ANP-10278NP
Revision 1

# U.S. EPR Realistic Large Break Loss of Coolant Accident

## Topical Report

**January 2010**

AREVA NP Inc.

## Nature of Changes

| Item | Section(s) or Page(s) | Description and Justification |
|---|---|---|
| 1-1. | 1.0 | Added link to methodology changes. |
| 1-2. | 1.0 | Updated to reflect additional Appendices. |
| 1-3. | 2.2.1 | Clarification of Safety Injection System description. |
| 1-4. | 2.2.2 | Clarification regarding cross-connect between LHSI lines. |
| 1-5. | 3.0 | Separation of Codes and Methods section into individual sections, which changes section numbers. |
| 1-6. | 3.1 | Added to describe use of stand-alone RODEX3A in RLBLOCA calculation. |
| 1-7. | 3.2 | Reorganized S-RELAP5 description into this dedicated section. |
| 1-8. | 3.2.1 | Added to describe use of S-RELAP5/RODEX3A fuel model. |
| 1-9. | 3.3 | Reorganized ICECON description into this dedicated section. |
| 1-10. | 4.0 | Reorganized description of RLBLOCA methodology to this dedicated section. |
| 1-11. | 4.1 | Added to describe modifications to RLBLOCA EM. |
| 1-12. | - | Deleted Event Description section because the information in this section could be found in other sections in the report. |
| 1-13. | 4.2 | Reorganized description of LBLOCA scenario to this dedicated section. Clarification added regarding reactor signal initiation. |
| 1-14. | 4.3 | Clarified that the RLBLOCA model demonstrates compliance to first three criteria of 10 CFR50.46. |
| 1-15. | 4.4 | Updated to reflect use of 124 cases. |
| 1-16. | 4.4.1 | Added to explain use of first cycle fuel. |
| 1-17. | 4.5 | Clarified the description of the single failure. |
| 1-18. | 4.6.1 | Addition description of break size spectrum considered in the analysis. |
| 1-19. | 4.6.2 | Addition description of power shapes considered in the analysis. |
| 1-20. | 4.7.1 | Added explanation of why only LOOP cases are considered in the analysis. |
| 1-21. | Table 4-1 | Clarification of sampled parameters considered in the analysis. |
| 1-22. | Table 4-2 | Added description of when LOOP occurs and added to discussion of partial cooldown. Status of MHSI and LHSI clarified. |
| 1-23. | - | Figure 3-1 moved to Appendix B, which describes the ICECON containment model. |
| 1-24. | 5.1 | Added link to methodology changes section regarding discussion of fuel rod stored energy. Added to discussions of core post-CHF |

| | | |
|---|---|---|
| | | heat transfer and rewet. |
| 1-25. | 5.2 | Added to discussion of downcomer entrainment/de-entrainment and downcomer condensation. |
| 1-26. | 5.3 | Modified discussion of upper plenum entrainment/de-entrainment. Added description of RCP treatment in the analysis. Added discussion of downcomer nodalization as it pertains to downcomer boiling. Modified the discussion of oscillations. |
| 1-27. | 6.2 | Updated IRWST temperatures to be consistent with updated analysis values. |
| 1-28. | 6.2 | Added to discussion of partial cooldown. |
| 1-29. | 7.0 | Updated conclusions to reflect methodology changes. |
| 1-30. | Appendix A | Updated sample problem to reflect methodology changes. |
| 1-31. | Appendix B | Added to describe ICECON containment model |
| 1-32. | Appendix C | Added to describe decay heat modeling. |
| 1-33 | Document | Grammatical and punctuation changes made throughout |

# Contents

**APPENDIX C– Justification of the Decay Heat Modeling in S-RELAP5**

## List of Tables

Page



## List of Figures

Page

## Nomenclature

| Acronym | Description |
|---|---|
| CCFL | Counter Current Flow Limit |
| CCTF | Cylindrical Core Test Facility |
| CCW(S) | Component Cooling Water (System) |
| CHF | Critical Heat Flux |
| CLPS | Cold Leg Pump Suction |
| CSAU | Code Scaling, Applicability, and Uncertainty |
| CVCS | Chemical and Volume Control System |
| DC | Downcomer |
| DCD | Design Control Document |
| DEGB | Double-Ended Guillotine Break |
| DESB | Double-Ended Split Break |
| DNB(R) | Departure from Nucleate Boiling (Ratio) |
| EBS | Extra Borating System |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| EM | Evaluation Model |
| FLECHT | Full Length Emergency Cooling Heat Transfer |
| HFP | Hot Full Power |
| HHSI | High Head Safety Injection |
| HZP | Hot Zero Power |
| IRWST | In-Containment Refueling Water Storage Tank |
| LBLOCA | Large Break LOCA |
| LHGR | Linear Heat Generation Rate |
| LHSI | Low Head Safety Injection |
| LOCA | Loss-of-Coolant Accident |
| LOFT | Loss-of-Fluid Test |
| LOOP | Loss of Offsite Power |
| MER | Mass and Energy Release |
| MFW(S) | Main Feedwater System |
| MHSI | Medium Head Safety Injection |
| MSLB | Main Steam Line Break |
| MSRT | Main Steam Relief Train |
| MTC | Moderator Temperature Coefficient |
| NSSS | Nuclear Steam Supply System |
| PCT | Peak Cladding Temperature |
| PDTF | Product Development Test Facility |
| PIRT | Phenomena Identification and Ranking Table |
| PLHGR | Planar Linear Heat Generation Rate |
| PWR | Pressurized Water Reactor |
| PZR | Pressurizer |

| Acronym | Description |
|---|---|
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| RHR(S) | Residual Heat Removal (System) |
| RLBLOCA | Realistic Large Break LOCA |
| RPV | Reactor Pressure Vessel |
| RV | Reactor Vessel |
| SBO | Station Blackout |
| SEASET | System Effects and Separate Effects Tests |
| SCTF | Slab Core Test Facility |
| SG | Steam Generator |
| SIS | Safety Injection System |
| SMART | Small Array Reflood Test |
| SS | Steady-State |
| THTF | Thermal-Hydraulic Test Facility |
| UFM | Ultrasonic Flow Meter |
| UH | Upper Head |
| UPTF | Upper Plenum Test Facility |
| W/EPRI | Westinghouse/Electric Power Research Institute |
| $\Delta P$ | Pressure Difference |

# 1.0 INTRODUCTION

The purpose of this report is to describe and demonstrate the applicability of AREVA NP's previously approved Realistic Large Break LOCA Evaluation Model to the U.S. EPR design. This methodology is described in detail in the topical report EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors" (Reference 1). In subsequent sections of this report, this Realistic LBLOCA (RLBLOCA) methodology will be demonstrated to be applicable to the U.S. EPR reactor.

The U.S. EPR reactor is an evolutionary Pressurized Water Reactor (PWR) design and retains the principal features of existing 4-loop plants and fuel designs. Section 2.0 provides a brief overview of the U.S. EPR design, focusing on features important for mitigation of LBLOCA events.

Sections 3.0 through 6.0 of this report contain an overview of the LBLOCA codes and methods, a discussion of important phenomena, and the bases for applying the code and methods to the U.S. EPR design. The applicability of the methodology is demonstrated in part by comparing physical characteristics of existing plants and fuel designs, for which the same general methodology has been applied, to the corresponding physical characteristics of the U.S. EPR reactor. In addition, applicability is demonstrated by showing that phenomena occurring in existing plants are the same as those for the U.S. EPR reactor, and that the phenomena are adequately modeled by the codes.

A number of changes have been made to the methodology relative to that described in the NRC-approved topical report EMF-2103(P)(A) for application to the U.S. EPR™. These changes are described in Section 4.1.

Report conclusions appear in Section 7.0. Appendix A contains LBLOCA sample calculations. Appendix B describes the ICECON containment model. Appendix C describes decay-heat modeling in S-RELAP5.

## 2.0 U.S. EPR DESIGN OVERVIEW

The U.S. EPR is an evolutionary PWR with a rated core thermal power of 4590 MWt. The primary system design, loop configuration, and main components are similar to those of currently operating PWRs, thus forming a proven foundation for the design.

The U.S. EPR has a 4-loop Reactor Coolant System (RCS) composed of a Reactor Pressure Vessel (RPV) that contains 241 fuel assemblies, a pressurizer (PZR) including control systems to maintain system pressure, one Reactor Coolant Pump (RCP) per loop, one Steam Generator (SG) per loop, associated piping, and related control and protection systems.

The RCS is contained within a concrete containment building. The containment building is enclosed by a shield building with an annular space between the two buildings. The pre-stressed concrete shell of the containment building has a steel liner and the shield building wall is reinforced concrete. The Containment and Shield Buildings comprise the Reactor Building. The Reactor Building is surrounded by four Safeguard Buildings and a Fuel Building (see Figure 2-1). The internal structures and components within the Reactor Building, Fuel Building, and two Safeguard Buildings (including the plant Control Room) are protected against aircraft hazard and external explosions. The other two Safeguard Buildings are not protected against aircraft hazard or external explosions; however, they are separated by the Reactor Building, which restricts damage from these external events to a single safety division.

Four 100% capacity safety systems are separated into four divisions (one per Safeguard Building). The four divisions of safety systems are consistent with an N+2 safety concept. With four divisions, one division can be out-of-service for maintenance and one division can fail to operate, while the remaining two divisions are available to perform the necessary safety functions even if one is ineffective due to the initiating event.

In the event of a loss of offsite power (LOOP), each safeguard division is powered by a separate Emergency Diesel Generator (EDG). In addition to the four safety-related diesels that power various safeguards, two independent diesel generators are available to power essential equipment during a postulated Station Blackout (SBO) event—loss of offsite AC power with coincident failure of all four EDGs.

Water storage for safety injection is provided by the In-containment Refueling Water Storage Tank (IRWST). Also inside containment, below the RPV, is a dedicated spreading area for molten core material following a postulated worst-case severe accident.

The fuel pool is located outside the Reactor Building in a dedicated building to simplify access for fuel handling during plant operation and handling of fuel casks. As stated previously, the Fuel Building is protected against aircraft hazard and external explosions. Fuel pool cooling is assured by two redundant, safety-related cooling trains.

Although the U.S. EPR embodies a number of improvements on existing PWR designs, these improvements are evolutionary and U.S. EPR design conditions are similar to operating PWRs. Reference 2 (Tables 2-1 through 2-4) contains comparisons of U.S. EPR design parameters and those of contemporary plants.

## 2.1 *U.S. EPR Plant Design and Features*

Reference 2 describes the U.S. EPR core design, the RCS and its principal components, overpressure (primary and secondary) protection, and the principal fluid systems. That discussion will not be repeated here. However, there have been changes to the Safety Injection System/Residual Heat Removal System (SIS/RHRS) since Reference 2 was published. Since these changes affect the LBLOCA response, a description of the SIS/RHRS and the modifications follow.

## 2.2 *Safety Injection System/Residual Heat Removal System (SIS/RHRS)*

The SIS/RHRS performs normal shutdown cooling, as well as emergency coolant injection and recirculation functions to maintain reactor core coolant inventory and provide adequate decay heat removal following a LOCA. The SIS/RHRS also can maintain RCS inventory following a Main Steam Line Break (MSLB).

### 2.2.1 *SIS/RHRS Description and Operation*

The SIS/RHRS consists of four independent trains, with one train dedicated to each of the four RCS loops. Each train provides injection capability using an accumulator pressurized with nitrogen gas, a Medium Head Safety Injection (MHSI) pump, and a Low Head Safety Injection (LHSI) pump. The LHSI pumps also perform the operational functions of the RHRS. (Figure 2-2 is a flow schematic of a single train of the SIS/RHRS.) Each of the four SIS trains is provided with a separate suction connection to the IRWST. Guard pipes are provided for sump suction piping between the sump connection and the suction isolation valve. The sumps are provided with a series of screens, providing protection of the SIS pumps against debris entrained with IRWST fluid.

Each pump is provided with a miniflow (minimum flow) line routed to the IRWST. The miniflow lines prevent pump dead-heading when the RCS pressure is greater than the pump discharge pressure. The LHSI/RHR pump miniflow line also provides cooling and mixing of the IRWST.

In the injection mode, the MHSI and LHSI/RHR pumps take suction from the IRWST and inject into the RCS through nozzles located in the side of the piping. These pumps are located in the Safeguard Buildings, close to the containment. The LHSI/RHR pumps and the MHSI pumps normally inject into the cold legs. In the long term following a LOCA, the LHSI discharge can be switched over to the hot legs to limit the boron concentration in the core, thus reducing the risk of crystallization in the upper part of the core.

An LHSI/RHR heat exchanger is located downstream of each LHSI/RHR pump. These heat exchangers are installed in the Safeguard Buildings and cooled by the CCWS. The accumulators are located inside the containment and inject into the RCS cold legs when the RCS pressure falls below the accumulator pressure, using the same injection nozzles as the LHSI/RHR and MHSI pumps.

During RHR operation, the LHSI/RHR pumps take suction from the RCS hot leg and discharge through the LHSI/RHR heat exchangers back to the RCS cold leg. During shutdown, the LHSI/RHR pump is used in the RHR mode, but the MHSI pump remains available for water makeup in the event of a LOCA.

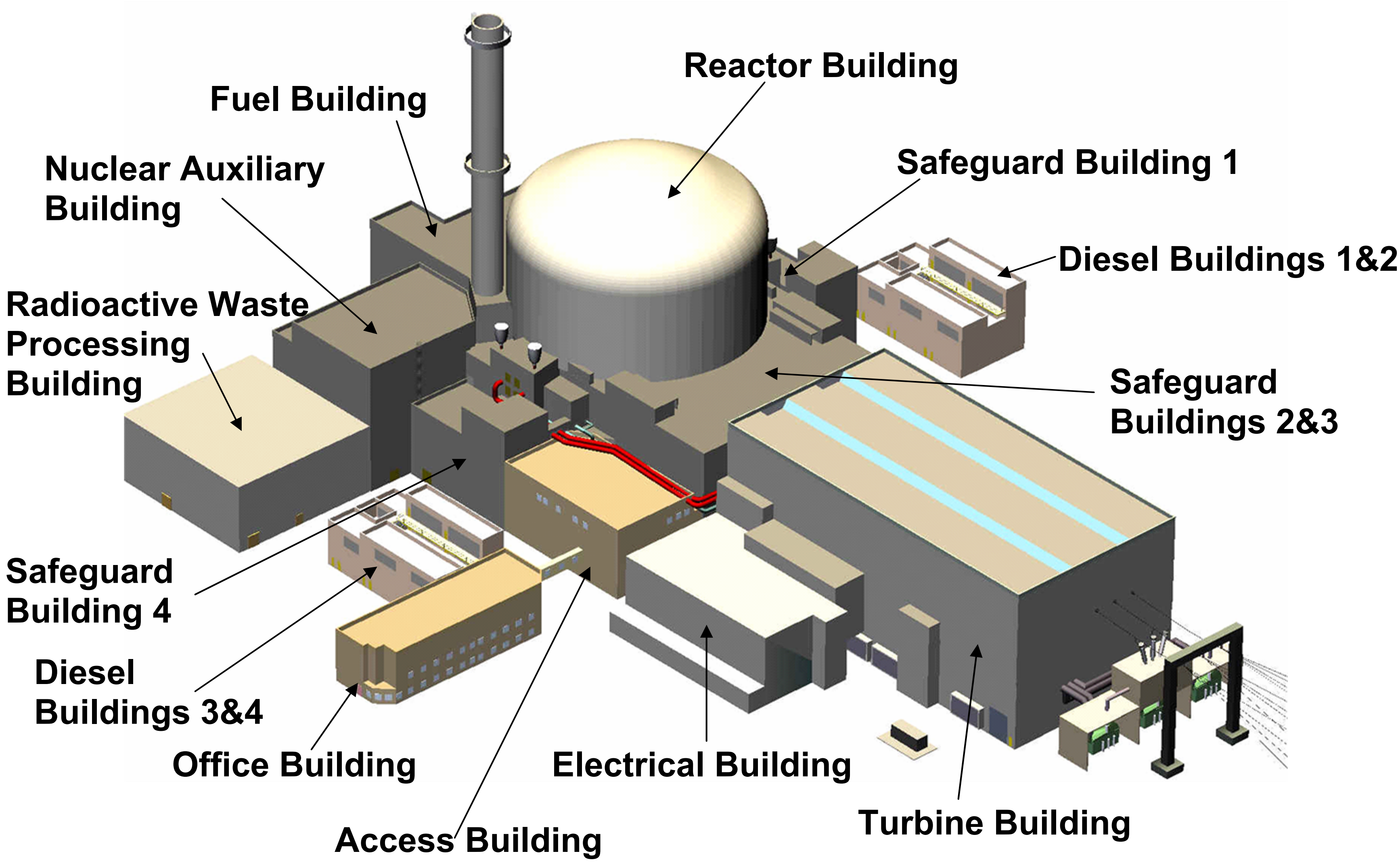
All four SIS/RHRS trains are powered from separate emergency buses, each backed by an EDG. The LHSI/RHR pumps in Trains 1 and 4 are also backed-up by the SBO diesels. One SIS/RHRS train is located in each of the Safeguard Buildings, thereby providing separation and/or physical protection from external and internal hazards.

### 2.2.2 *SIS/RHRS Modifications*

A subsequent change to the design that potentially impacts LBLOCA response is the addition of cross-connects in the SIS/RHRS. Under normal operating conditions, all four trains of SIS/RHRS are separate and independent. However, during online maintenance of an LHSI train, cross-connect valves are opened to connect the LHSI discharge lines of train 1 to train 2 and of train 3 to train 4. In the unlikely event of a LBLOCA coincident with maintenance of an LHSI train, and with an assumed single failure of one train of pumped safety injection, the cross-connected LHSI linesi promote a more even distribution of safety injection to the cold legs.

Figure 2-3 shows the location of the cross-connects for the four SIS/RHRS trains. Figure 2-2 notes the cross-connect attachment location in the flow schematic of a complete SIS/RHRS train. The cross-connect attachment points are made to the LHSI piping upstream of the LHSI/MHSI connection. Check valves prevent MHSI flow from entering the cross-connects. Therefore, only the LHSI (not MHSI) is split between cold legs. The cross-connects are active only when one LHSI train has been taken out of service for maintenance. Isolation valves are installed on the cross-connect piping and are open only during maintenance.

**Figure 2-1 General U.S. EPR Layout**



## Figure 2-2 Safety Injection System/Residual Heat Removal System

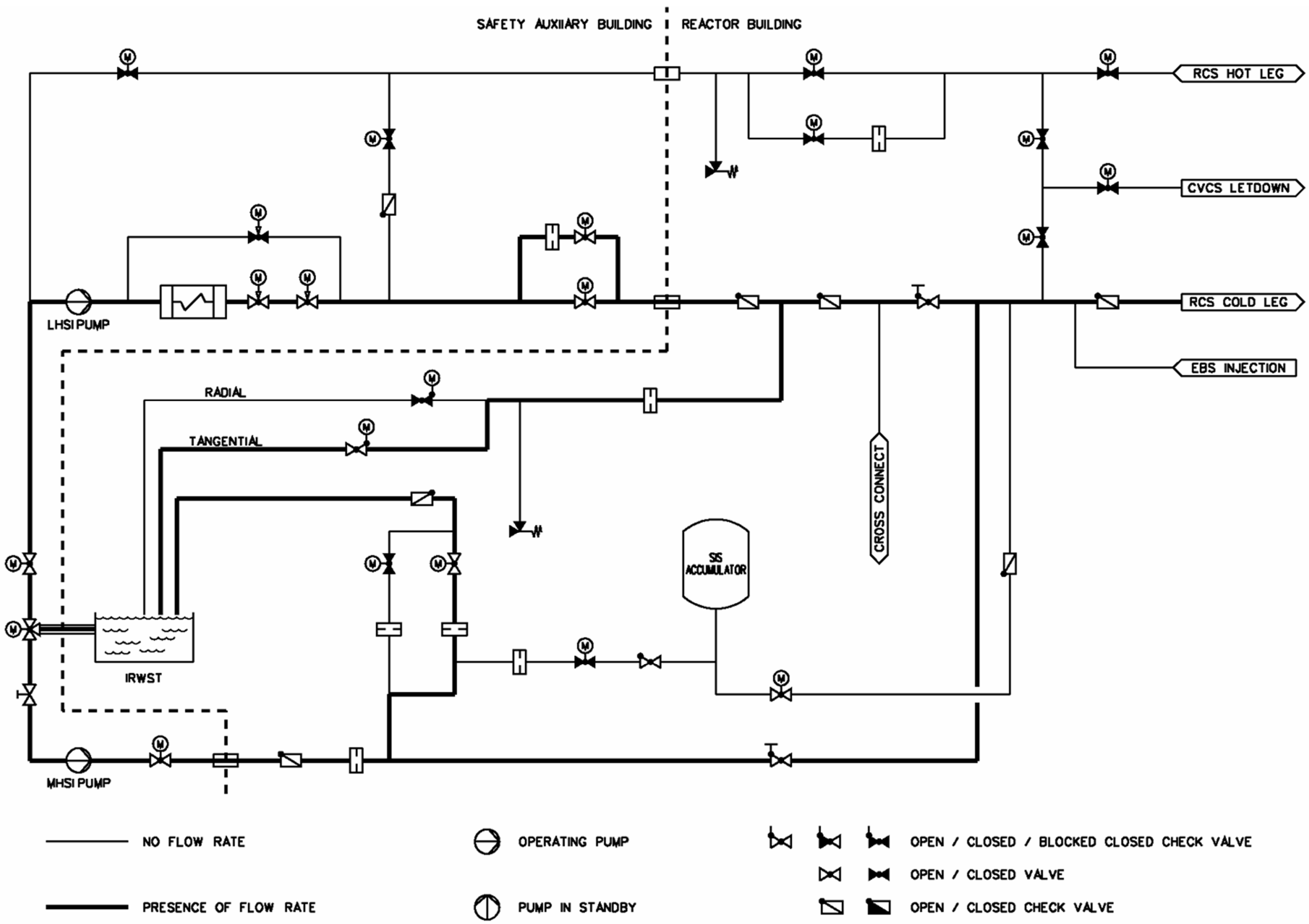
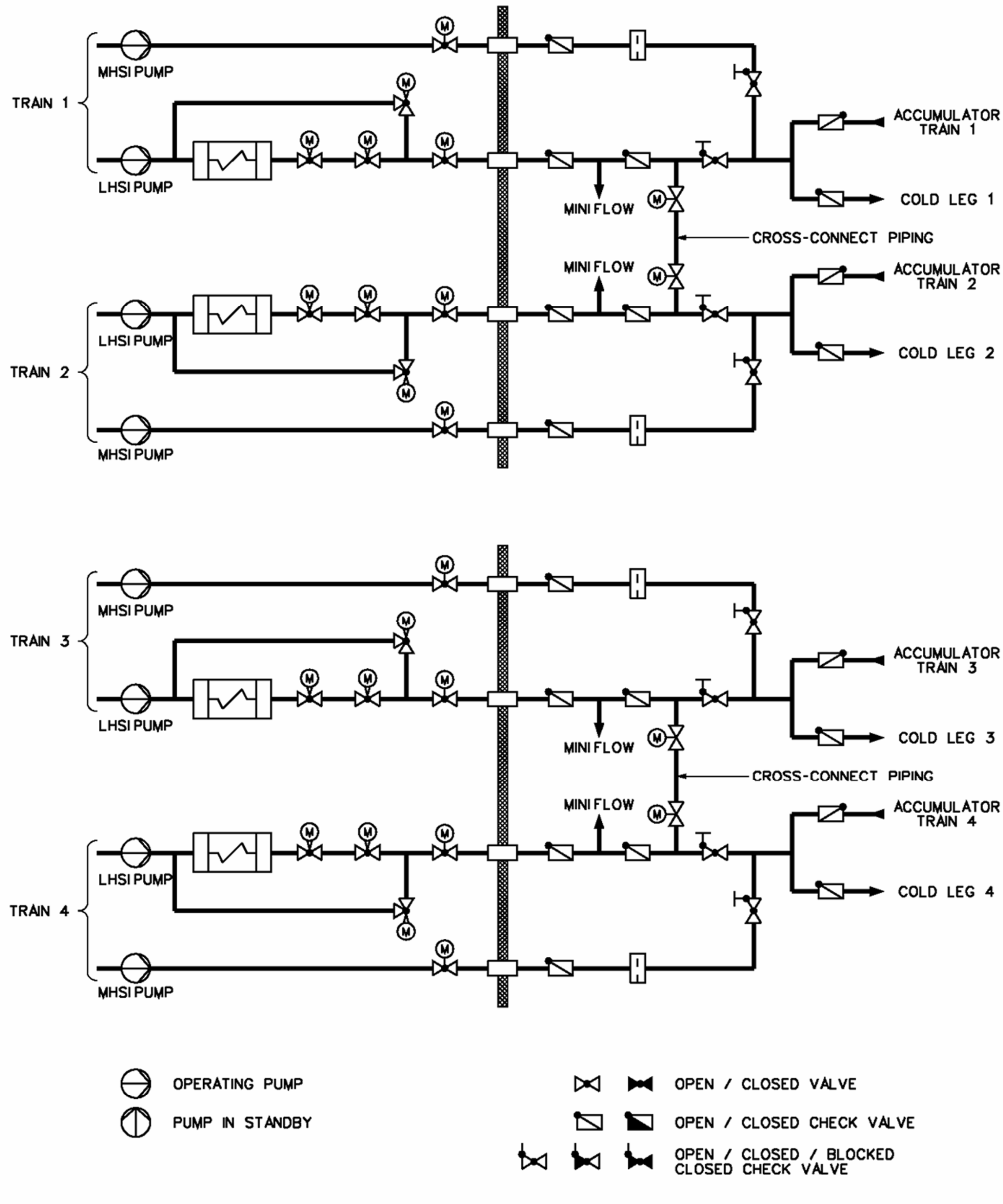


**Figure 2-3 Cross-Connect Piping Location**



# 3.0 LARGE BREAK LOCA CODES

The RLBLOCA methodology consists of the following computer codes, which are NRC-approved within the context of the RLBLOCA evaluation model:

- RODEX3A for computation of the initial fuel properties to be used by the fuel performance models in S-RELAP5 for computation of stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 for system thermal-hydraulic calculations. Containment back-pressure calculations are performed by an ICECON module (based on CONTEMPT LT-22) within S-RELAP5.

## 3.1 *RODEX3A*

RODEX3A is used to compute the initial fuel state at the burnup for the RLBLOCA case being considered. The RODEX3A calculations are used to generate fuel data to transfer to the downstream S-RELAP5 calculations. These calculations are performed primarily to determine the fuel-stored energy using best-estimate values for the axial power shape. Values of these parameters are time-in-cycle-dependent. The result of the RODEX3A calculations is a binary file of fuel rod data that is automatically transferred to the S-RELAP5 steady-state initialization process.

The phenomena modeled in the full RODEX3A code can be divided into two general categories: burnup-dependent effects and power-dependent effects. Burnup-dependent effects include cladding creep deformations, pellet densification and swelling deformations, and fission gas release. Power-dependent effects include thermal expansion of pellet and cladding, elastic deformation of cladding, gas pressure in the gap, gap width and conductance, and fuel and cladding thermal conductivity and heat capacity.

Based on the aforementioned categories, RODEX3A is used in two ways in the RLBLOCA methodology. First, the full RODEX3A code is used to expose the fuel rods being modeled through a desired power and burnup history. This exposure analysis treats both burnup and power-dependent effects.

Second, a subset of RODEX3A is included in S-RELAP5 to calculate only power-dependent effects. The values used for permanent phenomena are obtained from the full RODEX3A exposure analysis and are treated as constant values thereafter. Power-dependent phenomena are updated by the RODEX3A models in S-RELAP5 throughout both steady-state and transient analyses. The data transferred from the full RODEX3A calculation to S-RELAP5 describes the fuel at zero power. The steady-state S-RELAP5 calculation establishes the fuel state at power.

The fuel regions for RODEX3A calculation are the hot rod; the hot assembly; six high-powered surrounding assemblies; the average core, which in the U.S. EPR design represents 186 assemblies; and the low-powered, outer ring of assemblies, which represents 48 assemblies. Additional fuel regions are added for each gadolinia rod concentration. These regions correspond to the regions modeled in the S-RELAP5 input. That is, there is a one-to-one correspondence between the fuel rods modeled in RODEX3A and those modeled in S-RELAP5. The RODEX3A input for each region is consolidated so that only one RODEX3A computer run is required for each case in the RLBLOCA analysis.

## 3.2 *S-RELAP5*

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

The two-fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on another are accounted for by interfacial friction and heat and mass transfer interaction terms in the conservation 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 enable accurate accounting for physical dimensions and the dominant phenomena expected during LBLOCA. The basic building blocks for modeling are the hydraulic volumes for fluid paths and the heat structures for heat transfer surfaces. In addition, special purpose components exist to represent specific components such as the pumps or the steam generator separators. Plant geometry is modeled at the resolution necessary to resolve the flow field and the phenomena being modeled within practical computational limitations.

A typical calculation for each of the “sampled” cases 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 accommodate operation within plant technical specifications or plant-specific measured data. Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by simulating a break in the cold leg, at the pump discharge of the loop with the pressurizer. The break is located close to the reactor vessel, between the Emergency Core Cooling System (ECCS) injection nozzle and the cold leg nozzle. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5.

### 3.2.1 *S-RELAP5/RODEX3A Fuel Model*

As described in Section 3.1, the S-RELAP5/RODEX3A model does not calculate the burnup response of the fuel. Instead, fuel conditions of interest at the burnup are transferred via a binary data file from RODEX3A to S-RELAP5, establishing the initial state of the fuel prior to the transient.

The RODEX3A models in S-RELAP5 are identical to those in the RODEX3A code except for the following:

- The cladding thermal property routine was changed to compute properties at the higher temperatures possible in a transient.
- A cladding ballooning and rupture strain model was added to calculate deformations that can occur at elevated cladding temperatures when the internal rod pressure exceeds the coolant channel pressure. The RLBLOCA Evaluation

Model (EM) conservatively neglects swelling and rupture.

- The RODEX3A quasi-steady-state temperature solution algorithm was replaced by the S-RELAP5 transient temperature solution algorithm. The S-RELAP5 algorithm accounts for the thermal capacitance of the fuel.

The RODEX3A models in S-RELAP5 calculate fuel thermal properties at each time step. The S-RELAP5 thermal conduction solution uses thermal conductivities from RODEX3A to calculate new fuel temperatures each time step. RODEX3A uses new fuel temperatures from S-RELAP5 conduction solution to calculate fuel thermal properties for the next time step.

### 3.3 *ICECON Containment Module*

The S-RELAP5 ICECON containment module and the U.S. EPR containment model are described in Appendix B.

# 4.0 LARGE BREAK LOCA EVALUATION MODEL

The AREVA NP RLBLOCA EM is a best-estimate methodology formulated using nonparametric statistics. The methodology follows the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology (Reference 3). The CSAU method outlines an approach for defining and qualifying a best-estimate thermal-hydraulic code and quantifies the uncertainties in a LOCA analysis. Some three dozen key phenomenological and plant parameters are randomly sampled (see Table 4-1) for each case in the case-set. In the RLBLOCA EM, Peak Cladding Temperature (PCT) is predicted with greater than 95 percent probability and 95 percent confidence. The computer code is S-RELAP5, an AREVA NP-developed code that is based on INEL's RELAP5/MOD2 and /MOD3 code series. The methodology complies with 10CFR50.46 requirements.

The methods used in the application of S-RELAP5 to large break LOCA are fully described in Reference 1. A detailed assessment of this computer code was made through comparisons to experimental data. These assessments were used to develop quantitative estimates of the ability of the code to predict important physical phenomena in a PWR large break LOCA. The final step of the RLBLOCA methodology is to combine the uncertainties related to the code and plant parameters, as well as estimates of the peak cladding temperature, peak local oxidation, and core-wide oxidation. The steps taken to derive the uncertainty estimates are summarized below:

- Base Plant Input File Development

  First, base RODEX3A and S-RELAP5 input files for the plant, including the containment input file, are developed based on plant-specific information. Code input development guidelines are applied to make the model nodalization consistent with the model nodalization used in the code validation.

- Sampled Case Development

  The nonparametric statistical approach requires that many "sampled" cases be created and processed. For every RODEX3A and S-RELAP5 input created, including the ICECON containment input file, each key LOCA parameter is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided through plant technical specifications, data, etc.). The key LOCA parameters are listed in Table 4-1. Table 4-1 includes both parameters related to LOCA phenomena (based on the Phenomena Identification and Ranking Table (PIRT) provided in Reference 1) and parameters related to plant operation.

- Determination of ECCS Adequacy

  The adequacy of the ECCS is demonstrated when the PCT, peak local oxidation, and total oxidation satisfy the criteria set forth in 10CFR50.46. The coolable geometry and long-term cooling requirements are not addressed in this report.

## 4.1 *Modifications to RLBLOCA EM*

As noted in Section 1.0, the purpose of this report is to describe and demonstrate the applicability of AREVA NP's Realistic LBLOCA Evaluation Model (RLBLOCA EM, Reference 1) to the U.S. EPR reactor. Currently, the methodology is approved for application to Westinghouse 3- and 4-loop plants and Combustion Engineering (CE)-designed plants. Changes to the approved methodology made for use on the U.S. EPR plant are described below. The RLBLOCA EM with the noted changes relative to the methodology approved in EMF-2103(P)(A) is applicable to the U.S. EPR design because of its similarity to current 4-loop plants in design, geometry, functionality, and phenomenological response to a LBLOCA.

- Core Power

  The assumed reactor core power for the U.S. EPR realistic large break loss-of-coolant accident is 4612 MWt. This value represents the plant-rated thermal power of 4590 MWt with a maximum power measurement uncertainty of 0.48 percent (22 MWt) added to the rated thermal power. The value of 0.48 percent is based on the use of a Caldon CheckPlus™ ultrasonic flow meter (UFM) to measure main feedwater flow. Use of the Caldon CheckPlus™ UFM is approved as noted in NRC Regulatory Issue Summary 2007-24. The U.S. EPR design bases consider a maximum heat balance measurement uncertainty of ±22 MWt (0.48 percent or ≈0.5 percent of rated power). This uncertainty was verified by a calculation of core thermal power with a secondary side heat balance. The reactor core power for the U.S. EPR RLBLOCA analysis is not sampled.

- Quenching Criteria

  The RLBLOCA analysis is 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 900ºF before the rod is allowed to quench.

- Modification to the Application of Forslund-Rohsenow

  The RLBLOCA analysis is 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.

- Treatment of Split Breaks

  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 breaks range in area from 10 percent of $A_{PIPE}$ to twice the cross-sectional area of the 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.

- Increased Number of Sampled Cases

  The RLBLOCA analysis uses 124 cases instead of the 59 cited in the Reference 1 methodology. The evaluation extracts the figure of merit for each of three acceptance criteria of 10CFR50.46; i.e., peak clad temperature, local oxidation, and total hydrogen generation from the 124$^{th}$ ordered case using separate ordering for each of the three parameters.

- Treatment of Offsite Power

  GDC 35 states that the emergency core cooling system must function for both onsite power available (offsite power unavailable) and offsite power available (onsite power unavailable). By design, there is no significant difference in results between the loss-of-offsite power (LOOP) and non-LOOP cases for the U.S. EPR reactor. The reactor is designed with an automatic reactor-coolant pump trip on coincident safety-injection signal and low-RCP differential pressure. This feature causes the reactor coolant pumps to trip in the event of a LOCA, even if offsite power is available. Furthermore, the LOOP condition produces conservative PCT results because the delays for commencing ECCS injection are greater than those in the non-LOOP condition. Therefore, this analysis assumes only LOOP.

