Attachment A to AEP:NRC:0745F
Revisions to Proposed Technical Specifications
(As Found in Attachment A to AEP:NRC:0745C)

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	
$F_Q(Z,\ell) \leq [\frac{1.97}{P}]\ [K(Z)]$	$F_Q(Z,\ell) \leq [\frac{F^L_Q(E_\ell)}{P}]\ [K(Z)]$	$P > 0.5$
$F_Q(Z,\ell) \leq [3.94]\ [K(Z)]$	$F_Q(Z,\ell) \leq 2\ [F^L_Q(E_\ell)\ K(Z)]$	$P \leq 0.5$

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

$F^L_Q(E_\ell)$ 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 $\overline{R}$.

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	
$F_Q^M(Z) \leq \left[\frac{1.97}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$F_Q^M(Z) \leq \left[\frac{F_Q^L(Z)}{P \times E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$P > 0.5$
$F_Q^M(Z) \leq \left[\frac{3.94}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$F_Q^M(Z) \leq \left[\frac{2\ F_Q^L(Z)}{E_p(Z)}\right] \frac{K(Z)}{V(Z)}$	$P \leq 0.5$

SURVEILLANCE REQUIREMENTS (Continued)

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:

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

Exxon Nuclear Co Fuel

$$\left[\left[\text{max. over Z of } \frac{F_Q^M(Z) \times V(Z) \times E_p(Z)}{\frac{1.97}{P} \times [K(Z)]}\right] - 1\right] \times 100$$

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.

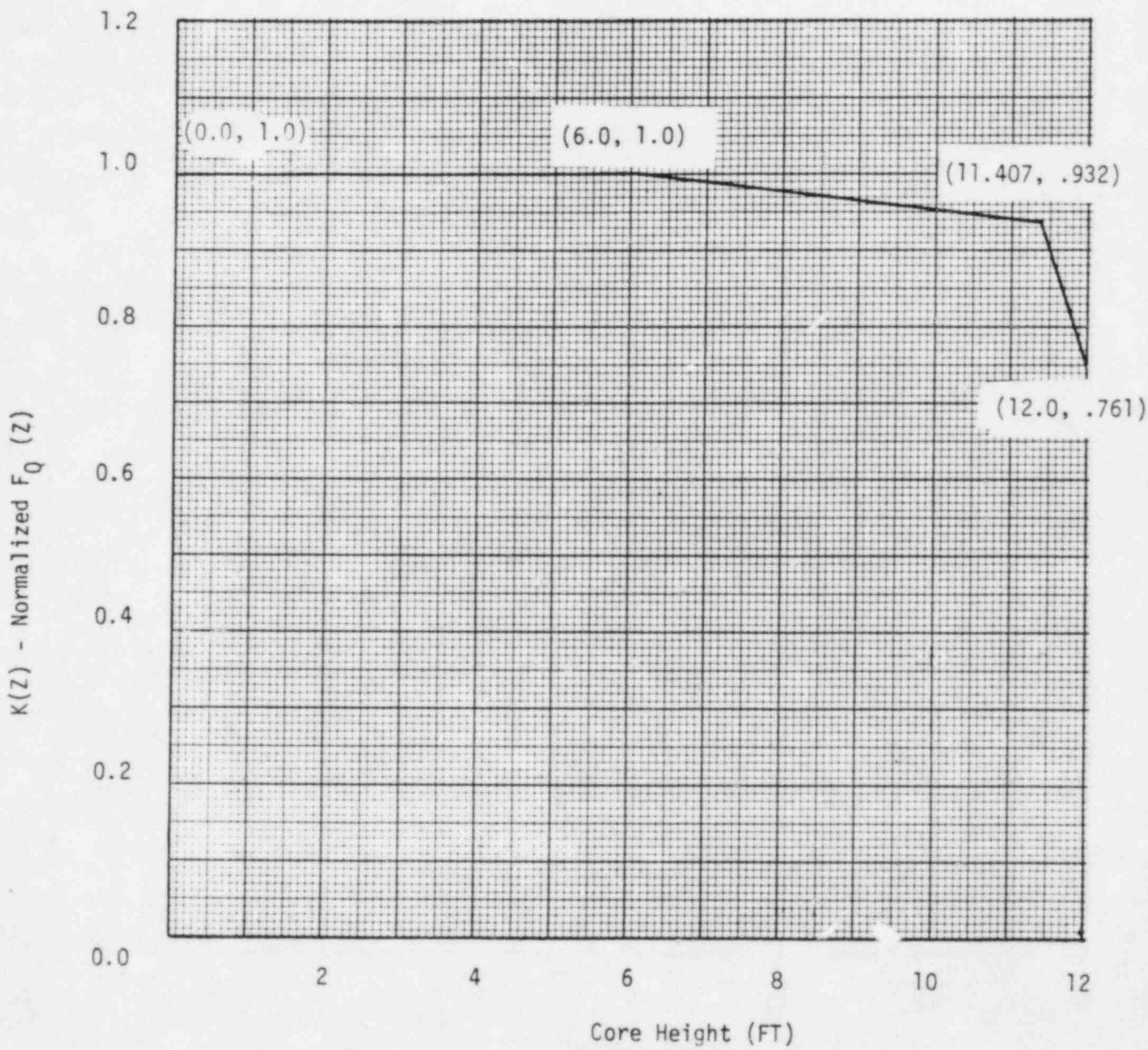

Figure 3.2-3 K(Z) - Normalized F_Q (Z) As A Function of Core Height For Westinghouse Fuel

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

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

Westinghouse Fuel

$$[F_j(Z)]_s = \frac{[1.97]\ [K(Z)]}{(\overline{R_j})(P_L)(1.03)(1 + \sigma_j)(1.07)F_p}$$

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. $\overline{R_j}$, 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.

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

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.*

$$APL = \text{min over Z of } \frac{1.97 \times K(Z)}{F_Q(Z,\ell) \times V(Z)} \times 100\ \%$$ Westinghouse Fuel

$$APL = \text{min over Z of } \frac{F_Q^L(E_\ell) \times K(Z)}{F_Q(Z,\ell) \times V(Z) \times E_p(Z)} \times 100\ \%$$ 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 Factc... Report. The above limit is not applicable in the following core plane ... ns.

1. Lower co... ion 0% to 10% inclusive.
2. Upper core re... 0% to 100% inclusive.

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

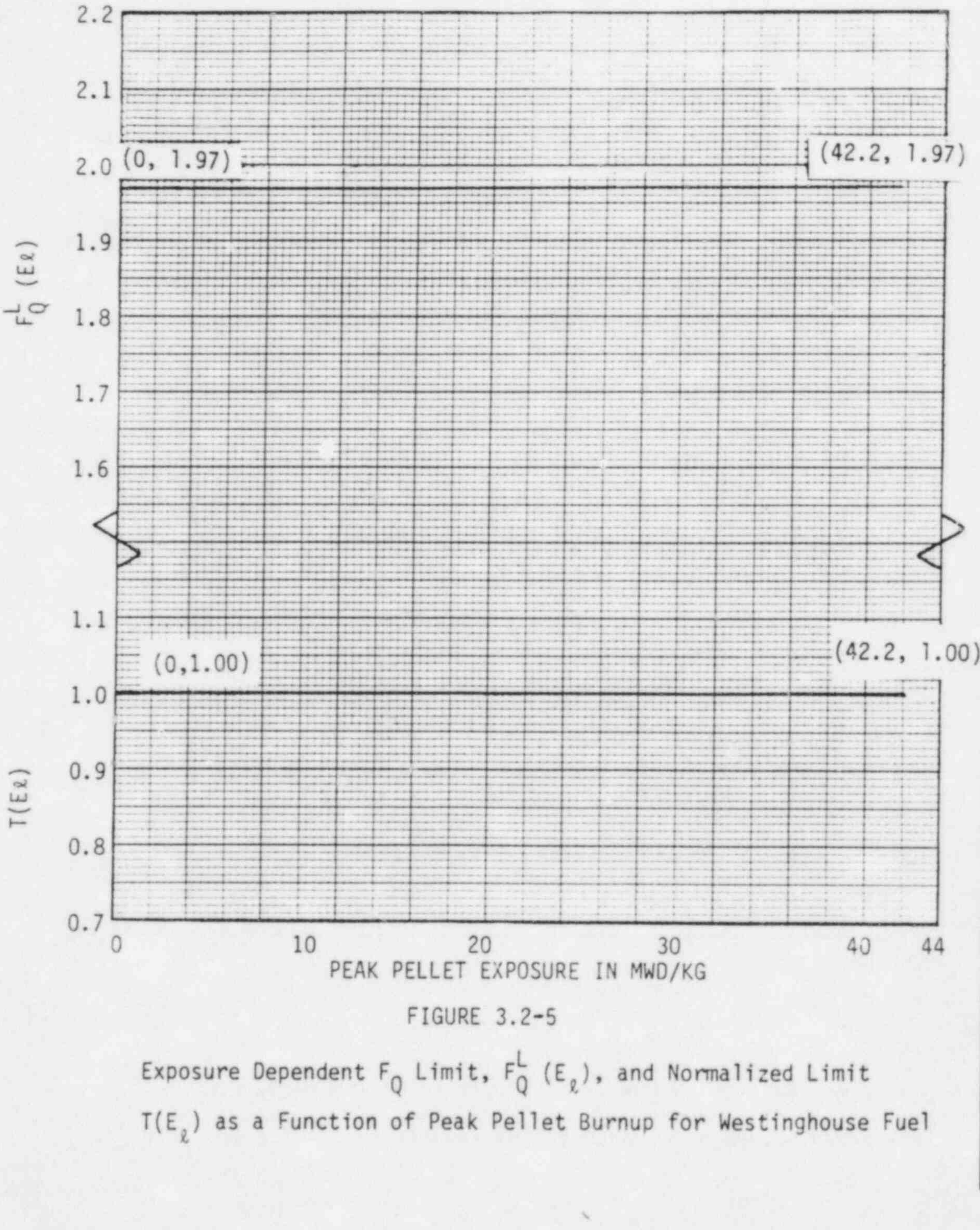

FIGURE 3.2-5

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

Attachment B to AEP:NRC:0745F
Updated Safety Evaluation
(Replacing that in Attachment B to AEP:NRC:0745C)

1.0 INTRODUCTION

D. C. Cook Unit 1 is operating with an all Exxon Nuclear Company (ENC) fueled core during Cycle 7. For subsequent cycles, it is planned to refuel Unit 1 with 15x15 optimized fuel assembly (OFA) regions supplied by the Westinghouse Electric Corporation (W). As a result, future core loadings would range from approximately a 40 percent OFA and 60 percent ENC fueled core to eventually an all OFA fueled core. The W 15x15 OFA fuel design is similar to the W 15x15 LOPAR (low parasitic) fuel which has had substantial operating performance in a number of nuclear plants. The major difference introduced by the W 15x15 OFA design is the use of five intermediate Zircaloy grids replacing five intermediate Inconel grids for the LOPAR fuel. The 15x15 Zircaloy grid design is similar to the W 17x17 OFA grid design. The W 17x17 OFA design has been generically approved by the NRC via their review of the W 17x17 OFA Reference Core Report.(1) Operating experience has been obtained for six demonstration 17x17 OFAs which contain Zircaloy intermediate grids.(2) Two assemblies have satisfactorily completed three cycles of irradiation to about 28,000 MWD/MTU burnup, two have completed two cycles to about 19,400 MWD/MTU, and two have completed one cycle in excess of 9,000 MWD/MTU. The demonstration OFAs have been examined and provide reason to expect good performance from the 15x15 OFA design.

This report summarizes the results of the W analyses which justify the transition from an all ENC core, through a mixed OFA/ENC fueled core to an all OFA core. Although it is planned to operate D. C. Cook Unit 1 Cycle 8 at the current licensed maximum power level of 3250 MWt, the core evaluations/analyses summarized in this report have been performed at a reactor power level of 3411 MWt, with the exception of the large break LOCA which was analyzed at 3250 MWt. This conservative design basis provides early identification of those safety/accident analysis limits for a potential uprating.

All analyses were performed utilizing W standard methods, which are described in the W Reload Safety Evaluation Methodology Topical.[3] The approved Westinghouse Improved Thermal Design Procedure (ITDP) is used in the DNB analyses of both W and ENC fuel. The W WRB-1 correlation is used in the OFA DNB analyses. Both the ITDP and WRB-1 correlation were previously used to license D. C. Cook Unit 2 operation. The ENC fuel is analyzed using the W-3 DNB correlation. Other features being introduced with the Cycle 8 reload include the Westinghouse Wet Annular Burnable Absorber (WABA) rods and a revision to the Westinghouse fuel thermal safety model (PAD Code) used in the safety analyses. In addition, for the large break LOCA the worst break was reanalyzed with the currently NRC approved PAD fuel thermal safety model[22] determining the initial fuel rod conditions. Westinghouse has submitted topical reports[4,5] on these subjects and is supporting the NRC's generic review, in order to obtain approval well before the planned Cycle 8 startup.

2.0 SUMMARY AND CONCLUSIONS

The Westinghouse Reload Safety Evaluation Methodology[3] was used to evaluate the transition from ENC fuel to W 15x15 OFA fuel for D. C. Cook Unit 1. Parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate the safety evaluation of future reload cores. Transition core effects were considered in the mechanical, thermal and hydraulic, nuclear, and accident evaluations described in Chapter 18 of Reference 1. The summary of these evaluations for the D. C. Cook Unit 1 transition to an all W 15x15 OFA core is given in the following sections of this submittal.

The transition design and safety evaluations are based on the following maximum power conditions: 3411 MWt reactor power and 577.1°F vessel average temperature, with the exception of the large break LOCA which was analyzed at 3250 MWt.

The results of evaluations/analyses and tests discussed in this report lead to the following conclusions:

1. The Westinghouse OFAs are mechanically and hydraulically compatible with the ENC fuel assemblies, control rods, and reactor internals interfaces.

2. Changes in the nuclear characteristics due to the transition from ENC to W 15x15 OFA fuel will be within the normal variations from cycle-to-cycle due to fuel management effects. W 15x15 OFA fuel up to and including a 4.00% nominal enrichment can be stored in the fresh and spent fuel areas.

3. Demonstration experience with W 17x17 OFAs containing Zircaloy grids provides reason to expect satisfactory operation from 15x15 OFA Zircaloy grids.

4. The WABA rod, as described in its generic topical[4], is compatible with the W 15x15 OFA and satisfies all performance requirements for its design life.

5. The proposed Technical Specification changes presented in Attachment A are applicable to cores containing any combination of W 15x15 OFA and EXC fuel.

6. All design criteria for the W 15x15 OFA fuel are satisfied.

7. A reference is established upon which to base future cycle safety evaluations for W OFA reload fuel.

3.0 MECHANICAL EVALUATION

The mechanical design requirements and criteria for the 17x17 OFA design are described in Reference 1, which was approved by the NRC. The 15x15 OFA design meets these same basic requirements and criteria.

ENC, in estabishing their assembly design, demonstrated their fuel's compatibility with the W LOPAR design which was the initial D. C. Cook Unit 1 fuel. W has demonstrated compatibility of its 15x15 OFA design with its LOPAR design. Compatibility of the OFA and ENC fuel is thereby demonstrated.

Figure 1 and Table 1 present a comparison of the W 15x15 OFA and ENC fuel assemblies. The W and ENC fuel rods have similar length and clad OD dimensions. The W 15x15 OFA rods have the same design as the LOPAR W 15x15 fuel rods which have exhibited good in-core performance in many operating reactors.

The top and bottom Inconel grids of the OFA are the same as the Inconel grids of a W LOPAR fuel assembly. The five intermediate OFA Zircaloy-4 grids have thicker and wider straps than the OFA Inconel grids (See Figure 1) in order to closely duplicate the Inconel grid strength. The ENC assembly grids are bimetallic, consisting of Zircaloy-4 straps with Inconel grid springs. Both the OFA Zircaloy and ENC bimetallic grids e grid heights of 2.25 inches. Elevation of the grids was established to ensure satisfactory axial alignment during operation.

Due to thicker Zircaloy grid straps and a resulting reduced cell size, the OFA guide thimble tube ID (above dashpot) has a 12 mil reduction compared to the ENC thimble tube ID of 0.511 inches. Below the dashpot, the OFA and ENC fuel thimble tubes have the same dimensions. The OFA guide tube thimble ID provides sufficient nominal diametral clearance for control rods as well as source rods, burnable absorber rods, and OFA thimble plugs. Due to reduced OFA diametral clearance, the control rod

scram time to the dashpot is increased from the current 1.8 seconds to 2.4 seconds. This increase in rod drop time was determined from conservative analytical calculations. The 2.4 second scram time is used in all the accident reanalyses.

The OFA design has minor differences in the overall height of the top and bottom nozzles, the adapter plate flow-slot configuration and hold-down leaf springs as compared to the ENC fuel assembly design. These minor differences have no adverse impact on the interaction of W 15x15 OFA and ENC assemblies during fuel handling operations or reactor operations. The W 15x15 OFA design uses a 3-leaf holddown spring design compared to the 2-leaf springs in the ENC assembly. The W OFA 3-leaf spring has been previously used in 15x15 LOPAR assemblies, as well as on the 17x17 OFA demonstration assemblies. The 3-leaf spring provides additional holddown force margin compared to the 2-leaf spring. The OFA bottom nozzle has similar design features and dimensions compared to the ENC nozzle. The OFA bottom nozzle design has a reconstitutable feature, as shown in Figure 2, which allows it to be easily removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the standard W LOPAR nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

As stated in the 17x17 OFA Reference Core Report [1], for a given burnup, the magnitude of rod bow for the W OFA is conservatively assumed to be the same as that of a W LOPAR fuel assembly. The most probable causes of significant rod bow are rod-grid and pellet-clad interaction forces and wall thickness variation. Since the OFA fuel rods are the same as the W LOPAR fuel rods, there will be no difference in predicted bow due to rod considerations. The OFA design will have reduced grid forces due to the Zircaloy grid springs. Therefore, this component is predicted to decrease OFA rod bow compared to LOPAR fuel.

The wear of fuel rod cladding is dependent on both the support provided by the grids and the flow environment to which it is subjected. OFA and ENC assembly flow test results were evaluated. ENC hydraulic test results show the crossflow between ENC and W 15x15 LOPAR assemblies is very similar to that obtained during W flow tests on side-by-side W 15x15 OFA and W 15x15 LOPAR assemblies. These tests showed only a small crossflow between assemblies and no significant fuel rod wear due to rod vibration. Extrapolation of the results from flow tests involving OFA and LOPAR assemblies shows that fuel rod wear would be less than ten (10) percent of the cladding thickness for at least 48 months of reactor operation. This assures that clad wear will not impair fuel rod integrity.

The above conclusions on OFA rod wear and integrity have also been supported by analytical results. The analysis accounted for rod vibrations caused by both axial and crossflows, and the effect of potential fuel rod to grid gaps.

4.0 NUCLEAR EVALUATION

The nuclear design of cores with W OFA and ENC fuel is accomplished by using the standard calculational methods as described in the W Reload Safety Evaluation Methodology(3). The dimensional and material differences between the W and ENC assemblies are small so that the W computer codes and methods are also valid for the ENC fuel. Dimensions and composition for each of the two fuel designs were used to establish the models. The burnup distribution of the ENC fuel assemblies remaining in Cycle 8 has been obtained by depleting the loading patterns from earlier cycles using two-dimensional and three-dimensional models of the applicable cores.

Changes in the nuclear characteristics during the transition cycles from an ENC fueled core to a W 15x15 OFA core will be primarily due to fuel management considerations (number of feed assemblies, feed enrichment, cycle burnup, etc.) and not due to the differences in fuel assembly design. Each reload core design will be evaluated to assure that design and safety limits for the OFA and ENC fuel are satisfied according to the W reload safety evaluation methodology. For the evaluation of the worst-case $F_Q(Z)$ envelope, axial power shapes are synthesized with the limiting F_{xy} values chosen over three overlapping burnup windows during the cycle. The design and safety limits will be documented in each cycle specific reload safety evaluation report which serves as the basis for any significant changes requiring NRC review.

In order to accommodate potential increases in future feed enrichments, a criticality analysis of the fuel storage areas was performed for nominal enrichments up to and including 4.00 wt.% U-235 in W 15x15 OFA fuel. These analyses confirm that all current safety criteria applicable to fuel storage are satisfied(6).

5.0 THERMAL AND HYDRAULIC EVALUATION

Results of hydraulic compatibility tests performed by the Exxon Nuclear Company for the ENC and W 15x15 LOPAR assemblies were compared to hydraulic test data for the W 15x15 LOPAR and OFA assemblies. The data show that the W 15x15 OFA fuel assemblies are hydraulically compatible with the ENC fuel assemblies. Pressure drop data were obtained over a range of fluid temperatures and flow rates. Pressure drop values were then extrapolated to core operating conditions. At typical reactor conditions, the ENC fuel assembly has a pressure drop within 0.7 percent of the W 15x15 OFA pressure drop.

The thermal hydraulic design of this core is conservatively analyzed at 3411 MWt core 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 core power and a 567.8°F vessel average temperature. The analyses employed the Improved Thermal Design Procedure[7] (ITDP) and the THINC IV[8,9] computer code. The WRB-1[10] DNB correlation was used in the W 15x15 OFA analyses, whereas the W-3 correlation was used to analyze the ENC fuel. The thermal-hydraulic design criteria remain the same as those presented in the D. C. Cook Unit 1 Updated FSAR[11]. All design criteria are satisifed.

The design method employed to meet the DNB design basis is the ITDP[7]. 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 probability that the minimum DNBR will be greater than or equal to the limit DNBR for the peak power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a design DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition,

the limit DNBR values are increased to values designated as the safety analysis limit DNBRs. The plant allowance available between the safety analysis limit DNBR values and the design limit DNBR values is not required to meet the design basis.

In this application, the WRB-1 DNB correlation[10] is employed in the thermal hydraulic design of the W 15x15 OFA fuel. Due to an improvement in the accuracy of the critical heat flux prediction with the WRB-1 correlation compared to previous DNB correlations, a correlation limit DNBR of 1.17 is applicable. The W-3 DNBR correlation [12,13] was used in the design of the ENC fuel assembly. A W-3 correlation limit DNBR of 1.30 is applicable.

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

	W 15x15 OFA		ENC 15x15	
	Typical	Thimble	Typical	Thimble
Correlation Limit	1.17	1.17	1.30	1.30
Design Limit	1.32	1.31	1.58	1.50
Safety Analysis Limit	1.69	1.69	1.58	1.50

The margin to the safety analysis DNBR limit is more than sufficient to cover the maximum 12.5 percent rod bow penalty at full flow Conditions[14] and a 5 percent transition core penalty, both applied to the OFA only. An additional rod bow penalty of 2.4 percent DNBR at loss-of-flow conditions[14] is covered explicitly in the loss-of-flow analysis for the W 15x15 OFA. The 5 percent transition penalty was determined by analyzing W 15x15 OFA and ENC assembly loading patterns at various core conditions in the same manner as the W 17x17 OFA/LOPAR fuel analysis which was reviewed and

approved by the NRC[15]. The 5 percent transition penalty for OFA is due to the higher OFA mixing vane loss coefficient compared to that of the ENC fuel. This results in localized flow redistribution from the OFA to the ENC assembly near mixing vane grid positions. When the full transition is complete (all ENC assemblies removed from core), the transition core penalty will no longer apply to OFA assemblies.

The ENC fuel assembly would be expected to have less gap closure than the W 15x15 OFA, due to the ENC fuel's thicker cladding, as shown in Reference 16. Data obtained by other investigations[17,18] show that gap closures up to 55 percent have no measurable effect on DNB. Therefore, no resultant rod bow DNBR penalty is required for ENC 15x15 fuel.

6.0 ACCIDENT ANALYSES AND EVALUATION

6.1 NON-LOCA ACCIDENT ANALYSES AND EVALUATION

The effects of the transition from the resident ENC fuel to W OFA on the non-LOCA accident analyses have been addressed. The standard Westinghouse reload methodology described in Reference 3 was used. All of the non-LOCA accidents* in the D. C. Cook FSAR were reanalyzed to include three major design changes:

1. The analyses were performed at a conservative reactor power level of 3411 MWt. This affects all of the transients that are limiting at full power.

2. The ITDP was used with both the WRB-1 and WRB-3 DNB correlations. This impacts all of the DNB limited accidents. A conservative set of core thermal safety limits, overtemperature delta T, and overpower delta T setpoints were generated that are applicable for both the transition and complete OFA cores. These limits are valid for reactor power levels up to and including 3411 MWt.

3. The control rod scram time to the dashpot is increased 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 analyses requiring this parameter.

Also included in the analyses were fuel temperatures based on the revised PAD code. A +5 pcm/°F moderator temperature coefficient (MTC) existing at full power was conservatively used for heatup events. This is conservative since the Technical Specifications require a non-positive MTC at or above seventy (70) percent power.

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*With exception of startup on an inactive loop. This transient cannot occur above 10 percent rated thermal power and thus was not reanalyzed.

The acceptance criterion used in the non-LOCA safety analyses is independent of the fuel vendor. Thus, the results of the FSAR Chapter 14 accident reanalysis and evaluation, which are contained in Attachment C, show that the transition to OFAs can be accommodated with margin to the applicable FSAR safety limits for power levels up to and including 3411 MWt.

6.2 LARGE BREAK LOCA (at 3250 MWt)

Description of Analysis Assumptions for W 15x15 OFA Fuel, Including Transition Impact

The large break loss-of-coolant accident (LOCA) analysis for D. C. Cook Unit 1, applicable to a full W 15x15 OFA core, was analyzed to develop W 15x15 OFA fuel specific peaking factor limits. This analysis is consistent with the methodology employed in Reference 1. The currently approved 1981 large break ECCS evaluation model[19] was utilized for a spectrum of cold leg breaks. The revised PAD fuel thermal safety model[5] generated the initial fuel rod conditions.* The D. C. Cook Unit 1 analysis was performed for an assumed steam generator tube plugging level of five (5) percent, and was analyzed for both minimum and maximum safeguards (safety injection flows) assumptions, in accordance with Reference 20. A revised FSAR chapter 14.3.1.1, given in Attachment D, contains a full description of the analysis and assumptions utilized for the W OFA ECCS LOCA analysis. The ENC fuel ECCS analysis contained in FSAR section 14.3.1.2 remains unchanged.

When assessing the LOCA impact of transition cores, it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than either a complete core of the reference fuel

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* The worst break was reanalyzed with the currently NRC approved PAD fuel thermal safety model[22] initial fuel rod conditions.

design or a complete core of the new fuel design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch.

For the ENC and W 15x15 OFA designs, this difference in fuel assembly resistance (K/A^2), is less than one percent. The different flow resistances for the two assembly designs impact two portions of the LOCA analysis model. One is the reactor coolant system (RCS) blowdown portion of the transient, analyzed with the SATAN VI computer code, where the higher resistance W OFA assembly has less cooling flow than the ENC assembly. While the SATAN VI computer code models the crossflow between the average core flow channel (N-1 fuel assemblies) and a hot assembly flow channel (one fuel assembly), experience has shown that the SATAN VI results are not significantly affected by small differences in the hydraulic resistance ($\pm$ 10 percent) between these two channels. Since small resistance mismatches in the core are insignificant when compared to the total system resistance, and since the total core resistance is uniformly distributed in the SATAN VI code, the effect on the large break LOCA blowdown transient of modeling hydraulic resistance mismatch can be neglected. Therefore, it is not necessary or meaningful to perform a new SATAN VI analysis for this transition core configuration because the hydraulic resistance mismatch is much less than $\pm$ 10 percent.

The other portion of the LOCA evaluation model impacted by the hydraulic resistance difference is the core reflood transient. Since the hydraulic mismatch is so small, only crossflows due to the smaller rod size and different grid designs need to be evaluated. The maximum reflood axial flow reduction for the W 15x15 OFA fuel at any location in the core, resulting from crossflows to adjacent ENC assemblies, has been conservatively calculated to be three percent. Analyses have been per-

formed, which demonstrate that a reduction of five (5) percent in reflood axial flow rate results in a 19°F PCT increase. Therefore, the maximum PCT penalty possible for W 15x15 OFA fuel during the transition period is 12°F. After this transition, the W ECCS analysis will apply to a full core without the crossflow penalty.

The resident ENC fuel is shown to have axial flow rates always greater than the nominal design flow rate, for core axial elevations where PCT's can possibly occur. Therefore, the ENC ECCS analysis is not detrimentally affected by assembly crossflow and remains applicable to the ENC fuel for transition cycles.

The method of analysis, including assumptions and codes used, are described in detail in the revised FSAR Chapter 14.3.1.1 provided in Attachment D.

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment D.

Conclusions

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one (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 seventeen (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.

The time sequence of events for all breaks analyzed is shown in Table 14.3.1-6 of the revised FSAR Chapter 14.3.1.1, presented in Attachment D.*

The large break W 15x15 OFA LOCA analysis for D. C. Cook Unit 1 utilizing the currently approved 1981 evaluation models resulted in a PCT of 2170°F for the 0.4 C_D (discharge coefficient) LOCA Maximum Safeguards Injection (Max. SI) case at a total peaking factor of 2.00.**

The small impact of crossflow for transition core cycles is conservatively evaluated to be at most a 12°F effect on the W fuel, which is easily accommodated in the margin to 10 CFR 50.46 limits.

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* The worst break was reanalyzed with the currently NRC approved PAD fuel thermal safety model.[22] Results are shown in Table 14.3.1A-5 of Addendum A of Attachment D.

** The worst break was reanalyzed with the currently NRC approved PAD fuel thermal safety model[22] initial fuel rod conditions. The results are shown in Addendum A of Attachment D. This resulted in a PCT of 2162°F for the 0.4 C_D (discharge coefficient) LOCA Maximum Safeguards Injection (Max. SI) case at a total peaking factor of 1.97.

The ENC ECCS analysis is not detrimentally affected by assembly cross-flow; consequently the ENC peaking factor limits remain valid for the ENC fuel during the transition period.

It can be seen from the results contained in Chapter 14.3.1.1 of the revised FSAR section that this ECCS analysis for D. C. Cook Unit 1 remains in compliance with 10 CFR 50.46 of Appendix K.

6.3 SMALL BREAK LOCA (at 3411 MWt)

Description of Analysis Assumptions for 15x15 OFA Fuel Including Transition Impact

The small break loss-of-coolant accident (LOCA) analysis for D. C. Cook Unit 1, applicable to a full W 15x15 OFA core, was analyzed to develop W 15x15 OFA fuel specific peaking factor limits. This is consistent with the methodology employed in Reference 1. The currently approved October 1975 small break ECCS evaluation model(21), was utilized for a spectrum of cold leg breaks. The revised PAD fuel thermal safety model(5), generated the initial fuel rod conditions. Revised FSAR chapter 14.3.2, given in Attachment E, contains a full description of the analysis and assumptions utilized for the W OFA ECCS LOCA analysis.

When assessing the impact of a LOCA on transition cores it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than either a complete core of the reference fuel design or a complete core of the improved fuel design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch.

The WFLASH computer code[21] is used to model the core hydraulics during a small break event. Only one core flow channel is modeled in WFLASH since the core flow rate during a small break is relatively low and this provides enough time to maintain flow equilibrium between fuel assemblies (i.e., crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break. Thus it is not necessary to perform a small break evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the W 15x15 OFA design.

The methods of analysis, including assumptions and codes used, are described in detail in the revised FSAR Chapter 14.3.2 in Attachment E.

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment E.

Conclusions

The small break optimized fuel LOCA analysis for D. C. Cook Unit 1, utilizing the currently approved 1975 Small Break Evaluation model, resulted in a peak clad temperature of 1630^{o}F for the 4-inch diameter cold leg break. The analysis assumed the worst small break power shape consistent with a LOCA F_Q envelope of 2.32 at core midplane elevation and 1.5 at the top of the core.

Analyses presented in the revised FSAR Chapter 14.3.2 show that the high head portion of the ECCS, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperature well below the required limits of 10 CFR 50.46. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

7.0 TECHNICAL SPECIFICATION CHANGES

Based on the preceding evaluations, a number of technical specification changes for D. C. Cook Unit 1 are required to support the transition to OFA. These changes are given in the proposed Technical Specification page changes (see Attachment A of this submittal).

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TABLE 1

Comparison of OFA and ENC Assembly Design

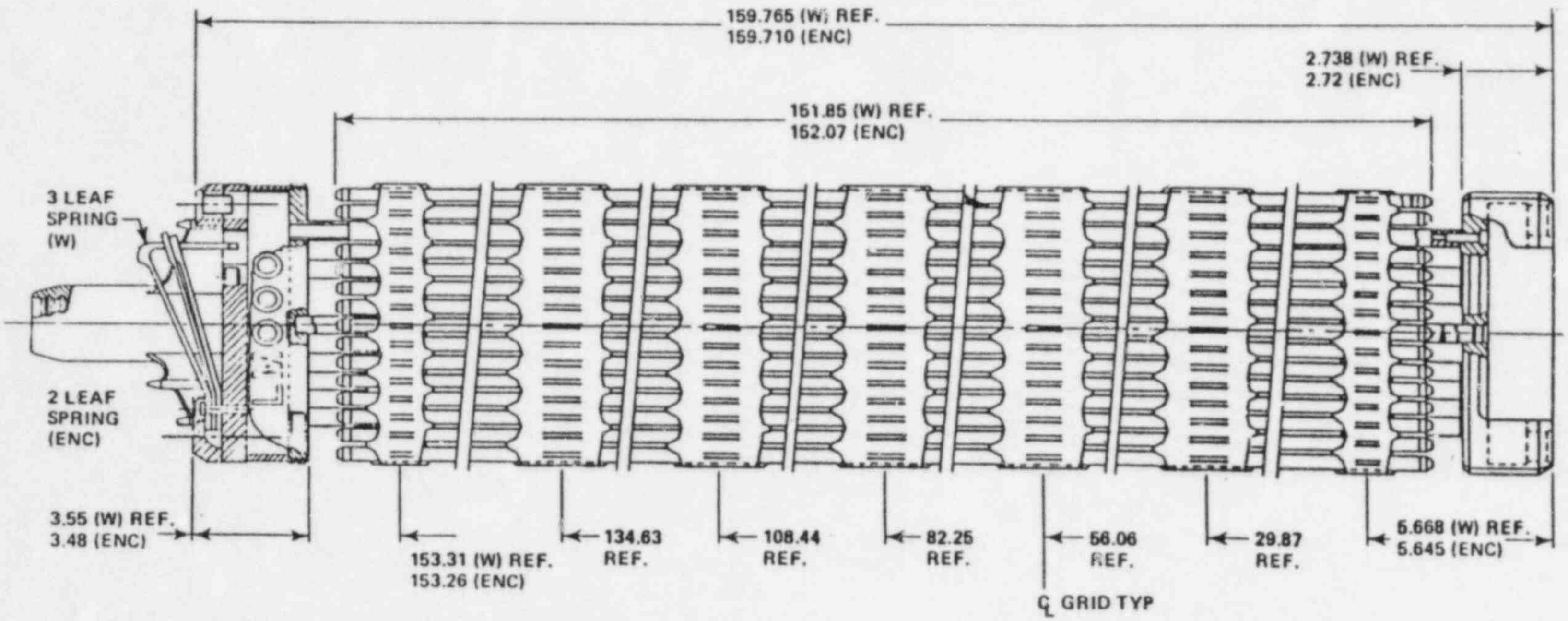
Parameter	15x15 W Optimized Fuel Assembly Design	15x15 ENC Fuel Assembly Design
Fuel Assembly Length, in.	159.765	159.71
Fuel Rod Length, in.	151.85	152.07
Assembly Envelope, in.	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in.	0.563	0.563
Number of Fuel Rods/Ass'y	204	204
Number of Guide Thimbles/Ass'y	20	20
Number of Instrumentation Tube/Ass'y	1	1
Compatible with Movable In-Core Detector System	Yes	Yes
Fuel Tube Material	Zircaloy-4	Zircaloy-4
Fuel Rod Clad OD, in.	0.422	0.424
Fuel Rod Clad Thickness, in.	0.0243	0.030
Fuel/Clad Gap, mil	7.5	7.5
Fuel Pellet Diameter, in.	0.3659	0.3565
Guide Thimble Material	Zircaloy-4	Zircaloy-4
Guide Thimble ID, in.*	0.499	0.511
Structural Material - Five Inner Grids	Zircaloy-4	Zircaloy-4 Straps Inconel Springs

*Above dashpot

TABLE 1 (Continued)

Parameter	15x15 W Optimized Fuel Assembly Design	15x15 ENC Fuel Assembly Design
Structural Material - 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

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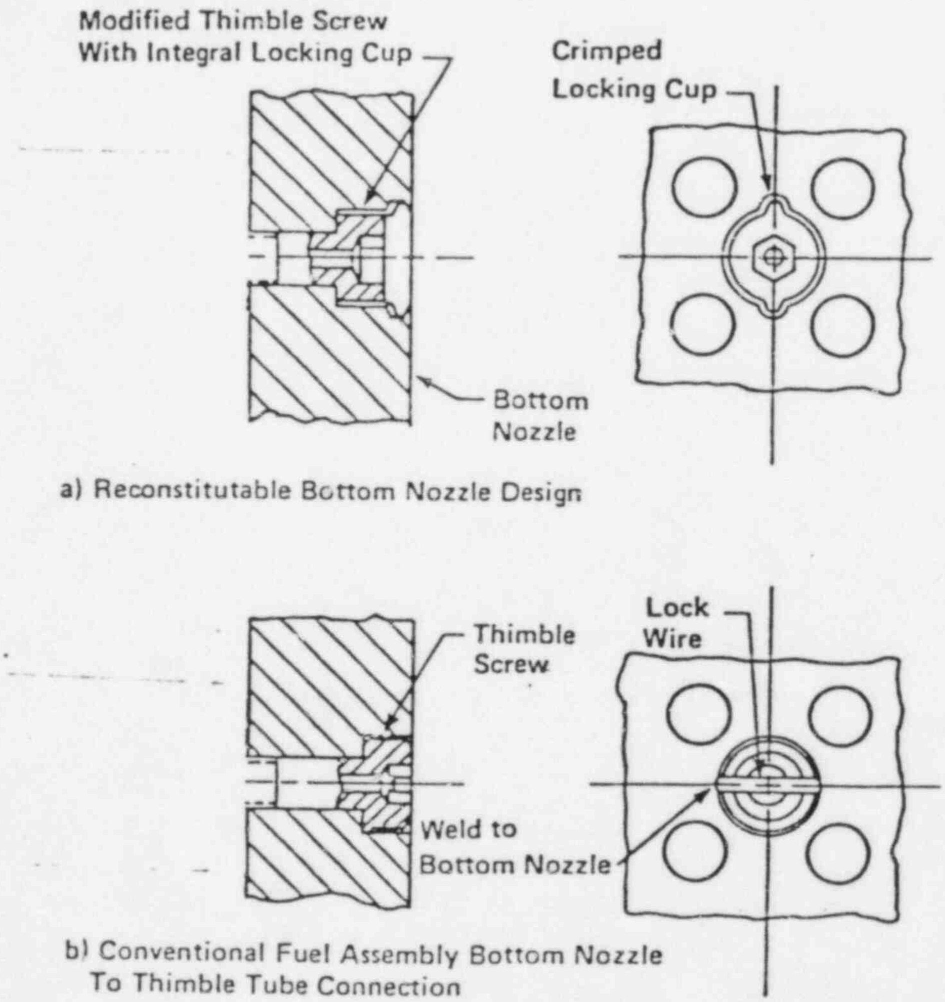
Figure 1 Comparison of ENC Fuel Assembly Dimensions With Westinghouse 15X15 OFA Schematic



BOTTOM NOZZLE TO THIMBLE TUBE CONNECTION

FIGURE 2

Attachment C to AEP:NRC:0745F
Updated Large Break LOCA Analysis
(Replacing that in Attachment D to AEP:NRC:0745C)

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 10CFR50. 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

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*The worst break (CD = 0.4 Max SI case) was reanalyzed with the currently NRC approved PAD fuel thermal safety model[16] initial fuel rod conditions. Results are in Addendum A.

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 phenomenon 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:

Figures 14.3.1-1 through 14.3.1-12	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.
Figures 14.3.1-13 through 14.3.1-24	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.
Figures 14.3.1-25 through 14.3.1-36	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.

Figures 14.3.1-37 through 14.3.1-44	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.

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*The worst break was reanalyzed with the currently NRC approved PAD fuel thermal safety model (16) initial fuel rod conditions. The results are in Addendum A. This resulted in a PCT of 2162°F for the 0.4 C_D (discharge coefficient) LOCA Maximum Safeguards Injection (Max SI) case at a total peaking factor of 1.97.

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16. Letter from J. F. Stoltz (NRC) to T. M. Anderson (Westinghouse); subject: Review of WCAP-8720, Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations, NRC SER dated March 27, 1980.

TABLE 14.3.1-1

LARGE BREAK

CONTAINMENT DATA

(ICE CONDENSER CONTAINMENT)

NET FREE VOLUME		
(Includes Distribution between Upper, Lower and Deadend compartments)	UC	746,829 ft^3
	LC	249,446
	DE	116,168
	IC	122,400
Initial Conditions		
Pressure		14.7 psia
Temperature for the Upper, Lower and Dead Ended Compartments	UC	100°F
	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 Upper and Lower Compartments	LC	2835 gpm
	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	.4848E+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	.3981E+07
.2600E+02	.5422E+04	.3398E+07
.2700E+02	.6389E+04	.3418E+07
.2800E+02	.6847E+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	.79[illegible]5E+03	.1672E+06
.4500E+02	.4830E+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+03	.3082E+08
.2000E+01	.4048E+05	.2495E+08
.4000E+01	.3507E+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	.8336E+04	.5534E+07
.2300E+02	.7786E+04	.4880E+07
.2400E+02	.7322E+04	.4503E+07
.2500E+02	.6390E+04	.3851E+07
.2600E+02	.5432E+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	.4954E+04	.1422E+07
.3600E+02	.3303E+04	.9903E+06
.3800E+02	.2519E+04	.6662E+06
.3950E+02	.6450E+03	.1585E+06
.4500E+02	.3425E+03	.2042E+06
.5000E+02	.3423E+03	.2042E+05
.5382E+02	.3425E+03	.2042E+05
.5412E+02	.2467E+03	.2588E+05
.5432E+02	.2467E+03	.2587E+05
.5442E+02	.3467E+03	.2586E+05
.5452E+02	.3467E+03	.2585E+05
.5462E+02	.3468E+03	.2575E+05
.6004E+02	.3790E+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	.1506E+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

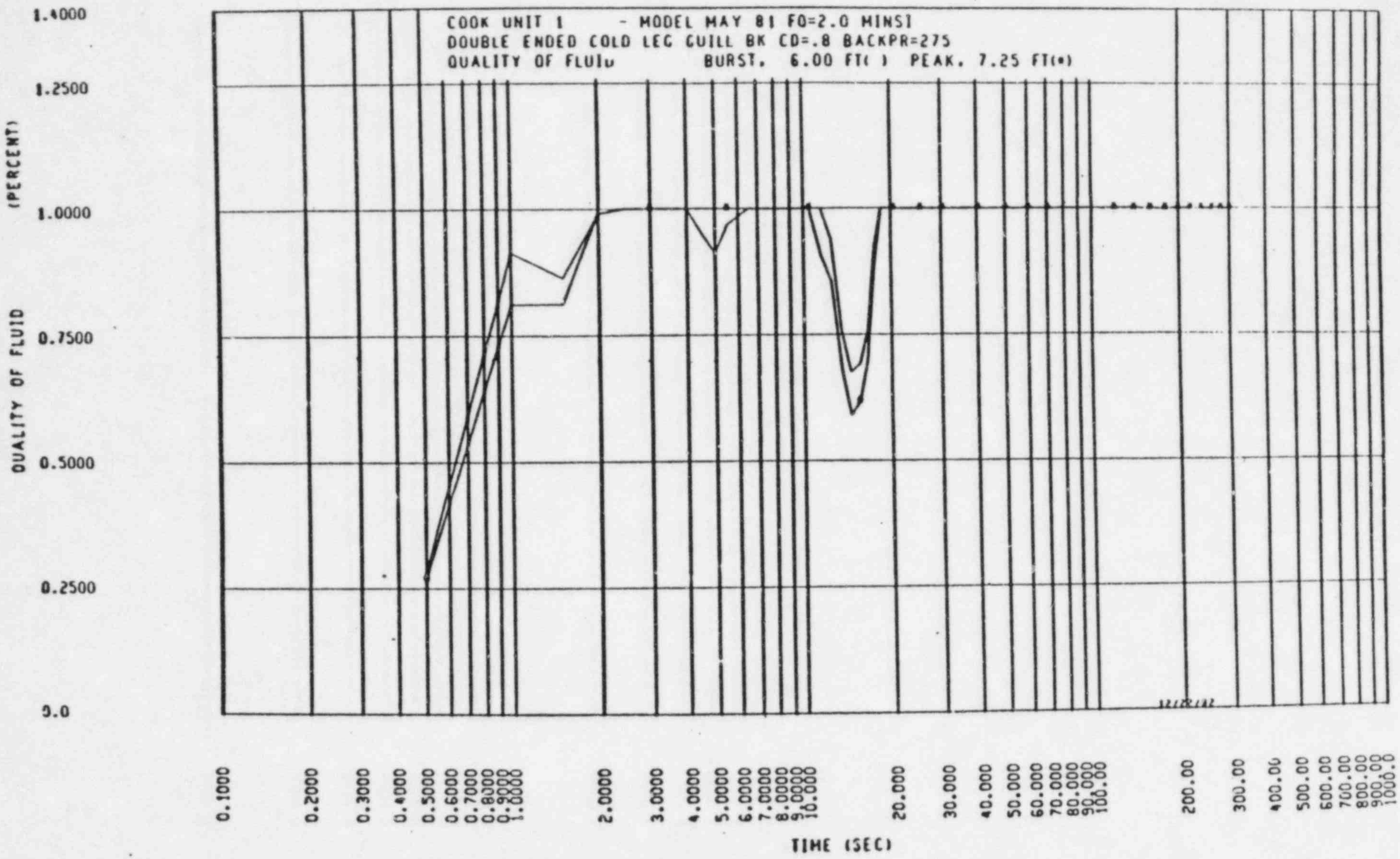


FIGURE 14.3.1-1 FLUID QUALITY
DECLG(CD = 0.8)

