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Revision 0

Robinson Nuclear Plant
Realistic Large Break LOCA Analysis

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Nature of Changes

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1.	All	This is a new document.

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Nomenclature

BOC	Beginning of Cycle
CFR	Code of Federal Regulations
DEGB	Double-Ended Guillotine Break
DESB	Double-Ended Split Break
ECCS	Emergency Core Cooling System
EOC	End of Cycle
F_Q^T	Total Peaking Factor
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor
FCTF	Fuel Cooling Test Facility
FANP	Framatome Advanced Nuclear Power
HFP	Hot Full Power
HHSI / HPSI	High Head / Pressure Safety Injection
LBLOCA	Large Break Loss of Coolant Accident
LHSI / LPSI	Low Head / Pressure Safety Injection
MOC	Middle of Cycle
MTC	Moderator Temperature Coefficient
NRC	U. S. Nuclear Regulatory Commission
PCT	Peak Clad Temperature
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RLBLOCA	Realistic Large Break LOCA
RV	Reactor Vessel
SI	Safety Injection

1.0 Introduction

This report describes a RLBLOCA analysis of a postulated large-break loss-of-coolant accident (LBLOCA) for the Robinson Nuclear Plant, which is a 3-loop PWR operating with Framatome ANP, Inc. (FANP) fuel.

The analysis supports operation during Cycle 23 and future cycles, 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 non-parametric statistical methods inherent in the FANP RLBLOCA 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 single-failure assumption is applied in which the negative effects of both the loss of a LPSI pump and the loss of a diesel generator are simulated. (Containment coolers and sprays are assumed to be fully functional in this analysis.) The effects of Gadolinia-bearing fuel rods and peak fuel rod exposures are also considered.

2.0 Summary

The limiting peak cladding temperature (PCT) analysis is based on the parameter specification given in Table 2.1, based on a set of 59 calculations. The limiting PCT, occurring on a 4 w/o Gd_2O_3 bearing rod, was calculated to be 1952°F.

The analysis assumes full-power operation at 2339 MW_t (plus uncertainties), a steam generator tube plugging level of up to 10% in any generator, a total peaking factor (F_Q) of 2.62 (including uncertainties, but no axial dependency), and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.80 (including uncertainty). This analysis also addresses typical operational ranges or technical specification limits (whichever is applicable) with regard to pressurizer pressure and level; accumulator pressure, temperature (containment temperature), and level; core average temperature; core flow; containment pressure and temperature; and refueling water storage tank temperature.

The FANP 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, which corresponds to a maximum assembly burnup of 57,000 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 (EFPH)	5821
Core Power (MW)	2336
Total Peaking (F_Q)	2.594
Radial Peak ($F_{\Delta h}$)	1.8
Axial Offset	+0.20
Break Type	DEGB
Break Size (ft^2/side)	3.64 (~88%)
Offsite Power Availability	Yes
Decay Heat Multiplier	1.02

3.0 Analysis

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

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel excluding the cladding surrounding the plenum volume, were to react.
- The calculated changes in core geometry shall be such that the core remains amenable to cooling.

Note that this analysis does not evaluate for changes in core coolability due to seismic events, nor does it evaluate the 10 CFR 50.46(b) long-term cooling criterion.

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 3-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 Reactor Coolant System (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 large 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

departure from nucleate boiling (DNB). Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect, and core fission ends. As heat transfer from the rods is reduced, the cladding temperature rises.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate, and may also lead to a period of positive core flow or reduced downflow as the reactor coolant pumps in the intact loops continue to supply water to the vessel. Cladding temperatures may be reduced, and some portions of the core may rewet during this period.

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

Mitigation of the LBLOCA begins when the safety injection actuation signal (SIAS) is tripped. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst single-failure be considered for ECCS safety analysis. This single-failure has been determined to be the loss of one ECCS train, including one high-pressure safety-injection (HPSI) pump and one low-pressure safety injection pump (LPSI) pump. The FANP RLBLOCA methodology conservatively assumes an on-time start and normal lineups of the containment spray and fan coolers to conservatively reduce containment pressure and increase break flow. Hence, the analysis assumes that one HPSI pump, one LPSI pump, two containment spray pumps, and four fan coolers are operating.

When the RCS pressure falls below the accumulator pressure, fluid from the accumulators is injected into the cold legs. In the early delivery of accumulator 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 accumulator water is exhausted and core recovery must rely on SI coolant delivery alone. As the accumulators empty, the nitrogen gas used to pressurize the accumulators 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 LPSI and the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the steam generator, and the reactor coolant pump before it is vented out 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 the LPSI.

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 methodology (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.

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

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the equations. The conservation

equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomenon expected during the LBLOCA event are captured. The basic building block for modeling is the hydraulic volume for fluid paths and the heat structure for a heat transfer surface. In addition, special purpose components exist to represent specific components such as the pumps or the steam generator separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

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. 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 the CONTEMPT-LT code (Reference 3).

The methods used in the application of S-RELAP5 to the large break LOCA are described in Reference 1. A detailed assessment of this computer code was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the code's ability to predict key physical phenomena in a PWR large break LOCA. 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 95% probability. 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 the 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% probability level with 95% confidence. Total oxidation and total hydrogen are based on the 95/95 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 Westinghouse designed pressurized water reactor (PWR), which has three loops, each with a hot leg, a U-tube steam generator, and a cold leg with a RCP. The RCS also includes one pressurizer. The ECCS includes one accumulator/LPSI and one HPSI injection path per RCS loop. The HPSI and LPSI feed into common headers which are connected to the accumulator lines.

The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and ECCS back to the common LPSI header and accumulators. 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 up to 10% per steam generator was assumed.

As described in the FANP RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of those parameters sampled is given in Table 3.1. The LBLOCA phenomenological uncertainties are provided in Reference 1. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 3.2. Plant data is analyzed to develop uncertainties for the process parameters sampled in the analyses. Table 3.3 presents a summary of the uncertainties used in the analyses. Two parameters (refueling water storage tank (RWST) temperature for SI flows and diesel start time) are set at conservative bounding values for all calculations. Where applicable, the sampled

parameter ranges are based on technical specification limits. Plant data are used to define range boundaries for some parameters, e.g., loop flow (high end) and containment temperature (low end).

The containment response was modeled conservatively relative to an Appendix K response. Specifically, Technical Specification minimum RWST temperature was used for the containment sprays along with an Uchida heat transfer coefficient multiplier of [ ]. The containment model initial pressure was set to a nominal value.

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 complied with during the application of the methodology as addressed in Table 3.4.

3.5 *Realistic Large Break LOCA Results*

A set of 59 transient calculations was performed sampling the parameters listed in Table 3.1. For each transient calculation, PCT was calculated for a UO_2 rod and for gadolinia bearing rods with concentrations of 2 w/o, 4 w/o, and 8 w/o Gd_2O_3. The limiting PCT (1952°F) occurred in Case 1 for a 4 w/o Gd_2O_3 rod. The major parameters for the limiting transient are characterized 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% limit. A nominal 50/50 PCT case was identified as Case 46. The nominal PCT is 1521°F. This result can be used to quantify the relative conservatism in the 95/95 result. In this analysis, it is 431°F.

