REGULATORY [illegible]RMATION DISTRIBUTION SYS[illegible] (RIDS)

ACCESSION NBR:8212210101 DOC.DATE: 82/12/14 NOTARIZED: NO DOCKET #
FACIL:50-389 St. Lucie Plant, Unit 2, Florida Power & Light Co. 05000389
AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards addl info re small feedwater line break analysis. Info neededed to close confirmatory issue addressed in Section 15.10.E of SER.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 19
TITLE: Licensing Submittal: PSAR/FSAR Amdts & Related Correspondence

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	A/D LICENSNG	1	0	LIC BR #3 BC	1	0
	LIC BR #3 LA	1	0	NERSES,V. 01	1	1
INTERNAL:	ELD/HDS2	1	0	IE FILE	1	1
	IE/DEP EPDS 35	1	1	IE/DEP/EPLB 36	3	3
	NRR/DE/AEAB	1	0	NRR/DE/CEB 11	1	1
	NRR/DE/EQB 13	2	2	NRR/DE/GB 28	2	2
	NRR/DE/HGEB 30	1	1	NRR/DE/MEB 18	1	1
	NRR/DE/MTEB 17	1	1	NRR/DE/QAB 21	1	1
	NRR/DE/SAB 24	1	1	NRR/DE/SEB 25	1	1
	NRR/DHFS/HFEB40	1	1	NRR/DHFS/LQB 32	1	1
	NRR/DHFS/OLB 34	1	1	NRR/DL/SSPB	1	0
	NRR/DSI/AEB 26	1	1	NRR/DSI/CPB 10	1	1
	NRR/DSI/CSB 09	1	1	NRR/DSI/ICSB 16	1	1
	NRR/DSI/METB 12	1	1	NRR/DSI/PSB 19	1	1
	NRR/DSI/RAB 22	1	1	NRR/DSI/RSB 23	1	1
	REG FILE 04	1	1	RGN2	3	3
	RM/DDAMI/MIB	1	0			
EXTERNAL:	ACRS 41	6	6	BNL(AMDTS ONLY)	1	1
	DMB/DSS (AMDTS)	1	1	FEMA-REP DIV 39	1	1
	LPDR 03	1	1	NRC PDR 02	1	1
	NSIC 05	1	1	NTIS	1	1

TOTAL NUMBER OF COPIES REQUIRED: LTTR 52 ENCL 45

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8212210101 DOC.DATE: 82/12/14 NOTARIZED: NO DOCKET #
FACIL:50-389 St. Lucie Plant, Unit 2, Florida Power & Light Co. 05000389
AUTH.NAME AUTHOR AFFILIATION
UHRIG,R.E. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards addl info re small feedwater line break analysis.
Info needed to close confirmatory issue addressed in
Section 15.10.F of SER.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR __ ENCL __ SIZE:__________
TITLE: Licensing Submittal: PSAR/FSAR Amdts & Related Correspondence

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	A/D LICENSNG	1	0	LIC BR #3 BC	1	0
	LIC BR #3 LA	1	0	NERSES,V. 01	1	1
INTERNAL:	ELD/HDS2	1	0	IE FILE	1	1
	IE/DEP EPDS 35	1	1	IE/DEP/EPLB 36	3	3
	NRR/DE/A/EAB	1	0	NRR/DE/CEB 11	1	1
	NRR/DE/EQB 13	2	2	NRR/DE/GB 28	2	2
	NRR/DE/HGEB 30	1	1	NRR/DE/MEB 18	1	1
	NRR/DE/MTEB 17	1	1	NRR/DE/QAB 21	1	1
	NRR/DE/SAB 24	1	1	NRR/DE/SEB 25	1	1
	NRR/DHFS/HFEB40	1	1	NRR/DHFS/LQB 32	1	1
	NRR/DHFS/OLB 34	1	1	NRR/DL/SSPB	1	0
	NRR/DSI/AEB 26	1	1	NRR/DSI/CPB 10	1	1
	NRR/DSI/CSB 09	1	1	NRR/DSI/ICSB 16	1	1
	NRR/DSI/METB 12	1	1	NRR/DSI/PSB 19	1	1
	NRR/DSI/RAB 22	1	1	NRR/DSI/RSB 23	1	1
	REG FILE 04	1	1	RGN2	3	3
	RM/DDAMI/MIB	1	0			
EXTERNAL:	ACRS 41	6	6	BNL(AMDTS ONLY)	1	1
	DMB/DSS (AMDTS)	1	1	FEMA-REP DIV 39	1	1
	LPDR 03	1	1	NRC PDR 02	1	1
	NSIC 05	1	1	NTIS	1	1

TOTAL NUMBER OF COPIES REQUIRED: LTTR 52 ENCL 45

OX 14000, JUNO BEACH, FL 33408

FLORIDA POWER & LIGHT COMPANY

December 14, 1982
L-82-542

Office of Nuclear Reactor Regulations
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

Re: ST. LUCIE UNIT NO. 2
DOCKET NO. 50-389
SMALL FEEDWATER LINE BREAK ANALYSIS

Attached please find additional information required by J. Guttman of your staff, which supplements the feedwater line break analysis submitted in Florida Power and Light Company letter L-82-533 dated December 9, 1982.

It is our understanding that this submittal completes the information required to close the confirmatory issue addressed in Section 15.10.2 of the Safety Evaluation Report.

If you have any questions regarding this submittal, please contact us accordingly.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President
Advanced Systems and Technology

REU/RJS/JES/ok

Attachment

B001

cc: J.P. O'Reilly, Region II
Harold F. Reis, Esquire

REANALYSIS OF SMALL BREAK LOSS OF FEEDWATER INVENTORY EVENTS WITH THE LIMITING SINGLE FAILURE AND OFFSITE POWER AVAILABLE

INTRODUCTION

Purpose

The purpose of this reanalysis is to show that the results of the small break loss of feedwater inventory event with the limiting single failure and offsite power available produce maximum pressures less than 110% of design.

Background

The loss of feedwater inventory event presented previously demonstrates that breaks of all sizes, when combined with the loss of offsite power, produce maximum pressures well below 120% of design. Based on the recurrence frequences provided in Reference 3, the NRC has concluded that the 120% of design maximum pressure criterion is appropriate for large break loss of feedwater inventory events, and small break loss of feedwater inventory events combined with the loss of offsite power. However, it must be shown that small break loss of feedwater inventory events with the limiting single failure and offsite power available meet the maximum pressure criterion of 110% of design.

In order to demonstrate compliance with this criterion, a reanalysis of small breaks with a modified methodology was required. The methodology used previously is applicable to the full spectrum of break sizes. However, it is extremely conservative when applied to the smaller break sizes. As a result, a new method of analysis which is still conservative was developed, and is discussed in the following section.

Since the recurrence frequencies presented in Reference 3 apply to pipes greater than 6 inches in diameter, the reanalysis need only consider breaks less than approximately 0.20 ft^2. This is the same break size presented previously as the limiting break with the original methodology. Therefore, in the following sections "small" breaks refer to those which are less than 0.20 ft^2.

