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MIDDLE SOUTH UTILITIES SYSTEM

L. V. MAURIN
Vice President Nuclear Operations

March 3, 1983

W3P83-0636
Q-3-A29.15

Director of Nuclear Reactor Regulation
ATTENTION: Mr. G. W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Agency
Washington, D.C. 20555

SUBJECT: Waterford SES 3
Docket No. 50-382
ECCS Reanalysis

Reference: W3P82-4063 from L. V. Maurin to T. M. Novak dated 12/22/82

Dear Sir:

The referenced letter documented LP&L discussion with the NRC which led us to perform an ECCS reanalysis addressing the as-built safety injection tank delivery line flow resistance and the effects of operation of the containment purge system on the minimum ECCS containment backpressure using the CE ECCS evaluation flow blockage model developed in response to the requirements of NUREG-0630. As noted in the referenced letter, we understand that NRC approval of this CE model will support the timely review of our submittal.

Please find enclosed a submittal of a draft amendment to the Waterford 3 FSAR. This draft amendment consists of the results of a revised large break LOCA ECCS performance analysis and updates FSAR Section 6.2.1.5 (minimum containment pressure analysis), Section 6.3.3.1 (introduction and summary), and section 15.6.3.3.1 (large break LOCA). In addition, several references are added to Section 15.6 defining the ECCS performance model revision used in this analysis to conform with the required clad deformation and flow blockage model guidelines of NUREG-0630. This information will be included in our next FSAR Amendment.

Yours very truly,

L V Maurin

L. V. Maurin

LVM/RMF/ssd

cc: E. L. Blake, W. M. Stevenson, J. Wilson (NRC), L. Constable

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Following closure of the MFIV's, there is an inventory of feedwater between the MFWIV and the ruptured steam generator. As the ruptured steam generator depressurizes, this inventory starts to boil. As steam in the line expands, this feedwater inventory is pushed into the steam generator and is boiled off by primary to secondary heat transfer. The expansion of the feedwater inventory into the ruptured steam generator has been considered in the analysis. The expansion is assumed to be isentropic.

6.2.1.4.5 Energy Inventories

An energy balance for the most severe secondary system pipe rupture is provided in Table 6.2-9.

6.2.1.4.6 Additional Information Required for Confirmatory Analyses

The flow area of the main steam lines are as indicated in Table 6.2-1. For the MSLB analysis, the postulated rupture is assumed to occur at the nozzle of one of the steam generators. Therefore, the fL/D from the ruptured unit to the break is conservatively assumed to be zero. In the MSLB analysis, the fL/D from the intact steam generator to the break is assumed to be 10.97 which is related to the flow condition of the 32 in. i.d. pipe at the steam generator nozzle.

Feedwater flow to the ruptured steam generator for the most severe MSLB cases listed in Subsection 6.2.1.1 are shown on Figures 6.2-13c and 6.2-13d.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on the Emergency Core Cooling System

6.2.1.5.1 Introduction and Summary

Appendix K to 10CFR50 provides the required and acceptable features of Emergency Core Cooling System (ECCS) evaluation models.[11] Included in this list is the requirement that the containment pressure assumed in the evaluation of ECCS performance not exceed a pressure calculated conservatively for that purpose. The ECCS performance analysis for Waterford-3 which is presented in Subsection 6.3.3, meets the minimum containment pressure requirement of Reference 11, Appendix K, Paragraph I.D.2. 9

6.2.1.5.2 Method of Calculation

The calculations reported in this section are performed using the large break evaluation model described in Reference 12, which was approved by the NRC in Reference 17. The CEFLASH-4A[13] computer program is used to determine the mass and energy released to the containment during the blowdown phase of a postulated LOCA, and the COMPERC-II[14] computer program is used to determine both the mass and energy released to the containment during the reflood phase and the minimum containment pressure response to be used in the evaluation of the effectiveness of the Emergency Core Cooling System.

~~The similarity which exists for the NSSS of the 34xx MWt reactor plants was recognized (San Onofre 2 and 3, Forked River, Waterford-3, and Pilgrim 2)[15,16], and a generic blowdown calculation was performed for all~~

~~of these reactors. However, due to the differences in containment design among these plants, the minimum containment pressure to be used in the ECCS performance analysis has been calculated specifically for Waterford-3.~~

6.2.1.5.3 Input Parameters

6.2.1.5.3.1 Mass and Energy Release Data

INSERT A

The mass and energy released to the containment for the most severe LOCA, the 0.8xDEG/PD break, is listed as a function of time in Table 6.2-19. ~~As discussed in Subsection 6.2.1.5.2, the mass and energy released during blowdown has been calculated generically for all 34xx MWt plants, and the mass and energy released during reflood is calculated specifically for Waterford-3~~. The quantity of safety injection fluid assumed to spill from the break is discussed in Subsection 6.2.1.5.3.5. 23

6.2.1.5.3.2 Initial Containment Internal Conditions

The initial containment internal conditions which have been assumed for this analysis are:

Temperature - 80 F (Minimum)

Pressure - ~~14.2~~ 14.37 psia (Minimum)

Relative Humidity - 100 percent (Maximum)

For each parameter, the conservative direction with respect to minimizing the containment pressure appears in parentheses.

6.2.1.5.3.3 Containment Volume

The net free containment volume assumed for this analysis is 2,677,000 ft^3.

6.2.1.5.3.4 Active Heat Sinks

For this analysis, it is conservative to maximize the heat removal capacity of the containment active heat sinks; thus, both the containment sprays and all four containment fan coolers are assumed to actuate in the shortest possible time following the break and to operate at their maximum capacity, assuming the minimum temperature of both the stored water and cooling water. To minimize the actuation time, offsite power is assumed to be available for all active heat sinks. (It should be noted that offsite power is assumed to be unavailable for the SIS.)

The Safety Injection System equipment assumed to be operable for this analysis is discussed in Subsection 6.3.3.2.1.

The heat removal rate of the containment fan coolers is shown as a function of containment temperature in Figure 6.2-30a. The operating parameters assumed for the containment sprays are as follows:

Insert A

The mass and energy release during blowdown and reflood has been calculated specifically for Waterford-3. The blowdown calculation was performed using NSSS system design data generic for all 34XX Mwt reactor plants (San Onofre 2 and 3, Waterford-3, and Pilgrim 2) with the exception of safety injection tank flow resistance factors (K-factors). Based on results from Waterford-3 SIT blowdown tests, K-factors specific to the Waterford-3 SIT blowdown lines have been conservatively used for the CEFLASH-4A and COMPERC-II computer program calculations.

Flow - 4180 gpm (Maximum)

Temperature - ~~70~~ 55 F (Minimum)

6.2.1.5.3.5 Steam-Water Mixing

The effect of mixing and condensation of containment steam with spilled ECCS water upon the containment pressure is calculated in the manner described in Section III.D.2 of Reference 12. The effective ECCS spillage rate is shown as a function of time in Figure 6.2-30b.

6.2.1.5.3.6 Passive Heat Sinks

The surface areas and thicknesses of all exposed containment passive heat sinks are listed in Table 6.2-7. To conservatively maximize the heat transfer to these passive sinks, the surface areas have been assumed to be at the maximum of their uncertainty ranges, and their thermal properties (conductivity and heat capacity) have been maximized. The thermal properties assumed for this analysis are:

Material	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)	
Paint - Containment Vessel interior	1.67	26.5	9
Paint - Containment Vessel exterior	1.67	19.75	
Paint - Steel Structure	0.235	49.9	
Paint - Concrete	0.156	47.1	
Carbon Steel	25.9	53.57	
Stainless Steel	9.8	54.0	9
Concrete	1.0	31.9	

6.2.1.5.3.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients between the containment atmosphere and the passive heat sinks have been calculated in the manner described in Section III.D.2 and Figure III.D.2-2 of Reference 12. The variation of the condensing heat transfer coefficients as a function of time is shown quantitatively in Figure 6.2-30c. 9

~~6.2.1.5.3.8 Containment Purge System~~ INSERT B

~~The analysis presented in this subsection has been performed assuming complete containment integrity. The Containment Purge System has, therefore, been assumed not to be operating at the time of the postulated LOCA. Containment purging during normal operation will occur only during a small~~

Insert B

6.2.1.5.3.8 Containment Purge System

The analysis presented in this subsection includes the effects of the containment purge system which is assumed to be operating at the time of the postulated LOCA. The purge system isolation vent lines are 48 inches in diameter. The butterfly-type purge system isolation valves are mechanically limited to a maximum open position of 40°. From this 40° open position, the purge system isolation valves are assumed to be fully closed 5.0 seconds after actuation of the ESF relays in response to the high containment pressure (18 psia) signal. It is conservatively assumed that only dry air is removed from the containment atmosphere.

~~percentage of the calendar year. The chance of a very low probability accident such as a postulated LOCA occurring during purging operations would, therefore, be even smaller.~~

6.2.1.5.4 Results

For the most severe LOCA, the 0.8xDEG/PD break, the minimum containment 23
pressure response is shown in Figure 6.2-31a. As required by 10CFR50 9
Appendix K, the containment pressure used in the ECCS performance evalua-
tion does not exceed this pressure (see Figure 15.6-~~107~~). The responses 23
of the containment atmosphere and SIS (recirculation) sump temperatures are
shown in Figures 6.2-31b and 6.2-31c, respectively. The containment 9
pressure responses for breaks other than the worst break are presented
in the figures in Subsection 15.6.3.3.

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TABLE 6.2-19
WATERFORD 3
BLOWDOWN AND REFLOOD MASS AND ENERGY RELEASE DATA
0.8 DEG/PD

Time (sec)	Mass Flow (lbm/sec)		Energy Release (BTU/sec)		Integral of Mass Flow (lbm)		Integral of Energy Rel (BTU)	
0	0		0		0		0	
.05	7.8158	$\times 10^4$	4.3144	$\times 10^7$	3.0860	$\times 10^3$	1.7017	$\times 10^6$
.10	7.2280		3.9883		6.8735	$\times 10^3$	3.7923	
.15	7.4455		4.1085		1.0589	$\times 10^4$	5.8427	
.20	7.3490		4.0561		1.4296		7.8885	
.25	7.4343		4.1079		1.8008		9.9386	$\times 10^6$
.35	7.2477		4.0101		2.5315		1.3979	$\times 10^7$
.45	7.1457		3.9570		3.2507		1.7960	
.60	7.1030		3.9366		4.3182		2.3875	
.80	7.0232		3.8955		5.7322		3.1714	
1.0	6.9711		3.8701		7.1304		3.9473	
1.4	6.7457		3.7546		9.8787	$\times 10^4$	5.4149	
1.8	6.2446		3.4848		1.2477	$\times 10^5$	6.9231	
2.2	5.5424		3.0984		1.4841		8.2435	
2.6	5.0841		2.8497		1.6957		9.4280	$\times 10^7$
3.0	4.8849		2.7481		1.8939		1.0541	$\times 10^8$
3.4	4.6788		2.6449		2.0857		1.1622	
3.8	4.5061		2.5673		2.2693		1.2664	
4.4	4.0764		2.3675		2.5274		1.4146	
5.2	3.4701		2.1035		2.8287		1.5932	
6.0	3.0157		1.8955		3.0875		1.7531	
6.8	2.6896		1.7225		3.3146		1.8974	
7.6	2.4949		1.5949		3.5204		2.0294	
8.4	2.4069		1.5184		3.7167		2.1540	
9.2	2.2829		1.4335		3.9045		2.2721	
10.0	2.1177	$\times 10^4$	1.3371	$\times 10^7$	4.0808	$\times 10^5$	2.3830	$\times 10^8$

Time	Mass Flow		Energy Release		Integral of Mass Flow		Integral of Energy Rel.	
(sec)	(lbm/sec)		(BTU/sec)		(lbm)		(BTU)	
11.0	1.8876	$\times 10^4$	1.2120	$\times 10^7$	4.2813	$\times 10^5$	2.5105	$\times 10^8$
12.0	1.6347	↓	1.0863	$\times 10^7$	4.4579	↓	2.6255	↓
13.0	1.2134	$\times 10^4$	9.3528	$\times 10^6$	4.6027		2.7269	
14.0	7.2684	$\times 10^3$	7.4923	↓	4.6961		2.8107	
15.0	6.8125	↓	6.0932		4.7654		2.8779	
16.0	4.4891		4.6906		4.8212		2.9318	
17.0	3.0473		3.4461		4.8589		2.9723	
18.0	2.5587		2.7927		4.8862		3.0032	
19.0	3.0987		2.9197		4.9143		3.0323	
20.0	3.7365		2.7633		4.9484		3.0608	
21.0	3.8978		2.3402		4.9871		3.0865	
22.0	3.7667		1.9115	$\times 10^6$	5.0234		3.1067	
23.0	1.4441	$\times 10^3$	5.2159	$\times 10^5$	5.0489		3.1199	
24.0	0		0		5.0586		3.1252	
25.0	0		0		5.0586		3.1252	
26.0	0		0		5.0586		3.1252	
27.0	0		0		5.0586		3.1252	
27.8	0		0		5.0586	$\times 10^5$	3.1252	$\times 10^8$
← Time of Annulus Downflow								
Start of Reflood (Values below are for steam only)								
45.8	0		0		5.0586	$\times 10^5$	3.1252	$\times 10^8$
55.8	0		0		5.0586	↓	3.1252	↓
65.8	0		0		5.0586		3.1252	
75.8	0		0		5.0586		3.1252	
85.8	0		0		5.0586		3.1252	
95.8	0		0		5.0586	$\times 10^5$	3.1252	$\times 10^8$

Time (sec)	Mass Flow (lbm/sec)		Energy Release (BTU/sec)		Integral of Mass Flow (lbm)		Integral of Energy Rel (BTU)	
105.8	0		0		5.0586	$\times 10^5$	3.1252	$\times 10^8$
125.8	1.6640	$\times 10^2$	2.1718	$\times 10^5$	5.0696	↓	3.1395	↓
145.8	1.6142	↓	2.1069	↓	5.1022		3.1821	
165.8	1.8361		2.3965		5.1347		3.2246	
185.8	1.8509		2.4158		5.1715		3.2726	
205.8	1.8562		2.4227		5.2085		3.3209	
225.8	1.8660		2.4355		5.2457		3.3694	
245.8	1.8698		2.4405		5.2830		3.4181	
265.8	1.8777		2.4508		5.3204		3.4669	
285.8	1.8800		2.4537		5.3581		3.5161	
305.8	1.8905		2.4675		5.3957		3.5652	
325.8	1.8855		2.4609		5.4334		3.6143	
345.8	1.8970		2.4760		5.4711		3.6636	
365.8	1.8912		2.4684		5.5089		3.7129	
385.8	1.8969		2.4758		5.5468		3.7624	
405.8	1.9076		2.4898		5.5848		3.8119	
445.8	1.9006		2.4807		5.6609		3.9113	
485.8	1.9207		2.5069		5.7374		4.0112	
525.8	1.9291		2.5179		5.8142		4.1115	
565.8	1.9293		2.5182		5.8913		4.2121	
605.8	1.9112	↓	2.4945	↓	5.9683	↓	4.3125	↓
645.8	1.9108	$\times 10^2$	2.4940	$\times 10^5$	6.0452	$\times 10^5$	4.4129	$\times 10^8$

