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SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGY

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ABSTRACT

This report is presented to demonstrate TU Electric's application of USNRC-approved Siemens Power Corporation's (SPC) Emergency Core Cooling Systems (ECCS) Evaluation Model EXEM PWR Small Break Model, to the Comanche Peak Steam Electric Station (CPSES).

This report contains a description of the EXEM PWR Small Break methodology which includes the computer codes, the details of the nodalization schemes, and the calculational procedures followed during all phases of the Loss-of-Coolant Accident (LOCA) analyses. The methodology is used to perform small break LOCA-ECCS licensing analyses that comply with USNRC regulations contained in 10 CFR 50.46 and 10 CFR 50, Appendix K. The method also satisfies the requirements of NUREG-0737, TMI Action Item II.K.3.30.

In order to comply with a 10 CFR 50, Appendix K requirement, a spectrum of small breaks, ranging from 2 through 4 inches in diameter, is examined.

Although higher peak clad temperatures (PCT) are usually associated with beginning of life (BOL) fuel because of the higher stored energy, a fuel burnup study is also conducted. This is done in order to confirm that the end of cycle (EOC) pin pressures, which are higher than those encountered early in life and which foster a higher driving force for rod burst, do indeed result in lower PCT for the fuel under consideration.

All system analyses were performed with ANF-RELAP using an explicit representation of the CPSES-1 4 loops. However, because CPSES-1 has no significant loop asymmetries, a model using the more customary 2-loop representation, with a broken loop and a lumped intact loop has also been developed. The limiting analysis was repeated using this 2-loop model and yielded essentially identical results to the 4-loop model.

In order to further support the "robustness" of the findings, two additional types of sensitivity studies were performed. The first was a time step study demonstrating that all break spectrum results are converged. The second was the [

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The methodology presented herein — including all codes, results, input decks, inferrences and conclusions presented within this report — will be used to perform small break LOCA analyses and evaluations in compliance with 10 CFR 50.46 criteria and 10 CFR 50, Appendix K requirements, for fuel cycle analyses and to address pertinent licensing issues, for Comanche Peak Steam Electric Station Unit One and Unit Two.

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CHAPTER 1

INTRODUCTION

The main objective of this work is to obtain approval of TU Electric's application of Siemens Power Corporation's (SPC) methodology — including all codes, all input decks, all results, all inferences and conclusions — so that it may be applied to the Comanche Peak Steam Electric Station Unit One and Unit Two for fuel cycle analyses and to address pertinent licensing issues.

This report describes TU Electric's application of SPC's USNRC-approved (Reference 1.1) Emergency Core Cooling Systems (ECCS) Evaluation Model, entitled "EXEM PWR Small Break Model", to the Comanche Peak Steam Electric Station Unit One (CPSES-1).

The methodology is used to perform the Small Break LOCA-ECCS licensing analyses that comply with USNRC regulations contained in 10 CFR 50.46, 10 CFR 50, Appendix K, and the requirements of NUREG-0737, TMI Action Item II.K.3.30.

The analyses presented in this report include a description of the current EXEM PWR Small Break LOCA methodology (Chapter 2 and Appendix), including the details of the various nodalization schemes and procedures followed during all phases of the LOCA analyses, which is postulated to occur with the plant in normal operation. Each calculation is performed in compliance with the explicitly approved EXEM PWR Small Break LOCA methodology. Three principal computer codes are used. RODEX2 provides the initial fuel conditions. ANF-RELAP calculates the system thermal-hydraulic response including core boundary conditions. TOODEE2 is used to calculate hot rod behavior. Five types of sensitivity studies are presented in Chapter 3.

The first is a break spectrum study. Breaks ranging from 2 through 4 inches in diameter are examined in order to comply with 10 CFR 50.46 (a)(1)(i).

The second type of sensitivity is a time step study. These are performed for each of the cases within the break spectrum, in order to verify that a converged solution has been obtained.

The third type of sensitivity study identifies the bounding [ ]. This study is required by the SPC Safety Evaluation Report (SER) (Reference 1.1) and consists of re-analyzing the most limiting break size with the [ ].

All ANF-RELAP analyses discussed above and presented in Chapter 3 are performed using an explicit representation of the CPSES-1 4 loops. In the fourth type of sensitivity study, because CPSES-1 has no significant loop asymmetries, an ANF-RELAP model using the industry-wide customary representation, a 2-loop model with a single broken loop and a lumped intact loop, has also been developed and is presented in Chapter 2. In Chapter 3, the limiting analysis is repeated using this 2-loop model in order to show that it yields results essentially identical to the explicit 4-loop model.

The fifth type of sensitivity is a fuel exposure study. This is done in order to find out whether beginning of life (BOL) or end of cycle (EOC) result in lower PCT for the fuel under consideration.

In chapter 4, key results from base case analyses and sensitivity studies are summarized. Chapter 4 also summarizes how the most limiting small break LOCA case for the EXEM PWR Small Break methodology is determined, how the PCT is computed, and how compliance with the LOCA-ECCS criteria in 10 CFR 50, Appendix K for CPSES-1 and CPSES-2 is demonstrated.

The Appendix provides a description of the codes used in the EXEM PWR Small Break methodology, their interfaces, interrelationships, and respective inputs and outputs.

CHAPTER 2

DESCRIPTION OF THE METHOD

This report describes the application of USNRC-approved, Siemens Power Corporation's latest ECCS Evaluation model, entitled "EXEM PWR Small Break Model" (Reference 2.1), to the Comanche Peak Steam Electric Station Unit 1

The EXEM PWR Small Break methodology is illustrated schematically in Figure 2.1. For presentation purposes the methodology can be said to embody three basic types of calculations: (1) Determination of Initial Fuel Conditions (RODEX2), (2) System Thermal-Hydraulic Response (ANF-RELAP), and (3) Hot-Rod Thermal Response and Cladding Heatup (TOODEE2). These are discussed in the sections that follow. Additional details of the codes used in these calculations including interfaces, interrelationships, inputs and outputs are provided in the Appendix.

2.1 DETERMINATION OF INITIAL FUEL CONDITIONS

Calculations are required to determine initial fuel conditions for both ANF-RELAP and TOODEE2. [

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2.2 SYSTEM THERMAL-HYDRAULIC RESPONSE ANALYSIS

The system thermal-hydraulic response is analyzed using ANF-RELAP, a modified version of RELAP5/MOD2.

The explicit 4-loop ANF-RELAP system model used for CPSES-1 is described in detail in Section 2.4.1 and the lumped 2-loop model is described in detail in Section 2.4.2. The initial conditions of the ANF-RELAP fuel rod model, i.e. [

], as mentioned in Section 2.1 and described in the Appendix. The RELAP5/MOD2 code is described in detail in Reference 2.2. In addition to less significant changes and corrections, RELAP5/MOD2 has been modified in three major ways to produce ANF-RELAP:

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The ANF-RELAP calculation provides the thermal-hydraulic boundary conditions for the fuel thermal response analysis, which is performed using the TOODEE2 code (Reference 2.5). [

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2.3 HOT ROD THERMAL RESPONSE ANALYSIS

TOODEE2 (Reference 2.5) is used to calculate the hot fuel rod heatup during the entire accident. It is part of the original WREM package approved by the NRC (Reference 2.6) and is also part of the TU Electric Large Break LOCA methodology (Reference 2.7) currently used in performing large break loss-of-coolant accident analyses that comply with 10 CFR 50.46 and Appendix K thereto.

TOODEE2 is a two-dimensional, time-dependent fuel rod thermal and mechanical analysis program. TOODEE2 models the fuel rod as radial and axial nodes with time-dependent heat sources. Heat sources include both decay heat and heat generation via reaction of water with zircaloy. The code considers conduction within solid regions of the fuel, radiation and conduction across gap regions, and convection and radiation to the coolant and surrounding rods.

The outputs of TOODEE2, namely: peak clad temperature, percent local cladding oxidation and percent pin-wide cladding oxidation are compared to the 10 CFR 50.46 (b) (1) through (3) criteria. Regarding (3), if pin-wide oxidation is less than 1% it is concluded that the criteria of less than 1% core-wide oxidation is met.

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2.4 DESCRIPTION OF THE MODELS

2.4.1 CPSES-1 4-LOOP ANF-RELAP NSSS MODEL

The Comanche Peak Steam Electric Station has two Westinghouse pressurized water reactors. Both units are typical 4-loop plants with a rated thermal power of 3411 MWt each.

The CPSES-1 ANF-RELAP NSSS model reflects a considerable amount of engineering insight and experience and incorporates:

a. Information from the most recent plant drawings, design basis documents, vendor documents, Technical Specifications and Final Safety Analysis Report.

b. Careful consideration of the guidelines set forth by SPC for the application of their methodology (Reference 2.8).

This section describes the explicit 4-loop version of the ANF-RELAP base input model for the Comanche Peak Steam Electric Station Unit 1 (CPSES-1). The discussion of this model is divided into the following sub-sections:

1. Volumes, junctions and heat structures
2. Core power
3. Emergency core cooling systems
4. Trips and delays

Since there are no significant loop asymmetries only loop-1 will be discussed in this section, i.e., where the corresponding information for the other three loops is the same, the redundant information is omitted.

2.4.1.1 VOLUMES, JUNCTIONS AND HEAT STRUCTURES

Figure 2.2 shows the CPSES-1 nodalization diagram for the ANF-RELAP entire explicit 4-loop base input model. Table 2.1 identifies the volumes, junctions, and heat structures associated with the reactor vessel, pressurizer, loop-1 and other systems. It also provides node numbers for cross-referencing with Figure 2.3, which, in order to allow better visibility, shows only loop-1 and the reactor vessel. Table 2.2 summarizes key parameter values for the reactor vessel, pressurizer, loop-1 and other systems of the CPSES-1 ANF-RELAP NSSS explicit 4-loop base input model.

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Steam generator models include both primary and secondary sides. An appropriately detailed nodalization of the steam generator secondary has been implemented in order to insure realistic heat transfer behavior across the steam generator tubes.

Steam generator pressure relief is obtained by simulation of the safety valves only (Table 2.9), i.e. no credit is taken for the heat removal capability of the steam dump and bypass system nor the atmospheric relief valves. Five percent of the steam generator tubes are assumed plugged. This assumption is required by the methodology. It is made in these analyses in order to support the potential need for operation under these circumstances and is a conservative assumption for fewer obstructed tubes.

Reactor coolant pumps are modeled using Westinghouse homologous curves in the single phase regime combined with homologous difference and multiplier curves for the CE-EPRI tests in the two-phase regime. The CE-EPRI reactor coolant pump data were reviewed and found to be applicable to CPSES reactor coolant pumps.

The containment is represented by a time-dependent volume (TMDPVOL) with constant atmospheric pressure.

2.4.1.2 CORE POWER

The total core power during transients is determined by the point reactor kinetics model in ANF-RELAP. Conservative input data are entered for this model in order to compute the fission power with a 1.02 multiplier and decay heat with a 1.2 multiplier, per 10 CFR 50, Appendix K requirements. The model accounts for the reactivity effects associated with scram, change in moderator density and in fuel temperature. The effects are evaluated on a core average, cycle specific basis using the reactor physics methodology and associated uncertainty factors presented in Reference 2.9 to assure conservatism. For the analyses presented herein, reactivity feedbacks representative of the CPSES-1 core for cycle 5 have been selected and are shown in Tables 2.3, 2.4 and 2.5 for moderator density effects, fuel temperature effects and scram, respectively.

All core power is conservatively assumed to be generated in the fuel, i.e. none is deposited in moderator, cladding, or passive heat structures. This power is distributed according to the nodal power factor (NPF) entered for each active heat structure that represents a portion of UO_2 fuel. End of cycle convertion ratios are used to maximize actinide decay heat.

2.4.1.3 EMERGENCY CORE COOLING SYSTEMS

The CPSES ECC system is arranged into four subsystems: (1) the charging/safety injection, (2) high head safety injection, (3) low head residual heat removal injection, and (4) accumulators.

There are two safety injection trains. Each train contains one centrifugal charging pump, one high head safety injection pump, and one low head residual heat removal pump with associated piping, valves, controls, and instrumentation.

Loss of offsite power is assumed to occur coincidentally with the reactor trip. One diesel generator train is removed on the assumption that it (one train) fails to start. Therefore, only one train of safety systems are represented in the present NSSS model. This assumption is made in order to satisfy the single failure criterion, as discussed in Chapter 3.

All pumped systems take suction from the refueling water storage tank (RWST) during the injection phase. In the present analyses the RWST water temperature is taken at the maximum value (120 degrees F) allowed by the Technical Specifications. This is conservative since it minimizes heat removal by sensible heat transfer to injected fluid.

The pumped ECCS mass flow rates for each loop, versus pressure, for each injection system, which are given in Table 2.6, were derived from the values given in Reference 2.12.

The system contains four accumulators, one per loop. The minimum Technical Specifications (Reference 2.11) tank water volume (6119 gals. per tank) is used. The accumulators are modeled using a two-volume PIPE component (as opposed to the ACCUM component), per SPC methodology.

2.4.1.4 TRIPS AND DELAYS

The following trips and delays are used:

1. Reactor trip occurs on a low pressurizer pressure signal (1860 psia) plus a 2 second delay for signal processing. The 2.4 second rod travel time is accounted for by the scram reactivity (Table 2.5).

2. The reactor coolant pumps (RCP) are tripped at reactor trip, as discussed in Chapter 3. This trip occurs because at reactor trip offsite power is assumed lost. The RCPs cannot operate after a reactor trip if offsite power is not available.

3. Steam flow isolation is initiated at the time of reactor trip with the turbine stop valves taking 0.5 seconds to close (Table 2.7) following a 2 second signal processing delay. The steam dump and bypass system and the atmospheric relief valves are not credited. The safety valves operate as shown in Table 2.9.

4. Main feedwater isolation begins at the time of "S" signal plus a 7 second delay which includes 2 seconds for signal processing (Table 2.7).

5. SI Actuation Signal occurs on a low pressurizer pressure "S" signal (1715 psia) plus 2 seconds .

6. The delays for each of the pumped safety injection systems are given in Table 2.7.

7. Accumulators inject at set pressure (603 psia) without delay.

8. Available auxiliary feedwater (1 motor-driven pump) is assumed to be up and running 60 seconds after the "S" signal. "Cold" AFW injection is delayed for another 140 seconds, conservatively accounting for the flow travel time down the piping. During these 140 seconds, AFW is delivered at the higher main feedwater temperature. One motor-driven AFW pump is assumed lost due to the unavailability of offsite power, compounded with the failure of one diesel generator to start (single failure). Turbine-driven auxiliary feedwater (TDAFW) is not credited because it is difficult to demonstrate that it would be automatically activated on a steam generator Lo-Lo level signal prior to quenching of the fuel. However, preliminary calculations show that TDAFW shifts the most limiting break to a larger size (6 inches) and results in lower peak clad temperatures by approximately 250^{0}F. Thus, although the TDAFW is not considered in any of the

present analyses, it might be in future analyses, if adequate justification for its availability can be demonstrated.

2.4.2 CPSES-1 2-LOOP ANF-RELAP NSSS MODEL

The purpose of developing the 2-loop model is simply to obtain the same results as the 4-loop model but in considerably less computation time. Therefore, the 2-loop model duplicates the 4-loop model in every possible way.

The 2-loop version of the ANF-RELAP CPSES-1 NSSS model is derived from the 4-loop version in a direct manner: three of the four loops are assumed to be identical and are modeled as one lumped loop with appropriately scaled input. The pressurizer is connected to the intact lumped loop following usual modeling practices. The lumped loop represents the "unbroken" or "intact" loops. The "broken" loop remains the single loop-1 described in Section 2.4.1 and shown in Figure 2.3. Figure 2.4 shows the nodalization diagram for the entire 2-loop model. The data in Tables 2.1 and 2.2 also apply to this model except for the lumped loop data, which are not listed in order to avoid redundancy. The loop lumping is done using the standard modeling practice for deriving the lumped loop input from the single loop input as summarized below:

(1) Component flow areas are three times the area of the corresponding single loop component.

(2) Component lengths are identical in the lumped loop and in the corresponding single loop component.

(3) Component fluid volumes are three times the volume of the corresponding single loop component.

(4) Azimuthal angles remain zero.

(5) Inclination angles are identical in the lumped loop and in the corresponding single loop component.

(6) Elevation changes are identical in the lumped loop and in the corresponding single loop component.

(7) Wall roughnesses are identical in the lumped loop and in the corresponding single loop component.

(8) Hydraulic diameters are identical in the lumped loop and in the corresponding single loop component.

(9) Control flags are identical in the lumped loop and in the corresponding single loop component.

(10) Initial Conditions are identical in the lumped loop and in the corresponding single loop component.

(11) Junction flow areas are three times the area of the corresponding single loop junctions.

(12) Forward and reverse loss coefficients are identical in the lumped loop and in the corresponding single loop junctions.

(13) Junction flags are identical in the lumped loop and in the corresponding single loop junctions.

(14) Junction initial mass flow rates are three times the flow rates of the corresponding single loop junctions, whereas velocities are the same.

(15) The surface area factors for the heat conductors are three times the area factors of the corresponding single loop heat conductors.

(16) All control system parameters are identical in the lumped loop and in the corresponding single loop.

All features of the 4-loop model described in Section 2.4.1, including core power, emergency core cooling systems, trips and delays, are preserved in the 2-loop model and therefore need not be repeated.

2.4.3 TOODEE2 MODEL

TOODEE2 is used to calculate the temperature distribution in the hot rod. Table 2.8 summarizes the fuel geometry data used in the TOODEE2 model.

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The first and last axial nodes are identified as the bottom and top of the fuel rod, respectively. The TOODEE2 hot rod axial nodalization diagram is shown in Figure 2.5.

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The thermal-hydraulic boundary conditions for the TOODEE2 calculations are those associated with the [ ] region of the ANF-RELAP model, as described in Section 2.1 and in the Appendix.

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TABLE 2.1

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive
REACTOR VESSEL (RV)							
RV DOWNCOMER (DC)							
o Bottom DC	108	5	0	4	0	0	5
RV LOWER PLENUM (LP)							
o Bottom LP	109	1	0	0	0	0	1
o Middle LP	110	1	0	3	0	0	1
o Top LP	111	1	0	4	0	0	1
RV CORE BYPASS & BARREL/BAFFLE (BYPASS)							
o Bottom Bypass	128	3	0	2	0	0	3
RV CORE ACTIVE FUEL REGION							

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive

REACTOR VESSEL (RV) (cont'd)

[ ]

[ ]

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive

REACTOR VESSEL (RV) (cont'd)

[ ]

[ ]

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive
REACTOR VESSEL (RV) (cont'd)							
[ ]							
UPPER PLENUM & GUIDE TUBES (UP, GTs)							
o Bottom UP	166	1	0	6	0	0	1
o Guide Tubes	170	1	0	0	0	0	1
o Middle UP I	173	1	0	1	0	0	1
o Middle UP II	174	1	0	5	0	0	1
o Top UP	178	1	0	0	0	0	1
UPPER HEAD (UH)							
o Bottom UH	180	1	0	0	0	0	1
o Middle UH	181	1	0	5	0	0	1
o Top UH	182	1	0	0	0	0	1
LOOP-1 PRIMARY							
LOOP-1 HOT LEG (L1 HL)							
o L1 HL #1	410	1	0	2	0	0	1
o L1 HL's # 2&3	414	2	0	1	0	0	2
o L1 SG Inlet &	422	1	0	3	0	0	0
Outlet Plena	426	1	0	2	0	0	0

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive
LOOP-1 PRIMARY (cont'd)							
L1 STEAM GENERATOR (SG)							
o IL SG U-Tubes	424	8	0	7	0	0	8
L1 CROSS-OVER LEG (XLG)							
o L1 XLG	460	4	0	4	0	0	4
L1 REACTOR COOLANT PUMP (RCP)							
o L1 RCP	475	1	0	2	0	0	0
L1 COLD LEG (CL)							
o IL CL # 1 RCP Side	480	1	0	1	0	0	1
o IL CL # 2 Middle	490	1	0	1	0	0	1
o IL CL # 3 RV Side	495	1	0	0	0	0	1
LOOP-1 SECONDARY							
L-1 MAIN AND AUXILIARY FEEDWATER (MFW & AFW)							
o L1 MFW Source	502	0	1	0	0	0	0
o L1 MFW Flow	506	0	0	0	1	0	0
o L1 AFW Source	520	0	1	0	0	0	0
o L1 AFW Flow	525	0	0	0	1	0	0

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive
		LOOP-1 SECONDARY (Cont'd)					
L1 SG Vessel							
o L1 SG Downcomer	510	4	0	3	0	0	0
o L1 DC to Boiler	515	0	0	1	0	0	0
o L1 SG Boiler	540	5	0	4	0	0	0
o L1 SG Separator	560	1	0	3	0	0	0
o L1 SG Steam Dome	570	1	0	0	0	0	0
L1 SG STEAM LINE AND SAFETY VALVES							
o L1 Steam Line	575	0	0	0	1	0	0
o L1 Steam Sink	580	0	1	0	0	0	0
o L1 Safety Valve	585	0	0	0	1	0	0
o L1 S.V. Steam Sink	590	0	1	0	0	0	0
PRESSURIZER (PRZR - is connected to LOOP-4 Hot Leg in explicit 4-loop model)							
o PRZR Surge-Line	603	0	0	1	0	0	0
o PRZR Surge-Line	610	1	0	1	0	0	0
o PRZR Tank	620	6	0	5	0	0	6
ACCUMULATORS (ACCUM)							
o L1 ACCUM	720	2	0	1	0	0	0

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

Region -OR- System	Noding Diagram Number	Number of Volumes		Number of Junctions		Number of Heat Structures	
		Active	TMDP	Active	TMDP	Active	Passive
ACCUMULATORS (ACCUM) (cont'd)							
o L1 ACCUM Surge-	735	0	0	1	0	0	0
Line & Flow	730	1	0	0	0	0	0
EMERGENCY CORE COOLING SYSTEM (ECCS)							
ECCS							
o L1 HHSI Source	745	0	1	0	0	0	0
o L1 CCP Source	750	0	1	0	0	0	0
o L1 CCP Flow	770	0	0	0	1	0	0
o IL HHSI Flow	775	0	0	0	1	0	0
BREAK & CONTAINMENT							
BREAK							
o Break Junction Valve	805	0	0	1	0	0	0
CONTAINMENT							
o Containment	810	0	1	0	0	0	0

Table 2.2

SUMMARY OF CPSES-1 ANF-RELAP SYSTEM MODEL COMPONENTS

Component Number/Type		Area (ft^2)	Length (ft)	Volume (ft^3)	Inclination (Degrees)	Elevation Change (ft)	Surface Roughness	D_{Hyd} (ft)	Flags
100	BRANCH	13.33	5.85	77.96	-90.0	-5.85	0.00	1.53	100
102	BRANCH	13.33	5.85	77.96	-90.0	-5.85	0.00	1.53	100
104	BRANCH	10.38	2.29	23.78	-90.0	-2.29	0.00	0.98	100
106	BRANCH	10.38	2.29	23.78	-90.0	-2.29	0.00	0.98	100
108-1	ANNULUS	33.24	3.93	133.89	-90.0	-3.93	0.00	1.57	100
108-2	ANNULUS	32.42	4.00	129.68	-90.0	-4.00	0.00	1.45	100
108-3	ANNULUS	32.42	4.00	129.68	-90.0	-4.00	0.00	1.45	100
108-4	ANNULUS	32.42	4.00	129.68	-90.0	-4.00	0.00	1.45	100
108-5	ANNULUS	32.42	4.05	136.01	-90.0	-4.05	0.00	1.42	100
109	SNGLVOL	47.44	2.96	140.41	90.0	2.96	0.00	7.77	000
110	BRANCH	112.92	2.96	334.25	90.0	2.96	0.00	11.99	000
111	BRANCH	95.91	4.23	405.71	90.0	4.23	0.00	11.05	000
128-1	ANNULUS	25.08	4.00	100.31	90.0	4.00	0.00	0.008	000
128-2	ANNULUS	25.08	4.00	100.31	90.0	4.00	0.00	0.008	000
128-3	ANNULUS	25.08	4.00	100.31	90.0	4.00	0.00	0.008	000

