TU ELECTRIC

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Executive Vice President

April 6, 1989

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
INFORMATION TO SUPPORT TECHNICAL SPECIFICATION
CHANGE FOR LOW-LOW STEAM GENERATOR WATER LEVEL

Gentlemen:

During the Technical Specification (T. S.) review meetings that took place from February 13 through 24, 1989, TU Electric agreed to submit a safety analysis which demonstrates the acceptability of a low-low steam generator water level safety analysis limit of 0% of the narrow range instrument span. This analysis is used in the development of the setpoints identified as Functional Unit 6.b in T. S. Table 3.3-3, sheet 4 and Functional Unit 13 in T. S. Table 2.2-1. In accordance with that agreement, enclosed is an advance description of a FSAR change which will be included in the upcoming FSAR Amendment 76.

If you have any questions on this material, please do not hesitate to contact me or my staff.

Sincerely,

William J Cahill

William J. Cahill, Jr.

RLA/vld
Enclosure

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

D029
1/1

ENCLOSURE TO TXX-89172

CONSISTS OF THE FOLLOWING

FSAR items:

pgs. 15.2-17, 18, 21, 23, 24
30 & 35

Table 15.0-4 (sh. 2)

Table 15.2-1 (sh. 5 to 9)

Figures 15.2-9 to 26A

Pg. 032-131

Pg. 212-139

plant thermal kinetics, RCS including the natural circulation, 57
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.

The assumptions used in the analysis are as follows:

1. The plant is initially operating at 102 percent of the engineered safety features (ESF) design rating.

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

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

4. Reactor trip occurs on steam generator low-low level. No credit is taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power. 57

5. 860 gpm of Auxiliary feedwater is delivered to ~~two~~ four steam generators against a steamline backpressure of 1236 psia. 73

6. Auxiliary feedwater is delivered by either: ~~the motor-driven~~ auxiliary feedwater pump or the turbine-driven auxiliary feedwater pump. 73

7. Secondary system steam relief is achieved through the steam generator safety valves. 57

i. Both of the motor-driven auxiliary feedwater pumps, or
ii. The turbine-driven auxiliary feedwater pump.

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73 8. The initial reactor coolant average temperature is 6.5°F higher than the nominal ESF value, and initial pressurizer pressure is 30 psi higher than nominal.

73 9. The lower auxiliary feedwater flow rate results in a larger amount of coolant expansion into the pressurizer. The pressurizer power operated relief valves are assumed to function normally to maintain the peak reactor coolant system pressure at or below the actuation setpoint (2350 psia) throughout the transient.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

15.2.6.2.2 Results

57 The transient response of the RCS following a loss of AC power is shown in Figures 15.2-9 and 15.2-10. The calculated sequence of events for this event is listed in Table 15.2-1.

74 The first few seconds of the transient following receipt of a reactor trip signal will closely resemble a simulation of the complete loss of flow incident (see Section 15.3.2), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core.

57 The LOFTRAN Code (3) results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

15.2.7.2 Analysis of Effects and Consequences

15.2.7.2.1 Method of Analysis

A detailed analysis using the LOFTRAN Code [3] is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator level, pressurizer water level, and reactor coolant average temperature.

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the engineered safety features (ESF) design rating.

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

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

4. The worst single failure in the Auxiliary Feedwater System occurs. 57

~~5. Auxiliary feedwater is delivered to four steam generators.~~ 57

5. 860 gpm of auxiliary feedwater is delivered to four steam generators against a steamline backpressure of 1236 psia.

An additional assumption made for the loss of normal feedwater evaluation is that only the pressurizer safety valves are assumed to function normally. Operation of the valves maintains peak RCS pressure at or below the actuation setpoint (2500 pounds per square inch absolute (psia)) throughout the transient. 57

Since the two Comanche Peak units will have different steam generators (see Section 5.4.2), the effect of this difference has been considered in the analysis. Both types of steam generators are integral preheater models. The major difference, from the standpoint of accident analysis for this event, is the slightly higher secondary side mass as a function of power for the D5 (Unit 2) model. In order to maximize the time until reactor trip on low-low steam generator level occurs and to insure that the analysis is valid for both units, the initial steam generator secondary mass was assumed to be 110% of the higher D5 mass. The low-low steam generator water level trip setpoint was assumed to be the same mass (lb. mass) for both units 49 (see Table 15.0-4). Note that while a higher secondary mass (larger heat sink) is, in general, a benefit for primary side heatup events, the assumption of a higher initial mass results in a delay of the trip signal, and thus produces a more severe transient. 5

In addition, all steam generators for both units will be equipped with separate feedwater connections for injection of auxiliary feedwater and main feedwater at low power operation. The major effect of injecting auxiliary feedwater into the upper section of the downcomer is that most of the flow will bypass the preheat region due to the higher resistance to flow in the preheater. This will result in a slight decrease in heat removal capability. However, the auxiliary feedwater injection point is now much closer to the steam generator, resulting in a much smaller volume of hot feedwater which must be purged before the colder auxiliary feed enters the units. 5

Further, since the bottom of the narrow range span in the Unit 1 (D4) steam generator is lower than in the Unit 2 (D5) steam generator, the Unit 1 low-low steam generator water level trip setpoint is used to provide a bounding analysis for both units.

Plant characteristics and initial conditions are further discussed in Section 15.0.3. Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. The Reactor Protection System is required to function following a loss of normal feedwater as analyzed here. The Auxiliary Feedwater System is
57 required to deliver a minimum auxiliary feedwater flow rate. The auxiliary feedwater flow rate assumed for the Loss of Normal Feedwater analysis is ~~600~~ 860 gal/min against a streamline backpressure of 1236 psia. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference [2].

Results

Figures 15.2-11 and 15.2-12 show the significant plant parameter transients following a loss of normal feedwater.

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

73 The auxiliary feedwater flow rate for this event is higher than that for the loss of nonemergency AC power event (section 15.2.6) due to the additional heat input to the coolant from the reactor coolant pumps.

8. A conservative feedline break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator. This minimizes the heat removal capability of the affected steam generator.

57 9. Reactor trip occurs on steam generator low-low level.

Therefore, the failure of the

, against a steamline backpressure of 1236 psia,

10. The Auxiliary Feedwater System is actuated by the low-low steam
73 generator water level signal. The Auxiliary Feedwater System is assumed to supply a total of 430 gallons per minute (gpm) between two unaffected steam generators from a motor-driven pump. ~~(Note~~
57 ~~that a~~ turbine-driven pump, which ~~is~~ is capable of supplying 430 gpm to three intact steam generators, ~~rather than to two intact steam generators.)~~ is considered to be the worst single active failure. A 60 second delay ~~was~~ is assumed following the low-low level signal to allow time for startup of the emergency diesel generators and the auxiliary feedwater pumps.

73 Approximately 106 seconds of main feedwater before ~~was assumed before~~ are required to purge the feedwater lines ~~were purged and the~~ relatively cold (120°F) auxiliary feedwater can ~~entered~~ enters the unaffected steam generators.

73 11. Thirty minutes after the reactor trip, an additional 265 gpm is assumed to be supplied to the third intact steam generator by operator action.

12. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

This assumption causes more severe analysis results than does the feeding of three intact steam generators.

RCS pressure will be maintained at the safety valve setpoint until safety injection flow is terminated by the operator or until AFW flow is increased to the intact steam generators as mentioned in Section 15.2.8.2. The reactor core remains covered with water throughout the transient. ~~and~~ water relief due to thermal expansion, is ~~prevented~~ by the heat removal capability of the Auxiliary Feedwater System, 57 71

through the pressurizer safety valves,

limited

and reactor coolant makeup is provided by the Safety Injection System.

The major difference between the two cases analyzed can be seen in the plots of hot and cold leg temperatures, Figures 15.2-16 through 15.2-18 (with offsite power available) and Figures 15.2-23 through 15.2-25 (without offsite power). It is apparent from the initial portion of the transient (<300 seconds) that the case without offsite power results in higher temperatures in the hot leg. For longer times, however, the case with offsite power results in a more severe rise in temperature until the coolant pumps are turned off and the Auxiliary Feedwater System is realigned. The pressurizer fills more rapidly for the case with power due to the increased coolant expansion resulting from the pump heat addition; ~~however, no~~ water is relieved for ~~either~~ case. As previously stated, the core remains covered with water for both cases. 57

the

with offsite power available; however,

15.2.8.3 Conclusions

Results of the analyses show that for the postulated feedwater line rupture, the assumed Auxiliary Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core.

15.2.8.4 Analysis of Radiological Effects and Consequences

Radioactivity doses from the postulated feedwater line rupture are less than those previously presented for the postulated steam line break. All applicable acceptance criteria are therefore met. 57

TABLE 15.0-4

(Sheet 2)

TRIP POINTS AND TIME DELAYS TO TRIP

ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analysis	Time Delays (Seconds)	
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1.0	
Undervoltage trip	68% nominal	1.5	
Turbine trip	Not applicable	2.0	
Low-low steam generator level	34.6%* (Unit 1) and 0% (Unit 2) 0% of narrow range level span		49
High steam generator level trip of the feedwater pumps and closure of feedwater systems valves, and turbine trip	90% (Unit 1) and 81% (Unit 2) of narrow range level span	2.0	49

* The basis for the Unit 1 limiting setpoint is the Loss of Normal Feedwater analysis. The setpoint used in the Feedline Break analysis was assumed to be 15% 73

TABLE 15.2-1

(Sheet 5 of 9)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)	
Loss of non-emergency AC power	Main feedwater flow stops	10.0	
	Low steam generator water level trip	~~55.4~~ 67.8	
	Rods begin to drop	~~57.4~~ 69.8	
	Reactor coolant pumps begin to coastdown	~~59.4~~ 71.8	
	Peak water level in pressurizer occurs	~~61.0~~ 72.0	73
	Four ~~Two~~ steam generators begin to receive 860 ~~430~~ gpm from auxiliary feedwater system	~~115.4~~ 127.8	73
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~~~1700~~ ~335	73

TABLE 15.2-1

(Sheet 6 of 9)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
Loss of normal feedwater flow	Main feedwater flow stops	10.0
	Low steam generator water level trip	~~55.4~~ 67.8
	Rods begin to drop	~~57.4~~ 69.8
	Peak water level in pressurizer occurs	~~61.0~~ 72.0
	Four steam generators begin to receive ~~600~~ 860 gpm from auxiliary feedwater system (see Note 2)	~~115.4~~ 127.8 | 73
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~~~1800~~ ~320

TABLE 15.2-1

(Sheet 7 of 9)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)	
Feedwater system pipe break			
1. With offsite power available	Main feedline rupture occurs	10	
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	~~32.7~~ 34.2	
	Rods begin to drop	~~34.7~~ 36.2	
	Pressurizer safety valve setpoint reached	~~38~~ 39.5	
	Steam generator safety valve setpoint reached in intact steam generators	~~38~~ 39.5	
	Auxiliary feedwater is delivered to two intact steam generators	~~92.7~~ 94.2	73 73
	Low steam line pressure setpoint reached in ruptured steam generator	~~368.9~~ 350.7	73
	All main steam line isolation valves close	~~375.9~~ 357.7	73
	Pressurizer water relief begins	1760	

