ATTACHMENT 10

ANP-2970(NP), Revision 0

Sequoyah Units 1 and 2 HTP Fuel Realistic Large Break LOCA Analysis

April 2011

(NON-PROPRIETARY VERSION)

AREVA NP Inc.

ANP-2970(NP)
Revision 0

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AREVA

Nature of Changes

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

Contents

This document contains a total of 152 pages.

Tables

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Nomenclature

AFD	Axial Flux Difference
ASI	Axial Shape Index
BLEU	Blended Low Enriched Uranium
BWR	Boiling Water Reactor
CCTF	Cylindrical Core Test Facility
CFR	Code of Federal Regulations
CPHS	Containment Pressure High Signal
CSAU	Code Scaling, Applicability, and Uncertainty
DC	Downcomer
DEGB	Double-Ended Guillotine Break
DFSS	Design For Six Sigma
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPH	Effective Full Power Hours
EM	Evaluation Model
FA	Fuel Assembly
FLECHT-SEASET	Full-Length Emergency Core Heat Transfer Separate Effects and Systems Effects Tests
F_Q	Total Peaking Factor
$F_{\Delta H}$	Nuclear Enthalpy Rise Factor
HPSI	High Pressure Safety Injection
HFP	Hot Full Power
LANL	Los Alamos National Laboratory
LEFM	Leading Edge Flow Meter
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LOFT	Loss of Fluid Test
LOOP	Loss of Offsite Power
LPSI	Low Pressure Safety Injection
MSIV	Main Steam Isolation Valve
NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System

Nomenclature (*Continued*)

PCT	Peak Clad Temperature
PIRT	Phenomena Identification and Ranking Table
PLHGR	Planar Linear Heat Generation Rate
PPLS	Pressurizer Pressure Low Signal
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RLBLOCA	Realistic Large Break Loss of Coolant Accident
RV	Reactor Vessel
RWST	Refueling Water Storage Tank
SGLS	Steam Generator Low (pressure) Signal
SIAS	Safety Injection Activation Signal
SI	Safety Injection
SER	Safety Evaluation Report
THTF	Thermal Hydraulic Test Facility
TVA	Tennessee Valley Authority
UHI	Upper Head Injection
w/o	Weight Percent

1.0 Introduction

This report describes and provides results from a RLBLOCA analysis for the Sequoyah Unit 1 and Unit 2[1] Stations. The plants are 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 19 and beyond with AREVA NP's 17x17 HTP fuel design using either BLEU or standard UO_2 fuel and M5 cladding. The analysis was performed in compliance with the U.S. Nuclear Regulatory Commission (NRC) approved RLBLOCA Evaluation Model (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 Units 1 and 2 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 diesel (loss of one train of the pumped ECCS) 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 for fresh and once-burned fuel.

The following are deviations from the approved RLBLOCA EM (Reference 1) that were requested by the NRC are referred to as the "Transition Package." The "Transition Package" is fully described in Section 4.

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.

[1] The analysis models the Unit 1 steam generator, which is identical to the Unit 2 replacement steam generator, scheduled to be installed in the fall of 2012. If the Unit 2 steam generator is not replaced in the fall 2012, which is the same outage for the HTP fuel transition, this RLBLOCA analysis will need to be dispositioned for the impact of the old Unit 2 steam generators.

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 guillotine break (DEGB) 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 4.6.2 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 cases was run with a LOOP assumption and a second set with a No-LOOP assumption. The set of 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 3-24. The effect on PCT results is expected to be minor.

During recent RLBLOCA EM modeling studies, it was noted that cold leg condensation efficiency may be under-predicted. Water entering the downcomer (DC) post-accumulator injection remained sufficiently subcooled to absorb DC wall heat release without significant boiling. However, tests (Reference 14) indicate that the steam and water entering the DC from the cold leg, subsequent to the end of accumulator injection, reach near saturation resulting from the condensation efficiency ranging between 80- to 100-percent. To assure that cold leg condensation would not be under-predicted, a RLBLOCA EM update was made. Noting that saturated fluid entering the DC agrees with industry tests, steam and liquid multipliers were developed so as to approximately saturate the cold leg fluid before it enters the DC. The multipliers were developed through scoping studies using a number of plant configurations—Westinghouse-designed 3- and 4-loop plants, and CE-designed plants. The results of the scoping study indicated that multipliers of 10 and 150 for liquid and steam, respectively, were appropriate to produce saturated fluid entering the DC. This RLBLOCA EM departure was recently discussed with the NRC and the NRC agreed that the approach described immediately above was satisfactory in the interim. The modification is implemented post-accumulator injection, 10 seconds after the vapor void fraction in the bottom of the accumulator becomes greater than 90-percent. Thus, the accumulators have injected all their water into the cold legs, and the nitrogen cover gas has entered the system and been mostly discharged through the break before the condensation efficiency is increased by the factors of 10 and 150, for liquid and vapor respectively. Providing saturated fluid conditions at the DC entrance conservatively reduces both the DC driving head and the core flooding rate. Recall that test results indicate that fluid conditions entering the DC range from saturated to slightly subcooled. Hence, it is conservative to force an approximation of saturated conditions for fluid entering the DC.

The NRC raised the issue concerning fuel thermal conductivity degradation as a function of burnup in Information Notice 2009-23. In order to manage this issue, AREVA Inc. is modifying the way RODEX3A temperatures are compensated in the RLBLOCA Transition Package methodology. In the current process, the RLBLOCA computes PCTs at many different times during an operating cycle. For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady-state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then two-transformation adjustments to the base fuel centerline temperature are computed. The first transformation is a linear adjustment for an exposure of 10 Mwd/MtU or higher. The second adjustment is performed in the S-RELAP5 initialization process for the transient case. In the new process, a polynomial transformation is used for the first transformation instead of a linear transformation. The rest of the RLBLOCA process for initializing the S-RELAP5 fuel rod temperature should not be altered and the rest of LOCA transient should also continue in the original fashion. Section 6 will provide additional information on the adjustment and adding once-burned fuel to the analysis. Note that these changes are also deviations required by the NRC that are departures from the approved evaluation model.

Recent NRC concerns raised in the form of RAIs are responded to in Section 6.

2.0 Summary

The limiting PCT analysis is based on the parameter specification given in Table 2-1 for the limiting case. The limiting PCT is 1941 °F for a fresh UO_2 hot rod (Case 86) with offsite power available (No-LOOP) conditions. From the same case, the PCT for a once-burned UO_2 rod is 1899 °F. Gadolinia-bearing rods of 2, 4, 6 and 8 w/o Gd_2O_3 were analyzed for fresh and once-burned fuel[2] in all cases. The limiting PCT for all once-burned rods in Case 86 was in a 6 w/o Gd_2O_3 rod (1917 °F). This RLBLOCA result is based on a case set of 93[3] individual transient cases for offsite power not available (LOOP) and 93 individual transient cases for offsite power available (No-LOOP) conditions. The core is composed only of AREVA NP 17x17 fuel and was modeled in a mixed-core configuration due to thermal-hydraulic differences in the Mark BW and HTP fuel designs.

The analysis assumes full core power operation at 3479 MWt (including uncertainties), a steam generator tube plugging level of 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 ($F_{\Delta H}$) of 1.7056 (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 Transition Package methodology (on a forward fit basis) explicitly analyzes fresh and once-burned fuel assemblies to respond to recent NRC RAIs. The second-burned fuel assemblies have minimal power and are typically located on the periphery; therefore would not be limiting in regards to PCT for a RLBLOCA analysis. The AREVA core management design process ensures that reinsert fuel does not have the limiting $F_{\Delta H}$. The analysis demonstrates that the 10 CFR 50.46(b) criteria listed in Section 3.0 are satisfied.

[2] The once-burned GAD rods were not reduced by the once-burned UO_2 peaking reduction (0.9168) shown in Figure 6-4 at the sampled time-in-life, thus producing higher PCT results.

[3] AREVA decided to run 93 cases (reporting the 92nd case) for the Sequoyah analysis for three main reasons: 1) current AOR Sequoyah RLBLOCA results, 2) high peaking values analyzed, and 3) the greater pressure drop in the 17x17 HTP fuel w/ integral flow mixers. The allowance to execute more cases is discussed in EMF-2103(P)(A) Section 5.2.1.

Table 2-1 Summary of Major Parameters for Limiting Transient

	Fresh UO_2 Fuel	Once-Burned UO_2 Fuel
Core Average Burnup (EFPH)	5400	5400[4]
Core Power (MWt)	3479	3479
Total Peaking (F_Q)	2.634	2.634
Radial Peak ($F_{\Delta H}$)	1.7056	1.5637[5]
Axial Offset	0.3182	0.3182
Break Type	Split	Split
Break Size (ft^2/side)	3.8673	3.8673
Offsite Power Availability	Available	Available
Decay Heat Multiplier	1.0	1.0

[4] The first cycle ends at 13200 EFPH and this time is from the start of 2nd cycle; the assembly total burnup is 18600 EFPH.

[5] Calculated using Fresh Fuel $F_{\Delta H}$ x K(Burnup) multiplier (0.9168 at 18600 EFPH) from Figure 6-4.

3.0 Analysis

The purpose of the analysis is to verify the adequacy of the ECCS and to demonstrate compliance to the 10CFR 50.46(b) criteria.

- 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 analyze a new fuel design for Sequoyah, the coolable geometry calculation (LOCA-seismic loads) will be revised and testing will validate the 4th criterion is met. The long-term cooling licensing bases would remain applicable. 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. Section 4 discusses the additional information provided under the "Transition Package" on EMF-2103. Section 5 provides the conclusions, Section 6 addresses recent NRC RAIs on RLBLOCA submittals and Section 7 contains the reference list.

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 diesel (one ECCS pumped injection train) with fully functional containment sprays. 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 cover 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 operational characteristics 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 3-1. This list includes both parameters related to LOCA phenomena (based on the PIRT provided in Reference 1) and to plant operating parameters.

3. Determination of Adequacy of ECCS

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

3.3 Plant Description and Summary of Analysis Parameters

The plant analysis presented in this report is for a Westinghouse-designed PWR, which has four loops, each with a hot leg, a U-tube steam generator, and a cold leg with a RCP[6]. The RCS also includes one pressurizer connected to a hot leg. The core contains (193) 17x17 AREVA 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.

[6] 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 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 Units 1 and 2 Stations. 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 Units 1 and 2 Stations. Both entities have ongoing processes that assure the ranges and values of input parameters for the Sequoyah Units 1 and 2 Stations 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 3-1. The LBLOCA phenomenological uncertainties are provided in Reference 1. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 3-2. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Table 3-3 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 6-2 (Reference 5) was used in setting the remainder of the containment model input parameters. As noted in Table 3-3, containment temperature is a sampled parameter. Containment pressure response is indirectly ranged by sampling the upper containment volume (Table 3-3). 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.

The containment initial conditions and boundary conditions are given in Table 3-8. 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. 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 actual 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 3-9. 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 3-9 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 Units (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 3-4.

Blowdown Quench (SER Restriction #7)

Item: 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 run corrected. A blowdown quench is characterized by a temperature reduction of the PCT node to saturation temperature during the blowdown period.

Treatment: Examine plots of PCT vs. Time at the PCT node for all cases for evidence of blowdown quench using the RLBLOCA definition of blowdown (i.e., accumulator discharge).

The highest clad surface temperatures independent of elevation and also at the PCT node are calculated as a function of time for all cases. These plots were examined and compared to the end of blowdown (EOB) to verify that no quench of the PCT node occurred prior to EOB. As a result of this inspection, several cases exhibit lingering cladding temperatures near saturation temperature during blowdown and were flagged for further evaluation. Six cases were "screened" for further evaluation regarding possible quench prior to EOB. The hot pin, PCT axial heat structure node, and the corresponding core fluid node for each case are identified. These parameters form the basis for illustrating the clad temperature, fluid (saturation) temperature, and accumulator injection transient at the PCT location for the screened cases.

Figure 3-25 through Figure 3-30 indicates that, of the six screened cases, evidence of blowdown quench is apparent in three of the cases, Cases 32, 35, and 68 (Note that these cases represent three of the lowest 4 cases sorted in order of decreasing PCT). Clad temperature superheat equivalents achieved for these cases range from about 200 to 450 °F, much lower than the approximately 900 °F blowdown superheat resulting from the limiting case, Case 86. As a result, the hot pins are more easily quenched by a minimal intrusion of liquid into the hot channel.

In comparison to the limiting case, the linear heat generation rates (LHGRs) and break areas are quite small and this explains why the blowdown clad temperature response is minimal. Given the combined low hot pin power and the small break area, with its comparatively low blowdown rate and slower transient, Case 32, 35, and 68 could not achieve PCTs that are limiting or even high in the order of cases ranked according to PCT.

Effectively, there is no blowdown quench and the cases of interest regarding blowdown quench have no impact. Therefore, compliance to the SER restriction has been demonstrated.

Bottom-Up Quench

Item: 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 down quench. A top-down quench is characterized by the quench front moving from the top to the bottom of the hot assembly.

Treatment: The reflood heat transfer model in S-RELAP5 and the prescribed upper plenum nodalization have been developed so that the peak clad temperature location quenches from the consequences of a bottom-up quench only. No additions to the calculation notebook or Design Report are required.

Several provisions have been implemented in the S-RELAP5 model to prevent top-down quench. These measures are:

- The CCFL model is applied on all core exit junctions.
- The reverse form loss at the hot channel and central core exit is increased by a factor of 1000 at the beginning of core reflood.

To confirm that top-down quench does not occur in the analysis, the liquid flow at the exit of the hot channel was plotted. Junction 065 is the hot channel exit junction designation for the exit. Because the liquid flow rate is not particularly steady, it was integrated to smooth the result.

Cases can exhibit periodic positive integrated flow (indicating flow in an upward direction), constant integrated flow (no flow), or negative integrated flow (flow moving from the top-down). Core flows can reverse at break initiation, during blowdown, and reverse flow can continue until the core begins to be recovered by accumulator and ECCS injection. Top-down quench, then, is judged as a potential problem if the hot channel exit flow reverses after the beginning of core recovery.

Using the beginning of core recovery (BOCR) for each case analyzed and examining the integrated hot channel exit flows, it is apparent that, for the most part, core exit post-BOCR liquid flows are positive, ruling out the potential for top-down quench. There are some cases, however, that exhibit temporary flow reversals. These cases are screened out and after close examination of all of the cases; it is shown that core exit reverse flow is not significant after BOCR and any top-down quench effects are not sufficient to have an effect on PCT predictions resulting from the analysis. Therefore, compliance to the SER restriction has been demonstrated.

3.5 Realistic Large Break LOCA Results

Two case sets of 93 transient calculations were performed sampling the parameters listed in Table 3-1 and the results of the 92nd case are reported. For each case set, PCT was calculated for a UO_2 rod and for Gadolinia-bearing rods with concentrations of 2, 4, 6 and 8 w/o Gd_2O_3. The limiting case set, that contained the PCT, was the set with no offsite power available. The limiting PCT (1941 °F) occurred in Case 86 for a fresh UO_2 rod. The major parameters for the limiting transient are presented in Table 2-1. The once-burned rod limiting PCT (1917 °F) occurred in Case 86 for a 6 w/o GAD rod[7]. Table 3-5 lists the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1-percent limit. The best-estimate PCT case is Case 72, which corresponded to the median case out of the 93-case set with no offsite power available. The nominal PCT was 1484 °F. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 457 °F.

The case results, event times and analysis plots for the limiting PCT case are shown in Table 3-5, Table 3-6, and in Figures 3-12 through 3-23. Figure 3-7 shows linear scatter plots of the key parameters sampled for the 93 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 93 calculations, respectively. Figures 3-10 and 3-11 show the maximum oxidation and total oxidation versus PCT scatter plots for the 93 calculations, respectively. Key parameters for the limiting PCT case are shown in Figures 3-12 through 3-23. 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 3-24.

[7] The once-burned GAD rods were not reduced by the once-burned UO_2 peaking reduction (0.9168) shown in Figure 6-4 at the sampled time-in-life (total = 18600 EFPH), thus producing higher PCT results. The burned UO_2 rod PCT for Case 86 was 1899 °F.

Table 3-1 Sampled LBLOCA Parameters

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties, burnup)
	Break type (guillotine versus split)
	Critical flow discharge coefficients (break)
	Decay heat[8]
	Critical flow discharge coefficients (surgeline)
	Initial upper head temperature
	Film boiling heat transfer
	Dispersed film boiling heat transfer
	Critical heat flux
	T_{min} (intersection of film and transition boiling)
	Initial stored energy
	Downcomer hot wall effects
	Steam generator interfacial drag
	Condensation interphase heat transfer
	Metal-water reaction
Plant[9]	
	Offsite power availability[10]
	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)

[8] Not sampled in analysis, multiplier set to 1.0.

[9] Uncertainties for plant parameters are based on typical plant-specific data.

[10] Not sampled, see Section 4.9.

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

	Event	Operating Range
1.0	**Plant Physical Description**	
	1.1 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	$\leq$ 15-percent
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal reactor power	3479 MWt[11]
	b) F_Q	≤ 2.65[12]
	c) $F_{\Delta H}$	≤ 1.7056[13]
	d) MTC	≤ 0 at HFP
	2.2 Fluid Conditions	
	a) Loop flow	131.6 Mlbm/hr $\leq$ M $\leq$ 152.8 Mlbm/hr
	b) RCS average temperature	578.2 °F $\leq$ T $\leq$ 583 °F
	c) Upper head temperature	~Tcold Temperature[14]

11 Includes uncertainties

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

13 Includes 4-percent measurement uncertainty

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

Table 3-2 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) 37 s (w/o offsite power)
	j) SI pump delay	37 s (w/ offsite power) 37 s (w/o offsite power)
	k) RHR pump delay	37 s (w/ offsite power) 37 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 3-3 Statistical Distributions Used for Process Parameters[15]

Parameter	Operational Uncertainty Distribution	Parameter Range	Measurement Uncertainty Distribution[16]	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[17]	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) 37 (w/o offsite power)	N/A	N/A
LHSI Pump Delay (s)	Point	37 (w/ offsite power) 37 (w/o offsite power)	N/A	N/A
RHR Pump Delay (s)	Point	37 (w/ offsite power) 37 (w/o offsite power)	N/A	N/A

15 Note that core power is not sampled, see Section 1.0

16 All measurement uncertainties were incorporated into the operational ranges

17 This is no longer a sampled parameter. One set of 93 cases is run with LOOP and one set of 93 cases is run with No-LOOP.

Table 3-4 SER Conditions and Limitations

SER Conditions and Limitations	Response
1. A CCFL violation warning will be added to alert the analyst to CCFL violation in the downcomer should such occur.	There was no significant occurrence of CCFL violation in the downcomer for this analysis. Violations of CCFL were noted in a statistically insignificant number of time steps.
2. AREVA NP has agreed that it is not to use 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 Units 1 and 2 (15.08 kW/ft) are 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[18].
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 and in response to RAI #25 for EMF-2103(P)(A) Rev. 0.
5. The model applies to 3 and 4 loop Westinghouse- and CE-designed nuclear steam systems.	Sequoyah Units 1 and 2 are Westinghouse 4-loop plants.
6. The model applies to bottom reflood plants only (cold side injection into the cold legs at the reactor coolant discharge piping).	Sequoyah Units 1 and 2 are bottom reflood plants.
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 six cases; these cases 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.

[18] In response to recent NRC RAI questions, a blowdown rupture check was performed and the results reported in Section 6. There were no cases that exhibited blowdown rupture.

Table 3-4 SER Conditions and Limitations *(Continued)*

SER Conditions and Limitations	Response
9. The model does not determine whether Criterion 5 of 10 CFR 50.46, long term cooling, has been satisfied. This will be determined by each applicant or licensee as part of its application of this methodology.	Long-term cooling was not evaluated in this analysis.
10. Specific guidelines must be used to develop the plant-specific nodalization. Deviations from the reference plant must be addressed.	The nodalization in the plant model is consistent with the Westinghouse 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 are slightly differently than the sample problem; 4) The hydraulic model of the core employs 22 axial nodes instead of 23; 5) There are no standpipes present in the Sequoyah Units 1 and 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 Units 1 and 2 are 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 3-7 presents the summary of the full range of applicability for the important heat transfer correlations, as well as the ranges calculated in the limiting case of this analysis. Calculated values for other parameters of interest are also provided. As is evident, the plant-specific parameters fall within the methodology's range of applicability.

Table 3-4 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 Units 1 and 2 plants have previously been operating with M5 clad fuel and thus this restriction has been satisfied.

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

Case #86 (offsite power available)	Fresh Fuel UO_2 Rod	Once-Burned Fuel 6 w/o GAD[19]
PCT		
Temperature	1941 °F	1917 °F
Time	265.9 s	265.9 s
Elevation	10.043 ft	10.043 ft
Metal-Water Reaction		
Pre-transient Oxidation (%)	0.7624	1.3144
Transient Oxidation Maximum (%)	2.9906	2.5376
Total Oxidation Maximum (%)	3.7530	3.8520
Total Whole-Core Oxidation (%)	0.0982	N/A

[19] The once-burned GAD rods were not reduced by the once-burned UO_2 peaking reduction (0.9168) shown in Figure 6-4 at the sampled time-in-life (total = 18600 EFPH), thus producing higher PCT results. The burned UO_2 rod PCT for Case 86 was 1899 °F.

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

Event	Time (s)
Break Opened	0.0
RCP Trip	N/A
SIAS Issued	0.0
Start of Broken Loop Accumulator Injection	9.5
Start of Intact Loop Accumulator Injection (Loops 2, 3 and 4 respectively)	12.1, 12.1, 12.1
Start of Charging	37.0
SI/RHR Available	37.0
Broken Loop SI/RHR Delivery Began	37.0
Intact Loop SI/RHR Delivery Began (Loops 2, 3 and 4 respectively)	37.0, 37.0, 37.0
Beginning of Core Recovery (Beginning of Reflood)	48.2
PCT Occurred	265.9
Broken Loop Accumulator Emptied	82.6
Intact Loop Accumulators Emptied (Loops 2, 3 and 4 respectively)	83.5, 83.8, 83.3
Transient Calculation Terminated	839.7

Table 3-7 Heat Transfer Parameters for the Limiting Case[20]

Table 3-8 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 x 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[24]	
Post-LOCA initiation at 600 s	
Total Flow = 120,000 cfm	

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

Table 3-9 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-ºF)	Volumetric Heat Capacity (BTU/ft3-ºF)
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

