ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

SEQUOYAH NUCLEAR PLANT UNIT 2
REALISTIC LARGE BREAK
LOSS OF COOLANT ACCIDENT ANALYSIS

ANP-2655(NP)
REVISION 1
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AREVA NP Inc.

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Revision 1

Sequoyah Nuclear Plant Unit 2 Realistic Large Break LOCA Analysis

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document.
2.	All	This revision replaces Revision 0 for Sequoyah Unit 2 completely.

Contents

Tables

Figures

This document contains a total of 104 pages.

Nomenclature

AFD	Axial Flux Difference
BLEU	Blended Low Enriched Uranium
CFR	Code of Federal Regulations
CCTF	Cylindrical Core Test Facility
CSAU	Code Scaling, Applicability, and Uncertainty
DEGB	Double-Ended Guillotine Break
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPH	Effective Full Power Hours
EM	Evaluation Model
F_Q	Total Peaking Factor
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor
HFP	Hot Full Power
LBLOCA	Large Break Loss of Coolant Accident
LANL	Los Alamos National Laboratory
LEFM	Leading Edge Flow Meter
LOCA	Loss of Coolant Accident
MSIV	Main Steam Isolation Valve
MTC	Moderator Temperature Coefficient
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
RLBLOCA	Realistic Large Break LOCA
RV	Reactor Vessel
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank

Nomenclature (cont'd)

SER	Safety Evaluation Report
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
TVA	Tennessee Valley Authority
UHI	Upper Head Injection

1.0 Introduction

This report describes and provides results from a RLBLOCA analysis for the Sequoyah Unit 2 Station. Revision 1 of the report replaces Revision 0 and includes the responses to the NRC's request for additional information sent by TVA on December 21, 2007 to the NRC in Section 6. The plant is a Westinghouse 4-loop design with a rated thermal power of 3455 MWt and ice condenser containment. The loops contain four RCPs, four U-tube steam generators and a pressurizer. The ECCS is provided by two independent injection trains and four accumulators.

The analysis supports operation for Cycle 16 and beyond with AREVA NP's Mark-BW fuel design using either BLEU or standard UO_2 fuel and M5 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 RLBLOCA 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 Sequoyah Unit 2 Station with AREVA NP fuel.

The non-parametric statistical methods inherent in the AREVA NP 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 loss of one train of the pumped ECCS injection is simulated. Regardless of the single-failure assumption, all containment pressure-reducing systems are assumed fully functional. The effects of Gadolinia-bearing fuel rods and peak fuel rod exposures are considered.

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

The assumed reactor core power for the Sequoyah realistic large break loss-of-coolant accident is 3479 MWt. This value represents the plant rated thermal power of 3455 MWt with a maximum power measurement uncertainty of 0.7 percent (24 MWt) added to the rated thermal power. The power measurement uncertainty assumption discussed in 10CFR50, Appendix K was previously reduced for Sequoyah from 2.0 percent of the plant rated thermal power to 0.7 percent based on the installation of a LEFM system to measure main feedwater flow. The improved feedwater flow measurement accuracy provided by the LEFM allowed for a power measurement uncertainty recovery of 1.3 percent. The basis for the current 0.7 percent measurement uncertainty assumption is documented in Topical Report No. WCAP-15669, Revision 0. The power was not sampled in the analysis. This is not expected to have an adverse effect on the PCT results.

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 33 percent of the DEGB area (see Section 6.6 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 3-23. The effect on PCT results is expected to be minor.

2.0 Summary

The limiting peak cladding temperature (PCT) analysis is based on the parameter specification given in Table 2-1 for the limiting case. The limiting PCT is 2002 °F for a UO_2 rod in a case with No-LOOP conditions. Gadolinia-bearing rods of 2, 4, 6 and 8 w/o Gd_2O_3 were also analyzed, but were not 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 17x17 thermal hydraulically compatible fuel designs; hence, there is no mixed core consideration.

The analysis assumes full core power operation at 3479 MWt (including uncertainties), a steam generator tube plugging level of up to 15 percent in all steam generators, a total peaking factor (F_Q) up to a value of 2.65 (including uncertainties, but no axial dependency), and a nuclear enthalpy rise factor ($F_{\Delta H}$) up to a value of 1.706 (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 (based on containment 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 (EFPH)	10200
Core Power (MWt)	3479
Total Peaking (F_Q)	2.568
Radial Peak ($F_{\Delta H}$)	1.706
Axial Offset	0.2613
Break Type	Split
Break Size (ft^2/side)	2.7259
Offsite Power Availability	Available
Decay Heat Multiplier	0.97322

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:

- 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.
- Long-term cooling is established and maintained after the LOCA.

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. Since the analysis purpose is solely to change the LBLOCA licensing basis (from deterministic to realistic) of Unit 2, prior coolable geometry (LOCA-seismic loads) and long-term cooling licensing bases remain unaffected and valid. Therefore, compliance with Criteria 4 and 5 is assured.

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 4-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 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 boiling and convection to steam. The coolant voiding creates a strong negative reactivity effect and core fission ends. As heat transfer from the fuel rods is reduced, the cladding temperature rises.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate, and leads 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 (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 reactor coolant pumps, reducing their effectiveness. Once again, the core flow reverses as most of the vessel fluid mass flows out through the broken cold leg.

Mitigation of the LBLOCA begins when the 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. 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 one charging pump, one SI pump, one RHR pump and two containment spray pumps 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 pumped ECCS 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 charging, SI and RHR 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 generator, and the reactor coolant pump before it is vented out the break. Some steam may 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 the RHR 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 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 (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 RODEX3A code provides initial conditions for the S-RELAP5 fuel models. Specific parameters are discussed in Section 3.3.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into one of the loops (specifically, the loop with the pressurizer). The evolution of the transient through blowdown, refill and reflood is computed continuously using S-RELAP5. 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) and has been updated for modeling ice condenser containments.

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 specific inputs supplied by TVA. 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 3-2. 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 Westinghouse-designed PWR, which has four loops, each with a hot leg, a U-tube steam generator, and a cold leg with a RCP[1]. The RCS also includes one pressurizer connected to a hot leg. The core contains (193) 17x17 thermal-hydraulic compatible AREVA Mark–BW fuel assemblies. The ECCS includes one charging and one accumulator/SI/RHR injection path per RCS loop. The SI and RHR feed into common headers which are connected to the accumulator lines. The charging pumps are also cross-connected. 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 accumulator lines. The charging injection flows are connected to the RCS, and the SI and RHR injection flows are connected to the accumulator lines, consistent with the plant layout. 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 15 percent per steam generator was assumed.

Plant input modeling parameters were provided by TVA specifically for the Sequoyah Unit 2 Station. By procedure, TVA maintains plant documentation current, and directly communicates with AREVA on plant design and operational issues regarding reload cores. TVA and AREVA will continue to interact in that fashion regarding the use of AREVA fuel in the Sequoyah Unit 2 Station. Both entities have ongoing processes that assure the ranges and values of input parameters for the Sequoyah Unit 2 Station RLBLOCA analysis bound those of the as-operated plant.

As described in the AREVA RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of those parameters is given in Table 3-2. 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 3-3. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Table 3-4 presents a summary of the uncertainties used in the analysis. Two parameters, RWST temperature for ECCS 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 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) containment parameters—containment pressure and temperature. In many instances, the conservative guidance of CSB 6-1 (Reference 5) was used in setting the remainder of the containment model input parameters. As noted in Table 3-4, containment temperature is a sampled parameter. Containment pressure response is indirectly ranged by sampling the upper containment volume (Table 3-4). The minimum value is carried over from use in the long-term containment integrity analysis of record for Sequoyah. The maximum value is a simplified value computed as the volume available within the upper dome of the containment and within the crane wall above the control rod drive missile shield with no accounting for internal structures and the volumes of the refueling canal and the annular region separating the ice compartments neglected. This volume is maximized by neglecting the volume of internal structures. The lower compartment volume is biased low in order to promote flow through the ice baskets. 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 Sequoyah through application of the process used in the RLBLOCA EM (Reference 1) sample problems. The ice condenser containment noding is shown in Figure 3-6. In the ice compartment, the water formed by melted ice and condensed steam flows to the lower ice compartment sump where it accumulates, if the ice bay drains are not large enough to accommodate the rate of water production. When the water level in the lower ice compartment sump rises above the bottom of the lower doors, water spillage through the lower doors occurs in addition to flow through the drain ports. The water drainage (spillage plus drainage) from the ice compartment falls through the lower compartment vapor. This condenses steam and reduces the containment pressure. The ice compartment drainage flow is treated as a 100 percent efficient spray during the post-blowdown period of the transient.

[1] The RCPs are Westinghouse 93A type pumps. The homologous pump performance curves for this type of pump were input to the S-RELAP5 plant model.

The initial conditions and boundary conditions are given in Table 3-9. The building spray is modeled at maximum heat removal capacity. While there is an option within the computer code model to deliver spray to the lower compartment, this option is not applicable to Sequoyah Unit 2. All spray flow is delivered to the upper compartment. Because the start time for the recirculation fan is 600 seconds, forced flow from the upper compartment to the lower compartment is not likely to occur during the time period analyzed. The flow of steam or air, from the lower compartment to the upper compartment, backwards through the back draft dampers, is not modeled (no reverse direction flow). This approach is conservative in that no bypass of the ice beds (from lower to upper compartments) is allowed, and all flow from the lower compartment is directed through the ice beds. The passive flow of air and steam, from the upper compartment to the lower compartment, is modeled however. This is a passive flow, which is only a function of the excess pressure of the upper compartment compared to the lower compartment, the flow area of the recirculation fan back draft dampers, and the loss coefficient of the dampers. The back draft dampers are designed such that reverse flow from the lower to the upper compartment is prevented. However, when the upper compartment pressure is at least 0.5 psi greater than the lower compartment, the dampers open and allow flow from the upper compartment to the lower compartment. Flow in this manner, from the upper to lower compartment, is modeled without this minimum pressure difference, i.e. any excess pressure is modeled as resulting in flow.

Passive heat sink parameters are listed in Table 3-10. Surface coatings, where they existed, were incorporated as an equivalent thickness of base material in order to eliminate any insulating effects on the exposed surfaces of the heat structures. Because the original basis for the size of each heat sink was biased low (for a different application), the values listed in Table 3-10 reflect a 10 percent increase in heat transfer surface area as compensation. Passive heat sinks were added to the lower containment to represent new sump screens being installed in the Sequoyah Unit 2 plant (17 ft^3 of steel). Additionally, all heat structure exposed surfaces remain available for condensing steam, even when they may become covered by ice melt or condensate.

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 3-5.

Six non-limiting PCT cases were potential candidates for blowdown quench (SER Item 7). The applicable heat structure temperature stays well above the local saturation temperature prior to accumulator injection in all of these cases. For this set of calculations, no evidence of blowdown quench was found and compliance to the SER restriction has been demonstrated.

Case 21 and Case 40 did exhibit blowdown quench at the end of blowdown. The applicable features for the cases that exhibited a quench of the PCT node before the end of blowdown are:

- relatively small break area,
- offsite power continues to be available to power Reactor Coolant Pumps, and,
- the temporary quench occurs at the time of the end of blowdown, when accumulators start injecting into the intact loops.

Prior to being quenched the two cases exhibited a relatively small heatup during blowdown. This compares to the limiting case which exhibited a heatup of over 1000 °F during blowdown. This reduced heatup for the two cases shows that these rods were more susceptible to being quenched. Mechanistically, the observed quench occurs because the small break area limits break flow. This reduces the rates at which pressure and flow decrease at the PCT location compared with the limiting case. In addition, the time of the quench coincides with the moment when accumulators start injecting into the intact loops, adding liquid mass into the system at a significant rate and possibly helping the quench through increased availability of the liquid at the quench location.

The void fraction at the PCT locations indicates that liquid is available for cooling. Both blowdown quench cases had off site power available. Thus the continued operation of the Reactor Coolant Pumps provides increased forced convection cooling. The resulting combination of higher core flow and pressure cools the cladding sufficiently to enable a return to nucleate boiling.

It is therefore concluded that the predicted blowdown quench is appropriate for these non-limiting cases and also that this behavior is not applicable in any way to the limiting case. The blowdown quench for these cases is justified and compliance to the SER restriction has been demonstrated.

Table 3-1 Comparison of Limiting Case to Non-Limiting Cases Exhibiting Blowdown Quench

Case	PCT (°F)	PCT Elevation (ft)	PCT Time (s)	Tmin Sampled (K)	Break Type	Break Size (ft^2/side)	Peak LHGR Sampled (kw/ft)	Offsite Power Available
20 (limiting case)	2002	9.83	130.7	652.97	split	2.7259	14.6133	Yes
21	940	9.83	107.7	638.92	guillotine	1.4005	13.8788	Yes
40	1282	9.83	129.2	637.01	guillotine	1.3839	14.8103	Yes

Several measures have been taken to prevent the top-down quench (SER Item 8). The upper plenum nodalization features include:

- the homogenous 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.

Seven non-limiting cases were closely examined for top-down quench. These cases exhibit short periods of decrease in the integrated mass flux at the hot assembly exit, indicating the possibility of a net downward flow, however this is happening after the PCT has occurred. The heat structure temperature displays a drop to saturation temperature starting at the bottom and sequentially progresses upward through the PCT elevation, the nodes above the PCT node experiencing the quench at times later than at the PCT node. In three of these cases, some of the nodes situated right above the PCT node are being quenched slightly earlier than the PCT location. For one case the net downward flow occurs right at the end of the transient, after the PCT location and the entire core have already quenched, and thus it does not raise any concern of top-down quench. For the other two cases, liquid down flow occurs well past PCT time due to the reduction in steam production in the hot assembly. The modeling precautions taken to prevent top-down quench are sufficient, 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 3-2. For each transient calculation, 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 offsite power available. The limiting PCT (2002 °F) occurred in Case 20 for a UO_2 rod. The major parameters for the limiting transient are characterized in Table 2-1. Table 3-6 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. A nominal best estimate PCT case was identified as Case 14, which corresponded to the median case out of the 59-case set with offsite power available. The nominal PCT was 1514 °F. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 488 °F.

The case results, event times and analysis plots for the limiting PCT case are shown in Table 3-6, Table 3-7, and in Figures 3-12 through 3-22. Figure 3-7 shows linear scatter plots of the key parameters sampled for the 59 calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter ranges used in the analysis. Figures 3-8 and 3-9 show the time of PCT and break size versus PCT scatter plots for the 59 calculations, respectively. Figures 3-10 and 3-11 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 Figures 3-12 through 3-22. Figure 3-12 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 3-23.

Table 3-2 Sampled LBLOCA Parameters

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties, burnup)
	Break type (guillotine versus split)
	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[1]	
	Offsite power availability[2]
	Break size
	Pressurizer pressure
	Pressurizer liquid level
	Accumulator pressure
	Accumulator liquid level
	Accumulator temperature (based on lower compartment containment temperature)
	Containment temperature
	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 6.10.

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

	Event	Operating Range
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.374 in.
	b) Cladding inside diameter	0.326 in.
	c) Cladding thickness	0.024 in.
	d) Pellet outside diameter	0.3195 in.
	e) Pellet density	96 percent of theoretical
	f) Active fuel length	144 in.
	g) Resinter densification	[ ]
	h) Gd_2O_3 concentrations	2, 4, 6, 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	17x17
	e) SG tube plugging	≤ 15 percent
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal reactor power	3479 MWt[1]
	b) F_Q	≤ 2.65[2]
	c) $F_{\Delta H}$	≤ 1.706[3]
	d) MTC	≤ 0 at HFP
	2.2 Fluid Conditions	
	a) Loop flow	131.6 Mlbm/hr ≤ M ≤ 152.8 Mlbm/hr
	b) RCS average temperature	578.2 °F ≤ T ≤ 583 °F
	c) Upper head temperature	~Tcold Temperature[4]

1 Includes uncertainties

2 Ensures that a minimum 7 percent peaking margin is maintained to the Fq limits when operating at the positive or negative AFD limit

3 Includes 4 percent measurement uncertainty

4 Upper head temperature will change based on sampling of RCS temperature

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

	d) Pressurizer pressure	1859.7 psia ≤ P ≤ 2459.7 psia
	e) Pressurizer level	57 percent ≤ L ≤ 95 percent
	f) Accumulator pressure	614.7 psia ≤ P ≤ 697.7 psia
	g) Accumulator liquid volume	1004.6 ft^3 ≤ V ≤ 1095.4 ft^3
	h) Accumulator temperature	95 °F ≤ T ≤ 130 °F (coupled to containment lower volume temperature)
	i) Accumulator fL/D	As-built piping configuration
	j) Minimum ECCS boron	≥ 2400 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 area)	0.33 ≤ A ≤ 1.0 full pipe area (split) 0.33 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of one train of ECCS
	e) Offsite power	On or Off
	f) Charging pump flow	Bounding minimum of current pump delivery
	g) SI pump flow	Bounding minimum of current pump delivery
	h) RHR pump flow	Bounding minimum of current pump delivery
	h) ECCS pumped injection temperature	110 °F
	i) Charging pump delay	37 s (w/ offsite power) 27 s (w/o offsite power)
	j) SI pump delay	37 s (w/ offsite power) 27 s (w/o offsite power)
	k) RHR pump delay	37 s (w/ offsite power) 27 s (w/o offsite power)
	l) Containment pressure	14.3 psia, nominal value
	m) Containment upper compartment temperature	80 °F ≤ T ≤ 110 °F
	n)Containment lower compartment temperature	95 °F ≤ T ≤ 130 °F
	0) Containment sprays delay	8 s
	p) Containment spray water temperature	55 °F

Table 3-4 Statistical Distributions Used for Process Parameters[1]

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution[2]	Standard Deviation
Pressurizer Pressure (psia)	Uniform	1859.7 – 2459.7	N/A	N/A
Pressurizer Liquid Level (percent)	Uniform	57 – 95	N/A	N/A
Accumulator Liquid Volume (ft^3)	Uniform	1004.6 – 1095.4	N/A	N/A
Accumulator Pressure (psia)	Uniform	614.7 – 697.7	N/A	N/A
Containment Lower Compartment /Accumulator Temperature (°F)	Uniform	95 – 130	N/A	N/A
Containment Upper Compartment Temperature (°F)	Uniform	80 – 110		
Containment Upper Volume (ft^3)	Uniform	651,000 – 692,600	N/A	N/A
Initial RCS Flow Rate (Mlbm/hr)	Uniform	131.6 – 152.8	N/A	N/A
Initial RCS Operating Temperature (Tavg) (°F)	Uniform	578.2 – 583	N/A	N/A
RWST Temperature for ECCS (°F)	Point	110	N/A	N/A
RWST Temperature for Containment Sprays (°F)	Point	55	N/A	N/A
Offsite Power Availability[3]	Binary	0,1	N/A	N/A
Delay for Containment Cooling (s)	Point	8.0	N/A	N/A
Charging Pump Delay (s)	Point	37 (w/ offsite power) 27 (w/o offsite power)	N/A	N/A
LHSI Pump Delay (s)	Point	37 (w/ offsite power) 27 (w/o offsite power)	N/A	N/A
RHR Pump Delay (s)	Point	37 (w/ offsite power) 27 (w/o offsite power)	N/A	N/A

1 Note that core power is not sampled, see Section 1.0
2 All measurement uncertainties were incorporated into the operational ranges
3 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 3-5 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 nodalization 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 Sequoyah Unit 2 is lower than that used in the development of the RLBLOCA EM (Reference 1). An end-of-life calculation was not performed; thus, the need for a blowdown cladding rupture model was not reevaluated.
4.	Slot breaks on the top of the pipe have not been evaluated. These breaks could cause the loop seals to refill during late reflood and the core to uncover again. These break locations are an oxidation concern as opposed to a PCT concern since the top of the core can remain uncovered for extended periods of time. Should an analysis be performed for a plant with loop seals with bottom elevations that are below the top elevation of the core, 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.	The evaluation of slot breaks is documented in the AREVA RLBLOCA analysis guidelines.
5.	The model applies to 3 and 4 loop Westinghouse- and CE-designed nuclear steam systems.	Sequoyah Unit 2 is a Westinghouse 4-loop plant.
6.	The model applies to bottom reflood plants only (cold side injection into the cold legs at the reactor coolant discharge piping).	Sequoyah Unit 2 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. The possibility of Blowdown quench was observed in seven cases which are discussed in Section 3.4.
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 3-5 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 4-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, 3 and 4 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. Some minor differences that are included in the plant specific model include: 1) The RV upper internals are of the inverted top-hat type, therefore an additional node was added to the upper head volume in order to model the region situated below the top hat brim and above the upper support plate; 2) The plant was designed to use Upper Head Injection which utilized columns. However it was modified and the upper head safety injection was disconnected and capped. The flow path of the UHI Columns was modeled with an extra set of pipe components connecting the lower most volume of the upper head to the inlet into the corresponding radial region of the upper plenum; 3) The pumped piping branches into the accumulator discharge piping slightly differently; 4) The hydraulic model of the core employs 22 axial nodes instead of 23; 5) There are no standpipes present in the Sequoyah Unit 2 RV upper plenum; 6) The plant has safety grade charging which is included in the model; 7) The lower support plate that separates the lower plenum from the lower head of the reactor vessel is curved; 8) Sequoyah Unit 2 is a cold upper head type plant. 9) The ICECON noding is representative for an ice condenser plant and represents a change from Reference 1. 10) Component 154 has only one cell instead of the two in Reference 1.
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 3-8 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.