MIN
SI

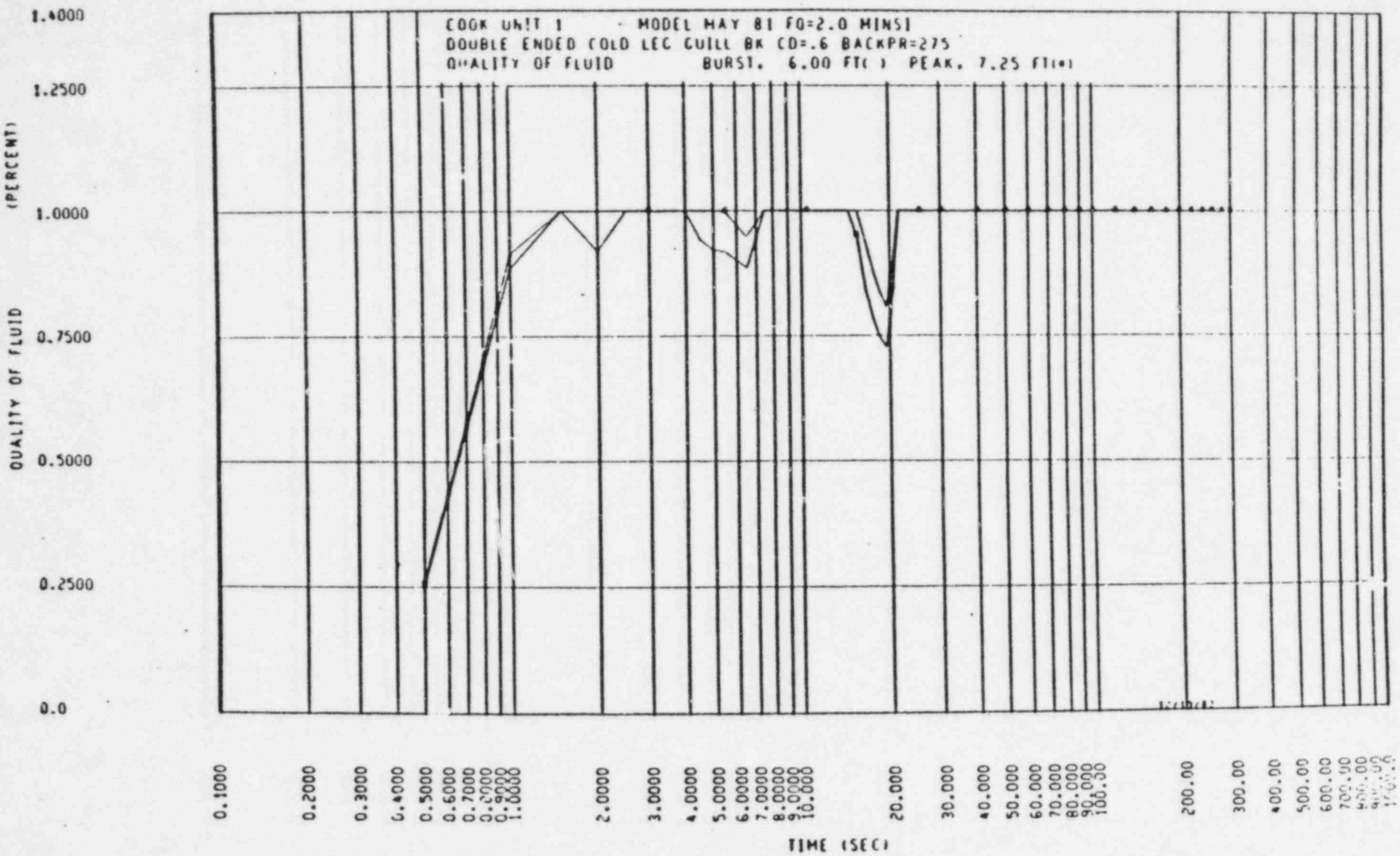


FIGURE 14.3.1-2 FLUID QUALITY DECLG(CD - 0.6) | MIN SI

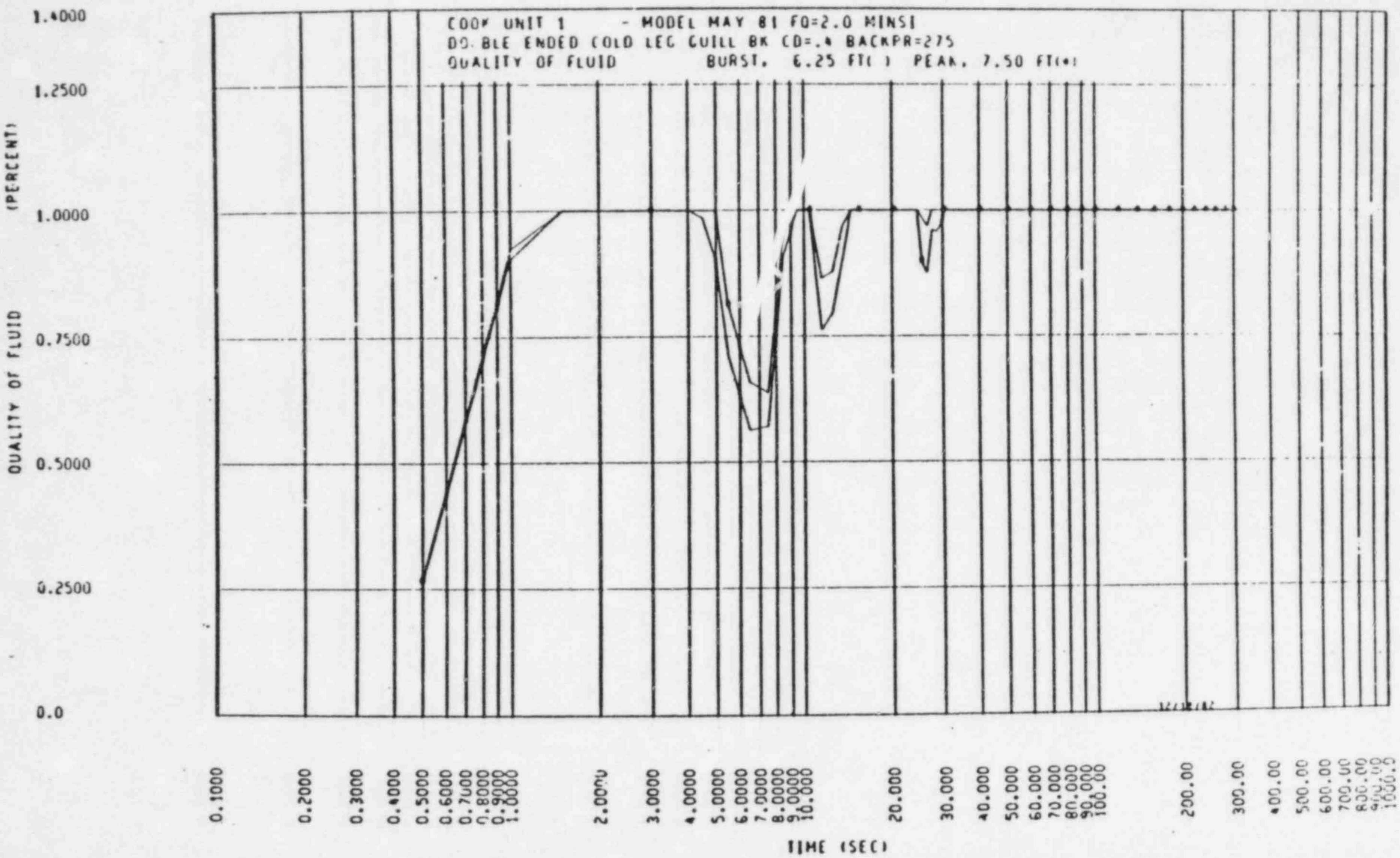


FIGURE 14.3.1-3 FLUID QUALITY MIN
DECLG(CD = 0.4) SI

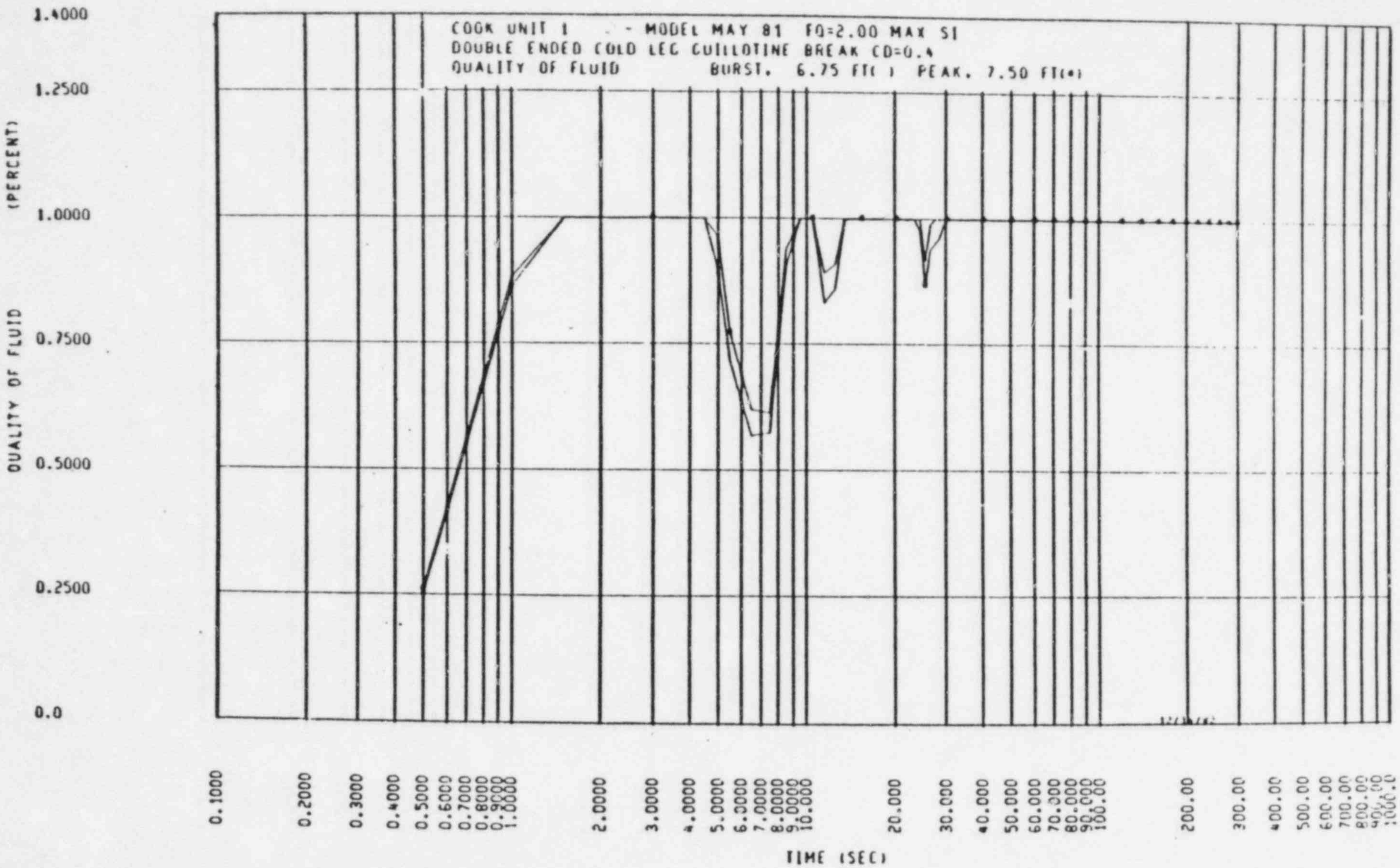


FIGURE 14.3.1-4 FLUID QUALITY MAX
DECLG(CD = 0.4) SI

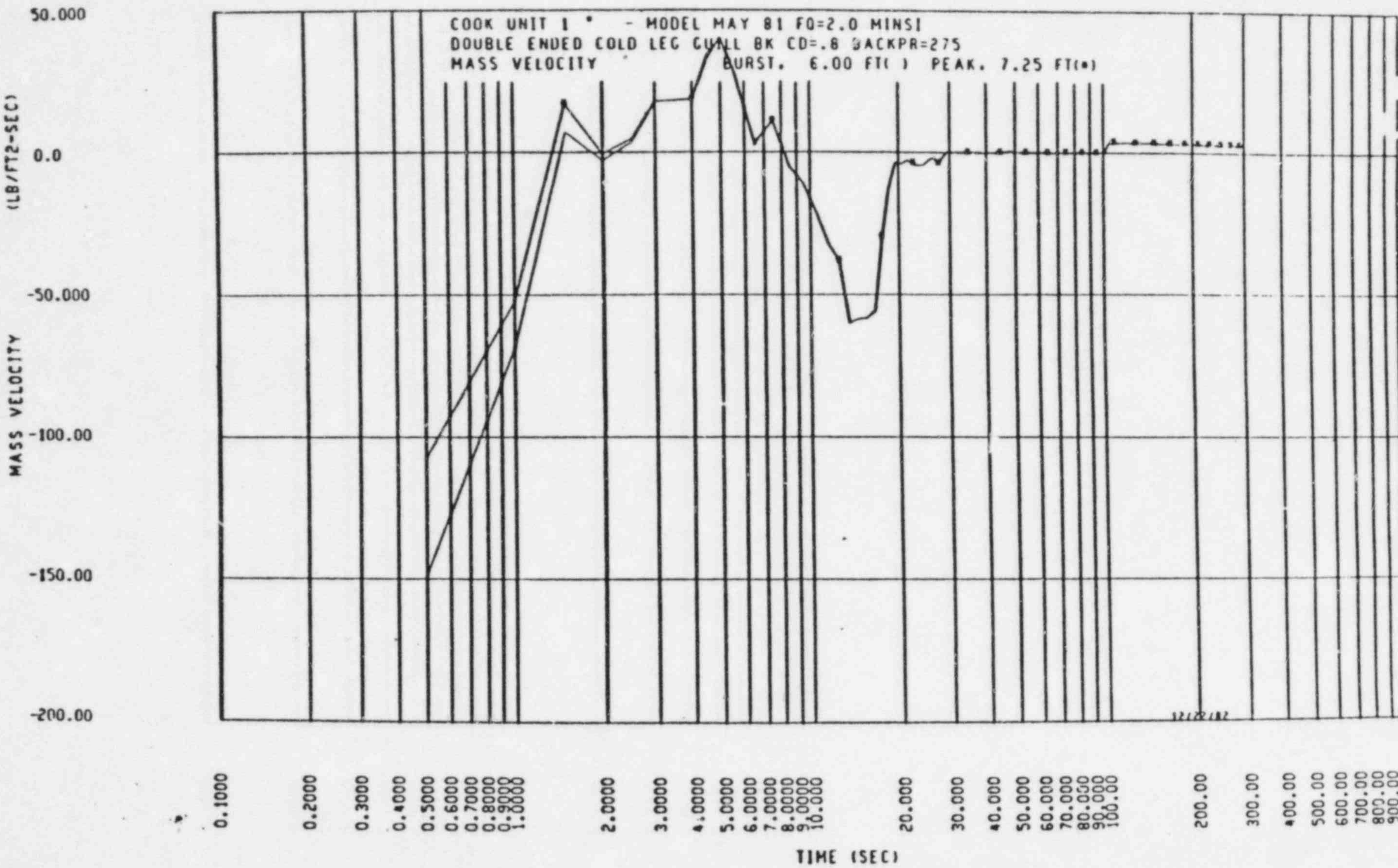


FIGURE 14.3.1-5 MASS VELOCITY DECLG(CD = 0.8) | MIN SI

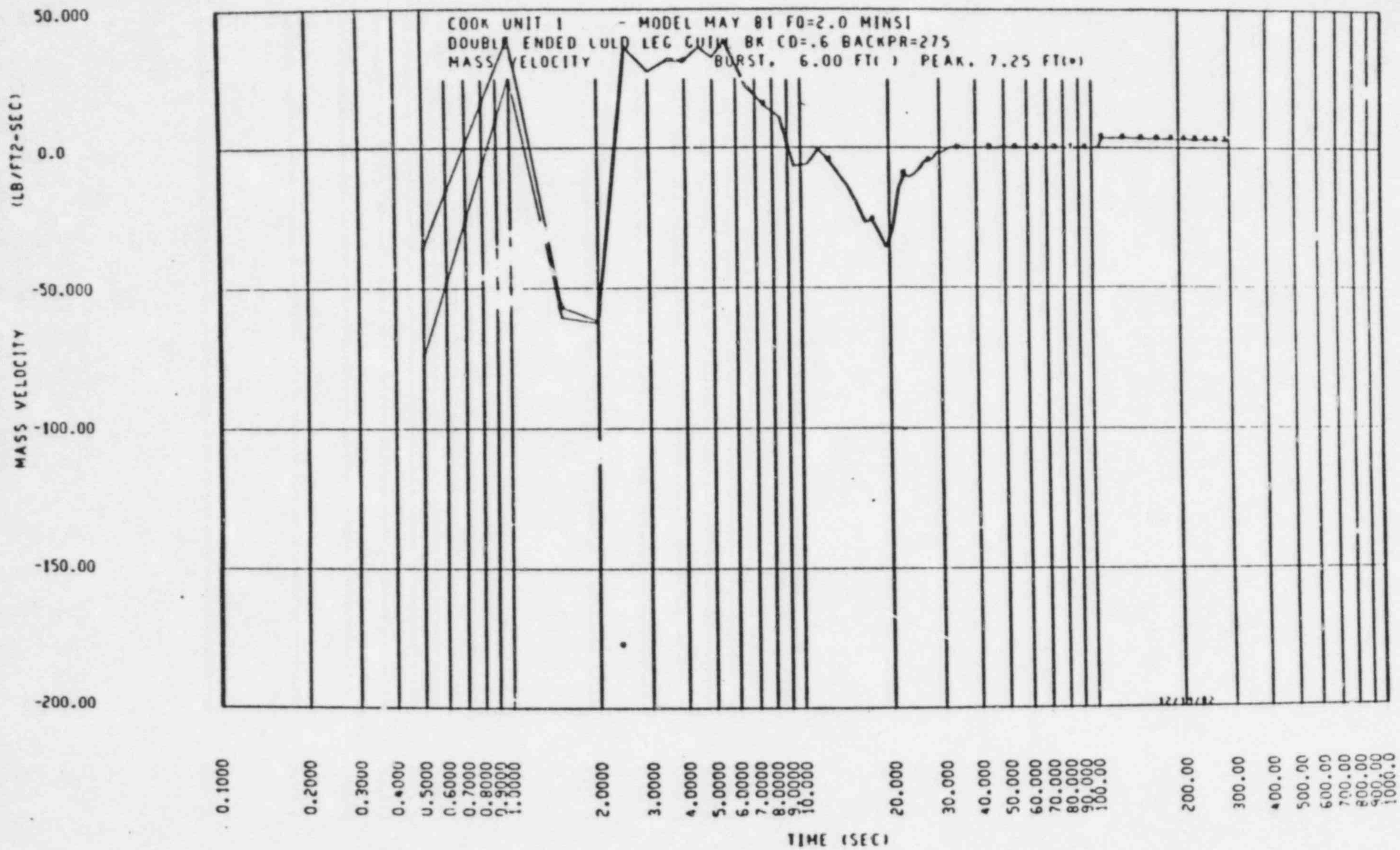


FIGURE 14.3.1-6 MASS VELOCITY DECLG(CD = 0.6) MIN SI

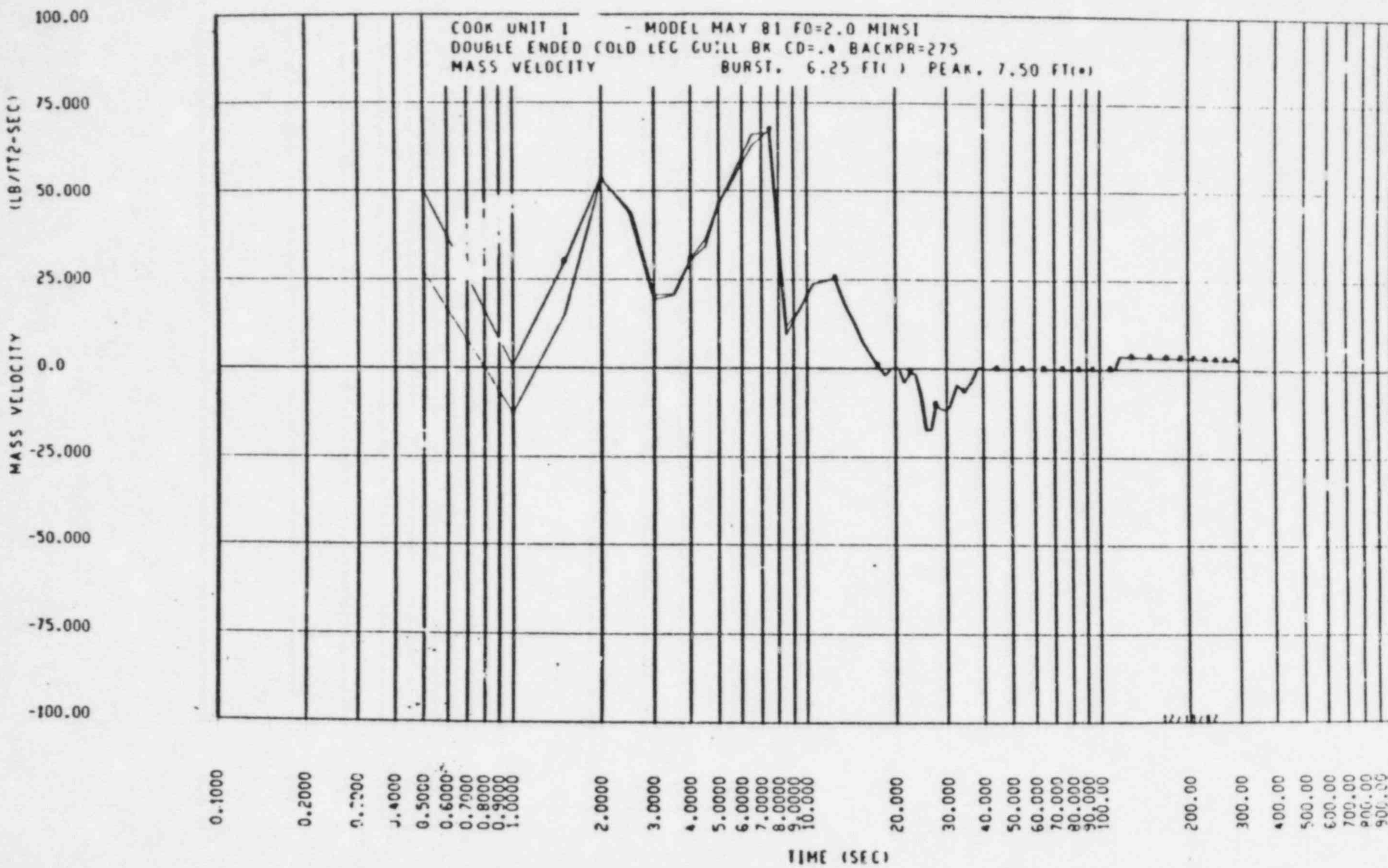


FIGURE 14.3.1-7 MASS VELOCITY MIN
DECLG(CD = 0.4) SI

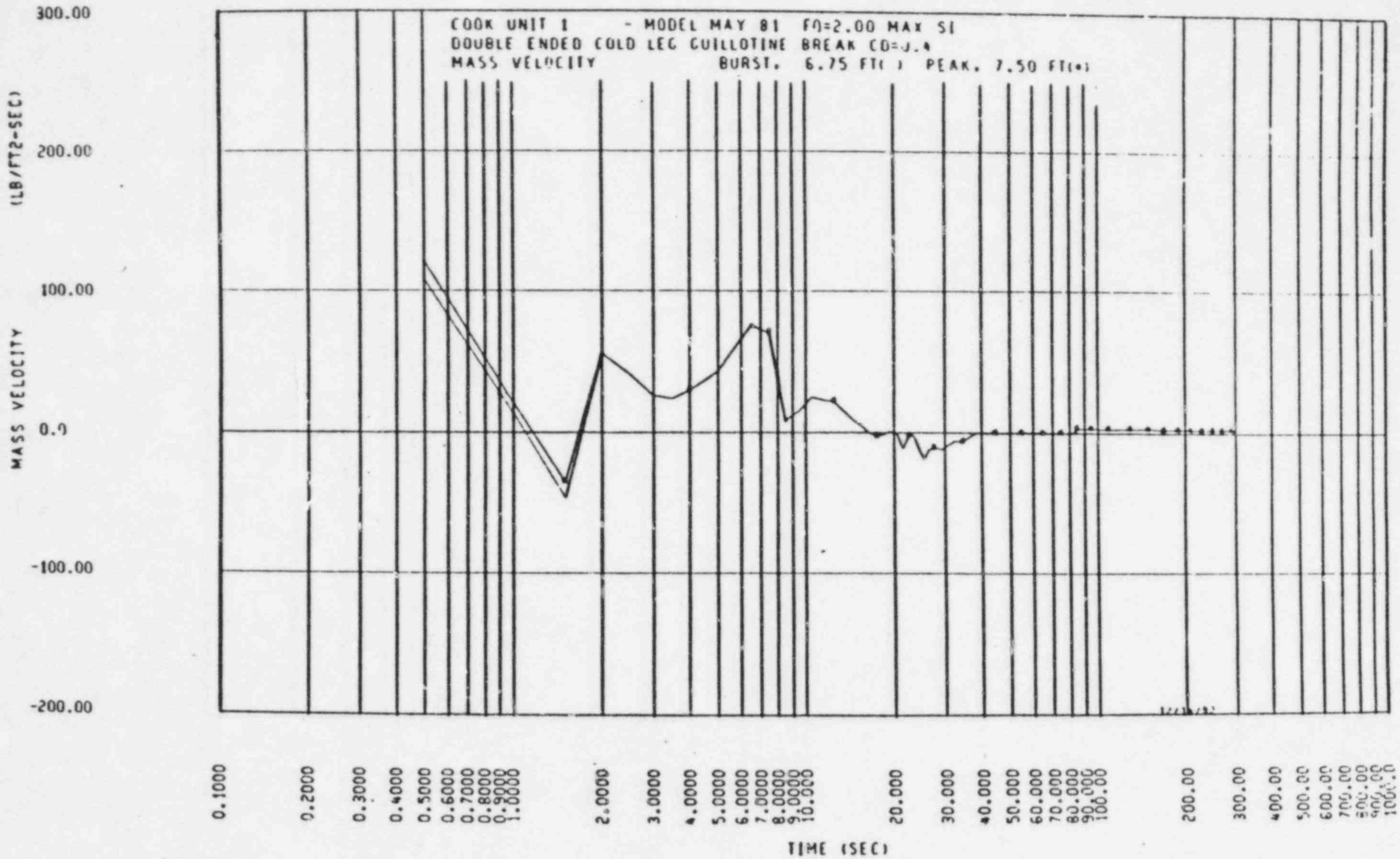


FIGURE 14.3.1-8 MASS VELOCITY MAX
DECLG(CD = 0.4) SI

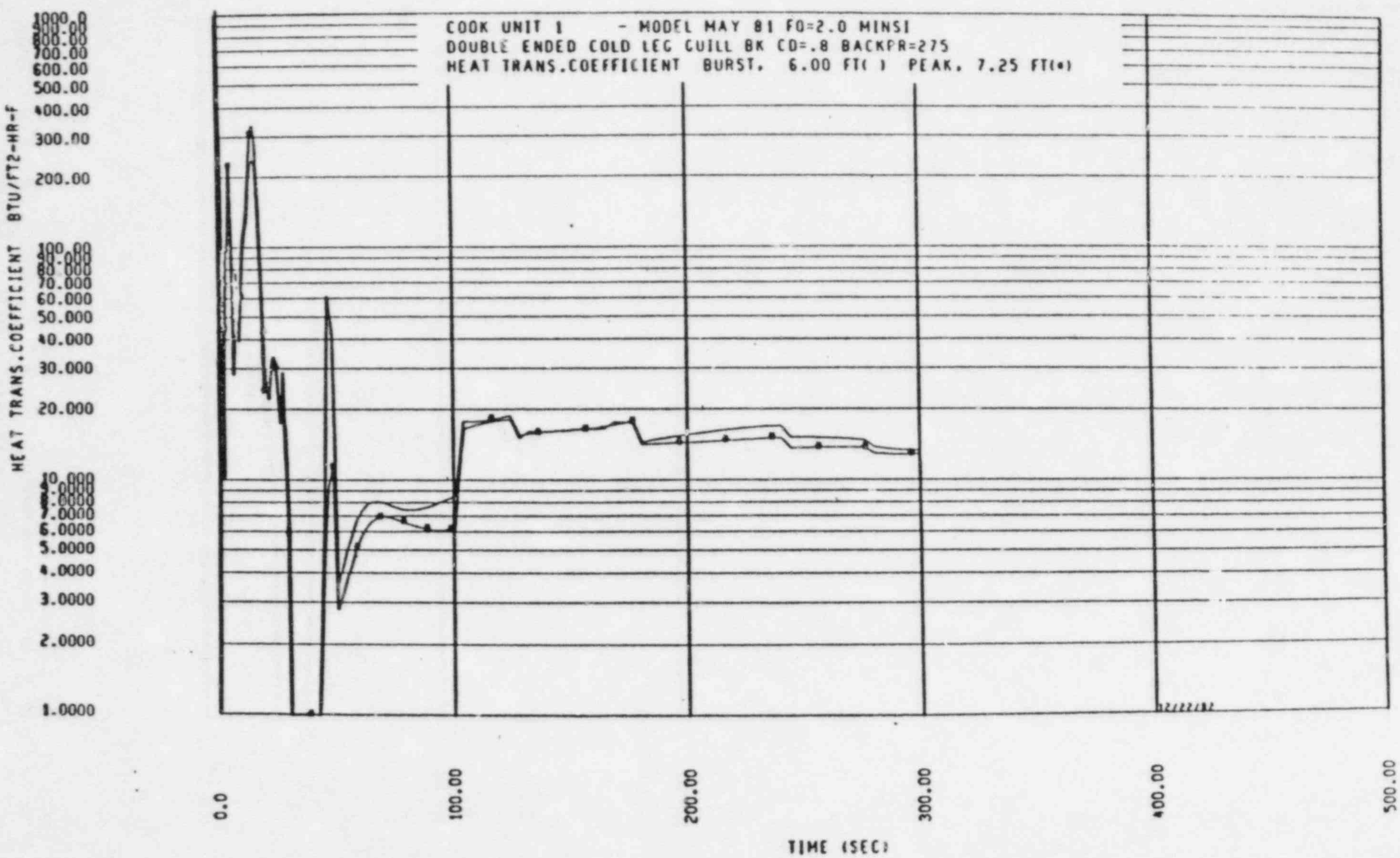


FIGURE 14.3.1-9 HEAT TRANSFER COEFFICIENT
DECLG(CD = 0.8) MIN SI

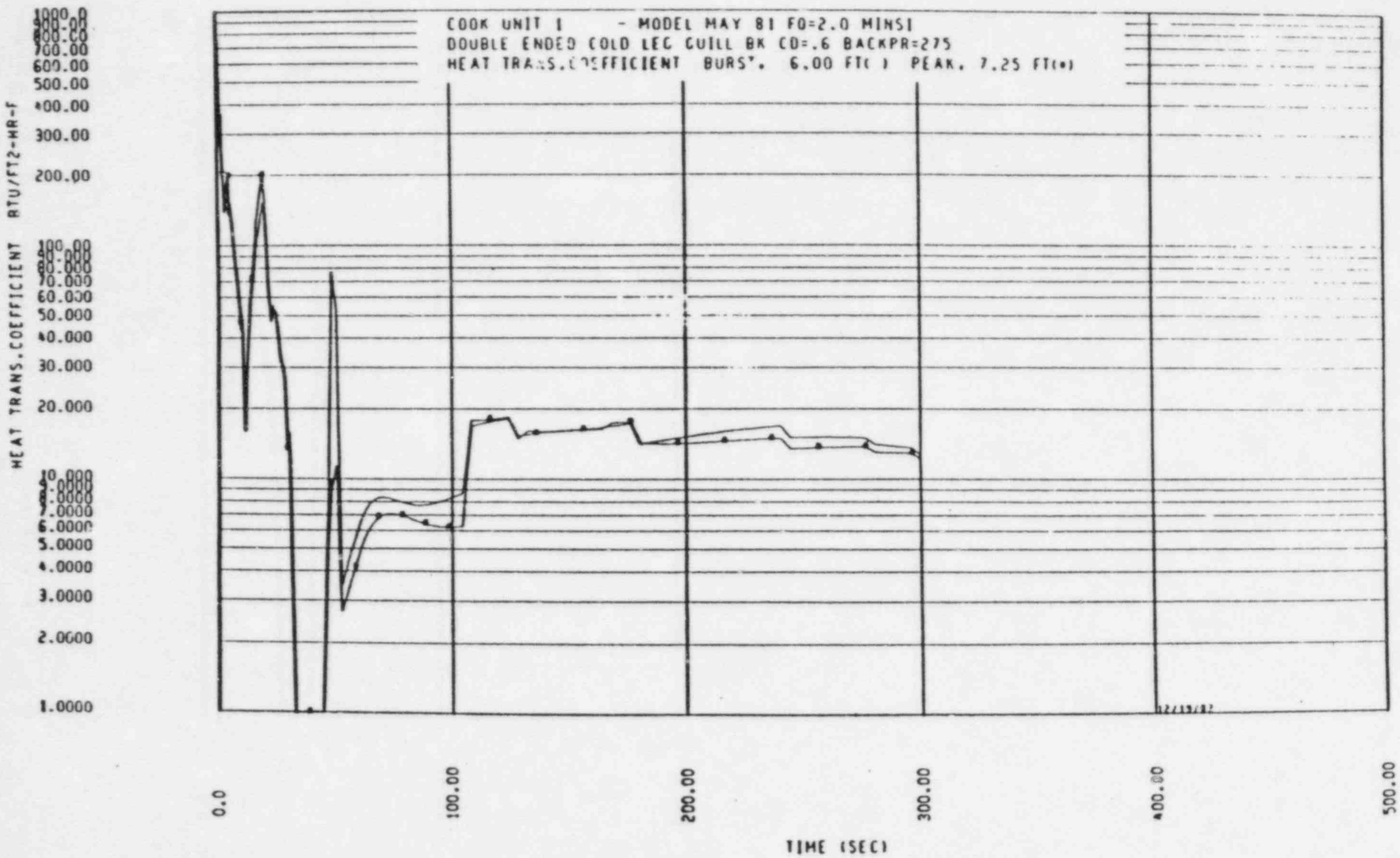


FIGURE 14.3.1-10 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.6) MIN SI

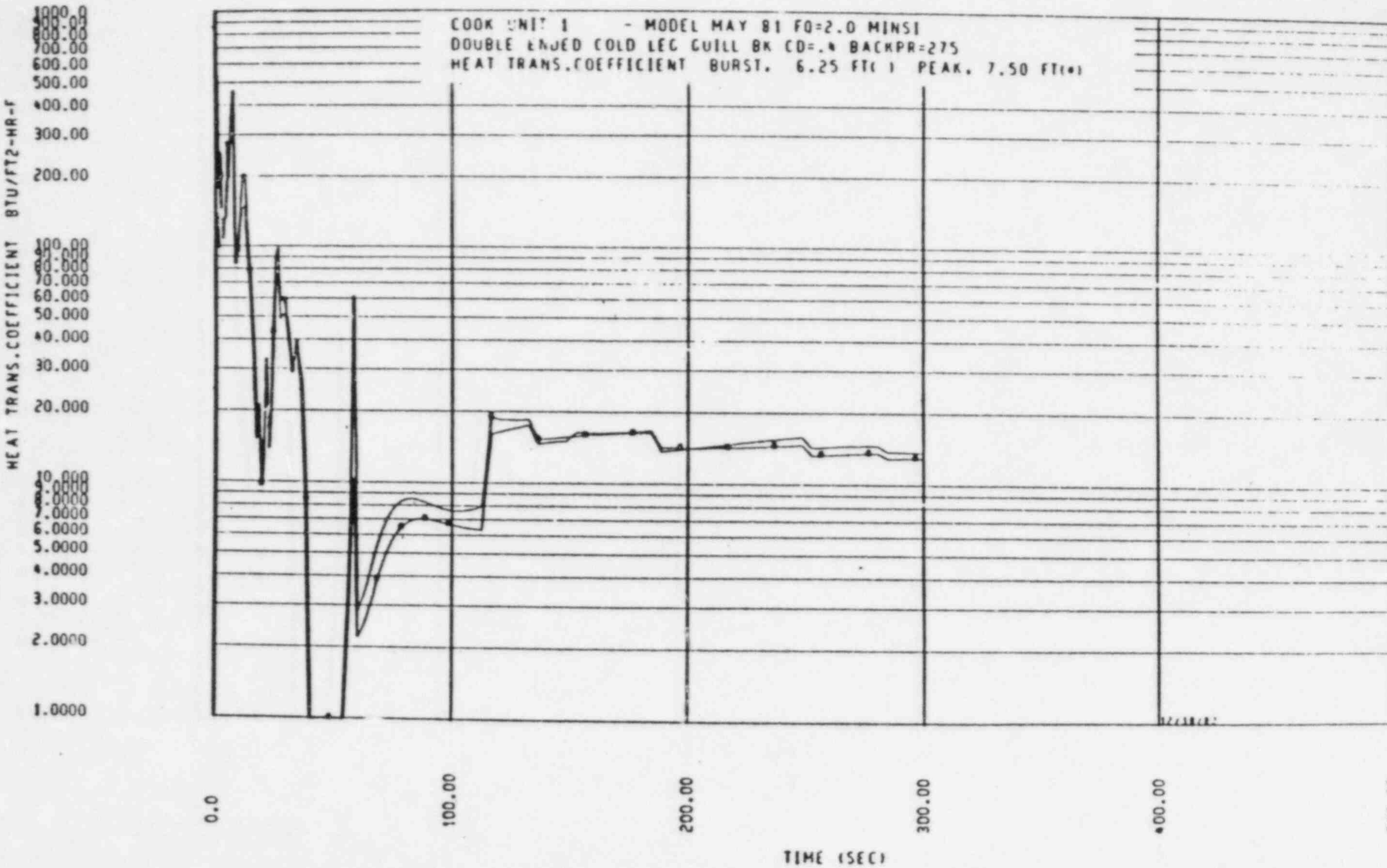


FIGURE 14.3.1-11 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.4) MIN SI

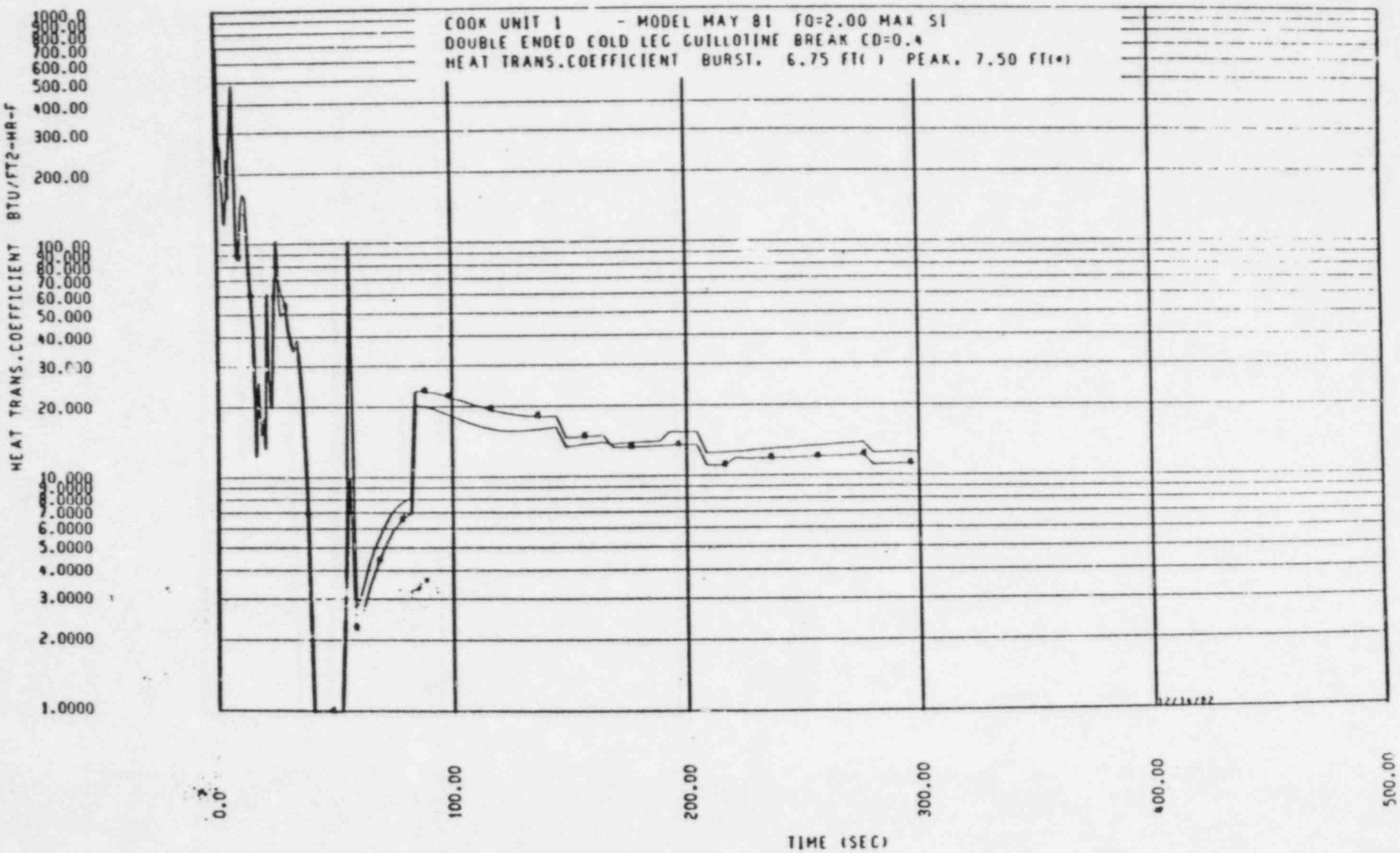


FIGURE 14.3.1-12 HEAT TRANSFER COEFFICIENT DECLG(CD=0.4) MAX SI

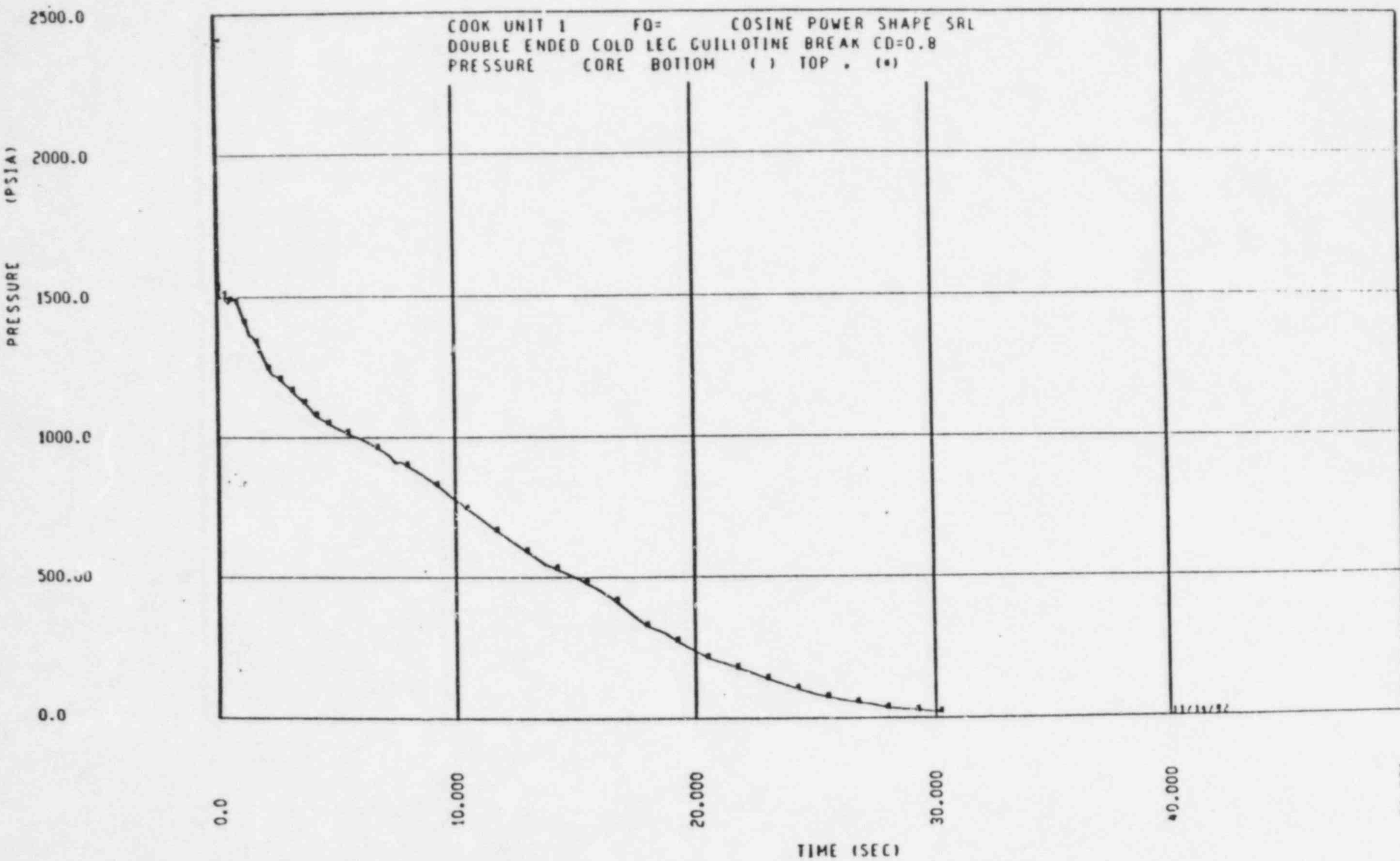


FIGURE 14.3.1-13 CORE PRESSURE MIN
DECLG(CD=0.8) SI

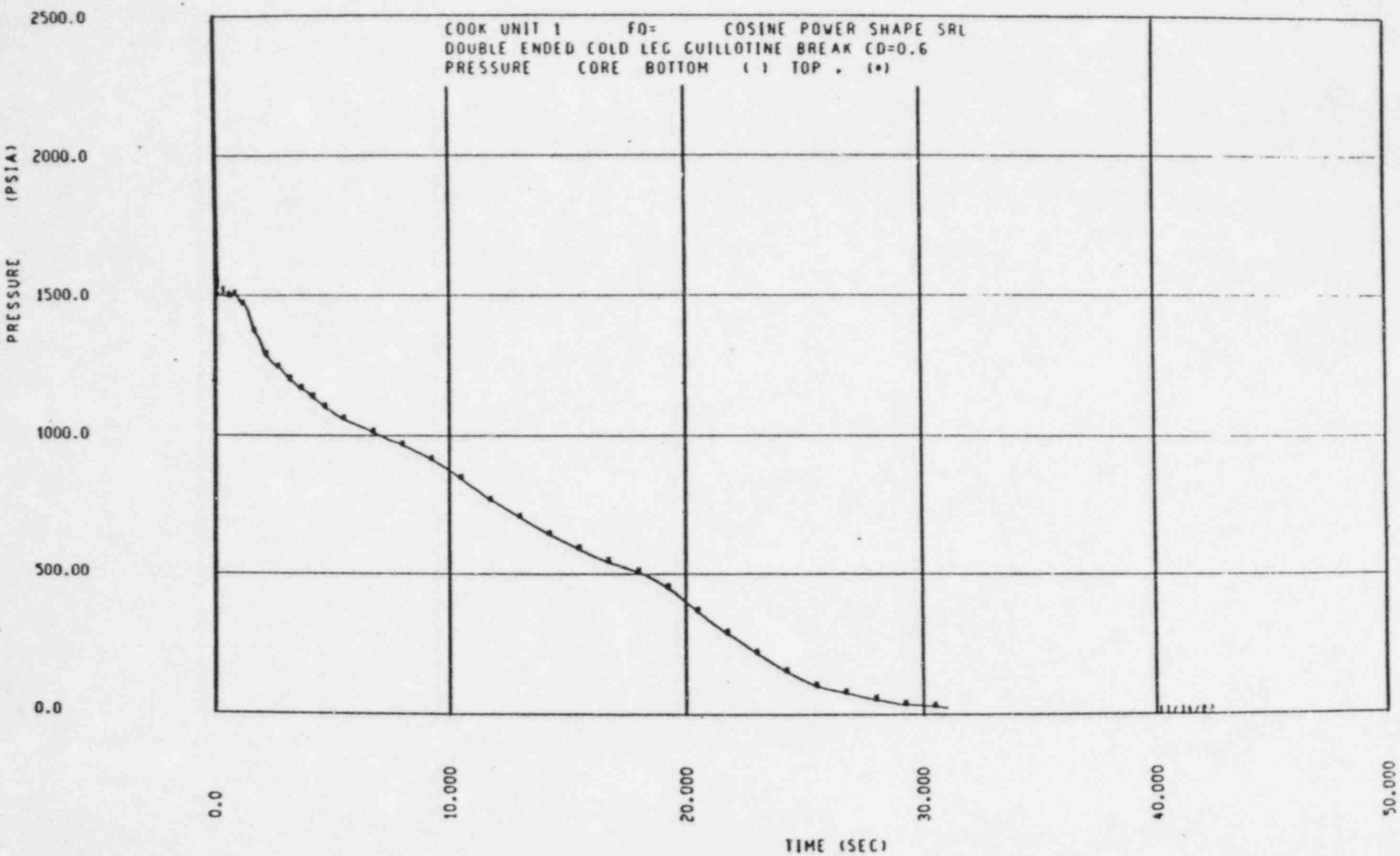


FIGURE 14.3.1-14 CORE PRESSURE DECLG(CD=0.6) MIN SI

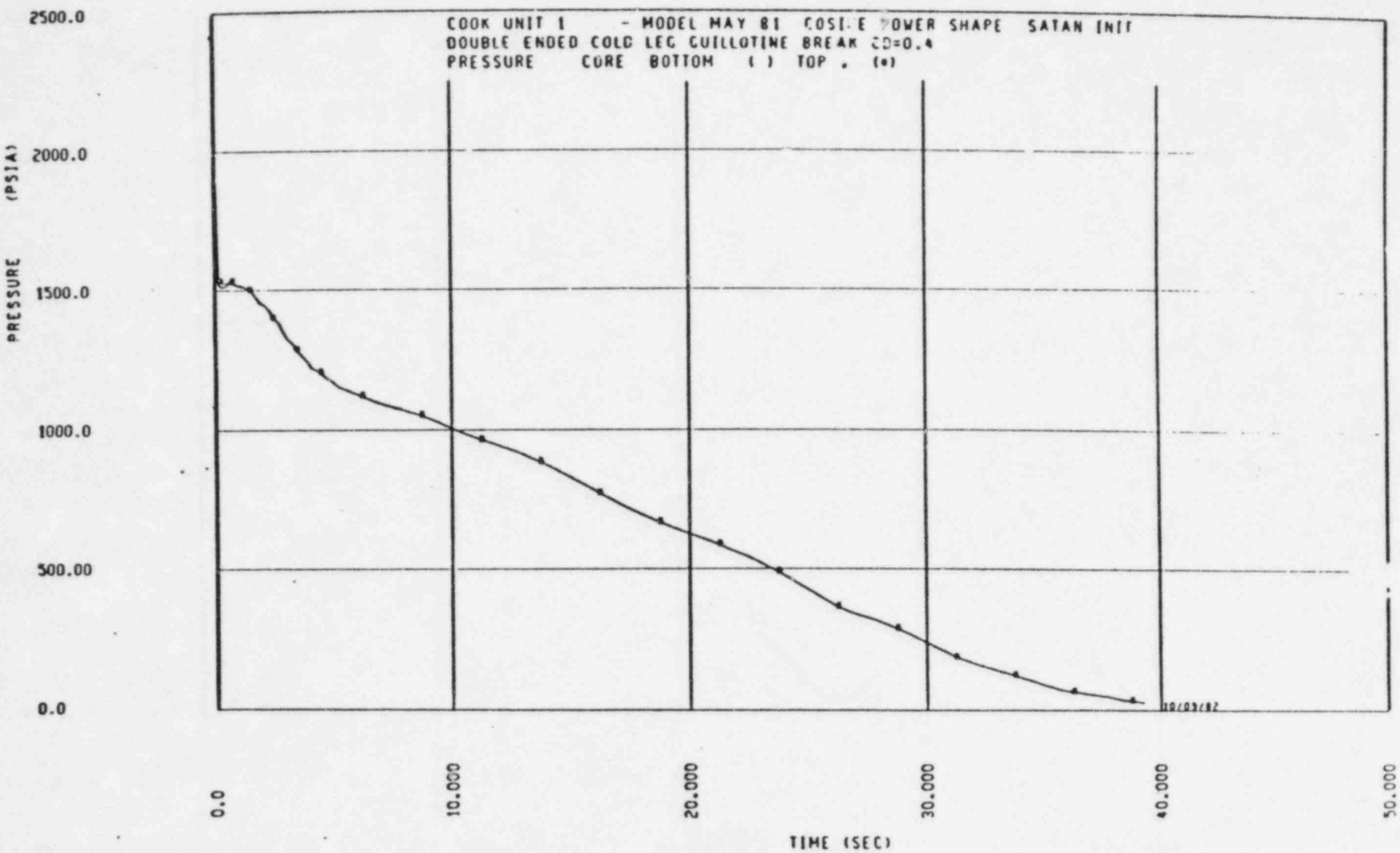


FIGURE 14.3.1-15 CORE PRESSURE DECLG(CD=0.4) MIN SI

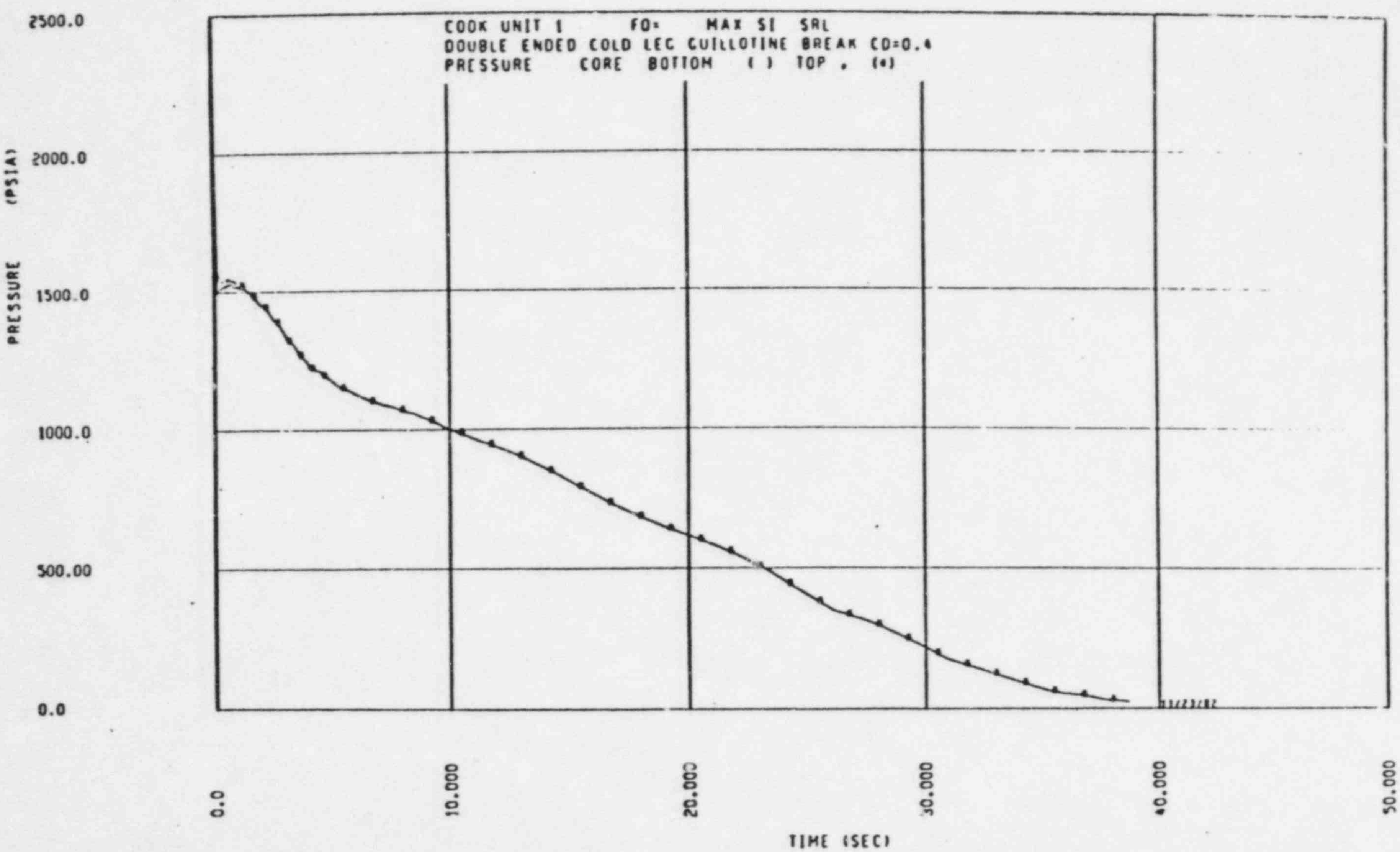


FIGURE 14.3.1-16 CORE PRESSURE MAX SI
DECLG(CD = 0.4)

