CEN-268, REVISION 1

SUPPLEMENT 1-NP, REVISION 1-NP

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON CEN-268

PREPARED FOR THE C-E OWNERS GROUP BY
TRANSIENT METHODS AND LOCA GROUP
NUCLEAR FUEL ENGINEERING

MAY 1987

COMBUSTION ENGINEERING

POWER SYSTEMS

RESPONSE TO NRC REQUEST
FOR ADDITIONAL INFORMATION
ON CEN-268

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1.0 INTRODUCTION

In July 1984, the NRC issued a second request for additional information (Reference 1) to help them complete the review of the trip two/leave two RCP trip strategy and trip setpoints report (Reference 2) submitted by the CEOG.

In order to resolve the large number of questions in an expeditious manner, a meeting was held on October 4, 1984 between representatives of the NRC, Los Alamos National Laboratory acting as NRC consultant, C-E Owners Group and C-E. The 59 questions documented in Reference 1 were discussed at the meeting. Responses to 33 questions were provided and found to be satisfactory to the NRC representatives at the meeting. The responses to these questions are documented in the NRC meeting minutes (Reference 3) and are included as Appendix A for convenience. The NRC stated that no further written responses to these questions would be necessary. The responses to the remaining 26 questions are provided in this supplement to the original report (Reference 2).

The first NRC request for additional information on the revised RCP trip strategy report (Reference 2) was issued May 3, 1984 (Reference 4). The response to the information request was transmitted to the NRC in a letter from the CEOG on June 25, 1984. (Reference 5). This transmittal is included as Appendix B for completeness.

2.0 RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON CEN-268

This section provides the written responses to 26 of the 59 questions as agreed at the October 4, 1984 meeting.

Question 7: Clarify what causes the spikes at about 500s in the liquid mass curves shown in Figures 3.1 and 3.4.

Response: The spikes in Figures 3.1 and 3.4 which start at about 480 sec into the transient are caused by an accumulation of liquid mass on the hot side of the reactor coolant system (RCS). At this time, the hot side of the steam generators (SGs) begins to drain very rapidly uncovering the flow path from the hot side to the cold side of the SGs. Thus, the normal flow path for the two-phase liquid to "exit" the hot side volume without going out the break is closed. However, for these two cases, two RCPs are still running and thereby forcing liquid into the reactor vessel (RV). The combination of liquid entering the RV, but not leaving the SGs creates an increase in liquid mass in the hot side of the RCS. The sudden increase in mass begins to decrease at about 500 sec when the void fraction in the RCPs rises to approximately 1.0. At this time, the flow into the core begins to decrease. The 20 sec duration of liquid mass accumulation in the hot side of the RCS (from 480 to 500 sec) is the time for the cold side of the SGs and the suction legs to essentially become void of two-phase liquid.

Question 8: What are the break flow and void fraction in the hot leg near the break for the two cases shown in Figs. 3-1 to 3-6? Provide the data in graphical form.

Response: As agreed at the October 4, 1984 meeting, plots of break flow and quality near the break would provide an acceptable response to this question. The plots which correspond to the two computer cases presented in Figures 3-1 to 3-6 for the 2700 MWt Reference plant with the steam bypass system operational and with the MSSVs in operation are provided in Figures 1 through 4, respectively. Refer to the CEFLASH-4AS nodal diagram provided in Figure 11 for the location of flow path 3.

Question 13: Provide plots of break flow and void fraction in the hot leg near the break as a function of time for the 0.05 ft^2 break case.

Response: As agreed at the October 4, 1984 meeting, plots of break flow and quality near the break would provide an acceptable response to this question. These plots for the 3410 MWt plant analysis are provided in Figures 5 and 6, respectively.

Question 19: In reference to the second tier of RCP trip setpoints, what is being recommended as a basis for loss of subcooling, hot leg or cold leg subcooling?

Response: RCS subcooling based upon pressurizer pressure and hot leg RTD temperature is recommended during forced circulation conditions (i.e., RCPs are operating). Forced circulation is the RCS condition when a determination of RCP trip is required. In many C-E plants, subcooling is displayed by the saturation margin monitor (SMM). In the C-E system, each channel of the SMM receives temperature input from one hot leg and two cold leg RTDs. The highest RTD temperature auctioneered from the three RTD inputs is used to compute the minimum RCS subcooling. In general, this means the subcooling displayed is based on the hot leg RTD temperature since the hot leg is typically hotter than the cold legs. The instrumentation existing in most C-E plants permits determination of subcooling based on hot leg temperature. This parameter was used in the evaluation of the RCP trip strategy.

Question 22: In Figure 5-2, why is the cold leg temperature slightly greater than the hot leg temperature after about 200 sec?

Response: After 200 sec, the cold and hot leg temperatures are virtually identical. Differences shown on the figure occurred as a result of transferring the cold and hot leg temperatures on to one plot. In general, the differences in temperature between the cold and hot leg were no greater than 1°F after 200 sec.

Question 23: In reference to the containment radiation alarm calculation, what are the assumptions used to get a radiation level alarm in 15 sec including break location, mixing in the containment and containment sensor locations? Can a break occur in a containment where it would take considerably longer to reach the sensor location?

Response: The use of containment radiation in the T2/L2 RCP trip strategy was reassessed and was determined not to have satisfactory sensitivity. Therefore, containment radiation is no longer used in the T2/L2 RCP trip strategy. Thus, this question is not applicable to CEN-268.

Question 25: Are there any non-LOCA events capable of activating the containment radiation alarms?

Response: Activation of the containment radiation alarms during non-LOCA events is very unlikely. However, a steam line break (SLB) inside containment can in principle activate the containment radiation alarms if it is assumed that a steam generator secondary side activity level at the technical specification limit of 0.1 μCi/gm exists. If a containment radiation alarm did occur, the second set of two RCPs would not be tripped since a SLB does not result in a loss of RCS subcooling.

Question 27: In reference to the plots for the inadvertent open PORV analysis, why do the hot and cold leg temperatures remain almost constant for the entire transient?

Response: The figure in question was mislabeled and is not representative of an inadvertent open PORV transient. The hot and cold leg temperatures provided in Figure 5-6 are representative of a much smaller break size than an open PORV and a transient which does not result in RCS depressurization. The hot and cold leg temperatures consistent with the other plots for the inadvertent open PORV analysis are provided in Figure 7. Note that the hot leg temperature decreases rapidly after reactor trip and saturation is reached at about 240 sec.

Question 29: In reference to the inadvertent open PORV analysis, clarify why the subcooling does not go to zero earlier when the pressure drops below 1543 psia.

Response: As stated in the response to question 27, the temperature plots provided in Figure 7 are consistent with the loss of subcooling shown in Figure 5-7 and the other plots for the inadvertent open PORV analysis. Based on the temperature data given in Figure 7, the hot leg subcooling would not be expected to go to zero before about 240 sec, as shown in Figure 5-7. For further clarification, the hot leg loss of subcooling for the inadvertent open PORV analysis is presented in Figure 8 on an expanded time scale.

Question 30: How long would it take for the pressurizer quench tank rupture disk to burst and produce a containment radiation alarm?

Response: The use of containment radiation in the T2/L2 RCP trip strategy was reassessed and was determined not to have satisfactory sensitivity. Therefore, containment radiation is no longer used in the T2/L2 RCP trip strategy. Thus, this question is not applicable to CEN-268.

Question 38: Page 5-8 states: "the lack of a containment radiation alarm would have resulted in the second two pumps not being tripped even if the subcooling criterion had not been met." Clarify if this is correct because it seems the ending should be "had been met."

Response: The statement should have been worded "had been met". The revised text of the second paragraph on page 5-8 is given below.

For this event, there would be no secondary side radiation alarm and most probably no containment radiation alarm. The first of these indications would signal a trip of the second two pumps if it were not for the subcooling criterion. The lack of a containment radiation alarm would have resulted in the second two pumps not being tripped even if the subcooling criterion had been met. Thus, the RCP trip strategy would result in manual tripping of the first two RCPs on low pressurizer pressure, and no manual tripping of the second two RCPs due to the presence of more than [20°F] subcooling in at least one hot leg. Since this is true for the DEGB SLB it would also be true for smaller SLBs.

Question 42: In reference to the SBLOCA licensing analysis (Section 7.1), what HPSI delivery flow rate was used? Provide the data in graphical or tabular form.

Response: The licensing analysis performed to show compliance with 10CFR50.46 used the HPSI flow characteristics presented in Table 1.

Question 43: Page 7-2 states: The active core starts to uncover at 731 sec and remains uncovered for approximately 1450 sec. The maximum depth of uncovery was 4.0 ft at 1228 sec. Figure 7-2 indicates the maximum depth of uncovery occurs at 1600 sec. Which is correct?

Response: For the 0.1 ft^2 hot leg break licensing analysis presented in Section 7.1, the maximum depth of uncovery is 4.0 ft and it occurs at 1228 sec after the start of the transient. The time scales on Figures 7-2 and 7-3 were mislabeled after 900 sec.

Question 44: As seen in Figs. 7-5 and 7-6, the minimum level and minimum inventory do not occur at the same time. Why not?

Response: The minimum two-phase mixture level in the inner reactor vessel occurs at 1228 sec compared to the minimum hot side liquid inventory which occurs at about 1050 sec. This difference is attributable to several effects.

The hot side liquid inventory is comprised of the liquid mass in the downcomer, inner reactor vessel, hot legs and hot side of the steam generators. Although the hot legs and steam generators have drained of liquid by approximately 820 sec, a large amount of liquid remains in the downcomer region throughout the transient. Thus, a comparison of hot side liquid inventory (mass) and inner reactor vessel two-phase mixture level is not on a common basis.

In addition, it should be noted that the two-phase level in the inner reactor vessel decreases from 17.6 ft at 1050 sec to 17.5 ft at 1228 sec. During this time period, the collapsed liquid level (which is more closely related to inventory) remains virtually constant. The collapsed liquid level in the inner reactor vessel varies from 14.55 ft at 1050 sec to 14.57 ft at 1230 sec while reaching a minimum value of 14.50 ft at about 1150 sec.

Question 45: Provide plots of the break flow and void fraction in the hot leg near the break for the licensing analysis case discussed in Section 7.1.

Response: As was agreed at the October 4, 1984 meeting, plots of break flow and quality near the break would provide an acceptable response to this question. These plots for the 0.1 ft^2 hot leg break licensing analysis are provided in Figures 9 and 10, respectively.

Questions 48-55: "Hence, the results of the previous analyses remain valid for the T2/L2 RCP trip scheme. The major analytical parameters used in the analyses included reactor power, break size, HPSI flow rate, SG safety relief valve setpoint, and overall SG heat transfer characteristics."

(Q.48) What previous analyses are referred to?

(Q.49) What are the steps and assumptions used in the calculation of the pressure setpoint?

(Q.50) What break sizes were considered?

(Q.51) What range of trip setpoints was analyzed?

(Q.52) What range of HPSI flow rates was considered?

(Q.53) What range of SG safety relief valve setpoints was considered?

(Q.54) What range of overall SG heat transfer characteristics was considered?

(Q.55) How are normal or adverse containment conditions accounted for in instrument uncertainties?

These questions pertain to the plant specific evaluation of the RCS pressure setpoint presented in the Appendix of Reference 2. Since the questions are related to the same analysis, the responses are provided together in this summary response.

Response: Introduction

Several analyses were performed for the C-E Owners Group to determine the minimum acceptable low RCS pressure setpoint for tripping all RCPs following a LOCA. These analyses were performed in response to the NRC request in IE Bulletin 79-06C. In addition, the NRC requested that the trip setpoint be at a pressure lower than the safety injection actuation signal (SIAS).

Analysis Description

The RCS pressure following a small break LOCA exhibits a characteristic "pressure plateau" shortly after reaching saturation conditions as a result of steam generator (SG) secondary behavior. Since all SBLOCAs exhibit this pressure plateau, it is necessary to determine a RCP trip setpoint above the plateau pressure. Therefore, in calculating the RCP trip pressure setpoint, a conservative set of input assumptions was used to maximize the magnitude of the pressure plateau.

The major assumptions used in the analyses include:

1. 4.3% of core power (this is the power level at the start of the pressure plateau.)
2. RCP heat of 20 MWt.
3. SG overall heat transfer coefficient of 300 BTU/hr-ft²-°F. (Analyses indicate the SG heat transfer coefficient is in the 500-600 BTU/hr-ft²-°F range during the pressure plateau period when the RCPs are operating.)
4. Heat transfer area for only one SG is used. CEN-114 analyses (Reference 6) have demonstrated that the maximum estimated reduction or degradation in SG heat transfer area due to a partial temporary loss in secondary level is 17%. Therefore, the 50% degradation in SG heat transfer area assumed in this analysis is conservative.
5. SG secondary side temperature based on the highest safety relief valve setpoint is used. The highest safety relief valve setpoint results in the highest SG sink temperature (T_{sec}) which is used in the analysis.
6. Flow from one HPSI pump was assumed. (The HPSI flow characteristics used in licensing analysis are provided in response to question 42.)

