Consumers Power Co. 50-255 Palisades

Rpt: Loss-of-Coolant Accident Analysis in conformance with 10 CFR 50, App K....

(Rec'd w/ltr 7-9-75.............#7364)

Palisades Core I Reanalysis ECCS Performance Results

I. Introduction and Summary

On January 4, 1974, the Atomic Energy Commission issued New Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Reactors(1). The analysis presented herein demonstrates that the Palisades Core I ECCS design satisfies these new criteria.

This analysis has been performed using the approved CE large break evaluation model(3) including proposed modifications(8). For comparison, the worst break calculations were repeated without the proposed modifications. The results of this analysis, which are presented in Section II, cover primary system ruptures in the pump discharge leg larger than the pipe flow area, 4.909 ft^2. As demonstrated in CENPD-137(2), smaller breaks are not limiting. Therefore, a small break spectrum analysis has not been performed.

The total system flow rate was reduced from 130 x 10^6 lbs/hr to 124 x 10^6 lbs/hr for this Core I reanalysis. This coolant flow was selected as the minimum which would be reached based on the latest measurements and includes allowances for measurement error.

Using the proposed modifications to the CE large break evaluation model, peak clad temperature calculations were performed for the entire spectrum of break sizes at a peak linear heat generation rate (LHGR) of 11.0 kw/ft. Since at this LHGR the peak clad temperature for the worst break (1.0 x DES/PD*) was only 2135°F, this case was rerun at a peak LHGR of 11.3 kw/ft. This latter calculation produced a peak clad temperature of 2198°F, which is still below the criteria limit of 2200°F, thus demonstrating that operation at a peak LHGR of 11.3 kw/ft is acceptable.

Using the approved CE large break evaluation model <u>without</u> proposed modifications, the revised peak clad temperature results for the worst break (1.0 x DES/PD) did not alter the determination of the acceptable peak LHGR of 11.3 kw/ft.

---

*DES/PD = Double-Ended Slot at the Pump Discharge

The results of this study supercede those reported in References 9 and 10 and show that the plant meets the AEC Acceptance Criteria published in the Federal Register on January 4, 1974. Conformance is summarized as follows:

Criterion (1) Peak Clad Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200^{O}F".

Using the approved CE evaluation model with or without proposed modifications, the spectrum analysis yielded a peak clad temperature of 2198^{O}F for the 1.0 x DES/PD break at a peak linear heat generation rate of 11.3 kw/ft.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation".

With the proposed modifications, the spectrum analysis yielded a local peak clad oxidation percentage of 2.67% for the 0.6 x DEG/PD break.

Without the proposed modifications the 0.6 x DEG/PD break yielded a local peak clad oxidation percentage of 5.74%.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react".

With the proposed modifications, the 0.6 x DEG/PD break produced the highest core-wide oxidation which was <0.421%.

Without the proposed modifications, the 0.6 x DEG/PD break produced the highest core-wide oxidation which was <0.529%.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling".

The clad swelling and rupture model which is part of the CE Evaluation Model[3] accounts for the effects of changes in core geometry if such changes are predicted to occur. With these core geometry changes, core cooling was enough to lower temperatures. No further rupture can occur since the calculations were carried to the point at which the temperatures were decreasing. Thus, a coolable geometry has been maintained.

Criterion (5) Long Term Cooling. "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core".

The spectrum analysis presented in this report shows that the rapid insertion of borated water from the ECCS will suitably limit the peak clad temperature and cool the core within a short period of time. Subsequently, the safety injection pumps would supply cooling water from the refueling water tank to remove decay heat resulting from the long-lived radioactivity remaining in the core. When the refueling water tank is nearly empty, the safety injection pumps would then be lined up to recirculate water from the containment sump. In this manner, the core would be cooled for an indefinite period of time.

II. Large Break Analysis

A. Method of Calculation

The calculations reported in this section were performed using Combustion Engineering's large break evaluation model[3] with the following proposed modifications[8]:

1. The containment wall noding technique has been revised in order to provide a converged wall temperature solution.

2. Based on a recent review of steam-water mixing data, the resistance across the ECCS injection section during the period after the safety injection tanks have emptied has been revised.

In order to confirm that the model changes did not affect the peak LHGR, two additional calculations were performed; the break producing the peak clad temperature (1.0 x DES/PD) and the break with the highest local and core-wide oxidation percentages (0.6 x DEG/PD) were repeated using the approved model without the proposed modifications.

In this model the CEFLASH-4A[4] computer program is used to determine the primary system flow parameters during the blowdown phase and the COMPERC-II[5] computer program is used to describe the system behavior during the refill and reflood phases. The core flow and thermodynamic parameters from these two codes are used as input to the STRIKIN-II program[6] which calculates the hot rod clad temperature transient. The peak clad temperature and peak local clad oxidation percentage are therefore obtained from the STRIKIN-II calculation. The core-wide clad oxidation percentage is obtained from the results of both the STRIKIN-II and the COMZIRC[5, Supplement 1] computer programs.

B. Emergency Core Cooling System Assumptions

The Emergency Core Cooling System consists of three high pressure pumps, two low pressure pumps and four safety injection tanks. Automatic operation of the pumps is actuated by either a low-low pressurizer pressure signal or a high containment pressure signal. Flow is initiated from the safety injection tanks when the cold leg pressure drops below 215 psia plus the elevation head. Parameters pertinent to the calculation of the LOCA are presented in Table II-1.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. It is assumed that two high pressure pumps are operable at the time of the accident. Furthermore, it is assumed that off-site power is lost and all pumps must await diesel startup before they can begin to deliver flow. (It is assumed, however, that off-site power is

available for the containment spray system). Also, it is assumed that all safety injection flow delivered to the broken cold leg is lost.

An analysis of the possible single failures that can occur within the ECCS has shown that the worst single failure for the large break results is the failure of one of the low pressure pumps to start[3]. Thus, only one low pressure pump is used in the current LOCA analysis for Palisades.

A review was made of the effects of a single failure or operator error that causes any manually-controlled, electrically-operated valve to move to a position that could adversely affect the ECCS. In conformance with Branch Technical Position EICSB 18[11], plant modifications and changes to the technical specifications have been initiated to protect against any loss of safety function caused by this type of failure.

The above assumptions lead to the conclusion that the following safety injection flows are available:

75% of the flow from two high pressure pumps (one of the three H.P. pumps is not energized on SIAS)

75% of the flow from one low pressure pump

Flow from three safety injection tanks

In the analyses reported in this section, no credit is taken for pump flow until the tanks are empty.

C. Core, System and Containment Parameters

The significant core and system parameters used in the large break calculations are presented in Table II-1. The peak linear heat rate was assumed to occur in the top of the core, the conservative location as identified in Section IV.A.4 of Ref. 3. A conservative beginning-of-life moderator temperature coefficient ($+.5 \times 10^{-4}$ $\Delta\rho/^{o}F$) was used for all cases.

The gap conductance at the hot spot, as determined by the FATES computer program[7], represents the minimum value for the remainder of the first cycle.

The study of peak clad temperature versus burnup presented in Ref. 3 shows that the peak clad temperature is maximized when the gap conductance is minimized.

Containment parameters as presented in Table II-2 are chosen to minimize containment pressure such that a conservative determination of core reflood rate is made. Pressure suppression equipment start-up times are selected at their minimum values corresponding to off-site power being available.

D. Break Spectrum

In general, all possible break locations are considered in a LOCA analysis. However, it was demonstrated in Reference 3 that ruptures in the cold leg pump discharge location produce the highest clad temperatures. This is due to the minimization of core flow and reflood rate for this break location. Since core flow is a function of the break size, the Palisades large break calculations have been performed for the cold leg pump discharge breaks for both guillotine and slot breaks over a range of break sizes. This range was selected in order to show that the peak clad temperature is maximized with respect to break size.

E. Results

Table II-3 presents a listing of the large break sizes analyzed in this study along with the figure number presenting the pertinent transient data for each break.

As noted in Table II-3 the results for each of the breaks analyzed are displayed graphically in Figures II.1 through II.10. For each of the breaks shown in Figures II.1 through II.8 the nine variables listed in Table II-4 are plotted as a function of time. For the break having the highest peak clad temperature (1.0 x Double-Ended Slot) the additional quantities listed in Table II-5 are also presented. For Figure II.9 (1.0 x DES/PD) only those quantities which were affected by the removal of the proposed modifications are presented. (Refer to Figure II.1 for typical behavior of the other variables). Only the local clad oxidation percentage is presented for the 0.6 x DEG/PD break using the approved model without the proposed modifications (Figure II.10-0). (Refer to Figure II.7 for typical behavior of the other variables).

Peak clad temperature calculations for the entire break spectrum are reported at a peak linear heat generation rate (LHGR) of 11.0 kw/ft. The calculation for the 1.0 x DES/PD was repeated at a peak LHGR of 11.3 kw/ft. The resulting peak clad temperature and local clad oxidation for this latter calculation are included in Figures II.1-H and II.1-0, respectively. This case was repeated using the approved model without the proposed modifications at a peak LHGR of 11.3 kw/ft., and the results are shown in Figure II.9.

Times of interest for the various breaks are shown in Table II-6 while Table II-7 summarizes peak clad temperatures and clad oxidation percentages. The STRIKIN-II calculations were run to 280 seconds with the proposed modifications and to 420 seconds without the modifications, in order to fully terminate the clad oxidation reaction.

The peak local clad oxidation results for a LHGR of 11.0 kw/ft, given in Table II-7, show that the 0.6 x DEG/PD break had the highest local percentage. To determine the maximum local clad oxidation percentage at the limiting LHGR of 11.3 kw/ft, the calculation for the 0.6 x DEG/PD break was repeated. A special figure (II.7-0) was inserted into the results for the 0.6 x DEG/PD break, showing the transient local clad oxidation percentage at the limiting LHGR. This case was also repeated without the proposed modifications at a peak LHGR of 11.3 kw/ft, and the results shown in Figure II.10-0.

It should be noted that the hot assembly region power in CEFLASH-4A was based on a peak LHGR of 11.91 kw/ft, thus conservatively allowing flexibility during the determination of the allowable peak LHGR based on the STRIKIN-II prediction of the hot rod thermal behavior. This procedure, however, leads to a conservative core-wide clad oxidation calculation since the CEFLASH-4A hot assembly fuel and clad temperatures are used to initialize COMZIRC at the beginning of the reflood. Thus, the actual values for core-wide clad oxidation would be less than those reported in Table II-7.

Figure II.11 shows peak clad temperature plotted versus break size and type. It is noticed that the worst break is the 1.0 x Double-Ended Slot break, which has a peak clad temperature of 2198^{o}F at 11.3 kw/ft.

Mass and energy release to the containment during blowdown is presented in Tables II-8 and II-9, for the cases with and without proposed modifications, respectively. Also shown in these tables is the steam expulsion data during reflood. The ECC water spillage and containment spray flow rates are presented graphically in Figure II.12.

III. Computer Code Version Identification

The following versions of the Combustion Engineering ECCS Evaluation Model computer codes were used for this analysis, including the proposed modifications:

CEFLASH-4A: Version No. 74329
STRIKIN-II: Version No. 75066
COMPERC-II: Version No. 75097
COMZIRC : Version No. 75055

For the calculations performed without the proposed modifications:

CEFLASH-4A: Version No. 74329
STRIKIN-II: Version No. 75066
COMPERC-II: Version No. 75055
COMZIRC : Version No. 75055

IV. References

1. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3 - Friday, January 4, 1974.

2. CENPD-137, "Calculation Methods for the CE Small Break LOCA Evaluation Model", Combustion Engineering Proprietary Report, August, 1974 (Proprietary).

3. CENPD-132, "Calculative Methods for the C-E Large Break LOCA Evaluation Model", August 1974. (Proprietary)

   CENPD-132, Supplement 1, "Updated Calculative Methods for the C-E Large Break LOCA Evaluation Model", December 1974 (Proprietary).

4. CENPD-133, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis", April 1974 (Proprietary).

   CENPD-133, Supplement 2, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modification)", December 1974 (Proprietary).

5. CENPD-134, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core", April 1974 (Proprietary).

   CENPD-134, Supplement 1, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modification)", December 1974 (Proprietary).

6. CENPD-135, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program, April 1974 (Proprietary).

   CENPD-135, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)", December 1974 (Proprietary).

7. CENPD-139, "CE Fuel Evaluation Model", July 1974 (Proprietary).

8. DP-606, Letter from F. M. Stern (CE) to Victor Stello, Jr. (NRC) dated April 14, 1975, re: Request for Modification of Approved C-E ECCS Model of November 27, 1974.

9. Letter from Consumers Power Company (R. B. Sewell) to the AEC (L. M. Muntzing) dated October 21, 1974, "Palisades Plants - Final Acceptance Criteria".

10. Letter from Consumers Power Company (R. B. Sewell) to the AEC (L. M. Muntzing) dated Dec. 16, 1974, Subject: Docket #50-255, License DPR-20, "Palisades Plant - ECCS Reanalysis".

11. Branch Technical Position EICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves".

Table II-1

Palisades Core I
General System Parameters

Quantity	Value	
Reactor Power Level (102% of Nominal)	2244	MWt
Average Linear Heat Rate (102% of Nominal)	4.73	kw/ft
Peak Linear Heat Rate	11.3	kw/ft
Gap Conductance at Peak Linear Heat Rate	403.3	BTU/hr-ft^2-^{o}F
Fuel Centerline Temperature at Peak Linear Heat Rate	3647	^{o}F
Fuel Average Temperature at Peak Linear Heat Rate	2617	^{o}F
Hot Rod Gas Pressure	190	psia
Moderator Temperature Coefficient at Initial Density	$+0.5 \times 10^{-4}$	$\Delta\rho/^o$F
System Flow Rate (Total)	124.0×10^6	lbs/hr
Core Flow Rate	120.3×10^6	lbs/hr
Initial System Pressure	1800	psia
Core Inlet Temperature	530	^{o}F
Core Outlet Temperature	576	^{o}F
Active Core Height	11.0	Ft.
Fuel Rod OD	0.4135	In.
Number of Cold Legs	4	
Number of Hot Legs	2	
Cold Leg Diameter	30	In.
Hot Leg Diameter	42	In.
Safety Injection Tank Pressure	215	psia
Safety Injection Tank Gas/Water Volume	908/1103	Ft^3

Table II-2

Palisades Core I

Containment Physical Parameters

Net Free Volume	1.64×10^6 ft^3
Initiation Time for:	
Spray Flow	30.0 sec.
Fan Coolers; 3 Fans	0.0 sec.
4^{th} Fan	27.5 sec.

Containment Initial Conditions:	
Temperature	90^o F.
Pressure	14.7 psia
Relative Humidity	50%

Containment Spray Water:	
Temperature	40^o F.
Flow Rate (Total, 2 pumps)	3250 gpm

Fan Air Cooler Capacity (per Cooler)

Vapor Temperature (^{o}F)	Capacity (BTU/sec.)
40	0
104	700
184	9530
244	22900
283	34600

Heat Sinks:

TABLE II-2 (Cont'd)

A. Heat Sink	Total Mass (Lbm)	Total Surface Area (Ft^2)	Net Surface Area (Ft^2)	Thickness (Inches)
1. Tanks and Piping	351,000	19,332	19,332	.453
2. Heating and Ventilation Ducts	80,000	20,072	20,072	.100
Reactor Crane	315,000	13,946	13,946	2.35
3. Internal Concrete Structures	11,500,000	55,845	9,401	33.0
4. Gratings	54,450	8,880	8,880	.154
Roof Trusses	91,600	12,116	12,116	.1875
5. Containment Dome		7,270	7,270	
Liner Plate	73,000			0.25
Concrete	338,000			36.0
6. Containment Dome Base		11,000	11,000	
Liner Plate	110,750			0.25
Concrete	12,750,000			92.75
7. Containment Wall		54,400	50,600	
Liner Plate	546,000			0.25
Concrete	29,775,000			42.0
8. Storage Pool Floor		821	821	.003
Liner Plate	6,200			0.1875
Concrete	470,000			45.8
9. Shielded Internal Walls		3,635	3,635	
Liner Plate	26,650			0.1875
Concrete	2,420,000			53.4
10. Containment Base Slab		8,229	8,229	
Liner Plate	83,000			0.25
Concrete	15,350,000			149.0
11. Biological Shield		2,340 (ID)	2,340 (ID)	
Steel Lining	38,300			0.406
Insulation	8,580			4.00
Concrete	3,720,000			90.0
12. Structural, Support Steel	574,000	26,320	26,320	5.40

B. Minimum for protective coatings with reference to the table on the previous page:

Item 1. Four safe inject, oper floor in dome inorganic zinc 3 M-DFT (mils, Dry Film Thickness) with one coat inorganic white (Titanium Dioxide) 1½ M-DFT. Steam piping insulation with aluminum cladding, no coating.

Item 2. H&V ductwork, no coating. Reactor crane, one coat inorganic zinc 3 M-DFT.

Item 3. Concrete floor, organic epoxy de-contamable coating, 15 M-DFT. Concrete walls six feet up from floor, coating 10 M-DFT.

Item 4. Grating, galvanized, no coating. Handrails, organic, alkyd coating, 3 M-DFT.

Item 5. Roof trusses, contain dome and liner* plate, inorganic zinc 3 M-DFT with 1½ M-DFT in-organic white.

*The liner plate at the floor and six feet above was finish coated over the inorganic zinc with six mils DFT of an epoxy organic coating.

All concrete, except floor and six foot wainscot, no coating.

Item 6. Same as Item 5.

Item 7. Same as Item 6.

Item 8. Storage pool stainless steel, no coating.

Item 9. Same as Item 6.

Item 10. Same as Item 5.

Item 11. No coating.

Item 12. Platform columns, supports, etc. alkyd organic primer and finish coat 3 mils DFT.

In Addition: Steam Generator Basis: organic epoxy six mils DFT.

Four air coolers, organic alkyd-epoxy
Four mils DFT
Four Hold-Up Tanks: no coating.

TABLE II-2 (Cont'd)

C. Thermal Conductivity (Maximum)

And

Volumetric Heat Capacity (Maximum)

Materials	Thermal Conductivity (Btu/hr-ft-°F)	Specific Heat (Btu/lbm-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Organic Protective Coatings	0.3	-	62
Inorganic Protective Coatings	2	-	62
Stainless Steel Liner Plate	11	0.12	59
Carbon Steel Liner Plate	28	0.12	59
Structural Concrete	0.9	0.23	33

D. Heat Transfer Coefficients

a. Containment atmosphere to sump: 500 BTU/hr-ft^2-^{0}F

b. Sump to base slab: 20 BTU/hr-ft^2-^{0}F

c. Containment structure to enclosure building atmosphere: 10.0 BTU/hr-ft^2-^{0}F

Table II-3

Palisades Core I
Large Break Spectrum

Approved CE Evaluation Model With Proposed Modifications

Break Size, Type and Location	Abbreviation	Figure
1.0 x Double-Ended Slot Break in Pump Discharge Leg	1.0 x DES/PD	II.1
0.8 x Double-Ended Slot Break in Pump Discharge Leg	0.8 x DES/PD	II.2
0.6 x Double-Ended Slot Break in Pump Discharge Leg	0.6 x DES/PD	II.3
0.5 x Double-Ended Slot Break in Pump Discharge Leg	0.5 x DES/PD	II.4
1.0 x Double-Ended Guillotine Break in Pump Discharge Leg	1.0 x DEG/PD	II.5
0.8 x Double-Ended Guillotine Break in Pump Discharge Leg	0.8 x DEG/PD	II.6
0.6 x Double-Ended Guillotine Break in Pump Discharge Leg	0.6 x DEG/PD	II.7
0.5 x Double-Ended Guillotine Break in Pump Discharge Leg	0.5 x DEG/PD	II.8

Approved CE Evaluation Model Without Proposed Modifications

1.0 x Double-Ended Slot Dreak in Pump Discharge Leg	1.0 x DES/PD	II.9
0.6 x Double-Ended Guillotine Break in Pump Discharge Leg	0.6 x DEG/PD	II.10

Table II-4

Variables Plotted as a Function of Time

for Each Large Break in the Spectrum

Variable	Figure Designation
Core Power	A
Pressure in Center Hot Assembly Node	B
Leak Flow	C
Hot Assembly Flow (below hot spot)	D.1
Hot Assembly Flow (above hot spot)	D.2
Hot Assembly Quality	E
Containment Pressure	F
Mass Added to Core During Reflood	G
Peak Clad Temperature	H

Table II-5

Additional Variables Plotted as a Function of Time for the Large Break Having the Highest Clad Temperature

Variables	Figure Designation
Mid Annulus Flow	I
Qualities Above and Below the Core	J
Core Pressure Drop	K
Safety Injection Tank Flow into Intact Discharge Legs	L
Water Level in Downcomer During Reflood	M
Gap Conductance	N
Local Clad Oxidation	O
Clad Temperature, Centerline Fuel Temperature, Average Fuel Temperature and Coolant Temperature for Hottest Node	P
Hot Spot Heat Transfer Coefficient	Q
Hot Spot Heat Transfer Coefficient During Reflood	R
Containment Temperature	S
Sump Temperature	T

Table II-6

Palisades Core I

Times of Interest for Each Large Break
(Seconds)

Approved CE Model With Proposed Modifications

Break	Hot Rod* Rupture	SI Tanks On	End of Bypass	End of Blowdown	Start of Reflood	SI Tanks Empty
1.0 x DES/PD	-	16.2	16.2	18.9	30.65	56.0
0.8 x DES/PD	-	16.5	16.5	19.2	30.95	56.4
0.6 x DES/PD	-	17.6	17.6	20.3	30.05	57.4
0.5 x DES/PD	-	18.7	18.7	21.5	33.28	58.5
1.0 x DEG/PD	-	16.3	16.3	19.0	30.76	56.1
0.8 x DEG/PD	-	16.7	16.7	19.5	31.28	56.5
0.6 x DEG/PD	-	17.9	17.9	20.6	32.33	57.5
0.5 x DEG/PD	-	19.5	19.5	22.3	34.03	58.2

Approved CE Model Without Proposed Modifications

Break	Hot Rod* Rupture	SI Tanks On	End of Bypass	End of Blowdown	Start of Reflood	SI Tanks Empty
1.0 x DES/PD	-	16.2	16.2	18.9	30.43	54.72
0.6 x DEG/PD	-	17.9	17.9	20.6	32.12	56.39

* Hot Rod Rupture is not Predicted to Occur

Table II-7

Palisades Core I
Peak Clad Temperatures and Oxidation Percentages
for the Break Spectrum

Approved CE Evaluation Model With Proposed Modifications

Break	Peak Clad Temperature (°F)	Clad Oxidation % Local	Clad Oxidation % Core-Wide
11.0 kw/ft			
1.0 x DES/PD	2135	2.14	<0.3761
0.8 x DES/PD	2124	1.76	<0.3099
0.6 x DES/PD	2111	1.69	<0.2906
0.5 x DES/PD	2044	1.69	<0.2501
1.0 x DEG/PD	2131	2.05	<0.3469
0.8 x DEG/PD	2118	2.01	<0.3426
[illegible] x DEG/PD	2083	2.42	<0.3867
0.5 x DEG/PD	1906	2.06	<0.1986
11.3 kw/ft			
1.0 x DES/PD	2198	2.35	<0.4116
0.6 x DEG/PD	2141	2.67	<0.4209

Approved CE Evaluation Model Without Proposed Modifications

Break	Peak Clad Temperature (°F)	Clad Oxidation % Local	Clad Oxidation % Core-Wide
11.3 kw/ft			
1.0 x DES/PD	2198	4.98	<0.5077
0.6 x DEG/PD	2141	5.74	<0.5286

TABLE 4.8

BLOWDOWN AND REFLOOD MASS AND ENERGY RELEASE DATA

1.0 DES/PD

APPROVED MODEL WITH PROPOSED MODIFICATIONS

TIME SEC	MASS FLOW LBm/SEC		ENERGY RELEASE BTU/SEC		Integral of MASS FLOW LBm		INTEGRAL OF ENERGY RELEASE BTU	
0.0	0.0		0.0		0.0		0.0	
0.05	6.3137	E+04	3.2654	E+07	3.2435	E+03	1.6792	E+06
0.10	6.9110		3.5857		6.5633		3.4002	
0.15	7.0144		3.6350		1.0066	E+04	5.2162	
0.20	6.6976		3.4696		1.3506		6.9989	
0.25	6.4822		3.3623		1.6775		8.6928	
0.35	6.1542		3.5089		2.3469		1.2170	E+07
0.45	6.4719		3.3649		3.0025		1.5576	
0.60	6.4662		3.3627		3.9847		2.0686	
0.80	6.3855		3.3238		5.2620		2.7330	
1.0	6.2853		3.2738		6.5268		3.3915	
1.4	6.2179		3.2451		9.0312		4.6972	
1.8	5.8704		3.0691		1.1469	E+05	5.9707	
2.2	4.9139		2.5664		1.3621		7.0955	
2.6	4.4042		2.3033		1.5472		8.0627	
3.0	4.1068		2.1495		1.7176		8.9542	
3.4	3.9093		2.0489		1.8776		9.7923	
3.8	3.7347		1.9605		2.0307		1.0595	E+08
4.4	3.4890		1.8390		2.2468		1.1731	
5.2	3.2666		1.7376		2.5162		1.3157	
6.0	3.0484		1.6398		2.7687		1.4058	
6.8	2.8567		1.5475		3.0048		1.5783	
7.6	2.6806		1.4630		3.2262		1.6986	
8.4	2.4673		1.3652		3.4323		1.8118	
9.2	2.2431		1.2610		3.6206		1.9169	
10.0	2.0495		1.1650		3.7922		2.0138	

TABLE II. 8 Continued

TIME SEC	MASS FLOW LBM/SEC		ENERGY RELEASE BTU/SEC		Integral of MASS FLOW LBM		Integral of ENERGY RELEASE BTU	
11.0	1.7864	E+04	1.0393	E+07	3.9844	E+05	2.1242	E+08
12.0	1.4617	↓	8.9831	E+06	4.1476	↓	2.2212	↓
13.0	9.5131	E+03	7.4932	↓	4.2690	↓	2.3031	↓
14.0	6.7981	↓	6.3137	↓	4.3483	↓	2.3721	↓
15.0	4.9876	↓	5.2183	↓	4.4071	↓	2.4298	↓
16.0	3.4950	↓	3.7987	↓	4.4494	↓	2.4748	↓
17.0	1.7564	↓	2.1858	↓	4.4755	↓	2.5051	↓
18.0	5.5621	E+02	6.4001	E+05	4.4879	↓	2.5205	↓
18.9	5.6209	↓	6.5653	↓	4.4931	↓	2.5262	↓

↑ Time of Annulus Downflow (18.9)

Start of Reflood (Values below are for steam only.) (30.6)

TIME SEC	MASS FLOW LBM/SEC		ENERGY RELEASE BTU/SEC		Integral of MASS FLOW LBM		Integral of ENERGY RELEASE BTU	
30.6	0.0		0.0		4.4931	E+05	2.5262	E+08
45.6	0.0		0.0		4.4931	↓	2.5262	↓
60.6	2.0308	E+02	2.6311	E+05	4.5030	↓	2.5333	↓
75.6	2.0607	↓	2.6699	↓	4.5347	↓	2.5725	↓
90.6	2.0069	↓	2.6001	↓	4.5696	↓	2.6120	↓
105.6	1.9464	↓	2.5218	↓	4.5934	↓	2.6504	↓
120.6	1.9172	↓	2.4839	↓	4.6223	↓	2.6879	↓
135.6	1.8758	↓	2.4303	↓	4.6508	↓	2.7248	↓
150.6	1.8588	↓	2.4083	↓	4.6789	↓	2.7612	↓
165.6	1.8511	↓	2.3983	↓	4.7068	↓	2.7974	↓
180.6	1.8416	↓	2.3860	↓	4.7346	↓	2.8334	↓
195.6	1.8503	↓	2.3973	↓	4.7622	↓	2.8692	↓
210.6	1.8508	↓	2.3980	↓	4.7898	↓	2.9049	↓
225.6	1.8419	↓	2.3864	↓	4.8174	↓	2.9407	↓
240.6	1.8476	↓	2.3938	↓	4.8450	↓	2.9765	↓

TABLE II.8 Continued

TIME SEC	MASS FLOW LBM/SEC		ENERGY RELEASE BTU/SEC		INTEGRAL OF MASS FLOW LBM		INTEGRAL OF ENERGY RELEASE BTU	
255.6	1.8325	E+02	2.3741	E+05	4.8727	E+05	3.0122	E+08
270.6	1.8461	↓	2.3918	↓	4.9003	↓	3.0481	↓
285.6	1.8540	↓	2.4021	↓	4.9280	↓	3.0839	↓
300.6	1.8396	↓	2.3833	↓	4.9557	↓	3.1198	↓
315.6	1.8592	↓	2.4088	↓	4.9834	↓	3.1557	↓
330.6	1.8426	↓	2.3873	↓	5.0111	↓	3.1916	↓
345.6	1.8443	↓	2.3895	↓	5.0389	↓	3.2276	↓
360.6	1.8539	↓	2.4019	↓	5.0667	↓	3.2637	↓
375.6	1.8327	↓	2.3745	↓	5.0945	↓	3.2997	↓
390.6	1.8540	↓	2.4020	↓	5.1222	↓	3.3356	↓
405.6	1.8649	↓	2.4162	↓	5.1500	↓	3.3716	↓
420.6	1.8528	↓	2.4005	↓	5.1778	↓	3.4076	↓
430.6	1.8533	↓	2.4011	↓	5.1964	↓	3.4316	↓

TABLE II.7

BLOWDOWN AND REFLOOD MASS AND ENERGY RELEASE DATA

1.0 DES/PD

APPROVED MODEL

TIME	MASS FLOW		ENERGY RELEASE		Integral of MASS FLOW		Integral of ENERGY Release	
SEC	LBM/SEC		BTU/SEC		LBM		BTU	
BLOWDOWN DATA the SAME as GIVEN IN TABLE II.8								
Start of Reflood (Values below are for steam only)								
30.4	0.		0.		4.4931	E+05	2.5262	E+08
45.4	0.		0.		4.4931	↓	2.5262	↓
60.4	1.7493	E+02	2.2664	E+05	4.5035	↓	2.5397	↓
75.4	1.5673	↓	2.0307	↓	4.5279	↓	2.5713	↓
90.4	1.5155	↓	1.9634	↓	4.5511	↓	2.6013	↓
105.4	1.4838	↓	1.9224	↓	4.5736	↓	2.6305	↓
120.4	1.4619	↓	1.8940	↓	4.5957	↓	2.6592	↓
135.4	1.4474	↓	1.8753	↓	4.6176	↓	2.6875	↓
150.4	1.4399	↓	1.8656	↓	4.6393	↓	2.7156	↓
165.4	1.4419	↓	1.8681	↓	4.6609	↓	2.7436	↓
180.4	1.4424	↓	1.8688	↓	4.6825	↓	2.7716	↓
195.4	1.4405	↓	1.8663	↓	4.7041	↓	2.7996	↓
210.4	1.4421	↓	1.8683	↓	4.7257	↓	2.8276	↓
225.4	1.4497	↓	1.8783	↓	4.7474	↓	2.8557	↓
240.4	1.4542	↓	1.8841	↓	4.7692	↓	2.8839	↓
255.4	1.4526	↓	1.8820	↓	4.7911	↓	2.9123	↓
270.4	1.4602	↓	1.8919	↓	4.8130	↓	2.9407	↓
285.4	1.4685	↓	1.9026	↓	4.8350	↓	2.9692	↓
300.4	1.4726	↓	1.9079	↓	4.8571	↓	2.9978	↓
315.4	1.4767	↓	1.9133	↓	4.8793	↓	3.0265	↓
330.4	1.4831	↓	1.9215	↓	4.9015	↓	3.0553	↓
345.4	1.4870	↓	1.9265	↓	4.9238	↓	3.0842	↓
360.4	1.4817	↓	1.9197	↓	4.9461	↓	3.1131	↓
375.4	1.4933	↓	1.9347	↓	4.9684	↓	3.1420	↓

TABLE II.9 Continued

TIME	MASS FLOW		ENERGY RELEASE		Integral of MASS FLOW		Integral of ENERGY RELEASE	
SEC	LBm/SEC		BTU/SEC		LBm		BTU	
390.4	1.4950	E+02	1.9369	E+05	4.9908	E+05	3.1710	E+08
405.4	1.4932	↓	1.9346	↓	5.0132	↓	3.2000	↓
420.4	1.4979	↓	1.9407	↓	5.0357	↓	3.2291	↓
430.4	1.4992	↓	1.9424	↓	5.0506	↓	3.2486	↓