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Table 2.2 (Cont'd)

SUMMARY OF CPSES-1 ANF-RELAP SYSTEM MODEL COMPONENTS

Component Number/Type	Area (ft^2)	Length (ft)	Volume (ft^3)	Inclination (Degrees)	Elevation Change (ft)	Surface Roughness	D_{Hyd} (ft)	Flags

[ ]

Table 2.2 (Cont'd)

SUMMARY OF CPSES-1 ANF-RELAP SYSTEM MODEL COMPONENTS

Component Number/Type	Area (ft^2)	Length (ft)	Volume (ft^3)	Inclination (Degrees)	Elevation Change (ft)	Surface Roughness	D_{Hyd} (ft)	Flags
166 BRANCH	84.84	1.28	108.60	90.0	1.28	0.00	0.0399	000
170 SNGLVOL	16.84	13.29	223.78	90.0	12.98	0.00	0.25	100
173 BRANCH	84.84	2.405	204.04	90.0	2.405	0.00	10.39	000
174 BRANCH	84.84	2.42	205.31	90.0	2.42	0.00	10.39	100
178 SNGLVOL	84.84	2.15	182.41	90.0	2.15	0.00	10.39	100
180 SNGLVOL	81.72	2.96	241.88	90.0	2.96	0.00	10.20	100
181 BRANCH	125.48	2.37	297.38	90.0	2.37	0.00	12.64	100

Table 2.2 (Cont'd)

SUMMARY OF CPSES-1 ANF-RELAP SYSTEM MODEL COMPONENTS

Component Number/Type		Area (ft^2)	Length (ft)	Volume (ft^3)	Inclination (Degrees)	Elevation Change (ft)	Surface Roughness	D_{Hyd} (ft)	Flags
182	BRANCH	80.26	4.53	363.60	90.0	4.53	0.00	10.11	100
410	BRANCH	4.58	5.20	23.82	0.0	0.00	1.5E-4	2.42	000
414-1	PIPE	4.59	8.88	40.77	0.0	0.00	1.5E-4	2.42	000
414-2	PIPE	4.59	7.59	34.84	0.0	0.00	1.5E-4	2.42	000
422	BRANCH	21.17	7.91	167.47	90.0	7.91	1.5E-4	5.19	000
424-1	PIPE	10.46	9.06	94.77	90.0	9.06	5.0E-6	0.0553	000
424-2	PIPE	10.46	7.25	75.84	90.0	7.25	5.0E-6	0.0553	000
424-3	PIPE	10.46	7.25	75.84	90.0	7.25	5.0E-6	0.0553	000
424-4	PIPE	10.46	4.44	46.44	90.0	4.44	5.0E-6	0.0553	000
424-5	PIPE	10.46	4.44	46.44	-90.0	-4.44	5.0E-6	0.0553	000
424-6	PIPE	10.46	7.25	75.84	-90.0	-7.25	5.0E-6	0.0553	000
424-7	PIPE	10.46	7.25	75.84	-90.0	-7.25	5.0E-6	0.0553	000
424-8	PIPE	10.46	9.06	94.77	-90.0	-9.06	5.0E-6	0.0553	000
426	BRANCH	21.17	7.91	167.47	-90.0	-7.91	1.5E-4	5.19	000
460-1	PIPE	5.24	7.67	40.19	-90.0	-5.81	1.5E-4	2.58	000
460-2	PIPE	5.24	7.07	37.05	-45.0	-4.50	1.5E-4	2.58	000
460-3	PIPE	5.24	3.52	18.44	0.0	0.00	1.5E-4	2.58	000
460-4	PIPE	5.24	7.07	37.05	45.0	4.50	1.5E-4	2.58	000
475	PUMP	10.68	7.36	78.60	90.0	5.81			00

Table 2.2 (Cont'd)

SUMMARY OF CPSES-1 ANF-RELAP SYSTEM MODEL COMPONENTS

Component Number/Type		Area (ft^2)	Length (ft)	Volume (ft^3)	Inclination (Degrees)	Elevation Change (ft)	Surface Roughness	D_{Hyd} (ft)	Flags
480	BRANCH	4.12	7.14	29.42	0.0	0.00	1.5E-4	2.29	000
490	BRANCH	4.12	15.76	64.96	0.0	0.00	1.5E-4	2.29	000
495	BRANCH	4.12	2.20	9.064	0.0	0.00	1.5E-4	2.29	000
502	TMDPVOL	100.00	10.00	1000.00	0.0	0.00	0.00	0.00	100
504	TMDPVOL	100.00	10.00	1000.00	0.0	0.00	0.00	0.00	100
510-1	PIPE	194.84	7.85	1529.53	-90.0	-7.85	1.5E-4	15.72	100
510-2	PIPE	5.67	7.25	41.11	-90.0	-7.25	1.5E-4	0.34	100
510-3	PIPE	5.67	7.25	41.11	-90.0	-7.25	1.5E-4	0.34	100
510-4	PIPE	5.67	9.06	51.37	-90.0	-9.06	1.5E-4	0.34	100
540-1	PIPE	54.32	9.06	492.18	90.0	9.06	1.5E-4	0.0972	000
540-2	PIPE	55.48	7.25	402.21	90.0	7.25	1.5E-4	0.0972	000
540-3	PIPE	55.48	7.25	402.21	90.0	7.25	1.5E-4	0.0972	000
540-4	PIPE	51.99	4.44	230.84	90.0	4.44	1.5E-4	0.0972	000
540-5	PIPE	87.33	3.41	297.79	90.0	3.41	1.5E-4	10.54	000
560	SEPARATR	47.77	23.74	1133.99	90.0	23.74	0.00	7.80	100
570	SNGLVOL	127.86	9.73	1244.05	90.0	9.73	1.5E-4	12.76	100

Table 2.2 (Cont'd)

SUMMARY OF CPSES-1 ANF-RELAP SYSTEM MODEL COMPONENTS

Component Number/Type		Area (ft^2)	Length (ft)	Volume (ft^3)	Inclination (Degrees)	Elevation Change (ft)	Surface Roughness	D_{Hyd} (ft)	Flags
580	TMDPVOL	100.00	10.00	1000.00	0.0	0.00	0.00	0.00	100
590	TMDPVOL	100.00	10.00	1000.00	0.0	0.00	0.00	0.00	100
610	BRANCH	0.683	67.49	46.10	90.0	27.89	0.00	0.683	000
620-1	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	000
620-2	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	000
620-3	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	000
620-4	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	000
620-5	PIPE	36.58	9.704	354.97	90.0	9.704	0.00	6.82	000
620-6	PIPE	36.58	9.704	354.97	90.0	9.704	0.00	6.82	000
720-1	PIPE	22.93	1.50	34.390	90.0	1.50	0.00	5.40	000
720-2	PIPE	80.46	16.35	1315.61	90.0	16.35	0.00	5.40	000
730	SNGLVOL	0.418	81.12	33.91	-90.0	-10.17	0.00	0.73	100
745	TMDPVOL	1.00	10.00	10.00	90.0	10.00	0.00	0.00	01
750	TMDPVOL	1.00	10.00	10.00	90.0	10.00	0.00	0.00	01
810	TMDPVOL	100.00	1.00	100.00	0.0	0.00	0.00	0.00	100

TABLE 2.3

DENSITY REACTIVITY TABLE (β=0.0044)

DENSITY (1bm/ft³)	REACTIVITY ($)
0.62	-185.331
6.24	-76.88
12.49	-34.001
18.73	-8.68
24.97	0.619
31.21	1.219
37.46	0.469
42.10	0.00
43.70	-0.161
46.14	-0.661
49.94	-1.461
62.43	-3.751

TABLE 2.4

DOPPLER REACTIVITY TABLE (β=0.0044)

TEMPERATURE (°F)	REACTIVITY ($)
400.0	3.481
650.0	3.481
800.0	2.841
1000.0	2.051
1200.0	1.311
1400.0	0.612
1450.0	0.441
1585.0	0.000
1600.0	-0.049
1800.0	-0.679
2000.0	-1.289
2200.0	-1.879
2400.0	-2.449
2600.0	-3.009
2800.0	-3.549
3000.0	-4.079

TABLE 2.5

SCRAM REACTIVITY TABLE (β=0.0044)

TIME (SECONDS)	REACTIVITY ($)
0.00	0.00
0.48	0.00
0.96	-0.053
1.44	-0.133
1.92	-0.400
2.16	-0.800
2.40	-1.813
2.64	-3.413
2.88	-4.800
3.12	-5.120
3.36	-5.227
3.60	-5.333
1.E6	-5.333

TABLE 2.6

ECCS FLOW FOR EACH LOOP VS. PRESSURE

RCS PRESSURE (psig)	CCP (1bm/sec)	HHSI (1bm/sec)
0	11.27	18.42
100	10.96	17.75
120	10.89	17.61
200	10.64	17.05
300	10.33	16.32
400	10.00	15.57
500	9.68	14.75
600	9.34	13.89
700	9.00	12.97
800	8.65	12.02
900	8.23	10.98
1000	7.90	9.68
1200	7.11	6.89
1300	6.71	5.05
1400	6.30	1.98
1500	5.87	0.0
1600	5.34	0.0
1800	4.02	0.0
2000	2.26	0.0
2100	1.25	0.0

TABLE 2.7

TRIPS AND DELAYS

ACTION	TRIPS	DELAYS (Sec)
o Lo-Przr Pressure signal	RCS @ 1860 psia	
o Reactor trip	Lo-Przr signal	2.0
o "S" signal	PRZR @ 1715 psia	
o Reactor Coolant Pumps trip	Reactor Trip	
o Main Steam Isolated	Reactor Trip	2.5
o Main Feedwater Isolated	"S" signal	7.0
o Auxiliary Feedwater Injects	"S" signal	200.0
o SI Actuation Signal	"S" signal	2.0
o Charging Pump Injects	"S" signal	17.0
o HPSI Pump Injects	"S" signal	22.0
o Accumulators Inject	RCS @ 603 psia	

TABLE 2.8

FUEL ASSEMBLY/ROD DATA

PARAMETER	VALUE
o Outer Diameter of Fuel Rod	0.360 in
o Active Fuel Height	144.0 in
o No. of Fuel Assemblies	193
o No. of Fuel Rods/Assy	264
o No. of Guide Thimbles/Assy	24
o No. of Instr. Tubes/Assy	1
o Cladding Thickness	0.025 in
o Diametral Gap	0.0065 in

TABLE 2.9

STEAM GENERATOR SAFETY VALVES FLOW RATES

Secondary Pressure (psia)	SV Flow Rate(1bm/sec)
0.0	0.0
1200.0	0.0
1236.0	0.0
1236.1	248.1
1246.3	248.1
1246.4	498.3
1256.6	498.3
1256.7	750.5
1266.9	750.5
1267.0	1004.8
1287.5	1004.8
1287.6	1263.2
2000.0	1263.2

Figure 2.1 Schematic of TU Electric's Small Break Model

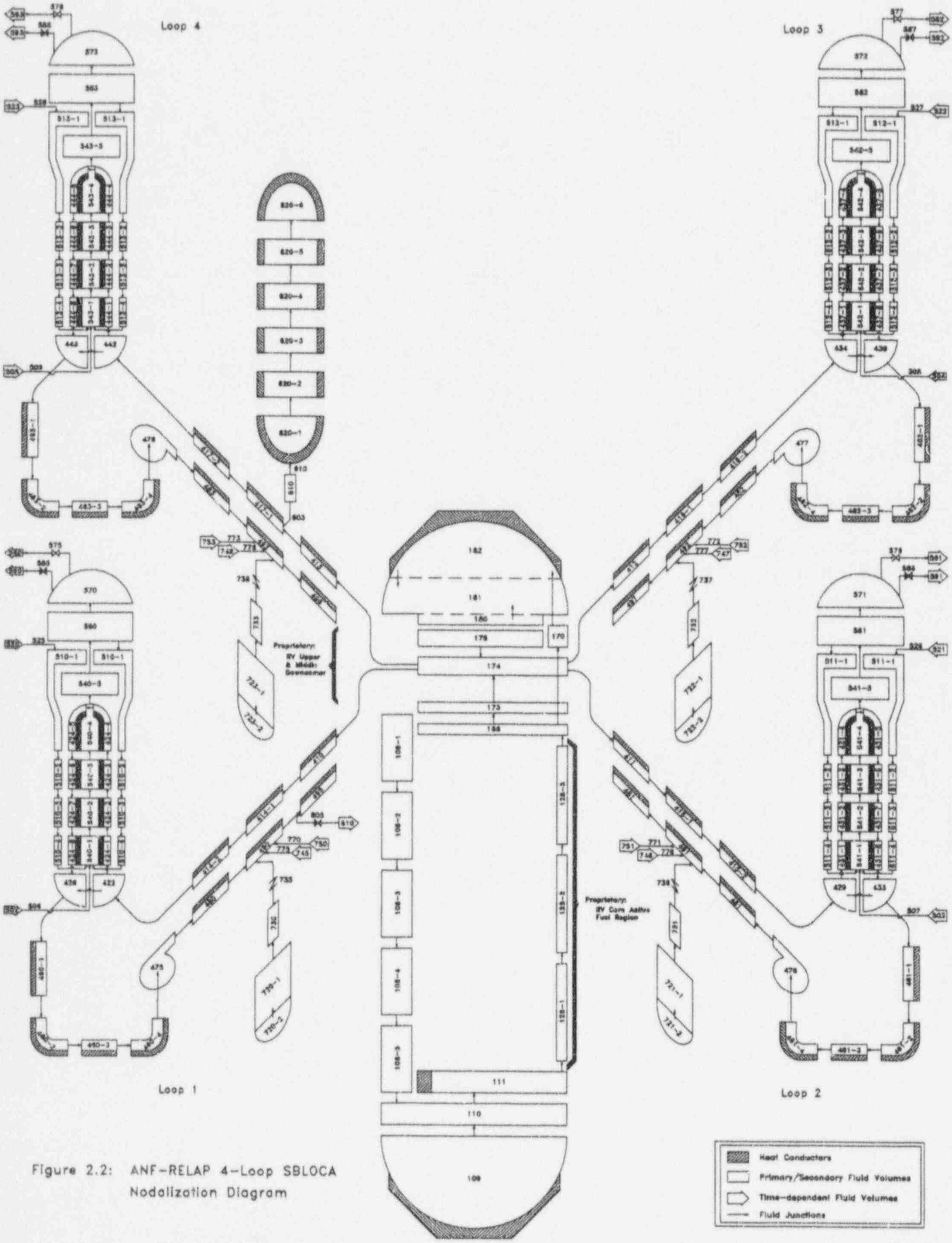

Figure 2.2: ANF-RELAP 4-Loop SBLOCA Nodalization Diagram

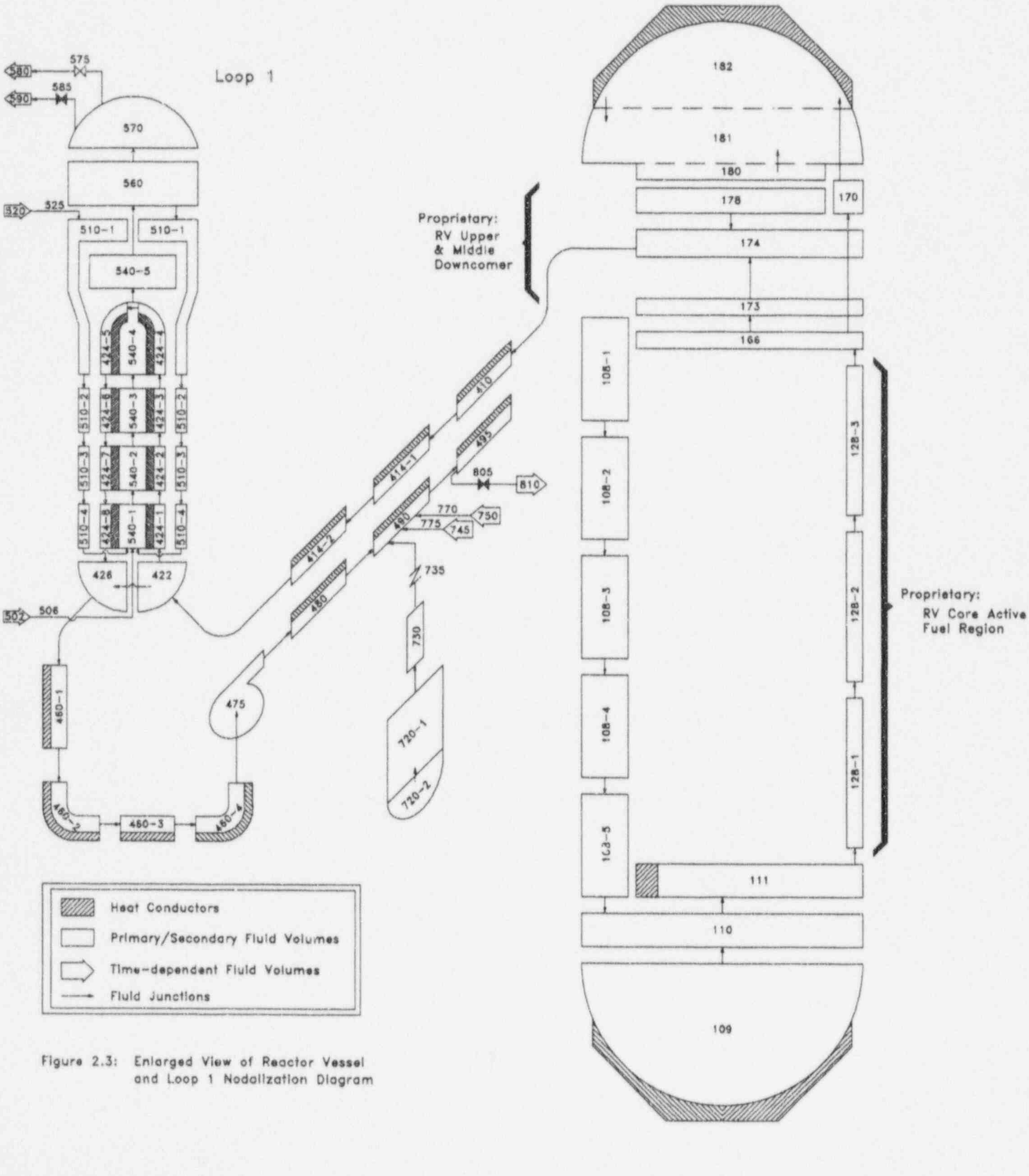

Figure 2.3: Enlarged View of Reactor Vessel and Loop 1 Nodalization Diagram

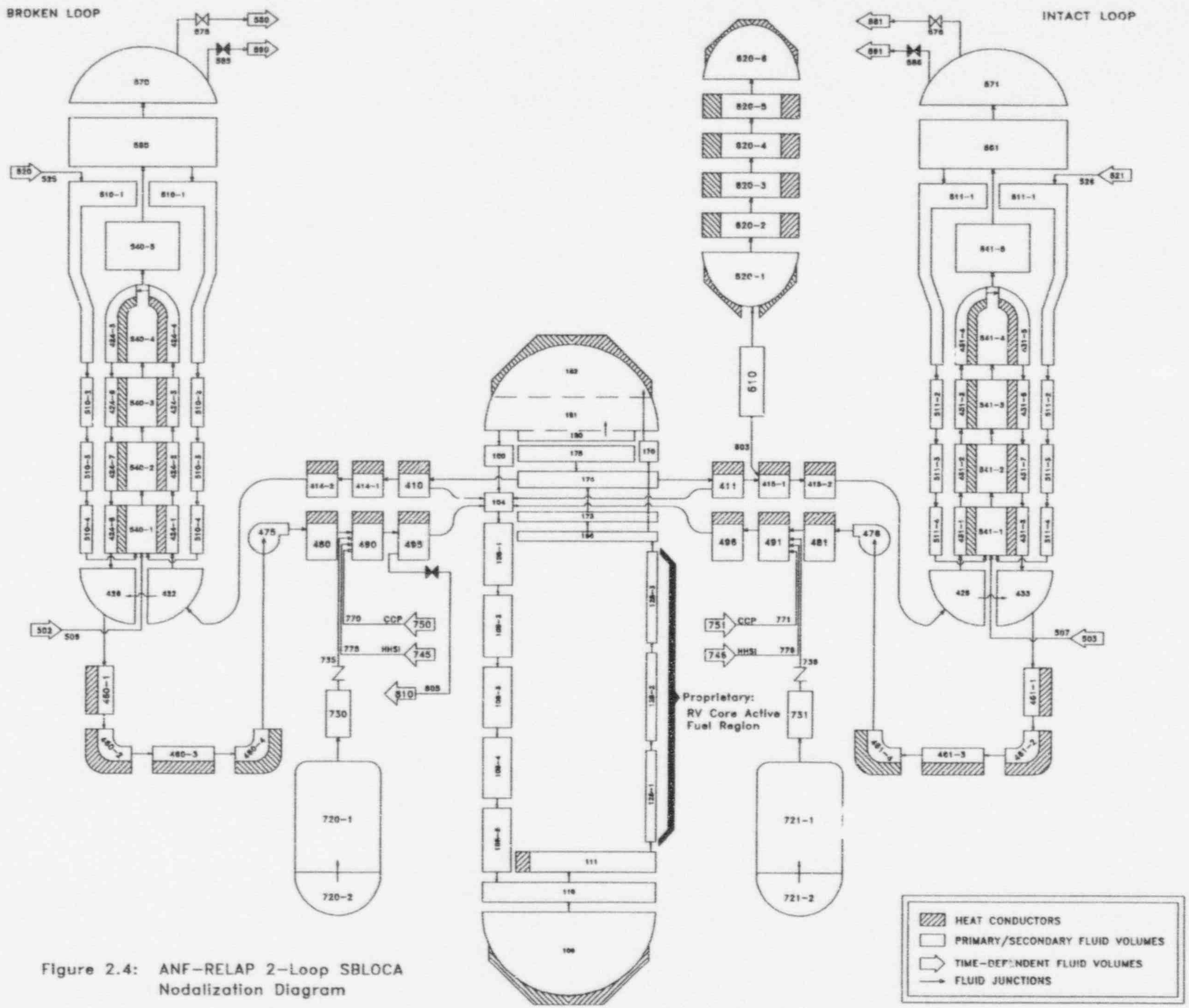

Figure 2.4: ANF-RELAP 2-Loop SBLOCA Nodalization Diagram

Figure 2.5: TOODEE2 Nodalization Diagram

CHAPTER 3

BASE CASE ANALYSIS AND SENSITIVITY STUDIES

Small break loss-of-coolant accident analyses frequently require the investigation of the impact of variations in several method- and plant-specific issues on the LOCA consequences.

Method-specific issues are suggested throughout 10 CFR 50.46, Appendix K thereto, and in NUREG-0737 II.K.3.30, and are addressed in Reference 2.1. The present work constitutes TU Electric's application of SPC's approved Evaluation Methodology (EM), using method-specific parameters as prescribed by the method developers (Reference 2.8). Hence, the effect of variations in method-specific parameters within the bounds of methodology recommendations has already been ascertained in Reference 2.1 and sensitivity studies for these variables need not be repeated here.

