

**St. Lucie Unit 1
Extended Power Uprate
Licensing Report**

**Attachment 5
Appendix C**

**St. Lucie Nuclear Plant Unit 1
EPU Cycle
Realistic Large Break LOCA Summary Report
With Zr-4 Fuel Cladding**

ANP-2903(NP) Revision 000

Areva NP Inc.

ANP-2903(NP)
Revision 000

St Lucie Nuclear Plant Unit 1 EPU Cycle
Realistic Large Break LOCA Summary Report with Zr-4
Fuel Cladding

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Nature of Changes

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

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Nomenclature

CCTF	Cylindrical Core Test Facility
CE	Combustion Engineering Inc.
CFR	Code of Federal Regulations
CSAU	Code Scaling, Applicability, and Uncertainty
DC	Downcomer
DEGB	Double-Ended Guillotine Break
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPH	Effective Full Power Hours
EPU	Extend Power Uprate
EM	Evaluation Model
F_Q	Total Peaking Factor
FP&L	Florida Power and Light Company
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor
HPSI	High Pressure Safety Injection
HFP	Hot Full Power
LANL	Los Alamos National Laboratory
LHR	Linear Heat Rate
RLBLOCA	Realistic Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MSIV	Main Steam Isolation Valve
NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PLHGR	Planar Linear Heat Generation Rate
PWR Pressurized	Water Reactor
RAS Recirculation	Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RV Reactor	Vessel
RWST	Refueling Water Storage Tank
SI Safety	Injection
SIAS	Safety Injection Activation Signal
SIT	Safety Injection Tank
SER	Safety Evaluation Report

1.0 Introduction

This report describes and provides results from a RLBL OCA analysis for the St Lucie Nuclear Plant Unit 1 Extended Power Uprate. The plant is a CE-designed 3020 MWt plant with a large dry containment. AREVA NP is the current fuel supplier. The plant is a 2 X4 loop design – two hot legs and four cold legs. The loops contain four RCPs, two U-tube steam generators and one pressurizer. The ECCS is provided by two independent safety injection trains and four SITs.

The analysis supports operation for EPU Cycle and beyond with AREVA NP's HTP 14X 14 fuel design using standard UO_2 fuel with 2%, 4%, 6% and 8% Gd_2O_3 and ZIRCALOY-4 cladding, unless changes in the Technical Specifications, Core Operating Limits Report, core design, fuel design, plant hardware, or plant operation invalidate the results presented herein. The analysis was performed in compliance with the NRC-approved RLBOCA EM (Reference 1) with exceptions noted below. Analysis results confirm the 10CFR50.46 (b) acceptance criteria presented in Section 3.0 are met and serve as the basis for operation of the St Lucie Nuclear Plant Unit 1 with AREVA NP fuel. Per RLBOCA EM (Reference 1), fuel assemblies residing in the core for more than one cycle will not be limiting. Therefore, this RLBOCA analysis covers the transition cycle with both fresh fuel and burned fuel.

The non-parametric statistical methods inherent in the AREVA NP RLBOCA methodology provide for the consideration of a full spectrum of break sizes, break configuration (guillotine or split break), axial shapes, and plant operational parameters. A conservative loss of a diesel assumption is applied in which LPSI inject into the broken loop and on one intact loop and HPSI inject into all four loops. 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.

The following are deviations from the approved RLBL OCA EM (Reference 1) that were requested by the NRC.

The assumed reactor core power for the St Lucie Unit 1 realistic large break loss-of-coolant accident is 3029 MWt. This value represents the 10% power uprate and 1.7% measurement uncertainty recapture (MUR) relative to the current rated thermal power of 2700 MWt plus 0.3% power measurement uncertainty. $(2700 \text{ MWt} \times (1+10\%) \times (1+1.7\%) \times (1+0.3\%)) = 3029 \text{ MWt}$

The RLBLOCA analysis was performed with a version of S-RELAP5 that requires both the void fraction to be less than 0.95 and the clad temperature to be less than 900 °F before the rod is allowed to quench. This may result in a slight increase in PCT results when compared to an analysis not subject to these constraints.

The RLBLOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than 15 percent of the total heat transfer at and above a void fraction of 0.9. This may result in a slight increase in PCT results when compared to previous analyses for similar plants.

The split versus double-ended break type is no longer related to break area. In concurrence with Regulatory Guide 1.157, both the split and the double-ended break will range in area between the minimum break area (A_{min}) and an area of twice the size of the broken pipe. The determination of break configuration, split versus double-ended, will be made after the break area is selected based on a uniform probability for each occurrence. A_{min} was calculated to be 26.7 percent of the DEGB area (see Section 4.6 for further discussion). This is not expected to have an effect on PCT results.

In concurrence with the NRC's interpretation of GDC 35, a set of 59 cases was run with a LOOP assumption and a second set with a No-LOOP assumption. The set of 59 cases that predicted the highest PCT is reported in Section 2 and Section 3, herein. The results from both case sets are shown in Figure 3-22. The effect on PCT results is expected to be minor.

During recent RLBLOCA EM modeling studies, it was noted that cold leg condensation efficiency may be under-predicted. Water entering the DC post-accumulator injection remained sufficiently subcooled to absorb DC wall heat release without significant boiling. However, tests (Reference 7) indicate that the steam and water entering the DC from the cold leg, subsequent to the end of accumulator injection, reach near saturation resulting from the condensation efficiency ranging between 80 to 100 percent. To assure that cold leg condensation would not be under-predicted, a RLBLOCA EM update was made. Noting that saturated fluid entering the DC is the most conservative modeling scheme, steam and liquid multipliers were developed so as to approximately saturate the cold leg fluid before it enters the DC. The multipliers were developed through scoping studies using a number of plant configurations—Westinghouse-designed 3- and 4-loop plants, and CE-designed plants. The results of the scoping study indicated that multipliers of 10 and 150 for liquid and steam, respectively, were appropriate to

produce saturated fluid entering the DC. This RLBLOCA EM departure was recently discussed with the NRC and the NRC agreed that the approach described immediately above was satisfactory in the interim. The modification is implemented post-accumulation injection, 10 seconds after the vapor void fraction in the bottom of the accumulator becomes greater than 90 percent. Thus, the accumulators have injected all their water into the cold legs, and the nitrogen cover gas has entered the system and been mostly discharged through the break before the condensation efficiency is increased by the factors of 10 and 150, for liquid and vapor respectively. Providing saturated fluid conditions at the DC entrance conservatively reduces both the DC driving head and the core flooding rate. Recall that test results indicate that fluid conditions entering the DC range from saturated to slightly subcooled. Hence, it is conservative to force an approximation of saturated conditions for fluid entering the DC.

AREVA Inc. has acknowledged an issue concerning fuel thermal conductivity degradation as a function of burnup as raised by the NRC. In order to manage this issue, AREVA Inc. is modifying the way RODEX3A temperatures are compensated in the RLBLOCA Revision 0/Transition package methodology. In the current process, the RLBLOCA computes PCTs at many different times during an operating cycle. For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then two-transformation adjustments to the base fuel centerline temperature are computed. The first transformation is a linear adjustment for an exposure of 10 Mwd/MtU or higher. The second adjustment is performed in the S-RELAP5 initialization process for the transient case. In the new process, a polynomial transformation is used for the first transformation instead of a linear transformation. The rest of the RLBLOCA process for initializing the S-RELAP5 fuel rod temperature should not be altered and the rest of LOCA transient should also continue in the original fashion.

2.0 Summary

The limiting PCT analysis is based on the parameter specification given in Table 2-1. The limiting PCT is 1672 °F for a 6% Gd₂O₃ Rod in a case with LOOP conditions (LOOP is loss of offsite power. No-LOOP is with offsite power available). UO₂ rods and Gadolinia bearing rods of 2, 4 and 8% were also analyzed, but, were not found to be limiting. This RLBLOCA result is based on a case set of 59 individual transient cases for LOOP and 59 individual transient cases for No-LOOP conditions. The core is composed only of AREVA NP HTP 14x14 thermal hydraulically compatible fuel designs; hence, there is no mixed core consideration.

The analysis assumed full core power operation at 3029 MWt. The value represents the 10 % power uprate and 1.7 % measurement uncertainty recapture (MUR) relative to the current rated thermal power (2700 MWt) plus 0.3% power measurement uncertainty. The analysis assumed a steam generator tube plugging level of 10 percent in all steam generators, a total of LHR of 15.0 kW/ft (no axial dependence), a total peaking factor (F_Q) up to a value of 2.161, and a nuclear enthalpy rise factor ($F_{\Delta H}$) up to a value of 1.749 (including 6% uncertainty). This analysis bounds typical operational ranges or technical specification limits (whichever is applicable) with regard to Pressurizer pressure and level; SIT pressure, temperature, and level; core average temperature; core flow; containment pressure and temperature; and RWST.

The AREVA RLBLOCA methodology explicitly analyzes only fresh fuel assemblies (see Reference 1, Appendix B). Previous analyses have shown that once- and twice-burnt fuel will not be limiting up to peak rod average exposures of 62,000 MWd/MTU. The analysis demonstrates that the 10 CFR 50.46(b) criteria listed in Section 3.0 are satisfied.

Table 2-1 Summary of Major Parameters for Limiting Transient

Core Average Burnup (EFPD)	6874.59
Core Power (MWt)	3029.06
Hot Rod LHR, kW/ft	14.6990
Total Hot Rod Radial Peak (F_r^T)	1.810
ASI	0.0393
Break Type	Guillotine
Break Size (ft ² /side)	4.1705
Offsite Power Availability	Not available
Decay Heat Multiplier	0.98429

3.0 Analysis

The purpose of the analysis is to verify typical technical specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10CFR 50.46(b) criteria are met:

- (1) The calculated maximum fuel element cladding temperature shall not exceed 2200 °F.
- (2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- (3) 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.
- (4) The calculated changes in core geometry shall be such that the core remains amenable to cooling.
- (5) Long-term cooling is demonstrated outside of this report for operation at EPU conditions.

The analysis did not evaluate core coolability due to seismic events, nor did it consider the 10CFR 50.46(b) long-term cooling criterion.

The RLBLOCA analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. AREVA NP has previously performed an analysis which demonstrates that for all cases of horizontal seismic and LOCA loads, the resulting loads are below the spacer grid elastic load limit and thus the grids sustain no permanent deformation.

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 to the SER is addressed in Section 3.4. Section 3.5 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 primary piping. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel 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 cold leg break is assumed to open instantaneously. For this break, a rapid 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 criticality ends. As heat transfer from the fuel rods is reduced, the cladding temperature increases.

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 leads to a period of positive core flow or reduced downflow as the RCPS in the intact loops continue to supply water to the RV (in No-LOOP conditions). Cladding temperatures may be reduced and some portions of the core may rewet during this period. The positive core flow or reduced downflow period ends as two-phase conditions occur in the RCPS, 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 the SIAS is issued. This signal is initiated by either high containment pressure or low Pressurizer pressure. Regulations require that a worst single-failure be considered. This single-failure has been determined to be the loss of one ECCS pumped injection train. The AREVA RLBLOCA methodology conservatively assumes an on-time start and normal lineups of the containment spray to conservatively reduce containment pressure and increase break flow. Hence, the analysis assumes that the loss of one emergency diesel generator, which takes one train of ECCS pumped injection out. LPSI inject into the broken loop and one intact loop, HPSI inject into all four loops, and all containment spray pumps are operating.

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 drive 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; thus, core heat transfer improves and cladding temperatures decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery continues relying solely on pumped ECCS 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 has been 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 more energy is removed from the core by the HPSI and LPSI while 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 generators, and the reactor coolant pumps before it is vented out the break. Some steam flow to the upper head and pass through the spray nozzles, which provide a vent path to the break. The resistance of this flow path to the steam flow is balanced by the driving force of water filling the downcomer. This resistance may act to retard the progression of the core reflood and postpone core-wide cooling. Eventually (within a few minutes of the accident), the core reflood 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 LPSI pumped injection system.

3.2 Description of Analytical Models

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

The RLBLOCA methodology consists of 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 (includes ICECON for containment response).
- AUTORLBLOCA for generation of ranged parameter values, transient input, transient runs, and general output documentation.

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 the other 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 phenomena expected during the LBLOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the RCPs or the steam generator separators. All geometries are modeled at the resolution 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 RODEX 3A 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. Containment pressure is also calculated by S-RELAP5 using containment models derived from ICECON (Reference 4), which is based on the CONTEMPT-LT code (Reference 3).

The methods used in the application of S-RELAP5 to the LBLOCA 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 St Lucie Unit 1 specific inputs. 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, base RODEX3A and S-RELAP5 input files for the plant (including the containment input file) are developed. Code input development guidelines are applied to ensure that model nodalization is consistent with the model nodalization 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. Total oxidation and total hydrogen are based on the limiting PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the criteria set forth in Section 3.0.

3.3 Plant Description and Summary of Analysis Parameters

The plant analysis presented in this report is for a CE-designed PWR, which has 2X 4-loop arrangement. There are two hot legs each with a U-tube steam generator and four cold legs each with a RCP¹. The RCS includes one Pressurizer connected to a hot leg. The core contains 217 thermal-hydraulic compatible AREVA HTP 14X14 fuel assemblies with 2%, 4%, 6% and 8% gadolinia pins. The ECCS includes one HPSI, one LPSI and one SIT injection path per RCS loop. The break is modeled in 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., RAS) for ECCS pumped injection need not be considered.

The S-RELAP5 model explicitly describes the RCS, RV, Pressurizer, and ECCS. The ECCS includes a SIT path and a LPSI/ HPSI path per RCS loop. The HPSI and LPSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary-side steam generator that is instantaneously isolated (closed MSIV and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 10 percent per steam generator was assumed.

As described in the AREVA RLBLOCA methodology, many parameters associated with RLBLOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of those parameters is given in Table 3-1. The RLBLOCA 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 uncertainties for the process parameters sampled in the analysis. Table 3-3 presents a summary of the uncertainties used in the analysis. Where applicable, the sampled parameter ranges are based on technical specification limits or supporting plant calculations that provide more bounding values.

For the AREVA NP RLBLOCA EM, dominant 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)

¹ The RCPs are Byron-Jackson Type DFSS pumps are specified by FP&L. The homologous pump performance curves were input to the S-RELAP5 plant model; the built-in S-RELAP5 curves were not used.

Containment parameters—containment pressure and temperature. In many instances, the conservative guidance of CSB 6-2 (Reference 5) 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 response is indirectly ranged by sampling the containment volume (Table 3-3). Containment heat sink data is given in Table 3-9. In accordance with Reference 1, the condensing heat transfer coefficient is intended to be closer to a best-estimate instead of a bounding high value. A [] Uchida heat transfer coefficient multiplier was specifically validated for use in St Lucie through application of the process used in the RLBLOCA EM (Reference 1) sample problems.

The initial conditions and boundary conditions are given in Table 3-8. The building spray is modeled at maximum heat removal capacity. All spray flow is delivered to the containment.

3.4 SER Compliance

A number of requirements on the methodology are stipulated in the conclusions section of the SER for the RLBLOCA methodology (Reference 1). These requirements have all been fulfilled during the application of the methodology as addressed in Table 3-4.

3.4.1 Item 7: Blowdown Quench

One case was potential candidate for blowdown quench and was closely inspected. For this calculation, no evidence of blowdown quench was observed. Therefore, compliance to the SER restriction has been demonstrated.

3.4.2 Item 8: Top-down Quench

Several provisions have been implemented in the S-REL AP5 model to prevent the top-down quench. The upper plenum nodalization features include:

- the homogeneous option is selected for the junction that connects the first axial level node above the hot channel to the second axial level node above the hot channel;
- no cross-flow is allowed between the first axial level Upper Plenum nodes above the hot channel to the average channel;
- the CCFL model is applied on all core exit junctions.

Six cases were closely examined for top-down quench. No evidence of top-down quench was observed. Therefore, compliance to the SER restriction has been demonstrated.

3.5 Realistic Large Break LOCA Results

Two case sets of 59 transient calculations were performed sampling the parameters listed in Table 3-1. For each case set, PCT was calculated for a UO_2 rod and for Gadolinia-bearing rods with concentrations of 2, 4, 6 and 8 w/o Gd_2O_3 . The limiting case set, that contained the PCT, was the set with no offsite power available. The limiting PCT (1672 °F) occurred in Case 3 for a 6% Gd_2O_3 rod. The major parameters for the limiting transient are presented in Table 2-1. Table 3-5 lists the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1 percent limit. The best-estimate PCT case is Case 17, which corresponded to the median case out of the 59-case set with no offsite power available. The nominal PCT was 1509 °F for a 6% Gd_2O_3 rod. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 163 °F.

The case results, event times and analysis plots for the limiting PCT case are shown in Table 3-5, Table 3-6, and Figure 3-11 through Figure 3-21. Figure 3-6 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 ranges used in the analysis. Figure 3-7 and Figure 3-8 show the time of PCT and break size versus PCT scatter plots for the 59 calculations, respectively. Figure 3-9 and Figure 3-10 show the maximum oxidation and total oxidation versus PCT scatter plots for the 59 calculations, respectively. Key parameters for the limiting PCT case are shown in Figure 3-11 through Figure 3-21. Figure 3-11 is the plot of PCT independent of elevation; this figure clearly indicates that the transient exhibits a sustained and stable quench. A comparison of PCT results from both case sets is shown in Figure 3-22.

Table 3-1 Sampled LBLOCA Parameters

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties, burnup)
	Break type (guillotine versus split)
	Critical flow discharge coefficients (break)
De	decay heat
	Critical flow discharge coefficients (surge line)
	Initial upper head temperature
	Film boiling heat transfer
	Dispersed film boiling heat transfer
	Critical heat flux
Initial	T_{min} (intersection of film and transition boiling)
	stored energy
	Downcomer hot wall effects
	Steam generator interfacial drag
	Condensation interphase heat transfer
Metal-water	reaction
Plant¹	
	Offsite power availability ²
Brea	break size
Pressu	pressurizer pressure
	Pressurizer liquid level
SIT	pressure
	SIT liquid level
	SIT temperature (based on containment temperature)
Contai	containment temperature
Contai	containment volume
	Initial RCS flow rate
	Initial operating RCS temperature
	Diesel start (for loss of offsite power only)

¹ Uncertainties for plant parameters are based on typical plant-specific data with the exception of "Offsite power availability," which is a binary result that is specified by the analysis methodology.

² Not sampled, see Section 4.9.

Table 3-2 Plant Operating Range Supported by the LOCA Analysis

	Event	Operating Range
1.0	Plant Physical Description	
1.1	<u>Fuel</u>	
	a) Cladding outside diameter	0.440 in.
	b) Cladding inside diameter	0.384 in.
	c) Cladding thickness	0.028 in.
	d) Pellet outside diameter	0.377 in.
	e) Pellet density	95.35 percent of theoretical
	f) Active fuel length	136.7 in.
	g) Resinter densification	[]
h)	Gd ₂ O ₃ concentrations	2, 4, 6, 8 w/o
1.2	<u>RCS</u>	
	a) Flow resistance	Analysis
	b) Pressurizer location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	14X14 AREVA NP HTP fuel
	e) SG tube plugging	10 percent ($\pm 2\%$ asymmetry) ¹
2.0	Plant Initial Operating Conditions	
	<u>2.1 Reactor Power</u>	
	a) Nominal reactor power	3029 MWt ²
	b) LHR	15.0 kW/ft
c)	F _Q 2.161	
d)	Fr	1.810 ³
	<u>2.2 Fluid Conditions</u>	
	a) Loop flow	140.8 Mlbm/hr $\leq M \leq$ 164.6 Mlbm/hr
	b) RCS Cold Leg temperature	548.0 °F $\leq T \leq$ 554.0 °F
	c) Pressurizer pressure	2210 psia $\leq P \leq$ 2290 psia
	d) Pressurizer level	62.6 percent $\leq L \leq$ 68.6 percent
	e) SIT pressure	214.7 psia $\leq P \leq$ 294.7 psia
	f) SIT liquid volume	1090 ft ³ $\leq V \leq$ 1170 ft ³

¹ In the RLBLOCA analysis, only the maximum 10% tube plugging in each steam generator was analyzed. By independently sampling the break loss discharge coefficients, any flow differences attributed to asymmetry in the SG tube plugging is covered by use of the RLBLOCA methodology.

² Includes 0.3% uncertainties

³ The radial power peaking for the hot rod is including 6% measurement uncertainty and 3.5% allowance for control rod insertion affect.

$$Fr_{\text{tech spec}} * (1 + \text{uncert_Fr}) * (1 + \text{uncert_cr_insertion}) = 1.65 * (1.0 + 0.06) * (1 + 0.035) = 1.810$$

Table 3-2 Plant Operating Range Supported by the LOCA Analysis (Continued)

	Event	Operating Range
	g) SIT temperature	115.5 °F ≤ T ≤ 124.5 °F (It's coupled with containment temperature)
	h) SIT resistance fL/D	As-built piping configuration
	i) Minimum ECCS boron	≥1900 ppm
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 cold leg pipe area)	0.2997 ≤ A ≤ 1.0 full pipe area (split) 0.2997 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of one emergency diesel generator
	e) Offsite power	On or Off
	f) ECCS pumped injection temperature	120 °F
	g) HPSI pump delay	19.5 (w/ offsite power) 30.0 (w/o offsite power)
	h) LPSI pump delay	19.5 (w/ offsite power) 30.0 (w/o offsite power)
	i) Containment pressure	14.7 psia, nominal value ¹
	j) Containment temperature	115.5 °F ≤ T ≤ 124.5 °F
	k) Containment sprays delay	0 s
	l) LPSI flow	BROKEN_LOOP * * LOOP-1A1 * * RCS pressure LPSI flow * ----- psia gpm 18.32 1287. 23.48 1261. 33.47 1210. 43.02 1158. 47.64 1132. 52.14 1107. 69.04 1005. 87.73 877. 103.73 748. 117.05 620. 127.72 492. 135.41 364. 140.64 236. 143.98 82. 144.37 31. 144.44 0. INTACT_LOOP1 * LOOP-1B1 *

¹ Nominal containment pressure range is -0.7 to 0.5 psig. For RLBOCA, a reasonable value between this range is acceptable.

		* RCS pressure	LPSI flow
		-----	-----
psia	gpm		
18.32	0.0		
23.48	0.0		
33.47	0.0		
43.02	0.0		
47.64	0.0		
52.14	0.0		
69.04	0.0		
87.73	0.0		
103.73	0.0		
117.05	0.0		
127.72	0.0		
135.41	0.0		
140.64	0.0		
143.98	0.0		
144.37	0.0		
144.44	0.0		
INTACT_LOOP2			
*			
* LOOP-1A2			
*			
		* RCS pressure	LPSI flow
		-----	-----
psia	gpm		
18.32	0.0		
23.48	0.0		
33.47	0.0		
43.02	0.0		
47.64	0.0		
52.14	0.0		
69.04	0.0		
87.73	0.0		
103.73	0.0		
117.05	0.0		
127.72	0.0		
135.41	0.0		
140.64	0.0		
143.98	0.0		
144.37	0.0		
144.44	0.0		
INTACT_LOOP3			
*			
* LOOP-1B2			
*			
		* RCS pressure	LPSI flow
		-----	-----
psia	gpm		
18.32	926.		
23.48	902.		
33.47	853.		
43.02	804.		
47.64	780.		
52.14	755.		
69.04	657.		
87.73	535.		
103.73	413.		
117.05	291.		
127.72	169.		

		135.41 47. 140.64 0. 143.98 0. 144.37 0. 144.44 0.
	m) HPSI flow	BROKEN_LOOP * RCS pressure HPSI flow * psia gpm 15. 160.0 315. 137.0 615. 109.0 815. 85.0 1015. 51.0 1115. 16.0 1125. 8.0 1129. 0.0 INTACT_LOOP1 * RCS pressure HPSI flow * ----- psia gpm 15. 151.7 315. 130.0 615. 103.7 815. 81.3 1015. 48.7 1115. 15.3 1125. 5.7 1129. 0.0 INTACT_LOOP2 * RCS pressure HPSI flow * ----- psia gpm 15. 151.7 315. 130.0 615. 103.7 815. 81.3 1015. 48.7 1115. 15.3 1125. 5.7 1129. 0.0 INTACT_LOOP3 * RCS pressure HPSI flow * ----- psia gpm 15. 0.0 315. 0.0 615. 0.0 815. 0.0 1015. 0.0 1115. 0.0 1125. 0.0 1129. 0.0

Table 3-3 Statistical Distributions Used for Process Parameters¹

Parameter	Operational Uncertainty Distribution	Parameter Range
Pressurizer Pressure (psia)	Uniform	2210 – 2290
Pressurizer Liquid Level (percent)	Uniform	62.6 – 68.6
SIT Liquid Volume (ft ³)	Uniform	1090.0 – 1170.0
SIT Pressure (psia)	Uniform	214.7 – 294.7
Containment Temperature (°F)	Uniform	115.5 – 124.5
Containment Volume (ft ³) Uniform		2.461E+6 – 2.637E+6
Initial RCS Flow Rate (Mlbm/hr)	Uniform	140.8 – 164.6
Initial RCS Operating Temperature (Tcold) (°F)	Uniform 548.0	– 554.0
RWST Temperature for ECCS (°F)	Point	104
Offsite Power Availability ² Binar	y	0,1
Delay for Containment Spray (s)	Point	0
LPSI Pump Delay (s)	Point	19.5 (w/ offsite power) 30.0 (w/o offsite power)
HPSI Pump Delay (s)	Point	19.5 (w/ offsite power) 30.0 (w/o offsite power)

¹ Note that core power is not sampled, see Section 1.0.

² This is no longer a sampled parameter. One set of 59 cases is run with LOOP and one set of 59 cases is run with No-LOOP.

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 CCFL violation in the downcomer should such occur.	There was no significant occurrence of CCFL violation in the downcomer for this analysis. Violations of CCFL were noted in a statistically insignificant number of time steps.
2. AREVA NP has agreed that it is not to use normalization with hot leg to downcomer nozzle gaps.	Hot leg nozzle gaps were not modeled.
3. If AREVA NP 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 clad 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 St Lucie Unit 1 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 spillunder (Top crossover pipe (ID) at the crossover pipes lowest elevation) that are below the top elevation of the core, AREVA NP 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.	For St Lucie unit 1, the elevation of the cross-over piping top (ID) relative to the cold leg center line is -57 inches, and the elevation of the top of the active core relative to the cold leg center line is -66.235 inches. Therefore, no evaluation is required.
5. The model applies to 3 and 4 loop Westinghouse- and CE-designed nuclear steam systems.	St Lucie Unit 1 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).	St Lucie Unit 1 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 or 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.	The limiting case did not show any evidence of a 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.	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 10 CFR 50.46, 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 in this analysis.
10. Specific guidelines must be used to develop the plant-specific nodalization. Deviations from the reference plant must be addressed.	The nodalization in the plant model is consistent with the CE-designed 2X4 loop sample calculation that was submitted to the NRC for review. Figure 3-1 shows the loop noding used in this analysis. (Note only Loop 1 is shown in the figure; Loops 2 and 3 are identical to loop 1, except that only Loop 1 contains the pressurizer and the break.) Figure 3-2 shows the steam generator model. Figures 3-3, 3-4, and 3-5 show the reactor vessel 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.	Simulation of clad temperature response is a function of phenomenological correlations that have been derived either analytically or experimentally. The important correlations have been validated for the RLBLOCA methodology and a statement of the range of applicability has been documented. The correlations of interest are the set of heat transfer correlations as described in Reference 1. 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 case of this analysis. Calculated values for other parameters of interest are also provided. As is evident, the plant-specific parameters fall within the methodology's range of applicability.
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 discussed in Section 3.5.
13. The licensee or applicant wishing to apply AREVA NP 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 10 CFR 50.46(a)(i) to include M5 cladding material has been completed.	Not applicable.