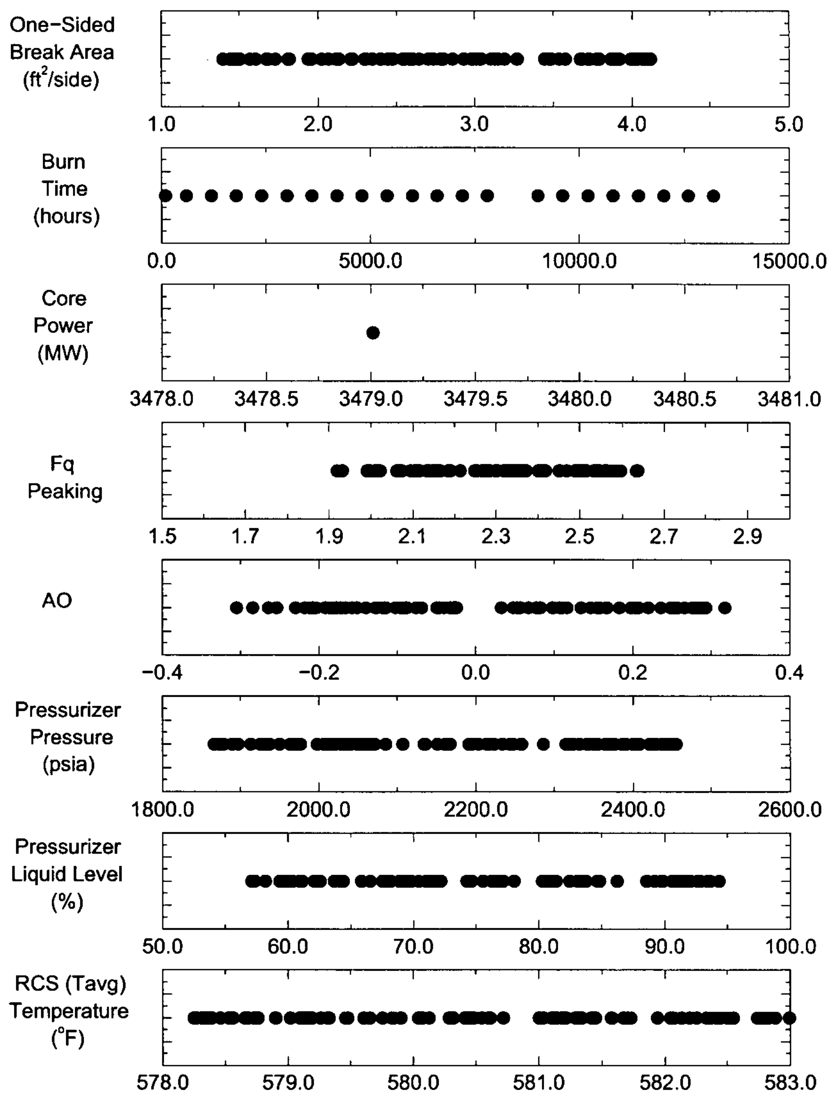

Figure 3-7 Scatter Plot of Operational Parameters

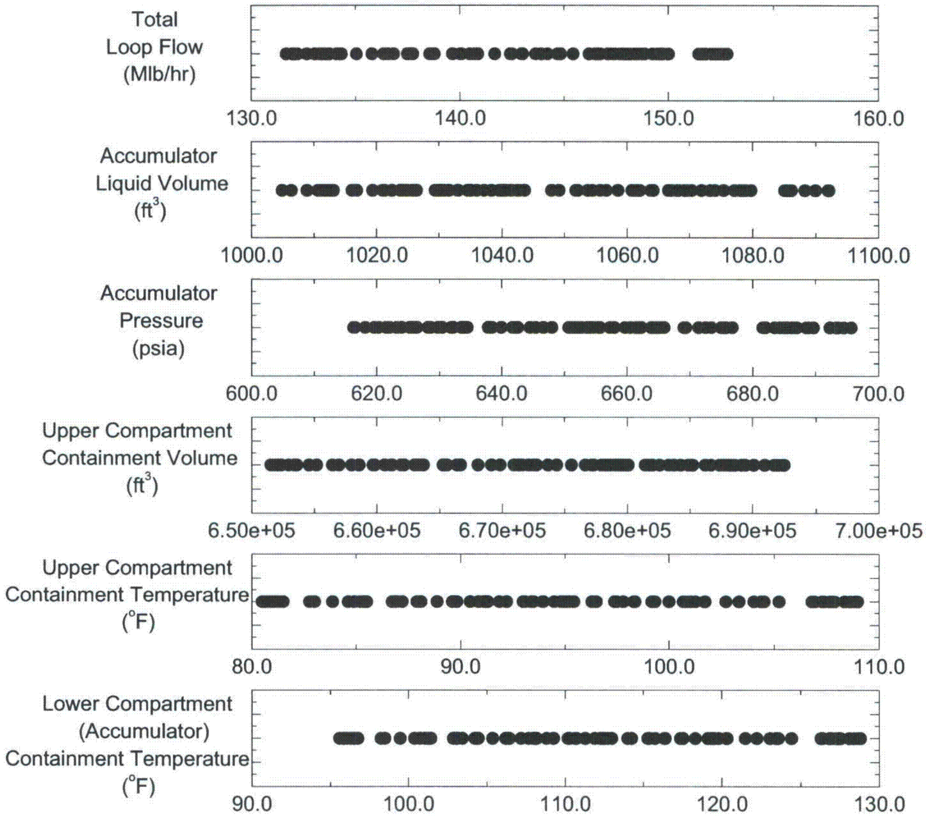

Figure 3-7 Scatter Plot of Operational Parameters ***(Continued)***

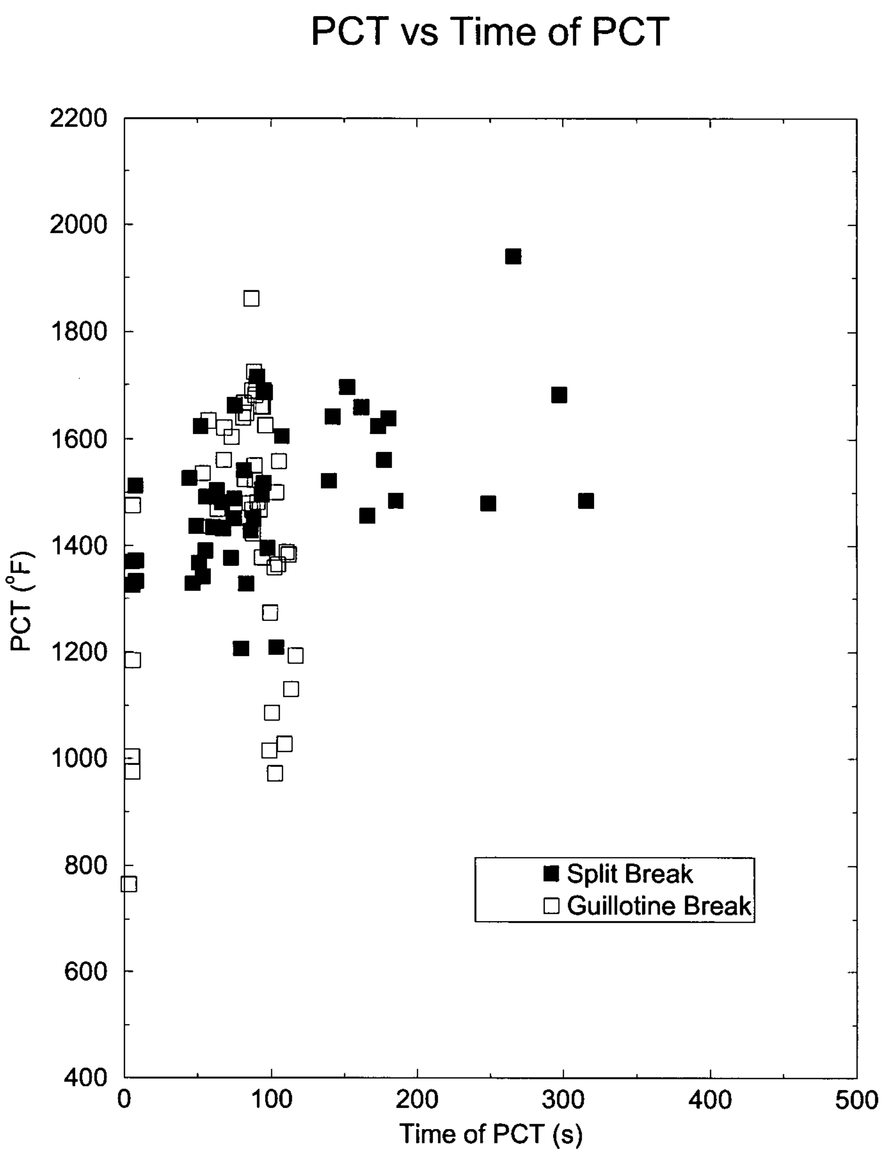

Figure 3-8 PCT versus PCT Time Scatter Plot

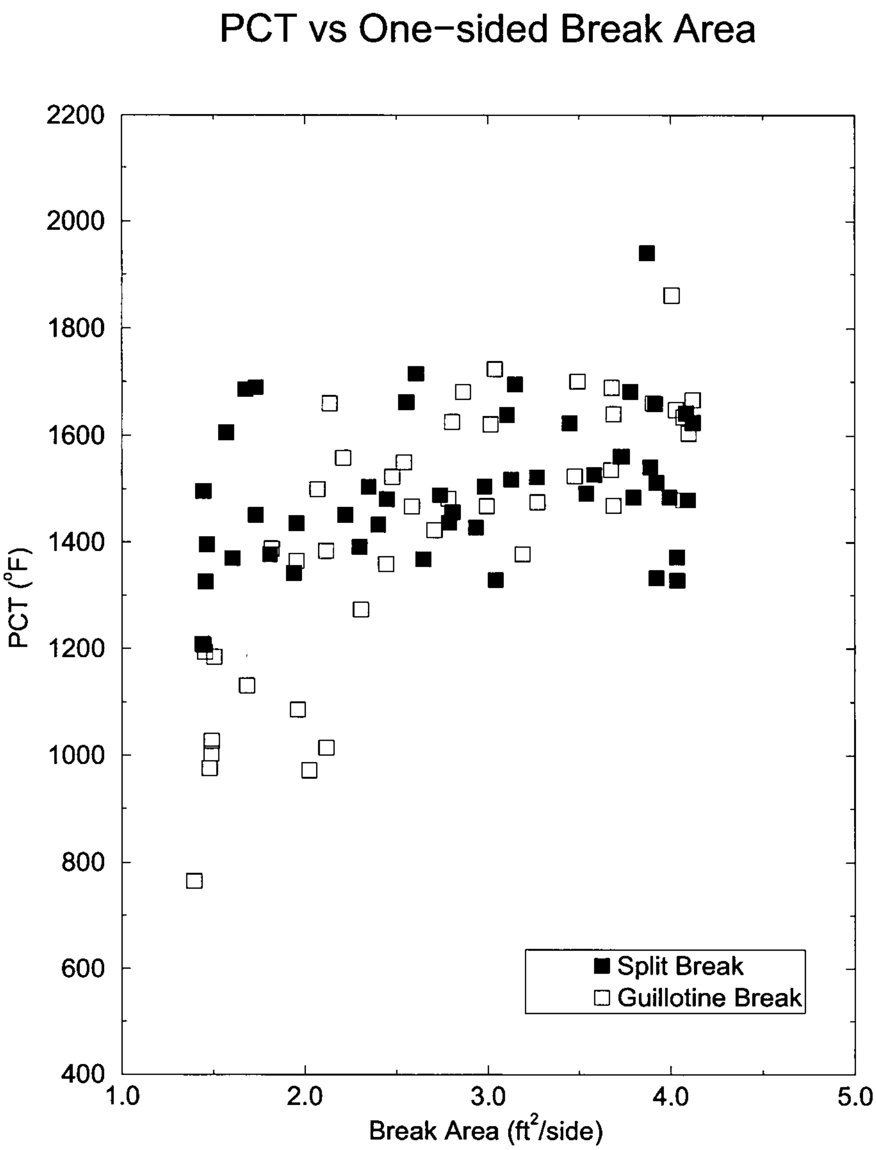

Figure 3-9 PCT versus Break Size Scatter Plot

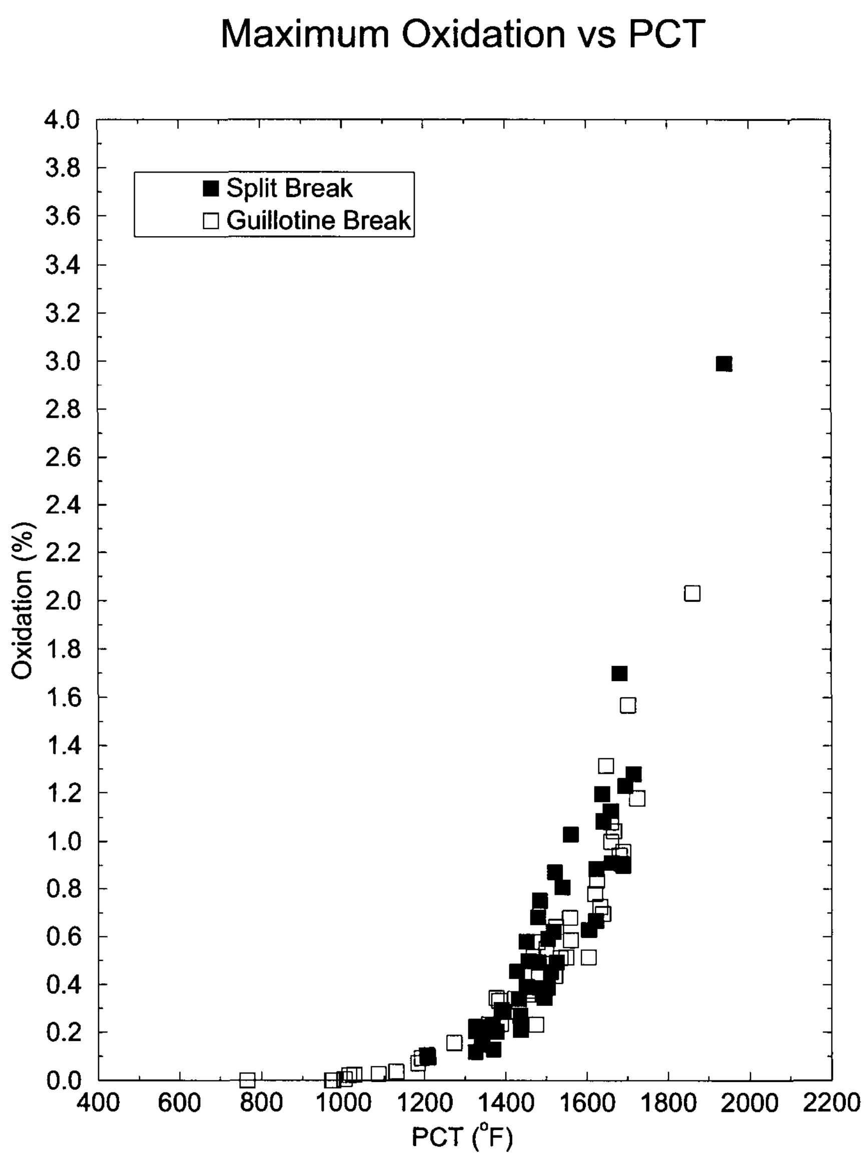

Figure 3-10 Maximum Transient Oxidation versus PCT Scatter Plot

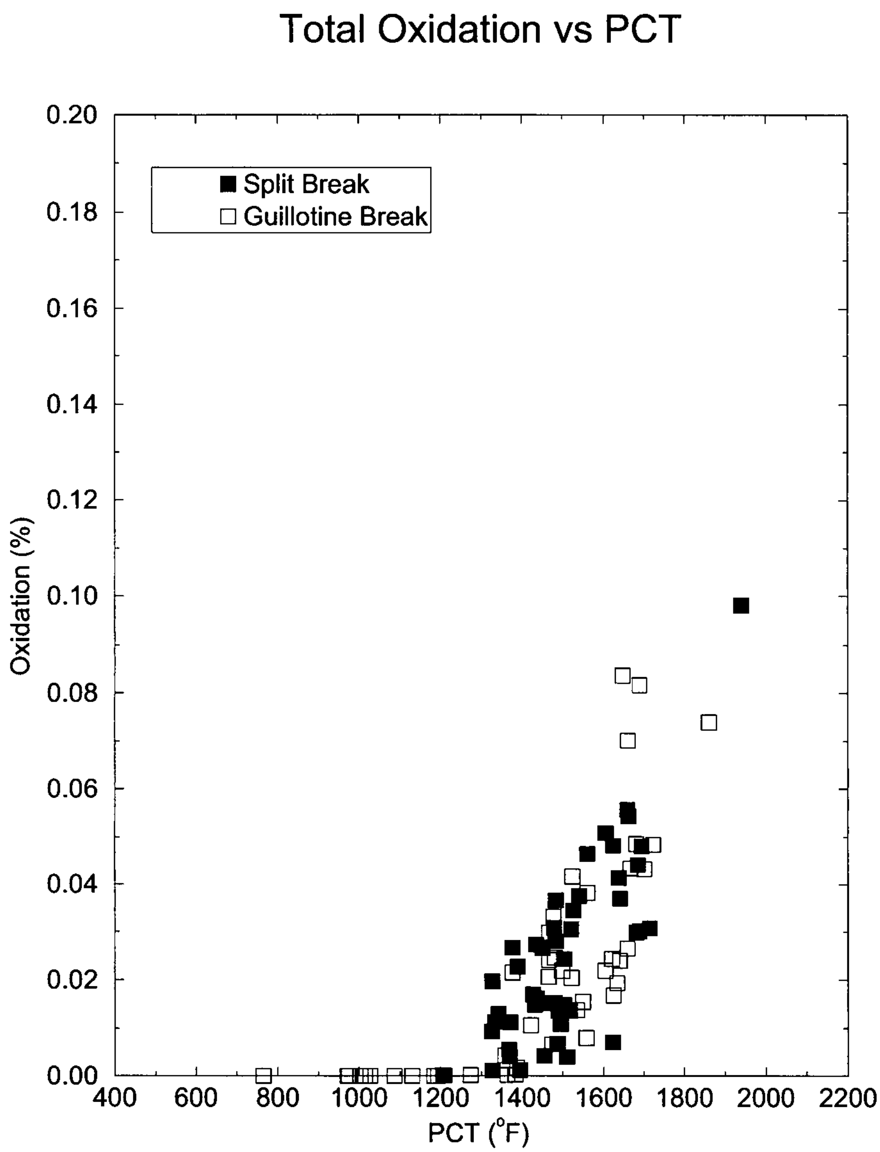

Figure 3-11 Total Oxidation versus PCT Scatter Plot

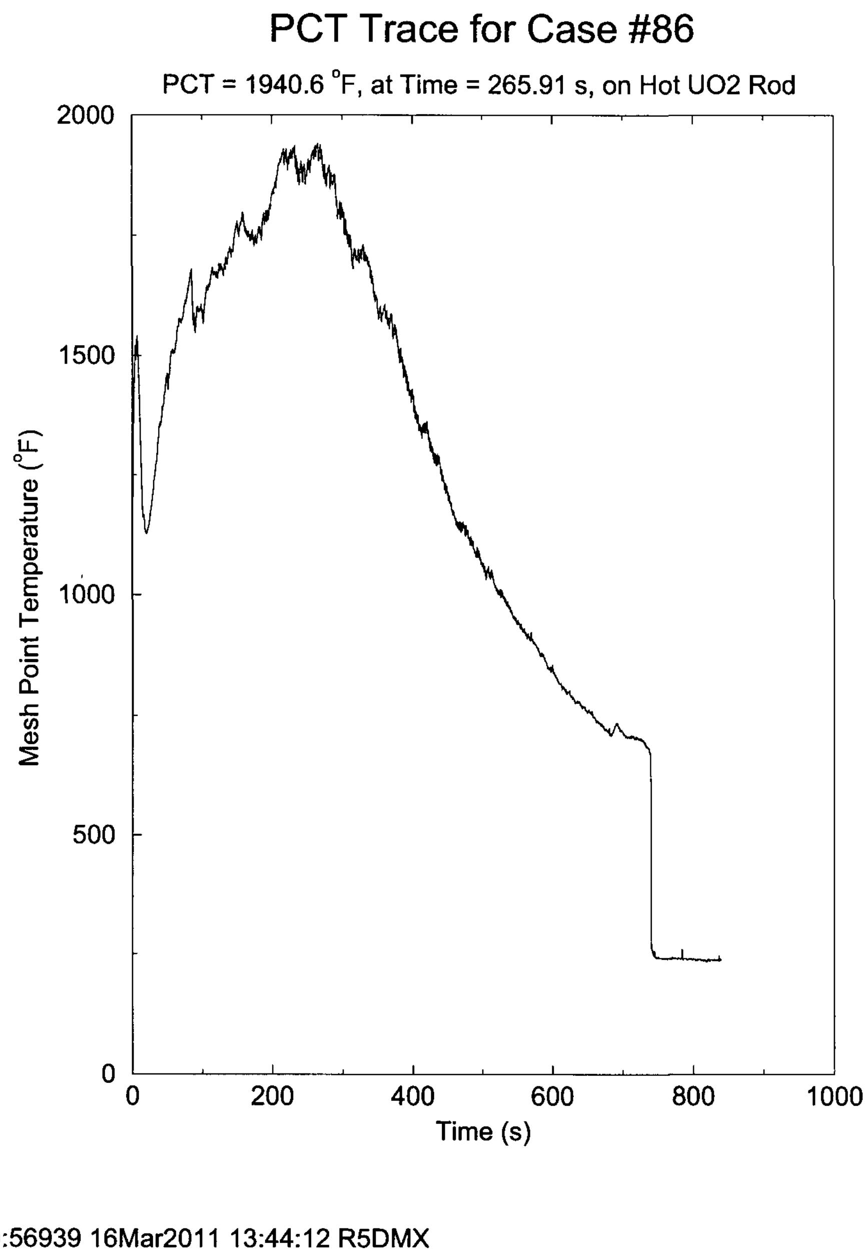

Figure 3-12 Peak Cladding Temperature (Independent of Elevation) for the Limiting Case

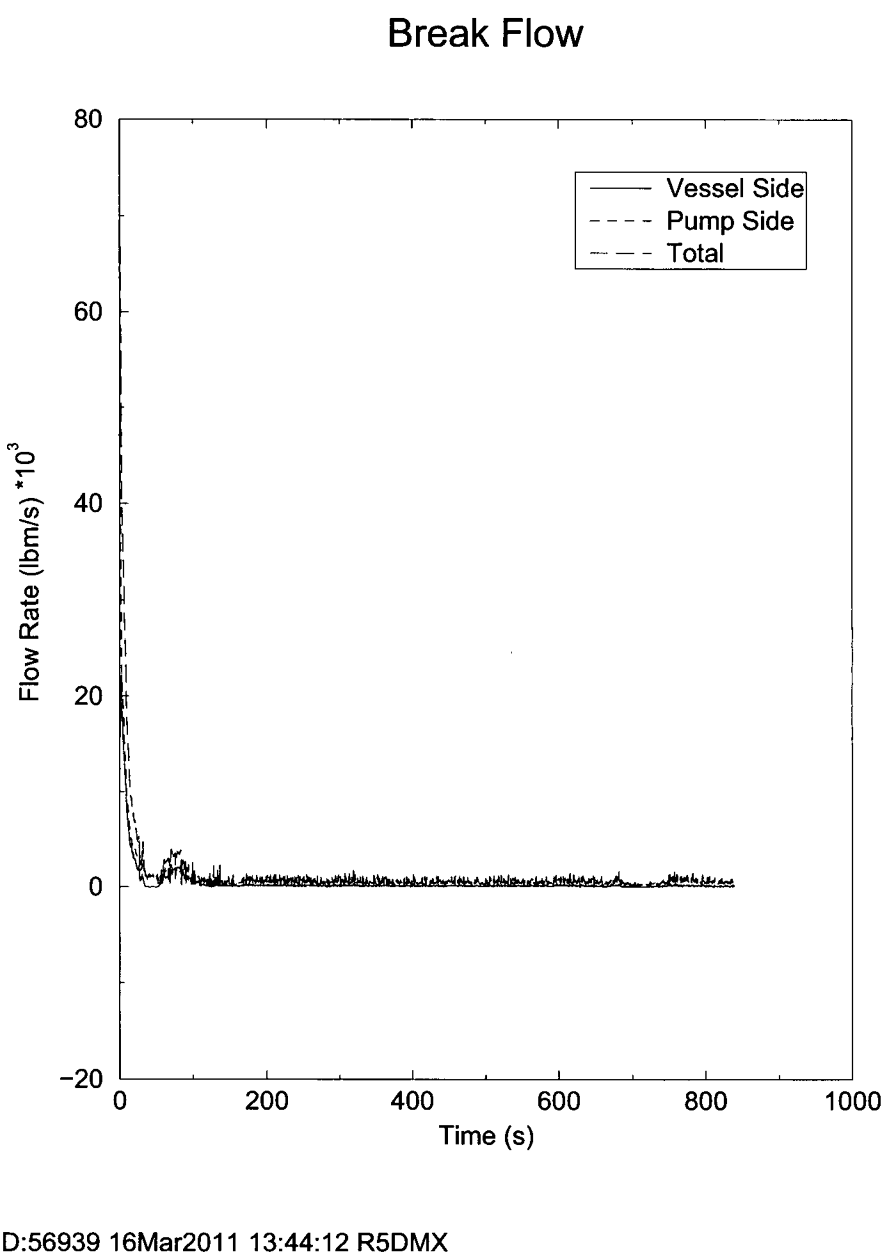

Figure 3-13 Break Flow for the Limiting Case

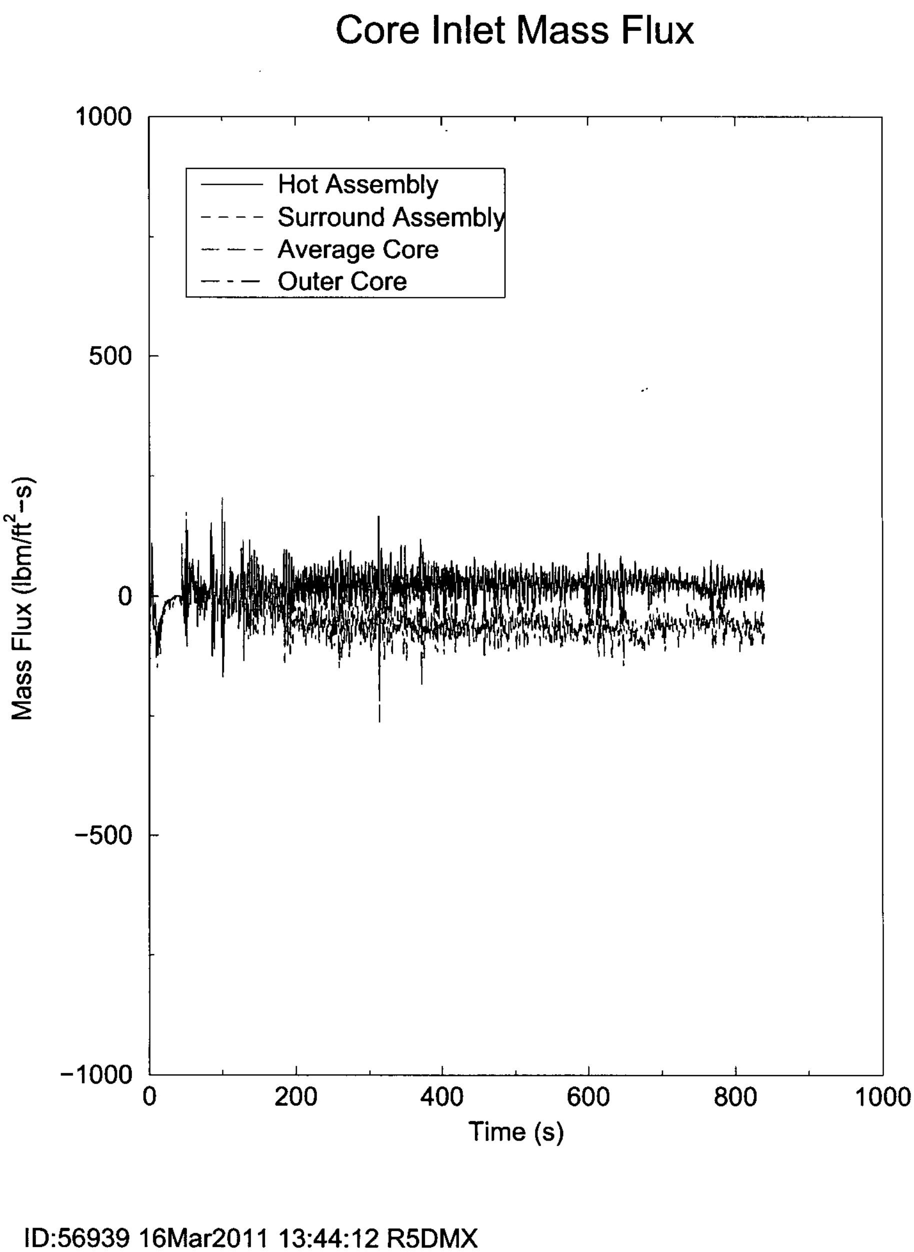

Figure 3-14 Core Inlet Mass Flux for the Limiting Case

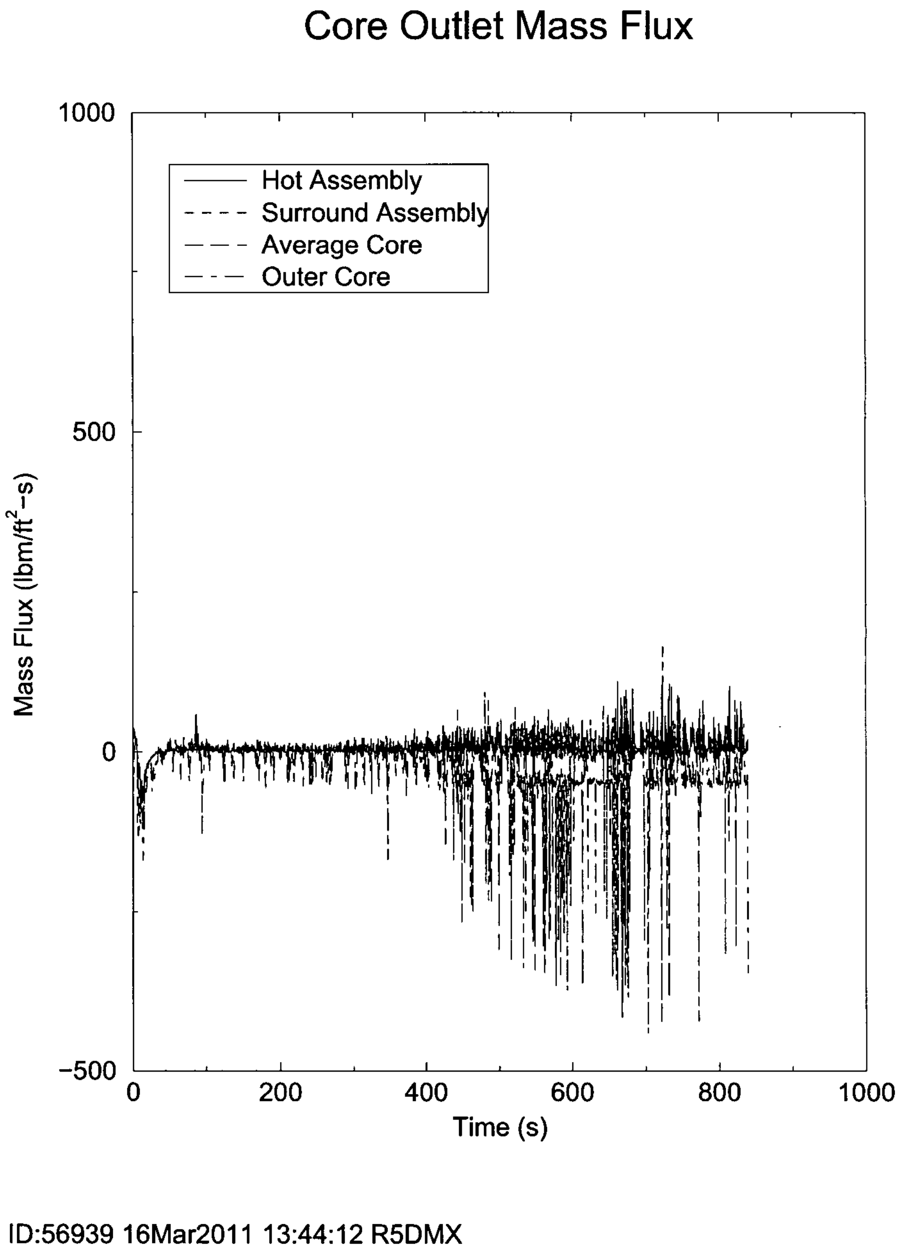

Figure 3-15 Core Outlet Mass Flux for the Limiting Case

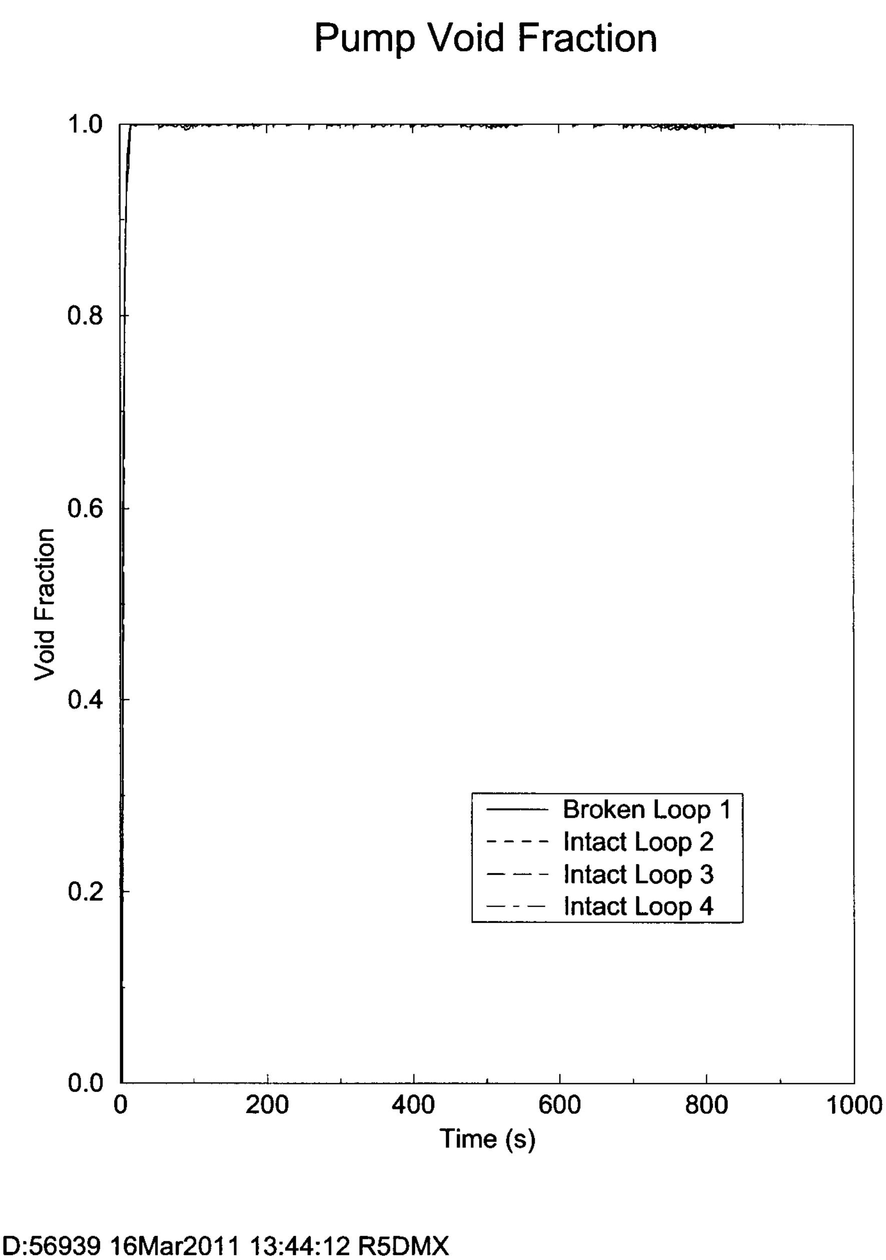

Figure 3-16 Void Fraction at RCS Pumps for the Limiting Case

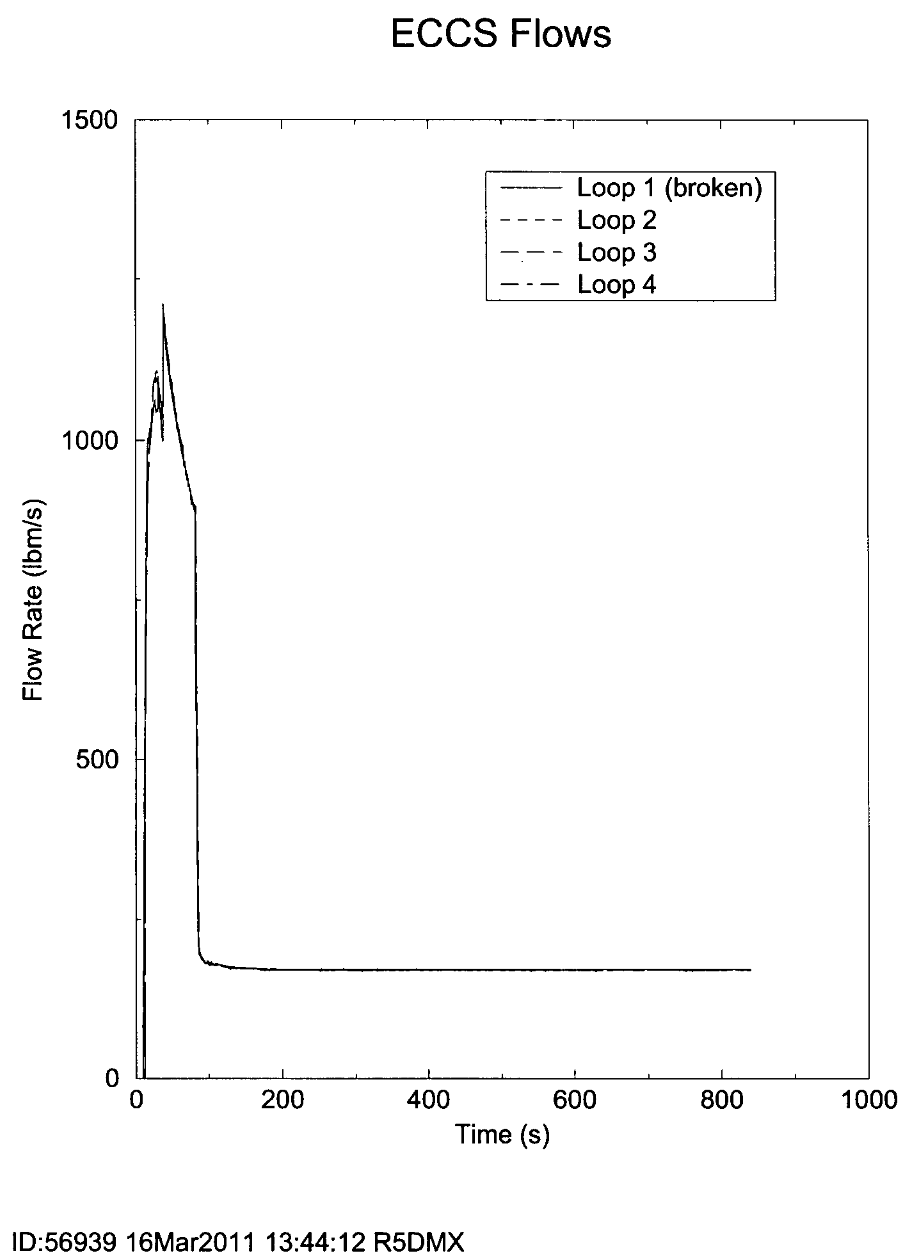

Figure 3-17 ECCS Flows (Includes Accumulator, Charging, SI and RHR) for the Limiting Case

