ATTACHMENT 6

Fort Calhoun Station Unit No. 1 – License Amendment Request to Support Use of AREVA Realistic Large Break Loss of Coolant Accident Methodology

AREVA Non-Proprietary Report

AREVA

SOURCE REFERENCE RECORD

Source Reference for: 103-2502-00 Fort Calhoun Station Realistic Large Break LOCA Summary Report
Document Number/Title

File No.	Subject	Remarks/Applicable Licensing Document
32-5063600-00	Fort Calhoun Cycle 24 S-RELAP5 Base Deck Input Development	
32-5063601-00	Fort Calhoun Cycle 24 RODEX3A Deck Development for RLBLOCA	
32-5063602-00	Fort Calhoun Cycle 24 ICECON Model for RLBLOCA	
32-5063603-00	Fort Calhoun Cycle 24 S-RELAP5 Steady State Initialization for RLBLOCA Analysis	
32-5063604-00	Fort Calhoun Cycle 24 RLBLOCA Uncertainty Analysis	

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.

Contents

Tables

Figures

Nomenclature

ASI	Axial Shape Index
CCFL	Counter Current Flow Limit
CE	Combustion Engineering, Inc.
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CL	Cold Leg
CSAU	Code Scaling, Applicability and Uncertainty
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EM	Evaluation Model
F_r^T	Total Integrated Radial Peaking Factor
FANP	Framatome Advanced Nuclear Power, Inc.
HFP	Hot Full Power
HPSI	High Pressure Safety Injection
LBLOCA	Large Break Loss-of-Coolant Accident
LHR/LHGR	Linear Heat Rate/Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MTC	Moderator Temperature Coefficient
NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPPD	Omaha Public Power District
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PORV	Power Operated Relief Valve
PWR	Pressurized Water Reactor

Nomenclature *(Continued)*

RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RLBLOCA	Realistic Large Break LOCA
RV	Reactor Vessel
SBLOCA	Small Break Loss-of-Coolant Accident
SER	Safety Evaluation Report
SG	Steam Generator
SIAS	Safety Injection Actuation Signal
SIRWT	Safety Injection and Refueling Water Tank
SIT	Safety Injection Tank

1.0 Introduction

This report describes and provides results from a RLBLOCA analysis for the Fort Calhoun Station. The plant is a CE-designed 1,525 MWt (as analyzed herein) PWR plant with a dry containment. FANP is the current fuel supplier. The plant is a 2x4-loop design—two hot legs and four cold legs. The loops contain four RCPs, two U-tube steam generators and a pressurizer. The ECCS is provided by two independent safety injection trains and four SITs.

The analysis herein supports operation for Cycle 24 and beyond with FANP's Advanced CE14 HTP fuel design using M5® cladding, unless invalidated by changes in Technical Specifications, Core Operating Limits Report, core design, fuel design, plant hardware or plant operation. The analysis represents a large break LOCA methodology change (from deterministic to realistic) and a fuel design change (from the current CE-HTP 14x14 design using Zr-4 cladding to the Advanced CE14 HTP design having M5® cladding). The core contains 133 FANP 14x14 Advanced CE14 HTP fuel assemblies with M5® cladding. The analysis was performed in compliance with The NRC-approved RLBLOCA EM (Reference 1). Analysis results confirm that the 10CFR50.46(b) acceptance criteria presented in Section 3.0 are met and serve as the basis for operation of the Fort Calhoun Station with FANP fuel.

The non-parametric statistical methods inherent in the FANP RLBLOCA methodology provide for consideration of a full spectrum of break sizes, break configuration (guillotine or split break), axial power shapes, and plant operational parameters. A conservative single-failure assumption is applied in which the negative effects of the loss of a train of ECCS pumped injection is simulated. Regardless of the single-failure assumption, all containment pressure-reducing systems are assumed fully functional. The effects of gadolinia-bearing fuel rods and peak fuel rod exposures are considered.

2.0 Summary

The limiting PCT is 1,675 °F; it is for a UO_2 rod. Gadolinia-bearing rods of 4 w/o and 8 w/o Gd_2O_3 were also analyzed, but were not limiting. This RLBLOCA result is based on a case set comprised of 59 individual transient cases. The core is composed of only FANP 14x14 thermal-hydraulically compatible fuel designs; hence, from the standpoint of LBLOCA analyses, no consideration of co-resident fuel (mixed core) is necessary. Table 2.1 gives the analysis parameters for the limiting (95/95) PCT case.

The analysis assumed full-power operation at 1,525 MWt (plus uncertainties), a steam generator tube plugging level of 10 percent in both steam generators, a total LHR of 15.5 kW/ft (technical specification value with no axial dependency), and an F_r^T of 1.86 (including uncertainty and control rod insertion effect). The analysis addresses typical operational ranges or technical specification limits (whichever are applicable) with regard to pressurizer pressure and liquid level; SIT pressure, temperature (set to containment temperature) and liquid level; core inlet temperature; core flow; containment pressure and temperature; and SIRW tank temperature.

The FANP RLBLOCA methodology explicitly analyzes only fresh fuel assemblies (Reference 1, Appendix B). Previous analyses showed that once- and twice-burnt fuel is not limiting up to peak rod average exposures of 62,000 MWd/MTU. The analysis demonstrates that the 10CFR50.46(b) criteria listed in Section 3.0 are satisfied.

Table 2.1 Summary of Major Parameters for the Limiting PCT Case

Core Average Burnup (EFPH)	3,513.1
Core Power (MWt)	1,527.8
Hot Rod LHR, kW/ft	15.12
Total Hot Rod Radial Peak (F_r^T)	1.862
ASI	-0.1585
Break Type	Guillotine
Break Size (ft^2/side)	2.537
Offsite Power Availability	Not Available
Decay Heat Multiplier	1.0524

3.0 Analysis

The purpose of the analysis is to verify the adequacy of the ECCS for the planned Cycle 24 plant configuration by demonstrating that the following criteria of 10CFR 50.46(b) are met:

- The calculated maximum fuel element cladding temperature shall not exceed 2,200 °F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel excluding the cladding surrounding the plenum volume were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling. The RLBLOCA analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Therefore, compliance with Criterion 1, demonstrating that the PCT is less than 2,200 F, assures that the core remains amenable to cooling and satisfies Criterion 4.

Section 3.1 of this report describes the postulated LBLOCA event. Section 3.2 describes the models used in the analysis. Section 3.3 describes the 2x4-loop PWR plant and summarizes the system parameters used in the analysis. Compliance with the RLBLOCA evaluation model SER is addressed in Section 3.4. Section 3.5 addresses the mixed core. Section 3.6 summarizes the results of the RLBLOCA analysis.

3.1 Description of the LBLOCA Event

A LBLOCA is initiated by a postulated large rupture of the RCS piping. Based on deterministic studies, the worst break location is in the cold leg piping between the RCP and the RV for the RCS loop containing the pressurizer. The break initiates a rapid depressurization of the RCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, reactor trip is conservatively neglected in the analysis. The reactor is shut down by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The large cold leg break is assumed to open instantaneously. For this break, a

rapid primary system depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience DNB. Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect and core fission ends. As heat transfer from the fuel rods is reduced, the cladding temperature rises.
Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate and may also lead to a period of positive core flow or reduced downflow as the RCPs in the intact loops continue to supply water to the vessel. Cladding temperatures may be reduced and some portions of the core may rewet during this period.

This positive core flow or reduced downflow period ends as two-phase conditions occur in the reactor coolant pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg.

Mitigation of the LBLOCA begins when an SIAS occurs. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst active single-failure be considered for ECCS safety analysis. This worst active single failure was determined generically in the RLBLOCA evaluation model to be the loss of one ECCS train. The FANP RLBLOCA methodology conservatively assumes a minimal time delay and a normal (no failure irrespective of the assumed worst single active failure) lineup of the containment sprays and fan coolers to reduce containment pressure and increase break flow. The analysis assumes that one HPSI pump, one LPSI pump, all containment spray pumps and all containment fan coolers are operational.

When the RCS pressure falls below the SIT pressure, fluid from the SITs is injected into the cold legs. In the early delivery of SIT water, high pressure and high break flow will cause some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core. This improves core heat transfer and cladding temperatures begin to decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery relies solely on ECCS pumped injection. As the SITs empty, the nitrogen gas used to pressurize the SITs exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas is expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperatures created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures

may increase for a short period until additional energy is removed from the core by the LPSI and the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the steam generator and the RCP before it is vented out the break. The resistance of this flow path to the steam flow (including steam binding effects) is balanced by the driving force of water filling the downcomer. This resistance (steam binding) may act to retard the progression of core reflooding and postpone core-wide cooling. Eventually (within a few minutes of the accident), core reflooding will progress sufficiently to ensure core-wide cooling. Full core quench occurs within a few minutes after core-wide cooling. Long-term cooling is then sustained with the LPSI.

3.2 Description of Analytical Models

The RLBLOCA methodology is documented in topical report EMF-2103, *Realistic Large Break LOCA Methodology* (Reference 1). The methodology follows the CSAU evaluation methodology (Reference 2). This method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LBLOCA analysis.

