Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

FEB 1 6 1995

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

WATTS BAR NUCLEAR PLANT (WBN) - CHANGES TO FINAL SAFETY ANALYSIS REPORT (FSAR) FOR REANALYSIS OF A SMALL-BREAK LOSS-OF-COOLANT ACCIDENT (SBLOCA) AND OTHER ACCIDENT EVENTS (TAC M89427)

This letter submits information describing the latest reanalyses for a SBLOCA and other postulated accident events at WBN. The information is presented in the enclosure as markups to various pages of FSAR Chapter 15 showing the changes that resulted from the SBLOCA reanalysis and related updates for other accident transients. TVA plans to include the changes in a future FSAR amendment, but is submitting the enclosed markups in advance to expedite review of this issue by the NRC staff.

TVA committed to perform a reanalysis of the SBLOCA event in a letter dated April 23, 1994. This reanalysis was subsequently performed by Westinghouse Electric Corporation in June 1994. The SBLOCA reanalysis incorporated various emergency core cooling system (ECCS) evaluation model changes that were described in the letter dated April 23, 1994, and which, cumulatively, exceeded the threshold for a "significant" change in the model as defined by 10 CFR 50.46. In addition to incorporating the ECCS model changes, WBN's latest SBLOCA reanalysis relaxed the setpoint tolerance for the main steam safety valves and the pressurizer safety valves from ±1% to ±3%. The reanalysis also used ECCS flow rates that were based on a detailed technical evaluation of as-installed ECCS pump flow characteristics and plant-specific piping arrangements, as confirmed by measured pump performance. The results of the SBLOCA reanalysis demonstrated that WBN satisfies the ECCS acceptance criteria of 10 CFR 50.46. Specifically, the reanalysis determined that a 4" break

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size is limiting for a SBLOCA event at WBN and the resulting peak cladding temperature for this limiting break is 1452°F.

Relaxing the setpoint tolerance for the main steam safety valves and the pressurizer safety valves from ±1% to ±3% also affected the following FSAR Chapter 15 accident analyses in addition to the SBLOCA analysis:

- Loss of external electrical load and/or turbine trip,
- Loss of normal feedwater,
- Major rupture of a main feedwater pipe,
- Single reactor coolant pump locked rotor.

Each of these accidents was reanalyzed by Westinghouse with acceptable results which indicate that the accident transient does not exceed any safety limits. Information on the reanalyses for the above accidents is presented in the enclosure as markups to appropriate pages of Chapter 15.

Two FSAR Chapter 6 page markups are include in the enclosure. They show a change in the approximate time required before ECCS switchover from cold leg recirculation to hot leg recirculation (i.e., 12 hours instead of 15 hours). This change in switchover timing resulted from using the revised ECCS flow rates described above for the SBLOCA reanalysis.

If you have any questions about the information provided in this letter, please telephone John Vorees at (615) 365-8819.

Sincerely,

Dwight E. Nunn
Vice President
New Plant Completion
Watts Bar Nuclear Plant

Enclosure
cc: See page 3

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LATEST REANALYSES FOR A SBLOCA AND
OTHER POSTULATED ACCIDENT EVENTS

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PAGE MARKUPS

FROM FSAR CHAPTERS 6 AND 15

FOR REVISED ACCIDENT ANALYSES

The time required to complete the sequence is essentially the time required for the operator to perform the accompanying manual operations. Controls for emergency core cooling system components are grouped together on the main control board. The component position lights verify when the function of a given switch has been completed.

After the injection operation, water collected in the containment sump is cooled and returned to the reactor coolant system by the low head/high head recirculation flow path. The low head recirculation flow path consists of the residual heat removal pumps taking suction from the containment sump and discharging the flow directly to the RCS through the residual heat exchangers and cold leg injection lines. The high head recirculation flow path consists of the residual heat removal pumps taking suction from the containment sump and discharging the flow through the residual heat exchangers to the suction of the centrifugal charging pumps and safety injection pumps. The flow from the centrifugal charging pumps and safety injection pumps is returned to the RCS through the cold leg injection lines. The latter mode of operation assures flow in the event of a small rupture where the depressurization proceeds more slowly such that the reactor coolant system pressure is still in excess of the shutoff head of the residual heat removal pumps at the onset of recirculation.

Approximately 15 hours after the switchover to recirculation, hot leg recirculation will be initiated to assure against an excessive buildup of boric acid in the core.

The containment sump isolation valve is interlocked with its respective pump suction/refueling water storage tank isolation valve to the residual heat removal system. The interlock is provided with redundant signals from each isolation valve. This interlock prevents remote manual opening the sump isolation valve when the refueling water storage tank isolation valves are open and thus prevents dumping the refueling water storage tank contents into the containment sump. However, when an accident signal is present, this interlock is bypassed to allow initiation of the switchover sequence.

The refueling water storage tank is protected from back flow of reactor coolant from the reactor coolant system. All connections to the refueling water storage tank are provided with check valves to prevent back flow. When the reactor coolant system is hot and pressurized there is no direct connection between the refueling water storage tank and the reactor coolant system. When the reactor coolant system is being cooled and the residual heat removal system is placed in service, the residual heat removal system is isolated from the refueling water storage tank by a motor-operated valve in addition to a check valve.

Redundancy in the external recirculation loop is provided by the inclusion of duplicate charging, safety injection, and residual heat removal pumps and residual heat exchangers. Inside the containment, the charging pump and safety injection pump discharge is piped separately into all four cold legs for the charging pumps and into all four cold legs and all four hot legs for the safety injection pumps. The low head pumps take suction through redundant lines from the containment sump and discharge through separate paths to the reactor coolant system. The containment sump design is shown in Figure 6.3-6.

TABLE 6.3-3

(Sheet 3 of 3)

SEQUENCE OF CHANGE-OVER OPERATION, INJECTION TO RECIRCULATION (Cont'd)

To align No. 2 RHR pump to spray (complete Steps 3 and 4).

3. Close the valve between the No. 2 RHR pump discharge and two RCS cold legs (FCV-63-94).

4. Open the valve between the discharge of the No. 2 RHR pump and the No. 2 RHR spray header (FCV-72-41).

   Note: The valve between the discharge of the No. 2 RHR pump and the No. 2 RHR spray header is interlocked such that it cannot be opened unless the containment sump valve to the No. 2 RHR pump is open.

The following switching operation should be used approximately 15 12 hours after initiation of cold leg recirculation when realigning the ECCS from the cold leg recirculation mode to the hot leg recirculation mode. In the event a train cannot be aligned in the hot leg recirculation mode, it will be realigned to the cold leg recirculation.

1. Stop the safety injection pumps.

2. Close the safety injection pumps discharge crossover valves (FCV 63-152 and -153). [8821A and B]

3. If available, close the safety injection pump cold leg injection valve (FCV 63-22). [8835]

4. Open the safety injection pump hot leg injection valves (FCV 63-156 and -157). [8802A and B]

5. Start the safety injection pumps.

If RHR hot leg injection is desired:

6. Close train A or B cold leg injection valve (FCV 63-93 or -94). [8809A, 8809B]

7. Open the RHR crossover valve (FCV 74-33 or -35). [8716B, 8716A]

8. Open the RHR hot leg injection valve (FCV 63-172). [8840]

9. Close the other train RHR pump cold leg injection valve (FCV-63-94 or -93). [8809B, 8809A]

TABLE 15.1-3

(Sheet 1 of 2)

TRIP POINTS AND TIME DELAYS TO TRIP
ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed in Analysis	Time Delay (Seconds)
Power Range High Neutron Flux, High Setting	118%	0.5
Power Range High Neutron Flux, Low Setting	35%	0.5
Overtemperature ΔT	Variable (see Figure 15.1-1)	7.0*
Overpower ΔT	Variable (see Figure 15.1-1)	7.0*
High Pressurizer Pressure	2445 psig	2.0
Low Pressurizer Pressure	~~1845~~ 1910 psig	2.0

* Total time delay (including RTD time response and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

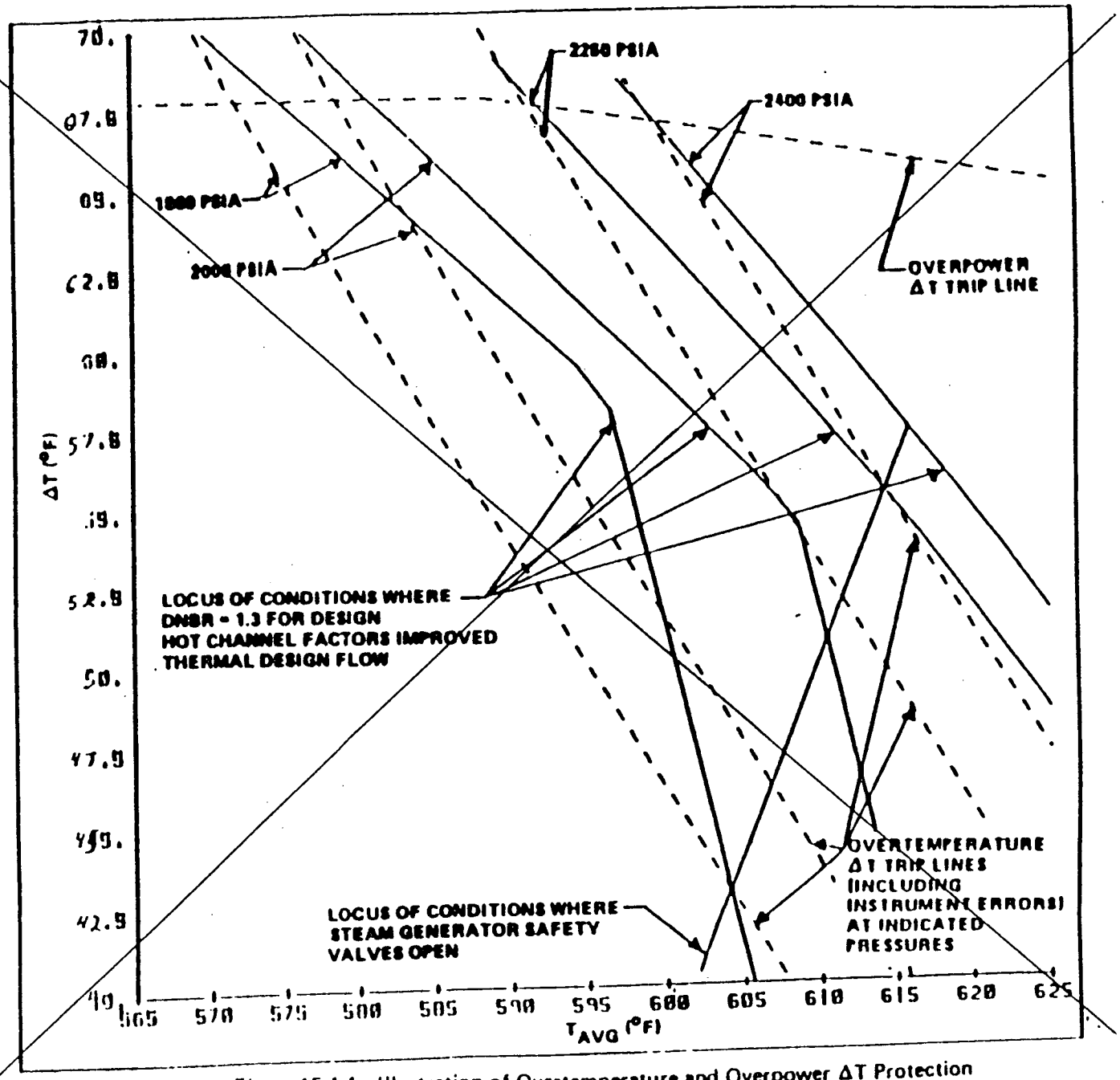

Figure 15.1-1. Illustration of Overtemperature and Overpower ΔT Protection

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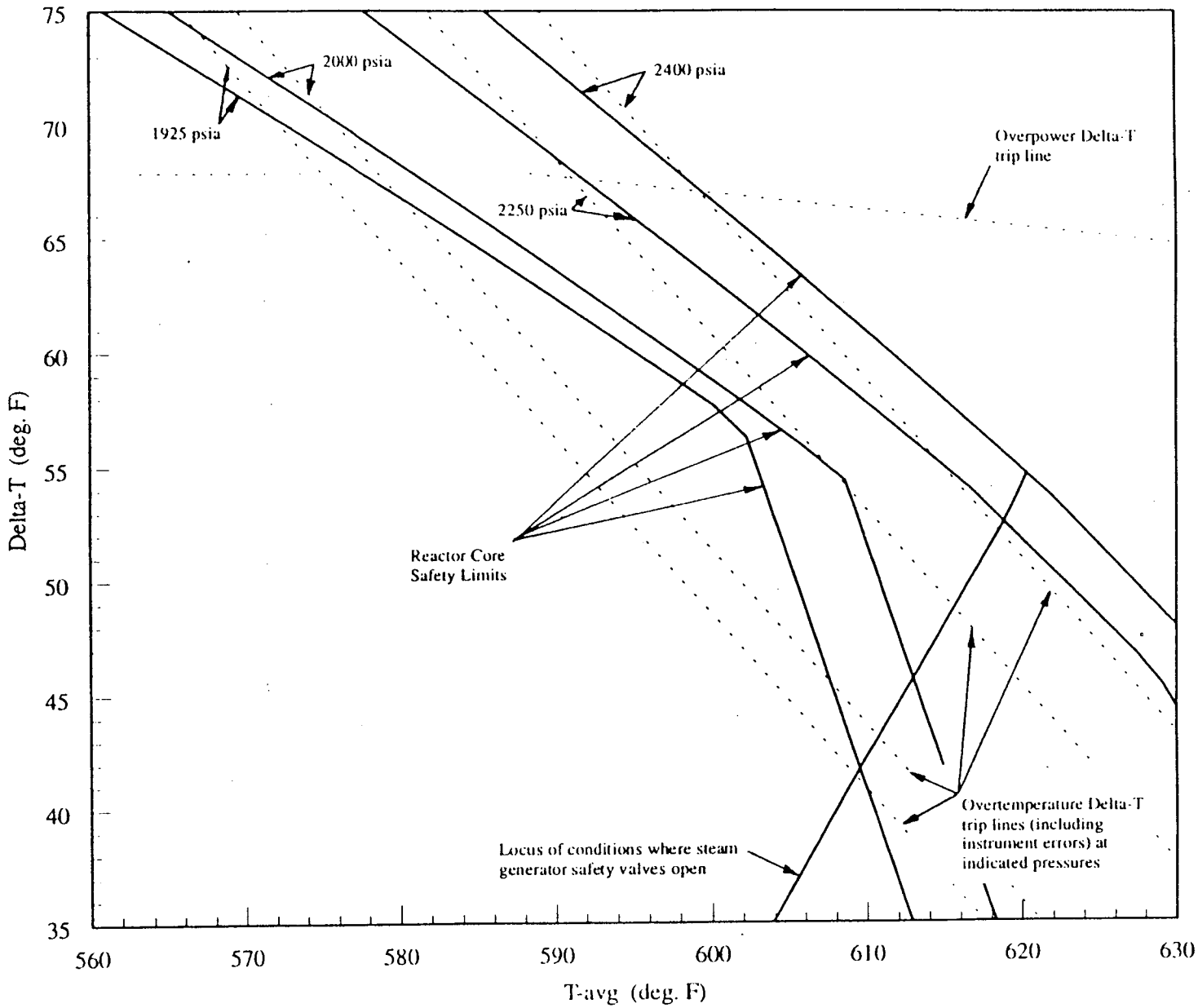


Figure 15.1-1. Illustration of Overtemperature and Overpower Delta-T Protection

Typical assumptions are:

1. Initial Operating Conditions - the initial reactor power and RCS temperatures are assumed at their maximum values consistent with the steady state full power operation including allowances for calibration and instrument errors. The initial RCS pressure is assumed at a minimum value (pressurizer pressure -46 psi allowance for steady-state fluctuations and measurement error) consistent with steady-state full-power operation including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the accident.

2. Moderator and Doppler Coefficients of Reactivity - the total loss of load is analyzed for both the beginning-of-life and end-of-life conditions. The least negative moderator temperature coefficients at beginning-of-life and a large (absolute value) negative value at end-of-life are used. A conservatively large (absolute value) Doppler power coefficient is used for all cases.

3. Reactor Control - from the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control.

4. Steam Release - no credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoints where steam release through safety valves ~~limits~~ secondary steam pressure, ~~at the setpoint value.~~ occurs to limit the

5. Pressurizer Spray and Power-Operated Relief Valves - two cases for both the beginning and end-of-life are analyzed:

   a. Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure.

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

6. Feedwater Flow - main feedwater flow to the steam generators is assumed to be lost at the time of loss of external electrical load.

Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on turbine trip.

Results

The transient responses for a total loss of load from 102% of full power operation are shown for four cases; two cases for the beginning of core life and two cases for the end of core life, in Figures 15.2-19 through 15.2-26. The calculated sequence of events for the accident is shown in Table 15.2-1.

Figures 15.2-19 and 15.2-20 show the transient responses for the total loss of steam load at beginning-of-life with a least negative moderator temperature coefficient assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the high pressurizer pressure signal trip channel. The minimum DNBR is well above the limiting value.

Figures 15.2-21 and 15.2-22 show the responses for the total loss of load at end-of-life assuming a large (absolute value) negative moderator temperature coefficient. All other plant parameters are the same as the above. The DNBR increases throughout the transient and never drops below its initial value. The reactor is tripped by the low-low steam generator water level signal.

The pressurizer safety valves are not actuated in the transients shown in Figures ~~15.2-19 through~~ 15.2-21 and 15.2-22.

The total loss of load accident was also studied assuming the plant to be initially operating at 102% of full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2-23 and 15.2-24 show the beginning-of-life transients with a least negative moderator coefficient. The neutron flux remains constant at 102% of the full power until the reactor is tripped. The DNBR increases throughout the transient. In this case the pressurizer safety valves ~~is~~ are actuated.

Figures 15.2-25 and 15.2-26 are the transients at the end-of-life with the other assumptions being the same as in Figure 15.2-23 and 15.2-24. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated.

Reference [9] presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.7.3 Conclusions

Results of the analyses, including those in Reference [9], show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

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

15.2.8 LOSS OF NORMAL FEEDWATER

15.2.8.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden

pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Two cases are examined for a loss of normal feedwater event. The first is the case where offsite ac power is maintained, and the second is the case where offsite ac power is lost, which results in reactor coolant pump coastdown as described in Section 15.2.5.2.

The case where offsite ac power is lost is limiting with respect to over-pressurization of the RCS and loss of water from the reactor core due to the decreased capability of the reactor coolant pump to aid in residual core heat removal as a result of the reactor coolant pump coastdown.

Assumptions

1. The initial steam generator water level (in all steam generators) at the time of reactor trip is at a conservatively low level.

2. The plant is initially operating at 102% of the Nuclear Steam Supply System design rating. The heat added to the RCS by the reactor coolant pumps is assumed.

3. The core residual heat generation is based on the 1979 version of ANS 5.1[14] based upon long term operation at the initial power level. The decay of U-238 capture products is included as an integral part of this expression.

4. A heat transfer coefficient in the steam generator associated with RCS natural circulation.

5. Two motor-driven auxiliary feedwater pumps are available one minute after the accident. (Failure of the turbine-driven auxiliary feedwater pump is assumed since this failure provides minimum auxiliary feedwater flow.)

6. Constant auxiliary feedwater flow equal to 820 gpm from the two motor-driven auxiliary feedwater pumps is delivered to four steam generators.

7. Auxiliary feedwater temperature is 120 °F.

8. Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in fact, be through the power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis these have been assumed unavailable.

9. The initial reactor coolant average temperature is set conservatively higher than the nominal value ~~for the case where offsite ac power is maintained~~ since this results in a greater expansion of the RCS from pump and decay heat during the transient and, subsequently, a higher water level in the pressurizer. ~~For the case where offsite ac power is lost, the initial reactor coolant average temperature is set conservatively lower than the nominal value since this results in a greater density in the RCS and lower natural circulation.~~

10. The initial pressurizer pressure is 46 psi higher than nominal. This 46 psi allowance is for steady-state fluctuations and measurement error.

11. The low-low steam generator level trip setpoint is conservatively assumed to be 0.0% of narrow range span.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the reactor protection and engineered safeguards systems (e.g., the auxiliary feedwater system) in removing long term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above.

One such assumption is the loss of external (offsite) ac power. This assumption results in coolant flow decay down to natural circulation conditions reducing the steam generator heat transfer coefficient. Following a loss of offsite ac power, the first few seconds of a loss of normal feedwater transient will be virtually identical to the transient response (including DNBR and neutron flux versus time) presented in Section 15.3.4 for the complete loss of forced reactor coolant flow incident.

An additional assumption made for the loss of normal feedwater evaluation is that the pressurizer power-operated relief valves are assumed to function normally. If these valves were assumed not to function, the coolant system pressure during the transient would rise to the actuation point of the pressurizer safety valves ~~(2500 psia)~~ (nominally 2500 psia). The increased RCS pressure, however, results in less expansion of the coolant and hence more margin to the point where water relief from the pressurizer would occur.

Results

Figures 15.2-27a through 15.2-27i show the significant plant parameter transients following a loss of normal feedwater where offsite power is lost. The calculated sequence of events for this accident are listed in Table 15.2-1.

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

The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves.

From Figure 15.2-27~~f~~g, it can be seen that at no time is there water relief from the pressurizer. If the auxiliary feed delivered is greater than that of two motor-driven pumps, if the initial reactor power is less than 102% of the NSSS design rating, or if the steam generator water level in one or more steam generators is above the low-low level trip point at the time of trip, then the results of this transient will be bounded by the analysis presented.

The plant will slowly approach a stabilized condition at hot standby with auxiliary feedwater removing decay heat. The plant may be maintained at hot standby or further cooled through manual control of the auxiliary feed flow. The operating procedures also call for operator action to control RCS boron concentration and pressurizer level using the CVCS and to maintain steam generator level through control of the auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following reactor trip.

15.2.8.3 Conclusions

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

15.2.9 LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES (STATION BLACKOUT)

Analysis is now contained in Sections 15.2.7 and 15.3.4.

15.2.10 EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS

15.2.10.1 Identification of Causes and Accident Description

Additions of excessive feedwater cause increases in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (high neutron flux, overtemperature ΔT, and overpower ΔT trips) prevents any power increase which could lead to a DNBR less than the limiting value.

Excessive feedwater flow could be caused by a full opening of one or more feedwater control valves due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous addition of excessive feedwater is prevented by the steam generator high-high level trip, which closes the feedwater control and isolation valves.

TABLE 15.2-1

(Sheet 5 of 10)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS (Cont'd)

Accident	Event	Time (sec.)
Loss of External Electrical Load		
1. With pressurizer control (BOL)	Loss of electrical load	0
	High pressurizer pressure reactor trip point reached	~~6.5~~ 6.2
	Rods begin to drop	~~8.5~~ 8.2
	Minimum DNBR occurs	~~10.0~~ (1)
	Peak pressurizer pressure occurs	~~10.0~~ 9.0
2. With pressurizer control (EOL)	Loss of electrical load	0
	Low-low steam generator water level reactor trip point reached	~~66.4~~ 74.5
	Rods begin to drop	~~68.4~~ 76.5
	Minimum DNBR occurs	(1)
	Peak pressurizer pressure occurs	~~7.0~~ 9.0

(1) DNBR does not decrease below its initial value.

TABLE 15.2-1

(Sheet 6 of 10)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS (Cont'd)

Accident	Event	Time (sec.)
3. Without pressurizer control (BOL)	Loss of electrical load	0
	High pressurizer pressure reactor trip point reached	~~5.0~~ 4.7
	Rods begin to drop	~~7.0~~ 6.7
	Minimum DNBR occurs	(1)
	Peak Pressurizer pressure occurs	~~8.0~~ 7.5
4. Without pressurizer control (EOL)	Loss of electrical load	0
	High pressurizer pressure reactor trip point reached	~~5.0~~ 4.7
	Rods begin to drop	~~7.0~~ 6.7
	Minimum DNBR occurs	(1)
	Peak pressurizer pressure occurs	7.0

(1) DNBR does not decrease below its initial value.

TABLE 15.2-1

(Sheet 7 of 10)

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS (Cont'd)

Accident	Event	Time (sec.)
Loss of Normal Feedwater with Offsite Power Available (LONF)	Main feedwater flow stops	10.0
	Low-low steam generator water level reactor trip	~~68.5~~ 69.1
	Rods begin to drop	~~70.5~~ 71.1
	Peak water level in pressurizer occurs	~~73.0~~ 74.0
	Four steam generators begin to receive auxiliary feed from two motor-driven auxiliary feedwater pumps	~~128.5~~ 129.1
Loss of Normal Feedwater with Loss of Offsite Power (LOOP)	Main Feedwater Flow Stops	10.0
	Low-low steam generator water level reactor trip	~~70.2~~ 69.1
	Rods begin to drop	~~72.2~~ 71.1
	Reactor coolant pumps begin to coastdown	~~74.2~~ 73.1
	Peak water level in pressurizer occurs	~~76.0~~ 74.0
	Four steam generators begin to receive auxiliary feed from two motor-driven auxiliary feedwater pumps	~~130.2~~ 129.1

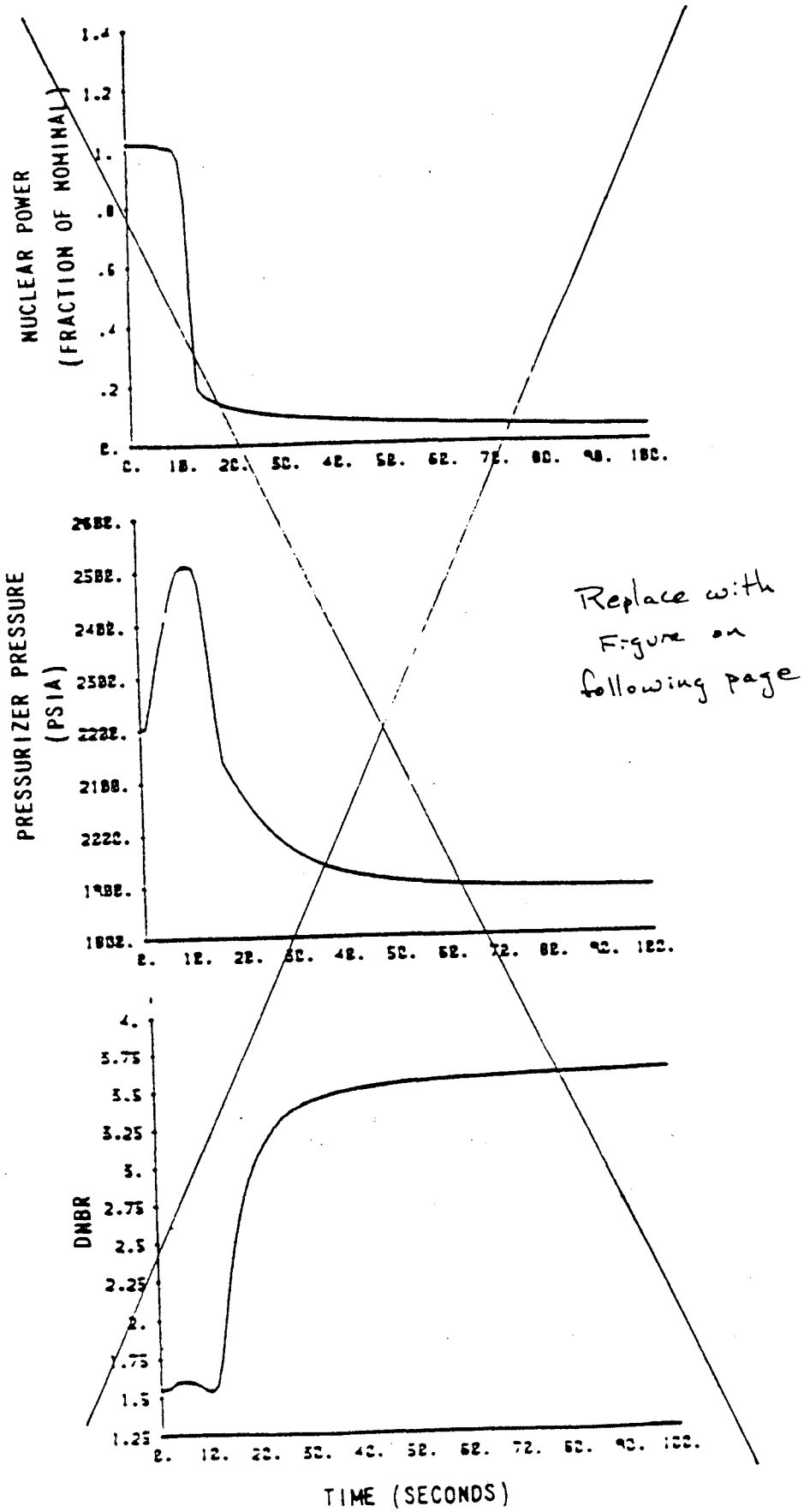
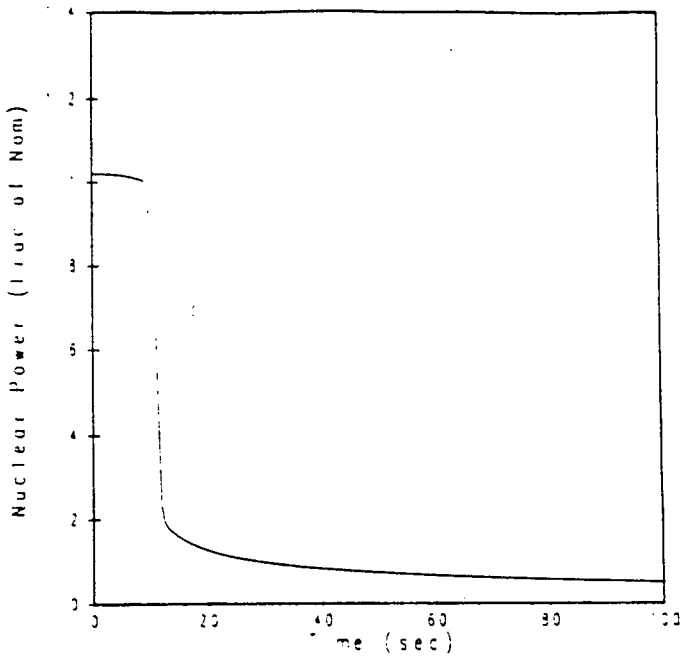
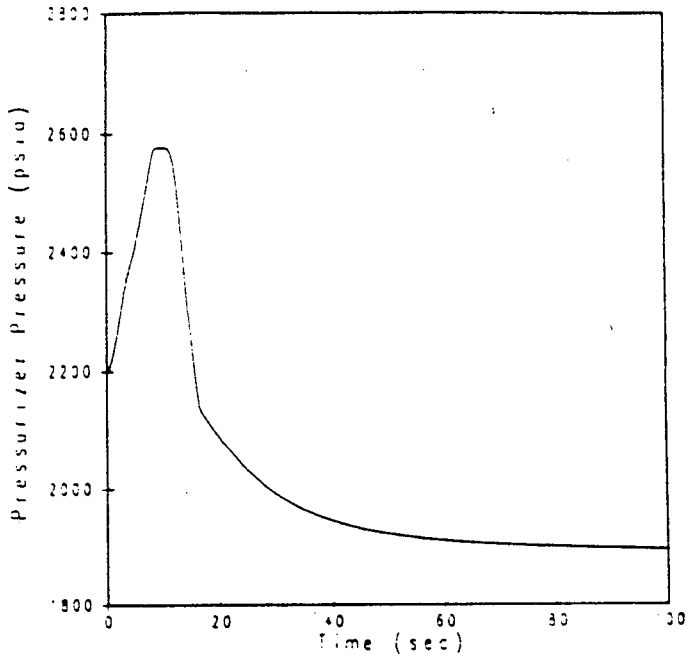
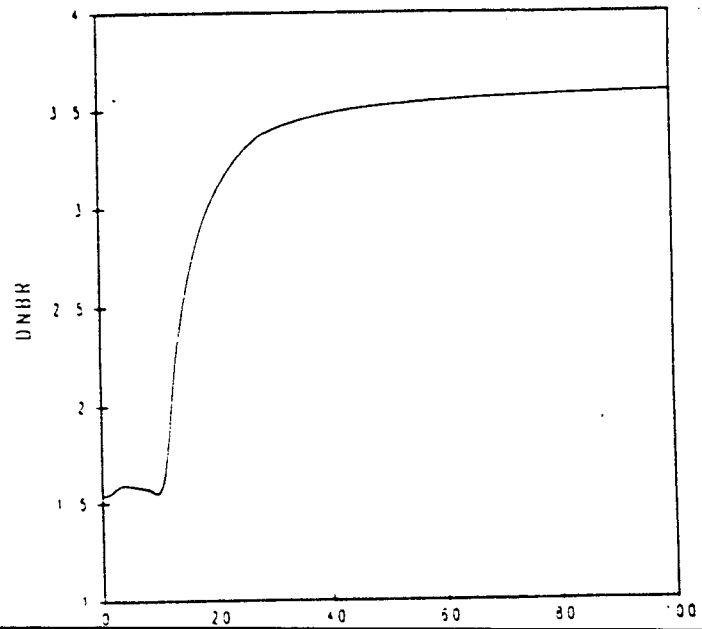


Figure 15.2-19 Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves, Beginning-of-Life







WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves, Beginning of Life
FIGURE 15.2-19

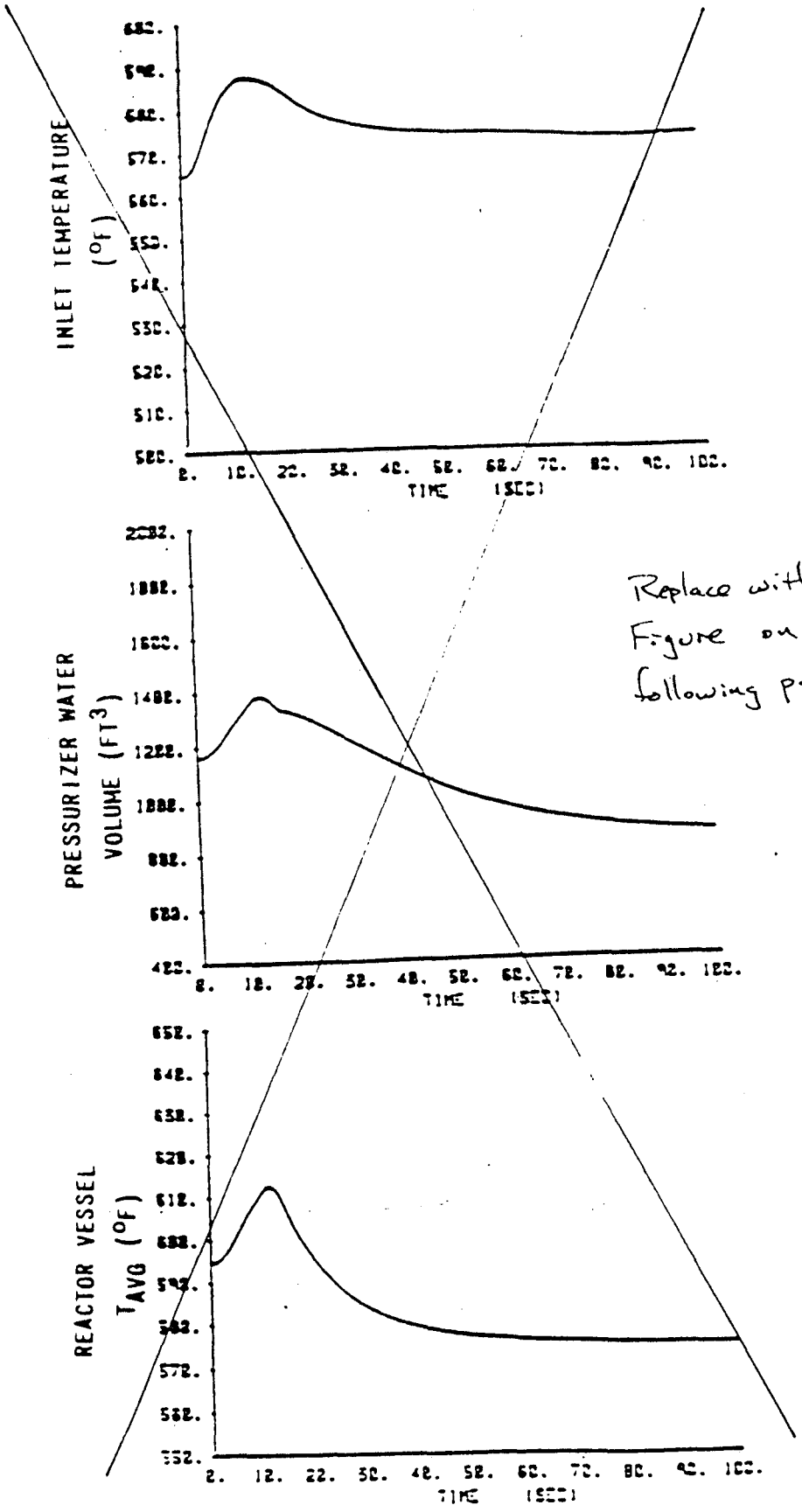


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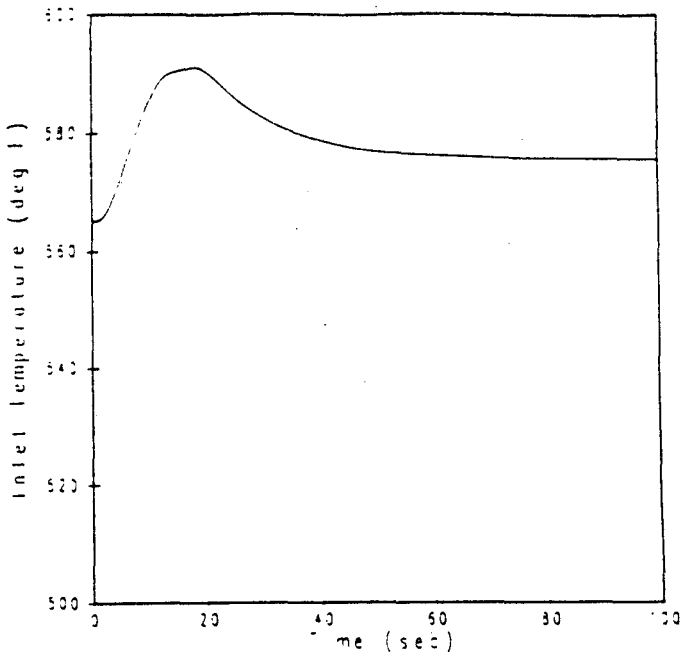
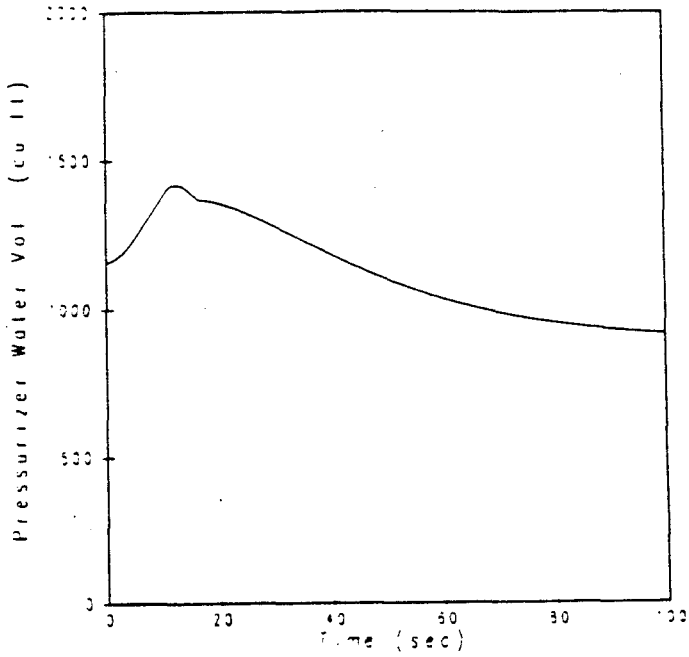
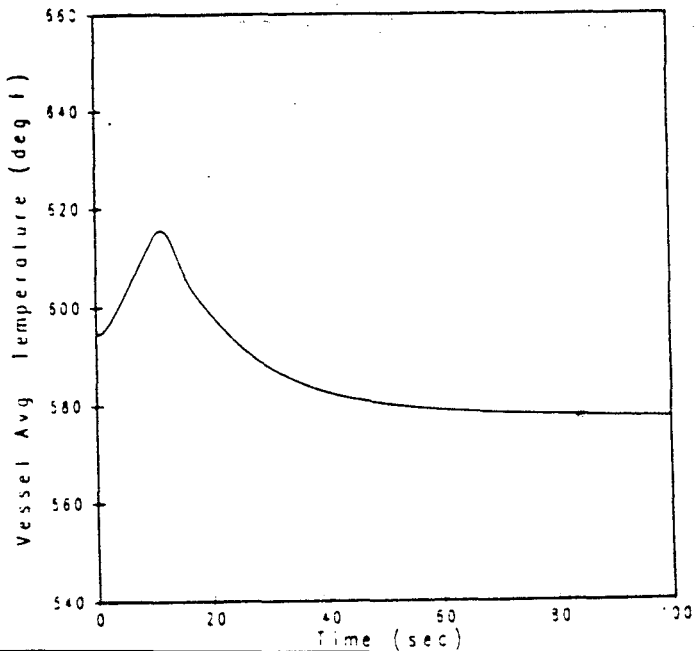
Figure 15.2-20 Loss of Load Accident with Pressurizer Spray and Power—Operated Relief Valves, Beginning-of-Life







WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves, Beginning of Life
FIGURE 15.2-20

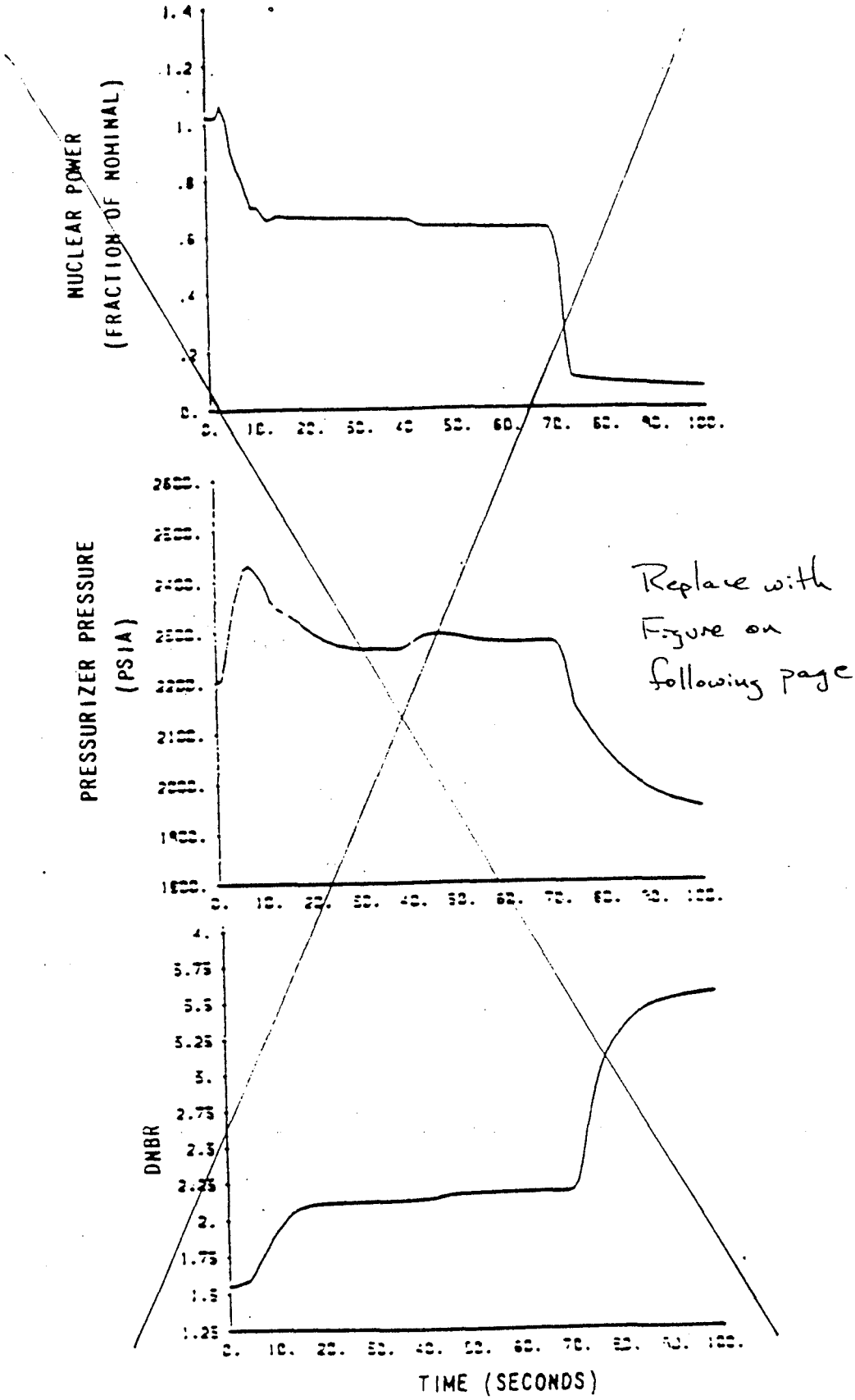
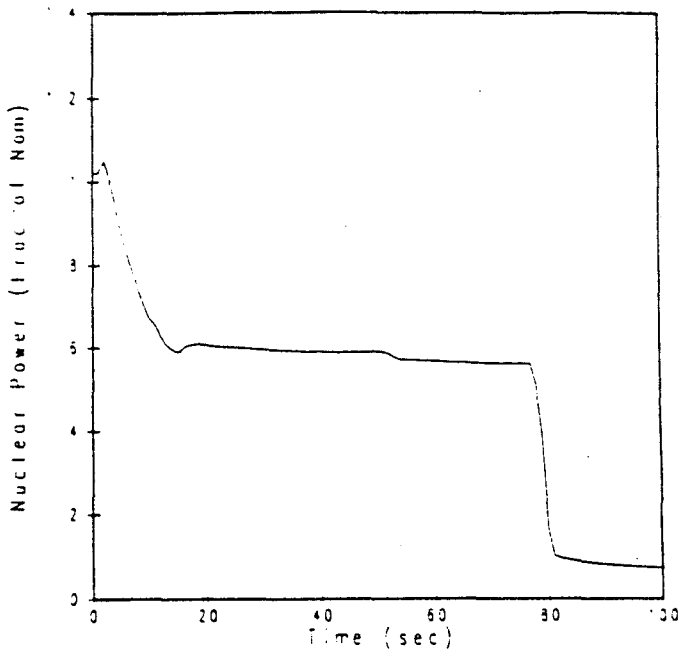
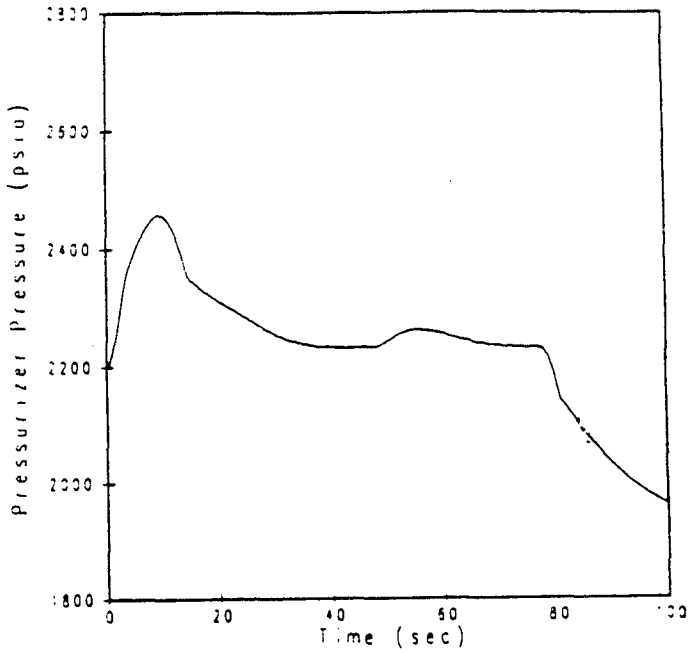
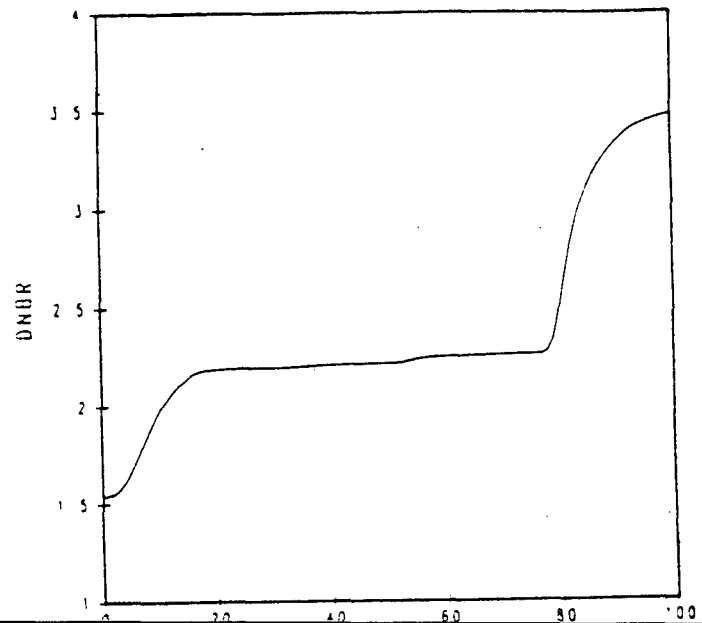


Figure 15.2-21 Loss of Load Accident with Pressurizer Spray and Power—Operated Relief Valves. End-of-Life







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Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves, End of Life
FIGURE 15.2-21

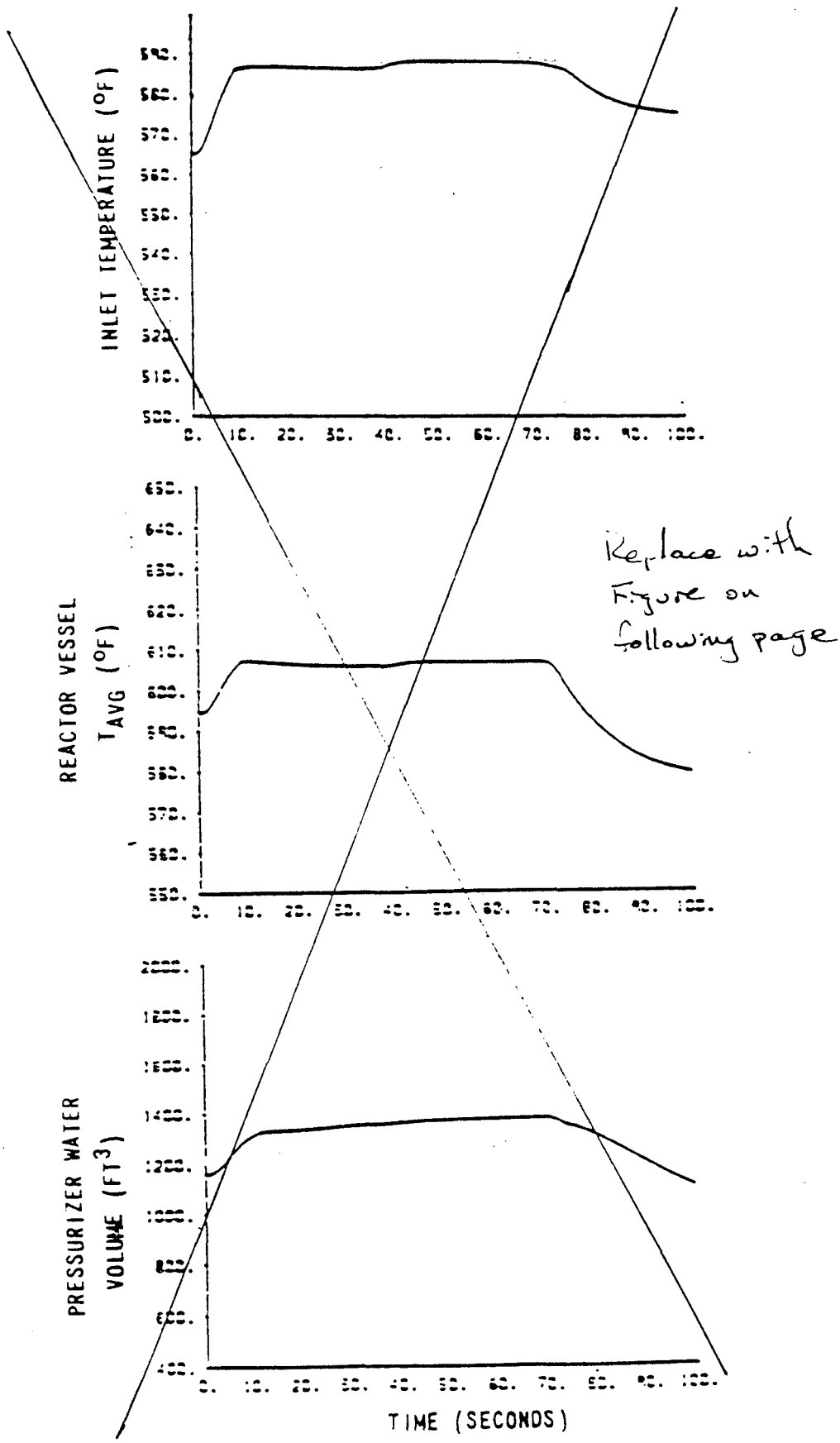
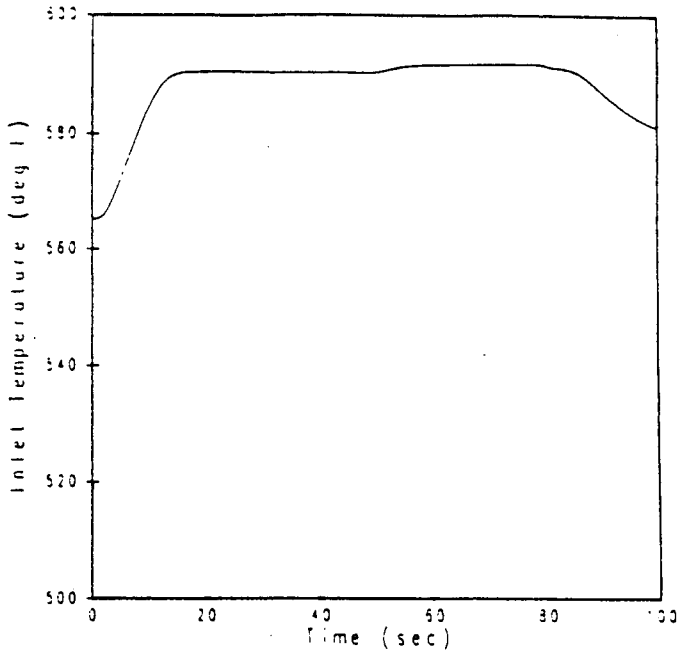
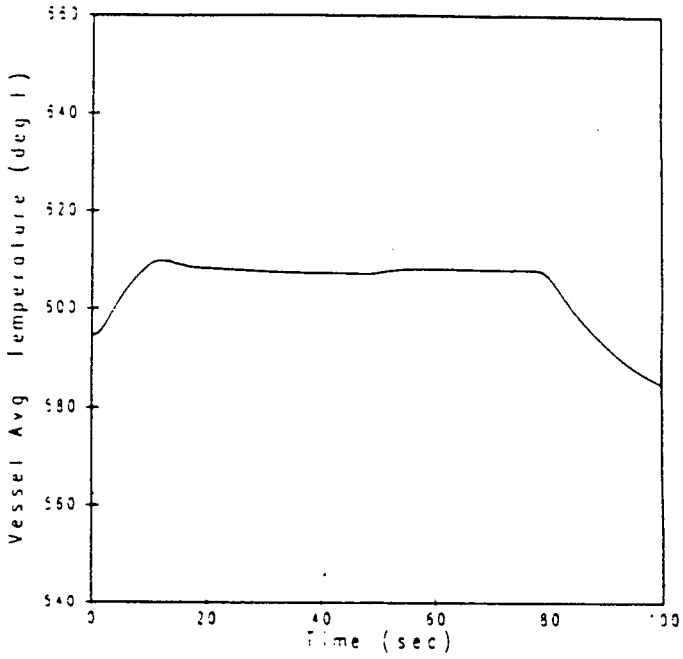
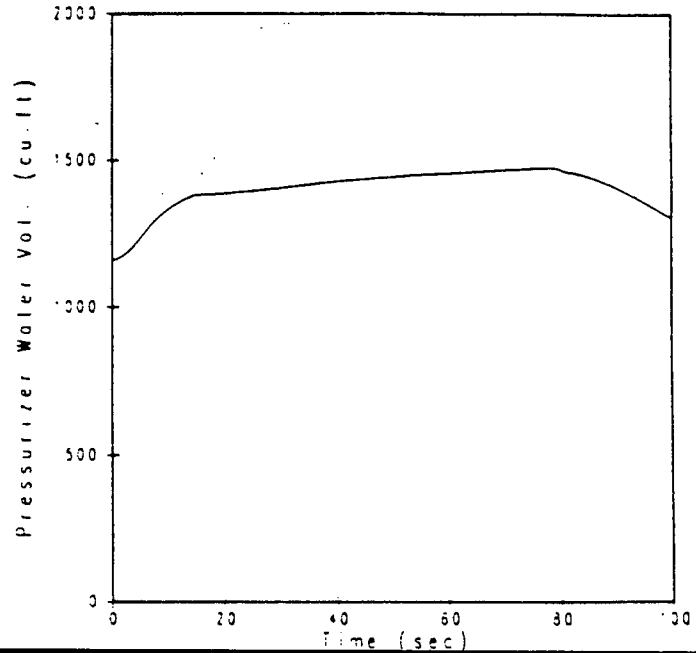