Table 3-5 SER Conditions and Limitations *(Continued)*

SER Conditions and Limitations	Response
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.	The Sequoyah Unit 2 plant has previously been operating with M5 clad fuel and thus this restriction has been satisfied.

Table 3-6 Summary of Results for the Limiting PCT Case

Case #	1
PCT	
Temperature	2002 °F
Time	130.7 s
Elevation	9.831 ft
Metal-Water Reaction	
percent Oxidation Maximum	3.4199
percent Total Oxidation	0.0200

Table 3-7 Calculated Event Times for the Limiting PCT Case

Event	Time (s)
Break Opened	0.0
RCP Trip	N/A[1]
SIAS Issued	0.1
Start of Broken Loop Accumulator Injection	12.8
Start of Intact Loop Accumulator Injection (Loop 2, 3 and 4 respectively)	14.7, 14.8, 14.8
Start of Charging	37.1
SI/RHR Available	37.1
Broken Loop SI Delivery Began	37.1
Intact Loop SI Delivery Began (Loop 2, 3 and 4 respectively)	37.1, 37.1, 37.1
Broken Loop RHR Delivery Began	37.1
Intact Loop RHR Delivery Began (Loop 2, 3 and 4 respectively)	37.1, 37.1, 37.1
Beginning of Core Recovery (Beginning of Reflood)	50.2
Broken Loop Accumulator Emptied	84.1
Intact Loop Accumulators Emptied (Loop 2, 3 and 4 respectively)	84.4, 84.6, 84.2
PCT Occurred	130.7
Transient Calculation Terminated	501.2

Notes: 1. The limiting 59-case set had offsite power available.

Table 3-8 Heat Transfer Parameters for the Limiting Case

Table 3-9 Containment Initial and Boundary Conditions

Containment Net Free Volume	**Volume (ft^3)**
Upper Compartment	651,000 – 692,600
Lower Compartment (minimum)	248,500
Ice Condenser	181,400
Dead Ended Compartments	129,900
Initial Mass of Ice	2.448×10^6 lbm
Initial Conditions	
Containment Pressure (nominal)	14.3 psia
Upper Containment Temperature	80 °F – 110 °F
Lower Containment Temperature	95 °F – 130 °F
Humidity	100 percent
Containment Spray	
Maximum Total Flow	2 x 7700 = 15,400 gpm
Minimum Spray Temperature	55 °F
Fastest Post-LOCA initiation of spray	10 s (ramped to full flow between 8 and 10 s)
Containment Air Return Fan[11]	
Post-LOCA initiation at 600 s	
Total Flow = 120,000 cfm	

[11] Due to the relatively late start of the recirculation fan, it is not modeled in this analysis.

Table 3-10 Passive Heat Sinks in Containment

Heat Sink	Area ft^2	Thickness ft	Inside Radius ft	Thickness ft	Height ft	Material	Left Side	Right Side
Reactor Cavity Walls	6438	2.02				concrete	Lower Comp.	insulated
Concrete Floor	4444	2.00				concrete	Lower Comp.	insulated
Interior Concrete	8464	1.00				concrete	Lower Comp.	insulated
Reactor Vessel Biological Shield Wall			11	6.0	19.88	concrete	Lower Comp.	Lower Comp.
Steel Lined Refueling Canal in LC			13.	0.02083	21.48	stainless steel	Lower Comp.	
				4.0	21.48	concrete		Lower Comp.
Crane Wall between LC & DE			41.5	3.0	33.72	concrete	Lower Comp.	Dead End
Crane Wall in LC			41.5	3.0	29.37	concrete	Lower Comp.	insulated
Crane Wall in UC			41.5	3.0	32.44	concrete	Upper Comp.	insulated
Refueling Canal in Contact with Upper and Lower Compartment	2551	0.02083				stainless steel	Upper Comp.	
		3.87				concrete		Lower Comp.
Refueling Canal in Contact with Annular Region	1,260	0.02083				stainless steel	Upper Comp.	
		3.0				concrete		annulus
Concrete Structure between Upper and Lower Compartment	13,081	2.34				concrete	Upper Comp.	Lower Comp.
Interior Concrete	2278	3.0				concrete	Upper Comp.	insulated
Containment Shell	24,646	0.05417				carbon steel	Upper Comp.	annulus
LC Steel Heat Sink	24,999	0.03674				carbon steel	Lower Comp.	insulated
UC Steel Heat Sink	11669	0.4229				carbon steel	Upper Comp.	insulated
Dead-End Steel Heat Sink	8610	0.074375				carbon steel	DE Comp.	insulated

Material Properties	Thermal Conductivity (BTU/hr-ft-°F)	Volumetric Heat Capacity (BTU/ft3-°F)
Concrete	0.84	30.24
Carbon Steel	27.3	59.2
Stainless Steel	9.87	59.22