- Decay Heat

  As described in Appendix C, this analysis treats total decay heat with a standard deviation uncertainty of ±2 percent. Appendix C also explains that the assumptions of infinite operating time at full power, all fission from U-235, and 200 MeV/fission (conservatively low value, which yields a higher fission rate and, thus, more fission products) support the overall adequacy of the RLBLOCA U.S. EPR decay heat modeling.

- Initial Stored Energy – Treatment of Average and Peripheral Assemblies

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**Figure 4-1 Comparison of RODEX3A Initial Stored Energy Adjustment Methods**

**Figure 4-2   Comparison of RODEX3A Adjusted Values (New Adjustment vs. Reference 1 Methodology)**

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- Downcomer Boiling

  The Reference 1 RLBLOCA EM was changed to increase the amount of cold leg condensation calculated in agreement with test data. Biasing towards saturated fluid conditions at the downcomer (DC) entrance provides an appropriate DC fluid temperature for prediction, which increases the potential for DC boiling.

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## 4.2 *Large Break LOCA Scenario*

The large break LOCA event is defined in Section 15.6.5 of the U.S. Nuclear Regulatory Commission's Standard Review Plan (SRP) (Reference 4) as follows:

"Loss-of-coolant accidents (LOCA) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished."

An LBLOCA is initiated by a postulated break in the RCS piping. Based on numerous industry studies, the limiting break location for current PWRs has been shown consistently to occur in the cold leg piping between the reactor coolant pump and the reactor vessel; nothing in the U.S. EPR design invalidates that conclusion. The plant is assumed to be operating normally at full power prior to the accident. A break in the cold leg piping, downstream from the pump, is assumed to open instantaneously. A rapid depressurization[1] of the primary system occurs, along with a core-flow stagnation and reversal. RCS depressurization, together with the core-flow stagnation and reversal, causes the fuel rods to experience departure from nucleate boiling (DNB). Subsequently, the limiting fuel rods are cooled by film and transition boiling heat transfer. Coolant voiding creates a strong negative reactivity effect and core fission ends. As heat transfer from the fuel rods is reduced, cladding temperatures rise. A reactor trip signal is initiated when the pressurizer or hot leg low-pressure trip setpoint is reached. For RLBLOCA analyses, reactor trip is conservatively neglected. The reactor is rapidly shut down via core coolant voiding.

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

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[1] Although the automatic partial cooldown system of the U.S. EPR design is available to cool and depressurize the RCS, it is not modeled. Per the RLBLOCA EM, the steam generators are conservatively isolated at break initiation.

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

Mitigation of the LBLOCA begins when the SIS is actuated on very low pressurizer pressure. A worst single failure is assumed for ECCS safety analysis. This single failure is the loss of one ECCS-pumped injection train, which equates to the loss of one MHSI pump and one LHSI pump. In addition, another train of MHSI and LHSI is assumed to be unavailable because of maintenance. This means both LHSI cross-connect lines are open. All four accumulators are available.

An on-time start and normal lineups of the containment spray, fan coolers (if present), or other cooling mechanisms are assumed in the EM. For the U.S. EPR plant, containment sprays are unavailable for a number of hours following LOCA initiation. Also, the U.S. EPR design has no fan coolers; hence, neither sprays nor fan coolers are present in the RLBLOCA model.

Cooling of the IRWST is performed by the LHSI pumps and associated heat exchangers. The LHSI pumps take suction from the IRWST and circulate a portion of the flow through minimum flow lines back to the IRWST. The minimum flow lines branch from the LHSI line downstream of the LHSI heat exchanger; thus, all LHSI flow is cooled by the LHSI heat exchanger, including flow through the minimum flow lines. Cooling of the IRWST does not significantly impact containment pressure during an RLBLOCA event.

When the RCS pressure falls below the accumulator pressure, fluid from the accumulators is discharged into the cold legs. In the early delivery of accumulator water, high pressure and high break flow will cause some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core; thus, core heat transfer improves and cladding temperatures decrease. Eventually, the relatively large volume of accumulator water is exhausted, and core recovery relies on pumped SI coolant delivery.

As the accumulators empty, the nitrogen gas used to pressurize the accumulators enters the RCS. The release of nitrogen gas causes a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas is expelled through the break, the ECCS may temporarily be unable to sustain full core cooling because of the core decay heat and the higher steam temperatures created by quenching lower portions of the core. Fuel rod cladding temperatures increase for a short period until additional energy is removed from the core by low-pressure safety injection, which is facilitated by continued decay heat reduction. Steam generated from fuel rod rewet entrains liquid and is carried around the loop before being vented out the break. The resistance of this flow path to the steam flow is balanced by the driving force of water filling the downcomer. It acts to retard the progression of core reflooding and postpones core-wide cooling.

Within minutes of accident initiation, core reflood progresses 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 low head safety injection.

## 4.3 *LOCA Acceptance Criteria*

A LBLOCA event is part of the LOCA definition in Section 15.6.5 of the U.S. Nuclear Regulatory Commission’s SRP (Reference 4). It is classified as a postulated accident and a Condition IV event. It is not expected to occur during the lifetime of the plant; however, it is considered a design basis accident.

The LBLOCA acceptance criteria, as stated in 10CFR50.46, are:

- The calculated fuel element cladding temperature shall not exceed 2,200 °F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum region, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The RLBLOCA model described in this report is used to demonstrate compliance with the first three criteria of 10 CFR50.46.

## 4.4 *Cases Analyzed*

In contrast to the Reference 1 RLBLOCA EM, which defines an analysis as a case set comprising a minimum of 59 individual cases, the U.S. EPR RLBLOCA analysis case set comprises 124 individual cases. Per the EM, all breaks are located at the pump discharge, the limiting location in the primary system. The values for the sampled parameters are chosen randomly for each case within a specified range based on plant operating limits and uncertainties.

### 4.4.1 *Initial Cycle Fuel*

Experiments and analysis have shown that fuel peak cladding temperatures in limiting conditions are sensitive to fuel rod power and stored energy. The AREVA NP RLBLOCA methodology treats fuel rod power and stored energy in a best-estimate manner accounting for plant technical specifications. The occurrence of fuel pin swelling and rupture has been shown experimentally and analytically to decrease local cladding temperature. This effect is conservatively neglected in the model.

The AREVA NP RLBLOCA methodology uses the statistical sampling of the time in cycle for fresh assemblies only. Implementation of a burnup-dependent model addresses the Regulatory Guide 1.157 statement that

> "The steady-state temperature distribution and stored energy in the fuel before the postulated accident should be calculated in a best-estimate manner for the assumed initial conditions, fuel conditions, and operating history."

Core loading experience supports the premise that $UO_2$-based fuel assemblies tend to give limiting peak-assembly powers during their initial cycle. Cycle-to-cycle burnup will impact PCT by changes in power and stored energy. Both of these factors rank high in the phenomenological importance and ranking table in Reference 1. Stored energy is strongly correlated to fuel rod power, and in general they both decrease with burnup, resulting in lower PCTs. However, degradation in fuel thermal-conductivity and changes in the axial power shape create the opportunity for burned or gadolinia-bearing fuel assemblies to become limiting. A reload check is performed to confirm that the first cycle fuel does produce the limiting PCT during LOCA. If this cannot be established, the analysis will include second-cycle fuel.

## 4.5 *Choice of Single Failure and Preventive Maintenance*

The U.S. EPR design contains four SIS/RHRS trains of pumped injection, each with its own diesel generator. The methodology assumes a conservative single failure—the failure of a train of Emergency Core Cooling (ECC) pumped injection, which for the U.S. EPR design equates to the loss of one MHSI pump and one LHSI pump. Total pump capacity is such that the four-train configuration allows performance of preventive maintenance on one complete train during normal operation. Hence, in addition to the single-failure loss of one complete SIS pumped injection train, a second train of pumped injection is assumed out of service for maintenance. The LBLOCA analysis, therefore, assumes only two of the four trains of pumped injection start and deliver flow.

One of the two operating trains is assumed conservatively to inject into the RCS cold leg with the break. Because the ECCS connection is near the break, all of the ECCS flow delivered to the broken RCS cold leg spills into the containment. A break in the ECCS line itself at the cold leg nozzle is less than 10 percent of the cold leg area and is analyzed as a small break LOCA.

As described in Section 2.2.2, the LHSI portions of the SIS have cross-connects that are open when a SIS/RHRS train is out of service for maintenance. The cross-connects connect Loops 1 and 2 and Loops 3 and 4. These connections make sure that at least one cold leg opposite the break provides LHSI flow to the downcomer. The intact cold leg with the active SIS is a sampled parameter in the RLBLOCA uncertainty analysis. The actual distribution of LHSI flow to the cold legs reflects the cross-connects and is determined dynamically by S-RELAP5 based on local fluid conditions. Because the cross-connects are upstream of the LHSI/MHSI junctions, only two cold legs (one broken and one intact) receive MHSI flow.

There are two reasons that accumulators are not selected as a single failure for RLBLOCA analysis:

1) Accumulators are passive devices. The only components located between the accumulator tanks and the respective RCS cold leg piping are check valves and motor-operated isolation valves. As a standard step in each plant startup, each isolation valve is stroked open, and the source of electric power is disconnected. On this basis, an inadvertently-closed isolation valve is not considered credible. Likewise, the accumulator check valves are regarded as passive and high-reliability components exempt from single-failure consideration in the short term. The check valves are treated as passive because there is no external force or interfacing system involved with their operation. Although check valves must open for them to deliver water, the pressure differential across the valves to open becomes increasingly large as the primary system depressurizes.

2) U.S. EPR Technical Specifications require all four ECCS accumulators to be operable.

## 4.6 *Initial Conditions and Key Input Parameters*

The S-RELAP5 sample problem model presented in Appendix A is based on U.S. EPR design information. The sample problem results provided in Appendix A are suitable for the intended purpose of illustrating expected plant LOCA performance using AREVA NP's previously approved RLBLOCA EM. Table 4-1 provides a list of the parameters sampled during the RLBLOCA analysis.

### 4.6.1 *Break Size*

Applying the RLBLOCA methodology to break sizes as low as the RLBLOCA EM cross-sectional break size of 0.1 x $A_{pipe}$ was reviewed to identify important phenomena and to verify that the phenomena were modeled properly. Some differences were identified between the larger breaks and those near the smallest applied break area. However, in all cases, the modeling approach incorporated within S-RELAP5 and the RLBLOCA methodology was verified as sufficient to evaluate pipe breaks near the lowest applied break area (0.1 x $A_{pipe}$).

A comparison between the RLBLOCA methodology and the SBLOCA modeling shows good agreement, indicating that at the transition either model is adequate to evaluate the LOCA. Thus, the RLBLOCA methodology can be applied to breaks as small as 0.1 x $A_{pipe}$ with the results correctly evaluating the LOCA consequences.

### 4.6.2 *Power Shape*

A sensitivity study documented in the Appendix B of Reference 1 showed that flatter radial power distribution profiles in the surrounding assemblies; average assemblies; and low-powered, outer ring of assemblies produce higher PCTs. As such, the procedure for ranging radial power is biased towards these flatter radial distributions to conservatively bound the selection of radial profiles.

### 4.7 *Equipment Status*

Equipment status is presented in Table 4-2.

#### 4.7.1 *Trips and Controls Credited in the RLBLOCA Analysis*

Under accident conditions, a reactor trip signal is generated when the pressurizer low-pressure trip setpoint is reached; however, this trip is conservatively neglected in a RLBLOCA analysis and the reactor is shutdown by core coolant voiding. Control rod insertion is not credited. A pumped safety-injection actuation signal is issued when the very-low-pressurizer pressure setpoint is reached; a maximum ECCS-pumped injection delay time is assumed. (The U.S. EPR design does not have a high-containment pressure trip to actuate pumped safety injection.) Accumulators automatically begin discharging into the cold legs once the primary system pressures fall below their pressure. The partial SG secondary-side cooldown system is conservatively not modeled for LBLOCA. The rapid depressurization of the primary system and steam generator isolation with break initiation preclude the need for its modeling.

The RLBLOCA EM samples LOOP, tripping the reactor coolant pumps at event initiation if LOOP is chosen. The U.S. EPR reactor uses a pump trip on low pump pressure difference (ΔP) in combination with a SIS actuation signal. If the safety injection actuation signal was generated and the pump ΔP falls below 75 percent of the ΔP across the pump in normal operations, the main coolant pumps are tripped. This feature causes the main coolant pumps to trip in the event of a LOCA even if offsite power is available. Therefore, there is no significant difference in the results between the LOOP and non-LOOP cases, and the U.S. EPR RLBLOCA analysis assumes only LOOP.

### 4.7.2 *Status of Key Plant Equipment*

The U.S. EPR reactor has four complete safety trains with each train comprising an accumulator, MHSI and LHSI pumped injection, and a diesel generator. The MHSI and LHSI injection lines tee into the accumulator line, which in turn connects to the cold leg piping downstream of the pump discharge. The LHSI has cross-connects that are opened when one safety train is down for maintenance. All four accumulators, passive devices, are functional. However, only two of the four trains of ECCS-pumped injection are assumed to function; the single failure removes one train and a second train is assumed down for maintenance. Containment sprays, part of the severe accident heat removal system, are unavailable for actuation until twelve hours after transient initiation and are not considered in the analysis.

**Table 4-1 Sampled Parameters in RLBLOCA EM**

| **Phenomenological** | |
|---|---|
| | Time in cycle (axial shape, rod properties, and burnup) |
| | Peaking factors |
| | Break type (guillotine versus split) |
| | Break size |
| | Critical flow discharge coefficients (break) |
| | Offsite power availability (fixed at LOOP for U.S. EPR design) |
| | Decay heat |
| | Critical flow discharge coefficients (surgeline) |
| | Initial upper head temperature |
| | Film boiling heat transfer |
| | Dispersed film boiling heat transfer |
| | Critical heat flux |
| | $T_{min}$ (intersection of film and transition boiling) |
| | Initial stored energy |
| | Downcomer hot wall effects |
| | Steam generator interfacial drag |
| | Condensation interphase heat transfer |
| | Metal-water reaction |
| **Plant**[1] | |
| | Core power (fixed at nominal power + uncertainty for U.S. EPR design) |
| | Initial flow rate |
| | Initial operating temperature |
| | Pressurizer pressure |
| | Pressurizer level |
| | Containment volume |
| | Containment temperature |
| | Accumulator pressure |
| | Accumulator system volume |
| | Intact cold leg with operational MHSI and LHSI |

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

**Table 4-2: Equipment Status**

| Plant Equipment or System | Status |
|---|---|
| SIS Actuation | SIS actuation is on the very low pressurizer pressure setpoint, 1667.9 psia (with an uncertainty of ± 25 psi for normal conditions and ± 55 psi for degraded conditions). |
| MHSI and LHSI | • One train out of service for preventive maintenance.<br>• One train out of service due to single failure.<br>• One MHSI pump to the broken cold leg.<br>• One LHSI pump to the broken cold leg.<br>• One MHSI pump to one of the intact cold legs (sampled).<br>• One LHSI pump to one of the intact cold legs (sampled – same cold leg receiving MHSI) and to another cold leg through a cross-connection. |
| Accumulators | All four accumulators are available. |
| Control Rod Scram | Rod insertion is not credited. |
| Reactor Coolant Pumps | The RCPs trip on LOOP or "on low ΔP over RCP and SIS signal," where the minimum ΔP over the RCP setpoint is defined as 75 percent of the nominal ΔP. LOOP occurs at the beginning of the transient. |
| Partial Cooldown | Per the RLBLOCA EM, SG isolation occurs at break initiation; hence, partial cooldown is not simulated. The S-RELAP5 model for the RLBLOCA analysis does not incorporate the partial cooldown feature. Neglecting the MSRT cooldown feature reduces the energy being removed from the primary system and, therefore, is conservative. Refer to Section 6.2. |
| Steam Generator Main Steam and Feedwater | Per the RLBLOCA EM, SG isolation occurs at break initiation. |

# 5.0 U.S. EPR LARGE BREAK PHENOMENA

The RLBLOCA EM was developed following the CSAU approach (Reference 3). A PIRT process was used to identify and rank key phenomena for each of the three phases—blowdown, refill, and reflood—of a large break LOCA transient (Reference 1, Table 3.4). The most important phenomena (ranking seven or higher), grouped by transient phase, are described in the following three sections, concluding that the U.S. EPR design would not change the outcome of the PIRT.

## 5.1 *Blowdown Phenomena*

- Fuel Rod Stored Energy: U.S. EPR fuel is the same, excepting active core length, as that used in current PWR plants analyzed by the RLBLOCA methodology. The longer core length is within the calculation capabilities of the codes (RODEX3A and S-RELAP5) and methodology to analyze, so the model parameters (see Table 4.19 in Reference 2) required in all RLBLOCA analyses are also applicable to the U.S. EPR design. The U.S. EPR design introduces no new methodological or phenomenological considerations with respect to fuel rod stored energy.

  The fuel rod stored energy treatment is described in Section 4.1.

- Core Departure from Nucleate Boiling (DNB): DNB is modeled in S-RELAP5 by the Biasi and modified Zuber Critical Heat Flux (CHF) correlations. The calculations are conservatively biased using a multiplier, and sensitivity studies concluded that DNB is not significant to LBLOCA PCT (see Table 4.1 in Reference 1). Hence, the correlations and multiplier are equally applicable to the U.S. EPR design. Further evidence of design applicability is shown in Appendix A, Table A-10, a plant-specific check for RLBLOCA core heat transfer range applicability. The U.S. EPR design introduces no new methodological or phenomenological considerations with respect to core DNB.

- Core Post-CHF Heat Transfer: Core post-CHF heat transfer was assessed by comparing Thermal Hydraulic Test Facility (THTF) and Full Length Emergency Cooling Heat Transfer (FLECHT)-Separate Effects Test (SEASET) test data with S-RELAP5 (Reference 1, Sections 4.3.1.1 and 4.3.3.2.5). The results defined the uncertainty ranges used in the RLBLOCA methodology. Those ranges are equally applicable to U.S. EPR calculations. There is nothing about the U.S. EPR design that would change or invalidate the Oak Ridge National Laboratory Thermal-Hydraulic Test Facility benchmark; again, note the comparisons presented in Appendix A, Table A-10. The U.S. EPR design introduces no new methodological or phenomenological considerations with respect to core post-CHF heat transfer.

- Rewet: S-RELAP5 benchmarks of test data exhibiting blowdown rewetting (Reference 1, Section 4.3.2.1.4) conservatively predicted the measured PCTs. Rewetting was not predicted everywhere it was observed; however, the calculated clad temperatures followed the data trends, and the predicted PCT was 1350°F compared to the measured PCT of 1236°F. The U.S. EPR design introduces no new methodology or phenomenological considerations with respect to blowdown rewet (quench).

  RLBLOCA restricts blowdown quench. A blowdown quench is characterized by a temperature reduction of the PCT node to saturation temperature during the blowdown period. This restriction applies to all PWRs, including the U.S. EPR reactor.

  The phenomenon of blowdown rewet (quench) is described further in Appendix A, Section A.3.0, as part of the LBLOCA sample calculations.

- Core Flow Reversal and Stagnation: Core flow reversal and stagnation are the result of break size and the rate of coolant loss versus the rate of coolant injection from the ECC systems. The methodology treats these items by ranging associated parameters such as break size; break coefficient; break type; RPV upper-head temperature; and accumulator pressure, volume and temperature (Reference 1, Section 4.3.3.1.3). For the U.S. EPR design, ranging of these parameters is still appropriate; the plant configuration presents no unique features in that regard. The design introduces no new methodological or phenomenological considerations with respect to core flow reversal and stagnation.

- Critical Flow at the Break: Critical flow at the break was assessed by comparison of S-RELAP5 with full-scale critical flow tests at the Marviken facility (Reference 1, Section 4.3.3.2.7). S-RELAP5 code predictions agreed well with the test data (Reference 1, Figure 4.99). Moreover, the U.S. EPR break geometry and fluid conditions are similar to those of current PWRs for which the RLBLOCA methodology applies. Hence, S-RELAP5 is capable of calculating critical flow, and the critical flow uncertainty parameters described in Reference 1 are also applicable to the U.S. EPR reactor.

- Flow Split Between Loops: The flow split between loops is controlled in the methodology by independently ranging the discharge coefficients of the two broken ends of the cold leg pipe in the RLBLOCA uncertainty analysis (Reference 1, Section 4.3.3.2.7). Ranging as specified in the methodology will be applied to the U.S. EPR design. Therefore, the code and methodology are also applicable to the U.S. EPR reactor.

## 5.2 *Refill Phenomena*

- Core Post-CHF Heat Transfer: Core post-CHF heat transfer was assessed by comparing THTF and FLECHT-SEASET test data with S-RELAP5 predictions (Reference 1, Sections 4.3.1.1 and 4.3.3.2.5). The results defined uncertainty ranges used in the RLBLOCA methodology. Those ranges are equally applicable to U.S. EPR calculations. There is nothing about the U.S. EPR design that would change or invalidate the above-mentioned benchmarks. Note the comparisons presented in Appendix A, Table A-10, for further evidence of applicability. The U.S. EPR design introduces no new methodological or phenomenological considerations with respect to core post-CHF heat transfer.

- Accumulator Discharge: Accumulator differences relative to current plants are discussed in Section 6.2. The differences regarding EM applicability are inconsequential to the U.S. EPR plant.

- Downcomer Entrainment/De-entrainment and Countercurrent, Slug and Non-equilibrium Flow: The radial width and hydraulic diameter of the lower downcomer region are somewhat larger than current 4-loop plants. Both small-scale (Loss-of-Fluid Test (LOFT) and Semiscale and full-scale (Upper Plenum Test Facility (UPTF)) data were used to evaluate ECC water penetration into the downcomer. The tests demonstrate that, with the RLBLOCA methodology plant lower plenum nodalization, very little water was delivered to the downcomer and lower plenum during the period when the intact cold legs were filling with ECC water. Only after the cold legs were filled did a significant amount of ECC penetration to the downcomer and lower plenum begin. With the RLBLOCA methodology plant lower plenum nodalization, the code conservatively predicted the entrainment of ECC water from the intact cold legs to the broken cold leg during the cold leg filling period, and correctly predicted full or partial entrainment of ECC water to the broken cold leg during the lower plenum refill period (Reference 1, Section 4.3.1.11.1). Due to the use of both small- and full-scale data, the effects of scale were taken into account in evaluating downcomer entrainment effects predicted by S-RELAP5 as it will be applied for the U.S. EPR design. Therefore, the evaluations are also applicable to U.S. EPR plants.

The dominant downcomer LBLOCA phenomena (condensation, hot wall effects, multidimensional flow, counter-current flow limit (CCFL), and entrainment) affect the refill period. These phenomena primarily influence the duration of ECCS bypass. It was shown that there is little sensitivity of these phenomena to downcomer nodalization for the U.S. EPR plant.

- Downcomer Condensation: The RLBLOCA EM was changed to increase the amount of cold leg condensation calculated in agreement with test data, as described in Section 4.1. The interfacial condensation heat-transfer coefficient applied in the cold legs and downcomer is a parameter that is varied in the RLBLOCA uncertainty analysis (Reference 1, Table 4.19).

- Downcomer 3-D Effects: Downcomer 3-D effects were evaluated by comparison of S-RELAP5 to the Upper Plenum Test Facility tests (Reference 1, Section 4.3.1.11). Results of those tests indicated that S-RELAP5 can calculate the 3-D effects, and that refilling of the downcomer is conservatively predicted. The codes and methodology are also capable of modeling the phenomenon for the U.S. EPR design.

- Loop Flow Oscillations: Comparisons of S-RELAP5 to UPTF Test 8 demonstrated that S-RELAP5 is capable of calculating the appropriate phenomena in a full-scale facility (Reference 1, Section 4.4.2.2.8). There is nothing about the U.S. EPR design that would change or invalidate the above-mentioned benchmark. Thus, the codes and methodology are also capable of modeling the phenomena for the design.

- Flow Split Between Loops: Flow split between the loops is controlled in the methodology by independently ranging the discharge coefficients of the two broken ends of the cold leg pipe in the RLBLOCA uncertainty analysis (Reference 1, Section 4.3.3.2.7). Ranging as specified in the methodology will be applied to the U.S. EPR design; therefore, the code and methodology are also applicable.

## 5.3 *Reflood Phenomena*

- Fuel Rod Oxidation: Fuel parameters affecting fuel rod oxidation are the same for the U.S. EPR design as for the applicable current PWRs.

- Fuel Rod Decay Heat: Fuel parameters affecting fuel rod decay heat are the same for the U.S. EPR design as for the applicable current PWRs.

- Core Post-CHF: Core Post-CHF was assessed by comparing THTF and FLECHT-SEASET test data with S-RELAP5. The results defined uncertainty ranges used in the RLBLOCA methodology (Reference 1, Section 4.3.3.2.5). Those ranges will be applied to U.S. EPR calculations. Thus, the Post-CHF model is applicable to the U.S. EPR design. Refer to Appendix A, Table A-10, for further validation.

- Core Reflood Heat Transfer and Quench: During reflood, S-RELAP5 maps the appropriate reflood heat transfer regime along the axis of the core. This model was assessed as a best-estimate model against both separate effects and integral effects tests, including FLECHT-SEASET, Cylindrical Core Test Facility (CCTF), Slab Core Test Facility (SCTF), LOFT and Semiscale. The results of those evaluations demonstrated that the models for the phenomena used in S-RELAP5 can be applied to full-scale PWR LBLOCA events (Reference 1, Section 4.4.2.1). The S-RELAP5 model is conservatively biased in sampling of the heat transfer coefficients. Core quenching is no different for the U.S. EPR reactor than for similar PWRs. The models used in S-RELAP5 will be applied to the U.S. EPR reactor consistent with their application in current PWRs (refer to Appendix A, Table A-10). Therefore, the code and methodology are capable of modeling the phenomena for the U.S. EPR design.

- Core 3-D Flow, Void Distribution and Generation: Code assessments demonstrated best-estimate performance of S-RELAP5 for core 3-D flow, void distribution and generation (Reference 1, Section 4.3.3.1.1). Core 3-D effects are influenced by the initial power distributions and the size of the break. Power distributions and break size, type, and discharge coefficient are randomly varied in the RLBLOCA uncertainty analysis—as they will be for the U.S. EPR. The codes and methodology are both capable of modeling the phenomena for the U.S. EPR design.

- Core Entrainment/De-entrainment: Reference 1, Section 4.3.3.1.2, notes that liquid entrainment in the core was demonstrated to be conservatively calculated by the S-RELAP5 code. As noted in Reference 1, Section 4.4.2.2.2, the determinants of the model applicability to PWR LBLOCA events for models affecting core entrainment are primarily local and, in the core, are principally related to the conditions within the flow channel between the fuel rods. The U.S. EPR flow channels are within the range of plant types applicable to the RLBLOCA methodology. The tests used in assessing core entrainment used full-length (12-foot) fuel rods, and comparisons of S-RELAP5 to the data demonstrated that the core entrainment model in S-RELAP5 is conservative and scales suitably to full-scale PWR LBLOCA events. Since the U.S. EPR design conforms to the nodalization and modeling specified in the methodology, the codes and methodology are capable of modeling the phenomena for the design.

- Upper Plenum Entrainment/De-entrainment: The U.S. EPR reactor has a primary system volume-to-core power ratio similar to that of current PWR designs. Since the upper plenum of the design is within the range of applicable plant types, and the design conforms to the approach and modeling prescribed in the methodology, it is concluded that S-RELAP5 and the methodology are also capable of calculating the phenomena for the U.S. EPR reactor.

- Upper Plenum Draining and Fall-Back: The methodology conservatively disallows radial cross-flow in the first axial level of the upper plenum above the hot assembly, creating a virtual standpipe that restricts fall-back into the hot assembly. Since the U.S. EPR design conforms to the approach and modeling prescribed in the methodology, it is concluded that S-RELAP5 and the methodology are also capable of calculating the phenomena for the design.

- Steam Generator Steam Binding: Containment pressure during a LBLOCA is generally higher for the U.S. EPR reactor than for current PWR plants. A major impact of higher containment pressure is that it will tend to reduce steam binding in the steam generators due to higher steam density. As shown in the PIRT, steam binding has a relatively strong impact during the reflood phase, and reducing steam binding tends to reduce PCT.

  High steam-generator operating pressure and temperature tend to increase steam binding. This phenomenon was shown to be conservatively predicted by S-RELAP5 (Reference 1, Section 4.3.1.11.3), and those tendencies will be the same for the U.S. EPR reactor as they are for current PWRs. Moreover, the design response is conservatively modeled because the SG partial cooldown feature is not credited.

  Due to the conservative range of variation in containment pressure and conservative modeling in the methodology and codes, steam binding is conservatively biased in the RLBLOCA methodology and codes as they are applied to current PWRs and to the U.S. EPR plant.

- RCP Differential Pressure Form Loss: U.S. EPR pump-specific homologous curves are used in accordance with the methodology specifications. Thus, the code and methodology are both applicable to the U.S. EPR design.
  Because the design includes a safety-grade pump trip, the RLBLOCA EM evaluates only scenarios in which the RCPs are depowered at time of reactor trip. The S-RELAP5 model calculates pump speed dynamically with consideration of flywheel inertia and dynamic forces from friction and fluid interaction. Pump seizure is not considered part of a best-estimate LBLOCA scenario and for this reason is not considered in the RLBLOCA EM.

- Noncondensable Gas: Modeling of the U.S. EPR accumulators conforms to the methodology requirements. As noted in Section 6.2, the nitrogen pressurization of the U.S. EPR accumulators is within the range of applicability to current PWRs. Thus, the S-RELAP5 code and methodology are equally capable of calculating the noncondensable gas effects for the U.S. EPR design.

- Accumulator Discharge: Accumulator differences are discussed in Section 6.2.

- Downcomer Boiling: A study was performed to demonstrate acceptability of the nodalization to capture downcomer boiling for the U.S. EPR design. The sensitivity study employed the condensation multipliers as described in the discussion of downcomer boiling in Section 4.0. The study comprised a radial-mesh sensitivity and an axial-mesh sensitivity. The results of the radial mesh portion are consistent with the exact solution study for generic 4-loop PWRs. The results of the axial-mesh sensitivity found that the current scheme, [ ] is sufficient to accurately resolve void distributions within the downcomer. Thus, the current downcomer model is sufficient to predict the downcomer driving head and the resolution of the downcomer boiling effects.

- Reflood Oscillations: The sensitivity of cladding temperatures to flow oscillations is assessed through both analysis and evaluation of experiments. In the gravity-fed LOFT and CCTF considered in the developmental assessment of S-RELAP5, the calculated flow oscillations did not lead to premature quenching of the hot fuel rods (and hence an underestimation of the peak cladding temperature). To complement this existing code developmental assessment of flow oscillations, a sensitivity study using an S-RELAP5 model of the FLECHT-SEASET facility was performed, specifically isolating core flow oscillations at a cycle period and magnitude similar to those seen in the U.S. EPR RLBLOCA analysis. The S-RELAP5 model predicted higher cladding temperatures when flow oscillations occur.

The review of flow oscillations using code assessment and sensitivity studies confirms that the current treatment of this phenomenon with the S-RELAP5 RLBLOCA methodology is sufficient and conservatively predicts cladding temperatures.

Based on the above considerations, it is concluded that the existing RLBLOCA methodology is suitable for simulating the various phenomena that occur during a large break transient as it will be analyzed for the U.S. EPR design. The design introduces no new phenomenological considerations that would require RLBLOCA EM modifications.

# 6.0 S-RELAP5 CODE VALIDATION FOR U.S. EPR LARGE BREAK LOCA ANALYSES

This section covers validation of the RLBLOCA S-RELAP5-based EM for application to the U.S. EPR design. It is concluded that the design contains no features or transient phenomena requiring additional benchmarks or methodology changes.

## 6.1 *S-RELAP5 Acceptance for LBLOCA Analysis*

Table 5-1 summarizes the benchmarks used to assess or otherwise confirm that S-RELAP5 adequately simulates the important LBLOCA phenomena discussed in Section 5.0. No additional benchmarks are required to demonstrate U.S. EPR applicability.

## 6.2 *S-RELAP5 Acceptability for U.S. EPR LBLOCA Analysis*

This section discusses design differences between the U.S. EPR reactor and current PWR plants that are relevant to Chapter 15 RLBLOCA safety analyses. Disposition arguments that justify the applicability of the RLBLOCA methodology to the U.S. EPR design are provided for those differences that could potentially have a significant impact on the valid application of the codes, model, or other aspects of the EM.

- <u>High Containment Pressure</u>: For large break LOCA, the U.S. EPR transient containment pressure is expected to be higher than in current plants. This is because the U.S. EPR reactor does not have fan coolers, and containment sprays (reserved for severe accidents) are not activated until approximately 12 hours after transient initiation. Containment pressure is a significant PIRT-identified factor during refill and reflood. The methodology treats containment pressure as a statistically varied parameter by randomly sampling containment volume. The magnitude of the pressure does not require ICECON or other methodology changes. The RLBLOCA EM is adequate and appropriate. The higher U.S. EPR containment pressures are within the S-RELAP5 (ICECON module) code and methodology modeling capabilities.

- Containment Heat Removal System: As noted previously, the U.S. EPR containment design does not have fan coolers; neither does it activate containment sprays for LBLOCA. Though the absence of containment spray in the ICECON calculations differs from the RLBLOCA analyses for most PWRs, it is within the code modeling capabilities and within the methodology. No ICECON or other methodology changes are required; the RLBLOCA EM remains applicable.

- In-containment Refueling Water Storage Tank: The IRWST is an open pool within the containment building that partly immerses a portion of the containment building structure. The open pool covers about two-thirds of the floor area at the bottom of the containment building. The IRWST functions as both the external water storage tanks and internal sumps of current PWRs. Additionally, there is a heat exchanger downstream of the LHSI pump that provides safety-grade cooling of the LHSI for the SIS. There is also a minimum flow line downstream of the heat exchanger that flows back to the IRWST and provides cooling of the IRWST, including during LOCA events.

  Being inside the containment, the IRWST water temperature variation is treated as the same as normal operational containment temperature, 100°F to 131°F. The RLBLOCA methodology requires the tank water temperature for pumped safety injection to be set equal to the Technical Specification maximum value. Conforming to that requirement, the energy contained within the total RCS liquid mass in the primary system after depressurization will be mixed with the IRWST water, using the Technical Specification maximum IRWST water temperature. This approach results in a conservatively elevated temperature for the pumped SIS water, above the technical specification maximum IRWST temperature. LHSI heat exchanger cooling is conservatively neglected. The outlined procedure provides a means of determining a maximum IRWST water temperature in compliance with EM requirements.

The RLBLOCA methodology also stipulates that containment cooling supplied by the IRWST not be modeled with the same initial temperature assumed for safety injection. For the U.S. EPR reactor, which does not make use of sprays, this relates to the heat transfer between the containment atmosphere and the IRWST water. For that calculation, cooled LHSI flow recirculated back to the IRWST is neglected. To the time of PCT, the amount of recirculation flow is negligible relative to the volume of fluid in the IRWST. Summarizing with regards to the IRWST, the RLBLOCA EM is capable of acceptably modeling the U.S. EPR configuration without change.

- MHSI: Unlike current PWRs, the U.S. EPR design uses MHSI pumps instead of high head safety injection (HHSI) pumps; however, the RLBLOCA EM is capable of modeling this configuration of ECC-pumped injection without change.

