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SMALL BREAK LOSS OF COOLANT ACCIDENT ANALYSIS METHODOLOGY

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ABSTRACT

This report is presented to demonstrate the application of the USNRC-approved Advanced Nuclear Fuels (ANF) Corporation's 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 LOCA. 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 full spectrum of small breaks, ranging from 4 to 8 inches in diameter, is examined.

Furthermore--in order to support the technical specification linear heat generation rate (LHGR) limit as a function of height-- several potentially limiting power shapes are considered. Analyses are presented for the chopped cosine and two additional shapes, one of them most limiting according to

the vendor's FSAR analyses and a similar one which was developed using the methods described in Reference 1.2.

The present methodology—including all codes, input decks and conclusions reached within this report—will be applied to subsequent fuel cycles for the Comanche Peak Steam Electric Station Unit One and Unit Two. Evaluations will be performed on the basis of the cycle-specific parameters to verify that the results of the present analyses remain bounding.

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CHAPTER 1

INTRODUCTION

The present report describes the application of the USNRC-approved (Reference 1.1) Advanced Nuclear Fuels (ANF, formerly Exxon Nuclear) Corporation's ECCS Evaluation Model, entitled "EXEM PWR Small Break Model", to the Comanche Peak Steam Electric Station Unit One (CPSES-1).

The method is used to perform the Small Break LOCA-ECCS (Emergency Core Cooling Systems) 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 EXEM PWR Small Break methodology (Chapter 2), including the details of the nodalization schemes and procedures followed during all phases of the LOCA, which is postulated to occur with the plant in normal operation. Each calculation is performed in exact compliance with the explicitly approved EXEM PWR Small Break methodology. Regarding features of the calculation procedure which are "implied" in the approval, there has been but one deviation: the thermal-hydraulic calculations represent the average core

region using nine axial nodes (rather than the three shown in ANF's submittal). This deviation has been made in order to increase accuracy.

Two types of sensitivity studies are presented in Chapter 3. The first is a break spectrum study. Breaks ranging from 4 to 8 inches in diameter are examined in order to comply with 10 CFR 50.46 (a)(1)(i).

The second type of sensitivity study examines all realistic potentially limiting axial power shapes in order to support the LHGR limit as a function of height and to satisfy the requirements of 10 CFR 50, Appendix K Part I, A, 1. This is done as follows: First the population of shapes is developed through the axial power distribution control analysis described in Reference 1.2. Then, the shapes which are closest to the Technical Specification LHGR limit are selected. After that, the selected shapes are adjusted upward until the axial power shape curve touches the curve representing the Technical Specification LHGR limit as a function of core height. Finally, the shapes which are most likely to have the highest integrated power up to the PCT elevation are selected. Analyses are presented for the chopped cosine (Shape 2, Figure 3.1) and two additional shapes, one of them most limiting according to the vendor's FSAR analyses (Shape 1, Figure 3.1) and a similar one (Shape

3, Figure 3.1) which was developed using the method described above. These are the most likely candidates to yield the highest PCT according to the criterion just described.

In chapter 4, results from these sensitivity studies are used to establish the most limiting small break LOCA case for the EXEM PWR Small Break methodology and to show compliance with the LOCA-ECCS criteria in 10 CFR 50, Appendix K for CPSES-1.

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

The main objective of the work performed in connection with the present report is to obtain approval of this methodology — including all codes, input decks, inferences and conclusions — so that it may be applied to the Comanche Peak Steam Electric Station Unit One and Unit Two for subsequent fuel cycle analyses and to address pertinent licensing issues at any time. Evaluations will be performed on the basis of specific parameters to insure that results of the present analyses remain bounding, whenever such analyses are warranted.

CHAPTER 2

DESCRIPTION OF THE METHOD

The present report describes the application of Advanced Nuclear Fuels' ECCS Evaluation model, entitled "EXEM PWR Small Break Model", to the Comanche Peak Steam Electric Station Unit One.

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 Parameters, (2) System Thermal-Hydraulic Response Analysis and (3) Fuel Thermal Response Analysis. These are discussed in the sections that follow. 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 PARAMETERS

Calculations are required to determine initial fuel conditions for both ANF-RELAP and TOODEE2. For ANF-RELAP these are peak stored energy in the form of fuel temperatures. For TOODEE2 these also include: fission gas

inventory, gap width, crack and plenum volume. These calculations are performed using the RODEX2 code. This code is also part of the EXEM/PWR 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.

Four RODEX2 calculations are normally performed. The first is to determine the time of peak stored energy. The next two calculations determine the initial fuel temperatures to be used to initialize ANF-RELAP fuel temperatures in each of the two core regions, using the previously determined maximum stored energy and the power in each region. The fourth calculation sets the initial conditions for TOODEE2.

2.2 SYSTEM THERMAL-HYDRAULIC RESPONSE ANALYSIS

The system Thermal-hydraulic response is analyzed using ANF-RELAP, a modified version of RELAP5/MOD2, INEL Cycle 36.02.

The ANF-RELAP system model used for CPSES-1 is described in detail in Section 2.4.1. The initial conditions of the ANF-RELAP fuel rod model, i.e. stored energy are determined in a separate set of calculations using RODEX2, as described in Section 2.1 and 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:

(1) The Moody critical flow model (Reference 2.3) substitutes the RELAP5/MOD2 critical flow model during two-phase discharge in order to comply with the related requirement of 10 CFR 50, Appendix K, Section I, C, 1, b.

(2) The RELAP5/MOD2 mixture level calculation is modified with the objective of producing a two-phase level more suitable for the TOODEE2 fuel rod thermal analysis.

(3) A counter-current flow limitation (CCFL) constitutive equation based on a Kutateladze formulation with constants adjusted on the basis of the S-UT-08 test (Reference 2.4) is made available for use instead of the mechanistic interphase drag models in vertical junctions which can be selected by the user.

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). Seven sets of parameters are supplied from ANF-RELAP as shown in Figure 2.1:

(1) Saturation temperature entering the central core.

(2) Vapor mass flow rate exiting the central core.

(3) Normalized core power.

(4) Position of the two-phase central core mixture level.

(5) Average quality of the fluid below central core mixture level.

(6) Mass flow of liquid entering the central core.

(7) Temperature of the liquid entering the central core.

2.3 FUEL ROD THERMAL RESPONSE ANALYSIS

TOODEE2 (Reference 2.5) is used to calculate the hot fuel rod heat up 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 EXEM/PWR 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 energy equation is solved to determine the fuel rod thermal response. 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, viz., 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 (3) is met.

2.4 DESCRIPTION OF THE MODELS

2.4.1 CPSES-1 ANF-RELAP NSSS MODEL

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

The CPSES-1 ANF-RELAP NSSS model results from a considerable amount of engineering insight and experience gained over the past four years 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 ANF for the application of their methodology (Reference 2.8).

This section describes the ANF-RELAP base input model for the Comanche Peak Steam Electric Station Unit One (CPSES-1). The discussion of this model is divided into the following subsections:

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

2.4.1.1 VOLUMES, JUNCTIONS AND HEAT STRUCTURES

Figure 2.2 shows the CPSES-1 nodalization diagram for the ANF-RELAP base input model which is comprised of 114 regular volumes, 15 time dependent volumes, 124 regular junctions, 15 valve and time dependent junctions, 18 active heat structures representing nine axial nodes in two different regions of the core and 60 passive heat structures. Table 2.1 identifies the volumes, junctions, and heat structures associated with each region or system and provides their node number for cross-referencing with Figure 2.2. Table 2.2 summarizes key

parameter values for each of the CPSES-1 ANF-RELAP NSSS model components.

Three of the four loops are assumed to be identical and are modeled as one intact loop with appropriately scaled input. The pressurizer is connected to this loop following ANF procedures (Reference 2.8). These represent the "unbroken" loops. The "broken" loop is a single loop.

The reactor core is divided into two concentric radial sections: a central region and a peripheral region. The peripheral annulus produces approximately 70% of the power and the central region 30%. Both, the central region and the peripheral annulus are divided into nine axial volumes. Crossflow between these concentric regions is taken into consideration by flow junctions in the ANF-RELAP model which are established as shown in Figure 2.2. The fuel in the core is represented by two sets of heat structures. One set represents the fuel in the peripheral region and is associated with those fluid volumes. The other set of heat structures represents the hotter assemblies in the central region. All core fuel assembly heat structures are divided into nine axial elements.

Steam generator models include both primary and secondary sides. A 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. credit is not taken for the heat removal capability of the steam dump and bypass sys[illegible]ve percent of the steam generator tubes are ass[illegible]ged. This assumption is made 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. This is done for simplicity.

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 and decay heat 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 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.

2.4.1.3 EMERGENCY CORE COOLING SYSTEMS

The CPSES ECC system is arranged into four subsystems: (1) the high head charging/safety injection, (2) intermediate

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 intermediate head safety injection pump, and one low head residual heat removal pump with associated piping, valves, controls, and instrumentation.

Since loss of offsite power is assumed to occur coincidentally with the reactor trip, only one train is represented in the present NSSS model. The other train is removed on the assumption that one diesel generator train fails to start. 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 stage. 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 flow per each loop versus pressure values for each injection system, which are given in Table 2.6 were derived from the values given in Reference 2.10. These values

conservatively underestimate injection into the RCS because they incorporate spillage of injection to the containment while inventory depletion associated with spillage is already accounted for in the ANF-RELAP break flow calculation. In order to provide additional margin, the injection capacities of Reference 2.10 are reduced by 2% in deriving the values of Table 2.6.

The system contains four accumulators, one per loop. The three intact loop accumulators are lumped. 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) for simplicity, per ANF 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, Reference 2.8) 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.

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). The steam dump and bypass system is not credited. The safety valves operate as shown in Table 2.9.

4. Main feedwater isolation begins at the time of "S" signal and the 2 second signal processing time (Table 2.7).

5. SI Actuation Signal occurs on a low pressure "S" signal (1715 psia, Reference 2.8) followed by a 2 second delay for signal processing.

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

7. Accumulators inject at set pressure (648 psia) without delay.

8. Available auxiliary feedwater (1 turbine-, 1 motor-driven) pumps are assumed to be up and running 60 seconds after the "S" signal. Actual AFW injection

is delayed for another 98 seconds, conservatively accounting for the flow travel time down the piping. 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).

2.4.2 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.

The present TOODEE2 model divides the fuel rod into 24 axial and 10 radial nodes.

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 2.3.

The fuel pellet is divided into 8 radial rings (nodes) in which the last radial line location includes the gap. The first inner fuel pellet is node 2, and gridline 1 is identified as the pellet centerline. The last gridline is identified as the cladding outer radius. The cladding is

divided into 2 radial rings as required by EXEM PWR Small Break Model. The radial nodalization scheme is also shown in Figure 2.3.

The thermal-hydraulic bour`ary conditions for the TOODEE2 calculation are those associated with the central core region of the ANF-RELAP model, as described in Section 2.1 and in the Appendix. The power history for TOODEE2 is also obtained from ANF-RELAP.

The initial conditions of the fuel rod, i.e. stored energy, fission gas inventory, gap width, rod plenum and crack volumes are determined in a separate set of calculations using RODEX2, as described in Section 2.1 and in the Appendix.

TABLE 2.1

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

REGION OR SYSTEM	NODING DIAGRAM NUMBER	NUMBER OF VOLUMES ACTIVE	NUMBER OF VOLUMES TMDPVOL	NUMBER OF JUNCTIONS ACTIVE	NUMBER OF JUNCTIONS TMDPJUN	NUMBER OF HEAT STRUCTURE ACTIVE	NUMBER OF HEAT STRUCTURE PASSIVE
		REACTOR VESSEL (RV)					
RV DOWNCOMER (DC)							
o Upper DC	100	1	0	0	0	0	1
o Middle DC	104	1	0	3	0	0	1
o Bottom DC	108	5	0	4	0	0	5
RV LOWER PLENUM (LP)							
o Bottom LP	116	1	0	0	0	0	1
o Middle LP	120	1	0	3	0	0	1
o Top LP	124	1	0	3	0	0	1
RV CORE BYPASS & BARREL/BAFFLE (BYPASS)							
o Bottom Bypass	128	3	0	2	0	0	3
RV CORE ACTIVE FUEL REGION							
o Hot Central Core # 1	130	1	0	2	0	1	0
o Hot Central Core # 2	132	1	0	2	0	1	0
o Hot Central Core # 3	134	1	0	2	0	1	0
o Hot Central Core # 4	136	1	0	2	0	1	0
o Hot Central Core # 5	138	1	0	2	0	1	0
o Hot Central Core # 6	140	1	0	2	0	1	0
o Hot Central Core # 7	142	1	0	2	0	1	0
o Hot Central Core # 8	144	1	0	2	0	1	0
o Hot Central Core # 9	146	1	0	1	0	1	0

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

REGION OR SYSTEM	NODING DIAGRAM NUMBER	NUMBER OF VOLUMES ACTIVE	NUMBER OF VOLUMES TMDPVOL	NUMBER OF JUNCTIONS ACTIVE	NUMBER OF JUNCTIONS TMDPJUN	NUMBER OF HEAT STRUCTURE ACTIVE	NUMBER OF HEAT STRUCTURE PASSIVE
REACTOR VESSEL (RV) (Cont'd)							
o Average Core # 1	150	1	0	1	0	1	0
o Average Core # 2	152	1	0	1	0	1	0
o Average Core # 3	154	1	0	1	0	1	0
o Average Core # 4	156	1	0	1	0	1	0
o Average Core # 5	158	1	0	1	0	1	0
o Average Core # 6	160	1	0	1	0	1	0
o Average Core # 7	162	1	0	1	0	1	0
o Average Core # 8	164	1	0	1	0	1	0
o Average Core # 9	165	1	0	0	0	1	0
RV UPPER PLENUM & GUIDE TUBES (UP, GTs)							
o Bottom UP	166	1	0	5	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	3	0	0	0
o Top UP	178	1	0	0	0	0	1
RV UPPER HEAD (UH)							
o Bottom UH	180	1	0	0	0	0	1
o Middle UH	181	1	0	4	0	0	1
o Top UH	182	1	0	0	0	0	1

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

REGION OR SYSTEM	NODING DIAGRAM NUMBER	NUMBER OF VOLUMES ACTIVE	NUMBER OF VOLUMES TMDPVOL	NUMBER OF JUNCTIONS ACTIVE	NUMBER OF JUNCTIONS TMDPJUN	NUMBER OF HEAT STRUCTURE ACTIVE	NUMBER OF HEAT STRUCTURE PASSIVE
INTACT LOOP (IL) PRIMARY							
IL HOT LEGS (HLs)							
o IL HL #1	210	1	0	1	0	0	1
o IL HL's # 2&3	214	2	0	1	0	0	2
o IL HL to SG	217	0	0	1	0	0	0
IL STEAM GENERATORS (SGs)							
o IL SG U-Tubes & Inlet/Outlet Plena	220	10	0	9	0	0	8
IL CROSS-OVER LEG'S (XLGs)							
o SG to RCP Inlet	223	0	0	1	0	0	0
o IL XLG	226	3	0	2	0	0	3
IL REACTOR COOLANT PUMPS (RCPs)							
o IL RCP	230	1	0	2	0	0	0
IL COLD LEG'S (CLs)							
o IL CL # 1 RCP Side	236	1	0	1	0	0	1
IL CL # 2&3 RV Side	240	2	0	1	0	0	2
INTACT LOOP SECONDARY							
IL MAIN AND AUXILIARY FEEDWATER (MFW & AFW)							
o IL MFW Source	302	0	1	0	0	0	0
o IL MFW Flow	303	0	0	0	1	0	0
o IL AFW Source	304	0	1	0	0	0	0
o IL AFW Flow	305	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 ACTIVE	NUMBER OF VOLUMES TMDPVOL	NUMBER OF JUNCTIONS ACTIVE	NUMBER OF JUNCTIONS TMDPJUN	NUMBER OF HEAT STRUCTURE ACTIVE	NUMBER OF HEAT STRUCTURE PASSIVE
INTACT LOOP SECONDARY (Cont'd)							
IL SG Vessel							
o IL SG Downcomer	310	4	0	3	0	0	0
o IL DC to Boiler	315	0	0	1	0	0	0
o IL SG Boiler	330	5	0	4	0	0	0
o IL SG Separator	350	1	0	3	0	0	0
o IL SG Steam	360	1	0	0	0	0	0
IL SG STEAM LINE AND SAFETY VALVES							
o IL Steam Line	365	0	0	0	1	0	0
o IL Steam Sink	370	0	1	0	0	0	0
o IL Safety Valve	375	0	0	0	1	0	0
o IL S.V. Steam Sink	380	0	1	0	0	0	0
BROKEN LOOP (BL) PRIMARY							
BL HOT LEGS (HLs)							
o BL HL # 1	410	1	0	1	0	0	1
o BL HLs # 2&3	414	2	0	1	0	0	2
o BL HL To SG	417	0	0	1	0	0	0
BL STEAM GENERATOR (SG)							
o BL SG U-Tubes & Inlet/Outlet Plena	420	10	0	9	0	0	8
BL CROSS-OVER LEG (XLG)							
o SG to RCP Inlet	423	0	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 ACTIVE	NUMBER OF VOLUMES TMDPVOL	NUMBER OF JUNCTIONS ACTIVE	NUMBER OF JUNCTIONS TMDPJUN	NUMBER OF HEAT STRUCTURE ACTIVE	NUMBER OF HEAT STRUCTURE PASSIVE
BROKEN LOOP (BL) PRIMARY (Cont'd)							
o BL XLC	426	3	0	2	0	0	3
BL REACTOR COOLANT PUMP (RCP)							
BL RCP	430	1	0	2	0	0	0
BL COLD LEG (CL)							
o BL CL # 1 RCP Side	436	1	0	1	0	0	1
o BL CL # 2 Middle Volume	440	1	0	1	0	0	1
o BL CL # 3 RV Side	441	[illegible]	0	0	0	0	1
BROKEN LOOP SECONDARY							
BL MAIN AND AUXILIARY FEEDWATER (MFW & AFW)							
o BL MFW Source	502	0	1	0	0	0	0
o BL MFW Flow	503	0	0	0	1	0	0
o BL AFW Source	504	0	1	0	0	0	0
o BL AFW Flow	505	0	0	0	1	0	0
BL SG VESSEL							
o BL SG DC	510	4	0	3	0	0	0
o SG DC to Boiler	515	0	0	1	0	0	0
o BL SG Boiler	530	5	0	4	0	0	0
o BL SG Separator	550	1	0	3	0	0	0
o BL SG Steam Dome	560	1	0	0	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 STRUCTURE	
		ACTIVE	TMDPVOL	ACTIVE	TMDPJUN	ACTIVE	PASSIVE
BROKEN LOOP SECONDARY (Cont'd)							
BL SG STEAM LINE AND SAFETY VALVES							
o BL Steam Flow	565	0	0	0	1	0	0
o BL Steam Sink	570	0	1	0	0	0	0
o BL Safety Valve	575	0	0	0	1	0	0
o BL S.V. Steam Sink	580	0	1	0	0	0	0
PRESSURIZER (PRZR)							
o PRZR Surge-Line Flow	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 IL ACCUMs	700	2	0	1	0	0	0
o IL ACCUM Surge Line & Flow	703	0	0	1	0	0	0
	704	1	0	0	0	0	0
	705	0	0	0	1	0	0
o BL ACCUMs	720	2	0	1	0	0	0
o BL ACCUM Surge Line & Flow	723	0	0	[illegible]	0	0	0
	724	1	0	0	0	0	0
	725	0	0	0	1	0	0
EMERGENCY CORE COOLING SYSTEM (ECCS)							
ECCS							
o IL RHR Source	748	0	1	0	0	0	0
o IL HHSI Source	749	0	1	0	0	0	0
o IL CCP Source	750	0	1	0	0	0	0

TABLE 2.1 (Cont'd)

CPSES-1 ANF-RELAP NSSS NODALIZATION SUMMARY

REGION OR SYSTEM	NODING DIAGRAM NUMBER	NUMBER OF VOLUMES ACTIVE	NUMBER OF VOLUMES TMDPVOL	NUMBER OF JUNCTIONS ACTIVE	NUMBER OF JUNCTIONS TMDPJUN	NUMBER OF HEAT STRUCTURE ACTIVE	NUMBER OF HEAT STRUCTURE PASSIVE
			ECCS (Cont'd)				
o IL CCP Flow	751	0	0	0	1	0	0
o IL HHSI Flow	752	0	0	0	1	0	0
o IL RHR Flow	753	0	0	0	1	0	0
o BL RHR Source	768	0	1	0	0	0	0
o BL HHSI Source	769	0	1	0	0	0	0
o BL CCP Source	770	0	1	0	0	0	0
o BL CCP Flow	771	0	0	0	1	0	0
o BL HHSI Flow	772	0	0	0	1	0	0
o BL RHR Flow	773	0	0	0	1	0	0
			BREAK & CONTAINMENT				
BREAK							
o Break Junction Valve	805	0	0	0	1	0	0
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)	VERT. ANGLE (DEGREES)	ELEV. CHANGE (FT)	ROUGH. (FT)	HYD. DIAM. (FT)	FLAGS
100	SNGLVOL	26.65	5.85	155.92	-90.0	-5.85	0.00	1.53	100
104	BRANCH	20.74	2.29	47.49	-90.0	-2.29	0.00	0.98	100
108-1	ANNULUS	34.07	3.93	133.89	-90.0	-3.93	0.00	1.57	100
108-2	ANNULUS	33.24	4.00	132.96	-90.0	-4.00	0.00	1.52	100
108-3	ANNULUS	33.24	4.00	132.96	-90.0	-4.00	0.00	1.52	100
108-4	ANNULUS	33.24	4.00	132.96	-90.0	-4.00	0.00	1.52	100
108-5	ANNULUS	33.63	4.05	136.21	-90.0	-4.05	0.00	1.42	100
116	SNGLVOL	47.91	2.96	141.80	90.0	2.96	0.00	7.81	100
120	BRANCH	114.12	2.96	337.81	90.0	2.96	0.00	12.05	100
124	BRANCH	96.86	4.23	409.73	90.0	4.23	0.00	11.11	100
128-1	ANNULUS	25.08	4.00	100.31	90.0	4.00	0.00	0.008	100
128-2	ANNULUS	25.08	4.00	100.31	90.0	4.00	0.00	0.008	100
128-3	ANNULUS	25.08	4.00	100.31	90.0	4.00	0.00	0.008	100
130	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
132	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
134	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
136	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
138	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
140	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
142	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
144	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000

