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UHRIG,R.G. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Forwards response to request for analysis of limited small feedwater line break w/assumption of single failure & availability of offsite power,to verify that sys pressures will not exceed 120% of design pressure.

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	NRR/DSI/CSB	09	1	1	NRR/DSI/ICSB	16	1	1
	NRR/DSI/METB	12	1	1	NRR/DSI/PSB	19	1	1
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Florida Power & Light Co.

Division of Licensing

SUBJECT: Forwards response to request for analysis of limited small bore pipe break w/consideration of single failure & availability of onsite power, to verify that sys pressures will not exceed 120% of design pressure.

[illegible]

[illegible]

BOX 14000, JUNO BEACH, FL 33408

FPL

FLORIDA POWER & LIGHT COMPANY

December 9, 1982
L-82-533

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 Florida Power and Light Company's (FPL) response to the NRC staff request that the limiting small feedwater line break be analyzed assuming a single failure and the availability of offsite power to verify that system pressures will not exceed 120% of design pressure. Previous analyses had demonstrated that a loss of offsite power and the limiting break size did not exceed 120% of design pressure. This NRC request is summarized in Section 15.10.2 of the Safety Evaluation Report.

Attachment 1 provides the SAR change package documenting the results of the small FWLB re-evaluation. As expected, the peak reactor coolant system pressure (2621 psia) is less than 110% of design. The writeup in Attachment 1 refers to CESSAR for the description of the revised methodology. Because CESSAR is in the process of being amended to address small FWLB methodology, and in order to ensure an expeditious review, the pertinent CESSAR material is provided in Attachment 2.

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

Very truly yours,

Jack E. Vessely for

Robert E. Uhrig
Vice President
Advanced Systems and Technology

B001

REU/RJS/cab

Attachment

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

ATTACHMENT 1

15C.9 REANALYSIS OF SMALL BREAK LOSS OF FEEDWATER INVENTORY EVENTS WITH THE LIMITING SINGLE FAILURE AND OFFSITE POWER AVAILABLE

15C.9.1 INTRODUCTION

15C.9.1.1 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.

15C.9.1.2 Background

The loss of feedwater inventory event presented in Section 15.2.5.2 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 1, 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, as is stated in Reference 2, 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 in Section 15.2.5.2 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 1 apply to pipes greater than 4 inches in diameter, the reanalysis need only consider breaks less than approximately 0.1 ft^2. However, this range is conservatively extended up to 0.25 ft^2 (i.e., the same break size presented in Section 15.2.5.2 as the limiting break with the original methodology). Therefore, in the following sections "small" breaks refer to those which are less than 0.25 ft^2.

15C.9.2 METHOD OF ANALYSIS

15C.9.2.1 Mathematical Models

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 15.0.4 and the small break methodology described in the CESSAR-FSAR Appendix 15B (Reference 3).

The methods found in Appendix 15B of the CESSAR-FSAR, when applied to the St. Lucie 2 steam generator design produce a more realistic, but still conservative treatment of heat transfer and downcomer water level behavior in the affected steam generator

(i.e., the generator nearest the pipe break) when compared to the original methods utilized in Section 15.2.5.2. Those original methods conservatively assumed that the affected steam generator heat transfer degradation (due to high fluid quality) and low water level trip were delayed until the generator's liquid inventory was completely depleted. However, use of the steam generator model described in Reference 3 indicates that at least 21,000 lbm of liquid remain in the steam generator prior to heat transfer degradation. The steam generator model also indicates that the steam generator low level reactor trip (a downcomer liquid level of approximately 26 feet above the tubesheet) corresponds to a liquid inventory of over 70,000 lbm. However, the reanalysis of the small break loss of feedwater inventory event conservatively delays the low level trip until heat transfer degradation begins with 21,000 lbm of inventory remaining in the affected steam generator.

15C.9.2.2 Input parameters and initial conditions

The input parameters and initial conditions used to analyze the NSSS response are discussed in Section 15.0.3. The initial conditions for the principal process variables were varied within the range given in Table 15.0-9 to determine the set of initial conditions shown in Table 15C.9-1.

In addition to conservatively delaying steam generator low level trip coincident with the assumed heat transfer degradation, the initial primary system pressure was adjusted within the range specified in Table 15.0-9 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.

To determine the limiting single failure of the loss of feedwater inventory event with offsite power available, Table 15.0-6 was used. There are no single failures identified in this table which can adversely impact the consequences (i.e., pressurization) associated with the loss of feedwater inventory event. 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 affected 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 0.25 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 15C.9-1 which plots maximum primary pressure vs. break size. As can been seen, the limiting break size is the 0.25 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 affected 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 affected steam generator inventory decreased to 21,000 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.

15C.9.3 RESULTS

The dynamic behavior of the important NSSS parameters following the small loss of feedwater inventory event with the failure to fast transfer to offsite power following turbine trip is presented in Figures 15C.9-2 to 9. The sequence of events provided in Table 15C.9-2 summarizes the important results of this event.

A 0.25 ft^2 rupture in the main feedwater line is assumed to instantaneously terminate feedwater flow to both steam generators,

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(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 (See Subsection 15.0.1.5). Accordingly, spurious closure of a main steam isolation valve is not considered credible during the loss of feedwater inventory event.

and establish critical flow from the generator nearest the break at an initial rate of 2219 lbm/sec. This causes a decrease in steam generator liquid mass as shown by Figure 15C.9-9.

The break discharge enthalpy is assumed to remain that of saturated liquid until the affected 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 20.3 seconds which reduces the primary-to-secondary heat transfer rate. Rising primary coolant temperatures and pressures result. The temperature reactivity feedback during this period is extremely small as the assumed MTC is 0.0 $\Delta\rho$ /°F.

