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TXU

REVISED
LARGE BREAK
LOSS OF COOLANT ACCIDENT
ANALYSIS METHODOLOGY

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REACTOR ENGINEERING

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ABSTRACT

This report is presented to demonstrate the application TXU Electric's evaluation model based on USNRC-approved Siemens Power Corporation's (SPC) Emergency Core Cooling Systems (ECCS) Evaluation Model SEM/PWR-98 for Large Break Loss of Coolant Accident (LBLOCA), to the Comanche Peak Steam Electric Station (CPSES), and is intended to replace TXU Electric's current methodology (Reference 1.3).

This report contains a description of TXU Electric's application of the SEM/PWR-98 methodology which includes the computer codes, the details of the nodalization schemes, and the calculational procedures followed during all phases of the Loss-of-Coolant Accident (LOCA) analyses. The methodology is used to perform LBLOCA ECCS licensing analyses that comply with USNRC regulations contained in 10 CFR 50.46 and Appendix K.

In order to comply with a 10 CFR 50, Appendix K requirement, a spectrum of breaks, ranging from 0.6 through 1.0 double-ended guillotine (DEG) as well as 1.2 to 2.0 split (S), was examined. Although higher peak clad temperatures (PCTs) are usually associated with beginning of life (BOL) fuel because of the higher stored energy, a fuel exposure study was also conducted. This is done in order to confirm that the end of life (EOL) pin pressures, which are higher than those encountered early in life and which foster a higher driving force for rod burst, do indeed result in lower PCTs for the fuel under consideration.

Two additional types of sensitivity studies were performed. The first was a single failure study to confirm that the most limiting single failure is used. The second was a convergence criterion study, demonstrating that the value used for this parameter is adequate to produce converged results.

The methodology presented herein — including all codes, results, input decks, inferences and conclusions presented within this report — will be used to perform LBLOCA analyses and evaluations in compliance with 10 CFR 50.46 criteria and 10 CFR 50, Appendix K requirements, for fuel cycle analyses and to address pertinent licensing issues for Comanche Peak Steam Electric Station Unit 1 and Unit 2.

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CHAPTER 1

INTRODUCTION

The main objective of this work is to obtain approval of TXU Electric's evaluation model based[1] on Siemens Power Corporation's (SPC) methodology — including all codes, all input decks, all results, all inferences and conclusions — so that it may be applied to the Comanche Peak Steam Electric Station Unit 1 and Unit 2 for fuel cycle analyses and to address pertinent licensing issues. The methodology is used to perform the Large Break LOCA-ECCS licensing analyses that comply with USNRC regulations contained in 10 CFR 50.46 and 10 CFR 50 Appendix K.

This report describes TXU Electric's application of SPC's USNRC-approved (Reference 1.1) Emergency Core Cooling Systems (ECCS) Evaluation Model, entitled "SEM/PWR-98 ECCS Evaluation Model for PWR LBLOCA Applications", to the Comanche Peak Steam Electric Station Unit One (CPSES-1) for the current operating cycle (Cycle 8). The present Evaluation Model is a modification of the previous EXEM/PWR (Reference 1.2) methodology, on which TXU Electric's USNRC-approved LBLOCA analysis methodology (Reference 1.3) has been based.

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See footnote in Chapter 4 for differences between SPC's and TXU Electric's application.

The changes made in EXEM/PWR to create SEM/PWR-98 are discussed and referenced in SPC documentation (Reference 1.4). Changes were made to reduce variability in results. The Dougall-Rohsenowcorrelation was replaced with a revised film boiling correlation named Richert-Franz. The calculation of end-of-bypass was revised to remove unnecessary conservatism and to take credit for residual primary coolant calculated to remain in the system at the time of end-of-bypass. A new reactor coolant pump flow degradation model was developed based on tests conducted at Combustion Engineering (CE) and EPRI. SPC revised the application for the measured resinter density to conform to Regulatory Guide 1.126. SPC implemented a revised cladding creep model in its fuel rod code. Several additional minor changes were also made.

TXU Electric demonstrates in this report that it has applied the revised methodology as intended by SPC (Reference 1.4) and as approved by the USNRC (Reference 1.1). The present results are intended to become the analysis of record for CPSES-1, Cycle 8. It is also TXU Electric's intention to apply this methodology to CPSES-2 for the upcoming Cycle 6 and thus replace that current analysis of record as well.

This report includes a description of the current SEM/PWR-98 LBLOCA methodology (Chapter 2), including the details of the various nodalization schemes and procedures followed during all phases of the LOCA analyses. The changes from the previous EXEM/PWR methodology are indicated as appropriate (Chapter 2). The accident is postulated to occur with the plant in normal operation. Each calculation is performed in compliance with the explicitly approved SEM/PWR-98 LBLOCA methodology. Four types of sensitivity studies are presented in Chapter 3.

The first is a break spectrum study. Double-Ended Guillotine (DEG) breaks with discharge coefficients of 0.6, 0.8 and 1.0 were analyzed. In addition, longitudinal split breaks, having areas equal to: 2 times, 1.6 times and 1.2 times the cross-sectional area of the cold leg were also examined. This study satisfies a 10 CFR 50, Appendix K requirement.

The second type of sensitivity is an exposure or burnup study. This consists of examining the beginning of life (BOL), middle of life (MOL) and end of life (EOL) fuel condition for the most limiting break size and type as determined in the previous break spectrum sensitivity study.

The third type of sensitivity is a single failure study. The competing single failures for the large break loss-of-coolant accident analyses have been determined by experience (Reference 1.4). These are either: (a) the loss of one ECCS injection train or, (b) the loss of 1 train of low pressure injection. A sensitivity study is performed to confirm that the most limiting of these two is being used.

Finally, the fourth is a convergence criterion study, demonstrating that the value used for this parameter is adequate to produce converged results.

In Chapter 4, key results from base case analyses and sensitivity studies are summarized. Chapter 4 also summarizes how the most limiting large break LOCA case for the SEM/PWR-98 methodology is determined, how the PCT is computed, and how compliance with the LOCA-ECCS criteria in 10 CFR 50, Appendix K for CPSES-1 Cycle 8 is demonstrated.

CHAPTER 2

DESCRIPTION OF THE METHOD

This report describes the application of USNRC-approved, Siemens Power Corporation's ECCS Evaluation model, entitled "SEM/PWR-98" (Reference 1.4), to the Comanche Peak Steam Electric Station Unit 1 (CPSES-1) for the current operating cycle (Cycle 8).

2.1 BACKGROUND

In 1975, the NRC approved use of the Exxon Nuclear Company (ENC) WREM-based generic PWR ECCS Evaluation Model (Reference 2.4). This LOCA Evaluation Model was based on the NRC-developed Water Reactor Evaluation Model (WREM) (Reference 2.9).

In 1976, the ENC PWR model was updated resulting in the ENC WREM-II Evaluation Model (Reference 2.10). The ENC-WREM-II model differed 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 (Reference 2.1). The WREM-IIA differed 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 (Reference 2.3)-EM/FLOOD (WREM-II) calculation was replaced by a similar calculation using REFLEX (Reference 2.7).

In July 1986, the NRC accepted the EXEM/PWR (Reference 2.15) large break ECCS Evaluation Model for referencing of related licensing topical reports. EXEM/PWR is based on WREM-IIA PWR ECCS EM (Reference 2.1). EXEM/PWR updated WREM-IIA in four phases of the transient calculation: (a) stored energy and fission gas release models were revised in the fuel rod model in the RODEX2 code, (b) the NUREG-0630 clad rupture/blockage and a new fuel rod models were added to the RELAP4-EM system blowdown calculation, (c) leakage flow from upper plenum to downcomer was allowed, and new split break and core outlet enthalpy models were used along with a revised carryout rate fraction correlation in the REFLEX code for the reflood period, and (d) the heatup model in TOODEE2 (Reference 2.14) included a revised steam cooling model, NUREG-0630 clad rupture/blockage, a revised radiation heat transfer model, and a revised reflood heat transfer correlation.

In 1998, SPC consolidated the resolution of several issues resulting from the NRC's 1997 inspection (Reference 2.8), and other concerns raised in that time frame, in a revised evaluation model called SEM/PWR-98 (Reference 1.4). Changes were made to address code variability, which included replacing the numerical solution scheme, redefining the calculation of volume flow, combining the system blowdown and hot channel calculations, eliminating the use of the enthalpy transport model by increasing the number of nodes in the heat transfer regions, and improving consistency between code interfaces. The Dougall-Rohsenow correlation was replaced with a revised film boiling correlation named Richert-Franz. The calculation of end-of-bypass was changed to remove unnecessary conservatism and to take credit for residual primary coolant calculated to remain in the system at the time of end-of-bypass. The reactor coolant pump flow degradation model based on Semiscale pumps was replaced with a new model based on tests conducted by CE and EPRI. A new cladding creep model was implemented into RODEX2 to remove some contradictions that existed between the way the re-sinter density was used in EXEM/PWR and the requirements of Regulatory Guide 1.126.

Two concerns with EXEM/PWR were addressed differently by TXU Electric and SPC. These were both in the FCTF correlation within the TOODEE2 code. One concern involved the correlation itself; the other involved an error in the calculation of one of the parameters of the correlation (the so called "Z-equivalent error"). TXU Electric evaluated the effect of the "Z-equivalent error" and concluded it was in the conservative direction and elected not to correct it because the ensuing PCT reduction would be greater than 50^0F and thus require replacement of the Dougall-Rohsenow correlation. In the case of the FCTF correlation, the resolution was the same as that implemented by SPC, i.e, the implementation of a modified 1986 model which involved a linear interpolation of the dependency of the heat transfer coefficient with flooding rate within a certain range of flooding rates. However, TXU Electric implemented its own interpolation because the SPC version of TOODEE2 that had the interpolation also had corrected the Z-equivalent error. These differences become moot when TXU Electric switches to the version of TOODEE2 which is included in SEM/PWR-98.

The present report describes the application of this latest large break ECCS Evaluation model, SEM/PWR-98, to the Comanche Peak Steam Electric Station Unit 1, Cycle 8, the current operating cycle.

2.2 OVERVIEW OF THE METHOD

Like its predecessor, EXEM/PWR, the SPC SEM/PWR-98 methodology consist of a series of computer codes which are linked together to perform loss-of-coolant analysis to demonstrate plant and fuel design conformance to 10 CFR50.46 criteria and Appendix K requirements. The SEM/PWR-98 computer codes and the information transfers are illustrated schematically in Figure 2.1. After the (1) initialization phase, the accident is divided into three additional phases: (2) blowdown, (3) refill, and (4) reflood, for a total of four phases. Typically, each phase consists of a calculation of a system transient behavior and conditions which in turn provide boundary conditions for the detailed core or hot rod calculation. Phases (2) and (3) are separated by the End-of-Bypass (EOBY) event, and the Bottom of Core Recovery (BOCREC) event separates phases (3) and (4).

The discussion that follows will emphasize the changes made to EXEM/PWR to generate SEM/PWR-98. While an overview of the entire accident and methodology are given in Sections 2.2.1 through 2.2.5, this summary is not intended to provide a comprehensive description of the phenomena, issues and/or regulations pertaining to calculations of each phase and/or event. A road map for that purpose is available elsewhere (e.g., Reference 1.4).

Most of the changes are internal to the codes and are automatically incorporated into TXU Electric's methodology simply by use of the updated code versions. All those changes have already been approved by the NRC (Reference 1.1) and are explained in detail in Reference 1.4 and its references. Another type of change is the automation of data transfer between codes, which TXU Electric has developed independently. Finally, other changes involve application of the method and/or input considerations. All changes are summarized in Section 2.3.

2.2.1 INITIALIZATION

The initialization step includes core physics calculations to provide reactor kinetics input (Reference 2.17) and to generate all realistic potentially limiting axial power shapes in order to support the linear heat generation (LHGR) limit as a function of height. The population of shapes is developed through the axial power distribution control analysis (Reference 3.5).

Then, in compliance with the SEM/PWR-98 methodology, the shape peaking at the highest elevation in the core is selected. After that, the selected shape is adjusted upward until the axial power shape curve touches the curve representing the Technical Specification LHGR limit as a function of core height. This final adjustment is done differently by TXU Electric and SPC.

Calculations are required to determine initial fuel conditions for both RELAP4-EM and TOODEE2. These include: fission gas inventory, gap width, crack and plenum volumes. These calculations are performed using the RODEX2 code. The updated version of this code is also part of the SEM/PWR-98 methodology. Steady-state initialization calculations are performed with RELAP4-EM to verify that the desired initial conditions are being computed by the code. These include initial stored energy, flow, and initial pressure distributions. In addition, the steady-state energy balance for each core and steam generator volume is checked.

2.2.2 BLOWDOWN

The analysis of the large break LOCA begins with the analysis of the blowdown phase, which extends from accident initiation to EOBY. RELAP4-EM computes the thermal-hydraulic conditions of the primary and secondary systems during the depressurization following the LOCA. As previously done in EXEM/PWR, RELAP4-EM also computes the thermal performance of fuel rods during the blowdown portion of the accident. In that way 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 EOBY for the Fuel Rod Thermal

Analysis, and (c) to provide average core, hot assembly, and hot rod cladding and fuel temperatures for the reflood calculation.

The SEM/PWR-98 RELAP4-EM system model used for the CPSES-1analysis presented in this report 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 two calculations (system and hot channel) were run separately in the previous methodology but are now run simultaneously in SEM/PWR-98. The RELAP4-EM system calculation also provides the data necessary to compute the EOBY time, mass and energy releases to the containment up to EOBY, and the initial system conditions for the reflood analysis. During blowdown, the containment pressure is computed using a simple RELAP4-EM containment model. The containment pressure obtained from the more accurate refill calculation using ICECON is compared to this pressure and the blowdown calculation is iterated if necessary.

As before, RODEX2 is used to determine the initial stored energy. In SEM/PWR-98, RELAP4-EM now implements a fuel rod model consistent with RODEX2 models for both initial conditions and for the transient conditions that occur during the blowdown. The RODEX2 gap conductance model is now used including the radiation parameters. The basic RELAP4-EM calculation of fuel pellet temperatures accounts for radial variation of fuel density on pellet thermal conductivity and flux depression. The RELAP4-EM fuel rod plenum temperature was updated to better account for the thermal inertia of the plenum spring.

An error in the re-normalization of the actinide multiplier was corrected in the RELAP4-EM decay heat model. The volume average temperature model was also modified to correct an error. The previous model incorrectly included half of the volume of the adjacent gap node. The corrected model only includes the fuel volume region. In EXEM/PWR, the engineering peaking factor was conservatively applied to all rods in the hot assembly. However, by definition, it is a hot spot peaking factor, so that in SEM/PWR-98 it is only applied to the hot spot (actually to the hot rod). TXU Electric has not taken advantage of this last improvement because TXU Electric's engineering hot spot factor is conservatively included into the enthalpy rise factor, which is applied to the entire hot assembly, not just the hot rod.

2.2.3 END-OF-BYPASS

The revised end-of-bypass model is similar to that in EXEM/PWR. Both models define the inlet line volume as the total cold leg volume. Whereas the previous model discarded both the ECCS and non-ECCS water in the entire system at the end-of-bypass time, the new model discards only the ECCS water remaining in the system (i.e., the intact cold leg and downcomer volumes). The calculated residual primary system water is retained. In EXEM/PWR, the end-of-bypass time is defined as the time that sustained positive flow occurs from the upper to the lower downcomer volume less the time required to fill the cold leg from the ECCS injection point to the reactor vessel. In SEM/PWR-98, the end-of-bypass time is defined as the minimum of the time that sustained positive flow occurs either from upper to lower downcomer as above or, alternatively, from broken cold leg to upper downcomer less the time required to fill the portion of the cold leg from the ECCS injection point to the reactor vessel which is not occupied by residual coolant water.

2.2.4 REFILL

The period between the end-of-bypass and the beginning-of-core-recovery(BOCREC), or the start of reflood, is termed the refill portion of a LBLOCA. The start of reflood begins when the lower plenum and the downcomer below the core inlet fill with liquid. 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 accounted for in the calculations by delaying ECCS injections. Once the initial time of reflood has been established, the ECCS injection rates to the core during reflood can be obtained from the total injection rate of ECCS fluid into the downcomer. The flow to the downcomer differs from the ECCS injection rate due to the changing level of water stored in the cold legs. The SEM/PWR-98 refill model is essentially unchanged from EXEM/PWR.

The determination of the containment backpressure for the refill period is done by ICECON/CONTEMPT-LT (Reference 2.5), which is included in the RFPAC (Reference 2.16) code.

2.2.5 BOTTOM OF CORE RECOVERY (BOCREC)

Following the EOBY as determined in the RELAP4-EM system blowdown calculation, downflow is calculated in the downcomer region of the reactor vessel. ECCS 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 (Reference 2.6), which is also a part of the RFPAC code. 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 fraction, 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.

2.2.6 REFLOOD

This calculation considers the rate of reflooding of the reactor core and establishes core fluid conditions for the heatup calculations. The REFLEX code, also a part of RFPAC, is used to perform the reflood analysis.

REFLEX utilizes a quasi-steady state solution of the mass, momentum and energy equations for PWR reactor systems. The code is designed to be applicable for analysis of a PWR loop configuration. 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 as required.

The rod thermal analysis during the refill and reflood period is performed using the TOODEE2 computer code. TOODEE2 uses the EOBY temperatures from the hot channel and performs an adiabatic heatup calculation, 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 Zircaloy. The energy equation is solved to determine the fuel rod thermal response. The code considers conduction within solid regions of the fuel, radiation and conduction across gap regions, and convection and radiation to the coolant and surrounding rods, respectively. Radiation and convective heat transfer are assumed never to occur at the same time at any given axial node. Radiation is considered only until the convective heat transfer surpasses it. Based upon the calculated stress in the cladding (due to the differential pressure across the clad) and the cladding temperature, the code determines whether the clad has swelled and ruptured. Whenever rupture is calculated and the flooding rate drops below 1 in/sec, only steam cooling is allowed downstream of the ruptured node. This assumption 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, namely: 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).

TOODEE2 thus calculates the temperature distribution in the hot rod during the refill and reflood portion of the transient. The TOODEE2 calculation begins at EOBY time. External radiation heat transfer is modeled during the refill and reflood periods until the convective heat transfer coefficient becomes larger than the radiation coefficient at any given axial node. TOODEE2 rod temperatures are initialized with the hot rod temperature distribution from the blowdown calculation at EOBY. Radiation sink temperatures are computed from RELAP4-EM core temperatures at the EOBY time and the total pin power distribution.

With the exception of code changes discussed in Section 2.3, the overall SEM/PWR-98 reflood model is essentially unchanged from EXEM/PWR.

2.3 OVERVIEW OF THE CHANGES

The basic reason for the changes to EXEM/PWR ECCS evaluation model was to reduce the relatively large variability in PCT results due to small changes in the RELAP4-EM input. However, in reducing the large PCT variability, results showed that PCTs for some plants would decrease in excess of 50 ^{0}F. According to the current ECCS rule, a decrease of this magnitude required that the non-conservatism of the Dougall-Rohsenow film boiling heat transfer correlation used in EXEM/PWR also be addressed. Thus, additional changes were needed to revise or replace the Dougall-Rohsenow correlation and to ameliorate the adverse effect of implementing a more conservative film boiling correlation. In addition, the 1997 NRC inspection at SPC resulted in additional new commitments for revised ECCS models which were implemented with the revised model. The following sections describe the changes made to reduce variability, the revised film boiling heat transfer correlation and an offsetting revision to the calculation of end-of-bypass, an improved RCS pump degradation model to address issues from the inspection, and revised creep coefficient input for RODEX2 which will be implemented concurrently with changes that fully implement NRC Regulatory Guide 1. 126 for densification.

2.3.1 VARIABILITY CHANGES

Although this has not been the case for CPSES, SPC encountered analyses using EXEM/PWR where small inconsequential changes in input or calculated transient conditions resulted in large calculated changes in PCT. The changes made to reduce variability were intended to eliminate discontinuities which were identified to be a source of PCT variability.

2.3.1.1 Numerical Scheme

The revised numerical scheme reduced variability by providing a more efficient and consistent calculation which assures convergence, eliminates the "water packing" phenomena, and is more robust. This solution method includes logic to measure the degree of convergence at the end of each time step. The measured level of convergence is used to determine an optimal time step. Time step variation was found to trigger variability in EXEM/PWR. These occurred because: (1) time step convergence had to be determined by the user, (2) the "water packing" phenomena tended to make the calculation oscillatory which reduced time steps when cold ECCS water was being injected, and (3) code failures were frequent, requiring multiple restarts with varying time steps to run the calculations. SPC experience indicates that the revised numerical scheme provides much more consistent, converged and reliable calculation results with reduced time step variability.

The numerical solution method previously used in RELAP4-EM was based on a method proposed by Porsching, et al (Reference 2.12). The Porsching method is limited by its use of previous time step values for the junction enthalpies and by not requiring the achievement of a certain degree of convergence before advancing to the next time step. Both of these limitations were eliminated in the RELAP4-EM upgrade by extending the Porshing technique to represent a Newton's Method type of iterative solution where advanced time steps values are used and by adding logic to RELAP4-EM to measure the degree of convergence at the end of each time step. The term "convergence" is used here in the sense that for the given time step size, the nodal pressure, mass and energy terms in the finite difference form of the mass, energy and momentum equations satisfy these equations to within a specified maximum error. There are two aspects that need to be tested. The first is the error introduced by the use of previous iterate junction enthalpies in the energy conservation equation. The second is the determination of the degree to which the pressure linearization assumptions used in the RELAP4-EM solution method agree with the actual non-linear pressure as determined from the equation of state. Obviously, these are aspects of convergence with respect to time step, but not convergence with respect to nodalization.

The Porsching technique is based on approximating the finite difference form of the mass, energy and momentum equations as linear functions of junction flows. It is therefore important to test the accuracy of this approximation by comparing the linearized pressure used in the flow solution matrix with the equation of state pressure obtained from the nodal mass and energy resulting from these flows.

The measure of convergence of the energy equation is obtained by substituting updated values of junction enthalpies and flows into the non-linear form of the energy equation and comparing this with the rate of change of energy resulting from the solution of the linearized energy equation.

The measured level of convergence is used to determine an optimal time step. This method is based on comparing the measured convergence with a reference value equal to half the user specified maximum convergence criteria. If the measured convergence is smaller than the reference value, the time step is increased, while if the measured convergence is greater than the reference value, the time step is decreased.

The RELAP4-EM upgrade contains logic to recover from previously fatal occurrences of exceeding the bounds of the water property tables, as well as failure to converge in a given number of iterations. This involves trapping these occurrences and resetting the conditions to the beginning of the time step, including the restoration of the special evaluation model heat transfer flags to their condition at the beginning of the time step. The calculation then proceeds with a smaller time step. A lower bound of .000005 seconds is applied to the time step in this process.

2.3.1.2 <u>Volume Flow Definition</u>

Volume average flow is used in the momentum equations and in the heat transfer and CHF calculations. Both of these definitions were changed. The new definitions are summarized in (Reference 1.4).

