ISL-NSAD-TR-05-11

DRAFT REPORT

TRACE SMALL BREAK LOSS-OF-COOLANT ACCIDENT ANALYSES FOR A WESTINGHOUSE THREE-LOOP PLANT

by

Mark Bolander
C. Don. Fletcher

Information Systems Laboratories, Inc.
Idaho Falls, Idaho

prepared for

U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555

under

Contract No. NRC-04-04-065
Maintenance, Applications and Nuclear Assessment

Task Order No. 3:
Break Size Redefinition Supporting Calculations

JCN Number N6111
RES ID: RES-C05-021

July 2005

ABSTRACT

The U. S. Nuclear Regulatory Commission (NRC) is evaluating potential changes in the loss-of-coolant accident analysis requirements for the licensing of pressurized water reactor plants. To support these NRC evaluations, analyses are performed for a variety of small-break loss-of-coolant accident (SBLOCA) event sequences in a typical Westinghouse three-loop plant. SBLOCA simulations are performed with the TRACE computer code using a set of modeling assumptions consistent with typical current plant licensing-basis calculations. The TRACE calculations are then repeated including the assumption of an additional 50 s delay in the time required for diesel-electric generator availability.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

TRACE calculations were performed to simulate 11 small break loss-of-coolant accidents (SBLOCAs) using conditions and assumptions consistent with a typical Westinghouse three-loop pressurized water reactor licensing basis. These calculations were repeated with the assumption that the startup of the diesel-electric generator was delayed by an additional 50 s. The break that produced the highest peak cladding temperature (PCT) was the 2.5 inch cold leg break. The PCTs predicted by TRACE were below the temperature where significant fuel rod oxidation due to metal-water reaction is encountered.. The additional 50 s delay in startup of the diesel-electric generator resulted in no significant changes in the PCTs.

The emergency core cooling system (ECCS) was able to mitigate the fuel rod temperature excursion, quench the rods and refill the core in all of the transients simulated. Coolant from the high pressure injection (HPI) system was capable of refilling the core for the 2 inch cold leg break. HPI and accumulator flow refilled and quenched the core for break sizes between 2.25 and 4 inches. Coolant from all of the ECCS (HPI, accumulators and LPI) was required to terminate the core heatup for break sizes 6 inches and larger.

Difficulties were encountered in obtaining SBLOCA PCTs with TRACE that were as high as those provided in the FSARs for Westinghouse three-loop plants. Two model revisions which are considered beyond normal variation were made for the purpose of increasing TRACE-calculated PCTs into the PCT range normally encountered in the licensing-basis calculations. These revisions included significantly biasing downward the elevation of two of the loop seals and increasing the reactor coolant pump stopped rotor flow resistance. That these revisions were required suggests they are compensating for some shortcoming in the plant input model, in the TRACE code, or some significant difference between the TRACE code and the codes used for the licensing-basis calculations.

The PCT results from the TRACE calculations are summarized as follows.

Break Location and Size	**Peak Cladding Temperature (°F)**	
	Current Licensing Basis Assumption	**Additional 50 s diesel-electric generator delay**
Cold Leg 2.0 in Diameter	1,191	1,195
Cold Leg 2.25 in Diameter	1,514	1,511
Cold Leg 2.5 in Diameter	1,704	1,730
Cold Leg 2.75 in Diameter	1,619	1,605
Cold Leg 3.0 in Diameter	1,568	1,517
Cold Leg 4.0 in Diameter	1,252	1,266
Cold Leg 6.0 in Diameter	1,172	1,209
Pressurizer Surge Line (10.5 in diameter) Double-Ended	No Heatup	No Heatup
Safety Injection Line (Nominal Area, 8.5 in diameter) Double-Ended	1,240	1,232

Safety Injection Line (80% Nominal Area,) Double-Ended	1,311	1,333
Safety Injection Line (120% Nominal Area) Double-Ended	1,363	1,471

NOMENCLATURE

AFW	auxiliary feedwater
ANS	American Nuclear Society
CB	control block
CL	cold leg
DC	downcomer
DEGB	double-ended guillotine break
ECC	emergency core coolant
ECCS	emergency core cooling system
FA	flow area
F_Q	ratio of peak-to-average fuel rod power
FSAR	final safety analysis report
FW	feedwater
HFP	hot full power
HL	hot leg
HPI	high pressure injection
LANL	Los Alamos National Laboratory
LBLOCA	large break loss-of-coolant accident
LOCA	loss-of-coolant accident
LP	lower plenum
LPI	low pressure injection
MFW	main feedwater
P	pressure
PCT	peak cladding temperature
P/I	proportional-integral control function
PWR	pressurized water reactor
RC	reactor coolant
RCP	reactor coolant pump
RCS	reactor coolant system
RPS	reactor protection system
RV	reactor vessel
SBLOCA	small break loss-of-coolant accident
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SV	signal variable
T	temperature
T_{avg}	average reactor coolant system temperature
T_{cold}	reactor coolant system cold leg temperature
T_{hot}	reactor coolant system hot leg temperature
TRACE	TRAC/RELAP Advanced Computational Engine
TRAC-P	TRAC-PF1/MOD1 computer code
USNRC	U. S. Nuclear Regulatory Commission

1.0 INTRODUCTION

Currently, U. S. nuclear power plant licensing methods include many significant conservatisms related to plant conditions, analysis assumptions, equipment availability and system response times. Using these methods, the plant response during design-basis accidents is required to meet specific limits regarding fuel rod cladding temperature, localized oxidation and core-wide oxidation. The U. S. Nuclear Regulatory Commission (USNRC) is exploring various options by which nuclear plant licensing requirements might be revised for pressurized water reactors (PWRs).

Options under consideration include a relaxation of the maximum break size that must be considered in analysis of design basis loss-of-coolant accidents (LOCAs). The current requirement is that plants must be evaluated and meet the acceptance criteria for break sizes up to as large as a 100% double-ended offset shear of one cold leg. By reducing the maximum break size that must be considered, the minimum safety margins to the licensing acceptance limits would be increased. In response, plants could uprate their reactor power and/or relax analysis assumptions related to systems availability and response times. Should such an option be implemented: (1) the new maximum break size limit for which the acceptance criteria must be met will be a key factor affecting the benefits to be gained and (2) for break sizes larger than the new maximum break size limit and smaller than a 100% double-ended offset shear of a cold leg the risks associated with not meeting the acceptance criteria must be accounted for.

The work reported here supports the USNRC evaluation of break size redefinition options. A variety of small break loss-of-coolant accidents (SBLOCAs) in a Westinghouse three-loop PWR are analyzed here using the TRACE computer code and plant conditions and assumptions consistent with typical current final safety analysis report (FSAR) licensing-basis calculations. The TRACE calculations are then repeated including an assumption of an additional 50 s delay in starting the diesel-electric generator. **Table 1-1** lists the typical plant conditions and assumptions used in the analyses.

The purpose of these analyses is to define plant response for SBLOCAs of various locations and sizes and determine the effects of the diesel-electric generator delay time on the SBLOCA plant response. The primary parameter of interest is the peak cladding temperature (PCT) experienced during the accident sequences. The analyses are performed using the TRAC/RELAP Advanced Computational Engine (TRACE) computer code (Reference 1, Version 4160 plus modifications improving time step control in problems with check valves and tighter outer iteration convergence criteria) and a model of a typical three loop Westinghouse PWR.

Section 2 describes the three-loop Westinghouse TRACE plant model. The analyses of the SBLOCAs using conditions and assumptions consistent with a three loop Westinghouse licensing basis and an additional 50 s delay in the startup of the diesel-electric generator, are described in Section 3. Conclusions are given in Section 4 and references are provided in Section 5.