- SIS/RHRS: The SIS/RHRS has four trains; as long as all four trains are available, they are independent. If one train is down for maintenance, piping cross-connects are opened between Loops 1 and 2 and Loops 3 and 4, providing multiple injection points for the LHSI. Assuming a single failure and preventive maintenance, the U.S. EPR design has two remaining SIS/RHRS trains for pumped injection. One train injects into the broken cold leg and into an intact cold leg through a cross-connect, and the second train injects into an intact cold leg (sampled) and into another cold leg (which could be the broken leg) through a cross-connect. This configuration results in pumped injection into the cold leg(s) opposite the break and is similar to current PWR designs for which the RLBLOCA EM has been used. Therefore, the RLBLOCA EM is capable of modeling SIS pumped injection for the U.S. EPR design.

- Accumulators: U.S. EPR accumulators are configured similarly to those in current plants. Their construct is such that they are neither subject to a single failure nor allowed out of service for preventive maintenance. Thus, all four accumulators are available for accident mitigation. Their large capacity leads to a faster reactor vessel refill and higher flooding rates—all favorable trends that minimize the PCT.

S-RELAP5 was benchmarked against ACHILLES tests and shown not to over-predict the nitrogen-induced surge of water into the core and its resulting core cooling (Reference 1, Section 4.3.1.4). The tests demonstrate that the effects of nitrogen transport, including that occurring in the U.S. EPR design, will be adequately predicted by S-RELAP5.

Condensation due to ECCS injection into the cold legs is also an important refill and reflood phenomenon. It was found to be appropriately treated by S-RELAP5 and the RLBLOCA methodology in benchmarks of the Westinghouse/EPRI 1/3-scale tests using the cold-leg nodalization specified by the methodology and the bias and uncertainty range of the interfacial condensation heat-transfer coefficient in the ECC/steam mixing process (Reference 1, Section 4.3.3.2.9). Pressure and fluid oscillations in the loops caused by ECC injection into the cold legs were evaluated via the full-scale UPTF Test 8 (Reference 1, Section 4.4.2.2.8). Since the U.S. EPR design conforms to the nodalization guidelines specified in the methodology, the range of tests from part-scale to full-scale demonstrate that the results and conclusions are applicable. The features of the U.S. EPR accumulators can be modeled appropriately using the RLBLOCA EM without modification.

- <u>Preventive Maintenance</u>: The ramifications of preventive maintenance were previously discussed in SIS/RHRS and Accumulator items, as well as in Section 4.5. It was concluded that the preventive maintenance impact on equipment operation was within the modeling and calculation capabilities of the RLBLOCA EM.

- <u>Large Primary System Component Sizing</u>: The larger (relative to current 4-loop plants) size of the U.S. EPR RCS and primary system components can affect the time to empty the pressurizer, the end of blowdown time, and the core bypass time. The sizing of U.S. EPR components is included in the S-RELAP5 model.

The larger RV downcomer was previously discussed in Section 5.2 and its scale found to be properly accounted for in the RLBLOCA EM. The U.S. EPR pressurizer volume relative to total RCS volume is larger than for current PWRs. This results in differences in the time to empty the pressurizer. However, the larger U.S. EPR pressurizer volume is included in the S-RELAP5 model, and the code is fully capable of predicting differences in event timing.

Thus, the effects of larger components are within the capabilities of the methodology and codes to analyze.

- <u>Large Reactor Vessel Free Volume between the Vessel Nozzles and Top of Active Core</u>: The added distance between the top of the active core and the RV nozzles relative to current PWRs provides a taller head for core reflooding during LBLOCA events. The S-RELAP5 model reflects the difference in core elevation relative to the vessel nozzle. This difference, relative to current plants, does not produce conditions that are outside the capability of the methodology and codes to analyze.

- <u>Heavy Reflector</u>: The U.S. EPR design differs from current PWRs in that it uses a heavy reflector—an all stainless-steel structure between the multi-cornered periphery of the core and the core barrel. The heavy reflector effectively takes the place of the core baffle and eliminates the need for a thermal shield or neutron pads. The location of the metal mass differs from thermal shields or neutron pads in that it is located inside the core barrel, rather than outside the core barrel in the downcomer. The metal mass and volume of the heavy reflector are modeled in S-RELAP5. The location of the larger metal mass between the core and core barrel relative to the core baffle is a plant-specific difference that is not significant to the LBLOCA event. Flow through the axial cooling holes that are used to cool the heavy reflector are included in the bypass flow modeled in S-RELAP5. These differences, while requiring minor noding changes relative to the noding for the current core baffle arrangement, are within the capability of the methodology and codes to analyze.

- Long Core: The RLBLOCA methodology does not impose limits on core height. To accommodate the 14-foot U.S. EPR core and preserve the level of detail in core modeling, the number of axial nodes was increased.

  The scalability of the RLBLOCA methodology, and the ability to model cores of different lengths, has been demonstrated by comparison of S-RELAP5 to experiments of different scales. These experiments include those performed in the Semiscale (1/1600 scale), LOFT (1/50 scale), CCTF (1/21 scale), and UPTF (1/1 scale) facilities. The scaling indicated for each facility is relative to a 4-loop plant. The facilities have core lengths ranging from 5.5 feet for LOFT and Semiscale to 12 feet for CCTF. Taken together, the experiments covered all three phases of the LBLOCA: blowdown, refill, and reflood. The Semiscale and LOFT experiments covered all three phases, the CCTF covered the reflood and to a lesser extent the refill phases, and the UPTF primarily addressed the refill phase. The good agreement between the experimental data and the calculation results for all of these facilities demonstrate that the methodology is scalable.

  The scalability experiments have a range of core heights that differs by more than a factor of 2. The 14-foot core represents only a small increase in core height (approximately 17 percent) relative to the 12-foot core included in the methodology assessments. Thus, the approved methodology is judged to be applicable to 14-foot cores.

- Fuel Rod Lower Plenum and Isolation Pellet: U.S. EPR fuel rods include a lower plenum without a plenum spring and a non-fuel isolation pellet that separate the active fuel pellets from the lower plenum. The RODEX3A code and RLBLOCA model include the capability of modeling the U.S. EPR fuel rod lower plenum without modification. The isolation pellet has no effect on the RODEX3A modeling of the fuel rod. Therefore, the fuel rod is within the capability of the methodology and codes to analyze properly.

- Partial Cooldown: The U.S. EPR steam generator partial cooldown system is designed to cool the primary system via the MSRTs, thereby lowering the RCS pressure during various events, including LOCA. The S-RELAP5 model for the RLBLOCA analysis does not incorporate the partial cooldown feature. Neglecting the MSRT cooldown feature reduces the energy being removed from the primary system and is, therefore, conservative.

- Steam Generators Axial Economizer: U.S. EPR axial economizer steam generators differ from those of current U.S. 4-loop plants. The U.S. EPR design (shown in Figure A-2) physically separates the lower half of the downcomer into a cold half and a hot half. Feedwater is injected into only the cold half of the downcomer, while about 90 percent of the hot recirculation fluid is deposited into the hot half of the downcomer. The hot and cold division is continued up through about two-thirds of the tube region. Separators and dryers are also somewhat different in size and location than in current U.S. plants. While noding changes are necessary to represent the U.S. EPR steam generator configuration properly, no differences introduce hardware, phenomena, or range of applicability issues not previously assessed during the development of the RLBLOCA methodology. The LBLOCA is insensitive to the treatment of the steam generators. The axial economizer and other steam generator design details are within the capabilities of the methodology to model and its codes to analyze.

- High Steam Generator Operating Pressure and Temperature: The U.S. EPR steam generators operate at higher pressure and temperature than typical PWRs. The higher steam generator operating pressures and temperatures tend to increase steam binding during reflood. As noted in Section 5.3, this difference is within the capability of the methodology and codes to analyze.

- RCP Trip on Low ΔP Over RCP and SIS signal: The RLBLOCA methodology statistically samples loss of offsite power, tripping the RCPs at event initiation for LOOP, but not tripping pumps if offsite power is available. For the U.S. EPR design, the "RCP Trip on Low ΔP Over RCP and SIS signal" (see Table 3-2) is applied in the RLBLOCA uncertainty analysis cases when offsite power is available. It would be nonconservative to continue supplying forced RCS flow. While a unique U.S. EPR feature, the trip occurs early in the event and presents no challenging or new analysis features. Thus, this trip is a plant-specific difference that is within the capability of the methodology and codes to model and analyze.

- Lack of SIS Initiation Trip on High Containment Pressure: Current U.S. PWRs typically have both high containment pressure and low pressurizer pressure trips to initiate SIS. Generally, the high containment pressure trip is first to actuate (usually within about one second after transient initiation) during LBLOCA. The U.S. EPR design does not have a high containment pressure trip. The RLBLOCA methodology provides for SIS initiation on either high containment pressure or low pressurizer pressure. For the U.S. EPR design, the SIS is initiated on low pressurizer pressure (see Table 3-2). The delayed (several seconds relative to a high containment pressure trip) SIS initiation has no significant effect on RLBLOCA cases with or without LOOP. This is a plant-specific difference that is within the capability of the RLBLOCA methodology and codes to analyze.

**Table 6-1: Assessment Matrix Tests and Phenomena Addressed**

# 7.0 CONCLUSIONS

The U.S. EPR plant was evaluated from a phenomenological viewpoint in Section 5.0 and from a design viewpoint in Section 6.0. The Section 5.0 review concluded that the U.S. EPR response during a LBLOCA involves no additional phenomena beyond those already considered by the existing RLBLOCA EM methodology (Reference 1). Moreover, the ranges of fluid conditions encountered are similar to those for current U.S. PWR plants and within the range of applicability of the EM methodology.

The Section 5 review identified design differences between the U.S. EPR reactor and current U.S. PWR plants. It was concluded that the features of the U.S. EPR design can be acceptably modeled and analyzed using the existing RLBLOCA EM in conjunction with the described changes.

In summary, the previously approved RLBLOCA EM in conjunction with the described changes is applicable to the U.S. EPR plant.

## 8.0 REFERENCES

1. EMF-2103(P)(A), Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” FANP Richland, Inc., April 2003.

2. ANP-10263(P)(A), Revision 0, “Codes and Methods Applicability Report for the U.S. EPR,” AREVA NP Inc., August 2007.

3. NUREG/CR-5249, EGG-2552, “Quantifying Reactor Safety Margins,” Technical Program Group, December 1989.

4. NUREG-0800, LWR edition, Revision 2, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, July 1987.

# APPENDIX A

# LARGE BREAK LOCA SAMPLE CALCULATIONS

# APPENDIX A

## TABLE OF CONTENTS



## List of Tables



## List of Figures

## A.1.0 *Introduction and Summary*

The purpose of this appendix is to describe the structure and implementation of an S-RELAP5-based large break LOCA plant model, termed a sample problem. It is based on AREVA NP's previously approved RLBLOCA evaluation model (Reference A-1). Accident behavior reported in this appendix is representative of the U.S. EPR final design analyzed under LBLOCA conditions. Sections 3 through 7 describe the RLBLOCA methodology and codes as well as a generalized large break LOCA scenario, and justify the application of the RLBLOCA EM without modification to a U.S. EPR plant. The following summarizes the application of the RLBLOCA methodology to the U.S. EPR design and the results of that application.

## A.2.0 *Application Analysis Results*

The U.S. EPR reactor is a 4-loop plant with U-tube steam generators, similar in most facets to the current generation of 4-loop PWR plants. It is designed to operate at a core thermal power of 4590 MWt. The steam generators include an axial economizer for optimum thermal efficiency. The plant contains four safety trains. Each train contains its own MHSI and LHSI pumps and diesel generator; per-pump injection rates are provided in Tables A-1 and A-2. Diesel start time is set consistent with the loss-of-offsite-power assumption for ECCS-pumped injection. The core is composed of 241, 17 x 17, thermal-hydraulically compatible fuel assemblies, containing $UO_2$ fuel rods with gadolinia enrichments up to 8 weight percent. The fuel rods and grids use AREVA NP's advanced M5® material. The active core is slightly less than 14 feet. The plant is bottom reflooded. The containment is a double-walled, cylindrical vessel with a domed head.

The S-RELAP5 RLBLOCA plant model specifically represents the reactor vessel with internals and core, hot and cold leg primary system piping, main reactor coolant pumps, pressurizer and pressurizer surge line, steam generator primary and secondary sides, and ECCS (both pumped injection and accumulators). The containment heat structures use the Uchida correlation with a multiplier of 1.7.[1] The tube plugging is a uniform 5 percent in all four steam generators. The analysis is for an equilibrium fuel cycle representative of an 18-month core. Table A-3 and Table A-4 list many of the important modeling parameters.

The U.S. EPR accumulator lines in Loops 1 and 4 are about 10 ft longer in total length than the corresponding lines in Loops 2 and 3. This difference is reflected in the S-RELAP5 model of the respective accumulator lines. The section of piping between the accumulator discharge nozzle and safety injection connection point consists of two sizes of pipe: the flow area of one is 0.7530 $ft^2$, the other is 0.5592 $ft^2$. To preserve the total volume and length of piping in the S-RELAP5 control volumes, an effective area is calculated: 0.7238 $ft^2$ for Loops 1 and 4 and 0.7140 $ft^2$ for Loops 2 and 3. Flow resistance is also preserved. The resulting model is a correct representation of the U.S. EPR accumulator system.

The U.S. EPR S-RELAP5 plant model system nodalization details are shown in Figures A-1 through A-6. The model configuration is essentially the same as the 4-loop sample problem provided in Reference A-1 with changes incorporated to reflect current modeling guidelines and U.S. EPR-specific hardware; e.g., the axial economizer steam generator design. Noding changes are addressed in Table A-5, Item 10, as part of the RLBLOCA EM SER compliance.

As described in the RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. Table A-6 presents process parameters and statistical distributions used in the analyses. The LBLOCA phenomenological uncertainties are provided in Reference A-1.

For the AREVA NP RLBLOCA evaluation model, significant containment parameters, as well as NSSS parameters, were established via a PIRT process. Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to PCT) containment parameters—containment pressure and

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[1] For U.S. EPR licensing applications such as the Design Control Document (DCD), the 1.7 Uchida multiplier is confirmed as described in Appendix B.

temperature. In many instances, the conservative guidance of BTP-6-2 CSB 6-1 (Reference A-2) was used in setting the remainder of the containment model input parameters. As noted in Table A-6, containment temperature is a sampled parameter. Containment pressure is indirectly ranged by sampling the containment volume.

The RLBLOCA analysis uses 124 cases to obtain the highest PCT, maximum local oxidation, and maximum hydrogen generation. The highest PCT case is Case 38, which has PCT of 1625ºF. A nominal 50/50 PCT case is Case 13, with a PCT of 1243ºF. This result can be used to quantify the relative conservatism in the limiting PCT case result. In this analysis, it is 382ºF. Case 38 also reported the highest maximum local oxidation, with a value of 0.92 percent. The fraction of total hydrogen generated is not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation. Case 2 reports the highest total oxidation, with a value of 0.02 percent.

A summary of the major input parameters for the limiting PCT transient is presented in Table A-7. The hot fuel rod results and event times for the limiting PCT case are shown in Table A-8 and Table A-9. Figure A-7 and Figure A-8 show linear scatter plots of the important parameters sampled for the 124 calculations. These figures show the parameter ranges used in the analysis. Figures A-9 through A-23 show PCT scatter plots versus various sampled parameters. Maximum oxidation and total oxidation are shown versus PCT in Figure A-20 and Figure A-23, respectively. Figures A-24 through A-50 illustrate important parameters from the S-RELAP5 limiting PCT calculation. Figure A-24 is the plot of PCT independent of elevation.

The analysis reported herein is for a rated core thermal power level of 4590 MWt; a complete core of AREVA NP fuel; a steam generator tube plugging level of 5 percent in each generator; a total peaking factor ($F_Q$) of 2.60, U.S. EPR Technical Specification value; and a radial power peaking ($F_{\Delta H}$) of 1.70, Technical Specification value of 1.633 plus uncertainty of 4.1 percent. The analysis considers no $K_Z$ constraint on axial peaking; that is, $K_Z$ is set equal to one for all core elevations. Also, no core peaking burnup constraints are applied.

This U.S. EPR RLBLOCA sample problem, which is for the equilibrium fuel cycle, yields a limiting PCT of 1625°F. Maximum oxidation thickness and hydrogen generation are well within regulatory requirements.

Specifically, it is concluded for this U.S. EPR RLBLOCA sample problem that:

1. The calculated PCT for the limiting PCT case is less than 2200°F.
2. The maximum calculated local clad oxidation is less than 17 percent.
3. The maximum amount of core-wide oxidation does not exceed 1 percent of the fuel cladding.

## A.3.0 *SER Compliance*

The Conditions and Limitations imposed by the RLBLOCA EM SER are addressed in Table A-4. This U.S. EPR application complies with all SER Conditions and Limitations.

One nonlimiting PCT case exhibited a blowdown quench (SER Item 7); a discussion of this behavior follows.

One case out of the 124 that were analyzed as part of the RLBLOCA sample problem exhibited a quench of the PCT node before the end of blowdown. This case is distinguished as having a relatively small break area towards the lower end of the break spectrum analyzed. This case produces a PCT of 771ºF, well below the limiting PCT of 1625ºF. The limiting case (a split break with a large break area) did not exhibit a blowdown quench.

Mechanistically, the observed quench occurs because the small break area limits break flow. This reduces the rates at which pressure and flow decrease at the PCT location as compared to the limiting case. The resulting combination of higher core flow and pressure cools the clad sufficiently to enable a return to nucleate boiling.

A factor contributing to the occurrence of blowdown quench in these U.S. EPR cases is the low peak power density of 13.63 kW/ft. In comparison, the analyses presented in the Reference A-1 sample problem, which did not exhibit blowdown quench, had a peak power density of 15.7 kW/ft. The lower peak power density results in less severe clad heatup and facilitates quenching. It also contributes to a lower maximum PCT of 1625°F versus 1853°F for the Reference A-1 analysis.

It is therefore concluded that the predicted blowdown quench behavior is appropriate for these nonlimiting cases, and that the previously approved RLBLOCA EM (Reference A-1), in conjunction with the described changes, is applicable to the U.S. EPR design.

## A.4.0 *References*

A-1. EMF-2103(P)(A) Revision 0, “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” FANP Richland, Inc., April 2003.

A-2. NUREG-0800, LWR edition, Revision 2, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, July 1987.

**Table A-1 MHSI Flow Rates per Pump**

| Pressure Cold Leg[1] (psia) | Pressure SI Piping[2] (psia) | Flow Rate (lbm/s) |
|---|---|---|
| 14.5 | 21.2 | 130.1 |
| 145.0 | 151.6 | 117.3 |
| 290.0 | 296.6 | 105.5 |
| 435.0 | 441.5 | 94.0 |
| 580.0 | 586.5 | 82.3 |
| 725.0 | 731.5 | 70.0 |
| 870.0 | 876.4 | 56.7 |
| 1015.0 | 1021.0 | 41.8 |
| 1160.0 | 1166.0 | 24.1 |
| 1233.0 | 1239.0 | 13.0 |
| 1305.0 | 1306.0 | 0.0 |

**Table A-2 LHSI Flow Rates per Pump**

| Pressure Cold Leg[1] (psia) | Pressure SI Piping[2] (psia) | Flow Rate (lbm/s) |
|---|---|---|
| 14.5 | 36.4 | 312.2 |
| 58.0 | 76.8 | 273.2 |
| 87.0 | 104.0 | 248.4 |
| 116.0 | 131.4 | 223.4 |
| 145.0 | 159.0 | 197.6 |
| 174.0 | 186.6 | 170.7 |
| 203.0 | 214.4 | 141.9 |
| 232.0 | 242.4 | 110.5 |
| 261.0 | 270.5 | 75.1 |
| 290.0 | 298.9 | 32.3 |
| 311.0 | 316.3 | 0.0 |

1 Not used to specify injection flow. Location is downstream of injection points.

2 This is the local pressure within safety injection piping. Using Train 1 as typical or representative, this is the volume average pressure at volume 963-1 for LHSI and volume 965 for MHSI (see Figure A-6).

**Table A-3 RLBLOCA Analysis Plant Parameter Values**

| | Parameter Description | Parameter Value |
|---|---|---|
| 1.0 | Plant Physical Description | |
| | 1.1 Fuel | |
| | a) Cladding outside diameter | 0.374 in |
| | b) Cladding inside diameter | 0.329 in |
| | c) Cladding thickness | 0.0225 in |
| | d) Pellet outside diameter | 0.3225 in |
| | e) Pellet density | 96% of theoretical |
| | f) Active fuel length | 165.354 in |
| | g) $Gd_2O_3$ Concentration | 0 – 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 | 17 x 17 |
| | e) SG tube plugging | ≤ 5% |
| 2.0 | Plant Initial Operating Conditions | |
| | 2.1 Reactor Power | |
| | a) Core power | 4590 MWt + 22 MWt (heat balance uncertainty) = 4612 MWt |
| | b) Maximum core peaking (FQ) | ≤ 2.60[1] (normalized) |
| | c) Maximum pin radial peaking (FΔH) | ≤ 1.70 (normalized) |
| | d) MTC | ≤ 0 at HFP |
| | | |
| | 2.2 Fluid Conditions | |
| | a) Loop flow (total RCS flow) | 176.44 Mlbm/hr ≤ M ≤ 198.00 Mlbm/hr |
| | b) RCS average temperature | 590°F ≤ T ≤ 598°F |
| | c) Nominal upper head temperature | 594°F (average) |
| | d) Pressurizer pressure | 2214 psia ≤ P ≤ 2286 psia |
| | e) Pressurizer level | 49.3 % ≤ L ≤ 59.3 % |
| | f) Accumulator pressure | 653 psia ≤ P ≤ 711 psia |
| | g) Accumulator (one of four) liquid volume | 1236 $ft^3$ ≤ V ≤ 1413 $ft^3$ |
| | h) Accumulator temperature and containment liquid and vapor temperature (including IRWST liquid) | 100°F ≤ T ≤ 131°F (coupled to containment temperature) |
| | i) Accumulator line resistance | Design piping configuration |
| | | |

[1] Includes measurement and engineering uncertainties.

**Table A-4 RLBLOCA Analysis Plant Parameter Values (continued)**

| | Parameter Description | Parameter Value |
|---|---|---|
| 3.0 | Accident Boundary Conditions | |
| | a) Break location | Cold leg in loop containing the pressurizer |
| | b) Break type | Double-ended guillotine or split |
| | c) Break size per side (relative to cold leg pipe) | 0.05 ≤ A ≤ 1.0 full pipe area |
| | d) Worst single failure | Loss of one complete train of MHSI and LHSI |
| | e) Offsite power | Unavailable (refer to Section 4.1) |
| | f) Medium head safety injection flow | Minimum flow per pump w/o spillage (Table A-1) |
| | g) Low head safety injection flow | Minimum flow per pump w/o spillage (Table A-2), flow splits calculated by S-RELAP5 |
| | h) Pumped safety injection water temperature | 140°F[1] |
| | i) Safety injection delay | ≤ 15 seconds (with offsite power)<br>≤ 40 seconds (without offsite power) |
| | j) Containment pressure (initial) | 14.664 psia |
| | k) Containment sprays | N/A |

[1] The pumped safety injection draws water from the IRWST. In the S-RELAP5 model, the temperature of this source volume is set to 140ºF as an allowance for much hotter liquid spilling from the RCS and falling into the IRWST. However, the ICECON containment model applies the sampled value of containment temperature to the containment liquid, which represents the IRWST. The containment temperature sampling range is shown in 2.2h. The treatment of the IRWST liquid in the ICECON containment model is consistent with a minimum containment pressure calculation.

## Table A-5 SER Conditions and Limitations

| SER Conditions and Limitations | Response |
|---|---|
| 1. A CCFL violation warning will be added to alert the analyst to a CCFL violation in the downcomer should such occur. | There was no significant occurrence of CCFL violations in the downcomer for this analysis. |
| 2. 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 applied to an end-of-life analysis for which the pin pressure is significantly higher, the need for a blowdown cladding rupture model will be reevaluated. The evaluation may be based on relevant engineering experience and should be documented in either the RLBLOCA guideline or plant-specific calculation file. | The planar linear heat generation rate (PLHGR) for the U.S. EPR design is lower than the defined limit for the RLBLOCA EM (Reference A-1). An end-of-life calculation was not performed; thus, the need for a blowdown cladding rupture model was not reevaluated. |
| 4. Slot breaks on the top of the pipe have not been evaluated. These breaks could cause the loop seals to refill during late reflood and the core to uncover again. These break locations are an oxidation concern as opposed to a PCT concern since the top of the core can remain uncovered for extended periods of time. Should an analysis be performed for a plant with loop seals with bottom elevations that are below the top elevation of the core, AREVA NP will evaluate the effect of the deep loop seal on the slot breaks. The evaluation may be based on relevant engineering experience and should be documented in either the RLBLOCA guideline or plant-specific calculation file. | This evaluation is performed in accordance with the method documented in the RLBLOCA guideline. |
| 5. The model applies to 3- and 4-loop Westinghouse- and CE-designed nuclear steam systems. | The RLBLOCA EM is applicable to the U.S. EPR reactor, a 4-loop plant. This was addressed and justified in Section 5.0. |
| 6. The model applies to bottom reflood plants only (cold side injection into the cold legs at the reactor coolant discharge piping). | The RLBLOCA EM is applicable to the U.S. EPR design since it is a bottom reflood plant. |
| 7. The model is valid as long as blowdown quench does not occur. If blowdown quench occurs, additional justification for the blowdown heat transfer model and uncertainty are needed 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 PCT case showed no evidence of blowdown quench. Blowdown quenches were observed in a one case. An explanation of this behavior is provided in Section A.3.0. |
| 8. The reflood model applies to bottom-up quench behavior. If a top-down quench occurs, the model is to be justified or corrected to remove top quench. A top-down quench is characterized by the quench front moving from the top to the bottom of the hot assembly. | Examination of the case set showed that core quench initiated at the bottom of the core and proceeded upward. |
| 9. The model does not determine whether Criterion 5 of 10CFR50.46, long-term cooling, has been satisfied. This will be determined by each applicant or licensee as part of its application of this methodology. | Long-term cooling will be addressed in the Design Certification Application. |

| SER Conditions and Limitations | Response |
|---|---|
| 10. Specific guidelines must be used to develop the plant-specific nodalization. Deviations from the reference plant must be addressed. | The model nodalization is consistent with the sample calculations given in the RLBLOCA EM (Reference A-1), except for changes incorporated to reflect current modeling guidelines and U.S. EPR-specific hardware. Significant changes are noted below.<br><br>Accumulator Line: The accumulator line is shown in Figure A-6. For the U.S. EPR design, the accumulator line piping run between the check valve and the connection to the cold leg is quite long, about 25 feet. Hence, the normal single node (in the base EM) is divided into two nodes to minimize connecting nodes of disparate size.<br><br>ECCS Lines: The ECCS model is shown in Figure A-6. This model includes cross-connect piping for the LHSI system to provide injection to the opposite side of the reactor vessel from the broken loop, regardless of which intact SI train is randomly chosen to be active. The cross-connect isolation valves are open whenever one of the independent SI trains is out of service for maintenance. The cross-connect model involves additional piping to connect the LHSI systems to the accumulator discharge piping. Other than this difference, the modeling of the pumped SI sources is very similar to the base EM.<br><br>Steam Generator Axial Economizer: The SG is shown in Figure A-2. On the secondary side of the U.S. EPR U-tube SG, the bottom half of the downcomer and most of the tube region is physically divided into a hot and cold side. This, coupled with the component size and the location of the separators and dryers, requires obvious noding changes relative to the base EM for proper representation. Nevertheless, the changes were implemented mindful of maintaining conformity with the base model concept.<br><br>Inverted Top Hat: The RV upper head (UH) is shown in Figure A-3. The U.S. EPR UH is configured in what is commonly termed an “inverted top hat.” A section of the UH extends below the top of the downcomer into what is usually the top of the upper plenum (UP). To properly model the UH, the region extending into what is usually the top of the UP is modeled as a separate node. Hence, the U.S. EPR UH consists of three nodes instead of the two nodes in the base EM. |

| SER Conditions and Limitations | Response |
|---|---|
| | Heavy Reflector: The heavy reflector is also shown in Figure A-3. It consists of a set of massive plates surrounding the fuel assemblies. It replaces both the former and core baffle plates. The heavy reflector contains a series of flow holes, allowing fluid to both cool the reflector and bypass the core. The reflector is properly configured in the U.S. EPR model—both as a heat structure and a core flow bypass device. |
| 11. A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the topical report approval process must be provided. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effects of that deviation. | Table A-9 presents the summary of the full range of applicability for the important heat transfer correlations, as well as the ranges calculated in the limiting analysis case. Calculated values for other parameters of interest are also provided. As is evident, the plant-specific parameters fall within the applicability range of the methodology. This is evidence of the applicability of the previously approved RLBLOCA EM to the U.S. EPR plant. |
| 12. The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses, including the calculated worst break size, PCT, and local and total oxidation. | Analysis results are presented in Section A.2.0. |
| 13. Applicants or licensees wishing to apply the 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 10CFR50.46(a)(i) to include M5® cladding material has been completed. | AREVA NP understands that an exemption request is required for the use of M5® cladding. An exemption request is planned as part of Design Certification. |

## Table A-6 Statistical Distributions Used for Process Parameters

| Parameter | Operational Uncertainty Distribution | Parameter Range |
|---|---|---|
| Core Power (%) | Uniform | 100.48 |
| Total Initial Flow Rate (Mlbm/hr) | Uniform | 176.44–198.00 |
| Initial Average Operating Temperature (°F) | Uniform | 590–598 |
| Pressurizer Pressure (psia) | Uniform | 2214–2286 |
| Pressurizer Level (%) | Uniform | 49.3–59.3 |
| Containment Volume (x $10^6$ $ft^3$) | Uniform | 2.888–3.934[1] |
| Containment Temperature (°F) | Uniform | 100–131 |
| Accumulator Pressure (psia) | Uniform | 652.7–710.7 |
| Accumulator (one of four) Volume ($ft^3$) | Uniform | 1236–1412.6 |
| Intact Loop Number | Uniform | 1, 2, and 4 |

## Table A-7 Summary of Major Parameters for the Limiting PCT Transient

| | |
|---|---|
| Burn Time (hrs) | 10,469 |
| Hot Assembly Burnup (MWd/MTU) | 20,600 |
| Average Ring and Low-Powered, Outer Ring Burn Time (hrs) | 3039 |
| Average Ring and Low-Powered, Outer Ring Burnup (MWd/MTU) | 5900 |
| Core Power (MWt) | 4612.03 |
| Core Peaking ($F_Q$) | 2.54 |
| Axial Skew | Top |
| Break Type | DESB |
| Break Size per Side ($ft^2$) | 4.9901 (~97%) |
| Offsite Power Availability | No |
| Decay Heat Multiplier | 0.98778 |

[1] The lower bound is a nominal value representing the combined volumes of gas and water; maximum value is the sum of the lower bound volume and the volume of the containment heat sinks. Refer to Appendix B.

## Table A-8 Summary of Results for the Limiting Cases

| | Highest Cases | | |
|---|---|---|---|
| | **PCT** | **% Oxidation Maximum** | **% Total Oxidation** |
| **Case Number** | **38** | | **2** |
| Fuel Type (hot rod) | 8% Gadolinia | | 8% Gadolinia |
| PCT | | | |
| Temperature | 1625°F | | 1573°F |
| Time | 8.5 s | | 7.6 s |
| Elevation | 11.6 ft | | 11.6 ft |
| Metal-Water Reaction | | | |
| % Oxidation Maximum | 0.92% | | 0.81% |
| % Total Oxidation | 0.021% | | 0.023% |

**Table A-9 Calculated Event Times for the Limiting PCT Case**

| Event | Time (sec) |
|---|---|
| Begin analysis | 0 |
| Break opened | 0 |
| RCP tripped | 0 |
| PCT occurred (1625°F) | 8.5 |
| SIAS issued | 10.3 |
| Start of broken loop accumulator injection (loop 3) | 11.6 |
| Start of intact loop accumulator injection | 13.2 |
| Beginning of core recovery (beginning of reflood) | 28.4 |
| Start of MHSI | 50.3 |
| Broken loop MHSI delivery began (loop 3) | 50.3 |
| Intact loop MHSI delivery began (loop 4) | 50.3 |
| LHSI available | 50.3 |
| Broken loop LHSI delivery began (loop 3) | 50.3 |
| LHSI train 4 starts to deliver flow | 50.3 |
| Intact loop accumulator emptied (loops 1, 2, and 4 respectively) | 55.2, 55.5, 56.2 |
| Broken loop accumulator emptied (loop 3) | 59.1 |
| Transient calculation terminated | 738.8 |

**Table A-10 Heat Transfer Parameters for the Limiting PCT Case[1]**

[1] Values in brackets show full range of applicability as documented in Reference A-1.
[2] Conservatively biased parameter as per the AREVA NP RLBLOCA methodology (Reference A-1).
[3] 2269 psia is the initial upper plenum pressure for this case.
[4] Not important in pre-CHF heat transfer.