Following a SBLOCA, the RCS pressure stabilizes at a pressure plateau sufficiently above the secondary side pressure to remove the decay heat from the RCS. Since the RCS is in a quasi-steady state condition at this time, the RCS depressurization rate is essentially zero. Thus, the following expression was derived which iteratively computes a conservative upper bound of the magnitude of the pressure plateau.

$$Q_{core} + Q_{RCP} - UA\,\frac{(T_{pri} - T_{sec})}{3600} = f[W_{leak}(P)] + g[W_{HPSI}(P)]$$

where f and g are functions (energy flows) of the leak flow, $W_{leak}(P)$, and the high pressure safety injection flow, $W_{HPSI}(P)$, (which themselves are functions of RCS pressure) respectively in lb/sec.

Q_{core} = core decay heat (BTU/sec)

Q_{RCP} = RCP heat (BTU/sec)

U = SG overall heat transfer coefficient (BTU/hr-ft²-°F)

A = SG heat transfer area (ft²)

T_{pri} = RCS temperature (°F)

T_{sec} = SG secondary side temperature (°F)

Hot leg break sizes in the range from 0.1 ft² to 0.02 ft² for the 2700 MWt Reference plant have been shown (Reference 7) to increase the amount of core uncovery if the RCPs continue to operate during the transient. The 0.02 ft² break size was used in the Reference plant analysis since the smaller break sizes result in higher pressure plateaus.

As stated in the Appendix to Reference 2, instrument uncertainties were not accounted for in this evaluation.

Question 57: How much flow goes back through the cold legs in which the pumps are not running when only two of the pumps are kept running?

Response: Transient analyses were performed for small break LOCAs, SLBs and SGTRs in which two RCPs were operating while the remaining two RCPs were tripped. Reactor coolant system (RCS) transient flow data for the four cold legs are provided in tabular form for two SBLOCAs and one SLB. The transient RCS flow data for the SGTR case presented are not readily available as agreed at the October 4, 1984 meeting.

The SBLOCA analyses presented in Section 3.0 of Reference 2 are applicable to the request for RCS flow data with two RCPs on and two RCPs off. The other SBLOCAs presented in Reference 2 are not applicable to this RCP operating configuration. The flows through the four RCS loops for the 2700 MWt Reference plant analysis (Section 3.4.1) are presented in Table 2. The loop flow data for the 3410 MWt plant analysis (Section 3.4.3) are provided in Table 3. The flow path numbers referred to in Tables 2 and 3 are found in Figure 11. Positive flow is in the normal operating RCS flow direction, (i.e., into the reactor vessel).

The RCS flow data for the SLB transient discussed in Section 5.3 are given in Table 4. The flow path numbers referred to in Table 4 are described in Figure 13.

Question 58: Clarify with system noding diagrams what models were used for each of the transients discussed.

Response: A best estimate CEFLASH-4AS system model was used for the SBLOCA analyses. The CEFLASH-4AS node and flow path diagram is provided in Figure 11.

The CESEC computer code was used for the SGTR and SLB analyses. The CESEC noding diagram and flow path modeling scheme are given in Figures 12 and 13, respectively.

Question 59: We will be performing independent, confirmatory audit analyses of selected transients. In order to benchmark our calculation, please provide the following plots for the 2700 MWt reference plant, for the 0.1 ft^2 hot leg SBLOCA and for the DEGB SGTR cases (where noted by (*), these data were provided for the SBLOCA in the submittal):

1. Secondary side pressures, intact and faulted loops,
2. Hot leg and cold leg temperatures,
3. Core mixture level,*
4. Small break or SGTR mass flow,
5. Integrated small break or SGTR mass flow,
6. Rod temperatures,*
7. Reactor coolant system loop mass flows,
8. Steam and feedwater (main and auxiliary) flows, and
9. Safety injection and accumulator flows.

Response: The available information requested above for the 0.1 ft^2 hot leg break with 2 RCPs on and 2 RCPs off presented in Section 3.4.1 of Reference 2 is provided as described below.

DATA FOR THE 2700 MWt ANALYSIS OF THE 0.1 FT^2 HOT LEG SBLOCA

	Parameter	Comment
1.	Secondary side pressure	Figure 14, Note 1
2.	Hot and cold leg temperatures	Figure 15
3.	Core mixture level	Figure 3-2 in Reference 2
4.	Small break mass flow	Figure 1
5.	Integrated mass flow	Figure 16
6.	Rod temperatures	Note 2
7.	RCS loop mass flow	Table 2
8.	Steam and feedwater flow	Not applicable
9.	(a) Safety injection flow	Figure 17, Note 3
	(b) Safety injection tank (accumulator) flow	Not applicable

Notes:

1. The secondary side pressures for both loops are virtually identical.
2. Fuel rod temperature data for this case is not applicable since there was no significant core uncovery resulting in fuel heatup.
3. The safety injection flow provided is the flow rate into each of the four cold legs.

The SGTR data requested is not readily available as was discussed at the October 4, 1984 meeting. It was agreed at the meeting, that further action on this issue will be delayed until after LANL performs their calculations and their results are evaluated.

3.0 REFERENCES

1. NRC letter from James R. Miller (NRC) to R. Wells (Chairman, CEOG), dated July 2, 1984.
2. Combustion Engineering, Inc., "Justification of the Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients," CEN-268, March 1984.
3. NRC memorandum from E. D. Throm to B. W. Sheron, "Meeting Summary on Reactor Coolant Pump Trip, Combustion Engineering Owners Group, 10/04/84, Revision 1, Incorporates C-E/CEOG Comments," November 20, 1984.
4. NRC letter from James R. Miller (NRC) to R. Wells (Chairman, CEOG), dated May 3, 1984.
5. Letter from R. W. Wells (Chairman, CEOG) to James R. Miller (NRC), "Request for Additional Information on CEN-268," RWW-84-44, June 25, 1984.
6. Combustion Engineering, Inc., "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems," CEN-114-P (Amendment 1-P), July 1979 (Proprietary).
7. Combustion Engineering, Inc., "Response to NRC IE Bulletin 79-06C, Items 2 and 3 for C-E Nuclear Steam Supply Systems," CEN-115-P, August 1979 (Proprietary).

Table 1

HIGH PRESSURE SAFETY INJECTION (HPSI) FLOW
2700 MWt SBLOCA LICENSING ANALYSIS

RCS Pressure (psia)	Flow (gpm)
1225	0
1200	59.4
1100	148.4
1000	204.0
900	244.8
800	278.2
700	304.2
600	326.4
500	356.1
400	378.4
300	400.6
200	422.0

Table 2

2700 MWt PLANT ANALYSIS

0.1 FT² HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

RCS LOOP FLOWS

Time (sec)	RCS FLOW (lbm/sec) Path 17	Path 32*	Path 15*	Path 12
0	9658	9658	9658	9658
20	9603	9603	9517	9517
40	9292	9292	9280	9280
60	9033	9033	9159	9159
80	9824	5871	5831	9911
100	12885	-2914	-3007	12937
150	11425	-3461	-3463	11337
200	10081	-3024	-3029	9938
250	8626	-2576	-2574	8439
300	7385	-2254	-2247	7152
400	4933	-1764	-1730	4526
500	2721	-1292	-1270	2209
600	496	- 290	- 406	386
800	324	- 368	- 206	254
1000	196	- 175	- 181	196
1200	150	- 155	- 161	154
1400	147	- 106	- 111	137

*RCP operation in these flow paths was terminated at 77 sec.

Table 3

3410 MWt PLANT ANALYSIS

.05 FT² HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

RCS LOOP FLOWS

Time	RCS FLOW (lbm/sec)			
(sec)	Path 17	Path 32*	Path 15*	Path 12
0	10278	10278	10278	10278
20	10260	10260	10193	10193
40	10235	10235	10182	10182
60	9977	9977	9997	9997
80	9244	9244	9738	9738
100	13777	-1406	-1539	13885
120	15058	-4255	-4270	15058
140	15000	-4241	-4253	14978
160	14896	-4218	-4229	14871
180	14801	-4198	-4209	14772
200	14712	-4180	-4190	14684
250	14512	-4137	-4147	14484
300	13920	-4011	-4021	13886
400	12271	-3657	-3663	12210
600	9940	-2909	-2910	9851
800	7292	-2250	-2247	7165
1000	5330	-1823	-1806	5097
1200	2752	-1335	-1301	2475
1400	461	- 419	- 417	461
1600	512	- 510	- 322	368
1800	407	- 381	- 382	407
2000	343	- 319	- 319	342
2200	352	- 321	- 322	350

*RCP operation in these flow paths was terminated at 84 sec.

Table 4

DOUBLE ENDED GUILLOTINE SLB
RCS LOOP FLOWS

Time (sec)	RCS FLOW (lbm/sec) Path 2	Path 14*	Path 28*	Path 16
0	9686	9686	9686	9686
10	10253	10253	9813	9813
20	10691	10691	9795	9795
30	10960	10960	9892	9892
40	11110	11110	10042	10042
50	11208	11208	10188	10188
55	11630	9608	8850	10615
60	13118	3132	3282	12323
65	13856	- 211	13	13125
70	14133	-1744	-1457	13444
75	14269	-2784	-2493	13643
80	14340	-3481	-3222	13779
85	14344	-3748	-3566	13845
90	14313	-3722	-3665	13869
100	14240	-3752	-3667	13883
110	14177	-3736	-3667	13893
120	14124	-3722	-3667	13899
150	14006	-3691	-3663	13899
200	13886	-3658	-3651	13866
295	13747	-3619	-3622	13774

*RCP operation in these flow paths were terminated at 54 sec.