The hot fuel rod results, event times and analysis plots for the limiting PCT case are shown in Table 3.5, Table 3.6, and in Figures 3.10 through 3.20. 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. Figures 3.7 and 3.8 show PCT scatter plots vs. the time of PCT and vs. break size from the 59 calculations, respectively. Figure 3.9 shows the maximum oxidation vs. PCT for the 59 calculations. Figures 3.10 through 3.20 show key parameters from the S-RELAP5 calculation. Figure 3.10 is the plot of PCT independent of elevation.

Table 3.1 Sampled LBLOCA Parameters

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

[a] 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.

Table 3.2 Plant Operating Range Supported by the LOCA Analysis

	Event	Operating Range
1.0	Plant Physical Description	
	1.1 Fuel	
	a) Cladding outside diameter	0.424 in.
	b) Cladding inside diameter	0.364 in.
	c) Cladding thickness	0.030 in.
	d) Pellet outside diameter	0.357 in.
	e) Pellet density	95% of theoretical
	f) Active fuel length	144 in.
	g) Resinter densification	[ ]
	h) Gd_2O_3 concentrations	0, 2, 4, 8 w/o
	1.2 RCS	
	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	15x15
	e) SG tube plugging	≤ 10%
2.0	Plant Initial Operating Conditions	
	2.1 Reactor Power	
	a) Nominal reactor power	2339 MWt
	b) F_Q	≤ 2.62[a]
	c) $F_{\Delta H}$	≤ 1.8[b]
	d) MTC	≤ 0 at HFP
	2.2 Fluid Conditions	
	a) Loop flow	97.3 Mlbm/hr ≤ M ≤ 113 Mlbm/hr
	b) RCS average temperature	569.9 ≤ T ≤ 581.9°F[c]
	c) Upper head temperature	< Core Outlet Temperature

[a] Includes 5% measurement uncertainty and 3% manufacturing uncertainty.

[b] Include 4% measurement uncertainty.

[c] Sampled range of ±6°F includes both operational tolerance and measurement uncertainty.

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

	d) Pressurizer pressure	2200 ≤ P ≤ 2300 psia[a]
	e) Pressurizer level	43.3% ≤ L ≤ 63.3%
	f) Accumulator pressure	615 ≤ P ≤ 675 psia
	g) Accumulator liquid volume	825 ≤ V ≤ 841 ft^3
	h) Accumulator temperature	80 ≤ T ≤ 130°F (coupled to containment temperature)
	i) Accumulator fL/D	As-Built piping configuration
	j) Minimum ECCS boron	≥ 1950 ppm
3.0	Accident Boundary Conditions	
	a) Break location	Any RCS piping location
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe)	0.05 ≤ A ≤ 0.5 full pipe area (split) 0.5 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of one LPSI and one HPSI
	e) Offsite power	On or Off
	f) LPSI flow	Bounding minimum of current pump delivery[b]
	g) HPSI flow	Bounding minimum of current pump delivery[c]
	h) Safety injection temperature	≤ 100°F
	i) HPSI delay	≤ 20.5 (w/ offsite power) ≤ 40 (w/o offsite power)
	j) LPSI delay	≤ 29 (w/ offsite power) ≤ 44 (w/o offsite power)
	k) Containment pressure	Bounding current configuration
	l) Containment temperature	80 ≤ T ≤ 130°F
	m) Containment sprays delay	≥ 0 seconds

[a] Based on representative plant data, including measurement uncertainty.

[b] Per the direction given in Reference 4, the LPSI pump delivery curve provided in Reference 5 was degraded by 5% and used in the subject analysis. Flow spilts were calculated by S-RELAP5.

[c] Total HPSI pump delivery the one-pump HPSI curve provided in Reference 5, with uniform delivery to each loop.

Table 3.3 Statistical Distributions Used for Process Parameters

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution	Standard Deviation
Core Power (%)	Uniform	99.5 - 100.5	Normal	0.3
Pressurizer Pressure (psia)	Uniform	2220 - 2300	N/A	N/A
Pressurizer Level (%)	Uniform	43.3 – 63.3	N/A	N/A
Accumulator Liquid Volume (ft^3)	Uniform	825 - 841	N/A	N/A
Accumulator Pressure (psia)	Uniform	615 - 675	N/A	N/A
Containment/Accumulator Temperature (°F)	Uniform	80 - 130	N/A	N/A
Containment Volume ($x10^6$ ft^3)	Uniform	1.95 - 2.23	N/A	N/A
Initial Flow Rate (Mlbm/hr)	Uniform	97.3 - 113	N/A	N/A
Initial Operating Temperature (°F)	Uniform	569.9 – 581.9	N/A	N/A
RWST Temperature (°F)	Point	100	N/A	N/A
Offsite Power Availability[a]	Binary	0,1	N/A	N/A
Delay for Containment Cooling (s)	Point	0.0	N/A	N/A
HHSI Delay (s)	Point	20.5 (w/ offsite power) 40 (w/o offsite power)	N/A	N/A
LHSI Delay (s)	Point	29 (w/ offsite power) 44 (w/o offsite power)	N/A	N/A

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

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. Occurrences of CCFL were noted in a statistically insignificant number of opportunities.
2. Framatome ANP has agreed that it is not to use nodalization with hot leg to downcomer nozzle gaps.	Hot leg nozzle gaps were not modeled.
3. If Framatome ANP 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 is consistent with the 3-loop sample problem for the Reference 1 methodology. An end-of-life calculation was not explicitly performed thus the need for a blowdown clad rupture model was not reevaluated.
4. Slot breaks on the top of the pipe have not been evaluated. These breaks could cause the loop seals to refill during late reflood and the core to uncover again. These break locations are an oxidation concern as opposed to a PCT concern since the top of the core can remain uncovered for extended periods of time. Should an analysis be performed for a plant with loop seals with bottom elevations that are below the top elevation of the core, Framatome ANP 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.	The evaluation of slot breaks is documented in the FANP RLBLOCA analysis guidelines.
5. The model applies to 3 and 4 loop Westinghouse- and CE-designed nuclear steam systems.	The plant is a Westinghouse 3-loop plant.
6. The model applies to bottom reflood plants only (cold side injection into the cold legs at the reactor coolant discharge piping).	The plant is a bottom reflood plant.
7. The model is valid as long as blowdown quench does not occur. If blowdown quench occurs, additional justification for the blowdown heat transfer model and uncertainty are needed if the calculation is corrected. A blowdown quench is characterized by a temperature reduction of the peak cladding temperature (PCT) node to saturation temperature during the blowdown period.	Examination of the cases showed no evidence of blowdown quench.
8. The reflood model applies to bottom-up quench behavior. If a top-down quench occurs, the model is to be justified or corrected to remove top quench. A top-down quench is characterized by the quench front moving from the top to the bottom of the hot assembly.	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 Westinghouse 3-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 RV noding diagrams.
11. A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical report approval process must be provided. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation.	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.