2.3.1.3 <u>Combined System and Hot Channel Calculations</u>

This model change is in the methodology for applying the SEM/PWR-98 codes. That is, there are no changes to the analytical models or simulation of the basic physics associated with this change. The reason for separate system blowdown and core blowdown (hot channel) calculations is historical. At the time the original Exxon Nuclear Company Water Reactor Evaluation Model (Reference 2.4) was developed, computer capabilities significantly limited the models. The concept was that a blowdown calculation could be performed for the entire primary coolant system using an average core representation and relatively large time steps. The results of this calculation could then be imposed as time dependent boundary conditions on a separate but more detailed core calculation including the hot rod. This core or hot channel calculation could be performed at the much smaller time steps required to calculate the detailed core heat transfer. Also for relatively small changes in core parameters, the system blowdown calculation is often nearly unaffected. Thus, with the separated model, sensitivity studies could often be conducted varying only the hot channel model and using a single set of system boundary conditions. This reduced the number of system blowdown calculations necessary to do an analysis with the separated model.

The concept has been integral to SPC's PWR LOCA methodology for over 20 years since the original WREM models were approved, and generally has worked satisfactorily and served its purpose. However, computer capabilities have changed dramatically during this time, and the limitations necessitating separate calculations no longer exist.

Also, there is a downside to the use of separate core calculations which can be eliminated by combining the calculations. Imposing vessel plenum boundary conditions from a system calculation on the core calculation is an approximation and the boundary conditions from the system plot restart file must be interpolated to provide appropriate time values for the hot channel calculation which is using different time steps. This technique introduces some variability into the LOCA calculations due to the output frequency to the plot restart file and the relative time steps in the system blowdown and hot channel calculations. The separate models also require more computer runs to be made, increased setup time, and more data handling than the combined calculation. All of these factors are now unnecessary and contribute to the potential for errors.

Implementation of the combined calculation is directly through input of the detailed core model into the system blowdown calculation. The only code changes needed to do this are increased dimensions to allow the combined calculation to be input. These increased dimensions were also required for other changes.

2.3.1.4 Elimination of Enthalpy Transport Model (Increased Nodalization)

The enthalpy transport model was a significant source of variability in the EXEM/PWR model. Because of the computer limitations existing at the time the original WREM model was approved, the degree of nodalization of the system and the core was minimized. This resulted in the EXEM/PWR core nodalization consisting of 6 fluid nodes (10 for CPSES)[2] broken down into 2 radial regions: one, representing a single hot assembly and the other, the remainder of the core, each with 3 large axial nodes (5 for CPSES) in each radial region. With this very

2

It is possible that the larger number of nodes used in the CPSES Model contributed significantly to the fact that the effect of the enthalphy transport model on CPSES LBLOCA results was very small.

coarse nodalization, it was necessary to apportion the heat added to a fluid volume to provide appropriate fluid volume temperatures and achieve calculated steady-state conditions. The enthalpy transport model was included for this purpose. Experience has shown that while the enthalpy transport model achieves a reasonable steady state, under some transient conditions the use of this model yields unrealistic results. It has also been observed that for some conditions, the RELAP4-EM code is programmed to turn off the enthalpy transport model during a transient. This results in a discontinuity which leads to a significant variation in results.

The proposed resolution of this issue is the approach of the more recent codes such as TRAC and RELAP5 which simply use more detailed axial nodalization to provide a good simulation of the initial steady-state conditions without requiring an enthalpy transport model. With the increased computer capabilities, this can now also be done in RELAP4-EM by simply discontinuing use of the enthalpy transport model, and using a significantly increased number of axial nodes in the core and steam generators. Both of these changes are implemented through input to the existing codes, with the only required code changes being to increase the dimensions of appropriate variables to permit the greater number of fluid volumes, junctions, heat structures, etc. associated with the greater number of nodes.

Eliminating the enthalpy transport model removes the discontinuities and unrealistic behavior observed in some calculations. Increasing the nodalization not only allows suitable steady-state conditions, but also provides a more accurate simulation of core power distributions and fluid conditions in both the core and steam generators. The 3 axial nodes (5 for CPSES) in the previous core model were expanded to 12 or more axial (16 for the current CPSES-1 appliation) nodes for the revised model. This core axial nodalization was determined optimal in order to achieve convergence based on sensitivity calculations conducted by SPC (Reference 1.4).

2.3.1.5 Consistency Improvements Between Codes: RODEX2, RELAP4-EM and TOODEE2

The consistency between RODEX2 and RELAP4-EM fuel rod models was improved. For the EXEM/PWR model, the RODEX2 fuel rod input was only applied to the hot channel calculation. The new model permits the RODEX2 model to be applied to fuel rods representing the average core, the average rod in the hot assembly and the hot rod. Provision was also made to model a second rod in the hot assembly. This feature was added for possible analysis of gadolinia bearing rods. RODEX2 data are transferred to RELAP4-EM for each fuel node. In the EXEM/PWR model, the hot channel calculation obtained RODEX2 data at the elevations corresponding to the RELAP4-EM fuel nodes. The new model supplies RODEX2 data computed at the average power in each RELAP4-EM node. This eliminates possible inconsistencies in the application of RODEX2 data in the RELAP4-EM calculation and makes all RELAP4-EM calculations consistent with the approved RODEX2 models. The new model includes use of an emissivity value of 0.90 for the inner clad surface as is currently programmed in the approved RODEX2 model in contrast with a value of 0.67 used in previous approved EXEM/PWR models.

The transfer of the fuel density and power depression data from the RODEX2 code was modified. Previously RODEX2 used a different radial nodalization of the fuel region than did RELAP4-EM. Therefore, both the RODEX2 normalized radial nodalization and the corresponding fuel density and power depression data were required by RELAP4-EM. RELAP4-EM would then build fuel density and power depression tables for the RELAP4-EM nodalization by interpolation. The RODEX2 code was modified to provide fuel model data consistent with the RELAP4-EM fuel rod nodalization, thus eliminating the need for interpolation in RELAP4-EM.

The ability to specify the rod type for the RODEX2 model was added to the input. The rod type specifies the corresponding fuel density and power depression data tables to be used. Four rod types are allowed, with type 1 denoting an average rod, type 2 denoting an average rod in the hot bundle, and type 3 denoting the hot rod in the hot bundle. Type 4 is available for future use.

Logic was added to prevent convergence failure in the RODEX gap conductance model when the gap is calculated to be open or closed on consecutive iterations. Convergence failure is caused by the physical discontinuity in the gap conductance between the open and closed states. Iteration is required between the fuel average temperature and the gap conductance since each is a function of the other. The converged value must satisfy both relationships simultaneously. However, if the fuel temperature from a closed gap results in the gap opening and the open gap conductance raises the fuel temperature to close the gap, simultaneous solution of these relationships is impossible. Therefore, if after more than 5 iterations, a gap opens that was closed on the previous iteration, the gap will be held closed on all subsequent iterations at this time step.

The pellet and cladding surface roughness factors were added to the user supplied input. These were previously set internally.

The EXEM/PWR LBLOCA methodology computed plenum spring constants using RODEX2 clad nodal temperatures and passed these values to RELAP4-EM. The value for the top axial node is used by RELAP4-EM. RODEX2 was changed to compute the plenum spring constant based only on the fuel rod plenum temperature and to pass this value to RELAP4-EM.

The RODEX2 fuel models are not incorporated in TOODEE2; however, RODEX2 does supply input to the TOODEE2 code. One minor change to the input was implemented. The RODEX2 code was modified to compute the cold gap width for TOODEE2. The gap width for the peak power node is passed to TOODEE2. Previously, the EXEM/PWR LBLOCA calculations incorrectly computed the gap width for TOODEE2 at hot zero power conditions and effectively double accounted for cladding elastic deformation in TOODEE2.

2.3.2 REVISION OF DOUGALL-ROHSENOW CORRELATION

10 CFR 50 Appendix K requires that when changes or error corrections are made to evaluation models which reduce the calculated peak clad temperatures (PCTs) by more than 50 ^{0}F, the known non-conservatism associated with the use of the Dougall-Rohsenow film boiling heat transfer correlation must be removed. For some plants, the variability changes decreased PCT by 50 ^{0}F or more; therefore, to address the non-conservatism of the Dougall-Rohsenow correlation, SPC developed a replacement correlation, which they called the Richert-Franz correlation.

10 CFR Part 50, Appendix K states, in part:

"**Post-CHF Heat Transfer Correlations.** Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analysis. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data."

In consideration of these requirements, key elements in the development of the Richert-Franz correlation were:

a. The correlation must be conservative to the measured data as defined in 10 CFR 50 Appendix K.

b. The correlation must exhibit no discontinuities between heat transfer correlations when qualities change from less than zero to zero and from one to greater than one for filmboiling calculations in RELAP4-EM. Once RELAP4-EM switches to film-boiling within a node, the only other possible heat transfer mechanism allowed is single phase steam convective heat transfer. If single phase steam is no longer present in the node, RELAP4 can only go back to film-boiling. The code can alternate between single phase steam and film-boiling but never out of film-boiling to something else once that lockout has occurred. It is assumed that values for film-boiling convective heat transfer coefficients are conservative with respect to all other methods of heat transfer and therefore the correlation bounds all other states excluding single phase steam convection.

c. The correlation interface must be compatible with implementation in RELAP4-EM.

d. The correlation should never give a non-physical result for film-boiling phenomena within the RELAP4-EM parametric space. This means that the correlation should calculate increasing or decreasing values of h when the system would truly exhibit that behavior (i.e. when the flow rate goes up, the value of the heat transfer coefficient should go up). The correlation should also not calculate a trend which is not expected in the real system.

e. [

]

The actual correlation and its validation can be studied elsewhere (Reference 1.4).

[

] Therefore, the implementation is also a conservative approach to the transition between Richert-Franz and Dittus-Boelter.

2.3.3 REVISED END-OF-BYPASS MODEL

The Code of Federal Regulations, 10 CFR 50 Appendix K (Section C.1.c), defines how the end-of-bypass time is to be determined:

"**End of Blowdown.** For postulated cold leg breaks, all emergency water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculation be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the "end of bypass," after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining the "end of bypass" include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number."

The end-of-bypass model in SEM/PWR-98 is similar to that in EXEM/PWR. Both models define the inlet line as the total cold leg. Whereas, the previous model discarded both the ECCS and non-ECCS water in the entire system at the end-of-bypass time, the new model discards only the ECCS water remaining in the system (i.e., intact cold leg and downcomer volumes). The calculated residual primary system water in the system is retained. The previous model defines the end-of-bypass time to be the time that sustained positive flow occurs from the upper to the lower downcomer volume, less the time required to fill the cold leg from the ECCS injection point to the reactor vessel. The revised model defines the end-of-bypass time to be the minimum of the time that sustained positive flow occurs from the upper to the lower downcomer volume or the time that sustained positive flow occurs from the broken cold leg to upper downcomer volume, less the time required to fill the portion of the cold leg from the ECCS injection point to the reactor vessel which is not occupied by residual coolant water.

To compute the end-of-bypass time the following steps are taken:

a. The times that sustained positive flow occurs from the upper to the lower downcomer volume and from the broken cold leg to the upper downcomer volume are determined by RELAP4-EM. The minimum of these two times is used to set the end-of-bypass time. Two conditions must be satisfied for either time to be used: 1) the intact loop accumulator flow must have begun, and 2) the flow must remain positive for 0.5 seconds (the time period can be less than 0.5 seconds if the RELAP4-EM calculation has terminated on an end trip signal).

b. The total (steam plus liquid) residual non-ECCS water in the intact cold legs, broken cold leg and upper downcomer volumes are computed as a function of time. It is assumed that the incoming flows to a RELAP4-EM volume mix perfectly with the water in the volume, and leaves at the liquid fraction in the perfectly mixed volume (i.e., homogeneous model). For these calculations, the average of RELAP4-EM junction inlet flows at the start and end of the current time step are used. The total outlet flow is computed such that a mass balance is retained in the volume.

c. Since only the liquid fraction of the total water is required, the liquid fractions values are computed.

d. The end-of-bypass time is set to be the minimum of the two flow reversal times less the time required for the ECCS fluid to fill the portion of the intact cold leg volume not occupied by non-ECCS water from the injection point to the vessel entrance. The end-of-bypass time is set to the end-of-bypass time computed for the intact loop. To compute the time required to fill the cold leg, a table is first created of the volume of ECCS water added to the primary system versus time. The volume of ECCS is computed as the product of the total RELAP4-EM ECCS mass added to the cold leg and the cold leg specific volume (ft^3/lbm), where the specific volume is evaluated at the pressure in the cold leg at the flow reversal time. The time to fill the cold leg is set to the time when the volume of ECCS water at the flow reversal time less the volume at the end-of-bypass time equals the portion of the cold leg volume which is to be filled.

e. The residual water remaining in the primary system at the end-of-bypass time is calculated by summing the water masses in the lower downcomer and lower plenums, the non-ECCS water in the upper downcomer, and the non-ECCS water in the intact cold leg volumes. The residual water is then converted to an equivalent volume by dividing the mass by the saturated liquid density at the end-of-bypass containment pressure. The density at the end-of-bypass containment pressure is used since the RFPAC calculation sets the primary system pressure to the containment pressure during the refill portion of the transient.

2.3.4 REVISED PUMP DEGRADATION MODEL

SPC's EXEM/PWR evaluation model used a two-phase degradation model based on early Semiscale pump data. Since the time that the original WREM model was approved, two-phase testing has been performed on centrifugal pumps that are much more representative of reactor coolant pumps than the small scale pumps used for the Semiscale program.

During NRC's inspection at SPC in 1997, the applicability of the Semiscale pump degradation data became an issue. In response to the inspection (Reference 2.8), SPC agreed to address the pump degradation issue at the time the revised model, SEM/PWR-98, was submitted. To do this, the RELAP4-EM pump model was upgraded to apply a new two-phase pump performance degradation model based on data from the CE-EPRI Pump Two-Phase Performance Program. The CE-EPRI pump data are more representative of PWR reactor coolant pumps than the Semiscale pump data.

RELAP4-EM treats two-phase pump head degradation by first calculating the single-phase pump head and then subtracting a degradation correction term. This approach makes use of either user supplied single-phase homologous head curves and two-phase head difference curves or default curves which are built into RELAP4-EM. Default single-phase homologous curves are available for either Westinghouse or Bingham reactor coolant pumps. The built in difference curves supplied by RELAP4-EM are based on Semiscale two-phase degradation data. In the conventional RELAP4-EM formulation, the head difference multiplier, MH, is solely a function of pump void fraction and must be supplied by the user.

The CE-EPRI pump two-phase degradation test results are based on pumps more similar to PWR reactor coolant pumps, and make possible a more realistic analysis of reactor coolant pump two-phase flow degradation effects. The CE-EPRI test pump is more closely scaled to a PWR reactor coolant pump than the Semiscale pump in three key areas: pump size, pump type (radial vs. mixed flow), and pump specific speed. The new degradation model based on CE-EPRI data can be implemented in RELAP4-EM by use of the existing formulation with the exception of the head difference multiplier, MH, which in the new formulation includes a pump speed/flow and inlet pressure dependency.

RELAP4-EM calculates MH by using a formula, based on the CE-EPRI data, which incorporates all of the above mentioned dependencies.

2.3.5 CLADDING CREEP MODEL IN RODEX2

SPC changed the cladding creep model selection and revised the creep coefficients used in the RODEX2 code for PWR LBLOCA. The intent of this change is to introduce a thermal creep component for SPC's cold worked stress relieved Zircaloy-4 cladding. Application of this cladding creep model is intended to permit full compliance with the NRC guideline on densification (NRC Regulatory Guide 1.126) without unnecessarily penalizing LOCA analysis results. In conjunction with the creep model change in RODEX2, SPC modified the densification model described in Reference 2.2, Supplement 1, page 10 to be consistent with the Nuclear Regulatory Guide 1. 126. The value used in RODEX2 for the maximum in-reactor densification is equal to or greater than the upper one-sided 95/95 tolerance limit on the resintering density change measurements for a reload. The creep parameters were determined from a series of out-of-pile bi-axial burst and creep tests carried out on both non-irradiated and irradiated fuel rod cladding samples, as well as post irradiation creepdown results from SPC 17 x 17 rods irradiated for four cycles.

2.3.6 OTHER CHANGES

Other, less significant changes have been made in the models and code input. These include the changes described in the following paragraphs.

2.3.6.1 Inertial Flow Estimate for Critical Flow Model

The RELAP4-EM flow choking model requires an estimate of the end of time step flow to determine if the choking model is to be used for a junction. If the flow at the beginning of the time step is zero, the explicit form of the flow estimate will have zero friction, which results in a very large flow estimate. Since the explicit form of the estimate can produce erratic values, a test was included to limit the change in flow over a time step to between the maximum of five times the flow at the beginning of the time step and one one-hundredth of the corresponding choked flow.

When the beginning of time step flow is zero, this limit will always be less than the choked flow, and hence the choking model will not be applied. However, when the flow matrix is solved for the inertial flows, the resulting flow can exceed the corresponding choked value.

In order to avoid this situation, a backward differenced estimate of the flow was implemented. This correction of the inertial flow estimate has little effect on calculated results since it occurs when the flow at the beginning of the time step is near zero. Substantial flow develops over a few time steps, and thus the possibility for using inertial flow rather than choked flow would exist only over a very small time interval with no impact on overall system behavior and PCT.

2.3.6.2 Bubble Mass Integration Model

The RELAP4-EM Bubble Mass Integration Model was modified to solve a quadratic equation instead of two linear approximations for the determination of the bubble mass and the bubble density at the mixture level.

Logic was added to mitigate the consequences of bubble model degeneracies for the two special cases of a junction at the top of a nearly all liquid volume and a junction at the bottom of a nearly all vapor volume. In either case, an insignificant change in the mixture level would cause a large step change in junction enthalpy from saturated liquid to vapor or vice-versa. Such a discontinuity can introduce non-physical oscillations in the calculation and prevent convergence since the junction properties can alternate across the discontinuity on successive iterations. Therefore, the model was modified to ramp the junction properties to the volume homogeneous properties at very high or low average volume quality.

The smoothing function used is piecewise linear such that 90% of the transition occurs between .05 and .025 ft of the top or bottom of the volume, while the final 10% of the jump occurs over the final .025 ft. This provides a simple numerical simulation of the physical roughness of the liquid surface during this transition, and has a negligible impact on the solution since it is only applied over a small distance.

2.3.6.3 Pump Model

The pump model calculates the new time step pump head based on the previous volume average flow. However, the pump volume average flow is strongly dependent on the pump head, and the explicit coupling of the two may result in diverging values for pump head and pump volume flow. This behavior was eliminated by including the pump head term in the Jacobian matrix.

2.3.6.4 Inclusion of BLOCKPWR into RELAP4-EM

In EXEM/PWR the BLOCKPWR code generated cladding swelling and rupture tables for use in RELAP4-EM to calculate rupture and the pre-rupture cladding strain. These calculations are now included in RELAP4-EM to eliminate the need for a separate auxiliary code.

The additional input required for this optional model is:

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

Outside diameter of instrument tube

Outside diameter of guide tube

Cladding temperature

The model is based on hoop stress data at temperature ramp rates of 0.0 and 28.0 ^{0}C/sec and burst strain data at temperature ramp rates of 10.0 and 25.0 ^{0}C/sec. The model linearly interpolates data between the two ramp rates. A ramp rate of 0.0 ^{0}C/sec is the most conservative value since this leads to swelling and rupture at lower cladding hoop stresses.

2.3.6.5 <u>Actinide Multiplier Normalization Modification</u>

An error was identified in the re-normalization of the actinide multiplier in the RELAP4-EM decay heat model. The error was corrected by eliminating an erroneous normalization used in the calculation of coefficients for the decay heat and actinide terms.

Previously the actinide terms were multiplied by a factor greater than 1.0, depending on the user supplied value for UDUF (breeding ratio). Since this multiplier is not required by 10CFR50 Appendix K, three options were added. The first option removes the greater than one multiplier on the actinide coefficient (IOLD = 0). The second option retains the erroneous multiplier (IOLD = 1, for use in evaluating the error) The third option applies a true 1.2 multiplier on the actinide coefficients (IOLD = 2). SEM/PWR-98 uses the option to apply a 1.0 multiplier to the actinide decay heat. This is consistent with the post blowdown decay heat used in EXEM/PWR.

2.3.6.6 <u>Fuel Rod Volume Average Temperature Model</u>

The volume average temperature model was modified to correct an error. The previous model incorrectly included half of the volume of the adjacent gap node. The corrected model only includes the fuel region volume.

2.3.6.7 Engineering Factor

The engineering peaking factor accounts for manufacturing tolerances and other geometric uncertainties. In EXEM/PWR LBLOCA analyses, the engineering factor was conservatively applied to all rods in the hot assembly. However, by definition it is a hot spot peaking factor. Hence, for SEM/PWR-98 methodology calculations, it will be applied only to the hot spot or hot rod.

The engineering factor is applied in two places in the SEM/PWR-98 methodology. The first place is in defining the relative powers for the average core (PF_{ac}), hot assembly (PF_{ha}) and hot rod (PF_{hr}) regions for the RELAP4 blowdown analyses.

The second place the engineering factor is used is in defining the radial peaking map used to compute the radiation model constants and radiation sink temperatures for the TOODEE2 hot rod calculation. The EXEM/PWR LBLOCA calculations used a conservatively flat pin power distribution to compute the radiation model constants and radial sink temperatures. [

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In the present application for CPSES, the engineering factor was included with other factors and was not separated out as explained above. Although an assessment showed this effect to be negligible for CPSES (~ 2 ^{0}F in PCT), TXU Electric intends to follow the SPC approach, in future applications.

2.3.6.8 RFPAC SIS Flow Logic Correction

The PREFILL portion of RFPAC was originally designed assuming that the HPSI and LPSI flows would begin before the accumulators were exhausted. If the HPSI and/or LPSI flow is initiated after the accumulator flow ends, two potential problems were identified and corrected in RFPAC:

f. PREFILL uses an open channel hydraulic flow calculation to compute the level of water that exists in the cold leg based on the flow to the cold leg. PREFILL computes the flow from the cold leg to the downcomer as the sum of the inlet cold leg flow and the flow (averaged over the time step) required to change the mass in the cold leg to support the computed cold leg level. If the flow increases significantly during a time step (as occurs when the HPSI and/or LPSI flow begins after accumulator injection stops) , the open channel hydraulic flow calculation computes a higher liquid level in the cold leg. The result is that a negative flow to the downcomer is computed for the time step to account for the change in liquid level. To correct this problem, the flow rate to the downcomer is held constant at the flow rate prior to the increase in the cold leg level. This flow is maintained for the time period required by the new flow minus the old flow, to fill the cold leg to the new level. After this time period, the downcomer flows are computed as previously described.

g. To prevent non-condensibles from entering the primary system when the accumulators empty, a valve is added to the junction connecting the accumulator and accumulator line. This valve is closed when the liquid level reaches a prescribed level in the RELAP4-EM calculation. The PREFILL code empties the mass of water remaining in the accumulator and accumulator line by extending the accumulator flow tables using the last computed accumulator flow. The table is extended further to empty the cold leg to the new level computed by the open hydraulic channel calculation. A problem can occur if the extended accumulator flow ends near the time when the HPSI and/or LPSI flow begins. The calculation is performed in two passes. The first pass dumps the cold leg to the level supported by the cold leg flow immediately after the accumulator valve closes. The second pass dumps the cold leg to the level supported by the cold leg flow at the end of the extended flow period. If the cold leg level is higher at the start of the second pass than at the start of the first pass, the time for the extended accumulator flow is reduced. This problem occurs when the PREFILL code fails to reset the accumulator flow to zero between the times that the first and second passes terminate the extended accumulator flow.