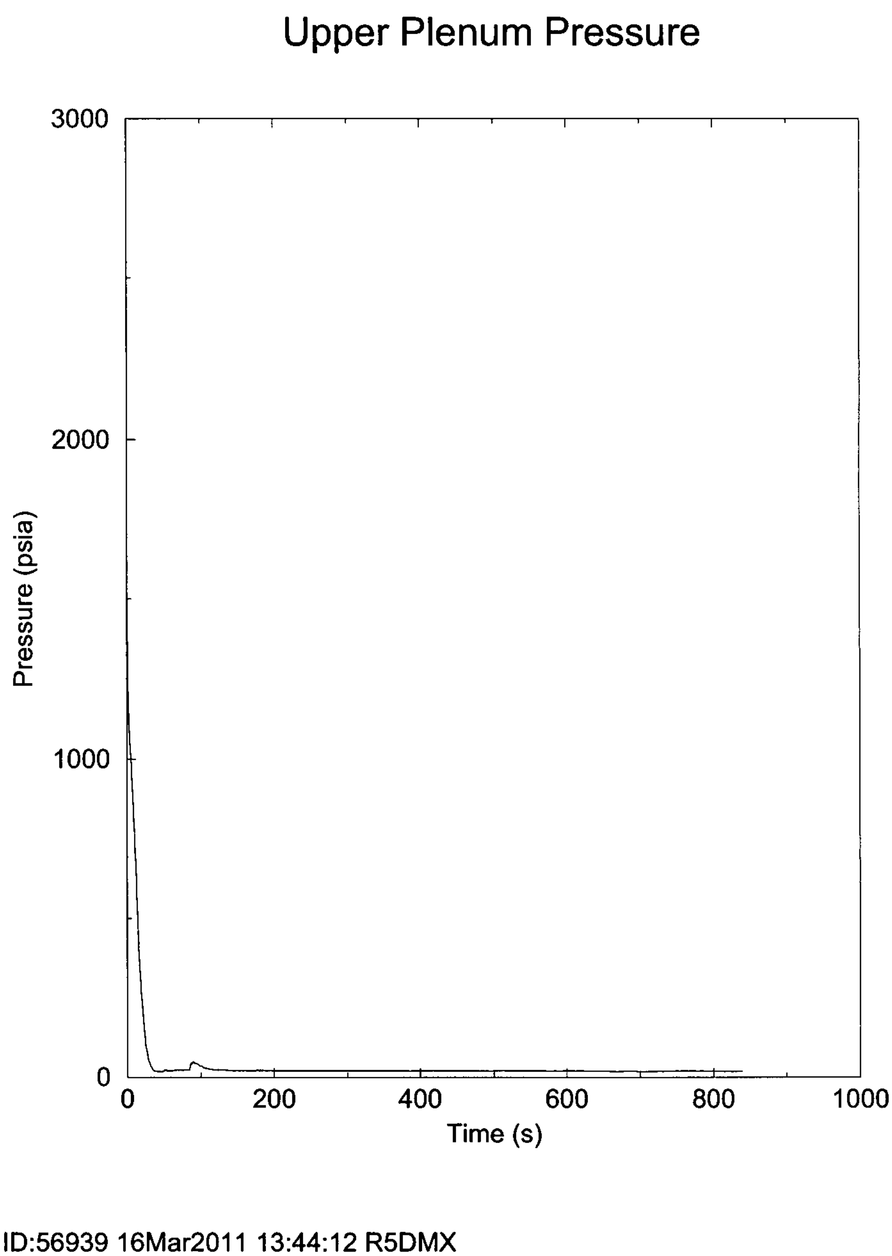

Figure 3-18 Upper Plenum Pressure for the Limiting Case

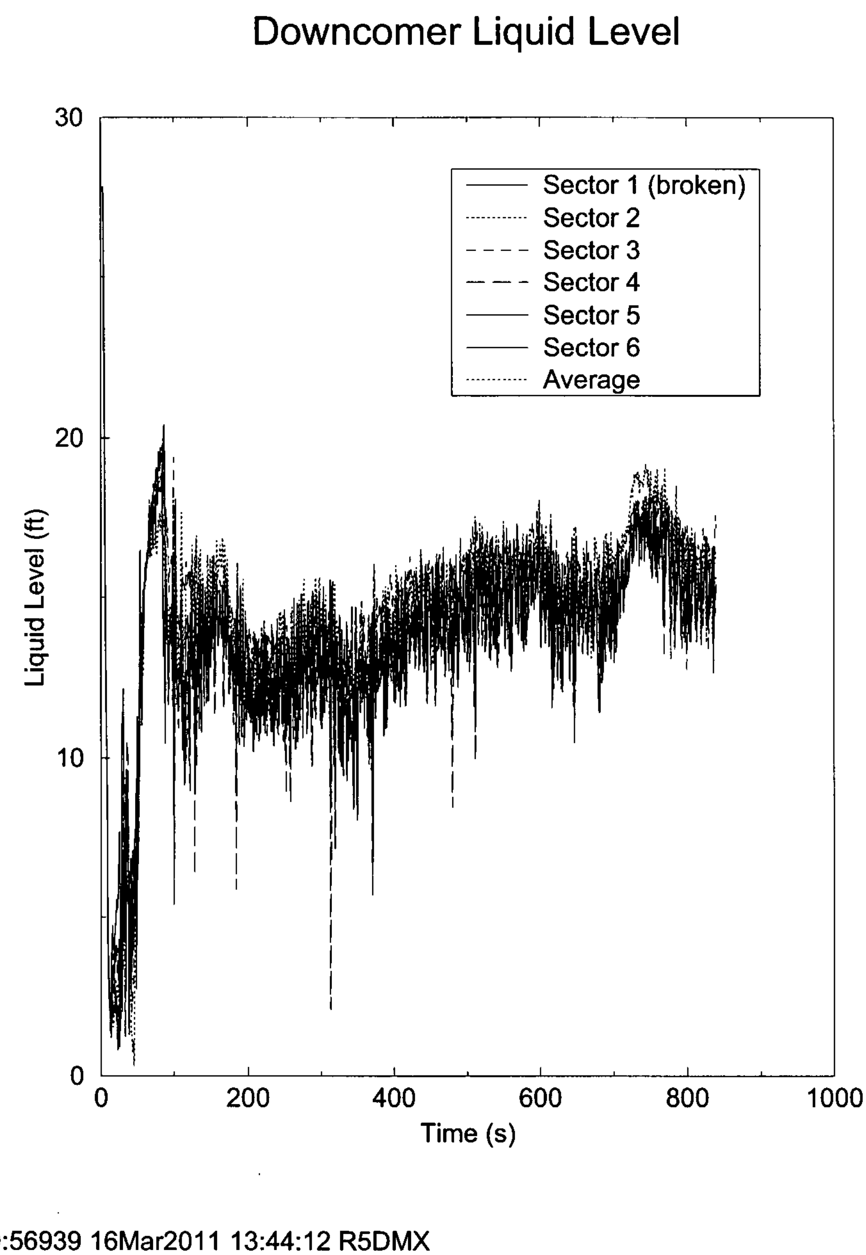

Figure 3-19 Collapsed Liquid Level in the Downcomer for the Limiting Case

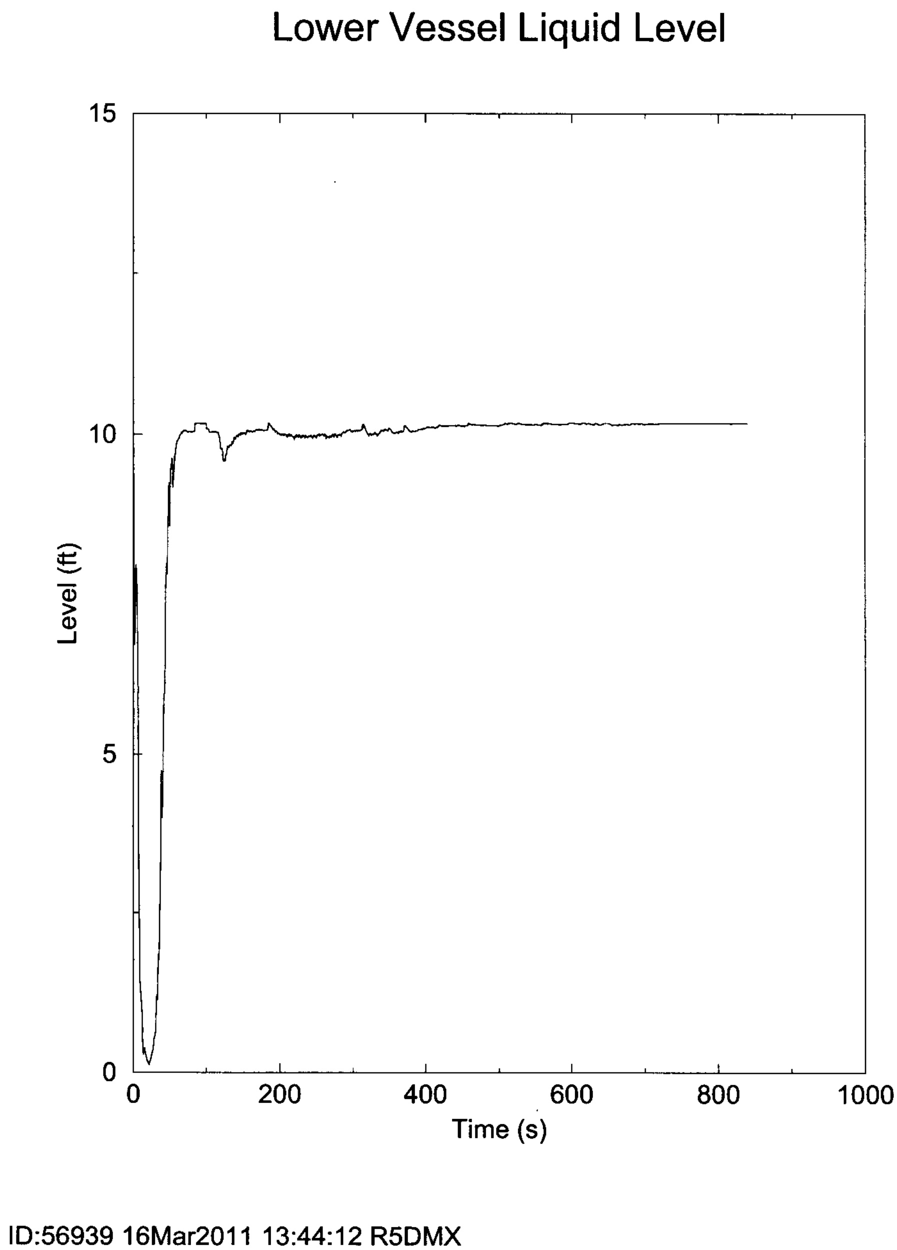

Figure 3-20 Collapsed Liquid Level in the Lower Plenum for the Limiting Case

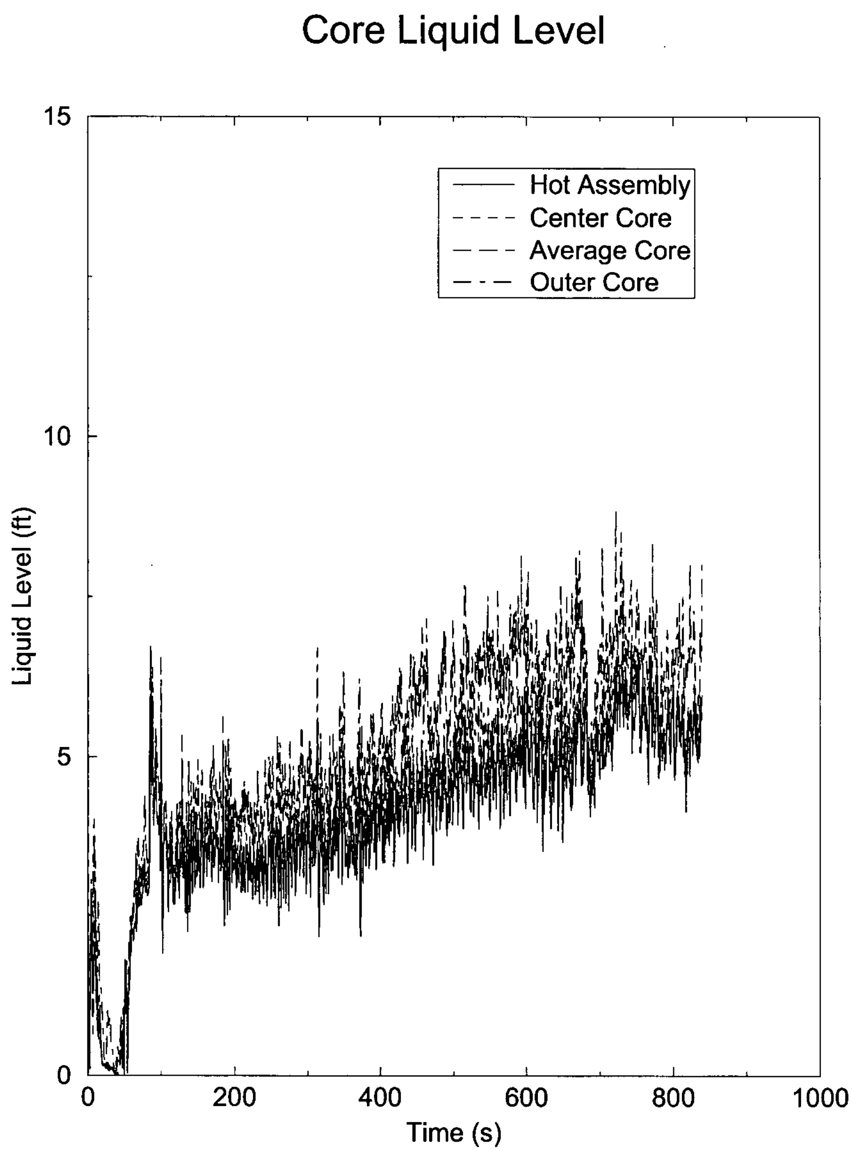

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Figure 3-21 Collapsed Liquid Level in the Core for the Limiting Case

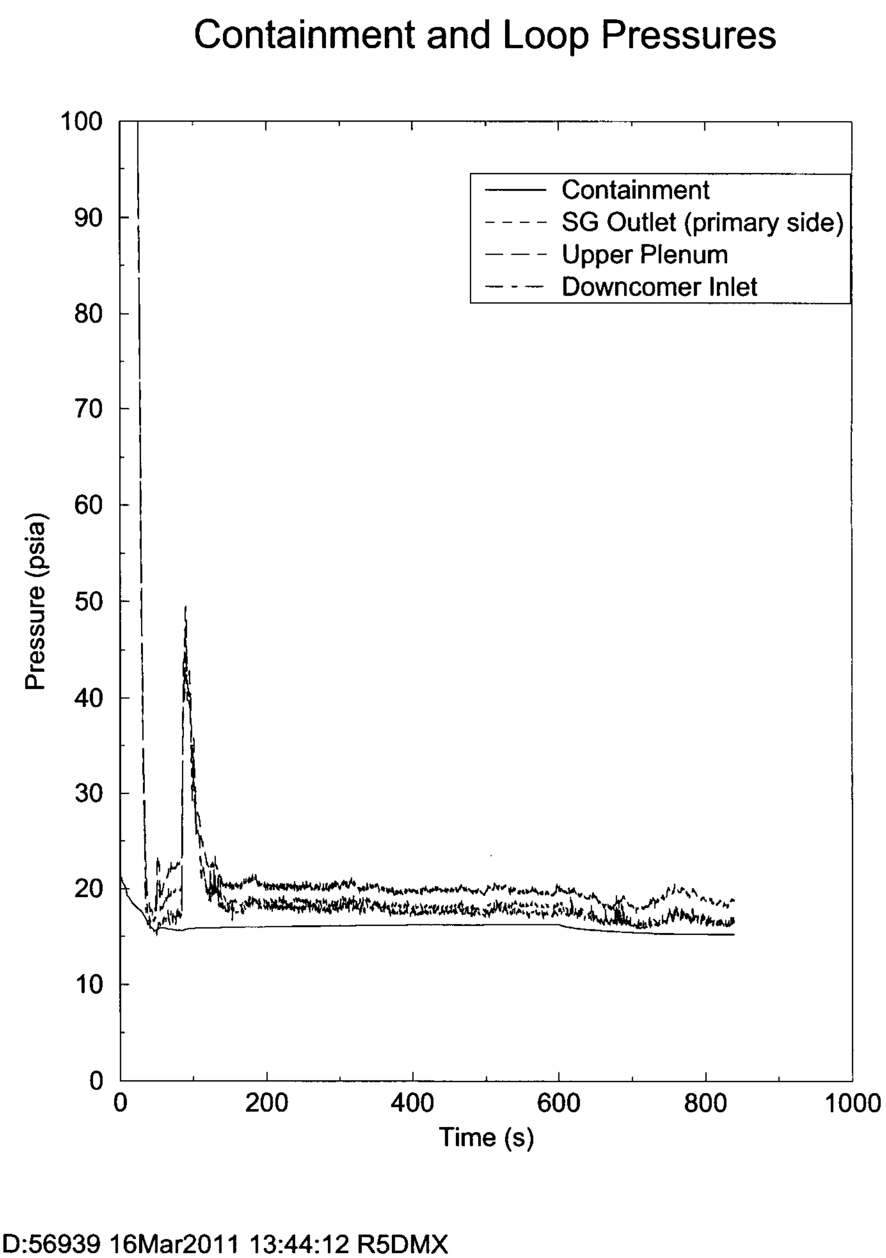

Figure 3-22 Containment and Loop Pressures for the Limiting Case

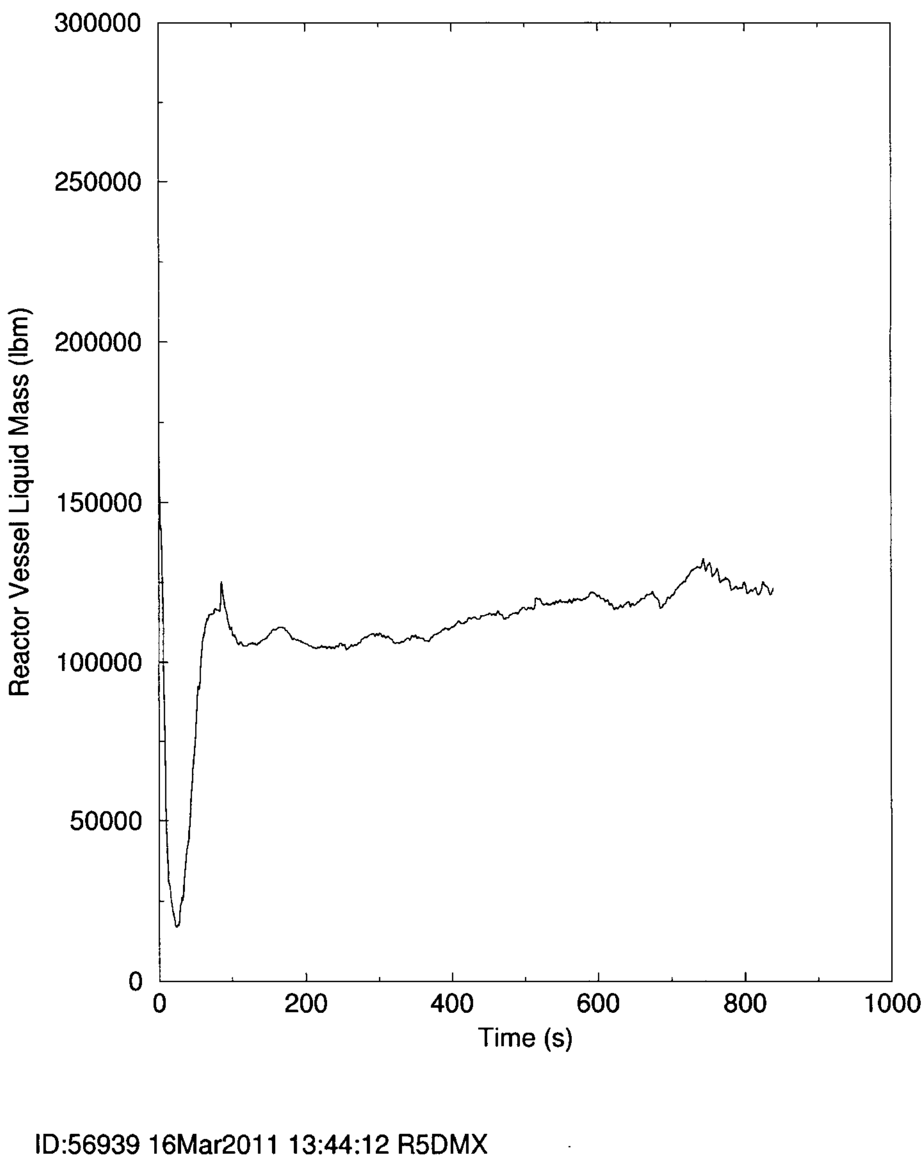

Figure 3-23 Reactor Vessel Liquid Mass (lbm) versus Time (sec)

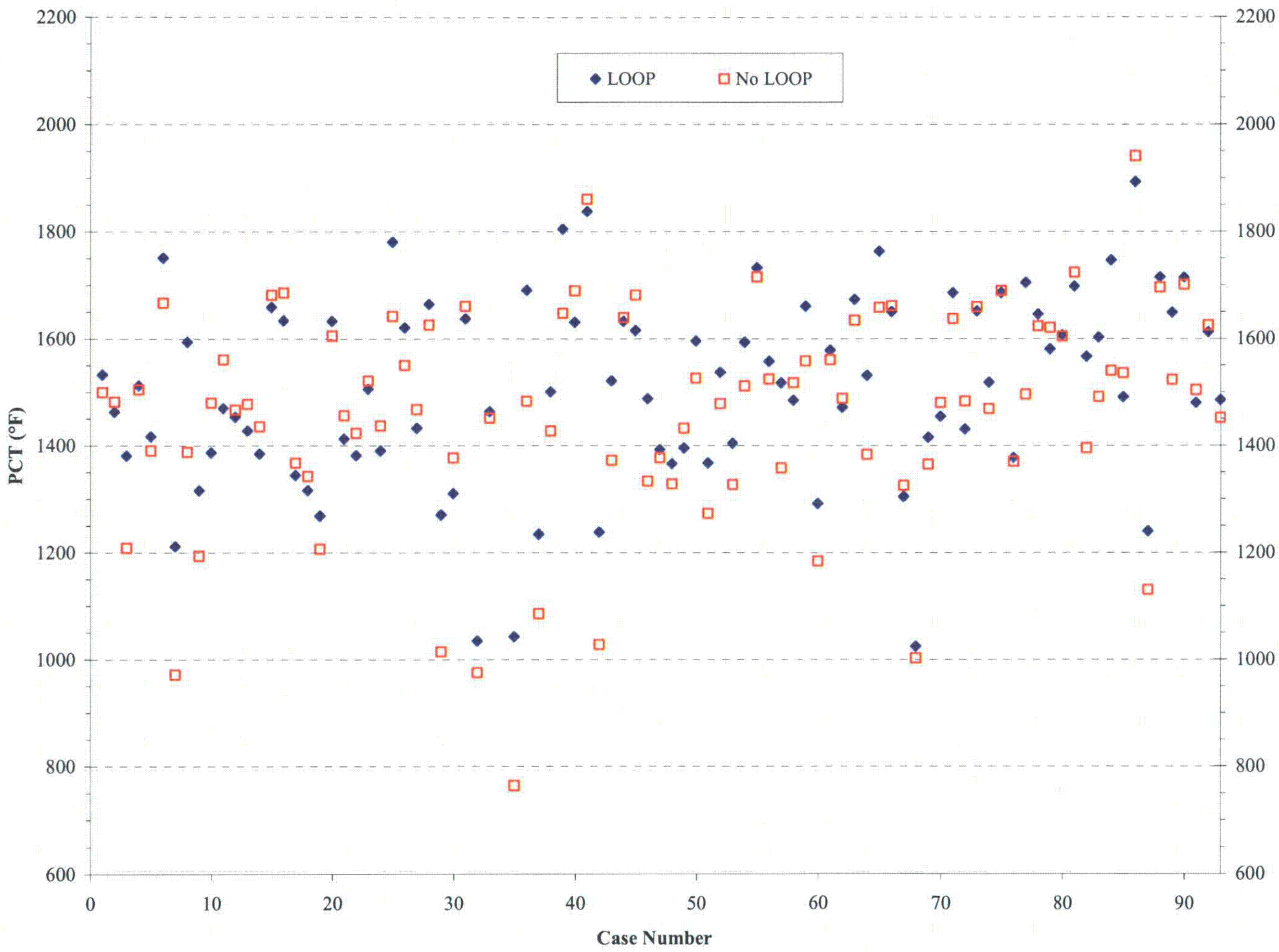

Figure 3-24 GDC 35 LOOP versus No-LOOP Cases

Case 29 TD Quench Screen − Limiting Rod

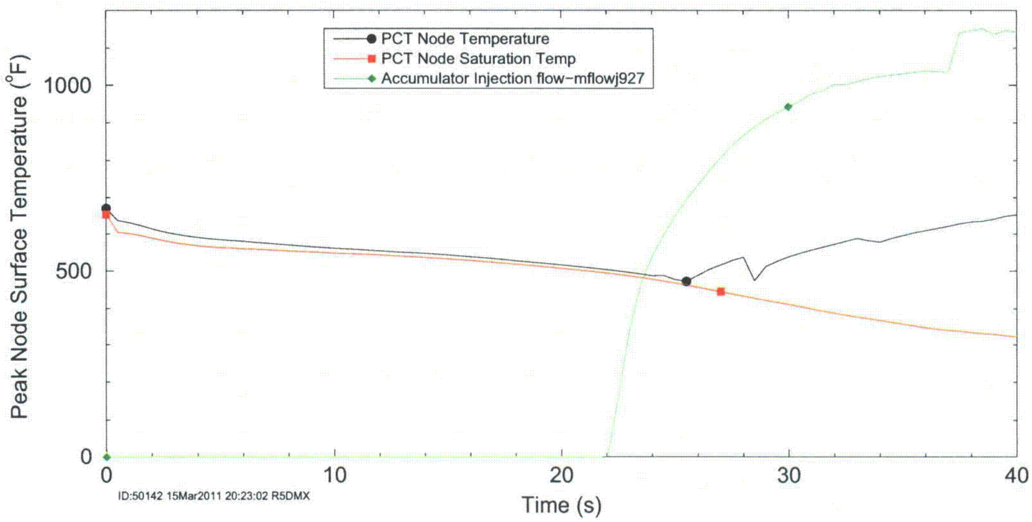

Figure 3-25: PCT Node Cladding Surface Temperature and Saturation Temperature, Case 29

Case 32 TD Quench Screen − Limiting Rod

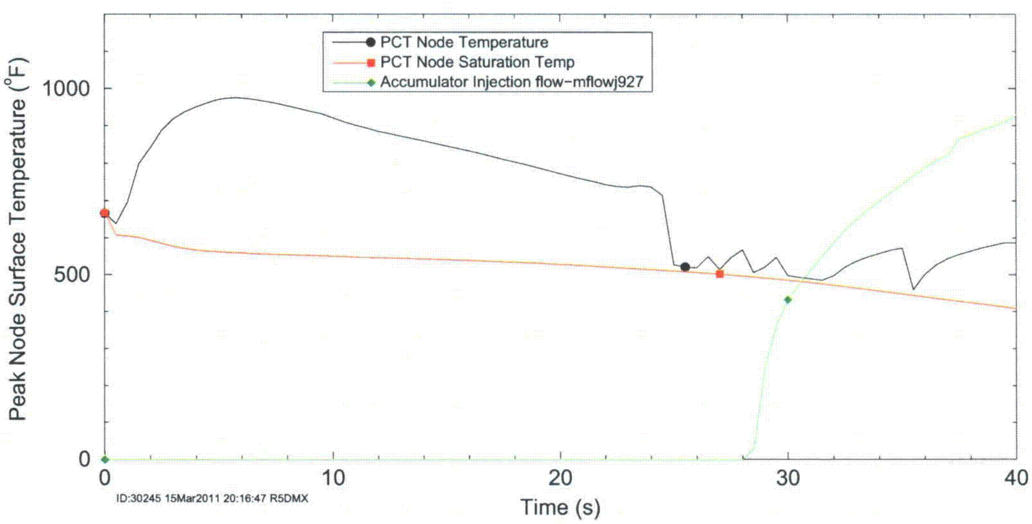

Figure 3-26: PCT Node Cladding Surface Temperature and Saturation Temperature, Case 32

Case 35 TD Quench Screen – Limiting Rod

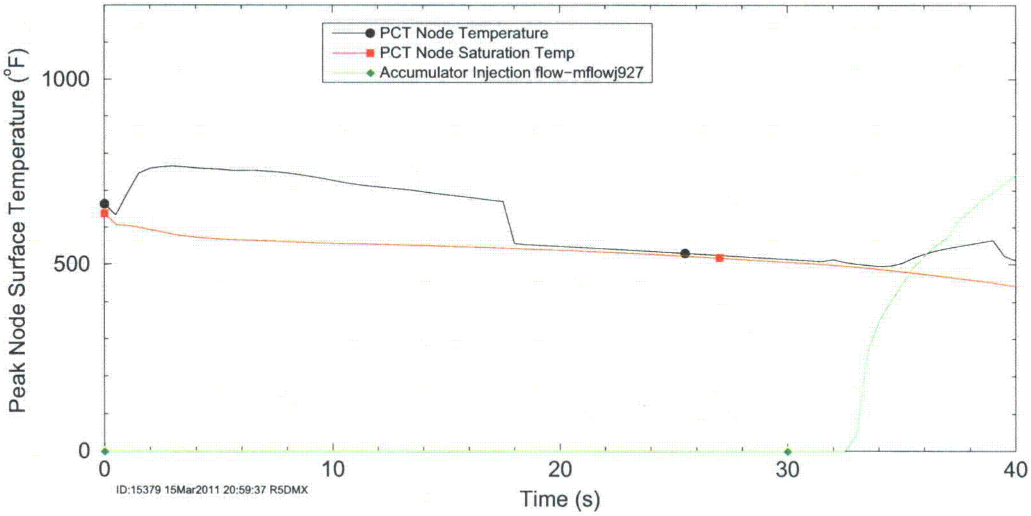

Figure 3-27: PCT Node Cladding Surface Temperature and Saturation Temperature, Case 35

Case 42 TD Quench Screen – Limiting Rod

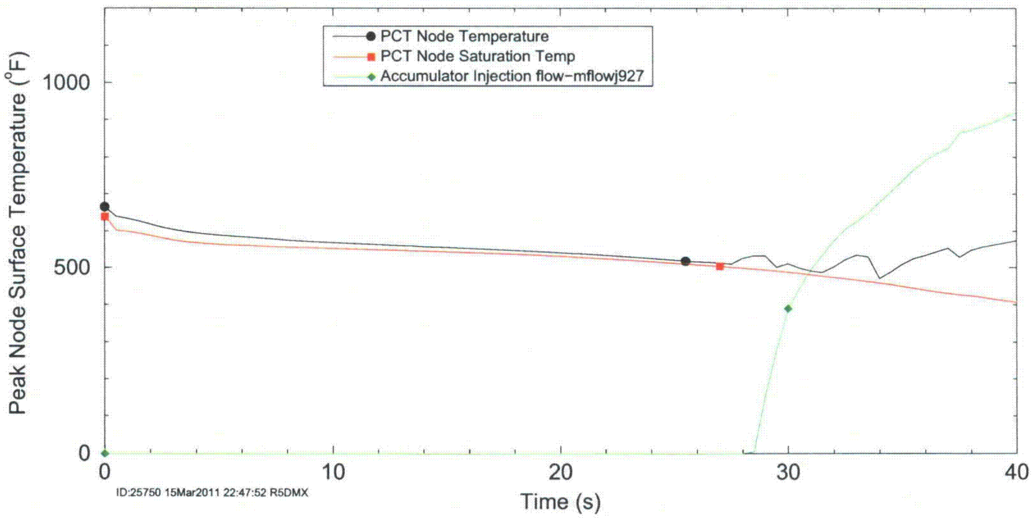

Figure 3-28: PCT Node Cladding Surface Temperature and Saturation Temperature, Case 42

Case 68 TD Quench Screen − Limiting Rod

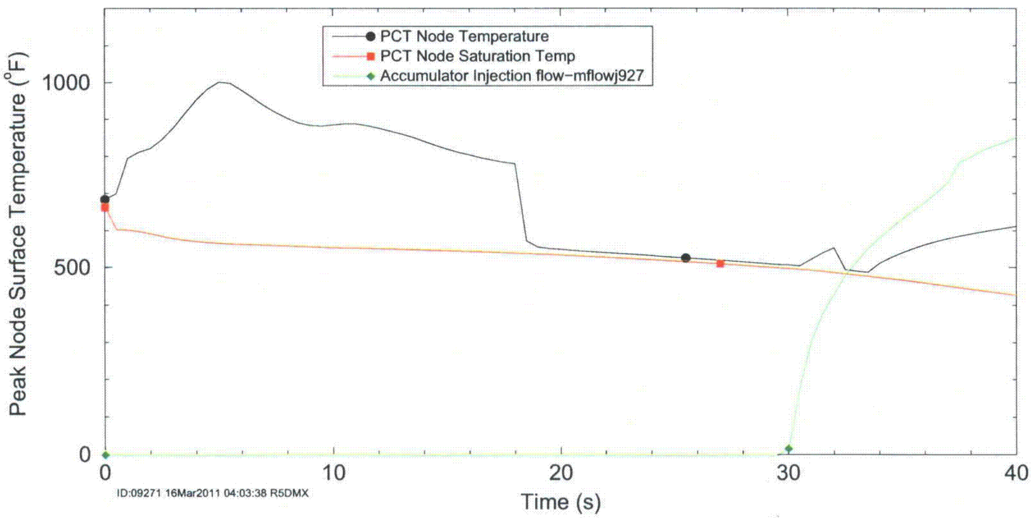

Figure 3-29: PCT Node Cladding Surface Temperature and Saturation Temperature, Case 68

Case 87 TD Quench Screen − Limiting Rod

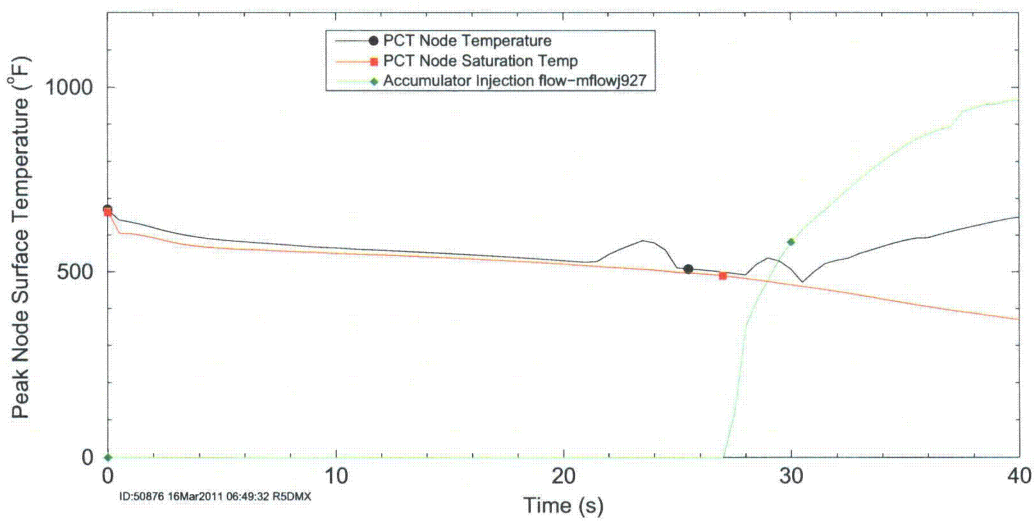

Figure 3-30: PCT Node Cladding Surface Temperature and Saturation Temperature, Case 87

4.0 Generic Support for Transition Package

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

4.1 Reactor Power

Question: *Reactor Power - Table 3-2, 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 . . ."*

Response: As indicated in Item 2.1 of Table 3-2 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. Not sampling the power level 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.