Table 3.5 Summary of Results for the Limiting PCT Case

Case #	1
PCT	
Temperature	1952°F
Time	29.6 seconds
Elevation	~9.4 ft
Metal-Water Reaction	
% Oxidation Maximum	1.82%
% Total Oxidation	0.04%
Total Hydrogen	0.68 lb

Table 3.6 Calculated Event Times for the Limiting PCT Case

Event	Time (sec)
Break Opened	0.0
RCP Trip	N/A
SIAS Issued	0.6
Start of Broken Loop Accumulator Injection	6.5
Start of Intact Loop Accumulator Injection	10.4, 10.4
Start of HHSI	21.1
End of Bypass / Beginning of Refill	23
Beginning of Core Recovery (Beginning of Reflood)	29.1
LHSI Available	29.6
PCT Occurred	29.6
LHSI Delivery Began (to common header)	37
Broken Loop Accumulator Emptied	47.5
Intact Loop Accumulators Emptied (Loop 2 and 3 respectively)	50.2, 47.9
Transient Calculation Terminated	414.8

Table 3.7 Heat Transfer Parameters for the Limiting Case[a]

[

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[a] Values in brackets show full range of applicability. Phasic data is provided regardless of the amount of that phase present during the respective period.

[

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Figure 3.1 Primary System Noding

Figure 3.2 Secondary System Noding

[

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Figure 3.3 Reactor Vessel Noding

Figure 3.4 Core Noding Detail

[

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Figure 3.5 Upper Plenum Noding Detail