There are, nevertheless, three exceptions: (1) a [

] is mandated in Reference 1.1 and, (2) a time step study conducted even though the threshold for this requirement per Reference 1.1, was not reached. In addition, (3) because CPSES has no significant loop asymmetries, an ANF-RELAP model using the more customary representation, where the intact loops are lumped leading to a 2-loop representation, is demonstrated, in another sensitivity study, to yield essentially identical results to the explicit 4-loop model. It is TU Electric's intention to utilize this 2-loop model in future analyses while CPSES continues to show no significant loop asymmetries.

The plant-specific issues which warrant investigation are given in the following passages from 10 CFR 50.46, Appendix K thereto and NUREG-0611, along with the approach taken in addressing each one.

10 CFR 50.46 (a)(1)(i), requires that "a number of postulated loss-of-coolant accidents of different sizes, locations and other properties" be calculated in sufficient amount "to provide assurances that the most severe postulated loss-of-coolant accidents are calculated." In compliance with this requirement, a break spectrum study has been conducted.

Although higher peak clad temperatures (PCT) are usually associated with beginning of life (BOL) fuel because of the higher stored energy, a fuel burnup study is also conducted. This is done in order to confirm that the end of cycle (EOC) pin pressures, which are higher than those encountered early in life and which foster a higher driving force for rod burst, do indeed result in lower PCT for the fuel under consideration.

10 CFR 50, Appendix K, Part I, A, (1) states: "A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime shall be studied and the one selected should be that which results in the most severe

calculated consequences for the spectrum of postulated breaks and single failures analyzed."

[

]

10 CFR 50, Appendix K, Part I, D, (1) states: "an analysis of possible failure modes of ECCS equipment and their effects on ECCS performance must be made. In carrying out the accident evaluation, the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place." The limiting single failure for the small break loss-of-coolant accident analyses in the CPSES-1 & 2 FSAR has been determined by the NSSS vendor (Reference 3.1). It is the loss of one ECCS injection train. Unless a common cause is established, the loss of one ECCS injection train involves multiple failures of ECCS equipment and

therefore is not a single failure. The required common cause is the loss of power to the train. In order to arrive at this condition consistently, it must also be assumed that both the preferred 345 KV and the alternate 148 KV offsite power sources are lost and that one emergency diesel generator fails to start. Hence, the most damaging single failure of ECCS equipment postulated for the present study is the failure of an emergency diesel generator to start. Offsite power (which is not ECCS equipment) unavailability is postulated in order to make the single failure meaningfull, i.e. the diesel generator is not needed if either the preferred 345 KV or the alternate 148 KV offsite power sources are available. Thus, one motor driven auxiliary feedwater pump, one high head centrifugal charging pump, one intermediate head safety injection pump and one low head residual heat removal (RHR) pump (which is not challenged in these analyses) as well as all four accumulators are available to mitigate the accident and are credited in all the calculations.

One additional conservatism is incorporated into all of the calculations in this work. That conservatism is that five percent of the steam generator tubes are assumed plugged. This assumption is made to support the potential need for operation under such circumstances and is a conservative assumption when fewer tubes are actually obstructed.

3.1 BASE CASE ANALYSIS

This section presents licensing analysis results for a 3.0 inch diameter break in the discharge line of the Reactor Coolant Pump. The axial power shape used for this base case is that determined as described in Section 3.0 as most limiting and is shown in Figure

3.1. The fuel rod exposure which maximizes stored energy is calculated by RODEX2 and occurs at 605 hours for the [ ] and 1214 hours for the [ ]. Fuel parameters used in this base case are consistent with this exposure.

The accident assumptions are summarized in Table 3.1 and the initial conditions are summarized in Table 3.2. Key fuel rod parameters are summarized in Table 3.3. Table 3.4 summarizes the timing of significant events for this base case.

[

]. When the neutronics models are activated at the start of the transient, there is a small reactivity imbalance which causes the power to rise. This effect is not significant because it is a slight linear increase (0% to 5%) over a short period of time (25 to 30 seconds). Furthermore, the effect is in the conservative direction and is therefore considered acceptable. Following this brief ramp, the power is soon seen (Figure 3.2) to drop off rapidly, at the time of reactor trip, due to the negative reactivity associated with control rod insertion. Reactor trip is activated by the pressurizer low pressure signal. After that, reactor power tapers off according to decay heat.

Figure 3.3 shows the primary and the secondary pressures and is used as a road map in the following discussion of system performance during this accident. The four accident periods (marked I through IV) in this figure have the following characteristics:

Period I - Depressurization:

The accident period marked I in Figure 3.3 corresponds to the early rapid depressurization which follows break opening. From the secondary side standpoint, period I includes: (1) the early pressure rise due to steam production in the steam generators while the main steam lines are isolated and the steam dump and bypass system is assumed to be inoperable and (2) part of the period where the steam generators are discharging through the safety valves.

Period II - Voiding:

Period I ends and period II begins when a substantial production of steam begins in the core and slows down the depressurization rate. This substantial steam production begins when the bottom of the core starts to boil. This indicates that the whole core is boiling. Thus, the onset of period II occurs when the lowest core nodes begin to develop a significant void fraction. This occurs at the same time, around 210 seconds, for [

] At this time then, the entire core is boiling, resulting in a large production of steam. The effect of this steam production is to reduce the net depressurization rate of the primary system. That in turn leads to the nearly flat primary system pressure trace, which characterizes period II, as seen in Figure 3.3. During period II, water is held up in the upper plenum (Figure 3.7) by the steam generated in the core.

[

] Near the end of period II, as steam production decreases, because less water is available, due to liquid boil off in the core, the broken loop seal clears (Figure 3.10). This allows the depressurization rate to increase, by clearing a vent path from the upper plenum to the break. The loop seal clearing also temporarily disrupts the [ ]: (1) flash either due to the depressurization associated with the clearing or by flowing back into the core and flashing there and/or, (2) to exit via break. The [ ], still in period II, but ends due to dry-out, just prior to the onset of period III. There is also an intermediate heat up and quenching of the clad (Figure 3.9), driven by redistribution of fluid in the core, induced by the loop seal clearing.

Period II from the secondary side point of view has two distinct behaviors. In the first part of period II all secondary pressures remain stable near the safety valves' set points. This is because in period I the steam generators' safety valves have opened due to the steam dump and bypass system unavailability, in order to discharge the steam produced. The atmospheric relief valves (ARVs) are not credited. The early part of period II continues this behavior. In the second part of period II, steam generator pressures in two loops begin to drop following the primary. Since this is also the secondary's behavior in period III, it is discussed in the next section.

Period III - Heatup:

The end of period II and beginning of period III starts with the end of significant steam production in the core caused by shortage of liquid, i.e. the onset of dryout. The end of period II and beginning of period III can be determined from the time at which the core collapsed level reaches the mid core height of 6 ft, indicating the top part of the core has dried out. This can be seen in Figure 3.8. Another indicator is when the top of the core void fractions jump to large values e.g. 0.9, also indicating dry out conditions there, as shown in Figures 3.4, 3.5 and 3.6. Thus, period III or the heat up period begins just before the hot rods enter into critical heat flux (CHF, Figure 3.9). The dropping of the collapsed core level to mid height (6 ft, Figure 3.8) and the rate at which it is dropping are indications that the core is drying out quickly and that steam production has become very low.

Period III is characterized by an increased depressurization rate from the compounded effects of: (1) the previously cleared loop seal (Figure 3.10) and, (2) the lack of steam generation which had been compensating for the energy discharge through the break and keeping the pressure fairly constant in period II.

From the secondary side point of view, period III (and the second part of period II), is characterized by two depressurization rates: one, almost non-existant for the broken loop 1 (cleared loop seal) and the loop 4 with the pressurizer and, another, following the primary, for the other two loops (2 & 3). Loops 2 and 3 depressurize because they receive auxiliary feedwater. Loops 1 and 4 do not receive auxiliary feedwater due to the failure of 1 motor driven pump resulting from the single failure of one diesel generator. This is why loops 1 and 4 do not depressurize, while loops 2 and 3 do, as shown in Figure 3.3.

It is during period III that the fuel experiences its temperature excursion as shown in Figure 3.9. The clad temperatures of Figure 3.9 start to rise right at the beginning of period III, because that is by definition when these axial locations dry out.

Period IV - Recovery:

Period III ends when the system pressure reaches the accumulator injection pressure. At that time, shown in Figure 3.11, the injection of accumulator water marks the onset of period IV. Accumulator injection causes the core collapsed water level to rise (Figure 3.8) and clad temperatures to begin turning around. Figure 3.9 shows the clad temperature histories one node above, one below and at the PCT [ ] location as calculated by ANF-RELAP. The rods are quenched from the bottom up with [ ]. Steam produced in the rod quench process changes the primary system depressurization rate, which again becomes flat, as shown in Figure 3.3. Finally, Figures 3.12 and 3.13 show break flow and pumped injection flow, respectively. These show that pumped injection flow overcomes break flow in the middle of period IV, indicating stable recovery is underway.

3.2 SENSITIVITY STUDIES

3.2.1 BREAK SPECTRUM

The most limiting break location has been determined in previous studies for this (Reference 3.1) and other similar plants (Reference 3.2) to be in the cold leg at the reactor coolant pump discharge. Therefore, this cold leg break location remains most limiting for the present evaluation and a worst break location search need not be repeated. This most limiting break location is the one considered in all cases discussed throughout this work.

According to the approved EXEM PWR Small Break Model, the break size is the first sensitivity issue addressed. The rationale for addressing break size first is that system thermal-hydraulic behavior is largely affected by break size and less dependent on other issues. Consequently, the break size is a first order effect, while the others are second order.

The break spectrum study is conducted using the [

] It is the same power shape used for the base case and the discussion on how it is obtained has been given in Section 3.0.

Three break sizes are analyzed in detail, namely: 3 inch (base case), 4 inch, and 2 inch. Larger sizes (6 inch and 8 inch) were found to be less limiting in preliminary calculations and therefore are not discussed in this document.

The accident assumptions for this and other studies are summarized in Table 3.1 and the initial conditions are summarized in Table 3.2. Key fuel rod parameters are summarized in Table 3.3. The sequence of events for the break spectrum study is summarized in Table 3.5.

The result of this study is that the most limiting break is the 3 inch break located in the reactor coolant pump discharge. The 4 inch and the 2 inch breaks result in lower peak clad temperatures than the base case. The 2 inch break shows no significant clad heatup. The other sensitivity studies use the limiting 3 inch break.

3 inch Break

This is the base case calculation described in Section 3.1. The ANF-RELAP PCT is calculated to be 1705.1°F in [ ]above the bottom of the core. The clad temperature history as calculated by the TOODEE2 code at the node where the PCT occurs is shown in Figure 3.14. The TOODEE2 PCT is 1779.8°F in [ ] The difference in elevations is due to the rupture of the TOODEE2 node corresponding to the highest powered node at the [ ] as also shown in Figure 3.14. TOODEE2 initial fuel conditions for this run correspond to beginning of cycle (BOL).

4 inch Break

The calculated system behavior for this case is similar to the base case (Section 3.1), although event durations are somewhat shorter due to the larger break size. The PCT is also lower. The ANF-RELAP PCT is calculated to be 1479.7°F also in [ ] above the bottom of the core. The clad temperature history as calculated by the TOODEE2 code at the node where the PCT occurs is shown in Figure 3.27. The TOODEE2 PCT for the 4 inch case is 1571.7°F in [ ] above the top of the core. Since the highest powered node did not rupture, the ANF-RELAP and the TOODEE2 PCT occur at the same elevation. TOODEE2 initial fuel conditions for this run also correspond to beginning of cycle (BOL).

Figure 3.15 shows the power behavior subject to the same mechanisms of the base case calculation.

Figure 3.16 shows the primary and the secondary pressures. The same four accident periods (also marked I through IV in this figure) are used in the following discussion of the 4 inch break.

Period I - Depressurization:

As in the base case the accident period marked I in Figure 3.16 corresponds to the depressurization of the primary system due to the break while the secondary pressure rises to and remains at the safety valves' set point. There are no major distinctions between system behavior during this period between the 4 inch break and the 3 inch base case except that the depressurization rate is higher for the larger break.

Period II - Voiding:

As in the base case, period I ends and period II begins when a substantial production of steam starts in the core and slows down the depressurization rate. This substantial steam production also begins with the formation of void at the bottom core elevation. This indicates the entire core is boiling. It occurs at the same time, around 105 seconds, for [

] The 3 inch base case discussion for this period applies to the 4 inch break as well. In this case two loop seals clear, the broken loop 1 and loop 2 (only loop 1 clears in the 3 inch case), in the middle of the period, most likely as a result of a larger break size. Loop seal clearing, as in the 3 inch case, also leads to an increase in the primary system depressurization rate. The [

], which is temporarily disrupted by the loop seal clearing, and reinstated until the onset of period III (Figure 3.20). As in the base case, in the 4 inch break there is also an intermediate heatup and quenching of the clad, driven by redistribution of fluid in the core, induced by the loop seal clearing (Figure 3.22). Secondary side behavior is similar to the 3 inch case.

Period III - Heatup:

As in the 3 inch discussion, the end of period II and beginning of period III occurs when the core collapsed level drops below the mid core elevation of 6 ft (Figure 3.21). The dropping of this level to about 6 ft means the top half of the core is dry, and steam production has been substantially reduced. The jump in top core elevations' void fractions to high (0.9) values also signals the onset of dryout in this case. The primary system pressure continues to drop significantly as two loop seals remain clear and steam production is low. It is also in period III that the fuel experiences its temperature excursion as shown in Figure 3.26. For the 4 inch break the loop seals are also clear before the beginning of period III. Secondary side behavior is also similar to the 3 inch case.

Period IV - Recovery:

As in the base case, period III ends when the system pressure reaches the accumulator injection pressure. At that time, shown in Figure 3.24, the injection of accumulator water marks the onset of period IV. Accumulator injection causes the core collapsed water level to rise (Figure 3.21) and clad temperatures to begin to turn around. Figure 3.22 shows the clad temperature histories one node above, one below and at the PCT [ ] location as calculated by ANF-RELAP. The rods are quenched from the bottom up with [

] Steam produced in the rod quench process changes the primary system depressurization rate, which again becomes flat, as shown in Figure 3.16. Finally, Figures 3.25 and 3.26 show break flow and pumped injection flow, respectively. Pumped injection flow overcomes break flow also in the middle of period IV, indicating stable recovery is underway.

The same conclusion drawn for the base case applies to the 4 inch calculation. The pumped injection flows (Figure 3.26) cannot keep up with the break flow (Figure 3.25) during periods I , II and III. Still, the accumulator injection pressure is reached well before the clad temperatures are too high and the temperatures are effectively turned around.

2 inch Break

This calculation is somewhat different from the base case calculation. The difference is that there is no period III, i.e., no heatup period. As in the 3 inch base case, the voiding period II is also interrupted when the broken loop seal clears. At that time, the same perturbations observed in the 4 inch and 3 inch cases occur here as well. The core collapsed level dips and recovers, the [ ] is interrupted and reinstated, there is a brief spike in clad temperatures (Figure 3.28) and the primary system pressure begins to drop more quickly. The important difference between this case and the other two is that, the increased depressurization rate associated with loop seal clearing drops the break flow to less than the total pumped ECCS flow rate before any core heatup takes place. Thus, the transient is essentially over after the loop seal clears, in the sense that the possibility of heatup is eliminated. There is never a sustained heatup of the clad for the 2 inch break, only the spike associated with the clearing of the loop seal, which is shown in Figure 3.28. Since there is no sustained heatup for the 2 inch break, a TOODEE2 calculation is unecessary.

Table 3.8 provides a summary of the PCTs for the break spectrum study. Figures 3.30, 3.31 and 3.32 summarize clad temperatures' histories for this study.

3.2.2 [ ]

The [

.] This study is performed for the most limiting break determined in the break spectrum study (3 inch, Section 3.2.1). [

]

The sequence of events for the two sensitivity cases are summarized and compared to the nominal case in Table 3.6. Figure 3.29 overlays the calculated ANF-RELAP clad temperatures for all three cases.

The conclusion, as seen in Figure 3.29, is that there is little difference in clad temperature history associated with these [ ] for the CPSES model. In any case, [

] used in the base case calculation are the most limiting, as indicated in the PCT summary of Table 3.9.

3.2.3 [ ]

[

.]

Nevertheless, preliminary calculations revealed that clad temperature profiles were not converged if the maximum time step was too large. Therefore, it became necessary to find the largest time step at which results were essentially unaffected by further reductions in time step. The objective of this time step study was to find that optimum time step value.

This study was performed for all three breaks in the break spectrum study. The main convergence criterion was a visual inspection of the behavior throughout the transient, of the most sensitive variable: the clad temperature. In addition to this visual criteria, in order to be deemed "converged", a run must also exhibit the same sequence of events of a smaller time step run. For example, if accumulator injection precedes the PCT in the smaller time step, this must also be the case for a larger time step to be acceptable. Thus, similar clad temperature histories are considered necessary but not sufficient conditions. Finally, although not a requirement, a maximum time step that was consistent throughout the break spectrum study was felt to be desirable if reasonably achievable.

Figure 3.30 shows six time step runs (0.05, 0.25, 0.010, 0.005, 0.0025 and 0.00125 seconds) for the base case 3 inch break. The three smallest show nearly identical results and event sequences. Thus it is concluded that a maximum time step of 0.005 seconds is adequate. It should be noted that three larger time steps of: 0.050, 0.025 and 0.010 seconds were also tried. The two largest show discrepancies in the visual comparison to the three smallest. The 0.01 seconds time step appears acceptable but was rejected here because: (a) The PCT occurred prior to accumulator injection and (b) this time step is too large for the 4 inch break described below.

Figure 3.31 shows four time steps (0.010, 0.005, 0.0025 and 0.00125 seconds) for the 4 inch break. The three smallest are extremely close until significantly after the peak clad temperature has been reached. In this case, the 0.010 second time step did not meet the visual convergence criterion, as shown in Figure 3.31.

Figure 3.32 shows two time steps (0.010 and 0.005 seconds) for the 2 inch break. Except around the spike associated with the loop seal clearing, these are identical. Therefore, for the sake of consistency with the 3 and 4 inch cases, and considering there is no sustained heatup for this case, the 0.005 second time step is considered adequate for the 2 inch break also.

In summary, all cases were certainly converged at 0.005 seconds. The base case 3 inch break might have been called converged at 0.010 seconds, except that the PCT occurred prior to accumulator injection in that case. The 4 inch break was clearly not converged at 0.010 seconds. The 2 inch break seems converged at 0.010 seconds. Therefore, the 0.005 second time step utilized is adequate for applications of the CPSES model.

Although converged at 0.005 seconds, the actual numerical value of the PCT can vary somewhat as the time step is reduced further. This can be interpreted as a convergence band. If the band is to the right of (i.e. higher than) the PCT, there could be a concern that the "actual" PCT would be larger. This issue was examined. For the limiting break, this variation is concluded to be between 20^{0}F and 30^{0}F. [

.]

[ ] studies will only be conducted in future applications if:

(1) [ ] apply or,

(2) if breaks larger than 4 inch are analyzed, or

(3) if [ ] are utilized.

3.2.4 TWO LOOP VERSUS FOUR LOOP ANF-RELAP MODEL

All analyses discussed up to this point in Chapter 3 have been conducted using the fully explicit four loop ANF-RELAP model discussed in Section 2.4.1.

However, CPSES has no significant loop asymmetries and the fully explicit four loop model is cumbersome to execute, particularly at the small time steps required.

Therefore, it is reasonable to expect that the industry-wide conventional 2-loop representation, with a broken loop and a lumped intact loop, would yield results substantially identical to the fully explicit four loop model.

In order to test this hypothesis a 2-loop ANF-RELAP model was developed as described in Section 2.4.2. Figure 3.33 compares clad temperature histories as calculated with the 2-loop and 4-loop models, for the 3 inch base case. Both used [ ] There are no significant differences in the transient as calculated with either model. The ANF-RELAP PCT is 1705.1°F for the 4-loop and 1709.7°F for the 2-loop. TOODEE2 PCTs are given in Table 3.11.

Therefore, as a result of this finding, TU Electric intends to utilize the two-loop model in future applications of this methodology.

3.2.5 EXPOSURE STUDY

This exposure study is done by performing an additional TOODEE2 calculation using the same ANF-RELAP boundary conditions as the base case 3 inch run, but for which RODEX2 initial fuel parameters are generated at EOC instead of the BOL conditions used for the base case.

Figure 3.34 compares TOODEE2 BOL and EOC temperature histories for the base case 3 inch break. The highest clad temperature corresponds to the BOL case. The lowest set of curves in Figure 3.34 correspond to the ruptured node. PCTs for this sensitivity study are summarized in Table 3.12.

TABLE 3.1

SUMMARY OF CPSES-1 SMALL BREAK LOCA ANALYSIS

ASSUMPTIONS FOR BASE CASE AND SENSITIVITY STUDIES

1. The initial power is 3479 MWt, which is 2% above the licensed power level of 3411 MWt, to account for calorimetric measurement uncertainty.

2. 5% of the steam generator tubes are plugged.

3. Break in reactor coolant pump discharge occurs at 0.0 s.

4. Reactor trips due to a Lo-Pressurizer pressure signal.

5. Loss of offsite power coincides with reactor trip.

6. The reactor coolant pumps (RCP) are tripped at reactor trip since RCP cannot operate without offsite power after a reactor trip.

7. Steam flow isolation is initiated at the time of reactor trip. The steam dump and bypass system is not credited.

8. Main feedwater isolation is initiated 7 seconds after "S" signal.

9. Failure of one diesel generator to start takes out one high head centrifugal charging pump, one intermediate head safety injection pump, one RHR pump and one motor-driven AFW pump. This is the single failure assumed for compliance with 10 CFR 50, Appendix K, Part D.

10. One high head centrifugal charging pump, one intermediate head safety injection pump inject on demand after the appropriate delays, at conservative flow rates.

11. One of the two motor-driven AFW pumps is credited, but injection is conservatively delayed in order to account for flow travel time.

12. All accumulators inject on demand.

TABLE 3.2

SUMMARY OF INITIAL CONDITIONS FOR CPSES-1

SMALL BREAK LOCA BASE CASE AND SENSITIVITY STUDIES

DESCRIPTION	VALUE
o Core Power	3479 Mwt
o Power Calorimetric Uncertainty Multiplier	1.02
o Power Shape Analyzed	[ ]
o Peak Linear Power [ ] (includes 102% factor)	13.12 KW/ft
o Fraction of heat deposited in fuel	0.974
o Total Peaking Factor, F^T_Q (flat segment of K(z))	2.42
o Total Peaking Factor, F^T_Q [ ]	[ ]
o Accumulator Water Volume per Tank	6119 gals
o Accumulator Cover Gas Pressure	603 psia
o Accumulator Water Temperature	150 °F
o Safety Injection Pumped Flow	Table 2.6
o Refueling Water Storage Tank Temperature	120 °F
o Initial Loop Flow	9661 lbm/sec
o Vessel Inlet Temperature	566 °F
o Vessel Outlet Temperature	626 °F
o Reactor Coolant Pressure	2280 psia
o Pressurizer Water Volume	1116 ft^3
o Steam Pressure	914 psia
o Auxiliary Feedwater Flow to each of SGs 2 & 3	29.6 lb/sec
o Auxiliary Feedwater Flow to each of SGs 1 & 4	0.00 lb/sec
o Steam Generator Tube Plugging Level	5%
o Steam Generator Safety Valves Set Points & Flows	Table 2.9
o Fuel Parameters	Table 3.3

TABLE 3.3

SUMMARY OF FUEL PARAMETERS FOR

BASE CASE SMALL BREAK LOCA ANALYSIS

PARAMETERS	VALUES			
Fuel Rod Geometry Data	Table 2.8			
Time to Maximum Stored Energy Exposure	[			]
Fuel Rod Composition:				
	[			]
[			]	
Average fuel temperature at peak stored energy (°F)	1532	2236	2245	

TABLE 3.4

SEQUENCE OF EVENTS FOR BASE CASE[1] SMALL BREAK LOCA

EVENT	TIME (SECONDS)
1. Break opens (period I begins)	0.0
2. Reactor Trip Signal	25.1
3. RCP tripped	27.1
4. MSIV closed	27.6
5. "S" Signal	34.3
6. MFW isolated	41.3
7. Centrifugal charging pumps inject	51.3
8. Safety injection pumps inject	56.3
9. Entire core boils (period II begins)	~210
10. Auxiliary Feedwater reaches SGs 2 & 3	234.3
11. Broken Loop 1 seal clears	~730
12. Critical Heat Flux at PCT node (period III begins)	~1050
13. Accumulator injection (period IV begins)	1824
14. Peak clad temperature reached	1824
15. Pumped ECCS flow exceeds break flow	~1600
16. Calculation ends	2100.0

1 3 inch break, [ ], maximum ANF-RELAP time step of [ ], 4-loop explicit ANF-RELAP model, beginning of cycle exposure.