At 20.3 seconds the affected 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 affected steam generator begins to degrade due to insufficient inventory. This degradation initiates a rapid heat up and pressurization of the reactor coolant system. At 20.7 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 unaffected 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 22.1 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 23.5 seconds, the surge line flow reaches its maximum value of 1108 lbm/sec.

The reactor coolant system pressure continues to increase to a maximum of 2621 psia at 24.4 seconds. At that time 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 23.3 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 unaffected steam

generator. The unaffected generator is forced to a maximum of 1005 psia at 25.4 seconds before the heat transfer begins to decrease. The core-to-steam generator heat rate mismatch is reduced sufficiently by 29.0 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 affected 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 unaffected steam generator from the affected steam generator and the break. The unaffected 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 unaffected steam generator for heat removal. By 1800 seconds the operator initiates a controlled cooldown to shutdown cooling utilizing the atmospheric dump valves.

15C.9.4 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 for Section 15C.9

1. "Response to NRC Question 440.81 on the St. Lucie 2 FSAR".

2. "Safety Evaluation Report Related to the Operation of St. Lucie Plant, Unit No. 2", NUREG-0843, October 1981. (Section 15.10.2)

3. "Reanalysis of Small Break Loss of Feedwater Inventory Events with the Limiting Single Failure and Offsite Power Available", CESSAR FSAR Appendix 15B (Section 15B.6).

TABLE 15C.9-1

ASSUMPTIONS FOR THE REANALYSIS OF THE SMALL BREAK

LOSS OF FEEDWATER INVENTORY EVENT

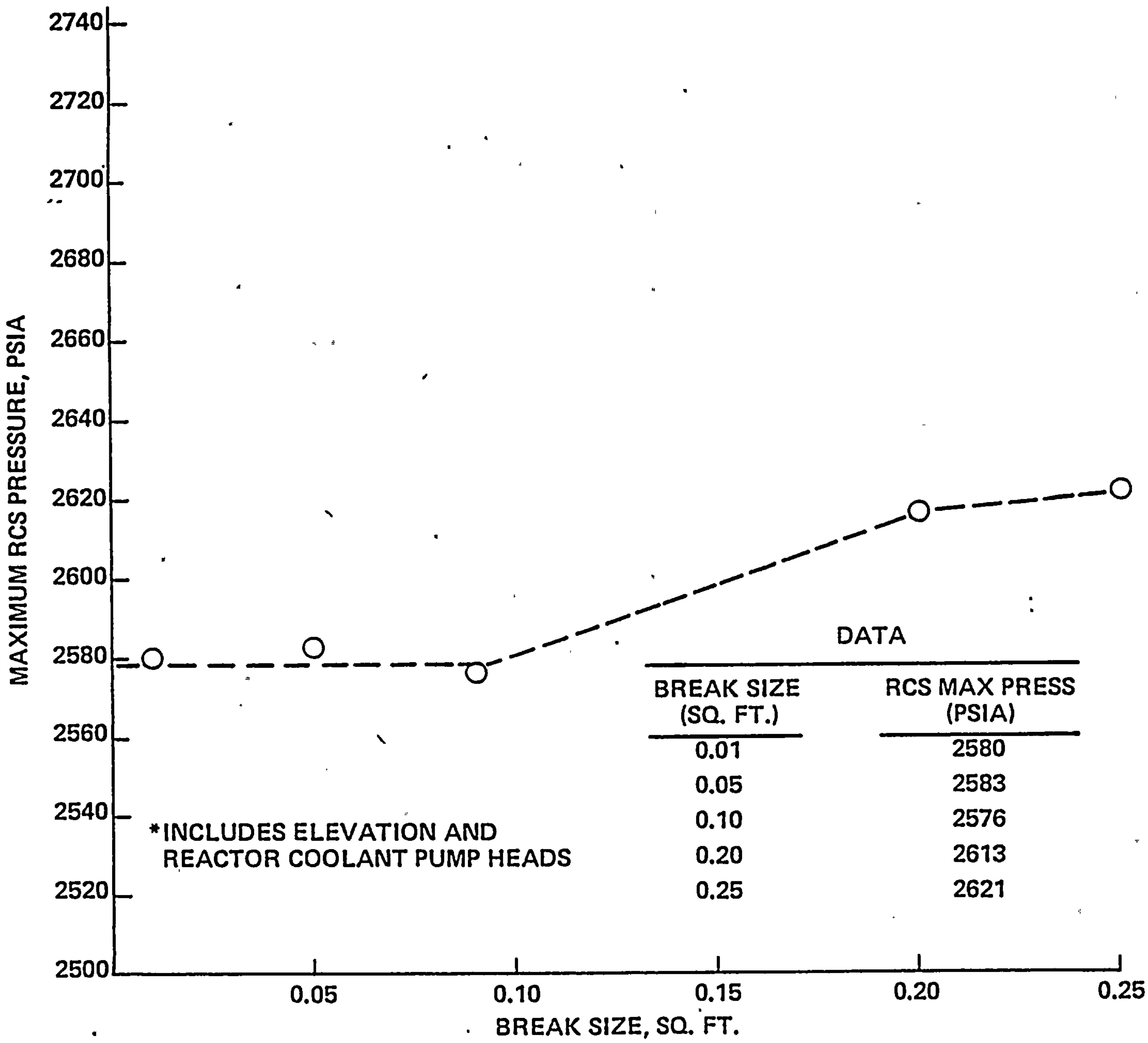
Parameter	Assumption
Initial core power, MWt	2630
Core inlet temperature, °F	551
Core mass flowrate, 10^6 lbm/hr	139
Reactor coolant system pressure, psia	2280
Steam generator pressure, psia	805
Moderator temperature coefficient, 10^{-4} $\Delta\rho$/°F	0.0
CEA worth for trip, 10^{-2} $\Delta\rho$	-5.5
Steam Bypass Control System	Manual
Pressurizer Pressure Control System	Manual
Pressurizer Level Control System	Manual
Power Operated Relief Valves	Manual
Feedwater line break area, ft^2	0.25
Initial steam generator total inventory, lbm	113,550