2.3.6.9 SWMDEN

SWMDEN is defined as the asymptotic fuel density after full densification and full accommodation of the solid swelling by the as-fabricated fuel porosity. SWMDEN is an input parameter in RODEX2. The value of SWMDEN is determined by the processes used when fabricating the fuel. [

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2.3.6.10 Minor Code Modifications

A number of minor code modifications were made in RELAP4-EM. These are not model changes and will not affect calculated results.

a. RELAP4-EM was re-dimensioned to allow up to 150 volumes, 200 junctions, 50 heat conductor geometries, 150 core conductors and 200 heat conductors. Leak and fill tables were increased to allow 50 entries each. The pump homologous curve tables were increased to allow 20 data pairs per table.

b. All common block coding was moved to INCLUDE files, and all subroutines using DATA statements to initialize data in common blocks were changed to BLOCK DATA routines to conform with ANSI FORTRAN77 requirements.

c. Minor modifications were made to provide variable naming consistency and to add new plot variables to the plot/restart file.

d. The "plotr4m" header file was renamed, and the plot smoothing routines were re-dimensioned to accommodate larger plot data records.

In addition, dimensions were increased in other SEM/PWR-98 codes to accommodate the increased detail in the calculations, and plot/restart files were changed to include a job identification number, computer hardware identification, and date and time stamp to improve traceability of the calculation.

2.4 DESCRIPTION OF THE MODELS

2.4.1 CPSES-1 RELAP4-EM SYSTEM BLOWDOWN MODEL

The Comanche Peak Steam Electric Station has two Westinghouse pressurized water reactors. Both units are typical 4-loop plants. Their current rated thermal power are 3411 MWt for CPSES-1 and 3445 MWt for CPSES-2.

The CPSES-1 RELAP4-EM system blowdown model reflects a considerable amount of engineering insight and experience and incorporates:

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

b. Careful consideration of the guidelines set forth by SPC for the application of their methodology (Reference 1.4).

This section describes the RELAP4-EM system blowdown base input model for the Comanche Peak Steam Electric Station Unit 1 (CPSES-1). The discussion of this model is divided into the following sub-sections:

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

2.4.1.1 Volumes, Junctions and Heat Structures

Figure 2.2 shows the CPSES-1 nodalization diagram for the base input model which is comprised of 79 volumes, 104 junctions and 114 heat structures. The model is consistent with that approved by the NRC in connection with SEM/PWR-98 (Reference 1.1). Table 2.1 identifies the particular volumes, junctions, and heat structures associated with the important regions or systems. Table 2.2 summarizes the most important parameters of the CPSES-1 NSSS model volumes and junctions. These parameters were calculated using information from plant drawings, design basis documents, vendor documents, Technical Specifications and Final Safety Analysis Report.

2.4.1.2 Core Power

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 with a 1.02 multiplier to the CPSES-1 power level of 3411 MWt and a 1.01 multiplier to the CPSES-2 power level 3445 MWt so that both plants are effectively analysed at an initial power of 3479.5 Mwt. (See Section 2.5.3 for a discussion on these multipliers).

A decay heat 1.2 multiplier is also applied, per 10 CFR 50, Appendix K requirements. The model accounts for the reactivity effects associated with change in moderator density and in fuel temperature. The effects are evaluated on a core average, cycle specific basis using the reactor physics methodology and associated uncertainty factors presented in Reference 2.17 to assure conservatism. For the analyses presented herein, moderator and fuel temperature reactivity feedbacks representative of the CPSES-1 core for Cycle 8 have been selected and are shown in Tables 2.3 and 2.4. Scram reactivity is conservatively neglected in the model.

2.4.1.3 Emergency Core Cooling Systems

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

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

Two competing "single failures" are considered in this report. In future analyses, only the most limiting of the two will be utilized.

The first involves a loss of offsite power, assumed to occur coincidentally with the break. One diesel generator train is removed on the assumption that it fails to start. Therefore, only one train of safety systems are represented in the present NSSS model. This loss of 1 train is the assumption made in order to satisfy the single failure criterion and it will include the failure of 1 train of containment spray pumps.

The second single failure assumption considered is the loss of 1 train of RHR pumps only. In this case all containment spray trains function.

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 ^{0}F) in order to minimize the containment back-pressure. The flow versus pressure values for each injection system, which are given in Table 2.5, reflect spillage of injection to the broken loop.

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 PCTs. Accumulator water temperature is assumed to be 88 ^{0}F, based on the lowest containment temperature recorded over a several year period. The minimum Technical Specifications (Reference 3.4) tank water volume (6119 gal.) is also used.

2.4.1.4 <u>Trips and Delays</u>

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 2.6.
6. Accumulators inject at the minimum accumulator set pressure.

2.4.1.5 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 Figure 2.4. The input is identical to that of the system volumes. The cold legs are time-dependent volumes with pressures initially set by the previous blowdown calculation and at containment pressure during the transient.

2.4.1.6 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.

2.4.1.6 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 14.2 psia (conservatively low, since the containment is not sub-atmostpheric), temperature at 88 ^{0}F (based on the lowest containment temperature recorded over a several year period), and relative humidity at 100%. A conservatively large containment volume of 3.063E6 ft^3 is used. 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.

2.4.1.7 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.

2.4.1.8 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 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.

2.4.1.9 <u>REFLEX</u>

The nodalization diagram for REFLEX is shown in Figure 2.5. 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 2.2 and 2.5. 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 (Reference 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.

2.4.1.10 <u>TOODEE2 Model</u>

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 2.7 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 (Figure 2.3).

The fuel pellet is divided into 8 radial rings (nodes) in which the last radial line location includes the gap. The first inner fuel pellet is node 2, and gridline 1 is identified as the pellet centerline. The last gridline is identified as the clad outer radius. The cladding is divided into 2 radial rings as required by EXEM/PWR. The radial nodalization scheme is shown in Figure 2.6.

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2.5 PLANT SPECIFIC CONSIDERATIONS

2.5.1 SOFTWARE QUALITY ASSURANCE

As part of its integrated Appendix B Quality Assurance program, TXU Electric has procedures in place for identifying, approving, implementing and reporting evaluation model modifications. These procedures also provide guidance for implementation of software developments/modifications, a process for software validation and installation of test cases and a configuration baseline package.

A configuration baseline calculation package was prepared, consistent with these procedures, which verifies the installation of SEM/PWR-98 codes (RELAP4-EM, RDX2LSE, RFPAC, FISHEX, TOODEE2 and TEOBY) on the TXU computing platforms and confirms that the codes' results are substantially identical to those obtained by SPC on their platforms.

2.5.2 MIXED CORE ANALYSES

One of the main purposes of performing "in-house" reload evaluations is the ability to optimize core design. This results in the co-existence of fuel assemblies by different vendors and even slightly different designs from the same vendor. Therefore, a combination of: (a) a representative mixed core input and, (b) multiple calculations, is performed, as required, in order to realistically account for multiple fuel types. Multiple calculations will be performed such that each fuel type is analyzed as the "Hot Assembly"[3]. A representative mixed core input will be prepared for the RELAP4-EM "average core" blowdown calculation. As a practical matter, it has been TXU Electric's experience that fuel types are (by design) hydraulically similar and that differences in PCT resulting from blowdown calculations of RELAP4-EM "average cores" whose inputs are based on either type or combination of types are second order and not significant in comparison with differences that can occur from using

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3 The fuel type of the hot rod is the same as that of the hot assembly (same as hot channel).

different fuel types for the hot assembly (and the hot rod), where the thermal-mechanical considerations dominate. As a result of this experience, TXU Electric intends to perform the blowdown portion of the calculation using thermal-hydraulic inputs for the RELAP4-EM "Average Core" that are based on the fuel type which has the most assemblies for the cycle under consideration. This "Average Core" input is then run with "Hot Channel" (and "Hot Rod"[2]) inputs for each of the fuel types in the core, including the type which was used to prepare the "Average Core" input. An exception can be made if all assemblies of a particular fuel type can be demonstrated to be located in regions of the core that are not at or adjacent to potential "Hot Channel" locations. In other words, if a particular fuel type is located only in lower power regions of the core, that fuel type cannot be a "Hot Channel" and need not be analyzed as such. Also, existing previous cycle calculations may be used to establish that a particular fuel type cannot be the most limiting "Hot Channel". Whenever this can be done, the analysis of that fuel type as a "Hot Channel" may not be necessary.

2.5.3 POWER MEASUREMENT UNCERTAINTY

Currently, 10CFR50 Appendix K states in part that: "... it shall be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for such uncertainties as instrumentation error)...". In September 1999, it was proposed to amend this portion of Appendix K by adding "...An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error...". Although this change is not in effect at the time this topical is being submitted. TXU Electric is able to effectively take advantage of the new wording because of an exemption that TXU Electric obtained in May 1999 (Reference 3.6). The exemption allows TXU Electric to use a multiplier that corresponds to the actual uncertainty in the power level measurement, demonstrated to the NRC's satisfaction, rather than the fixed 1.02 multiplier. Therefore, when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flow meter (LEFM√), this uncertainty is such that a 1.01 multiplier is currently applied. That value has been shown to bound the power level measurement uncertainty. If in the future, the power level measurement uncertainty is such that a lower multiplier is demonstrated as per the revised Appendix K wording above, TXU Electric may use the lower number.

It should be pointed out that both CPSES-1 and CPSES-2 have usually been and continue to be analyzed at approximately 3479 Mwt. When the power level uncertainty is 1.01 the plant (currently CPSES-2 only) has a rated thermal power of 3445 Mwt, when the power level uncertainty is 1.02 the plant has a rated thermal power of 3411 Mwt (currently CPSES-1). This is implemented in the technical specifications (Reference 3.4) in part via the statement: "When an initial power level of 102% of rated power is specified, 101% of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flow meter (LEFM√). When feedwater flow measurements from the LEFM√ are not available, the originally approved initial power level of 102% of rated thermal power shall be used."

TABLE 2.1

CPSES-1 Cycle 8 NSSS Nodalization Summary

Component Description	Volume No.
Downcomer	35, 36
Lower Plenum	37, 80, 81
Average Core	38 to 53
Hot Assembly	54 to 69
Core Bypass	78
Upper Head	1
Upper Core	82
Upper Plenum	2
Guide Tubes	83
Containment	79

	Intact Loop	Broken Loop
RCPs	16	32
Hot Leg	3	19
Intermediate Leg	14, 15	30, 31
Cold Leg	17, 18	33, 34
S/G - Primary	4 to 13	20 to 29
S/G - Secondary	77	78
Accumulator	73	75
SI Discharge Line	74	76
Pressurizer	71	
Surge Line	72	
Total = 83		

Heat Conductor Description	No. Of Conductors
Average Core	16
Hot Assembly	16
Hot Rod	34
S/G Tubes per loop	8
Containment	5
Vessel, Piping, etc.	27
Total = 114	

Fill Junction Description	Junction No. Intact Loop	Junction No. Broken Loop
Centrifugal Charging Pumps	110	111
Safety Injection Pumps	112	113
Low Pressure Injection Pumps	114	115
Main Feedwater	106	107
Auxiliary Feedwater	108	109
Steam Line Valve	104	105
Total = 12		

TABLE 2.2
SUMMARY OF CPSES-1 CYCLE 8 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	30.9750
02	UNDER PLENUM	672.7352	7.9750	1.0+06	1.5991	23.0000
82	UPPER CORE	74.6550	1.2769	1.0+06	0.0704	21.7231
83	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	200.2450	6.7369	29.7238	0.0553	35.5916
06	SG TUBES	200.2450	6.7369	29.7238	0.0553	42.3285
07	SG TUBES	200.2450	7.2926	29.7238	0.0553	49.0654
08	SG TUBES	200.2450	7.2926	29.7238	0.0553	56.3580
09	SG TUBES	200.2450	7.2926	29.7238	0.0553	56.3580
10	SG TUBES	200.2450	7.2926	29.7238	0.0553	49.0654
11	SG TUBES	200.2450	6.7369	29.7238	0.0553	42.3285
12	SG TUBES	200.2450	6.7369	29.7238	0.0553	35.5916
13	SG OUTLET	538.6653	7.9114	68.0871	5.3756	27.6802
14	INTERM. LEG	231.7245	5.7917	15.7242	2.5833	15.3125
15	INTERM. LEG	166.4985	4.0415	15.7242	2.5833	15.3125
16	PUMP	5.8000	7.3615	32.0316	3.6871	21.1042
17	COLD LEG	5.0970	2.2917	12.3741	2.2917	25.7709
18	COLD LEG	5.0970	2.2917	12.3741	3.0124	25.7709
19	HOT LEG	99.3765	3.6457	4.5869	2.5282	25.7083
20	SG INLET	179.5551	7.9114	22.6957	5.3756	27.6802
21	SG TUBES	66.7483	6.7369	9.9079	0.0553	35.5916
22	SG TUBES	66.7483	6.7369	9.9079	0.0553	42.3285
23	SG TUBES	66.7483	7.2926	9.9079	0.0553	49.0654
24	SG TUBES	66.7483	7.2926	9.9079	0.0553	56.3580
25	SG TUBES	66.7483	7.2926	9.9079	0.0553	56.3580
26	SG TUBES	66.7483	7.2926	9.9079	0.0553	49.0654
27	SG TUBES	66.7483	6.7369	9.9079	0.0553	42.3285
28	SG TUBES	66.7483	6.7369	9.9079	0.0553	35.5916
29	SG OUTLET	179.5551	7.9114	22.6957	5.3756	27.6802
30	INTERM. LEG	77.2415	14.0415	5.2414	2.5833	15.3125
31	INTERM. LEG	55.4995	5.7917	5.2414	2.5833	15.3125

TABLE 2.2 (Continued...)
SUMMARY OF CPSES-1 CYCLE 8 RELAP4-EM SYSTEM MODEL VOLUMES

VOLUME NUMBER	REGION DESCRIPTION	VOLUME (FT3)	VOLUME LENGTH (FT)	FLOW AREA (FT2)	HYDRAULIC DIAMETER (FT)	ELEV. (FT)
32	PUMP	78.6000	7.3615	10.6772	3.6871	21.1042
33	COLD LEG	51.6990	2.2917	4.1247	2.2917	25.7709
34	COLD LEG	51.6990	2.2917	4.1247	3.0124	25.7709
35	UPPER DOWNCOMER	388.8486	14.0000	1.0+06	1.4145	19.9167
36	LOWER DOWNCOMER	472.8934	14.3333	35.6714	1.6363	5.5834
80	LOWER HEAD	120.2742	2.5126	47.8684	3.3216	-0.4292
81	LOWER PLENUM	460.6664	3.5000	1.0+06	5.1105	2.0834
37	L CORE SUPT PLT	335.9651	4.1397	1.0+06	0.0691	5.5834
38	CORE 1 AVG	40.3298	0.750	53.7730	0.0397	9.7231
39	CORE 2 AVG	40.3298	0.750	53.7730	0.0397	10.4731
40	CORE 3 AVG	40.3298	0.750	53.7730	0.0397	11.2231
41	CORE 4 AVG	40.3298	0.750	53.7730	0.0397	11.9731
42	CORE 5 AVG	40.3298	0.750	53.7730	0.0397	12.7231
43	CORE 6 AVG	40.3298	0.750	53.7730	0.0397	13.4731
44	CORE 7 AVG	40.3298	0.750	53.7730	0.0397	14.2231
45	CORE 8 AVG	40.3298	0.750	53.7730	0.0397	14.9731
46	CORE 9 AVG	40.3298	0.750	53.7730	0.0397	15.7231
47	CORE 10 AVG	40.3298	0.750	53.7730	0.0397	16.4731
48	CORE 11 AVG	40.3298	0.750	53.7730	0.0397	17.2231
49	CORE 12 AVG	40.3298	0.750	53.7730	0.0397	17.9731
50	CORE 13 AVG	40.3298	0.750	53.7730	0.0397	18.7231
51	CORE 14 AVG	40.3298	0.750	53.7730	0.0397	19.4731
52	CORE 15 AVG	40.3298	0.750	53.7730	0.0397	20.2231
53	CORE 16 AVG	40.3298	0.750	53.7730	0.0397	20.9731
54	CORE 1 HOT	0.2098	0.750	0.2797	0.0399	9.7231
55	CORE 2 HOT	0.2098	0.750	0.2797	0.0399	10.4731
56	CORE 3 HOT	0.2098	0.750	0.2797	0.0399	11.2231
57	CORE 4 HOT	0.2098	0.750	0.2797	0.0399	11.9731
58	CORE 5 HOT	0.2098	0.750	0.2797	0.0399	12.7231
59	CORE 6 HOT	0.2098	0.750	0.2797	0.0399	13.4731
60	CORE 7 HOT	0.2098	0.750	0.2797	0.0399	14.2231
61	CORE 8 HOT	0.2098	0.750	0.2797	0.0399	14.9731
62	CORE 9 HOT	0.2098	0.750	0.2797	0.0399	15.7231
63	CORE 10 HOT	0.2098	0.750	0.2797	0.0399	16.4731
64	CORE 11 HOT	0.2098	0.750	0.2797	0.0399	17.2231
65	CORE 12 HOT	0.2098	0.750	0.2797	0.0399	17.9731
66	CORE 13 HOT	0.2098	0.750	0.2797	0.0399	18.7231
67	CORE 14 HOT	0.2098	0.750	0.2797	0.0399	19.4731
68	CORE 15 HOT	0.2098	0.750	0.2797	0.0399	20.2231
69	CORE 16 HOT	0.2098	0.750	0.2797	0.0399	20.9731
70	BYPASS	298.5298	13.3750	22.3200	0.7918	9.3750
71	PRESSURIZER	1836.2393	49.9218	36.7823	6.8434	55.3308
72	PZR SURGE LINE	46.6806	27.8893	0.6827	0.9323	27.4415
73	ACCUMULATOR IL	4050.0000	10.8152	226.9008	9.8132	33.5775
74	DISCH LINE IL	95.4600	7.8067	1.2528	0.7292	25.7709
75	ACCUMULATOR BL	1350.0000	10.8152	75.6336	9.8132	42.9908
76	DISCH LINE BL	40.0400	17.2200	0.4176	0.7292	25.7709
77	STEAM GENERATOR	17862.0000	41.8300	169.3512	0.1234	35.5916
78	STEAM GENERATOR	5954.0000	41.8300	56.4504	0.1234	35.5916
79	CONTAINMENT	3.063+06	299.00	10244.1500	114.21	-31.0000

TABLE 2.2 (Continued...)
SUMMARY OF CPSES-1 CYCLE 8 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
98	DWNCMR/UHEAD	33.9167	0.1898	0.6981	1.4946	1.4722	0.1667
01	UHEAD/GUIDE	36.2900	0.4550	0.5199	7.256320	7.402039	0.4617
101	UPCORE/GUIDE	23.0000	0.4748	11.9831	0.2636	0.1984	3.9061
102	UPCORE/UPLNM	23.0000	0.0842	28.8708	1.2060	0.9942	6.0629
103	GUIDE/UPLNM	24.2391	0.5103	11.5647	1.402923	1.402923	3.8373
02	UPLENUM/HL	26.9167	0.7834	13.7607	0.2674	0.4978	2.4167
03	HL/SG	28.5238	0.8239	13.7607	0.3007	0.2893	2.4167
04	SG/TUBES	35.5916	0.1714	29.7238	1.8080	2.6070	3.5518
05	TUBES/TUBES	42.3285	0.2360	29.7238	1.0-07	1.0-07	3.5518
06	TUBES/TUBES	49.0654	0.2360	29.7238	1.0-07	1.0-07	3.5518
07	TUBES/TUBES	56.3580	0.2360	29.7238	1.0-07	1.0-07	3.5518
08	TUBES/TUBES	61.4666	0.2360	29.7238	3.7818	3.7818	3.5518
09	TUBES/TUBES	56.3580	0.2360	29.7238	1.0-07	1.0-07	3.5518
10	TUBES/TUBES	49.0654	0.2360	29.7238	1.0-07	1.0-07	3.5518
11	TUBES/TUBES	42.3285	0.2360	29.7238	1.0-07	1.0-07	3.5518
12	TUBES/SG	35.5916	0.1714	29.7238	2.6070	1.8080	3.5518
13	SG/IL	28.5238	0.5267	15.7242	0.4485	0.5231	2.5833
14	IL/IL	16.6042	0.8053	15.7242	1.0-07	1.0-07	2.5833
15	IL/RCP	21.1042	0.4905	15.7242	0.1591	0.1591	2.5833
16	RCP/CL	27.1250	0.6602	10.8375	1.0-07	1.0-07	2.1447
17	CL/CL	26.9167	0.9239	12.3741	1.0-07	1.0-07	2.2917
18	CL/DWNCMR	26.9167	0.4859	12.3741	1.0660	0.4398	2.2917
19	UPLENUM/HL	26.9167	2.3502	4.5869	0.2674	0.4978	2.4167
20	HL/SG	28.5238	2.4716	4.5869	0.3007	0.2893	2.4167
21	SG/TUBES	35.5916	0.5143	9.9079	1.8080	2.6070	3.5518
22	TUBES/TUBES	49.0653	0.7080	9.9079	1.0-07	1.0-07	3.5518
23	TUBES/TUBES	49.0653	0.7080	9.9079	1.0-07	1.0-07	3.5518
24	TUBES/TUBES	49.0653	0.7080	9.9079	1.0-07	1.0-07	3.5518
25	TUBES/TUBES	49.0653	0.7080	9.9079	3.7818	3.7818	3.5518
26	TUBES/TUBES	49.0653	0.7080	9.9079	1.0-07	1.0-07	3.5518
27	TUBES/TUBES	61.4666	0.7080	9.9079	1.0-07	1.0-07	3.5518
28	TUBES/TUBES	49.0653	0.7080	9.9079	1.0-07	1.0-07	3.5518
29	TUBES/SG	35.5916	0.5143	9.9079	2.6070	1.8080	3.5518
30	SG/IL	28.5238	1.5801	5.2414	0.4485	0.5231	2.5833
31	IL/IL	16.6042	2.4159	5.2414	1.0-07	1.0-07	2.5833
32	IL/RCP	21.1042	1.4714	5.2414	0.1591	0.1591	2.5833
33	RCP/CL	27.1250	1.9806	3.6125	1.0-07	1.0-07	2.1447
34	BREAK VALVE	26.9167	2.7716	4.1247	1.0-07	1.0-07	2.2917
35	CL/DWNCMR	26.9167	1.4576	4.1247	1.0660	0.4398	2.2917
36	U/L DWNCMR	19.9167	0.4226	35.6714	1.0-07	1.0-07	6.7393
37	DWNCMR/LPLN	5.5834	0.2866	26.6891	0.3697	0.2422	5.8294
99	LHEAD/LPLNM	2.0834	0.0402	82.0641	0.0000	0.0000	10.2219
100	LPLNM/LCSP	5.5834	0.1327	49.9264	0.6628	0.6960	7.9730
38	LCSP/1AVG	9.7231	0.0252	53.7730	4.7230	4.7230	8.2744
39	1/2AVG	10.4731	0.0139	53.7730	0.5520	0.5520	8.2744
40	2/3AVG	11.2231	0.0139	53.7730	0.5520	0.5520	8.2744
41	3/4AVG	11.9731	0.0139	53.7730	0.5520	0.5520	8.2744
42	4/5AVG	12.7231	0.0139	53.7730	0.5520	0.5520	8.2744
43	5/6AVG	13.4731	0.0139	53.7730	0.5520	0.5520	8.2744
44	6/7AVG	14.2231	0.0139	53.7730	0.5520	0.5520	8.2744
45	7/8AVG	14.9731	0.0139	53.7730	0.5520	0.5520	8.2744
46	8/9AVG	15.7231	0.0139	53.7730	0.5520	0.5520	8.2744
47	9/10AVG	16.4731	0.0139	53.7730	0.5520	0.5520	8.2744
48	10/11AVG	17.2231	0.0139	53.7730	0.5520	0.5520	8.2744
49	11/12AVG	17.9731	0.0139	53.7730	0.5520	0.5520	8.2744
50	12/13AVG	18.7231	0.0139	53.7730	0.5520	0.5520	8.2744
51	13/14AVG	19.4731	0.0139	53.7730	0.5520	0.5520	8.2744
52	14/15AVG	20.2231	0.0139	53.7730	0.5520	0.5520	8.2744
53	15/16AVG	20.9731	0.0139	53.7730	0.5520	0.5520	8.2744
54	16AVG/UPCR	21.7231	0.0195	53.7730	0.5520	0.5520	8.2744