Table 1-1. Summary of Westinghouse Three-Loop PWR Model Assumptions for the TRACE SBLOCA Calculations

Parameter	TRACE SBLOCA Modeling Assumption
Initial core power	102% of current plant rated core power, typical of FSAR calculations.
Fraction of core power deposited in the fuel	97.4%, typical of FSAR calculations.
Initial RCS pressure	Typical of FSAR calculations.
Initial RCS average temperature	Typical of FSAR calculations.
Initial RCS total loop flow	Typical of FSAR calculations.
Initial pressurizer level	Typical of FSAR calculations.
Initial SG secondary pressure	Typical of FSAR calculations.
Initial SG secondary mass	Typical of FSAR calculations.
Initial SG secondary recirculation ratio	Typical of FSAR calculations.
Radial core power distribution	Three radial core rings (12 axial and 6 azimuth sections in each) with relative powers of 1.2, 1.177 and 0.4.
Axial core power distribution	1.288 relative power axial peak at X/L 0.833. FSAR axial power profile for SBLOCAs is typically also top peaked.
Hot rod modeling	Hot rods modeled in all three core rings. Hot rod relative peaking factors provide F_Q of 2.50, which is typical for FSAR SBLOCA calculations.
Core decay heat	120% of 1973 ANS Standard, typical of FSAR calculations.
SG tube plugging assumption	6%, typical of FSAR calculations.
Off-site power and diesel-electric generator availability	Off-site power assumed lost at the time of reactor trip, causing turbine and RC pump trips. One diesel-electric generator assumed available. Typical of FSAR SBLOCA calculations.
ECC failure assumption	Loss of one HPI pump and one LPI pump assumed, typical of FSAR SBLOCA calculations.
Number of HPI pumps assumed available	One, typical of FSAR SBLOCA calculations.
Number of LPI pumps assumed available	One, typical of FSAR SBLOCA calculations.
Number of accumulators assumed available	Three (one per coolant loop), typical of FSAR SBLOCA calculations.

Parameter	TRACE SBLOCA Modeling Assumption
Delay for ECC injection	HPI flow available at 28.5 s and LPI flow available at 44.7 s after safety injection actuation signal, typical of FSAR calculations. Fifty seconds added to these availability times for cases assuming an additional diesel-electric generator delay.
HPI and LPI temperatures	Typical of FSAR calculations.
Accumulator initial levels, pressures and temperatures	Typical of FSAR calculations.
Containment pressure response	Atmospheric pressure assumed at the break over the full length of the TRACE SBLOCA calculations. Typical of FSAR SBLOCA calculations.
RC pump behavior after trip	Pump speed specified as function of time, with the rotors assumed stopped 200 s after the pump trip. FSAR calculation assumptions vary.

2.0 MODEL DESCRIPTION AND REVISIONS

The TRACE model for a Westinghouse three-loop PWR used for the SBLOCA calculations in this report is described here.

Despite using assumptions and boundary conditions consistent with current plant licensing bases, in general difficulties were encountered in obtaining SBLOCA PCTs with TRACE that were as high as those provided in the FSARs for Westinghouse three-loop plants. Many of the model revisions described below were implemented in order to increase the TRACE-calculated PCTs. Of these, it is noted that most represent reasonable adjustments of model parameters based on plant-to-plant variations. However, two of the revisions made for the purpose of increasing PCTs may place the model beyond what are considered normal adjustments. First, a large (four-foot) downward biasing of the elevations of two of the three loop seals was implemented in order for the TRACE simulation of smaller break sizes to represent the clearing of only one loop seal. While loop seal biasing is often encountered in licensing basis calculations, it is more typically accomplished using only a one-foot downward bias. Second, a large increase in the reactor coolant pump stopped-rotor flow resistance was implemented to increase the pressure drop in the coolant loops following pump trip and coast down. This model revision may have been necessary in order to compensate for some other shortcoming of the TRACE code or the plant facility model.

Section 2.1 provides an overview description of the model and the starting point plant input model from which the model was developed. Section 2.2 describes the changes implemented in the full-power steady-state model and Section 2.3 describes the changes implemented in order to perform the transient SBLOCA simulations.

2.1 Overview Model Description

The plant model used for the calculations in this report was derived from a steady-state TRAC-PF1/MOD1 model (Reference 2) of a typical Westinghouse three-loop PWR originally developed at Los Alamos National Laboratory. That model was subsequently revised and updated by Information Systems Laboratories, Inc. for performing TRACE large break loss-of-coolant accident (LBLOCA) calculations (Reference 3). Additional revisions to the steady-state plant model for performing the SBLOCA calculations in this report are described in Section 2.2.

The TRACE steady-state plant model represents the reactor vessel, the three coolant loops [each including a steam generator (SG) and a reactor coolant pump (RCP) and one also including the pressurizer], emergency core cooling systems (ECCS) including accumulator, high pressure injection (HPI) and low pressure injection (LPI) systems, and the main feedwater (MFW) and steam line systems. The reactor vessel portion of the model utilizes the TRACE three-dimensional VESSEL component and contains 20 axial levels, four radial rings (three in the core region and one representing the downcomer), and six azimuthal sectors.

Heat structure models are employed to represent the core fuel rods, SG tubes, vessel and piping walls and major internal structures of the reactor coolant system (RCS) components. Control

models are included to facilitate the steadying of the model at a set of desired initial conditions and to represent the actions of the various plant control systems during the simulation of transient accident scenarios.

Nodalization diagrams for the TRACE Westinghouse three-loop plant model used for the calculations in this report are provided in **Figures 2-1** through **2-8**.

2.2 Steady-State Plant Model Revisions for SBLOCA Simulation

This section describes the revisions made to the TRACE steady-state Westinghouse three-loop plant model for performing the SBLOCA calculations. The SBLOCA transient calculations are performed as restart calculations that begin from the conditions present at the end of the TRACE calculation of full-power steady plant operation. The model modifications were generally minor and were made to facilitate performing the SBLOCA transient calculations and to update the model to be consistent with typical FSAR SBLOCA licensing-basis calculations.

2.2.1 Primary Coolant Loop Modifications

The small breaks analyzed in this report are assumed to occur at three different locations in the RCS piping: (1) the Loop-2 cold leg, between the ECCS injection nozzle and the reactor vessel, (2) the pressurizer surge line, and (3) the Loop-2 ECCS injection line. The cold leg breaks are assumed to be located on the bottom of the cold leg pipe. The pressurizer surge line and ECCS line breaks are assumed to be double-ended guillotine ruptures of those lines.

TEE 28 in the Loop-2 cold leg was originally modeled using four cells in the main flow path. To facilitate adding cold leg breaks to the model at the start of the SBLOCA transient calculations, Cells 3 and 4 of the TEE 28 main path are split off and made into new two-cell PIPE 29 (the revised cold leg noding is shown in **Figure 2-3**). The cold leg cell volumes, flow areas, lengths, etc. were preserved in this process. The cold leg breaks are added at PIPE 29 Cell 1, as described further in Section 2.3.

The four-cell side tube of TEE 30 in the original model represents the pressurizer surge line. To facilitate adding a pressurizer surge line break to the model, Cells 3 and 4 of the TEE 30 side tube were split off into new two-cell PIPE 31 (the revised surge line noding is shown in **Figure 2-4**). The pressurizer surge line cell volumes, flow areas, lengths, etc. were preserved in this process. The surge line break is added between the free end of the TEE 30 side tube and PIPE 31, as described further in Section 2.3.