TABLE 15C.9-2

SEQUENCE OF EVENTS FOR THE REANALYSIS OF THE LIMITING SMALL BREAK

LOSS OF FEEDWATER INVENTORY EVENT

Time	Event	Setpoint or Value
0.0	Rupture in the main feedwater line, ft^2	0.25
0.0	Complete loss of feedwater to both S. G.	------
0.0	Initial steam generator break flow, lbm/sec	2219
19.6	High pressurizer pressure trip condition, psia	2455
20.3	High pressurizer pressure trip signal	------
20.3	Low steam generator water level trip signal in affected steam generator	------
20.3	Heat transfer degradation begins in affected steam generator	------
20.7	Reactor trip breakers open	------
20.7	Turbine trip on reactor trip	------
20.7	Fast transfer failure-two reactor coolant pumps coastdown	------
21.4	CEAs begin to drop into core	------
22.1	Pressurizer safety valves open, psia	2525
23.3	Main steam safety valves open, psia	987
23.5	Maximum surge flow, lbm/sec	1108
24.4	Maximum RCS pressure, psia	2621
25.4	Maximum steam generator pressure, psia	1005
27.4	Affected steam generator dries out	------
29.0	Pressurizer safety valves close, psia	2450
30.0	Minimum pressurizer steam volume, ft^3	290



DATA

BREAK SIZE (SQ. FT.)	RCS MAX PRESS (PSIA)
0.01	2580
0.05	2583
0.10	2576
0.20	2613
0.25	2621

*INCLUDES ELEVATION AND REACTOR COOLANT PUMP HEADS

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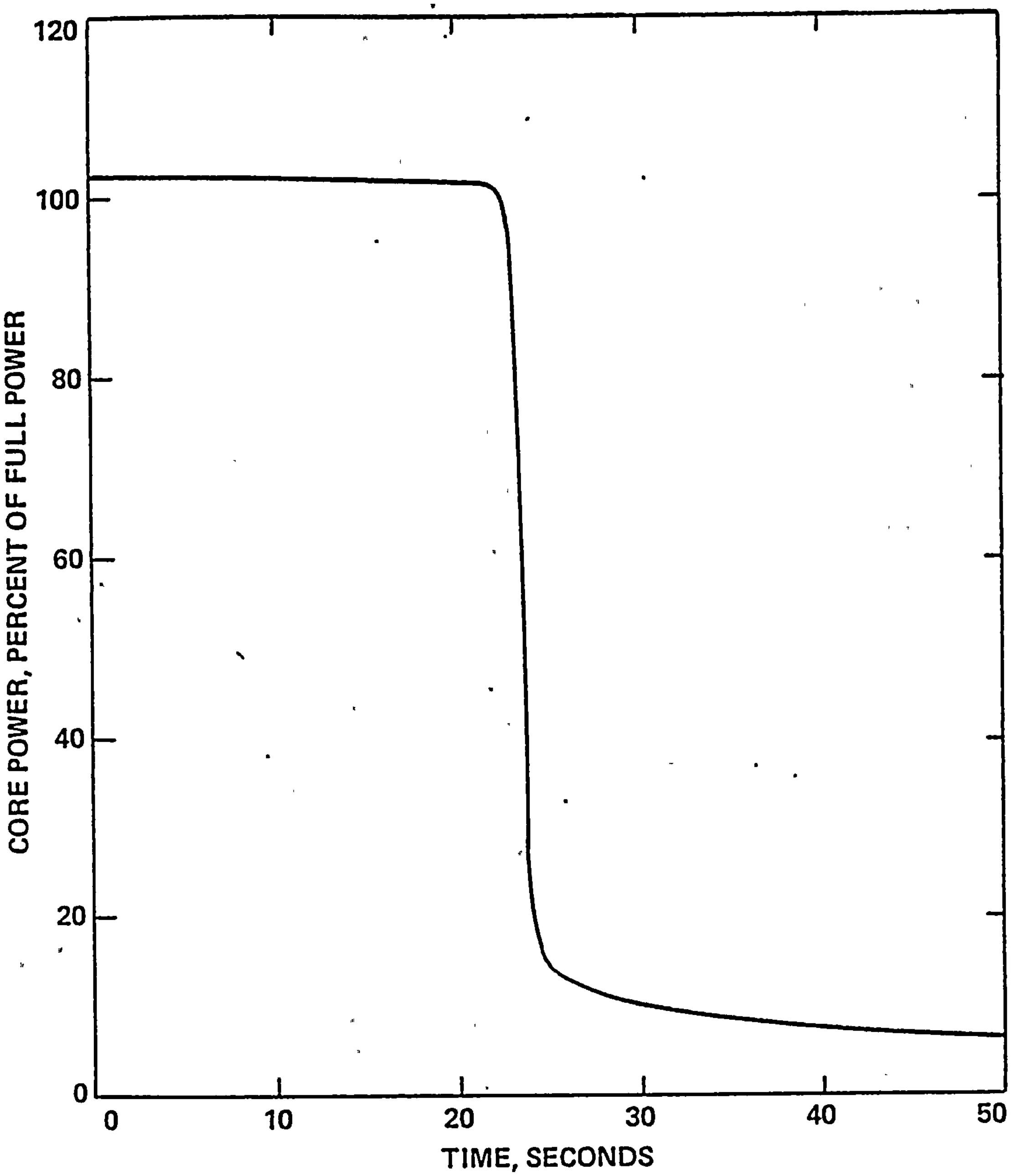
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
MAXIMUM RCS PRESSURE* vs BREAK SIZE
FIGURE 15C.9-1