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)	VERT. ANGLE (DEGREES)	ELEV. CHANGE (FT)	ROUGH. (FT)	HYD. DIAM. (FT)	FLAGS
146	BRANCH	15.32	1.33	20.38	90.0	1.33	5.E-6	0.0365	000
150	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
152	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
154	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
156	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
158	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
160	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
162	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
164	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
165	BRANCH	35.75	1.33	47.55	90.0	1.33	5.E-6	0.0365	000
166	BRANCH	84.84	1.28	108.60	90.0	1.28	0.00	0.0365	100
170	SNGLVOL	16.84	13.29	223.78	90.0	1.28	0.00	0.25	100
173	BRANCH	84.84	2.405	204.04	90.0	2.405	0.00	10.39	100
174	BRANCH	84.84	2.42	205.31	90.0	2.42	0.00	10.39	100
178	SNGLVOL	84.84	2.55	216.34	90.0	2.55	0.00	10.39	100
180	SNGLVOL	84.84	2.15	182.41	90.0	2.15	0.00	10.20	100
181	BRANCH	125.48	2.37	297.38	90.0	2.37	0.00	11.04	100
182	BRANCH	80.26	4.53	363.60	90.0	6.90	0.00	11.04	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)	VERT. ANGLE (DEGREES)	ELEV. CHANGE (FT)	ROUGH. (FT)	HYD. DIAM. (FT)	FLAGS
210	BRANCH	13.74	5.20	71.46	0.0	0.00	1.5E-4	2.42	100
214-1	PIPE	13.77	8.88	122.31	0.0	0.00	1.5E-4	2.42	100
214-2	PIPE	14.32	7.59	104.52	0.0	0.00	1.5E-4	2.42	100
220-1	PIPE	63.52	7.91	502.41	90.0	7.91	1.5E-4	5.19	100
220-2	PIPE	31.38	9.06	284.31	90.0	9.06	5.0E-6	0.0553	100
220-3	PIPE	31.38	7.25	227.52	90.0	7.25	5.0E-6	0.0553	100
220-4	PIPE	31.38	7.25	227.52	90.0	7.25	5.0E-6	0.0553	100
220-5	PIPE	31.38	4.44	139.32	90.0	4.44	5.0E-6	0.0553	100
220-6	PIPE	31.38	4.44	139.32	-90.0	-4.44	5.0E-6	0.0553	100
220-7	PIPE	31.38	7.25	227.52	-90.0	-7.25	5.0E-6	0.0553	100
220-8	PIPE	31.38	7.25	227.52	-90.0	-7.25	5.0E-6	0.0553	100
220-9	PIPE	31.38	9.06	284.31	-90.0	-9.06	5.0E-6	0.0553	100
220-10	PIPE	63.52	7.91	502.41	-90.0	-7.91	1.5E-4	5.19	100
226-1	PIPE	15.72	10.32	162.23	-90.0	-10.32	1.5E-4	2.58	100
226-2	PIPE	15.72	10.50	165.06	0.0	0.00	1.5E-4	2.58	100
226-3	PIPE	15.72	4.51	70.90	90.0	4.51	1.5E-4	2.58	100
230	PUMP	32.04	7.36	235.80	90.0	5.81			00
236	BRANCH	12.36	7.14	88.26	0.0	0.00	1.5E-4	2.29	100
240-1	PIPE	12.36	15.76	194.88	0.0	0.00	1.5E-4	2.29	100
240-2	PIPE	12.36	2.20	27.192	0.0	0.00	1.5E-4	2.29	100
302	TMDPVOL	300.00	10.00	3000.00	0.0	0.00	0.00	0.00	100
304	TMDPVOL	300.00	10.00	3000.00	0.0	0.00	0.00	0.00	100
310-1	PIPE	584.53	7.85	4588.59	-90.0	-7.85	1.5E-4	15.72	100
310-2	PIPE	17.01	7.25	123.33	-90.0	-7.25	1.5E-4	0.34	100
310-3	PIPE	17.01	7.25	123.33	-90.0	-7.25	1.5E-4	0.34	100
310-4	PIPE	17.01	9.06	154.11	-90.0	-9.06	1.5E-4	0.34	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)	VERT. ANGLE (DEGREES)	ELEV. CHANGE (FT)	ROUGH. (FT)	HYD. DIAM. (FT)	FLAGS
330-1	PIPE	162.97	9.06	1476.54	90.0	9.06	1.5E-4	0.0972	100
330-2	PIPE	166.43	7.25	1206.63	90.0	7.25	1.5E-4	0.0972	100
330-3	PIPE	166.43	7.25	1206.63	90.0	7.25	1.5E-4	0.0972	100
330-4	PIPE	155.97	4.44	692.52	90.0	4.44	1.5E-4	0.0972	100
330-5	PIPE	261.99	3.41	893.37	90.0	3.41	1.5E-4	10.54	100
350	SEPARATR	143.30	23.74	3401.97	90.0	23.74	0.00	7.80	100
360	SNGLVOL	383.5[illegible]	9.73	3732.15	90.0	9.73	1.5E-4	12.76	100
370	TMDPVOL	300.00	10.00	3000.00	0.0	0.00	0.00	0.00	100
380	TMDPVOL	300.00	10.00	3000.00	0.0	0.00	0.00	0.00	100
410	BRANCH	4.58	5.20	23.82	0.0	0.00	1.5E-4	2.42	100
414-1	PIPE	4.59	8.88	40.77	0.0	0.00	1.5E-4	2.42	100
414-2	PIPE	4.59	7.59	34.84	0.0	0.00	1.5E-4	2.42	100
420-1	PIPE	21.17	7.91	167.47	90.0	7.91	1.5E-4	5.19	100
420-2	PIPE	10.46	9.06	94.77	90.0	9.06	5.0E-6	0.0553	100
420-3	PIPE	10.46	7.25	75.84	90.0	7.25	5.0E-6	0.0553	100
420-4	PIPE	10.46	7.25	75.84	90.0	7.25	5.0E-6	0.0553	100
420-5	PIPE	10.46	4.44	46.44	90.0	4.44	5.0E-6	0.0553	100
420-6	PIPE	10.46	4.44	46.44	-90.0	-4.44	5.0E-6	0.0553	100
420-7	PIPE	10.46	7.25	75.84	-90.0	-7.25	5.0E-6	0.0553	100
420-8	PIPE	10.46	7.25	75.84	-90.0	-7.25	5.0E-6	0.0553	100
420-9	PIPE	10.46	9.06	94.77	-90.0	-9.06	5.0E-6	0.0553	100
420-10	PIPE	21.17	7.91	167.47	-90.0	-7.91	1.5E-4	5.19	100
426-1	PIPE	5.24	10.32	54.08	-90.0	-10.32	1.5E-4	2.58	100
426-2	PIPE	5.24	10.50	55.02	0.0	0.00	1.5E-4	2.58	100
426-3	PIPE	5.24	4.51	23.63	90.0	4.51	1.5E-4	2.58	100
430	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)	VERT. ANGLE (DEGREES)	ELEV. CHANGE (FT)	ROUGH. (FT)	HYD. DIAM. (FT)	FLAGS
436	BRANCH	4.12	7.14	29.42	0.0	0.00	1.5E-4	2.29	100
440	BRANCH	4.12	15.76	64.96	0.0	0.00	1.5E-4	2.29	100
441	BRANCH	4.12	2.20	9.064	0.0	0.00	1.5E-4	2.29	100
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
530-1	PIPE	54.32	9.06	492.18	90.0	9.06	1.5E-4	0.0972	100
530-2	PIPE	55.48	7.25	402.21	90.0	7.25	1.5E-4	0.0972	100
530-3	PIPE	55.48	7.25	402.21	90.0	7.25	1.5E-4	0.0972	100
530-4	PIPE	51.99	4.44	230.84	90.0	4.44	1.5E-4	0.0972	100
530-5	PIPE	87.33	3.41	297.79	90.0	3.41	1.5E-4	10.54	100
550	SEPARATR	47.77	23.74	1133.99	90.0	23.74	0.00	7.80	100
560	SNGLVOL	127.86	9.73	1244.05	90.0	9.73	1.5E-4	12.76	100
570	TMDPVOL	100.00	10.00	1000.00	0.0	0.00	0.00	0.00	100
580	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	100
620-1	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	100
620-2	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	100
620-3	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	100
620-4	PIPE	36.58	7.627	279.00	90.0	7.627	0.00	6.82	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)	VERT. ANGLE (DEGREES)	ELEV. CHANGE (FT)	ROUGH. (FT)	HYD. DIAM. (FT)	FLAGS
620-5	PIPE	36.58	9.704	354.97	90.0	9.704	0.00	6.82	100
620-6	PIPE	36.58	9.704	354.97	90.0	9.704	0.00	6.82	100
700-1	PIPE	226.90	1.50	340.35	[illegible]	1.50	0.00	9.81	100
700-2	PIPE	226.90	16.35	3709.65		16.35	0.00	9.81	100
704	SNGLVOL	1.254	81.12	101.73	-90.0	-10.17	0.00	0.73	100
720-1	PIPE	75.63	1.50	113.45	90.0	1.50	0.00	9.81	100
720-2	PIPE	75.63	16.35	1236.55	90.0	16.35	0.00	9.81	100
724	SNGLVOL	0.418	81.12	33.91	-90.0	-10.17	0.00	0.73	100
748	TMDPVOL	3.00	10.00	30.00	90.0	10.00	0.00	0.00	001
749	TMDPVOL	3.00	10.00	30.00	90.0	10.00	0.00	0.00	001
750	TMDPVOL	3.00	10.00	30.00	90.0	10.00	0.00	0.00	001
768	TMDPVOL	1.00	10.00	10.00	90.0	10.00	0.00	0.00	001
769	TMDPVOL	1.00	10.00	10.00	90.0	10.00	0.00	0.00	001
770	TMDPVOL	1.00	10.00	10.00	90.0	10.00	0.00	0.00	001
810	TMDPVOL	100.00	1.00	100.00	0.0	0.00	0.00	0.00	100

TABLE 2.3

DENSITY REACTIVITY TABLE

DENSITY lbm/ft^3	REACTIVITY ($)
0.62	-54.65
6.24	-32.46
12.49	-17.63
18.73	-9.38
24.97	-4.56
31.21	-1.82
37.46	-0.47
43.87	0.00
46.82	0.15
49.94	0.30
62.43	0.60

TABLE 2.4

DOPPLER REACTIVITY TABLE

TEMPERATURE (F)	REACTIVITY ($)
200.0	1.691
400.0	1.283
600.0	0.919
800.0	0.589
1000.0	0.284
1200.0	-0.000
1400.0	-0.267
1600.0	-0.519
1800.0	-0.759
2000.0	-0.988
2200.0	-1.207
2400.0	-1.417
2600.0	-1.620
2800.0	-1.816
3000.0	-2.006
3200.0	-2.189
3400.0	-2.367
3600.0	-2.541
3800.0	-2.709
4000.0	-2.874

TABLE 2.5

SCRAM REACTIVITY TABLE

TIME (SEC)	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 PER EACH LOOP VS. PRESSURE (98%)

RCS PRESSURE (psig)	CCP (lbm/sec)	HPSI (lbm/sec)	RHR (lbm/sec)	TOTAL (lbm/sec)
0	13.42	19.73	128.54	161.29
100	12.86	19.01	16.95	48.82
120	12.75	18.86	0.0	31.61
200	12.30	18.26	0.0	30.56
300	11.72	17.50	0.0	29.22
400	11.13	16.69	0.0	27.82
500	10.54	15.84	0.0	26.38
600	9.92	14.92	0.0	24.84
700	9.29	13.98	0.0	23.27
800	8.65	12.96	0.0	21.61
900	7.98	1[illegible].82	0.0	19.80
1000	7.30	10.58	0.0	17.88
1200	5.88	7.41	0.0	13.29
1300	5.13	5.19	0.0	10.32
1400	4.35	1.40	0.0	5.75
1500	3.55	0.0	0.0	3.55
1600	2.71	0.0	0.0	2.71
1800	0.84	0.0	0.0	0.84
2000	0.0	0.0	0.0	0.0
2100	0.0	0.0	0.0	0.0

TABLE 2.7

TRIPS AND DELAYS

ACTION	TRIPS AND DELAYS (sec)
Lo-Przr Pressure signal	RCS @ 1860 psia
Reactor trip	2 sec after Lo-Przr signal
"S" signal	Pressurizer @ 1715 psia
Reactor Coolant Pumps trip	Reactor Trip
Main Steam Isolated	0.5 sec after Reactor Trip
Main Feedwater Isolated	2 sec after "S" signal
Auxiliary Feedwater Injects	158 sec after "S" signal
SI Actuation Signal	2 sec after "S" signal
Charging Pump Injects	17 sec after "S" signal
HPSI Pump Injects	22 sec after "S" signal
Accumulators Inject	RCS @ 648 psia

TABLE 2.8

FUEL ASSEMBLY/ROD DATA

PARAMETER	VALUE
Outer Diameter of Fuel Rod	0.374 in
Active Fuel Height	144.0 in
No. of Fuel Assemblies	193
No. of Fuel Rods/Assy	264
No. of Guide Thimbles/Assy	24
No. of Instr. Tubes/Assy	1
Cladding Thickness	0.0225 in
Diametral Gap	0.0065 in
Outer Dia. of Guide Thimble	0.482 in

TABLE 2.9

STEAM GENERATOR SAFETY VALVES FLOW RATES

Secondary Pressure (psia)	SV Flow Rate (lbm/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

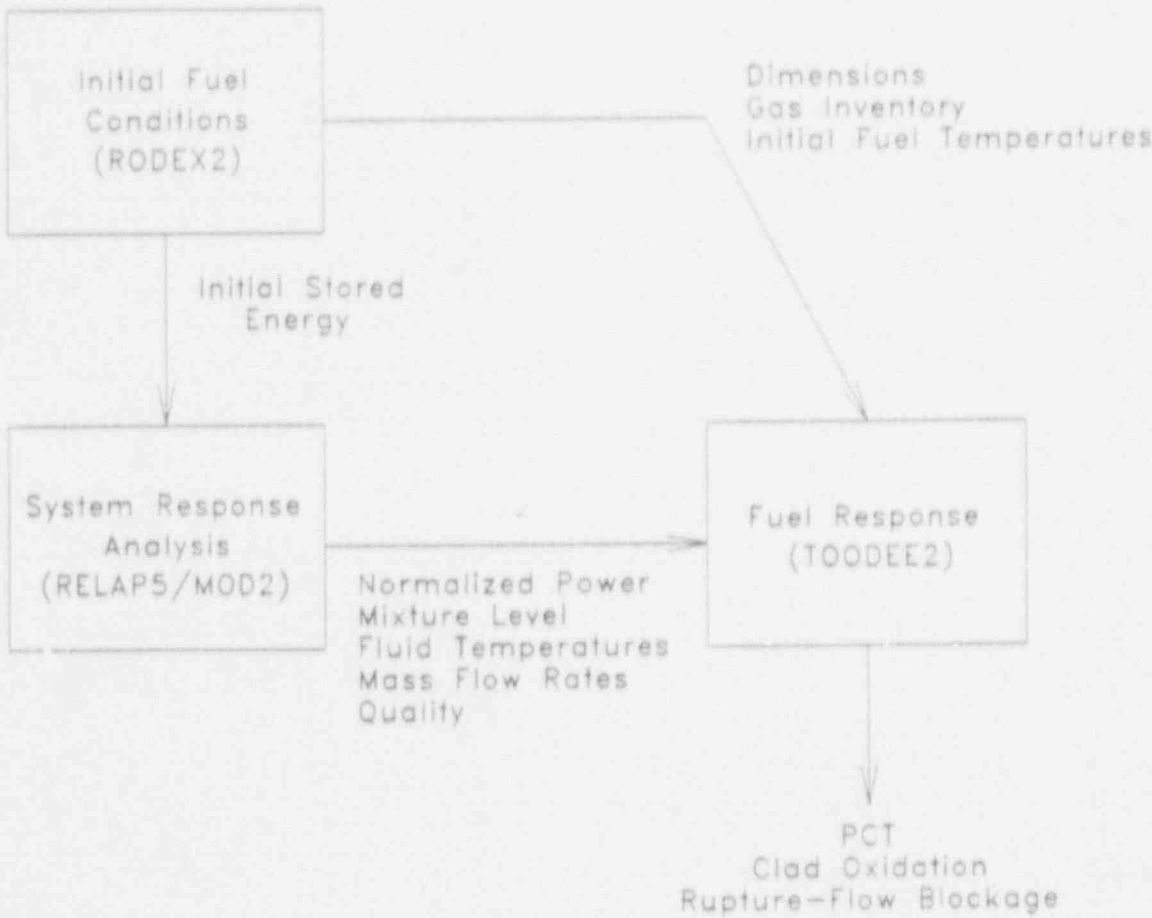


Figure 2.1 Schematic of ENC Small Break Model

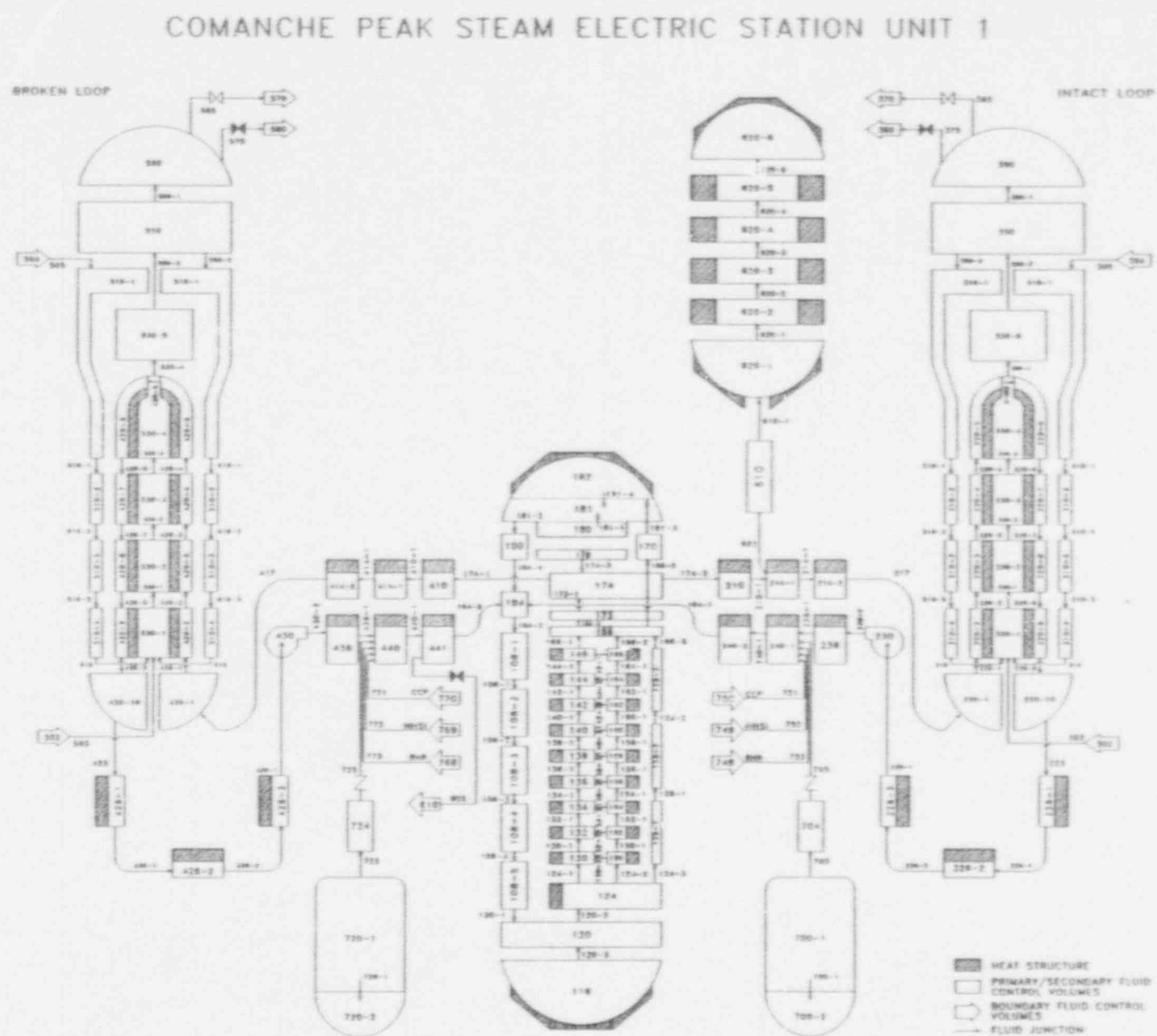
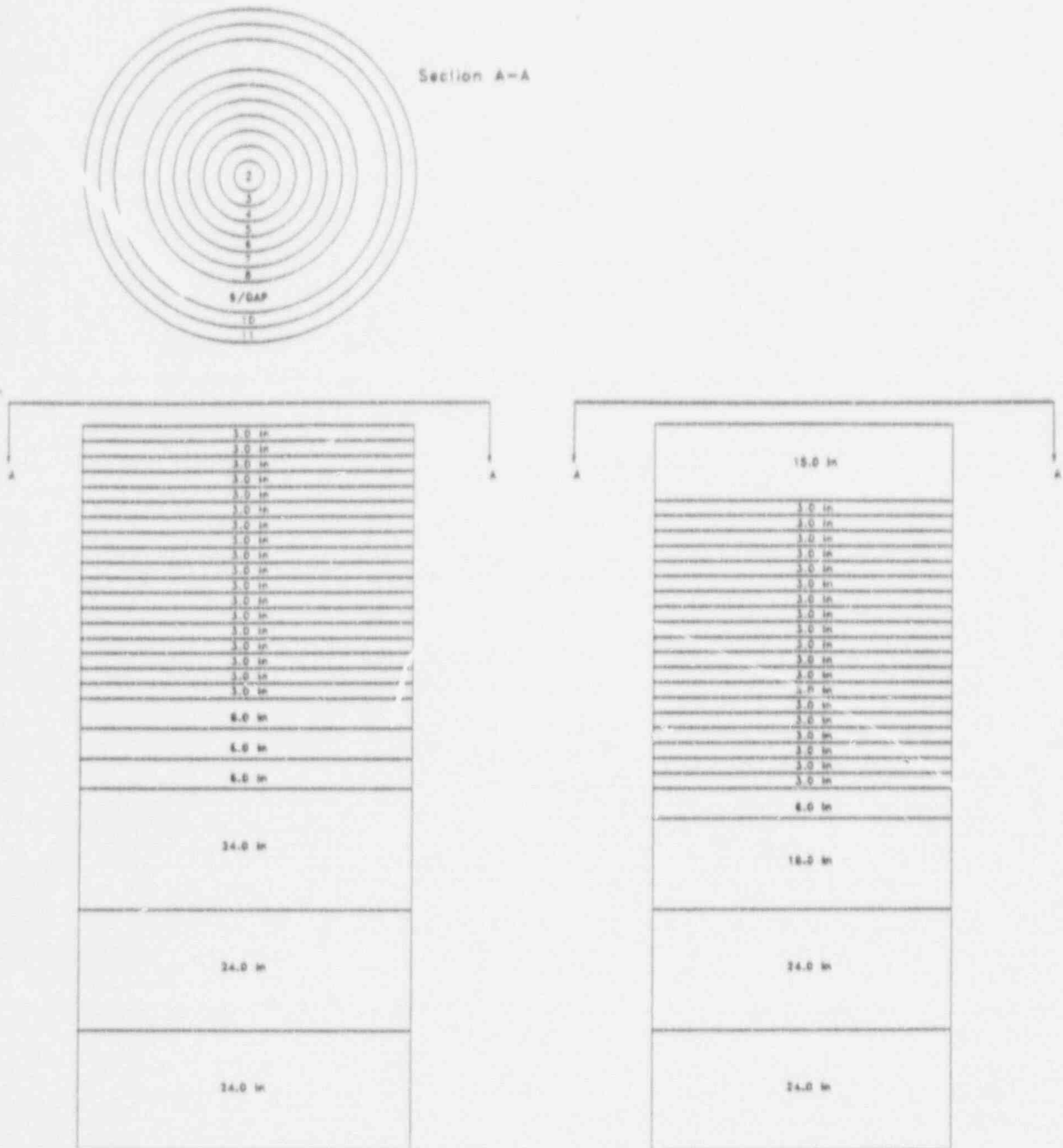