**Figure A-1 RLBLOCA Loop Noding Diagram**

**Figure A-2 Secondary Noding**

**Figure A-3 RLBLOCA RV Noding Diagram**

**Figure A-4 Core Noding Detail**

**Figure A-5 Upper Plenum Noding Detail**

**Figure A-6 Nodalization for S-RELAP5 ECCS Model**

**Figure A-7 Scatter Plot of Operational Parameters**

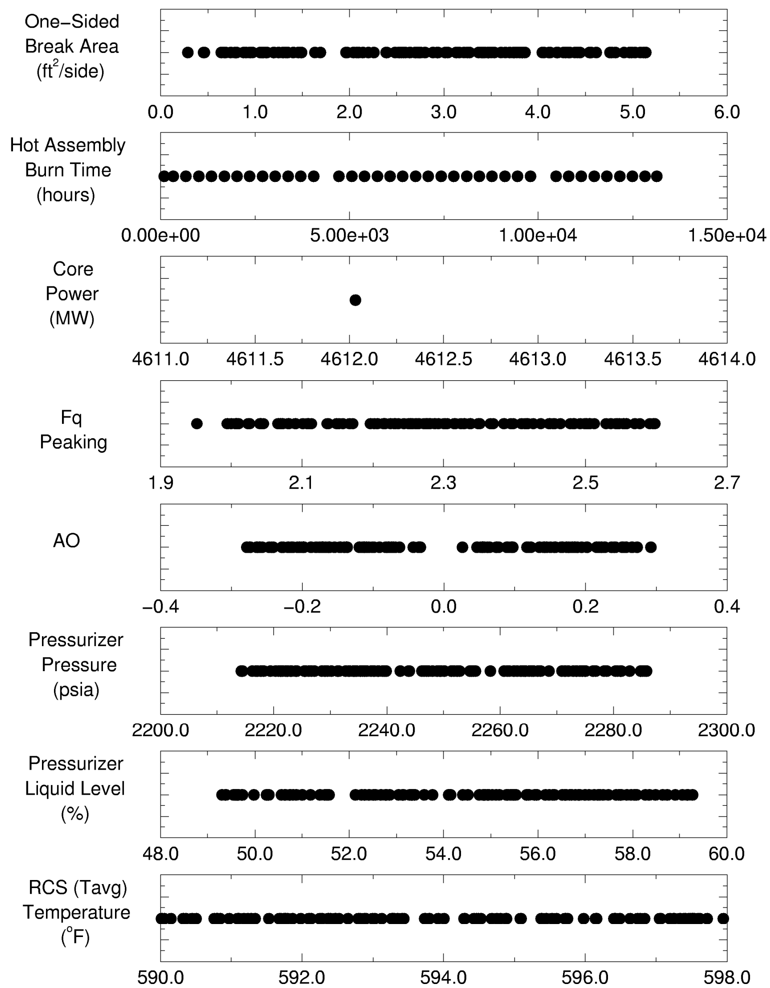


**Figure A-8 Scatter Plot of Operational Parameters (continued)**

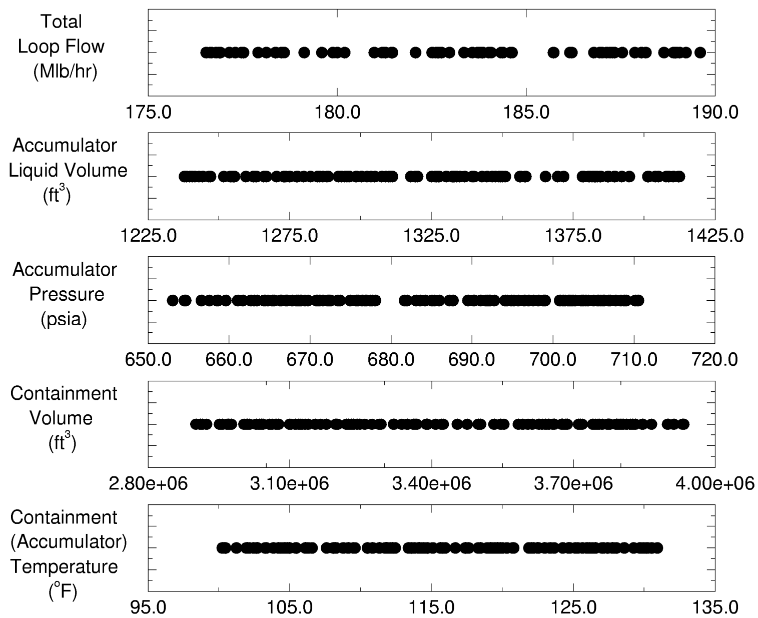


**Figure A-9 PCT versus Accumulator Liquid Volume Scatter Plot**

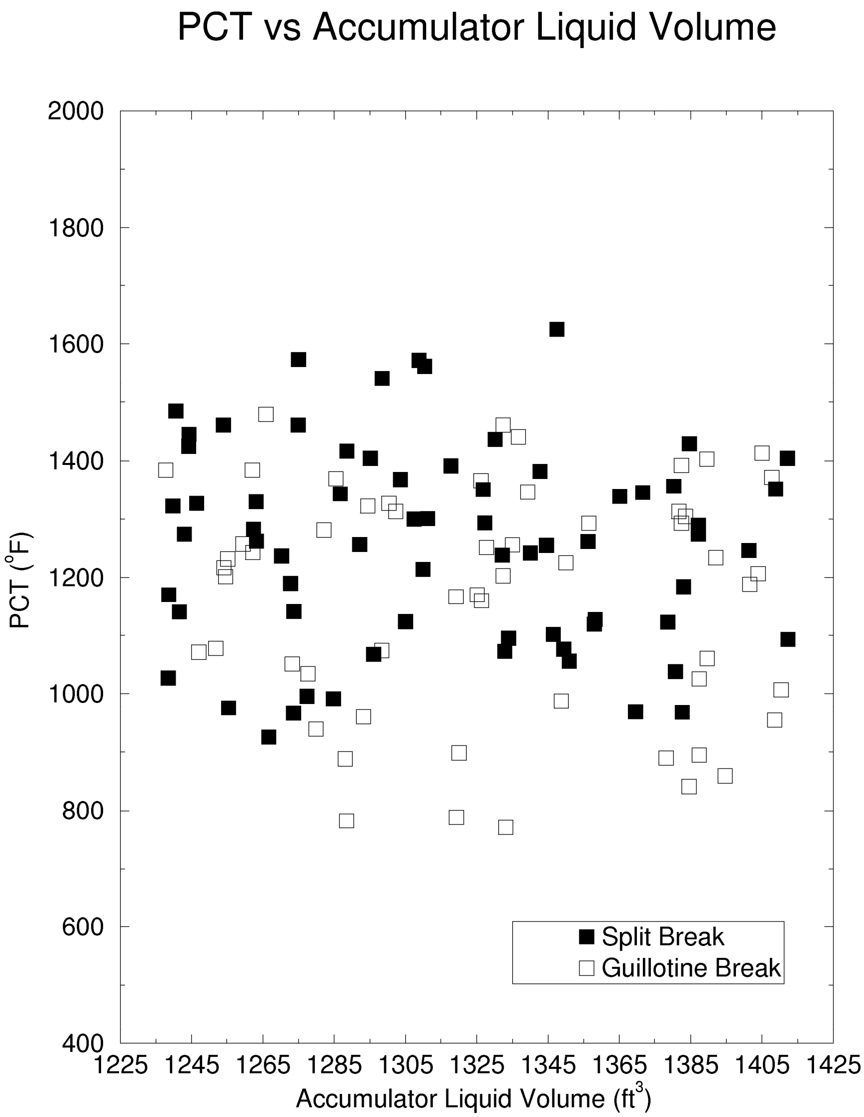


**Figure A-10 PCT versus Axial Offset Scatter Plot**

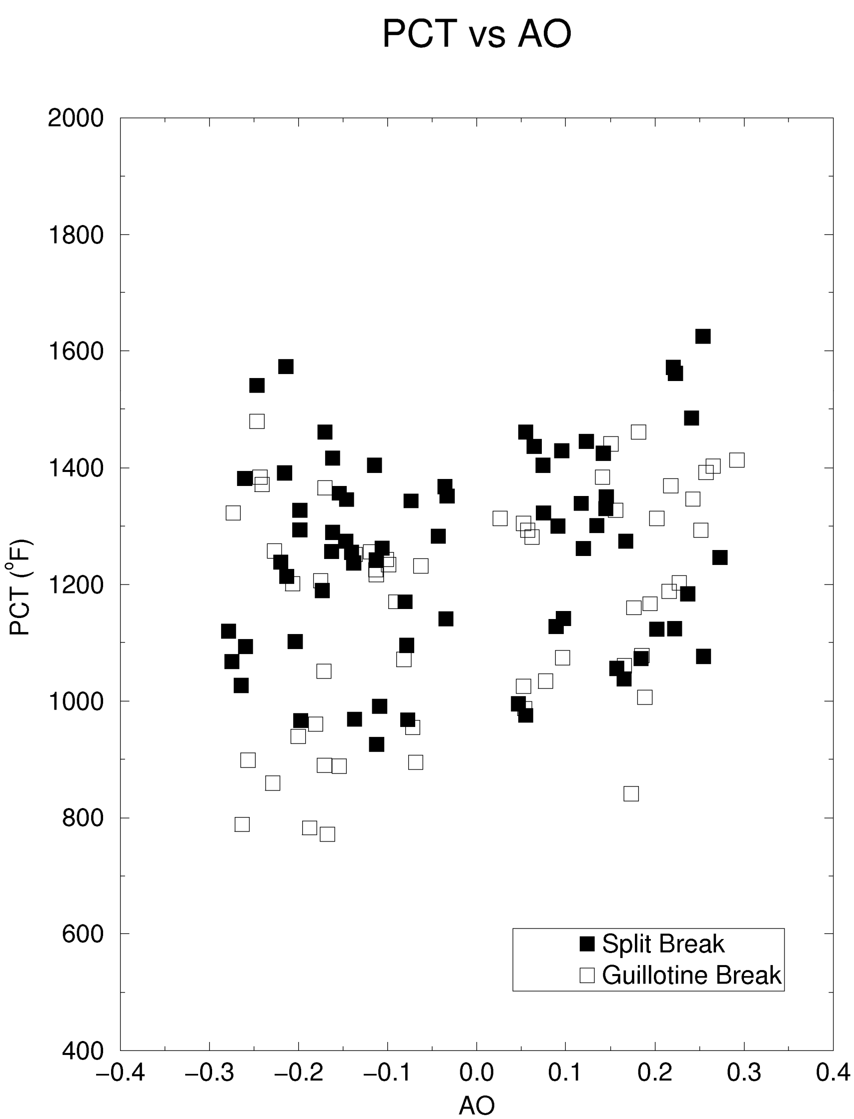


**Figure A-11 PCT versus One-Sided Break Area Scatter Plot**

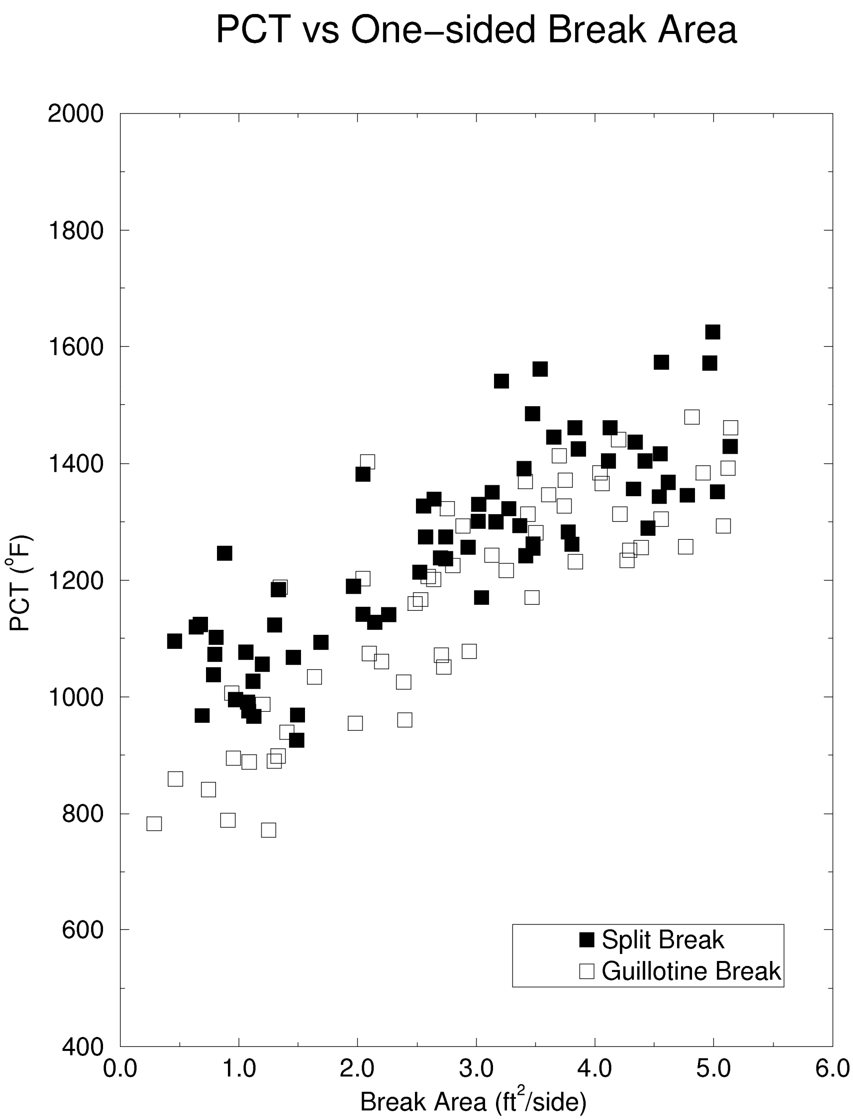


**Figure A-12 PCT versus Containment Temperature Scatter Plot**

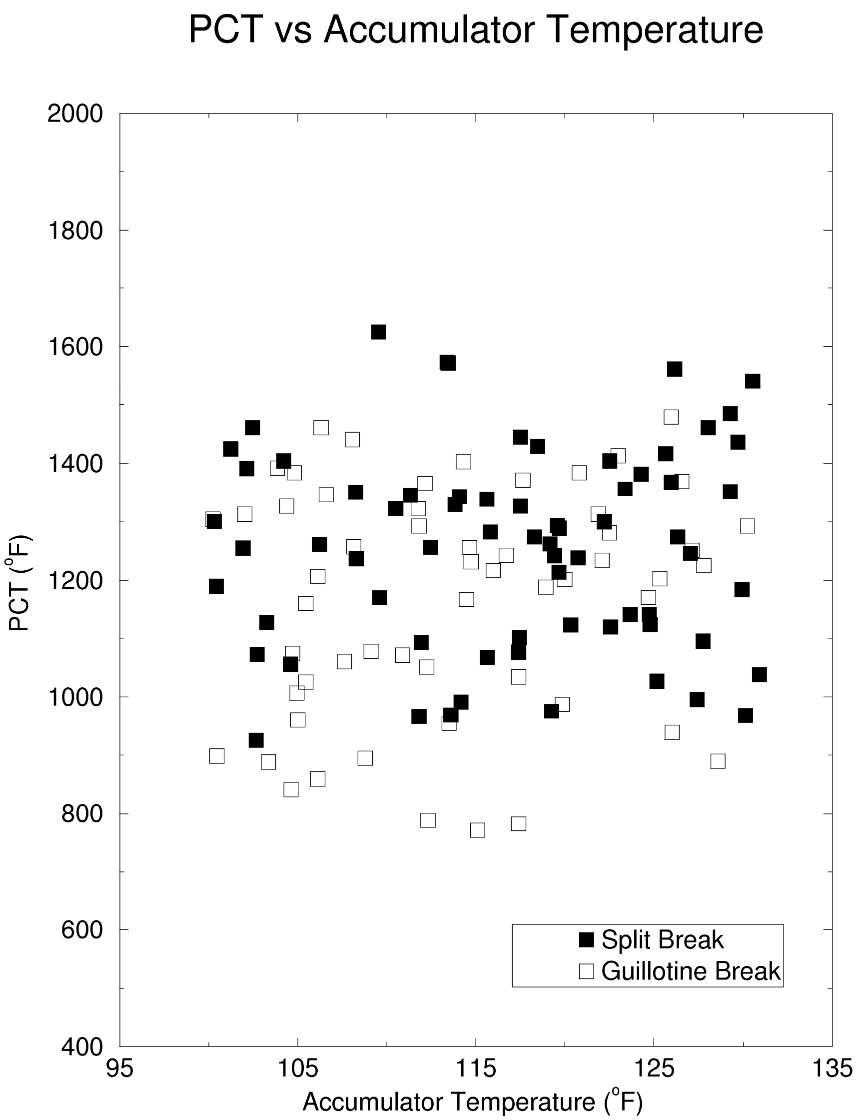


**Figure A-13 PCT versus Containment Volume Scatter Plot**

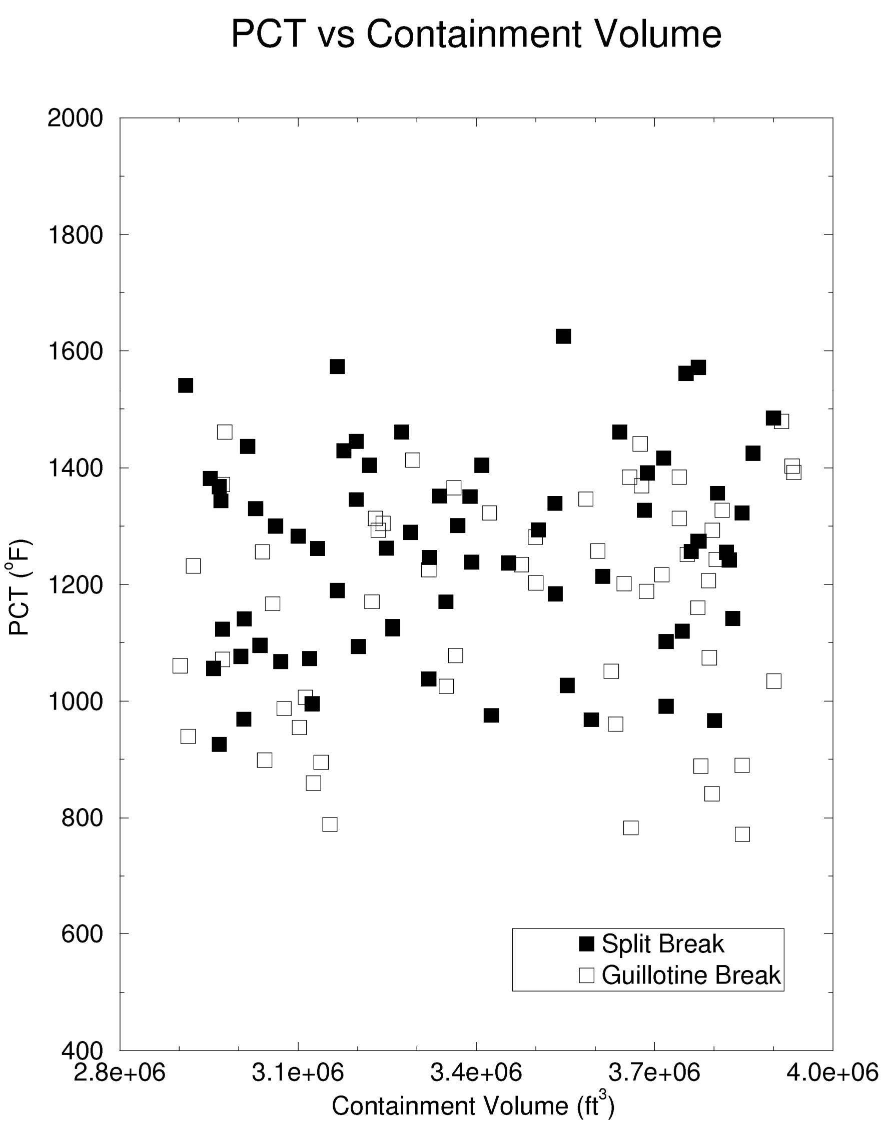


**Figure A-14 PCT versus Decay Heat Multiplier Scatter Plot**

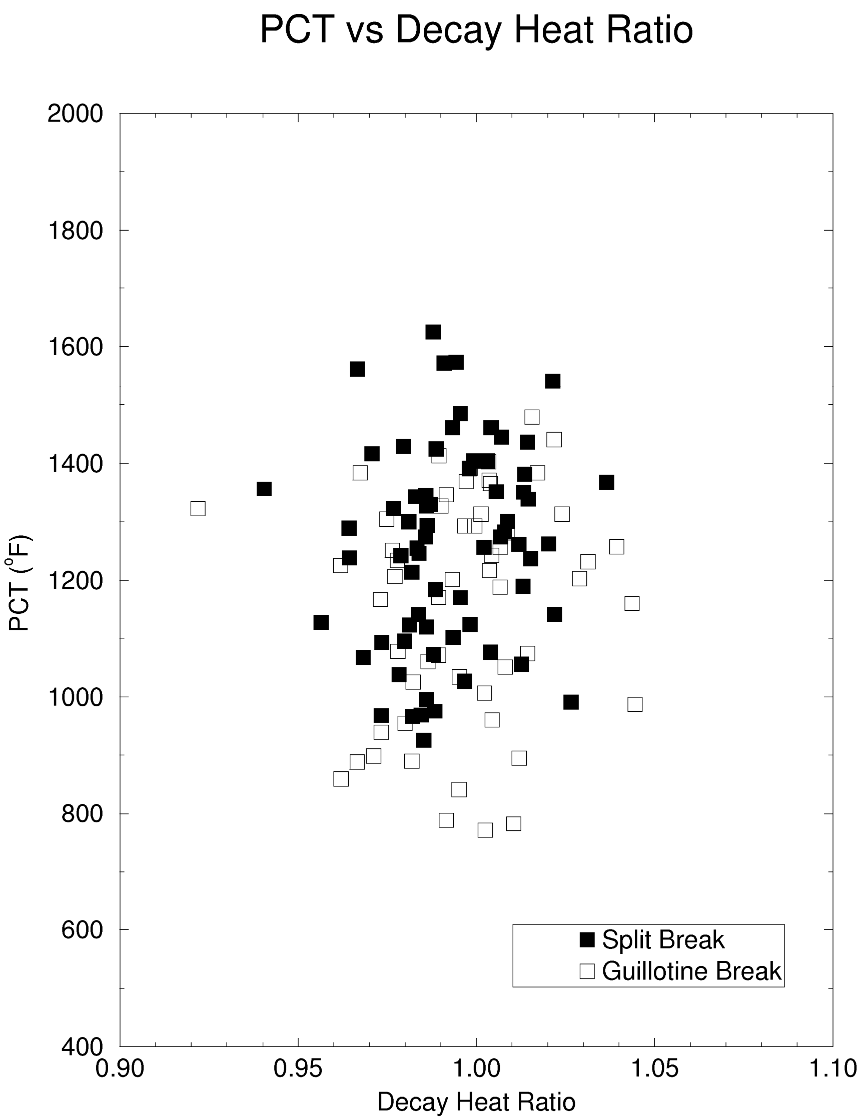


**Figure A-15 PCT versus Forslund Rohsenow Multiplier Scatter Plot**

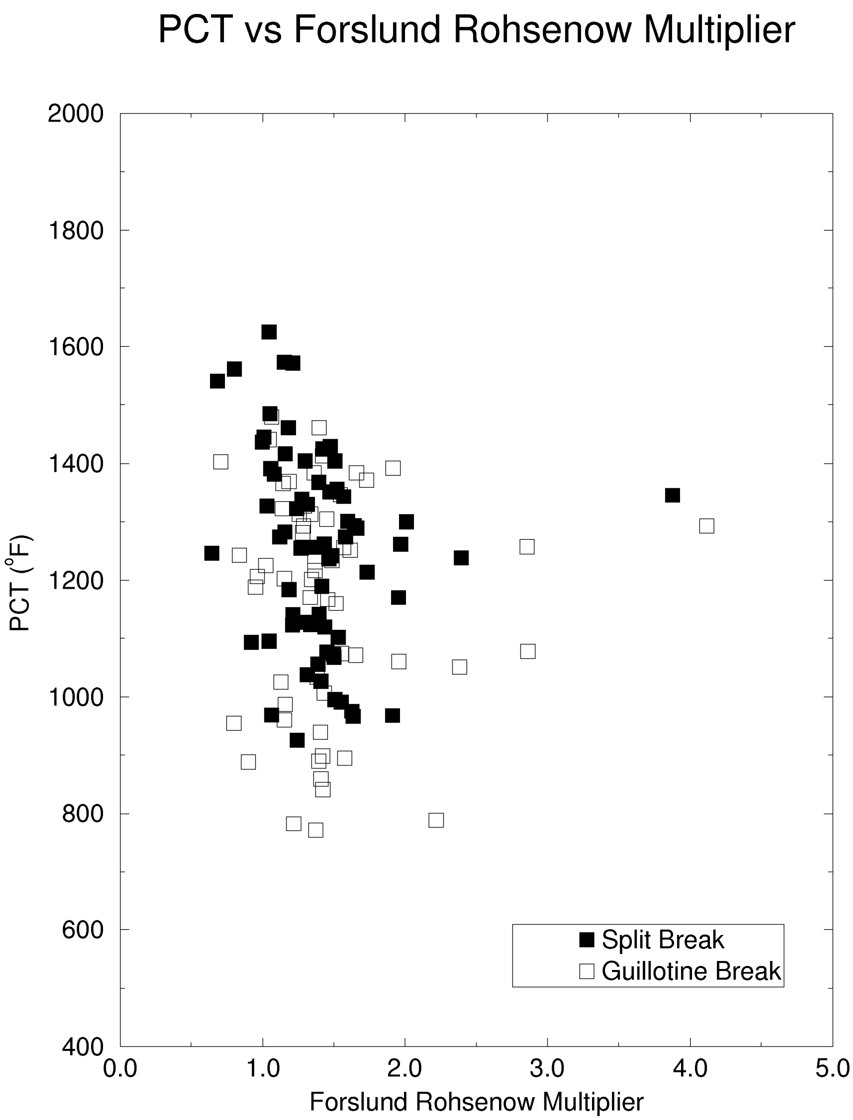


