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LARGE 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 large break (Emergency Core Cooling Systems) ECCS Evaluation Model entitled EXEM/PWR, to the Comanche Peak Steam Electric Station (CPSES).

This report contains a description of the EXEM/PWR 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 the LOCA-ECCS licensing analyses that comply with USNRC regulations contained in 10 CFR 50.46 and Appendix K thereto.

In order to comply with a 10 CFR 50, Appendix K requirement, a full spectrum of large breaks, ranging from 0.6 to 1.0 discharge coefficients for Double-Ended Guillotine breaks (DEG) and 1.0 for a longitudinal split break, is examined.

Furthermore—in order to support the Technical Specification linear heat generation rate (LHGR) limit as a function of core height—all realistic potentially limiting axial power shapes are considered, and analyses are presented for the chopped cosine and two top skewed axial power profiles.

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

This 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 (Ref. 1.1) Advanced Nuclear Fuels (ANF, formerly Exxon Nuclear) Corporation's large break ECCS Evaluation Model, entitled EXEM/PWR, to the Comanche Peak Steam Electric Station Unit One (CPSES-1).

The method is used to perform the LOCA-ECCS (Emergency Core Cooling Systems) licensing analyses that comply with USNRC regulations contained in 10 CFR 50.46 and Appendix K thereto.

The analyses presented in this report include a description of the EXEM/PWR 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 methodology. Regarding features of the calculation procedure which are "implied" in the approval, there is but one deviation: the thermal-hydraulic calculations represent the core region using five axial nodes (rather than the three shown in ANF's submittal). This deviation has been made in order to increase accuracy.

Three types of sensitivity studies are presented in Chapter 3.

The first is a break spectrum study. Large breaks ranging, from 0.6 to 1.0 discharge coefficients for Double-Ended Guillotine (DEG) and 1.0 for a longitudinal split break, are examined in order to comply with 10 CFR 50, Appendix K.

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. This is done as follows: First the population of shapes is developed through the axial power distribution control analysis described in Reference 3.5. 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 the most likely to have the highest integrated power up to the PCT elevation are selected. Analyses are presented for the chopped cosine and two top skewed profiles. These are the most likely candidates to yield the highest PCT according to the criterion just described.

The third type of sensitivity (burnup study) consists of examining the EOC fuel condition for the most limiting break and power shape as determined in the previous sensitivity studies (in which BOC fuel is used).

In Chapter 4, results from all these sensitivity studies are used to establish the Design Basis Accident (i.e., most limiting LOCA case) for the EXEM/PWR 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 methodology, their interfaces, interrelation-ships, and respective inputs and outputs.

The 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 the above may be applied to the Comanche Peak Steam Electric Station Unit One and Unit Two for subsequent fuel cycle analyses and to address any applicable 10 CFR 50, Appendix K issues. Evaluations will be performed on the basis of specific parameters to insure that results of the present analyses remain bounding.

CHAPTER 2

DESCRIPTION OF THE METHOD

2.1 BACKGROUND

In 1975, the NRC approved use of the Exxon Nuclear Company (ENC) WREM-based generic PWR ECCS Evaluation Model (Ref. 2.4). This LOCA Evaluation Model is based on the NRC-developed Water Reactor Evaluation Model (WREM) (Ref. 2.8).

In 1976, the ENC PWR model was updated resulting in the ENC WREM-II Evaluation Model (Ref. 2.9). The ENC-WREM-II model differs from the ENC-WREM model in four areas: (a) flow reduction due to blockage during reflood at rates less than 1 in/sec, (b) FLECHT multipliers for low reflood rates, (c) ice condenser containment pressure, and (d) hot wall delay.

In 1979, WREM-II was updated, leading to the WREM-IIA model (Ref. 2.1). The WREM-IIA differs from the WREM-II only with respect to evaluation of the reflood portion of the LOCA transient. During this portion of the transient, the RELAP4-EM/FLOOD (WREM-II) calculation is replaced by a similar calculation using REFLEX.

In July 1986, the NRC accepted the EXEM/PWR (Ref. 2.14) large break ECCS Evaluation Model for referencing of related licensing topical reports. EXEM/PWR is based on ENC WREM-IIA PWR ECCS EM (Ref. 2.1).

EXEM/PWR updates WREM-IIA in four phases of the transient calculation: (a) stored energy and fission gas release models are revised in the fuel rod model in the RODEX2 code, (b) the NUREG 0630 clad rupture/blockage and a new fuel rod model are added to the RELAP4-EM system blowdown calculation, (c) leakage flow from upper plenum to downcomer is allowed, also new split break and core outlet enthalpy models are used along with a revised carryout rate fraction correlation in the REFLEX code for the reflood period, and (d) the heatup model in TOODEE2 includes a revised steam cooling model, NUREG-0630 clad rupture/blockage, a revised radiation heat transfer model, and a revised reflood heat transfer correlation.

The present report describes the application of Exxon Nuclear Company's large break ECCS Evaluation model, entitled EXEM/PWR, to the Comanche Peak Steam Electric Station Unit One.

2.2 OVERVIEW OF THE METHOD

The EXEM/PWR methodology is illustrated schematically in Fig. 2.2.1. The accident is divided into three phases: blowdown, refill, and reflood. These phases are separated by two key events: End-of-Bypass (EOBY) and Bottom of Core Recovery (BOCREC). For presentation purposes, it is also appropriate to distinguish two types of calculations performed over these periods: Thermal-Hydraulic and Fuel Rod Thermal Analysis. These are discussed in the sections that follow.

2.2.1 THERMAL-HYDRAULIC ANALYSIS

2.2.1.1 BLOWDOWN

The analysis of the large break LOCA begins with the hydraulic analysis of the blowdown phase, noted as Step (1) on Fig. 2.2.1. RELAP4-EM computes the thermal-hydraulic conditions of the primary and secondary systems during the depressurization following the LOCA. The RELAP4-EM system model used for CPSES-1 is described in detail in Section 2.3.1. The RELAP4-EM system calculation determines the time dependent boundary conditions for the blowdown portion of the hot channel calculation. These are: (a) the core inlet and outlet plenum conditions and (b) the core power level. RELAP4-EM system calculation also provides the End-of-Bypass time (EOBY), mass and energy releases to the containment up to EOBY, and initial system conditions for the reflood analysis.

2.2.1.2 END-OF-BYPASS

The time at which downward flow through the downcomer is sustained for at least one second, less the time for accumulator fluid to flow from the intact cold leg injection point to the downcomer, is the calculated time for End-by-Bypass. This time signals the end of the blowdown as well as the start of the refill period.

2.2.1.3 REFILL

The rate at which the ECCS fluid is injected into the primary system intact recirculation lines during refill is determined by the ACCUM-SIS calculation (Step (3) in Fig. 2.2.1). This calculation uses a RELAP4-EM model which is essentially identical to the ECCS portion of the RELAP4-EM system blowdown model. The ACCUM-SIS calculation determines the ECCS flow rates to the cold legs after the End-of-Bypass period (EOBY). The intact loop ECCS boundary conditions for the ACCUM-SIS calculation are taken from the RELAP4-EM system blowdown calculation up to EOBY and assumed to be constant and equal to the containment pressure at EOBY thereafter.

Therefore, this calculation repeats the system calculation out to EOBY.

The determination of the containment backpressure for the refill period is done by ICECON/CONTEMPT-LT (Ref. 2.5), which is included in the RFPAC code.

The power generated in the core during the refill and reflood portions of the transient is calculated using a one-volume RELAP4-EM model and the FISHEX code, as shown in Step (4) in Fig. 2.2.1. The RELAP4-EM code is used to calculate the delayed fission contribution to the normalized decay power. Since RELAP4-EM computes total power, the fission contribution is obtained within FISHEX by subtracting fission product decay heat from the RELAP4-EM total power. Then the 20% multiplier is applied only to the fission product decay heat and not to the actinide decay heat, in compliance with the 10 CFR 50, Appendix K requirements.

2.2.1.4 <u>Bottom of Core Recovery (BOCREC)</u>

Following the EOBY as determined in the RELAP4-EM system blowdown calculation, downflow is calculated in the downcomer region of the reactor vessel. ECC water injected into the intact loops of the reactor will flow to the lower plenum under the influence of gravity forces. The time at which the water level reaches the bottom of the active fuel is called the Bottom of Core Recovery (BOCREC) and signals the start of the reflood portion of the transient.

The time to begin reflood, the ECCS flow rates to be used in the reflood analysis, and the temperature at which the ECCS fluid enters the core at the start of reflood are calculated in PREFILL, which is also a part of the RFPAC code (Step (5) in Fig. 2.2.1). The initial and boundary conditions to the PREFILL code are obtained from RELAP4-EM system blowdown results, the intact loop ACCUM-SIS calculation and the ICECON/CONTEMPT-LT calculation. The phenomena addressed by PREFILL are: (a) hot wall delay period, (b) free-fall delay time, (c) extended accumulator flows, (d) open channel flow spill, and (d) core inlet subcooling.

The start of reflood (BOCREC) is calculated by integrating in time the allowed flow rate of the ECCS water to the appropriate intact cold leg volume frac n to the lower plenum, and to the downcomer volume below the core inlet until they become liquid full. The time required for the ECCS water to fall from the bottom of the cold leg pipe to the core inlet (i.e., the free-fall delay time) is added to the time needed to fill the volumes listed above, yielding the actual BOCREC time.

When the ECCS fluid is injected into the downcomer, the fluid experiences a hot wall delay. Steam upflow created at the hot walls limits the downflow of ECCS fluid in the downcomer. During the hot wall delay period, the level in the downcomer may rise above the bottom of the broken loop cold leg, and liquid can flow out the break. In this situation, the break flow is calculated by a hydraulic model which includes open channel flow. If the ECCS flow is higher than the maximum flow allowed by the hot wall phenomenon then the allowed flow into the system is adjusted to account for the spillage. The adjusted flow rates are the ones used in the previously described integration process which determines BOCREC.

2.2.1.5 REFLOOD

This calculation considers the rate of reflooding of the reactor core (Step (5) in Fig. 2.2.1) and establishes core fluid conditions for the heatup calculations. The REFLEX code is used to perform the reflood analysis. In the ANF reflood calculations, the initial fuel rod temperatures for the average core are used. These are obtained from the RELAP4-EM hot channel calculation at EOBY (Step (2) in Fig. 2.2.1). The SHAPE/REFLOOD code calculates the fuel rod temperatures at BOCREC with the assumption of adiabatic heatup.

The REFLEX program calculates core reflood rates. This program is built upon a RELAP4 skeleton. The RELAP4 system equations are simplified in REFLEX in the interest of computational speed as follows:

The core neutronics, transient heat conduction and critical flow tables are omitted.

Acceleration pressure losses are omitted in the flow equations. Mass accumulation and gravitational losses are also omitted in all systems components except in the core and downcomer nodes and in the cold leg piping to the break during the accumulator discharge phase.

The fluid state equations are based on analytical fits to property tables over a limited pressure range, 10-100 psia. This method is faster than the previous table look-up process.

The numerical scheme of RELAP4 is replaced for the flow calculation by the linear theory method (Ref. 2.10), using a Gauss-Jordan elimination method (Ref. 2.11).

The core outlet enthalpy is conservatively assumed to be determined by steam generator secondary temperature and containment pressure in order to yield a conservatively high upper plenum pressure for reflood.

2.2.2 FUEL ROD THERMAL ANALYSIS

The fuel rod thermal analysis encompasses the three time periods outlined above, viz. blowdown, refill, and reflood, using two computational tools, viz. RELAP4-EM hot channel and TOODEE2.

2.2.2.1 Blowdown

The RELAP4-EM computer program is also used to perform the Hot Channel analysis which is identified as Step (2) in Figure 2.2.1. It is used: (a) to calculate the heatup transient during the blowdown phase, (b) to establish the temperature profile and extent of the metal-water reaction at the End-of-Bypass (EOBY) for the Fuel Rod Thermal Analysis described in Section 2.2.2, and (c) to provide average core, hot assembly, and hot rod cladding and fuel temperatures for the reflood calculation. Boundary conditions from the system blowdown calculation are used in performing these calculations. The RELAP4-EM Hot Channel model used for CPSES-1 is described in detail in Section 2.3.2.

2.2.2.2 Refill and Reflood

The rod thermal analysis during the refill and reflood period is performed by TOODEE2 computer code. TOODEE2 uses the EOBY temperatures from the hot channel analysis (Step (6) in Fig. 2.2.1) and performs an adiabatic heatup, except for radiation, which continues until BOCREC.

The reflood rates, as calculated in REFLEX, provide the remaining boundary conditions to complete the hot rod temperature analysis from BOCREC through the reflood period until core quench.

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 zircalloy. 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. Whenever rupture is determined and the flooding rate drops below 1 in/sec, only steam cooling is allowed downstream of the ruptured node. This is in compliance with the related Appendix K requirement. The effect of clad strain on pellet-to-clad gap heat transfer and on the thinning of the oxide layer on the outside of the cladding is considered. Once fuel rod rupture is determined, the code calculates both inside and outside metal water heat generation. Fuel rod rupture reduces the subchannel flow area at the rupture and diverts flow from the hot rod subchannel to neighboring subchannels.

Flow recovery is allowed above the rupture. The effect of flow diversion on heat transfer to the coolant is accounted for. The TOODEE2 code calculates heat transfer coefficients as a function of fluid condition or via reflood data-based correlations.

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 criteria (if pin-wide oxidation is less than 1% it is concluded that the criteria of less than 1% core-wide oxidation is met).

2.3 DESCRIPTION OF THE MODELS

2.3.1 CPSES-1 RELAP4-EM SYSTEM BLOWDOWN MODEL

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

This section describes the RELAP4-EM system blowdown base input model for the Comanche Peak Steam Electric Station Unit One (CPSES-1). The components of this model are as follows:

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

2.3.1.1 VOLUMES, JUNCTIONS AND HEAT STRUCTURES

Figure 2.3.1 shows the CPSES-1 nodalization diagram for the base input model which is comprised of 50 volumes, 74 junctions and 50 heat structures. Except for the number of axial core nodes, the model is identical to that approved by the NRC in connection with EXEM/PWR (Ref. 1.1). Table 2.3.1 identifies the particular volumes, junctions, and heat structures associated with the important regions or systems.

Table 2.3.2 summarizes the most important parameters of the CPSES-1 NSSS model volumes and junctions. These parameters were calculated using information from the most recent plant drawings, design basis documents, vendor documents, Technical Specifications and Final Safety Analysis Report.

2.3.1.2 <u>CORE POWER</u>

The total core power during transients is determined by the point reactor kinetics model in RELAP4-EM. Conservative input data are entered for this model in order to compute the fission power and decay heat per 10 CFR 50, Appendix K. The model accounts for the reactivity effects associated with the 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 factor presented in References 2.18 to assure conservatism. For the analyses presented herein, reactivity feedbacks representative of the CPSES-1 core have been selected and are shown in Tables 2.3.3 and 2.3.4 for moderator density effects and fuel temperature effects, respectively. Scram reactivity is conservatively neglected in the model.

2.3.1.3 EMERGENCY CORE COOLING SYSTEMS

The ECCS 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 (Fig. 2.3.3). 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. Only one train is represented in the present NSSS model. The other train is taken out by the single failure criterion in compliance with 10 CFR 50, Appendix K. 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 its minimum value (40 degrees F) in order to minimize the containment back-pressure. The flow versus pressure values for each injection system, which are given in Table 2.3.4, reflect spillage of injection to the broken loop. The injection capacities were obtained from Ref.3.1.

The system contains four accumulators, one per loop. The minimum accumulator set pressure is used in all calculations in this report. A sensitivity study using the highest accumulator set pressure allowed by Technical Specifications yielded insignificant differences in the fuel temperatures. Accumulator water temperature is assumed to be 90 degrees F for consistency with the initial containment temperature. The minimum Technical Specifications (Ref. 3.4) tank water volume (6119 gals.) is also used.

2.3.1.4 TRIPS AND DELAYS

The following trips and delays are used in the blowdown model:

1. Reactor coolant pumps trip at time of break.
2. Steam flow is isolated at time of break.
3. Main feedwater is isolated at time of break.
4. SI signal is generated at time of high containment pressure.
5. The delays following the SI signal for each of the pumped safety injection systems are given in Table 2.3.6.
6. Accumulators inject at the minimum accumulator set pressure.

2.3.2 RELAP4-EM HOT CHANNEL MODEL

This model is used for the determination of the thermal response of the hot rod during blowdown. The hot channel nodalization diagram for the chopped cosine axial power shape calculations is shown in Fig. 2.3.2. The nodalization of the hot rod heat structures may vary for other power shapes. Figure 2.3.2 shows that five fluid volumes are used to represent the average core, five fluid volumes for the hot channel and one fluid volume for each (inlet and outlet) plenum. Five heat structures are used to model the average core, five to model the hot assembly and twenty-four to model the hot rod. Crossflow between the average core and the hot channel is represented as required in 10 CFR 50, Appendix K. The present nodalization differs from EXEM/PWR (Ref. 2.1) in the number of core axial nodes. EXEM/PWR utilizes only three, while the present calculation uses five volumes. This is done in order to increase accuracy.

The lower and upper plenum volumes in the hot channel calculation are time dependent volumes. Their pressures and properties are read from a file containing their values.

This file is generated in a previously performed system blowdown calculation. The power level is also read from the system blowdown. All the initial conditions for the hot channel calculation are set identical to those of the corresponding system blowdown.

2.3.3 ACCUM-SIS MODEL

The objective of the ACCUM-SIS calculation is to determine the ECCS flow rates to the lumped intact loop cold leg and to the containment after EOBY.

The ACCUM-SIS calculation is essentially an application of RELAP4-EM. The nodalization diagram for this calculation is given in Fig. 2.3.3. The input is identical to that of the system volumes. The cold legs are time dependent volumes with pressures set by the previous blowdown calculation.

2.3.4 RFPAC MODELS

As previously described, RFPAC combines the four codes used to perform the refill and reflood thermal-hydraulic analyses (ICECON/CONTEMPT-LT, PREFILL, SHAPE/REFLOOD, and REFLEX) and eliminates the need for data transfer between codes. The input for each of these codes is described in detail in Ref. 2.15.