Figure 15.2-22 Loss of Load Accident with Pressurizer Spray and Power—Operated Relief Valves, End-of-Life







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Loss of Load Accident with Pressurizer Spray and Power-Operated Relief Valves, End of Life
FIGURE 15.2-22

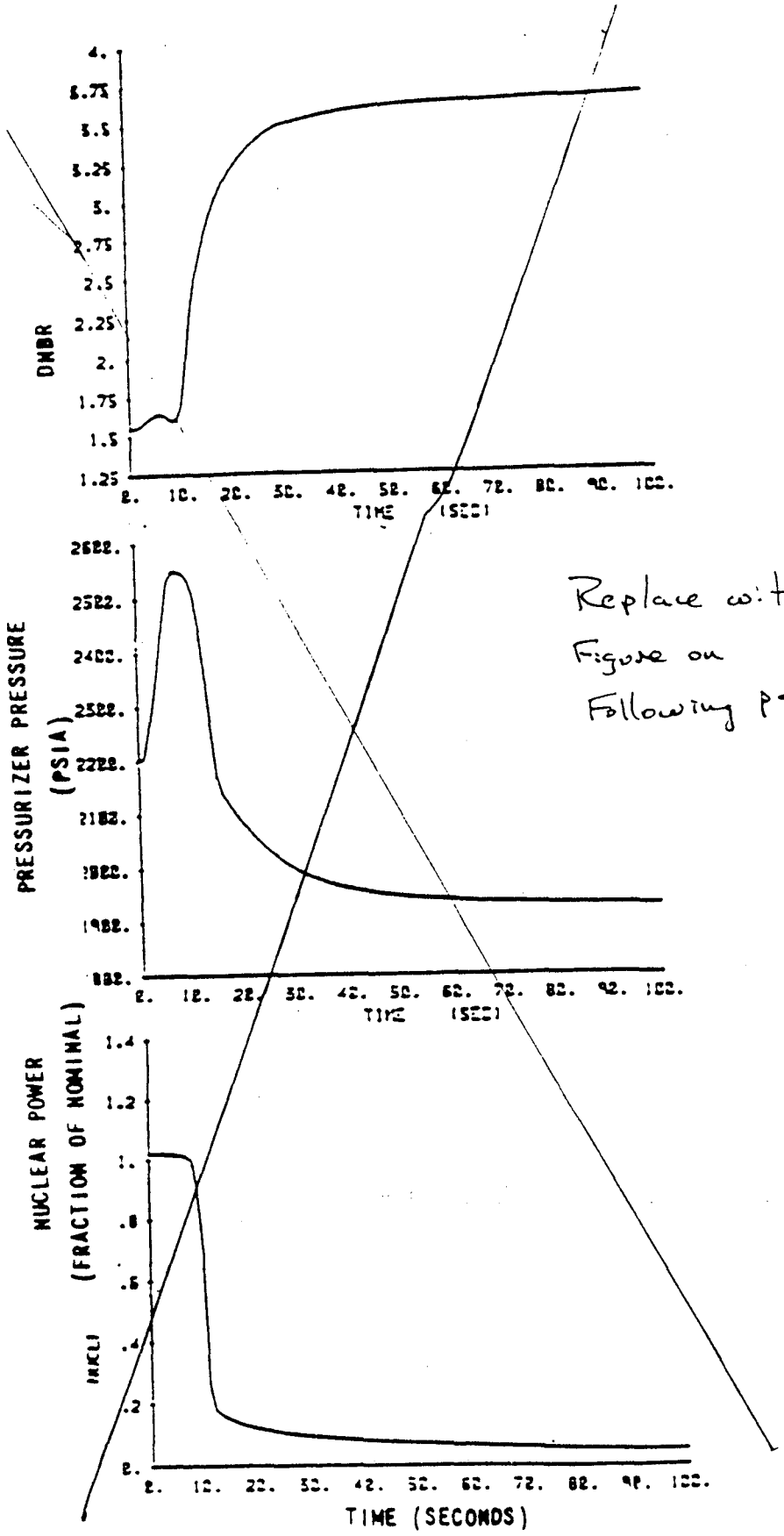

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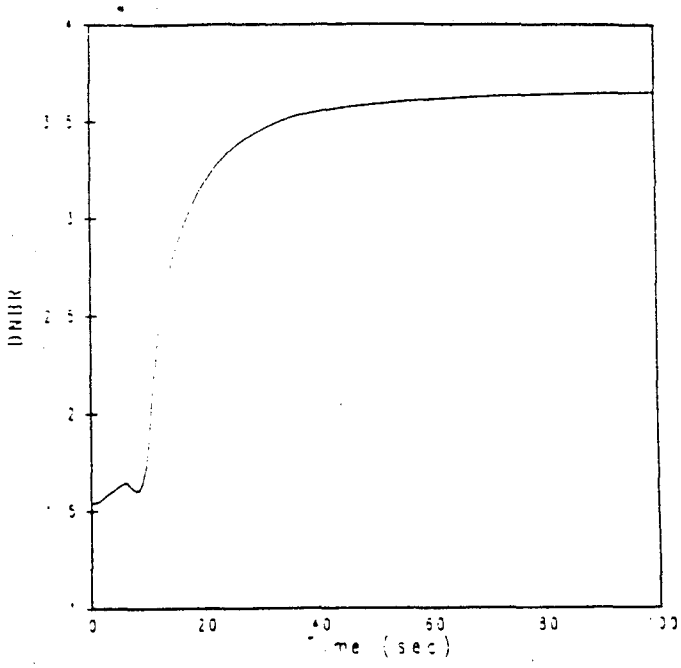
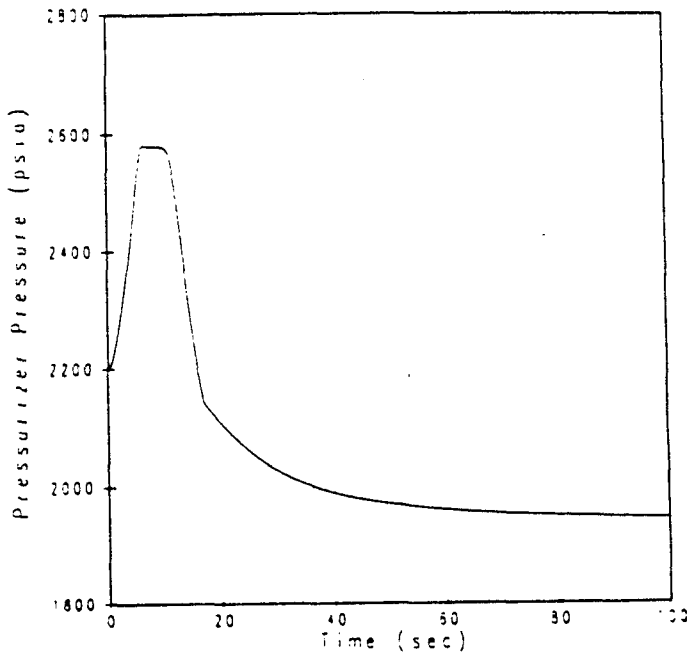
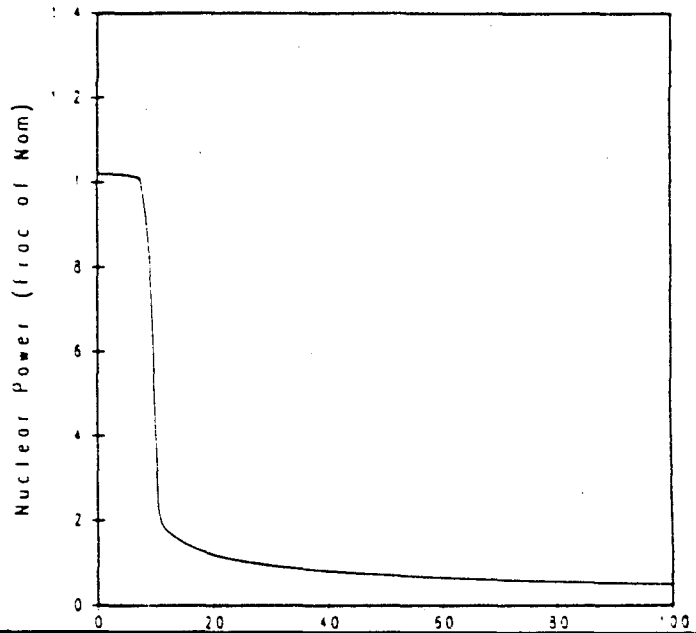
Figure 15.2-23 Loss of Load Accident Without Pressurizer Spray and Power – Operated Relief Valves, Beginning-of Life







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Loss of Load Accident without Pressurizer Spray and Power-Operated Relief Valves, Beginning of Life
FIGURE 15.2-23

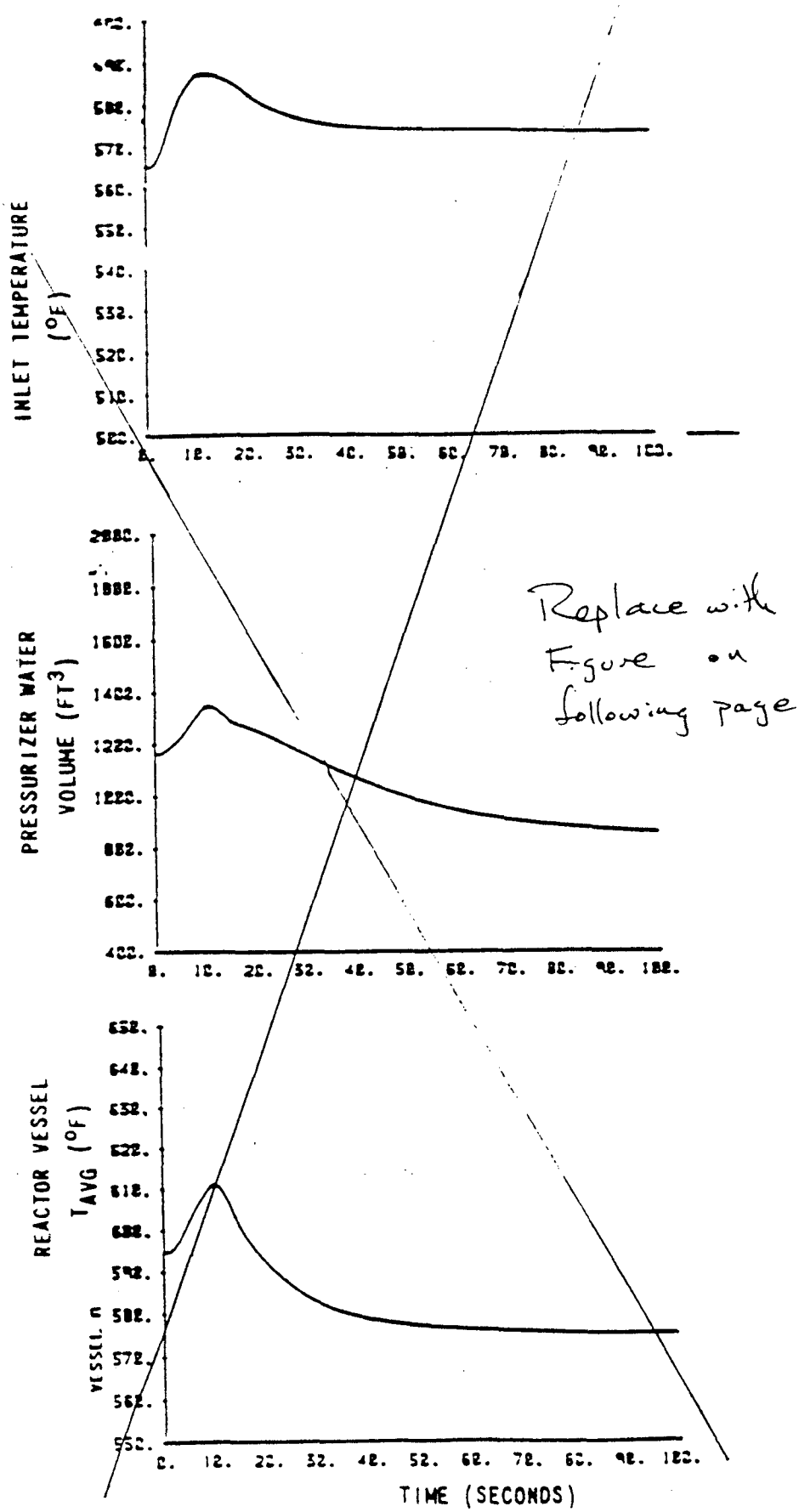

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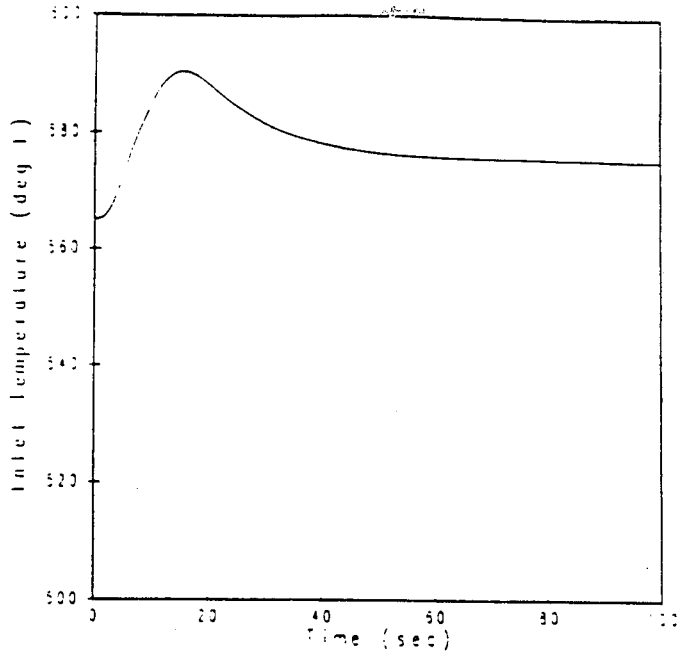
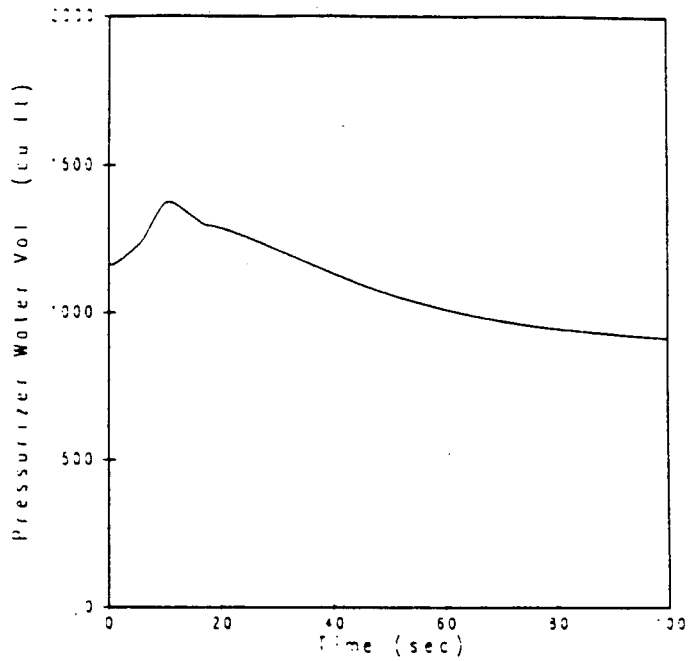
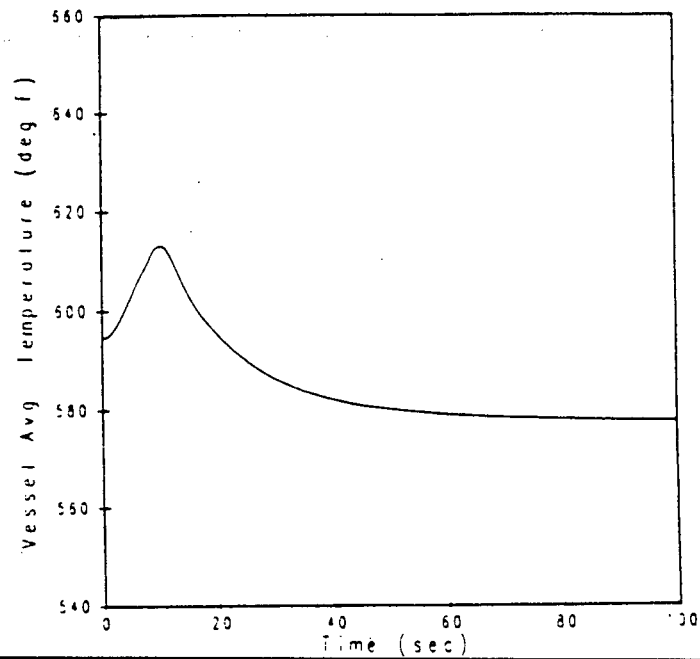
Figure 15.2-24 Loss of Load Accident Without Pressurizer Spray and Power – Operated Relief Valves, Beginning-Of-Life







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Loss of Load Accident without Pressurizer Spray and Power-Operated Relief Valves, Beginning of Life
FIGURE 15.2-24

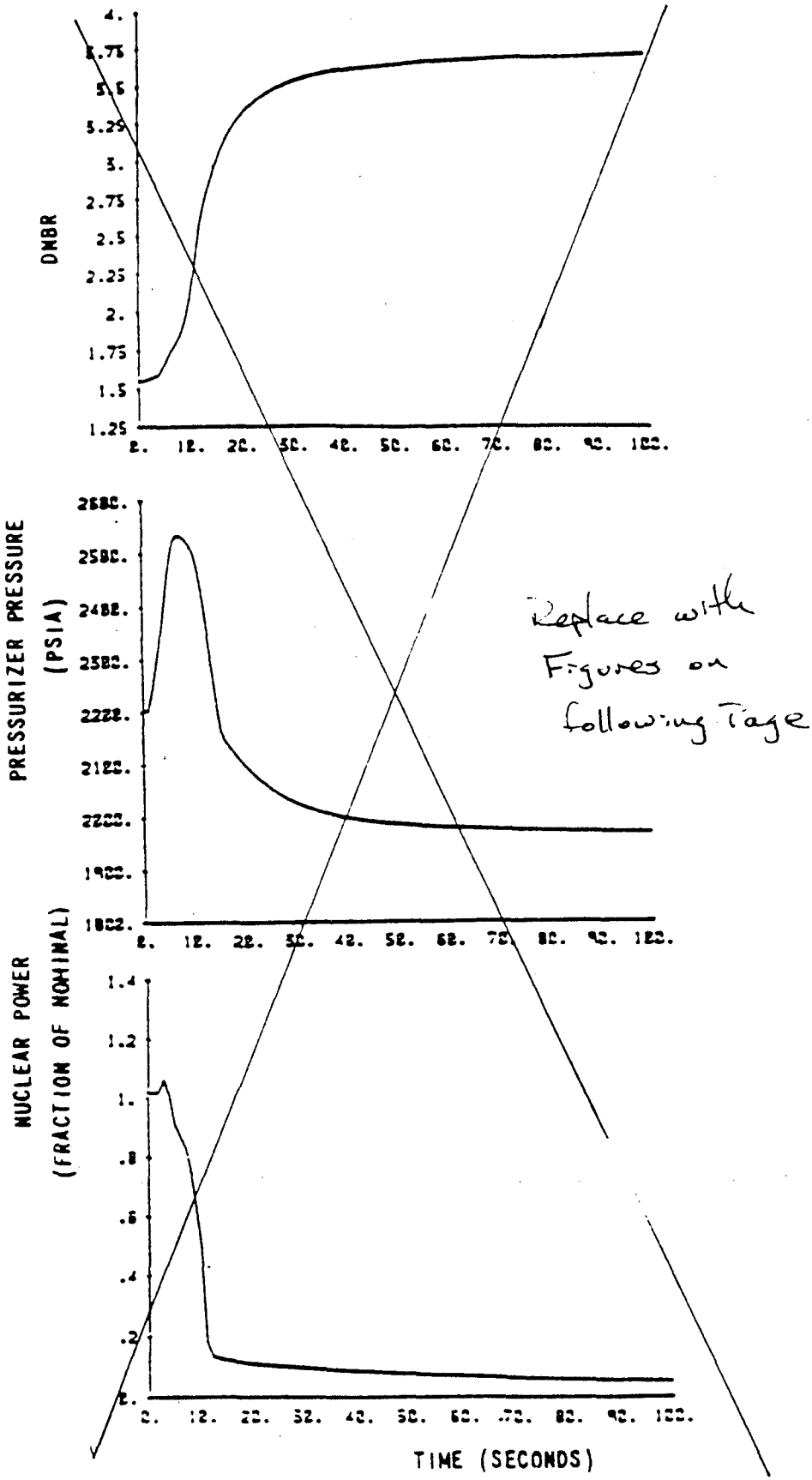
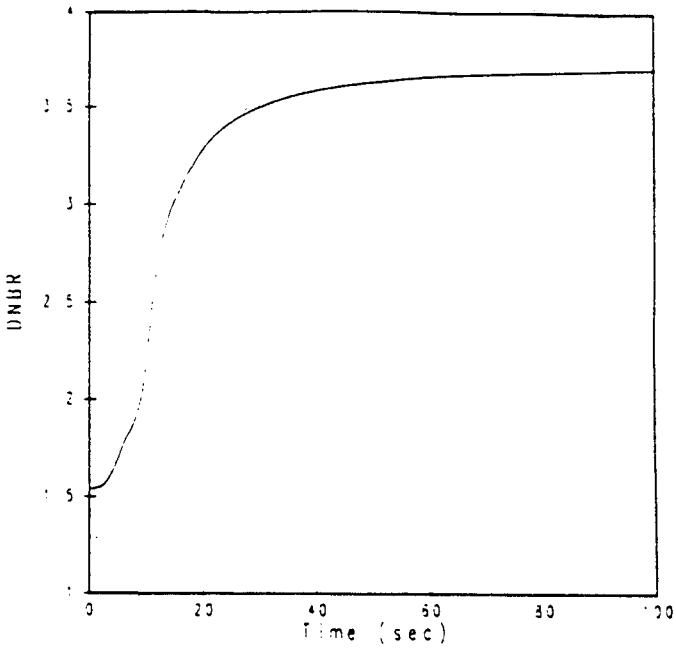
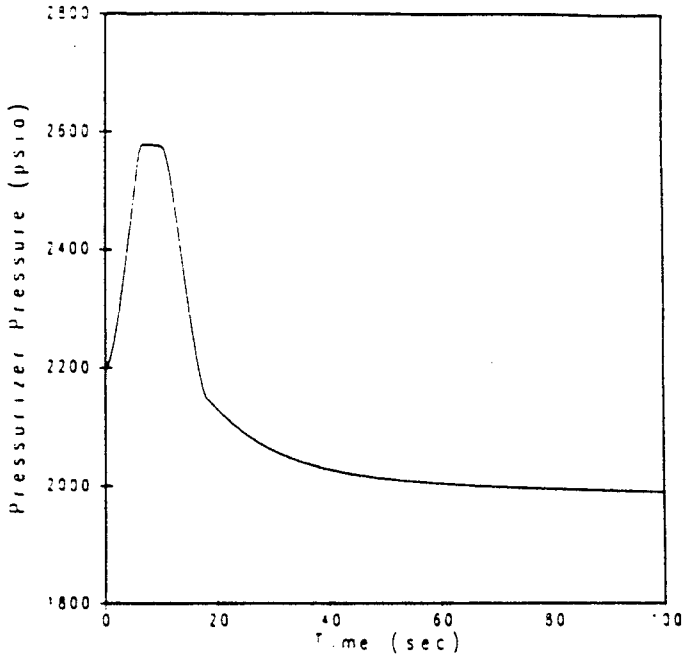
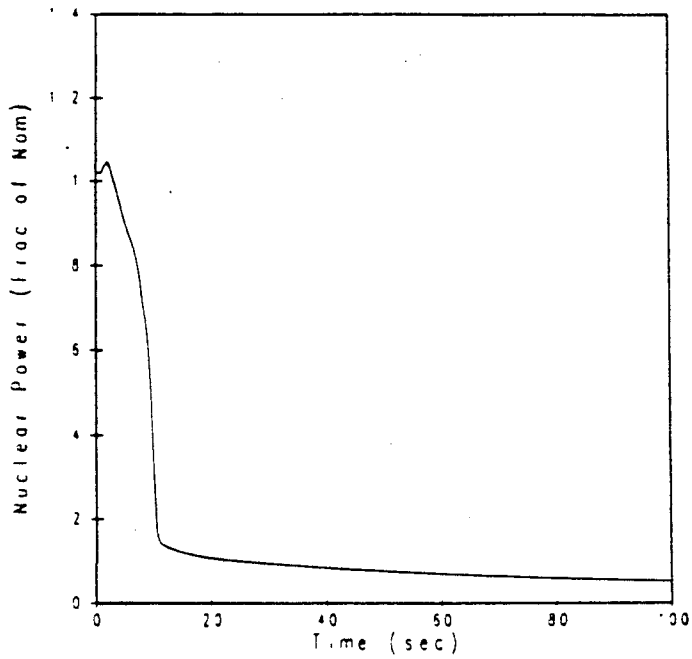


Figure 15.2 25 Loss of Load Accident Without Pressurizer Spray and Power Operated Relief Valves, End-of-Life







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Loss of Load Accident without Pressurizer Spray and Power-Operated Relief Valves, End of Life
FIGURE 15.2-25

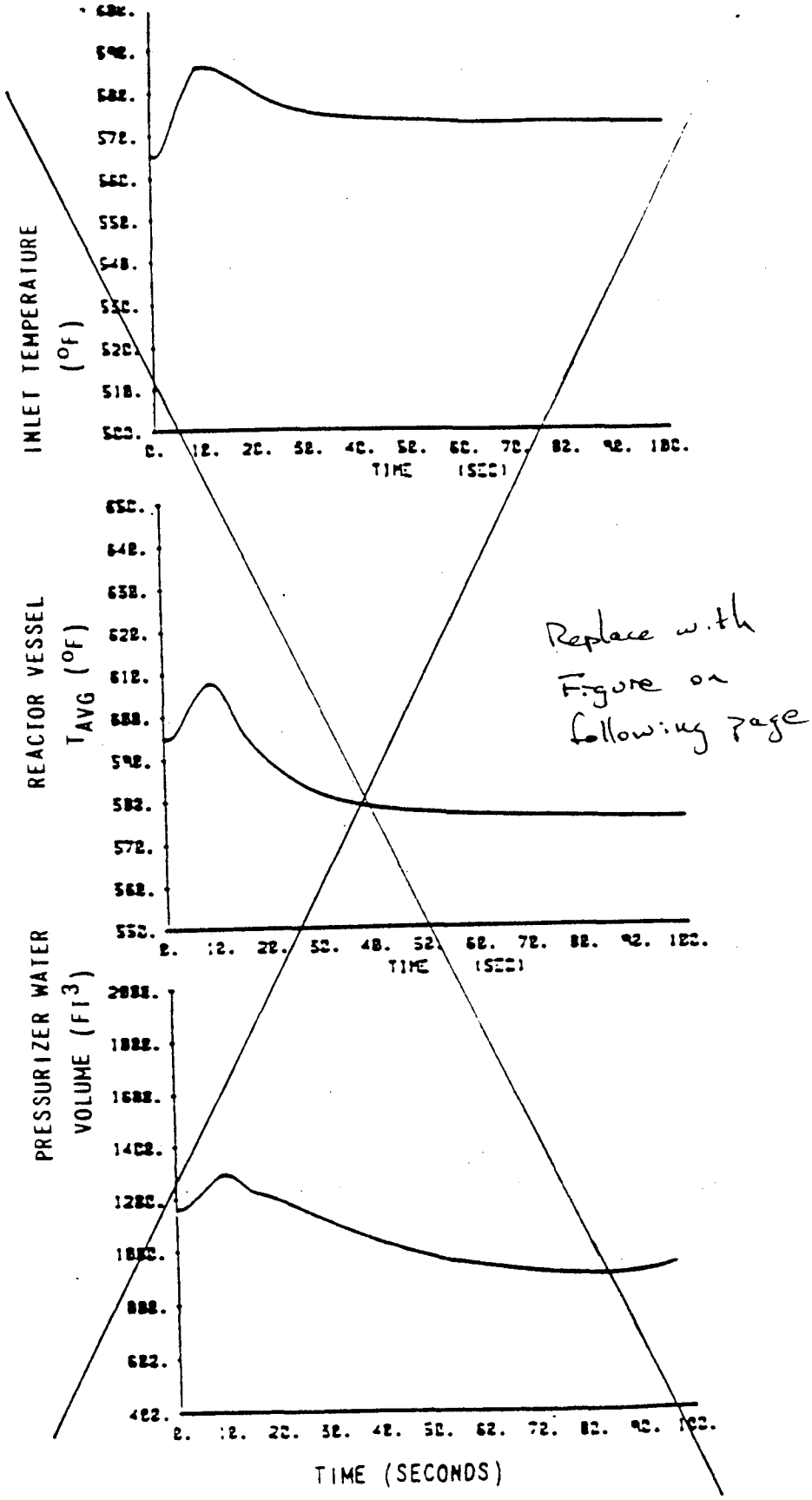
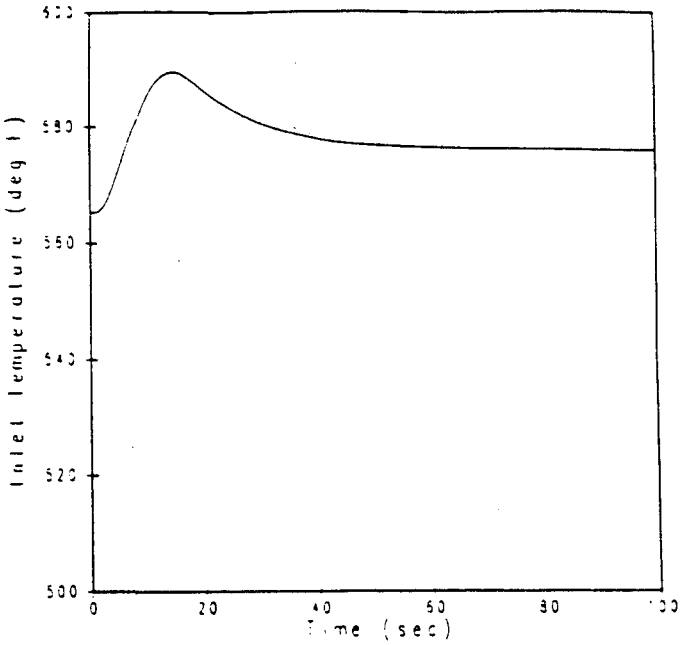
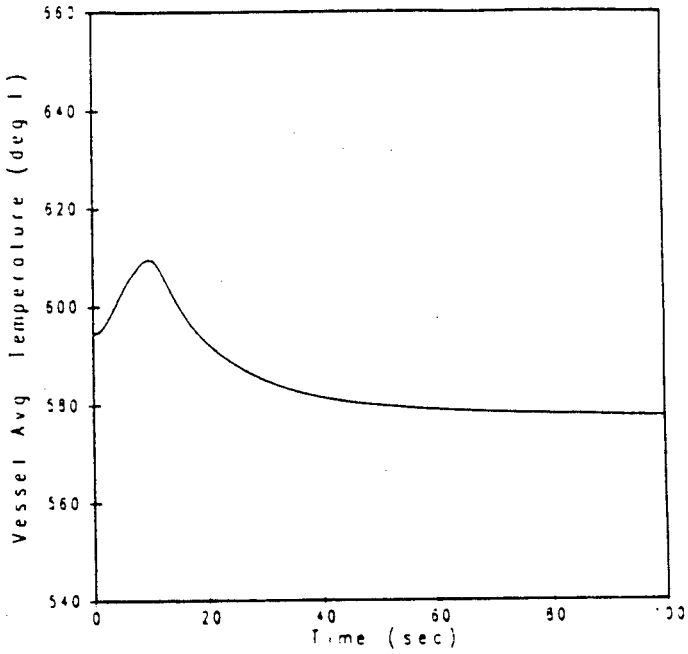
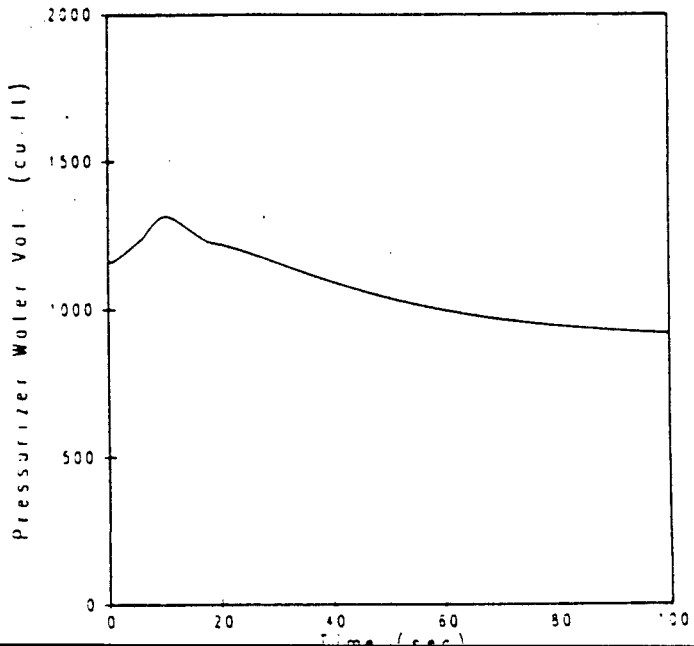


Figure 15.2-26 Loss of Load Accident Without Pressurizer Spray and Power-Operated Relief Valves, End-Of-Life







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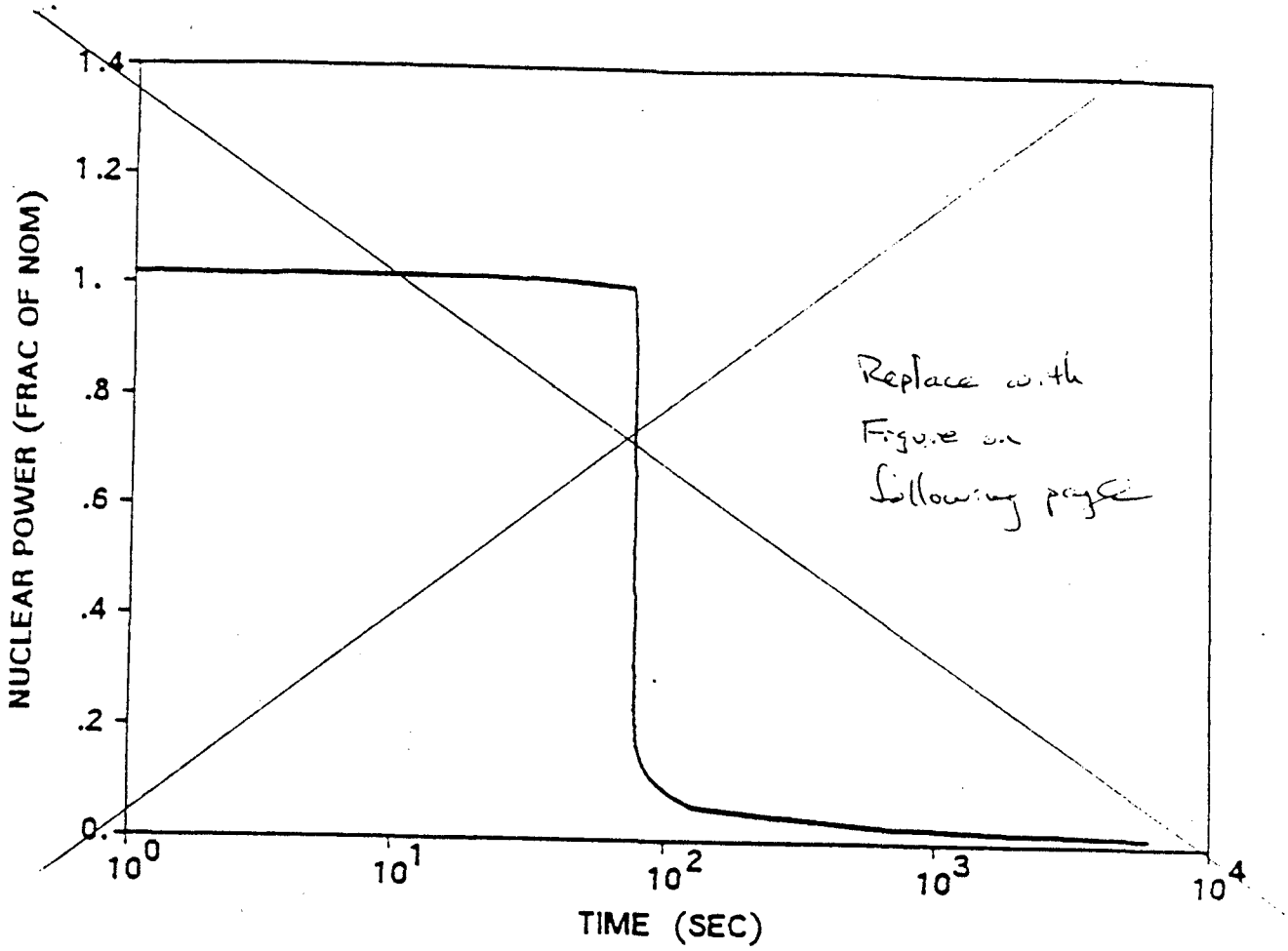
Loss of Load Accident without Pressurizer Spray and Power-Operated Relief Valves, End of Life
FIGURE 15.2-26



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NUCLEAR POWER TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27A

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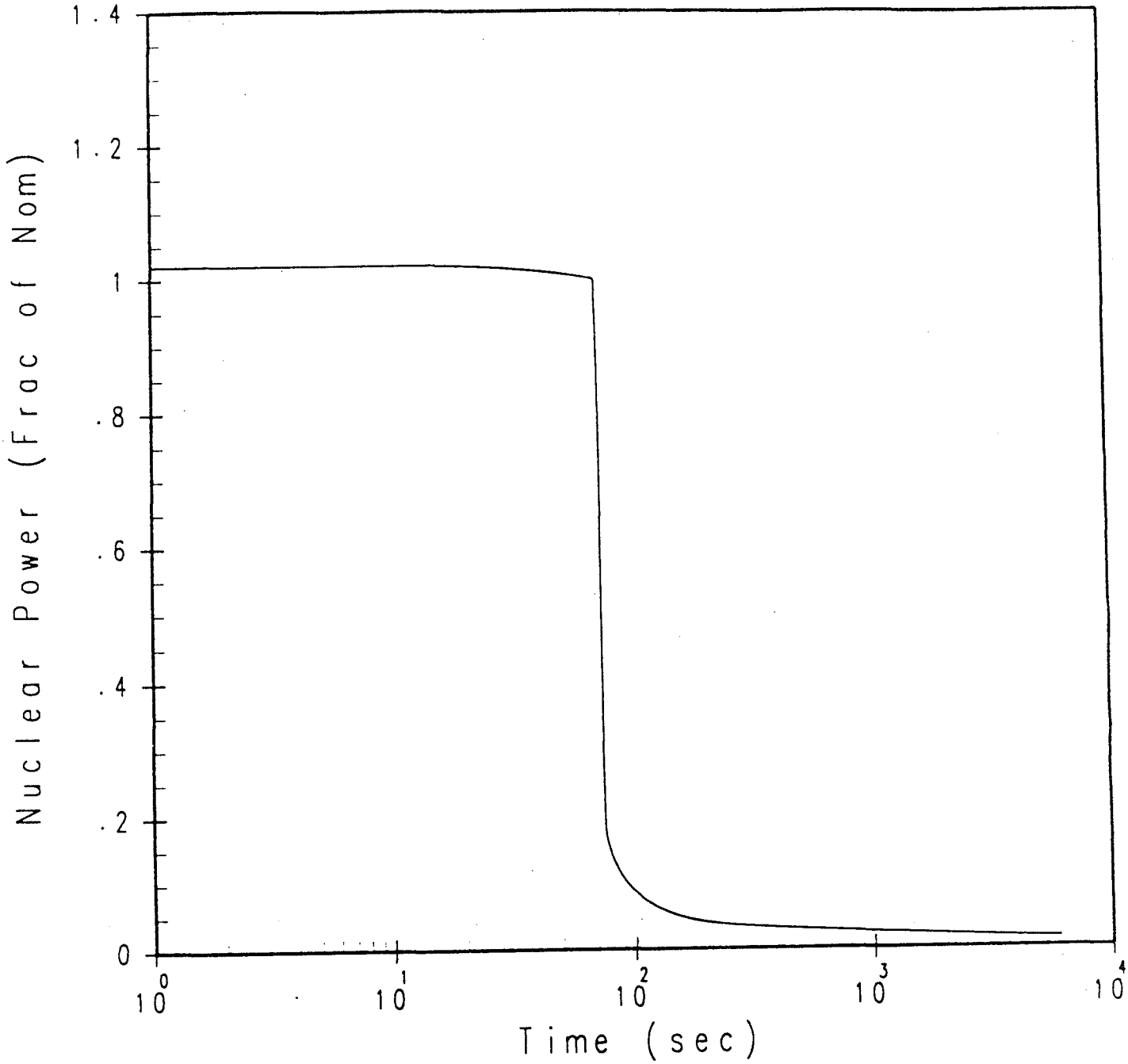

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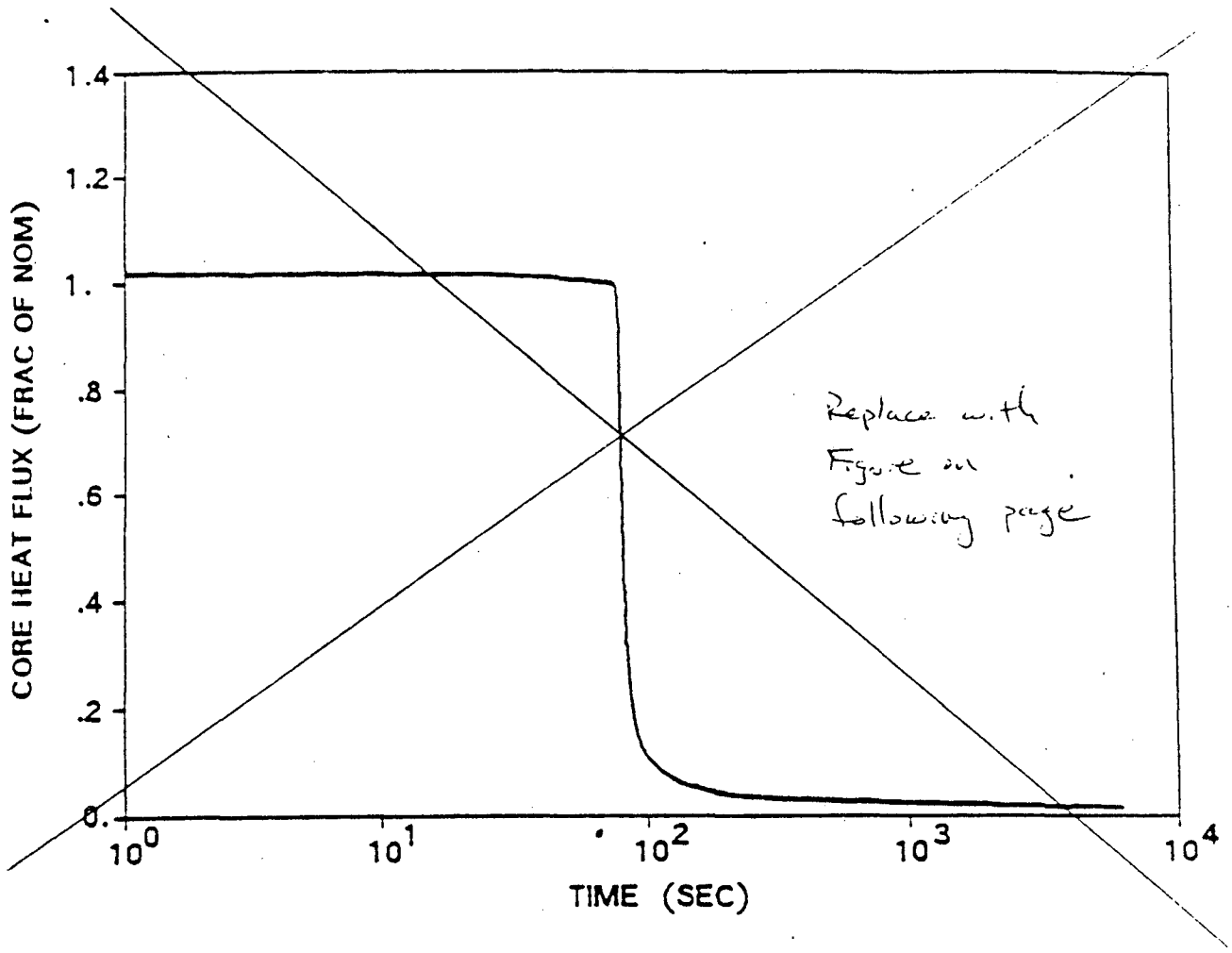
Nuclear Power Transient
for Loss of
Normal Feedwater
FIGURE 15.2-27A



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CORE HEAT FLUX TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27B

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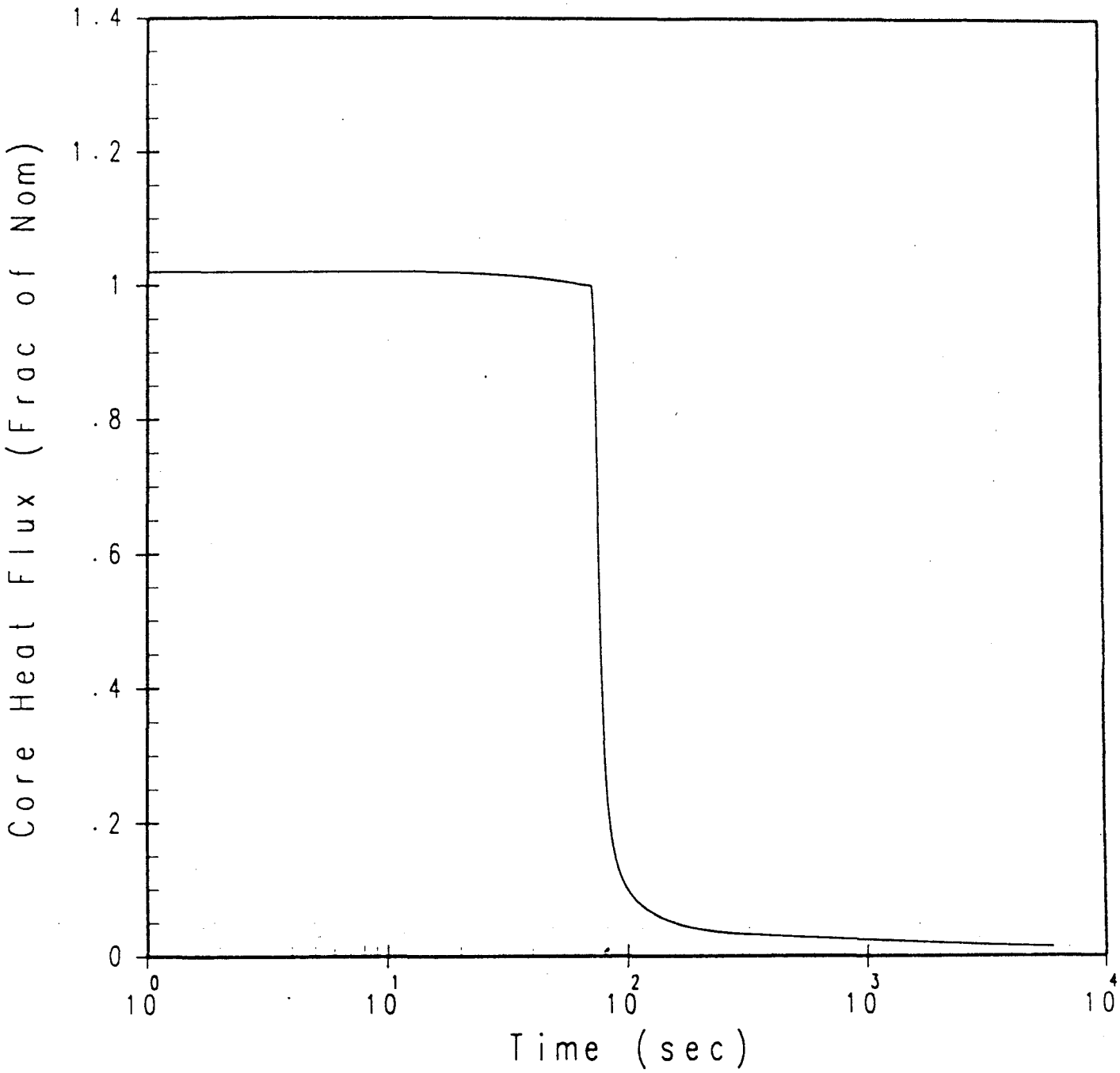

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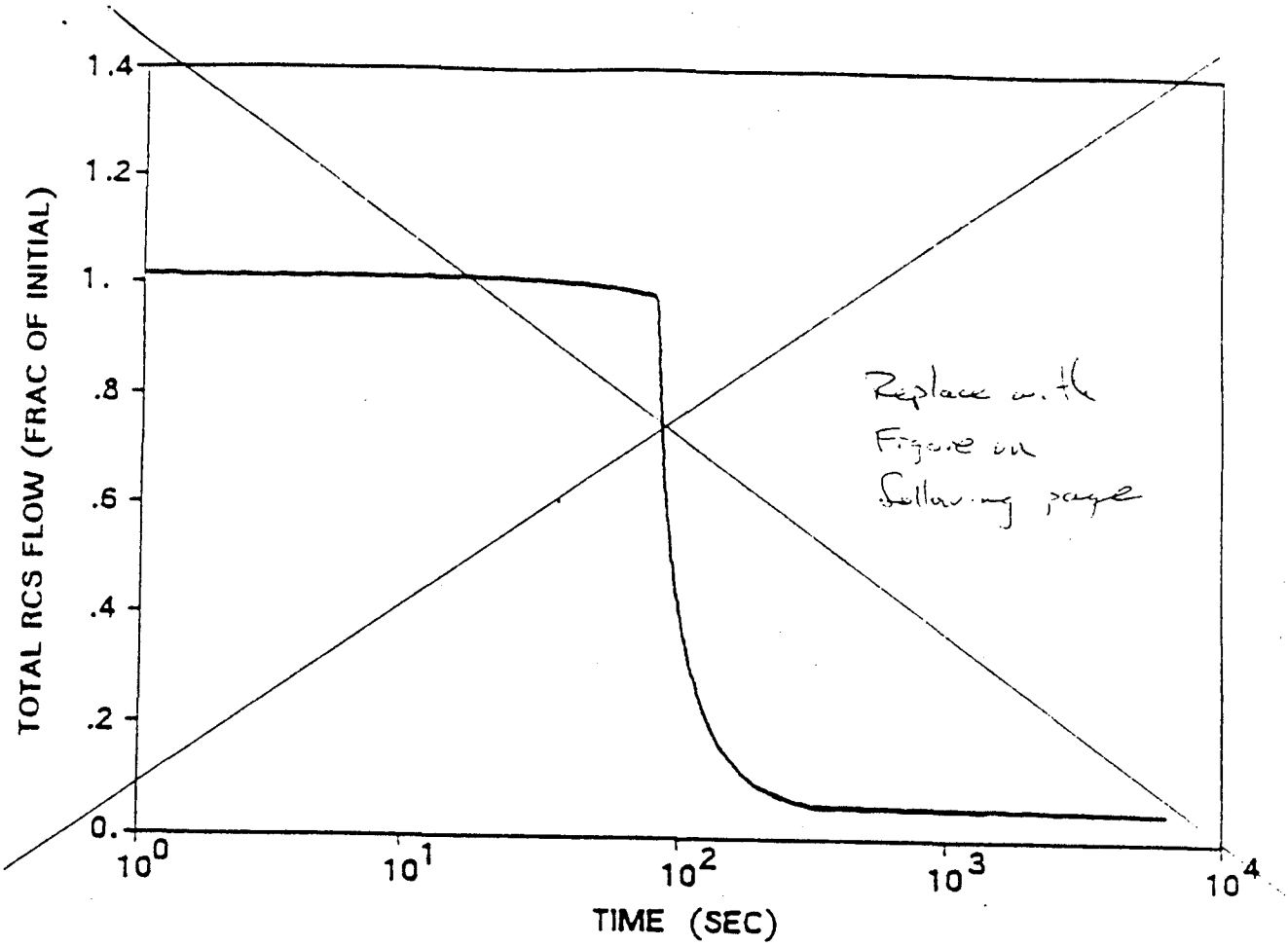
Core Heat Flux Transient
for Loss of
Normal Feedwater
FIGURE 15.2-27B



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FLOW TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27C