FIGURE 1

2700 MWt PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

BREAK FLOW

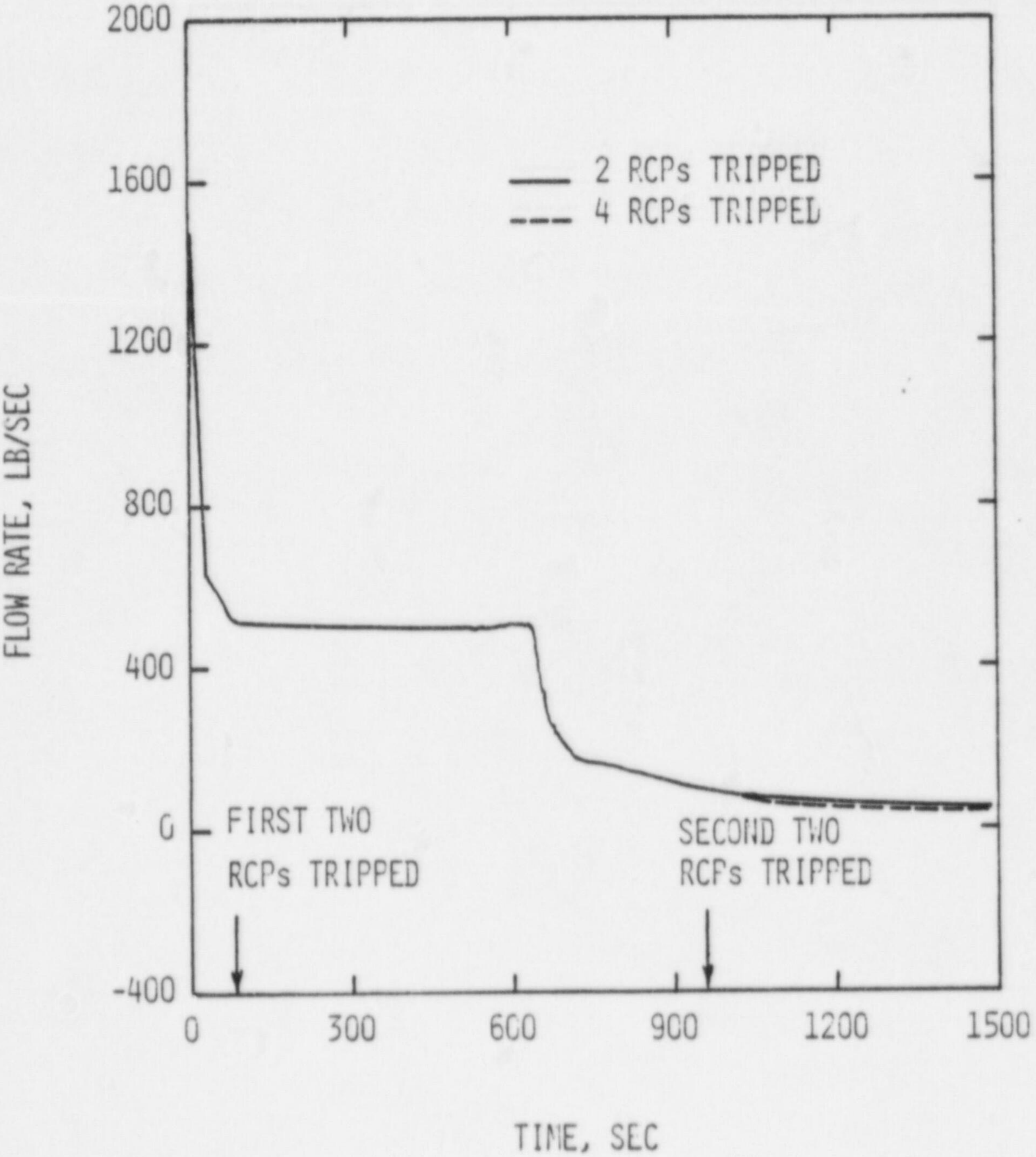


FIGURE 2

2700 MWT PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

QUALITY IN FLOW PATH 3

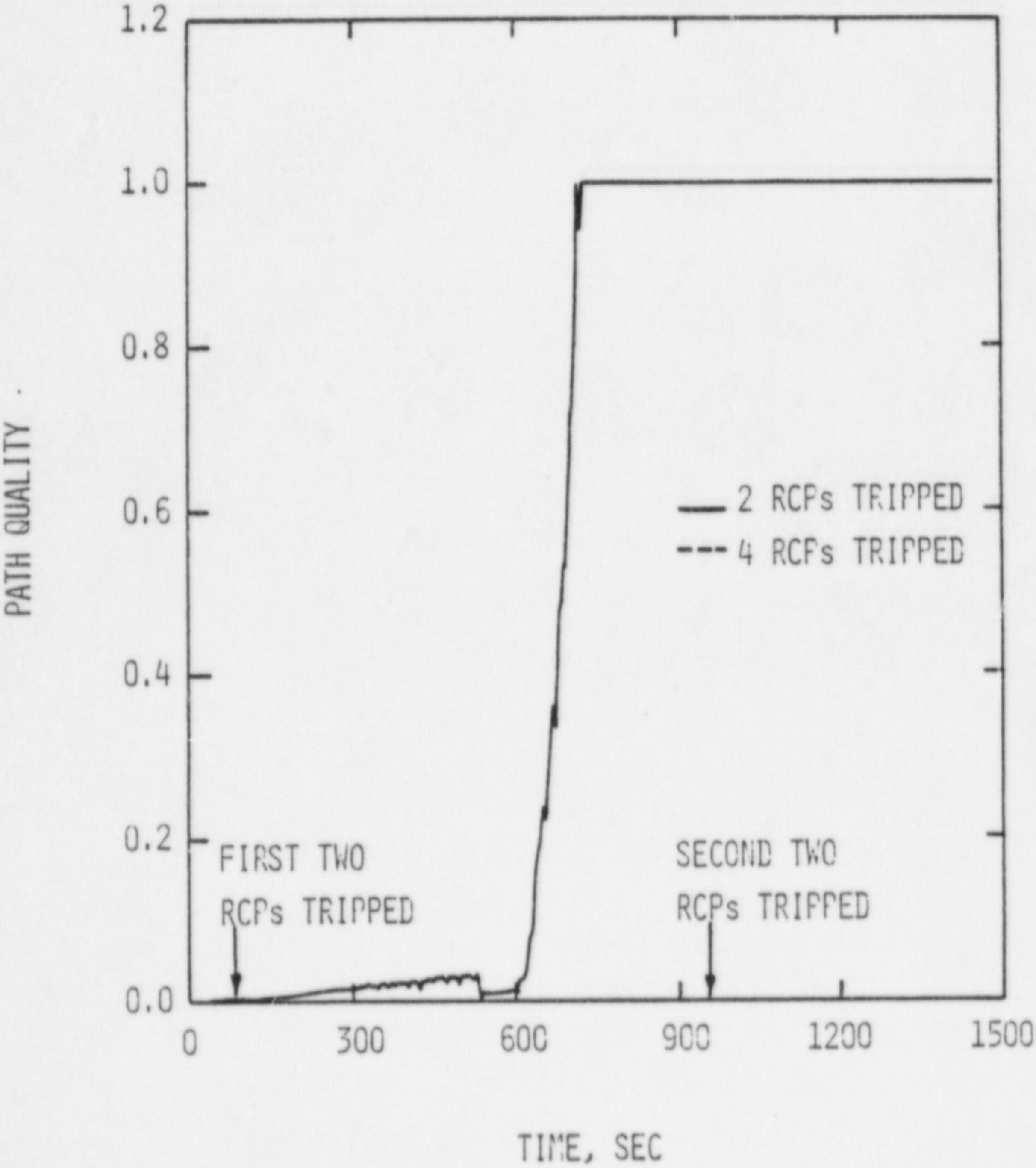


FIGURE 3

2700 MWT PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

MSSVs IN OPERATION

BREAK FLOW

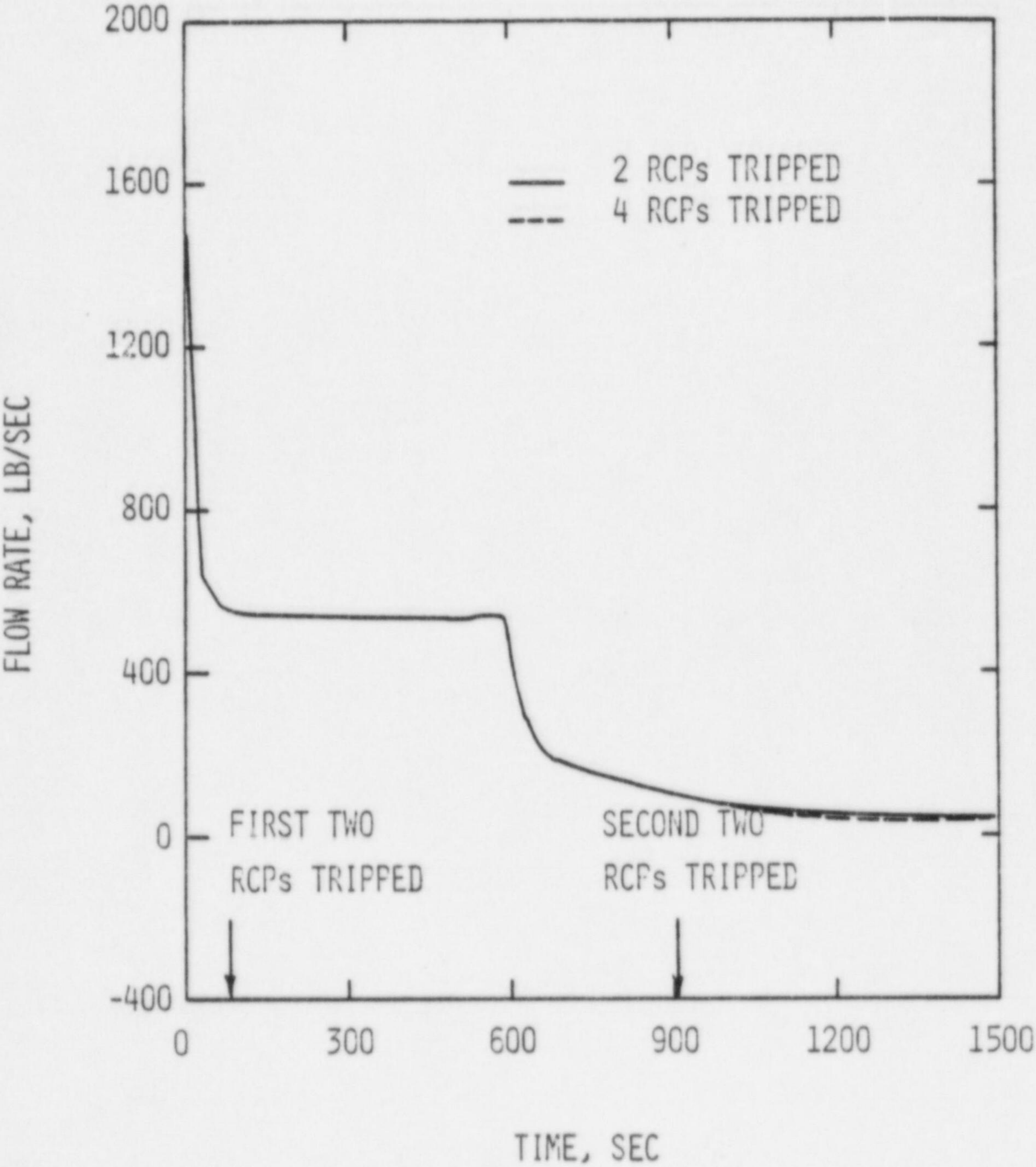


FIGURE 4

2700 MWT PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

MSSVs IN OPERATION