2.3.4.1 CONTAINMENT

ICECON/CONTEMPT-LT calculates the containment pressure response. The containment model is constructed so as to conservatively minimize containment pressure for the reflood calculations. The initial containment pressure is taken as 14.7 psia, temperature at 90° F, and relative humidity at 100%. The containment volume used is 3.063E6 ft^3. The spray system uses two spray pumps, so as to maximize containment heat removal. This model includes the maximum flow rates, minimum water temperature, and rated heat removal capacity for the fan coolers, which also maximizes containment heat removal.

2.3.4.2 PREFILL

The PREFILL code calculates (a) the time to beginning of reflood, (b) the ECCS injection flow rates for the refill analysis, and (c) the temperature at which ECCS fluid enters the core at the start of reflood. The transient specific input to this code is obtained from the RELAP4-EM blowdown results, the ACCUM-SIS results and ICECON results. The geometrical input involves a rearrangement of the information derived for the RELPAP4-EM system model.

2.3.4.3 SHAPE/REFLOOD

The SHAPE/REFLOOD calculation begins at BOCREC as determined by PREFILL. It uses the average core fuel and cladding temperatures from the RELAP4-EM hot channel calculation at EOBY to determine the average rod temperature at the peak power location at BOCREC time for use in the Fuel Cooling Test Facility (FCTF) reflood correlations. The power shape is transient specific; however, an evenly spaced 24 step axial profile is used.

2.3.4.4 REFLEX

The nodalization diagram for REFLEX is shown in Fig. 2.3.4. The present model uses 26 volumes and 24 junctions to represent the primary system. The REFLEX model is obtained by collapsing RELAP4-EM volumes as seen by comparison of Figures 2.3.1 and 2.3.4. The intact and broken loop secondary sides are represented by 3 and 2 volumes, respectively. The core bypass flow area is included in the downcomer annulus area for downcomer liquid level calculations as prescribed in Ref. 2.1. The angle between the cold leg and the ECCS line penetration is 45 degrees. The ECCS mixing pressure drop penalties for this case are 0.6 psi during accumulator injection and 0.15 psi afterwards.

2.3.5 TOODEE2 MODEL

TOODEE2 calculates the temperature distribution in the hot rod during refill and reflood. TOODEE2 calculations begin at end-of-bypass (EOBY). Only radiation heat transfer is allowed during the refill period. Only steam cooling is allowed downstream of the ruptured node following clad rupture for reflood rates less than one inch per second. Table 2.3.7 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 axial nodalization of the heat structures for the hot rod in the TOODEE2 model is identical to that of the hot rod in the RELAP4-EM Hot Channel model (Fig. 2.3.2). The TOODEE2 hot rod axial nodalization diagram for the chopped cosine axial power shape calculations is shown in Fig. 2.3.5. The nodalization may vary for other power shapes. Different axial nodalizations are discussed in the sections describing the calculations to which they apply.

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 clad outer radius. The cladding is divided into 2 radial rings as required by EXEM/PWR. The radial nodalization scheme is shown in Fig. 2.3.6.

TABLE 2.3.1

CPSES-1 NSSS Nodalization Summary

Component Description	Volume No.	
Downcomer	27, 28	
Lower Plenum	29, 50, 51	
Average Core	30 to 34	
Hot Assembly	35 to 39	
Core Bypass	40	
Upper Head	1	
Upper Core	52	
Upper Plenum	2	
Guide Tubes	53	
Containment	49	
	Intact Loop	**Broken Loop**
RCPs	12	24
Hot Leg	3	15
Intermediate Leg	10, 11	22, 23
Cold Leg	13, 14	25, 26
S/G - Primary	4 to 9	16 to 21
S/G - Secondary	47	48
Accumulator	43	45
SI Discharge Line	44	46
Pressurizer	41	
Surge Line	42	
Total = 53		

Heat Conductor Description	Conductor No.
Average Core	5
Hot Assembly	5
S/G per loop	4
Containment	5
RCS Piping	31
Total = 50	

Fill Junction Description	Junction No. Intact Loop	Junction No. Broken Loop
Centrifugal Charging Pumps	69	70
Safety Injection Pumps	71	72
Low Pressure Injection Pumps	73	74
Main Feedwater	65	66
Auxiliary Feedwater	67	68
Steam Line Valve	64	63
Total = 12		

TABLE 2.3.2

SUMMARY OF CPSES-1 RELAP4-EM SYSTEM MODEL VOLUMES

VOLUME NUMBER	REGION DESCRIPTION	VOLUME (FT3)	VOLUME LENGTH (FT)	FLOW AREA (FT2)	HYDRAULIC DIAMETER (FT)	ELEV. (FT)
01	UPPER HEAD	892.2414	9.8400	90.6749	1.9476	30.975
02	UNDER PLENUM	672.7352	7.9750	1.0+06	1.5991	23.0000
52	UPPER CORE	74.6550	1.2769	1.0+06	1.0704	21.7231
53	GUIDE TUBES	220.3825	13.2900	16.0829	0.3372	23.0000
03	HOT LEG	298.1295	3.6457	13.7607	2.5282	25.7083
04	SG INLET	538.6653	7.9114	68.0871	5.3756	27.6802
05	SG TUBES	422.7393	13.4737	31.3752	0.0553	35.5916
06	SG TUBES	422.7393	14.5852	31.3752	0.0553	49.0653
07	SG TUBES	422.7393	14.5852	31.3752	0.0553	49.0653
08	SG TUBES	422.7393	13.4737	31.3752	0.0553	35.5916
09	SG OUTLET	538.6653	7.9114	68.0871	5.3756	27.6802
10	INTERM. LEG	231.7245	5.7917	15.7242	2.5833	15.3125
11	INTERM. LEG	166.4985	4.0415	15.7242	2.5833	15.3125
12	PUMP	5.8000	7.3615	32.0316	3.6871	21.1042
13	COLD LEG	5.0970	2.2917	12.3741	2.2917	25.7709
14	COLD LEG	5.0970	2.2917	12.3741	3.0124	25.7709
15	HOT LEG	99.3765	3.6457	4.5869	2.5282	25.7083
16	SG INLET	179.5551	7.9114	22.6957	5.3756	27.6802
17	SG TUBES	140.9131	13.4737	10.4584	0.0553	35.5916
18	SG TUBES	140.9131	14.5852	10.4584	0.0553	49.0653
19	SG TUBES	140.9131	14.5852	10.4584	0.0553	49.0653
20	SG TUBES	140.9131	13.4737	10.4584	0.0553	35.5916
21	SG OUTLET	179.5551	7.9114	22.6957	5.3756	27.6802
22	INTERM. LEG	77.2415	14.0415	5.2414	2.5833	15.3125
23	INTERM. LEG	55.4995	5.7917	5.2414	2.5833	15.3125

TABLE 2.3.2 (Continued...)

SUMMARY OF CPSES-1 RELAP4-EM SYSTEM MODEL VOLUMES

VOLUME NUMBER	REGION DESCRIPTION	VOLUME (FT3)	VOLUME LENGTH (FT)	FLOW AREA (FT2)	HYDRAULIC DIAMETER (FT)	ELEV. (FT)
24	PUMP	78.6000	7.3615	10.6772	3.6871	21.1042
25	COLD LEG	51.6990	2.2917	4.1247	2.2917	25.7709
26	COLD LEG	51.6990	2.2917	4.1247	3.0124	25.7709
27	UPPER DOWNCOMER	392.6160	14.0000	1.0+06	1.4145	19.9167
28	LOWER DOWNCOMER	479.1362	14.3333	35.6714	1.6363	5.5834
50	LOWER HEAD	120.2742	2.5126	47.8684	3.3216	-0.4292
51	LOWER PLENUM	460.6664	3.5000	1.0+06	5.1105	2.0834
29	L CORE SUPT PLT	335.9651	4.1397	1.0+06	0.0691	5.5834
30	CORE 1 AVG	122.0097	2.4000	50.8738	0.0363	9.7231
31	CORE 2 AVG	122.0097	2.4000	50.8738	0.0363	12.1231
32	CORE 3 AVG	122.0097	2.4000	50.8738	0.0363	14.5231
33	CORE 4 AVG	122.0097	2.4000	50.8738	0.0363	16.9231
34	CORE 5 AVG	122.0097	2.4000	50.8738	0.0363	19.3231
35	CORE 1 HOT	0.6350	2.4000	0.2646	0.0365	9.7231
36	CORE 2 HOT	0.6350	2.4000	0.2646	0.0365	12.1231
37	CORE 3 HOT	0.6350	2.4000	0.2646	0.0365	14.5231
38	CORE 4 HOT	0.6350	2.4000	0.2646	0.0365	16.9231
39	CORE 5 HOT	0.6350	2.4000	0.2646	0.0365	19.3231
40	BYPASS	298.5298	13.3750	22.3200	0.7762	9.3750
41	PRESSURIZER	1836.2393	30.5397	36.7823	6.8434	55.3308
42	PZR SURGE LINE	46.6806	27.8893	0.6827	0.9323	27.4415
43	ACCUMULATOR IL	4050.0000	10.8152	226.9008	9.8132	33.5775
44	DISCH LINE IL	95.4600	7.8067	1.2528	0.7292	25.7709
45	ACCUMULATOR BL	1350.0000	10.8152	75.6336	9.8132	42.9908
46	DISCH LINE BL	40.0400	17.2200	0.4176	0.7292	25.7709
47	STEAM GENERATOR	17862.0000	41.8300	169.3512	0.1234	35.5916
48	STEAM GENERATOR	5954.0000	41.8300	56.4504	0.1234	35.5916
49	CONTAINMENT	3.063+06	299.00	10244.1500	114.21	-31.0000

TABLE 2.3.2 (Continued...)

SUMMARY OF CPSES-1 RELAP4-EM SYSTEM MODEL JUNCTIONS

JUNCTION NUMBER	JUNCTION LOCATION	ELEV (FT)	L/A (FT-1)	AREA (FT2)	FORWARD LOSS COEF	REVERSE LOSS COEF	HYDRAULIC DIAMETER
57	DWNCMR/UHEAD	33.9167	0.1898	0.6981	1.4946	1.4722	0.1667
01	UHEAD/GUIDE	36.2900	0.4550	0.5199	6.84023	7.09124	0.4617
60	UPCORE/GUIDE	23.0000	0.4748	11.9831	0.7321	0.6669	3.9061
61	UPCORE/UPLNM	23.0000	0.0842	28.8708	1.7018	1.4852	6.0629
62	GUIDE/UPLNM	24.2391	0.5103	11.5647	1.34902	1.34902	3.8373
02	UPLENUM/HL	26.9167	0.7834	13.7607	0.2424	0.4844	2.4167
03	HL/SG	28.5238	0.8239	13.7607	0.3292	0.2272	2.4167
04	SG/TUBES	35.5916	0.2728	31.3752	1.8828	2.6029	3.6491
05	TUBES/TUBES	49.0653	0.4294	31.3752	1.0-07	1.0-07	3.6491
06	TUBES/TUBES	61.4666	0.4294	31.3752	4.48907	4.48907	3.6491
07	TUBES/TUBES	49.0653	0.4294	31.3752	1.0-07	1.0-07	3.6491
08	TUBES/SG	35.5916	0.2728	31.3752	2.6029	1.8828	3.6491
09	SG/IL	28.5238	0.5267	15.7242	0.4485	0.5419	2.5833
10	IL/IL	16.6042	0.8053	15.7242	1.0-07	1.0-07	2.5833
11	IL/RCP	21.1042	0.4905	15.7242	0.1591	0.1591	2.5833
12	RCP/CL	26.9167	0.6602	12.3741	1.0-07	1.0-07	2.2917
13	CL/CL	26.9167	0.9239	12.3741	1.0-07	1.0-07	2.2917
14	CL/DWNCMR	26.9167	0.4541	12.3741	1.29431	0.46451	2.2917
15	UPLENUM/HL	26.9167	2.3502	4.5869	0.2424	0.4844	2.4167
16	HL/SG	28.5238	2.4716	4.5869	0.3292	0.2272	2.4167
17	SG/TUBES	35.5916	0.8185	10.4584	1.8828	2.6029	3.6491
18	TUBES/TUBES	49.0653	1.2883	10.4584	1.0-07	1.0-07	3.6491
19	TUBES/TUBES	61.4666	1.2883	10.4584	4.48907	4.48907	3.6491
20	TUBES/TUBES	49.0653	1.2883	10.4584	1.0-07	1.0-07	3.6491
21	TUBES/SG	35.5916	0.8185	10.4584	2.6029	1.8828	3.6491
22	SG/IL	28.5238	1.5801	5.2414	0.4485	0.5419	2.5833
23	IL/IL	16.6042	2.4159	5.2414	1.0-07	1.0-07	2.5833
24	IL/RCP	21.1042	1.4714	5.2414	0.1591	0.1591	2.5833
25	RCP/CL	26.9167	1.9806	4.1247	1.0-07	1.0-07	2.2917
26	BREAK VALUE	26.9167	2.7716	4.1247	1.0-07	1.0-07	2.2917
27	CL/DWNCMR	26.9167	1.4523	4.1247	1.29431	0.46451	2.2917
28	U/L DWNCMR	19.9167	0.4010	35.6714	1.0-07	1.0-07	6.7393
29	DWNCMR/LPLN	5.5834	0.2702	26.6891	0.3552	0.0826	5.8294
58	LHEAD/LPLNM	2.0834	0.0402	82.0641	0.0000	0.0000	10.2219
59	LPLNM/LCSP	5.5834	0.1327	49.9264	0.6628	0.6960	7.9730

TABLE 2.3.2 (Continued...)

SUMMARY OF CPSES-1 RELAP4-EM SYSTEM MODEL JUNCTIONS

JUNCTION NUMBER	JUNCTION LOCATION	ELEV (FT)	L/A (FT-1)	AREA (FT2)	FORWARD LOSS COEF	REVERSE LOSS COEF	HYDRAULIC DIAMETER
30	LCSP/1AVG	9.7231	0.0418	50.8738	4.5720	5.0613	8.0483
31	1/2AVG	12.1231	0.0472	50.8738	1.4020	1.4020	8.0483
32	2/3AVG	14.5231	0.0472	50.8738	1.4020	1.4020	8.0483
35	5AVG/UPCR	21.7231	0.0361	50.8738	1.4020	1.4020	8.0483
33	3/4AVG	16.9231	0.0472	50.8738	1.4020	1.4020	8.0483
[illegible]	4/5AVG	19.3231	0.0472	50.8738	1.4020	1.4020	8.0483
36	LCSP/1HOT	9.7231	4.5533	0.2646	4.5720	5.0613	0.5804
37	1/2HOT	12.1231	9.0703	0.2646	1.4020	1.4020	0.5804
38	2/3HOT	14.5231	9.0703	0.2646	1.4020	1.4020	0.5804
39	3/4HOT	16.9231	9.0703	0.2646	1.4020	1.4020	0.5804
40	4/5HOT	19.3231	9.0703	0.2646	1.4020	1.4020	0.5804
41	5HOT/UPCR	21.7231	4.5476	0.2646	1.00065	0.91189	0.5804
42	LCSP/BYPSS	9.3750	0.2870	5.3294	43.3410	46.2161	2.6049
43	BYPSS/UPCR	22.7500	0.2813	3.7661	21.8091	22.1510	2.1898
44	CROSSFLW 1	10.9231	0.4167	1.6592	9.5220	9.5220	1.4535
45	CROSSFLW 2	13.3231	0.4167	1.6592	9.5220	9.5220	1.4535
46	CROSSFLW 3	15.7231	0.4167	1.6592	9.5220	9.5220	1.4535
47	CROSSFLW 4	18.1231	0.4167	1.6592	9.5220	9.5220	1.4535
48	CROSSFLW 5	20.5231	0.4167	1.6592	9.5220	9.5220	1.4535
49	PRZR/SURGE	55.3308	50.7563	0.6827	0.8675	1.3377	0.9323
50	SURGE/HL	27.7711	50.8541	0.6827	0.7017	3.2479	0.9323
51	AT/ATDL	33.5775	30.4552	1.2528	3.9754	3.9754	0.7292
52	ATDL/CL	25.7709	30.9223	1.2528	2.4044	2.4044	0.7292
53	AT/ATDL	42.990	114.9168	0.4176	4.0102	4.0102	0.7292
54	ATDL/CL	25.770	116.3182	0.4176	2.4044	2.4044	0.7292
55	CL/CNTNMNT	26.9167	1.5340	4.1247	1.00	0.50	2.2917
56	CL/CNTNMNT	26.9167	1.2668	4.1247	0.50	1.00	2.2917
65	MFW FILL	40.5916	0.0000	3.0000	0.0000	0.0000	1.1284
66	MFW FILL	40.5916	0.0000	1.0000	0.0000	0.0000	1.1284
67	AUX FILL	73.5916	0.0000	3.0000	0.0000	0.0000	1.1284
68	AUX FILL	73.5916	0.0000	1.0000	0.0000	0.0000	1.1284
69	CCP/FILL	26.6873	0.0000	3.0000	0.0000	0.0000	1.1284
70	CCP/FILL	26.7913	0.0000	1.0000	0.0000	0.0000	1.1284
71	HHP/FILL	26.6873	0.0000	3.0000	0.0000	0.0000	1.1284
72	HHP/FILL	26.7913	0.0000	1.0000	0.0000	0.0000	1.1284
73	RHR/FILL	26.6873	0.0000	3.0000	0.0000	0.0000	1.1284
74	RHR/FILL	26.7913	0.0000	1.0000	0.0000	0.0000	1.1284
64	TSV FILL	95.7583	0.0000	3.0000	0.0000	0.0000	1.1284
63	TSV FILL	95.7583	0.0000	1.0000	0.0000	0.0000	1.1284

TABLE 2.3.3

DENSITY REACTIVITY TABLE

NORMAL DENSITY	REACTIVITY ($)
0.01	-54.65
0.1422	-32.46
0.2845	-17.63
0.4267	-9.38
0.5690	-4.56
0.7112	-1.82
0.8535	-0.47
1.0000	0.00
1.0669	0.15
1.1380	0.30
1.4225	0.60