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)	
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~~~1870~~ ~1856	
2. Without offsite power	Main feedline rupture occurs	10	
	Low-low steam generator level reactor trip setpoint reached in ruptured steam generator	~~32.7~~ 34.2	
	Rods begin to drop, power lost to the reactor coolant pumps	~~34.7~~ 36.2	
	Pressurizer safety valve setpoint reached	~~38~~ 39.5	
	Steam generator safety valve setpoint reached in intact steam generators	~~38~~ 39.0	
	Auxiliary feedwater is delivered to two intact steam generators	~~92.7~~ 94.2	73 73

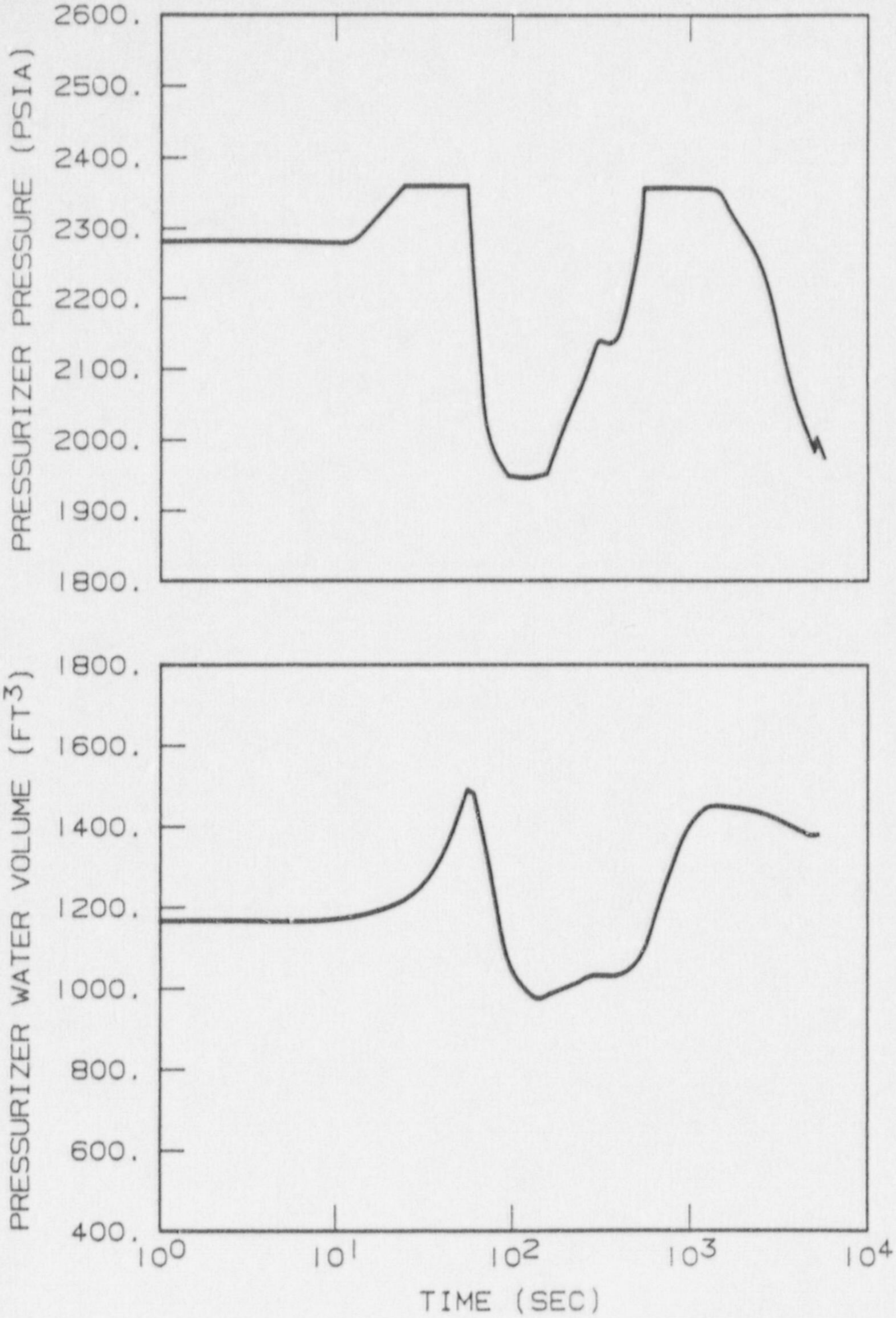
TABLE 15.2-1
(Sheet 9 of 9)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)	
	Low steam line pressure setpoint reached in ruptured steam generator	~~433.7~~ 417	73
	All main steam line isolation valves close	~~440.7~~ 424	73
	Core decay heat plus pump heat decreases to auxiliary feedwater heat removal capacity	~1850	

Note 1: DNBR does not decrease below its initial value.

Note 2: Analyses assume 600 gpm for conservatism during accident conditions. Four steam generators would receive more flow from the Auxiliary Feedwater System.



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Pressurizer Pressure and Water Volume Transient for Loss of Offsite Power

FIGURE 15.2-9

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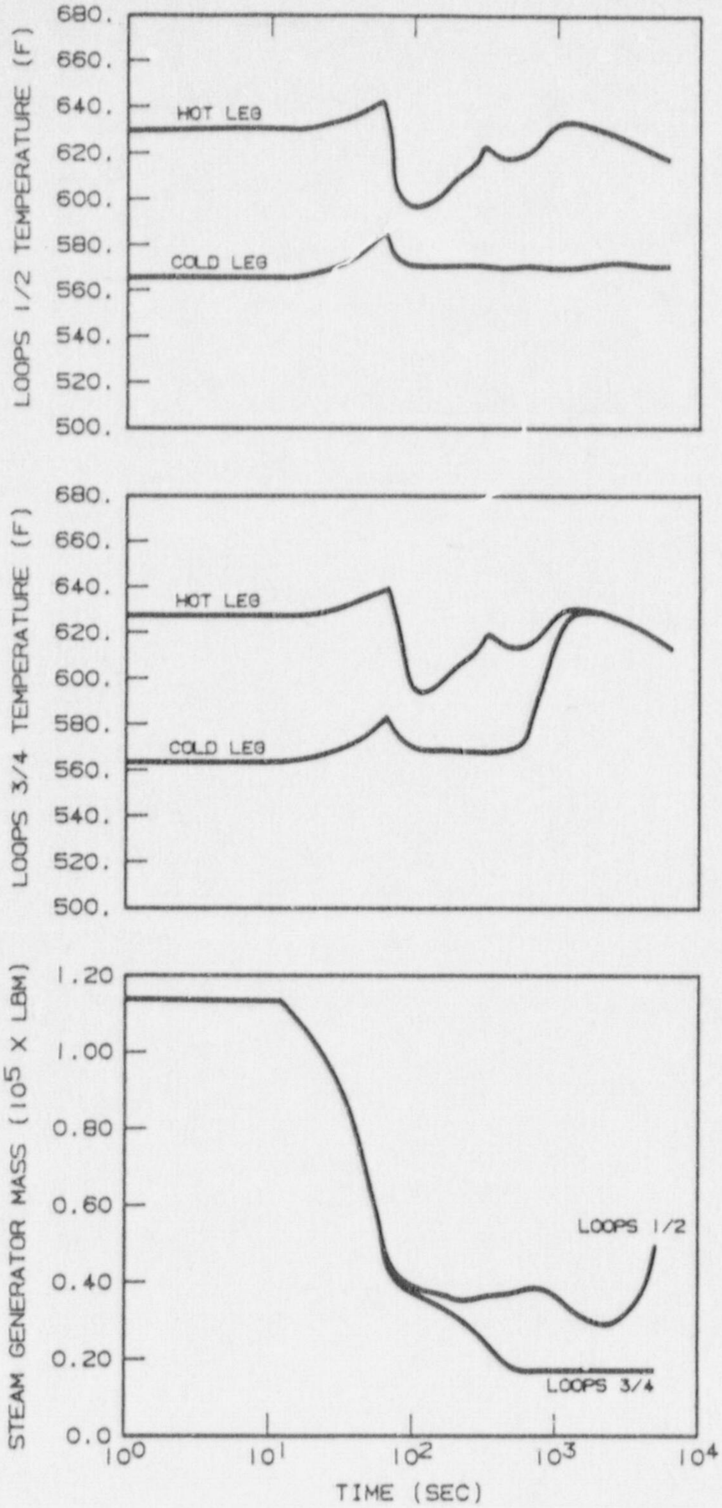
August 5, 1988

052-A-25890-1



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Steam Generator Mass and Reactor Coolant Loop Temperature for Loss of Offsite Power

FIGURE 15.2-10

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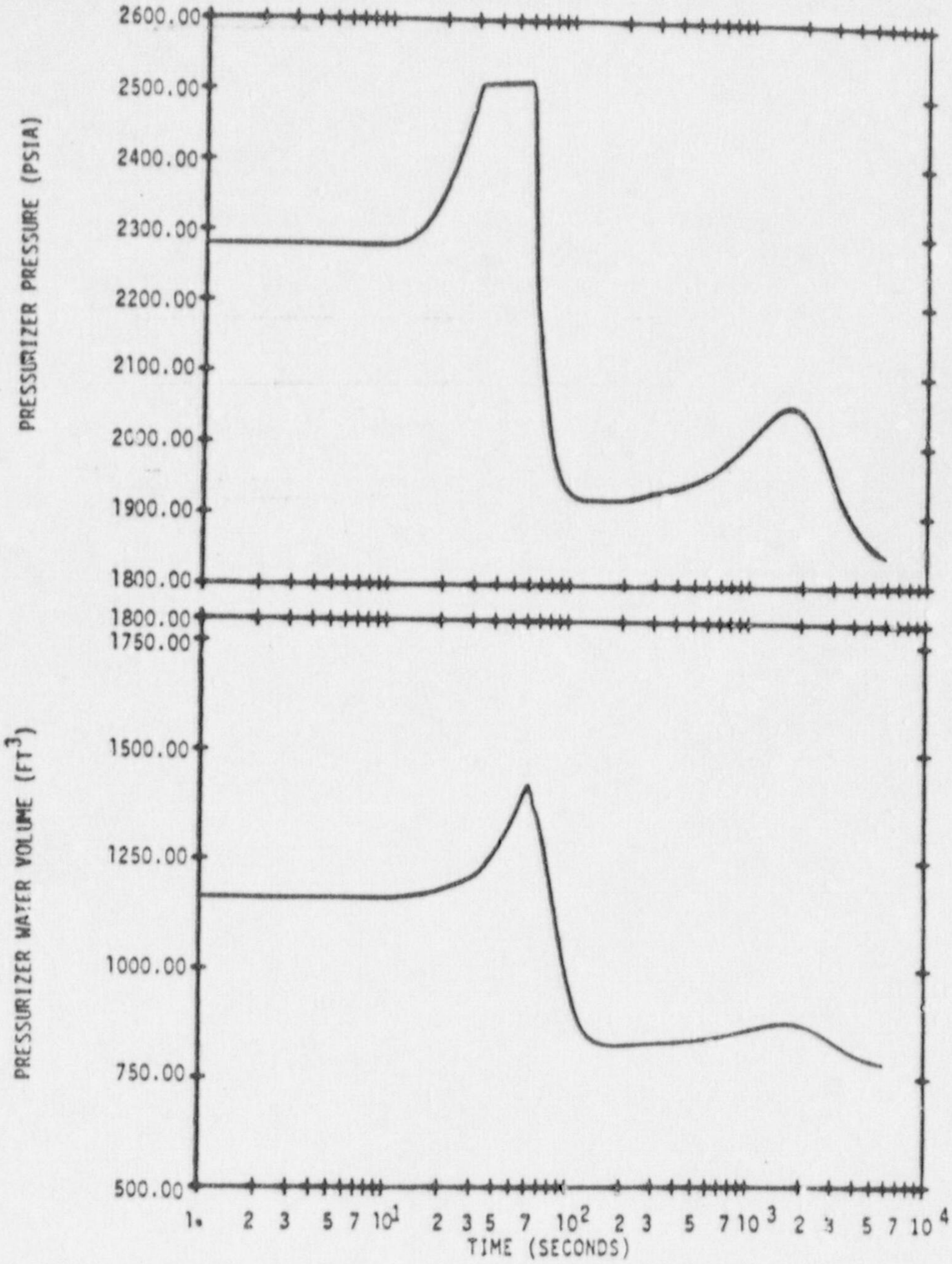
052-A-25890-2