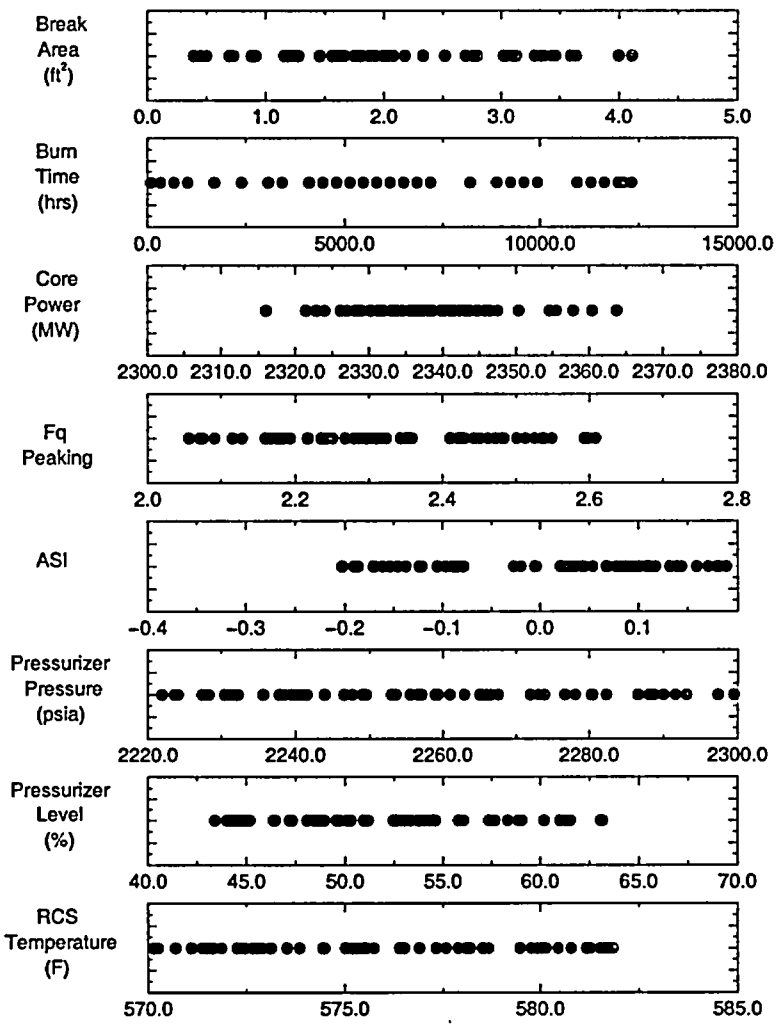

Figure 3.6 Scatterplot of Operational Parameters

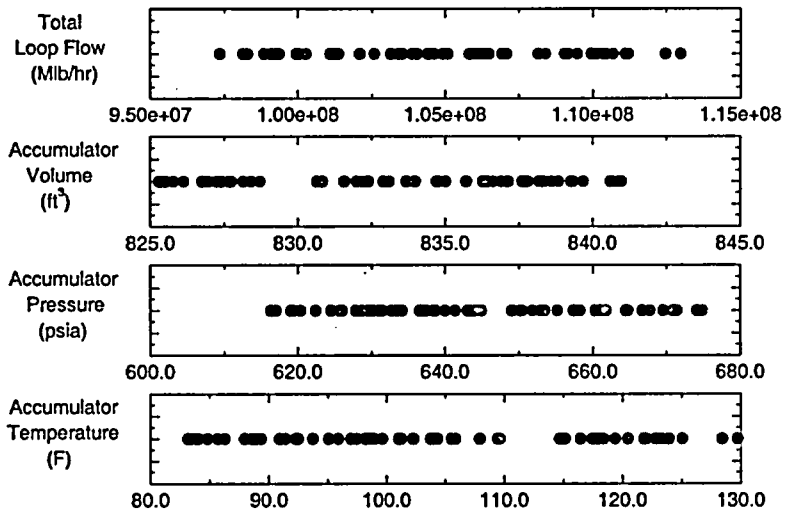

Figure 3.6 Scatterplot of Operational Parameters *(Continued)*

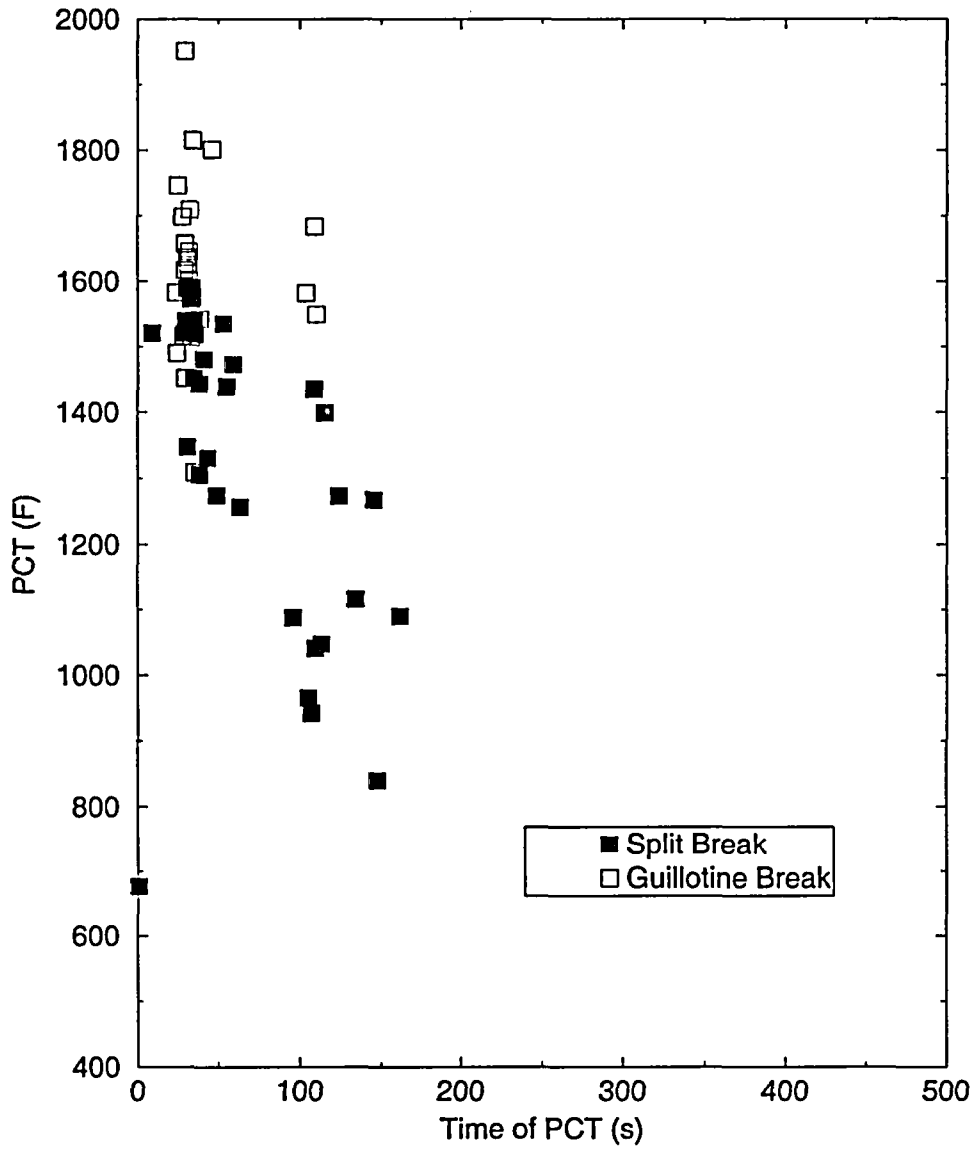

Figure 3.7 PCT vs. PCT Time Scatterplot From 59 Calculations

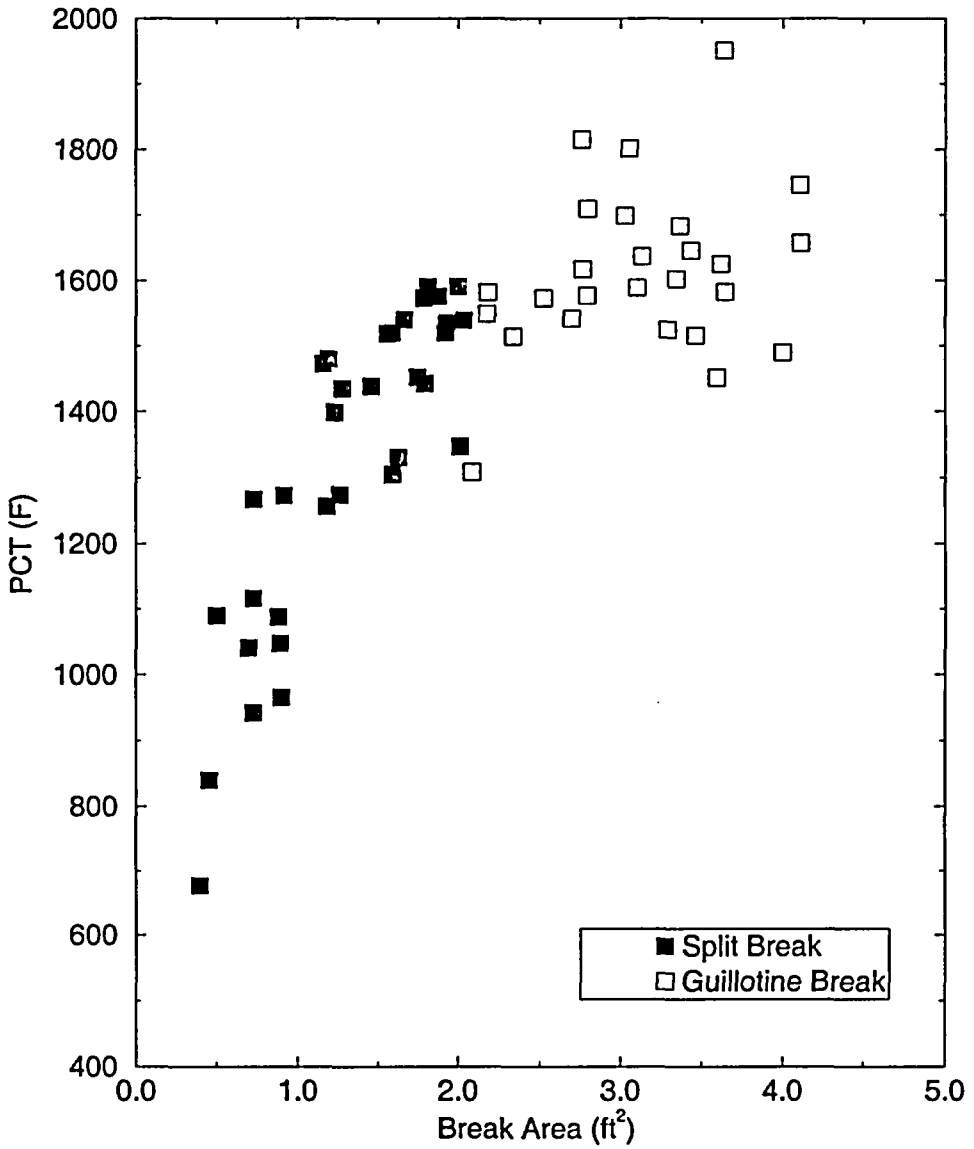