FIGURE II.1-A

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CORE POWER

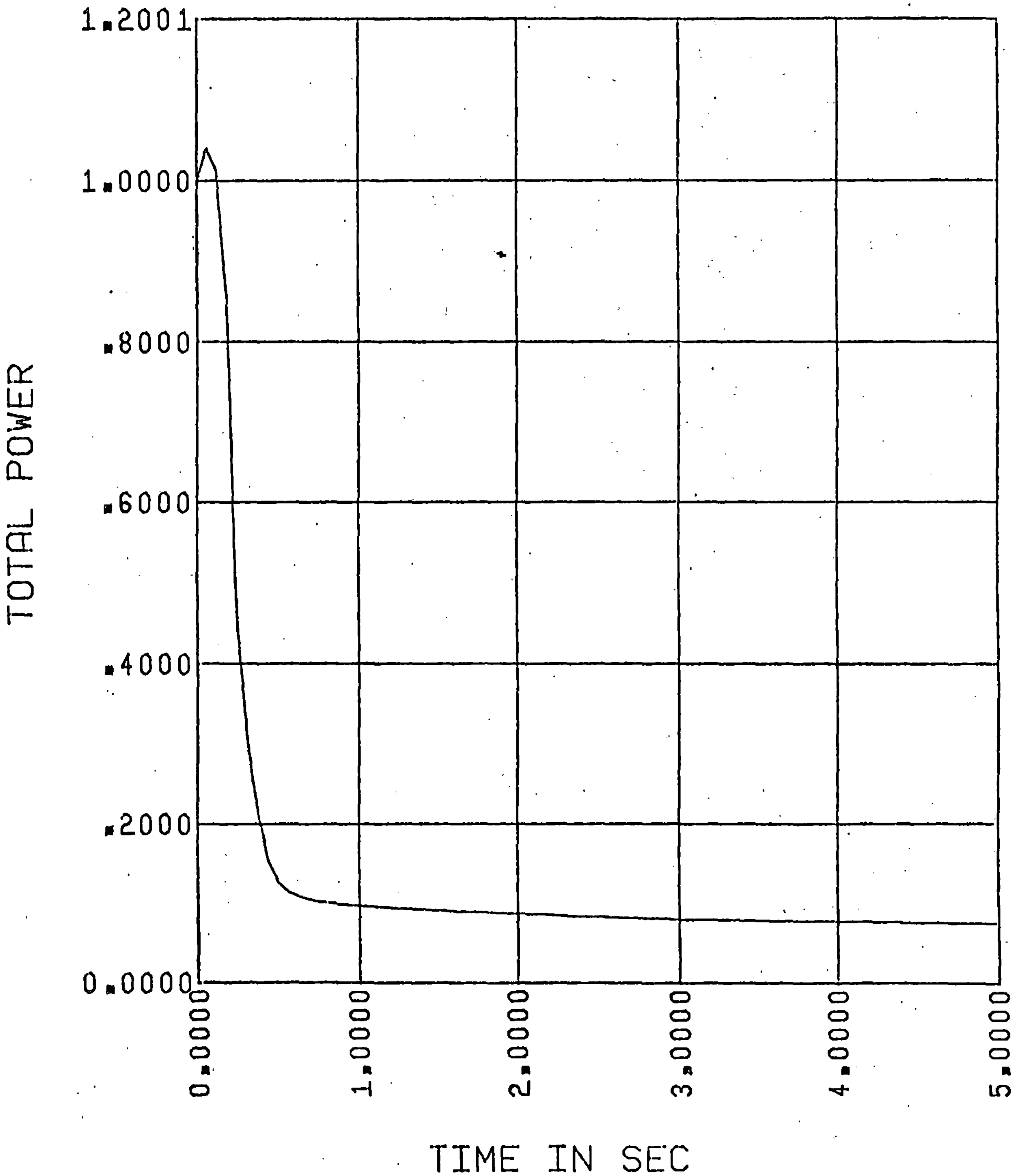


FIGURE II.1-B

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

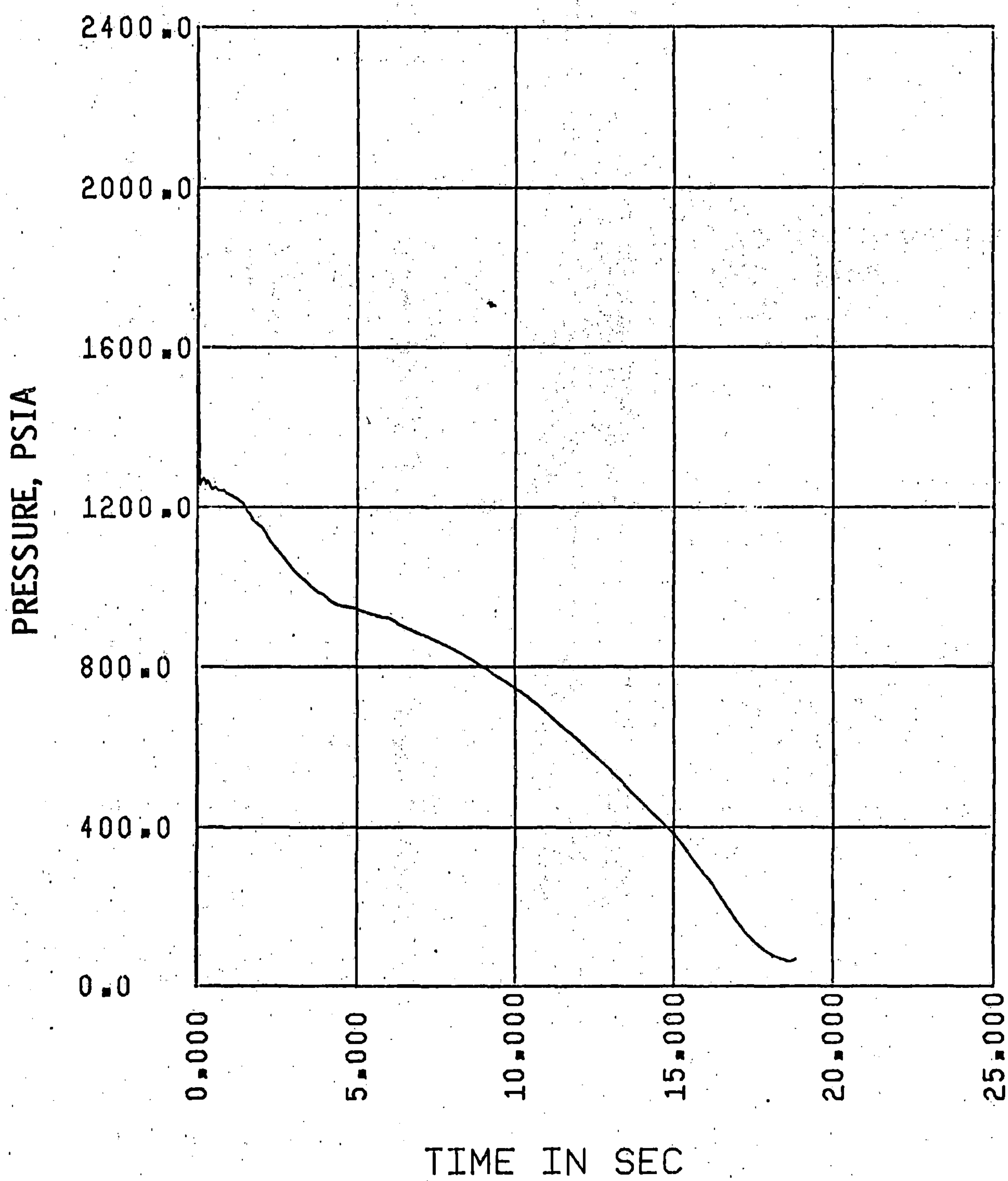


FIGURE II.1-C

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

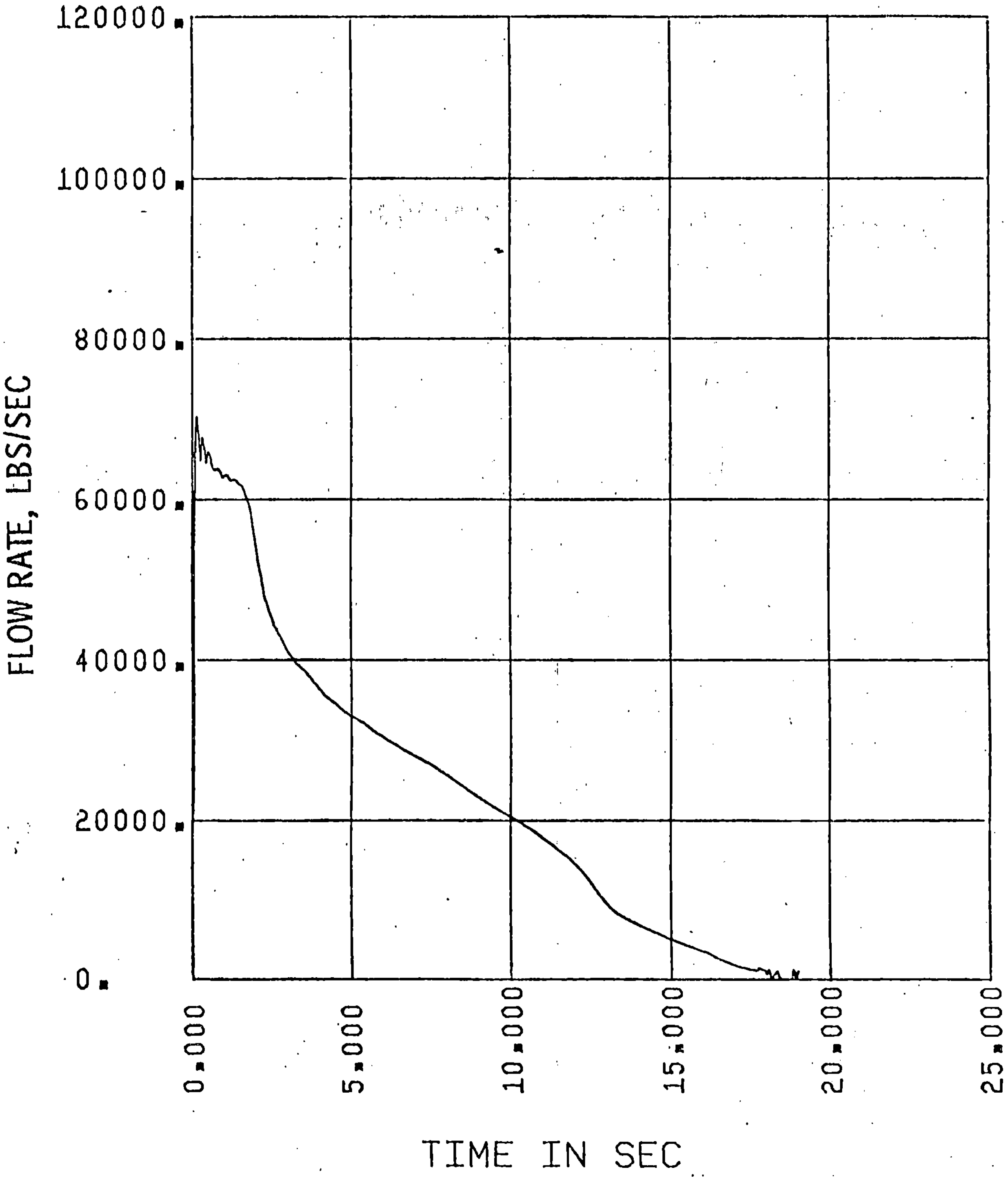


FIGURE II.1-D.1

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

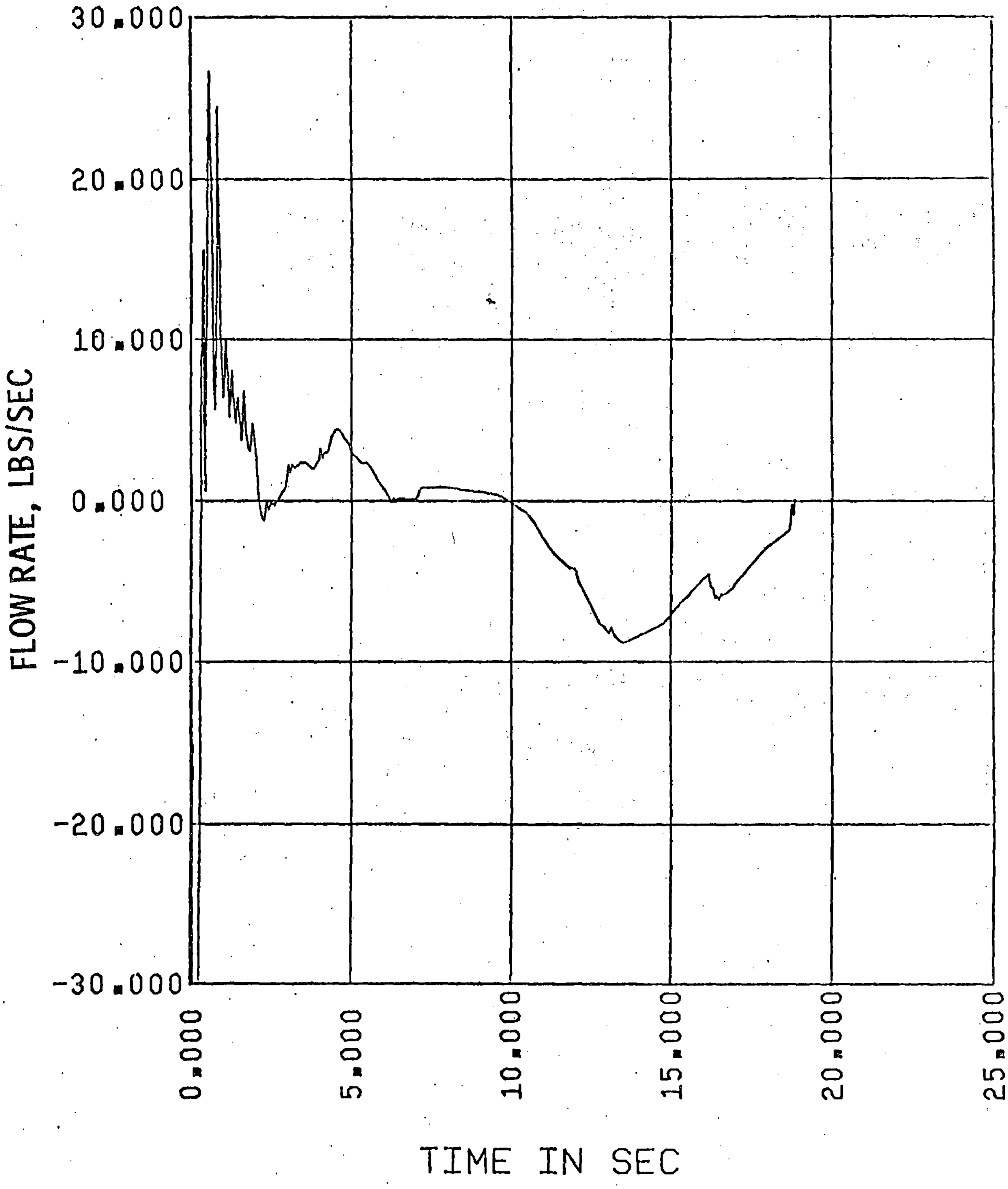


FIGURE II.1-D.2

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

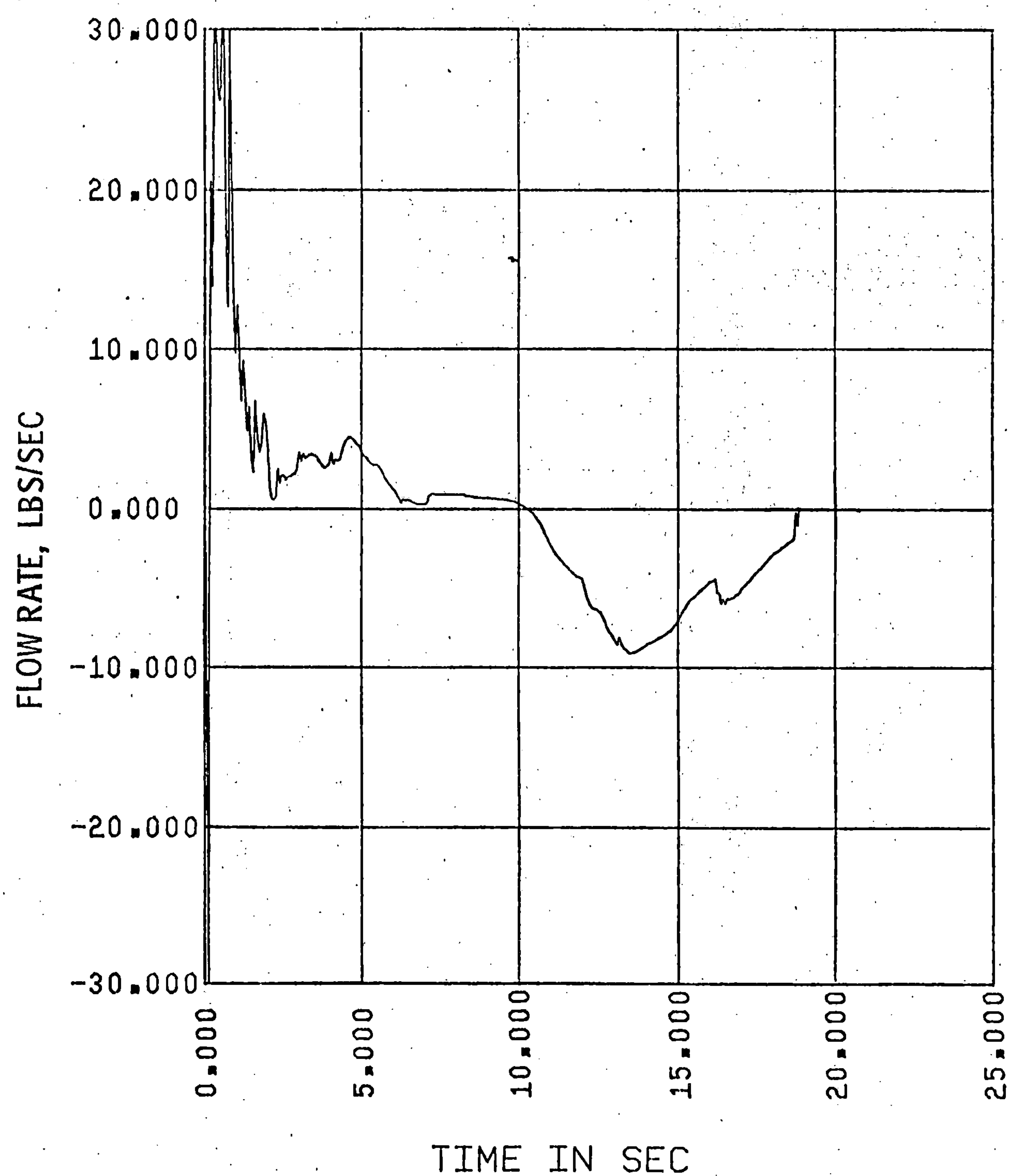


FIGURE II.1-E

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

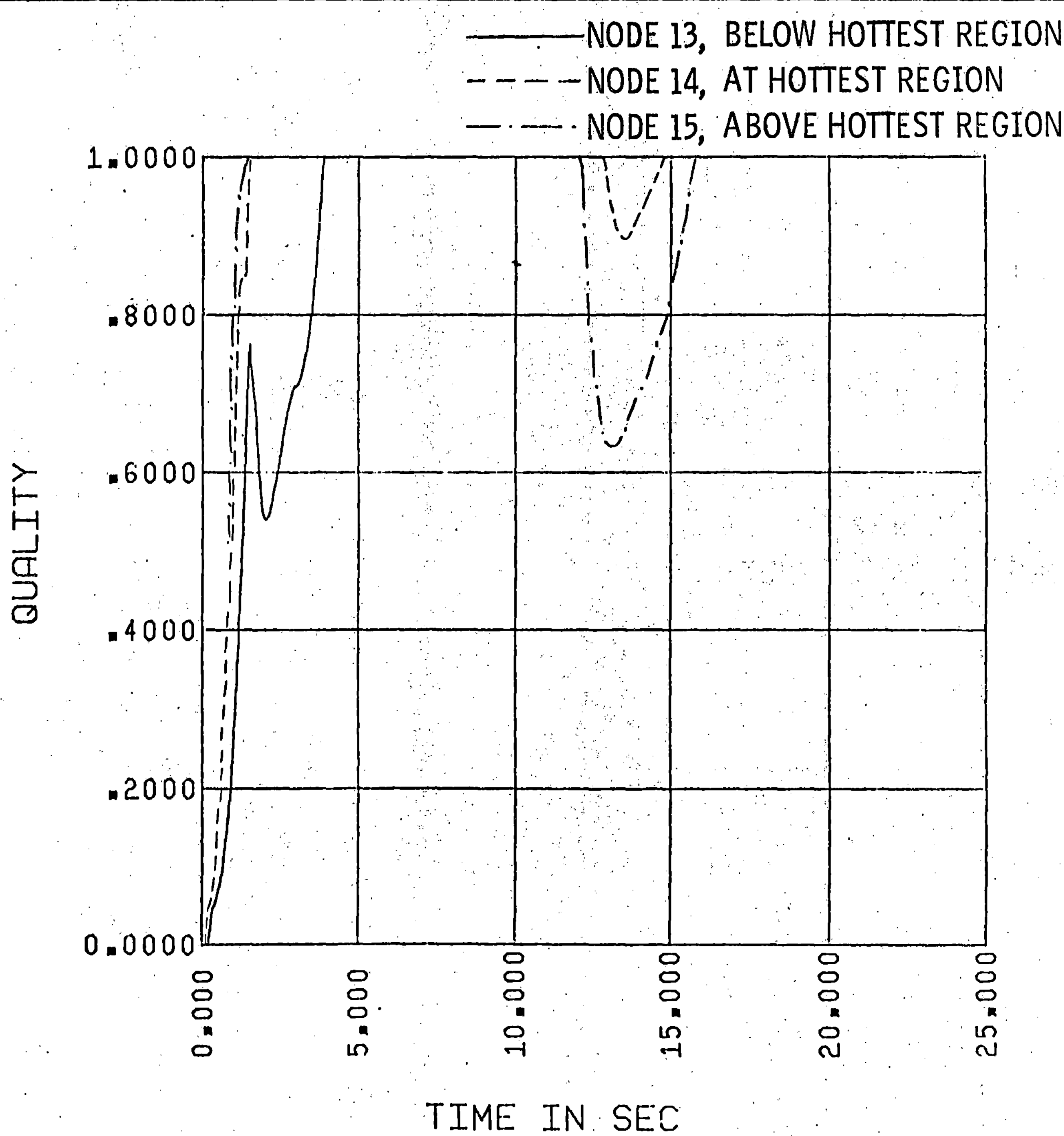


FIGURE II.1-F

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

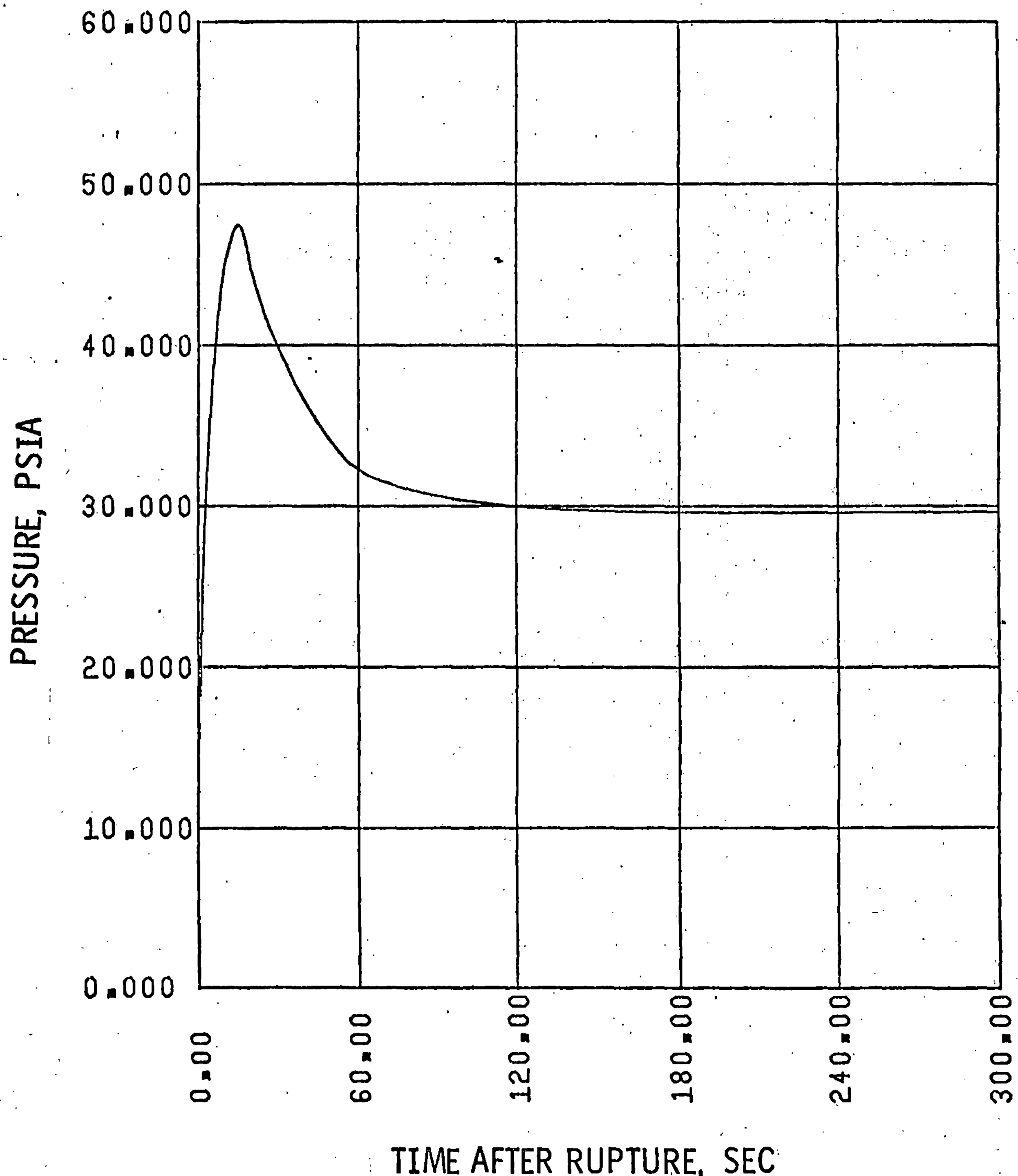


FIGURE II.1-G

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

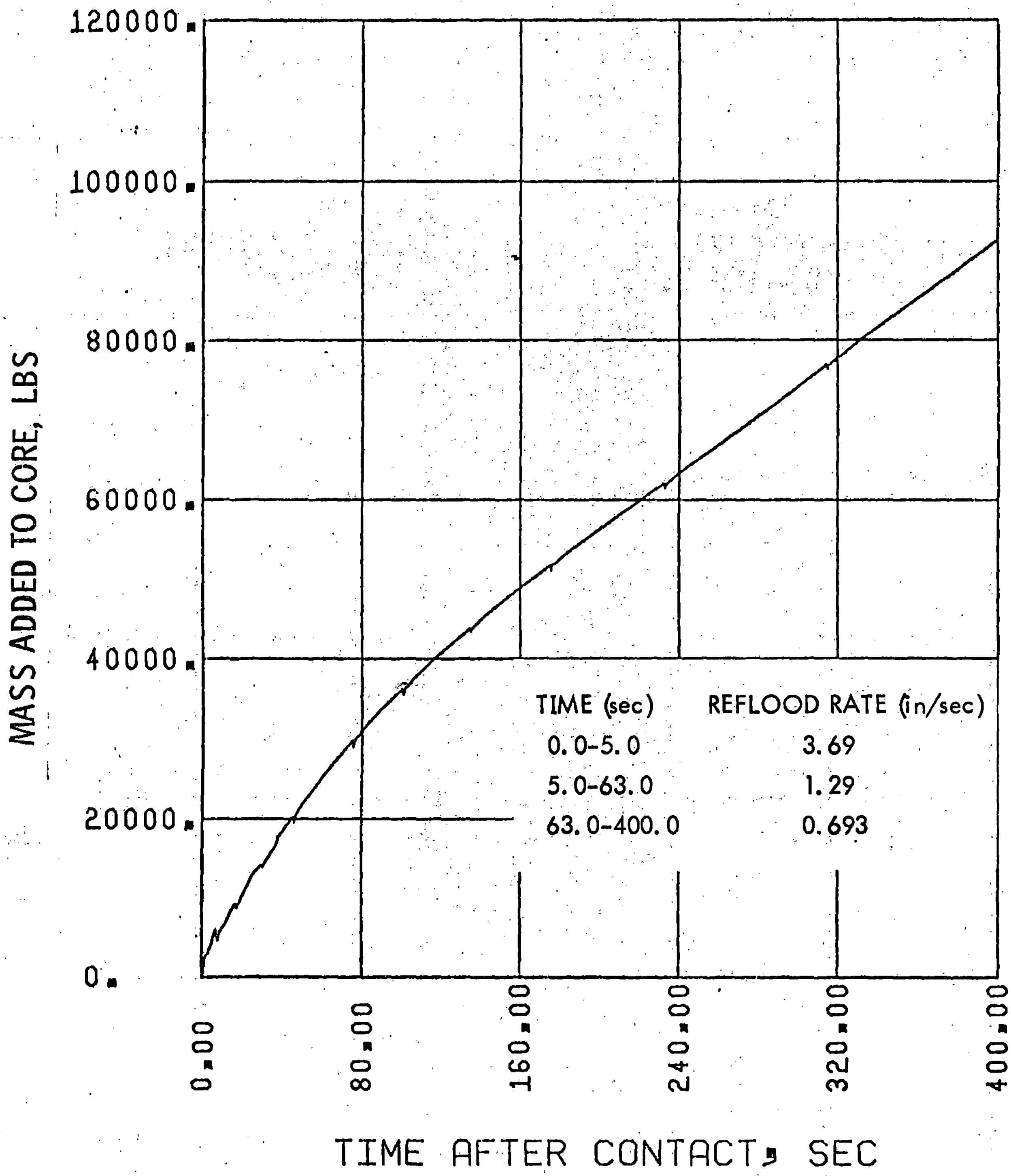


FIGURE II.1-H

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

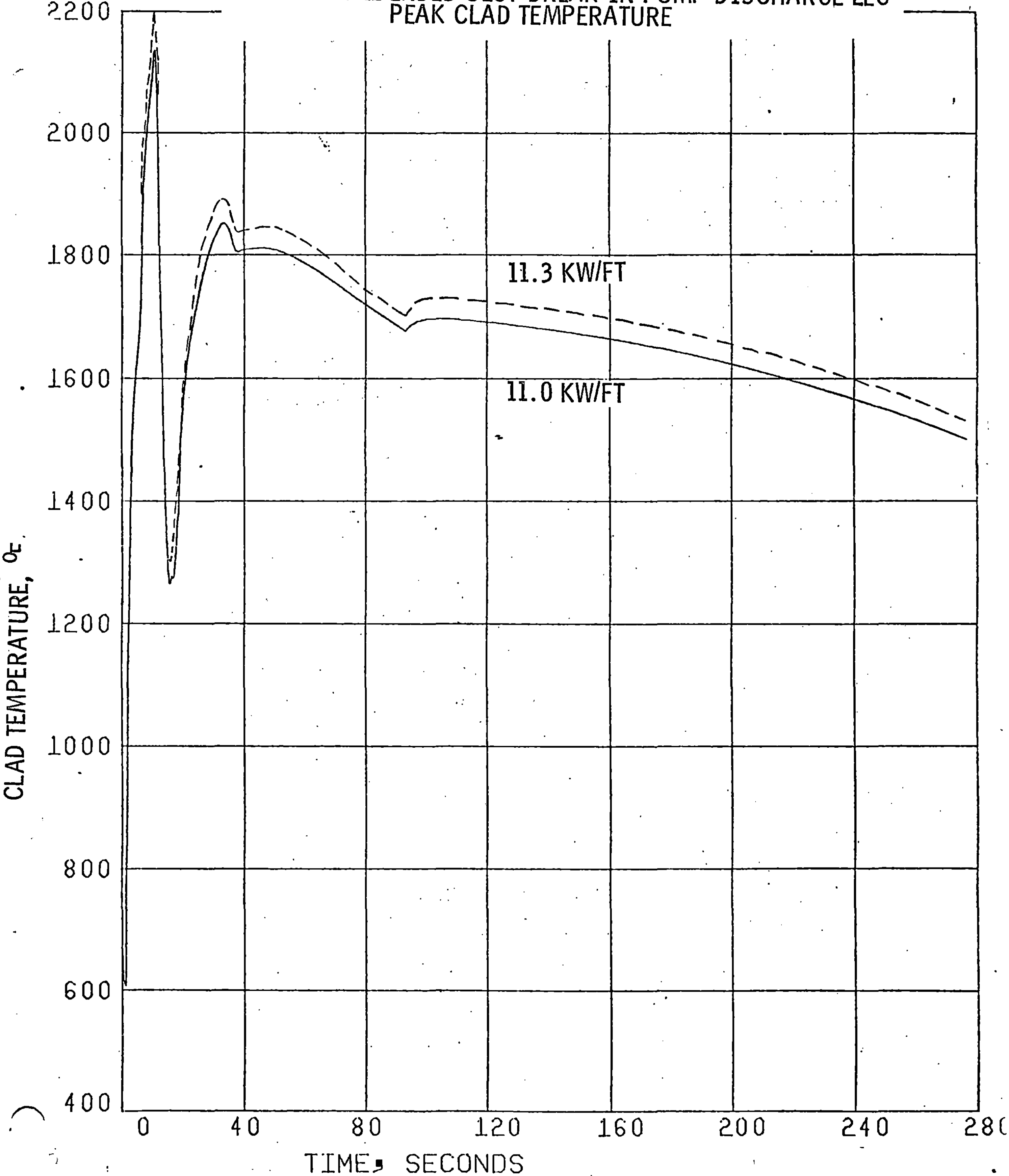


FIGURE II.1-I

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MID ANNULUS FLOW

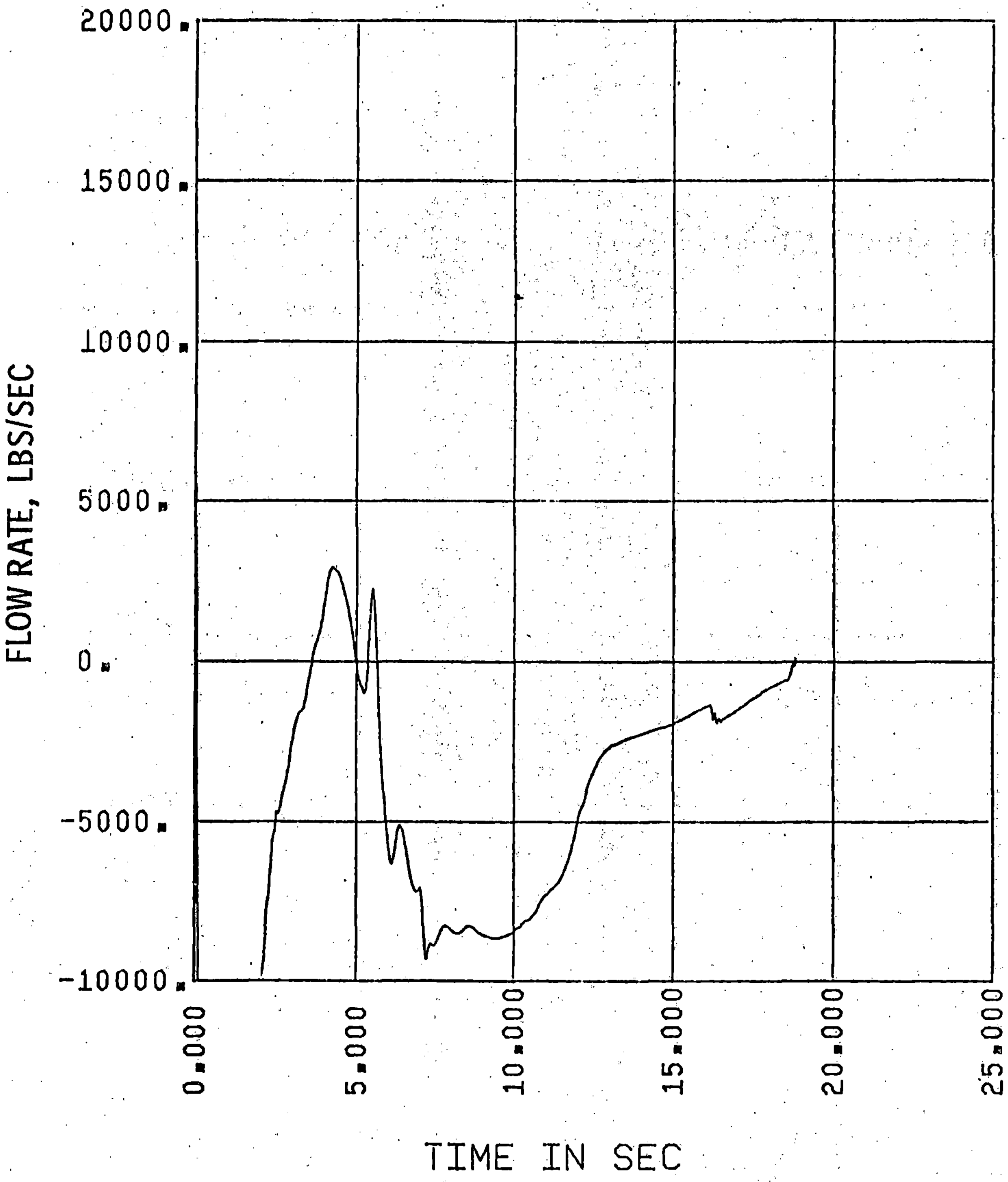


FIGURE II.1-J

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
QUALITIES ABOVE AND BELOW THE CORE

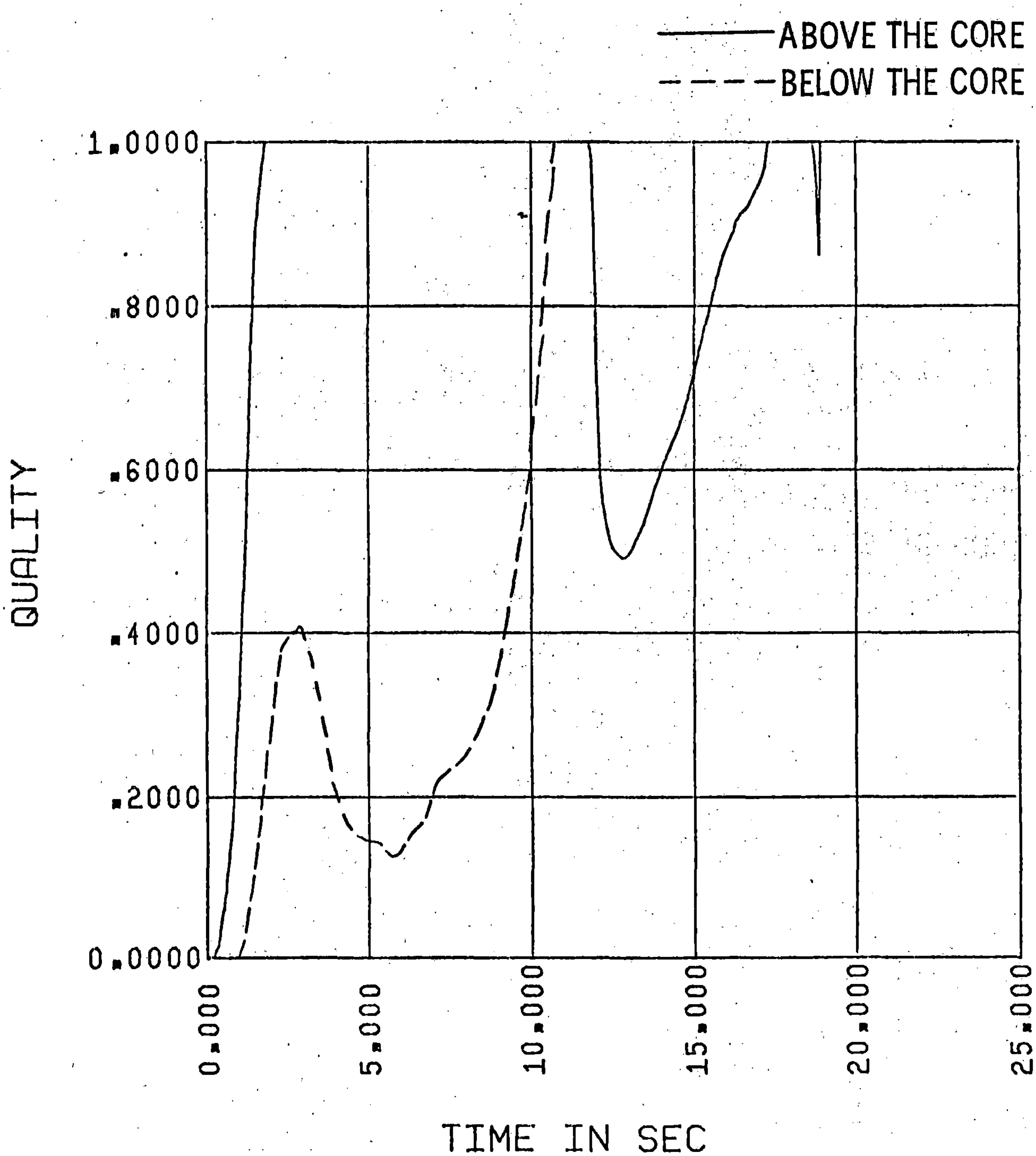


FIGURE II.1-K

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CORE PRESSURE DROP

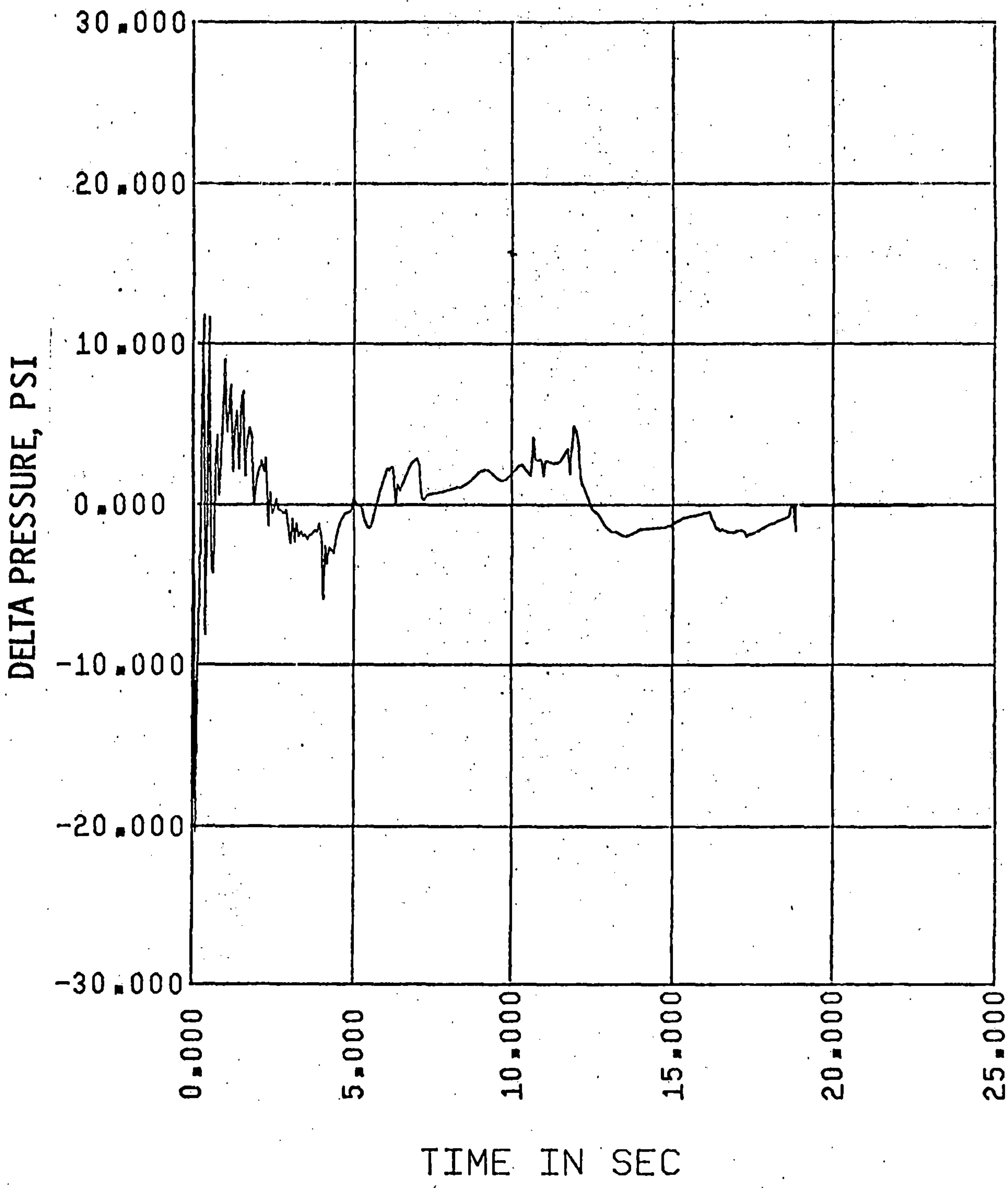


Figure II.1-L

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
SAFETY INJECTION TANK FLOW INTO INTACT DISCHARGE LEGS

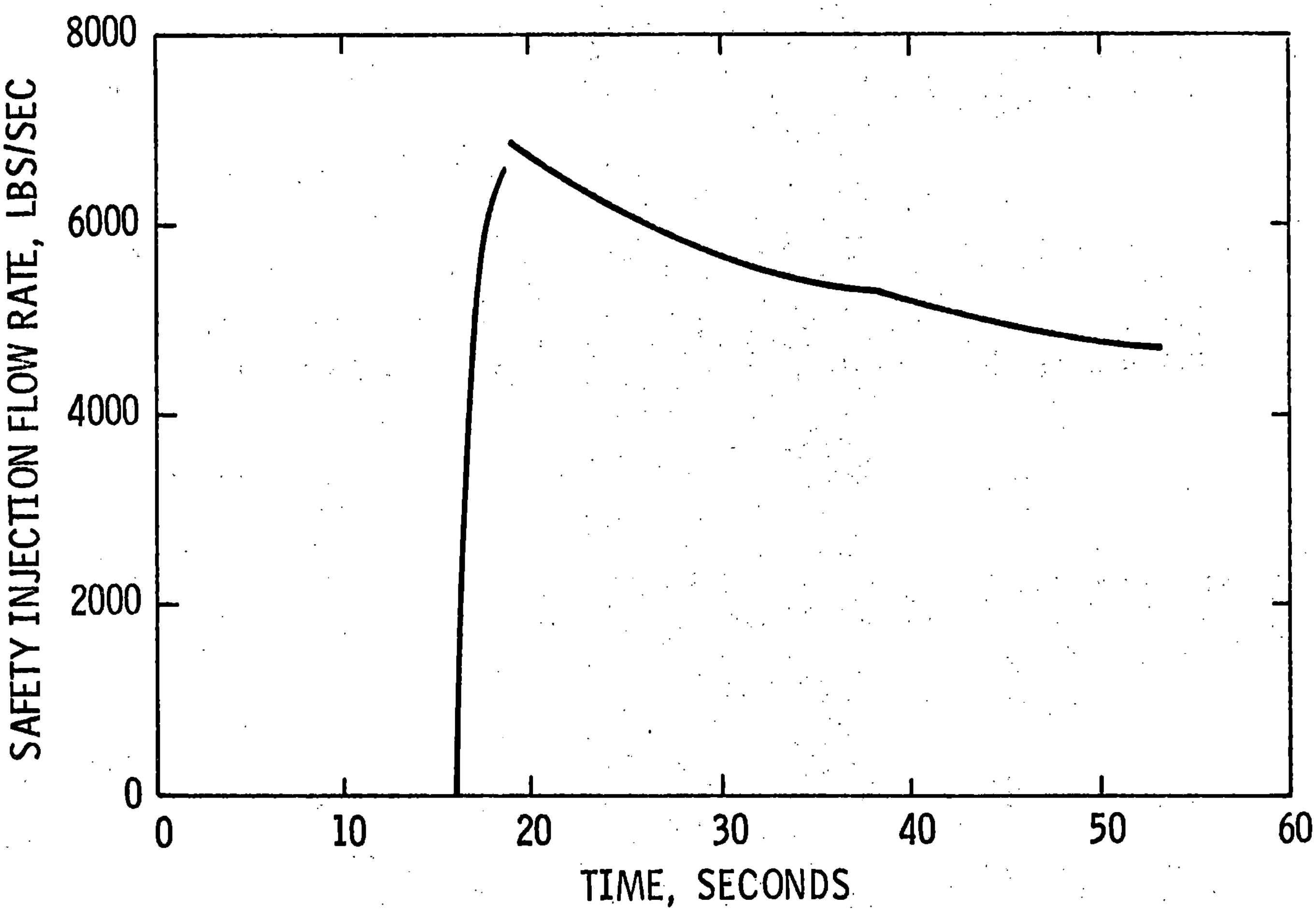


FIGURE II.1-M

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
WATER LEVEL IN DOWNCOMER DURING REFLOOD

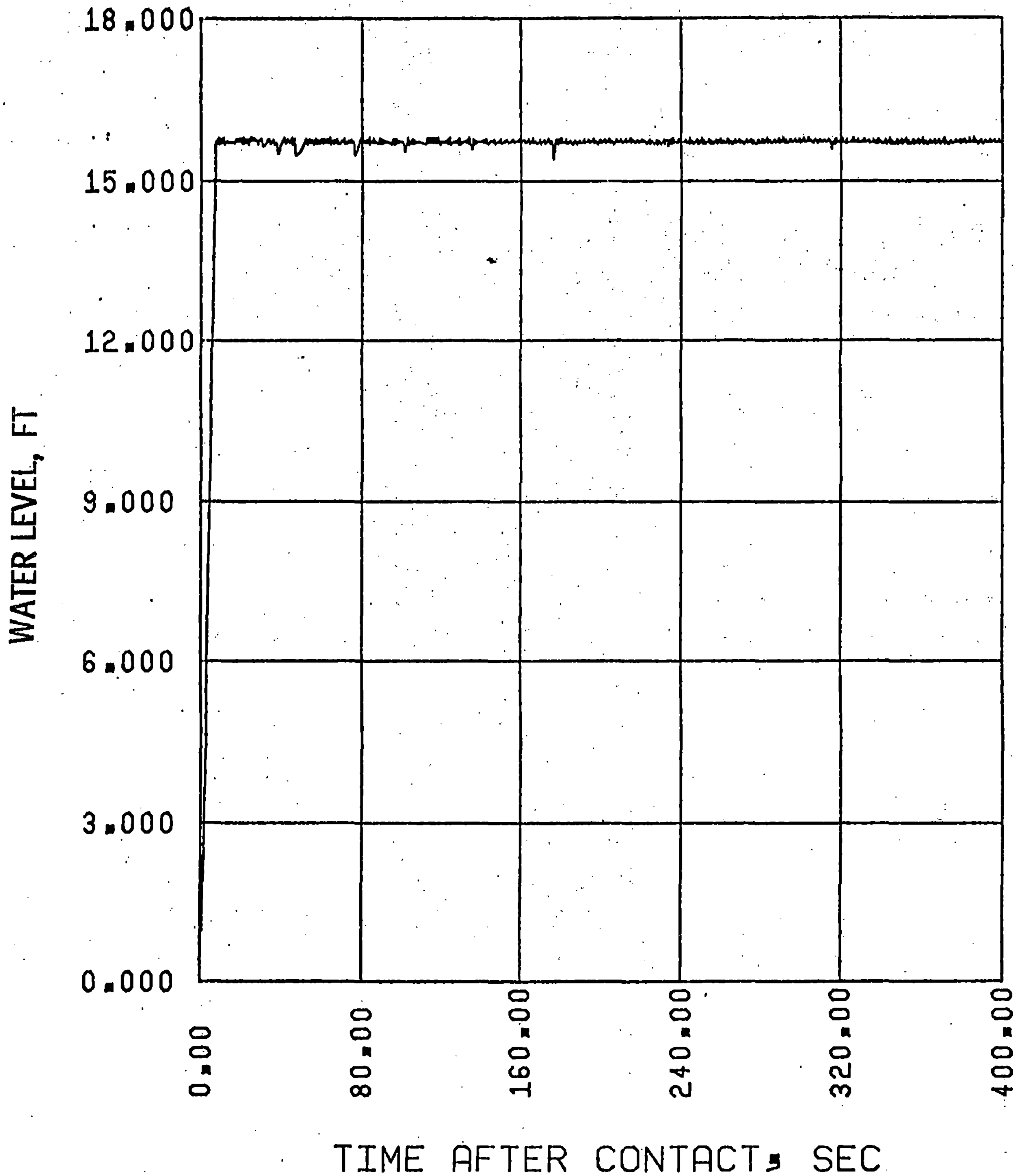


FIGURE II.1-N

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
GAP CONDUCTANCE

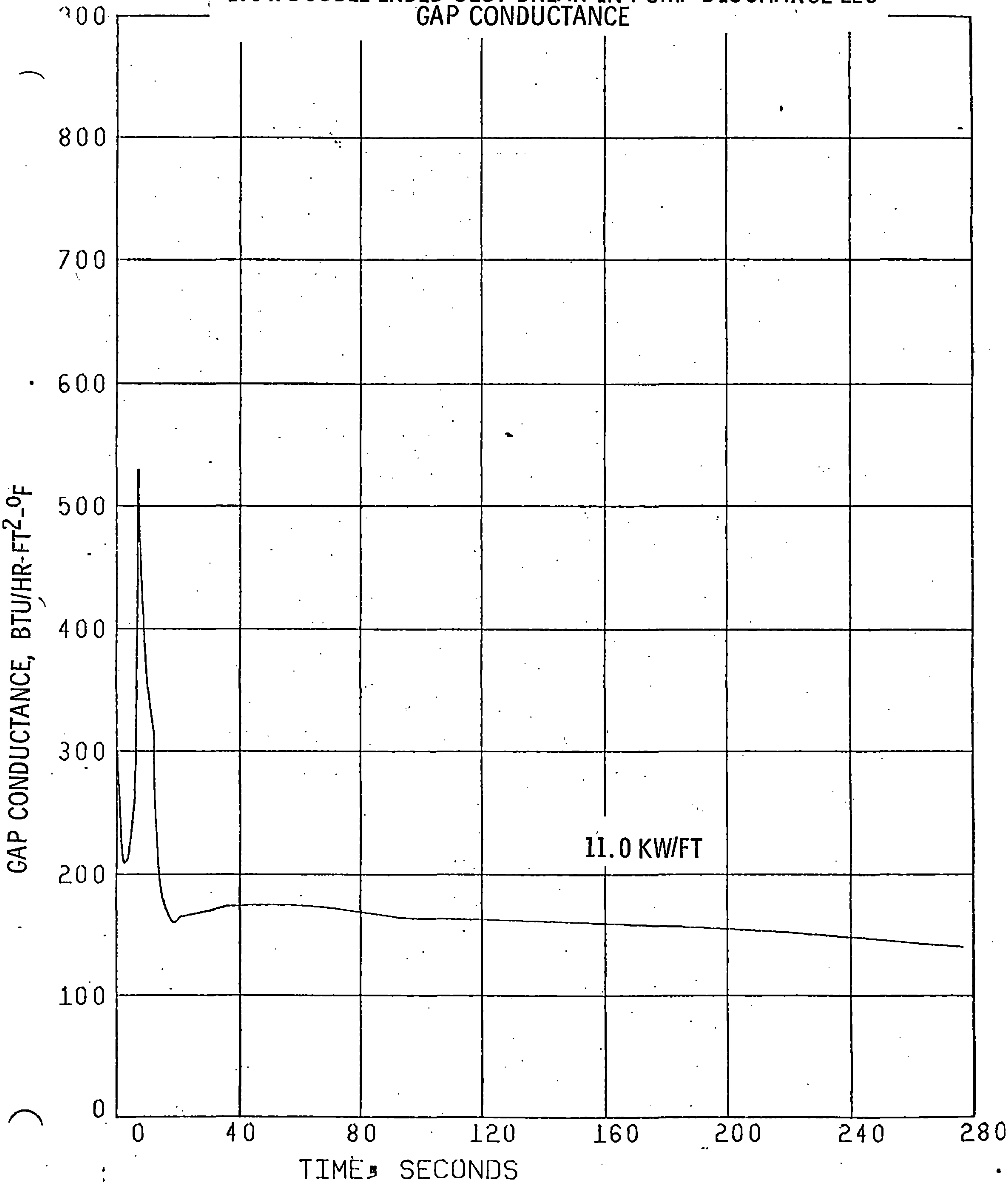


FIGURE II.1-0

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
LOCAL CLAD OXIDATION

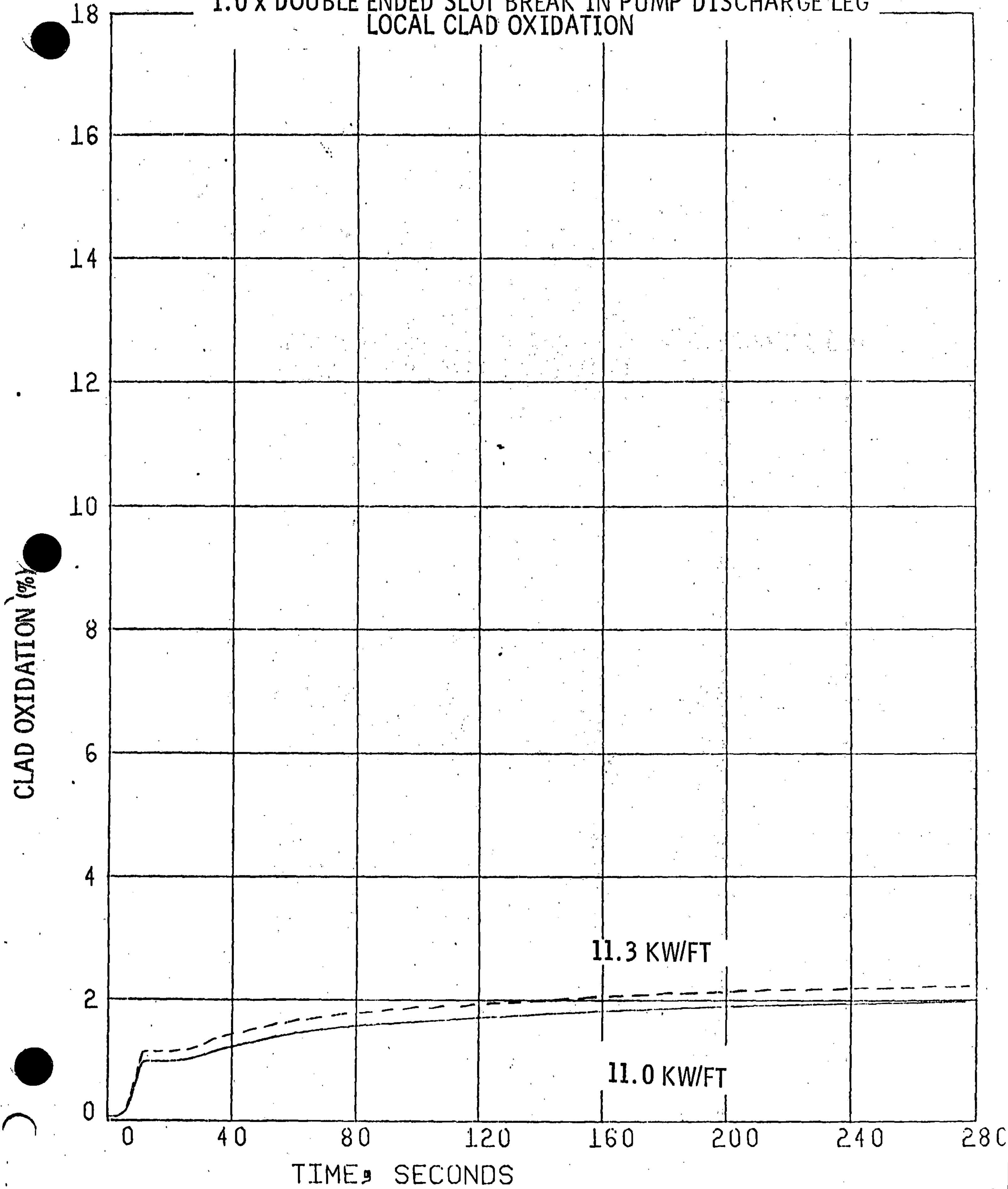


FIGURE II.1-P

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CLAD TEMPERATURE, CENTERLINE FUEL TEMP., AVG. FUEL TEMP., AND COOLANT TEMP. FOR HOTTEST NODE