TABLE 2.3.4

DOPPLER REACTIVITY TABLE

TEMPEATURE (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 2.3.5

ECCS FLOW VS. PRESSURE

RCS PRESSURE (psia)	CCP (1) (lblm/sec)	HPSI (1) (lbm/sec)	RHR (1) (lbm/sec)	TOTAL (lbm/sec)
0.0	13.70	20.26	131.13	165.09
14.7	13.70	20.26	131.13	165.09
34.7	13.58	20.13	123.27	156.98
54.7	13.47	19.99	114.80	148.26
114.7	13.13	19.60	34.60	67.33
154.7	12.90	19.22	0.00	32.12
214.7	12.55	18.66		31.21
414.7	11.37	16.79		28.16
614.7	10.13	14.53		24.66
1014.7	7.45	8.57		16.02
1614.7	2.77	0.00		2.77
2814.7	0.00			0.00

TABLE 2.3.6

TIME DELAY FOR EACH SYSTEM

ACTION	TIMF DELAY AFTER SI SETPCINT REACHED (sec,
SI actuation signal	2
Charging pumps up to speed	17 (Fill Table 1 initiated)
HPSI pumps up to speed	22 (Fill Table 2 initiated)
RHR pumps up to speed	27 (Fill Table 3 initiated)
Containment Spray	34

TABLE 2.3.7

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

FIG. 2.2.1
SCHEMATIC REPRESENTATION OF THE EXEM/PWR
ECCS EVALUATION MODEL

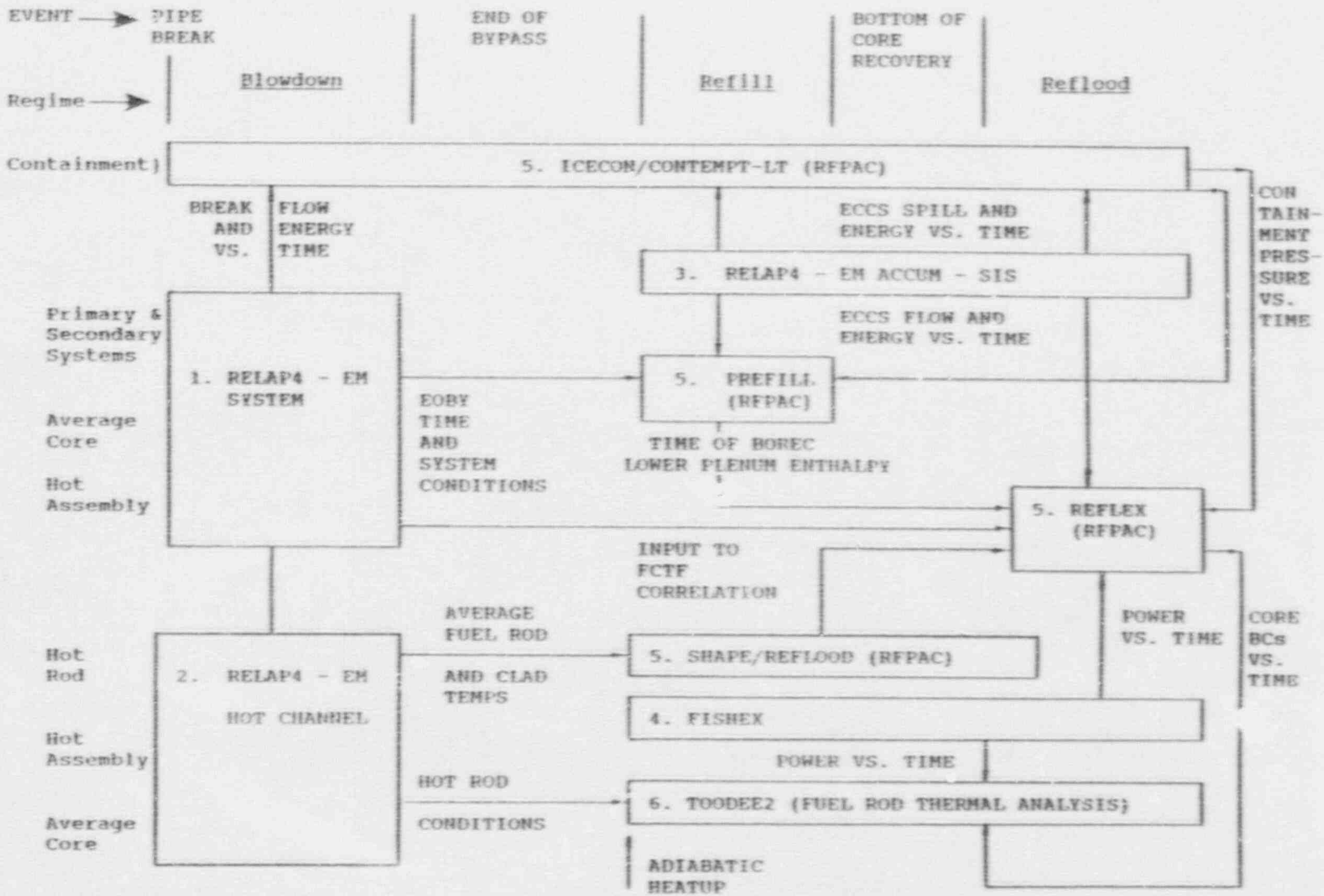


FIGURE 2.3.1

CPSES1 RELAP4 SYSTEM BLOWDOWN MODEL

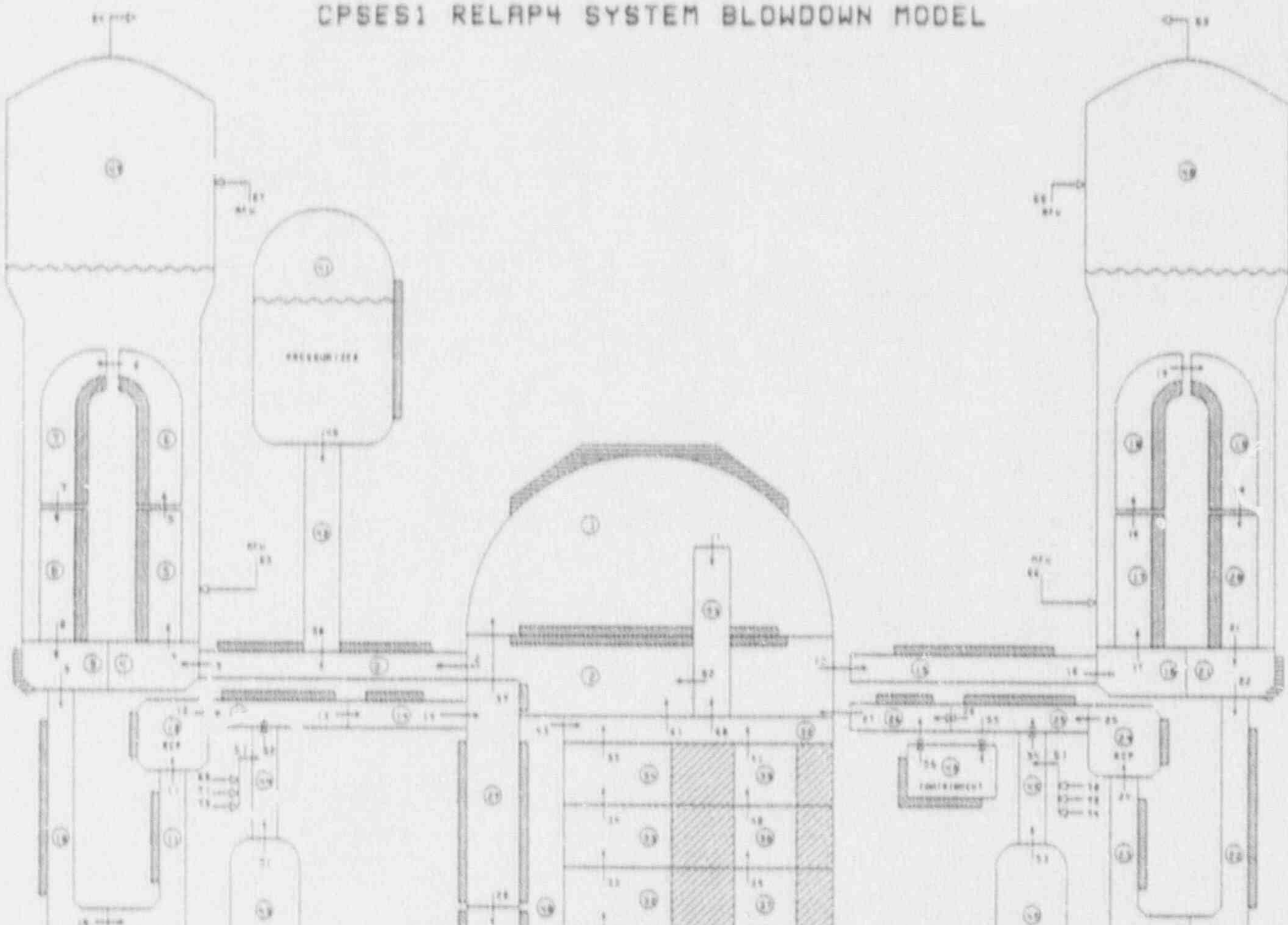
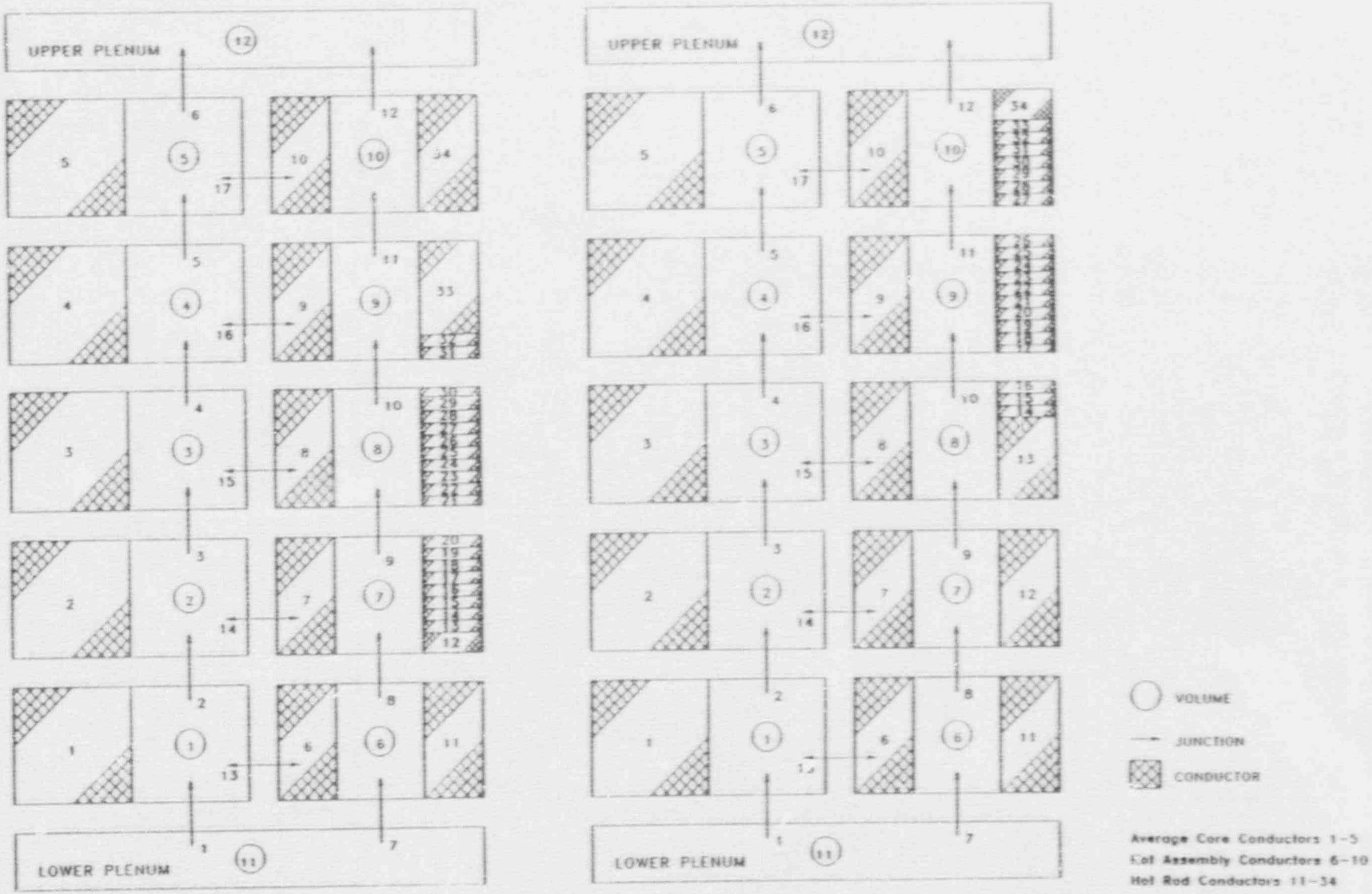


FIGURE 2.3.2 RELAP4 HOT CHANNEL NODALIZATION



A) Chopped Cosine

B) Top Skewed

FIGURE 2.3.3
ACCUM-SIS NODALIZATION

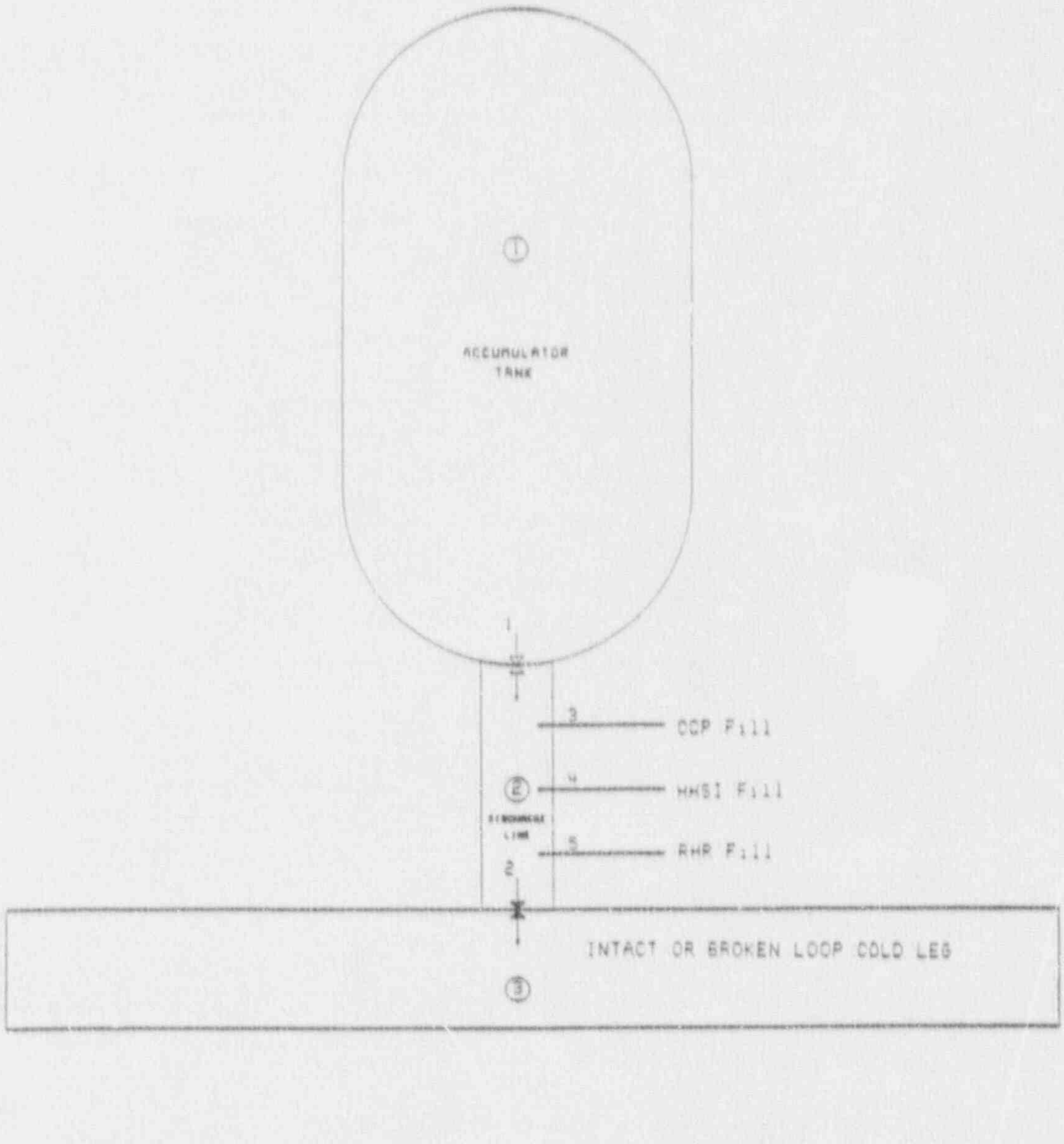


FIGURE 2.3.4

REFLEX NODALIZATION FOR CPSES UNIT 1

(W/O SPRAY NOZZLES)