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 Containment Noding Diagram

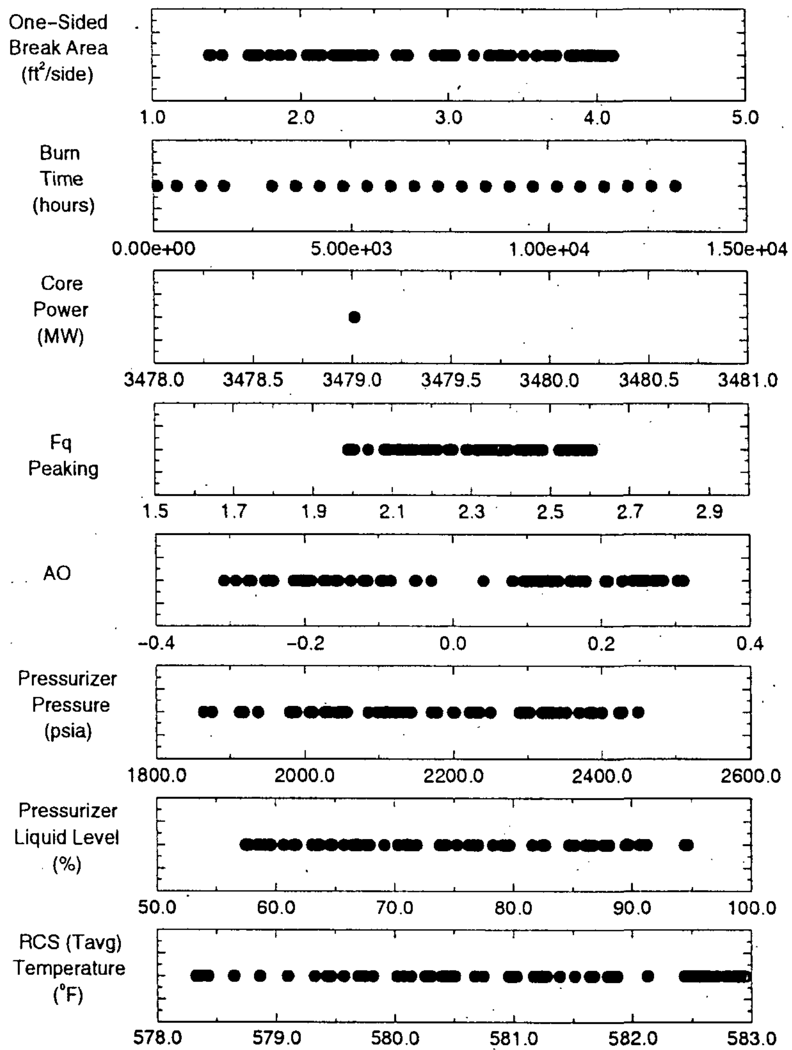

Figure 3-7 Scatter Plot of Operational Parameters

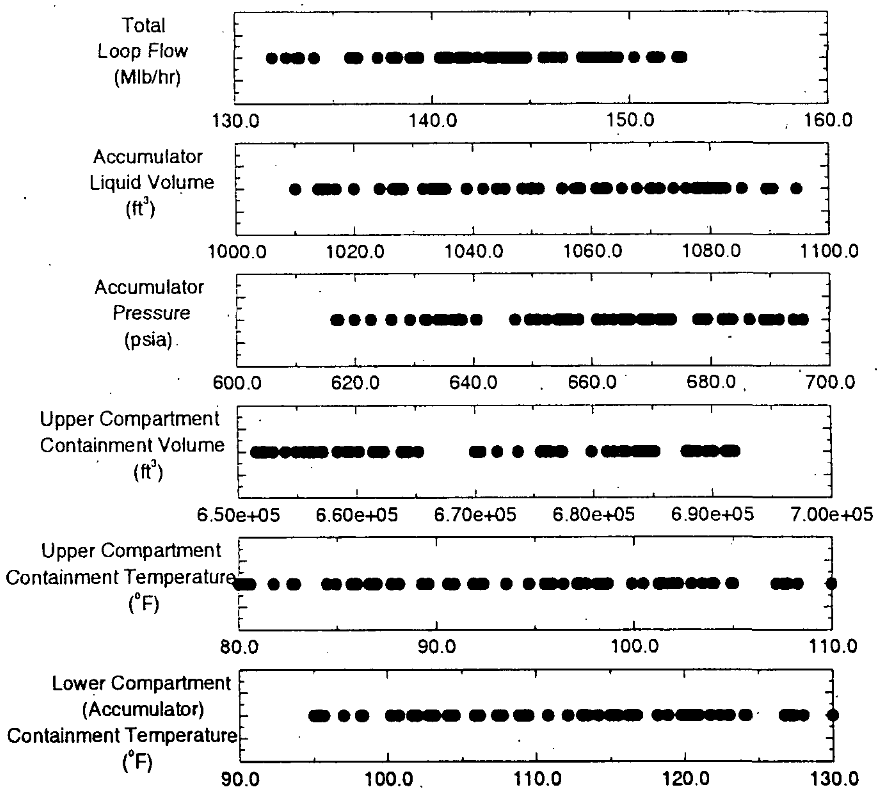

Figure 3-7 Scatter Plot of Operational Parameters ***(Continued)***

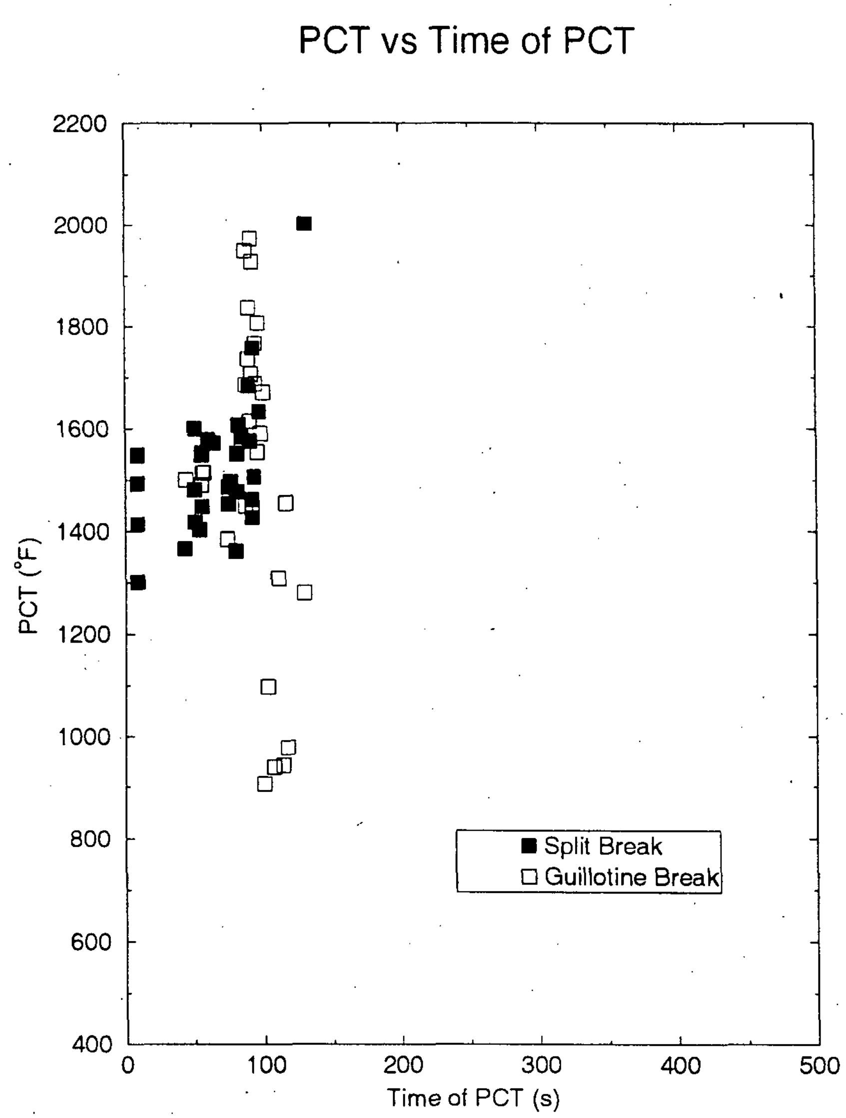

Figure 3-8 PCT versus PCT Time Scatter Plot from 59 Calculations

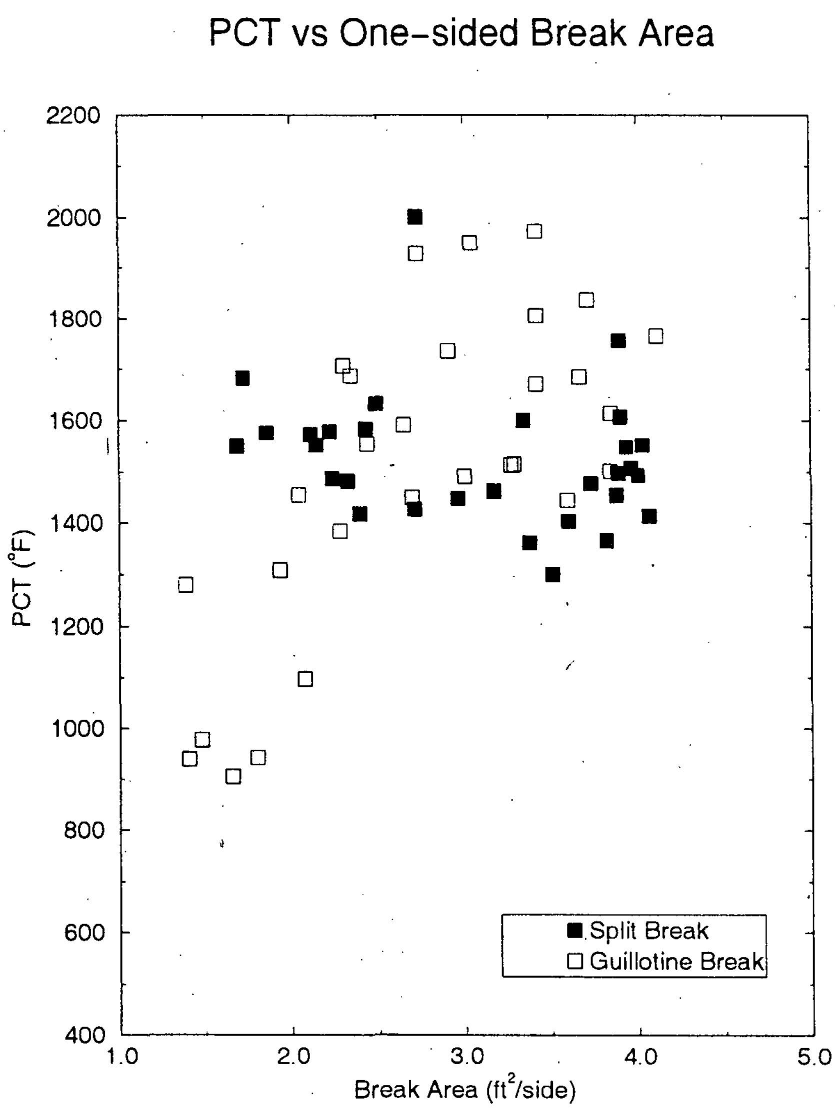

Figure 3-9 PCT versus Break Size Scatter Plot from 59 Calculations

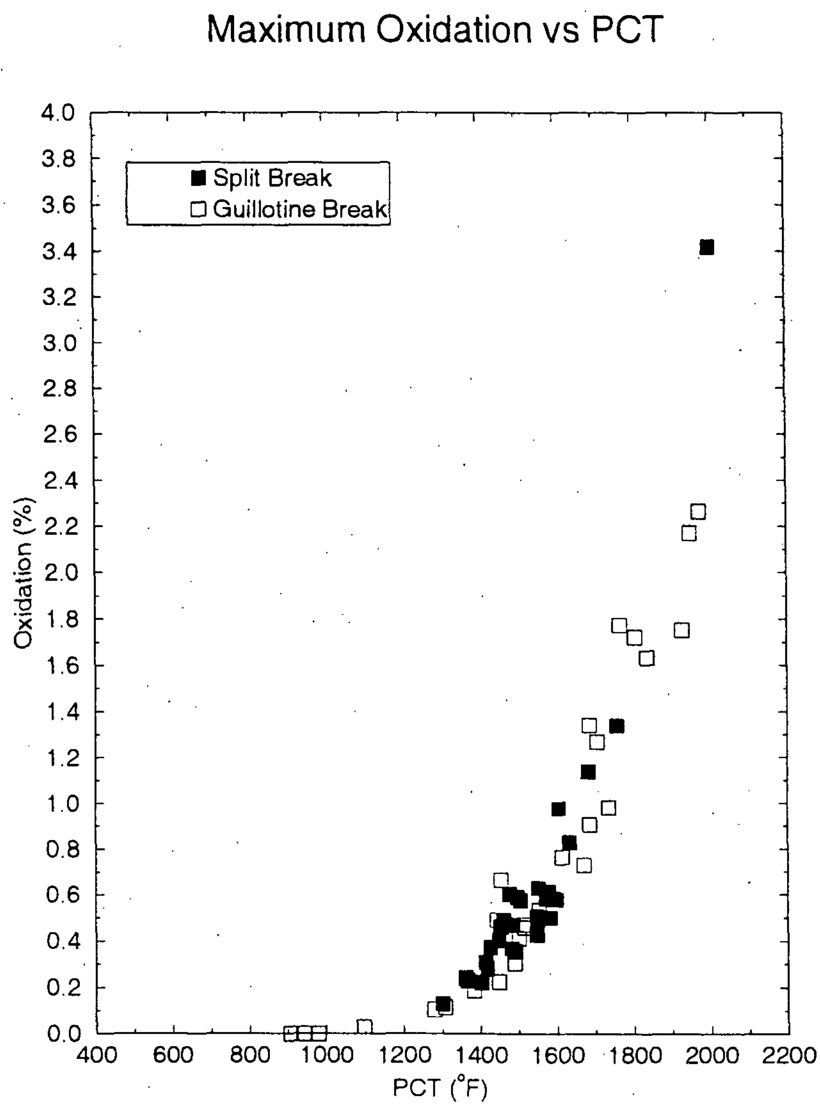

Figure 3-10 Maximum Oxidation versus PCT Scatter Plot from 59 Calculations

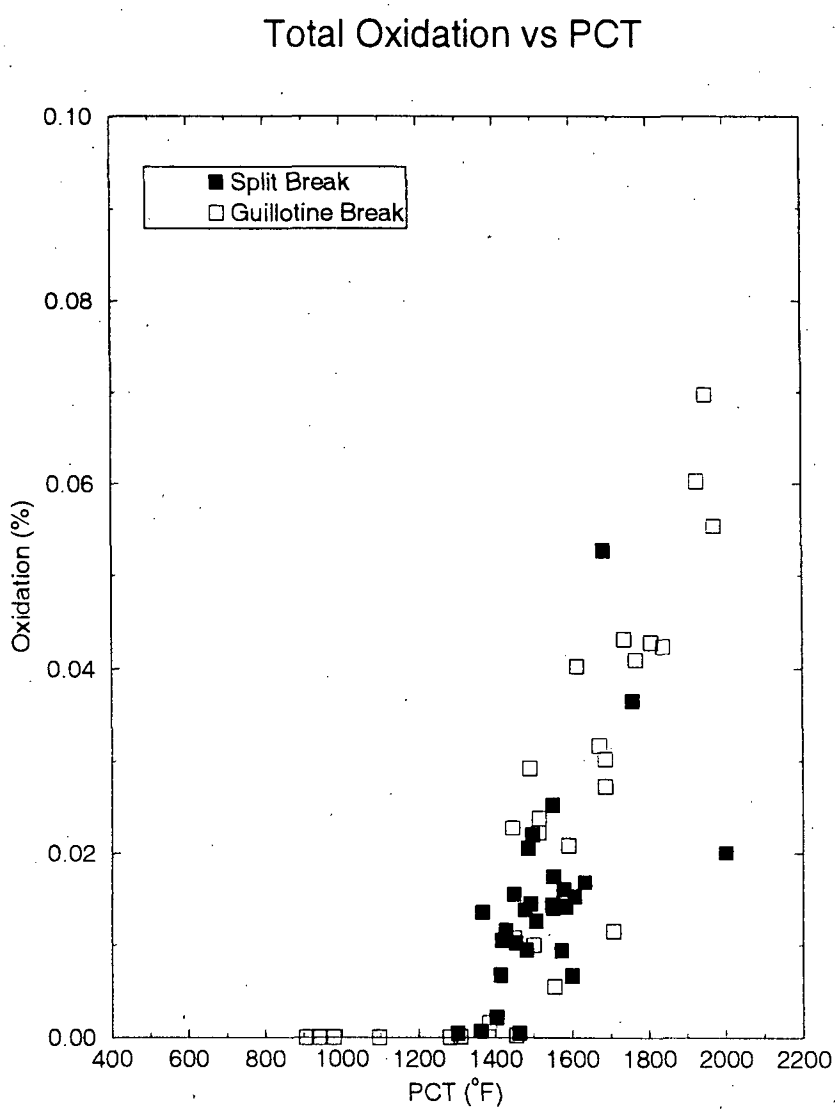

Figure 3-11 Total Oxidation versus PCT Scatter Plot from 59 Calculations

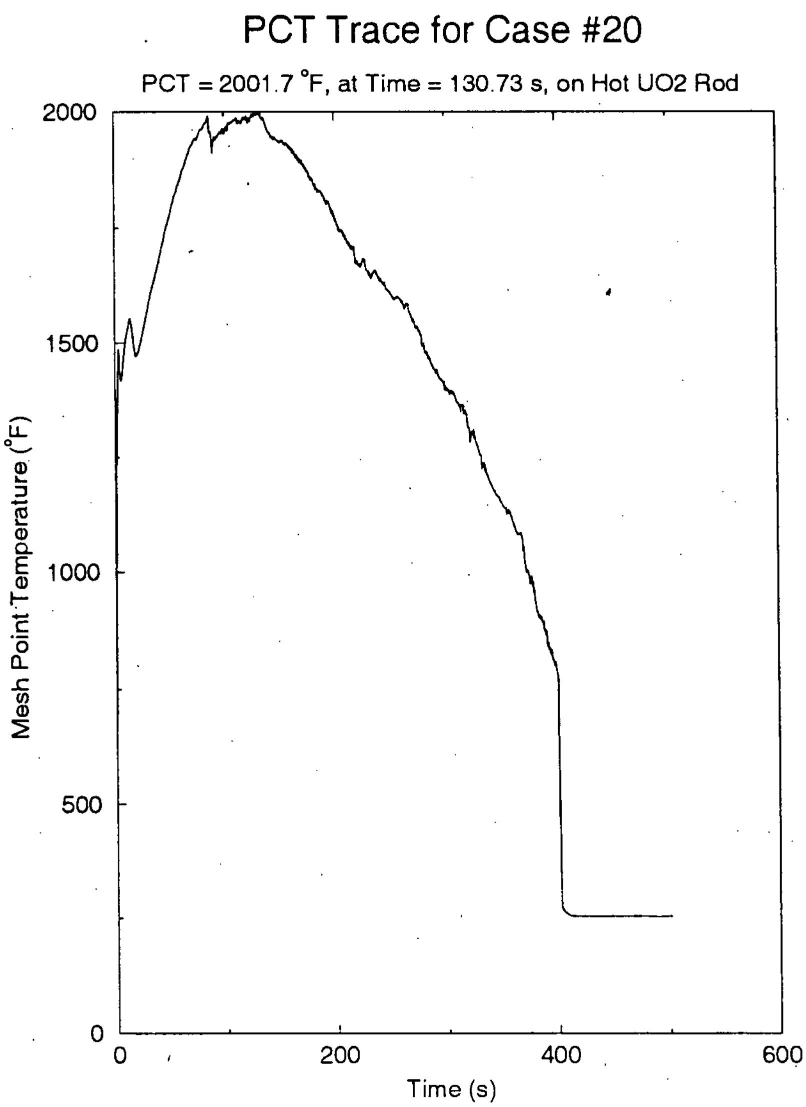

Figure 3-12 Peak Cladding Temperature (Independent of Elevation) for the Limiting Case

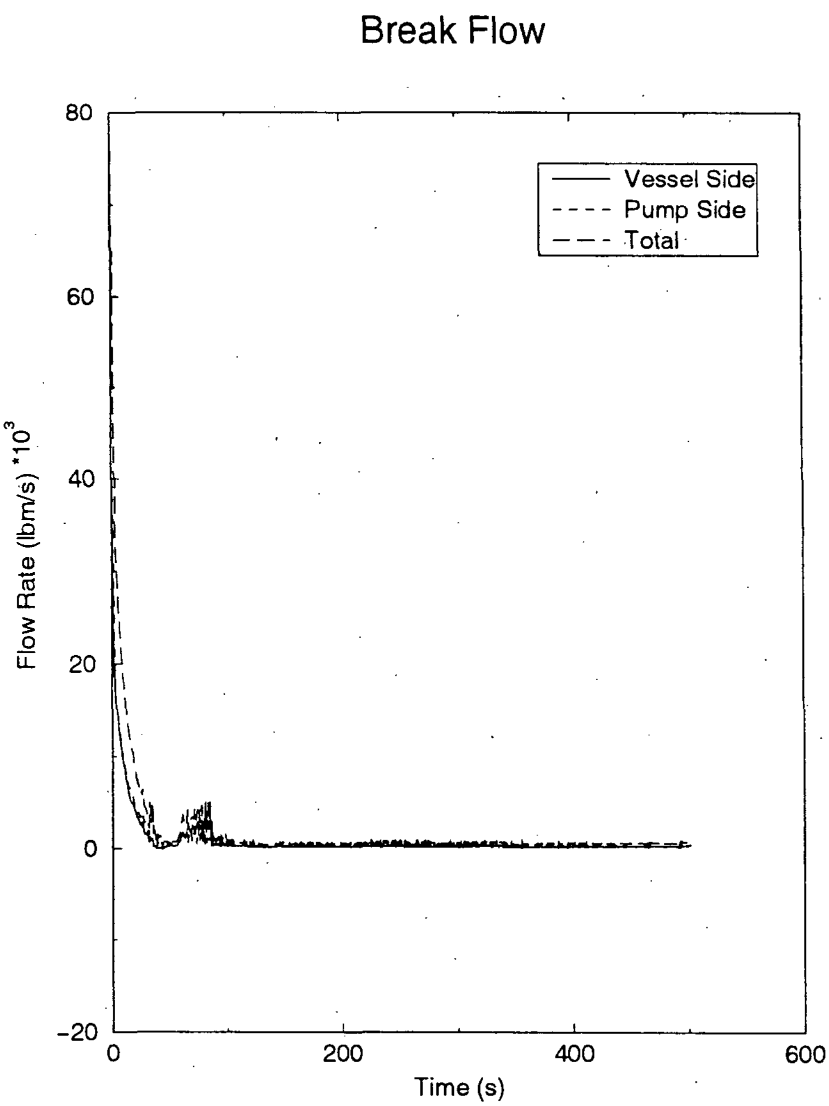

Figure 3-13 Break Flow for the Limiting Case

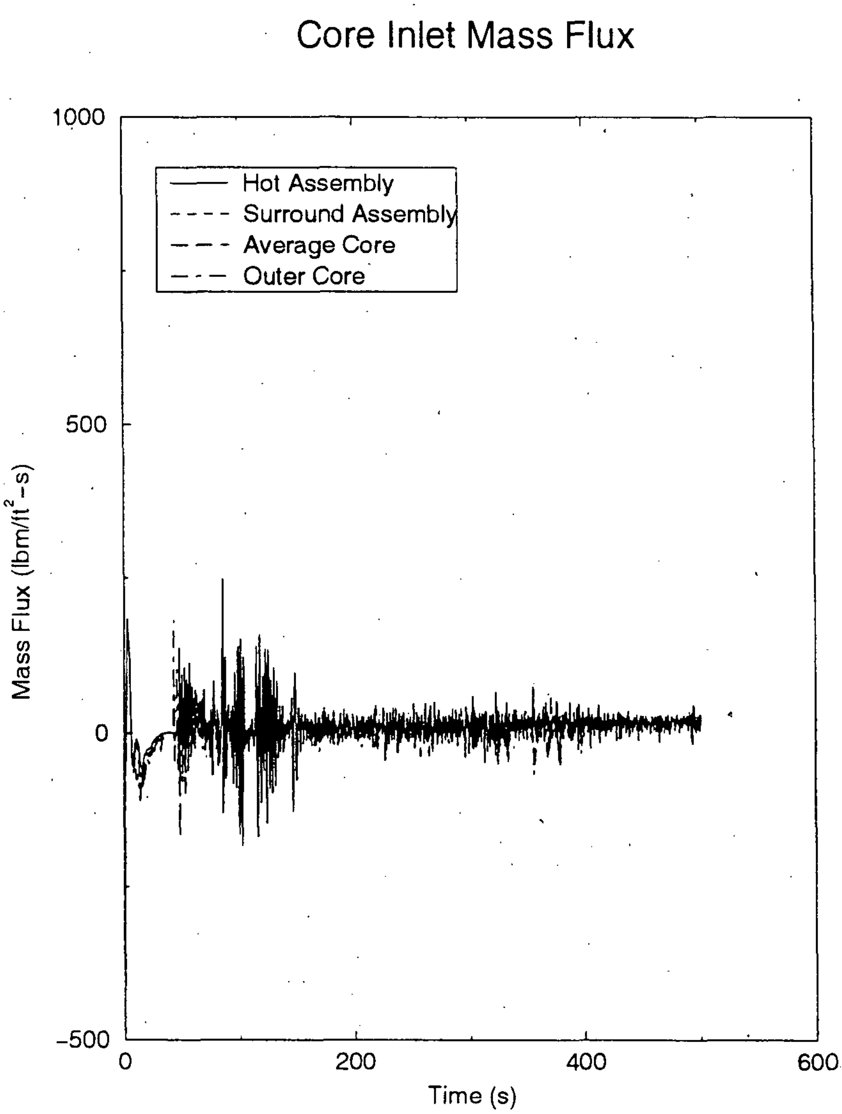

Figure 3-14 Core Inlet Mass Flux for the Limiting Case

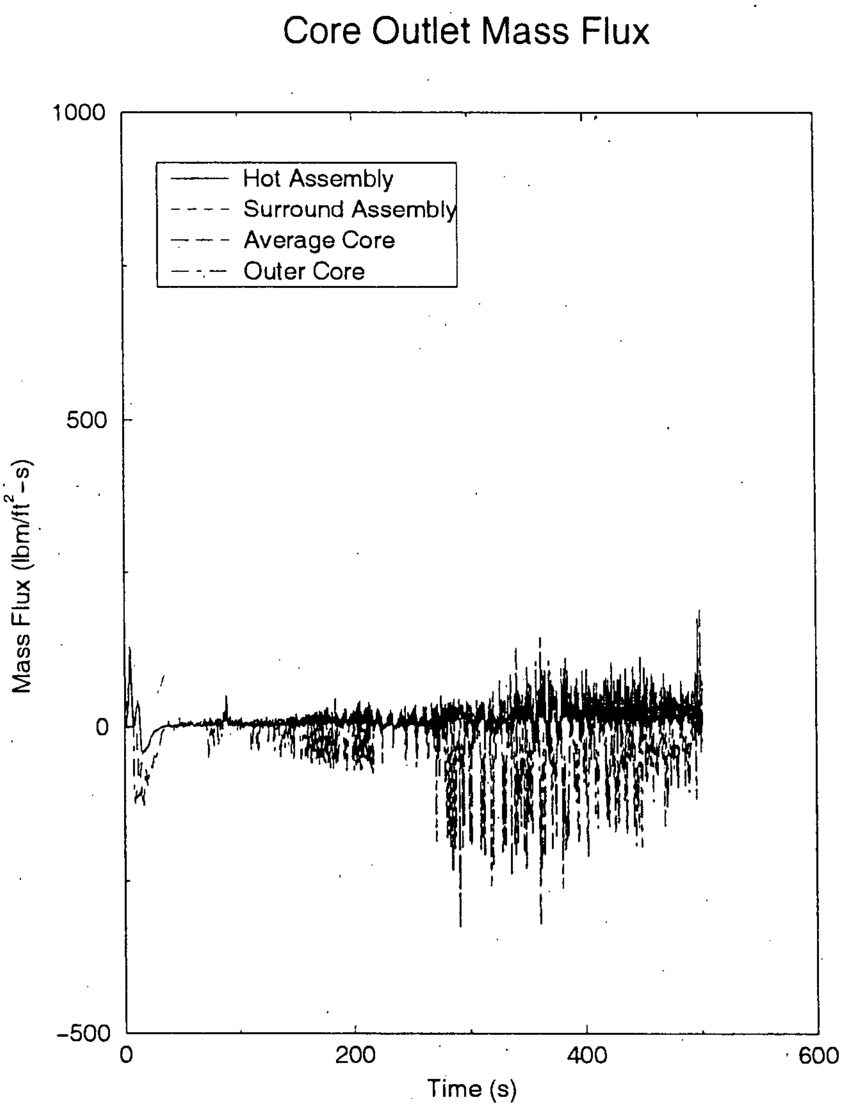

Figure 3-15 Core Outlet Mass Flux for the Limiting Case

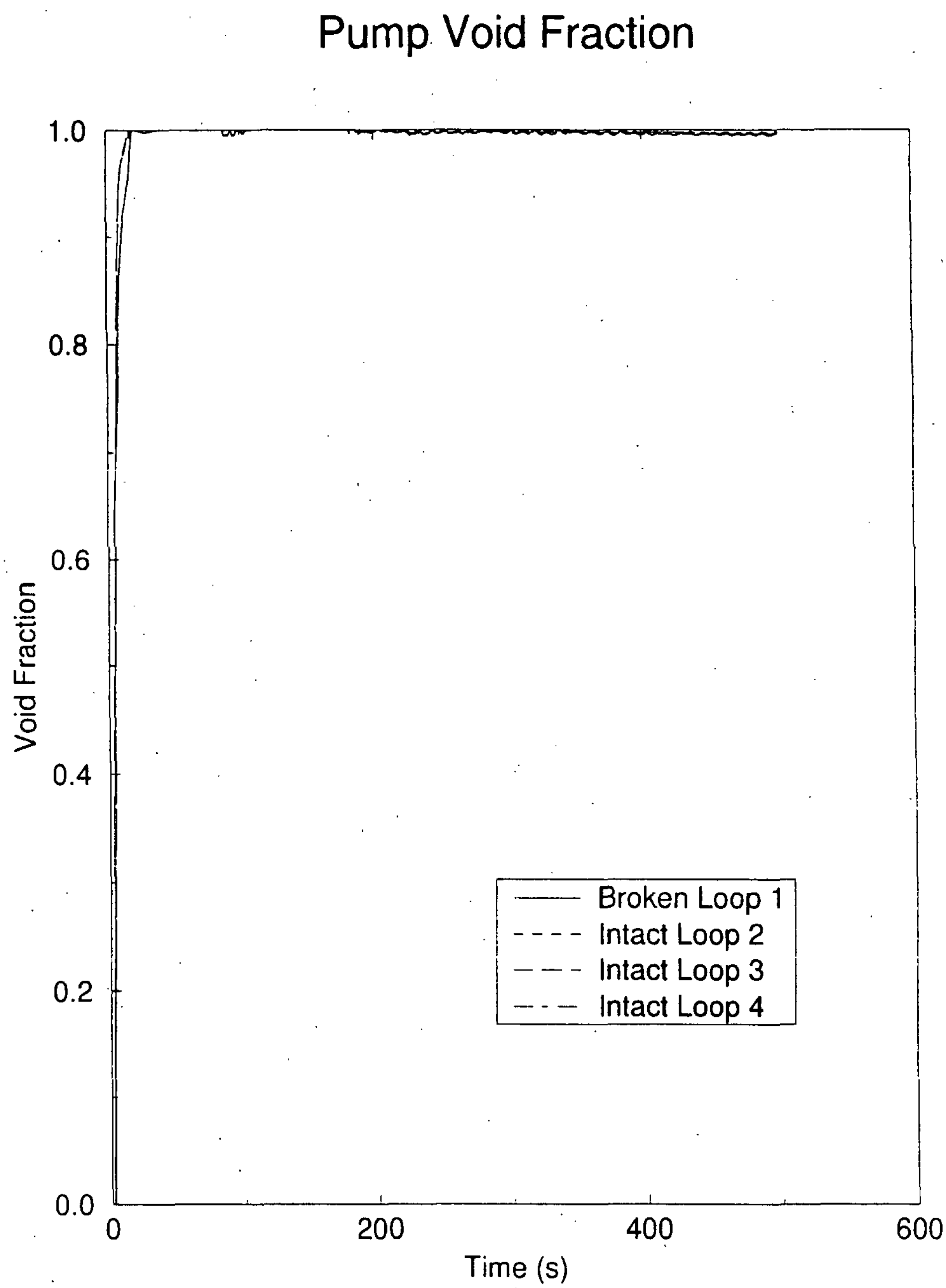

Figure 3-16 Void Fraction at RCS Pumps for the Limiting Case

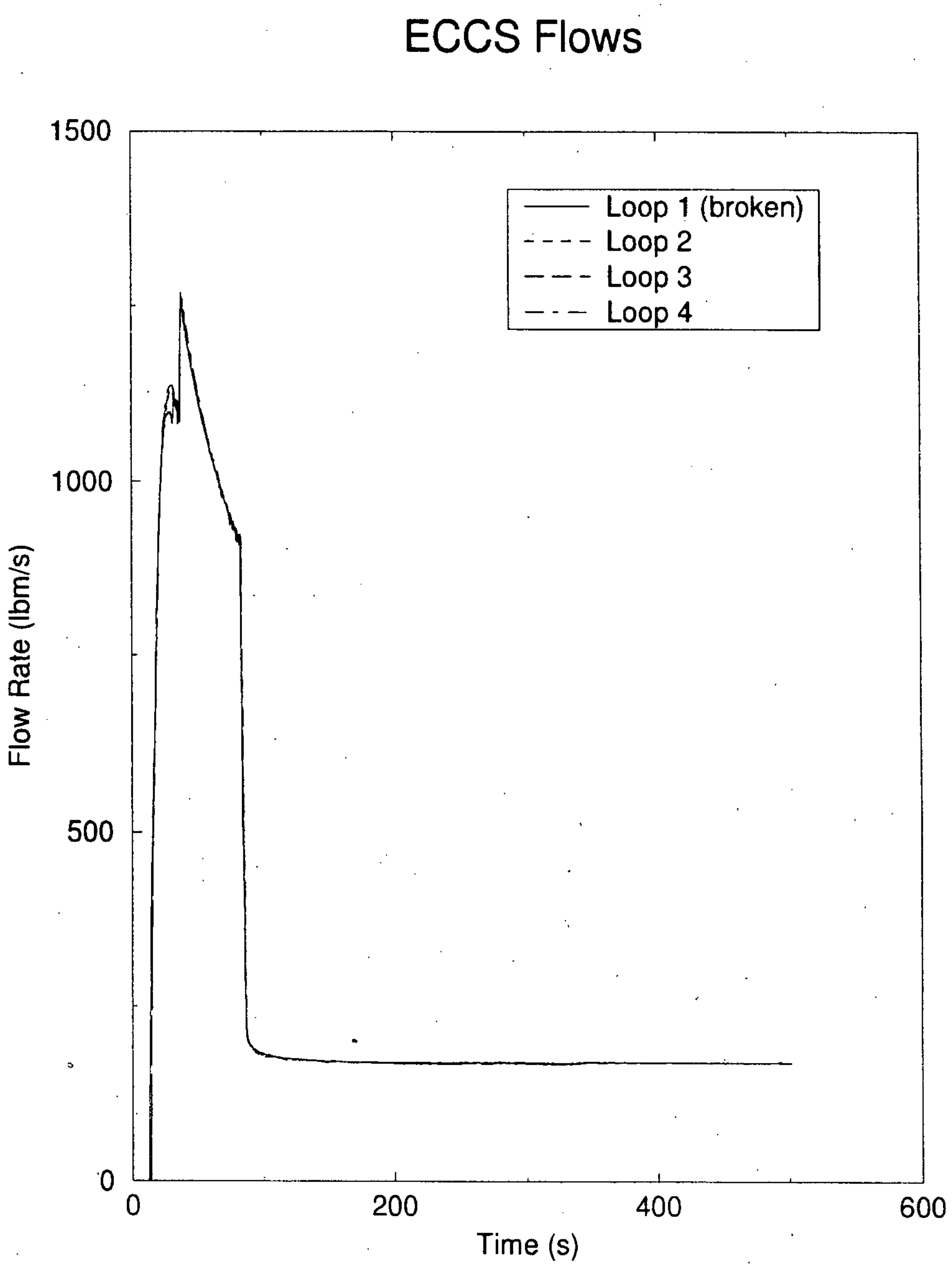

Figure 3-17 ECCS Flows (Includes Accumulator, Charging, SI and RHR) for the Limiting Case