Preliminary calculations showed that following reactor coolant pump trip the pump rotors did not completely stop turning as expected. In fact, low velocity steam was able to keep the rotors rotating slowly. User-supplied homologous curves (defining the relationships between pump head, flow and speed) are input for the reactor coolant pump components of the model. Following pump trip, the resistance for flow through the pumps is calculated from these homologous curves and the current pump speed, flow rate and head. With the rotors continuing

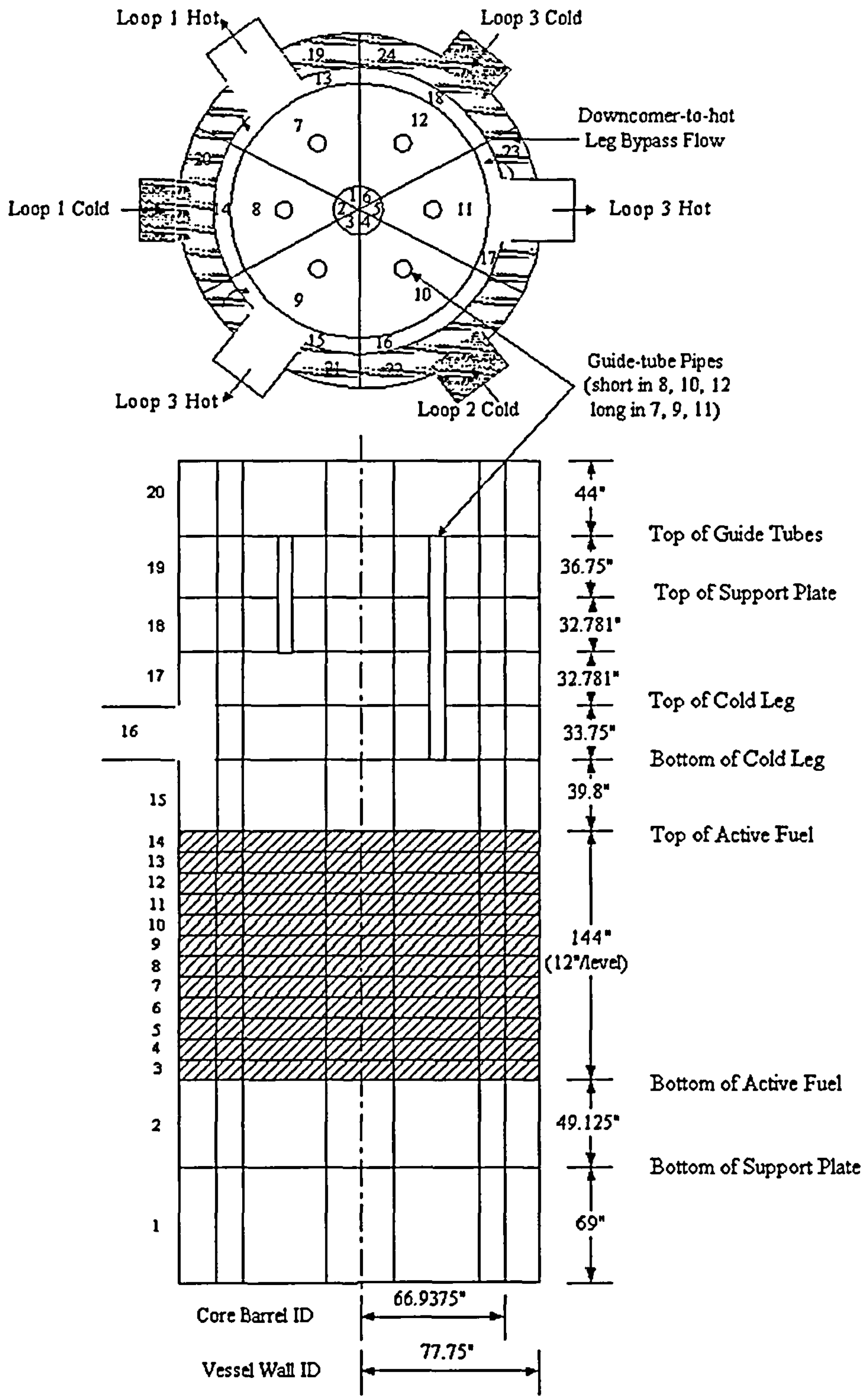


Figure 2-1. Nodalization of the Westinghouse Three-Loop Plant Reactor Vessel (TRACE Component 1)

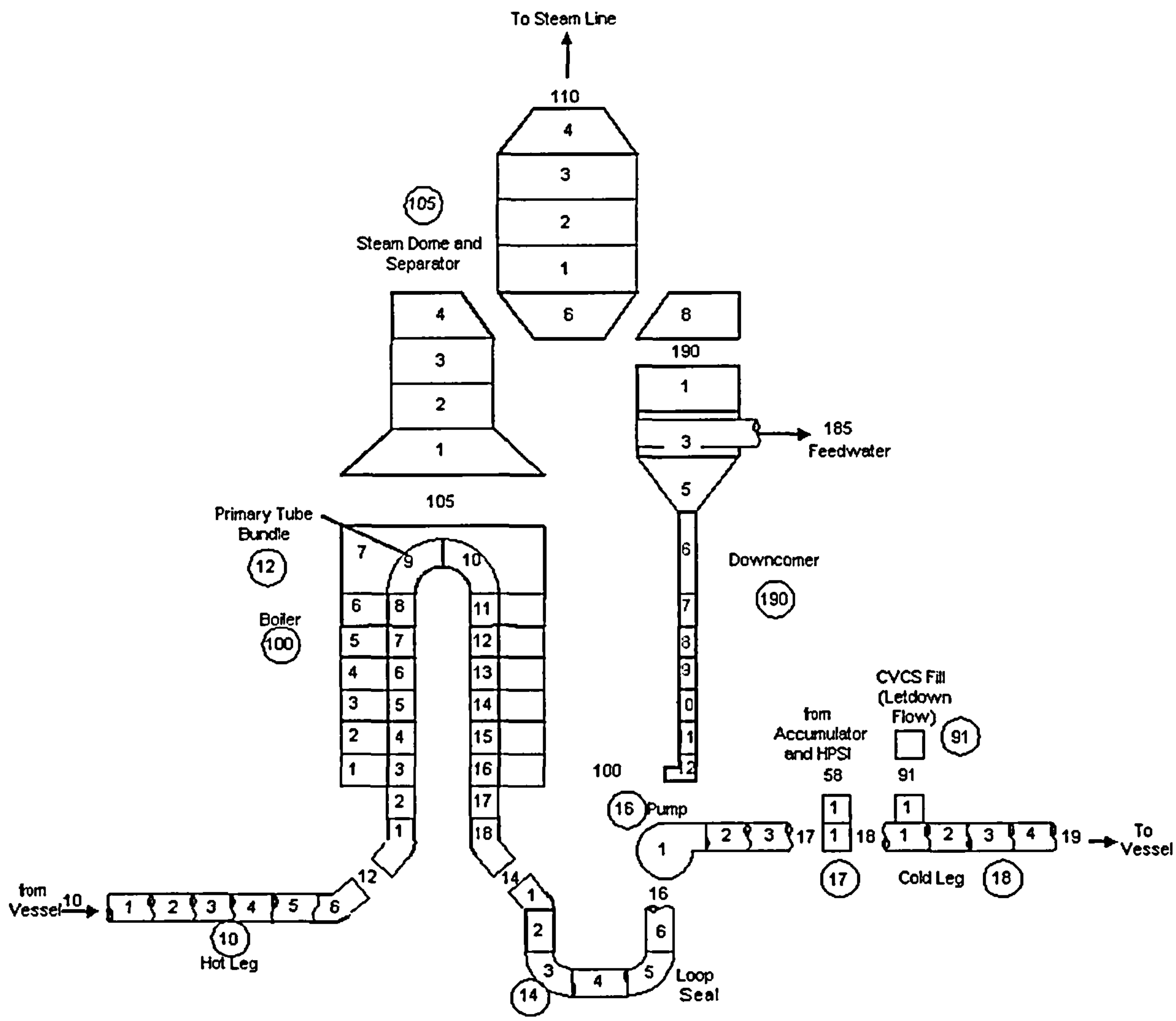


Figure 2-2. TRACE Nodalization of the Westinghouse Three-Loop Plant Coolant Loop 1

