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Omaha Public Power District

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May 15, 1980

Mr. K. V. Seyfrit, Director
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region IV
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

RECEIVED
MAY 19 1980

Reference: Docket No. 50-285

Dear Mr. Seyfrit:

In response to IE Bulletin 80-04, the Omaha Public Power District submitted a letter to the Commission, dated May 8, 1980, providing a commitment and schedule for the analyses required by the bulletin. In accordance with our May 8, 1980, letter, the analysis for reactivity increase from a Main Steam Line Break with continued feedwater is attached.

Sincerely,

W C Jones

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/BJH/TLP:jmm

Attach.

cc: U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Operation Inspection
Washington, D. C. 20555

LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

ATTACHMENT I

The Steam Line Rupture (SLB) event was analyzed for Ft. Calhoun Unit 1, Cycle 6 using reload licensing assumptions and methods except for automatic initiation of auxiliary feedwater flow in 3 minutes from initiation of the event.

The analysis assumed that the event is initiated by a circumferential rupture of a 26-inch (inside diameter) steam line at the steam generator main steam line nozzle. This break size is the most limiting, since it causes the greatest rate of temperature reduction in the reactor core region. The break outside containment cases were not analyzed since these result in less adverse reactivity transient because the break size is smaller due to flow venturis in each steam line. For conservatism no credit was taken for the existence of the venturi flow restrictors in the analyses.

The SLB event was analyzed with the assumption of a three minute delay between the time of transient initiation and time when Auxiliary Feedwater (AFW) flow is delivered to the affected steam generator. This is conservative with respect to the expected time of AFW initiation since the generation of the AFW signal actually occurs at the time of the low steam generator water level trip signal, and AFW flow is initiated three minutes following this signal. The analysis assumes, therefore, that AFW flow is delivered to the steam generator sooner than the flow is actually available resulting in a conservative prediction of the resulting cooldown.

A conservatively high value of the AFW flow was calculated assuming that all auxiliary feedwater pumps are operable. An AFW flow of 10.5% of full power feedwater flow was used in the analysis. This value accounts for pump run-out due to reduced back pressure. In addition, the analysis conservatively assumed that all the AFW flow is fed only to the damaged steam generator.

The analysis conservatively assumed that there is no main feedwater isolation when the Containment Isolation Actuation Signal (CIAS) is actuated. Hence, the main feedwater flow is ramped down to 5% of full power feedwater flow in 60 seconds (A more realistic main feedwater flow would be stopped in 20 seconds). This assumption is conservative because it prolongs the cooldown of the RCS and thus results in a more severe reactivity transient.

The two steam line rupture cases considered in conjunction with automatic initiation of auxiliary feedwater flow are:

1) 2 Loop - Full Load (1530 MWt)
2) 2 Loop - No Load (1 MWt)

Two Loop - Full Load

The Two Loop - 1530 (includes 2% power measurement uncertainties) MWt case was initiated at the conditions listed in Table 1. The Moderator Temperature Coefficient (MTC) of reactivity assumed in the analysis corresponds to end of life, since this MTC results in the greatest positive reactivity change during the RCS cooldown caused by the Steam Line Rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator temperatures covered in the analysis, a curve of reactivity insertion versus temperature, rather than a single value of MTC, is assumed in the analysis. The moderator cooldown curve assumed in given in Figure 1. The moderator cooldown curve given in Figure 1 was conservatively calculated assuming that on reactor scram, the highest worth Control Element Assembly is stuck in the fully withdrawn position.

The reactivity defect associated with fuel temperature decreases is also based on end of life Doppler defect. The Doppler defect based on an end of life Fuel Temperature Coefficient (FTC), in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the Steam Line Rupture event. The uncertainty on the FTC assumed in the analysis is given in Table 1. The β fraction assumed is the maximum absolute value including uncertainties for end of life conditions. This too is conservative since it maximizes the subcritical multiplication and thus, enhances the potential for Return-To-Power (R-T-P).

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip at the maximum allowed power level is 5.81%$\Delta\rho$, assuming that the most reactive CEA is stuck in the fully withdrawn position during a scram.

The analysis conservatively assumed that on Safety Injection Actuation Signal one High Pressure Safety Injection Pump and one Low Pressure Safety Injection pump fail to start. A conservative value for the boron reactivity worth of -1.0%$\Delta\rho$ per 87 PPM was assumed in the analysis. In addition, no credit is taken for any boron injected via charging pumps taking suction from the Boric Acid makeup tanks.

The conservative assumptions on feedwater flow were discussed previously. The feedwater flow and enthalpy as a function of time are presented in Figures 2 and 3 respectively.

Table 2 presents the sequence of events for the full power case initiated at the conditions given in Table 1. The reactivity insertion as a function of time is presented in Figure 4. The response of the NSSS during this event is given in Figures 5 through 9.

The results of the analysis show the affected steam generator blows dry at 69.1 seconds and thus terminates the initial cooldown of the RCS. The peak reactivity attained prior to delivery of Auxiliary Feedwater Flow is -.097%$\Delta\rho$ which occurs at 72.6 seconds. The corresponding peak Return-To-Power due to subcritical multiplication attained prior to delivery of auxiliary feedwater flow is 14% at 73.1 seconds. The delivery of boron via the High Pressure Safety Injection pump inserts negative reactivity and the core power decreases to the decay power level as the reactivity becomes nore negative.

The delivery of auxiliary feedwater flow starting at 180.0 seconds initiates a further cooldown of the RCS which results in more positive reactivity insertion and causes the core to approach criticality. However, the addition of boron via the high and low pressure safety injection pumps (one HPSI and one LPSI pump are assumed operable) terminates the approach to criticality and the core remains subcritical. The peak total reactivity attained following AFW is -.32%$\Delta\rho$ at 418.2 seconds. Since the core never reaches criticality, and, in the absence of subcritical multiplication, there is no Return-To-Power.

The results of the analysis show that the Return-To-Power prior to AFW is less than the Return-To-Power for the 2 Loop-Full power case presented in the FSAR. Since the critical heat flux was not exceeded in the FSAR case, it is concluded that the critical heat flux is not exceeded for the present case. Hence, the consequences of the event with AFW are no more adverse than the 2 loop full power MSLB case presented in the FSAR, without AFW.

Two Loop- No Load

The two loop- no load case was initiated at the conditions given in Table 3. The moderator cooldown curve is given in Figure 1. The cooldown curve corresponds to an end of life MTC. An end of life FTC was also used for the reasons previously discussed in connection with the two loop - 1530 MWt case.

The minimum CEA shutdown worth available is conservatively assumed to be 4.2%$\Delta\rho$, assuming the most reactive CEA is stuck in the fully withdrawn position during a scram. A maximum inverse boron worth of 87 PPM/%$\Delta\rho$ was conservatively assumed for the safety injection during the no load case. The feedwater flow and the enthalpy used in the analysis are presented in Figures 10 and 11 respectively.

Table 4 presents the sequence of events for the 2 loop- HZP case initiated from the conditions given in Table 3. The reactivity insertion as a function of time is presented in Figure 12. The NSSS response during this event is given in Figures 13 to 17.

The results of the analysis show that the affected steam generator blows dry at 121.7 seconds. The peak reactivity attained during this time period of -.27%$\Delta\rho$. The addition of boron from the high pressure safety injection adds negative reactivity and thus the core reactivity becomes more negative.

At 180 seconds the auxiliary feedwater flow is delivered to the affected steam generator. This initiates a further cooldown of the RCS. The cooldown of the RCS inserts more positive reactivity. However, Low Pressure Safety Injection flow is initiated at 102.6 seconds which injects additional boron. The negative reactivity added due to boron injection via the LPSI's more than offsets the positive reactivity inserted by the added cooldown of the RCS. Hence, the core never reaches criticality after initiation of auxiliary feedwater flow.

The two loop- no load case attains a peak total reactivity of -.27%$\Delta\rho$ which occurs before AFW. Hence, the results of 2 loop- no load SLB event with automatic initiation of auxiliary feedwater are no worse than the 2 loop- no load case analyzed for the FSAR, without AFW.

TABLE 1

KEY PARAMETERS ASSUMED IN THE MAIN STEAM LINE BREAK EVENT WITH AUTOMATIC INITIATION OF AUXILIARY FEEDWATER

(2 LOOP - FULL LOAD CONDITION)

Parameters	Units	FSAR Values	Present Analysis Values
Initial Core Power Level	MWt	1420	1530
Initial Core Inlet Temperature	°F	547	547
Initial RCS Pressure	psia	2100.0	2175.0
Initial Steam Generator Pressure	psia	770.0	880.5
Low Steam Pressure Trip Setpoint	psia	478.0	478.0
Safety Injection Actuation Setpoint	psia	1578	1578
High Pressure Safety Injection Flow Delivery	psia	not available	1390.0
Low Pressure Safety Injection Flow Delivery	psia	201*	201
CEA Worth at Trip	$\%\Delta\rho$	-5.0	-5.81
Moderator Cooldown Curve	$\%\Delta\rho$ vs. °F	Figure 1	Figure 1
Doppler Multiplier		1.20	1.15
Inverse Boron Worth	PPM/$\%\Delta\rho$	80.0	87.0
Feedwater Flow	BTU/sec vs. Sec	Figure 2	Figure 2
Feedwater Enthalpy	BTU/lbm vs. Sec	Figure 3	Figure 3

* No credit for Low Pressure Safety Injection was taken in the FSAR analysis.