TABLE 2.2 (Continued...)
SUMMARY OF CPSES-1 CYCLE 8 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
55	LCSP/1HOT	9.7231	1.3589	0.2646	4.7230	4.7230	0.5804
56	1/2HOT	10.4731	2.6814	0.2797	0.5520	0.5520	0.5968
57	2/3HOT	11.2231	2.6814	0.2797	0.5520	0.5520	0.5968
58	3/4HOT	11.9731	2.6814	0.2797	0.5520	0.5520	0.5968
59	4/5HOT	12.7231	2.6814	0.2797	0.5520	0.5520	0.5968
60	5/6HOT	13.4731	2.6814	0.2797	0.5520	0.5520	0.5968
61	6/7HOT	14.2231	2.6814	0.2797	0.5520	0.5520	0.5968
62	7/8HOT	14.9731	2.6814	0.2797	0.5520	0.5520	0.5968
63	8/9HOT	15.7231	2.6814	0.2797	0.5520	0.5520	0.5968
64	9/10HOT	16.4731	2.6814	0.2797	0.5520	0.5520	0.5968
65	10/11HOT	17.2231	2.6814	0.2797	0.5520	0.5520	0.5968
66	11/12HOT	17.9731	2.6814	0.2797	0.5520	0.5520	0.5968
67	12/13HOT	18.7231	2.6814	0.2797	0.5520	0.5520	0.5968
68	13/14HOT	19.4731	2.6814	0.2797	0.5520	0.5520	0.5968
69	14/15HOT	20.2231	2.6814	0.2797	0.5520	0.5520	0.5968
70	15/16HOT	20.9731	2.6814	0.2797	0.5520	0.5520	0.5968
71	16HOT/UPCR	21.7231	1.3532	0.2646	2.2800	2.2800	0.5968
72	LCSP/BYPSS	9.3750	0.2870	5.3294	41.3487	41.3487	2.6049
73	BYPSS/UPCR	22.7500	0.2813	3.7661	26.3112	26.3112	2.1898
74	CROSSFLW 1	10.0981	1.3333	0.5780	8.0852	8.0852	0.8579
75	CROSSFLW 2	10.8481	1.3333	0.5780	8.0852	8.0852	0.8579
76	CROSSFLW 3	11.5981	1.3333	0.5780	8.0852	8.0852	0.8579
77	CROSSFLW 4	12.3481	1.3333	0.5780	8.0852	8.0852	0.8579
78	CROSSFLW 5	13.0981	1.3333	0.5780	8.0852	8.0852	0.8579
79	CROSSFLW 6	13.8481	1.3333	0.5780	8.0852	8.0852	0.8579
80	CROSSFLW 7	14.5981	1.3333	0.5780	8.0852	8.0852	0.8579
81	CROSSFLW 8	15.3481	1.3333	0.5780	8.0852	8.0852	0.8579
82	CROSSFLW 9	16.0981	1.3333	0.5780	8.0852	8.0852	0.8579
83	CROSSFLW 10	16.8481	1.3333	0.5780	8.0852	8.0852	0.8579
84	CROSSFLW 11	17.5981	1.3333	0.5780	8.0852	8.0852	0.8579
85	CROSSFLW 12	18.3481	1.3333	0.5780	8.0852	8.0852	0.8579
86	CROSSFLW 13	19.0981	1.3333	0.5780	8.0852	8.0852	0.8579
87	CROSSFLW 14	19.8481	1.3333	0.5780	8.0852	8.0852	0.8579
88	CROSSFLW 15	20.5981	1.3333	0.5780	8.0852	8.0852	0.8579
89	CROSSFLW 16	21.3481	1.3333	0.5780	8.0852	8.0852	0.8579
90	PRZR/SURGE	55.3308	50.7563	0.6827	0.8675	1.3377	0.9323
91	SURGE/HL	27.7711	50.8541	0.6827	0.7017	3.2479	0.9323
92	AT/ATDL	33.5775	30.4552	1.2528	3.9754	3.9754	0.7292
93	ATDL/CL	25.7709	30.9223	1.2528	2.4044	2.4044	0.7292
94	AT/ATDL	42.9908	114.9168	0.4176	4.0102	4.0102	0.7292
95	ATDL/CL	25.7709	116.3182	0.4176	2.4044	2.4044	0.7292
96	CL/CNTNMNT	26.9167	1.5340	4.1247	1.00	0.50	2.2917
97	CL/CNTNMNT	26.9167	1.2668	4.1247	0.50	1.00	2.2917
106	MFW FILL	40.5916	0.0000	3.0000	0.0000	0.0000	1.1284
107	MFW FILL	40.5916	0.0000	1.0000	0.0000	0.0000	1.1284
108	AUX FILL	73.5916	0.0000	3.0000	0.0000	0.0000	1.1284
109	AUX FILL	73.5916	0.0000	1.0000	0.0000	0.0000	1.1284
110	CCP/FILL	26.6873	0.0000	3.0000	0.0000	0.0000	1.1284
111	CCP/FILL	26.7913	0.0000	1.0000	0.0000	0.0000	1.1284
112	HHP/FILL	26.6873	0.0000	3.0000	0.0000	0.0000	1.1284
113	HHP/FILL	26.7913	0.0000	1.0000	0.0000	0.0000	1.1284
114	RHR/FILL	26.6873	0.0000	3.0000	0.0000	0.0000	1.1284
115	RHR/FILL	26.7913	0.0000	1.0000	0.0000	0.0000	1.1284
105	TSV FILL	95.7583	0.0000	3.0000	0.0000	0.0000	1.1284
104	TSV FILL	95.7583	0.0000	1.0000	0.0000	0.0000	1.1284

TABLE 2.3
CPSES-1 CYCLE 8 DENSITY REACTIVITY TABLE

NORMALIZED DENSITY	REACTIVITY ($)
0.0000	-80.89
0.0142	-72.57
0.1428	-35.87
0.1785	-30.82
0.2858	-15.60
0.4286	-6.63
0.5714	-2.49
0.7144	-0.35
0.8572	0.29
1.0000	0.00
1.0572	-0.33
1.1428	-1.03

TABLE 2.4
CPSES-1 CYCLE 8 DOPPLER REACTIVITY TABLE

TEMPERATURE (F)	REACTIVITY ($)
650.0	3.223
800.0	2.536
1000.0	1.689
1200.0	0.907
1400.0	0.175
1450.0	0.00
1600.0	-0.514
1800.0	-1.168
2000.0	-1.800
2200.0	-2.411
2400.0	-3.007
2600.0	-3.586
2800.0	-4.150
3000.0	-4.695

TABLE 2.5
ECCS FLOW VS. PRESSURE (PER LOOP)

RCS PRESSURE (psia)	CCP (2) (1blm/sec)	HPSI (2) (1bm/sec)	RHR (1) (1bm/sec)	TOTAL (1bm/sec)
0.0	14.89	22.88	131.13	168.90
14.7	14.89	22.88	131.13	168.90
34.7	14.82	22.70	123.26	160.78
54.7	14.76	22.53	114.80	152.09
114.7	14.57	22.01	34.60	71.18
154.7	14.43	21.64	0.00	36.07
214.7	14.23	21.08		35.31
414.7	13.51	19.11		32.62
614.7	12.76	16.87		29.63
1014.7	11.13	11.29		22.42
1614.7	8.28	0.00		8.28
2314.7	0.00			0.00

TABLE 2.6
TIME DELAY FOR EACH SYSTEM

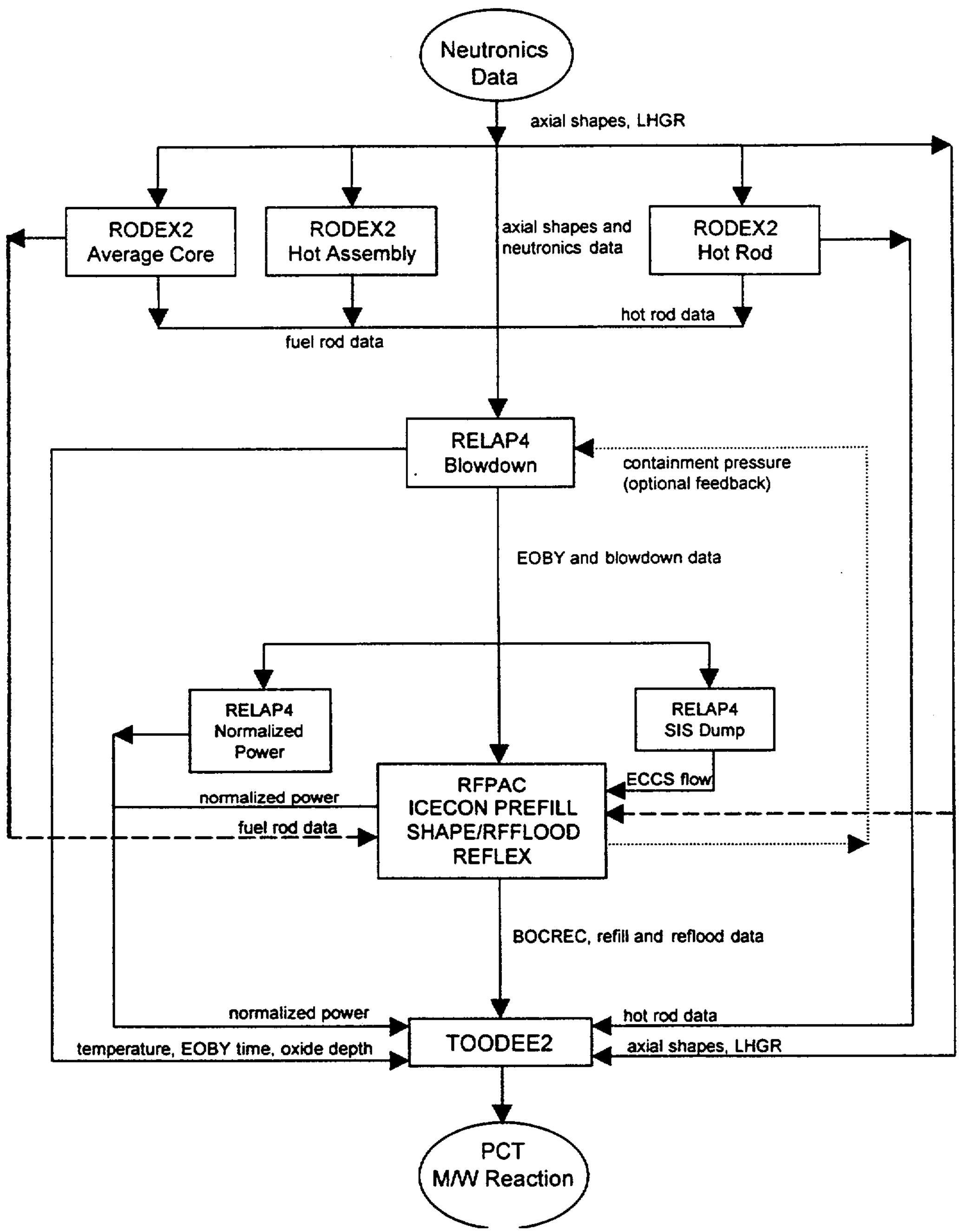
ACTION	TIME DELAY AFTER SI SETPOINT REACHED (sec)
SI actuation signal	2
Charging pumps up to speed	17 (Fill Table 1 initiated)
HPSI pumps up to speed	22 (Fill Table 2 initiated)
RHR pumps up to speed	27 (Fill Table 3 initiated)
Containment Spray	34

TABLE 2.7
CPSES-1 CYCLE 8 FUEL ASSEMBLY/ROD DATA

PARAMETER	VALUE
Outer Diameter of Fuel Rod	0.36 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	0.025 in
Diametral Gap	0.0065 in
Outer Dia. of Guide Thimble	0.48 in

Figure 2.1

Schematic Representation of the SEM/PWR-98 Methodology



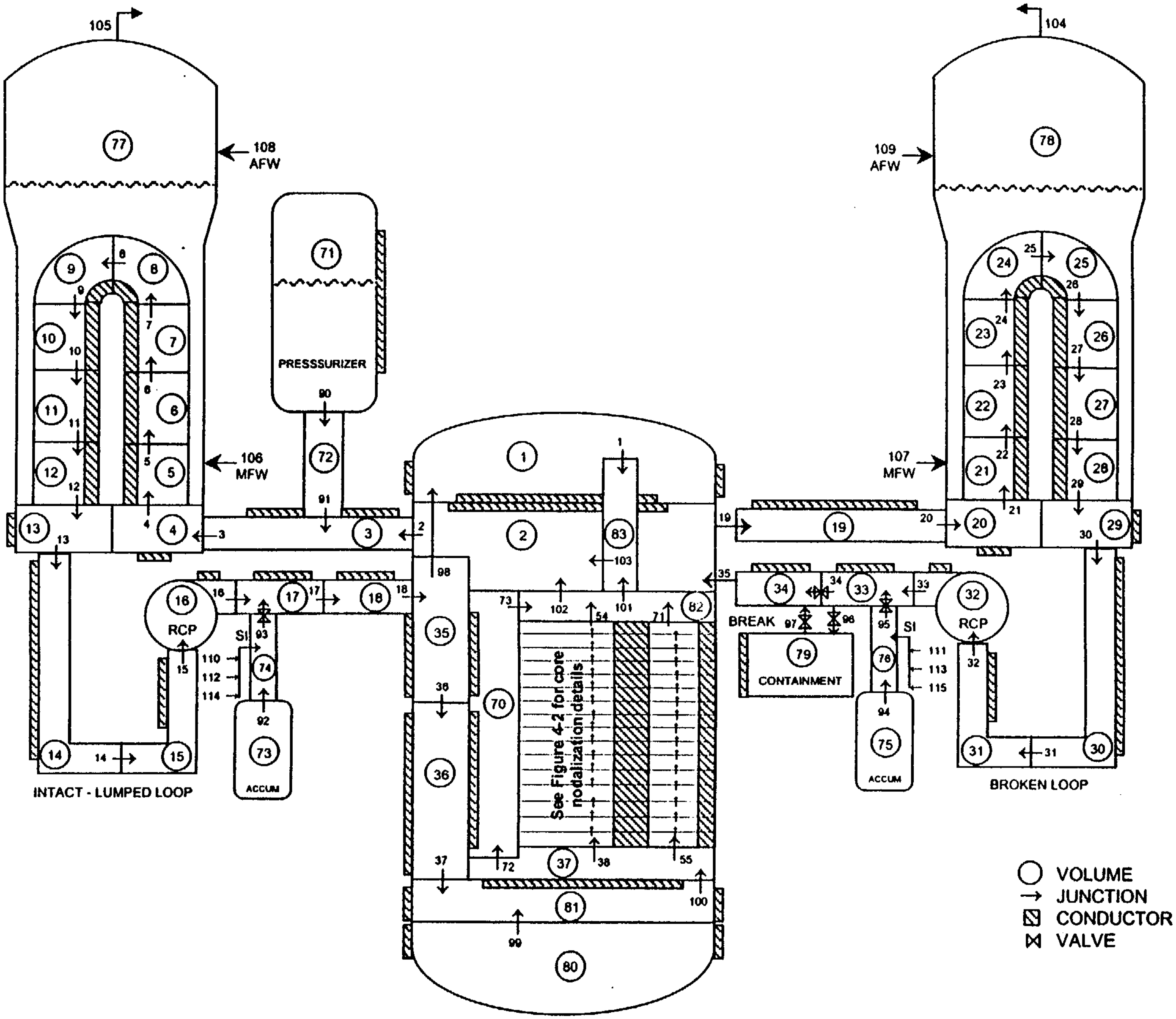


Figure 2.2 CPSES RELAP4 System Blowdown Model

Average Core & Hot Assembly Volumes	AC & HA HS	Grid Elevation (in)	Hot Rod HS
		144.0	
AC Vol. 53 HA Vol. 69 (9 in)	*AC HS 16* *HA HS 32*	135.0	*HS 66* *(9 in)*
AC Vol. 52 HA Vol. 68 (9 in)	*AC HS 15* *HA HS 31*	132 129 126.0	*HS 65 (3 in)* *HS 64 (3 in)* *HS 63 (3 in)*
AC Vol. 51 HA Vol. 67 (9 in)	*AC HS 14* *HA HS 30*	123 120 117.0	*HS 62 (3 in)* *HS 61 (3 in)* *HS 60 (3 in)*
AC Vol. 50 HA Vol. 66 (9 in)	*AC HS 13* *HA HS 29*	114 111 108.0	*HS 59 (3 in)* *HS 58 (3 in)* *HS 57 (3 in)*
AC Vol. 49 HA Vol. 65 (9 in)	*AC HS 12* *HA HS 28*	105 102 99.0	*HS 56 (3 in)* *HS 55 (3 in)* *HS 54 (3 in)*
AC Vol. 48 HA Vol. 64 (9 in)	*AC HS 11* *HA HS 27*	96 93 90.0	*HS 53 (3 in)* *HS 52 (3 in)* *HS 51 (3 in)*
AC Vol. 47 HA Vol. 63 (9 in)	*AC HS 10* *HA HS 26*	87 84 81.0	*HS 50 (3 in)* *HS 49 (3 in)* *HS 48 (3 in)*
AC Vol. 46 HA Vol. 62 (9 in)	*AC HS 9* *HA HS 25*	78 75 72.0	*HS 47 (3 in)* *HS 46 (3 in)* *HS 45 (3 in)*
AC Vol. 45 HA Vol. 61 (9 in)	*AC HS 8* *HA HS 24*	69 66 63.0	*HS 44 (3 in)* *HS 43 (3 in)* *HS 42 (3 in)*
AC Vol. 44 HA Vol. 60 (9 in)	*AC HS 7* *HA HS 23*	60 57 54.0	*HS 41 (3 in)* *HS 40 (3 in)* *HS 39 (3 in)*
AC Vol. 43 HA Vol. 59 (9 in)	*AC HS 6* *HA HS 22*	45.0	*HS 38* *(9 in)*
AC Vol. 42 HA Vol. 58 (9 in)	*AC HS 5* *HA HS 21*	36.0	*HS 37* *(9 in)*
AC Vol. 41 HA Vol. 57 (9 in)	*AC HS 4* *HA HS 20*	27.0	*HS 36* *(9 in)*
AC Vol. 40 HA Vol. 56 (9 in)	*AC HS 3* *HA HS 19*	18.0	*HS 35* *(9 in)*
AC Vol. 39 HA Vol. 55 (9 in)	*AC HS 2* *HA HS 18*	9.0	*HS 34* *9 in)*
AC Vol. 38 HA Vol. 54 (9 in)	*AC HS 1* *HA HS 17*	0.0	*HS 33* *(9 in)*

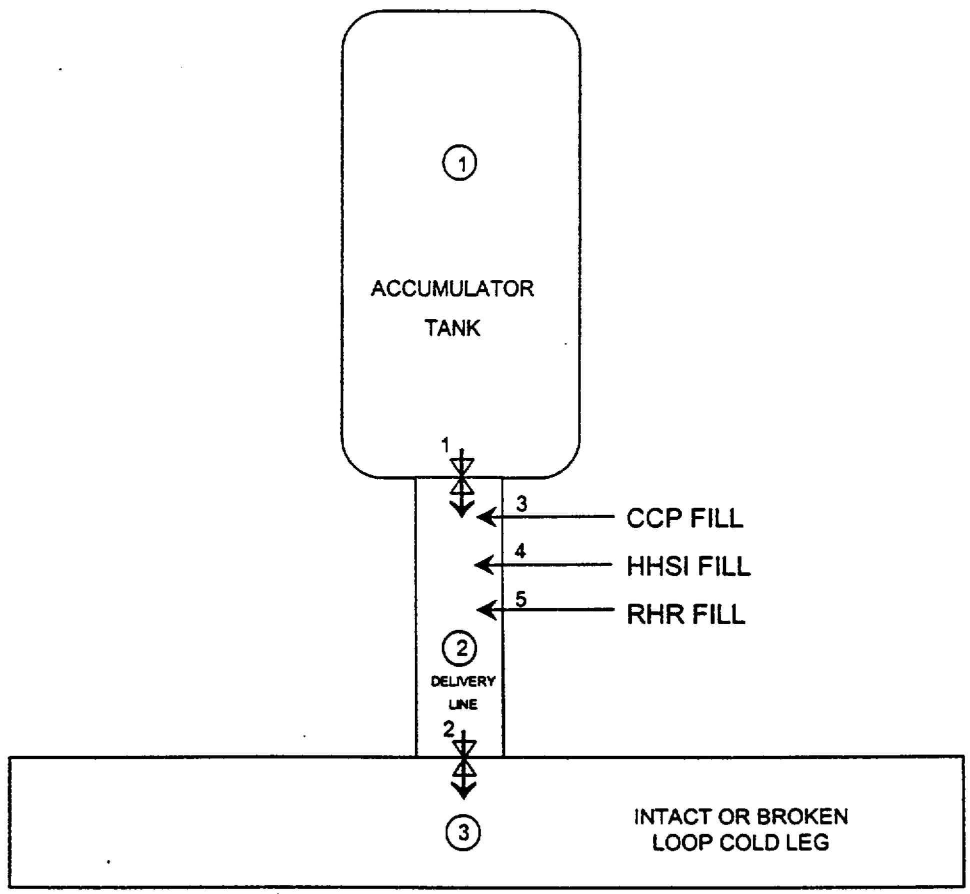
AC Axial Junctions: 38 through 54
HA Axial Junctions: 55 through 71
Core X-Flow Junctions: 74 through 89

AC Average Core
HA Hot Assembly
HS Heat Slab

Figure 2.3 RELAP4-EM Average Core, Hot Assembly and Hot Rod Nodalization

Figure 2.4

ACCUM-SIS Nodalization (Unchanged)



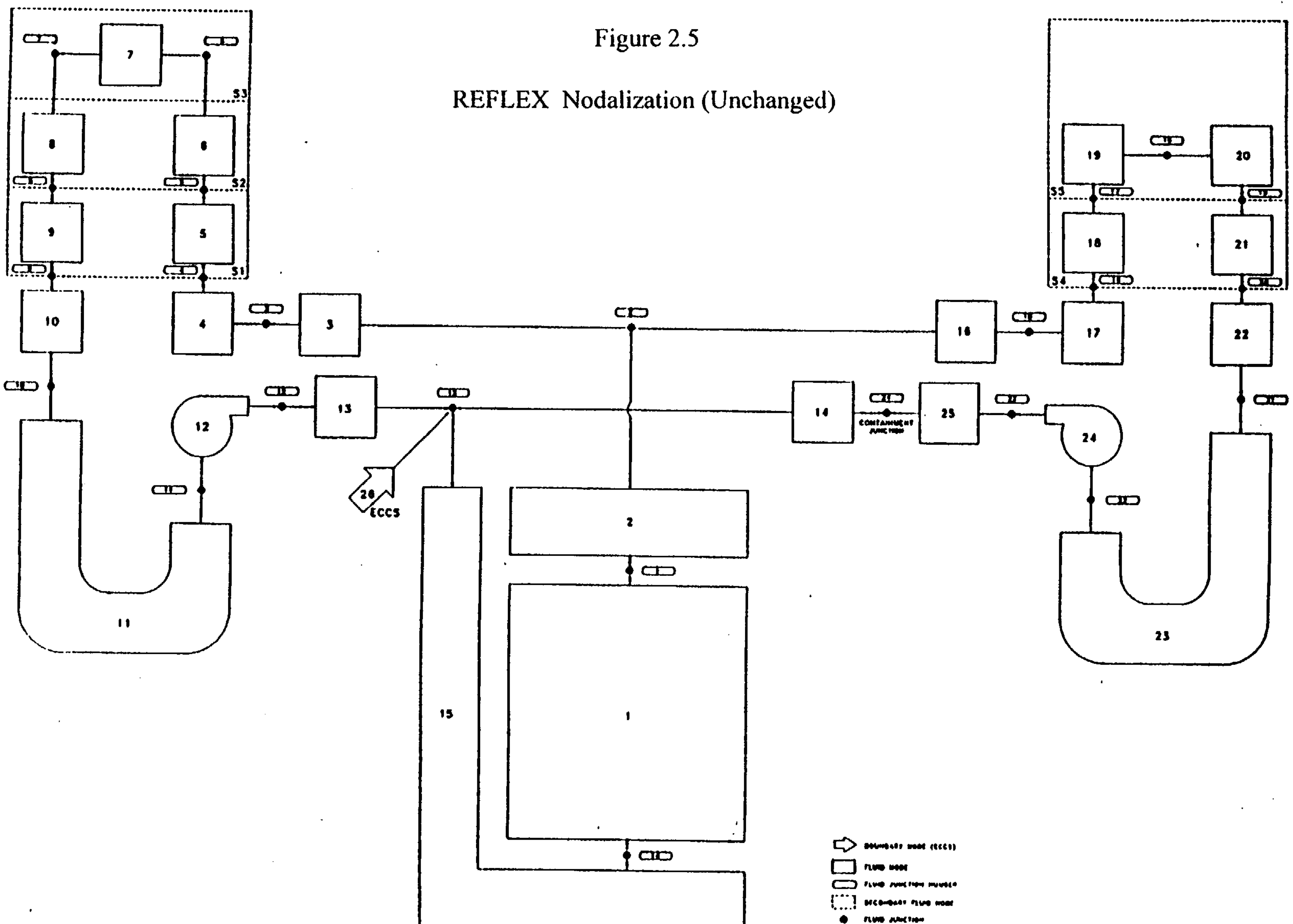


Figure 2.5

REFLEX Nodalization (Unchanged)

Figure 2.6

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.