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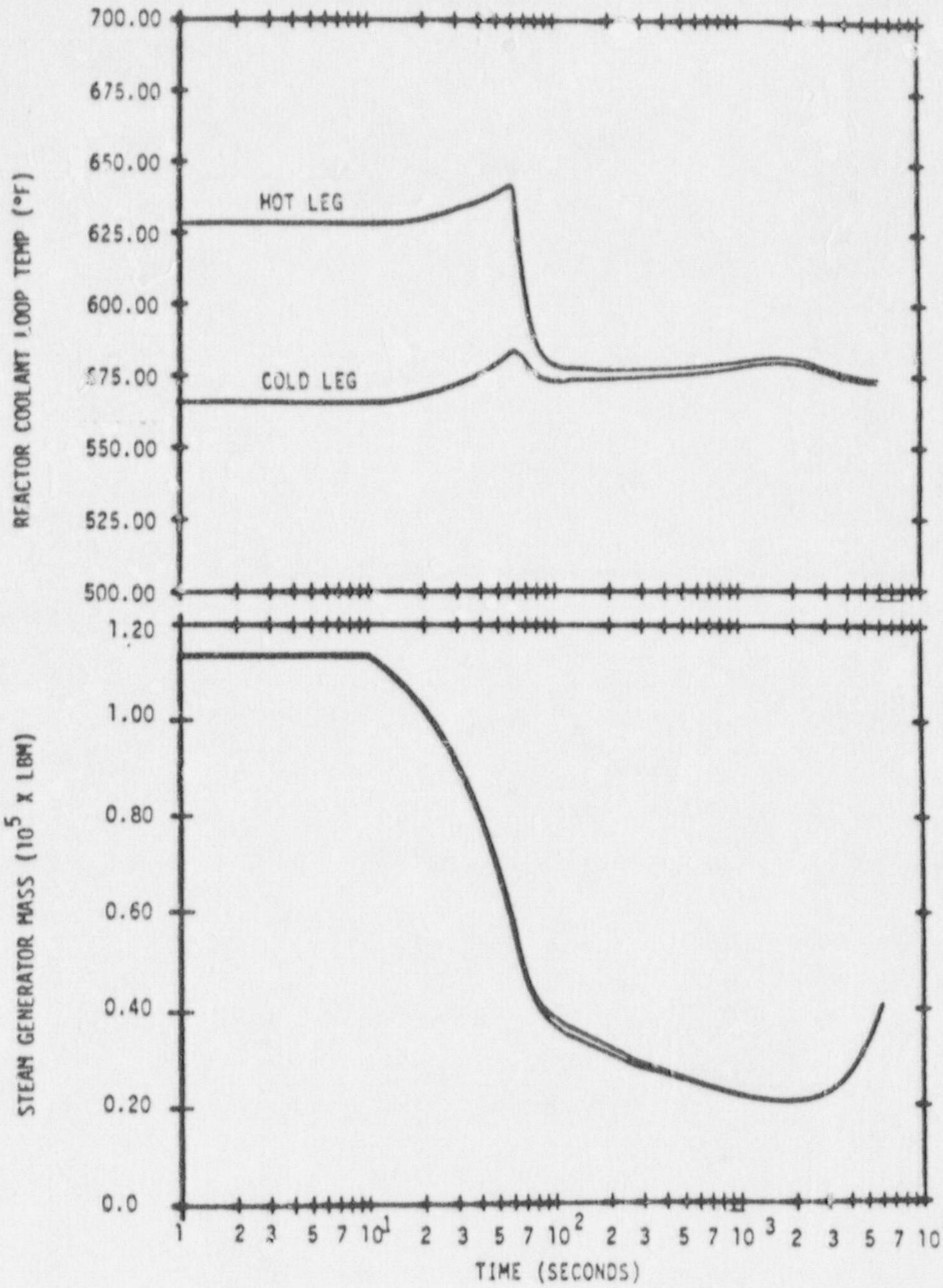
PRESSURIZER PRESS. & WATER VOL. TRANS. FOR LOSS OF NORMAL FEED.

FIGURE 15.2-11

AMENDMENT 57
DECEMBER 20, 1985



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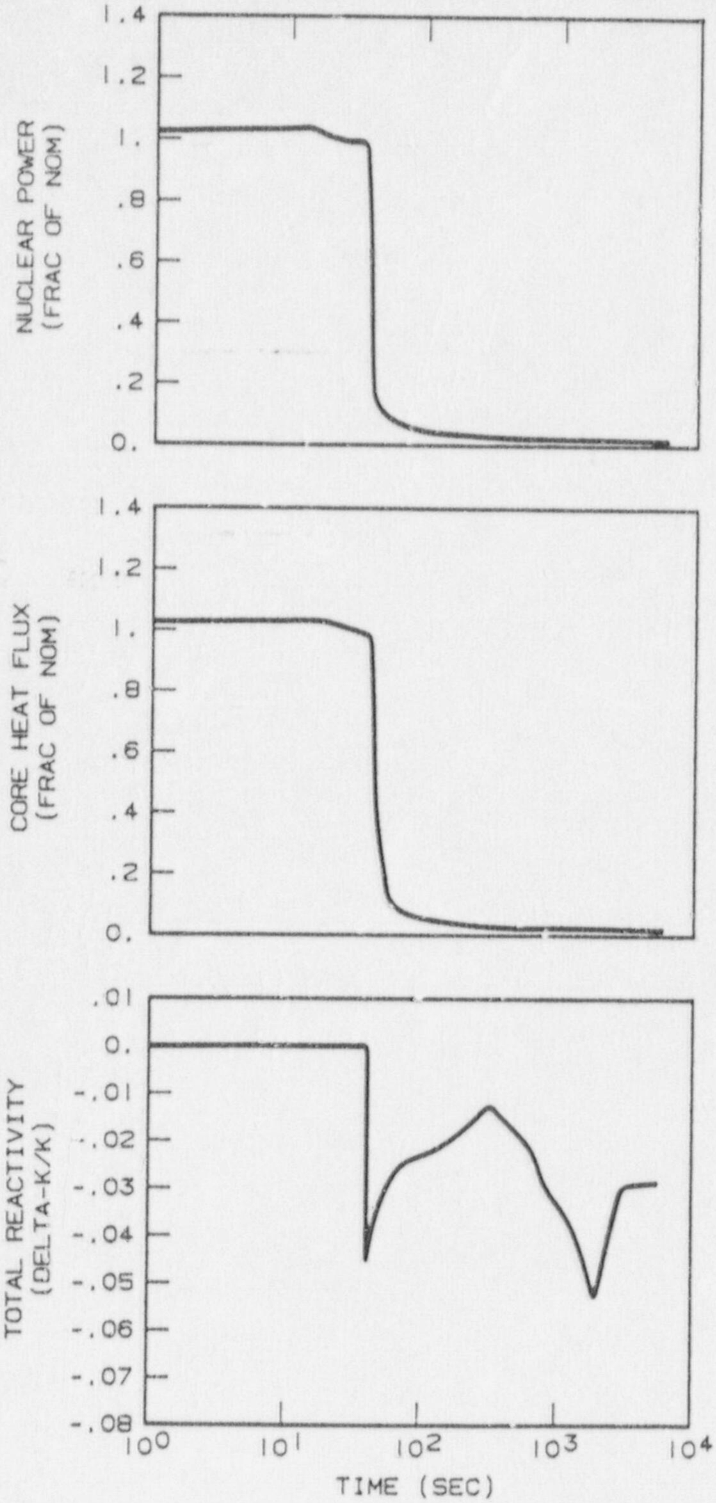
STEAM GENERATOR MASS AND REACTOR COOLANT LOOP TEMPERATURE FOR LOSS OF NORMAL FEEDWATER

AMENDMENT 57
DECEMBER 20, 1985

FIGURE 15.2-12



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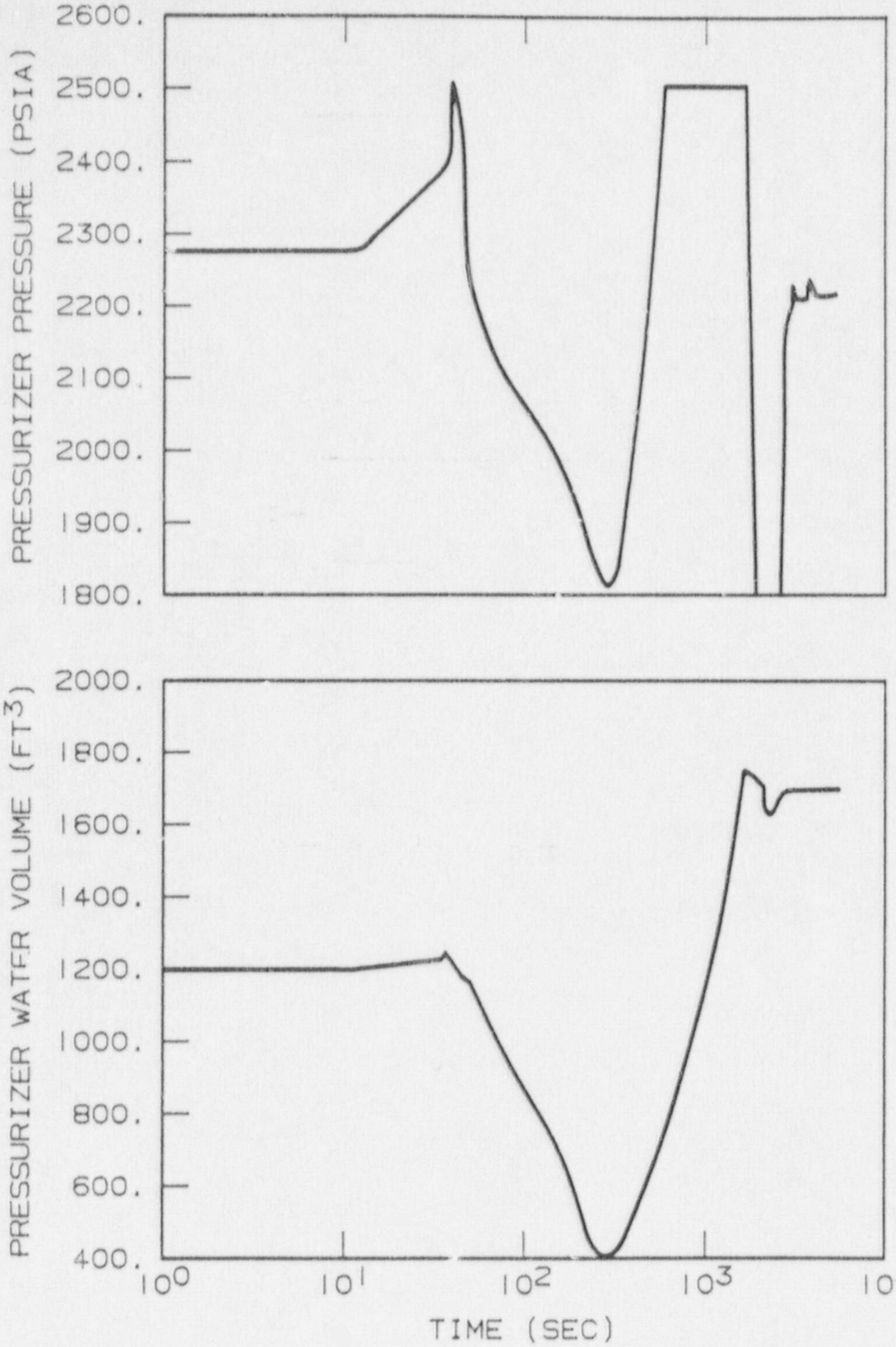
Main Feedline Rupture With Offsite Power Available