TABLE 2

Sequence of Events for the Main Steam Line Break Event with Automatic Initiation of Auxiliary Feedwater Flow (Full Load, Two-Loop Condition, Nozzle Break)

Time (sec.)	Event	Safety System Initiated	Setpoint or Value
0.0	Initiation of break	----	----
3.5	Low steam generator Pressure trip signal occurs, MSIS initiated and Main Steam Isolation Valves begin to close.	Reactor Protection System Main Steam Isolation System	478 psia
4.4	Trip breakers open	----	----
6.9	CEAs at 90% Insertion	Reactor Protection System	----
8.4	Complete closure of Main Steam Isolation Valves to terminate blowdown from the intact steam generator	----	----
13.8	Low RCS pressure, SIAS Initiated	Safety Injection System	1578 psia
14.5	Pressurizer empties	----	----
21.8	High Pressure Safety Injection flow Initiated	Safety Injection System	1390 psia
64.4	Main feedwater flow completes ramp down to 5%	----	----
69.1	Affected steam generator liquid inventory depleted and beginning of blowdown of feedwater only	----	----
73.1	Peak return-to-power* occurs with a peak reactivity of -.097%$\Delta\rho$		14%

* return-to-power includes decay heat and subcritical multiplication

TABLE 2 (Continued)

Time (sec.)	Event	Safety System Initiated	Setpoint or Value
135.4	Boron from safety injection reaches core mid-plane	----	----
180.0	Auxiliary Feedwater flow to affected steam generator initiated	----	----
418.2	Low Pressure Safety flow initiated	Safety Injection System	201 psia
418.2	Peak reactivity post auxiliary feedwater delivery	----	$-.32\%\Delta\rho$

TABLE 3

KEY PARAMETERS ASSUMED IN THE MAIN STEAM LINE BREAK EVENT WITH AUTOMATIC INITIATION OF AUXILIARY FEEDWATER

(2 LOOP - NO LOAD CONDITION)

Parameters	Units	FSAR Values	Present Analysis Values
Intial Core Power Level	MWt	1	1
Initial Core Inlet Temperature	°F	532.0	532.0
Initial RCS Pressure	psia	2100.0	2175.0
Initial Steam Generator Pressure	psia	900.0**	900.0
Low Steam Pressure Trip Setpoint	psia	478.0	478.0
Safety Injection Actuation Setpoint	psia	1578	1578
High Pressure Safety Injection Flow Delivery	psia	not available	1390.0
Low Pressure Safety Injection Flow Delivery	psia	201*	201
CEA Worth at Trip	$\%\Delta\rho$	-2.4	-4.2
Moderator Cooldown Curve	$\%\Delta\rho$ vs. °F	Figure 1	Figure 1
Doppler Multiplier		1.20	1.15
Inverse Boron Worth	PPM/$\%\Delta\rho$	80.0	87.0
Feedwater Flow	BTU/sec vs. Sec	Figure 10	Figure 10
Feedwater Enthalpy	BTU/lbm vs. Sec	Figure 11	Figure 11

* No credit for Low Pressure Safety Injection was taken in the FSAR analysis.

** approximate value

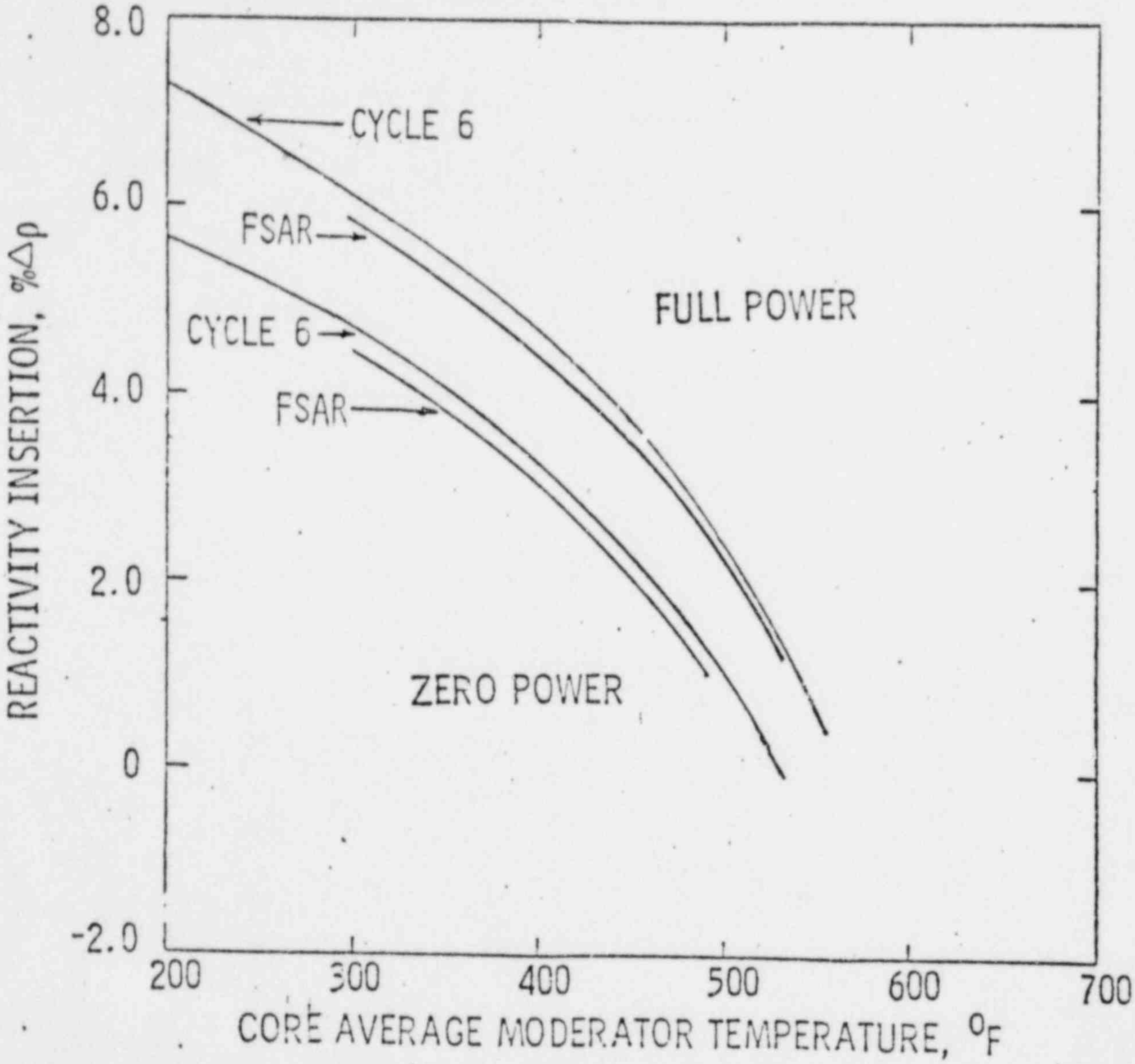
TABLE 4

Sequence of Events for the Main Steam Line Break Event
with Automatic Initiation of Auxiliary Feedwater Flow
(No Load, Two-Loop Condition, Nozzle Break)

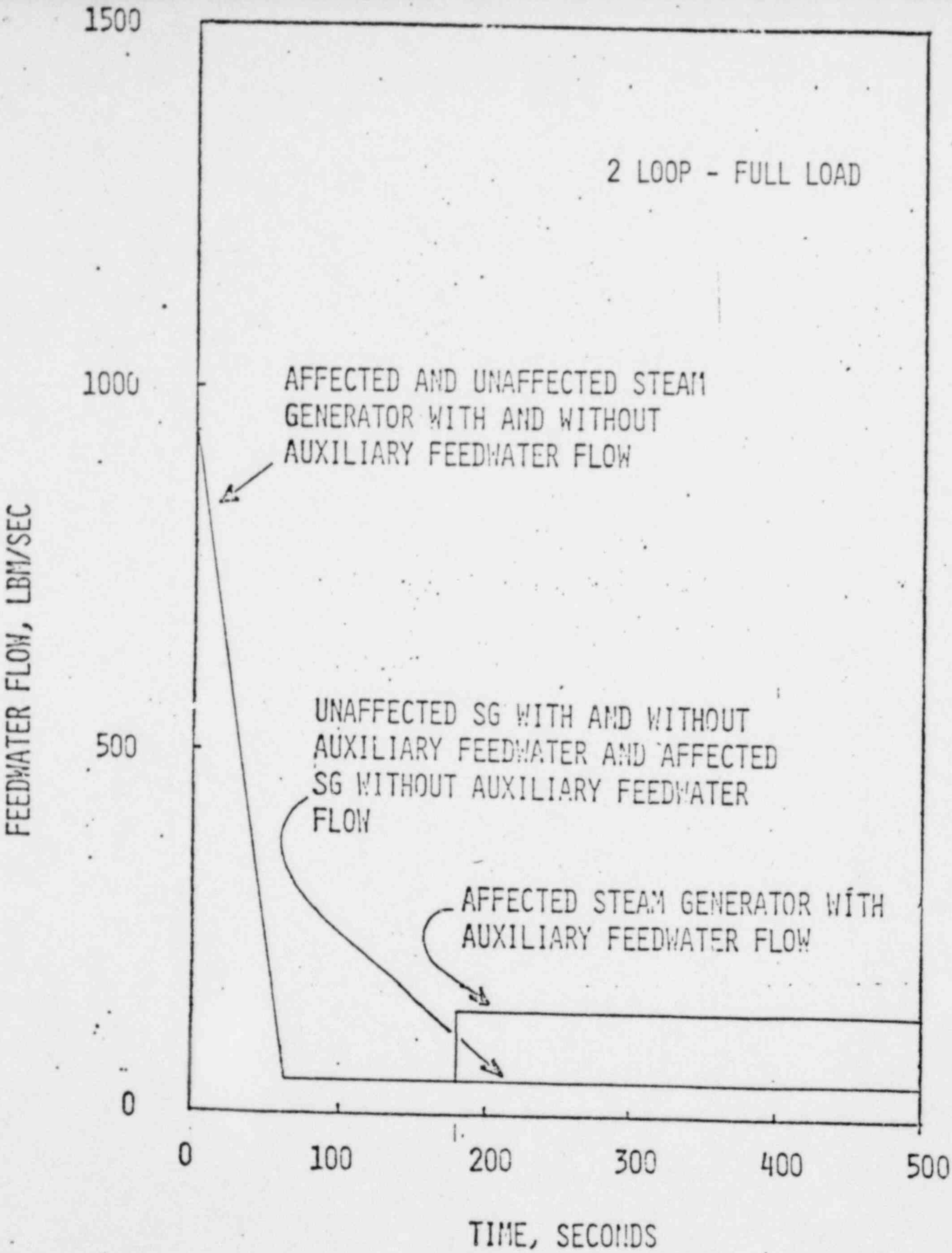
Time (sec.)	Event	Safety System Initiated	Setpoint or Value
0.0	Initiation of break	----	----
3.9	Low steam Generator Pressure trip signal occurs, MSIS initiated and Main Steam Isolation Valves begin to close.	Reactor Protection System Main Steam Isolation System	478 psia
4.8	Trip breakers open	----	----
7.3	CEAs at 90% Insertion	Reactor Protection System	----
8.4	Complete closure of Main Steam Isolation Valves to terminate blowdown from the intact steam generator	----	----
10.4	Pressurizer empties	----	----
13.4	Low RCS pressure, SIAS Initiated	Safety Injection System	1578 psia
21.4	High Pressure Safety Injection flow Initiated	Safety Injection System	1390 psia
102.6	Low Pressure Safety Injection Flow Initiated	Safety Injection System	201 psia
115.0	Boron from safety injection reaches mid-plane	----	----
121.7	Affected steam generator liquid inventory depleted and beginning of blowdown of feedwater only	----	----
124.5	Peak Reactivity		$-.27\%\Delta\rho$

