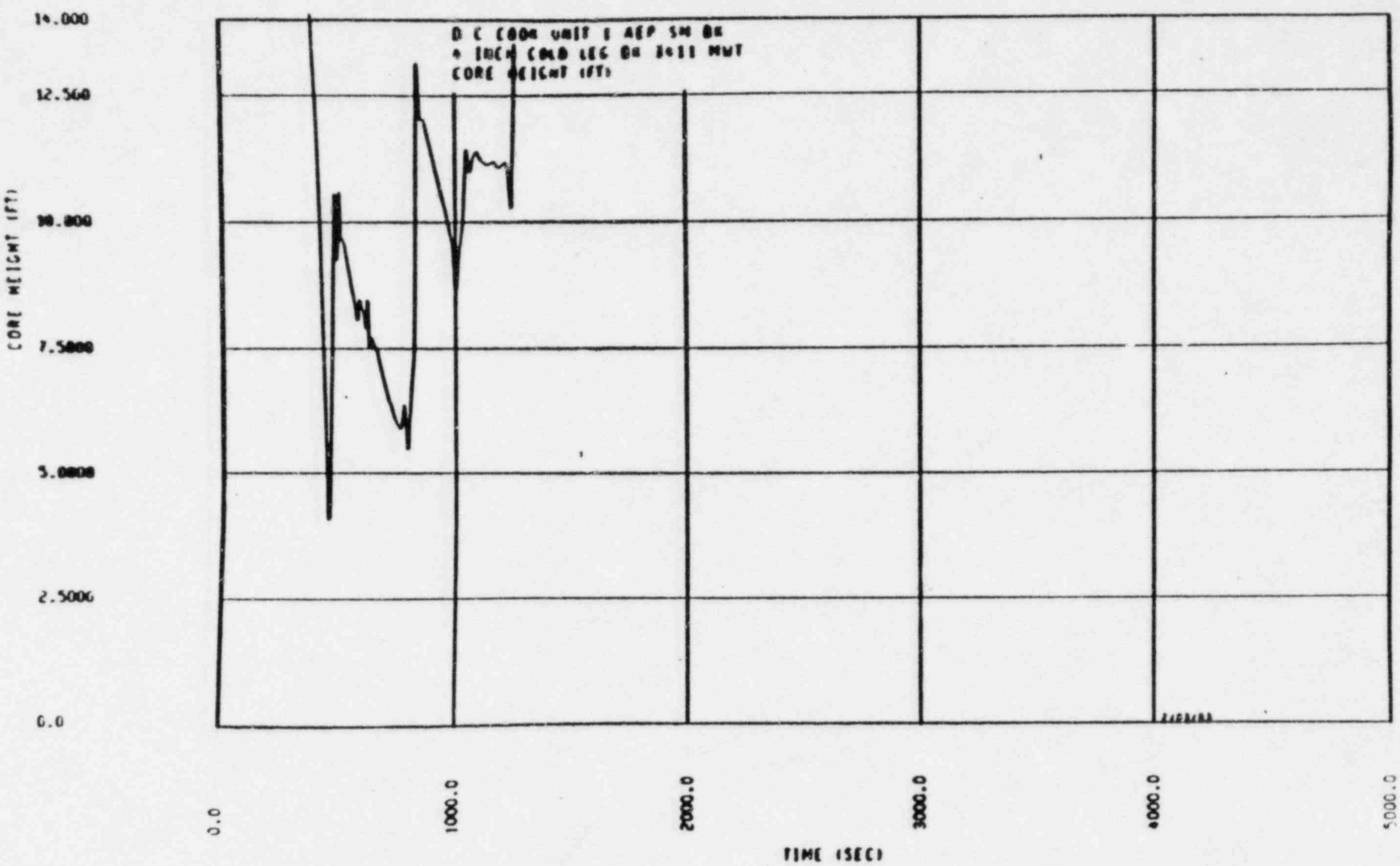


FIGURE 14.3.2-3 CORE MIXTURE HEIGHT - 4 IN. BREAK

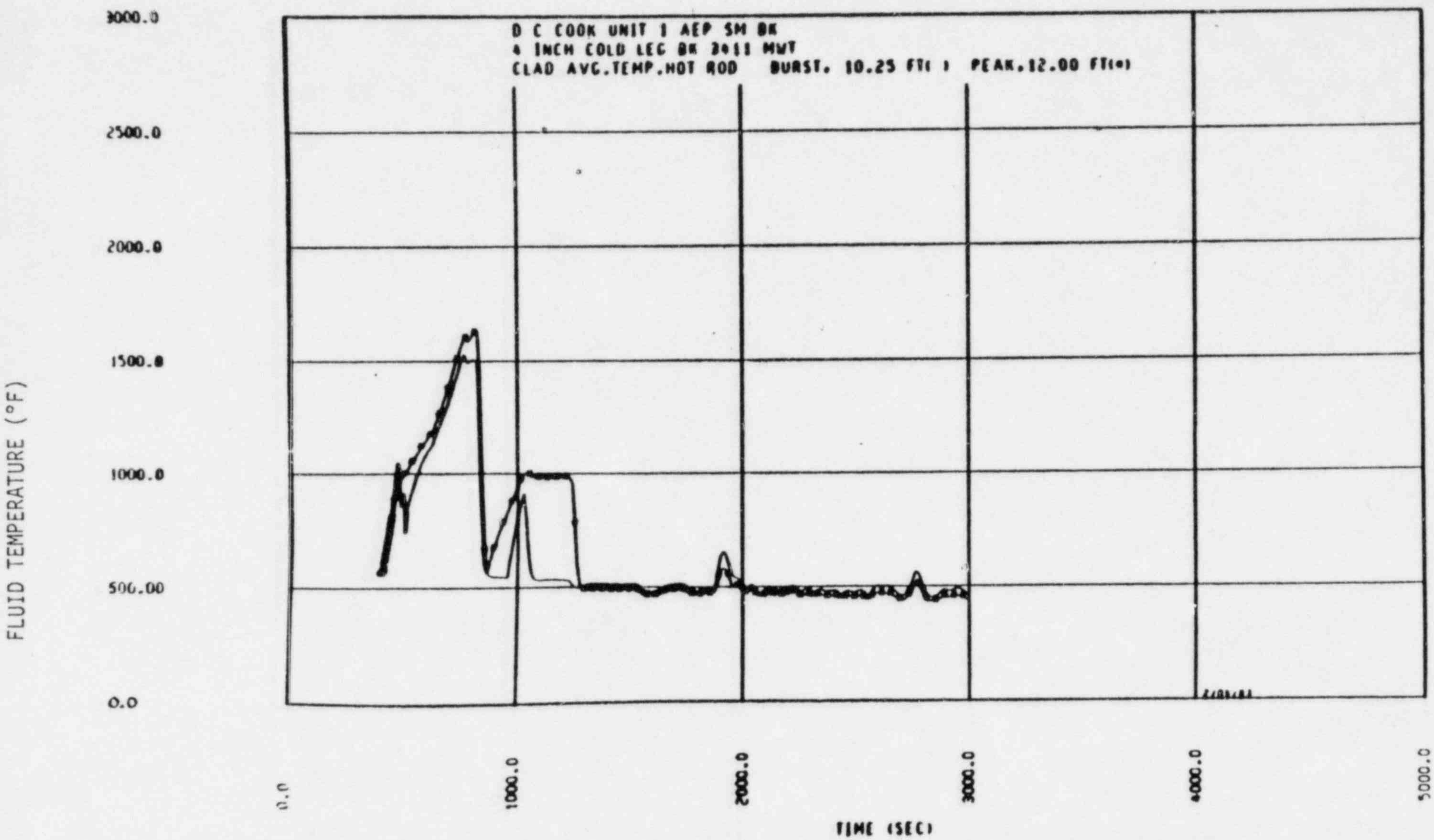


FIGURE 14.3.2-4 HOT SPOT CLAD TEMPERATURE - 4 IN. BREAK

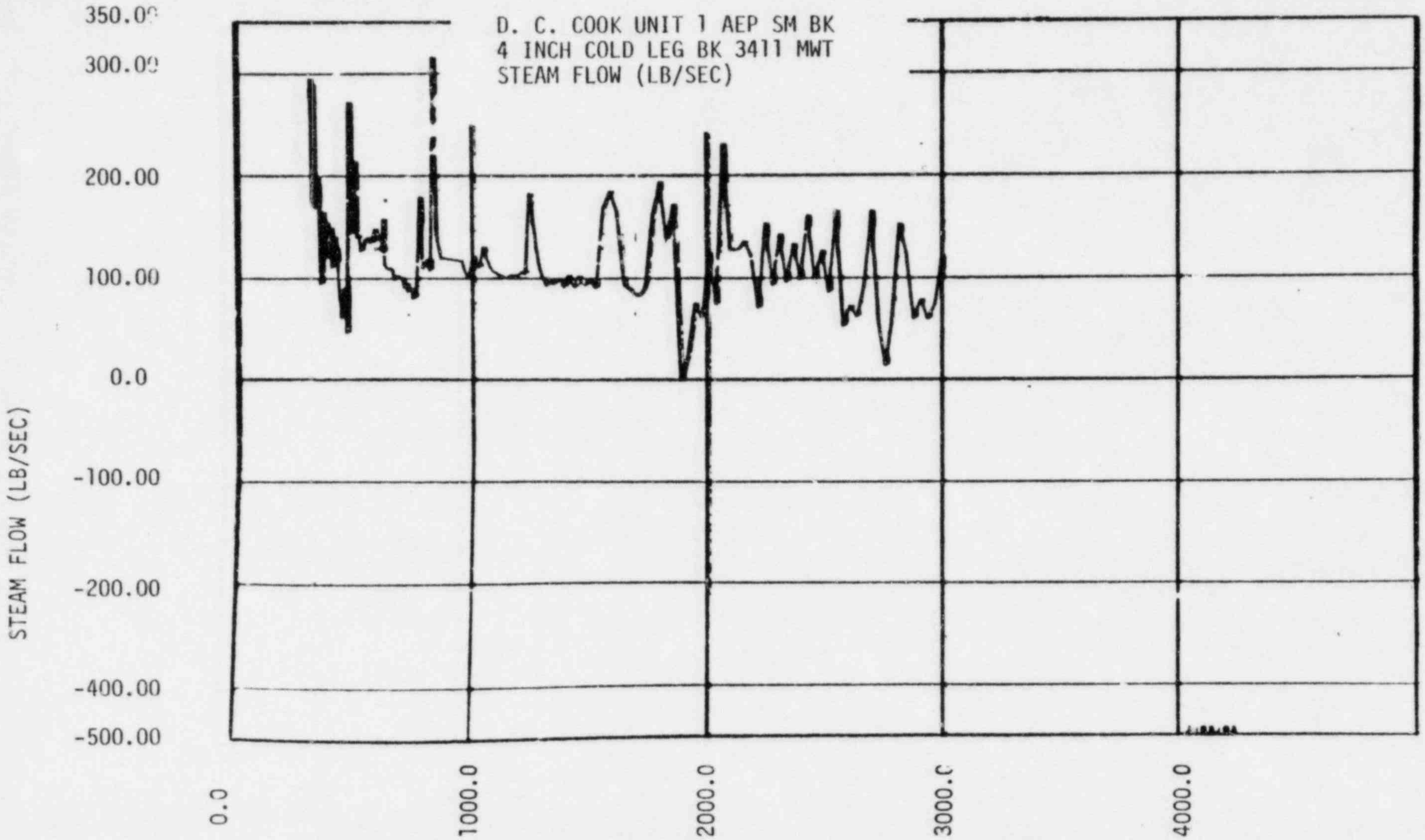


FIGURE 14.3.2-5 CORE STEAM FLOW RATE - 4 IN. BREAK

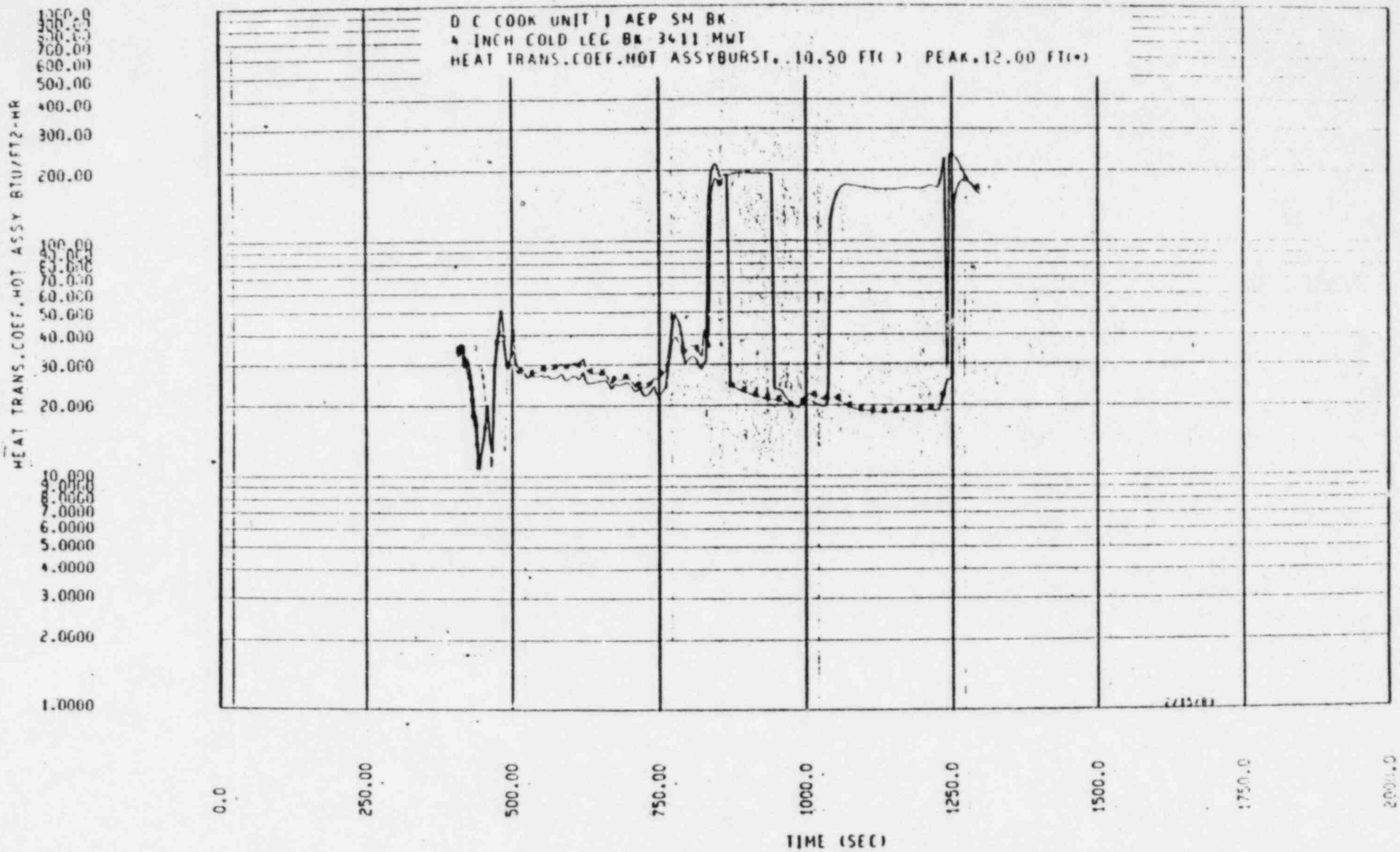


FIGURE 14.3.2-6 HOT SPOT HEAT TRANSFER COEFFICIENT - 4 IN. BREAK

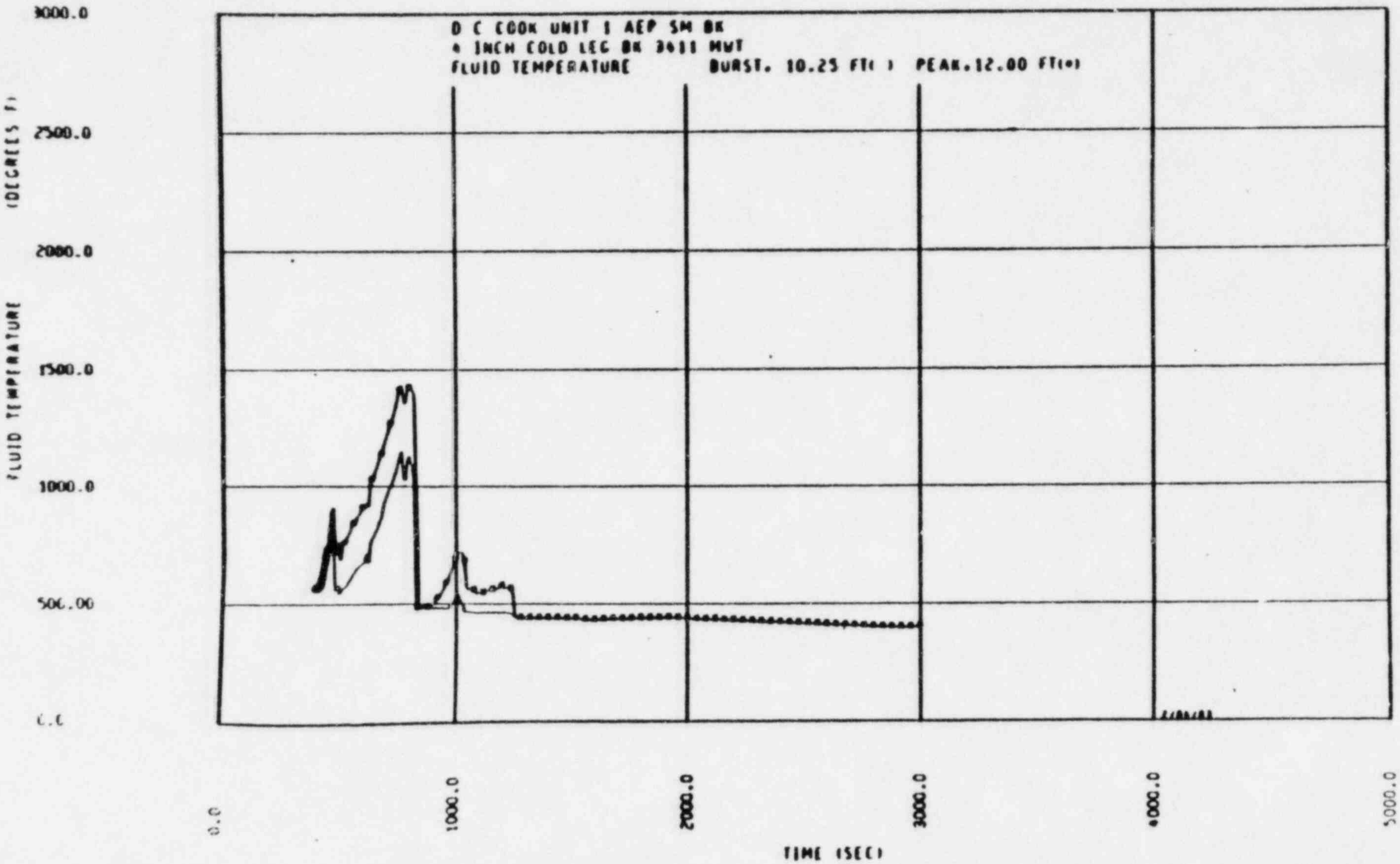


FIGURE 14.3.2-7 HOT SPOT FLUID TEMPERATURE - 4 IN. BREAK

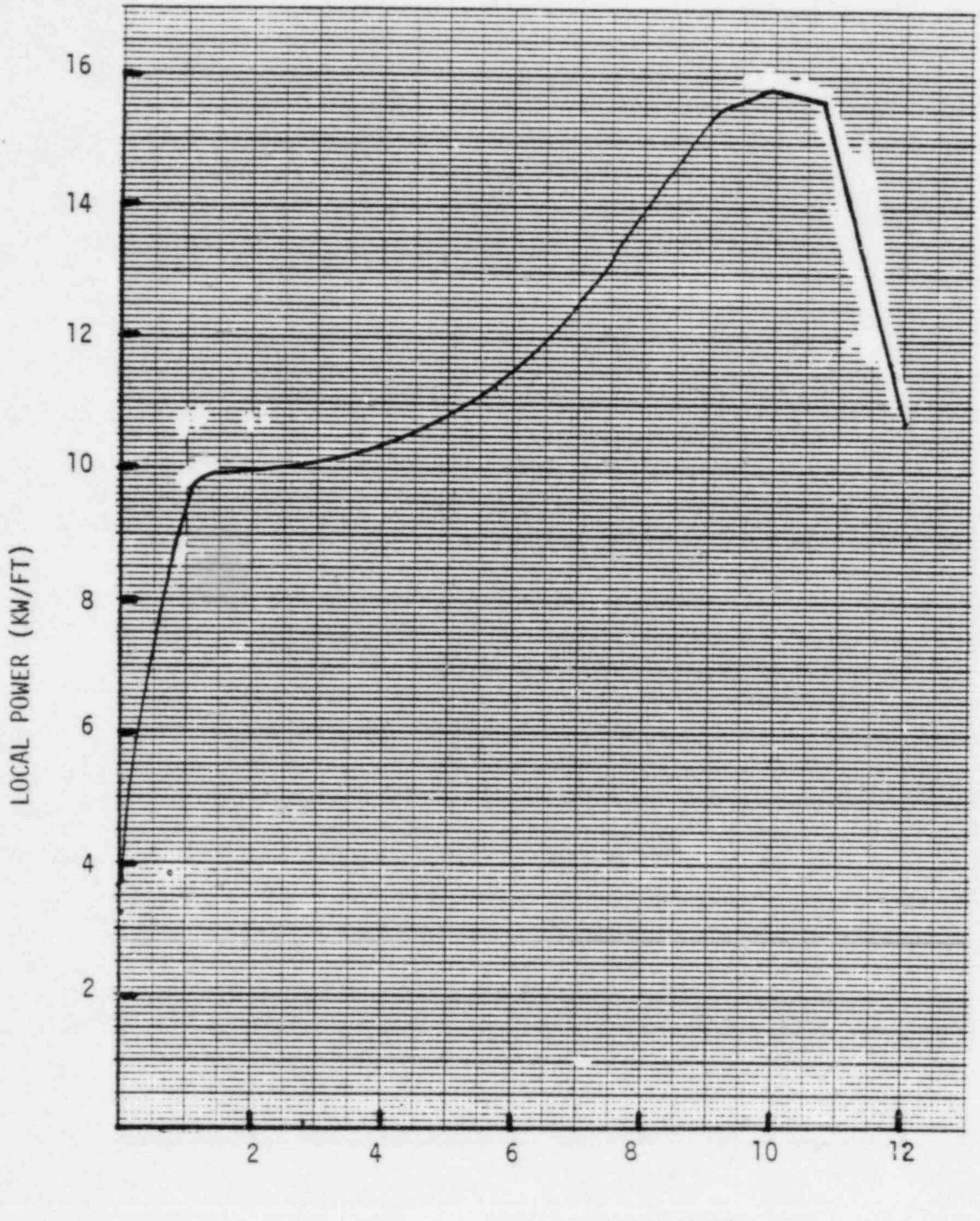


FIGURE 14.3.2-8 HOT ROD POWER DISTRIBUTION

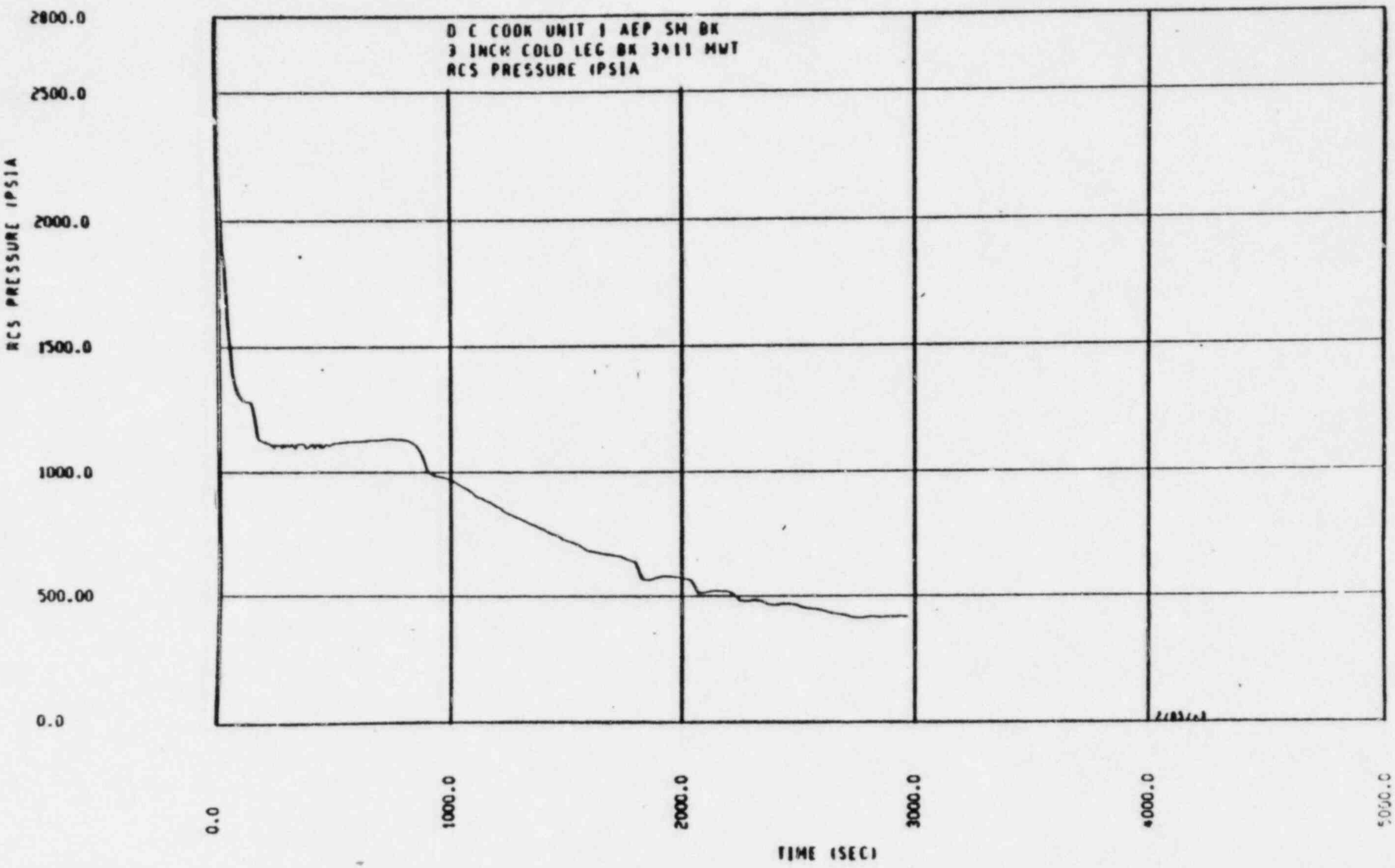


FIGURE 14.3.2-9 RCS PRESSURE - 3 IN BREAK

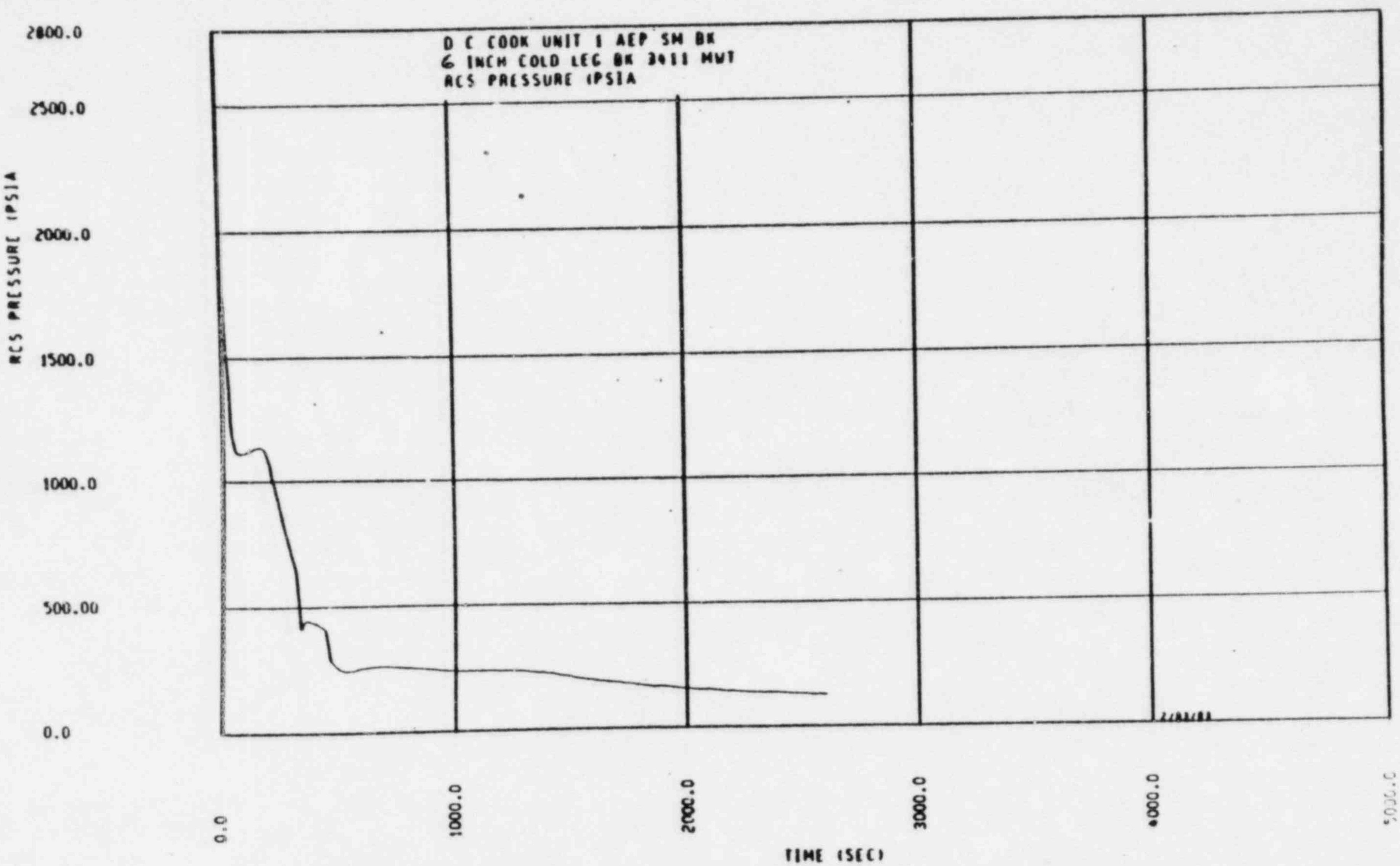