Table 3-5 Summary of Results for the Limiting PCT Case

Case # 3	1
PCT	
Temperature	1672 °F
Time	26.6 s
Elevation	3.406 ft
Metal-Water Reaction	
Percent Oxidation Maximum	0.6517
Percent Total Oxidation	0.0381

Table 3-6 Calculated Event Times for the Limiting PCT Case

Event	Time (s)
Break Opened	0.0
RCP Trip	N/A
SIAS Issued	1.0
Start of Broken Loop SIT Injection	14.6
Start of Intact Loop SIT Injection (Loops 2, 3 and 4 respectively)	17.3, 17.3 and 17.3
Broken Loop LPSI Delivery Began	31.0
Intact Loop LPSI Delivery Began (Loops 2, 3 and 4 respectively)	N/A, N/A and 31.0
Broken Loop HPSI Delivery Began	31.0
Intact Loop HPSI Delivery Began (Loops 2, 3 and 4 respectively)	31.0, 31.0 and N/A
Beginning of Core Recovery (Beginning of Reflood)	26.9
Broken Loop SIT Emptied	58.1
Intact Loop SITs Emptied (Loops 2, 3 and 4 respectively)	56.1, 58.6 and 60.9
PCT Occurred	26.6
Transient Calculation Terminated	553.5

Table 3-7 Heat Transfer Parameters for the Limiting Case

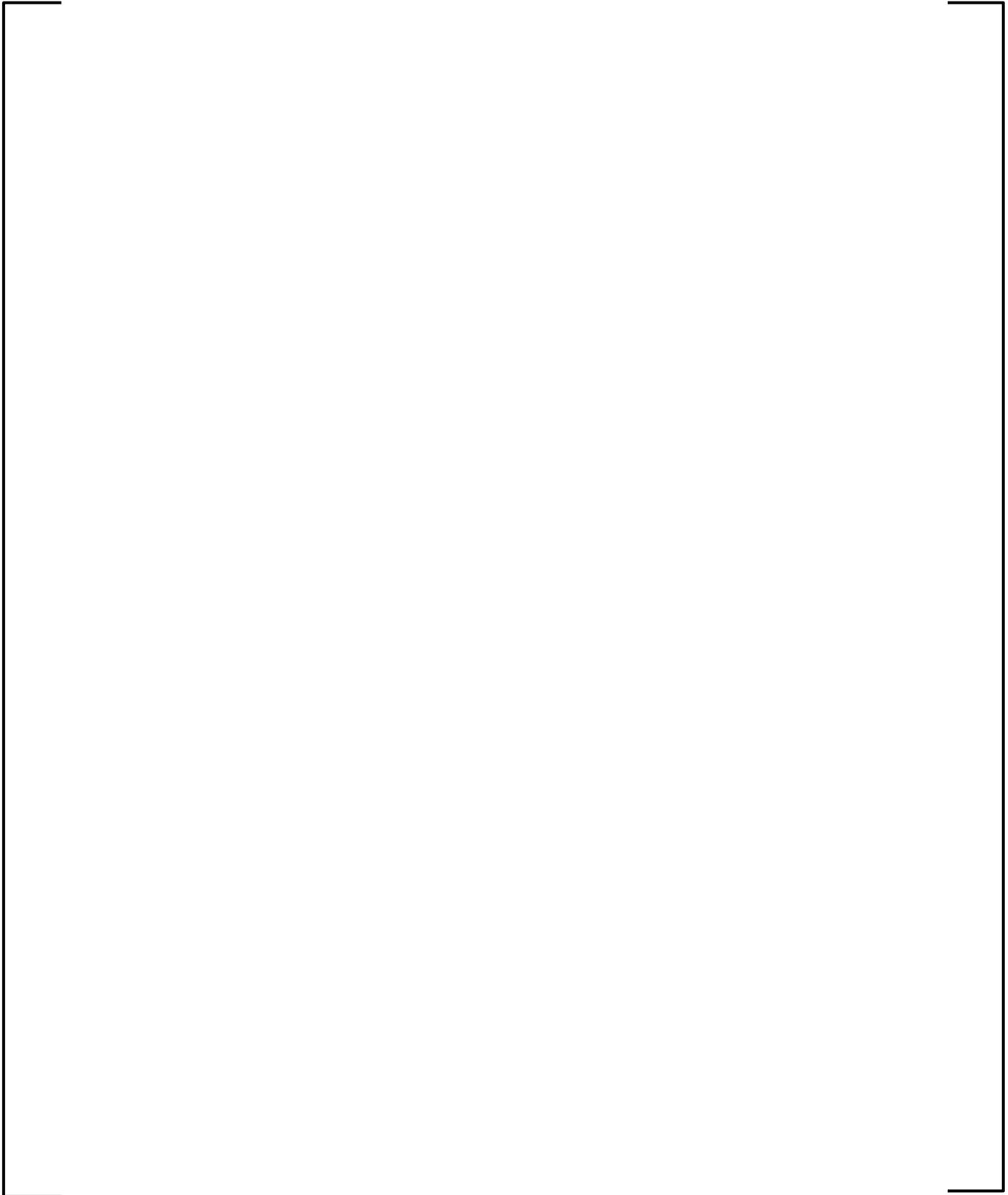
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Table 3-8 Containment Initial and Boundary Conditions

Containment Net Free Volume (ft³)	2,460,780 – 2,636,550
Initial Conditions	
Containment Pressure (nominal)	14.7 psia
Containment Temperature	115.5 °F – 124.5 °F
Outside Temperature	38 °F
Humidity 1.0	
Containment Spray	
Number of Pumps operating	2
Spray Flow Rate (Total, both pumps)	9,000 gpm
Minimum Spray Temperature	36 °F
Fastest Post-LOCA initiation of spray	0 s
Containment Fan Coolers	
Number of Fan Coolers Operating	4
Minimum Post Accident I initiation Time of Fan Coolers (sec)	0
Fan Cooler Capacity (1 Fan Cooler)	
Containment Temperature (F)	Heat Removal Rate (BTU/sec)
60	0
120	3472
180	8865
220	13,933
264	25,000

Table 3-9 Passive Heat Sinks in Containment¹

Heat Sink	Area (ft ²)	Thickness (ft)	Material
Containment Shell	86700	0.1171	C Steel
Floor Slab	12682	20.0	Concrete
Misc Concrete	87751	1.5	Concrete
Galvanized Steel	130000	0.0005833	Zinc
	130000	0.01417	C Steel
Carbon Steel	25000	0.03125	C Steel
Stainless Steel	22300	0.0375	S Steel
Misc Steel	40000	0.0625	C Steel
Misc Steel	41700	0.02083	C Steel
Misc Steel	7000	0.17708	C Steel
Imbedded Steel	18000	0.0708	C Steel
	18000	7.07	Concrete
Sump (GSI-191)	7414	0.02895	C Steel
Material Properties	Thermal Conductivity (BTU/hr-ft-°F)	Volumetric Heat Capacity (BTU/ft ³ -°F)	
Concrete	1.0	34.2	
Carbon Steel	25.9	53.57	
Stainless Steel	9.8	54.0	
Galvanizing	64.0	40.6	

¹ Passive heat sinks data listed in the table were used for RLBOCA analysis. Sensitivity studies were previously performed for the AREVA RLBLOCA Transition Package as applied to EMF-2103 to respond to the NRC's concerns. The results showed for a large dry containment, the PCT is not sensitive to change in containment back pressure. Hence, the heat sinks changes within ±5% range will not change the presented RLBLOCA results.

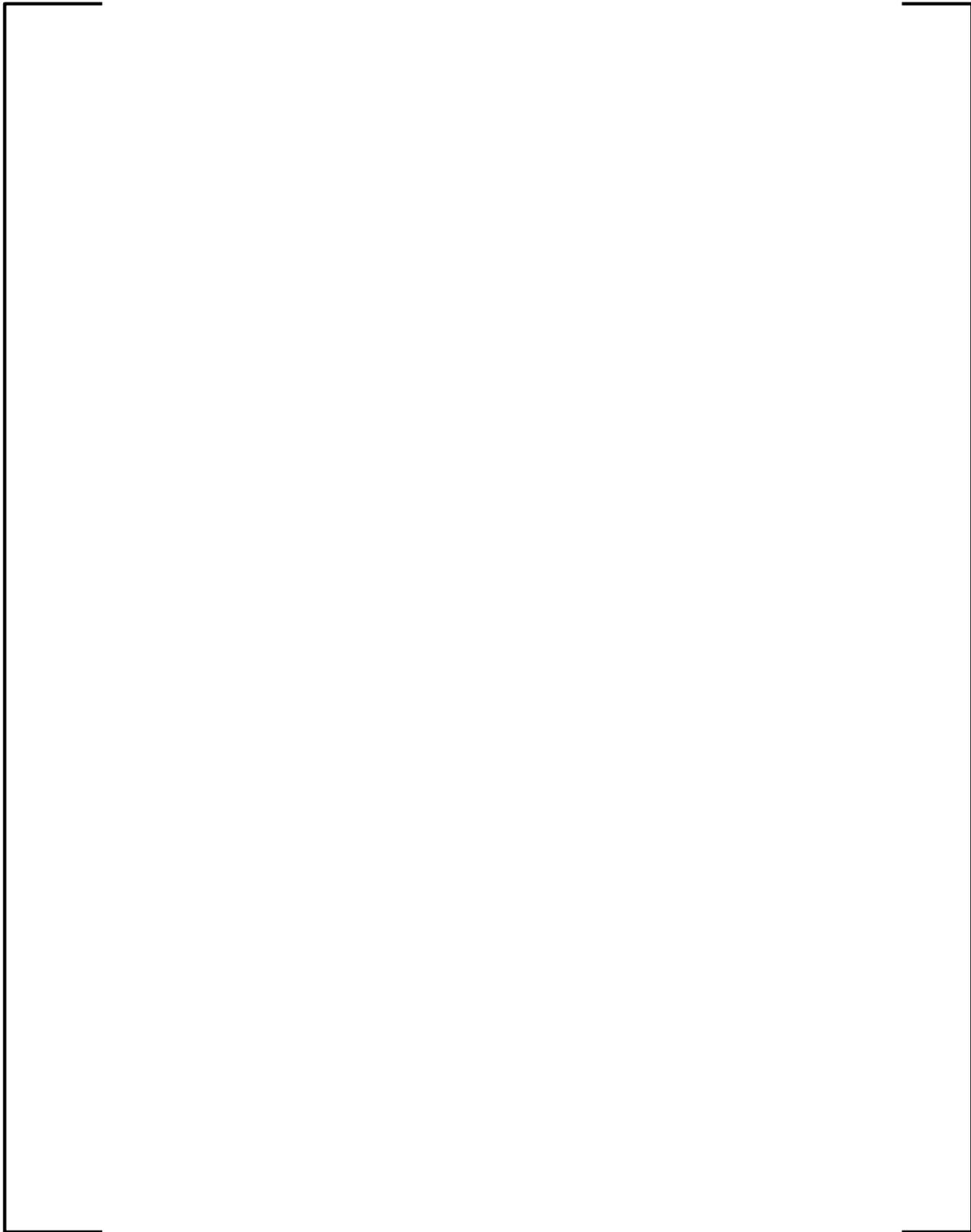


Figure 3-1 Primary System Noding

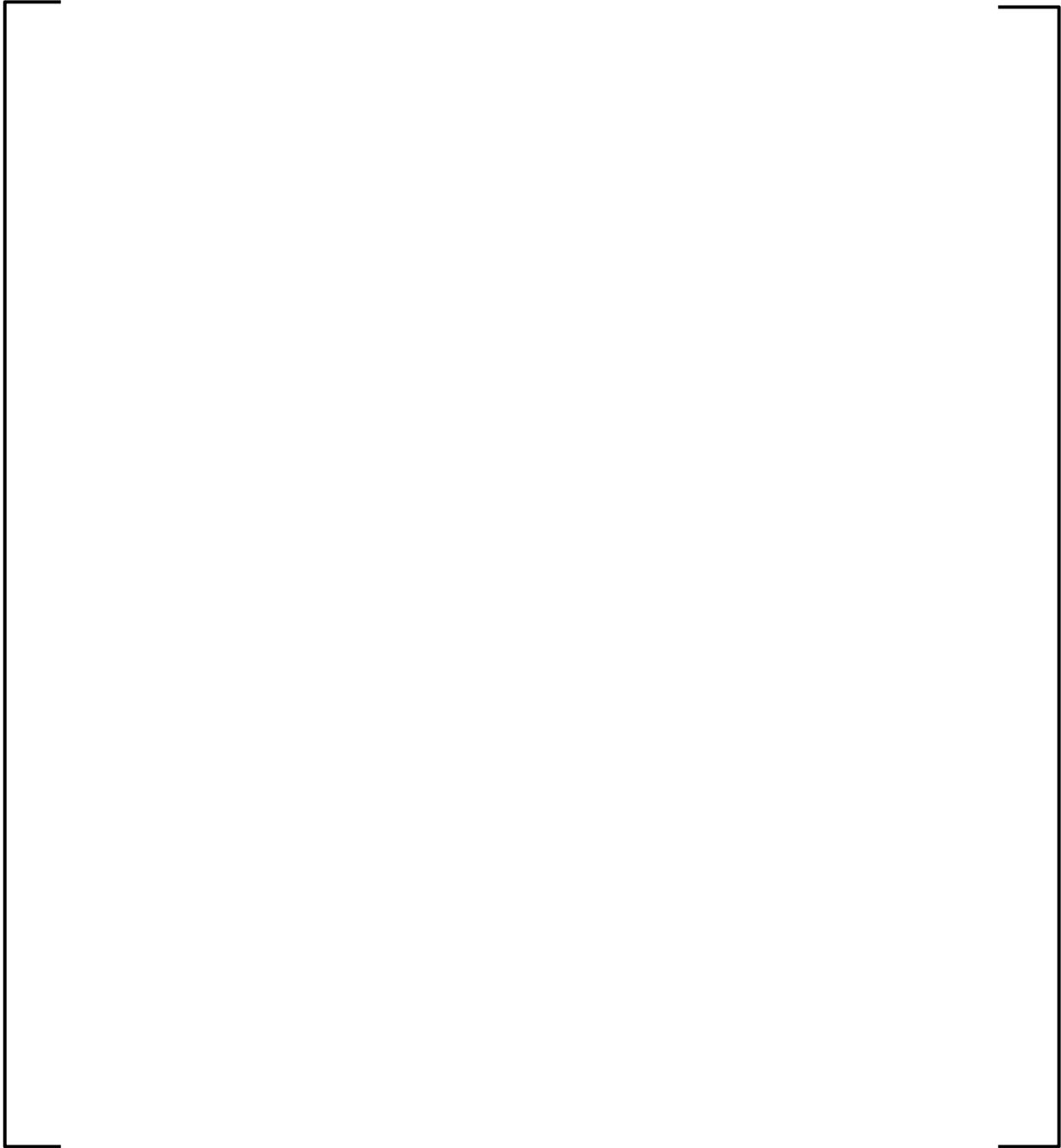


Figure 3-2 Secondary System Noding

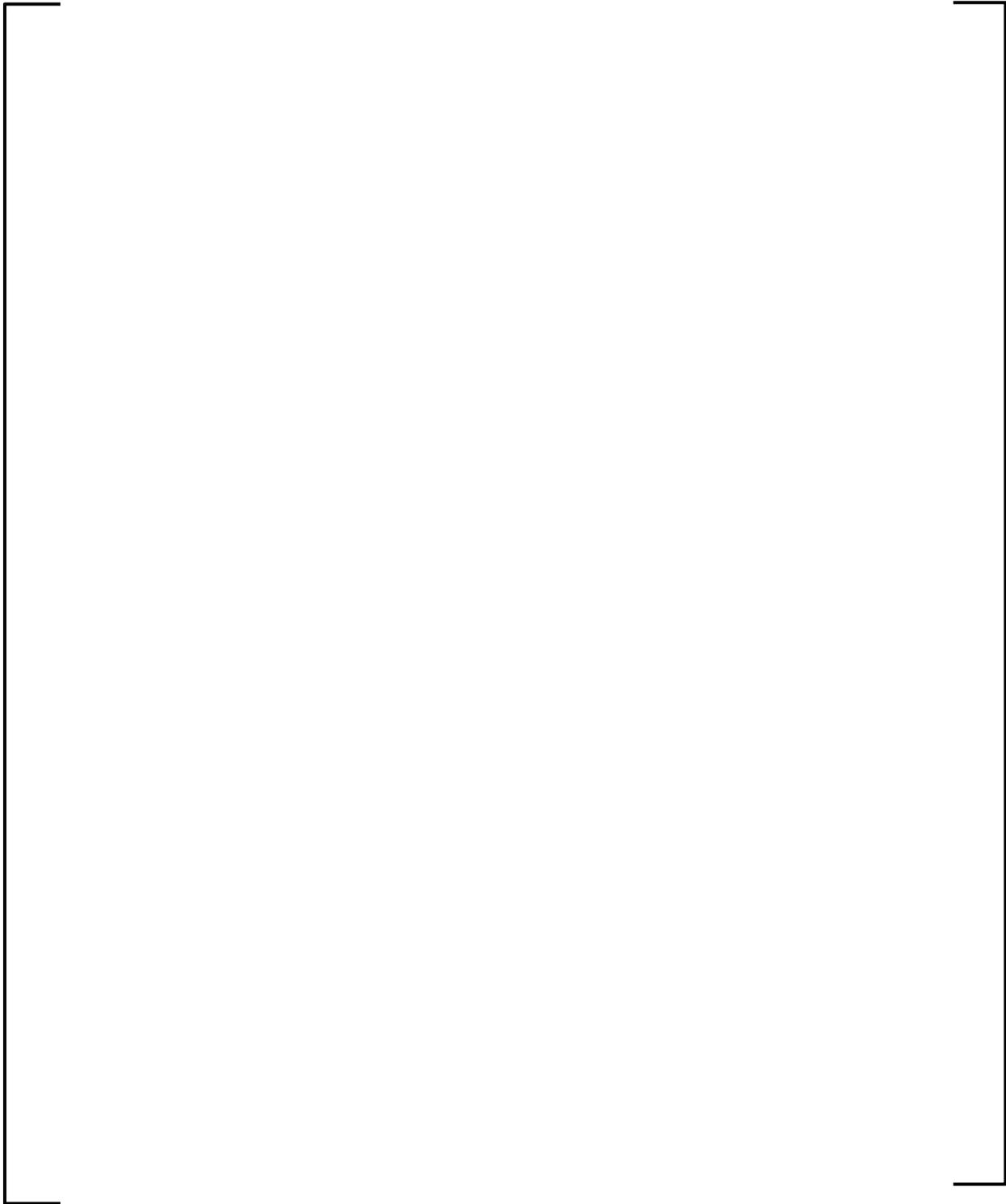


Figure 3-3 Reactor Vessel Noding

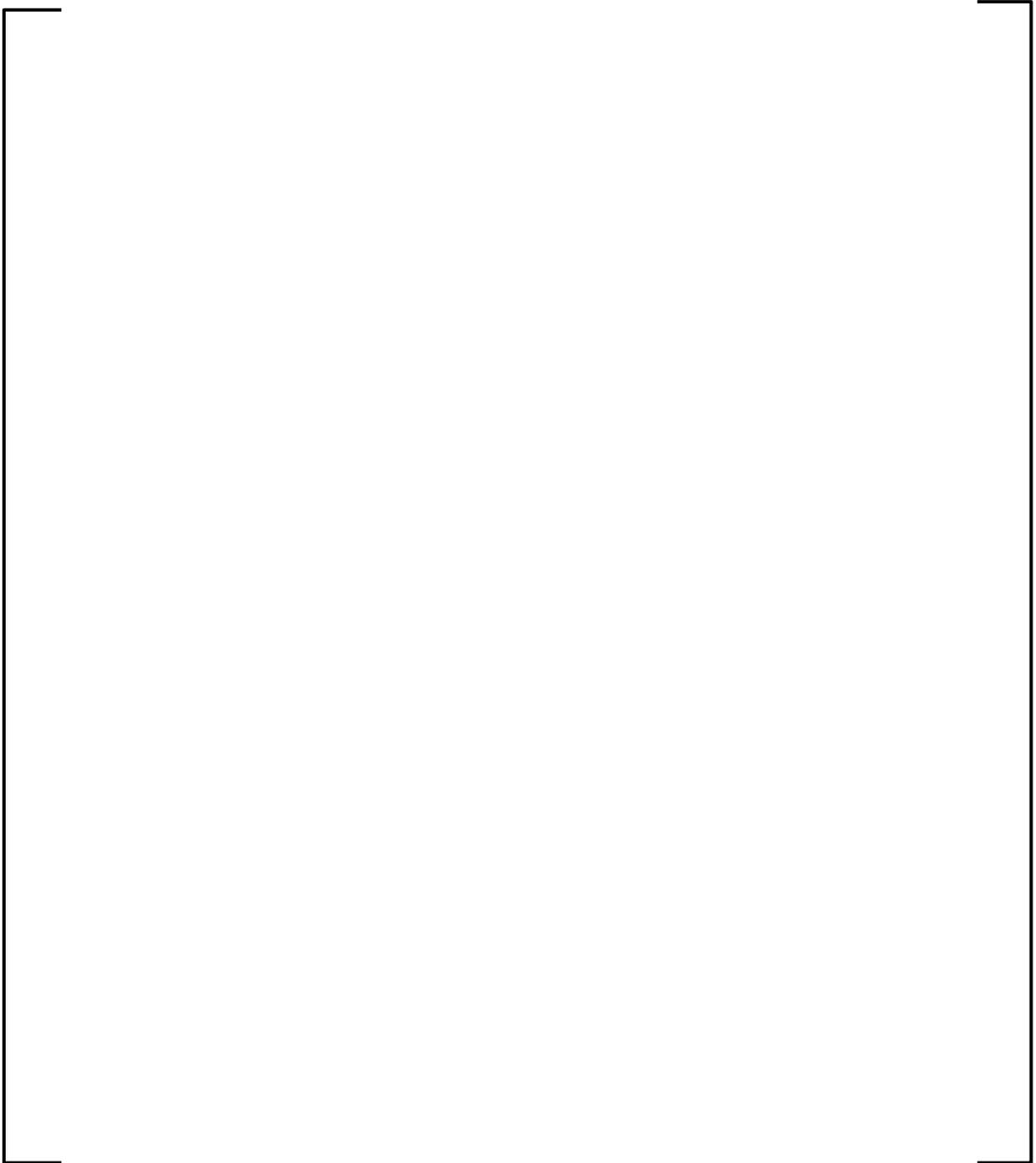


Figure 3-4 Core Noding Detail



Figure 3-5 Upper Plenum Noding Detail

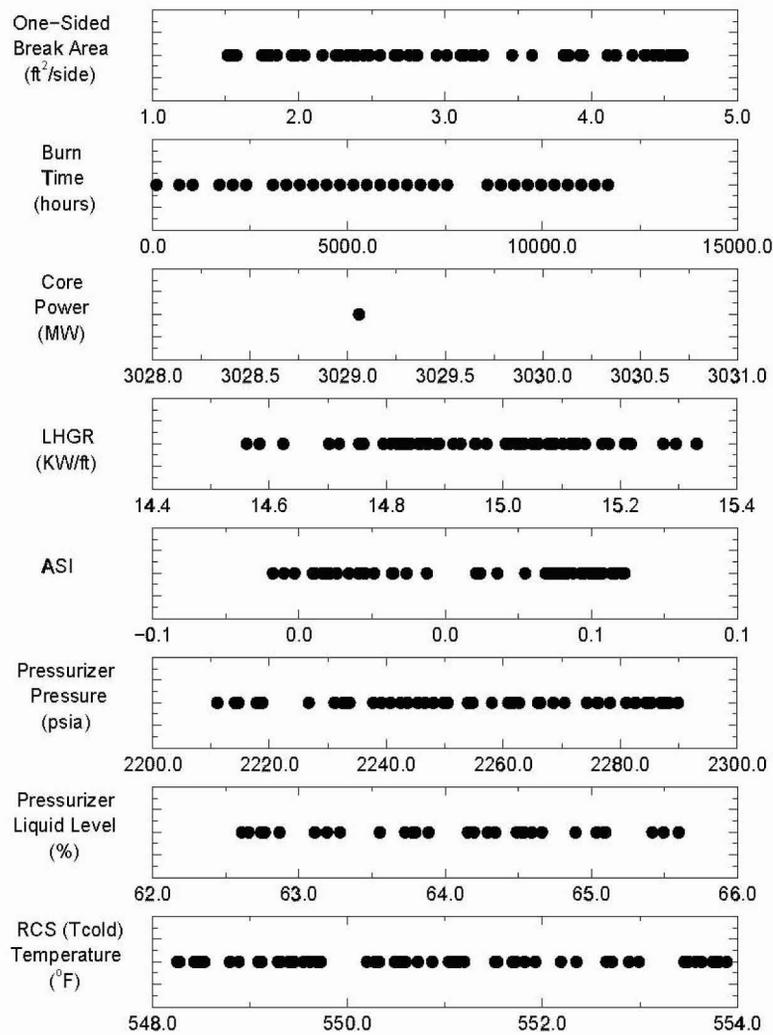


Figure 3-6 Scatter Plot of Operational Parameters

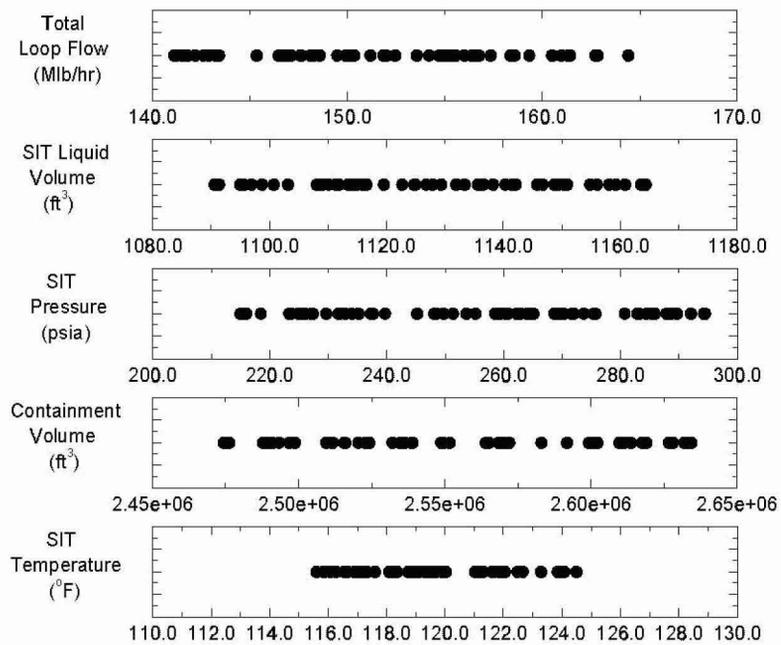


Figure 3-6 Scatter Plot of Operational Parameters (Continued)