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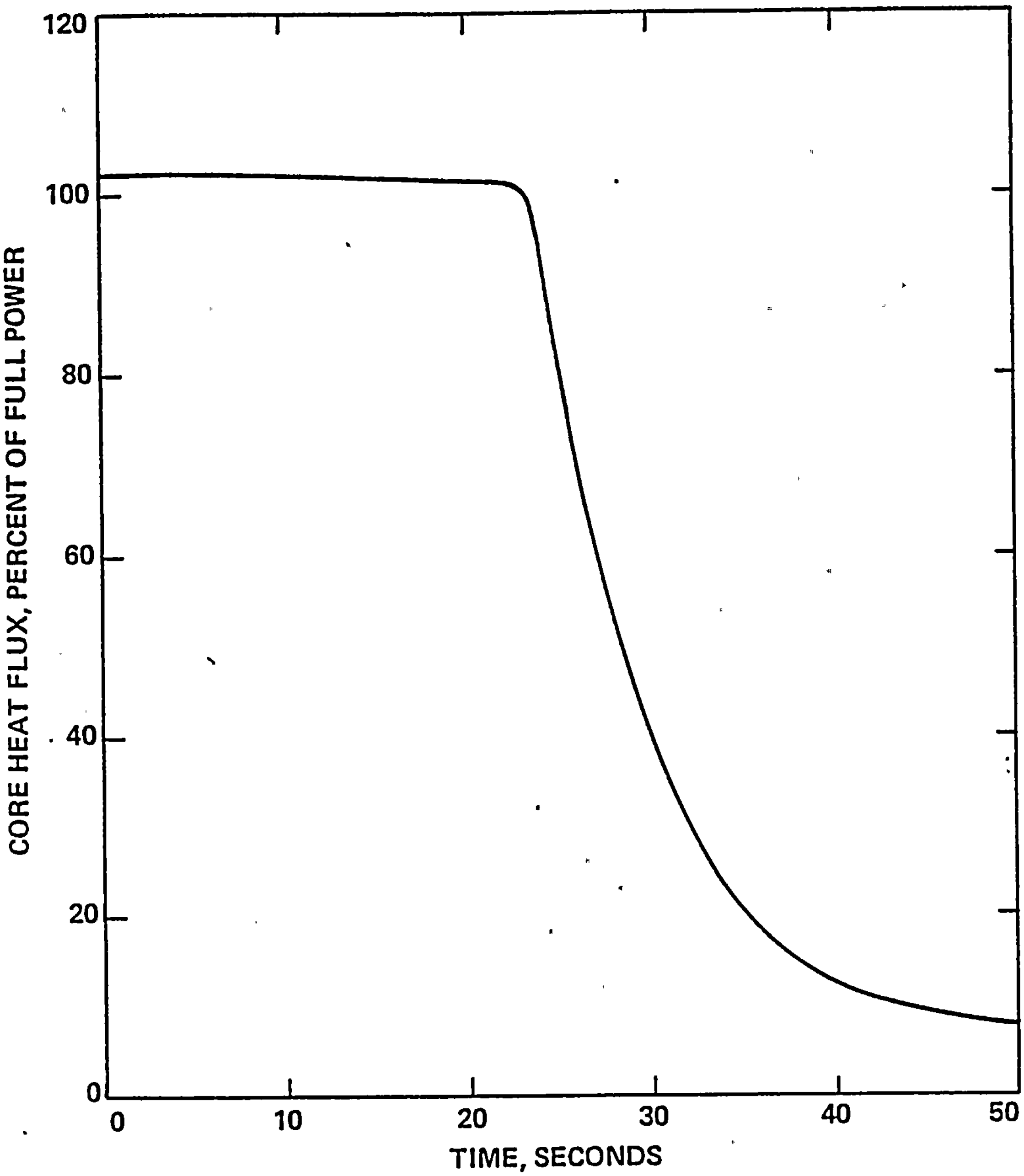
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
CORE POWER vs TIME
FIGURE 15C.9-2