Figure 3.8 PCT v. Break Size Scatterplot From 59 Calculations

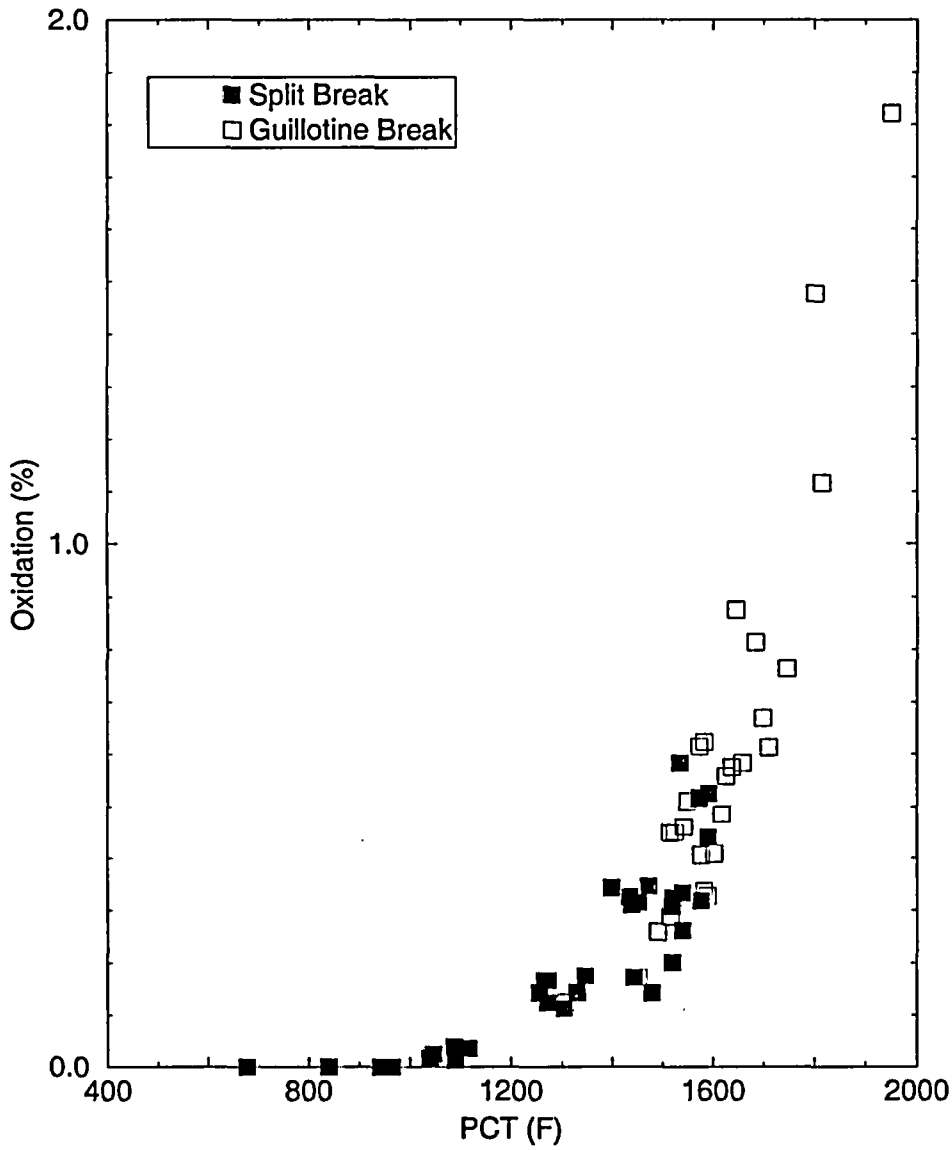

Figure 3.9 Maximum Oxidation vs. PCT Scatterplot From 59 Calculations

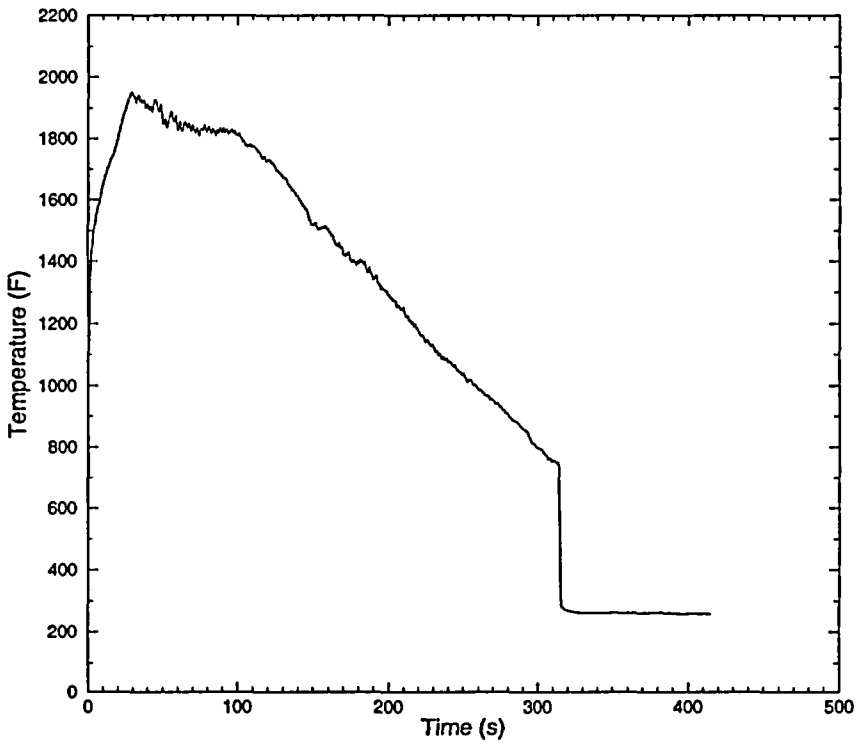

Figure 3.10 Peak Cladding Temperature (Independent of Elevation) for Limiting Break

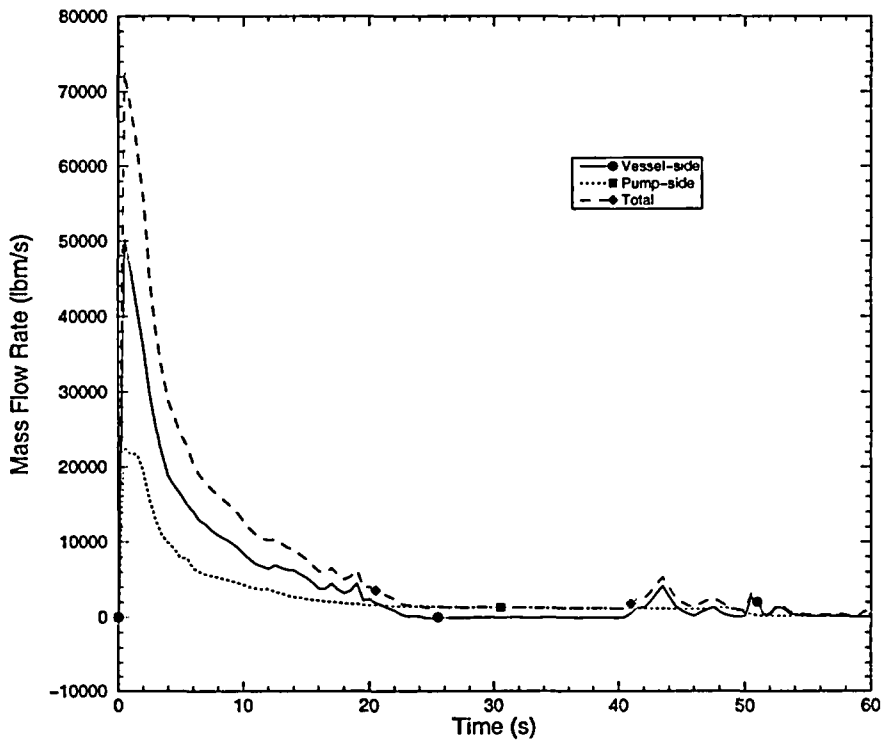

Figure 3.11 Break Flow for the Limiting Break (Early)