**Figure A-16 PCT versus Film Boiling Multiplier Scatter Plot**

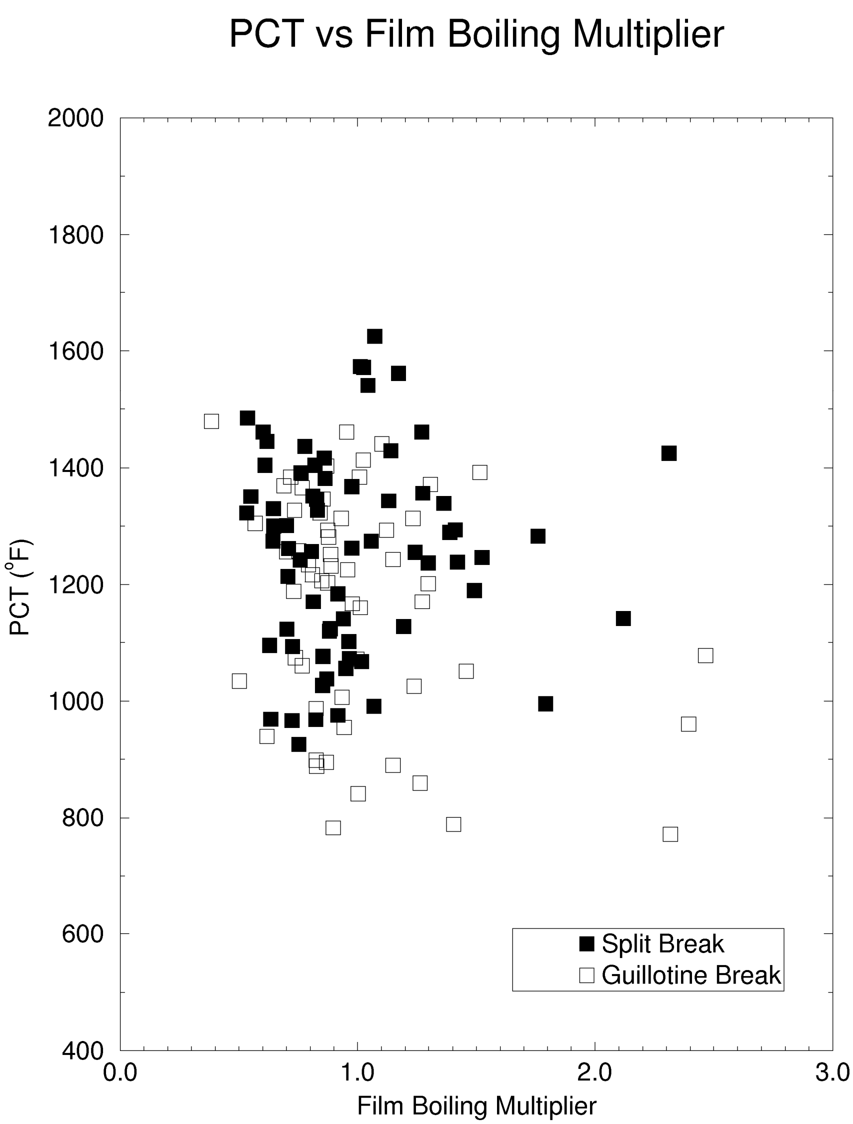


**Figure A-17 PCT versus Fq Scatter Plot**

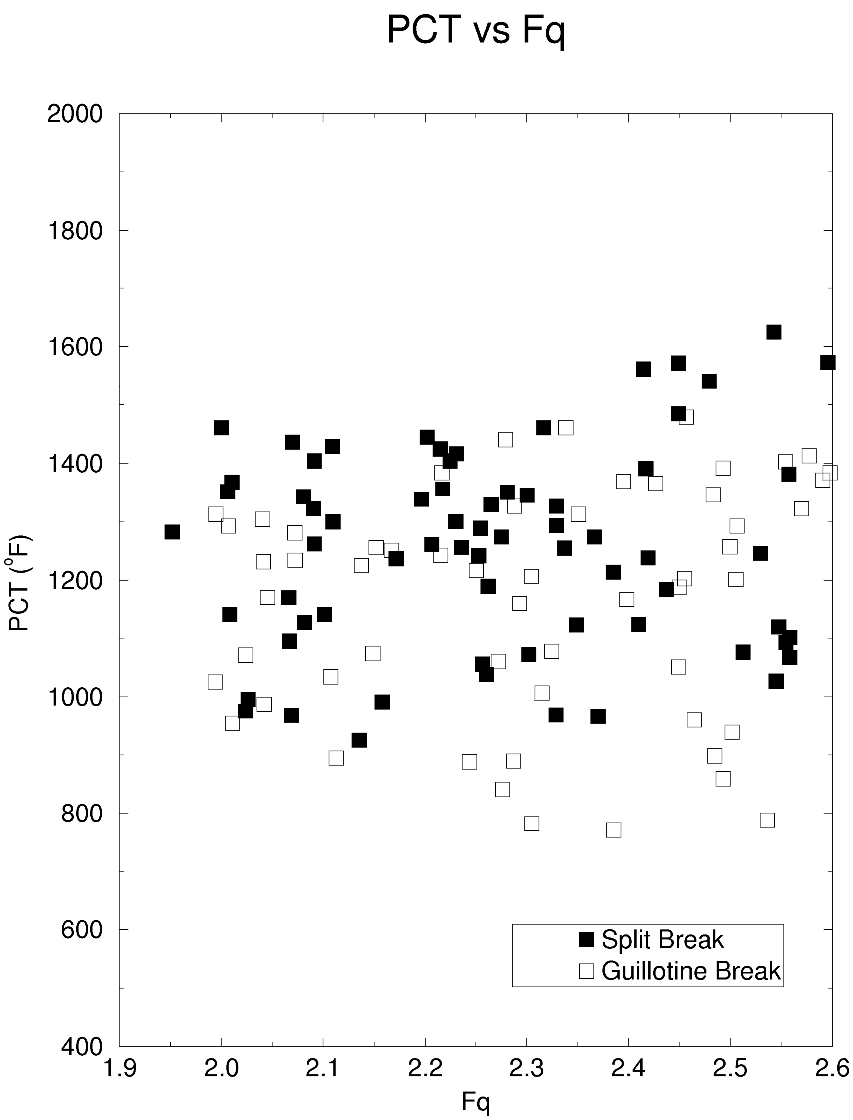


**Figure A-18 PCT versus Inner Ring Normalized Power Scatter Plot**

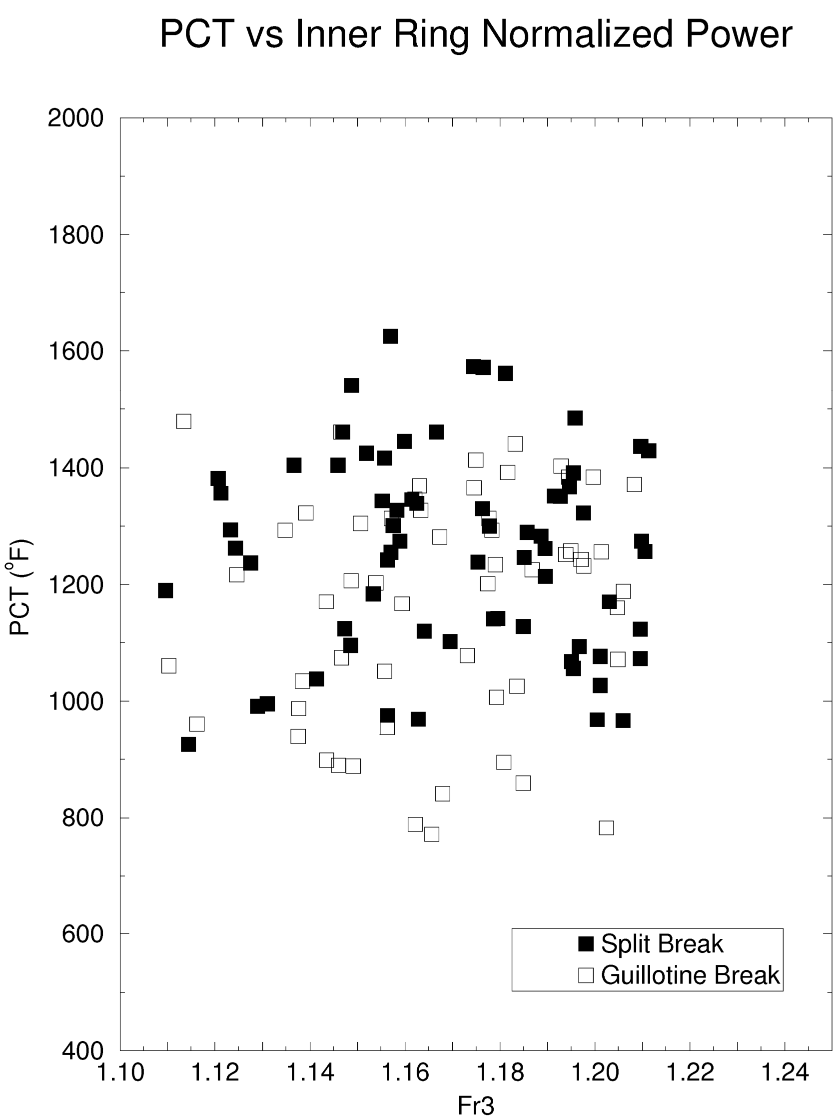


**Figure A-19 PCT versus Outer Ring Normalized Power Scatter Plot**

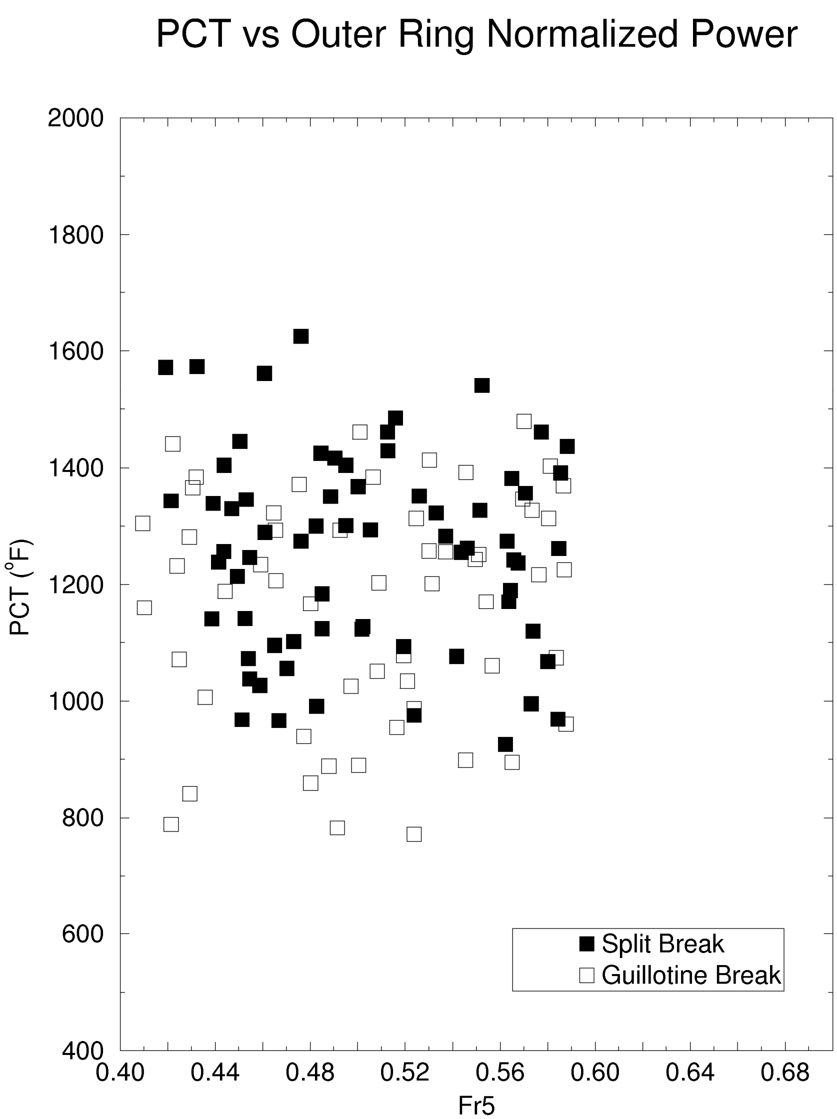


**Figure A-20 Maximum Oxidation versus PCT Scatter Plot**

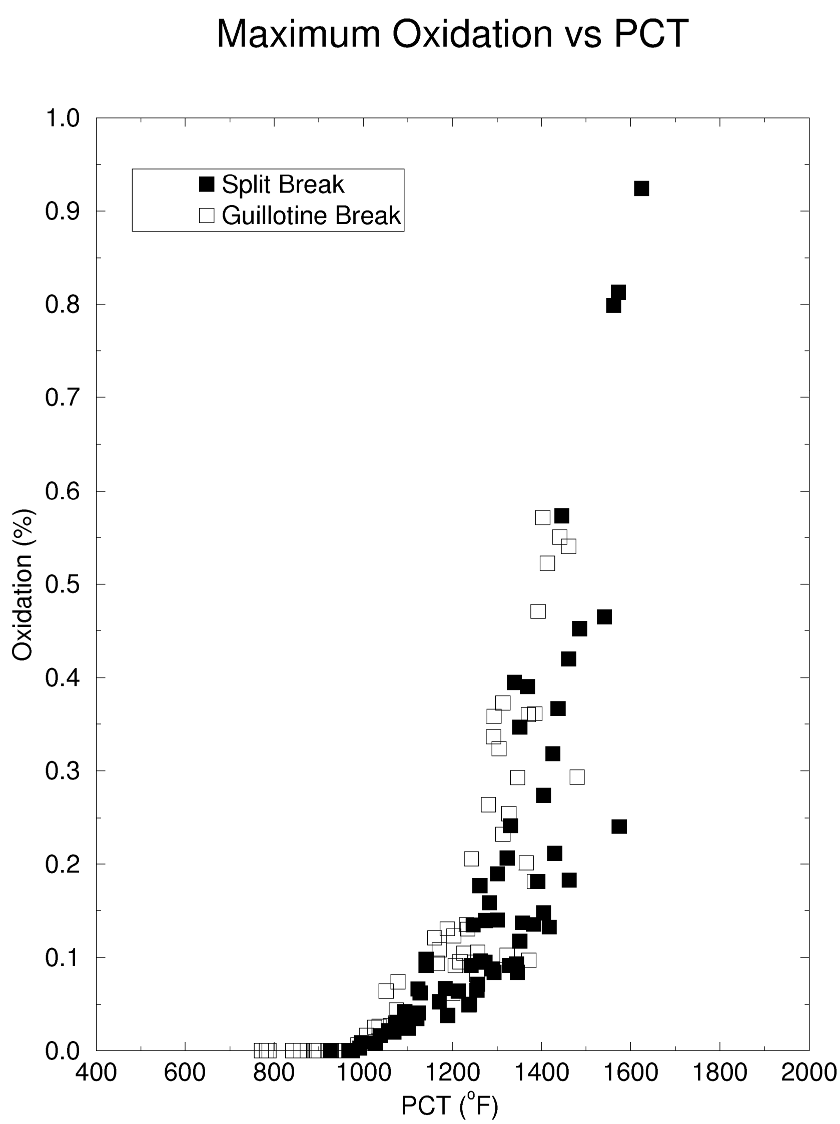


**Figure A-21 PCT versus Time of PCT Scatter Plot**

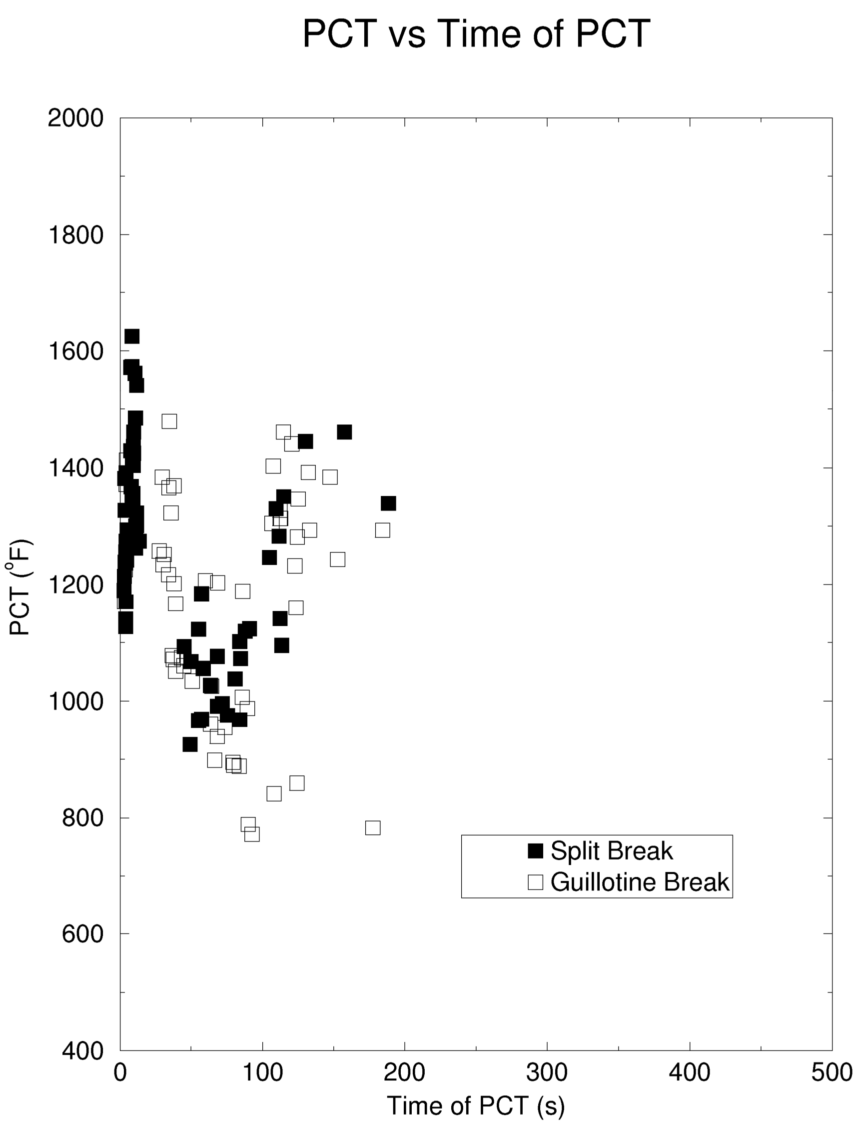


**Figure A-22 PCT versus Time in Cycle Scatter Plot**

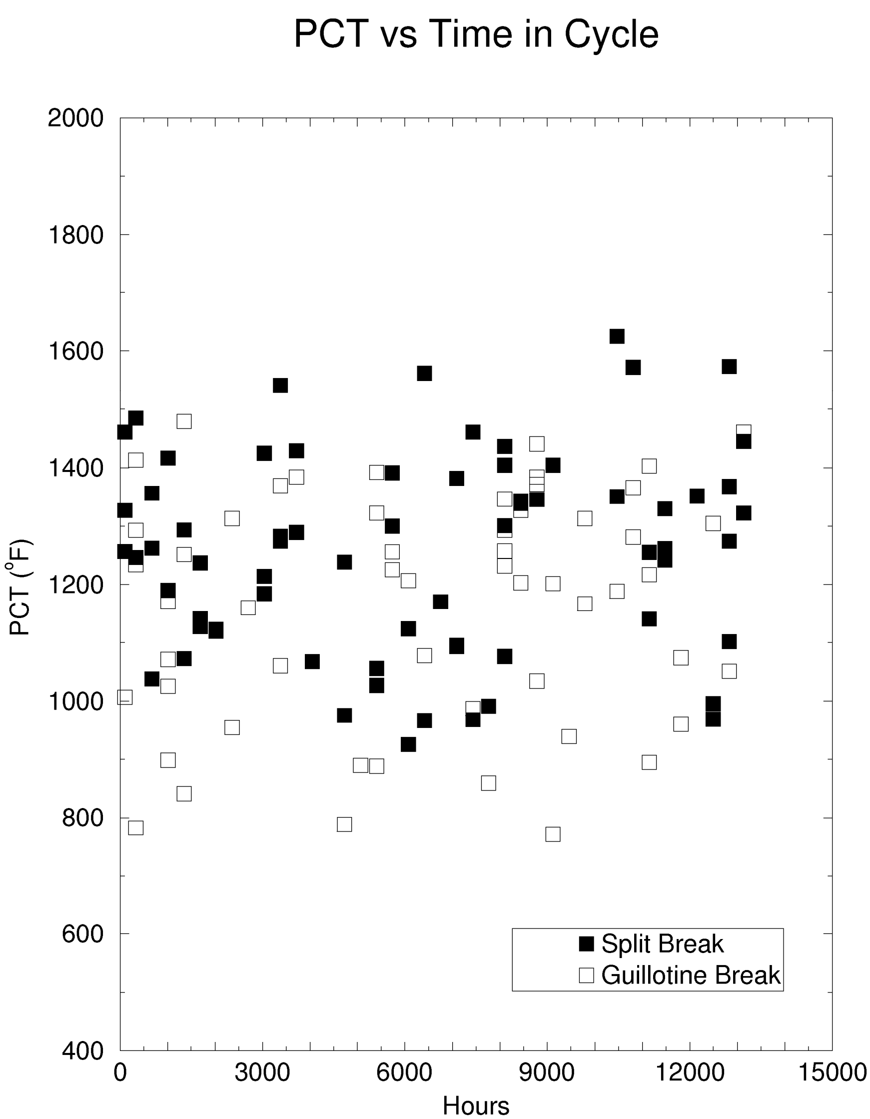


**Figure A-23 Total Oxidation versus PCT Scatter Plot**

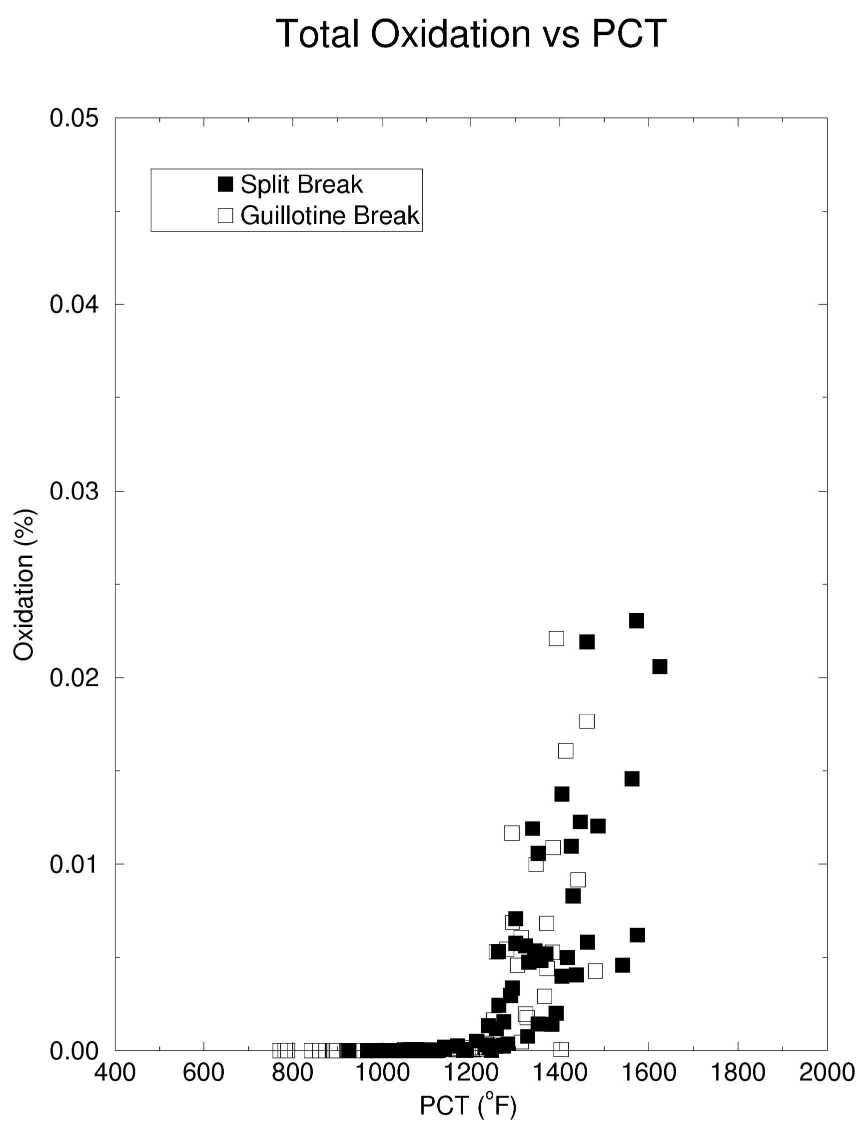


**Figure A-24  PCT Independent of Elevation for the Limiting PCT Case**

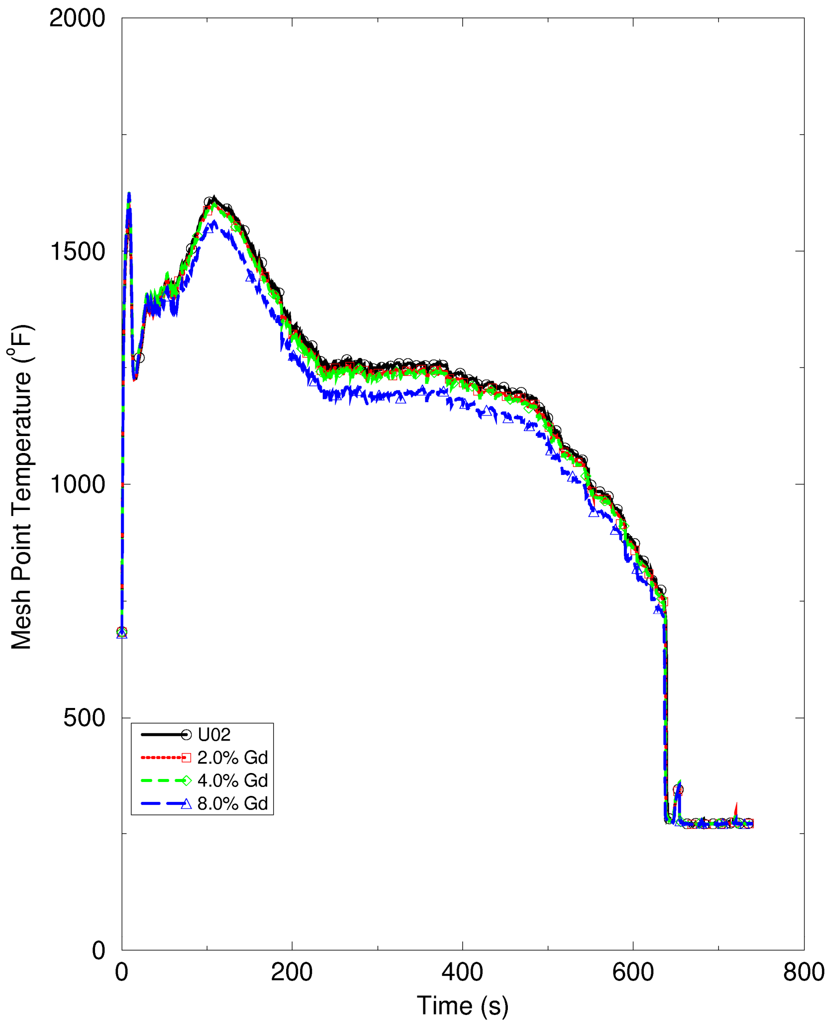


**Figure A-25 PCT Independent of Elevation for the Limiting PCT Case Hot Rod**

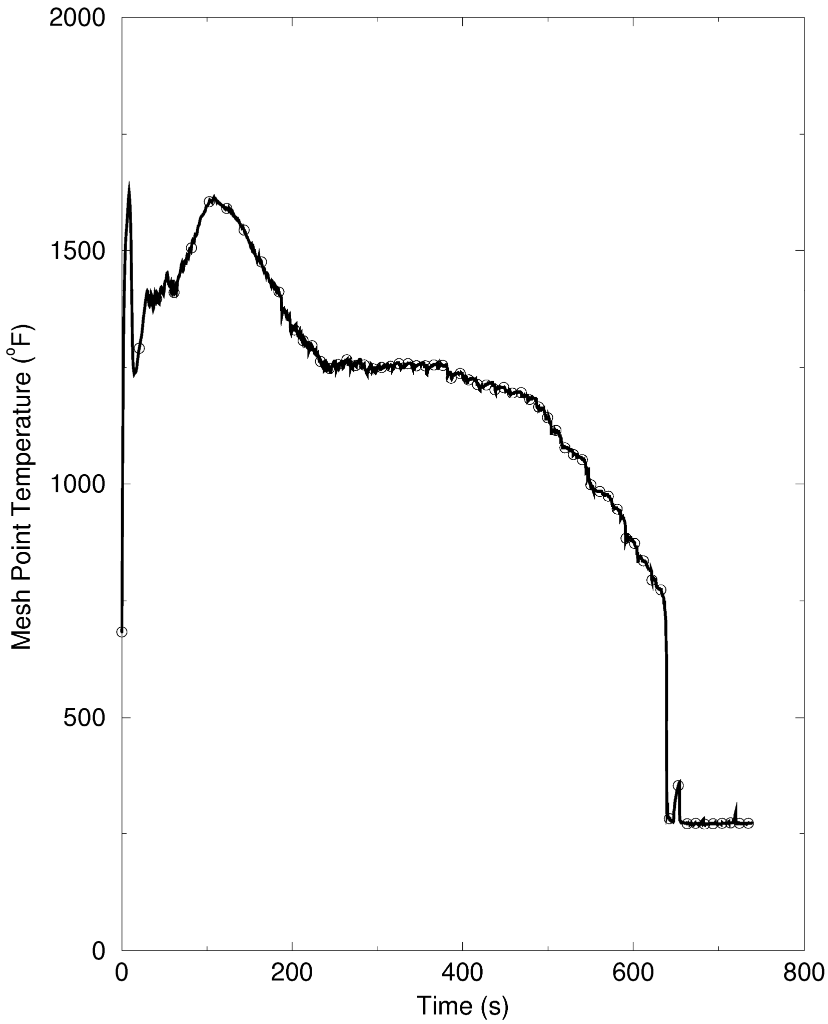


**Figure A-26 System Pressure for the Limiting PCT Case**

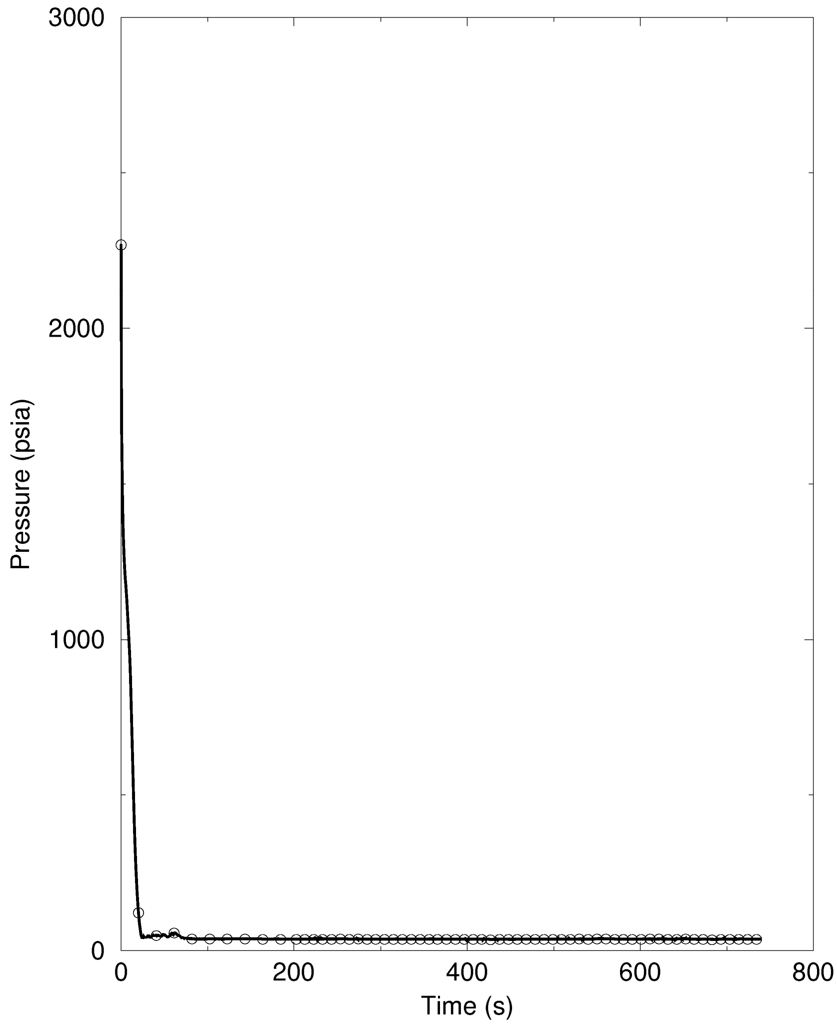


**Figure A-27 Flows Supplied to ECCS (includes Accumulator, MHSI and LHSI) for the Limiting PCT Case**

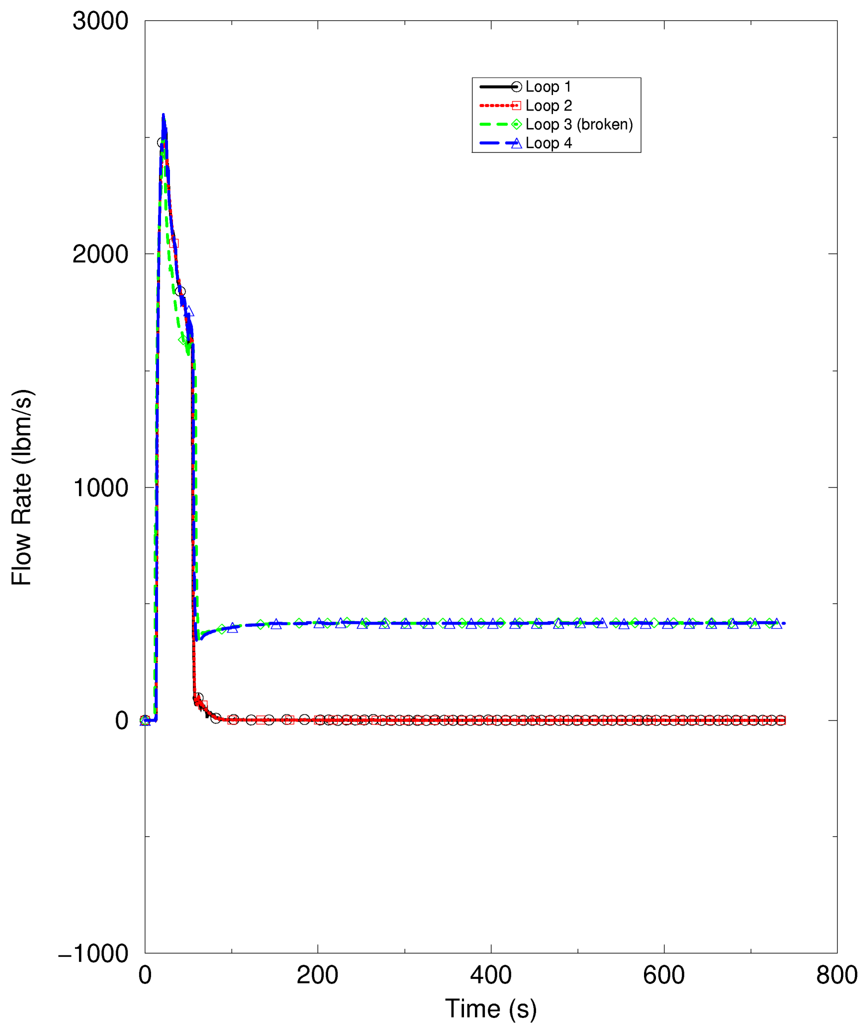