QUALITY IN FLOW PATH 3

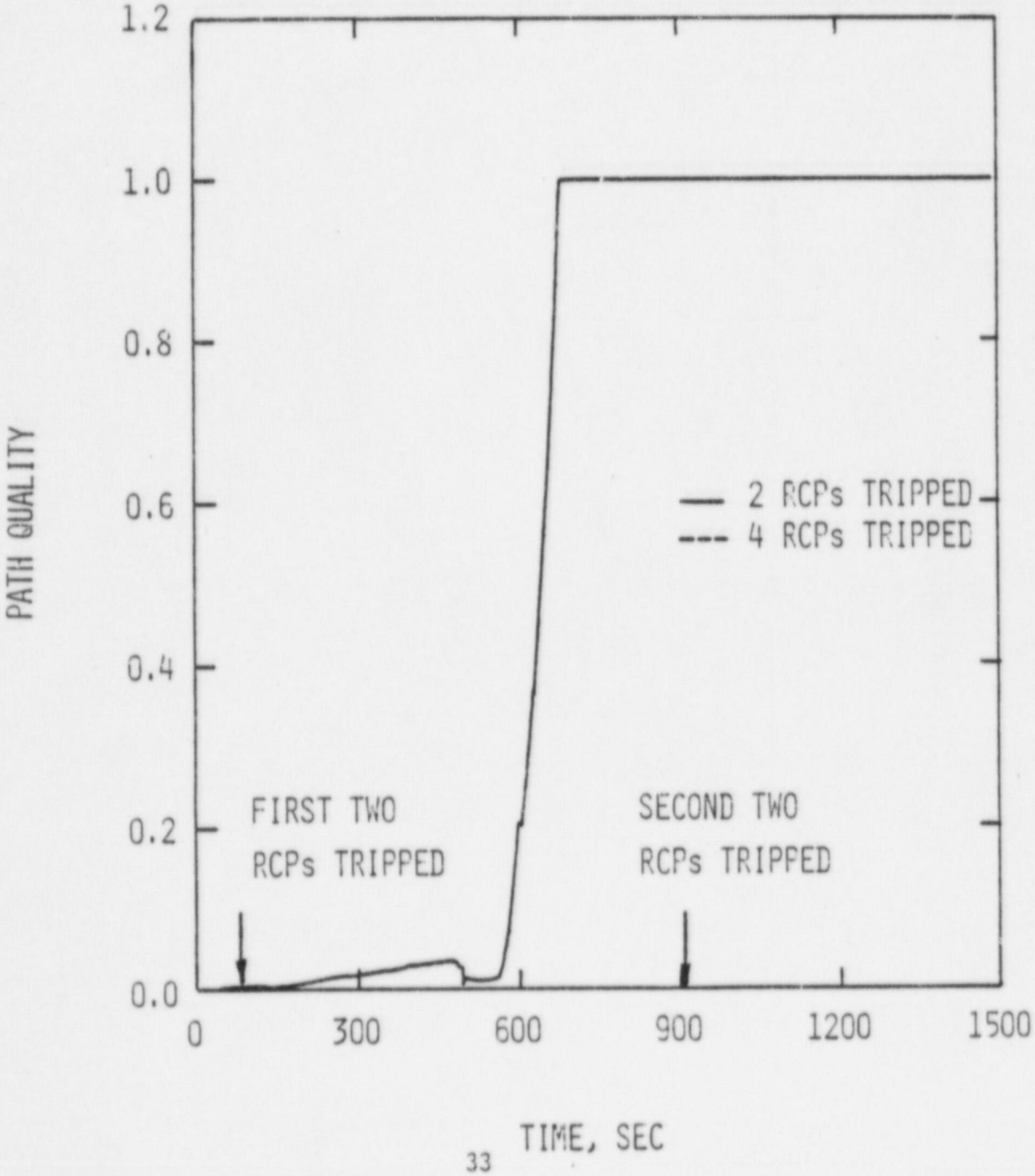


FIGURE 5

3410 MW_T PLANT ANALYSIS

.05 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

BREAK FLOW

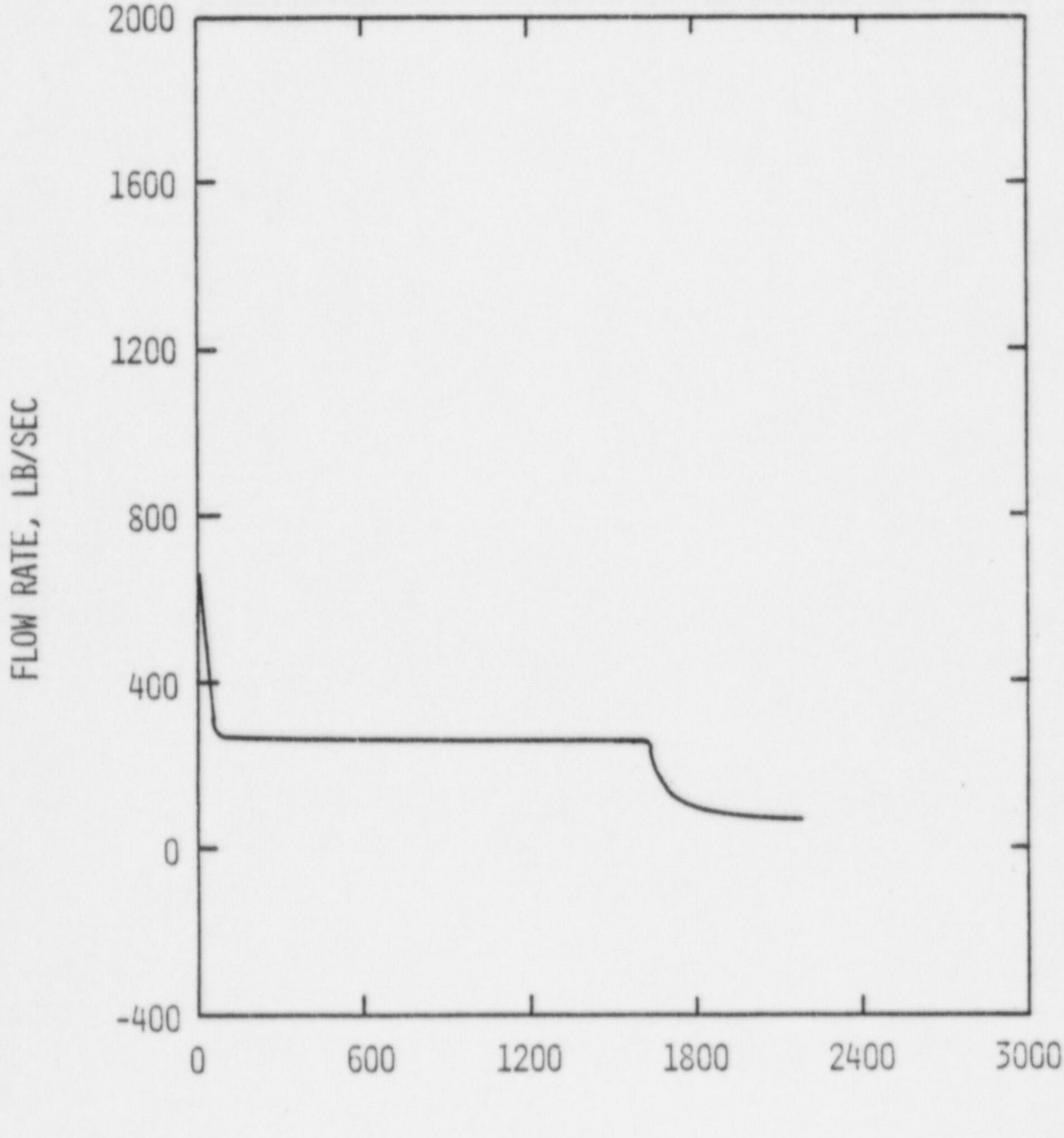


FIGURE 6
3410 MWт PLANT ANALYSIS
.05 FT^2 HOT LEG BREAK
STEAM BYPASS SYSTEM OPERATIONAL
QUALITY IN FLOW PATH 3

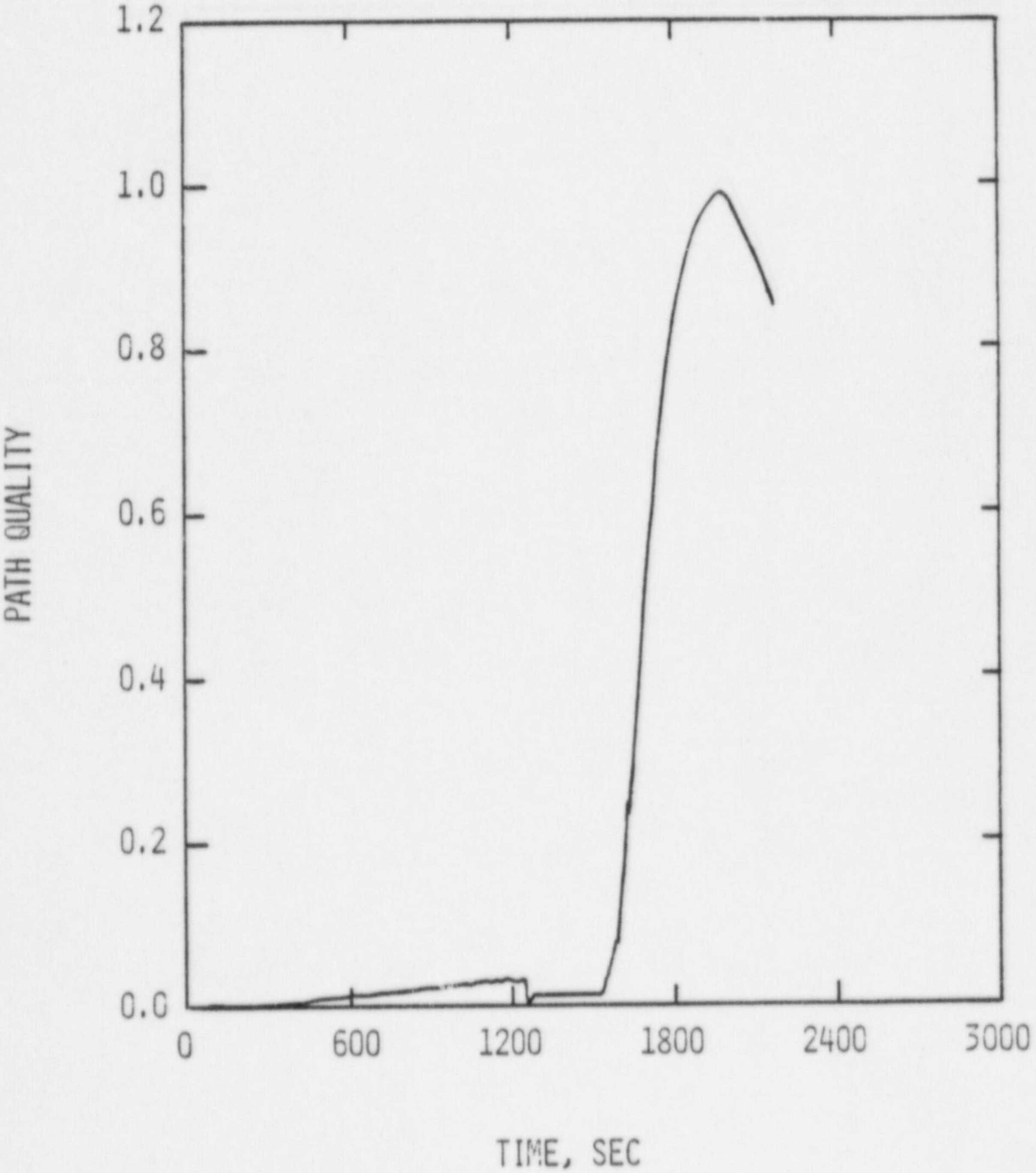


FIGURE 7

2700 MWt PLANT ANALYSIS
INADVERTANT OPEN PORV
RCS FLUID TEMPERATURES

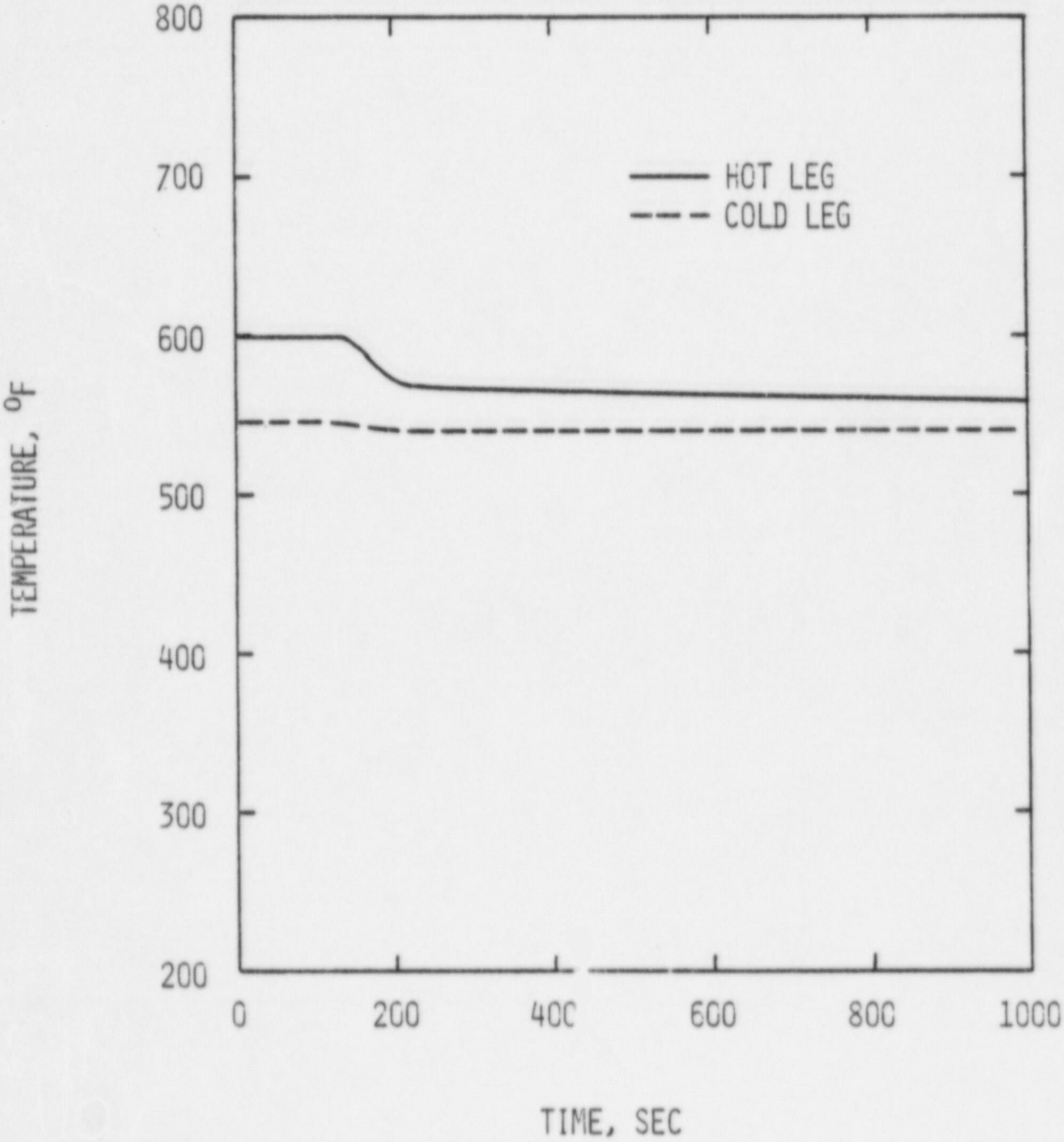


FIGURE 8