TABLE 3.5

SEQUENCE OF EVENTS FOR BREAK SPECTRUM[2] STUDY

EVENT	TIME (SECONDS)		
	3 inch	4 inch	2 inch
1. Break opens (period I begins)	0.0	0.0	0.0
2. Reactor Trip Signal	25.1	12.6	66.4
3. RCP tripped	27.1	14.6	68.4
4. MSIV closed	27.6	15.1	68.9
5. "S" Signal	34.3	21.2	75.9
6. MFW isolated	41.3	28.2	82.9
7. Centrifugal charging pumps inject	51.3	38.2	92.9
8. Safety injection pumps inject	56.3	43.2	97.9
9. Entire core boils (period II begins)	~210	~105	~450
10. AFW reaches SGs 2 & 3	234.3	221.2	276
11. Broken Loop 1 seal clears	~730	~397	~2014
12. Critical Heat Flux at PCT node (period III begins)	~1050	~660	No Heatup
13. Accumulator injection (period IV begins)	1824	889	N/A
14. Peak clad temperature reached	1824	898	No Heatup
15. Pumped ECCS flow exceeds break flow	~1600	~1000	Early
16. Calculation ends	2100.0	~1000	2400.0

---

2 All cases: nominal [ ], maximum ANF-RELAP time step of [ ], 4-loop ANF-RELAP explicit model, beginning of life fuel exposure.

TABLE 3.6

SEQUENCE OF EVENTS FOR [ ] STUDY[3]

EVENT	TIME (SECONDS)		
	[ ]	[ ]	[ ]
1. Break opens (period I begins)	0.0	0.0	0.0
2. Reactor Trip Signal	25.1	25.2	25.0
3. RCP tripped	27.1	27.2	27.0
4. MSIV closed	27.6	27.7	27.5
5. "S" Signal	34.3	34.5	34.3
6. MFW isolated	41.3	41.5	41.3
7. Centrifugal charging pumps inject	51.3	51.5	51.3
8. Safety injection pumps inject	56.3	56.5	56.3
9. Entire core boils (period II begins)	~210	~210	~210
10. Auxiliary Feedwater reaches SGs 2 & 3	234.3	234.5	234.3
11. Broken Loop 1 seal clears	~730	~730	~730
12. Critical Heat Flux at PCT node (period III begins)	~1050	~1050	~1050
13. Accumulator injection (period IV begins)	1824	1810	1802
14. Peak clad temperature reached	1824	1806	1804
15. Pumped ECCS flow exceeds break flow	~1600	~1600	~1600
16. Calculation ends	~2100	~2100	~2100

3 All cases: 3 inch break, maximum ANF-RELAP time step of [ ], 4-loop explicit ANF-RELAP model, beginning of life fuel parameters for ANF-RELAP, TOODEE2 runs not needed.

TABLE 3.7

SEQUENCE OF EVENTS FOR 2-LOOP TO 4-LOOP COMPARISON[4]

EVENT	TIME (SECONDS)	
	4-LOOP MODEL	2-LOOP MODEL
1. Break opens (period I begins)	0.0	0.0
2. Reactor Trip Signal	25.1	25.0
3. RCP tripped	27.1	27.0
4. MSIV closed	27.6	27.5
5. "S" Signal	34.3	34.2
6. MFW isolated	41.3	41.2
7. Centrifugal charging pumps inject	51.3	51.2
8. Safety injection pumps inject	56.3	56.2
9. Entire core boils (period II begins)	~210	~210
10. Auxiliary Feedwater reaches SGs 2 & 3	234.3	234.3
11. Broken Loop 1 seal clears	~730	~712
12. Critical Heat Flux at PCT node (period III begins)	~1050	~1050
13. Accumulator injection (period IV begins)	1824	1780
14. Peak clad temperature reached	1824	1786
15. Pumped ECCS flow exceeds break flow	~1600	~1600
16. Calculation ends	2100.0	2100.0

---

4 All cases: 3 inch break, nominal [ ], maximum ANF-RELAP time step [ ], beginning of life fuel exposure.

TABLE 3.8

PCT SUMMARY FOR BREAK SPECTRUM STUDY[5]

BREAK SIZE (INCHES)	ANF-RELAP PCT (^{0}F)	TOODEE2 PCT (^{0}F)
3.0	1705.1	1779.8
4.0	1479.7	1571.7
2.0	NO HEATUP	N/A

TABLE 3.9

PCT SUMMARY FOR [ ] STUDY[6]

[ ]	[ANF-RELAP PCT (^{0}F)]	[TOODEE2 PCT (^{0}F)]
[ ]	1705.1	1779.8
[ ]	1664.3	N/A
[ ]	1610.6	N/A

5 All cases: [ ], maximum ANF-RELAP time step of [ ], 4-loop ANF-RELAP explicit model, beginning of life fuel exposure.

6 All cases: 3 inch break, maximum ANF-RELAP time step of [ ], 4-loop explicit ANF-RELAP model, beginning of life fuel parameters for ANF-RELAP, TOODEE2 runs not needed.

TABLE 3.10

PCT SUMMARY FOR [ ][7]

3 INCH BREAK (SEE ALSO FIGURE 3.30)		
MAX ANF-RELAP [ ]	**ANF-RELAP PCT (^{0}F)**	**TOODEE2 PCT (^{0}F)**
[ ]	1747.2	1815.2
[ ]	1736.0	1758.9
[ ]	1726.6	1783.4
[ ]	**1705.1**	**1779.8**
[ ]	1722.6	1809.5
[ ]	1732.4	1795.2

4 INCH BREAK (SEE ALSO FIGURE 3.31)		
[ ]	**ANF-RELAP PCT (^{0}F)**	**TOODEE2 PCT (^{0}F)**
[ ]	1625.1	1705.1
[ ]	**1479.7**	**1571.7**
[ ]	1487.7	1579.1
[ ]	1459.6	1553.1

2 INCH BREAK (SEE ALSO FIGURE 3.32)		
[ ]	**ANF-RELAP PCT (^{0}F)**	**TOODEE2 PCT (^{0}F)**
[ ]	NO HEATUP	N/A
[ ]	**NO HEATUP**	**N/A**

---

7 All cases: [ ], 4-loop ANF-RELAP explicit model, beginning of life fuel exposure.

8 Not "converged".

9 Optimum time step.

TABLE 3.11

PCT SUMMARY FOR 2-LOOP MODEL VALIDATION STUDY[10]

ANF-RELAP NODALIZATION	ANF-RELAP PCT (^{0}F)	TOODEE2 PCT (^{0}F)
4-LOOP (FIGURE 2.2)	1705.1	1779.8
2-LOOP (FIGURE 2.4)	1709.7	1792.8

TABLE 3.12

PCT SUMMARY FOR EXPOSURE STUDY[11]