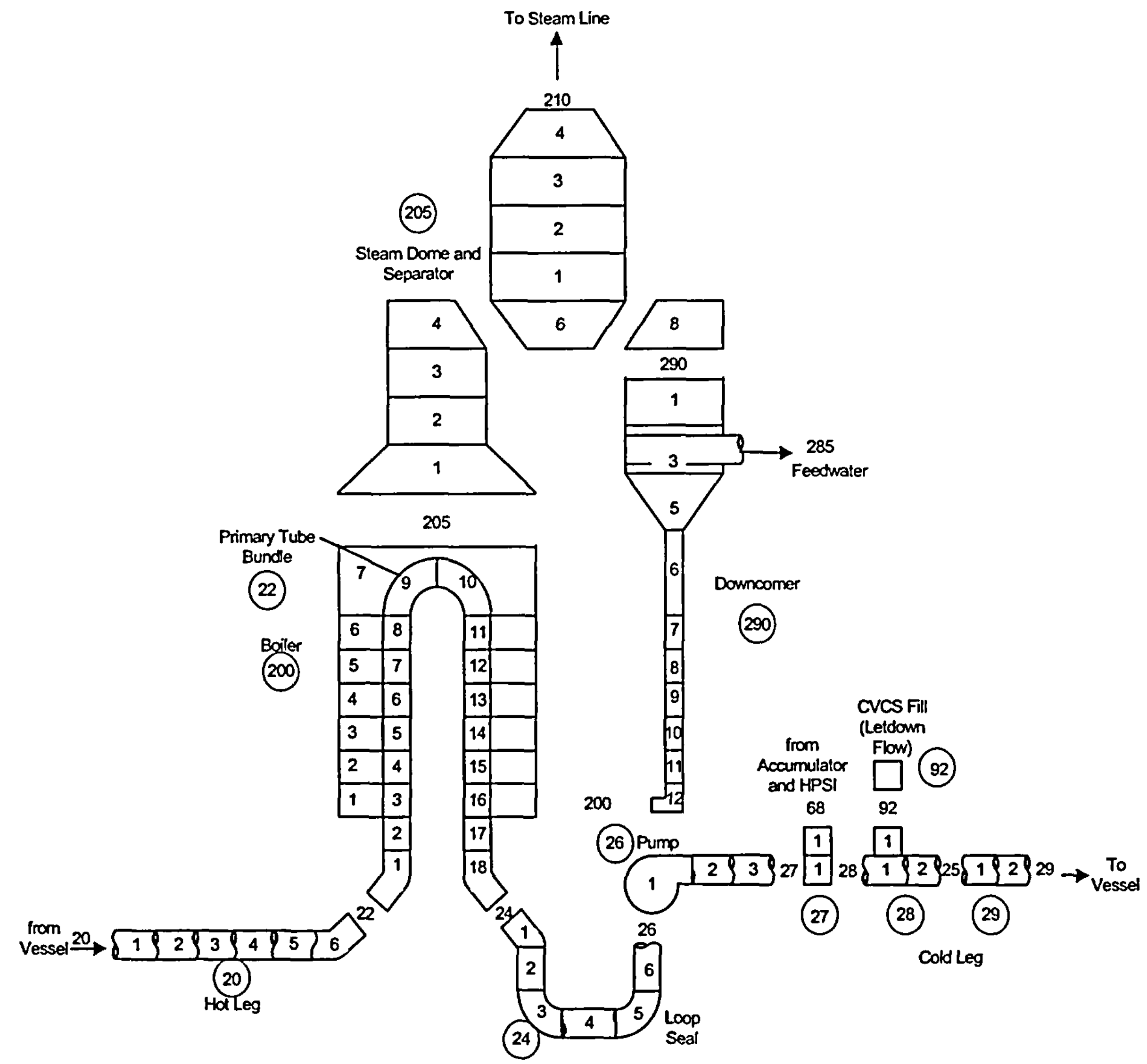


Figure 2-3. TRACE Nodalization of the Westinghouse Three-Loop Plant Coolant Loop 2