METHOD OF ANALYSIS

Mathematical Models

The methodology used in the reanalysis of small break loss of feedwater inventory events is the same as that applied and described previously with the exception of the treatment of steam generator heat transfer and reactor trip on steam generator low water level. Predictions of steam generator heat transfer and level behavior are based on the model documented in References 5 through 8. As discussed below, this model is conservative when applied to the small break loss of feedwater inventory events.

Steam Generator Heat Transfer

RCS pressurization is largely a function of the rate at which the ruptured steam generator's heat transfer decreases as its inventory is depleted. (The "ruptured" generator refers to the steam generator nearest the pipe break). Section 15B.3 (Ref. 4) documents the sensitivity of RCS pressurization to steam generator heat transfer behavior. The study verified that RCS pressurization is maximized by under-estimating the affected steam generator liquid mass corresponding to the initiation of heat transfer degradation (i.e., over-estimating the rate of heat transfer decrease). The original methodology took a simplistic and clearly conservative approach by assuming heat transfer degradation was instantaneous upon steam generator dryout. However, this approach is modified in order to more realistically predict the behavior.

A gradual heat transfer reduction is expected as the steam generator tubes are exposed to increasing void fractions which force the tubes from the normal nucleate boiling heat transfer regime into transition boiling and eventually into liquid deficient heat transfer. Transition boiling is anticipated when the local void fraction exceeds 0.9 (Reference 9). Liquid deficient heat transfer develops when local qualities approach 0.9. Under full power conditions and utilizing the steam generator model documented in References 5 through 8, the onset of these heat transfer regimes corresponds to steam generator liquid inventories of approximately 70,000 lbm and 35,000 lbm, respectively.
However, the referenced model conservatively ignores the transition boiling regime, thereby delaying heat transfer degradation until fluid conditions correspond to liquid deficient heat transfer. Thererfore, the modified treatment of steam generator heat transfer behavior is conservative, since it under-estimates the liquid mass associated with the initiation of heat transfer degradation.

Steam Generator Low Water Level Trip

As discussed in Section 15B.3 of Ref. 4, the original loss of feedwater inventory event method credited low water level trip in the ruptured steam generator only after its liquid inventory had been depleted. This assured conservative treatment of low level trip even if the loss of feedwater inventory event caused rapid steam generator depressurization (i.e., large breaks) and consequent swelling of the downcomer level due to flashing of the downcomer liquid. However, for sufficiently small breaks the steam generator pressure remains constant or increases prior to reactor trip and no downcomer level swell will occur due to flashing. Therefore, in the reanalysis of small break loss of feedwater inventory events steam generator low water level trip is credited with a larger liquid inventory remaining.

For the steam generators, the low level trip setpoint corresponds to a downcomer liquid level of approximately 24 feet above the tube sheet and a liquid inventory of over 70,000 lbm under full power conditions (based on the reference steam generator model). However, the reanalysis of small break loss of feedwater inventory events conservatively delays low level trip until heat transfer degradation begins with approximately 35,000 lbm of liquid remaining in the ruptured steam generator.

The NSSS response to the small break loss of feedwater inventory event with the limiting single failure and offsite power available, was modeled using the CESEC computer program described in Section 15.0. In addition, the input to the CESEC code was modified to account for the steam generator low level trip and heat transfer degradation methodology described in the previous paragraphs.

Input parameters and initial conditions

The input parameters and initial conditions used to analyze the NSSS response are discussed in Section 15.0. The initial conditions for the principal process variables were varied to determine the set of initial conditions shown in Table /

In addition to conservatively delaying steam generator low level trip coincident with the assumed heat transfer degradation, the initial primary system pressure was adjusted to achieve, where possible, a coincident reactor trip signal on high pressurizer pressure. This maximizes the primary pressurization potential of the small break loss of feedwater inventory event, by maximizing the primary system pressure at the time of the reactor trip.

As a result of the evaluation method applied to the loss of feedwater inventory analysis, the only mechanisms for mitigation of the reactor coolant system (RCS) pressurization are the pressurizer safety valves, the reactor coolant flow and the main steam safety valves. The last two influence the RCS-to-steam generator heat transfer rate.

There are no credible failures which can degrade pressurizer safety valve or main steam safety valve capacity. Nor are there any credible failures which can reduce steam flow to the ruptured steam generator.[1] A decrease in RCS to steam generator heat transfer due to reactor coolant flow coastdown can only be caused by a failure to fast transfer to offsite power or a loss of offsite power following turbine trip (i.e., two or four pump coastdown, respectively). Because offsite power is assumed to be available for this analysis, the failure to fast transfer is assumed following the turbine trip. This results in the coastdown of two reactor coolant pumps in diagonally opposite loops.

A spectrum of small breaks, of size less than or equal to 0.20 ft^2, were analyzed using the methodology described in the preceeding paragraphs to determine the limiting break size. The results of this analysis are provided in Figure 1 which plots maximum primary pressure vs. break size. As can been seen, the limiting break size is the 0.20 ft^2 break.

The reason that the largest break produces the most adverse pressurization is due to the more rapid degradation of heat transfer in the ruptured steam generator. The rate of heat transfer degradation is a major factor that determines the primary coolant pressurization of the event (i.e., the more rapid the reduction in steam generator heat transfer, the greater the primary pressurization). As was previously stated, heat transfer degradation is conservatively assumed to begin when the ruptured steam generator inventory decreased to 35,000 lbm. The larger break sizes require a shorter time interval to deplete this remaining inventory, resulting in a more rapid heat transfer degradation, and greater primary coolant pressurization.

Detailed results of this limiting break size are presented in the following section.

RESULTS

The dynamic behavior of the important NSSS parameters following the small break loss of feedwater inventory event with the failure to fast transfer to offsite power following turbine trip is presented in Figures 2 - 9. The sequence of events provided in Table 2 summarizes the important results of this event

(1) It should be noted that the coincident occurrences (failures) considered in Chapter 15 do not include spurious independent failures, only consequential failures and pre-existing failures. Accordingly, spurious closure of a main steam isolation valve is not considered credible during the loss of feedwater inventory event.

A 0.20 ft^2 rupture in the main feedwater line is assumed to instantaneously eliminate feedwater flow to both steam generators, and establish critical flow from the generator nearest the break at an initial rate of 1979 lbm/sec. This causes a decrease in steam generator liquid mass as shown by Figure 9.

The break discharge enthalpy is assumed to remain that of saturated liquid until the ruptured steam generator empties, at which time saturated vapor enthalpy is assumed.

The absence of subcooled feedwater flow causes a constant heatup and pressurization of the steam generators during the first 26.6 seconds which reduces the primary-to-secondary heat transfer rate. Rising primary coolant temperatures and pressures result. Due to the temperature reactivity feedback during this period core power is reduced from an initial value of 102% to 99.8% at 26.6 seconds.