Method-specific issues are suggested throughout 10 CFR 50.46 and Appendix K and are addressed in Reference 1.4. The present work constitutes TXU Electric's application of SPC's approved Evaluation Methodology (EM), using method-specific parameters as prescribed by the method developers. Hence, the effect of variations in method-specific parameters within the bounds of methodology recommendations has already been ascertained in Reference 1.4 and sensitivity studies for these variables need not be repeated here. There is one exception: the convergence criterion was varied to verify the robustness of results for the CPSES application.

The plant-specific issues which warrant investigation are given in the following passages from 10 CFR 50.46 and Appendix K thereto, along with the approach taken in addressing each one. 10 CFR 50.46 (a)(1)(i), requires that "a number of postulated loss-of-coolant accidents of different sizes, locations and other properties" be calculated in sufficient amount "to provide assurances that the most severe postulated loss-of-coolant accidents are calculated." In compliance with this requirement, a break spectrum study has been conducted.

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

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

This issue is resolved by selecting the axial power shape which has the peak power location highest in the core and then raising the peak node power to the technical specification limit (Reference 3.4) also known as K(z) curve. This selection criterion, is part of the SEM/PWR-98 methodology. The shape selected in this manner is most limiting because it puts the highest power region close to the top, where it is left uncovered for the longest period of time. The most limiting shape is selected from a vast population of possible shapes, which is developed on a cycle-specific basis, through standard reactor physics methodologies such as the power distribution control analysis, described in Reference 3.5. Shapes are specific to time in cycle, i.e., the power shape for the beginning of life exposure case is selected from a universe of beginning of cycle shapes; the shape for the middle of life case is selected from a middle of cycle universe of shapes and; the end of life case from an end of cycle set of shapes. Thus, within each interval in the cycle, beginning, middle and end, the highest peaking factor power shape of those shapes having the power peaking highest in the core is selected. The power in the peak node is then increased to the technical specification limit (Reference 3.4) for that elevation. The adjustment of the selected shape upward until the axial power shape curve touches the curve representing the Technical Specification LHGR limit as a function of core height is done differently by TXU Electric and SPC. SPC adjusts the entire curve upward to touch the Technical Specification LHGR limit. This approach necessarily entails distorting the shape in an arbitrary manner in order to preserve the total power (area under the power shape curve). The resulting shape would violate TXU Electric axial offset limits and would therefore not be possible for CPSES. Furthermore, arbitrarily re-shaping in its entirety a realistic power shape negates the reactor physics basis for that shape. The use of a shape that could be substantially different from what reactor physics would allow, would distort thermal hydraulics boundary conditions, in addition to affecting rod thermal performance. For these reasons, TXU Electric has chosen an approach that does not arbitrarily distort the thermal hydraulics of the transient and only raises the relative power in the node closest to the K(Z) curve. While this results in a local (negative) flux depression, flux depressions are not unphysical and exist throughout core, e.g. those caused by axial grids. TXU Electric rationalizes its approach simply as another depression (negative) at an axial location.

10 CFR 50, Appendix K, Part I, D, (1) states: "an analysis of possible failure modes of ECCS equipment and their effects on ECCS performance must be made. In carrying out the accident

evaluation, the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place." The competing single failures for the large break loss-of-coolant accident analyses have been determined by experience (Reference 1.4). These are either: (a) the loss of one ECCS injection train or, (b) the loss of 1 train of low pressure injection. A sensitivity study is performed to verify which of these is the most limiting.

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

3.1 BASE CASE ANALYSIS

This section presents licensing analysis results for a Double-Ended Guillotine break in the discharge line of the Reactor Coolant Pump. This break location has been generically shown to be the most limiting (e.g., Reference 1.4). The axial power shape used for this base case is that determined as described in Section 3.0 as most limiting for Unit 1 Cycle 8 (BOL) and is shown in Figure 3.1. The fuel rod exposure which maximizes stored energy is calculated by RODEX2. 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 3479.5 MWt. This power level is determined as explained in Section 2.5.2. ECCS injection into the broken loop is lost, and is postulated to spill directly to the containment. Loss of one train of low pressure pumped injection (residual heat removal pumps, RHR) is the postulated single failure as required by 10 CFR 50, Appendix K. (In a sensitivity study, an alternative single failure, the loss of a diesel-generator resulting in the loss of one full train of ECCS, is examined.) Thus, for this base case, two high head centrifugal charging pumps, two intermediate head safety injection pumps 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 (Reference 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 (see Section 2.4.1.6). Furthermore, containment safeguards are also assumed to function as designed while consistent with the single failure; i.e., two trains of containment sprays are available for the base case. (Only one train is considered in the single failure sensitivity case, the other taken out by a postulated failure of the diesel-generator.) The fan coolers are disabled on the SI signal as per design.

Ten 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 3.5 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 3 and 10 seconds the power goes through a local maximum because of an increase in reactivity, which in turn is caused by an increase in the liquid fraction in the center of the core (Figure 3.6). The increase in power results from a temporary coolant accumulation in that region, which is associated with a flow reversal (Figures 3.4 and 3.5). Beyond this time, core power follows the 1971 ANS Draft Standard decay heat values.

Figure 3.6 shows mid-core average quality. The figure indicates that core flashing takes place around 2.5 seconds. 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 approximately 15 seconds, Figure 3.11) the mid-core quality again drops quickly, but begins to increase again right after the drop and is back to 1.0 at approximately 24 seconds.

Figure 3.7 shows the downcomer liquid inventory. The downcomer remains nearly full until almost 5 seconds. As shown in Figure 3.4, the drainage coincides with the decrease and subsequent flow reversal which is caused by the break and occurs starting around 5 seconds as well. After that the downcomer is quickly depleted reaching a minimum inventory at the time the accumulators begin to inject, when it once again begins to fill quickly.

Figure 3.8 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.

Figure 3.9 and 3.10 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 35 psia, approximately 16 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.

ECCS flow rates are presented in Figures 3.11 through 3.13. The accumulators begin to inject at 15 seconds and are empty at 44 seconds. The centrifugal charging pumps begin to discharge around 19 seconds and the intermediate head safety injection pump at approximately 24. The low pressure injection system comes on at approximately 28 seconds and reaches full capacity at about 45 seconds.

Figure 3.14 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 3.15.

The core flooding rates are shown in Figure 3.16. The flooding rate does not drop below one inch per second until approximately 80 seconds. The PCT time is approximately 160 seconds.

The metal reaction depth at the hot spot is shown in Figure 3.17.

The PCT node clad temperature history is shown in Figure 3.18. The PCT is calculated to be 1963 °F at 162.7 seconds, at 11.125 ft. The ruptured node was at elevation 8.875 ft and it occurred at 44.28 seconds. The maximum nodal oxidation was 3.2% with maximum total pin oxidation 0.50%.

3.2 SENSITIVITY STUDIES

3.2.1 BREAK SPECTRUM

The most limiting break location has been generically determined (e.g., see Reference 1.4) 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.

The break size is the first sensitivity issue addressed. The rationale for addressing break size first is that system thermal-hydraulic behavior is largely affected by break size and less dependent on other issues, i.e., the break size is a first order effect, while the others are second order.

The break spectrum study is conducted using beginning of life exposure and a beginning of cycle axial power shape, developed as explained at the beginning of Chapter 3. This is the same power shape used for the base case (Figure 3.1). Thus, the first case in the break spectrum study has: (a) the above power shape, (b) the guillotine type break and, (c) beginning of life (BOL) fuel, i.e. it is the base case of Section 3.1.

Three DEG break sizes are examined by using the break discharge coefficient values of 1.0 (base case), 0.8 and 0.6, respectively.

Split type breaks are also analyzed. Three longitudinal split break sizes are examined: 2.0, 1.6 and 1.2 times the cold leg cross-section area, while maintaining the discharge coefficient at 1.0.

The accident assumptions for this and the other sensitivity 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 for BOL and in Table 3.4 for MOL and EOL.

The sequence of events for the break spectrum study is summarized in Table 3.5.

The results of the 0.8 DEG 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. The PCT node clad temperature history is shown in Figure 3.19. The PCT is calculated to be 1941.0 °F at 159.3 seconds, at 11.125 ft. The ruptured node was at elevation 8.875 ft and it occurred at 45.3 seconds. The maximum nodal oxidation was 2.9% with maximum total pin oxidation 0.48%.

The 0.6 DEG 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 Figure 3.20. The PCT node clad temperature history is shown in Figure 3.20. The PCT is calculated to be 1853.1 °F at 156.0 seconds, at 11.125 ft. The ruptured node was at elevation 8.875 ft and it occurred at 49.3 seconds. The maximum nodal oxidation was 2.1% with maximum total pin oxidation 0.39%.

The longitudinal split break calculation shows results that are respectively similar to the DEG. For example the 2.0 split PCT is 1959.6 °F (which is similar to the same break area/CD combination of the 1.0 DEG, with a PCT of 1963.2 °F). The 1.6 split PCT is 1930.6 °F (which is similar to the same break area/CD combination of the 0.8 DEG PCT, with a PCT 1941.0 °F). The 1.2 split PCT is 1867.6 °F (which is similar to the same break area/CD combination of the 0.6 DEG, with a PCT of 1853.1 °F).

Results of this sensitivity study are summarized in Table 3.8. The conclusion 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 and assuming a Double-Ended Guillotine break.

3.2.2 EXPOSURE

The exposure study is done to support operation to EOL 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. Furthermore, in order to verify that middle of life (MOL) operation is not a concern, that condition is examined as well. 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 EOL and even more so at MOL.

The fuel parameters for the BOL conditions are given in Table 3.3 and for MOL and EOL conditions in Table 3.4. The most limiting power shapes for each of these exposures are shown in Figure 3.1. The sequences of events for the exposure study are summarized in Table 3.6.

The PCT for the EOL case is calculated to be 1836.8 °F at 181.2 seconds, at 10.875 ft. The ruptured node was at elevation 9.625 ft and it occurred at 46.4 seconds. The maximum nodal oxidation was 2.0% with maximum total pin oxidation 0.35%. The clad temperatures are shown in Figure 3.21. The middle of life case is even less limiting with a PCT of only 1824.1 °F (Figure 3.22).

Results of this sensitivity study are summarized in Table 3.9. The conclusion from the burnup study is that all burnups are bounded by the beginning of life condition, since the two extremes (maximum stored energy and maximum pin pressure) as well as the intermediate combination have been examined. Therefore the BOL exposure will be used in future analyses.

3.2.3 SINGLE FAILURE

The competing single failures for the large break loss-of-coolant accident analyses have been determined by experience (Reference 1.4). These are either: (a) the loss of one ECCS injection train or, (b) the loss of 1 train of low pressure injection. A sensitivity study is performed to verify which of these scenarios is the most limiting. The base case analysis of Section 3.1 assumed the failure one train of low pressure pumped injection (1 residual heat removal pump, RHR) as the single failure required by 10 CFR 50, Appendix K. This sensitivity study examines an alternative single failure, namely a postulated failure of a diesel-generator. This assumed single failure will result in the loss of one full train of ECCS, assuming loss of offsite power. Thus, for this sensitivity case, 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 (Reference 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 and to be consistent with the assumed single failure; i.e., for this sensitivity case only one train of containment spray is available, the other is taken out by the postulated failure the diesel-generator. The fan coolers are also disabled in this case on the SI signal as per design. The rationale for selecting this case is to examine the trade-off between the deleterious effect on the peak clad temperature of: (a) a lower containment pressure as in the base case, where both trains of containment spray pumps work versus, (b) a lesser ECCS injection into the core as in this sensitivity case, but where containment back pressure can be higher due to the loss of one train of spray pumps.

The sequence of events for the single failure of 1 train of ECCS is summarized in Table 3.7. The PCT is calculated to be 1944.1 °F at 155.8 seconds, at 11.125 ft. The ruptured node was at elevation 8.875 ft and it occurred at 44.3 seconds. The maximum nodal oxidation was 3.0% with maximum total pin oxidation 0.49%.

Results of this sensitivity study are summarized in Table 3.10. The conclusion from the single failure study is that the single failure assumed in the base case is more limiting. Therefore the single failure of one low pressure injection pump (RHR pump) will be used in future analyses.

3.2.4 CONVERGENCE CRITERION

The base case analysis of Section 3.1 assumed a convergence criterion (see section 2.3.1.1 and Reference 1.4). This case simply varied the convergence criterion variable ESPW from 0.5 in the base case to 0.25 in this case in order to check the robustness of the methodology. The new PCT was less than 1 ^{0}F lower, so it is concluded that CPSES results are numerically robust and the recommended value for variable ESPW=0.5 is adequate. Results of this sensitivity study are summarized in Table 3.11.

3.2.5 SUMMARY OF MOST LIMITING SENSITIVITY STUDIES

The most limiting scenario is the base case scenario:

- 1.0 DEG at reactor coolant pump discharge.
- BOL fuel exposure.
- Assumed Single Active Failure of 1 RHR pump.

TABLE 3.1

SUMMARY OF CPSES-1 CYCLE 8 LARGE BREAK LOCA ANALYSIS ASSUMPTIONS FOR BASE CASE AND SENSITIVITY STUDIES

1. The initial power is 3479.5 MWt, which is the current rated thermal power plus an allowance for the power calorimetric measurement uncertainty. See Section 2.5.3 for the discussion on this subject.

2. 10% of the steam generator tubes are plugged.

3. Break in reactor coolant pump discharge occurs at 0.05 s.

4. No Credit taken for Reactor trip (no scram reactivity insertion).

5. Three accumulators inject into intact loops on demand.

6. Two high head centrifugal charging pumps, two intermediate head safety injection pumps and one low head high flow residual heat removal pump inject on demand after the appropriate delays. Assumed single failure: 1 train of low pressure injection (RHR). (In a sensitivity study an alternative single failure, namely the loss of one full train of ECCS, taken out by a postulated failure its diesel-generator, is examined.)

7. Containment pressure is minimized in accordance with branch Technical Position CSB 6-1 (Reference 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 and to be consistent with the single failure; i.e., two trains of containment sprays are available. The fan coolers are disabled on the SI signal per plant design. Passive heat structures are included. (Only one train is considered in the single failure sensitivity case, the other taken out by a postulated failure of the diesel-generator.)

8. No credit is given for Auxiliary Feedwater.

TABLE 3.2

SUMMARY OF INITIAL CONDITIONS FOR CPSES-1 CYCLE 8 LARGE BREAK LOCA BASE CASE AND SENSITIVITY STUDIES

DESCRIPTION	VALUE
o Core Power	3479.5 Mwt
o Power Calorimetric Uncertainty Multiplier	1.02 (see Section 3.0)
o Power Shape Analyzed	
o Peak Linear Power [@ 8.75 ft] (includes 1.02 factor)	12.952 KW/ft
o Fraction of heat deposited in fuel	0.974
o Total Peaking Factor, F^T_Q (flat segment of K(z))	2.42
o Total Peaking Factor, F^T_Q [(@8.75 ft)]	
o Accumulator Water Volume per Tank	6119 gals
o Accumulator Cover Gas Pressure	623 psig
o Accumulator Water Temperature	88 °F
o Safety Injection Pumped Flow	Table 2.6
o Refueling Water Storage Tank Temperature	40 °F
o Initial Loop Flow	9739 lbm/sec
o Vessel Inlet Temperature	559.6 °F
o Vessel Outlet Temperature	620 °F
o Reactor Coolant Pressure	2250 psia
o Pressurizer Water Volume	1123 ft^3
o Steam Pressure	928 psia
o Containment conditions	Table 3.1 item 7
o Steam Generator Tube Plugging Level	10%
o Single Failure	Loss of 1 Train of Low Pressure Injection
o Fuel Parameters	Unit 1 Cycle 8, Table 3.3

TABLE 3.3

SUMMARY OF FUEL PARAMETERS FOR BASE CASE (BOL) LARGE BREAK LOCA ANALYSIS

PARAMETERS	VALUES		
Fuel Rod Geometry Data	Table 2.8		
Time to Maximum Stored Energy Exposure			
Fuel Rod Composition:			
Average fuel temperature at peak stored energy (°F)	1484	1884	2125

TABLE 3.4

SUMMARY OF FUEL PARAMETERS FOR LARGE BREAK LOCA EXPOSURE STUDY (MOL AND EOL)

PARAMETERS	VALUES		
Fuel Rod Geometry Data		Table 2.8	
MIDDLE OF LIFE (MOL)			
Average fuel temperature at MOL stored energy (°F)	1108	1410	1665
END OF LIFE (EOL)			
Average fuel temperature at EOL stored energy (°F)	1128	1434	1720

TABLE 3.5

SEQUENCE OF EVENTS FOR BREAK SPECTRUM[4] STUDY

EVENT	TIME (SECONDS					
	1.0 DEG[5]	0.8 DEG	0.6 DEG	2.0 Split	1.6 Split	1.2 Split
1. Break opens	0.03	0.03	0.03	0.03	0.03	0.03
2. Main fedwater isolated	0.03	0.03	0.03	0.03	0.03	0.03
3. MSIV closed	0.03	0.03	0.03	0.03	0.03	0.03
5. High containment pressure Hi-1 signal	1.13	1.21	1.38	1.18	1.20	1.24
6. Accumulator injection, intact loop	14.93	15.08	16.74	15.15	15.21	15.57
7. Centrifugal charging pumps inject	18.13	18.21	18.38	18.18	18.20	18.25
8. End of Bypass	22.72	23.19	25.34	23.04	23.08	23.57
9. Safety injection pumps inject	23.13	23.21	23.38	23.18	23.20	23.25
10. Low pressure pump injects (RHR)	28.13	28.21	28.38	28.18	28.20	28.25
11. Bottom of Core Recovery (BOCREC)	36.71	37.25	39.50	37.07	37.10	37.62
12. Accumulator empty	43.75	43.95	45.70	44.10	44.15	44.60
13. Rod Burst	44.28	45.34	49.30	44.89	45.13	45.52
14. Peak clad temperature reached	162.73	159.29	155.95	161.14	154.58	140.67
15. Calculation ends	250.00	250.00	250.00	250.00	250.00	250.00

4 All cases: same limiting power shape peaking at 8.75 ft (Figure 3.1), single failure of 1 RHR pump, beginning of life fuel exposure.

5 Base case.

TABLE 3.6

SEQUENCE OF EVENTS FOR EXPOSURE STUDY[6]

EVENT	TIME (SECONDS)		
	BOL[7]	MOL	EOL
1. Break opens	0.03	0.03	0.03
2. Main fedwater isolated	0.03	0.03	0.03
3. MSIV closed	0.03	0.03	0.03
5. High containment pressure Hi-1 signal	1.13	1.14	1.13
6. Accumulator injection, intact loop	14.93	14.94	14.89
7. Centrifugal charging pumps inject	18.13	18.14	18.13
8. End of Bypass	22.72	22.70	22.68
9. Safety injection pumps inject	23.13	23.14	23.13
10. Low pressure pump injects (RHR)	28.13	28.14	28.13
11. Bottom of Core Recovery (BOCREC)	36.71	36.66	36.65
12. Accumulator empty	43.75	43.75	43.70
13. Rod Burst	44.28	45.85	46.43
14. Peak clad temperature reached	162.73	166.30	181.18
15. Calculation ends	250.00	250.00	250.00

6 All cases: 100 % double-ended guillotine break (1.0 DEG), single failure of 1 RHR pump.

7 Base case.

TABLE 3.7

SEQUENCE OF EVENTS FOR SINGLE FAILURE STUDY[8]

EVENT	TIME (SECONDS)	
	1 TRAIN RHR[9]	1 TRAIN ECCS
1. Break opens	0.03	0.03
2. Main fedwater isolated	0.03	0.03
3. MSIV closed	0.03	0.03
5. High containment pressure Hi-1 signal	1.13	1.13
6. Accumulator injection, intact loop	14.93	14.93
7. Centrifugal charging pumps inject	18.13	18.13
8. End of Bypass	22.72	22.73
9. Safety injection pumps inject	23.13	23.13
10. Low pressure pump injects (RHR)	28.13	28.13
11. Bottom of Core Recovery (BOCREC)	36.71	36.74
12. Accumulator empty	43.75	43.70
13. Rod Burst	44.28	44.28
14. Peak clad temperature reached	162.73	155.83
15. Calculation ends	250.00	250.00

8 All cases: same limiting power shape peaking at 8.75 ft (Figure 3.1), 100 % double-ended guillotine break (1.0 DEG) and beginning of life (BOL) exposure.