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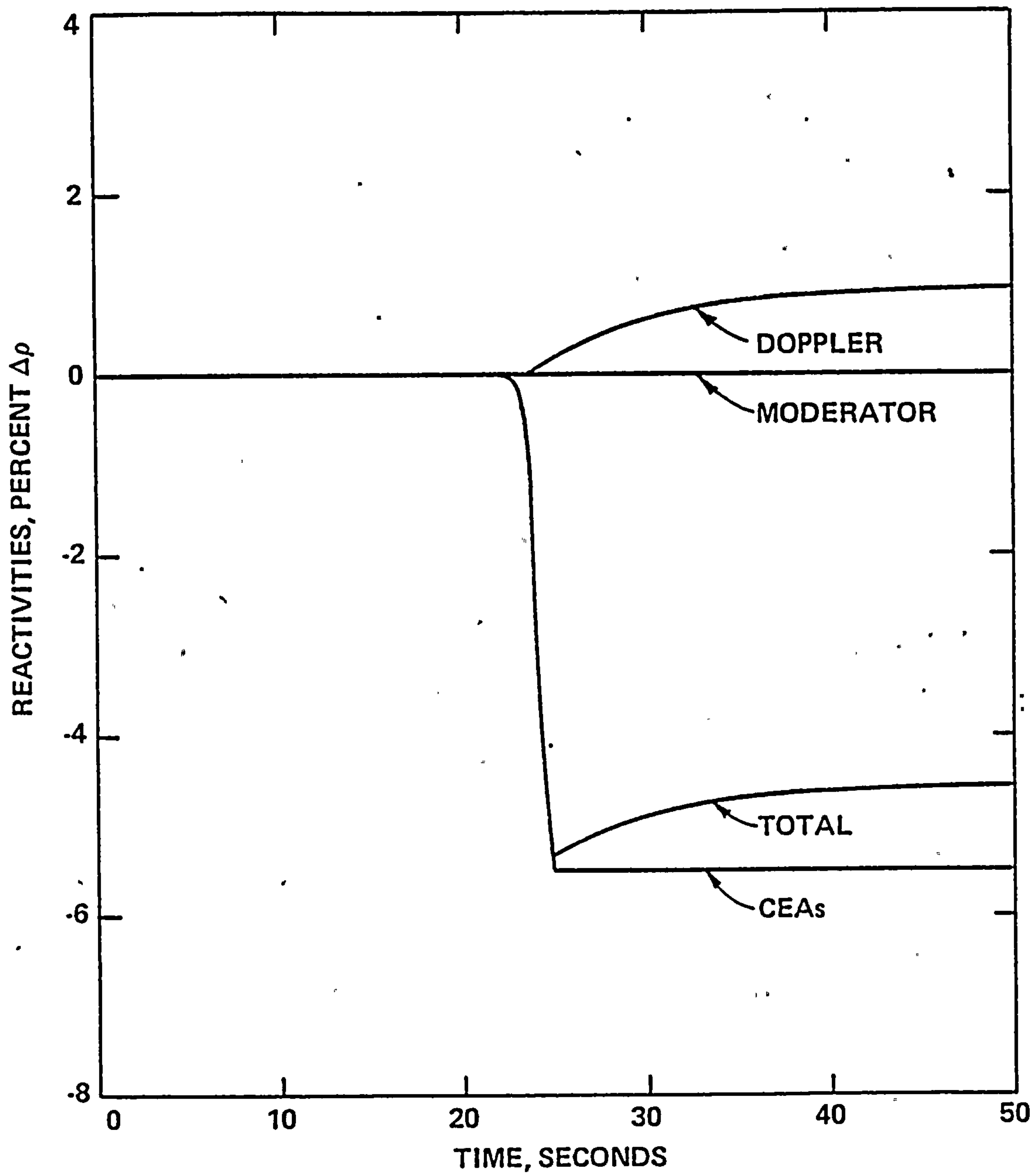
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
CORE HEAT FLUX vs TIME

FIGURE 15C.9-3



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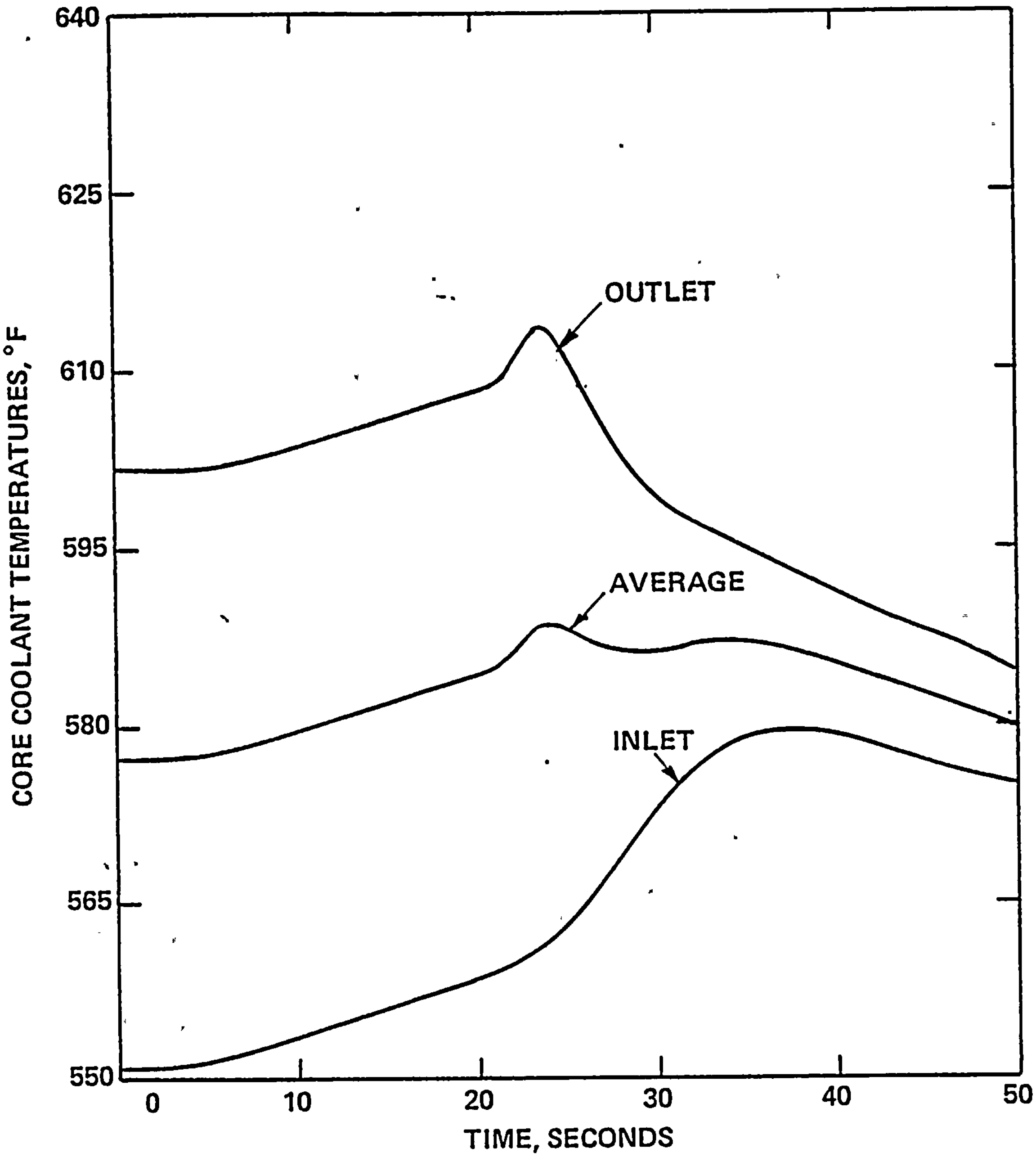
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
REACTIVITIES vs TIME
FIGURE 15C.9-4