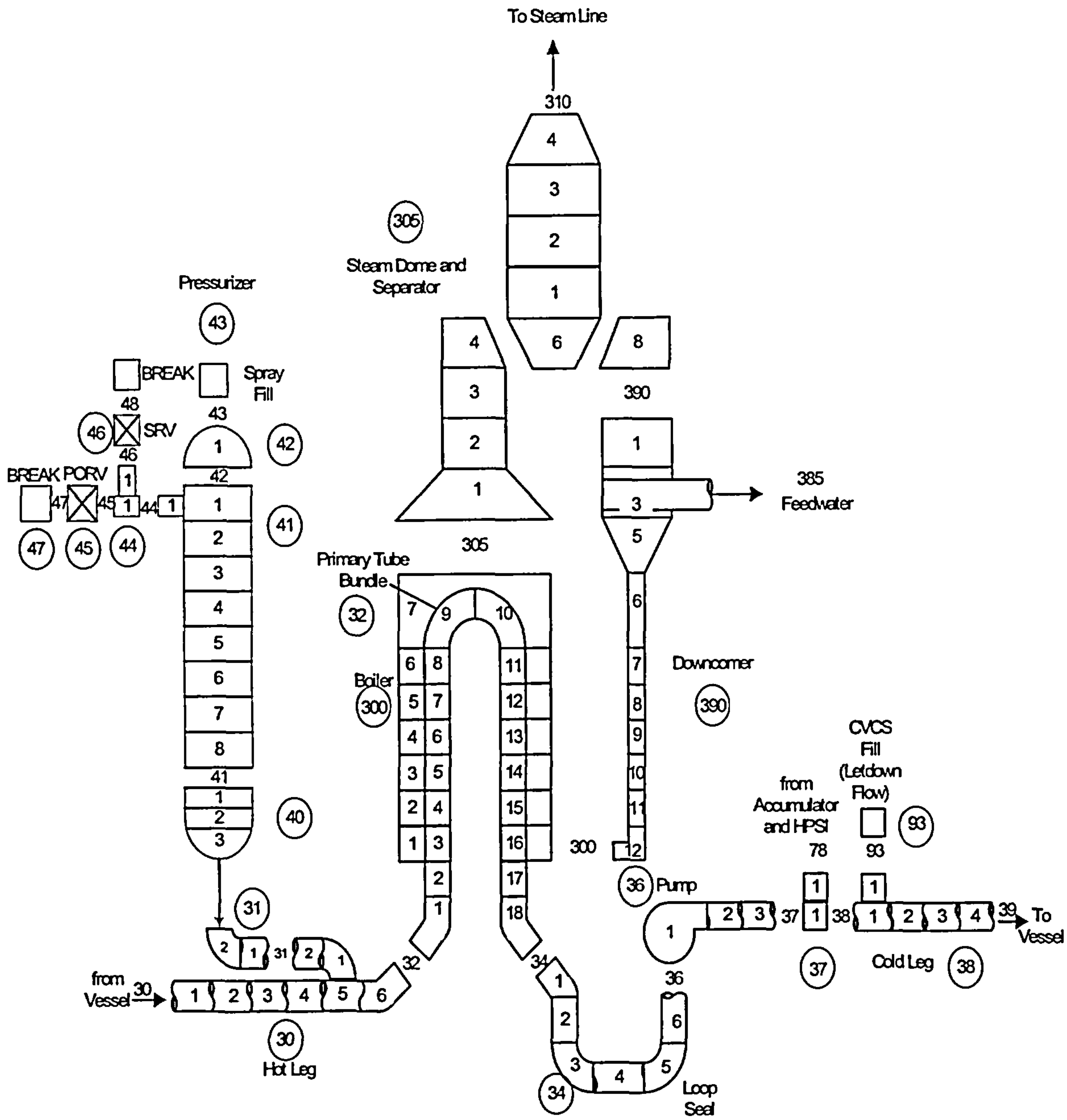
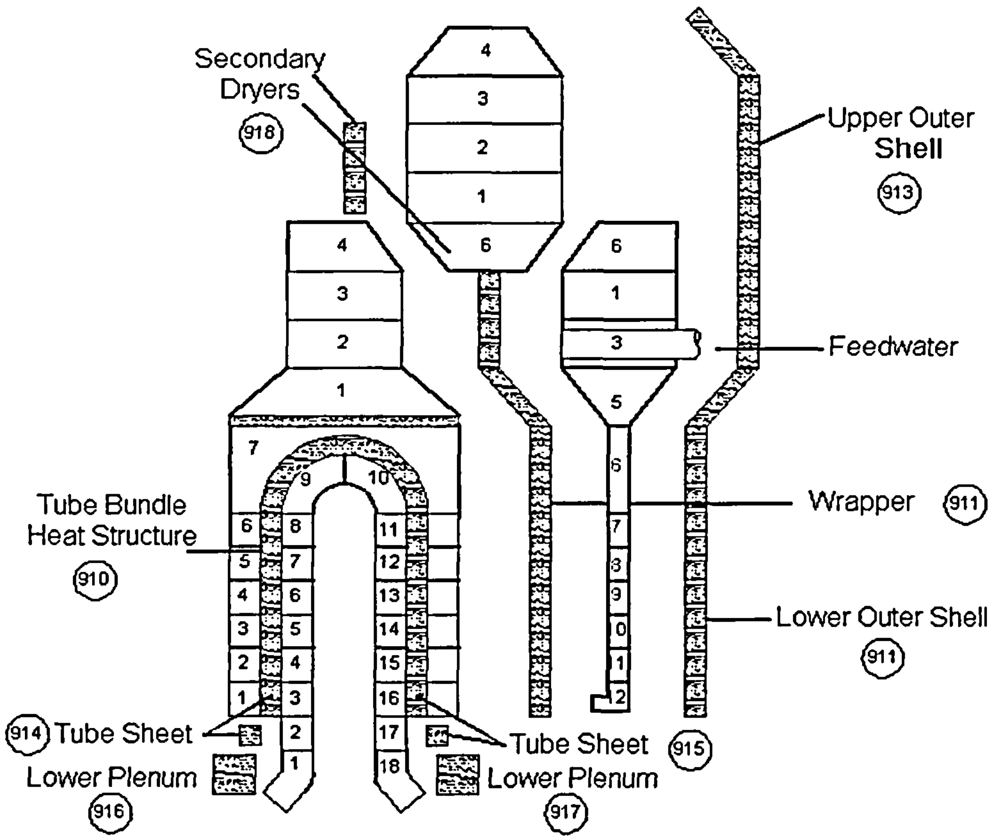


Figure 2-4. TRACE Nodalization of the Westinghouse Three-Loop Plant Coolant Loop 3



(Note: Heat structure component numbers are for loop-1 steam generator)

Figure 2-5. TRACE Nodalization of the Westinghouse Three-Loop Plant Steam Generator Secondary Region

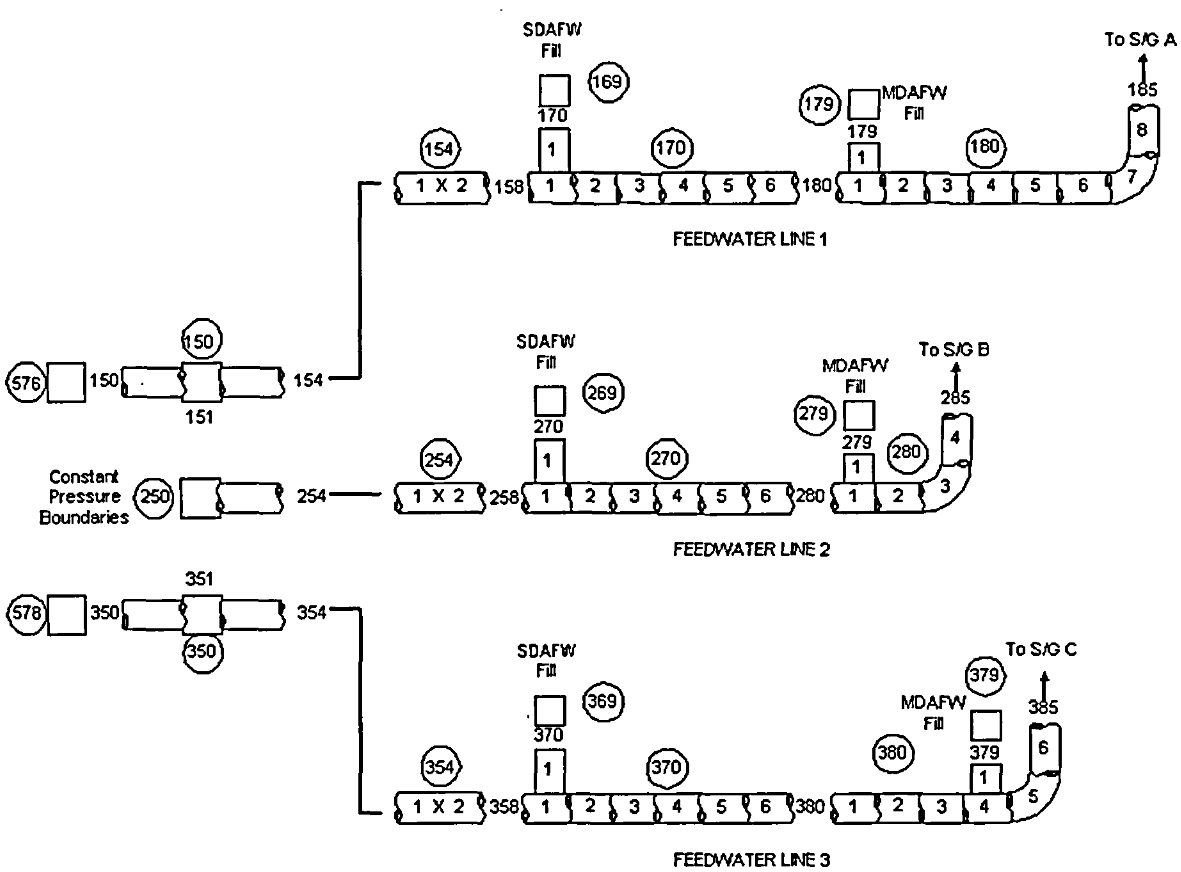


Figure 2-6. TRACE Nodalization of the Westinghouse Three-Loop Plant Main Feedwater System