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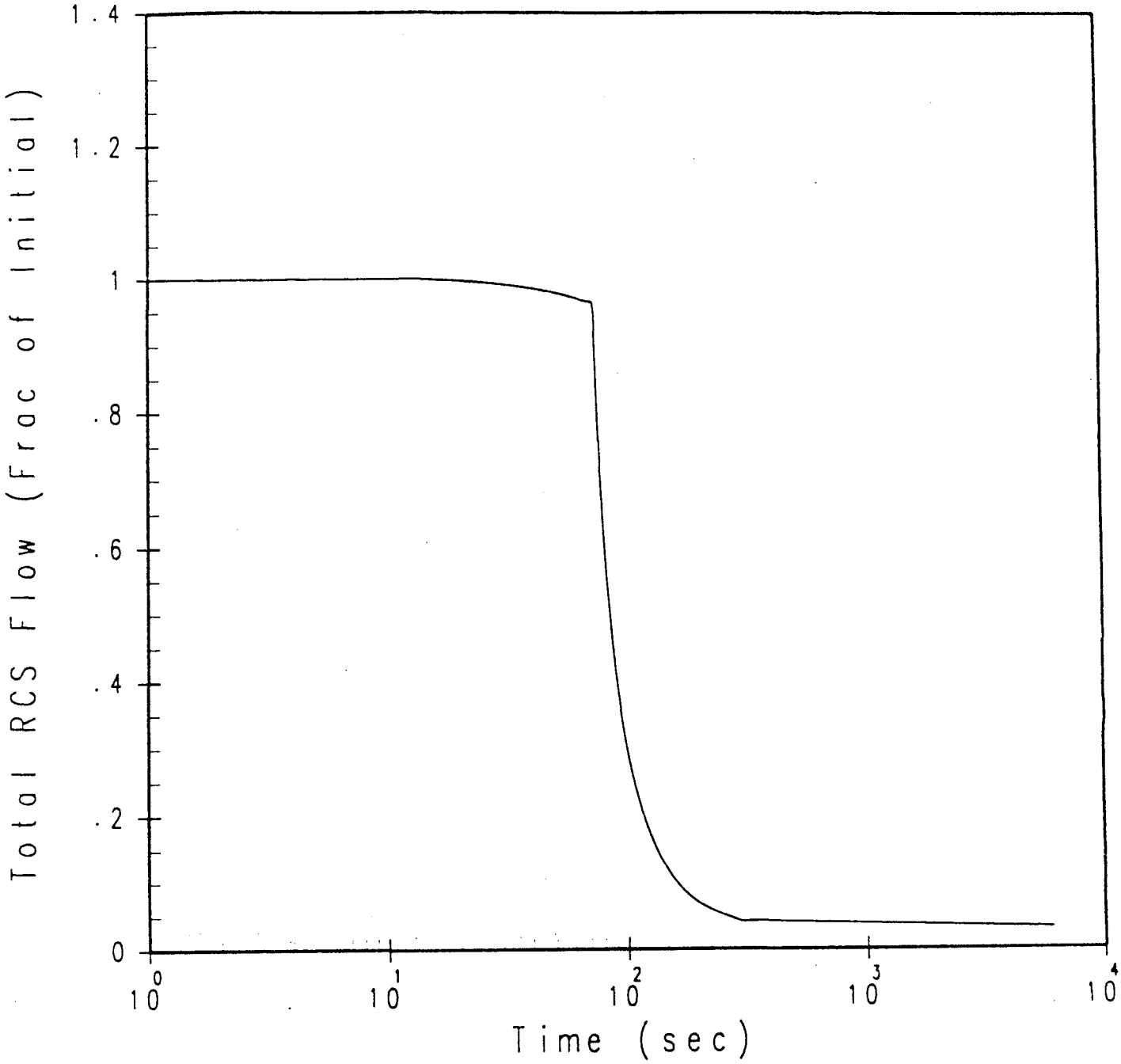

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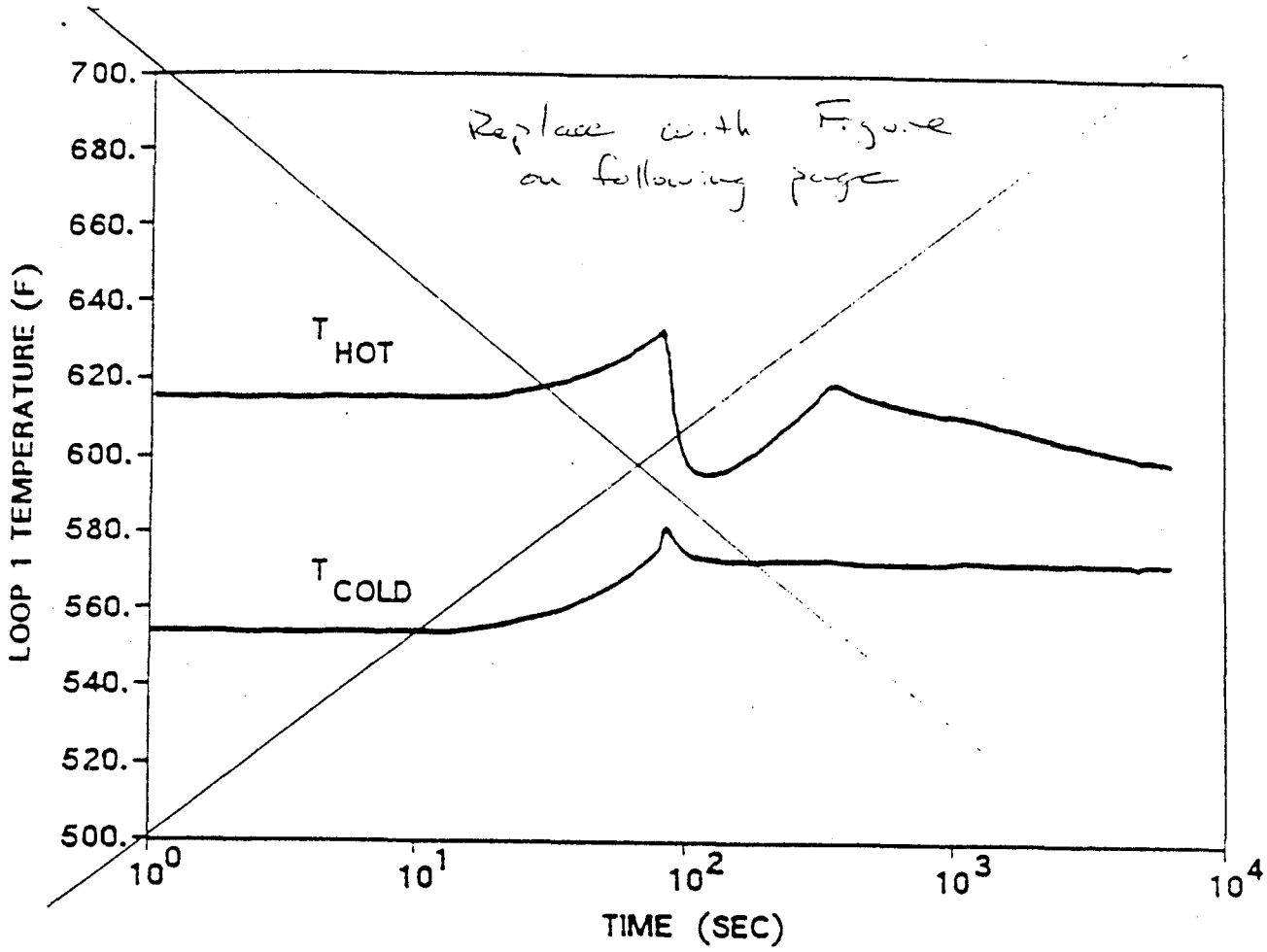
FlowTransient
for Loss of
Normal Feedwater
FIGURE 15.2-27C



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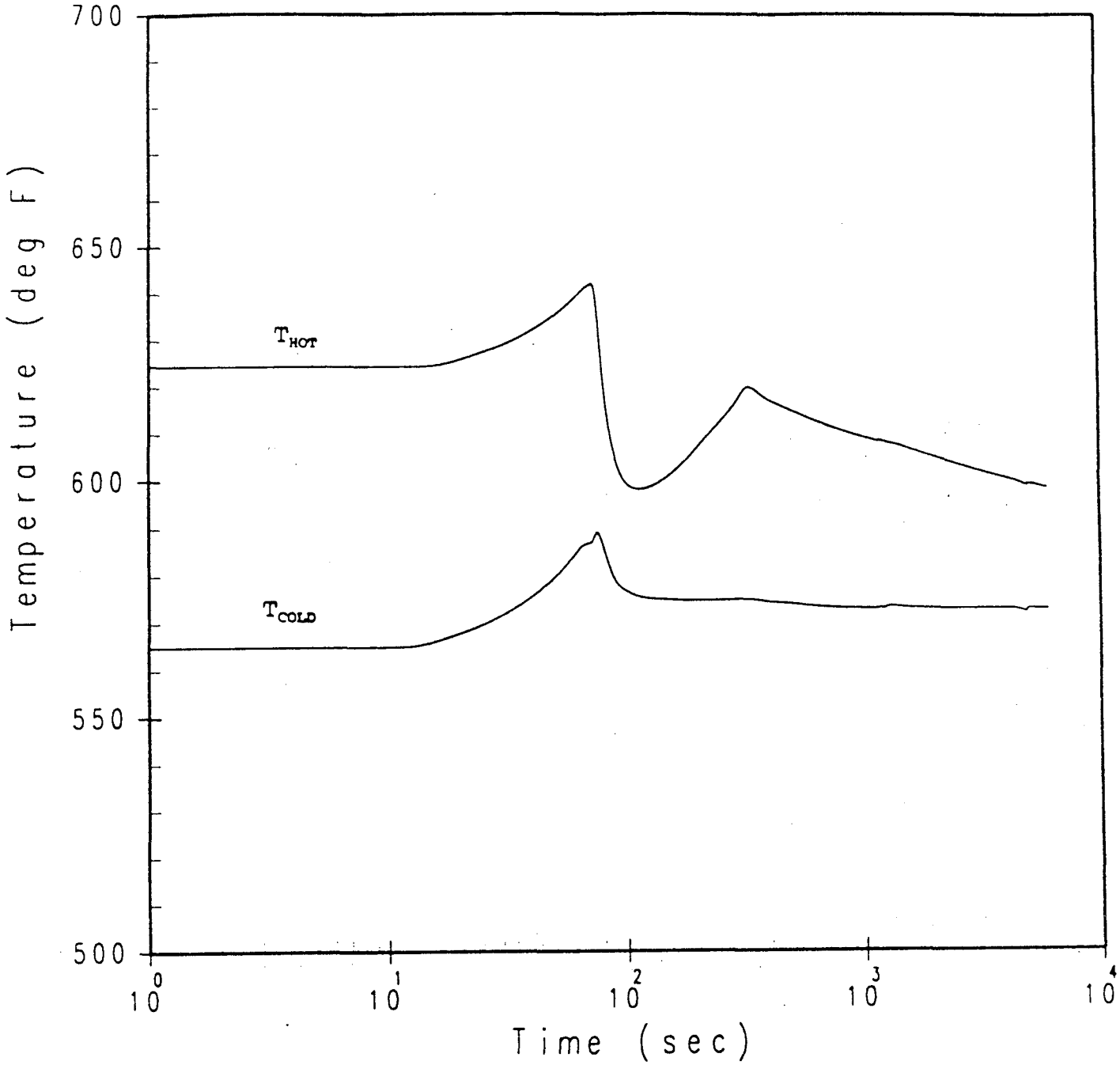
REACTOR COOLANT SYSTEM
TEMPERATURE TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27D



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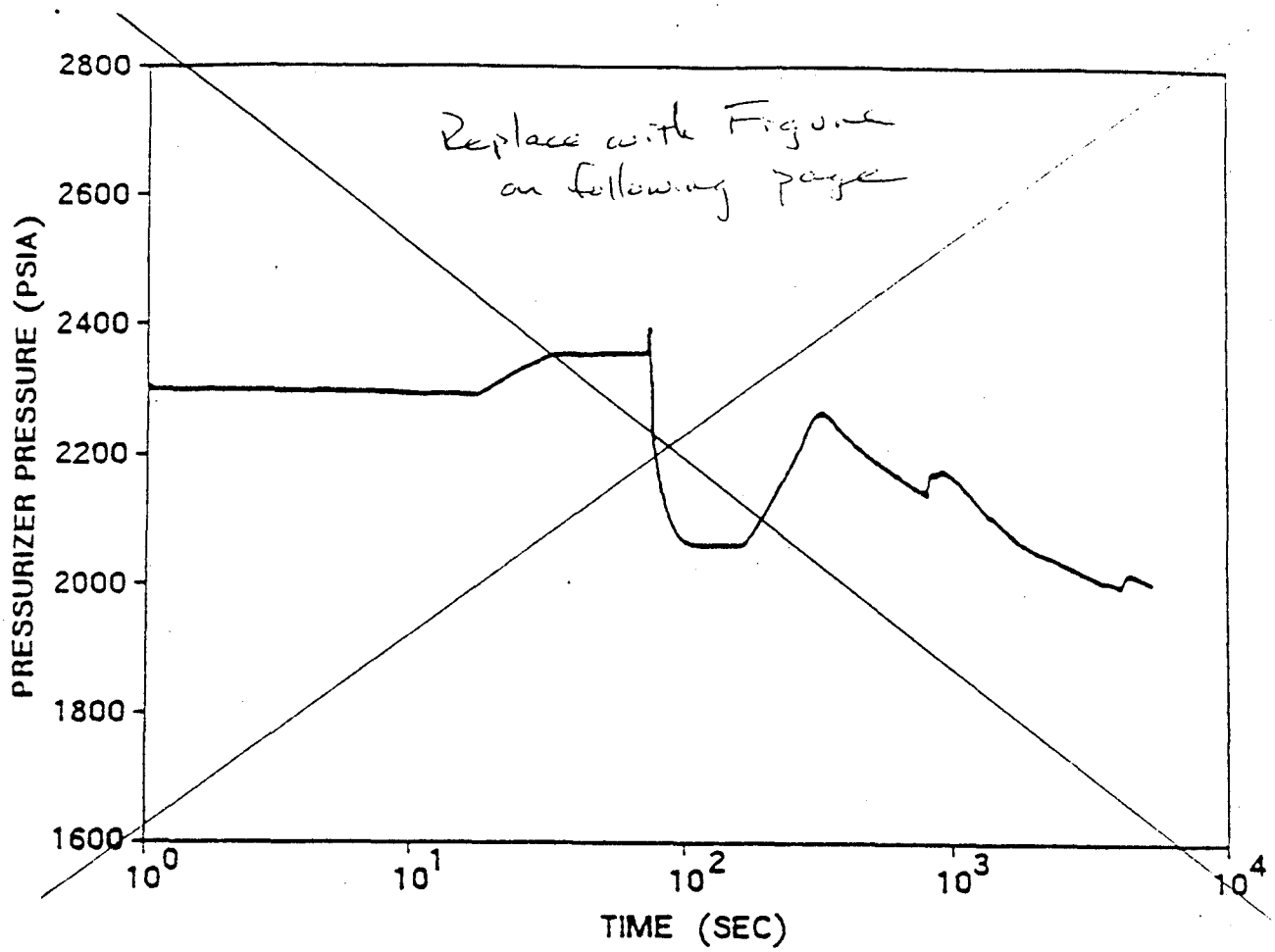
Reactor Coolant Temperature
Transient for Loss of
Normal Feedwater
FIGURE 15.2-27D



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PRESSURIZER PRESSURE TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27F

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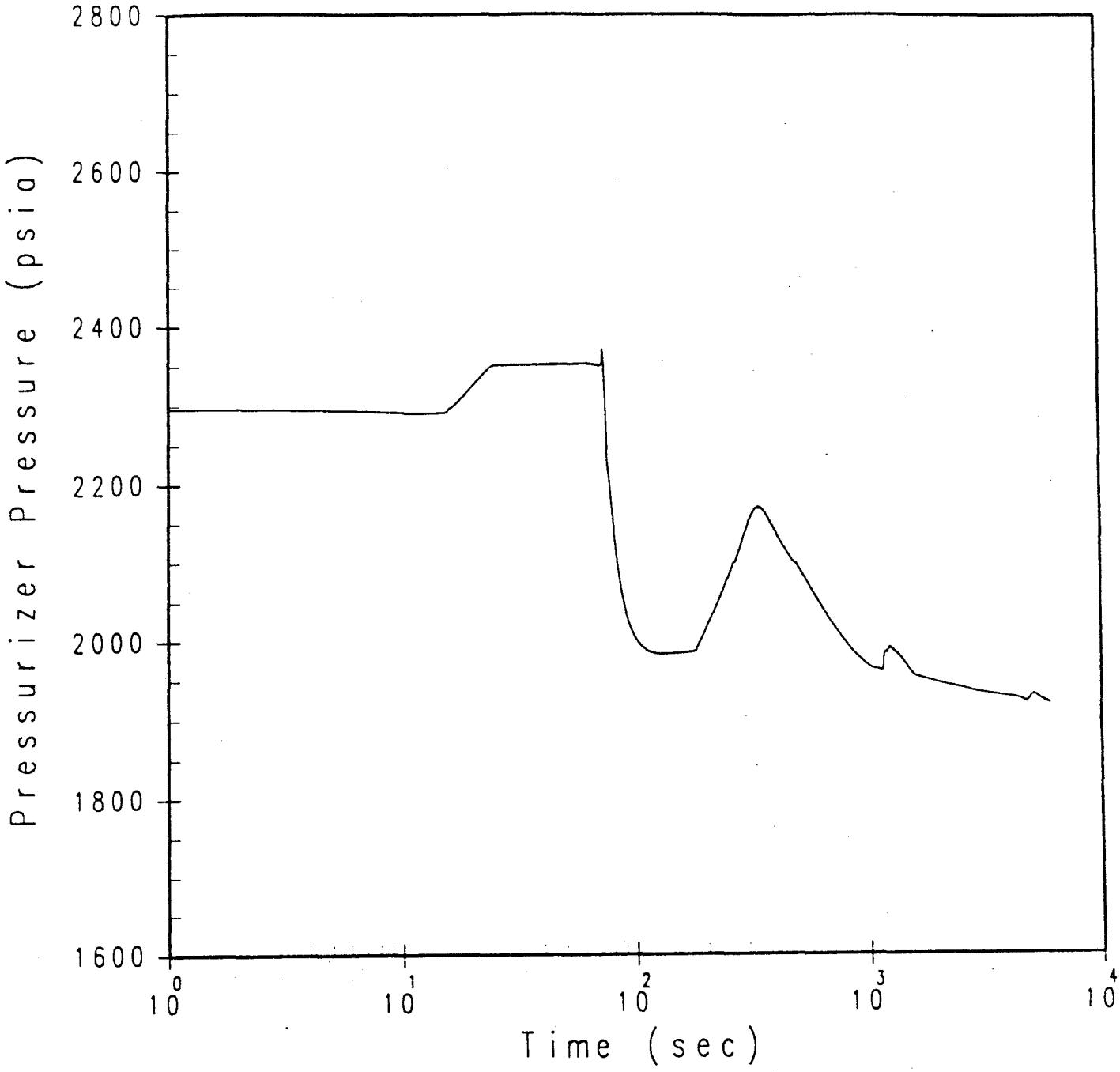

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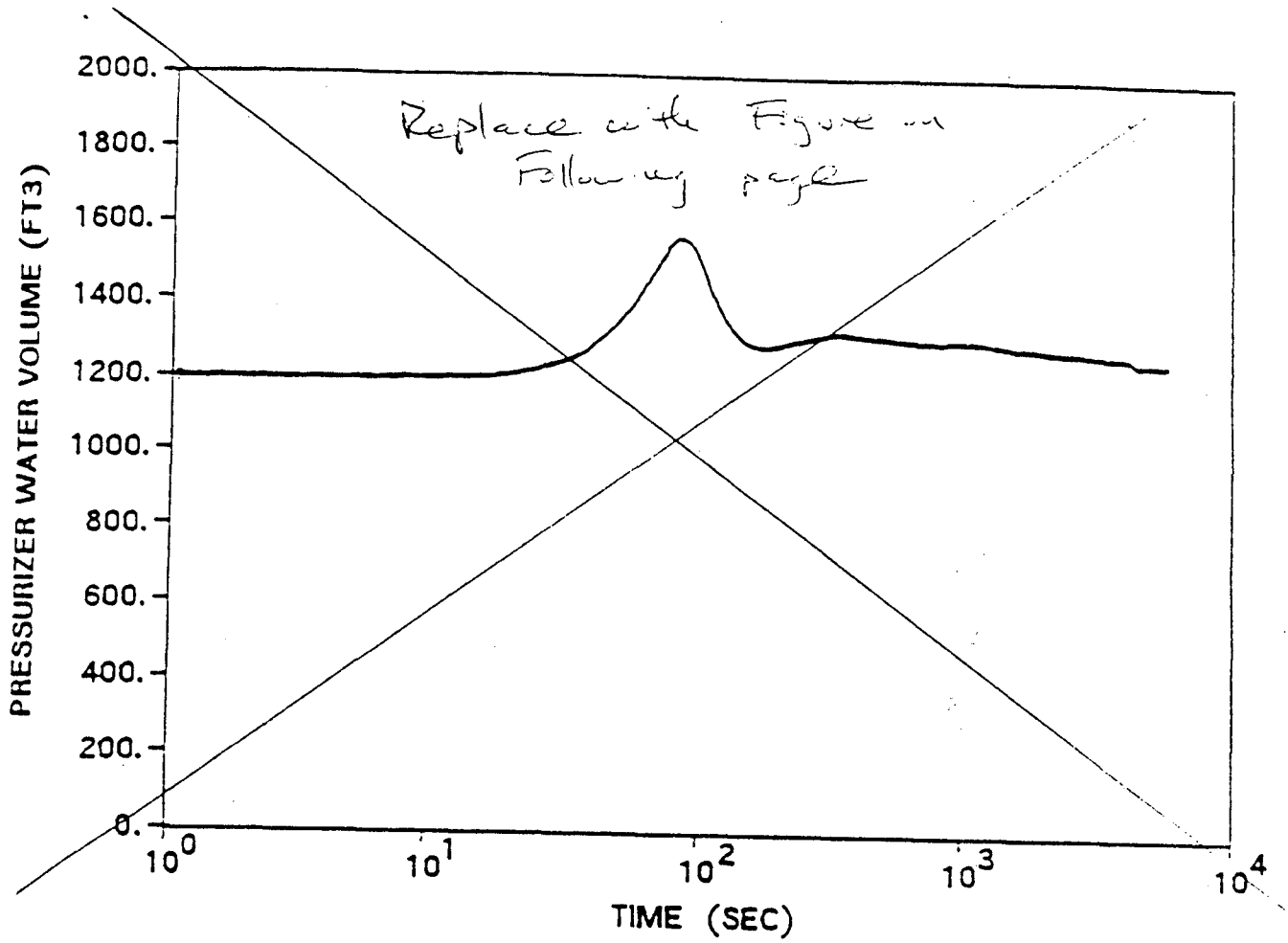
Pressurizer Pressure Transient
for Loss of
Normal Feedwater
FIGURE 15.2-27F



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PRESSURIZER WATER
VOLUME TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27G

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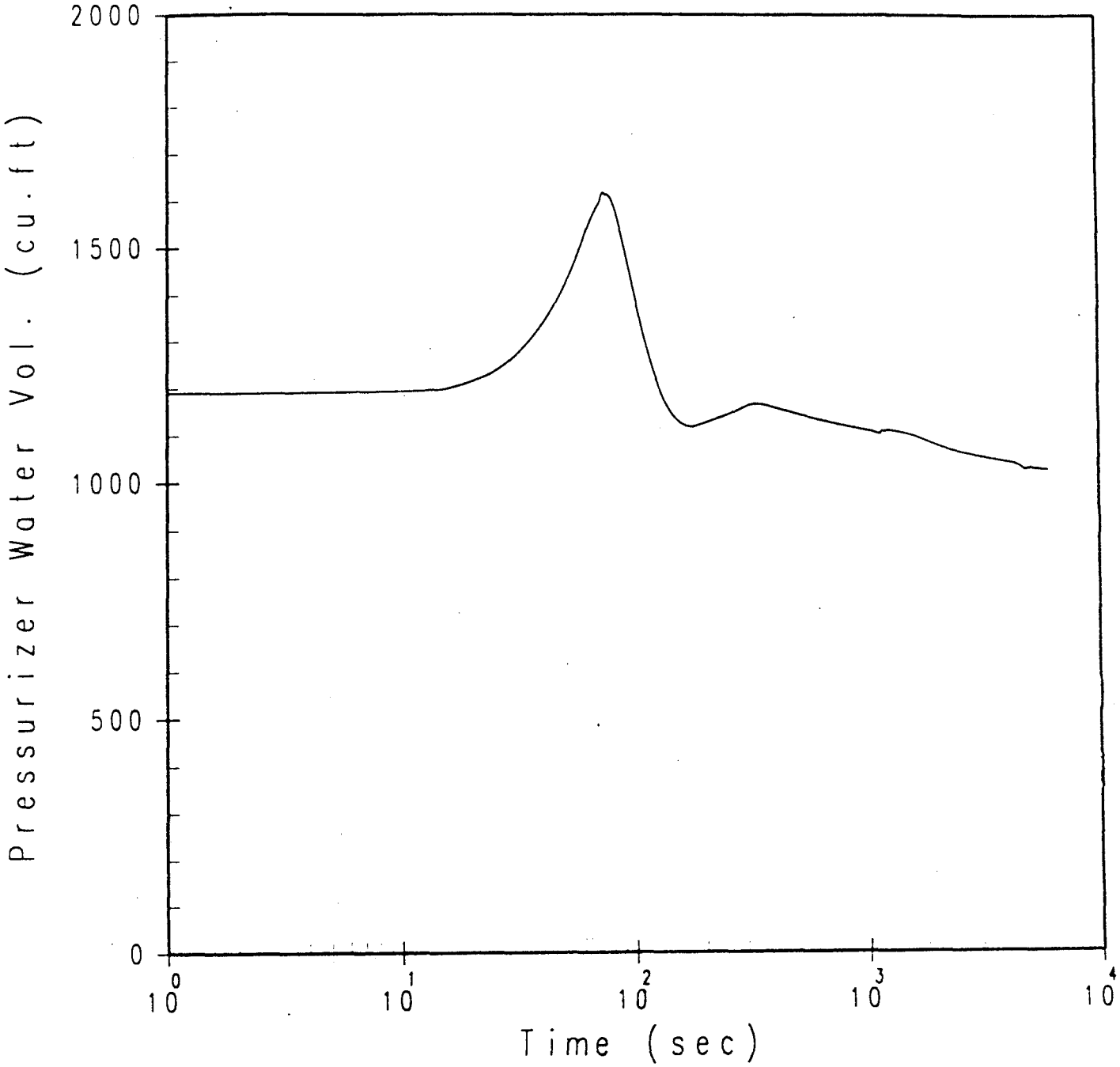

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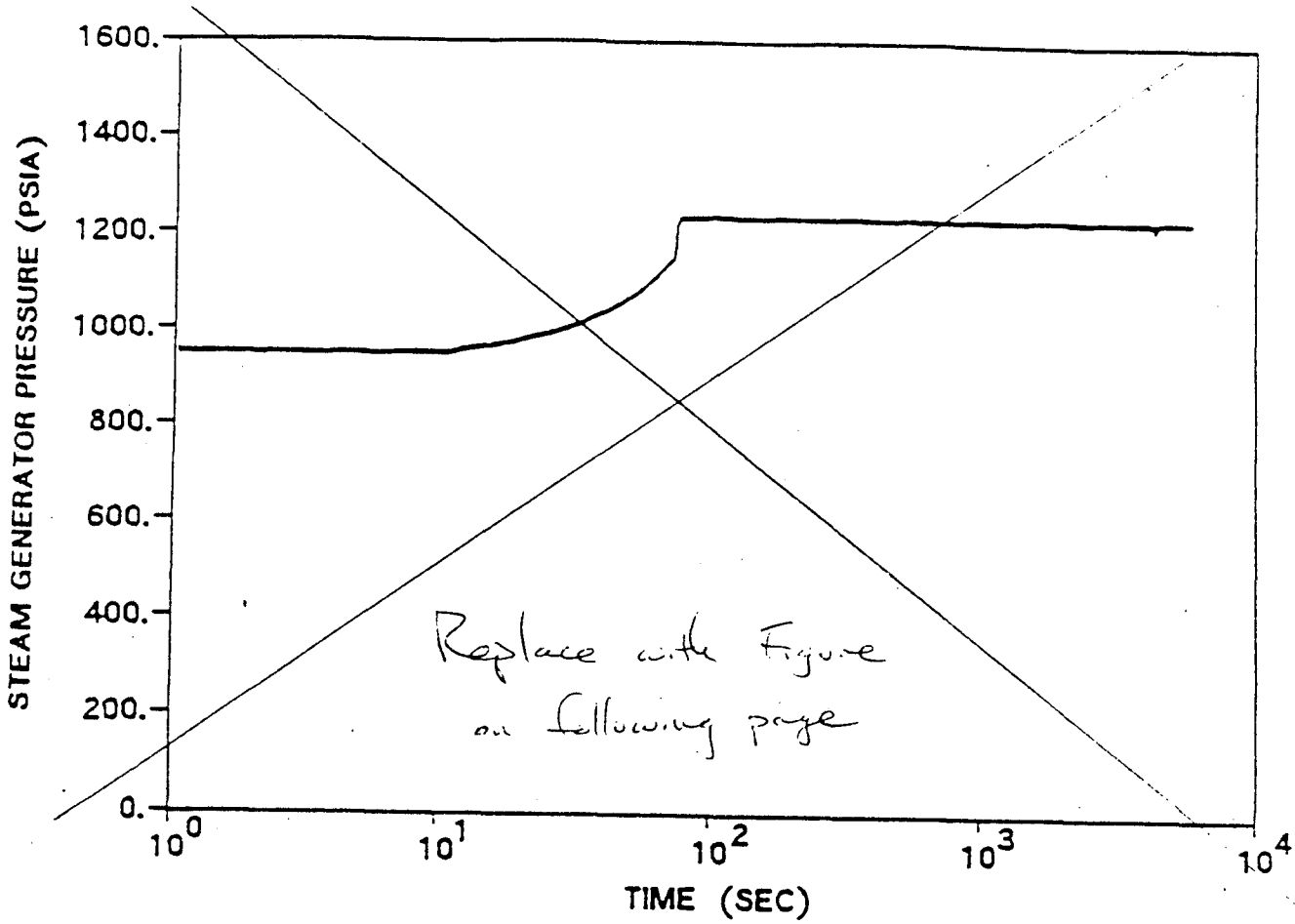
Pressurizer Water Volume
Transient for Loss of
Normal Feedwater
FIGURE 15.2-27G



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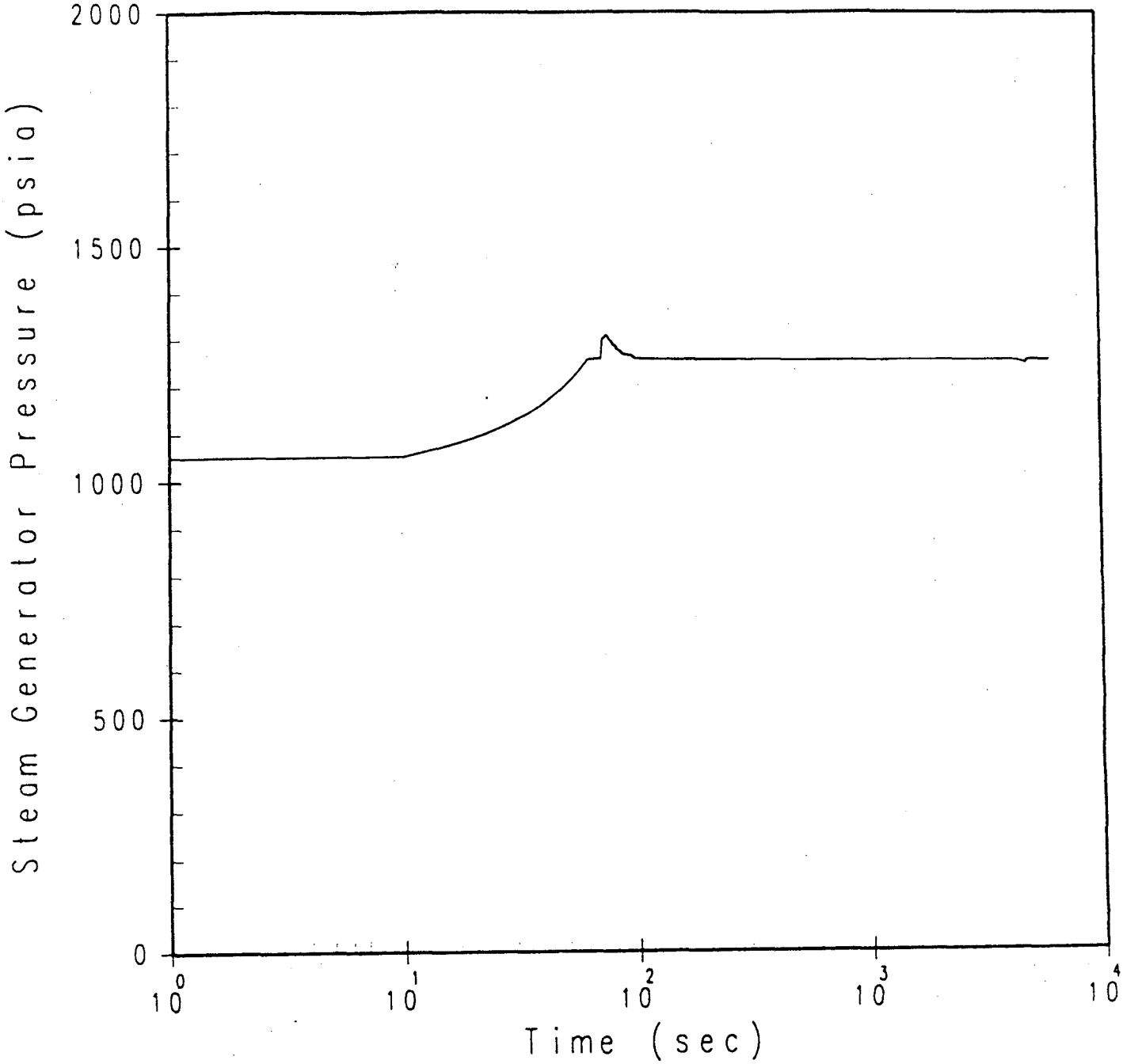
STEAM GENERATOR
PRESSURE TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27H



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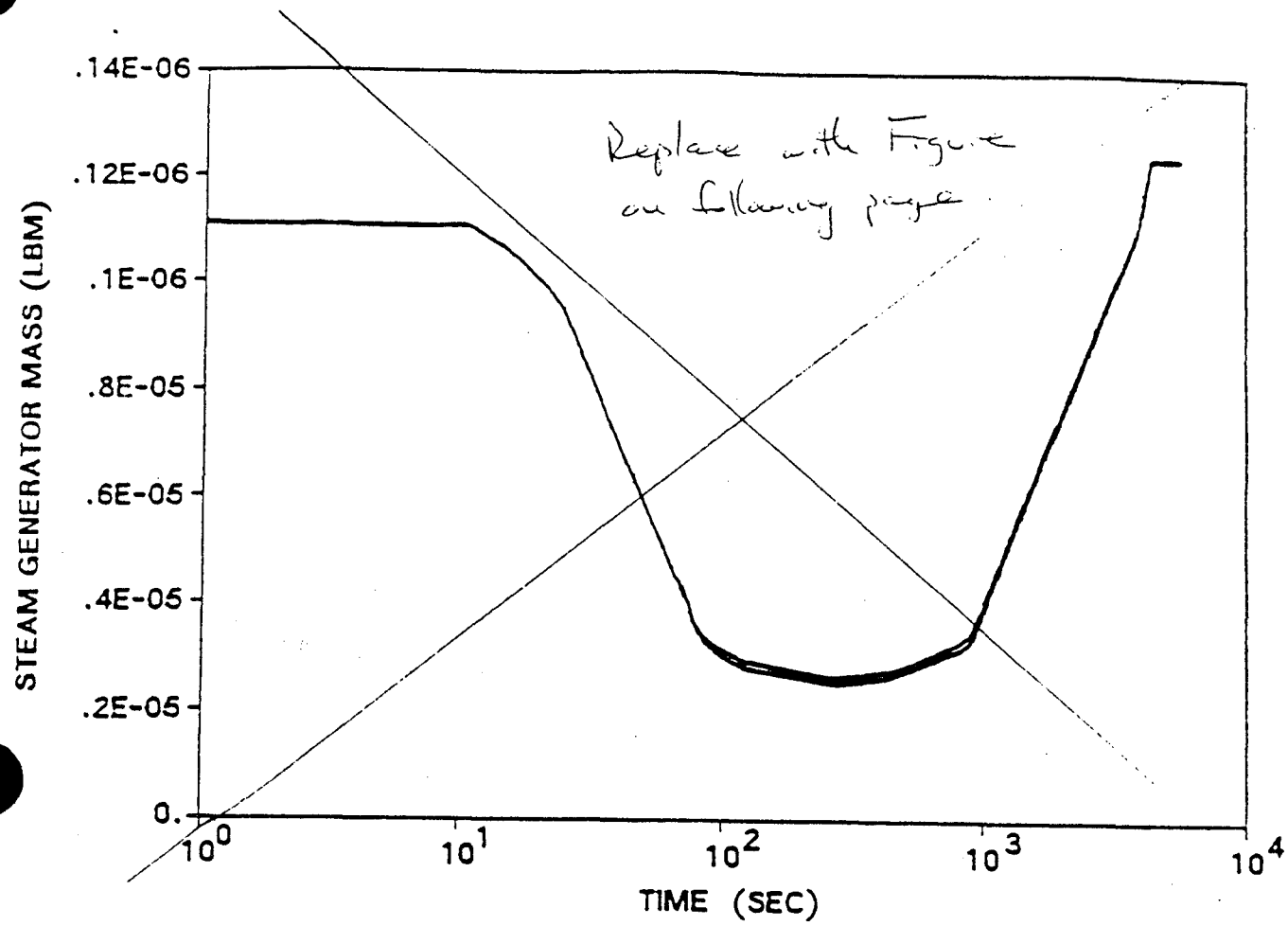
FINAL SAFETY ANALYSIS REPORT

Steam Generator Pressure
Transient for Loss of
Normal Feedwater
FIGURE 15.2-27H



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STEAM GENERATOR
MASS TRANSIENT
FOR LOSS OF
NORMAL FEEDWATER
FIGURE 15.2-27I

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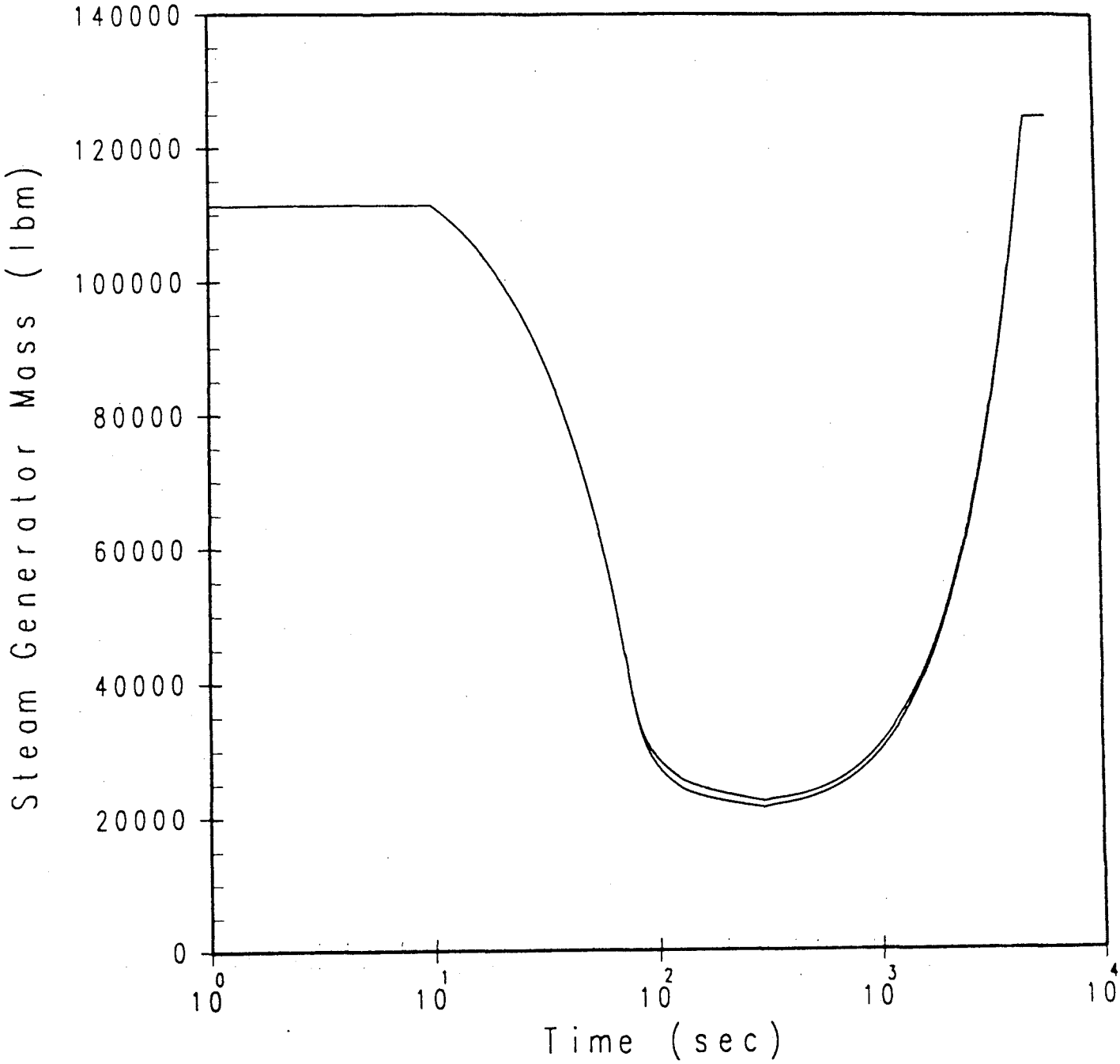

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Steam Generator Mass
Transient for Loss of
Normal Feedwater
FIGURE 15.2-27 I

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equation of mass, energy, and momentum applied throughout the system. A detailed description of NOTRUMP is given in References [1] and [2].

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

Cladding thermal analyses are performed with the LOCTA-IV[3] code which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the NOTRUMP hydraulic calculations as input.

A schematic representation of the computer code interfaces is given in Figure 15.3-1.

The small break analysis was performed with the approved Westinghouse ECCS Small Break Evaluation Model[1,2,3].

Safety injection flow rate to the RCS as a function of system pressure is an input parameter. The SIS is assumed to begin delivering full flow to the RCS 30 seconds after the generation of a safety injection signal.

Also, minimum safeguards ECCS capability and operability has been assumed in these analyses including conservative assumptions with regard to spillage of ECCS water from broken lines.

Hydraulic transient analyses are performed with the NOTRUMP code which determines the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core and mixture height history. The core thermal transient is performed with the LOCTA-IV[3] code. Both calculations assume the core is operating at 102% of licensed power.

15.3.1.3 <u>Reactor Coolant System Pipe Break Results</u>

A spectrum of break sizes was analyzed ~~Analyses were conducted~~ to determine the limiting break size in terms of the highest peak cladding temperature. ~~This section contains results for a downflow barrel/baffle reactor vessel internals design (initial configuration) and for an upflow barrel/baffle design (the present configuration) in order to capture the spectrum of break sizes analyzed as well as address reduced auxiliary feedwater and safety injection flows for the present configuration.~~

For all cases reported, during the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained.

These break sizes were 3, 4, and 6 inches.

The resultant heat transfer cools the fuel rod cladding to very near the coolant temperatures as long as the core remains covered by a two-phase mixture. When the mixture level drops below the top of the core, the steam flow computed with NOTRUMP provides cooling to the upper portion of the core.

The typical core power (dimensionless) transient following the accident (relative) to reactor scram time is shown in Figure 15.3-9. Also shown is the typical hot rod axial power shape in Figure 15.3-10.

The reactor scram delay time is equal to the reactor trip signal time plus ~~[illegible] for~~ control rod insertion time, or a total of 4.7 seconds. During this ~~[illegible]~~ delay period, the reactor is conservatively assumed to continue to operate at the initial rated power level.

~~For the analysis without reduced auxiliary feedwater and safety injection flow,~~ The safety injection flow is depicted in Figure 15.3-2 as a function of RCS pressure. ~~This figure represents injection flow based on pump performance curves degraded 5% from the design head.~~ Auxiliary feedwater flow is ~~[illegible]~~ 1050 gpm based on the ~~design capacity~~ operation of one motor-driven and ~~one~~ the turbine-driven auxiliary feedwater pump, each delivering to two steam generators.

The 30 second delay time includes the time for diesel generator startup, loading on the 6.9 kV shutdown board, and sequential loading of the centrifugal charging and safety injection pumps onto the emergency buses, with acceleration to full speed and capability for injection. Although included in the 30 second delay, the effect of the residual heat removal pump flow is not a factor in this analysis since their shutoff head is lower than RCS pressure during the time period for this transient.

Insert A

~~A spectrum of break sizes was analyzed for a downflow barrel/baffle region based on the initial reactor vessel internals design. These break sizes were 9.0, 4.0, and 6.0 inches. Figures 15.3-11 and 15.3-12 present the RCS pressure transients. Figures 15.3-13a and 15.3-13b present the core mixture height transients for the 3.0 and 6.0 inch breaks, respectively. The peak cladding temperatures for these cases are less than the peak cladding temperature of the 4.0 inch break, which was, therefore, determined to be the limiting break. The peak cladding temperatures for the 3.0 and 6.0 inch breaks are presented in Figures 15.3-14a and 15.3-14b.~~ The depressurization transient for the 4.0 inch break is shown in Figure 15.3-3. The extent to which the core is uncovered is shown in Figure 15.3-4. The peak cladding temperature transient is shown in Figure 15.3-5. The steam flow rate for this break is shown in Figure 15.3-6. The heat transfer coefficients for the rod for this phase of the transient are given in Figure 15.3-7, and the hot spot fluid temperature is shown in Figure 15.3-8~~a~~.

~~An additional break case of 4.0 inches was analyzed for an upflow barrel/baffle region, which is WBN's present configuration. Results for this break can be found in Table 15.3-1, Table 15.3-2, and Figures 15.3-8b through 15.3-8g.~~

~~For the analysis with reduced auxiliary feedwater and safety injection flows, only the upflow barrel/baffle reactor vessel internals configuration needed to~~

Insert B

Insert A

The 4 inch break was determined to be the limiting break size, with a peak cladding temperature of 1452°F. The transient results for the limiting 4 inch break are presented in Figures 15.3-3 to 15.3-8.

Insert B

The comparable transient results for the 3 inch break are presented in Figures 15.3-11 to 15.3-~~16~~ 11e and for the 6 inch break in Figures 15.3-~~17~~ 12 to 15.3-~~22~~ 12e.

~~be considered. Two break sizes were analyzed, the 4.0 inch which had been established in the previous analysis, and a 2.0 inch which was determined to be limiting in peak cladding temperature.~~

~~For this analysis, the safety injection flow is depicted in Figure 15.3-8h as a function of RCS pressure. A reduced auxiliary feedwater flow of 1050 gpm is utilized, supplied by one motor driven and one turbine driven auxiliary feedwater pump, each delivering to two steam generators.~~

~~Therefore, for the upflow barrel/baffle configuration with the assumed reduction in auxiliary feedwater and safety injection flows, the worst break size of 2 inches resulted in a maximum peak hot spot cladding temperature of 2009°F. Results for this break can be found in Table 15.3-1 and Table 15.3-2. Plots of the RCS upper plenum pressure (Figure 15.3-8b), mixture level (Figure 15.3-8i), peak cladding temperature (Figure 15.3-8j), steam flow (Figure 15.3-8k), heat transfer coefficients (Figure 15.3-8l) accumulator, and hot spot fluid temperatures (Figure 15.3-8m) are provided for this case.~~

It should be noted that all small break sizes presented here result in calculated peak cladding temperatures less than those calculated for large breaks (Section 15.4.1).

15.3.1.4 Conclusions - Thermal Analysis

For cases considered, the emergency core cooling system meets the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element cladding temperature provides margin to the limit of 2200°F, based on an F_q value of 2.40.

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

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

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

The time sequence of events is shown in Table 15.3-1. Table 15.3-2 summarizes the results of these analyses.

15.3.2 Minor Secondary System Pipe Breaks

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6 inch diameter break or smaller.

TABLE 15.3-1

SMALL BREAK LOCA ANALYSIS

TIME SEQUENCE OF EVENTS

Event	Time (sec)				
Break Size:	3 inch[3]	4 inch[3]	6 inch[3]	4 inch[1]	3-inch[2]
Break initiation	0.0	0.0	0.0	0.0	0.0
Reactor trip signal	8.36	4.72	2.39	4.735	31.5
Safety injection signal	17.97	13.19	9.19	12.915	41.0
Top of core uncovery	1322	665	271	643.0	894
Accumulator injection begins	2836	890	375	847.0	1643
Peak cladding temperature occurs	2143	1050	455	935.5	1701
Top of core recovered	2770	1575	477	1440.0	N/A

(1) Upflow barrel/baffle region assumed.
(2) Upflow barrel/baffle region assumed with reduced auxiliary feedwater and safety injection flows.
(3) Downflow barrel/baffle region assumed.

TABLE 15.3-1

SMALL BREAK LOCA ANALYSIS

TIME SEQUENCE OF EVENTS

Event	Time (seconds)		
Break Size:	3 Inch	4 Inch	6 Inch
Break Initiation	0.0	0.0	0.0
Reactor trip signal	20.7	11.9	6.4
Safety injection signal	30.2	20.5	13.1
Top of core uncovery	985	535	289
Accumulator injection begins	2197	816	375
Peak cladding temperature occurs	1733	956	444
Top of core recovered	2618	1734	477

TABLE 15.3-2

SMALL BREAK ANALYSIS SUMMARY OF RESULTS

Parameter	Value				
Break Size:	3 inch[3]	4 inch[3]	6 inch[3]	4 inch[1]	3-inch[2]
Peak cladding temperature (°F)	1179	1549	1146	1446.1	2089
Elevation (ft)	12.0	12.0	12.0	12.0	12.0
Max. local Zr/H_2O reaction (%)	0.12	0.74	0.04	0.32	15.5
Elevation (ft)	12.0	12.0	12.0	12.0	12.0
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	<0.3	<1.0
Hot rod burst time (sec)	N/A	N/A	N/A	N/A	N/A

Boundary Condition Assumptions

NSSS power	102% of 3411 MW
Core power (rod heatup analysis)	102% of 3411 MW
Peak linear power	12.84 kW/ft[4]
Cold leg accumulators:	
Water volume (each)	1050 ft^3
Pressure	600 psia

(1) Upflow barrel/baffle region assumed.
(2) Upflow barrel/baffle region assumed with reduced auxiliary feedwater and safety injection flows.
(3) Downflow barrel/baffle region assumed.
(4) Peak linear power for this analysis is determined from average linear power of 5.553 kW/ft and a limiting F_Q = 2.311. The value F_Q = 2.311 was established from the intersection of the normalized F_Q vs. core height (see Figure 4.3-21), where 1.0 is the peak F_Q of 2.4, and power shape [K(z)] vs. core height.

TABLE 15.3-2

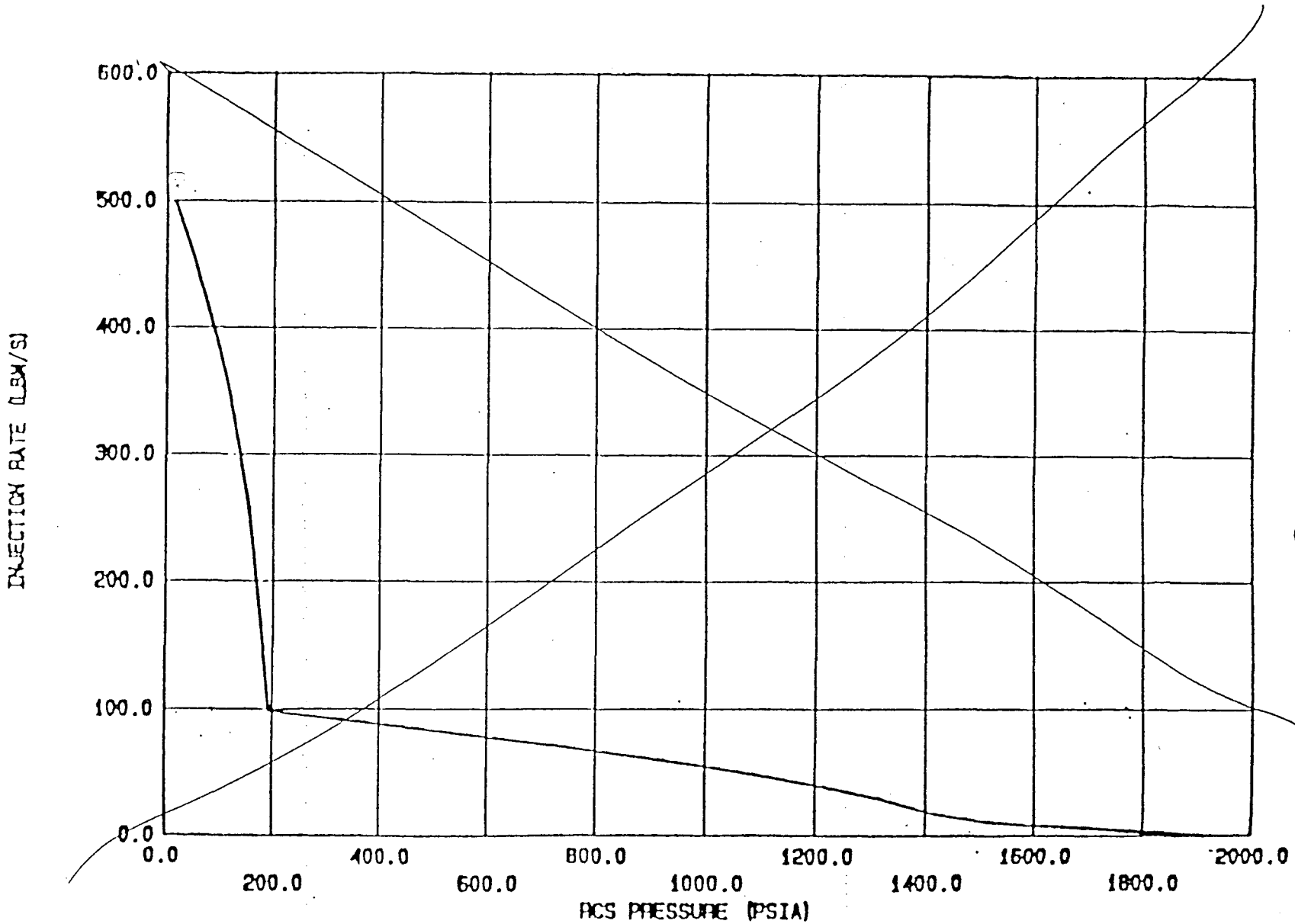
SMALL BREAK LOCA ANALYSIS SUMMARY OF RESULTS

Parameter	Value		
Break Size:	3 Inch	4 Inch	6 Inch
Peak cladding temperature (°F)	1303	1452	911
Elevation (ft)	11.50	11.50	10.50
Max. local Zr-H_2O reaction (%)	0.224	0.370	0.002
Elevation (ft)	11.50	11.25	10.75
Total Zr-H_2O reaction (%)	<1	<1	<1
Hot rod burst time (sec)	N/A	N/A	N/A

Boundary Condition Assumptions	
NSS power	102% of 3411 MW
Core power (rod heatup analysis)	102% of 3411 MW
Peak linear power	12.77 kW/ft[1]
Cold leg accumulators:	
Water volume (each)	1050 ft^3
Pressure	600 psia

[1] The hot rod linear power shape used for this analysis is shown in Figure 15.3-10. The peak linear power of 12.77 kW/ft corresponds to F_Q = 2.30 at that elevation. The value of F_Q = 2.30 at the peak linear power elevation was determined from the normalized F_Q as a function of core height (Figure 4.3-21), where normalized F_Q = 1.0 corresponds to the peak F_Q = 2.40.