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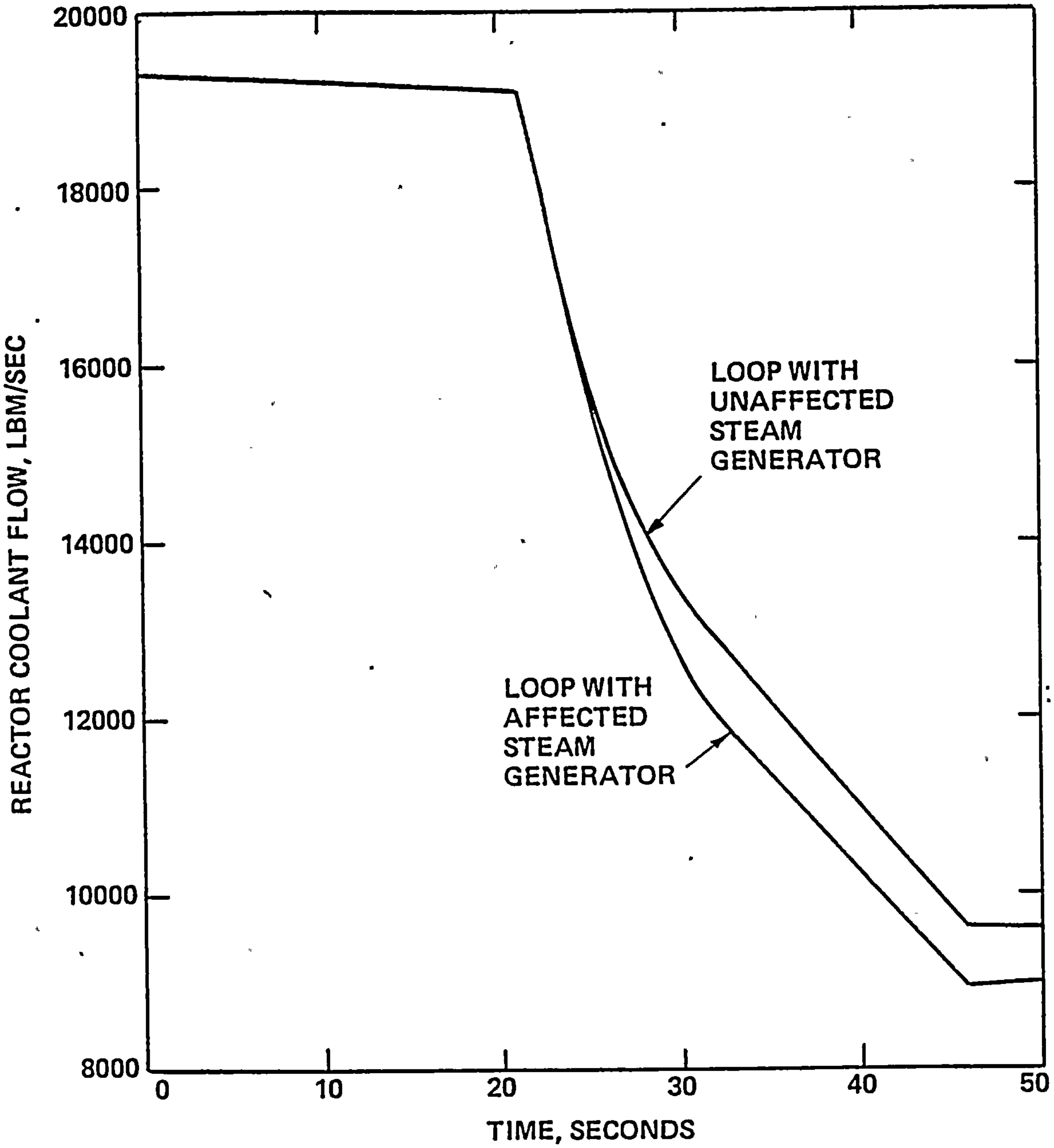
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
CORE COOLANT TEMPERATURES vs TIME
FIGURE 15C.9-5