FIGURE 6.2-30B

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

EFFECTIVE SPRAY AND SPILLAGE TO CONTAINMENT

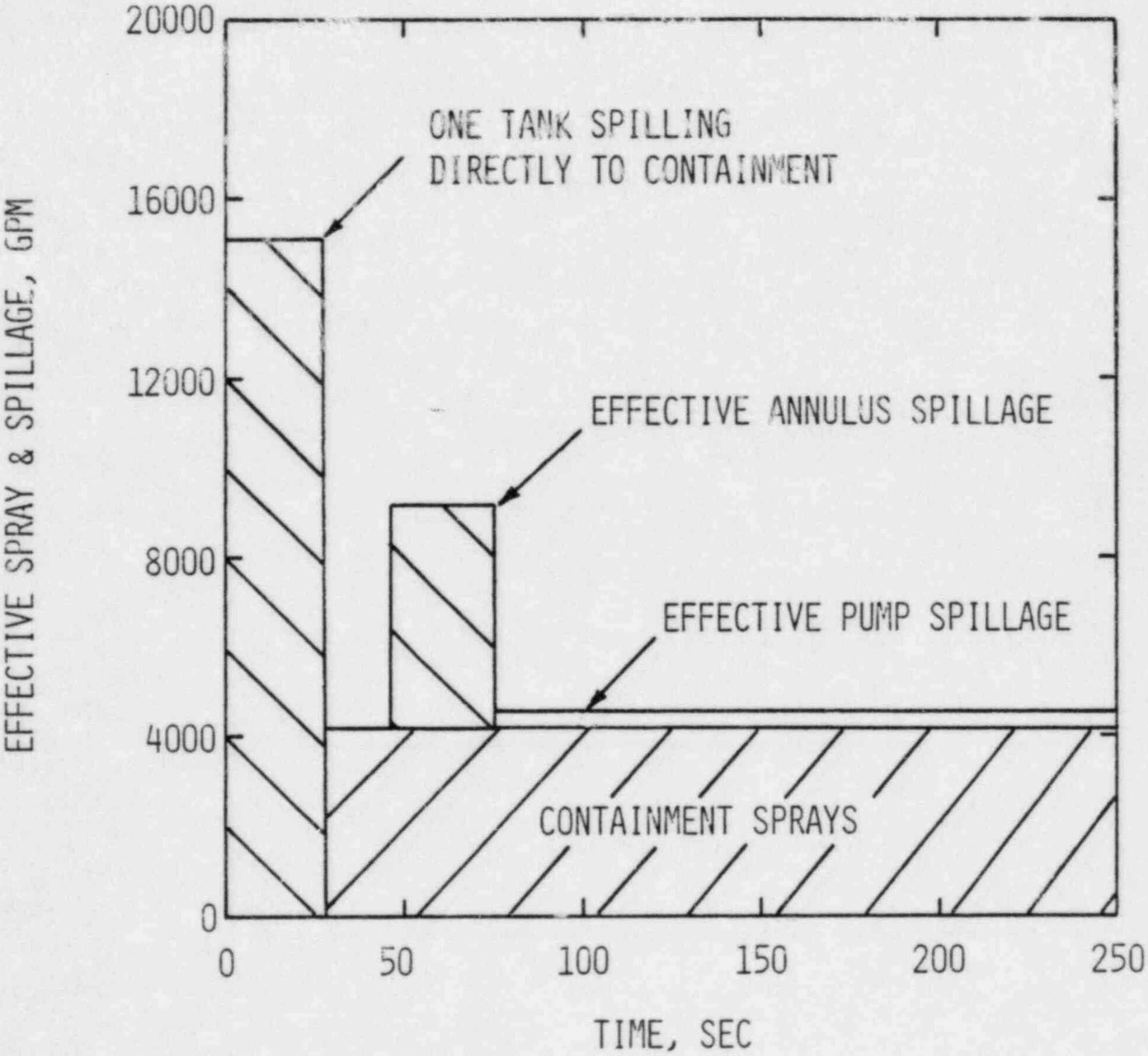


FIGURE 6.2-31A
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MINIMUM CONTAINMENT PRESSURE

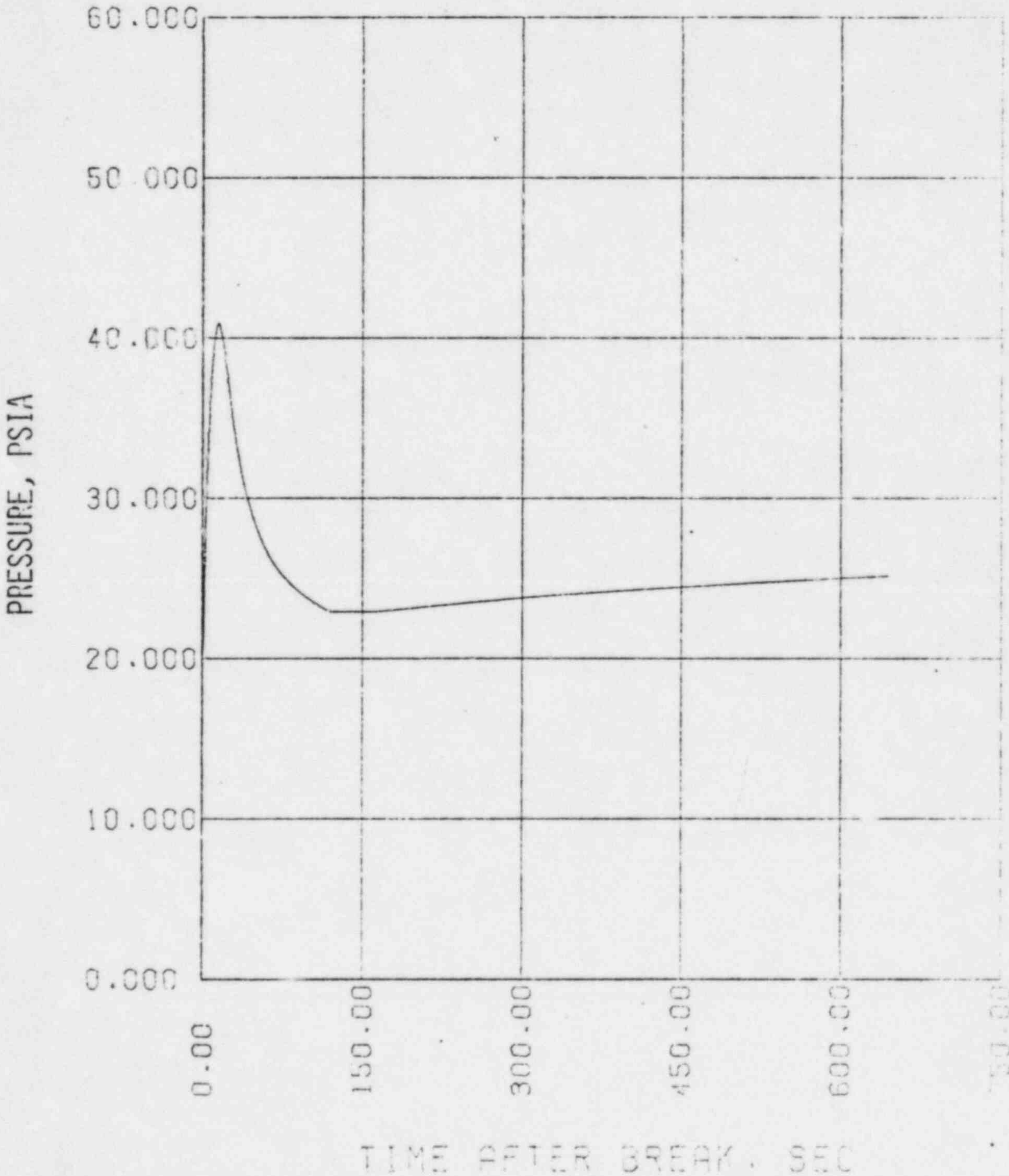


FIGURE 6.2-31B
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
TEMPERATURE, CONTAINMENT ATMOSPHERE

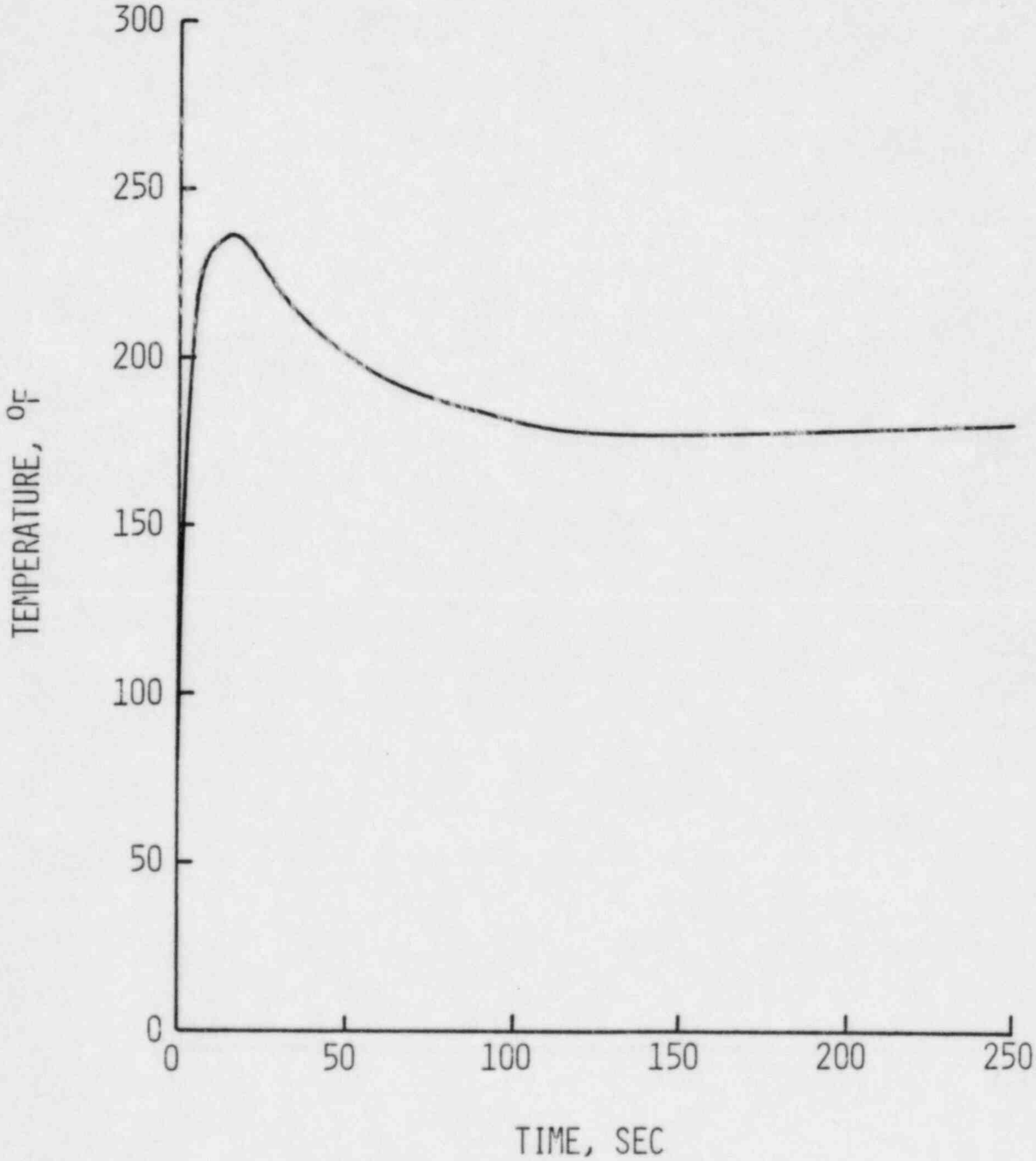
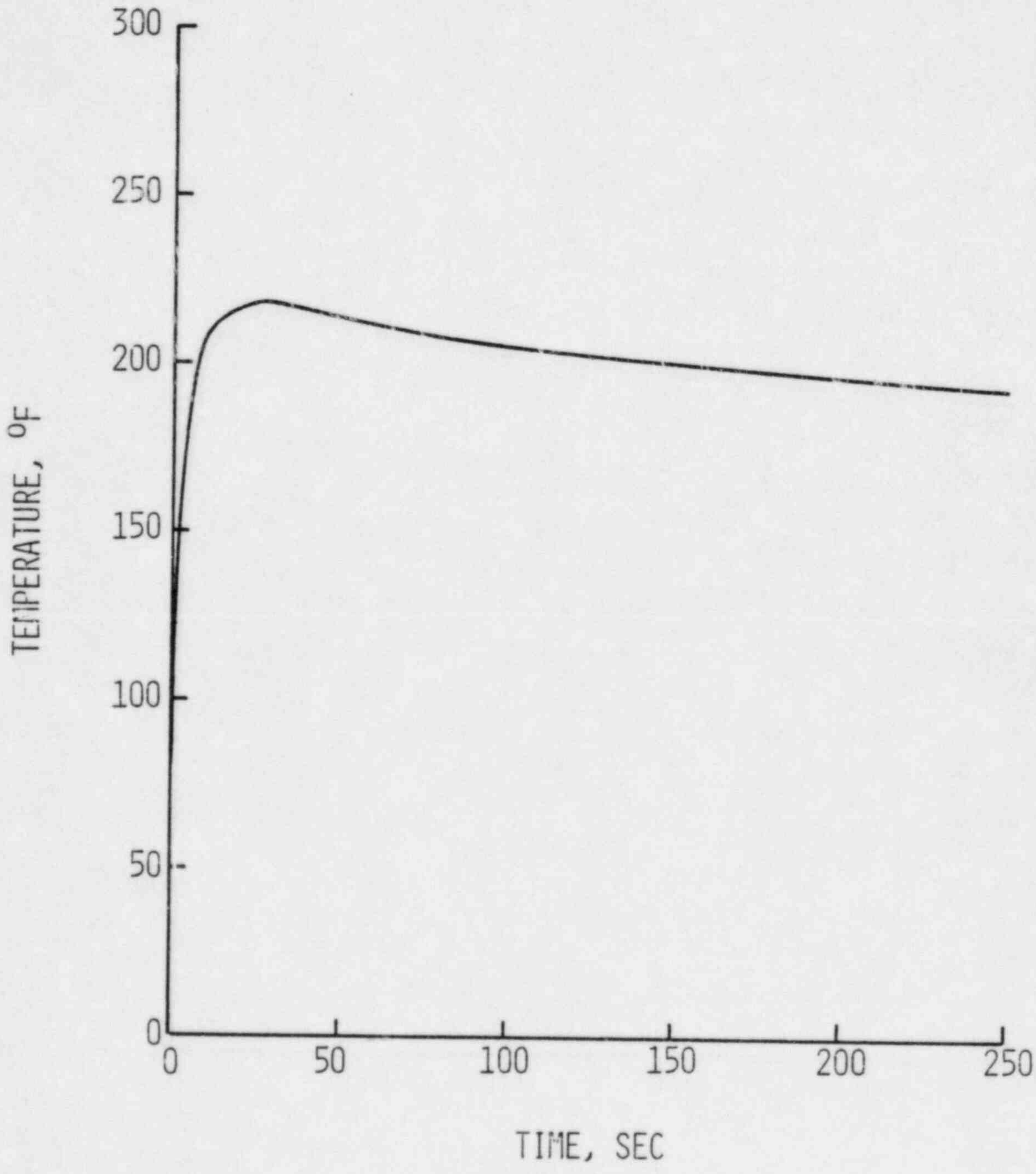


FIGURE 6.2-31C
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
TEMPERATURE, SUMP



The worst single failure of an active component for a large break is the failure of a LPSI pump. For conservatism, it is assumed that the operating LPSI pump is connected to the broken leg and to one intact leg. As explained above, the flow to the broken loop is the same as that to the intact loop.