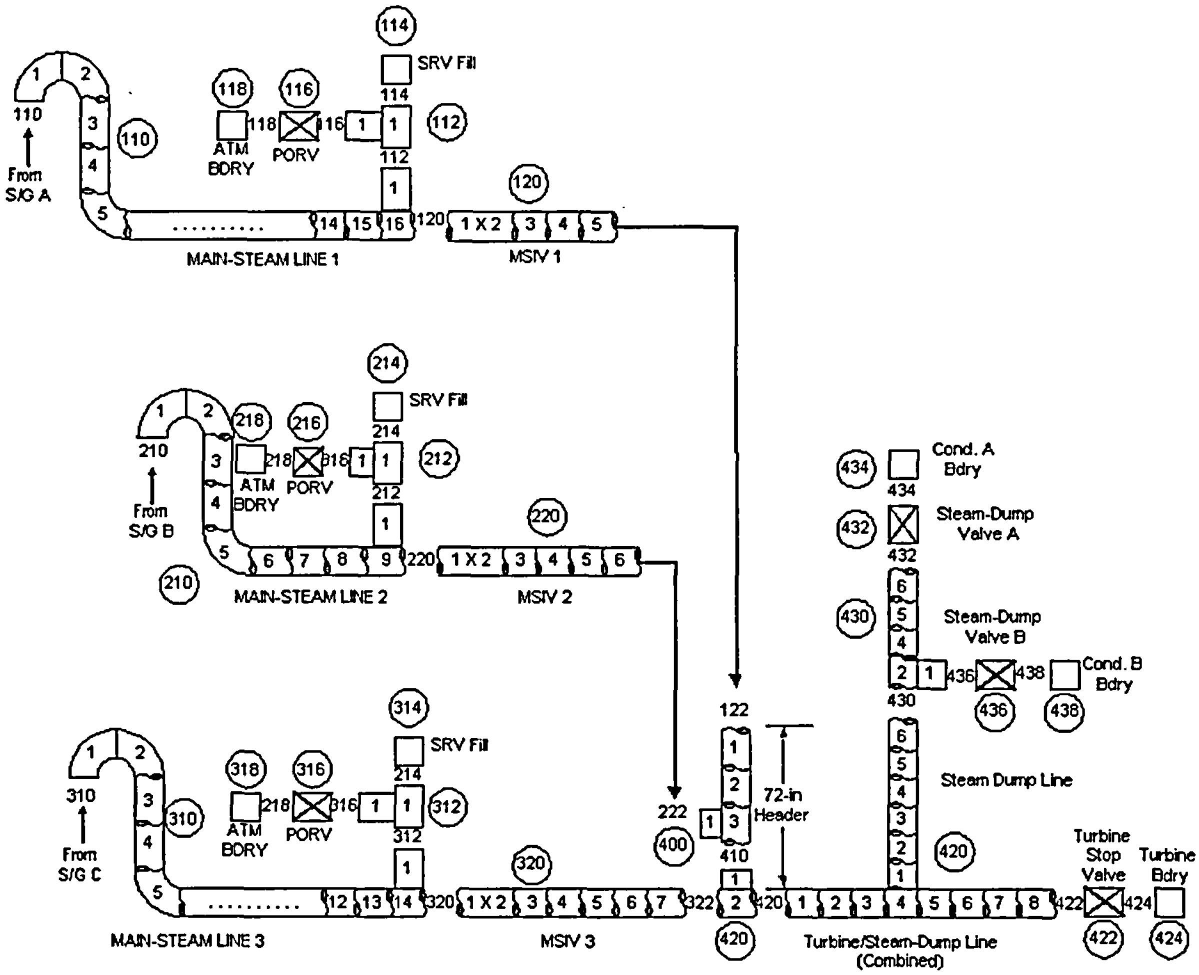


Figure 2-7. TRACE Nodalization of the Westinghouse Three-Loop Plant Main Steam System

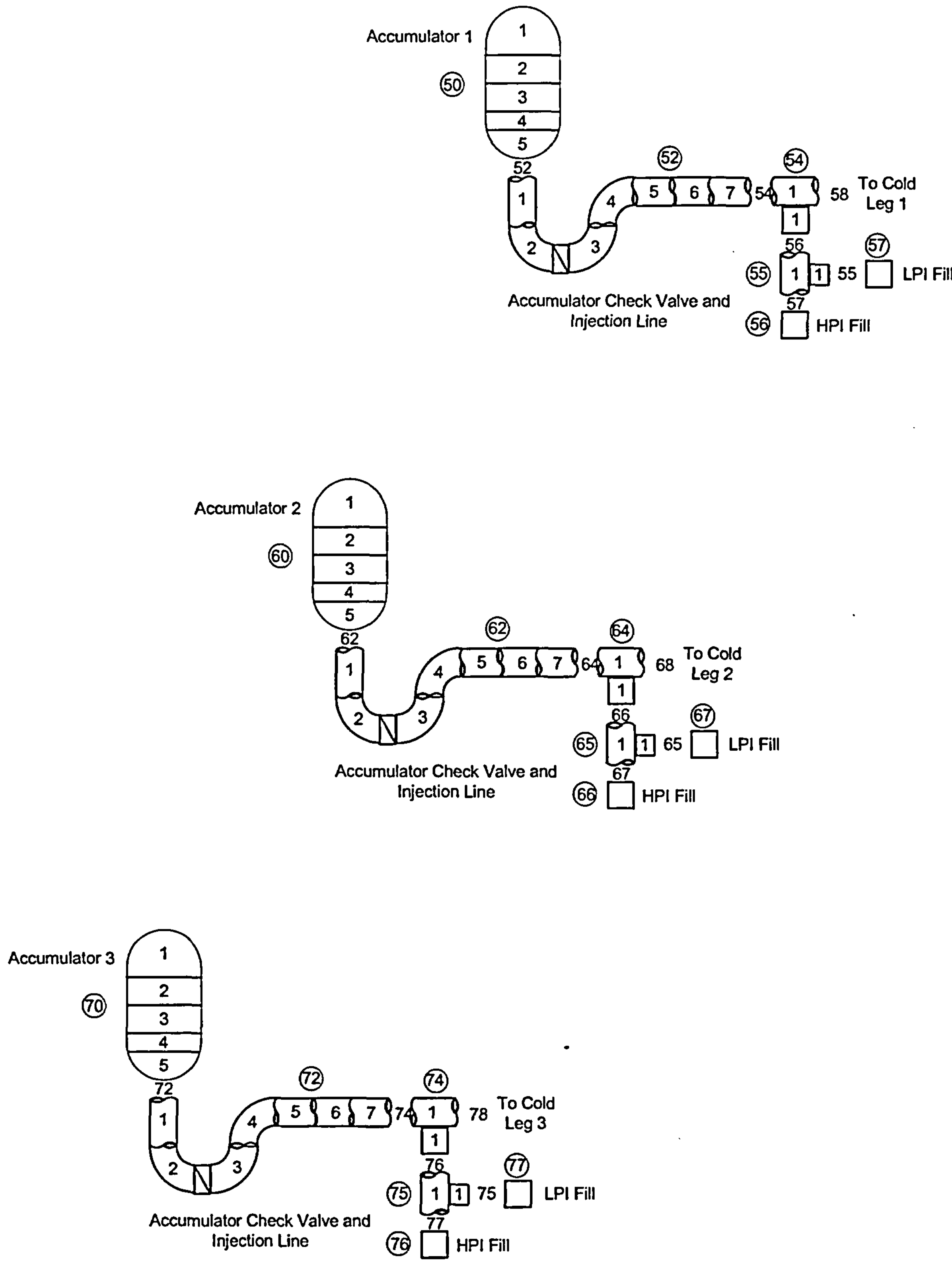


Figure 2-8. TRACE TRACE Nodalization of the Westinghouse Three-Loop Plant ECC Systems

to turn (even slowly), a much smaller pressure drop is calculated for flow through the pump than for a case where the impellers are completely stopped. Increasing this pressure drop tends to depress the core level, increasing the depth and length of the core uncovery and the extent of the fuel rod heatup.