FIGURE 15.3-2 SAFETY INJECTION VS. RCS PRESSURE



See new 15.3-2

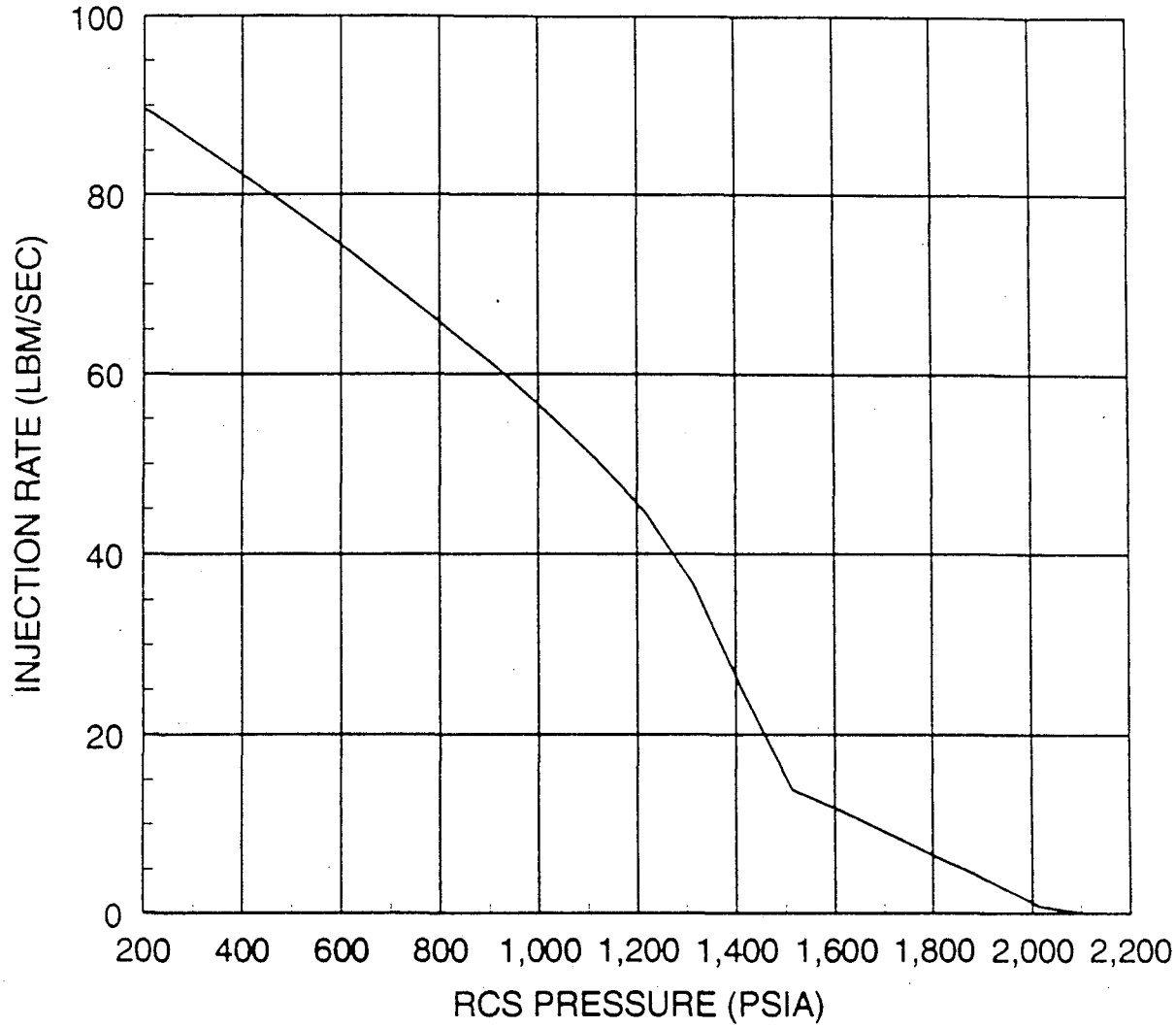


Figure 15.3-2 PUMPED SAFETY INJECTION FLOW RATE VS. RCS PRESSURE

See new 15.3-3

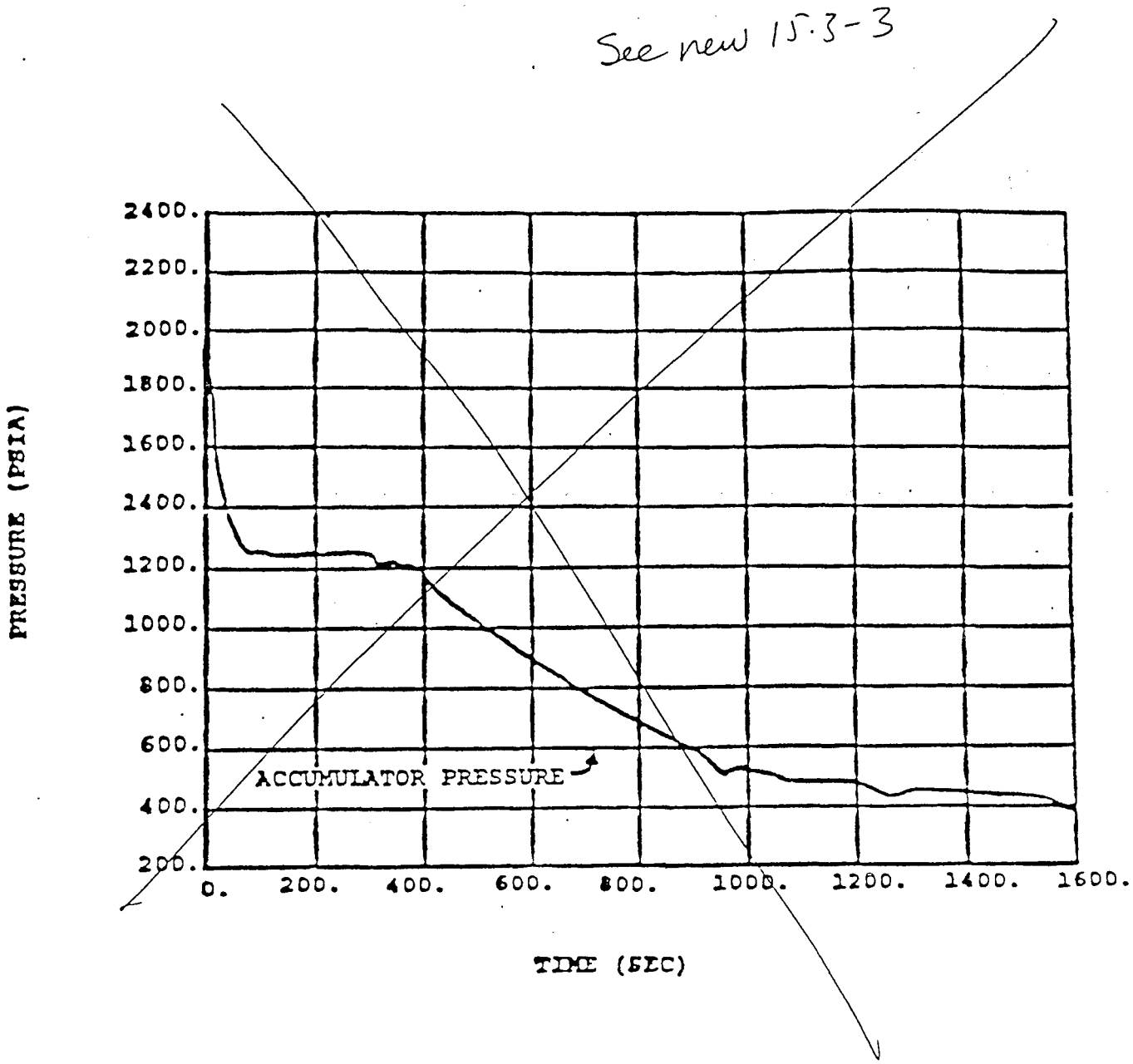


FIGURE 15.3-3 REACTOR COOLANT SYSTEM PRESSURE FOR THE 4-INCH BREAK

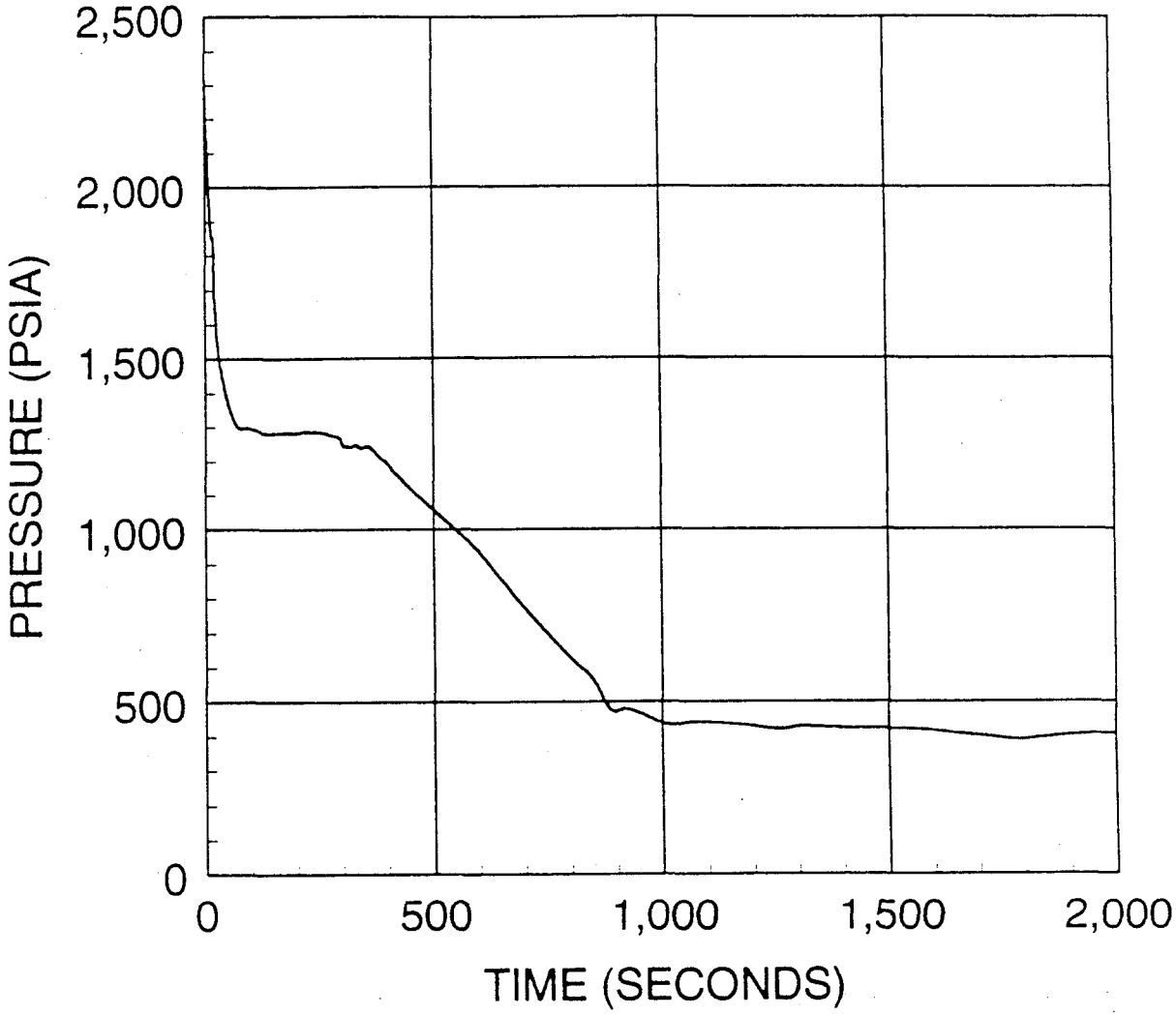


Figure 15.3-3 REACTOR COOLANT SYSTEM PRESSURE FOR 4 INCH BREAK

See New 15.3-4

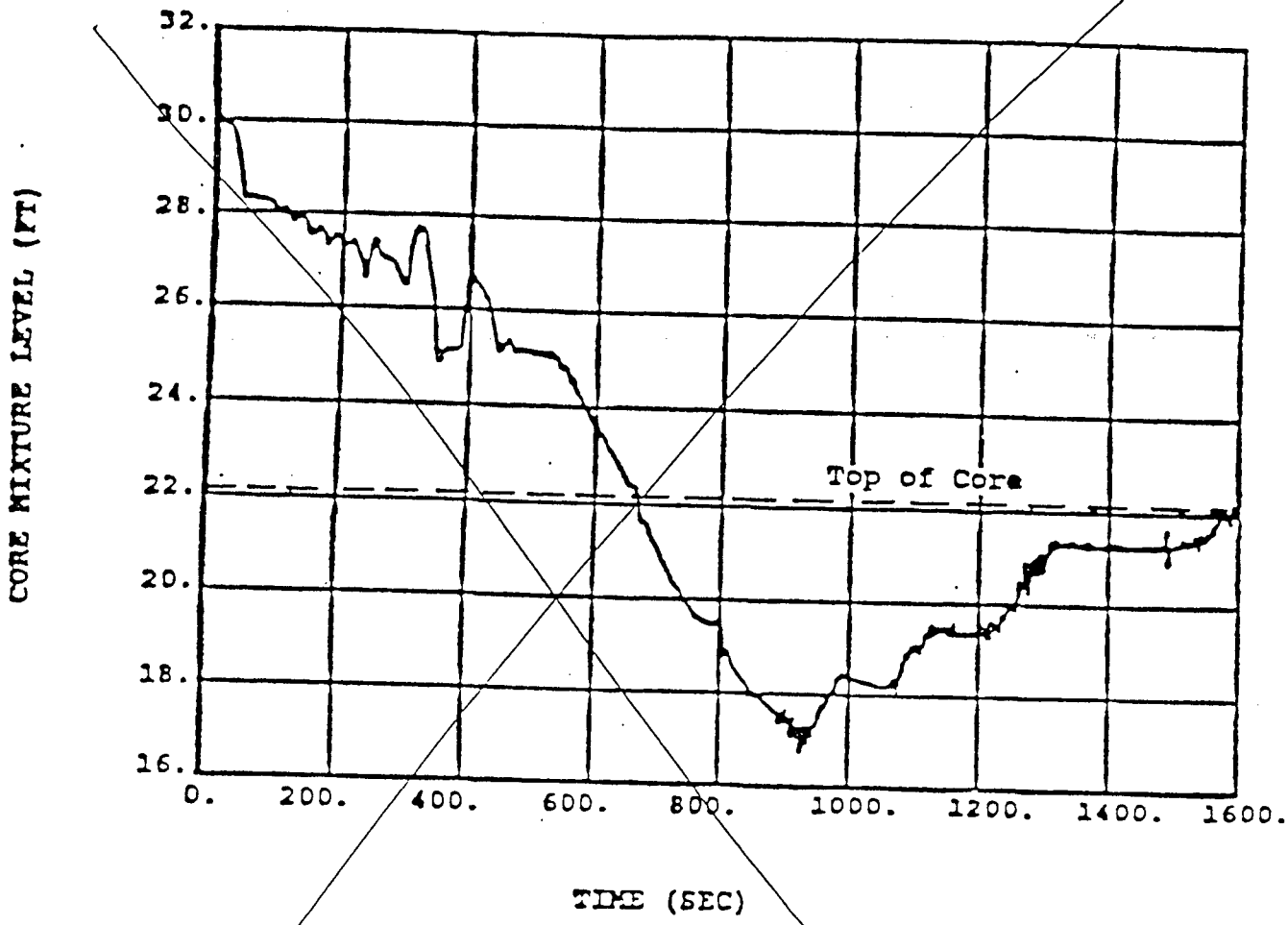


FIGURE 15.3-4 CORE MIXTURE HEIGHT TRANSIENT FOR THE 4-INCH BREAK

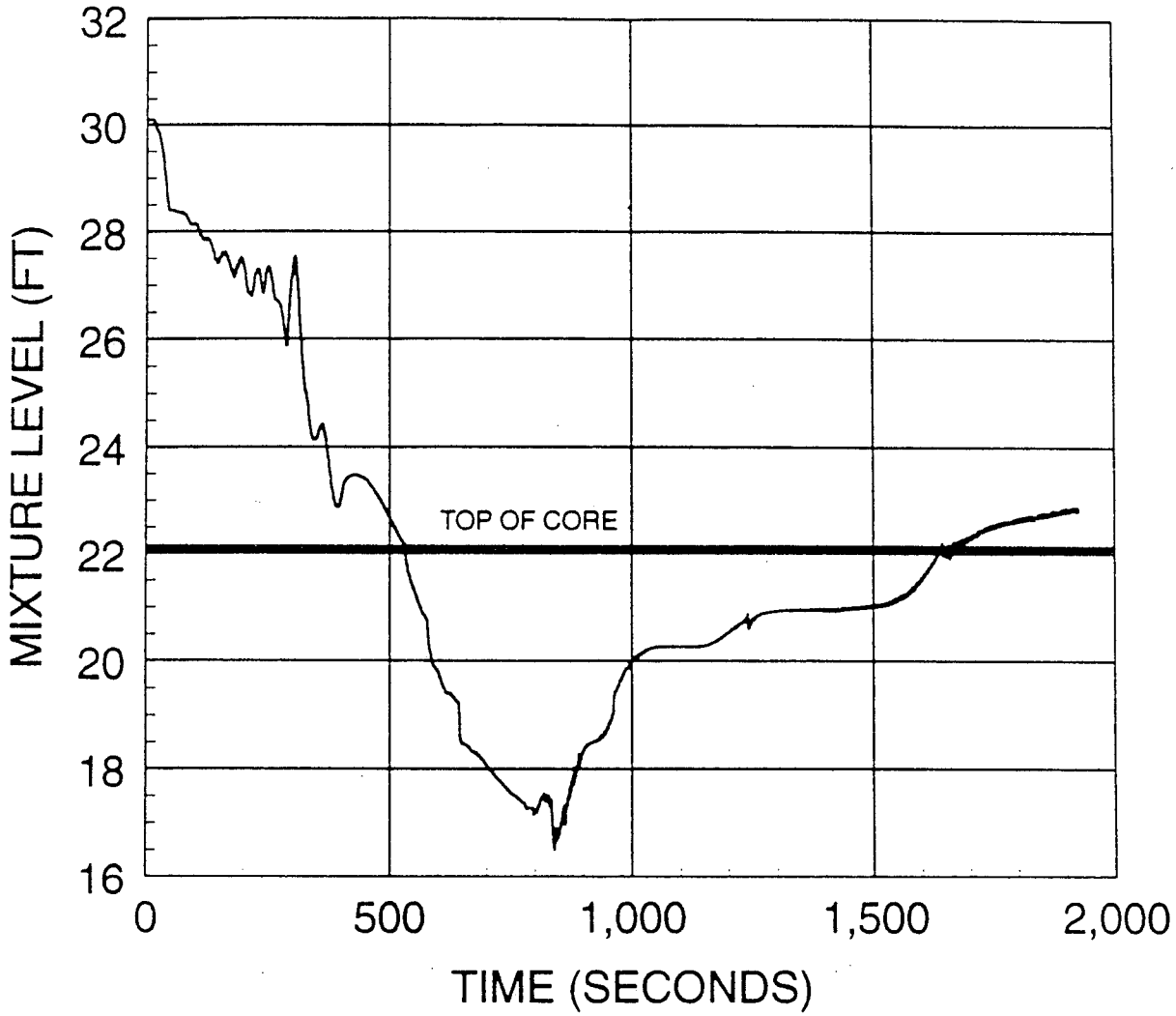


Figure 15.3-4 CORE MIXTURE LEVEL TRANSIENT FOR 4 INCH BREAK

See New 15.3-5

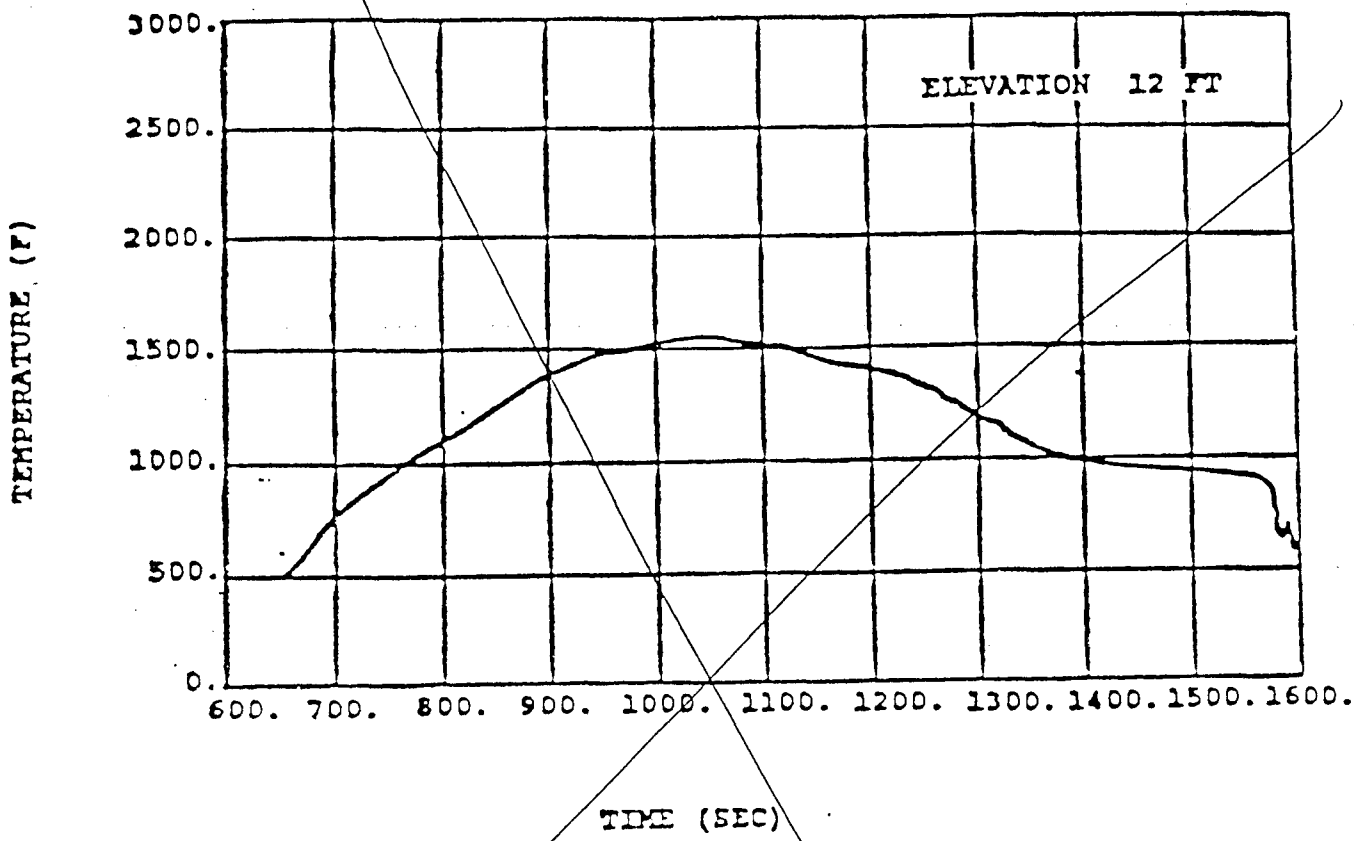


FIGURE 15.3-5 CLAD TEMPERATURE TRANSIENT FOR THE 4-INCH BREAK

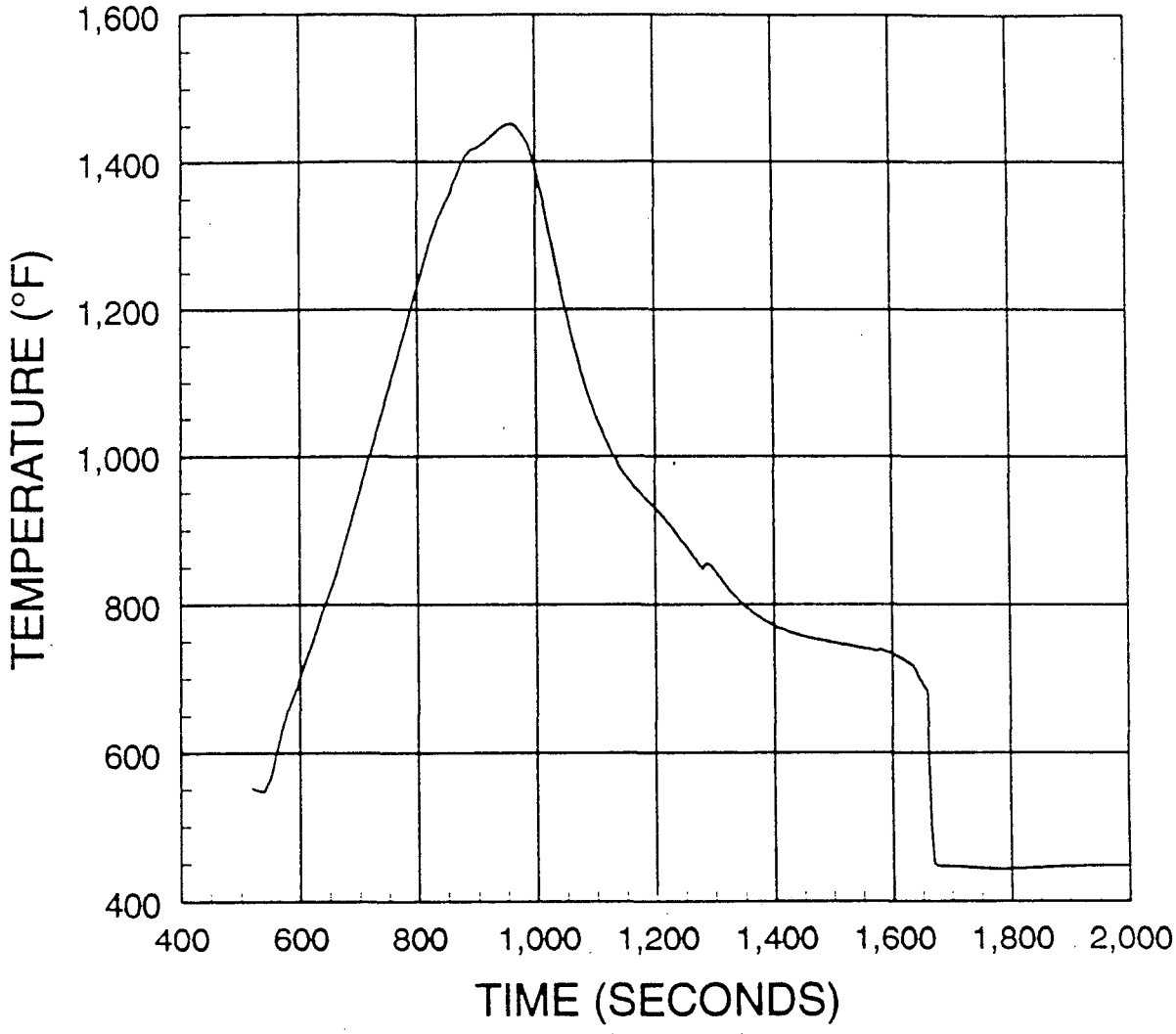


Figure 15.3-5 CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION FOR 4 INCH BREAK

See New 15.3-6

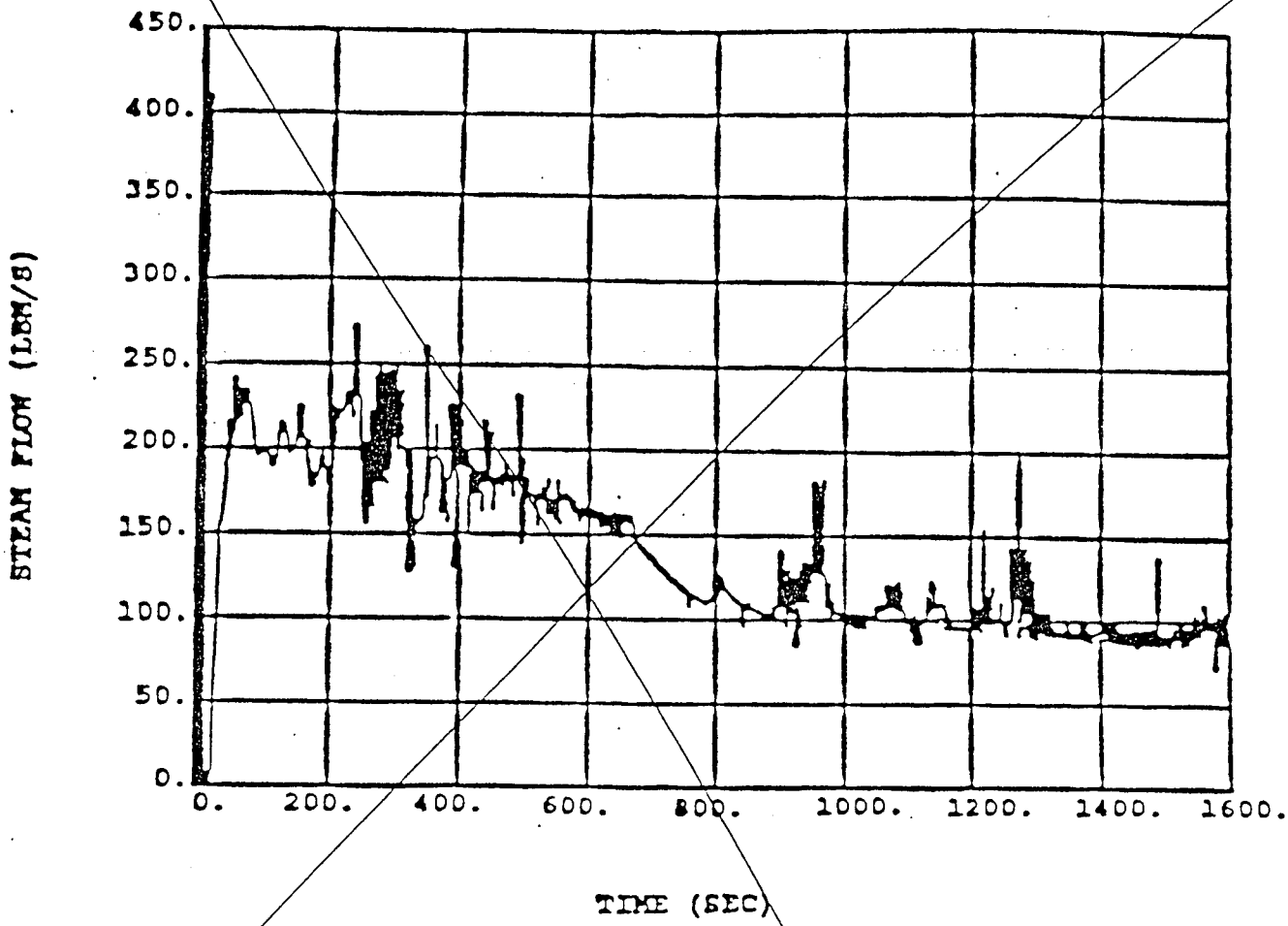


FIGURE 15.3-6 CORE STEAM FLOW RATE FOR THE 4-INCH BREAK

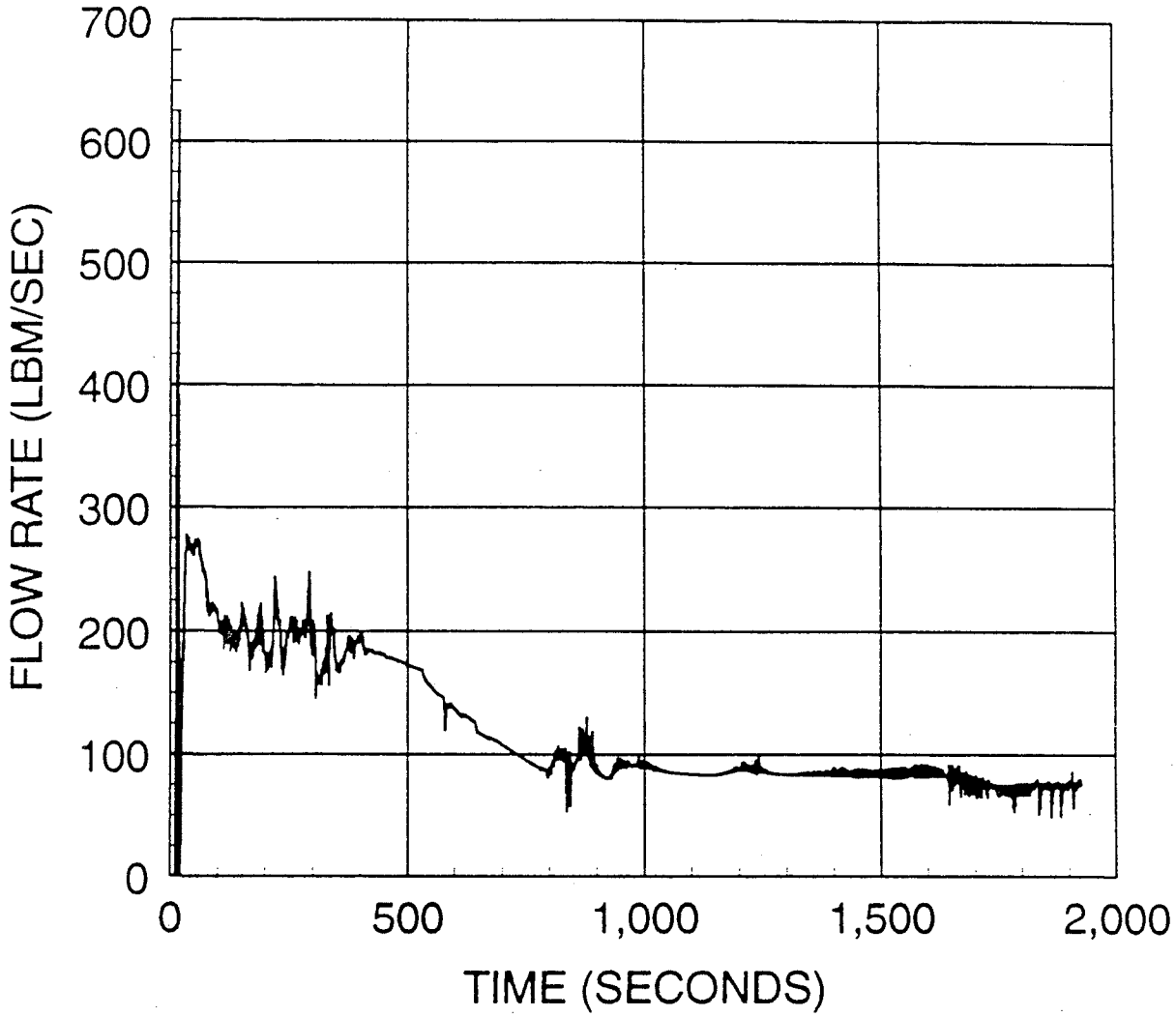


Figure 15.3-6 CORE OUTLET STEAM FLOW RATE FOR 4 INCH BREAK

See New 15.3-7

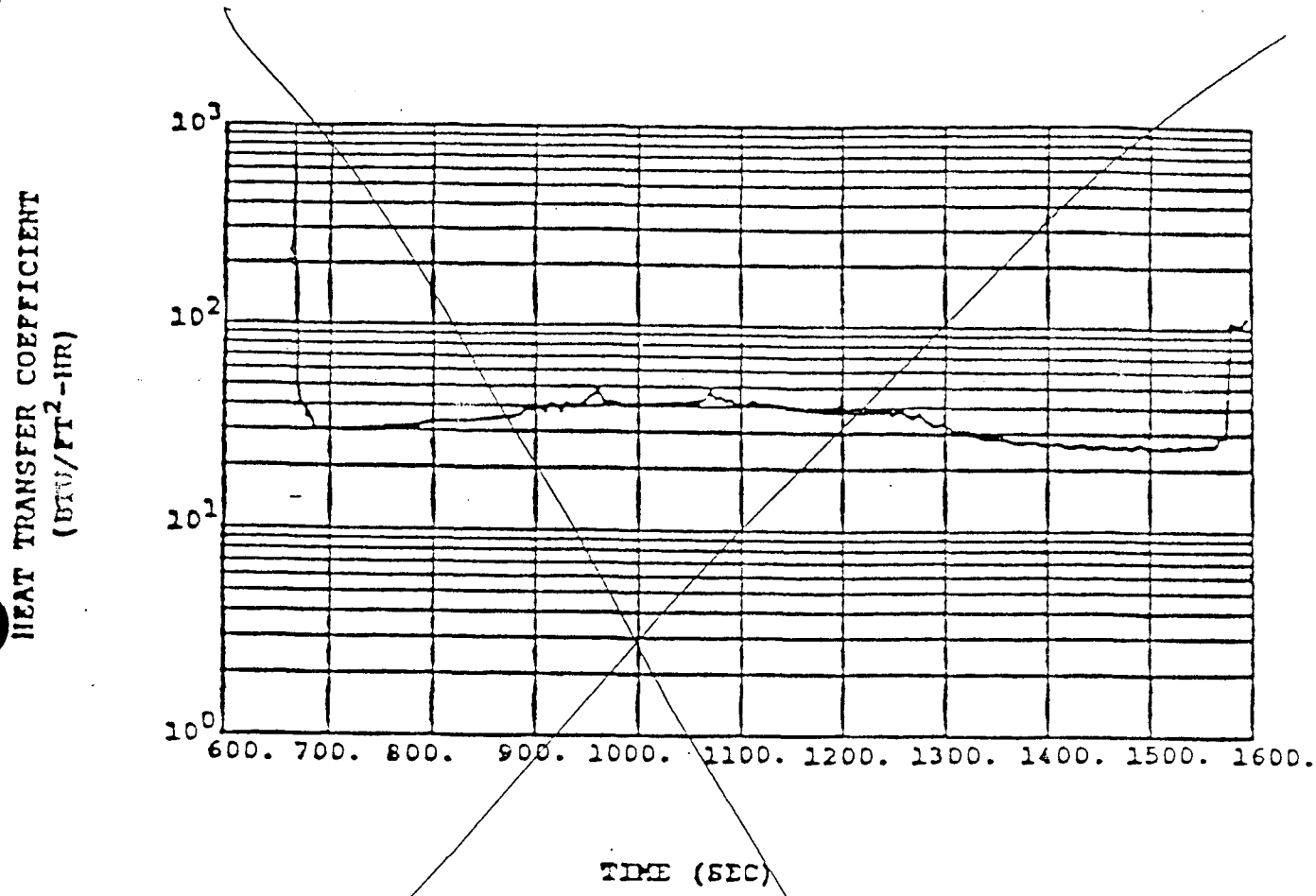


FIGURE 15.3-7 ROD HEAT TRANSFER COEFFICIENT AT THE PEAK TEMPERATURE ELEVATION FOR THE 4-INCH BREAK

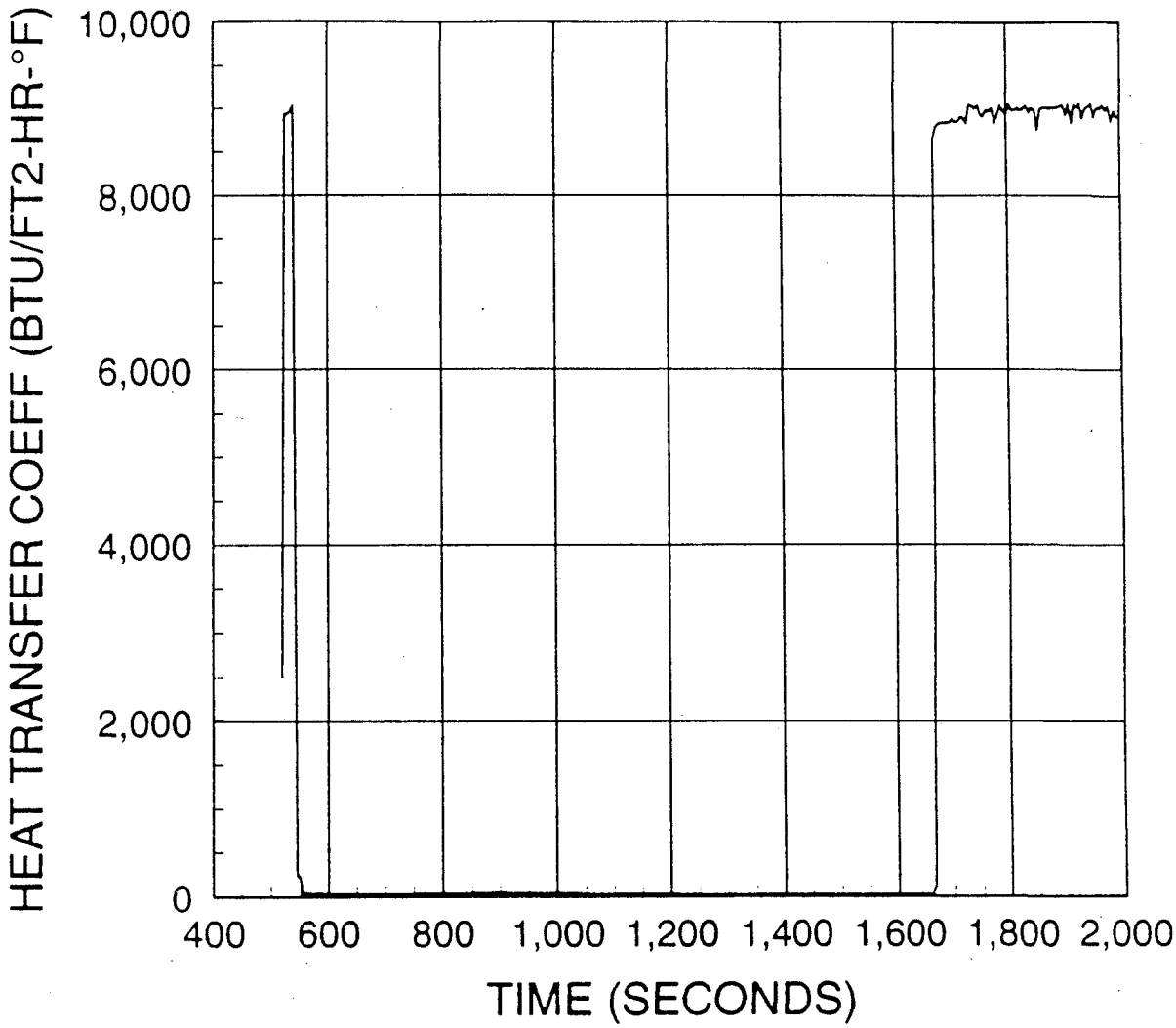


Figure 15.3-7 CLAD SURFACE HEAT TRANSFER COEFFICIENT AT PEAK TEMPERATURE ELEVATION FOR 4 INCH BREAK

See New 15.3-8
(replaces former 15.3-8a)

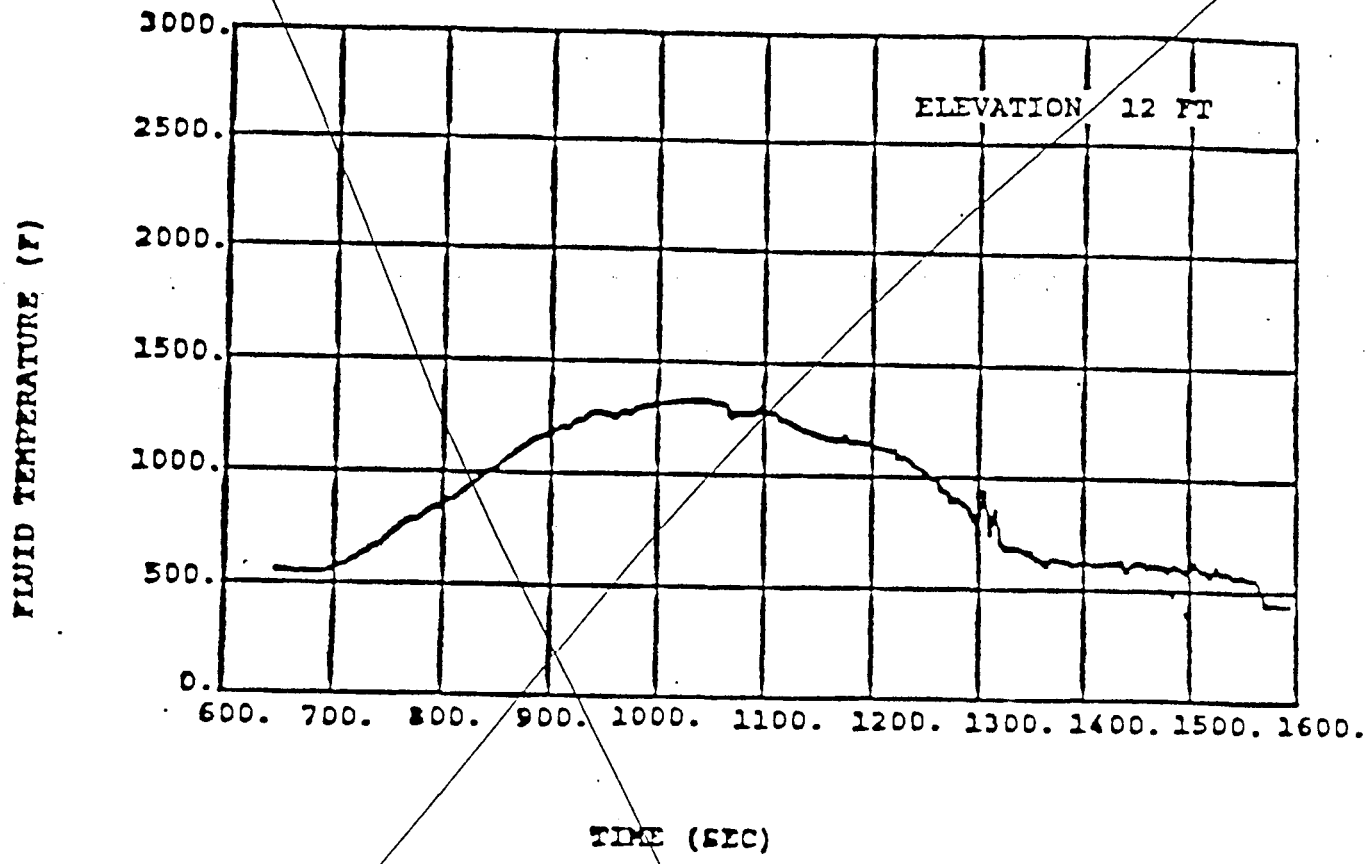


FIGURE 15.3-8a HOT SPOT FLUID TEMPERATURE FOR THE 4-INCH BREAK

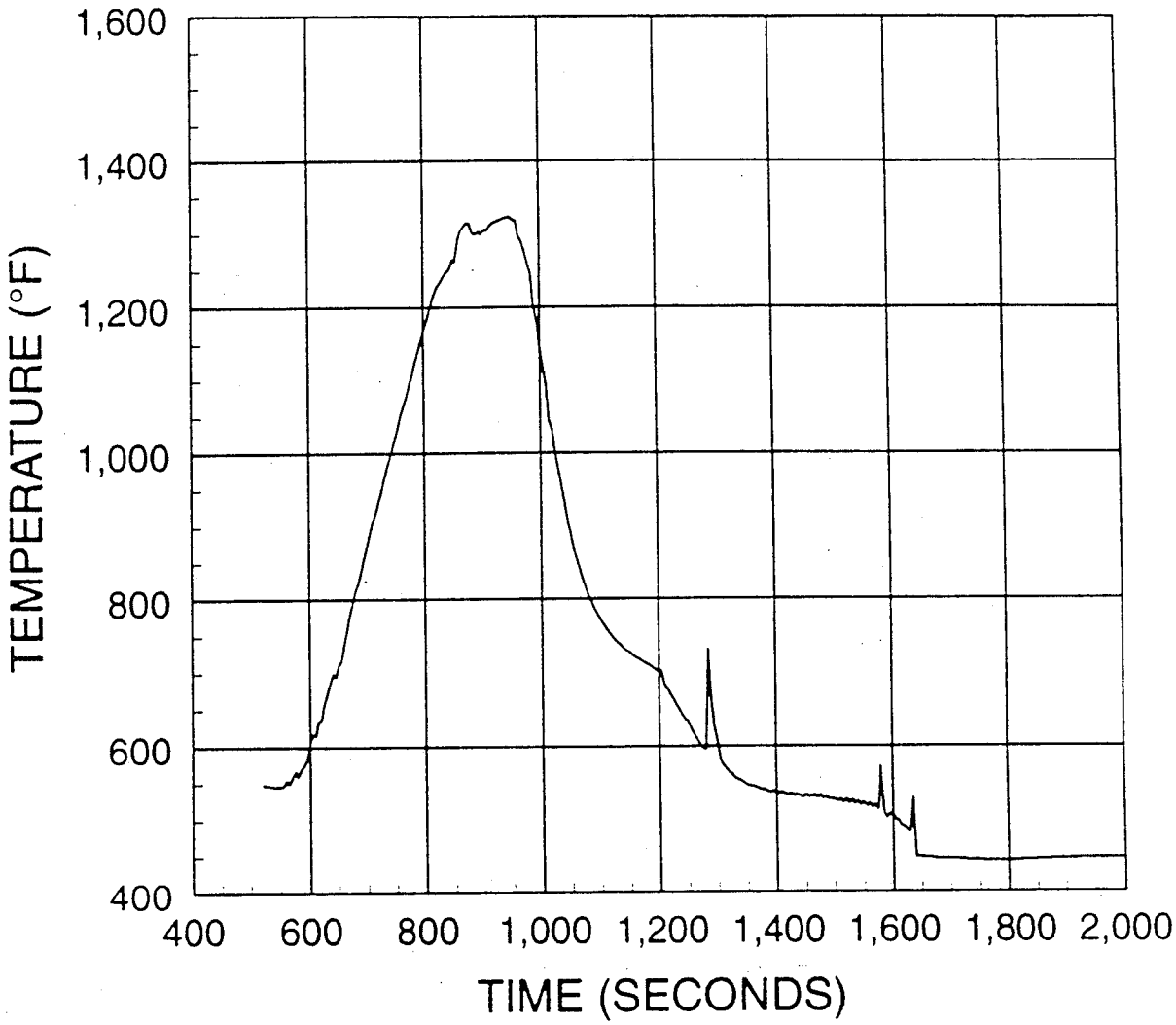


Figure 15.3-8 FLUID TEMPERATURE AT PEAK CLAD TEMPERATURE ELEVATION FOR 4 INCH BREAK

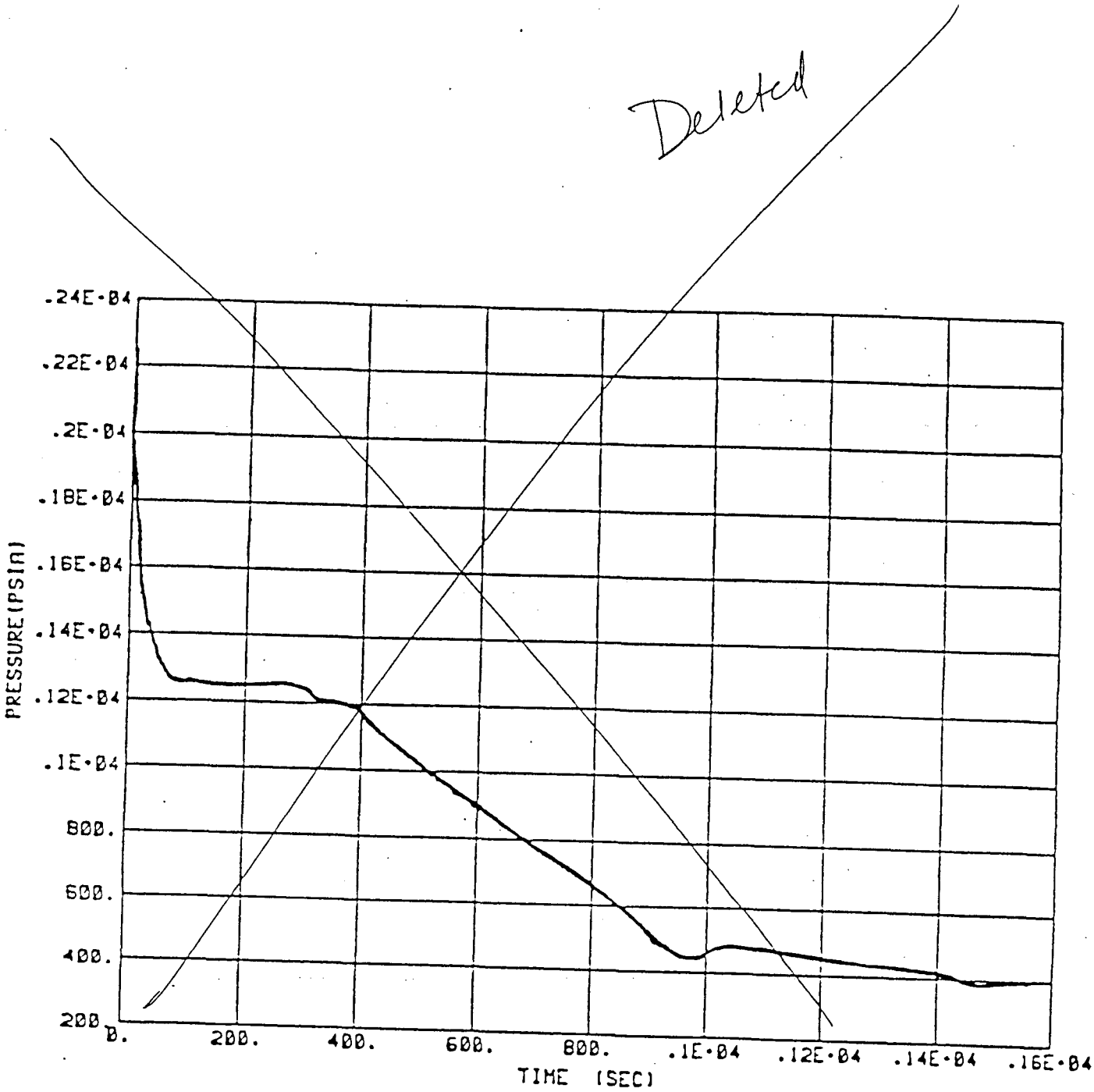


FIGURE 15.3-8b REACTOR COOLANT SYSTEM PRESSURE, 4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION

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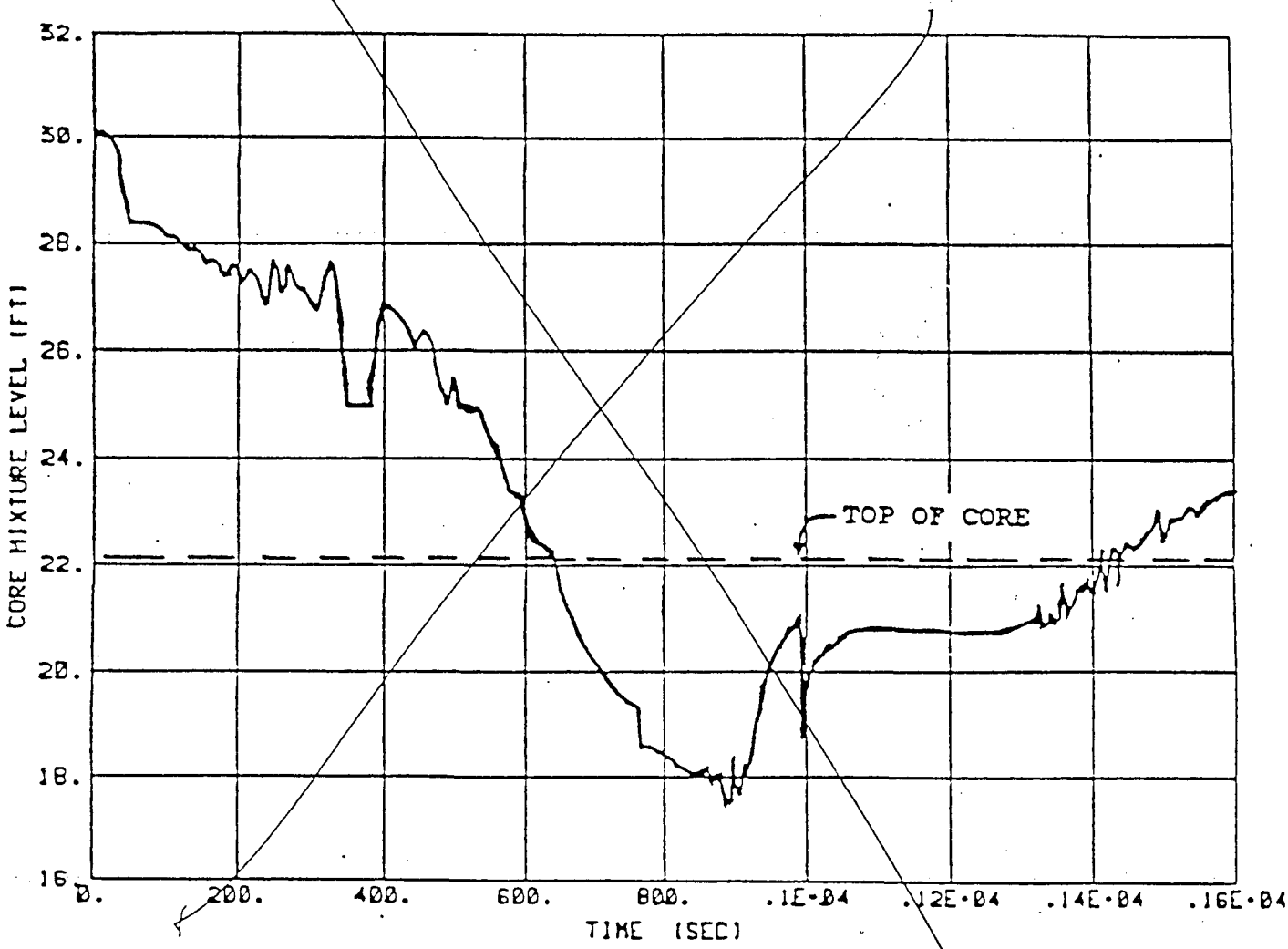

FIGURE 15.3-8c CORE MIXTURE HEIGHT TRANSIENT, 4-INCH BREAK, UPFLOW BARREL BAFFLE REGION

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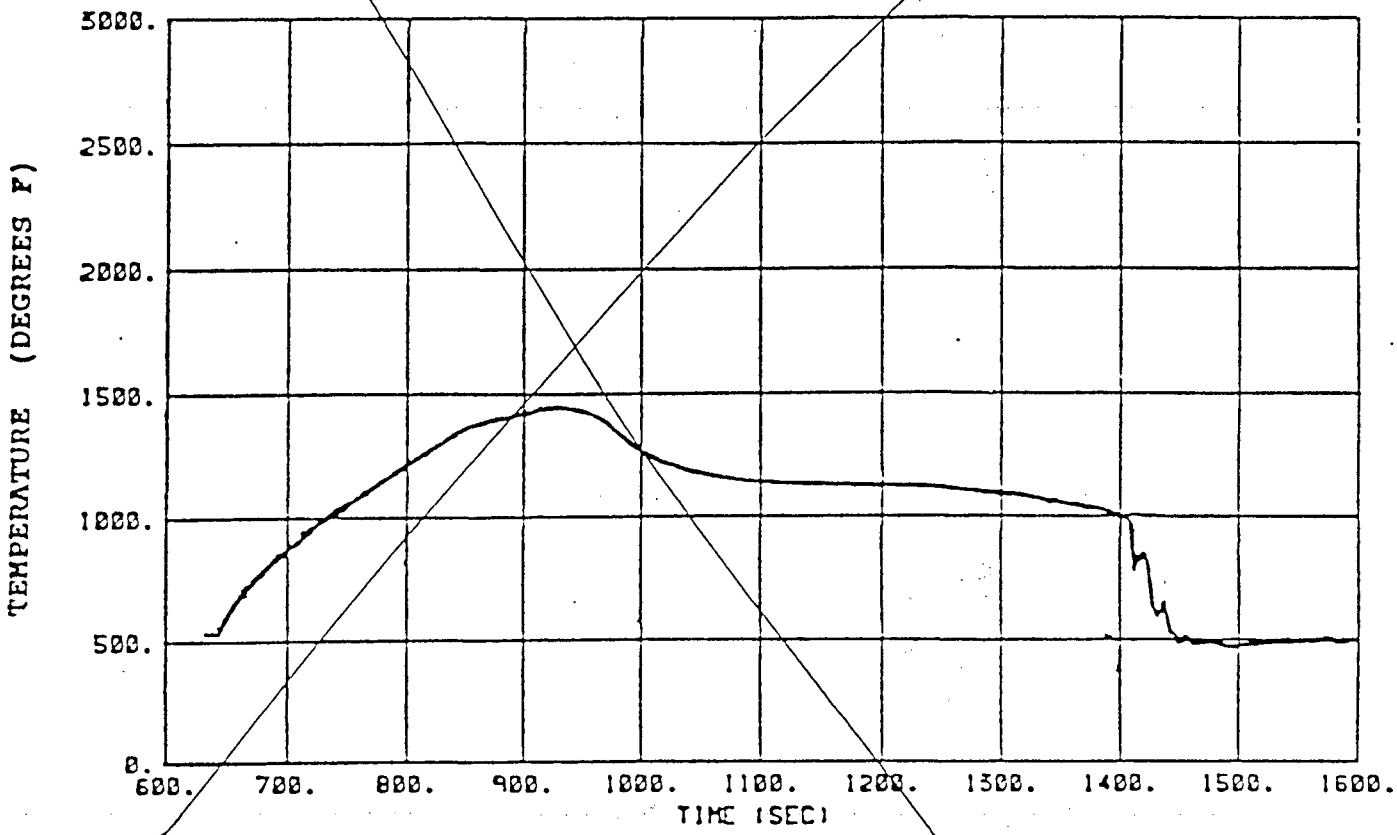

FIGURE 15.3-8d CLAD TEMPERATURE TRANSIENT,
4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION

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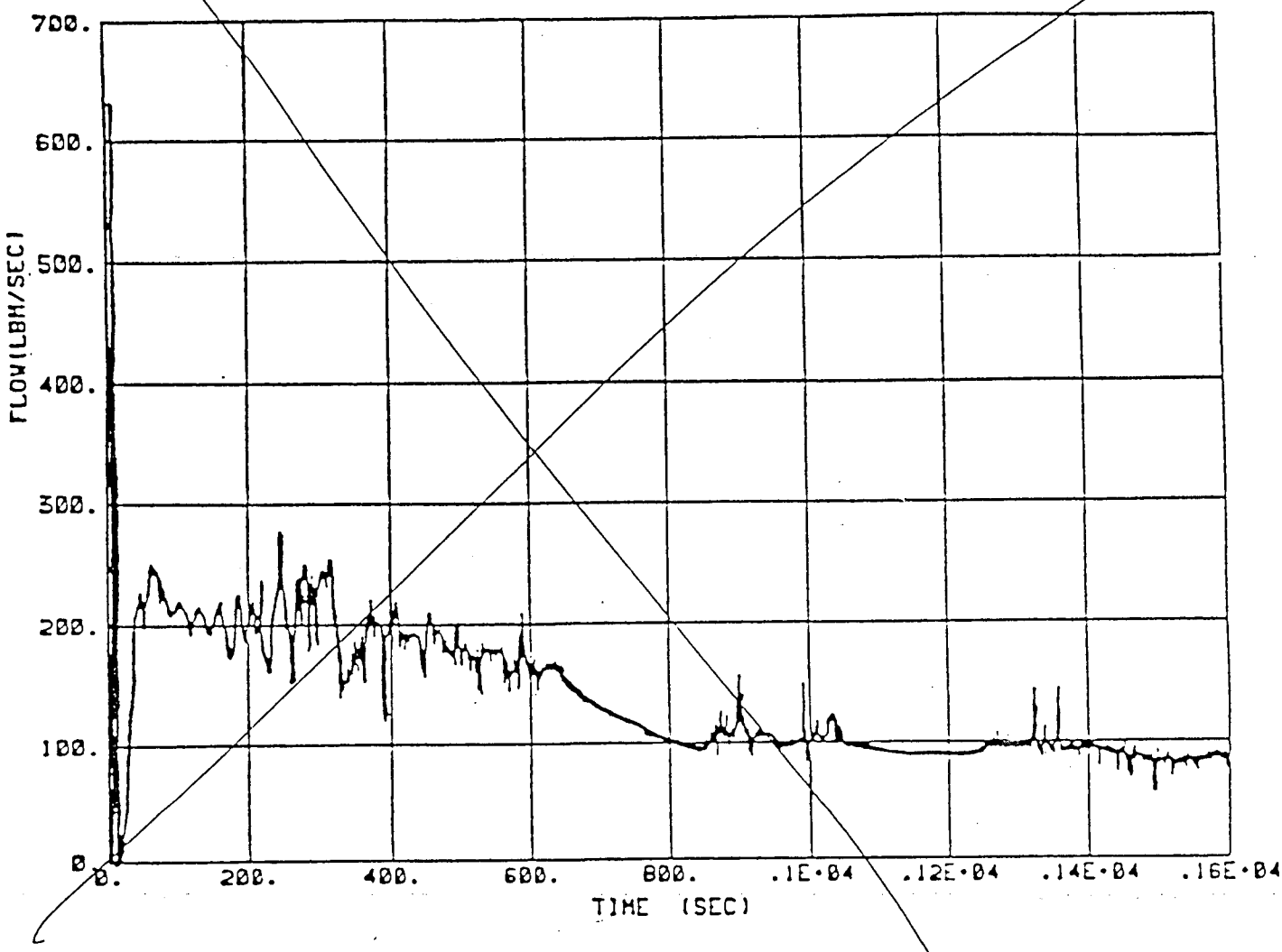

FIGURE 15.3-8e CORE STEAM FLOWRATE,
4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION

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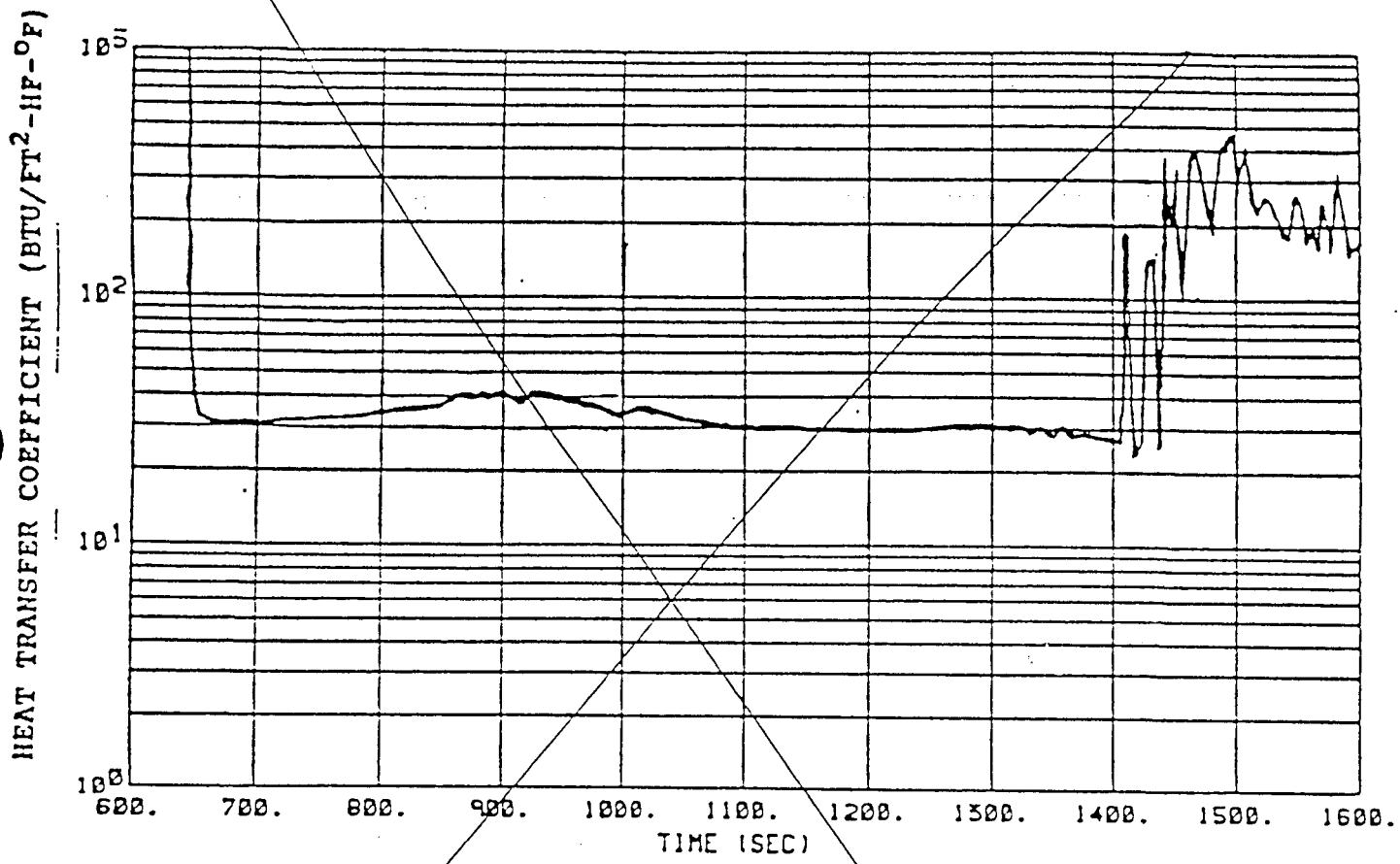

FIGURE 15.3-81 FUEL ROD HEAT TRANSFER COEFFICIENT,
4-INCH BREAK, UPFLOW BARREL/BAFFLE

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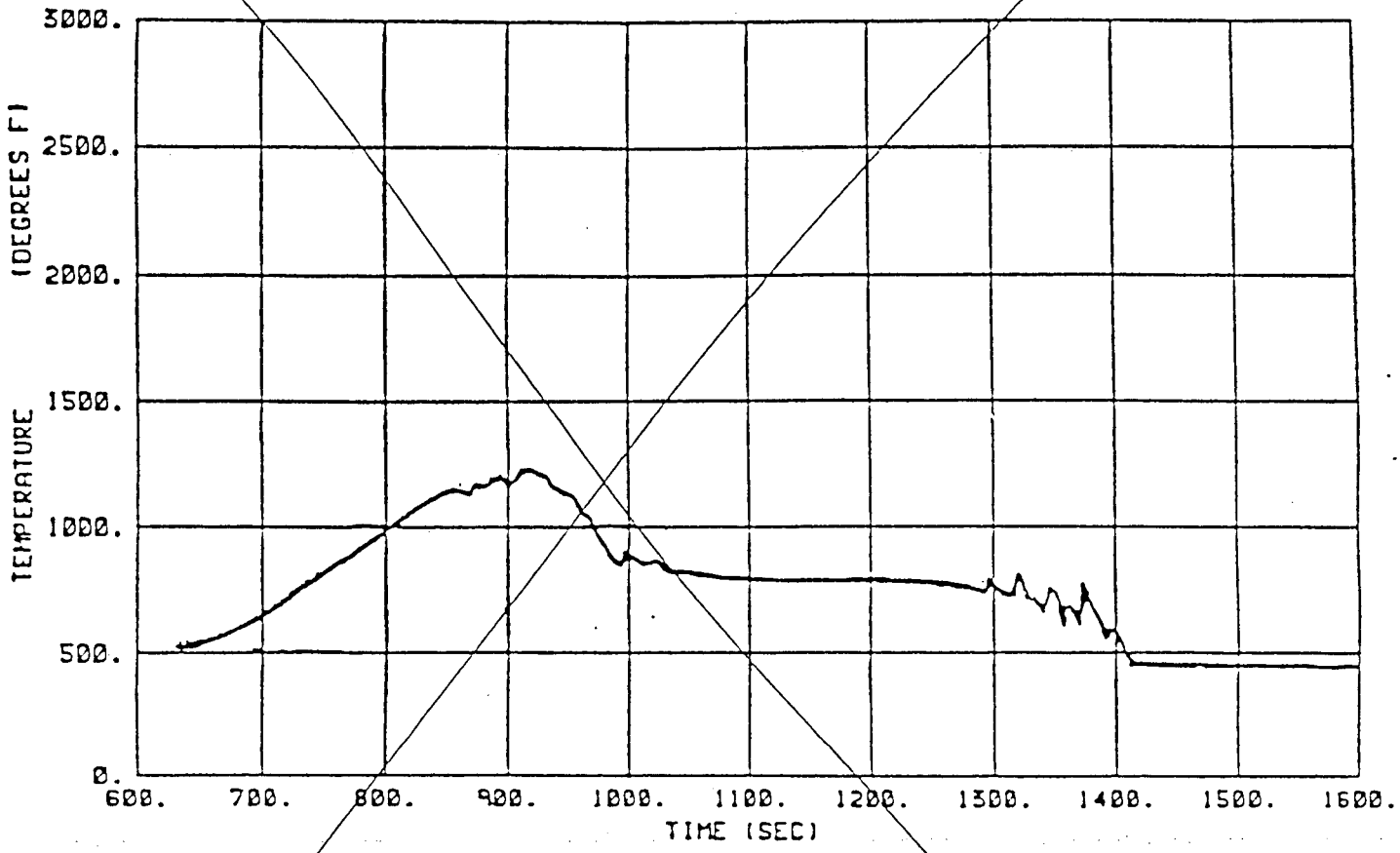
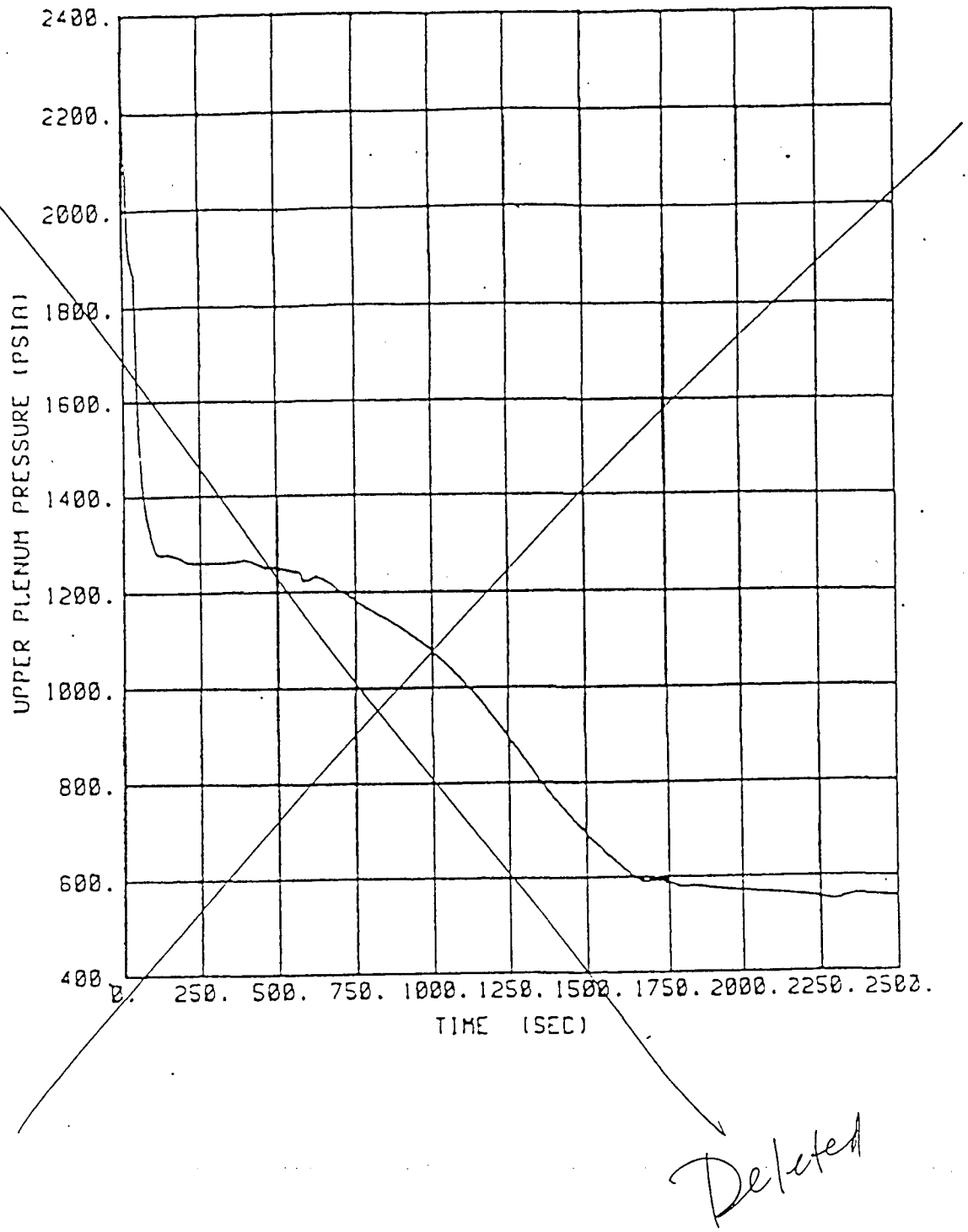


FIGURE 15.3-8g HOT SPOT FLUID TEMPERATURE, 4-INCH BREAK, UPFLOW BARREL/BAFFLE REGION



AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

REACTOR COOLANT SYSTEM
PRESSURE, 3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8h

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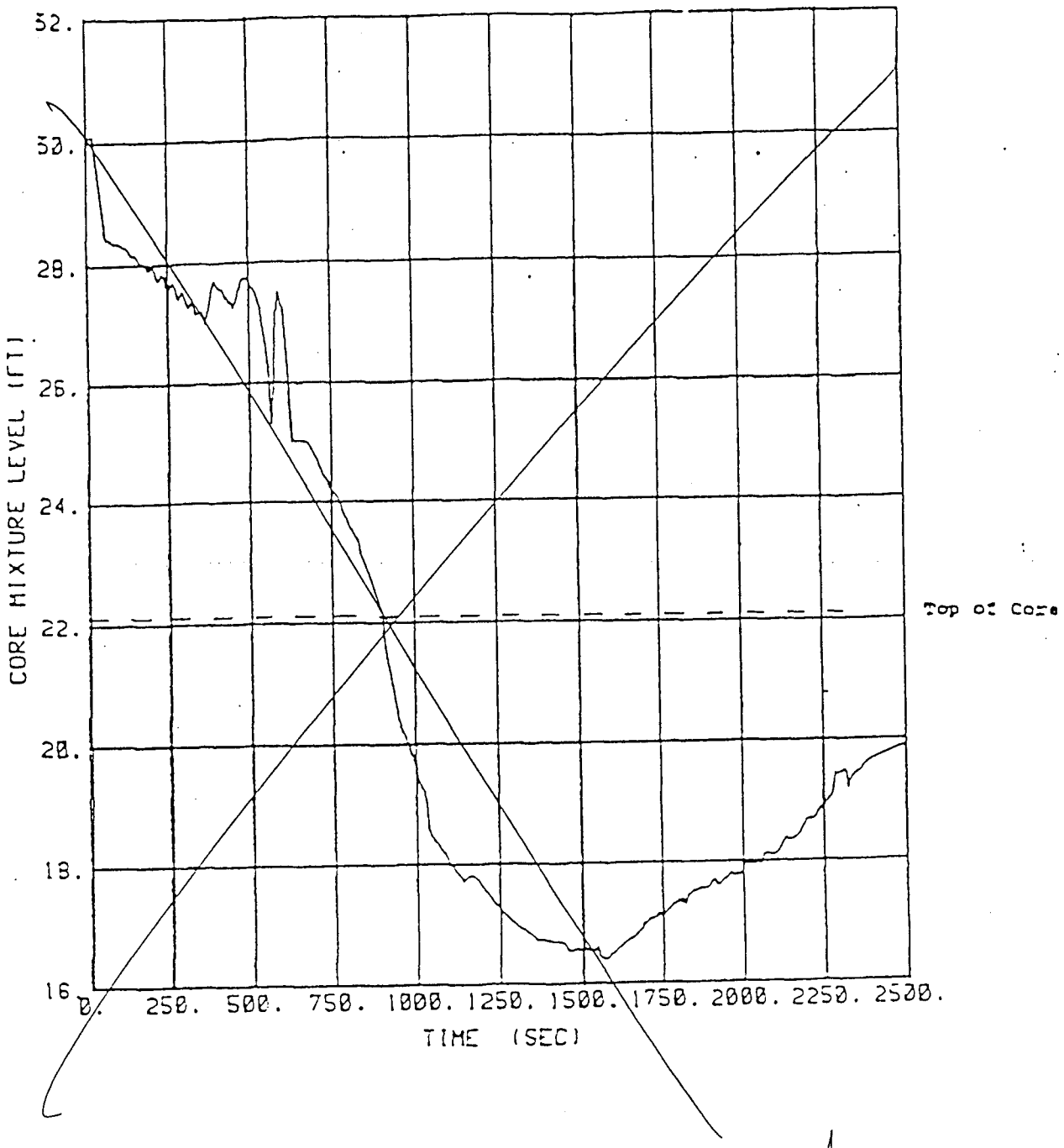

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AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

MIXTURE LEVEL TRANSIENT
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION

FIGURE 15.3-8i

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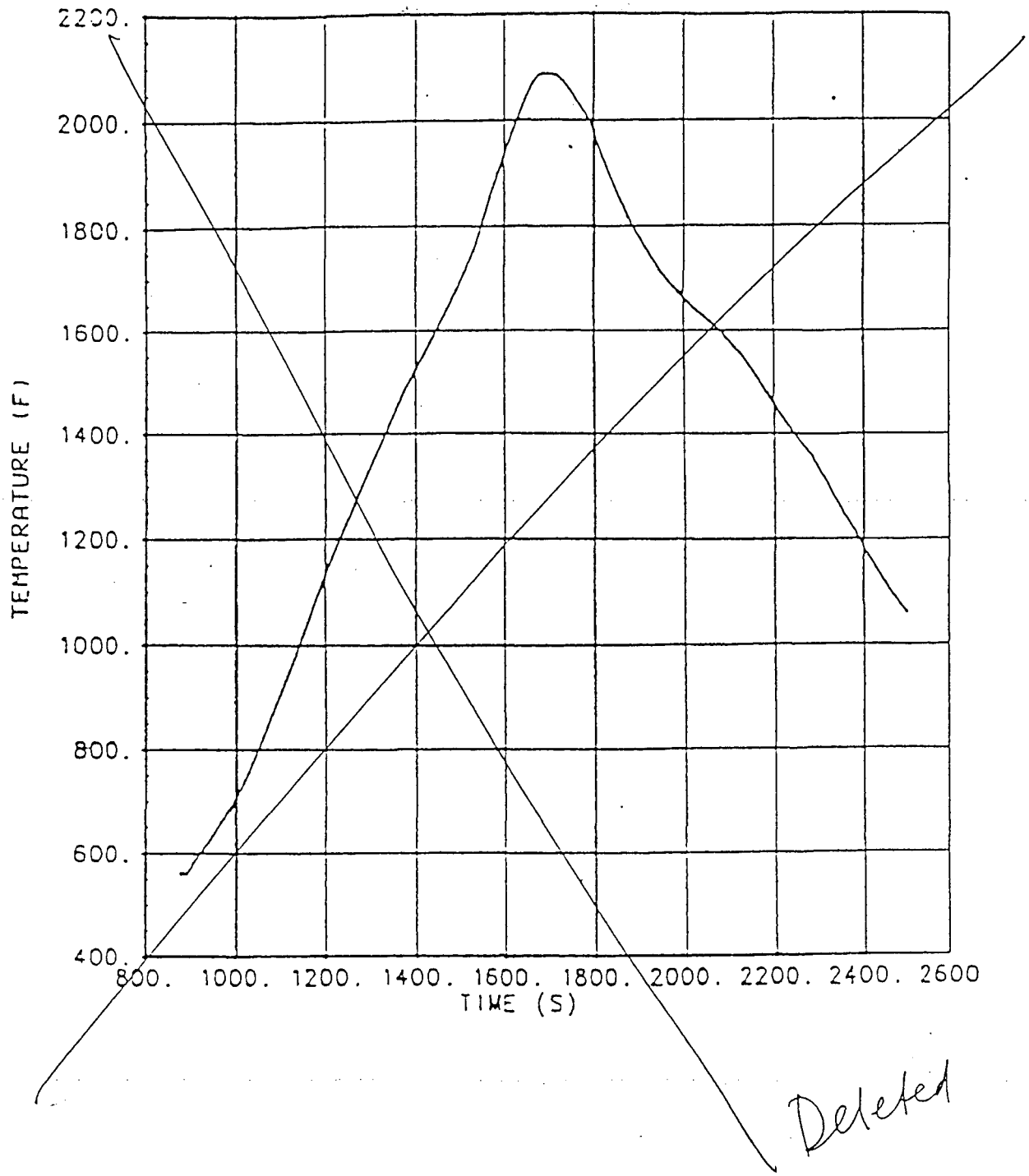

AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

PEAK CLAD TEMPERATURE TRANSIENT
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8j

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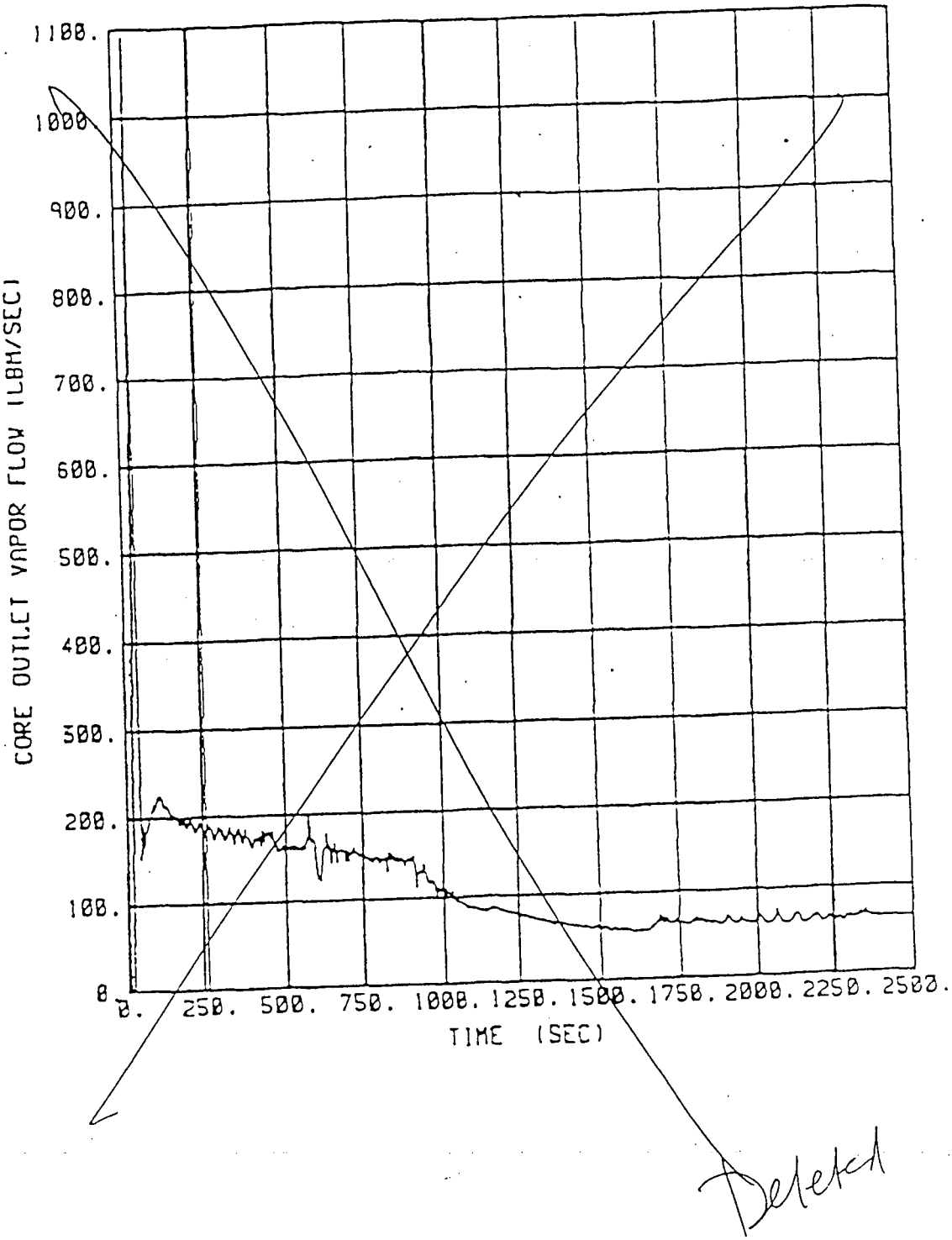

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AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

STEAM FLOW RATE
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8k

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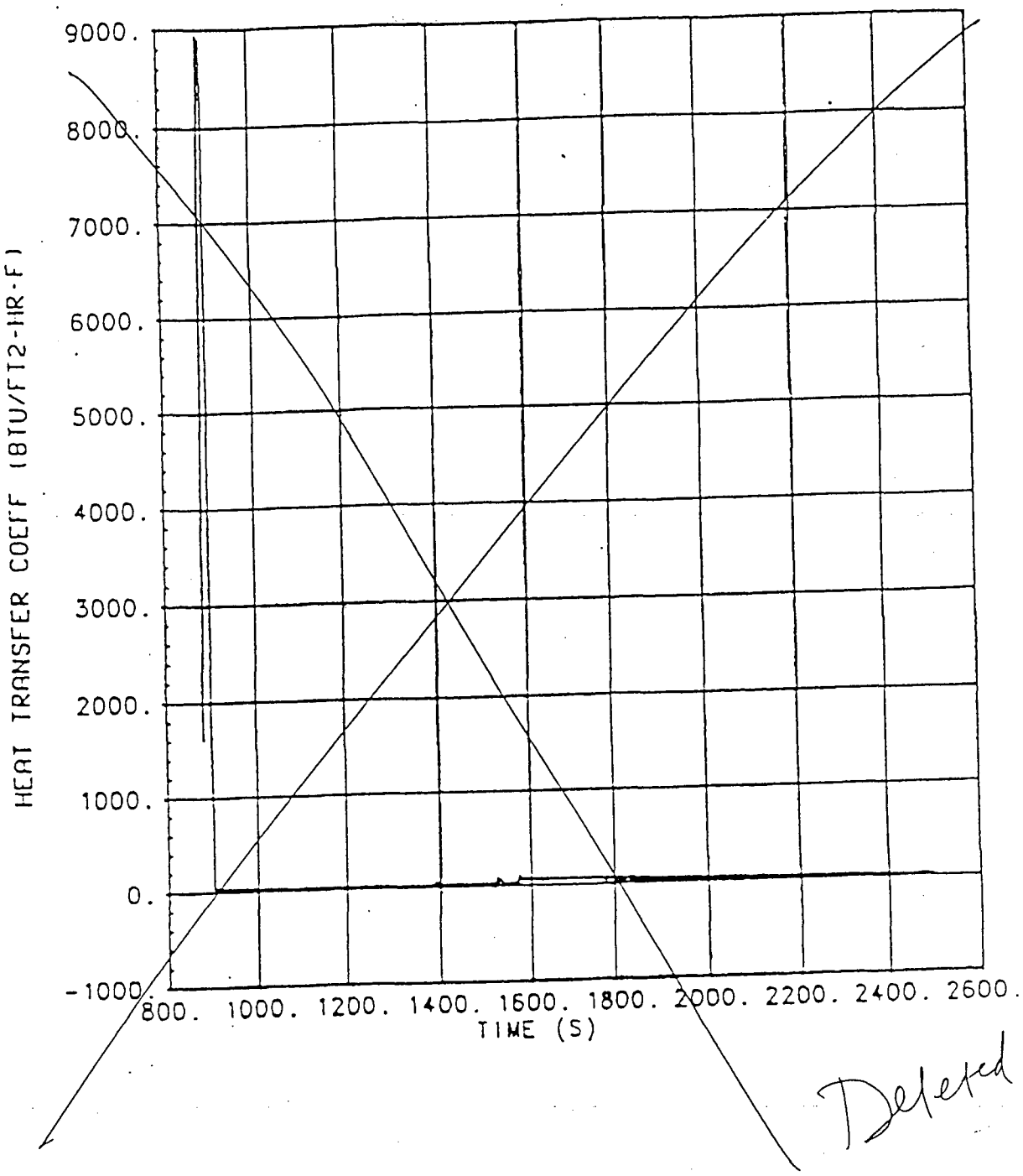

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AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

HEAT TRANSFER COEFFICIENT
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8L

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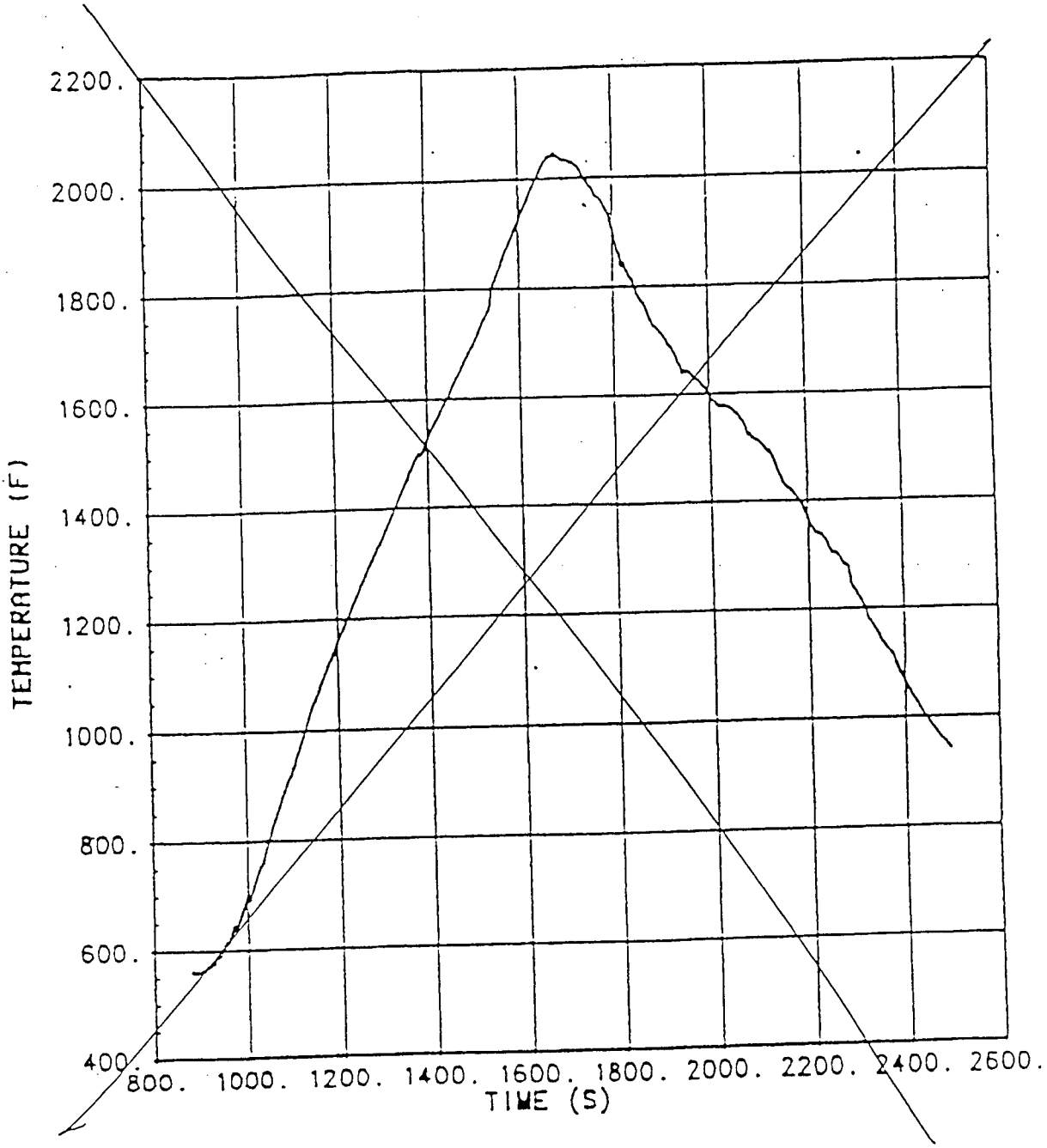

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AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

HOT SPOT FLUID TEMPERATURE
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8m

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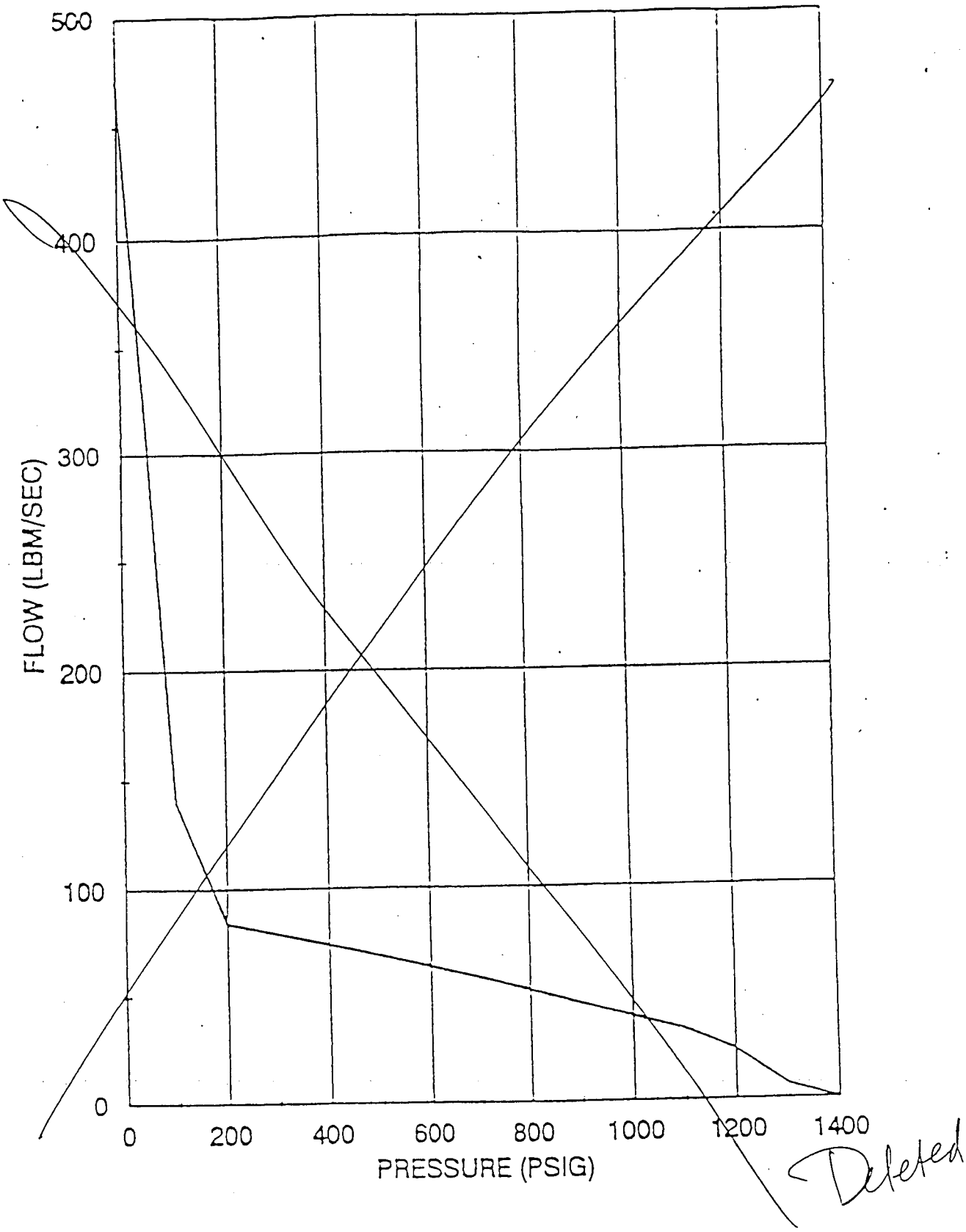

AMENDMENT 80

WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

SAFETY INJECTION VS. RCS PRESSURE
3-INCH BREAK
UPFLOW BARREL/BAFFLE REGION
FIGURE 15.3-8n

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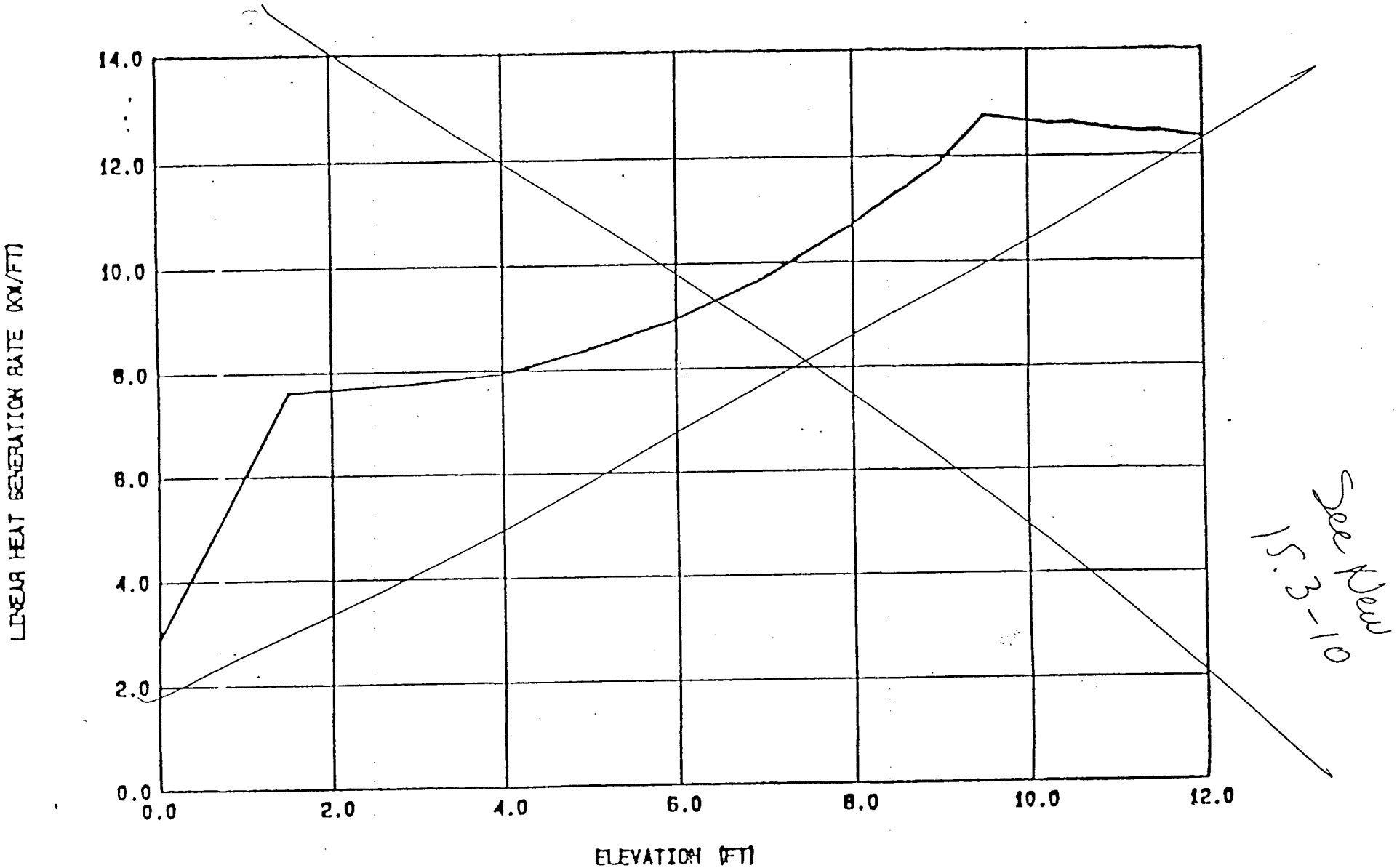
FIGURE 15.3-10 HOT ROD AXIAL POWER SHAPE