FIGURE 15.2-13

052-A-25890-3



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Main Feedline Rupture With Offsite Power Available

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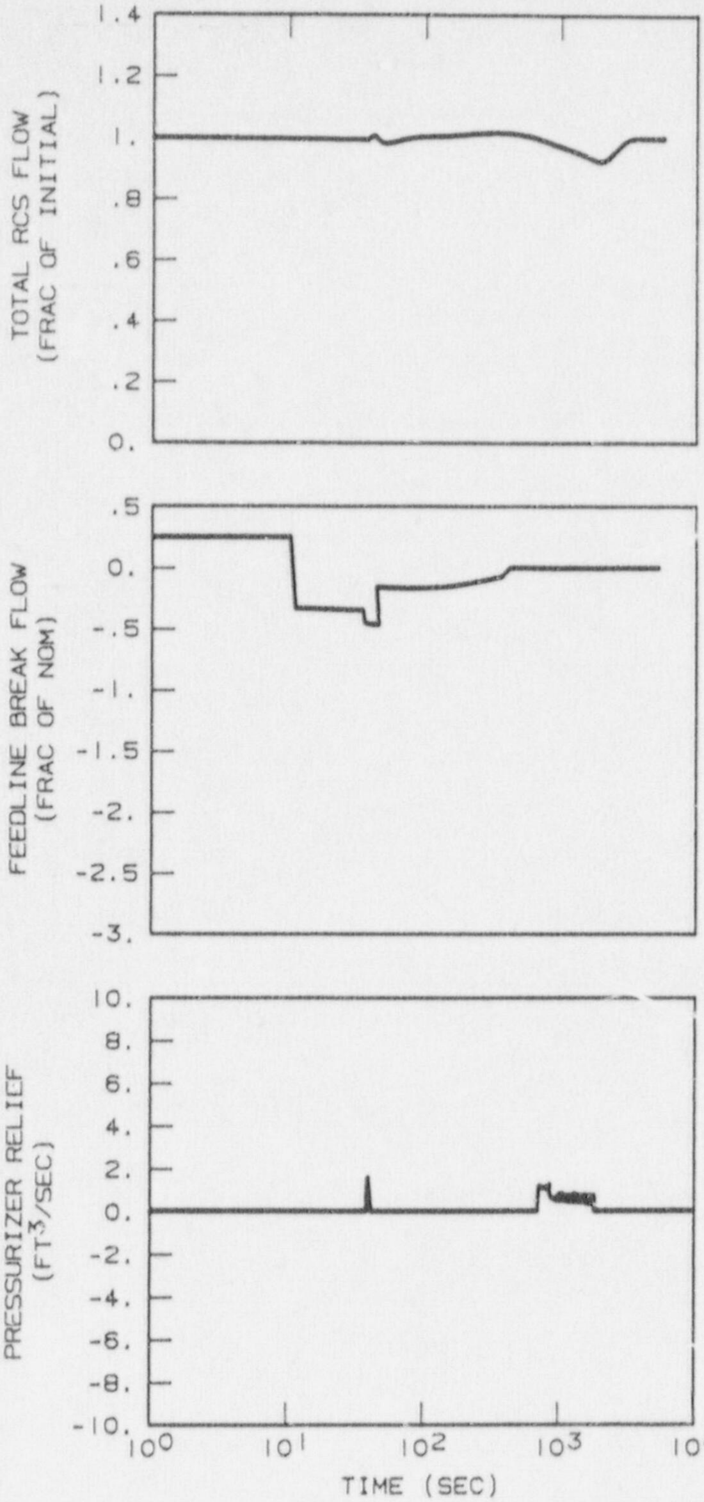
FIGURE 15.2-14

052-A-25890-4



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Main Feedline Rupture With Offsite Power Available

FIGURE 15.2-15

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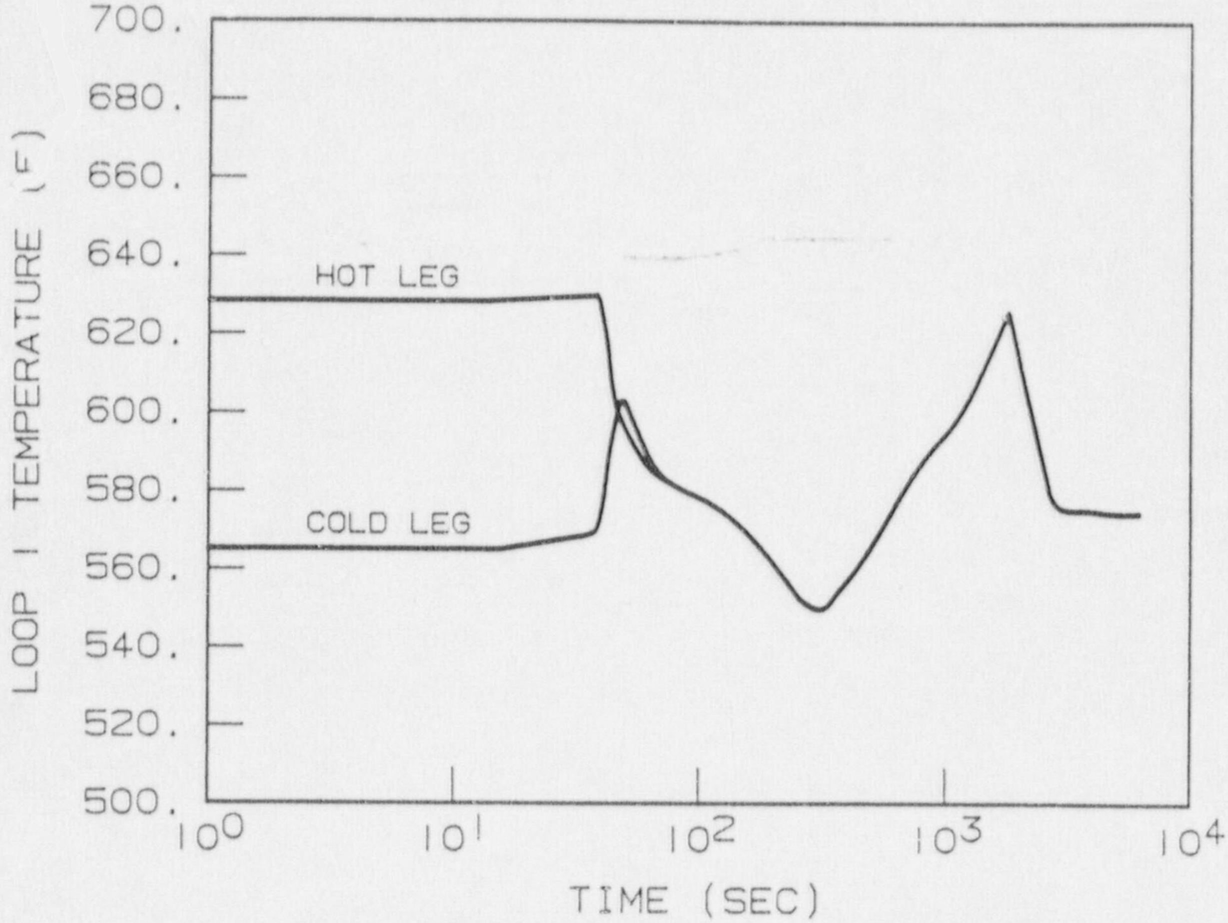
052-A-25890-5



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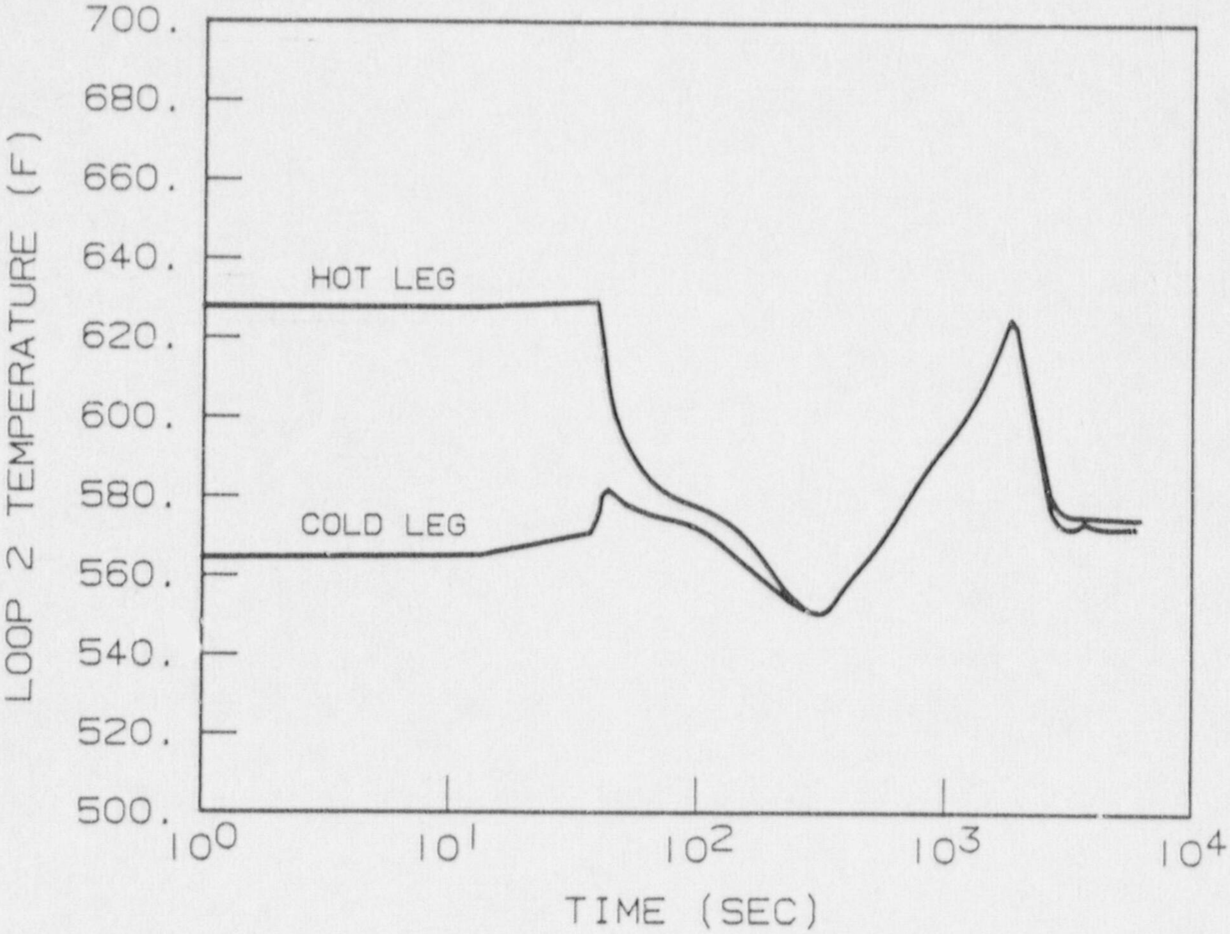
Main Feedline Rupture With Offsite Power Available