In the Small Break LOCA Evaluation the pressure between the three (3) intact legs and the broken leg is no greater than about 5 psi. Therefore, all lines see essentially the same back pressure and the flow is split evenly between them.

The worst single failure for a small break is the failure to start of one diesel generator. Therefore, only one HPSI and one LPSI pump will operate. It is assumed that the operating LPSI pump is connected to the broken leg and to one intact leg. As in the large break, the flow to the broken loop is the same as that to the intact loop.

6.3.3 PERFORMANCE EVALUATION

6.3.3.1 Introduction and Summary

The acceptance criteria for Emergency Core Cooling Systems for Light Water Cooled Reactors are set forth in 10CFR 50.46 (Reference 1). The analyses presented in this subsection and in Subsection 15.6.3.3 demonstrate that the Waterford-3 ECCS design satisfies these criteria.

The ECCS performance was evaluated for a spectrum of break sizes ranging from a full double ended guillotine break to a 0.01 ft^2 break. At 13.7 Kw/ft, the break yielding the highest peak clad temperature and local clad oxidation was identified as the 0.8xDEG/PD[a]. 23

INSERT C

(a) DEG/PD = Double Ended Guillotine at the Pump Discharge. 23

Insert C

The ECCS performance was reevaluated for the 0.8 x DEG/PD break at 13.4 Kw/ft using actual flow resistance K-factors for the SIT injection lines and assuming that the containment purge system is in operation at the time of the postulated LOCA.

The results of the ECCS performance analyses show that the plant meets the 10CFR50.46 Acceptance Criteria at a peak linear heat generation rate of ~~13.7~~ 13.4 Kw/ft. Conformance is summarized as follows: || 23

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

The analysis yielded a peak clad temperature of ~~2118~~ 2188 F for the 0.8xDEG/PD BREAK. || 23

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

The analysis yielded a local peak clad oxidation percentage of ~~16.69~~ 8.7 for the 0.8xDEG/PD BREAK. || 23

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 one percent 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."

The analysis yielded a peak core-wide oxidation of ~~[illegible]~~ less than 0.805 for the ~~1.0~~ 0.8xDEG/PD BREAK. || 23

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 evaluation model (4) 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 swelling and 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 intitial operation of the ECCS, the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

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15.6.3.3 Loss of Coolant Accident (LOCA)

15.6.3.3.1 Identification of Causes and Frequency Classification

The estimated frequency of a LOCA classifies it as a limiting fault as defined in Reference 1 of Section 15.0. A LOCA is defined as a hypothetical break in a pipe in the reactor coolant pressure boundary resulting in the loss of reactor coolant at a rate in excess of the capability of the coolant makeup system.[2] For this analysis, the particular breaks assumed are described in Subsections 6.3.3.2.3 and 6.3.3.3.3. 9

15.6.3.3.2 Sequence of Events and Systems Operations

The transient behavior during a LOCA is as follows. During the blowdown phase, the primary system depressurizes as primary coolant is ejected through the break into the containment, and the reactor is shutdown either by moderator voiding, or by CEA insertion. Following depressurization, 9
emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core. When the core has been completely recovered, the long-term cooling mechanisms described in Subsection 6.3.3.4 will maintain acceptable core temperatures until the plant is secured.

The sequence of important events which occur in the short-term is listed in Table 15.6-12 for large-break LOCAs and in Table 15.6-12a for small break LOCAs. 9
The sequence of events for long-term cooling is discussed in Subsection 6.3.3.4.

15.6.3.3.3 Core and System Performance

15.6.3.3.3.1 Large Break LOCA

← INSERT ***

15.6.3.3.3.1.1 Mathematical Model 9

The spectrum of large break calculations reported in this section are performed using the CE large break evaluation model described in References 3 and 4. In the CE model, the CEFLASH-4A[1] 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 determine the system behavior during the refill and reflood phases. The core flow and thermodynamic parameters from these two codes are used as input in the STRIKIN-II[6] program, which is used to calculate the hot rod clad temperature transient and peak local clad oxidation percentage. Except for the 0.5 ft^2 S/PD BREAK, the steam cooling heat transfer coefficients calculated by the PARCH code[7] were used for the time interval during which the reflood rate was less than 1.0 inch/second. For the 0.5 ft^2 S/PD BREAK, a minimum steam cooling heat transfer coefficient of 5.0 Btu/hr-ft^2-F was used for conservatism, as 23
described in CENPD-132, Supplement 1 (Reference 3) for the same time interval. The STRIKIN-II version used is identified in CENPD-135, Supplement 5 (Reference 6). The core-wide clad oxidation percentage is obtained from the results of both the STRIKIN-II and COMZIRC[5, Suppl. 1] computer programs.

INSERT D

Insert ***

The large break LOCA calculations described in the following subsections pertain to both a spectrum of large breaks and to a worst break (0.8 x DEG/PD) reanalysis. The spectrum analysis contains consistent base data, input, and mathematical models. The worst break reanalysis utilized containment purge data, actual SIT discharge line flow resistance K-factor input, and the latest C-E ECCS Evaluation Model Flow Blockage Analysis in Reference 13. Based on the spectrum results, which show that containment pressure and SIT injection play a relatively minor role in differences in PCT for different break sizes or break locations, it is concluded that only the worst break (0.8 x DEG/PD) need be reanalyzed due to the effects of containment purge and changes in SIT discharge line K-factors.

Insert D

The worst break (0.8 x DEG/PD) ECCS reanalysis reported in this section is performed using the C-E ECCS Evaluation Model Flow Blockage Analysis described in Reference 13. In this C-E model, new rupture temperature, rupture strain, and flow blockage models, adopted from NUREG-0630 (Reference 14), are used in the STRIKIN-II and PARCH codes. Also the steam cooling heat transfer coefficients calculated by the PARCH code, for use during the less than 1.0 inch/second reflood rate time interval, are calculated using an explicit method for redistribution of steam flow around the blockage region, described in Reference 13. The core-wide clad oxidation percentage is obtained from the COMZIRC results of the spectrum analysis. Improvements in steam cooling heat transfer using the model of Reference 13, reduced the amounts of cladding oxidation such that the results of the spectrum analysis are bounding.

REFERENCES (Cont'd)

7. CENPD-138, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" (Proprietary).

CENPD-138, Supplement 1, "PARCH, A FORTRAN IV Digital Program to Evaluate Pool Boiling Axial Rod and Coolant Heating" (Modifications), February, 1975.

CENPD-138, Supplement 2, "PARCH, A FORTRAN IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup", January, 1977 (Proprietary). 9

8. J. J. DiNunno, et. al., "Calculation of Distance Factors for Power and Test Reactor Sites, "TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C., 1962.

9. "Calculative Methods for the CE Small Break LOCA Evaluation Model", CENPD-137, August, 1974 (Proprietary).
"Calculative Methods for the CE Small Break LOCA Evaluation Model", CENPD-137, Supplement 1, January 1977 (Proprietary).

10. Letter, O.D. Parr (NRC) to F.M. Stern (CE), June 13, 1975. 9

11. Letter, K Kniel (NRC) to A.E. Scherer (CE), September 27, 1977.

12. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", 1973.

13. Enclosure 1-P to LD-81-095, "C-E ECCS Evaluation Model Flow Blockage Analysis," December, 1981 (Proprietary).

14. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analyses," NRC report NUREG-0630, April, 1980.

TABLE 15.6-12

TIME SEQUENCE OF IMPORTANT EVENTS FOR LARGE LOCA
(SECONDS AFTER BREAK)

Break	SI Tanks On	End of Bypass	Start of Reflood	SI Tanks Empty	SI Pumps On	Hot Rod Rupture	
1.0 DES/PD[a]	13.7	19.8	32.8	76.1	76.1	63.1	
0.8 DES/PD	13.7	20.0	33.0	76.3	76.3	63.3	
0.6 DES/PD	14.8	21.2	34.3	77.6	77.6	72.9	
0.5 ft^2 S/PD	134.0	142.88	155.7	201.3	201.3	220.5	23
1.0 DEG/PD	13.6	19.7	32.7	76.2	76.2	60.5	
0.8 DEG/PD	13.9	20.1	33.1	76.5	76.5	59.0	
0.6 DEG/PD	15.5	21.9	35.0	78.3	78.3	70.9	
0.8 DEG/PD[b]	13.9	24.9	45.8	119.3	119.3	43.1	

(a) See Table 15.6-15 for an explanation of these abbreviations.

(b) Worst break reanalysis with containment purging and actual SIT discharge line flow resistance K-factors.

TABLE 15.6-13

GENERAL SYSTEM PARAMETERS AND INITIAL CONDITIONS
(LARGE BREAK ECCS ANALYSIS)

Quantity		Value	
Reactor power level, MWt (102% of nominal)		3,458	23
Average linear heat rate, kW/ft (102% of nominal)		5.6	
Peak linear heat rate, kW/ft	13.4	13.7	23
Gap conductance at peak linear heat rate[a] Btu/hr-ft^2-F	1,406	1,447	
Fuel centerline temperature at peak linear heat rate[a] F	3,319.7	3,377.8	23
Fuel average temperature at peak linear heat rate[a] F	2,131.4	2,157.6	23
Hot rod gas pressure[a], psia		1,113.3	23
Moderator temperature coefficient at initial density, $\Delta\rho$ /F		$+0.5 \times 10^{-4}$	23
System flowrate (total), lb/hr		148.0×10^6	
Core flowrate, lb/hr		144.15×10^6	23
Initial system pressure, psia		2,250	
Core inlet temperature, F		557.5	
Core outlet temperature, F		618.6	
Active core height, ft.		12.5	
Fuel rod OD, in.		0.382	
Number of cold legs		4	
Number of hot legs		2	
Cold leg diameter, in.		30	
Hot leg diameter, in.		42	

a. These quantities correspond to the burnup (676 MWD/MTU, hot rod average) yielding the highest peak clad temperature. 23

TABLE 15.6-13 (Cont'd)

Quantity	Value
Safety injection tank pressure, psia	609
Safety injection tank gas/water volume, ft^3	~~575~~ 572/1,679

TABLE 15.6-14

PEAK CLAD TEMPERATURES AND OXIDATION PERCENTAGES FOR THE LARGE BREAK SPECTRUM

Break	Peak Clad Temperature[a] (F)	Clad Oxidation (%) Local[b]	Clad Oxidation (%) Core-Wide[c]	
				23
1.0 DES/PD	2,107	16.2	0.787	
0.8 DES/PD	2,108	16.2	0.796	
0.6 DES/PD	2,092	15.3	0.740	
0.5 ft^2 S/PD	2,049	14.2	0.393	23
1.0 DEG/PD	2,115	16.6	0.805	
0.8 DEG/PD	2,118	16.7	0.805	
0.6 DEG/PD	2,094	15.1	0.685	
0.8 DEG/PD[d]	2188	8.7	<0.805	

(a) Acceptance Criteria is ≤2200 F 23

(b) Acceptance Criteria is ≤17%

(c) Acceptance Criteria is ≤1.0%

(d) Worst break reanalysis with containment purging and actual SIT ~~discharge~~ discharge line flow resistance K-factors.

TABLE 15.6-15

LARGE BREAK SPECTRUM

Break, Size, Type and Location	Abbreviation	Figure	
1.0 x double-ended slot break in pump discharge leg	1.0 x DES/PD	15.6-36 through 15.6-44	
0.8 x double-ended slot break in pump discharge leg	0.8 x DES/PD	15.6-45 through 15.6-53	
0.6 x double-ended slot break in pump discharge leg	0.6 x DES/PD	15.6-54 through 15.6-62	23
0.5 ft^2 slot break in pump discharge leg	0.5 ft^2 S/PD	15.6-83 through 15.6-91	
1.0 x double-ended guillotine break in pump discharge leg	1.0 x DEG/PD	15.6-92 through 15.6-100	
0.8 x double-ended guillotine break in pump discharge leg	0.8 x DEG/PD	15.6-101 through 15.6-109k	23
0.6 x double-ended guillotine break in pump discharge leg	0.6 x DEG/PD	15.6-110 through 15.6-118	23
Peak clad temperature vs. break area		15.6-128	
[illegible]			
0.8 x double-ended guillotine break in pump discharge leg[a]	0.8 x DEG/PD	15.6-186 through 15.6-205	

(a) Worst break reanalysis with containment purging and actual SIT discharge line flow resistance K-factors.