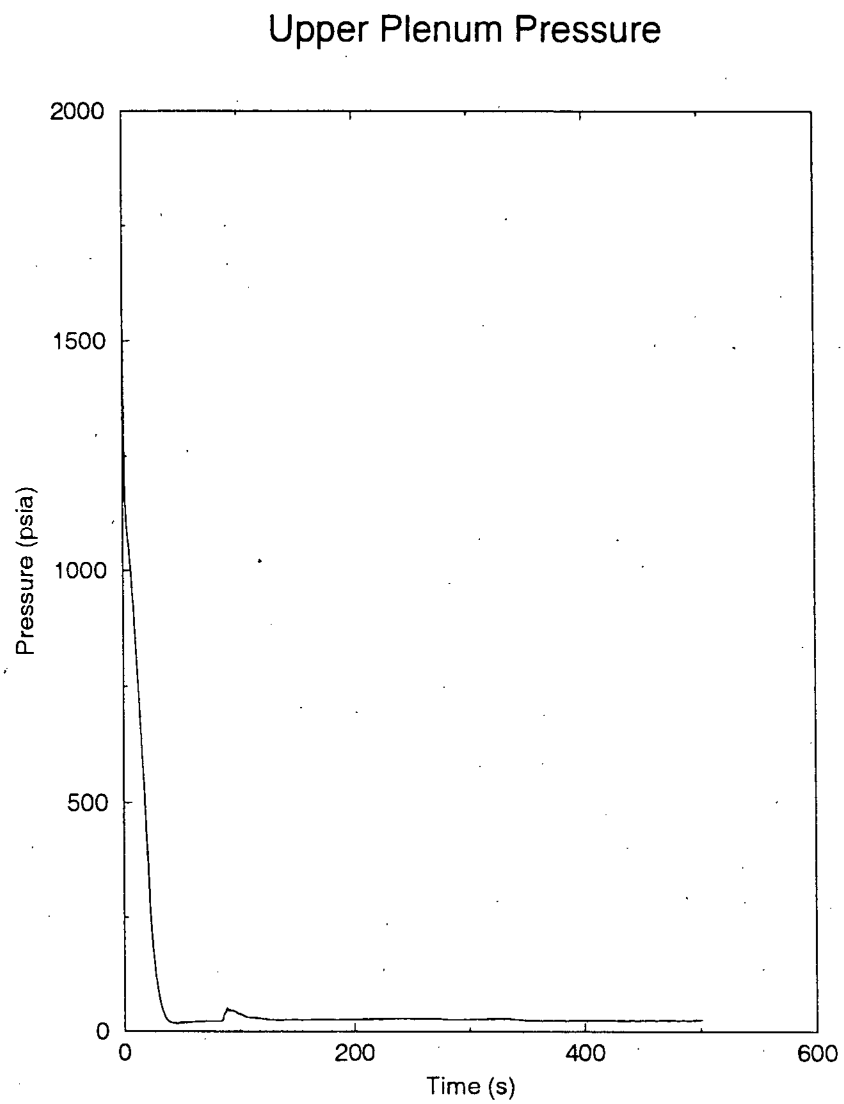

Figure 3-18 Upper Plenum Pressure for the Limiting Case

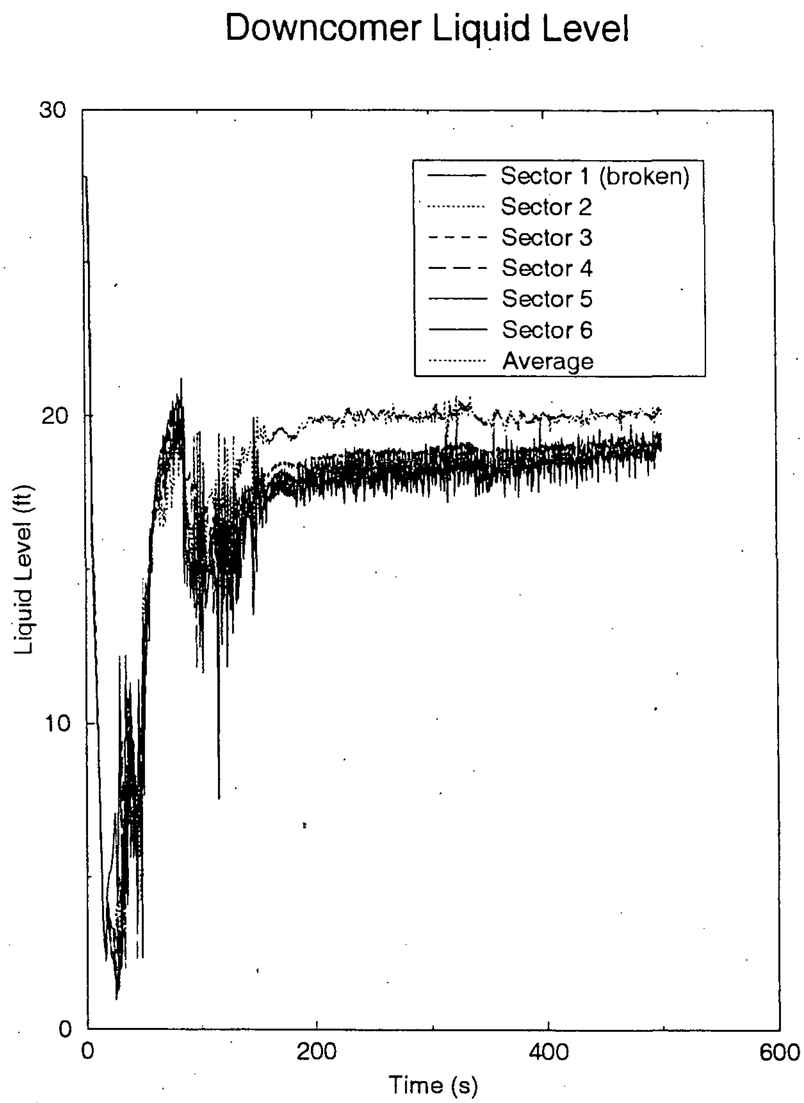

Figure 3-19 Collapsed Liquid Level in the Downcomer for the Limiting Case

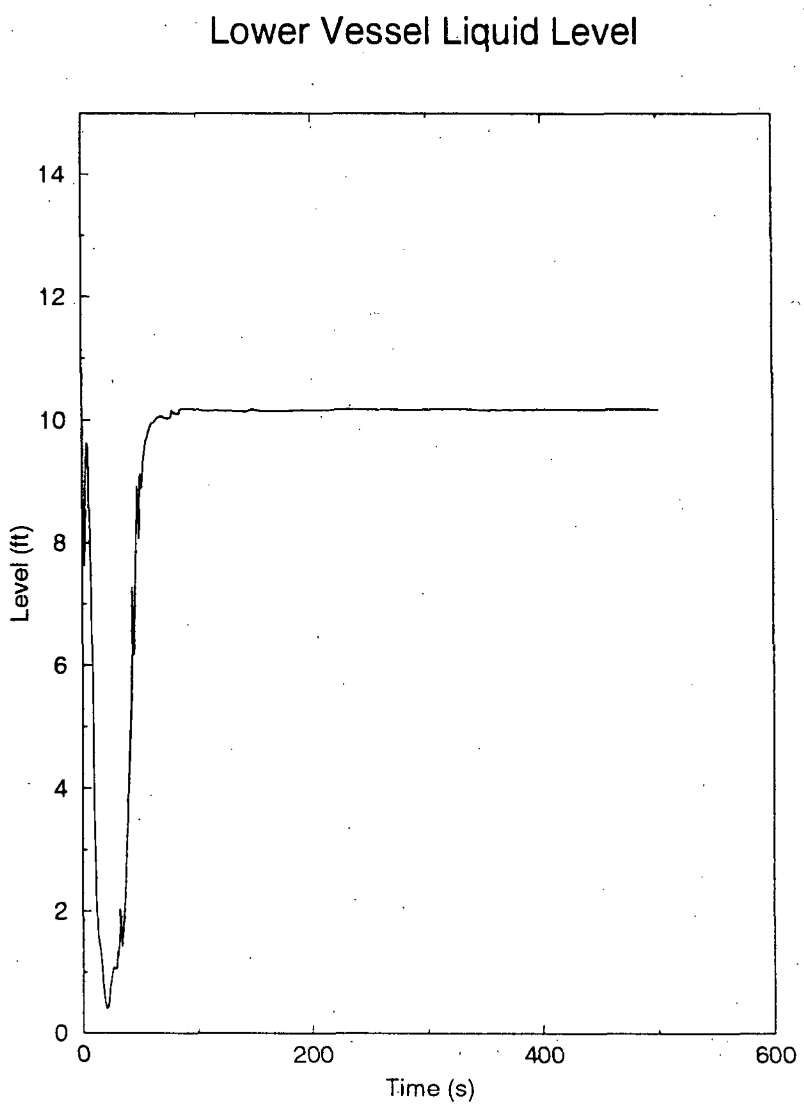

Figure 3-20 Collapsed Liquid Level in the Lower Plenum for the Limiting Case

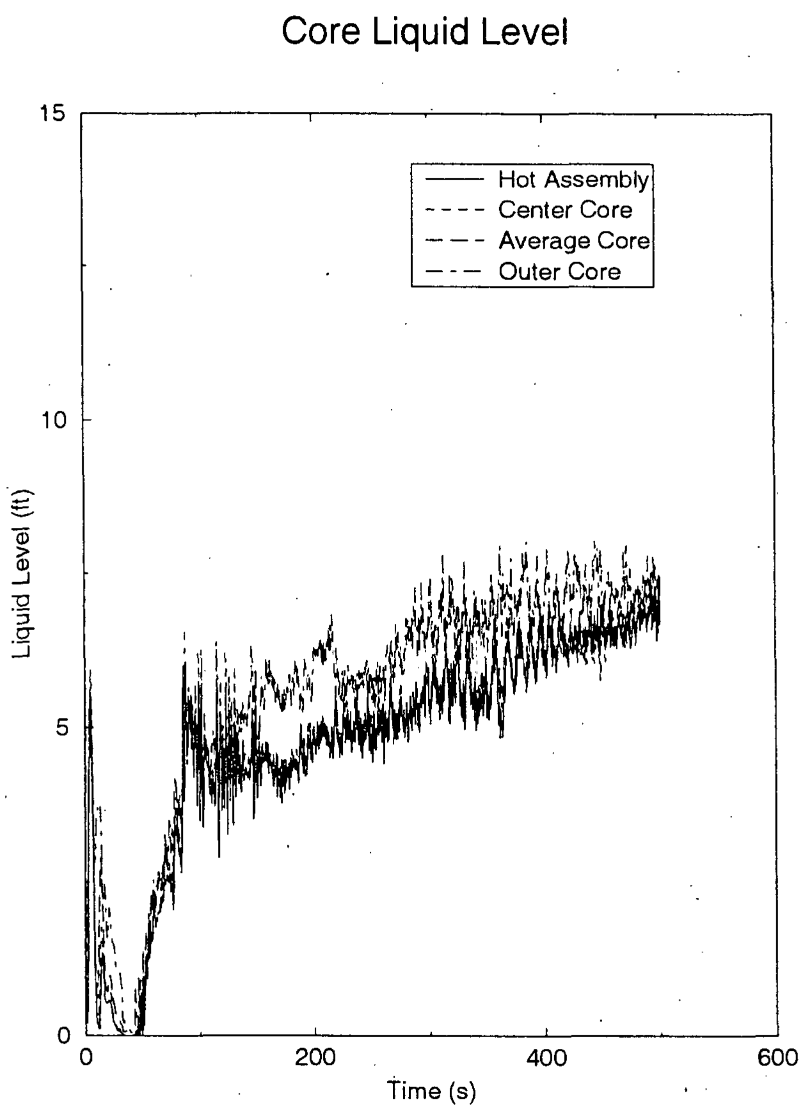

Figure 3-21 Collapsed Liquid Level in the Core for the Limiting Case

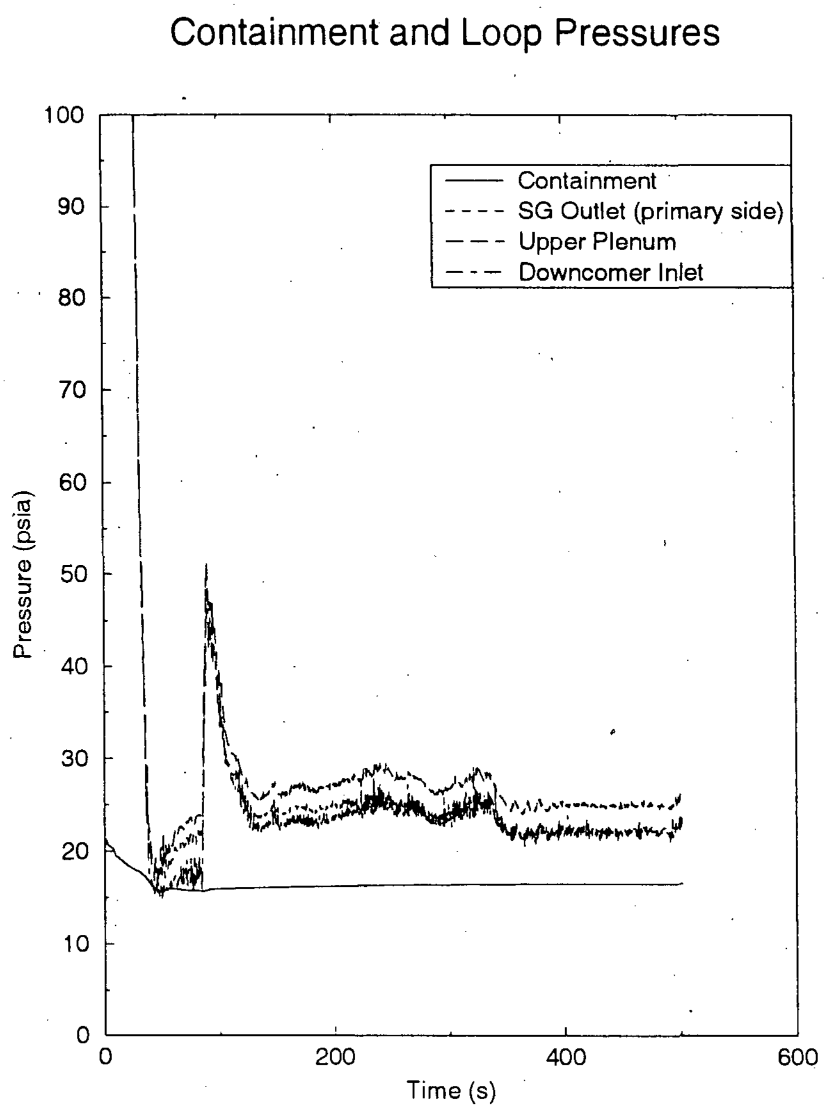

Figure 3-22 Containment and Loop Pressures for the Limiting Case

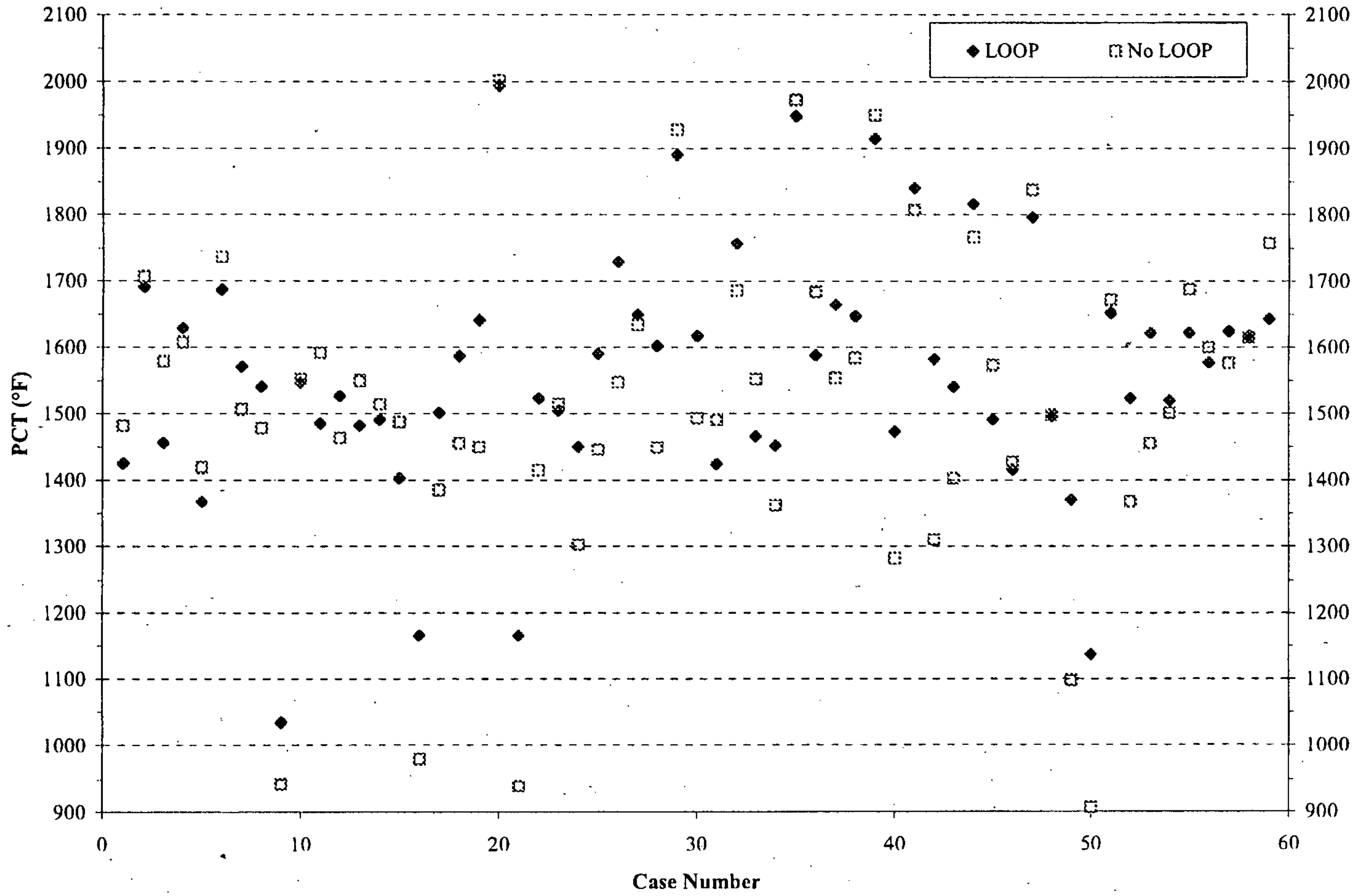

Figure 3-23 GDC 35 LOOP versus No-LOOP Cases

4.0 Conclusions

The results of the RLBLOCA analysis show that the limiting LOOP case has a PCT of 2002 °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 3479 MWt (including uncertainty), a steam generator tube plugging level of up to 15 percent in all steam generators, a total peaking factor (Fq) of 2.65 (including uncertainty) and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.706 (including uncertainty) with no axial or burnup dependent power peaking limit.

5.0 References

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8. NUREG/CR-0994, "A Radiative Heat Transfer Model for the TRAC Code" November 1979.

9. J.P. Holman, Heat Transfer, 4th Edition, McGraw-Hill Book Company, 1976

10. EMF-CC-130, "HUXY: A Generalized Multirod Heatup Code for BWR Appendix K LOCA Analysis Theory Manual," Framatome ANP, May 2001.

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.

13. Letter from Pedro Salas, Tennessee Valley Authority to U.S. NRC, TVA-SQN-TS-01-08, Sequoyah Nuclear Plant (SQN), Units 1 & 2, Technical Specification (TS) Change No. 01-08, "Increase Maximum Allowed Reactor Power Level to 3455 Mega-Watt Thermal (MWt)," November 15, 2001 (US NRC ADAMS Accession # ML013470345)

6.0 Addendum - Additional Information Supporting EMF-2103 Revision 0

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.

6.1 Reactor Power

<u>Question:</u> *Reactor Power - Table 3-3, Item 2.1, and its associated Footnote 1 indicate 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 . . ."*

<u>Response:</u> As indicated in Item 2.1 of Table 3-3 herein, the assumed reactor core power for the Sequoyah realistic large break loss-of-coolant accident is 3479 MWt. This value represents the plant rated thermal power (i.e., total reactor core heat transfer rate to the reactor coolant system) of 3455 MWt with a maximum power measurement uncertainty of 0.7 percent (24 MWt) added to the rated thermal power.

The power measurement uncertainty assumption discussed in 10CFR50, Appendix K was previously reduced for Sequoyah from 2.0 percent of the plant rated thermal power to 0.7 percent based on the installation of a leading edge flow meter (LEFM) system to measure main feedwater flow. The improved feedwater flow measurement accuracy provided by the LEFM allowed for a power measurement uncertainty recovery of 1.3 percent. This power level assumption is a change to the approved RLBLOCA EM (Reference 1).

The basis for the current 0.7 percent measurement uncertainty assumption is documented in Topical Report No. WCAP-15669, Revision 0. This report was submitted to NRC in Reference 13. NRC review and acceptance of the current power measurement uncertainty has been documented in Reference 7.

6.2 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 Sequoyah Unit 2 LAR 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 with a mean value of 626 K and a standard deviation of 33.6 K, making it very unlikely that Tmin would exceed 755 K (900 °F). For the Sequoyah Unit 2 cycle 16 case set T_{min} was never sampled above 690.7 K (783.6 °F). This is a change to the approved RLBLOCA EM (Reference 1).

6.3 Rod-to-Rod Thermal Radiation

Question: *Provide justification that the S-RELAP5 rod-to-rod thermal radiation model applies to the SQN-2 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 RLBLOCA 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 RLBLOCA 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.

6.3.1 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 6-1 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 6-1 Typical Measurement Uncertainties and Local Peaking Factors

Plant	$F_{\Delta H}$ 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

6.3.2 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 161 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 6-2, between the test bundle and a modern 17x17 assembly.

Table 6-2 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 6-3 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.