4.2 Rod Quench

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

Response: Yes, the version of S-RELAP5 employed for the Sequoyah 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 T_{min} would exceed 755 K (900 °F). For the Sequoyah case set T_{min} was never sampled above 703.4 K (806.5 °F). This is a change to the approved RLBLOCA EM (Reference 1).

4.3 Rod-to-Rod Thermal Radiation

Question: *Provide justification that the S-RELAP5 rod-to-rod thermal radiation model applies to the SQN 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 The Full-Length Emergency Core Heat Transfer Separate Effects and Systems Effects Tests (FLECHT-SEASET) 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 Thermal Hydraulic Test Facility (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), Loss of Fluid Test (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 (Reference 1). As expected, these figures demonstrate that the comparison between the code calculations and data is improved with the application of the derived biases. The CCTF, LOFT, and Semiscale benchmarks further indicate that, whatever consideration of rod-to-rod radiation is implicit in the S-RELAP5 reflood heat transfer modeling, it does not significantly effect code predictions under conditions where radiation is minimized. The measured PCTs in these assessments ranged from approximately 1,000 to 1,540 °F. At these temperatures, there is little rod-to-rod radiation. Given the good agreement between the biased code calculations and the CCTF, LOFT, and Semiscale data, it can be concluded that there is no significant over prediction of the total heat transfer coefficient.

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

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

As discussed above, the FLECHT-SEASET and THTF tests were selected to assess and determine the S-RELAP5 code heat transfer bias and uncertainty. A uniform radial power distribution was used in these test bundles. Therefore, the rod-to-rod temperature variation in the rods away from the vessel wall is caused primarily by the variation in the sub-channel fluid conditions. In the real operating fuel bundle, on the other hand, there can be 5- to 10-percent rod-to-rod power variation. In addition, the methodology includes a provision to apply the uncertainty measurement to the hot pin. Table 4-1 provides the hot pin measurement uncertainty and a representative local pin peaking factor for several plants. These factors, however, relate the pin to the assembly average. To more properly assess the conditions under which rod-to-rod radiation heat transfer occurs, a more local peaking assessment is required. Therefore, the plant rod-to-rod radiation assessments herein set the average pin power for those pins surrounding the hot pin at 96-percent of that of the peak pin. For pins further removed the average power is set to 94-percent.

Table 4-1 Typical Measurement Uncertainties and Local Peaking Factors

Plant	$F_{\Delta H}$ 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

4.3.2 Quantification of the Impact of Thermal Radiation using R2RRAD Code

The R2RRAD radiative heat transfer model was developed by Los Alamos National Laboratory (LANL) to be incorporated in the BWR version of the TRAC code. The theoretical basis for this code is given in References 8 and 11 and is similar to that developed in the HUXY rod heatup code (Reference 10, Section 2.1.2) used by AREVA for BWR LOCA applications. The version of R2RRAD used herein was obtained from the NRC to examine the rod-to-rod radiation characteristics of a 5x5 rod segment of the 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 4-2, between the test bundle and a modern 17x17 assembly.

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

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

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

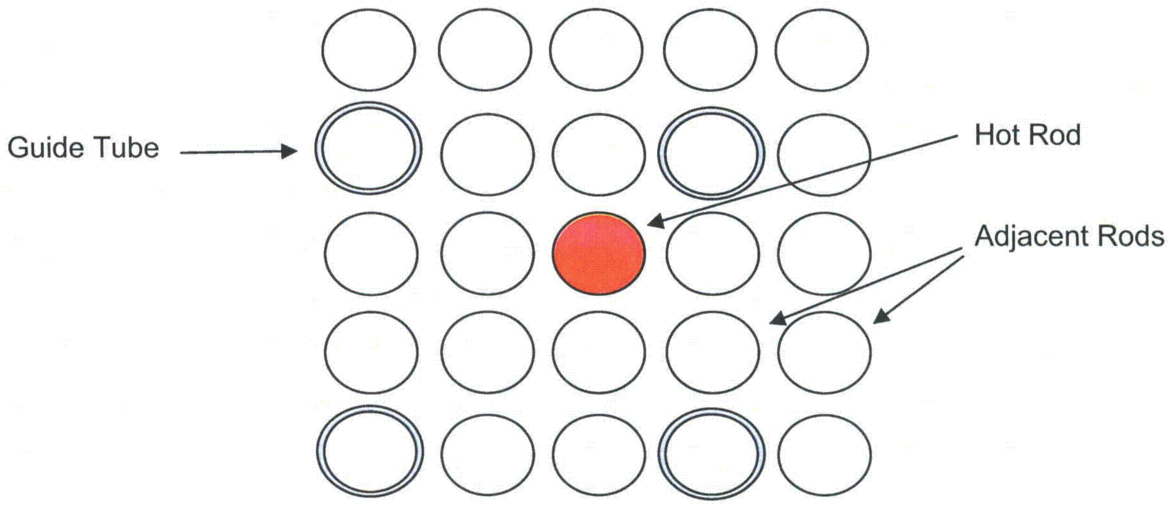

Figure 4-1 R2RRAD 5x5 Rod Segment

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. Because peak rod powers frequently occur at fuel assembly corners away from either guide tubes or instrument tubes and for added conservatism, the guide tubes (thimble tubes) were replaced by fuel rods in the input model described above. In line with the discussion in Section 4.3.1, the surrounding 24 rods were set to a temperature estimated for rods of 4-percent lower power. The boundary temperature was estimated based 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.

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

T_{sat} was taken as 270 °F.

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

Table 4-3 FLECHT-SEASET Test Parameters

Test	Rod 7J PCT at 6-ft (°F)	PCT Time (s)	HTC at PCT Time (Btu/hr-ft²-°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			

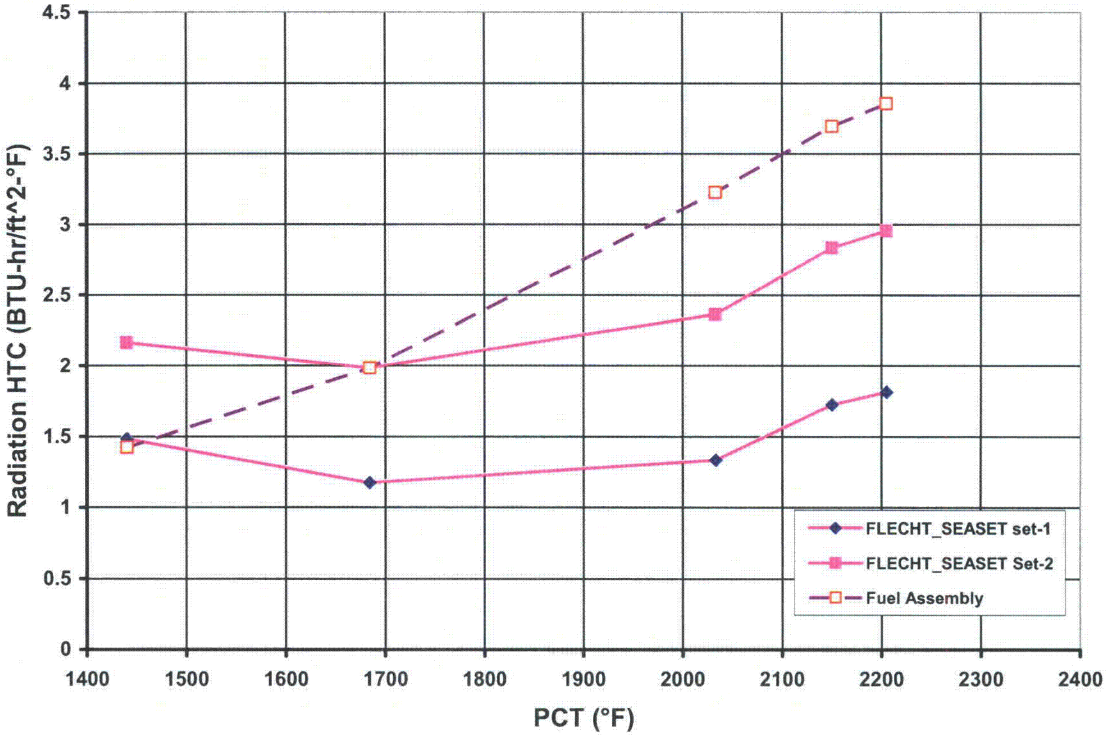

Figure 4-2 Rod Thermal Radiation in FLECHT-SEASET Bundle and in a 17x17 FA

4.3.3 Rod-to-Rod Radiation Summary

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

4.4 Film Boiling Heat Transfer Limit

Question: *In the SQN 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 Units 1 and 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).

4.5 Downcomer Boiling

Question: *If the PCT is greater than 1800°F or the containment pressure is less than 30 psia, has the Sequoyah Units 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 the Sequoyah Units has been established generically as adequate for the computation of downcomer phenomena including the prediction of potential local boiling effects. The model was benchmarked against the UPTF tests and the LOFT facility in the RLBLOCA methodology, Revision 0 (Reference 1). Further, AREVA addressed the effects of boiling in the downcomer in a letter, from James Malay to U.S. NRC, April 4, 2003. The letter cites the lack of direct experimental evidence but contains sensitivity studies on high and low pressure containments, the impact of additional azimuthal noding within the downcomer, and the influence of flow loss coefficients. Of these, the study on azimuthal noding is most germane to this question; indicating that additional azimuthal nodalization allows higher liquid buildup in portions of the downcomer away from the broken cold leg and increases the liquid driving head. Additionally, AREVA has conducted downcomer axial noding and wall heat release studies. Each of these studies supports the Revision 0 methodology and is documented later in this section.

This question is primarily concerned with the phenomena of downcomer boiling and the extension of the Revision 0 methodology and sensitivity studies to plants with low containment pressures and high cladding temperatures. Boiling, wherever it occurs, is a phenomenon that codes like S-RELAP5 have been developed to predict. Downcomer boiling is the result of the release of energy stored in vessel metal mass. Within S-RELAP5, downcomer boiling is simulated in the nucleate boiling regime with the Chen correlation. This modeling has been validated through the prediction of several assessments on boiling phenomenon provided in the S-RELAP5 Code Verification and Validation document (Reference 12).

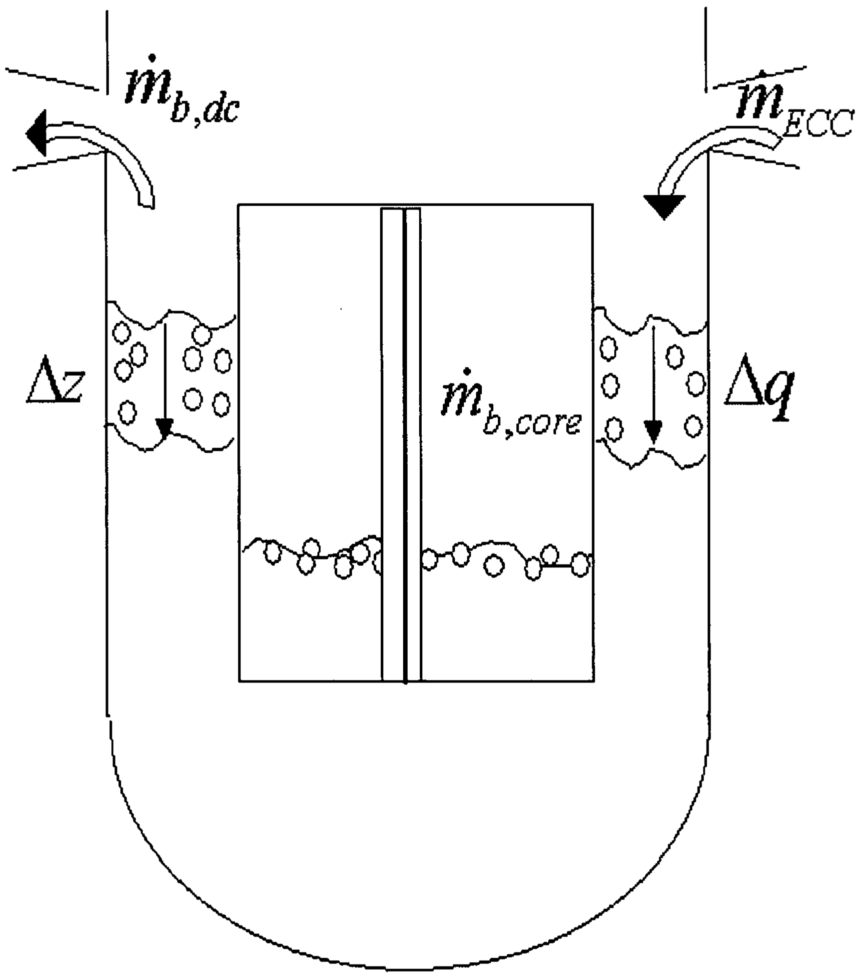

Figure 4-3 Reactor Vessel Downcomer Boiling Diagram

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

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

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

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

4.5.1 Wall Heat Release Rate

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

4.5.1.1 Exact Solution

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

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

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 4-4 shows the temperature distributions in the metal at 0.0, 100 and 300 seconds as calculated by using Equation 1 and S-RELAP5, respectively. The solutions are identical confirming the adequacy of the mesh spacing used in the downcomer wall.

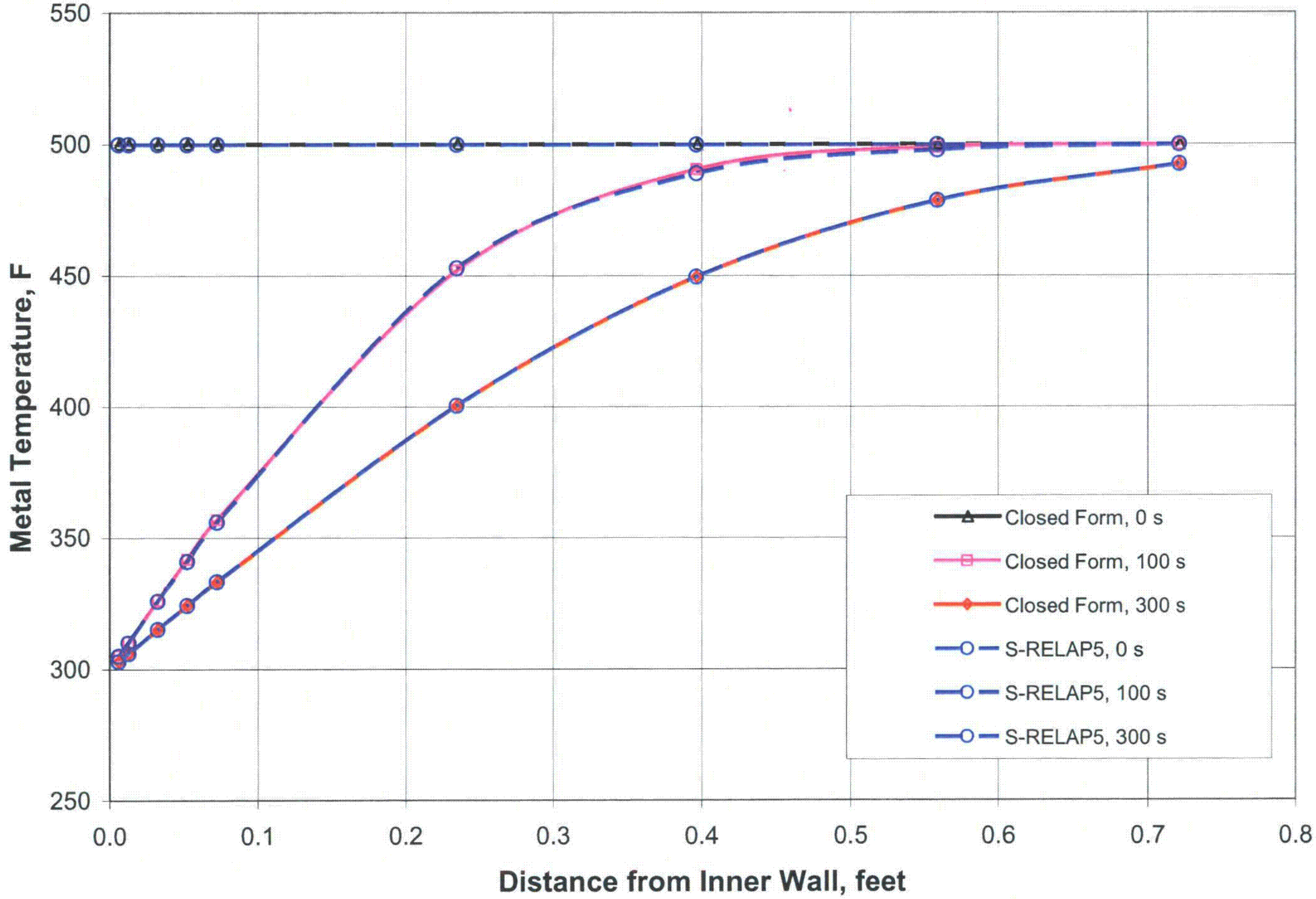

Figure 4-4 S-RELAP5 versus Closed Form Solution

4.5.1.2 Plant Model Sensitivity Study

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

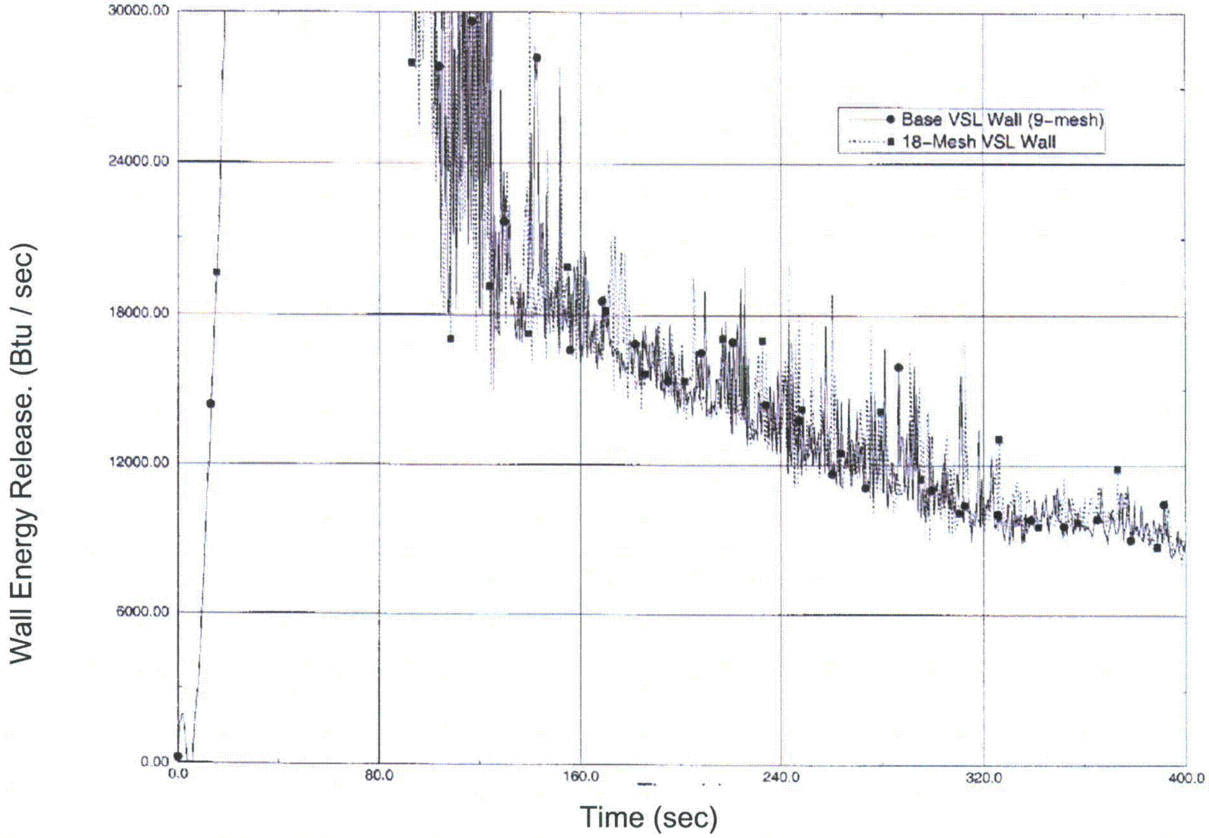

Figure 4-5 Downcomer Wall Heat Release – Wall Mesh Point Sensitivity

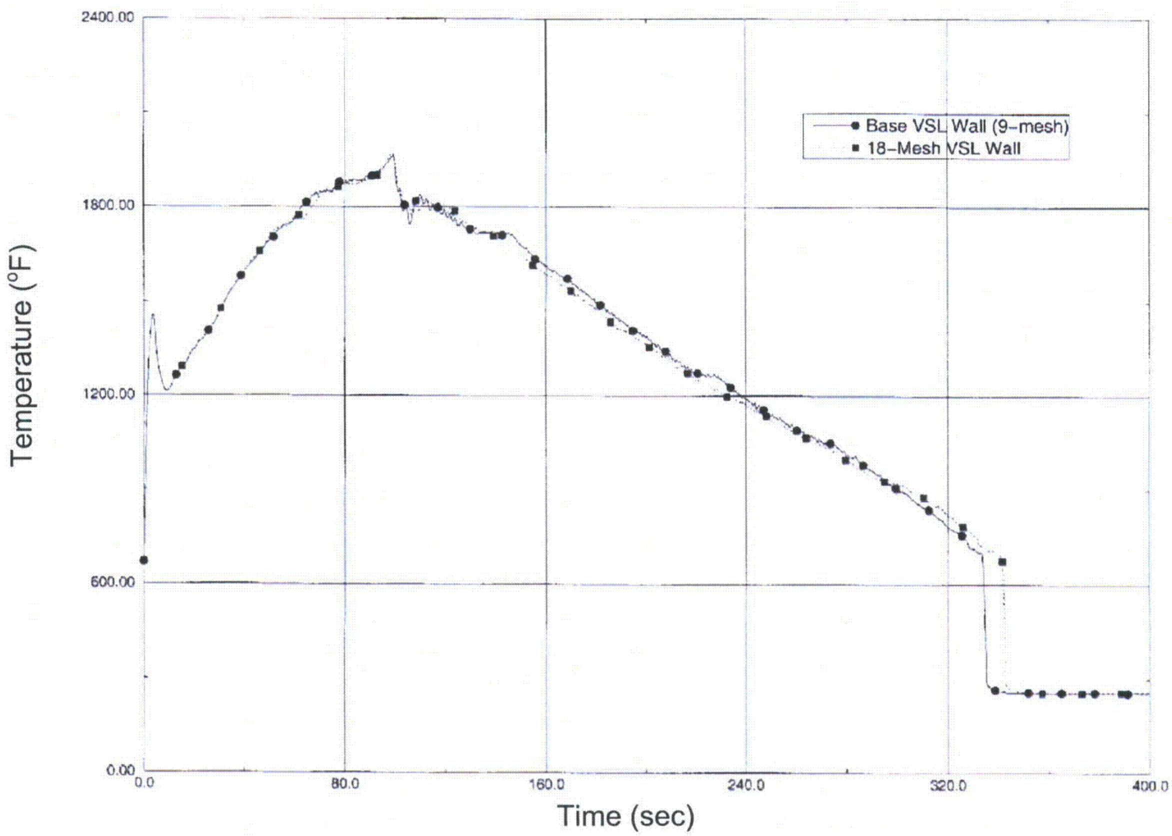

Figure 4-6 PCT Independent of Elevation – Wall Mesh Point Sensitivity

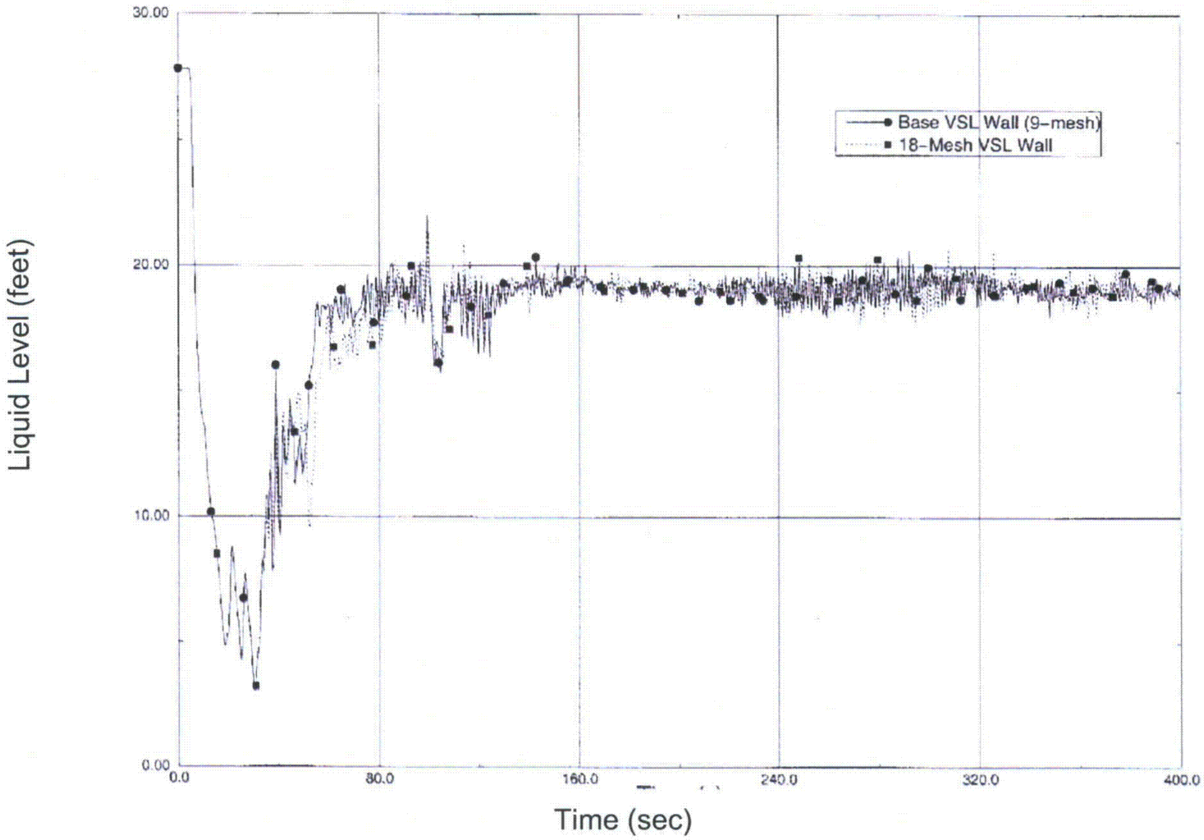

Figure 4-7 Downcomer Liquid Level – Wall Mesh Point Sensitivity

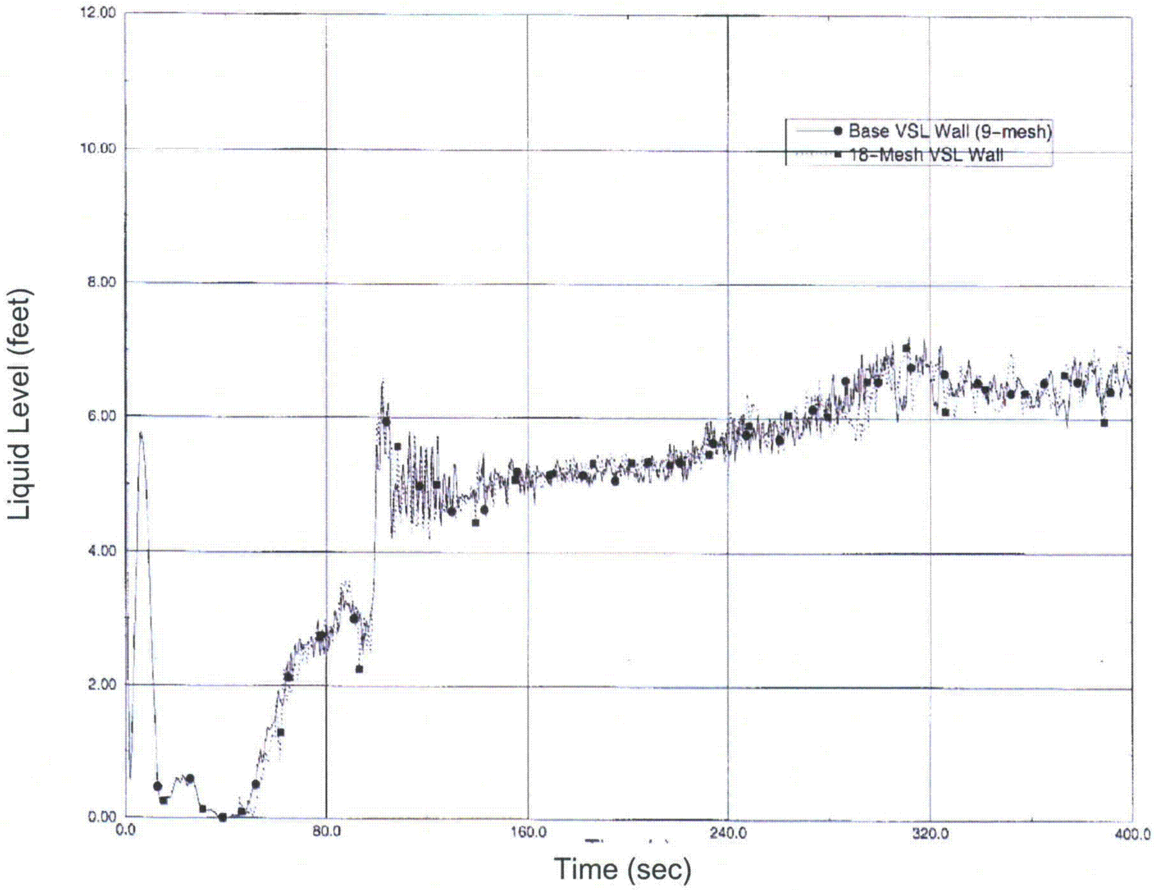

Figure 4-8 Core Liquid Level – Wall Mesh Point Sensitivity

4.5.2 Downcomer Fluid Distribution

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

4.5.2.1 Azimuthal Nodalization

In a letter to the NRC dated April 2003 (Reference 1), AREVA documented several studies on downcomer boiling. Of significance here is the study on further azimuthal break up of the downcomer noding. The study, based on a 3-loop plant with a containment pressure of approximately 30 psia during reflood, consisted of several calculations examining the 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 4-9). The base calculation simulated the limiting PCT calculation given in the EMF-2103 three-loop sample problem. This case was then repeated with the revised 6 x 9 downcomer noding.