FIGURE 15.6-128

PEAK CLAD TEMPERATURE vs. BREAK AREA

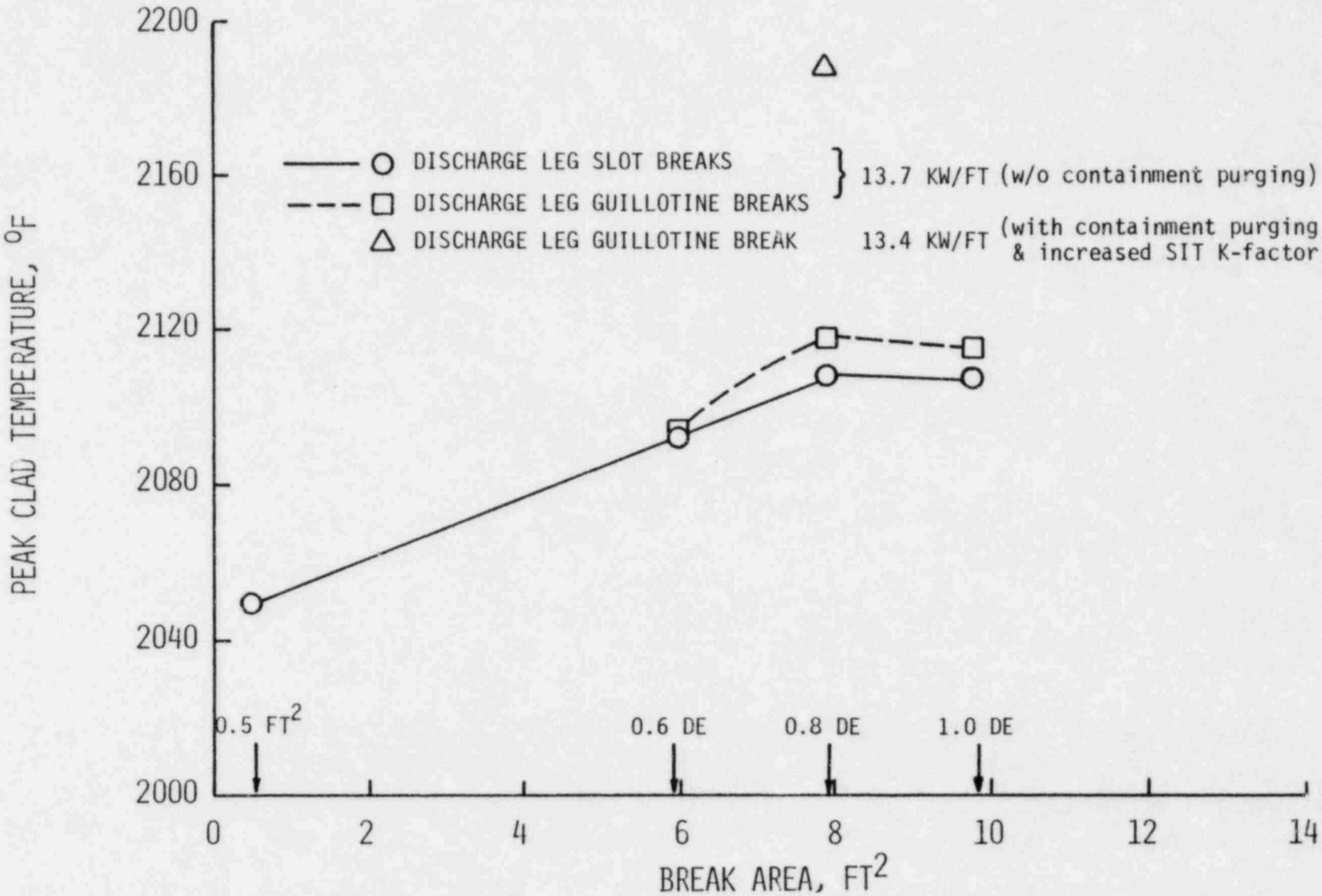

FIGURE 15.6-186
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE POWER

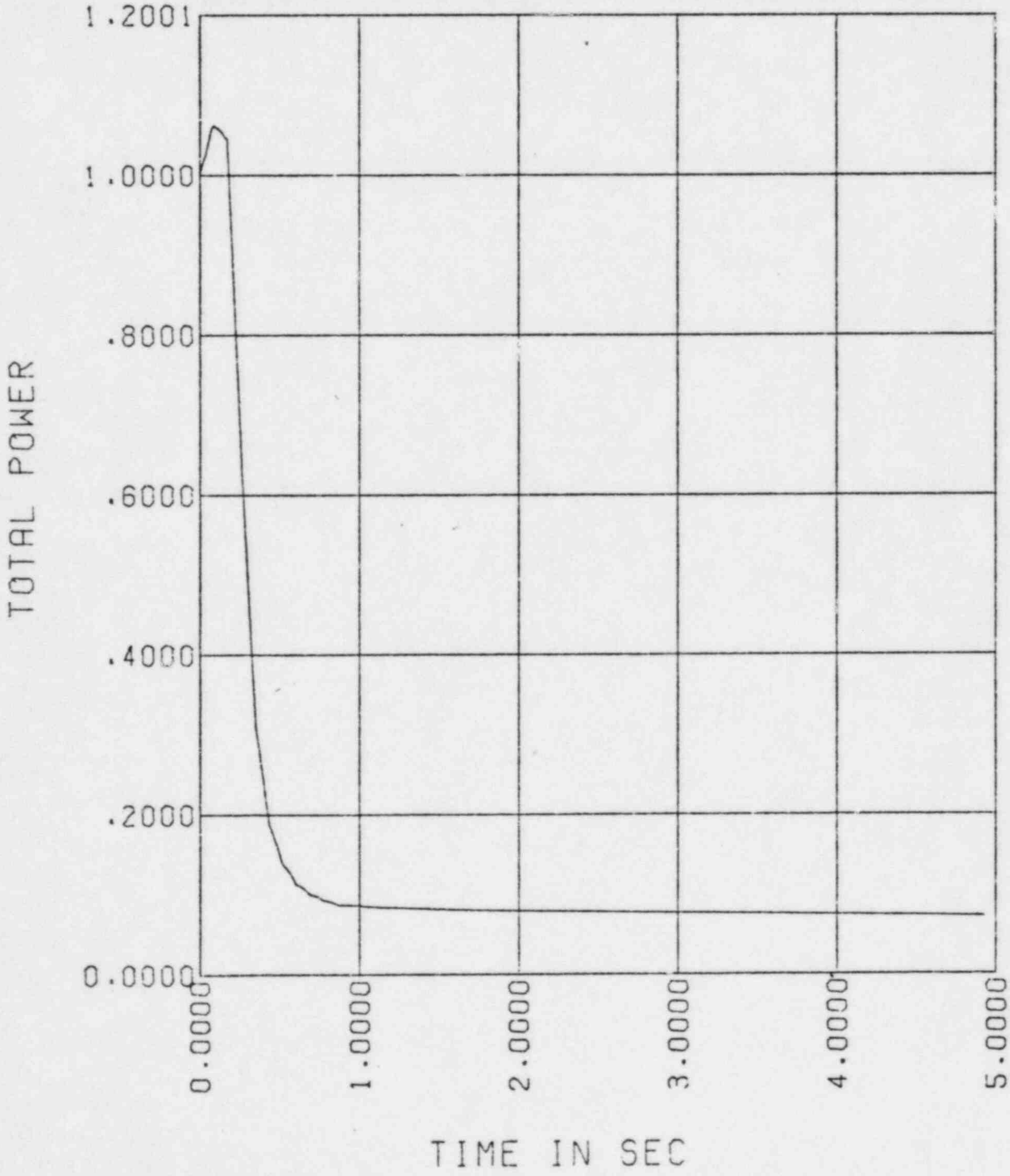

FIGURE 15.6-137
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
PRESSURE IN HOT ASSEMBLY NODE

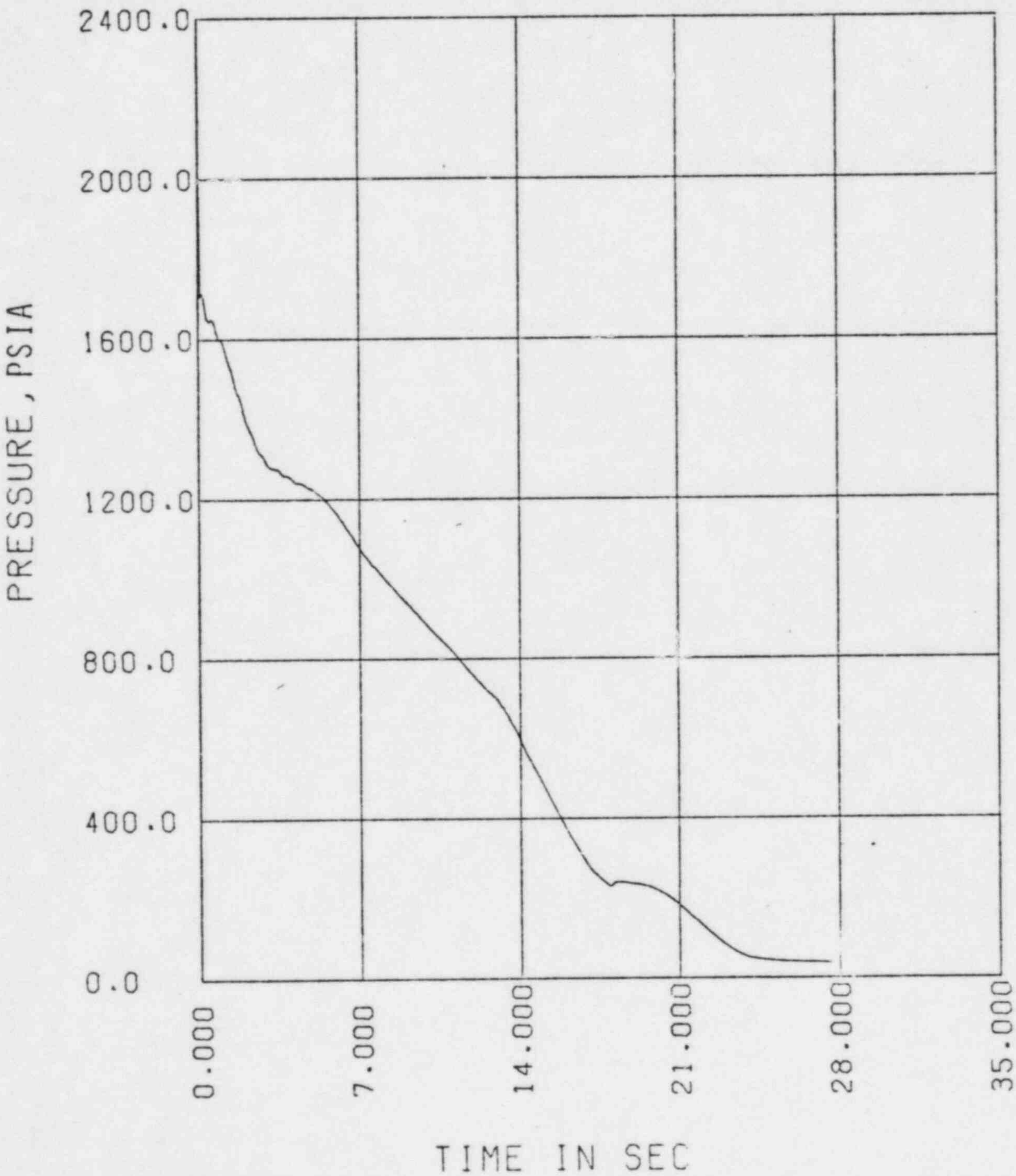

Figure 15.6-188
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
LEAK FLOW

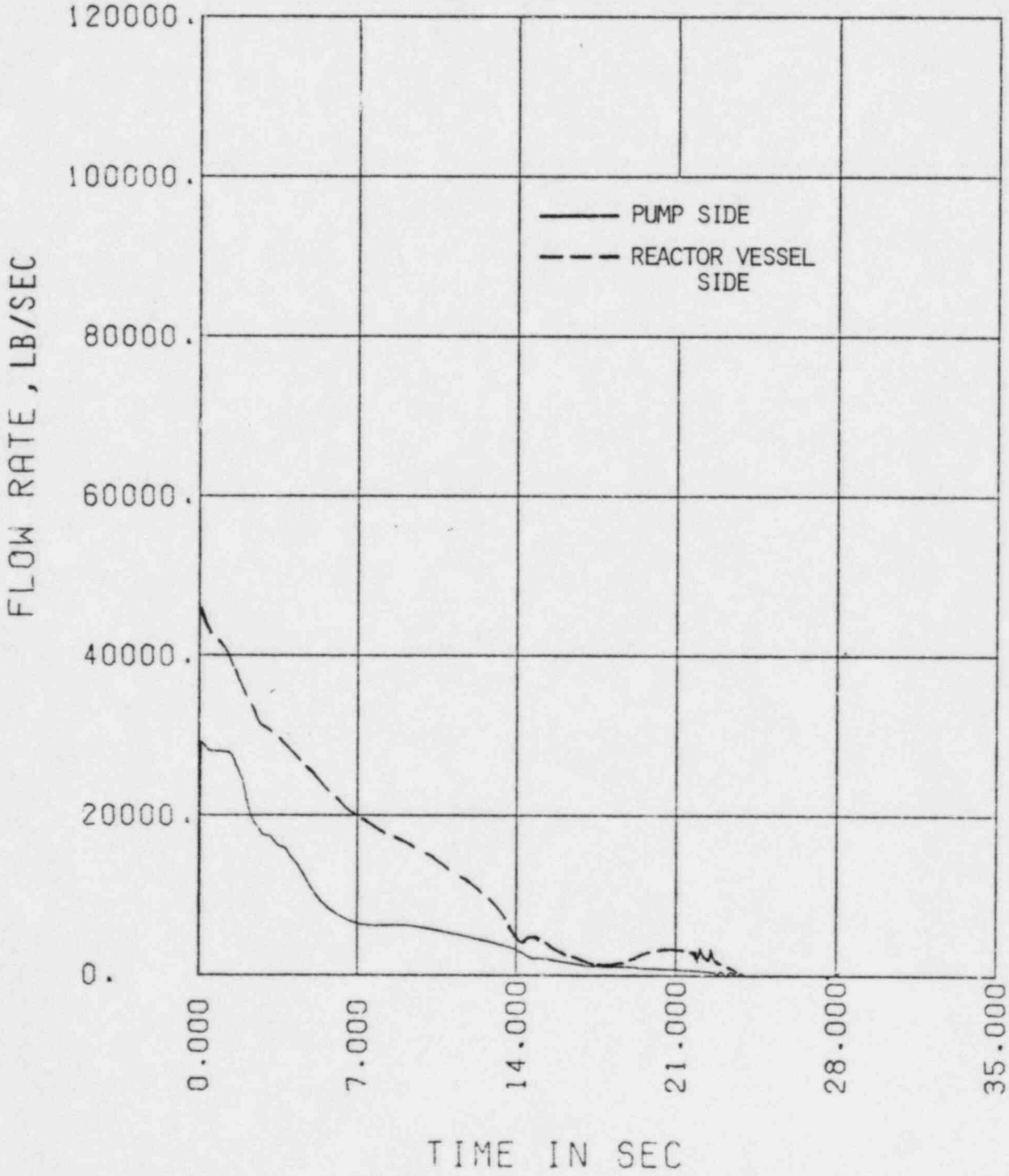

FIGURE 15.6-189
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 16, BELOW HOT SPOT