TABLE 4 (Continued)

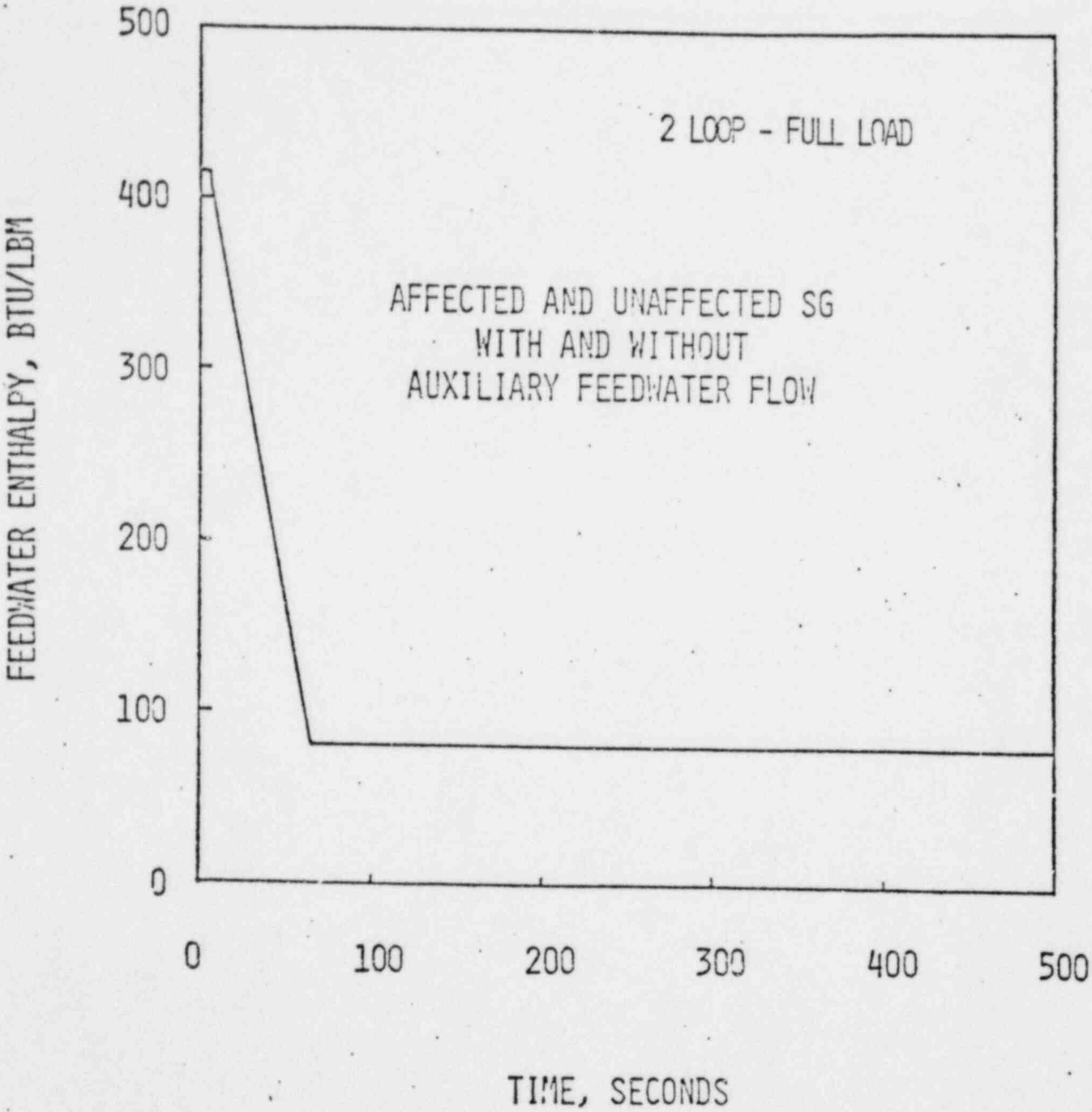
Time (sec.)	Event	Safety System Initiated	Setpoint or Value
180.0	Auxiliary Feedwater flow to affected steam generator initiated	----	----



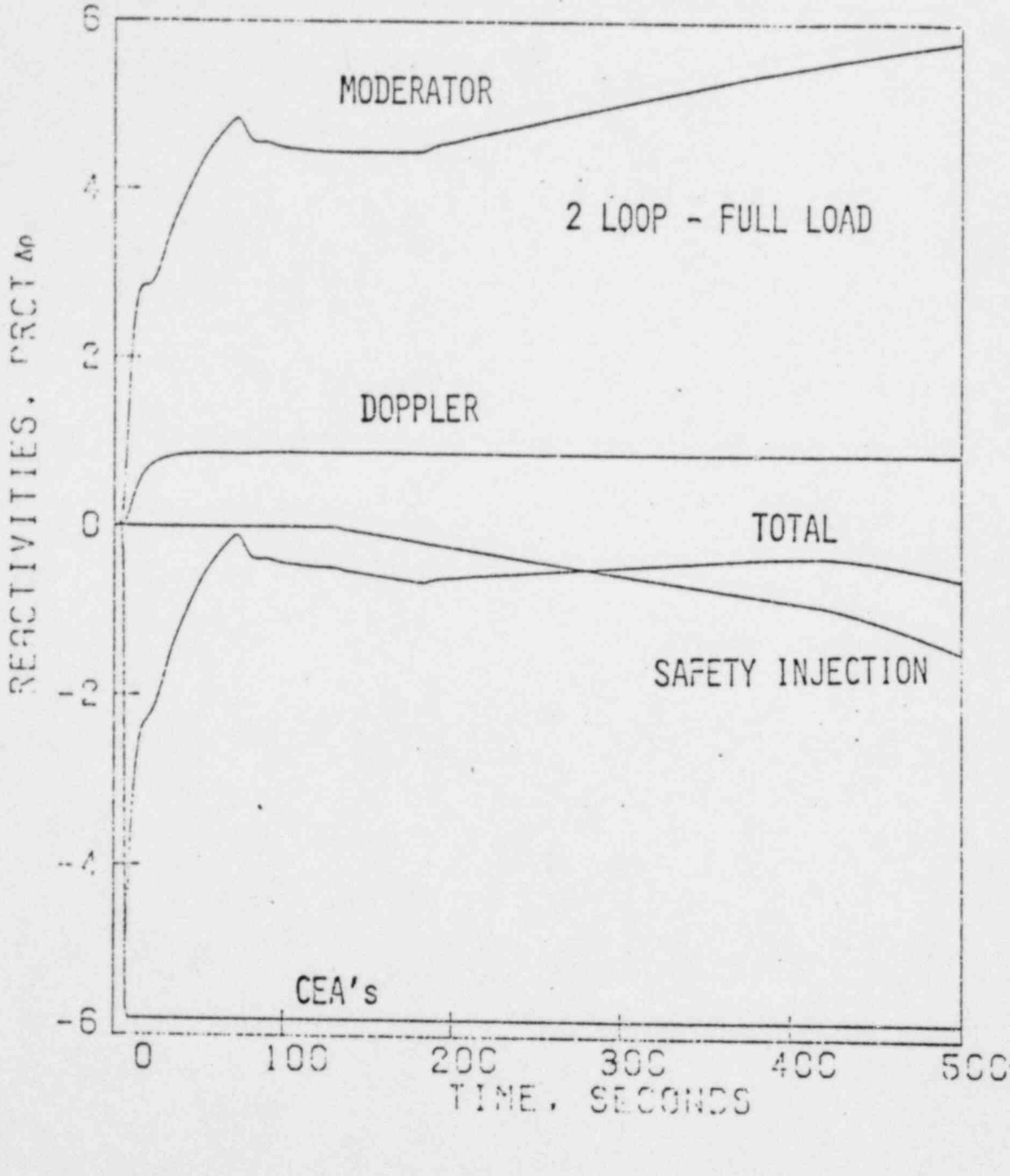
OPPD FT. CALHOUN-UNIT NO. 1	STEAM LINE RUPTURE REACTIVITY INSERTION VS MODERATOR TEMPERATURE	Figure 1