**Figure A-28 Flows Delivered by ECCS for the Limiting PCT Case**

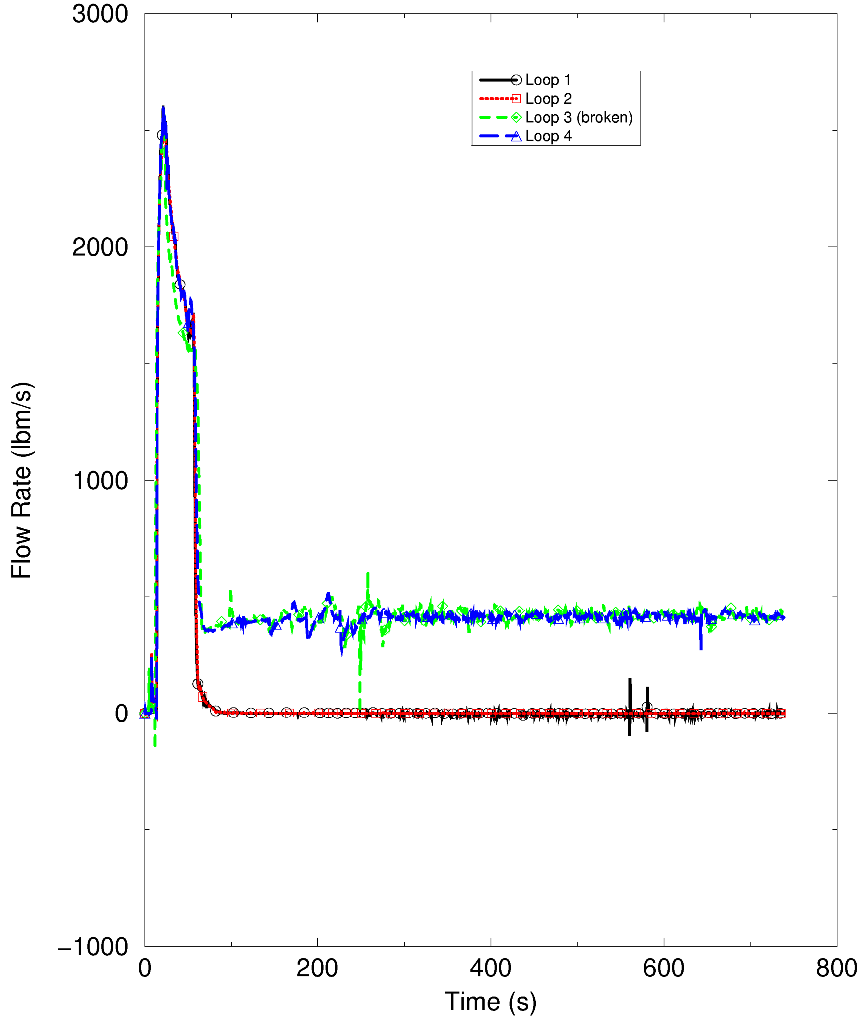


**Figure A-29 Core Inlet Flow for the Limiting PCT Case**

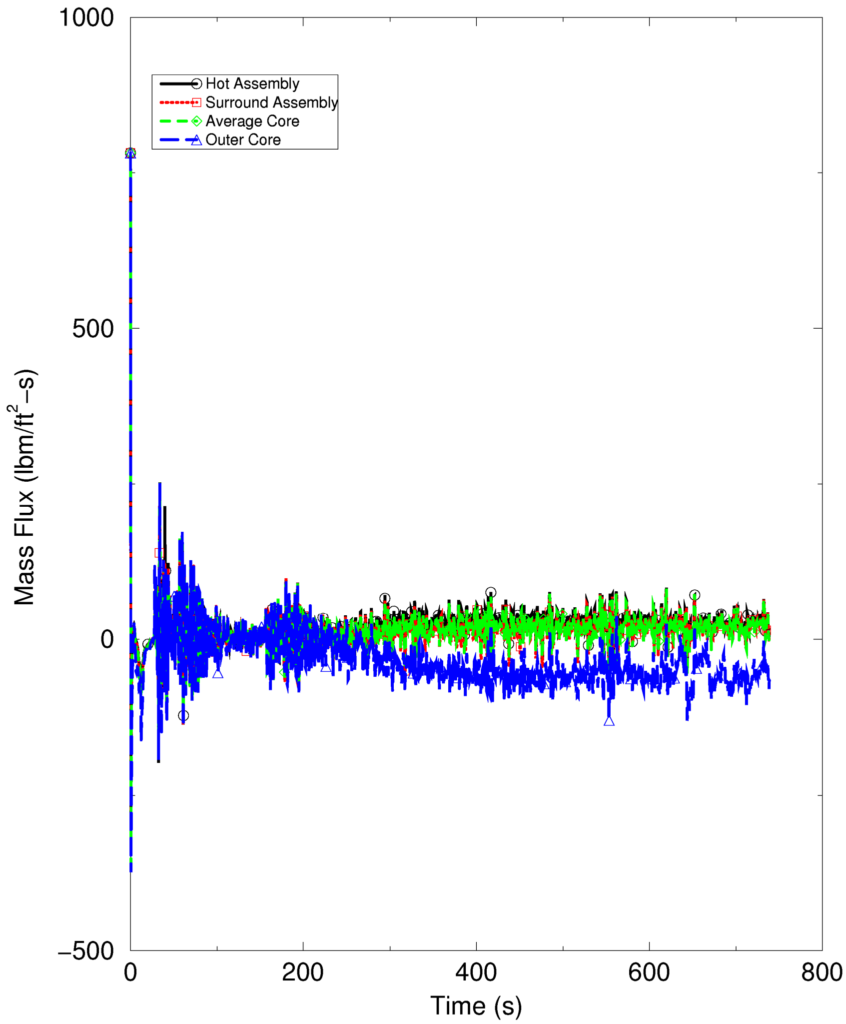


**Figure A-30 Core Outlet Flow for the Limiting PCT Case**

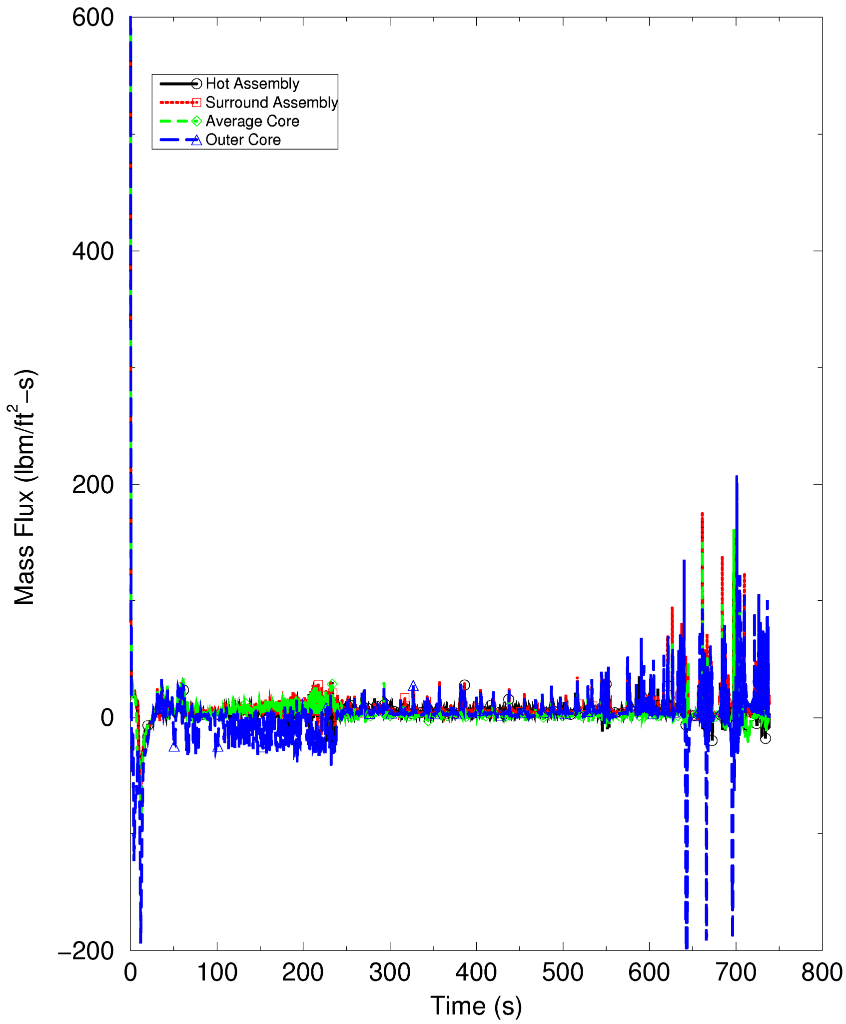


**Figure A-31  Break Flow for the Limiting PCT Case**

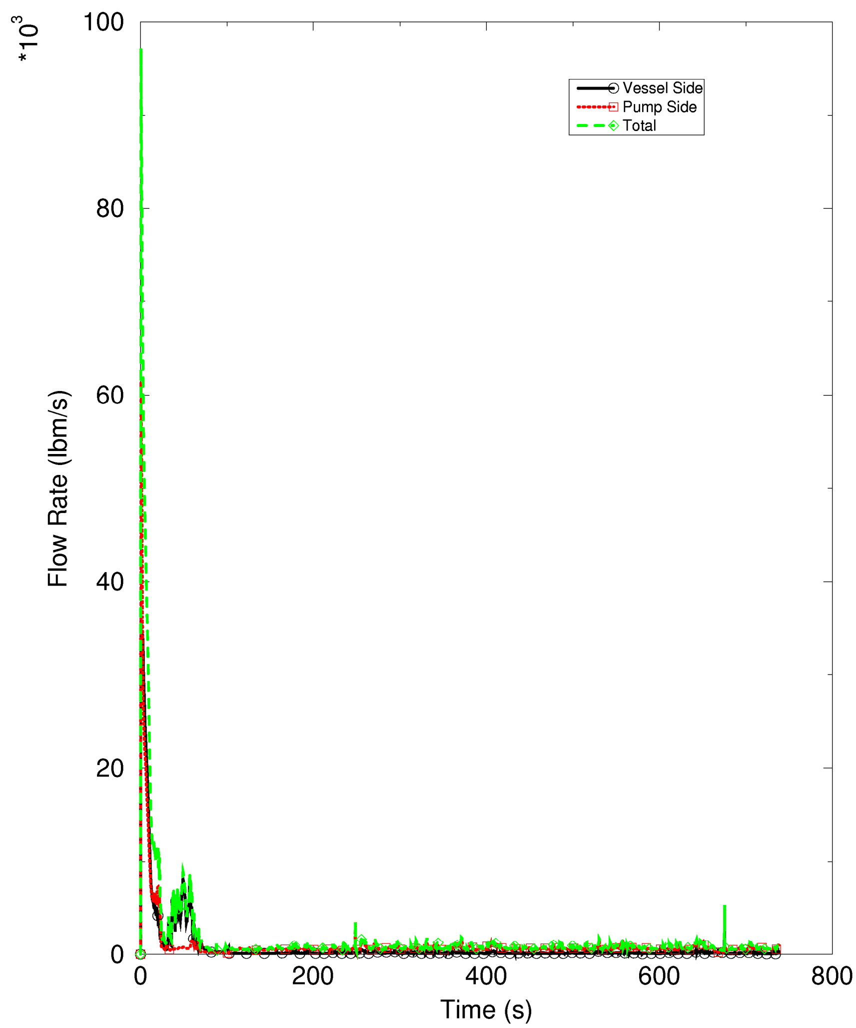


**Figure A-32 Collapsed Liquid Level in the Downcomer for the Limiting PCT Case**

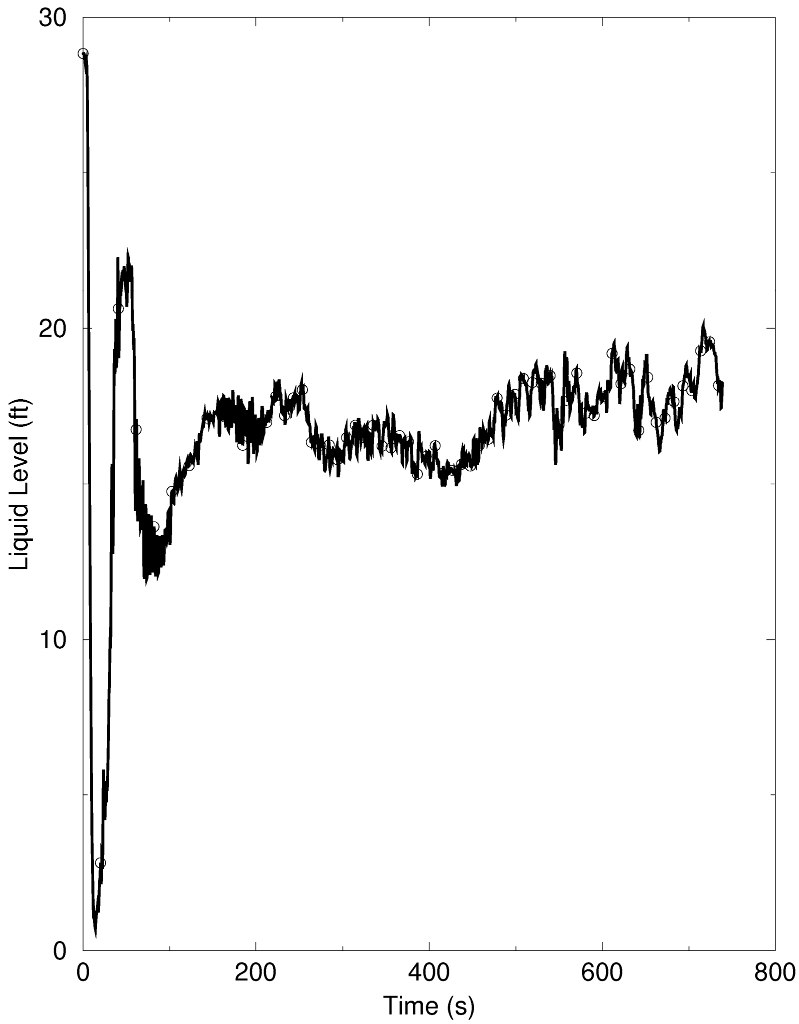


**Figure A-33 Core Liquid Level for the Limiting PCT Case**

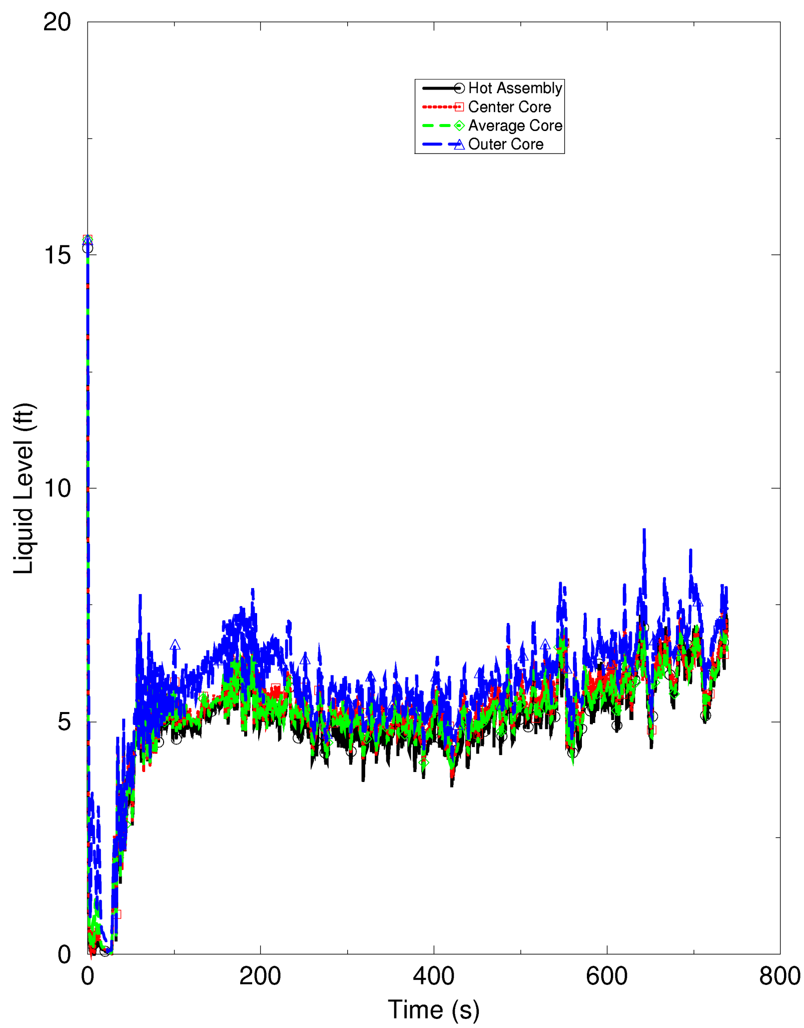


**Figure A-34 Reactor Power for the Limiting PCT Case**

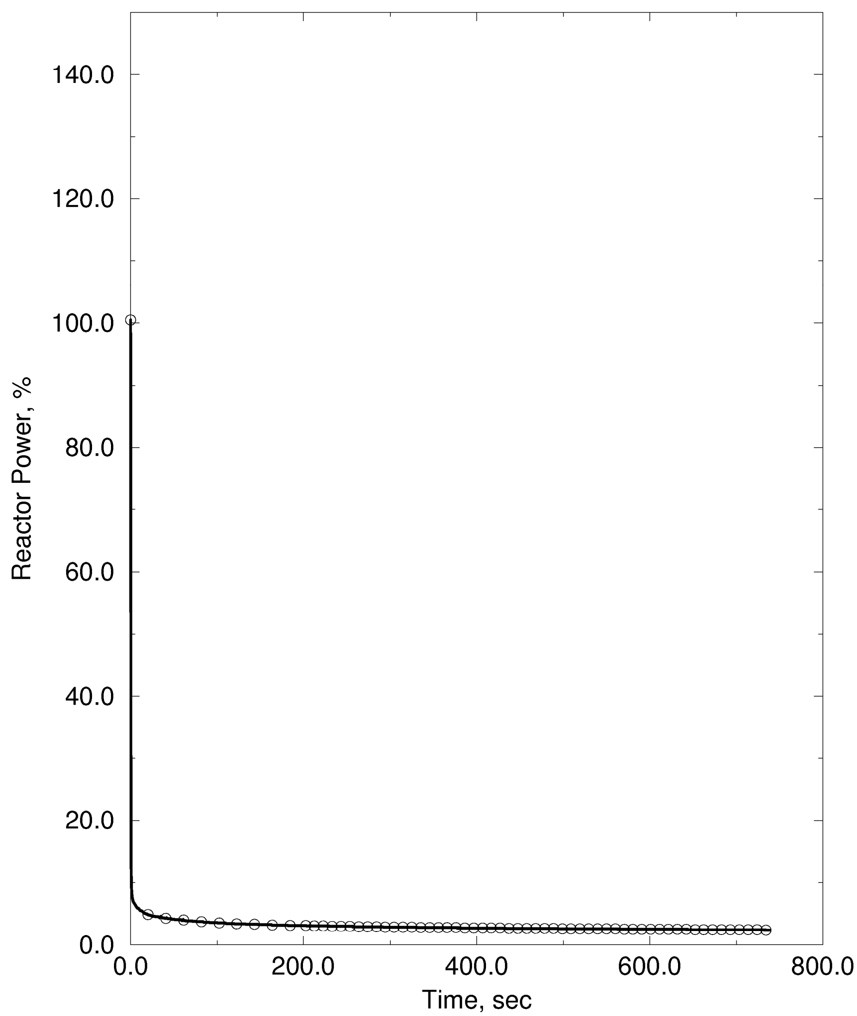


**Figure A-35 Secondary Pressure for the Limiting PCT Case**

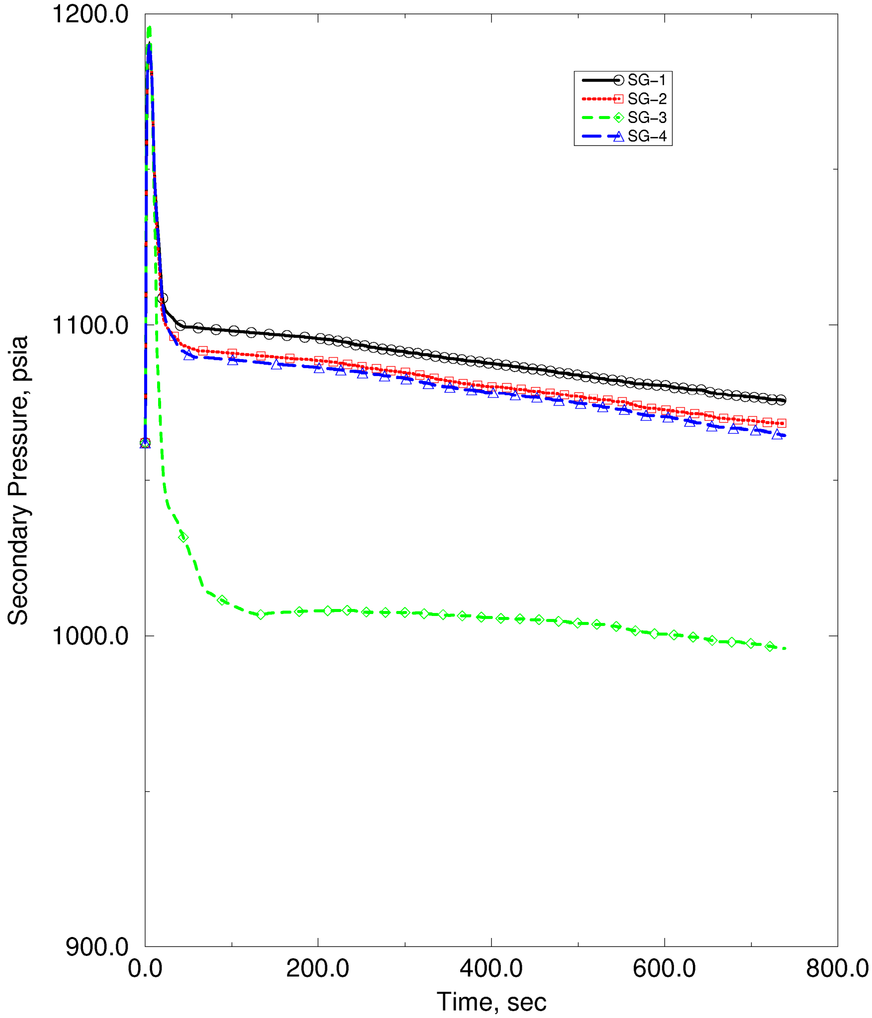


**Figure A-36 Downcomer Mass Flowrate for the Limiting PCT Case**

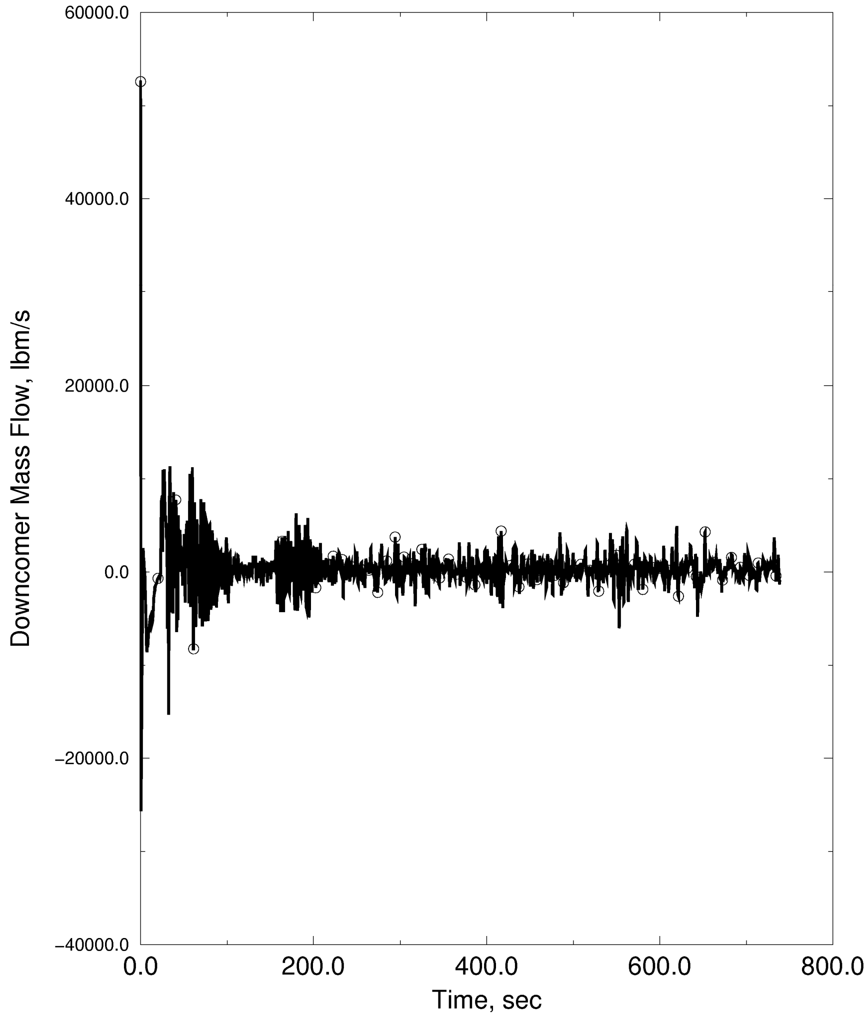


**Figure A-37 Core Inlet Temperature for the Limiting PCT Case**

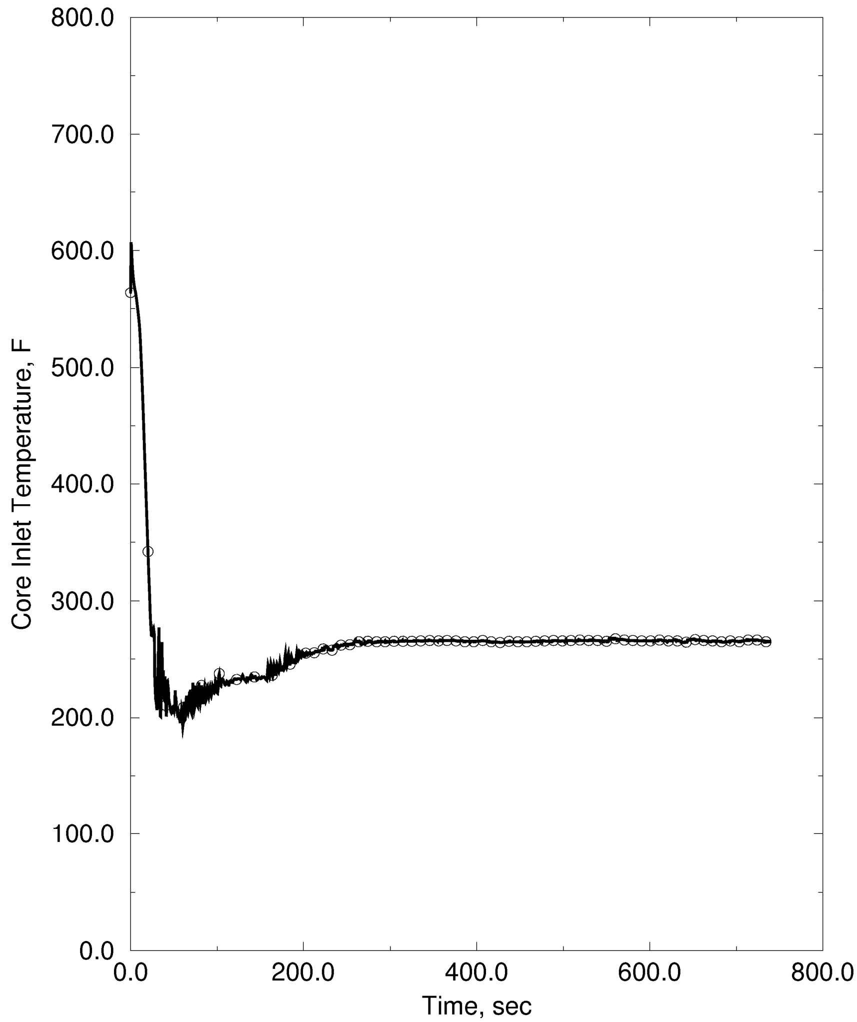


**Figure A-38 Core Inlet Quality for the Limiting PCT Case**

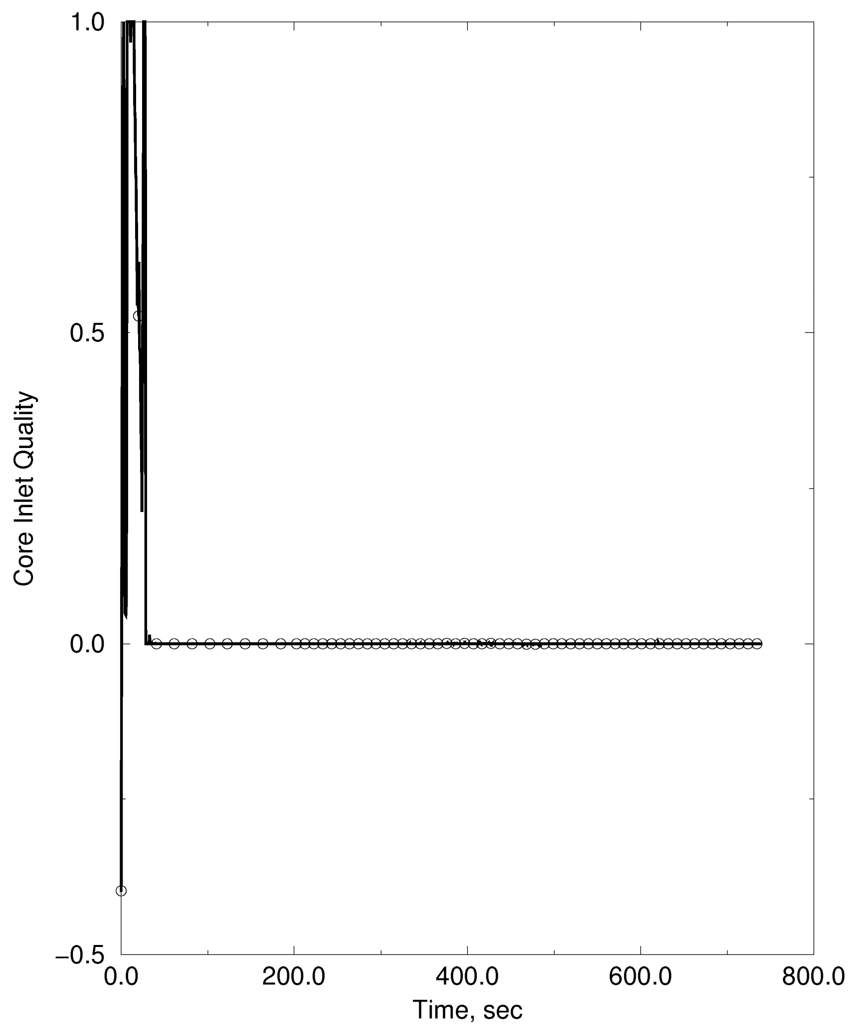
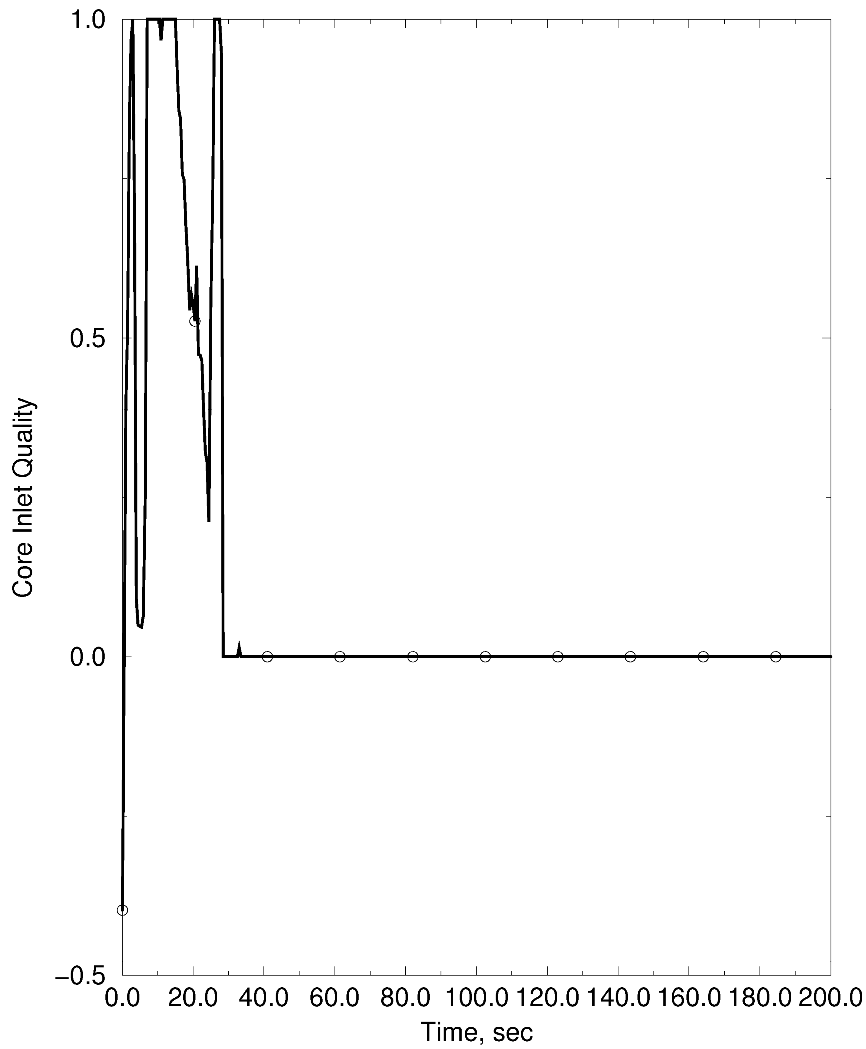