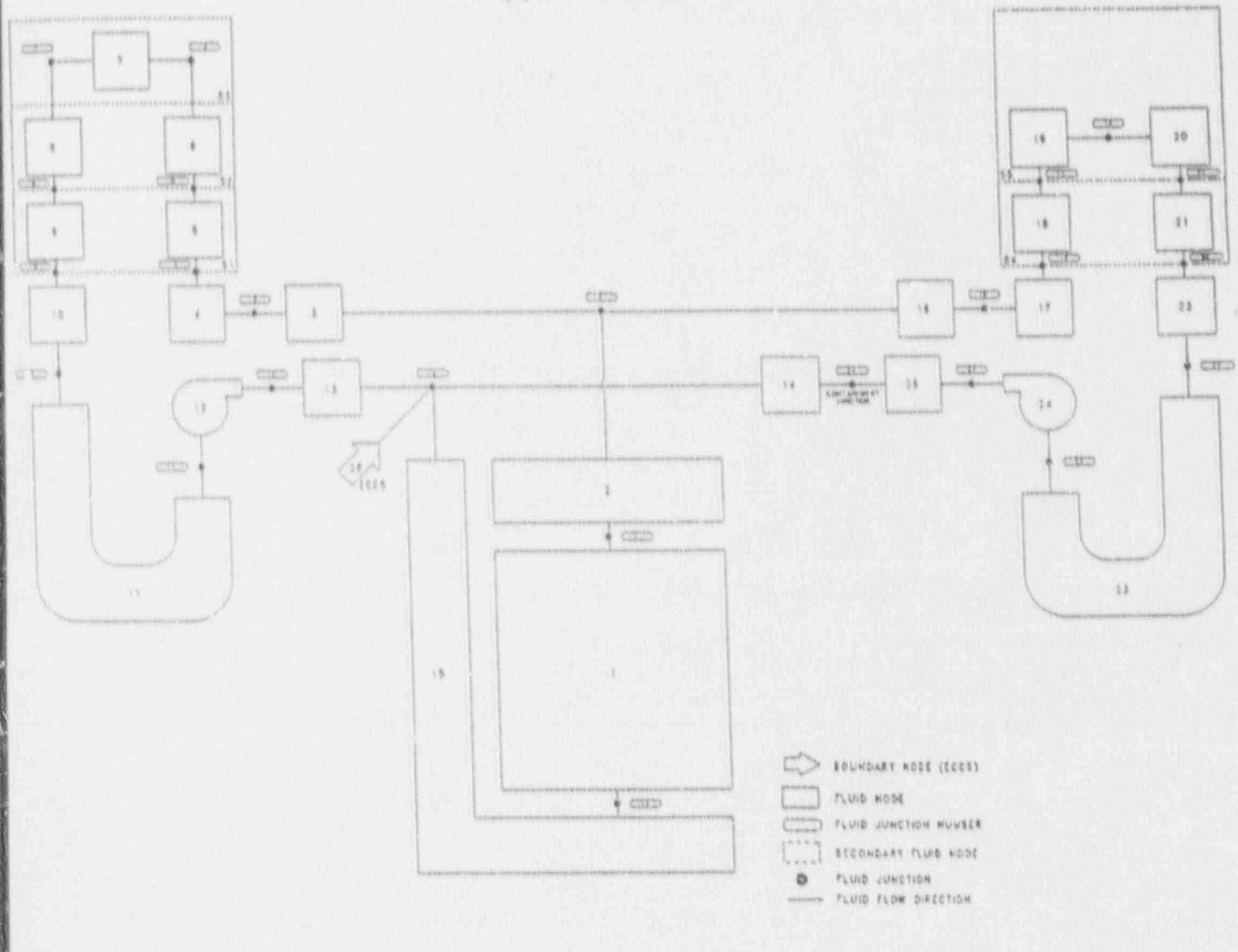


FIGURE 2.3.5 TOODEE2 HOT ROD NODALIZATION

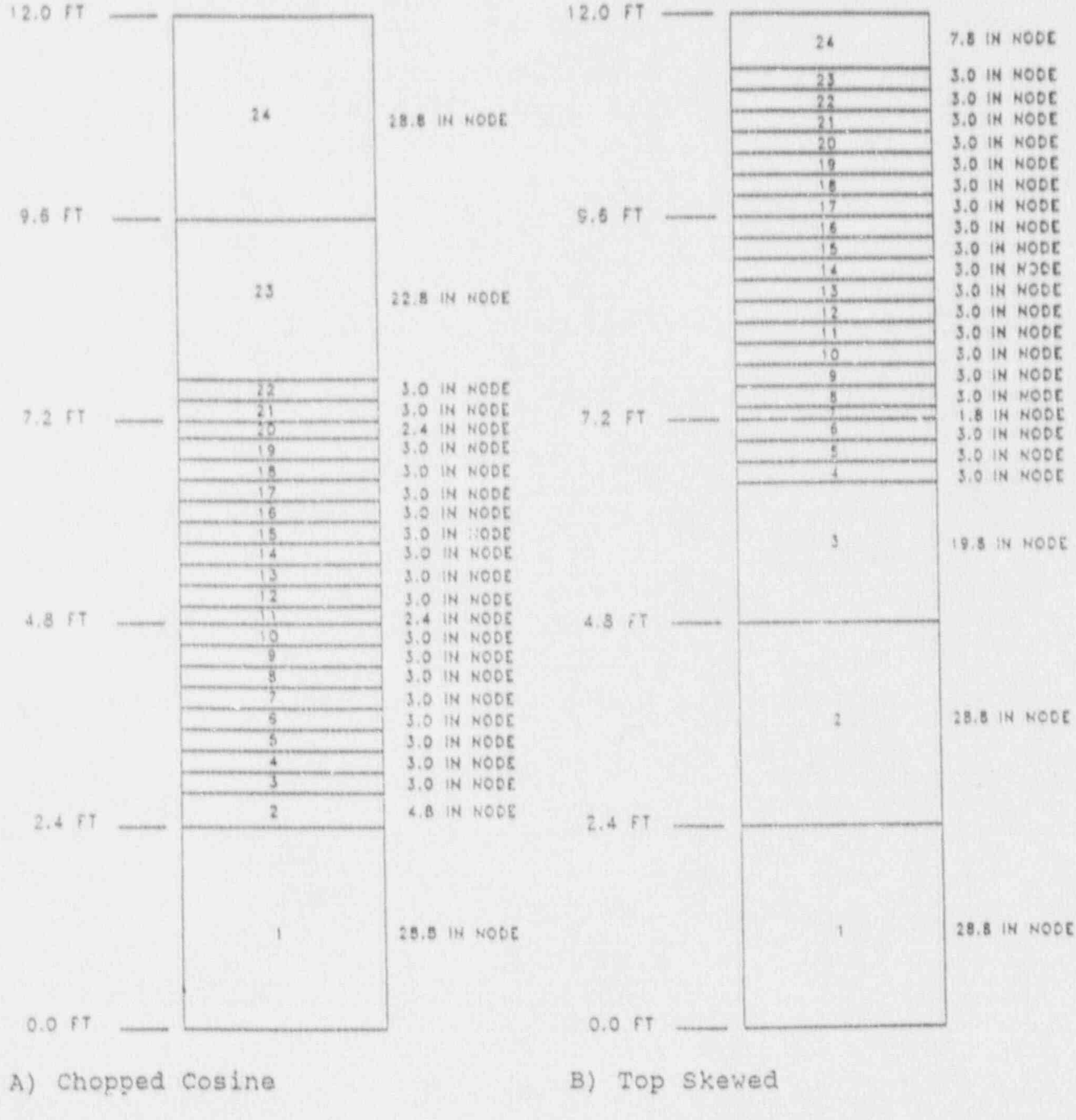


Fig. 2.3.6

TOODEE2 Radial Nodalization

```
Fuel Centerline
|———> Radial Direction

|-2-|-3-|-4-|-5-|-6-|-7-|-8-|-9-- + gap |-10-|-11-| Node No.
1   2   3   4   5   6   7   8           9    10   11 Gridline No
```

CHAPTER 3

BASE CASE ANALYSIS AND SENSITIVITY STUDIES

10 CFR 50, Appendix K requires the investigation of the impact of variations in several method- and plant-specific issues on the LOCA consequences.

The method-specific parameters requiring investigation are (a) nodalization and (b) time step. Such studies are conducted for methodology development and approval. The present work constitutes an application of an approved methodology using time step and nodalization as prescribed therein. Hence, the effect of variations in these parameters within the bounds of methodology recommendations has already been ascertained to be negligible, and sensitivity studies for these variables are not repeated here.

According to 10 CFR 50, Appendix K, the plant-specific issues which must be examined are (a) break spectrum (location, size, and type), (b) axial power shape, and (c) fuel type and exposure. These are the sensitivity studies examined in this chapter.

3.1 BASE CASE ANALYSIS

This section presents licensing analysis results for a Double-Ended Guillotine (DEG) break in the discharge line of the Reactor Coolant Pump. The chopped cosine axial power shape used for this base case is shown in Fig. 3.1. The fuel rod exposure which maximizes stored energy is calculated by RODEX and occurs at 613.8 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.

The major assumptions are that a DEG break occurs at 0.05 seconds with coincident loss of offsite power. The initial power level 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. ECCS injection into the broken loop is lost, and is postulated to spill directly to the containment. One pumped injection train is assumed lost due to failure of a diesel generator to start. This is the postulated single failure as required by 10 CFR 50, Appendix K. Thus, one high head centrifugal charging pump, one intermediate head safety injection pump and one low pressure high flow residual heat removal pump along with three accumulators are available to mitigate the accident. Containment pressure is minimized in accordance with Branch Technical Position CSB 6-1 (Ref. 3.2), "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Minimization of containment pressure is done by minimizing initial pressure and temperature and maximizing free volume and heat sinks. Furthermore, containment safeguards are also assumed to function as designed while consistent with the single failure; i.e., only one train of containment sprays is available. The other is taken out by the diesel-generator failure postulated above. The fan coolers are disabled on the SI signal as per design.

Five percent of the steam generator tubes are assumed plugged for this analysis. This assumption is made to support the potential need for operation under these circumstances and is a conservative assumption for fewer obstructed tubes.

Table 3.4 summarizes the timing of significant events for this case. This table should assist in the review of the following figures, which present key results.

Figures 3.2 and 3.3 show reactor power and net reactivity following the accident during the system blowdown phase. The reactor power decreases rapidly due to negative reactivity from core voiding. Between 4 and 5 seconds the power spikes mildly because of an increase in reactivity, which in turn is caused by an increase in the liquid fraction in the center of the core (Fig. 3.6). The increase in power results from a temporary coolant accumulation in that region, which is associated with a second flow reversal this time towards the normal flow direction, following the first reversal caused by the cold leg break location (Figs. 3.4 and 3.5). Beyond this time, core power follows the RELAP4-EM decay heat values (which represents 1.2 times 1971 ANS Decay Heat Standard).

Figures 3.6 and 3.7 show mid-core average quality and core inlet subcooling, respectively. Both figures indicate that core flashing takes place around 1 second. Again the quality falls between 4 and 10 seconds due to the flow reversal discussed above and evidenced in Figs. 3.4 and 3.5. Shortly after accumulator injection (at 15 seconds, Fig. 3.13) the mid-core quality drops quickly. The quality increases back to 1.0 at 24 seconds.

Figure 3.8 shows the downcomer liquid inventory. The downcomer remains full until 4 seconds. As shown in Fig.3.4, this time period corresponds to the period of flow reversal caused by the break. After that the downcomer is quickly depleted until 3 seconds after the accumulators begin to inject, when it once again begins to fill quickly.

Figure 3.9 shows the total break flow. The flow rapidly accelerates to two-phase critical flow (Moody model) in less than 0.1 second at the pump discharge. Rapid depressurization and flashing limit the initial break flow rates. The break flow rate gradually diminishes as volumes upstream of the break become void.

Fig. 3.10 and 3.11 show system and containment pressures respectively. Superimposed on the primary pressure is the seconday pressure showing that the heat transfer direction is reversed at approximately 8.0 seconds. The containment pressure peaks to about 36 psia, 19 seconds into the blowdown. The pressure turns around at this time due to steam condensation on equipment and concrete surfaces.

Containment spray comes into play only at approximately 34 seconds, injecting at a constant rate thereafter (Fig. 3.12).

ECCS flow rates are presented in Figs. 3.13 through 3.15. The accumulators begin to inject at 15 seconds and are empty at 44 seconds. The available centrifugal charging pump begins to discharge at 18 seconds and the intermediate head safety injection pump at 23. The low pressure injection system comes on at approximately 28 seconds.

Figure 3.16 shows the heat transfer coefficient at the peak clad temperature (PCT) node. Heat transfer is abruptly degraded as the core flashes at approximately one second into the accident. The blowdown clad temperatures at the PCT node are presented in Fig 3.17.

The core flooding rates are shown in Fig. 3.18. The flooding rate does not drop below one inch per second until 100 seconds. The PCT time is approximately 60 seconds.

The metal reaction depth at the hot spot is shown in Fig. 3.19.

The PCT node clad temperature history is shown in Fig. 3.20. The PCT is calculated to be 1959 °F in node 10 (Fig. 2.3.5), 4.7 ft above the bottom of the core. It is coincident with the ruptured node.

3.2 SENSITIVITY STUDIES

3.2.1 BREAK SPECTRUM

The most limiting break location has been determined in previous studies for this (Ref. 3.1) and other similar plants (Ref. 3.3) to be in the cold leg at the reactor coolant pump discharge. This determination results primarily from the loss of ECCS flow to the core associated with it. 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 methodology, the break size is the first sensitivity issue addressed, holding constant the axial power shape and the fuel exposure. The rationale for addressing break size first is that system thermal-hydraulic behavior during the blowdown period is largely affected by break size but is nearly independent of power shape and fuel exposure. Therefore the most limiting size for this shape and exposure will also be the most limiting size for other shapes and exposures.

The break spectrum study is conducted first for the guillotine type break with chopped cosine power shape and beginning of life (BOL) fuel. The reason for performing the break spectrum calculations with the other two parameters fixed to the values outlined above is that the large break LOCA analysis of record (Ref. 3.1) shows the most limiting break as a Double-Ended Guillotine type with chopped cosine axial power shape, and BOL fuel.

Three break sizes are examined by giving to the break discharge coefficient the values of 1.0 (base case, Section 3.1), 0.8 and 0.6, respectively.

Split type breaks are analyzed following the guillotine-type breaks. These analyses are expected to yield lower peak clad temperatures and are done to confirm this expectation for CPSES-1. Therefore, only the 1.0 discharge coefficient is examined for the longitudinal splits. It is noted that in EXEM/PWR the split break area is twice the ma.imum pipe area, as in the DEG.

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 Double-Ended Guillotine with a 1.0 discharge coefficient located in the main coolant pump discharge. Future studies will be performed using 1.0 as the limiting discharge coefficient.

3.2.1.1 DEG CD=1.0

This is the base case calculation described in Section 3.1. The PCT is calculated to be 1959 °F in node 10 at 4.7 ft above the bottom of the core.

3.2.1.2 DEG CD=0.8

The results of this calculation are quite similar to those of the base case (DEG CD=1.0, Section 3.1), during the various stages of the thermal-hydraulic analysis. However, during the fuel rod thermal analysis, the PCT node and the ruptured node do not coincide for this calculation, as shown in Fig. 3.21. The PCT is calculated to be 1870 °F in node 21 (7.3 ft above the bottom of the core per Fig. 2.3.5) and the ruptured node is node 16 (at 6.1 ft).

3.2.1.3 DEG CD=0.6

This calculation is nearly identical to the one discussed above (DEG CD=0.8). The PCT node and the ruptured node do not coincide for this calculation either, as shown in Fig. 3.22. The PCT is calculated to be 1768 °F in node 21 in this case (7.3 ft above the bottom of the core per Fig. 2.3.5) but the ruptured node is node 10 (at 4.7 ft).

3.2.1.4 SPLIT CD=1.0

The longitudinal split break calculation shows results similar to both the 0.8 and the 0.6 DEG. The PCT node and the ruptured node do not coincide for this calculation either, as shown in Fig. 3.23. The PCT is calculated to be 1901 °F in node 21 in this case (7.3 ft above the bottom of the core per Fig. 2.3.5) and the ruptured node is node 10 (at 4.7 ft).

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 (DEG CD=1.0, Section 3.2.1) and at the burnup yielding the highest stored energy. The maximum stored energy occurs at 613.8 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 3.5. For that analysis a prescribed series of load follow cases are modelled 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 Specification 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 Specifications (Ref. 3.4) limit at that location. 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 the most limiting power shape is the profile which peaks at 8.75 ft. shown in Figure 3.1. This result will be used in all other studies in the future.

3.2.2.1 <u>TOP PEAKED AT 8.75 FT AND CD=1.0</u>

As expected, there is minimal difference between blowdown results for this calculation and those for the chopped cosine and CD=1.0. Even clad temperatures for this period are nearly identical, as can be seen by comparison of Figs. 3.24 and 3.17. The ruptured node and the PCT node coincide in this calculation as well (Fig. 3.25) and correspond to node 13 (8.7 ft above the bottom of the core per Fig. 2.3.5). The calculated PCT is 2034 °F.

3.2.2.2 <u>TOP PEAKED AT 9.75 AND CD=1.0</u>

The PCT for this calculation is 1983 °F and occurs at node 22 (11.0 ft above the bottom of the core per Fig. 2.3.5). The rupture occurs at node 17 (9.7 ft above the bottom of the core per Fig. 2.3.5). Figure 3.26 shows the clad temperatures for this case.

3.2.3 <u>EXPOSURE</u>

The exposure study is done to support operation to EOC burnup levels. It is done because pin pressure increases with exposure, and higher pin pressures increase the driving force for rod burst, with the attendant effect of raising peak clad temperatures. It should be noted, however, that the stored energy effect tends to dominate the pin pressure effect so that a lower peak clad temperature is expected at EOC. Nevertheless, this study is done to confirm that that is indeed the case for the fuel under consideration.

The sequence of events for the burnup study is summarized in Table 3.7.

The clad temperatures are shown in Fig. 3.27.

The conclusion from the burnup study is that all burnups are bounded by the beginning of cycle (613.8 hours exposure) condition for the present fuel, since the two extremes (maximum stored energy and maximum pin pressure) have been examined. This exposure (613.8 hours) will be used in future studies unless fuel changes warranting a re-evaluation of this assumption occur.

TABLE 3.1

SUMMARY OF CPSES-1 LARGE BREAK LOCA ACCIDENT
ASSUMPTIONS FOR BASE CASE AND SENSITIVITY STUDIES

1. The initial power level is 104.5% above the 1.02 (calorimetric error factor) x 3411 MWt (i.e., 3636 MWt).

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

3. Reactor coolant pump discharge line break occurs at 0.05 seconds.

4. Loss of offsite power occurs coincident with break at 0.05 seconds.

5. Failure of one diesel generator to start takes out one high head centrifugal charging pump, one intermediate head safety injection pump and one low head high flow residual heat removal pump. This is the single failure assumption as required by 10 CFR 50, Appendix K.