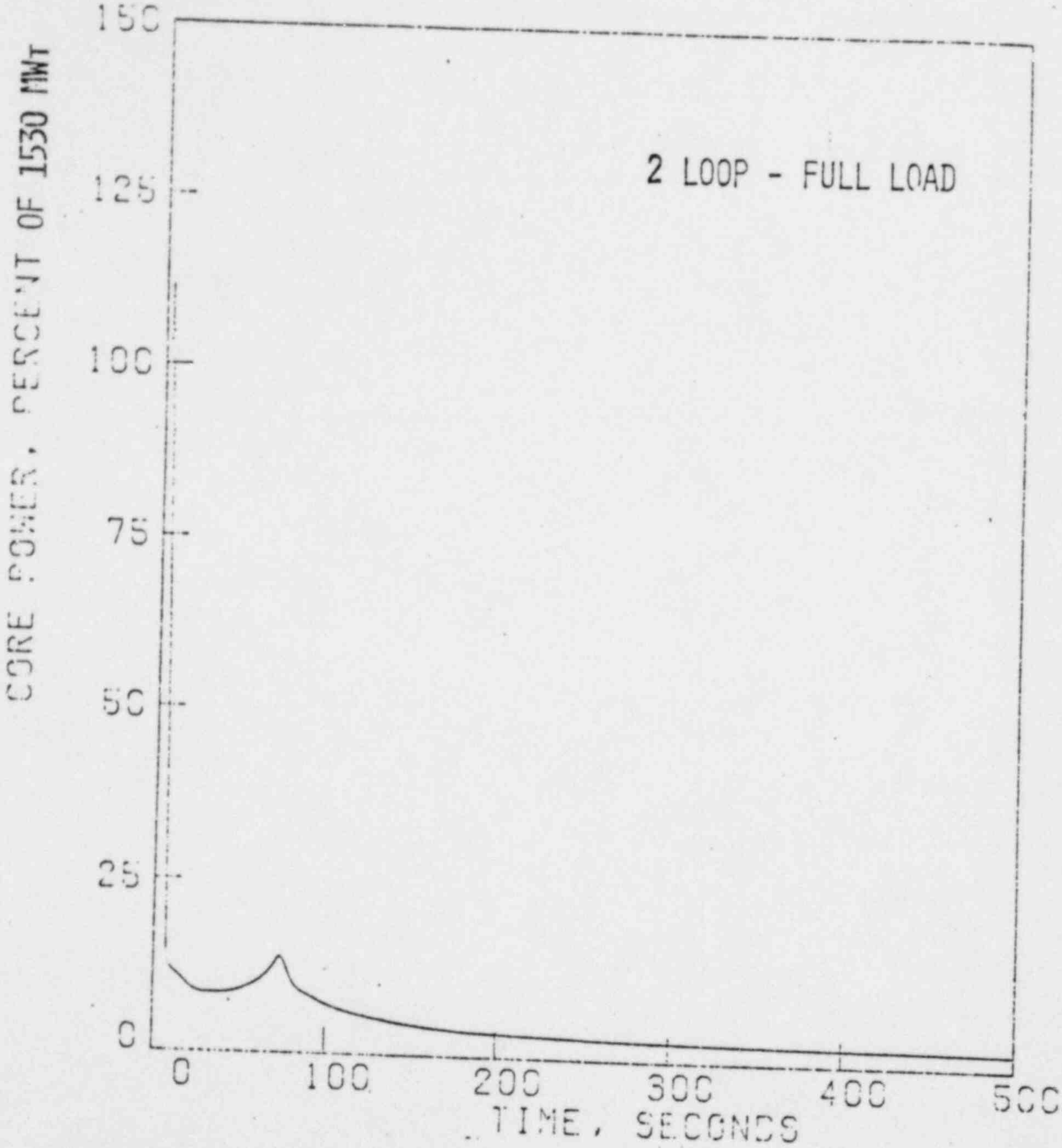
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE EVENT FEEDWATER FLOW VS TIME	Figure 2



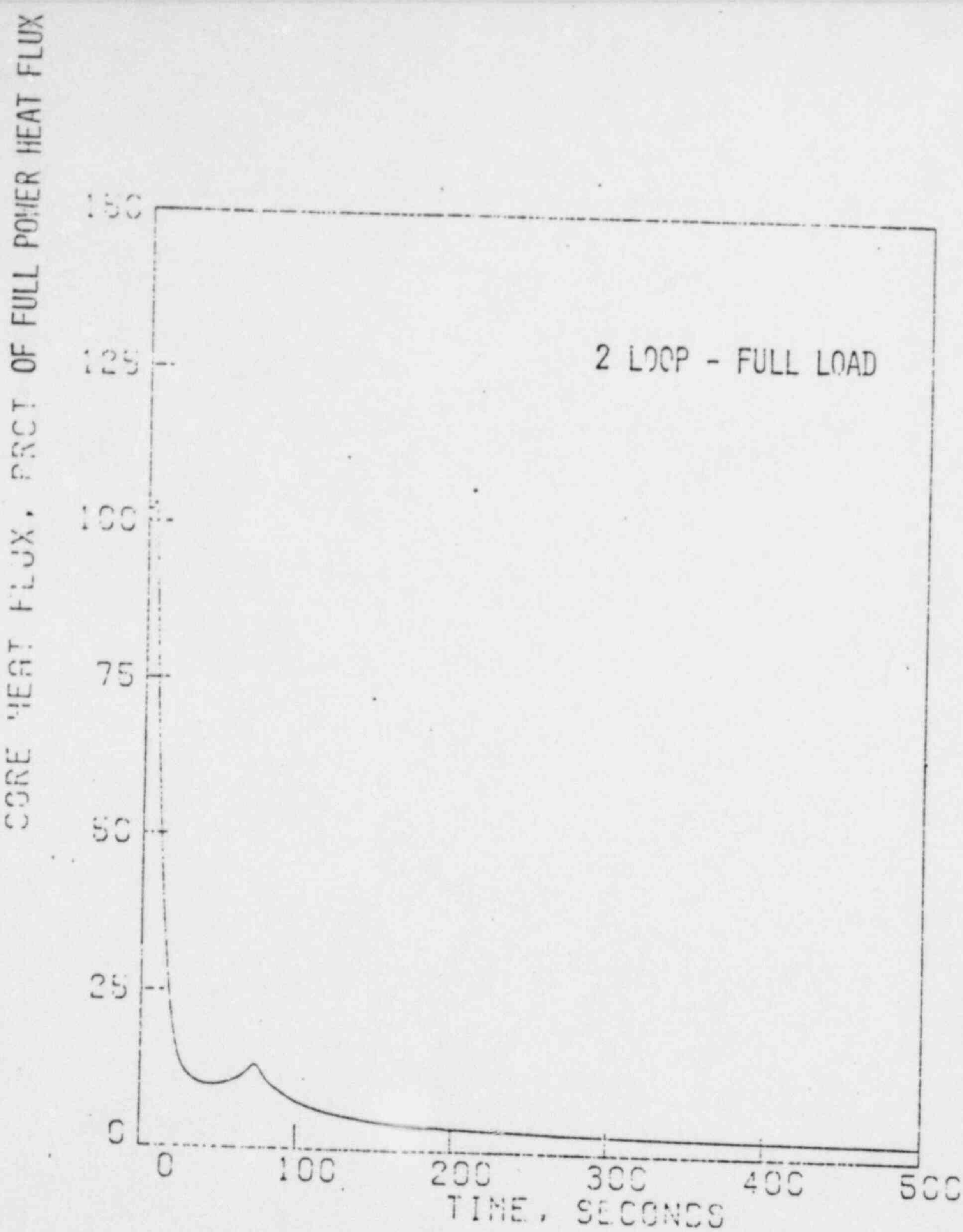
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE EVENT FEEDWATER ENTHALPY VS TIME	Figure 3



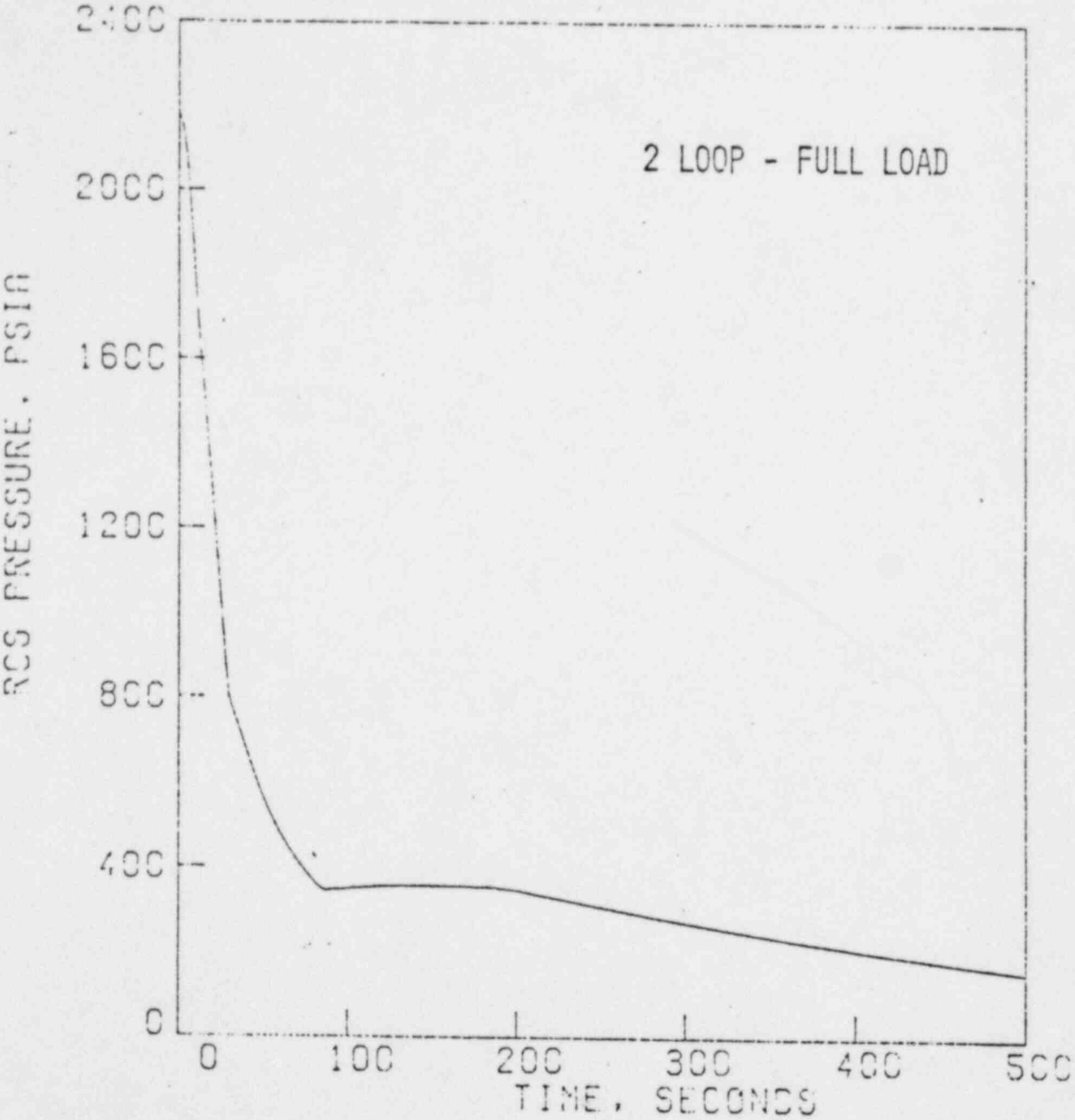
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE REACTIVITY CHANGES VS TIME	Figure 4