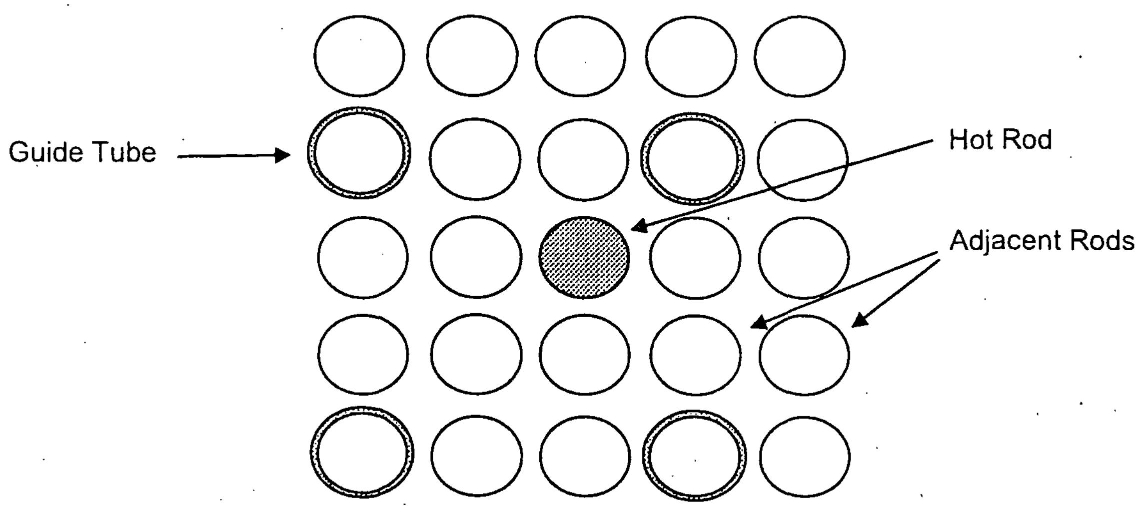


Figure 6-1 R2RRAD 5 x 5 Rod Segment

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 6.3.1, the surrounding 24 rods were set to a temperature estimated for rods of 4 percent lower power. The boundary temperature was estimated based 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.

$$T_{24\ rods} = 0.96 \cdot (PCT - T_{sat}) + T_{sat} \quad \text{and}$$
$$T_{rurrounding\ region} = 0.94 \cdot (PCT - T_{sat}) + T_{sat}.$$

T_{sat} was taken as 270 F.

Figure 6-2 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 6-3 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 7I (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			

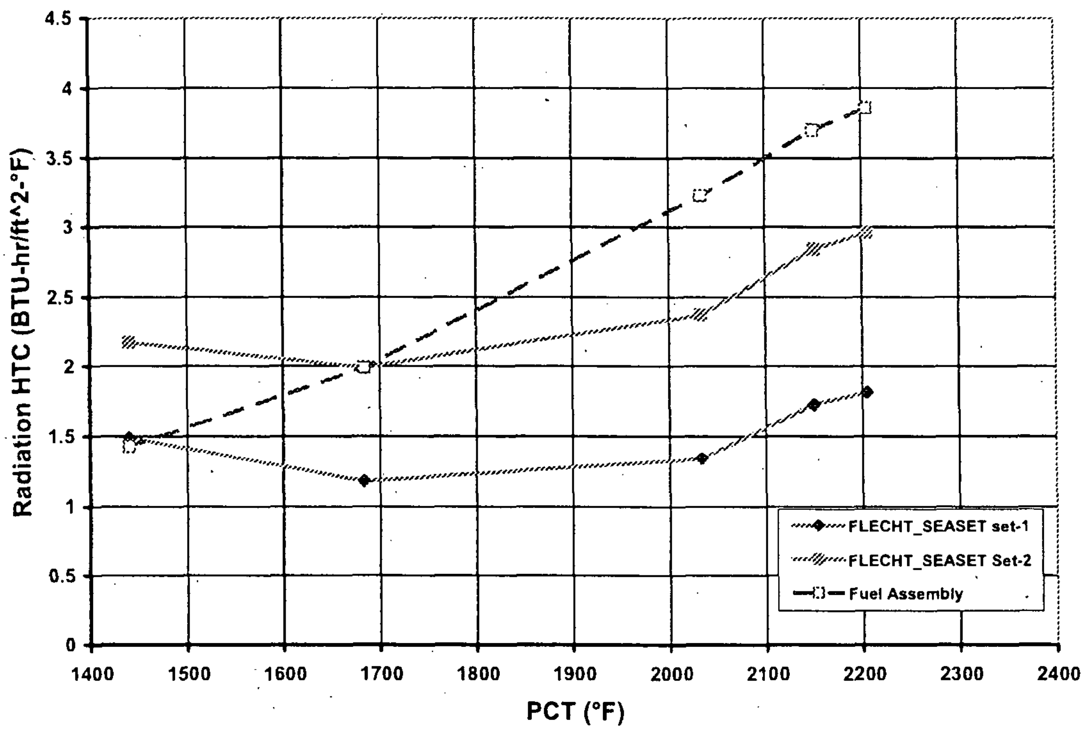


Figure 6-2 Rod Thermal Radiation in FLECHT-SEASET Bundle and in a 17x17 FA

6.3.3 <u>Rod-to-Rod Radiation Summary</u>

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.

6.4 Film Boiling Heat Transfer Limit

Question: *In the SQN-2 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 Sequoyah Unit 2 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).

6.5 Downcomer Boiling

Question: *If the PCT is greater than 1800°F or the containment pressure is less than 30 psia, has the Sequoyah Unit 2 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 Sequoyah Unit 2 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).

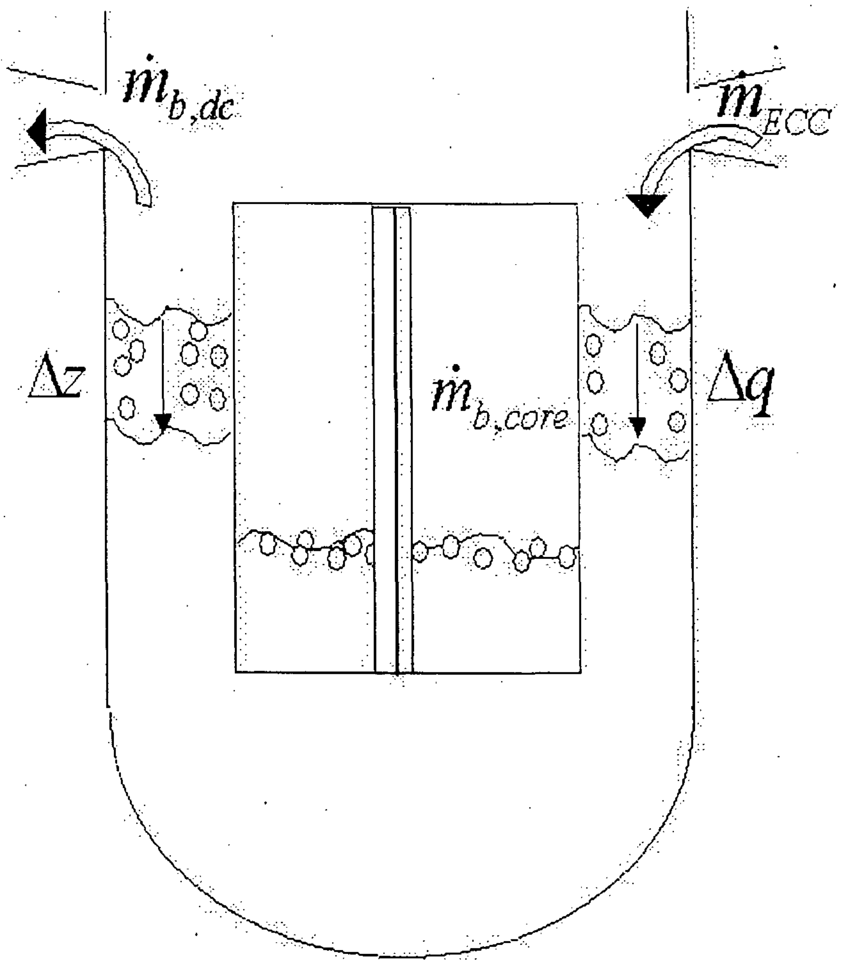


Figure 6-3 Reactor Vessel Downcomer Boiling Diagram

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.

6.5.1 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.

6.5.1.1 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

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

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

$\alpha = k/(\rho\, Cp)$,
k = thermal conductivity,
ρ = density,
Cp = specific heat, and
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 RLBLOCA model. Figure 6-4 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.

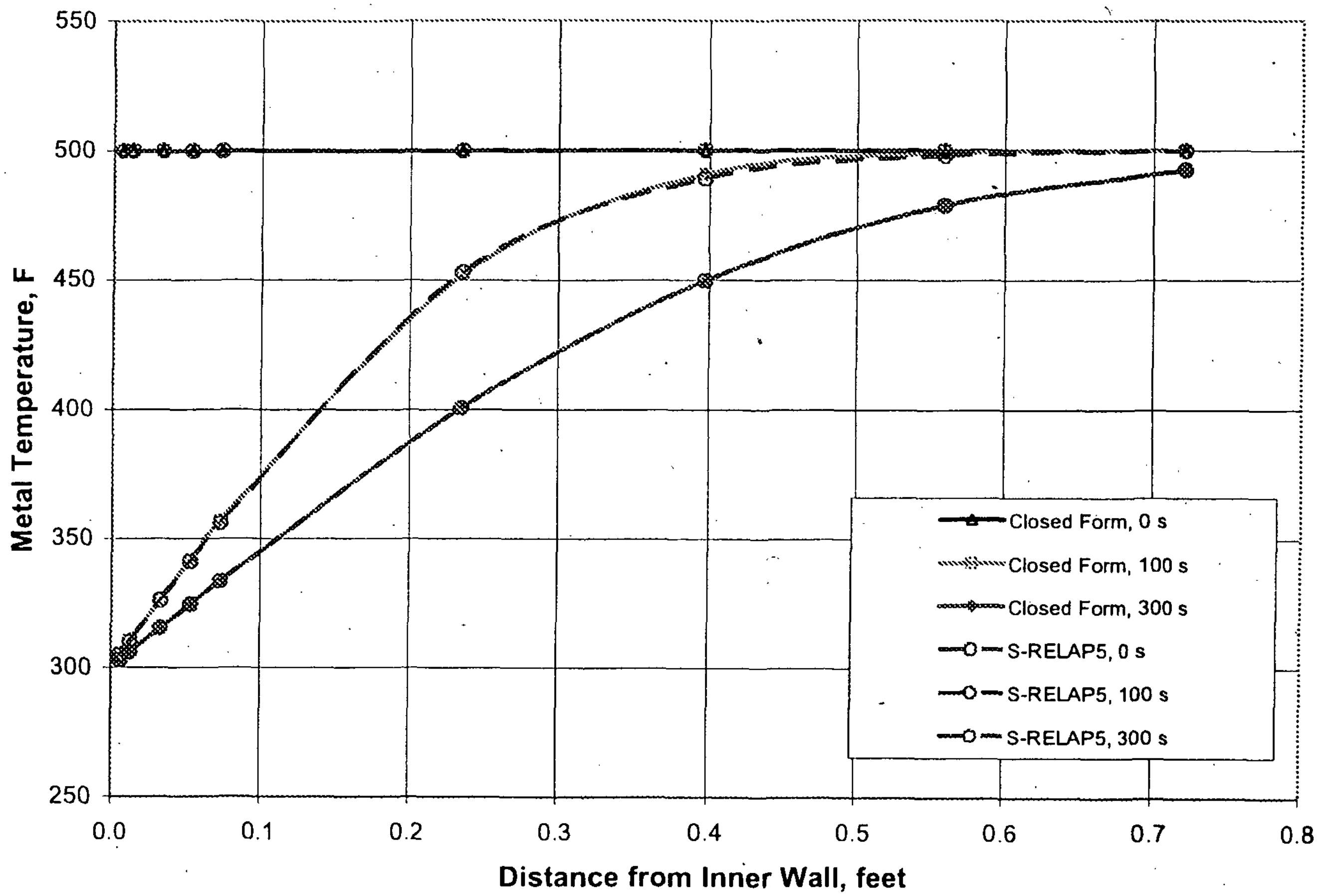


Figure 6-4 S-RELAP5 versus Closed Form Solution

6.5.1.2 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.