6. No credit is given for reactor scram.

7. Three accumulators inject into intact loops on demand.

8. One high head centrifugal charging pump, one intermediate head safety injection pump and one low head high flow residual heat removal pump inject on demand after the appropriate delays.

9. Containment pressure is minimized in accordance with branch Technical Position CSB 6-1 (Ref. 2.15), "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Minimization of containment pressure is done by minimizing initial pressure and temperature and maximizing free volume and heat sinks. Furthermore, containment safeguards are also assumed to function as designed while consistent with the single failure; i.e., only one train of containment sprays is available. The other is taken out by the diesel-generator failure postulated in assumption 5 above. The fan coolers are disabled on the SI signal as per design.

10. Passive heat structures are included.

11. No credit is given for Auxiliary Feedwater.

TABLE 3.2

SUMMARY OF INITIAL CONDITIONS FOR CPSES-1
LARGE BREAK LOCA BASE CASE AND SENSITIVITY STUDIES

DESCRIPTION	VALUE
Core Power	3636 MWt
Power Upgrade Multiplier	1.045
Power Calorimetric Uncertainty Multiplier	1.02
Reactor Coolant Pump Heat	20 MWt (approx)
Power Shapes Analyzed	Chopped Cosine
	Top skewed @ 8.75 ft
	Top skewed @ 9.75 ft
Peak Linear Power (includes 102% factor)	
Base Case (Fig. 3.1)	13.16 KW/ft
Top Peaked at 8.75 ft (Fig. 3.1)	12.71 KW/ft
Top Peaked at 9.75 ft (Fig. 3.1)	12.54 KW/ft
Total Peaking Factor, F^T_Q	
Base Case (Fig. 3.1)	2.32
Top Peaked at 8.75 ft (Fig. 3.1)	2.24
Top Peaked at 9.75 ft (Fig. 3.1)	2.21
Accumulator Water Volume	6119 gals/Accum
Accumulator Cover Gas Pressure	623 psig
Accumulator Water Temperature	90 ^{o}F
Safety Injection Pumped Flow	Table 2.3.4
Containment Parameters	Table 3.1, Item 9
Refueling Water Storage Tank Temperature	40 ^{o}F
Initial Loop Flow	9743 lbm/sec
Vessel Inlet Temperature	559.9 ^{o}F
Vessel Outlet Temperature	622.8 ^{o}F
Reactor Coolant Pressure	2250 psia
Steam Pressure	940 psia
Steam Generator Tube Plugging Level	5%
Fuel Parameters	Cycle 1, Table 3.3

TABLE 3.3

SUMMARY OF FUEL PARAMETERS FOR BOC and EOC
LARGE BREAK LOCA ANALYSIS

PARAMETERS			
Fuel Rod Geometry Data	Table 2.3.7		
	Beginning of Cycle (BOC)		
Time to Maximum Stored Energy Exposure	613.8 hours		
Fuel Rod Composition:			
	Average Core	Hot Assembly	Hot Rod
Gram Moles	0.02326	0.02327	0.02327
Helium fraction	0.96796	0.96793	0.96793
Argon fraction	0.00000	0.00000	0.00000
Hydrogen fraction	0.00000	0.00000	0.00000
Nitrogen fraction	0.03200	0.03200	0.03200
Krypton fraction	0.00001	0.00001	0.00001
Xenon fraction	0.00004	0.00006	0.00006
Effective cold plenum length (in)	5.923	5.979	5.981
Dish volume (in^3)	0.1419	0.1416	0.1414
	End of Cycle (EOC)		
Time to Maximum Fuel Pin Pressure	7542.0 hours		
Fuel Rod Composition:			
	Average Core	Hot Assembly	Hot Rod
Gram Moles	0.02329	0.02487	0.02520
Helium fraction	0.96703	0.90564	0.89349
Argon fraction	0.00000	0.00000	0.00000
Hydrogen fraction	0.00000	0.00000	0.00000
Nitrogen fraction	0.03197	0.02994	0.02954
Krypton fraction	0.00013	0.00852	0.01018
Xenon fraction	0.00087	0.05590	0.06680
Effective cold plenum length (in)	6.253	6.104	6.095
Dish volume (in^3)	0.1397	0.0837	0.0819

TABLE 3.4

SEQUENCE OF EVENTS FOR BASE CASE LARGE BREAK LOCA

EVENT	TIME (SECONDS)
1. Break opens	0.05
2. Loss of offsite power	0.05
3. Main feedwater isolated	0.05
4. MSIVs close	0.05
5. High containment pressure HI-1 signal	1.07
6. Accumulator injection, intact loop	14.90
7. Centrifugal charging pumps inject	18.07
8. End-of-Bypass	22.66
9. Safety injection pumps inject	23.07
10. Time of sustained downfall	24.88
11. Low pressure pumps inject	28.07
12. Bottom of Core Recovery	37.91
13. Rod burst	40.01
14. Accumulator empty	43.85
15. Peak clad temperature reached	60.06

TABLE 3.5

SEQUENCE OF EVENTS FOR BREAK SPECTRUM STUDY
(CHOPPED COSINE, BOC)

EVENT	TIME (SECONDS)			
	DOUBLE-ENDED GUILLOTINE			SPLIT
	CD=1.0	CD=0.8	CD=0.6	CD=1.0
1. Break opens	0.05	0.05	0.05	0.05
2. Loss of offsite power	0.05	0.05	0.05	0.05
3. Main feedwater isolated	0.05	0.05	0.05	0.05
4. MSIVs close	0.05	0.05	0.05	0.05
5. High containment pressure HI-1 signal	1.07	1.14	1.30	1.12
6. Accumulator injection	14.90	15.10	16.70	15.20
7. Centrifugal charging pumps inject	18.07	18.14	18.30	18.12
8. End-of-Bypass	22.66	23.13	25.29	22.91
9. Safety injection pumps inject	23.07	23.14	23.30	23.12
10. Time of sustained downfall	24.88	25.34	27.50	25.15
11. Low pressure pumps inject	28.07	28.14	28.30	28.12
12. Bottom of Core Recovery	37.91	38.43	40.63	38.19
13. Rod burst	40.01	47.18	52.19	41.[illegible]
14. Accumulator empty	43.85	43.95	45.65	44.15
15. Peak clad temperature reached	60.06	72.43	73.29	73.01

TABLE 3.6

SEQUENCE OF EVENTS FOR POWER SHAPE STUDY
(DEG, CD=1.0, BOC)

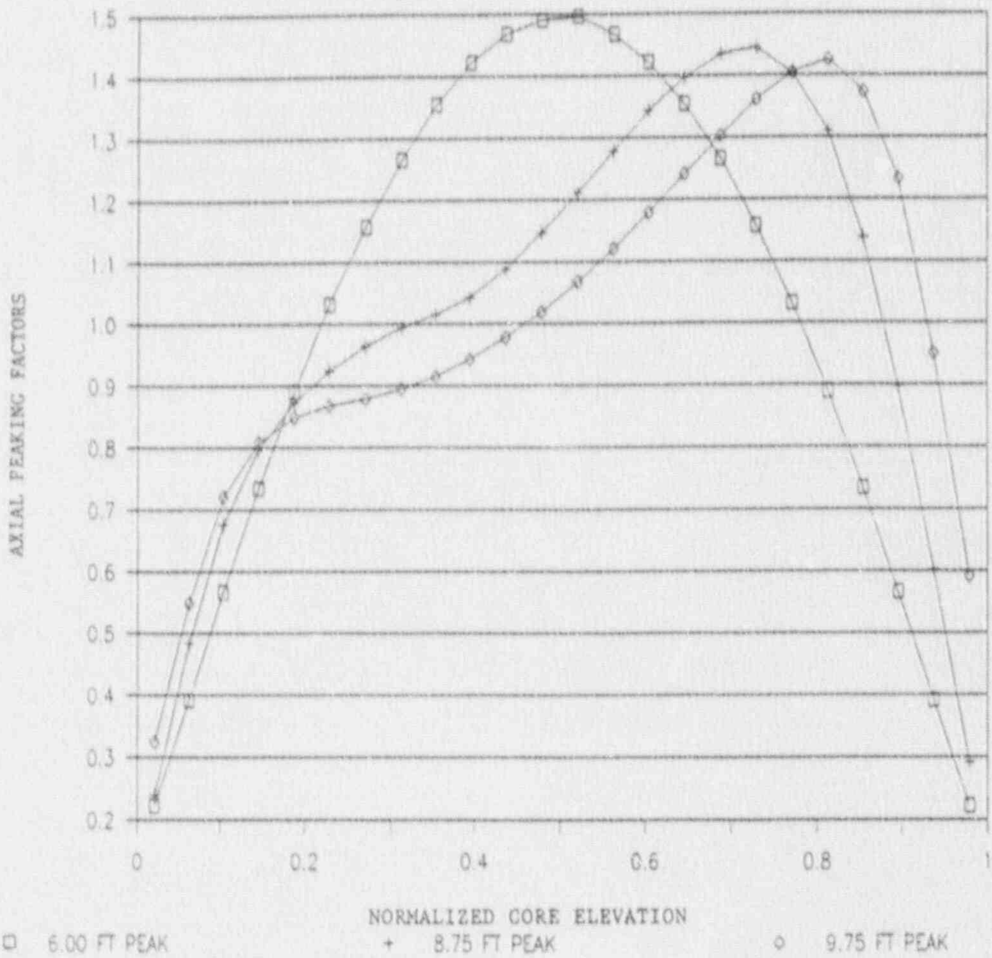
EVENT	TIME (SECONDS)		
	CHOPPED COSINE (BASE CASE)	TOP SKEWED AT 8.75'	TOP SKEWED AT 9.75'
1. Break opens	0.05	0.05	0.05
2. Loss of offsite power	0.05	0.05	0.05
3. Main feedwater isolated	0.05	0.05	0.05
4. MSIVs close	0.05	0.05	0.05
5. High containment pressure signal	1.07	1.07	1.07
6. Accumulator injection, intact loop	14.90	14.90	14.90
7. Centrifugal charging pumps inject	18.07	18.07	18.07
8. End-of-Bypass	22.66	22.64	22.64
9. Safety injection pumps inject	23.07	23.07	23.07
10. Time of sustained downfall	24.88	24.86	24.86
11. Low pressure pumps inject	28.07	28.07	28.07
12. Bottom of Core Recovery	37.91	37.93	37.91
13. Rod burst	40.01	44.69	44.99
14. Accumulator empty	43.85	43.70	43.80
15. Peak clad temperature reached	60.06	72.84	218.44

TABLE 3.7

SEQUENCE OF EVENTS FOR BURNUP STUDY
(TOP SKEWED AT 8.75', DEG, CD=1.0)

EVENT	TIME (SECONDS)	
	BOC	EOL
1. Break opens	0.05	0.05
2. Loss of offsite power	0.05	0.05
3. Main feedwater isolated	0.05	0.05
4. MSIVs close	0.05	0.05
5. High containment pressure HI-1 signal	1.07	1.07
6. Accumulator injection, intact loop	14.90	14.90
7. Centrifugal charging pumps inject	18.07	18.07
8. End-of-Bypass	22.64	22.64
9. Safety injection pumps inject	23.07	23.07
10. Time of sustained downfall	24.86	24.86
11. Low pressure pumps inject	28.07	28.07
12. Bottom of Core Recovery	37.93	37.93
13. Rod burst	44.69	51.14
14. Accumulator empty	43.70	43.70
15. Peak clad temperature reached	72.84	72.24

FIGURE 3.1 AXIAL POWER SHAPES



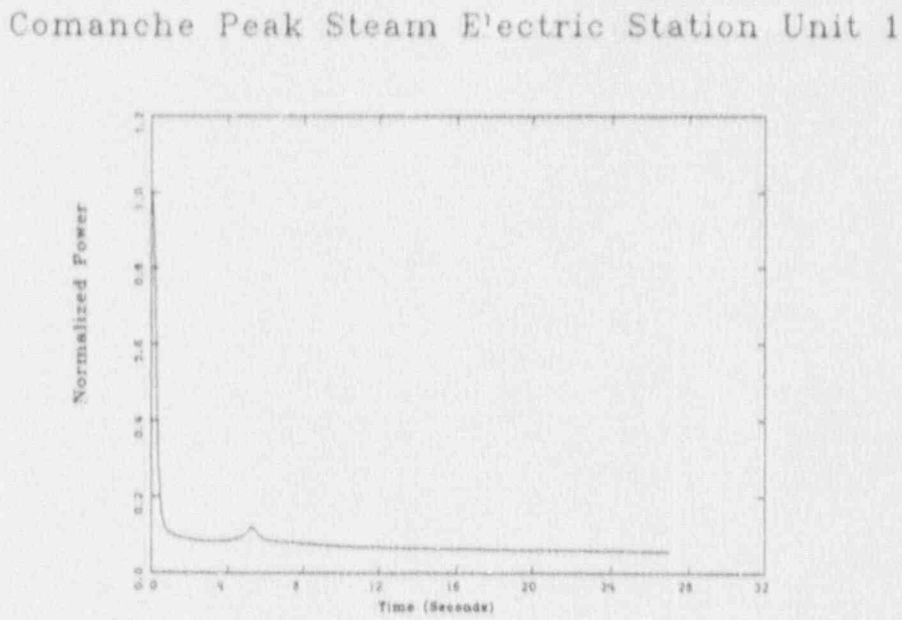


Figure 3.2 Normalized Power (Base Case)

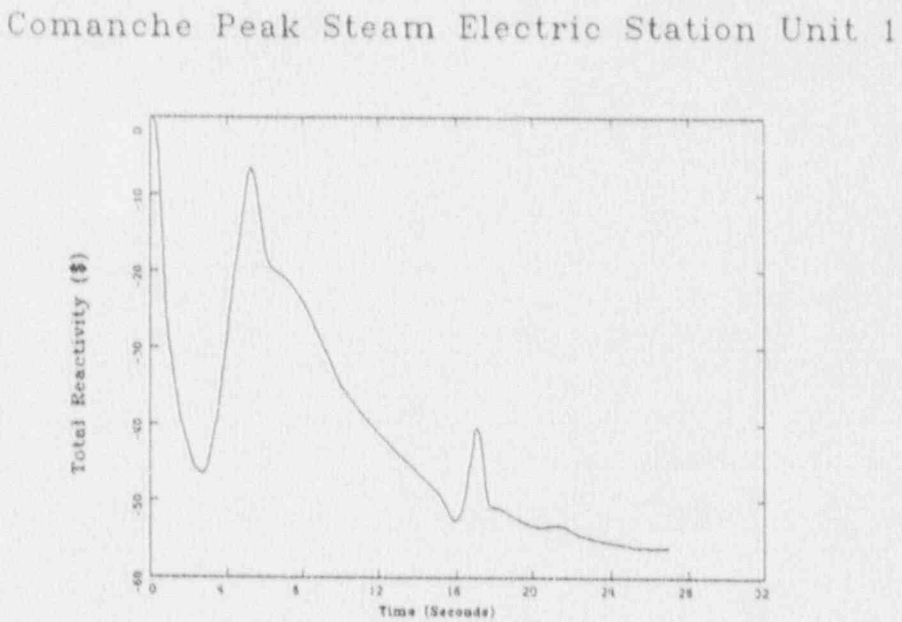


Figure 3.3 Total Reactivity (Base Case)

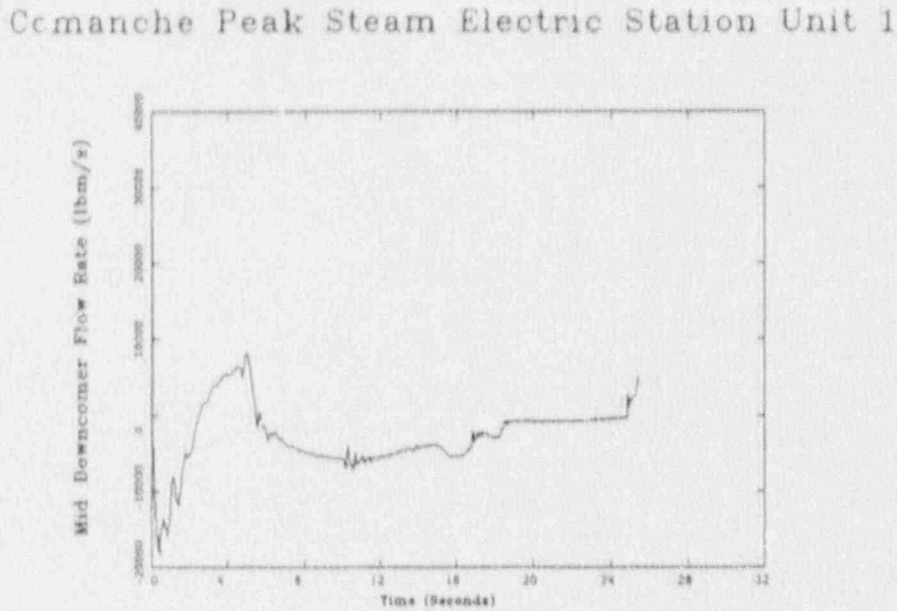


Figure 3.4 Downcomer Flow Rate (Base Case)

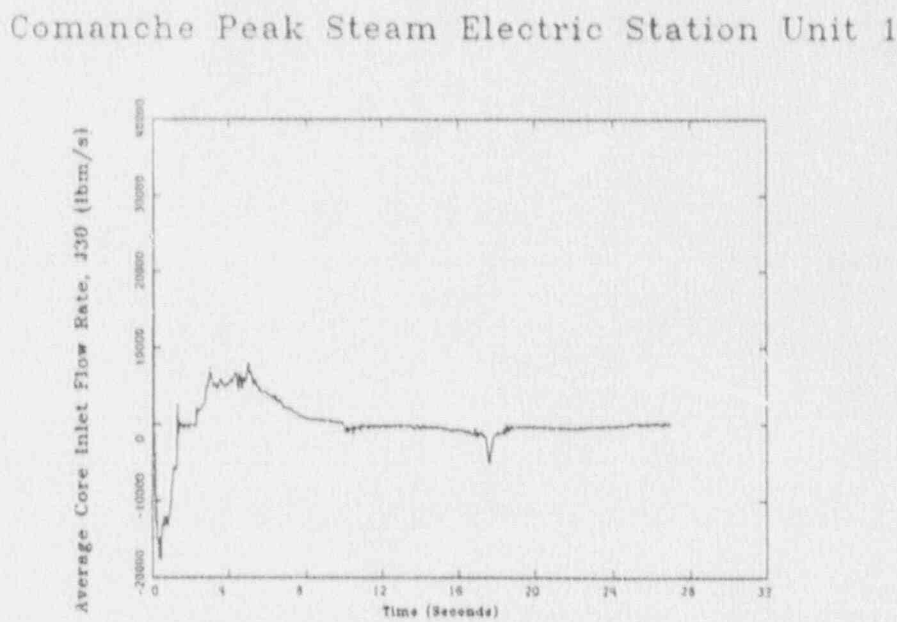


Figure 3.5 Average Core Inlet Flow Rate (Base Case)