The RLBLOCA methodology uses the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 for the system calculation, including the containment pressure response.

The governing two-fluid (plus non-condensibles) model with conservation equations for mass, energy and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heating.

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction, and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomenon expected during an LBLOCA event are captured. The basic building block for modeling is the hydraulic volume for fluid paths and the heat structure for a heat transfer surface. In addition, special purpose components exist to represent specific components such as the pumps or the

steam generator separators. All geometries are modeled at a level of detail necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

System nodalization details are shown in Figures 3.1 through 3.5. A point of clarification: in Figure 3.1, break modeling uses two junctions regardless of break type—split or guillotine; for guillotine breaks, Junction 151 is deleted, it is retained fully open for split breaks. Hence, total break area is the sum of the areas of both break junctions.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are chosen to reflect plant technical specifications or to match measured data. Additionally, the RODEX3A code provides initial conditions for the S-RELAP5 fuel models. Specific parameters are discussed in Section 3.3.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops (specifically, the loop with the pressurizer). The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Transient containment pressure is also calculated by S-RELAP5 using containment models derived from the CONTEMPT-LT code (Reference 3).

The methods used in the application of S-RELAP5 to the large break LOCA are described in Reference 1. A detailed assessment of this computer code was made through comparisons to experimental data, many benchmarks with cladding temperatures ranging from 1,700 °F (or less) to above 2,200 °F. These assessments were used to develop quantitative estimates of the ability of the code to predict key physical phenomena in a PWR LBLOCA. Various models—for example, the core heat transfer, the decay heat model and the fuel cladding oxidation correlation—are defined based on code-to-data comparisons and are, hence, plant independent.

The RV internals are modeled in detail (Figures 3.3 through 3.5) based on specific inputs supplied by OPPD. Nodes and connectivity, flow areas, resistances and heat structures are all accurately modeled. The location of the hot assembly/hot pin(s) is unrestricted; however, the channel is always modeled to restrict appreciable upper plenum liquid fallback.

The final step of the best-estimate methodology is to combine all the uncertainties related to the code and plant parameters and estimate the PCT at a high probability level. The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Base Plant Input File Development

   First, RODEX3A and S-RELAP5 base input files for the plant (including a containment input file) are developed. Code input development guidelines are followed to ensure that the model nodalization is consistent with that used in the code validation.

2. Sampled Case Development

   The non-parametric statistical approach requires that many "sampled" cases be created and processed. For every set of input created, each "key LOCA parameter" is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by plant technical specifications or data). Those parameters considered "key LOCA parameters" are listed in Table 3.1. This list includes both parameters related to LOCA phenomena (based on the PIRT provided in Reference 1) and to plant operating parameters.

3. Determination of Adequacy of ECCS

   The RLBLOCA methodology uses a non-parametric statistical approach to determine values of PCT at the 95 percent probability level with 95 percent confidence (95/95). Total oxidation and total hydrogen generation are based on the 95/95 PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the regulatory criteria set forth in Section 3.0.

3.3 Plant Description and Summary of Analysis Parameters

The plant analysis presented herein is for Fort Calhoun Station Unit No. 1, which has a 2x4-loop arrangement. There are two hot legs each with a U-tube steam generator and four cold legs each with a RCP[1]. The RCS also includes one pressurizer connected to a hot leg. The core contains 133 14x14 thermal-hydraulic compatible FANP HTP fuel assemblies. The ECCS includes four SIT lines, each connecting to a cold leg pipe downstream of the pump discharge. The HPSI and LPSI lines tee into the SIT lines prior to their connection to the cold legs. The ECCS HPSI pumps are cross-connected. The single failure assumption renders one LPSI pump, two LPSI injection MOVs, and a HPSI pump inoperable. This results in one LPSI pump injecting through two valves into cold legs 1A (leg containing the break) and 1B, and one HPSI pump injecting through four valves in all four of the cold legs. This models the break in

[1] The RCP are Byron-Jackson Type DFSS pumps as specified by OPPD. The homologous pump performance curves were input to the S-RELAP5 plant model; the built-in S-RELAP5 curves were not used.

the same loop as the pressurizer, as directed by the RLBLOCA methodology. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water (i.e. Recirculation Actuation Signal) for ECCS pumped injection need not be considered.

The S-RELAP5 model explicitly describes the RCS, RV, pressurizer, and the ECCS. The model also describes the steam generator secondary side that is instantaneously isolated (closed MSIV and feedwater pumps trip) at the time of the break. A symmetric steam generator tube plugging level of 10 percent per steam generator is assumed, to bound future fuel cycles.

Plant input modeling parameters were provided by OPPD specifically for Fort Calhoun Station. By procedure, OPPD maintains plant documentation current, and directly communicates with FANP on plant design and operational issues regarding reload cores. OPPD and FANP will continue to interact in that fashion regarding the use of FANP fuel in Fort Calhoun Station. Both entities have ongoing processes that assure the ranges and values of input parameters for the Fort Calhoun Station RLBLOCA analysis bound those of the as-operated plant values.

As described in the FANP RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. A list of the sampled parameters is given in Table 3.1. The LBLOCA phenomenological uncertainties are provided in Reference 1. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 3.2. Plant data are analyzed to develop uncertainty ranges for the process parameters sampled in the analyses. Uncertainty ranges capture all expected uncertainties, including plant operation and instrumentation measurements. Table 3.3 presents a summary of the uncertainties used in the analyses. Two parameters, SIRWT temperature for ECCS pumped injection flows and diesel start time, are set at conservative bounding values for all calculations. Where applicable, the sampled parameter ranges are based on technical specification limits. Plant and design data are used to define range boundaries for some parameters, for example, loop flow and containment temperature.

For the FANP RLBLOCA evaluation model, significant containment parameters, as well as NSSS parameters, were established via a PIRT process. Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to PCT) containment parameters—containment pressure and temperature. In many instances, the conservative guidance of CSB 6-1 (Reference 4) was used in setting the remainder of the containment model input parameters. As noted in Table 3.3, containment temperature is a sampled parameter. Containment pressure is indirectly ranged by sampling the upper containment volume (Table 3.3). Conservatively, a value below the containment-related technical specification minimum SIRWT

temperature was used for the building sprays. A Fort Calhoun Station-specific [ ] Uchida heat transfer coefficient multiplier was established through application of the process used in the RLBLOCA EM (Reference 1) sample problems. That process involves a comparison, shown in Figure 3.6, of pressure response curves generated from a RLBLOCA EM generically acceptable, best-estimate correlation and the plant-specific form of the Uchida correlation. The comparison demonstrates that the [ ] Uchida multiplier is within the RLBLOCA guidelines acceptance criterion and, therefore, validates the acceptability of the Fort Calhoun Station S-RELAP5 containment model.

3.4 NRC SER Compliance

The NRC SER on the RLBLOCA evaluation model stipulates a number of requirements (Reference 1). The application reported herein complies with all SER requirements. The requirements are addressed in Table 3.4.

3.5 Mixed-Core Considerations

The Fort Calhoun Station core model contains 133 14x14 FANP HTP fuel assemblies. All fuel assembly cages are similar in design, thermal-hydraulically compatible. Hence, due to the homogenous nature of the core fuel assemblies, no mixed-core evaluation need be done and no mixed-core penalty need be applied to the LBLOCA analysis.

3.6 Realistic Large Break LOCA Results

A case set comprising 59 transient calculations was performed sampling the parameters listed in Table 3.1. For each transient calculation, PCT was calculated for a UO_2 rod and for gadolinia-bearing rods with concentrations of 4 w/o and 8 w/o Gd_2O_3. The limiting PCT (1,675 °F) occurred in Case 10 for a UO_2 rod. The major parameters for the limiting transient are presented in Table 2.1. Table 3.5 lists the limiting PCT results for the hot fuel rods. The fraction of total hydrogen generated is conservatively bounded by the calculated total percent oxidation, which is well below the 1 percent limit. A nominal 50/50 PCT case, based on the UO_2 hot rod, was identified as Case 22. The nominal PCT is 1,366 °F. This result can be used to quantify the relative conservatism in the 95/95 result; in this analysis, it is 309 °F.

The hot fuel rod results are given in Table 3.5 and event times for the limiting PCT case are shown in Table 3.6, respectively. Figure 3.7 shows linear scatter plots of the key parameters sampled for the 59 calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter sample ranges used in the analysis.[2] Figures 3.8 and 3.9 are PCT scatter plots

[2] Figure 3.7, also Figure 3.9, presents the break flow area for only one break flow junction; total break flow area is the sum of the break flow areas from both break flow junctions (see break modeling in Figure 3.1).

versus the time of PCT and versus break size[3] from the 59 calculations, respectively. Figure 3.10 shows the maximum oxidation versus PCT for the 59 calculations. Figures 3.11 through 3.21 present transient results for key parameters from the S-RELAP5 limiting case. Figure 3.11 is a PCT elevation-independent plot; this figure clearly indicates that the transient exhibits a sustained and stable quench.