PCT vs Time of PCT

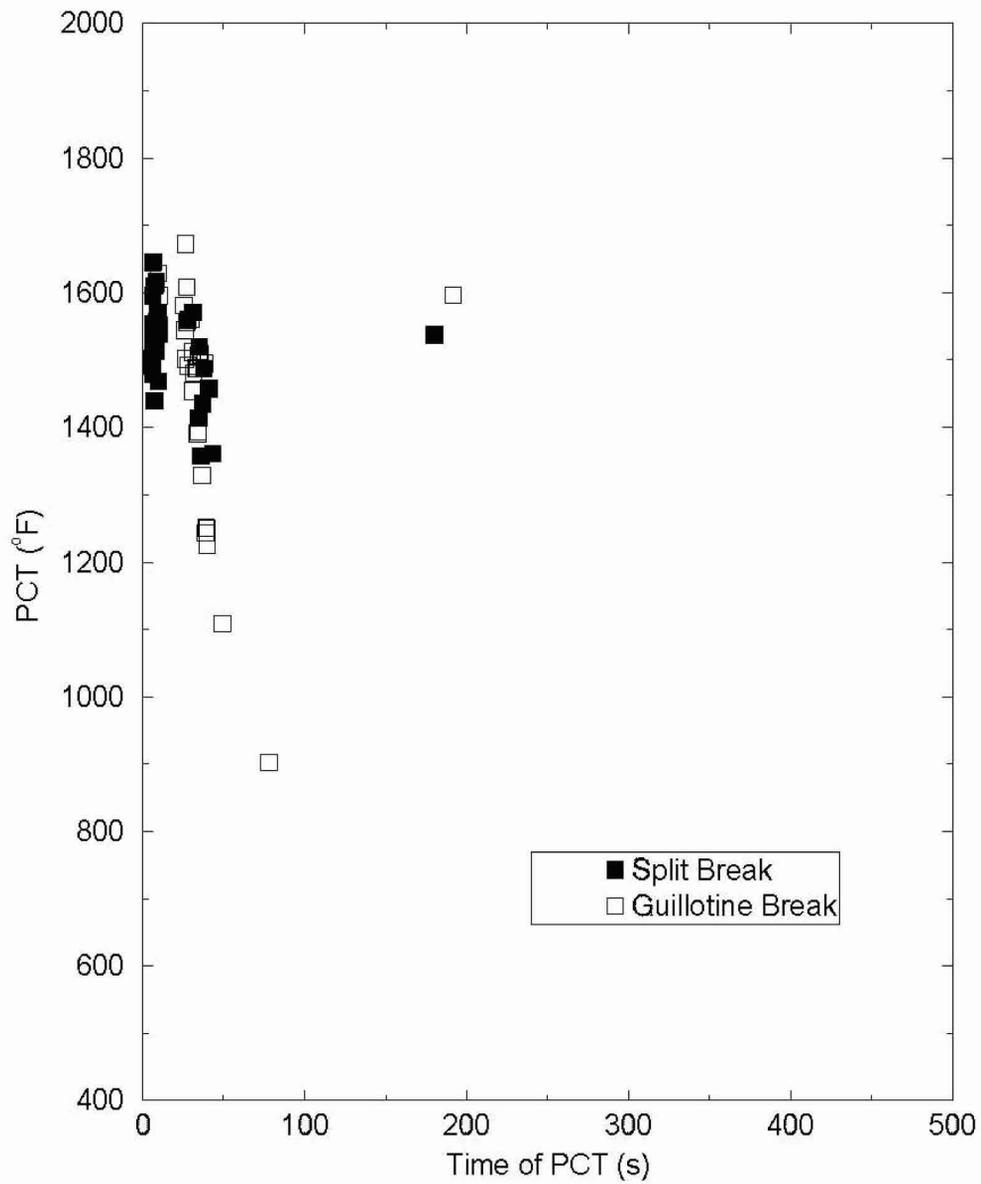


Figure 3-7 PCT versus PCT Time Scatter Plot from 59 Calculations

PCT vs One-sided Break Area

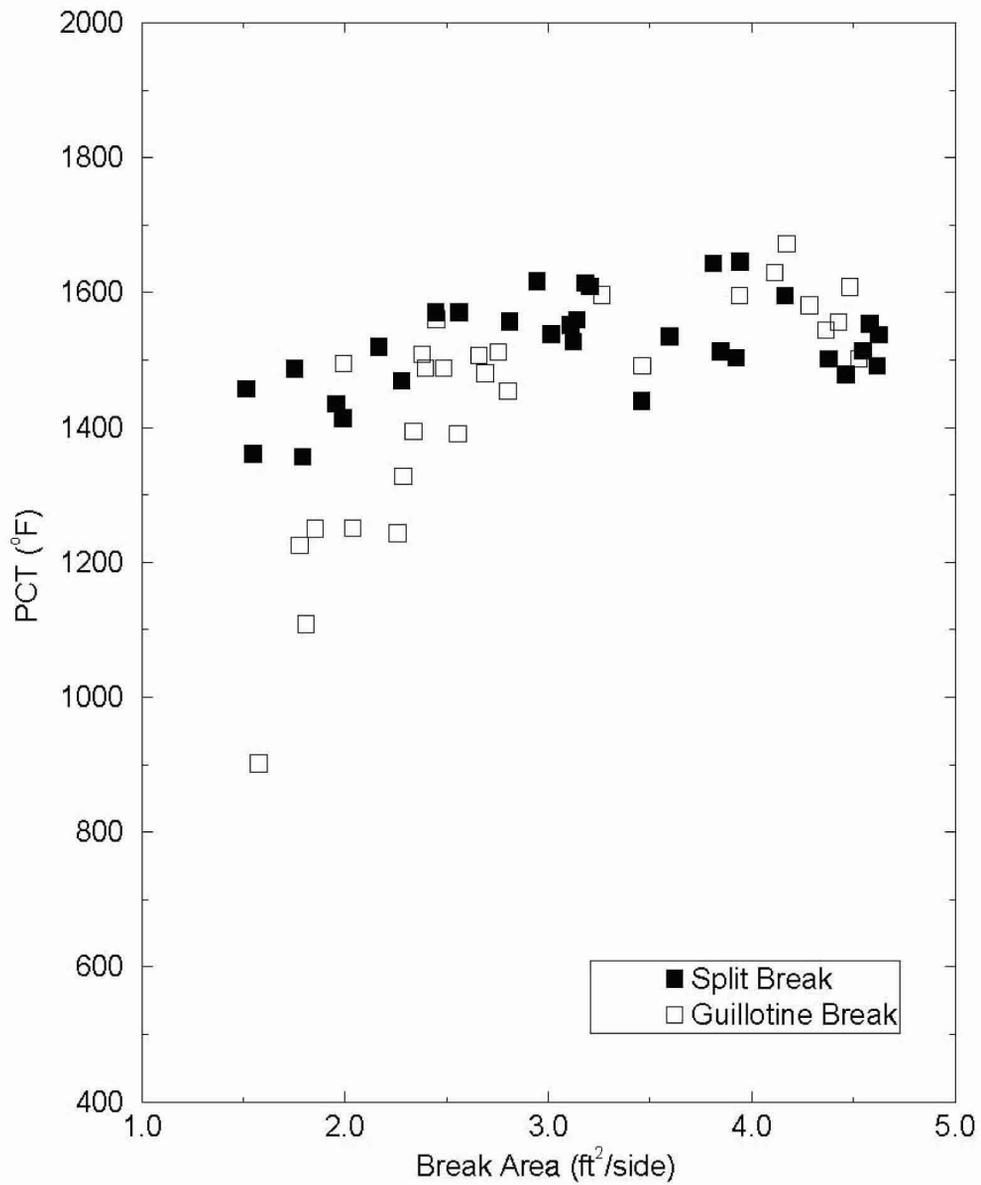


Figure 3-8 PCT versus Break Size Scatter Plot from 59 Calculations

Maximum Oxidation vs PCT

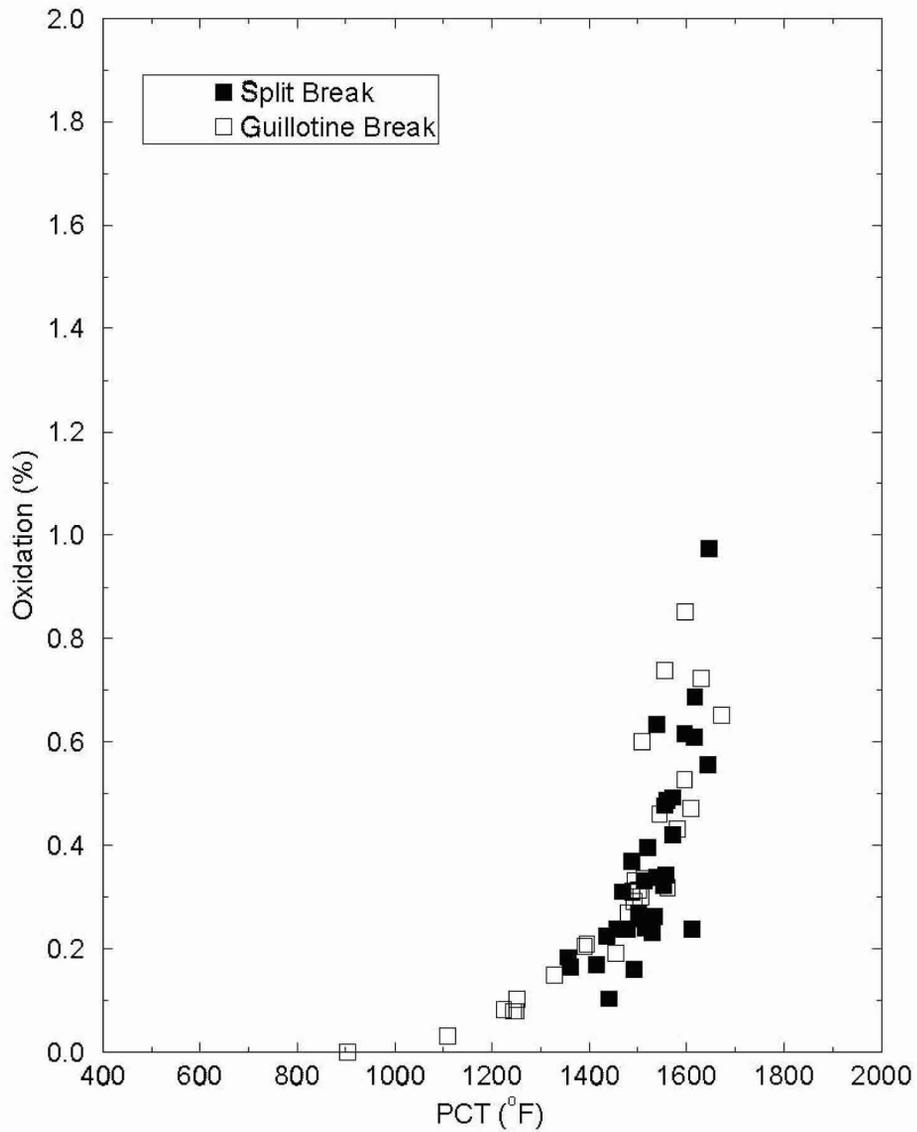


Figure 3-9 Maximum Oxidation versus PCT Scatter Plot from 59 Calculations

Total Oxidation vs PCT

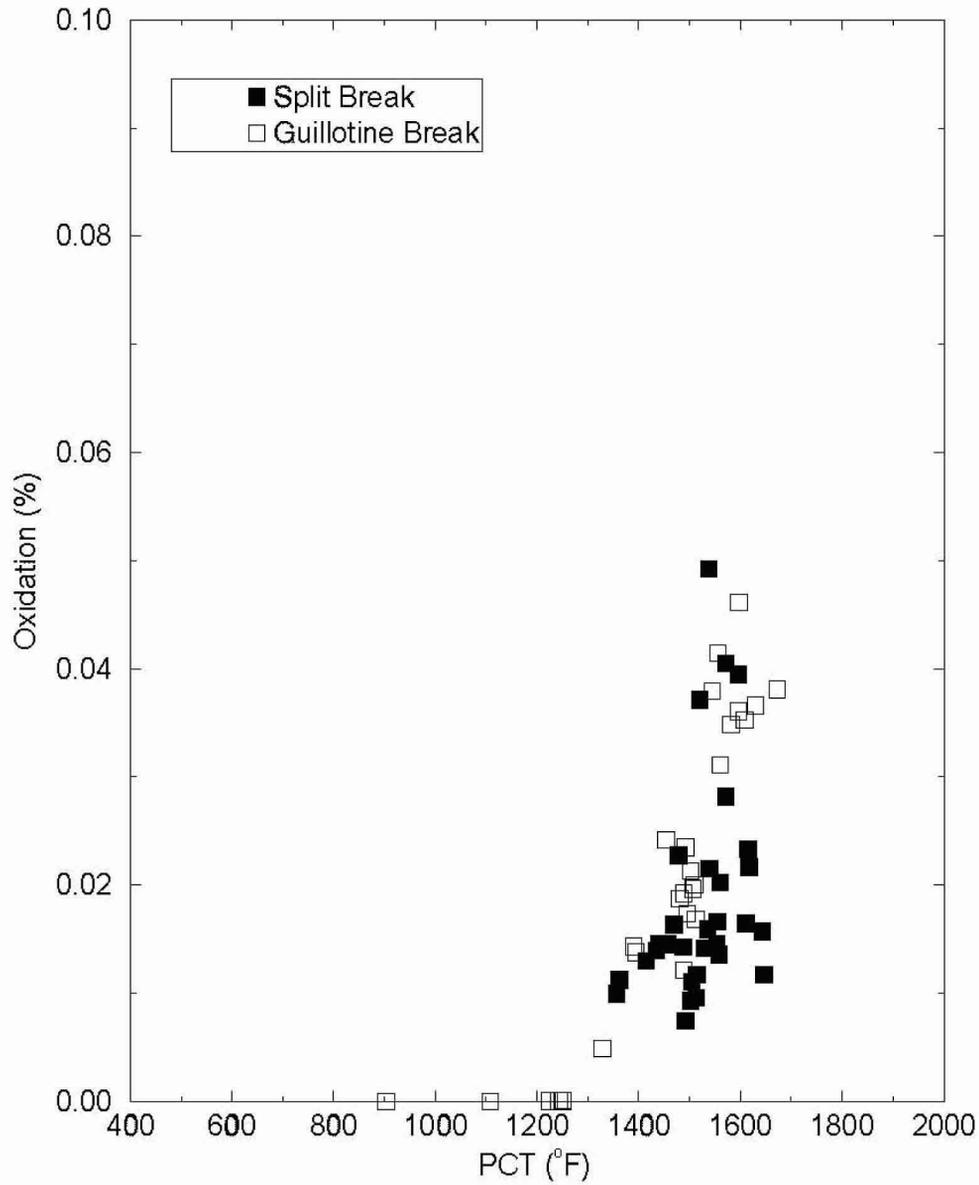
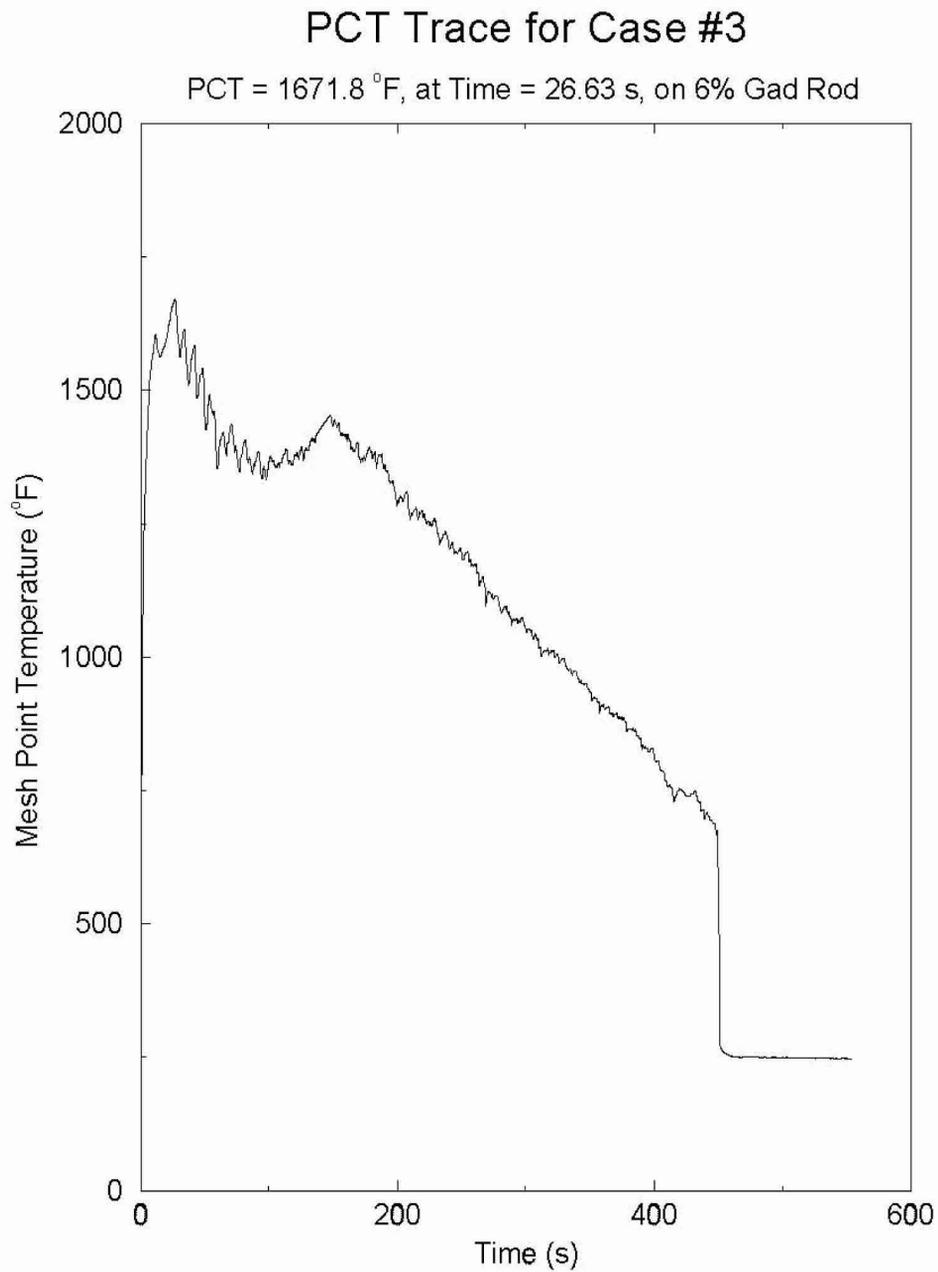
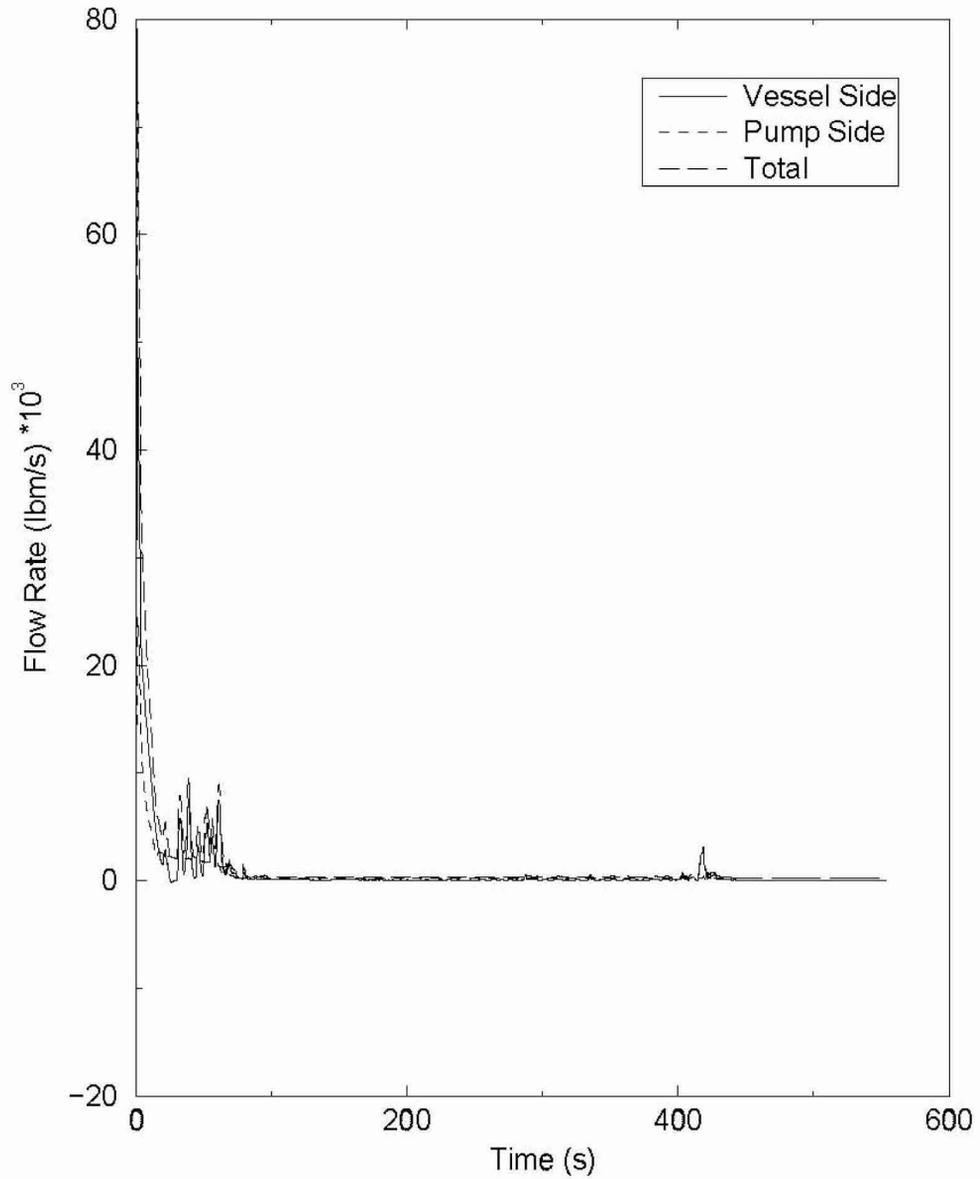


Figure 3-10 Total Oxidation versus PCT Scatter Plot from 59 Calculations



**Figure 3-11 Peak Cladding Temperature (Independent of Elevation)
for the Limiting Case**

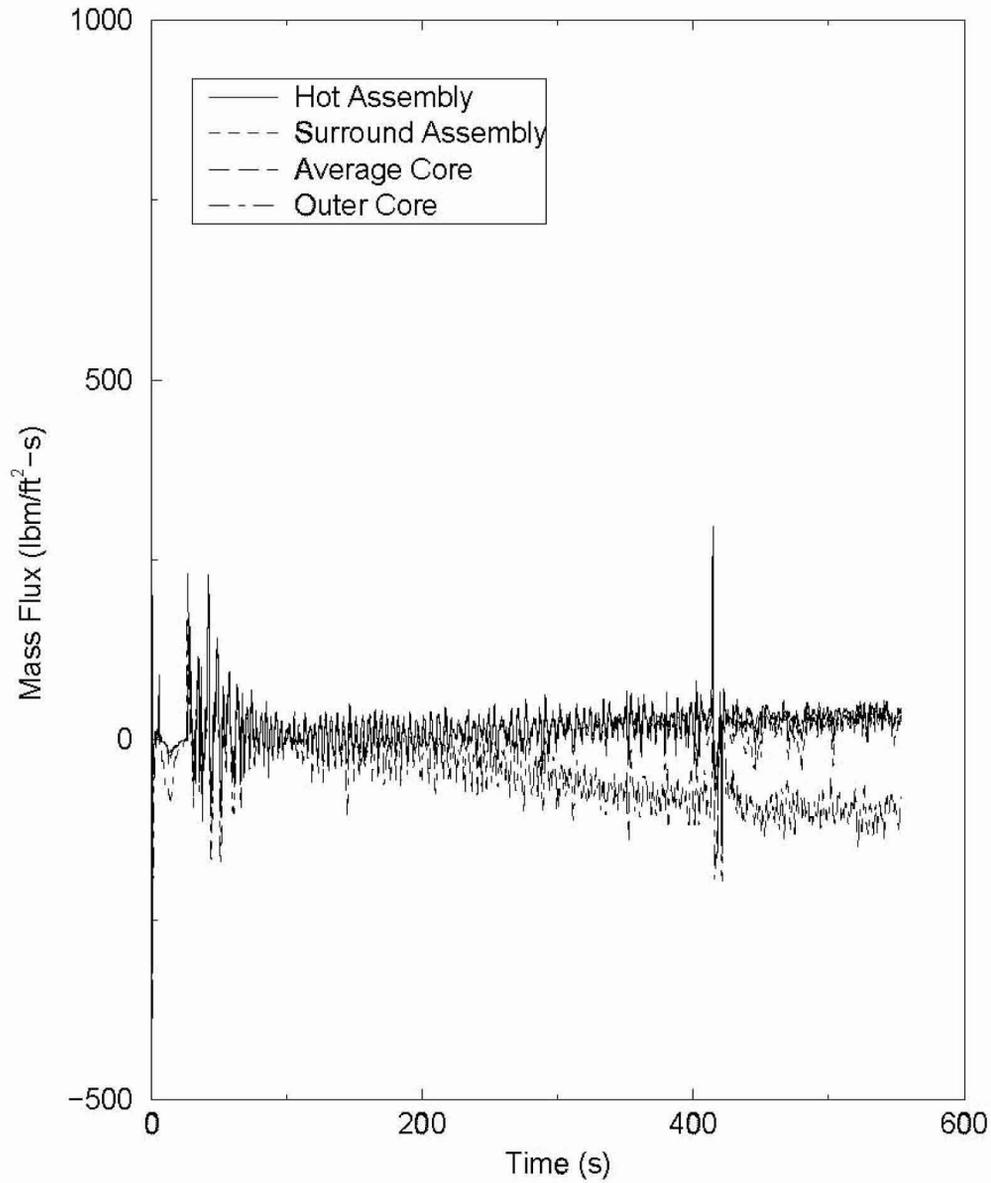
Break Flow



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Figure 3-12 Break Flow for the Limiting Case

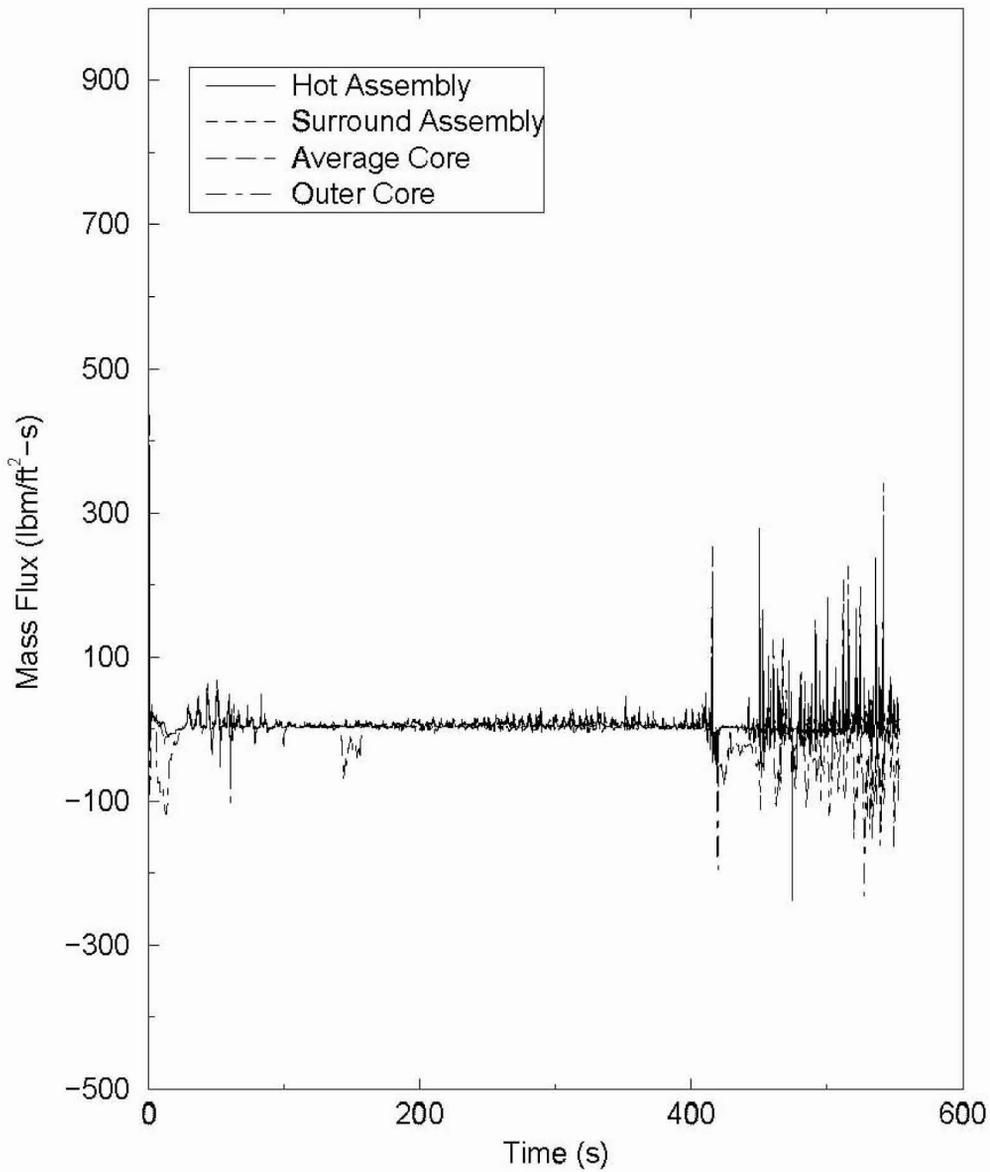
Core Inlet Mass Flux



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Figure 3-13 Core Inlet Mass Flux for the Limiting Case