Figure 2.2 ANF-RELAP SBLOCA Nodalization Diagram



A) Top Skewed

B) Chopped Cosine

Figure 2.3 TOODEE2 Nodalization Diagrams

CHAPTER 3

BASE CASE ANALYSIS AND SENSITIVITY STUDIES

Regulations pertaining to small break loss-of-coolant accident analyses 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 Chapter 5 and Appendix A of Reference 2.1. The present work constitutes an application of an approved Evaluation Methodology, 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 are not repeated here.

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 and the results are reported in Section 3.2.

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." A power shape study has been conducted and the results are reported in Section 3.3, in compliance with this requirement.

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 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 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 meaningful in a consistent manner, 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 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.

Two additional conservatism are incorporated into all of the calculations in this work.

The first is in the initial power level, which is taken to be 3636 MWt. This power level includes both a 1.02 multiplier

to account for calorimetric error and an increase of 4.5% above the licensed power level of 3411 MWt, representing a margin potentially available.

The second is that five percent of the steam generator tubes are assumed plugged. This assumption is made to support the potential need for operation under these 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 6.0 inch diameter break in the discharge line of the Reactor Coolant Pump. The axial power shape used for this base case is that determined by the vendor in the FSAR analysis (Reference 3.1) as the most limiting and is shown as power shape 1 in Figure 3.1. The fuel rod exposure which maximizes stored energy is calculated by RODEX2 and occurs at 678.6 hours. 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.

Figure 3.2 shows the core power dropping off quickly at first due primarily to the combined effects on reactivity of moderator voiding and control rod insertion and then tapering off according to decay power.

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 are characterized by their primary system depressurization rates and have the following characteristics:

Period I - Depressurization:

The accident period marked I in Figure 3.3 corresponds to the rapid depressurization which follows break opening. From the secondary side standpoint, period I corresponds to a 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 not operational, as indicated in Figure 3.4.

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 is evidenced in the rapid increase in the void fractions at mid core elevation for both the central (Figure 3.5) and average (Figure 3.6) core regions, shortly followed by a similar behavior in the corresponding lower nodes. The effect of this steam production is to reduce the depressurization rate of the primary system, resulting in the nearly flat system pressure which characterizes period II, as seen in Figure 3.3. Period II from the secondary side point of view begins when the pressure stabilizes near the safety valve's set point. During period II the steam generator safety valves open in order to discharge excess steam produced due to (a) the absence of feedwater between the main feedwater trip time and the time the auxiliary feedwater reaches the steam generators in combination with (b) the steam dump and bypass system unavailability. During period II there is still some heat transfer between the primary and the secondary, but this ends with period II.

Period III - Heat up:

The end of period II and beginning of period III is brought about by the end of significant steam production in the core, i.e. dryout. Thus, the end of period II and beginning of period III can be inferred from the time at which the downcomer and lower plenum levels begin to fall, as shown in Figures 3.7 and 3.8. The dropping of these levels imply the

core is dry and therefore steam production has essentially ceased. Therefore, period III is characterized by an increased depressurization rate which is due to the lack of steam generation to compensate for the energy discharge through the break. Conversely, from the secondary side point of view, period III is characterized by slow depressurization, as auxiliary feedwater injection removes sensible heat and raises steam generator inventories (Figure 3.9). At this time however, the primary can no longer benefit from the cooler secondary, since the primary pressure has dropped below the secondary pressure at the beginning of the period and the secondary is lost as a heat sink. It is in period III that the fuel experiences its temperature excursion as shown in Figure 3.10. The clad temperature of Figure 3.10 starts to rise slightly before the beginning of period III because its axial location dries out slightly before steam production ceases throughout the core. With the exception of a small perturbation in the broken loop, due to injection from the SI pumps, the loop seals are clear at the beginning of period III and remain clear throughout the accident, as evidenced by the steam generator tubes' water levels shown in Figure 3.11. Therefore, loop seal plugging has a small effect on the peak clad temperature for the base case analysis (6 inch break). The effect of loop seal plugging is more pronounced for the 4 inch break and is discussed in that section.

Period IV - Recovery:

Period III ends when the system pressure reaches the accumulator injection pressure. At that time, as shown in Figure 3.12, the massive injection of water reinstates steam production in the core causing the depressurization rate to level off again and marking the onset of period IV. Accumulator injection causes key water levels to rise (Figures 3.7, 3.8 and 3.13) and arrests the primary inventory depletion (Figure 3.14). The clad temperatures begin to turn around at accumulator injection time. Figure 3.10 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 node 7 quenching first and node 9 last.

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 ANF EXEM PWR Small Break Model, the break size is the first sensitivity issue addressed, holding constant the axial power shape. The rationale for addressing break size first is that system thermal-hydraulic behavior is largely affected by break size and less dependent on power shape and consequently the break size is a first order effect, while the power shape is second order.

The break spectrum study is conducted using the power shape determined by the vendor in the FSAR analysis (Reference 3.1) as the most limiting and at the burnup yielding the highest stored energy. The fuel rod exposure which maximizes stored energy is calculated by RODEX2 and occurs at 678.6 hours. Fuel parameters used in this base case are consistent with this exposure. In conclusion, the same power shape and exposure used for the base case analysis are used in the break spectrum study.

Three break sizes are examined, namely: 6 inch (base case), 4 inch and 8 inch, the last two sizes resulting in lower peak clad temperatures than the base case.

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 a 6 inch break located in the reactor coolant pump discharge. The power shape sensitivity study will be performed using the limiting break size.

<u>6 inch Break</u>

This is the base case calculation described in Section 3.1. The PCT is calculated to be 1812 °F in node 20 at 10.875 ft 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 3.17.

<u>4 inch Break</u>

The results of this calculation are similar to those of the base case (Section 3.1) so that only the key differences are pointed out here. The PCT is calculated to be 1585 °F in

node 20 in this case, 10.875 ft 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 3.33.

Figure 3.18 shows the power behavior subject to the same mechanisms which control the base case calculation.

Figure 3.19 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 3.19 corresponds to the depressurization of the primary system due to the break while the secondary pressure rises to the safety valve's set point. There are no major distinctions between system behavior during this period between the 4 inch break and the 6 inch base case except, of course, that the depressurization rate is lower.

Period II - Voiding:

As in the base case, 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 is evidenced in the rapid increase in the void fractions at mid core elevation for both the central (Figure 3.21) and average (Figure 3.22) core regions, shortly followed by a similar behavior in the corresponding lower nodes. The 6 inch base case discussion for this period applies to the 4 inch break as well.

Period III - Heat up:

As in the 6 inch discussion, the end of period II and beginning of period III is also inferred when the downcomer and lower plenum levels begin to fall, as shown in Figures 3.23 and 3.24. The dropping of these levels imply the core is dry, steam production has essentially ceased and the primary system pressure begins to drop significantly again. It is also in period III that the fuel experiences its temperature excursion as shown in Figure 3.26. As in the base case, the clad temperature of Figure 3.26 starts to rise slightly before the beginning of period III because its axial location dries out slightly before steam production ceases throughout the core. However, unlike the base case, in the 4 inch break there is an intermediate quenching of this clad location, which is driven by redistribution of fluid in the core induced by clearing of the loop seals (Figure 3.27). The quench is also seen as a small spike in the pressure curve at the end of period II. For the 4 inch break the loop

seals are also clear at the beginning of period III, but when the pressure drop associated with the end of significant steam production comes about and the high head safety injection pumps begin to discharge, some of that ECC water flows back accumulating in the broken loop seal. This accumulation can be inferred from the steam generator tube's water levels, shown in Figure 3.27. This effect is significant for the 4 inch break because it reduces the rate of depressurization of the system, thereby reducing pumped injection and delaying accumulator injection. Another contributor to the reduction of the primary depressurization rate is the bottled up secondary. In fact the secondary is being cooled by the primary during period III, as evidenced by the pinch point in the pressure histories (Figure 3.19) around 750 seconds.

Period IV - Recovery:

As in the base case, period III ends when the system pressure reaches the accumulator injection pressure. At that time, as shown in Figure 3.28, the massive injection of water reinstates steam production in the core causing the depressurization rate to level off again and marking the onset of period IV. Accumulator injection causes key water levels to rise (Figures 3.23, 3.24 and 3.29) and arrests the primary inventory depletion (Figure 3.30). The clad temperature begins to turn around at accumulator injection

time. Figure 3.26 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 node 7 quenching first and node 9 last.

Conclusion:

The same conclusion drawn for the base case applies to the 4 inch calculation. The pumped injection flows (Figure 3.31) cannot keep up with the break flow (Figure 3.32) during periods I and II. Still, the accumulator injection pressure is reached well before the clad temperatures are too high and the temperatures are effectively turned around. In compliance with ANF methodology, the clad temperatures are recalculated using the more conservative TOODEE2 code and are shown in Figure 3.33.

8 inch Break

This calculation is similar to the base case calculation as evidenced in Figures 3.34 through 3.48. The difference in results is primarily due to the faster depressurization associated with the 8 inch break, the earlier accumulator injection, higher ECC flow rates and lower PCT. The PCT is calculated to be 1750 °F in node 22 in this case, 11.625 ft 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 3.49.

3.2.2 AXIAL POWER SHAPE

The axial power shape study is performed to support the Technical Specification linear heat generation rate (LHGR) limit as a function of core height. This study is performed for the most limiting break determined in the break spectrum study (6 inch, Section 3.2.1) and at the burnup yielding the highest stored energy. The maximum stored energy occurs at 678.6 hours when maximum fuel densification occurs, resulting in the maximum gap width.

The population of axial power shapes is developed through the power distribution control analysis described in Reference 3.3. For that analysis a prescribed series of load follow cases are modeled which provide the maximum variation in axial shapes achieved within the allowed operating conditions.

The selection of the axial power shapes to be examined is a two-step process. The first step is selecting the power

shapes which are closest to the Technical Specifications limit curve for each elevation. The second step is selecting power shapes which have the highest integral power up to the PCT elevation. The selected shapes are subsequently renormalized so that the peak LHGR matches the Technical Specification (Reference 3.4) limit at that location. Analyses are performed for the chopped cosine (Shape 2, Figure 3.1) and two additional shapes, one of them most limiting according to the vendor's FSAR analyses (Shape 1, Figure 3.1) and one (Shape 3, Figure 3.1) developed using the method described above. These are the most likely candidates to yield the highest PCT according to the criterion just described. All these power shapes are shown in Figure 3.1.

The sequence of events for the axial power shape study is summarized in Table 3.6.

The conclusion to be drawn from the axial power shape study is that power shape 3, shown in Figure 3.1 is the most limiting. This result will be used in all other studies in the future.

POWER SHAPE 1 (6 inch, BOL)

This is the base case calculation described in Section 3.1. The PCT for this calculation is 1812 °F and occurs at node

21, 11.25 ft above the bottom of the core. The TOODEE2 calculation for the surface clad temperature at the node where the PCT occurs is shown in Figure 3.17.

CHOPPED COSINE (6 inch, BOL)

The PCT for this calculation is 1578 °F and occurs at node 20, 10.875 ft above the bottom of the core. The system performance throughout the accident is nearly identical to that described in Section 3.1 for the base case so that discussion need not be repeated here. The TOODEE2 calculation for the surface clad temperature at the node where the PCT occurs is shown in Figure 3.50.

POWER SHAPE 3 (6 inch, BOL)

The PCT for this calculation is 1837 °F and occurs at node 21, 11.25 ft above the bottom of the core. The system performance throughout the accident is nearly identical to that described in Section 3.1 for the base case so that discussion need not be repeated here. The TOODEE2 calculation for the surface clad temperature at the node where the PCT occurs is shown in Figure 3.51. This is the most limiting combination of break size and power shape.

TABLE 3.1

SUMMARY OF CPSES-1 SMALL BREAK LOCA ACCIDENT

ASSUMPTIONS FOR BASE CASE AND SENSITIVITY STUDIES

1. The initial power is 3636 MWt. This is 4.5% plus 2% for calorimetric error above the licensed power level.

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 on an "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 [illegible].

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 turbine- and one motor-driven AFW pumps are credited, but their injections are 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
Core Power Power Upgrade Multiplier Power Calorimetric Uncertainty Multiplier	3636 MWt 1.045 1.02
Power Shapes Analyzed	Chopped Cosine Shape 1 Fig. 3.1 Shape 3 Fig. 3.1
Peak Linear Power (includes 102% factor) Chopped Cosine (Fig. 3.1) Shape 1, Top Peaked at 9.25 ft (Fig. 3.1) Shape 3, Top Peaked at 10.75 ft (Fig. 3.1)	 13.16 KW/ft 12.59 KW/ft 12.36 KW/ft
Total Peaking Factor, F^T_Q Chopped Cosine (Fig. 3.1) Shape 1, Top Peaked at 9.25 ft (Fig. 3.1) Shape 3, Top Peaked at 10.75 ft (Fig. 3.1)	 2.32 2.22 2.18
Accumulator Water Volume per Tank	6119 gals
Accumulator Cover Gas Pressure	648 psia
Accumulator Water Temperature	90 °F
Safety Injection Pumped Flow	Table 2.6
Refueling Water Storage Tank Temperature	120 °F
Initial Loop Flow	9680 lbm/sec
Vessel Inlet Temperature	[illegible] °F
Vessel Outlet Temperature	627 °F
Reactor Coolant Pressure	2280 psia
Pressurizer Water Volume	1116 ft^3
Steam Pressure	894 psia
Auxiliary Feedwater Flow per SG	42.3 lb/sec
Steam Generator Tube Plugging Level	5%
Steam Generator Safety Valves Set Points And Flows	Table 2.9
Fuel Parameters	Cycle 1, 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	678.6 hours	
Fuel Rod Composition:		
	Average Core	Hot Central Core
Gram moles	0.02327	0.02327
Helium fraction	0.99495	0.99493
Argon fraction	0.00000	0.00000
Hydrogen fraction	0.00000	0.00000
Nitrogen fraction	0.00500	0.00500
Krypton fraction	0.00001	0.00001
Xenon fraction	0.00004	0.00006
Effective cold plenum length (in)	5.889	5.938
Dish volume (in^3)	0.1419	0.1411
Average fuel temp. at peak stored energy (°F)	1668	2104

TABLE 3.4

SEQUENCE OF EVENTS FOR BASE CASE SMALL BREAK LOCA
(6 inch, Power Shape 1, BOC)

EVENT	TIME (SECONDS)
1. Break opens (period I begins)	0.0
2. Reactor Trip Signal	3.2
3. RCP tripped	5.2
4. MSIV closed	5.7
5. "S" Signal	8.6
6. MFW isolated	10.6
7. Centrifugal charging pumps inject	25.6
8. Mid-elevation avg core boils (period II begins)	26.0
9. Safety injection pumps inject	30.6
10. Critical Heat Flux reached at PCT node	106.0
11. Onset of lower plenum depletion (period III begins)	163.0
12. Steam Generator tubes drained	165.0
13. Loop seals begin to fill	N/A
14. Auxiliary Feedwater injects	166.8
15. Accumulator injection (period IV begins)	329.9
16. Peak clad temperature reached	346.0
17. Peak clad temperature node quenched (period IV ends)	480.0
18. Calculation ends	600.0

TABLE 3.5

SEQUENCE OF EVENTS FOR BREAK SPECTRUM STUDY
(Power Shape 1, BOC)

EVENT	TIME (SECONDS)		
	6 inch	4 inch	8 inch
1. Break opens (period I begins)	0.0	0.0	0.0
2. Reactor Trip Signal	3.2	6.2	2.1
3. RCP tripped	5.2	8.2	4.1
4. MSIV closed	5.7	8.7	4.6
5. "S" Signal	8.6	13.6	4.9
6. MFW isolated	10.6	15.6	6.9
7. Centrifugal charging pumps inject	25.6	30.6	21.9
8. Mid-el avg core boils (period II begins)	26.0	47.0	7.8
9. Safety Injection pumps inject	30.6	35.6	26.9
10. Critical Heat Flux reached at PCT node	106.0	735.0	68.0
11. Lower plenum depletion begins (period III)	163.0	441.0	80.0
12. Steam Generator tubes drained	165.0	400.0	80.0
13. Loop seals begin to fill	N/A	400.0	N/A
14. Aux. Feedwater injects	166.8	171.8	163.1
15. Accumulator injects (period IV begins)	329.9	918.7	170.9
16. Peak clad temperature	346.0	932.0	186.0
17. Peak node quenched	480.0	1152.0	326.0
18. Calculation ends	600.0	1400.0	400.0