[3] The RLBLOCA approval provides for break size ranging down to 10 percent of the pipe cross-sectional area. Case set results were examined for the occurrence of phenomena characteristic of small break LOCA (loop seals, periods of natural circulation cooling, no rapid DNB immediately after transient initiation, etc.). The smallest break in the case set showed complete core voiding during blowdown and core refilling after the start of SIT injection—all LBLOCA characteristics. No characteristics exclusive to SBLOCA were observed in the Fort Calhoun Station case set results.

Table 3.1 Sampled LBLOCA Parameters

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties and burnup)
	Break type (guillotine versus split)
	Break size
	Critical flow discharge coefficients (break)
	Decay heat
	Critical flow discharge coefficients (surgeline)
	Initial upper head temperature
	Film boiling heat transfer
	Dispersed film boiling heat transfer
	Critical heat flux
	T_{min} (intersection of film and transition boiling)
	Initial stored energy
	Downcomer hot wall effects
	Steam generator interfacial drag
	Condensation interphase heat transfer
	Metal-water reaction
Plant[4]	
	Offsite power availability
	Core power and power distribution
	Pressurizer pressure
	Pressurizer liquid level
	SIT pressure
	SIT liquid level
	SIT temperature (based on containment temperature)
	Containment temperature
	Containment volume
	Initial flow rate
	Initial operating temperature
	Diesel start (for loss of offsite power only)

[4] Uncertainties for plant parameters are based on plant-specific values with the exception of "Offsite power availability," which is a binary result that is specified by the analysis methodology.

Table 3.2 Plant Operating Range Supported by the LOCA Analysis

	Event	Operating Range
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.440 in
	b) Cladding inside diameter	0.387 in
	c) Cladding thickness	0.0265 in
	d) Pellet outside diameter	0.3805 in
	e) Pellet density	96% of theoretical
	f) Active fuel length	129.3 in
	g) Resinter densification	[ ]
	h) Gd_2O_3 concentrations	4 and 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis considers plant-specific form and friction losses
	b) Pressurizer location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	FANP Advanced CE14 HTP
	e) SG tube plugging	10%
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal reactor power	1,525 MWt
	b) LHR	≤ 15.5 kW/ft[5]
	c) F_r^T	≤ 1.86[6]
	2.2 Fluid Conditions	
	a) Loop flow	74 Mlbm/hr ≤ M ≤ 81.3 Mlbm/hr
	b) RCS core inlet temperature	539 ≤ T ≤ 547 °F[7]
	c) Upper head temperature	< core outlet temperature
	d) Pressurizer pressure	2,053 ≤ P ≤ 2,172 psia
	e) Pressurizer liquid level	46% ≤ L ≤ 69.2%
	f) SIT pressure	254.2 ≤ P ≤ 289.2 psia
	g) SIT liquid volume	825 ≤ V ≤ 895.5 ft^3
	h) SIT temperature	83.44 ≤ T ≤ 120 °F (coupled to containment temperature)
	i) SIT fL/D	As-built piping configuration
	j) Minimum ECCS boron	≥ 1,900 ppm

[5] Includes a 6.2% local LHR measurement uncertainty, a 0.2% uncertainty due to fuel densification and thermal expansion, a 3% engineering uncertainty and a 1.3% power measurement uncertainty, Reference 5, Section 7.5.

[6] Includes a 6% measurement uncertainty plus a 1.4% control rod insertion effect.

[7] Sampled range of ±4 °F includes both operational tolerance and measurement uncertainty.

Table 3.2 Plant Operating Range Supported by the LOCA Analysis *(Continued)*

	Event	Operating Range
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge piping
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to CL pipe)	$0.05 \leq A \leq 0.5$ full pipe area (split) $0.5 \leq A \leq 1.0$ full pipe area (guillotine)
	d) Worst single-failure	Loss of one ECCS pumped injection train
	e) Offsite power	On or Off
	f) LPSI flow	Minimum flow
	g) HPSI flow	Minimum flow
	h) ECCS pumped injection temperature	105 °F
	i) HPSI delay time	12 (w/ offsite power) 30 seconds (w/o offsite power)
	j) LPSI delay time	12 (w/ offsite power) 30 seconds (w/o offsite power)
	k) Containment pressure	14.2 psia, nominal value
	l) Containment temperature	$83.44 \leq T \leq 120$ °F
	m) Containment spray/fan cooler delays	0/0 seconds

Table 3.3 Statistical Distributions Used for Process Parameters

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution	Standard Deviation
Core Power Operation (%)	Point	100	Normal	0.33
Pressurizer Pressure (psia)	Uniform	2,053 – 2,172	N/A	N/A
Pressurizer Liquid Level (%)	Uniform	46 – 69.2	N/A	N/A
SIT Liquid Volume (ft^3)	Uniform	825 – 895.5	N/A	N/A
SIT Pressure (psia)	Uniform	254.2 – 289.2	N/A	N/A
Containment/SIT Temperature (°F)	Uniform	83.44 – 120	N/A	N/A
Containment Volume[8] ($x10^6$ ft^3)	Uniform	1.02 – 1.16	N/A	N/A
Initial Flow Rate (Mlbm/hr)	Uniform	74 – 81.3	N/A	N/A
Initial Operating Temperature (°F)	Uniform	539 – 547	N/A	N/A
SIRWT Temperature (°F)	Point	105	N/A	N/A
Offsite Power Availability[9]	Binary	0,1	N/A	N/A
Delay for Containment Sprays (s)	Point	0	N/A	N/A
Delay for Containment Fan Coolers (s)	Point	0	N/A	N/A
HPSI Delay (s)	Point	12 (w/ offsite power) 30 (w/o offsite power)	N/A	N/A
LPSI Delay (s)	Point	12 (w/ offsite power) 30 (w/o offsite power)	N/A	N/A

[8] Uniform distribution for parameter with demonstrated PCT importance conservatively produces a wider variation of PCT results relative to a normal distribution. Treatment consistent with approved RLBLOCA evaluation model (Reference 1, Section 4.3.3.2.12).

[9] No data are available to quantify the availability of offsite power. During normal operation, offsite power is available. Since the loss of offsite power is typically more conservative (loss in coolant pump capacity), it is assumed that there is a 50 percent probability the offsite power is unavailable, upon reactor trip.

Table 3.4 SER Conditions and Limitations

SER Conditions and Limitations	Response
1. A CCFL violation warning will be added to alert the analyst to a CCFL violation in the downcomer should such occur.	There was no significant occurrence of CCFL violations in the downcomer for this analysis.
2. FANP has agreed that it is not to use nodalization with hot leg to downcomer nozzle gaps.	Hot leg nozzle gaps were not modeled.
3. If FANP applies the RLBLOCA methodology to plants using a higher planar linear heat generation rate (PLHGR) than used in the current analysis, or if the methodology is to be applied to an end-of-life analysis for which the pin pressure is significantly higher, then the need for a blowdown cladding rupture model will be reevaluated. The evaluation may be based on relevant engineering experience and should be documented in either the RLBLOCA guideline or plant specific calculation file.	The PLHGR for Fort Calhoun Station is lower than that used in the development of the RLBLOCA EM (Reference 1). An end-of-life calculation was not performed; thus, the need for a blowdown cladding rupture model was not reevaluated.
4. Slot breaks on the top of the pipe have not been evaluated. These breaks could cause the loop seals to refill during late reflood and the core to uncover again. These break locations are an oxidation concern as opposed to a PCT concern since the top of the core can remain uncovered for extended periods of time. Should an analysis be performed for a plant with loop seals with bottom elevations that are below the top elevation of the core, FANP will evaluate the effect of the deep loop seal on the slot breaks. The evaluation may be based on relevant engineering experience and should be documented in either the RLBLOCA guideline or plant-specific calculation file.	This is not applicable to the Fort Calhoun Station because it does not have "deep loop seals."
5. The model applies to 3- and 4-loop Westinghouse- and CE-designed nuclear steam systems.	The RLBLOCA evaluation model is applicable to Fort Calhoun Station since it is a CE-designed 2x4-loop plant.
6. The model applies to bottom reflood plants only (cold side injection into the cold legs at the reactor coolant discharge piping).	The RLBLOCA evaluation model is applicable to Fort Calhoun Station plant since it is a bottom reflood plant.
7. The model is valid as long as blowdown quench does not occur. If blowdown quench occurs, additional justification for the blowdown heat transfer model and uncertainty are needed if the calculation is corrected. A blowdown quench is characterized by a temperature reduction of the peak cladding temperature (PCT) node to saturation temperature during the blowdown period.	Examination of the case set showed no evidence of blowdown quench.
8. The reflood model applies to bottom-up quench behavior. If a top-down quench occurs, the model is to be justified or corrected to remove top quench. A top-down quench is characterized by the quench front moving from the top to the bottom of the hot assembly.	Examination of the case set showed that core quench initiated at the bottom of the core and proceeded upward.