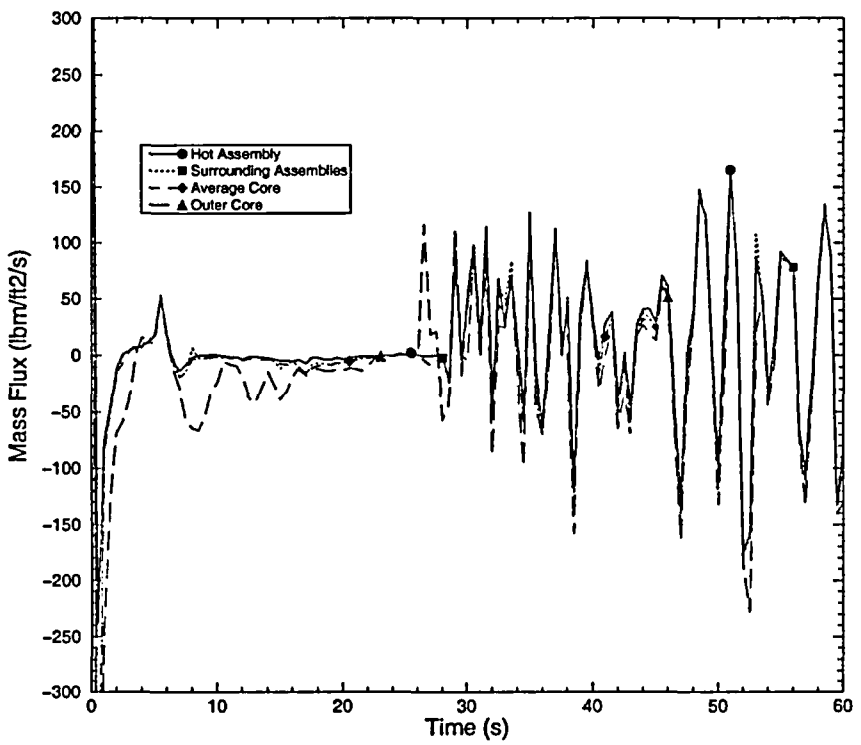

Figure 3.12 Core Inlet Mass Flux for Limiting Break (Early)

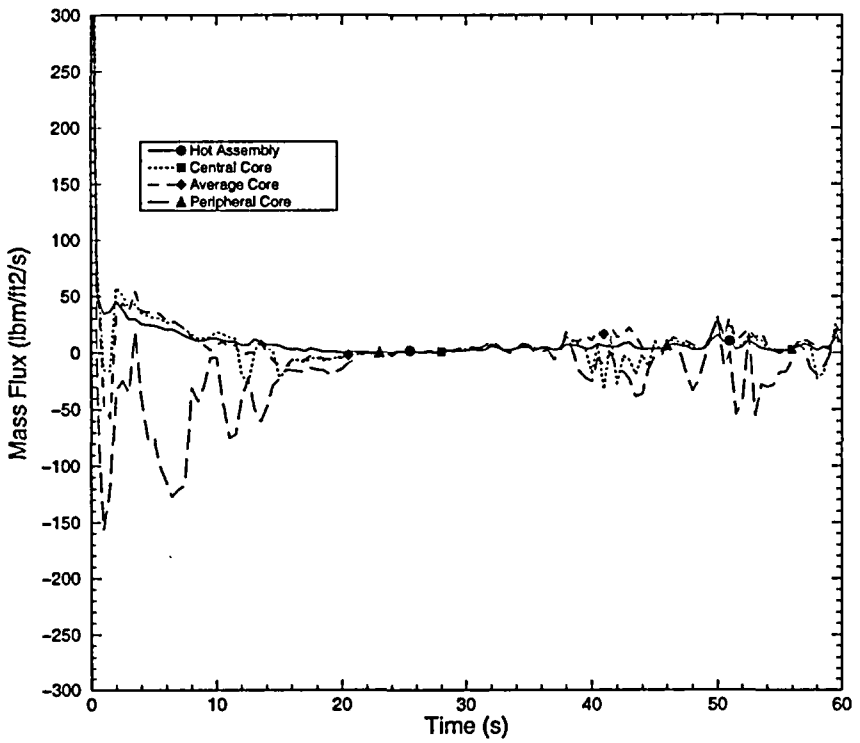

Figure 3.13 Core Outlet Mass Flux for Limiting Break (Early)