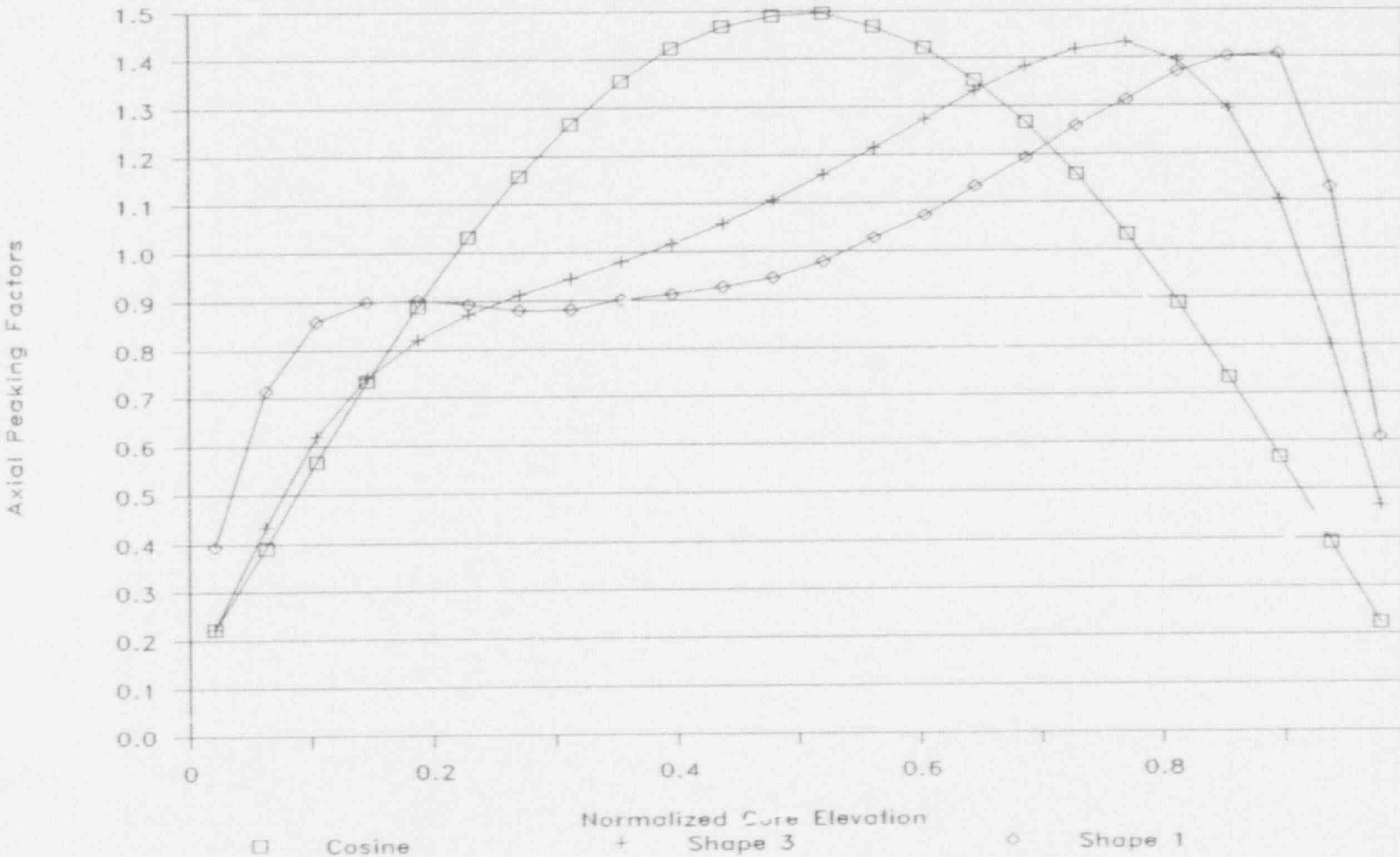
TABLE 3.6

SEQUENCE OF EVENTS FOR POWER SHAPE STUDY
(6 inch, BOC)

EVENT	TIME (SECONDS)		
	POWER SHAPE 1 (BASE CASE)	CHOPPED COSINE	POWER SHAPE 3
1. Break opens (period I begins)	0.0	0.0	0.0
2. Reactor Trip Signal	3.2	3.2	3.1
3. RCP tripped	5.2	5.2	5.1
4. MSIV closed	5.7	5.7	5.6
5. "S" Signal	8.6	9.1	8.5
6. MFW isolated	10.6	11.1	10.5
7. Centrifugal charging pumps inject	25.6	26.1	25.6
8. Mid elev. avg core boils (period II begins)	26.0	26.0	26.0
9. Safety injection pumps inject	30.6	31.1	30.6
10. Critical Heat Flux reached at PCT node	106.0	106.0	106.0
11. Lower plenum depletion begins (period III)	163.0	163.0	163.0
12. Steam Generator tubes drained	165.0	165.0	165.0
13. Loop seals begin to fill	N/A	N/A	N/A
14. Aux. Feedwater injects	166.8	166.7	167.3
15. Accumulator inject (period IV begins)	329.9	331.6	326.7
16. Peak clad temperature	346.0	356.0	346.0
17. Peak node quenched	480.0	480.0	480.0
18. Calculation ends	600.0	600.0	600.0