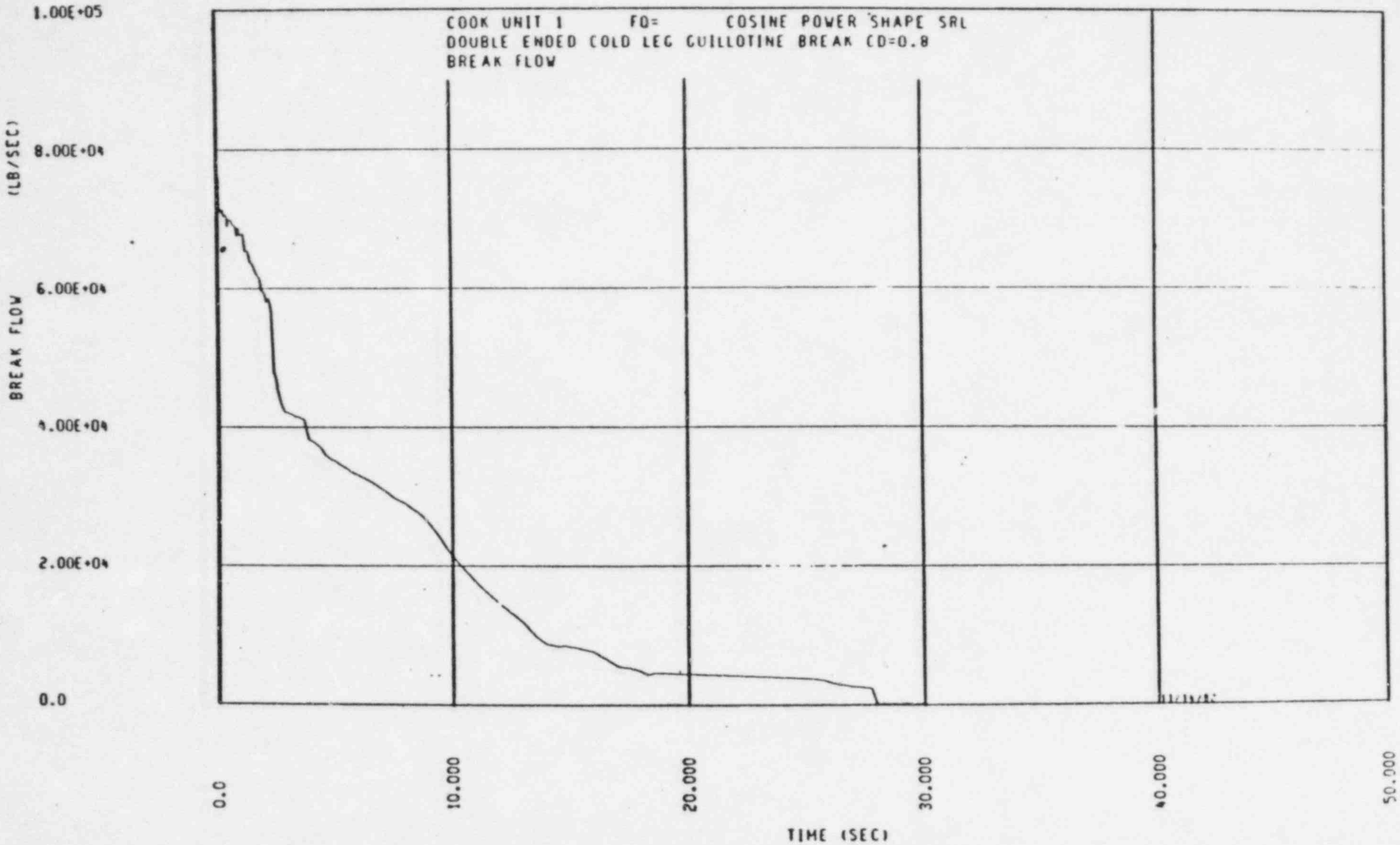


FIGURE 14.3.1-17 BREAK FLOW RATE MIN
DECLG (CD=0.8) SI

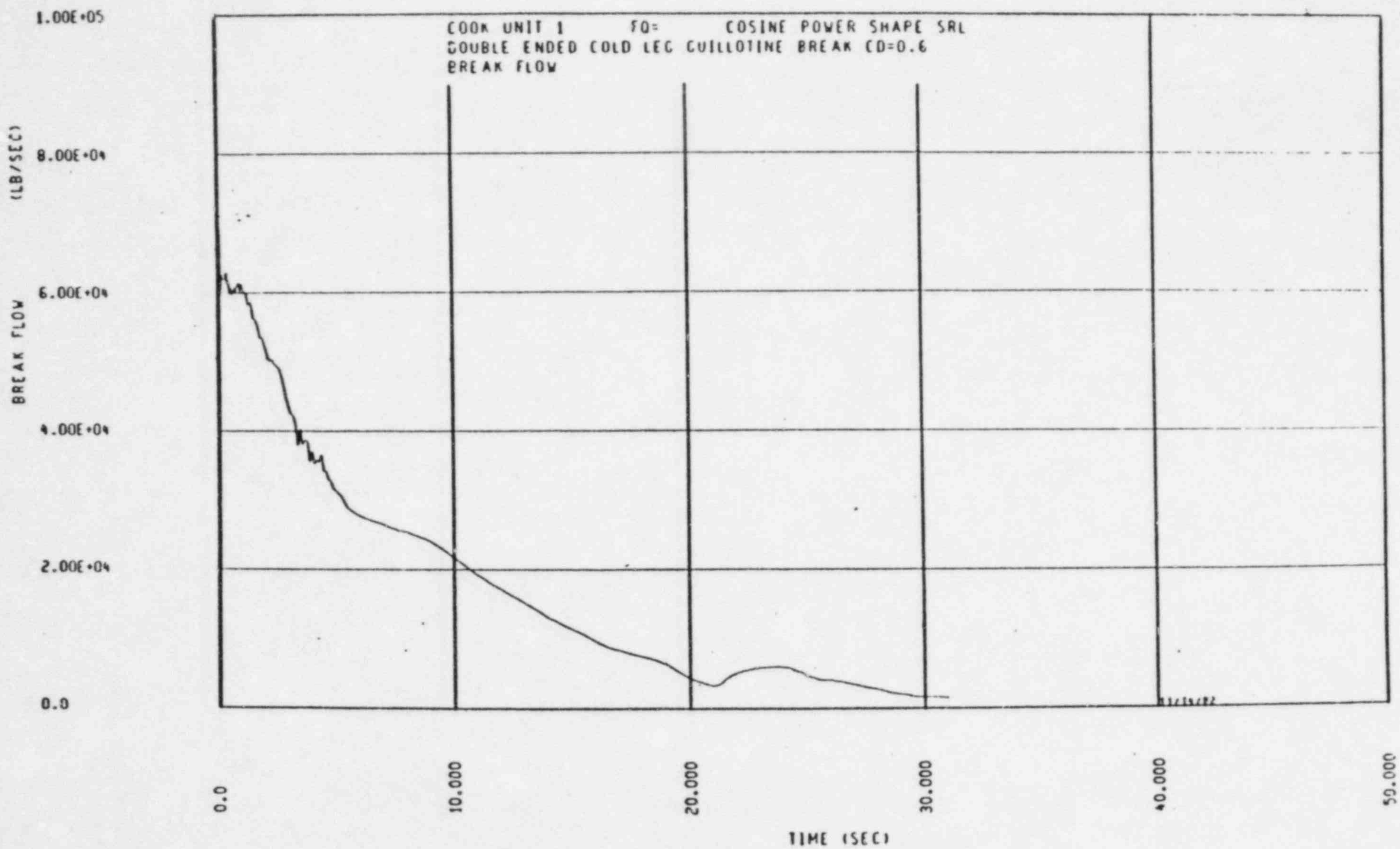


FIGURE 14.3.1-18 BREAK FLOW RATE DECLG(CD=0.6) MIN SI

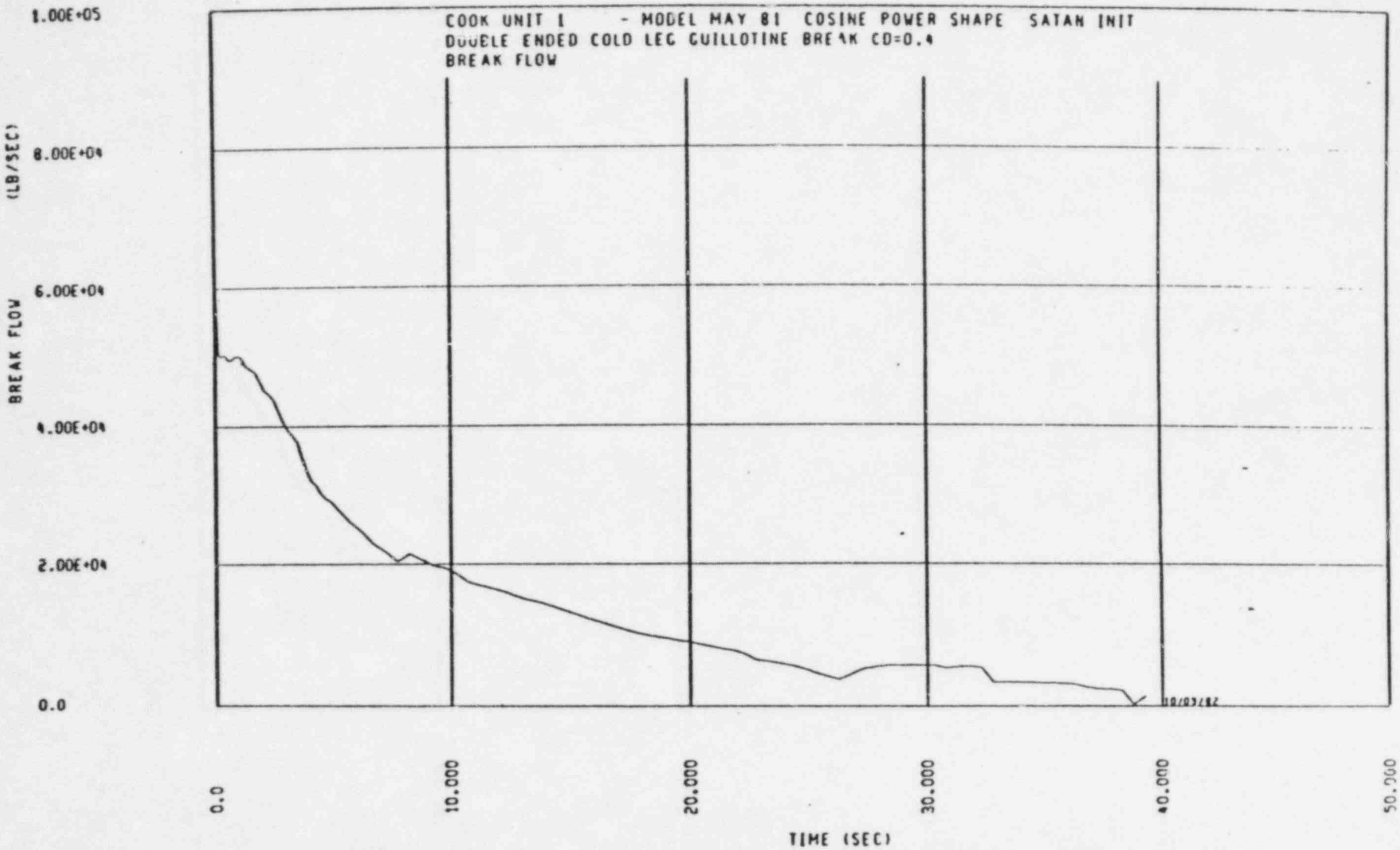


FIGURE 14.3.1-19 BREAK FLOW RATE MIN
DECLG(CD=0.4) SI

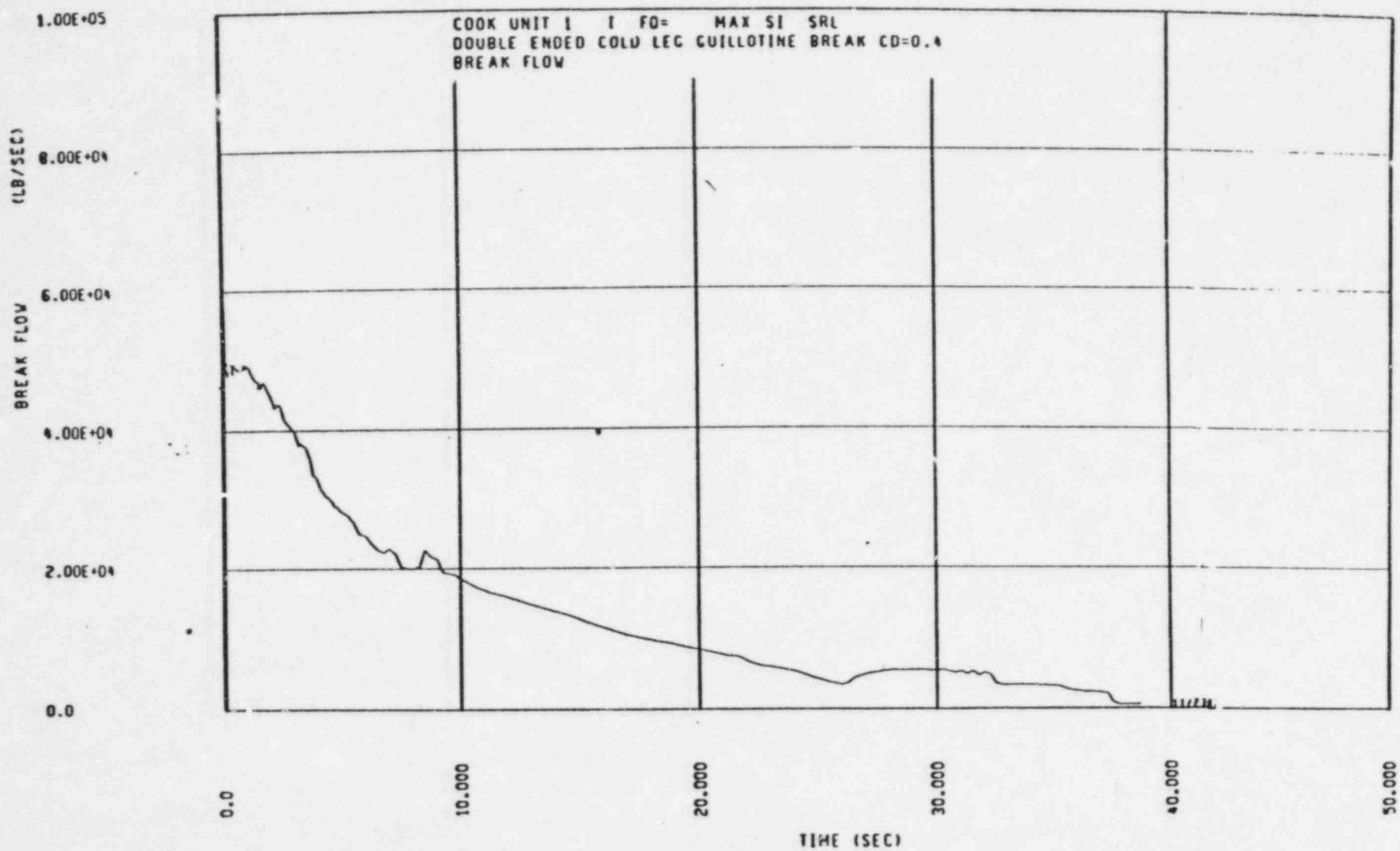


FIGURE 14.3.1-20 BREAK FLOW RATE MAX
DECLG(CD=0.4) SI

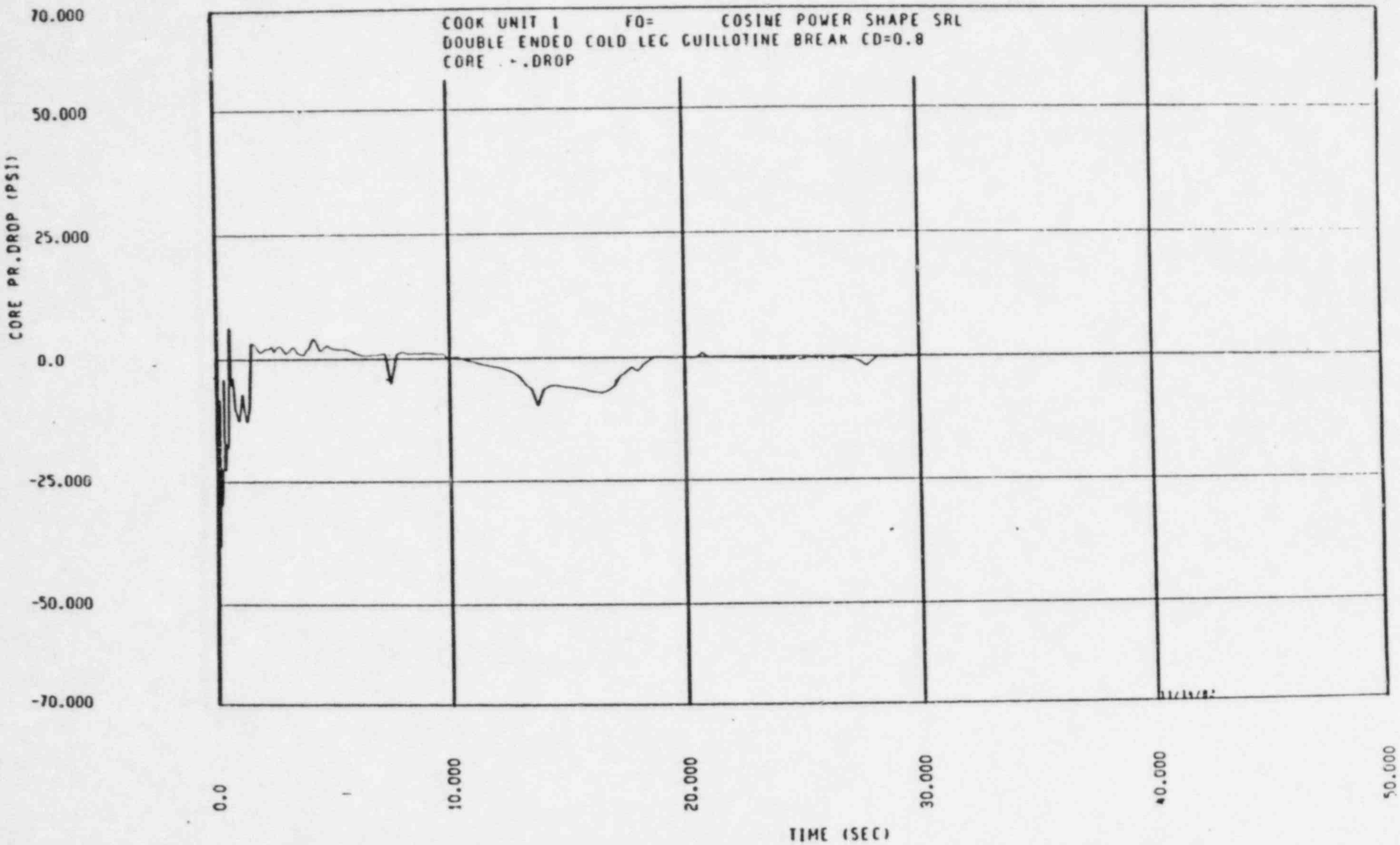


FIGURE 14.3.1-21 CORE PRESSURE DROP MIN
DECLG(CD = 0.8) SI

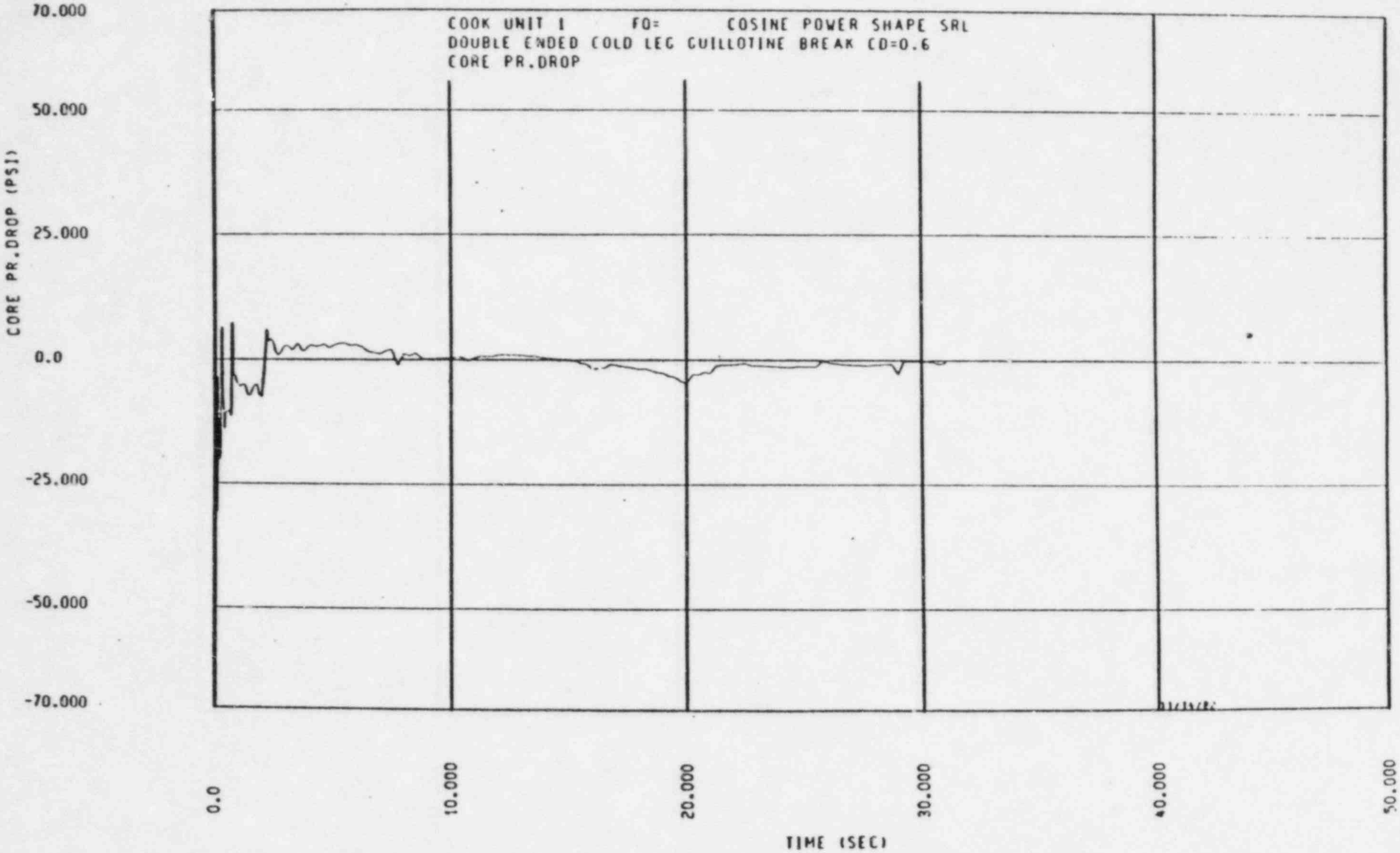


FIGURE 14.3.1-22 CORE PRESSURE DROP MIN
DECLG(CD=0.6) SI

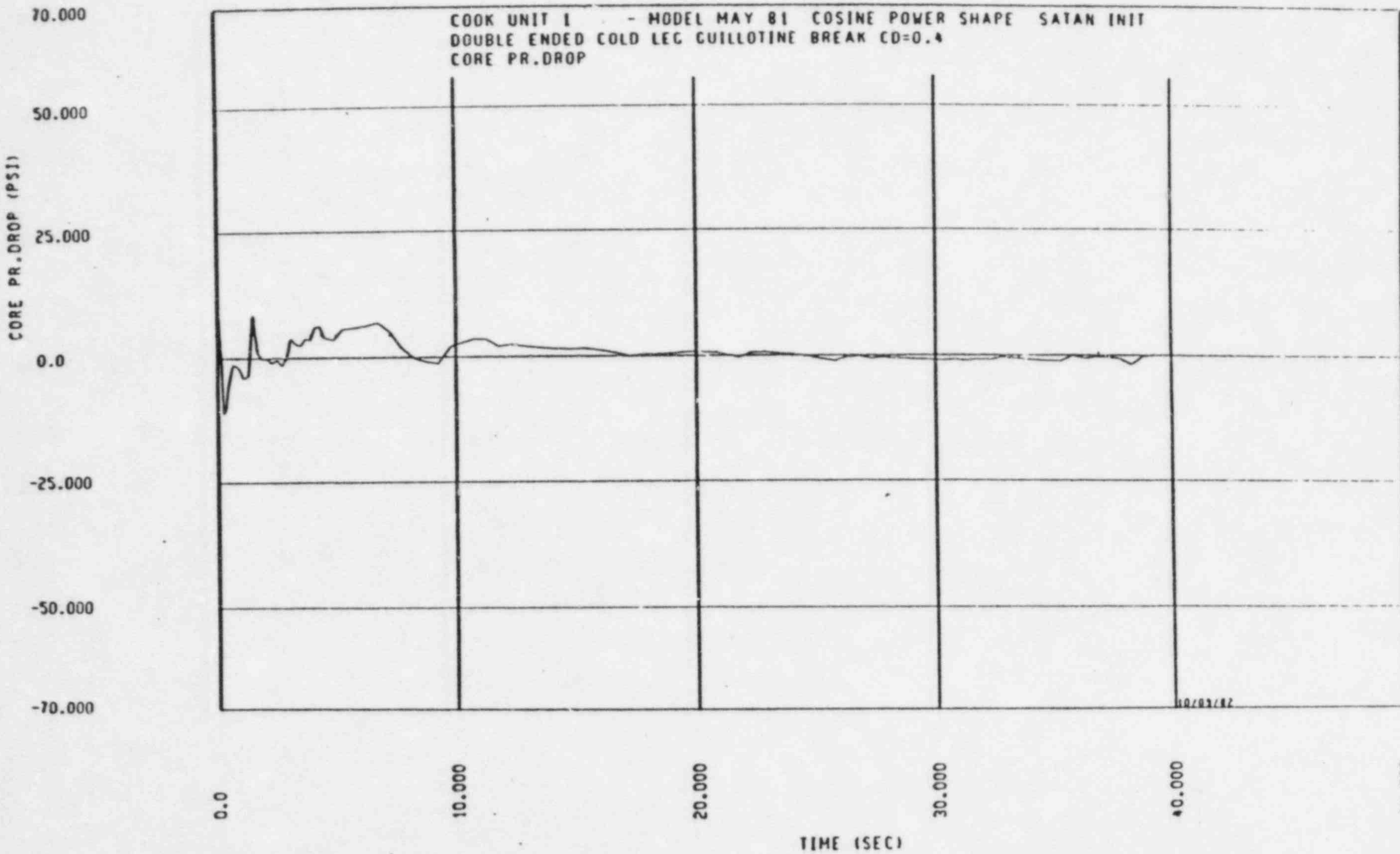


FIGURE 14.3.1-23 CORE PRESSURE DROP DECLG(CD = 0.4)

MIN SI

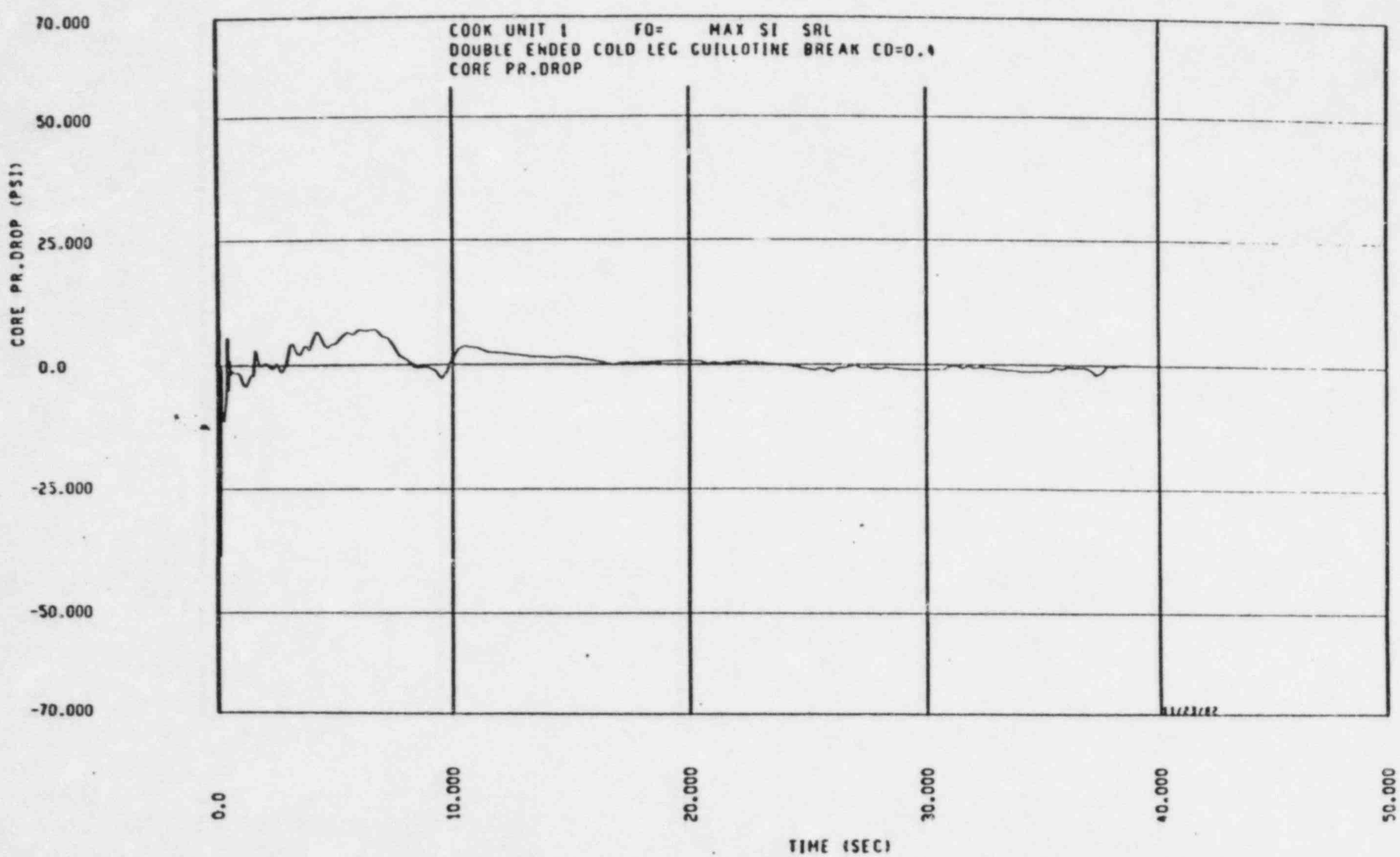


FIGURE 14.3.1-24 CORE PRESSURE DROP DECLG(CD = 0.4) MAX SI

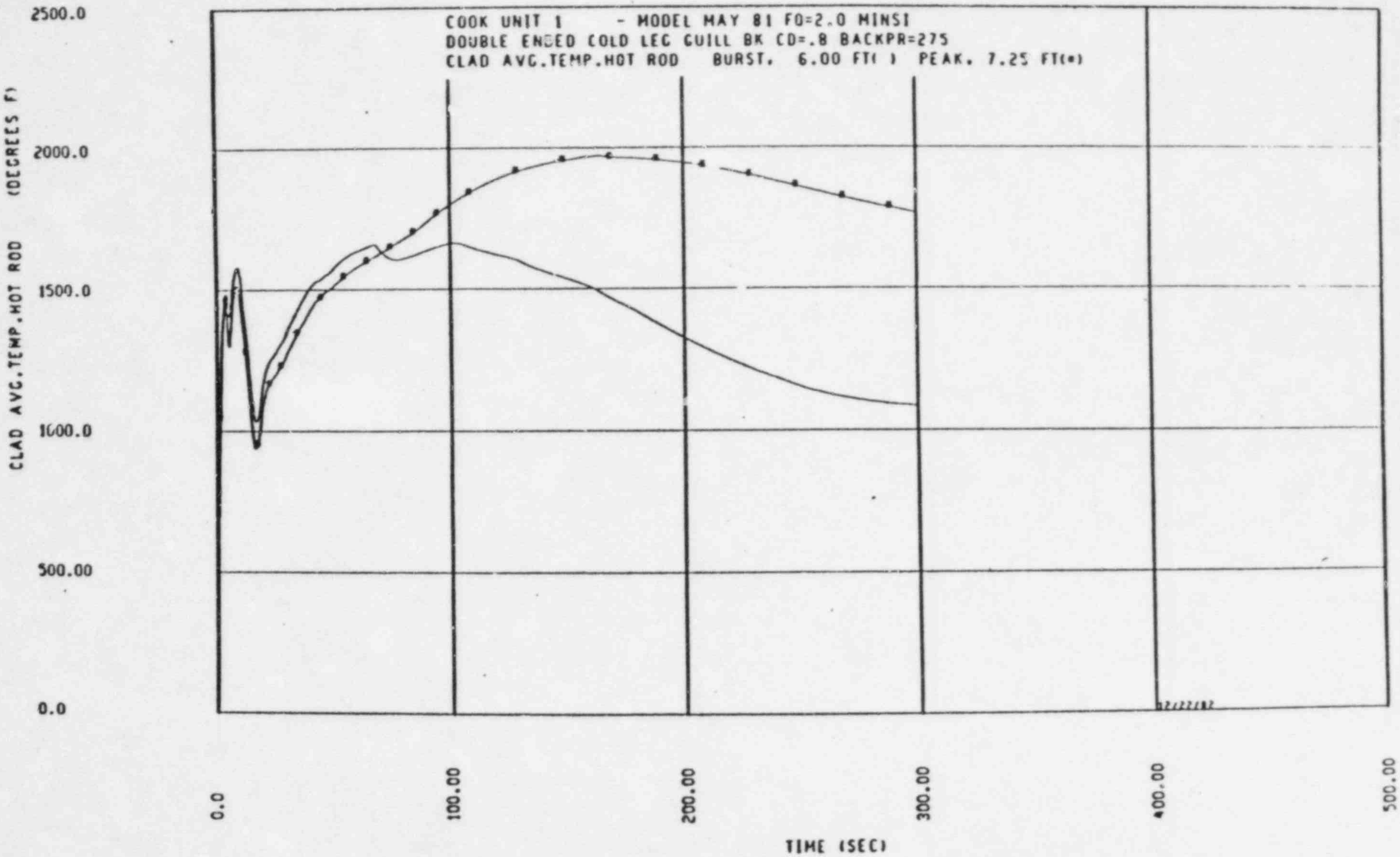


FIGURE 14.3.1-25 PEAK CLAD TEMPERATURE DECLG(CD = 0.8)

MIN SI

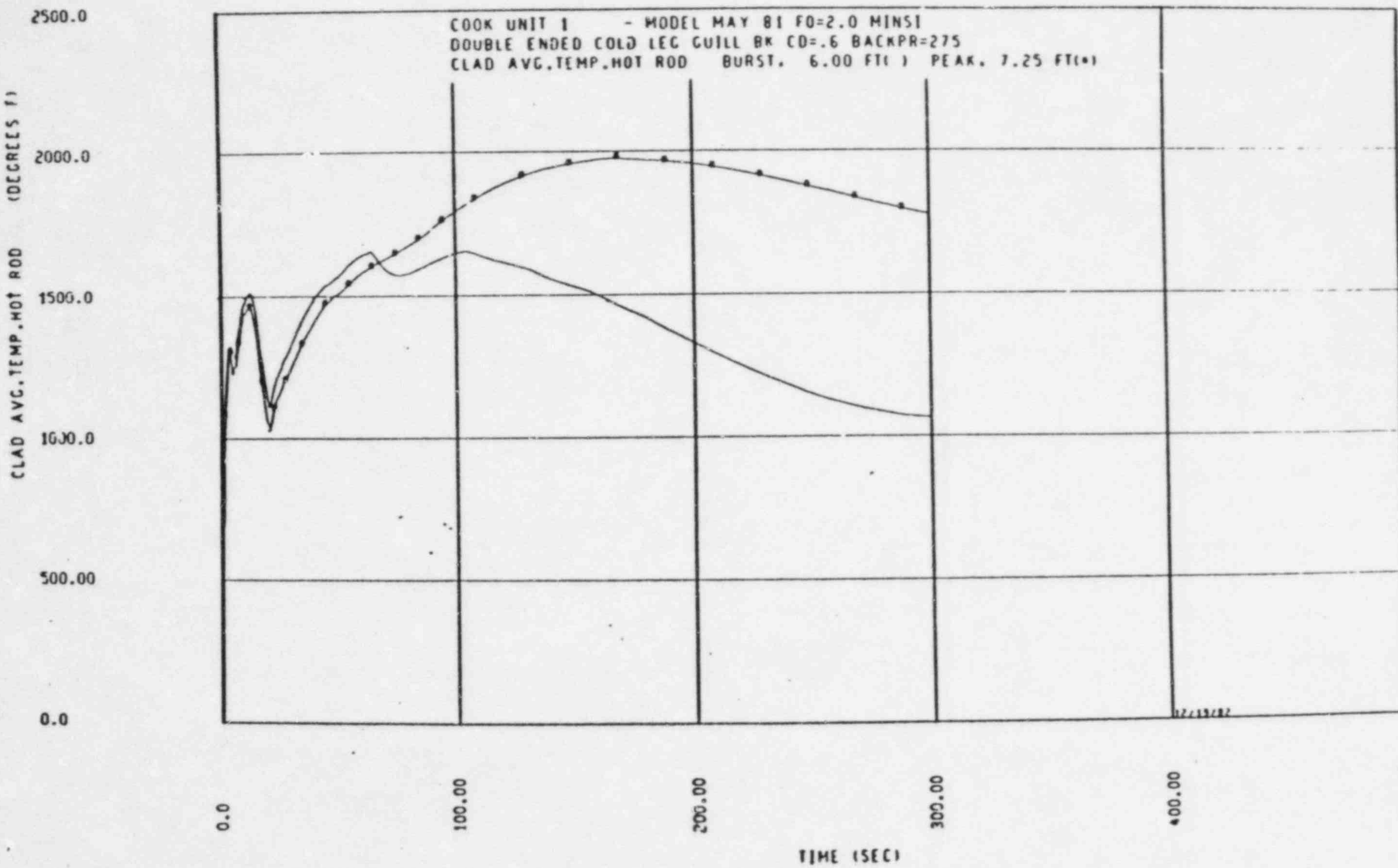


FIGURE 14.3.1-26 PEAK CLAD TEMPERATURE MIN
DECLG(CD = 0.6) SI

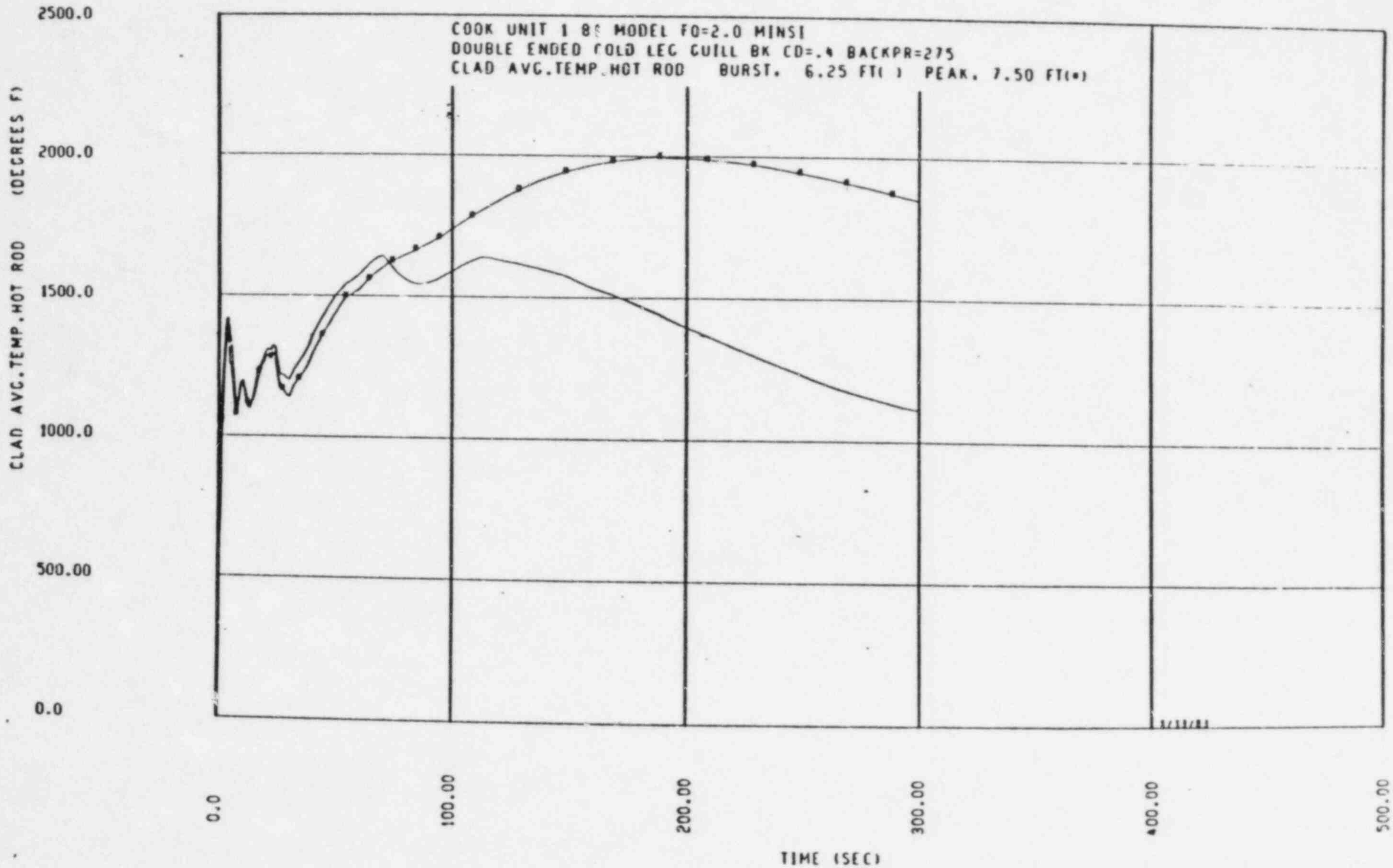


FIGURE 14.3.1-27 PEAK CLAD TEMPERATURE MIN SI
DECLG(CD = 0.4)

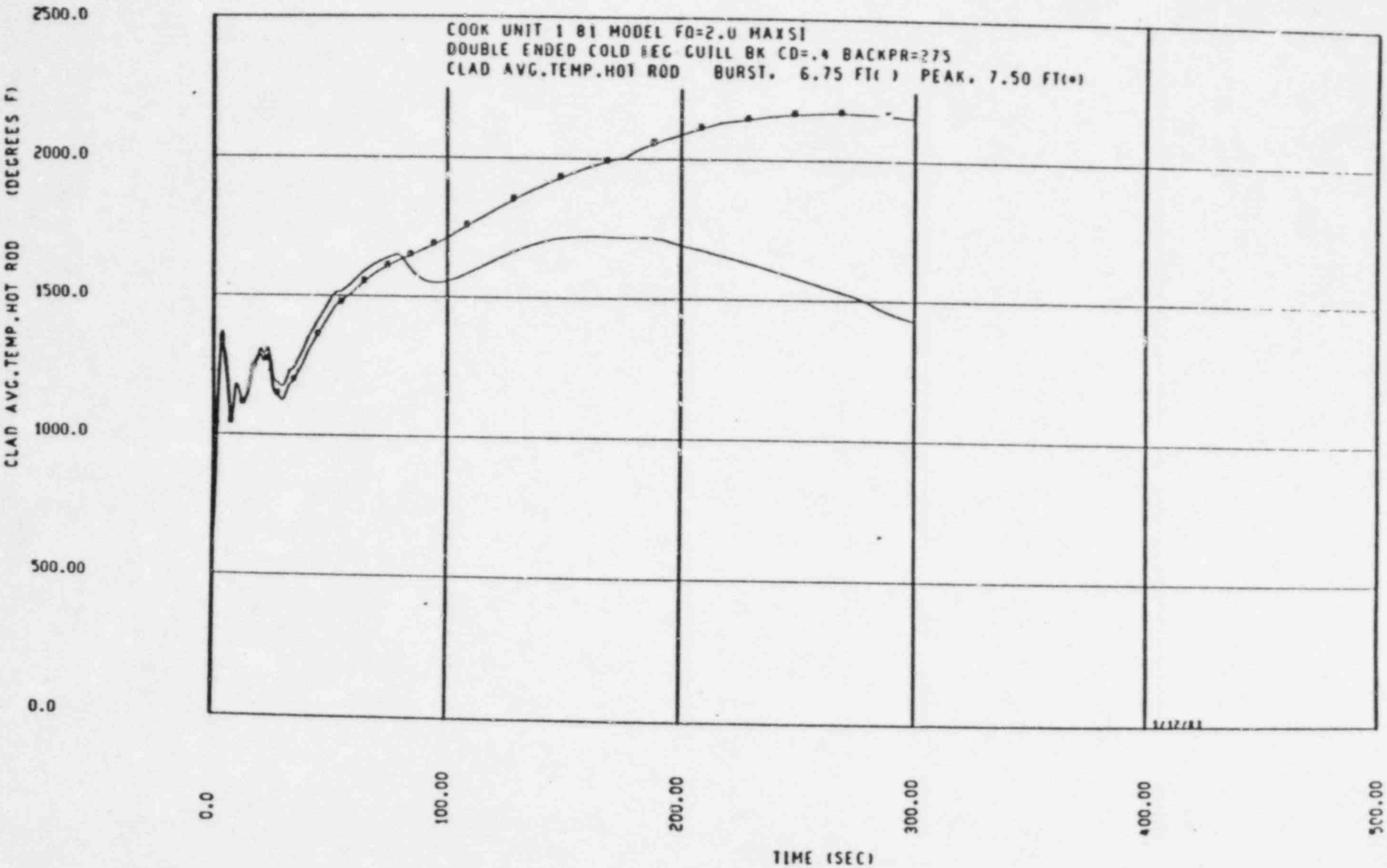


FIGURE 14.3.1-28 PEAK CLAD TEMPERAT URE DECLG(CD = 0.4)

MAX SI

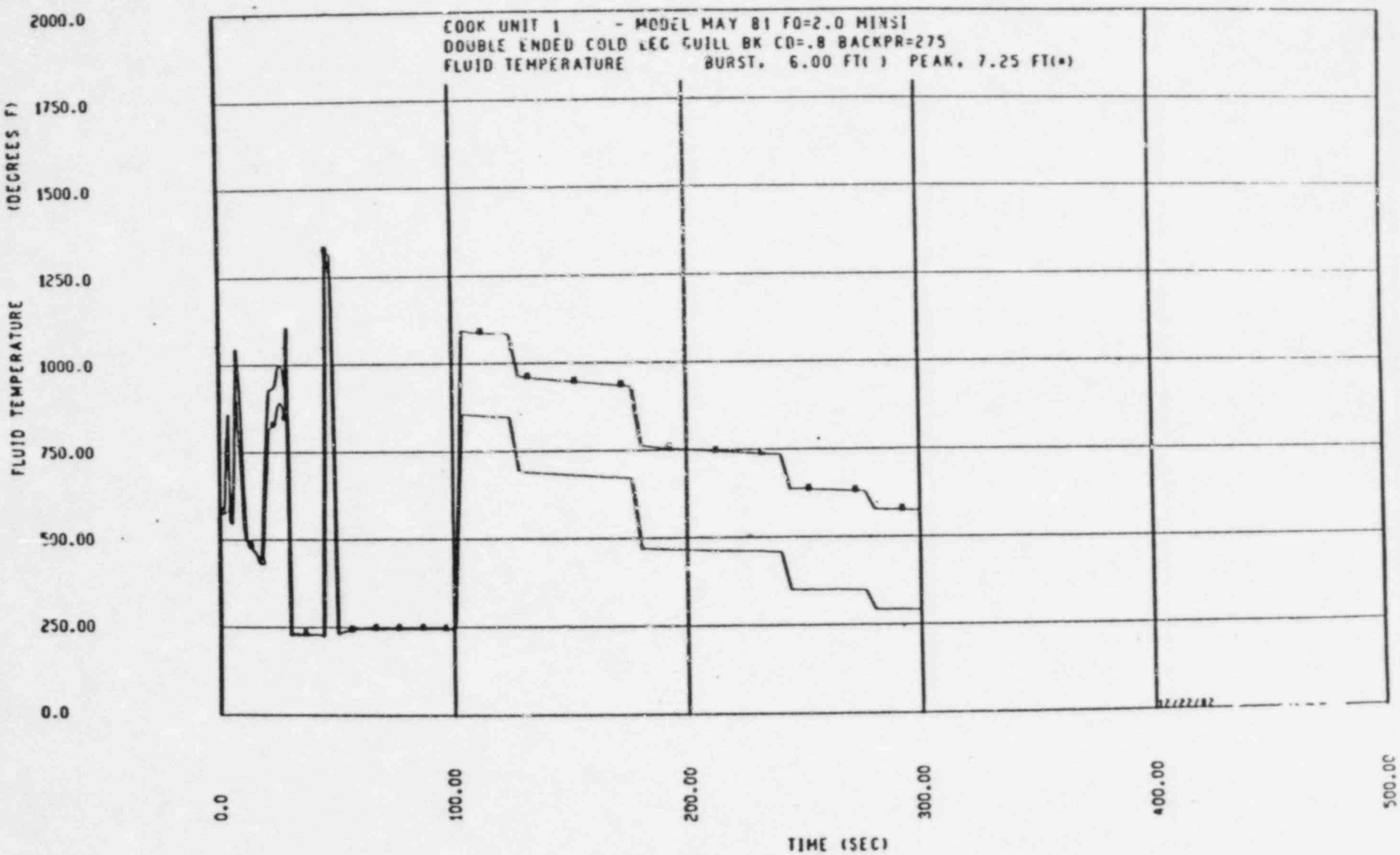


FIGURE 14.3.1-29 FLUID TEMPERATURE DECLG(CD = 0.8)