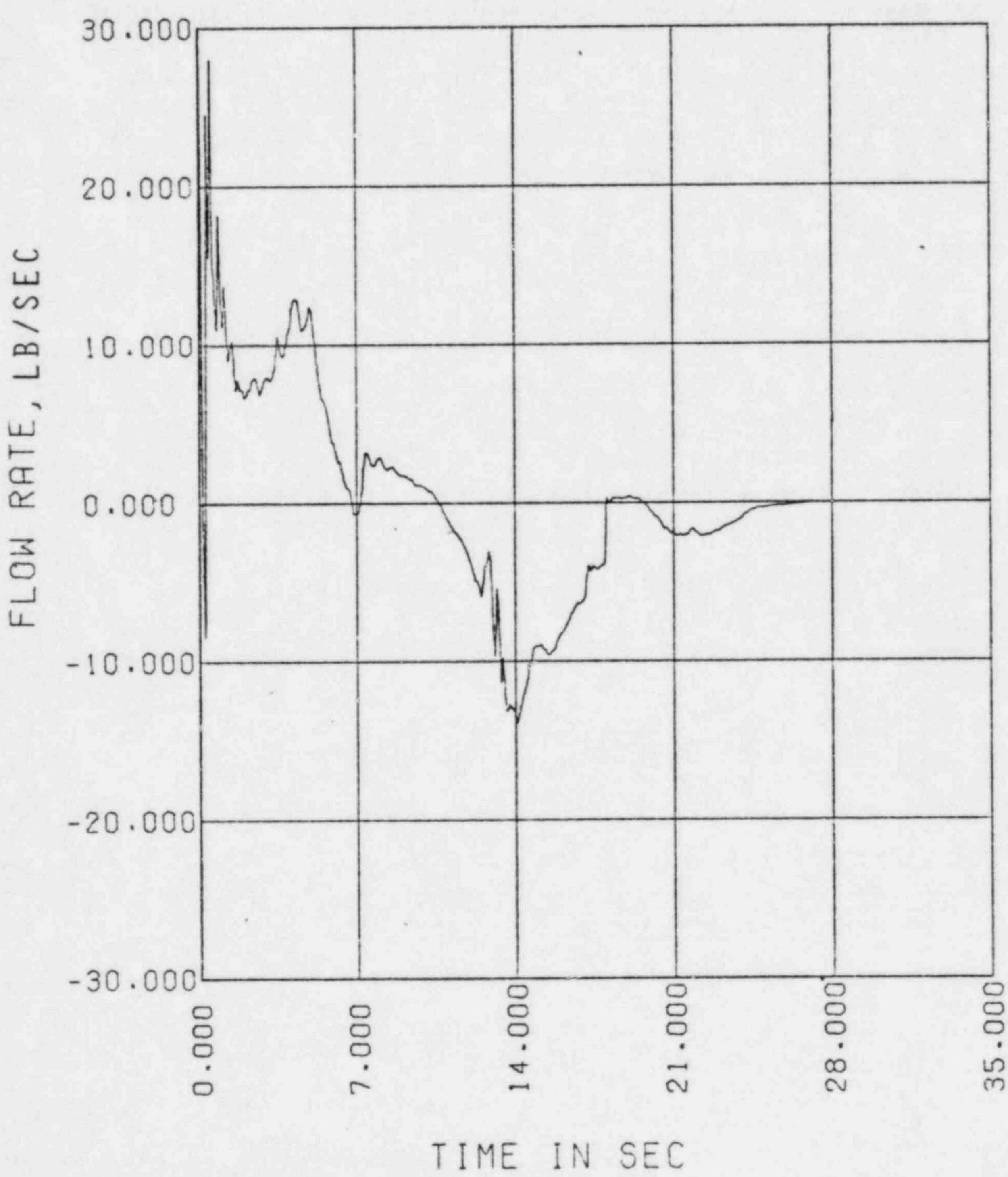

FIGURE 15.6-190
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
FLOW IN HOT ASSEMBLY - PATH 17, ABOVE HOT SPOT

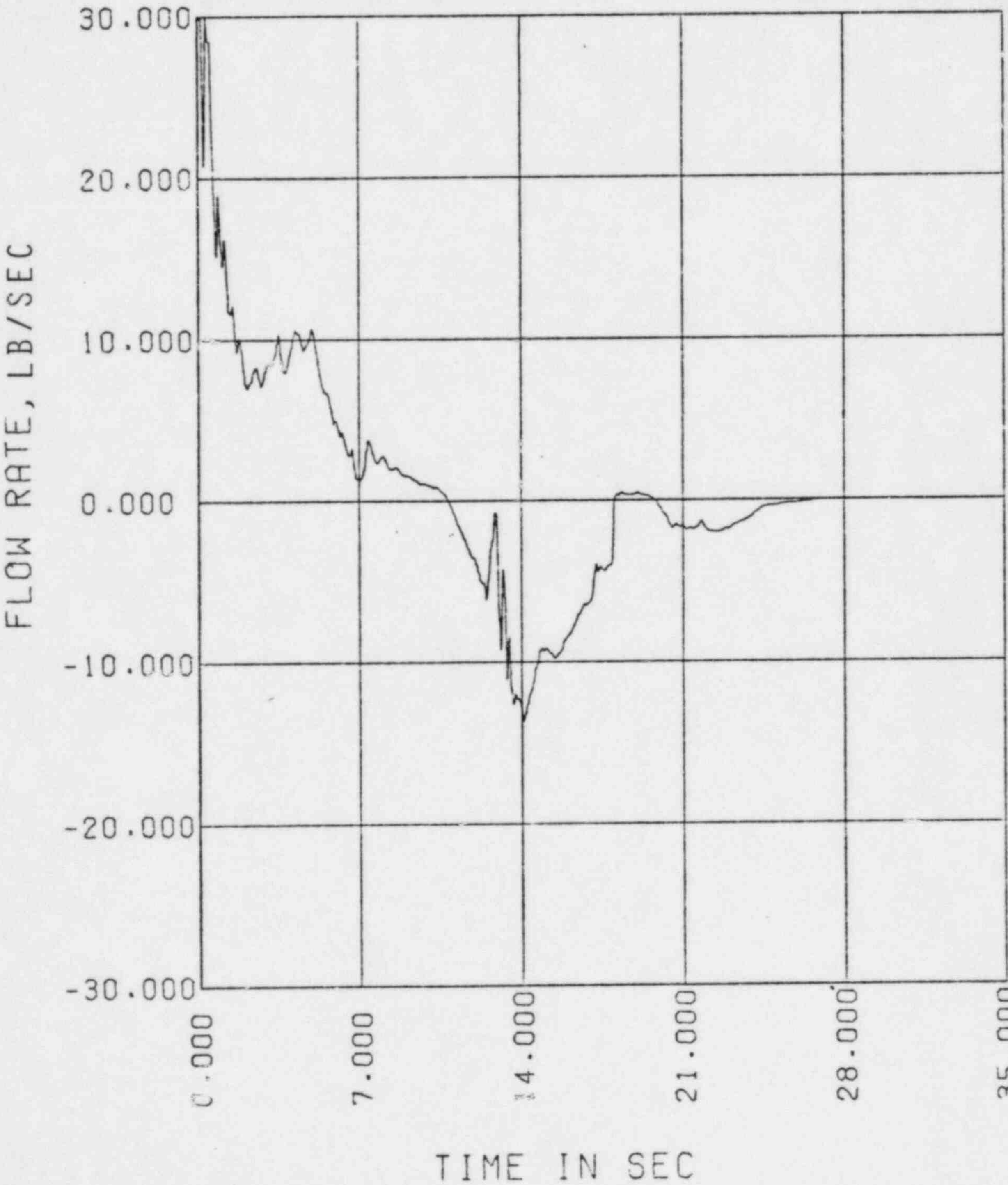

FIGURE 15.6-191
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
HOT ASSEMBLY QUALITY

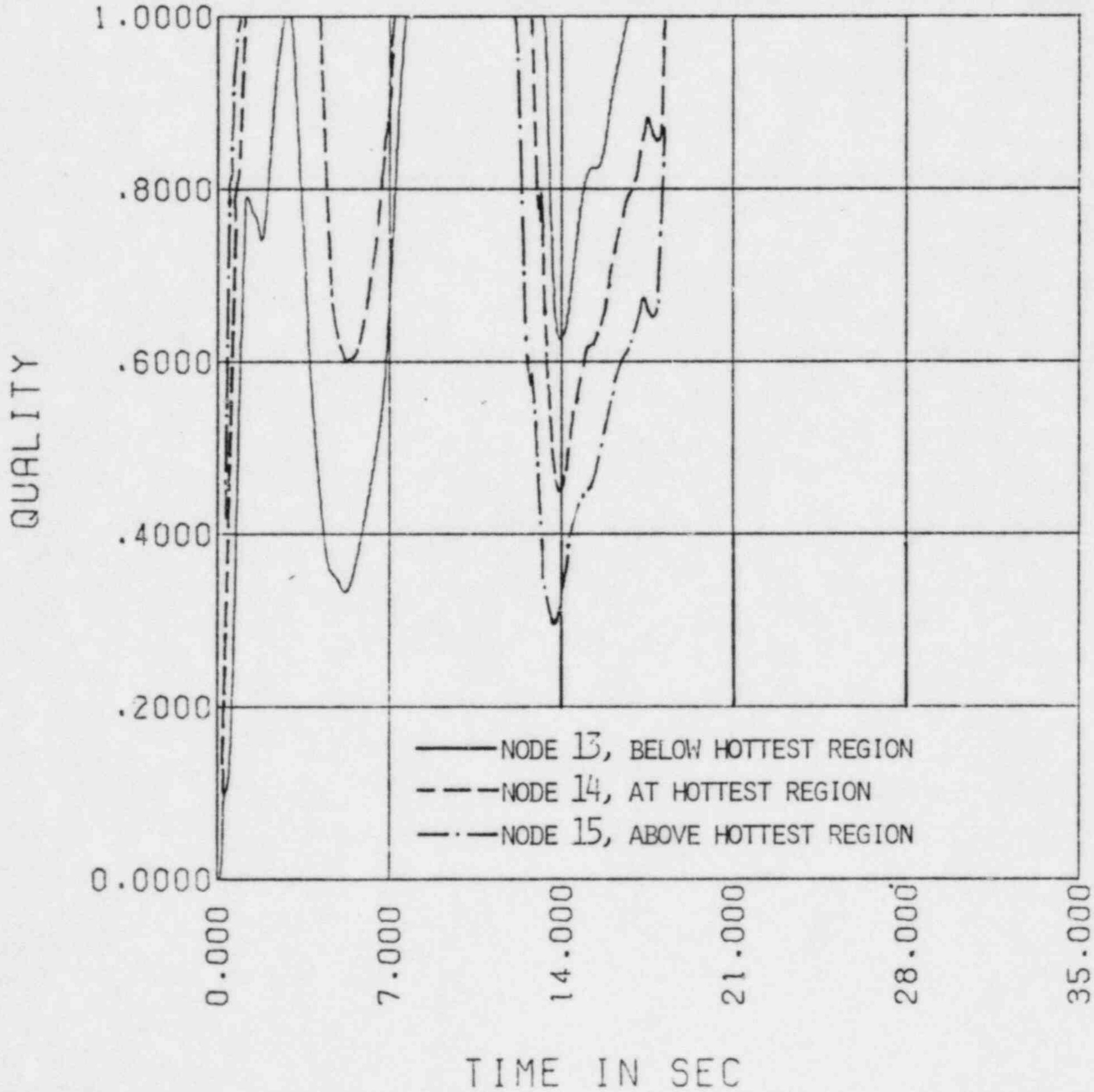

FIGURE 15.6-192
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CONTAINMENT PRESSURE

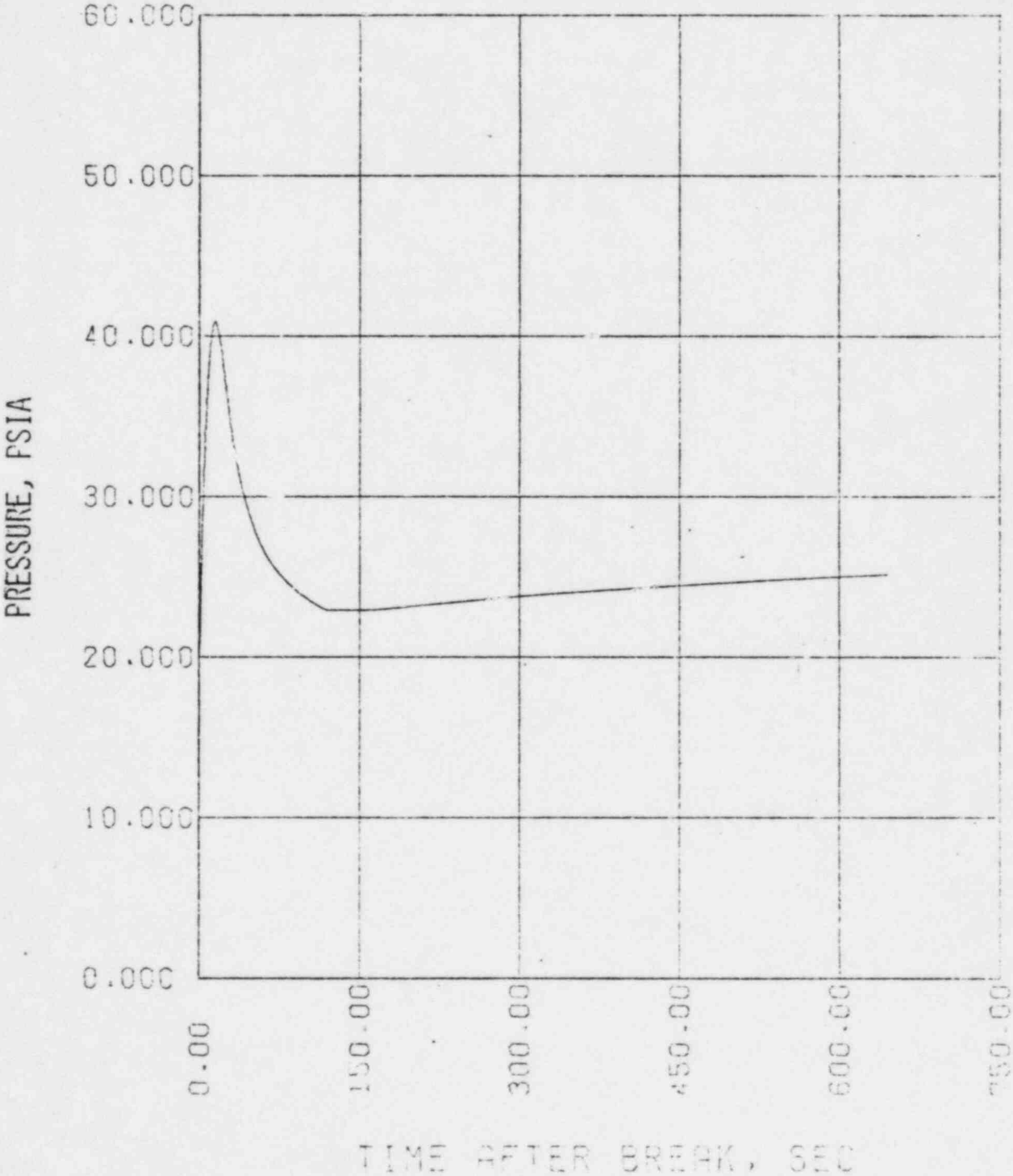

FIGURE 15.6-193
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MASS ADDED TO CORE DURING REFLOOD