The change resulted in an alteration of the blowdown evolution of the transient with little evidence of any 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.

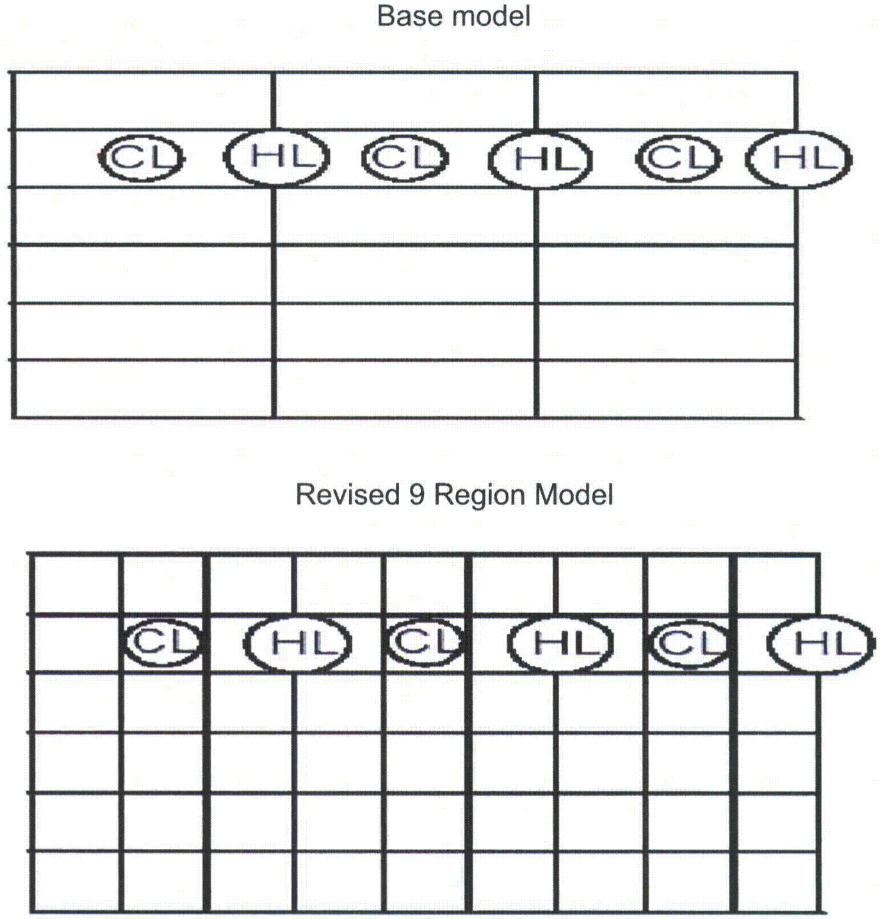

Figure 4-9 Azimuthal Noding

4.5.2.2 Axial Nodalization

The RLBLOCA methodology divides the downcomer into six nodes axially. In both 3-loop and 4-loop models, the downcomer segment at the active core elevation is represented by two equal length nodes. For most operating plants, the active core length is 12 feet and the downcomer segments at the active core elevation are each 6-feet high. (For a 14 foot core, these nodes would be 7-feet high.) The model for the sensitivity study presented here comprises a 4-loop plant with 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 pressure characteristics of ice condenser containments. Figure 4-10 provides the containment back pressure for the base modeling. Figures 4-11 through 4-14 show the total downcomer metal heat release rate, PCT independent of elevation, downcomer liquid level, and the core liquid level, respectively, for the base case and the modified case.

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

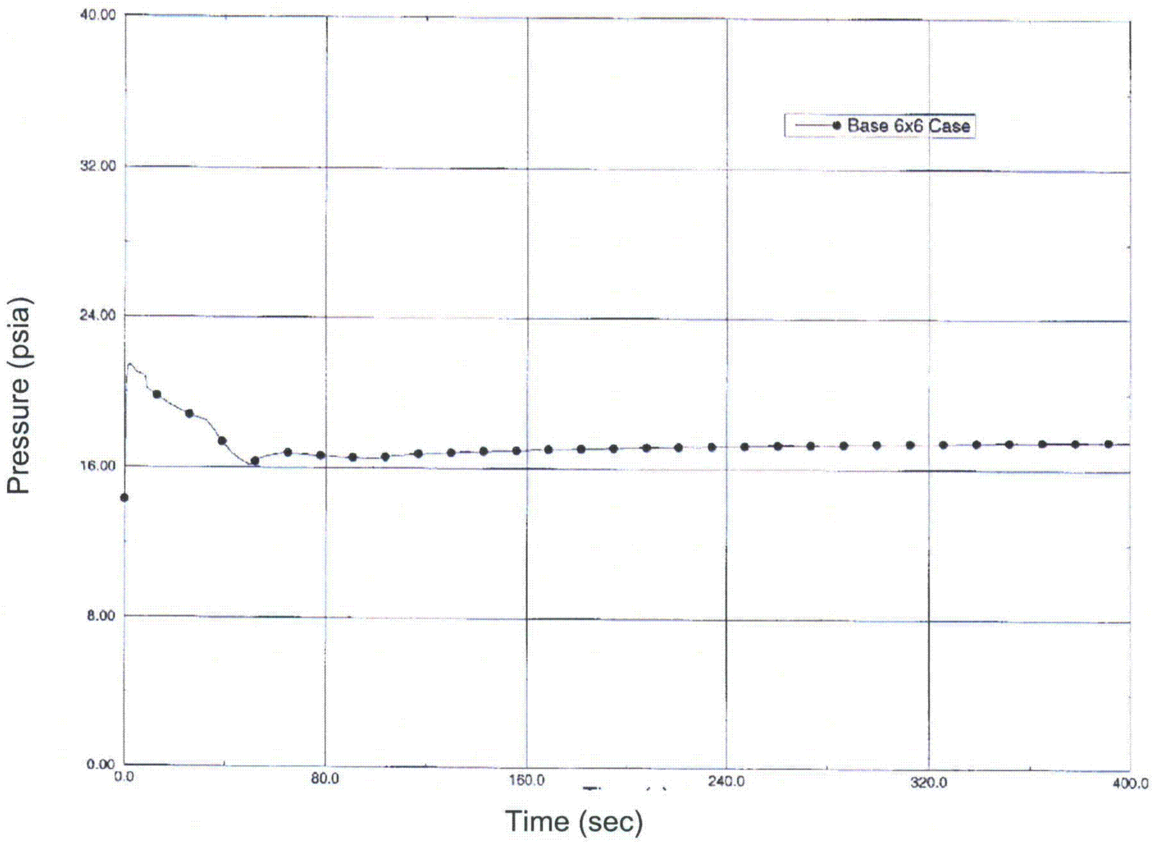

Figure 4-10 Lower Compartment Pressure versus Time

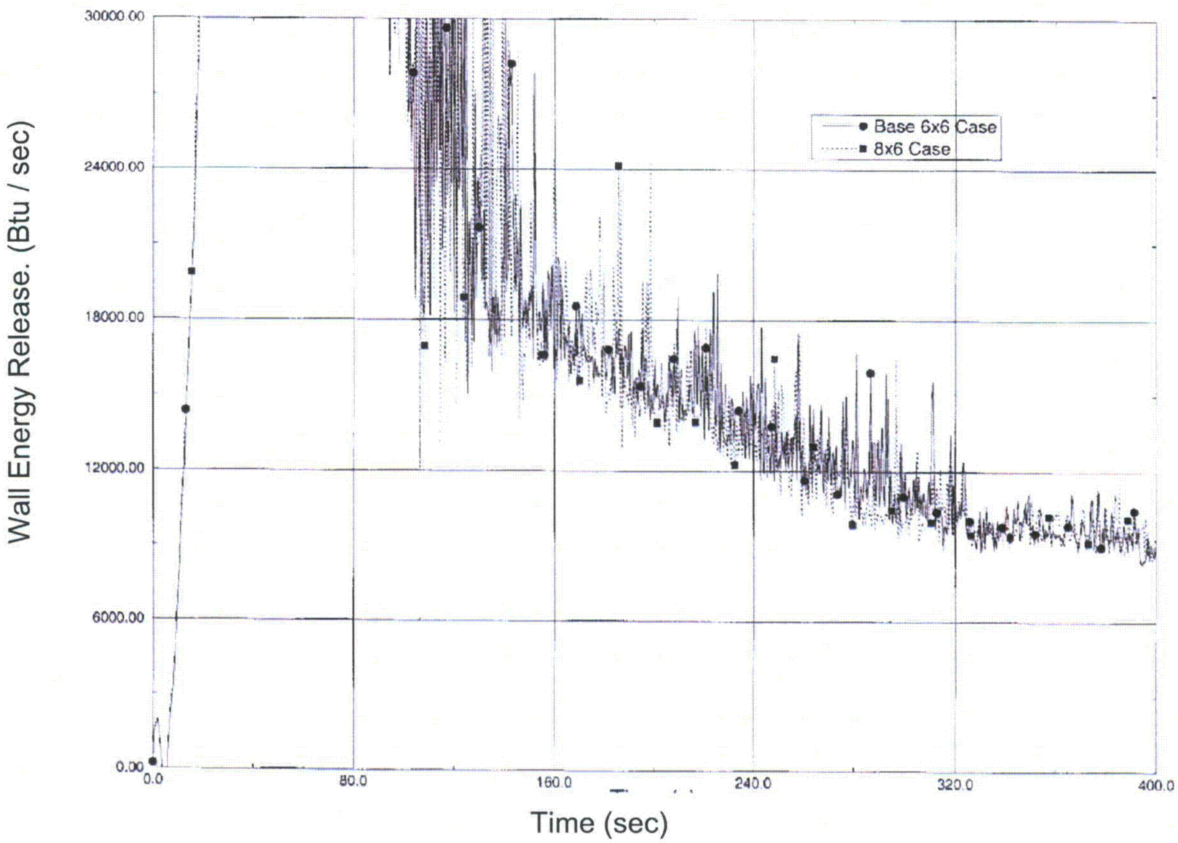

Figure 4-11 Downcomer Wall Heat Release – Axial Noding Sensitivity Study

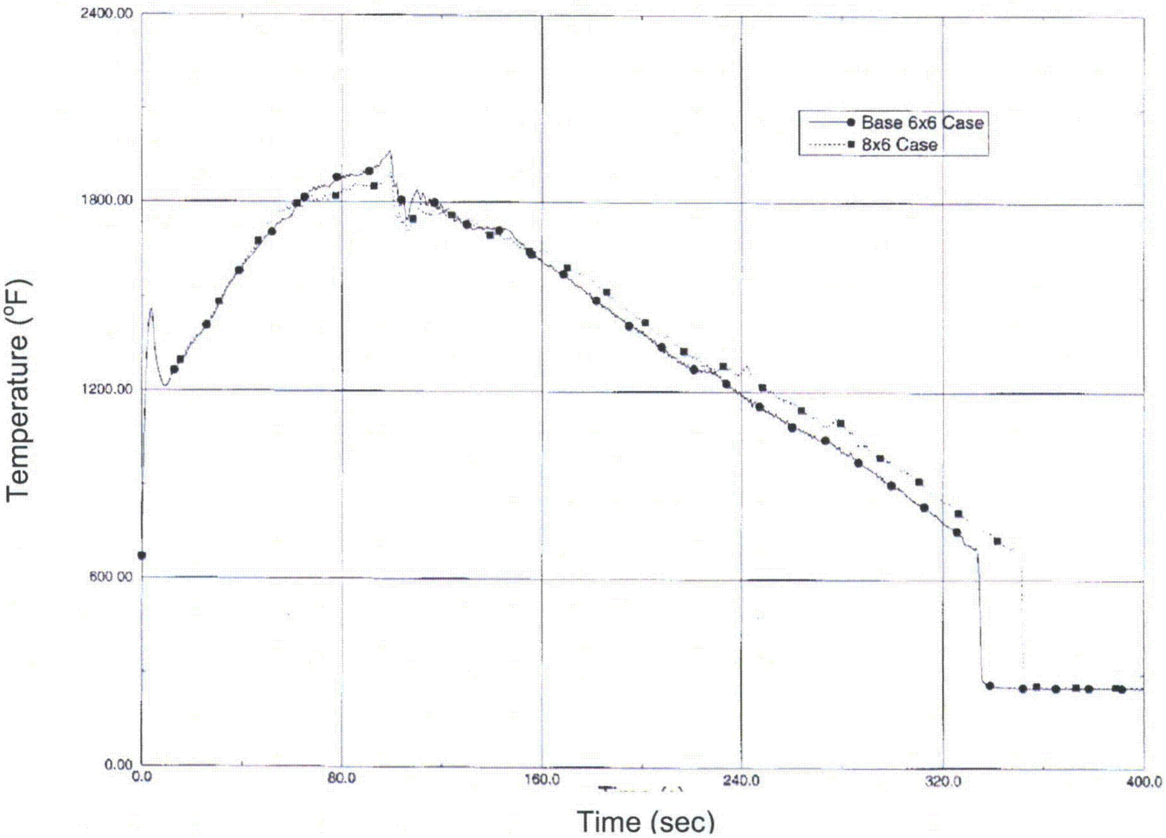

Figure 4-12 PCT Independent of Elevation – Axial Noding Sensitivity Study

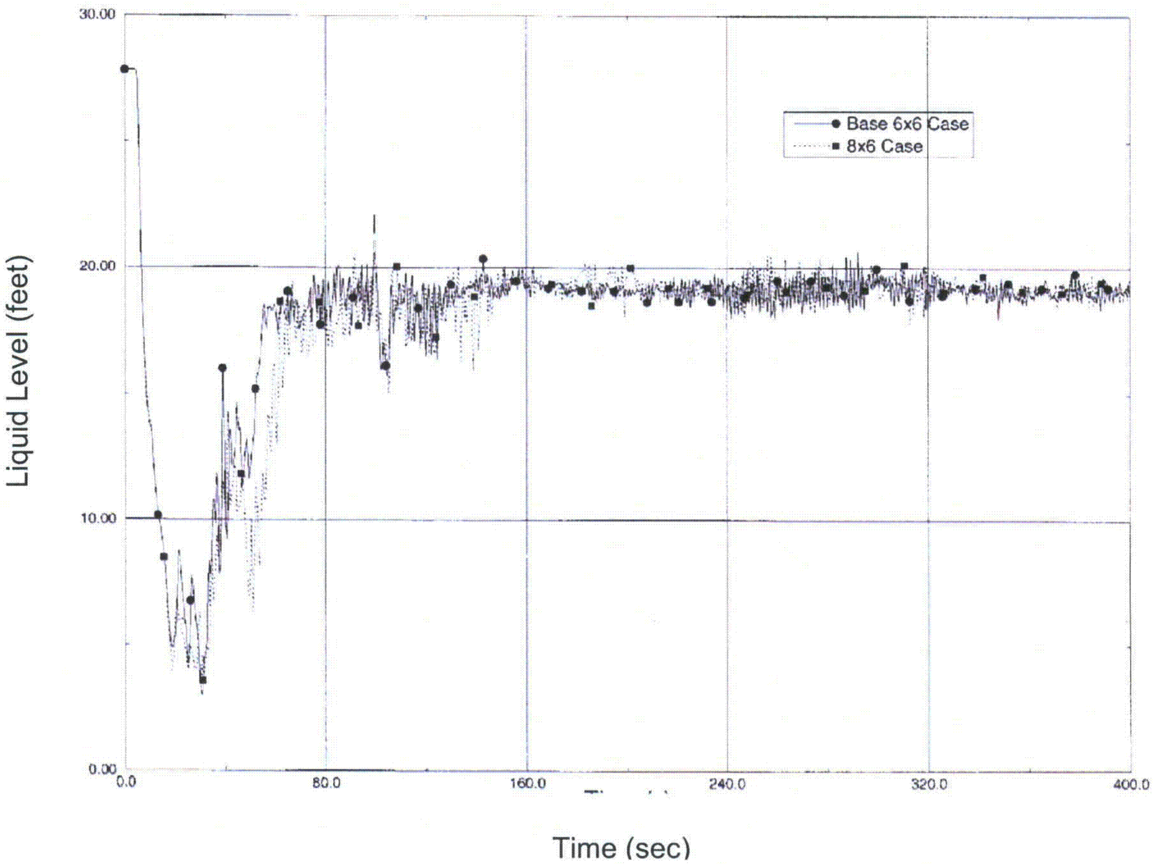

Figure 4-13 Downcomer Liquid Level – Axial Noding Sensitivity Study

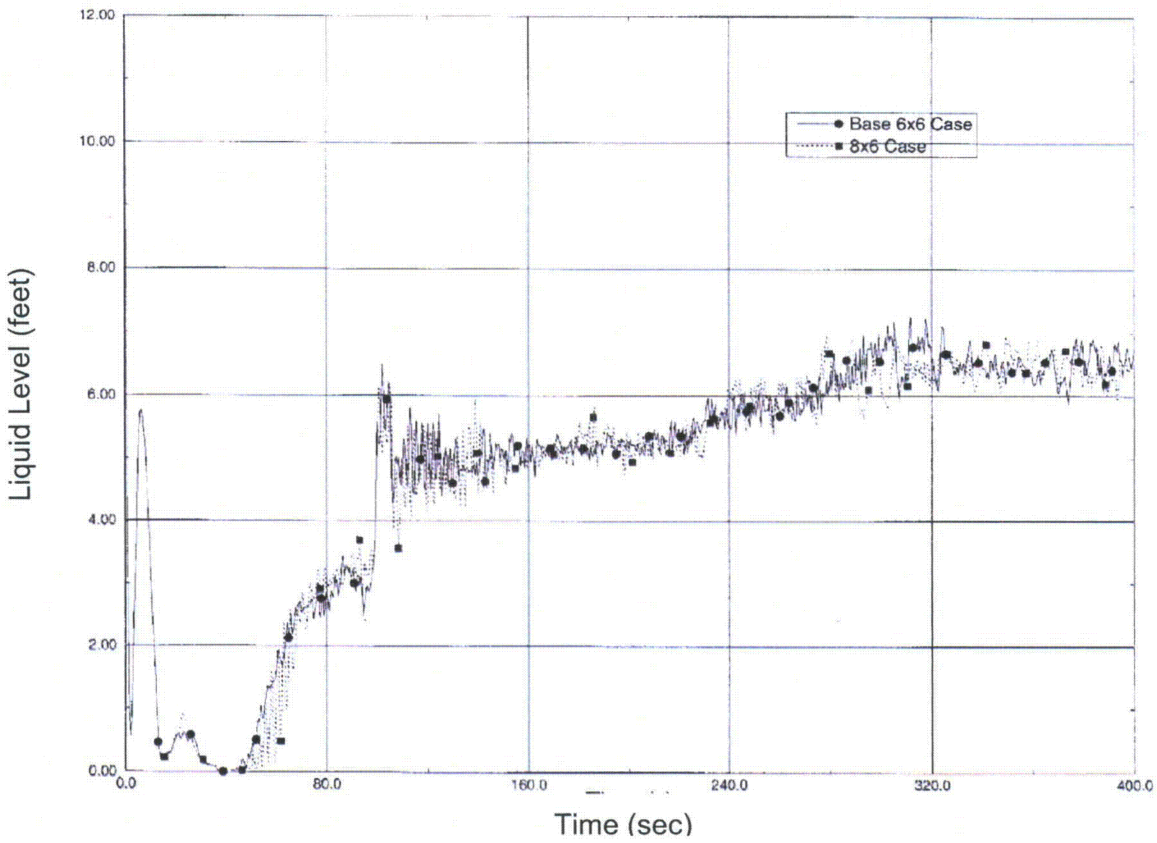

Figure 4-14 Core Liquid Level – Axial Noding Sensitivity Study

4.5.3 Downcomer Boiling Conclusions

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

4.6 Break Size

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

4.6.1 Break / Transient Phenomena

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

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

Refill is that period that starts with the end of blowdown, whichever definition is used, and ends when water is first forced upward 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 refilling of the core with water and ends with the achievement of complete core quench.

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

4.6.2 New Minimum Break Size Determination

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

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

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 ft^2 and the break area percentage, based on the full double-ended guillotine break total area, is 33-percent.

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

	Plant Description	System Pressure (psia)	Cold Leg Enthalpy (Btu/lbm)	Subcooled G_{break} (lbm/ft^2-s)	Saturated G_{break} (HEM) (lbm/ft^2-s)	No. of RCPs	RCP flow (lbm/s)	Spectrum Minimum Break Area (ft^2)	Spectrum Minimum Break Area (DEGB)
A	3-Loop W Design	2250	554.0	22198	6330	3	31558	2.21	0.27
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	2060	531.0	22068	5694	4	38277	2.76	0.28
F	2x4 CE Design	2250	544	22930	5834	4	41230	2.87	0.29
G	2x4 CE Design	2250	548	22637	6091	4	42847	2.98	0.30
H	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.

4.6.3 Intermediate Break Size Disposition

With the revision of the smaller break area for the RLBLOCA analysis, the break range for small breaks and large breaks are no longer contiguous. Typically the lower end of the large break spectrum occurs at between 0.2 to 0.3 times the total area of a 100-percent double-ended guillotine break (DEGB) and the upper end of the small break spectrum occurs at 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.

A variety of plant types for which analysis within the intermediate range have been completed were surveyed. Although statistical determinations are extracted from the consideration of breaks with areas above the intermediate range, the AREVA best-estimate methodology remains suitable to characterize the ECCS performance of breaks within the intermediate range. Table 4-4 provides a listing of the plant type, initial condition, and the fractional minimum RLBLOCA break area. Figures 4-15 through 4-20 provide the enlarged break spectrum results with the upper end of the small break spectrum and the lower end of the large break spectrum indicated by bars.

Table 4-5 provides differences between the true large break region and the intermediate break region (break areas between that of the largest SBLOCA and the smallest RLBLOCA). The minimum difference is 222 °F; however, this case is not representative of the general trend shown by the other comparisons. 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 eight plants, as at least 463 °F, and including this point would provide an average difference of 640 °F for the CE 2x4 design plants and a maximum difference of 840 °F for the 4-loop W plant design.

Thus, by both measures, the peak cladding temperatures within the intermediate break range will be several hundred degrees below those in the true large break range. Therefore, these breaks will not provide a limit or a critical measure of the ECCS performance. Given that the large break spectrum bounds the intermediate spectrum, the use of only the large break spectrum meets the requirements of 10CFR50.46 for breaks within the intermediate break LOCA spectrum, and the method demonstrates that the ECCS for a plant meets the criteria of 10CFR50.46 with high probability.

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

Plant Description	Generic Plant Label (Table 4-4)	Maximum PCT (°F) Intermediate Size Break	Maximum PCT (°F) Large Size Break	Delta PCT (°F)	Average Delta PCT (°F)
3-Loop W Design	A	1206	1930[25]	724	622
	B	1273	1951	678	
	C	1326	1789	463	
2x4 CE Design	D	984	1751	767	640
	E	1049	1740	691	
	F	791	1670	879	
	G	1464	1686	222[26]	
4-Loop W Design	H	1127	1967	840	840

[25] The analysis for this W 3-Loop plant was performed with the Transition methodology and no break sizes fell into the intermediate break range. The PCT value of 1206 °F is the closest point to the maximum end of the intermediate break spectrum.

[26] The analysis for this 2x4 CE plant was performed with the Transition methodology and no break sizes fell into the intermediate break range. The PCT value of 1464 °F is the closest point to the maximum end of the intermediate break spectrum. From the trends of the other 2x4 CE analyses, breaks falling within the intermediate break spectrum would be significantly lower.