EXPOSURE	ANF-RELAP PCT (^{0}F)	TOODEE2 PCT (^{0}F)
BEGINNING OF LIFE	1705.1	1779.8
END OF CYCLE	1705.1	1745.7

---

10 All cases: 3 inch break, [ ], maximum ANF-RELAP time step [ ], beginning of life fuel exposure. See also Figure 3.33

11 All cases: 3 inch break, [ ], 4-loop ANF-RELAP same run at BOL. See also Figure 3.34

Figure 3.2 Total Core Power

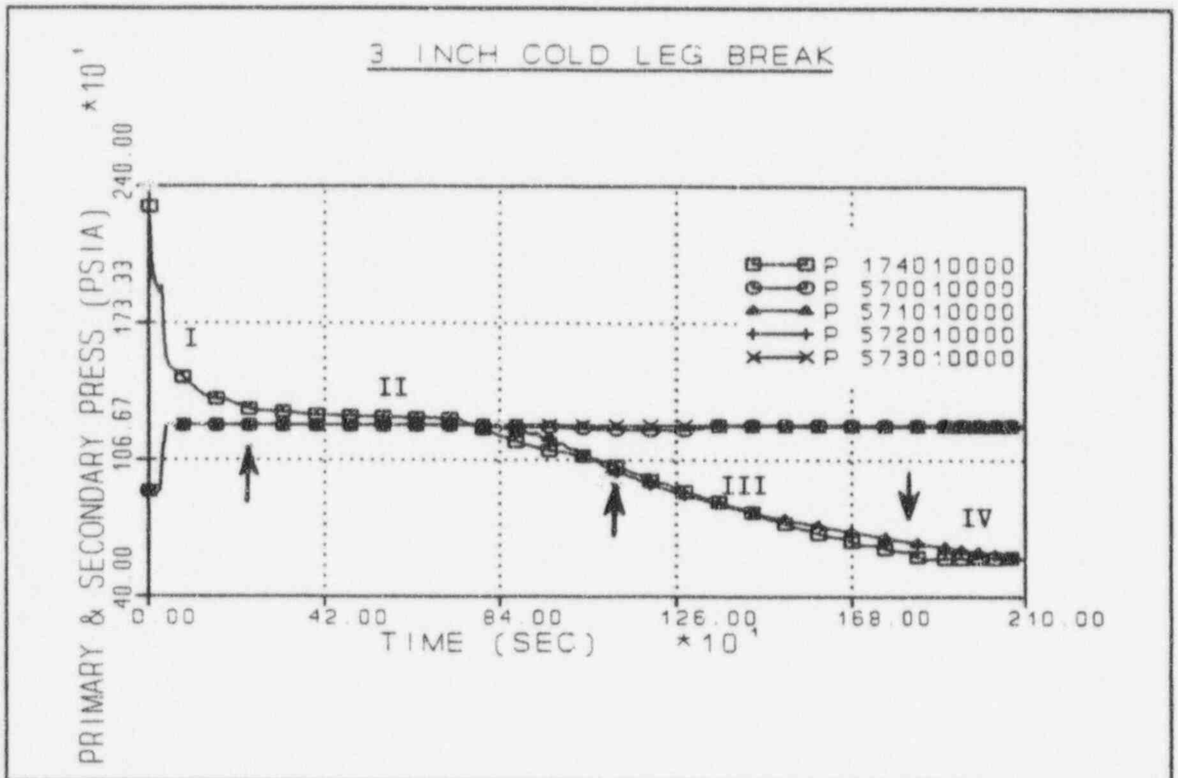

Figure 3.3 Primary & Secondary System Pressures

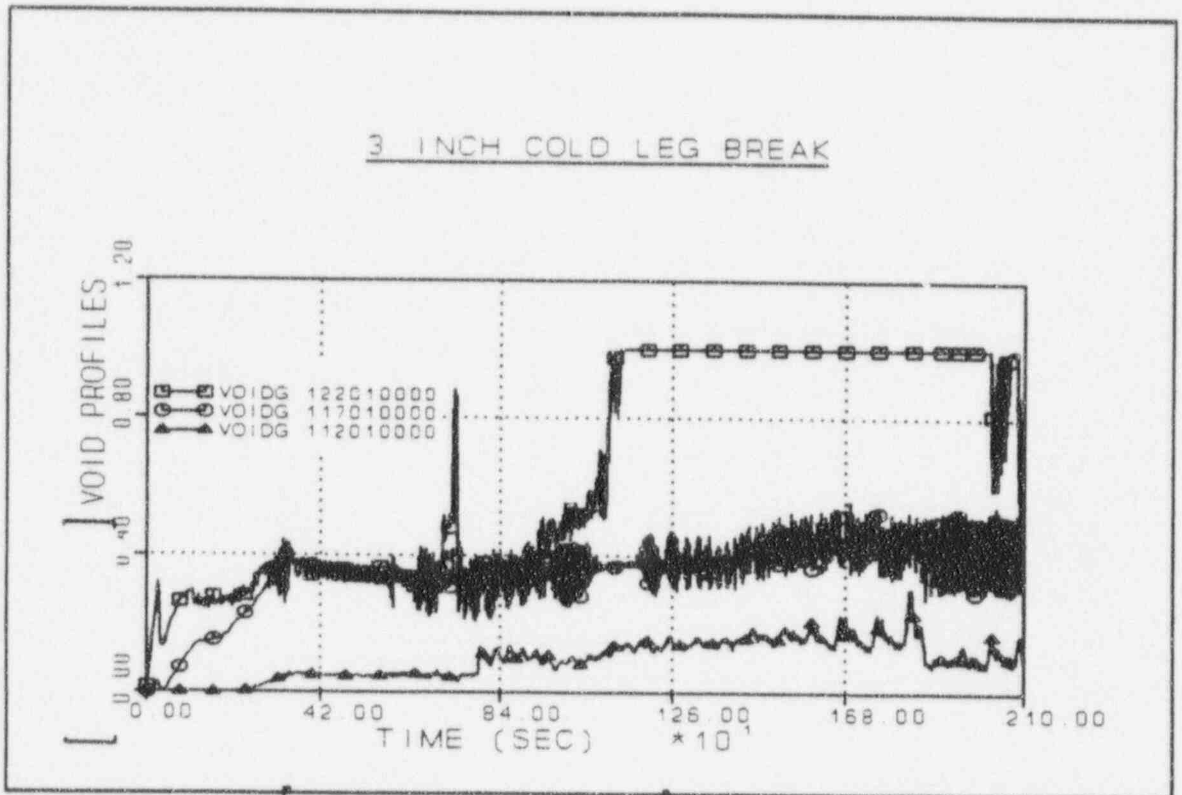

Figure 3.4 [ ] Void Fractions

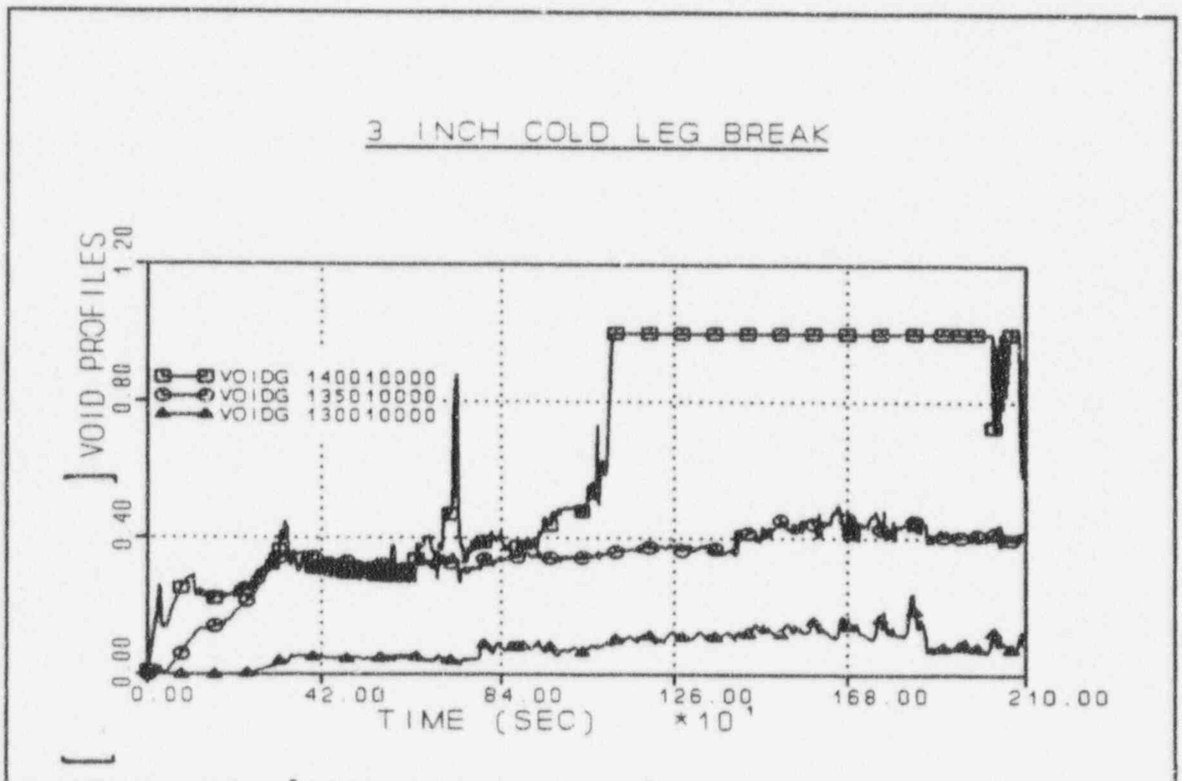

Figure 3.5 [ ] Void Fractions

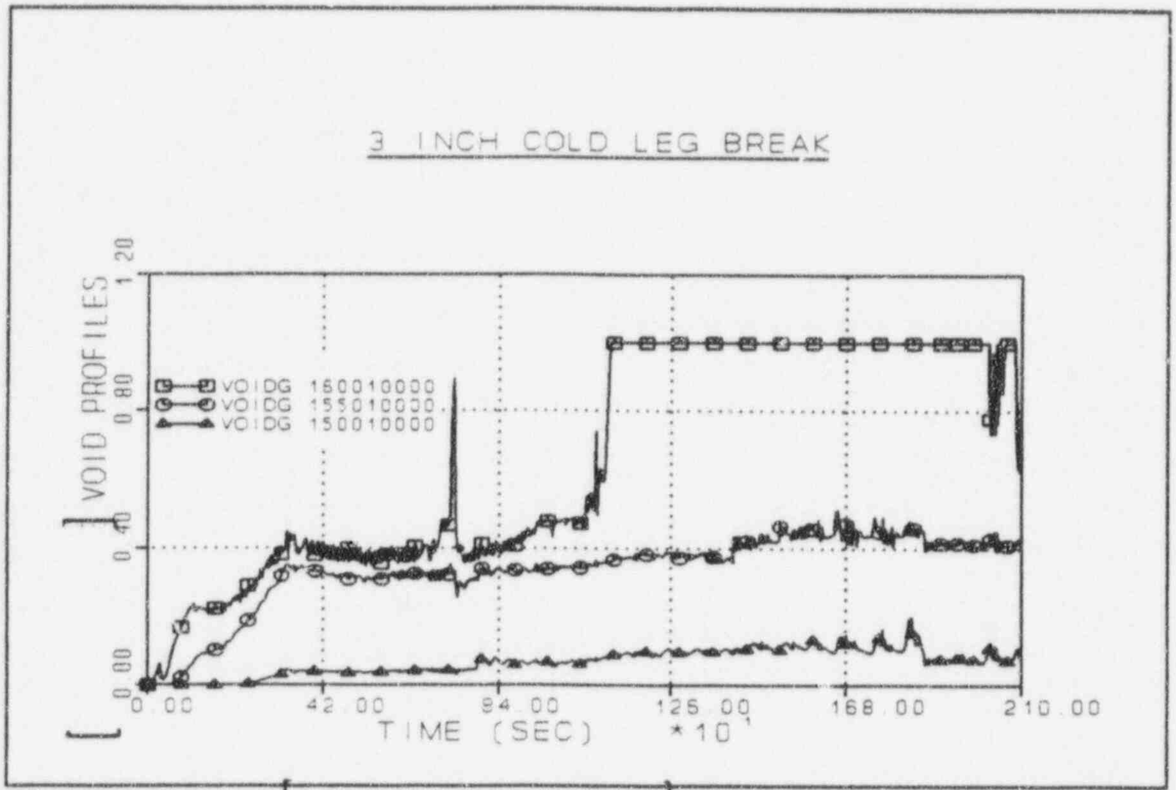

Figure 3.6 [ ] Void Fractions

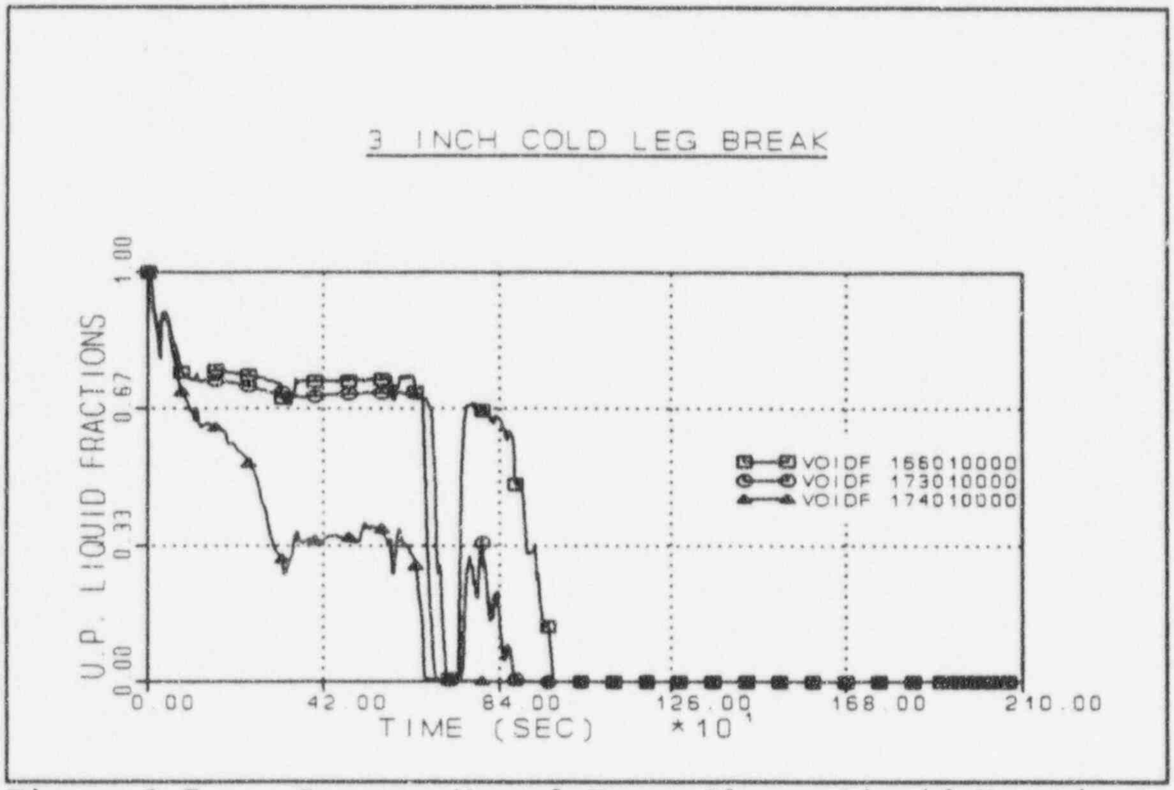

Figure 3.7 Reactor Vessel Upper Plenum Liquid Fractions

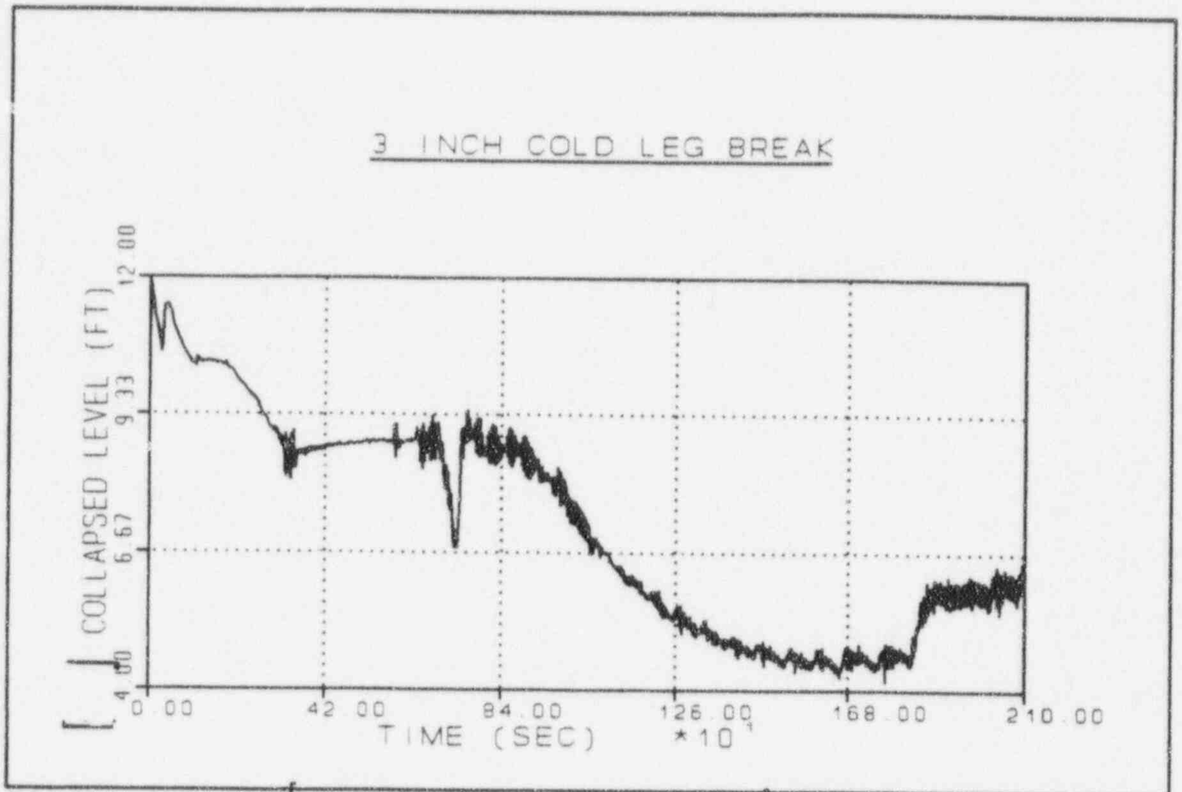

Figure 3.8 [ ] Collapsed Level

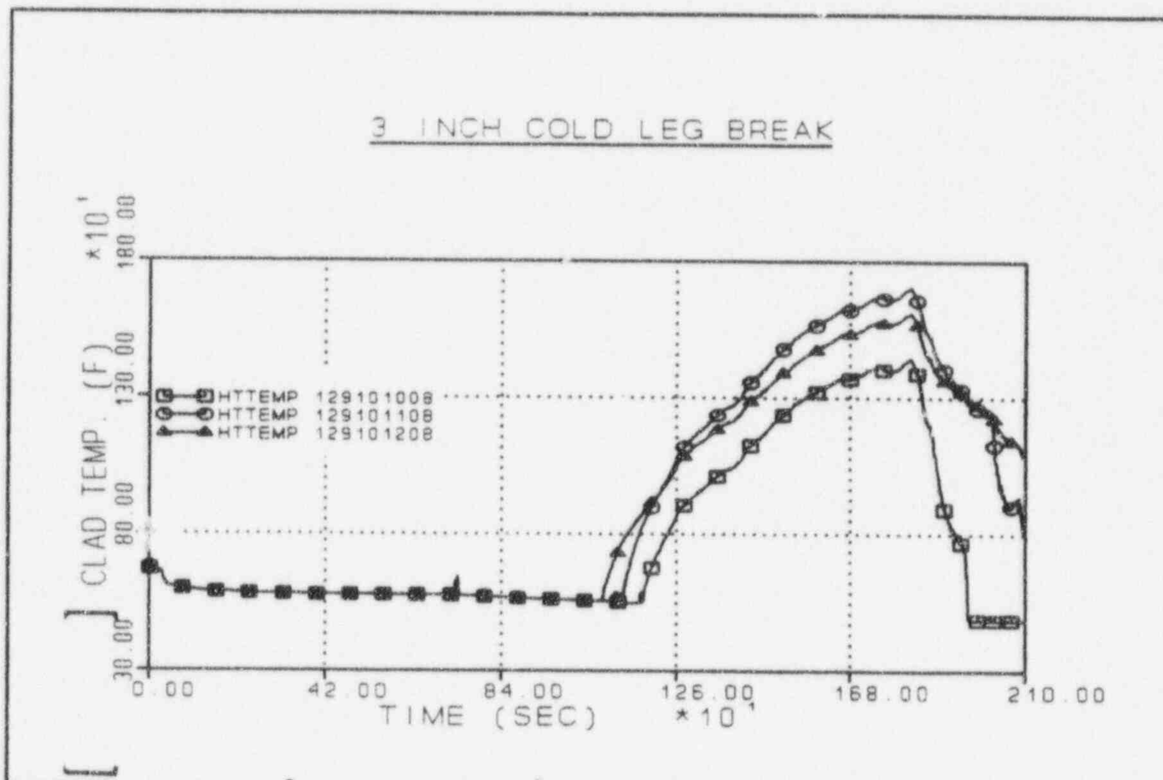