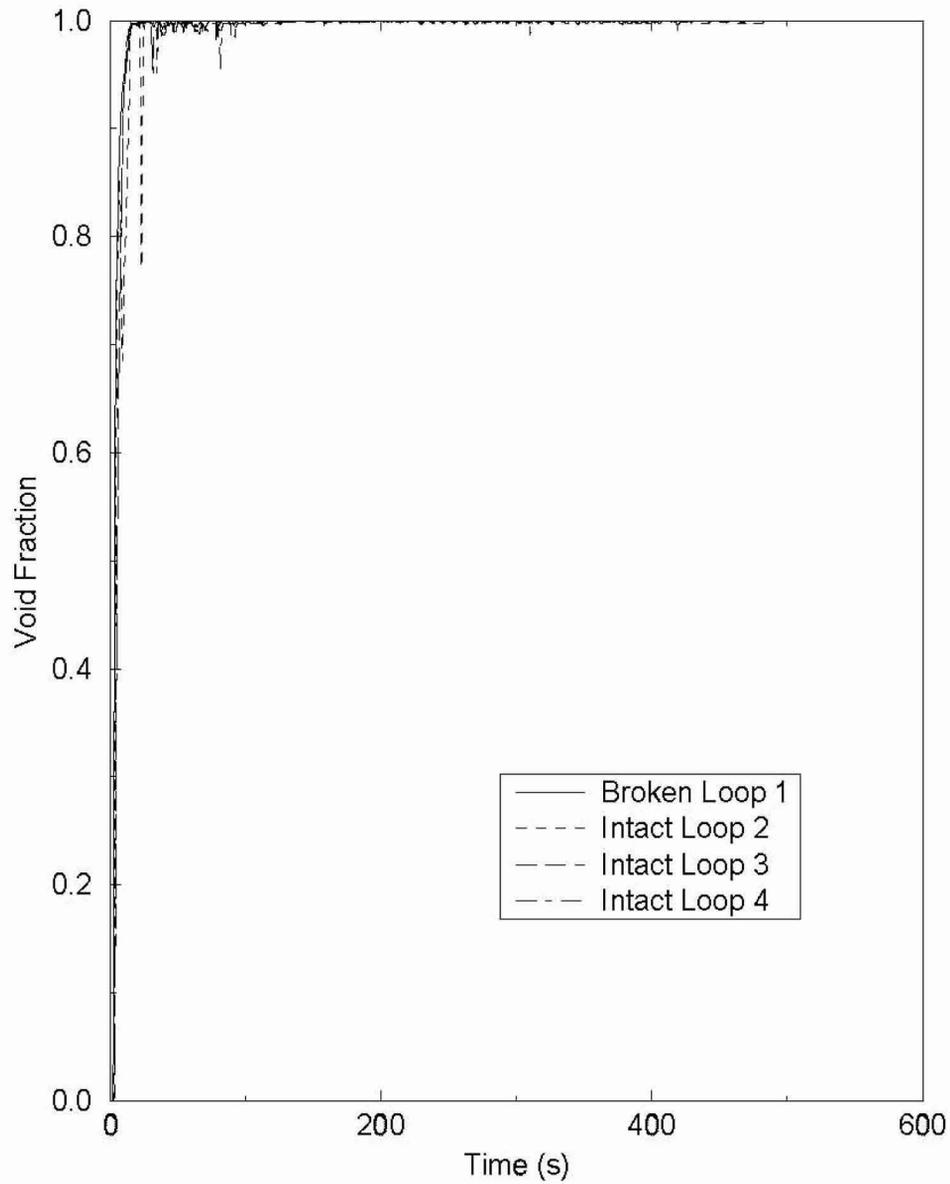
Core Outlet Mass Flux



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Figure 3-14 Core Outlet Mass Flux for the Limiting Case

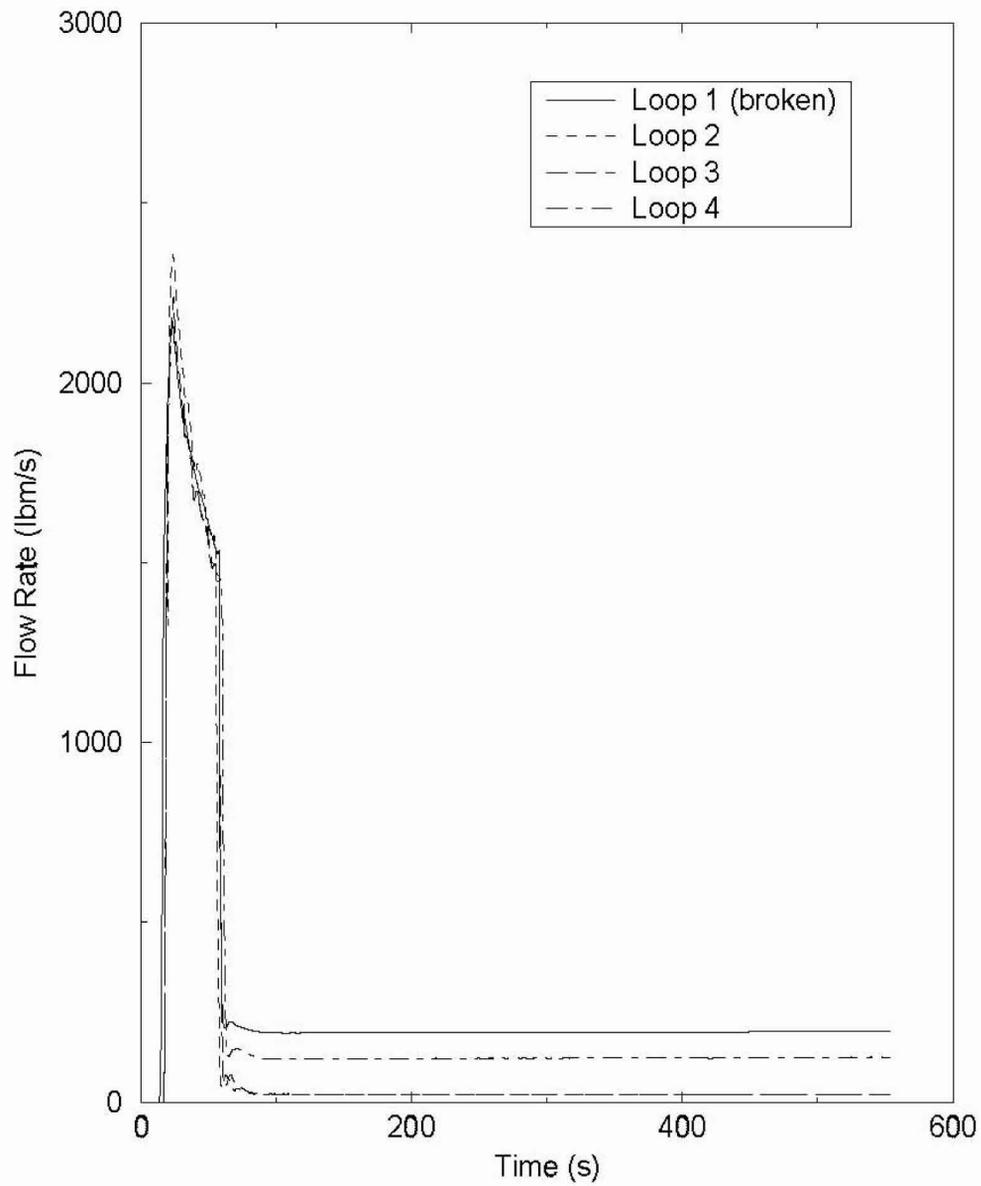
Pump Void Fraction



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Figure 3-15 Void Fraction at RCS Pumps for the Limiting Case

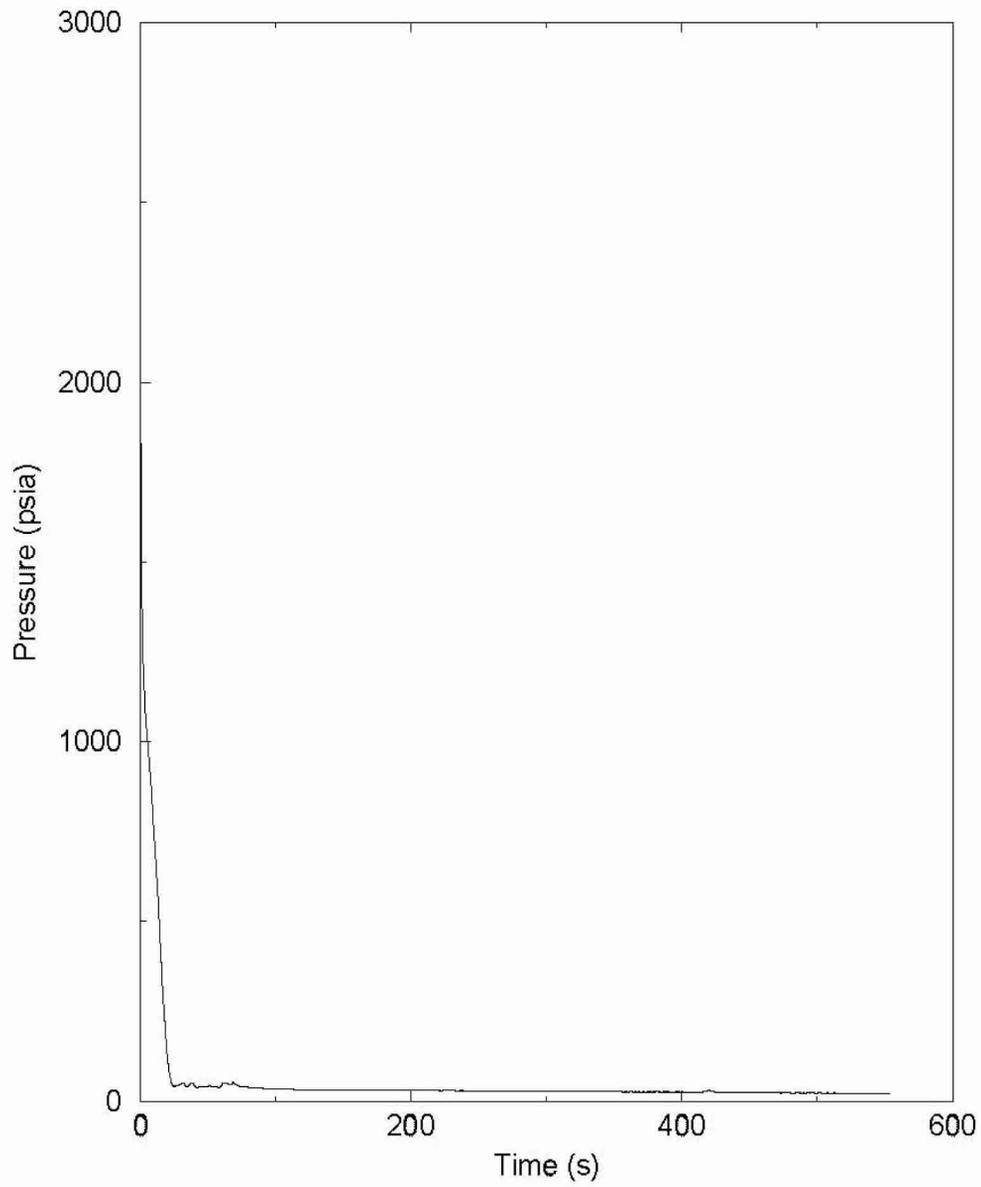
ECCS Flows



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Figure 3-16 ECCS Flows (Includes SIT, LPSI and HPSI) for the Limiting Case

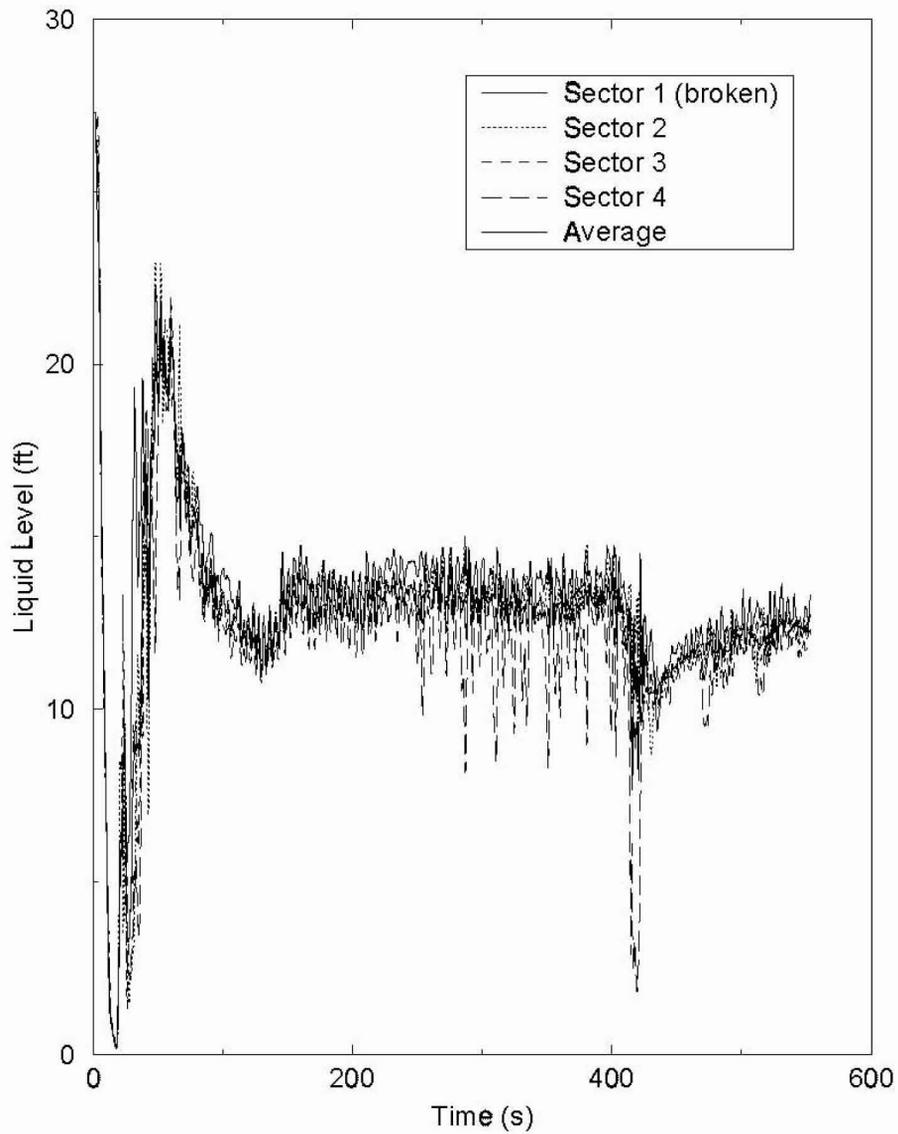
Upper Plenum Pressure



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Figure 3-17 Upper Plenum Pressure for the Limiting Case

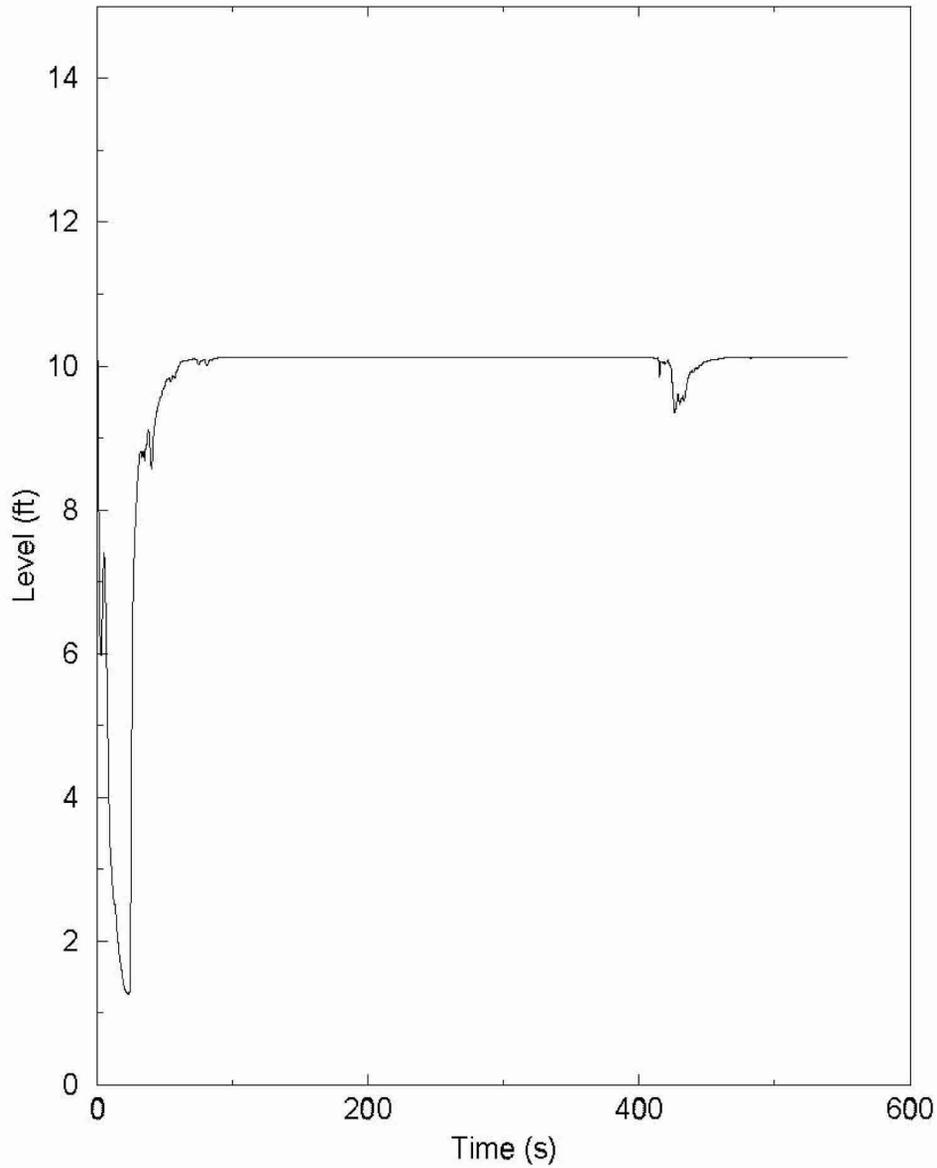
Downcomer Liquid Level



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**Figure 3-18 Collapsed Liquid Level in the Downcomer
for the Limiting Case**

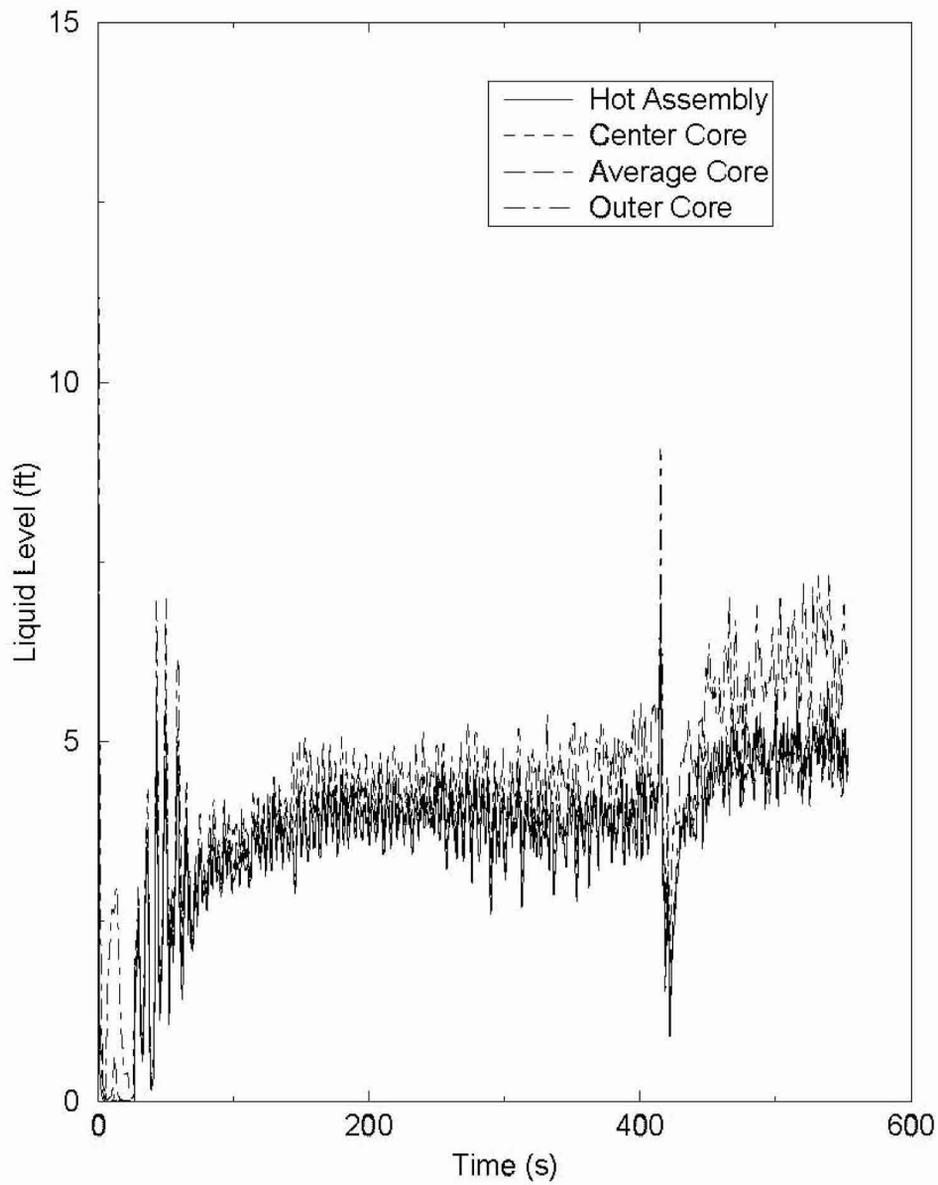
Lower Vessel Liquid Level



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**Figure 3-19 Collapsed Liquid Level in the Lower Plenum
for the Limiting Case**

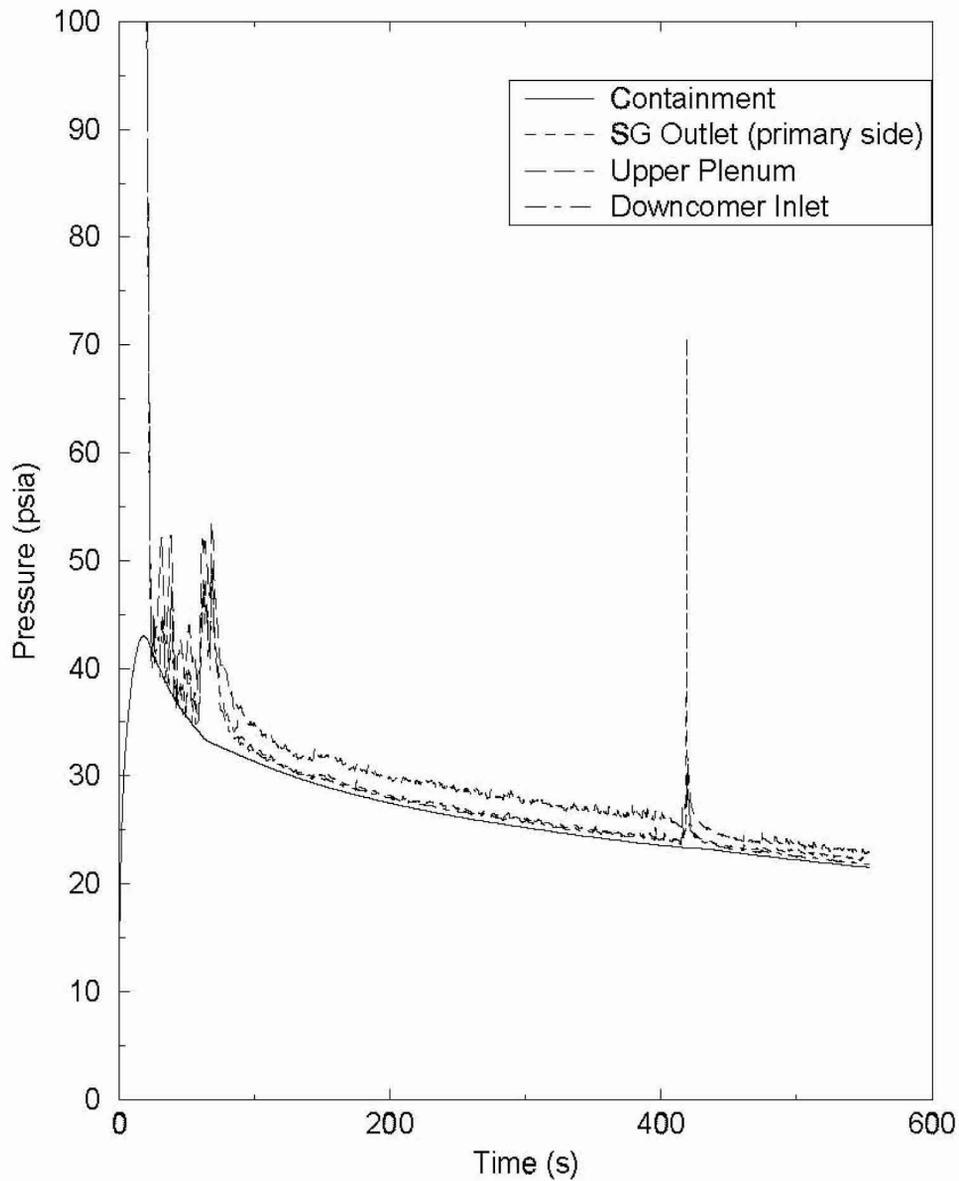
Core Liquid Level



ID:22950 16Dec2009 15:31:36 R5DMX

**Figure 3-20 Collapsed Liquid Level in the Core
for the Limiting Case**

Containment and Loop Pressures



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Figure 3-21 Containment and Loop Pressures for the Limiting Case

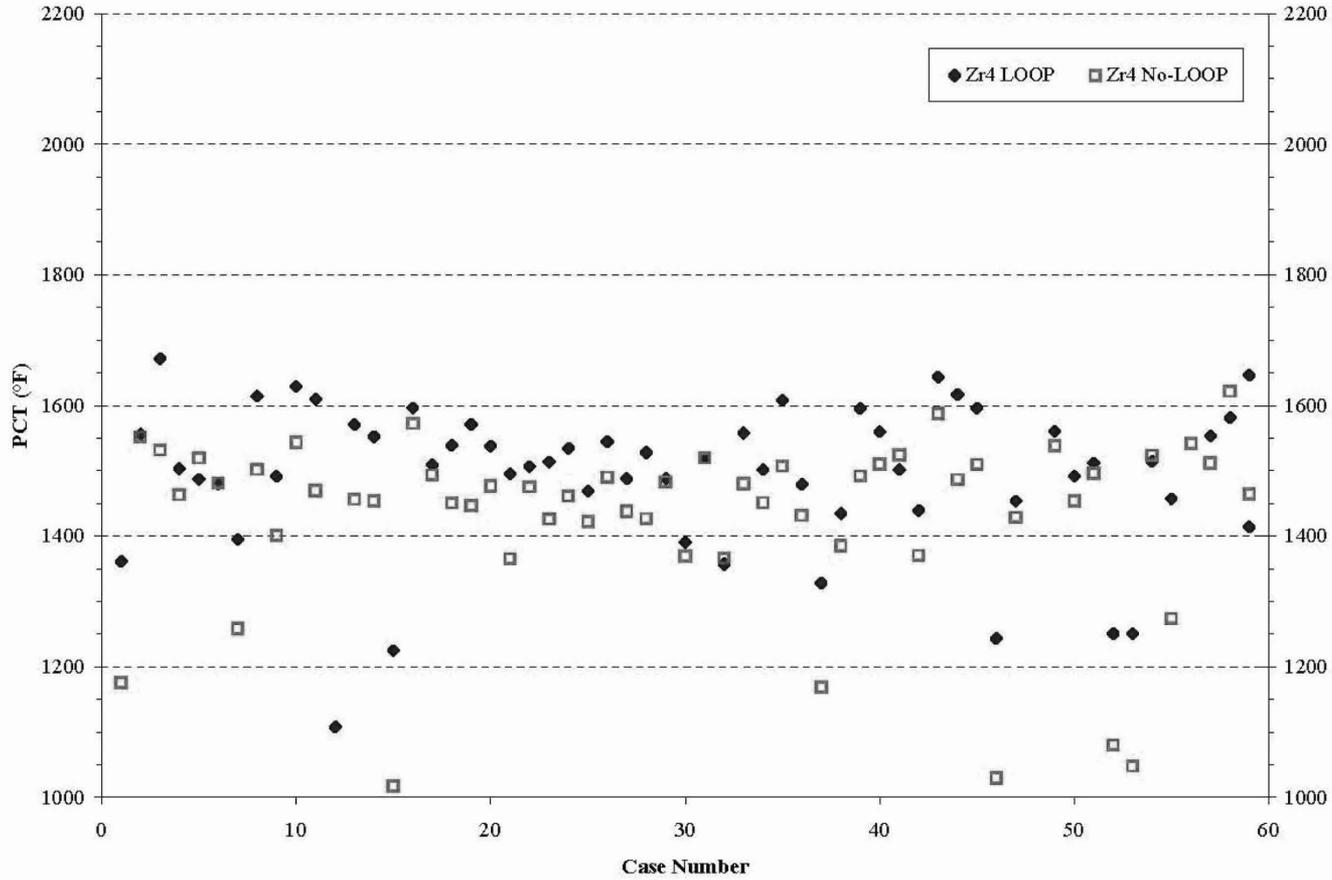


Figure 3-22 GDC 35 LOOP versus No-LOOP Cases

4.0 Generic Support for Transition Package

The following sections are responses to typical RAI questions posed by the NRC on EMF-2103 Revision 0 plant applications. In some instances, these requests cross-referenced documentation provided on dockets other than those for which the request is made. AREVA discussed these and similar questions from the NRC draft SER for Revision 1 of EMF-2103 in a meeting with the NRC on December 12, 2007. AREVA agreed to provide the following additional information within new submittals of a Realistic Large Break LOCA report.