Figure 3.1 SBLOCA Axial Profiles

CPSES Unit 1 Cycle 1



COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

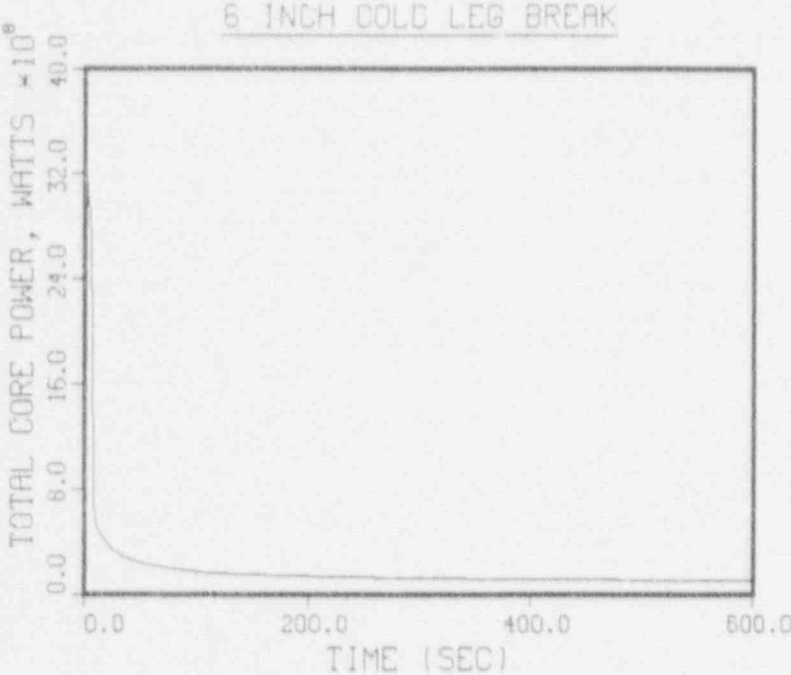


Figure 3.2 Core Power

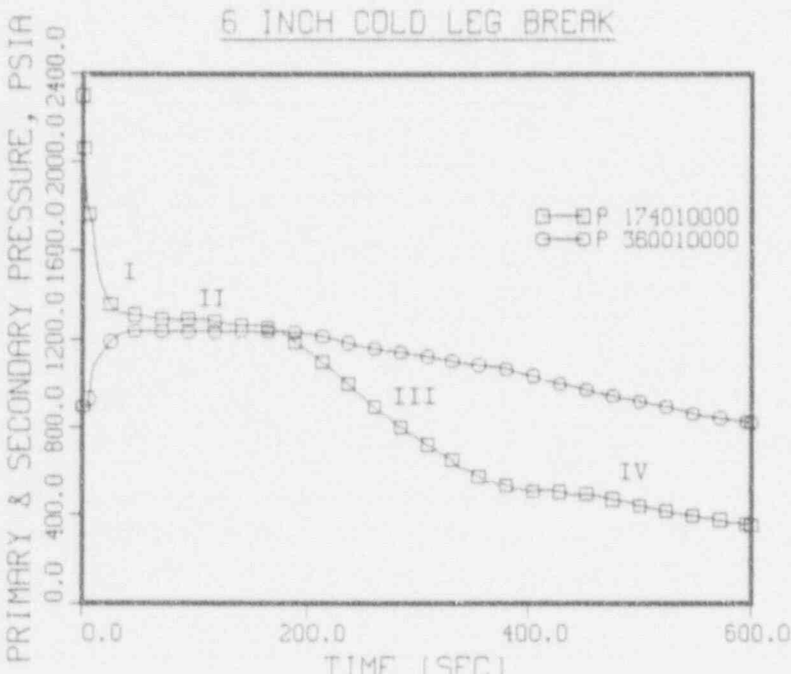


Figure 3.3 Primary and Secondary System Pressures

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

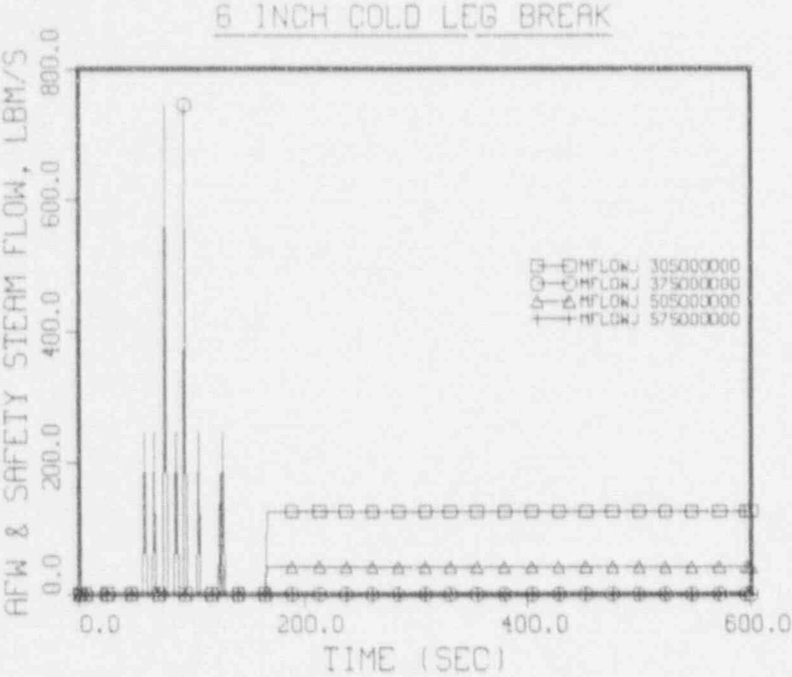


Figure 3.4 Intact and Broken Loop AFW and Steam Flows

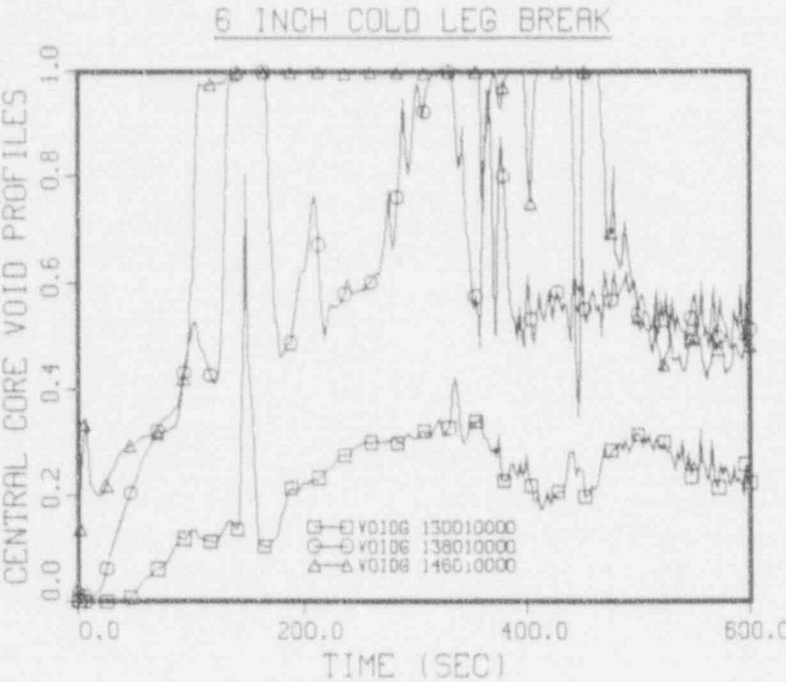


Figure 3.5 Central Core Region Void Fractions

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

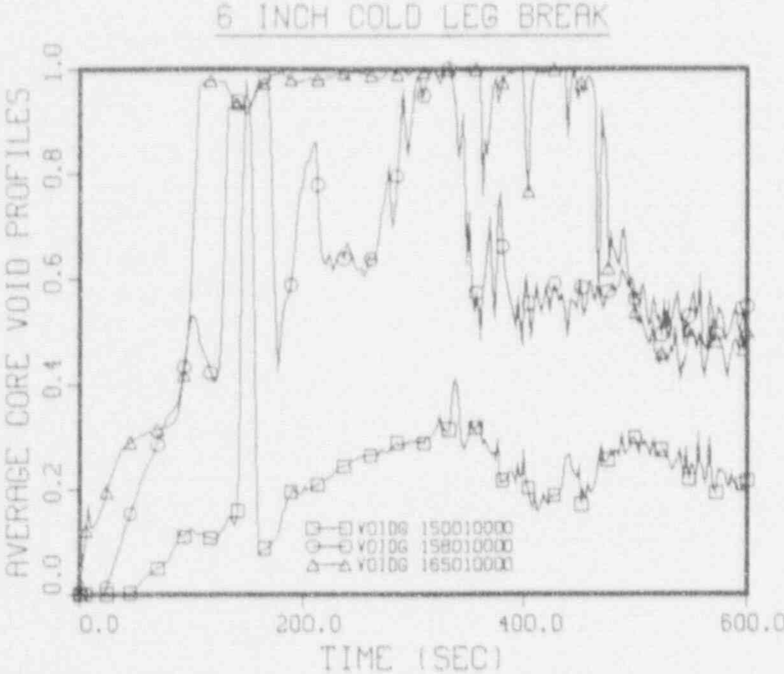


Figure 3.6 Average Core Region Void Fractions

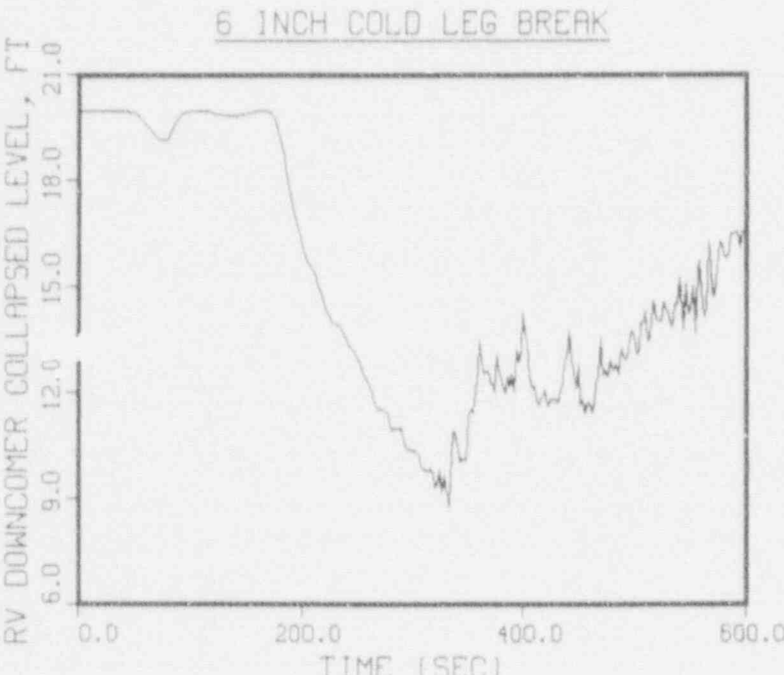


Figure 3.7 RV Downcomer Collapsed Water Level

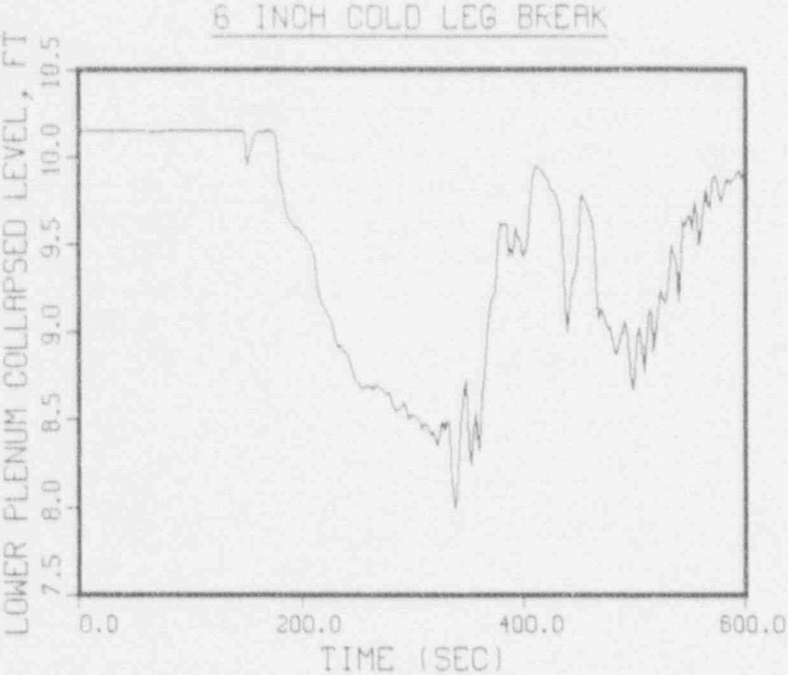


Figure 3.8 Lower Plenum Water Level

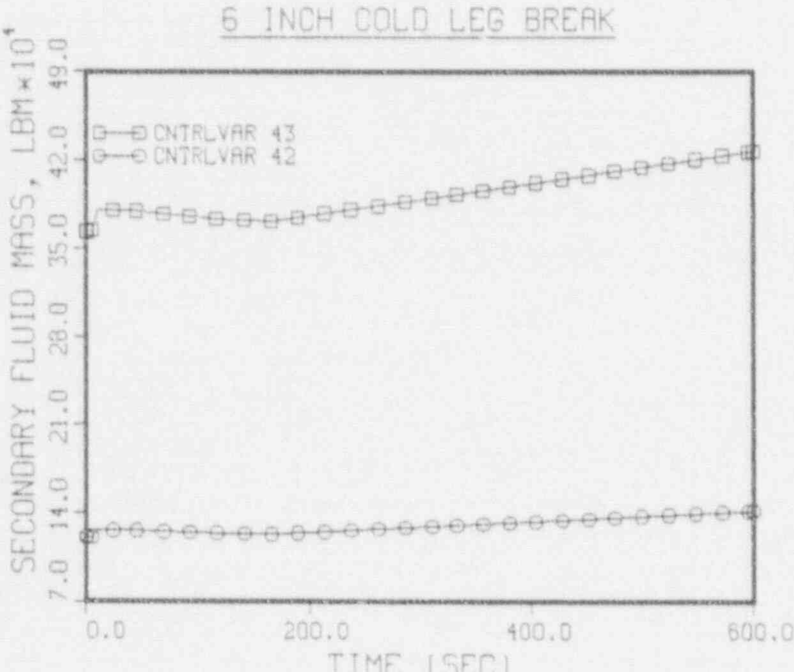


Figure 3.9 Intact and Broken Loop SG Inventories

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

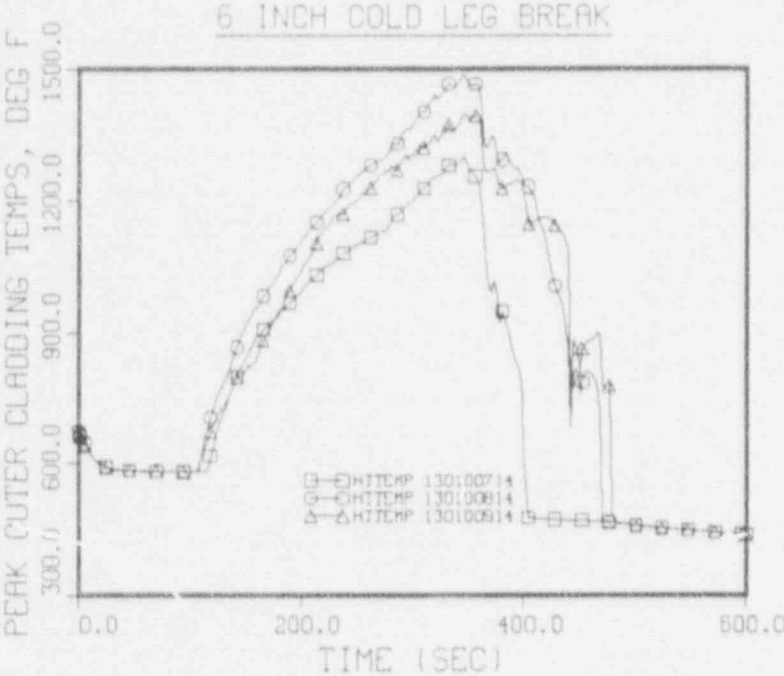


Figure 3.10 Central Core Cladding Surface Temperatures

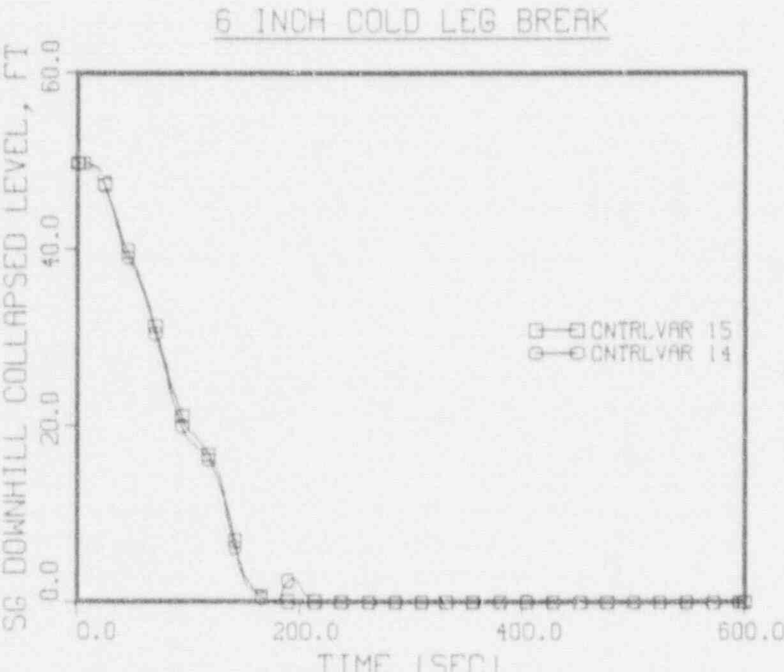


Figure 3.11 Intact and Broken Loop SG Downhill Collapsed Level

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

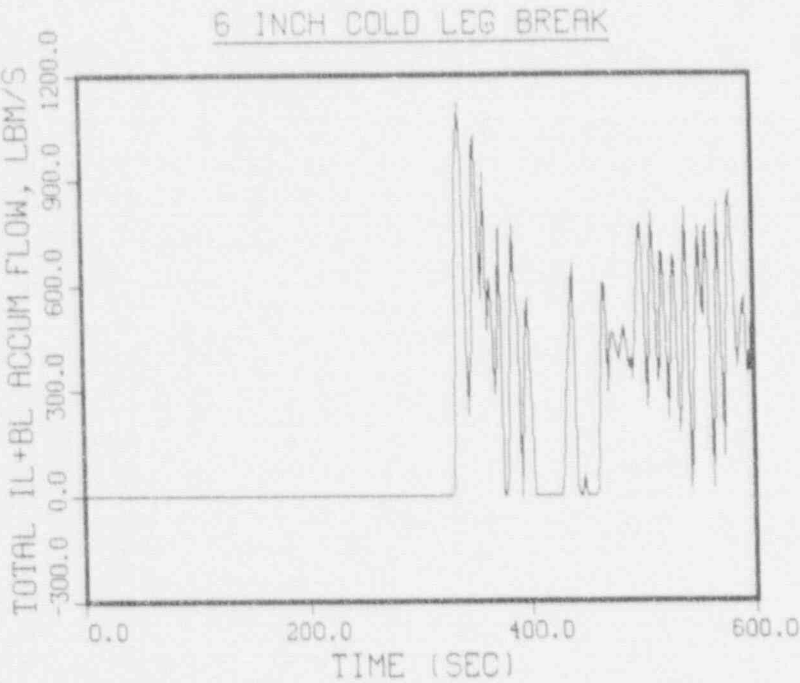


Figure 3.12 Accumulator Flow Rates

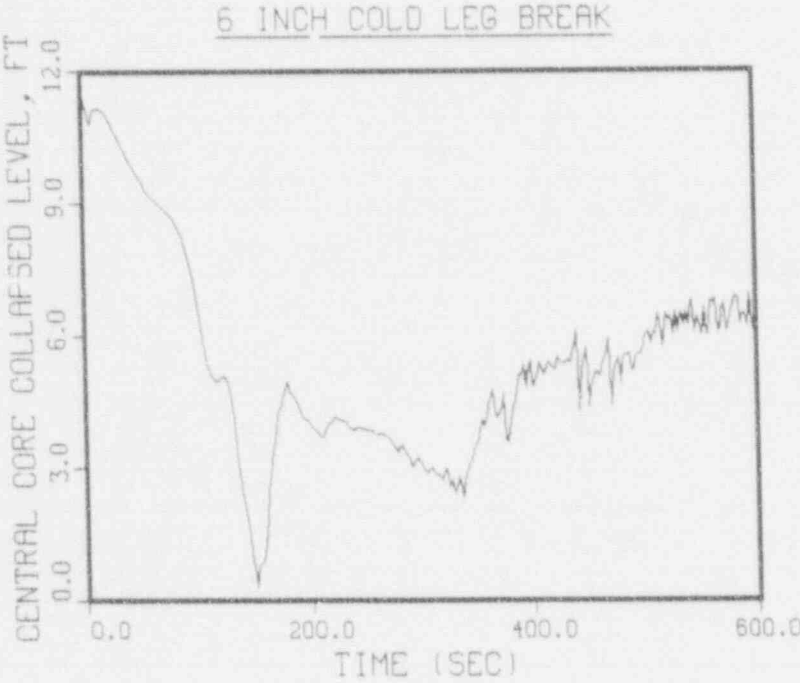


Figure 3.13 Collapsed Level in Central Core Region

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

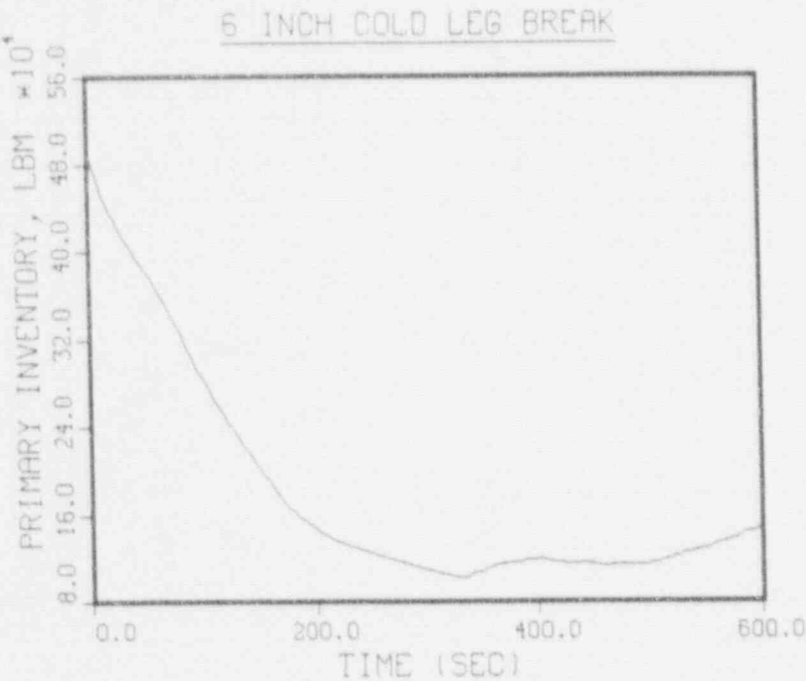


Figure 3.14 Primary System Inventory

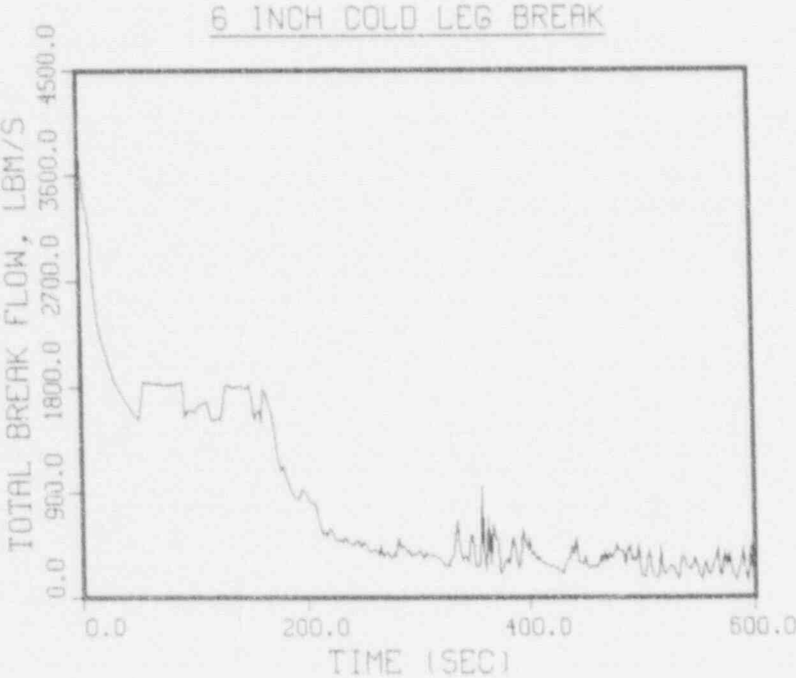


Figure 3.15 Break Flow Rate

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

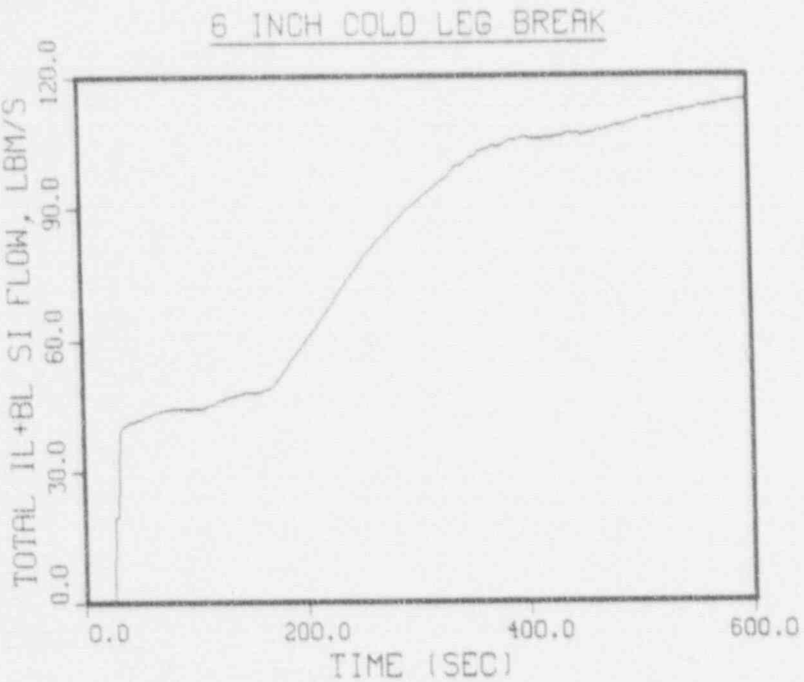


Figure 3.16 Pumped ECCS Injection Flow Rate

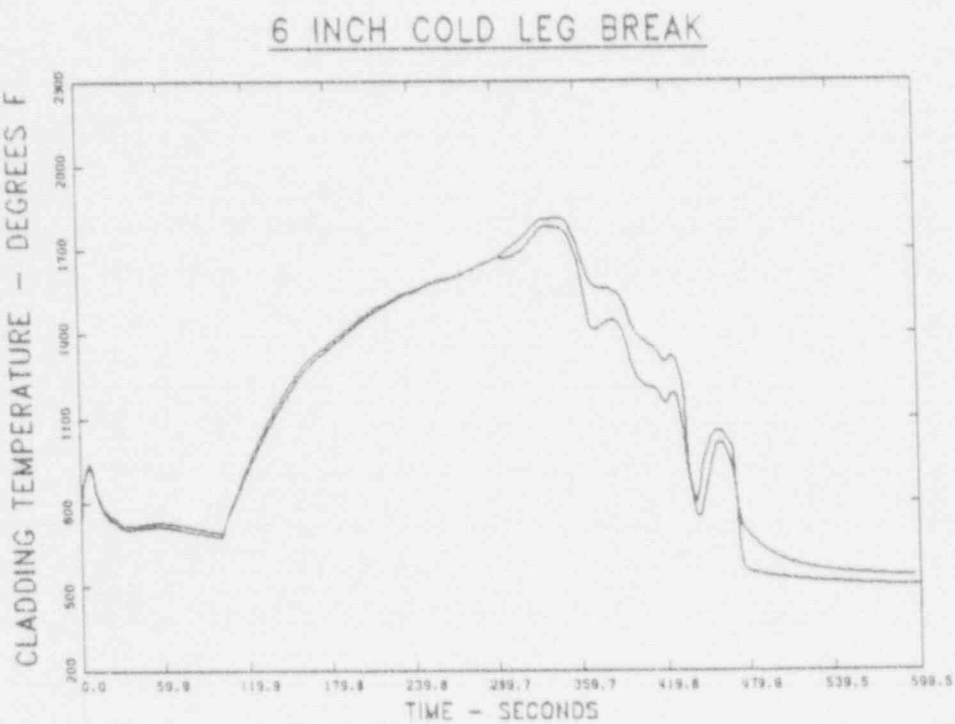


Figure 3.17 TOODEE2 Surface Cladding Temperatures

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

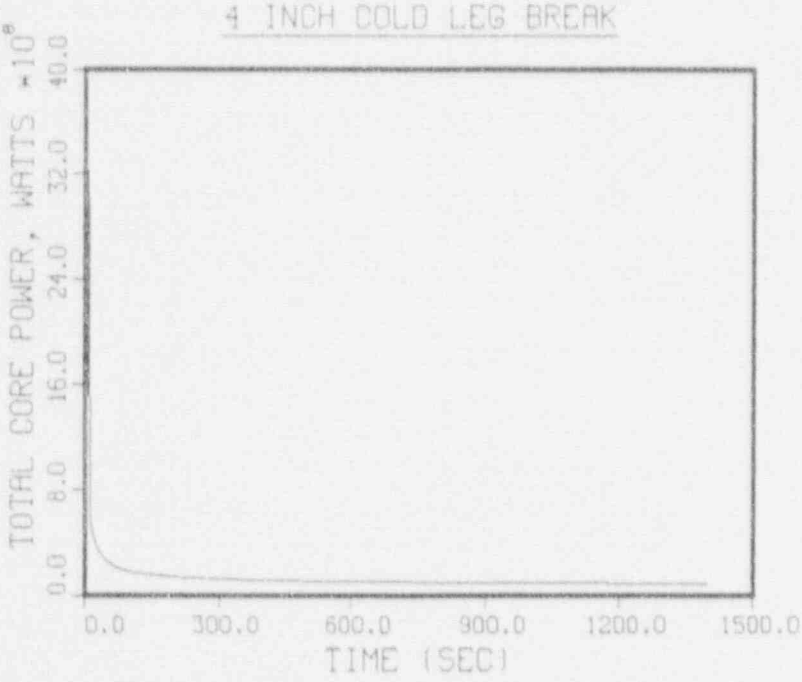


Figure 3.18 Core Power

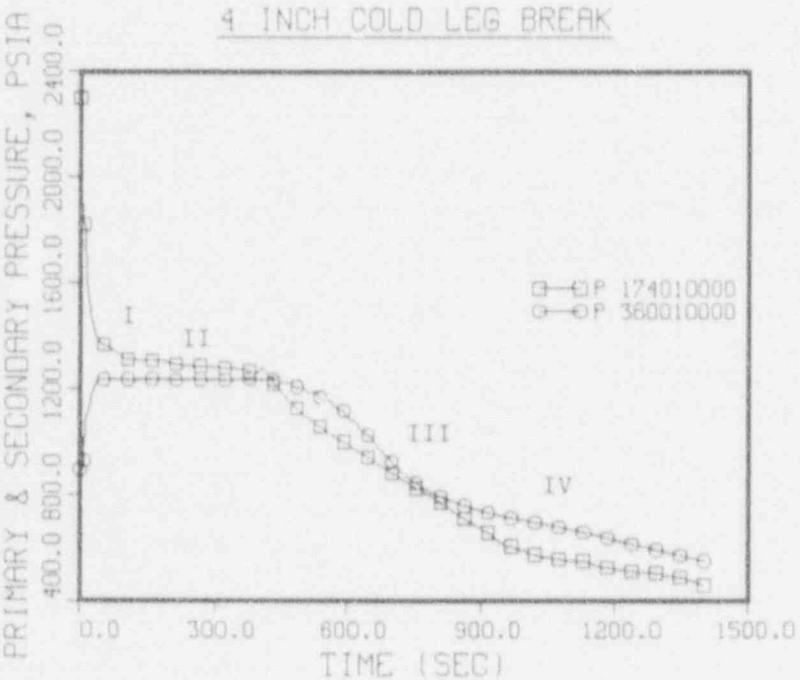


Figure 3.19 Primary and Secondary System Pressures

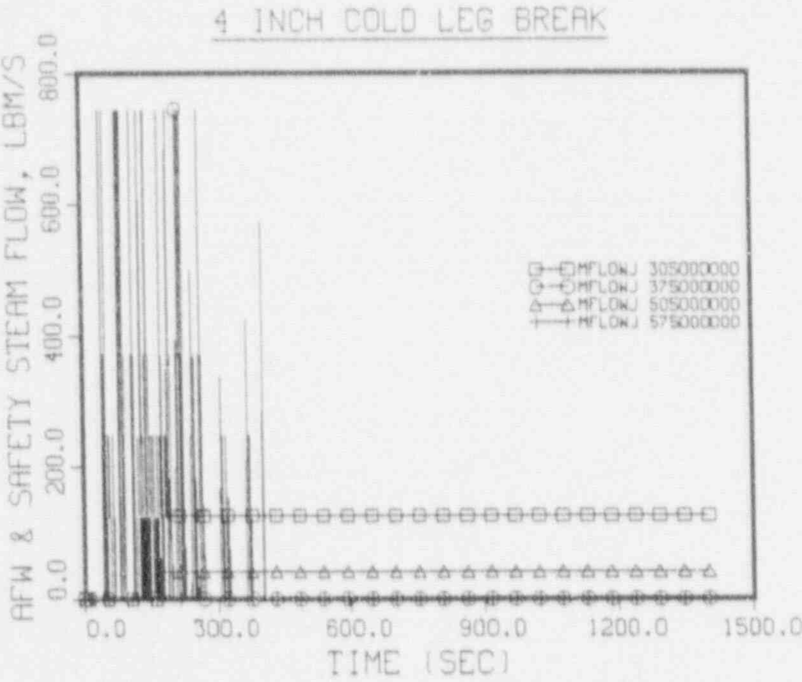


Figure 3.20 Intact and Broken Loop AFW and Steam Flows

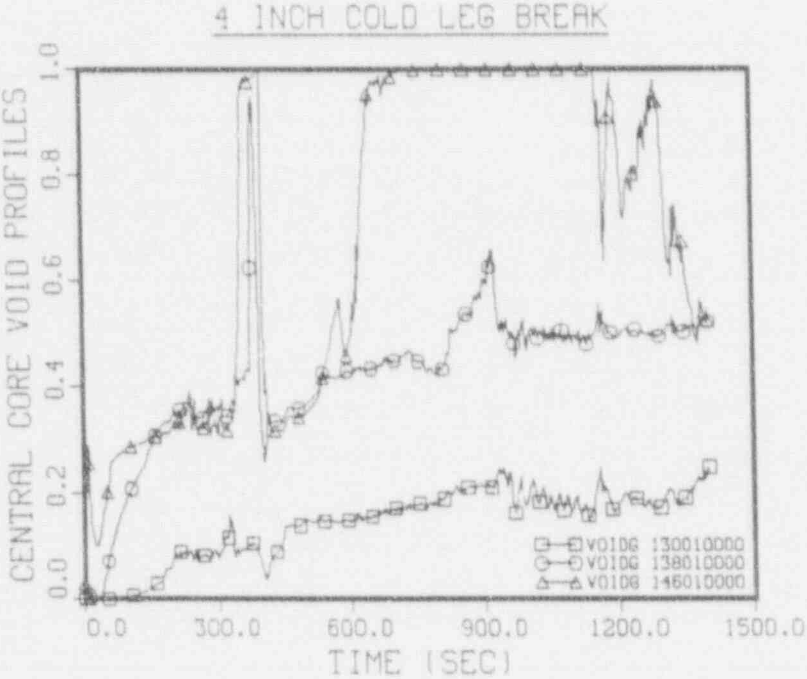


Figure 3.21 Central Core Region Void Fractions

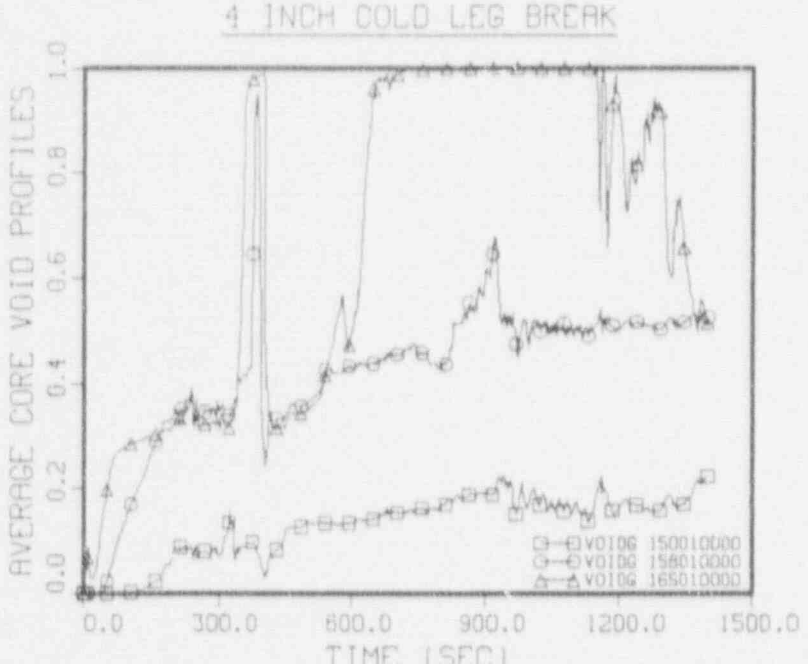


Figure 3.22 Average Core Region Void Fractions

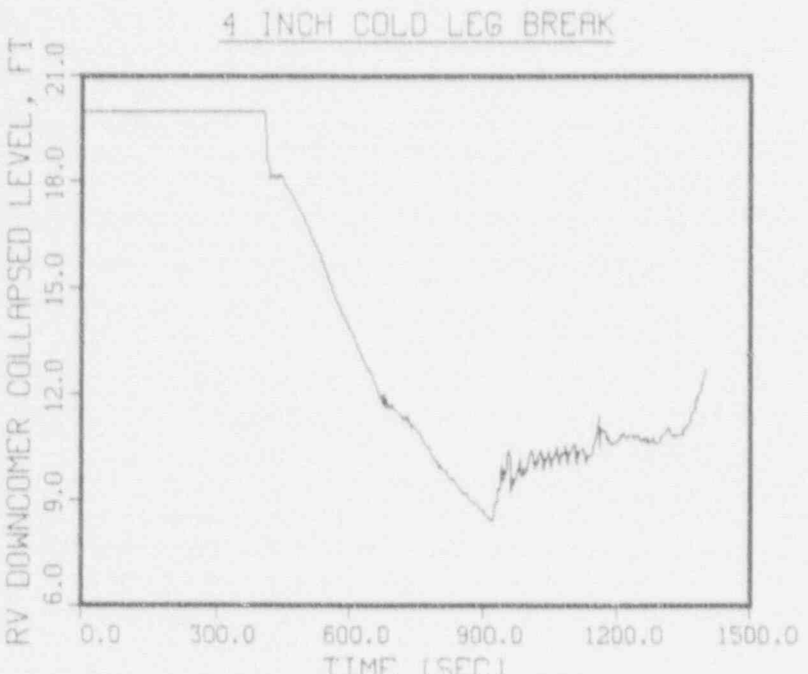


Figure 3.23 RV Downcomer Collapsed Water Level

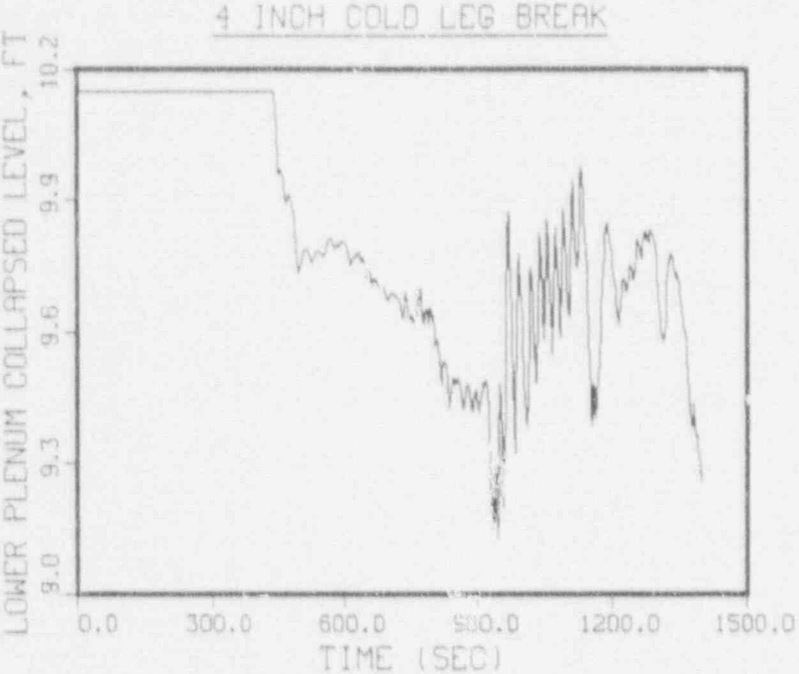


Figure 3.24 Lower Plenum Water Level

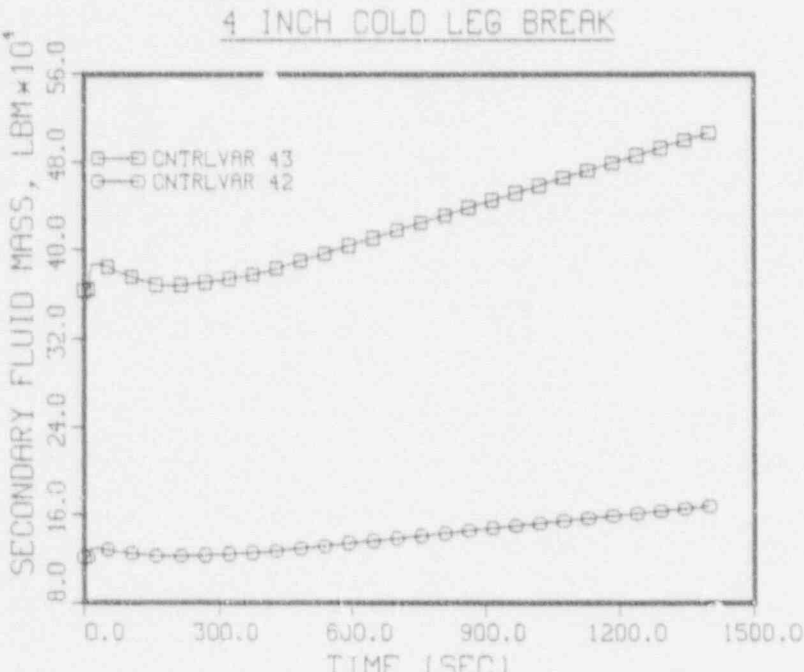


Figure 3.25 Intact and Broken Loop SG Inventories

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

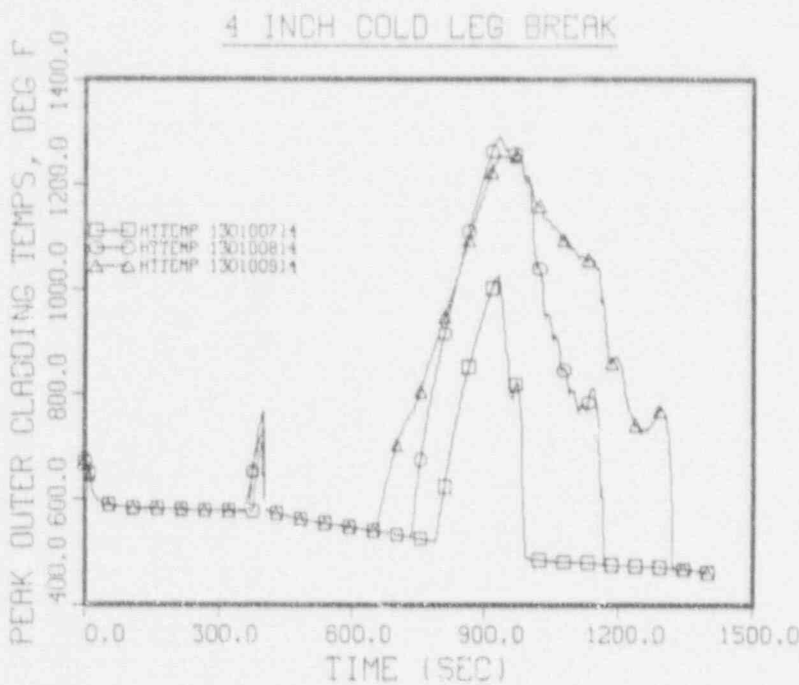


Figure 3.26 Central Core Cladding Surface Temperatures

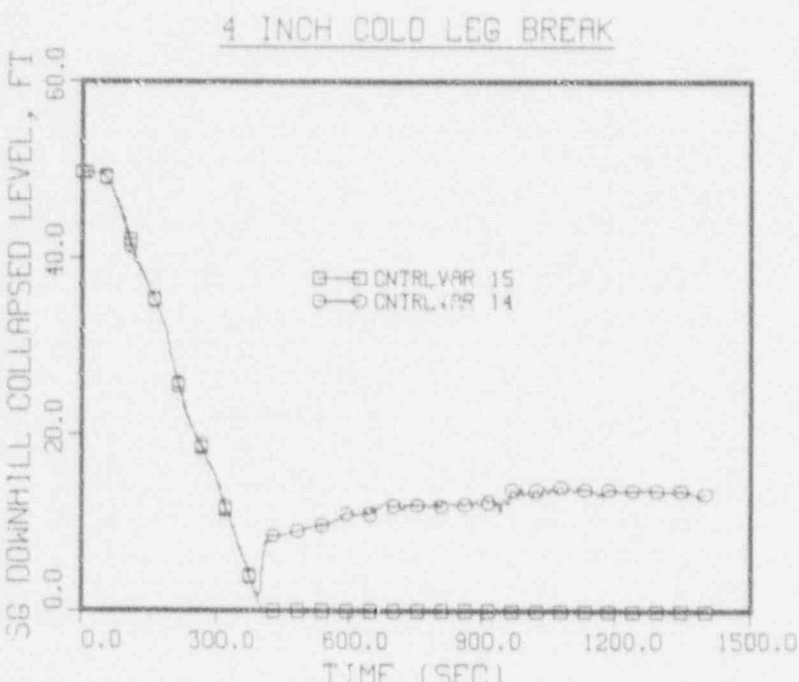


Figure 3.27 Intact and Broken Loop SG Downhill Collapsed Level

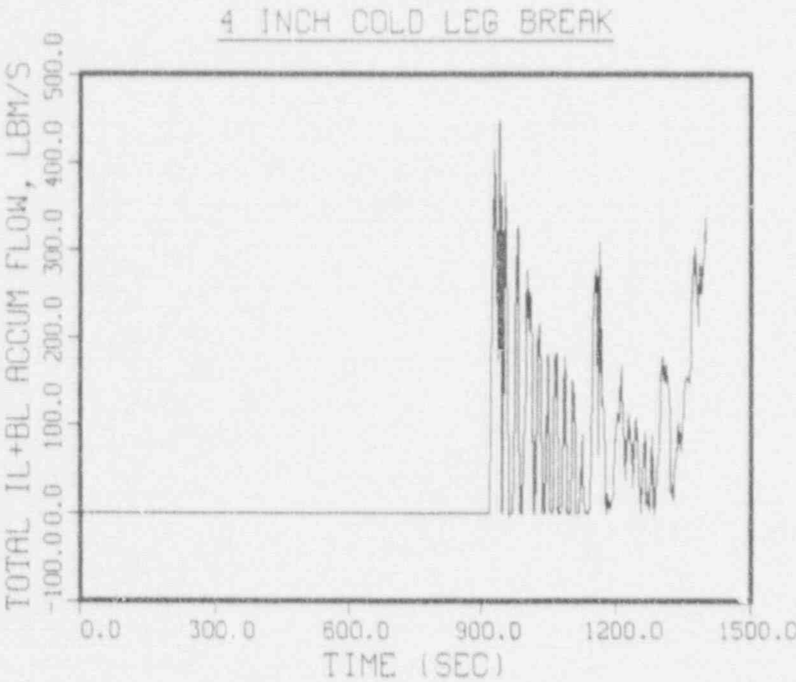


Figure 3.28 Accumulator Flow Rates

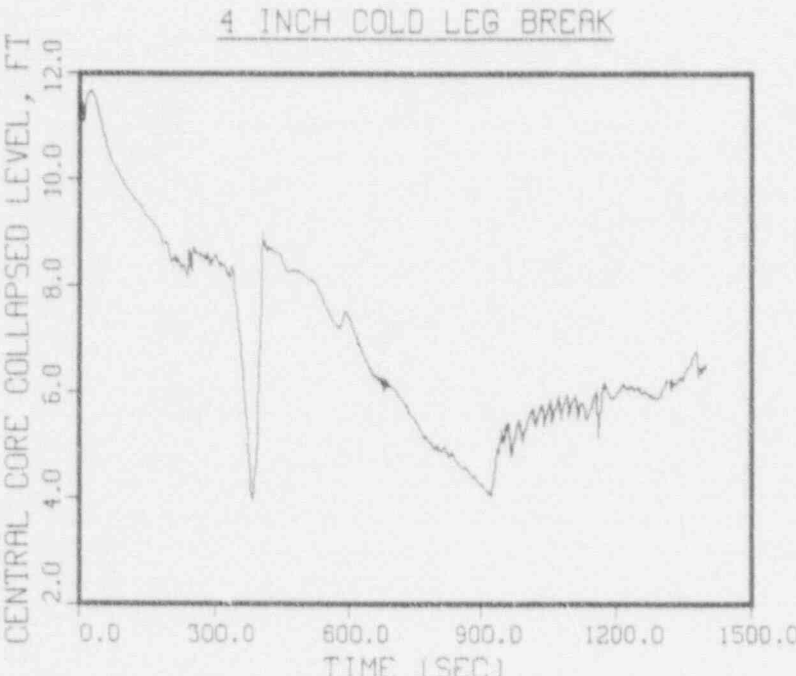


Figure 3.29 Collapsed Level in Central Core Region

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

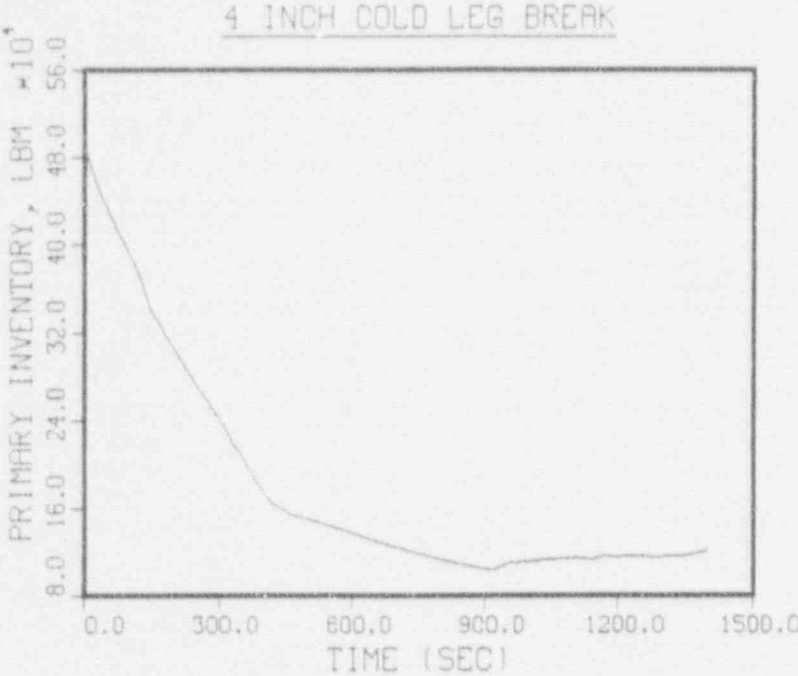


Figure 3.30 Primary System Inventory

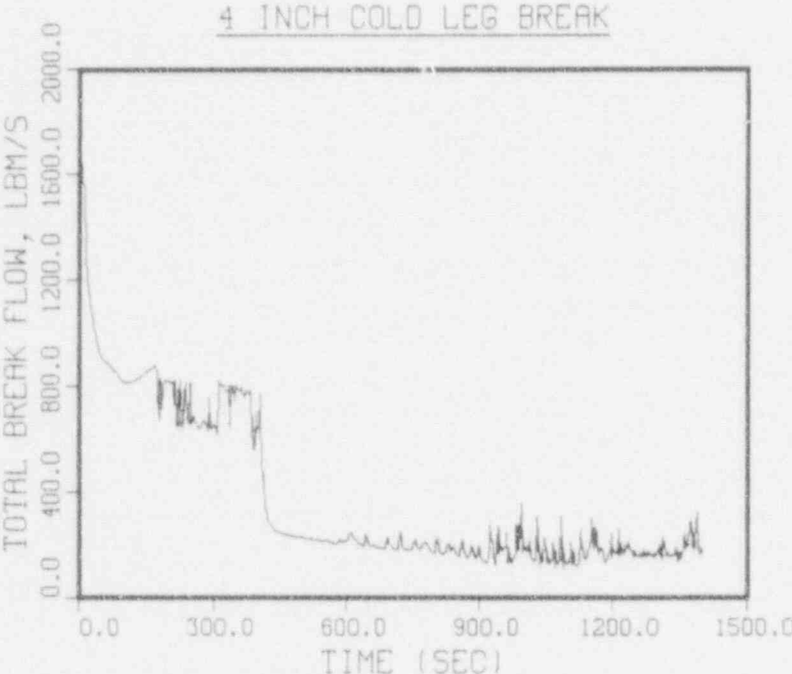


Figure 3.31 Break Flow Rate

4 INCH COLD LEG BREAK

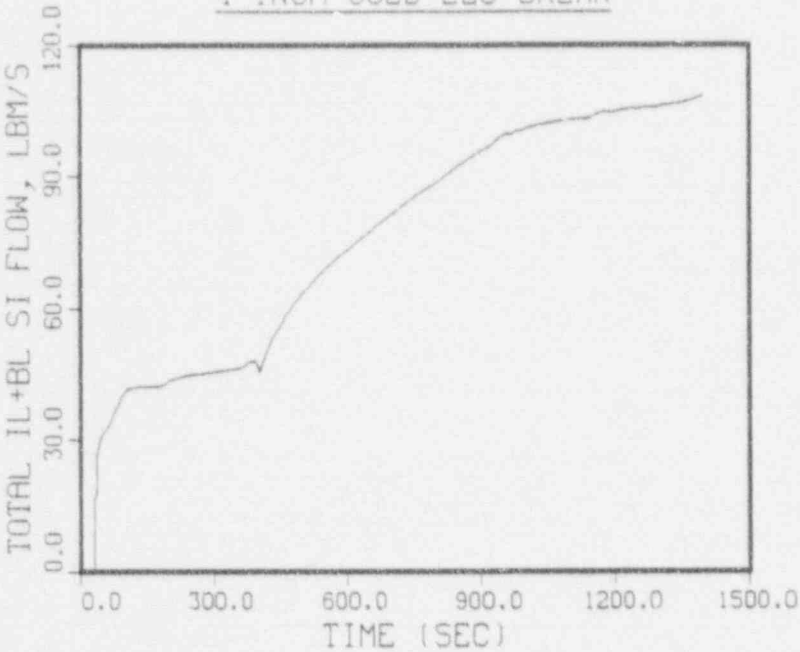


Figure 3.32 Pumped ECCS Injection Flow Rate

4 INCH COLD LEG BREAK

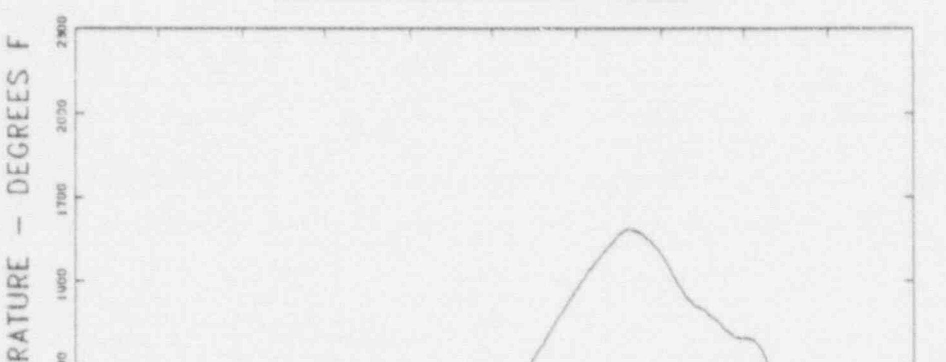


Figure 3.33 TOODEE2 Surface Cladding Temperature

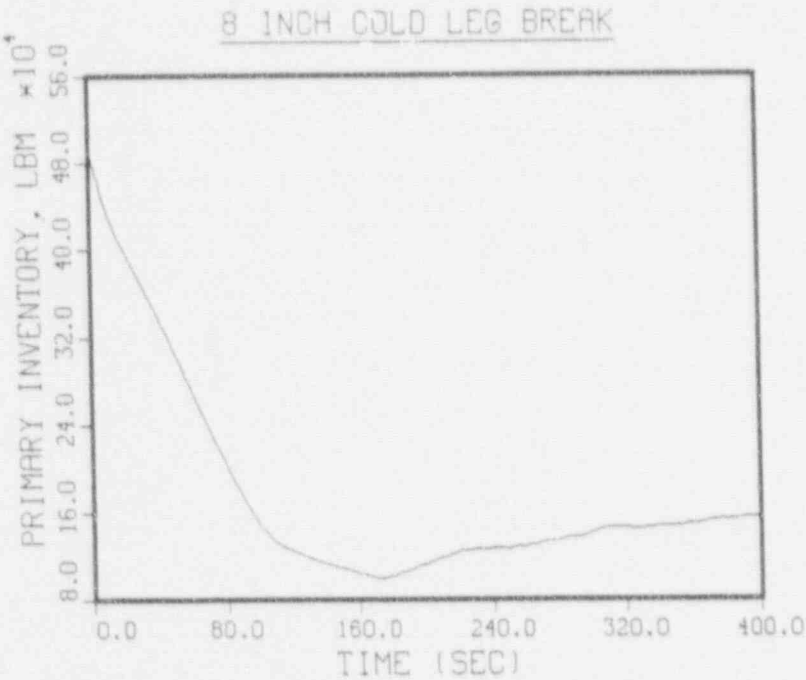


Figure 3.46 Primary System Inventory

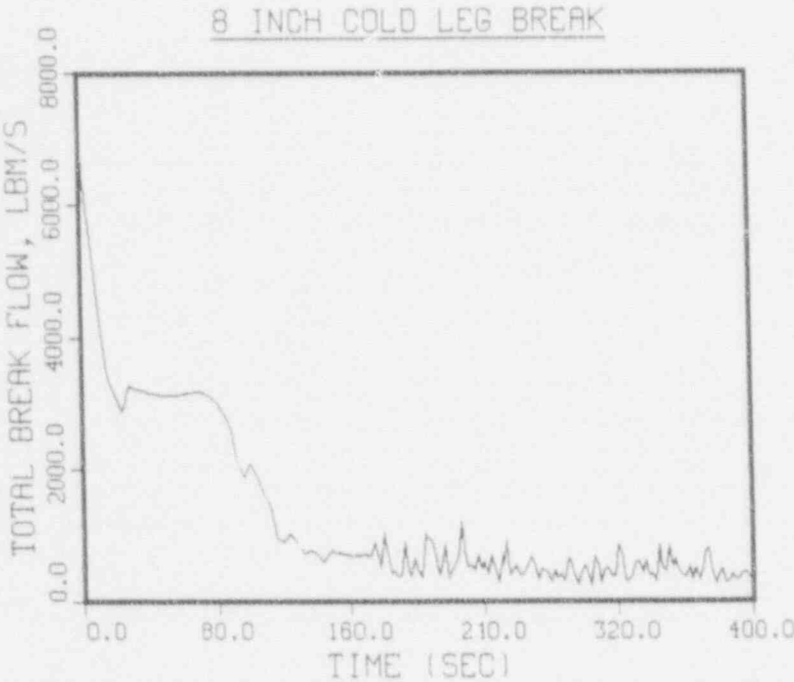


Figure 3.47 Break Flow Rate

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

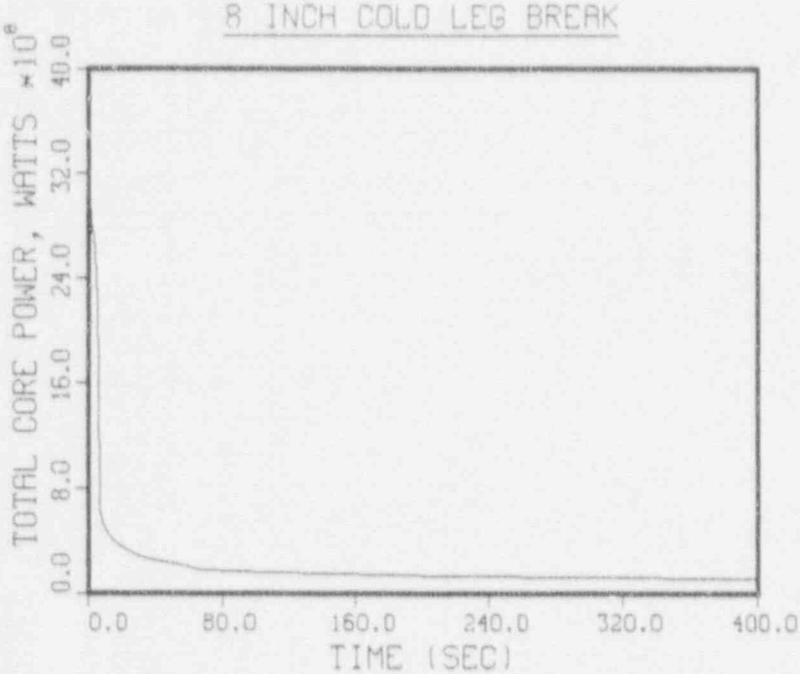


Figure 3.34 Core Power

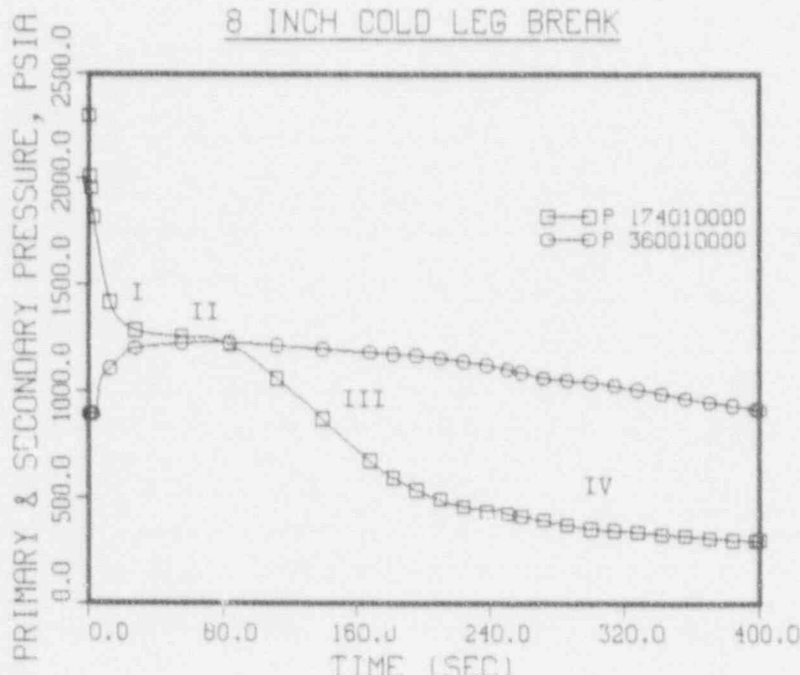


Figure 3.35 Primary and Secondary System Pressures

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

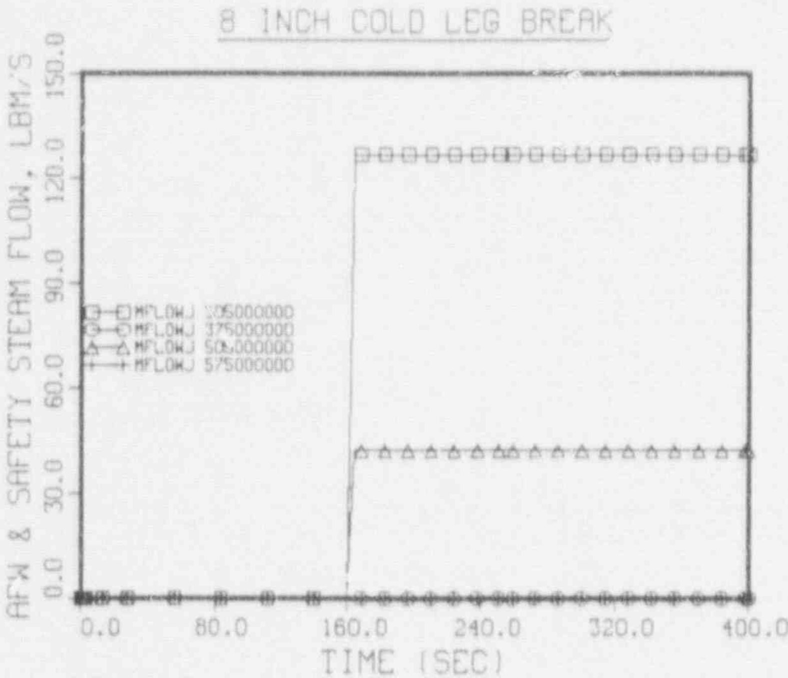


Figure 3.36 Intact and Broken Loop AFW and Steam Flows

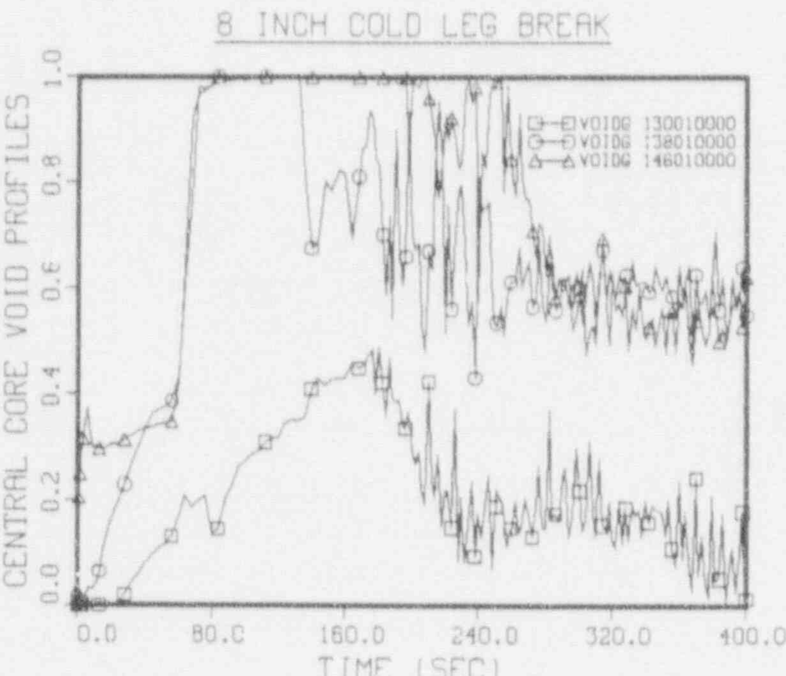


Figure 3.37 Central Core Region Void Fractions

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

8 INCH COLD LEG BREAK

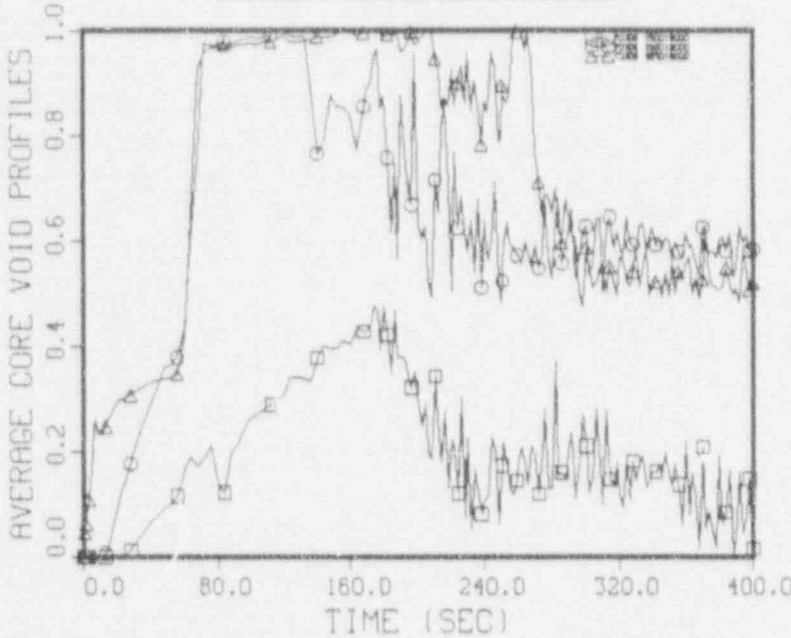