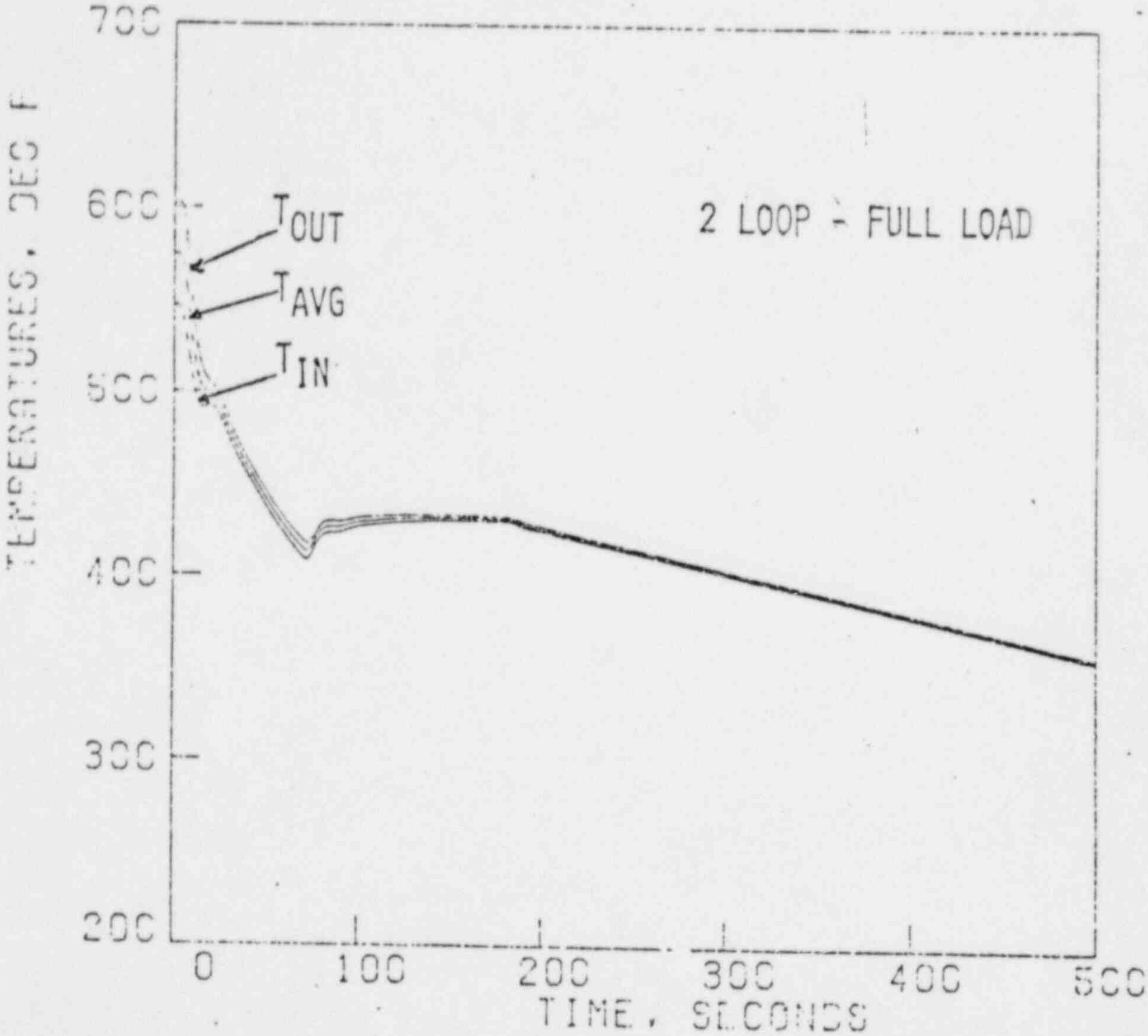
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE CORE POWER VS TIME	Figure 5



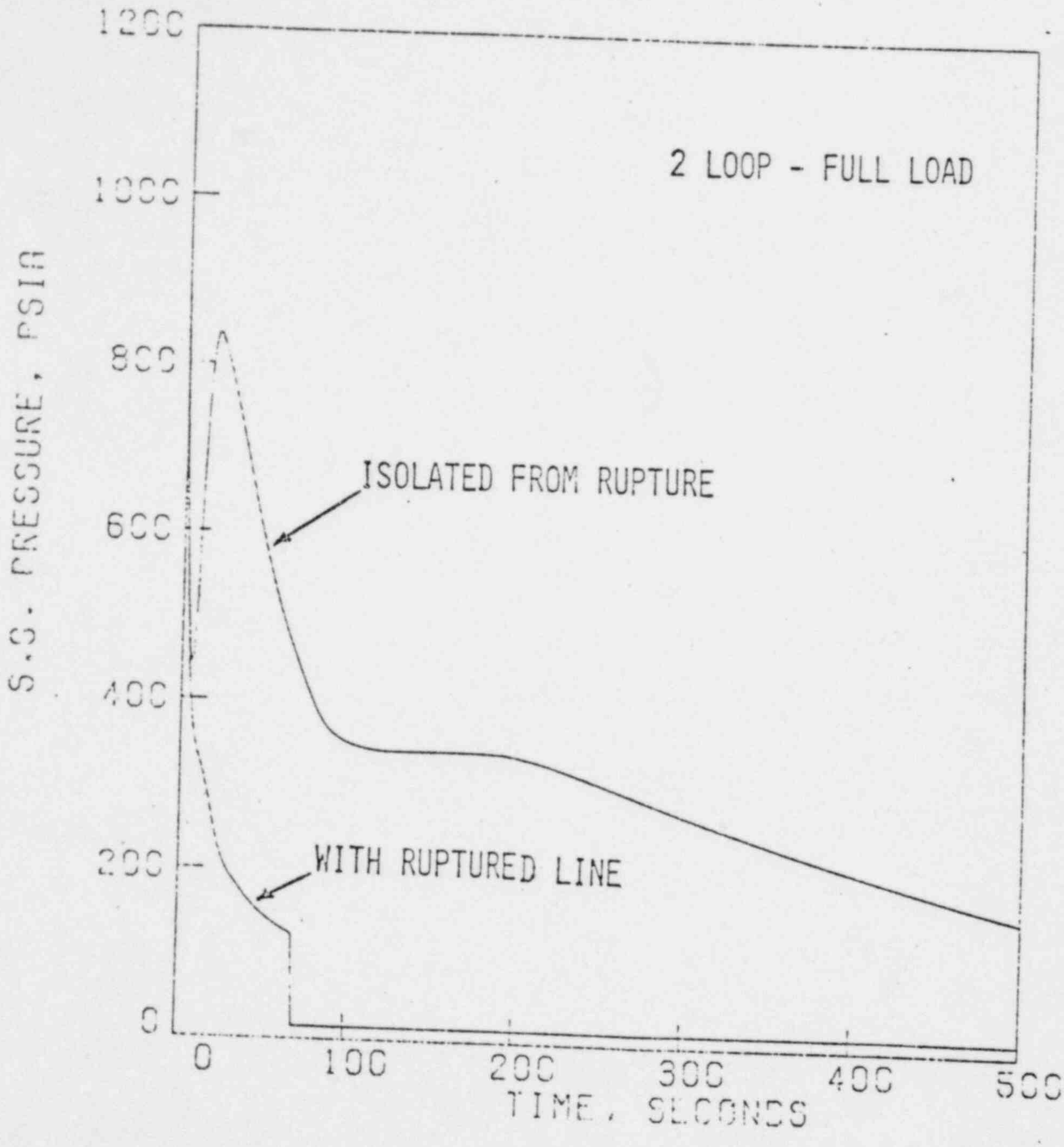
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE CORE AVERAGE HEAT FLUX VS TIME	Figure 6