At 26.6 seconds the ruptured steam generator produces a low water level reactor trip signal. This reactor trip signal is coincident with a high pressurizer pressure trip signal. Also at this time, heat transfer in the ruptured steam generator begins to degrade due to insufficient inventory. This degradation initiates a rapid heat up and pressurization of the reactor coolant system. At 27.5 seconds the reactor trip breakers open followed by an assumed instantaneous turbine trip. Immediately following turbine trip, the failure to fast transfer to offsite power occurs, resulting in the coastdown of two reactor coolant pumps. These occurrences further aggravate the primary pressurization.

Closure of the turbine leaves the pipe break as the only steam relief path, thereby reducing the energy flow from the intact steam generator below that of the primary-to-secondary heat transfer rate. The resulting steam generator pressurization reduces the primary-to-secondary temperature difference. In addition, the loss of reactor coolant flow following the loss of electrical power to two pumps decreases the heat transfer coefficient of the coolant in the steam generator tubes. A significant heat transfer reduction occurs.

Compression of the pressurizer steam volume due to the high insurge flow raises the pressure to the safety valve setpoint at 28.3 seconds. Thereafter, every increase in the surge flow causes a slight pressurization which opens the safety valves such that their volumetric discharge rate matches that of the insurge. At 30.2 seconds, the surge line flow reaches its maximum value of 1458 lbm/sec.

At this point in time, the reactor coolant system pressure is at a maximum of 2712 psia. Also, the increased pressure establishes a surge line pressure gradient which provides sufficient flow to allow the reactor coolant to expand under the existing heatup with no further pressurization. The rate of heatup decreases subsequent to core heat flux decay, causing primary pressures to drop.

At 30.0 seconds the main steam safety valves opened stabilizing the secondary side temperature and allowing the rising primary coolant temperature to develop greater heat transfer to the intact steam generator. The intact generator is forced to a maximum of 1342 psia at 33.8 seconds before the heat transfer begins to decrease. The core-to-steam generator heat rate mismatch is reduced sufficiently by 37.4 seconds to allow closure of the pressurizer safety valves, and the reactor coolant system enters a cooldown. Under the influence of steam blowdown through the ruptured steam generator to the break, the cooldown proceeds even after the steam generator safety valves close.

After this point, a main steam isolation signal is generated on low steam generator pressure which closes the main steam isolation valves, decoupling the intact steam generator from the ruptured steam generator and the break. The intact steam generator repressurizes, thereby reducing its heat transfer and eventually causing a primary system heatup. With the main steam safety valves re-opening, the primary-to-secondary heat inbalance is eliminated shortly thereafter. The NSSS enters into a quasi-steady state with a very gradual cooldown and depressurization due to decreasing core decay heat and with emergency feedwater flow maintaining an adequate liquid inventory within the intact steam generator for heat removal. By 1800 seconds the operator initiates a controlled cooldown to shutdown cooling utilizing the atmospheric dump valves.

CONCLUSION

This evaluation shows that the plant response to the limiting small feedwater line break event with the most adverse single failure with offsite power available produces a maximum RCS pressure which is within 110% of design (2750 psia).

References

1. "USNRC Standard Review Plan, Section 15.2.8, Feedwater System Pipe Breaks Inside and Outside Containment (PWR)", NUREG-75/087, November 24, 1975.

2. R.E. Henry, H.K. Fauske, "The Two Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes", Journal of Heat Transfer, Transactions of the ASME, May, 1971.

3. "Response to NRC Round One Question 440.42 on the CESSAR-FSAR".

4. CESSAR Final Safety Analysis Report, Docket STN-50-470F.

5. CENPD-107 Supplement 1, "ATWS Model modification to CESEC," September 1974. (Section 3.0).

6. CENPD-107 Supplement 1, Amendment 1-P, "ATWS model modifications to CESEC," November 1975. (Section 3.3).

7. CENPD-107 Supplement 3, "ATWS model modification to CESEC," August 1975. (Sections 240.8, 240.11 and 240.9).

8. CENPD-107 Supplement 4, "ATWS model modification to CESEC," December 1975. (Section 1.6, 1.8 and 4.2).

9. Forced Convection Boiling Studies, Final Report on Forced Convection Vaporization Project
V.E. Schrock and L.M. Grossman, TID-14632 (1959).

TABLE 1

ASSUMPTIONS FOR THE REANALYSIS OF THE LIMITING SMALL BREAK

LOSS OF FEEDWATER INVENTORY EVENT

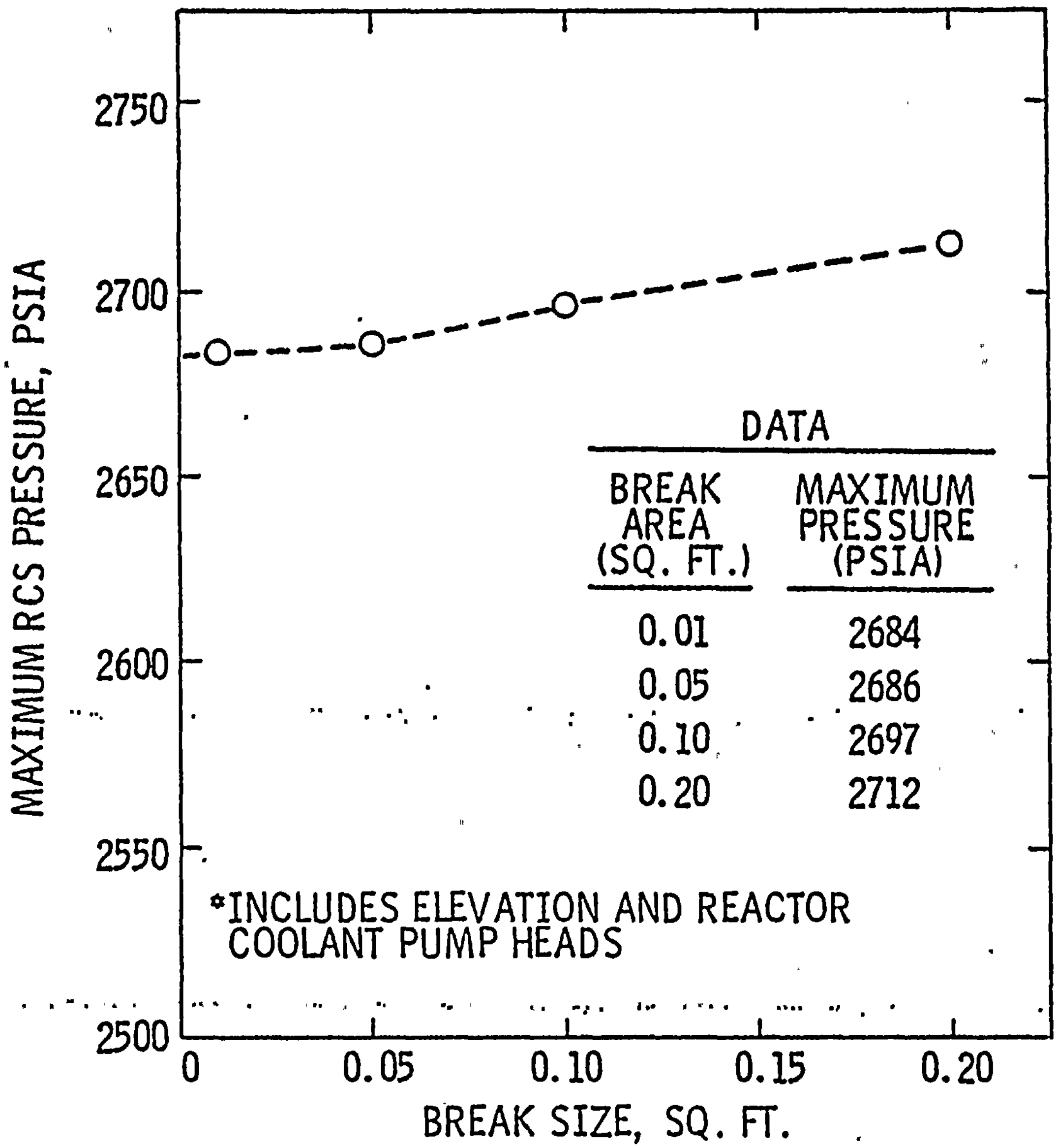
Parameter	Assumed Value
Core Inlet Temperature, ^{o}F	560
Core Mass Flowrate, 10^6 lbm/hr	164.9
Reactor Coolant System Pressure, psia	2115
Steam Generator Pressure, psia	1026
CEA Worth for Trip, 10^{-2} $\Delta\rho$	-10.0
Pressurizer Safety Valves Rated Flow, lbm/hr	460,000
Initial Pressurizer Liquid Volume, ft^3	1120
Initial Steam Generator Inventory, lbm	173,000
Feedwater Pipe Break Area, ft^2	0.20
Steam Bypass Control System	Manual
Pressurizer Pressure Control System	Manual
Pressurizer Level Control System	Manual