See New 15.3-10

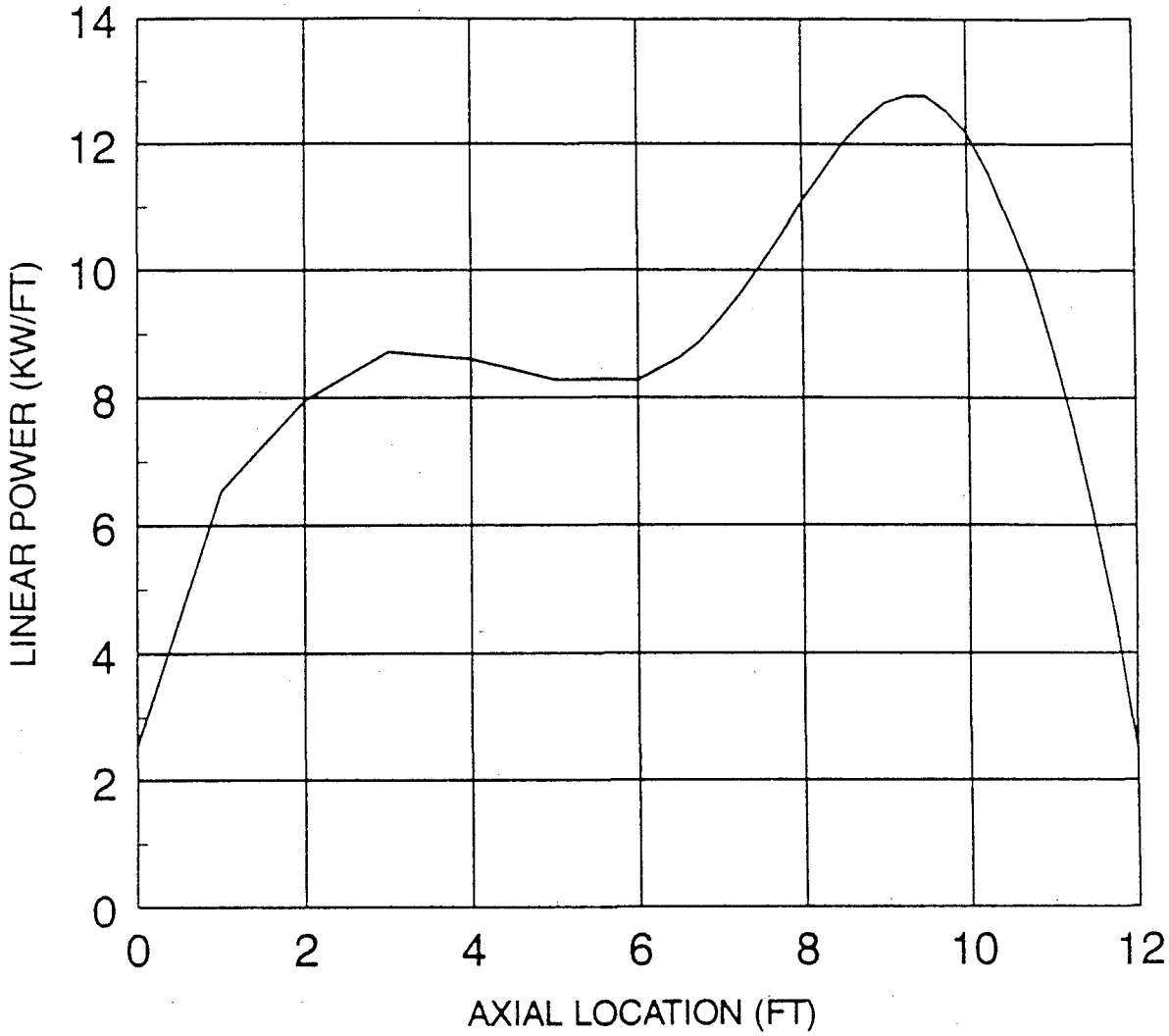


Figure 15.3-10 HOT ROD AXIAL POWER SHAPE

See Figures (new)
15.3-11 thru 15.3-11e

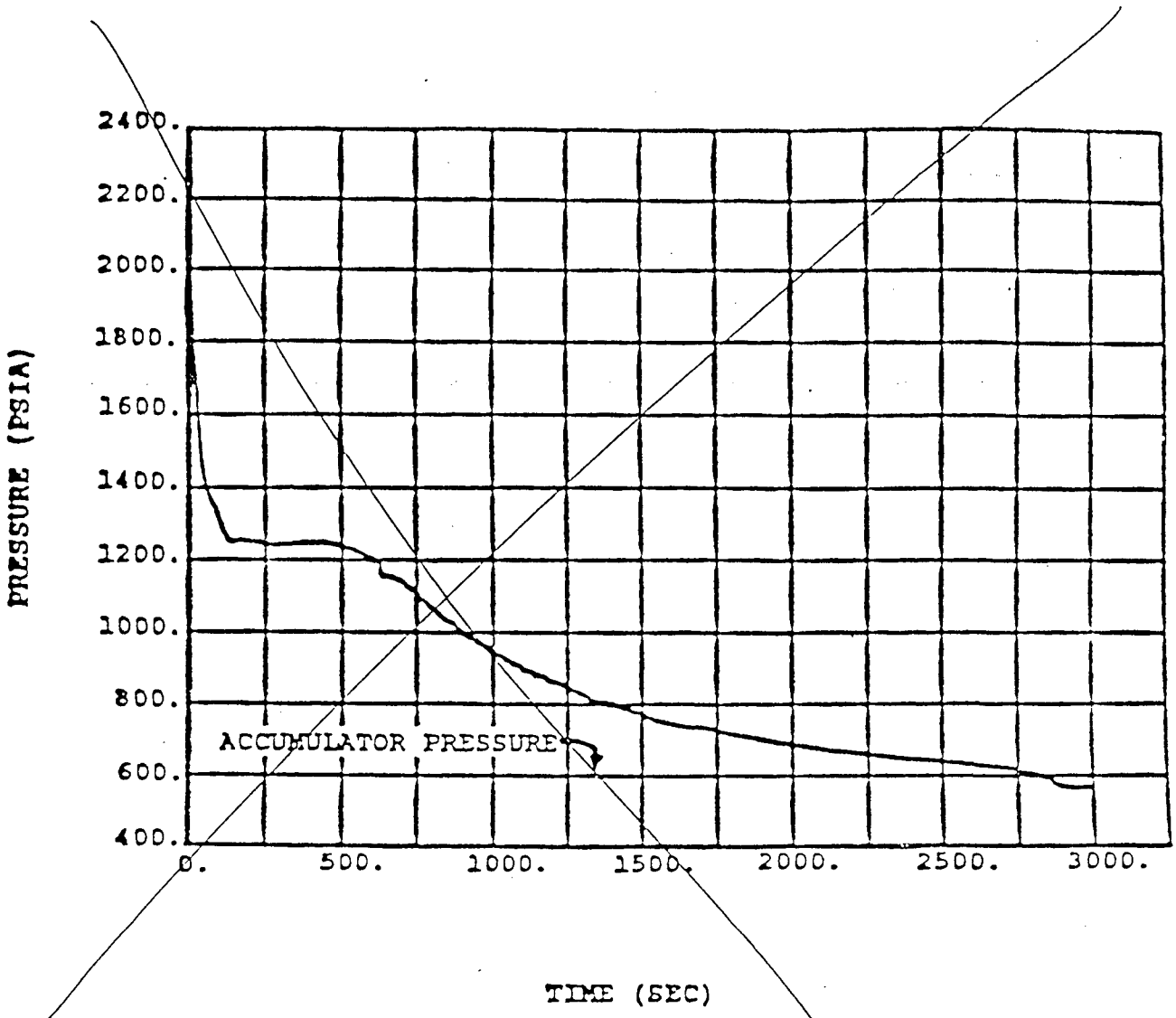


FIGURE 15.3-11 . REACTOR COOLANT SYSTEM PRESSURE FOR THE 3-INCH BREAK

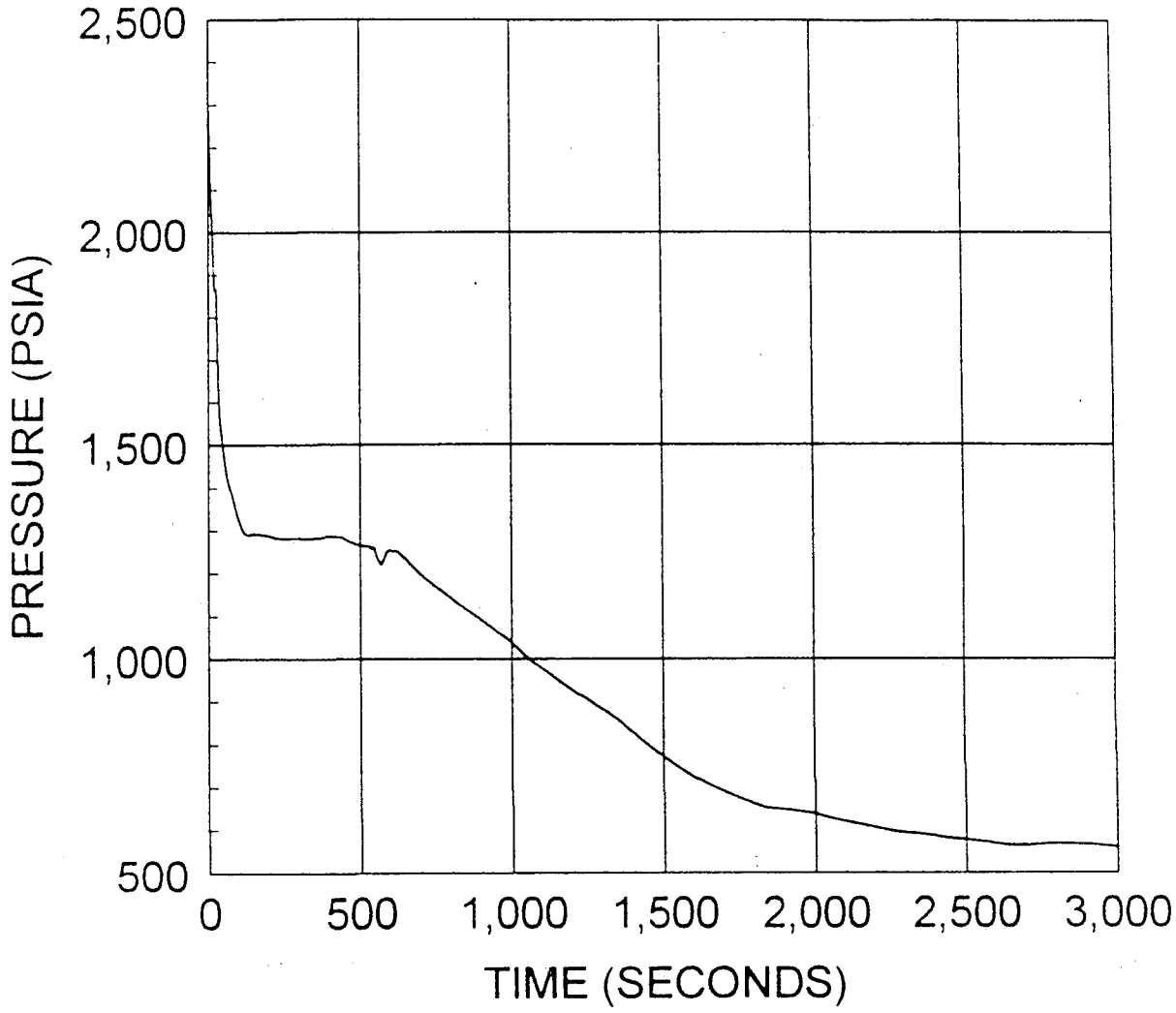


Figure 15.3-11 REACTOR COOLANT SYSTEM PRESSURE FOR 3 INCH BREAK

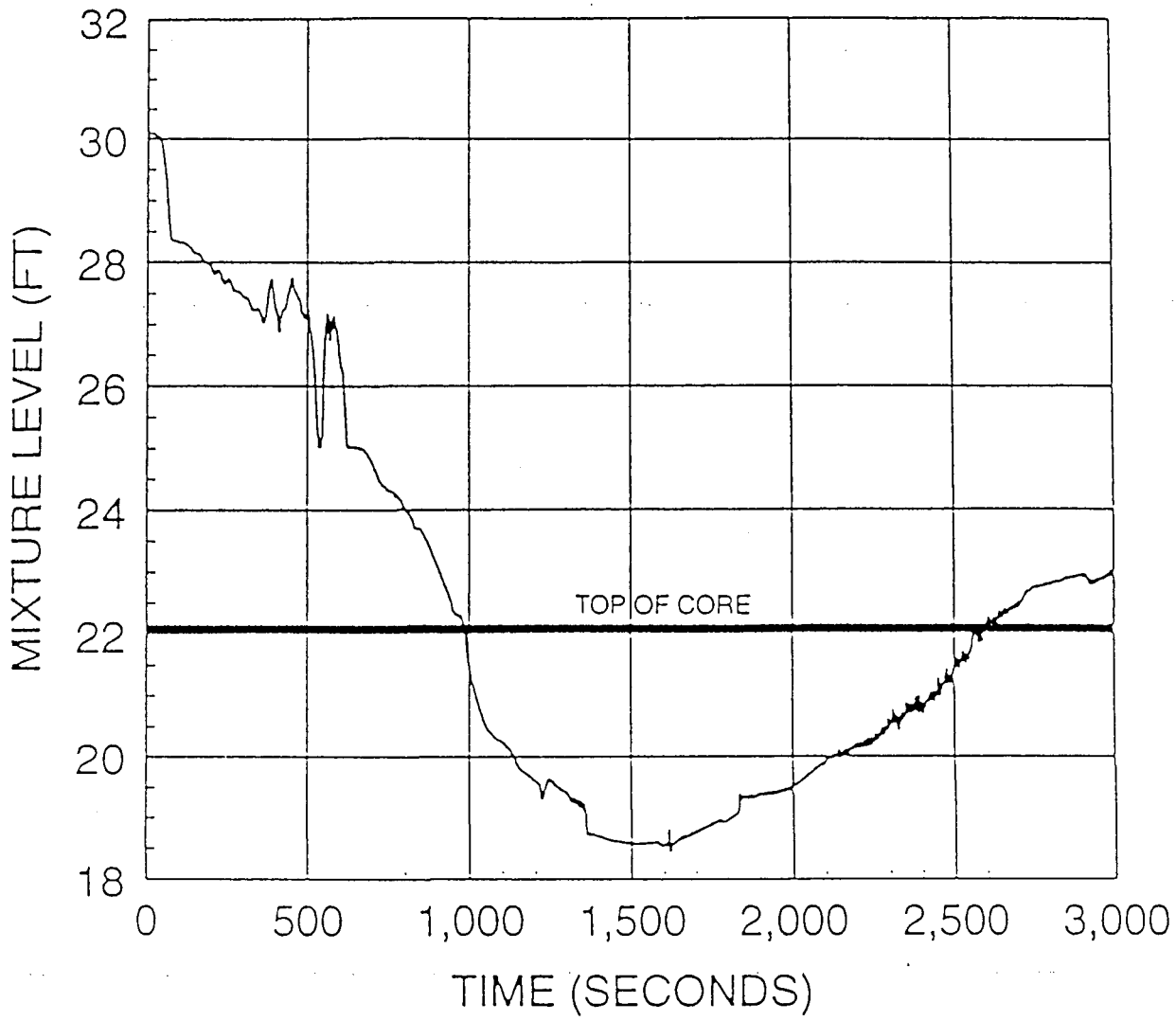


Figure 15.3-~~12~~ 11a CORE MIXTURE LEVEL TRANSIENT FOR 3 INCH BREAK

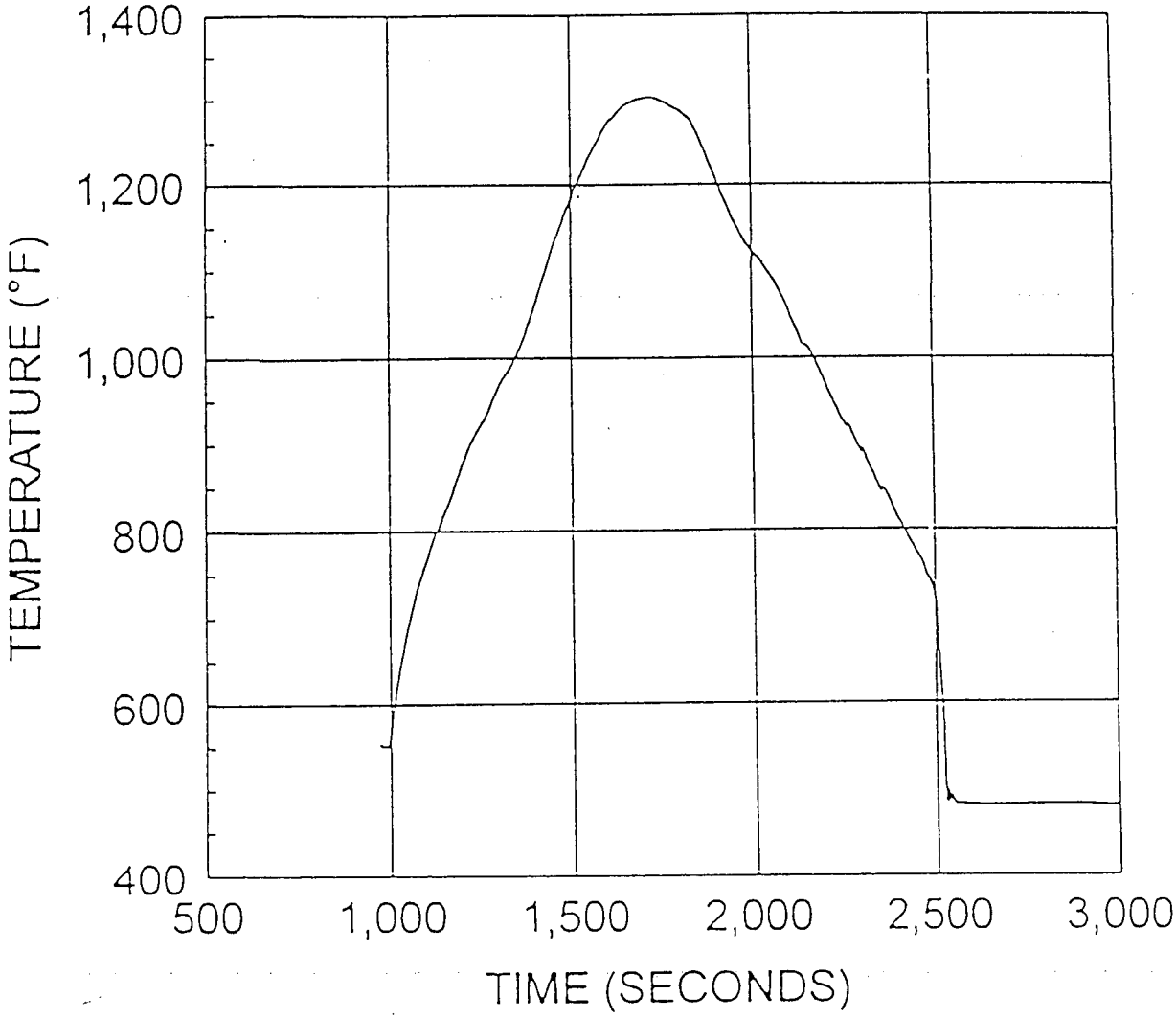


Figure 15.3-~~13~~ 11b CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION FOR 3 INCH BREAK

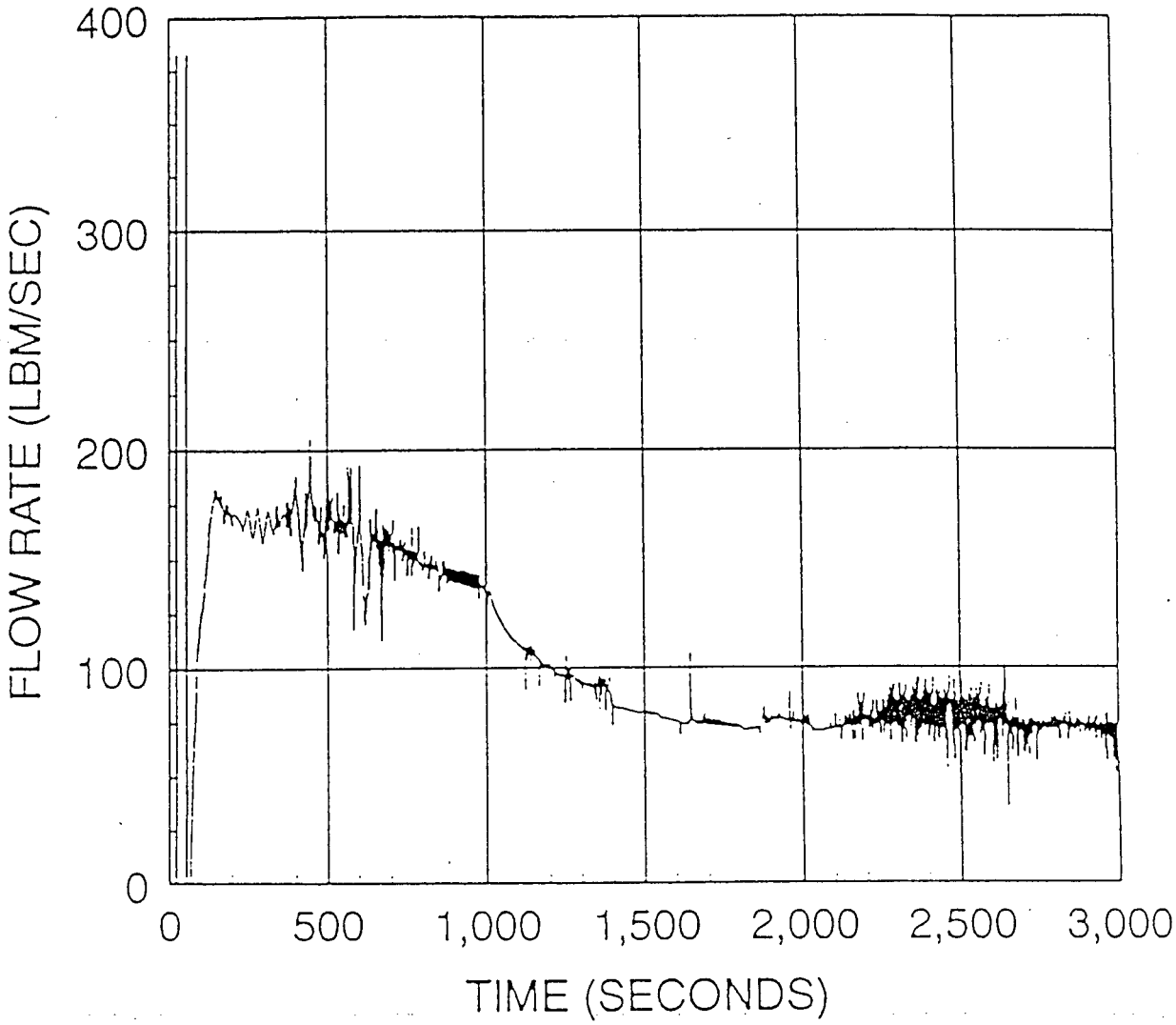


Figure 15.3-~~14~~ 11C CORE OUTLET STEAM FLOW RATE FOR 3 INCH BREAK

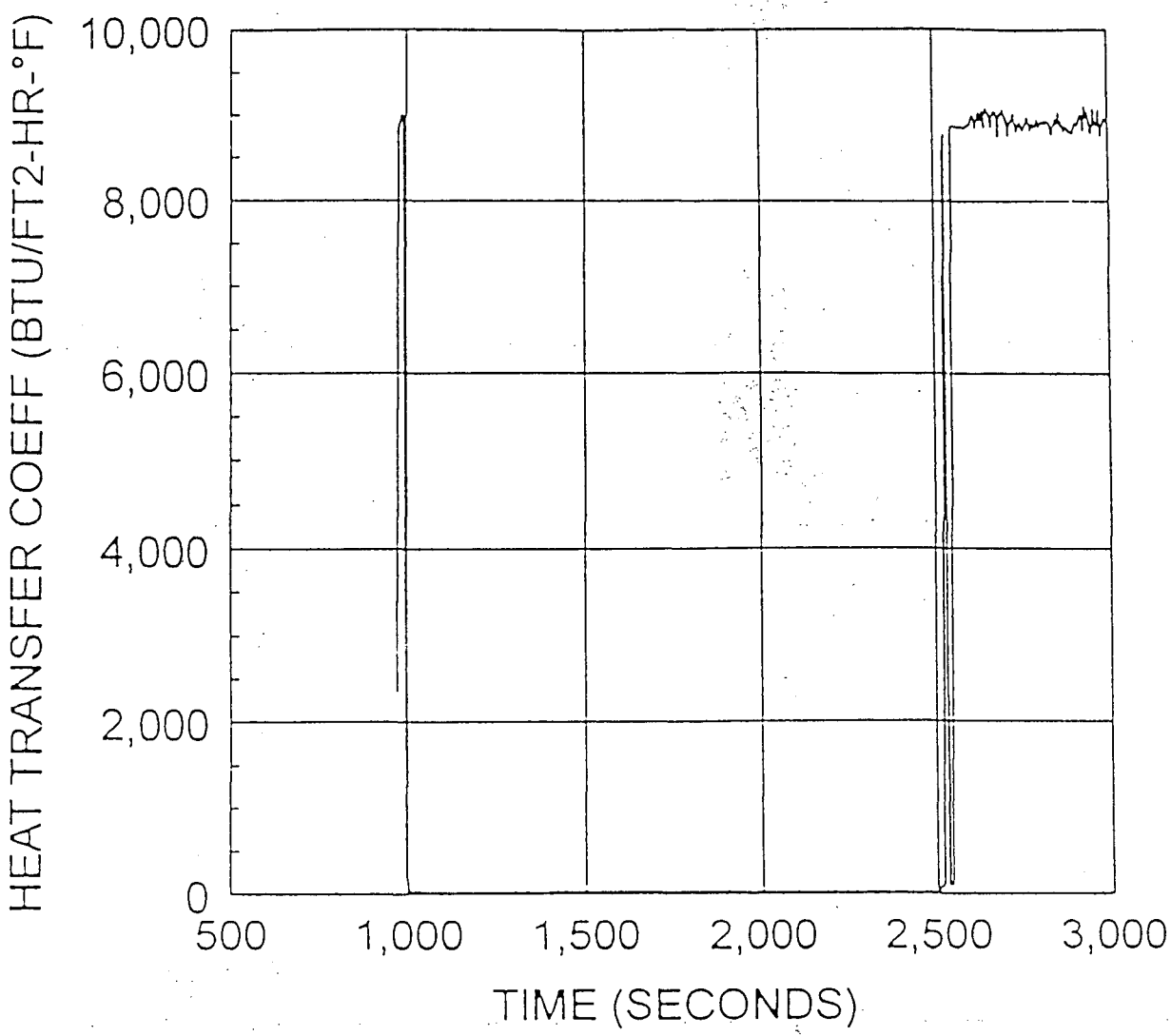


Figure 15.3-~~15~~ 11d CLAD SURFACE HEAT TRANSFER COEFFICIENT AT PEAK TEMPERATURE ELEVATION FOR 3 INCH BREAK

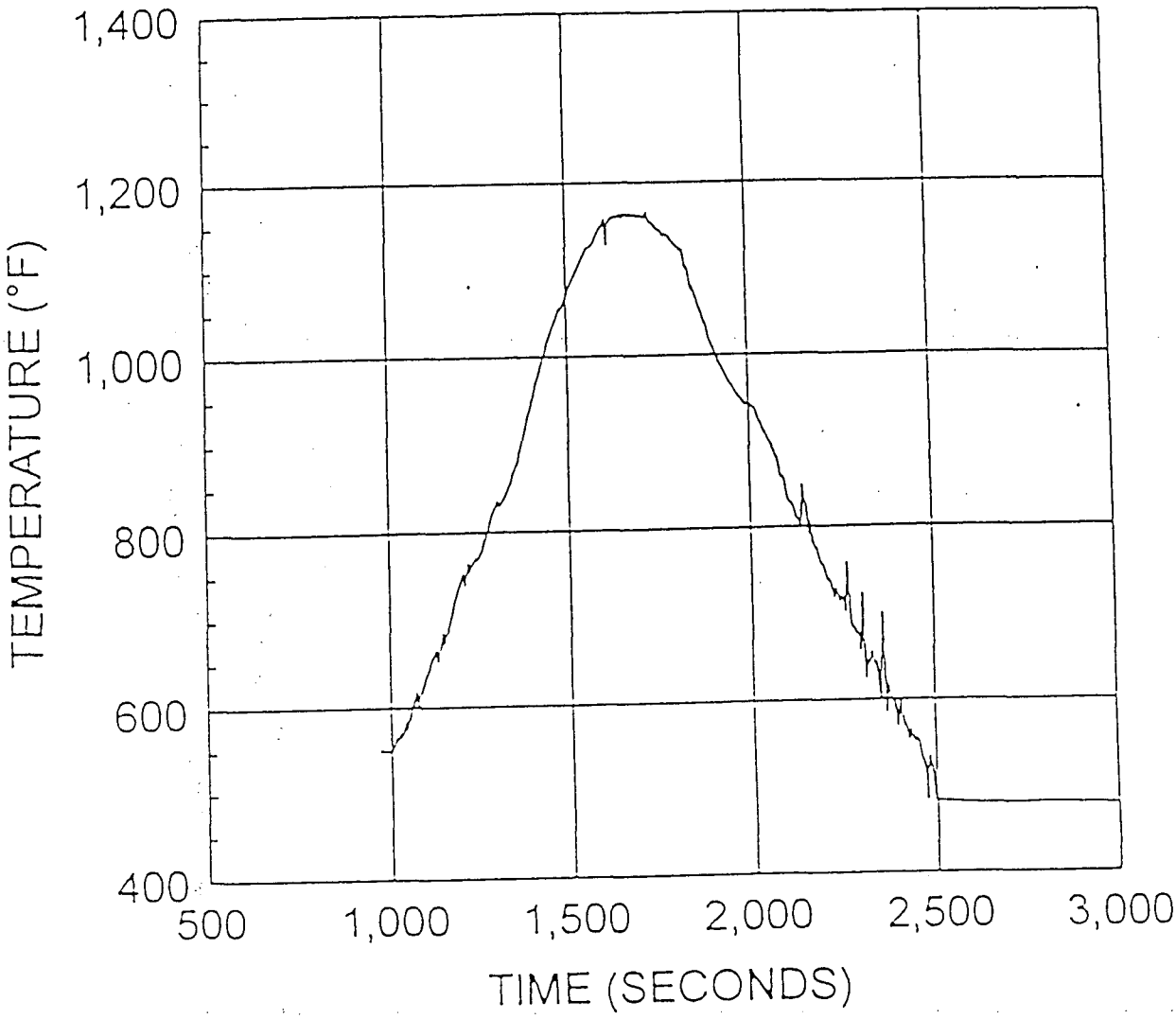


Figure 15.3-~~16~~ 11e FLUID TEMPERATURE AT PEAK CLAD TEMPERATURE ELEVATION FOR 3 INCH BREAK

See New Figures
15.3-12 thru 15.3-12e

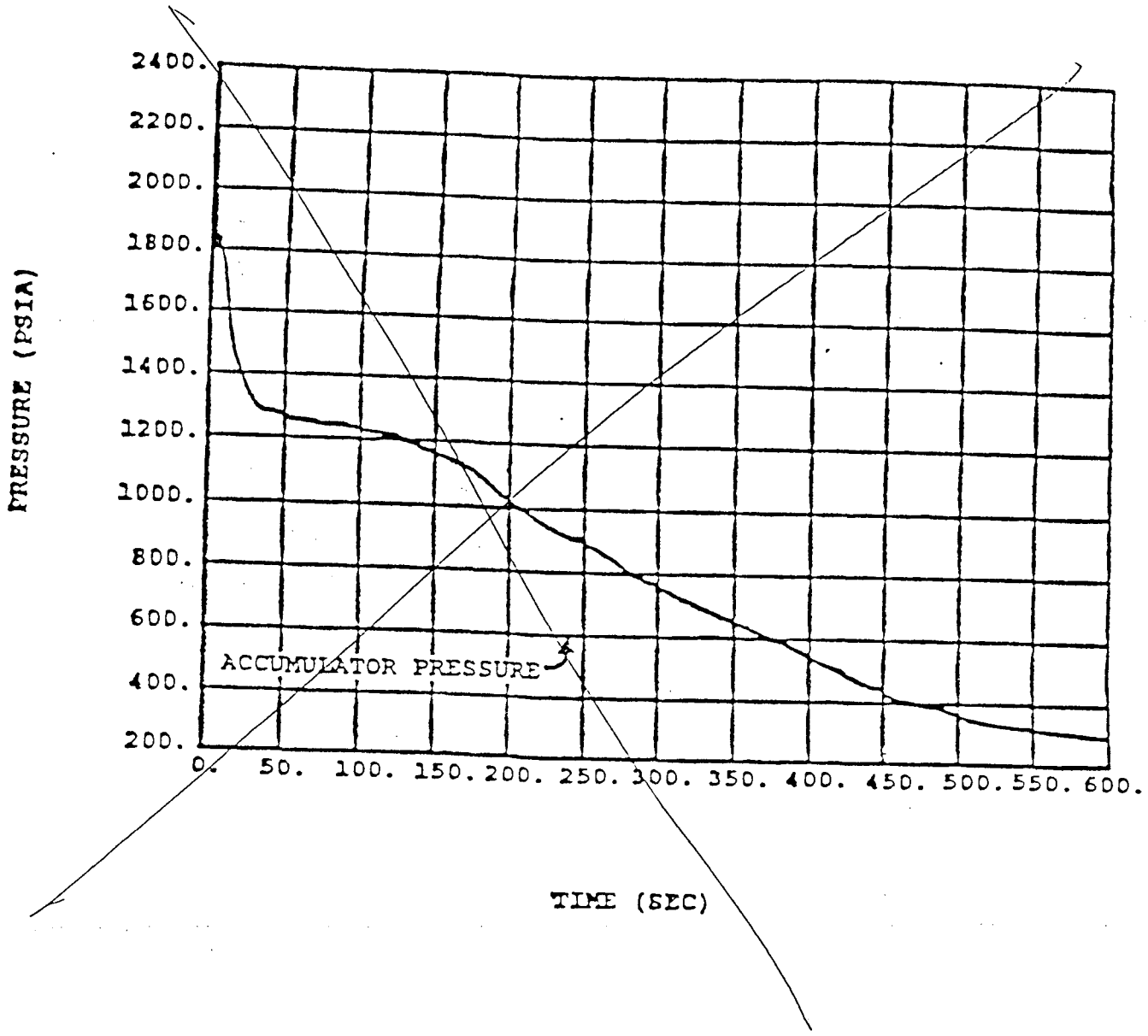


FIGURE 15.3-12 REACTOR COOLANT SYSTEM PRESSURE FOR THE 6-INCH BREAK

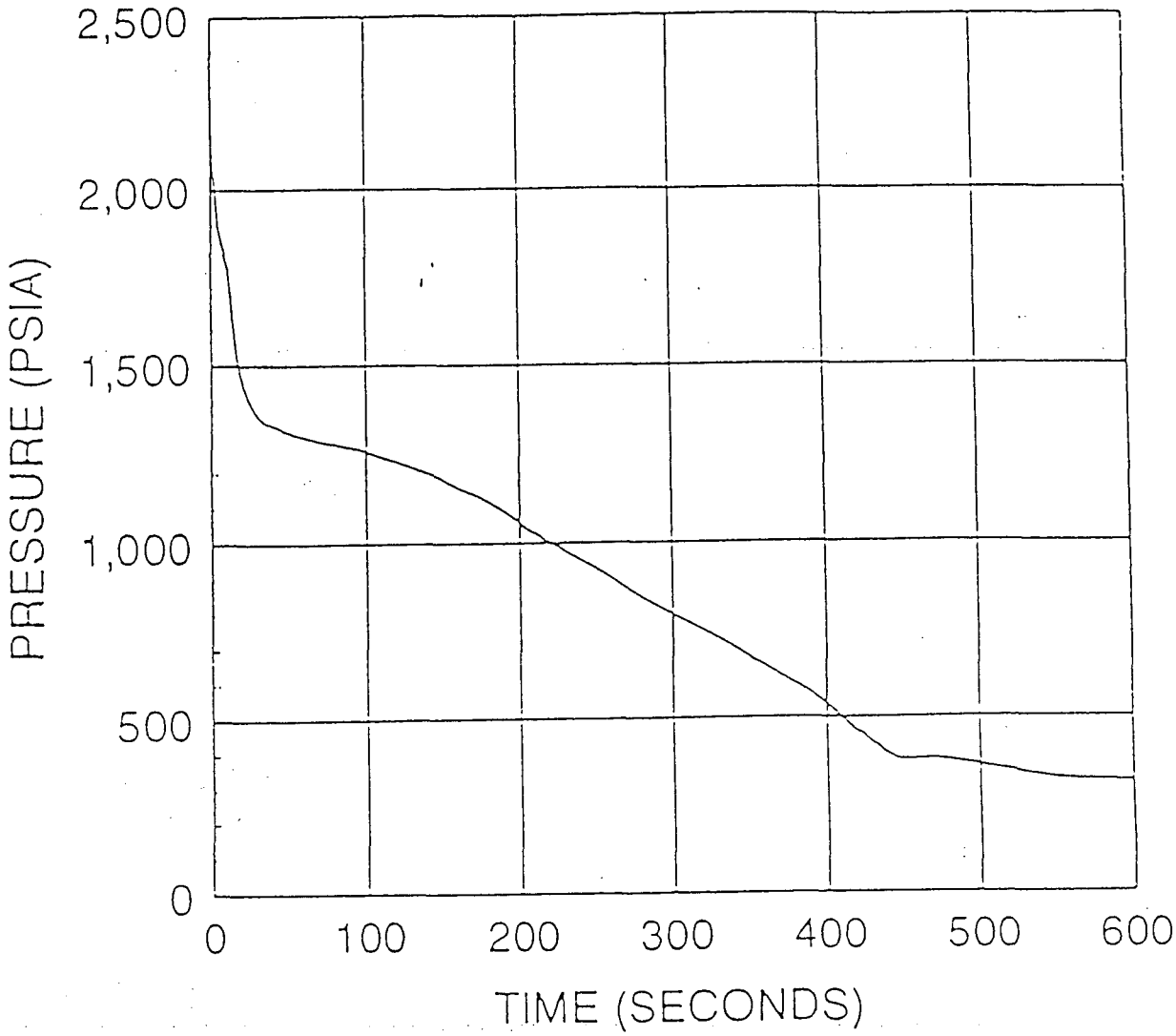


Figure 15.3-~~17~~ 12 REACTOR COOLANT SYSTEM PRESSURE FOR 6 INCH BREAK

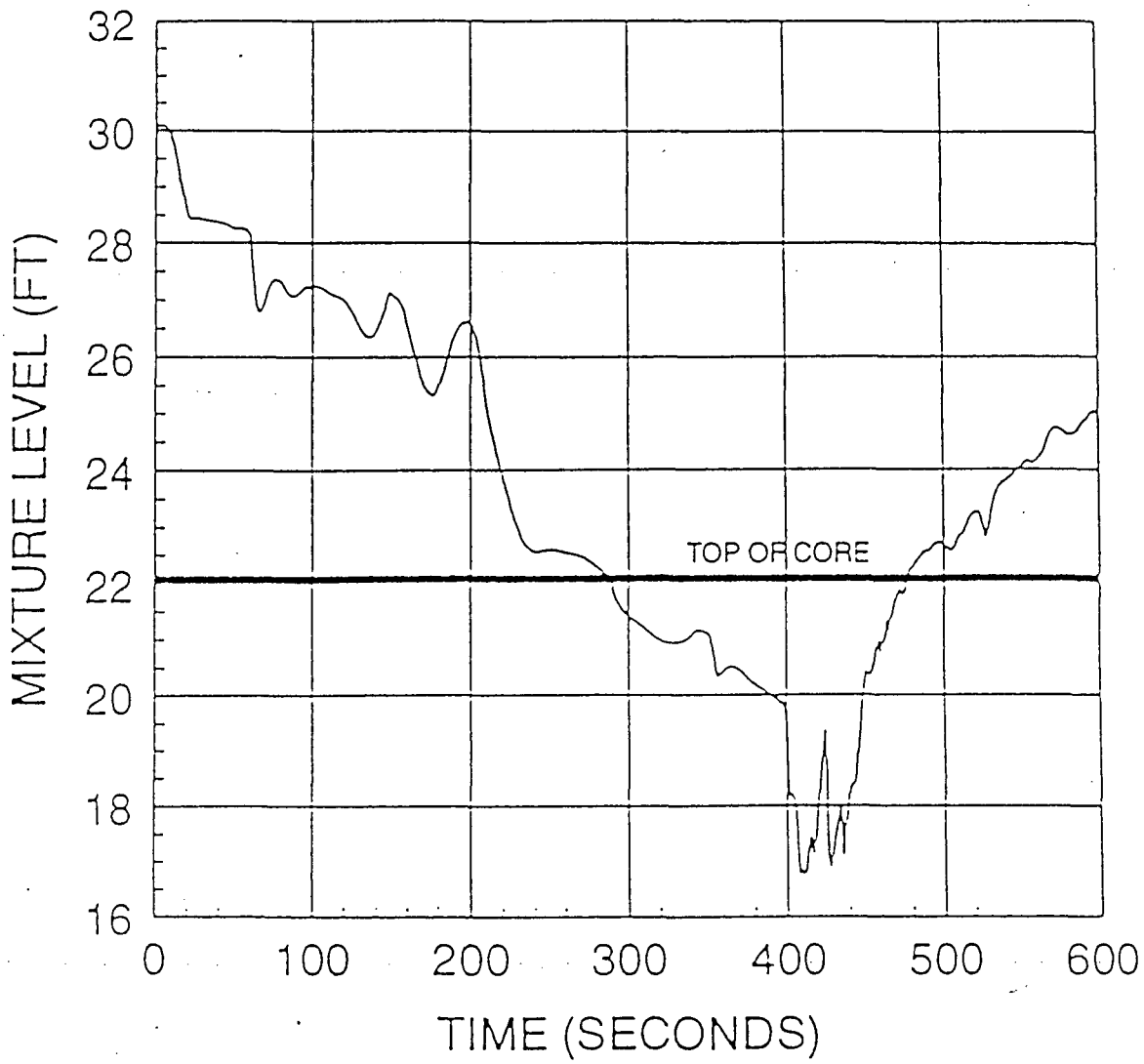


Figure 15.3-~~18~~ 12a CORE MIXTURE LEVEL TRANSIENT FOR 6 INCH BREAK

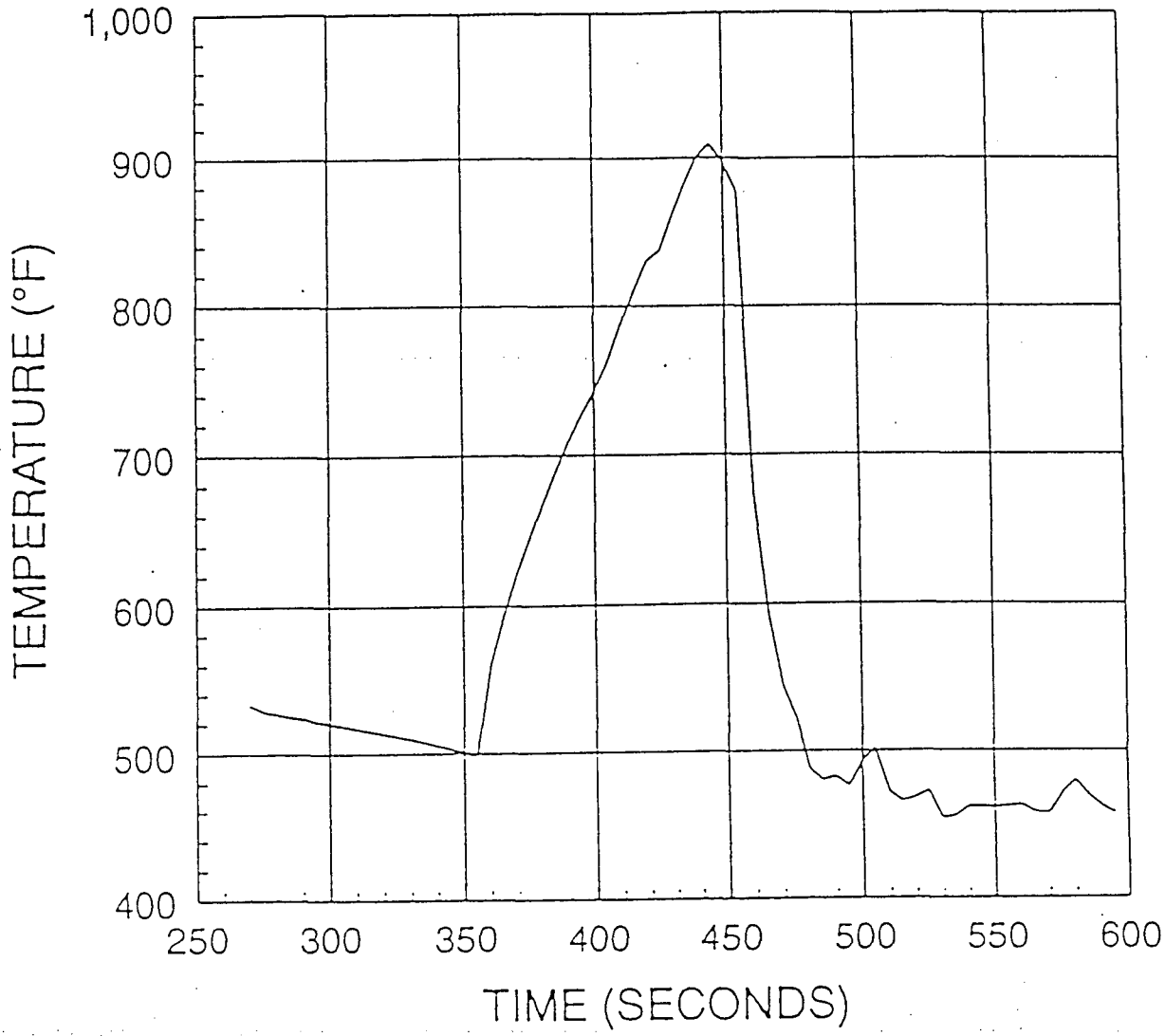
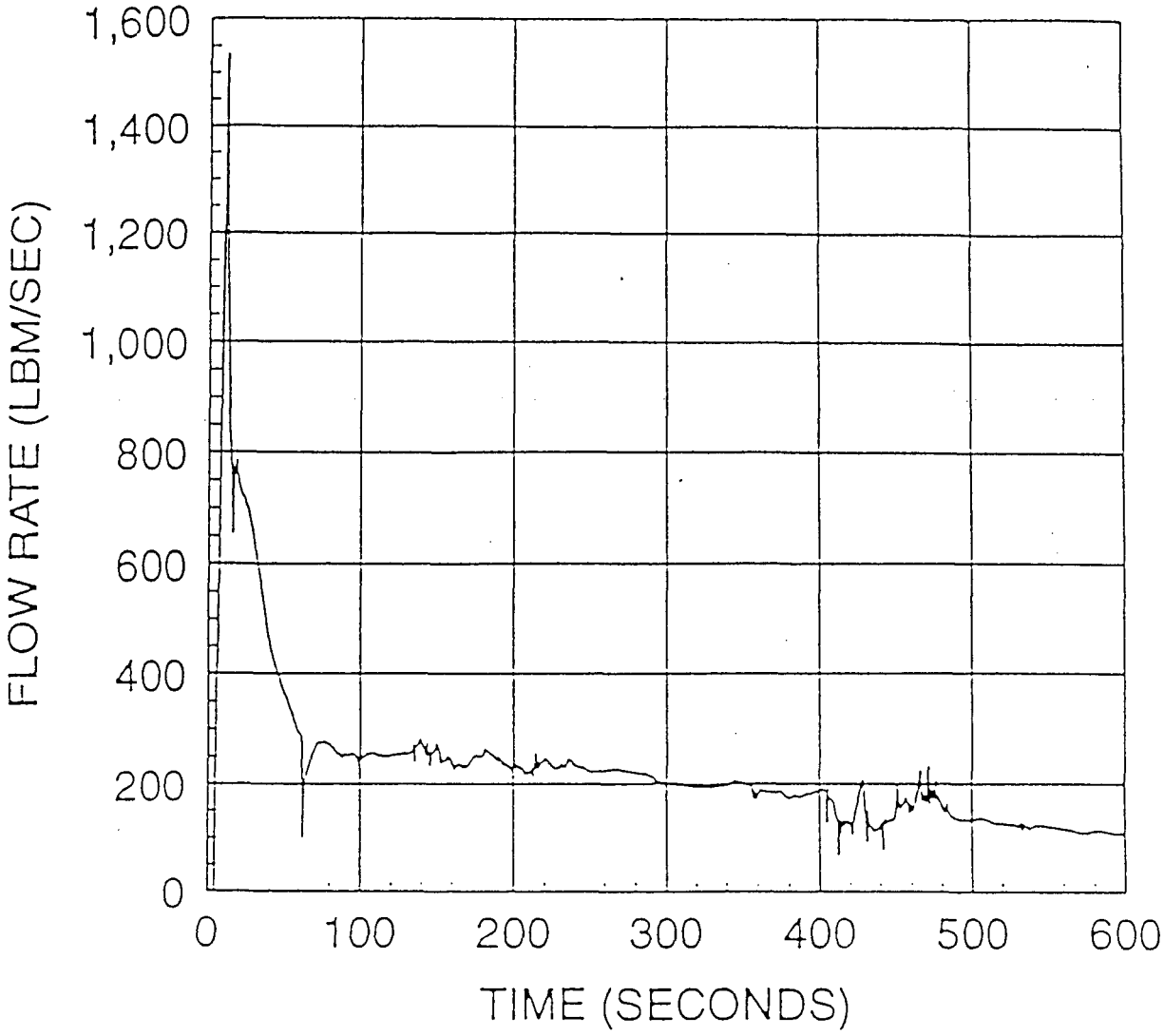


Figure 15.3-~~19~~ 12b CLAD TEMPERATURE TRANSIENT AT PEAK TEMPERATURE ELEVATION FOR 6 INCH BREAK



12C

Figure 15.3-20 CORE OUTLET STEAM FLOW RATE FOR 6 INCH BREAK

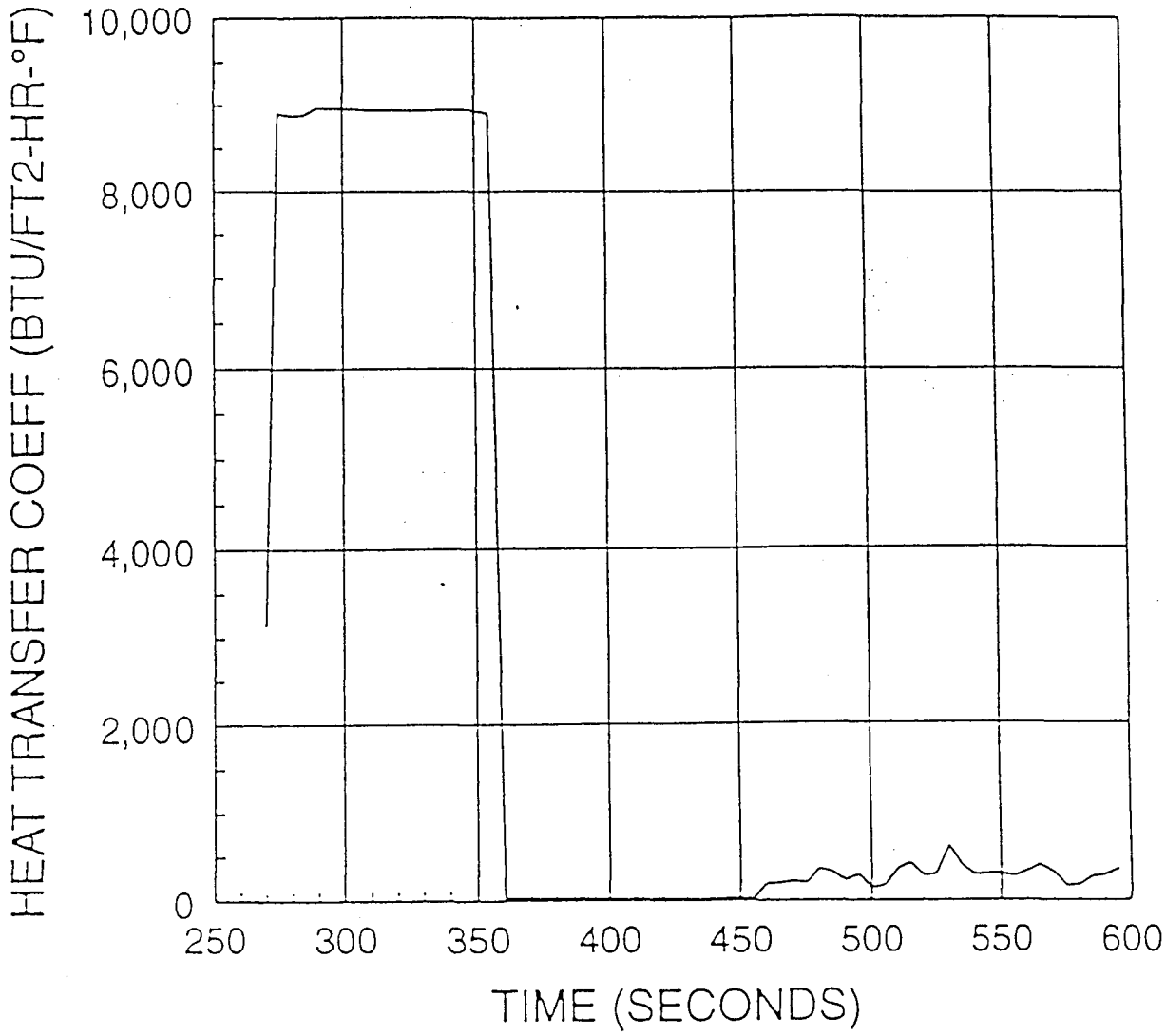


Figure 15.3-~~21~~ 12d CLAD SURFACE HEAT TRANSFER COEFFICIENT AT PEAK TEMPERATURE ELEVATION FOR 6 INCH BREAK

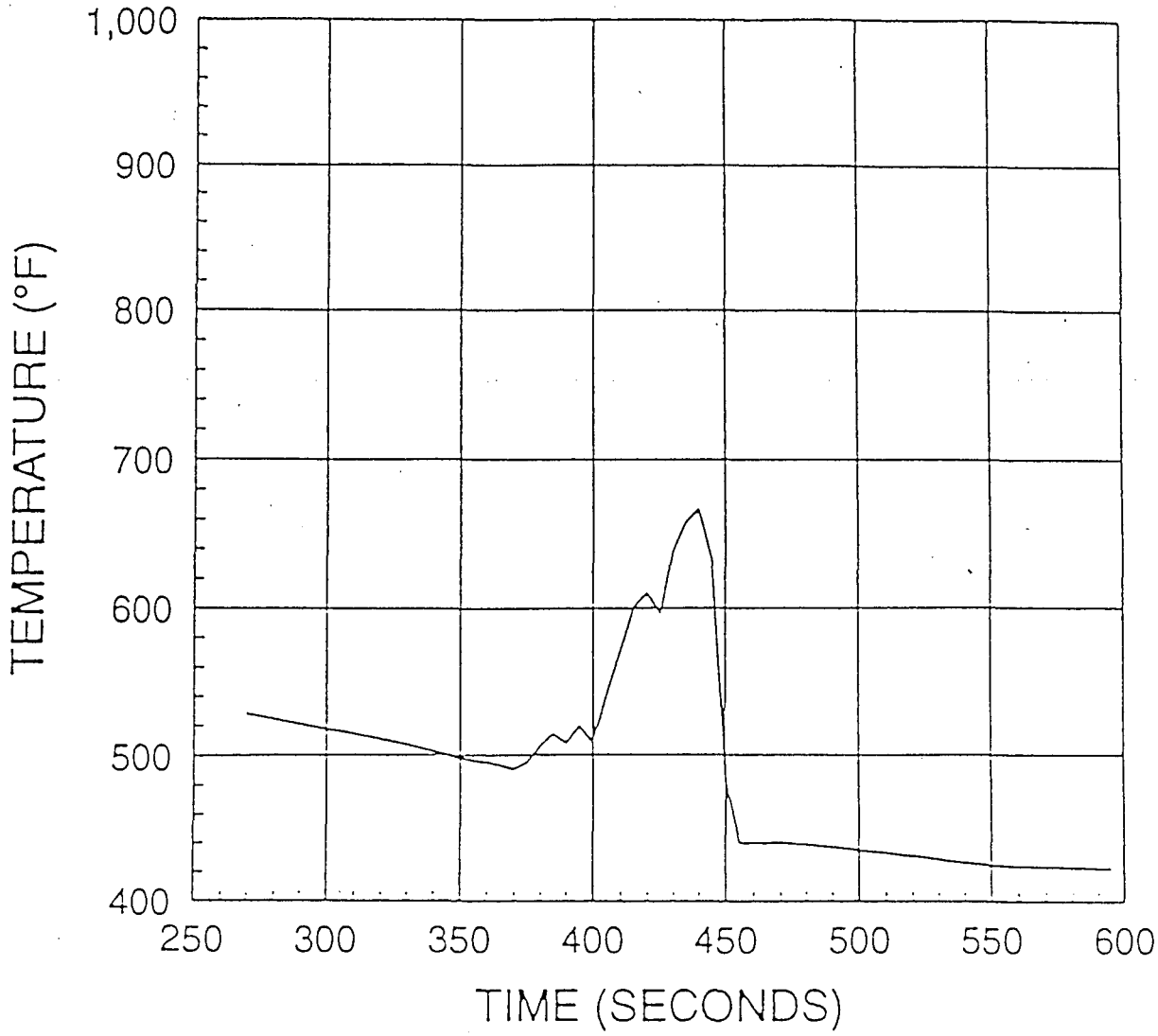


Figure 15.3-22 FLUID TEMPERATURE AT PEAK CLAD TEMPERATURE ELEVATION FOR 6 INCH BREAK

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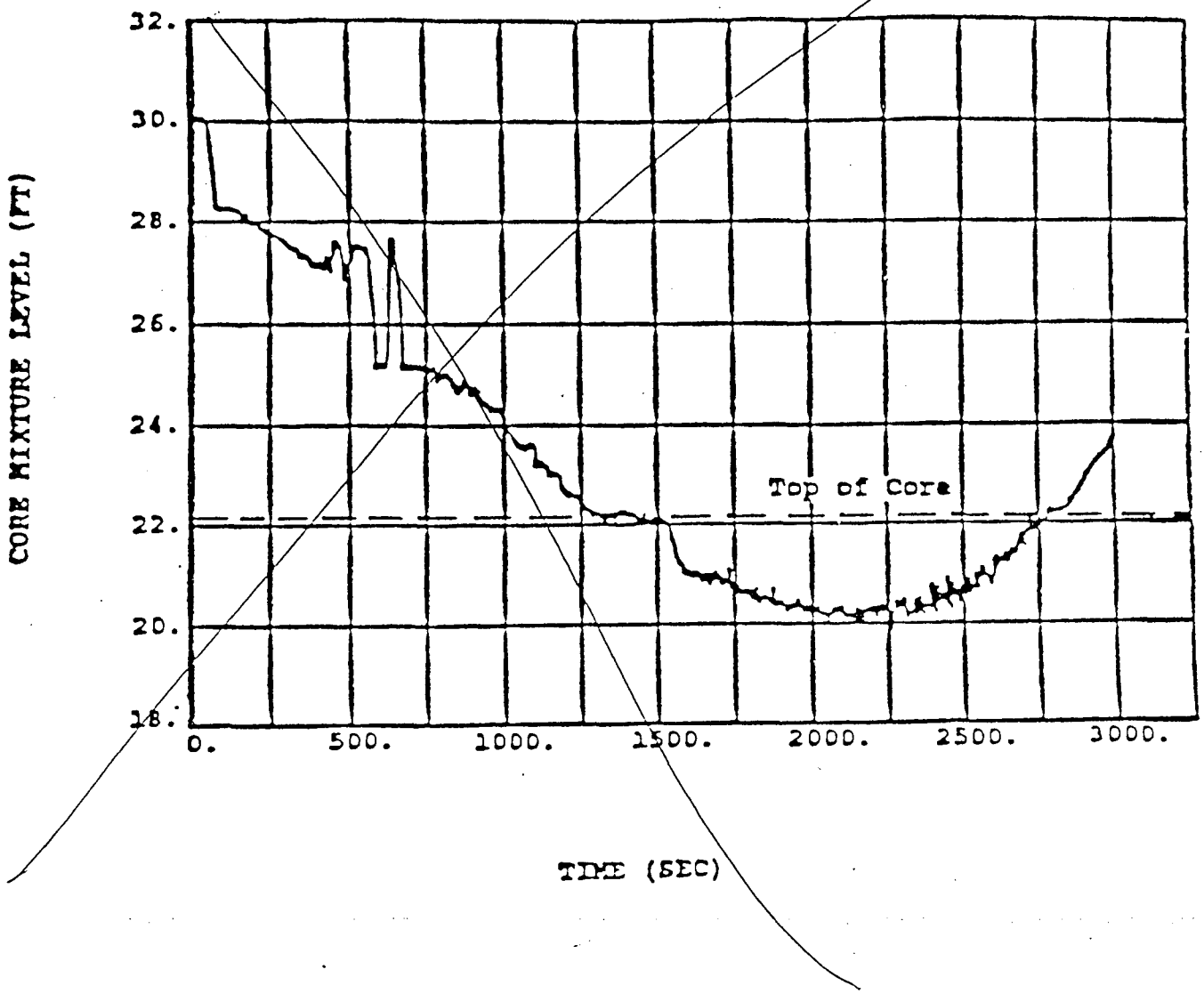