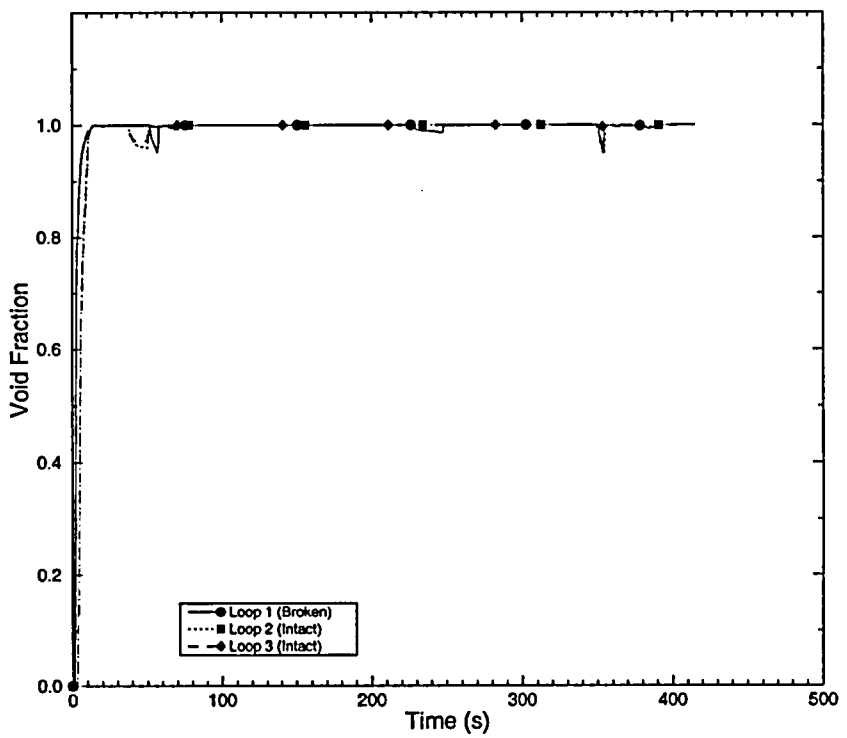

Figure 3.14 Void Fraction at RCS Pumps for Limiting Break

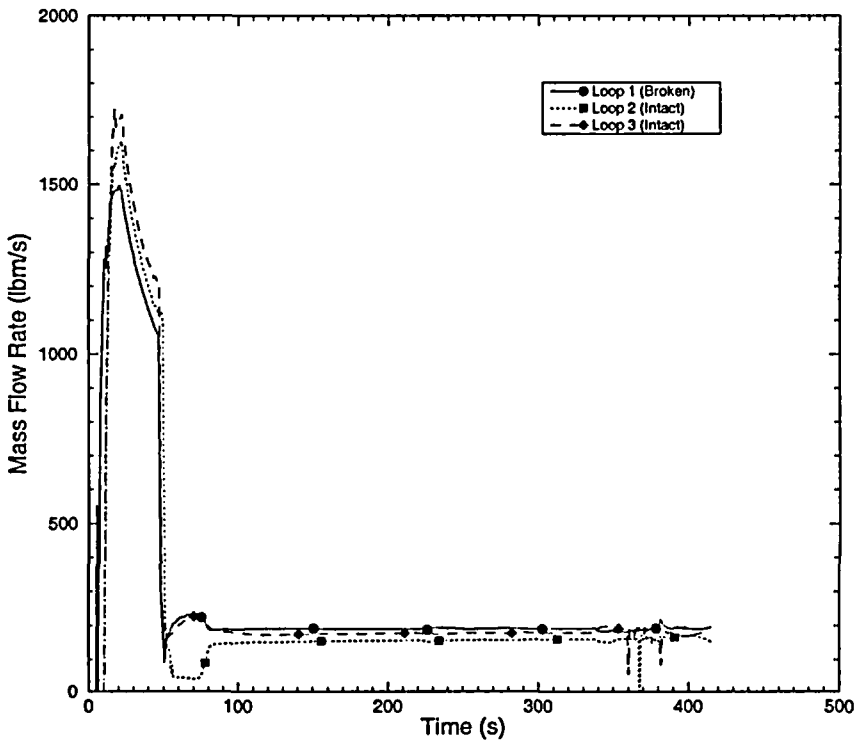

Figure 3.15 ECCS Flows (Includes Accumulator, HPSI, and LPSI) for Limiting Break

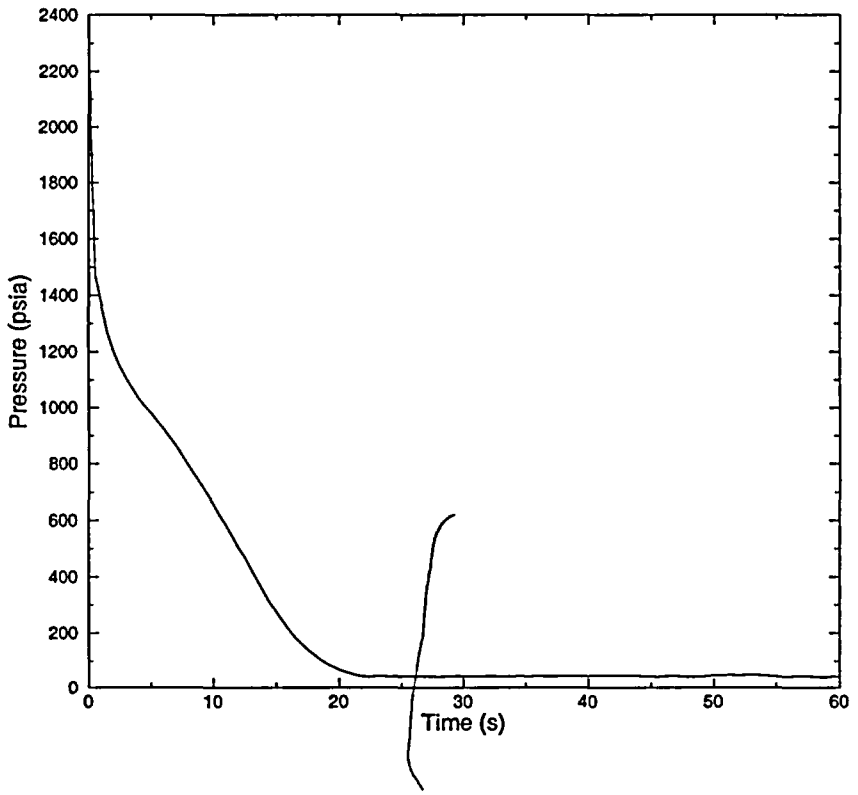

Figure 3.16 Upper Plenum Pressure for Limiting Break (Early)