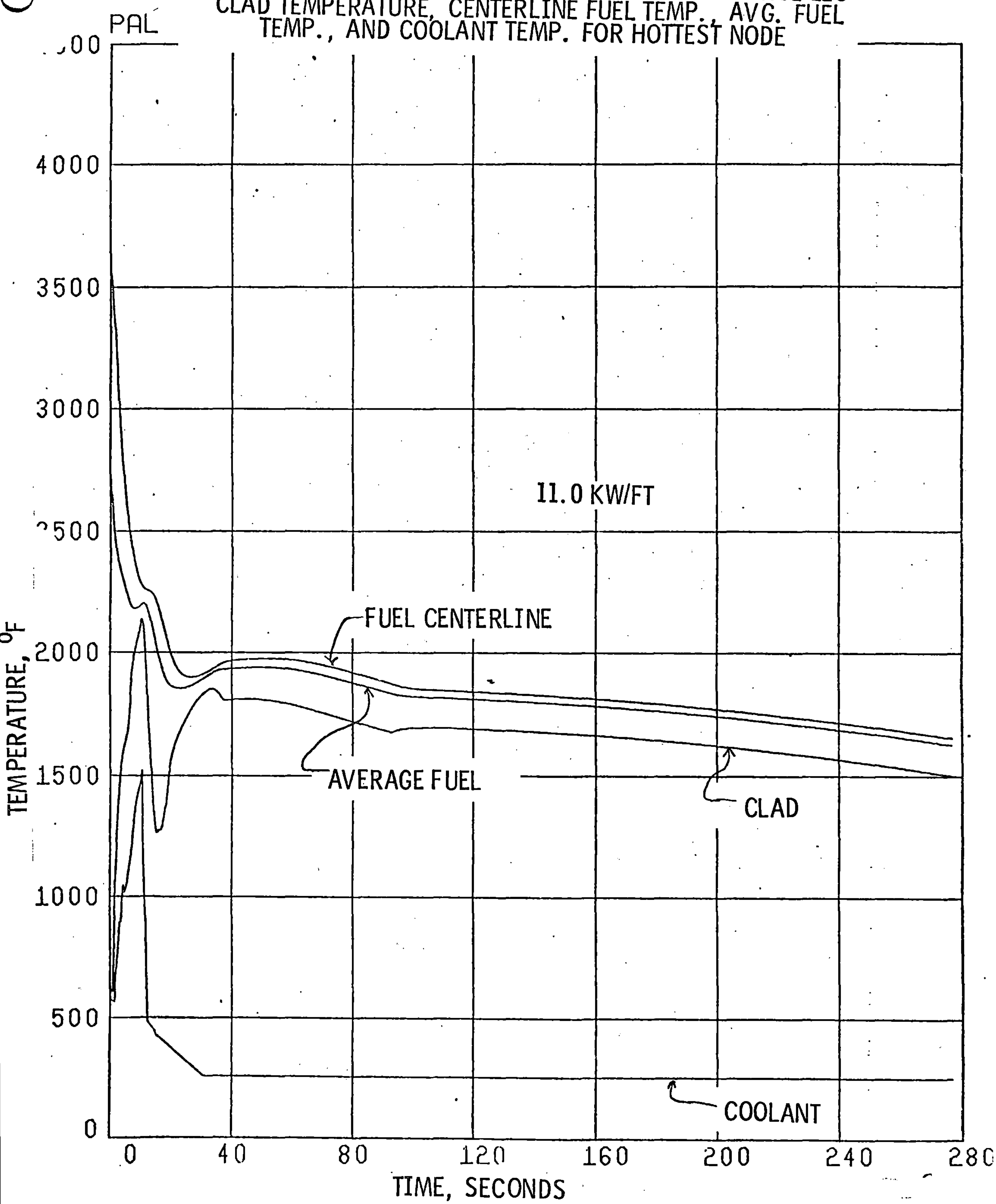


FIGURE II.1-Q

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT HEAT TRANSFER COEFFICIENT

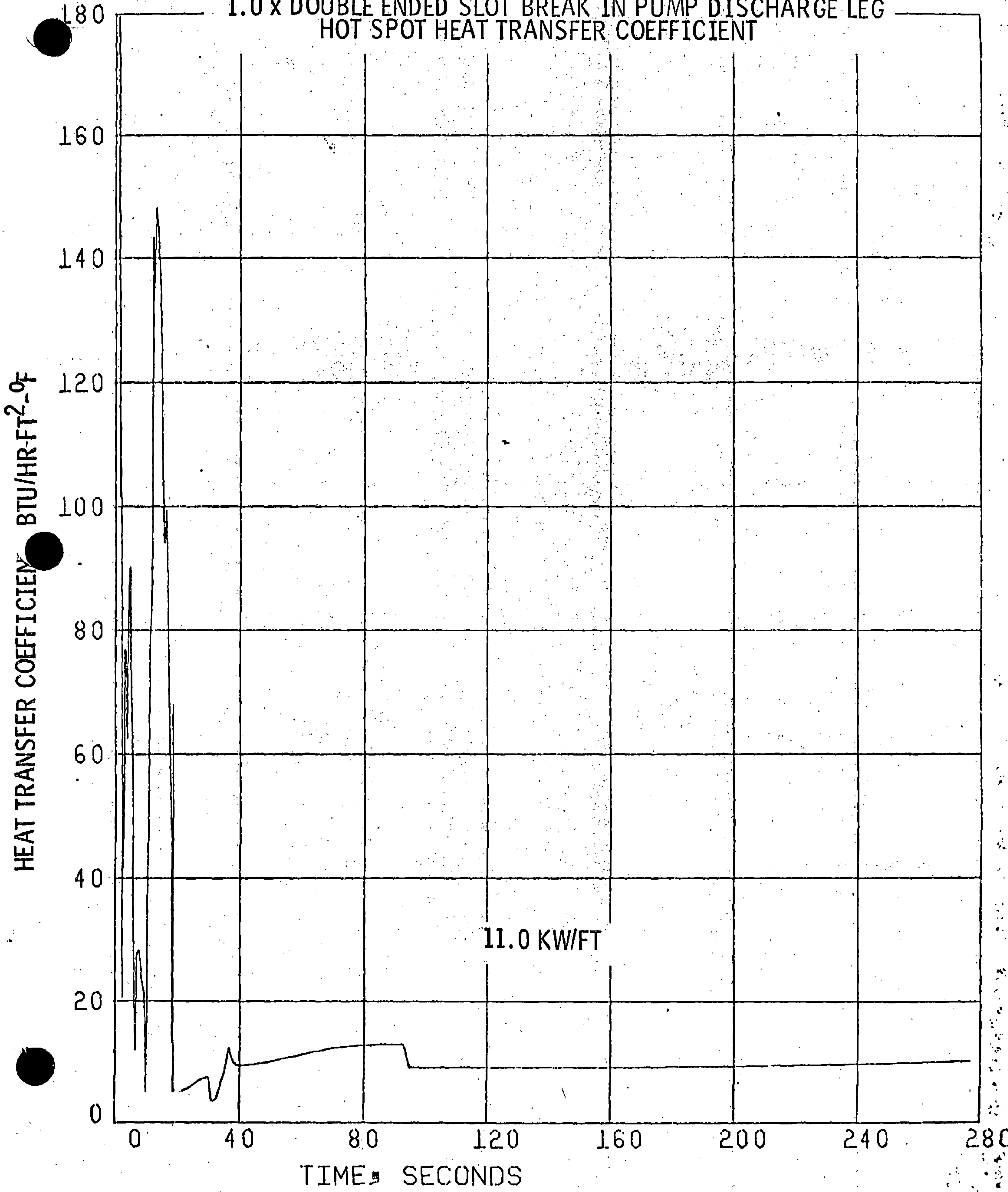


FIGURE II.1-R

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT HEAT TRANSFER COEFFICIENT DURING REFLOOD

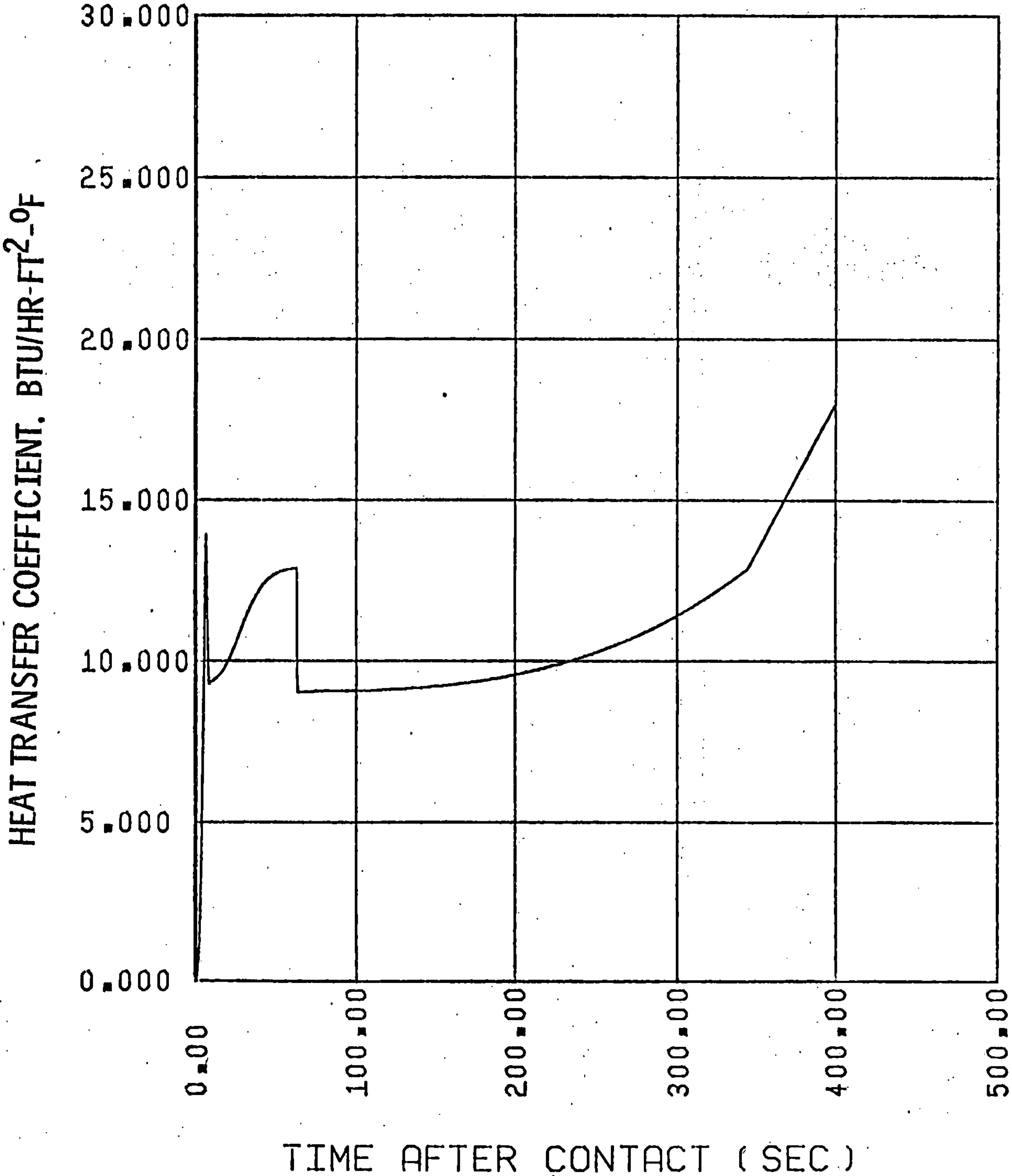


FIGURE II.1-S

PALISADES CORE I REANALYSIS

1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG

CONTAINMENT TEMPERATURE

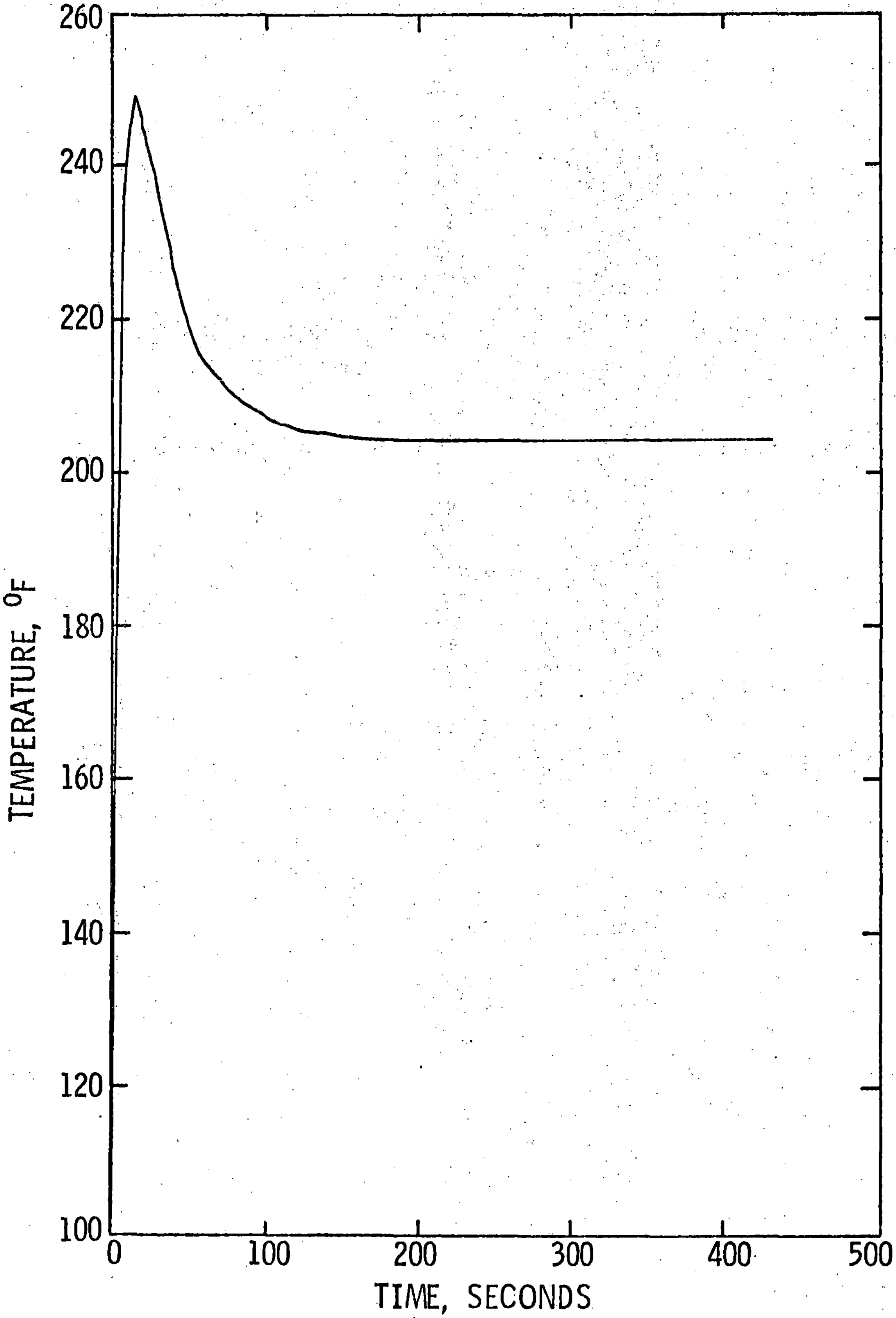


FIGURE II.1-T

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
SUMP TEMPERATURE

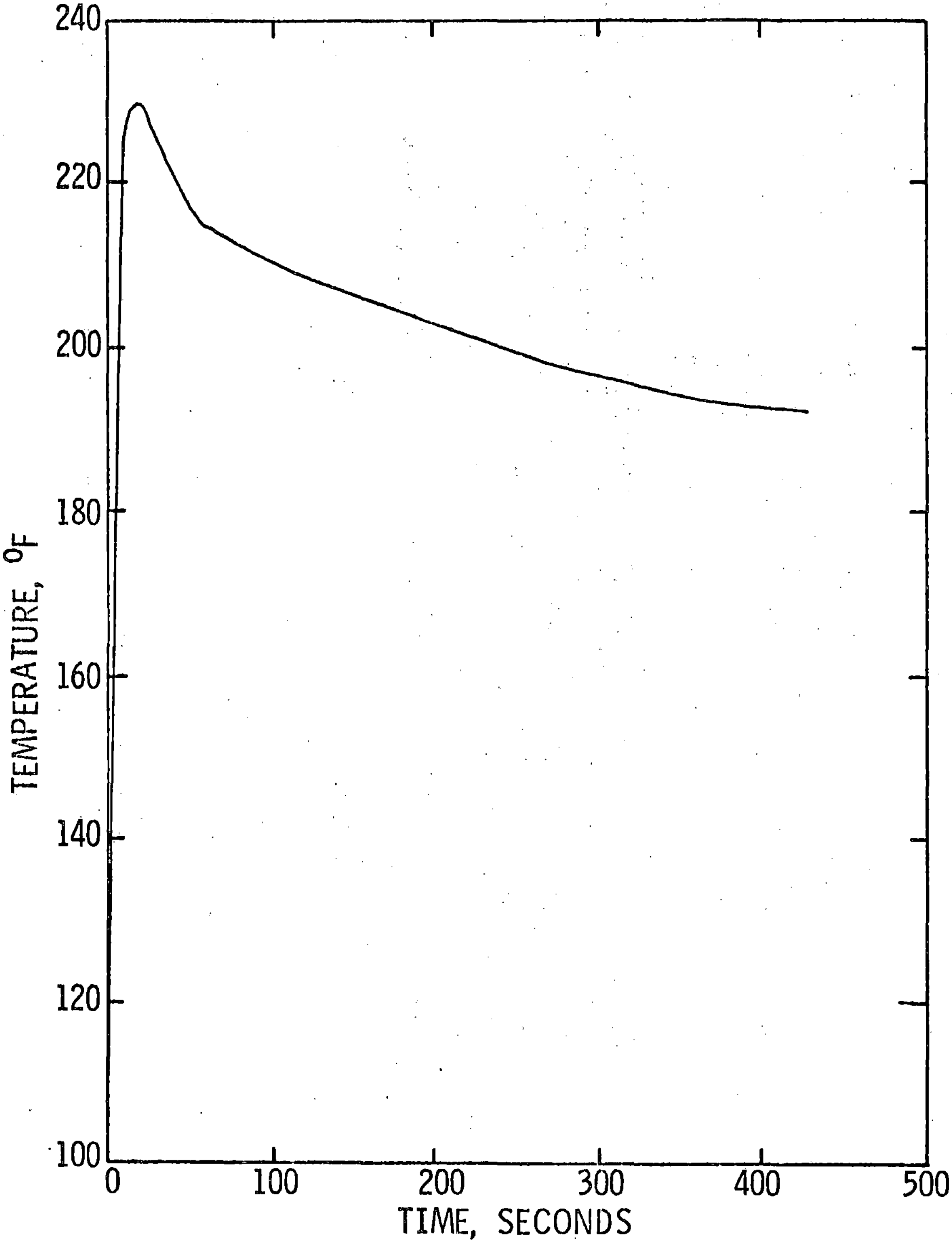


FIGURE II.2-A

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CORE POWER

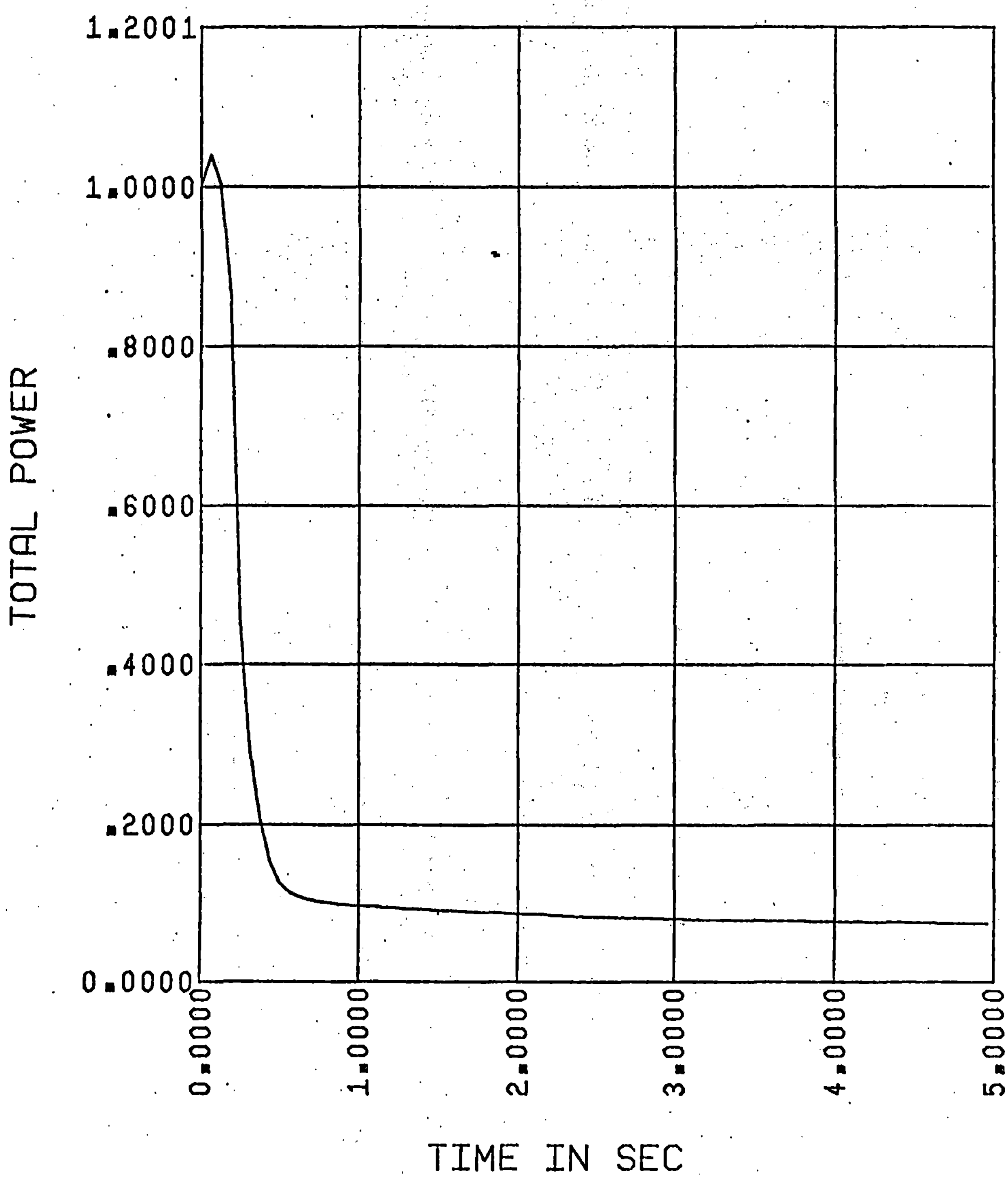


FIGURE II.2-B

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

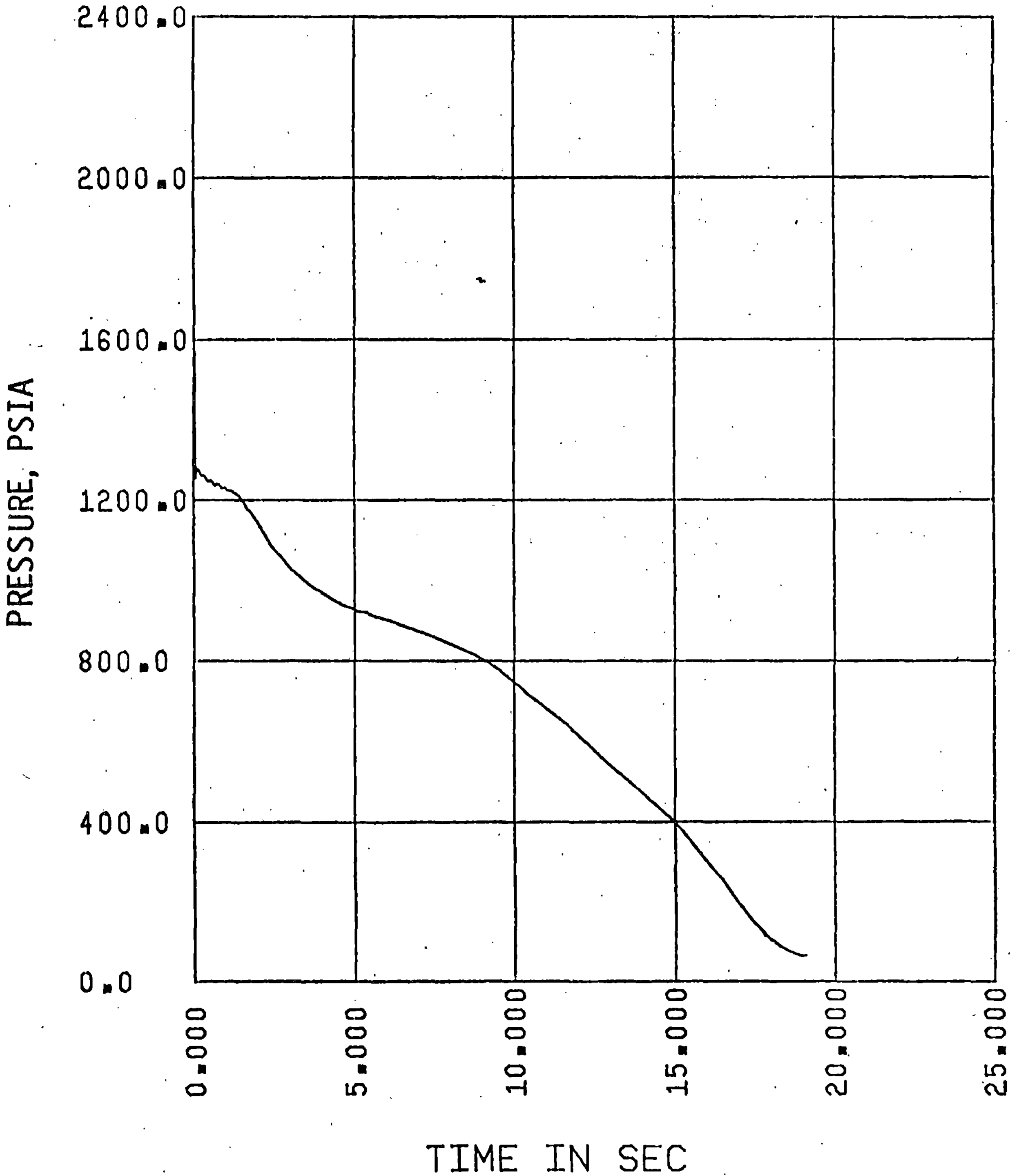


FIGURE II.2-C

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

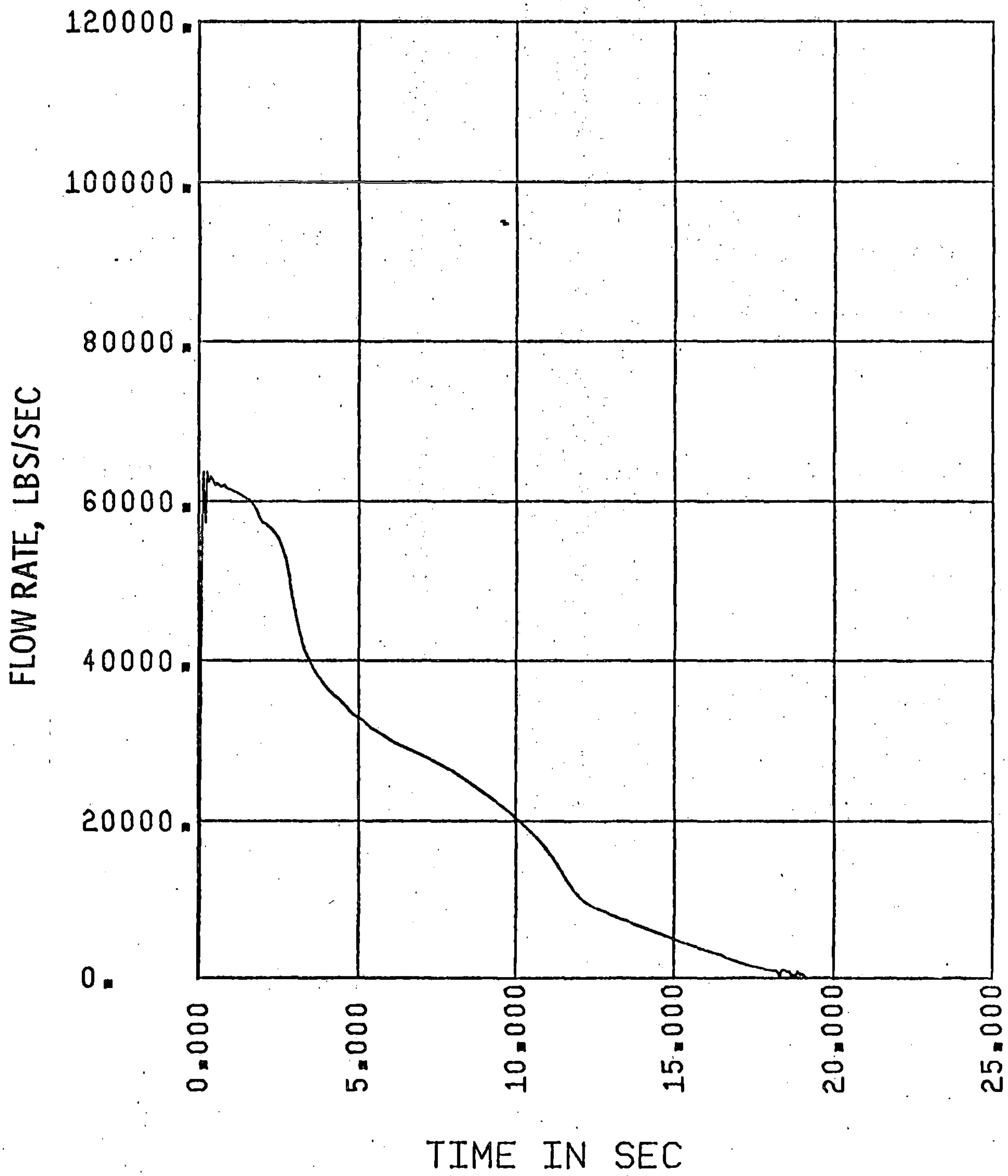


FIGURE II.2-D.1

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

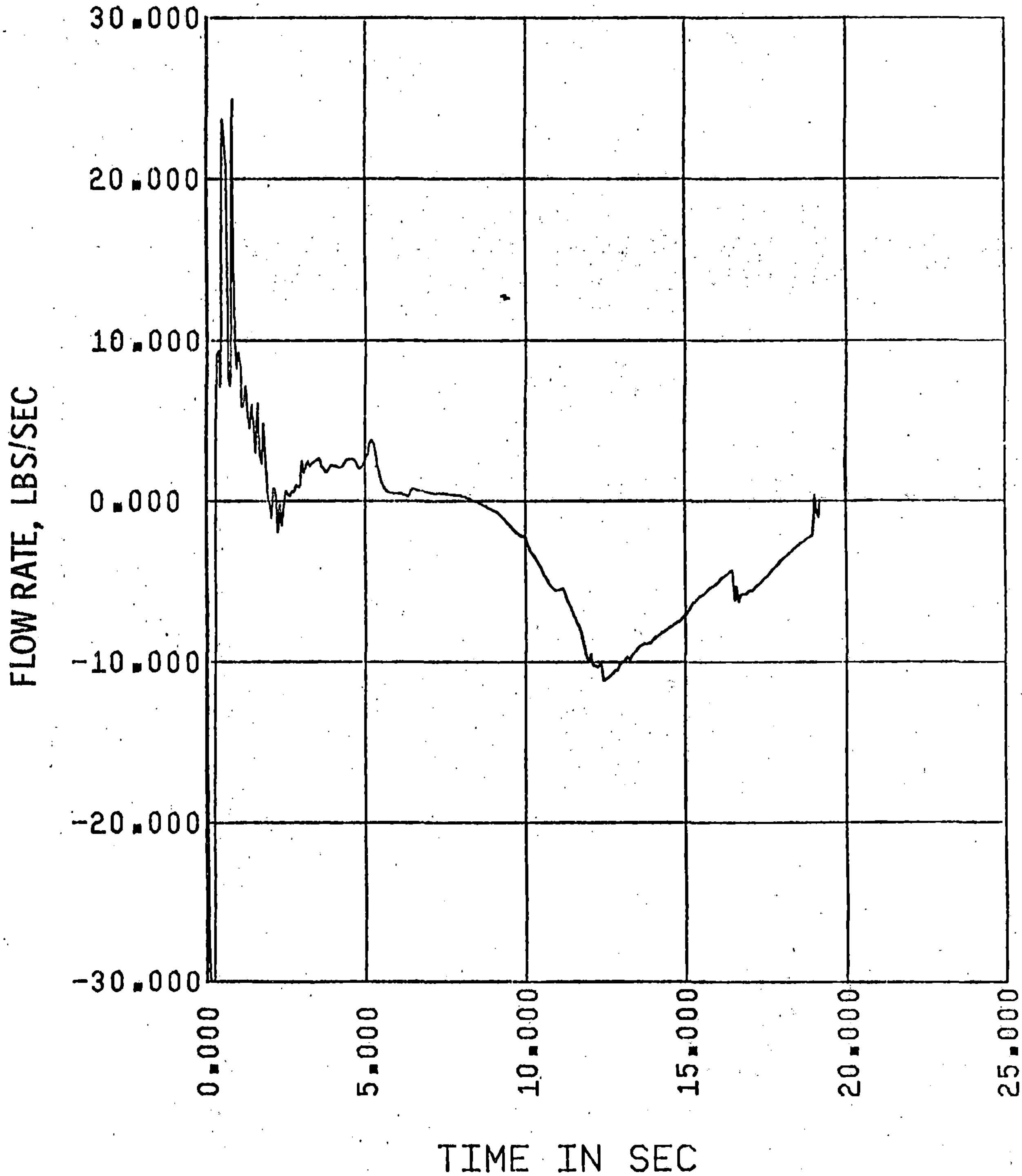


FIGURE II.2-D.2

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

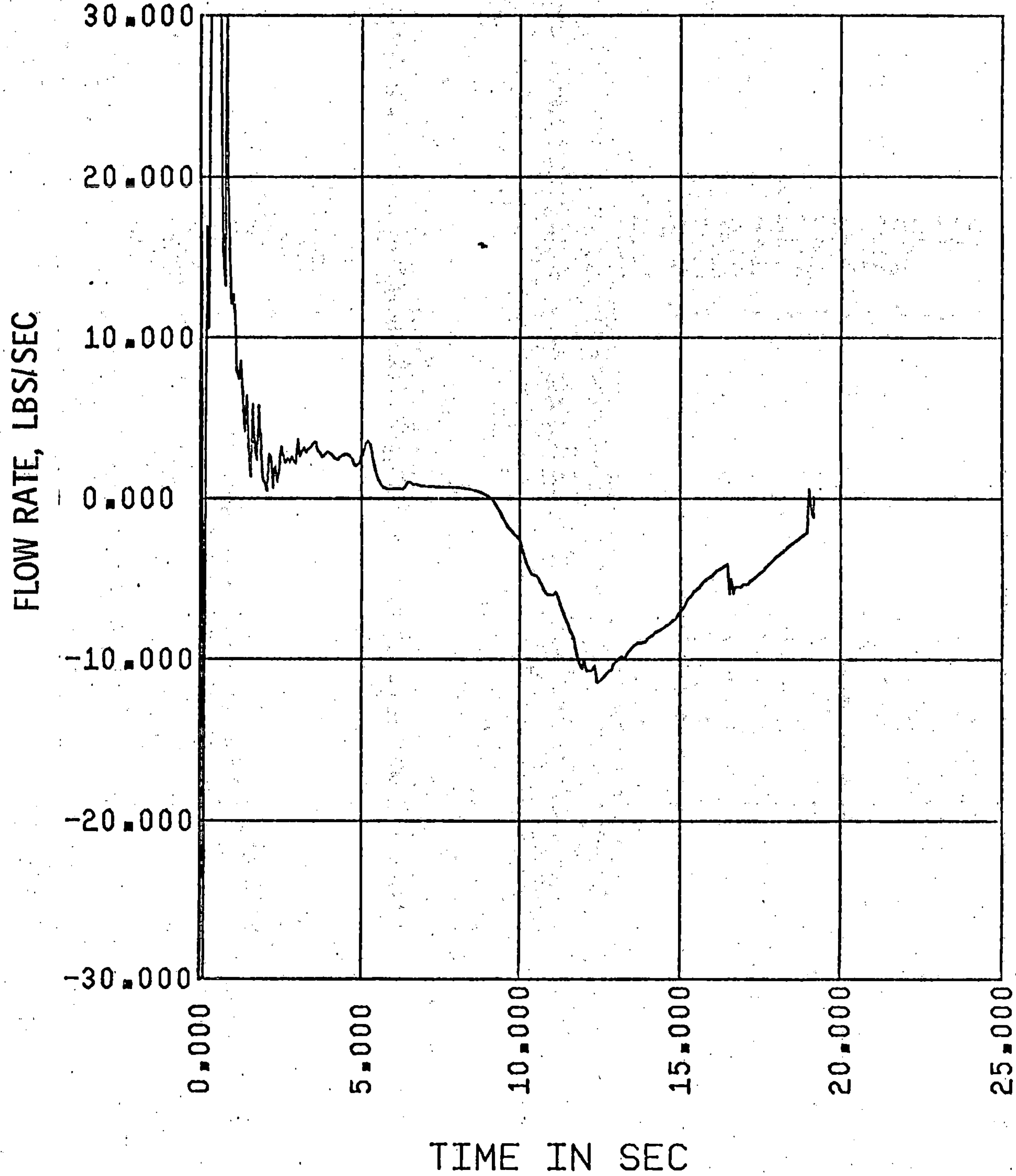


FIGURE II.2-E

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

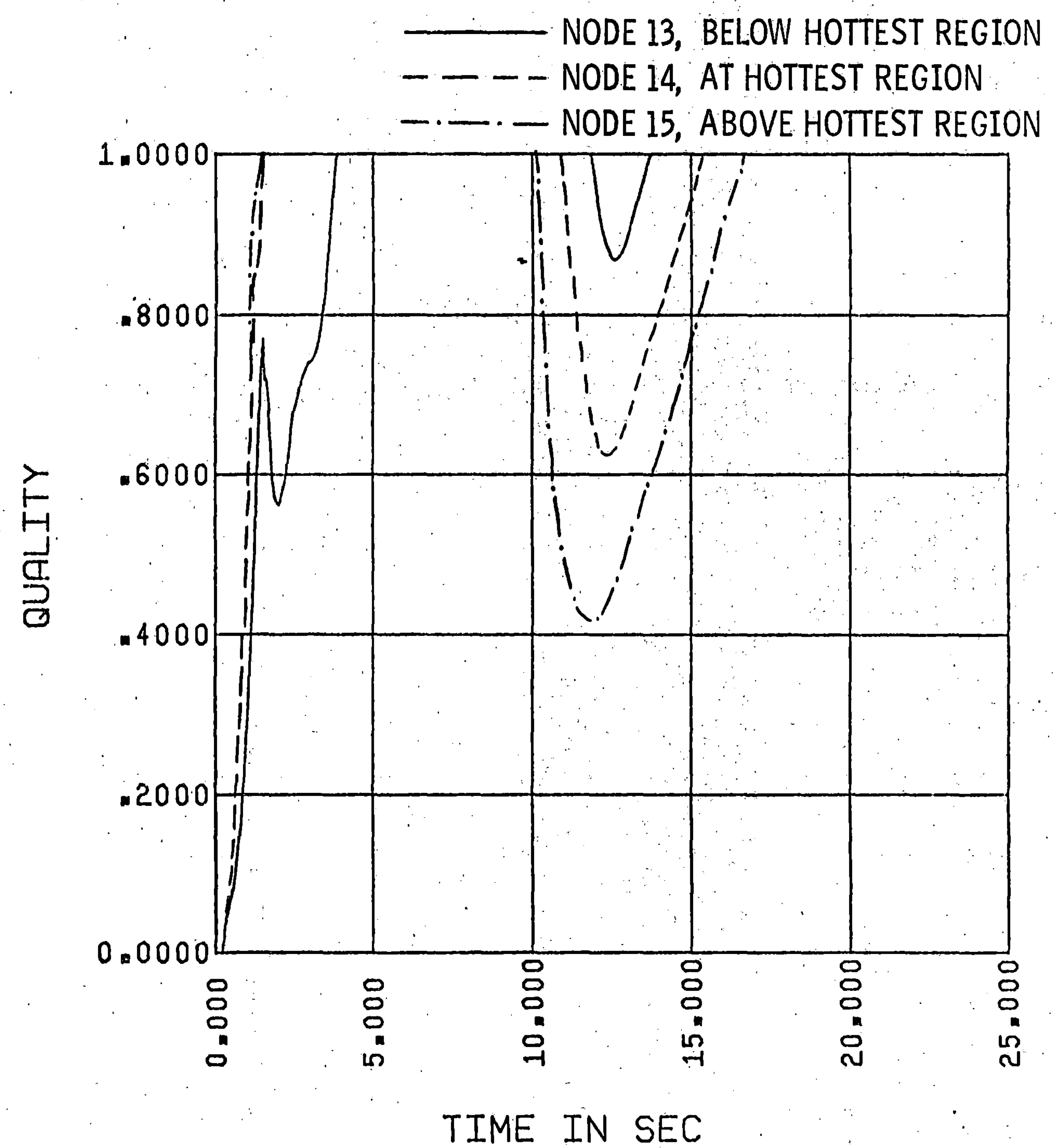


FIGURE II.2-F

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

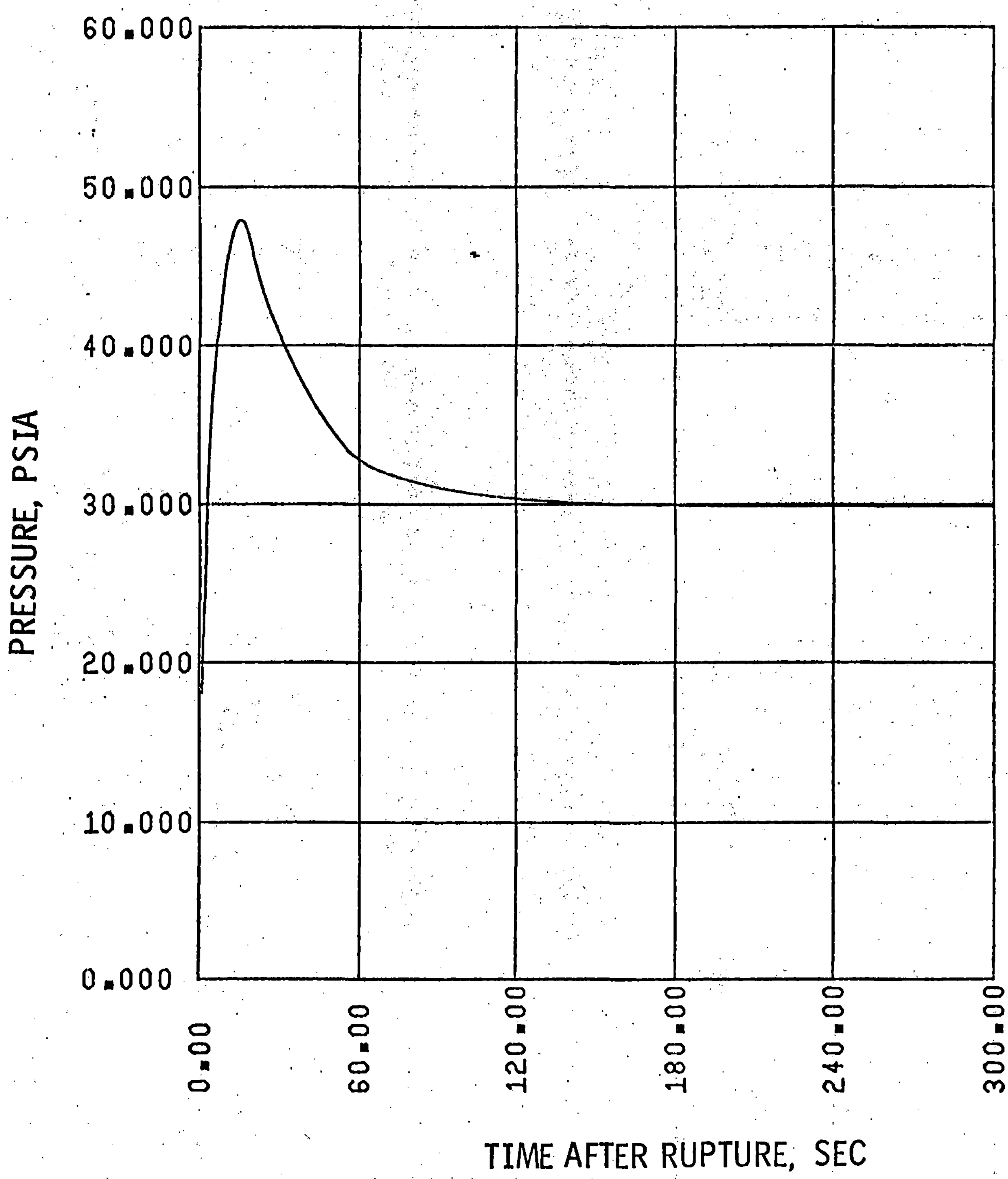


FIGURE II.2-G

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

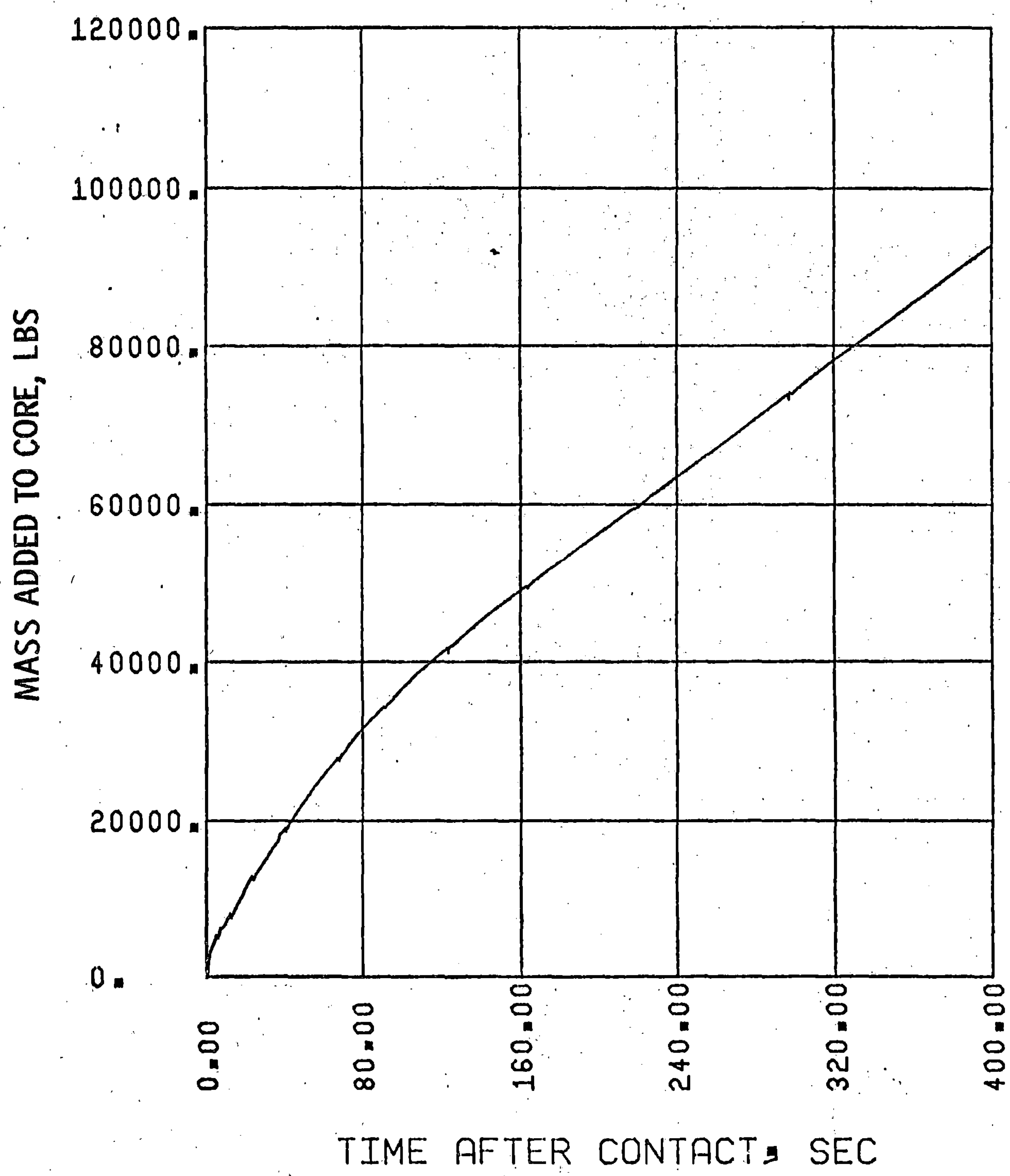


FIGURE II.2-H

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

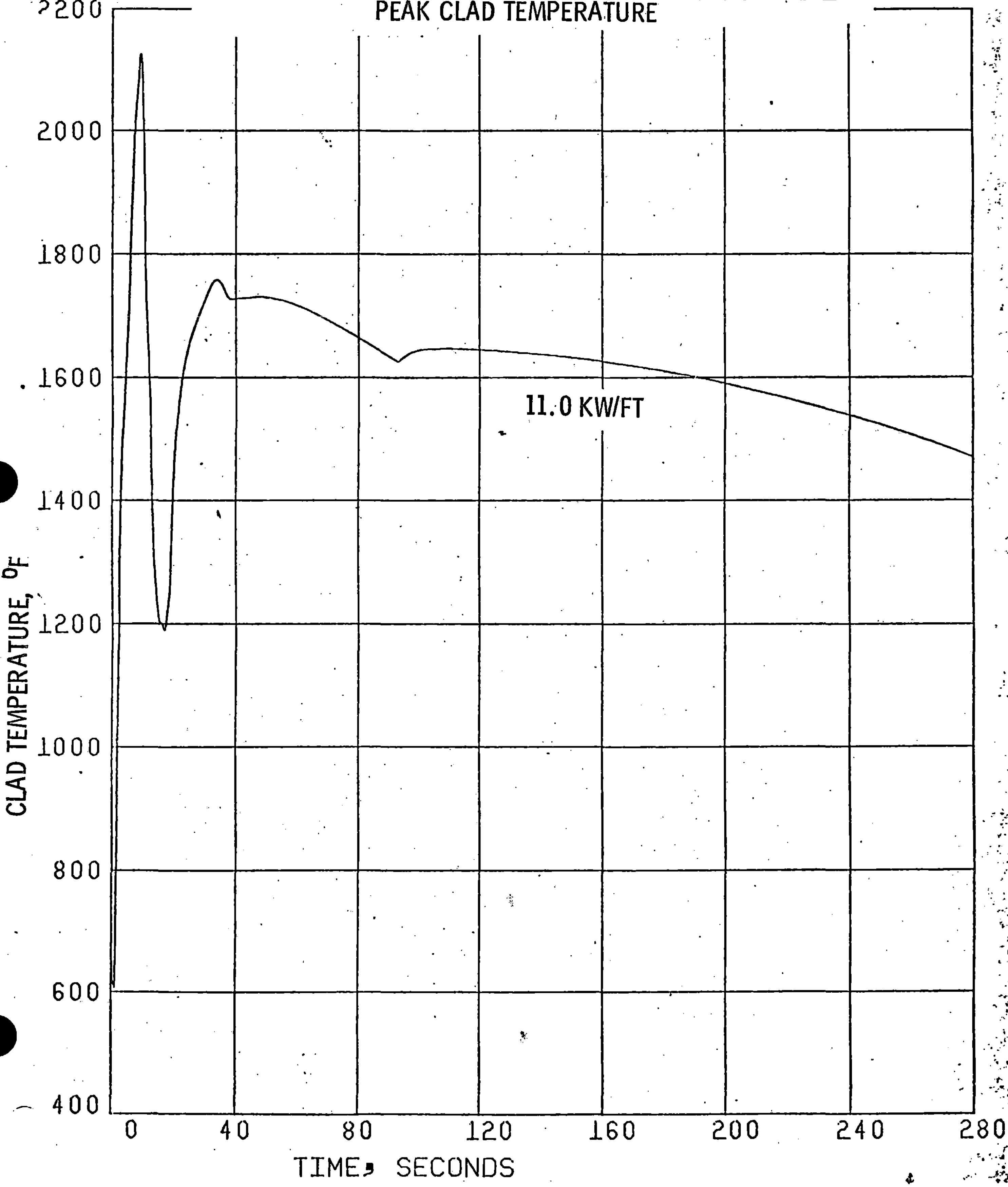


FIGURE II. 3-A

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CORE POWER

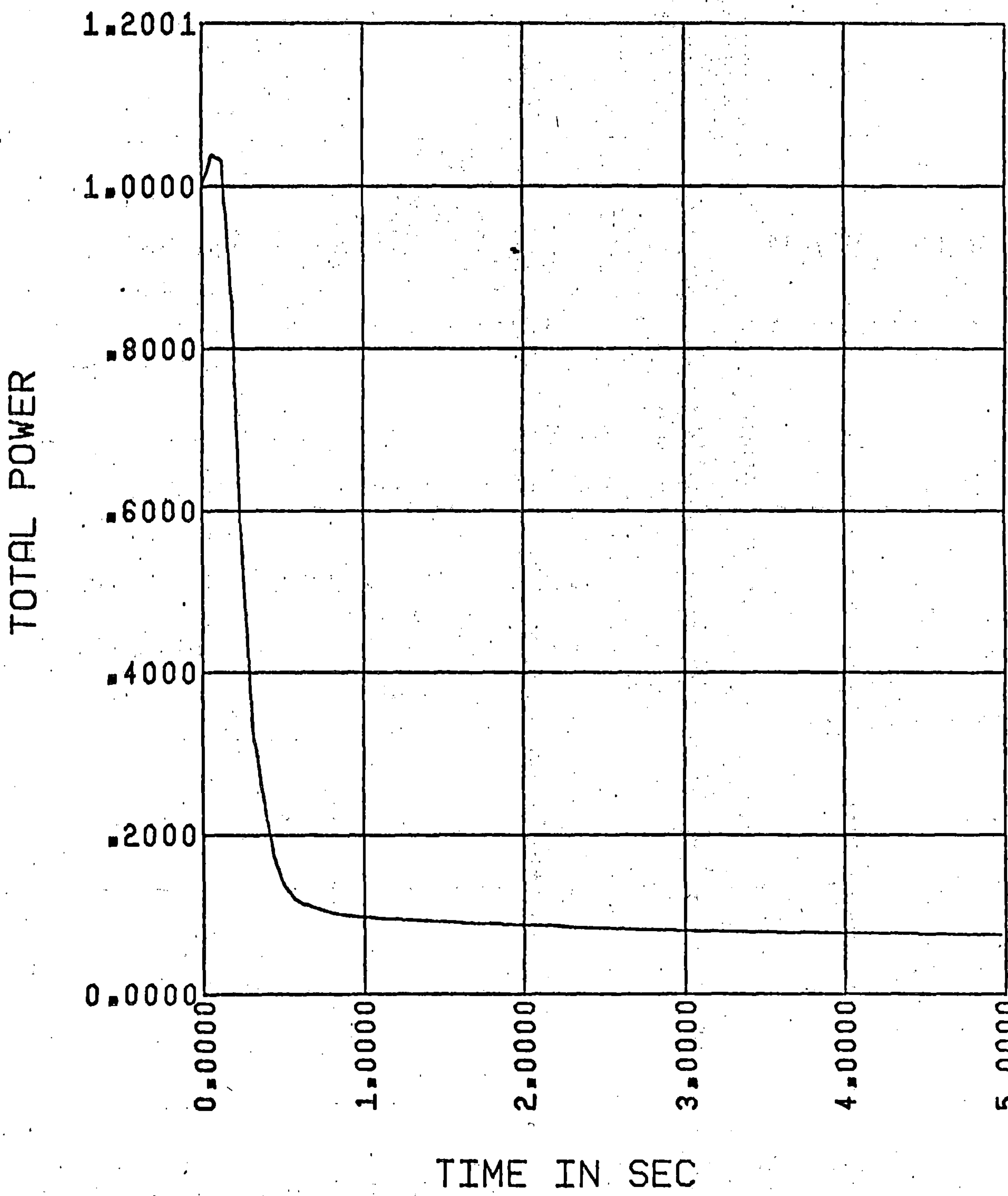


FIGURE II.3-B

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

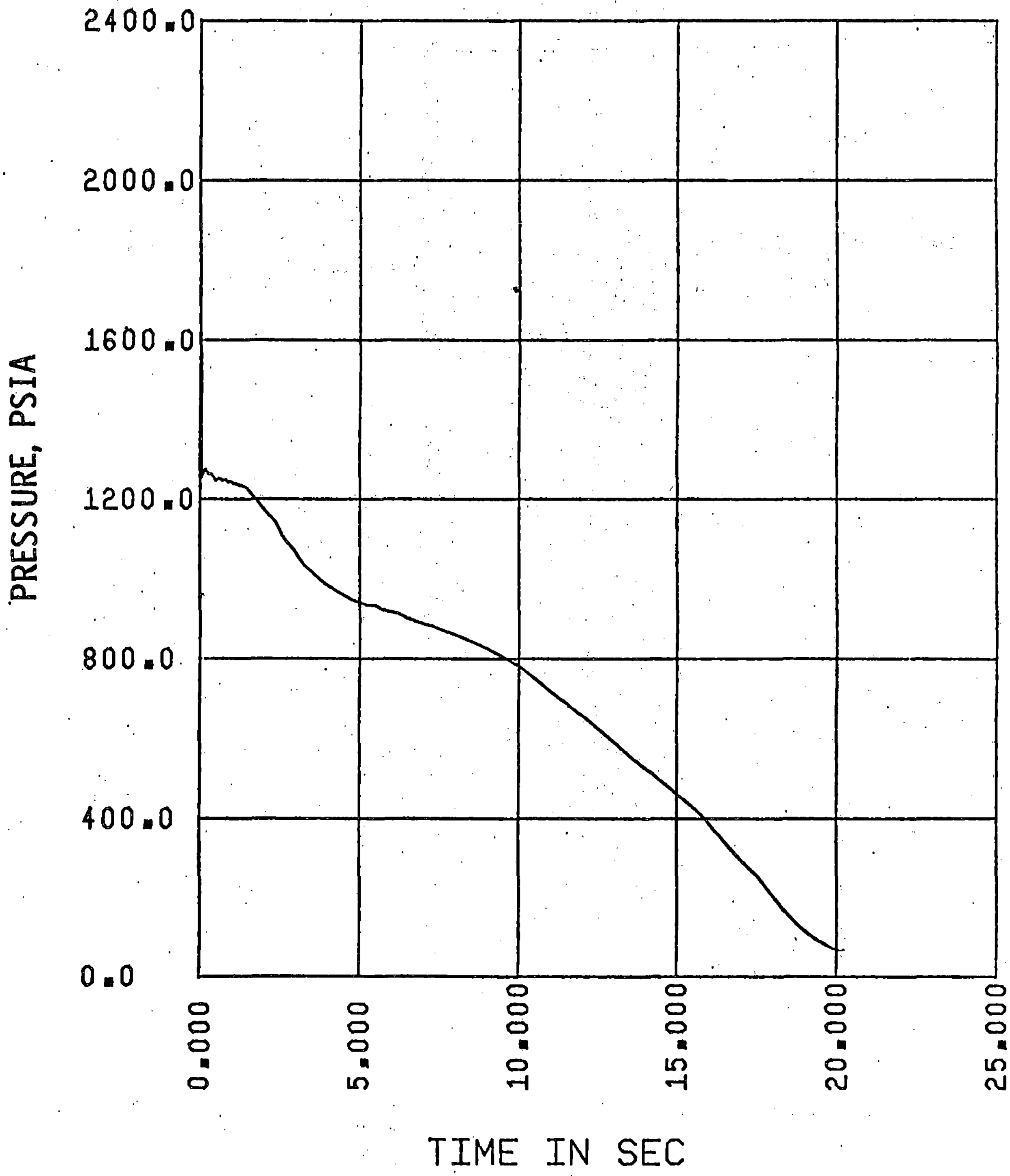


FIGURE II.3-C

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