2700 MW_T PLANT ANALYSIS
INADVERTENT OPEN PORV
HOT LEG SUBCOOLING

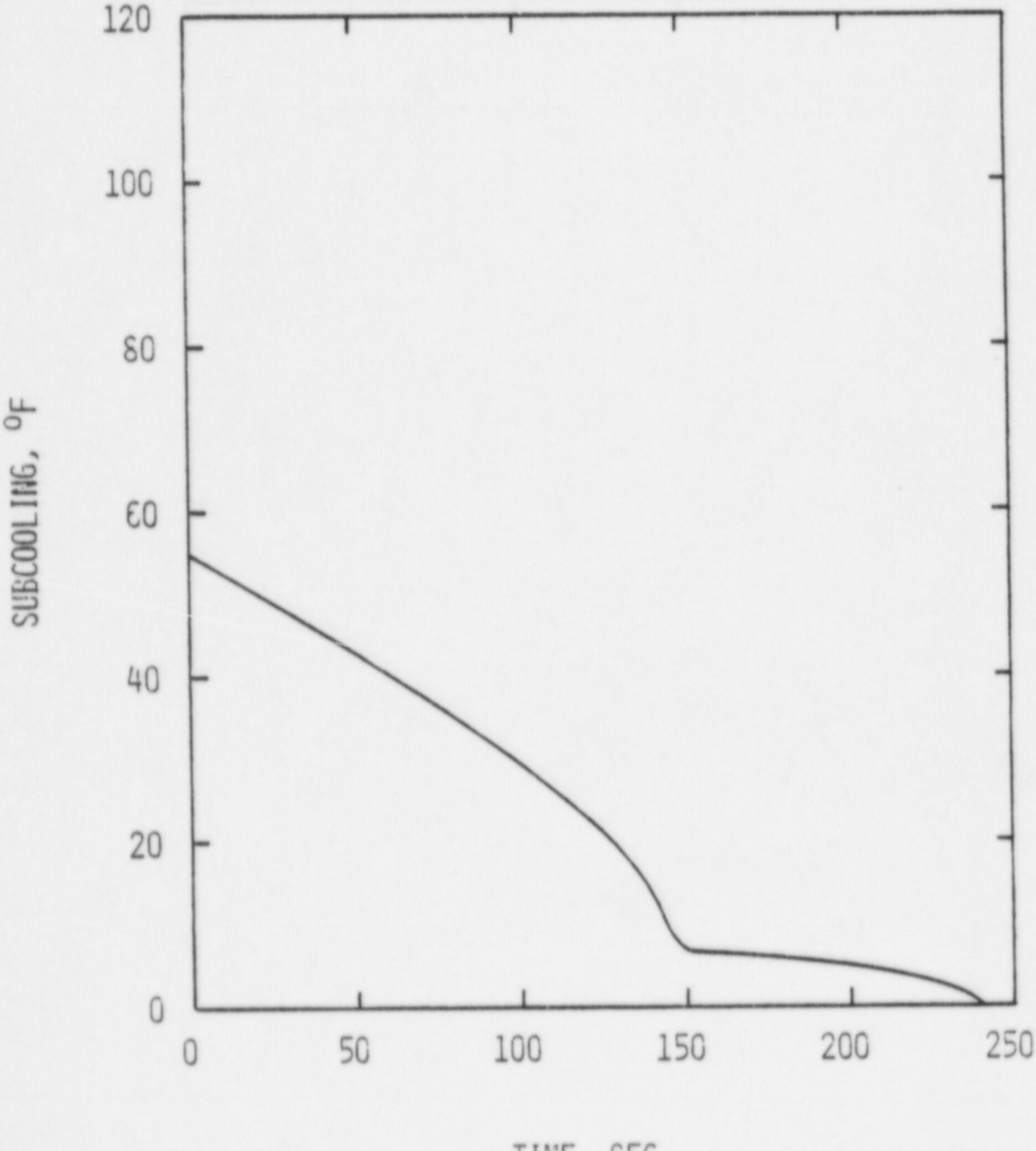


FIGURE 9

2700 MWt PLANT ANALYSIS
0.1 FT^2 HOT LEG BREAK LICENSING ANALYSIS
BREAK FLOW

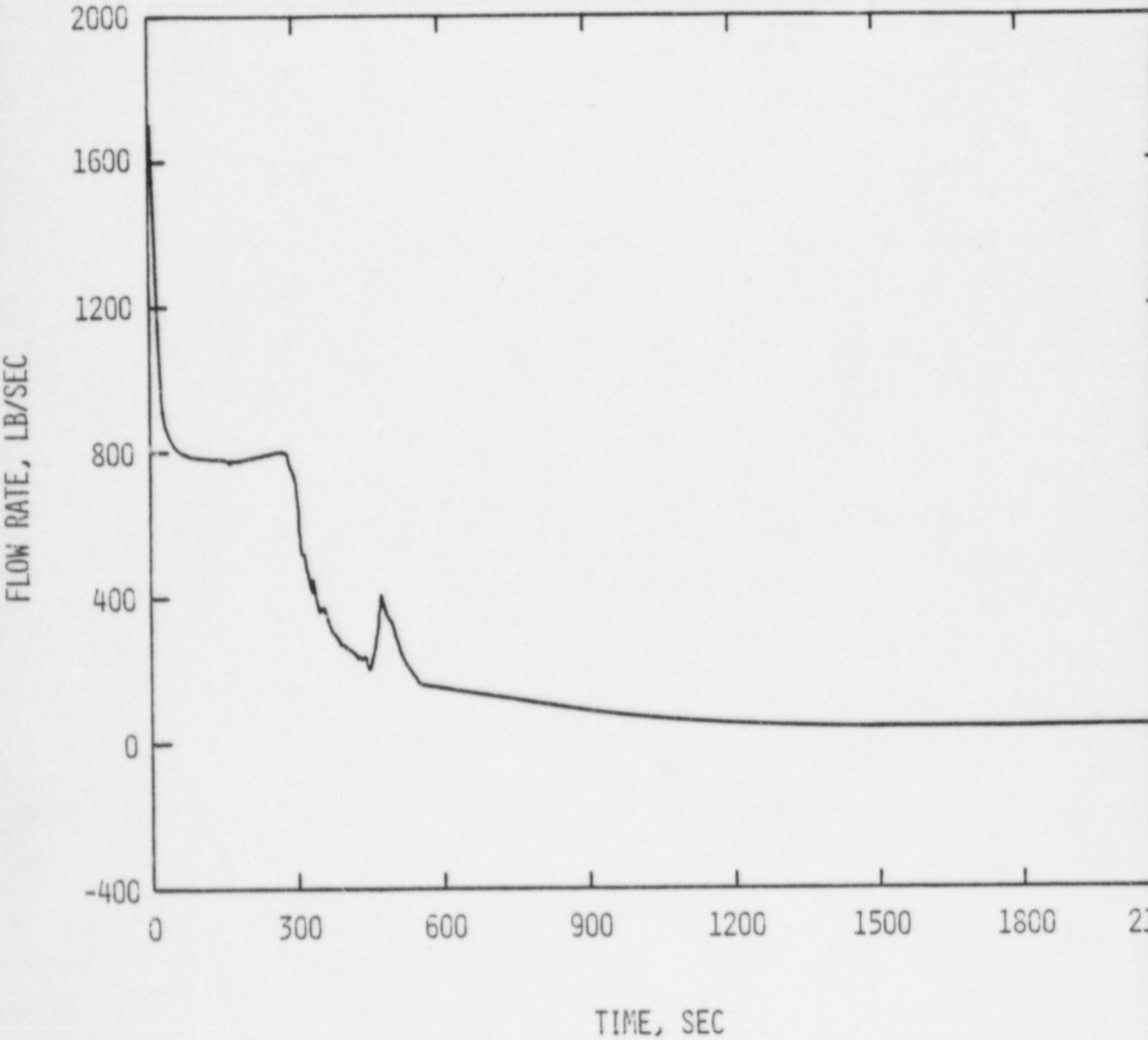


FIGURE 10

2700 MWT PLANT ANALYSIS
0.1 FT^2 HOT LEG BREAK LICENSING ANALYSIS
QUALITY IN FLOW PATH 3

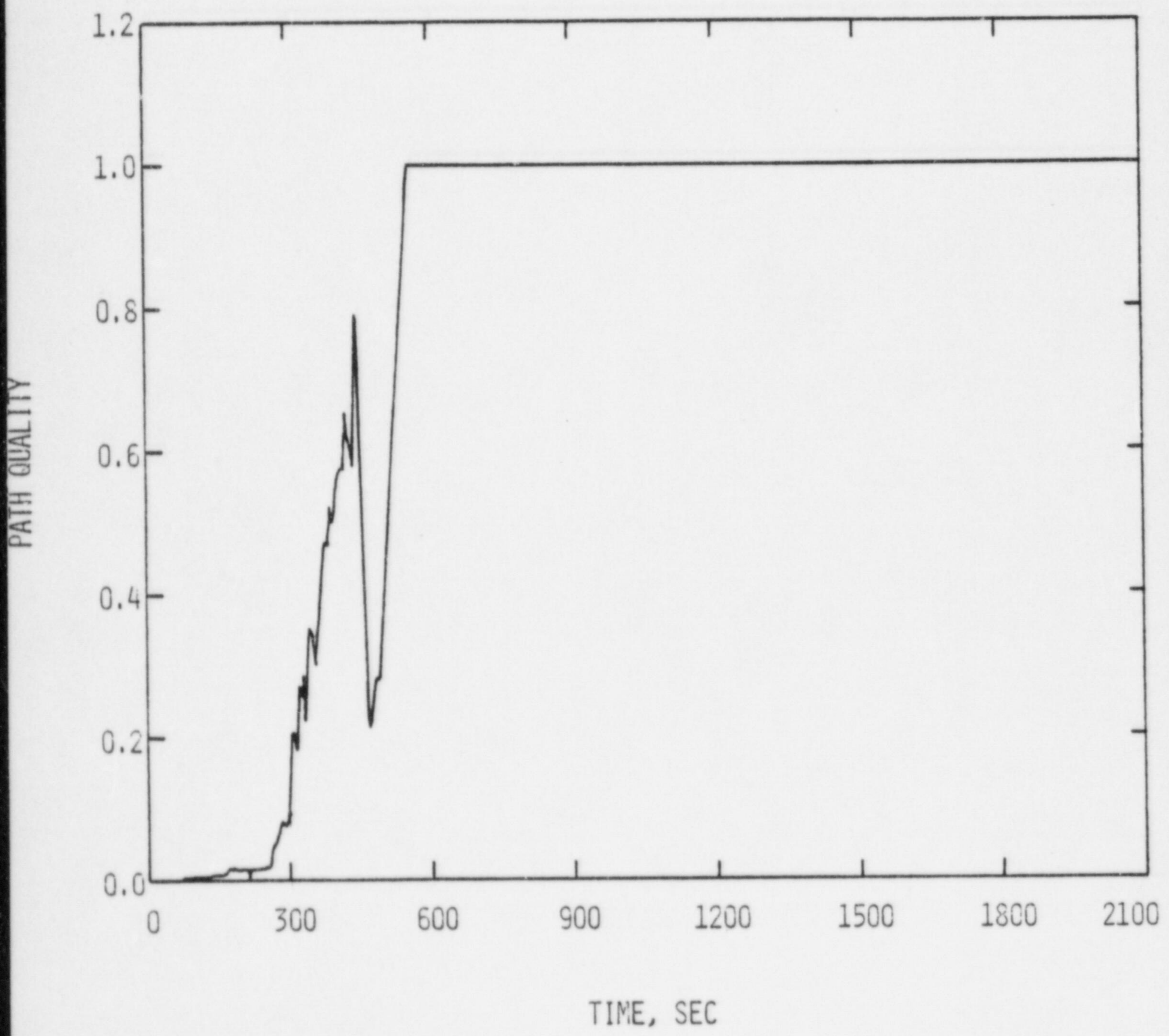


FIGURE 11

SBLOCA NODING DIAGRAM USED IN CEFLASH-4AS FOR RCP TRIP ANALYSES

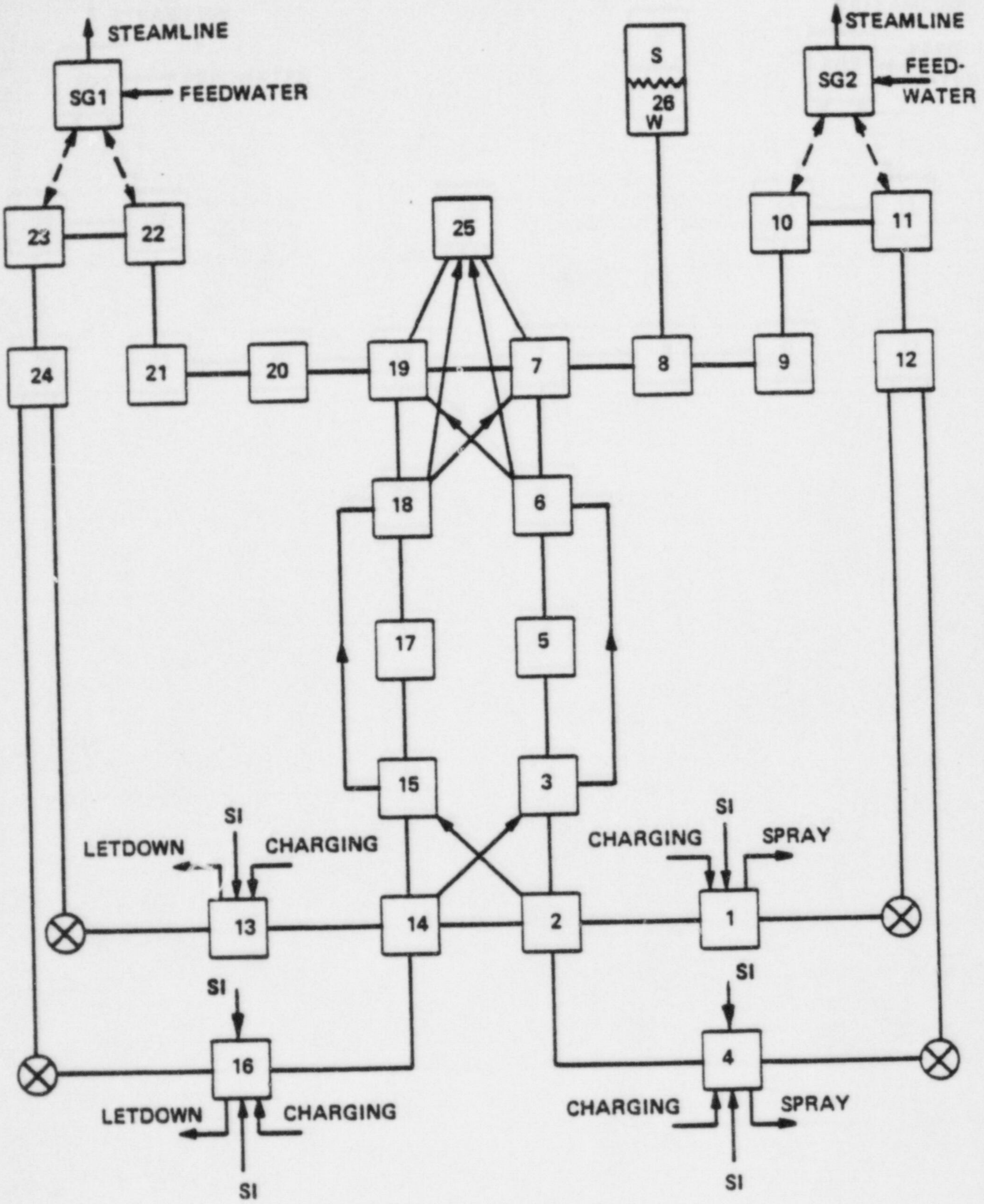
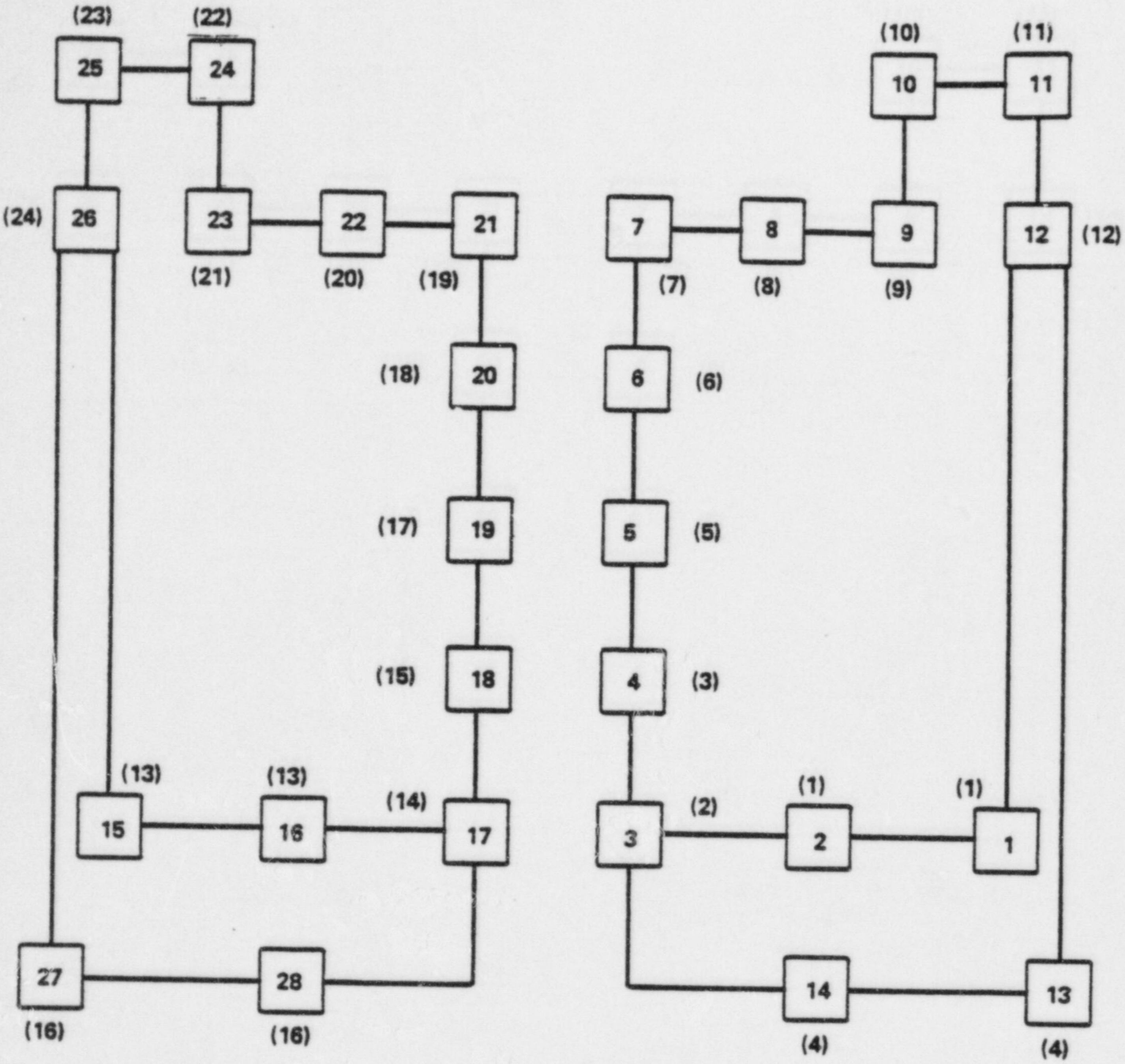


FIGURE 12

CESEC NODING DIAGRAM USED IN THE SGTR AND SLB ANALYSES FOR THE RCP TRIP STUDY



(1) NUMBER INSIDE THE PARENTHESIS REFERS TO THE CORRESPONDING THERMAL-HYDRAULIC NODE WHOSE PROPERTIES ARE USED IN THE CALCULATION

(2) THE AVERAGE OF THE PROPERTIES AND THE TOTAL FLOWS FROM PARALLEL NODES ARE USED FOR NODES REPRESENTING THE REACTOR VESSEL

(3) FOR NODES OTHER THAN THE REACTOR VESSEL AND THE REACTOR COOLANT PUMPS (1,13,15,27),THE AVERAGE OF THE UPSTREAM FLOW AND THE DOWNSTREAM FLOW IS USED