FIGURE 14.3.2-10 RCS PRESSURE - 6 IN. BREAK

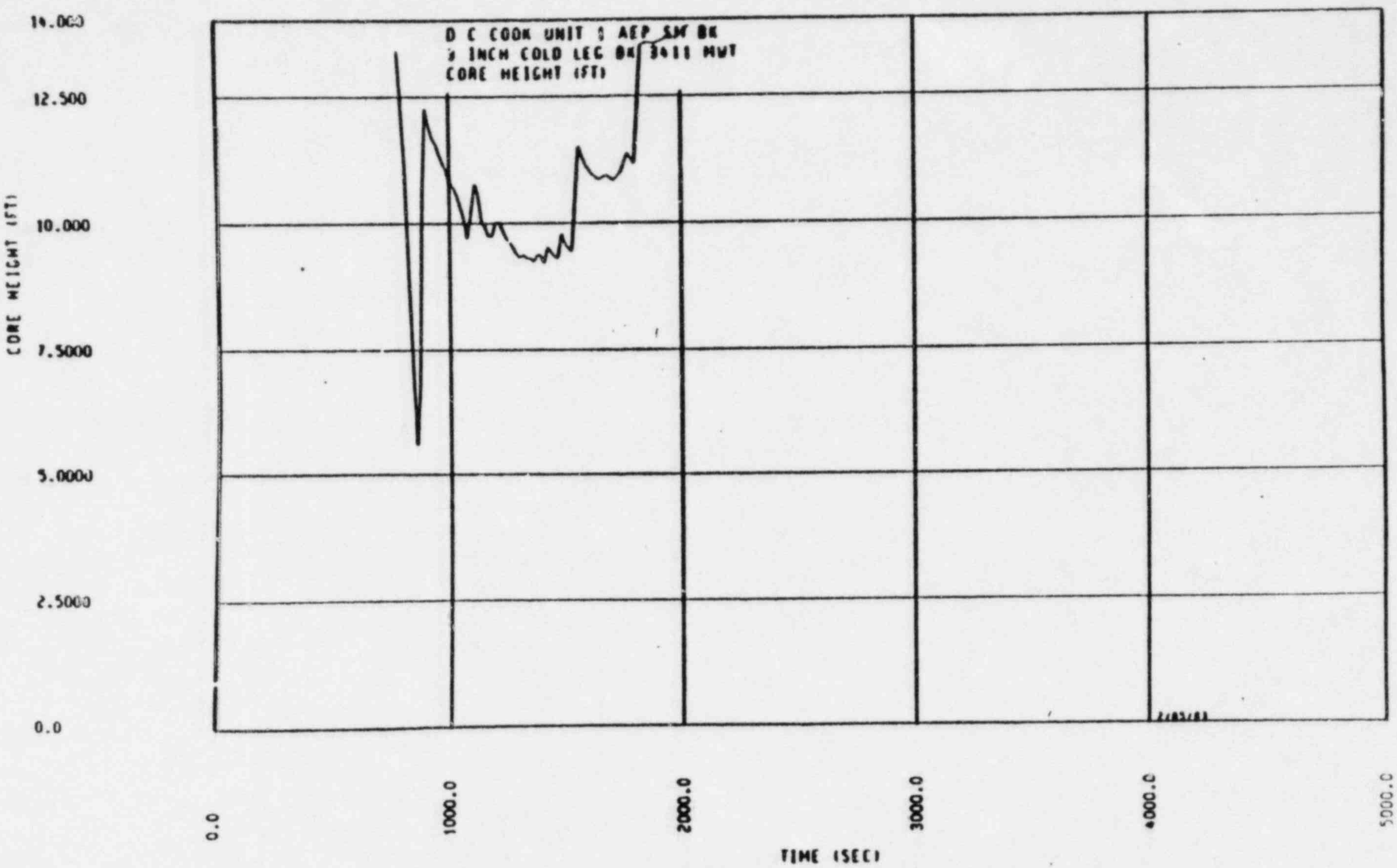


FIGURE 14.3.2-11 CORE MIXTURE HEIGHT - 3 IN. BREAK

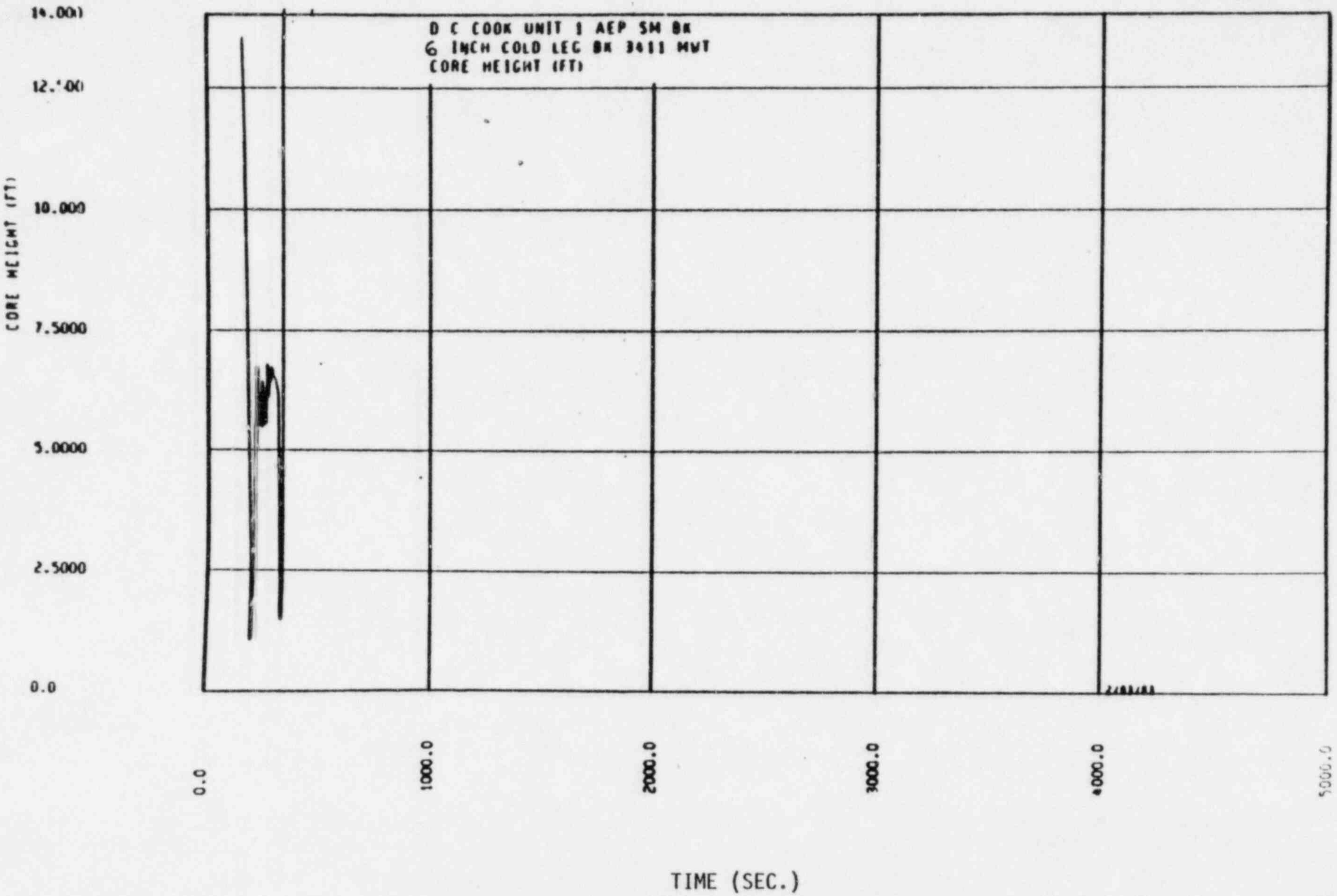


FIGURE 14.3.2-12 CORE MIXTURE HEIGHT - 6 IN. BREAK

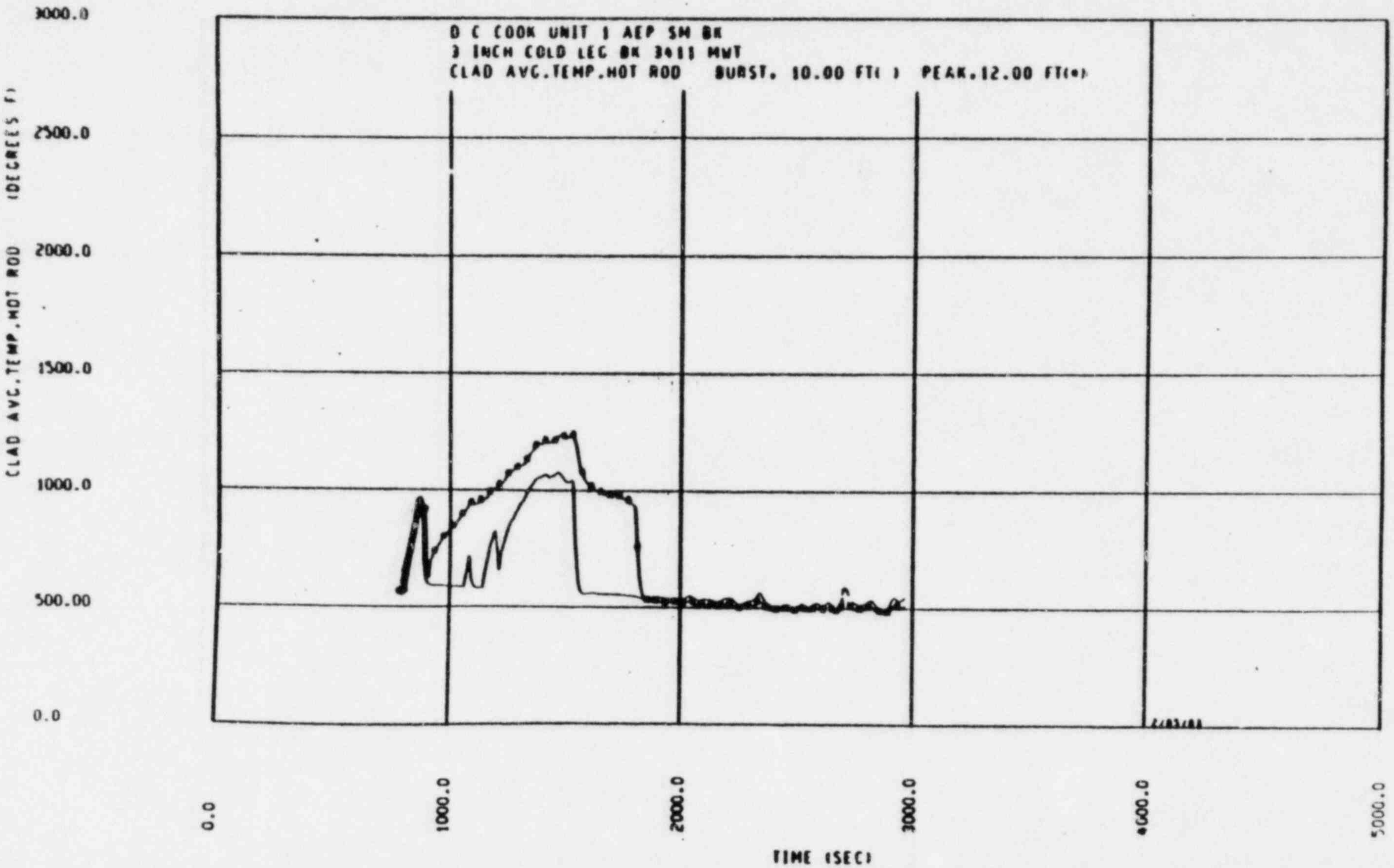


FIGURE 14.3.2-13 HOT SPOT CLAD TEMPERATURE - 3 IN. BREAK

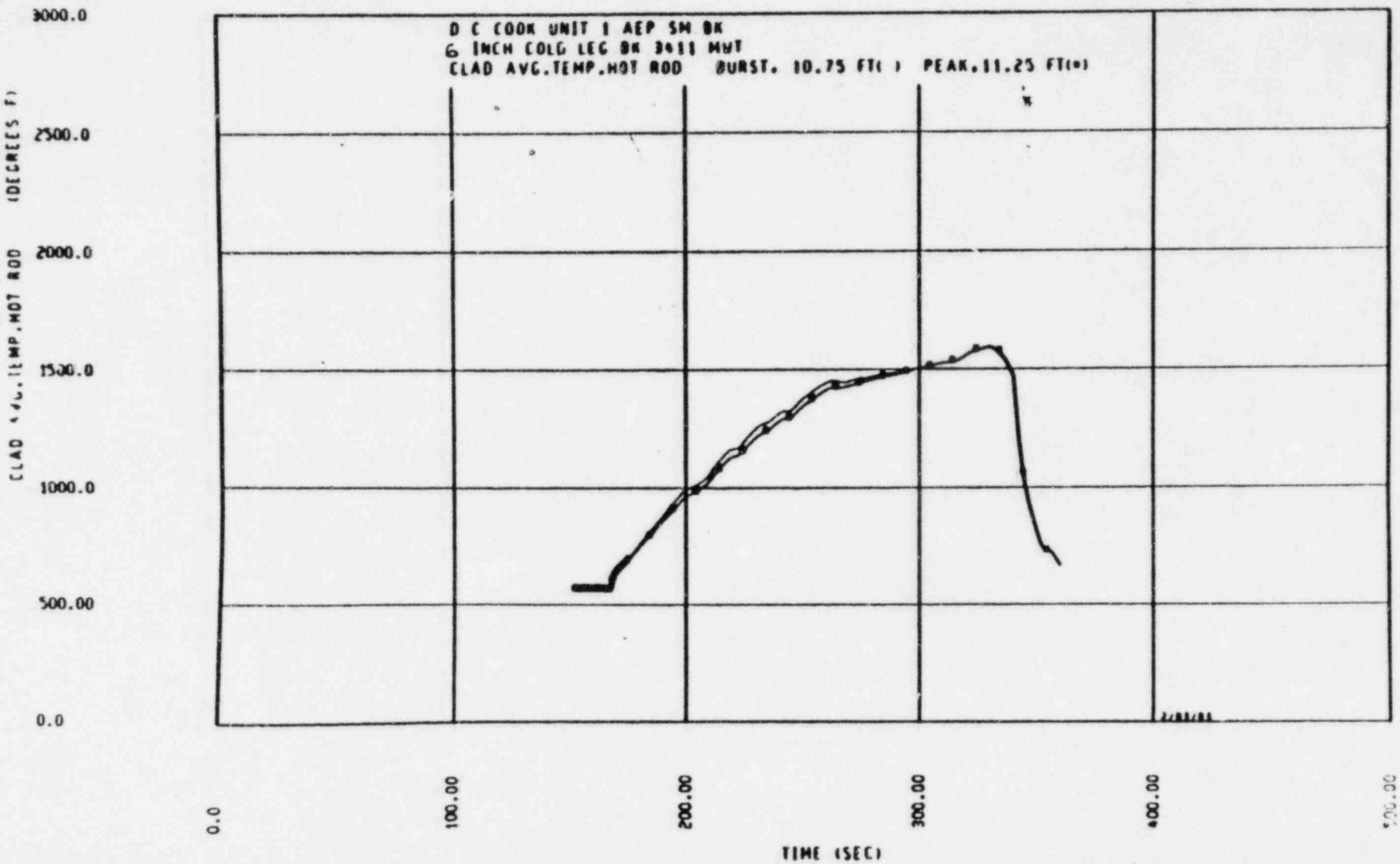


FIGURE 14.3.2-14 HOT SPOT CLAD TEMPERATURE - 6 IN. BREAK

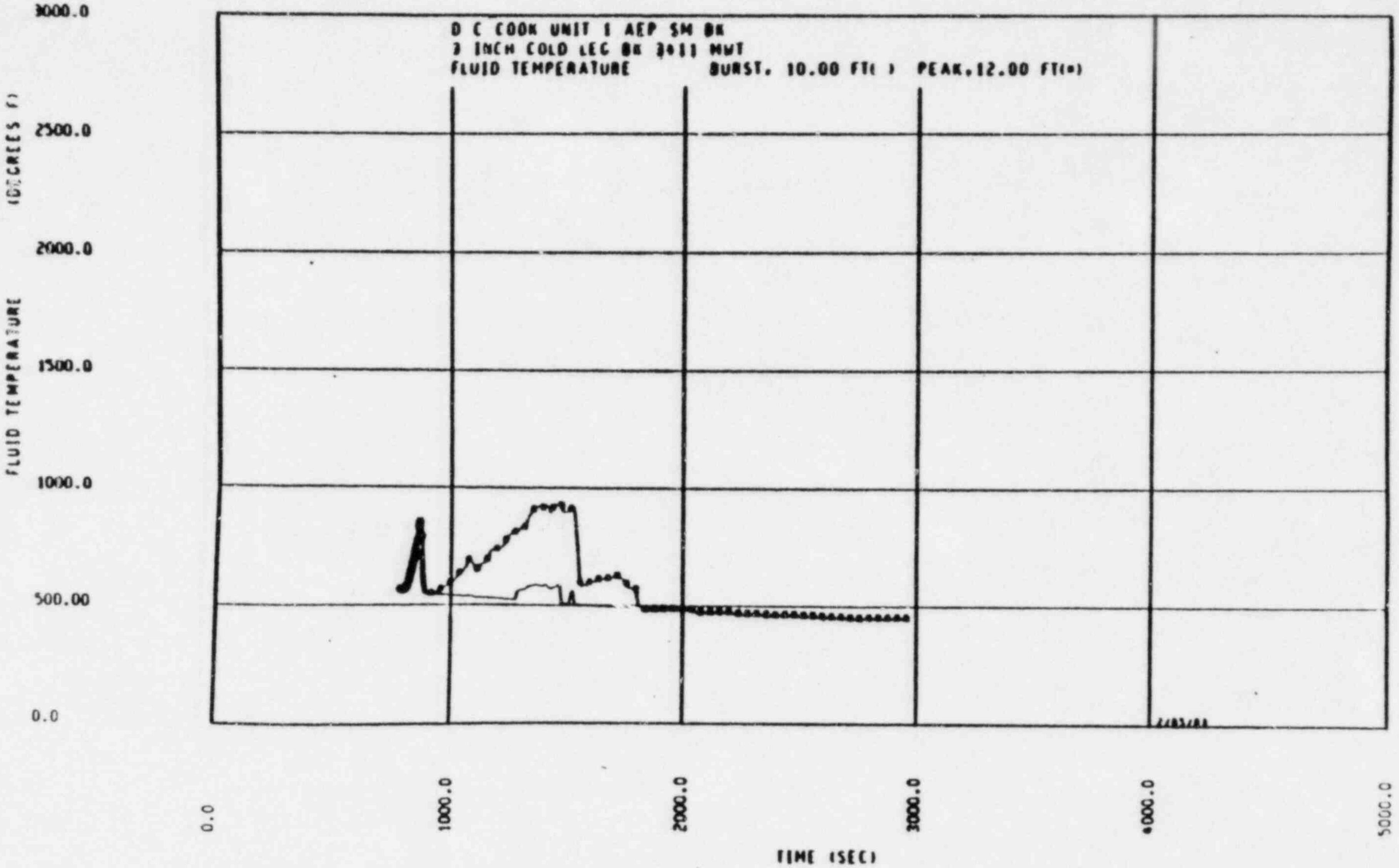


FIGURE 14.3.2-15 HOT SPOT FLUID TEMPERATURE - 3 IN. BREAK

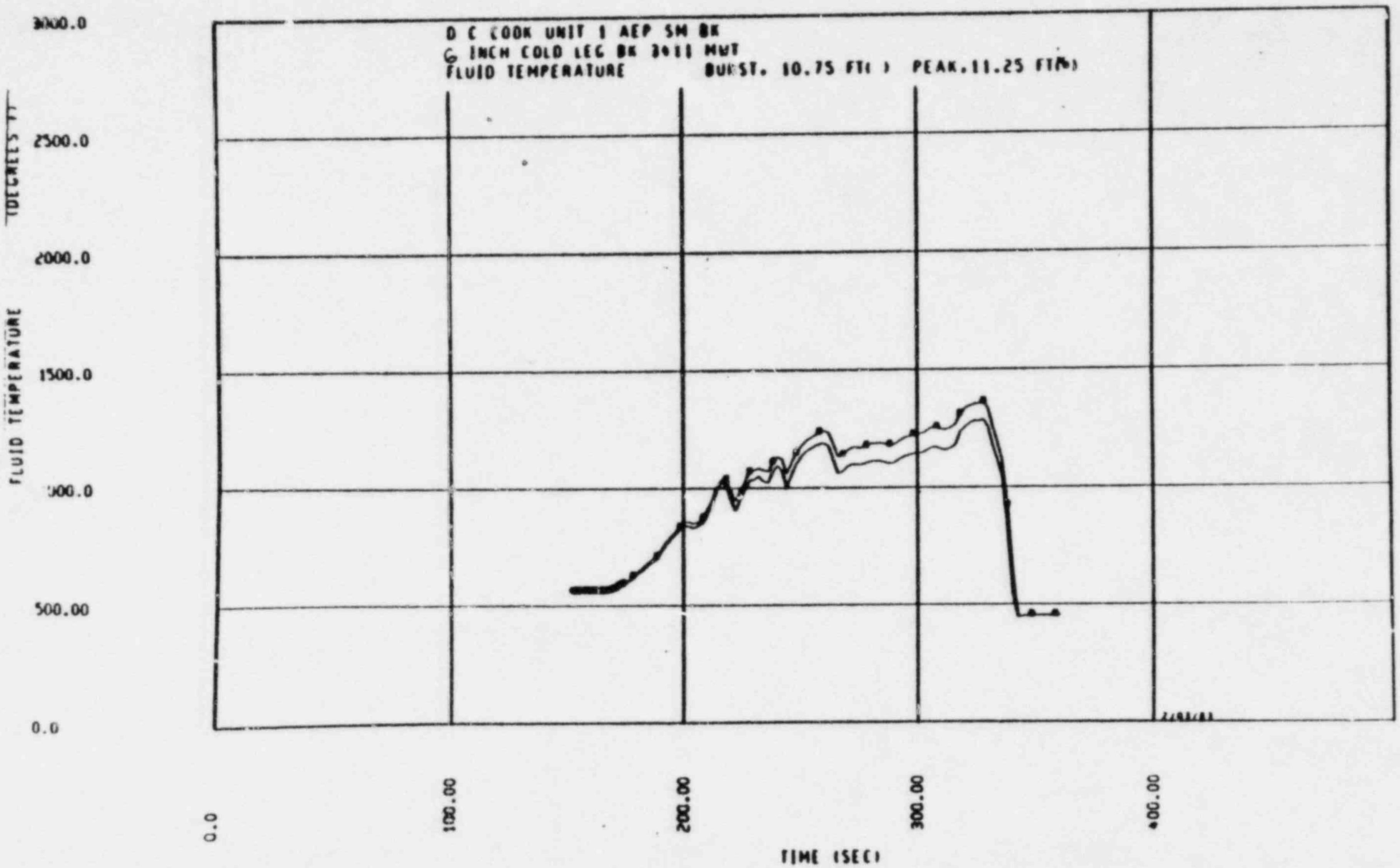


FIGURE 14.3.2-16 HOT SPOT FLUID TEMPERATURE - 6 IN. BREAK

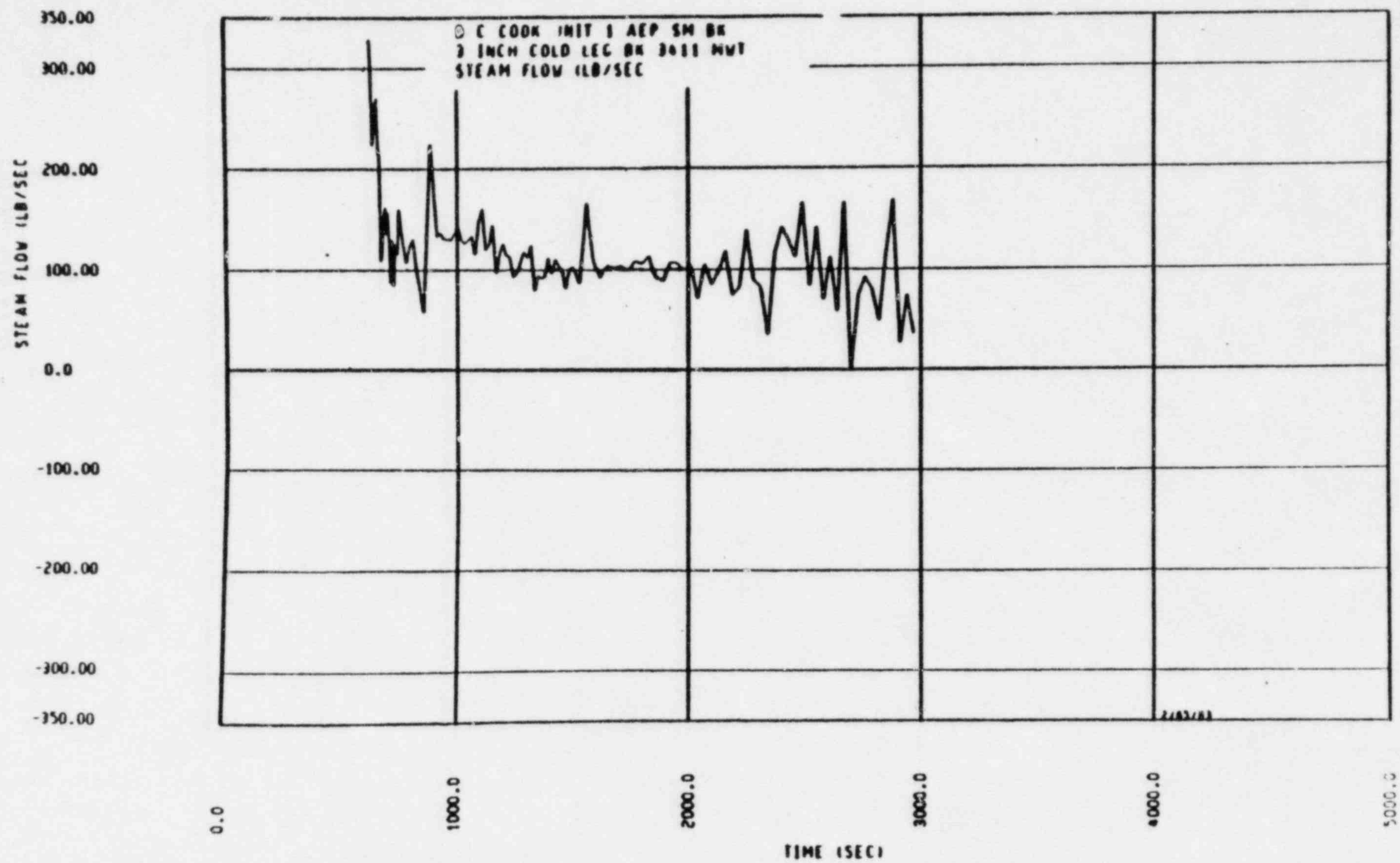


FIGURE 14.3.2-17 CORE STEAM FLOW RATE - 3 IN. BREAK