Table 3.4 SER Conditions and Limitations *(Continued)*

SER Conditions and Limitations	Response
9. The model does not determine whether Criterion 5 of 10CFR50.46, long-term cooling, has been satisfied. This will be determined by each applicant or licensee as part of its application of this methodology.	Long-term cooling was not evaluated herein.
10. Specific guidelines must be used to develop the plant-specific nodalization. Deviations from the reference plant must be addressed.	The Fort Calhoun Station model nodalization is consistent with the sample calculations given in the RLBLOCA evaluation model (Reference 1). Figure 3.1 shows the loop noding used in the analysis. Figure 3.2 shows the steam generator model. Figures 3.3, 3.4 and 3.5 show RV noding diagrams.
11. A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical report approval process must be provided. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.	Table 3.7 presents the summary of the full range of applicability for the important heat transfer correlations, as well as the ranges calculated in the limiting analysis case. Calculated values for other parameters of interest are also provided. As is evident, the plant-specific parameters fall within the applicability range of the methodology.
12. The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses, including the calculated worst break size, PCT and local and total oxidation.	Analysis results are presented in Section 3.6.
13. Applicants or licensees wishing to apply the Framatome ANP realistic large break loss-of-coolant accident (RLBLOCA) methodology to M5 clad fuel must request an exemption for its use until the planned rulemaking to modify 10CFR50.46(a)(i) to include M5 cladding material has been completed.	OPPD understands that an exemption request is required for the use of M5® cladding.

Table 3.5 Summary of Hot Rod Limiting PCT Results

Fuel Product	FANP Advanced CE14 HTP
Case Number	10
PCT	
Temperature	1,675 °F
Time	26.9 s
Elevation	8.597 ft
Metal-Water Reaction	
Oxidation Maximum	0.82%
Total Oxidation	0.02%

Table 3.6 Calculated Event Times for the Limiting PCT Case

Event	Time (sec)
Break Opened	0
RCP Trip	0
SIAS Occurs	0.7
Start of Broken Loop SIT Injection	13.4
Start of Intact Loop SIT Injection	16.6, 16.6, 16.6
Beginning of Core Recovery (Beginning of Reflood)	26.5
PCT Occurred	26.9
LPSI Available	30.7
Start of HPSI	30.7
Broken LPSI Delivery Began	30.7
Intact Loop LPSI Delivery Began (loops 1B, 2A and 2B, respectively)	N/A, N/A, 30.7
Broken HPSI Delivery Began	30.7
Intact Loop HPSI Delivery Began (loops 1B, 2A and 2B, respectively)	30.7, 30.7, 30.7
Broken Loop SIT Emptied	52.5
Intact Loop SIT Emptied (loops 1B, 2A and 2B, respectively)	51.5, 54.4, 54.1
Transient Calculation Terminated	381.3