TABLE 2

SEQUENCE OF EVENTS FOR THE REANALYSIS OF THE LIMITING SMALL BREAK

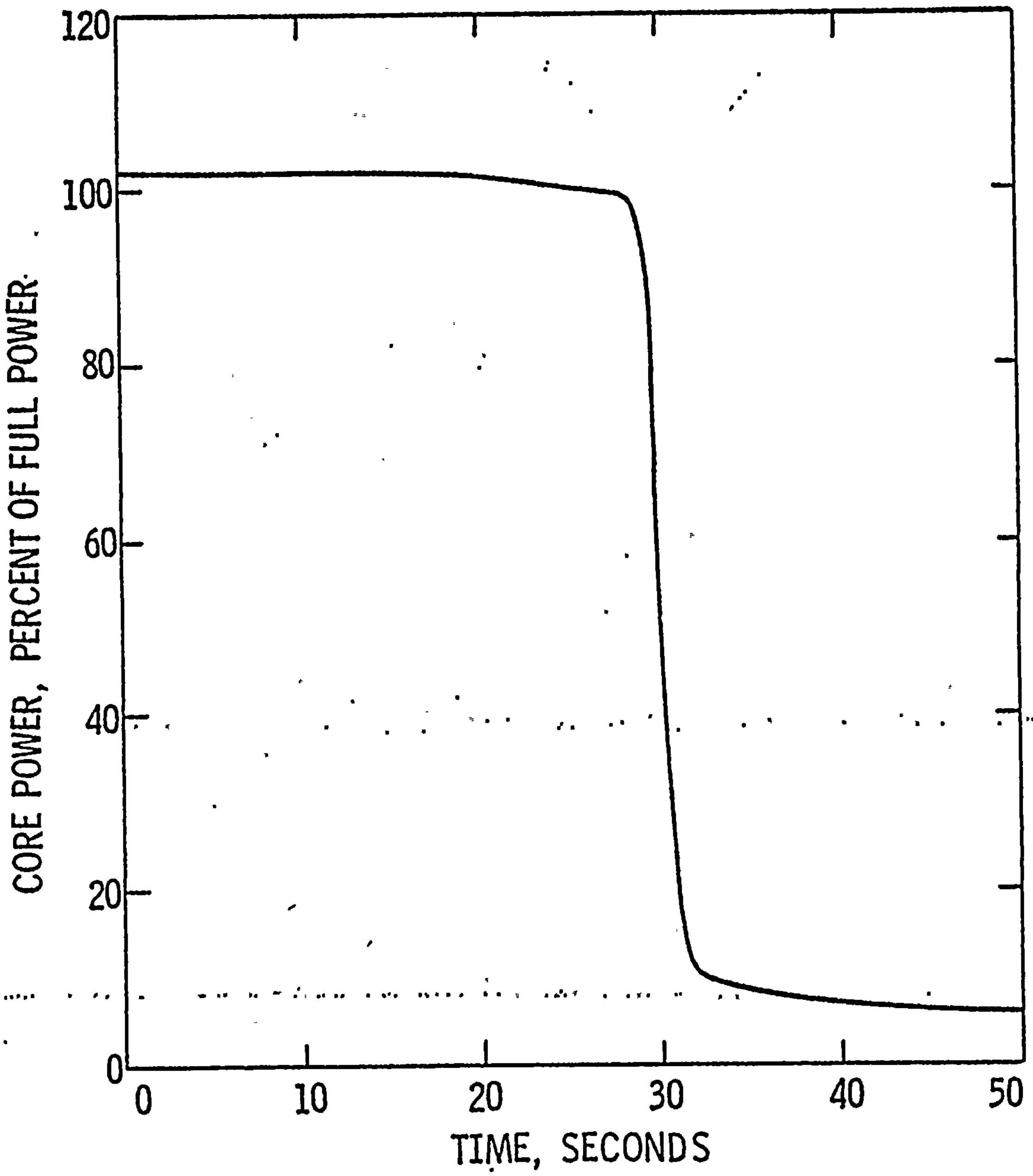
LOSS OF FEEDWATER INVENTORY EVENT

Time (sec)	Event	Setpoint or Value
0.0	Rupture in the Main Feedwater Line, ft^2	0.20
0.0	Complete Loss of Feedwater to Both Steam Generators	----
0.0	Initial Steam Generator Break Flow, lbm/sec	1979
26.0	High Pressurizer Pressure Trip Condition Reached, psia	2475
26.6	High Pressurizer Pressure Trip Signal Generated	----
26.6	Low Level Trip Signal in Ruptured SG	----
26.6	Heat Transfer Degradation in Ruptured SG Begins	----
27.5	Reactor Trip Breakers Open	----
27.5	Turbine Trip on Reactor Trip	----
27.5	Failure to Fast Transfer - Two Reactor Coolant Pumps Coast Down	----
27.8	CEAs Begin to Drop into Core	----
28.3	Pressurizer Safety Valves, psia	2525
30.0	Main Steam Safety Valves Open	1282
30.2	Maximum Surge Line Flow, lbm/sec	1458
30.2	Maximum RCS Pressure, psia	2712
33.8	Maximum Steam Generator Pressure, psia	1342
36.8	Ruptured SG Dries Out	----
37.4	Primary Safety Valves Close, psia	2523