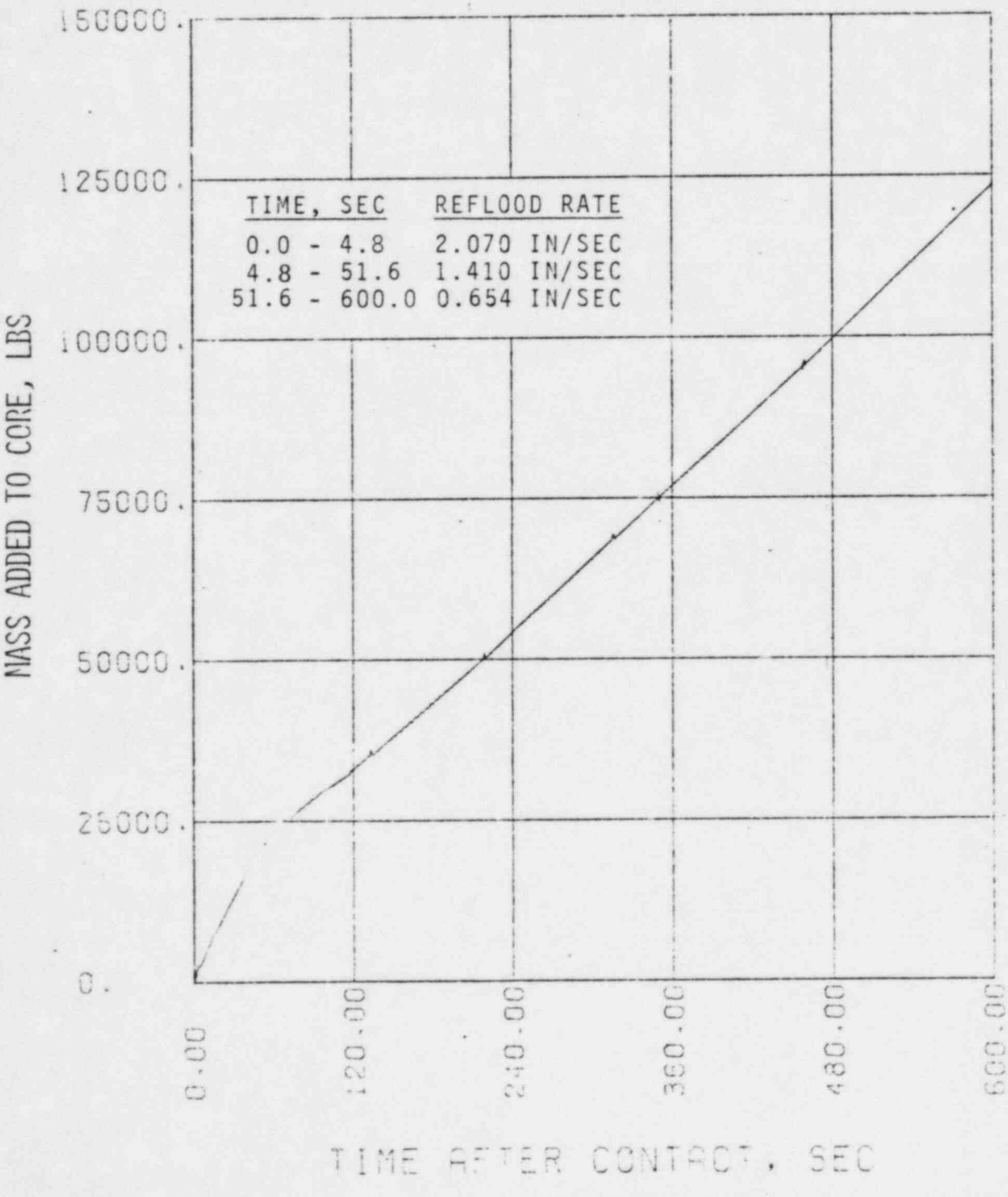

FIGURE 15.6-194

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

PEAK CLAD TEMPERATURE

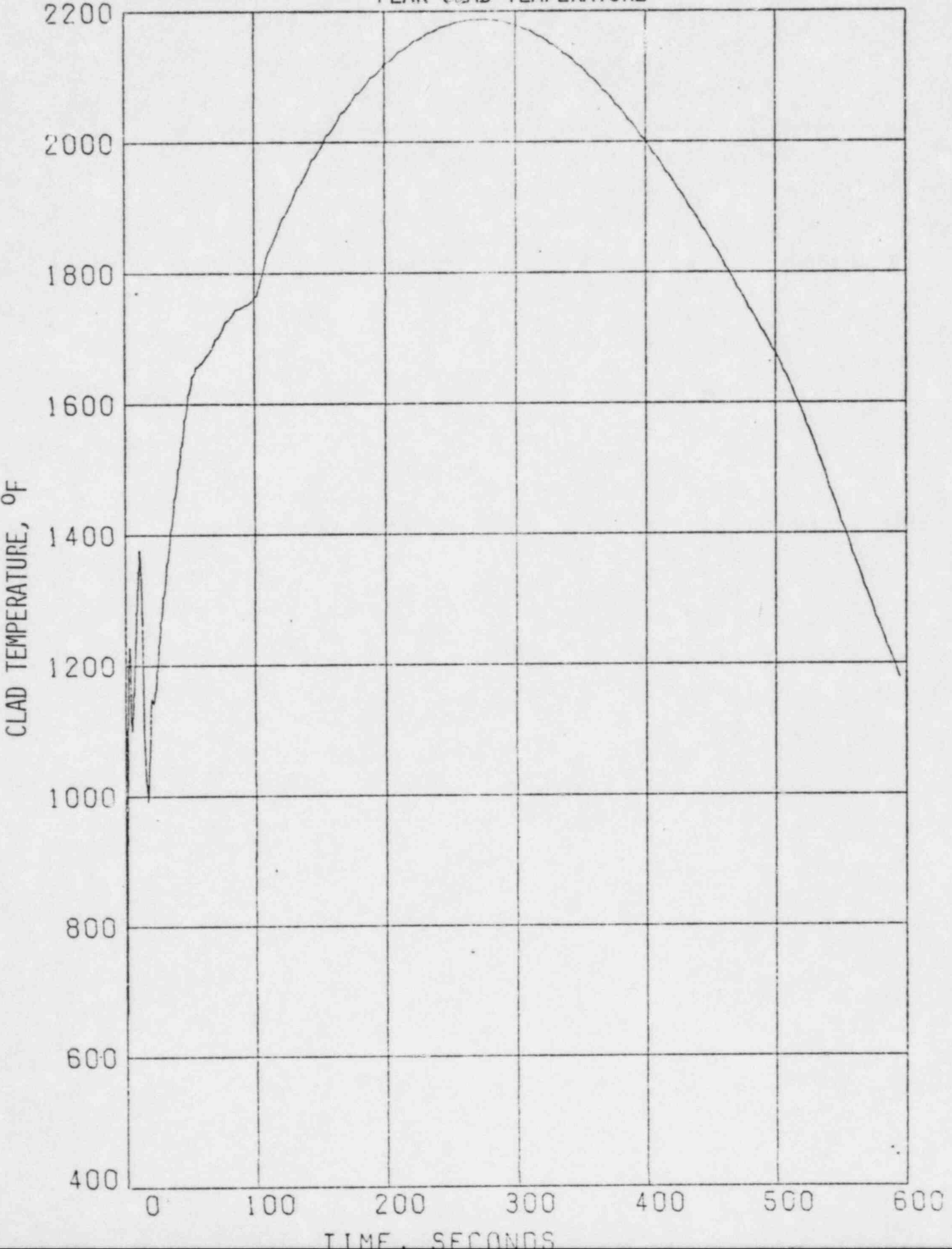

FIGURE 15.6-195
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
MID ANNULUS FLOW

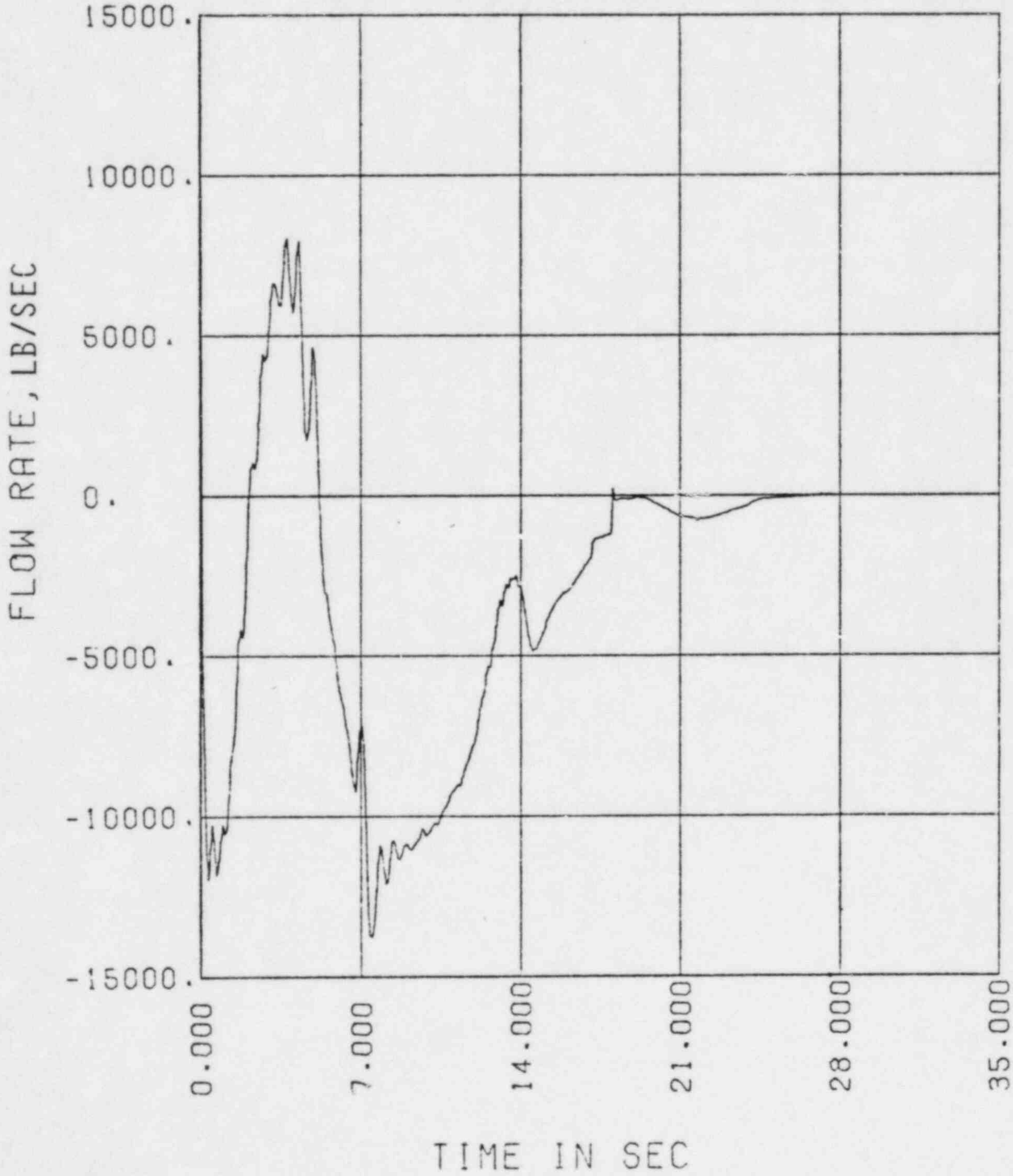

FIGURE 15.6-196

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

QUALITIES ABOVE AND BELOW CORE

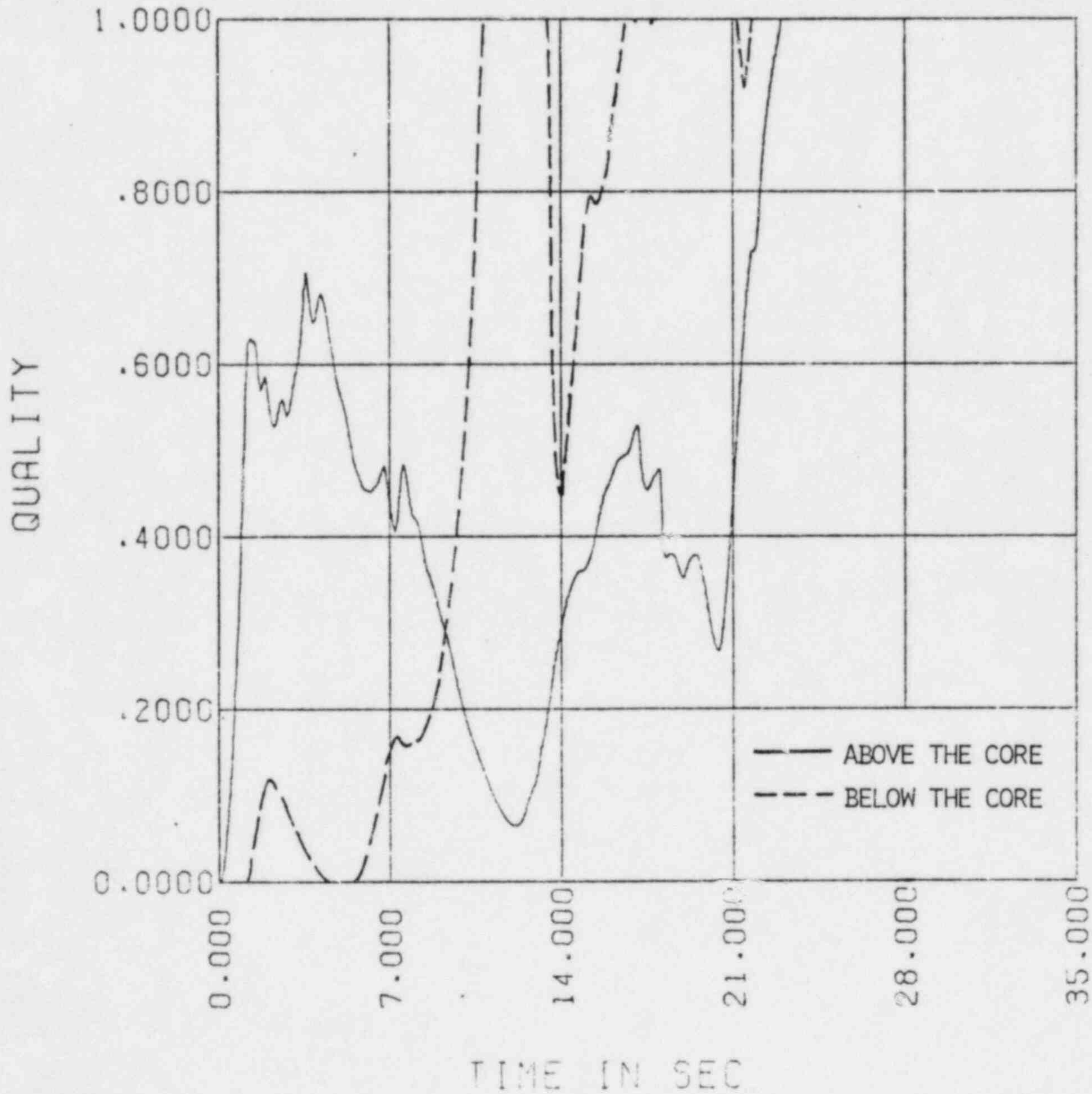

FIGURE 15.6-197
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE PRESSURE DROP