Table 3.7 Heat Transfer Parameters for the Limiting Case

Figure 3.1 Primary System Noding

Figure 3.2 Secondary System Noding

Figure 3.3 Reactor Vessel Noding

Figure 3.4 Core Noding Detail

Figure 3.5 Upper Plenum Noding Detail

Figure 3.6 S-RELAP5 Containment Pressure versus Best-Estimate Result

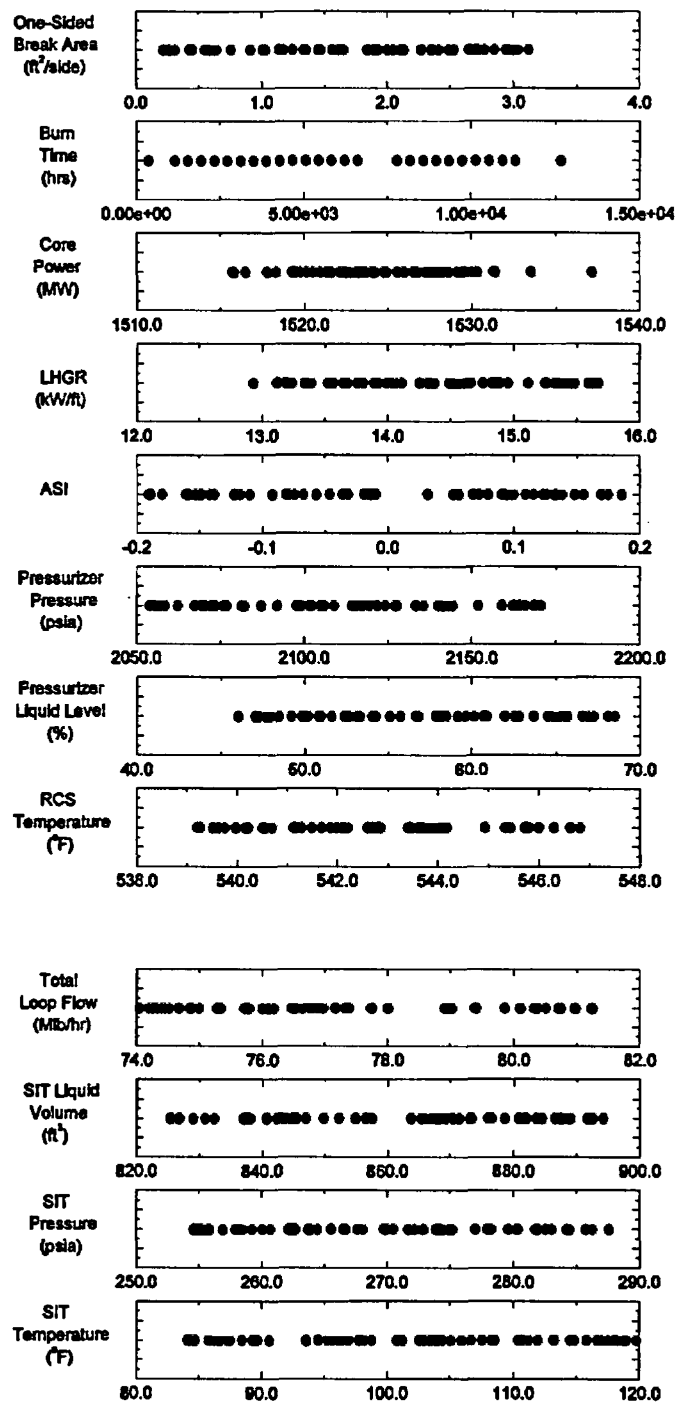

Figure 3.7 Scatter Plot of Operational Parameters, for all Cases

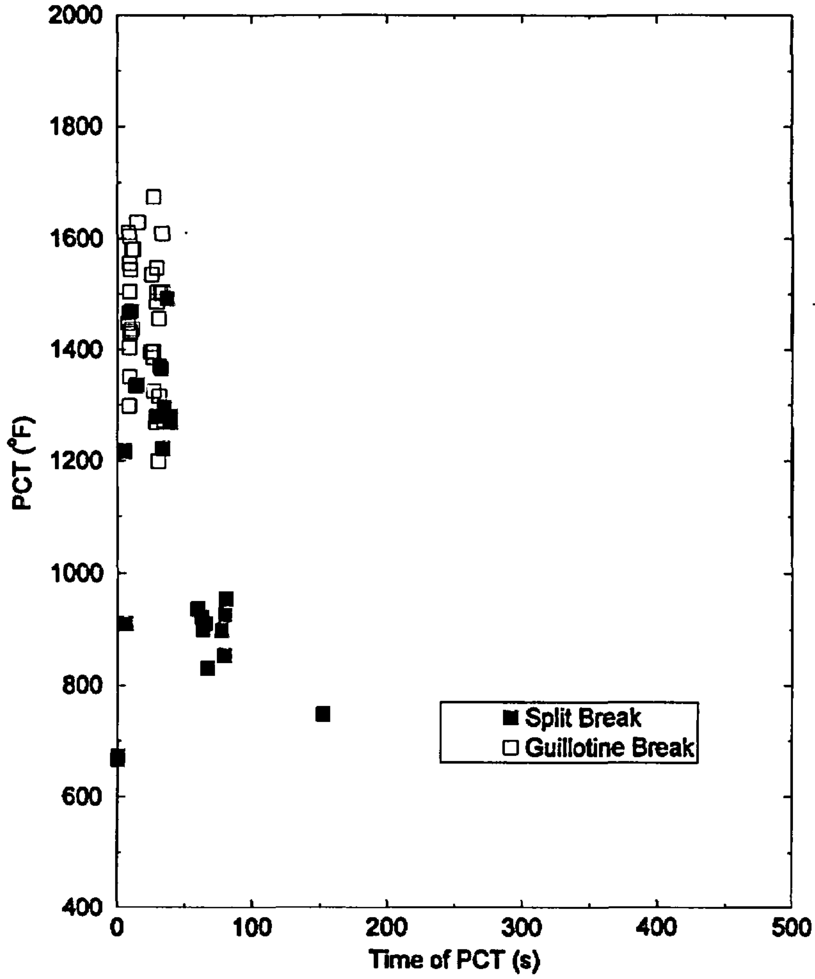

Figure 3.8 PCT versus PCT Time Scatter Plot from 59 Calculations

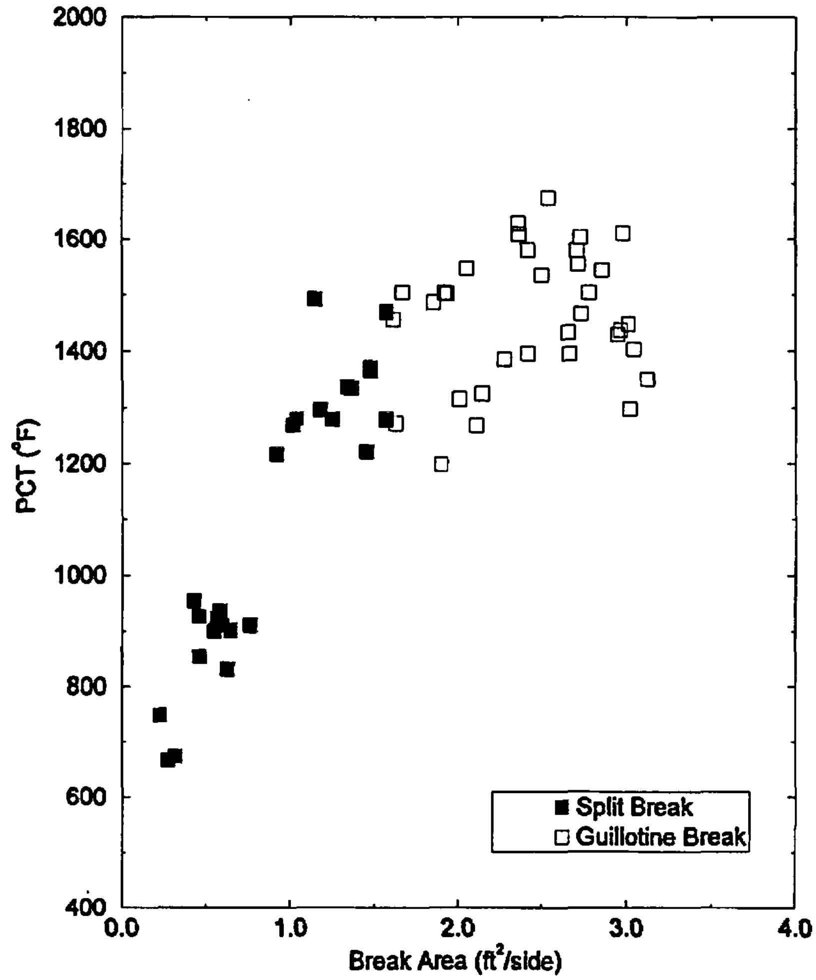

Figure 3.9 PCT versus Break Size Scatter Plot from 59 Calculations

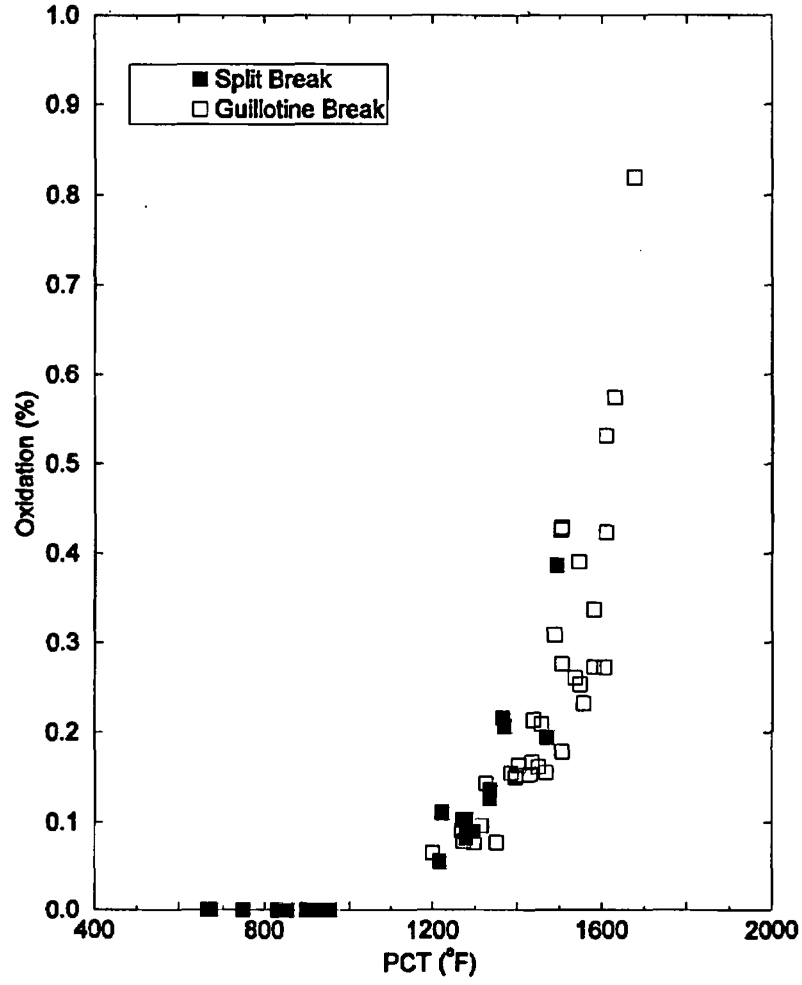

Figure 3.10 Maximum Oxidation versus PCT Scatter Plot from 59 Calculations

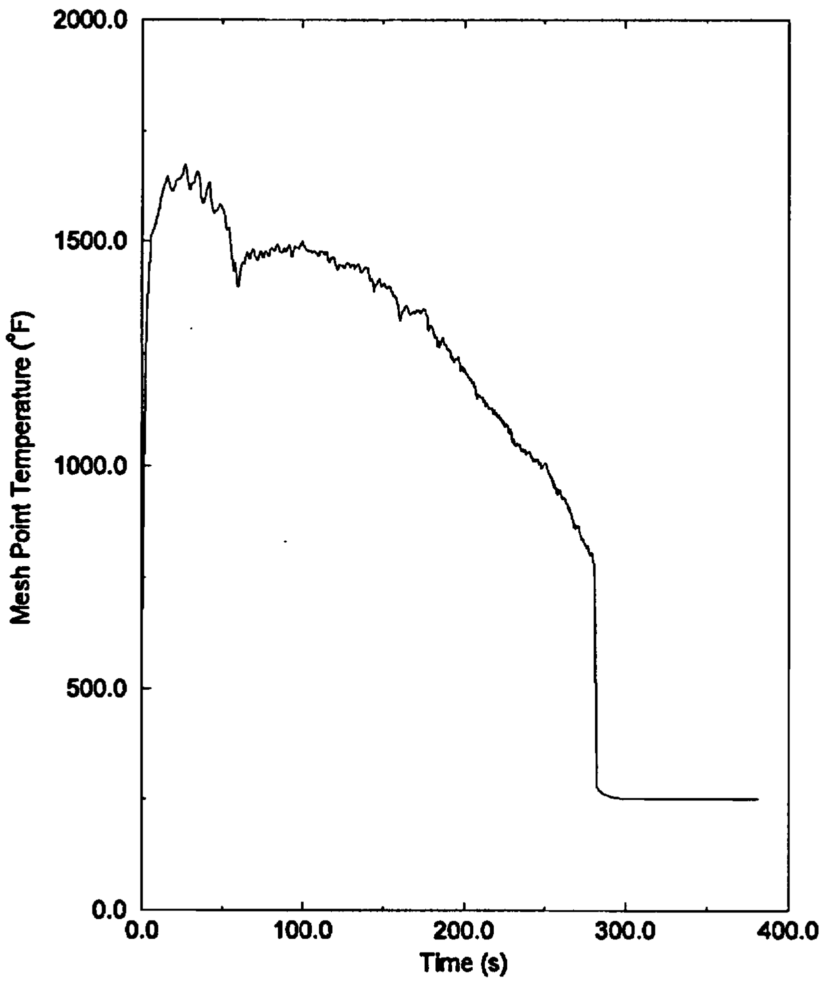

Figure 3.11 Peak Cladding Temperature (Independent of Elevation) for the Limiting Case