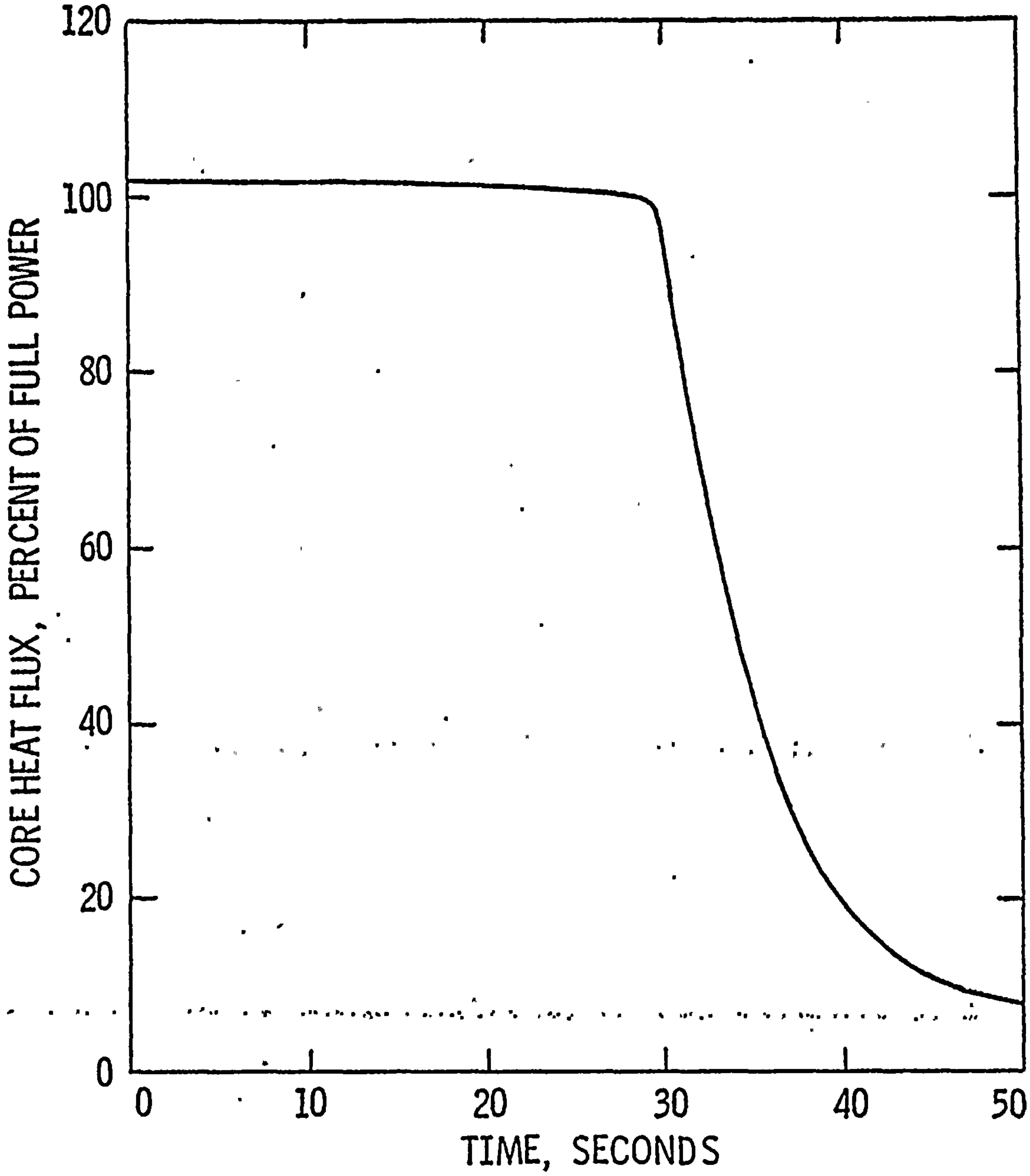


REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS
MAXIMUM RCS PRESSURE* vs BREAK AREA

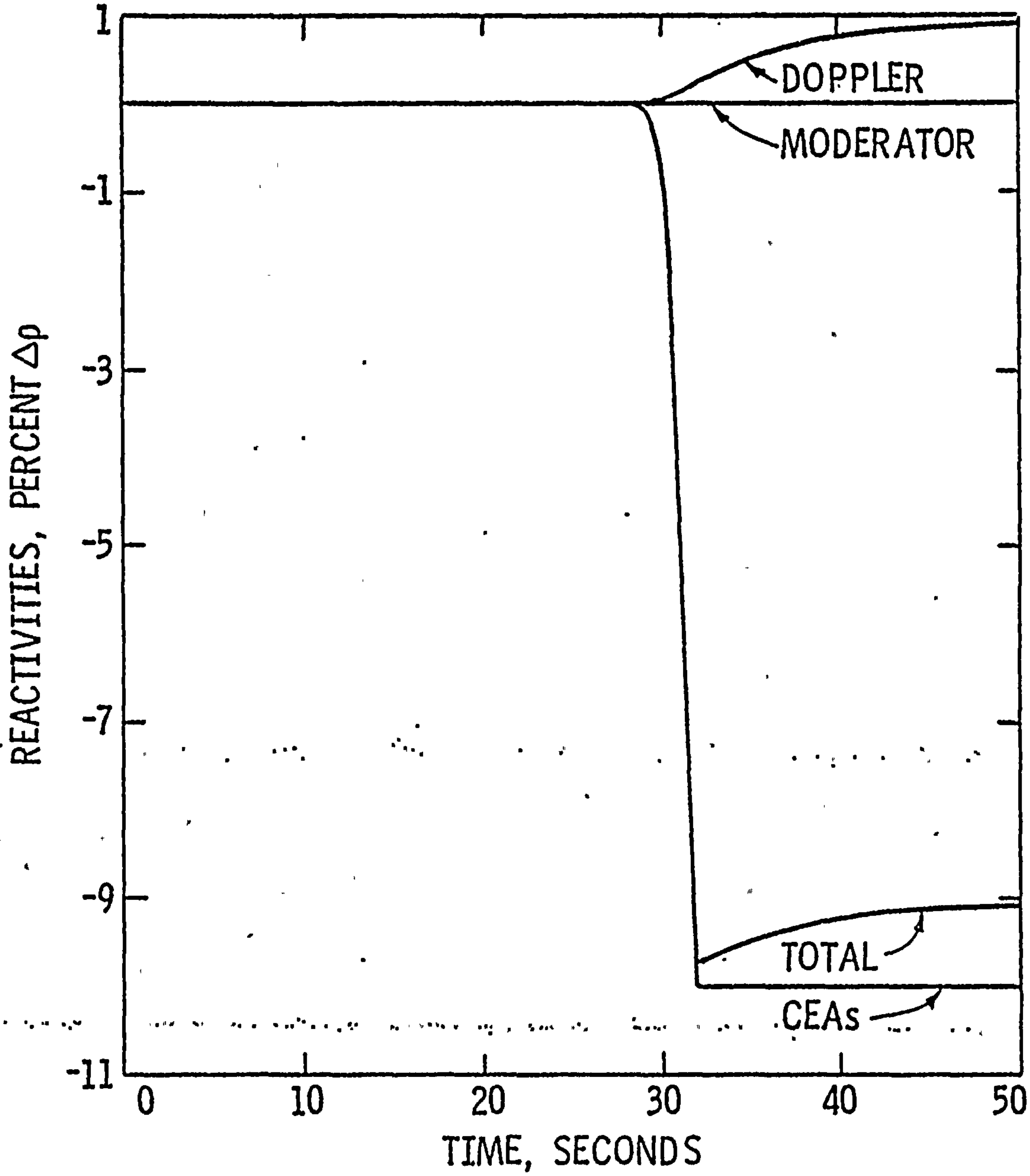
Figure 1



REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE CORE POWER vs TIME	Figure 2