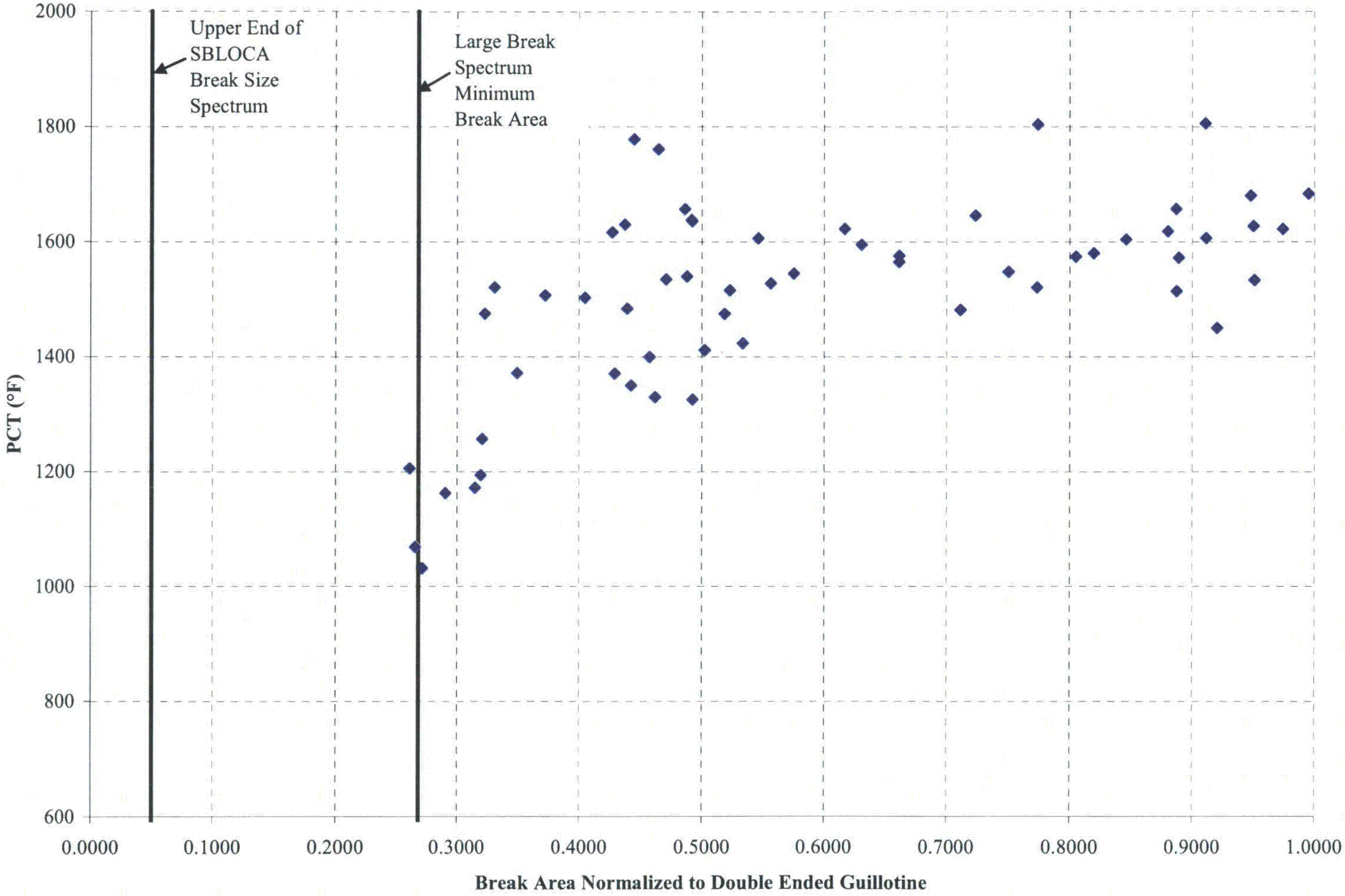

Figure 4-15 Plant A – Westinghouse 3-Loop Design

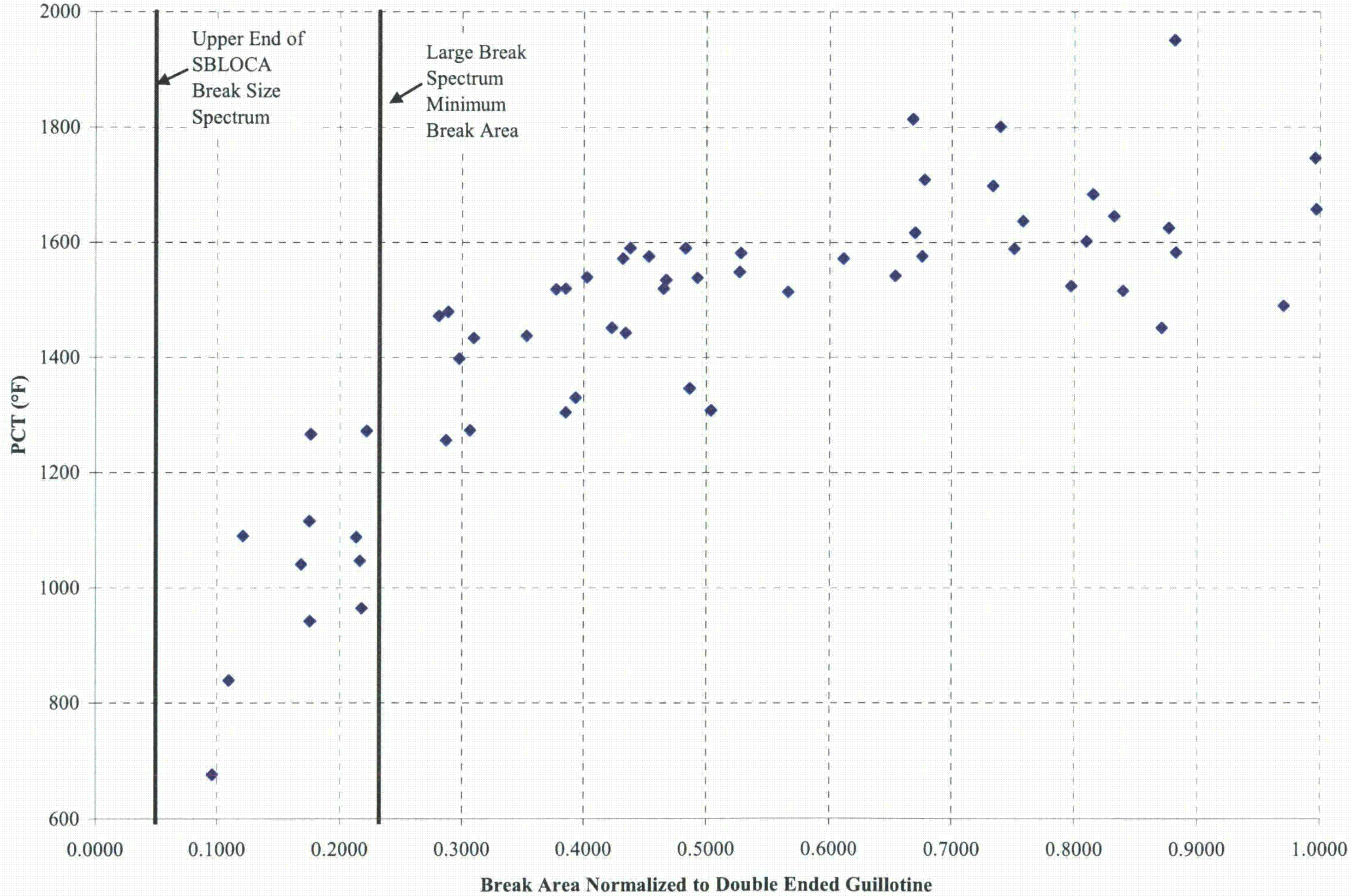

Figure 4-16 Plant B – Westinghouse 3-Loop Design

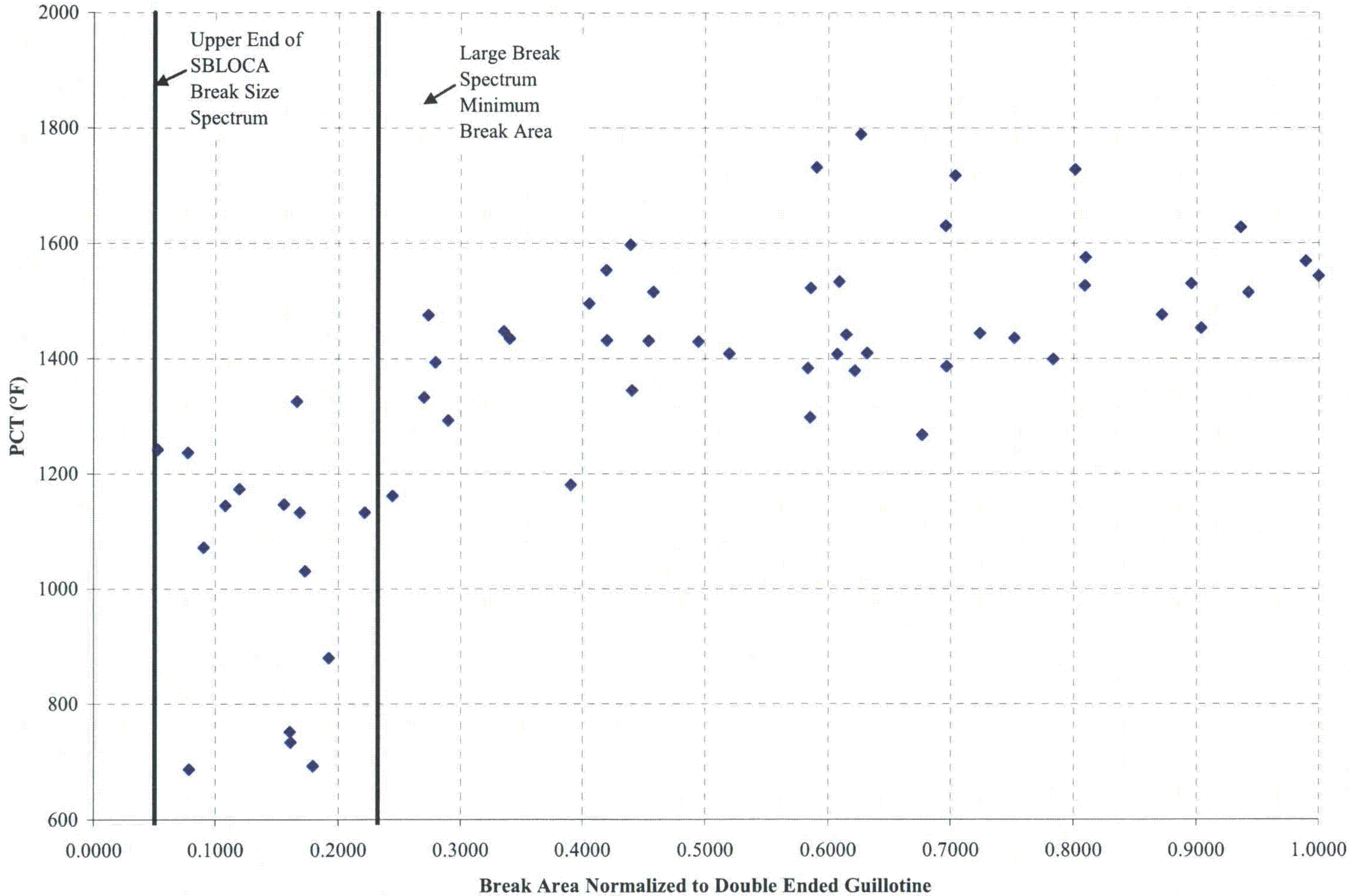

Figure 4-17 Plant C – Westinghouse 3-Loop Design

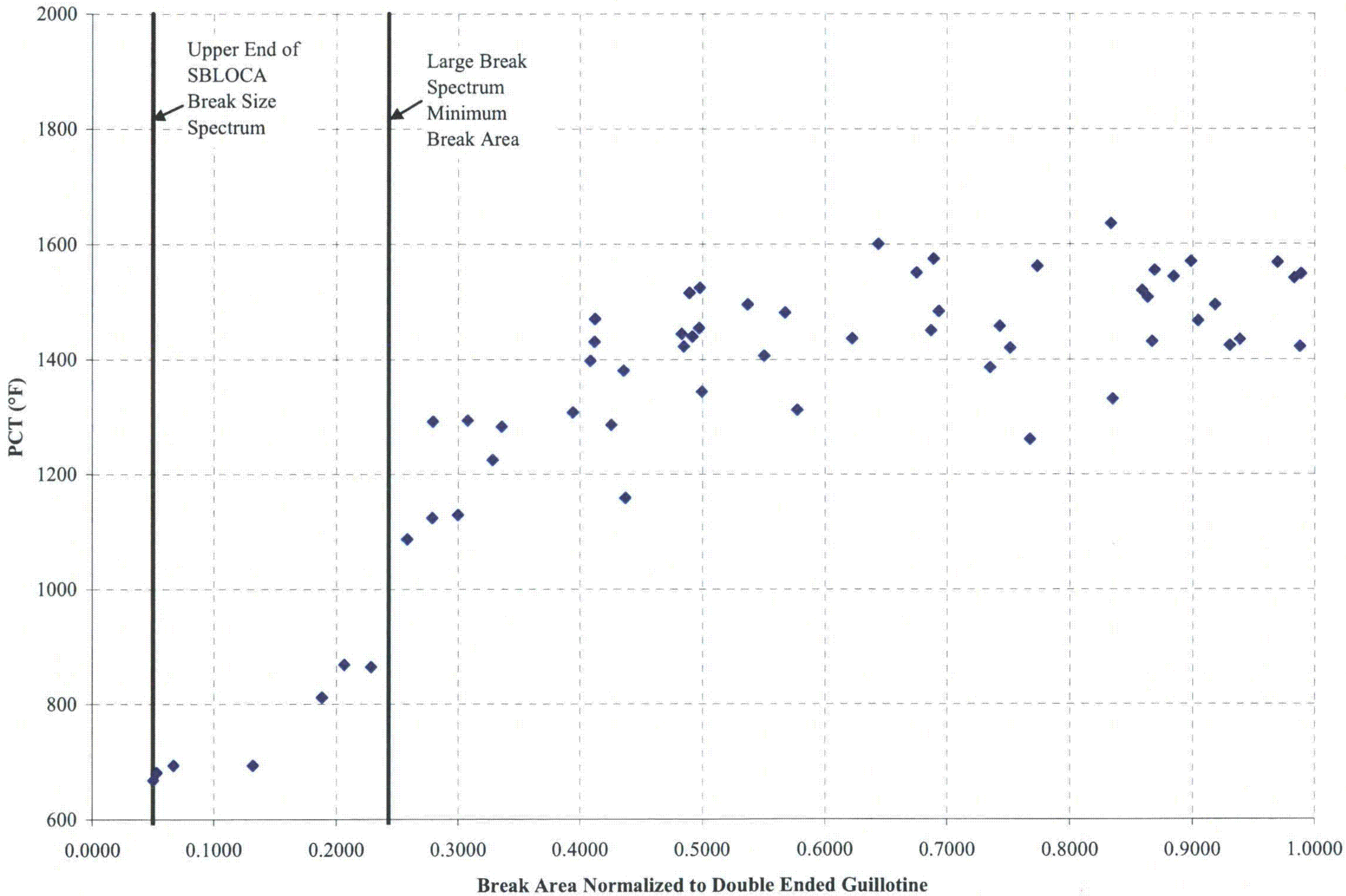

Figure 4-18 Plant D – Combustion Engineering 2x4 Design

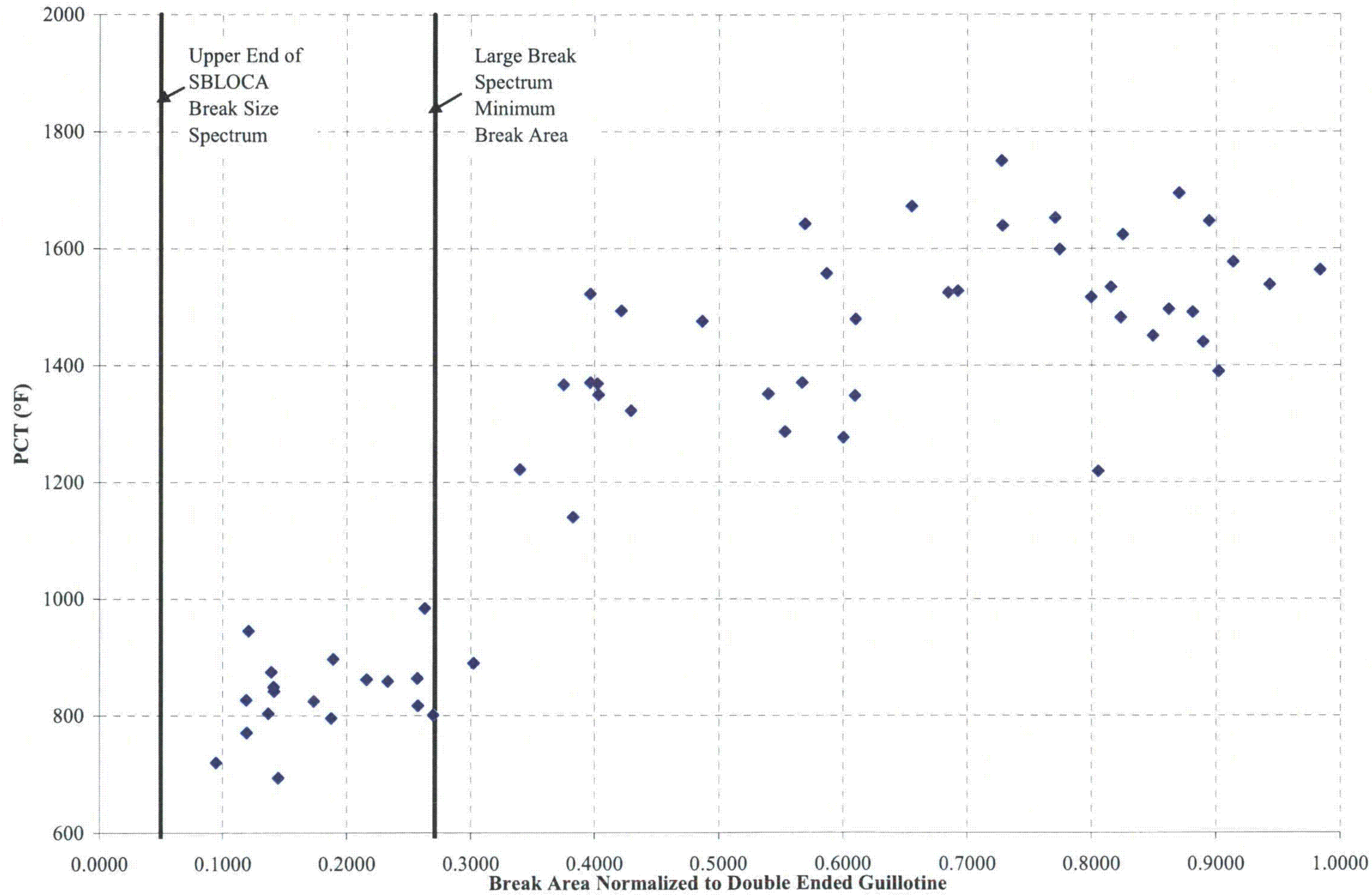

Figure 4-19 Plant E – Combustion Engineering 2x4 Design

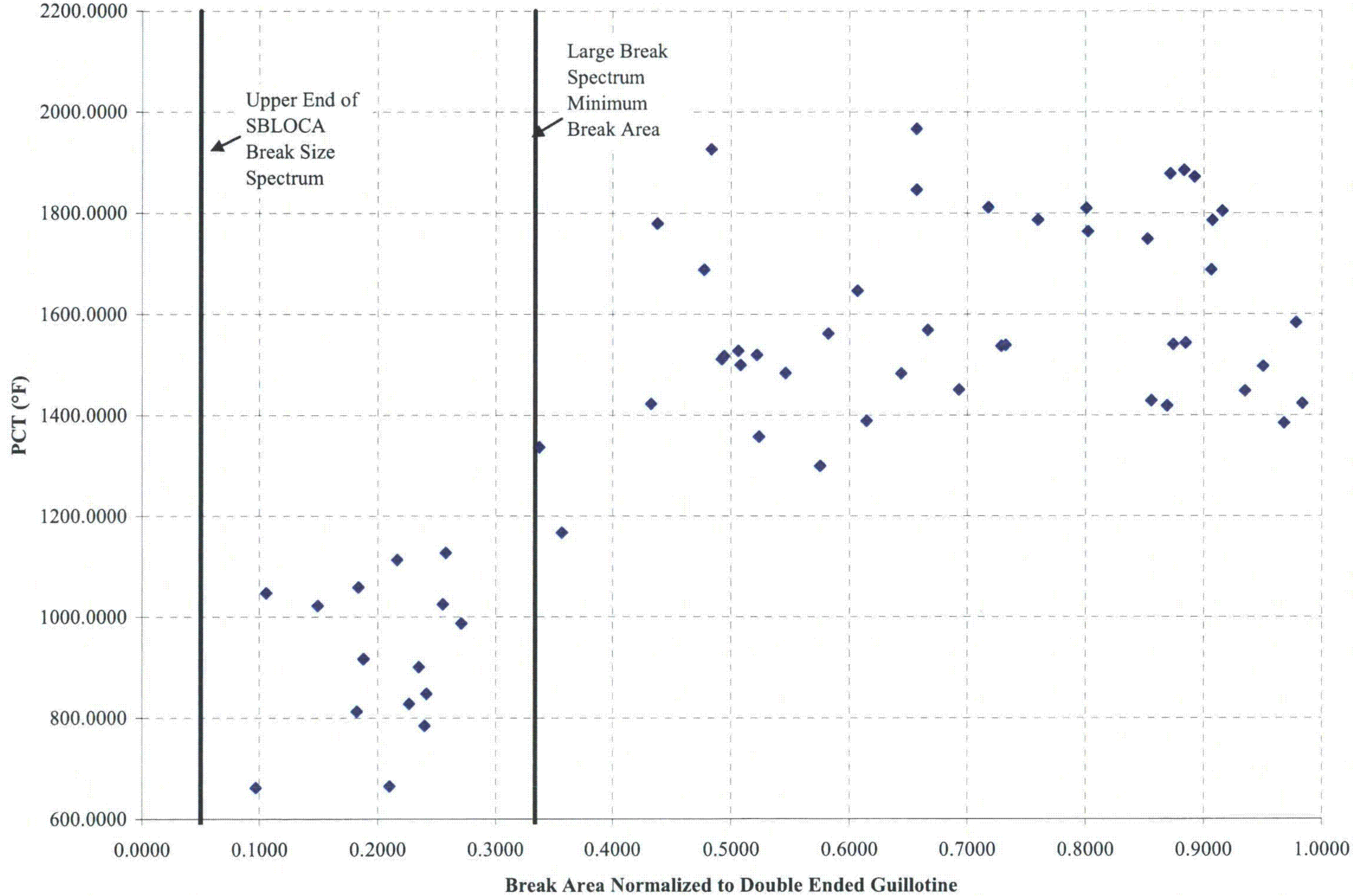

Figure 4-20 Plant H – Westinghouse 4-Loop Design

4.7 Detail information for Containment Model (ICECON)

Question: *Verify that the 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)."*

The AREVA RLBLOCA Report 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-2970(P) states that "in many instances" the guidance of NRC Branch Technical Position CSB 6-2 was used in determining the other containment parameters.

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

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

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

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

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

See Section 3.3 for discussion of questions (a) through (e). Containment initial conditions and cooling system information are provided in Table 3-8 and Heat Sinks are provided in Table 3-9. For Sequoyah Units 1 and 2, the scatter plots of PCT versus the sampled containment volumes and initial atmospheric temperature are shown in Figure 4-21 and Figure 4-22. Containment pressure as a function of time for limiting case is shown in Figure 4-23. Figures 4-24 through 4-32 are provided to supplement the NRC's review of the Sequoyah Units 1 and 2 RLBLOCA analysis.

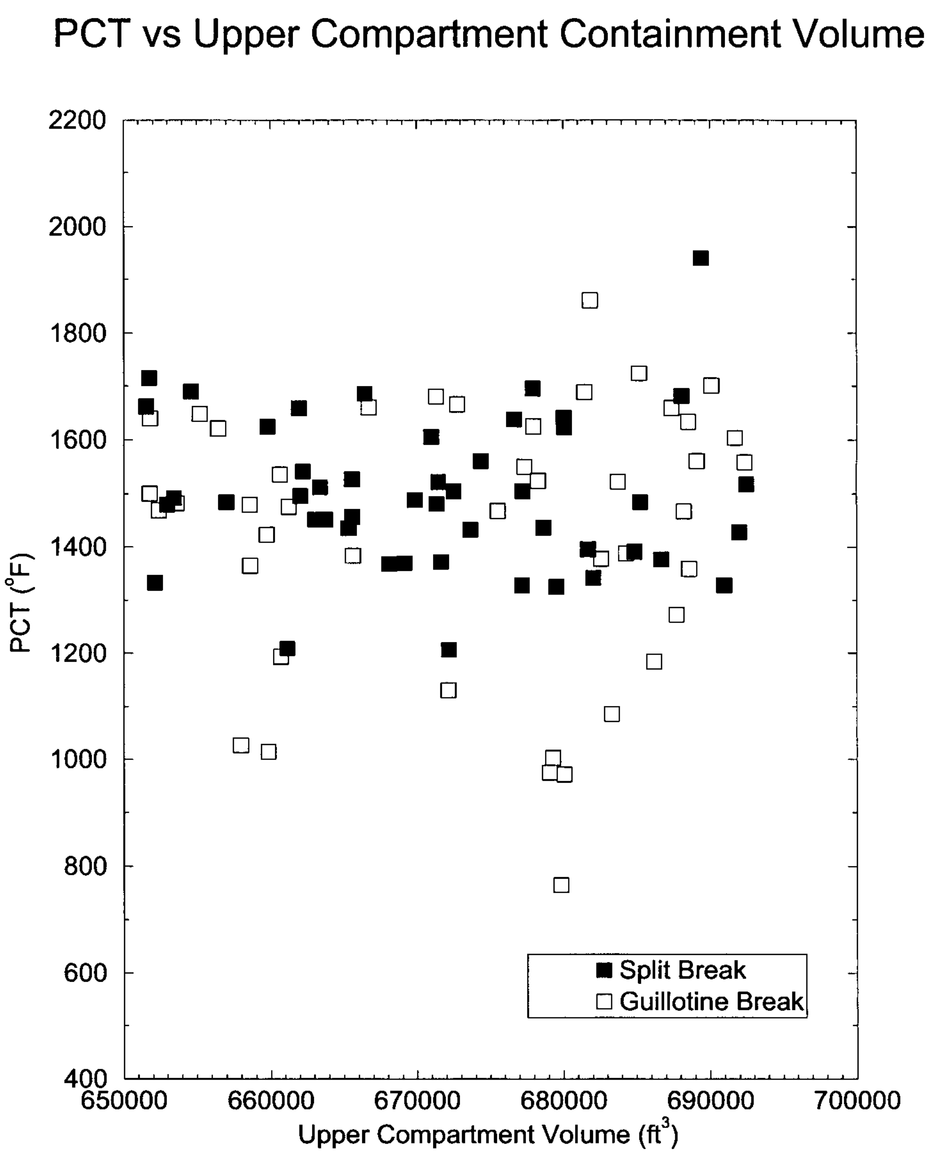

Figure 4-21 PCT vs. Containment Volume

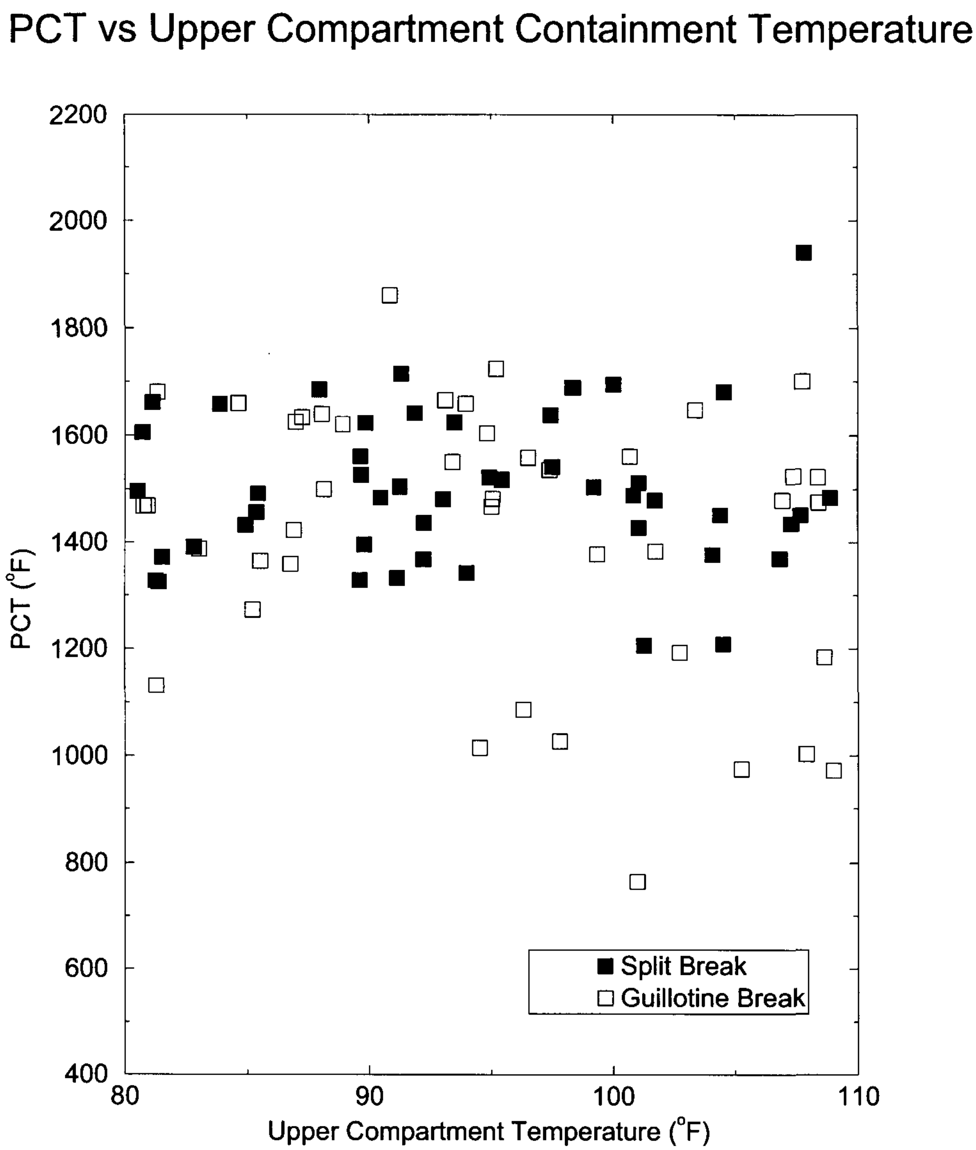

Figure 4-22 PCT vs. Initial Containment Temperature

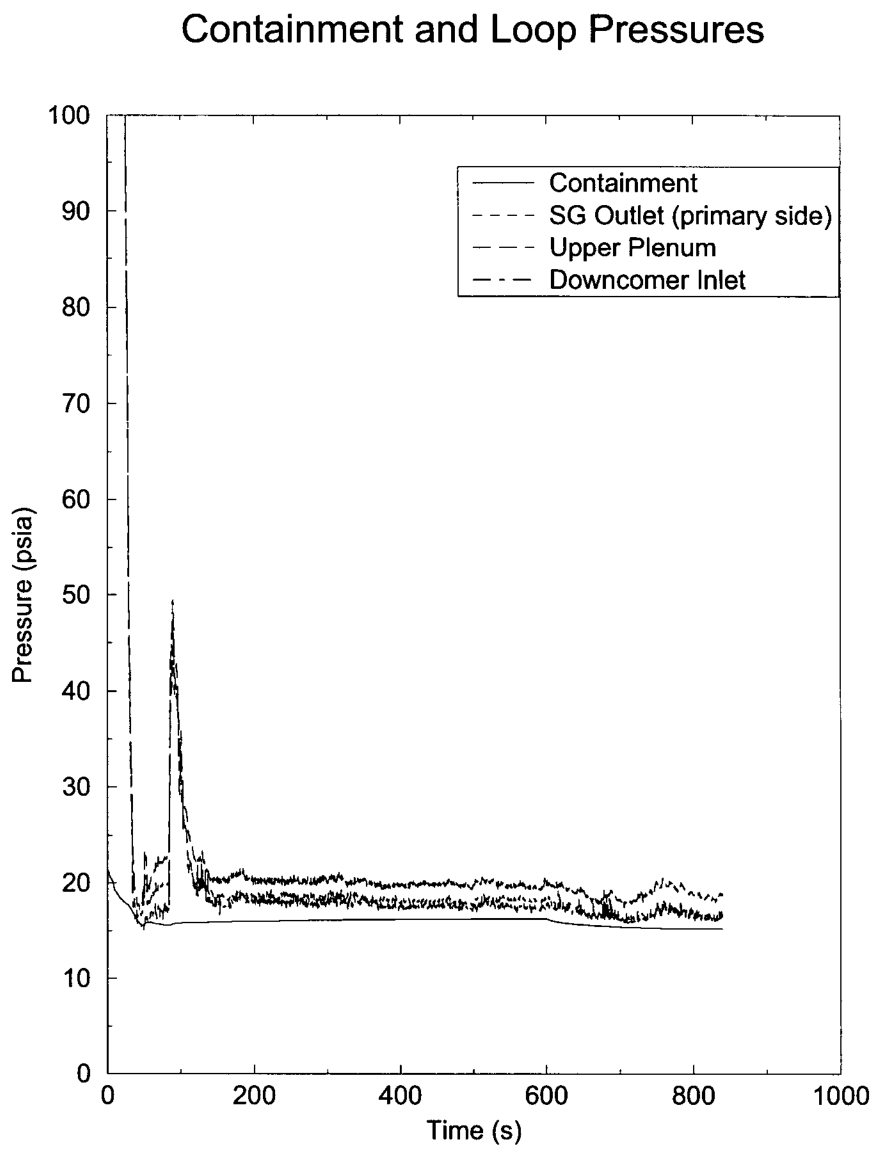

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Figure 4-23 Containment Pressure for Limiting Case