Figure 3.9 [ ] Cladding Temperature Profiles

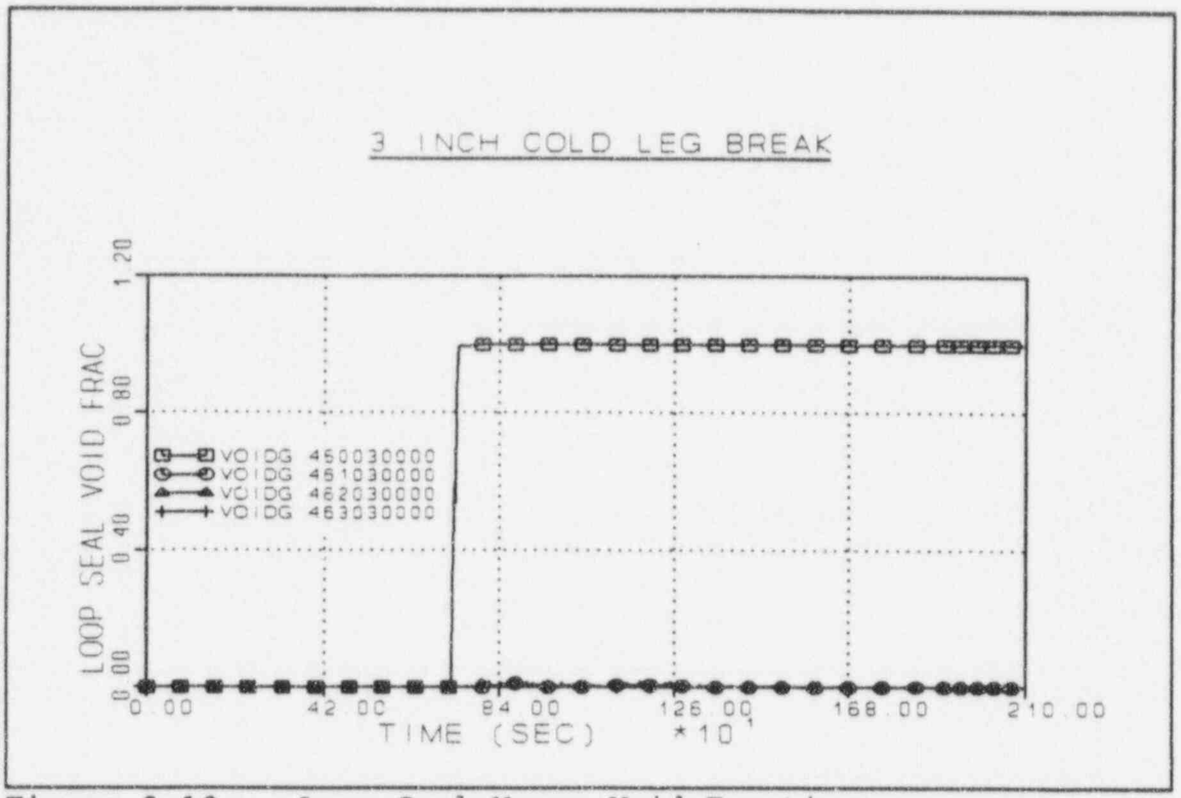

Figure 3.10 Loop Seal Vapor Void Fractions

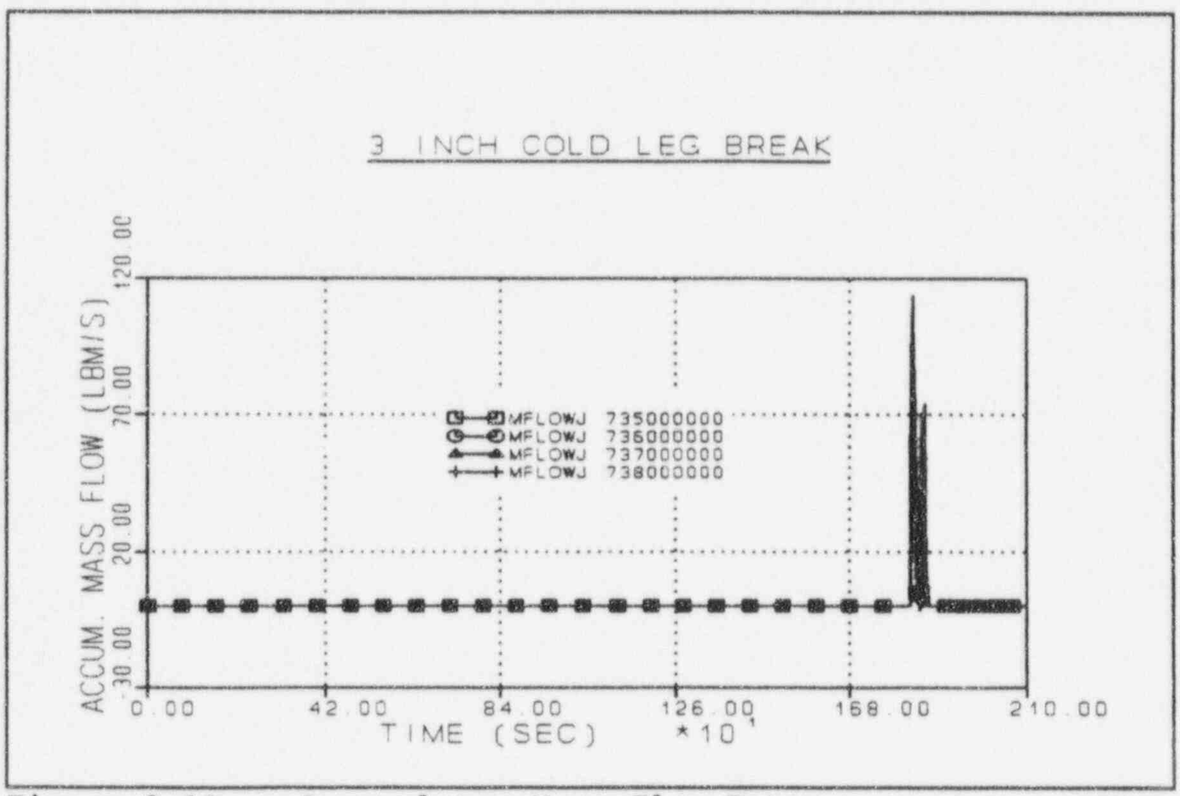

Figure 3.11 Accumulator Mass Flow Rates

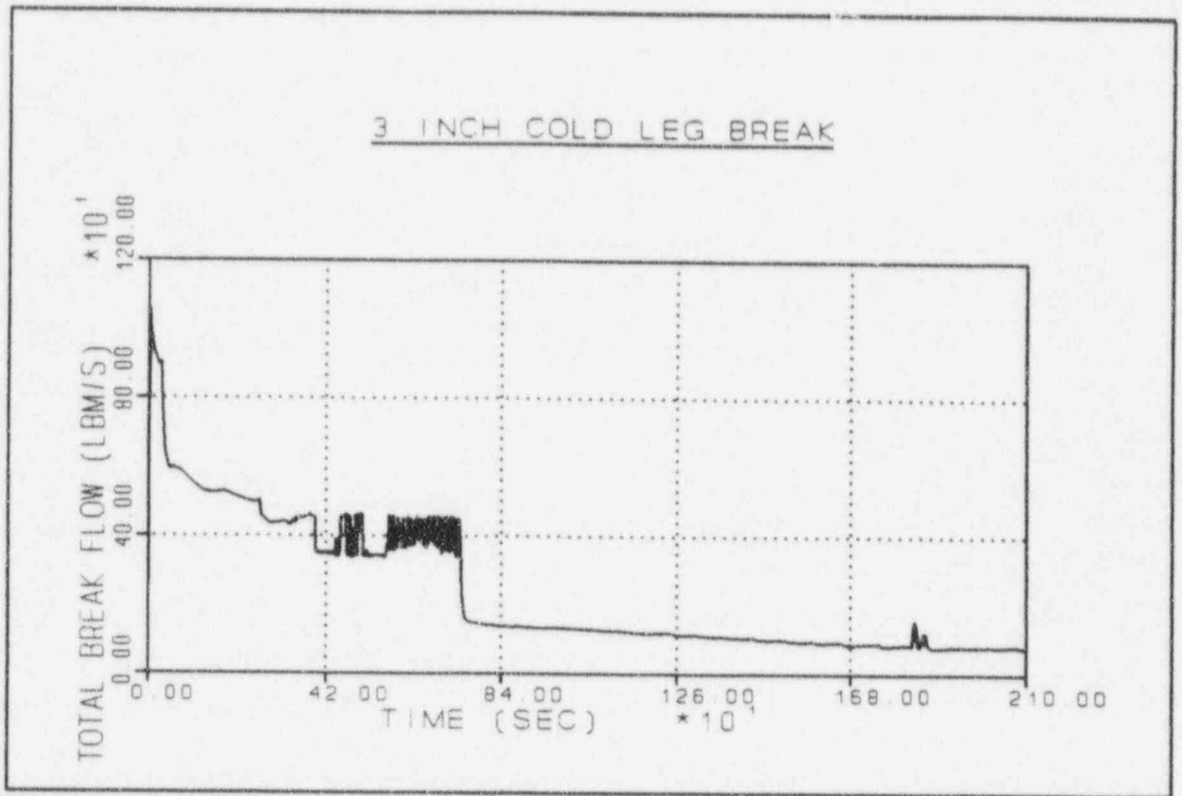

Figure 3.12 Total Break Mass Flow Rate

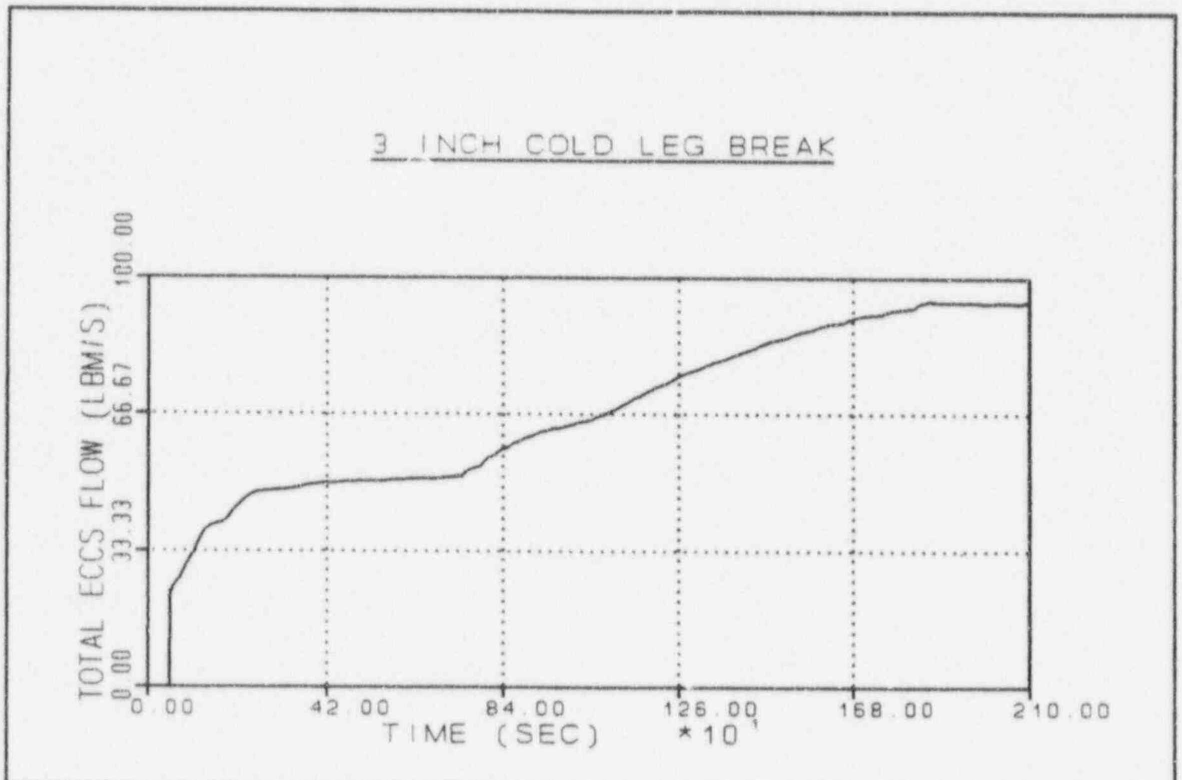

Figure 3.13 Total ECCS (single train) Mass Flow Rate

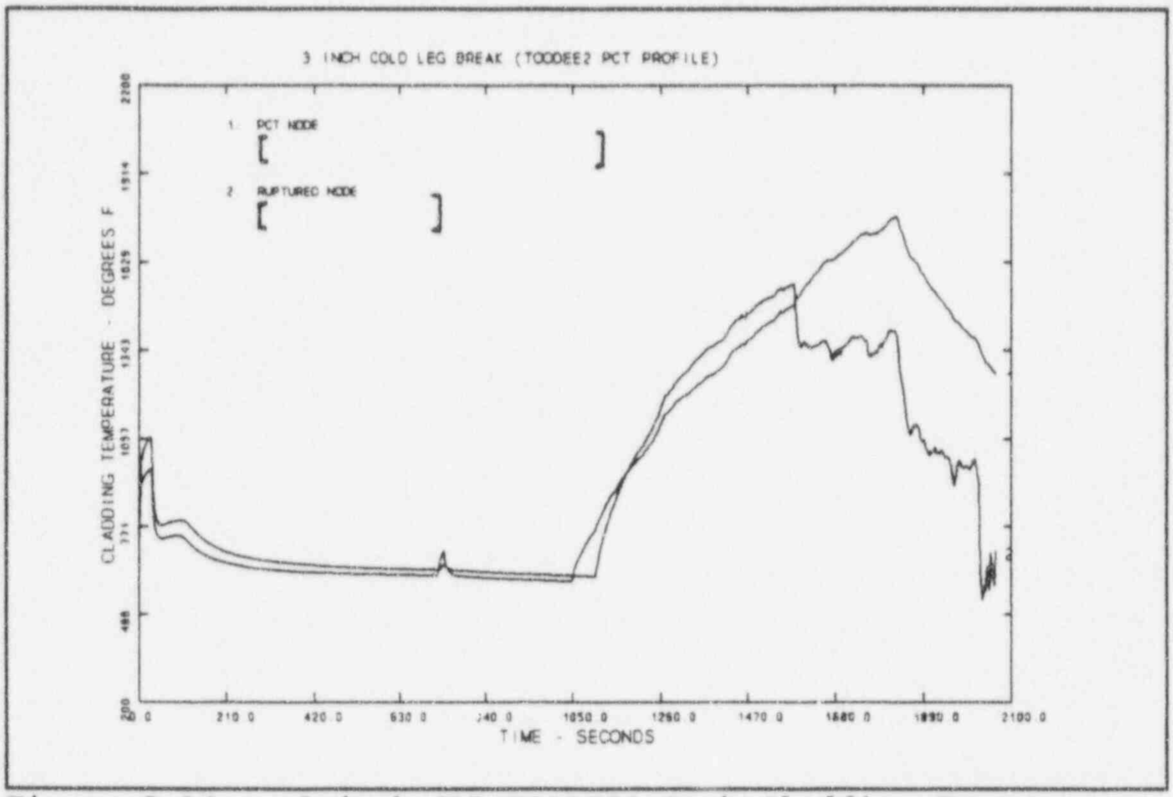

Figure 3.14 **3-inch CLB** TOODEE2 Peak Cladding Temperature

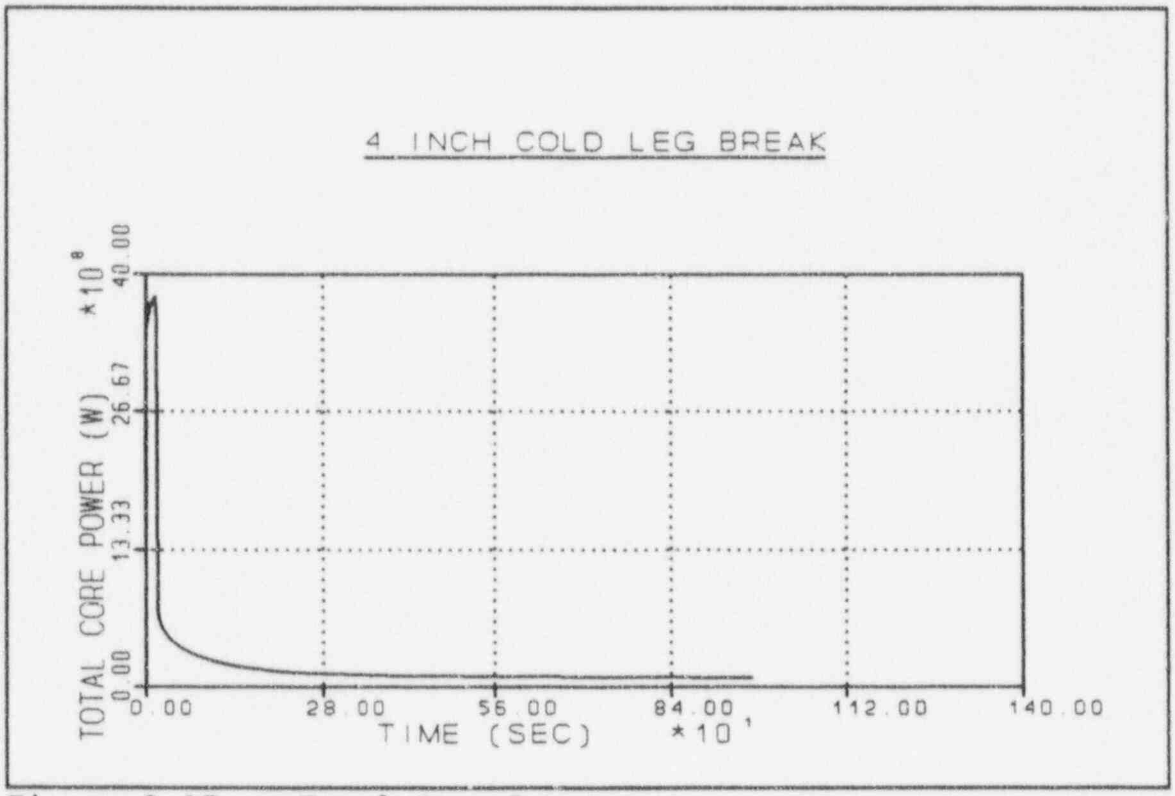

Figure 3.15 Total Core Power

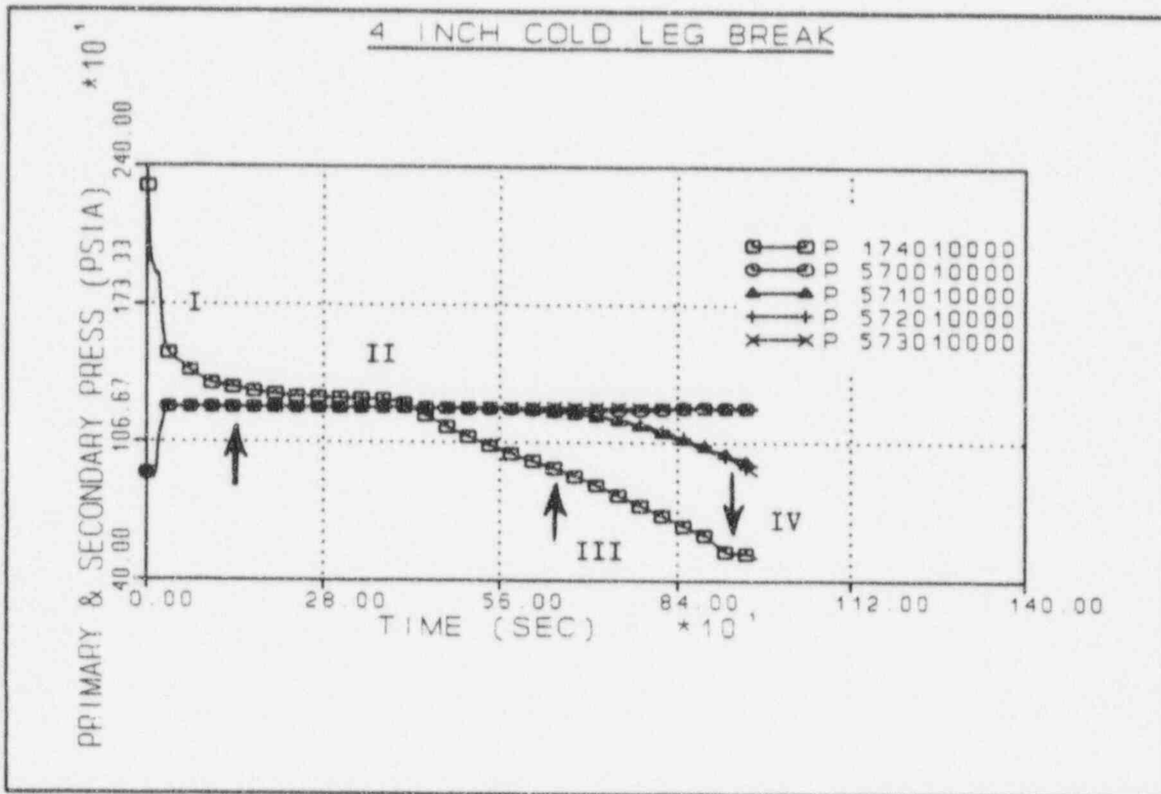

Figure 3.16 Primary & Secondary System Pressures

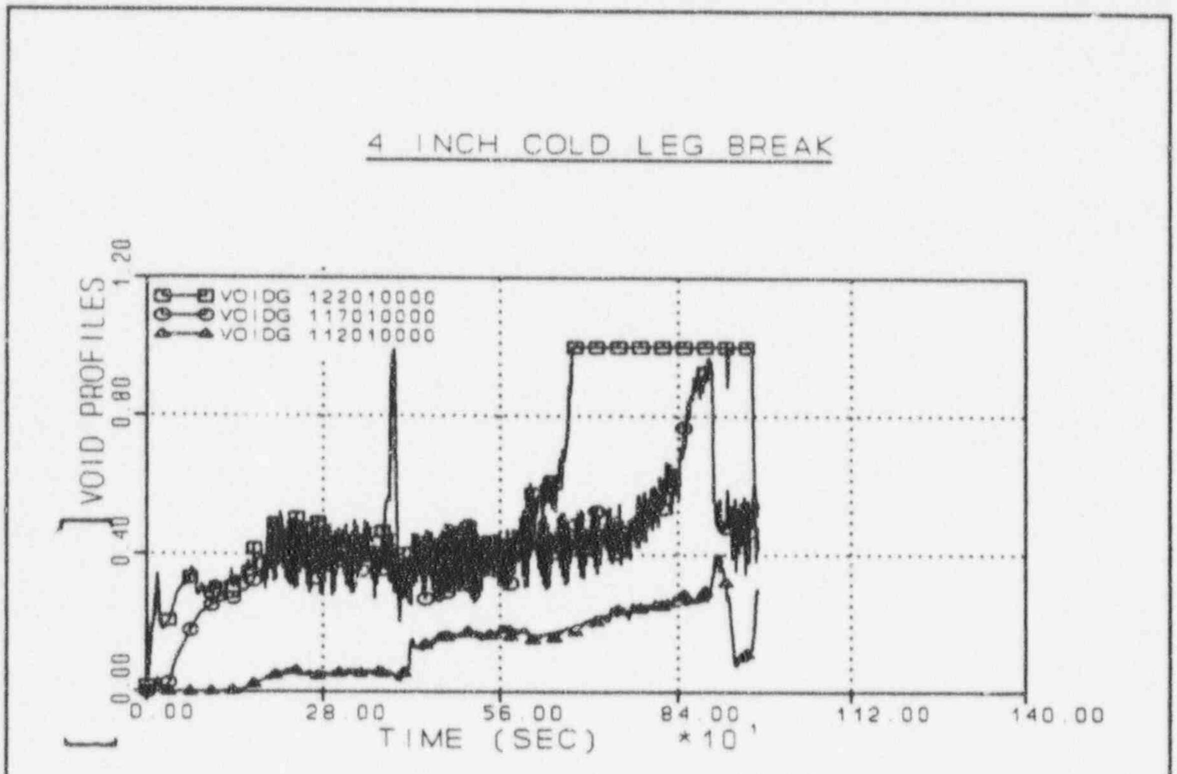

Figure 3.17 [ ] Void Fractions

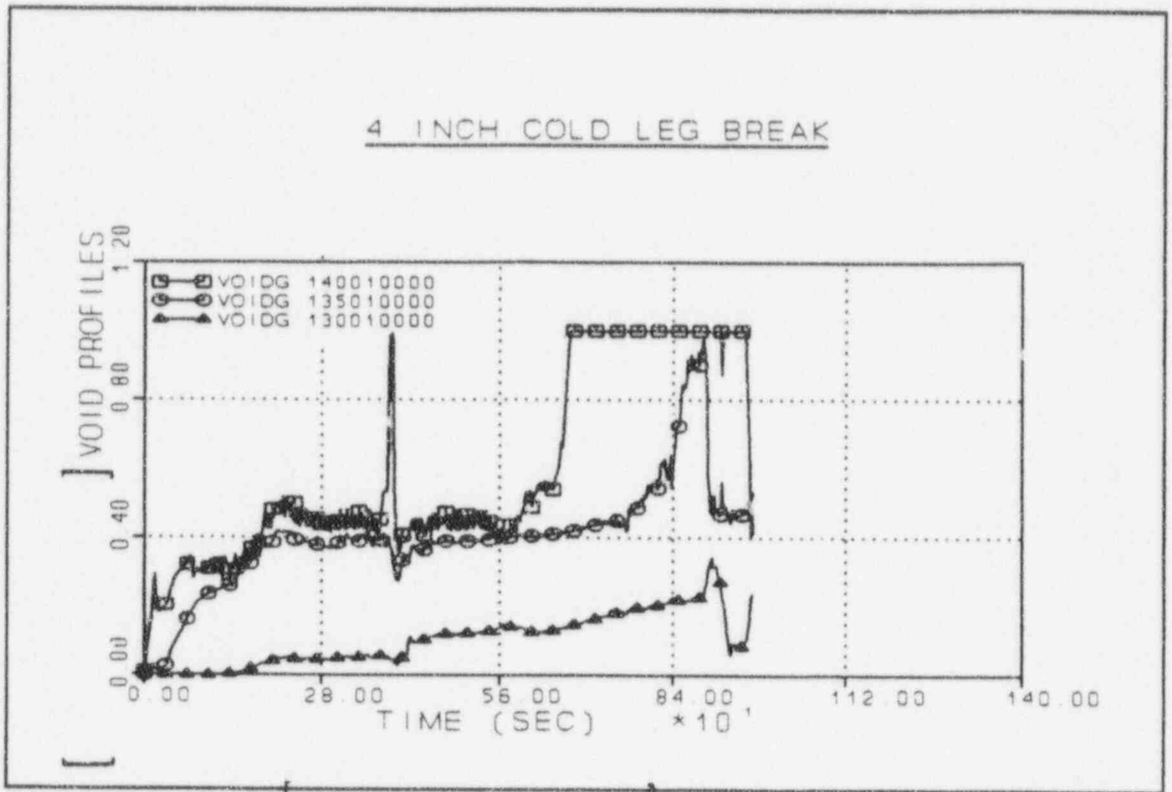

Figure 3.18 [ ] Void Fractions

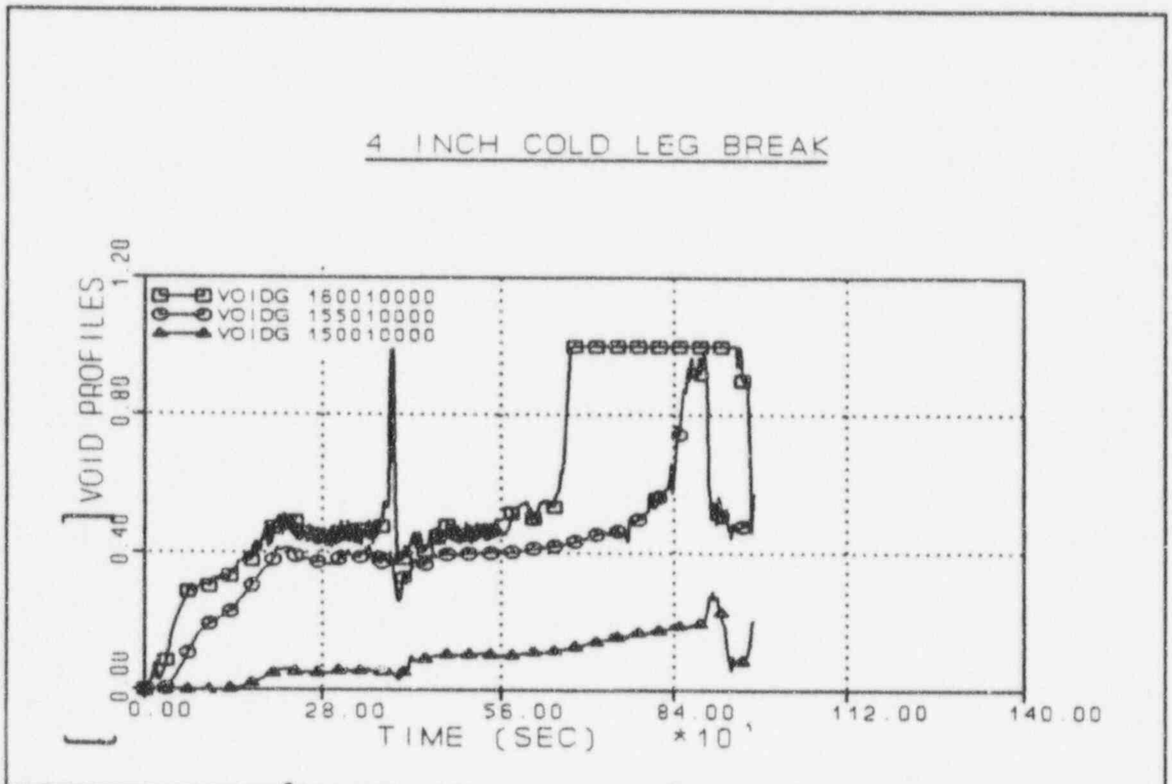

Figure 3.19 [ ] Void Fractions