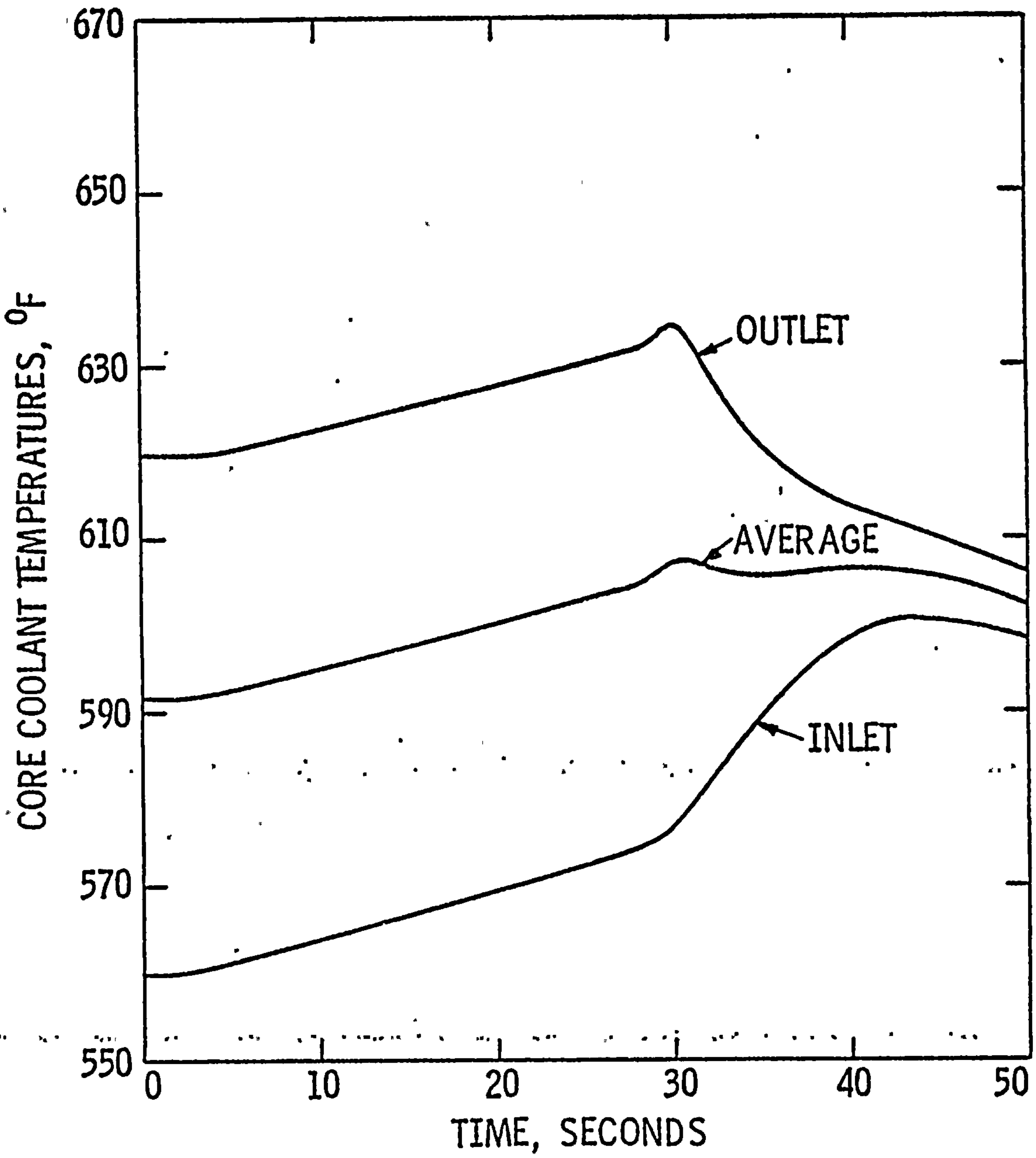


REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE CORE HEAT FLUX vs TIME	Figure 3



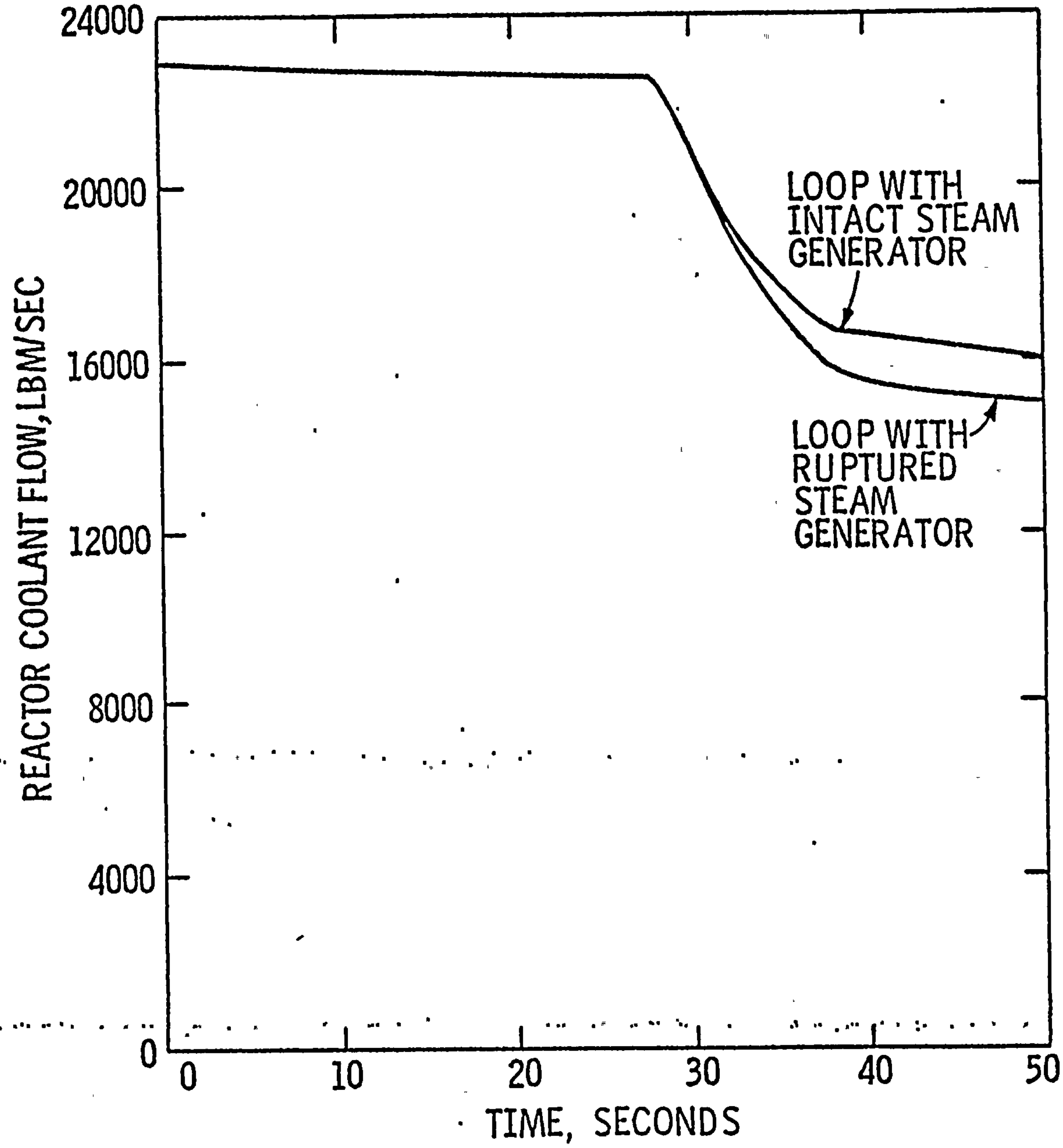
REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE REACTIVITIES vs TIME