FIGURE 15.2-16

052-A-25890-6



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Main Feedline Rupture With Offsite Power Available

FIGURE 15.2-17

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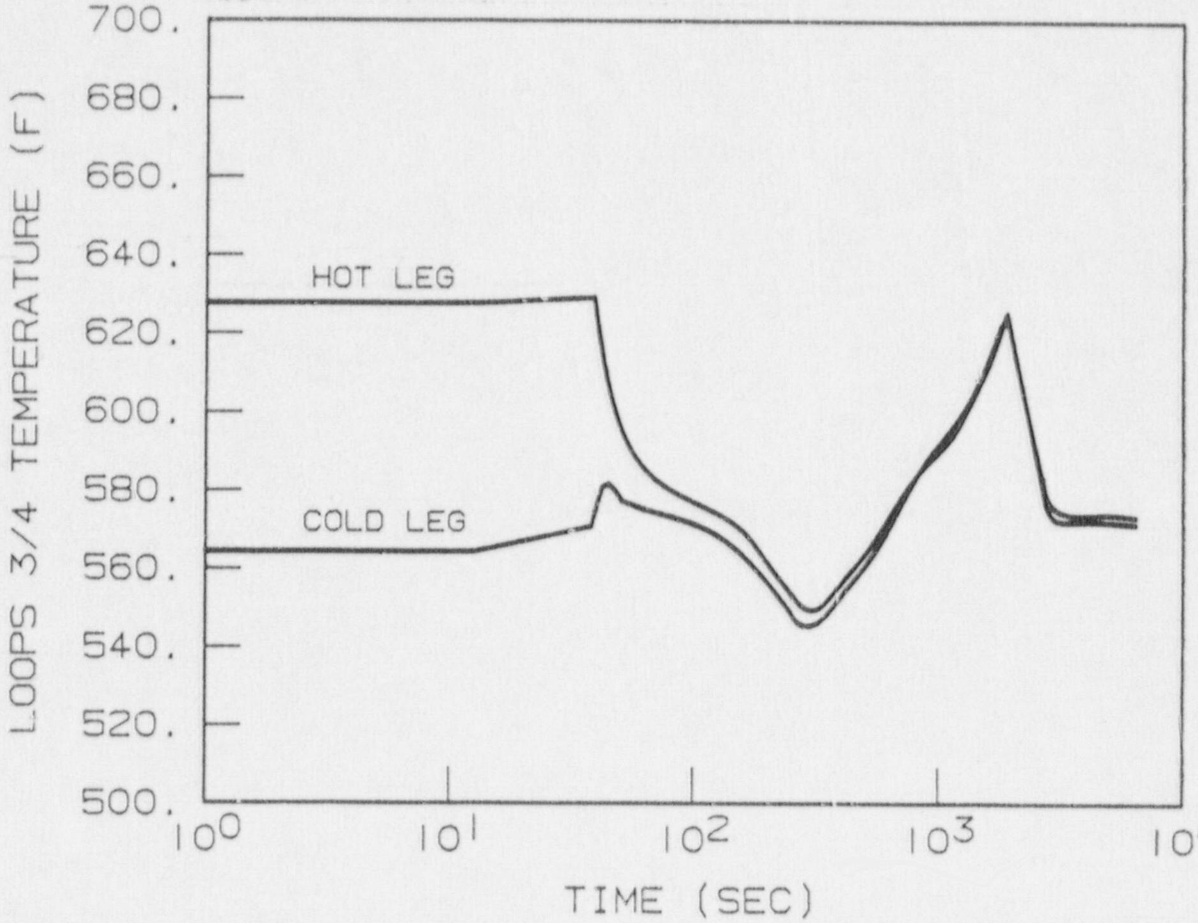
052-A-25890-7



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Main Feedline Rupture With Offsite Power Available

FIGURE 15.2-18

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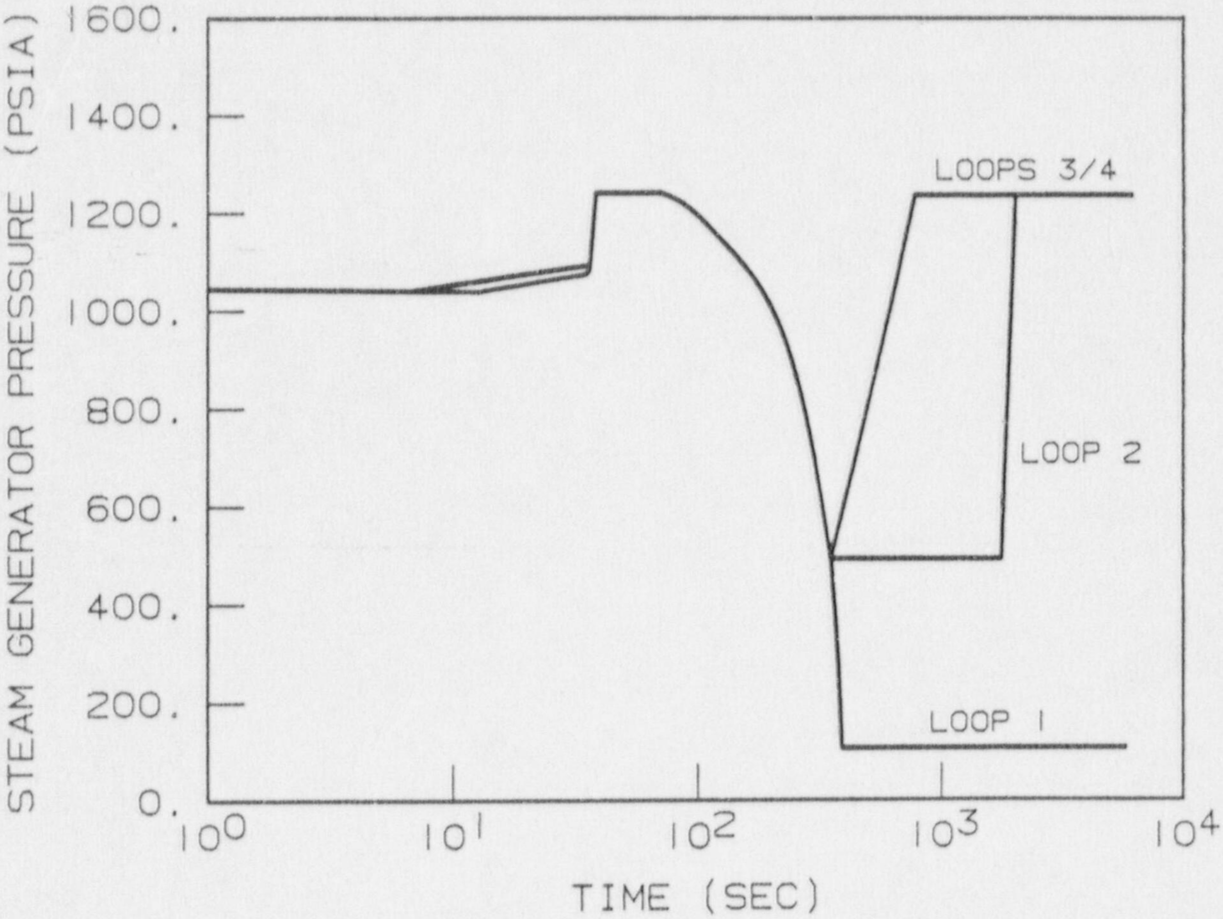
052-A-25890-8



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Main Feedline Rupture With Offsite Power Available

FIGURE 15.2-19

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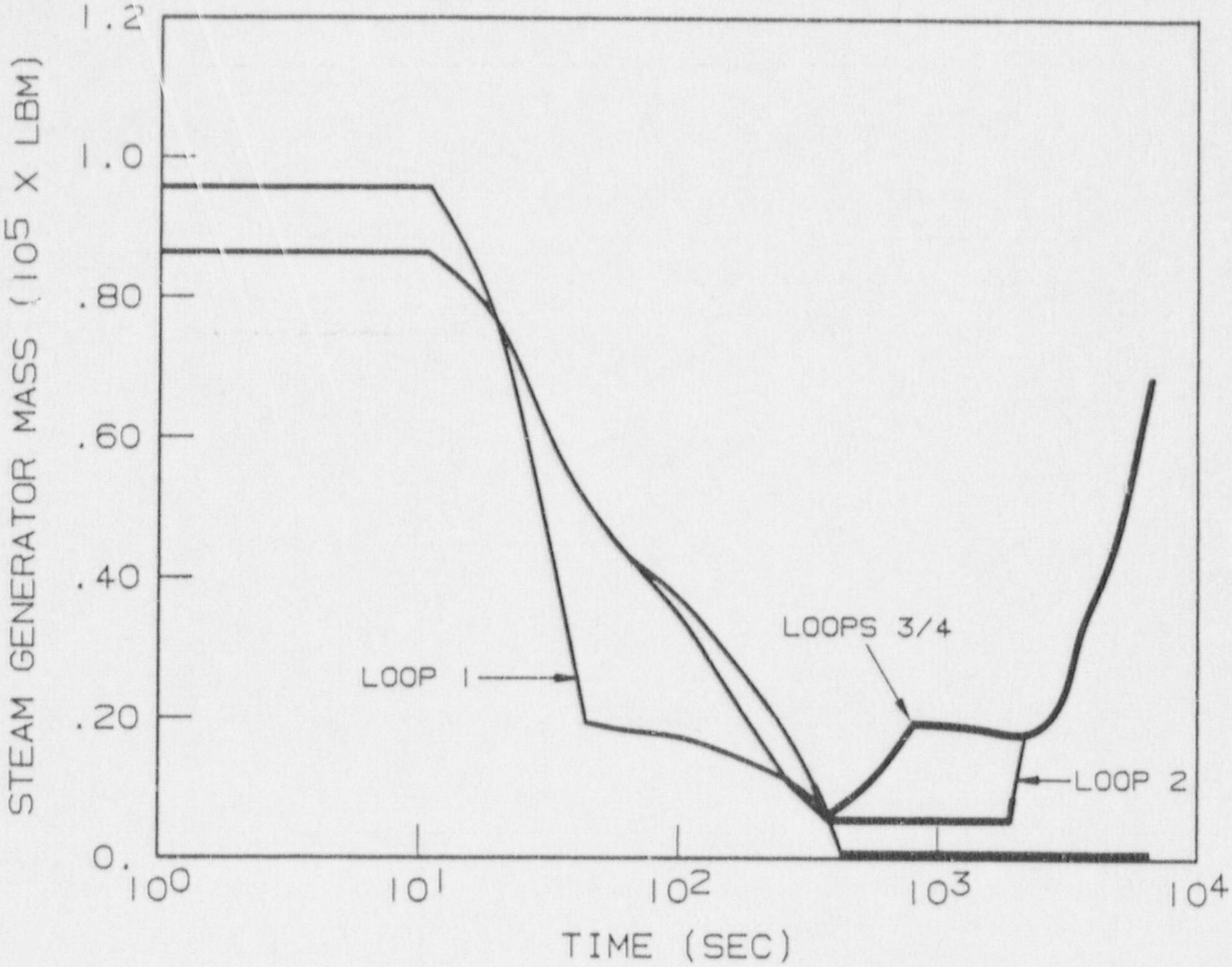