MIN
SI

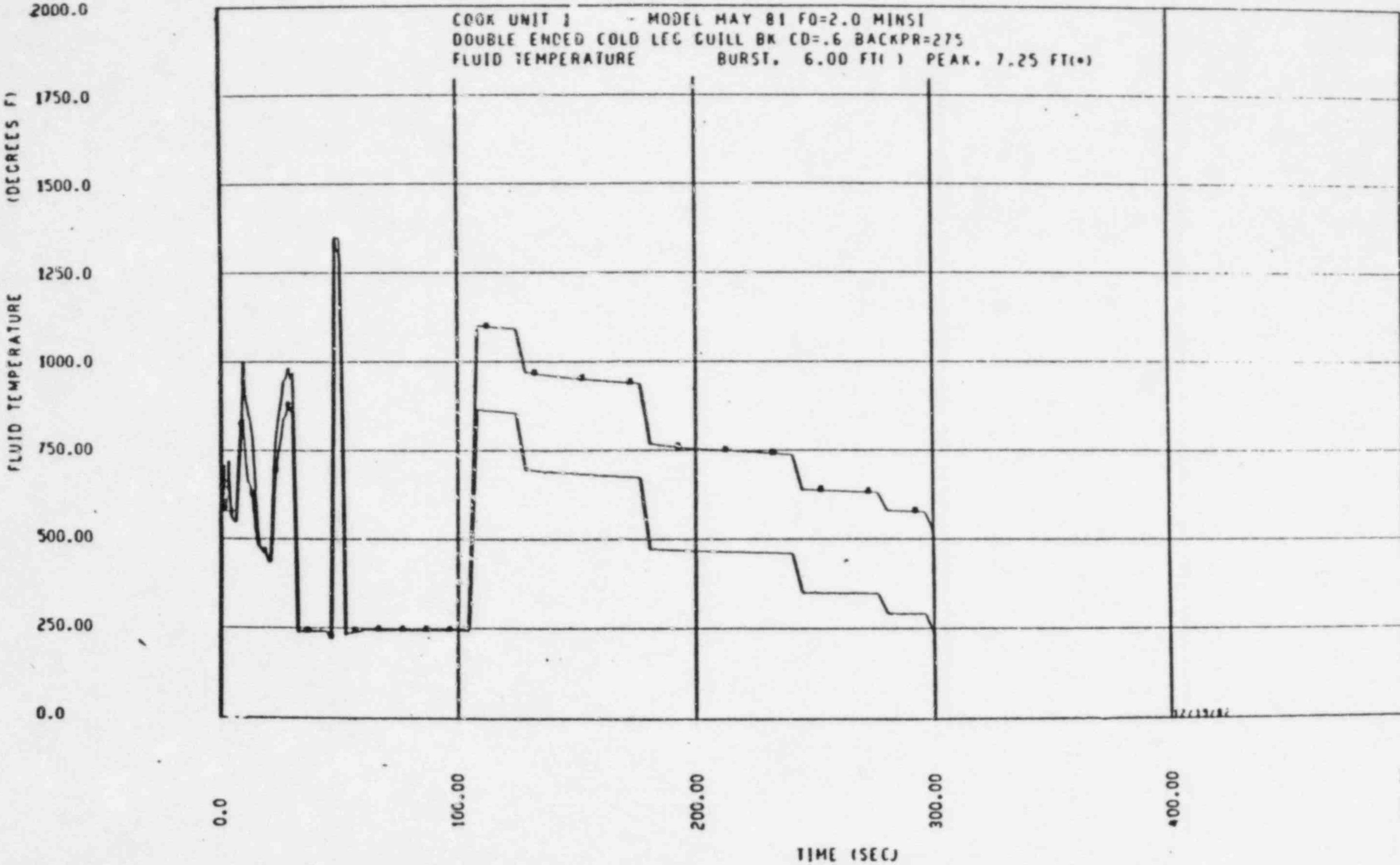


FIGURE 14.3.1-30 FLUID TEMPERATURE
DECLG(CD = 0.6)

MIN
SI

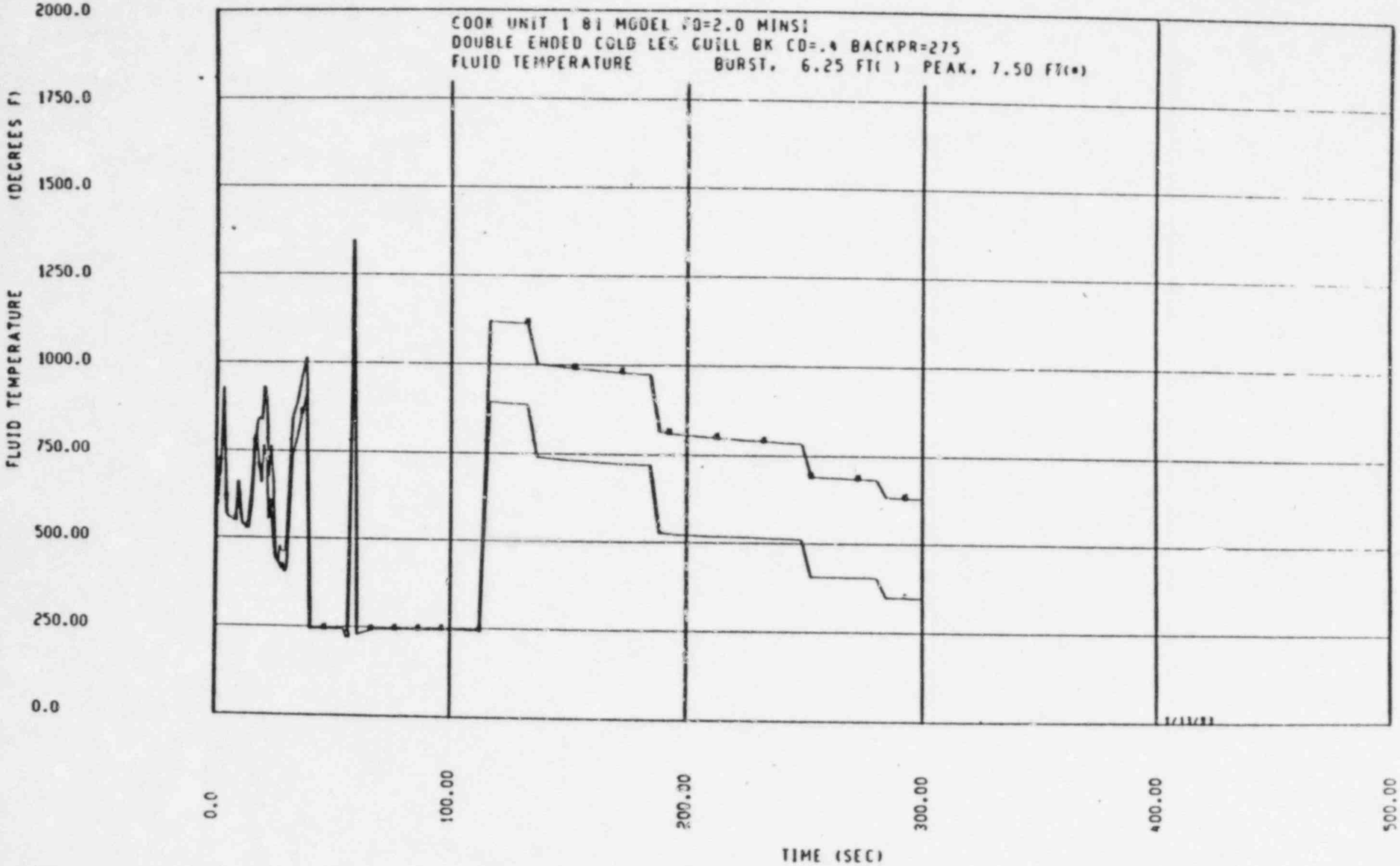


FIGURE 14.3.1-31 FLUID TEMPERATURE DECLG(CD = 0.4)

MIN SI

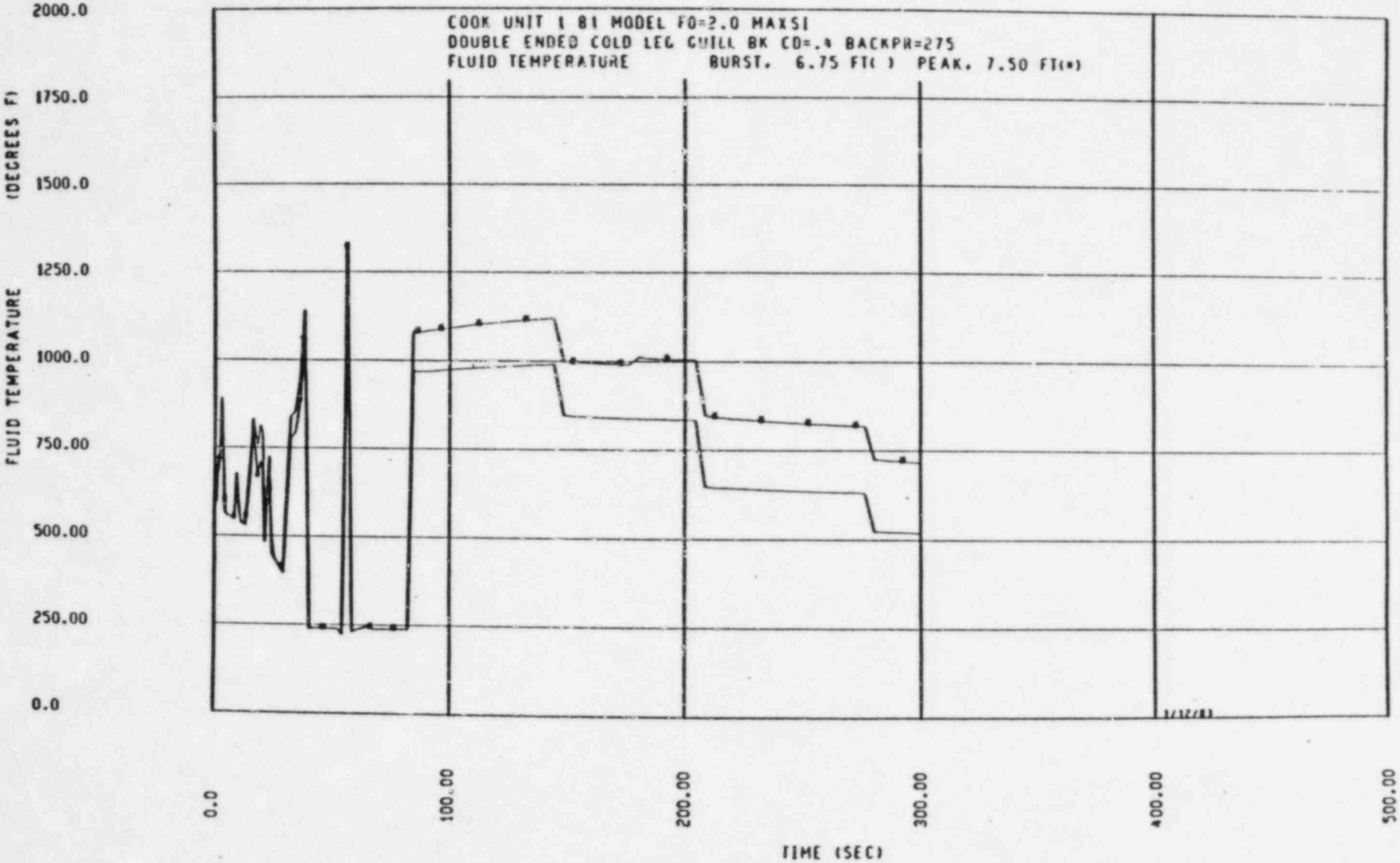


FIGURE 14.3.1-32 FLUID TEMPERATURE DECLG(CD = 0.4)

MAX SI

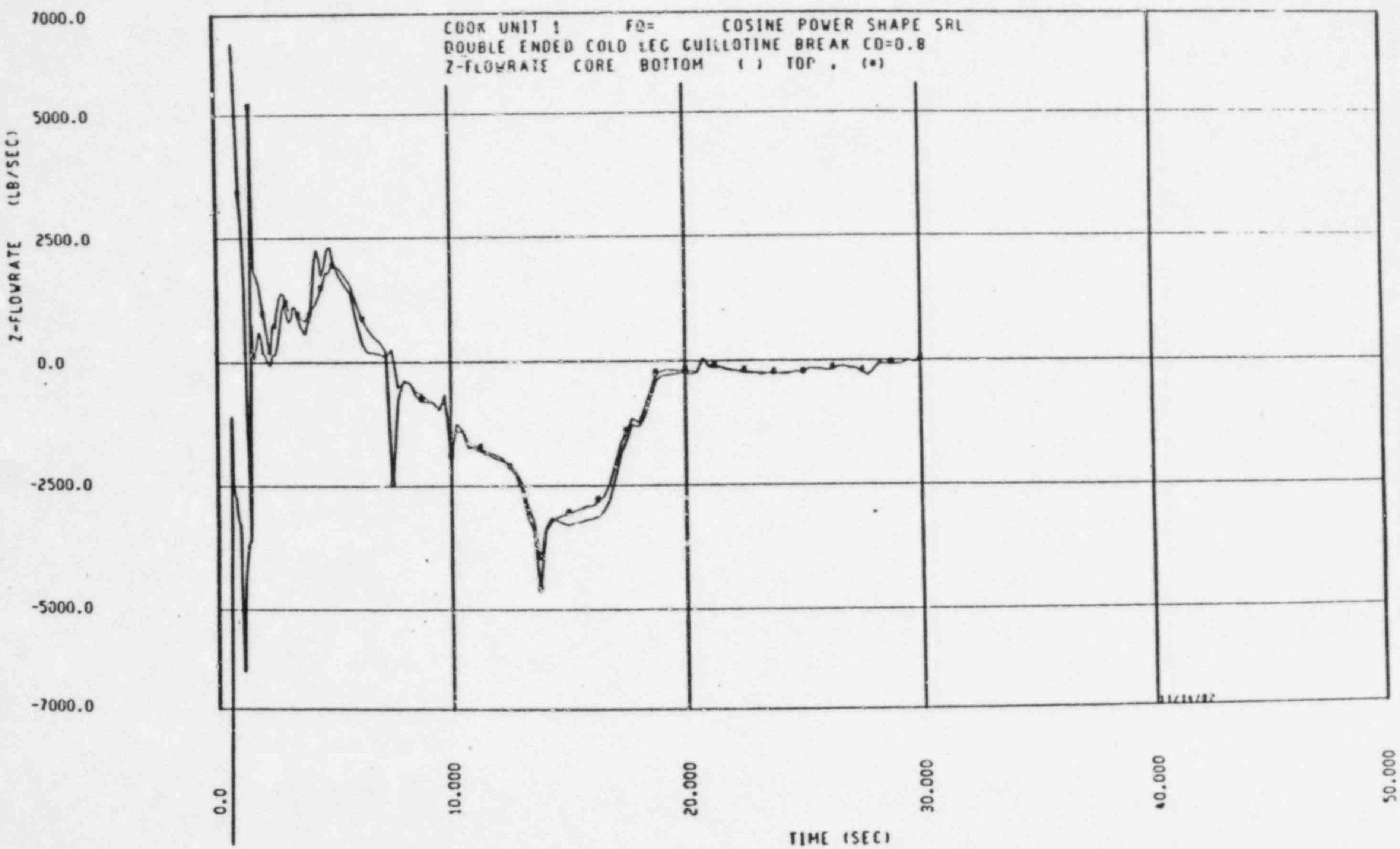


FIGURE 14.3.1-33 CORE FLOW (TOP AND BOTTOM) DECLG(CD = 0.8) MIN SI

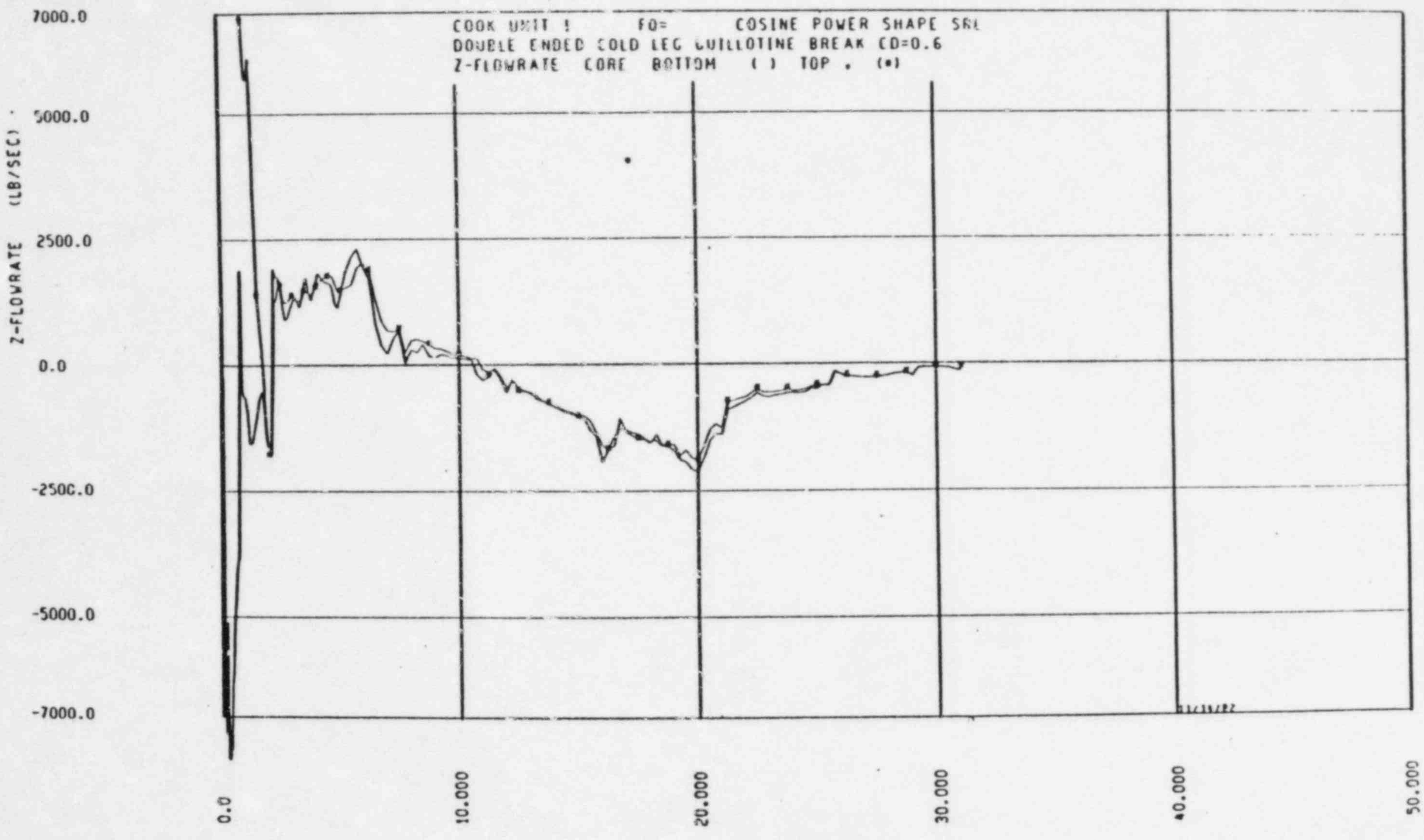


FIGURE 13.3.1-34 CORE FLOW (TOP AND BOTTOM) MIN
DECLG(CD = 0.6) SI

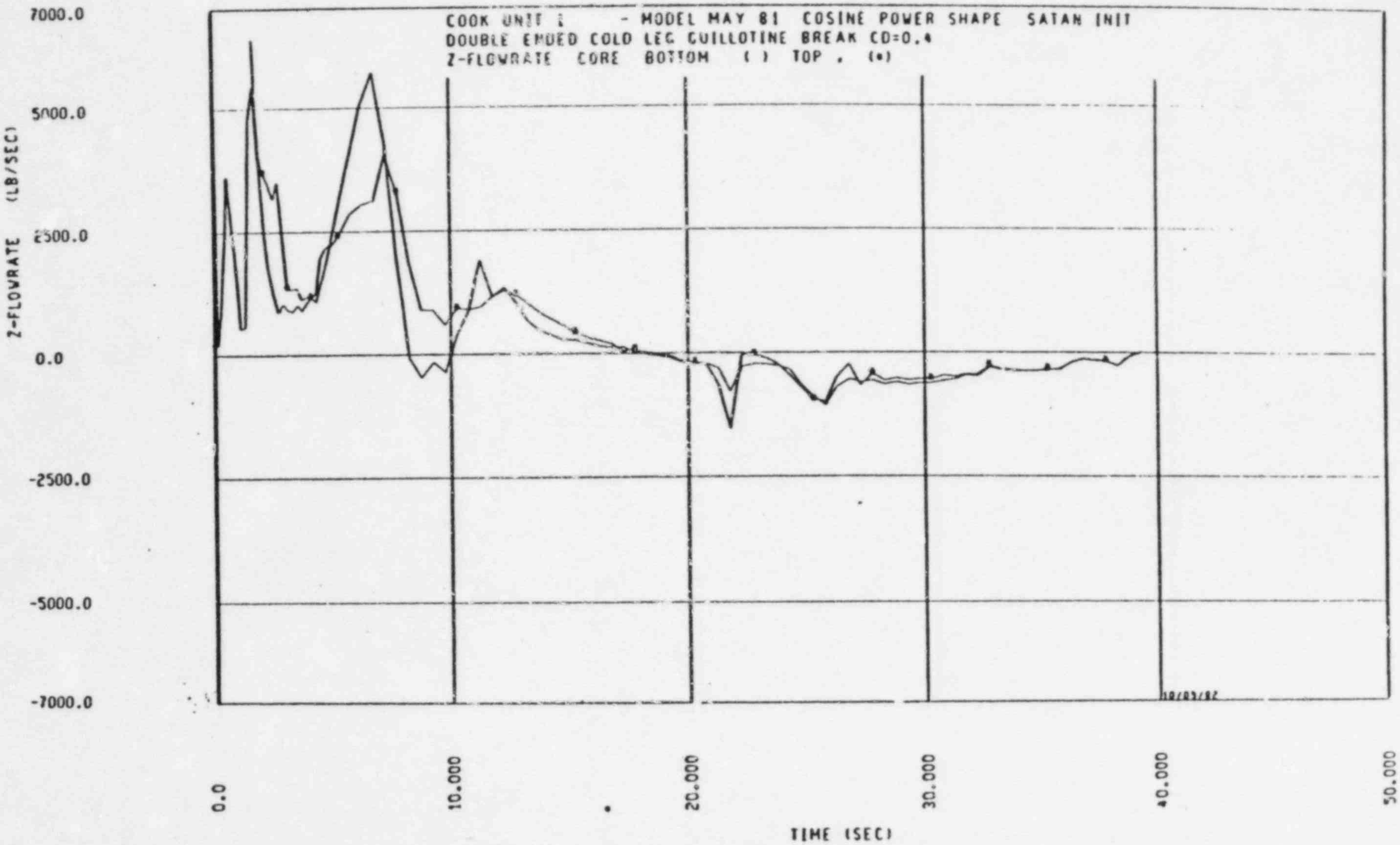


FIGURE 14.3.1-35 CORE FLOW (TOP AND BOTTOM) MIN SI
DECLG(CD = 0.4)

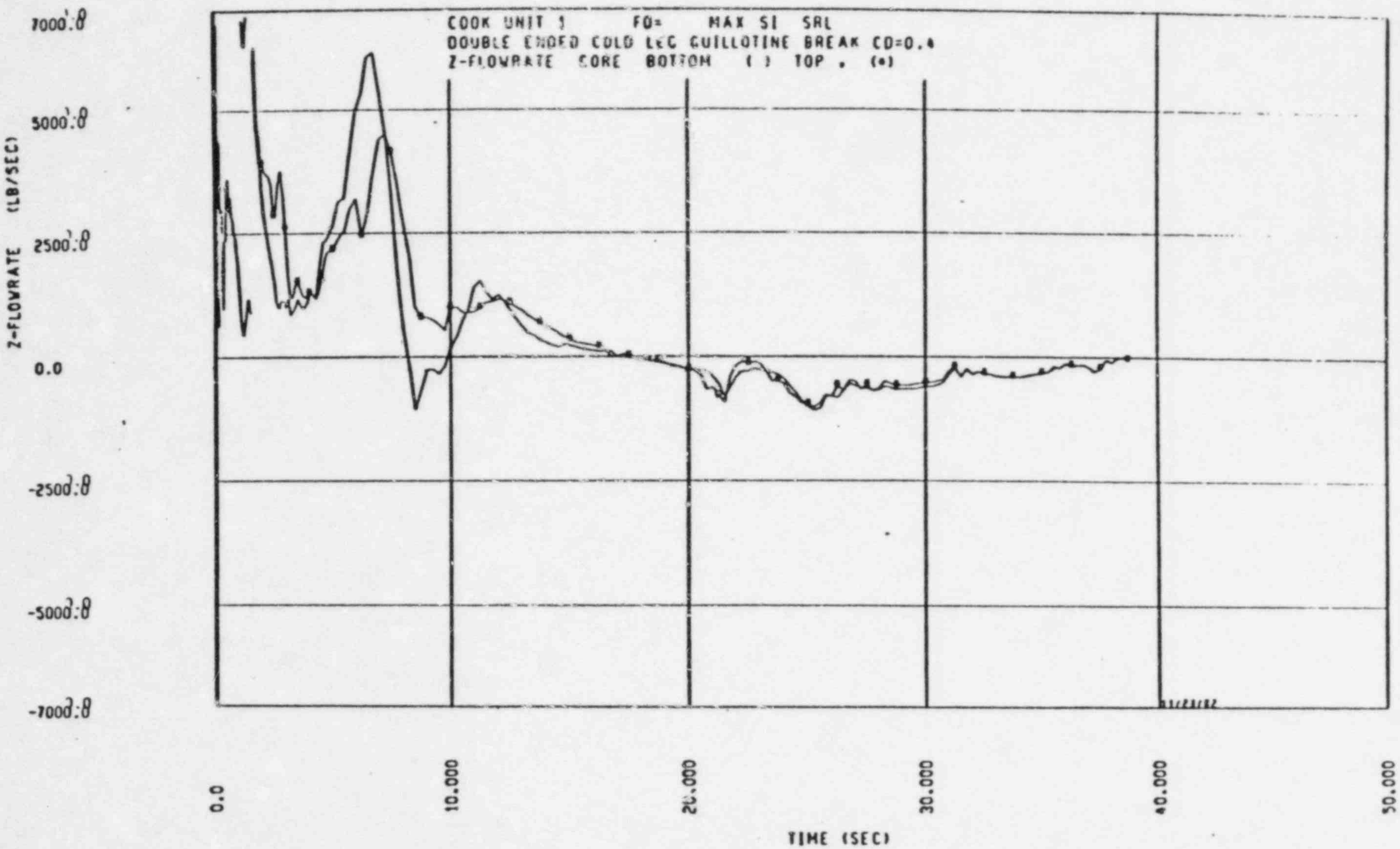


FIGURE 14.3.1-36 CORE FLOW (TOP AND BOTTOM)
DECLG(CD = 0.4)

MAX
SI

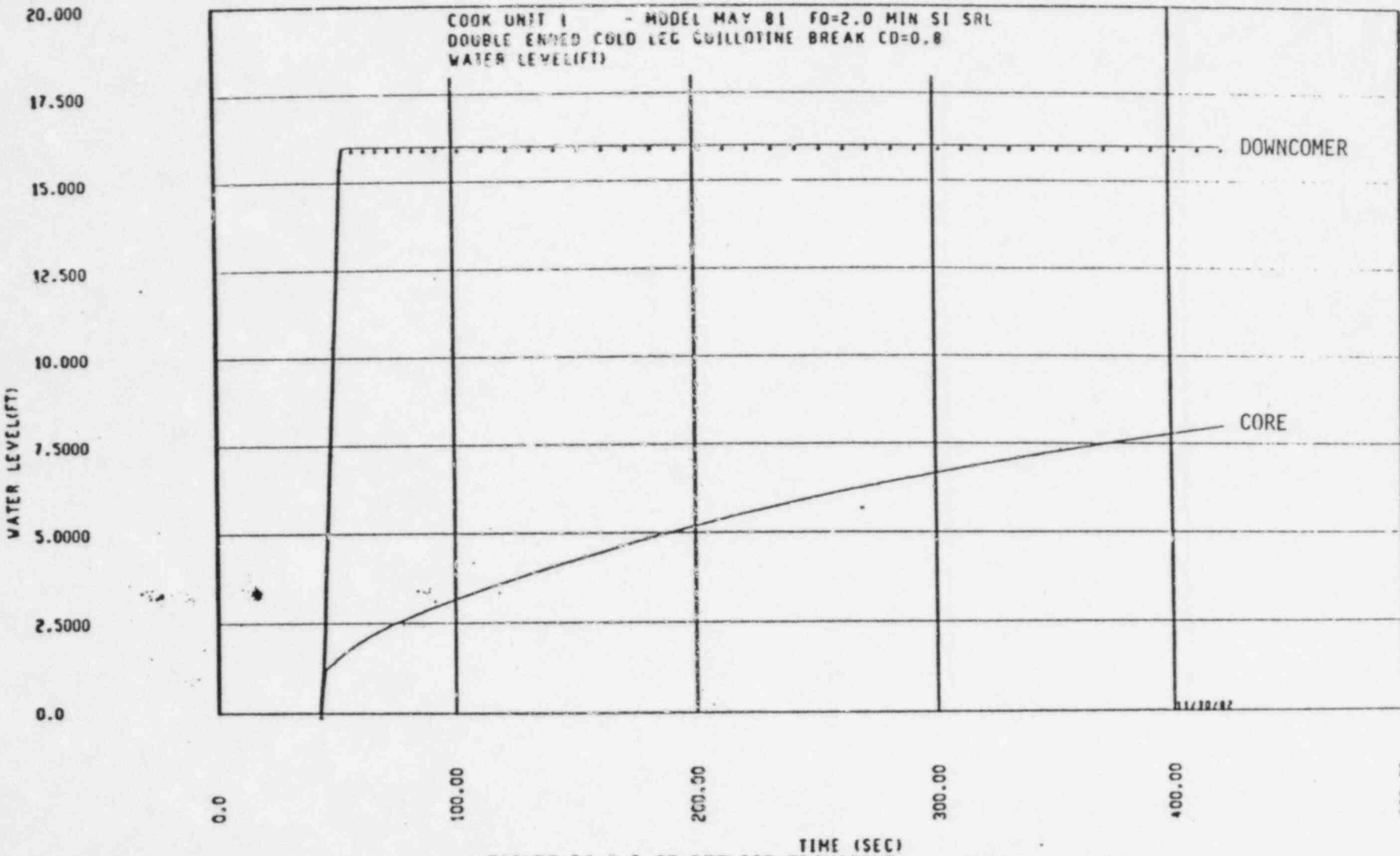


FIGURE 14.3.1-37 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.8)

MIN SI

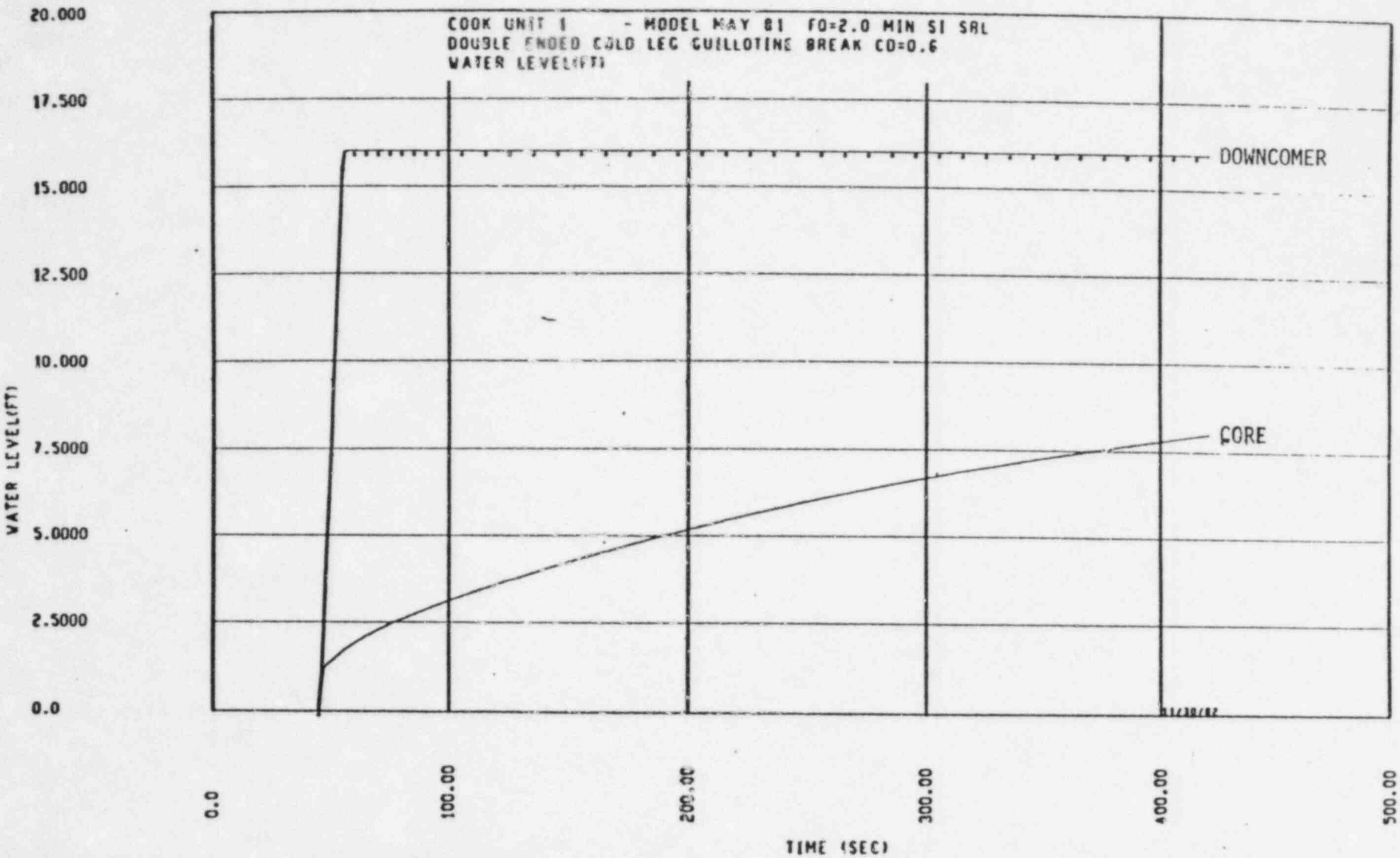


FIGURE 14.3.1-38 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.6)

MIN SI

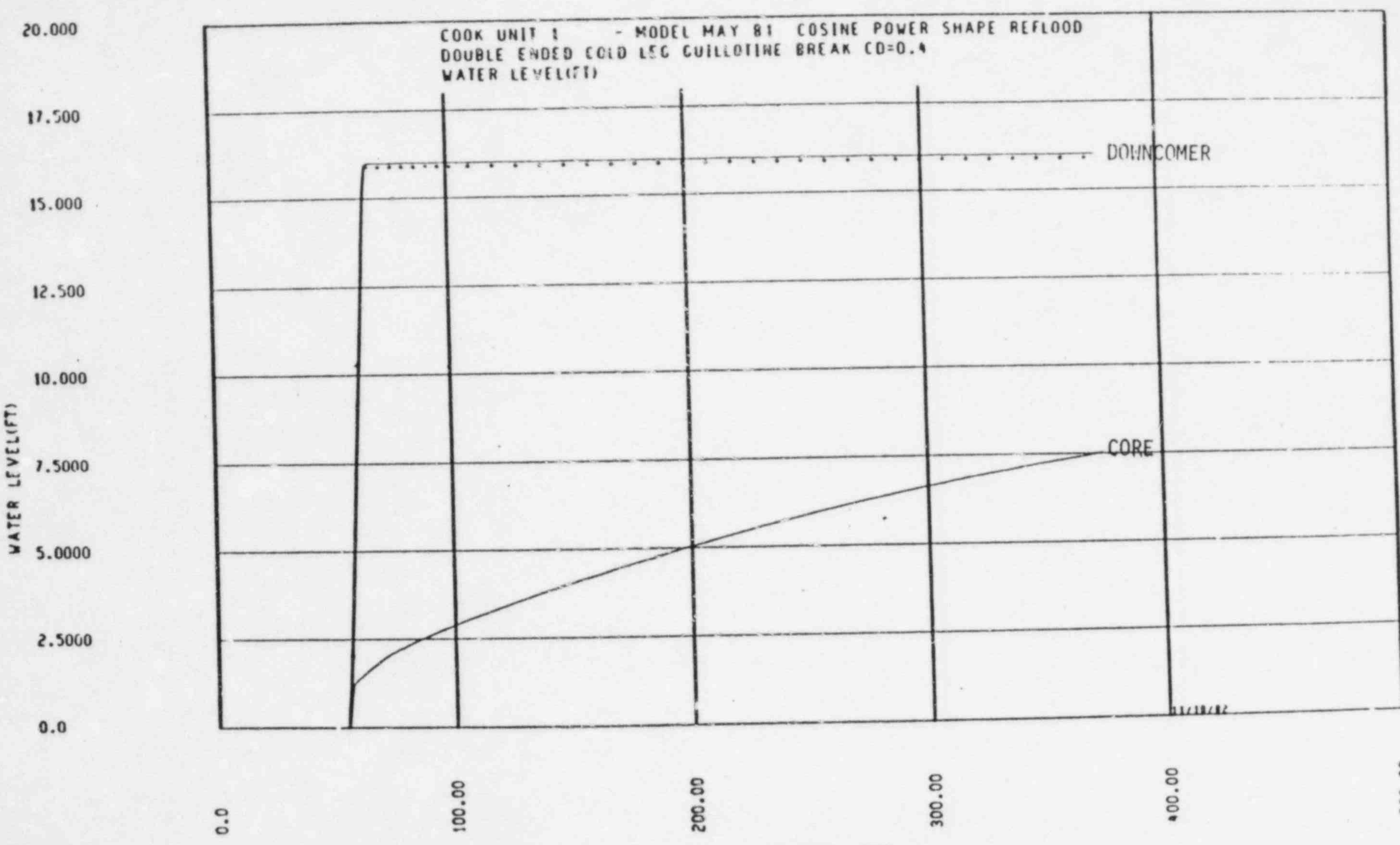


FIGURE 14.3.1-39 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4)

MIN SI

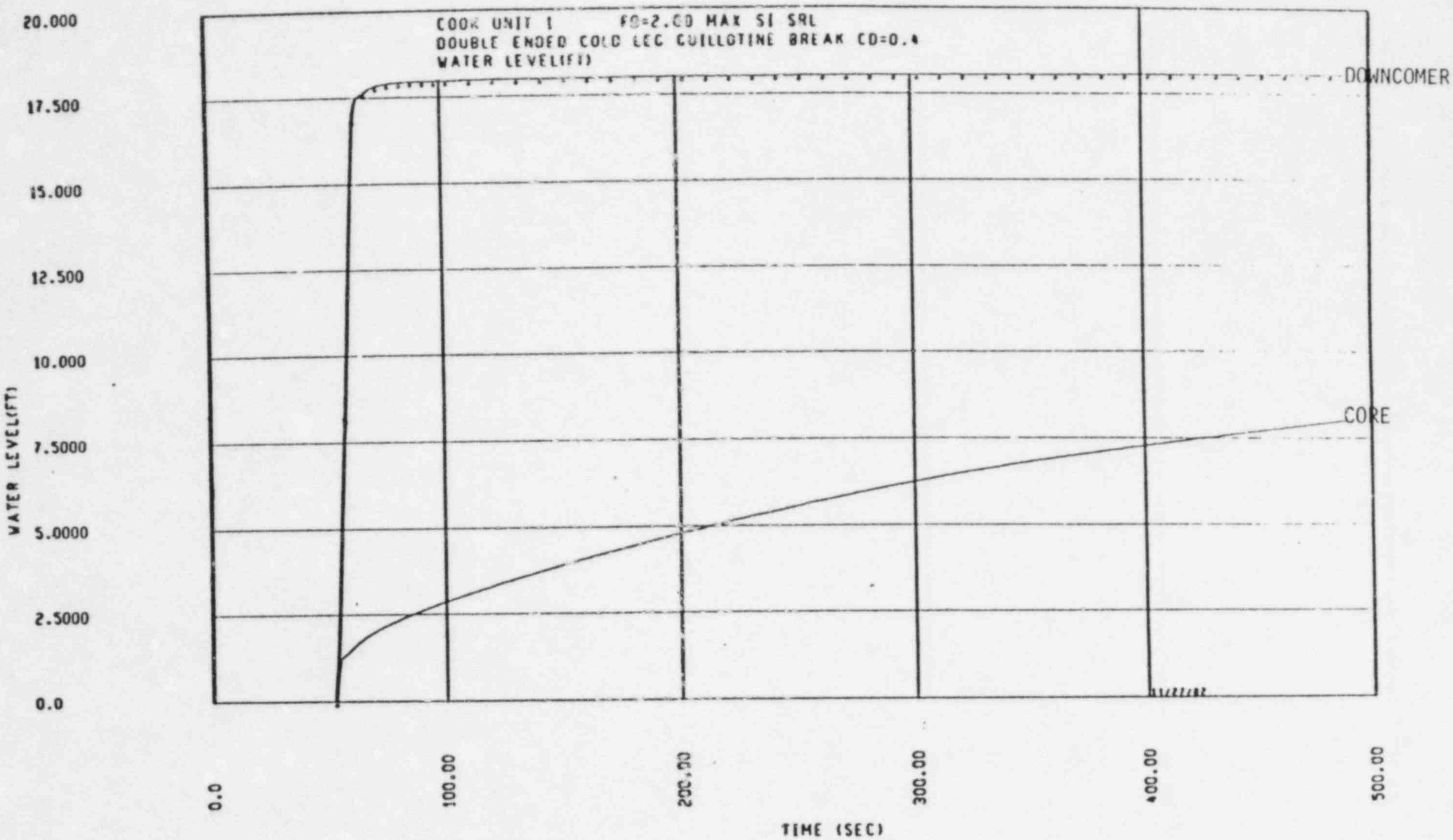


FIGURE 14.3.1-40 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4)

MAX SI

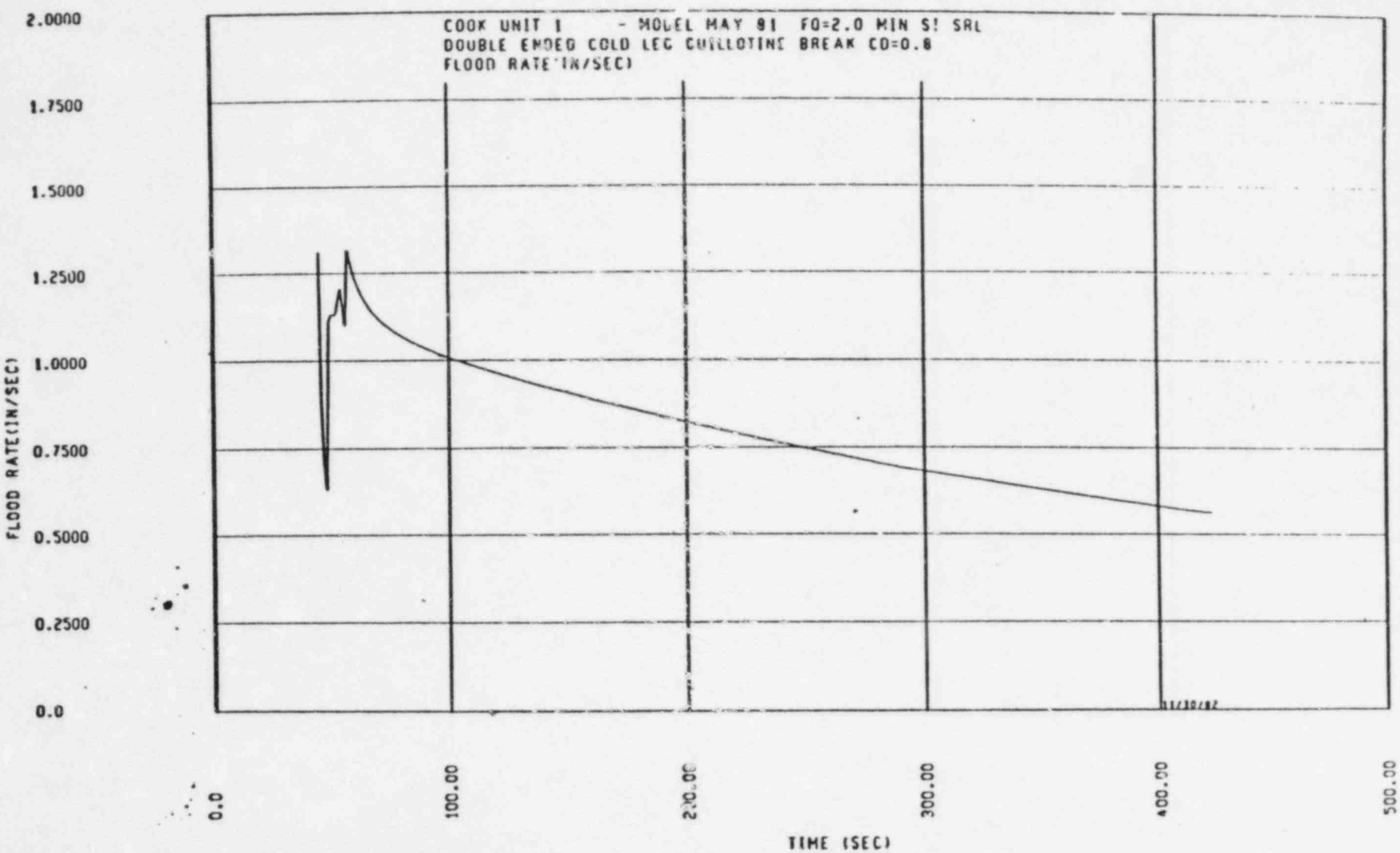


FIGURE 14.3.1-47 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.8)

MIN
SI

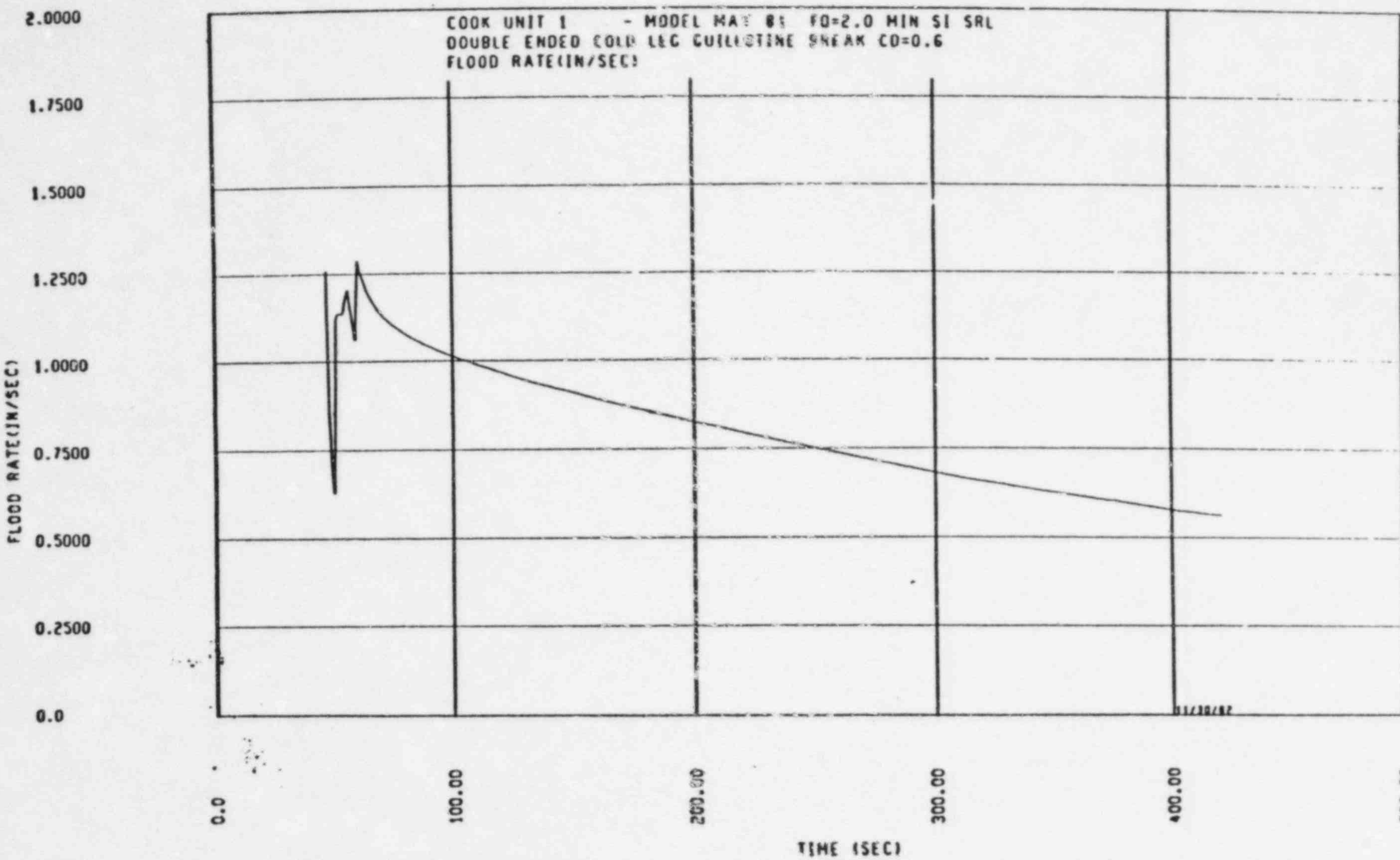


FIGURE 14.3.1-42 REFLOOD TRANSIENT MIN SI
CORE INLET VELOCITY
DECLG(CD = 0.6)