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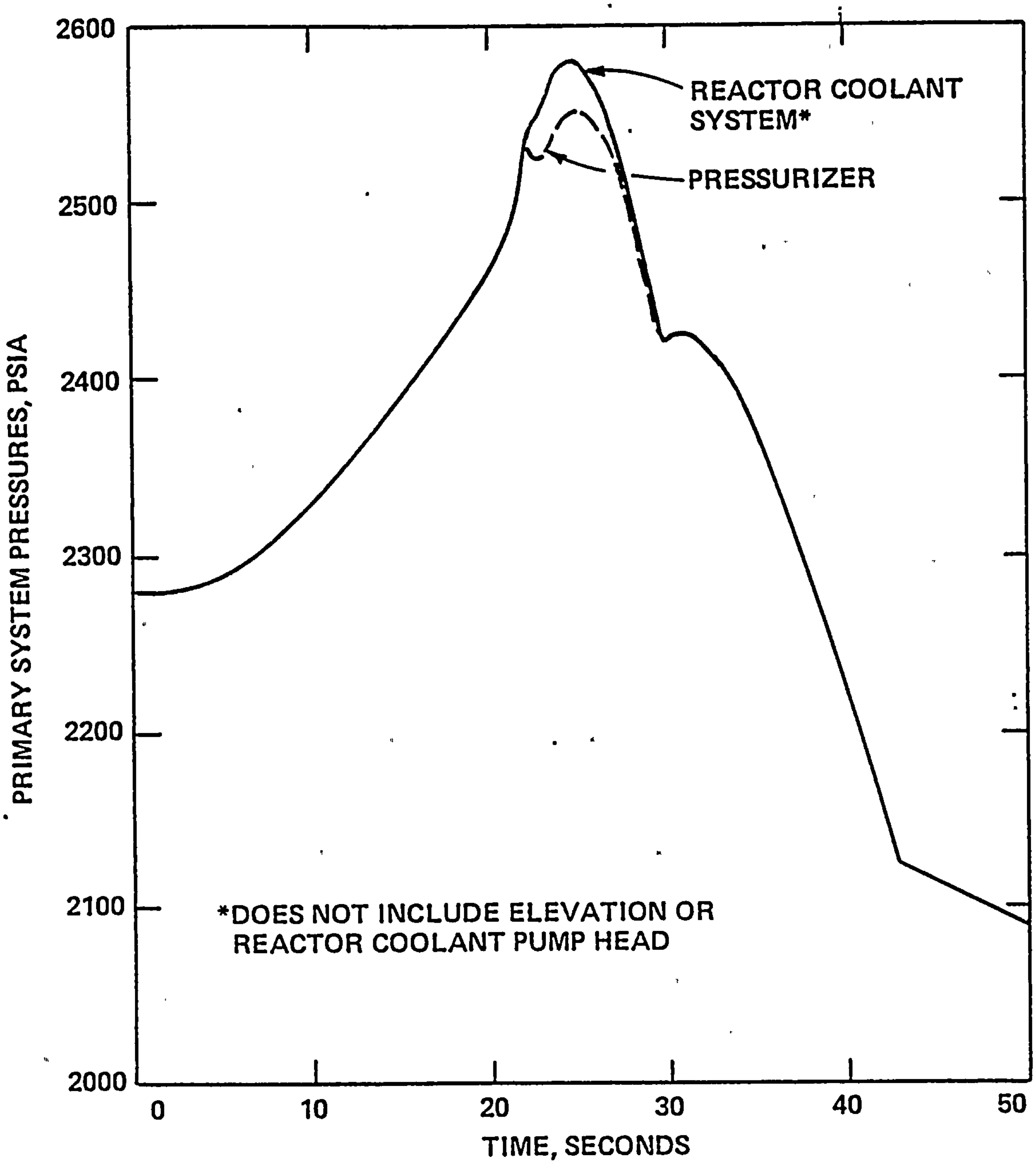
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
REACTOR COOLANT FLOW vs TIME
FIGURE 15C.9-6



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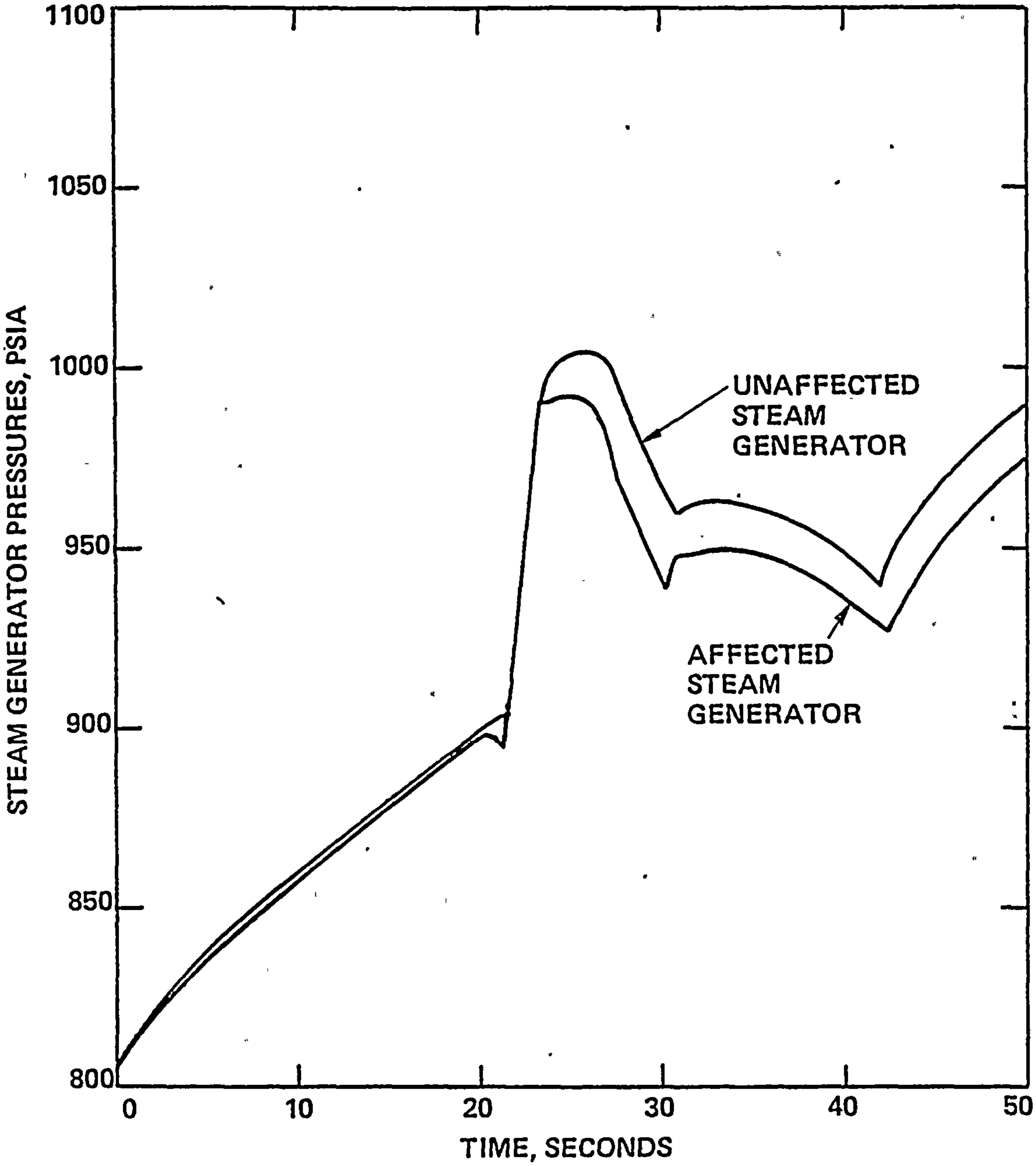
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT PRIMARY SYSTEMS PRESSURE vs TIME FIGURE 15C.9-7