9 Base case.

TABLE 3.8

RESULT SUMMARY FOR BREAK SPECTRUM STUDY[10]

BREAK DESCRIPTION	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
1.0 DEG	1963.2	3.2	0.50
0.8 DEG	1941.0	2.9	0.48
0.6 DEG	1853.1	2.1	0.39
2.0 Split	1959.6	3.2	0.49
1.6 Split	1930.6	2.8	0.46
1.2 Split	1867.6	2.3	0.42

TABLE 3.9

RESULT SUMMARY FOR EXPOSURE STUDY[11]

EXPOSURE	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
BOL	1963.2	3.2	0.50
MOL	1824.1	1.9	0.36
EOL	1836.8	2.0	0.35

10 All cases: same limiting power shape peaking at 8.75 ft, single failure of 1 RHR pump, beginning of life fuel exposure.

11 All cases: same limiting power shape peaking at 8.75 ft, single failure of 1 RHR pump, 100 % double-ended guillotine break (1.0 DEG).

TABLE 3.10

RESULT SUMMARY FOR SINGLE FAILURE STUDY[12]

SINGLE FAILURE	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
1 Train of LPI (RHR)	1963.2	3.2	0.50
1 Train of ECCS	1944.1	3.0	0.49

TABLE 3.11

RESULT SUMMARY FOR CONVERGENCE CRITERION STUDY[13]

SINGLE FAILURE	PCT (^{0}F)	% Oxidation (NODE)	% Oxidation (PIN)
EPSW = 0.5	1963.2	3.2	0.50
EPSW = 0.25	1962.4	3.2	0.50

12 **All cases: same limiting power shape peaking at 8.75 ft, 100 % double-ended guillotine break (1.0 DEG) and beginning of life (BOL) exposure.**

13 **All cases: same limiting power shape peaking at 8.75 ft, 100 % double-ended guillotine break (1.0 DEG), beginning of life (BOL) exposure and single failure of 1 train of low pressure injection (RHR).**

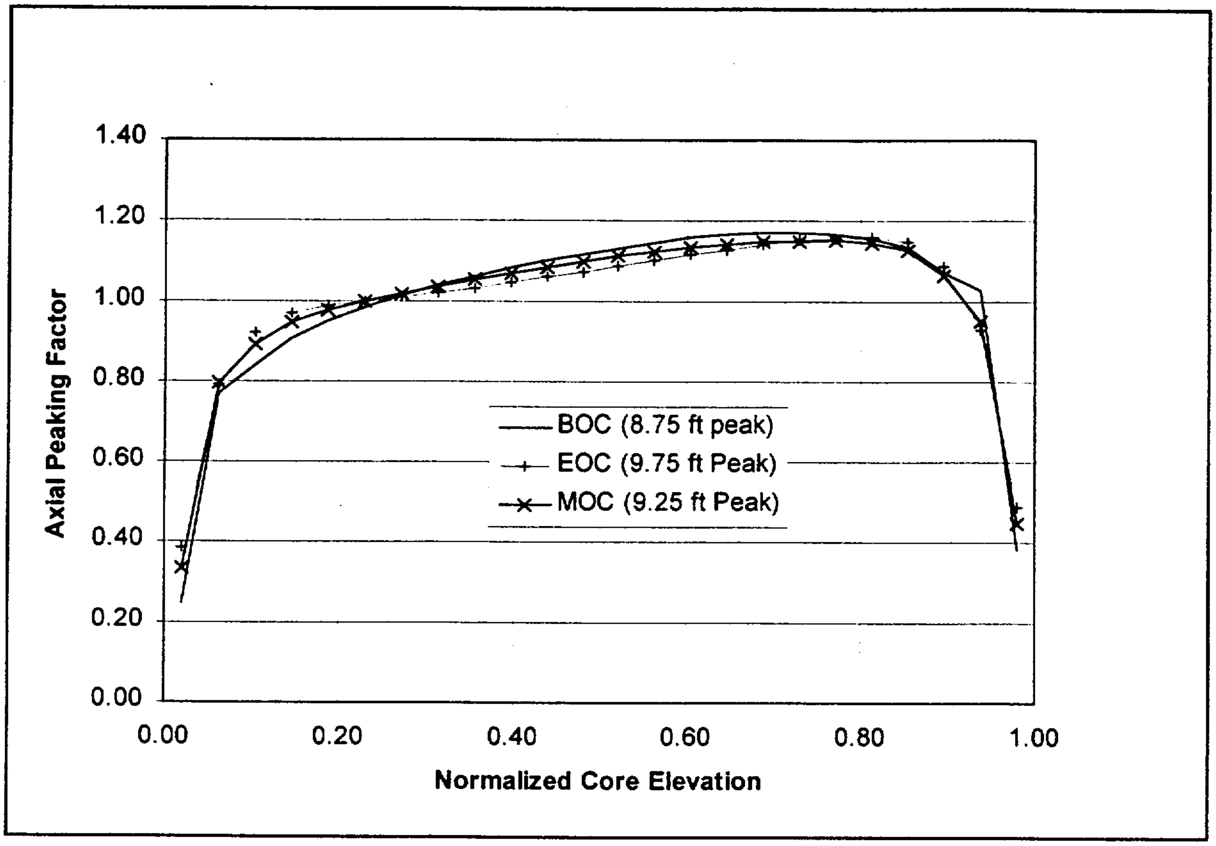


Figure 3.1 Axial Power Shapes

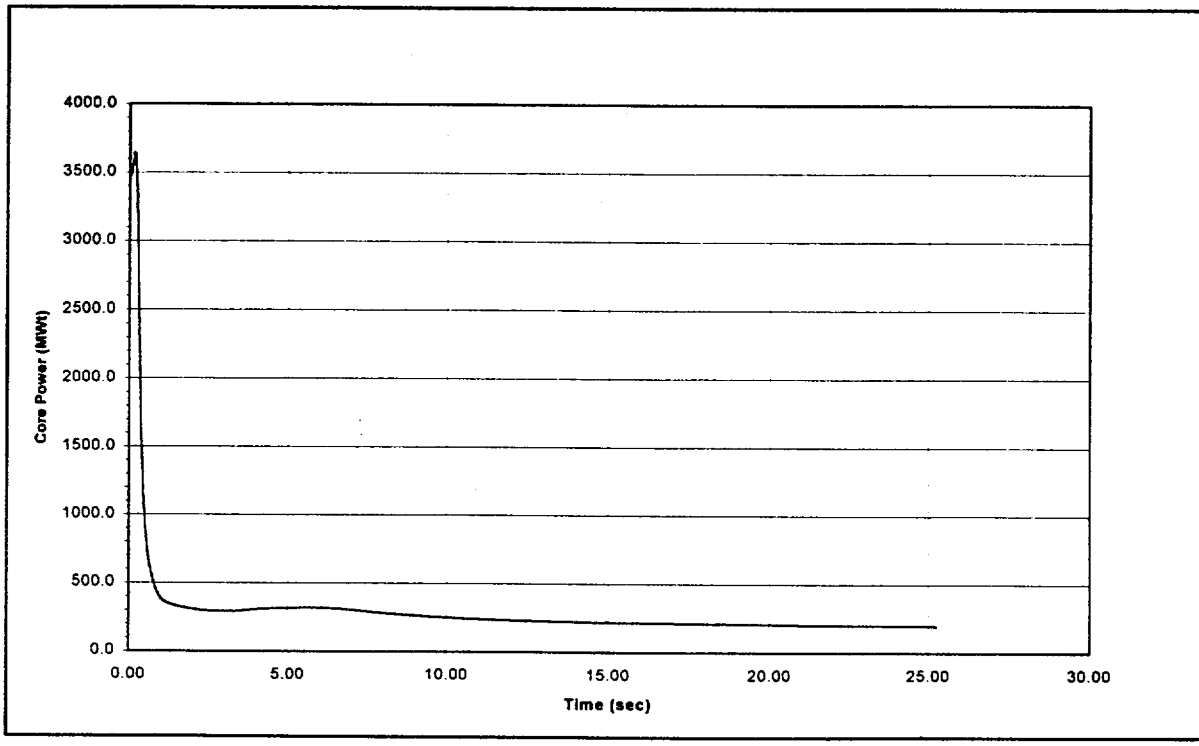


Figure 3.2 Core Total Power (Base Case)

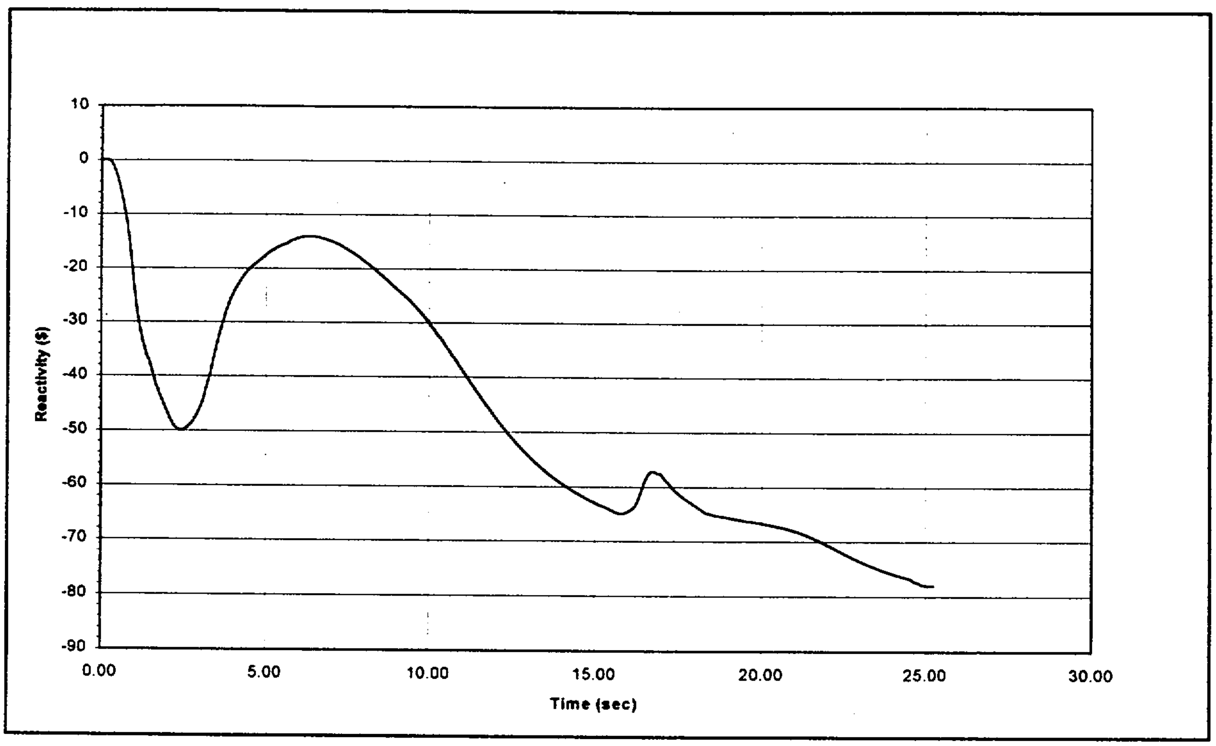


Figure 3.3 Total Reactivity (Base Case)

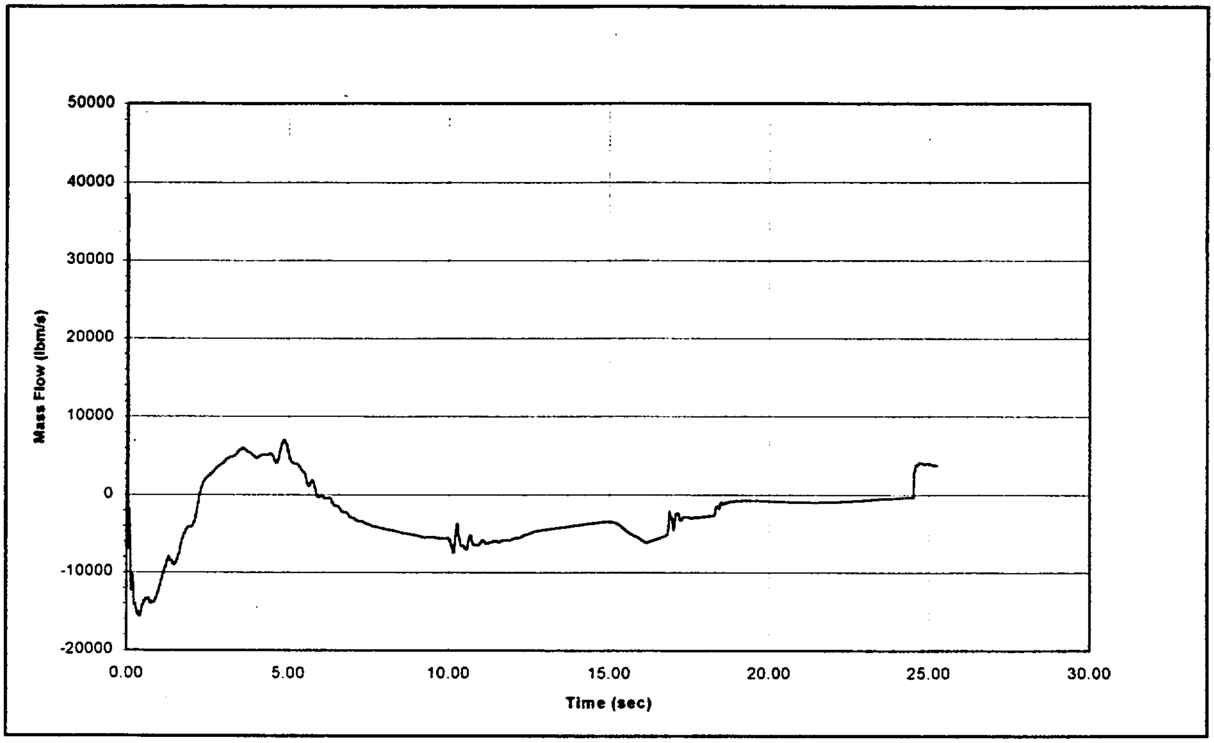


Figure 3.4 Downcomer Flow Rate (Base Case)

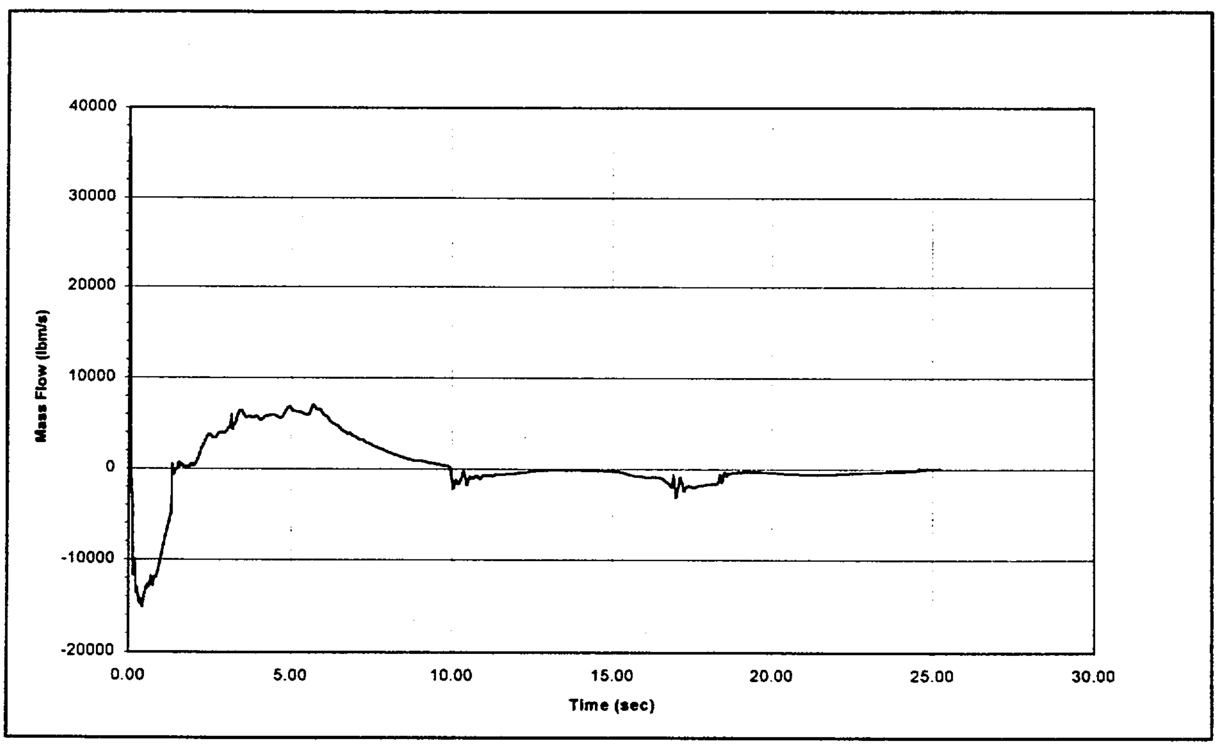


Figure 3.5 Average Core Inlet Flow Rate (Base Case)

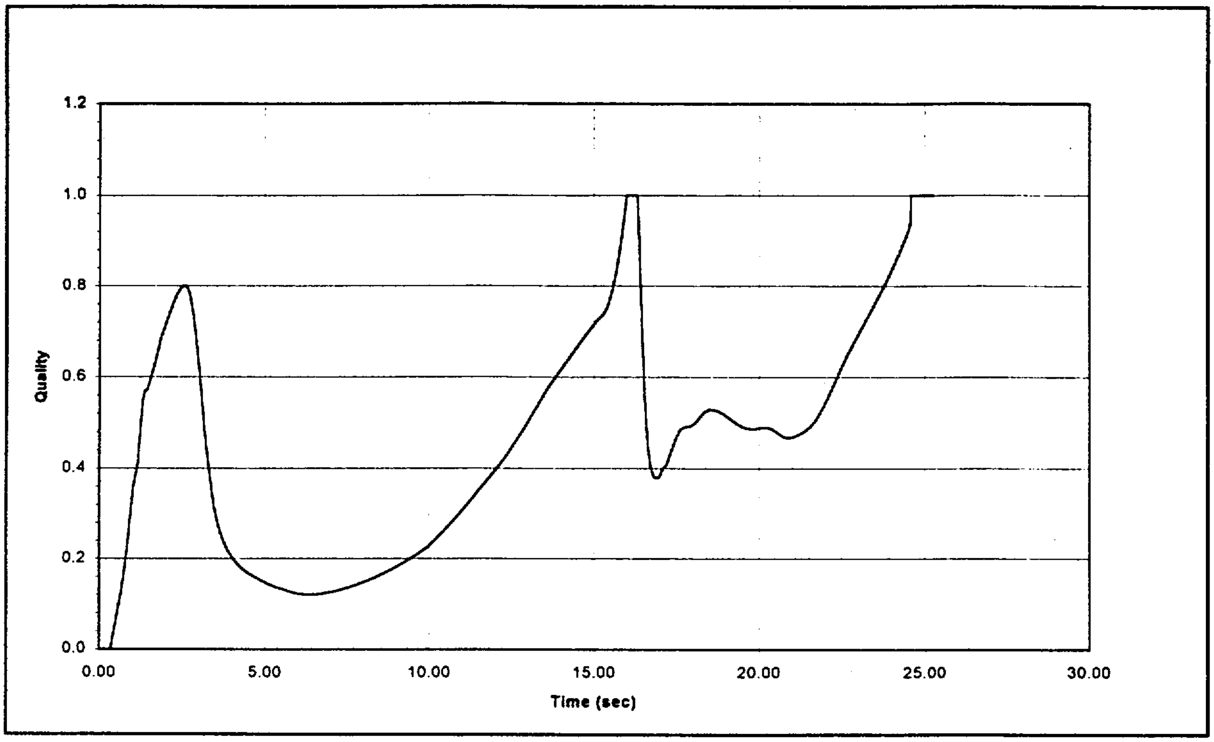


Figure 3.6 Average Core Mid Plane Quality (Base Case)

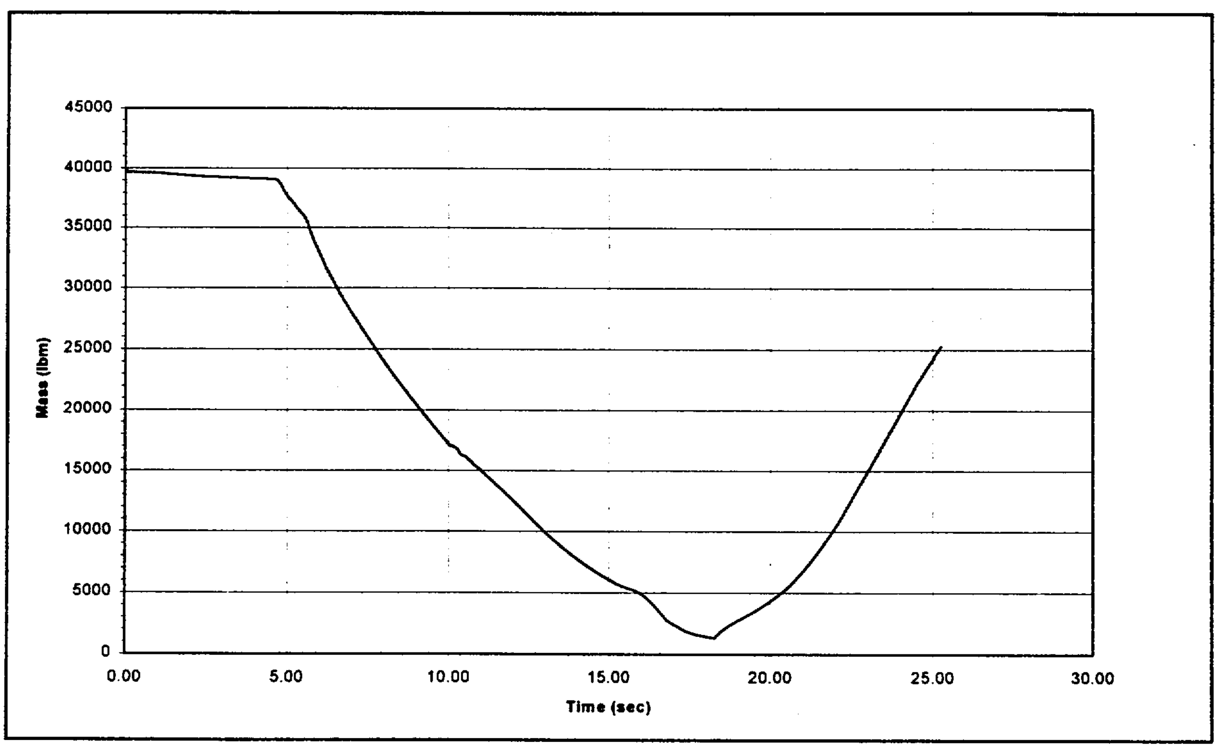


Figure 3.7 Downcomer Liquid Mass Inventory (Base Case)