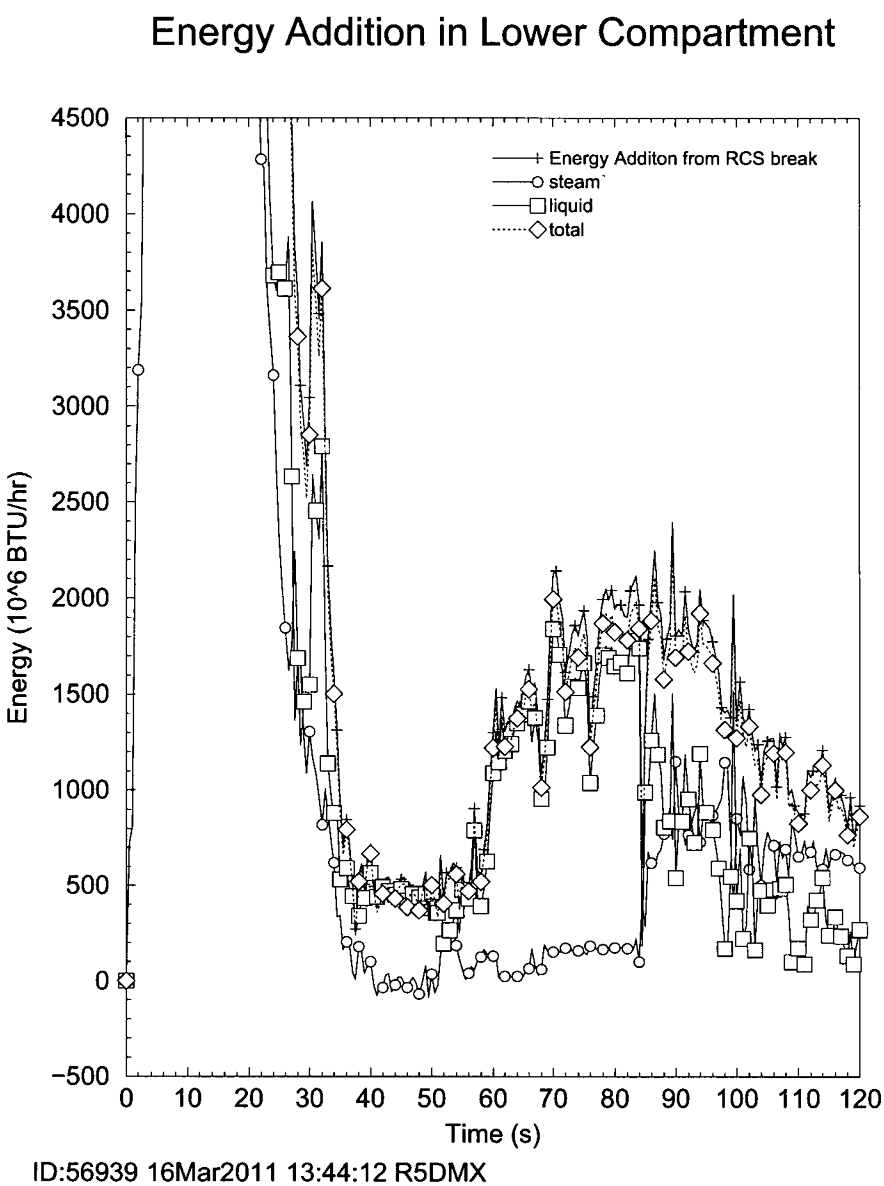

Figure 4-24 Energy Addition in Lower Compartment

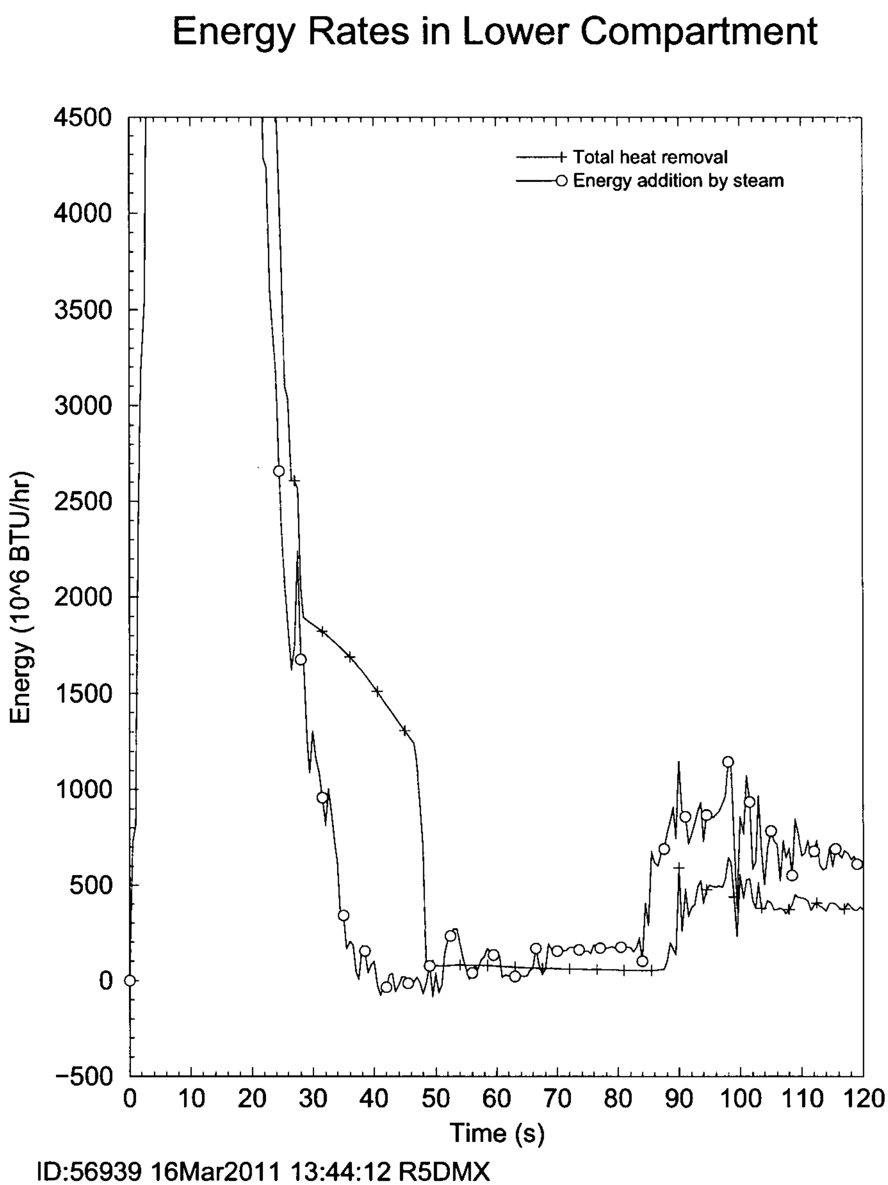

Figure 4-25 Energy Rates in Lower Compartment

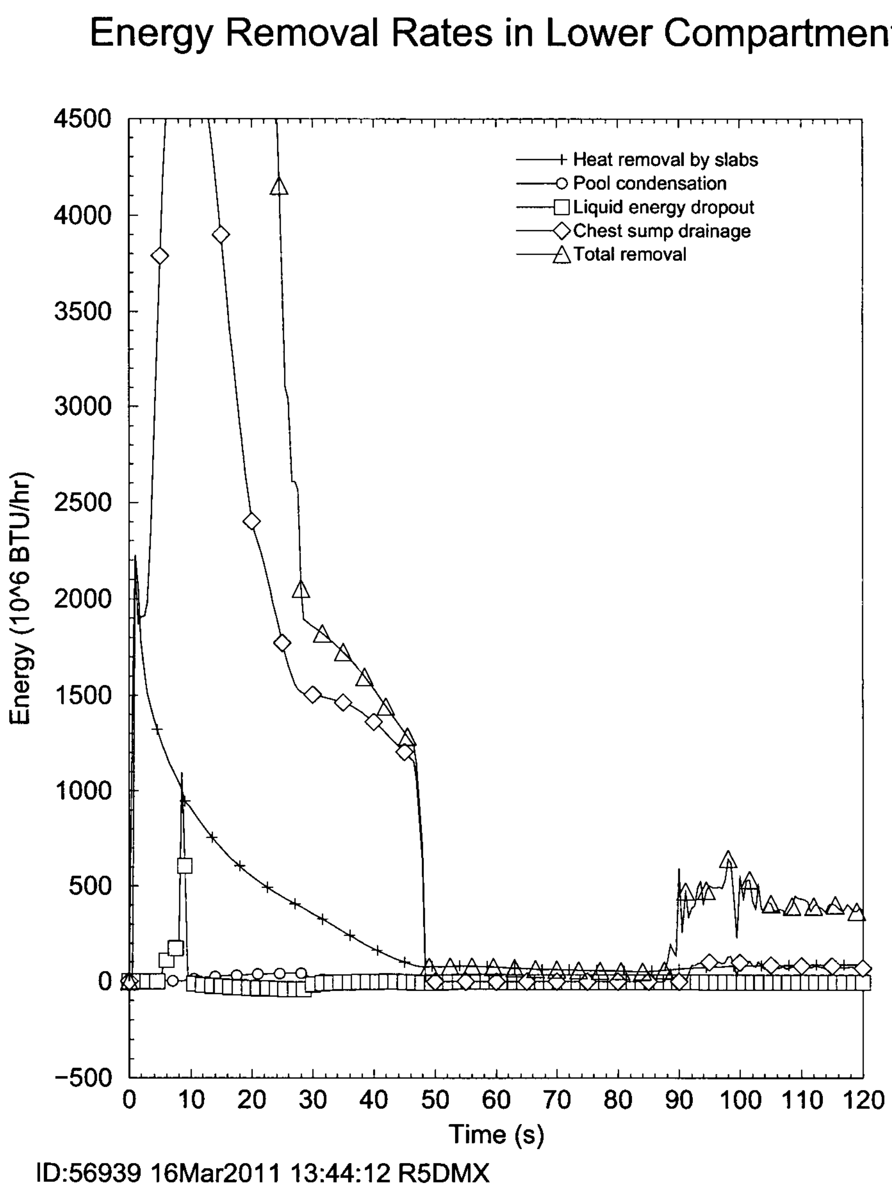

Figure 4-26 Energy Removal Rates in Lower Compartment

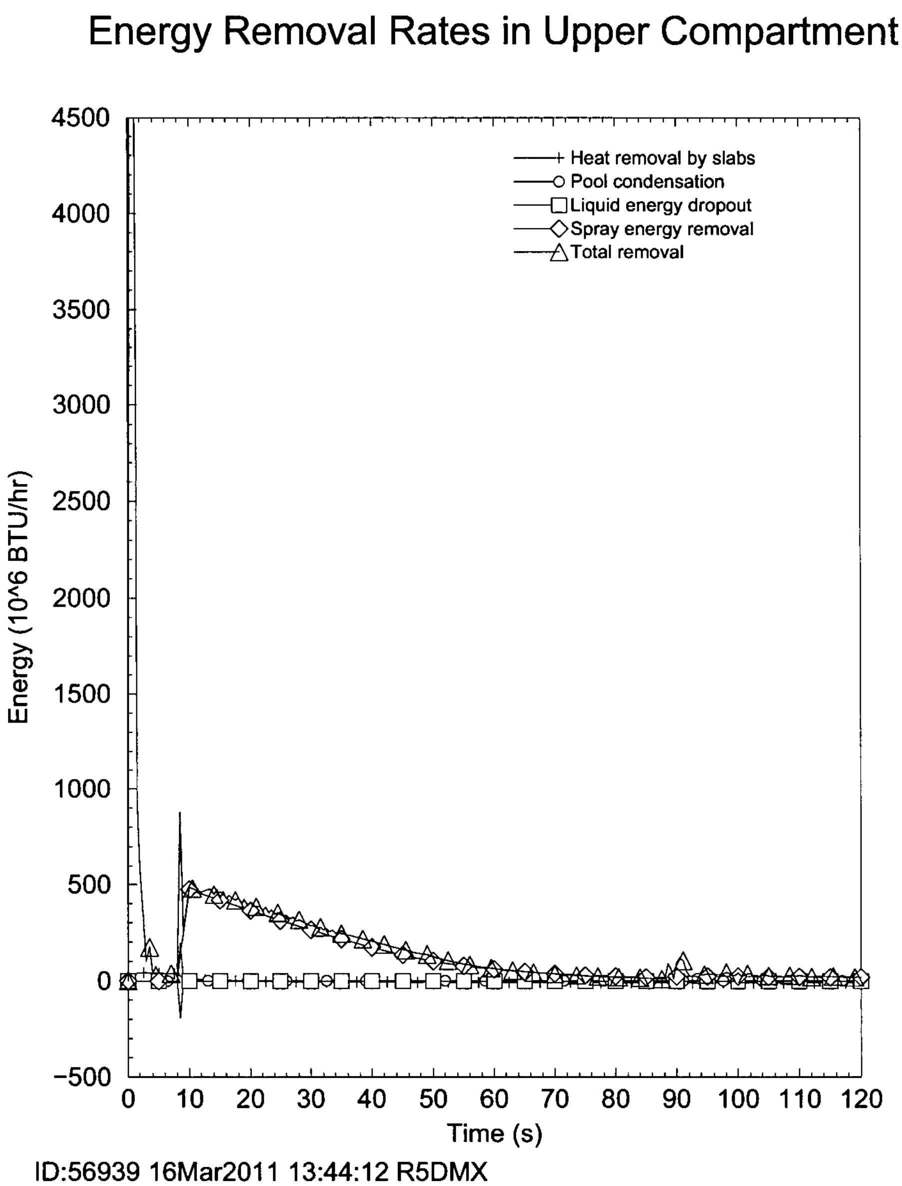

Figure 4-27 Energy Removal Rates in Upper Compartment

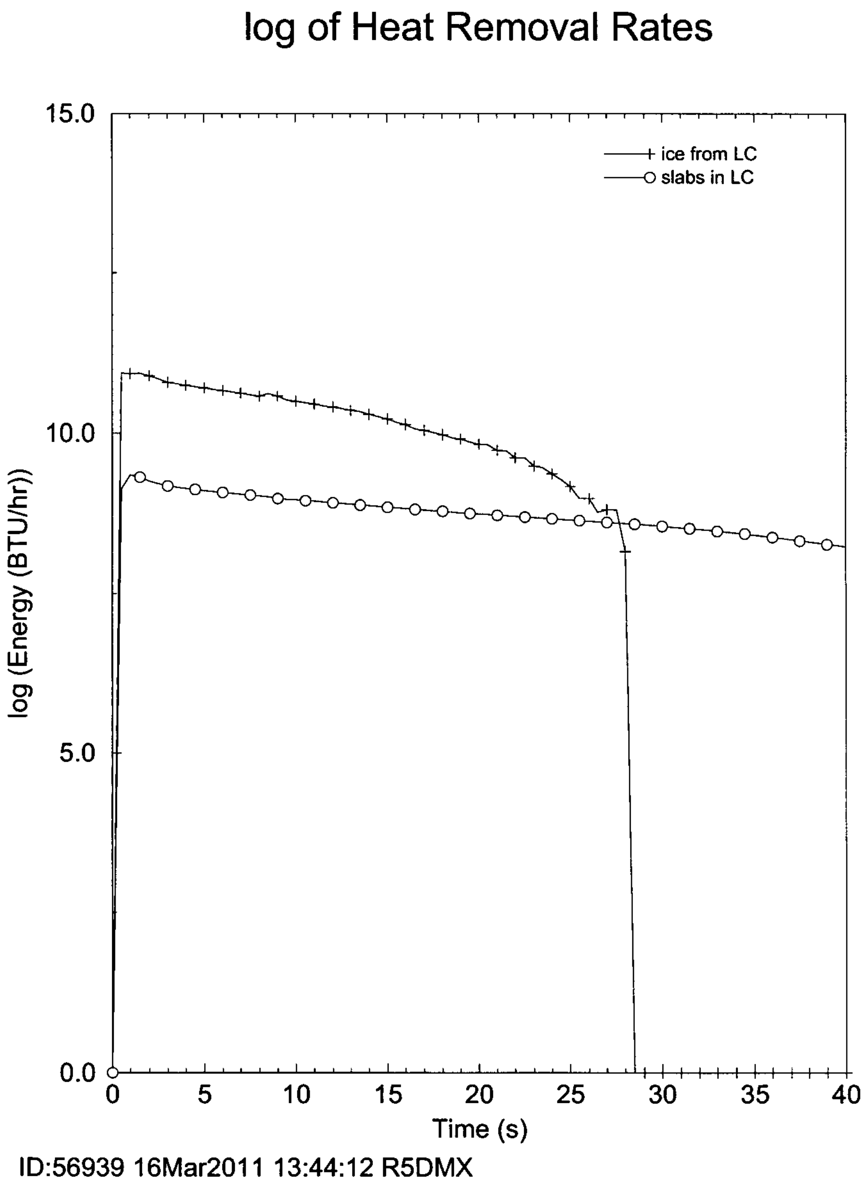

Figure 4-28 Heat Removal Rates (log)

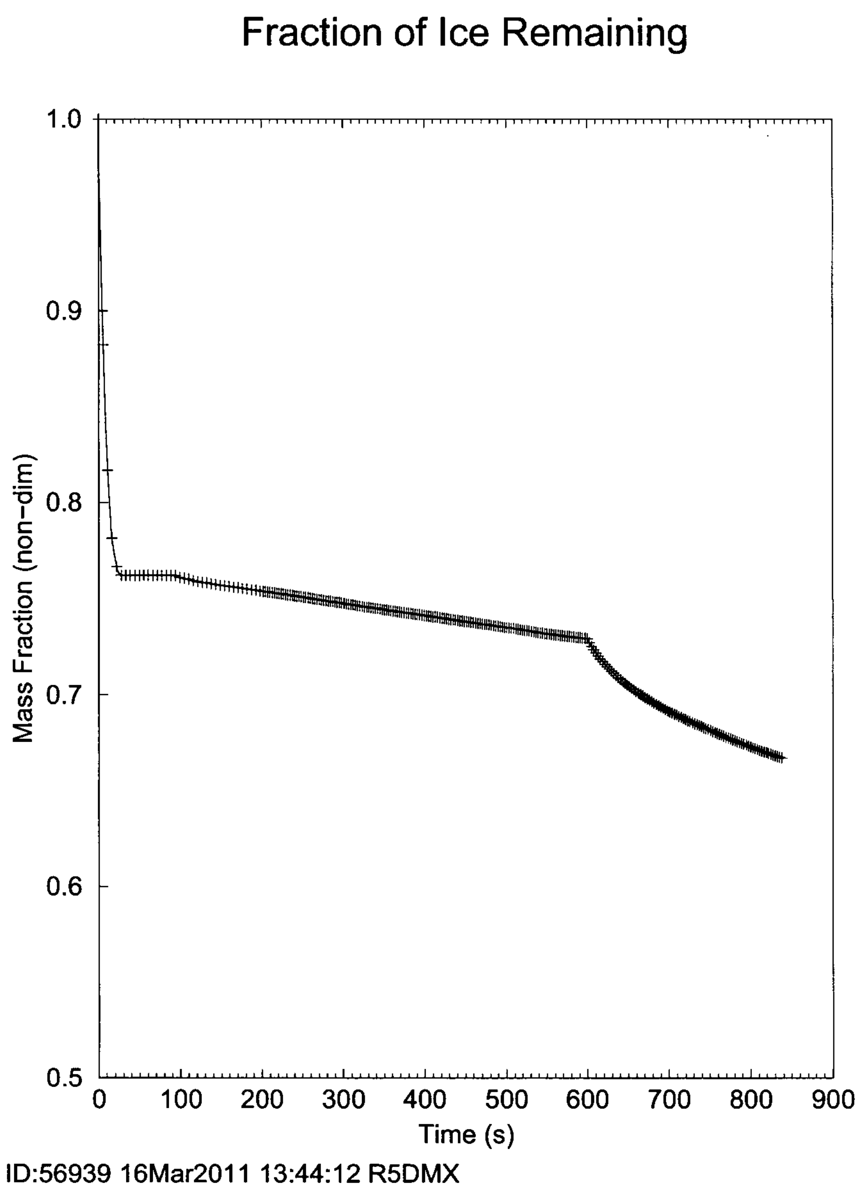

Figure 4-29 Fraction of Ice Remaining

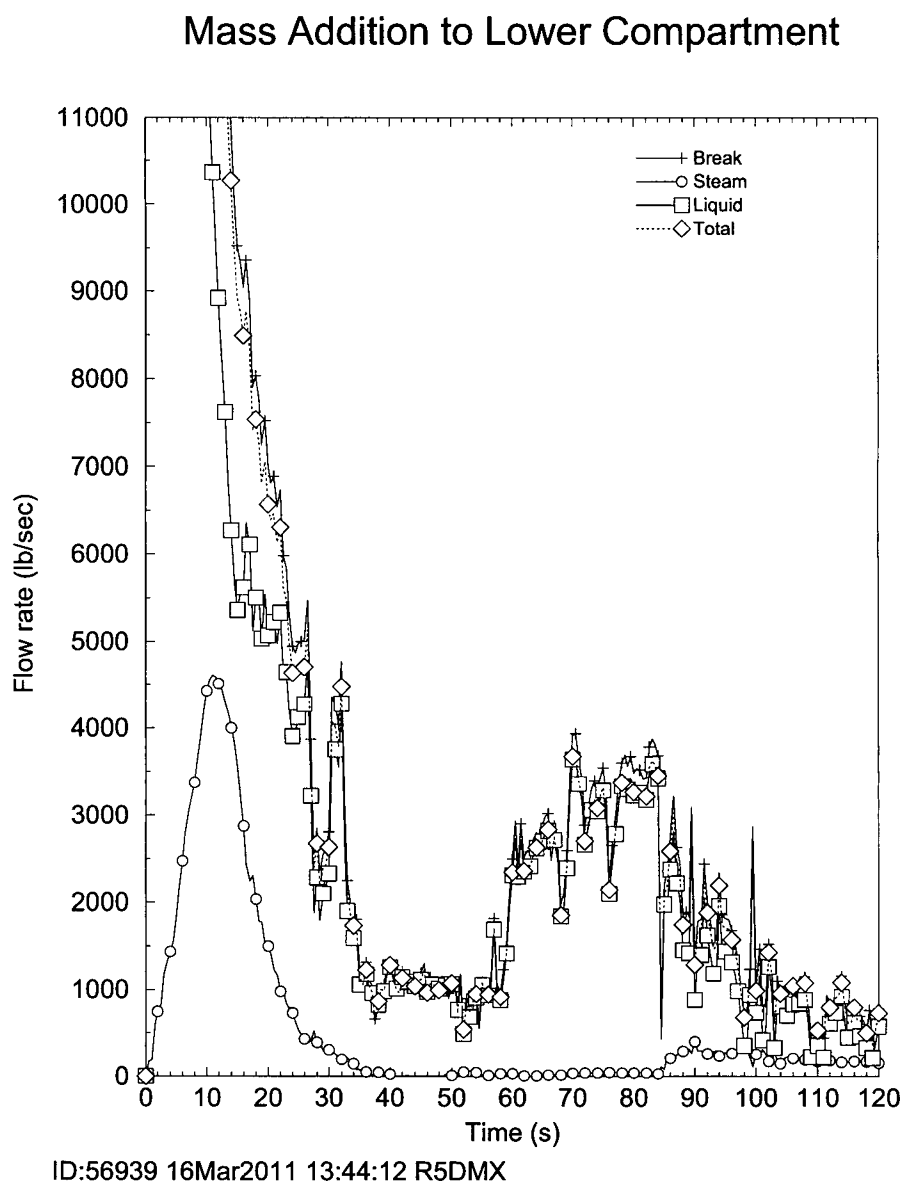

Figure 4-30 Mass Addition to Lower Compartment

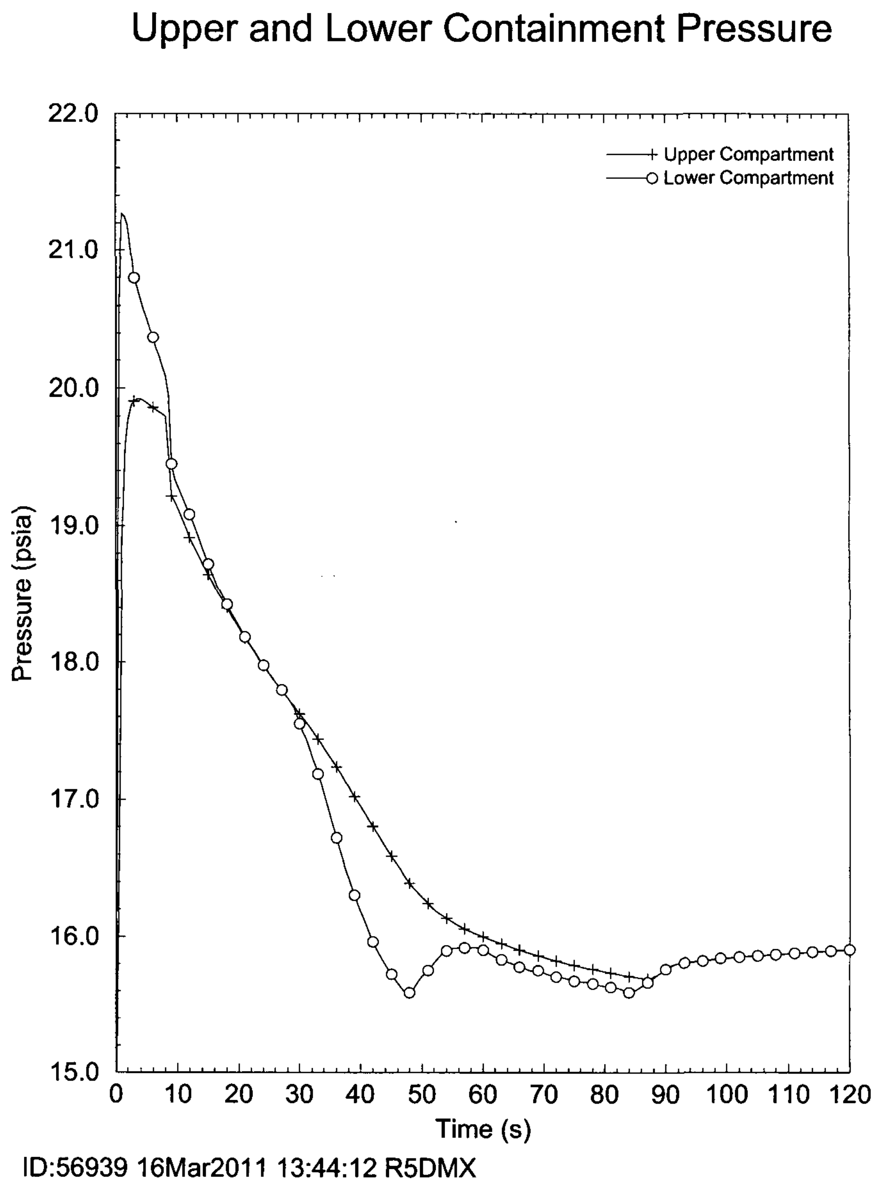

Figure 4-31 Upper Compartment and Lower Compartment Pressure

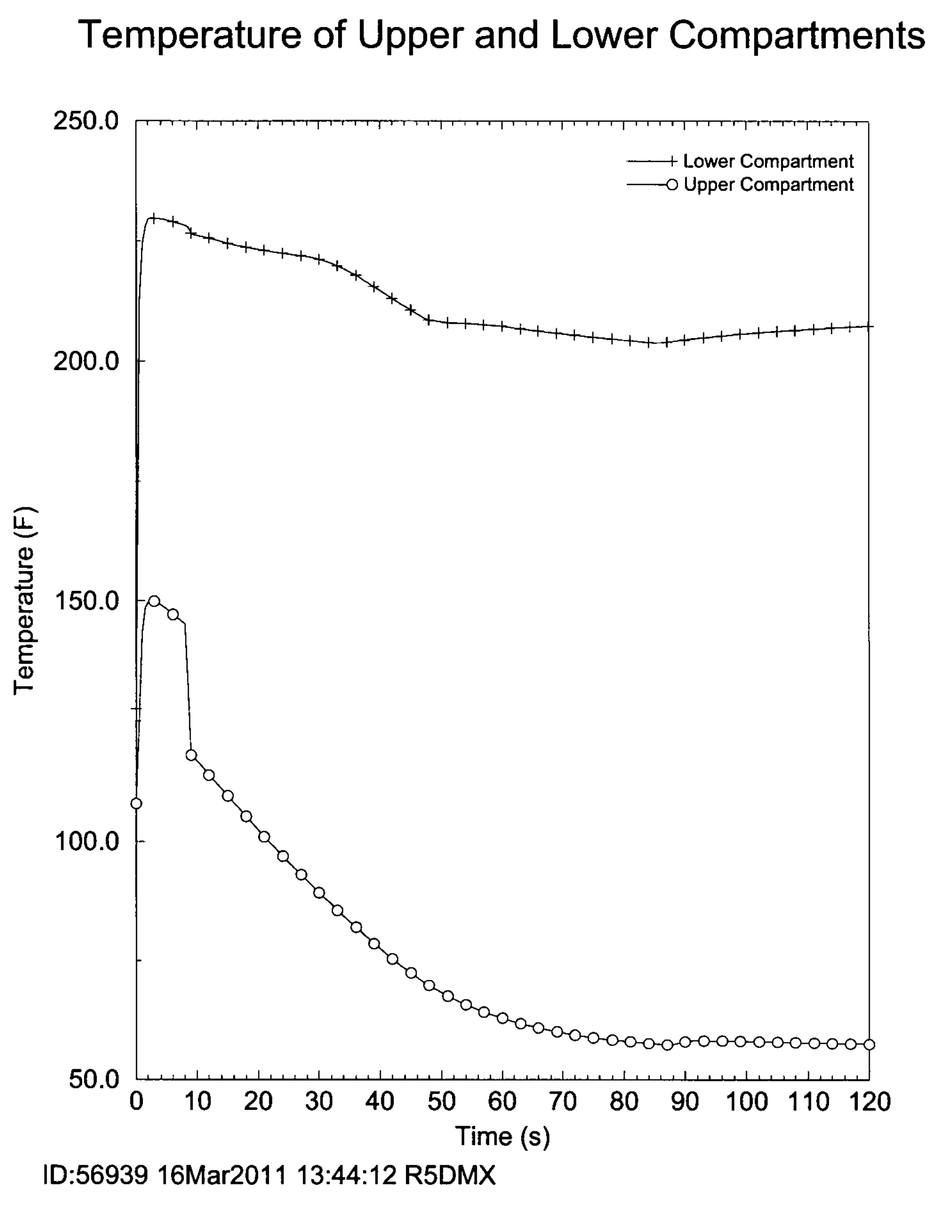

Figure 4-32 Temperature of Upper and Lower Compartments

4.8 Cross-References to North Anna

Question: *In order to conduct its review of the Sequoyah 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 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 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 Units 1 and 2.

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

4.9 GDC 35 – LOOP and No-LOOP Case Sets

In concurrence with the NRC's interpretation of GDC 35, a set of 93 cases each was run with a LOOP and No-LOOP assumption. The set of 93 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 3-24. This is a change to the approved RLBLOCA EM (Reference 1).

4.10 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 Sequoyah Units 1 and 2 LBLOCA analyses conservatively bound the values and ranges of those parameters for the as operated Sequoyah plants. 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 Units 1 and 2 realistic large break loss-of-coolant accident are verified as applicable 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;

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.

5.0 Conclusions

The results of the RLBLOCA analysis show that the limiting run (Case 86) had offsite power available and has a PCT of 1941 °F for a fresh UO_2 rod. The maximum oxidation thickness and hydrogen generation fall well within regulatory requirements.

The analysis supports operation at a 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 (F_Q) of 2.65 (including uncertainty) and a nuclear enthalpy rise factor ($F_{\Delta H}$) of 1.7056 (including uncertainty) with no axial dependent power peaking limit.

6.0 Recent NRC Request for Additional Information (RAI) and AREVA Responses

The NRC staff has found that strict adherence to currently referenced, or proposed for referencing AREVA methodologies are inconsistent with the NRC's requirements and review guidance without appropriate justification. This section addresses the NRC staff's concerns for the AREVA RLBLOCA methodology.

Question:

1. *Please provide more information about the management of the fuel thermal conductivity degradation issue identified in NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation." Specifically:*

   a. *Page 1-3, states, "For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady-state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then two-transformation adjustment to the base fuel centerline temperature is computed. The first transformation is a linear adjustment for an exposure of 10 MWd/MTU or higher. In the new process, a polynomial transformation is used in the first transformation instead of a linear transformation." Please clarify the following:*

      i. *Explain how the fuel pellet radial temperature profile is computed.*

      ii. *Explain which code is used to calculate this profile, both for initial conditions and through the postulated accident.*

      iii. *Explain whether the polynomial transformation is applied merely to the centerline temperature, or to the entire pellet temperature.*

   b. *Provide additional information to describe the polynomial transformation. Summarize data used to develop the polynomial transformation and discuss consideration of applicable uncertainties.*

Response:

The NRC concern covers a wide range of specific items but can be paraphrased as: "How does the AREVA RLBLOCA analysis for Sequoyah provide a licensing basis for fuel throughout its operational life with particular attention to the phenomena of thermal conductivity degradation with burnup?" In response, the following explanation of the methodology employed for Sequoyah is provided and followed by specific responses to each of the particular questions.

The AREVA transition package has been updated to specifically model both first and second cycle fuel rods. Third cycle fuel does not retain sufficient energy potential to achieve significant cladding temperatures nor cladding oxidation and is not included in the RLBLOCA individual pin calculations. The burnup for the individual first and second cycle rods analyzed is assigned according to the sampled time in cycle. The time in cycle is sampled once and is the same for both the fresh (first cycle) and once-burnt (second cycle) fuel. Burnup for the fresh and once-burnt rods is different in accordance with the cycle management. Likewise, pin pressure and thermal conductivity differ.

In addition to the thermal conductivity and fuel temperature adjustments for burnup, a burnup dependent reduction in allowed peaking is needed for the once-burnt fuel. For first cycle fuel, the RLBLOCA methodology increases the $F_{\Delta H}$ to the Technical Specification maximum (including uncertainty) for the first cycle hot rods in the model. Shortly into the cycle, once-burnt fuel has insufficient energy potential to achieve this peaking. A burnup dependent reduction in allowed peaking is therefore applied through an adjustment in the second cycle $F_{\Delta H}$. Not modeling the reduction would result in the simulation of an operational state for the once-burnt fuel that would be impossible to achieve. Figure 6-4 provides the bounding once-burnt fuel UO_2 power ratio curve. The curve expresses the relative power to which the once-burnt fuel pins will be controlled as a function of burnup during the cycle.

1.a.i The RODEX3 topical report, ANF-90-145(P)(A), Appendix B (Reference 16) details the calculation of the radial temperature distribution.

1.a.ii A portion of the RODEX3A fuel model was incorporated into the S-RELAP5 code to calculate fuel response for transient analyses. This coding, referred to as the S-RELAP5/RODEX3A model, deals only with transient predictions and does not calculate the burnup response of the fuel. Instead, fuel conditions at the burnup of interest are transferred via a binary data file from RODEX3A to S-RELAP5/RODEX3A, establishing the initial state of the fuel prior to the transient. The data transferred from RODEX3A describes the fuel at zero power. A steady-state S-RELAP5/RODEX3A calculation is required to establish the fuel state at power. The transient fuel pellet radial temperature profile is computed by solving the conduction equation in S-RELAP5. Material properties are calculated in S-RELAP5/RODEX3A.

1.a.iii The adjustment is applied to the entire fuel pellet. The polynomial transformation provides a bias adjustment to the fuel centerline temperature. A sampled parameter provides a random assessment and adjustment of the centerline temperature uncertainty. These are combined and the total adjustment is achieved by iterating a multiplicative adjustment to the fuel thermal conductivity until the desired fuel centerline temperature is reached.