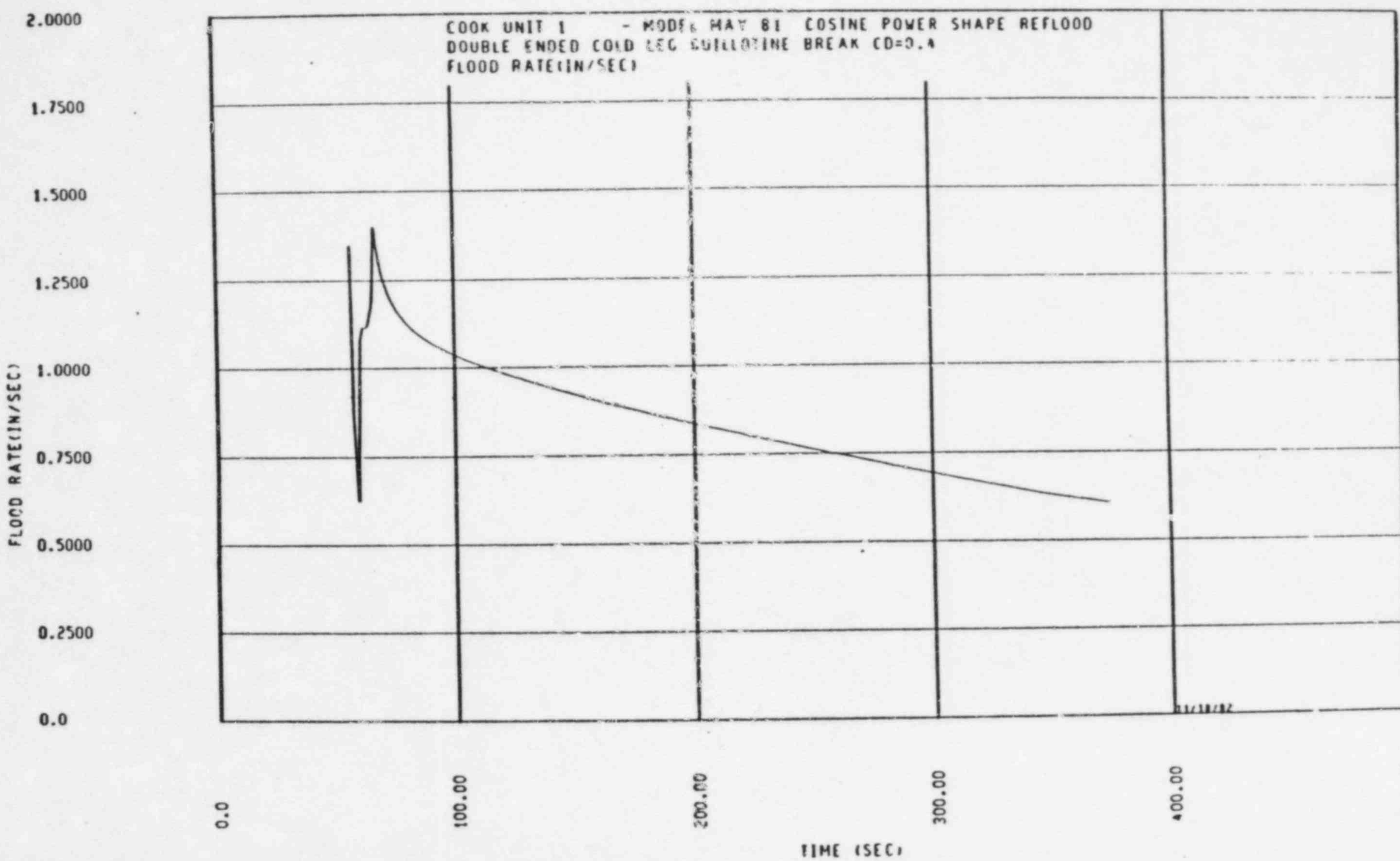


FIGURE 14.3.2-43 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.4)

MIN
SI

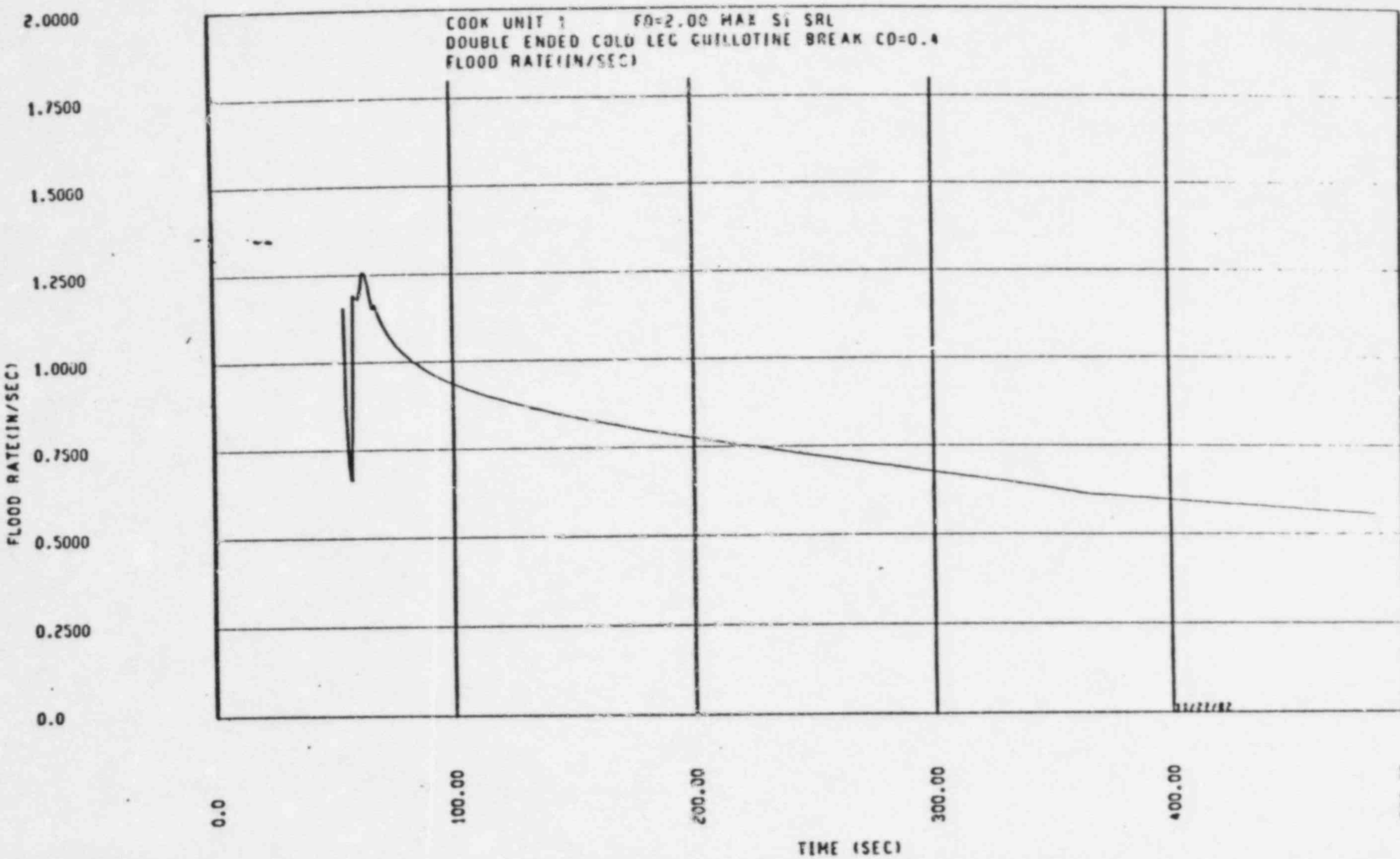


FIGURE 14.3.1-44 REFLOOD TRANSIENT
CORE INLET VELOCITY
DECLG(CD = 0.4)

MAX
SI

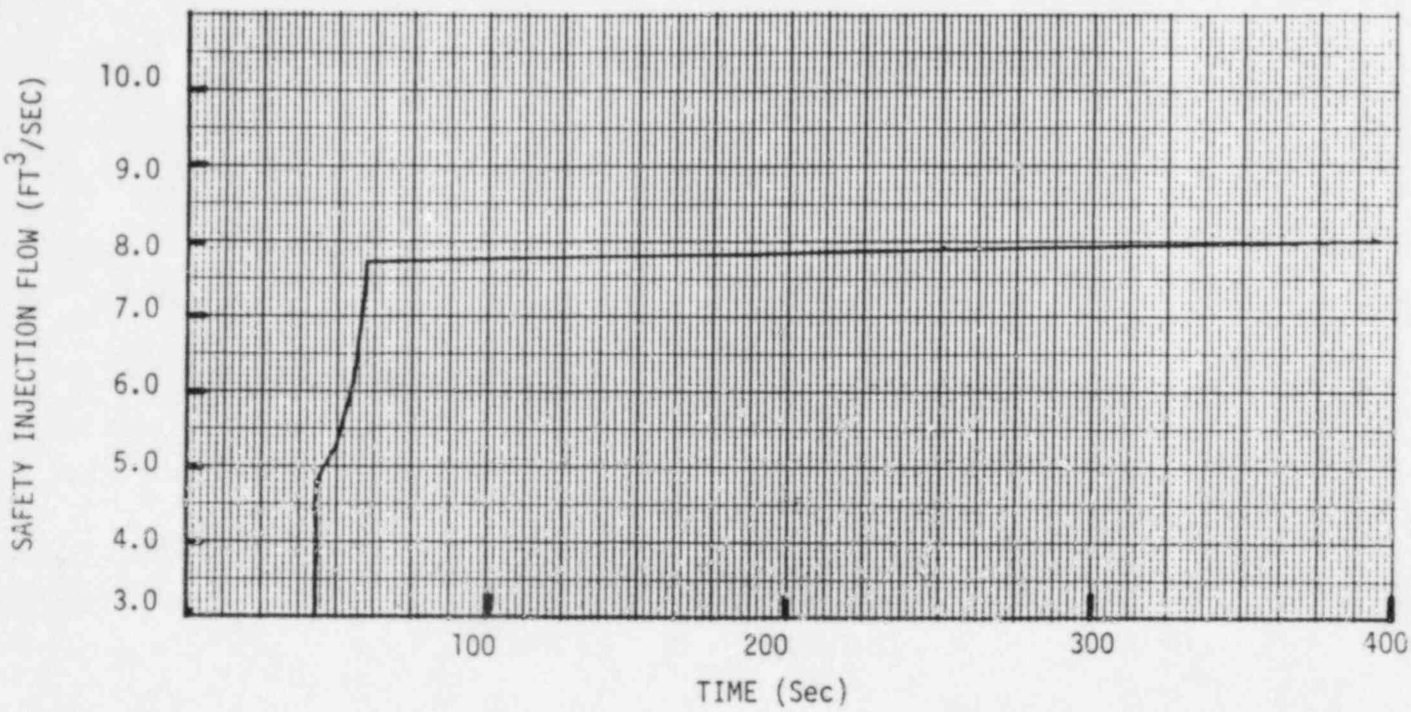


FIGURE 14.3.1-45 PUMPED ECCS FLOW (REFLOOD)
DECLG(CD = 0.8) MIN SI

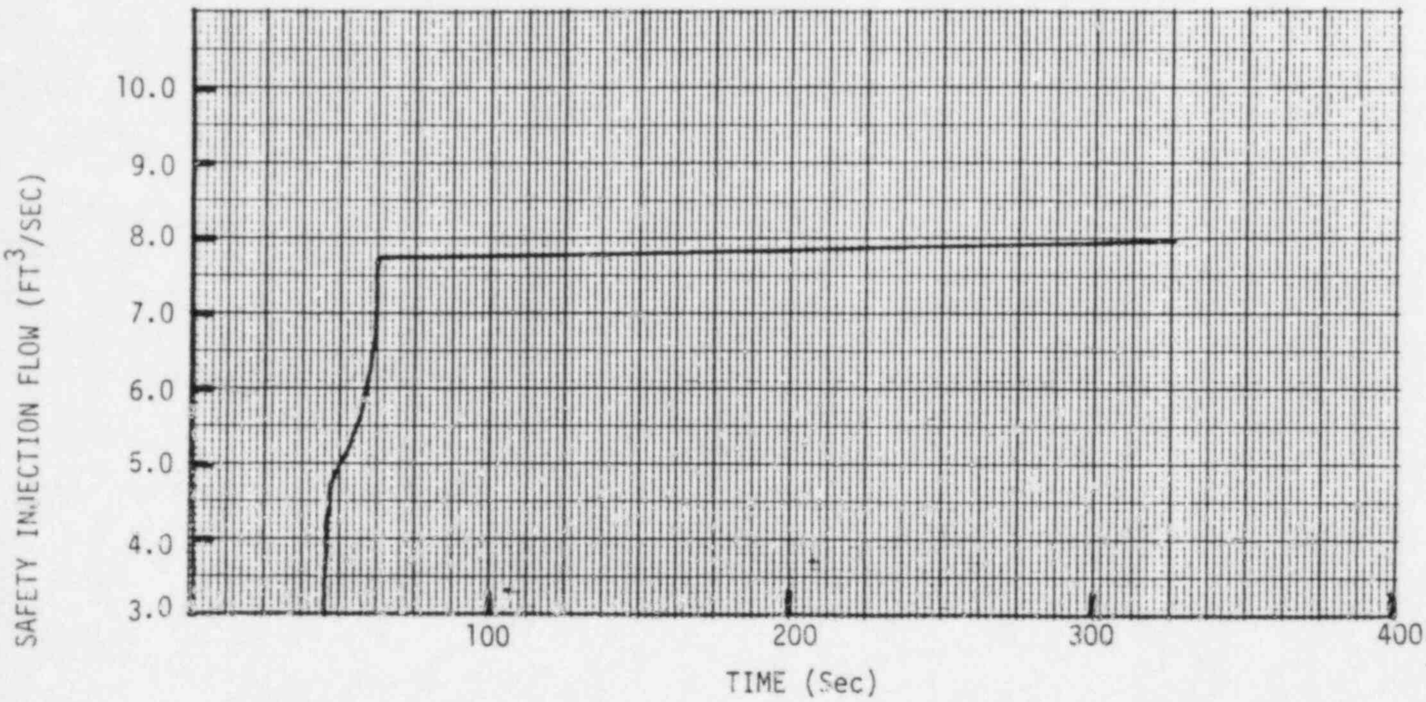


FIGURE 14.3.1-46 PUMPED ECCS FLOW (REFLOOD) DECLG(CD = 0.6)
MIN SI

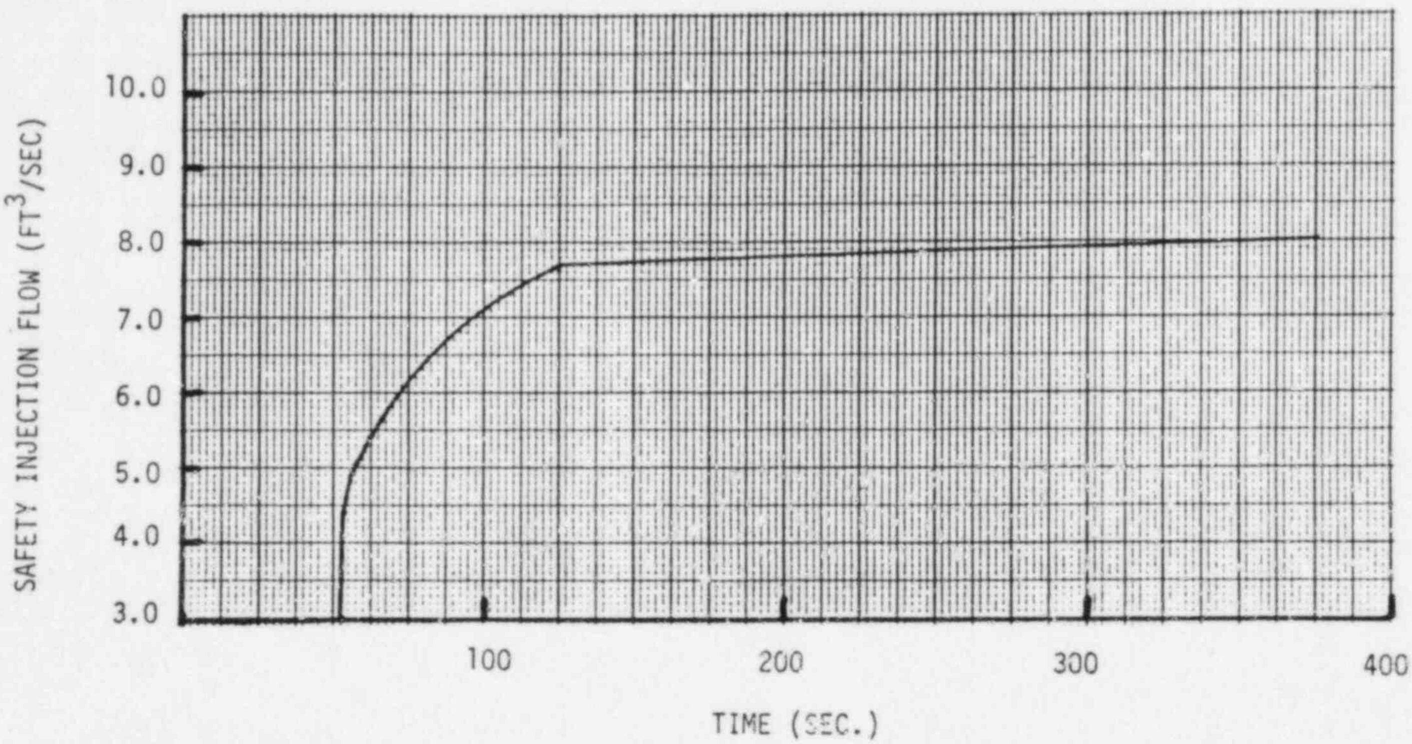


FIGURE 14.3.1-47 PUMPED ECCS FLOW (REFLOOD) DECLG(CD = 0.4)
MIN SI

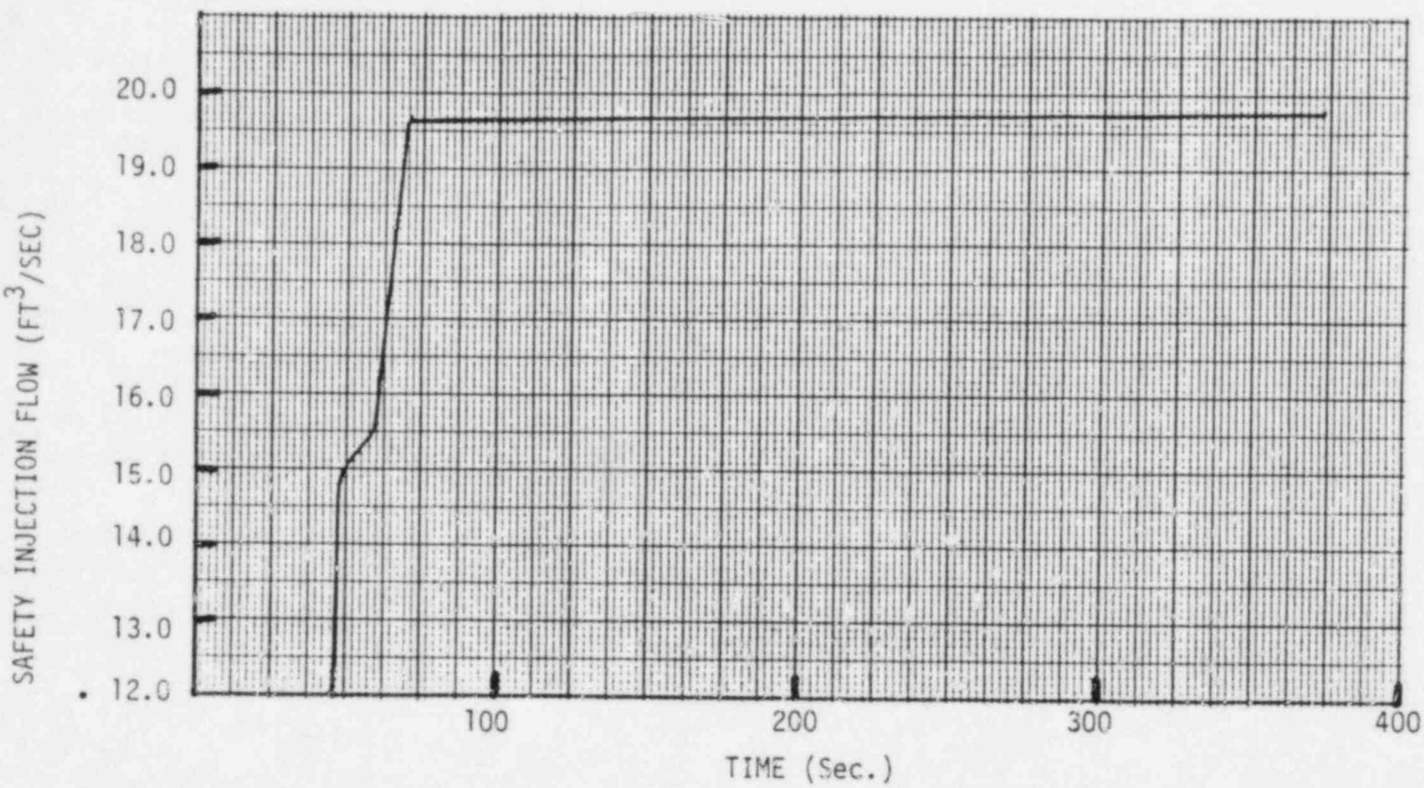


FIGURE 14.3.1-48 PUMPED ECCS FLOW (REFLOOD - DECLG (CD - 0.4)
MAX SI

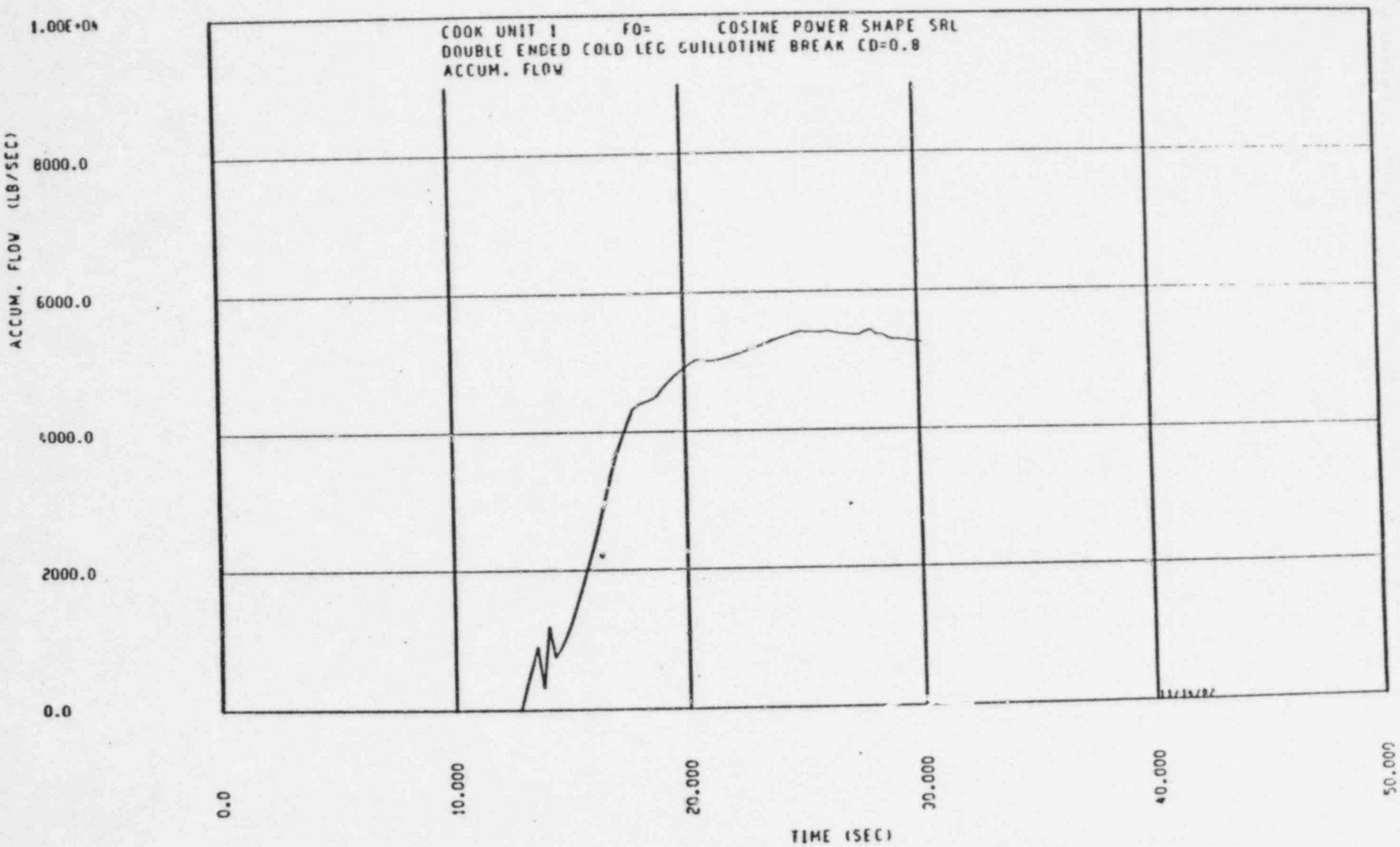


FIGURE 14.3.1-49 ACCUMULATOR FLOW (BLOWDOWN) MIN
DECLG(CD = 0.8) SI

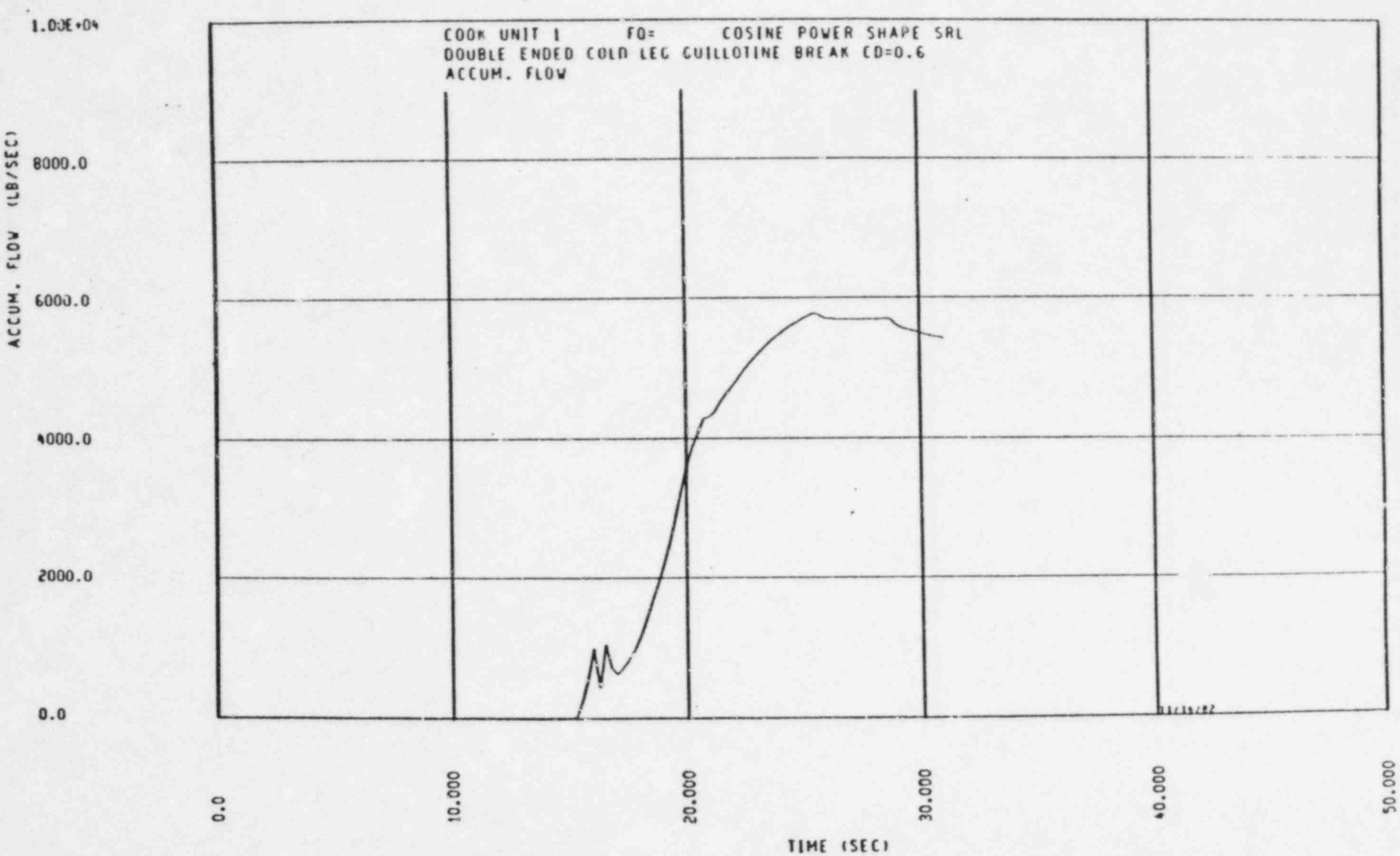


FIGURE 14.3.1-50 ACCUMULATOR FLOW (BLOWDOWN) MIN
DECLG(CD = 0.6) SI

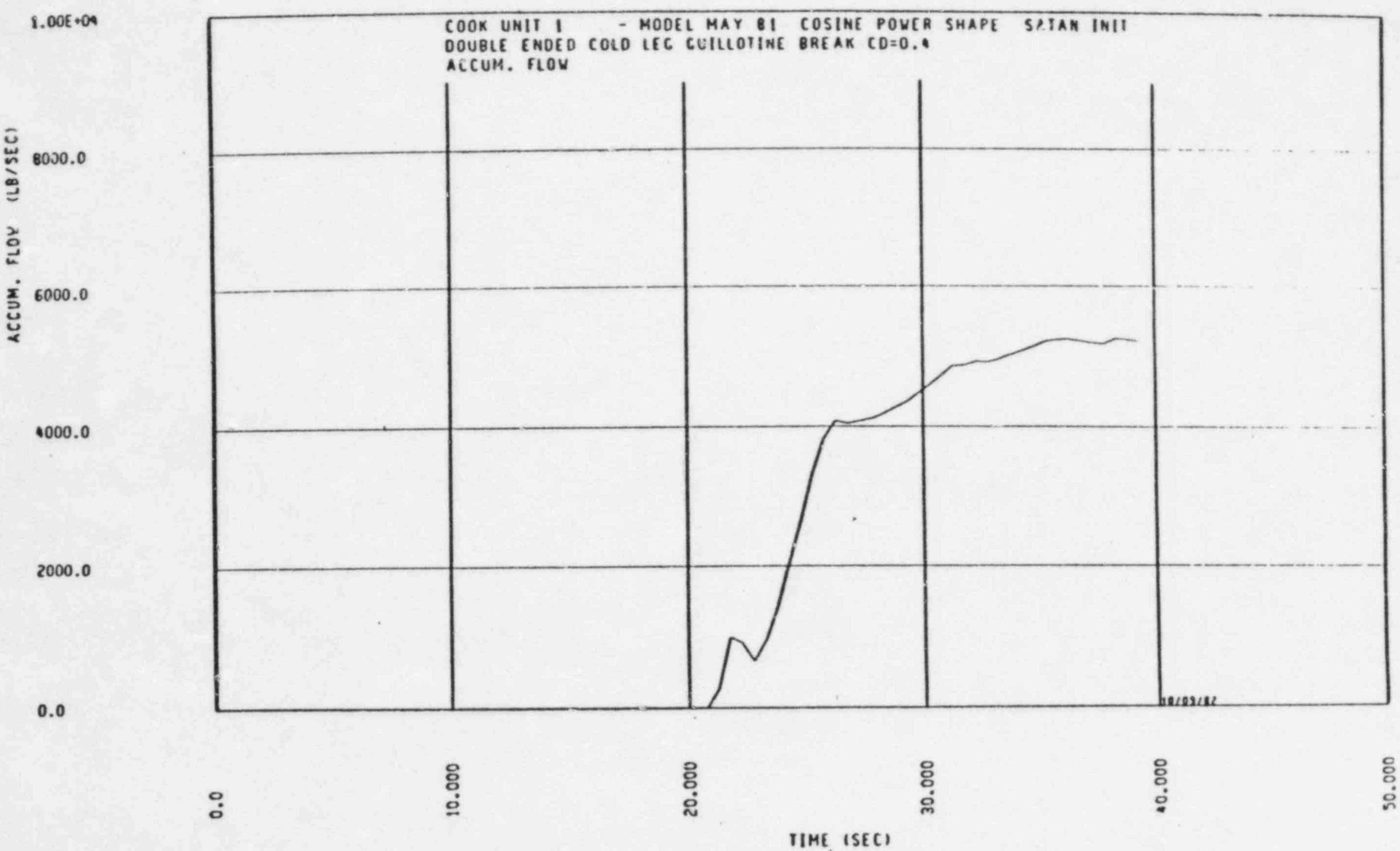


FIGURE 14.3.1-51 ACCUMULATOR FLOW (BLOWDOWN) MIN
DECLG(CD = 0.4) SI

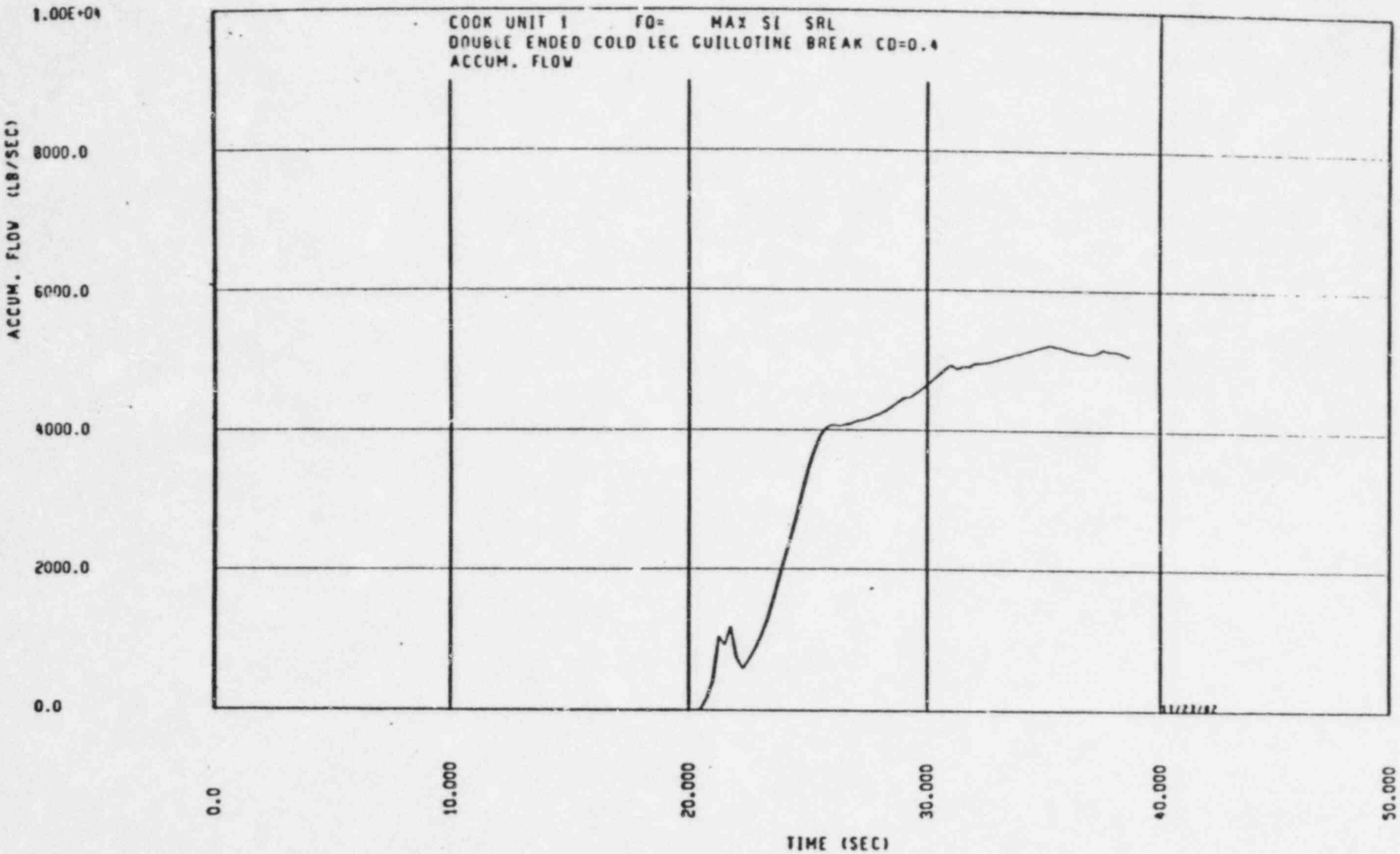
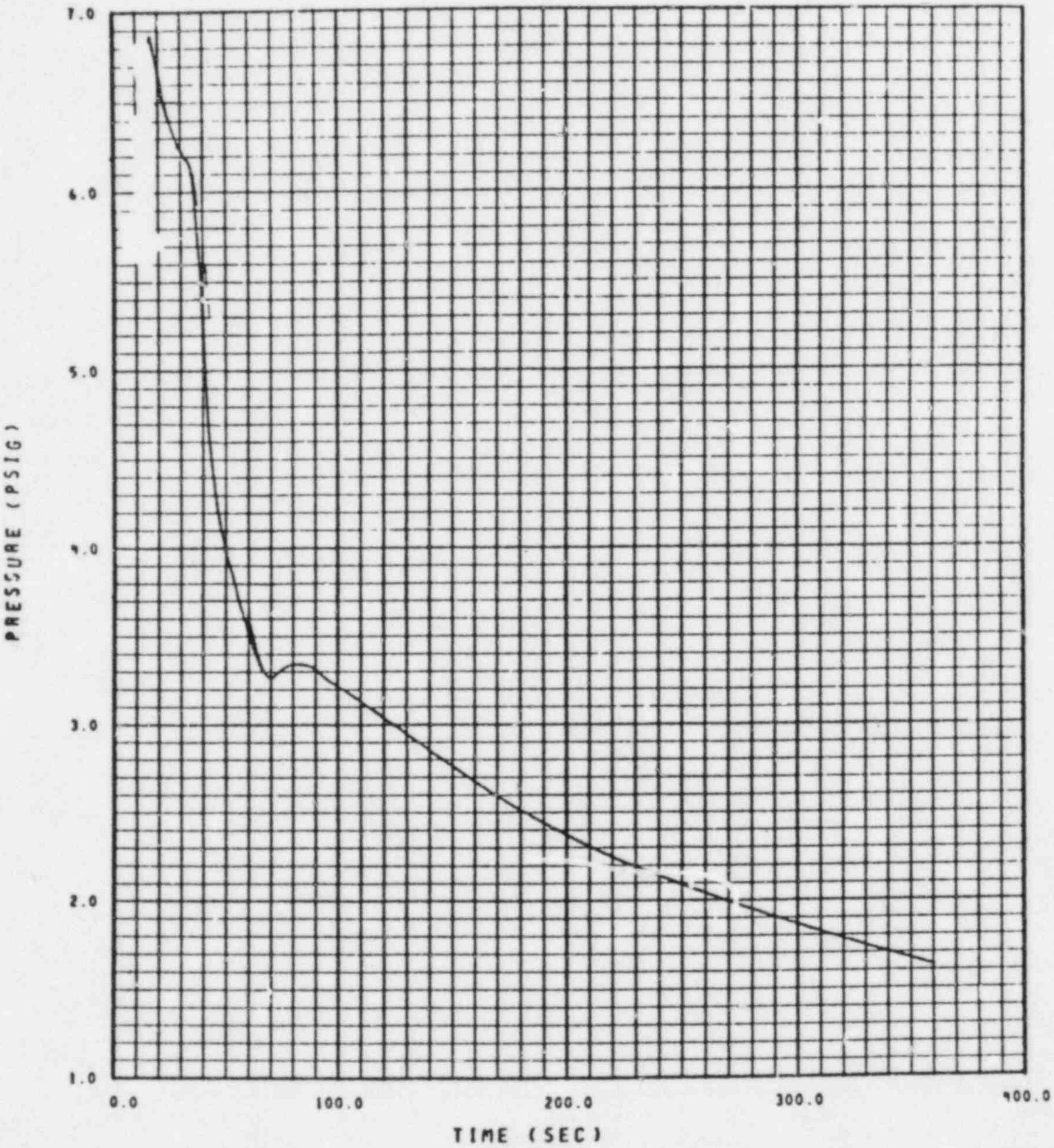


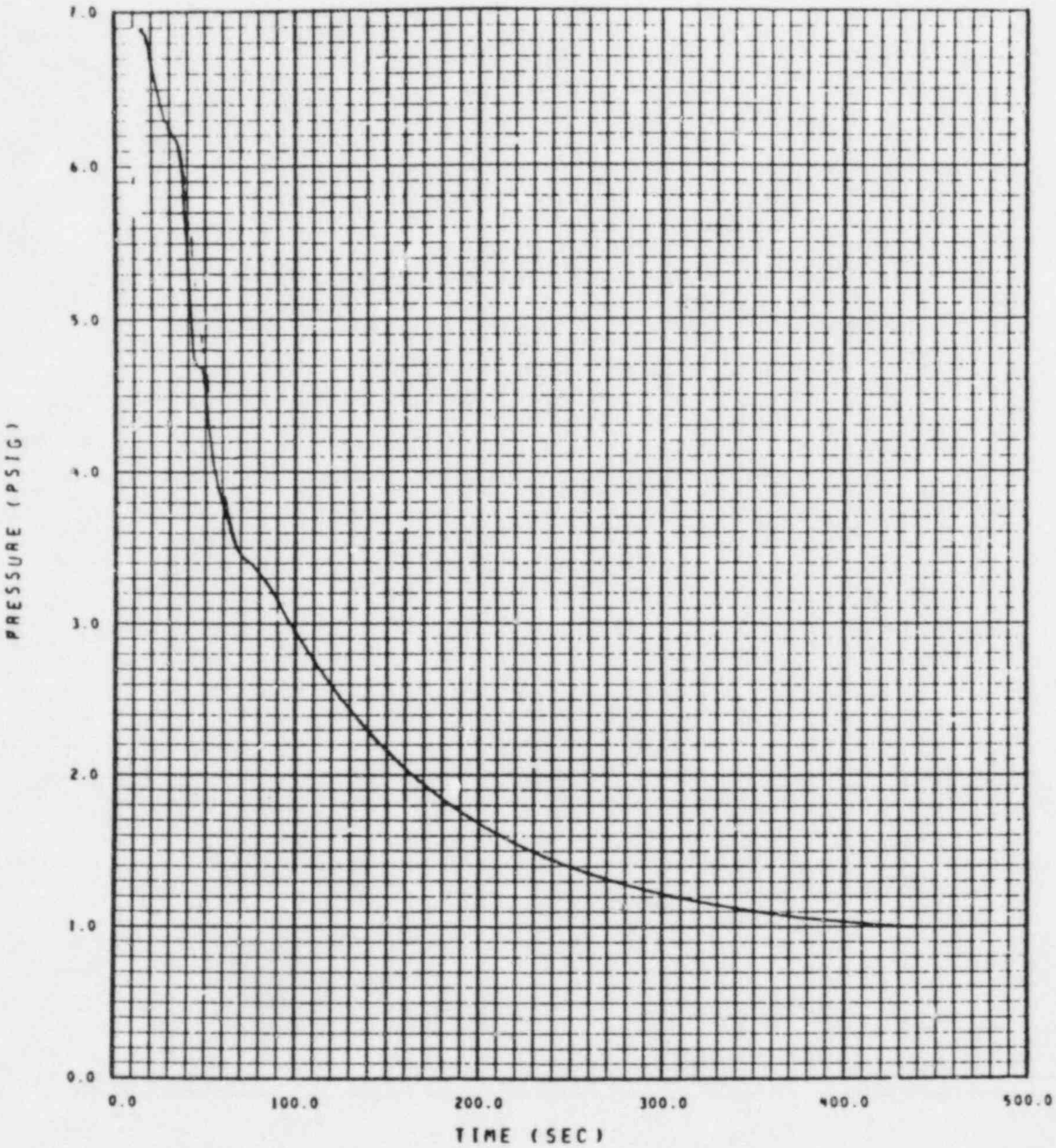
FIGURE 14.3.1-52 ACCUMULATOR FLOW (BLOWDOWN) MAX
DECLG(CD = 0.4) SI



COMPARTMENT PRESSURE

MINIMUM SI

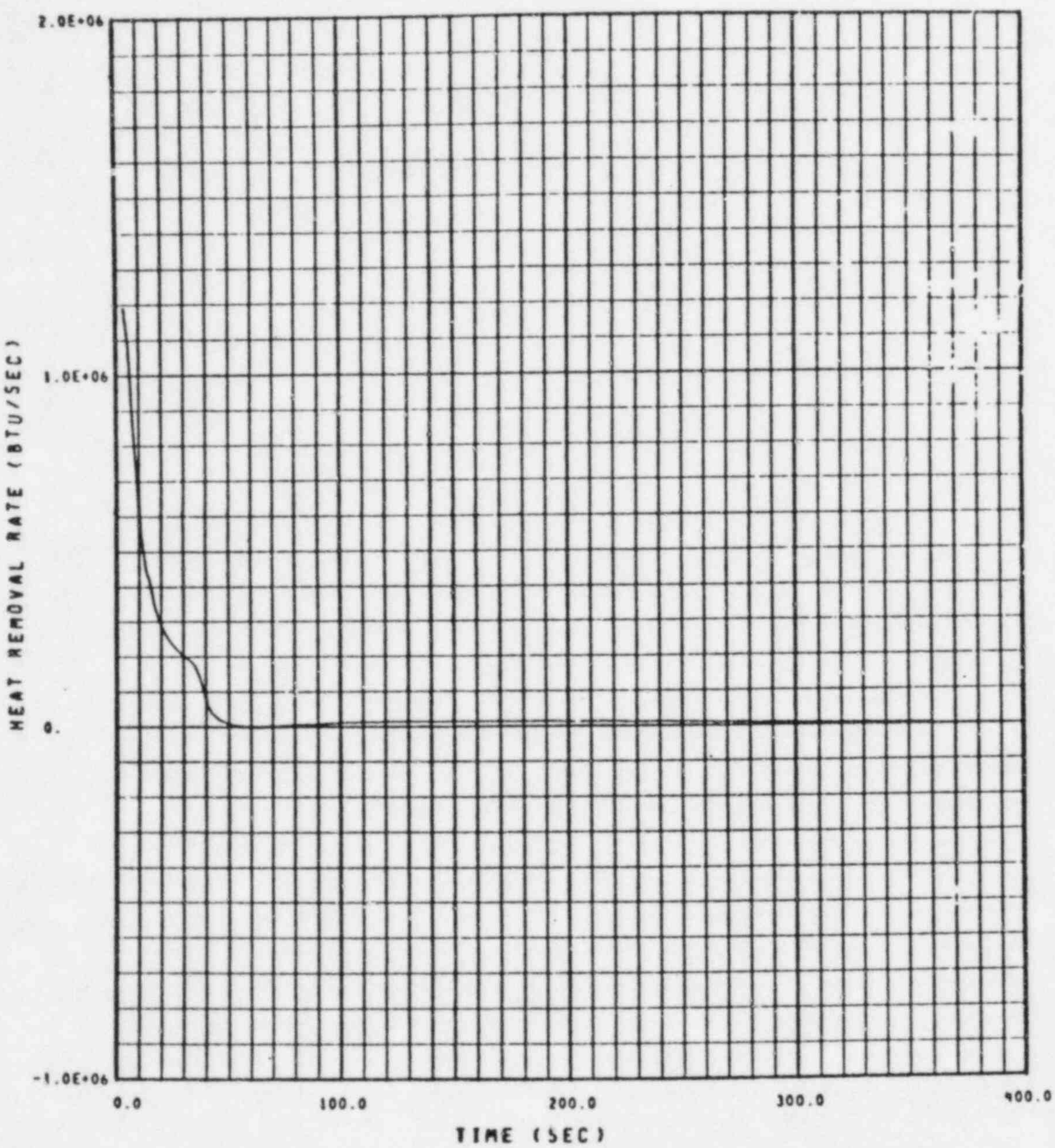
FIGURE 14.3.1-53 CONTAINMENT PRESSURE
DECLG(CD = 0.4)