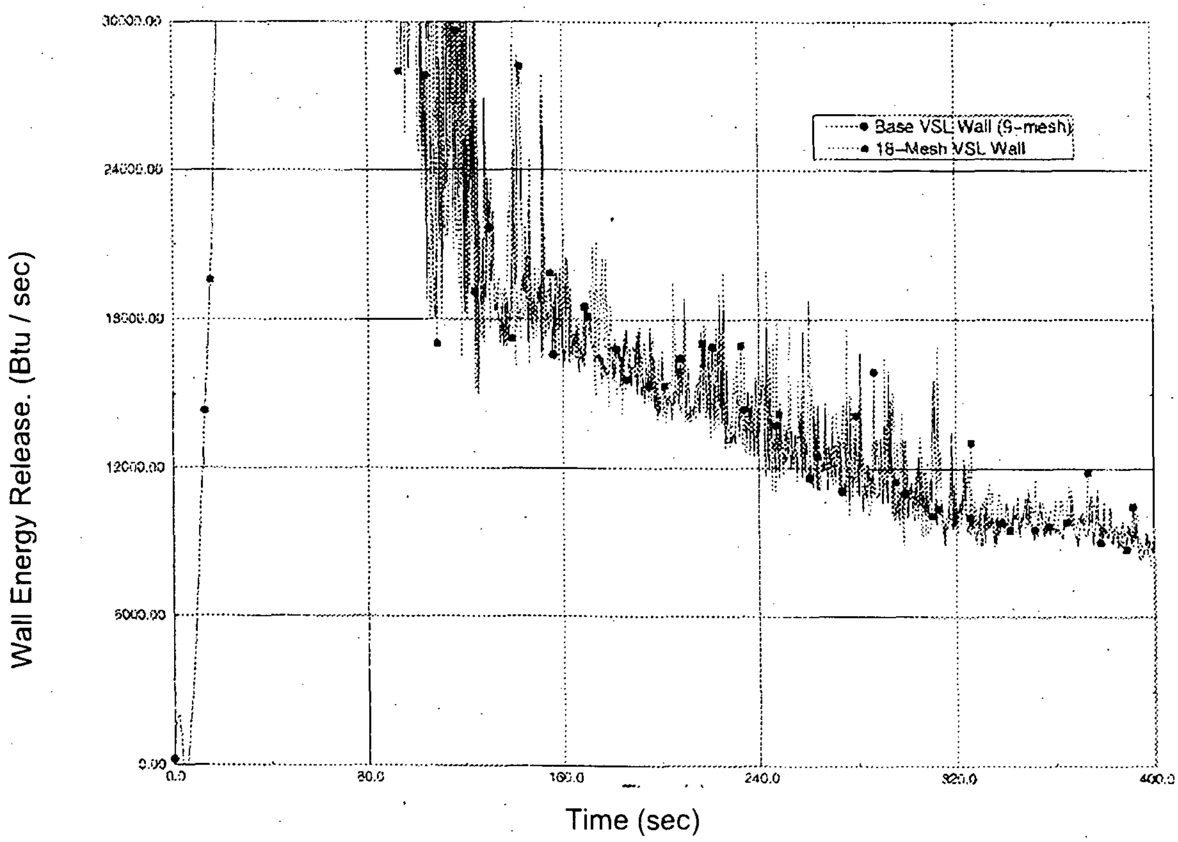


Figure 6-5 Downcomer Wall Heat Release – Wall Mesh Point Sensitivity

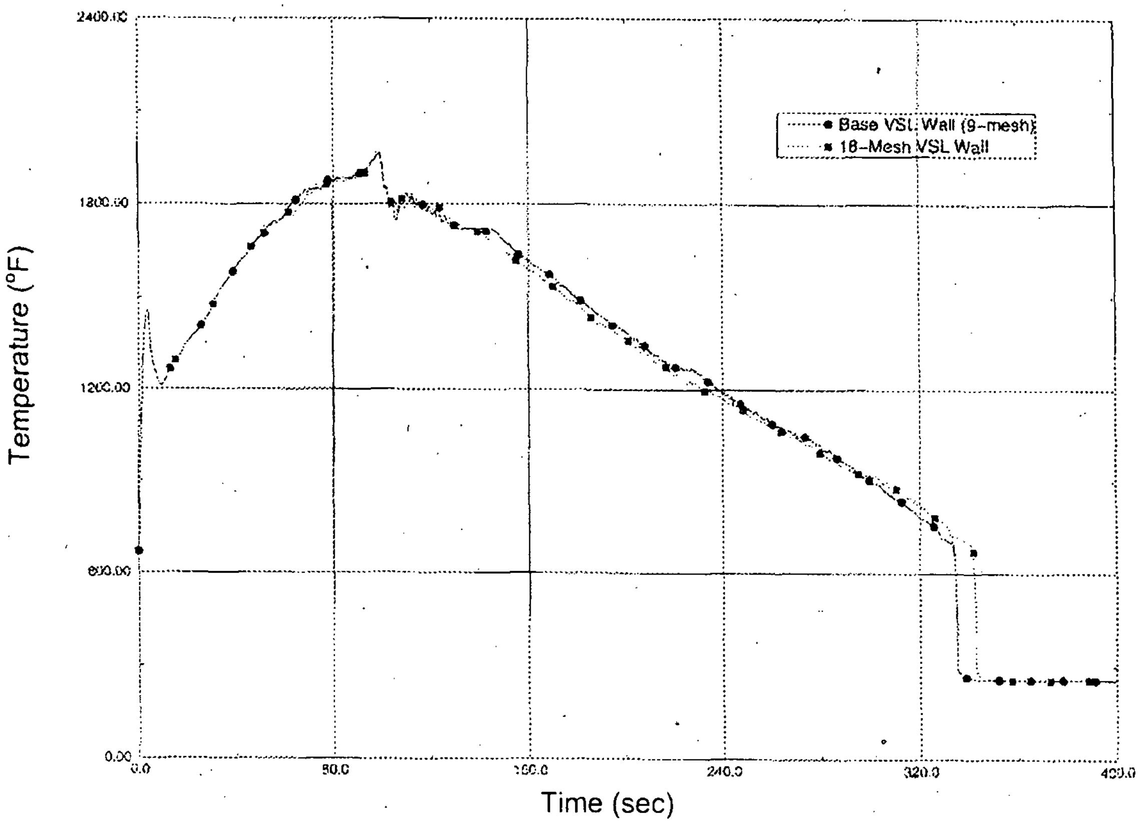


Figure 6-6 PCT Independent of Elevation – Wall Mesh Point Sensitivity

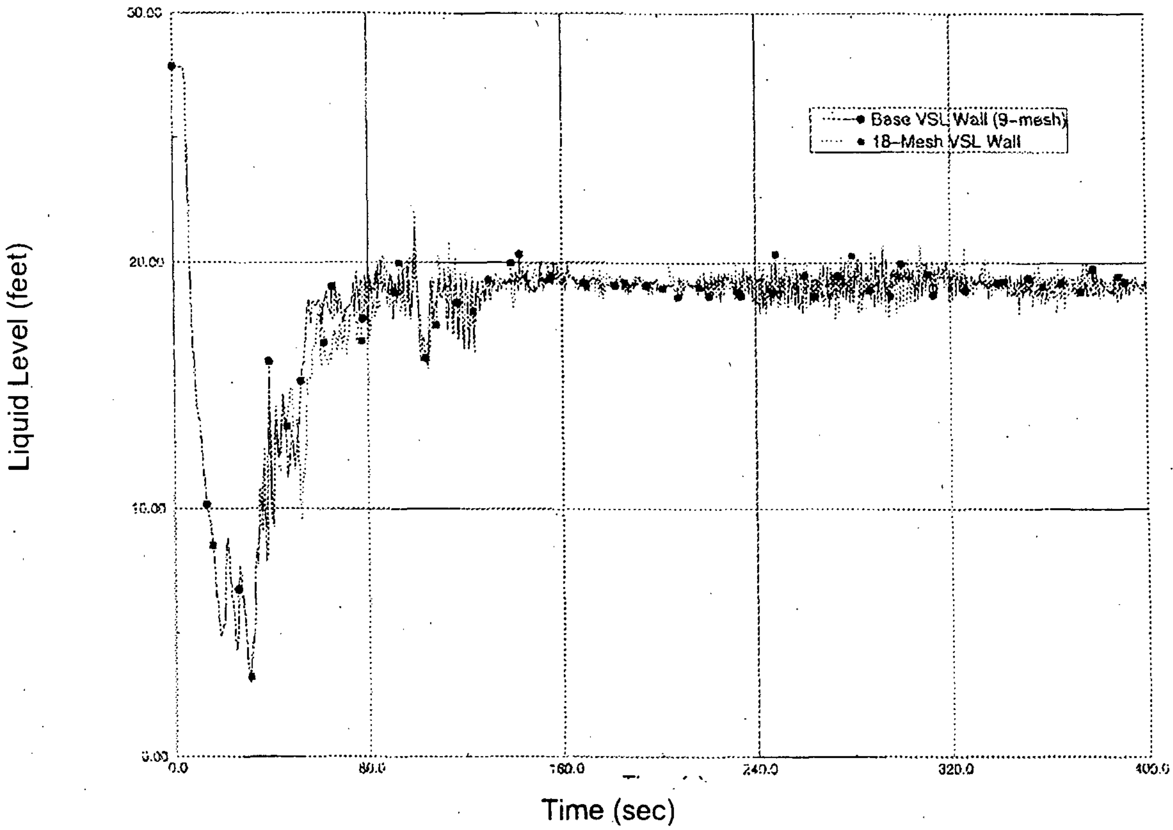


Figure 6-7 Downcomer Liquid Level – Wall Mesh Point Sensitivity

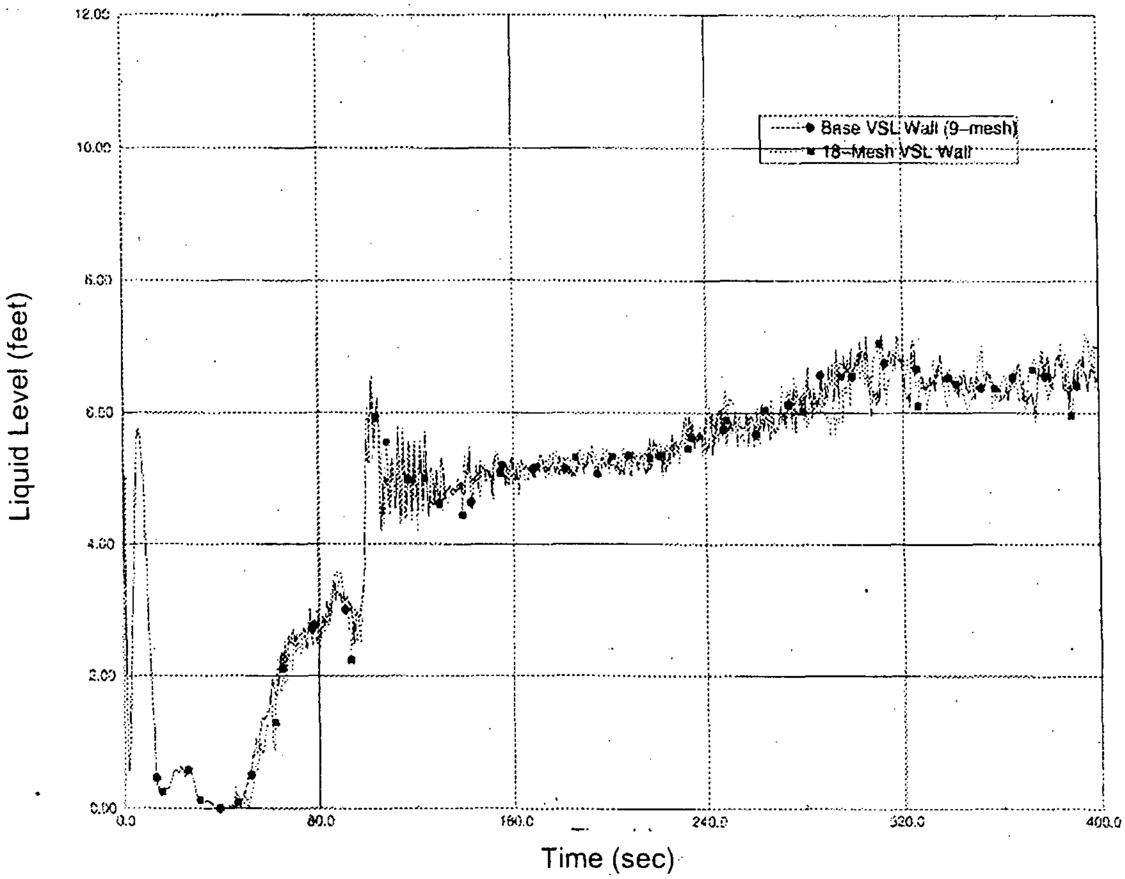


Figure 6-8 Core Liquid Level – Wall Mesh Point Sensitivity

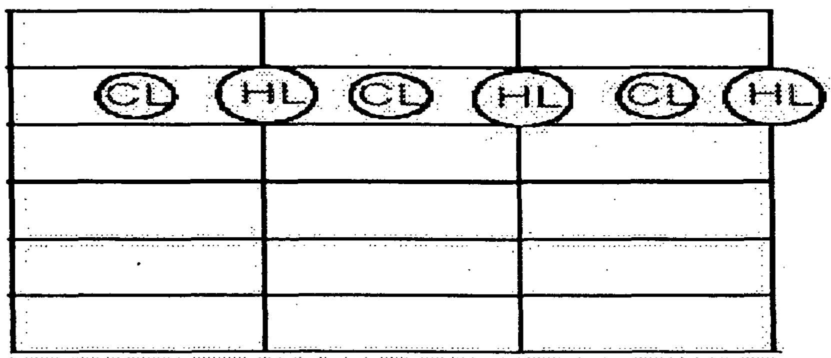
6.5.2 Downcomer Fluid Distribution

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.

6.5.2.1 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 affects 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 6-9). 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 affect 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.

Base model



Revised 9 Region Model

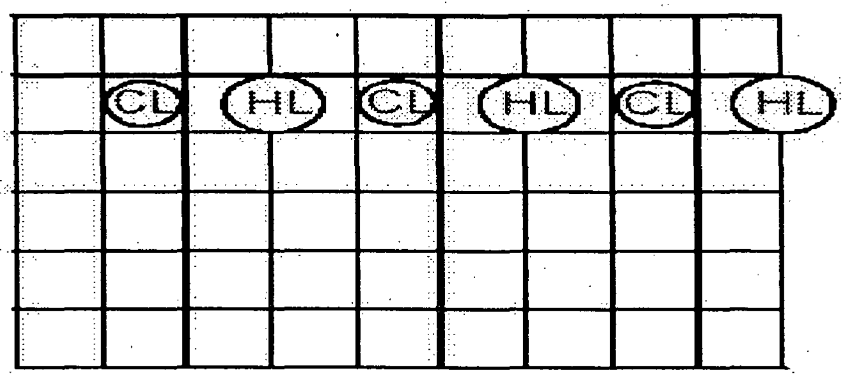


Figure 6-9 Azimuthal Noding

6.5.2.2 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 4-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 6-10 provides the containment back pressure for the base modeling. Figures 6-11 through 6-14 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.

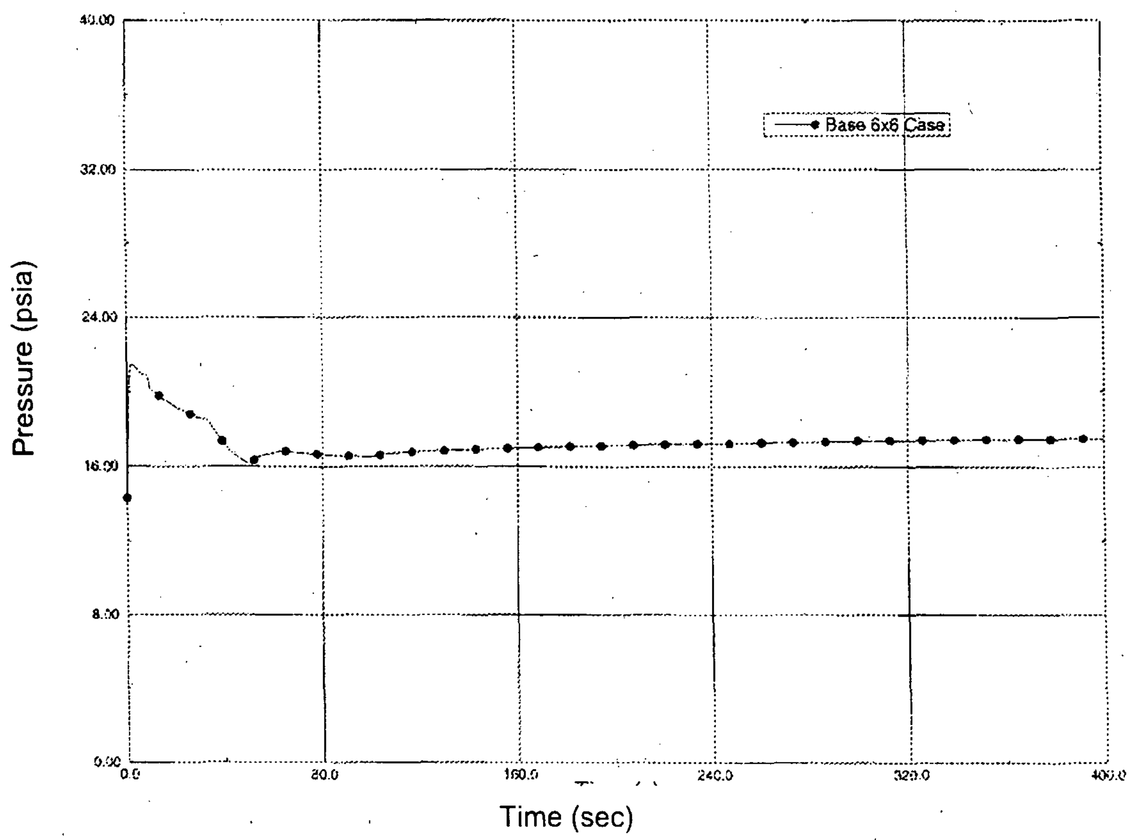


Figure 6-10 Lower Compartment Pressure versus Time

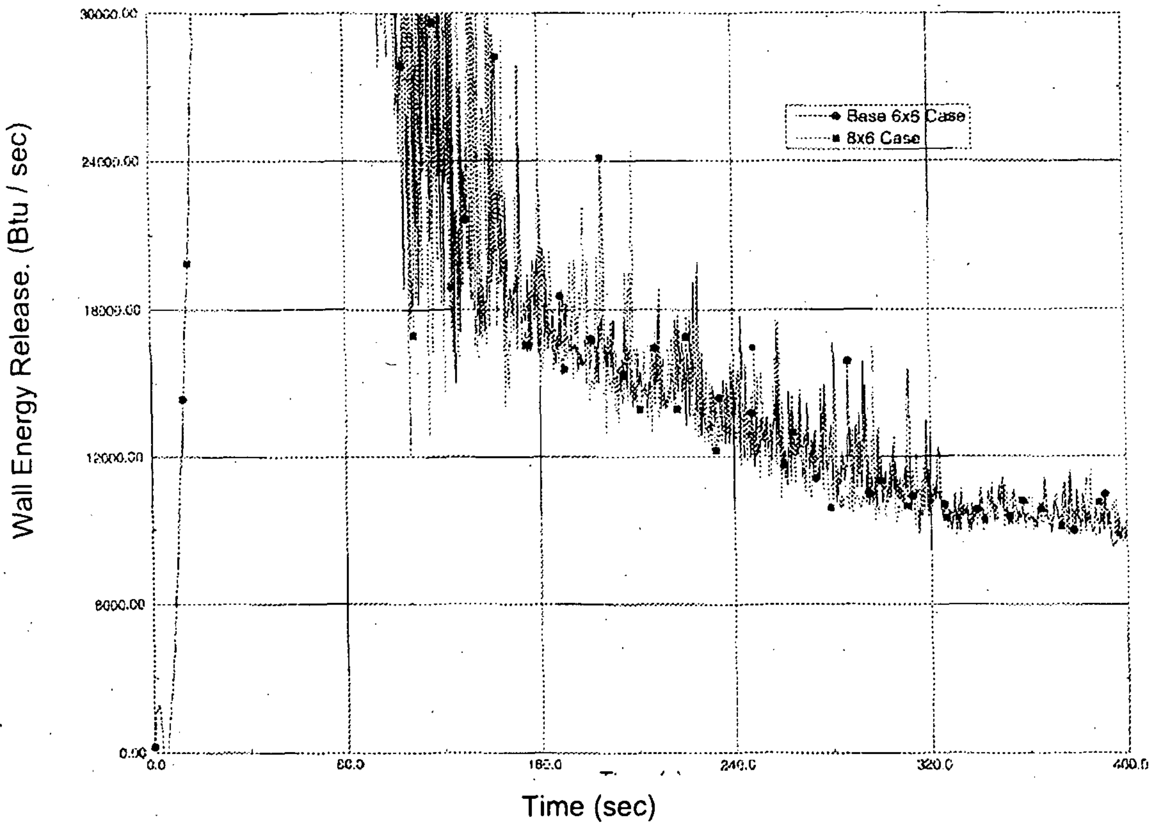


Figure 6-11 Downcomer Wall Heat Release – Axial Noding Sensitivity Study

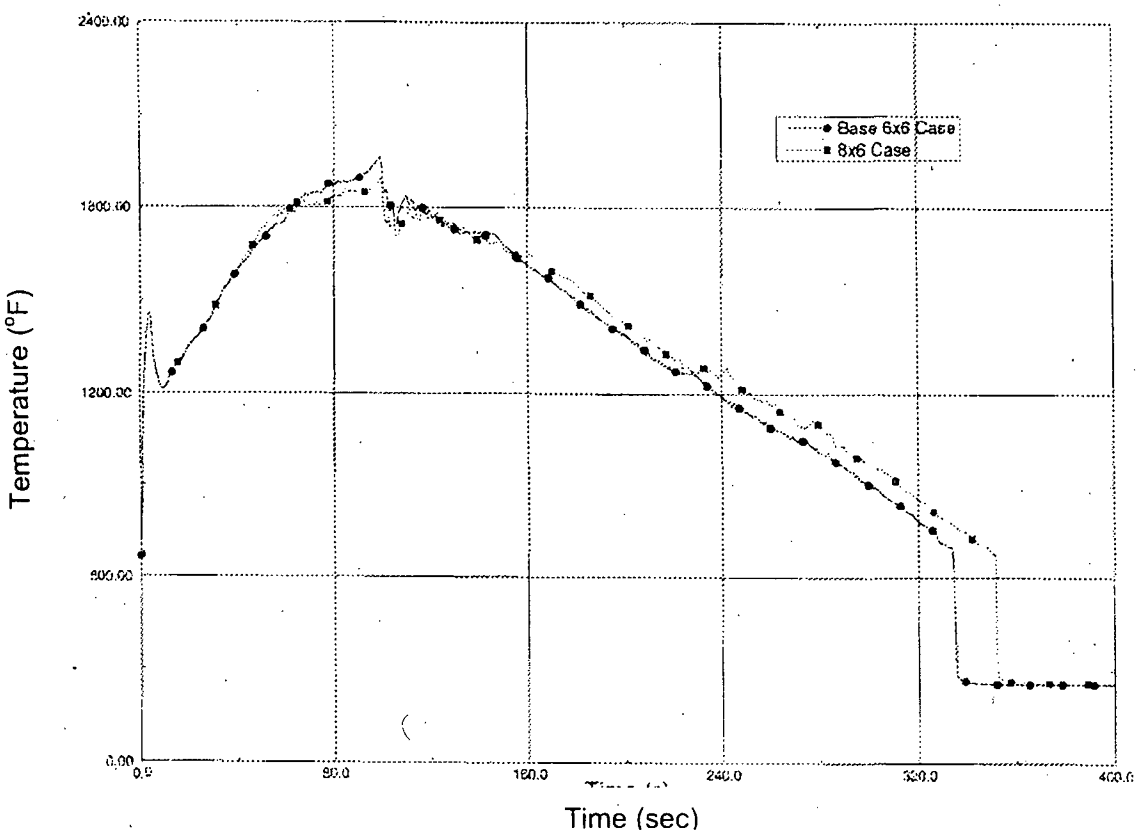


Figure 6-12 PCT Independent of Elevation – Axial Noding Sensitivity Study

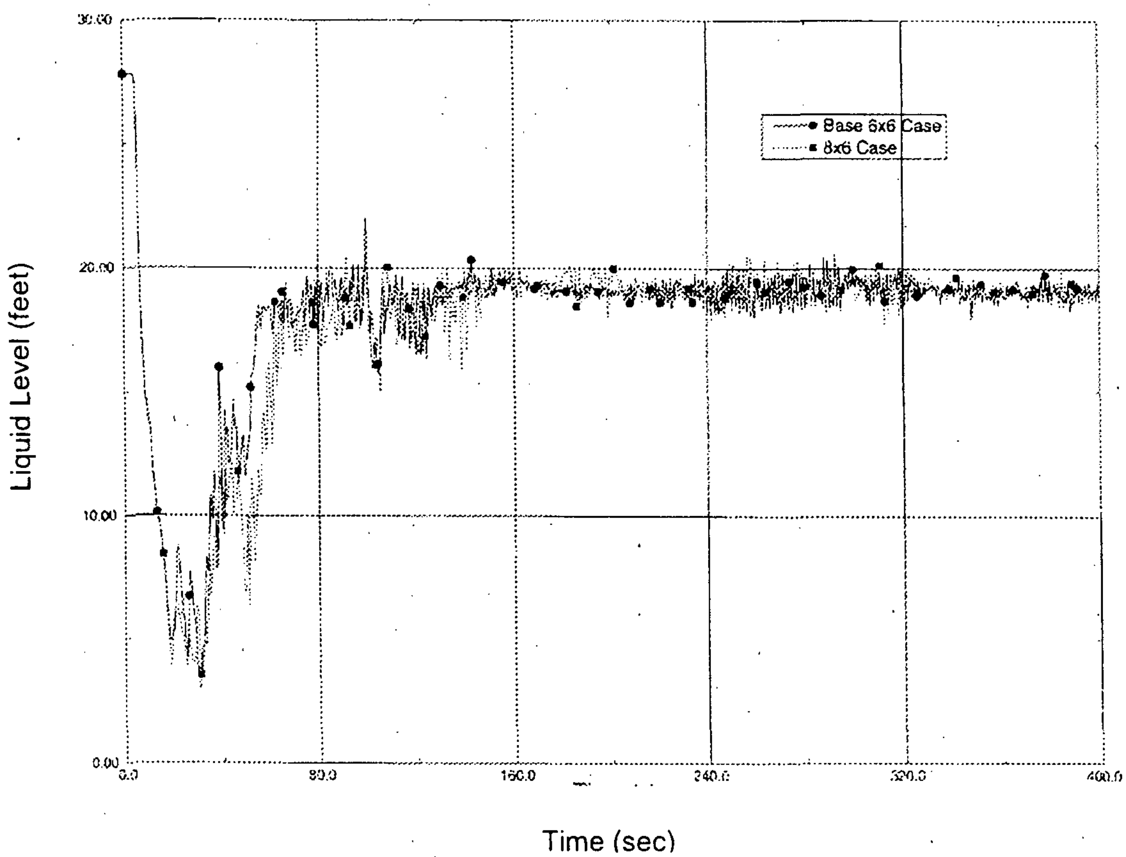


Figure 6-13 Downcomer Liquid Level – Axial Noding Sensitivity Study

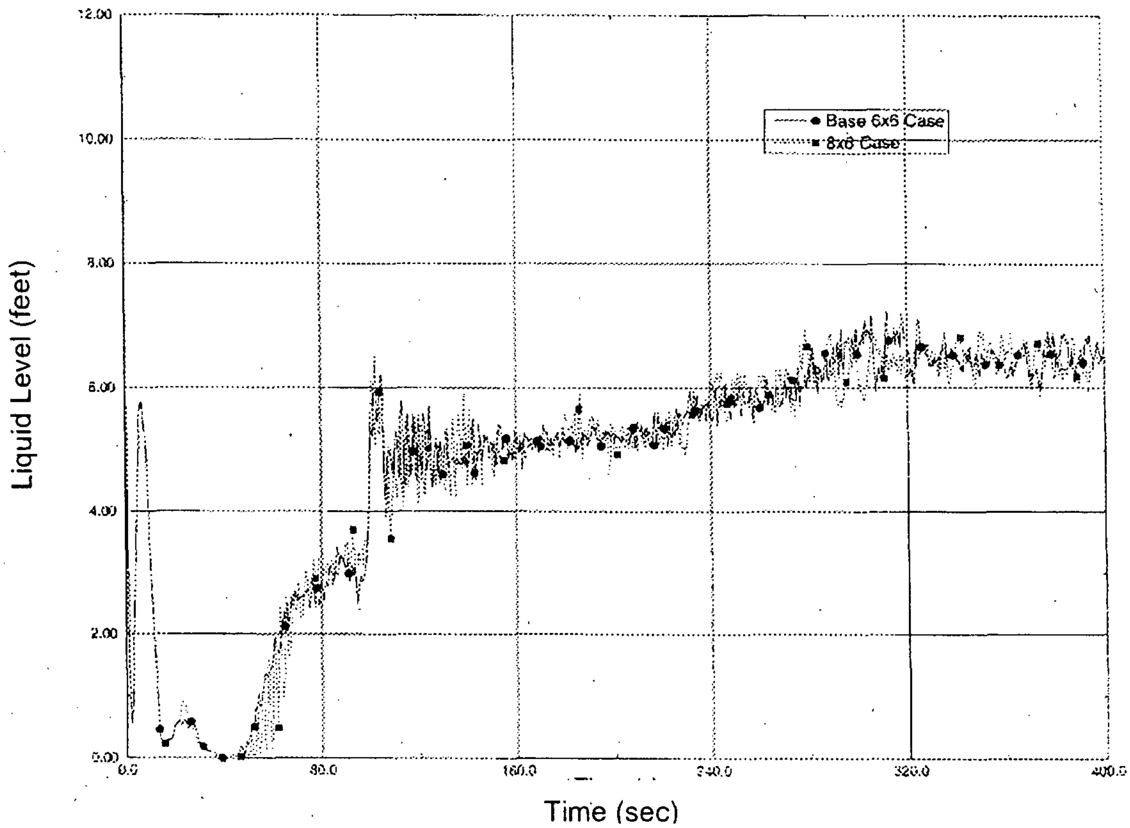


Figure 6-14 Core Liquid Level – Axial Noding Sensitivity Study

6.5.3 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.