4.1 Reactor Power

Question: *It is indicated in the RLBLOCA analyses that the assumed reactor core power "includes uncertainties." The use of a reactor power assumption other than 102 percent, regardless of BE or Appendix K methodology, is permitted by Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Appendix K.I.A, "Required and Acceptable Features of The Evaluation Models, 'Sources of Heat During a LOCA.'" However, Appendix K.I.A also states: "... An assumed power level lower than the level specified in this paragraph [1.02 times the licensed power level], (but not less than the licensed power level) may be used provided . . ." Please explain.*

Response: As indicated in Item 2.1 of Table 3-2 herein, the assumed reactor core power for the St Lucie Unit 1 Realistic Large Break Loss-of-coolant Accident is 3029 MWt. The value represents the 10% power uprate and 1.7% measurement uncertainty recapture (MUR) relative to the current rated thermal power (2700 MWt) plus 0.3% power measurement uncertainty.

4.2 Rod Quench

Question: *Does the version of S-RELAP5 used to perform the computer runs assure that the void fraction is less than 95 percent and the fuel cladding temperature is less than 900 °F before it allows rod quench?*

Response: Yes, the version of S-RELAP5 employed for the St Lucie Unit 1 requires that both the void fraction is less than 0.95 and the clad temperature is less than the minimum temperature for film boiling heat transfer (T_{min}) before the rod is allowed to quench. T_{min} is a sampled parameter in the RLBLOCA methodology that typically does not exceed 755 K (900

°F). This is a change to the approved RL BLOCA EM (Reference 1). This feature is carried forward into the UNOV07 version of S-RELAP5.

4.3 Rod-to-Rod Thermal Radiation

Question: *Provide justification that the S-RELAP5 rod-to-rod thermal radiation model applies to the St Lucie Unit 1 core.*

Response: The Realistic LBLOCA methodology, (Reference 1), does not provide modeling of rod-to-rod radiation. The fuel rod surface heat transfer processes included in the solution at high temperatures are: film boiling, convection to steam, rod to liquid radiation and rod to vapor radiation. This heat transfer package was assessed against various experimental data sets involving both moderate (1600 °F – 2000 °F) and high (2000 °F to over 2200 °F) peak cladding temperatures and shown to be conservative when applied nominally. The normal distribution of the experimental data was then determined. During the execution of an RL BLOCA evaluation, the heat transferred from a fuel rod is determined by the application of a multiplier to the nominal heat transfer model. This multiplier is determined by a random sampling of the normal distribution of the experimental data benchmarked. Because the data include the effects of rod-to-rod radiation, it is reasonable to conclude that the modeling implicitly includes an allocation for rod-to-rod radiation effects. As will be demonstrated, the approach is reasonable because the conditions within actual limiting fuel assemblies assure that the actual rod-to-rod radiation is larger than the allocation provided through normalization to the experiments.

The FLECHT-SEASET tests evaluated covered a range of PCTs from 1,651 to 2,239 °F and the THTF tests covered a range of PCTs from 1,000 to 2,200 °F. Since the test bundle in either FLECHT-SEASET or THTF is surrounded by a test vessel, which is relatively cool compared to the heater rods, substantial radiation from the periphery rods to the vessel wall can occur. The rods selected for assessing the RL BLOCA reflood heat transfer package were chosen from the interior of the test assemblies to minimize the impact of radiation heat transfer to the test vessel. The result was that the assessment rods comprise a set which is primarily isolated from cold wall effects by being surrounded by powered rods at reasonably high temperatures.

As a final assessment, three benchmarks independent of THTF and FLECHT-SEASET were performed. These benchmarks were selected from the Cylindrical Core Test Facility (CCTF), LOFT, and the Semiscale facilities. Because these facilities are more integral tests and

together cover a wide range of scale, they also serve to show that scale effects are accommodated within the code calculations.

The results of these calculations are provided in Section 4.3.4, Evaluation of Code Biases, page 4-100, of Reference 1. The CCTF results are shown in Figures 4.180 through 4.192, the LOFT results in Figures 4.193 through 4.201, and the Semiscale results in Figures 4.202 through 4.207. As expected, these figures demonstrate that the comparison between the code calculations and data is improved with the application of the derived biases. The CCTF, LOFT, and Semiscale benchmarks further indicate that, whatever consideration of rod-to-rod radiation is implicit in the S-RELAP5 reflood heat transfer modeling, it does not significantly effect code predictions under conditions where radiation is minimized. The measured PCTs in these assessments ranged from approximately 1,000 to 1,540 °F. At these temperatures, there is little rod-to-rod radiation. Given the good agreement between the biased code calculations and the CCTF, LOFT, and Semiscale data, it can be concluded that there is no significant over prediction of the total heat transfer coefficient.

Notwithstanding any conservatism evidenced by experimental benchmarks, the application of the model to commercial nuclear power plants provides some additional margins due to limitations within the experiments. The benchmarked experiments, FLECHET-SEASET and ORNL Thermal Hydraulic Test Facility (THTF), used to assess the S-RELAP5 heat transfer model, Reference 1, incorporated constant rod powers across the experimental assembly. Temperature differences that occurred were the result of guide tube, shroud or local heat transfer effects. In the operation of a pressurized water reactor (PWR) and in the RLBLOCA evaluation, a radial local peaking factor is present, creating power differences that tend to enhance the temperature differences between rods. In turn, these temperature differences lead to increases in net radiation heat transfer from the hotter rods. The expected rod-to-rod radiation will likely exceed that embodied within the experimental results.

4.3.1 Assessment of Rod-to-Rod Radiation Implicit in the RLBLOCA Methodology

As discussed above, the FLECHT-SEASET and THTF tests were selected to assess and determine the S-RELAP5 code heat transfer bias and uncertainty. Uniform radial power distribution was used in these test bundles. Therefore, the rod-to-rod temperature variation in the rods away from the vessel wall is caused primarily by the variation in the sub-channel fluid conditions. In the real operating fuel bundle, on the other hand, there can be 5 to 10 percent

rod-to-rod power variation. In addition, the methodology includes a provision to apply the uncertainty measurement to the hot pin. Table 4-1 provides the hot pin measurement uncertainty and a representative local pin peaking factor for several plants. These factors, however, relate the pin to the assembly average. To more properly assess the conditions under which rod-to-rod radiation heat transfer occurs, a more local peaking assessment is required. Therefore, the plant rod-to-rod radiation assessments herein set the average pin power for those pins surrounding the hot pin at 96 percent of that of the peak pin. For pins further removed the average power is set to 94 percent.

Table 4-1 Typical Measurement Uncertainties and Local Peaking Factors

Plant	F _H Measurement Uncertainty (percent)	Local Pin Peaking Factor (-)
1 4.0		1.068
2 4.0		1.050
3 6.0		1.149
4 4.0		1.113
5 4.25		1.135
6 4.0		1.058

4.3.2 Quantification of the Impact of Thermal Radiation using R2RRAD Code

The R2RRAD radiative heat transfer model was developed by Los Alamos National Laboratory (LANL) to be incorporated in the BWR version of the TRAC code. The theoretical basis for this code is given in References 8 and 11 and is similar to that developed in the HUXY rod heatup code (Reference 10, Section 2.1.2) used by AREVA for BWR LOCA applications. The version of R2RRAD used herein was obtained from the NRC to examine the rod-to-rod radiation characteristics of a 5x5 rod segment of the 16 1 rod FLECHT-SEASET bundle. The output provided by the R2RRAD code includes an estimate of the net radiation heat transfer from each rod in the defined array. The code allows the input of different temperatures for each rod as well as for a boundary surrounding the pin array. No geometry differences between pin locations are allowed. Even though this limitation affects the view factor calculations for guide tubes, R2RRAD is a reasonable tool to estimate rod-to-rod radiation heat transfer.

The FLECHT-SEASET test series was intended to simulate a 17x17 fuel assembly and there is a close similarity, Table 4-2, between the test bundle and a modern 17x17 assembly.

Table 4-2 FLECHT-SEASET & 17x17 FA Geometry Parameters

Design Parameter	FLECHT-SEASET	17x17 Fuel Assembly
Rod Pitch (in)	0.496	0.496
Fuel Rod Diameter (in)	0.374	0.374
Guide Tube Diameter (in)	0.474	0.482

Five FLECHT-SEASET tests (Reference 6) were selected for evaluation and comparison with expected plant behavior. Table 4-3 characterizes the results of each test. The 5x5 selected rod array comprises the hot rod, 4 guide tubes and 20 near adjacent rods. The simulated hot rod is rod 7J in the tests.

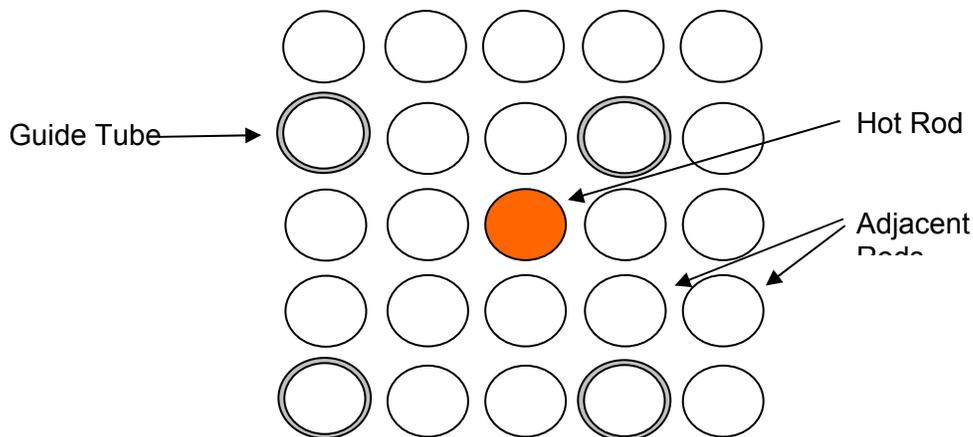


Figure 4-1 R2RRAD 5 x 5 Rod Segment

Two sets of runs were made simulating each of the five experiments and one set of cases was run to simulate the RLBLOCA evaluation of a limiting fuel assembly in an operating plant. For the simulation of Tests 31805, 31504, 31021, and 30817, the thimble tube (guide tube) temperatures were set to the measured values. For Test 34420, the thimble tube temperature was set equal to the measured vapor temperature. For the first experimental simulation set, the temperature of all 21 rods and the exterior boundary was set to the measured PCT of the simulated test. For the second experimental set, the hot rod temperature was set to the PCT

value and the remaining 20 rods and the boundary were set to a temperature 25 °F cooler providing a reasonable measure of the variation in surrounding temperatures. To estimate the rod-to-rod radiation in a real fuel assembly at LOCA conditions and compare it to the experimental results, each of the above cases was rerun with the hot rod PCT set to the experimental result and the remaining rods conservatively set to temperatures expected within the bundle. The guide tubes (thimble tubes) were removed for conservatism and because peak rod powers frequently occur at fuel assembly corners away from either guide tubes or instrument tubes. In line with the discussion in Section 4.3.1, the surrounding 24 rods were set to a temperature estimated for rods of 4 percent lower power. The boundary temperature was estimated based on an average power 6 percent below the hot rod power. For both of these, the temperature estimates were achieved using a ratio of pin power to the difference in temperature between the saturation temperature and the PCT.

$$T_{24 \text{ rods}} = 0.96 \cdot (PCT - T_{\text{sat}}) + T_{\text{sat}} \text{ and}$$

$$T_{\text{surrounding region}} = 0.94 \cdot (PCT - T_{\text{sat}}) + T_{\text{sat}}$$

T_{sat} was taken as 270 F.

Figure 4-2 shows the hot rod thermal radiation heat transfer for the two FLECHT-SEASET sets and for the plant set. The figure shows that for PCTs greater than about 1700 °F, the hot rod thermal radiation in the plant cases exceeds that of the same component within the experiments.

Table 4-3 FLECHT-SEASET Test Parameters

Test	Rod 7J PCT at 6-ft (°F)	PCT Time (s)	htc at PCTtime (Btu/hr-ft ² -°F)	Steam Temperature -at 7l (6-ft) (°F)	Thimble Temperature at 6-ft (°F)
34420 2205		34	10	1850	1850*
31805 2150		110	10	1800	1800
31504 2033		100	10	1750	1750
31021 1684		29	9	1400	1350
30817 1440		70	13	900	750
		* set to steam temp			

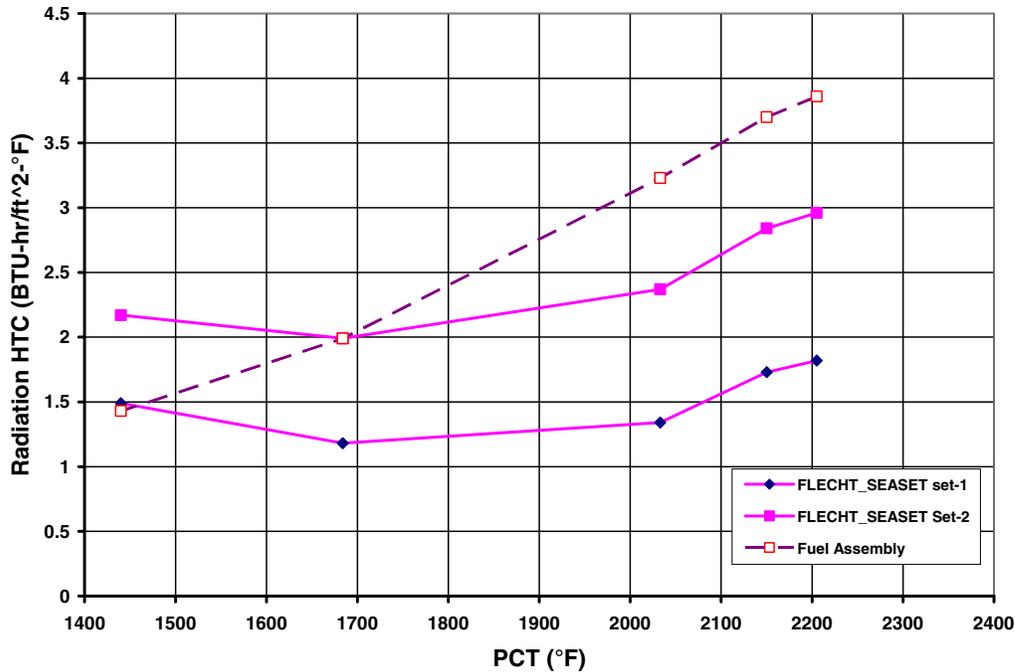


Figure 4-2 Rod Thermal Radiation in FLECHT-SEASET Bundle and in a 17x17 FA

4.3.3 Rod-to-Rod Radiation Summary

In summary, the conservatism of the heat transfer modeling established by benchmark can be reasonably extended to plant applications, and the plant local peaking provides a physical reason why rod-to-rod radiation should be more substantial within a plant environment than in the test environment. Therefore, the lack of an explicit rod-to-rod radiation model, in the version of S-RELAP5 applied for realistic LOCA calculations, does not invalidate the conclusion that the cladding temperature and local cladding oxidation have been demonstrated to meet the criteria of 10 CFR 50.46 with a high level of probability.

4.4 Film Boiling Heat Transfer Limit

Question: *In the St Lucie Unit 1 Cycle 24 calculations, is the Forslund-Rohsenow model contribution to the heat transfer coefficient limited to less than or equal to 15 percent when the void fraction is greater than or equal to 0.9?*

Response: Yes, the version of S-RELAP5 employed for the St Lucie Unit 1 RLBLOCA analysis limits the contribution of the Forslund-Rohsenow model to no more than 15 percent of the total heat transfer at and above a void fraction of 0.9. Because the limit is applied at a void fraction of 0.9, the contribution of Forslund-Rohsenow within the 0.7 to 0.9 interpolation range is limited to 15 percent or less. This is a change to the approved RLBLOCA EM (Reference 1). This feature is carried forward into the UNOV07 version of S-RELAP5.

4.5 Downcomer Boiling

Question: *If the PCT is greater than 1800°F or the containment pressure is less than 30 psia, has the St Lucie Unit 1 downcomer model been rebenchmarked by performing sensitivity studies, assuming adequate downcomer noding in the water volume, vessel wall and other heat structures?*

Response: The downcomer model for St Lucie Unit 1 has been established generically as adequate for the computation of downcomer phenomena including the prediction of potential local boiling effects. The model was benchmarked against the UPTF tests and the LOFT facility in the RLBLOCA methodology, Revision 0 (Reference 1). Further, AREVA addressed the effects of boiling in the downcomer in a letter, from James Malay to U.S. NRC, April 4, 2003. The letter cites the lack of direct experimental evidence but contains sensitivity studies on high and low pressure containments, the impact of additional azimuthal noding within the downcomer, and the influence of flow loss coefficients. Of these, the study on azimuthal noding is most germane to this question; indicating that additional azimuthal nodalization allows higher liquid buildup in portions of the downcomer away from the broken cold leg and increases the liquid driving head. Additionally, AREVA has conducted downcomer axial noding and wall heat release studies. Each of these studies supports the Revision 0 methodology and is documented later in this section.

This question is primarily concerned with the phenomena of downcomer boiling and the extension of the Revision 0 methodology and sensitivity studies to plants with low containment

pressures and high cladding temperatures. Boiling, wherever it occurs, is a phenomenon that codes like S-RELAP5 have been developed to predict. Downcomer boiling is the result of the release of energy stored in vessel metal mass. Within S-RELAP5, downcomer boiling is simulated in the nucleate boiling regime with the Chen correlation. This modeling has been validated through the prediction of several assessments on boiling phenomenon provided in the S-RELAP5 Code Verification and Validation document (Reference 12).

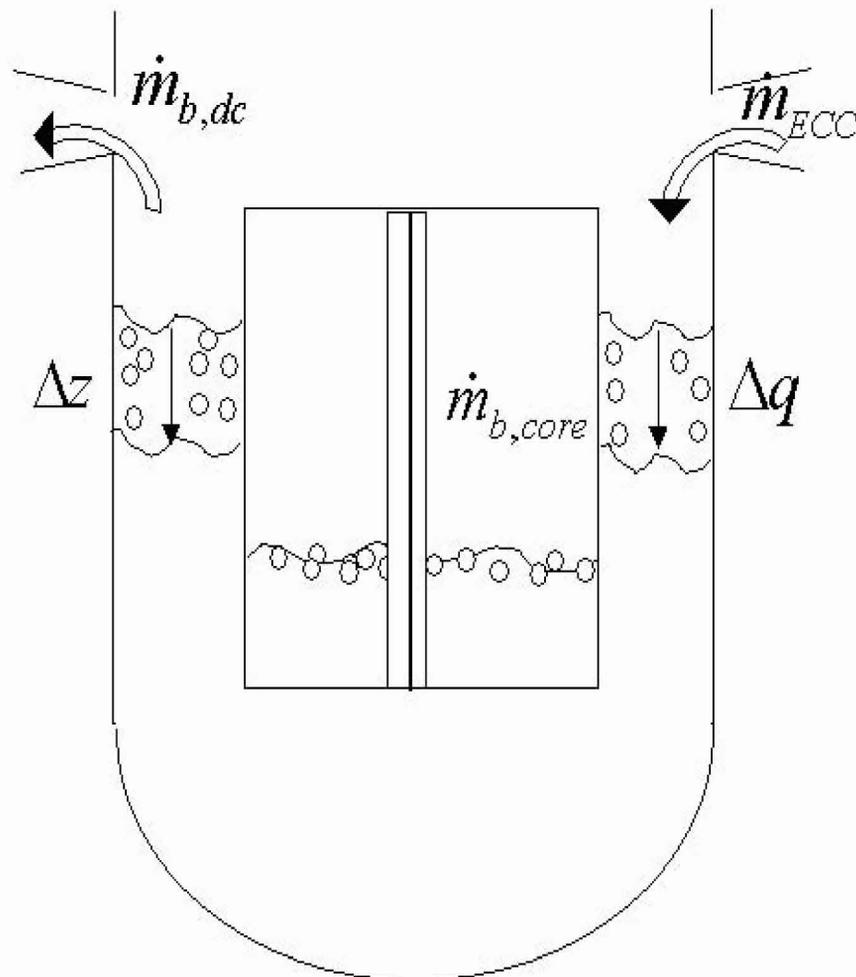


Figure 4-3 Reactor Vessel Downcomer Boiling Diagram

Hot downcomer walls penalize PCT by two mechanisms: by reducing subcooling of coolant entering the core and through the reduction in downcomer hydraulic head which is the driving force for core reflood. Although boiling in the downcomer occurs during blowdown, the biggest potential for impact on clad temperatures is during late reflood following the end of accumulator injection. At this time, there is a large step reduction in coolant flow from the ECC systems. As

a result, coolant entering the downcomer may be less subcooled. When the downcomer coolant approaches saturation, boiling on the walls initiates, reducing the downcomer hydraulic static level.

With the reduction of the downcomer level, the core inlet flow rate is reduced which, depending on the existing core inventory, may result in a cladding temperature excursion or a slowing of the core cooldown rate.

While downcomer boiling may impact clad temperatures, it is somewhat of a self-limiting process. If cladding temperatures increase, less energy is transferred in the core boiling process and the loop steam flows are reduced. This reduces the required driving head to support continued core reflood and reduces the steam available to heat the ECCS water within the cold legs resulting in greater subcooling of the water entering the downcomer.

The impact of downcomer boiling is primarily dependent on the wall heat release rate and on the ability to slip steam up the downcomer and out of the break. The higher the downcomer wall heat release, the more steam is generated within the downcomer and the larger the impact on core reflooding. Similarly, the quicker the passage of steam up the downcomer, the less resident volume within the downcomer is occupied by steam and the lower the impact on the downcomer average density. Therefore, the ability to properly simulate downcomer boiling depends on both the heat release (boiling) model and on the ability to track steam rising through the downcomer. Consideration of both of these is provided in the following text. The heat release modeling in S-RELAP5 is validated by a sensitivity study on wall mesh point spacing and through benchmarking against a closed form solution. Steam tracking is validated through both an axial and an azimuthal fluid control volume sensitivity study done at low pressures. The results indicate that the modeling accuracy within the RLBLOCA methodology is sufficient to resolve the effects of downcomer boiling and that, to the extent that boiling occurs; the methodology properly resolves the impact on the cladding temperature and cladding oxidation rates.

4.5.1 Wall Heat Release Rate

The downcomer wall heat release rate during reflood is conduction limited and depends on the vessel wall mesh spacing used in the S-RELAP5 model. The following two approaches are used

to evaluate the adequacy of the downcomer vessel wall mesh spacing used in the S-RELAP5 model.

4.5.1.1 Exact Solution

In this benchmark, the downcomer wall is considered as a semi-infinite plate. Because the benchmark uses a closed form solution to verify the wall mesh spacing used in S-RELAP5, it is assumed that the material has constant thermal properties, is initially at temperature T_i , and, at time zero, has one surface, the surface simulating contact with the downcomer fluid, set to a constant temperature, T_o , representing the fluid temperature. Section 4.3 of Reference 9 gives the exact solution for the temperature profile as a function of time as

$$(T(x,t) - T_o) / (T_i - T_o) = \text{erf} \{x / (2 \cdot (\alpha t)^{0.5})\}, \quad (1)$$

where, α is the thermal diffusivity of the material given by

$$\alpha = k / (\rho C_p),$$

k = thermal conductivity,

ρ = density,

C_p = specific heat, and

$\text{erf}\{\}$ is the Gauss error function (given in Table A-1 of Reference 9).

The conditions of the benchmark are $T_i = 500$ °F and $T_o = 300$ °F. The mesh spacing in S-RELAP5 is the same as that used for the downcomer vessel wall in the RLBL LOCA model. Figure 4-4 shows the temperature distributions in the metal at 0.0, 100 and 300 seconds as calculated by using Equation 1 and S-RELAP5, respectively. The solutions are identical confirming the adequacy of the mesh spacing used in the downcomer wall.

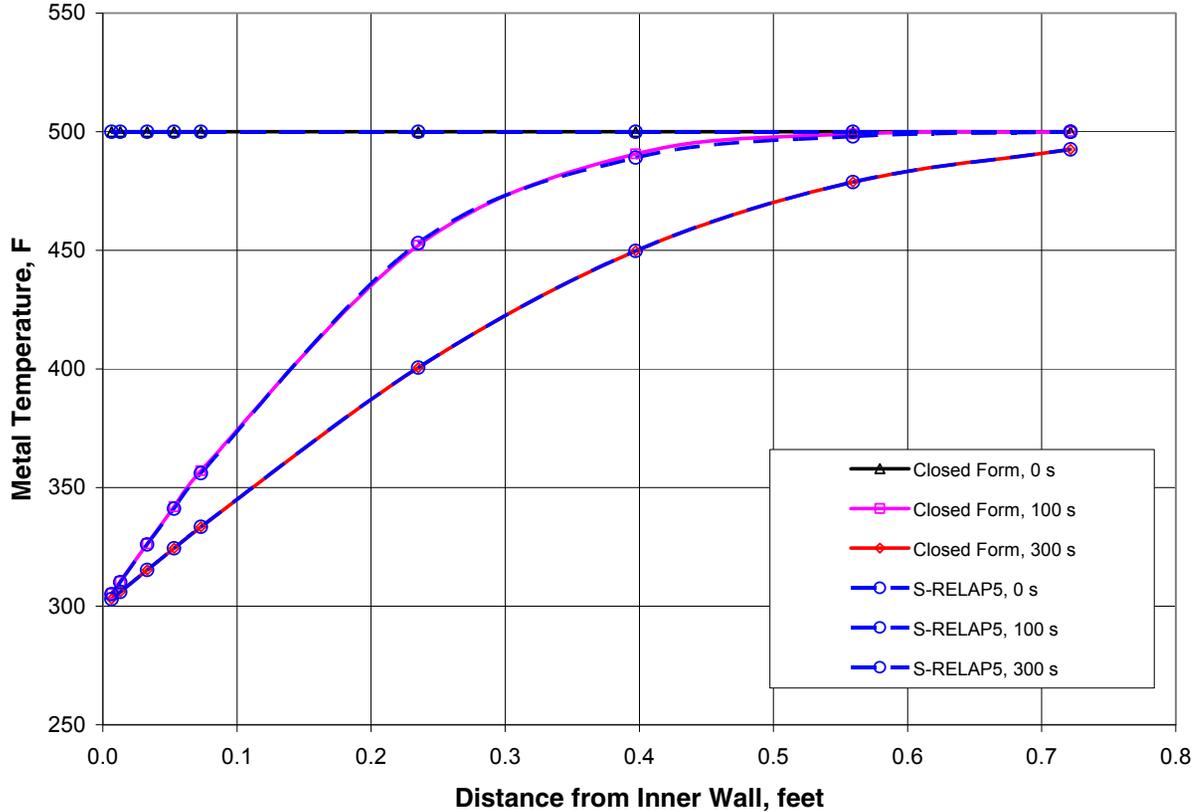


Figure 4-4 S-RELAP5 versus Closed Form Solution

4.5.1.2 Plant Model Sensitivity Study

As additional verification, a typical 4-loop plant case was used to evaluate the adequacy of the mesh spacing within the downcomer wall heat structure. Each mesh interval in the base case downcomer vessel wall was divided into two equal intervals. Thus, a new input model was created by increasing the number of mesh intervals from 9 to 18. The following four figures show the total downcomer metal heat release rate, PCT independent of elevation, downcomer liquid level, and the core liquid level, respectively, for the base case and the modified case. These results confirm the conclusion from the exact solution study that the mesh spacing used in the plant model for the downcomer vessel wall is adequate.