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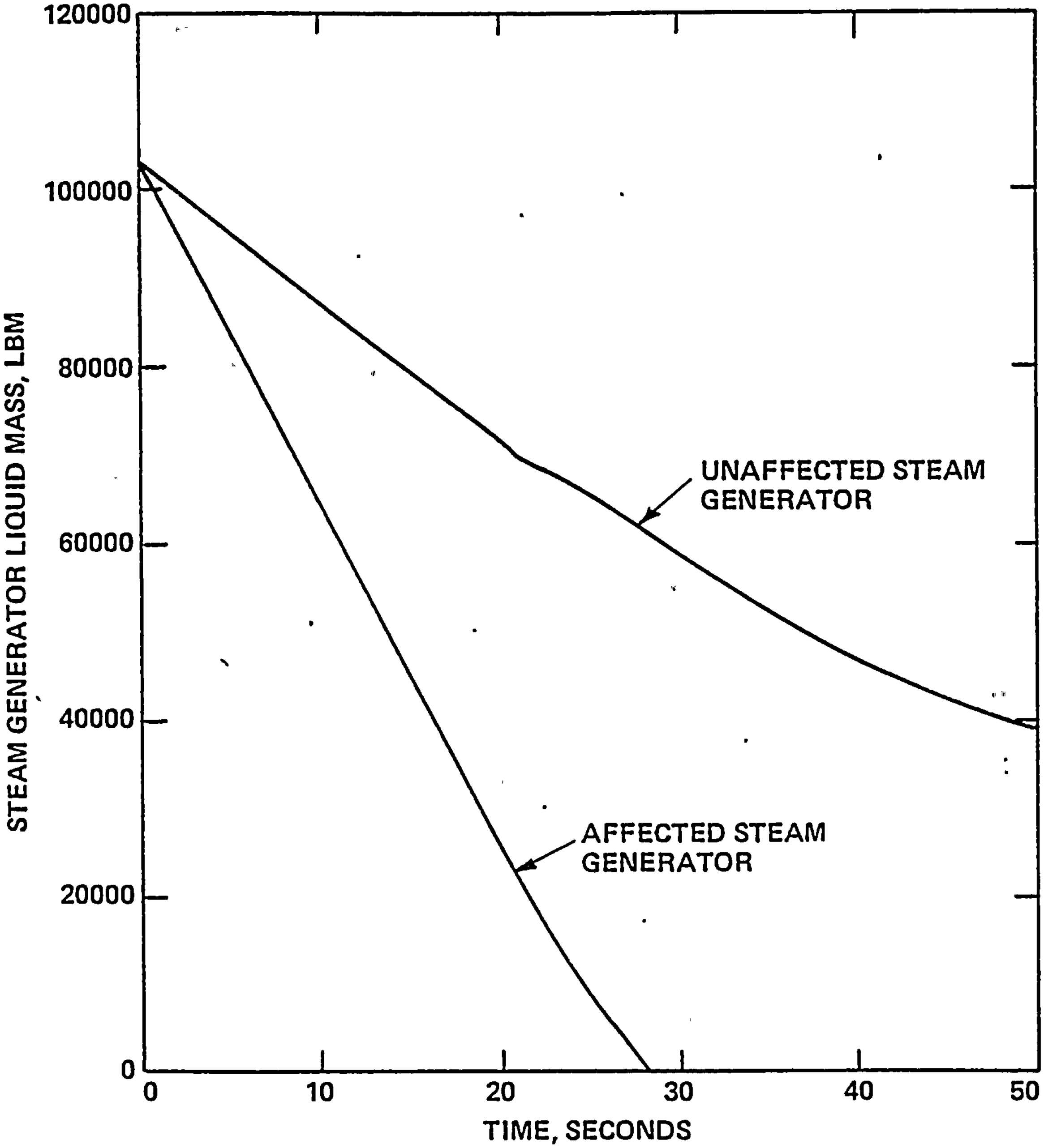
REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT STEAM GENERATOR PRESSURE vs TIME

FIGURE 15C.9-8



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REANALYSIS OF THE SMALL LOSS OF FEEDWATER INVENTORY EVENT
STEAM GENERATOR LIQUID MASS vs TIME

FIGURE 15C.9-9

ATTACHMENT 2

15B.6 REANALYSIS OF SMALL BREAK LOSS OF FEEDWATER INVENTORY EVENTS WITH THE LIMITING SINGLE FAILURE AND OFFSITE POWER AVAILABLE

15B.6.1 INTRODUCTION

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15B.6.1.1 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.

15B.6.1.2 Background

The loss of feedwater inventory event presented in Section 15B.4 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, as is stated in Reference 4, it ~~it is~~ 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 in Section 15B.4 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 in Section 15B.4 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.

15B.6.2 METHOD OF ANALYSIS

15B.6.2.1 Mathematical Models

The methodology used in the reanalysis of small break loss of feedwater inventory events is the same as that applied in Section 15B.4 and described in Section 15B.3 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.

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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 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 for the System 80 design. 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, 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 fowncomer 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 System 80 design 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

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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.

References for Appendix 15B

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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. "Safety Evaluation Report Related to the Final Design Approval of the Combustion Engineering Standard Nuclear Steam Supply System (CESSAR)" NUREG-0852 (Section 15.3.2).

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).