Figure 4

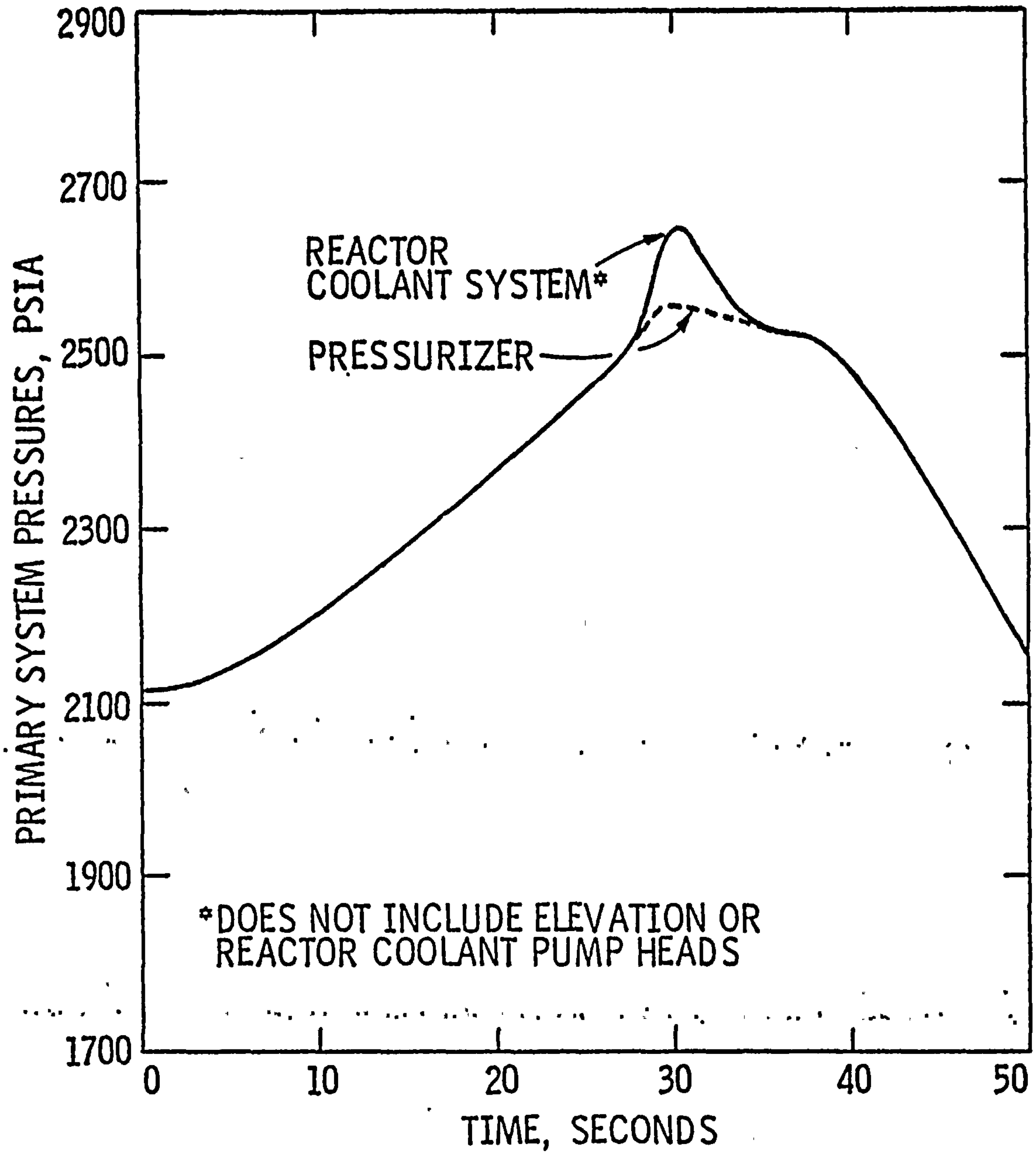


REANALYSIS OF SMALL LOSS OF
FEEDWATER INVENTORY EVENTS - LIMITING CASE
CORE COOLANT TEMPERATURES vs TIME

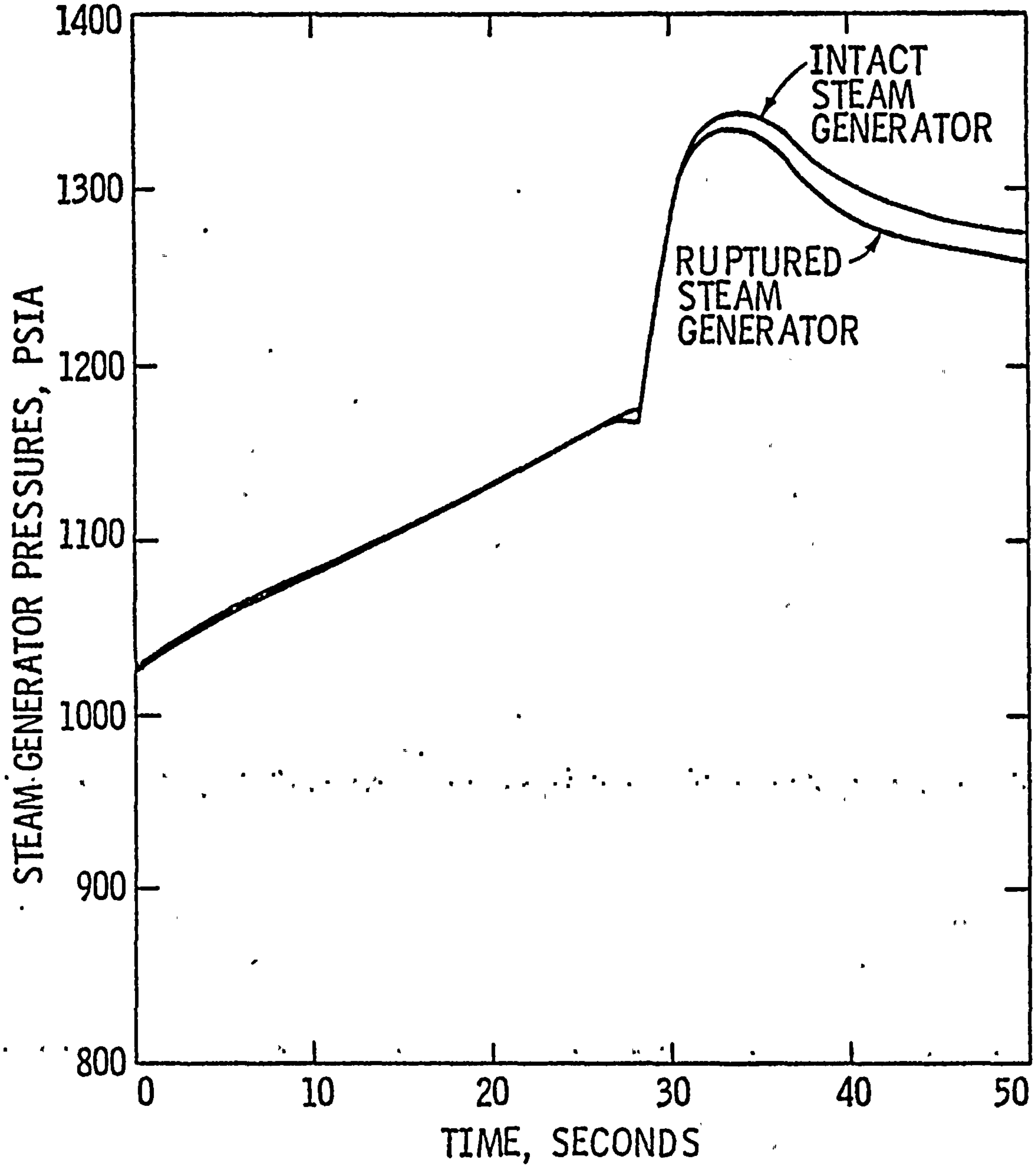
Figure 5



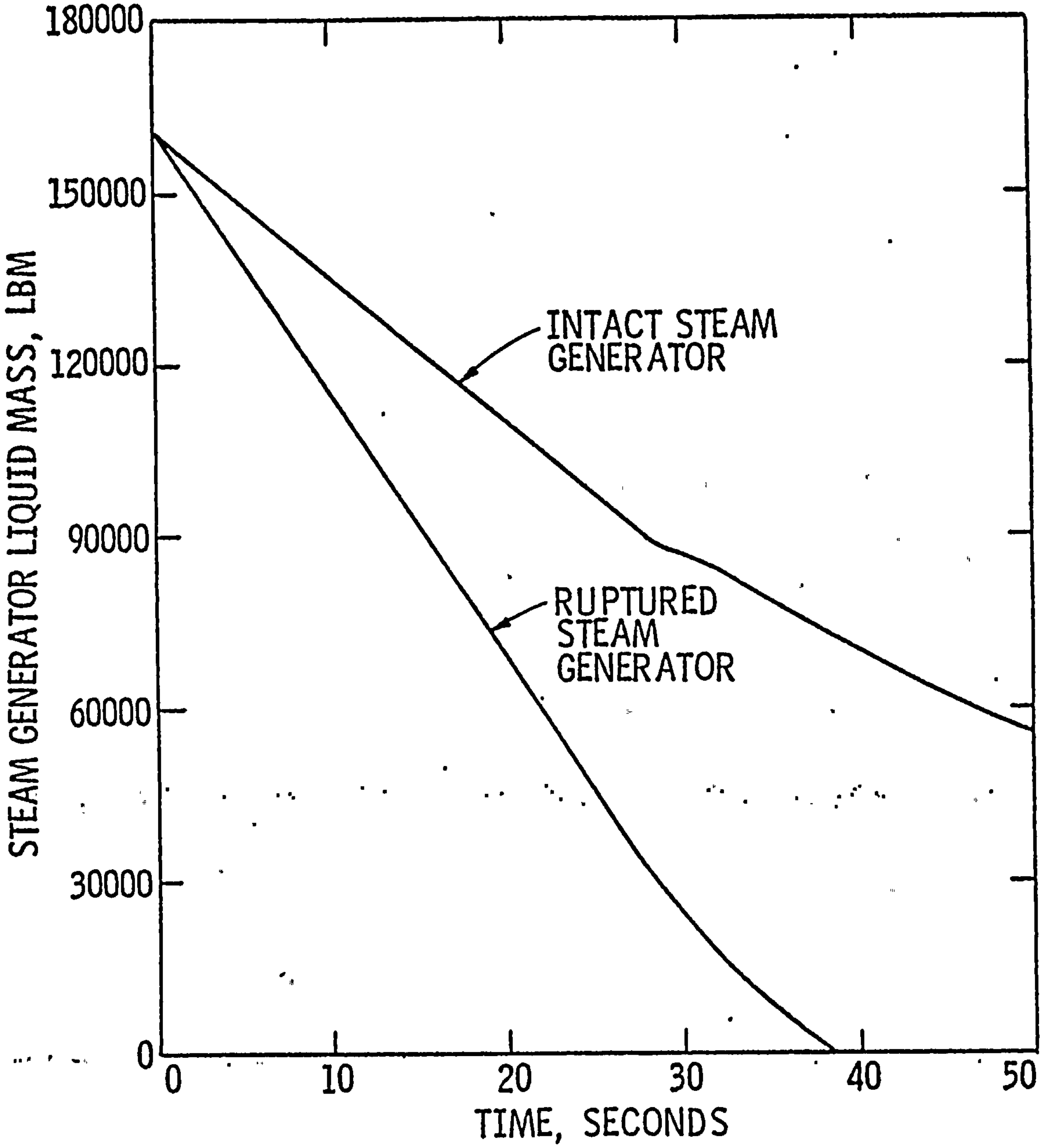
REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE REACTOR COOLANT FLOW vs TIME	Figure 6



REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE PRIMARY SYSTEM PRESSURES vs TIME	Figure 7



REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE STEAM GENERATOR PRESSURES vs TIME	Figure 8



REANALYSIS OF SMALL LOSS OF FEEDWATER INVENTORY EVENTS - LIMITING CASE STEAM GENERATOR LIQUID MASS vs TIME	Figure 9