FIGURE 15.3-13a CORE MIXTURE HEIGHT TRANSIENT FOR THE 3-INCH BREAK

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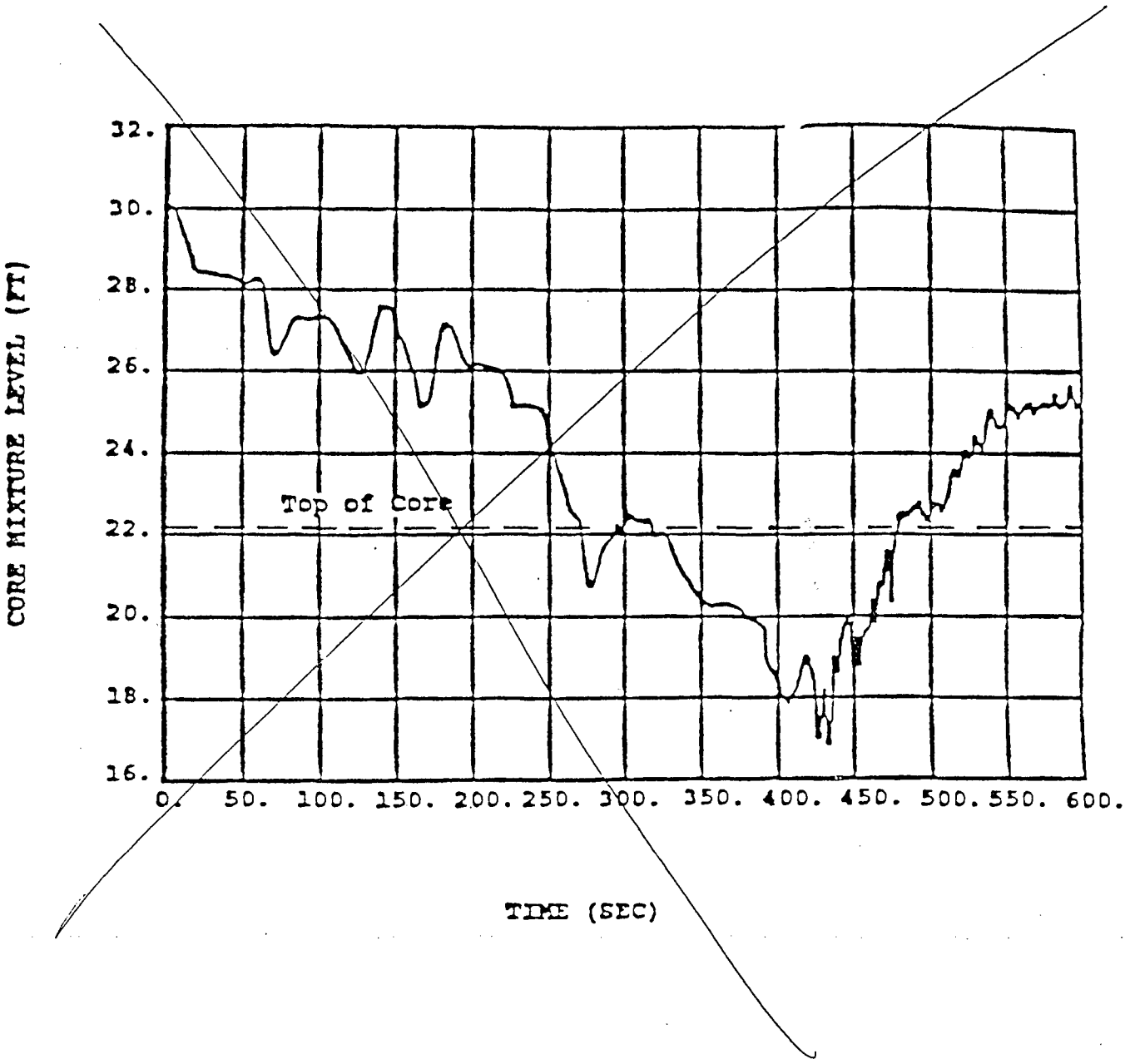

FIGURE 15.3-13b. CORE MIXTURE HEIGHT TRANSIENT FOR THE 6-INCH BREAK

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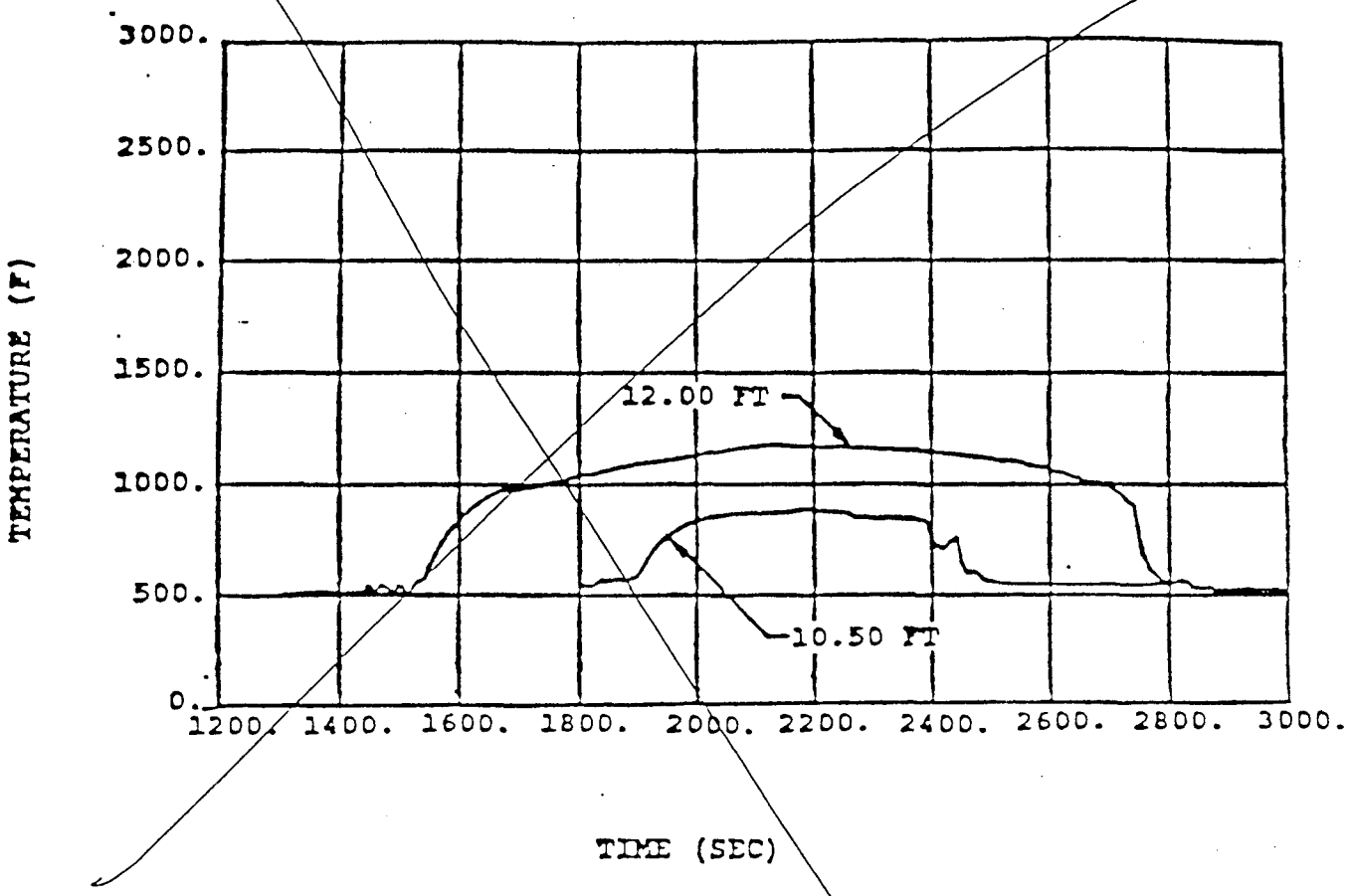

FIGURE 15.3-14 a CLAD TEMPERATURE TRANSIENT FOR THE 3-INCH BREAK

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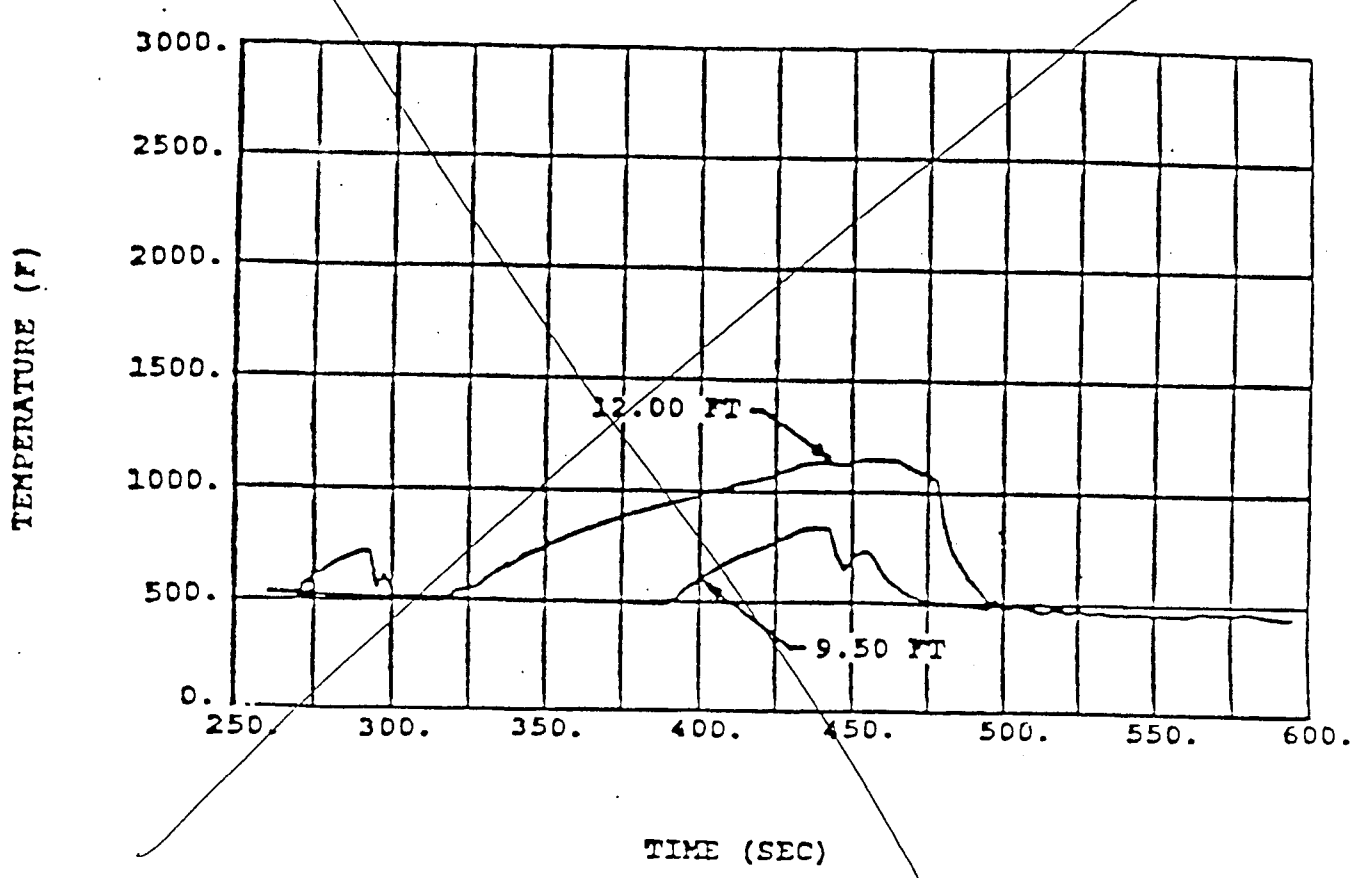


FIGURE 15.3-14 b CLAD TEMPERATURE TRANSIENT FOR THE 6-INCH BREAK

reactor coolant system peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[12] Code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN Code includes a film boiling heat transfer coefficient.

One reactor coolant pump seizure has been analyzed for a locked rotor/shaft break with four loops in operation.

The accident is evaluated without offsite power available. For the evaluation, power is assumed to be lost to the unaffected pumps instantaneously after reactor trip. At the beginning of the postulated locked rotor accident, i.e. at the time the shaft in one of the reactor coolant pumps in assumed to seize, the plant is assumed to be in operation under the most adverse steady state operating conditions, i.e., maximum steady state power level, maximum steady state pressure, and maximum steady state coolant average temperature.

When the peak pressure is evaluated, the initial pressure is conservatively estimated as 46 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response shown in Figure 15.4-15 is at the point in the reactor coolant system having the maximum pressure.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin 1.2 seconds after the flow in the affected loop reaches 87% to nominal flow. No credit is taken for the pressure reducing effect of the pressurizer relief valves, pressurizer spray, steam dump or controlled feedwater flow after plant trip.

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

The pressurizer safety valves are full open at ~~2575~~ 2580 psia and their capacity for steam relief is as described in Section 5.2.2.

Evaluation of DNB in the Core During the Accident

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

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN Code using the Bishop-Sandberg-Tong film boiling correlation[19]. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures).

TABLE 15.4-1

(Sheet 1 of 2)

TIME SEQUENCE OF EVENTS FOR CONDITION IV EVENTS

Accident	Event	Time (Seconds)
Major Reactor Coolant System Pipe Ruptures, Double-Ended Cold Leg Guillotine	See Table 15.4-17	
Major Secondary System Pipe Rupture		
1. Case A		
Complete severence of a pipe, loss of offsite power simultaneous with the break and initiation of safety injection signal	Steam Line Ruptures	0.0
	Low Steam Pressure Setpoint Reached	0.63
	Pressurizer Empties	11.0
	Criticality Attained	22.4
	Boron Reaches Core	56.8
2. Case B		
Complete severence of a pipe, offsite power available	Steam Line Ruptures	0.0
	Low Steam Pressure Setpoint Reached	0.63
	Pressurizer Empties	12.0
	Criticality Attained	24.8
	Boron Reaches Core	73.6
Reactor Coolant Pump Shaft Seizure (Locked Rotor/Broken Shaft)		
All pumps in operation, one shaft seizure without offsite power available	Rotor on one pump seizes	0
	Low flow trip point reached	0.02
	Rods begin to drop	1.22
	Undamaged pumps lose power and begin coasting down	1.22
	Maximum RCS pressure occurs	~~3.1~~ 3.6
	Maximum clad temperature occurs	4.0

TABLE 15.4-9

(Sheet 1 of 1)

TIME SEQUENCE OF EVENTS FOR FEEDLINE BREAK

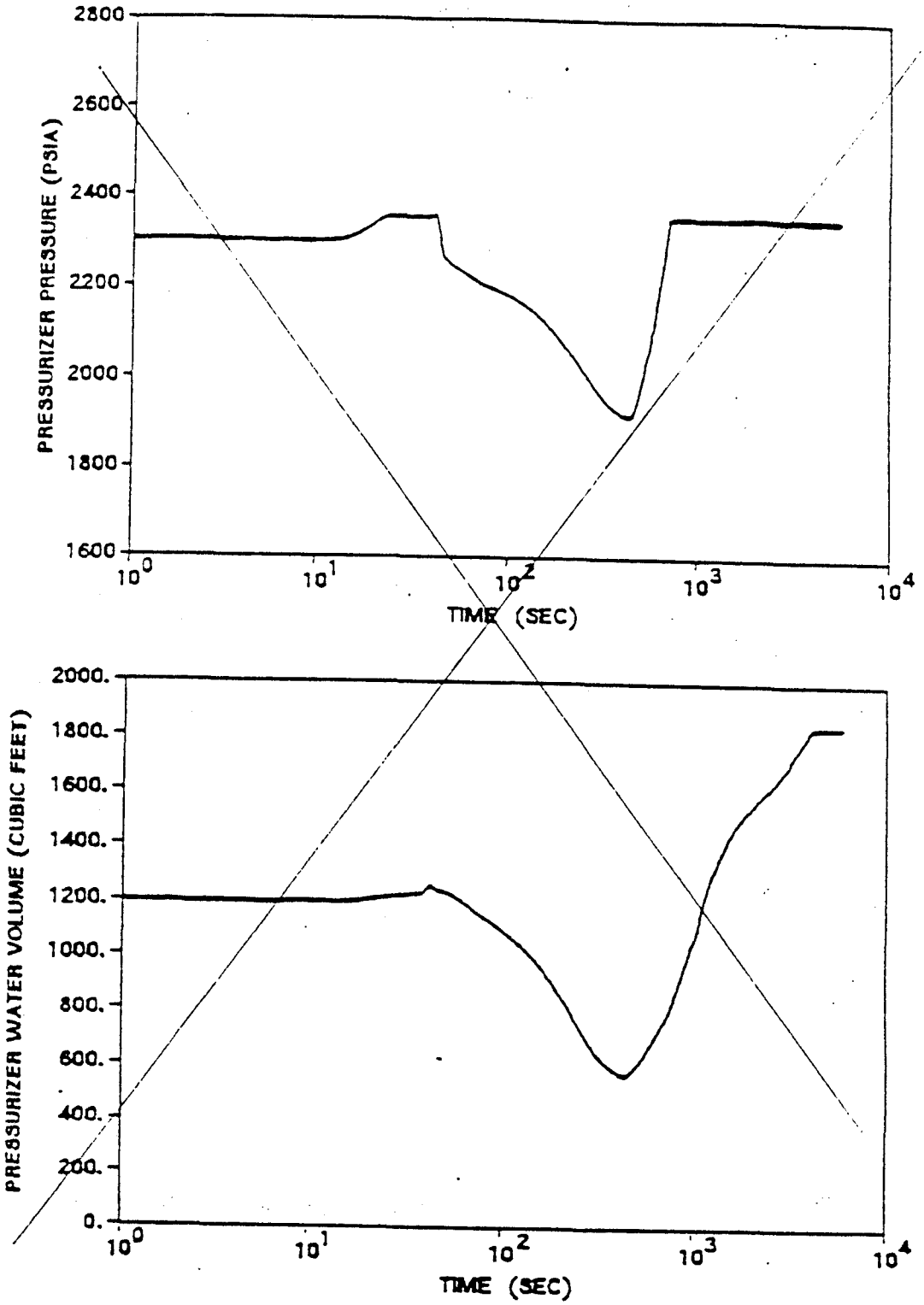
Event	Time (seconds) With Offsite Power	Time (seconds) Without Offsite Power
Feedline rupture occurs	10	10
Low-low steam generator level reactor trip and auxiliary feedwater pump start setpoint reached in affected steam generator	36	36
Rods begin to drop	38	38
Auxiliary feedwater starts to intact steam generators	96	96
Cold auxiliary feedwater reaches intact steam generators	260	260
Low steamline pressure setpoint reached	~~365~~ 366	~~437~~ 442
All main steam stop (main steam isolation) valves closed	~~370~~ 373	~~444~~ 449
Pressurizer relief valve setpoint reached	~~545~~ 22	~~240~~ 22
Pressurizer water relief begins	~~1710~~ 1734	~~4900~~ 5636
Core power decreases to auxiliary feedwater removal capacity	-4600	-1800

TABLE 15.4-10

(Sheet 1 of 1)

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS

	4 Loops Operating Initially
Maximum reactor coolant system pressure (psia)	~~2642~~ 2659
Maximum clad temperature at core hot spot (°F)	1795
Zr-H_2O reaction at core hot spot (% by weight)	0.3%



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WATTS BAR NUCLEAR PLANT
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ANALYSIS REPORT

PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS
FOR MAIN FEEDLINE RUPTURE
WITH OFFSITE POWER
FIGURE 15.4-14A

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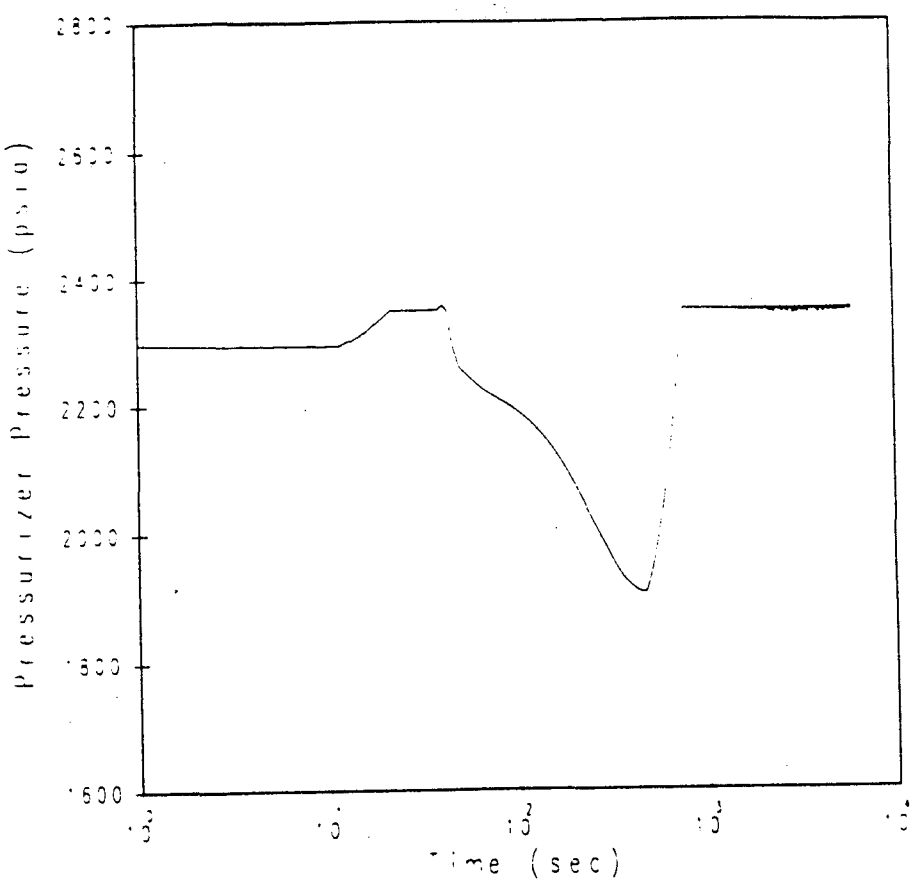

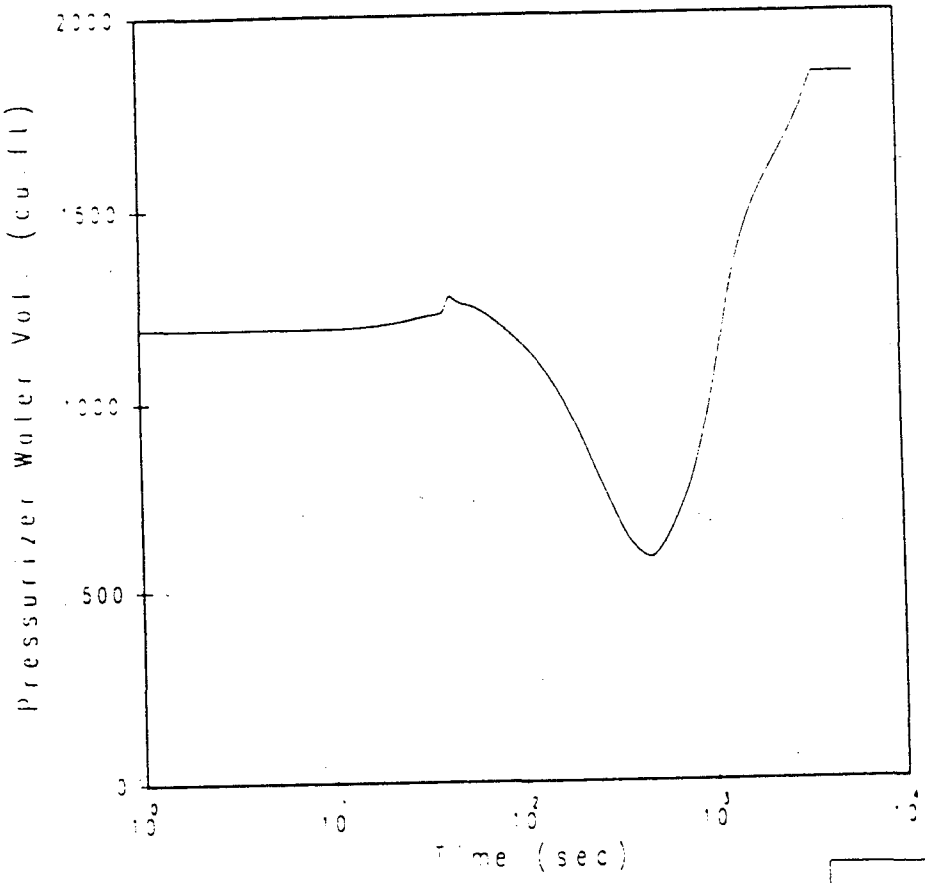

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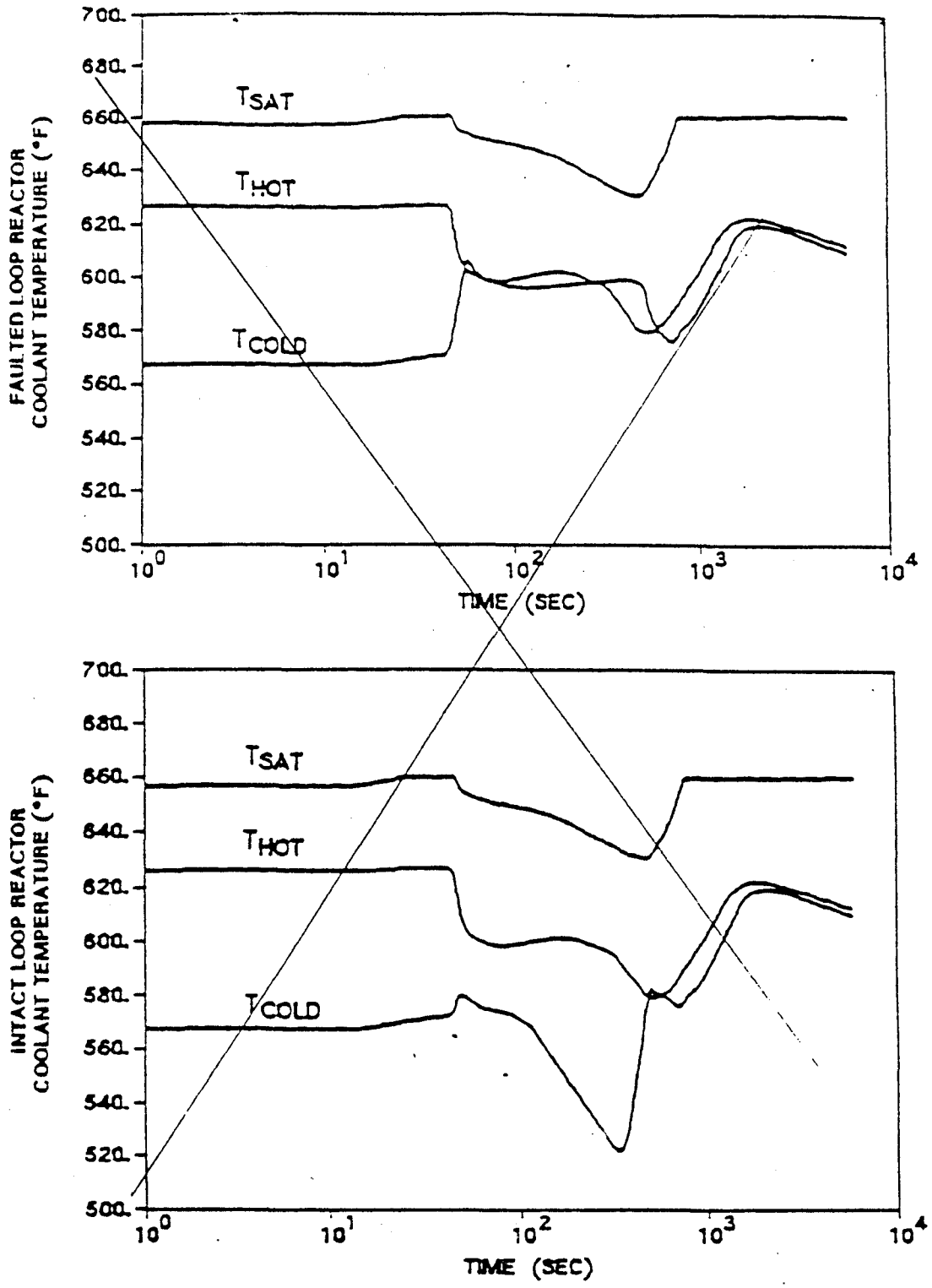
Pressurizer Pressure and Water Volume Transients for Main Feedline Rupture without Offsite Power
FIGURE 15.4-14A



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REACTOR COOLANT TEMPERATURE TRANSIENTS FOR THE FAULTED AND INTACT LOOPS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER
FIGURE 15.4-14B

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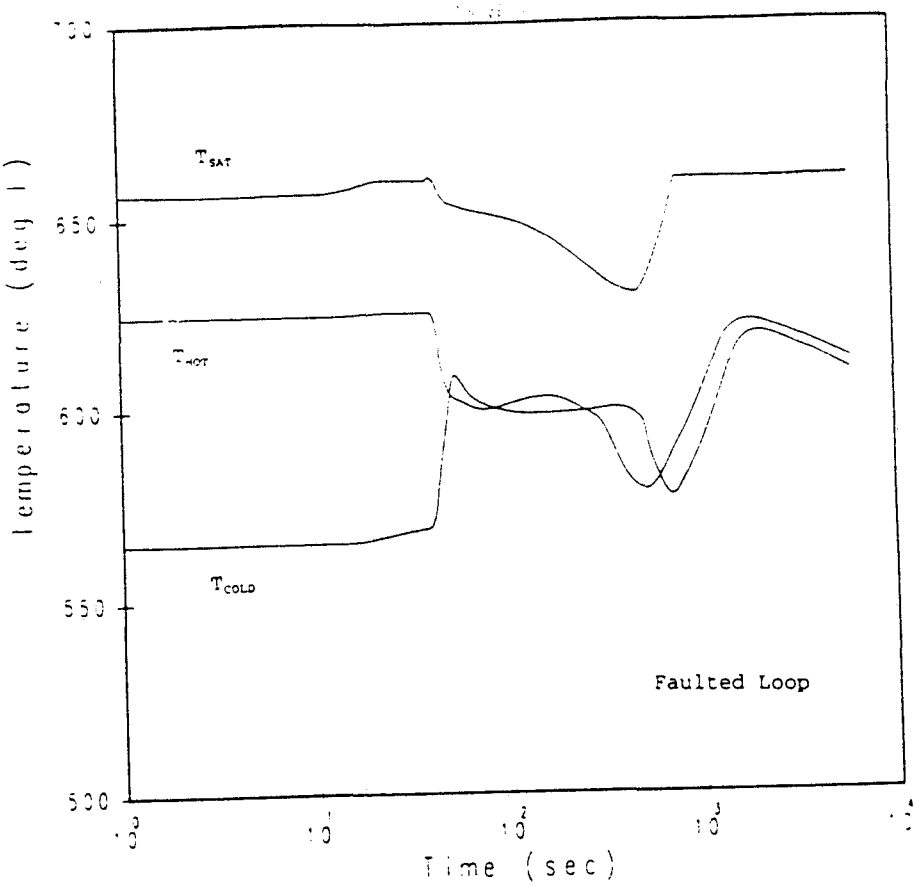

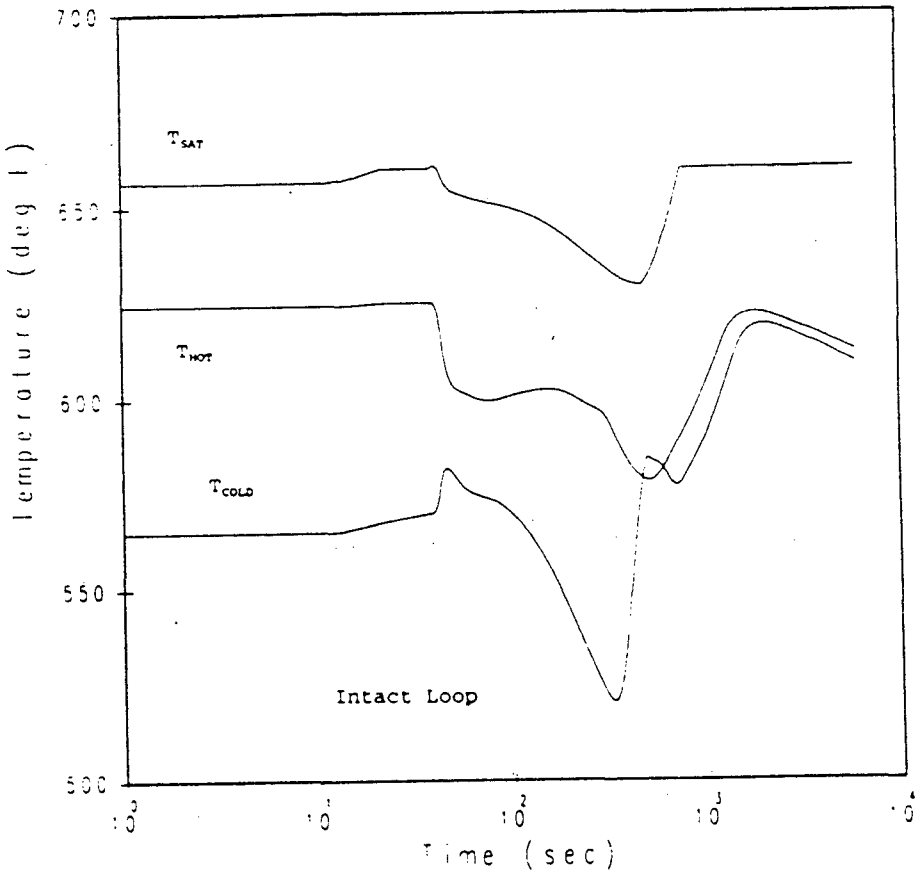

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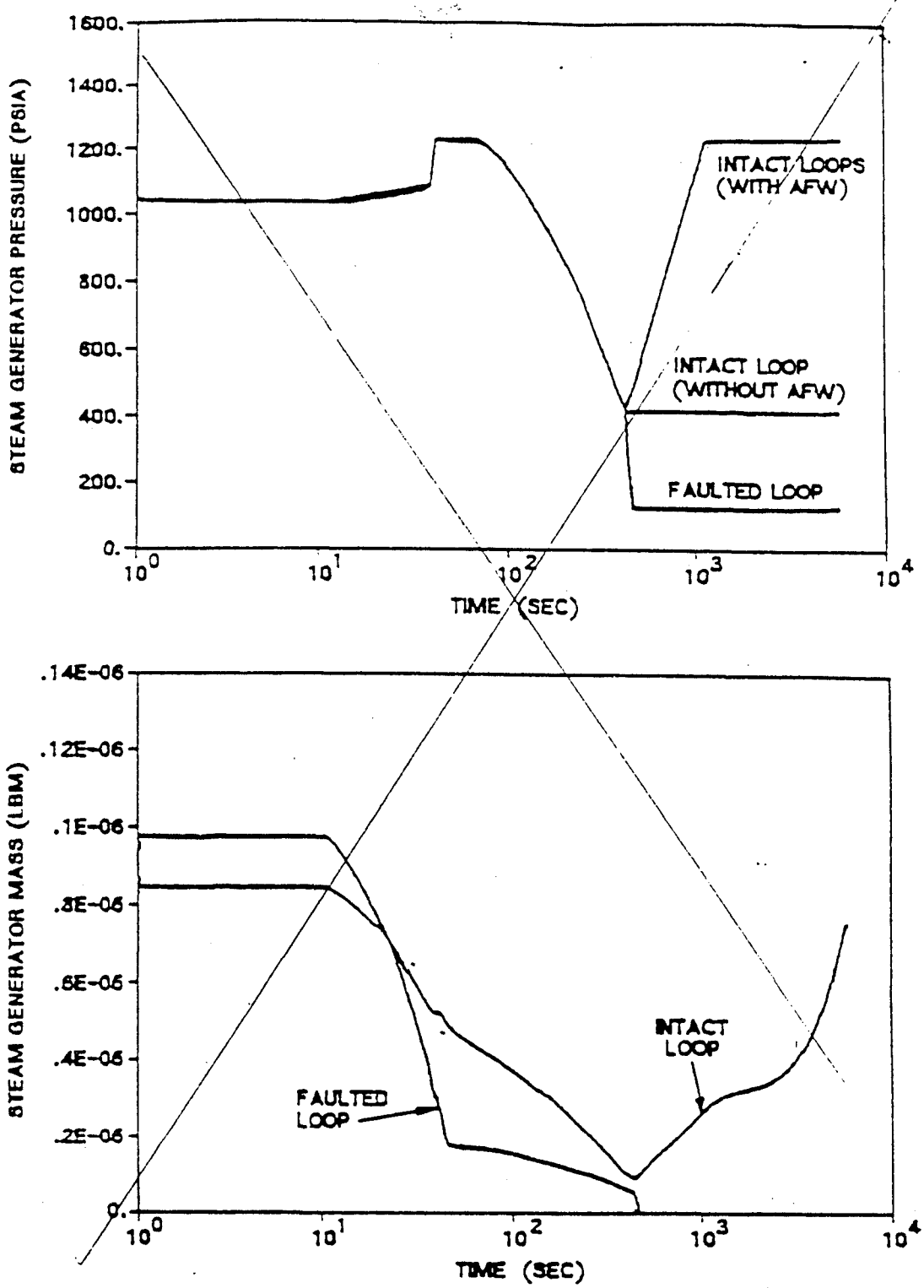
Reactor Coolant Temperature Transients for Main Feedline Rupture without Offsite Power
FIGURE 15.4-14B



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STEAM GENERATOR PRESSURE AND WATER MASS TRANSIENTS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER
FIGURE 15.4-14C

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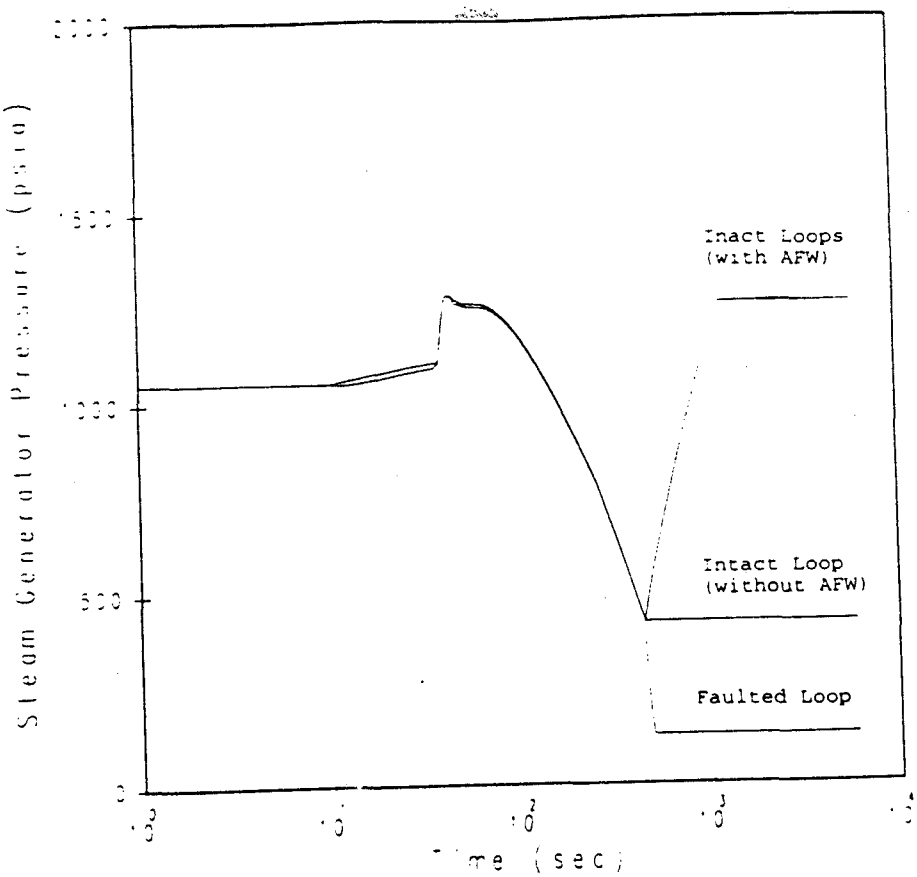

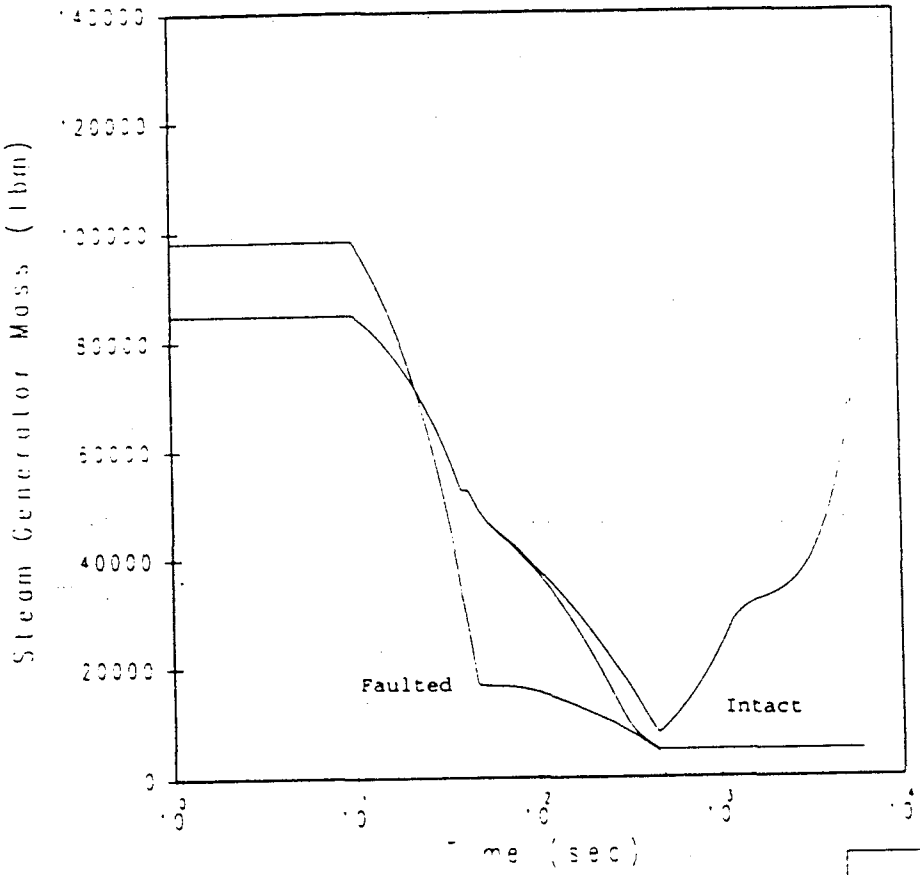

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Steam Generator Pressure and Mass Transients for Main Feedline Rupture without Offsite Power
FIGURE 15.4-14C

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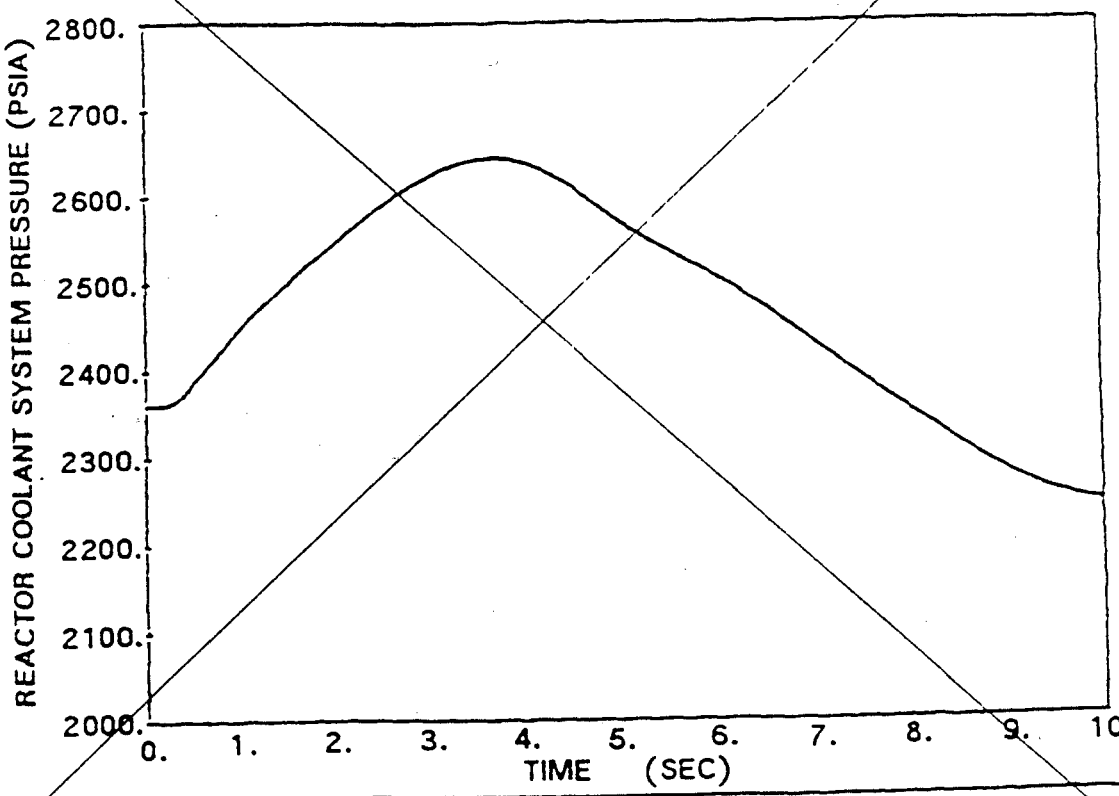

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RCS PRESSURE TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR
FIGURE 15.4-15

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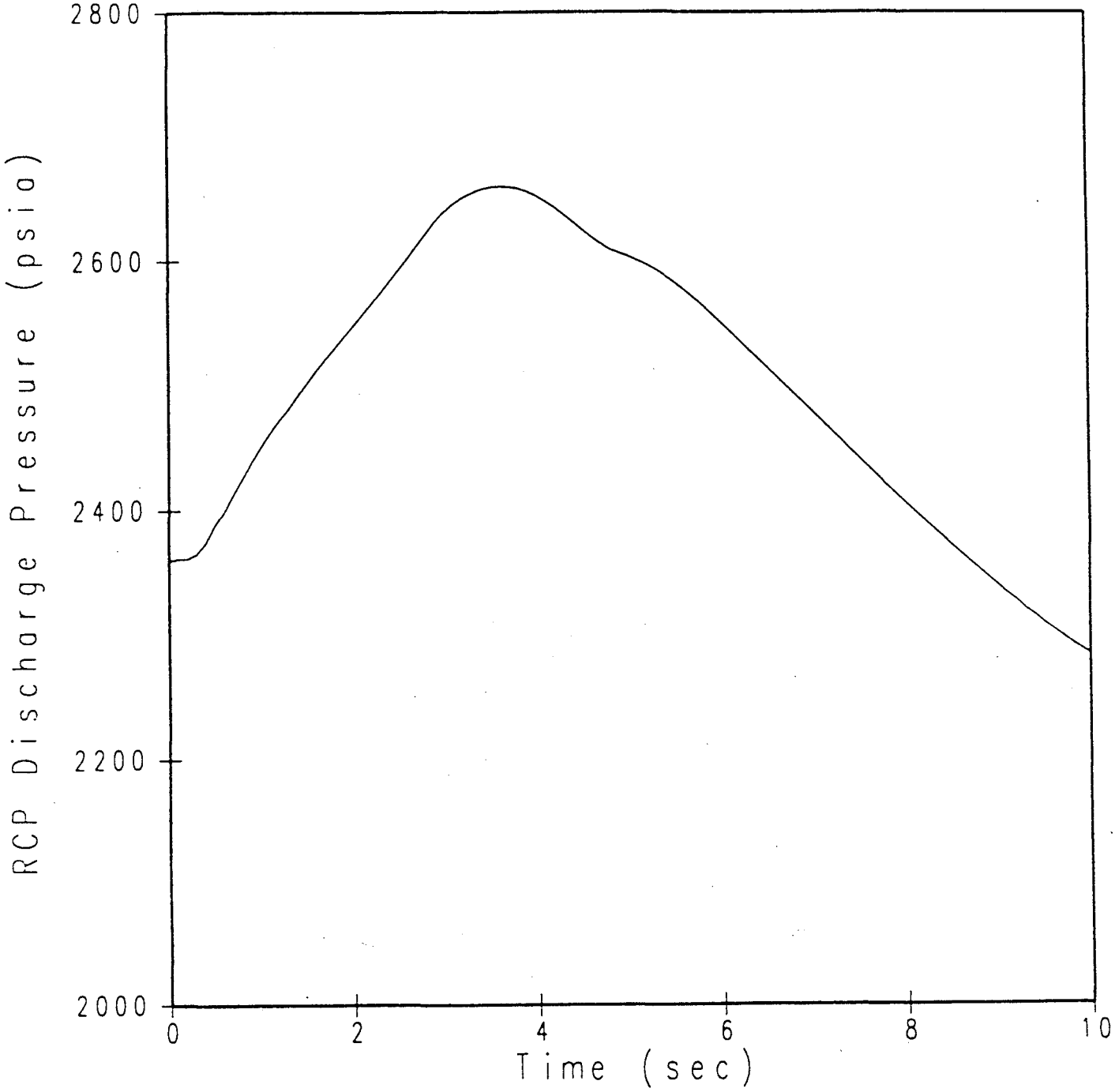

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RCS Pressure Transient
Four Pumps in Operation
One Locked Rotor
FIGURE 15.4-15

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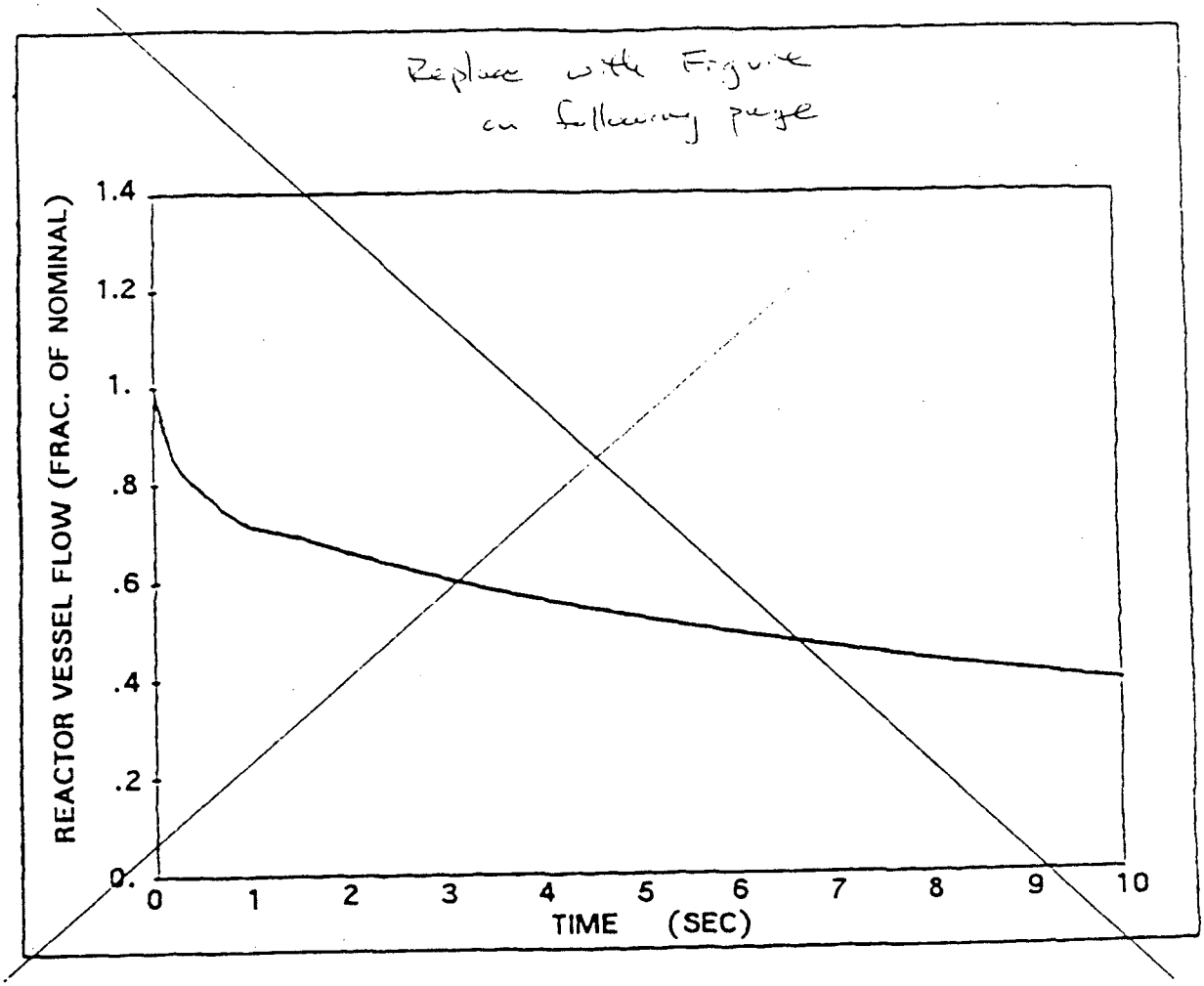

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ANALYSIS REPORT

REACTOR VESSEL FLOW TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR
FIGURE 15.4-17