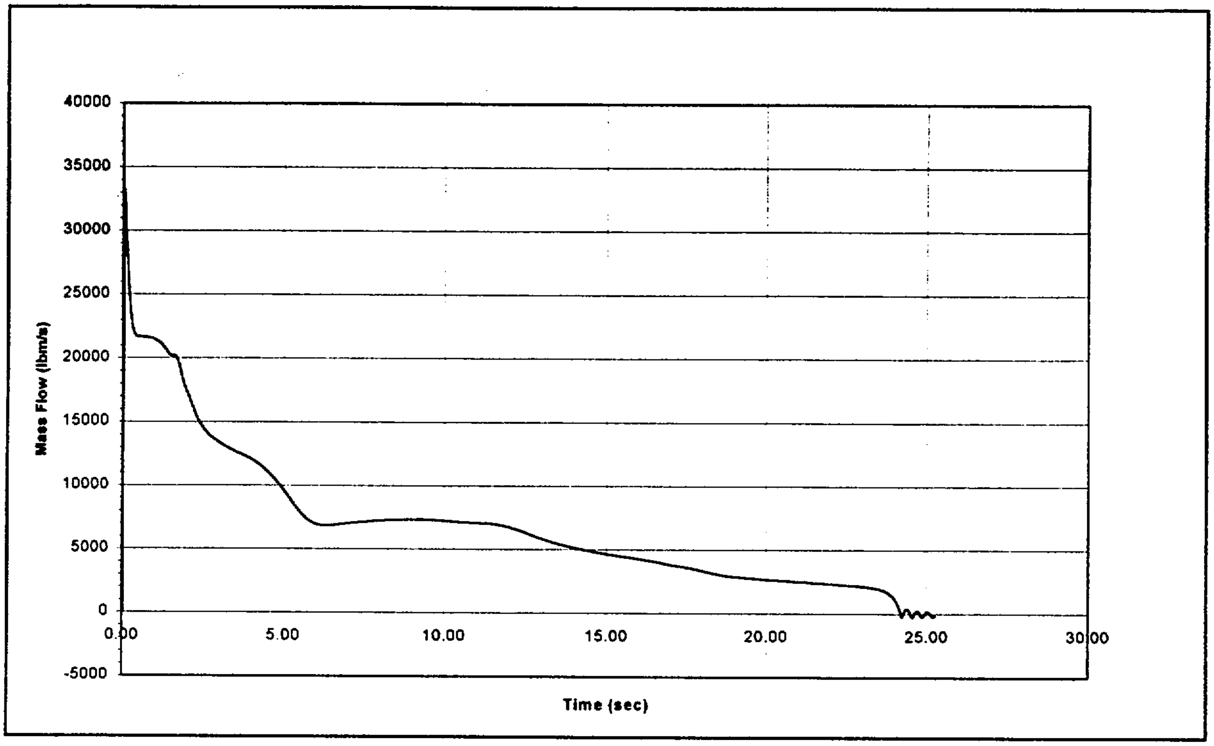


Figure 3.8 Total Break Flow (Base Case)

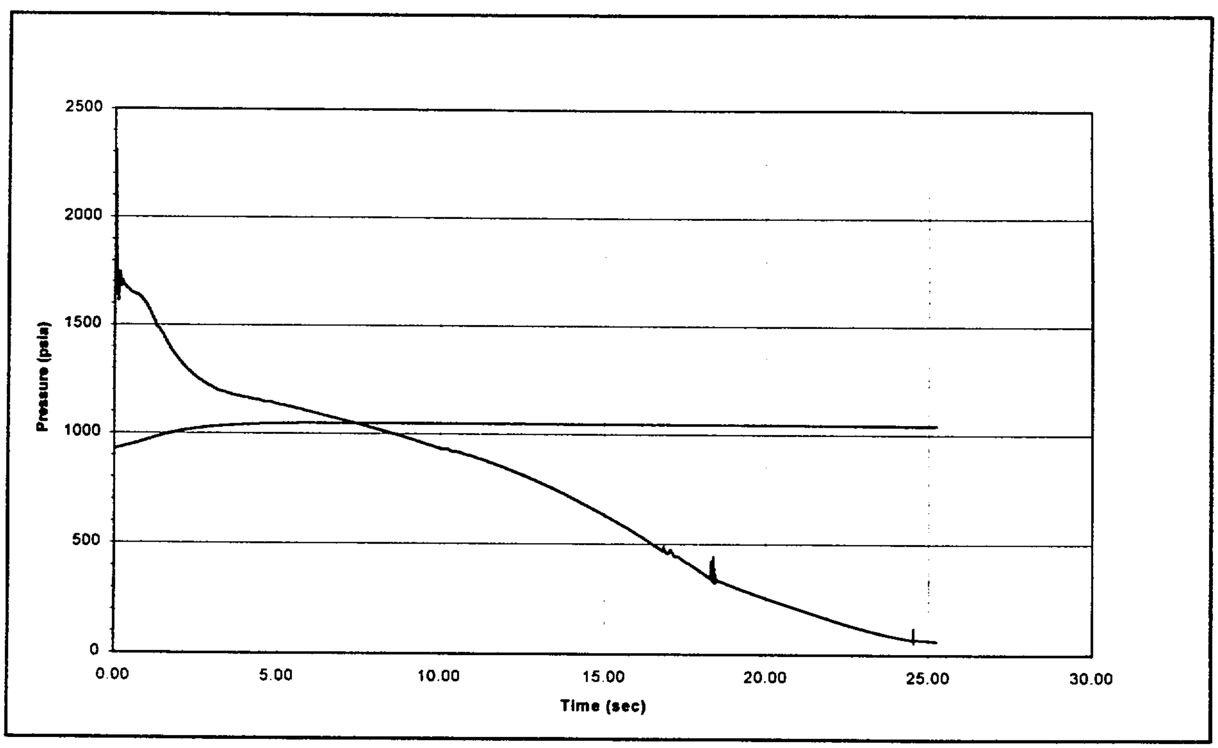


Figure 3.9 RCS and Secondary Pressures (Base Case)

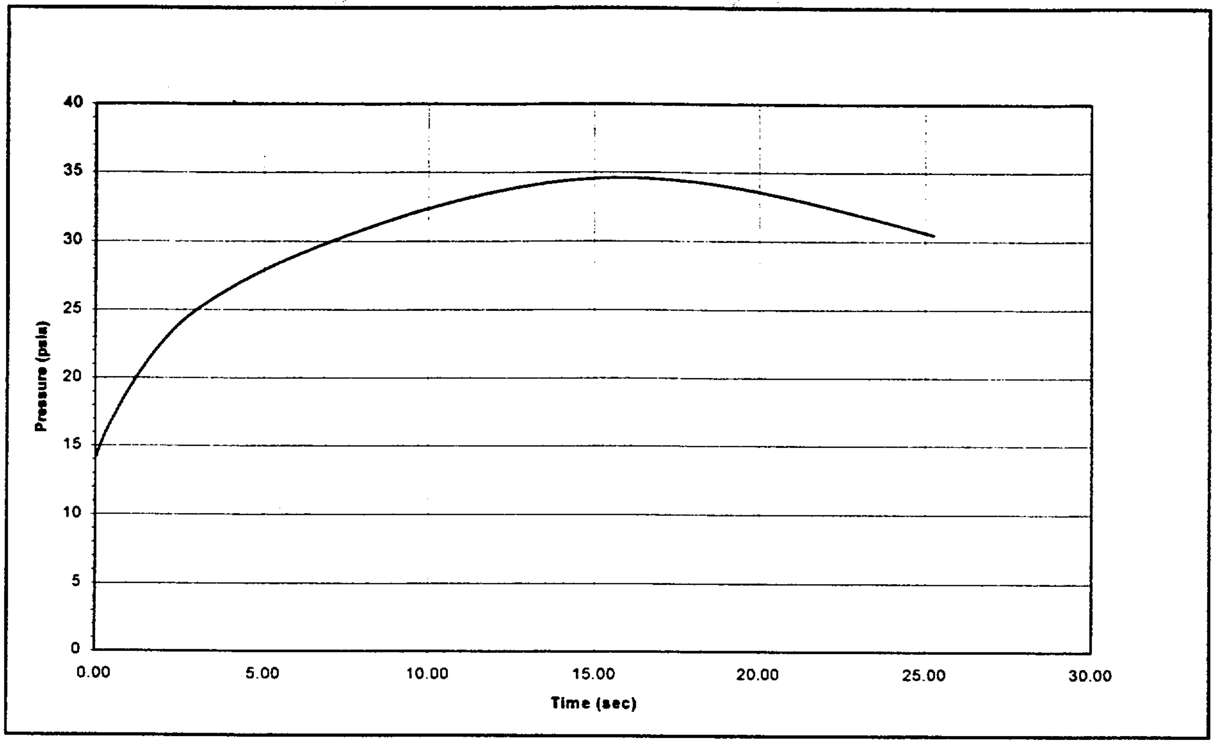


Figure 3.10 Containment Pressure (Base Case)

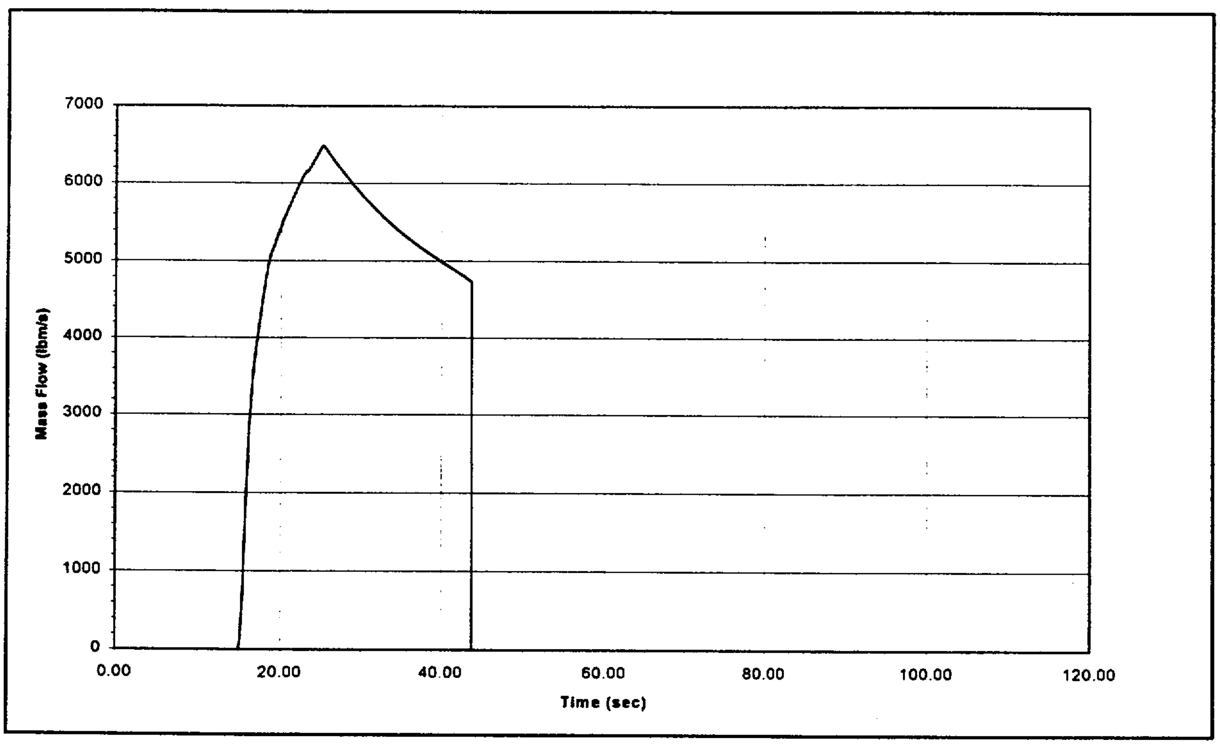


Figure 3.11 Accumulator Flow Rate (Base Case)

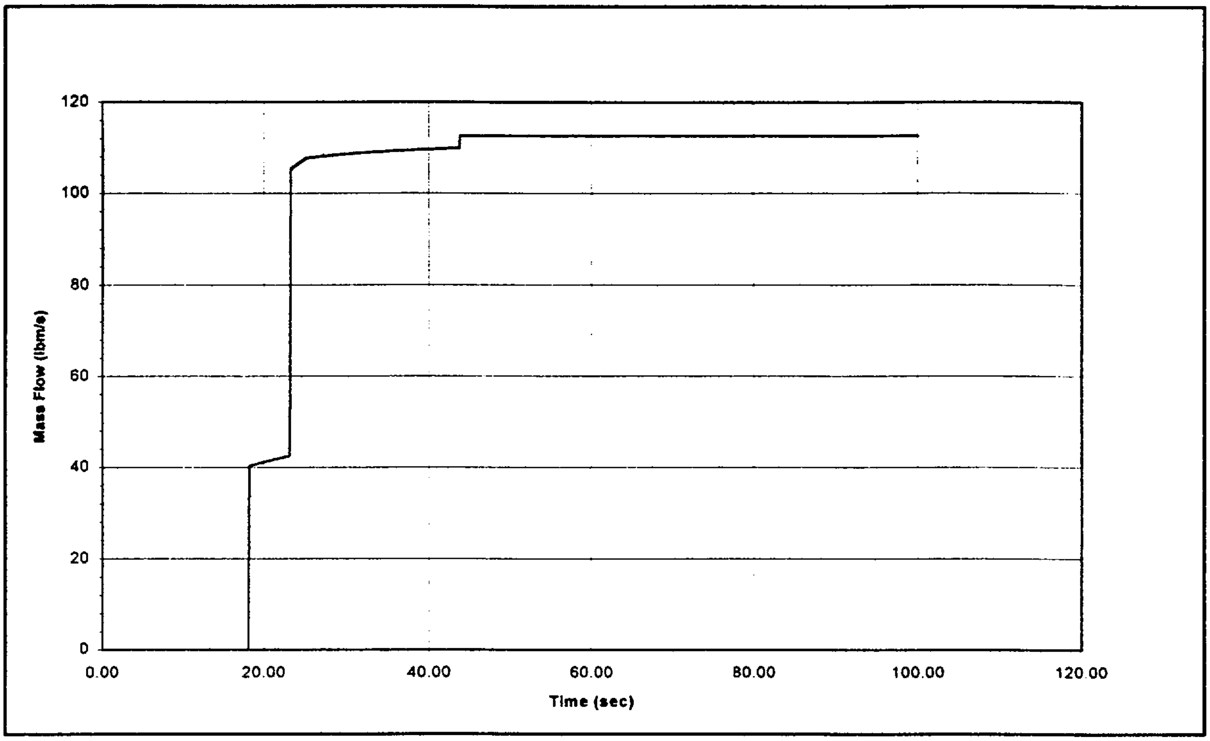


Figure 3.12 CCP and HHSI Pump Flow Rate (Base Case)

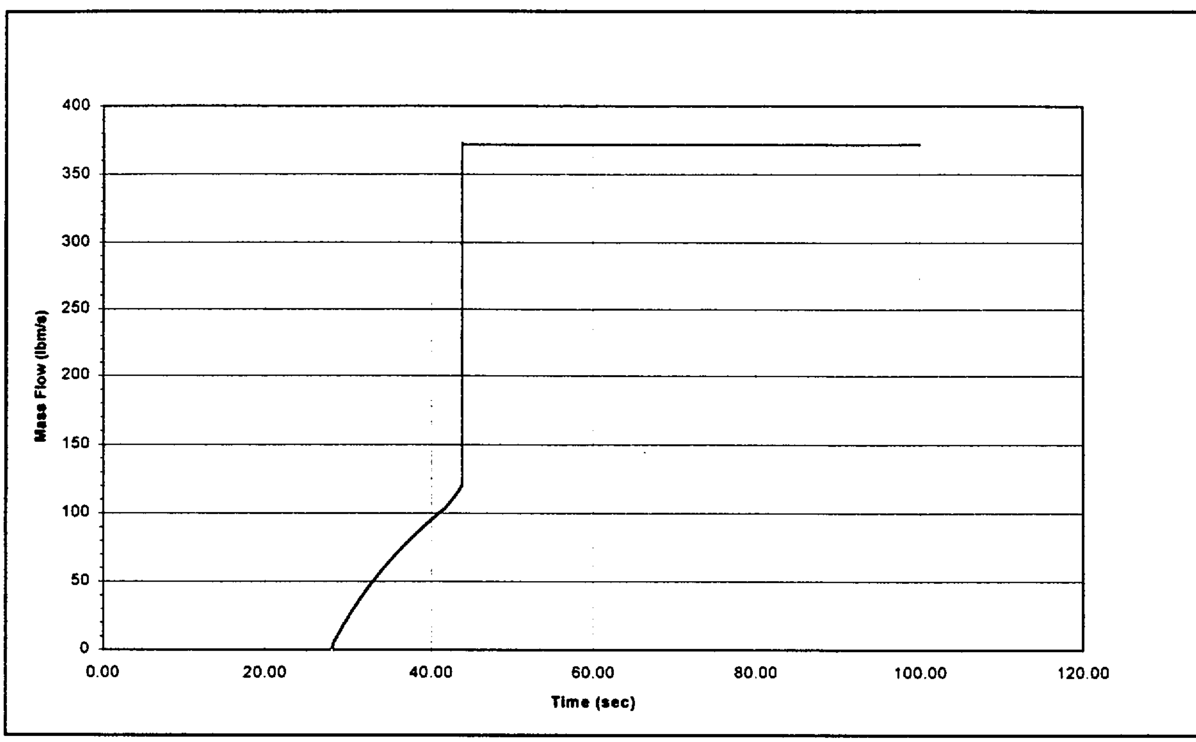


Figure 3.13 RHR Pump Flow Rate (Base Case)

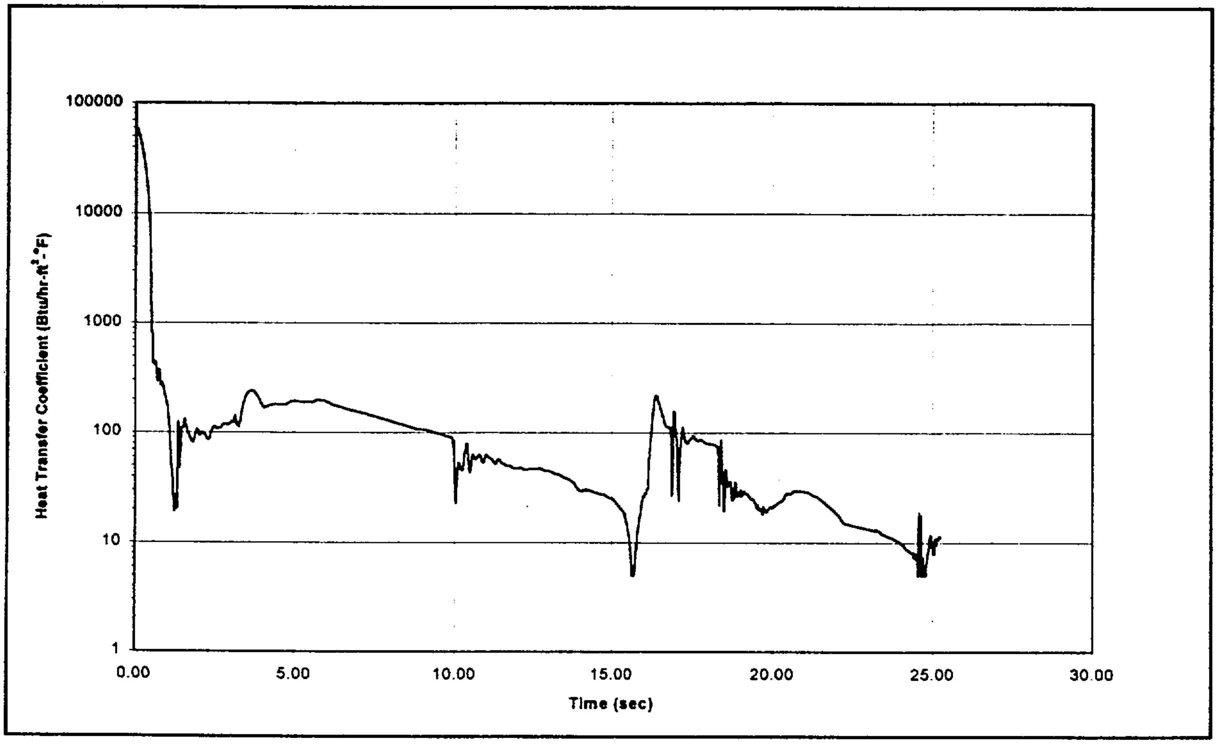


Figure 3.14 Hot Assembly Peak Power Node Heat Transfer Coeff. (Base Case)

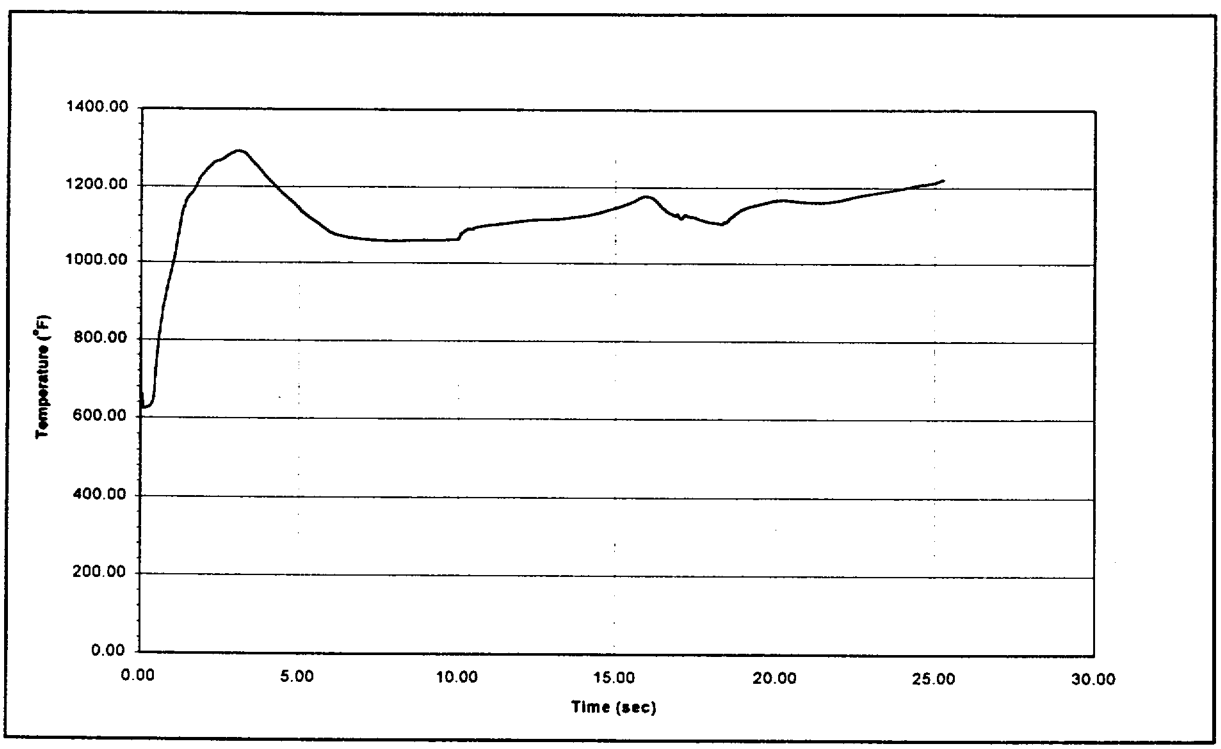


Figure 3.15 Hot Rod Temperature at PCT Node Elevation (Base Case)