Comanche Peak Steam Electric Station Unit 1

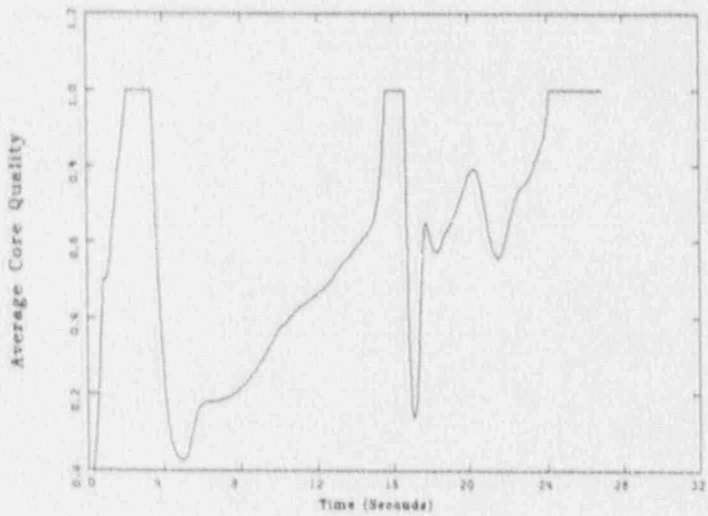


Figure 3.6 Average Core Mid Node Quality (Base Case)

Comanche Peak Steam Electric Station Unit 1

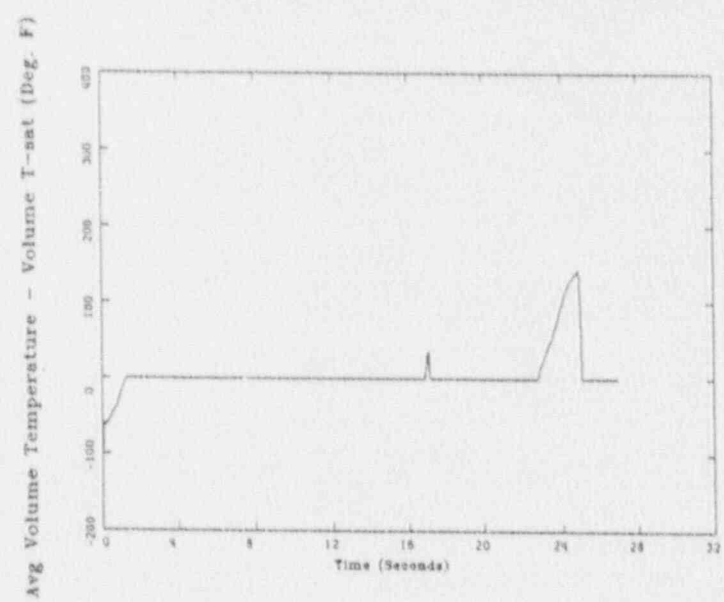


Figure 3.7 Core Inlet Subcooling (Base Case)

Comanche Peak Steam Electric Station Unit 1

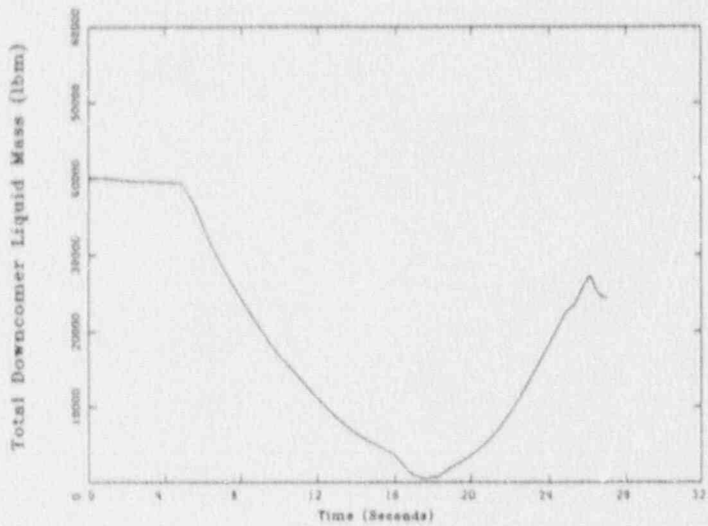


Figure 3.8 Downcomer Liquid Mass Inventory (Base Case)

Comanche Peak Steam Electric Station Unit 1

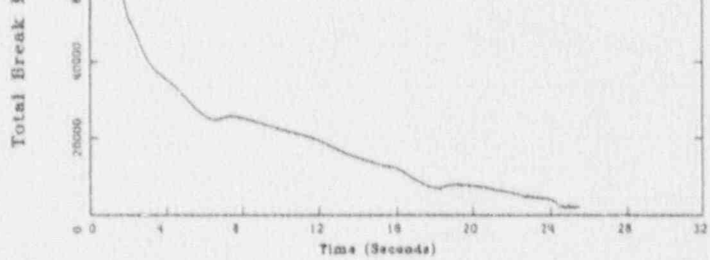


Figure 3.9 Total Break Flow (Base Case)

Comanche Peak Steam Electric Station Unit 1

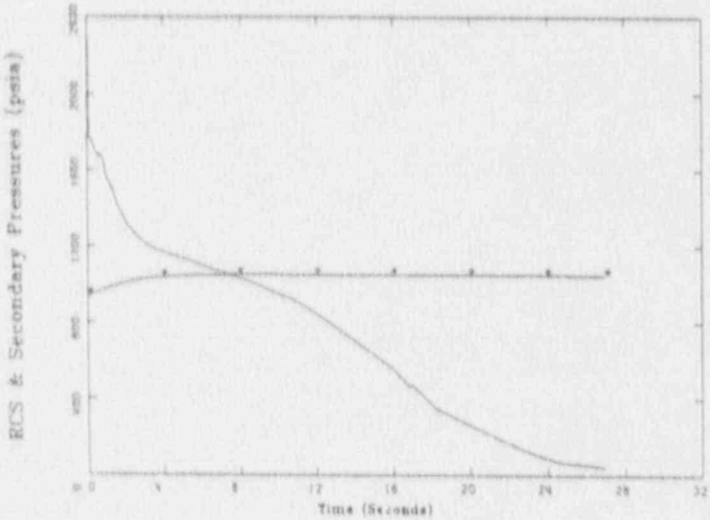


Figure 3.10 RCS & Secondary Pressures (Base Case)

Comanche Peak Steam Electric Station Unit 1

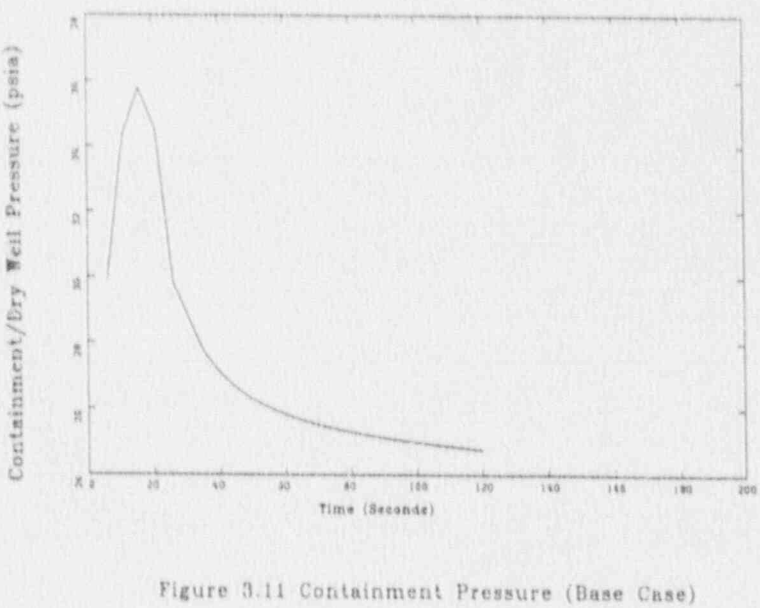


Figure 3.11 Containment Pressure (Base Case)

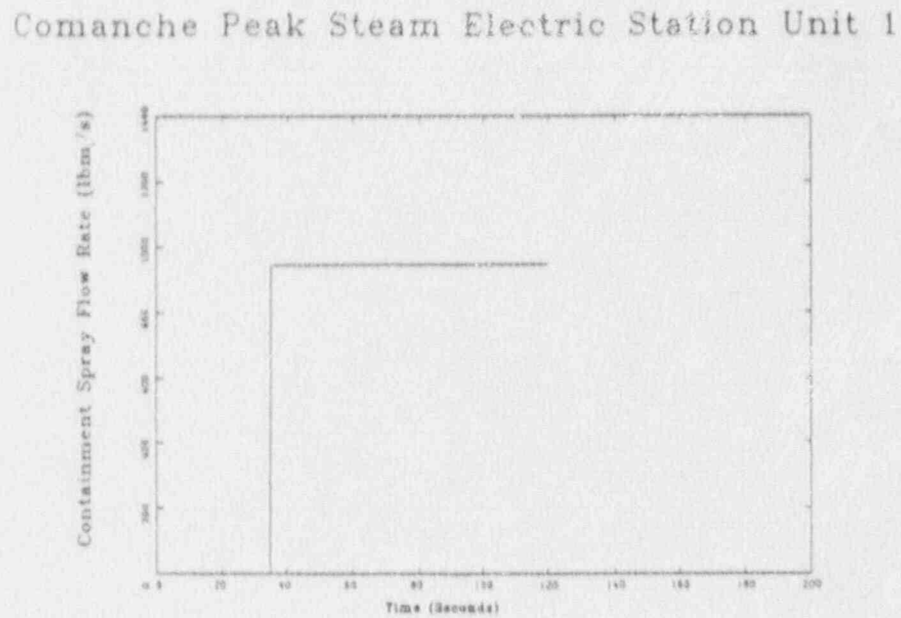


Figure 3.12 Containment Spray System Flow Rate (Base Case)

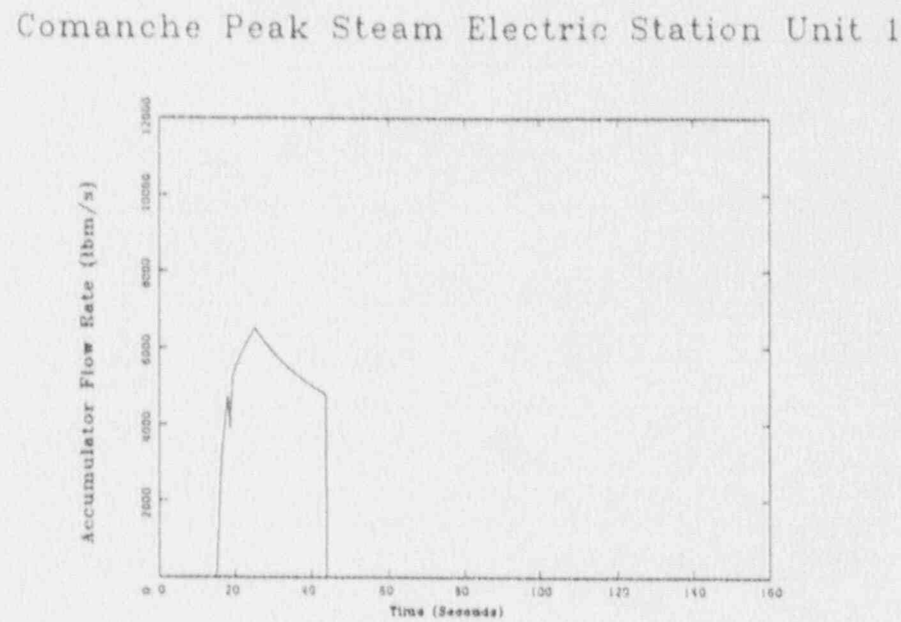


Figure 3.13 Accumulator Flow Rate (Base Case)

Comanche Peak Steam Electric Station Unit 1

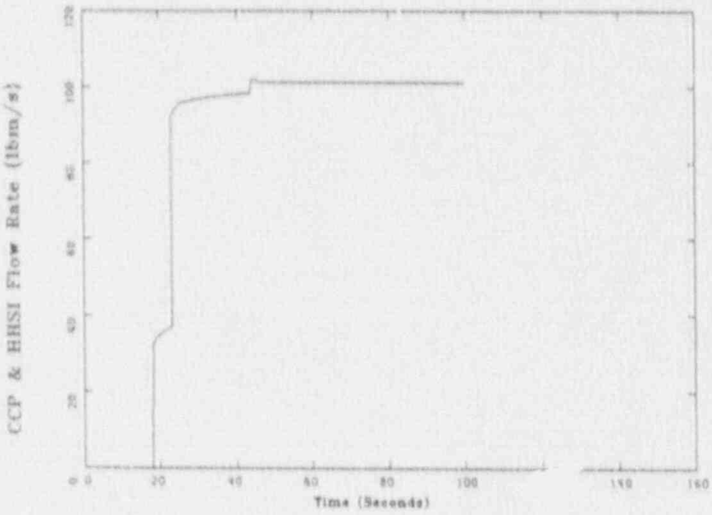


Figure 3.14 CCP & HHSI Pump Flow Rate (Base Case)

Comanche Peak Steam Electric Station Unit 1

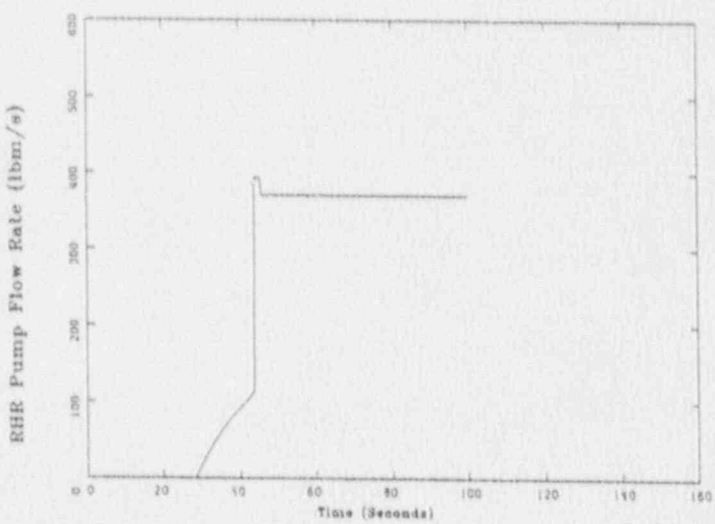


Figure 3.15 RHR Pump Flow Rate (Base Case)

Comanche Peak Steam Electric Station Unit 1

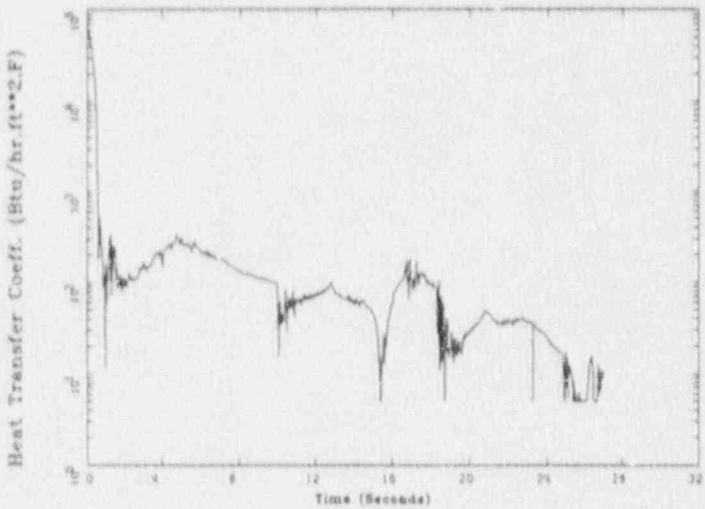


Figure 3.16 Mid Elevation Hot Assembly Heat Transfer Coeff. (Base Case)

Comanche Peak Steam Electric Station Unit 1

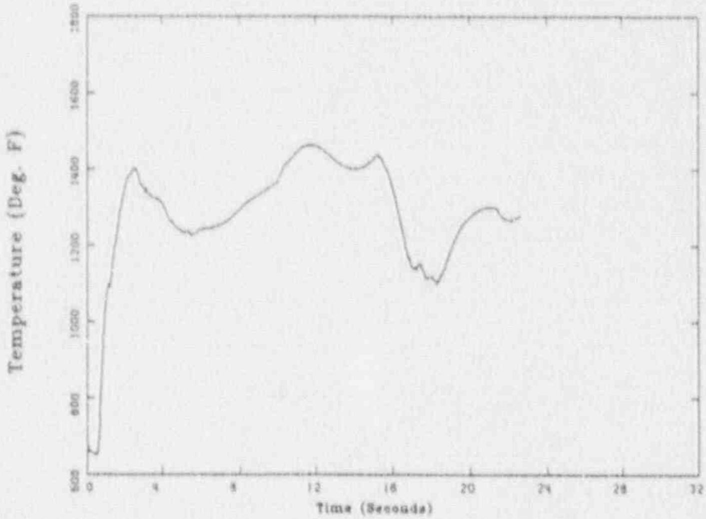


Figure 3.17 Hot Rod Temperature At PCT Node Elevation (Base Case)