6.6 Break Size

Question: *Were all break sizes assumed greater than or equal to 1.0 ft^2?*

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.

6.6.1 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 into 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.

6.6.2 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:

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

This gives

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

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.

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

The estimated minimum LBLOCA break area, A_{min}, is 2.76 ft2 and the break area percentage, based on the full double-ended guillotine break total area, is 33 percent.

Table 6-4 provides a listing of the plant type, initial condition, and the fractional minimum RLBLOCA break area, for all the plant types presented as generic representations in the next section.

Table 6-4 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)	No. of RCPs	RCP flow (lbm/s)	Spectrum Minimum Break Area (ft^2)	Spectrum Minimum Break Area (DEGB)
A	3-Loop W Design	2250	555.0	23190	5700	3	31417	2.18	0.26
B	3-Loop W Design	2250	544.5	23880	5450	4	28124	1.92	0.23
C	3-Loop W Design	2250	550.0	23540	5580	4	29743	2.04	0.25
D	2x4 CE Design	2100	538.8	22860	5310	4	21522	1.53	0.24
E	2x4 CE Design	2055	535.8	22630	5230	3	37049	2.66	0.27
F	4-Loop W Design	2160	540.9	23290	5370	3	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.

6.6.3 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 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 6-4 provides a listing of the plant type, initial condition, and the fractional minimum RLBLOCA break area. Figures 6-15 through 6-20 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 6-5 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 141 °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 6-15). 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, as 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 6-5 Minimum PCT Temperature Difference – True Large and Intermediate Breaks

Plant Description	Generic Plant Label (Table 6-4)	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	
4-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.