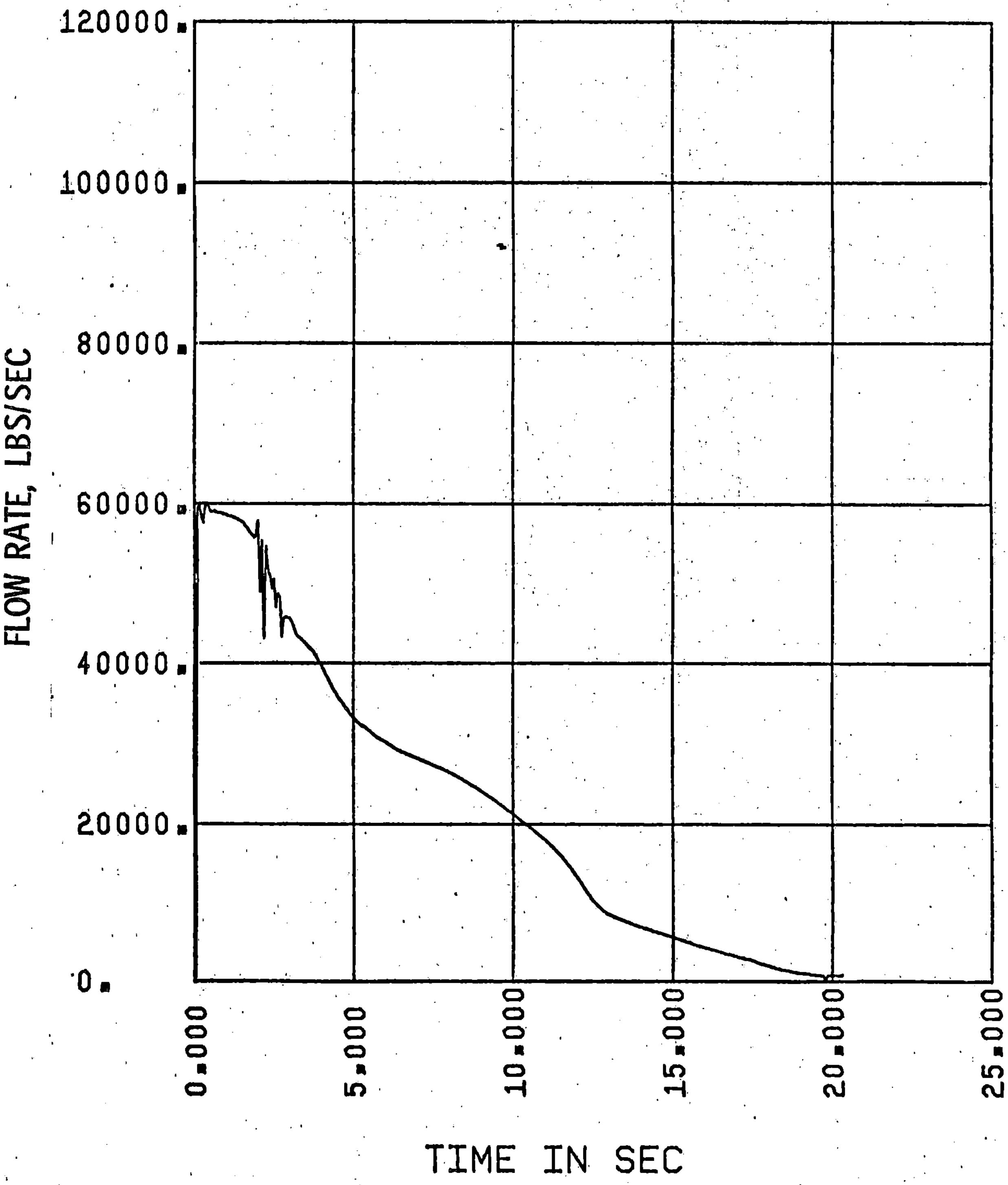


FIGURE II.3-D.1

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

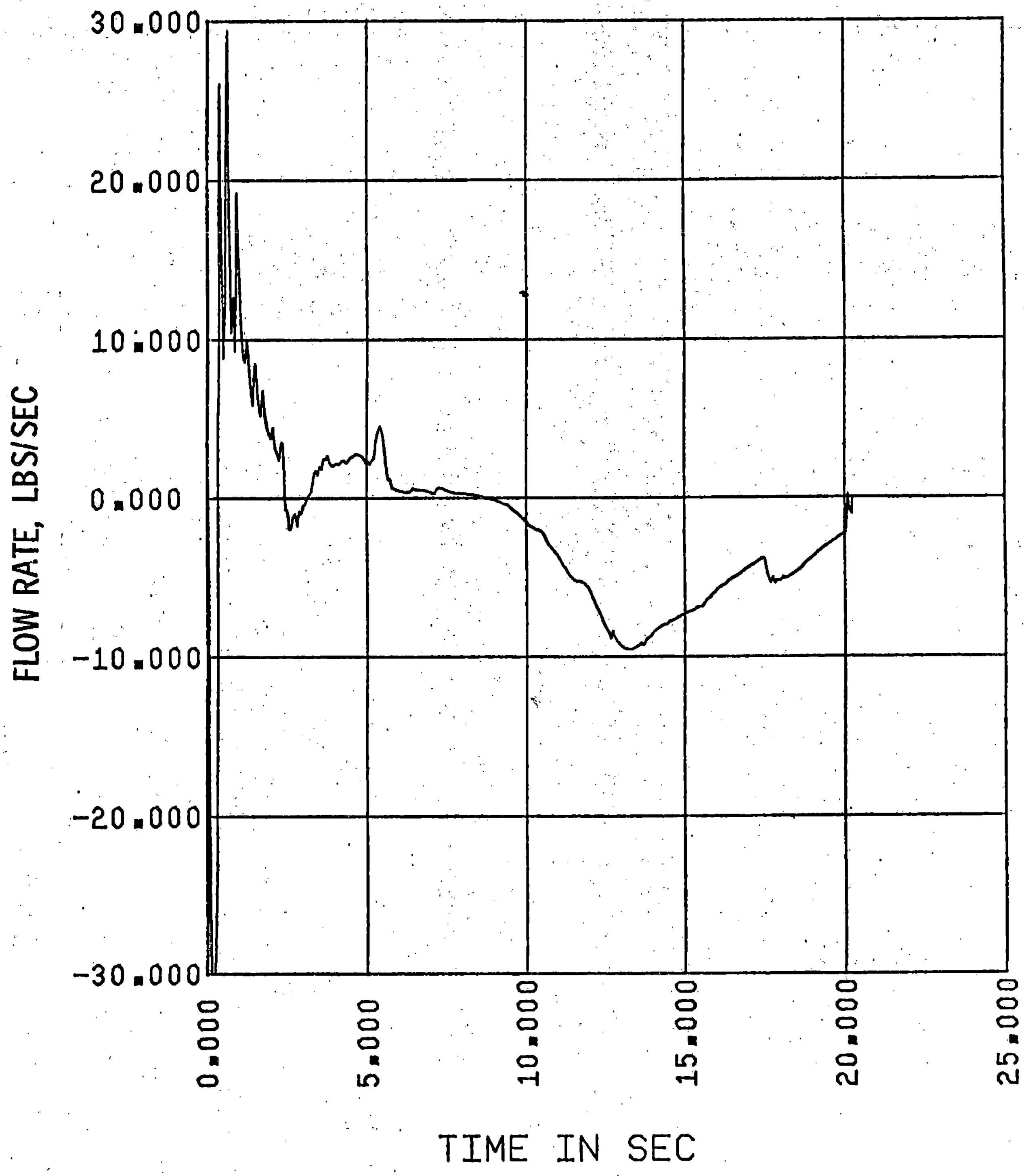


FIGURE II.3-D.2

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

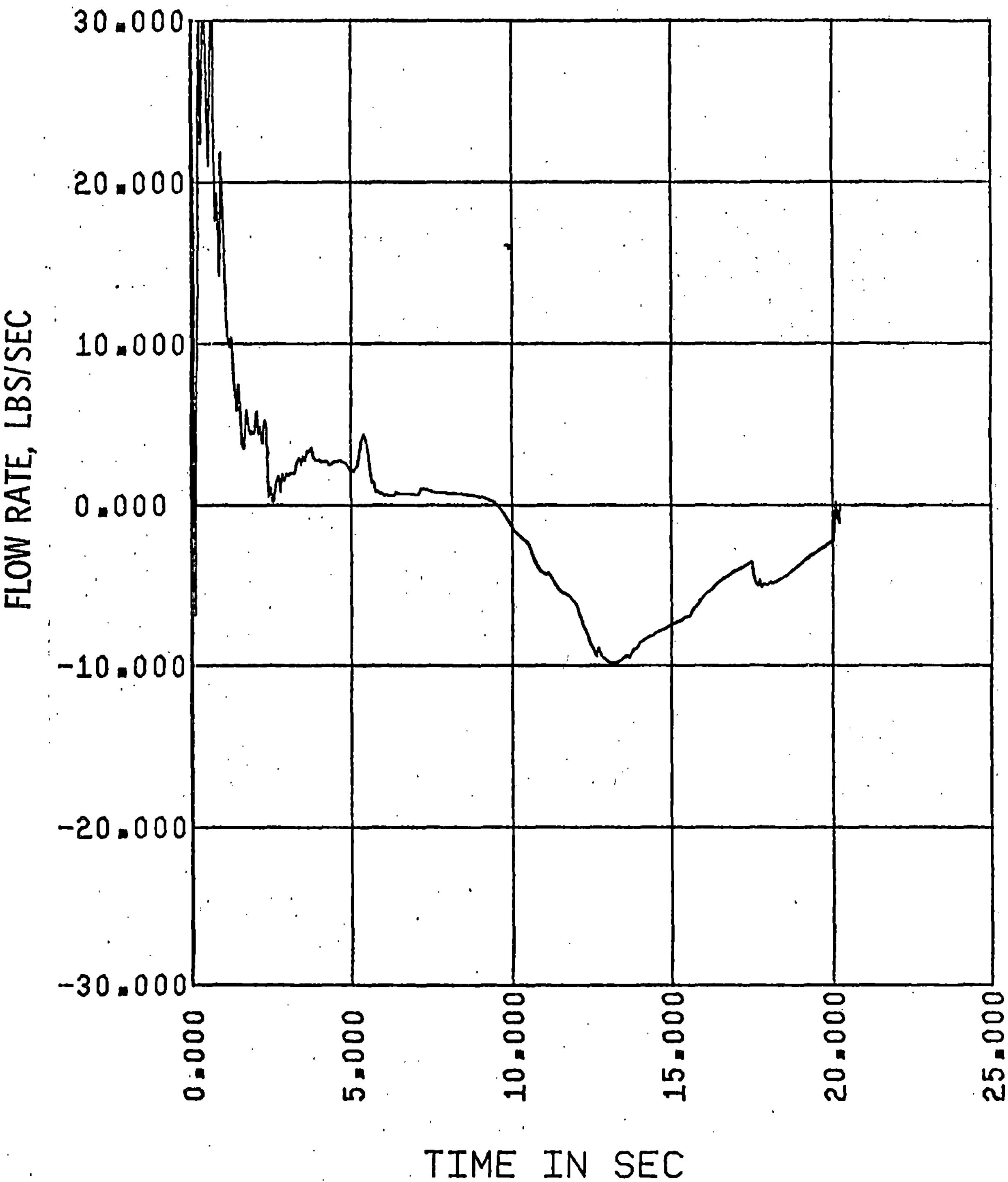


FIGURE II.3-E

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

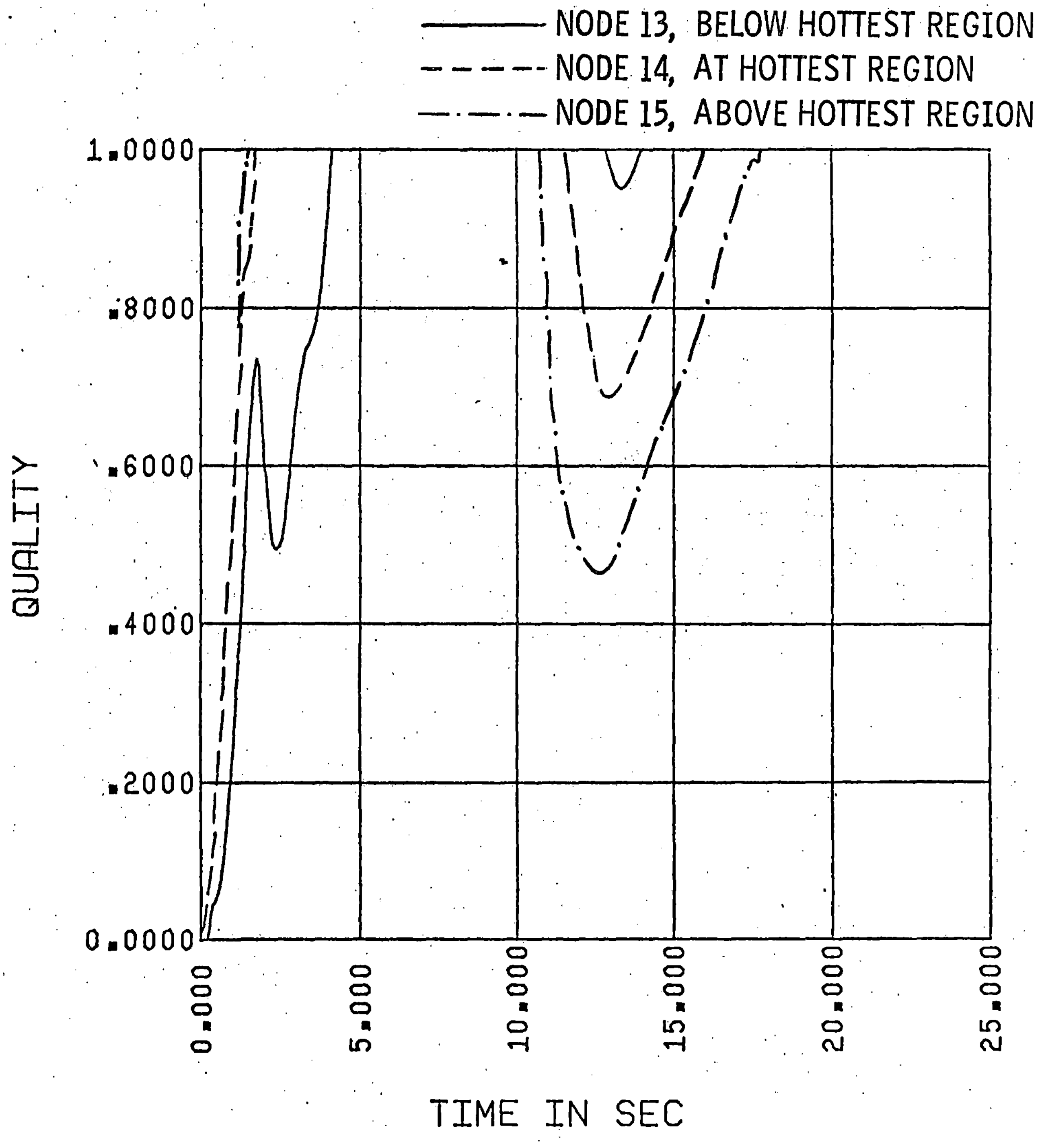


FIGURE II.3-F

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

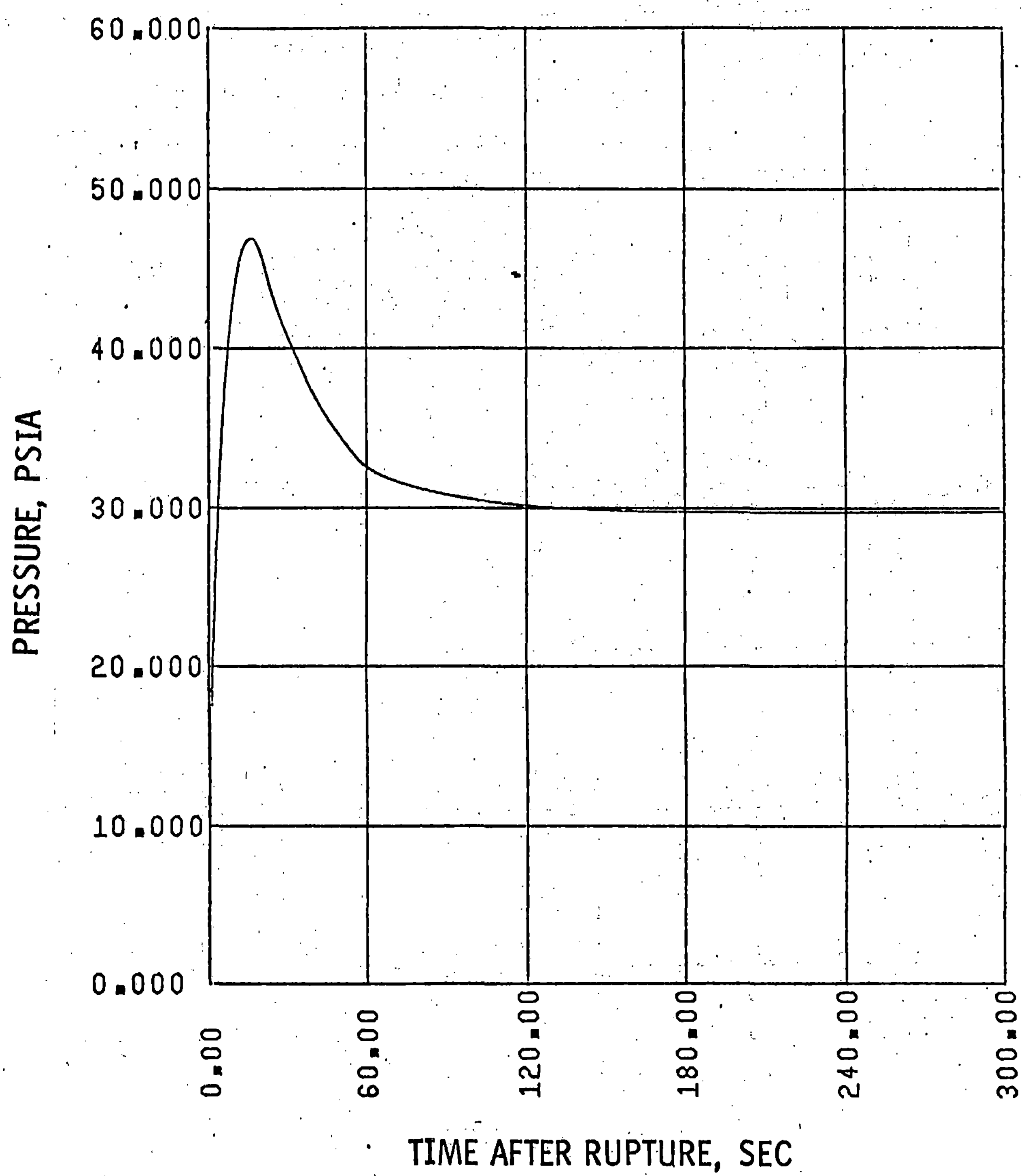


FIGURE II.3-G

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

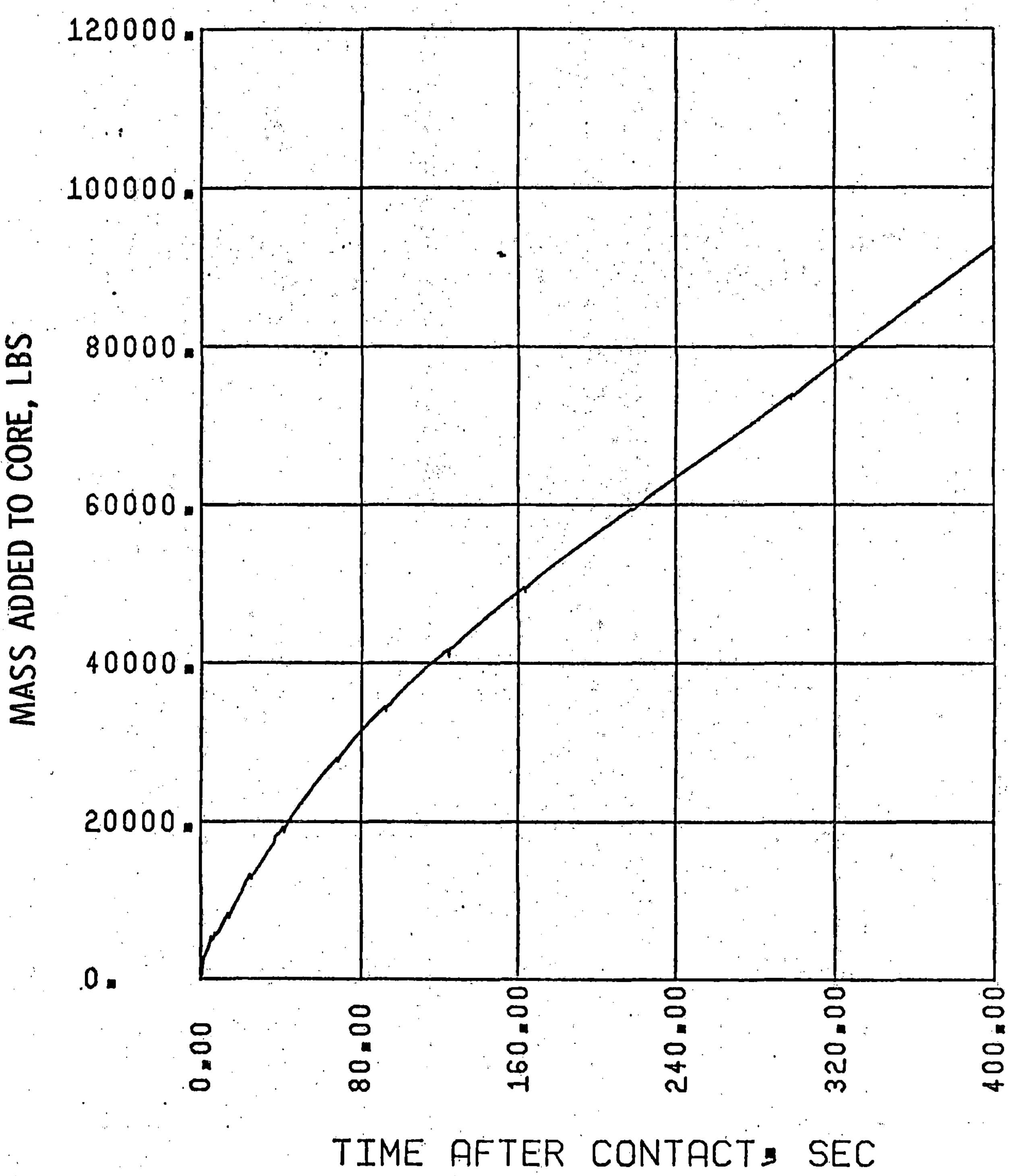


FIGURE II.3-H

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

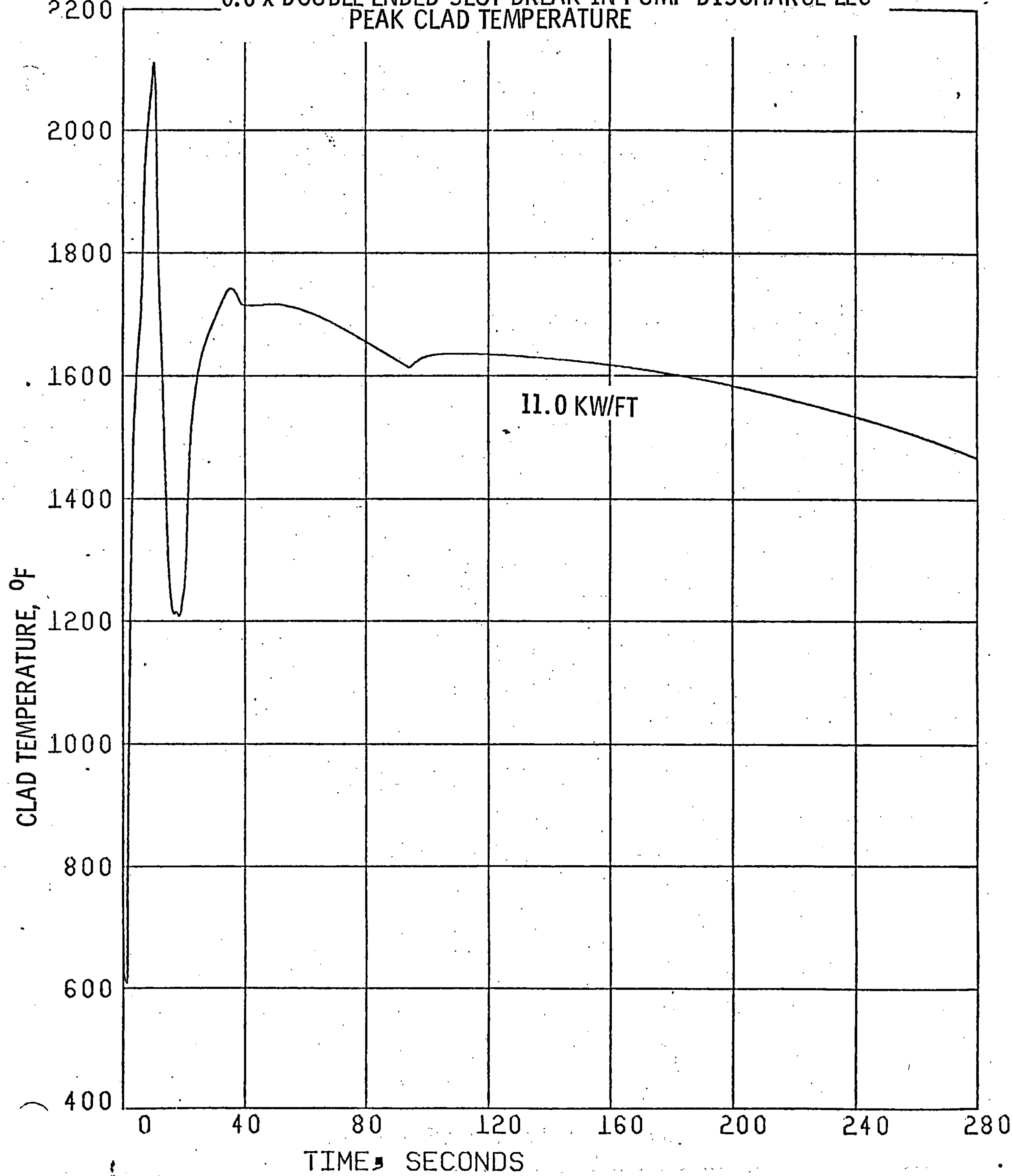


FIGURE II.4-A

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CORE POWER

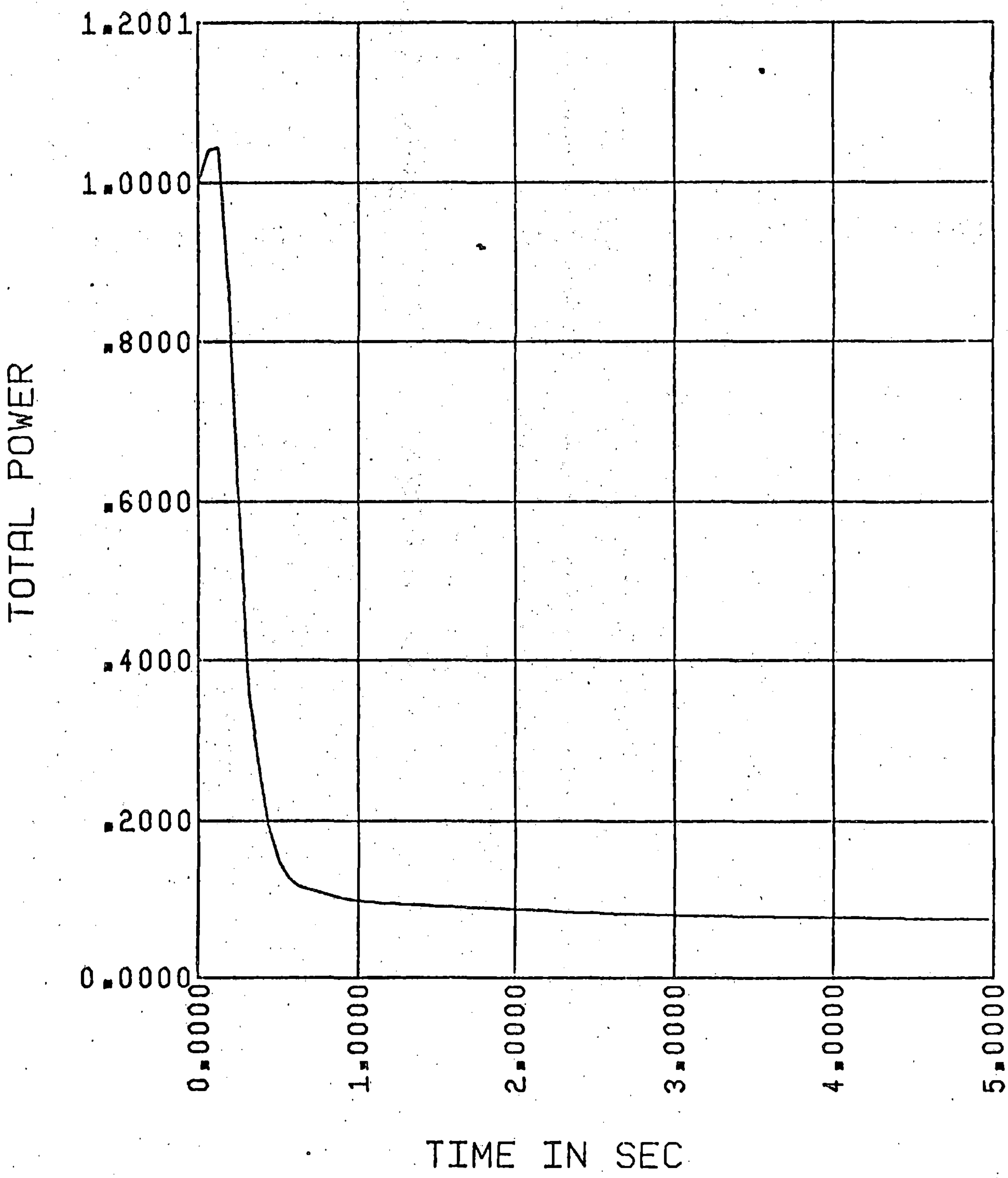


FIGURE II. 4-B

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

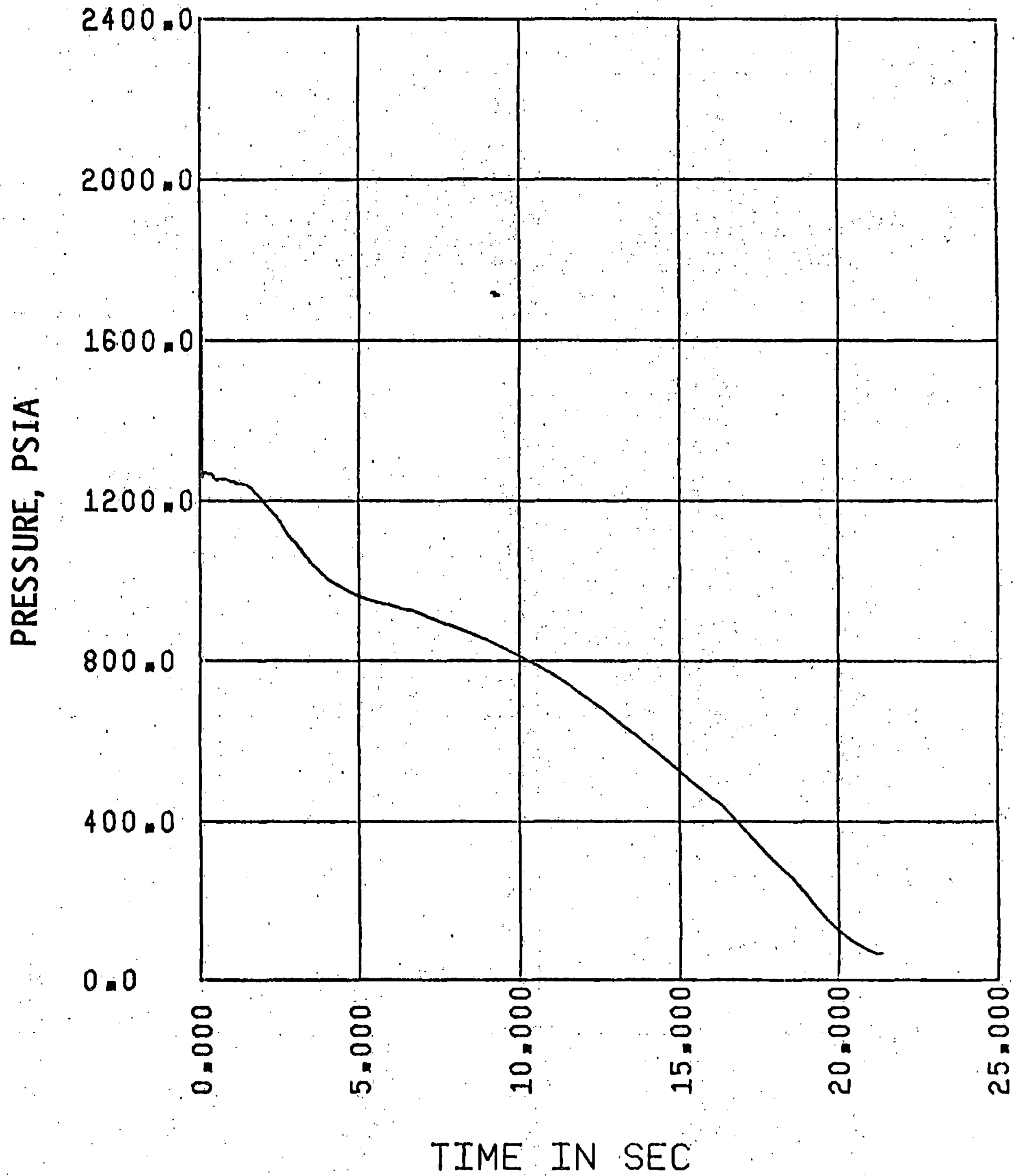


FIGURE II.4-C

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

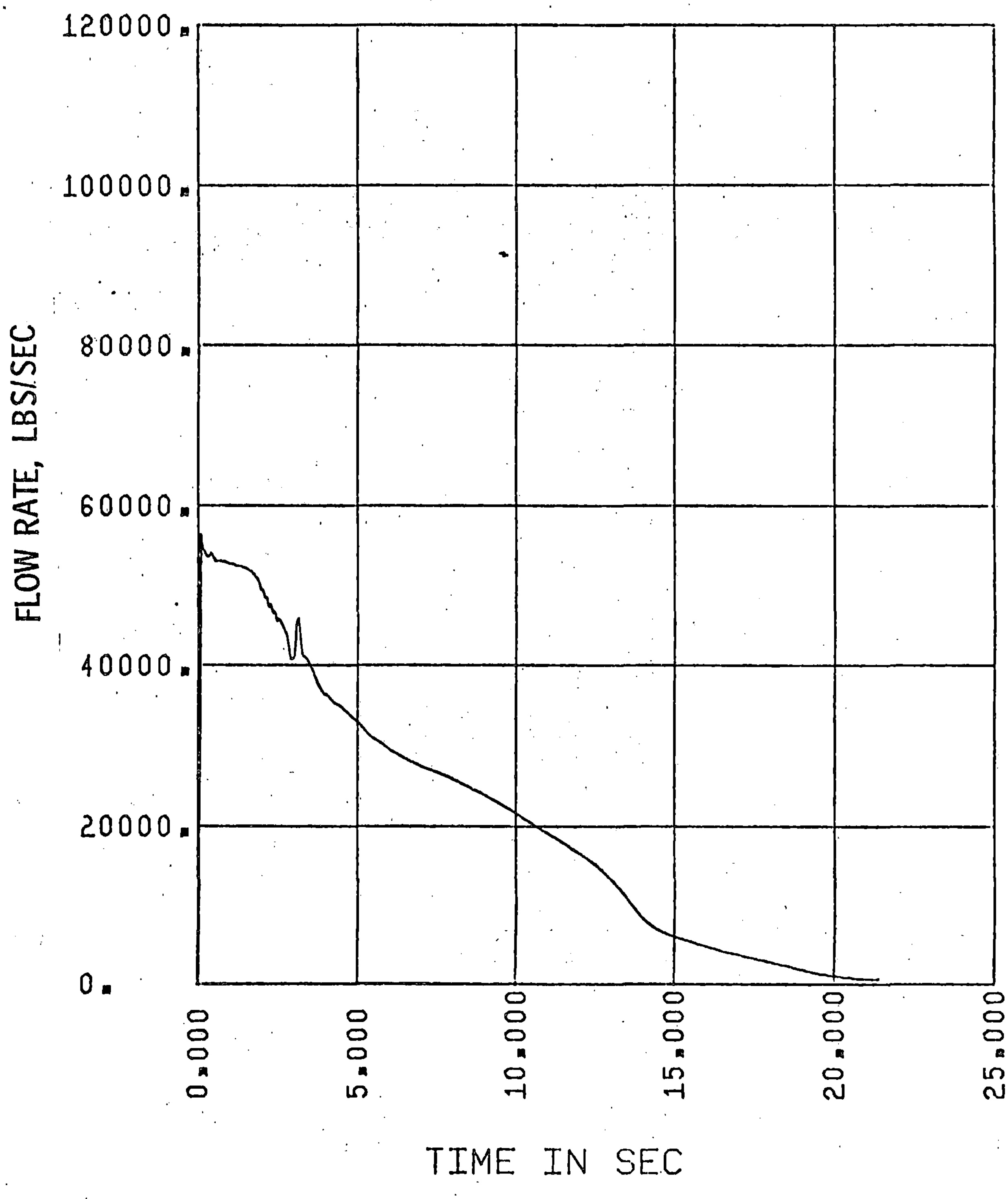


FIGURE II.4-D.1

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

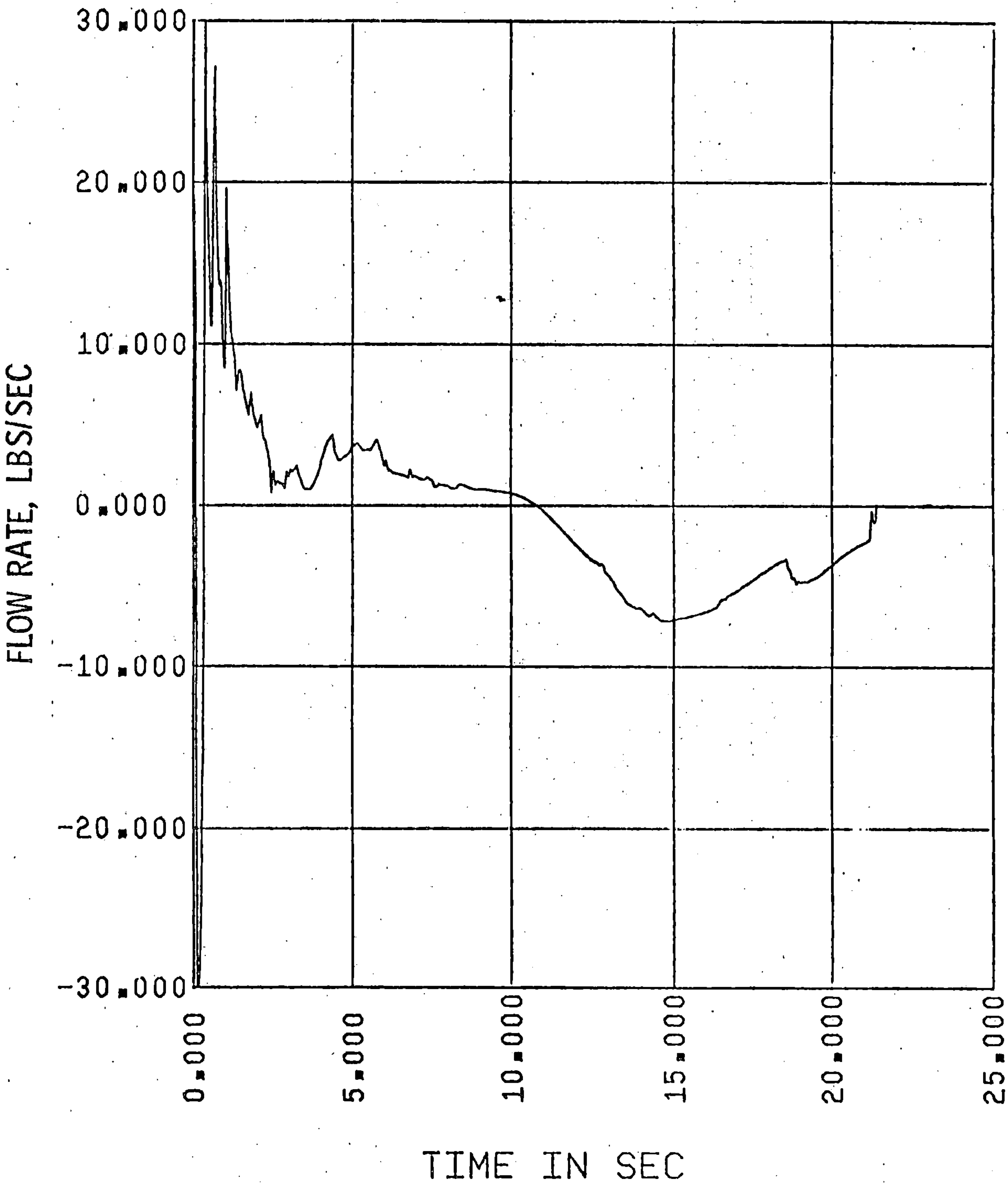


FIGURE II.4-D.2

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

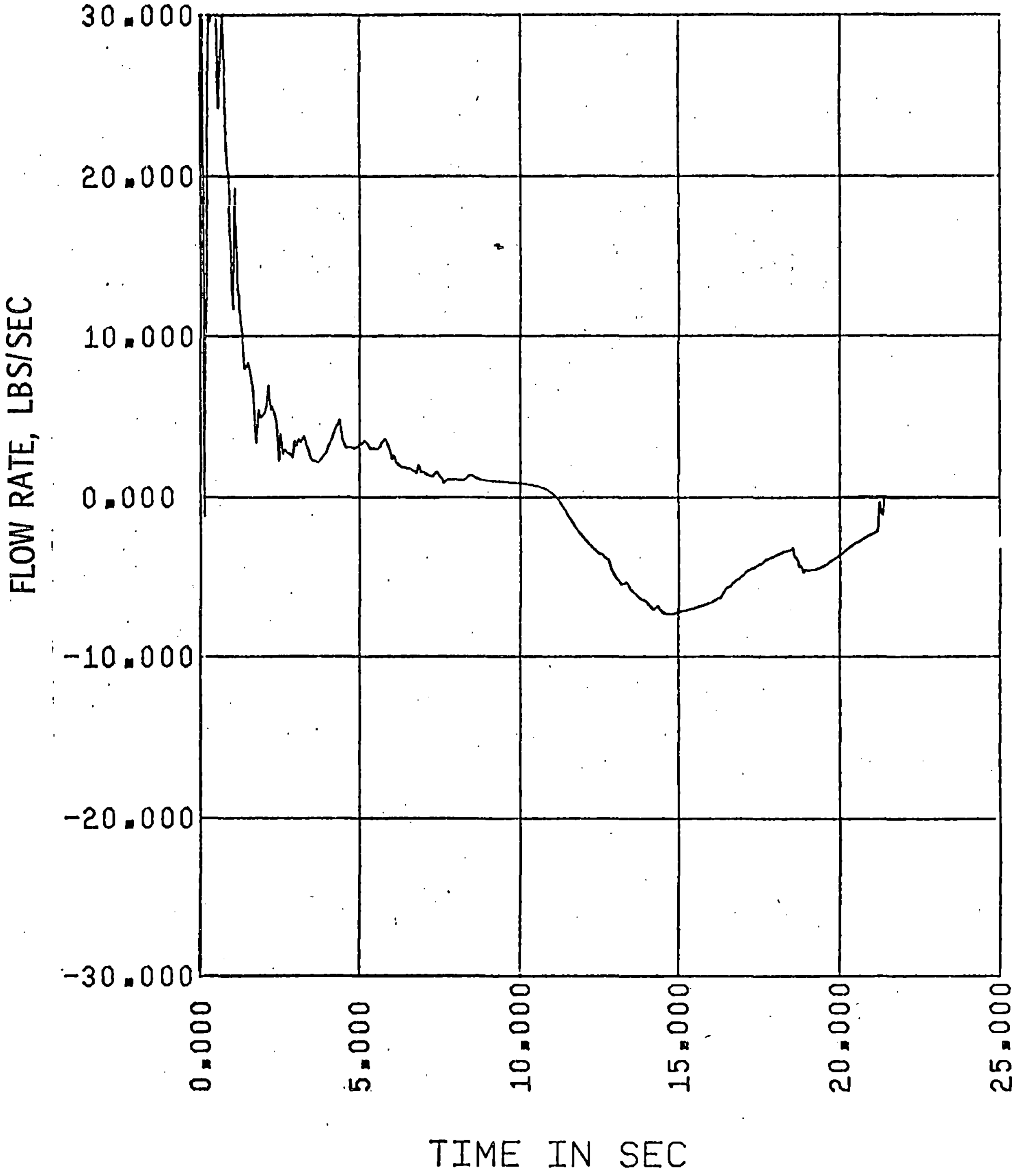


FIGURE II.4-E

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

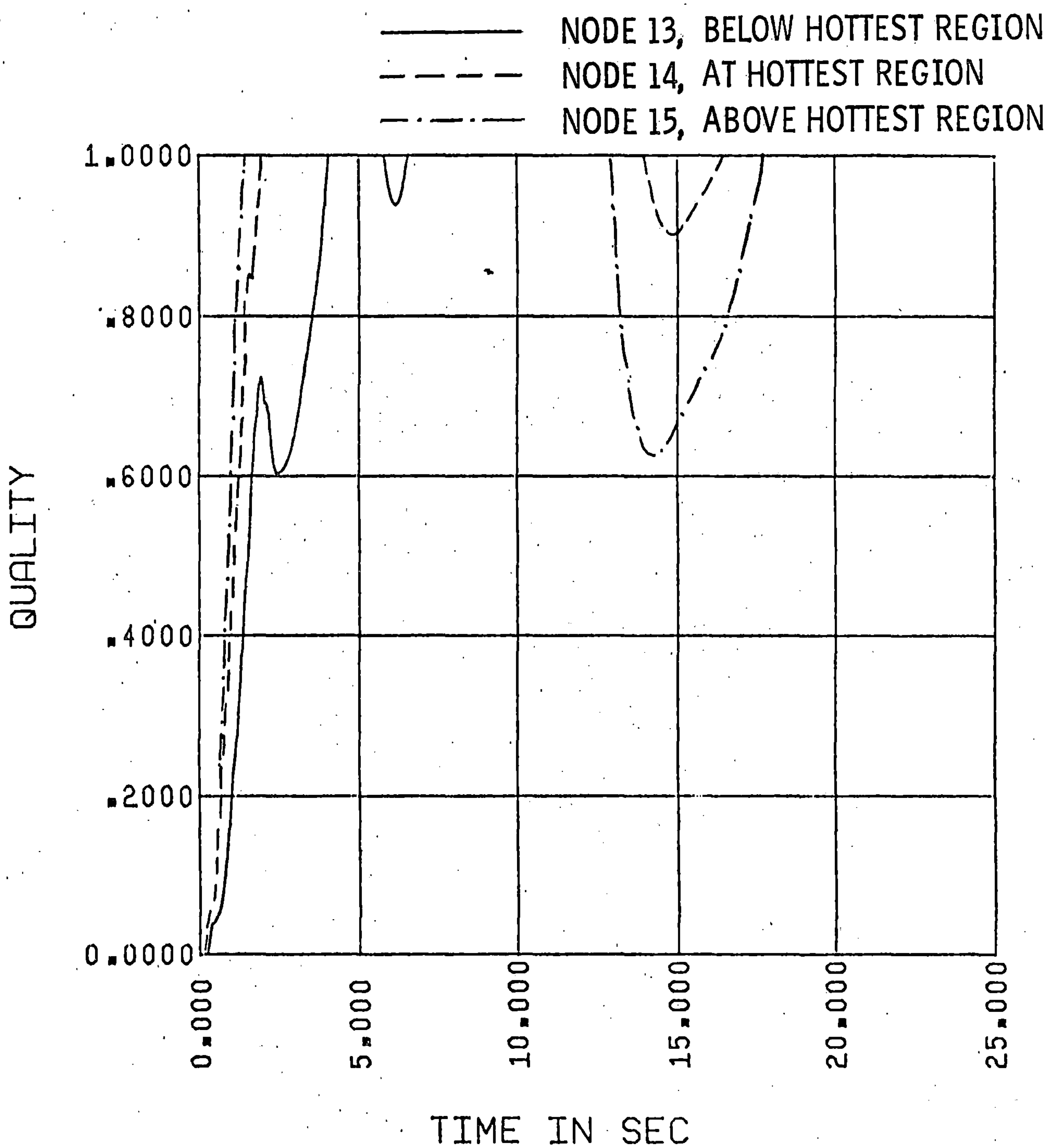


FIGURE II.4-F

PALISADES CORE I REANALYSIS

0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG

CONTAINMENT PRESSURE

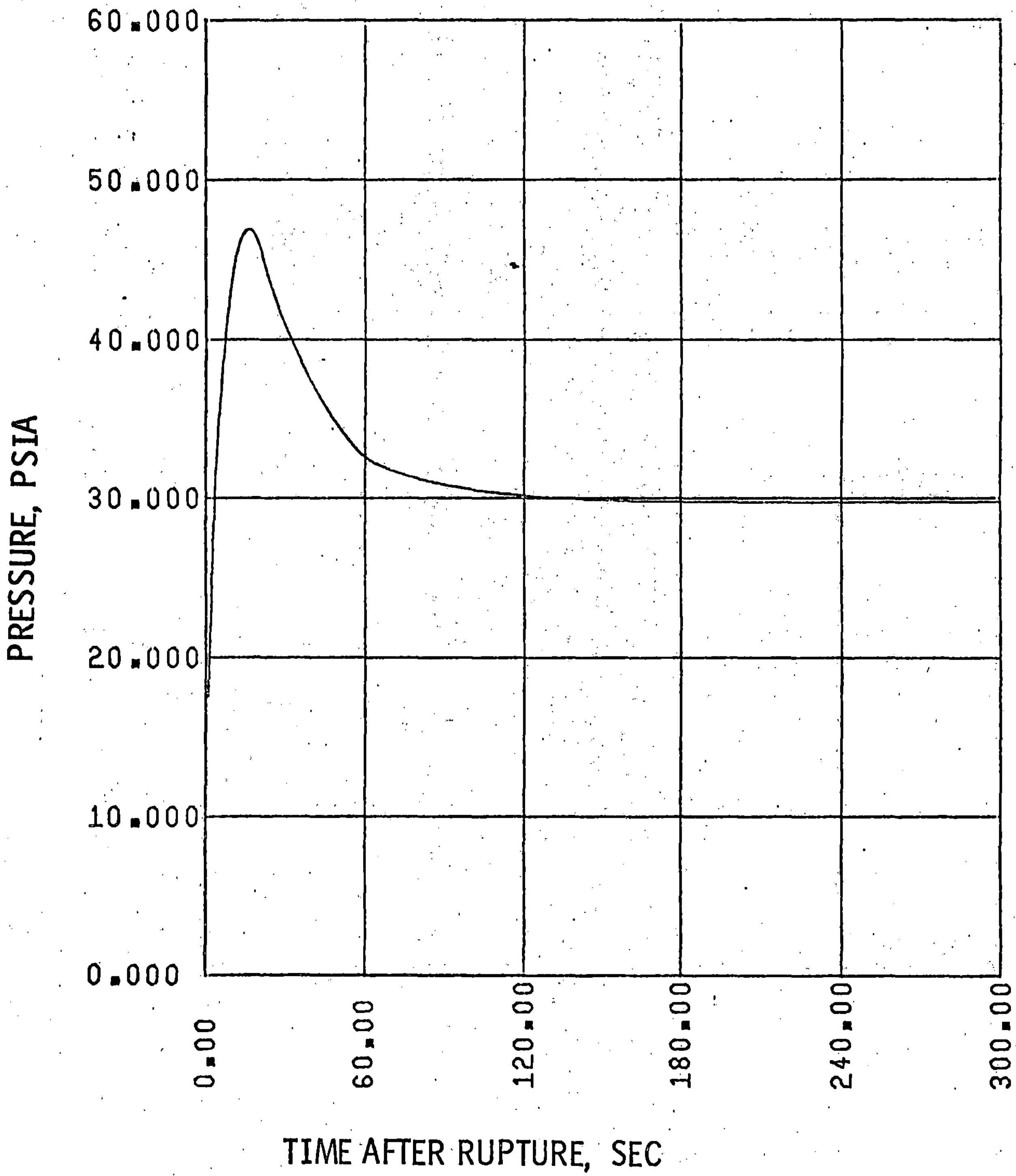


FIGURE II.4-G

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

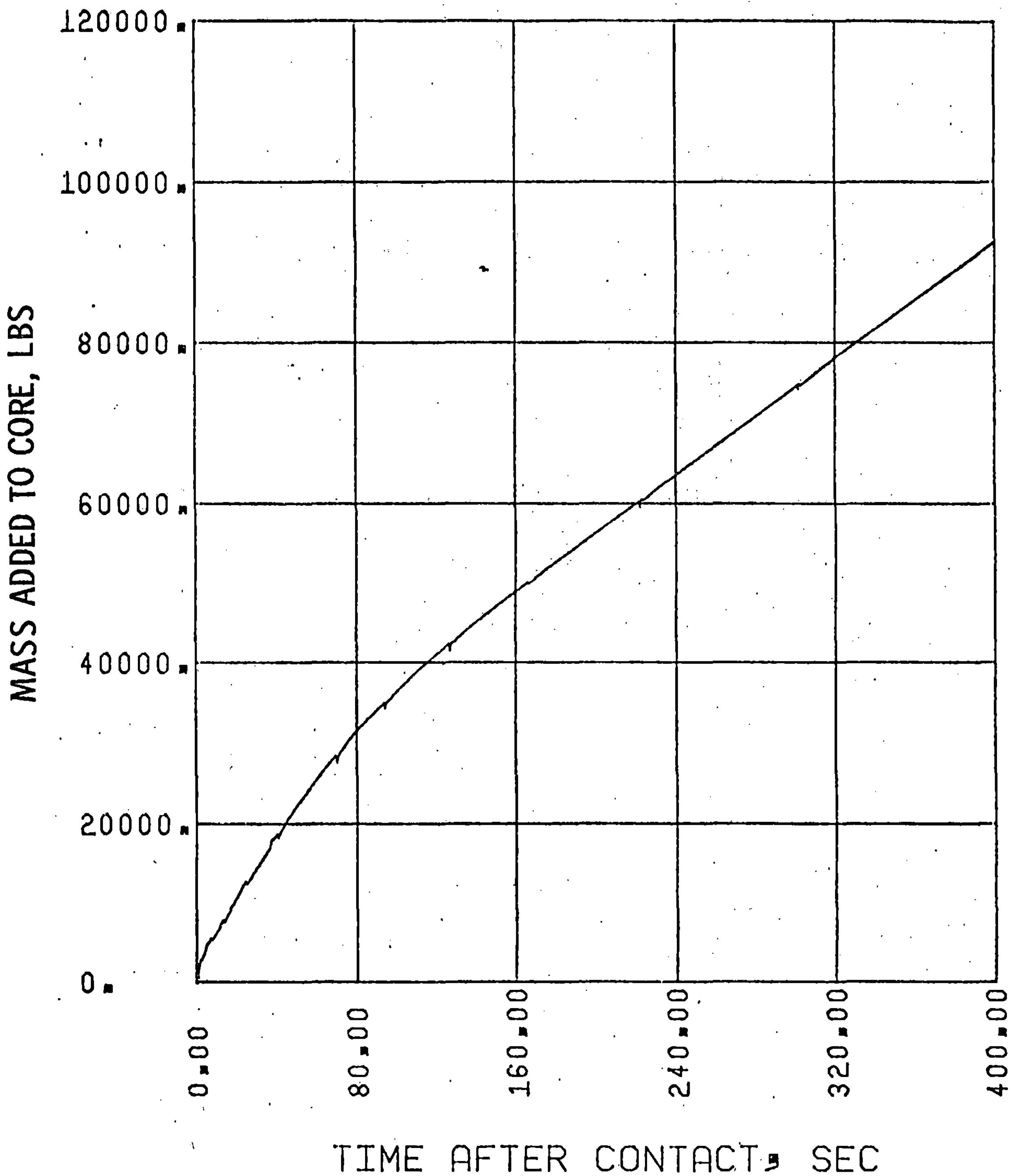


FIGURE II.4-H

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

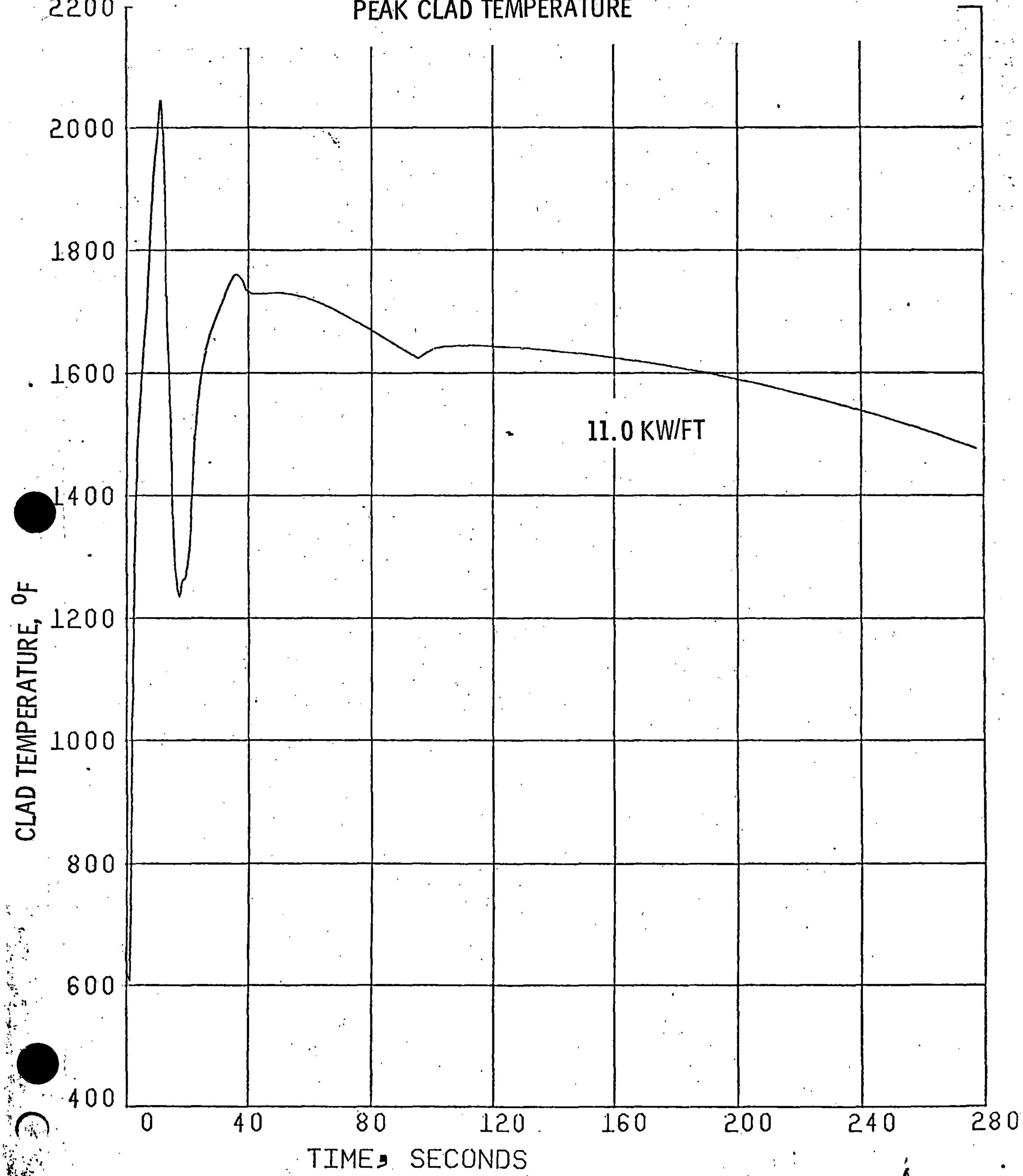


FIGURE II.5-A

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE POWER

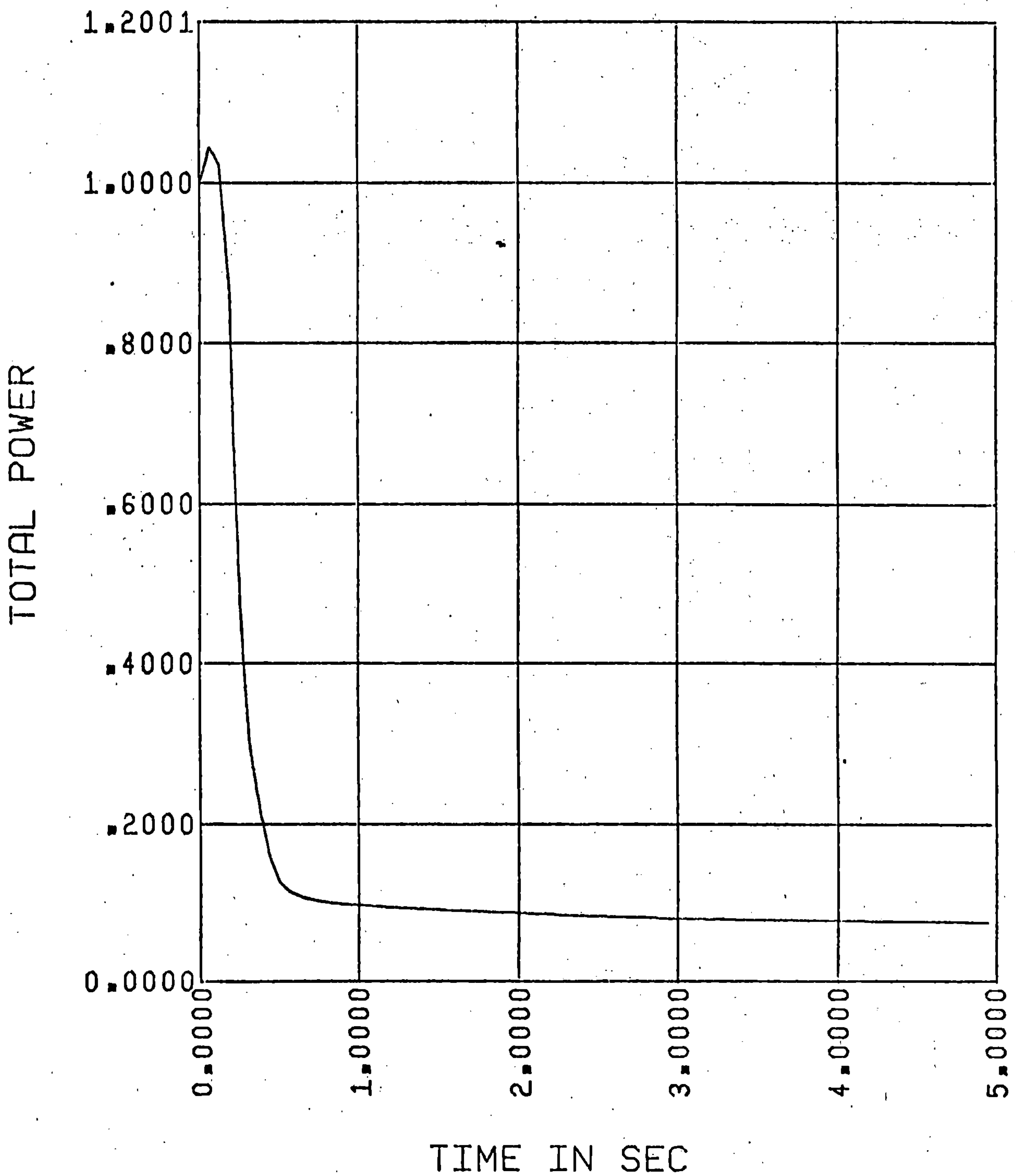


FIGURE II.5-B

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