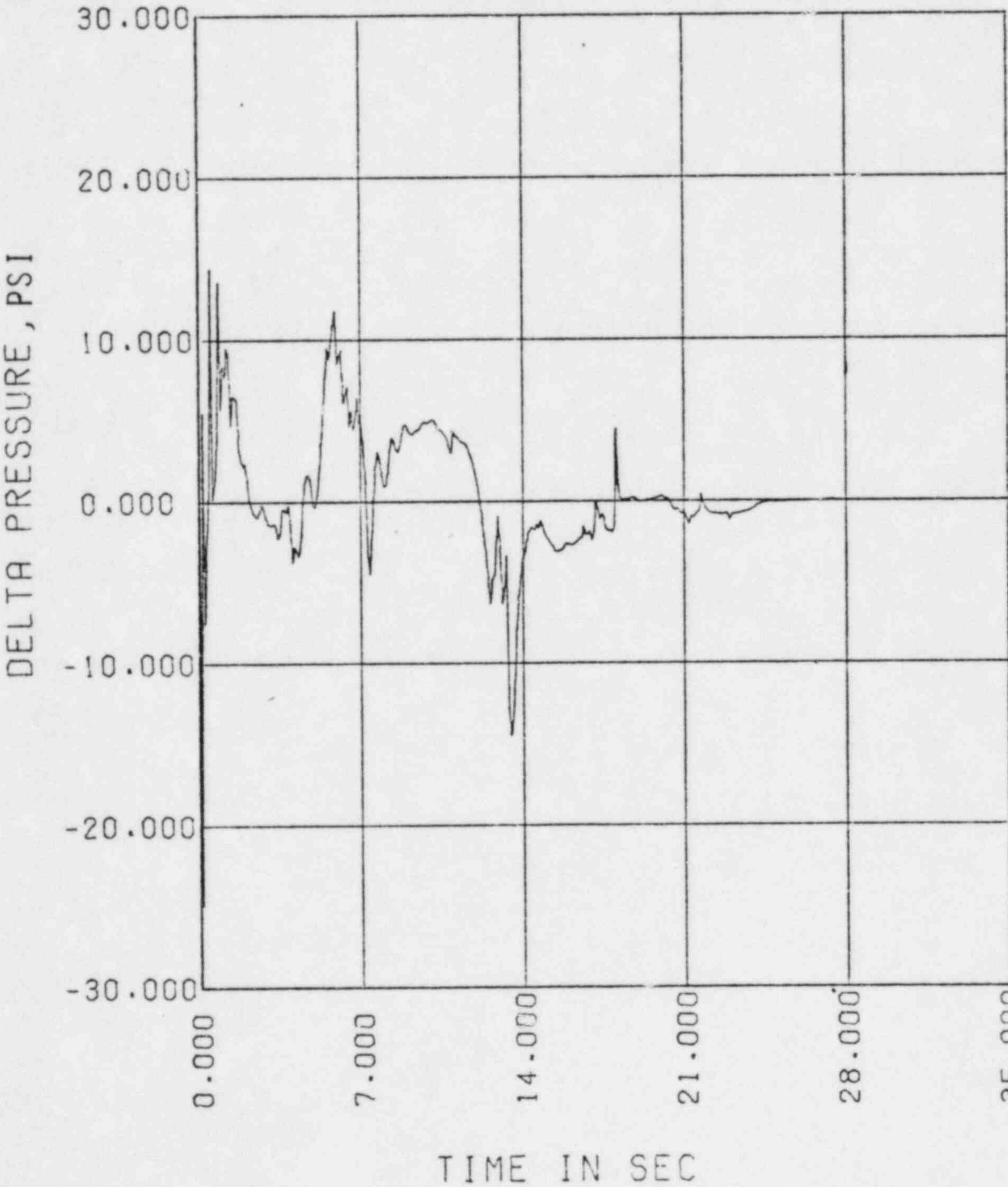

FIGURE 15.6-198

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

SAFETY INJECTION FLOW INTO INTACT DISCHARGE LEGS

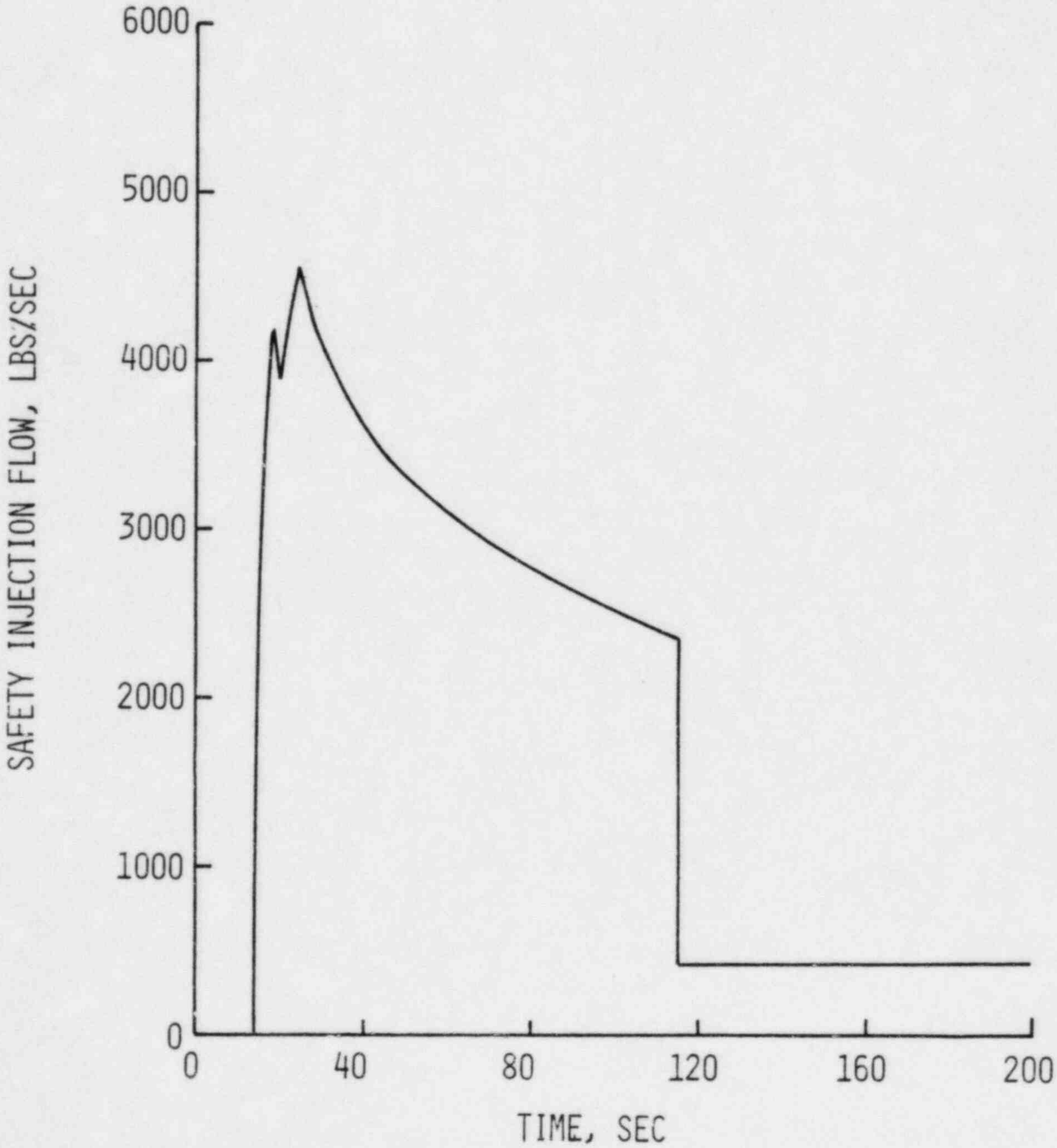

Figure 15.6-199
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
WATER LEVEL IN DOWNCOMER DURING REFLOOD