**Figure A-39  Core Inlet Quality for the Limiting PCT Case on Smaller Time Scale**



## Figure A-40 Core Outlet Temperature for the Limiting PCT Case

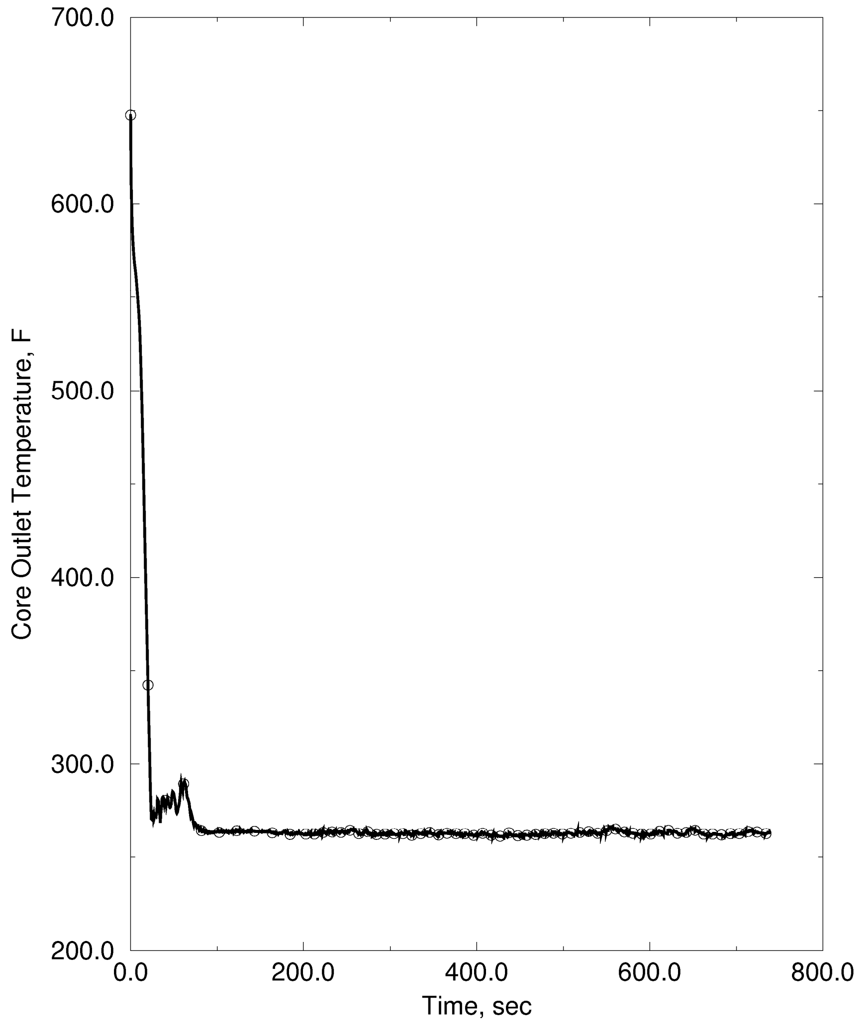


**Figure A-41 Core Outlet Quality for the Limiting PCT Case**

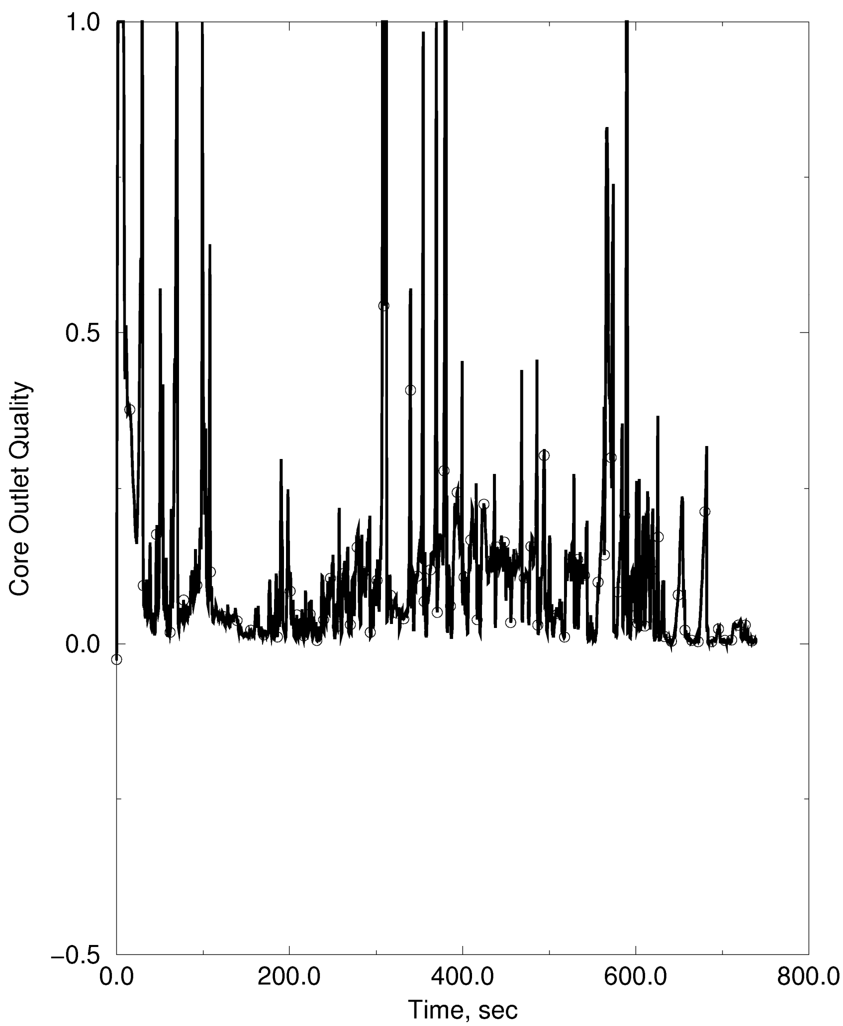


**Figure A-42 Core Outlet Quality for the Limiting PCT Case on Smaller Time Scale**

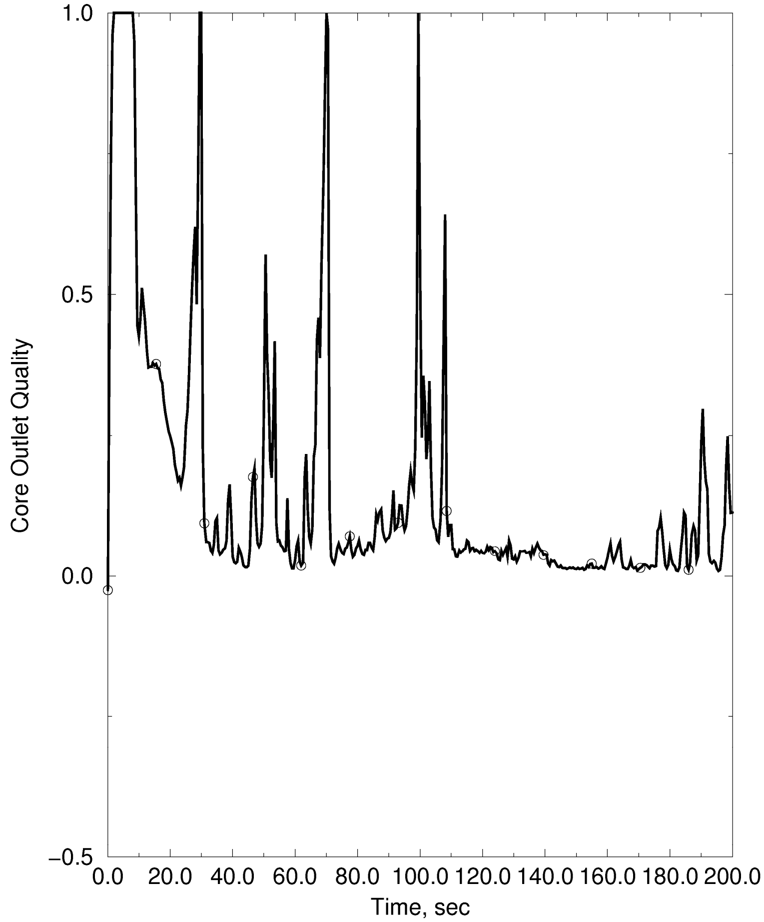


**Figure A-43 In-core Temperature for the Limiting PCT Case**

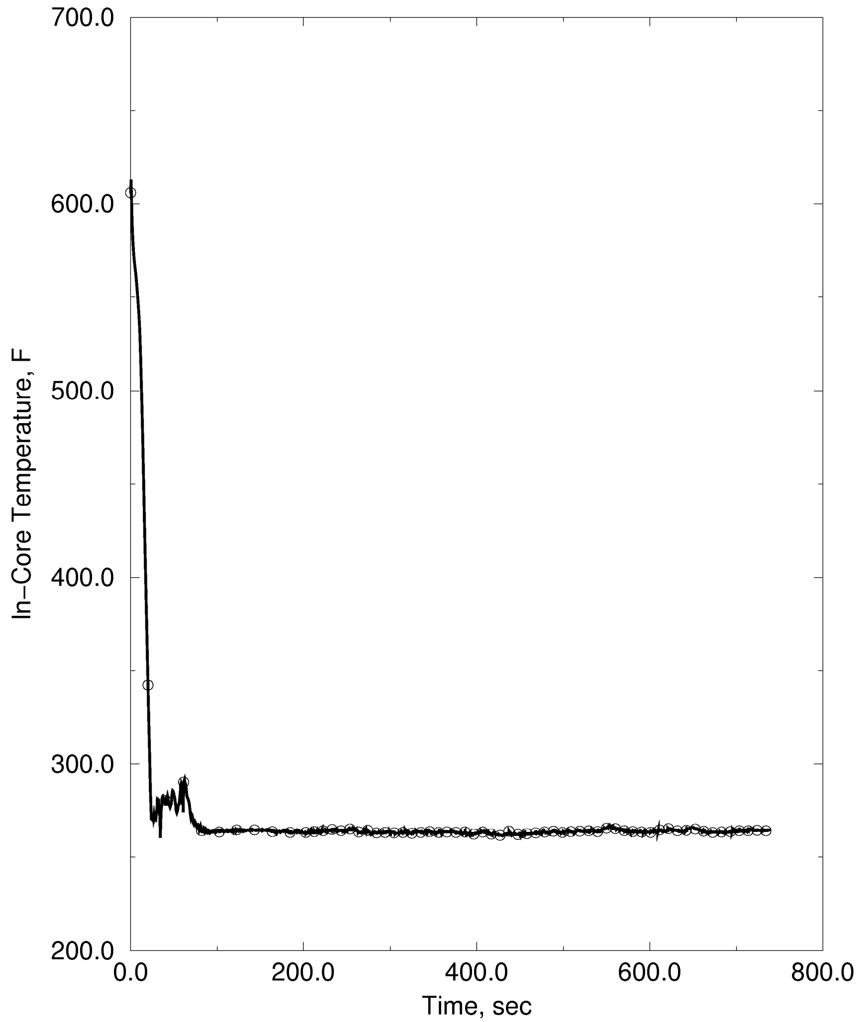


**Figure A-44  In-core Quality for the Limiting PCT Case**

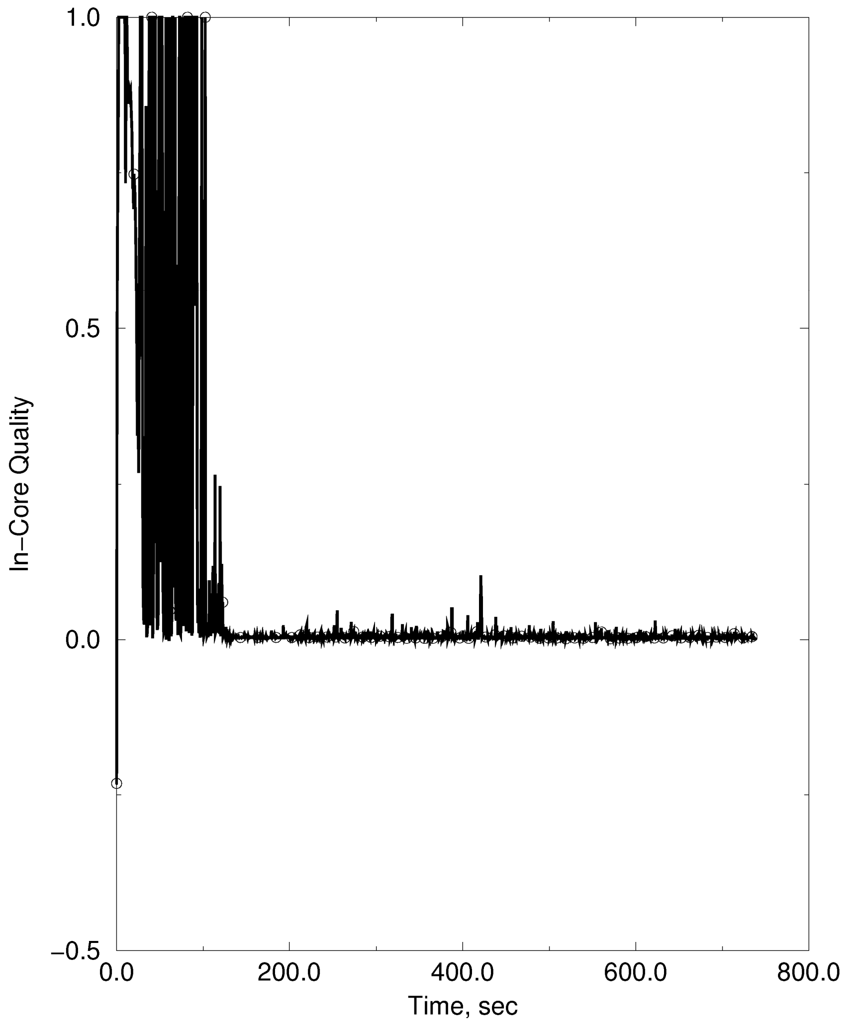


**Figure A-45 In-core Quality for the Limiting PCT Case on Smaller Time Scale**

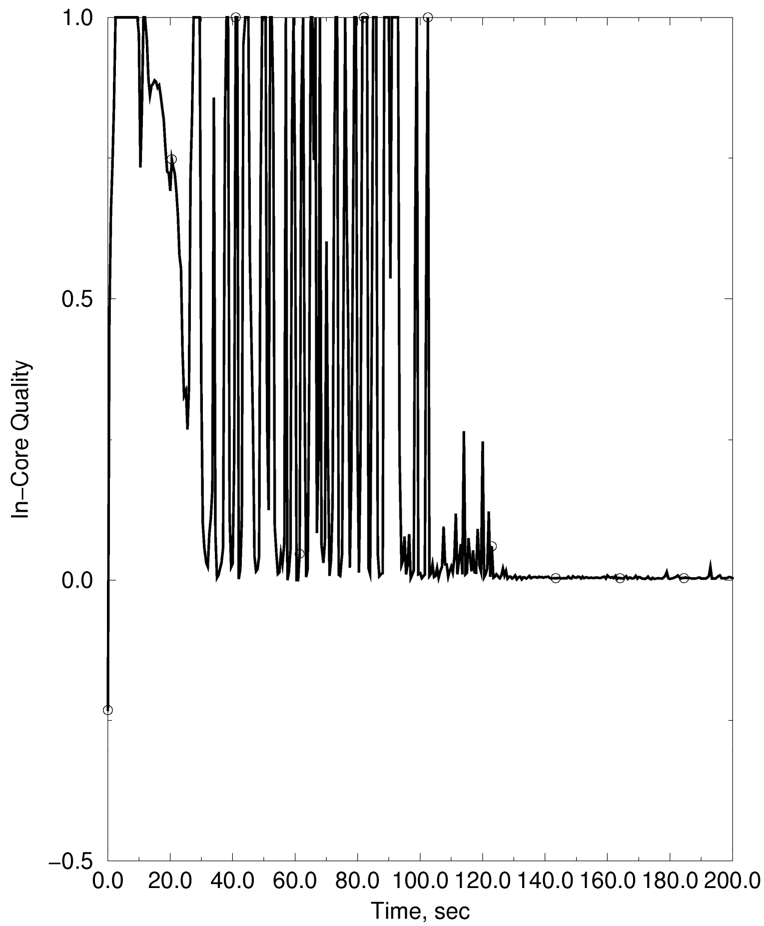


**Figure A-46 Cladding Temperature for the Limiting PCT Case**

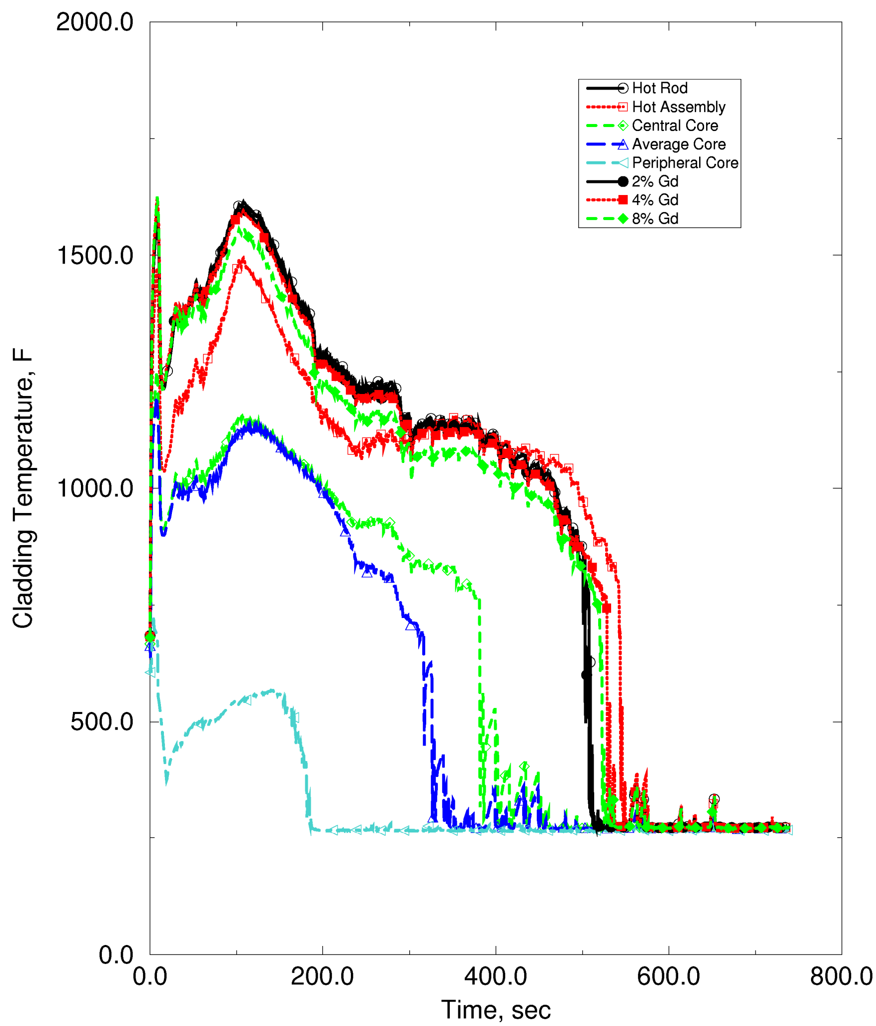


**Figure A-47 Heat Transfer Coefficient for the Limiting PCT Case**

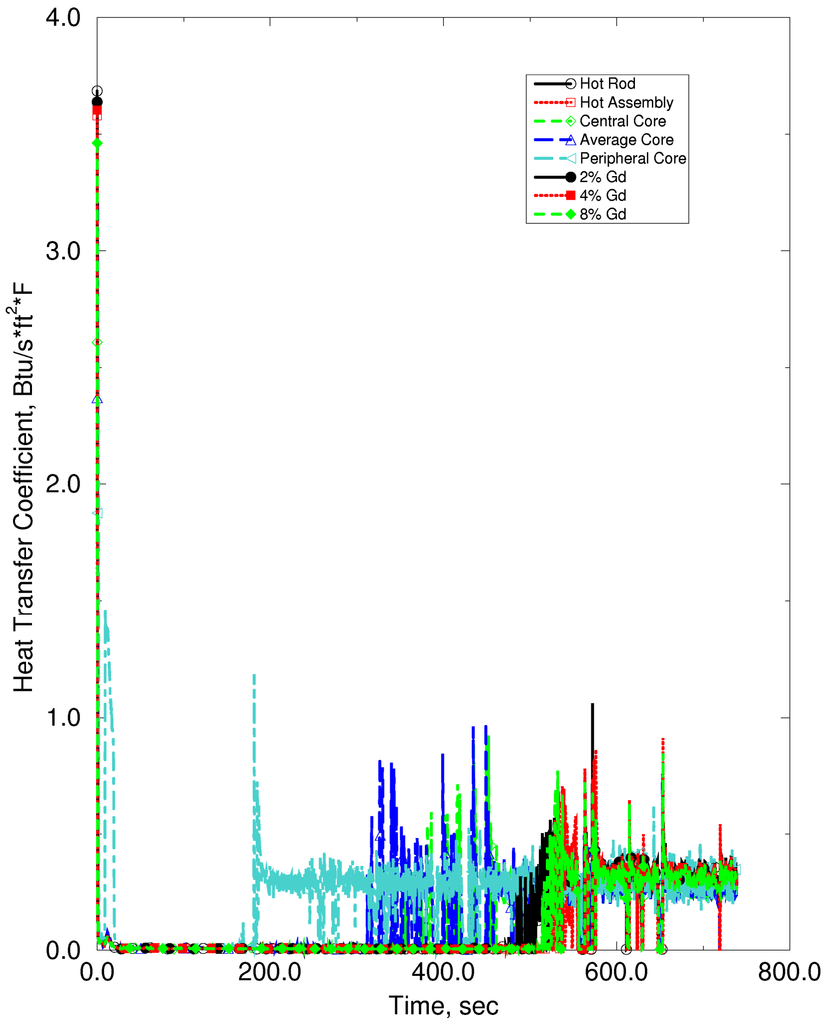


**Figure A-48 Primary to Secondary Heat Transfer Rate for the Limiting PCT Case**

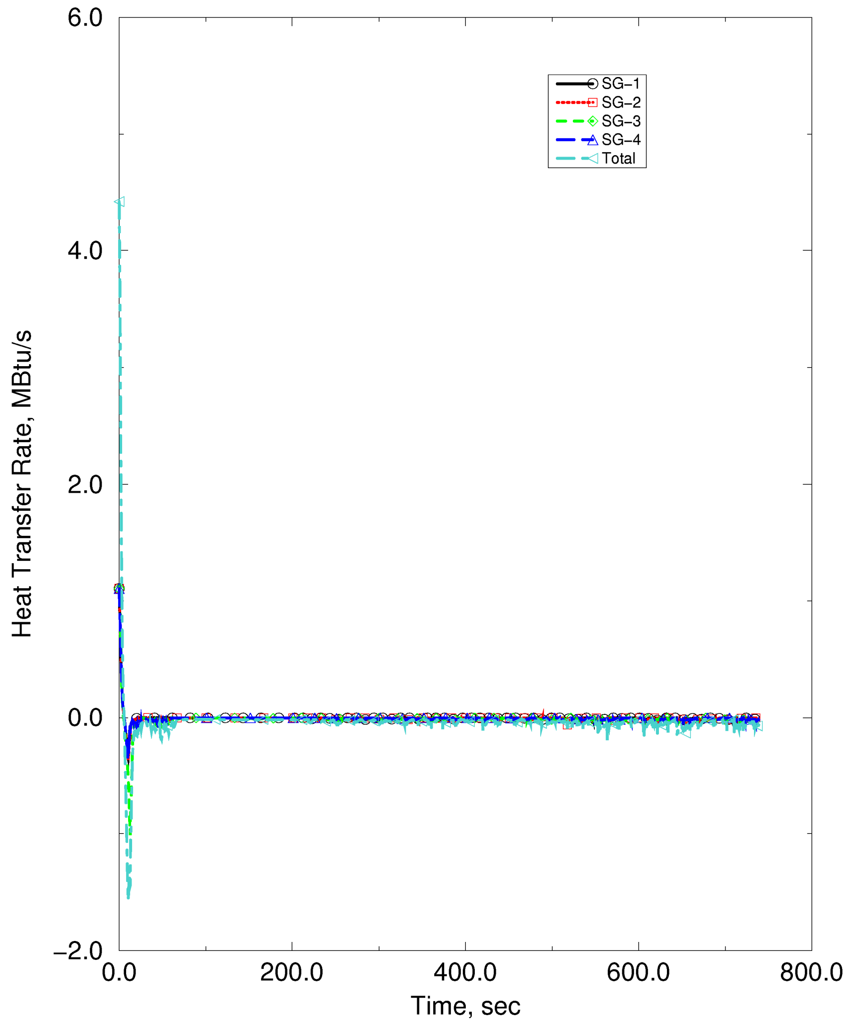


**Figure A-49 Pump Speed for the Limiting PCT Case**

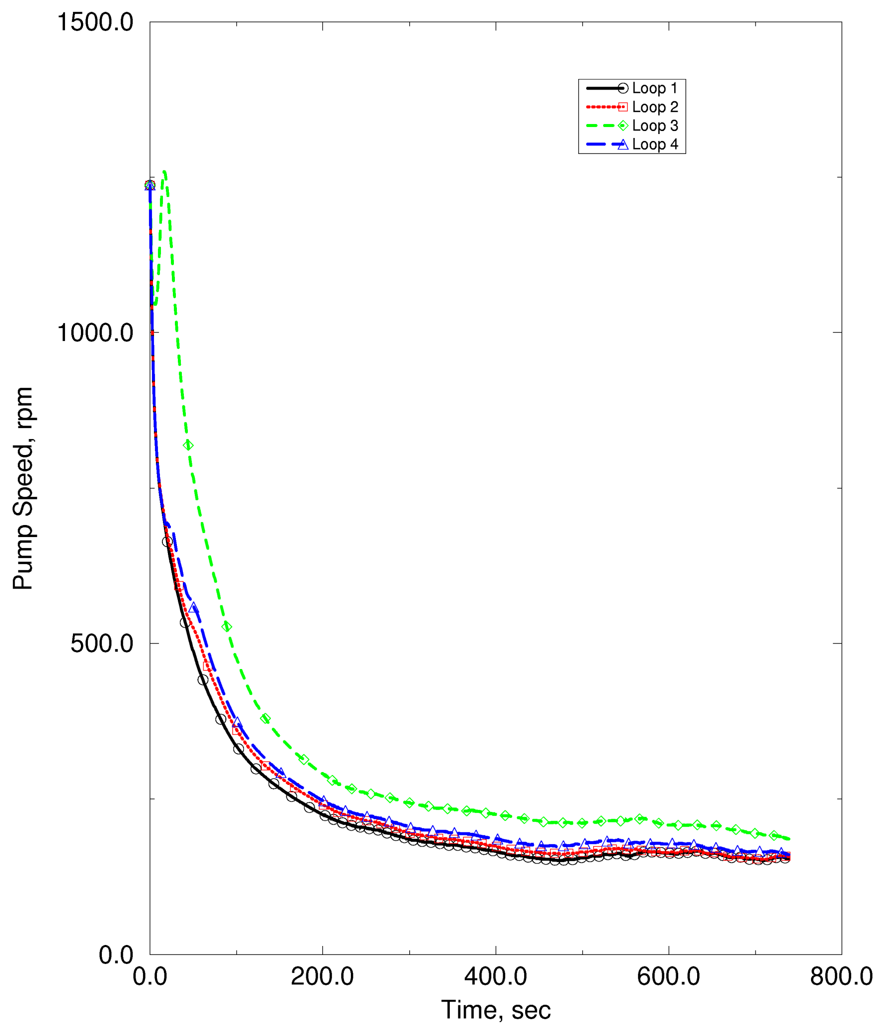
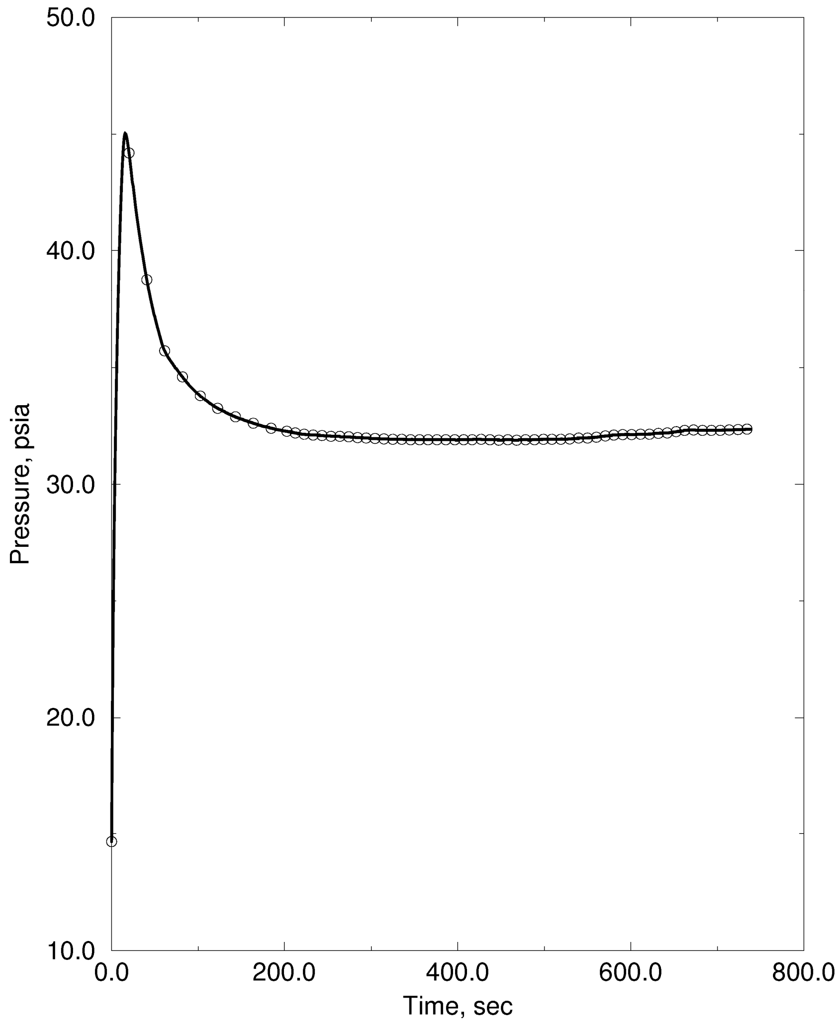


**Figure A-50  Containment Pressure for the Limiting PCT Case**



# APPENDIX B

# ICECON CONTAINMENT MODEL

# APPENDIX B

## TABLE OF CONTENTS

**Page**



## List of Tables



## List of Figures

## B.1.0 *Introduction and Summary*

The purpose of this appendix is to describe the U.S. EPR ICECON containment model used by the RLBLOCA EM to calculate containment back pressure during an LBLOCA. The dominant phenomenon of interest related to the containment model is the effect of containment pressure on PCT. Containment pressure is treated statistically in the RLBLOCA EM by ranging the containment volume from the best-estimate value to the maximum possible free volume. This biases the containment pressure response low, which is conservative for predicting PCT.

The containment pressure response is calculated dynamically coincident with the S-RELAP5 calculation using ICECON, a digital computer program written in FORTRAN IV, to predict the pressure response of PWR nuclear reactor containment systems to a postulated loss-of-coolant accident (LOCA). ICECON is capable of performing both best-estimate and Appendix K to 10 CFR 50 ECCS analyses through the choice of input. The calculated containment pressure is used to determine the back pressure for flow from the primary system to the containment during the LOCA transient. The fundamental basis of the ICECON computer code is the CONTEMPT/LT-022 code, to which an ice condenser model was added.

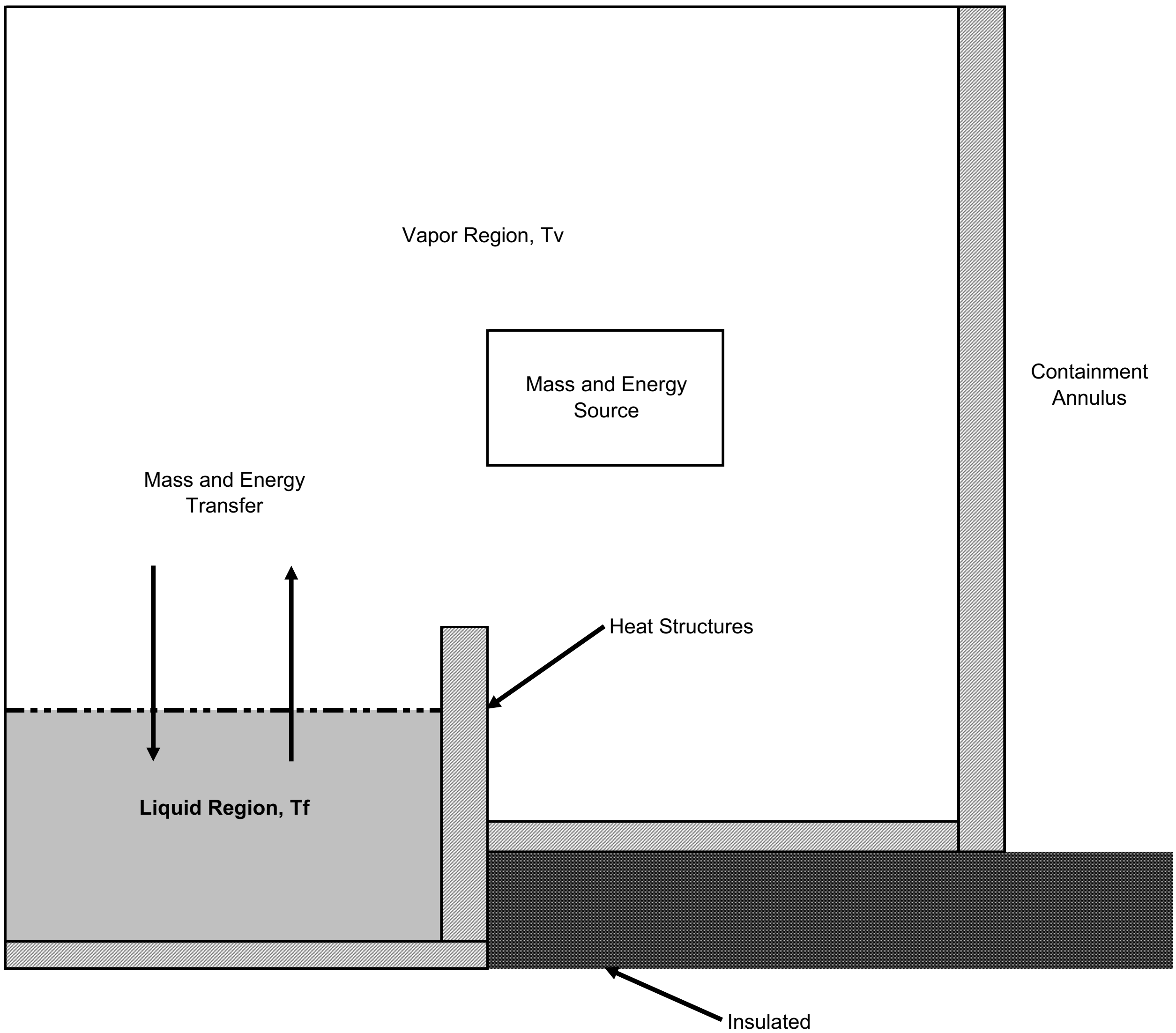
ICECON is capable of simulating pressure-temperature transients in both dry and ice condenser containments. The S-RELAP5/ICECON code interface enables ICECON to be run concurrently with S-RELAP5, providing a dynamic calculation of containment pressure that is consistent with the break-mass flow rate and specific enthalpy currently being generated by S-RELAP5. With the concurrent execution of S-RELAP5 and ICECON, a consistent break-pressure boundary condition is always available in S-RELAP5.

For the S-RELAP5/ICECON code interface, break-flow junction variables (velocities, specific enthalpies, densities, and void fractions) are transferred each time step from S-RELAP5 to ICECON. These variables are used in ICECON to generate a new containment pressure, which is transferred back to S-RELAP5 and used to alter the pressure in the time-dependent volumes representing the containment in the S-RELAP5 model.  At each time step, S-RELAP5 performs the necessary data transfers between the two codes.

The U.S. EPR containment in ICECON is modeled as a dry containment with only one compartment, the drywell compartment.  The reactor vessel and primary system are represented as a mass and energy source to the containment volume.  The ICECON computer code does not have the capability to model a multi-node drywell compartment.

Figure B-1 is a schematic of the U.S. EPR ICECON containment model showing the reactor vessel and containment volumes.  The containment building is modeled as being in contact with the containment annulus on the exterior side.

**Figure B-1 U.S. EPR ICECON Containment Model**




## B.2.0 *Assumptions*

The containment model assumes thermodynamic equilibrium between the spilled ECCS water and containment steam. Consequently, the spilled ECCS water and containment steam are well mixed, which is expected as the LBLOCA progresses through blowdown and reflood. Although the mixing between containment steam and spilled ECCS water is not a dominant phenomenon in determining the peak cladding temperature, it is conservative and bounding to assume thermodynamic equilibrium between the two phases. This assumption conservatively reduces containment pressure, which penalizes clad temperatures by increasing RCS voiding and break flow.

The noncondensable component of the two-phase mixture in S-RELAP5 is assumed to be in thermal and mechanical equilibrium with the vapor phase. The properties for the vapor phase are calculated assuming a Gibbs-Dalton mixture of steam and an ideal noncondensable gas.

ICECON's treatment of a two-component, two-phase mixture of liquid water, water vapor, and noncondensable gas is also based on the assumptions of the Gibbs-Dalton law.

The following assumptions are imposed to conservatively increase energy removal from the containment atmosphere:

- The nominal surface area of the containment heat sinks is conservatively increased by modeling the heat sinks representing the basemat floor in contact with the containment atmosphere.
- A passive heat sink representing uninsulated systems and components is added to the U.S. EPR ICECON model. The nominal surface area and thickness of this heat sink are consistent with the guidance of Reference B-4.
- All the nominal heat-transfer surface areas are increased by 10 percent.
- The air gap layer between the containment liner and concrete is neglected per Reference B-4, Section 3.B (iii). The paint layer, such as that on the containment walls, is assumed to have the material properties of steel. These assumptions eliminate any insulating effects on the exposed surfaces of the heat structures.

The material properties of the liner on the containment wall, the internal steel structures, and the concrete structures are consistent with the guidance of Table 2 of Reference B-4.

## B.3.0 *Containment Free Volume*

The total containment free gas volume, the nominal containment free volume, is 2,819,523 $ft^3$ (79,840 $m^3$). The nominal IRWST water volume is 68,405 $ft^3$ (1937 $m^3$). The sum of the nominal containment free volume and the nominal IRWST water volume yields a combined containment volume of 2,887,927 $ft^3$ (81,777 $m^3$).

The sum of the combined containment volume and the internal structure volume yields the maximum containment free volume, which is 3,933,665 $ft^3$. Using the heat structure volume in the maximum-containment free-volume calculation provides a conservative (bounding) estimate to cover the possibility of equipment removal.

The volume of “Internal Tanks and Equipment” is not considered in either the calculation of the nominal containment free volume or the maximum containment free volume.

## B.4.0 *Total Heat Transfer Surface Area*

The heat sinks represented in the ICECON model include the heat structure groups in the U.S. EPR GOTHIC containment model. The assumptions used to develop the GOTHIC model are reviewed for applicability to a conservative minimum back-pressure calculation. This assessment confirms that the ICECON model includes heat sinks that may be conservatively neglected in the GOTHIC model.

The updated ICECON model includes the following changes to heat structure surface area:

1. The ICECON model treats the containment walls and the in-containment refueling water storage tank (IRWST) walls as two-sided heat structures. The IRWST walls are in contact with water on one side and the containment atmosphere on the other. The containment walls are in contact with the containment annulus on one side and the containment atmosphere on the other.

    a. The containment annulus is modeled at 45ºF, which is the minimum winter design value. The heat transfer coefficient for heat transfer to the containment annulus is 5.0 Btu/hr-ft$^2$-ºF. The value of 5 Btu/hr-ft$^2$-ºF is used for free convection in air and is the upper range of values for a free convection application, as stated in Principles of Heat Transfer (Reference B-1).

2. The ICECON model considers an increase in the surface area of uninsulated systems and components. The surface area of this additional heat sink, named “Internal Tanks and Equipment,” is determined so that the total exposed internal heat sink area in ICECON is consistent with the total internal steel heat sink area recommended in Figure 1 of Reference B-4. As explained in Section B.3.0 the combined containment volume shown is 81,777 m$^3$. In accordance with Figure 1 of Reference B-4, the total internal heat sink area is 3.5x10$^4$ m$^2$ (376,737 ft$^2$) with a thickness of 0.375 in (0.03125 ft). It is assumed that the containment free volume in Figure 1 of Reference B-4 ranges from 0.0 m$^3$ to 1.2x10$^5$ m$^3$.

3. As stated in Section B.2.0 all the nominal heat transfer surface areas are increased by 10 percent to increase the energy removed from the containment atmosphere.

Containment and subcompartment design parameters are presented in Table B-1 and Table B-2.

## Table B-1 Initial Conditions

| Parameter | Additional Information | Value | Notes |
|---|---|---|---|
| Initial outside air (annulus) temperature (°F) | | 45 | Minimum annulus temperature based on winter design conditions |
| Initial outside air (annulus) absolute pressure (psia) | | 14.667 | Annulus pressure during normal operation (14.696 psia - 0.029 psi) |
| Initial outside air (annulus) humidity | | 0.7 | Annulus maximum relative humidity |
| Constant temperature (°F) of insulated boundary specified for heat structure | max (°F) | 131 | Value overridden by S-RELAP5, sampled parameter |
| | min (°F) | 100 | |
| Total compartment volume ($ft^3$) | nominal value ($ft^3$) | 2,887,927 | Combined containment volume |
| | bounding value ($ft^3$) | 3,933,665 | Maximum containment volume = (nominal total compartment volume) + (internal structure volume). Maximum volume lowers containment pressure, which reduces PCT margins. |
| Volume of liquid pool on floor | nominal value ($ft^3$) | 68,405 | Nominal IRWST water volume |
| Temperature of vapor region (°F) | max (°F) | 131 | Value overridden by S-RELAP5, sampled parameter |
| | min (°F) | 100 | |
| Temperature of liquid pool region (°F) | max (°F) | 131 | Value overridden by S-RELAP5, sampled parameter |
| | min (°F) | 100 | |
| Total compartment absolute pressure (psia) | nominal value (psia) | 14.664 | Average of nominal subpressure values for service (-0.8" $H_2O$) and equipment (-1.2" $H_2O$) compartments |
| Relative humidity of vapor region | bounding value | 1 | Assumed maximum |
| Horizontal cross-sectional area of compartment, used for liquid pool surface area and needed in evaporation model ($ft^2$) | | 6986.1 | IRWST pool surface area with 1.1 multiplier applied (IRWST slabs to liquid) |

## Table B-2 Heat Structure Mesh Point Intervals

<table>
<tr><th rowspan="2">Number</th><th rowspan="2">Heat Structure Description</th><th colspan="2">Thickness (ft)</th><th colspan="3">Additional Steel Thickness, $T_{ADD}$</th><th>Total Steel Thickness</th><th>Heat Structure Volume ($ft^3$)</th><th colspan="2">Surface Area ($ft^2$)</th><th colspan="2">Heat Transfer Boundary</th></tr>
<tr><th>Concrete</th><th>Steel</th><th>(ft)</th><th>Note</th><th>Layer</th><th>Steel+$T_{ADD}$</th><th>(Note F)</th><th>Nominal</th><th>(Nominal)*1.1</th><th>Outside</th><th>Inside</th></tr>
<tr><td>1</td><td>"Containment walls with Liner"</td><td>3.9892</td><td>0.0208</td><td>0.0007</td><td>E)</td><td>Paint</td><td>0.0215</td><td>395,948</td><td>98,778.3</td><td>108,656.13</td><td>Cont. annulus</td><td rowspan="2">Cont atmos.</td></tr>
<tr><td>2</td><td>"IRWST walls to atmos" (note A)</td><td>4.6</td><td></td><td>0.013</td><td rowspan="4">-</td><td>Steel</td><td>0.013</td><td rowspan="2">33,125</td><td>3166.6</td><td>3483.26</td><td rowspan="2">Cont atmos.</td></tr>
<tr><td>3</td><td>"IRWST walls to liquid" (note B)</td><td>4.6</td><td></td><td>0.013</td><td>Steel</td><td>0.013</td><td>4030.2</td><td>4433.22</td><td>Liquid pool</td></tr>
<tr><td>4</td><td>"IRWST slabs to atmos" (note C)</td><td>4.53</td><td></td><td>0.013</td><td>Steel</td><td>0.013</td><td>26,698</td><td>5890</td><td>6479</td><td rowspan="10">Insulated</td><td>Cont atmos.</td></tr>
<tr><td>5</td><td>"IRWST slabs to liquid" (note D)</td><td>13.12</td><td></td><td>0.013</td><td>Steel</td><td>0.013</td><td>83,343</td><td>6351</td><td>6986.1</td><td>Liquid pool</td></tr>
<tr><td>6</td><td>“Accessible area walls”</td><td>1.28</td><td></td><td>0.0013</td><td rowspan="8">E)</td><td>Paint</td><td>0.0013</td><td>114,278</td><td>89,284.5</td><td>98,212.95</td><td rowspan="8">Cont atmos.</td></tr>
<tr><td>7</td><td>“Non-accessible area walls”</td><td>1.64</td><td></td><td>0.0013</td><td>Paint</td><td>0.0013</td><td>195,255</td><td>118,886.3</td><td>130,774.93</td></tr>
<tr><td>8</td><td>“Accessible area slabs”</td><td>1.39</td><td></td><td>0.003</td><td>Paint</td><td>0.003</td><td>128,969</td><td>92,578.2</td><td>101,836.02</td></tr>
<tr><td>9</td><td>“Non-accessible area slabs”</td><td>1.72</td><td></td><td>0.003</td><td>Paint</td><td>0.003</td><td>60,106</td><td>34,905.2</td><td>38,395.72</td></tr>
<tr><td>10</td><td>“Thick steel”</td><td></td><td>0.13</td><td>0.0007</td><td>Paint</td><td>0.1307</td><td>3602</td><td>26,767.7</td><td>29,444.47</td></tr>
<tr><td>11</td><td>“Medium steel”</td><td></td><td>0.028</td><td>0.0007</td><td>Paint</td><td>0.0287</td><td>3955</td><td>142,410.8</td><td>156,651.88</td></tr>
<tr><td>12</td><td>“Thin steel”</td><td></td><td>0.005</td><td>0.0007</td><td>Paint</td><td>0.0057</td><td>459</td><td>92,998</td><td>102,297.8</td></tr>
<tr><td>13</td><td>“Internal Tanks and Equipment”</td><td></td><td>0.03125</td><td>0.0007</td><td>Paint</td><td>0.03195</td><td>-</td><td>114,590.5</td><td>126,049.55</td></tr>
</table>

Notes:
A) The vertical portion of the IRWST wall above the liquid pool
B) The vertical portion of the IRWST wall contacting the liquid pool
C) The concrete slab partially covering the IRWST liquid pool
D) The concrete slab covered by the IRWST liquid pool.
E) Paint has material properties of steel
F) Heat structure volume used in maximum containment volume calculation

### B.4.1 Heat Transfer to the IRWST

The ICECON model assumes the IRWST liquid is well mixed, so the liquid temperature in the containment vapor space and at the IRWST water interface are at the IRWST bulk liquid temperature. The heat transfer from the pool surface consists of:

- The sensible heat transferred by the temperature gradient.
- The latent heat of the mass transferred by the molar concentration gradient in the vapor.

In the RLBLOCA EM, the only liquid mass transferred to the pool is that which is calculated as condensate. In other words, other than the liquid transferred to the pool through condensation, all the liquid is held up in the atmosphere (ICECON option ALWAYS=0.0)

A sensitivity study performed during the U.S. EPR ICECON model development evaluated the effect of modeling liquid dropout from the atmosphere. For this study, the water drops from the atmosphere region each time-step in the post-blowdown period. Choosing this particular ICECON option deactivates the evaporation-condensation model. In other words, liquid mass transfer to the pool by condensation is not calculated, but the liquid dropout to the pool is modeled in the post-blowdown period (ICECON option ALWAYS=1.0).

The results of the sensitivity study showed that modeling liquid dropout produces a slight decrease in containment back-pressure, which is conservative for the PCT calculation (see Figure B-2). An examination of Figure B-3 shows that the difference in the pressure response begins when the liquid drops out at the end of blowdown time. Modeling liquid dropout increases the pool temperature in Figure B-3. In contrast, the pool temperature where the liquid is completely entrained in the atmosphere shows no discernable change in temperature. Therefore, modeling liquid dropout is conservative and appropriate for the U.S. EPR RLBLOCA analysis. Note that the legend in the figures shown below reflects that the current ICECON model in Revision 1 of Reference 1 models liquid dropout (ALWAYS=1.0).