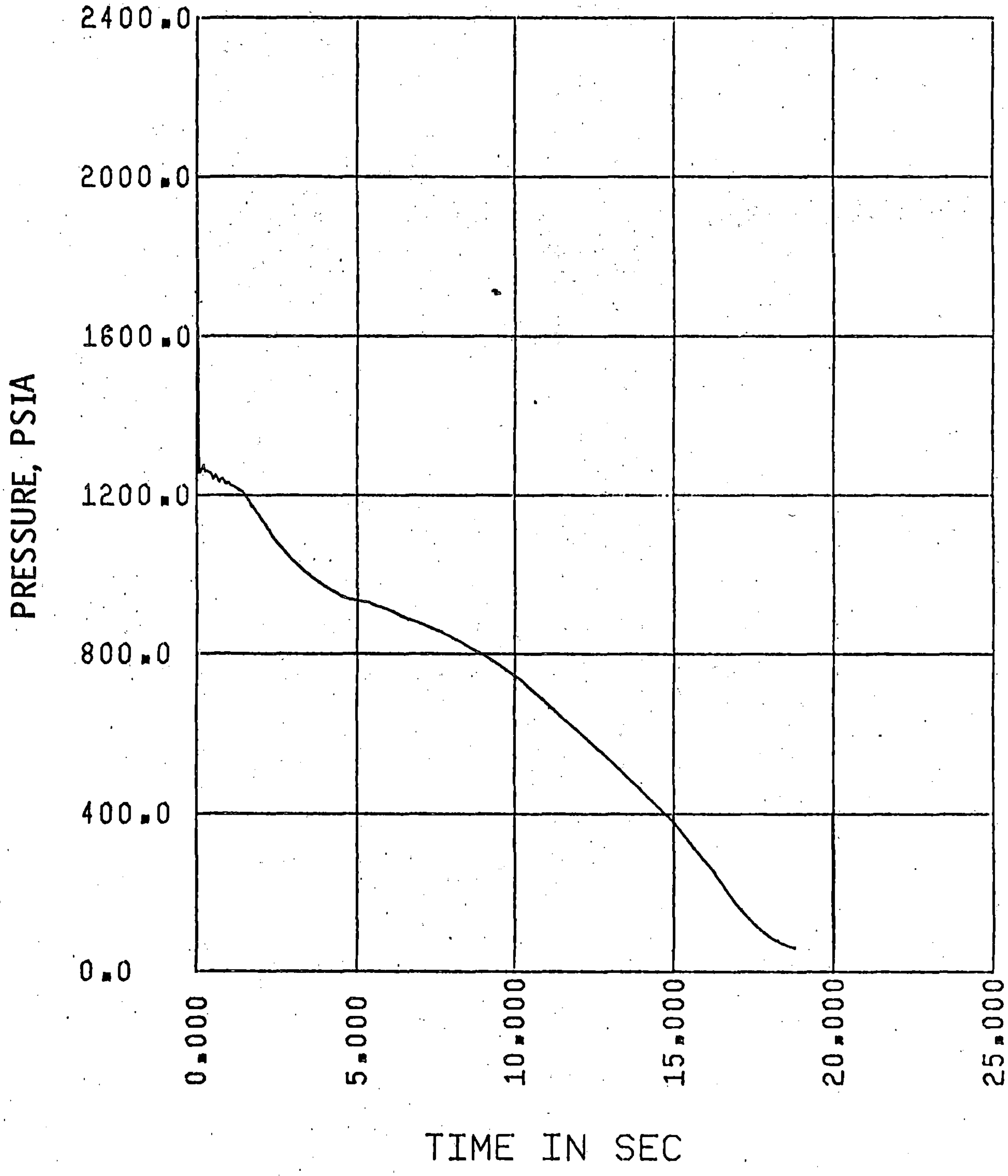


FIGURE II.5-C

PALISADES CORE I REANALYSIS

1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

LEAK FLOW

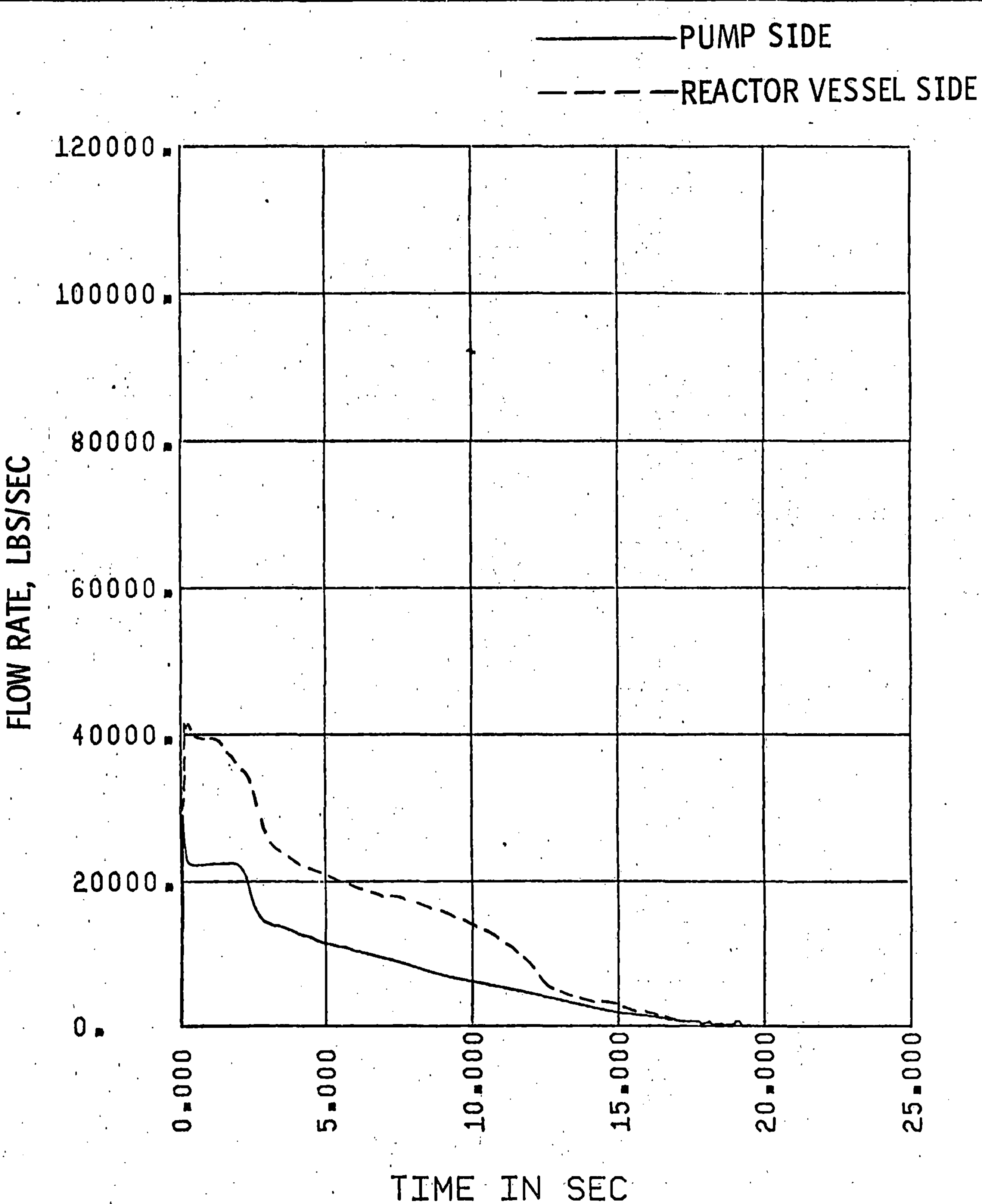


FIGURE II.5-D.1

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

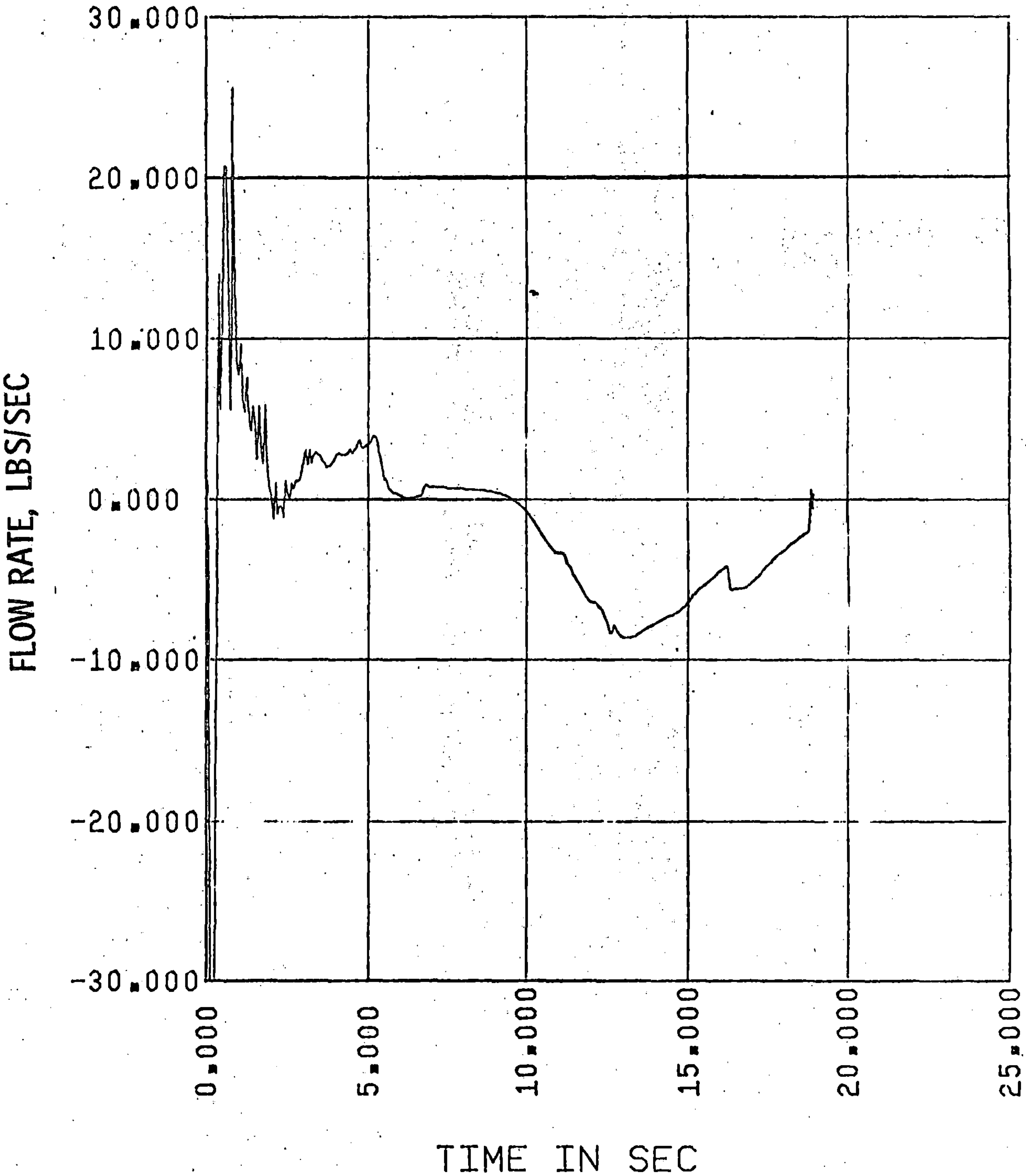


FIGURE II.5-D.2

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

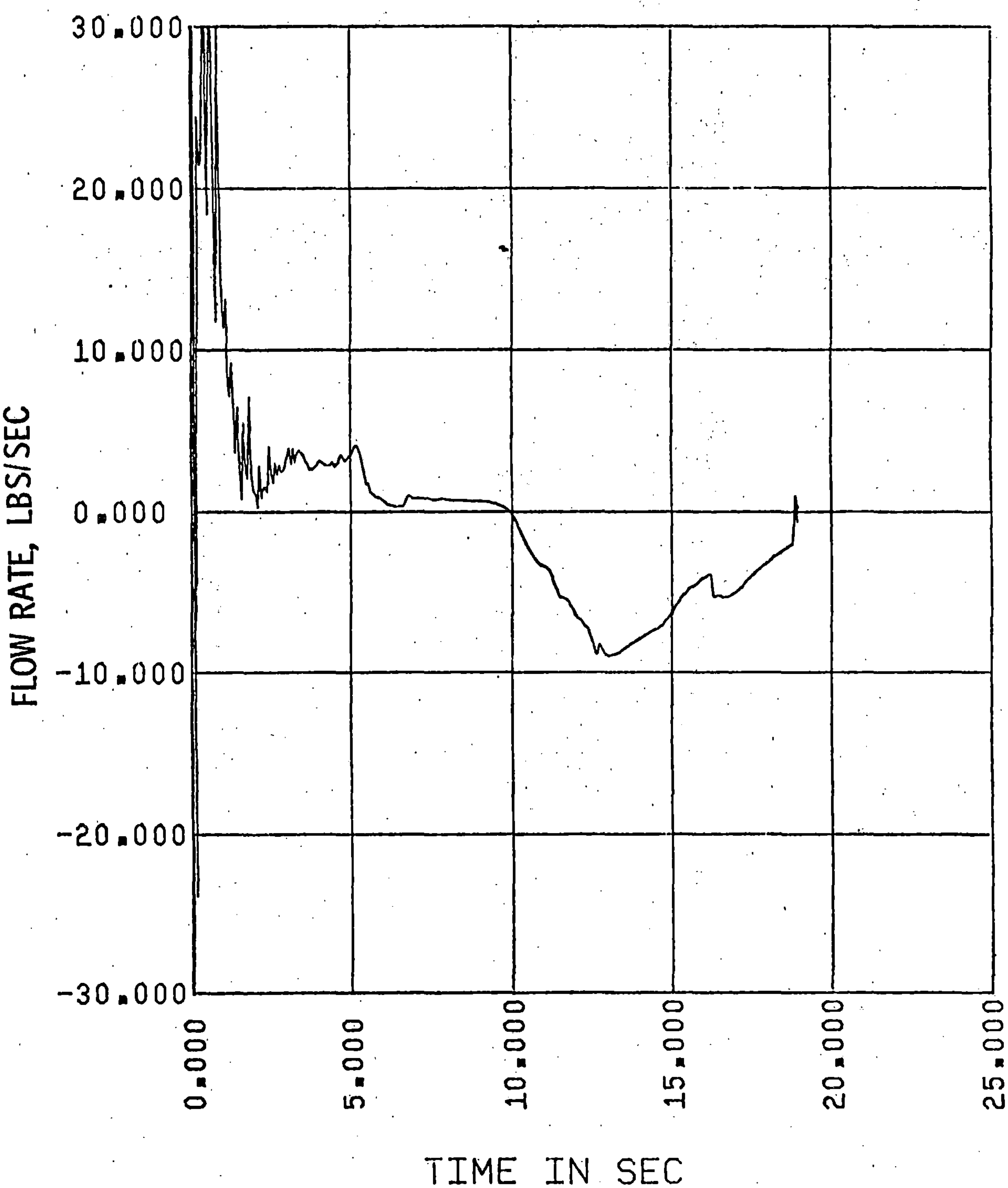


FIGURE II. 5-E

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

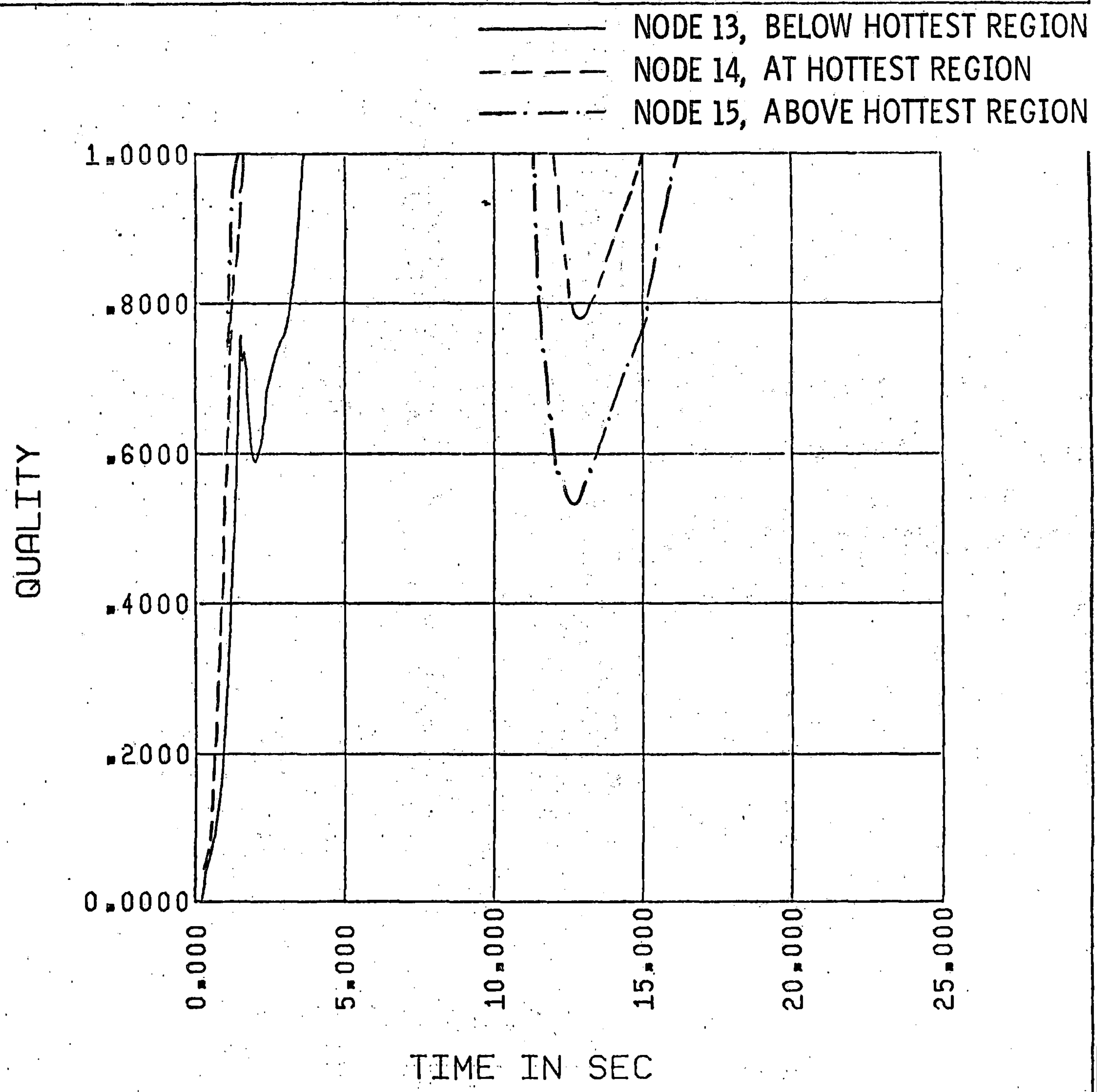


FIGURE II.5-F

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

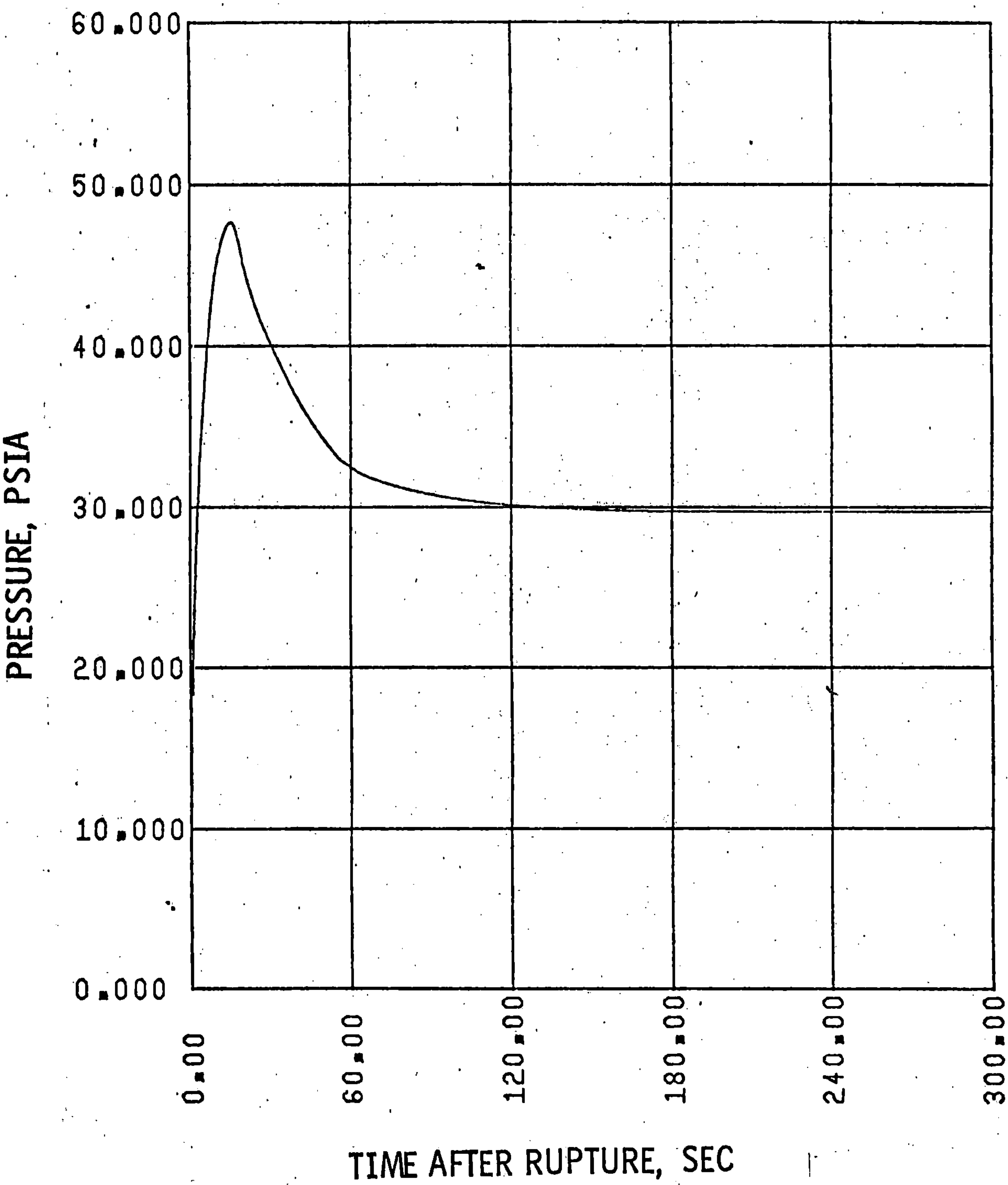


FIGURE II.5-G

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

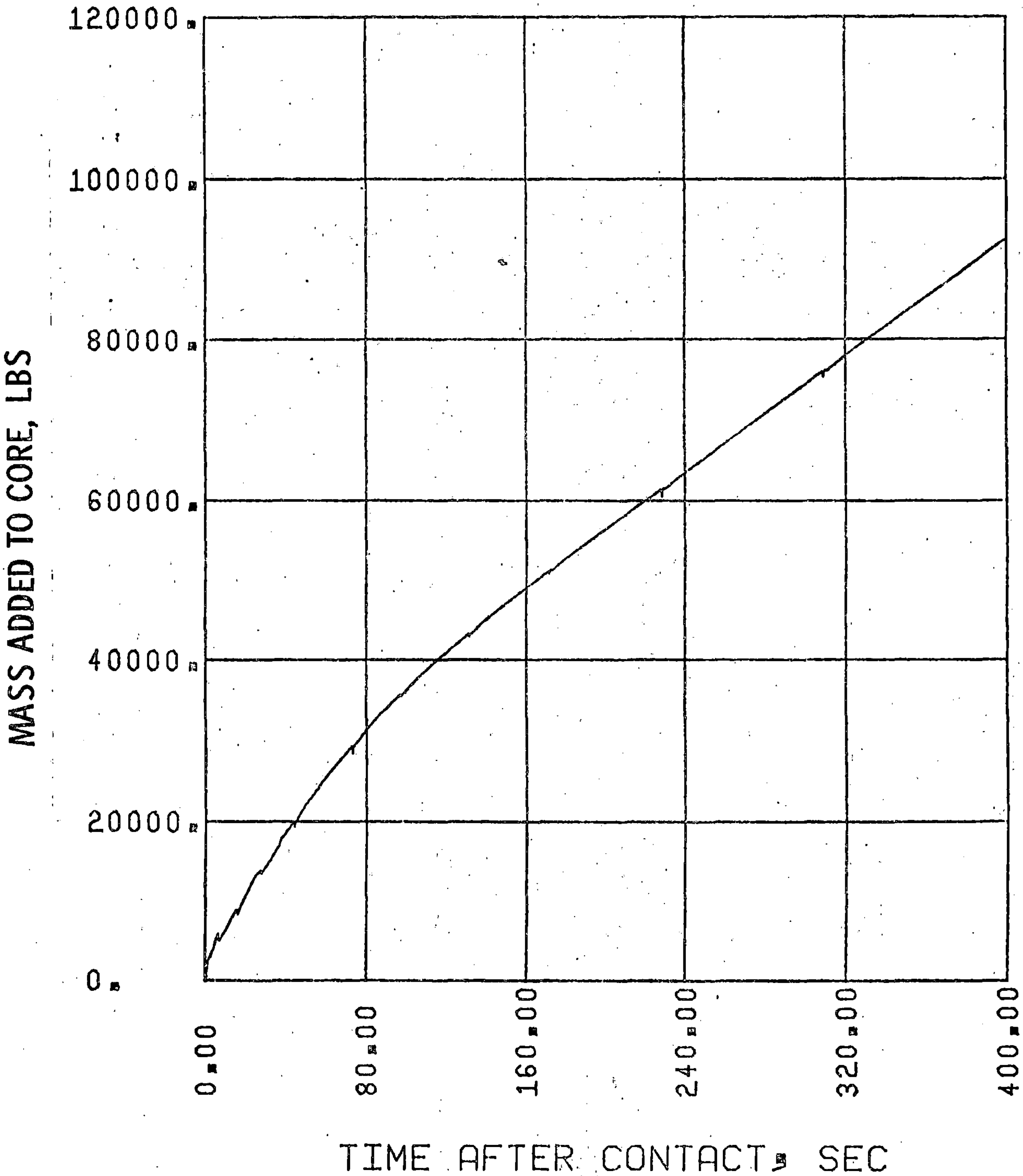


FIGURE II.5-H

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

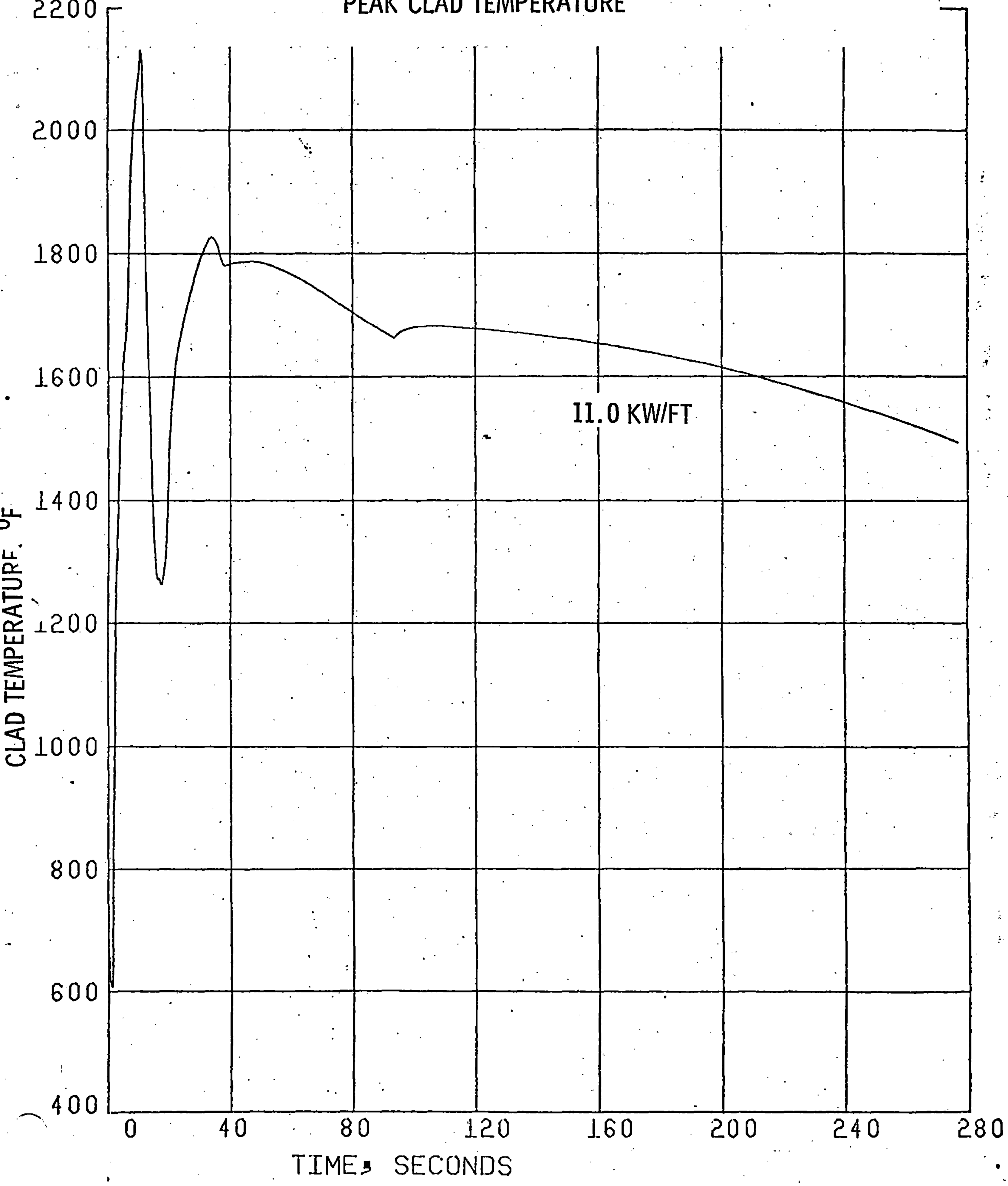


FIGURE II.6-A

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE POWER

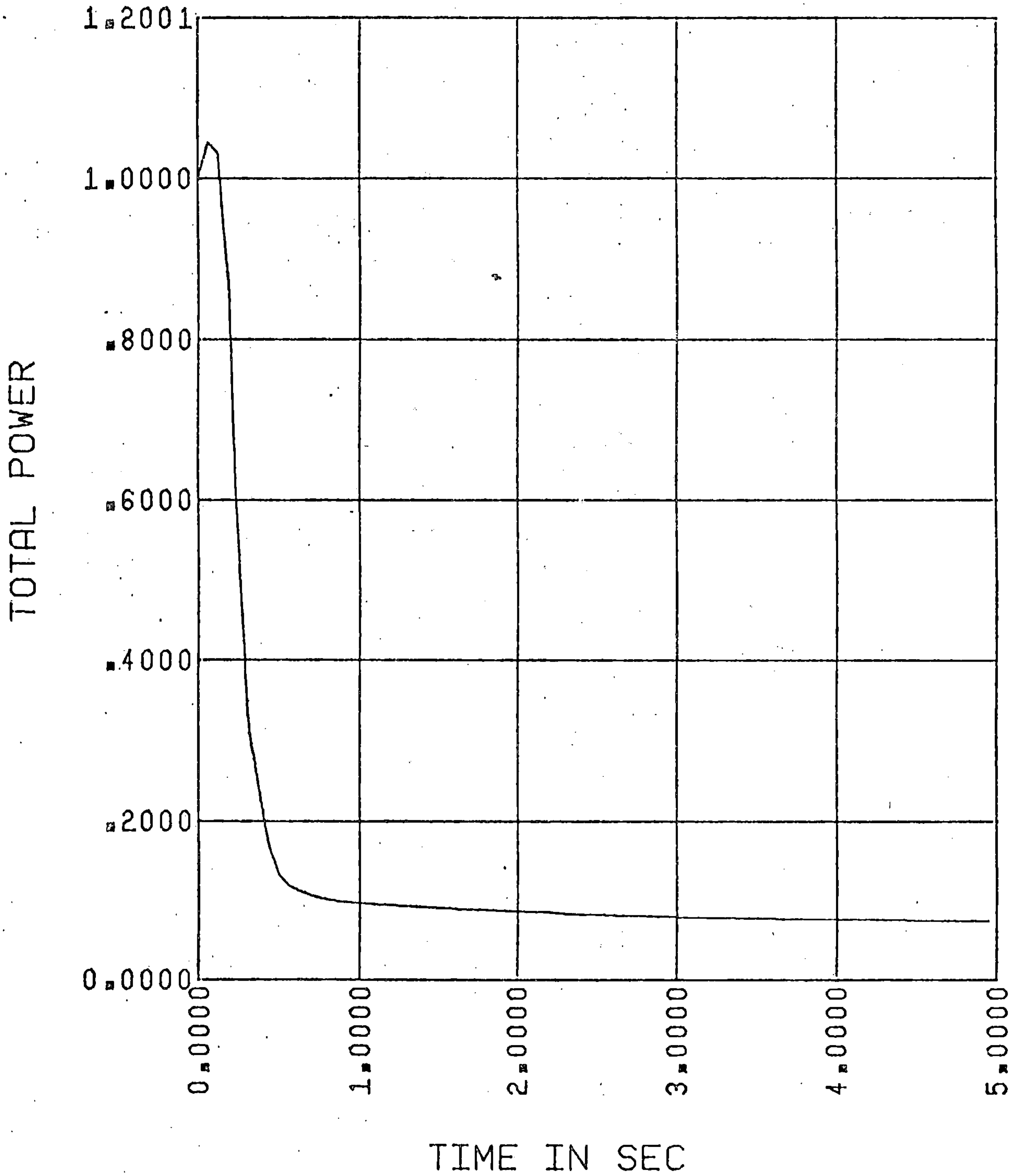


FIGURE II.6-B

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

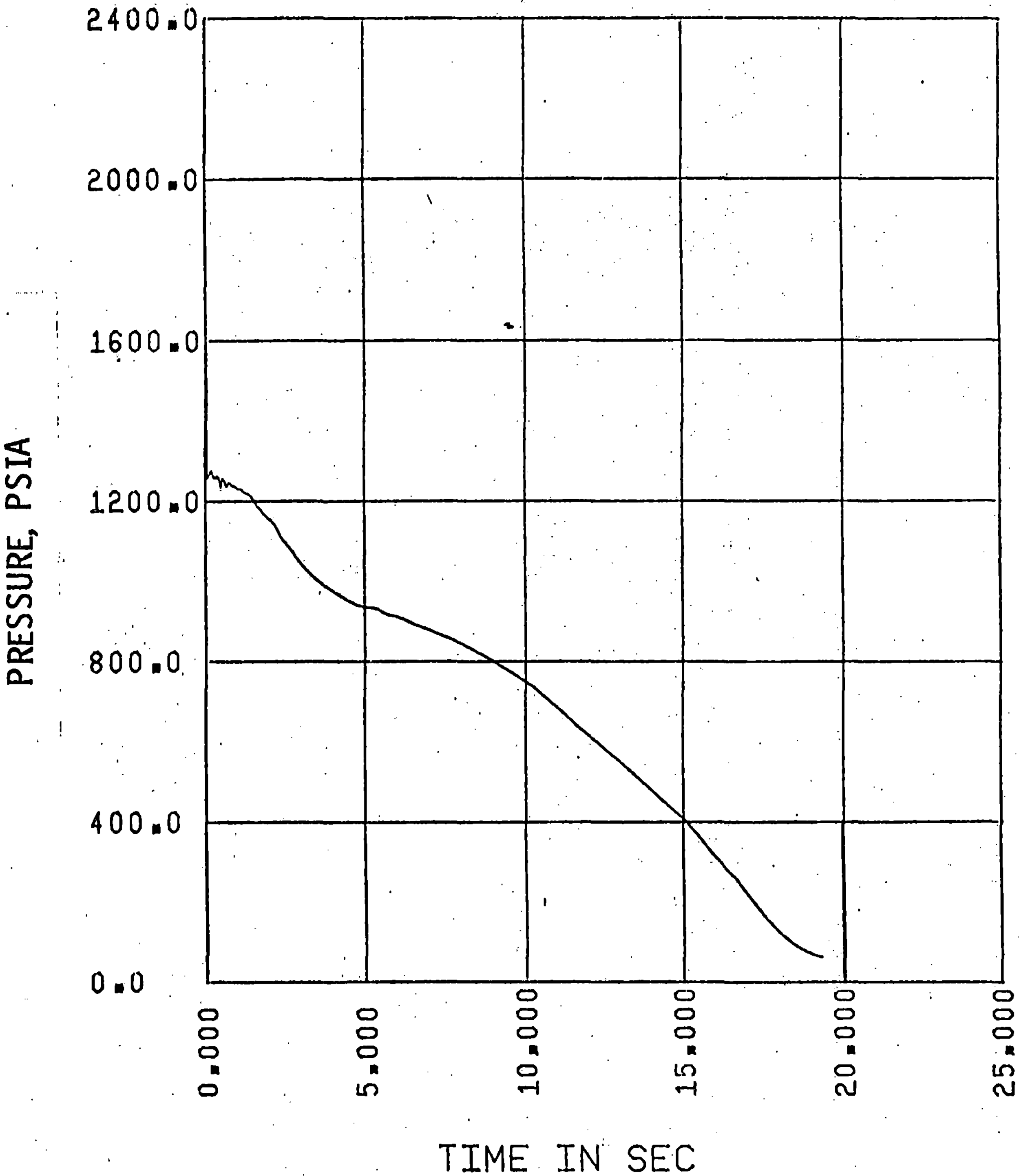


FIGURE II.6-C

PALISADES CORE I REANALYSIS

0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

LEAK FLOW

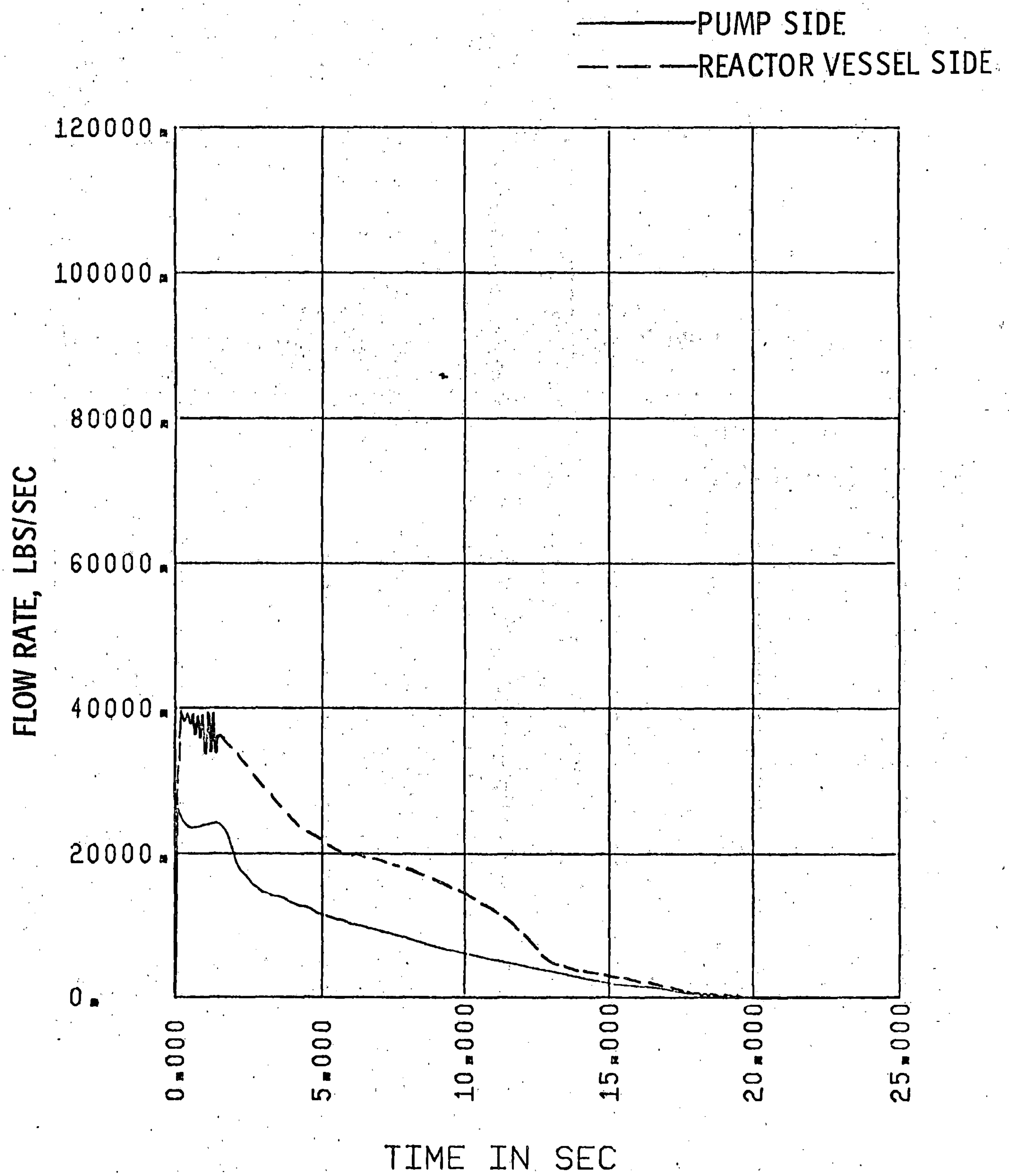


FIGURE II.6-D.1

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

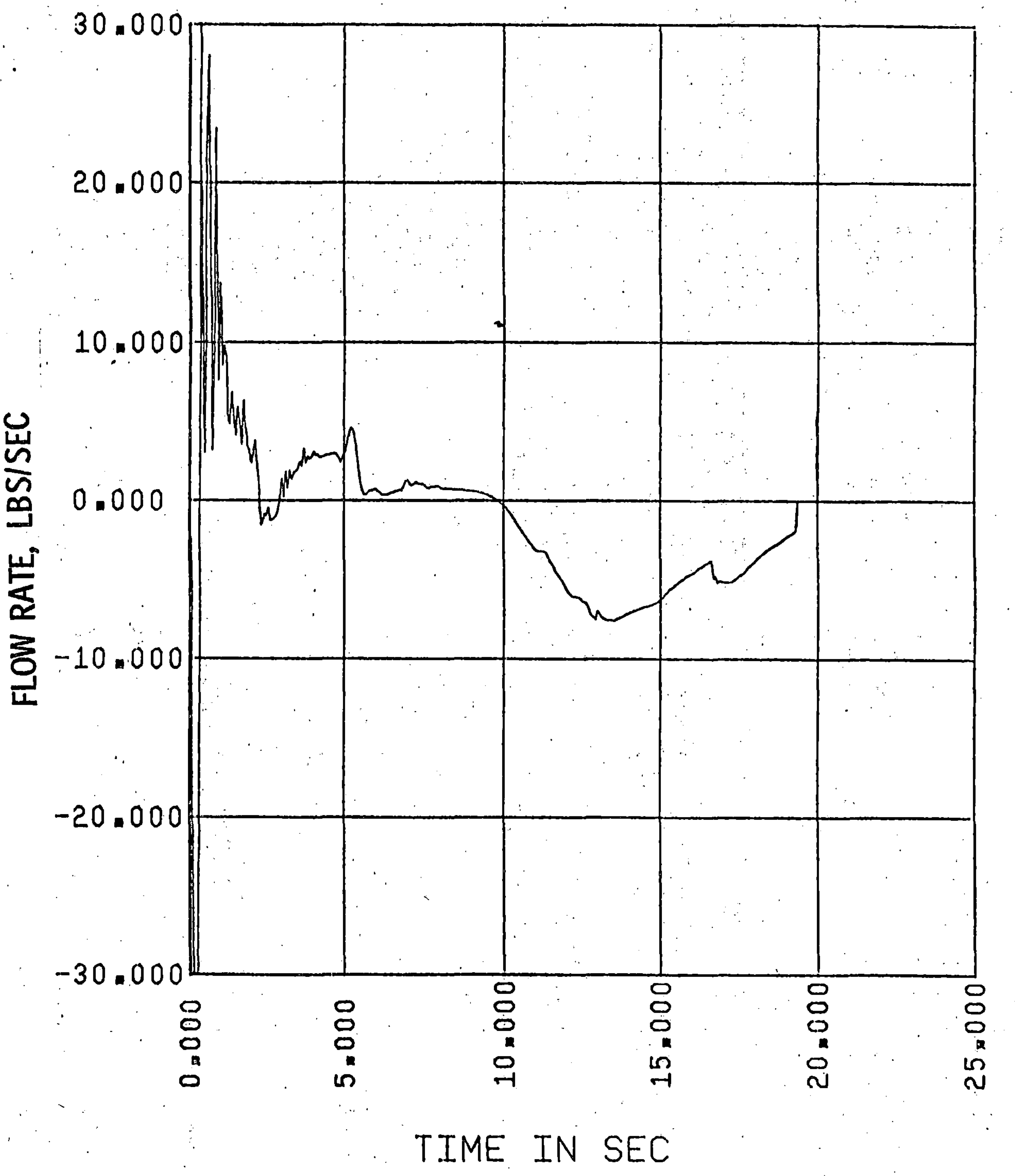


FIGURE II.6-D.2

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

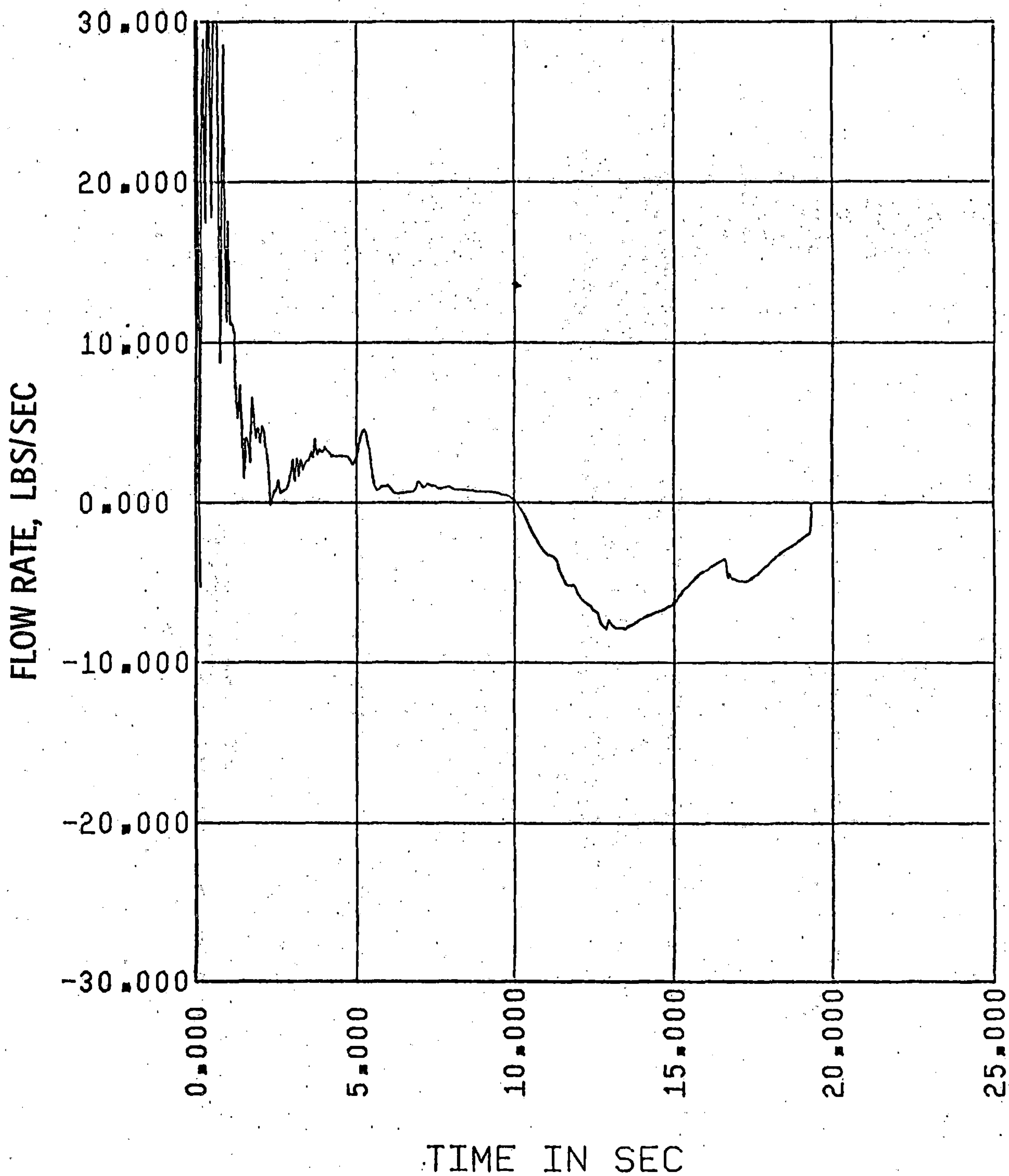


FIGURE II.6-E

PALISADES CORE I REANALYSIS

0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG

HOT ASSEMBLY QUALITY

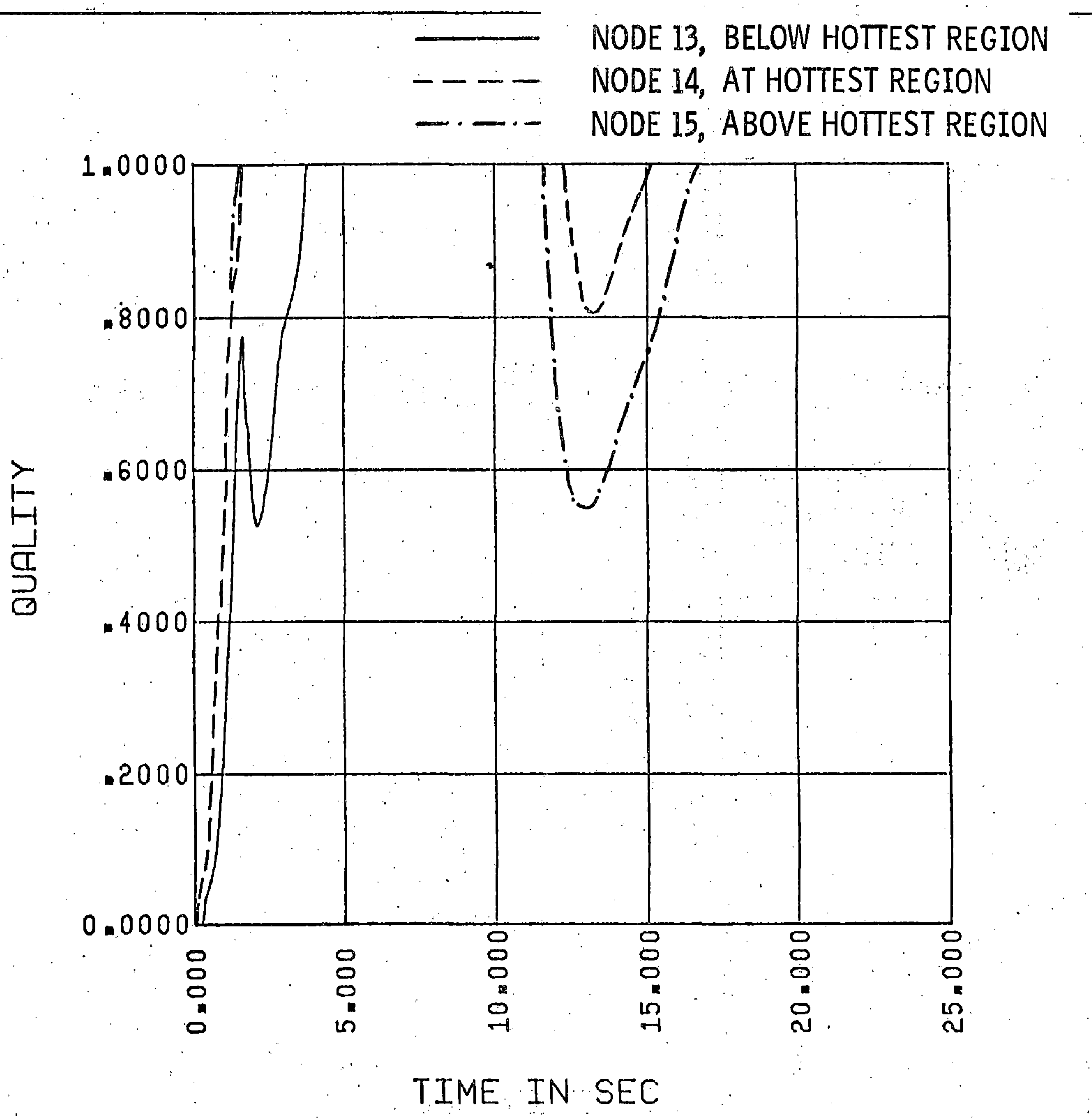


FIGURE II.6-F

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

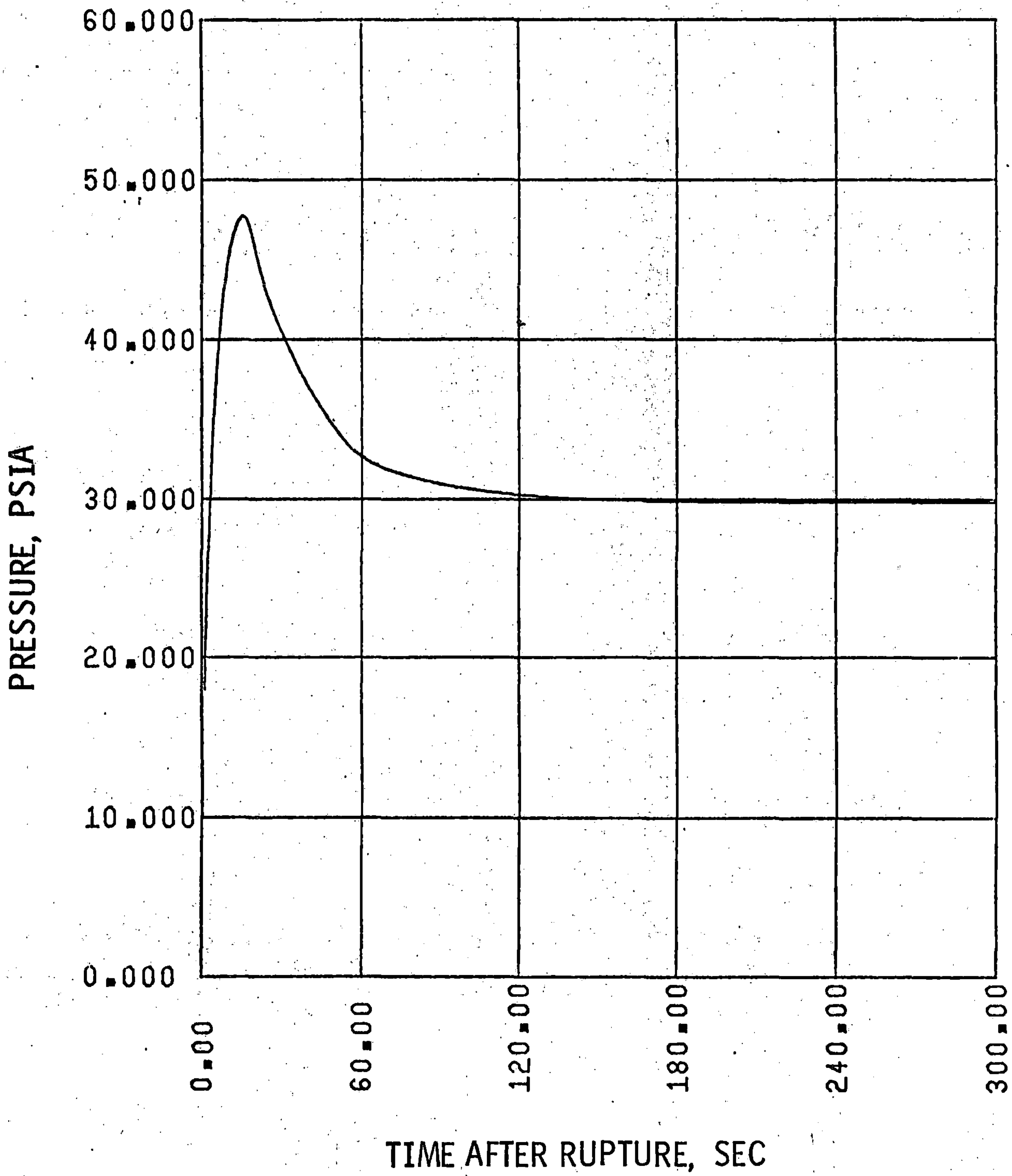


FIGURE II.6-G

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

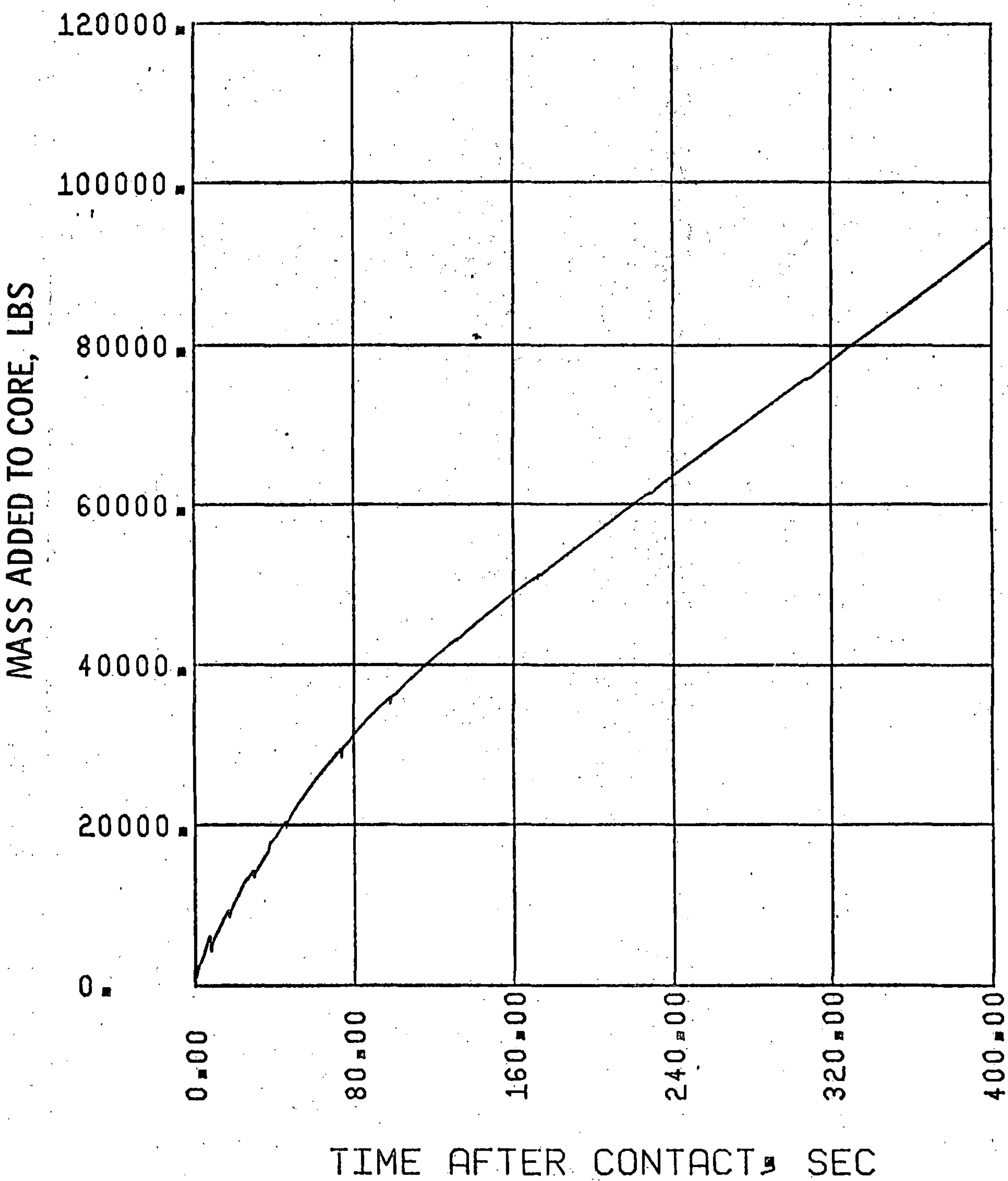


FIGURE II.6-H

PALISADES CORE I REANALYSIS
0.8 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