Figure 3.38 Average Core Region Void Fractions

8 INCH COLD LEG BREAK

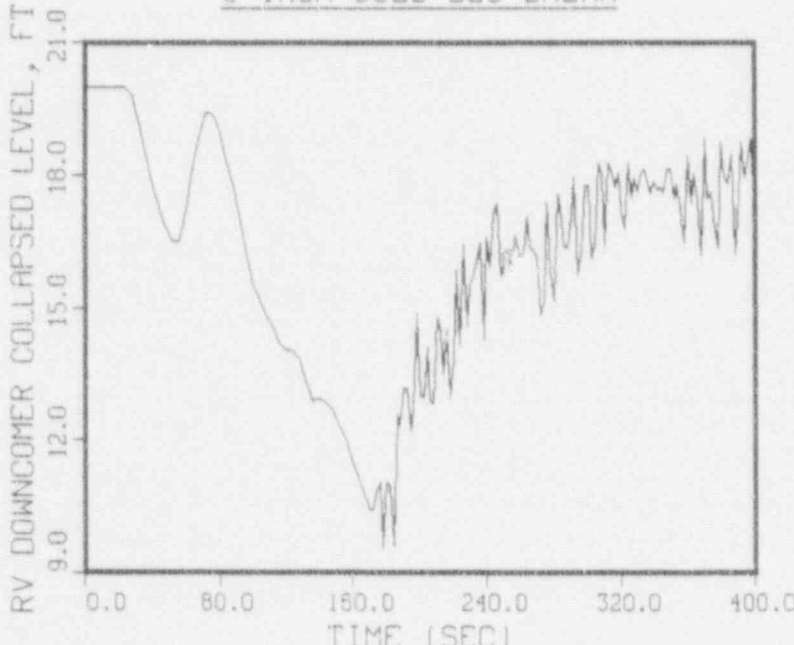


Figure 3.39 RV Downcomer Collapsed Water Level

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

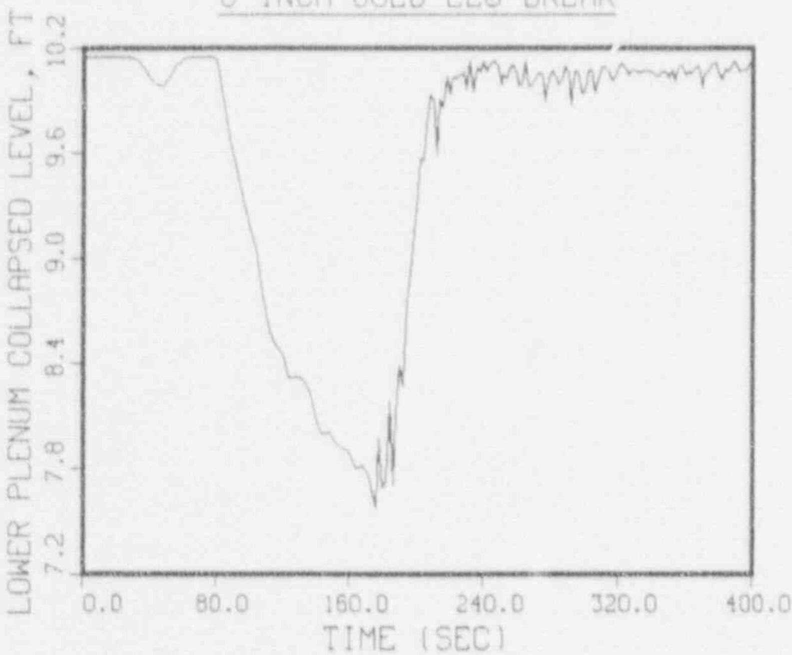


Figure 3.40 Lower Plenum Water Level

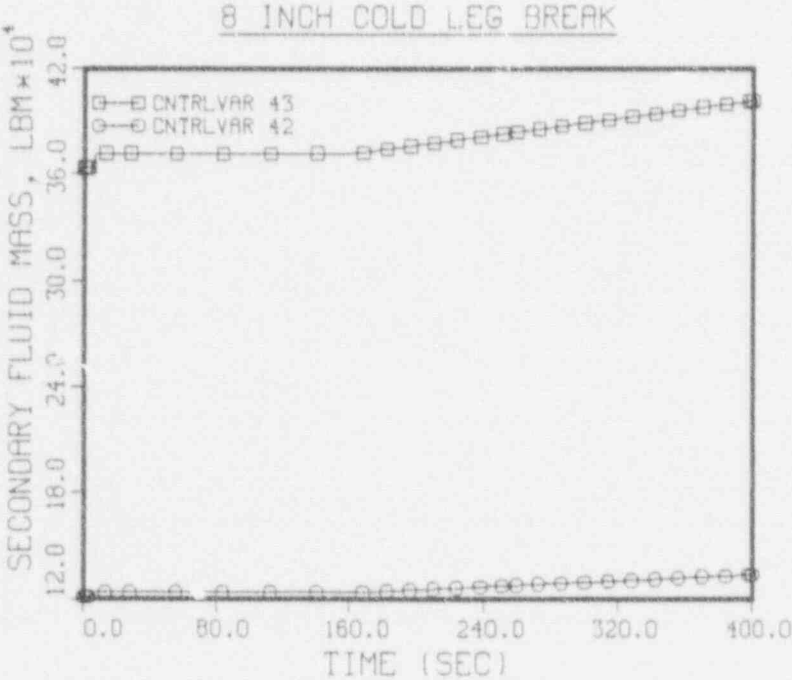


Figure 3.41 Intact and Broken Loop SG Inventories

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

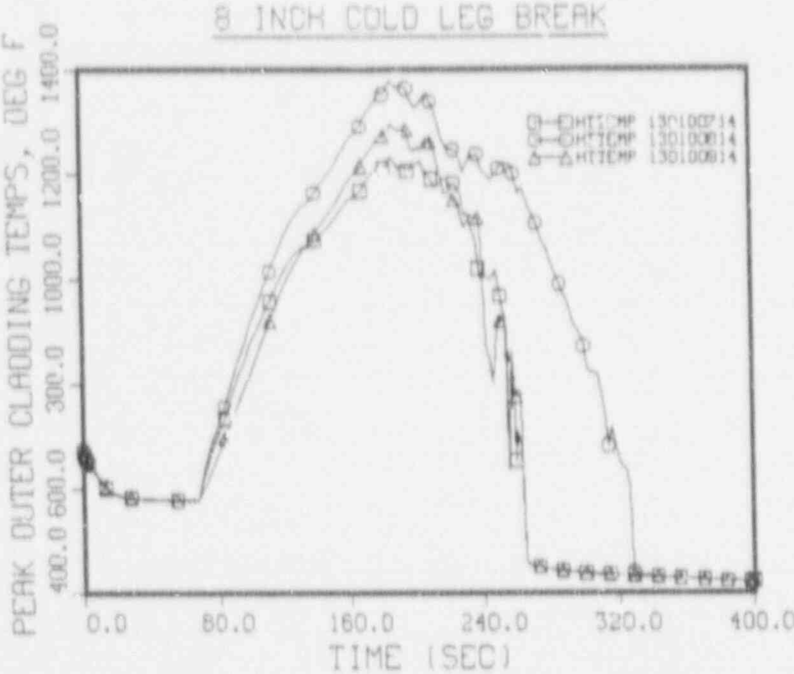


Figure 3.42 Central Core Region Cladding Surface Temperatures

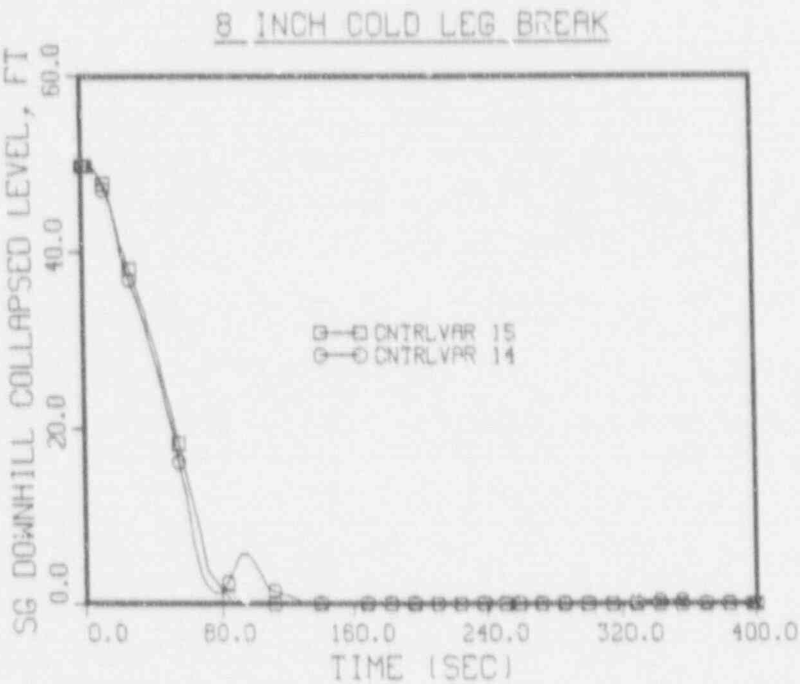


Figure 3.43 Intact and Broken Loop SG Downhill Collapsed Level

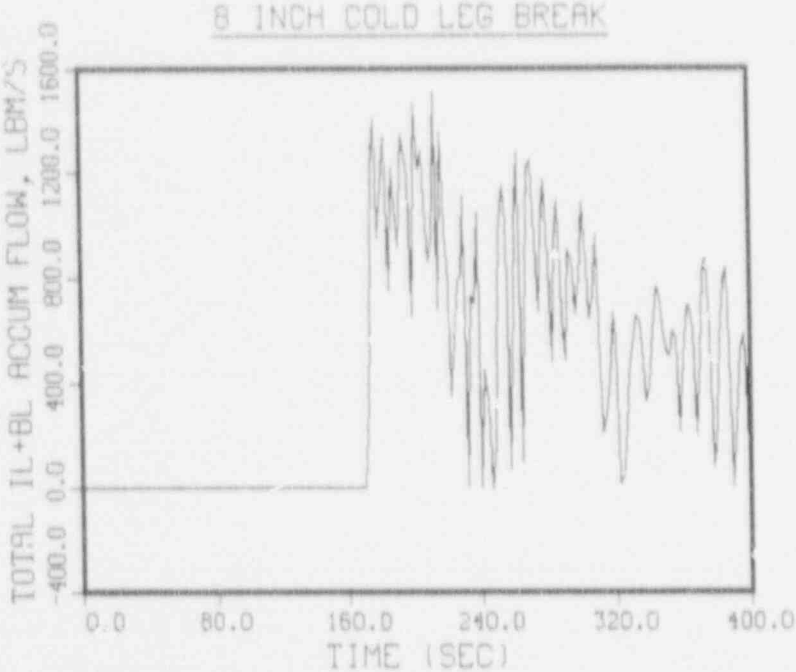


Figure 3.44 Accumulator Flow Rates

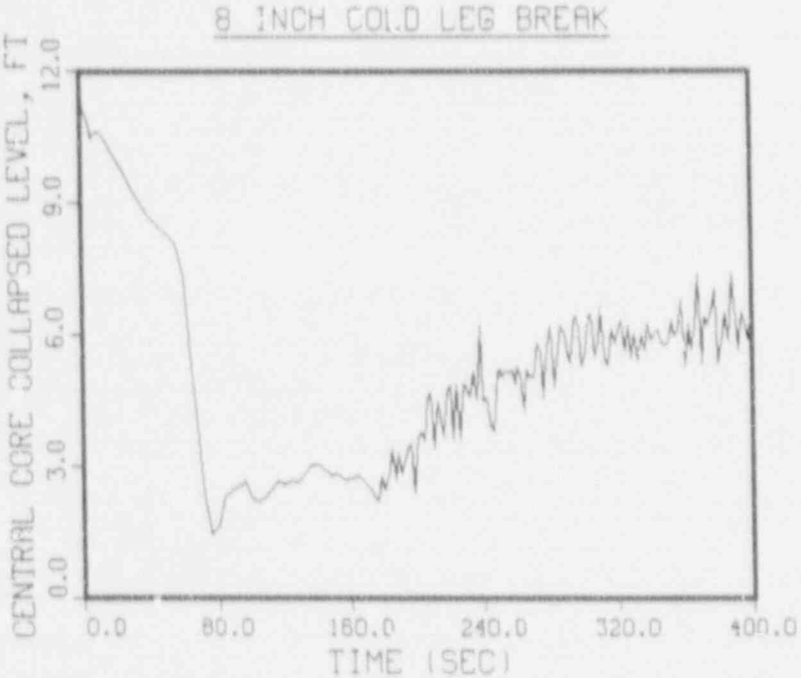


Figure 3.45 Collapsed Level in Central Core Region

8 INCH COLD LEG BREAK

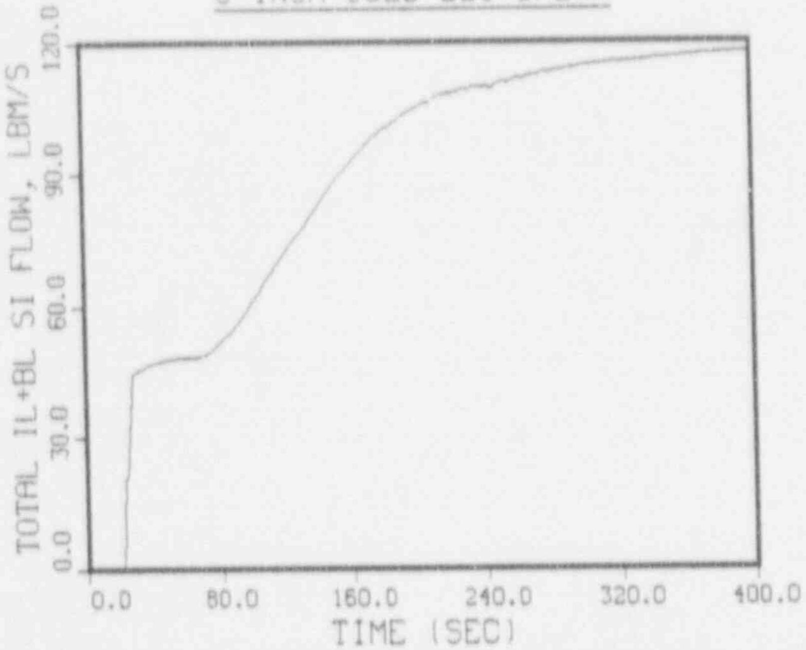


Figure 3.48 Pumped ECCS Injection Flow Rate

8 INCH COLD LEG BREAK

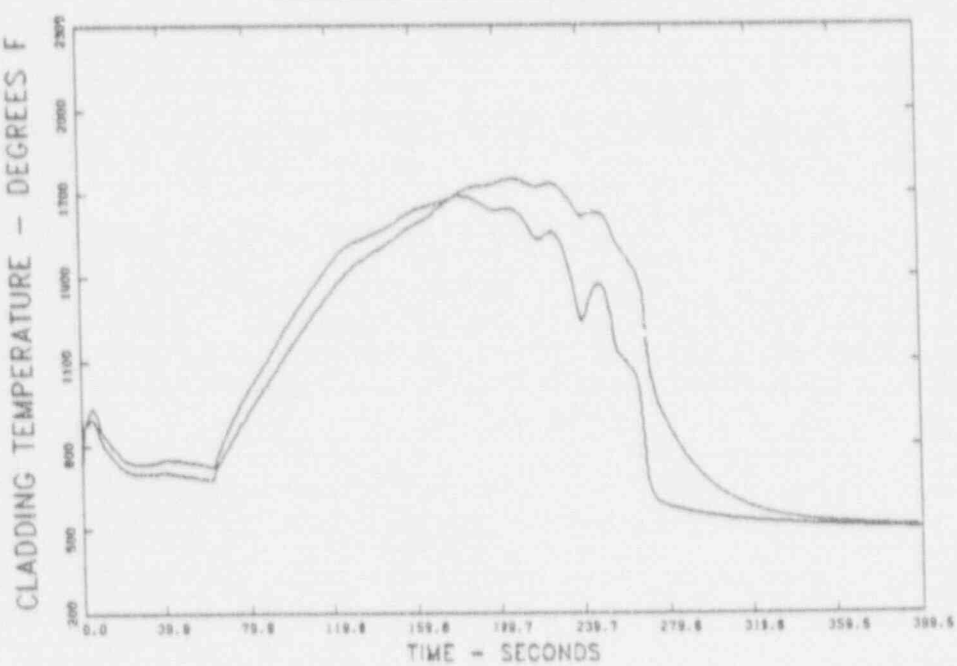


Figure 3.49 TOODEE2 Surface Cladding Temperatures

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1

6 INCH COLD LEG BREAK

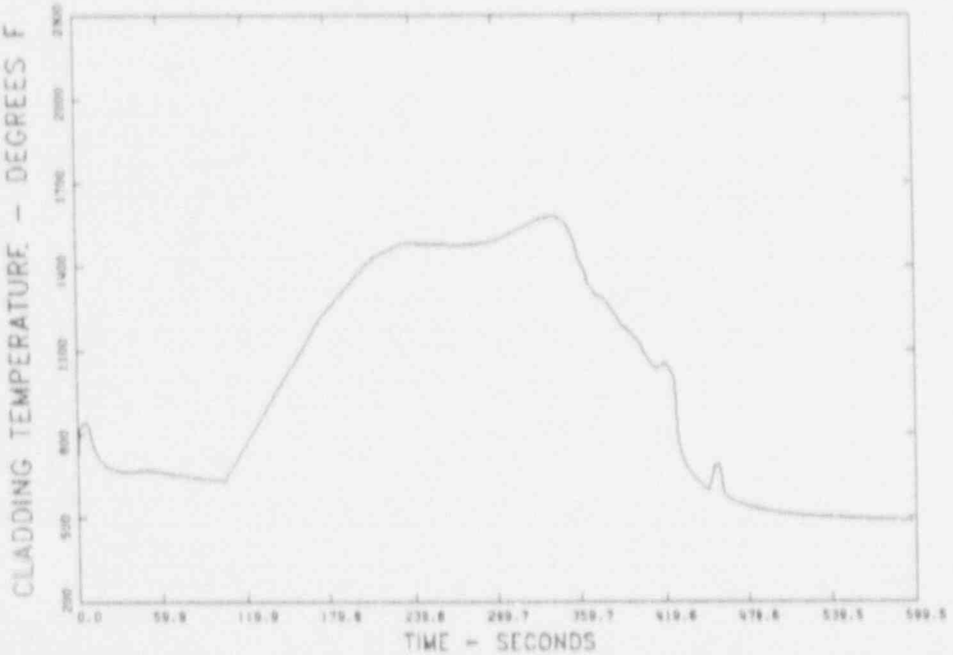


Figure 3.50 TOODEE2 Surface Cladding Temperature

6 INCH COLD LEG BREAK

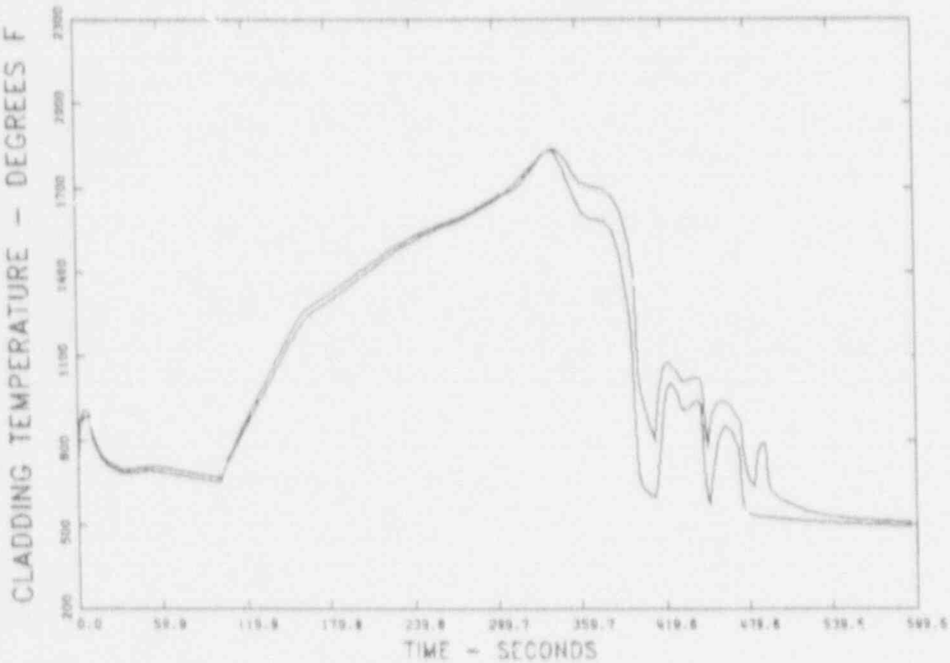


Figure 3.51 TOODEE2 Surface Cladding Temperatures

CHAPTER 4

CONCLUSION

The USNRC-approved (Reference 4.1) ANF Corporation'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 exact compliance with the explicitly approved EXEM PWR Small Break methodology. Regarding features of the calculation procedure which are "implied" in the approval, there has been but one deviation: the thermal-hydraulic calculations represent the average core region using nine axial nodes (rather than the three shown in ANF's submittal). This deviation has been made in order to increase accuracy.

Five calculations have been presented with two objectives:

1. To demonstrate Texas Utilities' 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 will be applied to subsequent fuel cycles for the Comanche Peak Steam Electric Station Unit One and Unit Two. Evaluations will be performed to verify that the results of the present analyses remain bounding.

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 degrees 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 established by 10 CFR 50.46 (b)(3).

4. No clad rupture is seen in these analyses. 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 fully quenched, the pressure level is near the RHR pumps set point 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 6 inch break in the main coolant pump discharge line.

2. The most limiting power profile is Power Shape 3, which is shown in Figure 3.1 and which corresponds to the most limiting Small Break LOCA shape as determined by the methods of Reference 3.3.

Texas Utilities 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 AND SENSITIVITY STUDIES

BREAK SIZE (INCHES)	AXIAL POWER SHAPE		
	SHAPE 1 (4)	CHOPPED COSINE	SHAPE 3
6.0	1812 °F (1) 3.12 % (2) 0.289 % (3)	1578 °F 0.593 % 0.193 %	1837 °F 3.45 % 0.399 %
4.0	1585 °F 0.685 % 0.097 %		
8.0	1750 °F 2.214 % 0.252 %		

NOTES:

(1) PEAK CLAD TEMPERATURE (DEGREES F)
(2) PERCENT LOCAL CLADDING OXIDATION
(3) PERCENT CORE-WIDE OXIDATION
(4) SEE FIG. 3.1 FOR SHAPES

CHAPTER 5

REFERENCES

Chapter 1:

1.1 Advanced Nuclear Fuels Corporation, "USNRC's Safety Evaluation of Advanced Nuclear Fuels' Small Break LOCA Evaluation Model ANF-RELAP and Acceptance for Referencing of Topical Report XN-NF-82-49, Revision 1," July 1988.

1.2 "Axial Power Distribution Control Analysis and Overtemperature and Overpower Trip Setpoint Methodology," RXE-90-006-P, to be published.

Chapter 2:

2.1 Exxon Nuclear Company, "Evaluation Model EXEM PWR Small Break Model," XN-NF-82-49, Revision 1, June 1986.

2.2 V. H. Ransom, et. al., "RELAP5/MOD2 Code Manual, Volume 1: Code Structure, Systems Models, and Solution Methods," NUREG/CR-4312, EGG-2396, August 1985.

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 B. D. Stitt, "Advanced Nuclear Fuels Corporation Procedure for PWR Safety Analysis Calculations, Section 10: Small Break LOCA Analysis," ANF-1238 (P), June, 1989.