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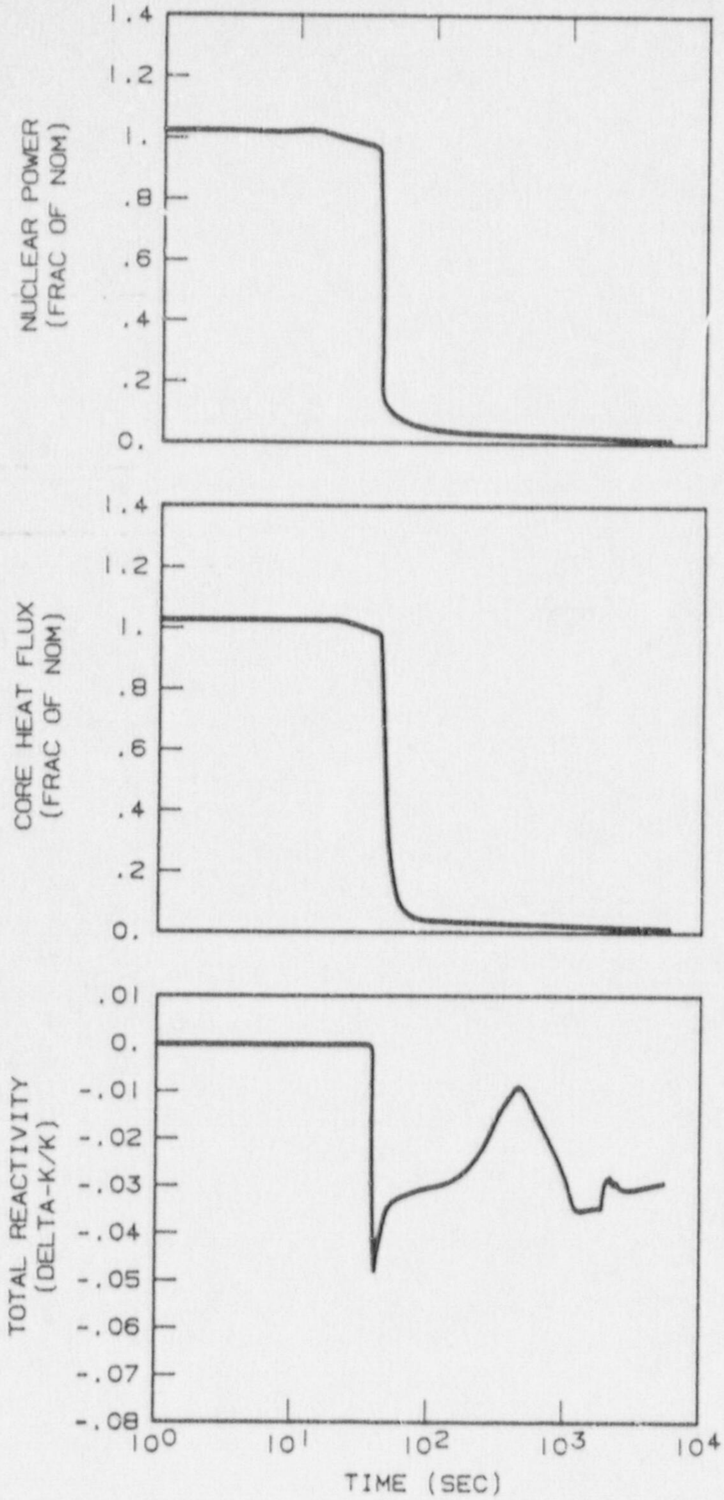
Main Feedline Rupture With Offsite Power

FIGURE 15.2-19A

052-A-25890-17



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Main Feedline Rupture Without Offsite Power

FIGURE 15.2-20

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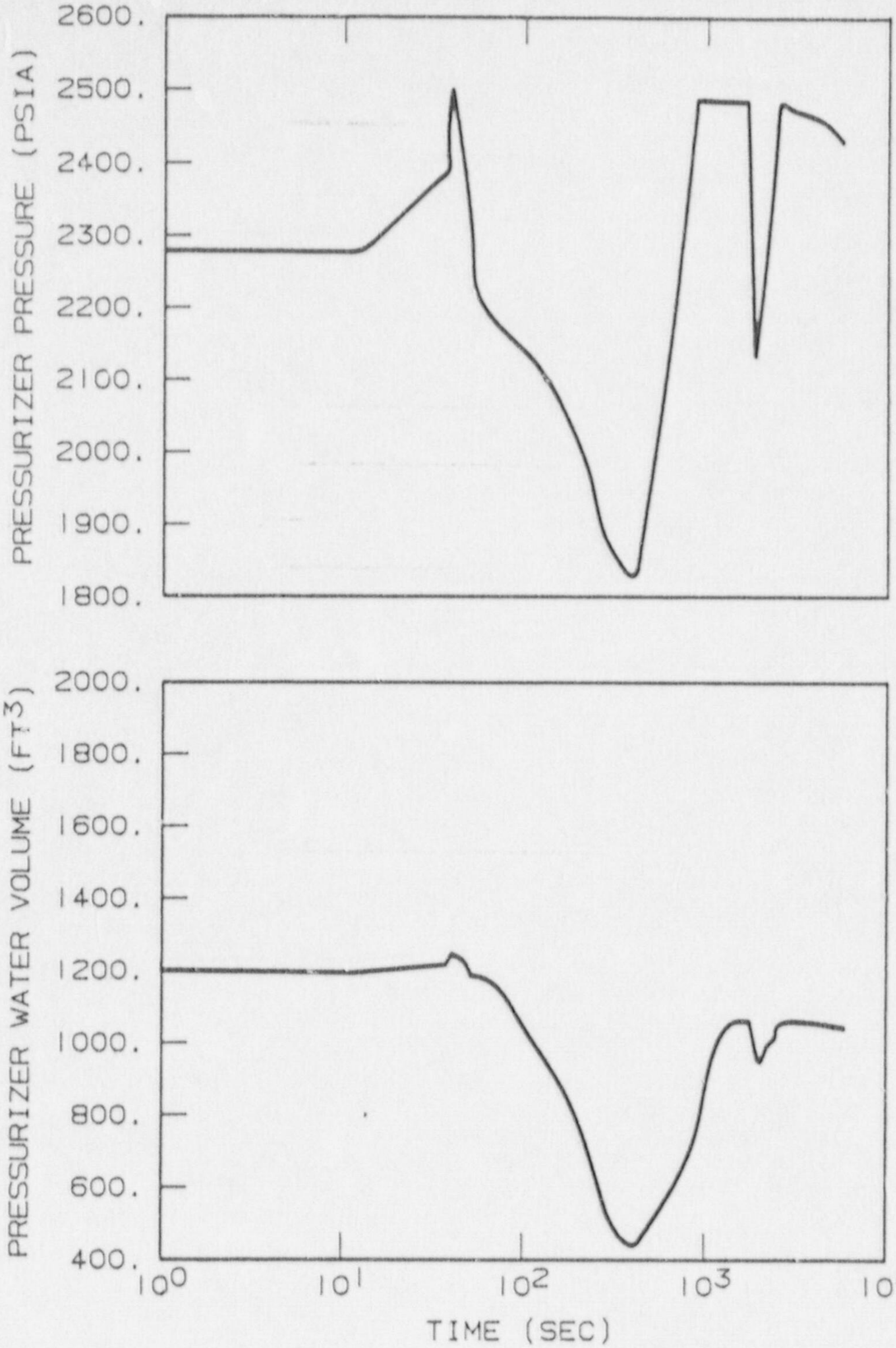

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Main Feedline Rupture Without Offsite Power

FIGURE 15.2-21

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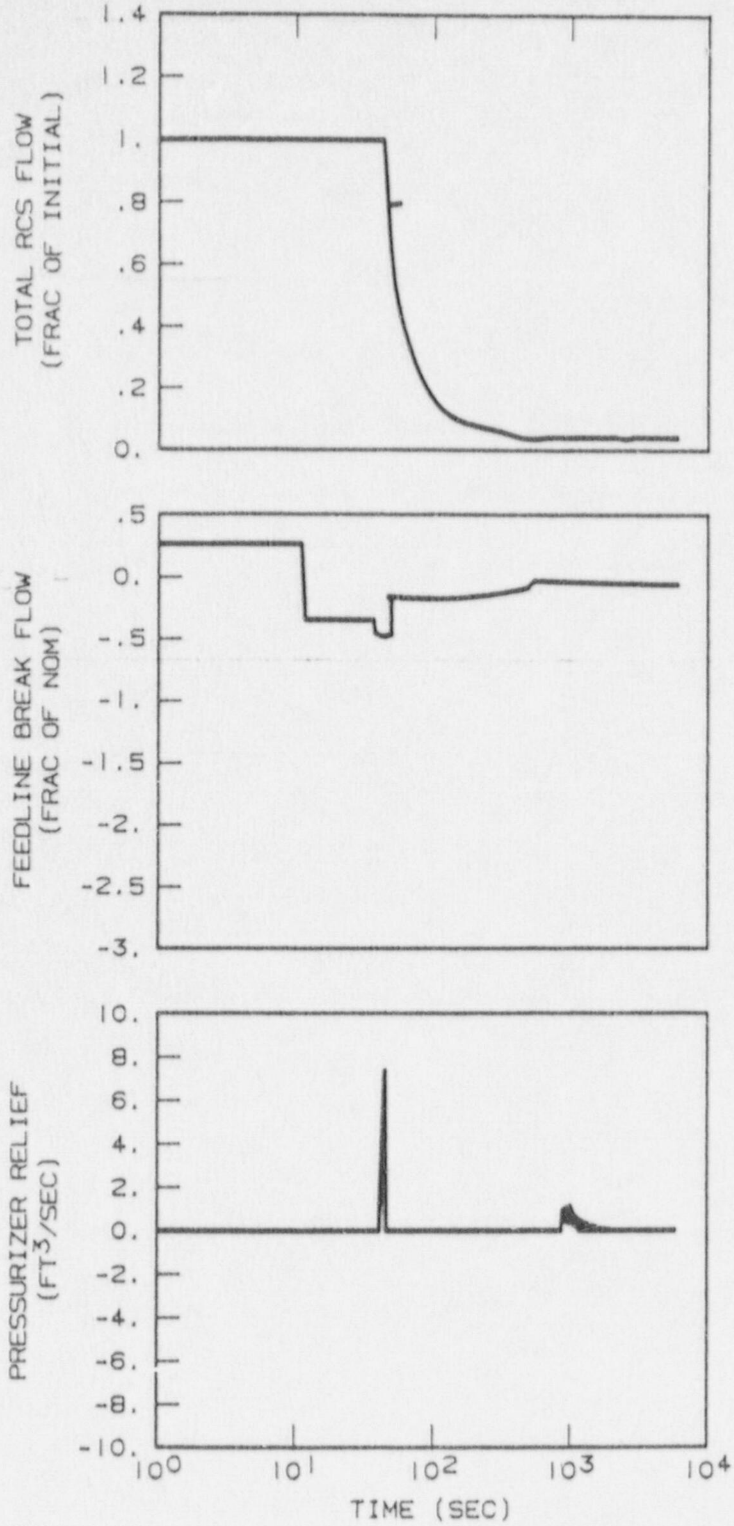
052-A-25890-11