Comanche Peak Steam Electric Station Unit 1

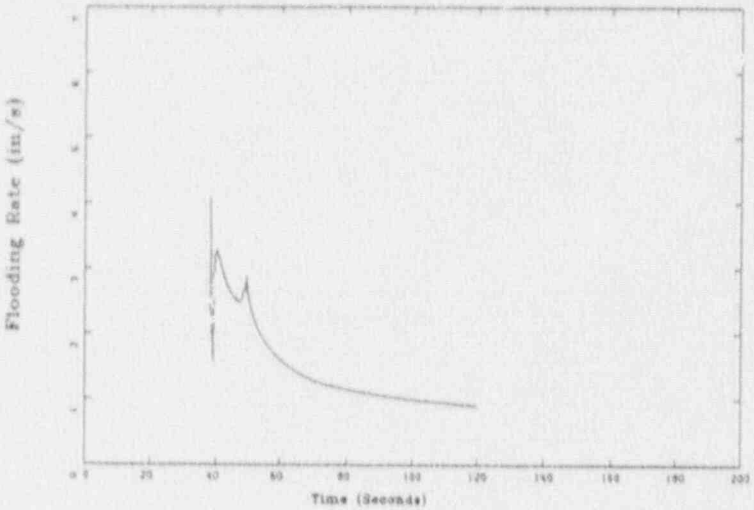


Figure 3.18 Core Flooding Rate (Base Case)

Comanche Peak Steam Electric Station Unit 1

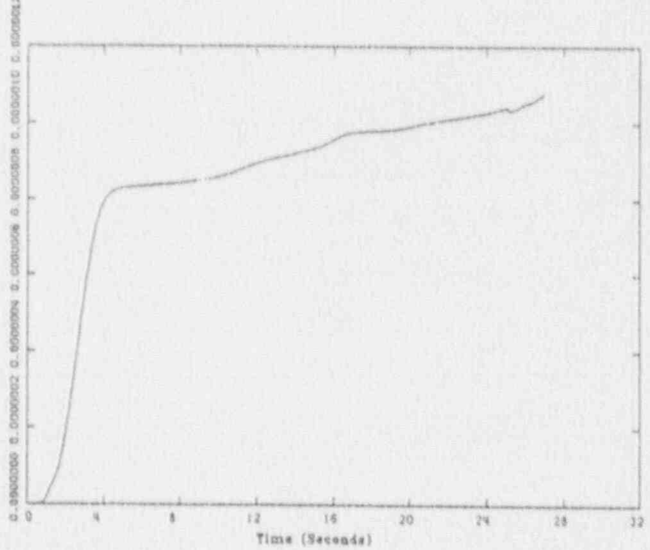


Figure 3.19 Mid Elevation Hot Assembly Zr/Water Reaction Depth (Base Case)

Comanche Peak Steam Electric Station Unit 1

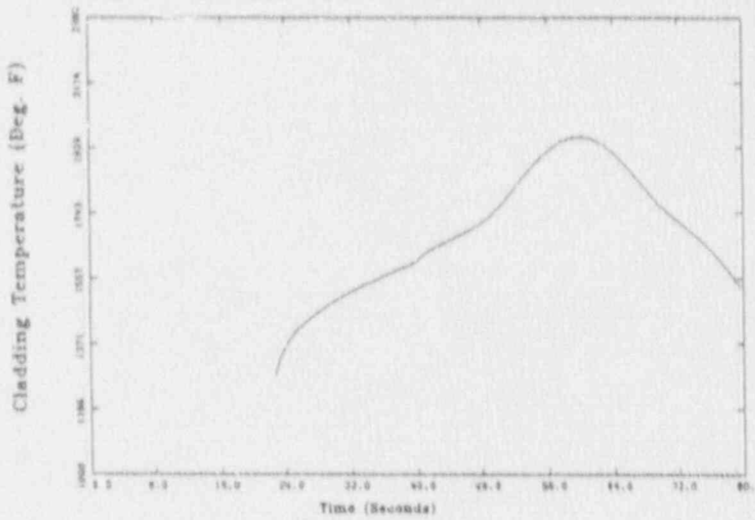


Figure 3.20 PCT/Ruptured Node Cladding Temperature (Base Case)

Comanche Peak Steam Electric Station Unit 1

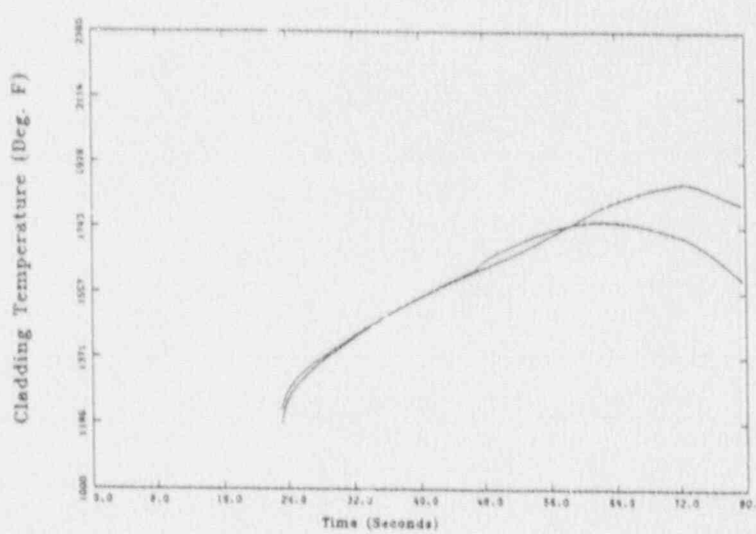


Figure 3.21 PCT/Ruptured Node Clad Temperature (Cosine/CD=0.8)

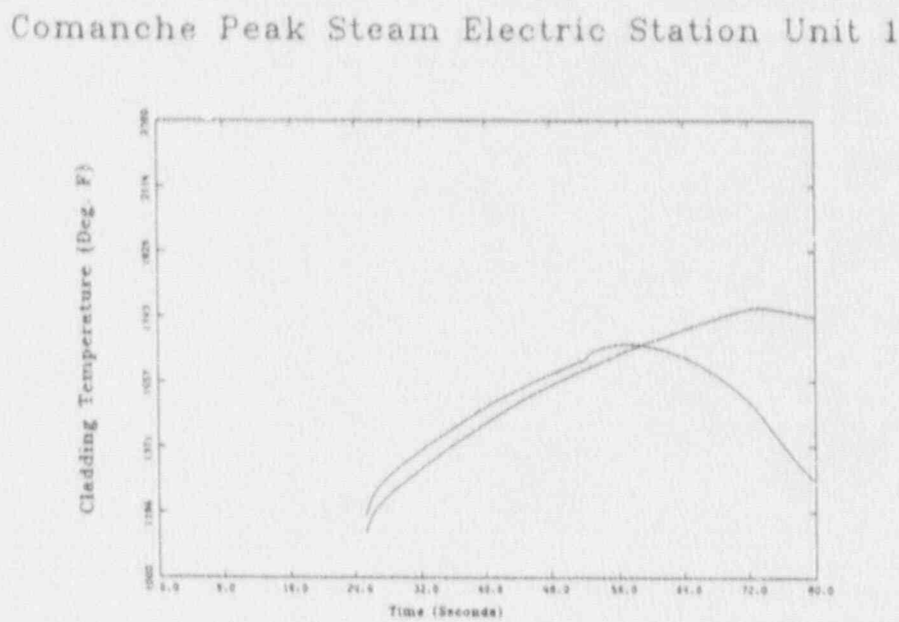


Figure 3.22 PCT/Ruptured Node Clad Temperature (Cosine/CD=0.6)

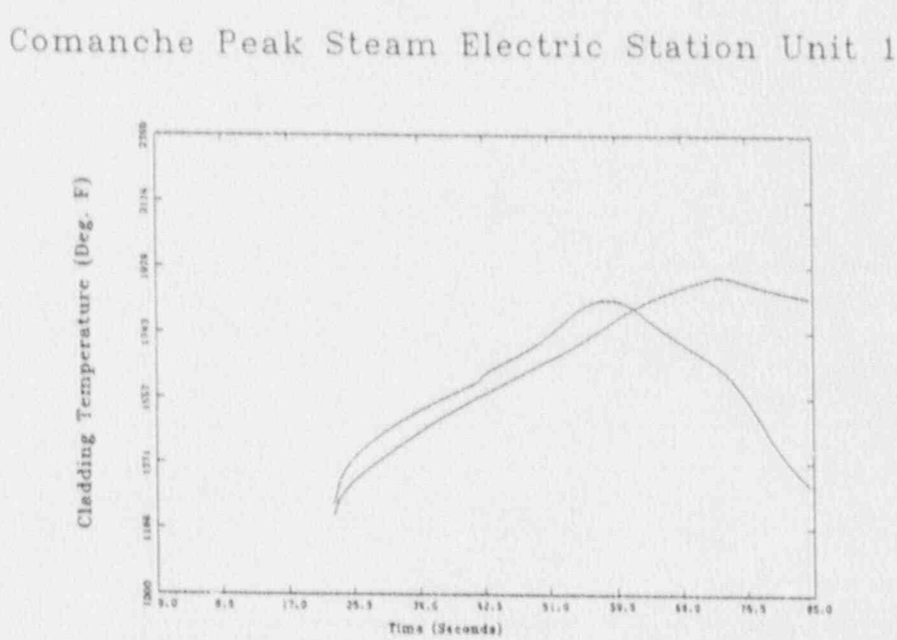


Figure 3.23 PCT/Ruptured Node Clad Temperature (Cosine/Split/CD=1.0)

Comanche Peak Steam Electric Station Unit 1

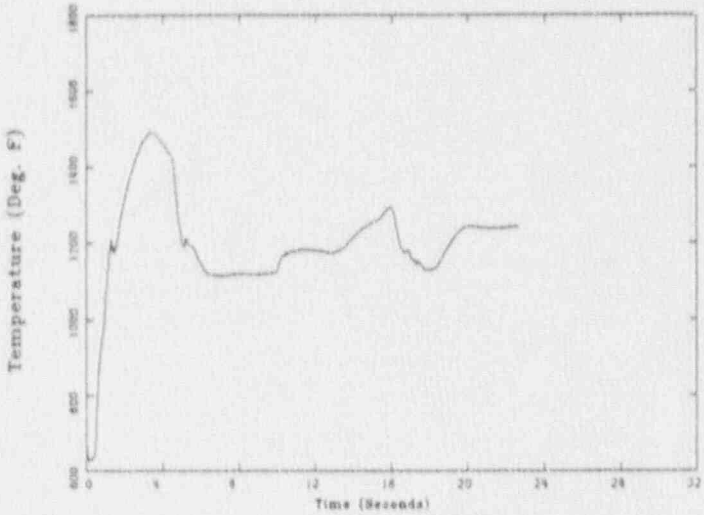


Figure 3.24 Hot Rod Temperature At PCT Node Elev. (Skewed @ 8.75 ft)

Comanche Peak Steam Electric Station Unit 1

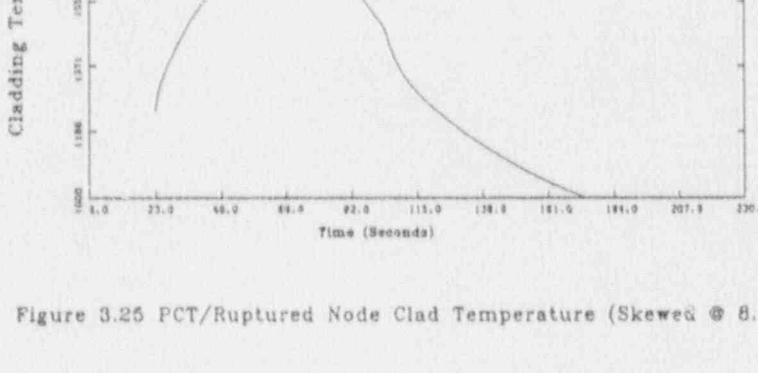


Figure 3.25 PCT/Ruptured Node Clad Temperature (Skewed @ 8.75 ft)

Comanche Peak Steam Electric Station Unit 1

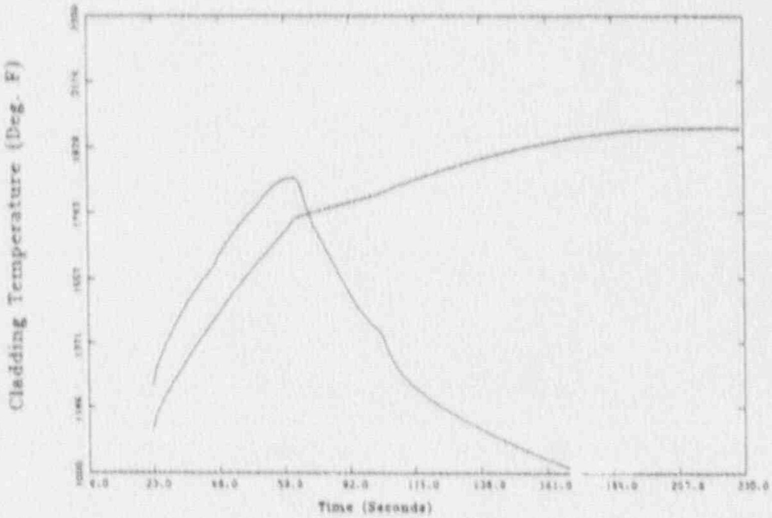


Figure 3.26 PCT/Ruptured Node Clad Temperature (Skewed @ 9.75 ft)

Comanche Peak Steam Electric Station Unit 1

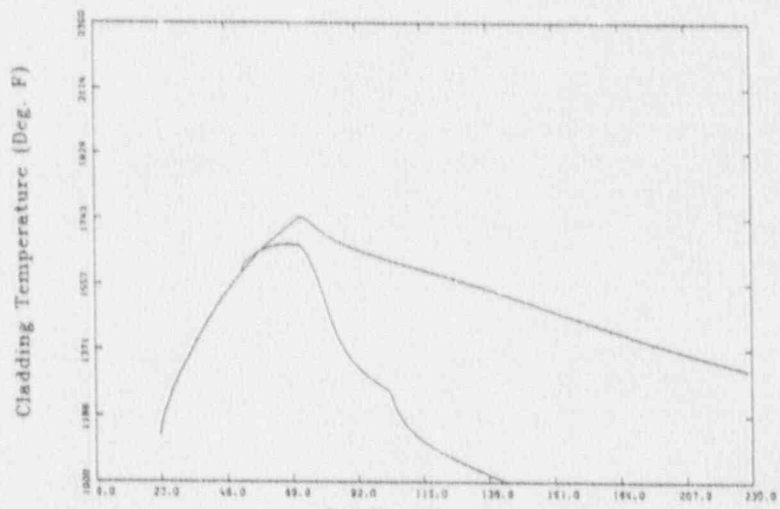


Figure 3.27 PCT/Ruptured Node Clad Temperature (Exposure/Skewed @ 8.75 ft)

CHAPTER 4

CONCLUSION

The USNRC-approved (Ref. 4.1) ANF Corporation's large break ECCS Evaluation model entitled EXEM/PWR 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 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 core region using five axial nodes (rather than the three shown in ANF's submittal). This deviation has been made in order to increase accuracy.

Seven calculations have been presented with two objectives:

1. To demonstrate Texas Utilities' ability to properly apply EXEM/PWR (Ref. 1.1); and

2. To demonstrate the development of up-to-date input decks and conclusions which are in compliance with 10 CFR 50, Appendix K. 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 (b) (1).

2. The total cladding oxidation at the peak location is under the 17% limit specified in 10 CFR 50 (b) (2).

3. The hydrogen generated in the core by cladding oxidation is less than the 1% limit established by 10 CFR 50 (b) (3).

4. Only hot channel rods experience clad rupture. The average core region undergoes only minor dimensional changes, but no clad ruptures are calculated to occur there. Thus, the coolable geometry criterion of 10 CFR 50 (b) (4) is satisfied.

5. The available ECCS is successfully initiated and the core is well cooled in less than 200 seconds. Therefore, the calculations comply with the long-term cooling criterion of 10 CFR 50 (b) (5).

Regarding the various sensitivity studies it has been found:

1. The most limiting break is a Double-Ended Guillotine rupture of the main coolant pump discharge line with a discharge coefficient of 1.0.

2. The most limiting Power Shape is the top skewed profile peaked at the elevation of 8.75 ft. which is shown in Figure 3.1.

3. The most limiting exposure occurs at 613.8 hours and is coincident with maximum stored energy in the fuel.

Texas Utilities will use the EXEM/PWR methodology including all codes, input decks, results, conclusions, and application procedures presented in this report to perform large break LOCA analyses and evaluations in compliance with 10 CFR 50 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

DOUBLE-ENDED GUILLOTINE BREAK CD	AXIAL POWER SHAPE			
	CHOPPED COSINE (4)	TOP PEAKED AT 8.75 FT		TOP PEAKED AT 9.75 FT
EXPOSURE →	BOL (5)	BOL (5)	EOL (6)	BOL (5)
1.0	1959 °F (1) 4.14 % (2) 0.56 % (3)	2034 °F 5.38 % 0.72 %	1745 °F 1.76 % 0.46 %	1983 °F 3.40 % 0.69 %
0.8	1870 °F 2.31 % 0.43 %			
0.6	1768 °F 1.27 % 0.30 %			

NOTES:

(1) PEAK CLAD TEMPERATURE (DEGREES F)
(2) PERCENT LOCAL CLADDING OXIDATION
(3) PERCENT CORE-WIDE OXIDATION
(4) BASE CASE
(5) MAXIMUM STORED ENERGY
(6) MAXIMUM PIN PRESSURE

LONGITUDINAL SPLIT BREAK CD	AXIAL POWER SHAPE
	CHOPPED COSINE
EXPOSURE →	BOL (5)
1.0	1901 °F 2.83 % 0.52 %

CHAPTER 5

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Chapter 3:

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APPENDIX

DESCRIPTION OF THE COMPUTATIONAL TOOLS

The EXEM/PWR Evaluation Model utilizes four basic computer codes:

1. RODEX2
2. RELAP4-EM
   1. SYSTEM BLOWDOWN
   2. HOT CHANNEL
   3. ACCUMULATOR-SIS
3. RFPAC
   1. PREFILL
   2. REFLEX
   3. ICECON/CONTEMPT-LT
   4. SHAPE/REFLOOD
4. TOODEE2

These basic codes address the various stages of the LOCA calculation as discussed in Section 2.2 and illustrated in Fig. 2.2.1. The codes, their interfaces, interrelationships and respective inputs and outputs are summarized in Fig. A.1 and Table A.1. The function of each code is described in the following sections.