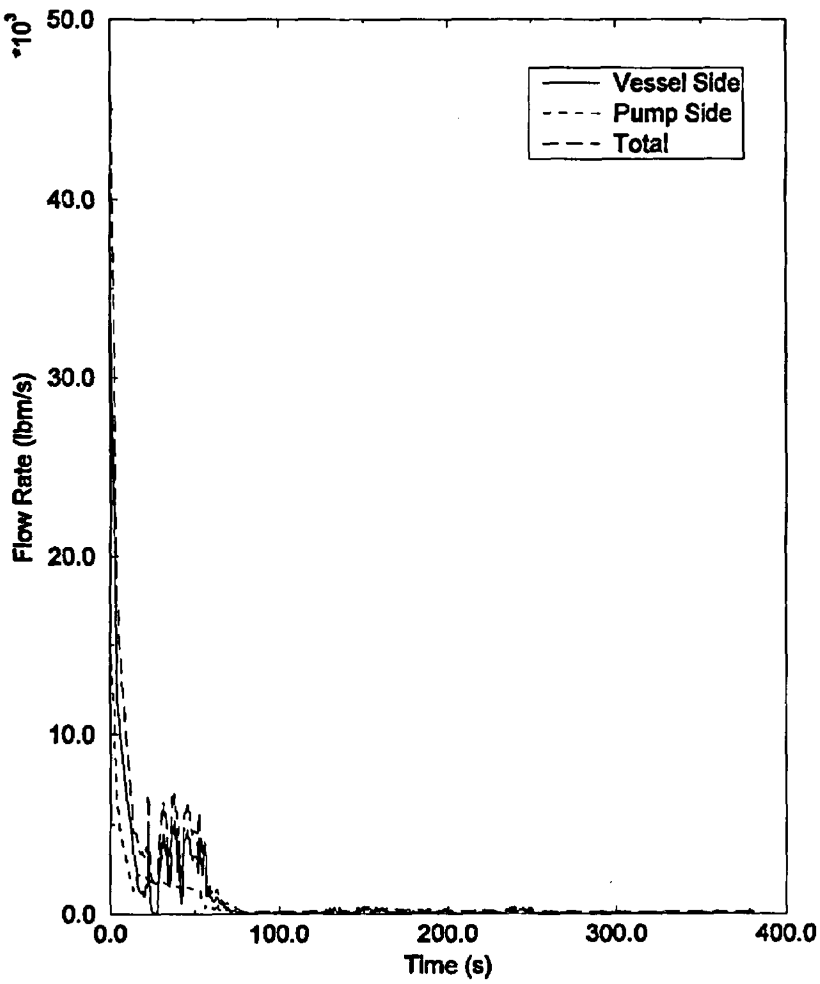

Figure 3.12 Break Flow for the Limiting Case

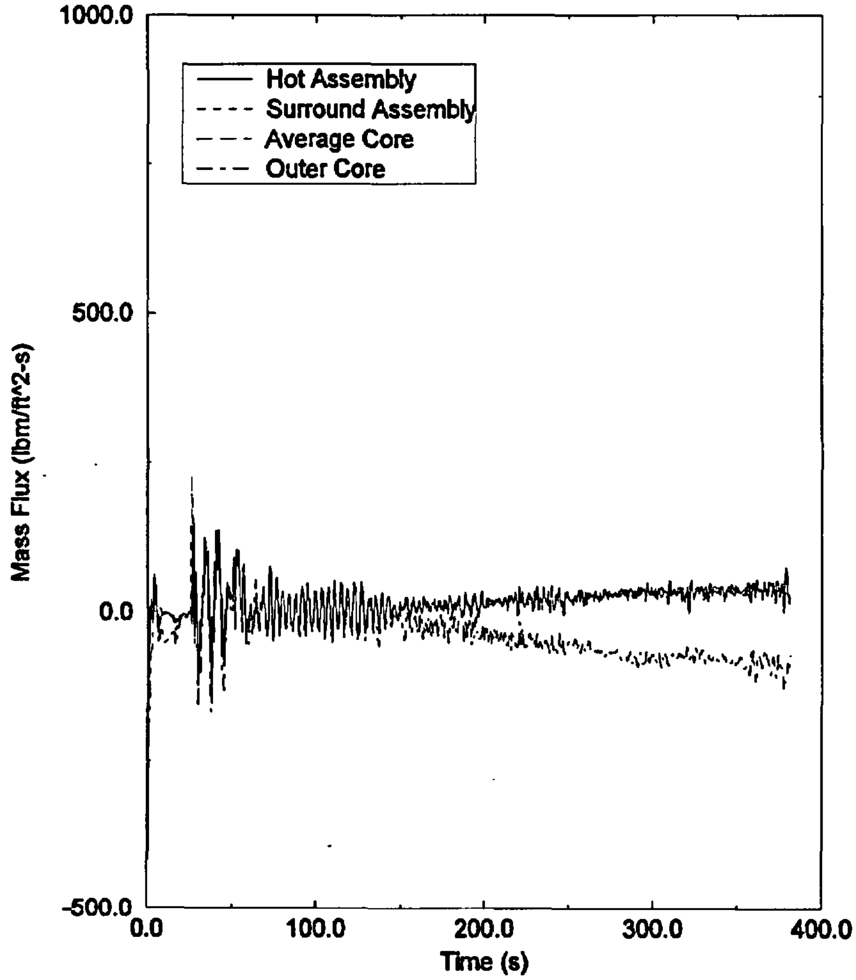

Figure 3.13 Core Inlet Mass Flux for the Limiting Case

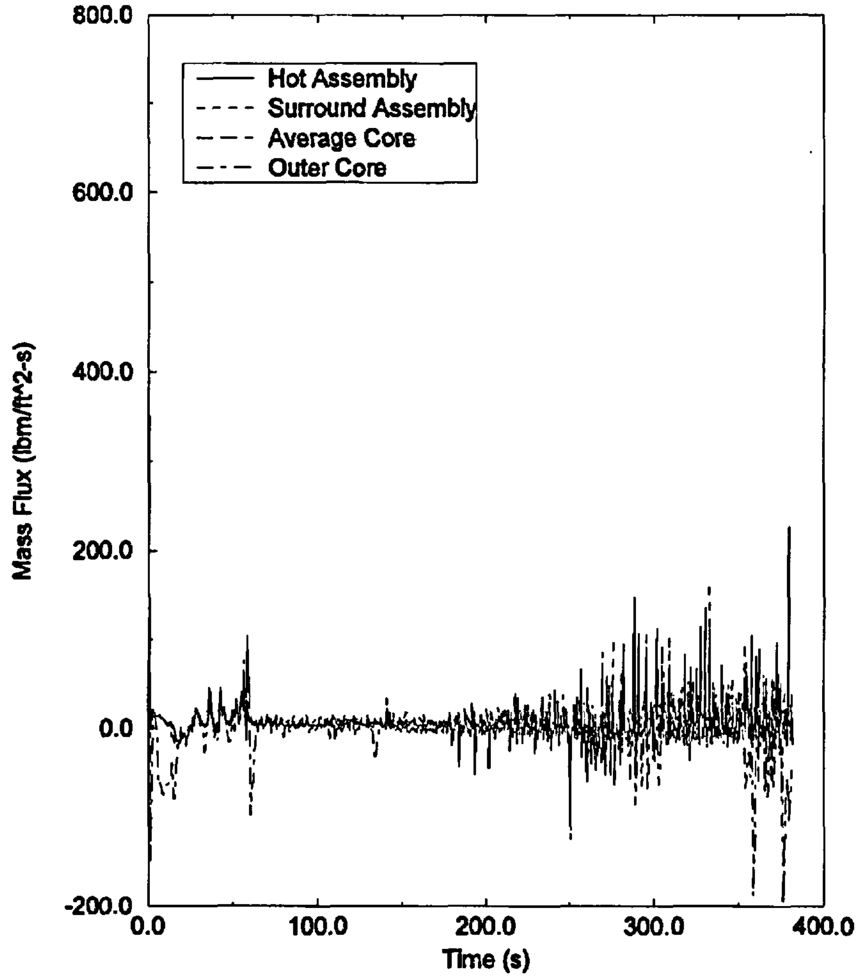

Figure 3.14 Core Outlet Mass Flux for the Limiting Case

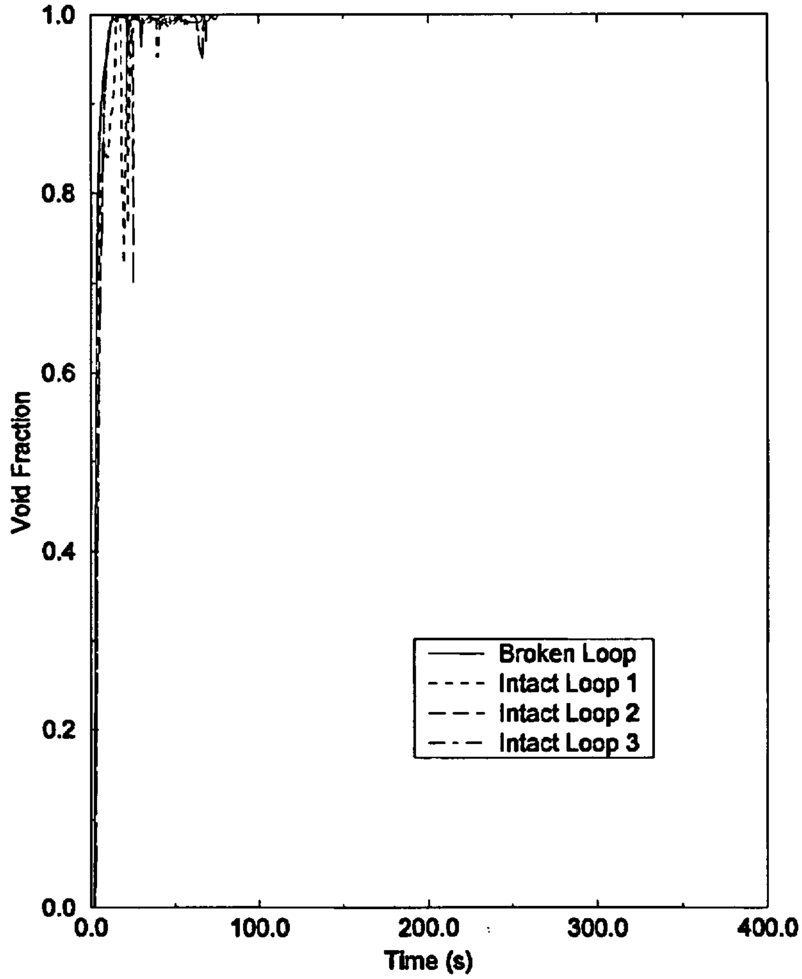

Figure 3.15 Void Fraction at RCS Pumps for the Limiting Case

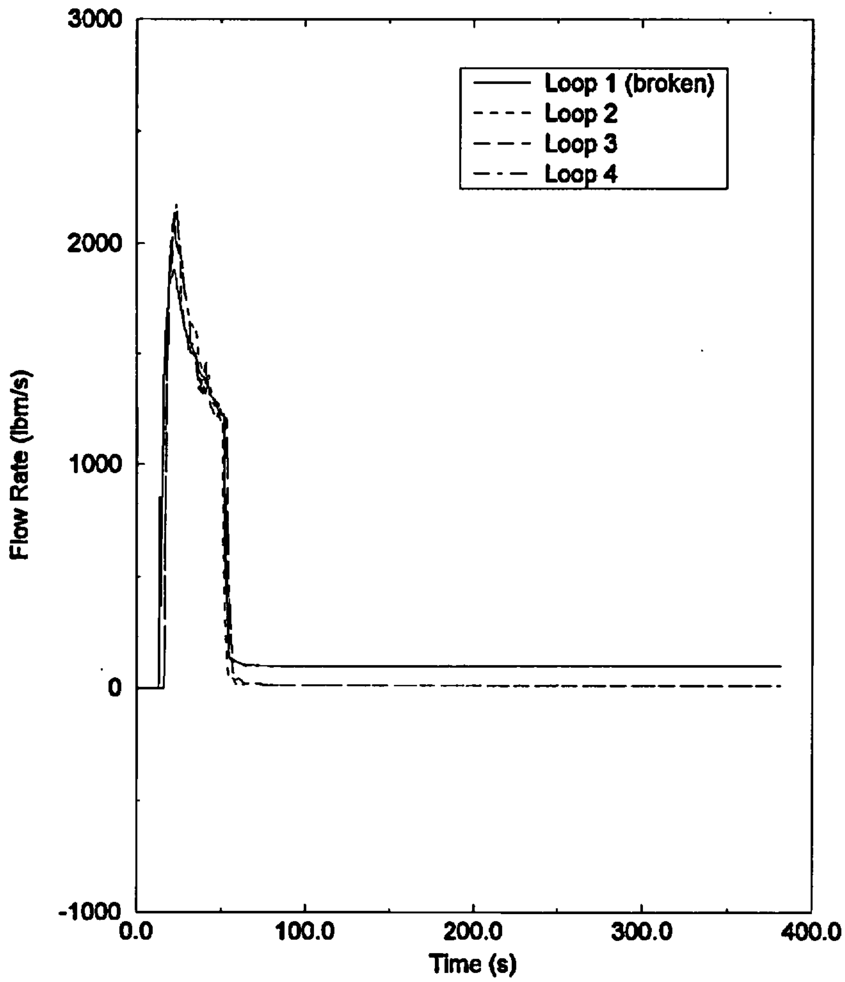

Figure 3.16 ECCS Flows (Includes SIT, HPSI and LPSI) for the Limiting Case

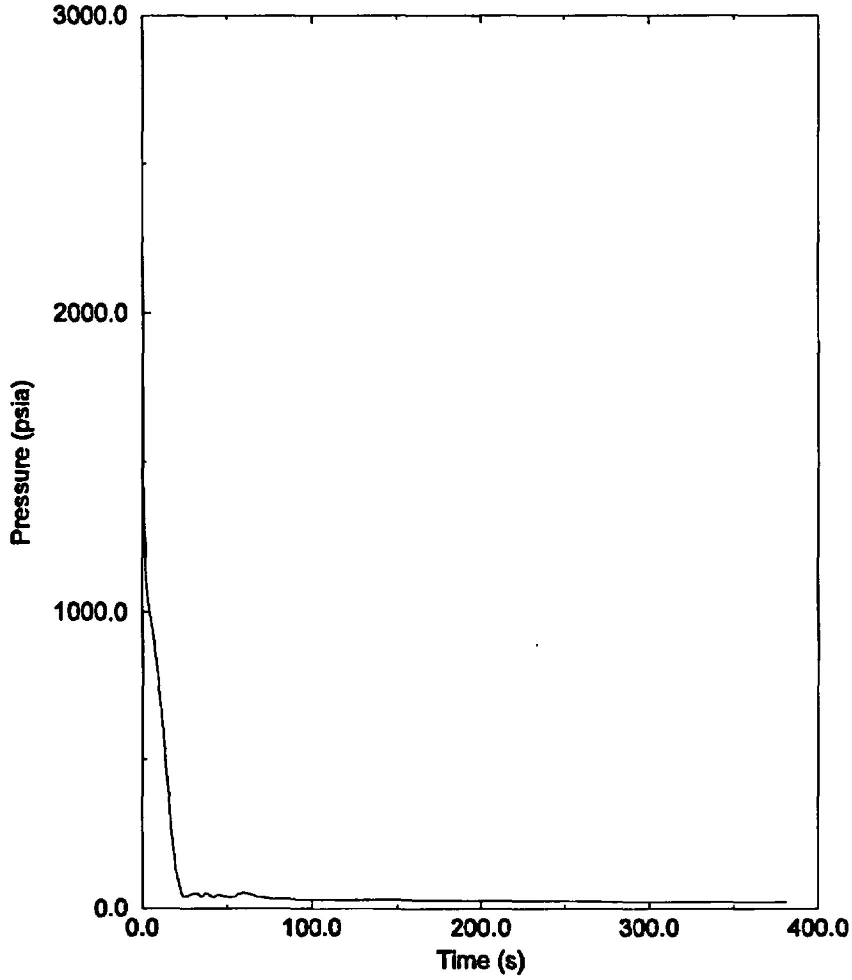

Figure 3.17 Upper Plenum Pressure for the Limiting Case

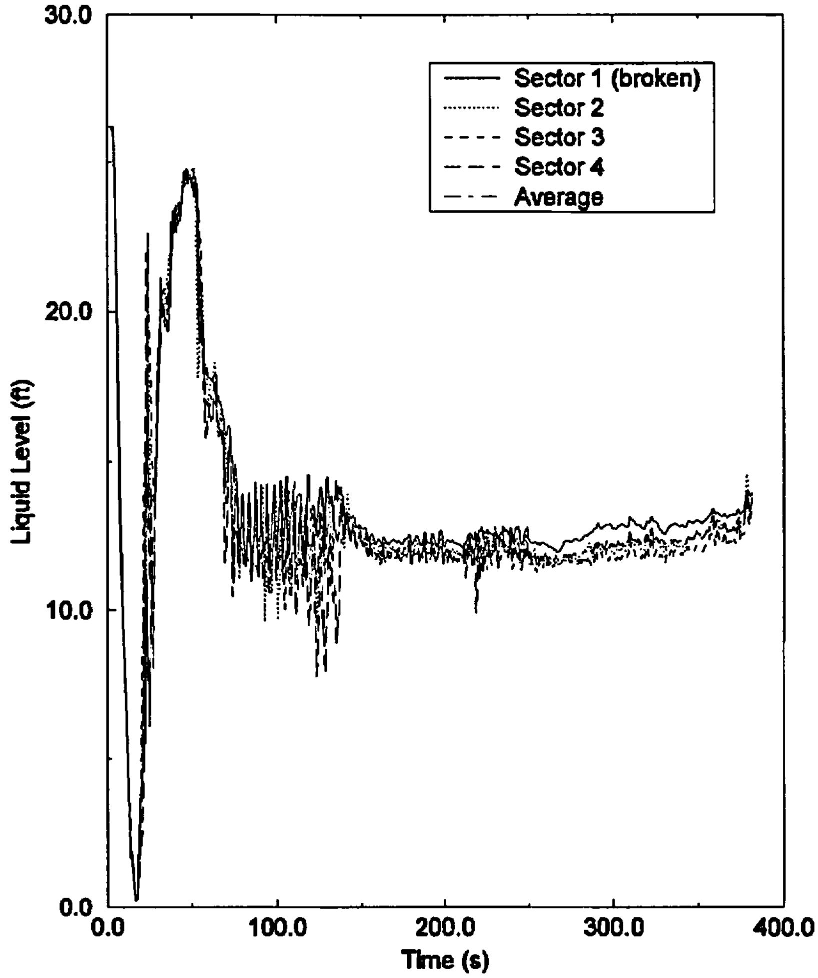

Figure 3.18 Collapsed Liquid Level in the Downcomer for the Limiting Case

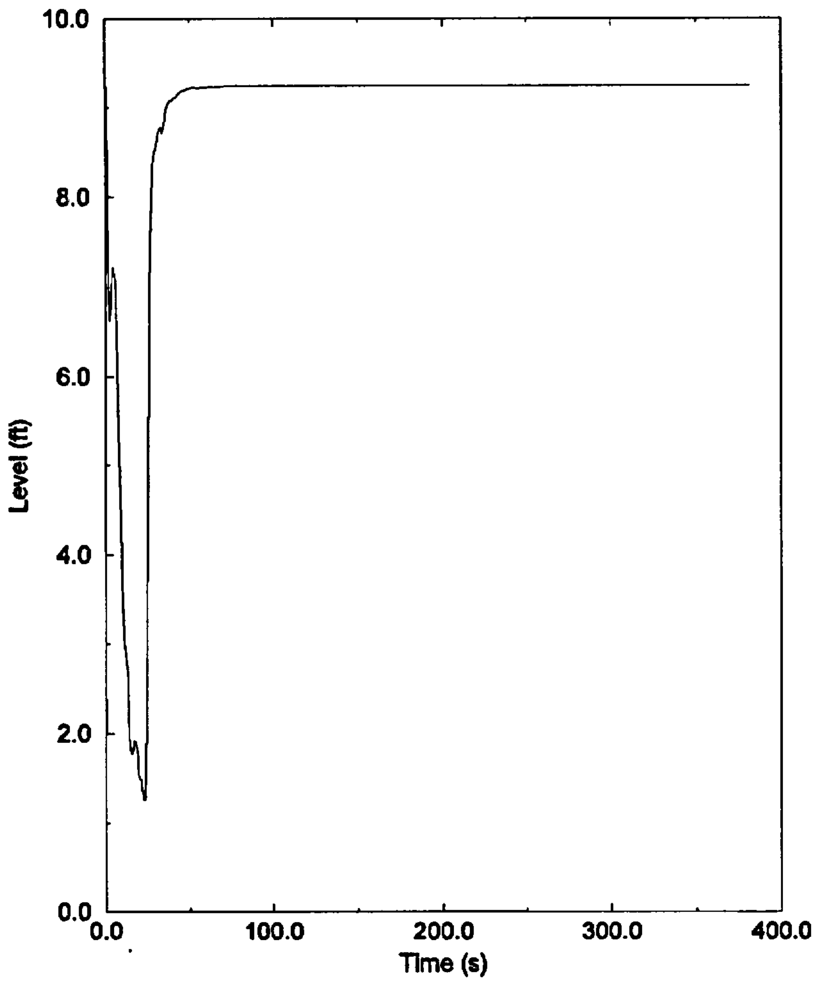

Figure 3.19 Collapsed Liquid Level in the Lower Plenum for the Limiting Case

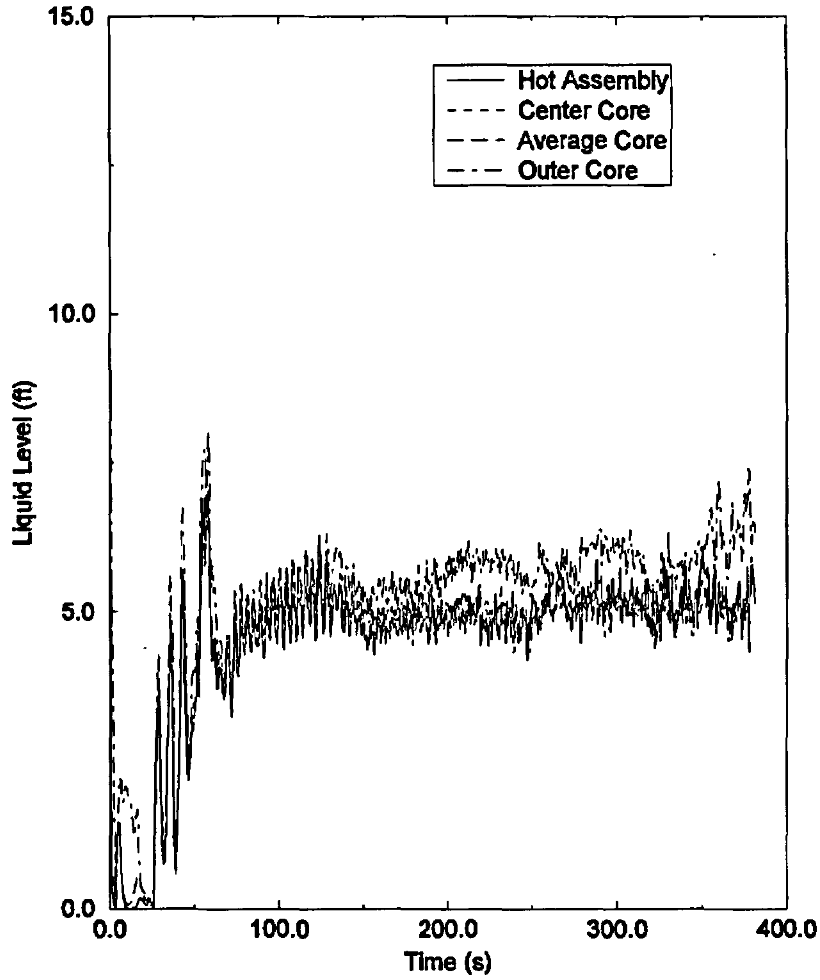