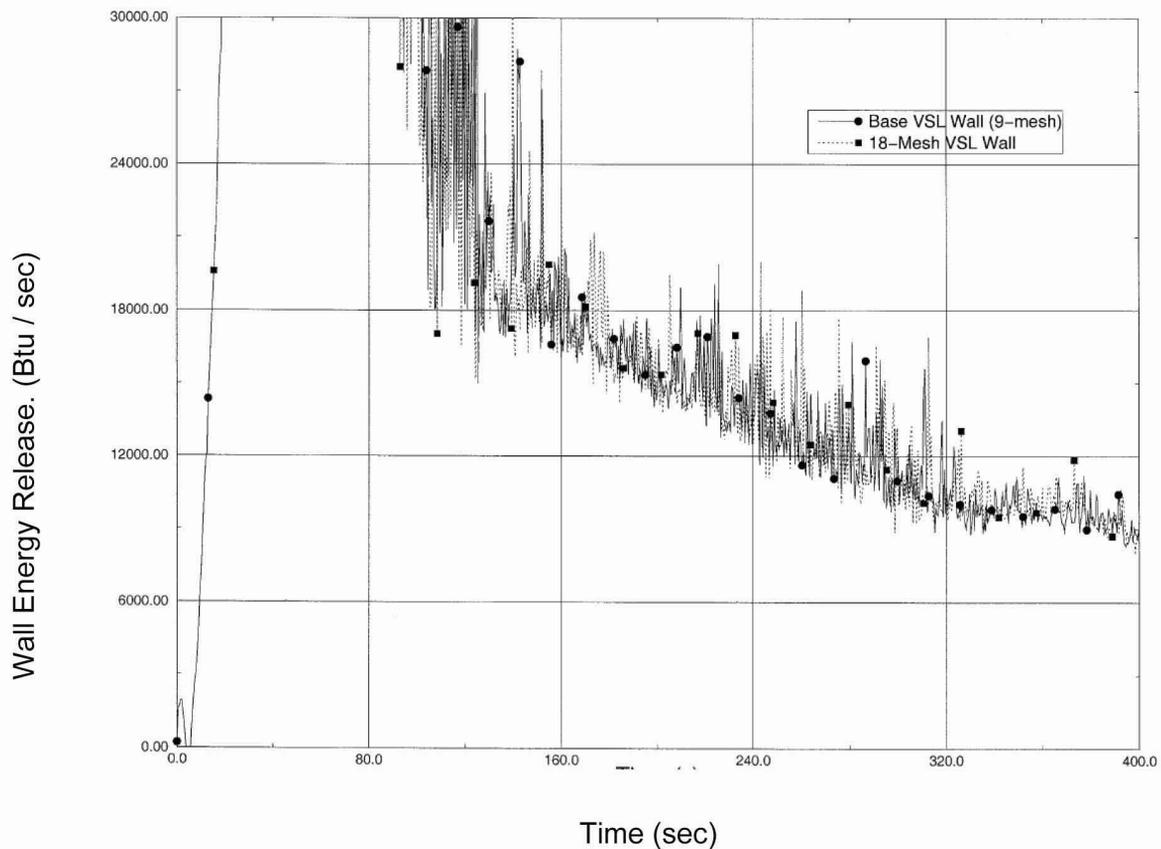


Figure 4-5 Downcomer Wall Heat Release – Wall Mesh Point Sensitivity

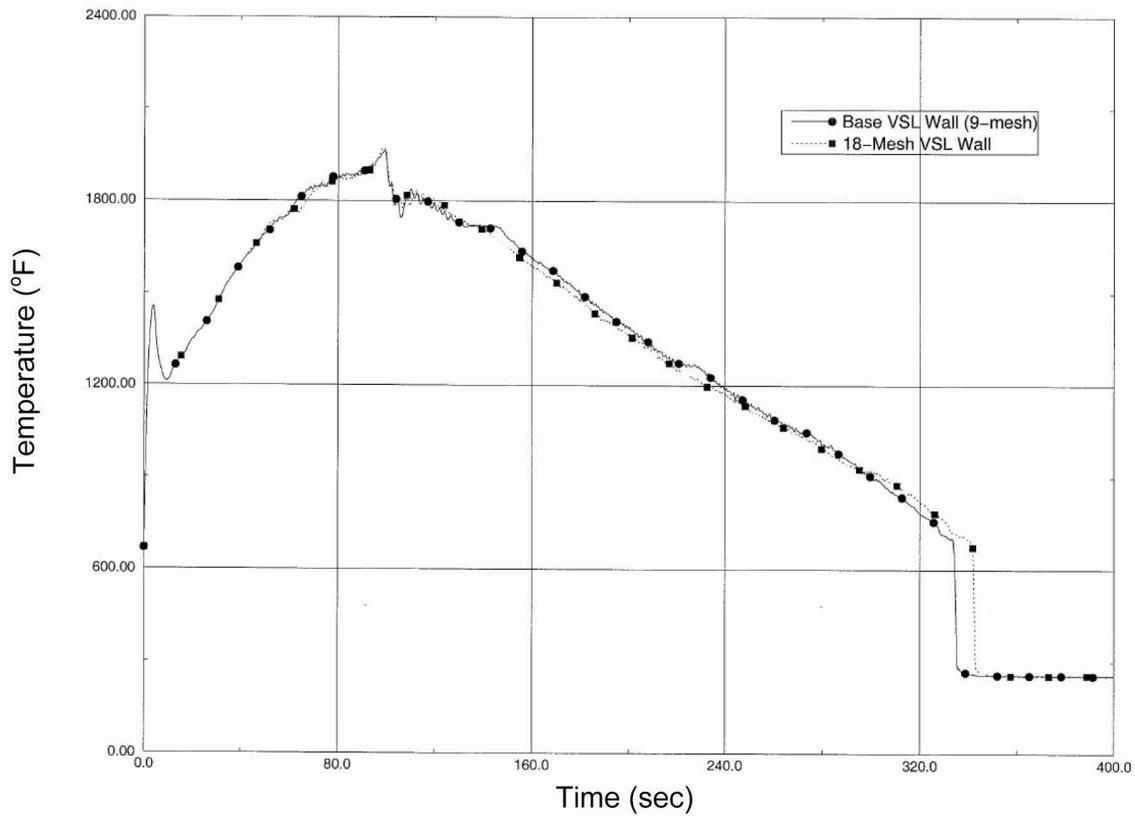


Figure 4-6 PCT Independent of Elevation – Wall Mesh Point Sensitivity

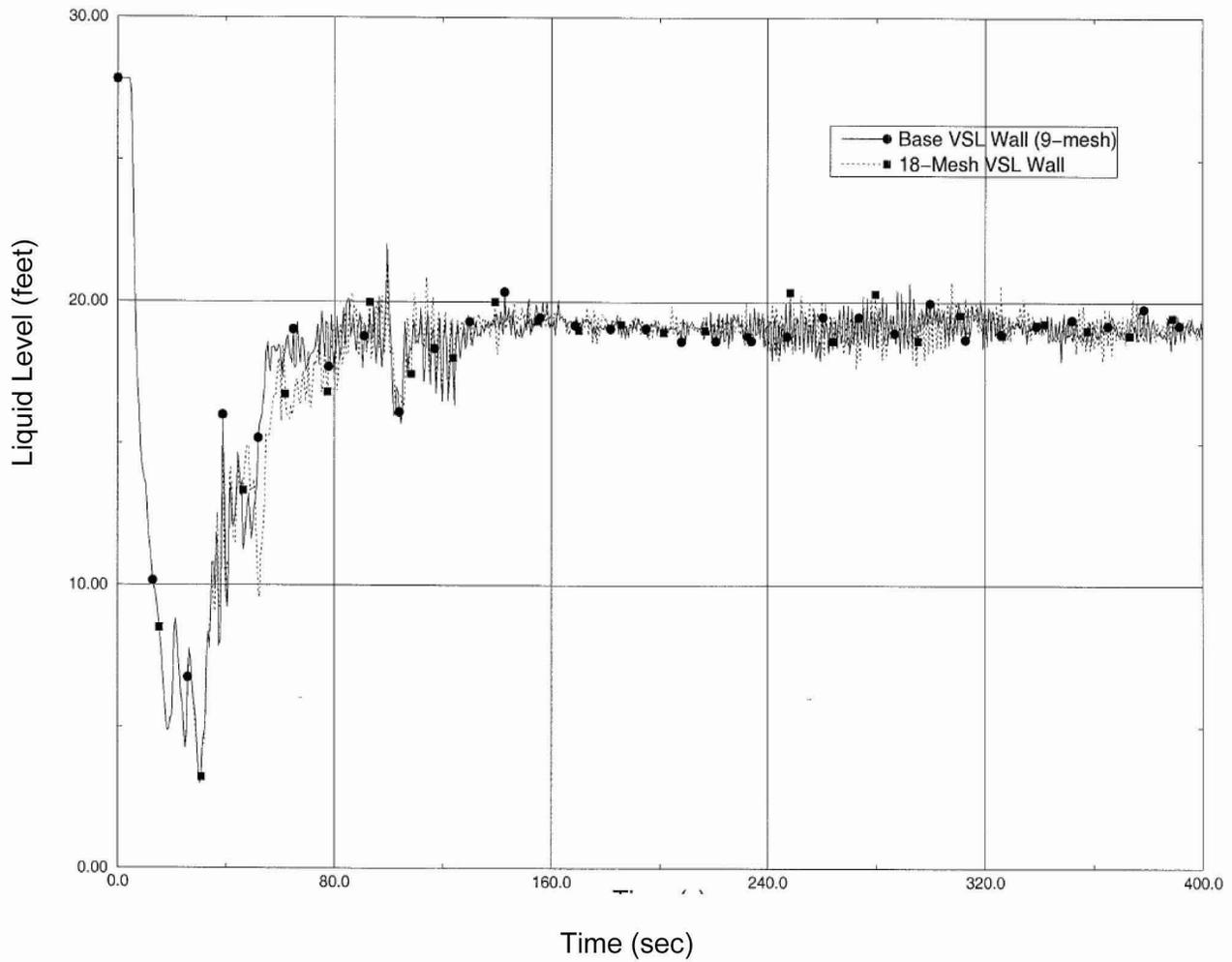


Figure 4-7 Downcomer Liquid Level – Wall Mesh Point Sensitivity

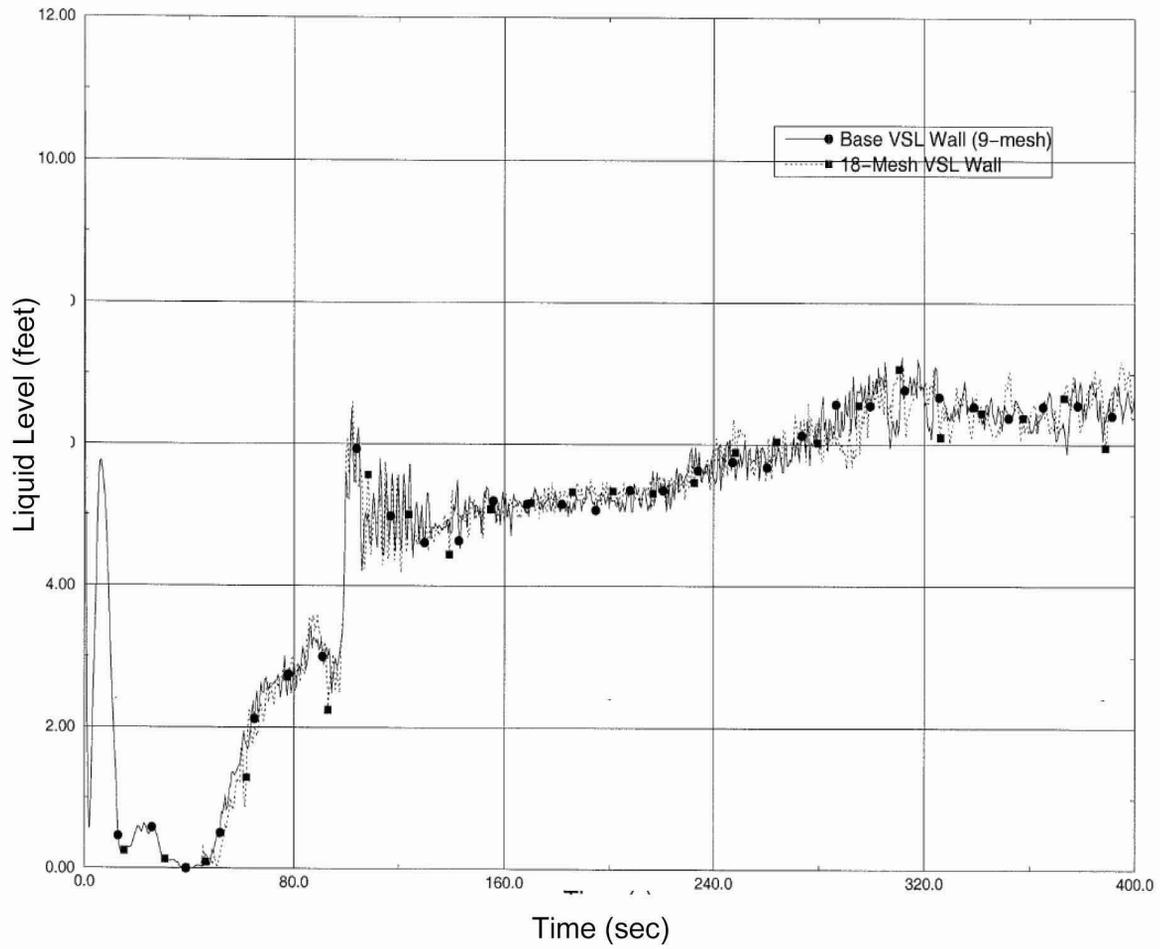


Figure 4-8 Core Liquid Level – Wall Mesh Point Sensitivity

4.5.2 Downcomer Fluid Distribution

To justify the adequacy of the downcomer nodalization in calculating the fluid distribution in the downcomer, two studies varying separately the axial and the azimuthal resolution with which the downcomer is modeled have been conducted.

4.5.2.1 Azimuthal Nodalization

In a letter to the NRC dated April, 2003 (Reference 1), AREVA documented several studies on downcomer boiling. Of significance here is the study on further azimuthal break up of the downcomer noding. The study, based on a 3-loop plant with a containment pressure of approximately 30 psia during reflood, consisted of several calculations examining the effects on clad temperature and other parameters.

The base model, with 6 axial by 3 azimuthal regions, was expanded to 6 axial by 9 azimuthal regions (Figure 4-9). The base calculation simulated the limiting PCT calculation given in the EMF-2103 three-loop sample problem. This case was then repeated with the revised 6 x 9 downcomer noding.

The change resulted in an alteration of the blowdown evolution of the transient with little evidence of any effect during reflood. To isolate any possible reflood impact that might have an influence on downcomer boiling, the case was repeated with a slightly adjusted vessel-side break flow. Again, little evidence of impact on the reflood portion of the transient was observed.

The study concluded that blowdown or near blowdown events could be impacted by refining the azimuthal resolution in the downcomer but that reflood would not be impacted. Although the study was performed for a somewhat elevated system pressure, the flow regimes within the downcomer will not differ for pressures as low as atmospheric. Thus, the azimuthal downcomer modeling employed for the RLBLOCA methodology is reasonably converged in its ability to represent downcomer boiling phenomena.

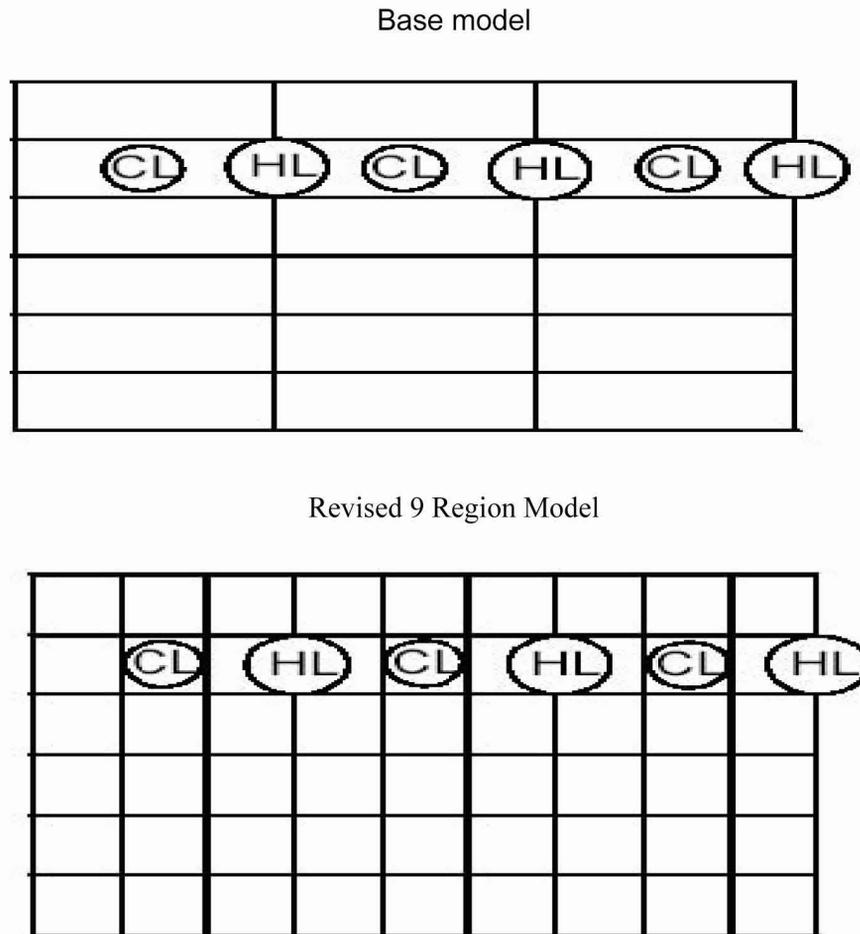


Figure 4-9 Azimuthal Noding

4.5.2.2 Axial Nodalization

The RLBLOCA methodology divides the downcomer into six nodes axially. In both 3-loop and 4-loop models, the downcomer segment at the active core elevation is represented by two equal length nodes. For most operating plants, the active core length is 12 feet and the downcomer segments at the active core elevation are each 6-feet high. (For a 14 foot core, these nodes would be 7-feet high.) The model for the sensitivity study presented here comprises a 3-loop plant with an ice condenser containment and a 12 foot core. For the study, the two nodes spanning the active core height are divided in half, revising the model to include eight axial

nodes. Further, the refined noding is located within the potential boiling region of the downcomer where, if there is an axial resolution influence, the sensitivity to that impact would be greatest.

The results show that the axial noding used in the base methodology is sufficient for plants experiencing the very low system pressures characteristic of ice condenser containments. Figure 4-10 provides the containment back pressure for the base modeling. Figure 4-11 through Figure 4-14 show the total downcomer metal heat release rate, PCT independent of elevation, downcomer liquid level, and the core liquid level, respectively, for the base case and the modified case.

The results demonstrate that the axial resolution provided in the base case, 6 axial downcomer node divisions with 2 divisions spanning the core active region, are sufficient to accurately resolve void distributions within the downcomer. Thus, this modeling is sufficient for the prediction of downcomer driving head and the resolution of downcomer boiling effects.