1.b. Paraphrased concern: Provide information on the treatment of thermal conductivity degradation.

Thermal conductivity degradation impacts the ability to transfer energy from within the pellet to the pellet surface and consequently through the cladding to the coolant. Both the initial pellet temperature and the transient release of energy from the pellet are affected. The impact of thermal conductivity changes with burnup are treated by applying a bias. This bias and a measure of the uncertainty in the data were determined by benchmarking the fuel performance code, RODEX3A, to a set of data that extends past the licensed burnup. The bias adjusts the initial fuel temperature to the mean of the benchmark results. The sampled uncertainty is used to provide for the variance of the benchmarks.

The database for the benchmarks is that used to qualify and approve the RODEX4 code (Reference 15). The data from three experimental rods (cases 432R2, 432R6, and 597R8) were not used in the benchmarks. Test 597R8 was not appropriate for this application. Cases 432R2 and 432R6 are rod studies that are not configured appropriately these types of comparisons. Essentially, these fuel rods are not representative of commercial PWR fuel. Part of the benchmark activity was to incorporate a fractional representation of difference between the RODEX3A calculated results and the data. The fractional adjustment provides a better adjustment over a range of initial temperatures. Therefore, for each benchmark case the $T_{fraction}$ was determined.

$$T_{fraction} = \frac{T_{rodex3A} - T_{data}}{T_{rodex3A}},$$

where:

$T_{fraction}$ = Delta fractional temperature of computed to data (K),

$T_{rodex3A}$ = Temperature computed by RODEX3A (K) and

T_{data} = Temperature from the RODEX4 database (K)

Figure 6-1 shows the RODEX3A benchmark results along with a polynomial fitted to the results using the least squares method. The negative of this polynomial is the bias which is added to RODEX3A predictions to achieve agreement with the data. Figure 6-2 shows the results of applying this bias in comparison to the results of applying the original RLBLOCA methodology Revision 0 bias. It is evident from Figure 6-2 that the bias makes the adjustment for burnup effects in accordance with the data.

The application of the bias within the methodology proceeds as follows: The burnup for the case hot rods, fresh and once burned, is determined by sampling the time in cycle and a RODEX3A calculation of the initial fuel centerline temperature performed. From the fit in Figure 6-1 an adjusted temperature is determined as per the equation below.

[ ]

where:

T_{new} = Adjusted fuel centerline temperature (K),

B = Burnup (Gwd/MtU or Mwd/KgU) and

$T_{original}$ = Unadjust RODEX3A fuel centerline temperature (K).

Figure 6-3 provides the bias adjustment $T_{new}/T_{original}$, as a function of burnup, using the above polynomial curve fit.

The uncertainty is determined from a Gaussian distribution characterized by a [ ] standard deviation and added to T_{new}. The fuel temperature calculation is then repeated with a multiplier, fuel K, on the code calculated fuel thermal conductivity. The fuel centerline temperature is compared to 'T_{new} + uncertainty' and the calculation is repeated with an adjusted fuel K as necessary. The process is continued until the calculated centerline fuel temperature matches 'T_{new} + uncertainty'. Since the process applies an adjustment to the fuel thermal conductivity, the temperature throughout the pellet is adjusted appropriately. The final multiplier is applied to the thermal conductivity throughout the transient.

Because the data fitting covers the complete range of applicable burnup it is applied as such and the zero bias offset used in Revision 0 for the first 10 GWd/mtU burnup is eliminated.

Follow-on Questions to #1:

The issue described in IN 2009-23 invalidates AREVA's generic disposition for analyzing fresh fuel only, which is based on sensitivity studies indicating that mid-second-cycle fuel had a PCT of 80°F lower than the limiting PCT. This work needs to be repeated accounting for fuel thermal conductivity degradation. Please provide several cases run at various times-in-life for once-burnt fuel, with information similar to the above list provided; burnup for the limiting rod is only necessary for the most limiting second-cycle case analyzed.

For the PCT-limiting RLBLOCA case, please provide:

a. *Corrected and uncorrected radial temperature profile of the hot rod at the time and location of peak cladding temperature.*
b. *Temperature vs. time for the limiting PCT case at the limiting location, including the fuel centerline, fuel average, and clad surface temperatures. Indicate the end of blowdown, start of refill, and start of reflood on this graph.*
c. *Burnup for the limiting rod.*

Response:

Figure 6-5 shows the corrected and uncorrected radial temperature profile for the limiting case hot rod at the initiation of the transient. Because the uncorrected radial profile is never used or recorded in the methodology, it cannot be provided. However, the uncorrected centerline temperature is available and shown on Figure 6-5. As the pellet power is not adjusted the radial temperature profile must follow the corrected profile closely and the two must converge at the surface of the pellet. Figure 6-6 shows the centerline, surface, and average fuel temperatures of the fresh UO_2 rod at the PCT elevation for the limiting PCT case. In this case, all of the fresh rods have higher PCTs than the once-burnt rods. The most limiting once-burnt rod is the UO_2 rod. With a cycle burnup of approximately 5400 EFPH, the fresh fuel has a burnup of 11.6 GWd/MTU while the once-burnt fuel has a burnup of ~30 GWd/MTU. A plot comparing the PCT of the fresh and once-burnt UO_2 rods for this case is shown in Figure 6-7.

Figure 6-1 Fractional Fuel Centerline Temperature Delta Between RODEX3A and Data

Figure 6-2 Fuel Centerline Temperature Delta of RODEX3A Calculations to Data (Original and Using the New Correlation)

Figure 6-3 Correction Factor (as applied for temperatures in Kelvin)

Figure 6-4 K(Burnup) Curve

Figure 6-5 Radial Temperature Profile for Hot Rod

Figure 6-6 Temperature versus Time for Fuel Centerline, Clad Surface, and Fuel Average

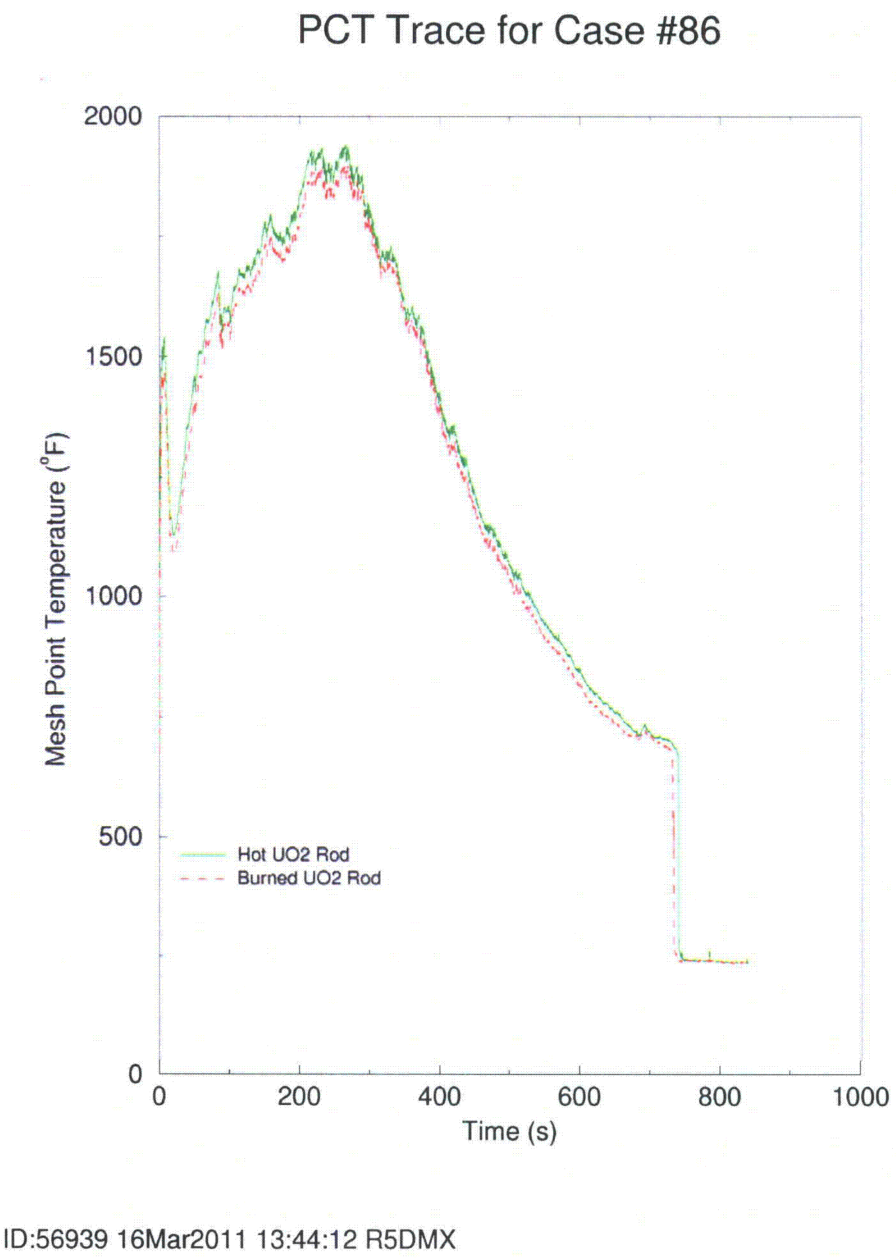

Figure 6-7 Fresh and Once-Burned UO_2 Rod PCT Trace

Question:

2. *The current licensing basis, deterministic loss of coolant accident (LOCA) analysis concluded that the limiting condition did not involve a worst-case single failure, but rather that it depended on injected coolant delivered in such a condition that the resultant containment environment, specifically the lower containment pressure, contributed to the limiting peak cladding temperature (PCT). Please provide information describing how this potentially limiting scenario was evaluated using the proposed best-estimate methodology.*

Response:

2. The current licensing basis for both Sequoyah Units is AREVA's NRC-approved RLBLOCA evaluation model and the worst single failure considered is loss of diesel with fully functional containment sprays. The EM also conservatively prescribes:

   (1) The use of full containment sprays without a time delay at the minimum technical specification temperature;
   (2) Pumped ECCS injection at the maximum technical specification temperature; and
   (3) Sampling of the containment volume (indirectly sampling containment pressure) from its nominal volume to its empty volume.

   Studies, comparing several failure assumptions, including a no-failure assumption (see EMF-2103(P)(A) Revision 0, RAI response Numbers 26 and 111) validate that the ECCS and containment modeling of the AREVA methodology trends to the conservative. The containment pressure response is indirectly ranged by sampling the containment volume. The possible range to be sampled from was 6.510E+5 to 6.926E+5 ft^3 for the Sequoyah Units upper containment volume. Figure 4-21 shows that there is little sensitivity between containment volume (indirectly pressure) and PCT for a statistical application. Thus, the methodology is responsive to the goal of a realistic evaluation, yet slightly conservative.

Question:

3. *Please provide additional information summarizing the single-failure evaluation performed to establish compliance with General Design Criterion (GDC) 35 requirements. Identify which single failures were considered, discuss whether each failure was evaluated or explicitly analyzed, and for those failures which were explicitly analyzed, explain whether they were analyzed in a reference case or explicitly as a part of the statistical methodology. Also discuss the basis for the single failure evaluation. For example, were single failures considered as a matter of experience with SEQUOYAH specifically, or with a generic Westinghouse nuclear steam supply system design?*

Response:

3. Section 4.9 discusses GDC 35. The single failure prescribed by EMF-2103(P)(A) (AREVA's RLBLOCA EM) is a loss of one train of ECCS (See response to RAI Number 2).

   AREVA satisfies the GDC-35 criteria by running one set of 93 cases with offsite power available and one set of 93 cases with no offsite power available. The sampling seeds are held constant between these two case sets, with the only difference being the offsite power assumption. The case set that produces the most limiting PCT is reported, for Sequoyah, this was offsite power available. Figure 3-24 in this document displays the results from the two case sets.

Follow-on Questions to #2 and #3:

a. *The staff also needs to understand how the limiting single failure for the 4-loop W NSSS was determined, since the basis for the RAI response defers to NRC-approved methodology. Poring through EMF-2103, the staff only located sensitivity results on 3-loop W systems. In some cases, the limiting failure would be a single LPSI and in others it was a diesel. The staff could not locate a clear, generic disposition for the single failure at any place in EMF-2103.*
b. *What was done under the auspices of EMF-2103 development to ensure that the containment analysis produced a sufficiently conservative prediction that a no failure, max SI spillage case, for a 4-loop W NSSS, is bounded by the chosen single failure? The staff will need to see that work.*

Response:

The definition for loss of a diesel scenario by itself would mean that in addition to loss of one LPSI and one HPSI pump, one train of containment spray would not be available. The current method models all containment pressure-reducing systems as fully functional. Containment fans start at time zero and containment sprays have a 10 second delay (Table 3-8).

The response to RAI #111 for EMF-2103 (Reference 26, Attachment 1 page 185 – 189) was based on sensitivities to 3-loop W plants. The Base Case, which produced the most limiting results, is described in the RAI #111 response as the loss of one diesel with full containment spray.

Figure 6-8 (recreated from RAI #111, Figure 111.2) shows that for the sample plant analysis, W 3-loop, the base case, AREVA ECCS failure assumptions, is 35 °F higher in PCT than a fully consistent loss of diesel and over 170 °F greater than the loss of one LPSI case.

Figure 6-8 Clad Temperature Response from Single Failure Study

A sensitivity study of the Sequoyah limiting case (Case 86) from the Analysis of Record (AOR) was conducted with "maximum" ECCS flow conditions to demonstrate that the minimum ECCS single failure assumption is conservatively bounding.

Sensitivity studies were run for the limiting case (Case 86) in both offsite power configurations with a maximum ECCS delivery. The loss of offsite power (LOOP) case for the max ECCS configuration had a PCT value of 1698 °F compared to AOR LOOP with minimum ECCS, which had a PCT of 1893 °F. The no loss of offsite power (NOLOOP) case for the max ECCS configuration had a PCT value of 1650 °F compared to AOR NOLOOP case with minimum ECCS, which had a PCT of 1941 °F. This demonstrates that the AREVA single failure assumption produces conservative results. Figures 6-9 through 6-12 show the respective PCT trace, containment and system pressure, ECCS injection rates, and downcomer level for both the AOR and the max ECCS sensitivity.

Figure 6-9 demonstrates that the maximum ECCS flow does not have a significant impact on the containment pressure up to about 100 seconds (approximately the time that the accumulator empties); the max ECCS containment pressure overlaps the AOR containment pressure. Figure 6-12 gives the downcomer level for both the AOR and the max ECCS case. It can be seen that the downcomer level in the max ECCS case is higher than the AOR, consequently providing more driving head for the reflood of the core. The higher driving head in the max ECCS case is enough to compensate for small differences in containment pressure (Figure 6-10) resulting in a faster post peak cooldown.

The AREVA RLBLOCA application, regardless of the loss of diesel assumption, models all containment pressure-reducing systems and conservatively assumes them to be fully functional. The AOR conservatively assumes an on-time start and normal lineups of the containment spray and fan coolers to conservatively reduce containment pressure and increase break flow. The results of the study demonstrate that the AOR ECCS configuration is PCT-limiting and oxidation-limiting.

Figure 6-9 Comparison of PCT Independent of Elevation for Max ECCS and Min ECCS

Figure 6-10 Comparison of Containment and System Pressure for Max ECCS and Min ECCS

Figure 6-11 Comparison of ECCS Flows for Max ECCS and Min ECCS

Figure 6-12 Downcomer Level

Question:

4. *Page 3-6 states, "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." For the limiting transient, the collapsed core liquid level from 200-350 seconds appears to trend downward (Figure 3-21). An indication of stable and increasing collapsed liquid level would substantiate the statement quoted above, but this is not the case for Figure 3-21. Is the SRELAP-5 model of the limiting case capable of generating credible results after 350s? If so, please provide results for a period of the transient sufficient to demonstrate that the core collapsed liquid levels are stable or increasing.*

Response:

4. Not applicable to the Sequoyah analyses; Figure 3-21 clearly shows a steady level increase from 200 seconds through transient termination.

Question:

5. *Please provide information to enable comparison between Technical Specifications (TS) requirements and analytic input parameters for Pressurizer Level. The TS requirement is given in inches and the input parameters are specified in percent span.*

Response:

5. Technical Specification LCO 3/4.4.4 states "The pressurizer shall be OPERABLE with a water volume of less than or equal to 1656 cubic feet (equivalent to an indicated level of less than or equal to 92% on narrow range instrumentation), and at least two groups of pressurizer heaters each having a capacity of at least 150 kw."

   The sampled range for the liquid level uncertainty in the pressurizer was 57- to 95-percent of span. The Technical Specifications for Sequoyah do not have the requirement in inches, just percent span.

Question:

6. *Please provide discussion to confirm that the assumed 80°F upper containment temperature and 95°F lower containment temperature are acceptable minimums without a TS requirement.*

Response:

6. Sequoyah does have an Technical Specification LCO for containment temperature. Technical Specifications LCO 3.6.1.5 states "between 85°F and 105°F in the containment upper compartment, and between 100°F and 125°F in the containment

lower compartment." The inputs TVA provided AREVA for the RLBLOCA analysis bound these LCO values.

Question:

7. *The TS minimum for the refueling water storage tank (RWST) temperature is 60°F. Previous, deterministic analyses demonstrated that minimum safety injection temperatures resulted in a limiting PCT. In light of this information, please explain why a minimum RWST temperature case was not evaluated, or if a minimum RWST temperature case was evaluated, please summarize the evaluation and discuss its conclusions.*

Response:

7. As stated in the response to Question 2 in Section 6, the NRC-approved RLBLOCA EM, EMF-2103(P)(A), prescribes use of the maximum temperature for the ECCS pumped injection and use of the minimum temperature for containment sprays. Sequoyah's temperatures were 110 °F for pumped injection and 55 °F for the containment sprays. While inconsistent, the choice of the two temperatures is conservative. AREVA's RLBLOCA analysis complies with, and does not deviate from, it's approved EM requirements.

Question:

8. *As noted in Section 1 of the AREVA RLBLOCA Summary Report, deviations from the approved RLBLOCA evaluation model (EMF-2103(P)(A), Revision 0) are necessary to demonstrate compliance with 10 CFR 50.46 requirements. Please provide a commitment to adhere to the deviations noted in Section 1 of AREVA RLBLOCA Summary Report until such time as:*

   a. *AREVA develops a new revision of EMF-2103,*

   b. *The NRC approves the new revision of EMF-2103, and*

   c. *Sequoyah implements the new, NRC-approved revision of EMF-2103.*

   The commitment should include language to indicate that meeting Conditions a, b, and c, above, or submitting a license action request to implement a different evaluation method, will obviate the need for this commitment.

Response:

8. TVA commits to the following:

Sequoyah Units 1 and 2 will adhere to the deviations noted in Section 1 of ANP-2970(P)(A) until such time as:

- AREVA develops a new revision of EMF-2103,
- The NRC approves the new revision of EMF-2103, and
- TVA implements the new, NRC-approved revision of EMF-2103.

This commitment will terminate when the above items are met or a license amendment is approved to permit the use of a different evaluation method to replace ANP- 2970(P)(A).

Question:

9. *The following questions are based on a July 14, 2009, letter from Gardner, AREVA NP, to the USNRC, re: Informational Transmittal Regarding Requested White Papers on the Treatment of Exposure Dependent Fuel Thermal Conductivity Degradation in Legacy Fuel Performance Codes and Methods.*

   A) AREVA postulates that clad swelling and rupture produces a benefit to PCT, and because of this, the realistic large break loss of coolant accident (RLBLOCA) model does not include a clad swelling and rupture model. Does this conjecture include consideration of test data, which has shown that following fuel rupture, the ballooned region fills with fuel fragments? What analytic studies support this conclusion? How are they applicable to Sequoyah? Please also address the potential for co-planar blockage with the fuel relocation evaluation.

   B) Since blowdown ruptures can occur at end of life conditions, show that blowdown ruptures do not occur at the end of life for the postulated Sequoyah large break LOCA.

Response:

[ ]

Experience with Appendix K methodologies has shown that the aggregate of these effects acts to decrease the cladding temperatures when no fuel relocation occurs. This was demonstrated in Appendix B, Section B.2 of RLBLOCA EM Topical (Reference 1) and the response to RAI 28 on the topical (page 79 of Amendment 1 to Reference 17) with sensitivity studies on both 3- and 4-loop PWRs with 15x15 and 17x17 fuel designs similar to the 17x17 fuel design used at Sequoyah. The studies included increased heat transfer surface area, increased local coolant velocities, a decrease in gap heat transfer, flow diversion, and interior cladding oxidation. The effects of increased turbulence, droplet shattering, and potential local quenching were not included within the modeling. Decrease in pellet thermal conductivity and a clad heating load increase also were not included since the studies were not meant to address fuel relocation. Even without half of the cooling mechanisms modeled, the cladding temperatures and local oxidations were reduced. This effect has also been observed experimentally in the FEBA (Reference 18) and FLECHT (Reference 19) test series.

Under a condition of fuel relocation, wherein the fuel above the ballooned region drops into the ballooned region, it has been postulated that increased decay heat generation will lead to an increase in cladding heat flux resulting in higher cladding temperatures. Various presentations (e.g., Reference 20 Articles 1 and 12) purport to show the effect. However, these studies have uniformly incorporated extreme assumptions on the conditions of relocation and the resultant heat transfer processes. Few include provisions for rupture-induced cooling mechanisms. Most assume that the cladding expands circularly without being encumbered by the surrounding pins in the fuel assembly. In fact, a free expansion of the fuel rod is only possible up to pin strains in the mid-30 percents. For higher strains the local gap volume no longer increases faster than the clad surface area. Finally, the packing factor of the rubble filling the ballooned region is over-predicted. If reasonable, yet conservative, assumptions are made, study results would lead to the expectation that fuel relocation, which is real, does not pose a condition by which the ruptured or ballooned regions will exceed the consequence of the non-ballooned regions of the hot pin.

The above conclusion was observed experimentally in the KfK experiments as reported in RAI 131 on the RLBLOCA EM topical report (page 120 of Amendment 1 to Reference 17). In the KfK in Pile Tests, fuel relocation into the ballooned area of the fuel rod occurred but did not adversely affect the subsequent clad temperature behavior. To determine when the fuel relocates two tests were performed with thermocouples located at the top of the pellet stack. One test comprised low burnup fuel, which maintained its pellet geometry after rupture. The other test was of higher burnup fuel which relocated. Relocation, for the test that relocated, was demonstrated by temperatures from the upper thermocouples showing a significant drop, loss of energy source, at the time of fuel rod rupture. For this test, the heatup rate, at the rupture elevation, following the rupture was reduced relative to the heatup rate prior to rupture. This reduction in heatup rate indicates that the PCT at the time of turnover would be less than what would have be reached if rupture had not occurred, even with the increase in localized decay heat from the pellet rubble residing at the ruptured region. Thus, the KfK experiments demonstrate that analyses which ignore the beneficial effects of swelling and rupture provide conservatively high clad temperature estimates for the ruptured region during reflood even when fuel relocation occurs.

[ ]

Question:

10. *Provide information to illustrate the conservative nature of the single-side only oxidation model and its application to the SEQUOYAH RLBLOCA analysis.*

Response:

10. AREVA's NRC-approved RLBLOCA EM uses the maximum un-ruptured cladding oxidation as representative or bounding of the oxidation that would have been computed at a rupture location. The position is supported by three aspects of the performed oxidation calculation.

- The cladding is initialized with no initial corrosion layer. Because the oxidation rate is inversely proportional to the oxidation layer present, the use of clean cladding at the start of the accident leads to substantially higher reaction rates. For corrosions in the range of the first cycle of M5 cladding, the difference in rate is a minimum of a 50-percent increase and increases during the cycle. The increase applies to both exterior and post-rupture interior oxidation.

- The cladding temperature even in the presence of fuel relocation is reduced for the ruptured region of the cladding. In the KfK experiments (page 210 of NRC:02:062 Attachment 1 to Reference 17 and included in Reference 18) the temperature drop at rupture was between 50 and 75 K. Since the oxidation rate is exponentially proportional to the cladding temperature, a drop of 50 to 75 K for Sequoyah provides an oxidation rate reduction of 50-percent or more.

- For ruptured cladding either the cladding interior oxidation rate is reduced by attached pellet fragments, moderate to highly burned fuel, or the cladding temperature decrease at rupture is much more than the 50 to 75 K explained in Item 2. In either case, an additional mechanism exists to reduce the local oxidation at the rupture location.

In conclusion, insights into the EM oxidation process and those that will evolve after rupture clearly identify differences that will reduce the oxidation at the rupture location to less than that which the EM calculates at un-ruptured locations. Thus, the RLBLOCA Revision 0 EM approach to reporting local oxidation is clearly appropriate to demonstrate compliance with the local oxidation criterion of 10CFR50.46.

Follow-on Question to #10:

NRC Generic letter 98-29 calls of the initial corrosion layer on the fuel pin to be included in the reported local oxidation value demonstrating compliance with the criteria of 10CFR50.46.

Response:

The initial corrosion layer was calculated to be 0.7624-percent for the Fresh UO_2 rod (at 15 GWd/MTU) and 1.3144-percent for the once-burned 6% GAD rod. The initial corrosion layer was added to the transient calculated value and the total is in Table 3-5.

Question:

11. *Provide additional information to justify the use of the selected analytic treatment for decay heat uncertainty in the RLBLOCA model.*

Response:

11. The RLBLOCA EM decay heat calculations are based on the 1979 ANSI/ANS standard (Reference 25). The standard is applicable to light water reactors containing low enriched uranium as the initial fissile material; all plants, to which the RLBLOCA EM is applicable, including Sequoyah, are such plants. The selected approach to simulate fission product decay assures a representative yet conservative treatment. The EM fission product decay heat uncertainty and the basis for the conservatism of the approach are outlined in the remainder of the response.

Non-Sampling Approach to Decay Heat

The RLBLOCA methodology proposed herein utilizes the U235 decay curve from the 1979 ANSI/ANS standard for fully saturated decay chains as the decay for all fission products. The fully saturated chains result from an assumption of infinite operation. The total energy per fission is assumed to be 200 MeV (Reference 25). No bias or uncertainty is assigned to the fission product decay heat. Differing from the base EMF-2103 evaluation model approach, the uncertainty for the decay heat parameter is set to zero and no sampling is done on this parameter, resulting in the decay heat being used with a 1.0 multiplier. The decay heat in the analysis is always the 1979 ANS standard for decay heat from U235 with fully saturated decay chains, corresponding to infinite operation, assuming 200 MeV per fission.

Conservatism in the Approach

In the approach used, the total energy per fission is assumed to be 200 MeV whereas a more accurate value for U235 would be greater than 202 MeV per fission. This imparts a direct 1-percent conservatism.

During irradiation, plutonium accumulates such that the ratio of plutonium-to-uranium fission-energy production rate is substantial and increasing. Because the decay energy resulting from plutonium fissions is less than that from U235, the decay energy is reduced from U235 fully saturated decay chains as the fuel is burned. Thus, as burnup increases, the RLBLOCA decay heat modeling with U235 only, accrues conservatism. This conservatism applies to all regions of the core according to the mix of burnups represented within each region.

The fresh fuel, hot pin and hot assembly, begin operation with no plutonium. Therefore, the reduction in decay heat due to plutonium build-up is not applicable to the low burnup fuel in the initial period of the cycle. However, for fresh fuel, the concentrations of long decay term fission products will not have built up. The lack of long decay term sources comprises a reduction in decay heat rate of several percent over the first year of operation, making the infinite operation assumption conservative while the plutonium concentration is accumulating.

Calculations of these considerations based on the 1979 ANS standard have been performed to demonstrate the conservatism of the selected approach. Figure 6-13 and Figure 6-14 show the decay heat versus time for:

1) Infinite Operation of U235 (the AREVA decay heat model)
2) Finite Operation to 0 GWD/mtU of all fissionable isotopes with uncertainties added
3) Finite Operation to 1 GWD/mtU of all fissionable isotopes with uncertainties added
4) Finite Operation to 1 GWD/mtU of all fissionable isotopes without uncertainties
5) Finite Operation to 20 GWD/mtU of all fissionable isotopes with uncertainties added
6) Finite Operation to 40 GWD/mtU of all fissionable isotopes with uncertainties added
7) Finite Operation to 60 GWD/mtU of all fissionable isotopes with uncertainties added

In order to treat the Plutonium buildup effect conservatively, the finite operations curves are based on cycle management and enrichment assumptions that minimize the build up of Plutonium. No uncertainty is included in the infinite operation curve. The uncertainties incorporated in the other curves are 2 sigma values for the individual isotopes as published in the 1979 ANS standard. This provides greater than a 95/95 confidence in each of the decay heat contributions. The contributions are added linearly according to the individual isotopes fractional occurrence of fission.

Because of the range of the decay heat parameter, the early comparison of the relationships is difficult to ascertain. Clearly the U235 infinite operation curve is conservative for all times after a few seconds (~2 seconds). To better demonstrate the relationships, Figure 6-15 and Figure 6-16 provide the ratios of the finite operation curves to the infinite operation curves. The curvature of the plotted ratios during the first 2 to 3 seconds is due to the increased uncertainties during this time phase. The 1979 ANS standard is based on measured data and the difficulty of measuring decay heat within a few seconds of shutdown is reflected in these uncertainties. The highest combined finite operation decay heat curve with uncertainties exceeds the AREVA decay heat curve by only 2.5 percent at shutdown and falls below the

AREVA curve in less than 2 seconds. Thus, there is only a 5 percent probability that the infinite operation curve of decay heat will be exceeded by up to 2.5 percent and that possibility exists for the firs 2 seconds of the transient. The potential accumulated underprediction is of short duration and of no consequence to the LOCA evaluation. The decay heat curve selected is suitable while somewhat conservative for the realistic evaluation of LOCA.

In conclusion, the choice of infinite operation with pure U235 fission product decay heat provides a base model that is conservative relative to the decay heat for finite operation. For RLBLOCA evaluation, the sampling of a decay heat multiplier has been removed such that the decay heat for all cases is 1.0 times the infinite operation U235 decay chain providing conservative treatment of the 1979 ANS standard with the assumption of 200 Mev/fission.

Follow-on Question to #11:

The NRC needs to understand the sensitivity that PCT has with respect to the decay heat uncertainty, please re-execute the limiting case with a 1.03 decay heat multiplier and report the results.

Response:

Not applicable to the Sequoyah analysis. The decay heat was not sampled.

Figure 6-13 Decay Heat Comparisons, Infinite Operation U235, Finite Operation All Isotopes (0.1 – 10 sec)

Figure 6-14 Decay Heat Comparisons, Infinite Operation U235, Finite Operation All Isotopes (10 – 1000 sec)

Figure 6-15 Decay Heat Ratios, Finite Operation over Infinite Operation U235 All Isotopes (0 – 10 sec)

[ ]

Figure 6-16 Decay Heat Ratios, Finite Operation over Infinite Operation U235 All Isotopes (>10 sec)

7.0 References

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

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ATTACHMENT 11

AREVA NP Affidavits

Attached are the affidavits supporting the request to withhold proprietary information (included in Attachments 5, 6, and 7) from the public.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) 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 ANP-2986(P), Revision 002, entitled "Sequoyah HTP Fuel Transition," dated June 2011 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 8th

day of June, 2011.

Kathleen A. Bennett

Kathleen Ann Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/11
Reg. # 110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2011

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
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CITY OF LYNCHBURG)

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3. I am familiar with the AREVA NP information contained in the report ANP-2971(P), Revision 1, entitled "Sequoyah Units 1 and 2 HTP Fuel S-RELAP5 Small Break LOCA Analysis," dated May 2011 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.

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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 12th

day of May 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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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-2970(P), Revision 0, entitled "Sequoyah Units 1 and 2 HTP Fuel Realistic Large Break LOCA Analysis," dated March 2011 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.

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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 28th

day of March 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014