A.1 RODEX2

RODEX2 is used within the EXEM/PWR framework to provide initial conditions for the RELAP4-EM system blowdown calculation. These conditions are (a) stored energy in the fuel, (b) gap gas composition, and (c) rod internal pressure. The stored energy is input iteratively by adjusting the fuel rod gap dimension in the RELAP4-EM system initialization calculation until the calculated stored energy matches the RODEX2 value. During the RELAP4-EM hot channel calculation, the built-in RODEX2 fuel models are activated so that gap adjustments are not needed there.

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 RELAP4-EM

A.2.1 SYSTEM BLOWDOWN

This code (Ref. 2.3) has been abundantly discussed in the literature. Only specific features of the EM version are briefly summarized in this section.

The fluid dynamics portion of the RELAP4 program solves the fluid mass, energy, and momentum equations. There is a choice of several fluid momentum equations in RELAP4. All of these are one-dimensional approximations and differ in the mathematical treatment of momentum flux. The form used in EXEM/PWR is the Incompressible Mechanical Energy Balance equation where the fluid dynamics in the vicinity of an area change is treated as incompressible. This formulation is used in all one-dimensional flow paths throughout the system piping and core. In the case of plena, the plenum area is specified as arbitrarily large, resulting in an equivalent stagnation volume treatment of the plenum. This modeling effectively eliminates the momentum flux portion of the equation as prescribed in Ref. 2.3. Similarly, in the case of crossflow, all momentum flux terms are deleted in the crossflow direction by assuming no area change (an open lattice) within the crossflow path.

A.2.2 HOT CHANNEL

This is not another code but an application of the RELAP4-EM code discussed above, to the hot channel. Boundary conditions from the system blowdown calculation are used. The main outputs from this calculation are the hot rod temperature distribution and oxide layer thickness at the End-of-Bypass. These are used to initialize the heatup calculation performed in the TOODEE2 code for the refill and reflood periods. The (RELAP4-EM) hot channel calculation is necessary in order to adequately account for crossflows in the hot assembly and for the hot rod which is not represented in the system blowdown calculation.

A.2.3 ACCUM-SIS

ACCUM-SIS is also an application of RELAP4-EM. The ACCUM-SIS calculation determines the ECCS flow rates to the cold legs after the end-of-bypass period (EOBY). The broken and intact loops are modeled, including accumulators, high, intermediate, and low pressure injection systems. Explicit modeling of the injection systems' piping is not done. These systems are modeled as fill junctions at the accumulator lines (Fig. 2.3.3). The broken loop flow is assumed to be lost to the containment and is included in the ICECON/CONTEMPT-LT input. The intact loop ECCS boundary conditions for the ACCUM-SIS calculation is taken from the RELAP4-EM system blowdown calculation up to EOBY and assumed to be constant and equal to the containment pressure at EOBY thereafter.

A.3 RFPAC

RFPAC combines the four codes used to perform the refill and reflood thermal-hydraulic analyses (ICECON/CONTEMPT-LT, PREFILL, SHAPE/REFLOOD, and REFLEX) and eliminates the need for data transfer between codes. In the context of the overall EXEM/PWR methodology, RFPAC serves as a bridge between the RELAP4 and supplying fluid boundary conditions to TOODEE2.

A.3.1 ICECON/CONTEMPT-LT

ICECON is essentially the same program as CONTEMPT-LT (Ref. 2.5). This is a computer program developed to describe the thermal-hydraulic behavior of reactor containment systems subjected to postulated accident conditions.

The code calculates the interrelated effects of reactor system blowdown, heat transfer, atmosphere leakage, safeguards system operation, pressure suppression system response, and miscellaneous mass and energy additions.

The code is used in the EXEM/PWR framework to provide containment pressure as a boundary condition for the PREFILL and REFLEX codes, which are used in reflood calculations. The mass and energy releases to the containment are input from the RELAP4-EM system calculation during the blowdown stage and from the broken loop ACCUM-SIS calculation after that, as described in Section 2.2.

A.3.2 PREFILL

The time between the system blowdown period as defined by the End-of-Bypass (EOBY) and the beginning of reflood when the water level reaches the bottom of the core (BOCREC) is the refill portion of the LOCA transient. The PREFILL code calculates the time to start of reflood and the flow of ECCS fluid to the core during reflood.

The phenomena addressed by PREFILL are (a) hot wall delay period, (b) free-fall delay time, (c) extended accumulator flows, (d) open channel flow spill, and (d) core inlet subcooling.

A.3.3 SHAPE/REFLOOD

SHAPE/REFLOOD uses the average core fuel and cladding temperatures at End-of-Bypass from the RELAP4-EM hot channel calculation to determine the average rod temperatures at the peak power location at the time of BOCREC for use in the Fuel Cooling Test Facility (FCTF) reflood correlations in the REFLEX code.

Injection of subcooled ECC water is possible. Steam condensation in the intact loops is accounted for, and spillage to the break from the downcomer is based on gravitational head forces developed between the downcomer and the break when the downcomer is full to the cold leg pipe level.

The steam-water interaction pressure loss penalty during pumped ECC injection is reduced to an injection-angle independent value of 0.15 psi, based upon EPRI data (Ref. 2.12). During the accumulator injection period the penalty is 0.6 psid.

Log-mean-temperature heat exchanger thermal balance equations are used for the heat transfer occurring in the steam generators instead of the RELAP4-EM conservation equations. This is faster and is justified by the slow thermodynamic changes occurring in the steam generators during reflood as compared to blowdown.

A.3.4 REFLEX

The REFLEX program calculates core reflood [illegible] This program is built upon a RELAP4 skeleton. The RELAP4 system equations are simplified in REFLEX in the interest of computational speed.

The system modeling detail and sophistication required for the blowdown calculation is not required for the somewhat slower reflood process. The code utilizes a quasi-steady state solution of the mass, momentum and energy equations for PWR reactor systems. Specific models were developed for the system, core downcomer annulus, ECC mixing location, and steam generators. An equation of state was developed to provide fluid properties. The modifications made to the original RELAP-EM/FLOOD code to produce REFLEX are as described in the following paragraphs.

The core neutronics, transient heat conduction and critical flow tables are omitted.

Acceleration pressure losses are omitted in the flow equations. Mass accumulation and gravitational losses are also omitted in all system components except in the core, downcomer nodes, and in the cold leg piping to the break during the accumulator discharge phase.

The fluid state equations are based on analytical fits to property tables over a limited pressure range, 10-100 psia. This method is faster than the previous table look-up process.

The numerical scheme of RELAP4 is replaced for the flow calculation by the linear theory method (Ref. 2.10), using a Gauss-Jordan elimination method (Ref. 2.11).

The core outlet enthalpy is conservatively assumed to be determined by steam generator secondary temperature and containment pressure in order to yield a conservatively high upper plenum pressure for reflood.

A.4 TOODEE2

TOODEE2 calculates the time dependent temperature distribution in the hot rod during the refill and reflood portions of the LOCA. The TOODEE2 calculation begins at End-of-Bypass (EOBY).

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 zircalloy. 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. Whenever rupture is determined and the flooding rate drops below 1 in/sec, only steam cooling is allowed downstream of the ruptured node. This is in compliance with the related Appendix K requirement. The effect of clad strain on pellet-to-clad gap heat transfer and on the thinning of the oxide layer on the outside of the cladding is considered. Once fuel rod rupture is determined, the code calculates both inside and outside metal water heat generation. Fuel rod rupture reduces the subchannel flow area at the rupture and diverts flow from the hot rod subchannel to neighboring subchannels.

Flow recovery is assumed above the rupture. The effect of flow diversion on heat transfer (both convection and radiation) to the coolant is accounted for. The TOODEE2 code calculates heat transfer coefficients as a function of fluid condition or via reflood data-based correlations.

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 criteria (if pin-wide oxidation is less than 1% it is concluded that the criteria of less than 1% core-wide oxidation is met).

A.5 DATA PREPARATION AND TRANSFER TOOLS

The EXEM/PWR methodology also includes four additional codes for preparing data and transferring results between the basic codes described above:

1. FISHEX
2. SHAPE
3. BLOCK
4. BLOWDOWN-ICECON

A.5.1 FISHEX

FISHEX is used to determine the normalized power (P(t)/P(0)) following the End-of-Bypass, for use with REFLEX and TOODEE2.

The code accounts for fission and decay heat, including heat from actinide decay. The 20% overpower factor included in the RELAP4 calculation is not applied in FISHEX to the decay heat from actinides, since this extra 20% of the decay heat from actinides is not required by 10 CFR 50.46 Appendix K.

A.5.2 SHAPE

SHAPE automates the building of portions of input decks to RELAP4, 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 RELAP4 and the axial power factors for input to the RODEX2 and TOODEE2 codes. The code can also set up certain blocks of input data to RELAP4, viz. reactivity coefficient data, core heat slab data, core section data, and core geometry data cards.

A.5.3 BLOCK

BLOCK generates the clad swelling and rupture tables for input to the RELAP4 code. The swelling and rupture model used in BLOCK is taken from TOODEE2. The model is based on data with temperature rates of 0.0 to 28.0 degree C per second and can interpolate the data between these two ramp rates.

The 0.0 degree C per second ramp rate is the most conservative because this rate leads to swelling and rupture at lower cladding hoop stresses. The 0.0 degree C per second value is used in the present LOCA/ECCS analyses. The tables generated by BLOCK are valid for calculating the fuel rod pre-rupture strain in RELAP4.

A.5.4 BLOWDOWN-ICECON

BLOWDOWN-ICECON is a data transfer tool. This code reads output files from the RELAP4 system calculation and transfers information to ICECON/CONTEMPT-LT and to FISHEX. The RELAP4 mass and energy releases to the containment are written in the appropriate format for CONTEMPT-LT use. Time-dependent reactivity calculated by RELAP4 is also read by BLOWDOWN-ICECON and converted into the appropriate format for input to FISHEX.

TABLE A.1

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

BLOCK
INPUT:
(1)* Number of fuel rods per assembly Number of instrument tubes per assembly Number of guide tubes per assembly Inside diameter of cladding Outside diameter of cladding Rod pitch Outside diameter of instrument tube Outside diameter of guide tube Cladding temperature ramp rate
OUTPUT:
(2) Rupture and blockage tables

SHAPE PUNCH
INPUT:
(3) 24 point axial power profile (Reactor Physics)
(4) Tech Spec peaking factor
(5) Renormalized 24 point axial power profile to Tech Spec peaking factor
OUTPUT:
(6) 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:
(6) 101 point axial power profile with Tech Spec peaking factor
(7) Adjusted axial peaking factor (tech spec) at peak node Axial nodalization to be used in RELAP4, TOODEE2, or RODEX2 Bundle geometry data (RELAP4)
OUTPUT:
(8) Reactivity weighting factors Power fraction data Core heat slab data Core section data Core geometry
(9) Power fraction data Core heat slab data Core section data Core geometry
(10) Axial core power factors Peak axial power location
(11) Axial power factors Axial grid locations

TABLE A.1 (Continued...)

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

RODEX2
INPUT: (11) Axial power factors Axial grid locations (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 (13) Core power history Average core Hot assembly Hot rod

TABLE A.1 (Continued...)

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

RODEX2
OUTPUT: (14) 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) Cladding + fuel surface roughness (15) Hot rod, hot assy, and average core gram-moles of gas (at exposure of interest) Hot rod, hot assy, and average core mole fractions of gas (at exposure of interest) Hot rod radially averaged density (at exposure of interest) Fuel model data cards, fuel density, and flux depression (at exposure of interest) Cold plenum length + dish volume (at exposure of interest) used to calculate cold plenum volume (16) Hot assy and average core gram-moles of gas (at exposure of interest) Hot assy and average core mole fractions of gas (at exposure of interest) Cold plenum length + dish volume (at exposure of interest) used to calculate cold plenum volume

TABLE A.1 (Continued...)

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

RELAP4-SYSTEM	
INPUT:	
(2)	Rupture and blockage tables
(8)	Reactivity weighting factors Power fraction data Core heat slab data Core section data Core geometry
(16)	Hot assy and average core gram-moles of gas (at exposure of interest) Hot assy and average core mole fractions of gas (at exposure of interest) Cold plenum length + dish volume (at exposure of interest) used to calculate cold plenum volume
(17)	NSSS definition (e.g. geometry, pump data, heat slabs, etc.) ECCS definition (e.g. accumulator volume, SI flow rate, etc.) Containment definition Fuel data Neutronics data

TABLE A.1 (Continued...)

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

RELAP4-SYSTEM
OUTPUT:
(18) Core inlet and outlet plenum data as boundary conditions Core power data EOBY time
(19) EOBY time
(20) Break mass and energy out to EOBY time Liquid remaining in the primary system at EOBY Reactivity versus time EOBY time
(21) RELAP4 ECCS model input
(22) Cold leg pressures (intact and broken loops) to EOBY time as boundary conditions Containment pressure at EOBY time EOBY time Time when the high containment pressure SIAS setpoint is reached
(23) Maximum downcomer or lower plenum slab temperature
(24) Steam generator secondary pressure and liquid mass

TABLE A.1 (Continued...)

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

RELAP4 HOT CHANNEL
INPUT:
(2) Rupture and blockage tables
(9) Power fraction data Core heat slab data Core section data Core geometry
(15) Hot rod, hot assy, and average core gram-moles of gas (at exposure of interest) Hot rod, hot assy, and average core mole fractions of gas (at exposure of interest) Hot rod radially averaged density (for all core nodes at exposure of interest) Fuel model data cards, fuel density, and flux depression (at exposure of interest) Cold plenum length + dish volume (at exposure of interest) used to calculate cold plenum volume
(18) Core inlet and outlet plenum data as boundary conditions Core power data EOBY time
OUTPUT:
(25) Fuel average temperatures and cladding temperatures for the 5 average core nodes
(26) Punch file created containing hot rod temperature distribution for the 24 axial node input Punch file created containing the oxide layer thickness for the 24 axial node input Hot rod internal pressure at the EOBY Hot rod cladding, hot assembly cladding, and average core cladding temperatures at the EOBY for calculation of radiation model sink temperatures for the 24 hot rod axial node input.

TABLE A.1 (Continued...)

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

BLOWDOWN ICECON
INPUT:
(20) Break mass and energy out to EOBY time Liquid remaining in the primary system at EOBY Reactivity versus time EOBY time
OUTPUT:
(27) Punch file created containing break mass flow rate and enthalpy versus time
(28) Punch file created containing reactivity versus time

ACCUMULATOR-SIS
INPUT:
(21) RELAP4 ECCS model input
(22) Cold leg pressures (intact and broken loops) to EOBY time as boundary conditions Containment pressure at EOBY time EOBY time Time when the high containment pressure SIAS setpoint is reached
OUTPUT:
(29) Broken loop ECCS flow rates and enthalpy to containment after EOBY
(30) Intact loop ECCS flow rates and enthalpy after EOBY

TABLE A.1 (Continued...)

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

CONTEMPT-LT
INPUT:
(27) Punch file created containing break mass flow rate and enthalpy versus time
(29) Broken loop ECCS flow rates and enthalpy to containment after EOBY
OUTPUT:
(31) Containment pressure and temperature response versus time

FISHEX
INPUT:
(28) Punch file created containing reactivity versus time
(32) Effective delayed neutron fraction divided by prompt neutron generation mean lifetime U238 atoms consumed per U235 atoms fissioned
OUTPUT:
(33) Punch file created containing normalized power versus time

TABLE A.1 (Continued...)

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

PREFILL
INPUT:
(23) Maximum downcomer or lower plenum slab temperature
(30) Intact loop ECCS flow rates and enthalpy after EOBY
OUTPUT:
(34) BOCREC time ECCS injection rates after BOCREC Temperature of ECCS fluid entering the core

SHAPE/REFLOOD
INPUT:
(25) Fuel average temperatures and cladding temperatures for the 5 average core nodes at EOBY
(33) Punch file created containing normalized power versus time
(39) Core power Core average linear heat generation rate
OUTPUT:
(35) Average rod temperature at the peak power location at BOCREC time for use in the FCTF reflood correlations

TABLE A.1 (Continued...)

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

REFLEX
INPUT: (10) Axial core power factors Peak axial power location (24) Steam generator secondary pressure and liquid mass (31) Containment pressure and temperature response versus time (34) BOCREC time ECCS injection rates after BOCREC Temperature of ECCS fluid entering the core (35) Average rod temperature at the peak power location at BOCREC time for use in the FCTF reflood correlations (36) Primary system geometry and loss coefficients based on the RELAP4 system deck OUTPUT: (37) Core coolant conditions versus time (core inlet flow, saturation temperature, effective core inlet flooding rate, and quench height) BOCREC time Time when instantaneous reflood rate drops below 1 inch/sec

TABLE A.1 (Continued...)

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

TOODEE2
INPUT:
(14) 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
(19) EOBY time
(26) Punch file created containing hot rod temperature distribution for the 24 axial node input Hot rod internal pressure at the EOBY Hot rod cladding, hot assembly cladding, and average core cladding temperatures at the EOBY for calculation of radiation model sink temperatures for the 24 hot rod axial node input
(33) Punch file created containing normalized power versus time
(37) Core coolant conditions versus time (core inlet flow, saturation temperature, effective core inlet flooding rate, and quench height) BOCREC time Time when instantaneous reflood rate drops below 1 inch/sec BOCREC time Core saturation temp at BOCREC Temperature of ECC water at BOCREC
(38) Axial power factors (from SHAPE) Axial grid locations (from SHAPE) Hot rod ALHGR (from SHAPE) Additional blockage data

TABLE A.1 (Continued...)

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

TOODEE2
OUTPUT: (40) Peak cladding temperature Percent local cladding oxidation Percent pin wide cladding oxidation

FIG. A.1

ANF EXEM/PWR METHODOLOGY FOR LARGE BREAK LOCA