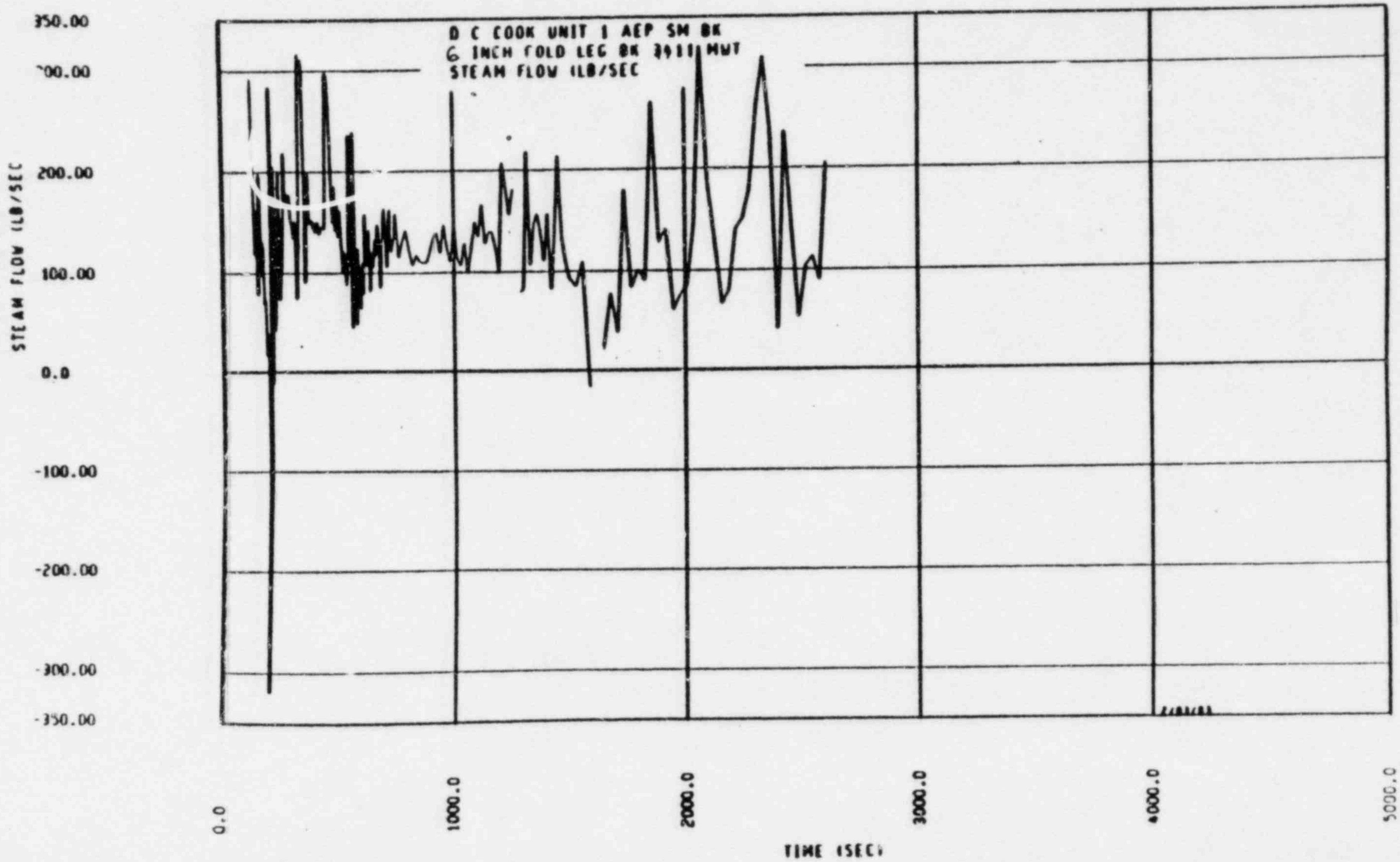


FIGURE 14.3.2-18 CORE STEAM FLOW RATE - 6 IN. BREAK

Attachment F to AEP:NRC:0745C
Description of Proposed Technical Specifications

JUSTIFICATION FOR D. C. COOK UNIT 1 (CYCLE 8) TECHNICAL SPECIFICATION CHANGES SUMMARY

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
1-5(a)	1.27	Introduces concept of DESIGN THERMAL POWER (DTP)	Necessary to take advantage of OTΔT margin gained by performing accident analysis at 3411 MWt core power.
2-2	Figure 2.1-1	Revised reactor core safety limits	New limits are based on ITDP at Design Thermal Power. These limits provide protection for DNB and exit boiling. The shape of the lines is consistent with current technical specifications. Limits are different, due to convoluting the plant uncertainties directly into the limit DNBR value.