COMPARTMENT PRESSURE

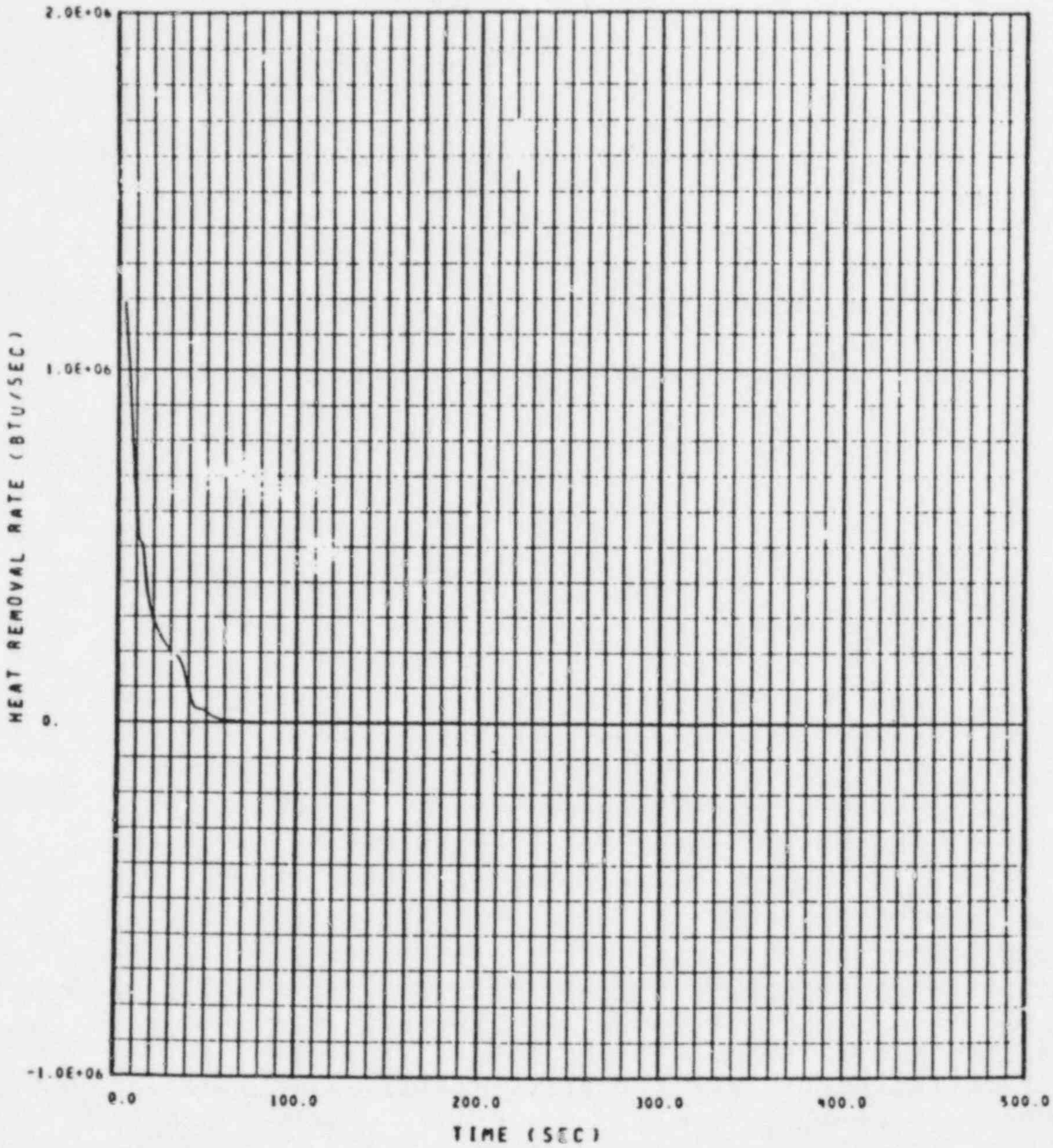
MAXIMUM SI

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



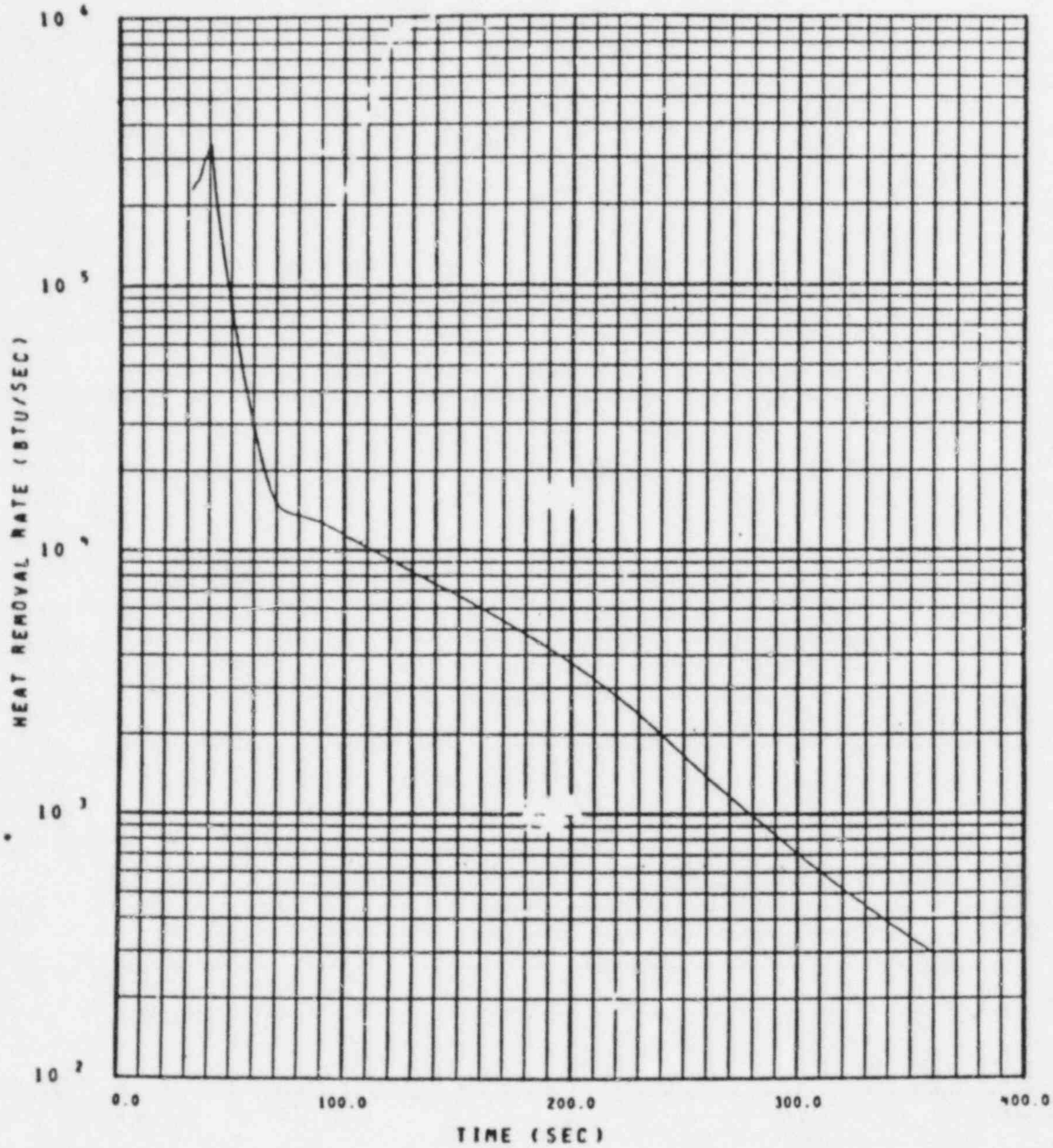
MINIMUM SI

FIGURE 14.3.1-55 LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



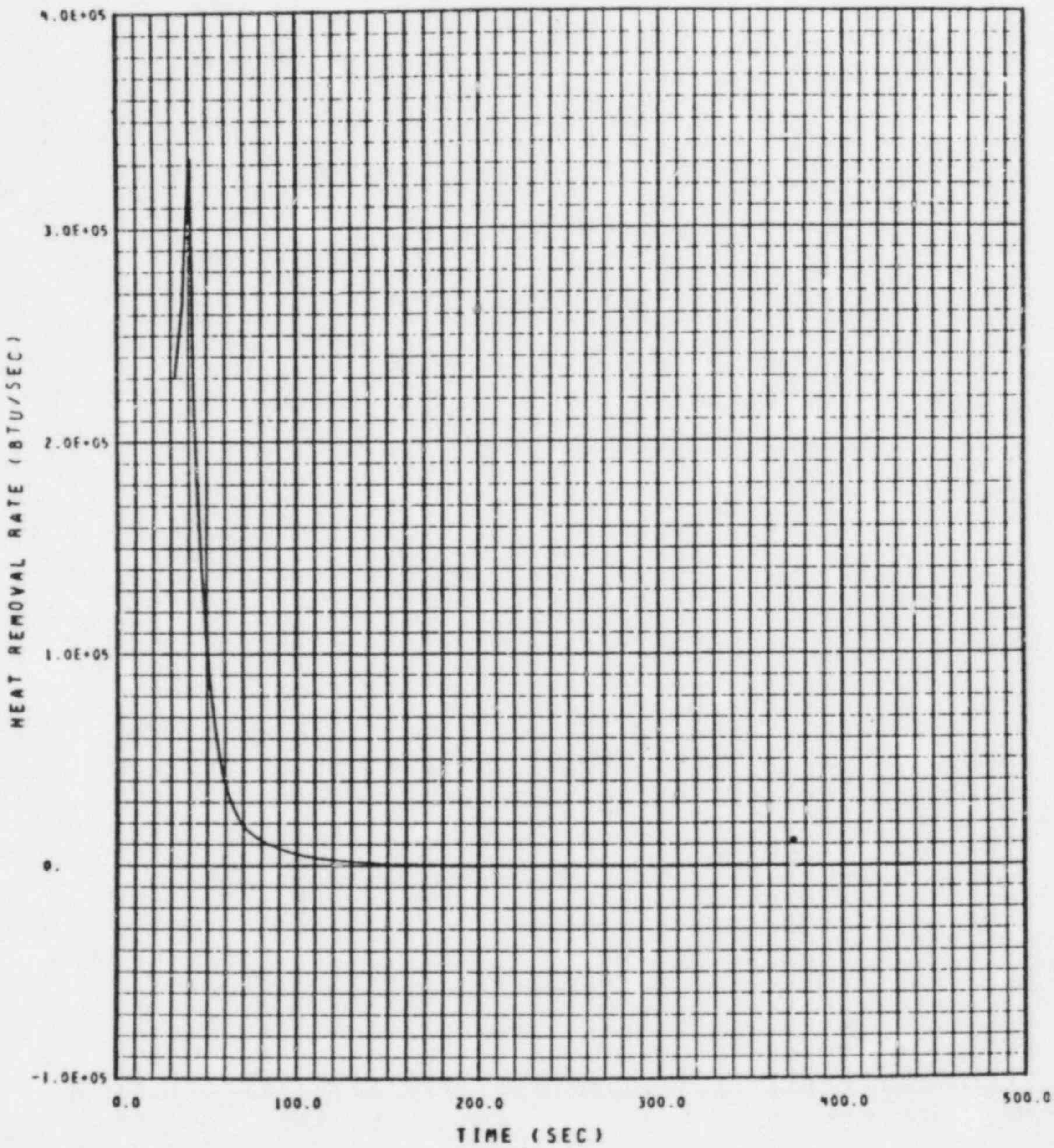
MAXIMUM SI

FIGURE 14.3.1-56 LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



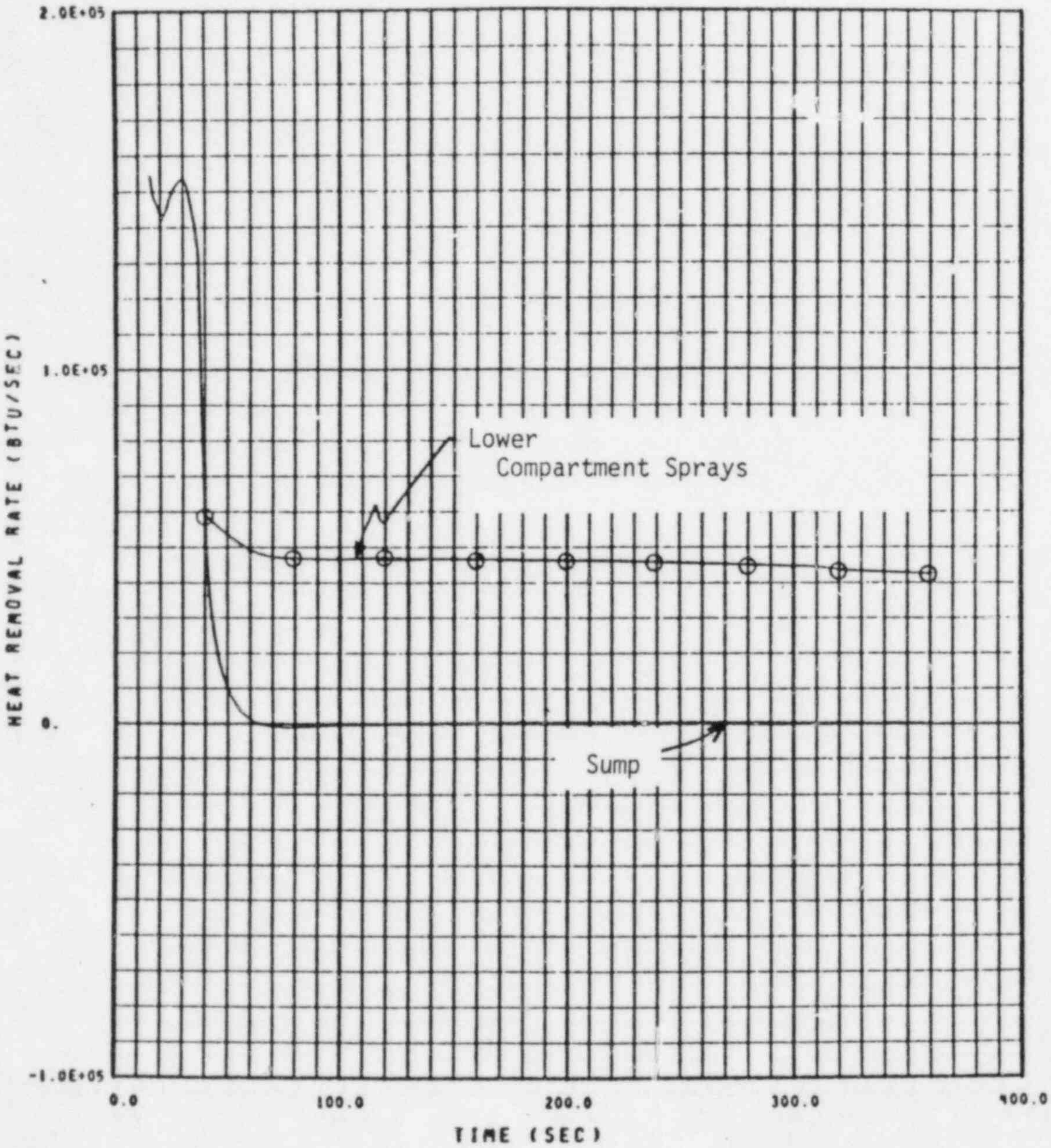
MINIMUM SI

FIGURE 14.3.1-57 HEAT REMOVAL BY LC DRAIN



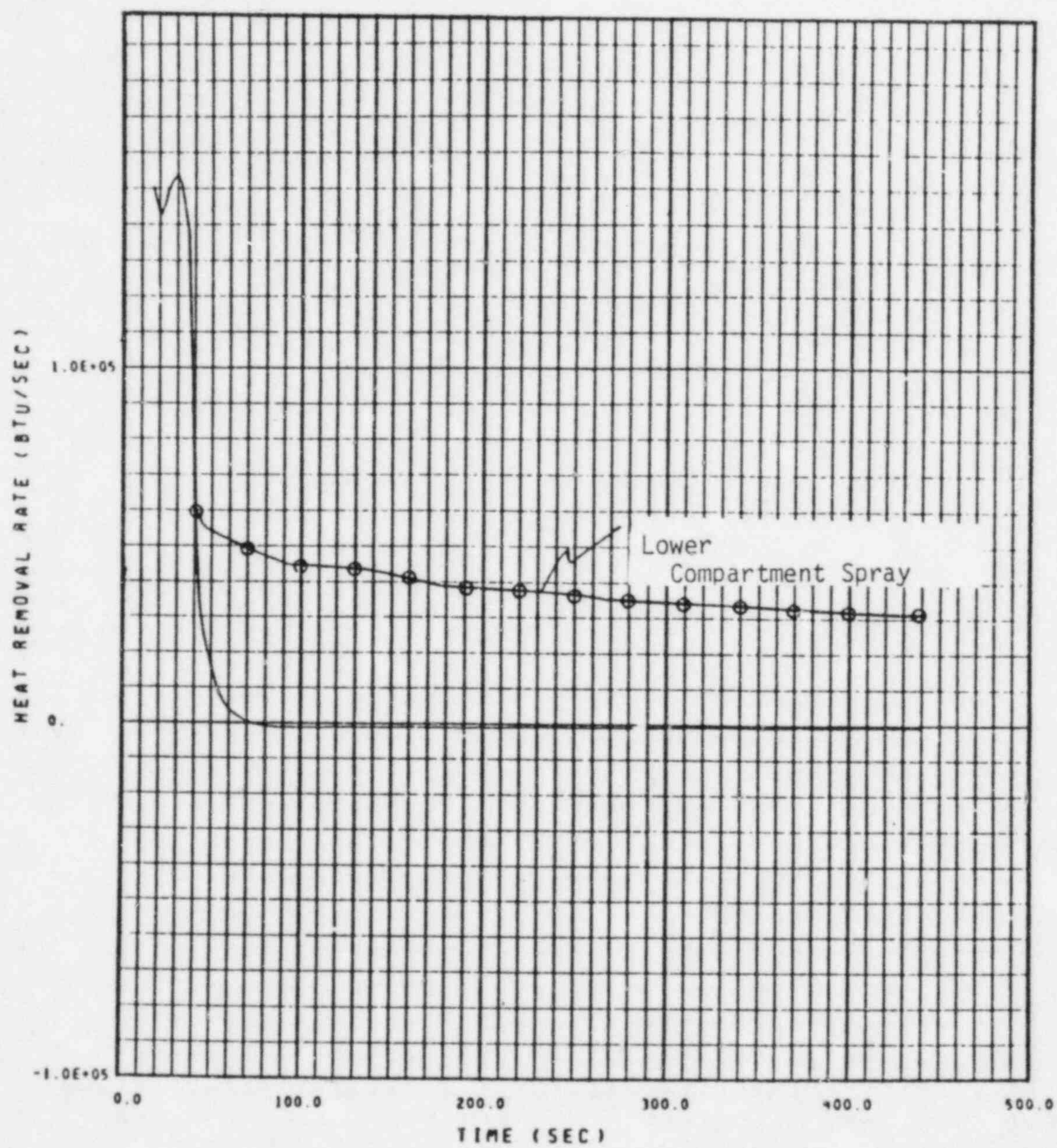
MAXIMUM SI

FIGURE 14.3.1-58 HEAT REMOVAL BY LC DRAIN



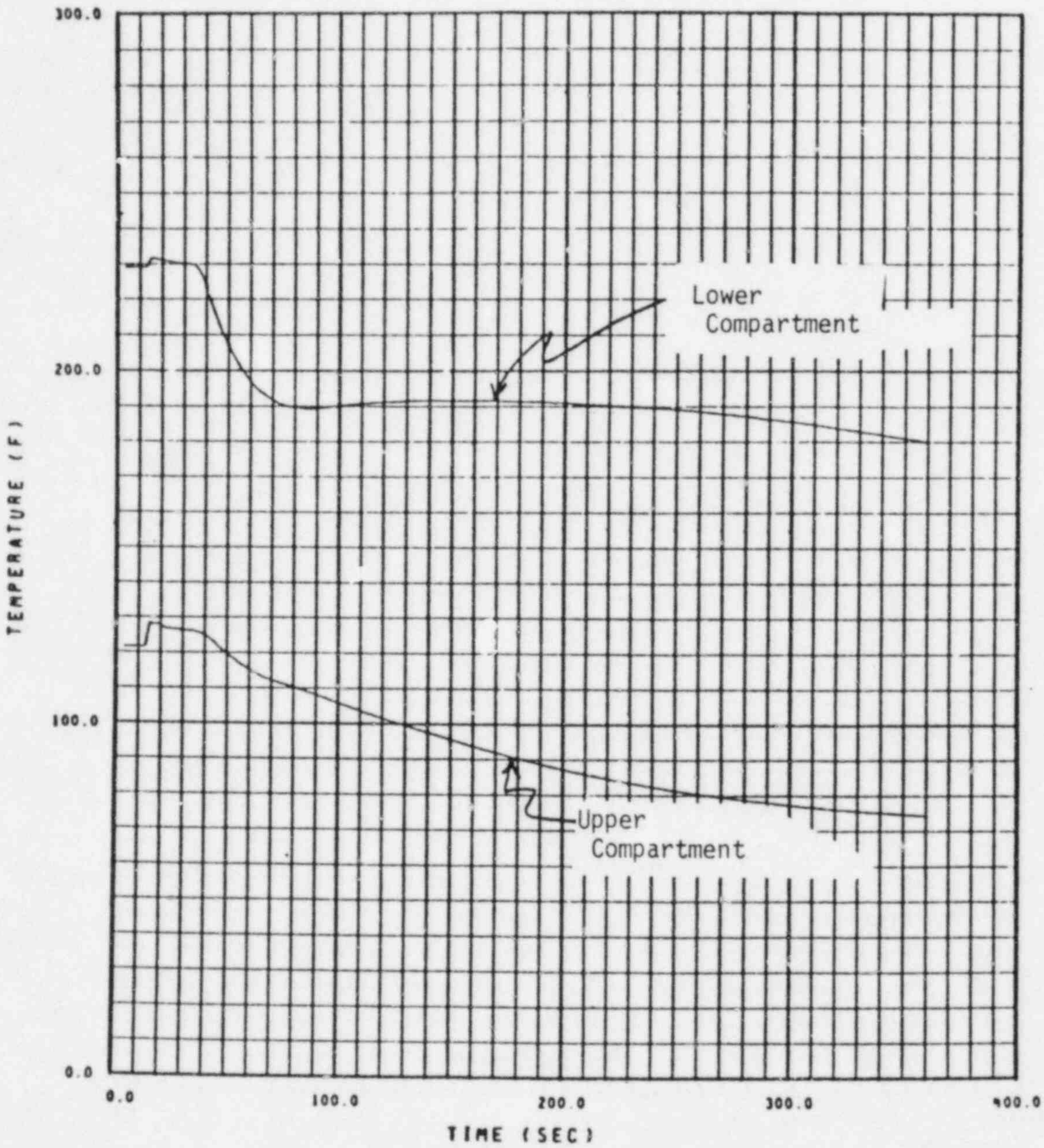
MINIMUM SI

FIGURE 14.3.1-59 HEAT REMOVAL BY SUMP AND LC SPRAYS



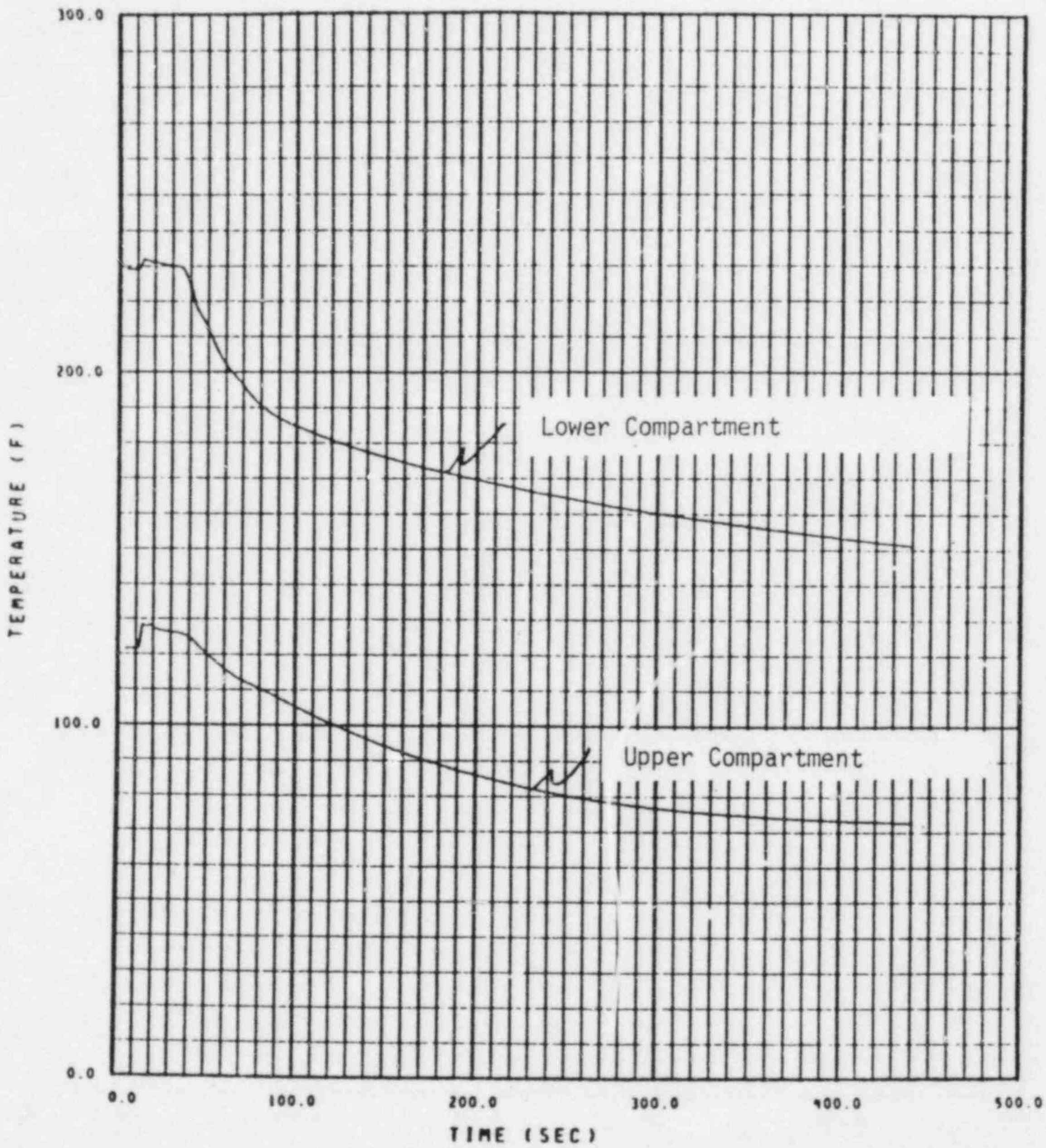
MAXIMUM SI

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

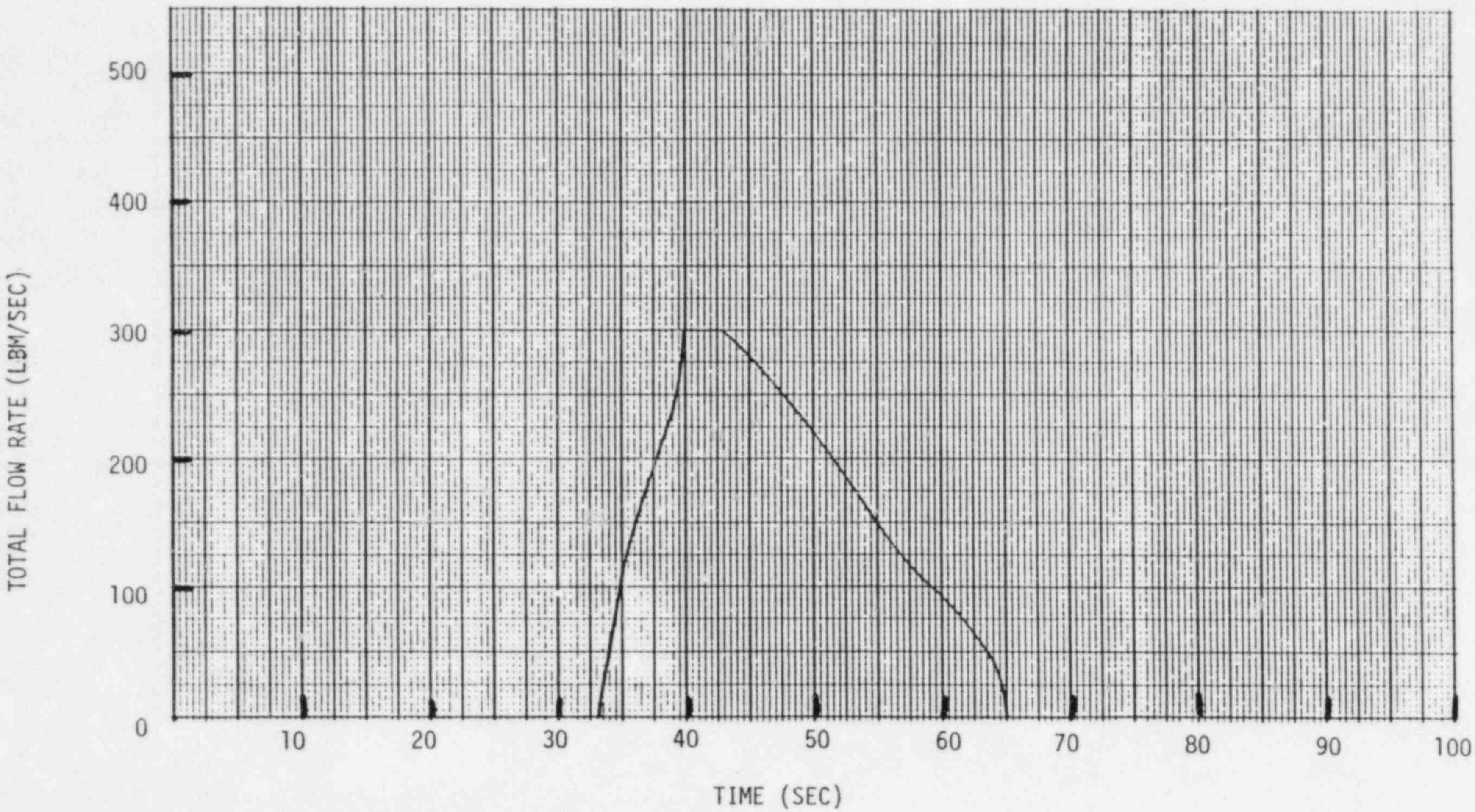


FIGURE 14.3.1-63 FLOW FROM THE UPPER TO LOWER COMPARTMENT
MIN. SI

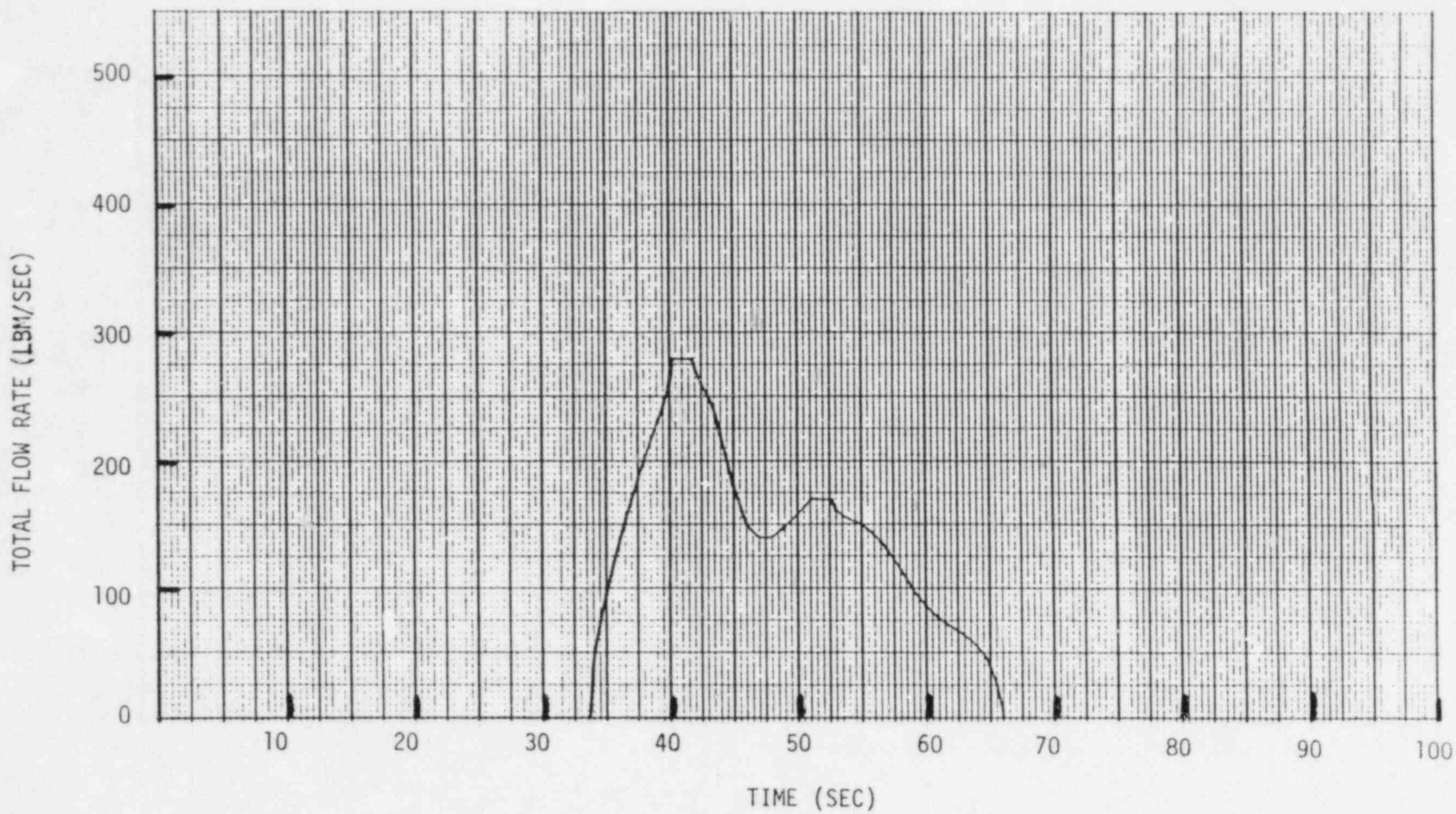


FIGURE 14.3.1-64 FLOW FROM THE UPPER TO LOWER COMPARTMENT
MAX. SI

ADDENDUM A

In order to assure a timely licensing approval for Cycle 8 startup operations, the worst break case was reanalyzed using the currently approved PAD fuel thermal safety model [16] initial fuel rod conditions. The worst break, which was reanalyzed, is the 0.4 C_D (discharge coefficient) Maximum Safeguards Injection (Max SI) case. Those results are presented in this Addendum.

Results of this calculation are summarized in Tables 14.3.1A-4 and 14.3.1A-5. Other tables and figures presented are analogous to those previously presented for the revised PAD analysis. The maximum clad temperature calculated for a large break is 2162°F, which is less than the acceptance criteria limit of 2200°F. The maximum local metal-water reaction is 6.58 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 periof of time will be provided.

TABLE 14.3.1-1A-1

LARGE BREAK

CONTAINMENT DATA

(ICE CONDENSER CONTAINMENT)

NET FREE VOLUME		
(Includes Distribution Between Upper, Lower,	UC	746,829 ft^3
and Dead-Ended 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 14.3.1-1A-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 14.3.1A-2

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	.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	.6398E+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	.4954E+04	.1422E+07
.3600E+02	.3380E+04	.9903E+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	.2575E+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-A-3

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-A-4

LARGE BREAK

Results	DECLG C_D=0.4 Max SI
Peak Clad Temp. °F	2162
Peak Clad Location Ft.	7.50
Local Zr/H_2O Reaction (Max)%	6.58
Local Zr/H_2O Location Ft.	7.50
Total Zr/H_2O Reaction %	< 0.3
Hot Rod Burst Time sec.	71.4
Hot Rod Burst Location Ft.	6.75

Calculation	
Licensed Core Power (Mwt) 102% of	3250
Peak Linear Power (kw/ft) 102% of	13.225
Peaking Factor (at License Rating)	1.97
Accumulator Water Volume (ft^3) per Accumulator	950

Cycle Analyzed Cycle 8

TABLE 14.3.1-A-5

LARGE BREAK

TIME SEQUENCE OF EVENTS

	Max SI DECLG C_D=0.4 (sec)
START	0.00
Reactor Trip Signal	0.60
Safety Injection Signal	4.05
Accumulator Injection	20.50
End of Blowdown	38.70
Bottom of Core Recovery	52.78
Accumulator Empty	67.45
Pump Injection	29.05