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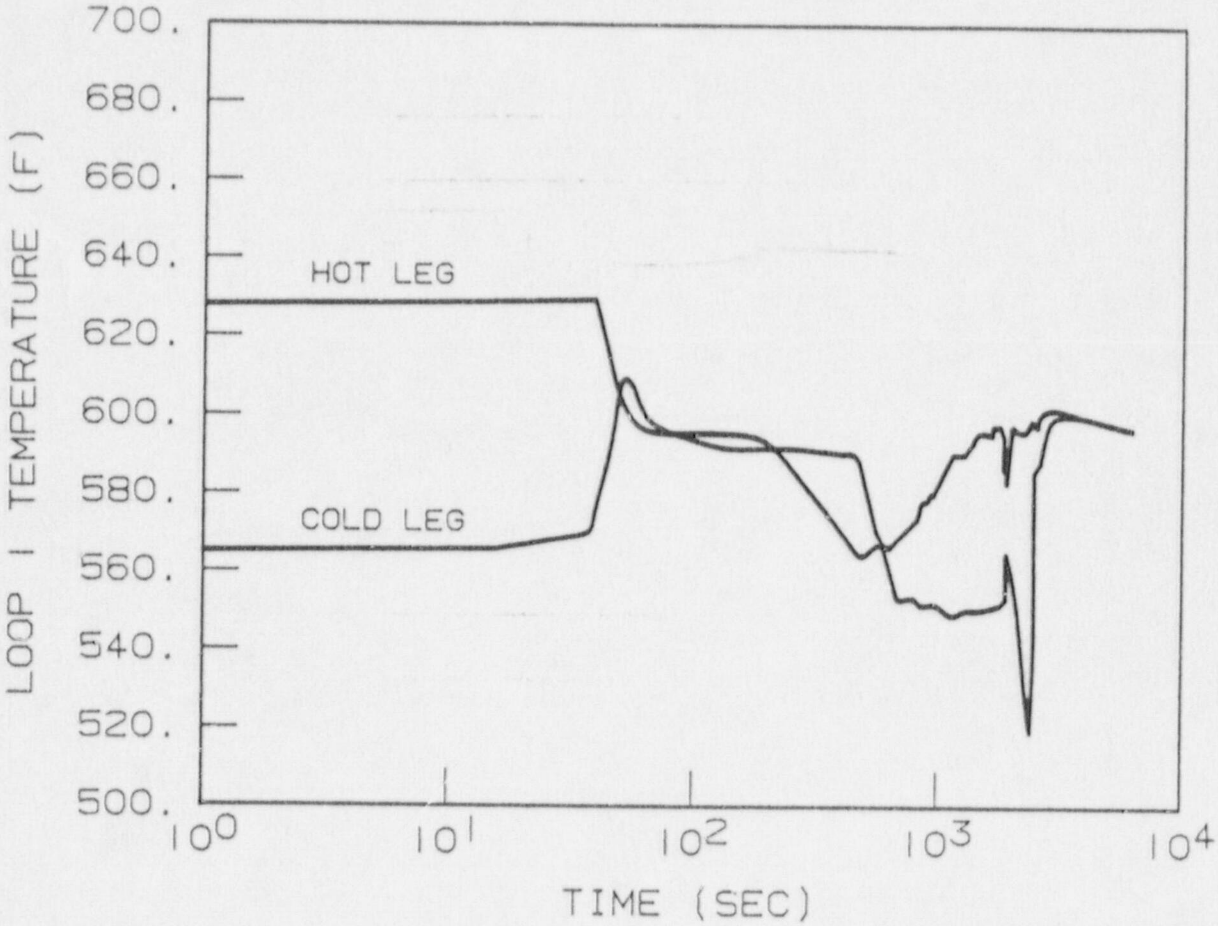
Main Feedline Rupture Without Offsite Power

FIGURE 15.2-22

052-A-25890-12



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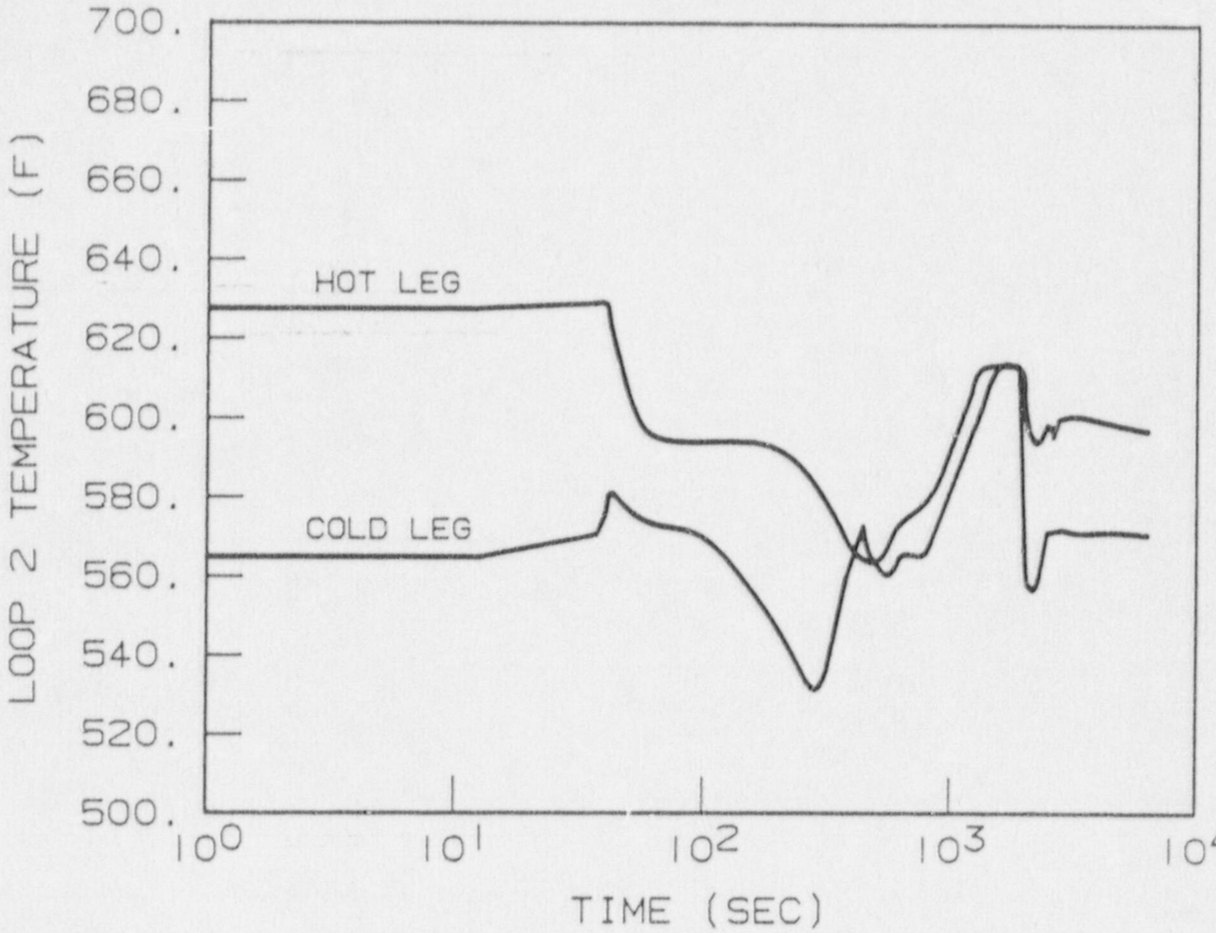
Main Feedline Rupture Without Offsite Power
FIGURE 15.2-23

052-A-25890-13



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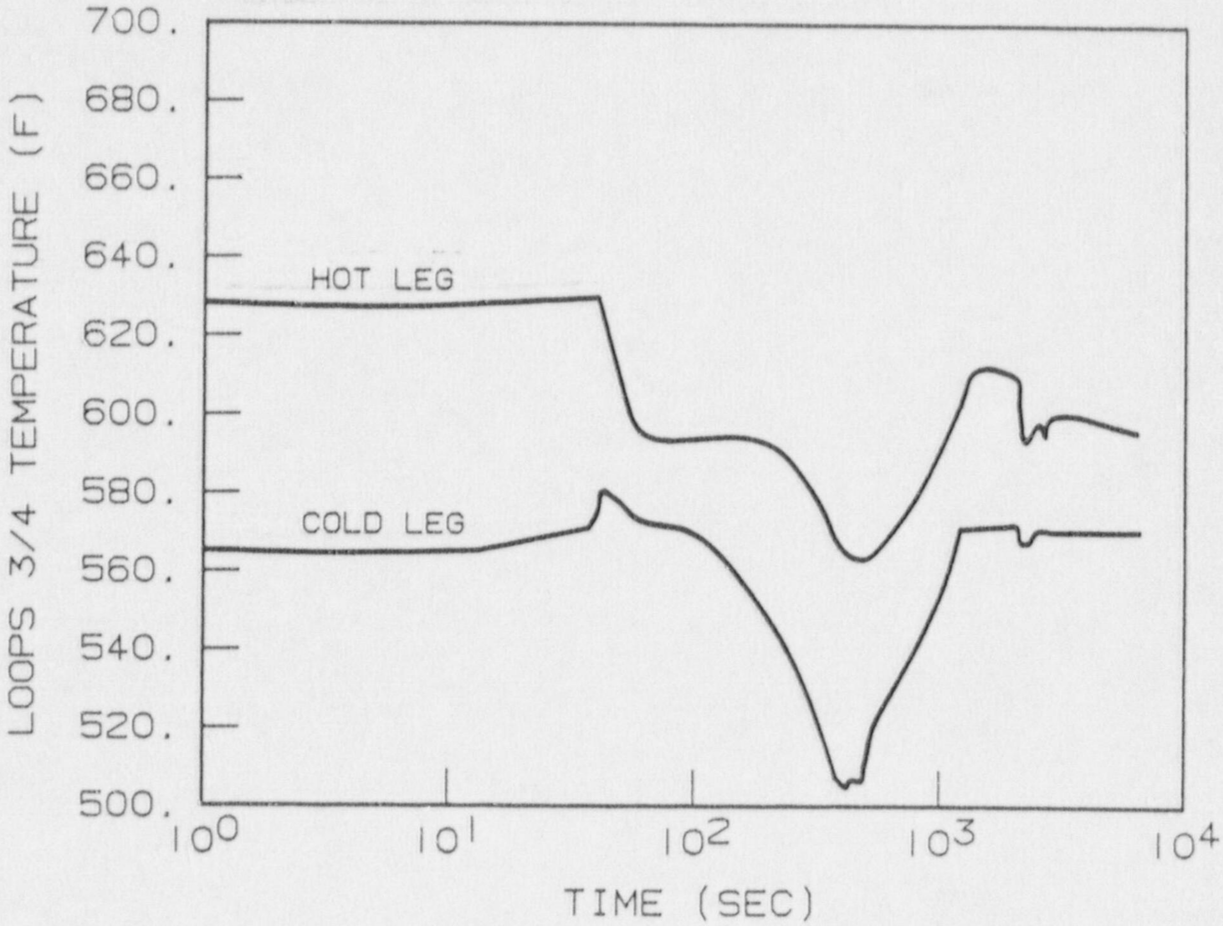
Main Feedline Rupture Without Offsite Power