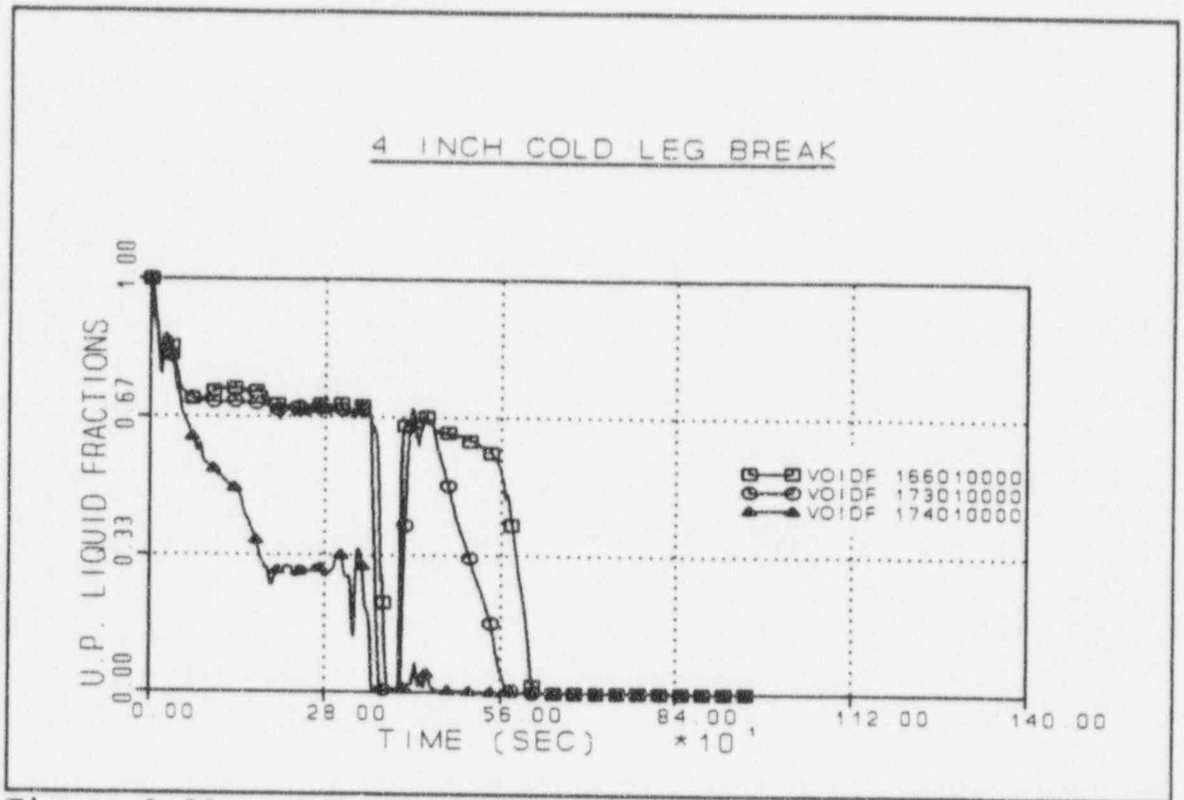

Figure 3.20 Reactor Vessel Upper Plenum Liquid Fractions

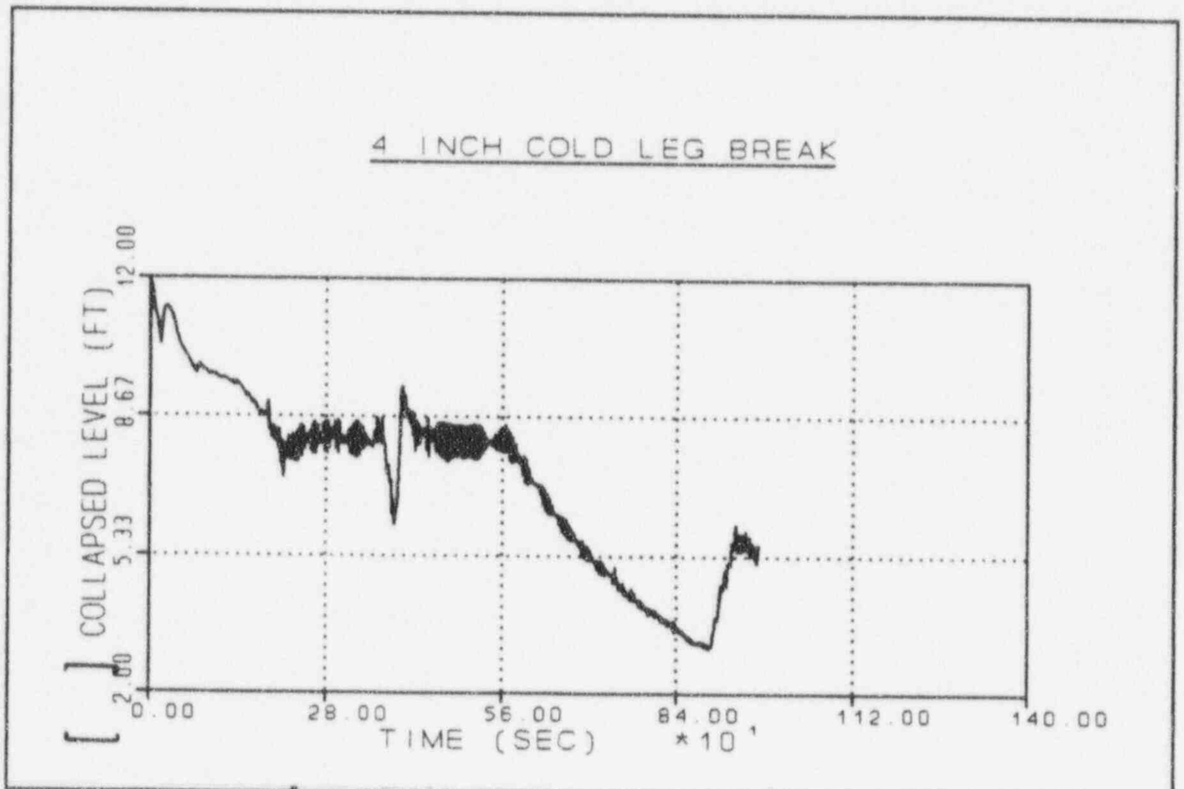

Figure 3.21 [ ] Collapsed Level

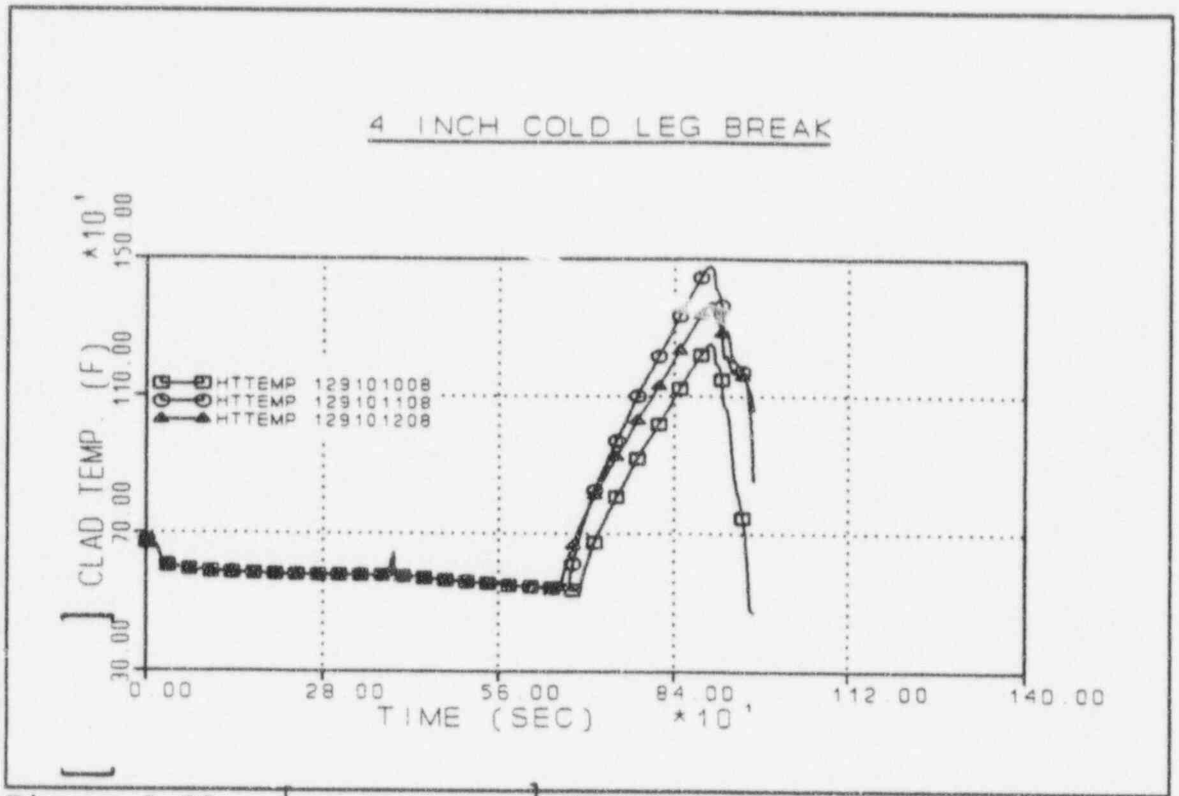

Figure 3.22 [ ] Cladding Temperature Profiles

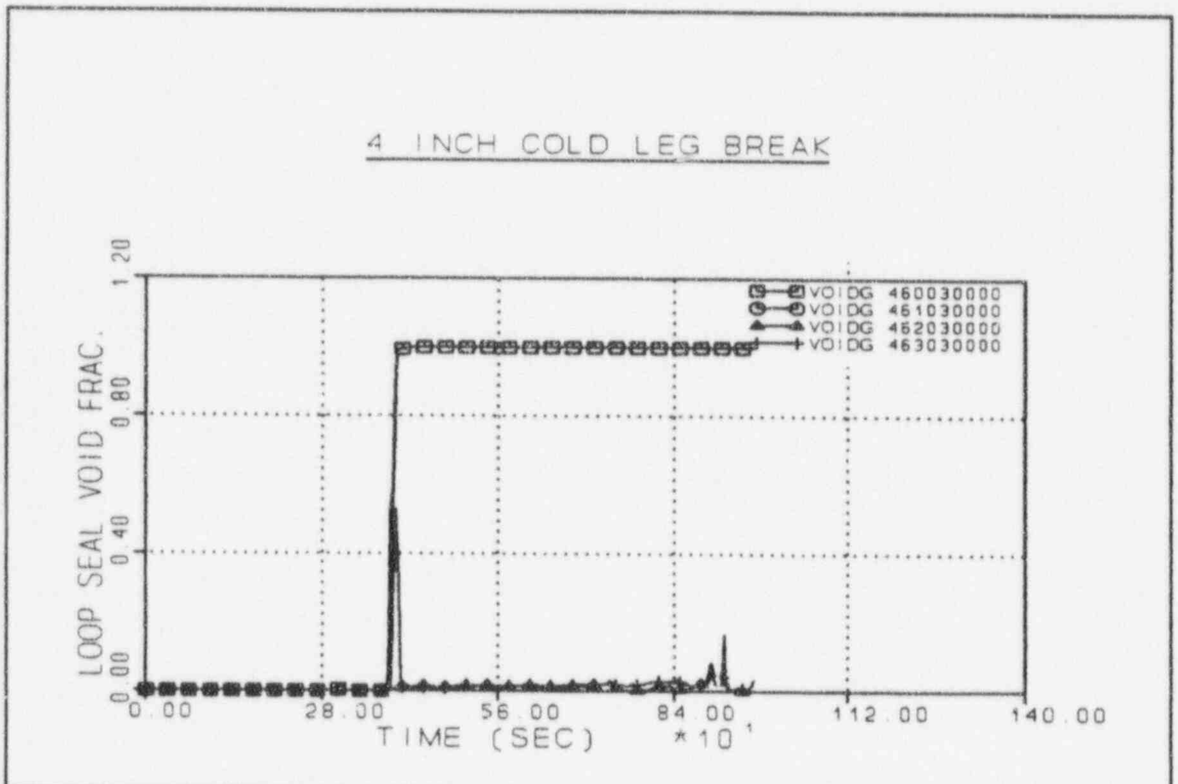

Figure 3.23 Loop Seal Vapor Void Fractions

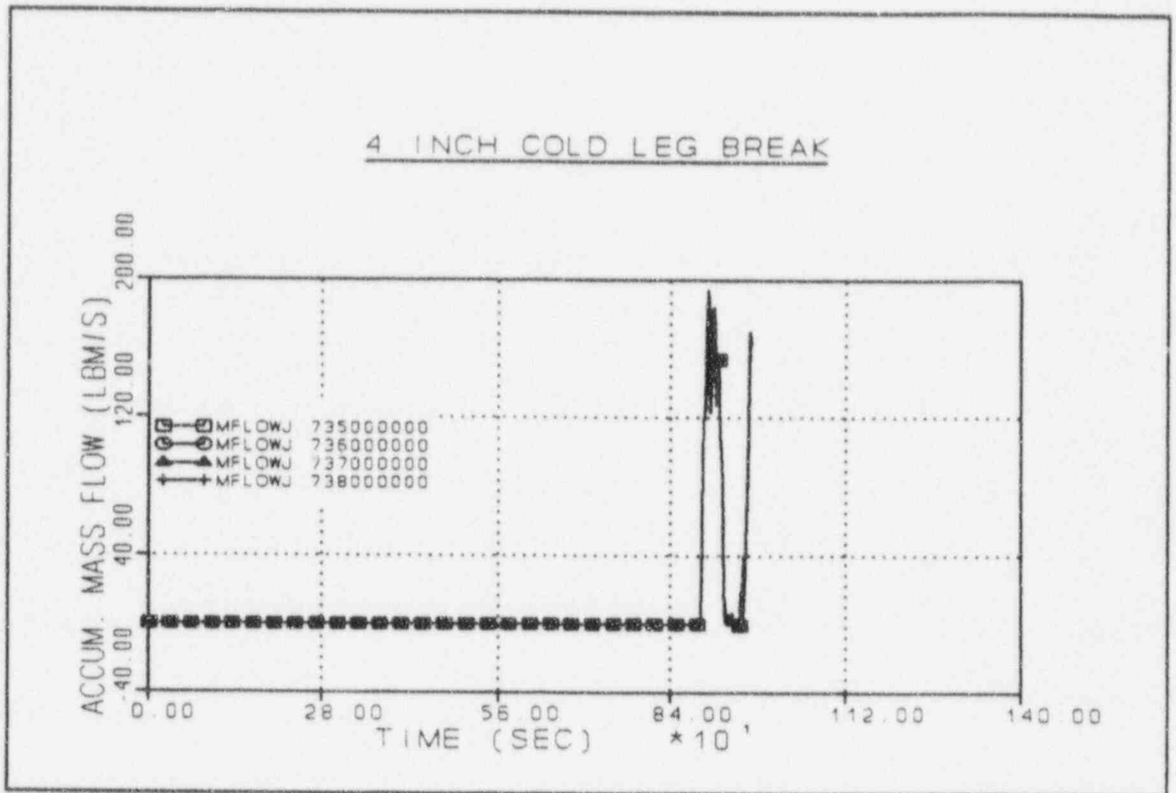

Figure 3.24 Accumulator Mass Flow Rates

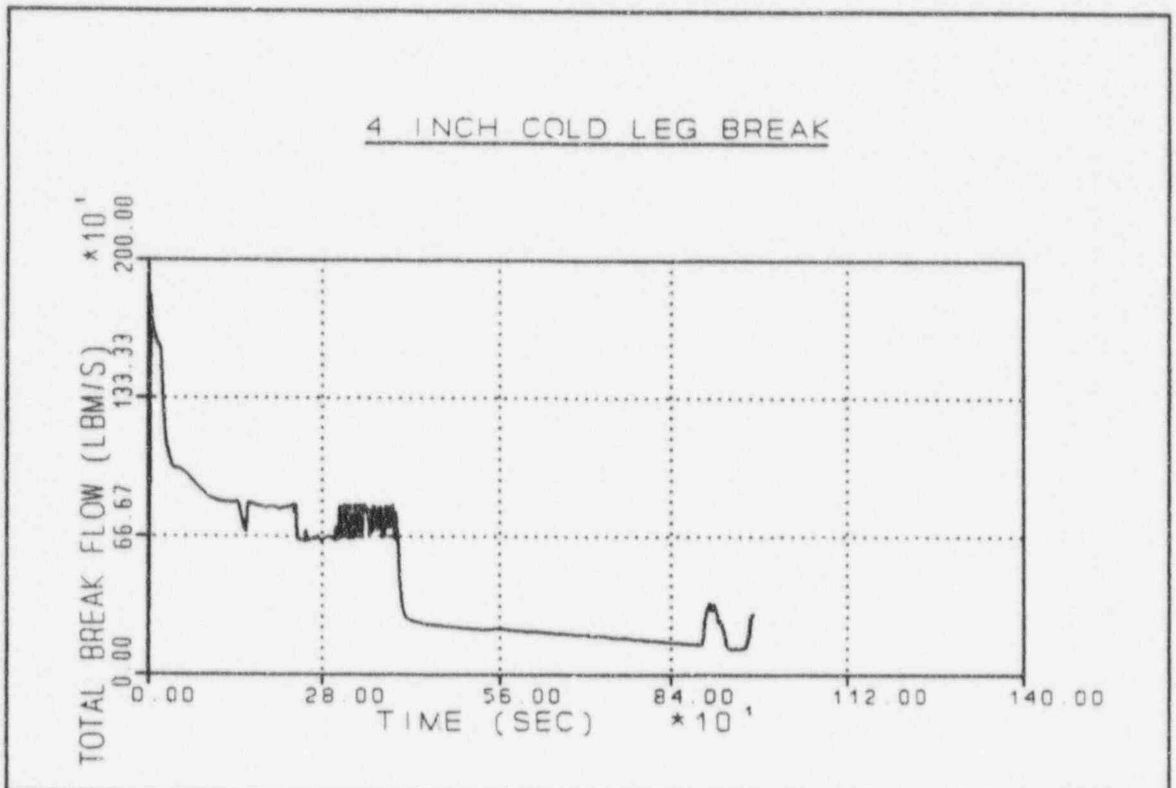

Figure 3.25 Total Break Mass Flow Rate

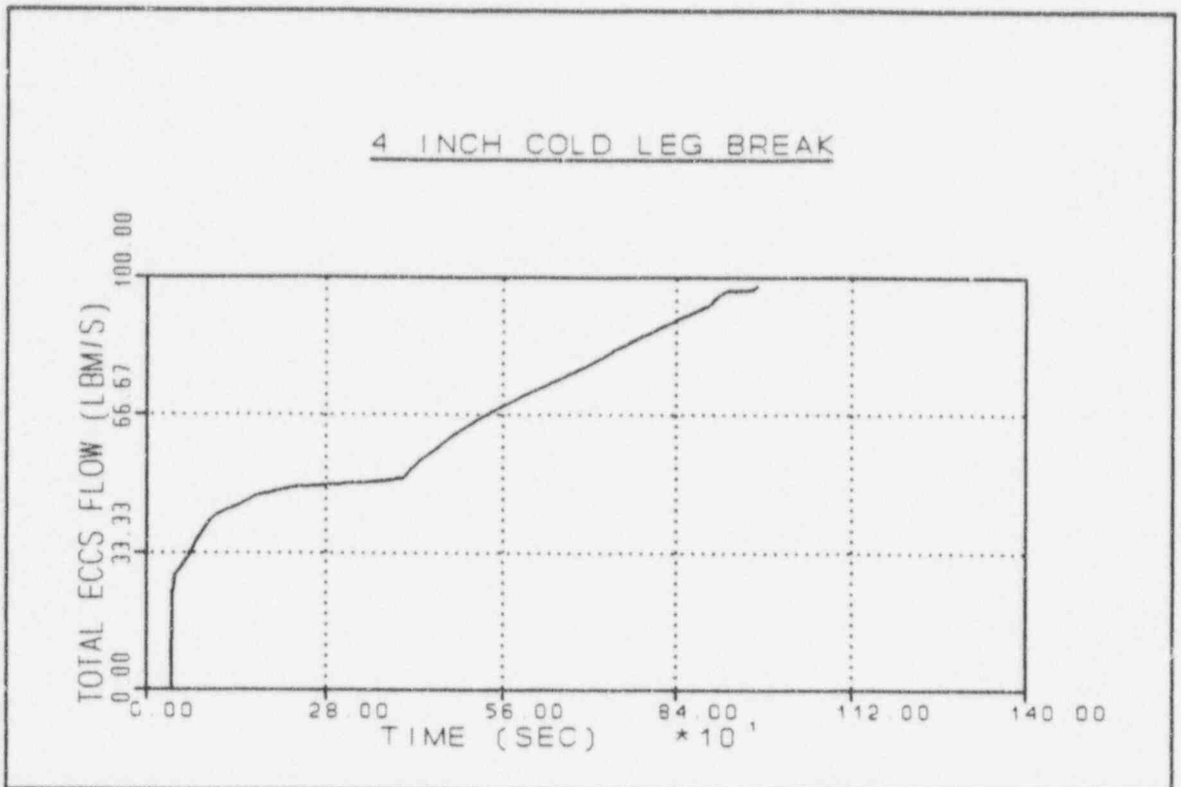

Figure 3.26 Total ECCS (single train) Mass Flow Rate

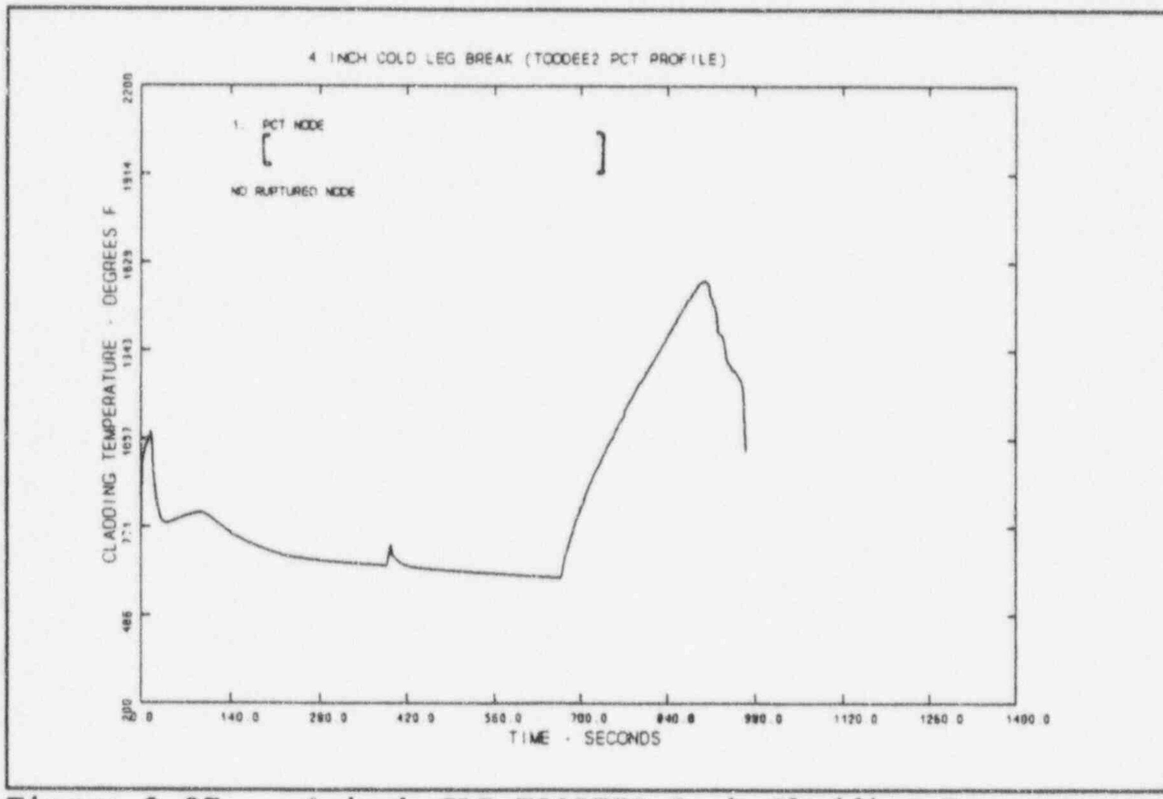

Figure 3.27 **4-inch CLB** TOODEE2 Peak Cladding Temperature

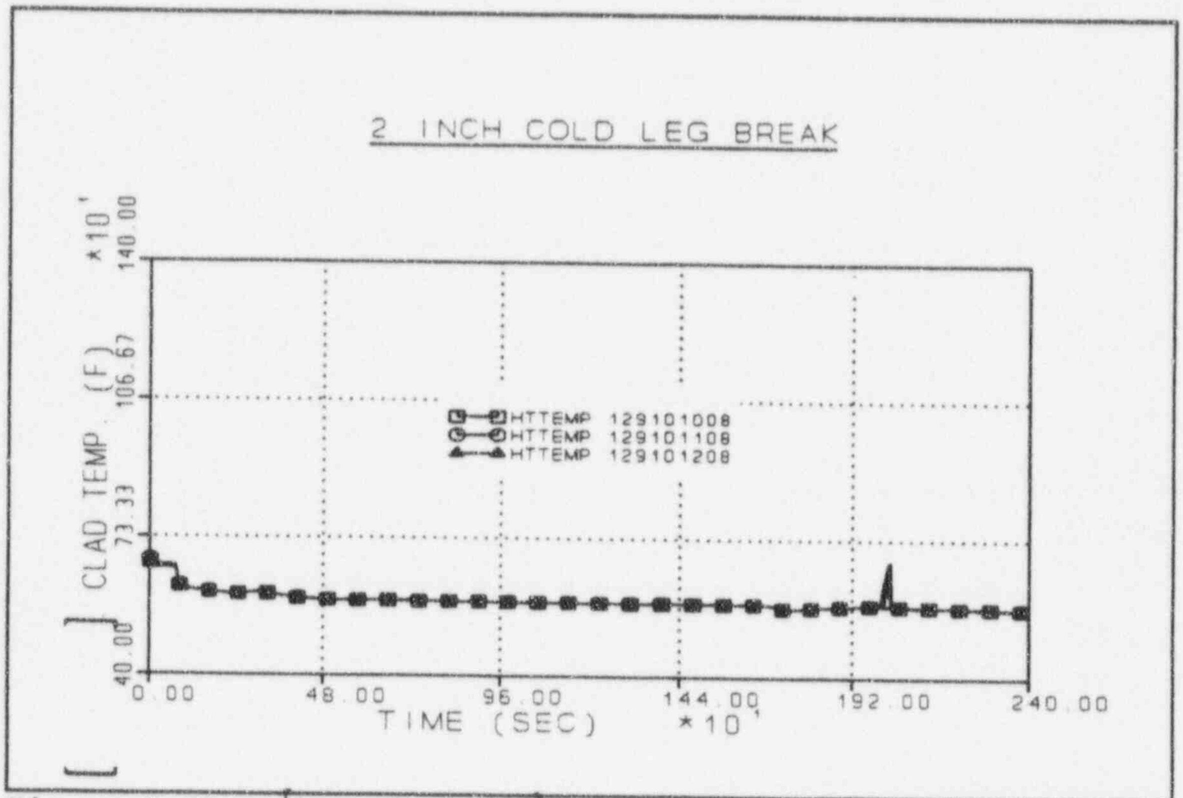

Figure 3.28 [ ] Cladding Temperature Profiles

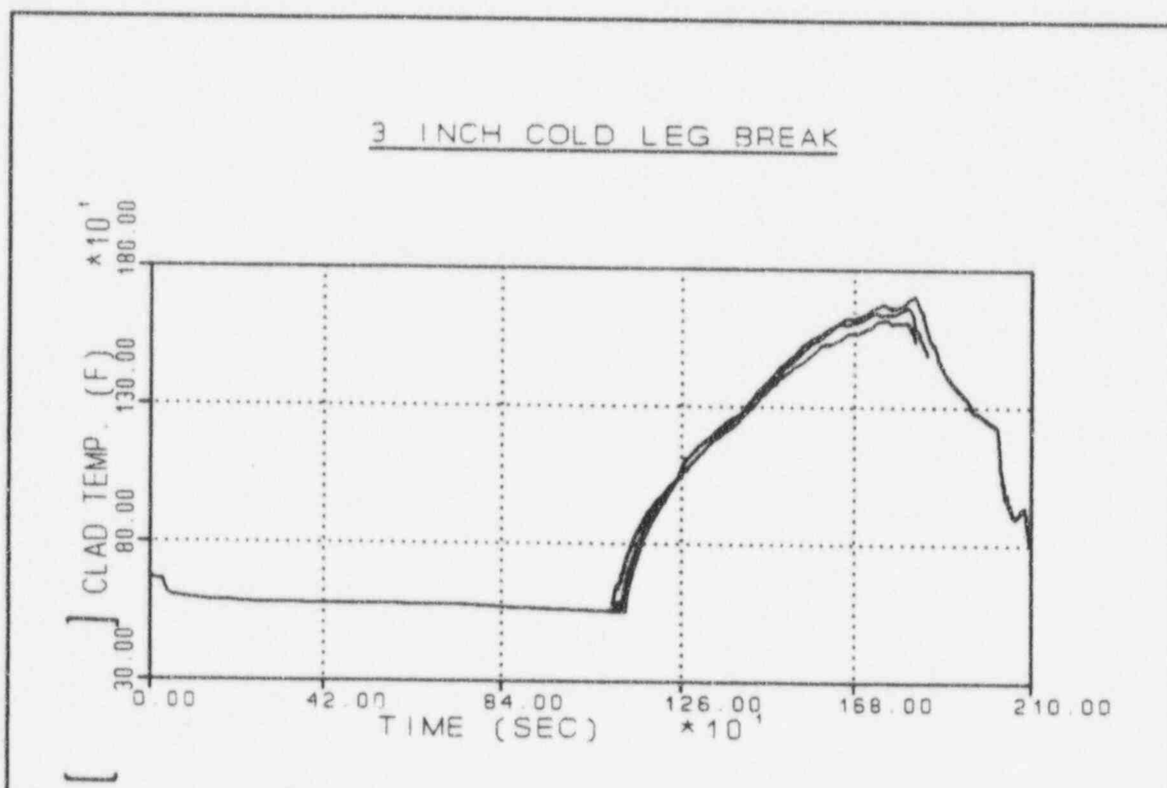

Figure 3.29 [ ] Clad Temp. for [ ]