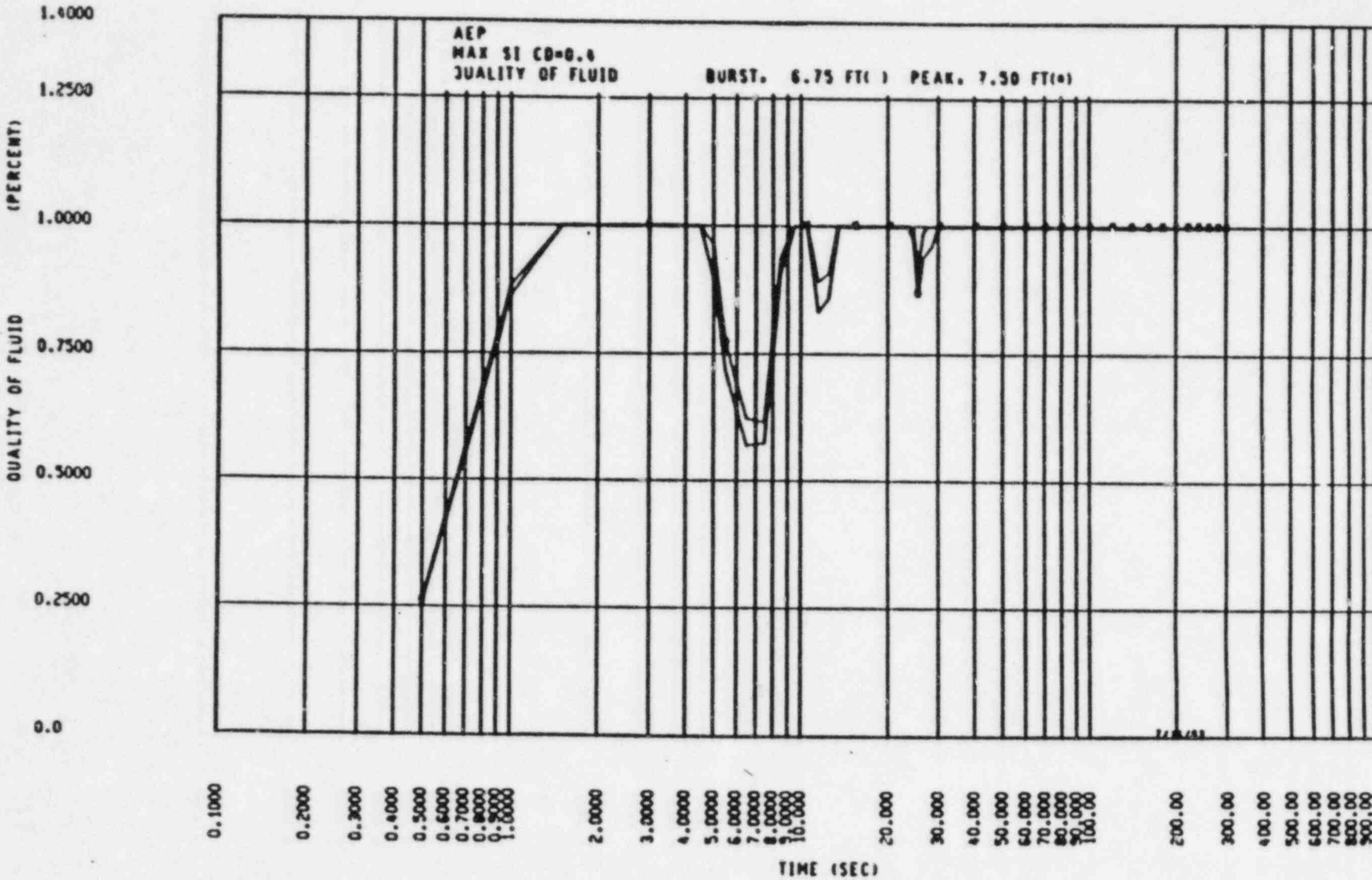


Figure 14.3.1A-1 FLUID QUALITY DECLG(CD = 0.4) MAX SI

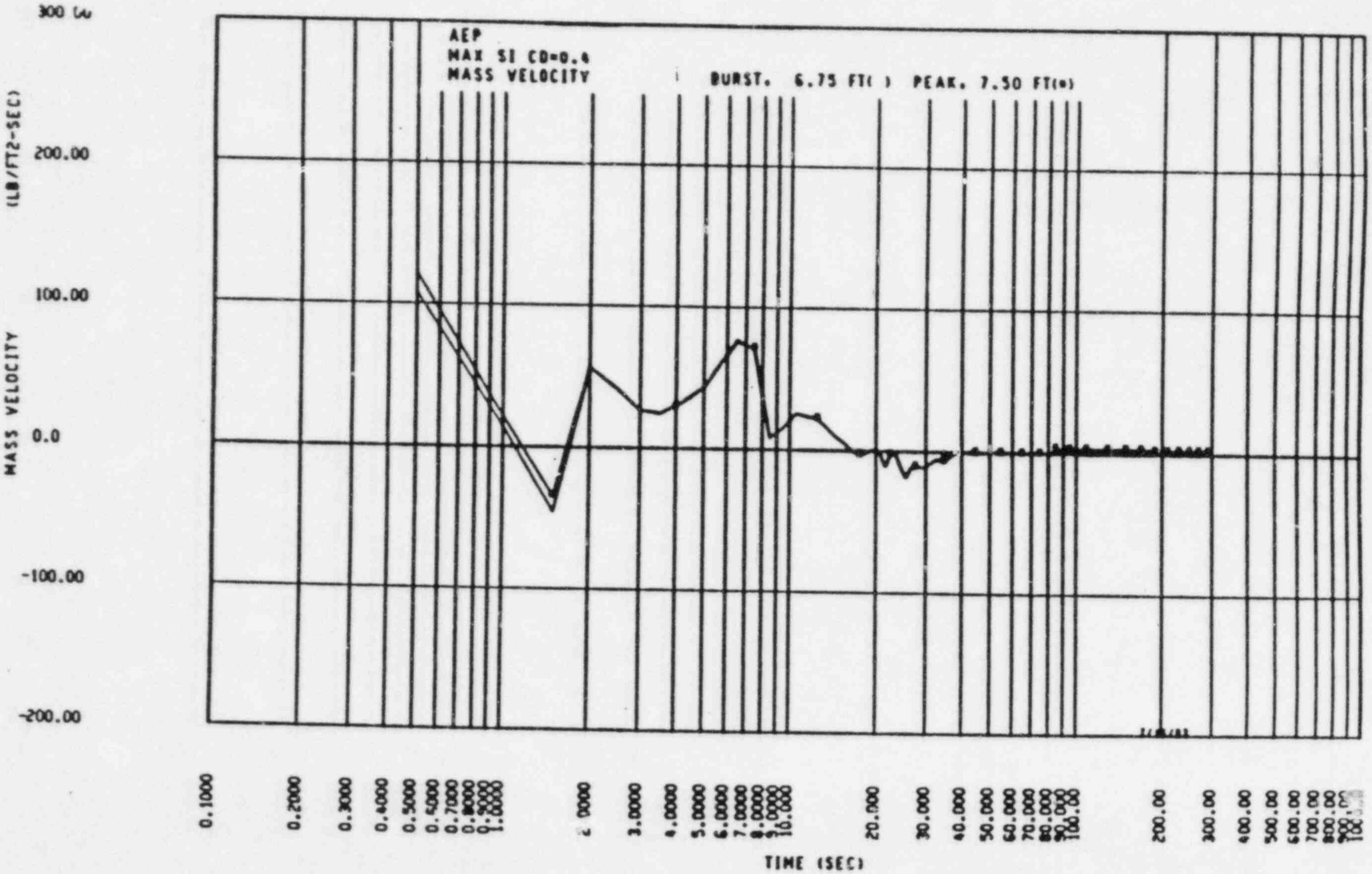


Figure 14.3.1A-2 MASS VELOCITY MAX
DECLG(CD = 0.4) SI

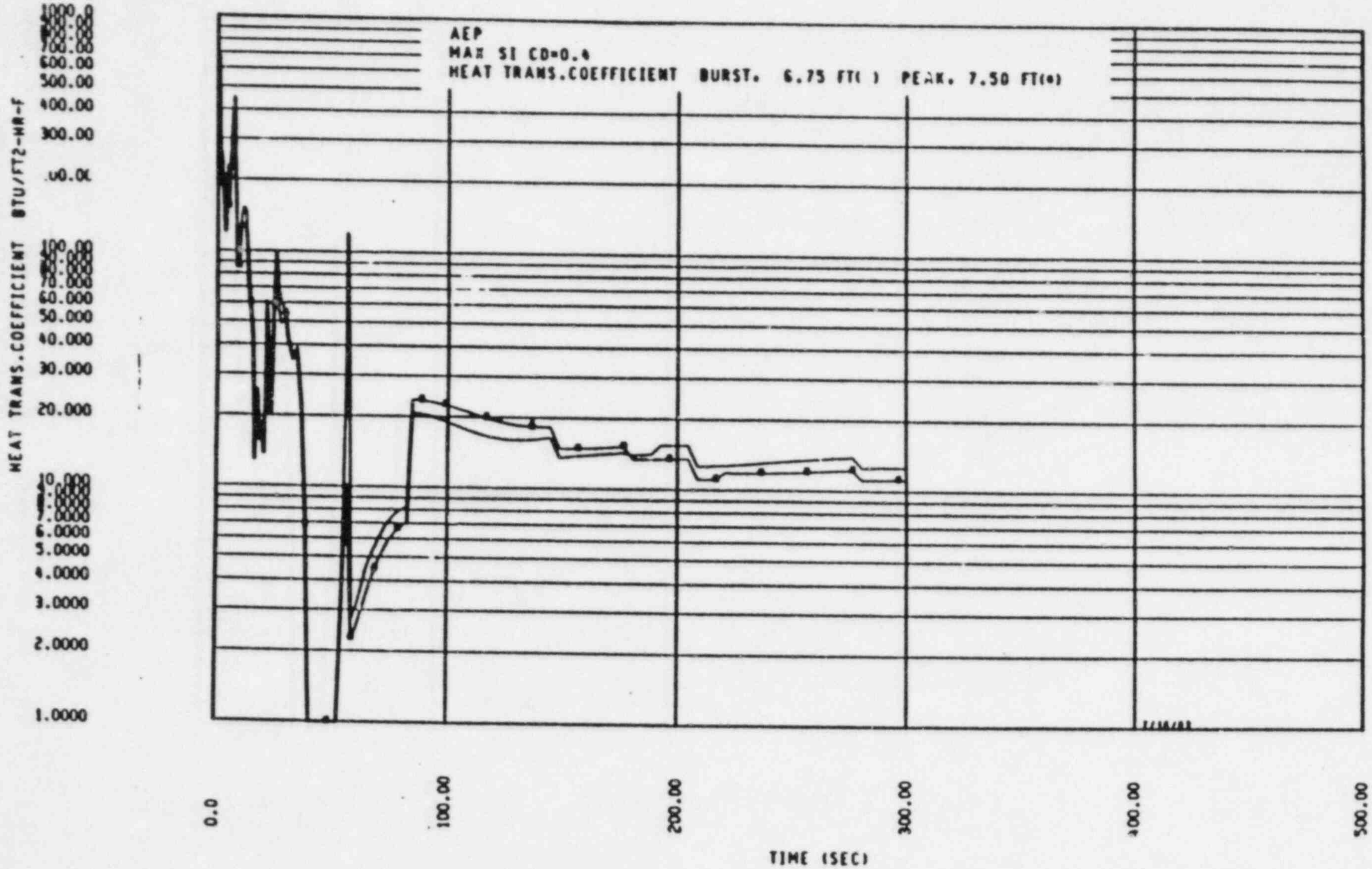


Figure 14.3.1A-3 HEAT TRANSFER COEFFICIENT
DECLG(CD=0.4) MAX SI

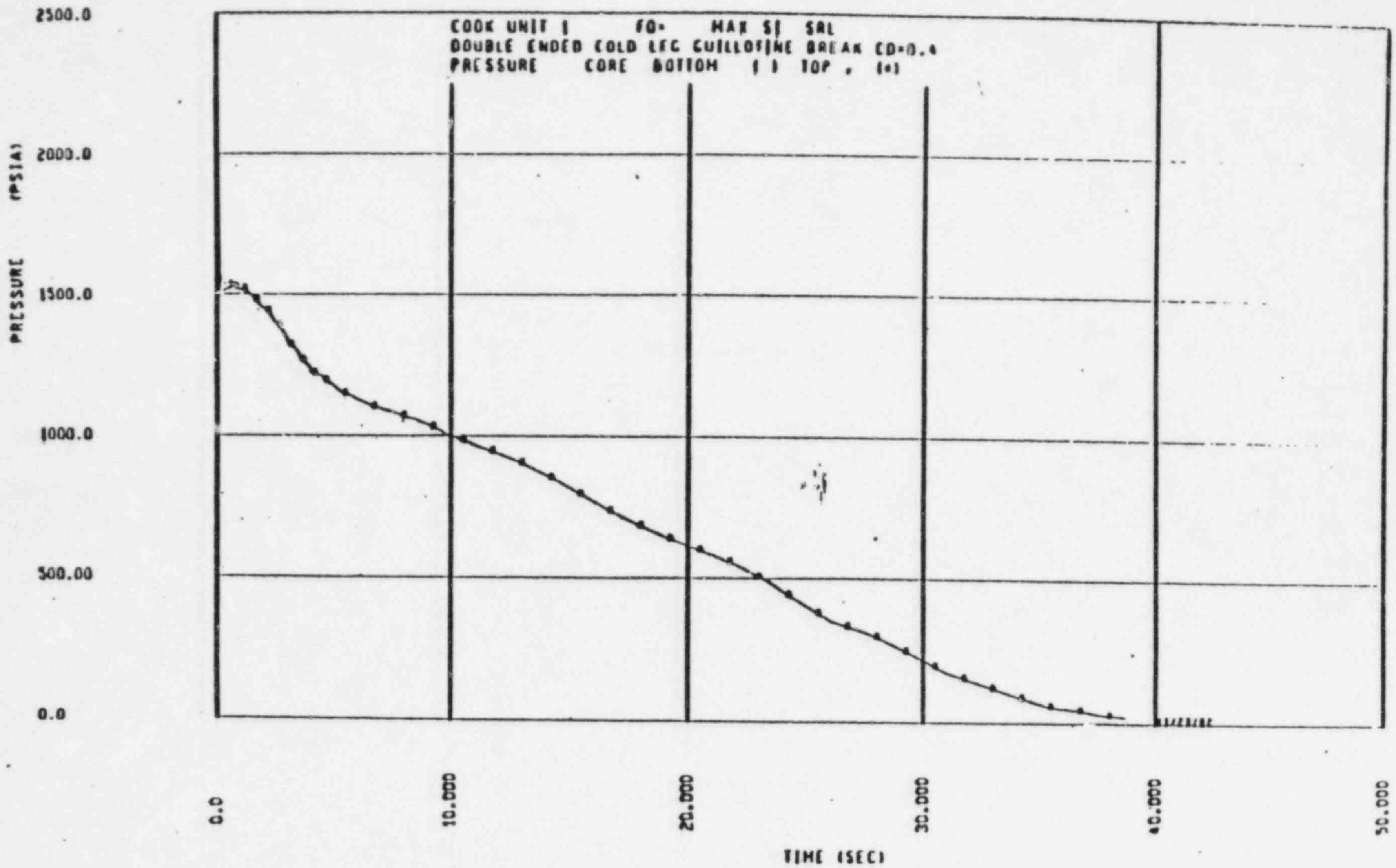


Figure 14.3.1A-4 CORE PRESSURE MAX
DECLG(CD = 0.4) SI

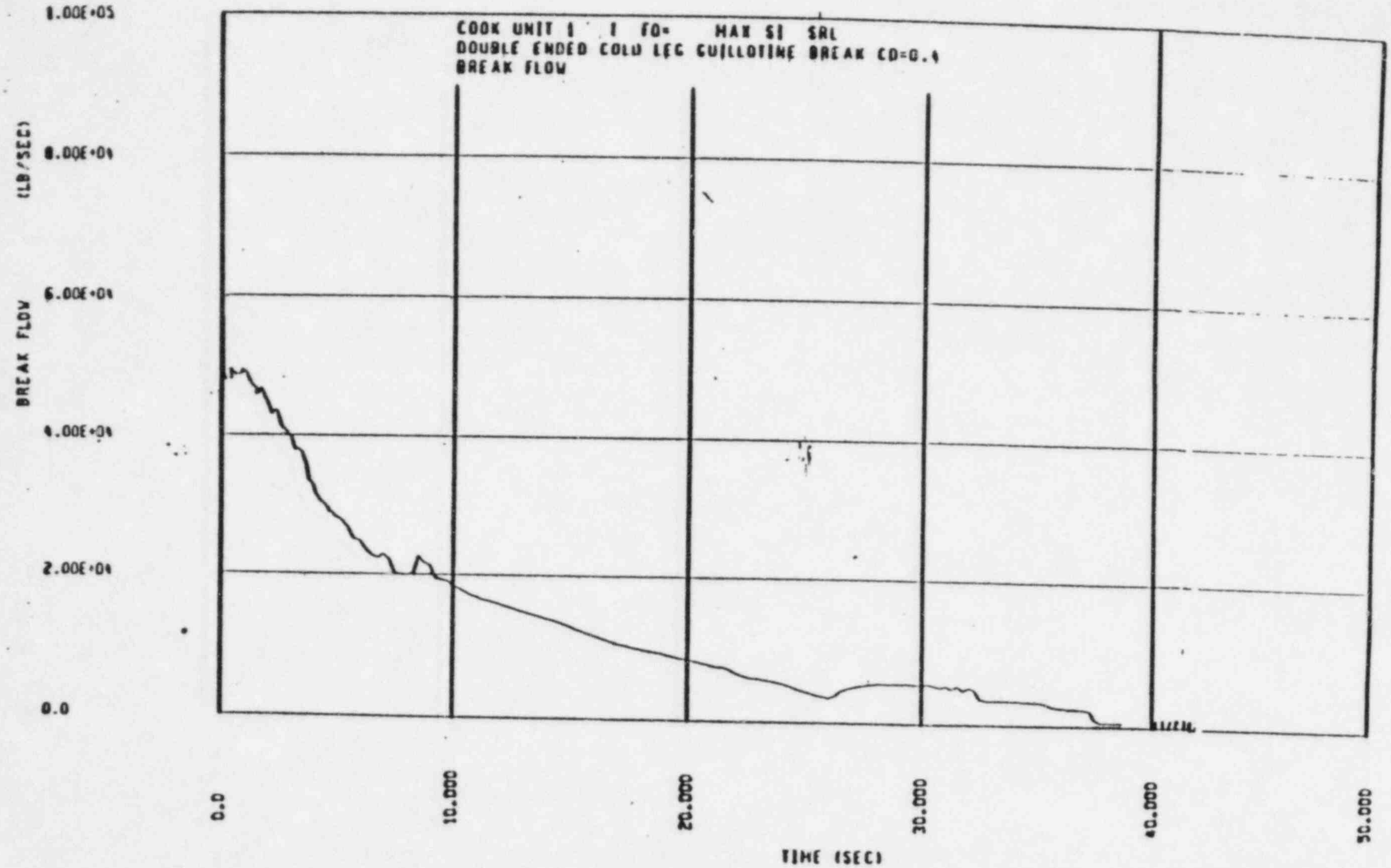


Figure 14.3.1A-5 BREAK FLOW RATE MAX
DECLG(CD=0.4) SI

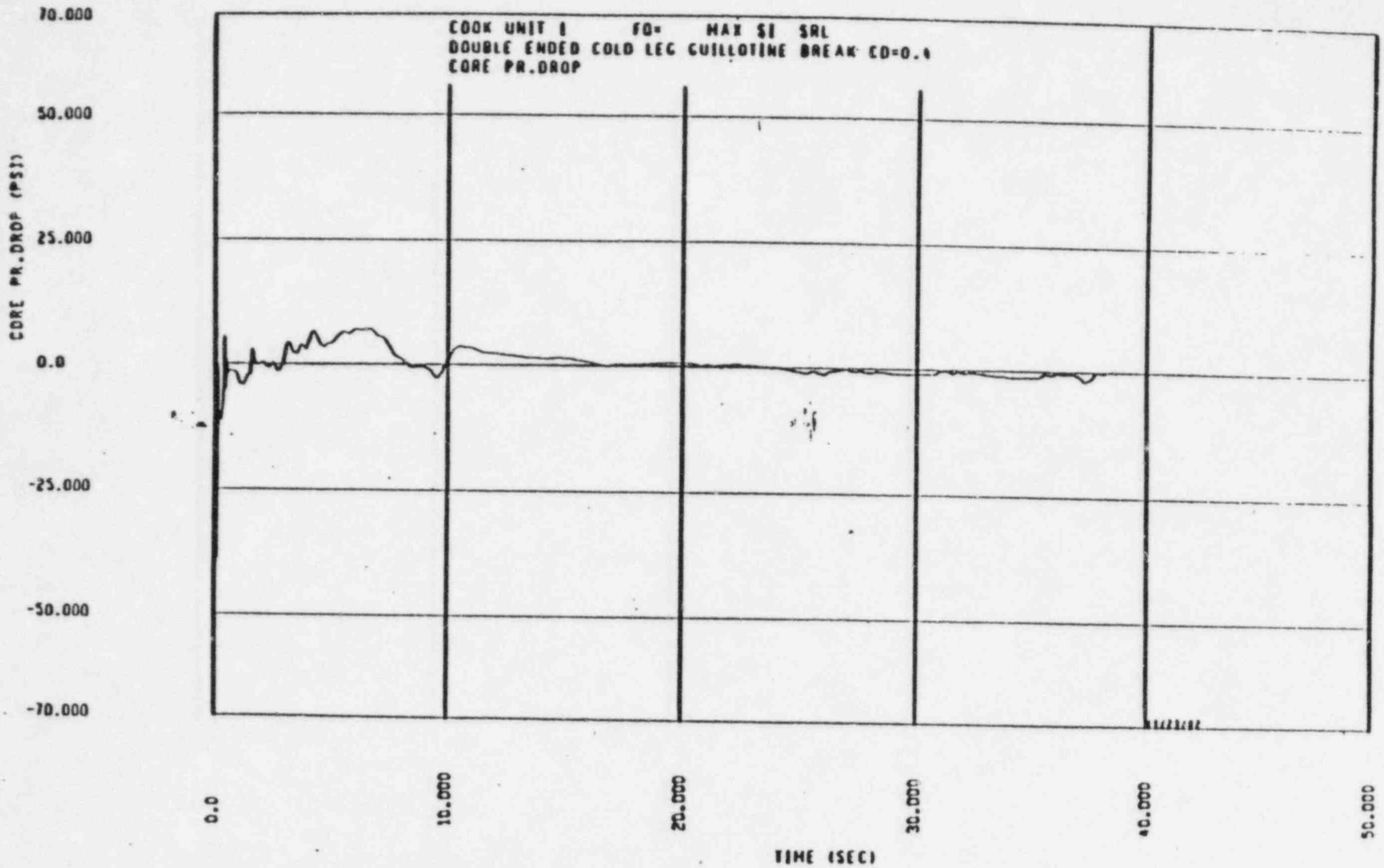


Figure 14.3.1A-6 CORE PRESSURE DROP DECLG(CD = 0.4) MAX SI

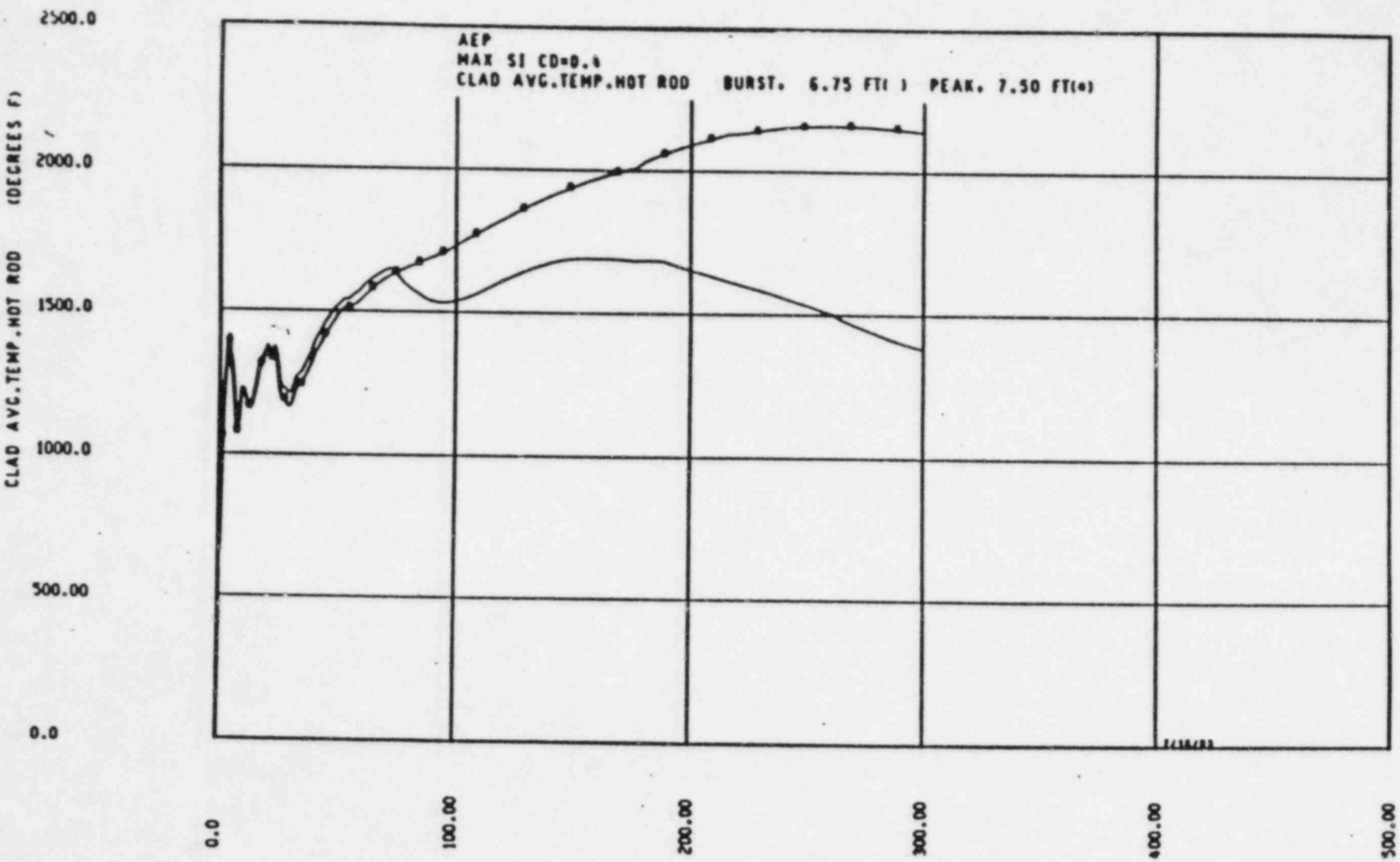


Figure 14.3.1A-7 PEAK CLAD TEMPERATAURE DECLG(CD = 0.4) MAX SI

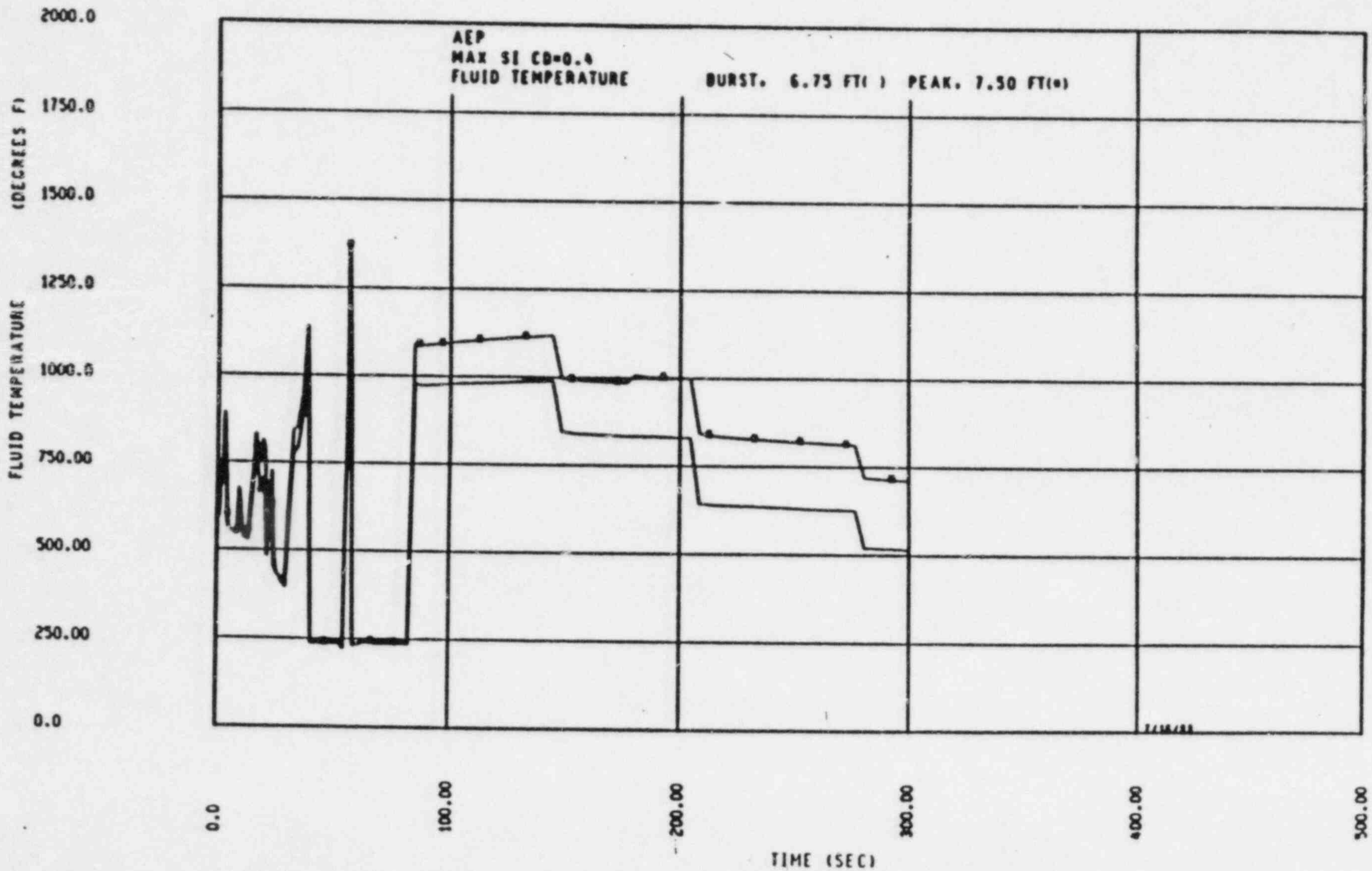


Figure 14.3.1A-8 FLUID TEMPERATURE DECLG(CD = 0.4) MAX SI

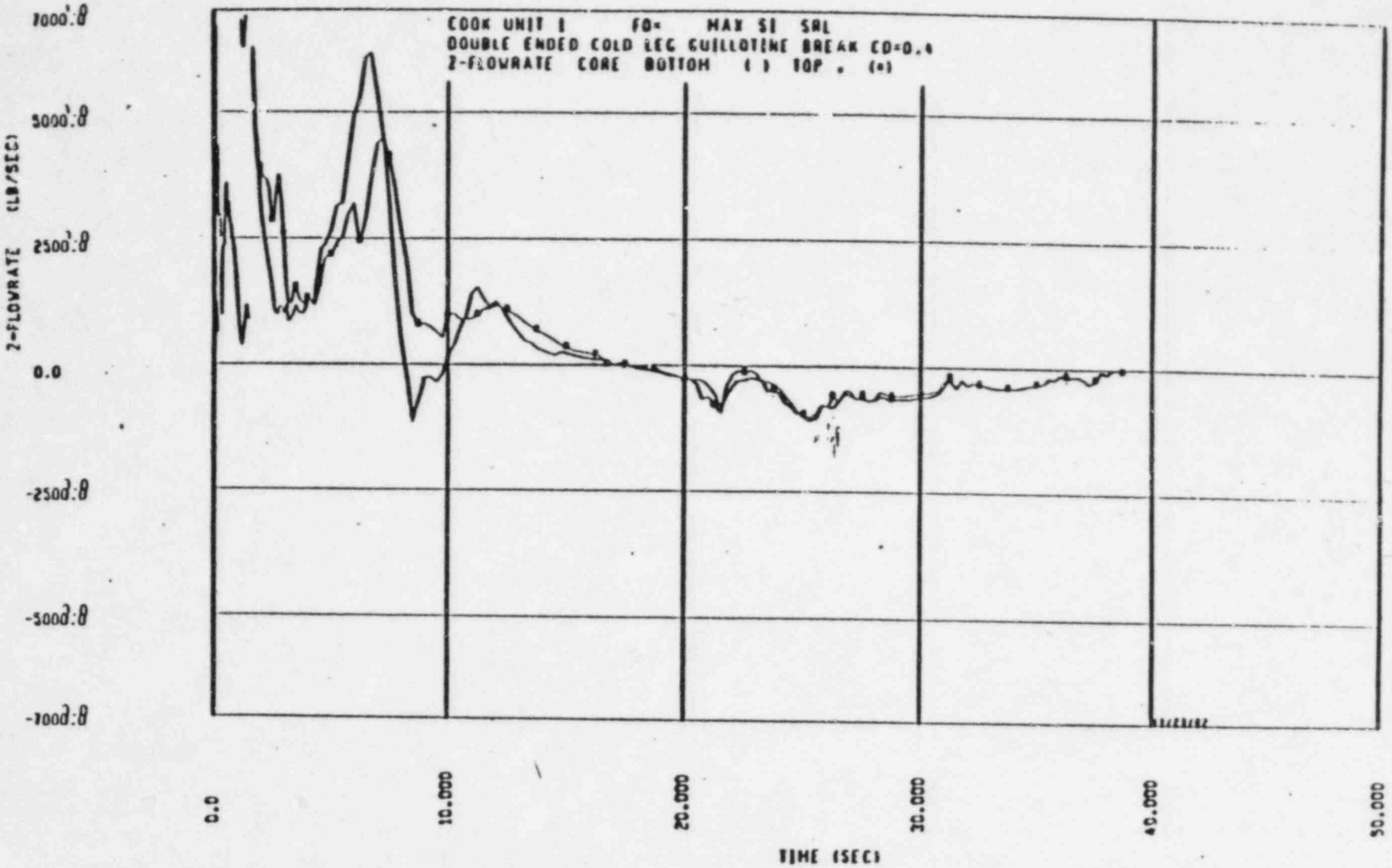


Figure 14.3.1A-9 CORE FLOW (TOP AND BOTTOM) DECLG(CD = 0.4) MAX SI

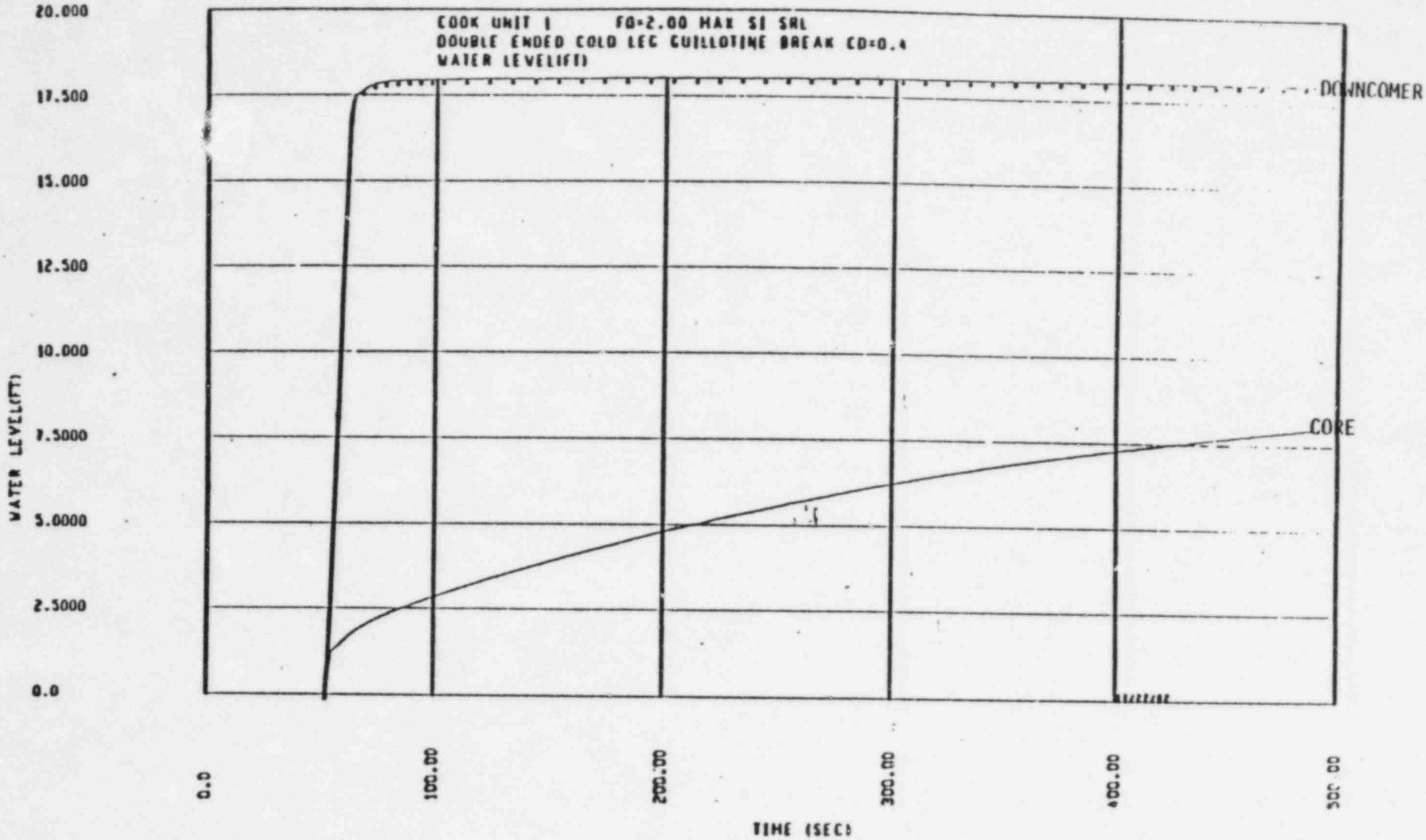


Figure 14.3.1A-10 REFLOOD TRANSIENT - CORE & DOWNCOMER WATER LEVELS DECLG(CD = 0.4) MAX SI

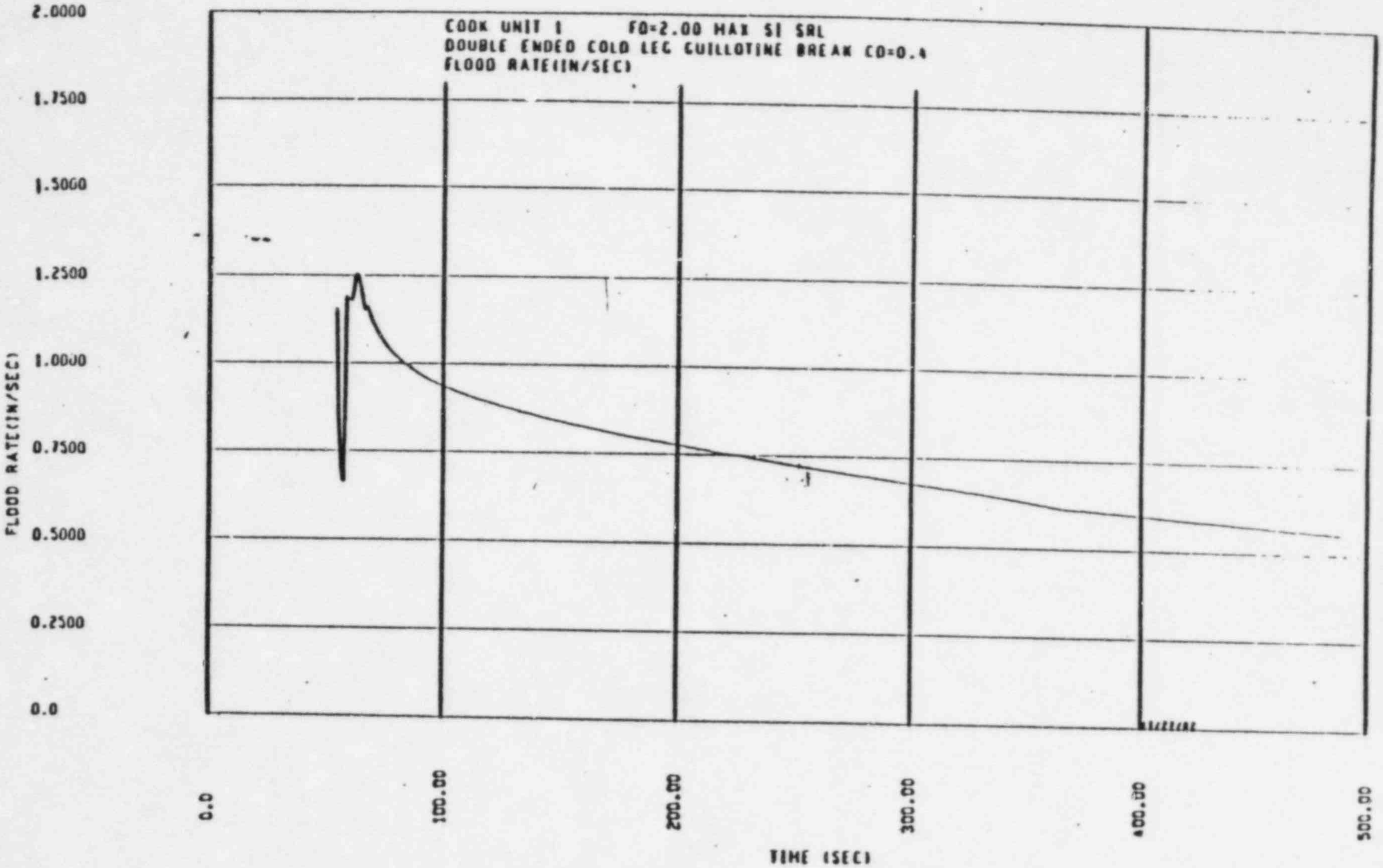


Figure 14.3.1A-11 REFLOOD TRANSIENT CORE INLET VELOCITY DECLG(CD = 0.4) MAX SI

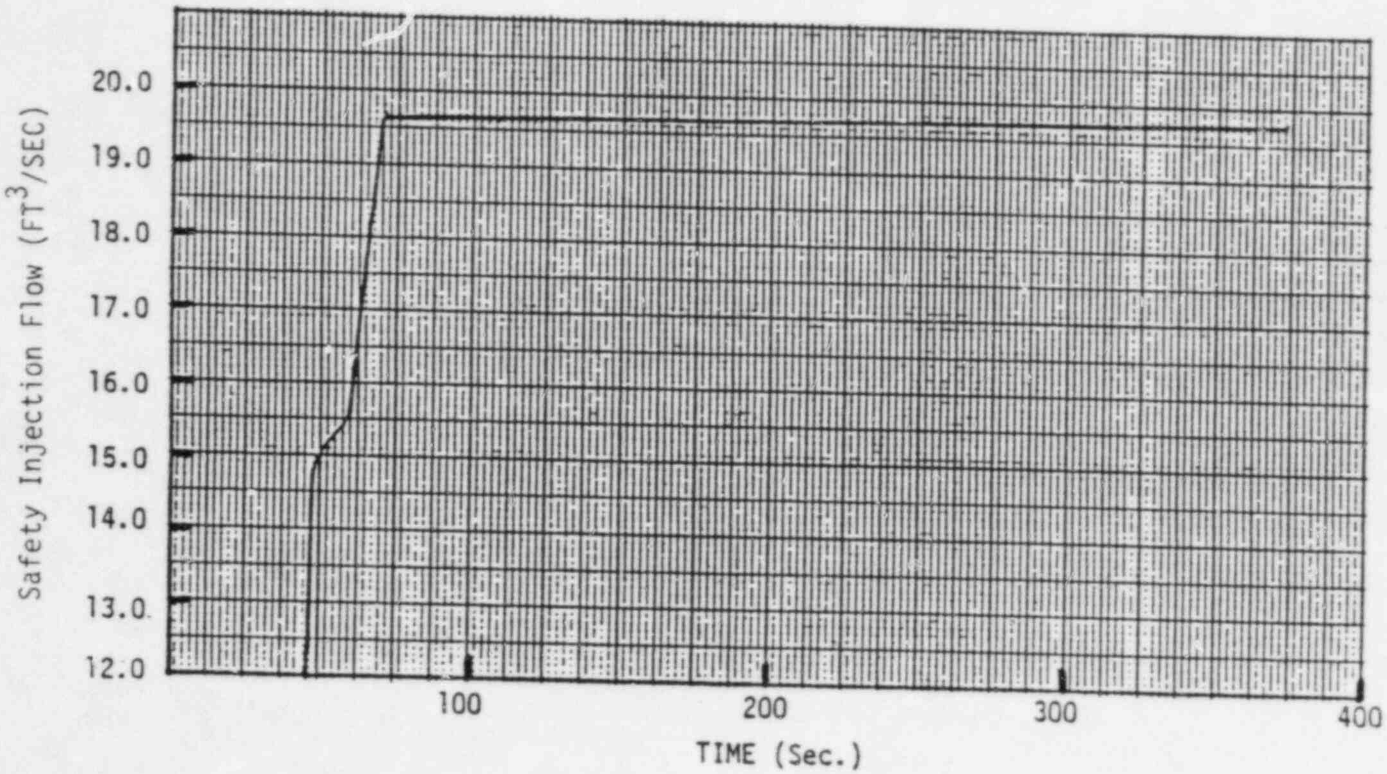


Figure 14.3.1A-12 PUMPED ECCS FLOW (REFLOOD - DECLG CD - 0.4) MAX SI

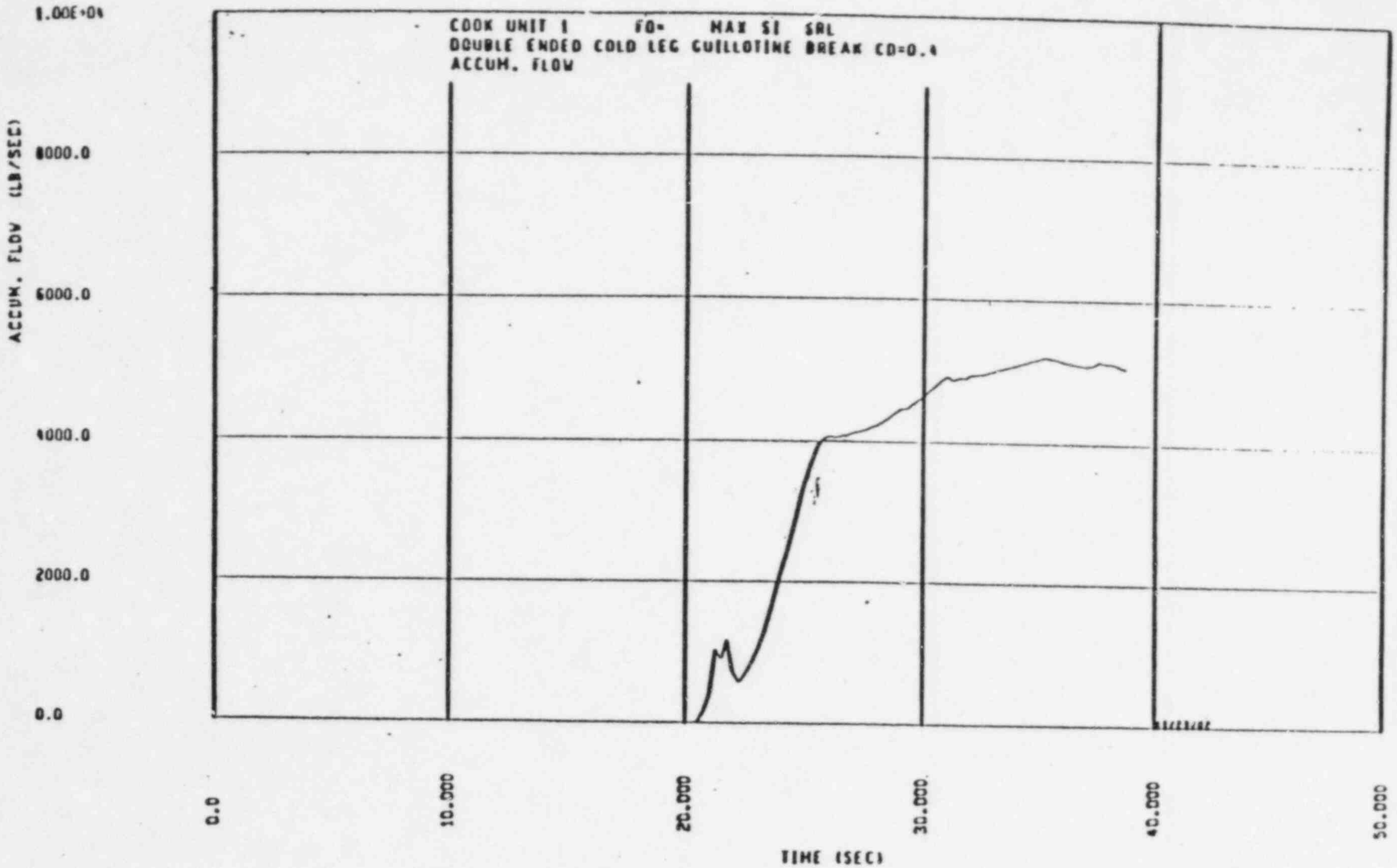
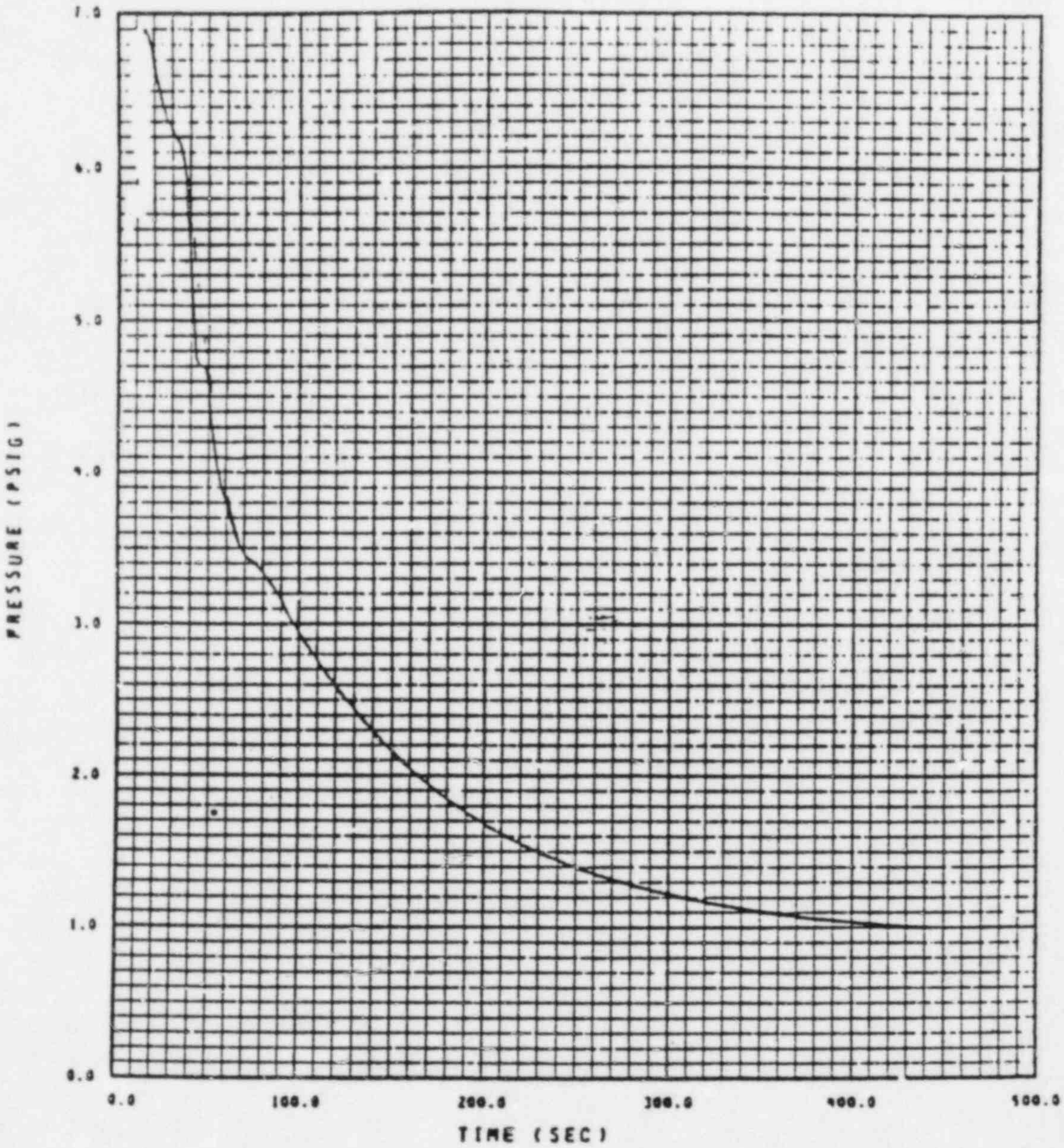


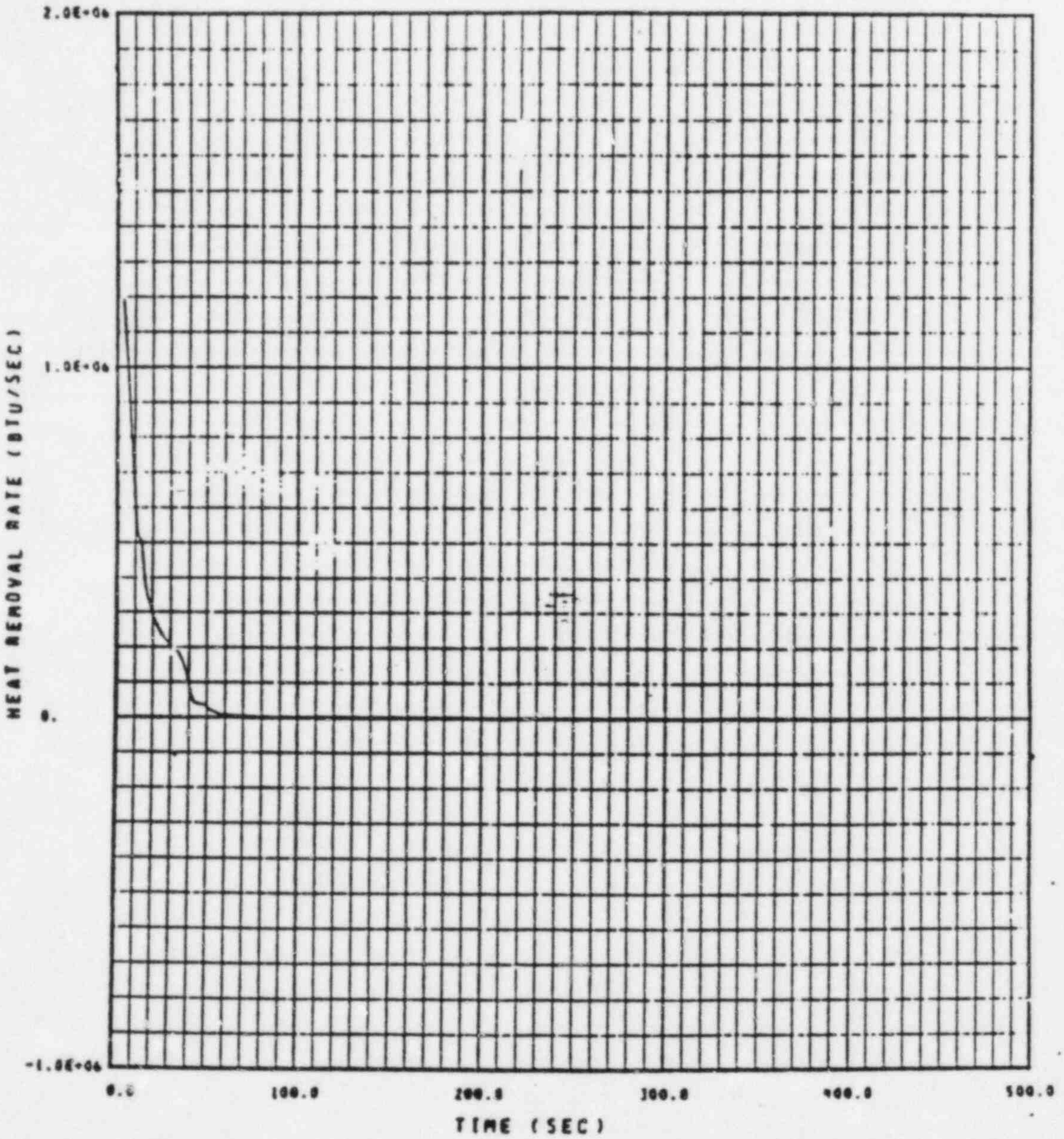
Figure 14.3.1A-13 ACCUMULATOR FLOW (BLOWDOWN) MAX
DECLG(CD = 0.4) SI



COMPARTMENT PRESSURE

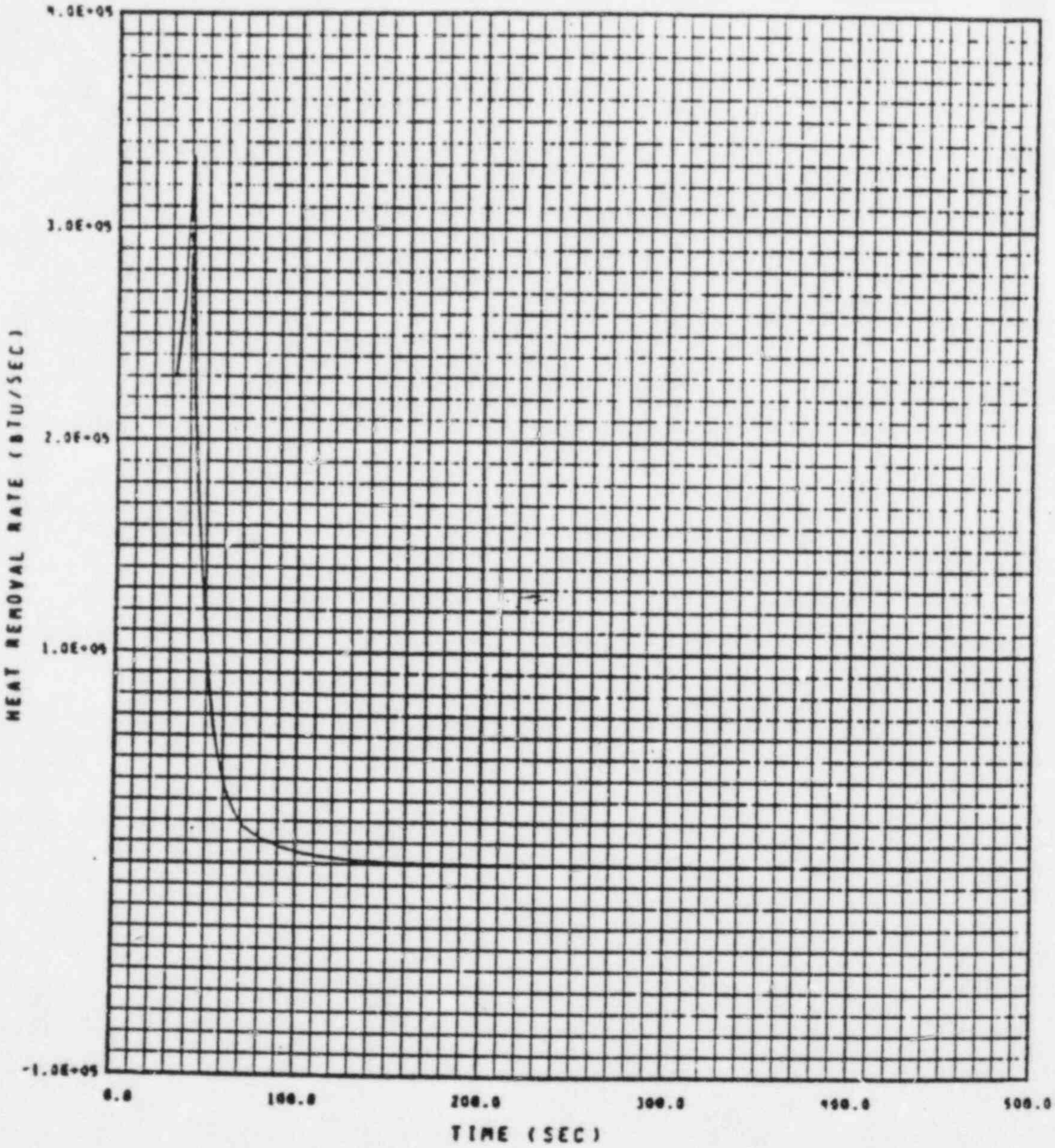
MAXIMUM SI

Figure 14.3.1A-14 CONTAINMENT PRESSURE
DECLG(CD = 0.4)



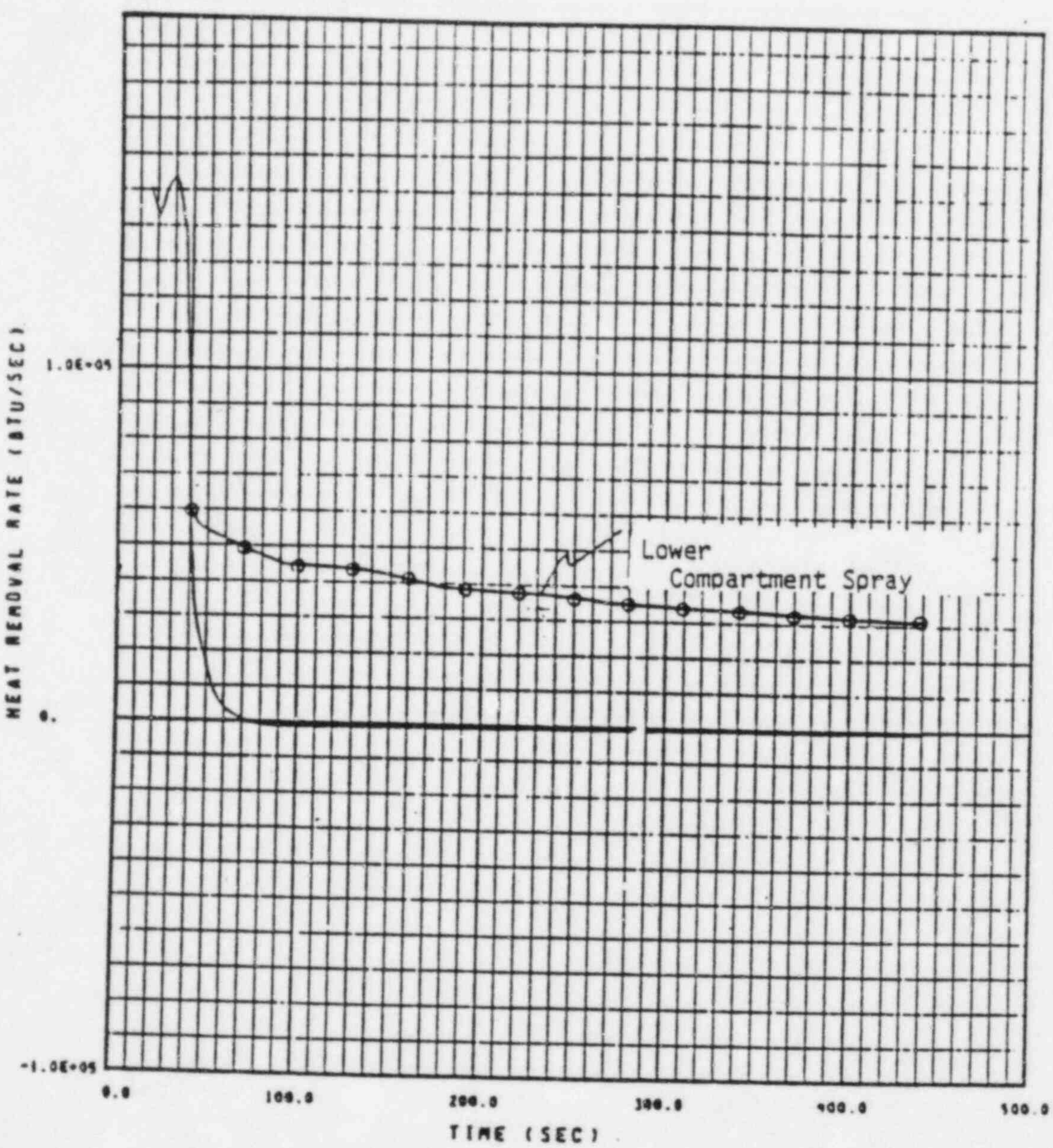
MAXIMUM SI

Figure 14.3.1A-15 LOWER COMPARTMENT STRUCTURAL HEAT REMOVAL RATE



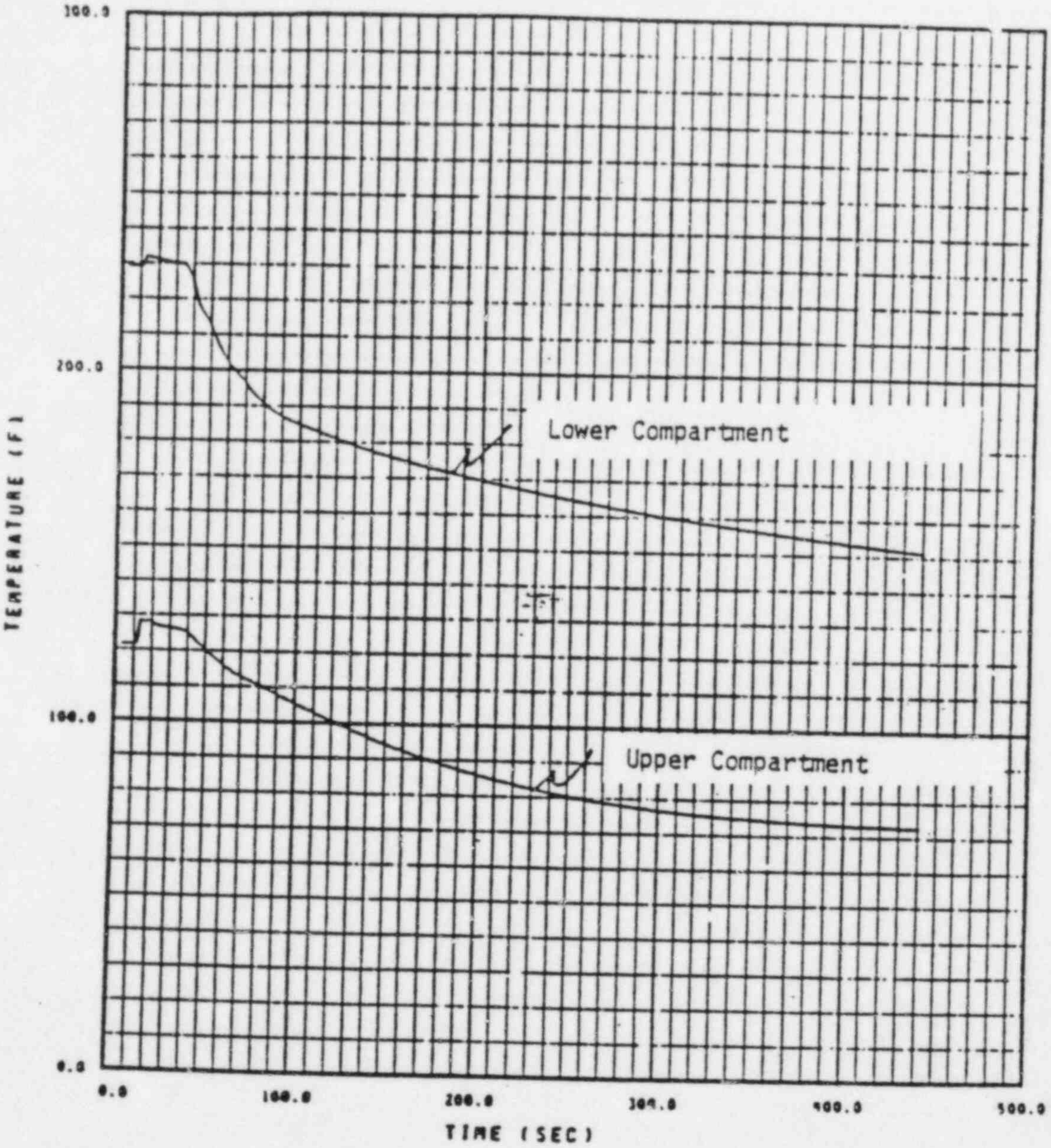
MAXIMUM SI

Figure 14.3.1A-16 HEAT REMOVAL BY LC DRAIN



MAXIMUM SI

Figure 14.3.1A-17 HEAT REMOVAL BY SUMP AND LC SPRAY



MAXIMUM SI

Figure 14.3.1A-18 COMPARTMENT TEMPERATURE

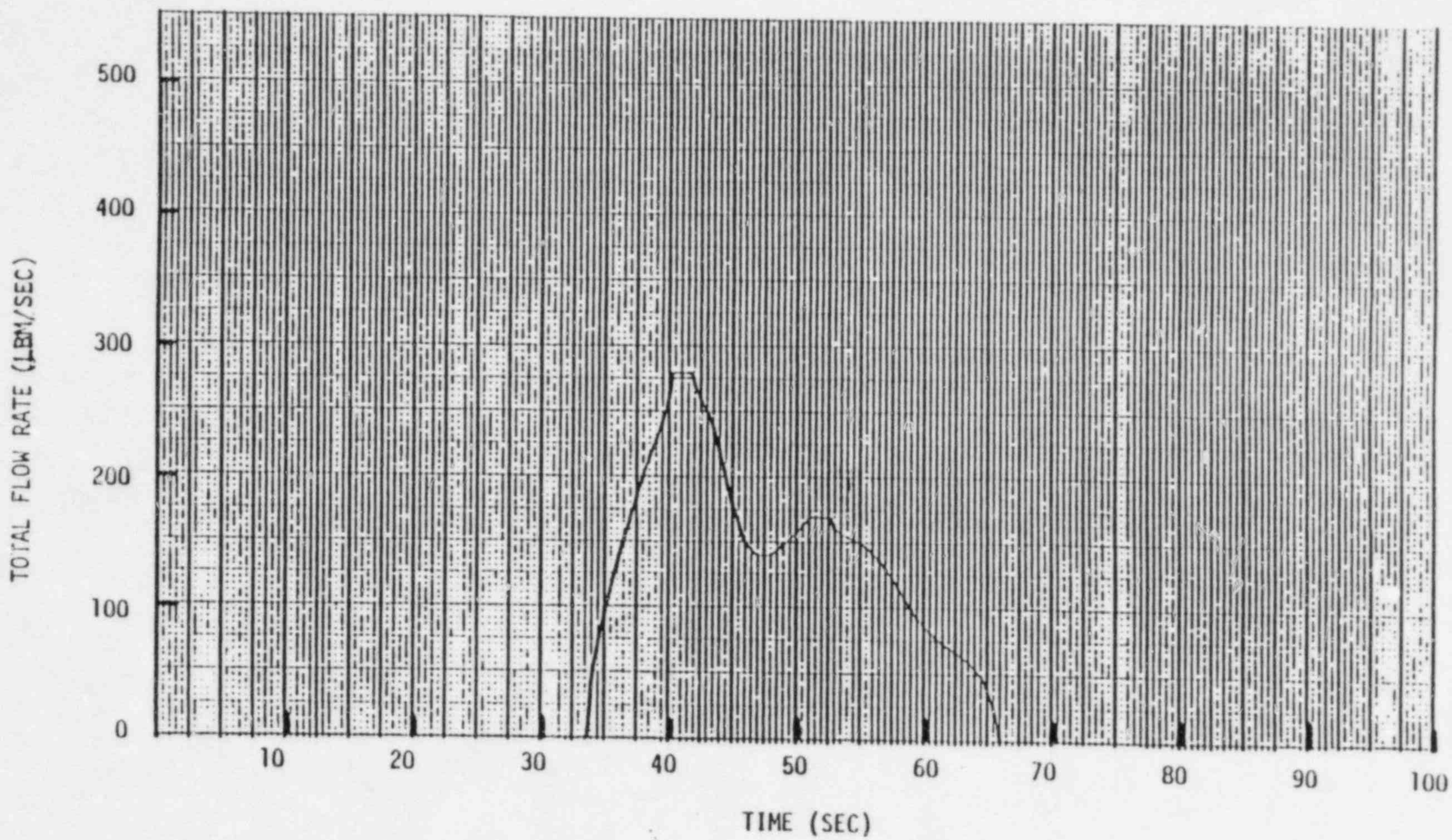


Figure 14.3.1A-19 FLOW FROM THE UPPER TO LOWER COMPARTMENT
MAX. SI

Attachment D to AEP:NRC:0745F
Revision to Description of Proposed Technical Specifications
(As Found in Attachment F to AEP:NRC:0745C)

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 1.97 (3.94 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 fuel
		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 Tec Spec and defined in peaking factor lir report. This report will be availabl in the licensee's offices 60 days pric 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

Attachment E to AEP:NRC:0745F
Responses to Reactor Physics Questions
in Letter, Varga to Dolan, June 29, 1983

Q1. Since this is apparently to be the reference cycle for the use of the Westinghouse OFA design fuel, please provide a summary table of core nuclear parameters against which future cycles may be compared. Also provide a comparison of these parameters to those for the present reference cycle.