Figure 3.20 Collapsed Liquid Level In the Core for the Limiting Case

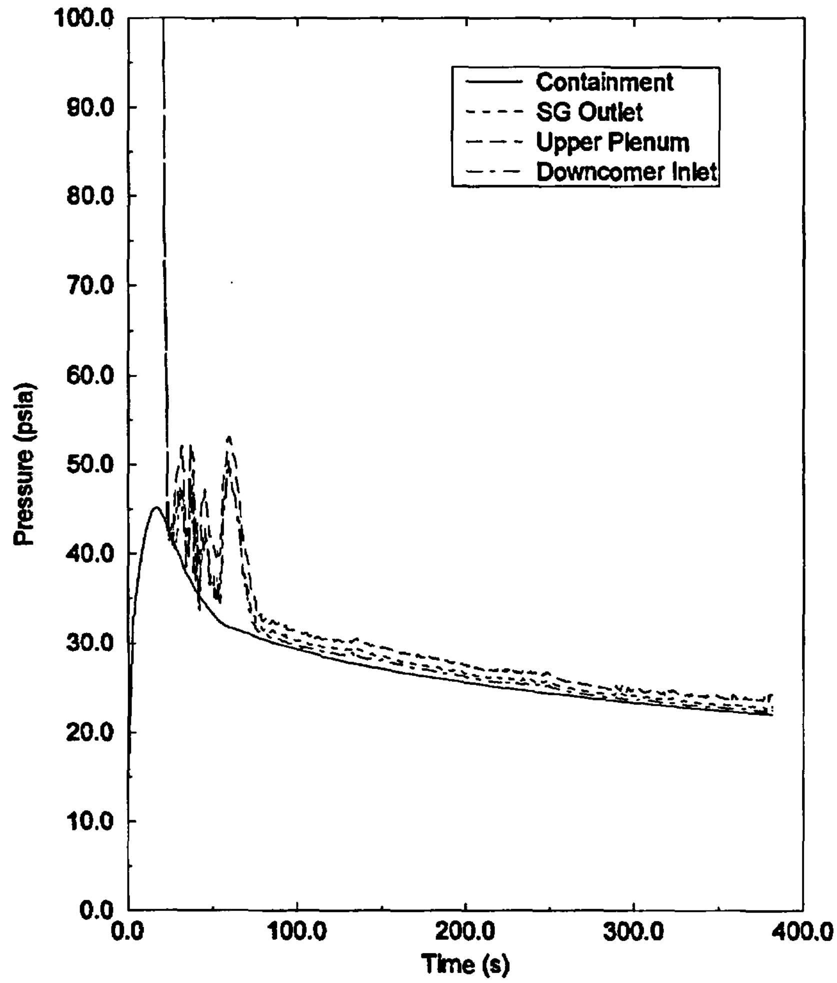

Figure 3.21 Containment and Loop Pressures for the Limiting Case

4.0 Conclusions

An RLBLOCA analysis was performed for the Fort Calhoun Station nuclear power plant using NRC-approved FANP RLBLOCA methods (Reference 1). Analysis results show that the limiting FANP fuel case has a PCT of 1,675 °F, and a maximum oxidation thickness and hydrogen generation that fall well within regulatory requirements. Mixed-core effects are a non-issue since the core is completely fueled with thermal-hydraulic compatible 14x14 FANP fuel assemblies.

The analysis supports operation at a nominal power level of 1,525 MWt (plus uncertainty), a steam generator tube plugging level of up to 10 percent in both steam generators, a linear heat rate of 15.5 kW/ft, an F_r^T of 1.86 with no axially-dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. No axial peaking reduction is imposed on the FANP fuel. For large break LOCA, all 10CFR50.46(b) criteria presented in Section 3.0 are met and operation of Fort Calhoun Station with FANP-supplied Advanced CE14 HTP M5® clad fuel is justified.

5.0 References

1. AREVA/FANP Document, EMF-2103(P)(A) Revision 0, *Realistic Large Break LOCA Methodology*, Framatome ANP, Inc., April 2003.

2. Technical Program Group, *Quantifying Reactor Safety Margins*, NUREG/CR-5249, EGG-2552, October 1989.

3. Wheat, Larry L., "*CONTEMPT-LT A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-Of-Coolant-Accident*," Aerojet Nuclear Company, TID-4500, ANCR-1219, June 1975.

4. U. S. Nuclear Regulatory Commission, NUREG-0800, Revision 2, Standard Review Plan, July 1981.

5. Fort Calhoun Station Updated Final Safety Analysis Report, Revision 14, September 14, 2004.