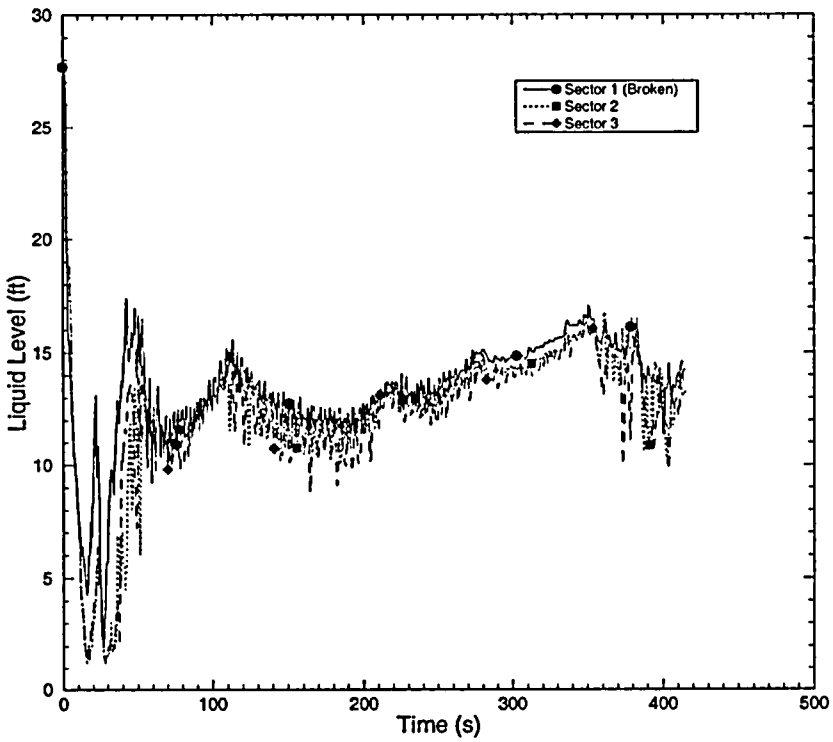

Figure 3.17 Collapsed Liquid Level in the Downcomer for Limiting Break

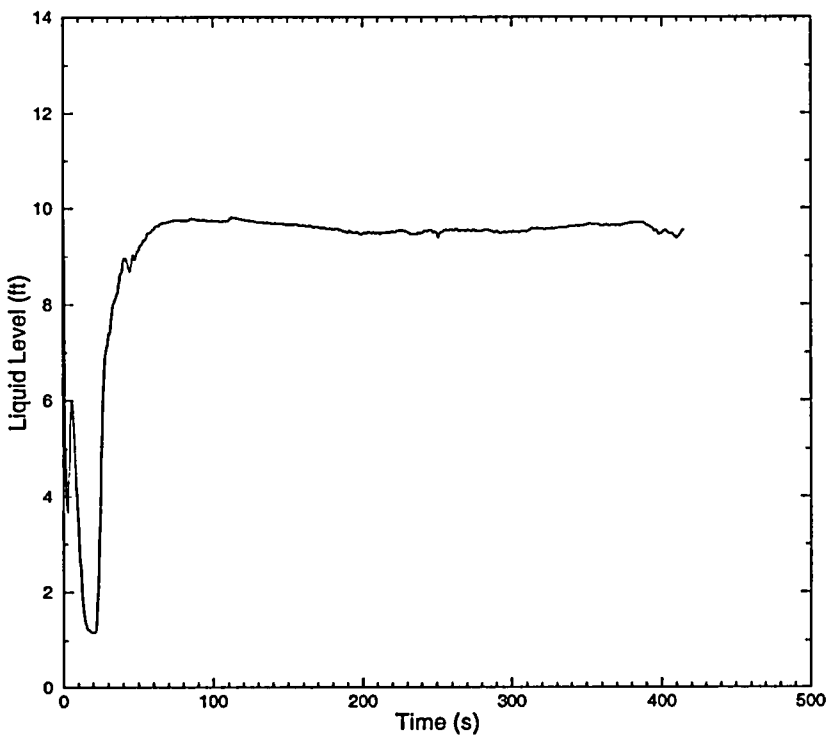

Figure 3.18 Collapsed Liquid Level in the Lower Plenum for the Limiting Break

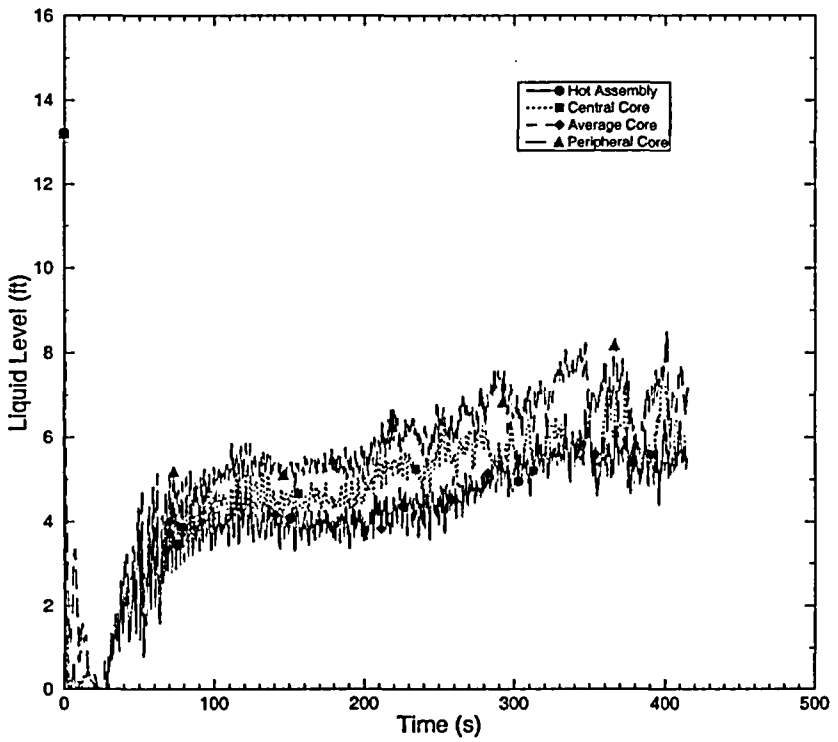

Figure 3.19 Collapsed Liquid Level in the Core for the Limiting Break

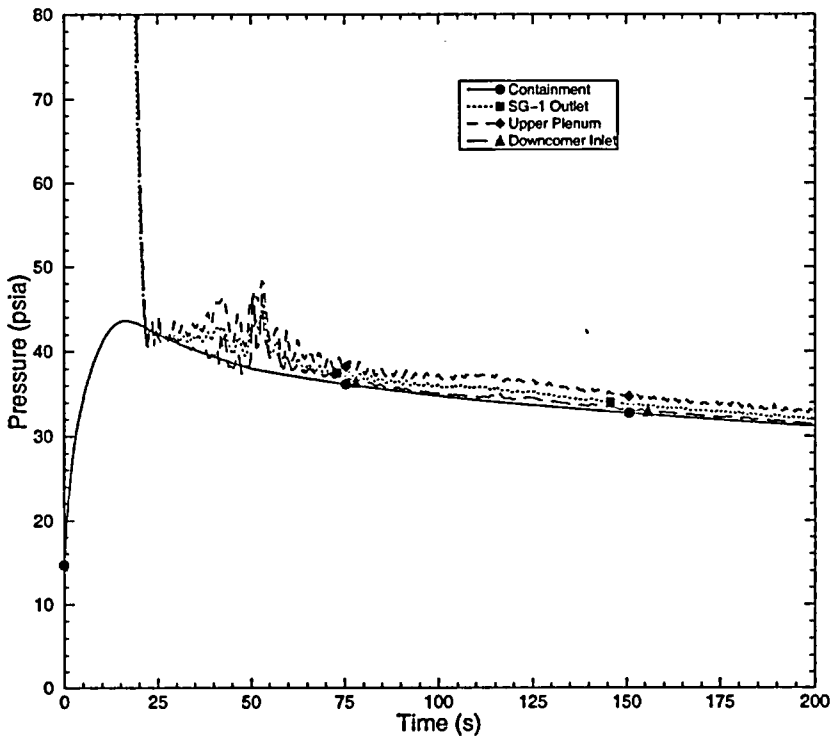

Figure 3.20 Containment and Loop Pressures for Limiting Break

4.0 Conclusions

The results of the RLBLOCA analysis show that the limiting case has a PCT of 1952°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 2339 MWt (plus uncertainty), a steam generator tube plugging level of up to 10% in any generator, a total peaking factor (F_Q^T) of 2.62 and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.80 with no axially-dependent power peaking limit.

5.0 References

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

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

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

4. E. J. Geyer (Progress Energy) letter to P. G. Newby (Framatome-ANP), "RNP Cycle 23 and RLBLOCA RHR Pump Performance, and RLBLOCA HPSI Delay," File NF-401.2604, Serial: NF-03A-0123, FANP Document: 38-5030411-00, July 14, 2003.

5. E. J. Geyer (Progress Energy) letter to P. G. Newby (Framatome-ANP), "Transmittal of Robinson Cycle 23 Plant Parameters Document," File NF-401.2604, Serial: NF-03A-0106, FANP Document: 38-5027076-01, July 25, 2003.

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING REVISION TO CORE OPERATING LIMITS REPORT (COLR) REFERENCES

EMF-3030(P)

Robinson Nuclear Plant
Realistic Large Break LOCA Analysis

February 2004

Proprietary Version

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REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING REVISION TO CORE OPERATING LIMITS REPORT (COLR) REFERENCES

Framatome ANP Affidavit Regarding Proprietary Attachment VI

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing in Regulatory Affairs, for FANP, and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by FANP to determine whether certain FANP information is proprietary. I am familiar with the policies established by FANP to ensure the proper application of these criteria.

3. I am familiar with the reports EMF-3030, Revision 0, entitled "Robinson Nuclear Plant Realistic Large Break LOCA Analysis" and EMF-3031, Revision 1, entitled "Harris Nuclear Plant Realistic Large Break LOCA Analysis" and referred to herein as "Documents." Information contained in these Documents have been classified by FANP as proprietary in accordance with the policies established by FANP for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by FANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure.

6. The following criteria are customarily applied by FANP to determine whether information should be classified as proprietary:

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(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

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8. FANP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 4th

day of March, 2004.

Ella F. Carr-Payne
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/05

ELLA F. CARR-PAYNE
Notary Public
Commonwealth of Virginia
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