FIGURE 13

CESEC FLOW MODEL NODAL SCHEME

FIGURE 14

2700 MW_T PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

SG SECONDARY SIDE PRESSURE

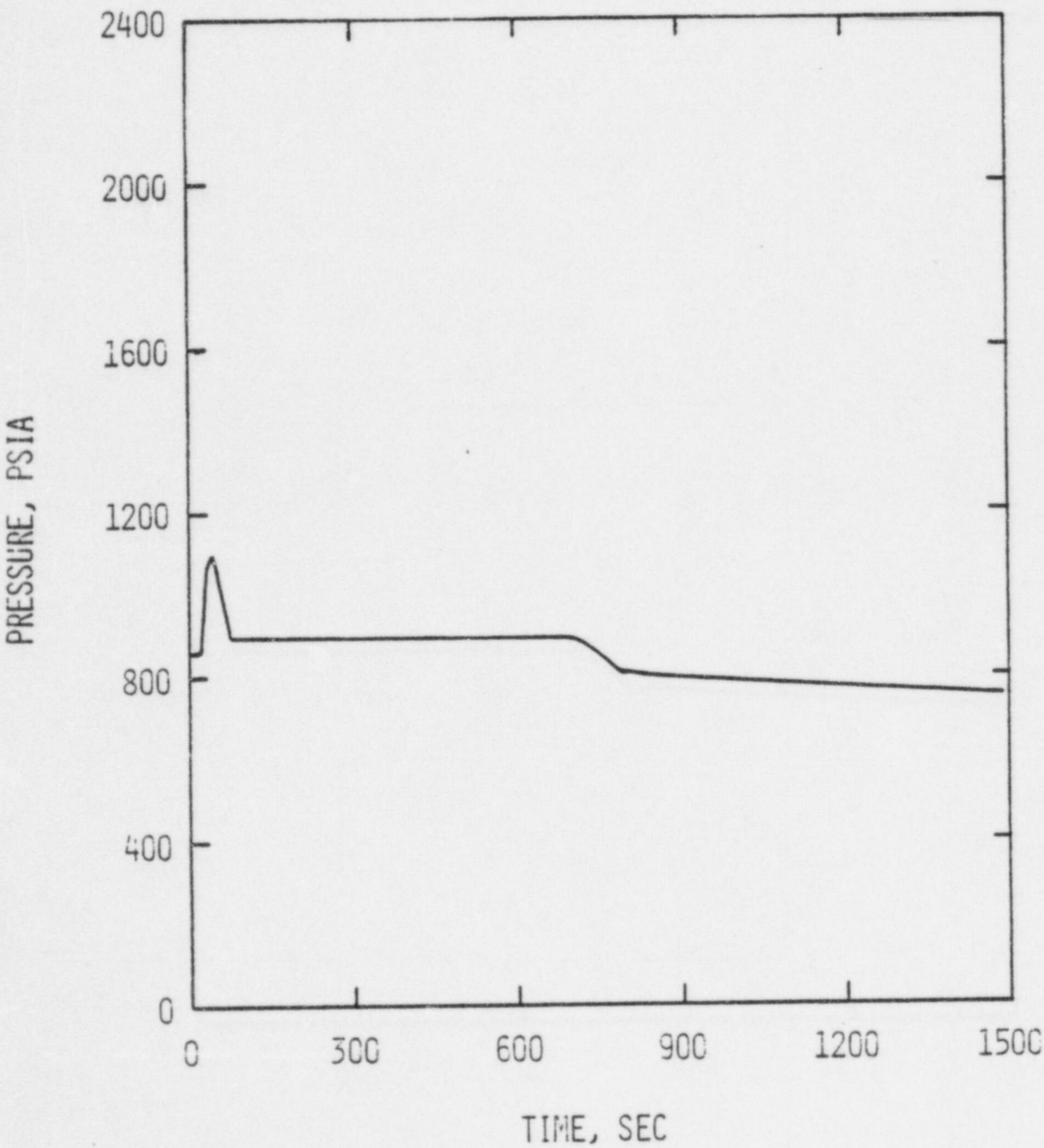


FIGURE 15

2700 MW_T PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

RCS TEMPERATURES

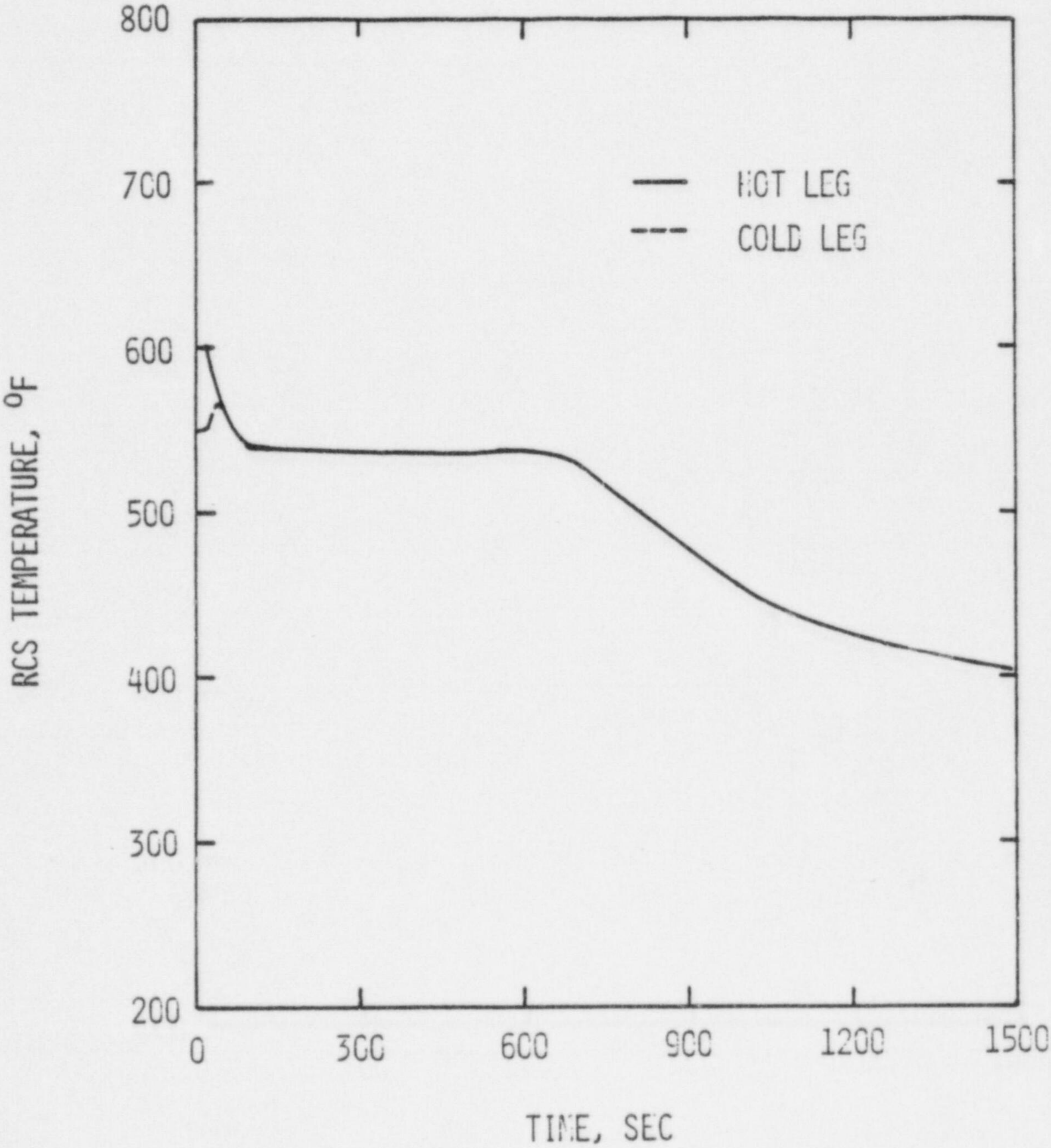


FIGURE 16

2700 MWT PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

INTEGRATED BREAK FLOW

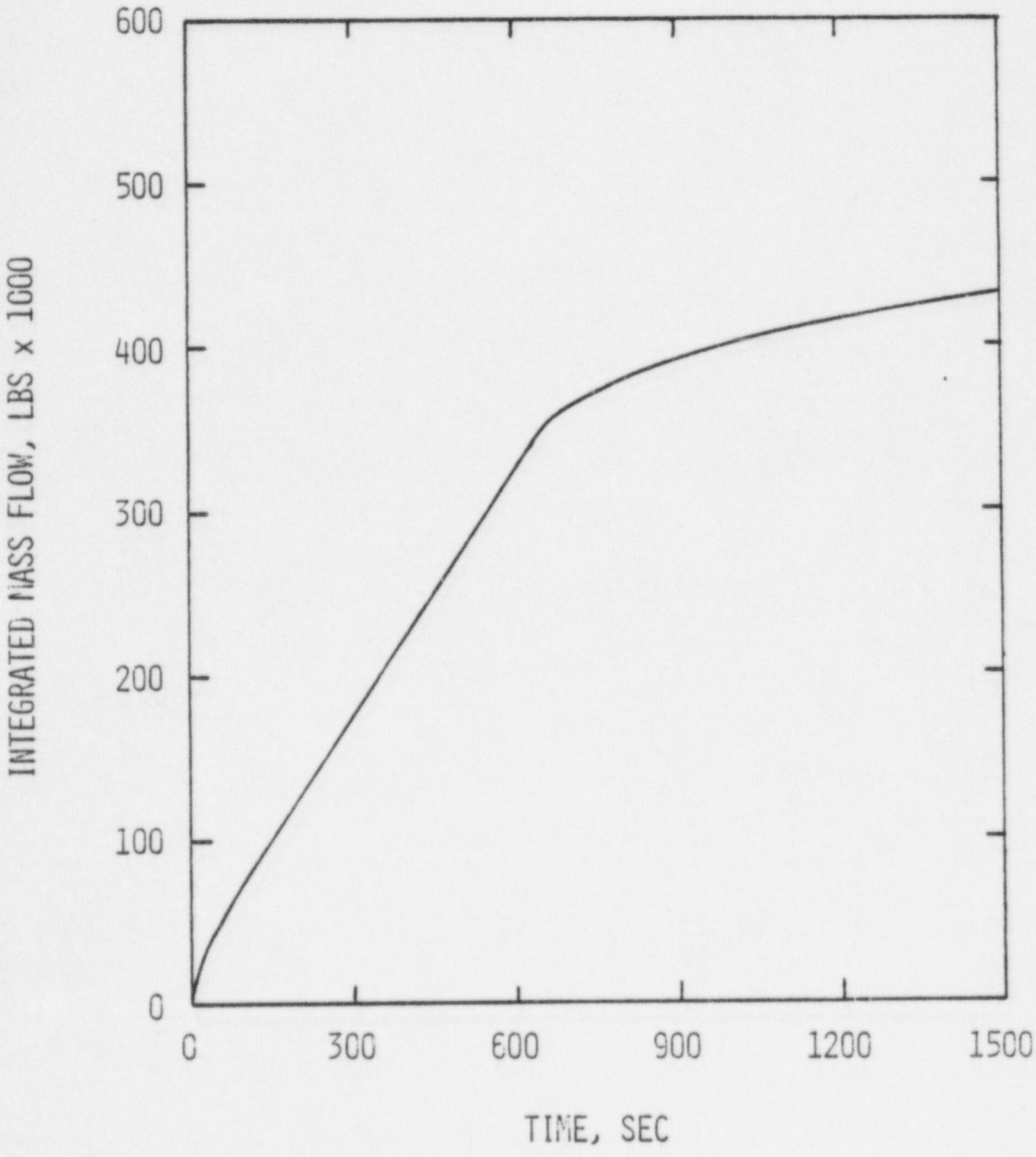


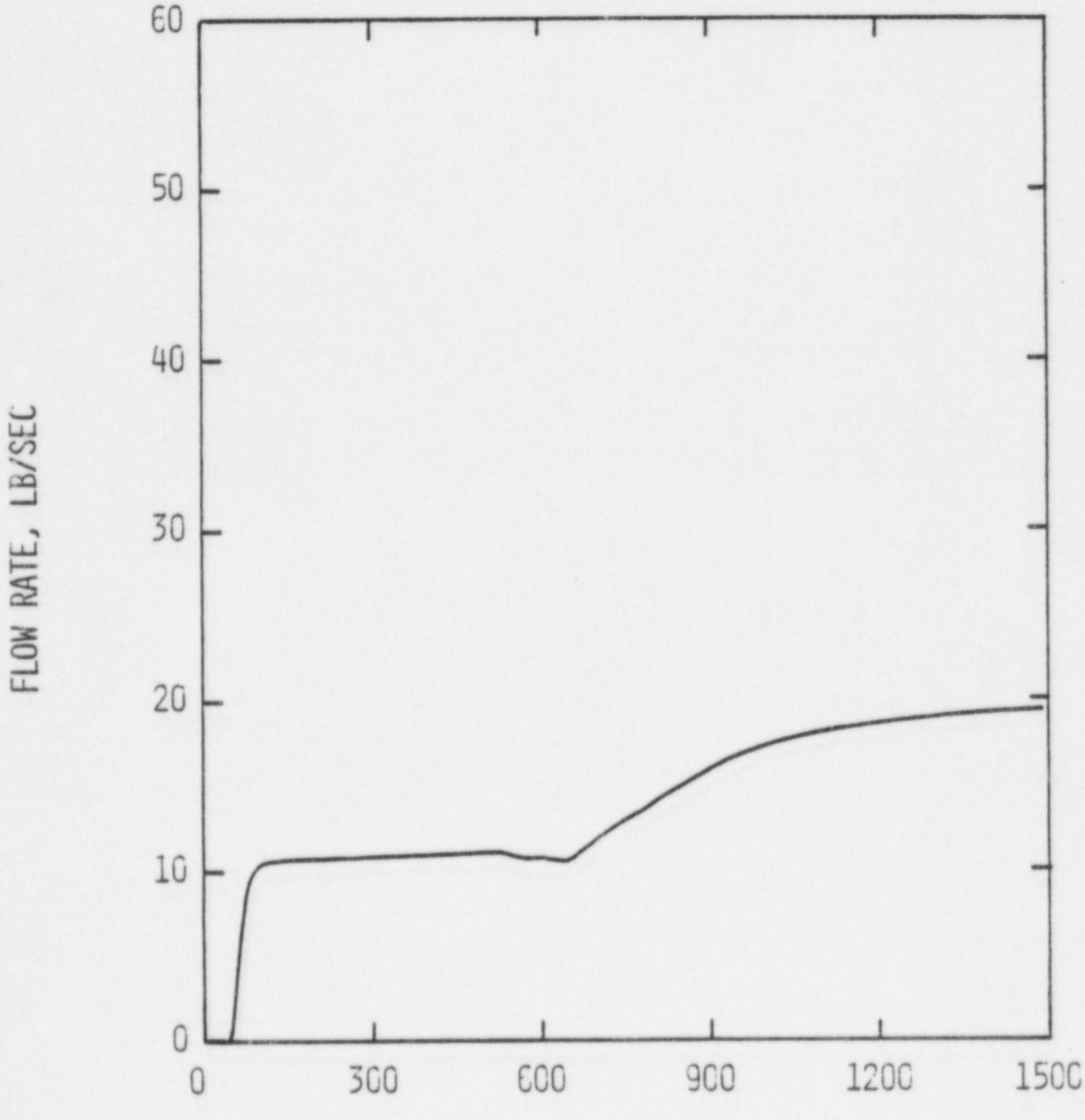
FIGURE 17

2700 MWt PLANT ANALYSIS

0.1 FT^2 HOT LEG BREAK

STEAM BYPASS SYSTEM OPERATIONAL

SAFETY INJECTION FLOW INTO EACH COLD LEG



APPENDIX A

MEETING SUMMARY ON REACTOR COOLANT PUMP TRIP
COMBUSTION ENGINEERING OWNERS GROUP 10/04/84

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 20 1984

MEMORANDUM FOR: Brian W. Sheron, Chief
Reactor Systems Branch
Division of Systems Integration

THRU: G. Norman Lauben, Section A
Section Leader
Reactor Systems Branch
Division of Systems Integration

FROM: Edward D. Throm, Section A
Reactor Systems Branch
Division of Systems Integration

SUBJECT: MEETING SUMMARY ON REACTOR COOLANT PUMP TRIP
COMBUSTION ENGINEERING OWNERS GROUP 10/04/84
REVISION 1, INCORPORATES CE/CEOG COMMENTS

The purpose of this memorandum is to provide a summary of the meeting held with the Combustion Engineering Owners Group concerning the Reactor Coolant Pump trip issue, MOR G-1. The meeting was held on October 4, 1984 in Hartford, Connecticut. A list of attendees is provided in enclosure 1.

The meeting was held to discuss the NRC Request for Additional Information on the Owners Group submittal, CEN-268, in response to Generic Letter 83-10. A copy of the questions is provided in enclosure 2, for ready access. The memorandum is intended to provided a record of the status of the Owners Group responses to these questions. Each question is addressed, and is identified as either complete(C), plant specific implementation(P), or requires written response (R) from the Owners Group.

1. (C) The Executive Summary of CEN-268 responds to the reasons for wanting the pumps available.

2. (C) The conservative best-estimate analysis assumes only one HPI pump is available.

3. (C) The conservative best-estimate decay heat is based on the 1979 ANS standard (nominal plus 2-sigma) with a 1.0 multiplier. The 1971 ANS decay heat, with a 1.2 multiplier, is used for Appendix K studies.

4. (C) Data provided in Owners Group letter RWW-84-44, June 25, 1984.

5. (C) The HPI flow characteristics used for the best-estimate analysis is a generic curve to cover all plants. For the evaluation model, the worst curve is used.

6. (C) The steam-generator primary-cold-side liquid is not considered because it will not drain into the reactor vessel via the hot legs when RCPs are turned off. The inventory does not provide core cooling.

7. (R) A written response will be provided.

8. (R) The Owners Group will provide quality data in lieu of void fraction data near the break location.

9. (C) The CE model does not directly use a hot rod peaking factor, but instead uses axial and radial power shapes. An equivalent hot rod peaking factor is 2.463.

10. (C) The accumulator pressures are 200 psig for the 2700 Mw(t) plant, and 600 psig for the 3410 Mw(t) plant.

11. (C) Page 3.3 of CEN-268 addresses this question.

12. (C) The accumulator safety injection tank pressure governs the limiting break size. The limiting break is just small enough to prohibit accumulator injection. For the 2700 Mw(t) plant, the size is 0.10 square feet, and for the 3410 Mw(t) plant, 0.05 square feet.

13. (R) (See 8. above)

14. (C) The appropriate safety injection curves are used for each plant type.

15. (C) Sensitivity studies performed in CEN-114 show the 0.1 square foot break to be the limiting size.

16. (C) Temperature data from both hot legs and all four cold legs are available for use in obtaining the reactor coolant system subcooling.

17. (P) The location of the temperature instruments and HPI locations is plant specific. Since it is expected that circulation will exist, either natural or reverse loop, the concern of false temperature data is not relevant (C).

18. (C) Flow is circulating in the loop with the tripped pump, at a level of about 10 to 20 per cent of nominal. (See 17. above)

19. (R) A written response will be provided.

20. (C) All computer program nodal models explicitly represent the hot and cold legs. No leg combining is used.

21. (C) Reactor coolant pump energy is not considered for the SBLOCA studies.

22. (R) A written response will be provided.

23. (R) A written response will be provided.

24. (P) Plant specific, to be addresses during implementation by each licensee.

25. (R) A written response will be provided.

26. (C) The reactor trip set-point is 1750 psia. HPI comes on at 1225 psia, at approximately 240 seconds.

27. (R) A written response will be provided.

28. (C) The study is based on a CEN-114 analysis (Case 4 of Appendix). A re-analysis was done only to the time when reactor coolant pump trip would occur.

29. (R) A written response will be provided.

30. (R) A written response will be provided.

31. (C) Data provided in Owners Group letter RWW-84-44, June 25, 1984. Letdown flow rate is based on the pressurizer level (linear model) and is isolated on a safety injection signal.

32. (C) Different plant models were used for each study. For the SBLOCA case, instrumentation uncertainties due to post-event environment result in using a lower pressure trip set-point.

33. (C) Simultaneous turbine trip on reactor trip was assumed for both cases.

34. (C) The EPGs require pump trip if pump integrity is challenged.

35. (C) The CESEC-III computer program and nodal model is used to study the steam-line break, and vessel mixing.

36. (C) Reactor coolant pump heat is considered for the steam-line break.

37. (C) (See 35. above)

38. (C) Text wil be corrected, as reflected in question.

39. (R) The limiting case is based on studies performed in CEN-128.

40. (C) A SBLOCA inside containment will set-off the radiation alarms.

41. (C) Letdown line breaks outside containment will be automatically isolated, or manually isolated by the operator.

42. (R) Data will be provided by the Owners Group.

43. (R) A written response will be provided.

44. (R) A written response will be provided.

45. (R) (See 8. above)

46. (C) Data is provided in CEN-114, Case P14. Plots are provided in CEN-115.

47. (C) Only one Appendix K calculation was required by Generic Letter 83-10.

48-55. (R) A written response will be provided.

56. (P) Plant specific, to be addresses during implementation by each licensee.