**Figure B-2 Containment Pressure**

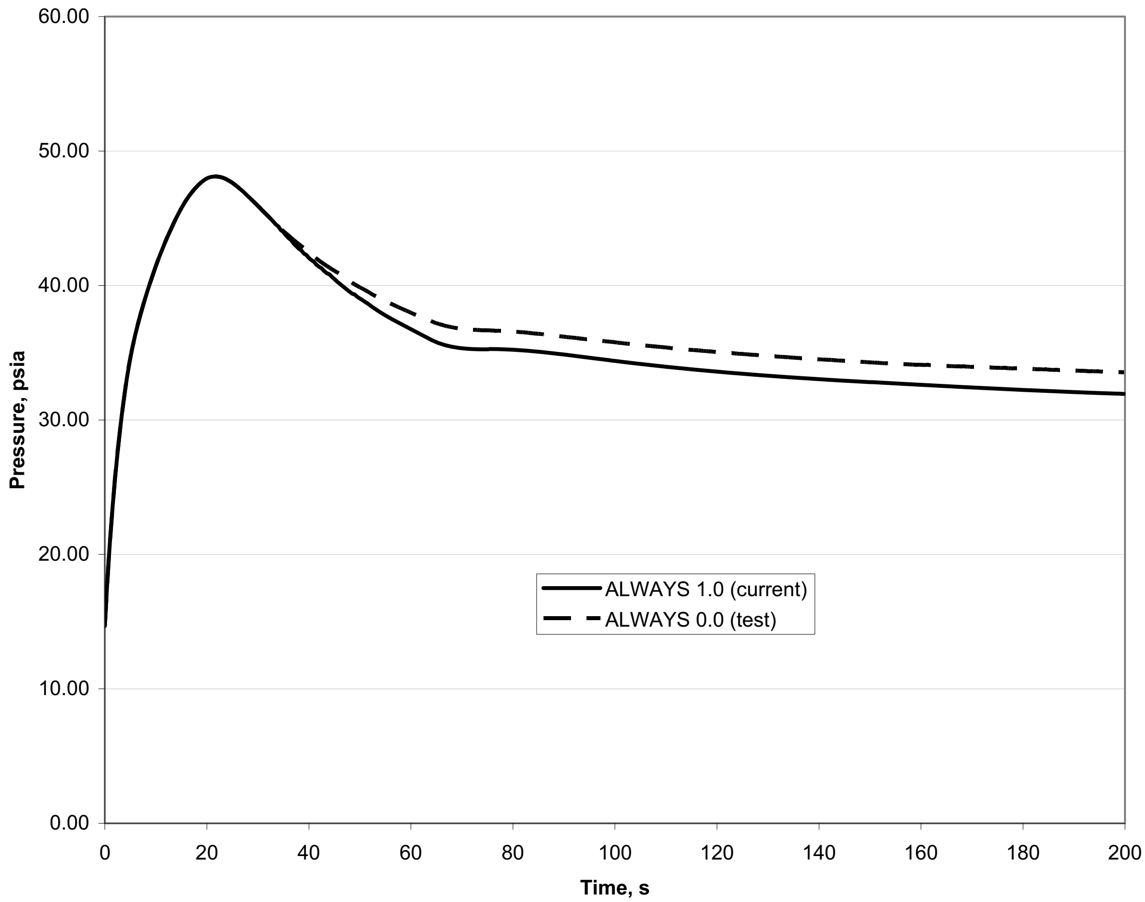
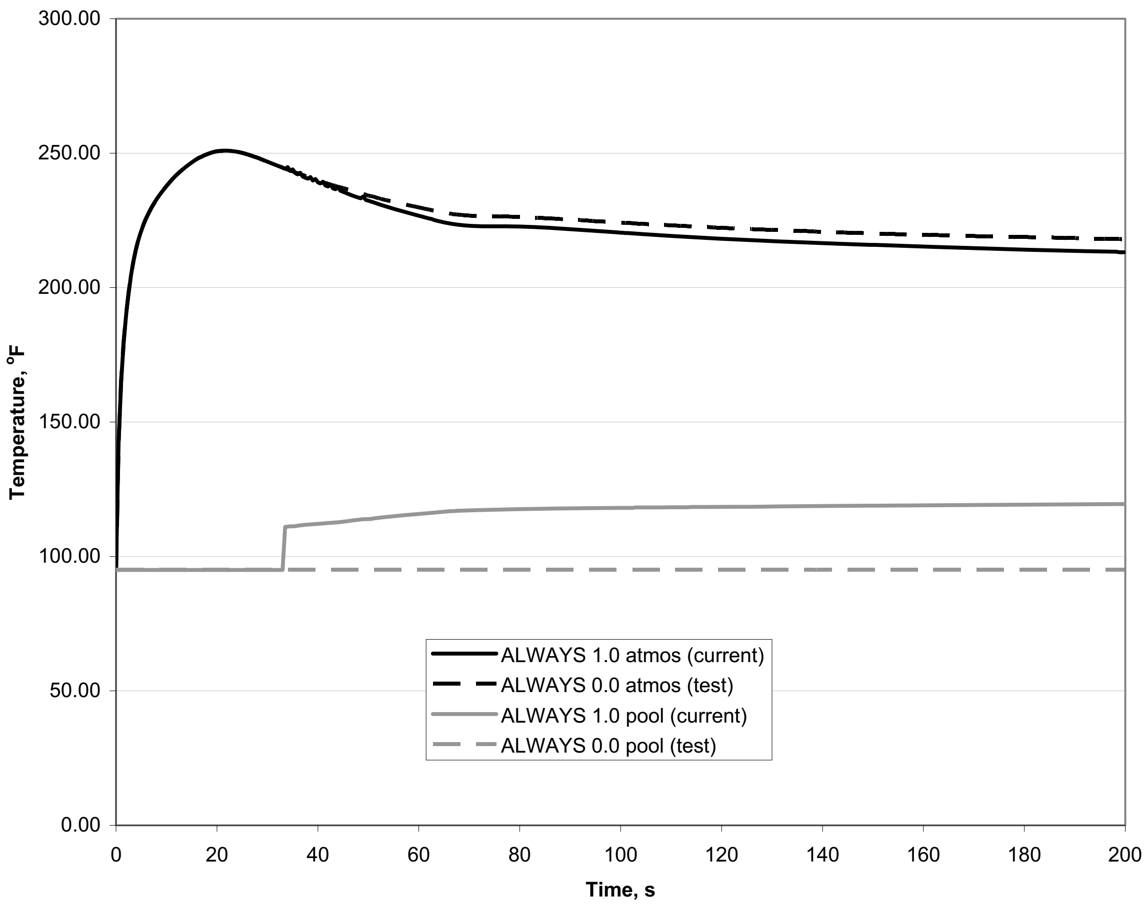


**Figure B-3 Atmosphere and Pool Temperature**




## B.5.0 *Model Validation*

The value of the Tagami coefficient is 72.5 Btu/hr-ft$^2$-$^o$F for the best-estimate Tagami correlation. As shown in Figure 2 of Branch Technical Position 6-2, the conservative Evaluation Model form of this coefficient uses an additional multiplier of 4.0, yielding an EM value of 290.0 Btu/hr-ft$^2$-$^o$F.

The best-estimate value of the Uchida multiplier is 1.0, and the EM value is 1.2. Section 3.4.2 of the RLBLOCA EM states that ICECON was originally approved for calculating a conservative containment back pressure under Appendix K rules, but that it can be used with realistic input to give an approximate realistic back-pressure calculation. The specific changes made to ICECON for a realistic calculation include removing the conservative evaluation model multipliers on Tagami and Uchida correlations and replacing the Tagami correlation with the equivalent Uchida correlation with a multiplier of 1.7. That is, the containment pressure comparison is made between 1.0 Tagami plus 1.0 Uchida and 1.7 Uchida.

Figure B-4 shows the results of a study used to confirm the 1.7 Uchida multiplier in the U.S. EPR ICECON containment model. Figure B-4 presents the containment pressure histories for five scenarios, which are described below.

1. **Full System, 1.7 Uchida** - This scenario transfers the break-flow junction mass flow rate, specific enthalpies, densities, and void fractions each time step from RELAP5 to ICECON. These variables are then used in ICECON to generate a new containment pressure, which is transferred back to RELAP5. This is the containment pressure shown in Figure B-4. The condensation heat transfer correlation is Uchida with a 1.7 multiplier.

2. **Uchida 1.7** – To exercise the full Tagami-Uchida correlation, the blowdown break-flow history must be known in advance, with the break mass and enthalpy history being supplied to the containment model via a set of direct heat addition input in ICECON. The enthalpy flow rate in this input is automatically integrated with respect to time until the end of blowdown time to obtain the blowdown energy deposition required to calculate the Tagami correlation. Therefore, the mass and energy release from the break in the full system calculation described above is tabulated. This table is used in a standalone ICECON model. This scenario uses the standalone ICECON model with a Uchida multiplier of 1.7. Based on the resolution of the tabulated mass and energy input, this pressure history is below that of the full system, described above.

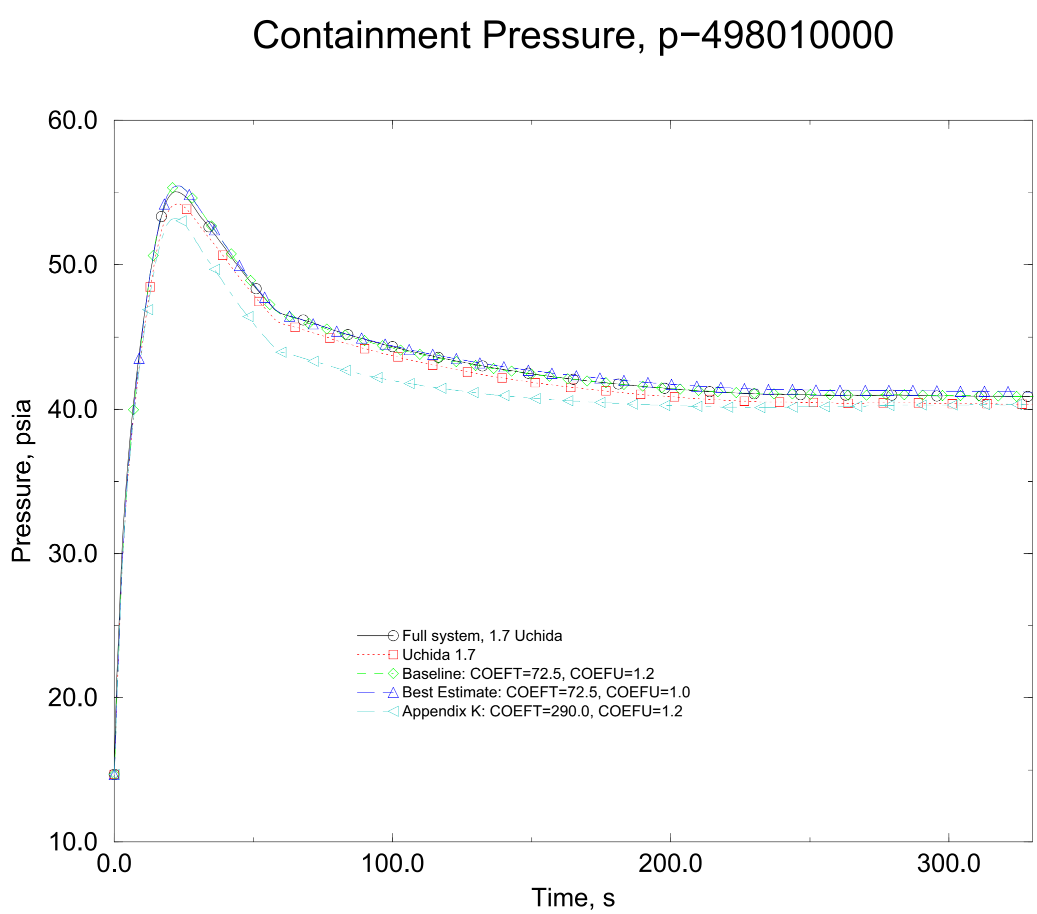
The 1.7 multiplier is a derived coefficient. Figure B-5 describes the iterative process by which this multiplier is tested and adjusted for application in the U.S. EPR RLBLOCA analysis. As shown in Figure B-6, the 1.7 coefficient on the Uchida correlation is conservative with respect to experimental data.

3. **Baseline, COEFT=72.5 plus COEFU=1.2** – This scenario uses the standalone ICECON model described in part 2 along with the best-estimate Tagami coefficient and the EM Uchida multiplier.

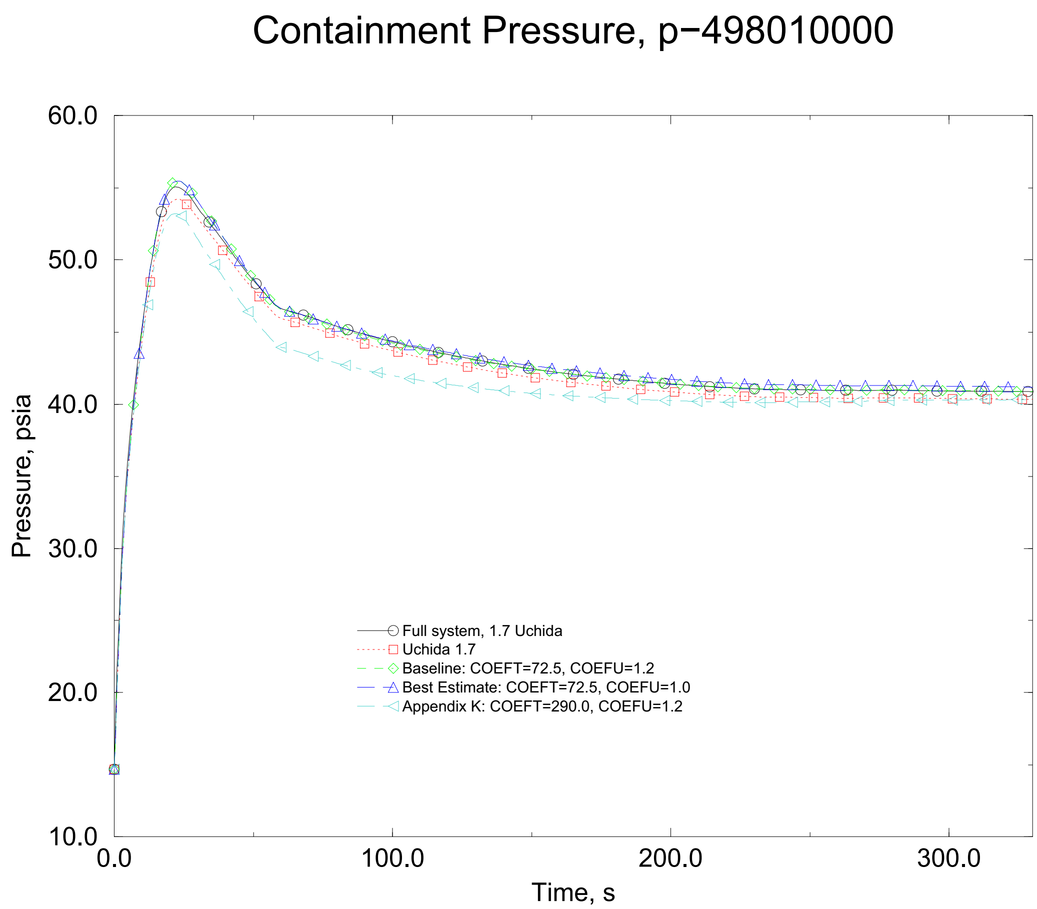
4. **Best Estimate, COEFT=72.5 plus COEFU=1.0** – This scenario uses the standalone ICECON model described in part 2 along with the best-estimate Tagami and Uchida coefficients. For a successful confirmation of the equivalent Uchida multiplier in the U.S. EPR design, the pressure history for “Uchida 1.7” is less than 1 psi greater than the benchmark case after the end of blowdown. As shown in Figure B-4, the equivalent Uchida multiplier is verified using either the Baseline or Best-Estimate correlation. The “Uchida 1.7” pressure history is below that of both the Baseline and Best-Estimate scenarios. It was decided that the Uchida multiplier benchmark will be performed using the best-estimate Tagami and Uchida coefficients (Figure B-5).

5. **Appendix K, COEFT=290.0 plus COEFU=1.2** – This scenario uses the EM values of Tagami and Uchida. As expected, this combination produces a conservative containment pressure response.

## Figure B-4 Comparison of Containment Pressure Histories



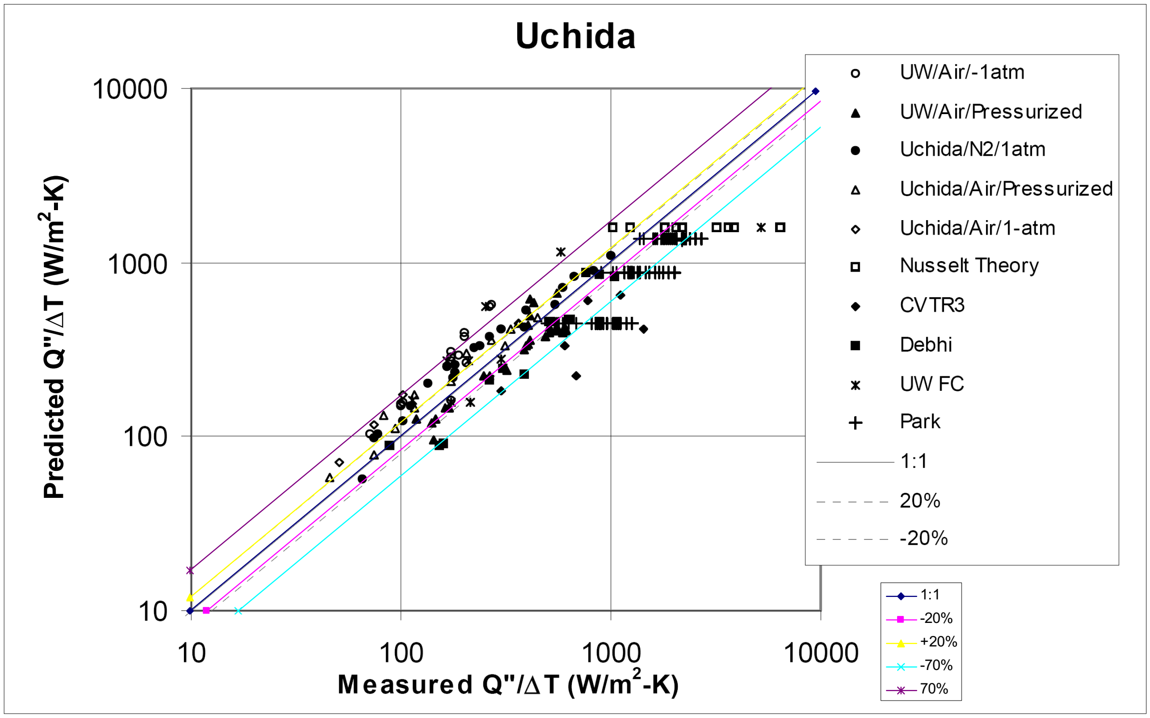

ID:21966 17May2007 09:02:59 ICECONc_4.dmx




ID:21966 17May2007 09:02:59 ICECONc_4.dmx

**Figure B-5 Uchida Multiplier Benchmark Flow Diagram**

**Figure B-6  Experiment Versus Uchida Heat and Mass Transfer Option**




### B.5.1.1 Comparison of ICECON Model to Equivalent Single-Node GOTHIC Model

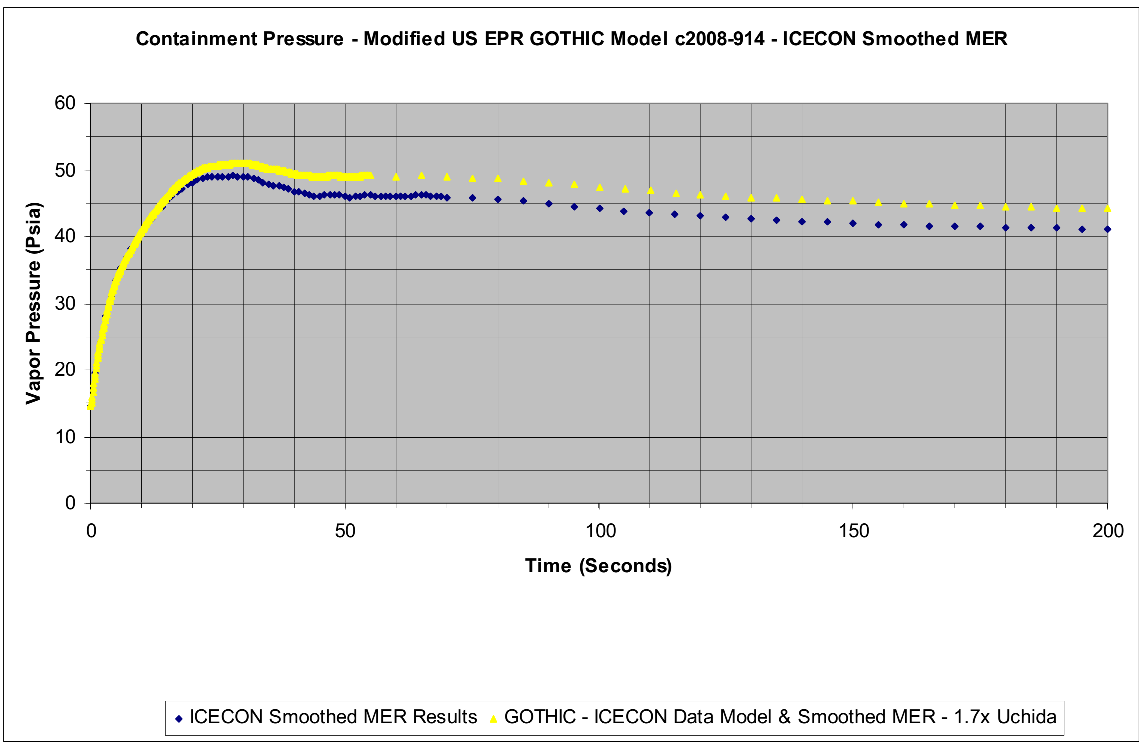
The U.S. EPR ICECON containment model is benchmarked to an equivalent GOTHIC model to confirm that ICECON properly predicts containment pressure for the U.S. EPR RLBLOCA analysis.  The benchmark uses the ICECON model from the sample problem presented in Appendix A.

The physical characteristics of the containment and associated thermal conductors in the GOTHIC model match those used in the ICECON model.  In the ICECON model, the air gap between the containment liner and concrete is neglected.  In the GOTHIC model, however, the air gap is modeled with steel properties.  This is not a significant difference.  In the GOTHIC model, only those boundary conditions necessary to mimic the mass and energy release in the ICECON model.

An integrated mass and energy release from S-RELAP5 is used to create the average mass flow rate and enthalpy input to both GOTHIC and ICECON. Using the integrated mass and energy release is consistent with the Design Certification containment analysis.

Figure B-7 provides the calculated U.S. EPR containment pressure response using the GOTHIC model based on use of input consistent with the ICECON model. Comparison of the GOTHIC and ICECON calculated U.S. EPR containment pressure profiles presented in Figure B-7 shows good agreement between the two codes. ICECON predicts a lower containment pressure, which is conservative for evaluating PCT.

**Figure B-7 Comparison of Containment Pressure Histories – GOTHIC to ICECON Benchmark**




### B.5.1.2 Comparison of ICECON Model to Equivalent Multi-Node GOTHIC Model

The single-node ICECON model also is compared to a multi-node GOTHIC model of the U.S. EPR design. The purpose of this comparison is to confirm the conservatism of the ICECON prediction of the containment back pressure, already demonstrated with the single-node GOTHIC.

The design basis multi-node GOTHIC containment model used for peak pressure in Reference 3 is modified to model a combination of minimum back pressure and best-estimate assumptions. The additional best-estimate assumptions include the following:

(1) Interfacial condensation on the surface of the IRWST pool is credited.

(2) The heat transfer correlations that provide best-estimate modeling of wall-fluid interactions, such as GOTHIC's DLM-F, DLM-M, and DLM-FM, are modeled for the containment heat sinks.

The effect of each parameter is evaluated in addition to the cumulative impact. The sequence of the benchmark cases associated with the individual parameter changes is described below.

- Base Case
  This case used the Mass and Energy (MER) tables from the single-node ICECON model and credited interfacial heat transfer at the surface of the IRWST pool.

- Control Volumes
  The total containment volume of the multi-node base case is increased by approximately 3 percent to match the value used in ICECON.

- Initial Conditions
  The initial conditions representing the containment pressure, vapor and liquid temperatures, relative humidity, and the thermal conductors' initial temperatures are set to the values used in the single-node ICECON model.

- Thermal Conductors
  The thermal conductors (heat sinks) are adjusted to reflect the conditions used in the single-node ICECON model. These adjustments include the addition of a new thermal conductor representing the internal tanks and

equipment, not previously modeled in the GOTHIC models, for conservatism, and a 10 percent surface area increase of all thermal conductors.

- Heat Transfer Coefficients
  The following describes the specific changes made to the heat transfer coefficients:
  i. Containment wall is modified such that its external boundaries are no longer insulated. This change is made to credit heat transfer to ambient and, thus, serves to minimize containment pressure.
  ii. IRWST wall is also modified such that its external boundary is no longer insulated but set to the same heat transfer coefficient as the internal boundary. This same heat transfer coefficient is modified from natural convection (to liquid) to DLM. This change is made to credit heat transfer from containment vapor to both surfaces of this heat structure to minimize containment pressure.
  iii. IRWST basemat internal boundary, in contact with IRWST water, is set to a heat-transfer coefficient value similar to that used in the single-node ICECON model.
- Material Properties
  The five material types used to model the containment heat sinks are modified to represent the properties used in the ICECON model.

All the changes previously described are made to reflect the single-node ICECON input parameters. One additional change is made: the DLM-FM heat transfer coefficient option is used instead of the DLM option that had been used in the preceding benchmark cases. From among the DLM options (F - film roughening, M – mist generation, and FM – film roughening and mist generation), DLM-FM is selected because it produces the lowest containment pressure prediction.

The pressure responses for the various multi-node GOTHIC benchmark cases are compared to the ICECON response in Figure B-8. The gradual decrease of containment pressure resulting from successive additive changes is captured in Figure B-8.

The final benchmark case, with DLM-FM, is compared with the ICECON containment pressure prediction in Figure B-9. This plot shows that during the initial blowdown phase, which lasts up to approximately 33 seconds, the multi-node GOTHIC model with DLM-FM predicts a lower pressure inside containment than that predicted by ICECON. However, past the blowdown phase, the pressure predicted by the multi-node GOTHIC with DLM-FM is higher than what the single-node ICECON predicts. This higher pressure is due to the imperfect mixing that occurs in a multi-node configuration compared to the near-perfect mixing inherent in a single-node model such as ICECON.

During the time period of interest, after blowdown, the multi-node GOTHIC benchmark model, with DLM-FM, predicts a higher pressure inside containment than that calculated by ICECON (the pressure difference is approximately 5 psi after 200 seconds). Therefore, it is concluded that the ICECON methodology for calculating containment backpressure following large break LOCA is conservative.

**Figure B-8 Containment Pressure for the Sequence of Benchmark Cases using the Multi-node GOTHIC Model - ICECON Prediction is also shown**

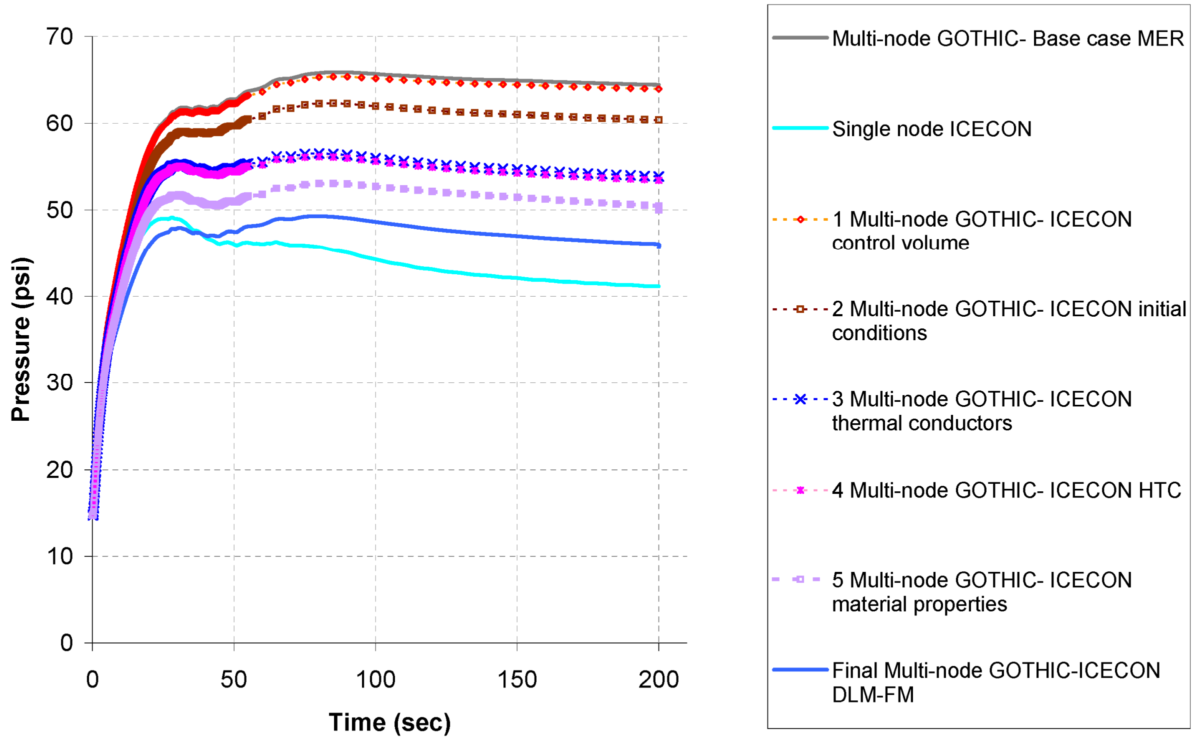
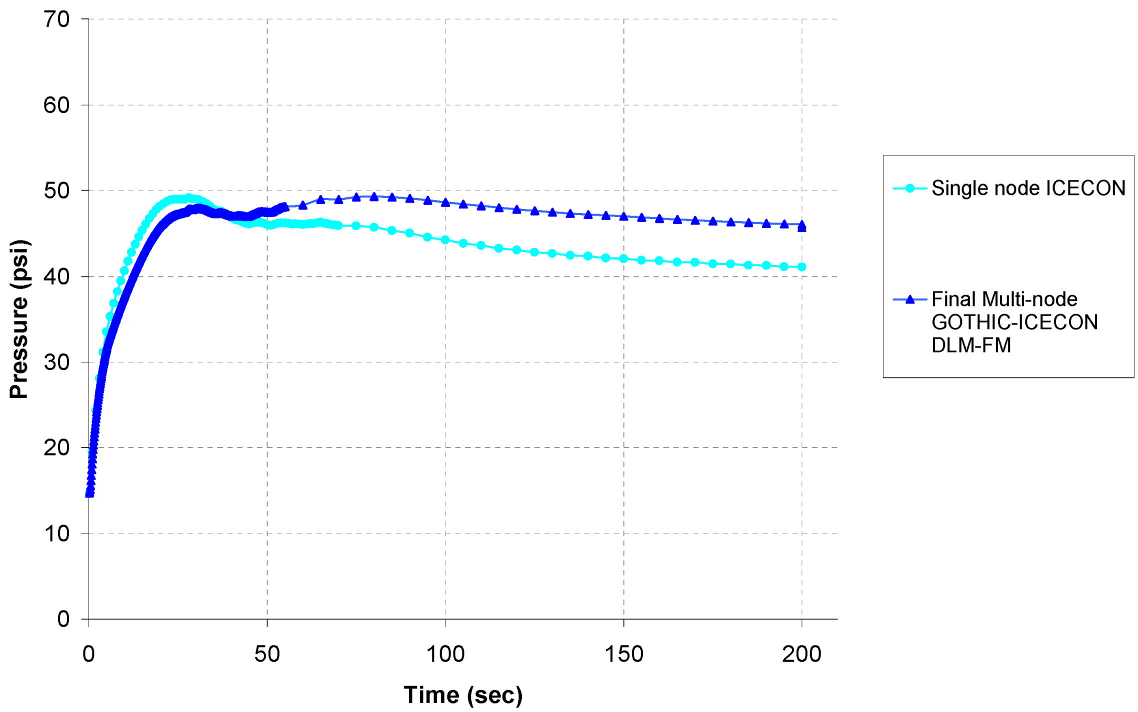


**Figure B-9 Comparison of Containment Pressure Predicted by the Multi-node GOTHIC with DLM-FM and Single Node ICECON**



## B.6.0 *References*

B-1. Frank Kreith, *Principles of Heat Transfer*, 3rd ed. (New York: Intext Educational Publishers, 1973), 14.

B-2 124-9057635-001, Revision 1, "AREVA NP Inc. U.S. EPR Final Safety Analysis Report," AREVA NP Inc., May 2009.

B-3 ANP-10299P, Revision 2, "Applicability of AREVA NP Containment Response Evaluation Methodology to the U.S. EPR™ for Large Break LOCA Analysis," AREVA NP Inc., December 2009.

B-4 NUREG-0800, NRC Standard Review Branch Technical Position 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," Revision 3, March 2007.

# APPENDIX C

# JUSTIFICATION OF THE DECAY HEAT MODELING IN S-RELAP5

# APPENDIX C

As noted in Sec. 4.3.3.2.3 of the main report, the AREVA NP Realistic Large-Break LOCA (RLBLOCA) evaluation model, S-RELAP5, calculates the decay heat based on the ANSI/ANS 5.1-1979 standard (Ref. C-1), along with the following assumptions (referred to as the Base Case in the rest of this Appendix):

(a) Infinite operating time at full power,

(b) All fissioning attributable only to U-235,

(c) Energy release of 200 MeV/fission,

(d) Inclusion of the decay heat generated from neutron capture in fission products,

(e) Inclusion of the decay heat contribution by Actinides (U-239, Np-239, etc.) generated as a result of neutron capture (based on a U-238 neutron-capture/fission ratio of 0.85), and

(f) Uncertainty in the fission-product decay heat based on randomly accessing a Gaussian probability density function with a standard deviation of 2 percent, in line with RG 1.157 (Ref. C-2).

These assumptions are supported as follows:

(a) The assumption of infinite operating time is equivalent to assuming an equilibrium fission-product inventory. This is realistic for short-lived fission products and conservative for the long-lived isotopes.

(b) In conjunction with the use of 200 MeV/fission, the assumption that U-235 is the only fissile isotope is a simplification that provides a slight conservatism over the direct inclusion of fission from U-238 and its fissionable derivatives.

(c) For U-235, the energy release per fission is 202 MeV. Assigning 200 MeV/fission is conservative since it leads to a corresponding increase (of 2/200 = 1 percent) in the fission-product decay heat.

(d) The decay heat generated from neutron capture in fission products is conservatively represented by setting the fissions per initial fissile atom equal to unity.

(e) The ANS Standard provides for the decay of U-239, Np-239 directly. The decay of these isotopes accounts for more than 90 percent of the decay energy contributed by all actinides. Setting the U-238 neutron-capture/fission ratio to 0.85, 10 percent above the actual capture to fission ratio, accounts for the actinides not directly simulated.

(f) The published standard deviation uncertainty for U-235 fission product decay is initially 3.3 percent and falls to below 2 percent by 8 seconds. Peak cladding temperatures typically occur in the post-blowdown period; i.e., at times greater than 30 seconds. By this time, the potential underprediction during the first 8 seconds caused by sampling with a 2 percent standard deviation has been offset by the overprediction following 8 seconds. PCT predictions during blowdown are dominated by the need to remove fuel-stored energy and are not substantially influenced by decay heat rates.

**References:**

C-1 ANSI/ANS-5.1-1979, “American National Standard for Decay Heat Power in Light Water Reactors,” American Nuclear Society, 1979.

C-2 Regulatory Guide 1.157, “Best Estimate Calculations of Emergency Core Cooling System Performance,” May 1989.