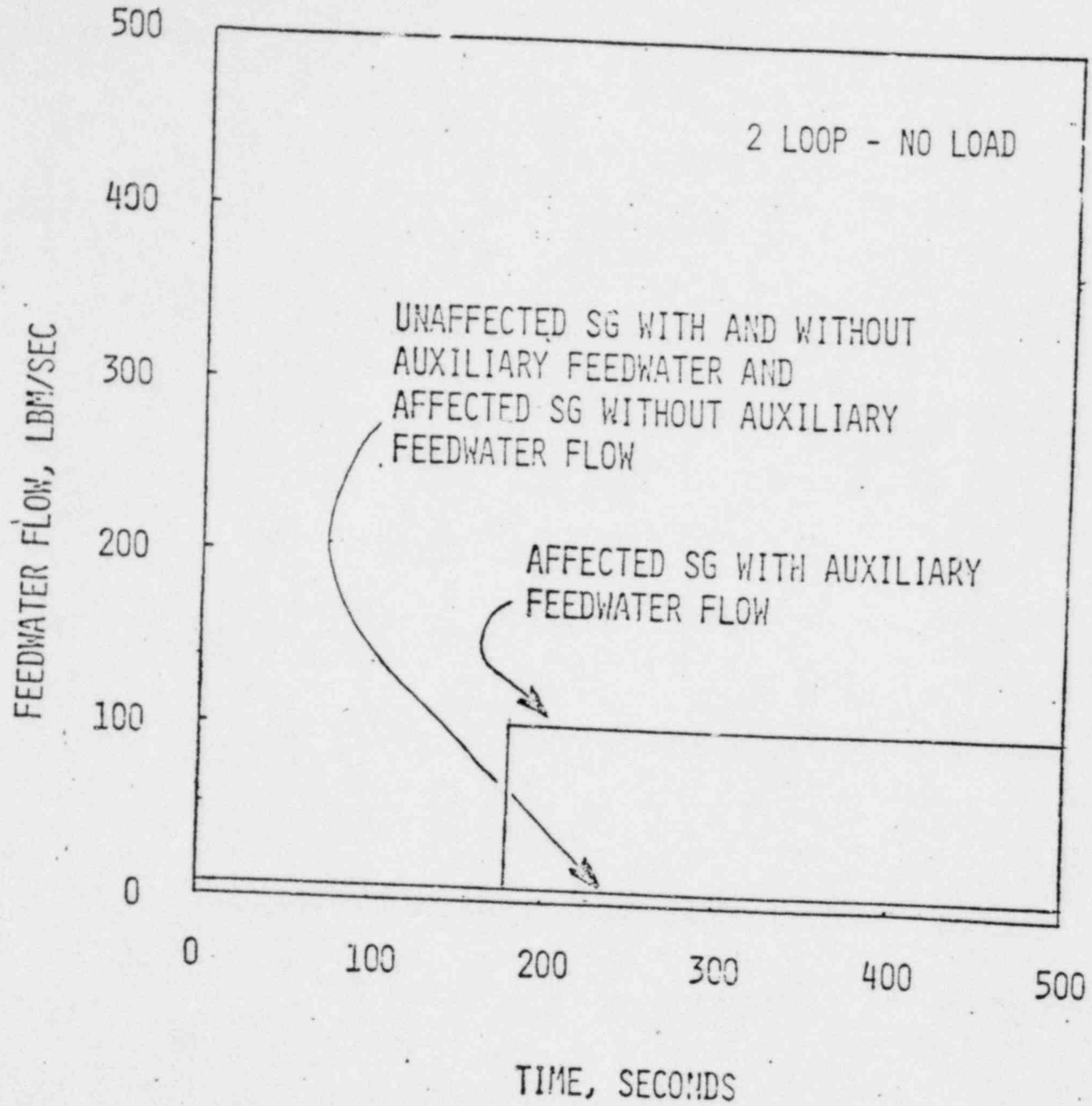
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE REACTOR COOLANT SYSTEM PRESSURE VS TIME	Figure 7



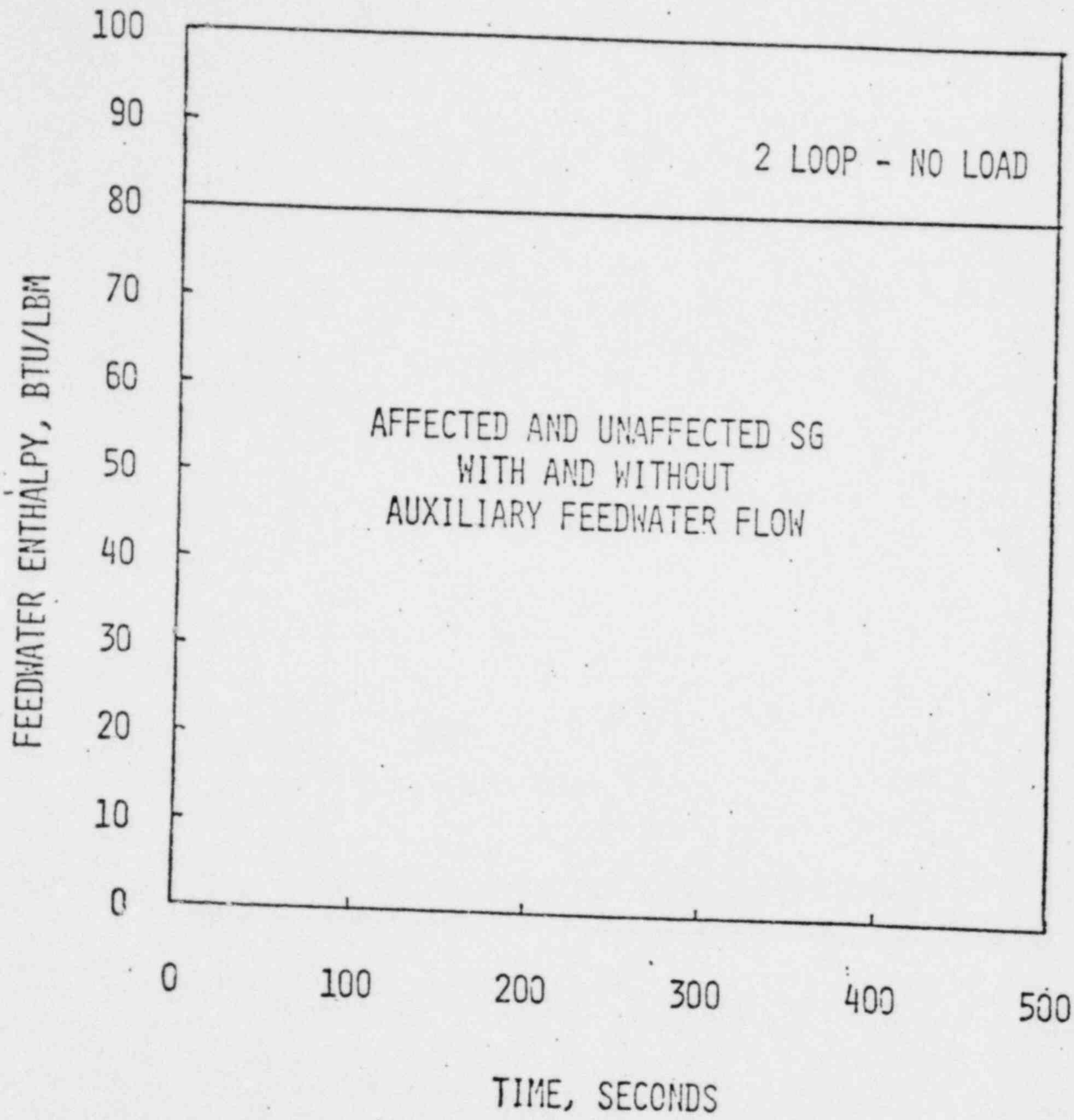
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE REACTOR COOLANT TEMPERATURE VS TIME	Figure 8



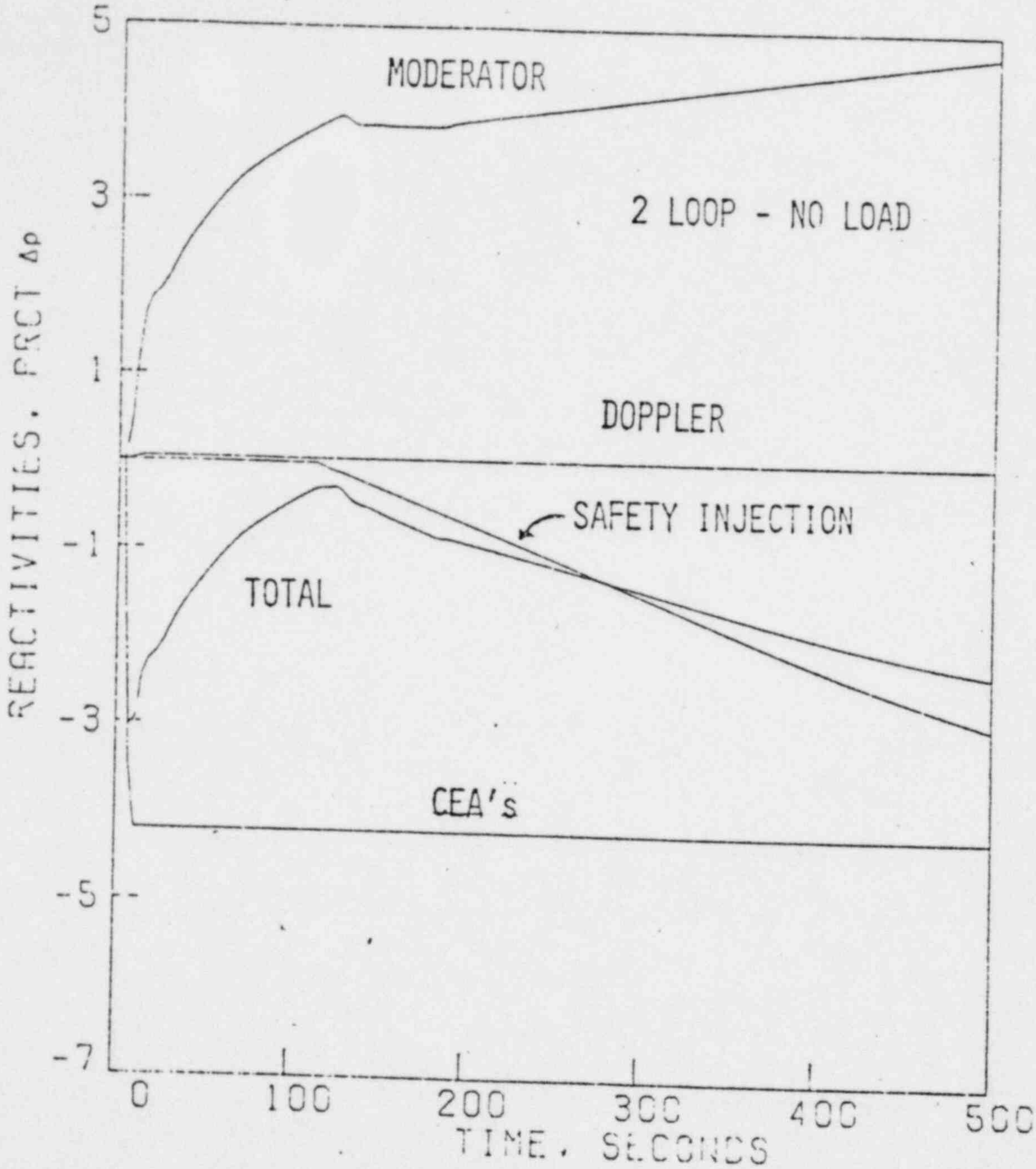
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE STEAM GENERATOR PRESSURE VS TIME	Figure 9