FIGURE 15.2-24

052-A-25890-14



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Main Feedline Rupture Without Offsite Power

FIGURE 15.2-25

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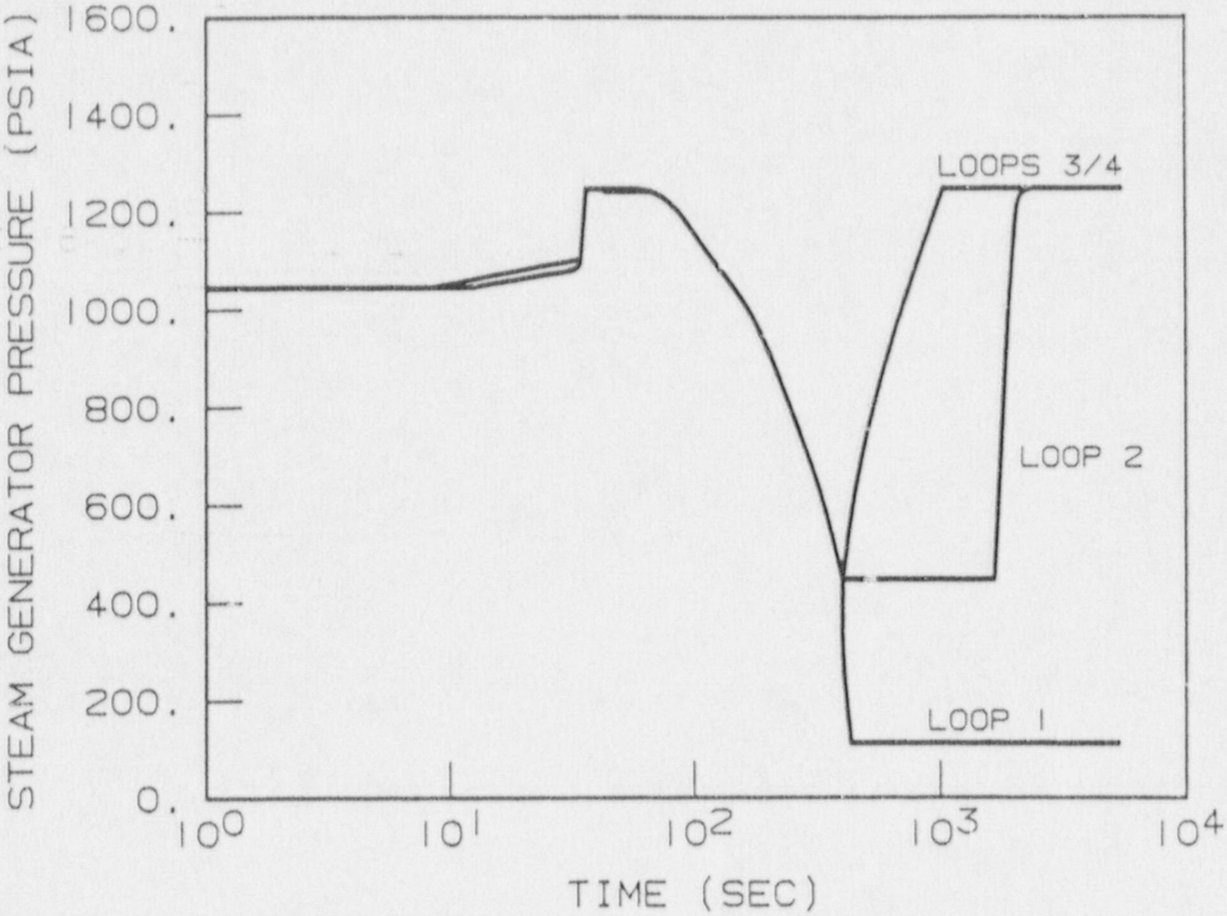

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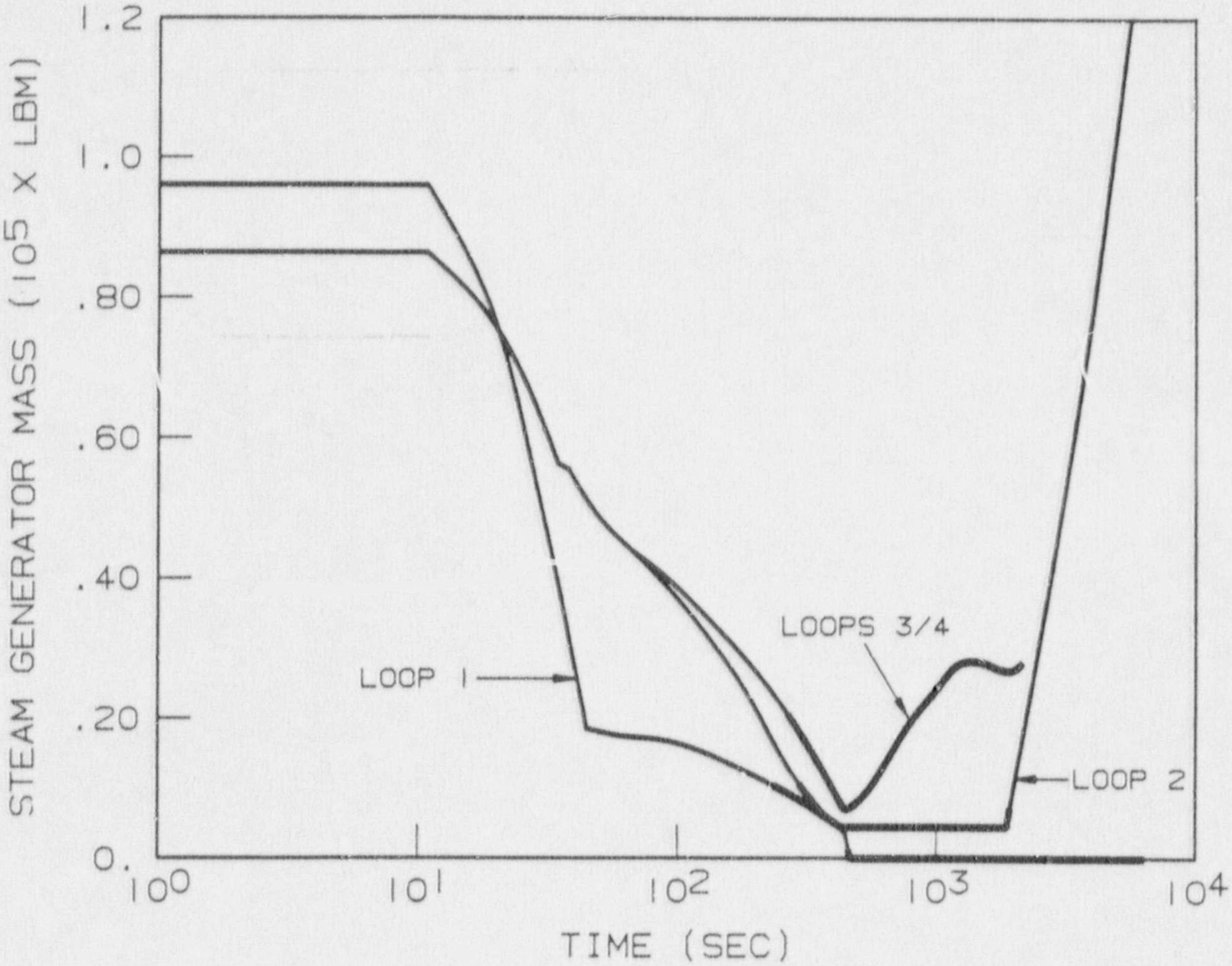
Main Feedline Rupture Without Offsite Power

FIGURE 15.2-26

052-A-25890-16



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Main Feedline Rupture Without Offsite Power

FIGURE 15.2-26A

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052-A-25890-18

Q212.75 For the reactor coolant pump locked rotor and shaft break events and for the feedwater line break event, does the analysis assume water relief from the pressurizer safety valves? If so, provide justification for the water relief rate assumed in the analysis and state if the basis for the hydraulic loads used to analyze the mechanical design of the valve, discharging piping, and their supports include water relief loads.

R212.75 No water relief through the pressurizer safety valves occurs for the reactor coolant pump locked rotor and shaft break events. For the feedline break events, water relief does occur. As shown on figures 15.2-12 and -13, the maximum water relief rate through the three (3) safety valves is a total of approximately 1 ft^3/sec. The flowrate is calculated based on the homogeneous equilibrium model for saturated fluid, which gives the most conservative relief rates. This model indicates a maximum available relief capacity of 17.4 ft^3/sec. at 2500 psia, much greater than that predicted in the analysis of the feedline break event.

The basis for analyzing the mechanical design of the Reactor Coolant Pressure Boundary Class 1 Piping and supports are discussed in Section 3.9N.1. Class 1 valves, such as the pressurizer safety valves, are discussed in Section 3.9N.1.4.5 and 5.4.13. The pressurizer safety valves discharge piping and relief tank are non-nuclear safety.

¶ The pressurizer safety valves at Comanche Peak have been analyzed for the MFLB conditions consistent with the work performed for WCAP-11677. The conclusion in WCAP-11677 that the Crosby 6M6 valves can pass slightly subcooled water ~~as a minimum up to~~ at least three times without damage applies to the Comanche Peak valves.

Westinghouse setpoint metholodolgy) within 5% of the top or bottom of the instrument range will respond any differently than any other protection function. 49

Because large steam generator pressure changes are not expected before trip, only the reference leg heatup effects need be considered, and not the effects of system pressure changes.

The basis for determination of the low-low setpoint is the Loss of Normal Feedwater and Feedline Break events. The setpoints were determined by considering the level used in each of the analyses for each unit.

	Unit 1	Unit 2
Loss of Normal Feedwater	~~34.6%~~ 0	0%
Feedline Break	~~2.0%~~ 0	0%

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For each unit, the setpoint was determined by considering the following errors for feedline break:

- Normal errors (normal channel accuracy, etc.)

- Post-Accident effects on transmitter (radiation and temperature)

- Reference leg effects (post-accident heatup)