Response: Listed below is a summary table of the core nuclear parameters against which future cycles may be compared. A comparison of these parameters to those for the reference cycle is not meaningful since all of the accidents described in the applicant's FSAR Chapter 14 were performed using the new parameters. Future cycles will be analyzed according to the methodology described in WCAP-9273 "Westinghouse Reload Safety Evaluation Methodology."

Kinetics Characteristics
D. C. Cook Unit 1

Moderator Temperature Coefficient ($pcm/^oF$)	-35 to 5.0 (power less than 70%) -35 to 0.0 (power greater than or equal 70%)
Doppler Coefficient ($pcm/^oF$)	-2.9 to 1.4
Delayed Neutron Fraction, β_{eff} (percent)	0.44 to 0.75
Maximum Prompt Neutron Lifetime (μsec)	26

Q2. In Table 2 of Attachment C, notes (2) and (3), the equations seem to be for power Doppler coefficient rather than for Doppler defect. Please confirm.

Response: We have confirmed this with our fuel vendor. Please see the revised Table 2, to appear on pages 55 and 56 of Attachment C to AEP:NRC:0745C. Please also note two other changes to this table:

1. Initial NSSS Thermal Power Output for Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power is now 3425/2055/343 instead of 3425/2124/411. These new powers are the correct input when using the Improved Thermal Design Procedure.

2. Initial NSSS Thermal Power Output for Uncontrolled Boron Dilution is now 0 and 3425 instead of 0 and 3415. This is a typographical correction.

Q3. In Section 6.3.2, what is the value of the conservatively large negative moderator coefficient?

Response; The negative moderator temperature coefficient used in the rod withdrawal from power analysis is $-35.0\ pcm/^oF$.

Q4. In Figures 10, 11, and 12 of Attachment C, the distinction between maximum and minimum feedback curves s not clear. Please clarify.

Response: The minimum feedback curves are the solid ones and the maximum feedback curves are the dashed ones for all three figures. Please see the revised like - numbered figures which clearly identifies this.

Q5. In Figures 4 through 9 of Attachment C are the reactivity values given as part of the title the total rod worths or should they be reactivity insertion rates in pcm/sec? Please clarify.

Response: The reactivity values given are reactivity insertion rates in pcm/sec. Please see enclosed revised Figure titles.

Q6. Please provide a loading diagram for Cycle 8 along with a table of the nuclear parameters for comparison to those used in the safety evaluation.

Response: A loading diagram and supplementary information are given in the enclosed Figure 1 and Table 1. The final core loading plan drawing is currently being checked and will be sent to you when available

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	3425/2055/343	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	NA
Uncontrolled Boron Dilution	NA*	NA	0 and 3425	NA	NA	NA
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 6.3.2
(2) Minimum Doppler power coefficient (pcm/%power) = -10.18 + 0.035Q where Q is in % power.
(3) Maximum Doppler power coefficient (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 6.3.2
(2) Minimum Doppler power coefficient (pcm/%power) = -10.18 + 0.035Q where Q is in % power.
(3) Maximum Doppler power coefficient (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.

FIGURE 10

RCCA WITHDRAWAL AT POWER 100% POWER

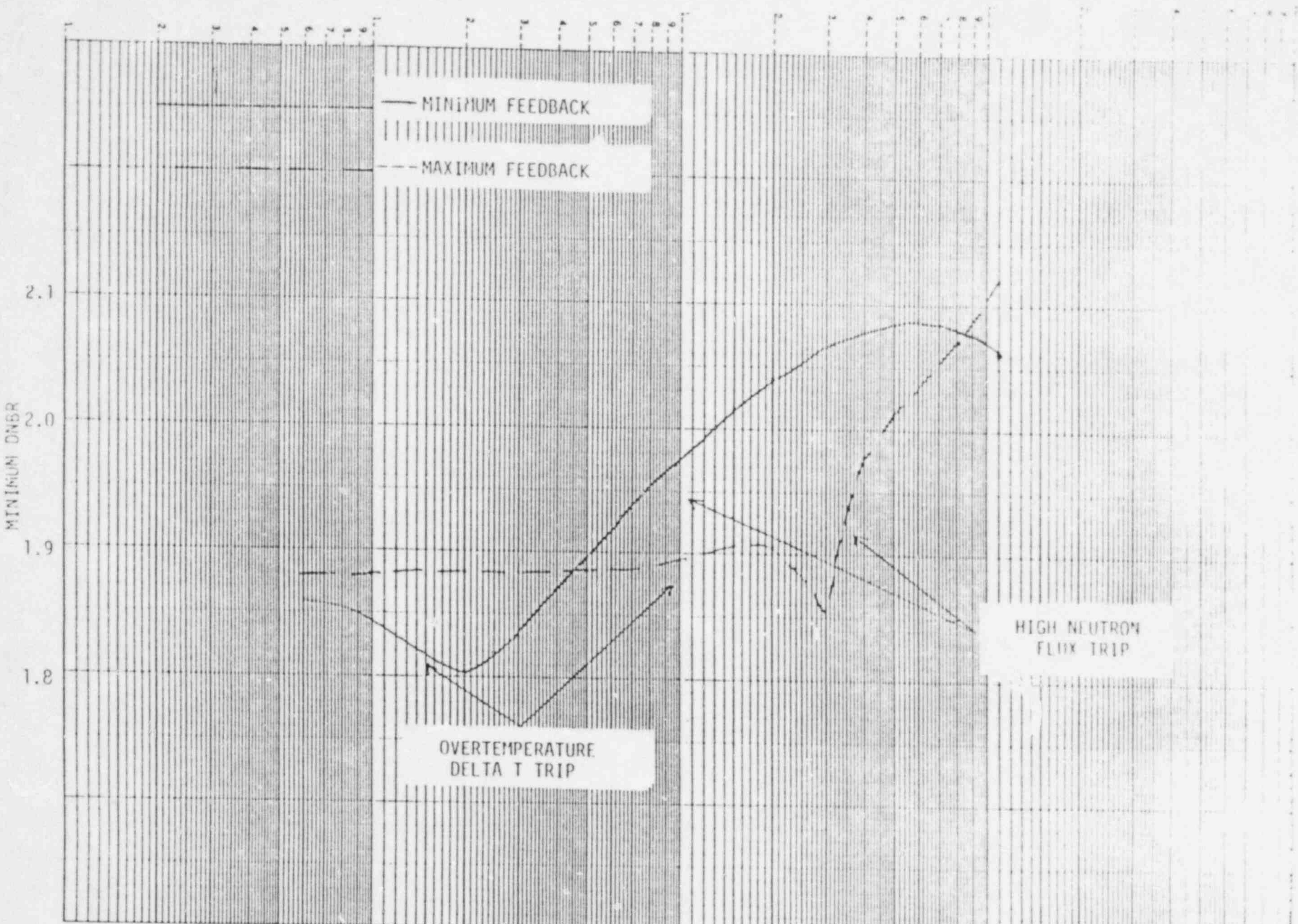


FIGURE 11

RCCA WITHDRAWAL AT POWER 60% POWER

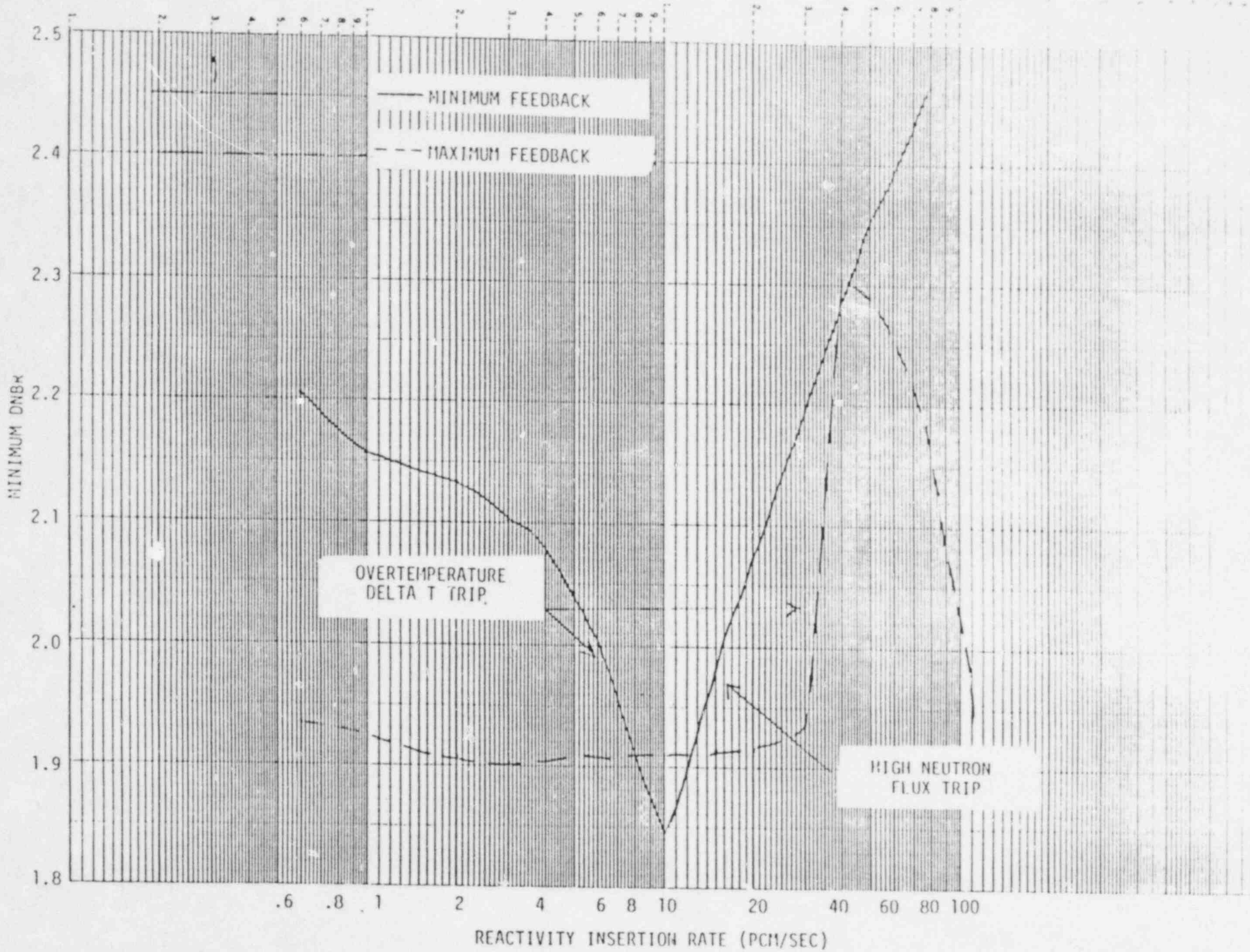
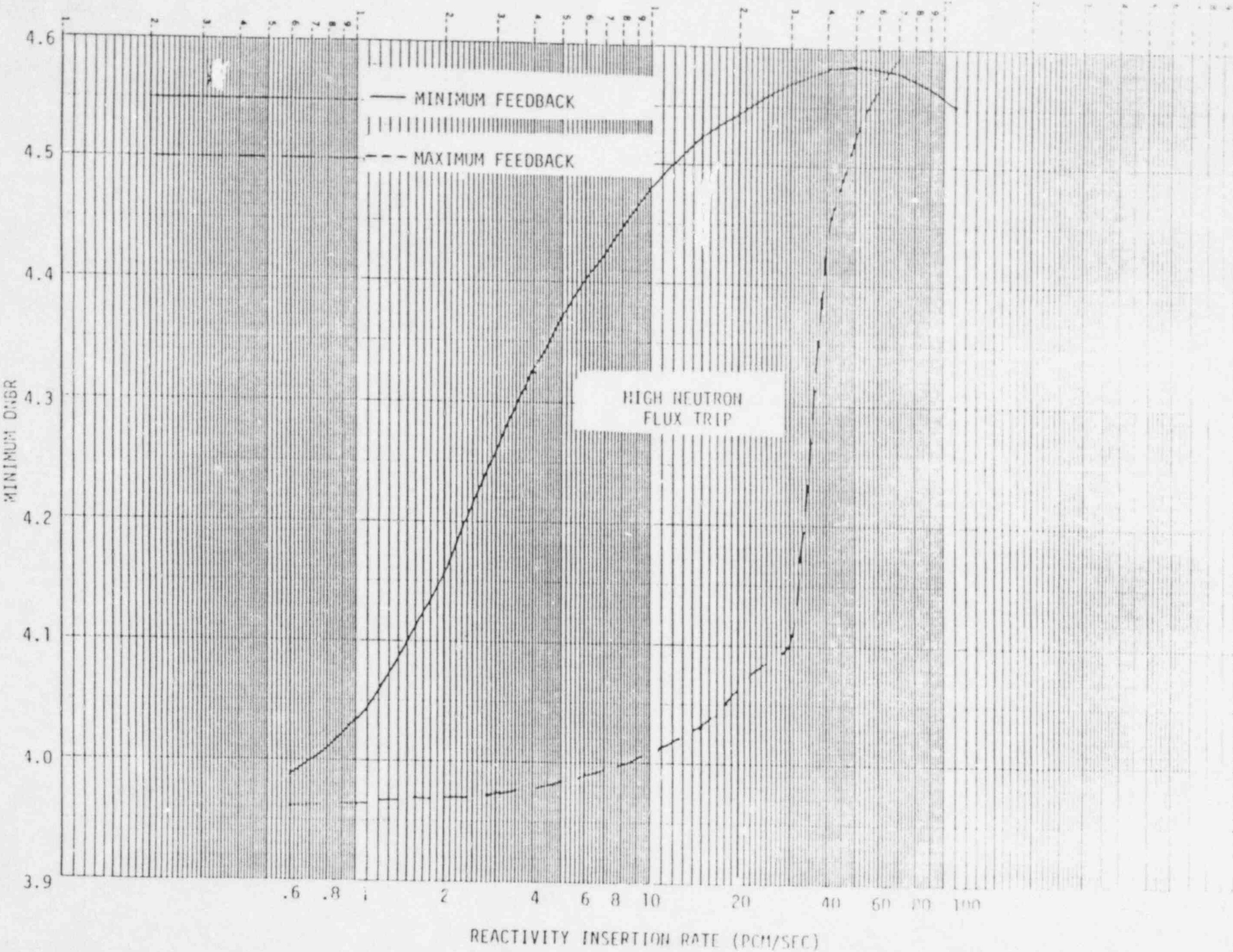


FIGURE 12

RCCA WITHDRAWAL AT POWER 10% POWER



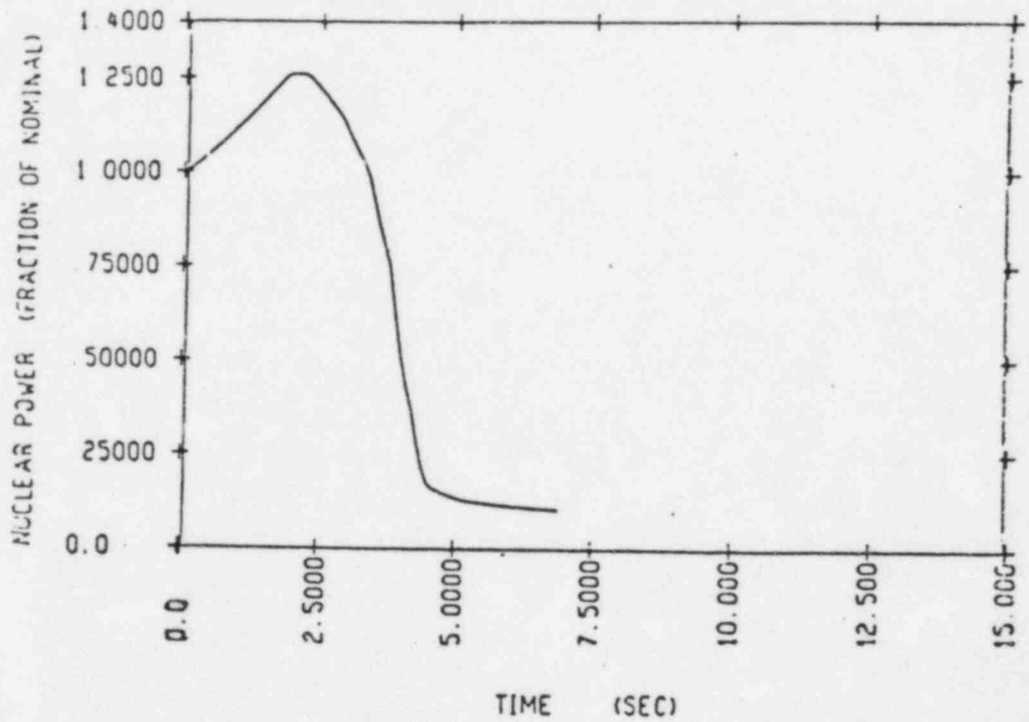


FIGURE 4

ROD WITHDRAWAL AT POWER
FULL POWER, 80 PCM/SEC INSERTION RATE,
MINIMUM REACTIVITY FEEDBACK
NUCLEAR POWER VERSUS TIME

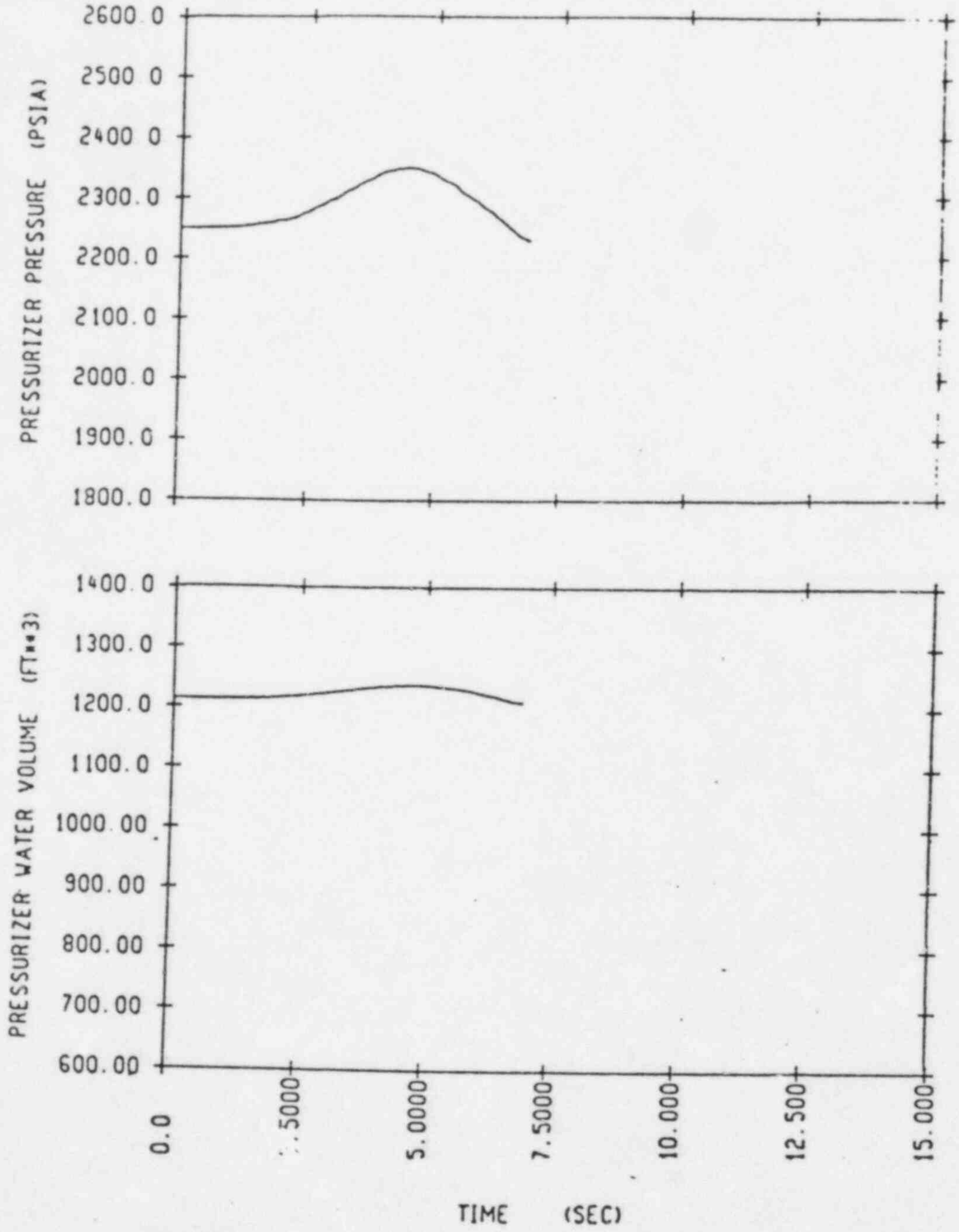


FIGURE 5

ROD WITHDRAWAL AT POWER

FULL POWER, 80 PCM/SEC INSERTION RATE, MINIMUM REACTIVITY FEEDBACK

PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME

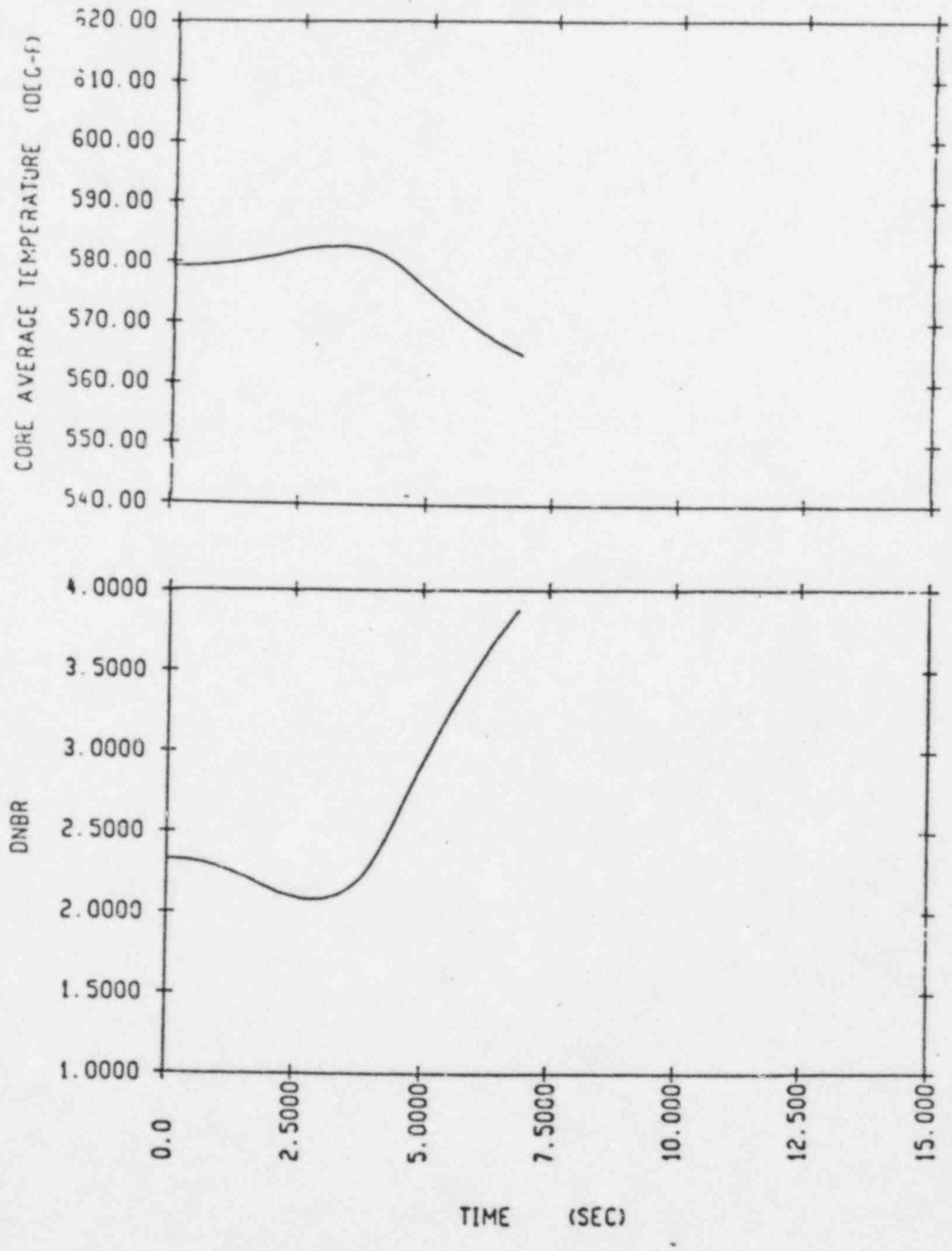


FIGURE 6

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

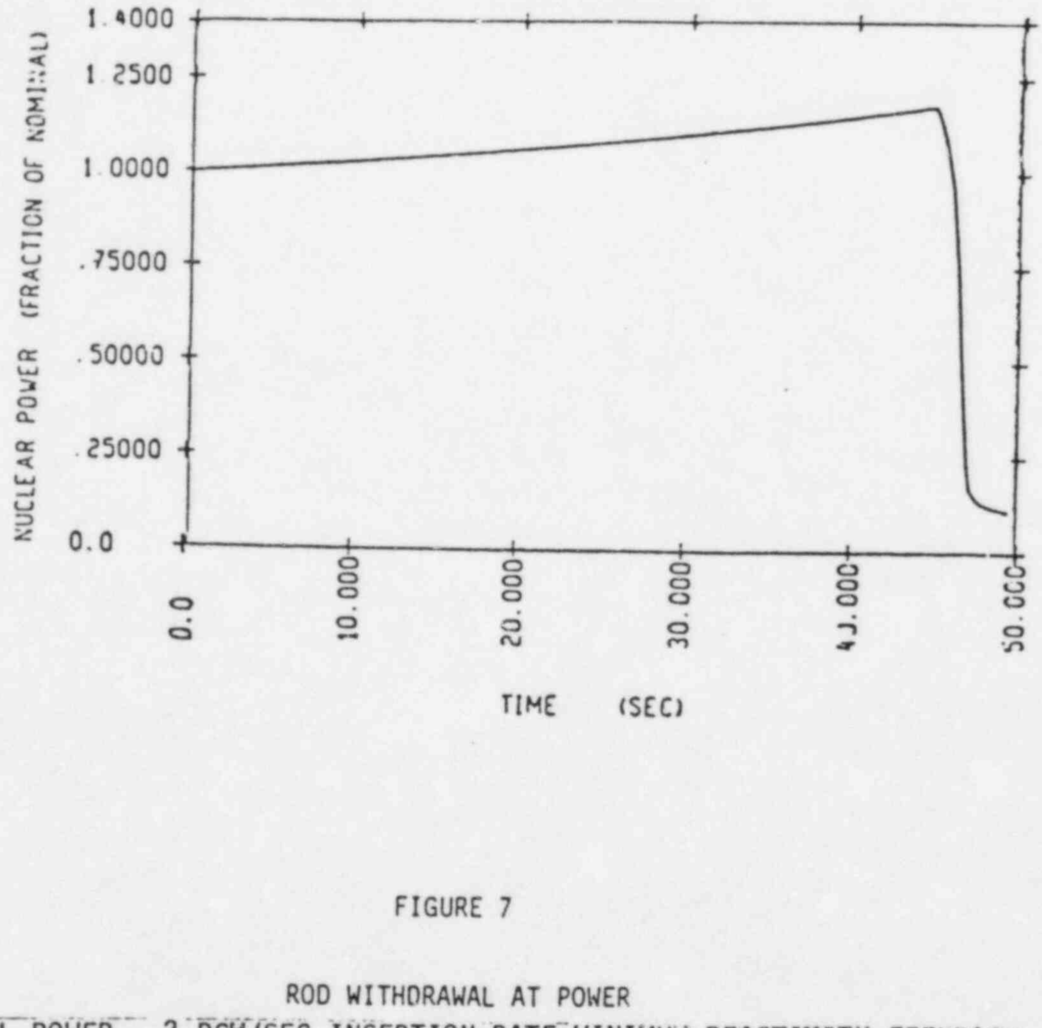


FIGURE 7

ROD WITHDRAWAL AT POWER

FULL POWER, 2 PCM/SEC INSERTION RATE, MINIMUM REACTIVITY FEEDBACK

NUCLEAR POWER VERSUS TIME

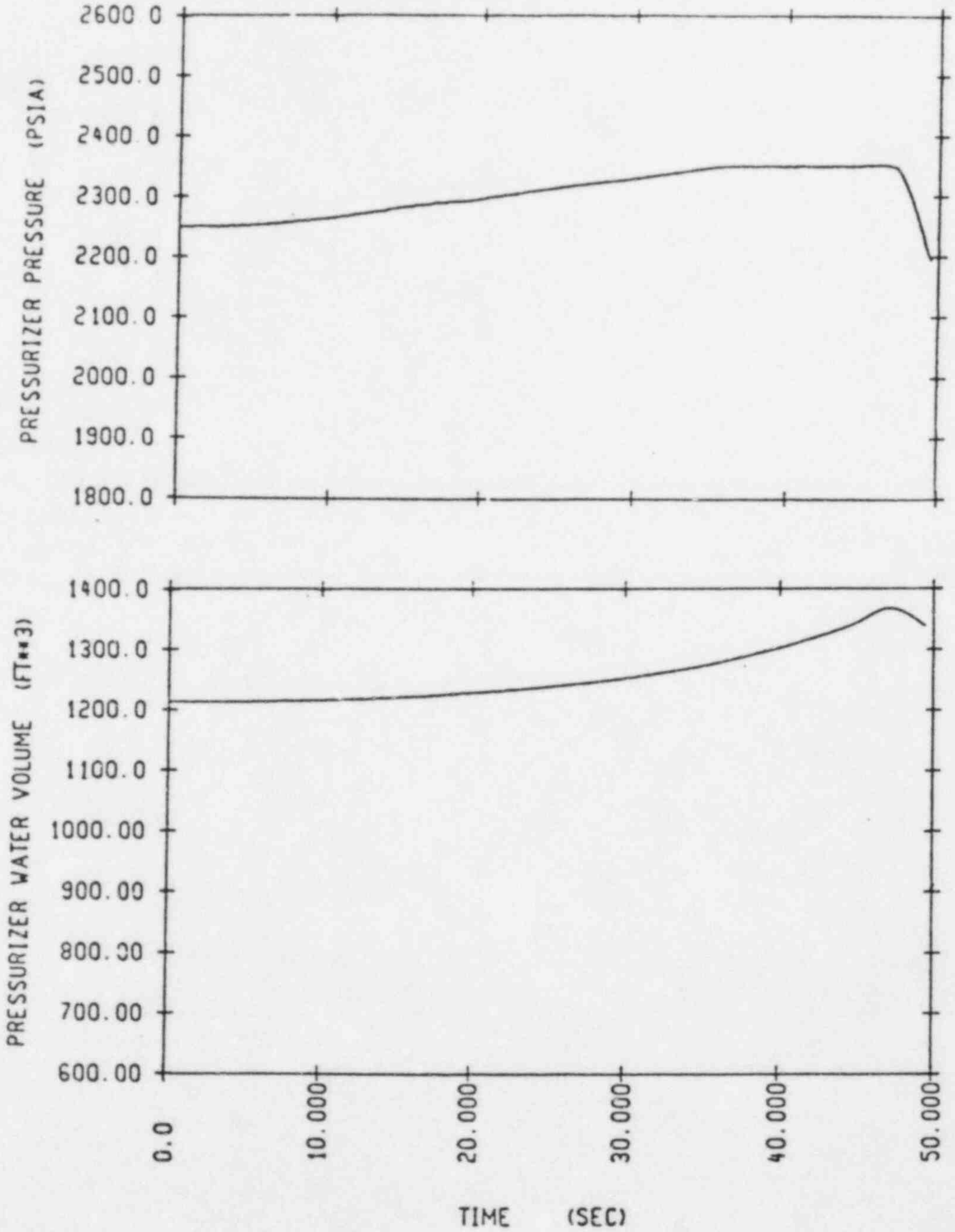


FIGURE 6

ROD WITHDRAWAL AT POWER

FULL POWER, 2 PCM/SEC INSERTION RATE, MINIMUM REACTIVITY FEEDBACK

PRESSURIZER PRESSURE AND WATER VOLUME VERSUS TIME

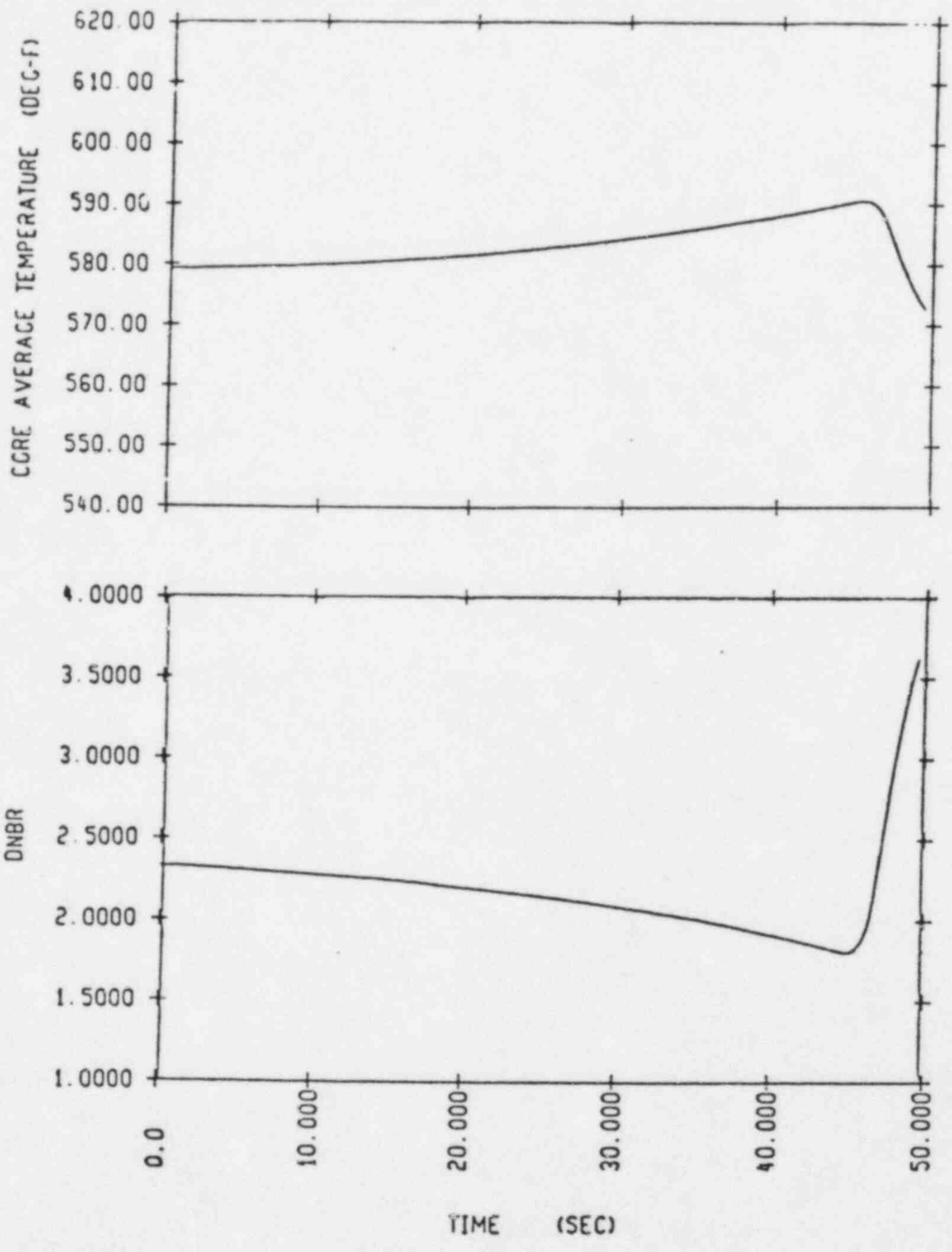


FIGURE 9

ROD WITHDRAWAL AT POWER

FULL POWER, 2 PCM/SEC INSERTION RATE, MINIMUM REACTIVITY FEEDBACK
CORE AVERAGE TEMPERATURE AND DNBR VERSUS TIME

FIGURE 1

CORE LOADING PATTERN
D. C. COOK UNIT 1, CYCLE 8

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
				8	10B SS*	10B 4	10B	10B 4	10B	8				
		8	9	10B 12	9 D	10A 12	8 SD	10A 12	9 D	10B 12	9	8		
	8	9 C	10B 12	9 A	10A 16	8 SD	10A 16	8 SD	10A 16	9 A	10B 12	9 C	8	
	9	10B 12	9 SD	9	9	10A 16	9 B	10A 16	9	9	9 SD	10B 12	9	
8	10B 12	9 A	9	8	10A 16	8	9	8 SD	10A 16	8	9	9 A	10B 12	8
10B	9 D	10A 16	9	10A 16	8 SD	10A 12	8 C	10A 12	8 SD	10A 16	9	10A 16	9 D	10B
10B 4	10A 12	8 SD	10A 16	8 SD	10A 12	9	9 SS	9	10A 12	8	10A 16	8 SD	10A 12	10B 4
10B	8 SD	10A 16	9 B	9	8 C	9	8 D	9	8 C	9	9 B	10A 16	8 SD	10B
10B 4	10A 12	8 SD	10A 16	8	10A 12	9	9	9	10A 12	8 SD	10A 16	8 SD	10A 12	10B 4
10B	9 D	10A 16	9	10A 16	8 SD	10A 12	8 C	10A 12	8 SD	10A 16	9	10A 16	9 D	10B
8	10B 12	9 A	9	8	10A 16	8 SD	9 SS	8	10A 16	8	9	9 A	10B 12	8
	9	10B 12	9 SD	9	9	10A 16	9 B	10A 16	9	9	9 SD	10B 12	9	
	8	9 C	10B 12	9 A	10A 16	8 SD	10A 16	8 SD	10A 16	9 A	10B 12	9 C	8	
		8	9	10B 12	9 D	10A 12	8 SD	10A 12	9 D	10B 12	9	8		
				8	10B	10B 4	10B	10B 4	10B SS*	8				

X YY	- Region Number - Number of Wet Annular Burnable Absorbers, or - Control Rod Locations (A, B, C, D, SD), or - Source Rods (unirradiated source, SS; irradiated source, SS*)

TABLE 1

FUEL ASSEMBLY DESIGN PARAMETERS
D. C. COOK UNIT 1 - CYCLE 8

Region	8	9	10A	10B
Fuel Type	ENC	ENC	W OFA	W OFA
Enrichment (w/o of U 235)*	2.905	2.903	3.30	3.60
Density (percent theoretical)*	94.0	94.0	94.5	94.5
Number of Assemblies	49	64	44	36
Burnup at Beginning of Cycle 8 (MWD/MTU) **	20300	9500	0	0
Fuel Stack Height (inches, cold)	144	144	144	144

* All values are nominal except for as-built enrichment values for Regions 8 and 9.

**Assumes a Cycle 7 nominal core average burnup of 10,300 MWD/MTU.

Attachment F to AEP:NRC:0745F
Responses to Fuels Questions
in Letter, Varga to Dolan, June 29, 1983

Q1. Please provide the projected maximum assembly average burnup of Exxon and Westinghouse fuel for Cycle 8. What will be maximum discharge batch average burnup at the end of Cycle 8?

R1. The Cycle 8 projected maximum assembly average burnup of Exxon fuel is 36,800 MWD/MTU. For Westinghouse fuel the maximum assembly burnup is 19,600 MWD/MTU. The discharged Exxon Region 8 fuel will have an average burnup of 32,600 MWD/MTU at the end of Cycle 8.

Q2. The Cycle 8 and future cores will consist of Exxon and Westinghouse fuel assemblies. Please provide an analysis of the structural adequacy of the fuel assemblies during the design seismic event for the mixed cores.

R2. We plan to attempt to provide a response to this question to the Nuclear Regulatory Commission by August 15, 1983.

Attachment G to AEP:NRC:0745F
Peaking Factor Limit Report

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Fuel Division

Box 3912
Pittsburgh Pennsylvania 15230

May 2, 1983

83AE*-G-035

W-AEP/0061

KEYWORDS: AEP
COOK-1
TECH-SPEC
CYCLE-8

Mr. Milton P. Alexich
Assistant Vice President
Nuclear Engineering
Indian and Michigan Electric Company
c/o American Electric Power Service Corporation
2 Broadway
New York, N.Y. 10004

5/13/83

MAY 12 1983

Attention: ~~Dr. J. J. Weiss~~ - Section Manager
Nuclear Materials and Fuel Management

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D.C. COOK UNIT 1
CYCLE 8 V(Z) FUNCTION REPLOT

Attached is a replot of the V(Z) function applicable to D.C. Cook Unit 1, Cycle 8. The function is replotted as a continuous function over the entire active core height and replaces the earlier version sent via our letter W-AEP/0049 dated March 24, 1983.

Numerical values are again provided on the figure to facilitate use on the plant computer.

Please call should you have any further comments on this figure.

Very truly yours,

R.T. Meyer

R. T. Meyer
Project Engineer
NFD Projects

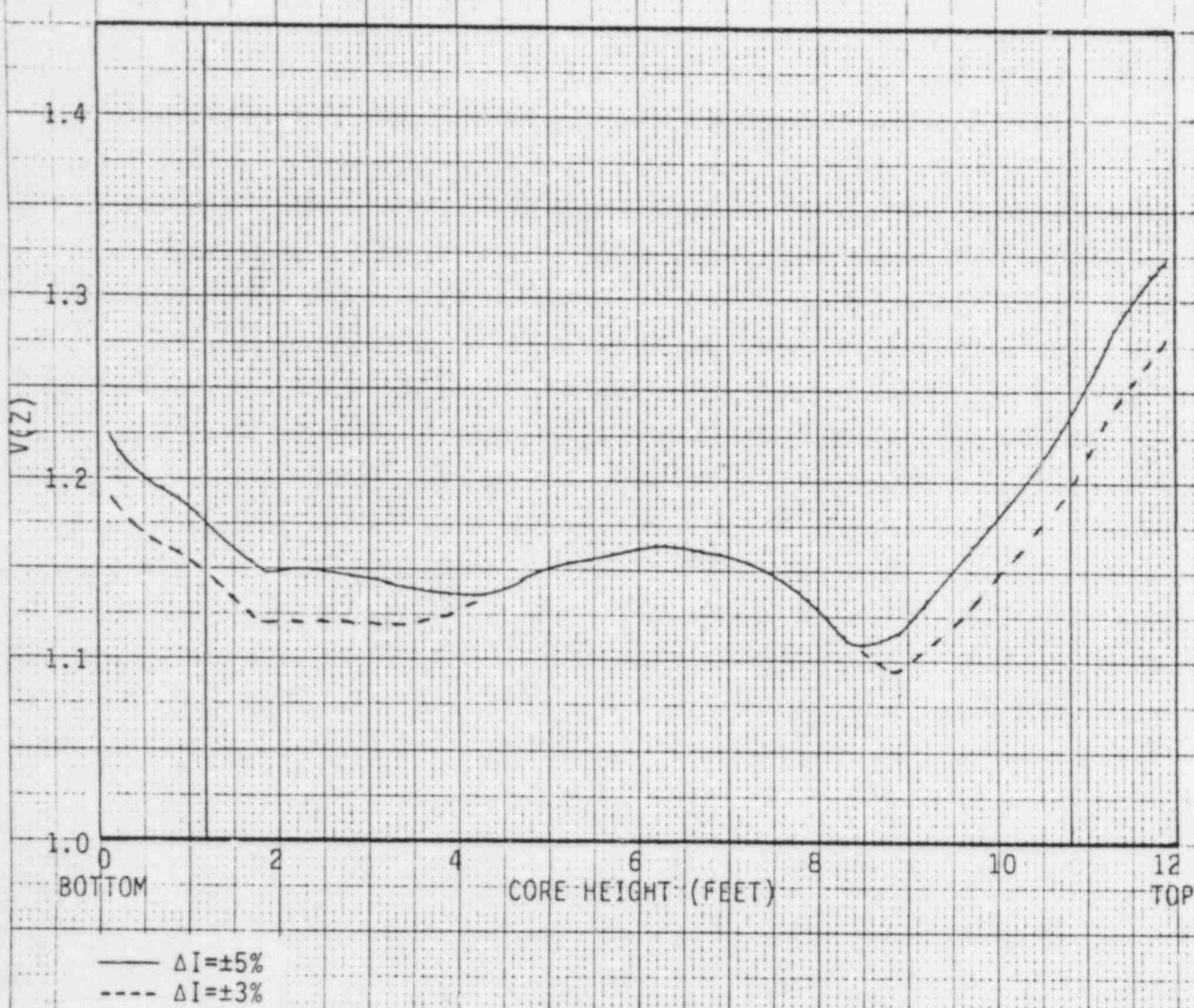
/mh

cc: H. L. Sobel
J. J. Weiss
J. Kern

Attachment

K+E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A. 46 1320

AEP CYCLE 8 - V(Z) FUNCTION



Core Height (Feet)	V(Z) ΔI=±5%	V(Z) ΔI=±3%
.13	1.222	1.190
.38	1.207	1.174
.88	1.190	1.158
1.38	1.166	1.137
1.88	1.147	1.120
2.38	1.149	1.121
2.88	1.145	1.120
3.38	1.141	1.119
3.88	1.137	1.122
4.38	1.137	1.137
4.88	1.149	1.149
5.38	1.155	1.155
5.88	1.160	1.160
6.38	1.163	1.163
6.88	1.159	1.159
7.38	1.152	1.152
7.88	1.134	1.134
8.38	1.108	1.108
8.88	1.115	1.095
9.38	1.143	1.114
9.88	1.173	1.141
10.38	1.204	1.167
10.63	1.220	1.183
10.88	1.241	1.201
11.38	1.290	1.246
11.63	1.309	1.261
11.88	1.321	1.279

Attachment H to AEP:NRC:0745F

Analysis Using the Standards in 10 CFR 50.92 about the Issue of No Significant Hazards Consideration for the License Amendment Request Addendum Contained in Letter No. AEP:NRC:0745F

Our analysis of the contents of the addendum to the license amendment request 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 addendum:

(1) does not involve a significant increase in the probability or consequences of an accident previously evaluated. As shown in Attachments B and C, to this letter, 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 use of standard calculational techniques approved by the NRC for cores reloaded by Westinghouse which have shown acceptable results.

(3) does not involve a significant reduction in a margin of safety. As shown in Attachments B and C to this letter the safety analyses performed in support of the license amendment addendum 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 addendum that would degrade the Plant's ability to safely control and mitigate any design basis accident.

One point is clarified further:

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 submitted a topical report (see Attachment B) on this matter and will support the NRC's generic review to obtain approval. In addition, the worst large break LOCA case has been reanalyzed using initial fuel rod conditions determined from the currently approved PAD fuel thermal safety model, in order to assure a timely licensing approval for Cycle 8 startup operations. We do not consider this point to involve any consideration of a significant safety hazard.