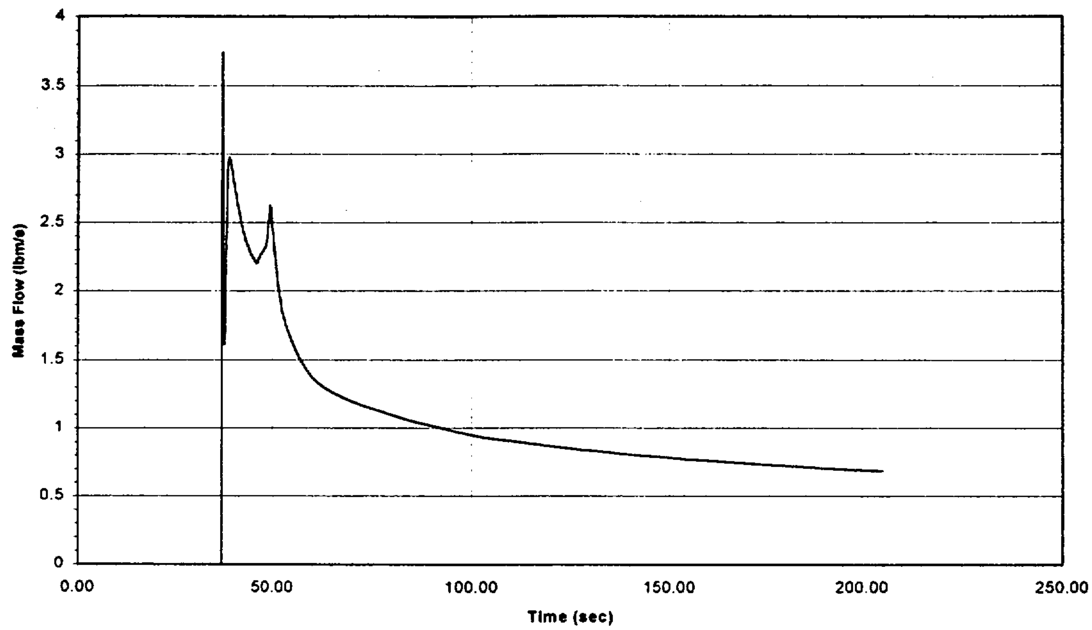


Figure 3.16 Core Flooding Rate (Inch/Seconds)

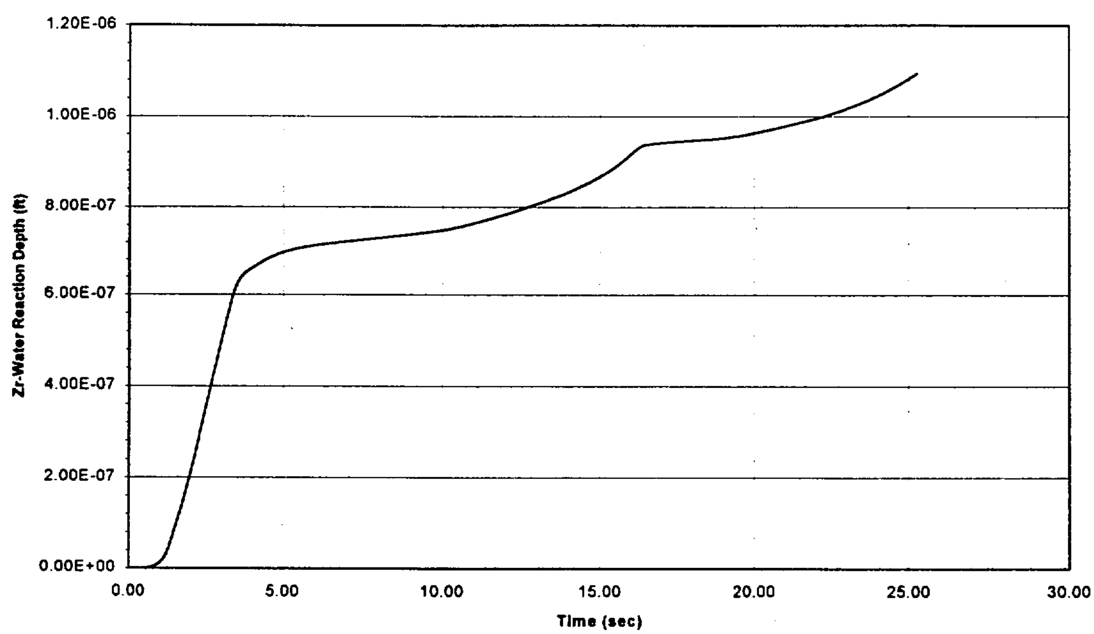


Figure 3.17 Hot Assembly Peak Power Node Zr/Water Reaction Depth (Base Case)

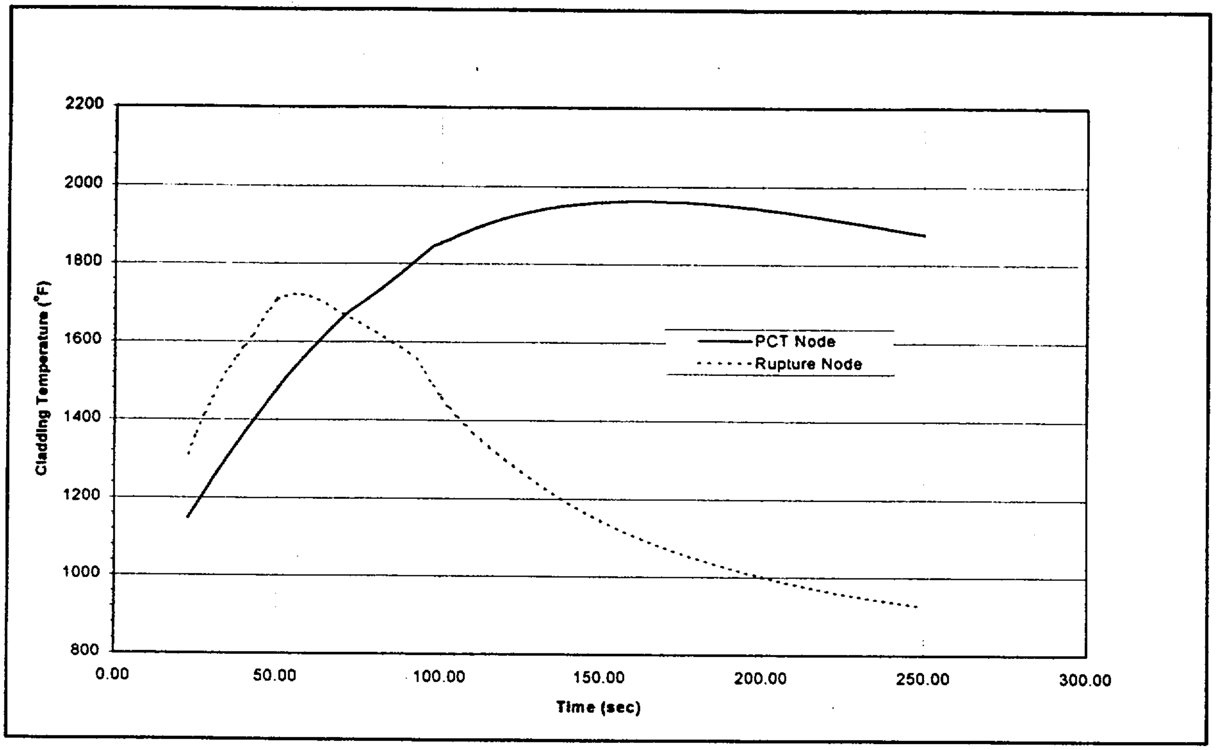


Figure 3.18 PCT/Ruptured Node Cladding Temperature (Base Case)

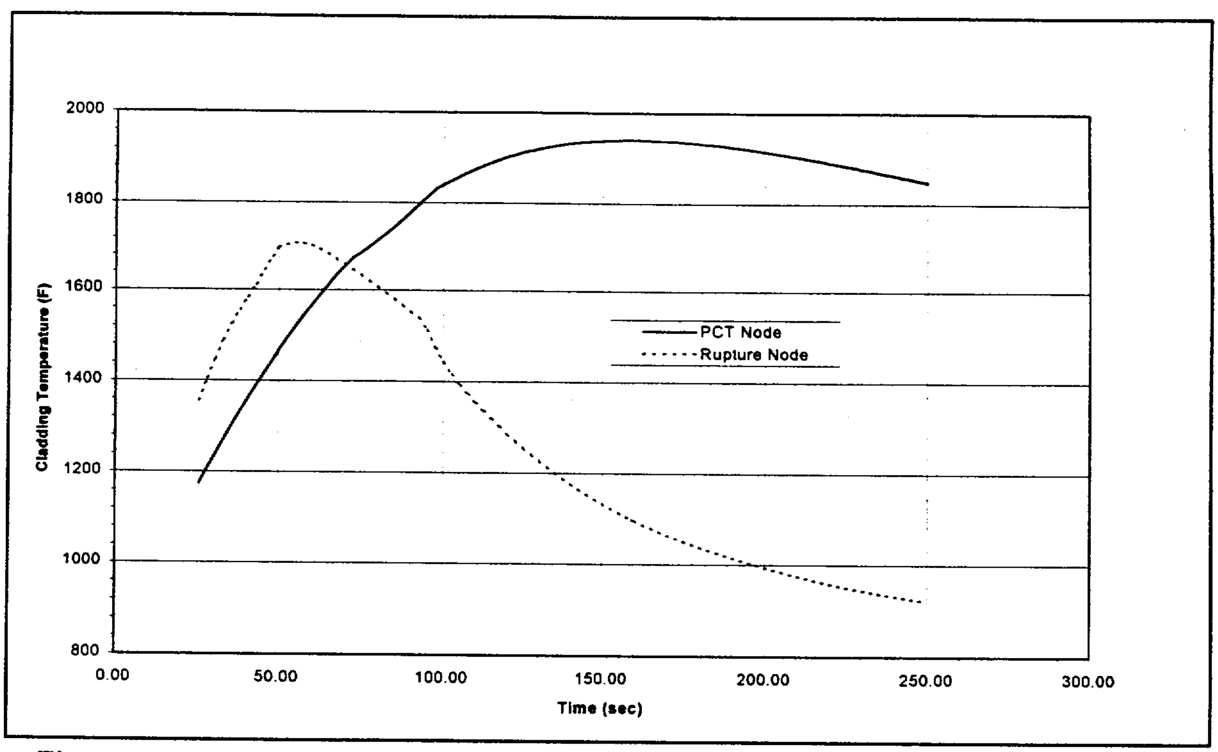


Figure 3.19 PCT/Ruptured Node Cladding Temperature (BOL Exposure, 0.8 DECLG)

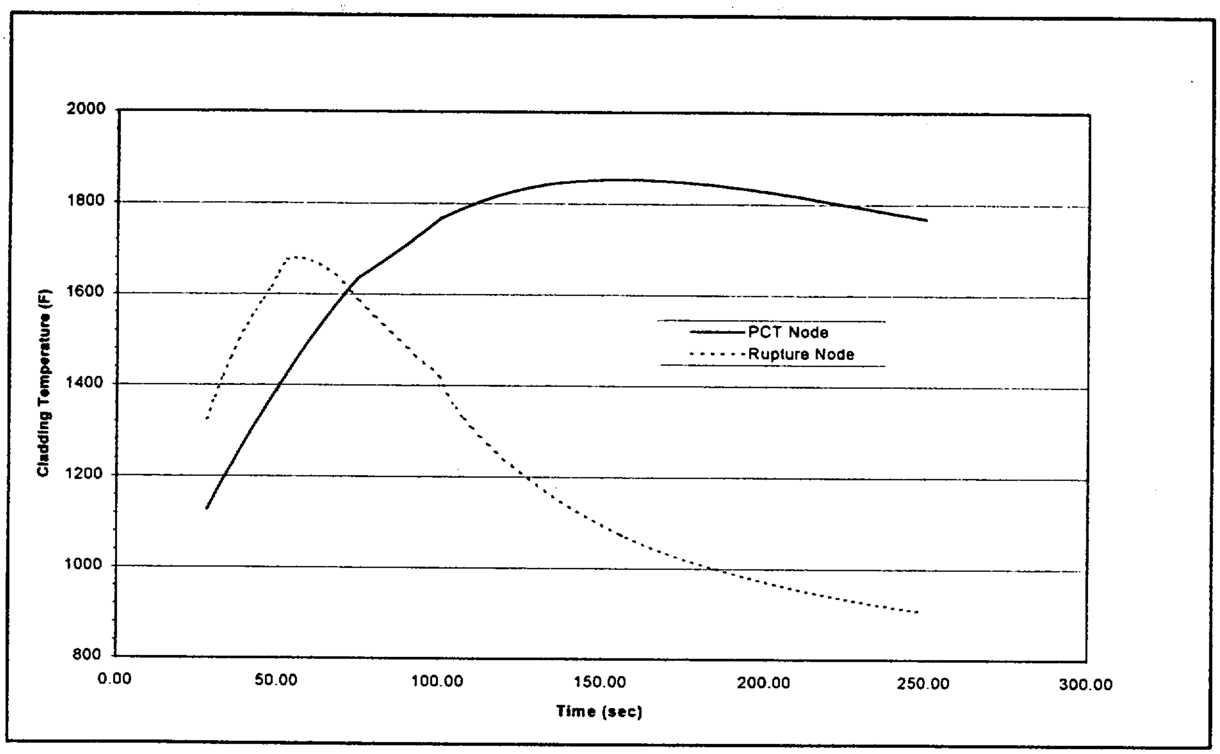


Figure 3.20 PCT/Ruptured Node Cladding Temperature (BOL Exposure, 0.6 DECLG)

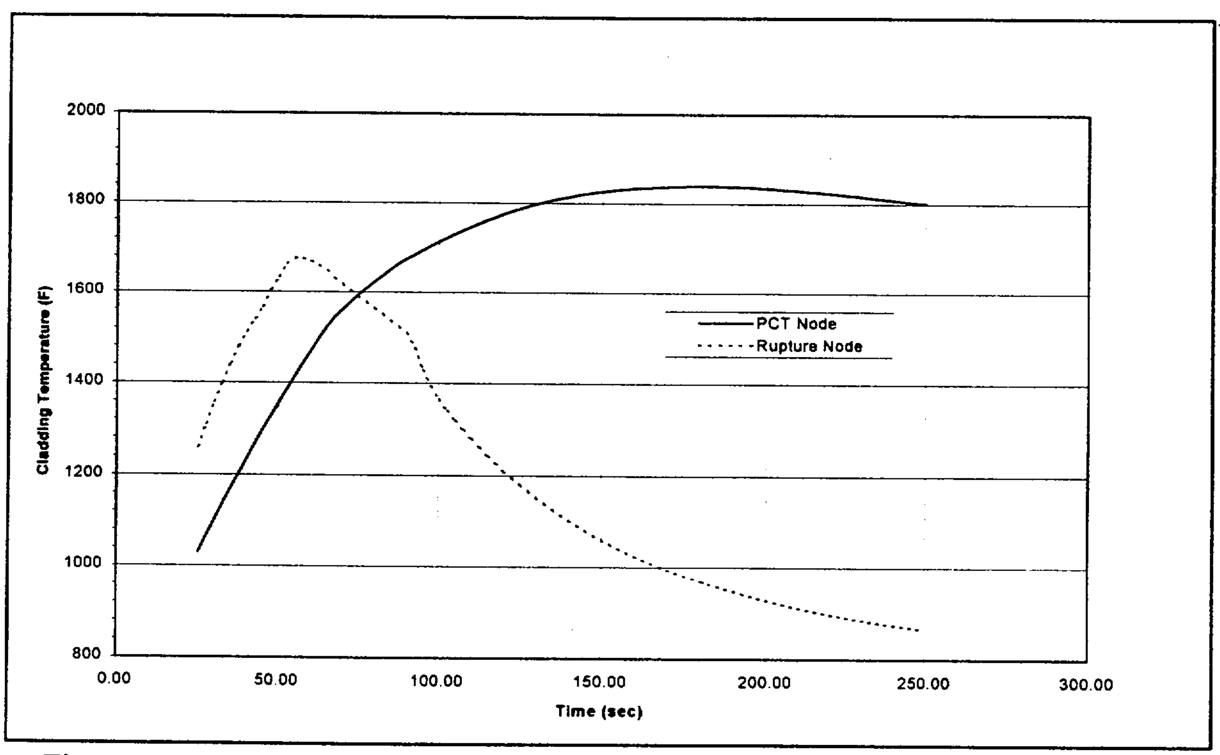


Figure 3.21 PCT/Ruptured Node Cladding Temperature (EOL Exposure, 1.0 DECLG)

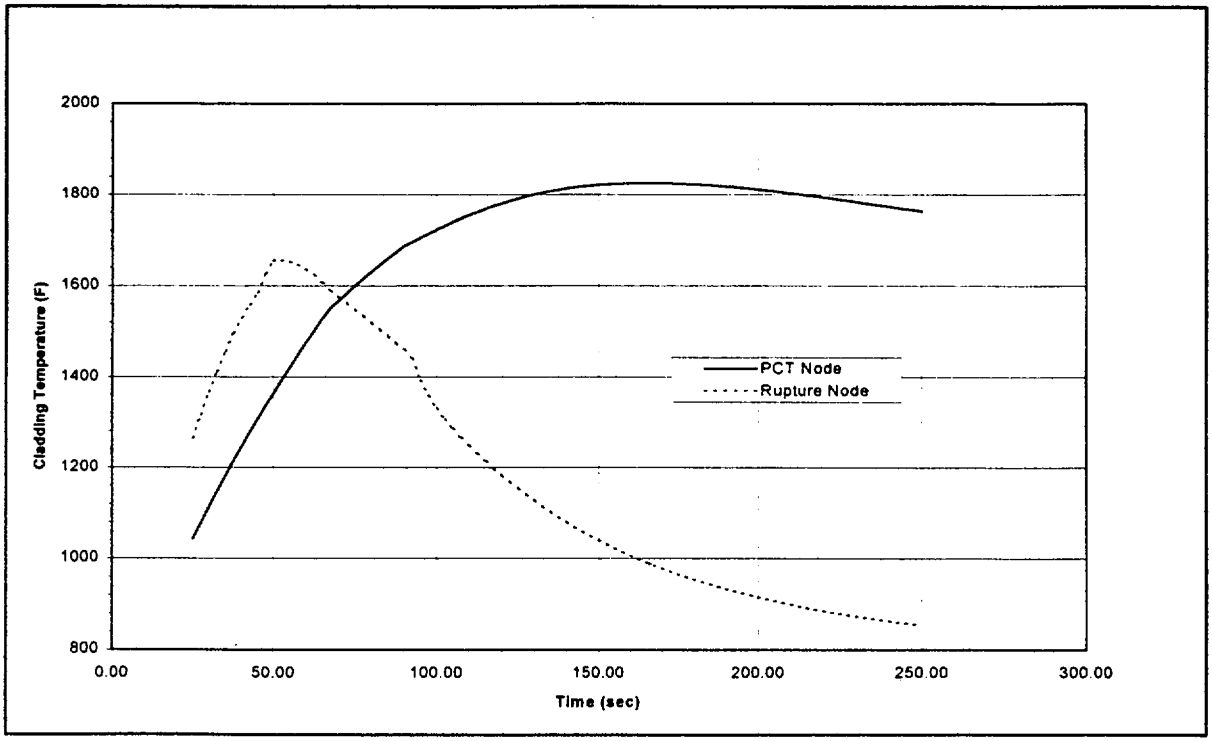


Figure 3.22 PCT/Ruptured Node Cladding Temperature (MOL Exposure, 1.0 DECLG)

CHAPTER 4

CONCLUSION

SPC's USNRC-approved (Reference 1.1) ECCS Evaluation model entitled SEM/PWR-98 has been applied to the Comanche Peak Steam Electric Station Unit One (CPSES-1)Cycle 8. Each calculation has been performed in close[14] compliance with the explicitly approved SEM/PWR-98 large break LOCA methodology. Ten calculations have been presented with three objectives:

1. To demonstrate TXU Electric's ability to properly apply SEM/PWR-98 large break LOCA Model (Reference 1.4),

2. To demonstrate the development of up-to-date input decks and conclusions which are in compliance with 10 CFR 50.46 and Appendix K thereto. Together, the codes, input decks and conclusions drawn from these calculations will be applied to perform subsequent fuel cycle analyses for the Comanche Peak Steam Electric Station Unit 1 and Unit 2; and

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The following differences exist between TXU Electric's and SPC's application: (1) the axial power shape adjustment approach, described in Section 2.2.1 and at the beginning of Chapter 3, (2) the automation of data transfer between codes, mentioned in Section 2.2, (3) the conservative inclusion of the engineering hot spot factor with the hot channel factors, described in Section 2.3.6.7 and, (4) the various plant specific considerations of Section 2.5.

3. To replace the analysis of record for Unit 1 Cycle 8 with the present analysis (base case) in fulfilment of a TXU Electric commitment.

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

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

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

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

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

5. The core is well cooled in less than 200 seconds. Therefore, the calculations comply with the long-term cooling criterion of 10 CFR 50.46 (b)(5).

Regarding the sensitivity studies it has been found:

1. The most limiting break is a Double-Ended Guillotine break in the main coolant pump discharge line with a 1.0 discharge coefficient (1.0 DEG).

2. The most limiting exposure is at beginning of life (BOL) and is coincident with the maximum stored energy in the fuel.

3. The most limiting single failure is that of one train of low pressure injection pumps (1 RHR pump).

4. The robustness of the methodology for the CPSES application was verified by running an additional case, identical to the base case, but with a convergence criterion variable EPSW = 0.25, instead of the nominal EPSW = 0.5. The PCT was less than 1°F lower.

TXU Electric will use the SEM/PWR-98 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 50.46 criteria and 10 CFR 50, Appendix K requirements, for both Comanche Peak Steam Electric Station Unit 1 and Unit 2.

TABLE 4.1

SUMMARY OF RESULTS FOR BASE CASE AND SENSITIVITY STUDIES

BREAK SIZE	EXPOSURE SENSITIVITY			Single Failure Sensitivity[15]: Loss of 1 Train of ECCS (1.0 DEG - BOL)
	BEGINNING Of life (BOL)	MIDDLE Of life (MOL)	END Of life (EOL)	
1.0 DEG	1963.2 ^{0}F (1) 3.2 % (2) 0.50 % (3)	1824.1 ^{0}F 1.9 % 0.36 %	1836.8 ^{0}F 2.0 % 0.35 %	1944.1 ^{0}F 3.0 % 0.49 %
0.8 DEG	1941.0 ^{0}F 2.9 % 0.48 %			
0.6 DEG	1853.1 ^{0}F 2.1 % 0.39 %			
2.0 Split	1959.6 ^{0}F 3.2 % 0.49 %			
1.6 Split	1930.6 ^{0}F 2.8 % 0.46 %			
1.2 Split	1867.6 ^{0}F 2.3 % 0.42 %			

NOTES:

ALL RESULTS FROM TOODEE2:

(1) PEAK CLADDING TEMPERATURE
(2) PERCENT LOCAL CLAD OXIDATION
(3) PERCENT CORE-WIDE[16] OXIDATION

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15 **All other cases have as single failure the loss of 1 train of low pressure injection (1 RHR pump).**

16 **Hot pin value is used as an upper bound for the core-wide value.**

CHAPTER 5

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