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 Unit One, Technical Specifications.

Chapter 3:

3.1 Comanche Peak Steam Electric Station Unit One, "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 TU Electric, "Steady State Reactor Physics Methodology," RXE-89-003-P, July 1989.

3.4 Comanche Peak Steam Electric Station Unit One, Technical Specifications.

Chapter 4:

4.1 Advanced Nuclear Fuels Corporation, "USNRC's Safety Evaluation of Advanced Nuclear Fuels' Small Break LOCA Evaluation Model ANF-RELAP and Acceptance for Referencing of Topical Report XN-NF-82-49, Revision 1," July 1988.

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 <u>RODEX2</u>

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 gap conductance model includes the effects of fill gas conduction, gap size, amount of fuel cracking and the fuel-cladding contact pressure.

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. The deformation calculations include consideration of densification, swelling, instantaneous plastic flow, creep, cracking and thermal expansion for the fuel pellet, and also consideration of creep, irradiation induced growth, and thermal expansion for the cladding.

A.2 ANF-RELAP

ANF-RELAP is a modified version of RELAP5/MOD2, INEL Cycle 36.02. The RELAP5/MOD2 code is described in detail in Reference 2.2. RELAP5/MOD2 has been modified in three major ways to produce ANF-RELAP:

(1) The Moody critical flow model (Reference 2.3) substitutes the RELAP5/MOD2 critical flow model during two-phase discharge in order to comply with the related requirement of 10 CFR 50, Appendix K, Section C.

(2) The RELAP5/MOD2 mixture level calculation is modified with the objective of producing a two-phase level, more suitable for the TOODEE2 fuel rod thermal analysis.

(3) A counter-current flow limitation (CCFL) constitutive equation based on a Kutateladze formulation with constants adjusted on the basis of the S-UT-08 test (Reference 2.4) is made available for use instead of the mechanistic interphase drag models in vertical junctions which can be selected by the user.

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 gap width is adjusted until ANF-RELAP fuel temperatures match.

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 element 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 energy equation is solved to determine the fuel rod thermal response. 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. Radiation and convective heat transfer are assumed never to occur at the same time at any given axial node. Radiation is considered only until the convective heat transfer surpasses it. 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, viz., 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 3 additional codes for obtaining input information and/or transferring results between the basic codes described above:

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

A.4.1 LOOPT

This code is used to determine initial thermal-hydraulic conditions for ANF-RELAP. Flows, pressure drops and temperatures are used to initialize the ANF-RELAP steady-state deck. These conditions are not necessarily 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 power fraction input data for ANF-RELAP and the axial power factors for input to the RODEX2 and TOODEE2 codes.

A.4.3 TIFRO

TIFRO (TOODEE2 Input From RELAP Output) takes the information necessary to prepare a table of mixture levels and steam flow rates for TOODEE2 input from the ANF-RELAP restart file.

TABLE A.1

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY
COMPUTER CODES (refer to FIGURE A.1)

SHAPE.PUN
INPUT:
(1)* 24 point axial power profile (Reactor Physics) renormalized to Tech Spec peaking factor
OUTPUT:
(2) 101 point axial power profile with Tech Spec peaking factor

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

TABLE A.1 (Continued...)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY

COMPUTER CODES (refer to FIGURE A.1)

SHAPE
INPUT:
(2) 101 point axial power profile with Tech Spec peaking factor
(3) Tech Spec axial peaking factor at peak node Axial nodalization to be used in ANF-RELAP, TOODEE2, or RODEX2 Bundle geometry data (ANF-RELAP)
OUTPUT:
(10) Core power and weighting fractions calculated from axial peaking factors
(4) 24 even node axial power profile
(7) 24 uneven node axial power profile 24 uneven axial grid locations

TABLE A.1 (Continued...)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY
COMPUTER CODES (refer to FIGURE A.1)

RODEX2
INPUT: (4) 24 even node axial power profile (12) Description of fuel, e.g. geometry, density, enrichment, etc. Cladding type and dimensions Initial mole fractions of fill gas Spring dimensions Hydraulic diameter, area, mass flux Axial nodalization Core power history for: - peripheral core region - hot central region OUTPUT: (9) Hot rod cold plenum length (at exposure of interest) used to calculate cold plenum volume Hot rod gram-moles of gas (at exposure of interest) Hot rod dish + crack volume (at exposure of interest) Hot rod variables (at exposure of interest) to calculate cladding diameter and cold gap width Hot rod mole fractions (at exposure of interest) Hot rod radially averaged density (at exposure of interest) Fuel temperature profile (8) Average fuel temperatures for the hot central region and for the peripheral region. The gap width is adjusted in ANF-RELAP to match these stored energies.

TABLE A.1 (Continued...)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY COMPUTER CODES (refer to FIGURE A.1)

LOOPT
INPUT: (5) Steady-state thermal-hydraulic parameters. OUTPUT: (6) Adjusted steady-state thermal-hydraulic parameters (e.g. for tube plugging percentage, pressure drops) (16) Coolant channel mass flux

TABLE A.1 (Continued...)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY
COMPUTER CODES (refer to FIGURE A.1)

ANF-RELAP
INPUT:
(10) Core power and weighting fractions
(8) Average fuel temperatures for the hot central region and for the peripheral region. The gap width is adjusted in ANF-RELAP to match these stored energies.
(11) NSSS information (Table 2.2) ECCS, SG AFW, safety valve flows (Tables 2.6 and 2.9) Trips and delays (Table 2.7) Fuel rod/assembly information (Table 2.8) Neutronics information (Tables 2.3, 2.4, 2.5)
OUTPUT:
(13) Saturation temperature entering the central core Vapor mass flow rate exiting the central core Normalized core power Position of the two-phase central core mixture level Average quality of the fluid below central core mixture level Mass flow of liquid entering the central core Temperature of the liquid entering the central core

TABLE A.1 (Continued...)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY
COMPUTER CODES (refer to FIGURE A.1)

TIFRO
INPUT: (13) ANF-RELAP restart file (See ANF-RELAP Output (13)) OUTPUT: (14) Saturation temperature entering the central core Vapor mass flow rate exiting the central core Normalized core power Position of the two-phase central core mixture level Average quality of the fluid below central core mixture level Mass flow of liquid entering the central core Temperature of the liquid entering the central core

TABLE A.1 (Continued...)

INPUT AND OUTPUT FOR THE EXEM/PWR METHODOLOGY COMPUTER CODES (refer to FIGURE A.1)

TOODEE2
INPUT:
(9) Hot rod cold plenum length (at exposure of interest) used to calculate cold plenum volume Hot rod gram-moles of gas (at exposure of interest) Hot rod dish + crack volume (at exposure of interest) Hot rod variables (at exposure of interest) to calculate cladding diameter and cold gap width (used for geometric definition of hot rod and blockage data) Hot rod mole fractions (at exposure of interest) Hot rod radially averaged density (at exposure of interest) Cladding + fuel surface roughness
(7) 24 uneven axial node power profile 24 uneven axial grid locations
(14) Saturation temperature entering the central core Vapor mass flow rate exiting the central core Normalized core power Position of the two-phase central core mixture level Average quality of the fluid below central core mixture level Mass flow of liquid entering the central core Temperature of the liquid entering the central core
OUTPUT:
(15) Peak cladding temperature and time Percent local cladding oxidation Percent pin wide cladding oxidation Rupture Node location and time

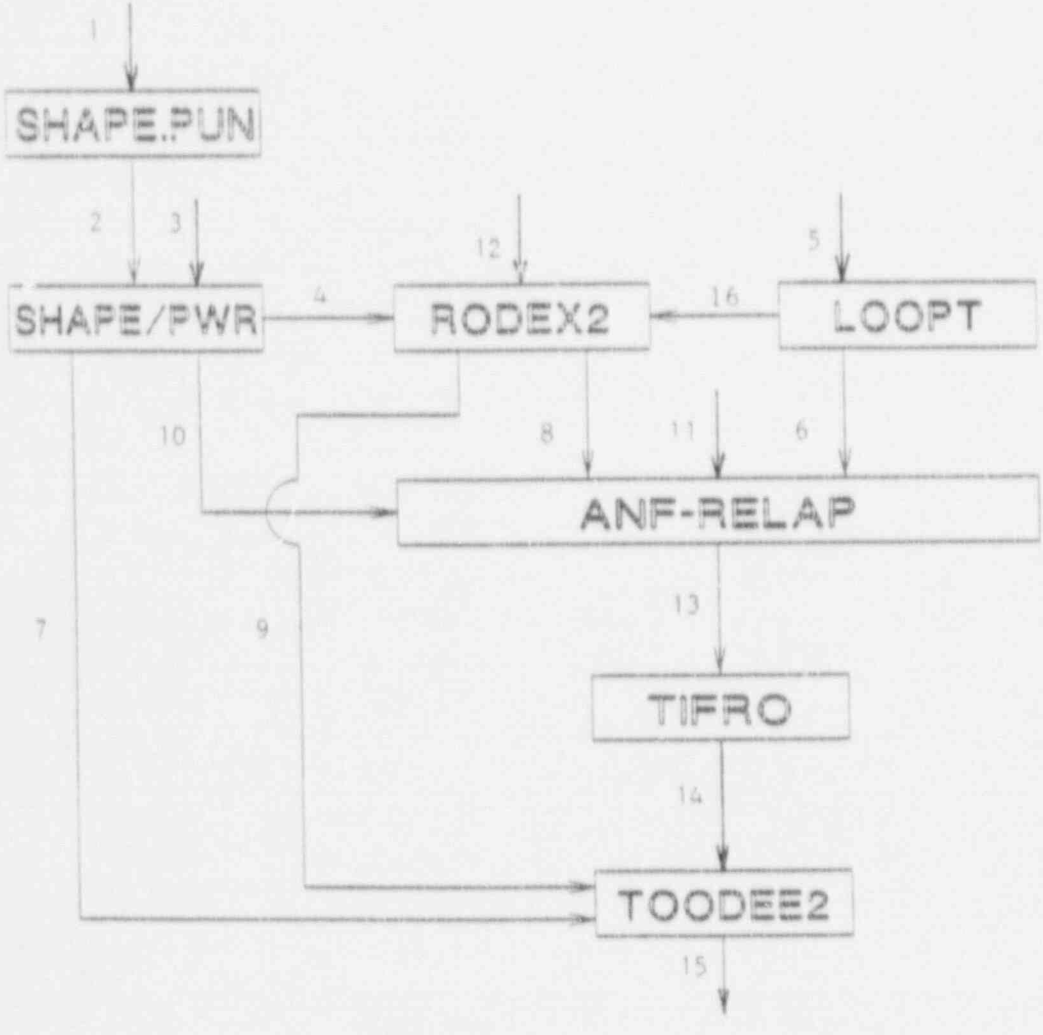


Figure A-1 ANF SBLOCA Computer Code Interfaces