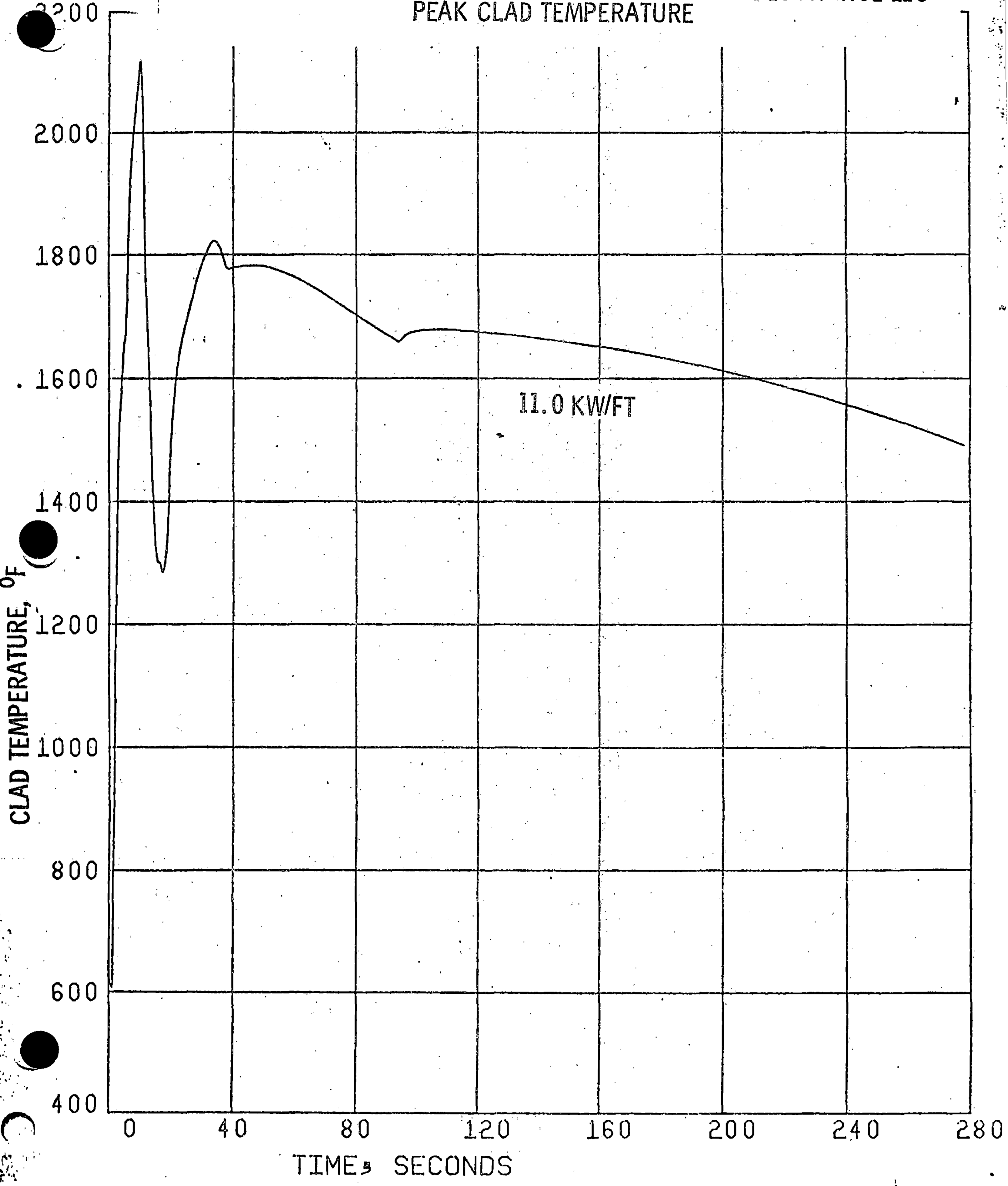


FIGURE II.7-A

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE POWER

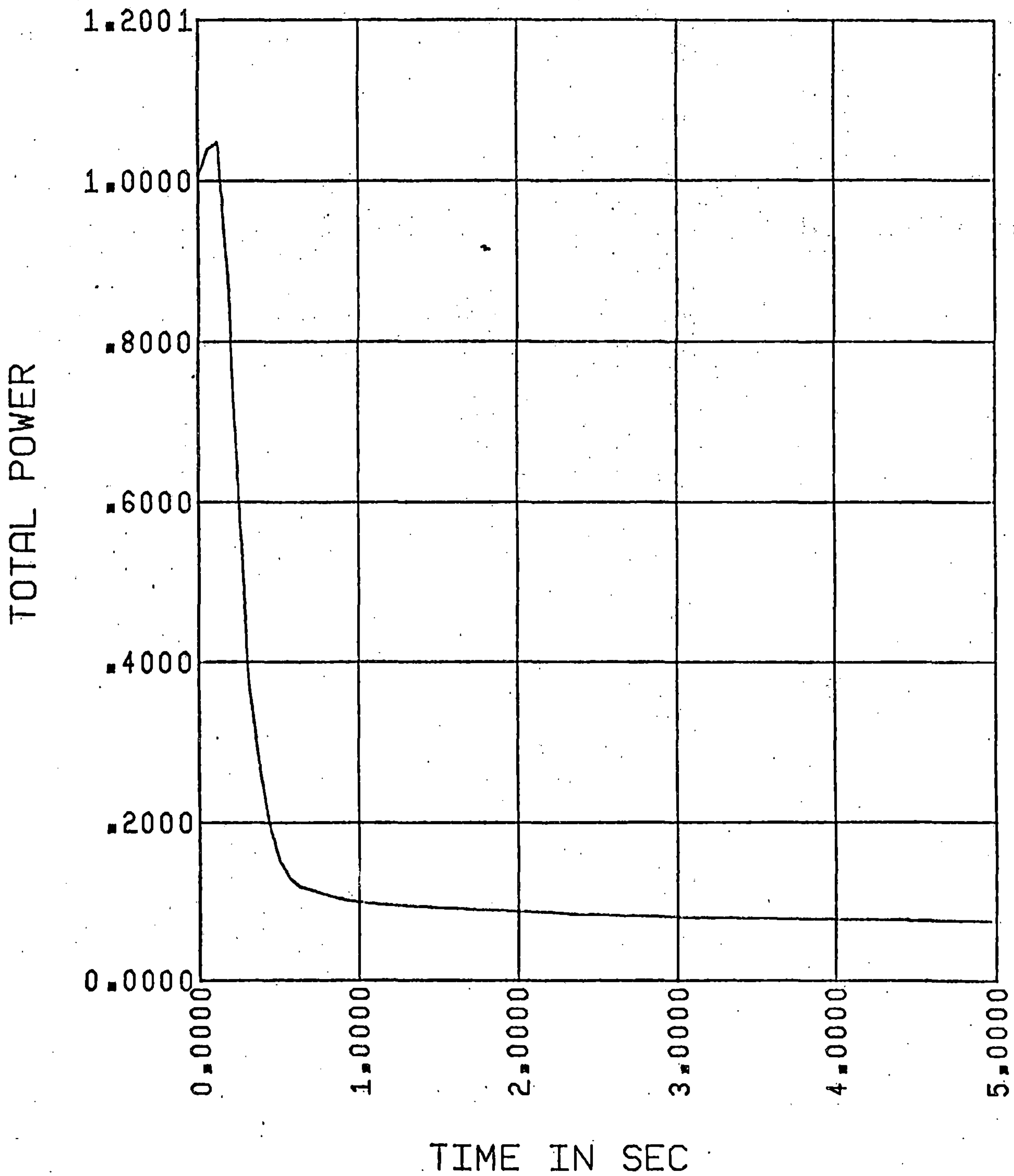


FIGURE II.7-B

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

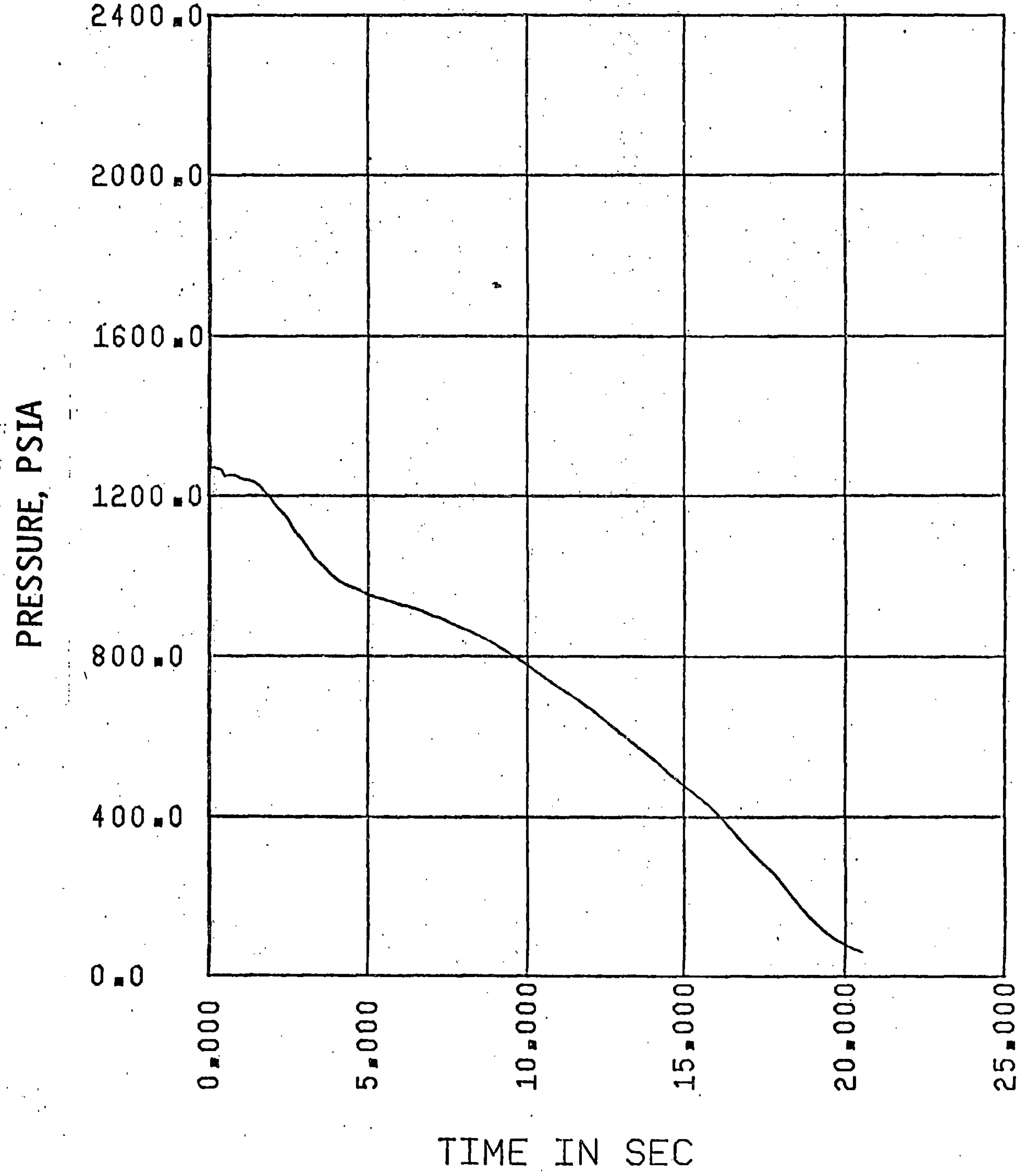


FIGURE II.7-C

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

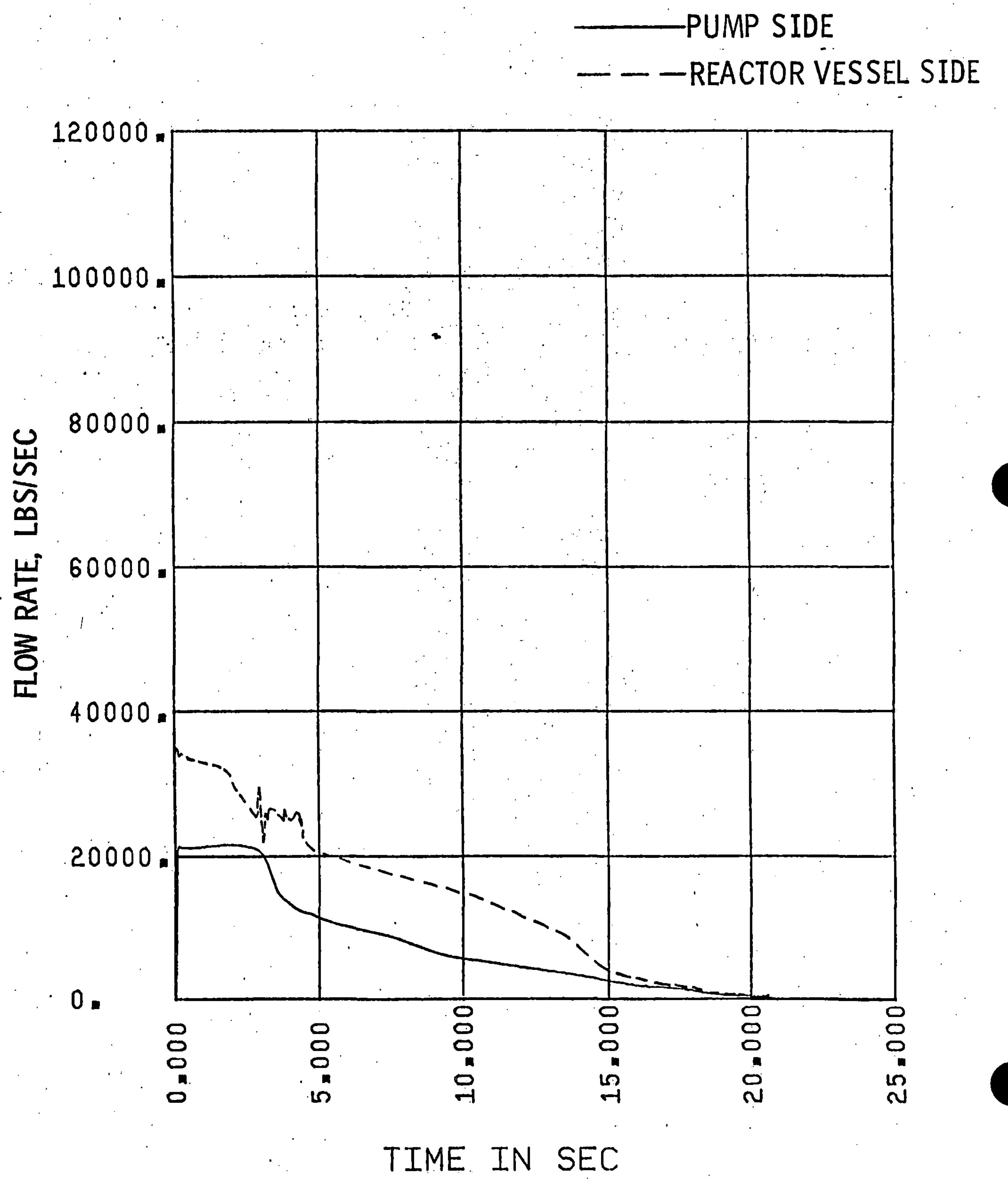


FIGURE II.7-D.1

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

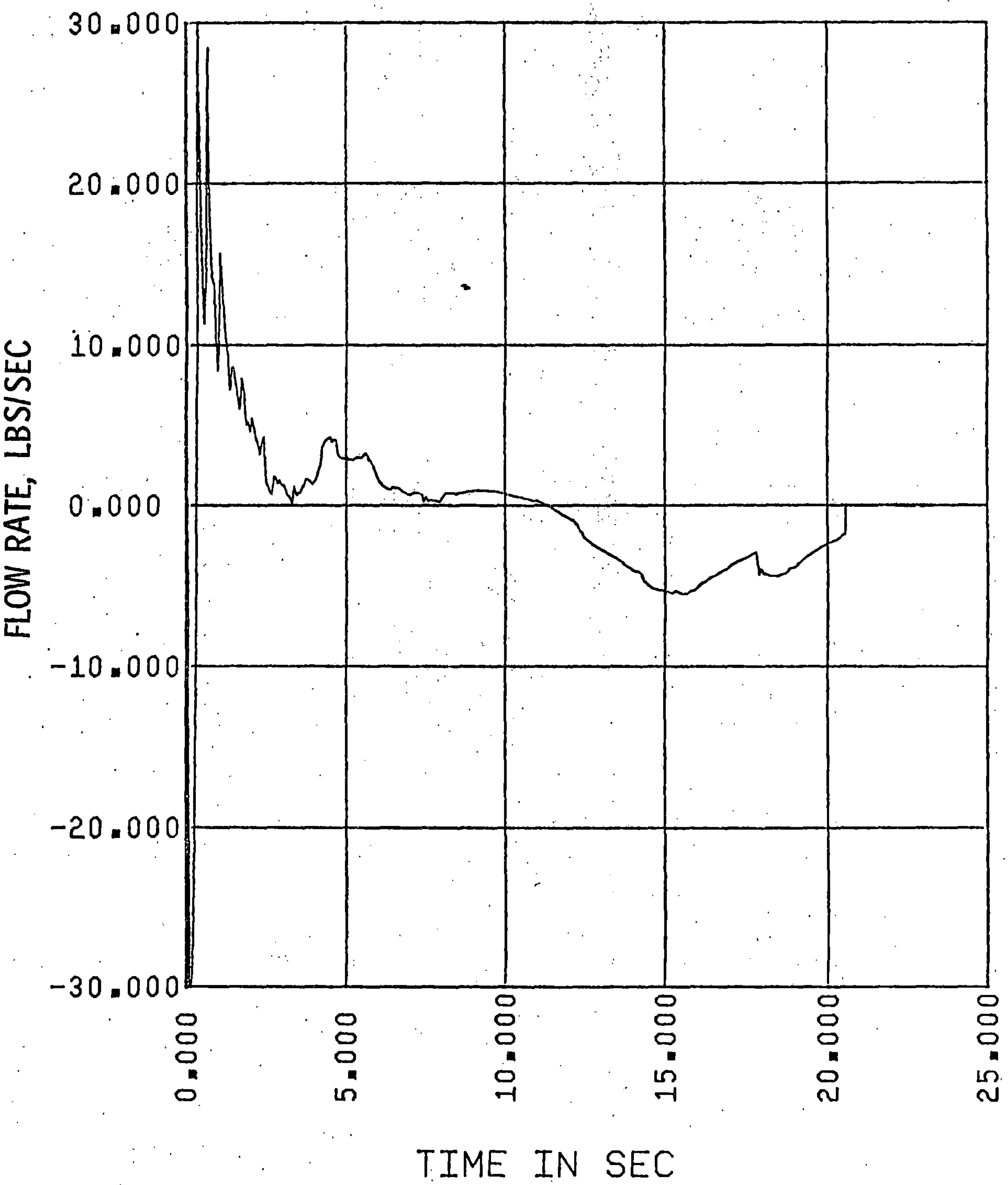


FIGURE II.7-D.2

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

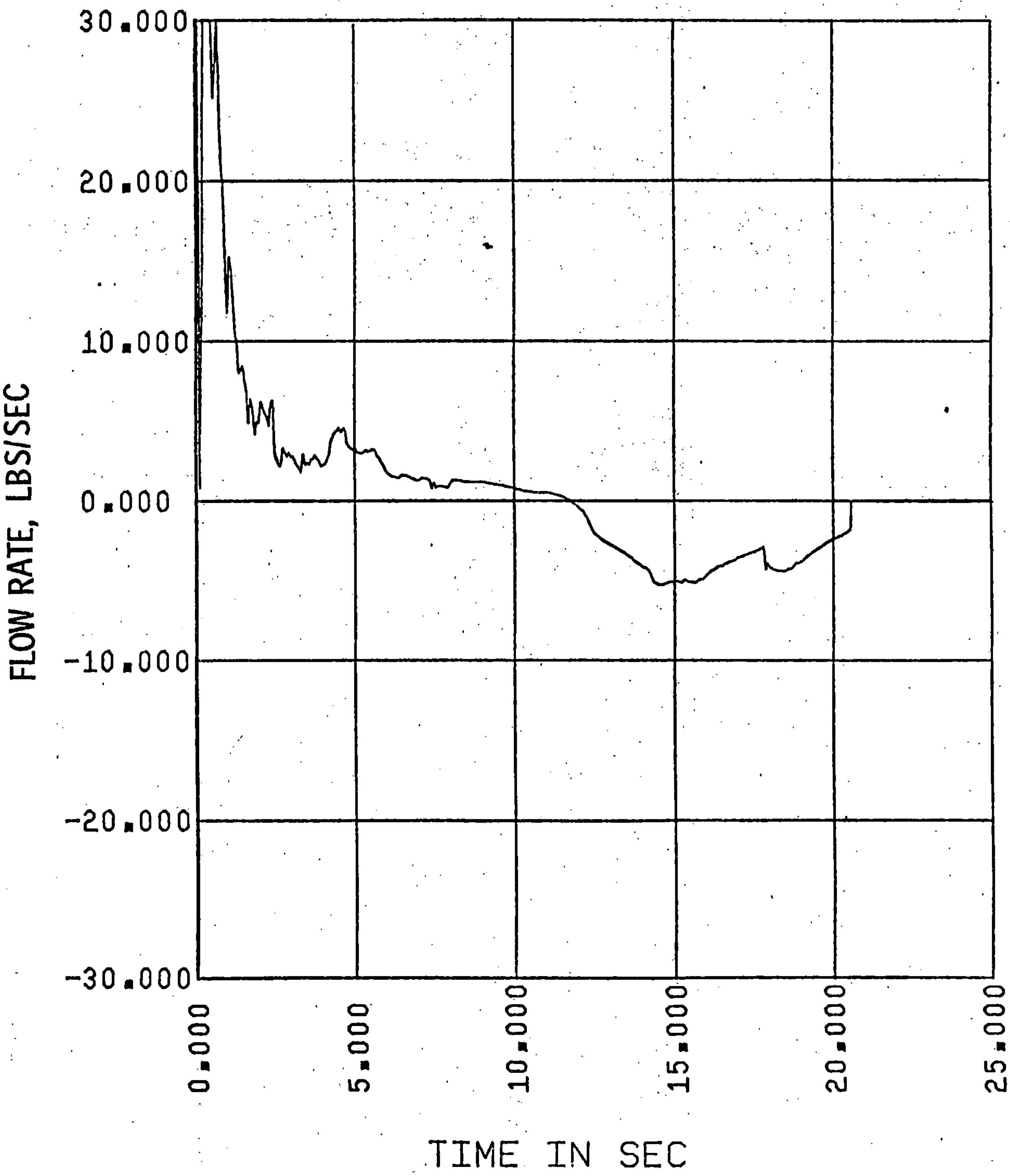


FIGURE II.7-E

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

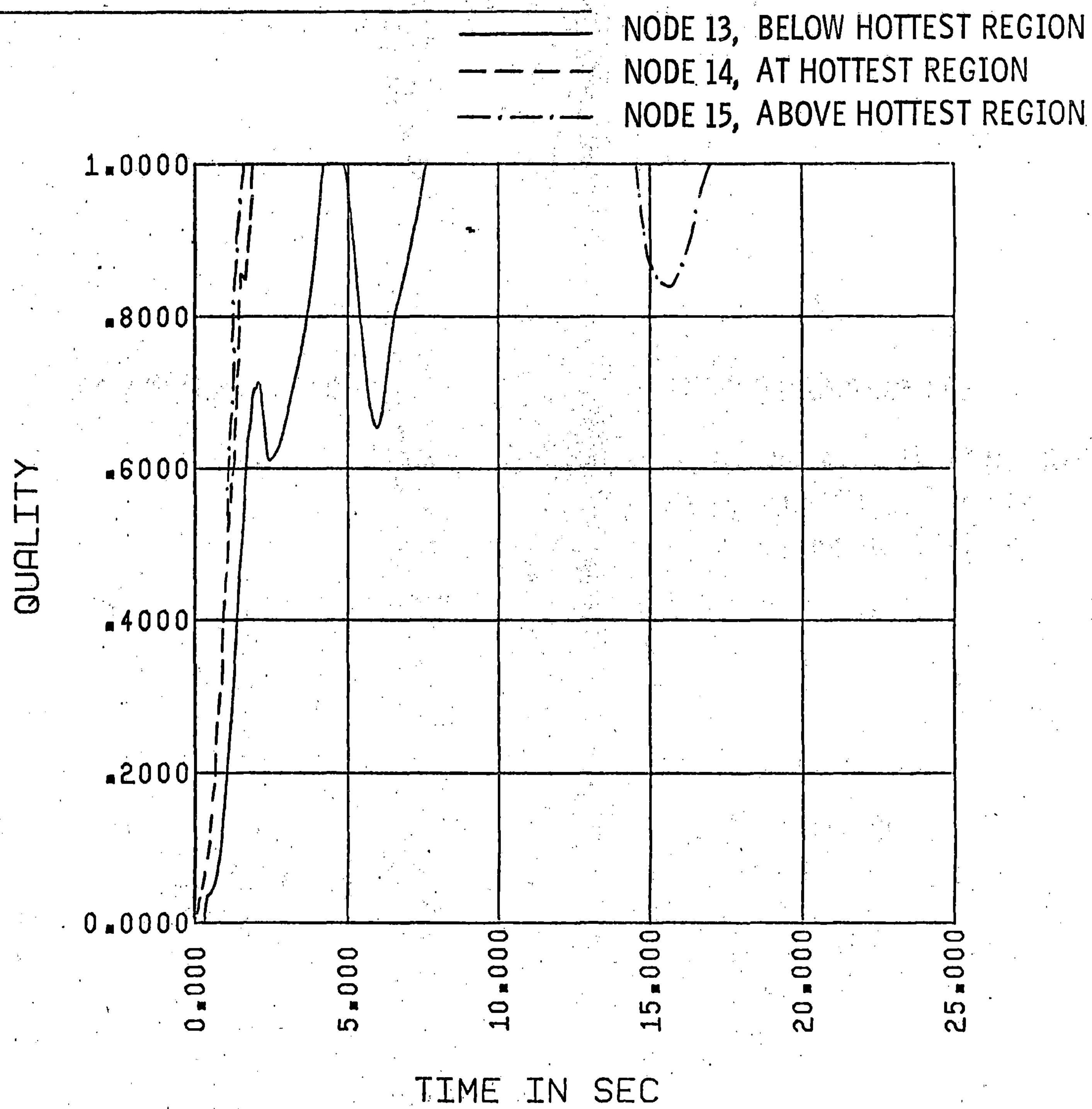


FIGURE II.7-F

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

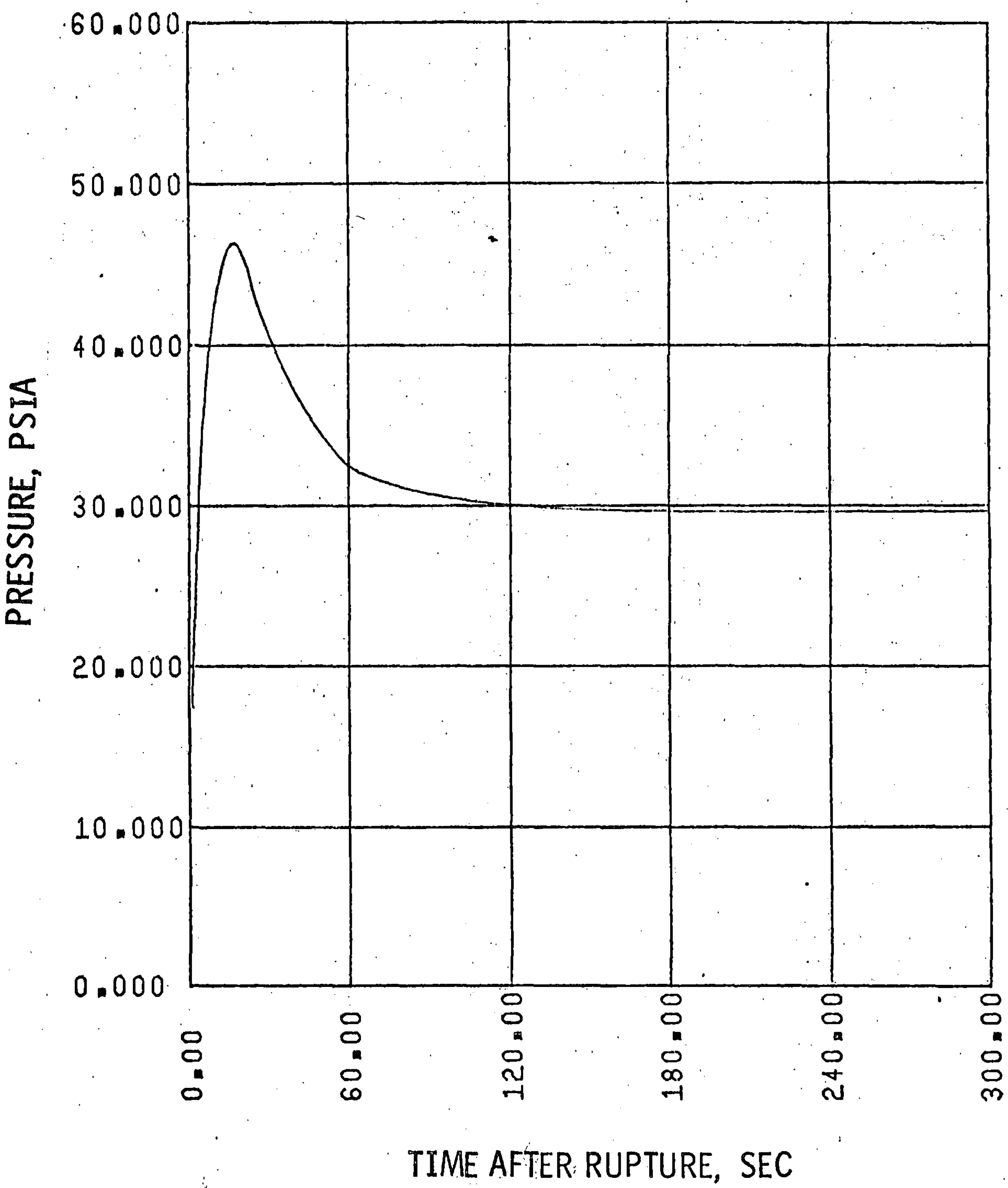


FIGURE II.7-G

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

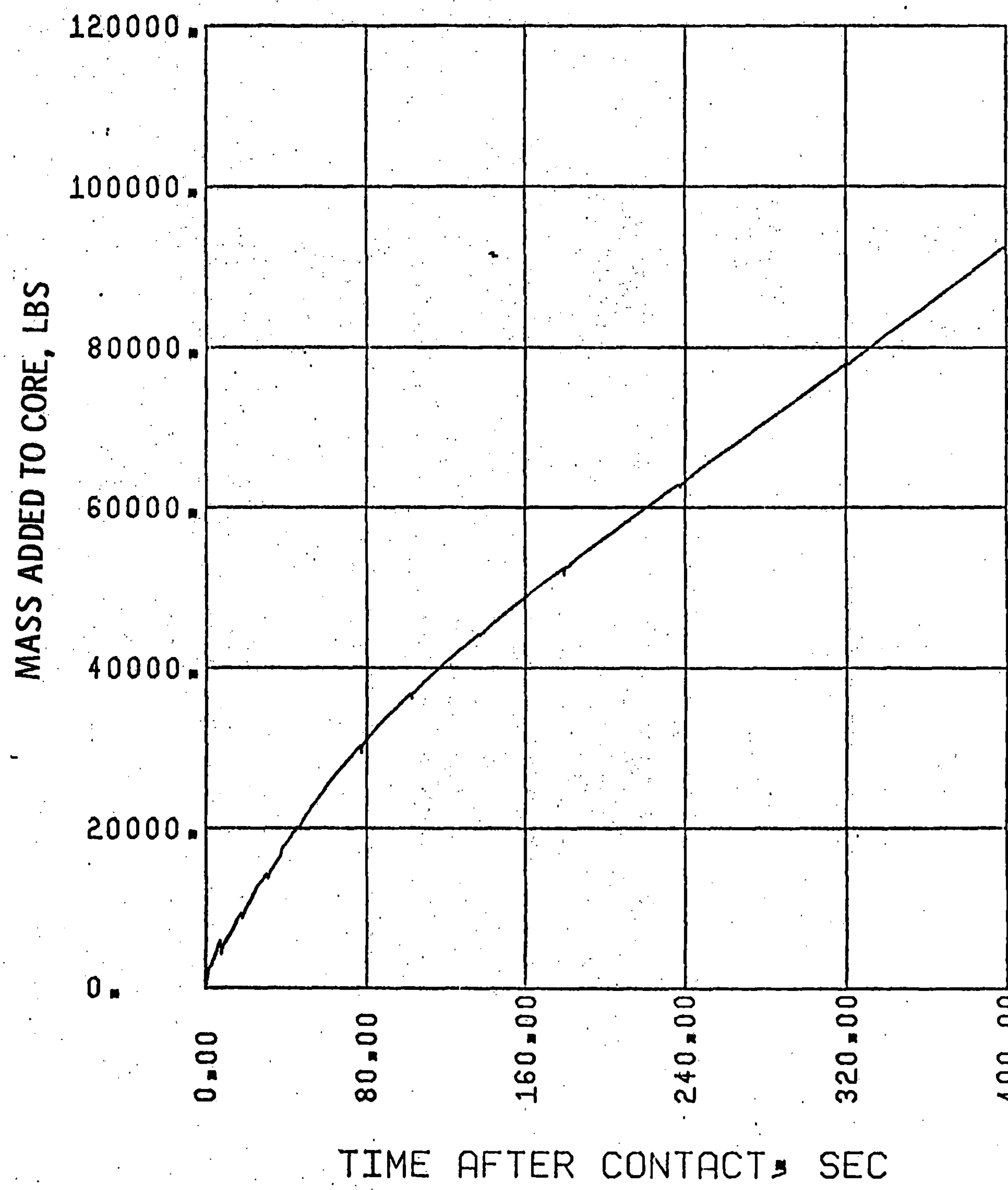


FIGURE II.7-H

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

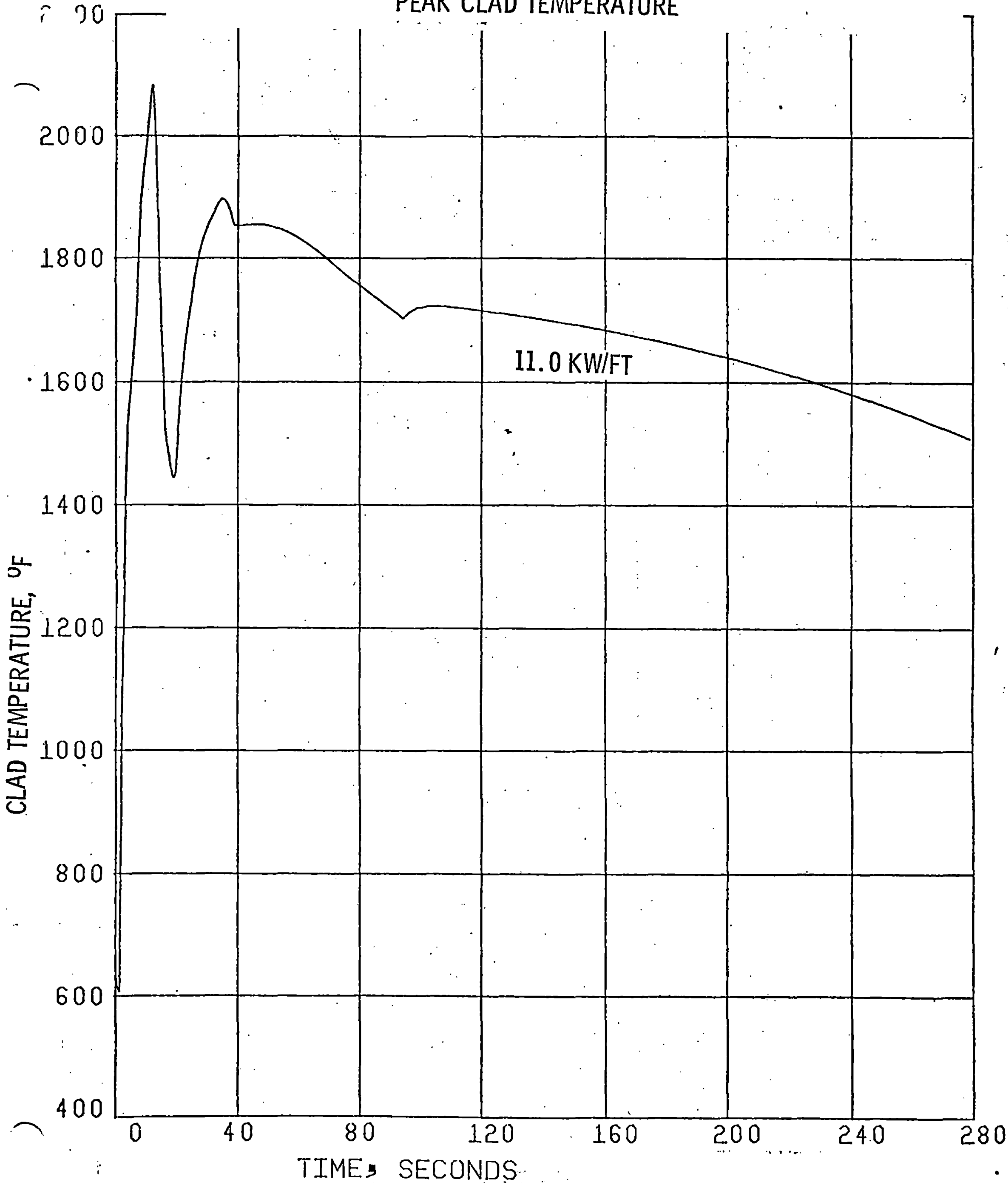


Figure II.7-0

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
LOCAL CLAD OXIDATION

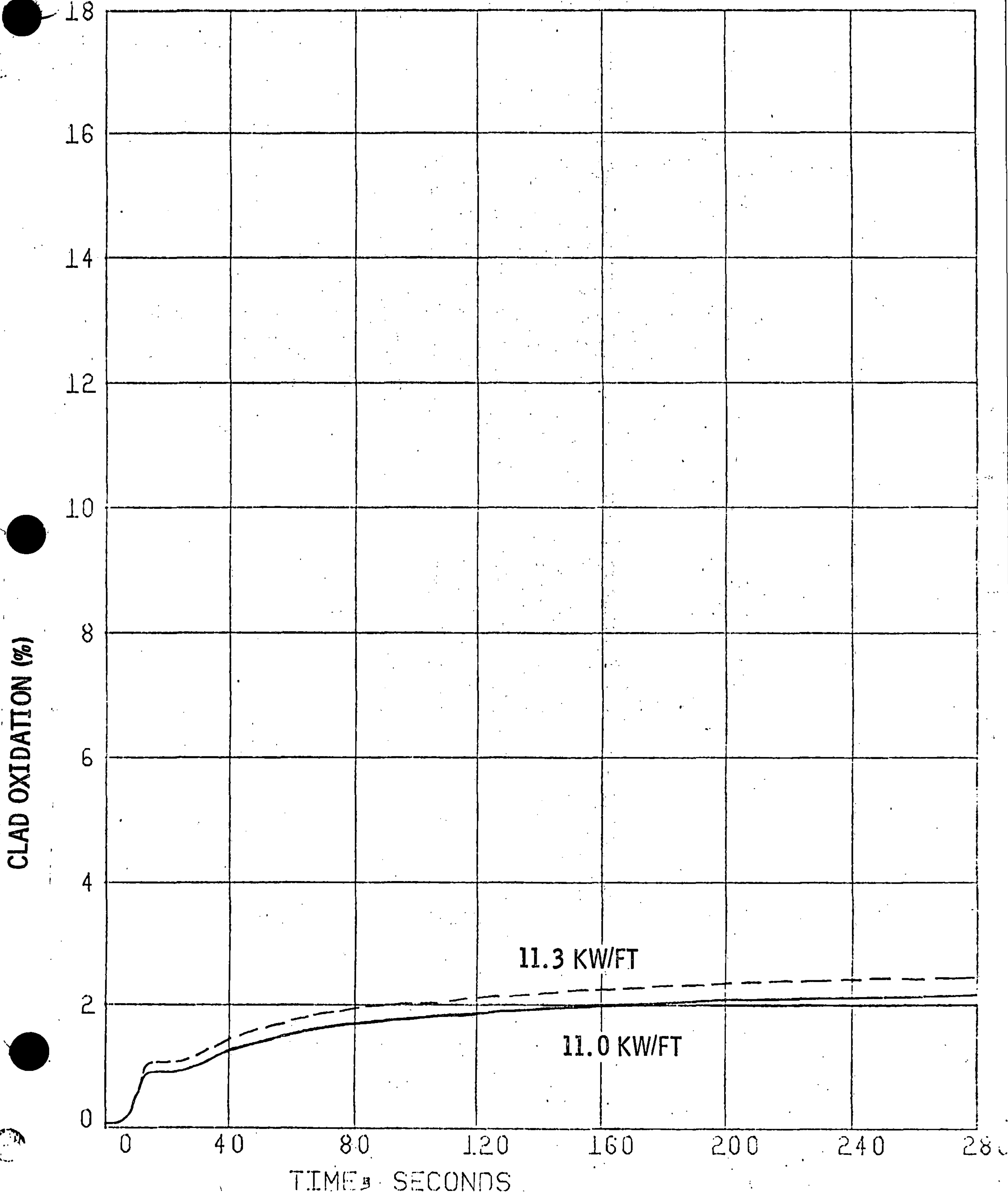


FIGURE II.8-A

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE POWER

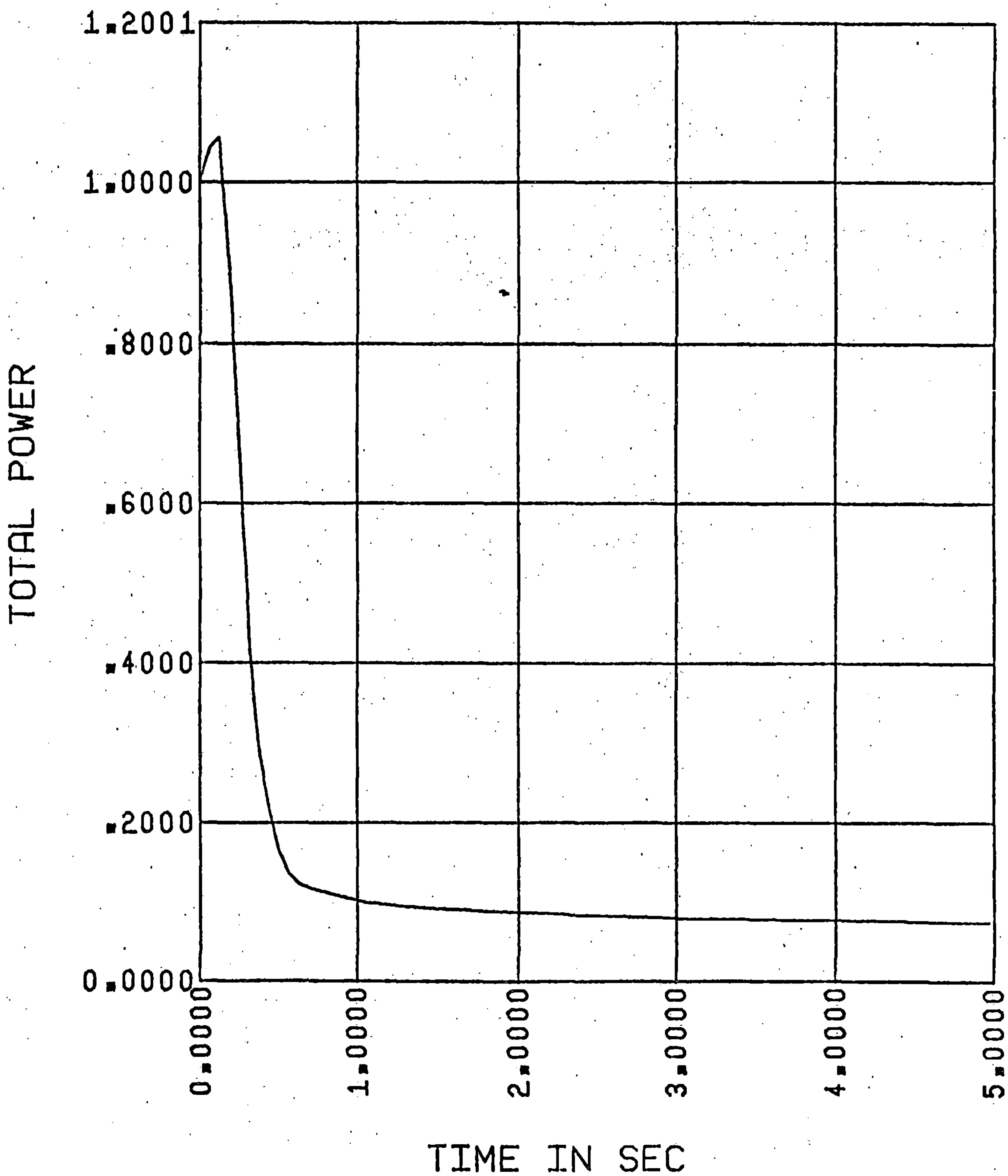


FIGURE II.8-B

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PRESSURE IN CENTER HOT ASSEMBLY NODE