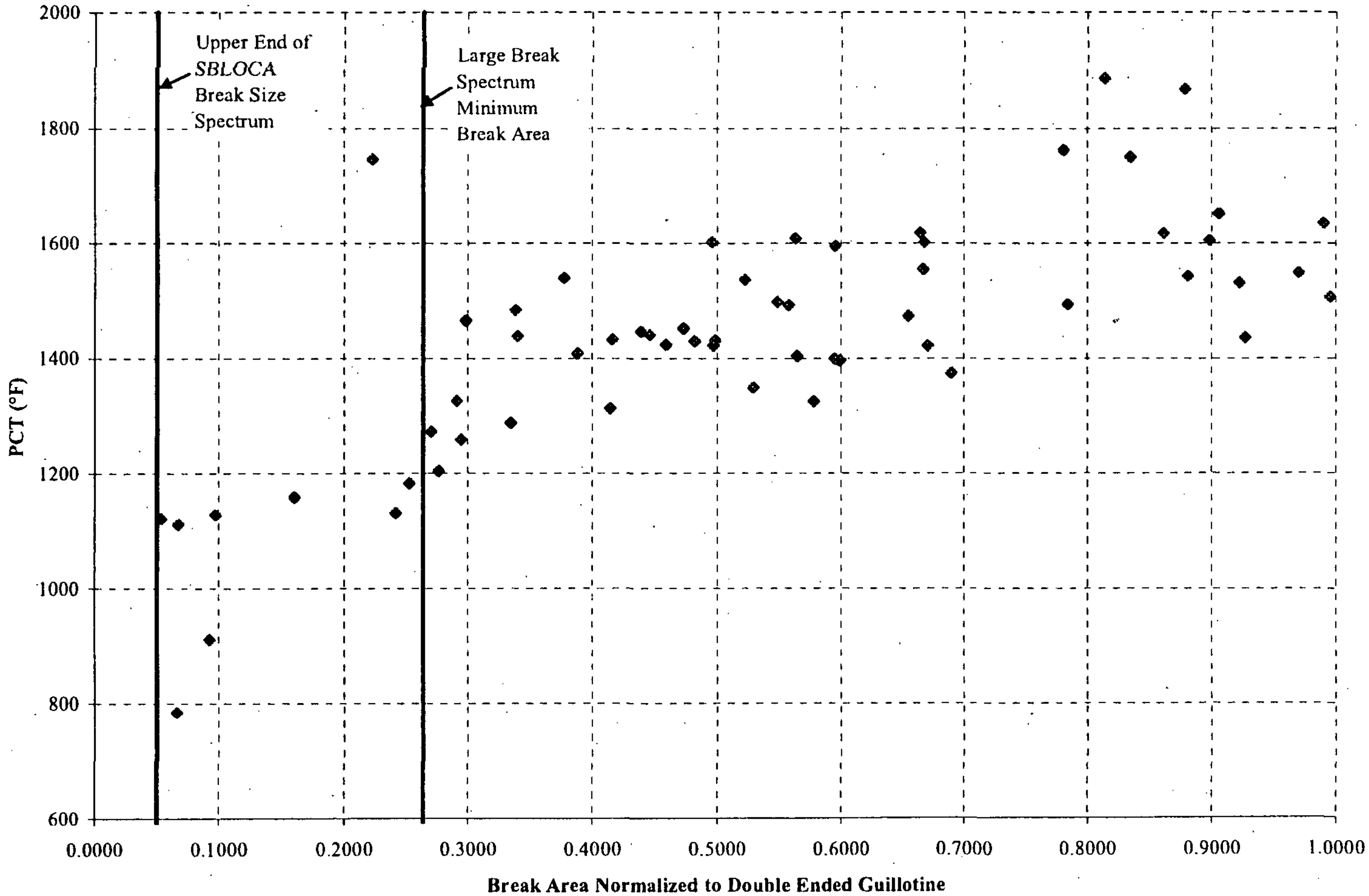


Figure 6-15 Plant A – Westinghouse 3-Loop Design

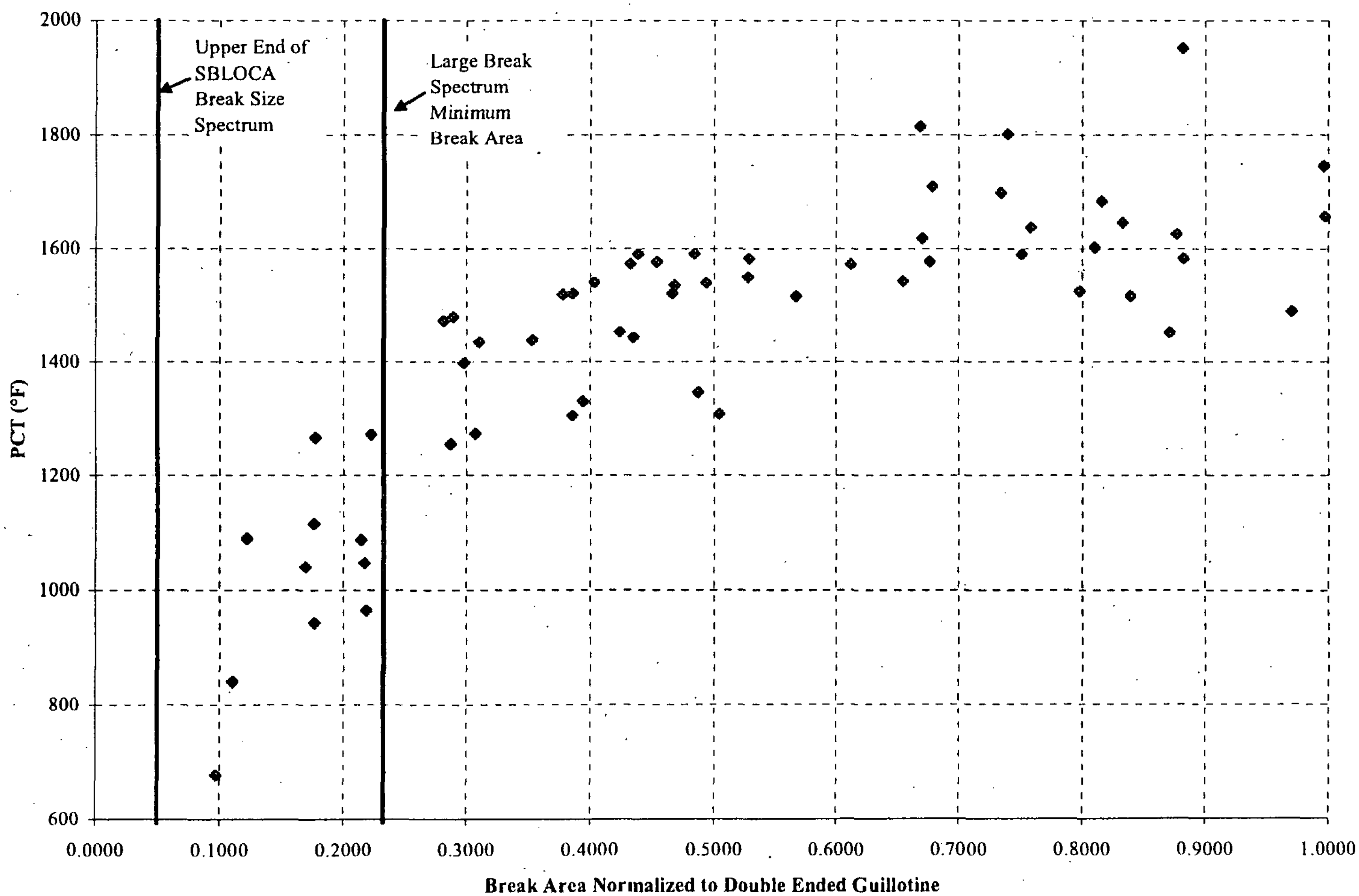


Figure 6-16 Plant B – Westinghouse 3-Loop Design

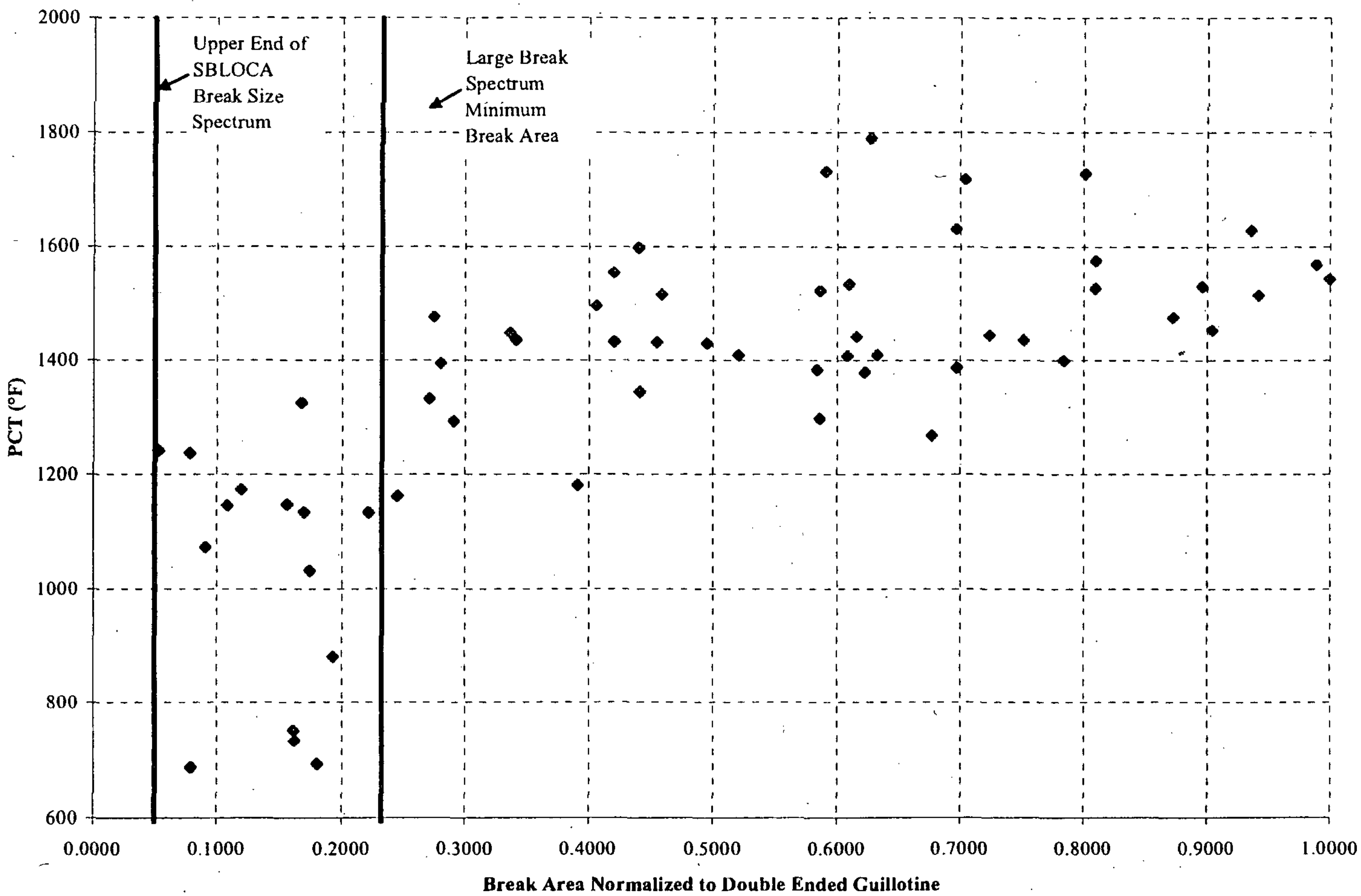


Figure 6-17 Plant C – Westinghouse 3-Loop Design

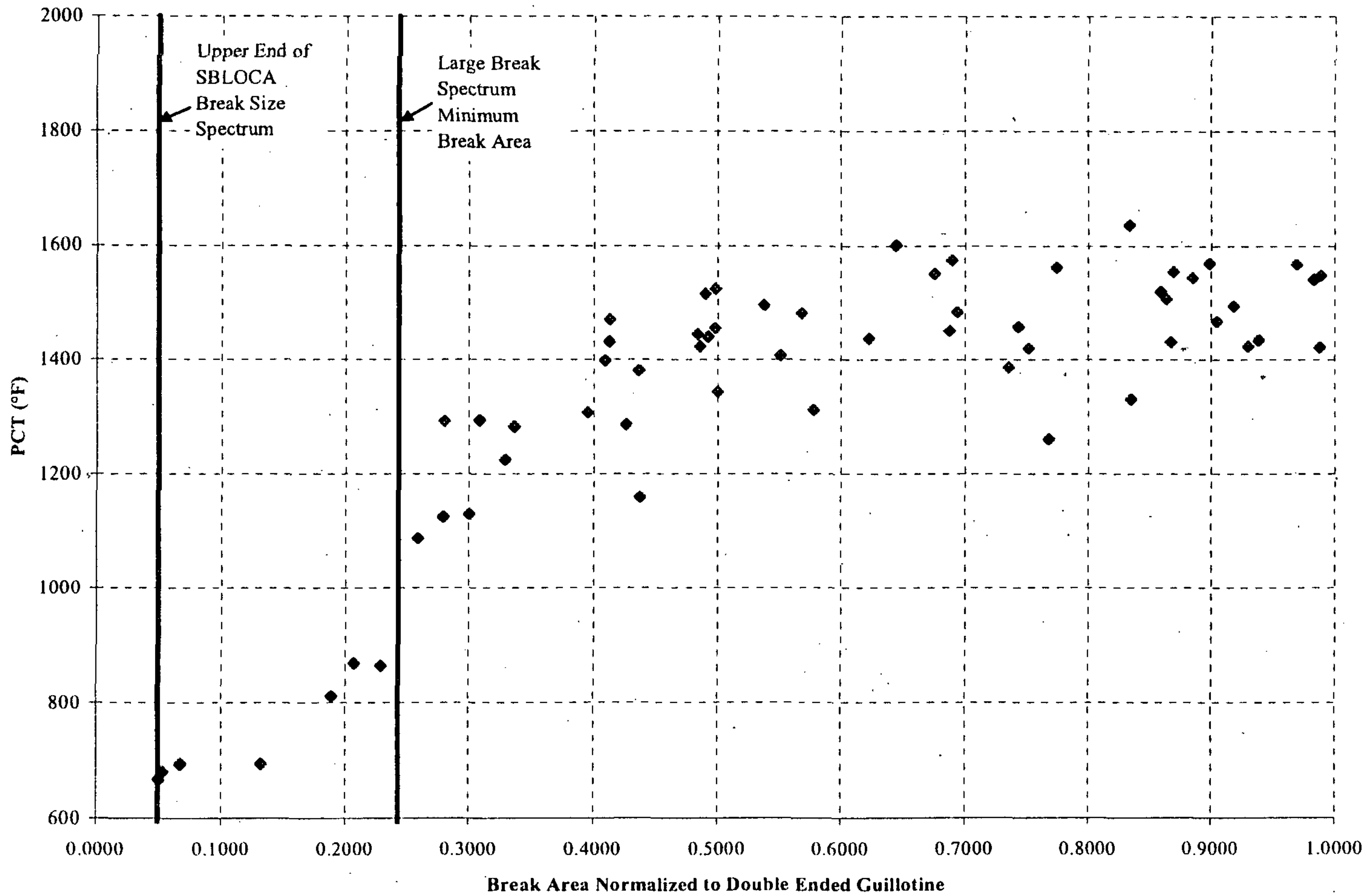


Figure 6-18 Plant D – Combustion Engineering 2x4 Design

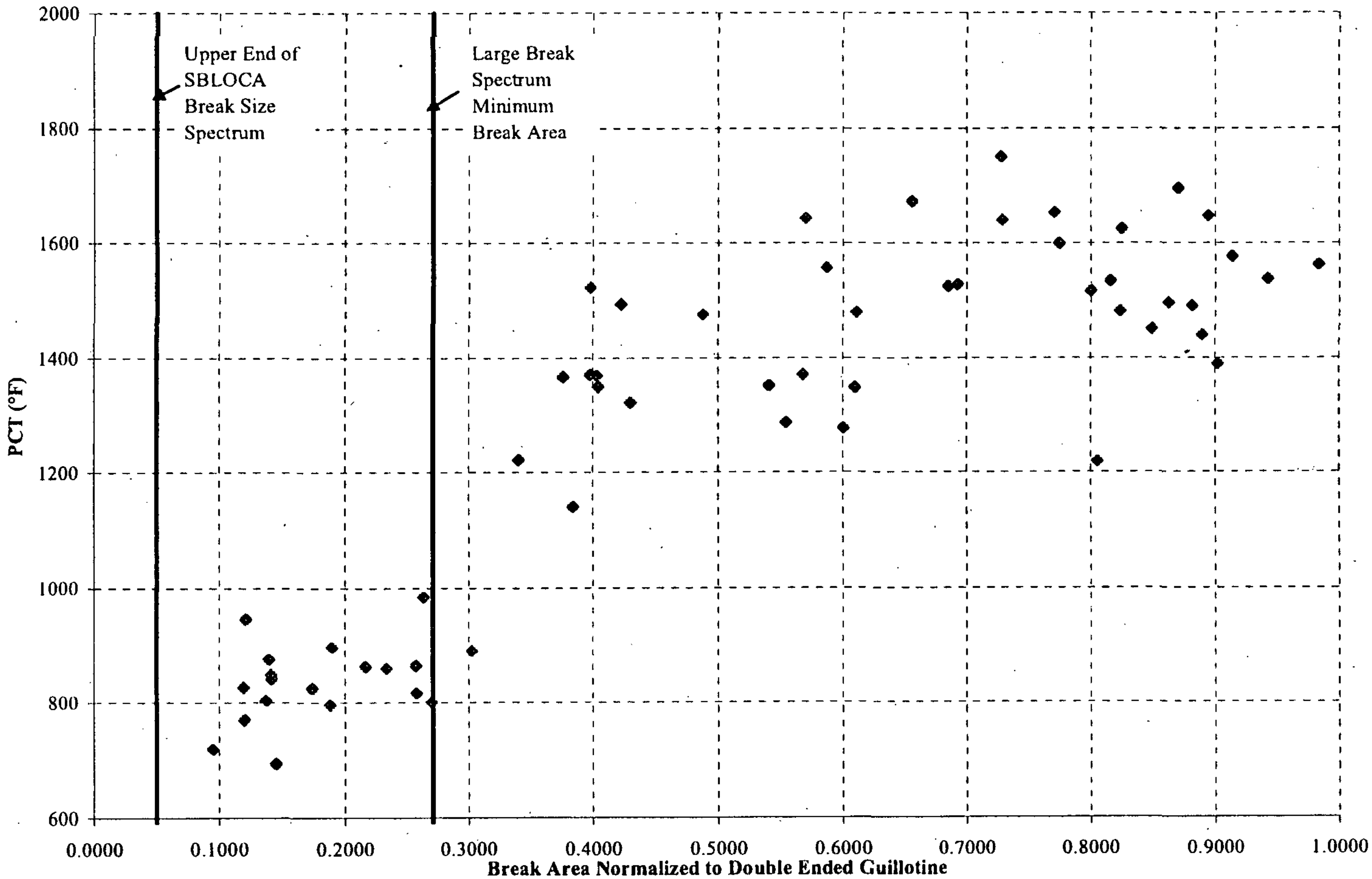


Figure 6-19 Plant E – Combustion Engineering 2x4 Design

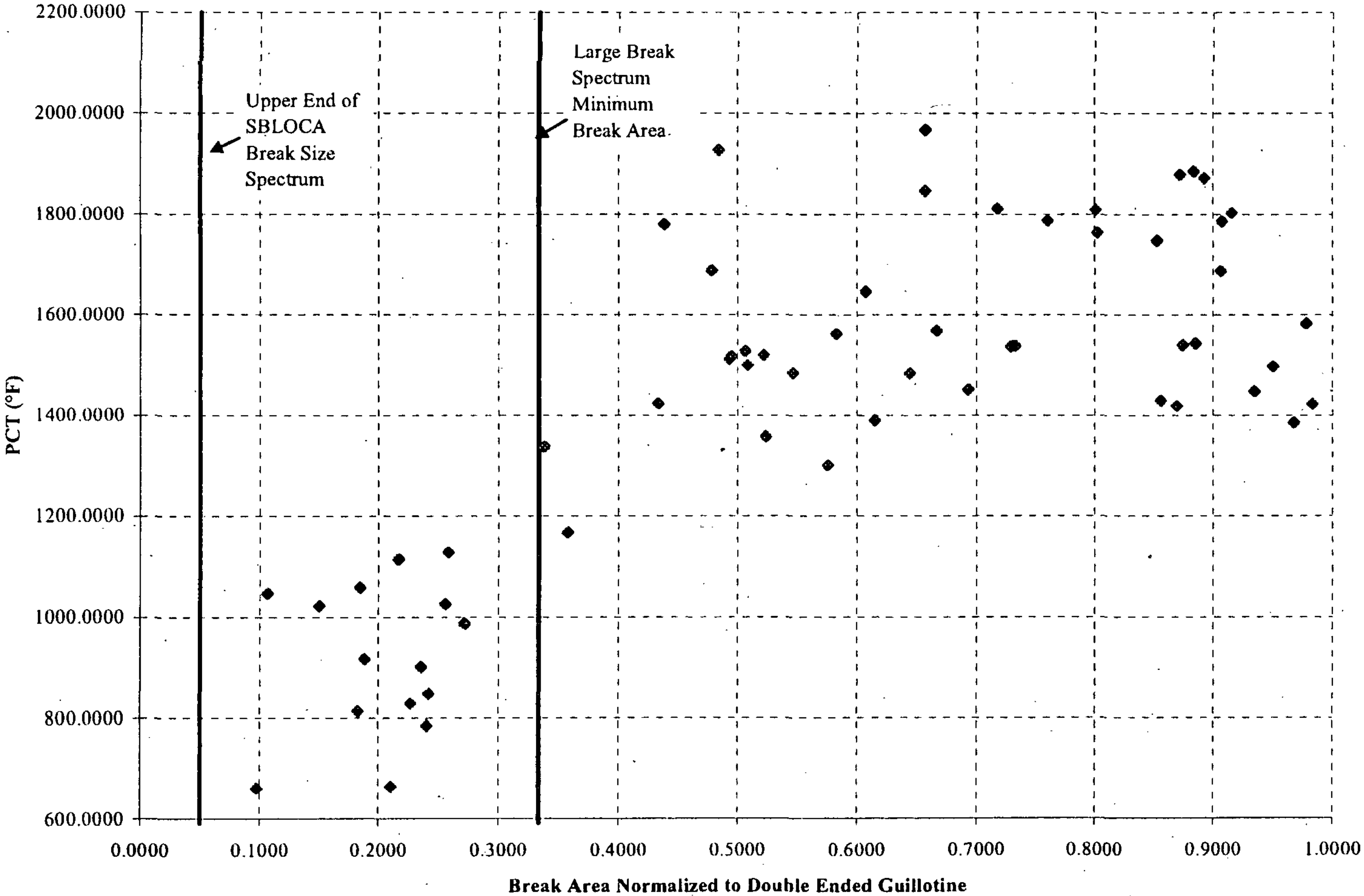


Figure 6-20 Plant F – Westinghouse 4-Loop Design

6.7 ICECON Model

Question: *Verify that the SQN-2 ICECON model is that shown in Figure 5.1 of EMF-CC-39(P) Revision 2, "ICECON: A Computer Program Used to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)."*

See Section 3.3.

6.8 Cross-References to North Anna

Question: *In order to conduct its review of the SQN-2 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 SQN-2 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 for the most part generic and related to the ability of ICECON to calculate containment pressures. Excepting as follows they are applicable to the Sequoyah Unit 2 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 - The direct response is completely applicable excepting that the reference to "North Anna Units 1 and 2" should be deleted. The statement in which the North Anna units are referenced is equally valid without identification of any specific plant.

The supplemental request and response are specific to North Anna and are not applicable to Sequoyah Unit 2.

The response provided to question 1 contains both generic and plant specific content. The portions that are generic remain applicable to Sequoyah Unit 2. However, the North Anna Units use sub-atmospheric containment designs and Sequoyah Unit 2 is of the ice condenser type. This leads to several differences in the way the information would be presented.

6.9 Containment Model

Question: *ANP-2695(P) shows that the containment parameters treated statistically are: (1) upper compartment containment volume, (2) upper compartment containment temperature, and (3) lower compartment containment temperature. ANP-2695(P) states that "in many instances" the guidance of NRC Branch Technical Position CSB 6-1 was used in determining the other containment parameters.*

[AREVA NP: Note that the same Containment System Branch Technical Position is now designated 6-2 instead of 6-1.]

(a) How is the mixing of containment steam and ice melt modeled so as to minimize the containment pressure?

See Section 3.3.

(b) Verify that all containment spray and fan coolers are assumed operating at maximum heat removal capacity.

See Section 3.3.

(c) Describe how the limits on the volume of the upper containment were determined.

See Section 3.3.

(d) How are the containment air return fans modeled and what is the effect of this modeling on the containment pressure?

See Section 3.3.

(e) Describe how passive heat sink areas and heat capacities are modeled so as to minimize containment pressure.

See Section 3.3.

The following are a set of containment plots that are produced to supplement the NRC's review of the Sequoyah Unit 2 RLBLOCA analysis.

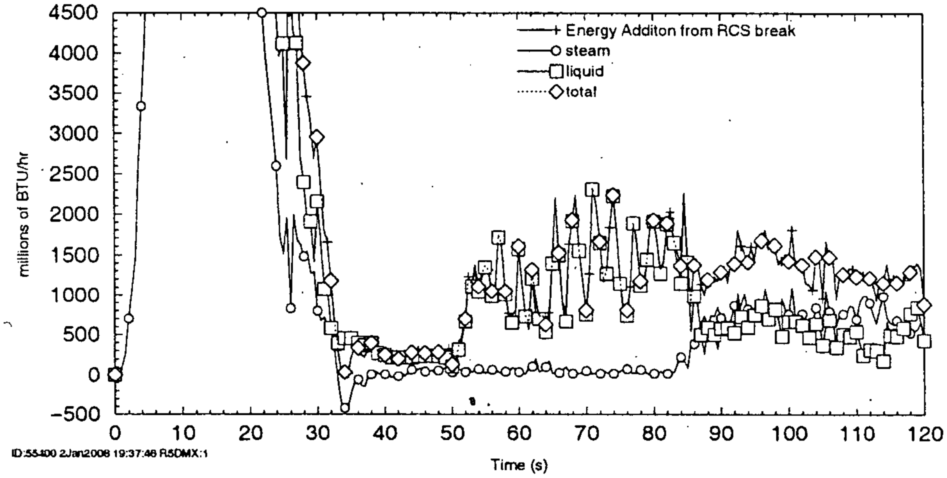


Figure 6-21 Energy Addition in Lower Compartment

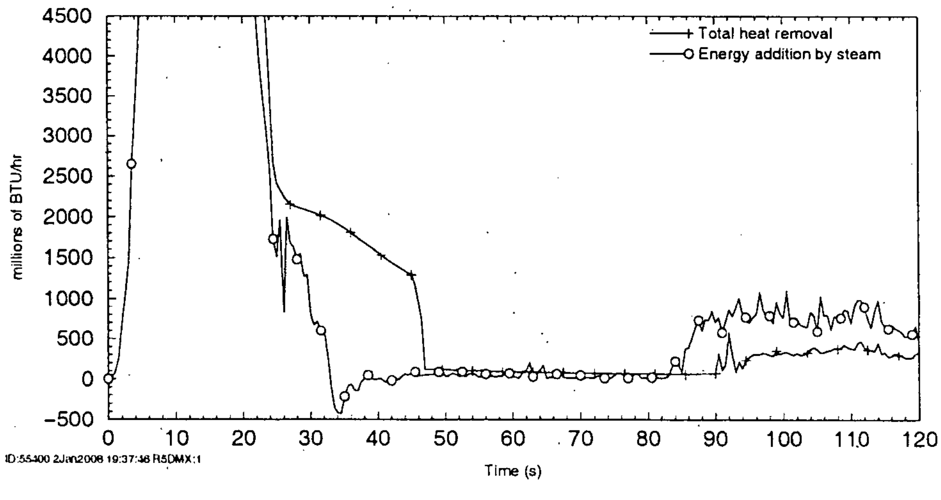


Figure 6-22 Energy Rates in Lower Compartment

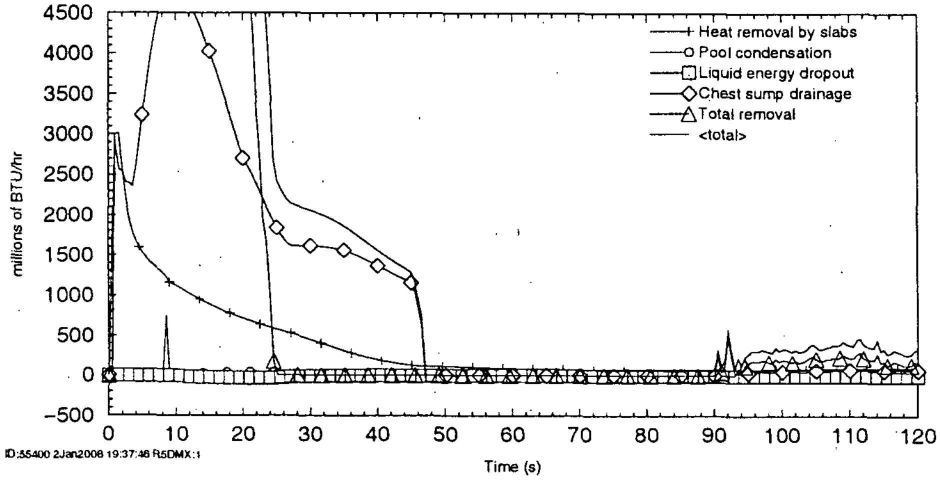


Figure 6-23 Energy Removal Rates in Lower Compartment

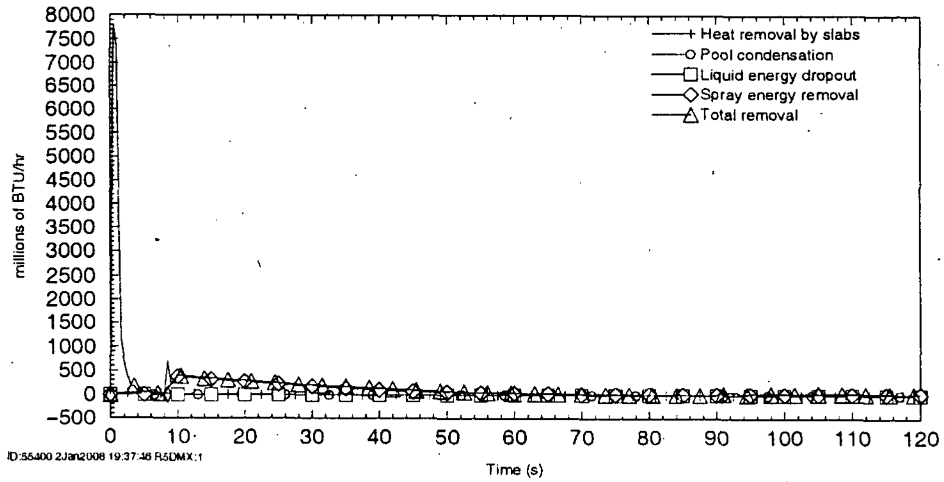


Figure 6-24 Energy Removal Rates in Upper Compartment

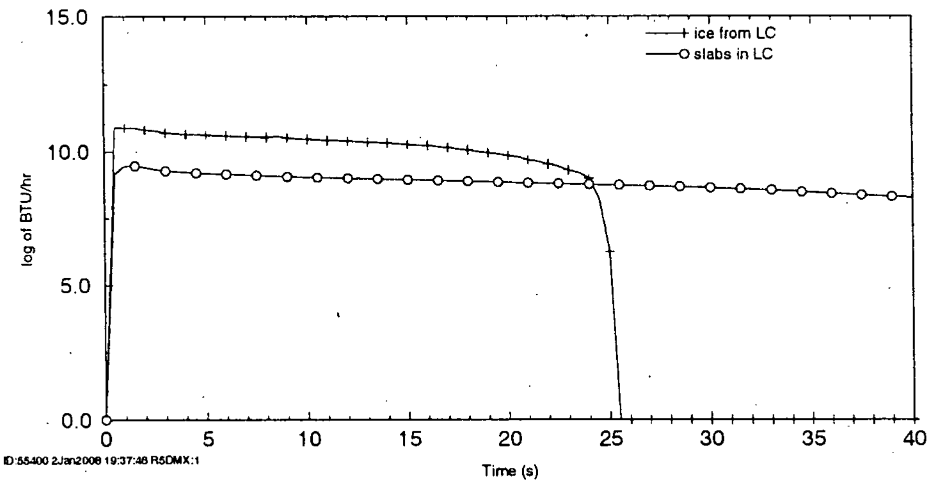


Figure 6-25 Heat Removal Rates (log)

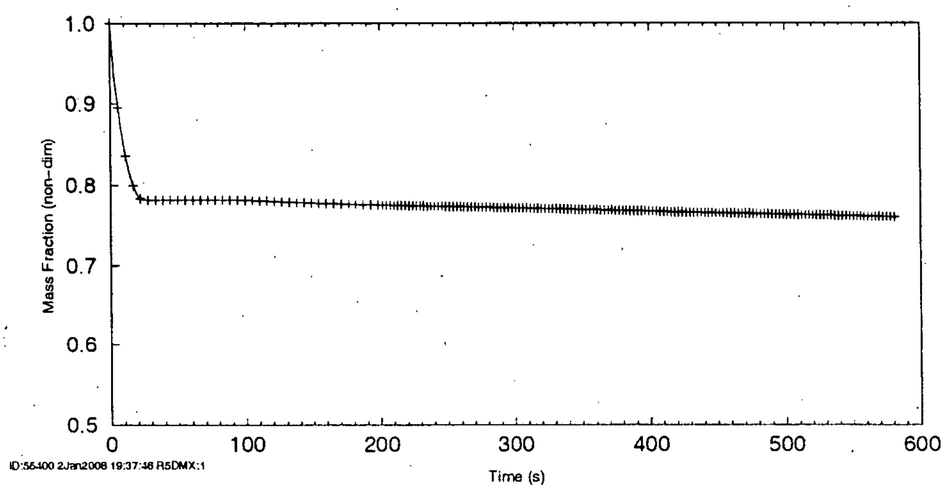


Figure 6-26 Fraction of Ice Remaining

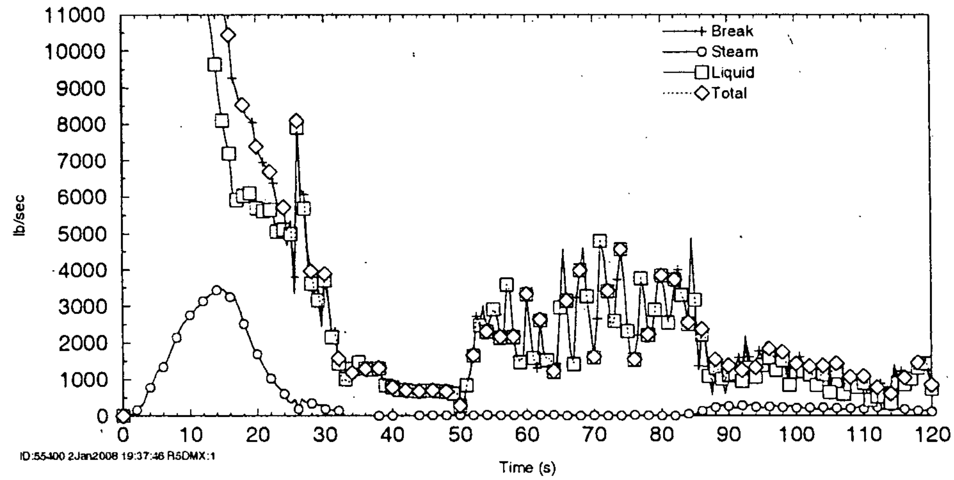


Figure 6-27 Mass Addition to Lower Compartment

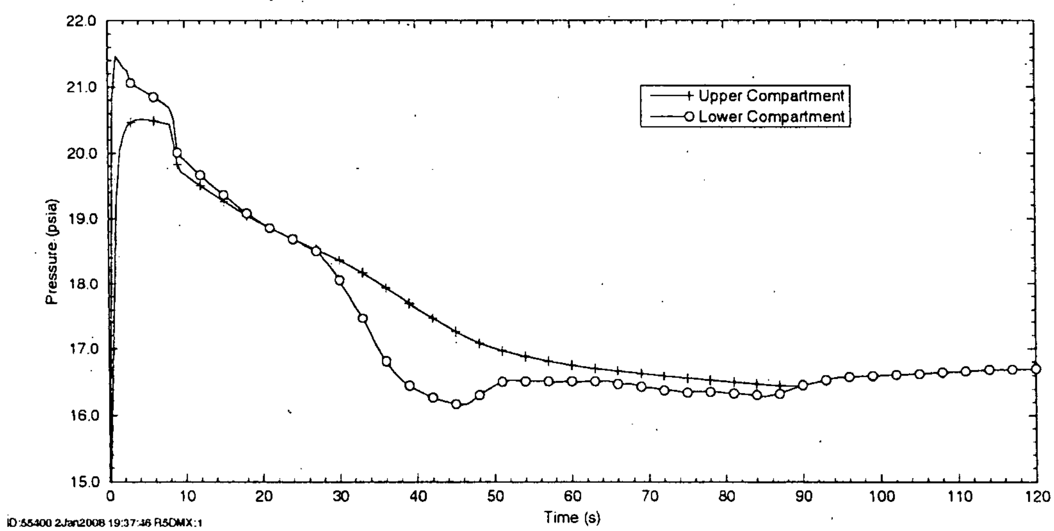


Figure 6-28 Upper Compartment versus Lower Compartment Pressure

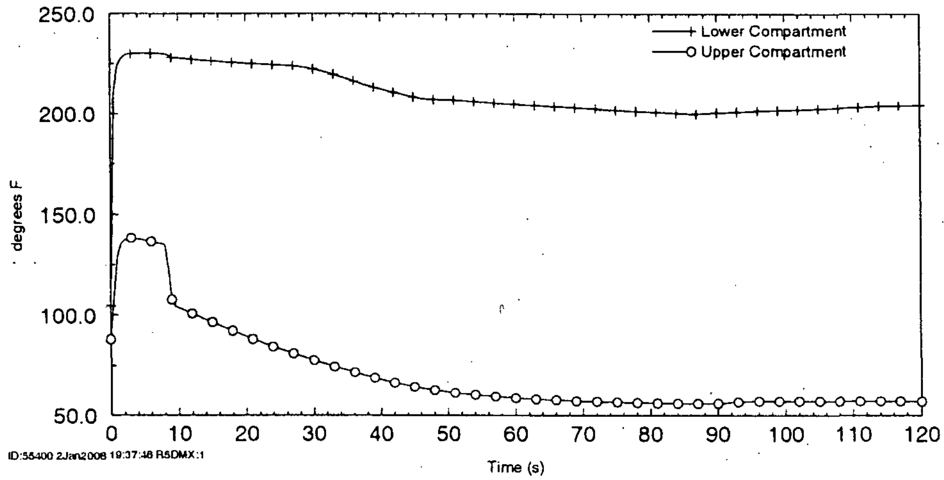


Figure 6-29 Temperature of Upper and Lower Compartments

6.10 GDC 35 – LOOP and No-LOOP Case Sets

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 3-23. This is a change to the approved RLBLOCA EM (Reference 1).

6.11 Statement

Question: *Provide a statement confirming that TVA and its LBLOCA analyses vendor have ongoing processes that assure that the input variables and ranges of parameters for the SQN-2 LBLOCA analyses conservatively bound the values and ranges of those parameters for the as operated SQN-2 plant. This statement addresses certain programmatic requirements of 10 CFR 50.46, Section (c).*

Response: TVA and the LBLOCA Analysis Vendor have an ongoing process to ensure that all input variables and parameter ranges for the Sequoyah Unit 2 realistic large break loss-of-coolant accident are verified as conservative with respect to plant operating and design conditions. In accordance with TVA 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 TVA and Vendor personnel in a formal analysis input summary document issued by the Analysis Vendor and 3) formal review and approval of the input summary document by TVA. Formal TVA approval of the input document serves as the release for the Vendor to perform the analysis.

Continuing review of the input summary document is performed by TVA 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 TVA. Revisions and updates to the analysis parameters are documented and approved in accordance with the process described above for the initial analysis.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

PROPRIETARY INFORMATION WITHHOLDING AFFIDAVIT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA NP information contained in the report ANP-2655(P), Revision 1, "Sequoyah Nuclear Plant Unit 2 Realistic Large Break LOCA Analysis," dated February 2008, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

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

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(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.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP 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 20th

day of February, 2008.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010