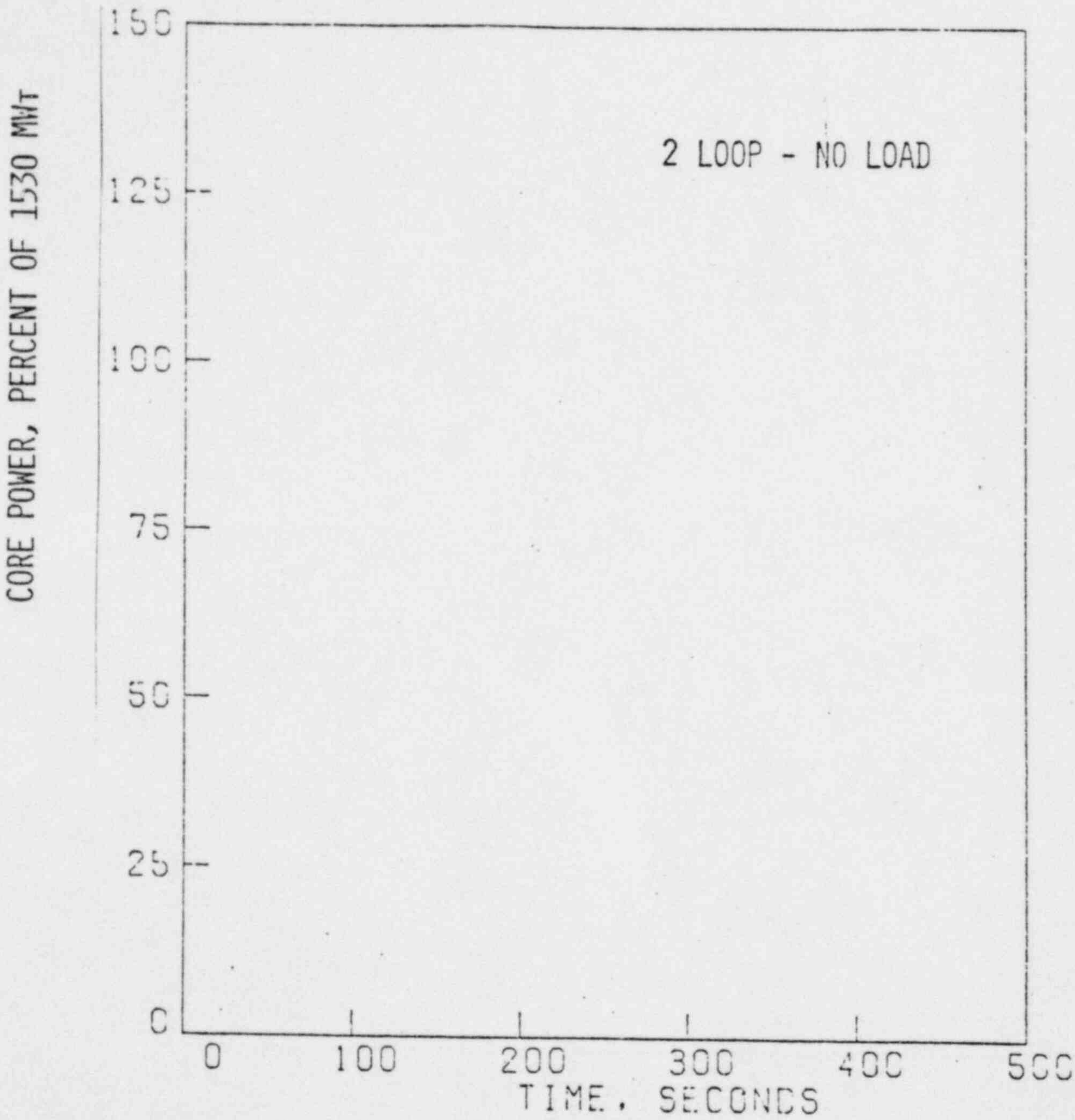
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE EVENT FEEDWATER FLOW VS TIME	Figure 10



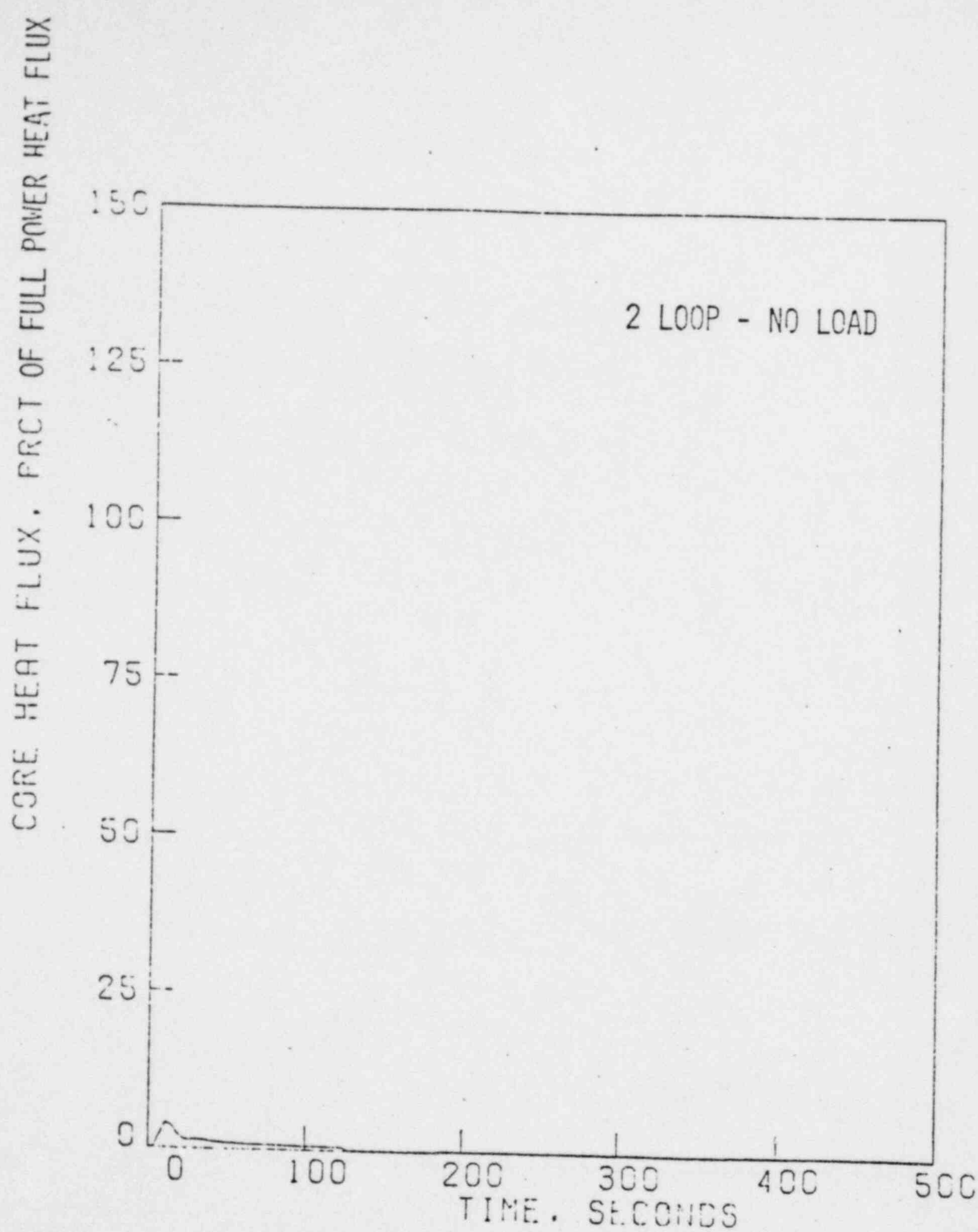
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE EVENT FEEDWATER ENTHALPY VS TIME	Figure 11



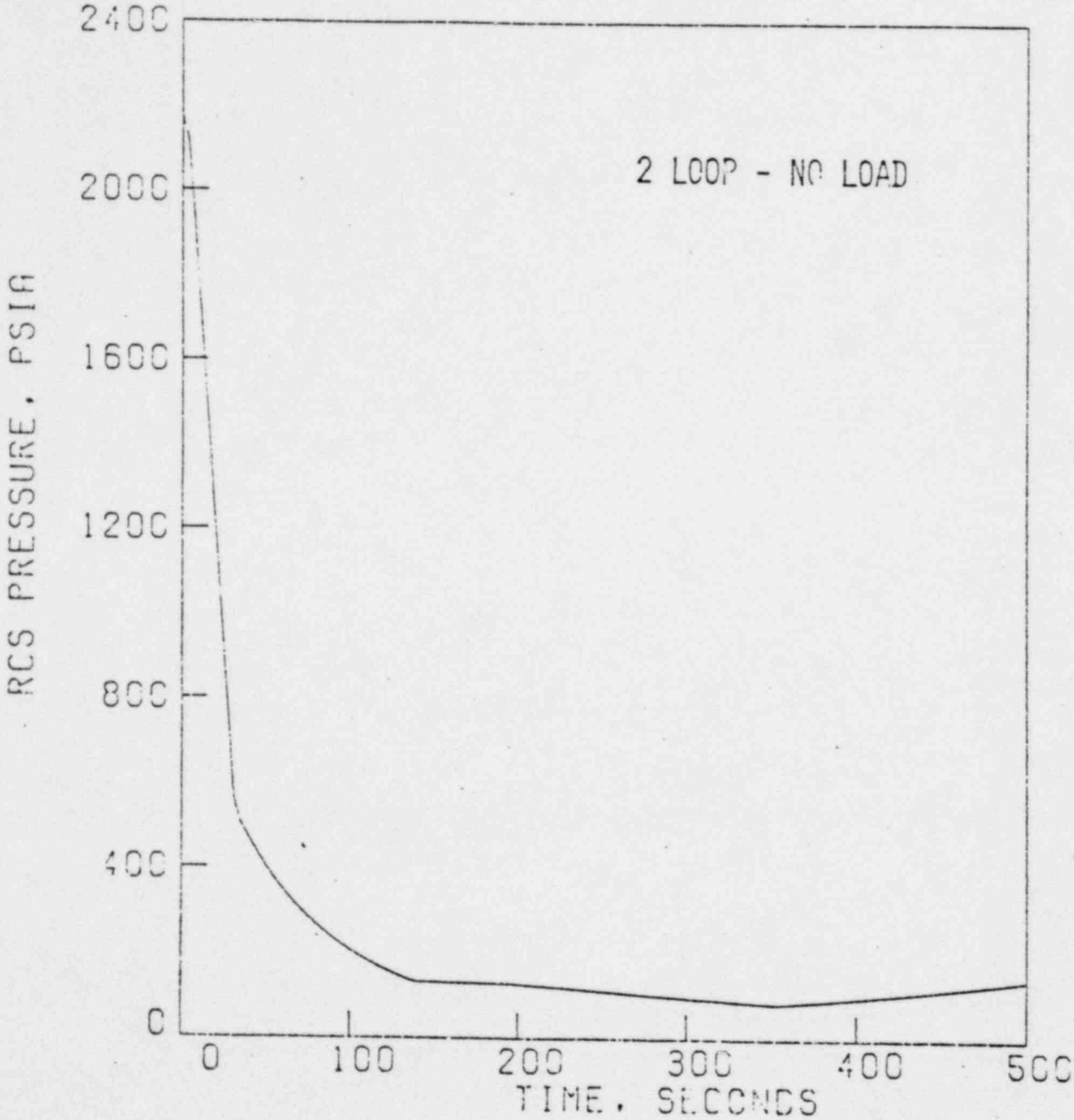
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE REACTIVITY CHANGES VS TIME	Figure 12