2-5	Table 2.2-1	Design flow is 91,600 gpm	Reflects use of ITDP.
2-7; 2-8; 2-9	Table 2.2-1	Definitions of ΔT_0, T', and T" are modified. Values of K_1, K_2, K_3, K_4 and K_6 are revised, and provisions of i, ii, and iii are modified	Change in setpoints reflects use of ITDP at design thermal power. The f(ΔI) reset function changed due to the DNB correlations used. The most restrictive axial offsets from the WRB-1 and W-3 correlations are used. With the WRB-1 correlation, top peak shapes (positive axial offsets) are penalized, thus resulting in a shift towards the negative side.
B 2-1	2.1.1 (Bases)	Deleted mention of W-3 DNB correlation; Discussed DNBR basis when using the Improved Thermal Design Procedure (ITDP)	The DNB correlations for each fuel type are specifically mentioned on page B 2-1 (a). These changes were made due to the use of the Improved Thermal Design Procedure (ITDP).

JUSTIFICATION FOR D. C. COOK UNIT 1 (CYCLE 8) TECHNICAL SPECIFICATION CHANGES SUMMARY

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
B 2-1(a)	2.1.1 (Bases)	Added table with limit DNBR values and appropriate DNB correlations for each fuel type. Revised "DNBR limit of 1.30" to "applicable design limit DNBR."	These changes were made due to the use of ITDP.