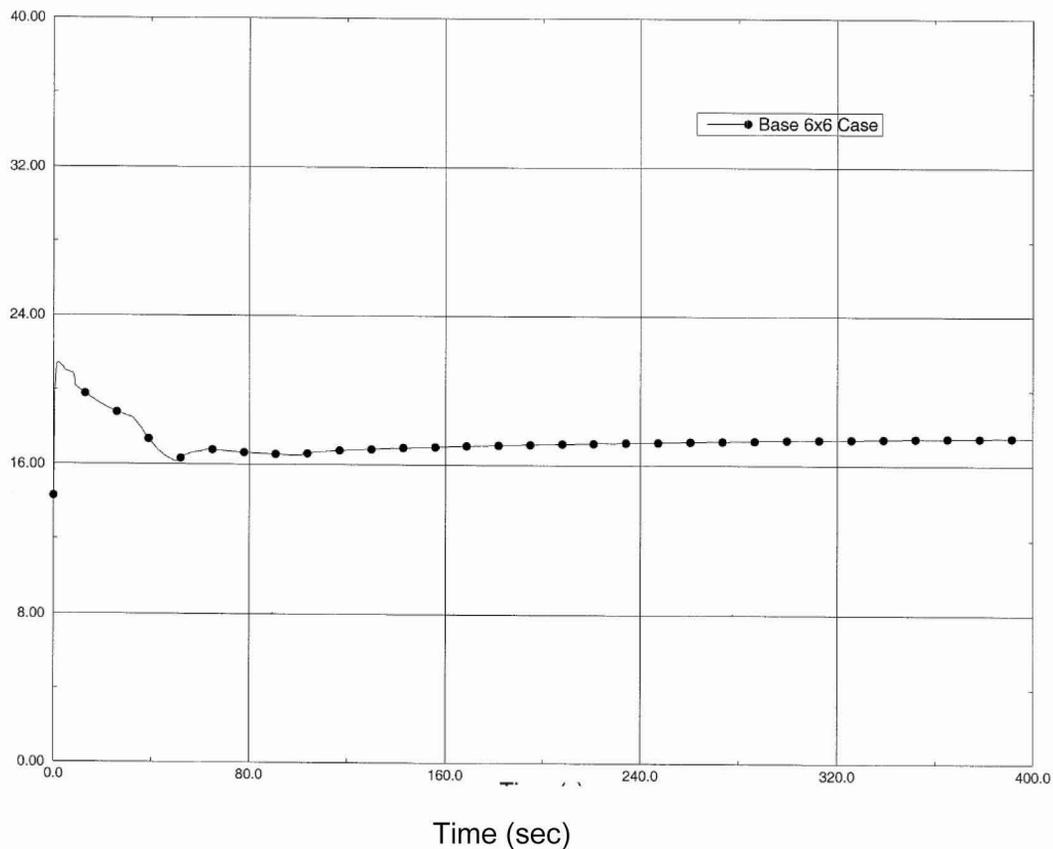


Figure 4-10 Lower Compartment Pressure versus Time

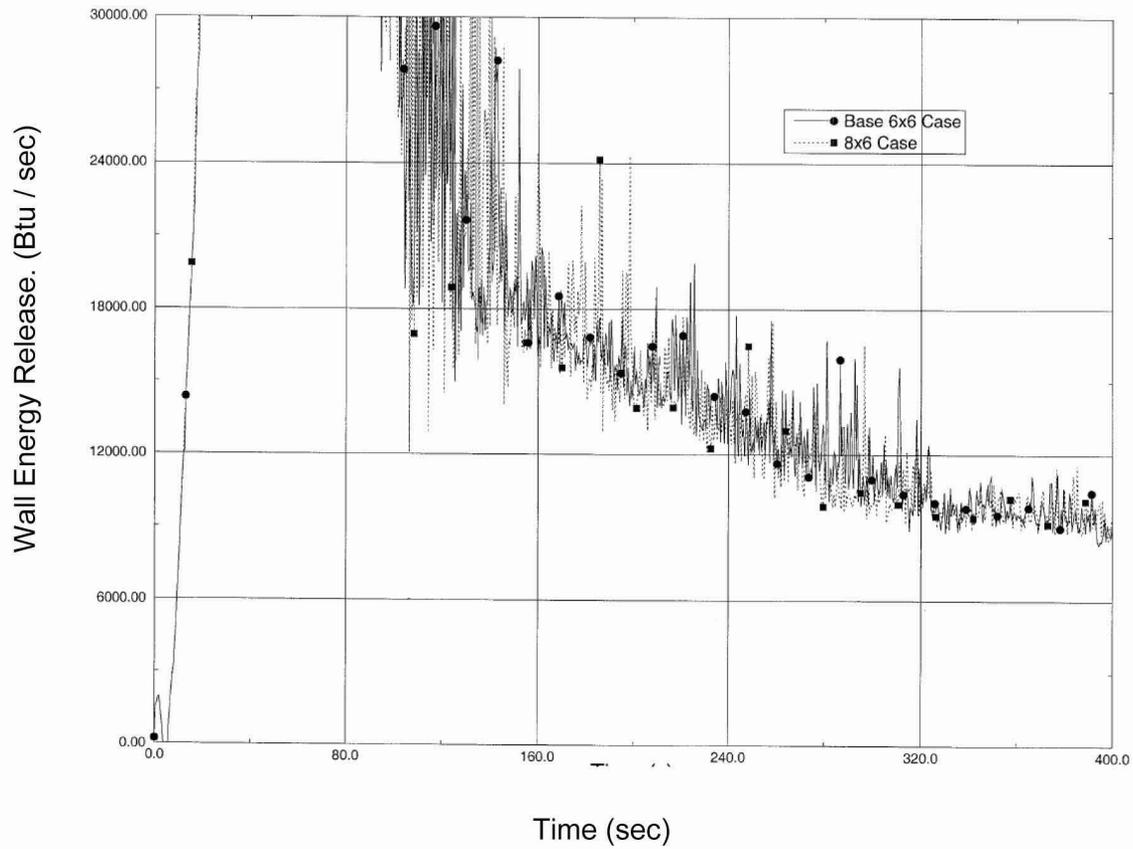


Figure 4-11 Downcomer Wall Heat Release – Axial Noding Sensitivity Study

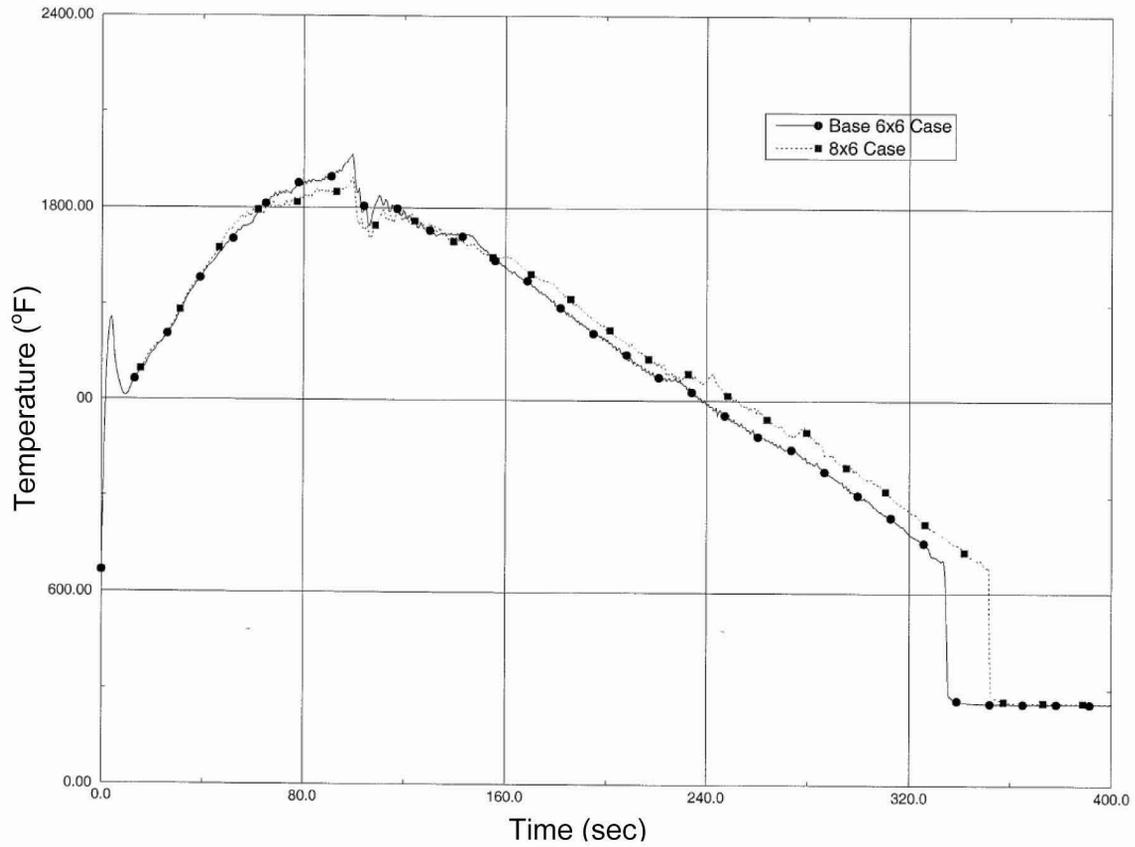


Figure 4-12 PCT Independent of Elevation – Axial Noding Sensitivity Study

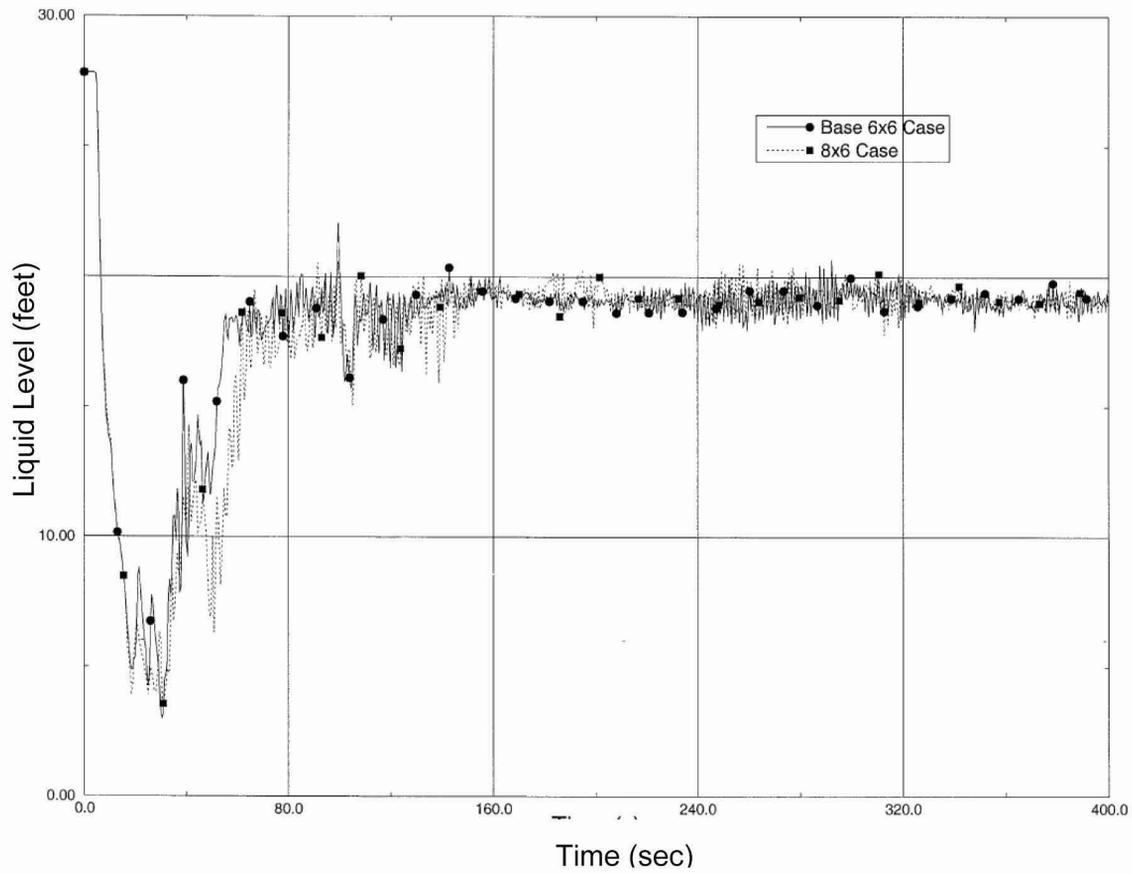


Figure 4-13 Downcomer Liquid Level – Axial Noding Sensitivity Study

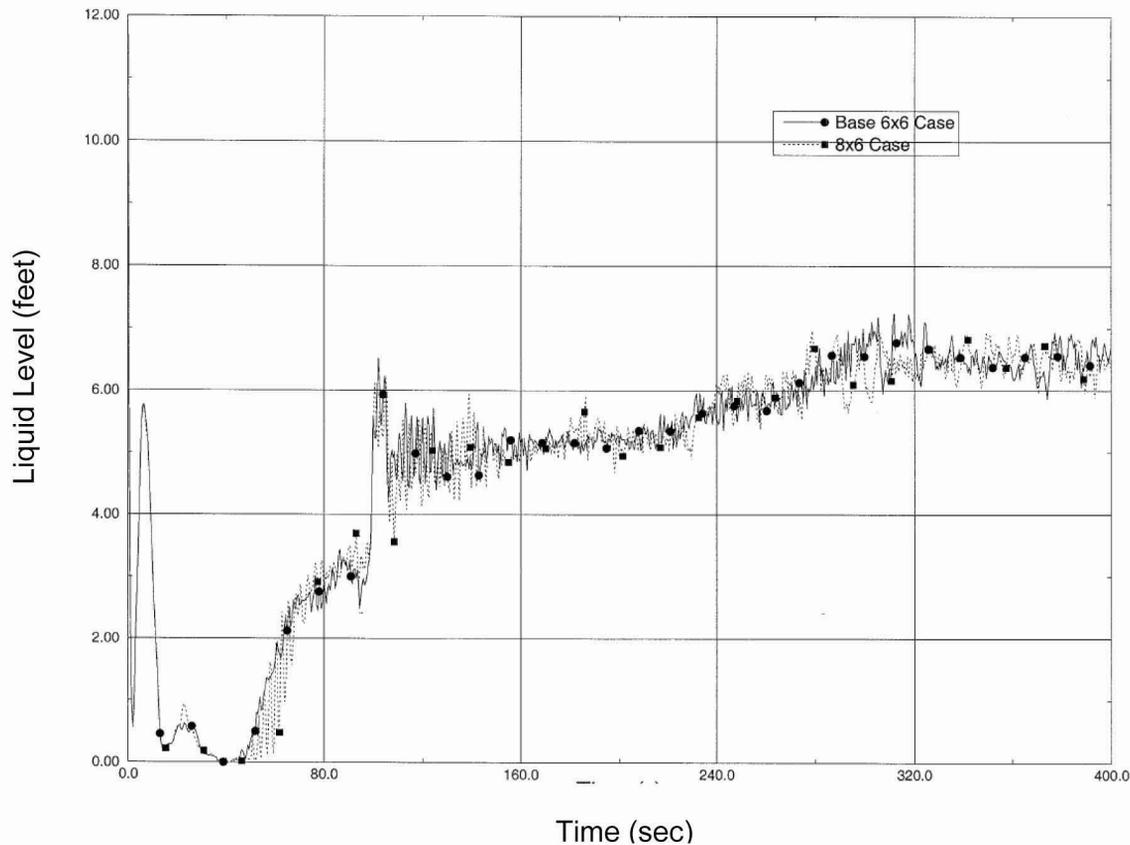


Figure 4-14 Core Liquid Level – Axial Noding Sensitivity Study

4.5.3 Downcomer Boiling Conclusions

To further justify the ability of the RLBLOCA methodology to predict the potential for and impact of downcomer boiling, studies were performed on the downcomer wall heat release modeling within the methodology and on the ability of S-RELAP5 to predict the migration of steam through the downcomer. Both azimuthal and axial noding sensitivity studies were performed. The axial noding study was based on an ice condenser plant that is near atmospheric pressure during reflood. These studies demonstrate that S-RELAP5 delivers energy to the downcomer liquid volumes at an appropriate rate and that the downcomer noding detail is sufficient to track the distribution of any steam formed. Thus, the required methodology for the prediction of downcomer boiling at system pressures approximating those achieved in plants with pressures as low as ice condenser containments has been demonstrated.

4.6 Break Size

Question: *Were all break sizes assumed greater than or equal to 1.0 ft²?*

Response: Yes.

The NRC has requested that the break spectrum for the realistic LOCA evaluations be limited to accidents that evolve through a range of phenomena similar to those encountered for the larger break area accidents. This is a change to the approved RLBLOCA EM (Reference 1). The larger break area LOCAs are typically characterized by the occurrence of dispersed flow film boiling at the hot spot, which sets them apart from smaller break LOCAs. This occurs generally in the vicinity of 0.2 DEGB (double-ended guillotine break) size (i.e., 0.2 times the total flow area of the pipe on both sides of the break). However, this transitional break size varies from plant to plant and is verified only after the break spectrum has been executed. AREVA NP has sought to develop sufficient criteria for defining the minimum large break flow area prior to performing the break spectrum. The purpose for doing so is to assure a valid break spectrum is performed.

4.6.1 Break / Transient Phenomena

In determining the AREVA NP criteria, the characteristics of larger break area LOCAs are examined. These LOCA characteristics involve a rapid and chaotic depressurization of the reactor coolant system (RCS) during which the three historical approximate states of the system can be identified.

Blowdown_____ The blowdown phase is defined as the time period from initiation of the break until flow from the accumulators begins. This definition is somewhat different from the traditional definition of blowdown which extends the blowdown until the RCS pressure approaches containment pressure. The blowdown phase typically lasts about 12 to 25 seconds, depending on the break size.

Refill _____ is that period that starts with the end of blowdown, whichever definition is used, and ends when water is first forced upward in to the core. During this phase the core experiences a near adiabatic heatup.

Reflood_____ is that portion of the transient that starts with the end of refill, follows through the filling of the core with water and ends with the achievement of complete core quench.

Implicit in this break-down is that the core liquid inventory has been completely, or nearly so, expelled from the primary system leaving the core in a state of near core-wide dispersed flow film boiling and subsequent adiabatic heatup prior to the reflood phase. Although this break down served as the basis for the original deterministic LOCA evaluation approaches and is valid for most LOCAs that would classically be termed large breaks, as the break area decreases the depressurization rate decreases such that these three phases overlap substantially. During these smaller break events, the core liquid inventory is not reduced as much as that found in larger breaks. Also, the adiabatic core heatup is not as extensive as in the larger breaks which results in much lower cladding temperature excursions.

4.6.2 New Minimum Break Size Determination

No determination of the lower limit can be exact. The values of critical phenomena that control the evolution of a LOCA transient will overlap and interplay. This is especially true in a statistical evaluation where parameter values are varied randomly with a strong expectation that the variations will affect results. In selecting the lower area of the RLBLOCA break spectrum, AREVA sought to preserve the generality of a complete or nearly complete core dry out accompanied by a substantially reduced lower plenum liquid inventory. It was reasoned that such conditions would be unlikely if the break flow rate was reduced to less than the reactor coolant pump flow. That is, if the reactor coolant pumps are capable of forcing more coolant toward the reactor vessel than the break can extract from the reactor vessel, the downcomer and core must maintain some degree of positive flow (positive in the normal operations sense). The circumstance is, of course, transitory. Break flow is altered as the RCS blows down and the RC pump flow may decrease as the rotor and flywheel slow down if power is lost. However, if the core flow was reduced to zero or became negative immediately after the break initiation, then the event was quite likely to proceed with sufficient inertia to expel most of the reactor vessel liquid to the break. The criteria base, thus established, consists of comparing the break flow to the initial flow through all reactor coolant pumps and setting the minimum break area such that these flows match. This is done as follows:

$$W_{\text{break}} = A_{\text{break}} * G_{\text{break}} = N_{\text{pump}} * W_{\text{RCP}}$$

This gives

$$A_{\text{break}} = (N_{\text{pump}} * W_{\text{RCP}}) / G_{\text{break}}$$

The break mass flux is determined from critical flow. Because the RCS pressure in the broken cold leg will decrease rapidly during the first few seconds of the transient, the critical mass flux is averaged between that appropriate for the initial operating conditions and that appropriate for the initial cold leg enthalpy and the saturation pressure of coolant at that enthalpy.

$$G_{\text{break}} = (G_{\text{break}}(P_0, H_{\text{CL0}}) + G_{\text{break}}(P_{\text{CLsat}}, H_{\text{CL0}})) / 2.$$

The estimated minimum LBLOCA break area, A_{min} , is 2.94 ft² and the break area percentage, based on the full double-ended guillotine break total area, is 29.97 percent.

Table 4-4 provides a listing of the plant type, initial condition, and the fractional minimum RLLOCA break area, for all the plant types presented as generic representations in the next section.

Table 4-4 Minimum Break Area for Large Break LOCA Spectrum

	Plant Description	System Pressure (psia)	Cold Leg Enthalpy (Btu/lbm)	Subcooled G_{break} (lbm/ft ² -s)	Saturated G_{break} (HEM) (lbm/ft ² -s)	RCP flow (lbm/s)	Spectrum Minimum Break Area (ft ²)	Spectrum Minimum Break Area (DEGB)
A	3-Loop W Design (sha)	2250	555.0 231	90	5700	31417	2.18	0.26
B	3-Loop W Design (rob)	2250	544.5 233	80	5450	28124	1.92	0.23
C	3-Loop W Design (nab)	2250	550.0 235	40	5580	29743	2.04	0.25
D	2x4 CE Design (ftc)	2100	538.8 223	60	5310	21522	1.53	0.24
E	2x4 CE Design (pal)	2055	535.8 226	30	5230	37049	2.66	0.27
F	4-loop W Design (seq2)	2160 540.9		23290	5370	39500	2.76	0.33

The split versus double-ended break type is no longer related to break area. In concurrence with Regulatory Guide 1.157, both the split and the double-ended break will range in area between the minimum break area (A_{min}) and an area of twice the size of the broken pipe. The determination of break configuration, split versus double-ended, is made after the break area is selected based on a uniform probability for each occurrence.

4.6.3 Intermediate Break Size Disposition

With the revision of the smaller break area for the RLBLOCA analysis, the break range for small breaks and large breaks are no longer contiguous. Typically the lower end of the large break spectrum occurs at between 0.2 to 0.3 times the total area of a 100 percent double-ended guillotine break (DEGB) and the upper end of the small break spectrum occurs at a approximately 0.05 times the area of a 100 percent DEGB. This leaves a range of breaks that are not specifically analyzed during a LOCA licensing analysis. The premise for allowing this gap is that these breaks do not comprise accidents that develop high cladding temperature and thus do not comprise accidents that critically challenge the emergency core cooling systems (ECCS). Breaks within this range remain large enough to blowdown to low pressures. Resolution is provided by the large break ECC systems and the pressure-dependent injection limitations that determine critical small break performance are avoided. Further, these accidents develop relatively slowly, assuring maximum effectiveness of those ECC systems.

A variety of plant types for which analysis within the intermediate range have been completed were surveyed. Although statistical determinations are extracted from the consideration of breaks with areas above the intermediate range, the AREVA best-estimate methodology remains suitable to characterize the ECCS performance of breaks within the intermediate range. Table 4-4 provides a listing of the plant type, initial condition, and the fractional minimum RLBLOCA break area. Figure 4-15 through Figure 4-20 provide the enlarged break spectrum results with the upper end of the small break spectrum and the lower end of the large break spectrum indicated by bars. Table 4-5 provides differences between the true large break region and the intermediate break region (break areas between that of the largest SBLOCA and the smallest RLBLOCA). The minimum difference is 14 °F; however, this case is not representative of the general trend shown by the other comparisons. The next minimum difference is 704 °F (see Figure 4-15). Considering this point as an outlier, the table shows the minimum difference between the highest intermediate break spectrum PCT and large break spectrum PCT, for the six plants, is at least 463 °F, and including this point would provide an average difference of 427 °F and a maximum difference of 840 °F.

Thus, by both measures, the peak cladding temperatures within the intermediate break range will be several hundred degrees below those in the true large break range. Therefore, these breaks will not provide a limit or a critical measure of the ECCS performance. Given that the

large break spectrum bounds the intermediate spectrum, the use of only the large break spectrum meets the requirements of 10CFR50.46 for breaks within the intermediate break LOCA spectrum, and the method demonstrates that the ECCS for a plant meets the criteria of 10CFR50.46 with high probability.

Table 4-5 Minimum PCT Temperature Difference – True Large and Intermediate Breaks

Plant Description	Generic Plant Label (Table 4-4)	Maximum PCT (°F) Intermediate Size Break	Maximum PCT (°F) Large Size Break	Delta PCT (°F)	Average Delta PCT (°F)
3-Loop W Design	A 1746	1	1887 141	1	427 ¹
	B 1273		1951	678	
	C 1326		1789	463	
2x4 CE Design	D 984		1751	767	767
	E 869		1636	767	
3-loop W Design	F 1127		1967	840	840

Note: 1. The 2nd highest PCT was 1183 °F. This changes the Delta PCT to 704 °F and the average delta increases to 615 °F.