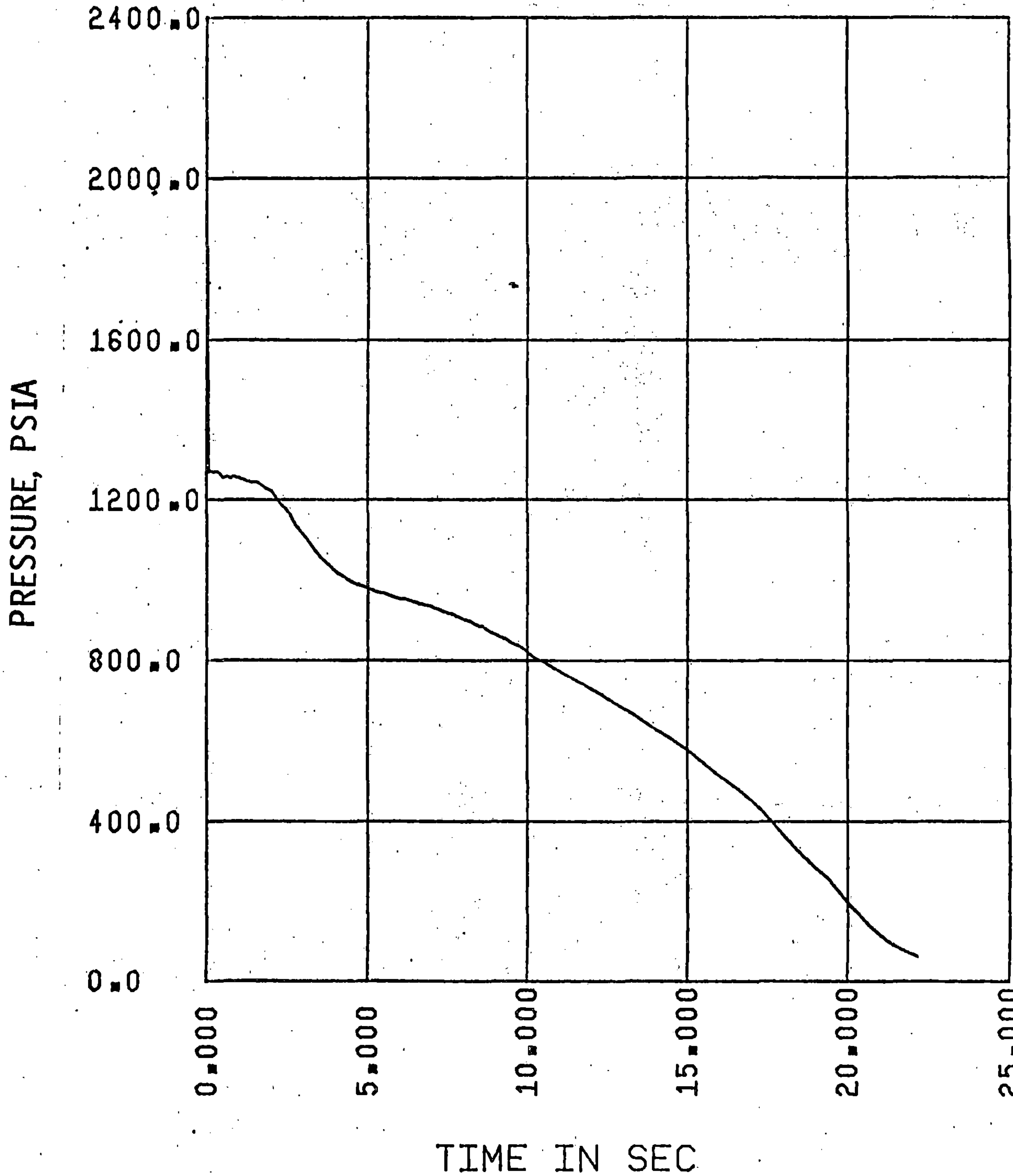


FIGURE II.8-C

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

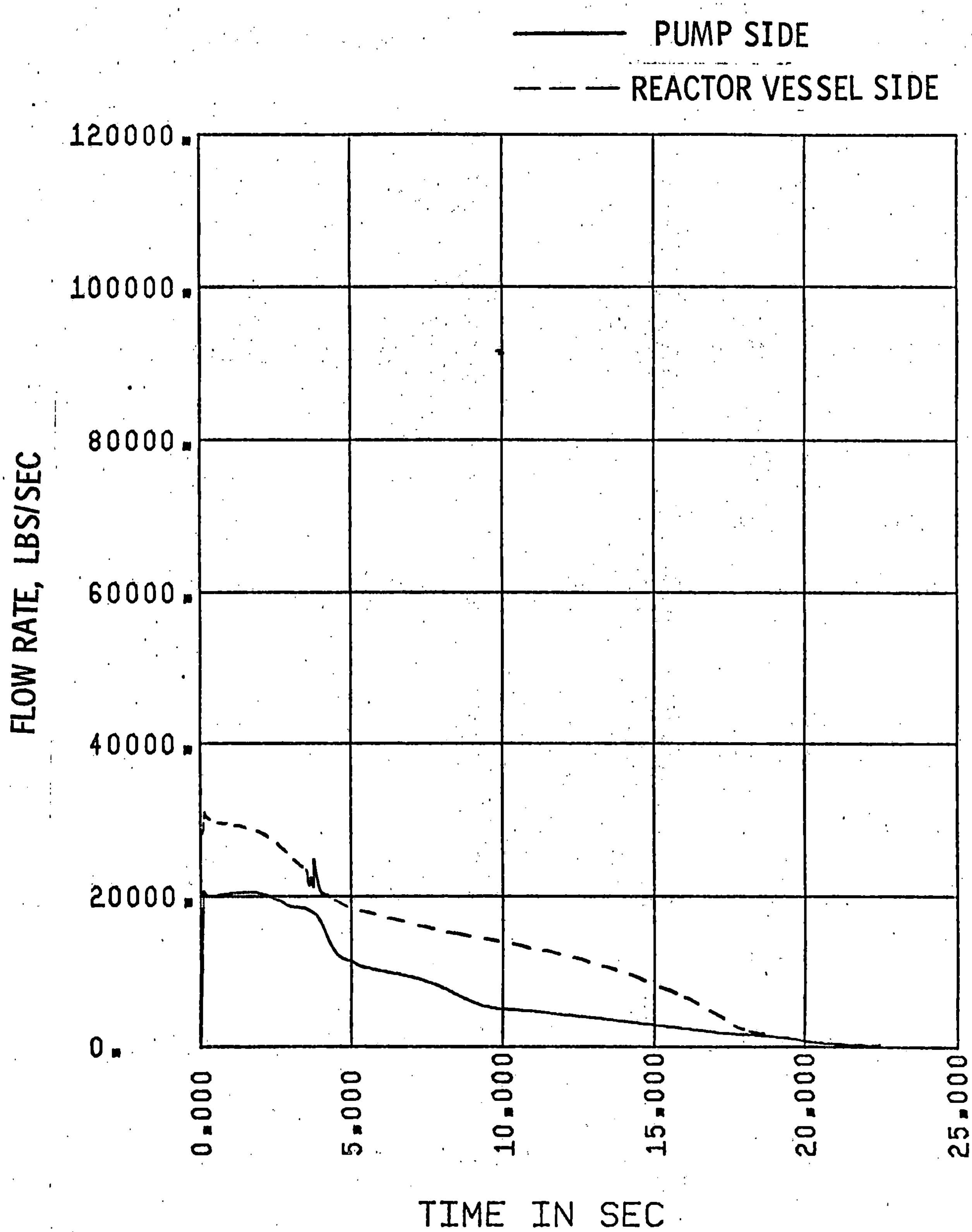


FIGURE II.8-D.1

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

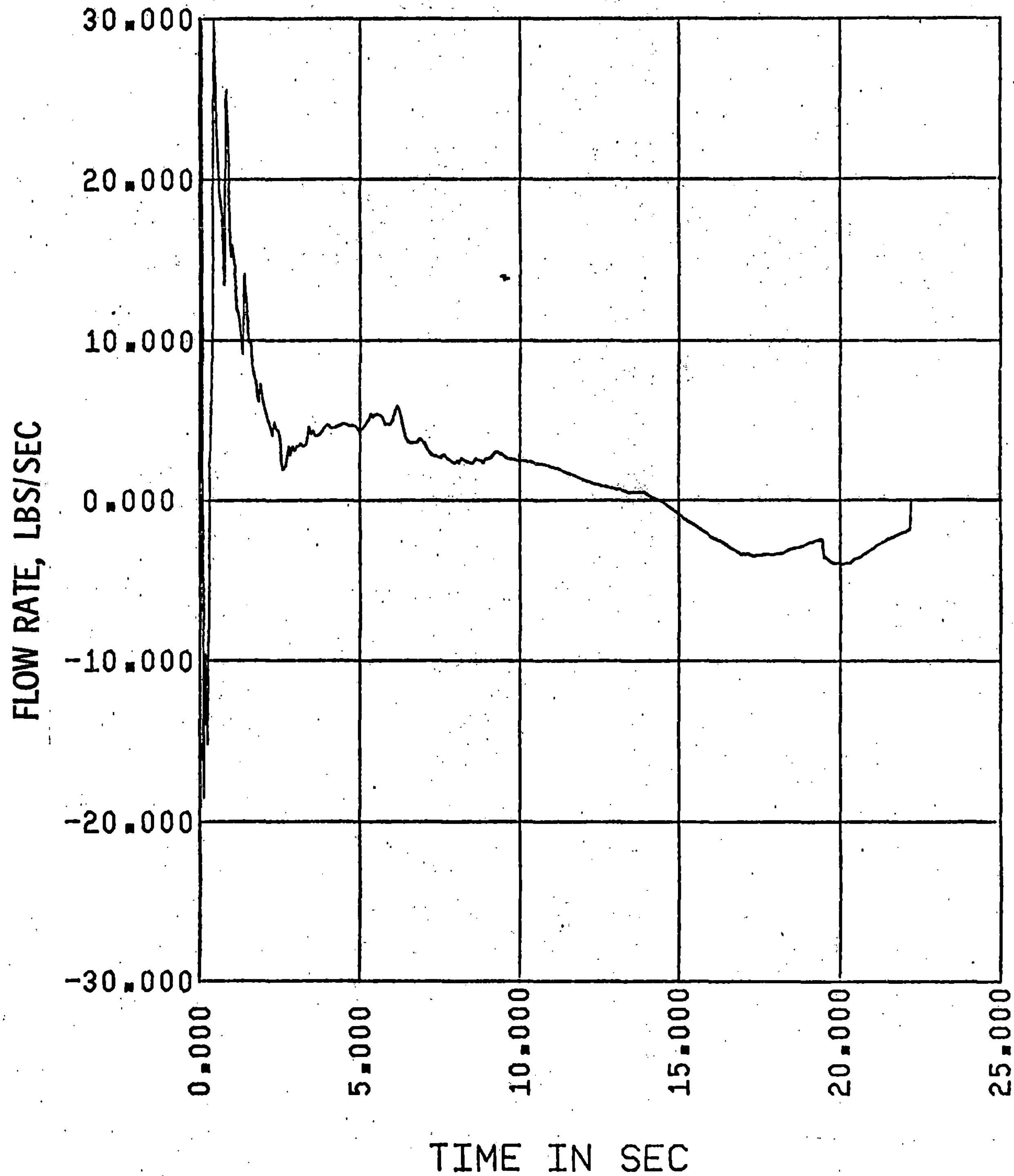


FIGURE II.8-D.2

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

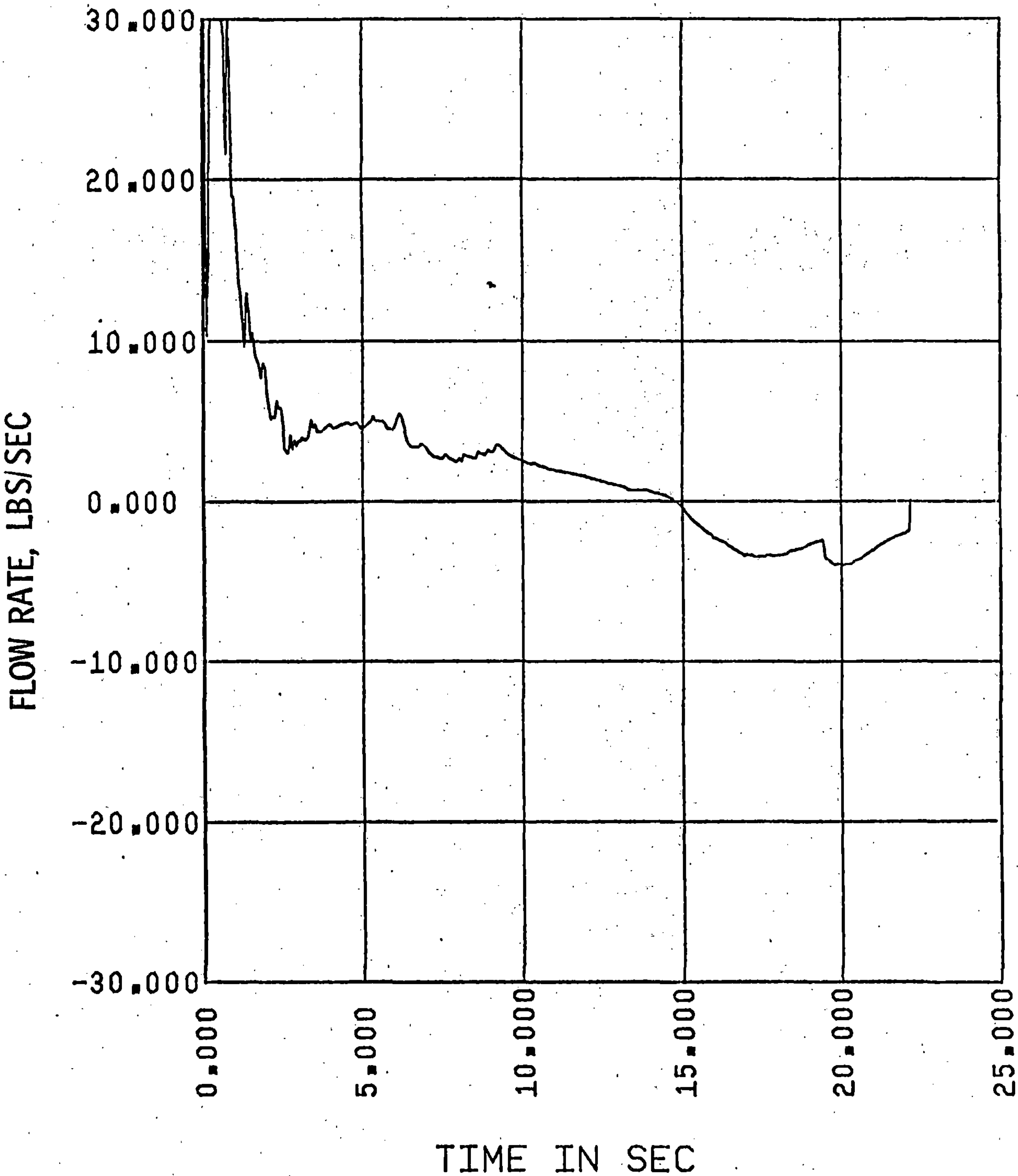


FIGURE II.8-E

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

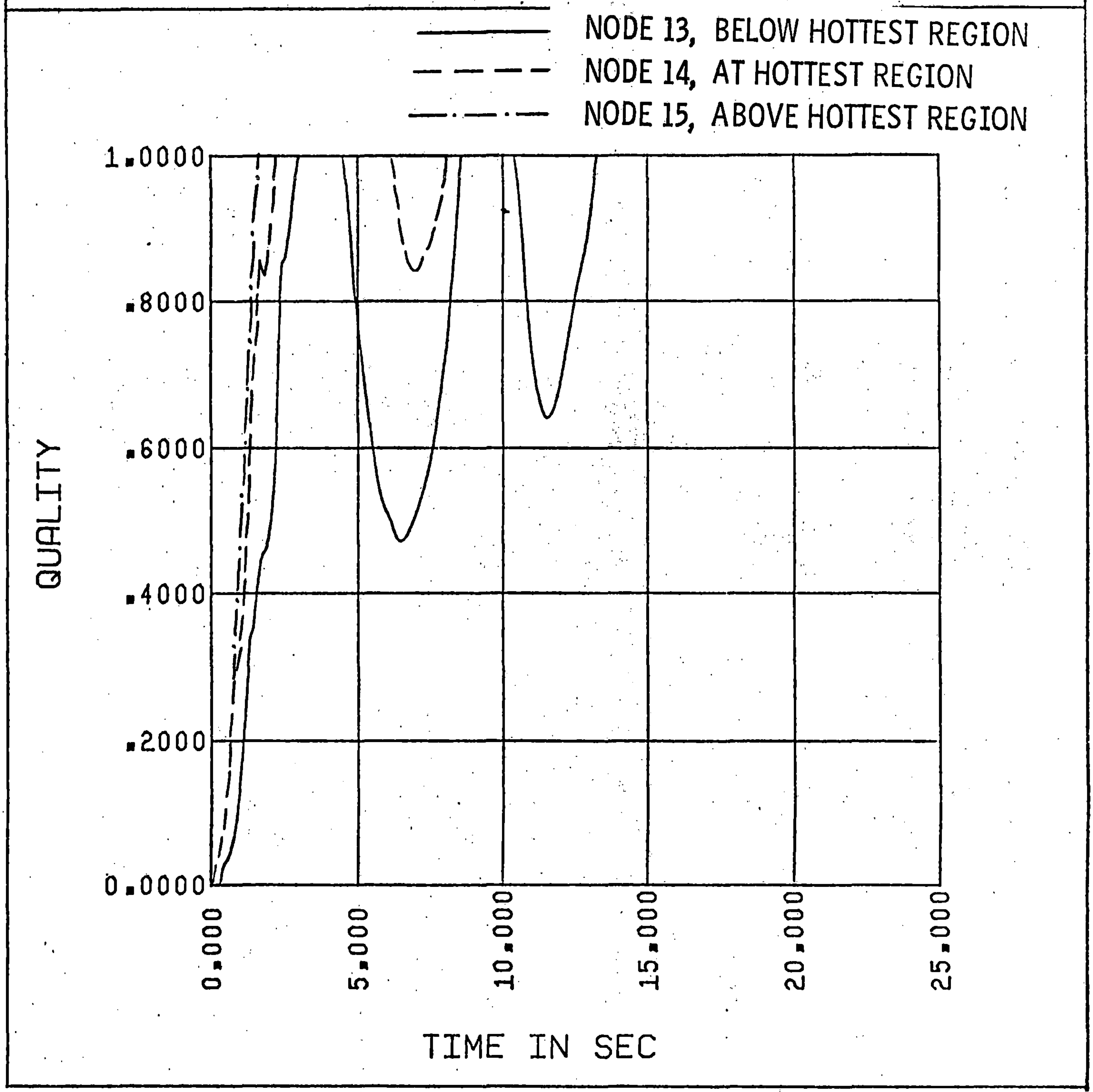


FIGURE II.8-F

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

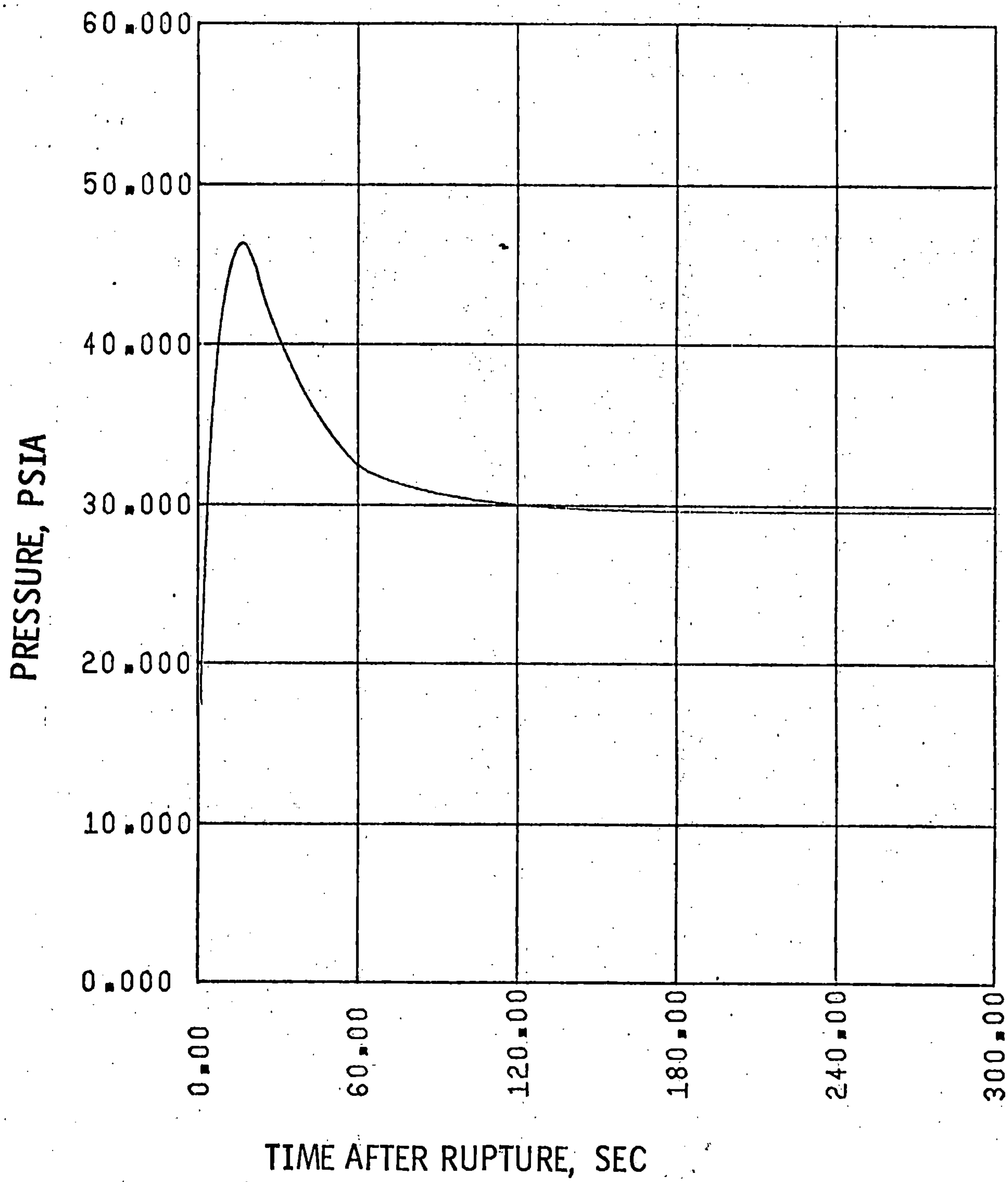


FIGURE II.8-G

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CURE DURING REFLOOD

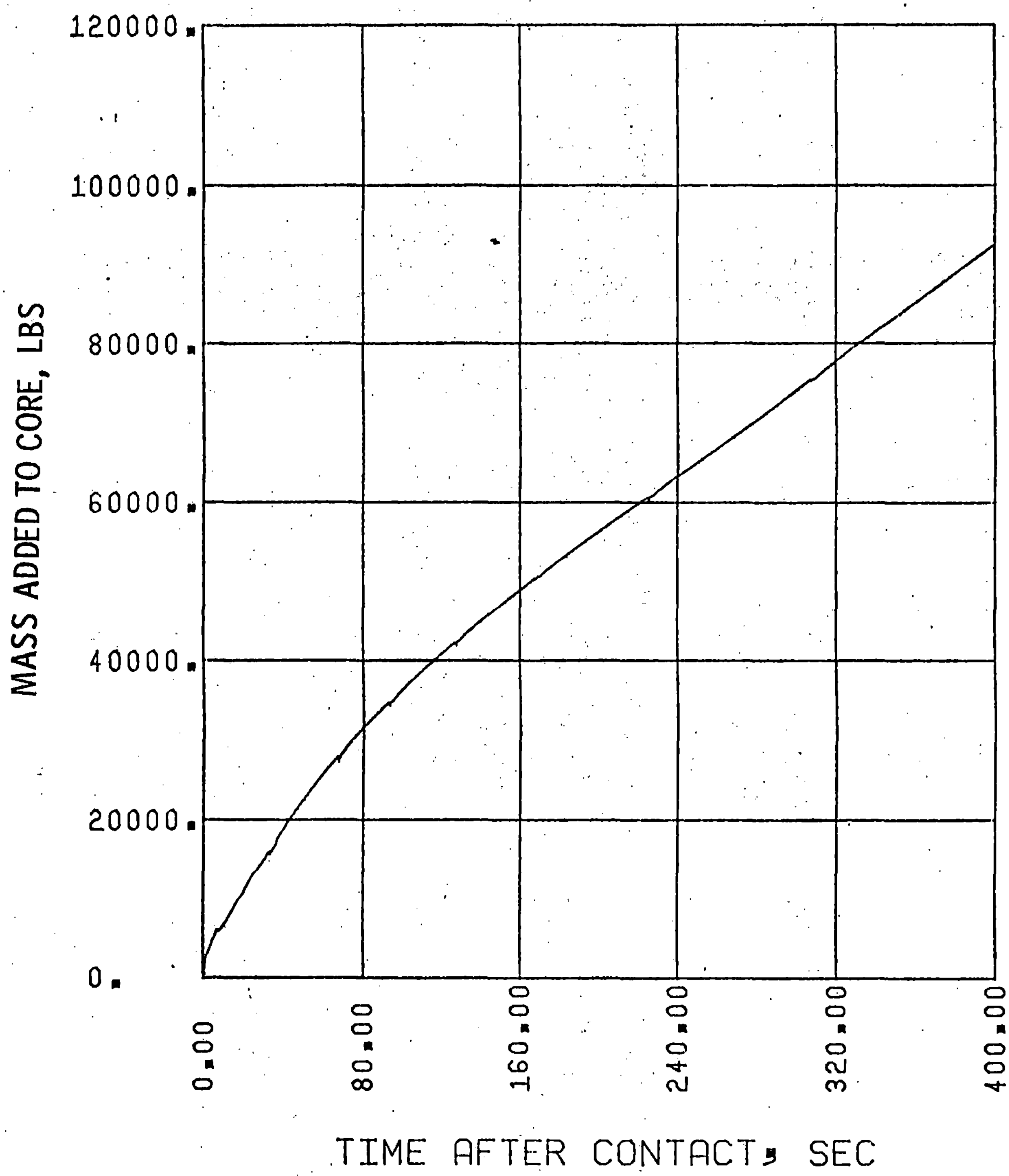


FIGURE II.8-H

PALISADES CORE I REANALYSIS
0.5 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

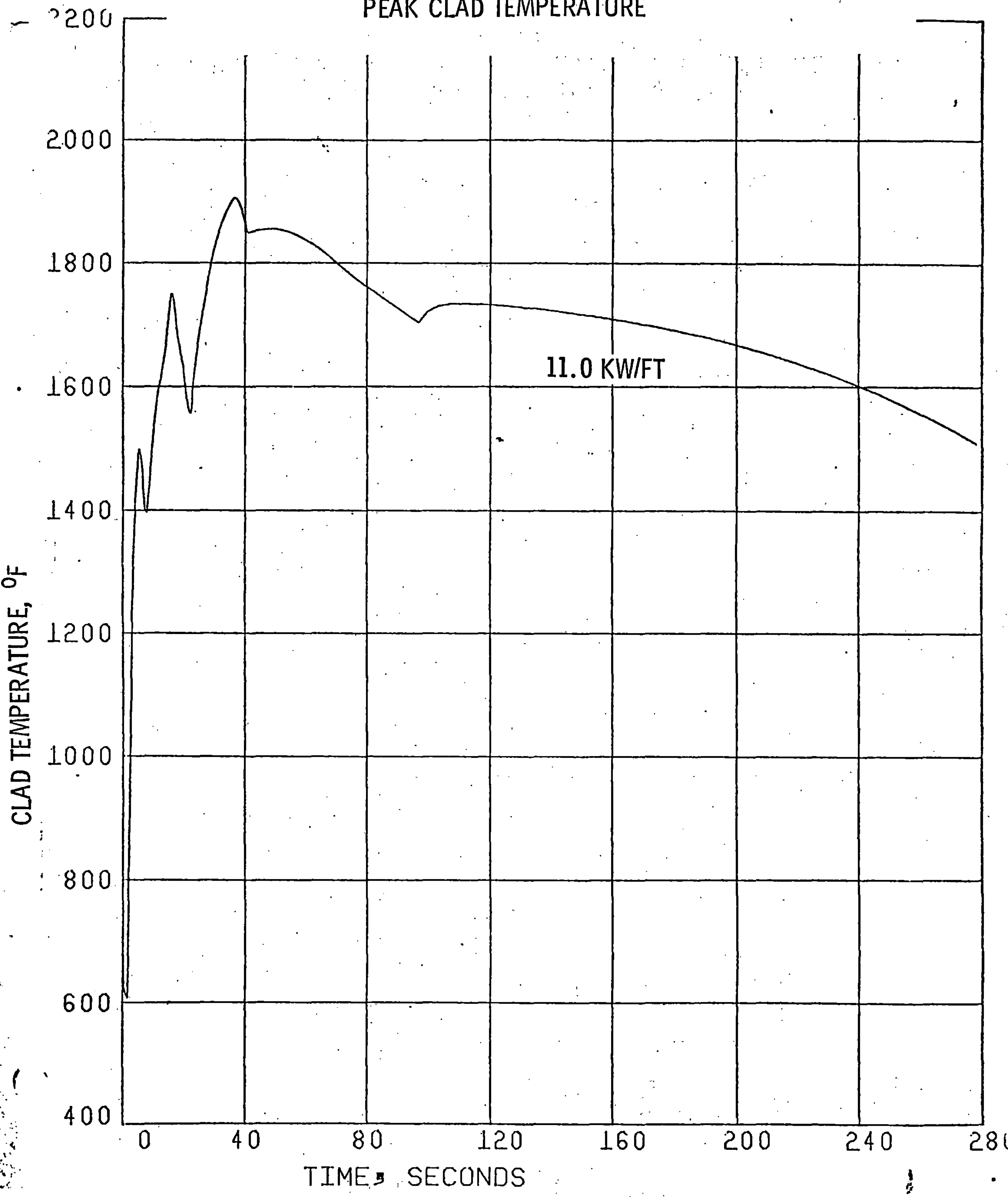


Figure II.9-F

PALISADES CORE REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

APPROVED MODEL

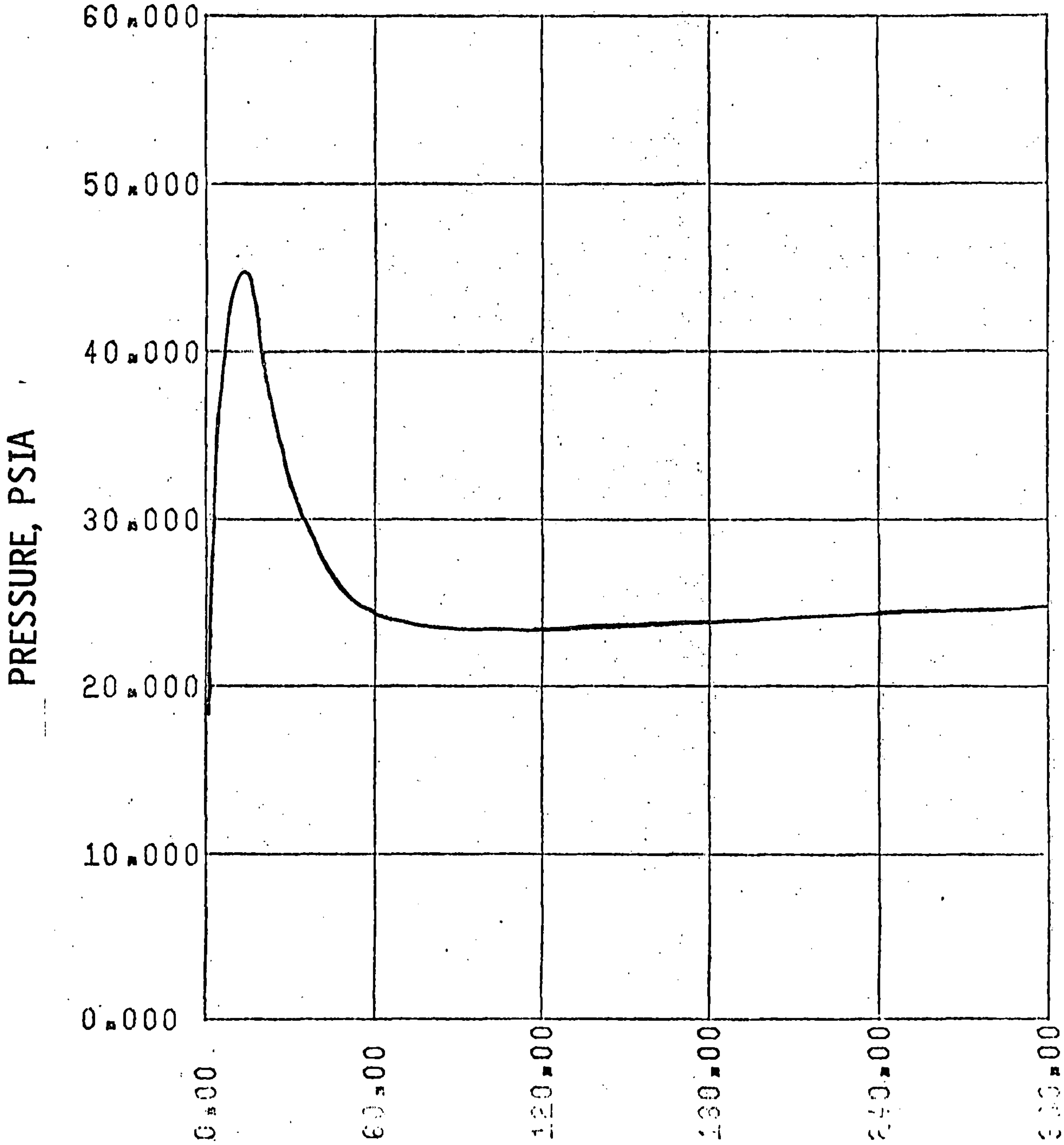


Figure II.9-G

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

APPROVED MODEL

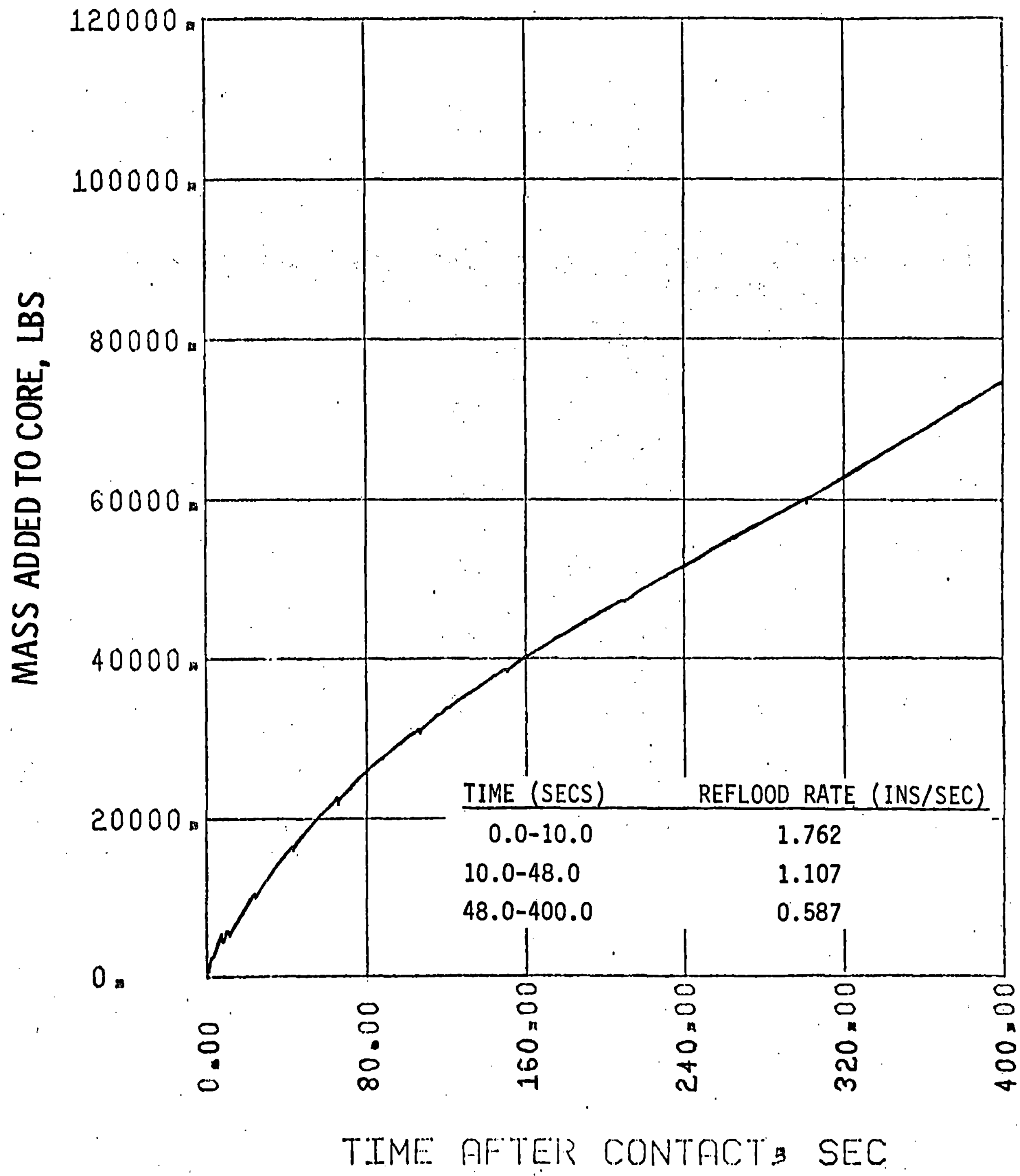


Figure II.9-H

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
PEAK CLAD TEMPERATURE

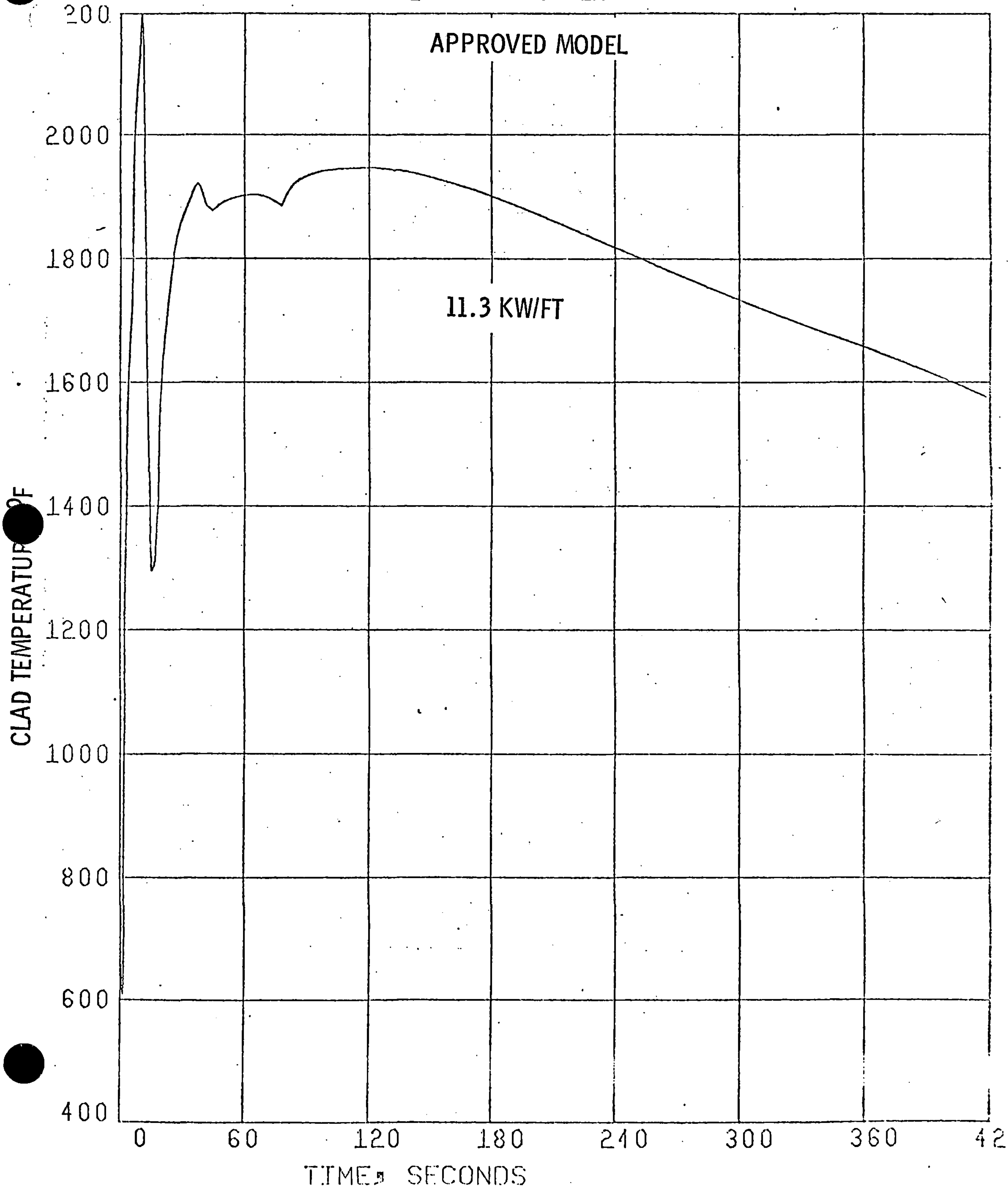


Figure II.9-L

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
SAFETY INJECTION TANK FLOW INTO INTACT DISCHARGE LEGS

APPROVED MODEL

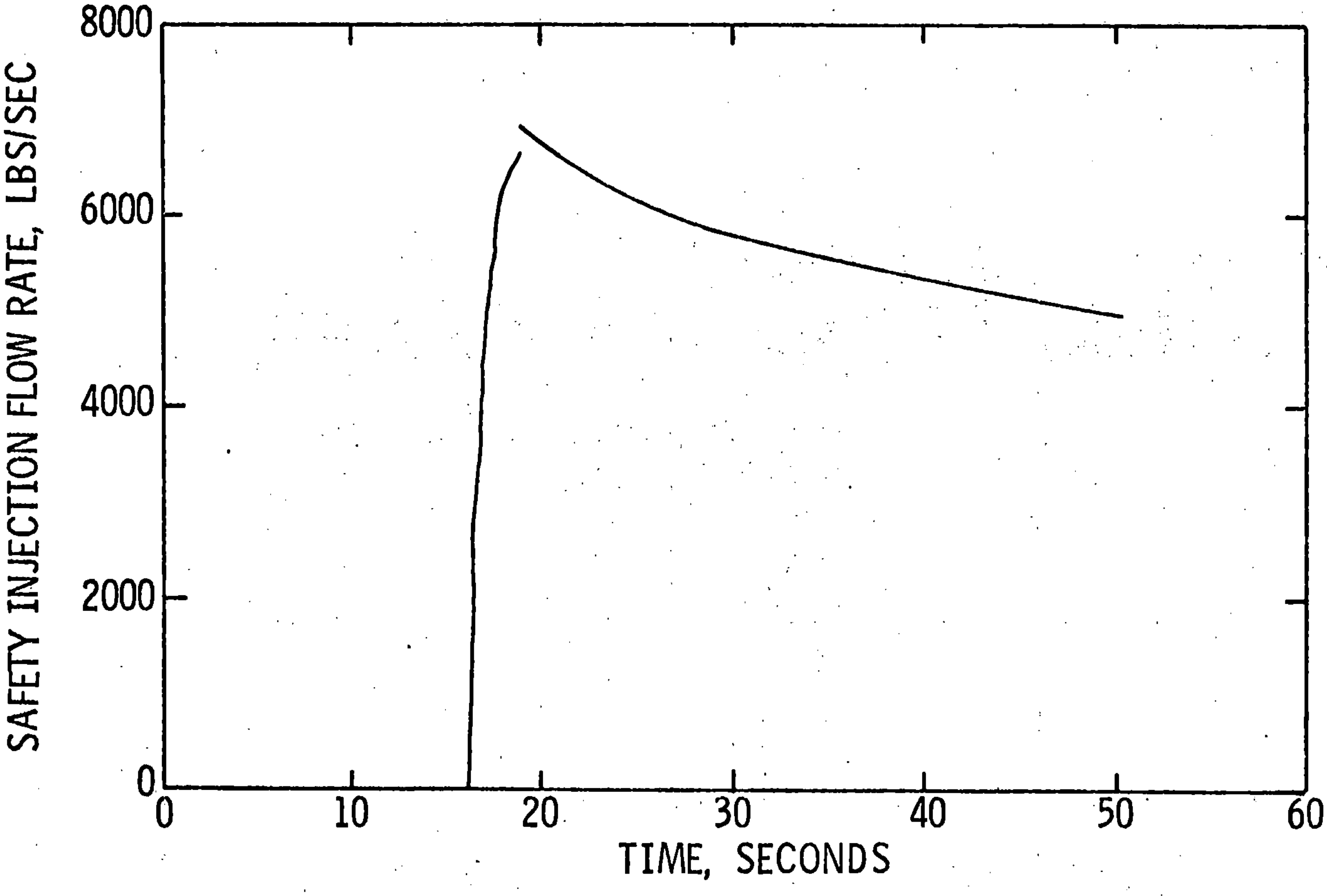


Figure II. 9-M

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
WATER LEVEL IN DOWNCOMER DURING REFLOOD

APPROVED MODEL

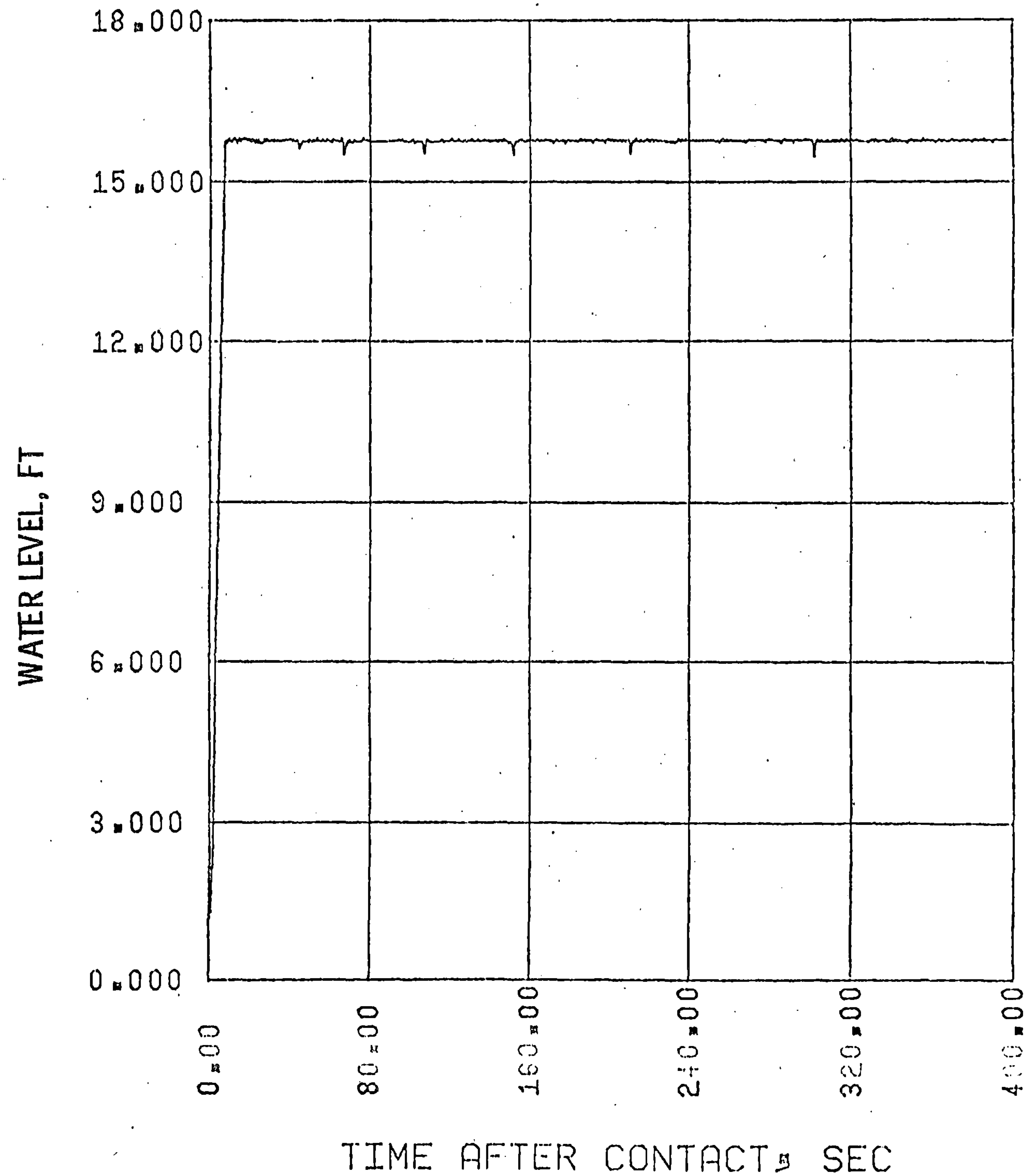


Figure II.9-N

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
GAP CONDUCTANCE

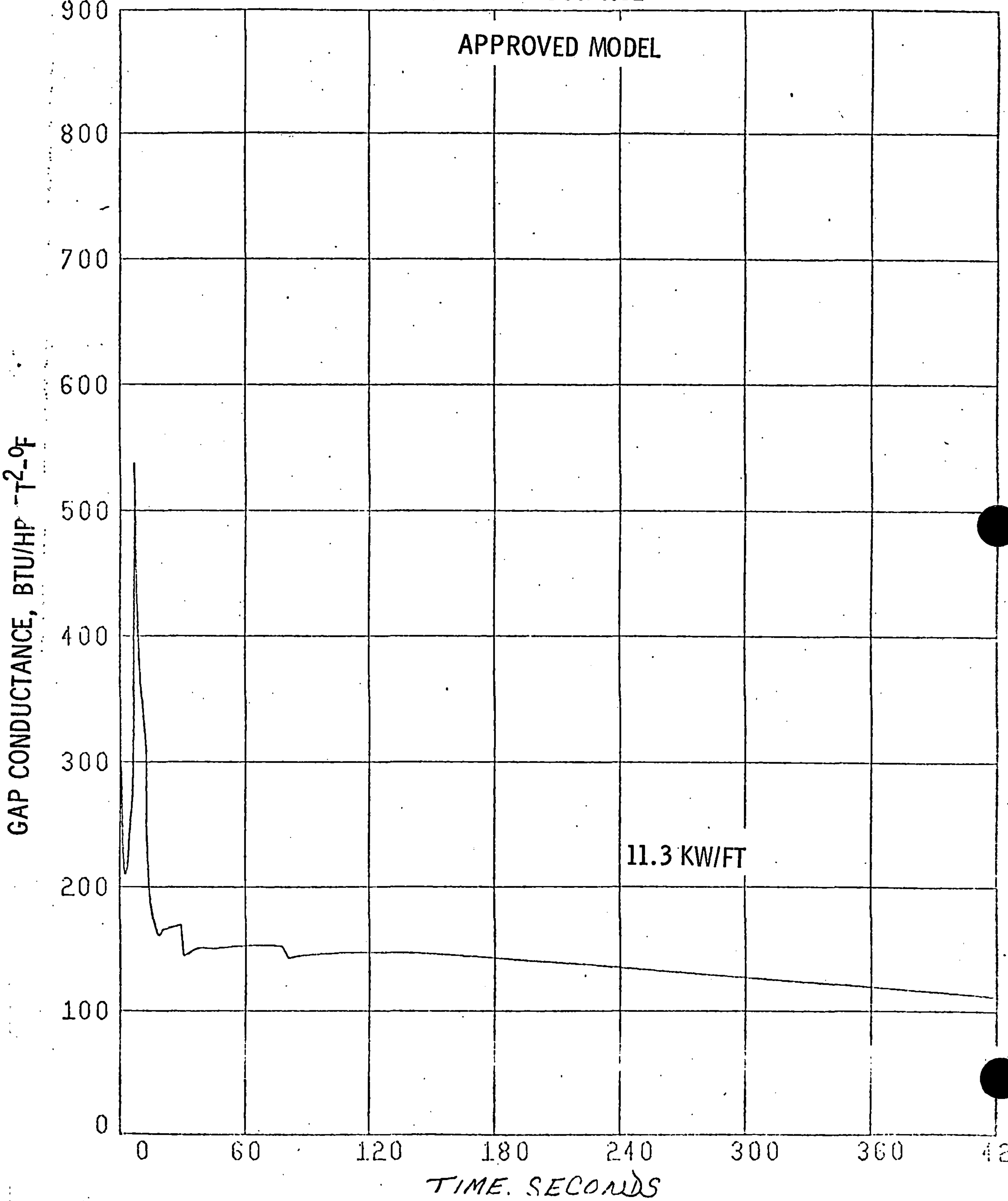


Figure II.9-0

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
LOCAL CLAD OXIDATION

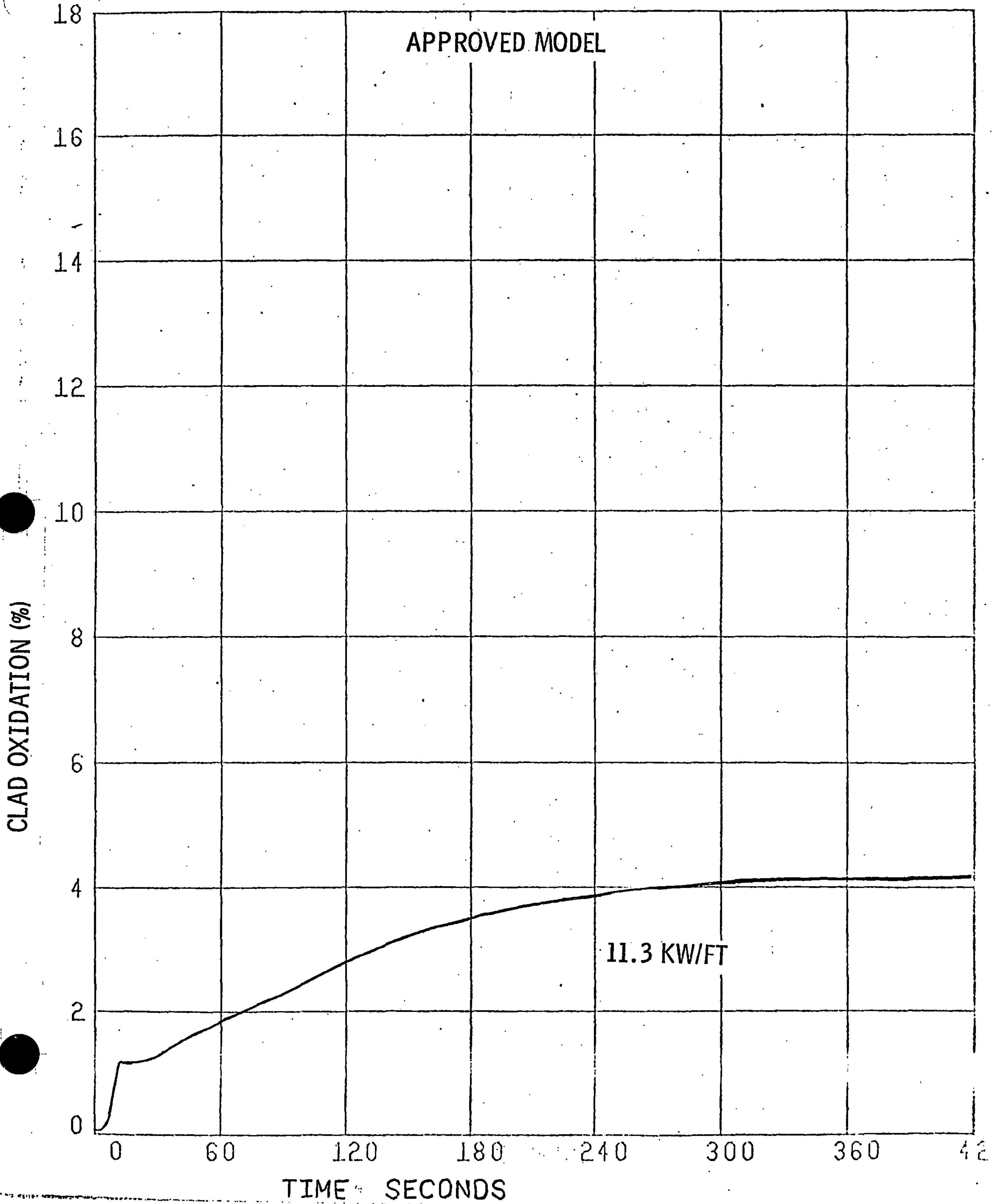


Figure II.9-P

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CLAD TEMPERATURE, CENTERLINE FUEL TEMP., AVG. FUEL TEMP.,
AND COOLANT TEMP. FOR HOTTEST NODE