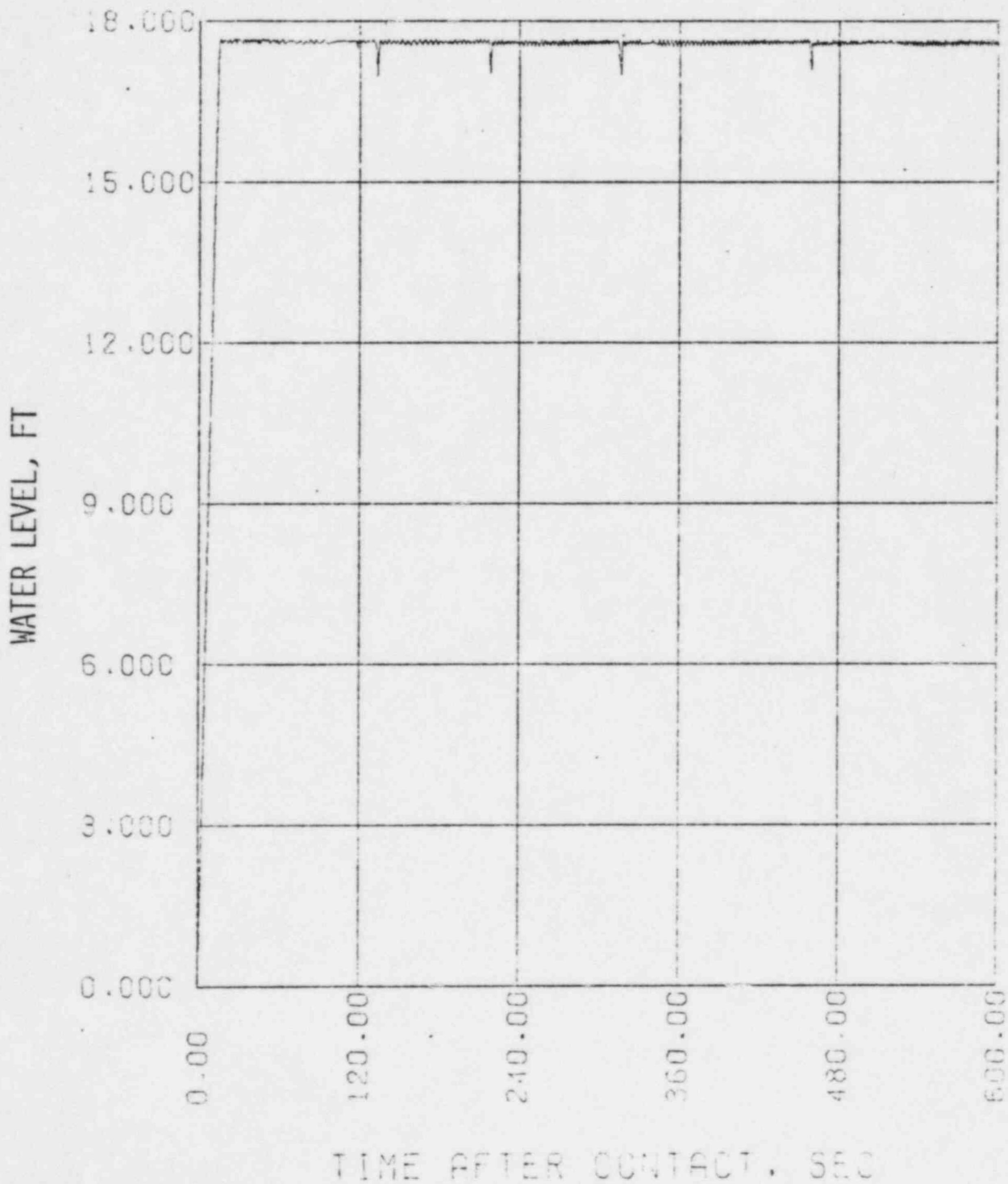

FIGURE 15.6-200

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

HOT SPOT GAP CONDUCTANCE

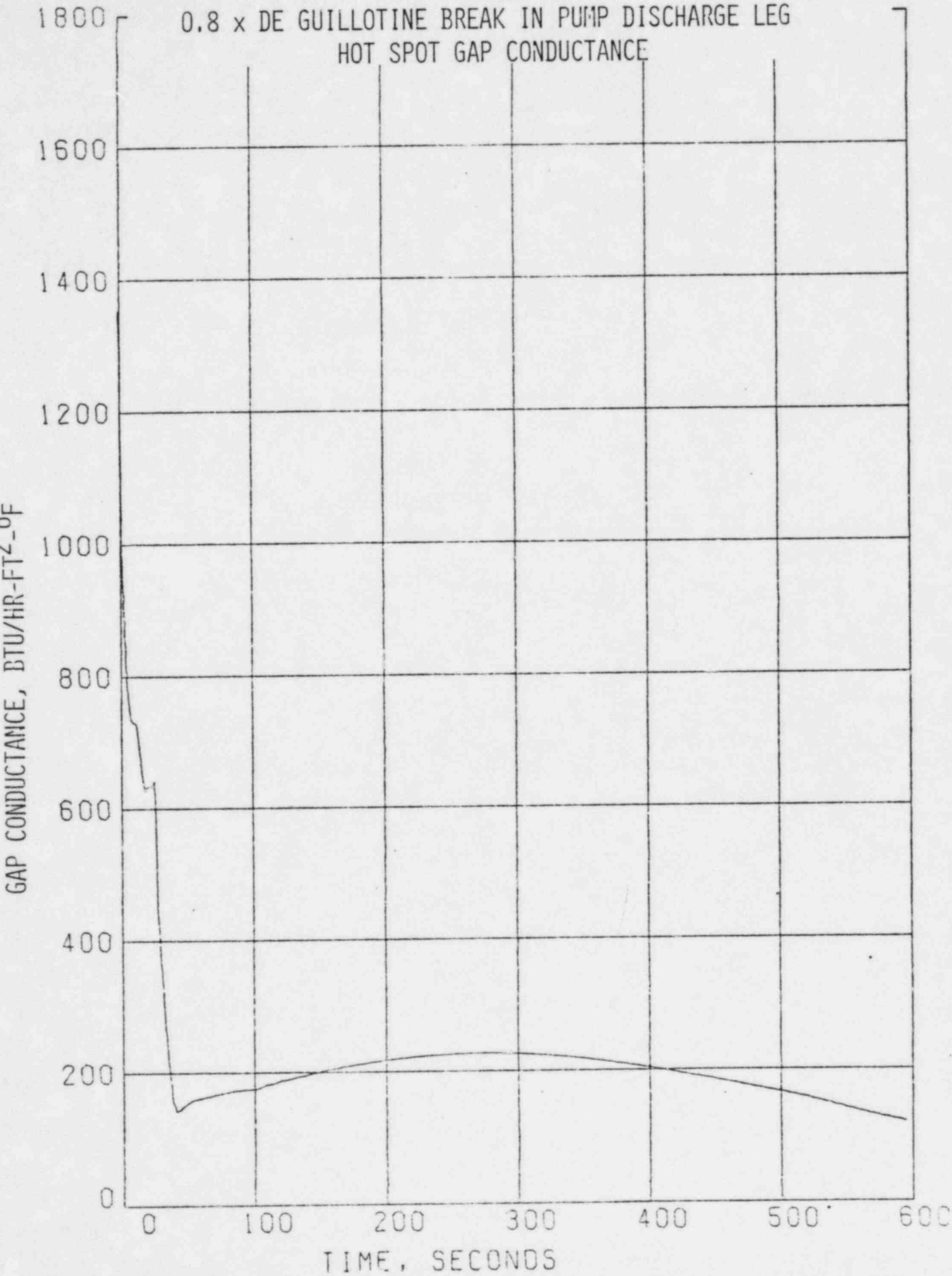

FIGURE 15.6-201

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

LOCAL CLAD OXIDATION

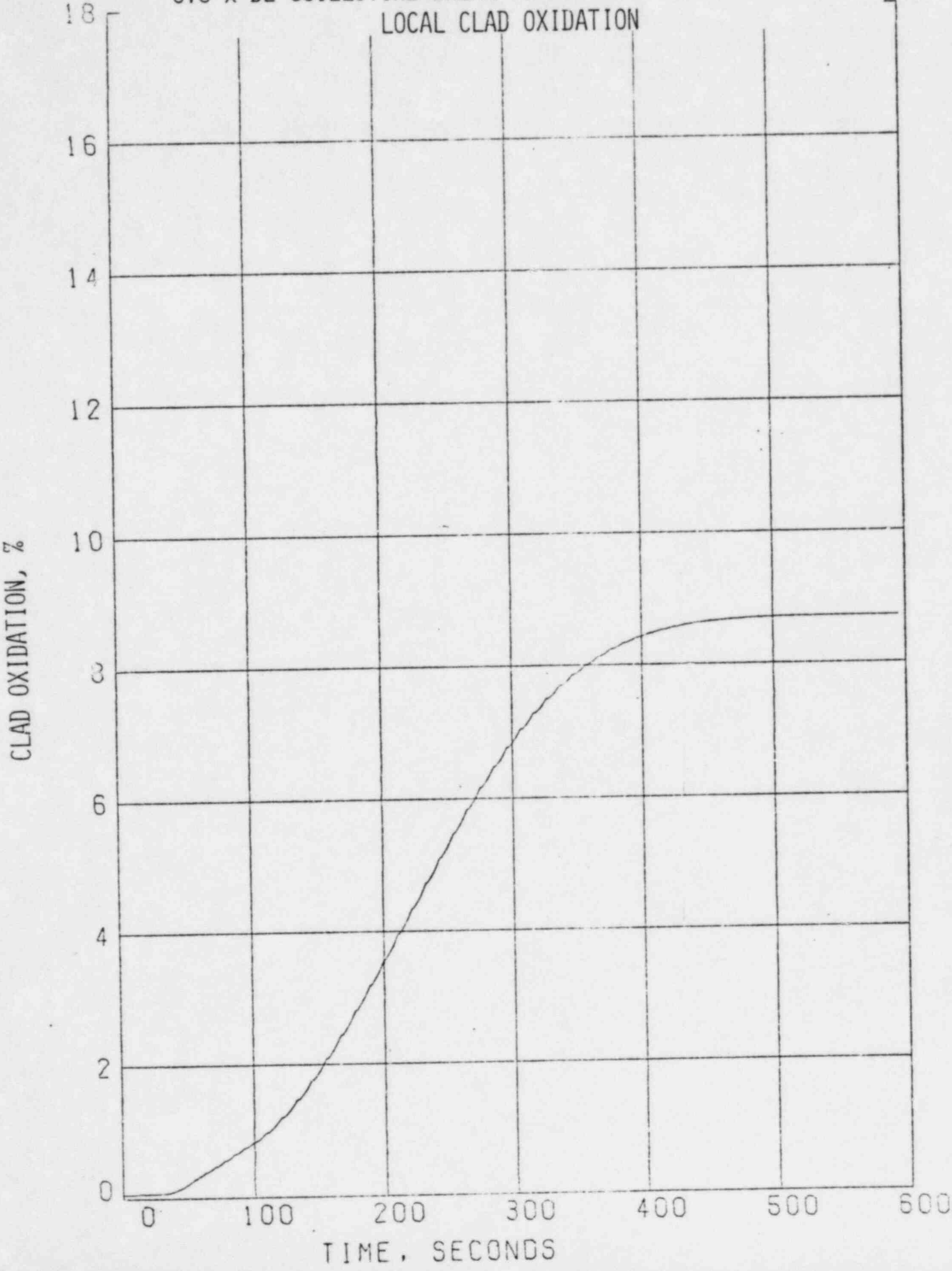

FIGURE 15.6-202

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CLAD TEMPERATURE, CENTERLINE FUEL TEMPERATURE,
AVERAGE FUEL TEMPERATURE AND COOLANT TEMPERATURE
FOR HOTTEST NODE

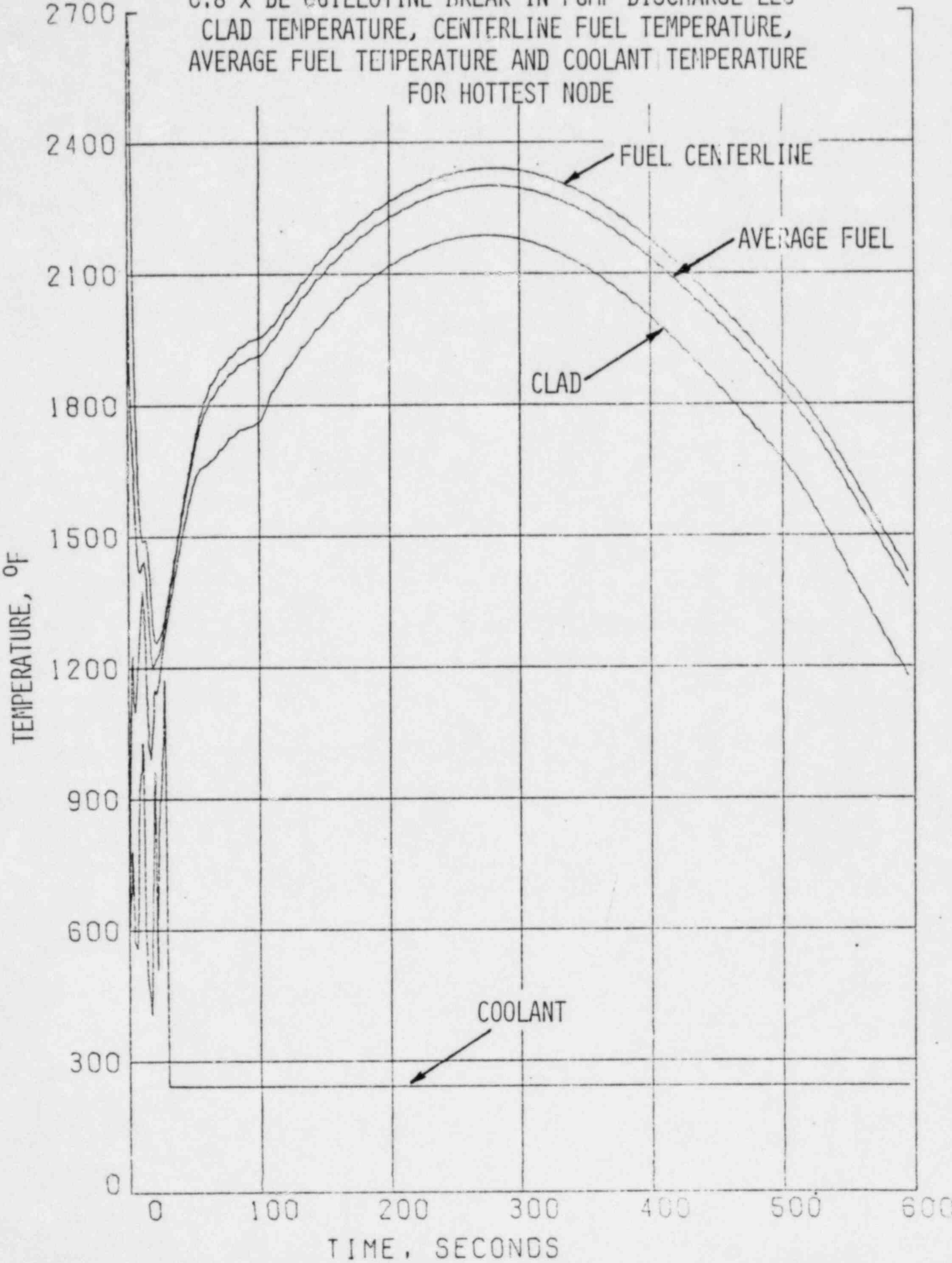

FIGURE 15.6-203

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

HOT SPOT HEAT TRANSFER COEFFICIENT

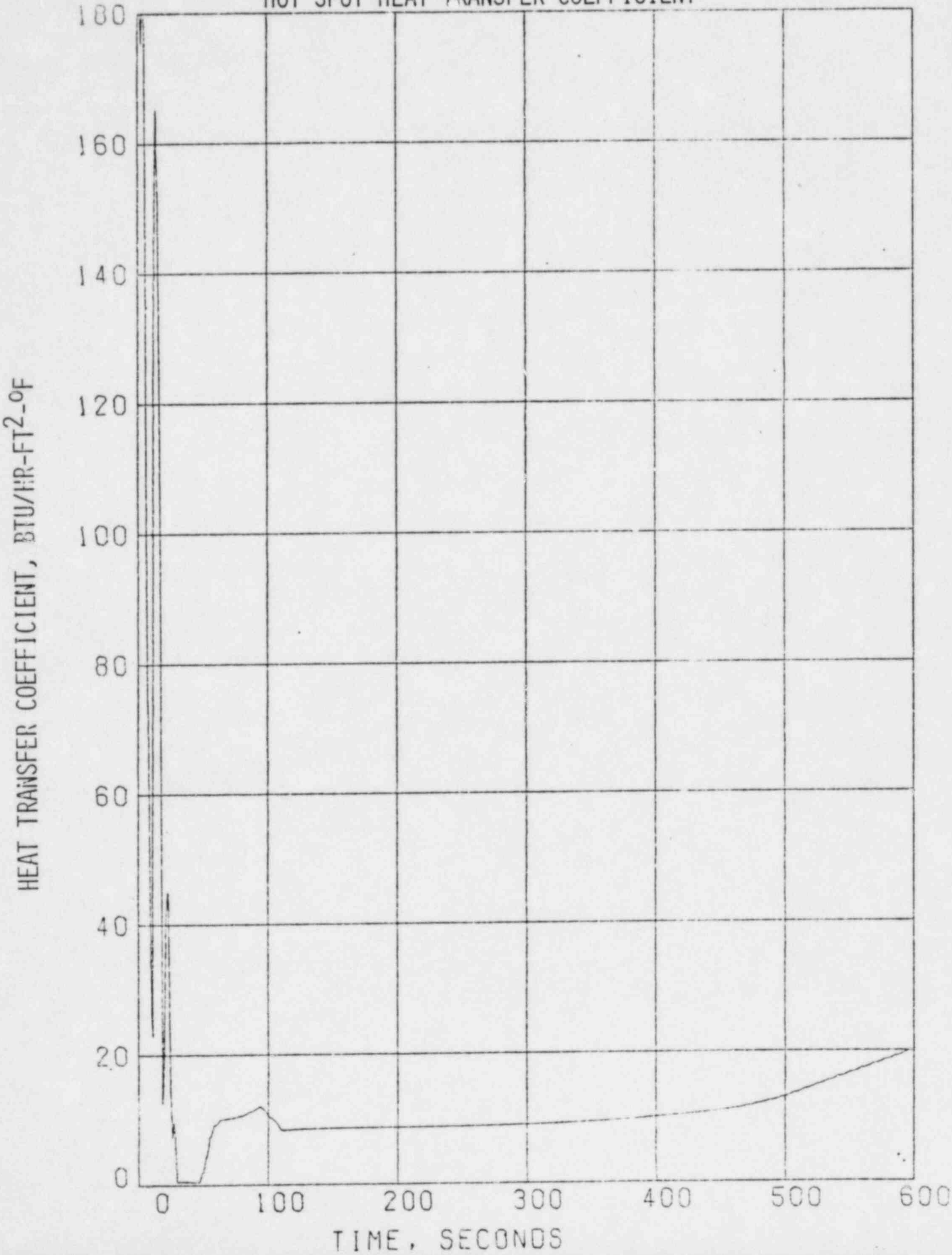

Figure 15.6-204

WATERFORD 3

0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG

HOT ROD INTERNAL GAS PRESSURE

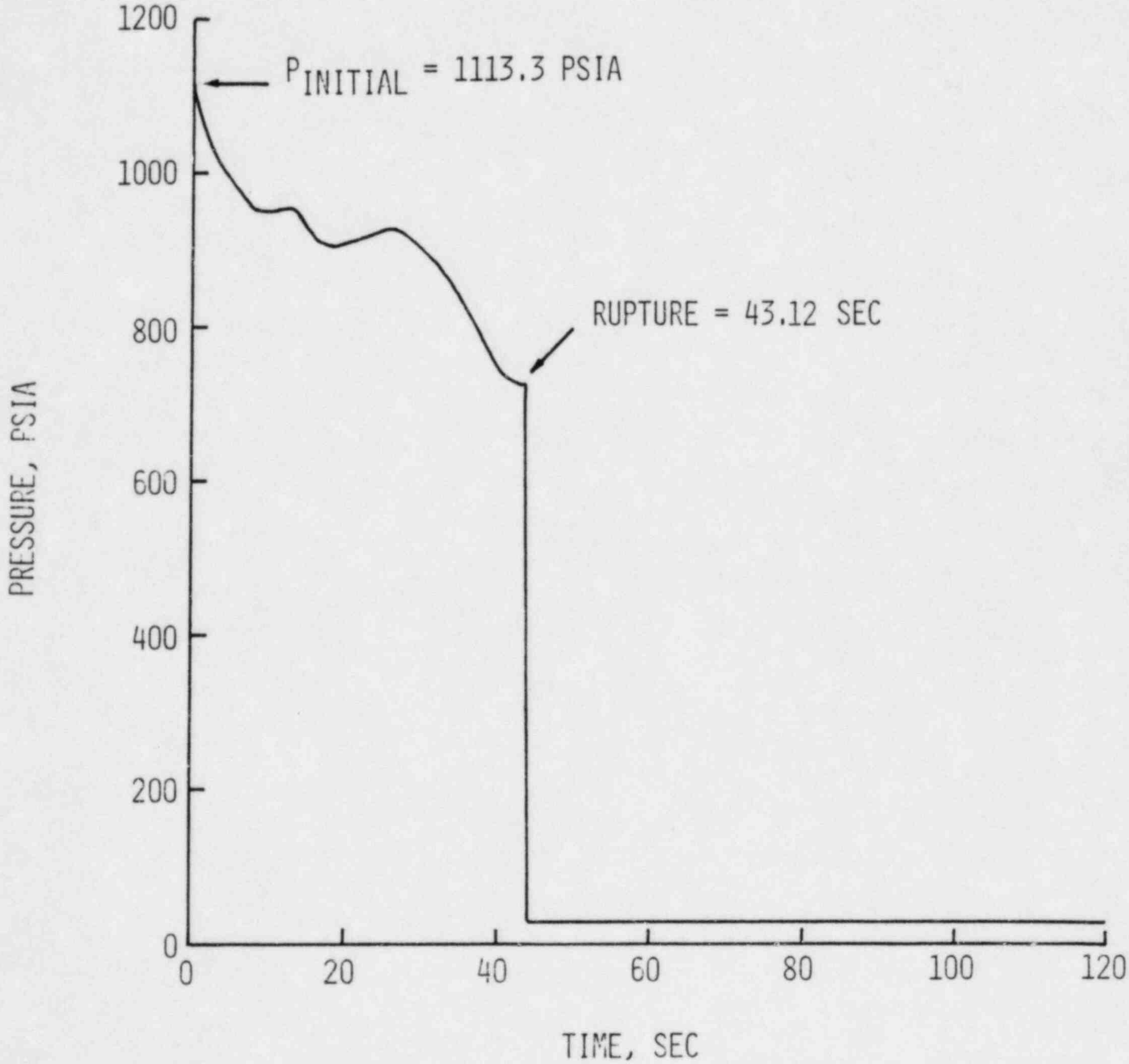

FIGURE 15.6-205
WATERFORD 3
0.8 x DE GUILLOTINE BREAK IN PUMP DISCHARGE LEG
CORE BULK CHANNEL FLOW RATE