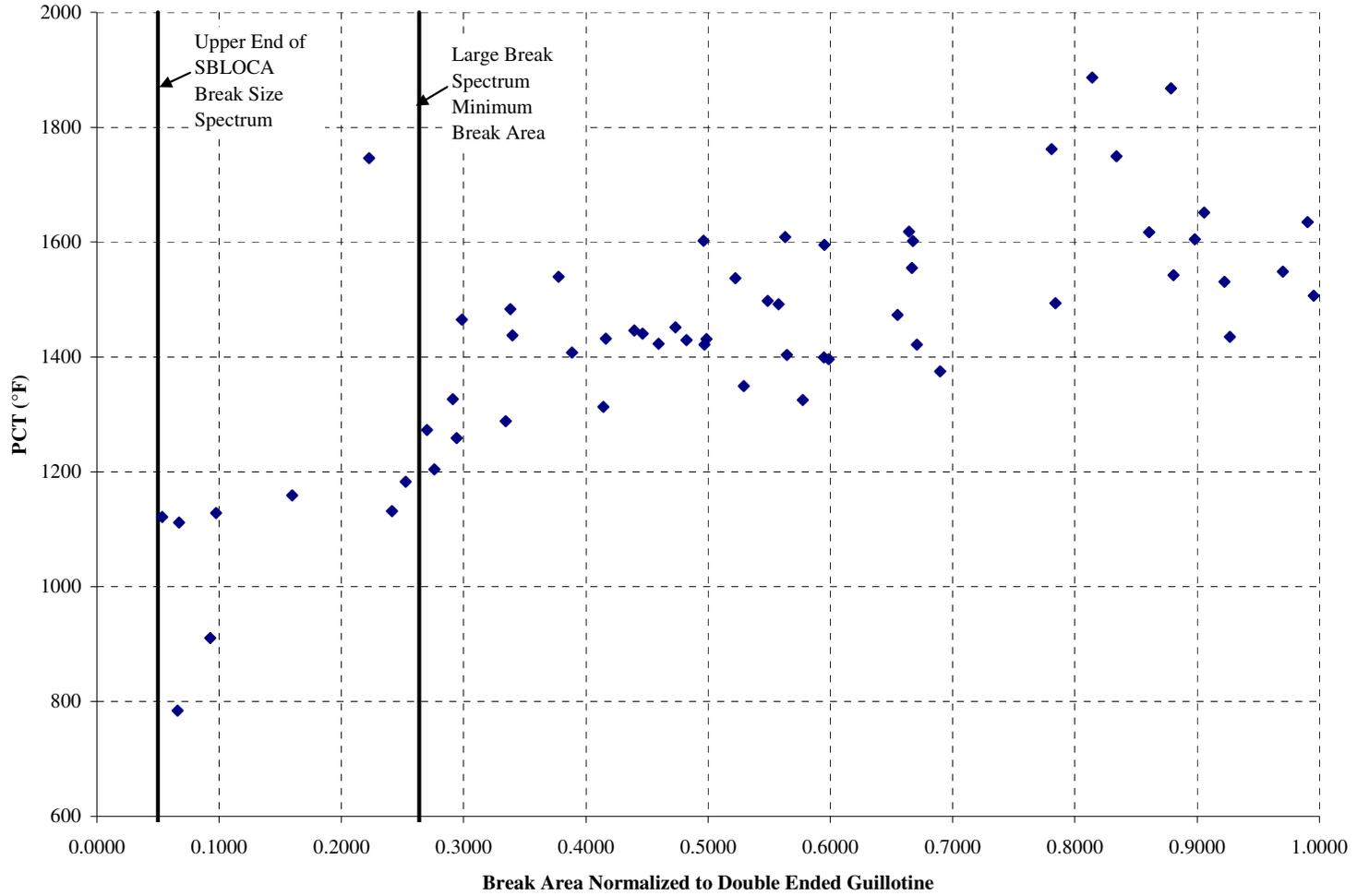


Figure 4-15 Plant A – Westinghouse 3-Loop Design

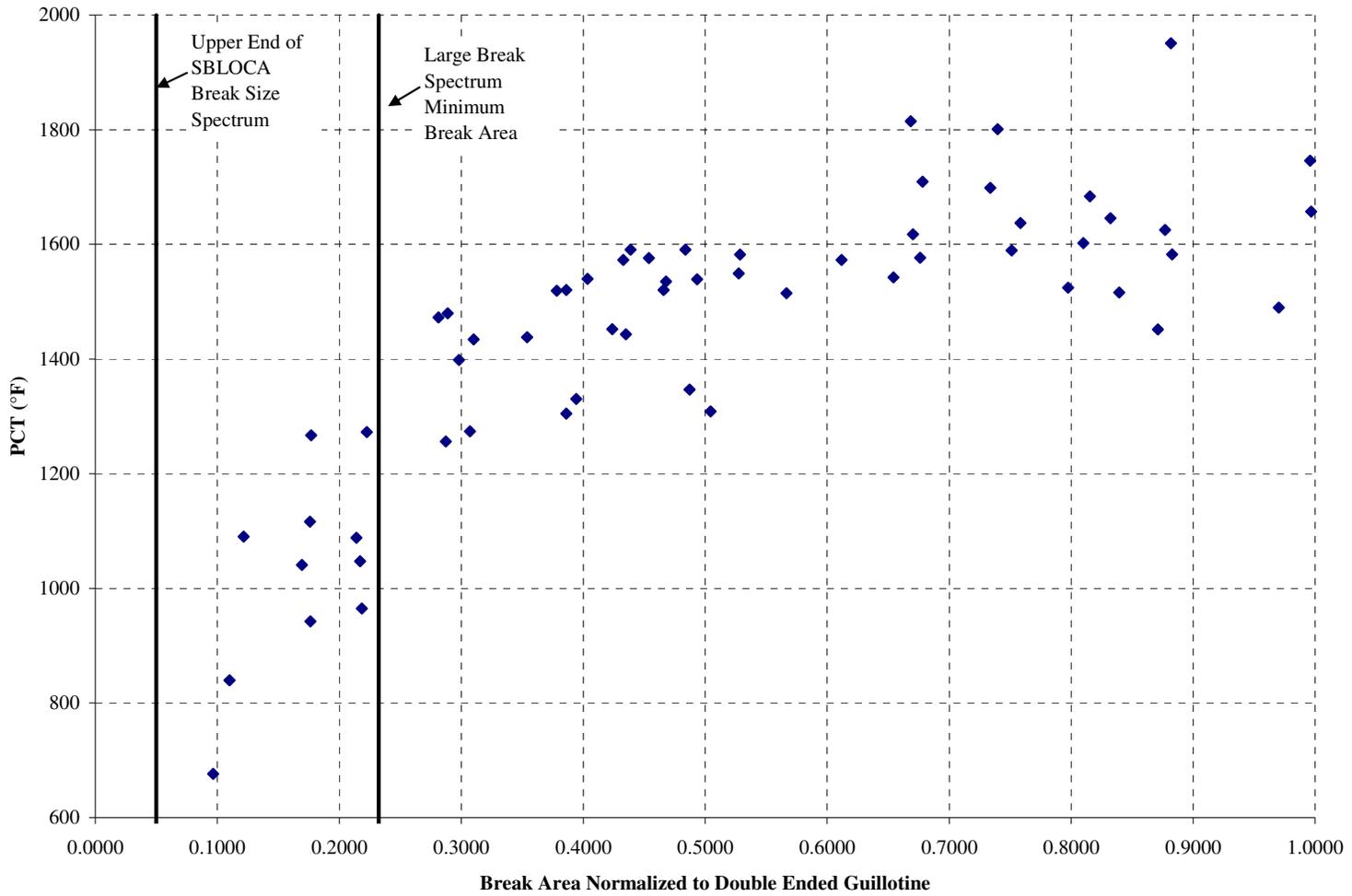


Figure 4-16 Plant B – Westinghouse 3-Loop Design

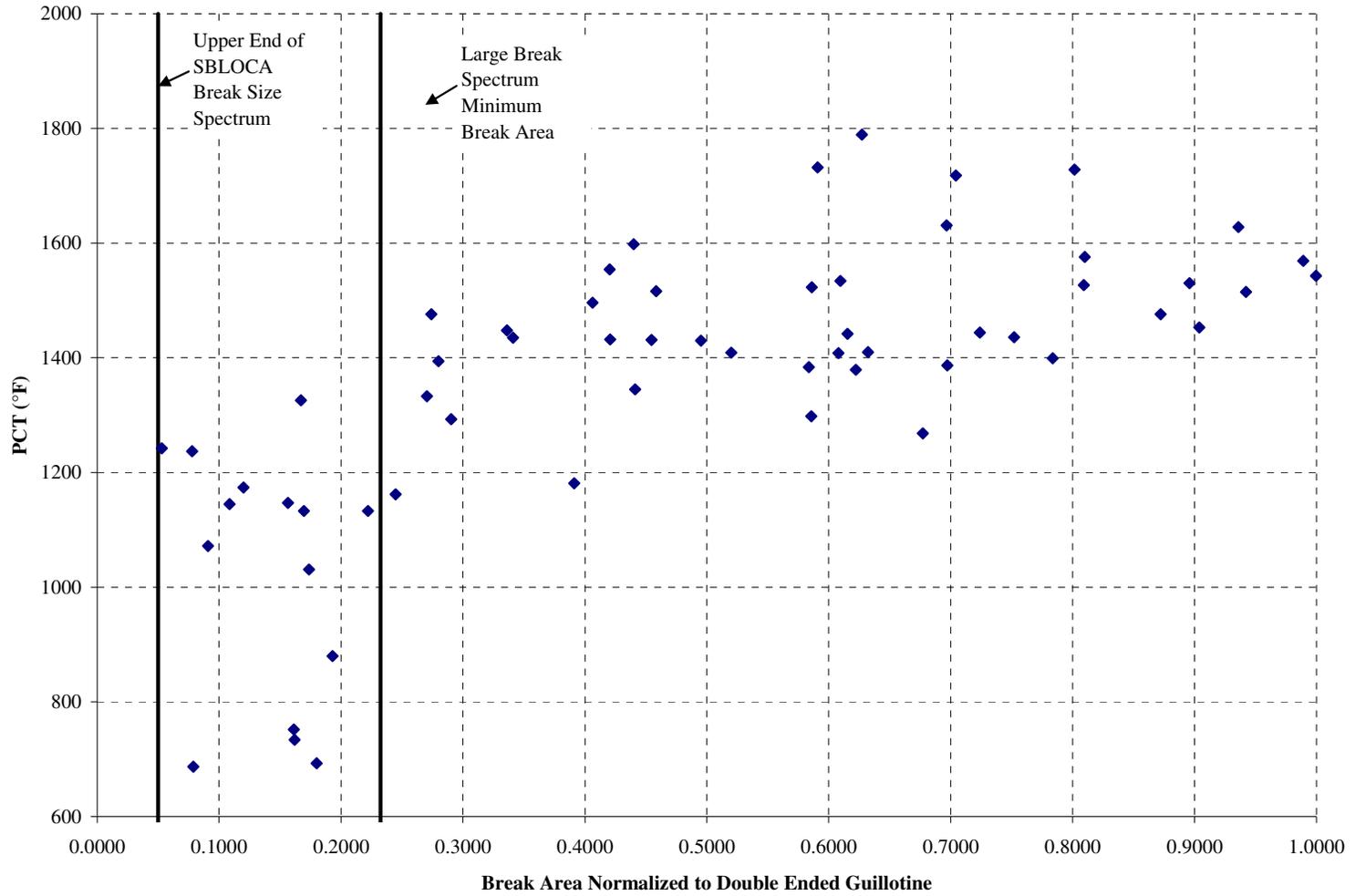


Figure 4-17 Plant C – Westinghouse 3-Loop Design

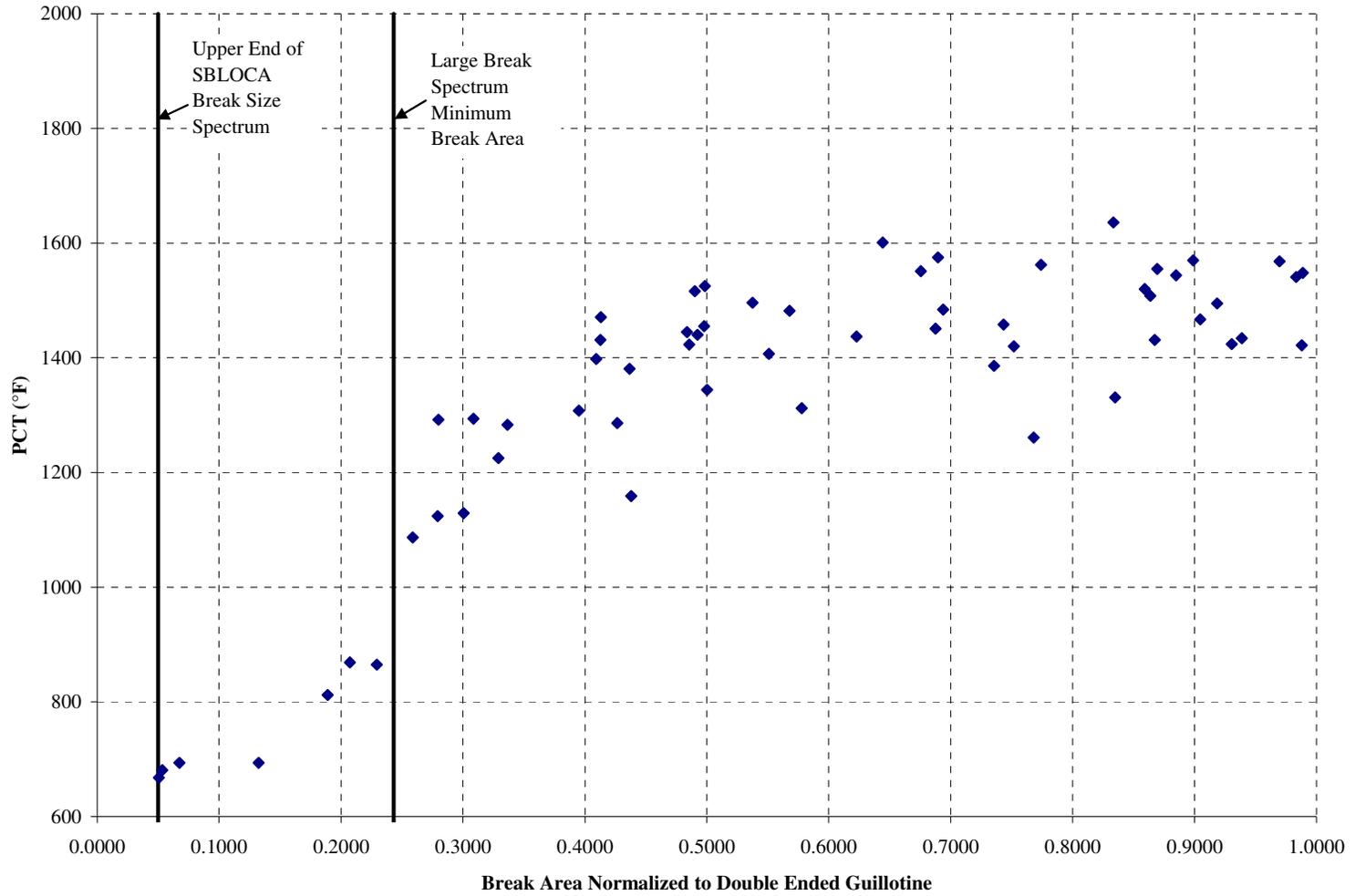


Figure 4-18 Plant D – Combustion Engineering 2x4 Design

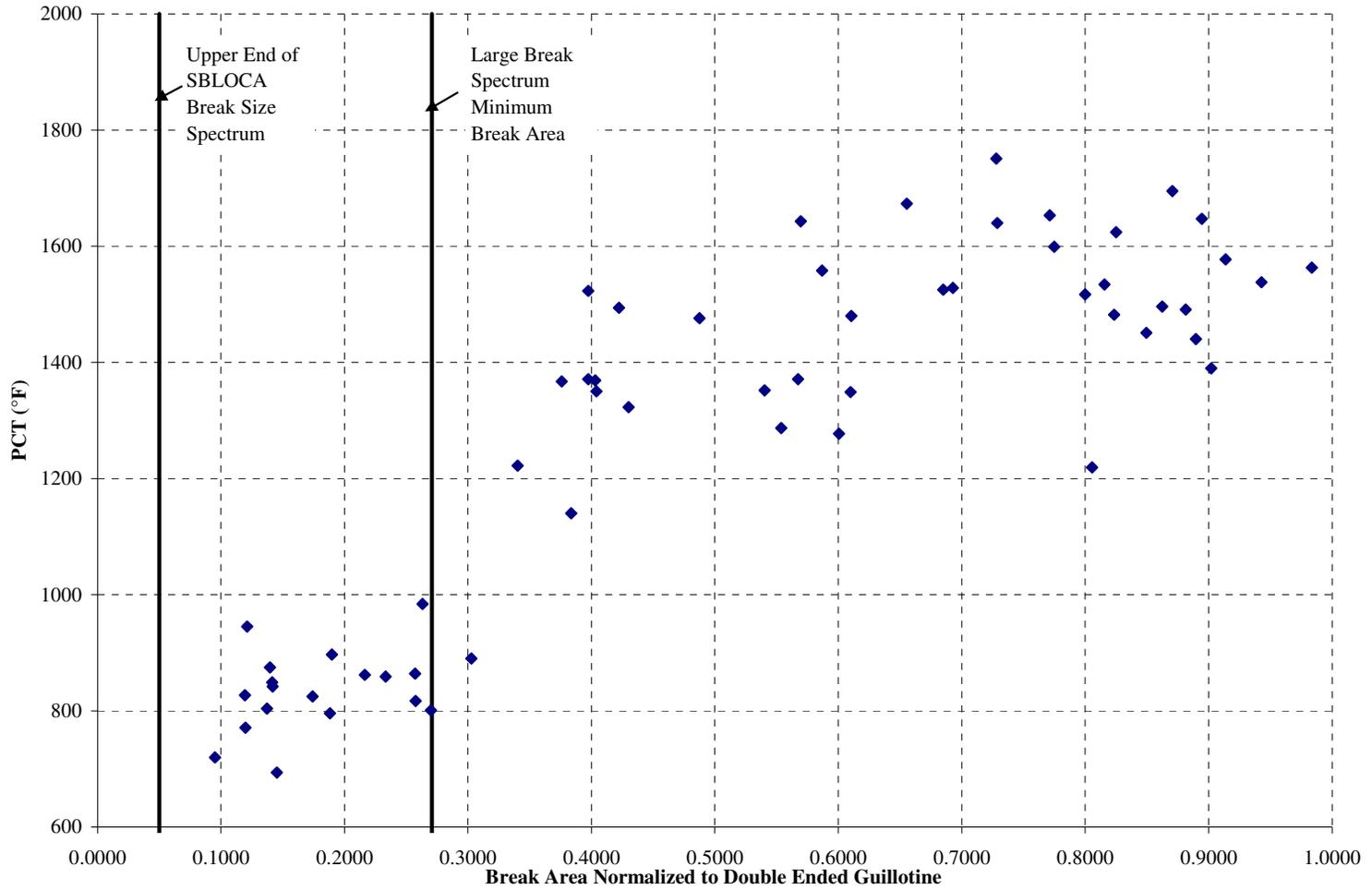


Figure 4-19 Plant E – Combustion Engineering 2x4 Design

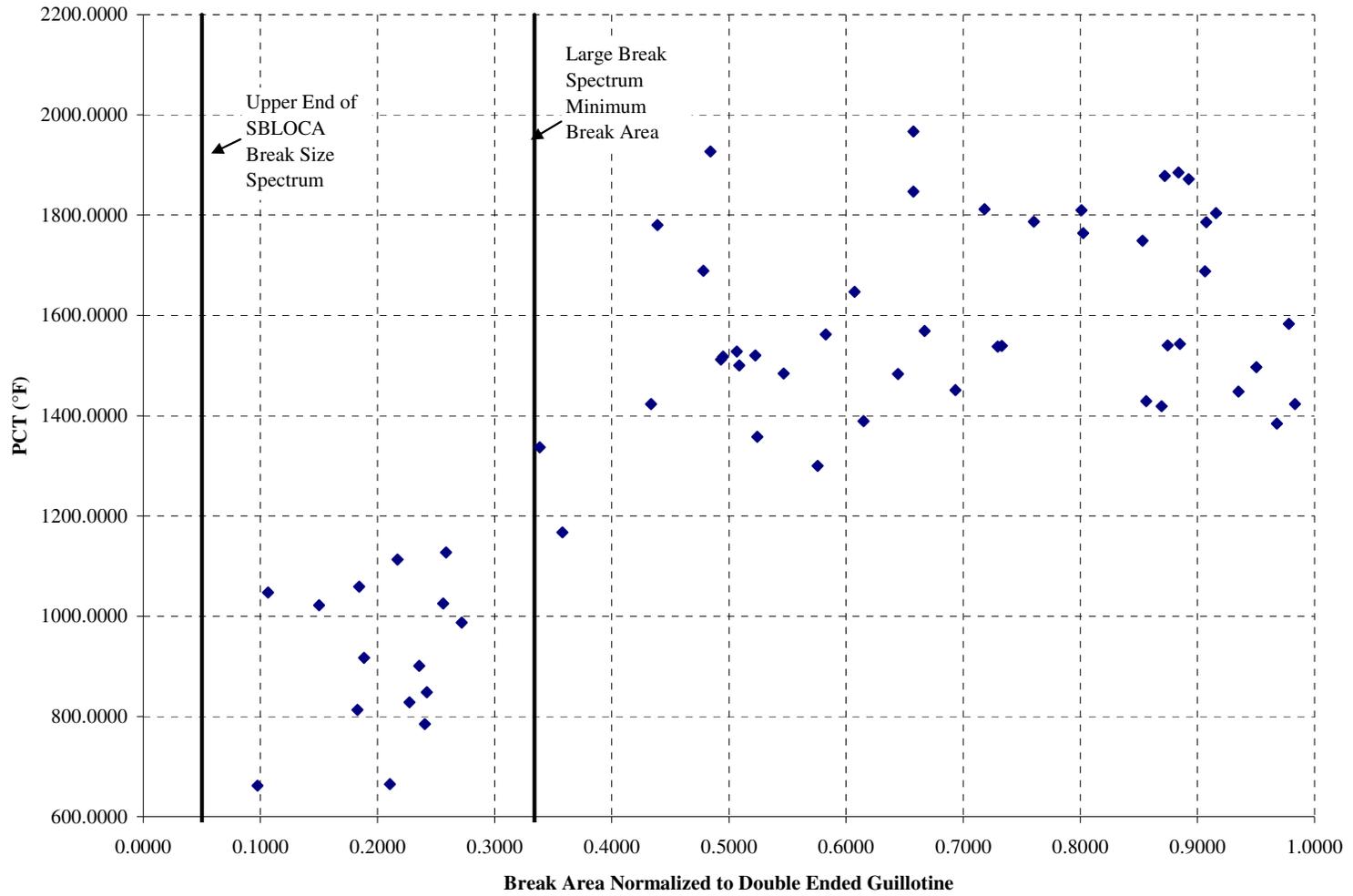


Figure 4-20 Plant F – Westinghouse 3-loop Design

4.7 Detail information for Containment Model

Containment initial conditions and cooling system information are provided in Table 3-8 and Heat Sinks are provided in Table 3-9. For St Lucie Unit 1, the scatter plots of PCT versus the sampled containment volumes and initial atmospheric temperature are shown in Figure 4-21 and Figure 4-22. Containment pressure as a function of time for limiting case is shown in Figure 4-23.

PCT vs Containment Volume

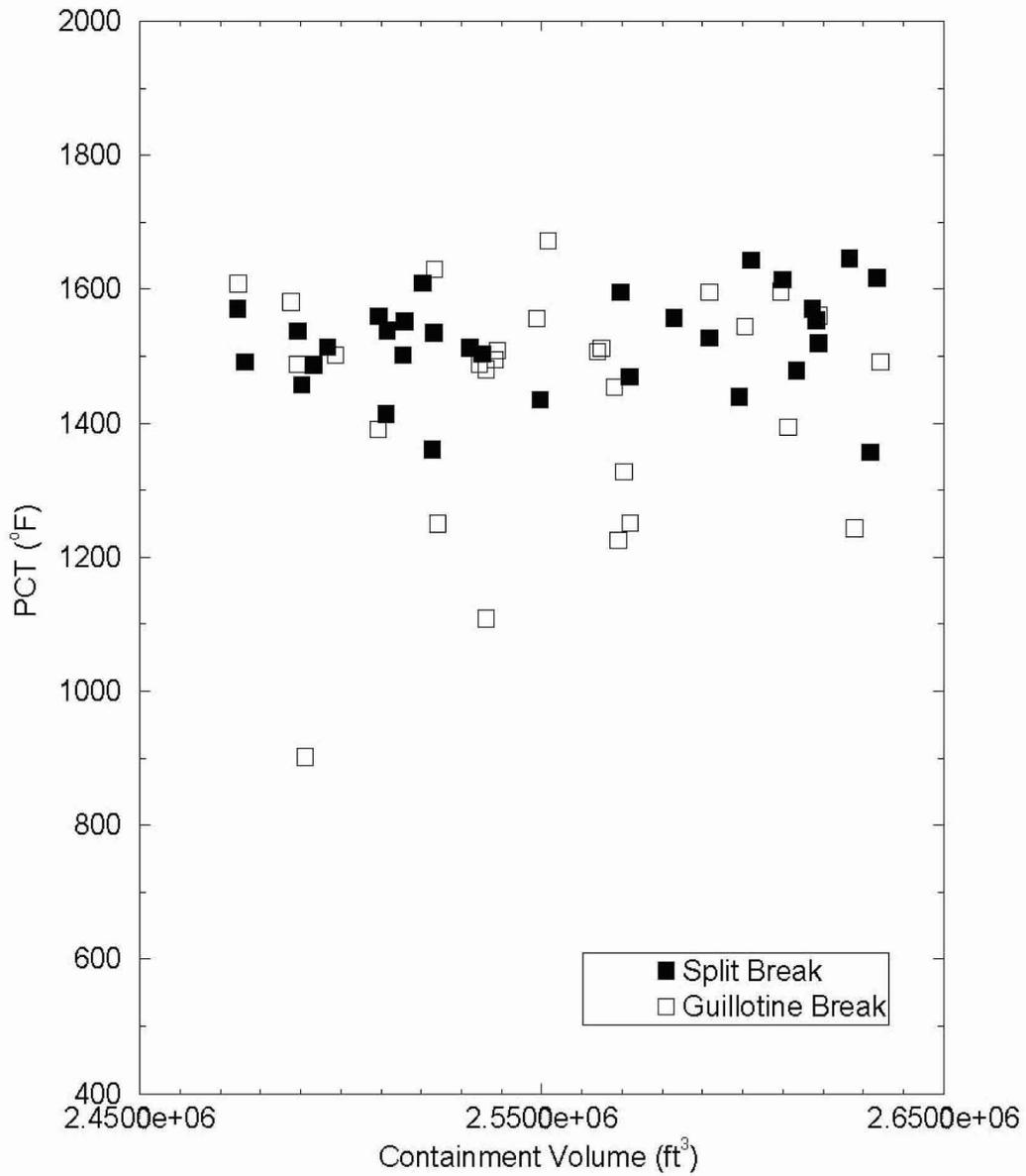


Figure 4-21 PCT vs. Containment Volume

PCT vs Containment Temperature

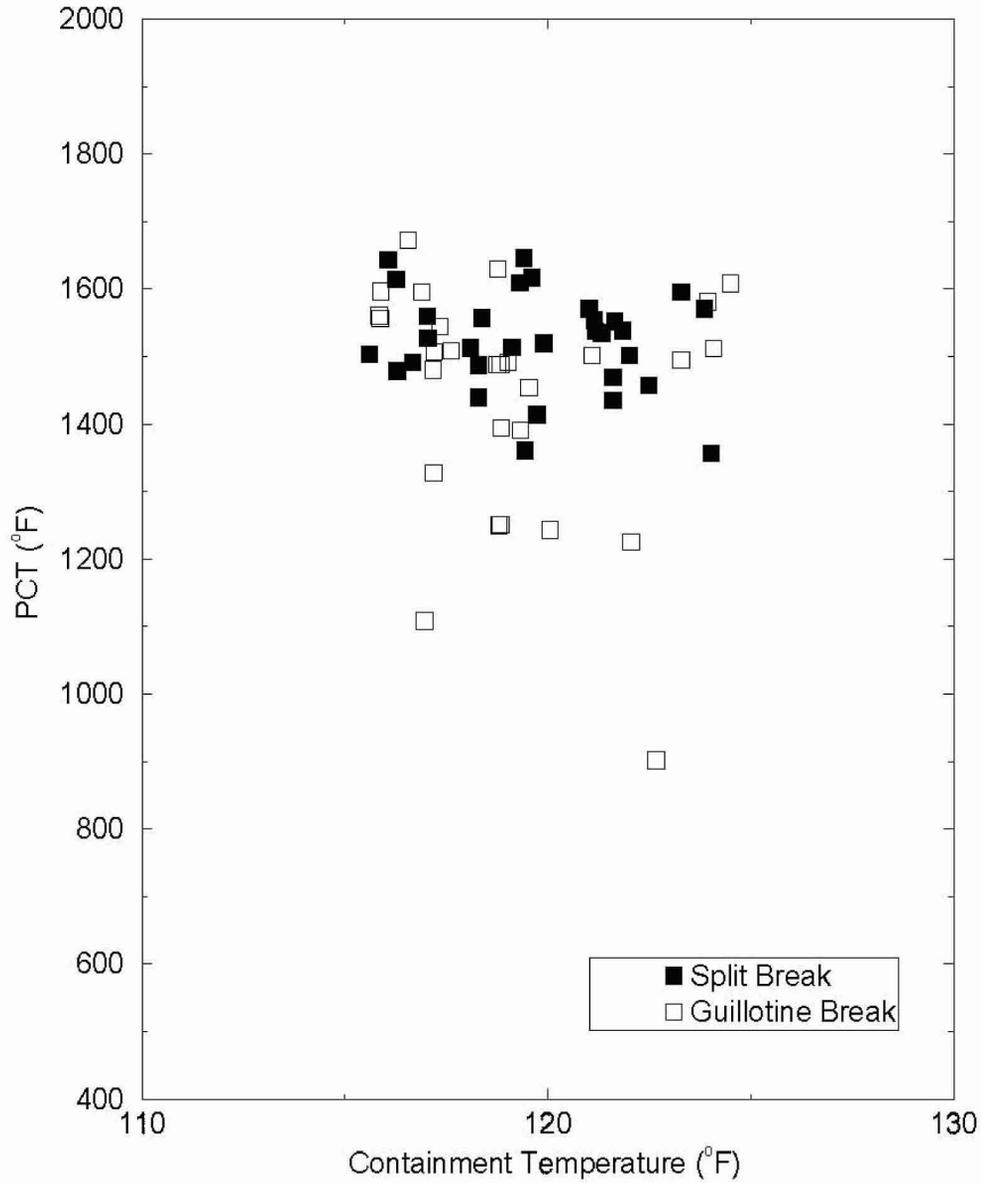


Figure 4-22 PCT vs. Initial Containment Temperature

Containment Pressures

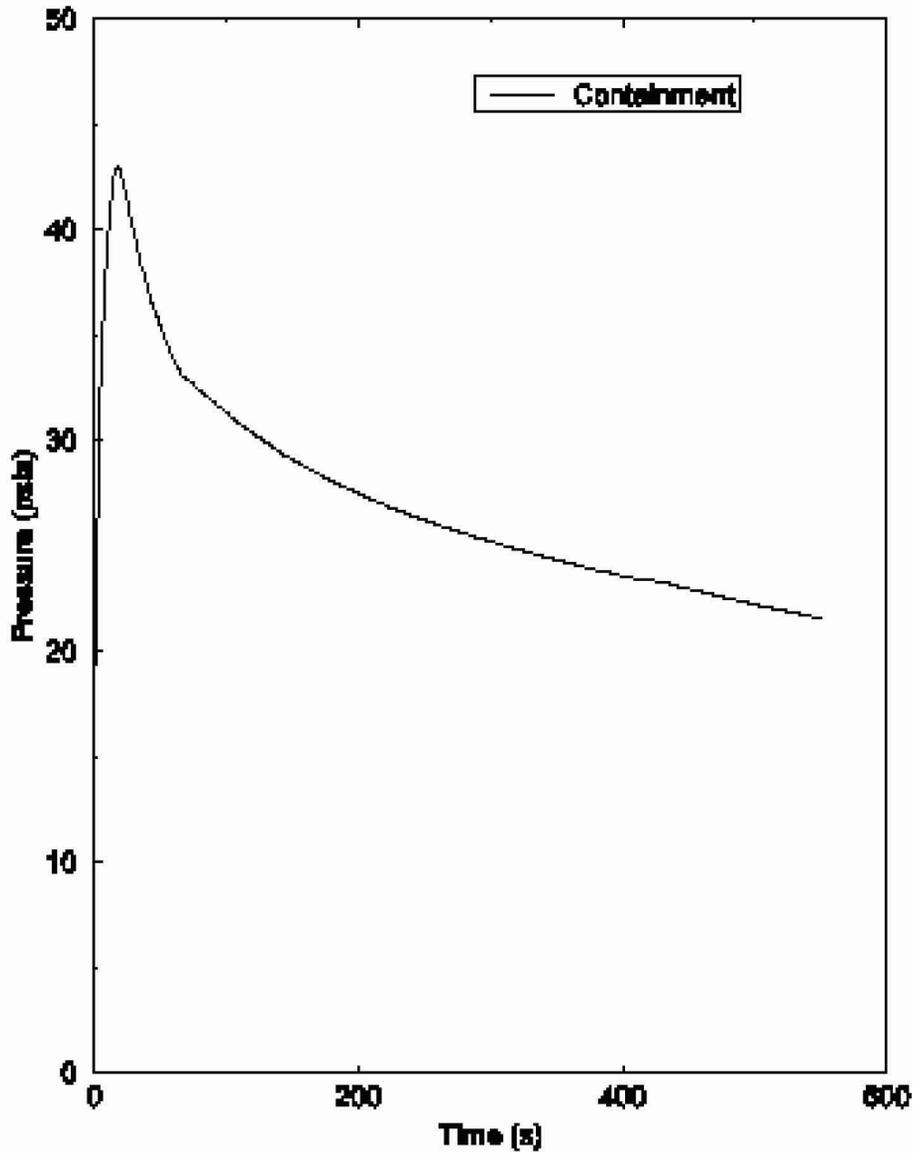


Figure 4-23 Containment Pressure as function of time for limiting case

4.8 Cross-References to North Anna

Question: *In order to conduct its review of the St Lucie Unit 1 application of AREVA's realistic LBLOCA methods in an efficient manner, the NRC staff would like to make reference to the responses to NRC staff requests for additional information that were developed for the application of the AREVA methods to the North Anna Power Station, Units 1 and 2, and found acceptable during that review. The NRC Staff safety evaluation was issued on April 1, 2004 (Agency-wide Documentation and Management System (ADAMS) accession number ML040960040). The staff would like to make use of the information that was provided by the North Anna licensee that is not applicable only to North Anna or only to subatmospheric containments. This information is contained in letters to the NRC from the North Anna licensee dated September 26, 2003 (ADAMS accession number ML032790396) and November 10, 2003 (ADAMS accession number ML033240451). The specific responses that the staff would like to reference are:*

September 26, 2003 letter: NRC Question 1

NRC Question 2

NRC Question 4

NRC Question 6

November 10, 2003 letter: NRC Question 1

Please verify that the information in these letters is applicable to the AREVA model applied to St Lucie Unit 1 except for that information related specifically to North Anna and to sub-atmospheric containments.

Response: The responses provided to questions 1, 2, 4, and 6 are generic and related to the ability of ICECON to calculate containment pressures. They are applicable to the St Lucie Unit 1 RLBLOCA submittal.

Question 1 – Completely Applicable

Question 2 – Completely Applicable

Question 4 – Completely Applicable (the reference to CSB 6-1 should now be to CSB Technical Position 6-2). The NRC altered the identification of this branch technical position in Revision 3 of NUREG-0800.

Question 6 - Completely applicable.

The supplemental request and response are applicable to St Lucie Unit 1.

4.9 GDC 35 – LOOP and No-LOOP Case Sets

Question: *10CFR50, Appendix A, GDC [General Design Criterion] 35 [Emergency core cooling] states that, “Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite electric power is not available) and for offsite electric power operation (assuming onsite power is not available) the system function can be accomplished, assuming a single failure.”*

The Staff interpretation is that two cases (loss of offsite power with onsite power available, and loss of onsite power with offsite power available) must be run independently to satisfy GDC 35.

Each of these cases is separate from the other in that each case is represented by a different statistical response spectrum. To accomplish the task of identifying the worst case would require more runs. However, for LBLOCA analyses (only), the high likelihood of loss of onsite power being the most limiting is so small that only loss of offsite power cases need be run. (This is unless a particular plant design, e.g., CE [Combustion Engineering] plant design, is also vulnerable to a loss of onsite power, in which situation the NRC may require that both cases be analyzed separately. This would require more case runs to satisfy the statistical requirement than for just loss of offsite power.)

What is your basis for assuming a 50% probability of loss of offsite power? Your statistical runs need to assume that offsite power is lost (in an independent set of runs). If, as stated above, it has been determined that Palisades, being of CE design, is also vulnerable to a loss of onsite power, this also should be addressed (with an independent set of runs).

Response: In concurrence with the NRC’s interpretation of GDC 35, a set of 59 cases each was run with a LOOP and No-LOOP assumption. The set of 59 cases that predicted the highest

figure of merit, PCT, is reported in Section 2 and Section 3, herein. The results from both case sets are shown in Figure 3-22. This is a change to the approved RLBLOCA EM (Reference 1).

4.10 Input Variables Statement

Question: *Provide a statement confirming that Florida Power & Light (FP&L) and its LBLOCA analyses vendor have ongoing processes that assure that the input variables and ranges of parameters for the LBLOCA analyses conservatively bound the values and ranges of those parameters for the operated St Lucie Nuclear Plant Unit 1 (SLA). This statement addresses certain programmatic requirements of 10 CFR 50.46, Section (c).*

Response: FP&L and the LBLOCA Analysis Vendor have an ongoing process to ensure that all input variables and parameter ranges for the SLA realistic large break loss-of-coolant accident are verified as conservative with respect to plant operating and design conditions. In accordance with FP&L Quality Assurance program requirements, this process involves

- 1) Definition of the required input variables and parameter ranges by the Analysis Vendor.
- 2) Compilation of the specific values from existing plant design input and output documents by FP&L and Vendor personnel in a formal analysis input summary document issued by the Analysis Vendor and
- 3) Formal review and approval of the input document by FP&L. Formal FP&L approval of the input document serves as the release for the Vendor to perform the analysis.

Continuing review of the input document is performed by FP&L as part of the plant design change process and cycle-specific core design process. Changes to the input summary required to support plant modifications or cycle-specific core alternations are formally communicated to the Analysis Vendor by FP&L. Revisions and updates to the analysis parameters are documented and approved in accordance with the processes described above for the initial analysis.

5.0 Conclusions

A RLBL OCA analysis was performed for the St Lucie Nuclear Plant Unit 1 using NRC – approved AREVA NP RLBLOCA methods (Reference 1). Analysis results show that the limiting LOOP case has a PCT of 1672°F, and a maximum oxidation thickness and hydrogen generation that fall well within regulatory requirements.

The analysis supports operation at a nominal power level of 3029 MWt (including 0.3 % uncertainty), a steam generator tube plugging level of up to 10 percent in all steam generators, a total LHR of 15.0 kW/ft, a total peaking factor (F_Q) up to a value of 2.161, and a nuclear enthalpy rise factor ($F_{\Delta H}$) up to a value of 1.749 (including 6% uncertainty) with no axial or burnup dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. For large break LOCA, the three 10CFR50.46 (b) criteria presented in Section 3.0 are met and operation of St Lucie Unit 1 with AREVA NP-supplied 14x14 Zircaloy-4 clad fuel is justified.

6.0 References

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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. XN-CC-39 (A) Revision 1, "ICECON: A Computer Program to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)," Exxon Nuclear Company, October 1978.
5. U. S. Nuclear Regulatory Commission, NUREG-0800, Revision 3, Standard Review Plan, March 2007.
6. NUREG/CR-1532, EPRI NP-1459, WCAP-9699, "PWR FLECHT SEASET Unblocked Bundle, Forced and Gravity Reflood Task Data Report," June 1980.
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8. NUREG/CR-0994, "A Radiative Heat Transfer Model for the TRAC Code" November 1979.
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11. D. A. Mandell, "Geometric View Factors for Radiative Heat Transfer within Boiling Water Reactor Fuel Bundles," Nucl. Tech., Vol. 52, March 1981.
12. EMF-2102(P)(A) Revision 0, *S-RELAP5: Code Verification and Validation*, Framatome ANP, Inc., August 2001.