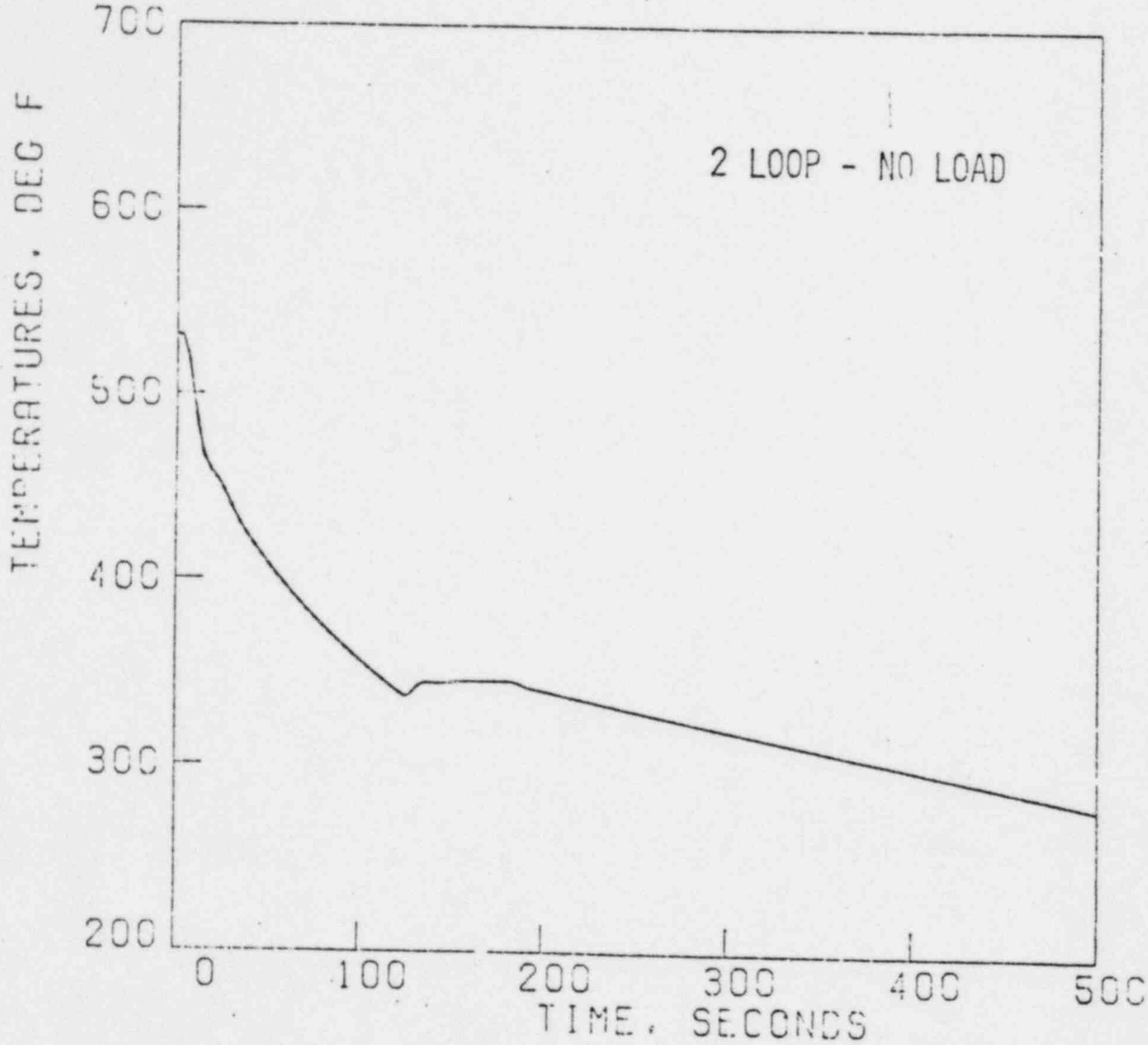
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE CORE POWER VS TIME	Figure 13



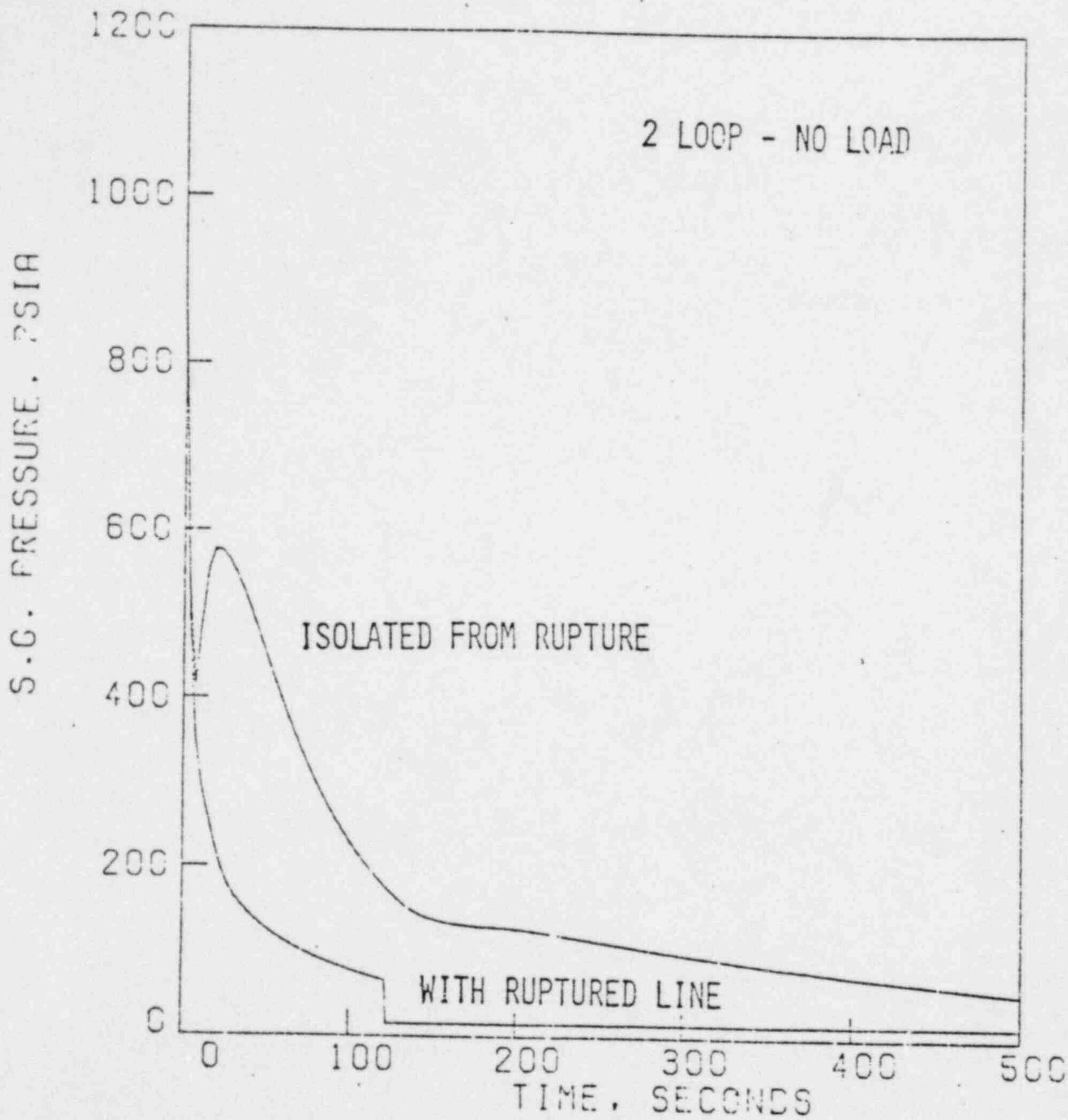
OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE CORE AVERAGE HEAT FLUX VS TIME	Figure 14



OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE REACTOR COOLANT SYSTEM PRESSURE VS TIME	Figure 15



OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE REACTOR COOLANT TEMPERATURES VS TIME	Figure 16



OPPD FT. CALHOUN-UNIT NO.1	STEAM LINE RUPTURE STEAM GENERATOR PRESSURE VS TIME	Figure 17