57. (R) A written response will be provided.

58. (R) The CESEC-III computer program nodal model was provided to the NRC contractor. The SBLOCA nodal model will be provided in a written response.

59. (R) The SGTR data is not readily available. If the NRC believes the information is required, the NRC will contact the Owners Group. The SBLOCA data will be provided in a written response.

At the end of the meeting, the Owners Group indicated that the written responses would be available in December 1984. Recent conversations with the Owners Group and CE indicates that the responses may be available in November 1984.

Edward D. Throm

Edward D. Throm, Section A
Reactor Systems Branch
Division of Systems Integration

Enclosures; As stated

cc: R. Bernero
R. Houston
G. Lainas
J. Miller
D. Jaffe
G. J. E. Willcutt, J., LANL
Rik W. Wells, Chairman, CE Owners Group
C. Molnar, CE

APPENDIX B

REQUEST FOR ADDITIONAL INFORMATION ON CEN-268

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 666-6911

June 25, 1984
RWW-84-44

Mr. James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing
U.S. Nuclear Regulatory Commission
7290 Norfolk Avenue
Bethesda, Maryland 20014

Subject: Request for Additional Information on CEN-268

References: (1) J. R. Miller letter to R. W. Wells dated May 3, 1984
(2) Report CEN-268, "Justification of Trip Two/Leave Two Reactor Coolant Pump Strategy During Transients", March 1984
(3) R. W. Wells letter to H. R. Denton, dated October 19, 1982, "Communications Between the Combustion Engineering Owners Group and the Nuclear Regulatory Commission"

Dear Mr. Miller:

The enclosed information is provided by the Combustion Engineering Owners Group (CEOG) in response to your request for additional information, Reference (1), concerning the evaluation of the trip two/leave two reactor coolant pump trip strategy and trip setpoints, Reference (2).

This letter is provided according to the terms stated in Reference (3), a copy of which is attached for your convenience. In particular, the information provided in this letter is not applicable to any licensee or licensee applicant until the letter is referenced by that licensee or licensee applicant for use in his docket. Please send copies of any correspondence concerning this submittal to individuals identified in the attached list.

If you have any additional comments or questions on this subject, please feel free to contact me at (203) 665-3614.

Sincerely,

Rik W. Wells

Rik W. Wells, Chairman
CE Owners Group

RWW/djr
Enclosure

ENCLOSURE

ADDITIONAL INFORMATION REQUESTED TO
EVALUATE CEN-268

The following information was requested (Reference 1) in order to perform confirmatory analyses to evaluate the adequacy of the trip two/leave two RCP trip strategy and trip setpoints (Reference 2). The data supplied reflect the input applicable to the Reference 2700 MWt plant used by C-E in the conservative best estimate (CBE) analyses of the small break loss-of-coolant accident (SBLOCA) and the representative steam generator tube rupture (SGTR) transient. Differences between the SBLOCA and SGTR input which affect the calculated results are indicated in the following data. The results from these analyses are summarized in Reference 2.

ITEM 1

Safety Injection (SI) Data.

(a) High Pressure Injection (HPI) and Coolant Charging (CC) flow rates to each loop, and spillage assumptions.
(b) HPI and CC temperatures.
(c) SI initiation signal and delay time.

RESPONSE

The total safety injection flow rate versus pressure used in the SBLOCA CBE analyses and the SGTR analysis is provided in Table 1. For the SBLOCA CBE analysis, one-quarter of the total flow (i.e., one high pressure safety injection (HPSI) pump and one low pressure safety injection (LPSI) pump) was assumed to flow into each of the four cold legs. For the SGTR analysis, one-quarter of the total safety injection flow (i.e., 2 HPSI pumps and 2 LPSI pumps) was assumed to flow into each of the four cold legs.

One-half of the constant charging total flow rate of 44 gpm from one pump was assumed to flow into each of two cold legs in opposite coolant loops in the SBLOCA analysis. The SGTR charging pump assumptions are included in the response to item 2.

Spillage of the safety injection and charging water was not assumed since the SBLOCAs analyzed were hot leg breaks and also spillage is not applicable to the analysis of SGTR transients.

The temperature used in the SBLOCA CBE analyses for the safety injection and charging water was 70°F. The SGTR analysis assumed 135°F safety injection water. The safety injection actuation signal (SIAS) was assumed to occur on low pressurizer pressure at the nominal setpoint of 1600 psia with a 0.9 sec instrument delay time.

ITEM 2

Is the Make-Up/Letdown system modeled?

RESPONSE

The make-up and letdown systems were not modeled for the SBLOCA analyses with the exception of one charging pump and isolation of letdown.

The following describes the modeling of the make-up and letdown systems for the SGTR analysis. Under normal operating conditions, the charging system supplies 44 gpm (prior to circulation through the regenerative heat exchanger) at 550°F to the RCS. Two standby charging pumps are activated sequentially if the pressurizer level deviation signal (calculated pressurizer level minus programmed pressurizer level setpoint (Figure 1)) decreases to less than -9 inches and -14 inches, respectively. Each standby charging pump then delivers 44 gpm. The two standby charging pumps are turned off if the deviation signal exceeds -4 inches and -6 inches, respectively. When a SIAS occurs all three charging pumps supply the previously designated flow (total flow of 132 gpm).

Letdown flow for the SGTR analysis varies linearly from 34 gpm to 132 gpm (as would be measured downstream of the letdown heat exchanger) as the pressurizer level deviation signal varies from -3.6 inches to 32 inches, respectively. The C-E analysis assumed the letdown flow matches the charging flow initially. Subsequently, letdown flow is reduced to the minimum limit (34 gpm) before it is automatically isolated upon SIAS.

ITEM 3

What is the low pressure reactor trip setpoint value?

RESPONSE

The reactor trip setpoint on low pressurizer pressure used in the SBLOCA CBE analyses was 1750 psia and in the SGTR analysis was 1875 psia.

ITEM 4

Where is the hot leg break located? Is it in the loop with the pressurizer?

RESPONSE

The location of the break in the SBLOCA analysis is at the bottom of the hot leg in the coolant loop which does not contain the pressurizer.

ITEM 5

Provide the reactivity coefficients used for density and moderator feedback, in tabular form.

RESPONSE

Values of the change in reactivity versus changes in moderator density and fuel temperature (Doppler) which were used in the analyses are provided in Tables 2 and 3, respectively. These values were normalized to yield zero reactivity change at the beginning of the transient.

ITEM 6

Is there steam generator isolation on containment over-pressure? If so, then the bypass valves would not be operable for SBLOCA analyses. No isolation is assumed for the generic analyses. Are there any licensees who have SG isolation on containment over-pressure, or are planning to install this feature?

RESPONSE

The steam bypass system was modeled since it is typically set in the automatic mode during normal full power operation and thus its operation is a valid CBE assumption for a SBLOCA. The CBE analyses for evaluation of the trip setpoints assumed that a high containment pressure signal would not be generated for the SBLOCA.

In addition, a SBLOCA analysis was made assuming the steam bypass system was in the manual mode and hence not operational, which is equivalent to steam generator isolation at the start of the SBLOCA. The results of this analysis were reported in Section 3 of CEN-268 and they demonstrate the acceptability of the trip two/leave two strategy. These results provide an upper bound on the analyses of the plants which have or may have steam generator isolation on high containment pressure.

REFERENCES

1. NRC Letter from James R. Miller (NRC) to R. Wells (Chairman, CEOG), dated May 3, 1984.

2. Report CEN-268, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients,", March 1974.

Table 1

Safety Injection Flow
2700 MWt Reference Plant Analysis

SBLOCA(1)		SGTR(2)	
RCS Pressure (psia)	Flow (gpm)	RCS Pressure (psia)	Flow (gpm)
1225	0	1165	0
1100	200	1105	272
1000	275	1008	452
900	330	929	540
800	375	824	632
700	410	711	720
600	440	591	812
500	480	445	900
400	510	290	992
300	540	166	1080
209	1282	135	2880
100	3312	105	3960
0	4535	0	6480

(1) SBLOCA CBE analyses assumed 1 HPSI pump and 1 LPSI pump.

(2) SGTR analysis assumed 2 HPSI pumps and includes 2 LPSI pumps.

<u>Table 2</u>

Change in Reactivity vs. Moderator Average Density
2700 MWt Reference Plant Analysis

Reactivity (Δk)	Average Moderator Density (lb/ft^3)
-0.23220	0.0
-0.12800	8.6144
-0.05570	17.230
-0.01400	25.844
-0.00333	30.151
-0.00191	31.013
0.000293	32.737
0.00171	34.459
0.00214	35.321
0.00253	37.044
0.00234	38.767
0.00144	40.920
0.0	43.074
-0.00176	45.228
-0.00660	51.688
-0.00718	53.843
-0.00677	55.997

Table 3

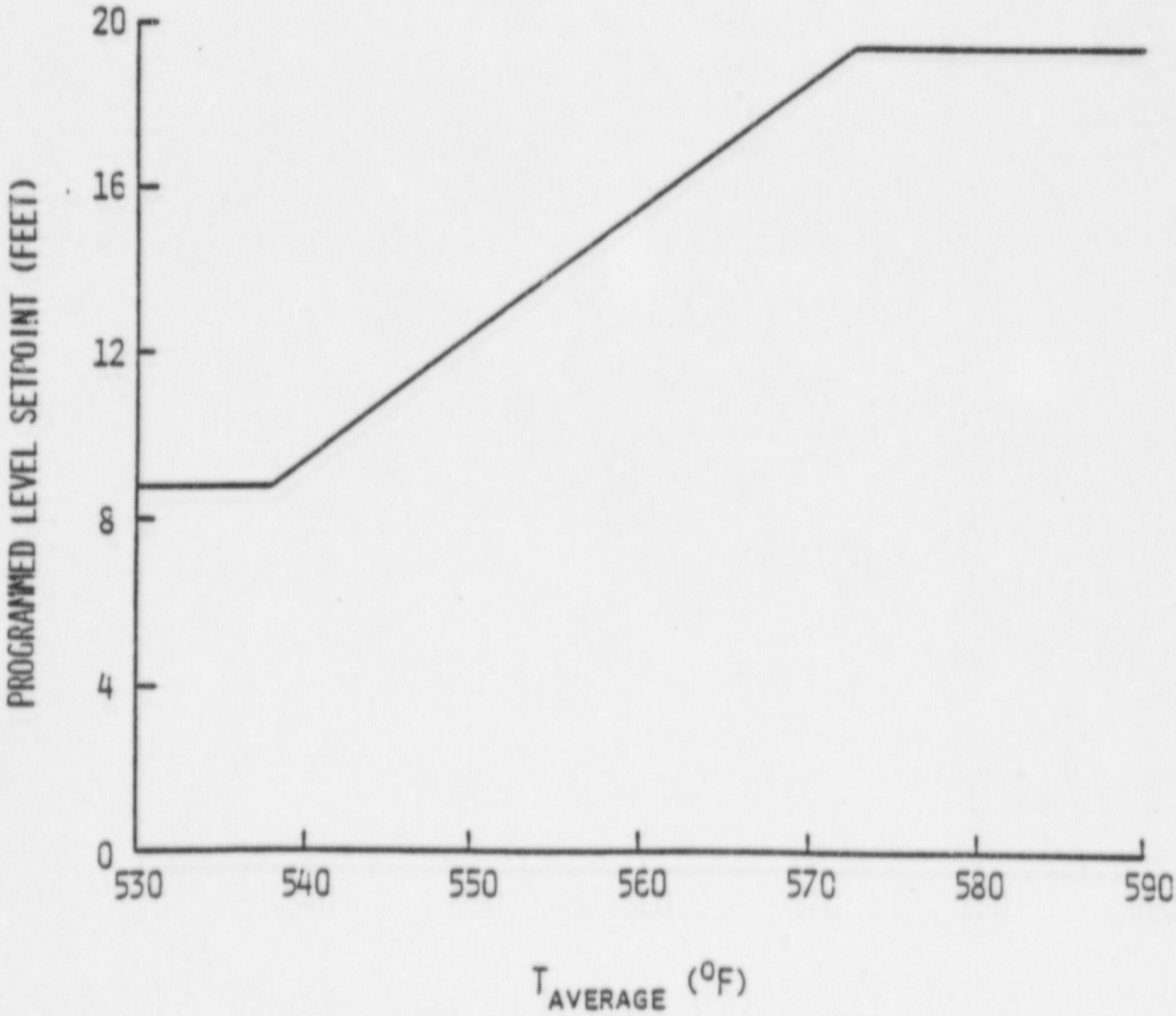
Change in Reactivity vs. Fuel Temperature
2700 MWt Reference Plant Analysis

Reactivity (Δk)	Core Average Fuel Temperature (°F)
0.022655	200.
0.014950	600.
0.011500	800.
0.008050	1000.
0.002300	1400.
-0.000340	1600.
-0.002125	1800.
-0.003825	2000.
-0.004420	2080.
-0.004420	5000.

FIGURE 1

PROGRAMMED PRESSURIZER LEVEL SETPOINT

COMBUSTION ENGINEERING REFERENCE 2700 MWT PLANT



COMBUSTION ENGINEERING, INC.