In order to allow the pump rotors to coast down to a complete stop, the pump-type option (IPMPTY) was changed from 2 to 1. With IMPTY = 2, the pump coastdown is calculated by the code based on inertial angular momentum considerations. With IPMPTY = 1, the pump coastdown is specified with a user-supplied speed-versus-time table. Prior to trip, the pump is assumed to operate at a constant speed. After trip, the pump rotor is assumed to coast down from initial to zero speed over a 200-s period. **Figure 2-9** shows the pump speed as a function of time after trip used in the SBLOCA calculations.

The total coolant loop resistance of the model during full power steady-state conditions with the pumps operating agrees well with typical plant data. However, preliminary calculations with the reactor coolant pump model modified so as to completely stop the rotors following trip showed that the coolant loop flow resistance was not as large as expected. To provide more flow resistance through the pumps when the rotors are stopped, the value for the pump homologous curve HSP2 point at zero speed (i.e., zero speed-to-flow ratio) was revised from -1.55 to -25.0 (head-to-flow ratio). This change therefore increases the loop flow resistance for situations after the pumps have tripped and the rotors have coasted to a stop, but does not otherwise affect the loop flow characteristics.

Loop seal behavior plays an important part in the prediction of the fuel rod PCTs. For smaller break sizes, FSAR calculations typically show only one loop seal clearing. However, in preliminary TRACE calculations (with the above changes to the pump speed control and stopped-rotor flow resistance) all three loop seals were clearing, even for the smaller break sizes. To attempt to minimize the number of loop seals calculated to clear, the loop seals in Loop 2 (the broken loop) and Loop 3 (the pressurizer loop) were artificially lowered by four feet. This was accomplished with two model revisions. First, referring to **Figures 2-3** and **2-4**, the GRAV terms at Cell Faces 4 and 6 in six-cell PIPEs 24 and 34 were changed from 0.56067 to 0.98929, effectively lowering the bottom of the loop seals by two feet. Second, two cells were added to the vertical sections of PIPEs 24 and 34 to bring the total elevation change to four feet. One cell was added after Cell 3 and the other cell was added after Cell 6 (these additional cells are not shown in **Figures 2-3** and **2-4**). Lowering the elevations of these two loop seals helped keep them from clearing (thereby increasing the PCTs) in the SBLOCA calculations for the smaller break sizes.

2.2.2 Vessel Component Modifications

The core region in VESSEL 1 is defined in Levels 3 through 14 and Rings 1 through 3 (see **Figure 2-1**). The pressure drop across the core is calculated between Levels 2 and 15. The lower core support plate, grid spacers, and upper tie plate reside between these levels. The locations of high flow resistance in this region of the vessel are at the lower core support plate and the upper tie plate. When the model was originally set up, the flow loss specified between

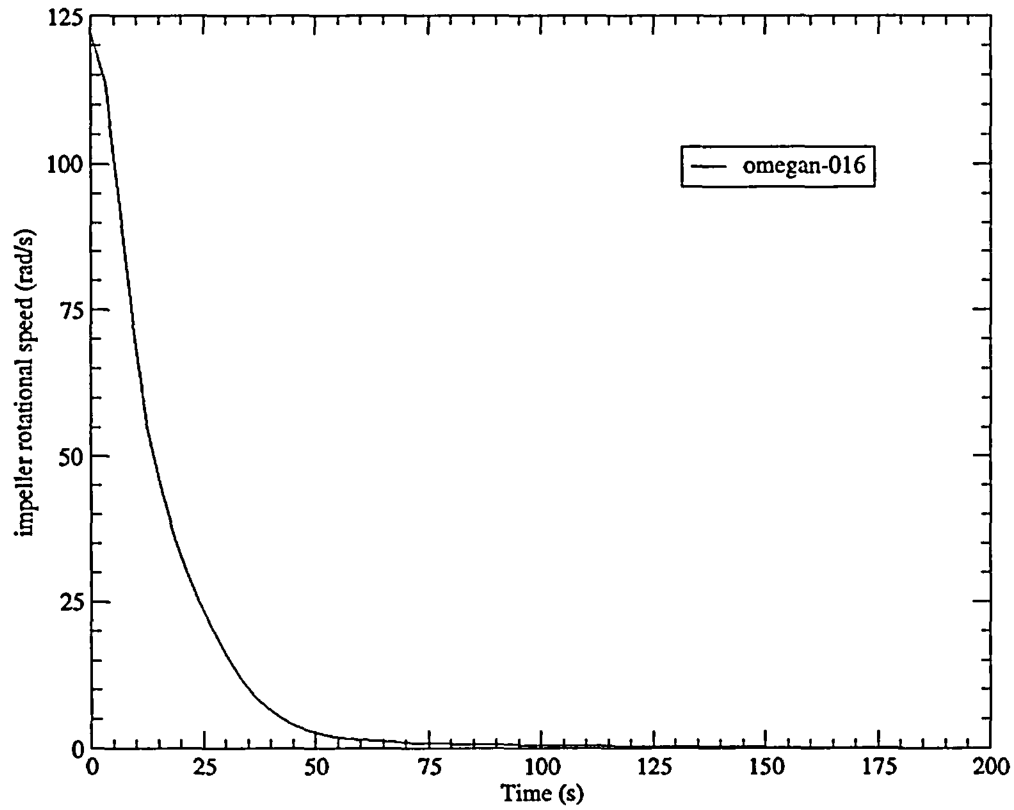


Figure 2-9. Reactor Coolant Pump Coastdown After Pump Trip

Levels 2 and 3 was relatively high in order to simulate the resistance across the lower core support plate. The remaining user-specified flow loss coefficients were selected in order to match the desired total core pressure drop and were evenly distributed from Levels 3 to 14 with the high flow resistance across the upper tie plate not modeled explicitly. For SBLOCAs, the core level depression is significantly affected by the flow resistance located at the upper core tie plate where the steam velocities are the highest. The flow loss coefficients in the core region of the model were rearranged so as to relocate a greater portion of the total flow loss to the upper tie plate. In this process, the total flow resistance and pressure drop were not changed. **Table 2-1** compares the distribution of the core region flow resistances in the SBLOCA model with that from the original LBLOCA model.

Table 2-1. Core Region Flow Loss Adjustment in VESSEL 1

Vessel Level	LBLOCA Model Flow Loss Coefficient	SBLOCA Model Flow Loss Coefficient
14 Upper Tie Plate	0.21797	2.20137
13	0.21797	0.10679
12	0.21797	0.10679

11	0.21797	0.10679
10	0.21797	0.10679
9	0.21797	0.10679
8	0.21797	0.10679
7	0.21797	0.10679
6	0.21797	0.10679
5	0.21797	0.10679
4	0.21797	0.10679
3	0.21797	0.10679
2 Lower Core Support Plate	2.49	1.62647

2.2.3 Fuel Rod HTSTR Modifications

In the original model, hot rods were included along with the average modeled rods for Ring 1 of the vessel. The modeling for Rings 2 and 3 was modified to also include hot rods in those rings. The HTSTR component numbers for the rods are 806, 807, 808, 809, 810 and 811 in Ring 2, and 812, 813, 814, 815, 816 and 817 in Ring 3. This modification required three changes to the HTSTR input for these components: input parameter NHOT (number of hot rods) was changed from 0 to 1, initial node temperatures for the hot rod were added to array RFTN, and fuel burnup data for the hot rods was added to array BURN.

2.2.4 Power Component Modifications

Peaking factors were added for the new hot rod heat structures in Rings 2 and 3; the F_Q for these rods was maintained at 2.5.