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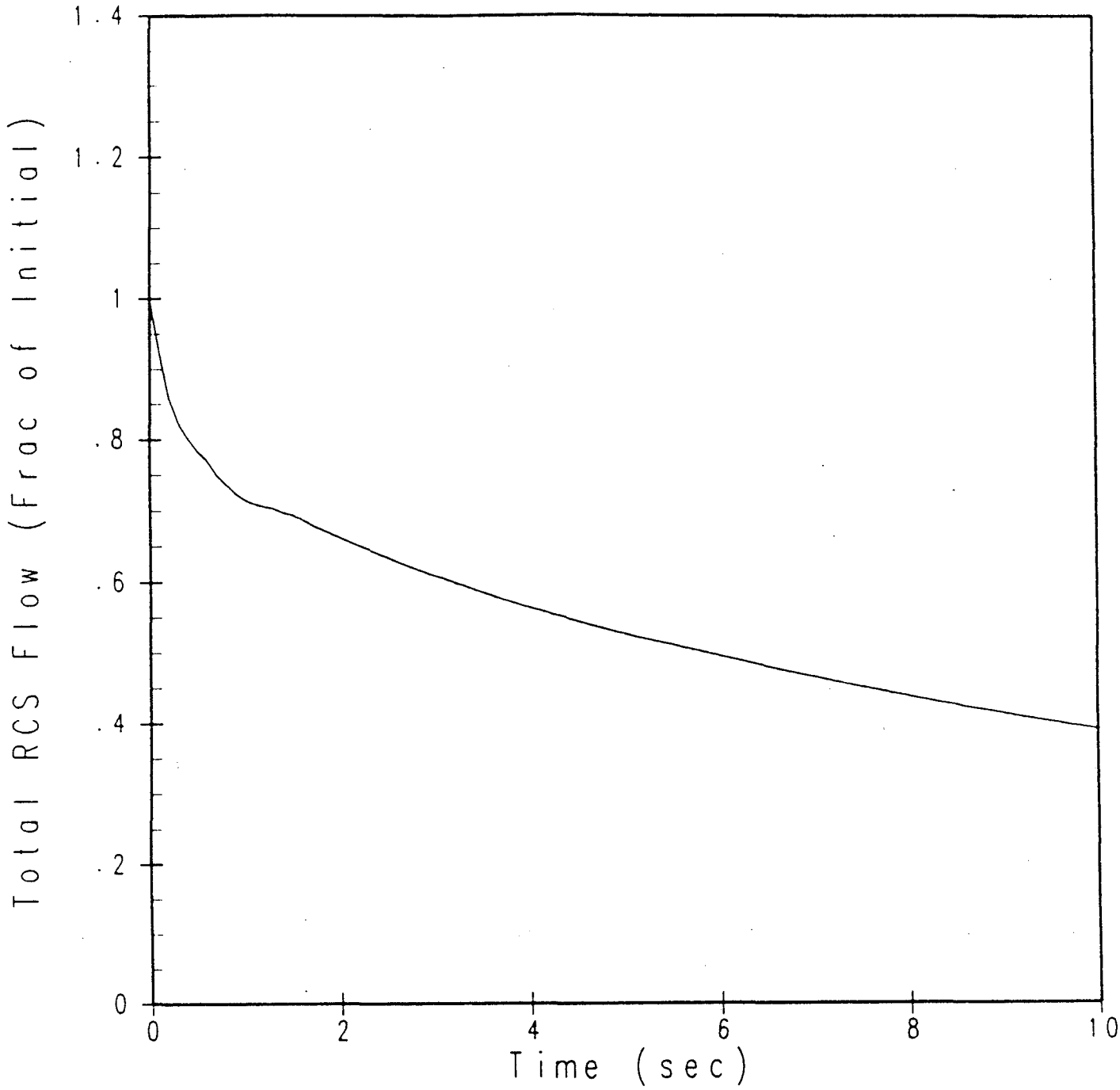

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Reactor Vessel Flow Transient
Four Pumps in Operation
One Locked Rotor
FIGURE 15.4-17

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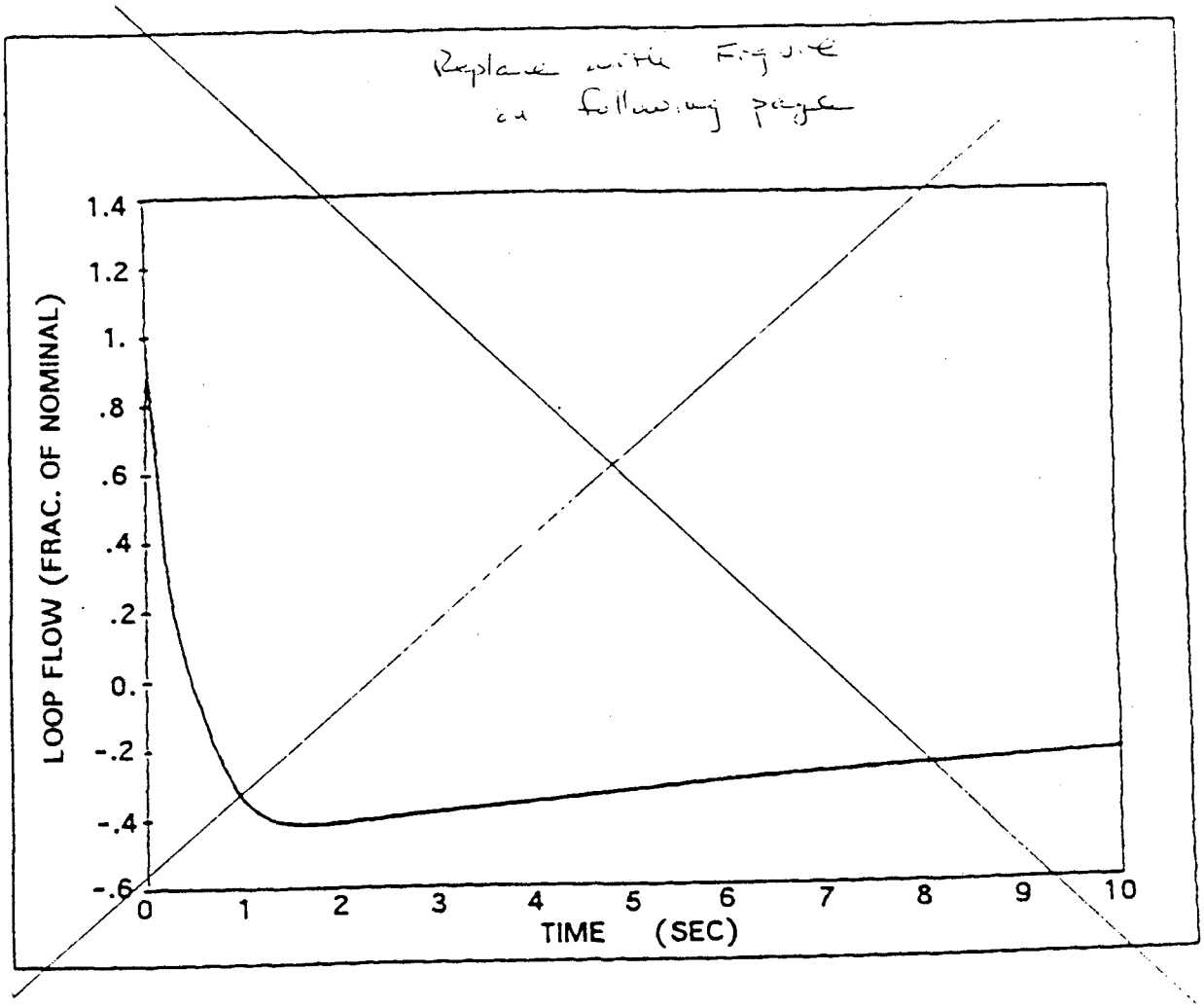

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LOOP FLOW TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR
FIGURE 15.4-18

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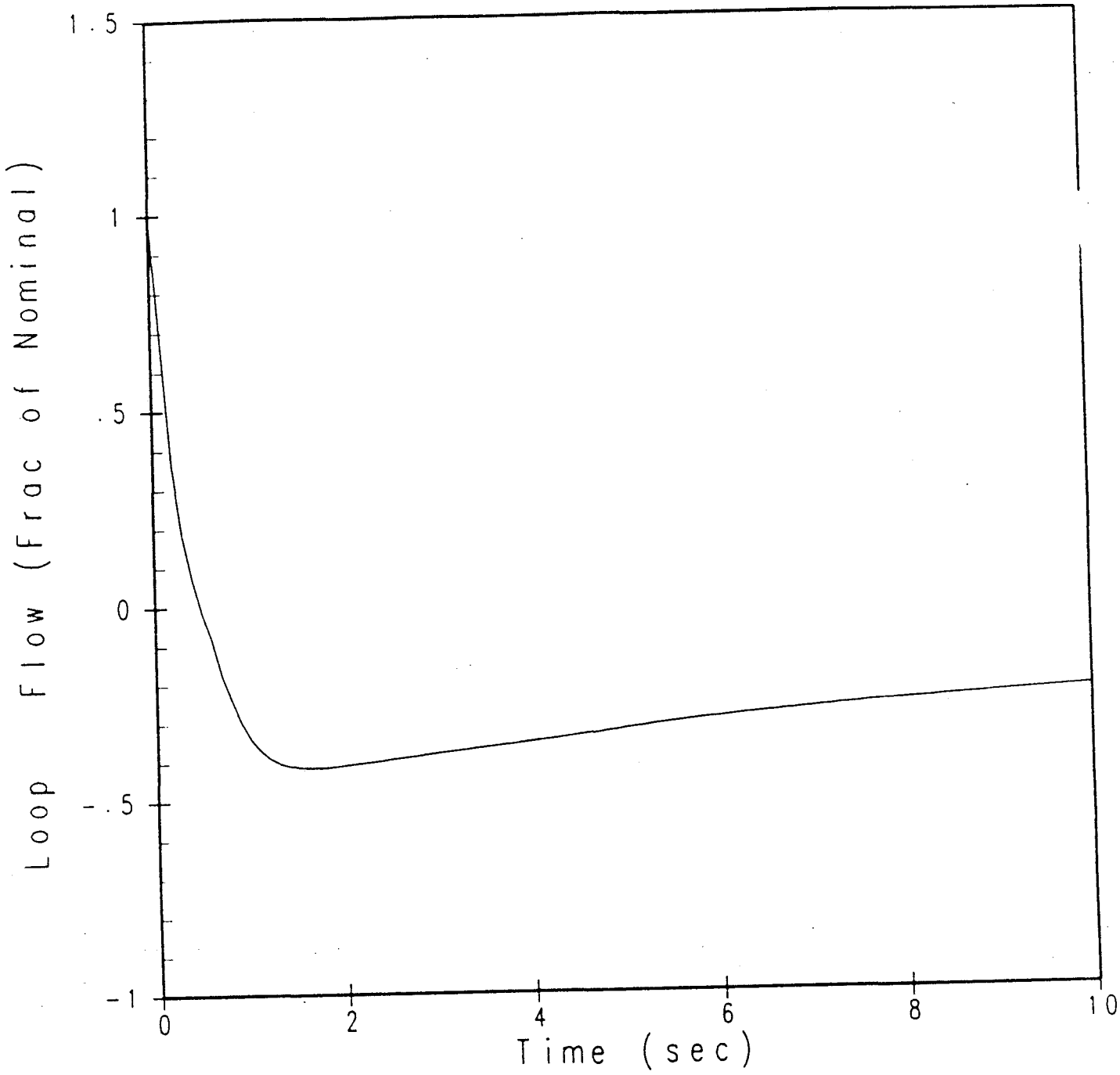

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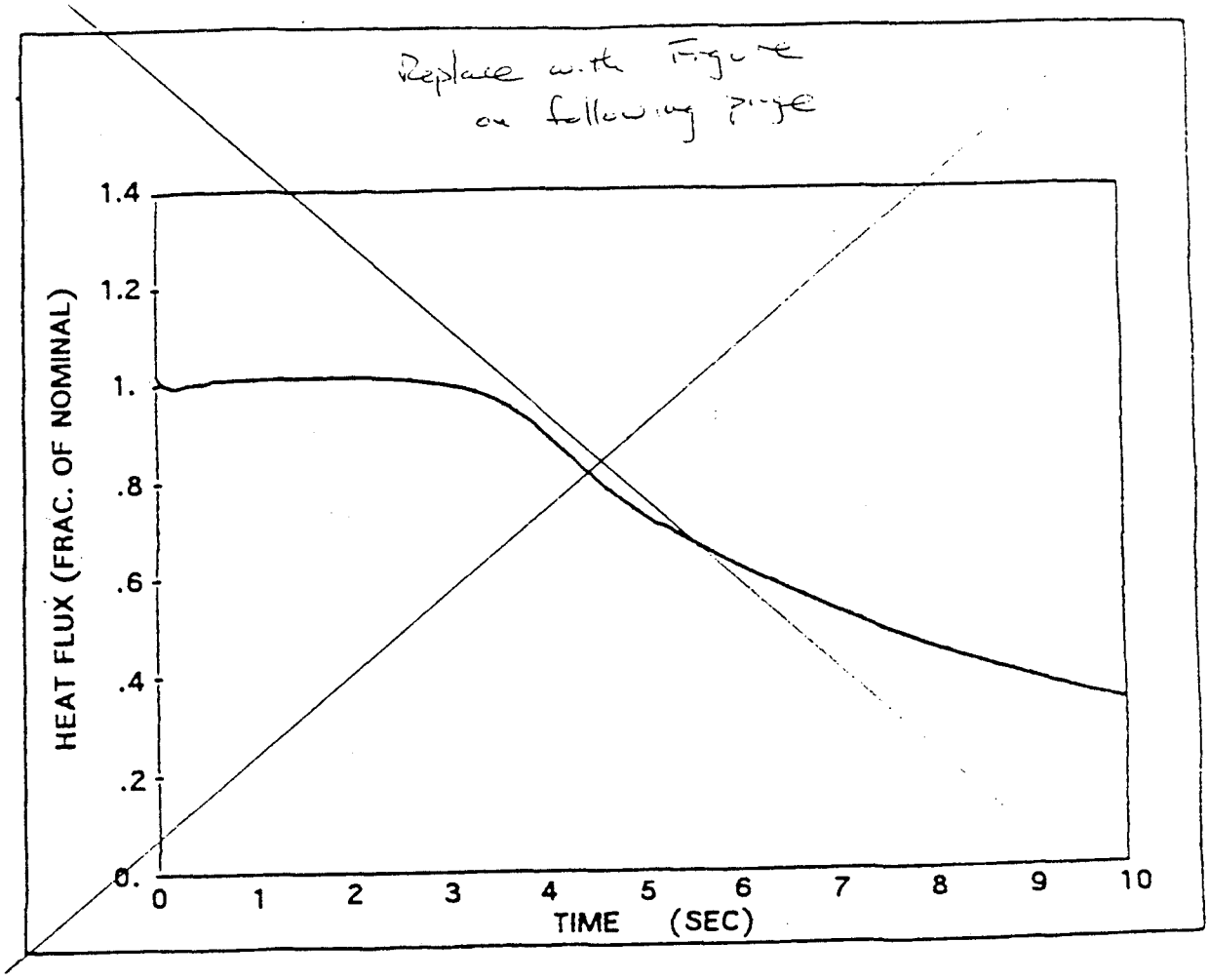
Faulted Loop Flow Transient
Four Pumps in Operation
One Locked Rotor
FIGURE 15.4-18



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HEAT FLUX TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR

FIGURE 15.4-19

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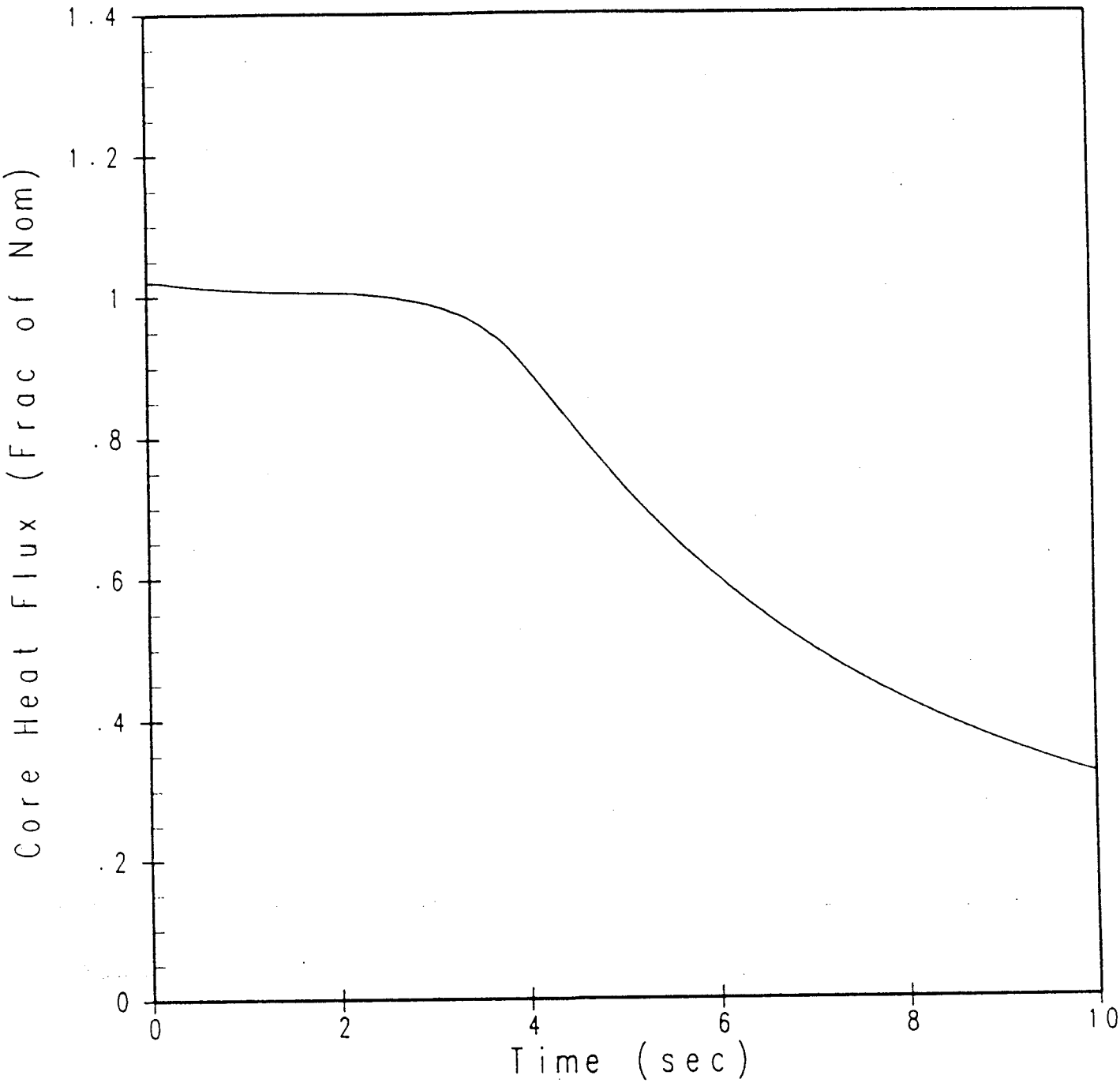

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FINAL SAFETY ANALYSIS REPORT

Heat Flux Transient
Four Pumps in Operation
One Locked Rotor
FIGURE 15.4-19

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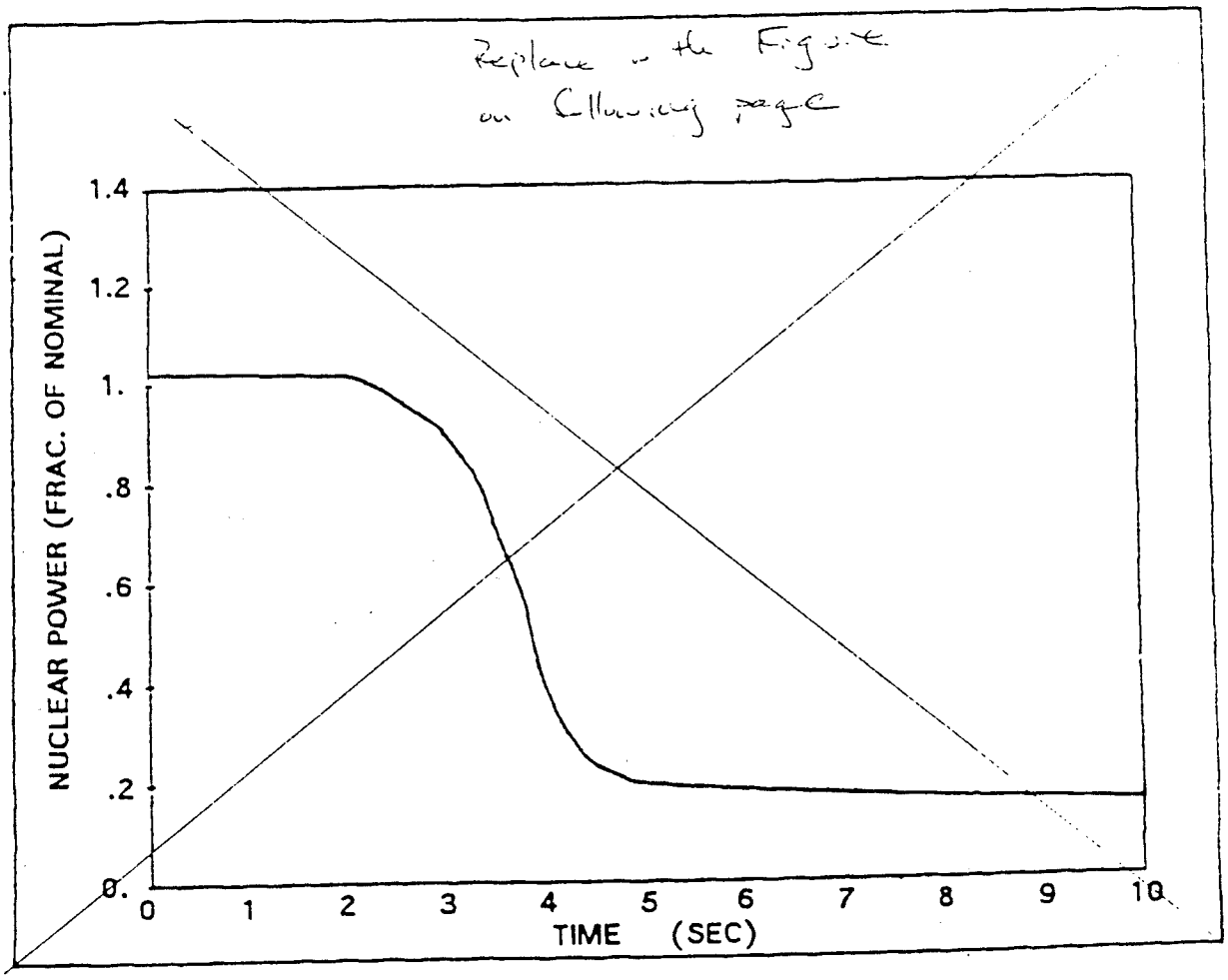

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NUCLEAR POWER TRANSIENT
FOUR PUMPS IN OPERATION
ONE LOCKED ROTOR
FIGURE 15.4-20

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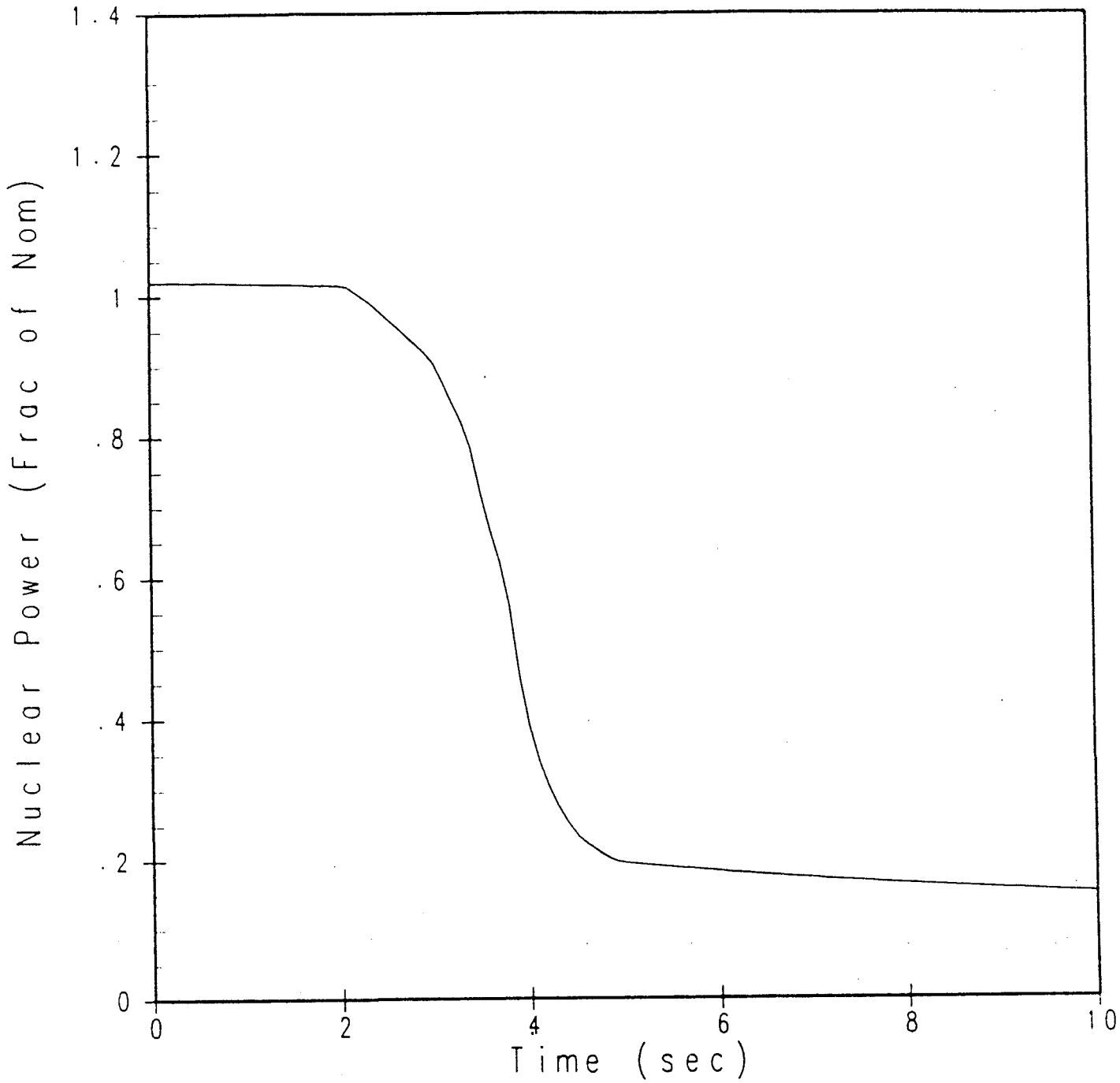

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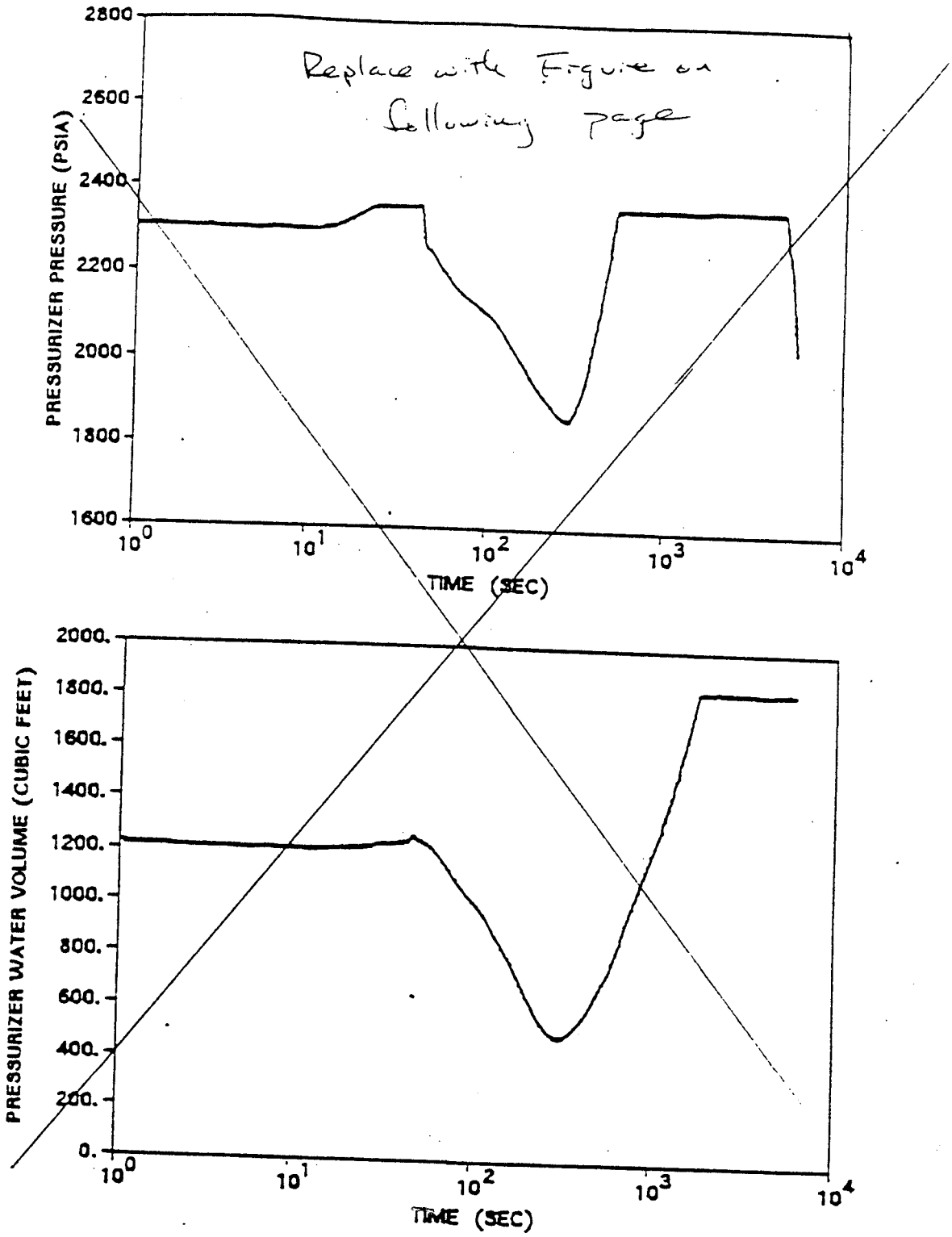
FINAL SAFETY ANALYSIS REPORT

Nuclear Power Transient
Four Pumps in Operation
One Locked Rotor
FIGURE 15.4-20



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WATTS BAR NUCLEAR PLANT
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ANALYSIS REPORT

PRESSURIZER PRESSURE AND
WATER VOLUME TRANSIENTS
FOR MAIN FEEDLINE RUPTURE
WITH OFFSITE POWER
FIGURE 15.4-13A

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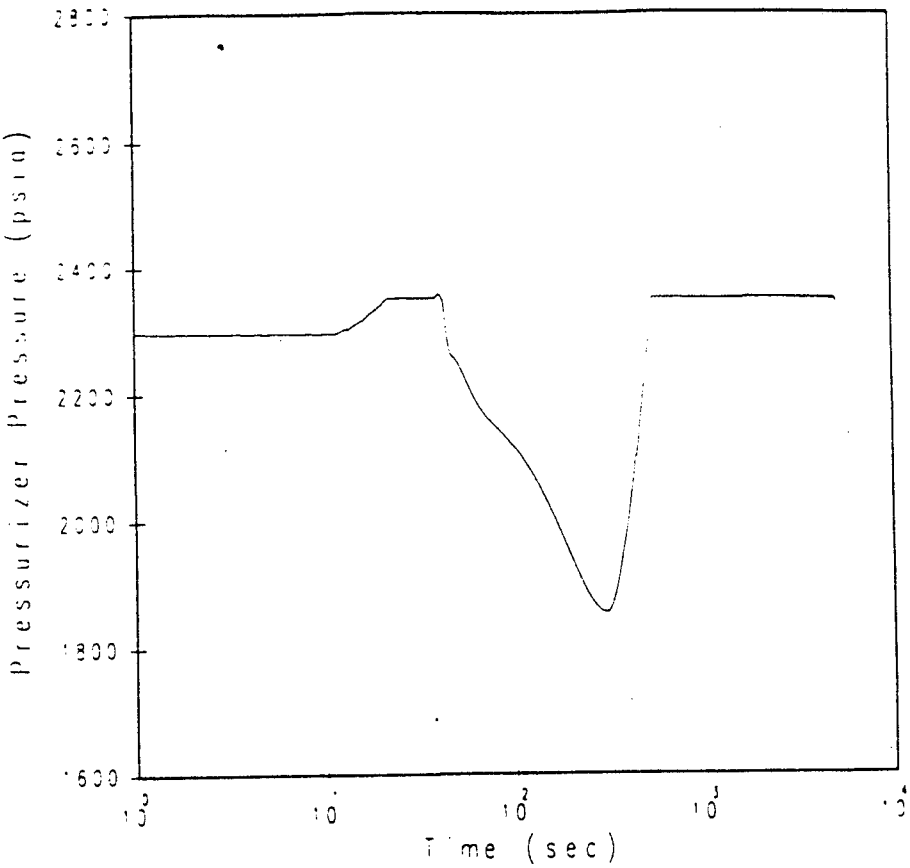
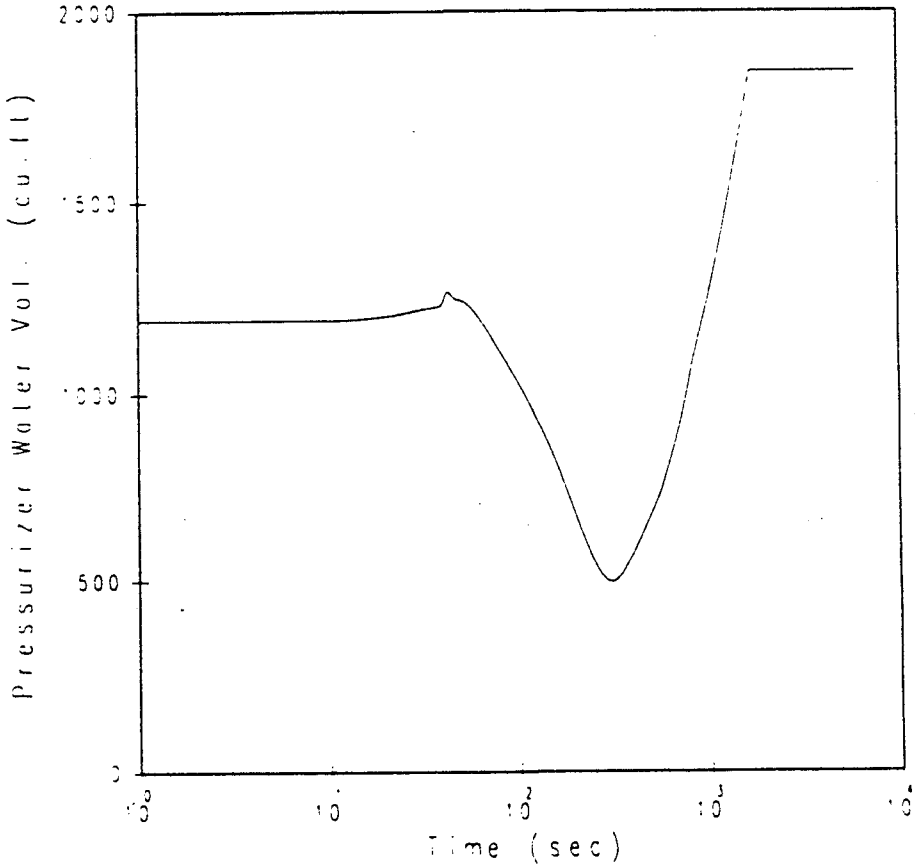
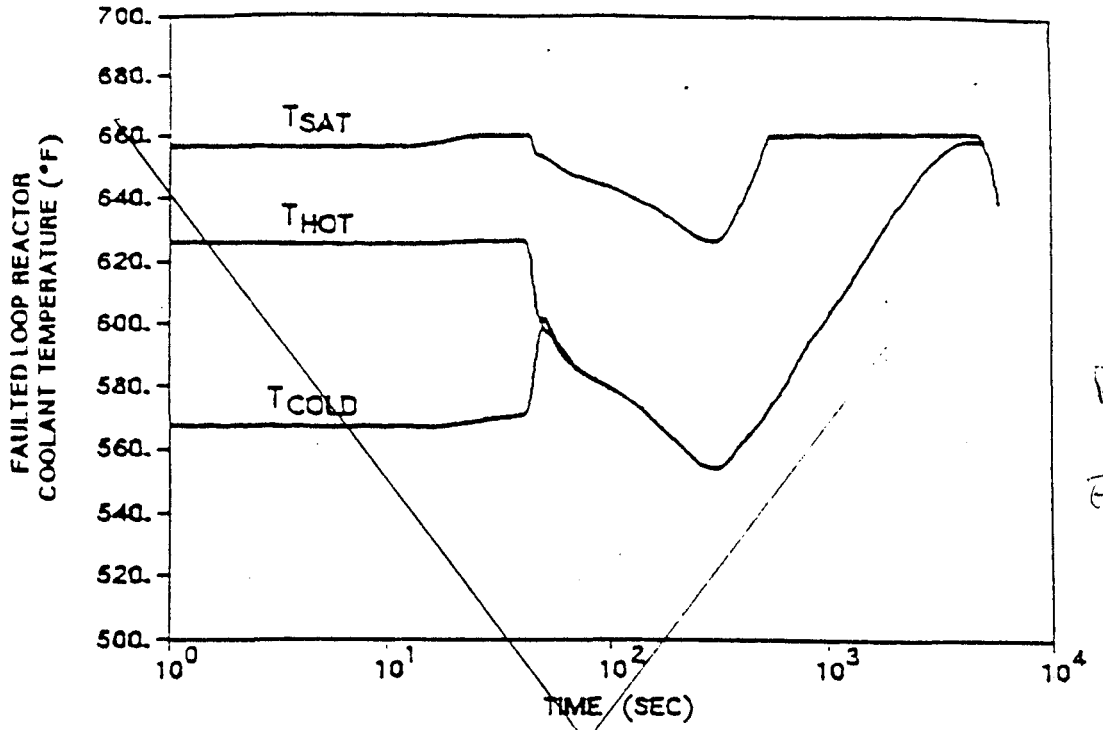




WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Pressurizer Pressure and Water Volume Transients for Main Feedline Rupture with Offsite Power
FIGURE 15.4-13A



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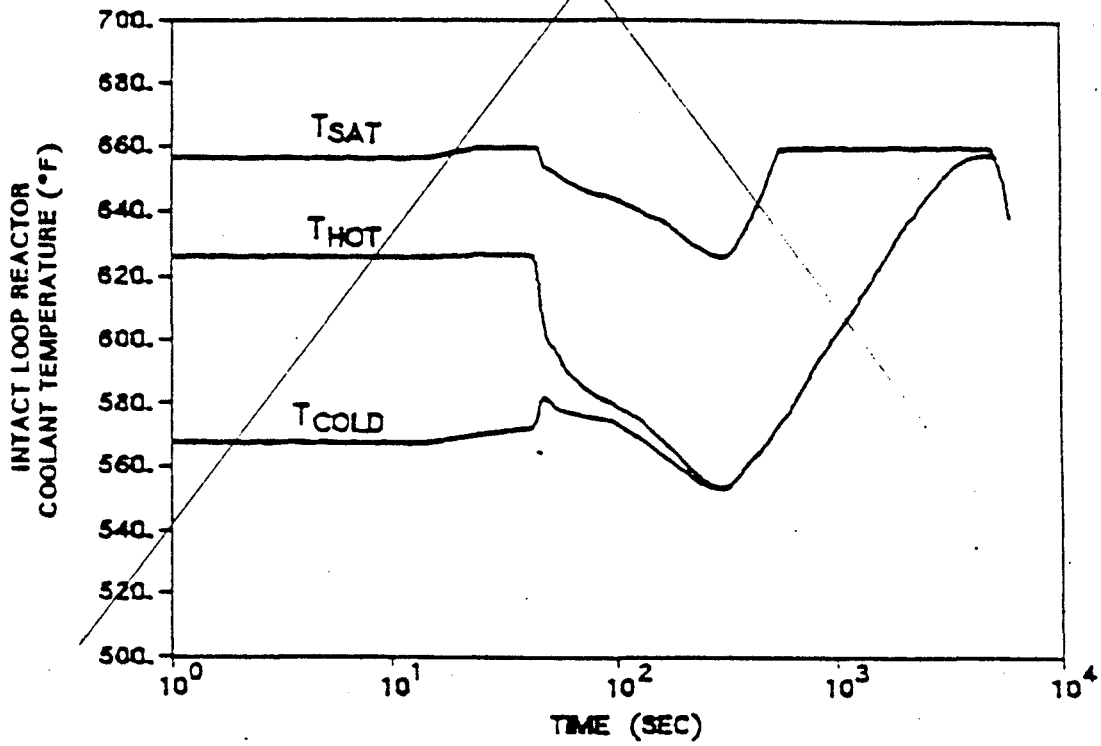

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WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

REACTOR COOLANT TEMPERATURE TRANSIENTS FOR THE FAULTED AND INTACT LOOPS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER
FIGURE 15.4-13B

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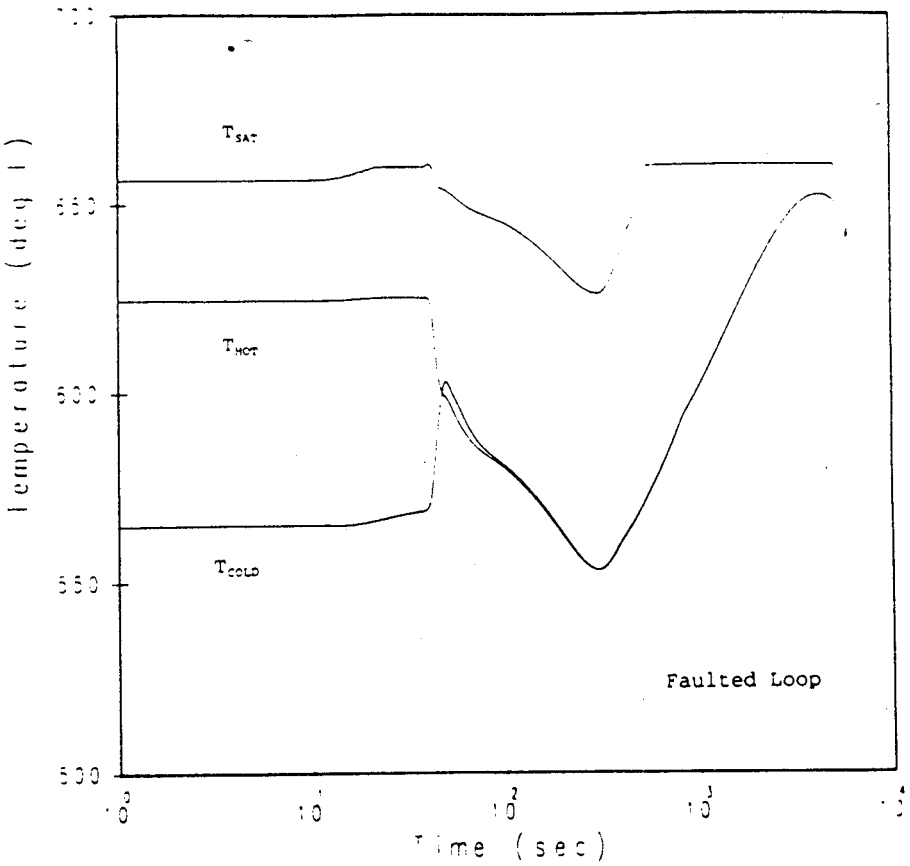
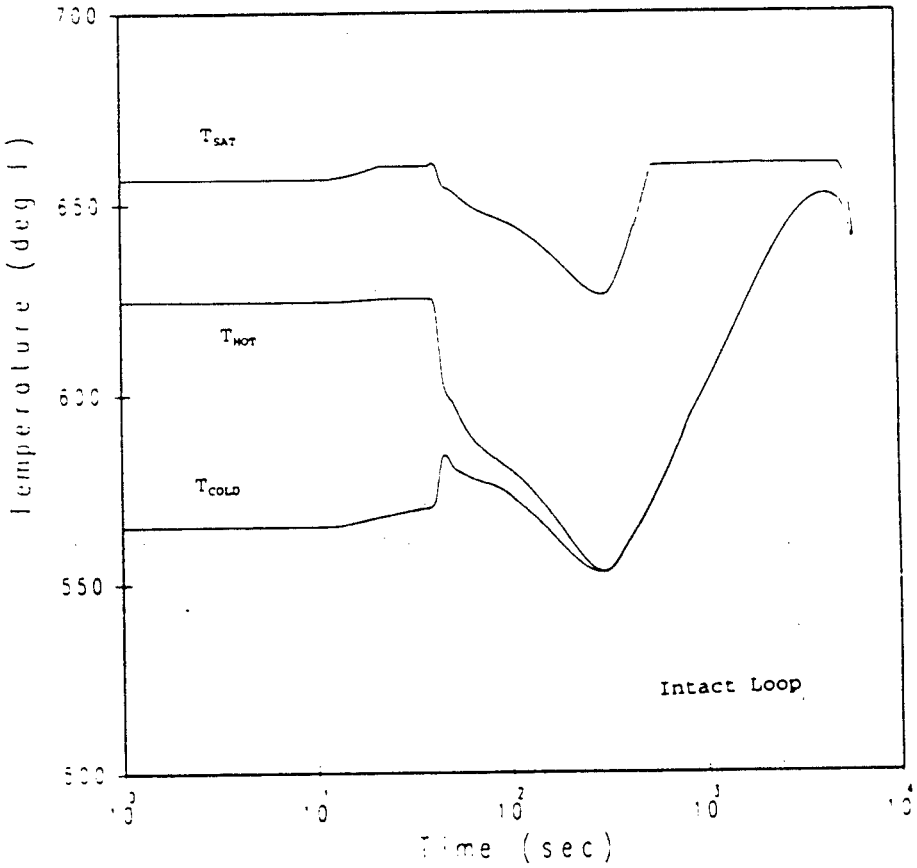
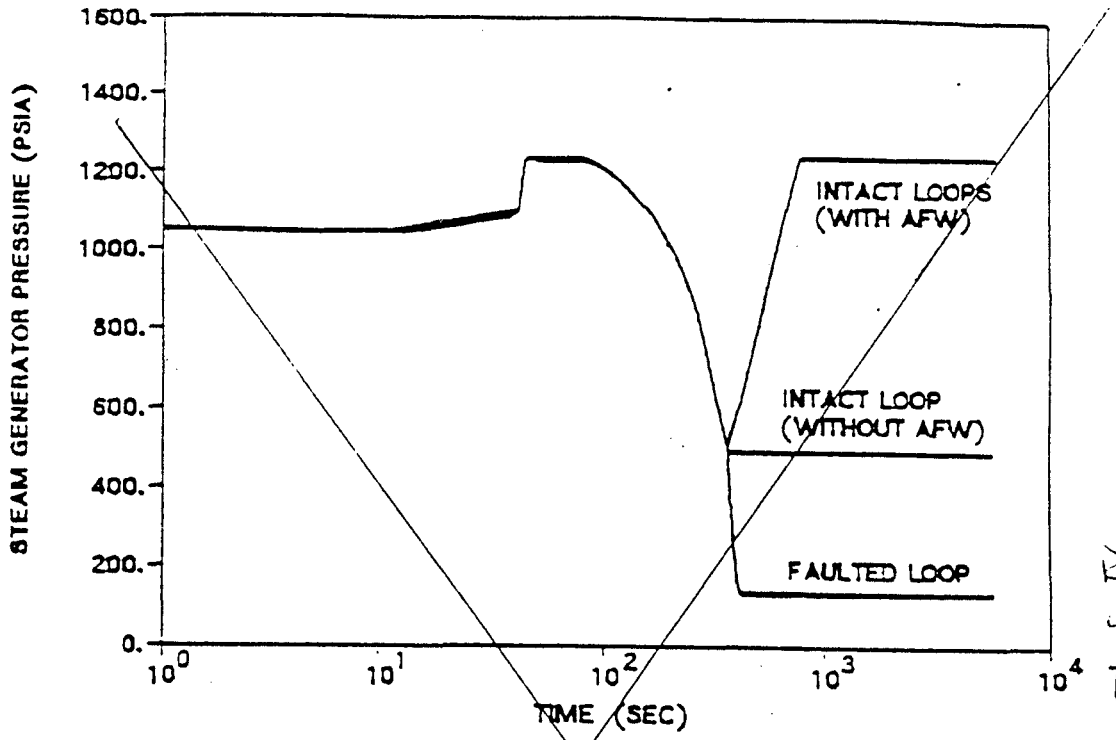




WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Reactor Coolant Temperature Transients for Main Feedline Rupture with Offsite Power
FIGURE 15.4-13B



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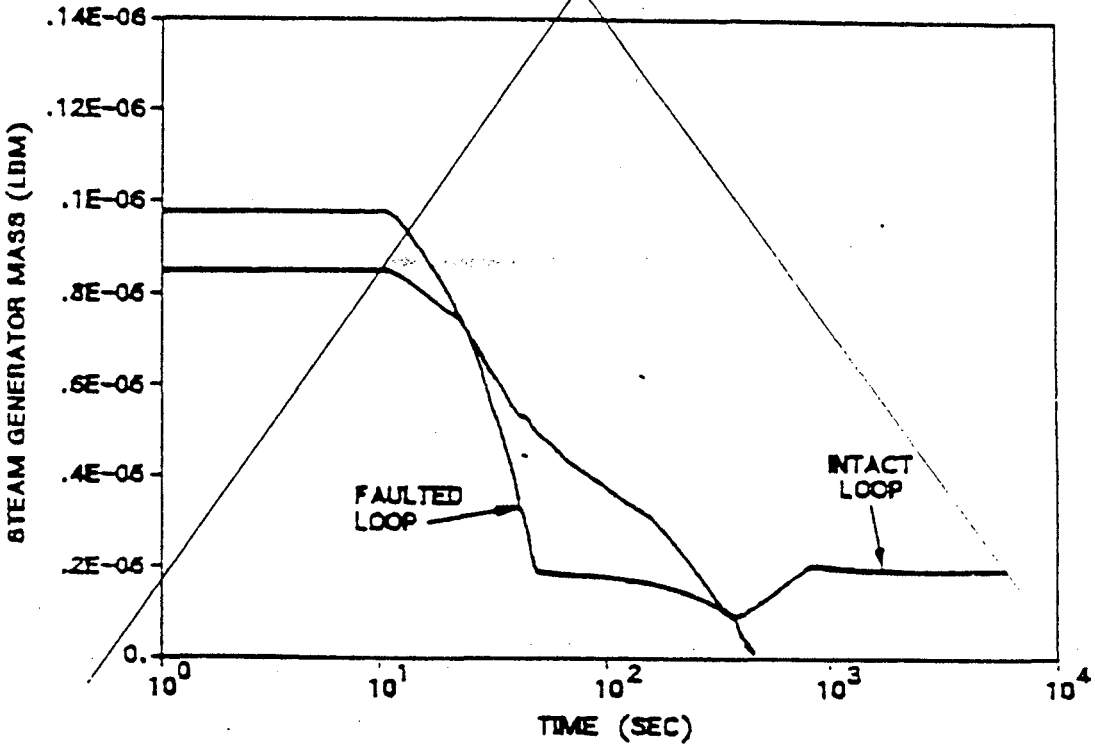

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STEAM GENERATOR PRESSURE AND WATER MASS TRANSIENTS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER
FIGURE 15.4-13C

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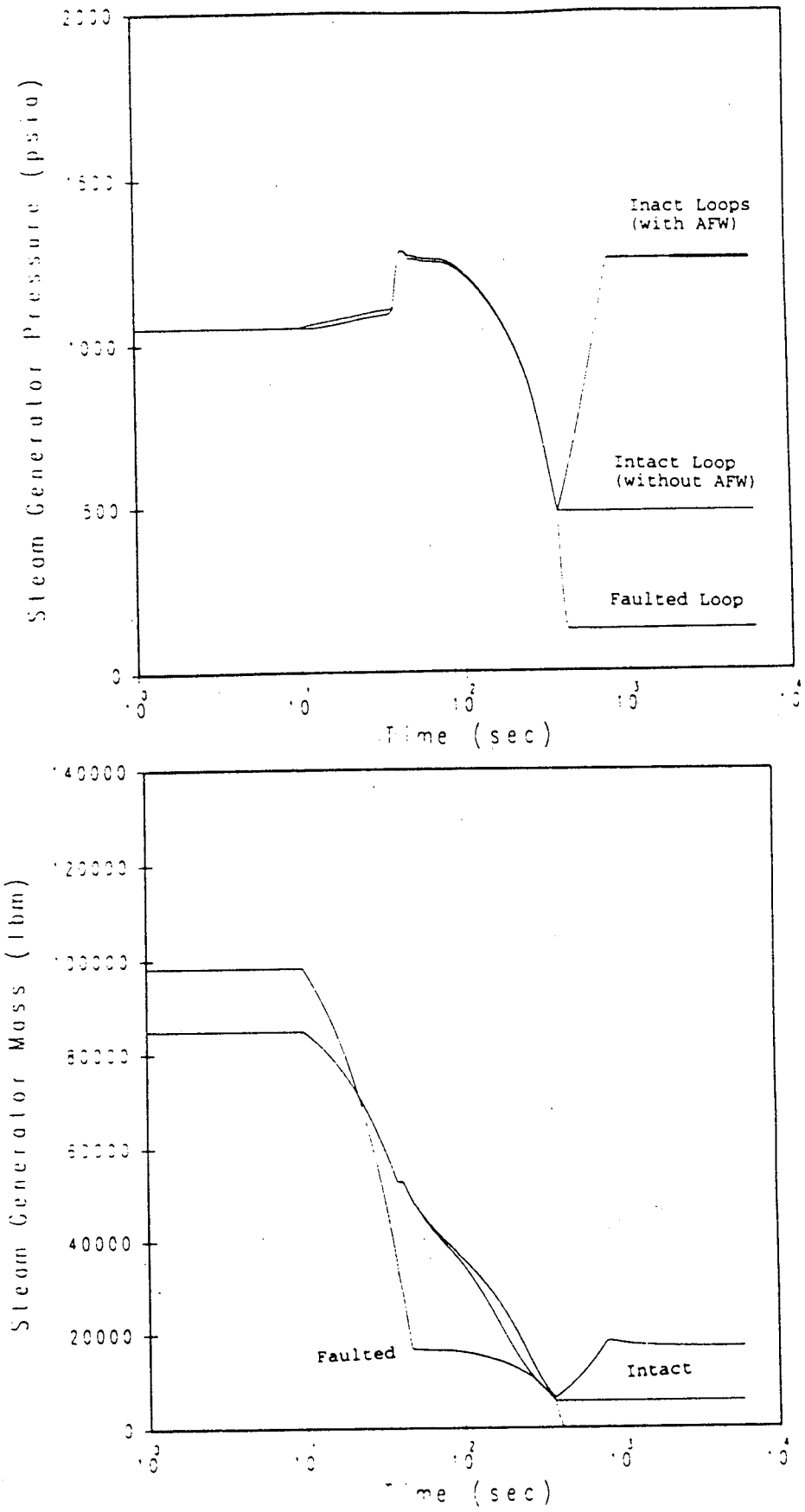

WATTS BAR NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Steam Generator Pressure and Mass Transients for Main Feedline Rupture with Offsite Power
FIGURE 15.4-13C