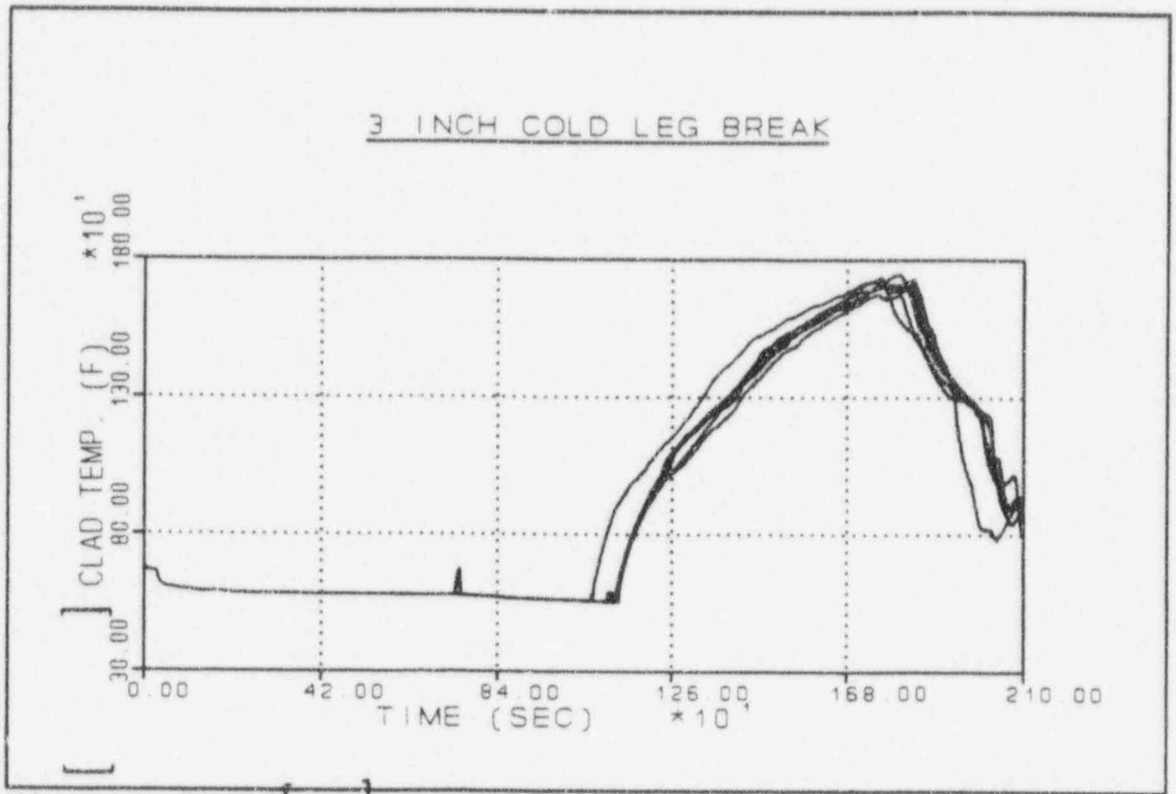

Figure 3.30 [ ] Cladding Temperature at Different Δts

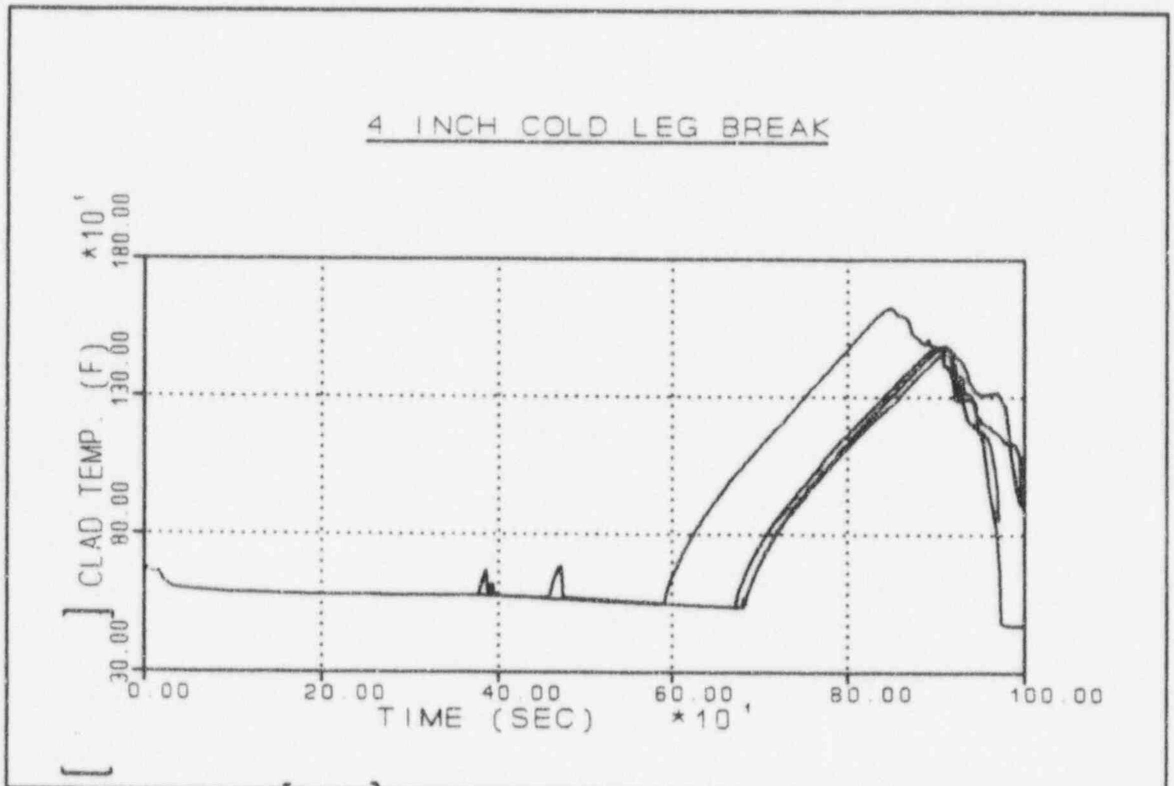

Figure 3.31 [ ] Cladding Temperature at Different Δts

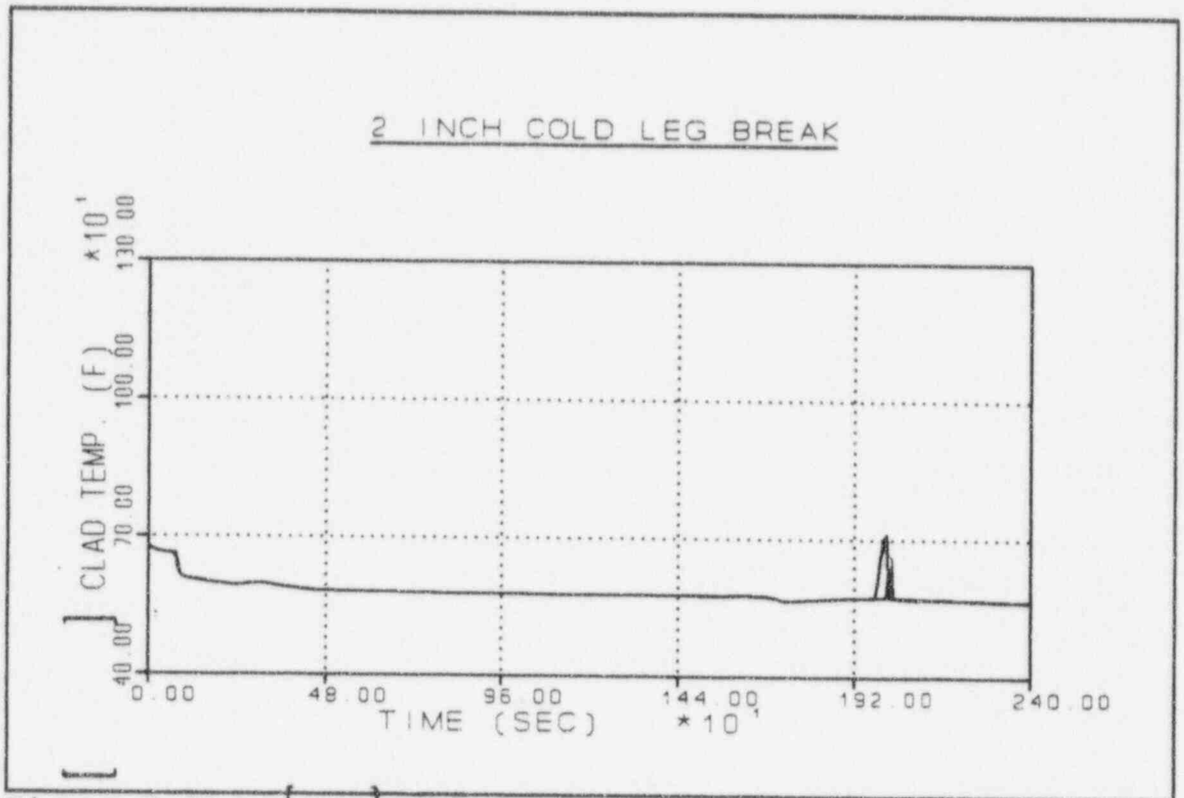

Figure 3.32 [ ] Cladding Temperature at Different Δts

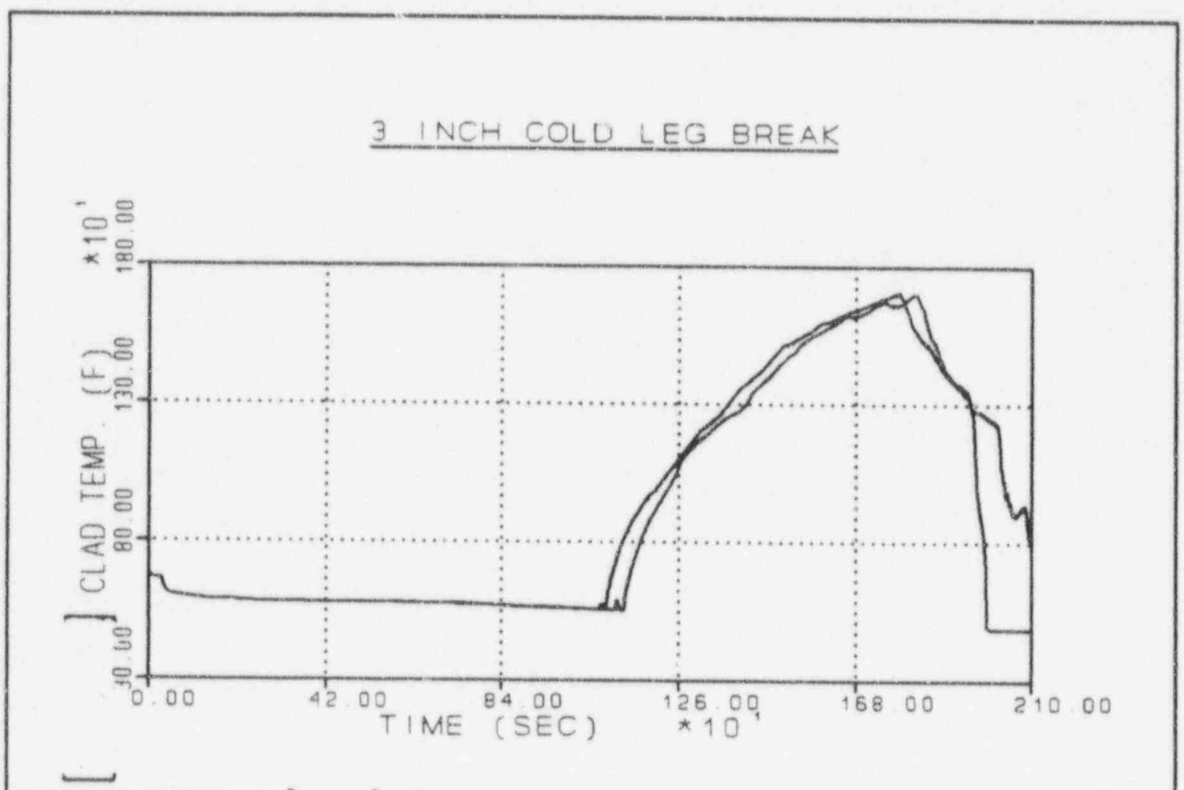

Figure 3.33 [ ] Cladding Temp. for 4- & 2-Loop Models

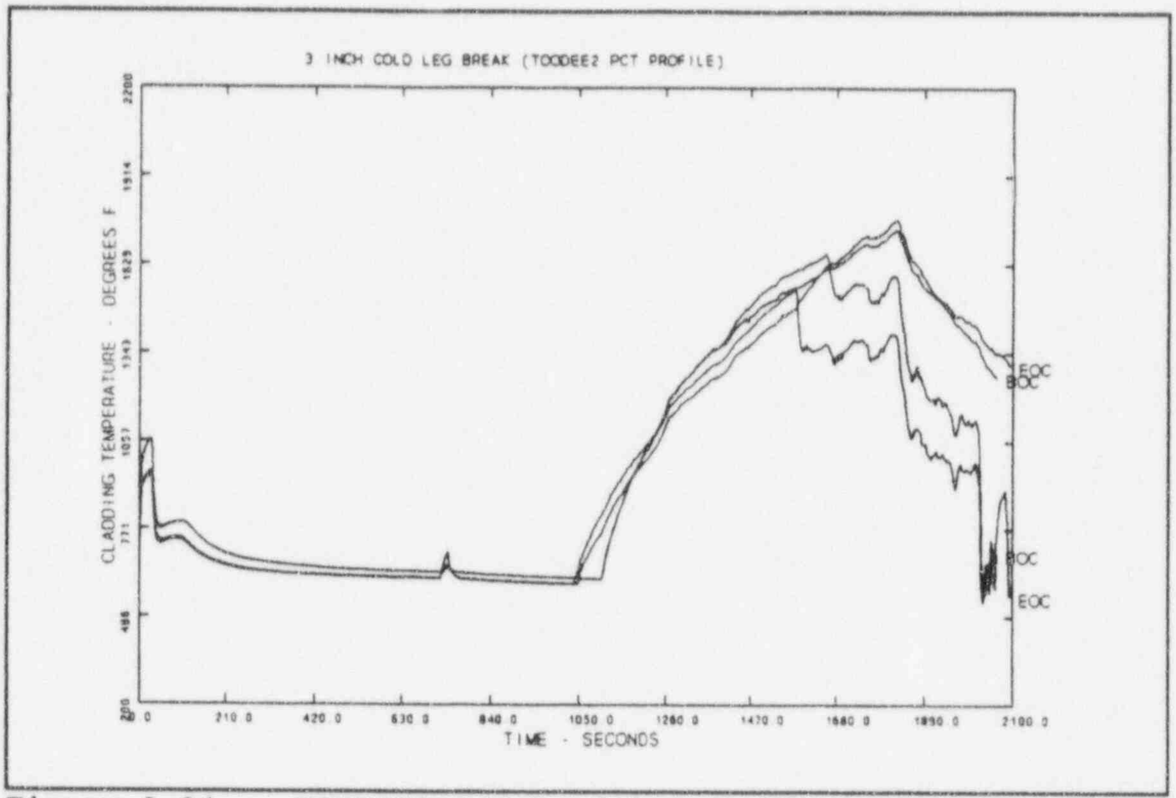

Figure 3.34 TOODEE2 PCT Profiles at BOC & EOC

CHAPTER 4

CONCLUSION

The USNRC-approved (References 1.1 and 2.1) SPC's ECCS Evaluation model entitled EXEM PWR Small Break Model has been applied to the Comanche Peak Steam Electric Station Unit One (CPSES-1).

Each calculation has been performed in close compliance with the explicitly approved EXEM PWR Small Break methodology. [

]. This was done in order to demostrate that the 2-loop model yields results which are essentially identical to the 4-loop model. TU Electric intends to use the 2-loop model in future calculations.

Six calculations, excluding [ ] studies, have been presented with two objectives:

1. To demonstrate TU Electric's ability to properly apply EXEM PWR Small Break Model (Reference 1.1); and

2. To demonstrate the development of up-to-date input decks and conclusions which are in compliance with 10 CFR 50.46 and Appendix K thereto. Together, the codes, input decks and conclusions drawn from these calculations will be applied to perform subsequent fuel cycle analyses for the Comanche Peak Steam Electric Station Unit One and Unit Two.

Table 4.1 summarizes the analyses and their key results. In each of the cases presented in this report, the calculated results show the following:

1. The calculated peak clad temperature is lower than the 2200^{0}F peak clad temperature limit set forth in 10 CFR 50.46 (b)(1).

2. The total cladding oxidation at the peak location is under the 17% limit specified in 10 CFR 50.46 (b)(2).

3. The hydrogen generated in the core by cladding oxidation is less than the 1% limit of 10 CFR 50.46 (b)(3).

4. The average core region undergoes only minor dimensional changes, no clad ruptures are calculated to occur there. Thus, the coolable geometry criterion of 10 CFR 50.46 (b)(4) is satisfied.

5. Following accumulator injection, the rods are quenched, the pumped ECCS flow exceeds the break flow and the core is well cooled thereafter. Therefore, the calculations comply with the long-term cooling criterion of 10 CFR 50.46 (b)(5).

Regarding the sensitivity studies it has been found:

1. The most limiting break is a 3 inch break in the main coolant pump discharge line.

2. The [ ] revealed little sensitivity to these [ ] for the CPSES model. [ ]

3. The optimum time step for these calculations is demostrated to be 0.005 seconds.

4. Although converged at 0.005 seconds, the actual numerical value of the PCT can vary slightly as the time step is reduced further. This can be interpreted as a convergence band. For the limiting break, this variation is concluded to be

between 20^0F and 30^0F. [

.]

5. The two-loop ANF-RELAP model yields results which are basically the same as those obtained with the explicit four-loop model. Peak clad temperature histories are nearly identical. Numerically, the two-loop ANF-RELAP PCT is 1710^0F and the four-loop PCT is 1705^0F. The corresponding TOODEE2 PCTs are 1780^0F for the 4-loop and 1793^0F for the 2-loop.

TU Electric will use the EXEM PWR Small Break model including all codes, input decks, results, conclusions, and application procedures presented in this report to perform small break LOCA analyses and evaluations in compliance with 10 CFR 50.46 criteria and 10 CFR 50, Appendix K requirements, for both Comanche Peak Steam Electric Station Unit One and Unit Two.

TABLE 4.1

SUMMARY OF RESULTS FOR BASE CASE,

2-LOOP MODEL AND EOC CASE

BREAK SIZE (INCHES)	EXPOSURE SENSITIVITY		2-LOOP ANF-RELAP MODEL
	BASE CASE (BOL)	END OF CYCLE (EOC)	
3.0	1779.8 ^{0}F (1) 1.947 % (2) 0.303 % (3)	1745.7 ^{0}F 2.513 % 0.310 %	1792.8 ^{0}F 2.061 % 0.327 %
4.0	1571.7 ^{0}F 0.379 % 0.044 %		
2.0	NO HEATUP		

<u>NOTES:</u>

ALL RESULTS FROM TOODEE2:

(1) PEAK CLADDING TEMPERATURE
(2) PERCENT LOCAL CLAD OXIDATION
(3) PERCENT CORE-WIDE[12] OXIDATION

---

[12] hot pin value is used as an upper bound for the core-wide value.

CHAPTER 5

REFERENCES

Chapter 1:

1.1 Letter,G.M.Holahan (USNRC) to R.A.Copeland, Siemens Power Corporation (SPC), "Acceptance for Referencing of the Topical Report XN-NF-82-49(P)(A), Revision 1, Supplement 1, Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model'(TAC No. M83302)," October 3, 1994.

Chapter 2:

2.1 Siemens Power Corporation, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," XN-NF-82-49 (P) (A), Revision 1 Supplement 1, May 1992.

2.2 V. H. Ransom, et. al., "RELAP5/MOD2 Code Manual, Volume 1: Code Structure, Systems Models, and Solution Methods," NUREG/CR-4312, Rev.1, March 1987.

2.3 F. J. Moody, "Maximum Flow Rate of a Single Component Two-Phase Mixture," J. Heat Transfer, Trans. ASME, 87, pp 134-142, February, 1965.

2.4 G. G. Loomis, "Summary of The Semiscale Program (1965 - 1986)," NUREG/CR-4945, EGG-2509, July 1987.

2.5 Division of Technical Review, Nuclear Regulatory Commission, "TOODEE2: A Two Dimensional Time Dependent Fuel Element Thermal Analysis Program," NUREG-75/057, May 1975.

2.6 Nuclear Regulatory Commission, Division of Technical Review, "WREM: Water Reactor Evaluation Model," NUREG-75/056 (Revision 1) May 1975.

2.7 Advanced Nuclear Fuels Corporation, "USNRC's Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Topical Reports," July 1986.

2.8 P. Salim, "Guidelines for PWR Safety Analysis Section 10.0: Small Break LOCA Analysis," EMF-1238 (P), Rev.1, Siemens Power Corporation, October, 1994.

2.9 TU Electric, "Steady State Reactor Physics Methodology," RXE-89-003-P, July 1989.

2.10 D. S. Huegel, C. M. Thompson, "Comanche Peak Unit 1 Accident Assumptions Checklists," WCAP-12368, August 1990, Revision 1.

2.11 Comanche Peak Steam Electric Station Units 1 and 2, Technical Specifications.

2.12 Westinghouse Letter, J.L. Vota (W) to W. J. Cahill (TUE) "Comanche Peak Units 1 & 2 Safety Evaluation for Reduced ECCS Flow to Prevent Charging/SI and HHSI Pump Runout During Recirculation," WPT-13963, September 25, 1991.

Chapter 3:

3.1 Comanche Peak Steam Electric Station Units One And Two, "Final Safety Analysis Report," Section 15.6, Amendment 78, January 15, 1990.

3.2 USNRC, "Water Reactor Evaluation Model (WREM): PWR Nodalization and Sensitivity Studies," - Technical Review U.S. Atomic Energy Commission, October 1974.

3.3 TUElectric, "Power Distribution Control Analysis And Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," RXE-90-006-P-A, June, 1994.

APPENDIX

DESCRIPTION OF THE COMPUTATIONAL TOOLS

The EXEM PWR Small Break Model consists of three basic computer codes:

1. RODEX2
2. ANF-RELAP
3. TOODEE2

The codes, their interfaces, interrelationships and respective inputs and outputs are summarized in Figure A.1 and Table A.1. The function of each code is described in the following sections.

A.1 RODEX2

RODEX2 is used within the EXEM PWR Small Break Model framework to provide initial conditions for the ANF-RELAP and TOODEE2 calculations, as illustrated in Figure A.1 and Table A.1.

RODEX2 describes the thermal-mechanical performance of fuel during its operational lifetime preceding the LOCA. The determination of stored energy for the LOCA analysis requires a conservative fuel rod thermal-mechanical model that is capable of calculating fuel and cladding behavior, including the gap conductance between fuel and cladding as a function of burnup. The parameters affecting fuel performance, such as fission gas release, cladding dimensional changes, fuel densification, swelling, and thermal expansion are accounted for.

RODEX2 provides an integrated evaluation procedure for considering the effect of varying temporal and spatial power histories on the temperature distribution, inert fission gas release, and deformation distribution (mechanical stress-strain and density state) within the fuel rod. The surface conditions for the fuel rods are calculated with a thermal-hydraulic model of a rod in a flow channel. [

]

The calculational procedure of RODEX2 is a time incremental procedure so that the power history and path dependent processes can be modeled. The axial dependence of the power and burnup distributions are handled by dividing the fuel rod into a number of axial segments which are modeled as radially dependent regions whose axial deformations and gas releases are summed. Power distributions can be changed at any time and the coolant and cladding temperatures are readjusted at all axial nodes. Deformation of the fuel and cladding and gas release are calculated using shorter time steps than those used to define the power generation. Gap conductance calculations are made for each of these incremental calculations based on gas released through the rods and the accumulated deformation at the mid point of each axial region within the fueled region of the rod. [

]

A.2 ANF-RELAP

ANF-RELAP is a modified version of RELAP5/MOD2, INEL Cycle 36.06. The RELAP5/MOD2 code is described in detail in Reference 2.2. RELAP5/MOD2 has been modified in [ ] major ways to produce ANF-RELAP:

[

]

The ANF-RELAP model is described in Section 2.4.1. Initial thermal-hydraulic conditions are determined using LOOPT (Section A.4.1) followed by initialization calculations which include a null transient run. Initial fuel rod stored energy is determined using RODEX2 (Section A.1). [

]

The ANF-RELAP calculation provides the thermal-hydraulic boundary conditions for the TOODEE2 code, as shown in Table A.1 and Figure A.1.

A.3 TOODEE2

TOODEE2 is a two-dimensional, time-dependent fuel rod thermal and mechanical analysis program. TOODEE2 models the fuel rod as radial and axial nodes with time-dependent heat sources. Heat sources include both decay heat and heat generation via reaction of water with zircaloy. The code considers conduction within solid regions of the fuel, radiation and conduction across gap regions, and convection and radiation to the coolant and surrounding rods, respectively. Based upon the calculated stress in the cladding (due to the differential pressure across the clad) and the cladding temperature, the code determines whether the clad has swelled and ruptured. Once fuel rod rupture is determined, the code calculates both inside and outside metal water heat generation.

The outputs of TOODEE2, namely: peak clad temperature, percent local cladding oxidation and percent pin-wide cladding oxidation are compared to the 10 CFR 50.46 (b) (1) through (3) criteria. Regarding (3), if pin-wide oxidation is less than 1% it is concluded that the criteria of less than 1% core-wide oxidation is met.

A.4 DATA PREPARATION AND TRANSFER TOOLS

Also used with the EXEM PWR Small Break model also are 2 additional codes for obtaining input information and/or transferring results between the basic codes described above:

1. LOOPT
2. SHAPE/PWR (SHAPE.PUN)

A.4.1 LOOPT

This code is used to determine initial thermal-hydraulic conditions for ANF-RELAP. It is needed because actual plant data, design data, or other safety analysis data are not necessarily available for the initial conditions desired. For example, 5% steam generator tube plugging. Flows, pressure drops and temperatures are used to initialize the ANF-RELAP steady-state deck. These LOOPT conditions are not exactly the initial conditions for the accident because ANF-RELAP initialization includes steady-state as well as a null transient calculation prior to initiation of the LOCA calculation.

A.4.2 SHAPE/PWR (SHAPE.PUN)

SHAPE automates the building of portions of input decks to ANF-RELAP, RODEX2, and TOODEE2. The code prepares input related to the axial power profile. The SHAPE code can alter and re-normalize a given axial power shape to a prescribed axial peaking factor. It then generates the axial power factors for input to the RODEX2 and TOODEE2 codes and power fractions for ANF-RELAP.

TABLE A.1

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

[ ]
INPUT:
(1)* [ ]
OUTPUT:
(2) [ ]

* The numbers in this table correspond to the numbers in Figure A.1.

TABLE A.1 (Cont'd)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

SHAPE/PWR
INPUT:
(2) [ ]
(3) [ ]
OUTPUT:
(10) [ ]
(4) [ ]
(7) [ ]

TABLE A.1 (Cont'd)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

RODEX2
INPUT:
(4) [ ]
(12) [ ]
OUTPUT:
(9) [ ]
(8) [ ].

TABLE A.1 (Cont'd)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

[ ]
INPUT:
(5) [ ]
OUTPUT:
(6) [ ]
(15) [ ]

TABLE A.1 (Cont'd)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

ANF-RELAP
INPUT: (10) Core power and weighing fractions (8) [ ] (11) o NSSS information (Table 2.2) o ECCS, SG AFW, safety valve flows (Tables 2.6 and 2.9) o Trips and delays (Table 2.7) o Fuel rod/assembly information (Table 2.8) o Neutronics information (Tables 2.3, 2.4, 2.5) **OUTPUT:** (13) [ ]

TABLE A.1 (Cont'd)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

TOODEE2
INPUT:
(9) [ ]
(7) [ ]
(13) [ ]
OUTPUT:
(14) [ ]

Figure A-1 TU Electric's SBLOCA Analysis Computer Code Interface
(Numbers in Circles Correspond to those in Table A-1)