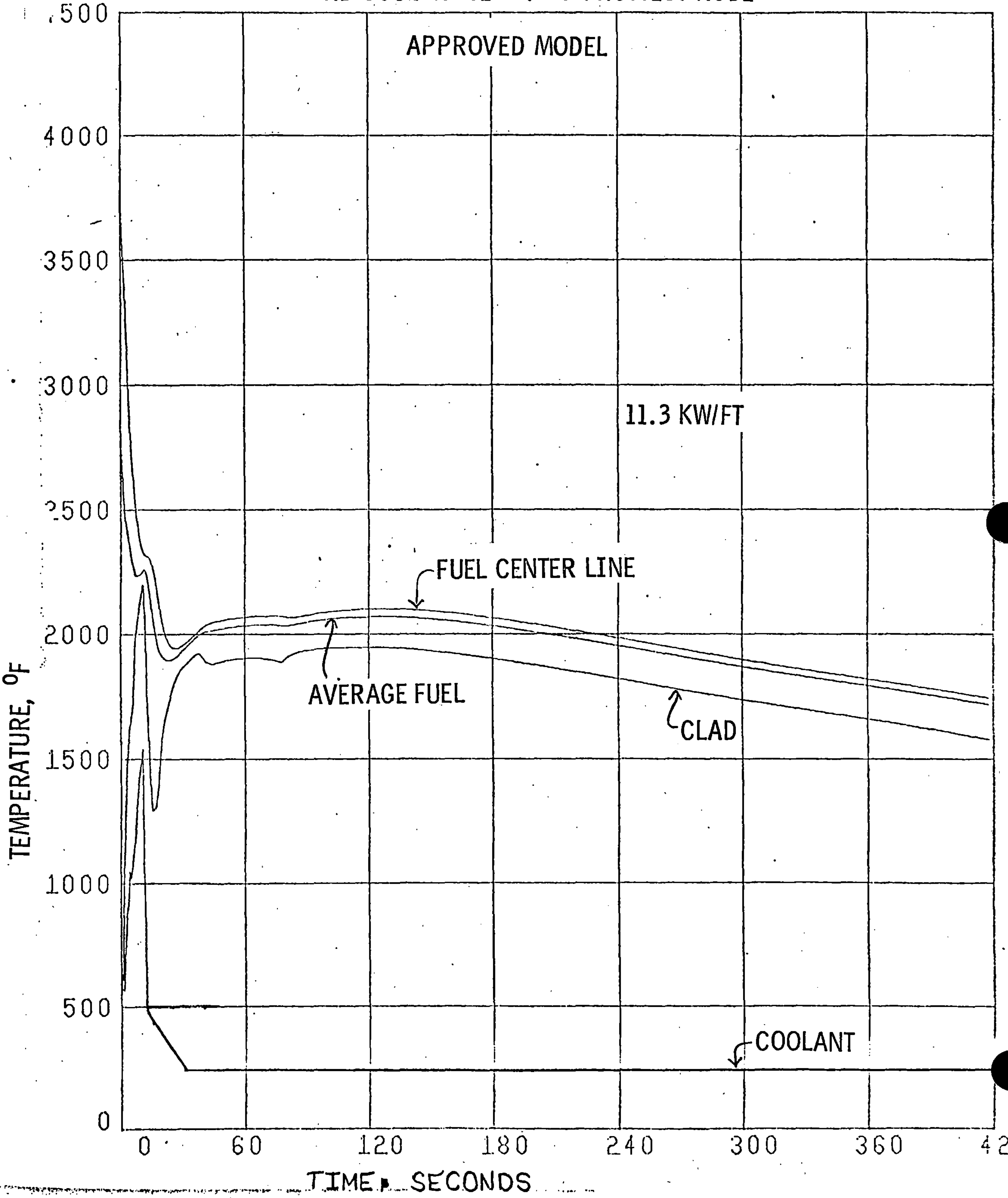


Figure II.9-Q

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT HEAT TRANSFER COEFFICIENT

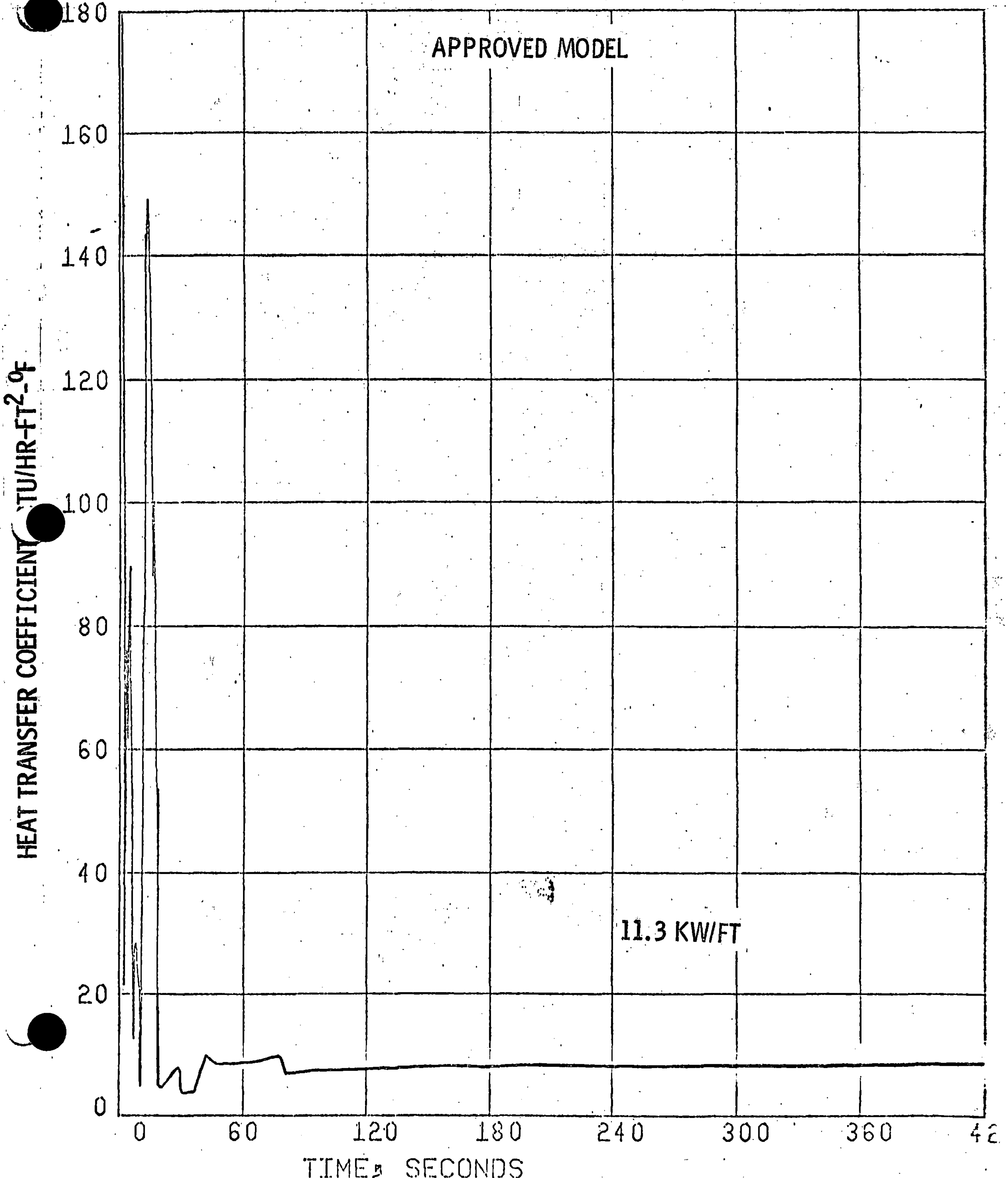


Figure II.9-R

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
HOT SPOT HEAT TRANSFER COEFFICIENT DURING REFLOOD

APPROVED MODEL

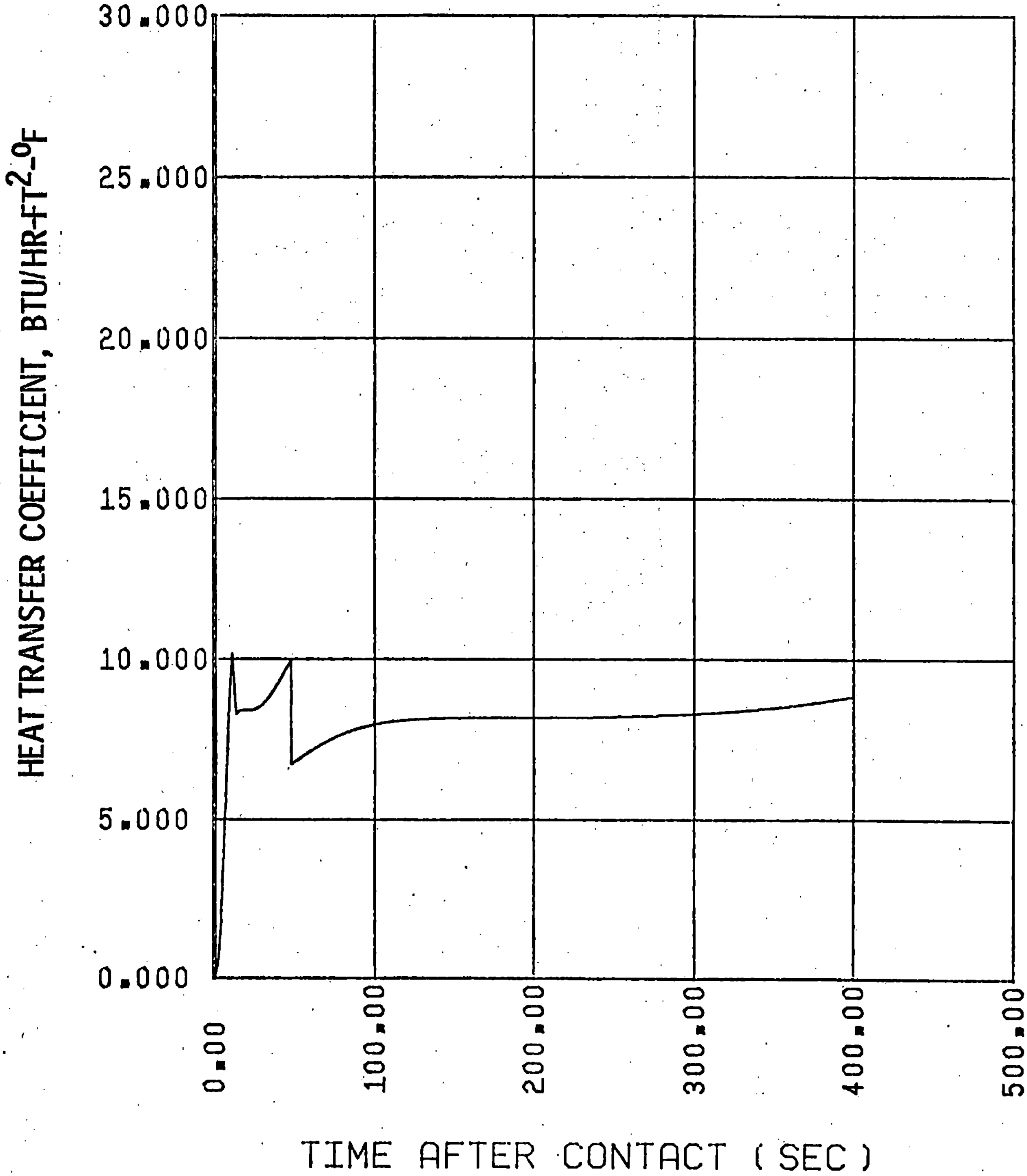


Figure II.9-S

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
CONTAINMENT TEMPERATURE
APPROVED MODEL

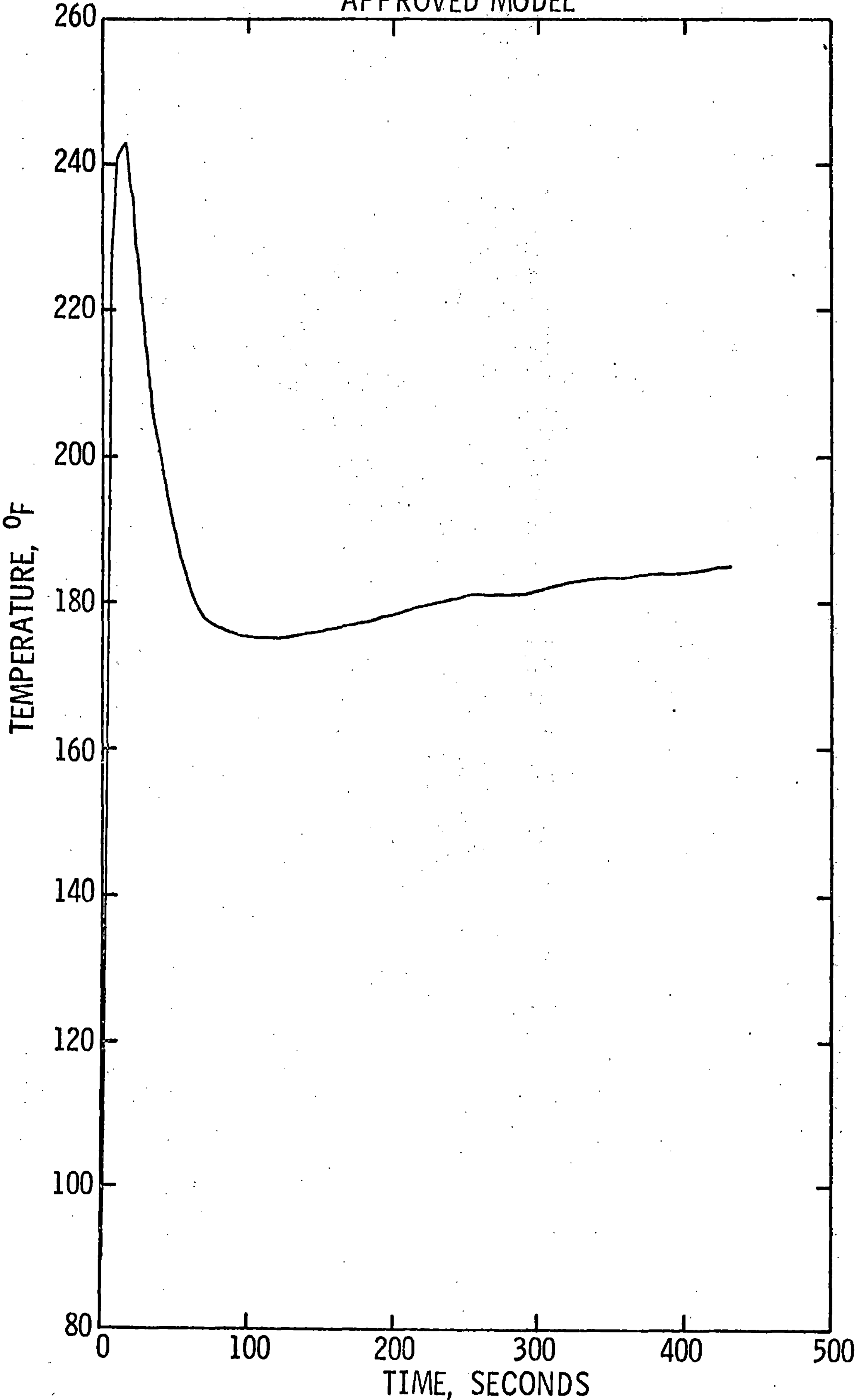


Figure II.9-T

PALISADES CORE I REANALYSIS
1.0 x DOUBLE ENDED SLOT BREAK IN PUMP DISCHARGE LEG
SUMP TEMPERATURE

APPROVED MODEL

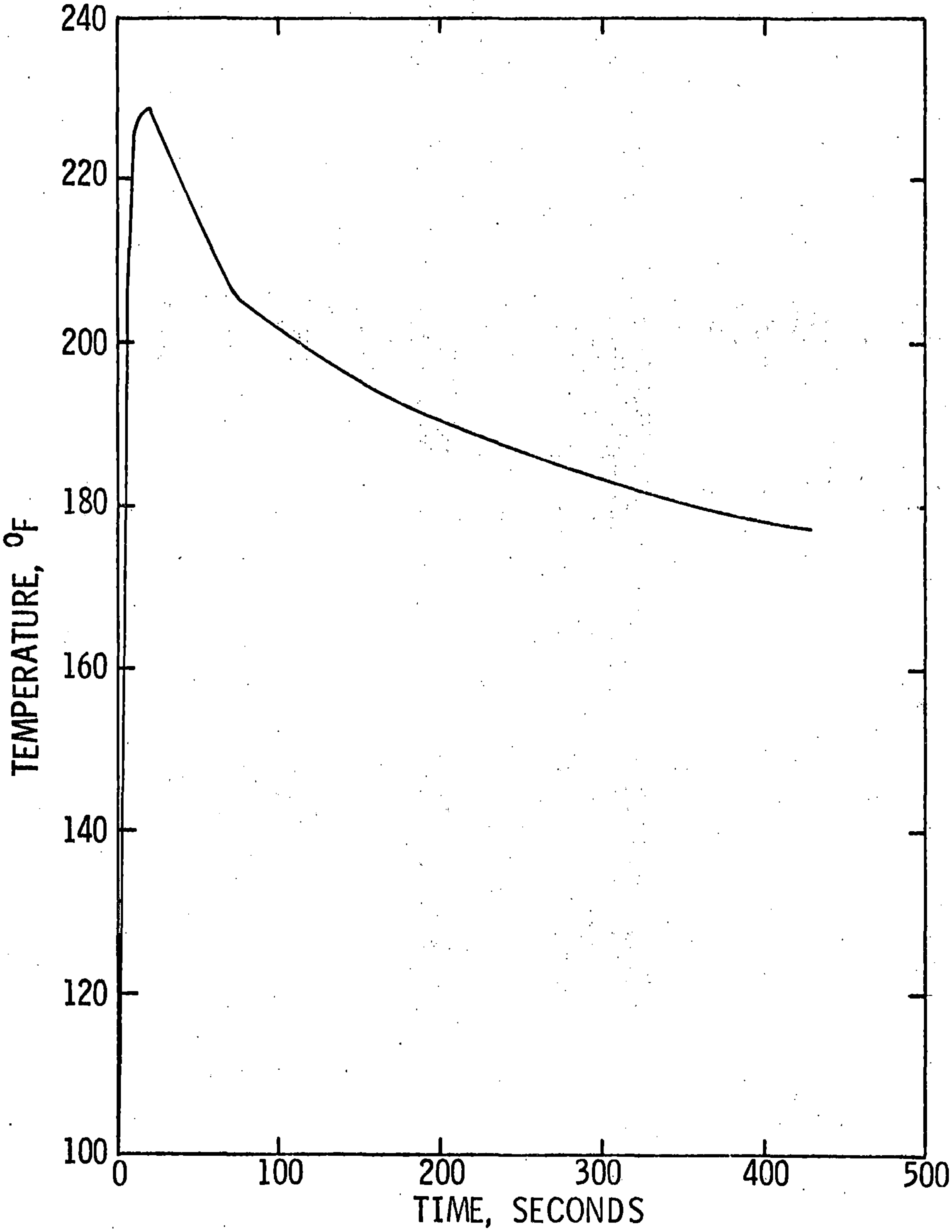


Figure II.10-0

PALISADES CORE I REANALYSIS
0.6 x DOUBLE ENDED GUILLOTINE BREAK IN PUMP DISCHARGE LEG
LOCAL CLAD OXIDATION

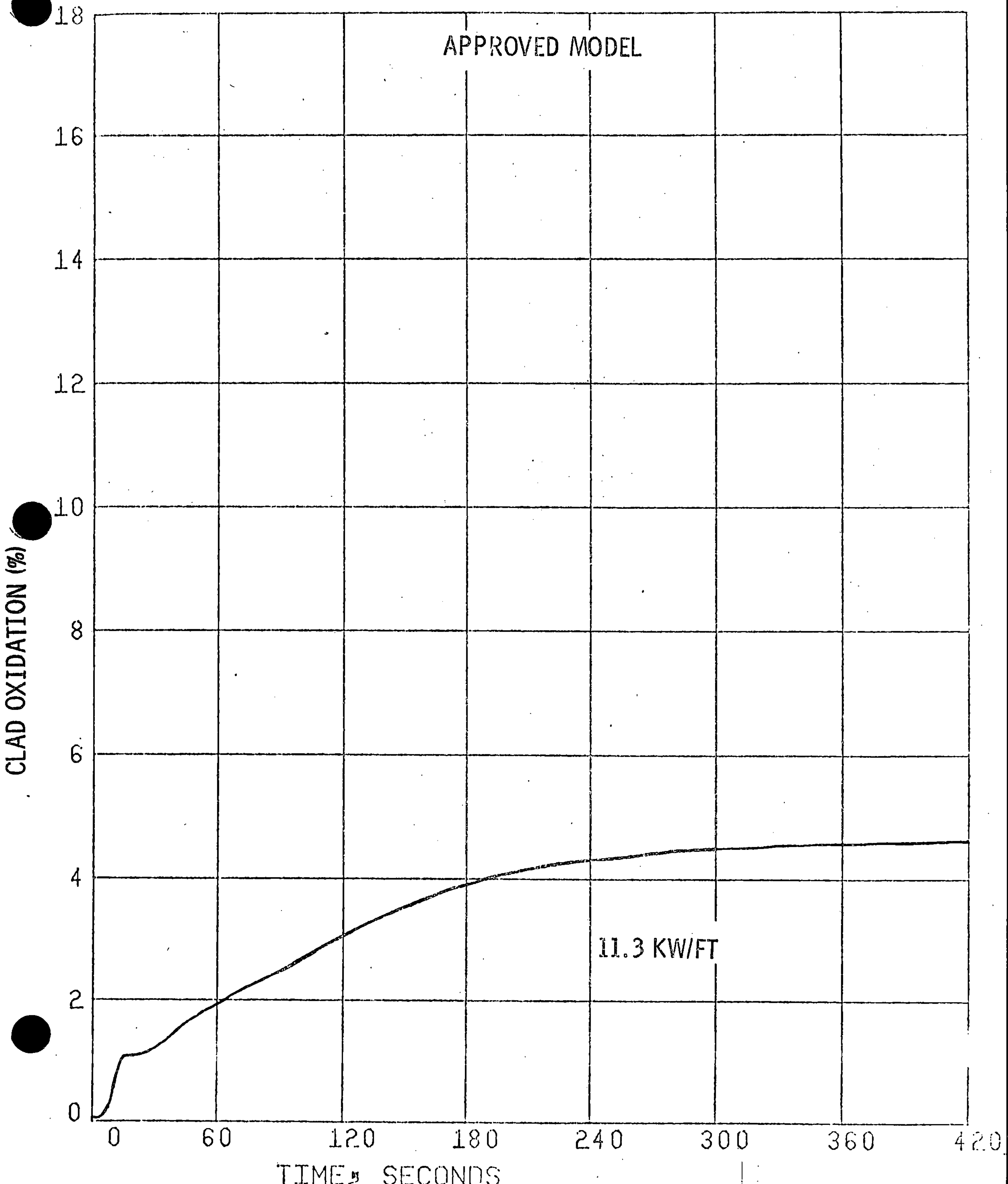


Figure II.11

PALISADES CORE I REANALYSIS
PEAK CLAD TEMPERATURE vs BREAK AREA

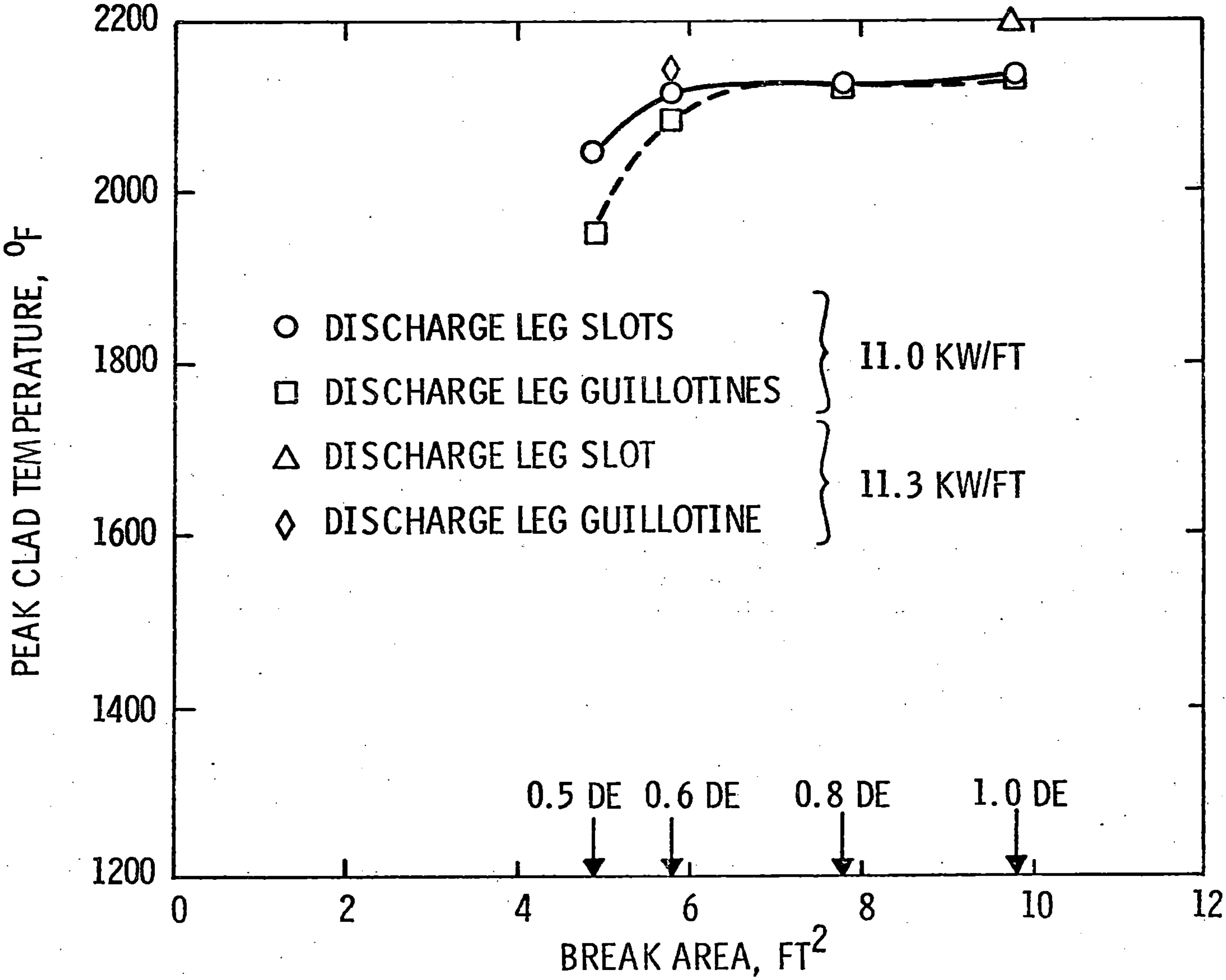
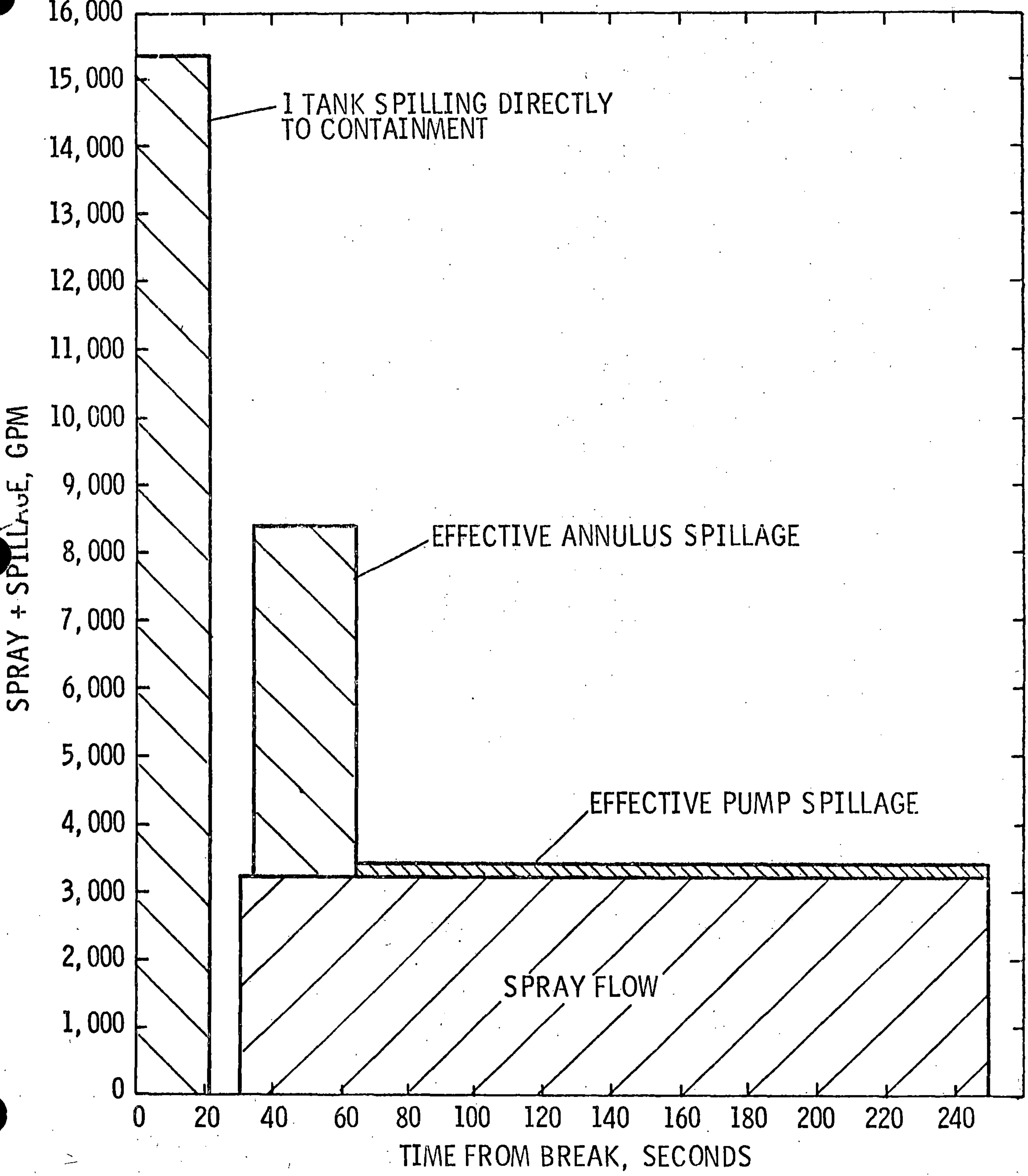


Figure II.12

PALISADES CORE I REANALYSIS
COMBINED SPILLAGE AND SPRAY INTO CONTAINMENT



CONSUMERS POWER COMPANY

DOCKET 50-255

REQUEST FOR CHANGE TO THE TECHNICAL SPECIFICATIONS

LICENSE DPR-20

For the reasons hereinafter set forth, it is requested that the Technical Specifications contained in Provisional Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on October 16, 1972 be changed as described below:

I. Changes

A. On Page ii, Table of Contents, add the following:

Section	Description	Page
3.18	Linear Heat Generation Rate Limits Associated With LOCA Considerations	3-82

B. Add a new Item d to Section 3.1.5 as follows:

"d. The primary to secondary leakage in a steam generator shall not exceed 0.3 gpm for any period greater than 24 consecutive hours."

C. Add a new paragraph at the end of the basis of Section 3.1.5 as follows:

"The 0.3 gpm primary to secondary leakage limit was originally provided to ensure that if clad collapse existed which in essence locked the fuel pellet stack that transients such as a loss of load transient accompanied by a secondary relief valve sticking open would not cause 10 CFR Part 20 limits to be exceeded at the site boundary. Fuel inspections performed during the fall of 1973 after a core exposure of approximately 7,000 MWd/MT have shown that the collapse of unpressurized fuel rod cladding is not likely to occur because significant pellet gaps do not exist. Therefore, from the standpoint of fuel densification the limit is no longer required. The limit is retained to provide a stringent limit on primary to secondary leakage during the transition from phosphate to volatile secondary water chemistry. This transition is due to the wastage attack experienced during 1973 due to the phosphate secondary water chemistry control."

D. Change Section 3.11 to read as follows:

"3.11 IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the in-core instrumentation system.

Objective

To specify the functional and operability requirements of the in-core instrumentation system.

Specification

a. Sufficient in-core instrumentation shall be operable whenever the reactor is operating at or above 75% rated power in order to: (1) Assist in the calibration of the out-of-core detectors, and (2) check gross core "power distribution. As a minimum, 10 individual detectors per quadrant, which shall include 2 detectors at each of the four axial levels, shall be operable.

b. For power operation above a power level of 85% of the level permitted by Section 3.18, in-core detector alarms generated by the data logger shall be set, based on the latest power distribution obtained, such that the peak linear power does not exceed the limit specified in Section 3.18. If four or more coincident alarms are received, the validity of the alarms shall be immediately determined and, if valid, power shall be immediately decreased below alarm set point and a power distribution map obtained. If a power distribution is not obtained within 24 hours of the alarm conditions, power shall be reduced to 85% of the value defined in Section 3.18.

c. The in-core detector alarm set points shall be established based on the latest power distribution maps, normalized to the kW/ft limit defined in Section 3.18.

d. Power distributions shall be evaluated every week or more often as required by plant operations.

e. The data logger can be inoperable for two hours. If at the end of two hours, it is not available, the power level shall not exceed 85% of the kW/ft limit defined in Section 3.18.

f. If the data logger for the in-cores is not operable for more than two hours, readings shall be taken and logged on a minimum of 10 individual detectors per quadrant at least each two hours thereafter or the reactor power level shall be reduced to less than 75% rated.

Bases

A system of 45 in-core flux detector and thermocouple assemblies and a data display, alarm and record functions has been provided.[1] The out-of-core nuclear instrumentation calibration includes:

"a. Calibration (axial and azimuthal) of the split detectors at initial reactor start-up and during the power escalation program.

b. A comparison check with the in-core instrumentation in the event abnormal readings are observed on the out-of-core detectors during operation.

c. Calibration check during subsequent reactor start-ups.

d. Confirm that readings from the out-of-core split detectors are as expected and that the ratio of the top and bottom detector readings is acceptable.

Core power distribution verification includes:

a. Measurement at initial reactor start-up to check that power distribution is consistent with calculations.

b. Subsequent checks during operation to insure that power distribution is consistent with calculations.

c. Indication of power distribution in the event that abnormal situations occur during reactor operation.

If the data logger for the in-core readout is inoperable, for more than two hours, power will be reduced to 85% of the limit specified in Section 3.18 to provide margin between the actual peak linear heat generation rates and the limit and the in-core readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the manpower available, the reactor power will be reduced to less than 75% rated to

minimize the probability of exceeding the peaking factors. The time interval of two hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the data logger is returned to service.

Reference - (1) FSAR, Section 7.4.2.4."

E. Add a new Section 3.18 as follows:

3.18 Linear Heat Generation Rate Limit Associated With LOCA Considerations

Applicability

Applies to fuel linear heat generation rates.

Objective

To delineate the requirements regarding fuel linear heat generation rates associated with a postulated Loss of Coolant Accident.

Specification

3.18.1 The linear heat generation rate with appropriate consideration of normal flux peaking, measurement-calculational uncertainity (10%), engineering factor (3%), increase in linear heat rate due to axial fuel densification (1.75%), power measurement uncertainity (2%), and flux peaking augmentation factors which vary from near 0% at the bottom of the core to approximately 3.8% at the top of the core, as shown in Figure 3-6, shall not exceed that limit which causes calculated ECCS performance, as predicted by an evaluation model approved by the NRC as satisfying the requirements of 10 CFR 50, Appendix K, to exceed the "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors" as given in 10 CFR 50.46(b)."

Bases

"To maintain the integrity of the fuel cladding under the conditions of a postulated Loss of Coolant Accident (LOCA), the Emergency Core Cooling Systems (ECCS) must satisfy certain criteria set forth by the US Nuclear Regulatory Commission in

Augmentation Factor Versus Height

Monte Carlo Computation for Palisades

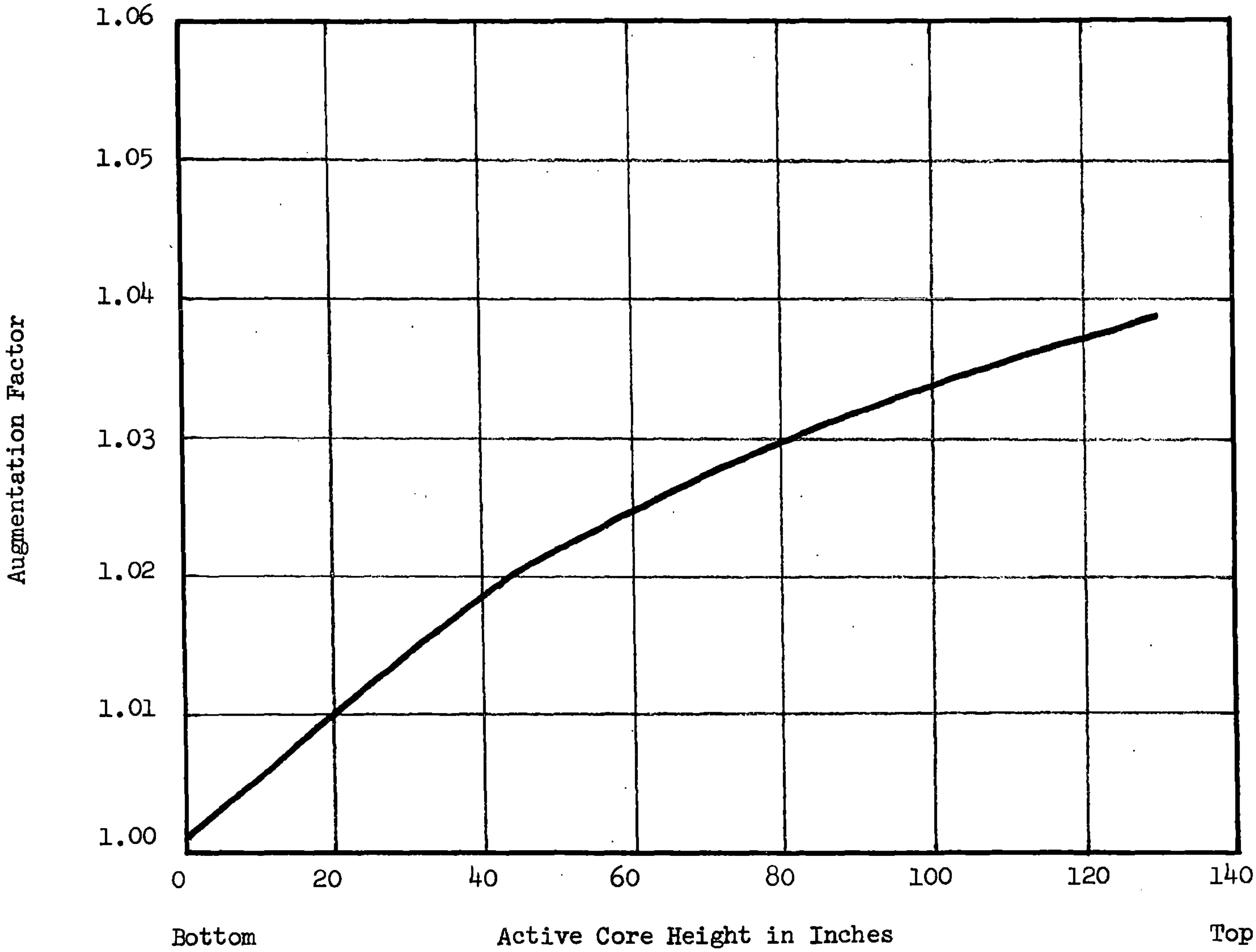


Figure 3 - 6

Augmentation Factor Versus Height

Monte Carlo Computation for Palisades

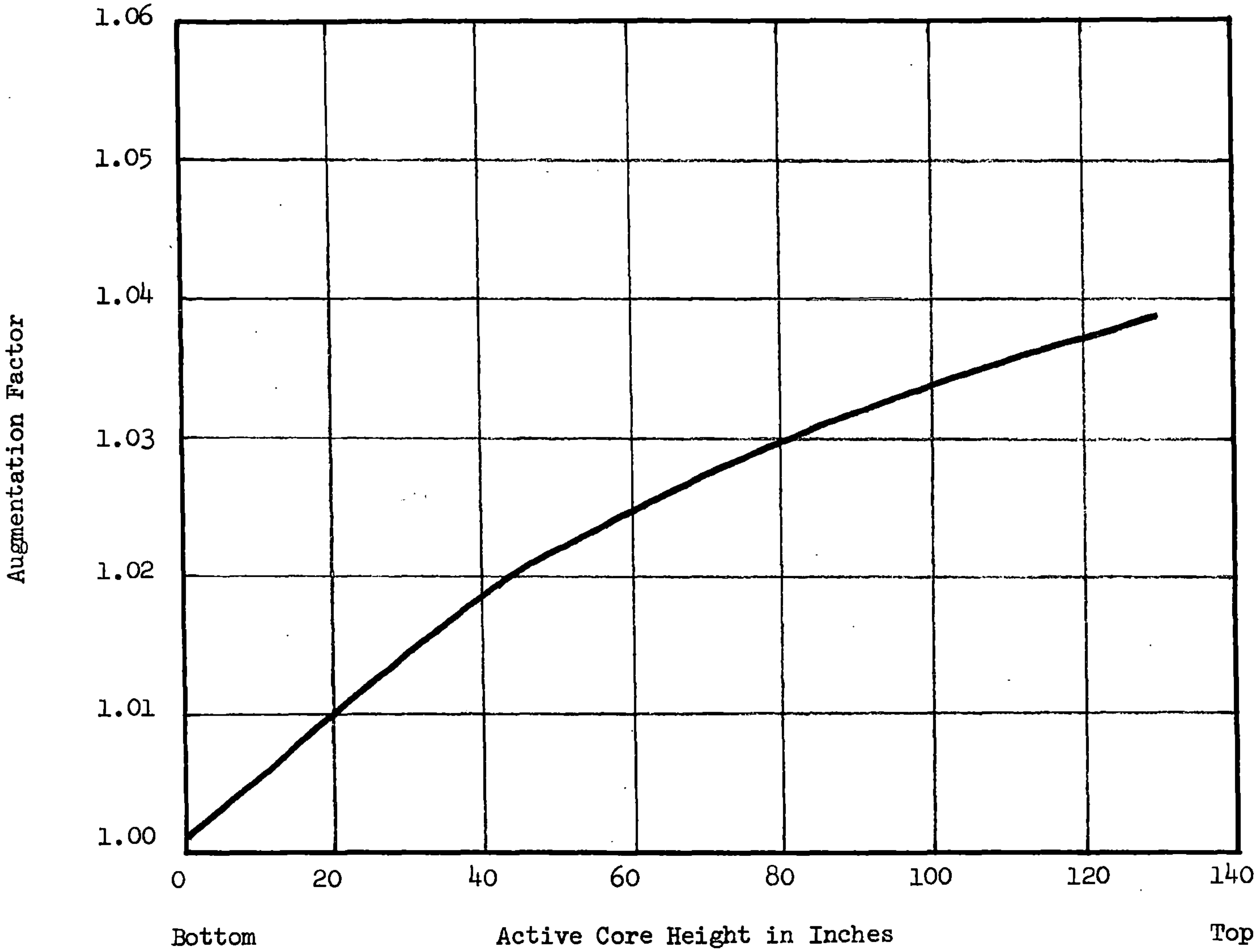


Figure 3 - 6

10 CFR 50.46(b). These criteria assure that under LOCA conditions the temperature and oxidation of the cladding will be controlled such that the uranium dioxide pellets will be maintained in a coolable geometry. These criteria are summarized below:

1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation, including effects of cladding thinning and rupture.
3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The computer codes which predict cladding temperature and oxidation under LOCA conditions must be approved by the US Atomic Energy Commission in accordance with Appendix K of 10 CFR 50. The results of these computer code calculations depend on ECCS performance characteristics and fuel design. Analyses performed with approved codes and techniques for each fuel design type taking into consideration anticipated operating conditions will be kept on file at the plant and at the General Office. These analyses provide safety limits given in terms of allowable linear heat generation rates in kW/ft for each fuel design type.

Appropriate factors for measurement-calculation uncertainty, engineering factor and shortening of the fuel pellet stack are specified to insure that linear heat generation rate limits are not exceeded during steady state operation."

F. Change Table 4.2.1 to read as follows:

"TABLE 4.2.1

Minimum Frequencies for Sampling Tests

		Test	Frequency	FSAR Section Reference
1.	Reactor Coolant Samples	Gross Gamma by Fission Product Monitor	Continuous (5)	None
		Quantitative gamma spectral analysis or gross beta gamma radioactivity analysis by internal proportional counter and qualitative gamma spectral analysis.	3 Times/Week (1)	
		Tritium Radioactivity	Weekly	None
		Chemistry (Cl and O_2)	3 Times/Week	None
		Radiochemical Analysis for $\bar{E}$ Determination	Semiannual (2)	None
2.	Reactor Coolant Boron	Boron Concentration	Twice/Week	None
3.	SIRW Tank Water Sample	Boron Concentration	Monthly	None
4.	Concentrated Boric Acid Tanks	Boron Concentration	Monthly	None
5.	SI Tanks	Boron Concentration	Monthly	6.1.2
6.	Spent Fuel Pool	Boron Concentration	Monthly	9.4
7.	Secondary Coolant	Gas Radioactivity by Air Ejector Gas Monitor	Continuous (6)	None
		Coolant Gross Radioactivity	Twice/Week (6)	None
		Iodine Concentration	Weekly (3)	None
8.	Liquid Radwaste	Radioactivity Analysis	Prior to Release of Each Batch	11.1
9.	Radioactive Gas Decay Tanks	Radioactivity Analysis	Prior to Release of Each Batch	11.1
10.	Stack-Gas Monitor Particulate Samples	Iodine 131 and Particulate Radioactivity	Weekly (4)	11.1

(1) When radioactivity level exceeds 10 percent of limits in Specification 3.1.4, or 3.1.5, the sampling frequency shall be increased to a minimum of once each day.

"TABLE 4.2.1 (Contd)

(2) Redetermine if: (a) The primary coolant radioactivity increases by more than 10 μCi/cc from the previous determination, and (b) upon each start-up only after a two-week equilibrium adjustment period shows a 10 μCi/cc increase from the previous determination in accordance with Specification 3.1.4.

(3) When radioactivity level exceeds 10 percent of limits in Specification 3.1.5, the sampling frequency shall be increased to a minimum of once each day.

(4) When iodine or particulate radioactivity levels exceed 10 percent of limit in Specifications 3.9.6 and 3.9.9, the sampling frequency shall be increased to a minimum of once each day.

(5) A daily sample shall be obtained and analyzed if fission product monitor is out of service.

(6) If the air ejector gas monitor is out of service, the secondary coolant gross radioactivity shall be measured once per day to evaluate steam generator leak tightness."

H. Delete Appendix B "Interim Special Technical Specifications" in its entirety.

II. Discussion

This proposed change to the Palisades Technical Specifications provides for safety limits governing reactor operation based upon the evaluation of the fuel and ECCS in accordance with the Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors as stated in 10 CFR 50.46(b) and Appendix K to 10 CFR 50.

The evaluation was performed by Combustion Engineering (CE) and transmitted to the Director of Regulation by letter dated July 9, 1975. The evaluation is for first core fuel operating at a peak linear heat generation rate of 11.3 kW/ft prior to any cladding collapse. The evaluation results and corresponding ECCS criteria for the worst primary pipe break (1.0x double-ended slot break) are as follows:

Parameter	CE Calculation	Criterion
Peak Clad Temperature	2198°F*	2200°F
Maximum Cladding Oxidation	0.061*	0.17

*These numbers were derived using a containment pressure evaluation based on Branch Technical Position CSB 6-1.

The 11.3 kW/ft linear heat generation rate has been shown to fulfill the requirements of 10 CFR 50.46(b) and Appendix K to 10 CFR 50; it will be used as an operational limit as specified in the proposed Section 2.1.2 of the Technical Specifications until such time as other appropriate calculations are performed.

During the Palisades Plant outage which lasted from August 1973 to September 1974, an interim fuel examination was conducted by Combustion Engineering at the request of Consumers Power Company. One facet of this fuel examination was a gamma scanning program to ascertain the location and size of any fuel column gaps that may have occurred as a result of densification and to determine fuel column lengths in order to clarify the extent of densification.

The gamma scanning equipment was positioned on the spent fuel racks in front of the fuel elevator at an angle of 60° to the south pool well. A sodium iodide crystal was used with a photomultiplier tube and a single channel pulse analyzer.

The analyzer was calibrated to detect peak gamma activity for the $Zr^{95} \rightarrow Nb^{95}$ decay at approximately 0.75 Mev. Calculations had been done to show that the activation of Zr^{95} in the cladding was negligible by comparison to the activity of the Zr^{95} fission product. This decay energy peak, therefore, provided a reliable indication of fuel pellet occupancy in the fuel rod. As a result, interpellet gaps were observed as sharp decreases in the recorded gamma signals. A strip chart recorder was used to record the gamma ray intensities and the movement of the fuel rod as it passed the collimator aperture. The aperture width was set at 0.060".

The four corner rods in each of twenty assemblies were scanned at a rate of 4" per minute. Ten rods were found to have gaps greater than 0.060" wide. A summary of those gaps and gap locations is presented in Table 1. With the exception of the large gap at the bottom of rod A-38 NW, these

gaps are quite small and infrequent. Because of the position of the gap in A-38 NW, it is not likely that this gap is due to densification.

TABLE 1

Summary of Fuel Stack Gaps

Greater Than 0.060"

Rod	Gap Width (Inches)	Gap Location (Inches From Bottom)
A-25 NE	0.680	114
A-12 NE	0.125	28.25
A-22 NE	0.125	36
B-50 SE	0.545 0.370	67.5 129.5
B-50 SW	0.490	65.5
B-01 NW	0.130	55.5
B-01 SW	0.330	127.0
B-67 SW	0.120	99.0
A-35 SE	0.140	130.0
A-38 NW*	2.160	1.25

*Possible Loading Anomaly

The infrequent occurrence of gaps, combined with the small size of the gaps observed, leads Consumers Power Company to the conclusion that a change in the densification and collapse model that has been applied to Palisades is appropriate.

A new model for calculation of flux peaking augmentation factors, based in part upon the data observed during this fuel examination is included in CENPD-139 "Fuel Evaluation Model." The proposed flux peaking augmentation factors are based upon this model.

We further believe that the size of gaps observed in Palisades fuel precludes the type of fuel rod collapse observed in selected other pressurized water reactors. Not a single gap which can be attributed to densification has been observed that is as long as 3/4 of an inch.

Clearly, the stability of small gaps to support collapse must be far superior to the 2" and greater gaps which have resulted in collapse at other pressurized water reactor facilities. Because of the results of the Palisades fuel examination, we do not believe fuel collapse will occur. Therefore, it is not necessary to maintain separate "post-collapse" rules for core average exposure above 10,265 MWd/MTU. We believe that the Appendix K criteria provide more than adequate protection for required fuel integrity in the unlikely event of a Loss of Coolant Accident. It follows from this reasoning that the analysis supplied by Combustion Engineering applies to Palisades fuel at least through the remainder of the first cycle of operation.

10 CFR 50, Appendix K, requires that the evaluations take into account the effects of possible fuel pellet shrinkage. Prior to the issuance of 10 CFR 50, Appendix K, fuel shrinkage was accounted for in calculations that were separate to the Interim Acceptance Criteria. As these calculations were performed after the initial evaluation of the Palisades Plant under the Interim Acceptance Criteria, and at the time the calculations were performed, it was assumed that at some point in the future the need for performing fuel shrinkage calculations would no longer exist or they would be combined with some other criteria; the operating limits associated with fuel shrinkage were kept separate from the original Palisades Technical Specifications. These operating limits were included in Appendix B to the Technical Specifications titled, "Interim Special Technical Specifications." As the evaluation performed in accordance with 10 CFR 50, Appendix K, includes the effects of fuel shrinkage, we have included in this Technical Specifications revision, changes which incorporate the still applicable "Interim Special Technical Specifications" limits into the original Technical Specifications. Requirements which were included in the Interim Special Technical Specifications which we deemed to be no longer applicable, have been deleted. These requirements are associated with postulated clad collapse; namely, limits of rate of power increase, limits on primary system operating pressure, flux peaking augmentation factor and the linear heat rate limits associated with the then postulated post-collapse operation.

The primary to secondary leakage limit, which was associated with collapsed clad, has been incorporated in the original Technical Specifications because of the tube wastage that has been previously experienced at the Palisades Plant. Even though the specific leakage limit is not justifiable solely on a technical basis, it is deemed prudent to continue a limit of this nature in effect until tube wastage has shown to have been halted.

III. Conclusion

This change has not been reviewed by our Palisades Plant Review Committee or the Safety and Audit Review Board; however, based on its similarity to the proposed change submitted by letter dated November 4, 1974, we believe that their review of this proposed change will result in the conclusion that this change does not involve a significant hazards consideration.

Consumers Power Company

R. A. Lamley

R. A. Lamley, Vice President

Sworn and subscribed to before me this 9th day of July 1975.

Sylvia B. Ball

Sylvia B. Ball
Notary Public, Jackson County, Michigan
My commission expires May 18, 1976