B 2-2; B 2-2(a)	2.1.1 (Bases)	Revised $F^N_{\Delta H}$ values for Westinghouse and Exxon Nuclear Company fuels.	$F^N_{\Delta H}$ values were revised, in order to define the different $F^N_{\Delta H}$ values for each fuel type.
B 2-4	2.2.1 (Bases)	Revised power range negative rate trip description; revised "DNBR limit of 1.30" to "applicable design limit DNBR."	The power range negative rate trip is to provide protection in the event of a dropped rod. This revision reflects the protection provided and the Westinghouse solution to the Dropped Rod issue as presented to the NRC. Note that the administrative restriction on the D bank applies until NRC approval removes it. These changes were made, due to the use of the Improved Thermal Design Procedure (ITDP).
B 2-6	2.2.1 (Bases)	For loss of flow description, revised "DNBR limit of 1.30" to "applicable safety analysis design limit DNBR for each fuel type."	These changes were made due to the use of ITDP.
3/4 1-1	3.1.1.1 4.1.1.1.1	Change in shutdown margin from 1.75% to 1.60% $\Delta k/k$	A reduction in the required shutdown margin to 1.60% (value for most W 4-loop plants) from 1.75% would make the shutdown margin easier to meet for all cycles. Cycle 8 design has been performed using 1.60%.
3/4 1-21	3.1.3.3	Rod drop time $\leq$ 2.4 secs.	Increased scram time due to smaller guide tube thimble in 15x15 OFA than current fuel assemblies.
3/4 1-24		Figure 3.1-1	Revised three loop rod insertion limits.
3/4 1-25		Figure 3.1-2	Revised four loop rod insertion limits.

JUSTIFICATION FOR D. C. COOK UNIT 1 (CYCLE 8) TECHNICAL SPECIFICATION CHANGES SUMMARY

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
B 3/4 1-1	3/4 1.1 (Bases)	Change in shutdown margin from 1.75% to 1.60% $\Delta k/k$	This supports changes made in Sections 3.1.1.1 and 4.1.1.1.1.
3/4 2-1; 3/4 2-2; 3/4 2-3; 3/4 2-4		No change	Included for completeness.
3/4 2-5	3.2.2	The LOCA FQ limit revised to 2.00 (4.00 for $P \leq .5$). Equation definition of $F_Q(Z,\ell)$	Resulted from 15x15 OFA reload LOCA analysis. Definition expanded to include both Westinghouse and ENC fuels.
		Word definition of F_Q^L ($E\ell$) and K(Z)	Definition expanded to include both Westinghouse and ENC fuels.
		"Setpoint reduction...with the reactor in at least HOT STANDBY."	Wording changed to be consistent with Westinghouse Tech Specs.
3/4 2-6	4.2.2.2.c	Equation definition of $F_Q^M(Z)$	Definition expanded to include both Westinghouse and ENC fuels.
3/4 2-7	4.2.2.2.c	Word definition V(Z)	V(Z) function will be removed from Tech Spec and defined in peaking factor limit report. This report will be available in the licensee's offices 60 days prior to cycle startup.
		Figure reference K(Z)	Reference expanded to include both Westinghouse and ENC fuels.
		Definition of $E_p(Z)$	Definition expanded to include Westinghouse fuel.
		Equation definition of $E_p(Z)$	Definitions expanded to include both Westinghouse and ENC Fuels

JUSTIFICATION FOR D. C. COOK UNIT 1 (CYCLE 8) TECHNICAL SPECIFICATION CHANGES SUMMARY

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
3/4 2-8	4.2.2.2.e	"...increase in peak pin power $F_{\Delta H}$..." changed to "...increase in max over Z of $[\frac{F_Q^M(Z)}{K(Z)}]$..."	Surveillance on $F_Q \cdot F_{\Delta H}$ variation is not always in the same direction as F_Q.
		"...that the peak pin power, $F_{\Delta H}$..." changed to "...that max over Z of $[\frac{F_Q^M(Z)}{K(Z)}]$..."	Surveillance on $F_Q \cdot F_{\Delta H}$ variation is not always in the same direction as F_Q.
3/4 2-9	4.2.2.2.f	"...and $P \geq 0.5$". An equation was added	For clarity, "$P \geq 0.5$" was removed from the equation and inserted in the text. Equations for Westinghouse and ENC fuels are separately defined.
3/4 2-10	Figure 3.2-2	Relabeled figure to denote ENC	--
3/4 2-11	Figure 3.2-3	The LOCA K(Z) curve was revised.	Resulted from 15x15 OFA reload LOCA analysis.
3/4 2-12	3.2.3	Revised $F_{\Delta H}$ values for Westinghouse and Exxon Nuclear Co. fuels	$F_{\Delta H}^N$ values were revised in order to define the different $F_{\Delta H}^N$ values for each fuel type.
3/4 2-13	4.2.3.2	Section deleted	Due to the use ITDP, the measured $F_{\Delta H}^N$ value is not increased by 4%, since this measurement uncertainty has been included in the design DNBR limit values.
3/4 2-14	3.2.4	Added "from RATED THERMAL POWER" after "3%" in two places	For clarity and consistency with Unit 2 Technical Specifications.
3/4 2-15		No change.	Added for completeness.
3/4 2-16	4.2.5.2	RCS flow rate to be measured once per month.	Required for use of ITDP.
3/4 2-17	Table 3.2-1	DNB parameters added in DTP column.	Parameters changed to reflect use of ITDP. Design Thermal Power column added to reflect increased Tavg at 3411 MWt.

JUSTIFICATION FOR D. C. COOK UNIT 1 (CYCLE 8) TECHNICAL SPECIFICATION CHANGES SUMMARY

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
3/4 2-18	3.2.6	Equation definition of $[F_j(Z)]_S$	Definition expanded to include both Westinghouse and ENC fuels.
3/4 2-18	3.2.6.c	Figure reference for K(Z)	Reference expanded to include both Westinghouse and ENC fuels.
3/4 2-19	3.2.6.e	Figure reference for F_Q^L (E_ℓ) and $T(E_\ell)$; word definition of $T(E_\ell)$	Reference and words expanded to include both Westinghouse and ENC fuel.
3/4 2-20	3.2.6.g	Equation definition of F_p	Definition expanded to include both Westinghouse and ENC fuel.
		Equation definition of APL	"
		"V(Z) is the function given in the peaking factor limit report"	V(Z) function will be removed from Tech Specs and defined in the peaking factor limit report.
3/4 2-21, 3/4 2-22		No change	Added for completeness.
3/4 2-23		Figure	Figure number changed.
3/4 2-24		Figure	Revised figure for Westinghouse fuel.
B 3/4 2-4	3/4 2.2 and 3/4 2.3 (Bases)	Revised explanation of $F_{\Delta H}$ uncertainty	Same justification as page 3/4 2-13.
B 3/4 2-5	3/4 2.2 and 3/4 2.3	Changed "effect" to "affect"	Grammatical change.
B 3/4 2-6	3/4 2.5 (Bases)	Revised "DNBR limit of 1.30" to "applicable design DNBR limit."	This change was made due to the use of ITDP.
B 3/4 3-[illegible]	3/4.3.3.6	Pages referred to in text were renumbered.	Consistency with previous page changes.
B 3/4 4-1	3/4.4.1 (Bases)	Revised "DNBR limit of 1.30" to "applicable design DNBR limit."	These changes were made due to the use of ITDP.

ATTACHMENT G TO AEP:NRC:0745C

ANALYSIS USING THE STANDARDS IN 10 CFR 50.92 ABOUT THE ISSUE OF NO SIGNIFICANT HAZARDS CONSIDERATION FOR THE LICENSE AMENDMENT REQUEST CONTAINED IN LETTER NO. AEP:NRC:0745C.

Our analysis of the contents of the license amendment requested in this letter shows that no significant hazards considerations are involved. In comparing the contents of the request against the standards of 10 CFR 50.92(c) we have found that the license amendment:

(1) does not involve a significant increase in the probability or consequences of an accident previously evaluated. As shown in Attachments B, C, D and E to this letter, the reloaded core is very similar in design to earlier cores and the results of the pertinent safety analyses show conformance with regulatory limits.

(2) does not create the possibility of a new or different kind of accident from any accident previously evaluated. This fact is true because of the similarity of designs between the new Westinghouse and the currently inserted Exxon fuel assemblies, because of the use of standard calculational techniques approved by the NRC for cores reloaded by Westinghouse which have shown acceptable results and because no modifications are being requested for any Plant component other than the replacement of eighty Exxon fuel assemblies by the same number of Westinghouse assemblies and,

(3) does not involve a significant reduction in a margin of safety. As shown in Attachments B, C, D and E to this letter the safety analyses performed in support of the license amendment show that sufficient margin exists to the 10 CFR 100, the 10 CFR 50.46 and the DNBR limits to ensure that the reloaded core will operate in a manner which is comparable in terms of safety to that of earlier cycles. No modifications are being requested in the license amendment that would degrade the Plant's ability to safely control and mitigate any design basis accident.

Three points are clarified further:

a) Cycle 8 of Unit 1 will employ burnable poison rods of the WABA design. This new burnable poison rod is compatible with the new Westinghouse fuel assemblies and satisfies all performance requirements for its design life. Westinghouse has submitted a topical report (see Attachment B) on the WABA design and is supporting the NRC's generic review in order to obtain approval prior to the Cycle 8 startup date. We do not consider this point to involve any consideration of a significant safety hazard.

b) The application for Cycle 8 of Unit 1 reload also employs a modified version of the PAD code to calculate fuel temperatures during the accident analysis. Westinghouse has similarly submitted a topical report (see Attachment A) on this matter and will support the NRC's generic review to obtain approval prior to the Cycle 8 startup date. We do not consider this point to involve any consideration of a significant safety hazard.

c) The average region discharge burnups are not expected to exceed 39,000 MWD/MTU. This limit is well within today's technology and design capabilities. The analyses where high burnup and corresponding clad characteristics were of importance, have accounted for the maximum expectable values and shown acceptable results. We do not consider this point to involve any consideration of a significant safety hazard.