An error in the axial power profile input table (ZPWTB) was corrected. Because of a confusion regarding the required input, the axial power shape had been inadvertently shifted downward 6 inches by entering a zero at the top of the axial power shape input instead of at the bottom.

2.2.5 Steam Generator Secondary and Balance-of-Plant Modifications

The SG power operated relief valves and the steam dump valves were modified to remain closed throughout the TRACE transient calculations. For licensing calculations, no credit is taken for operation of these valves.

The opening pressure setpoint for the SG safety relief valves was found to be too low. The pressure setpoint was adjusted to coincide with typical FSAR values. The revised main steam safety valve flow characteristics are shown in **Table 2-2**.

Table 2-2. Main Steam Safety Valve Flow as a Function of Pressure

Pressure (psia)	Mass Flow Rate (lbm/s) per SG
1,132.3	0.0
1,136.7	185.3
1,158.0	185.3
1,162.3	374.9
1,173.5	374.9
1,177.9	653.2
1,188.9	653.2
1,193.2	935.1
1,305.3	935.1

2.2.6 Emergency Core Cooling System Modifications

In some Westinghouse three-loop plants the accumulator and HPI and LPI injection lines connect to the cold legs through separate nozzles. In other Westinghouse three-loop plants the HPI and LPI injection lines connect to the accumulator discharge line, which terminates in a single ECCS nozzle on each cold leg. The second configuration, with the ECCS line connected on top of each cold leg, was assumed for the model used here (see **Figure 2-8**). In the original model, the ECCS line from the accumulator check valve to the cold leg was modeled horizontally and was assumed to be filled with subcooled water at or below the saturation temperature of the initial accumulator pressure (615 psia). During the blowdown phase of preliminary TRACE SBLOCA calculations, the ECCS lines were seen to drain and numerical instabilities were caused when accumulator injection began. To remedy the numerical instabilities a negative GRAV term of -0.84147 was placed at the last cell face of the side tubes in TEE 17, TEE 27, and TEE 37. This modification angled the ECCS lines slightly to the connections on the cold legs, preventing draining of the ECCS lines prior to accumulator injection and the numerical instabilities.

The steady-state input model for the LBLOCA analysis assumed HPI was available from two pumps. For SBLOCA licensing analyses, only one HPI pump may be credited. FILLs 56, 66, 76 (HPI flow as a function of cold leg pressure) were revised as shown in **Table 2-3** to represent the flow delivered from one HPI pump.

Table 2-3. HPI Flow Delivered as a Function of Cold Leg Pressure

Pressure (psia)	Flow (lbm/s) per Loop
14.7	22.8
65.0	22.4005
95.0	22.1496
120.0	21.8998
135.0	21.7997

138.0	21.7503
200.0	21.2495
400.0	18.7497
600.0	16.2994
800.0	13.7996
1,000.0	10.8500
1,200.0	7.2000
1,304.0	4.6000
1,380.0	0.0

2.2.7 Control System Modifications

The following control system modifications were implemented in order to correct errors, update setpoints to the typical FSAR values or add analysis conveniences:

- The pressurizer spray valves were not fully closing during preliminary transient simulations. Logic was added to terminate spray flow when the void in the Loop 3 cold leg was greater than 20%.
- The delay time for actuation of HPI was updated to 28.5 s after the Safety Injection actuation signal.
- The Safety Injection actuation signal is generated based on any one of three trips: Trip 24, Trip 38, or Trip 60. Trip 24 is the main steam isolation valve closure trip, Trip 38 is the high SG differential pressure trip, and Trip 60 is the low pressurizer pressure trip. Trip 60 was a time trip in the LBLOCA input model. For the SBLOCA analysis the setpoint condition for Trip 60 was modified to a pressurizer pressure below 1,615 psia (11.135 MPa).
- The primary coolant pump trip (Trip 22) was modified so that the pumps are tripped based on an assumed loss of offsite power coincident with the reactor scram trip (Trip 10).
- Loop T_{avg} is used in several controllers in the TRACE model. The T_{avg} calculator in the control system is a three step process. The first step calculated T_{avg} by averaging the loop T_{hot} and T_{cold}. The second step added the T_{avg} to itself and divided by two. The third step added one-half T_{avg} in step one to one half the value in step two. Steps 2 and 3 are redundant and were removed from the control system.
- Two signal variables were added to the control system to calculate the collapsed liquid levels in the downcomer and the core, SV4 and SV5 respectively. The reference points for the collapsed liquid level indications are Level 1 (bottom of the vessel) and Level 16 (the hot and cold leg connection level). These signal variables are used for convenience in tracking the collapsed liquid levels in the vessel.
- Control Blocks were added to calculate the total ECCS injected into the system. The total ECCS injected into the system is identified with control block 100 (CB100).

2.3 Model Modifications for the SBLOCA Transient Calculations

This section describes the model modifications made to implement the SBLOCA transients. These transients are divided into two categories: cold leg breaks and pressurizer surge line/ECCS line breaks. The small cold leg breaks were assumed to be located on the bottom of the Coolant Loop 2 pump discharge pipe, between the ECCS and charging injection nozzles and the reactor vessel. The pressurizer surge line and ECCS line breaks were assumed to be double-ended guillotine breaks.

2.3.1 Small Cold Leg Break Modifications

The recommendations for break modeling in the TRACE User Guide (Reference 4) were followed. The recommended method for modeling small breaks is to use a TEE component with a single convergent cell in the side tube. The side tube cell length is the pipe wall thickness. The entrance-to-exit flow area ratio of the side tube cell should be 3.0 (this is the ratio of FA at the internal-junction interface with main-tube cell JCELL to FA at the side-tube Cell 1 junction with the BREAK).

The small cold leg breaks were assumed to occur in Coolant Loop 2. PIPE component 29 (see **Figure 2-4**) was modified to a TEE component to model the small cold leg breaks. The side tube of the TEE was connected to Cell 1 of the main tube. The side tube cell length was set equal to the thickness of the cold leg pipe wall. The side tube was modeled using a single convergent cell. Choking was activated at the exit face of the side tube cell. **Figure 2-10** shows the cold leg small break configuration.

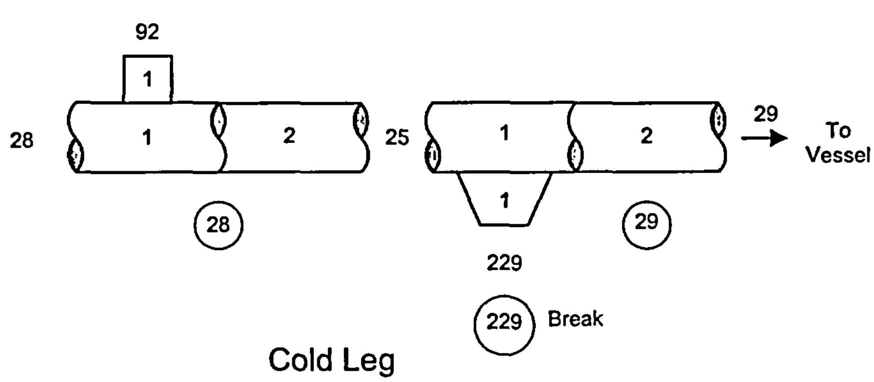


Figure 2-10. Small Cold Leg Break Configuration

2.3.2 Pressurizer Surge Line Break Modifications

The pressurizer surge line break was modeled by placing BREAK components at the exit of the side tube of TEE 30 (in the Loop 3 hot leg region) and the entrance of PIPE 31 (the remainder of

the pressurizer surge line). **Figure 2-11** shows the model configuration for the pressurizer surge line break.

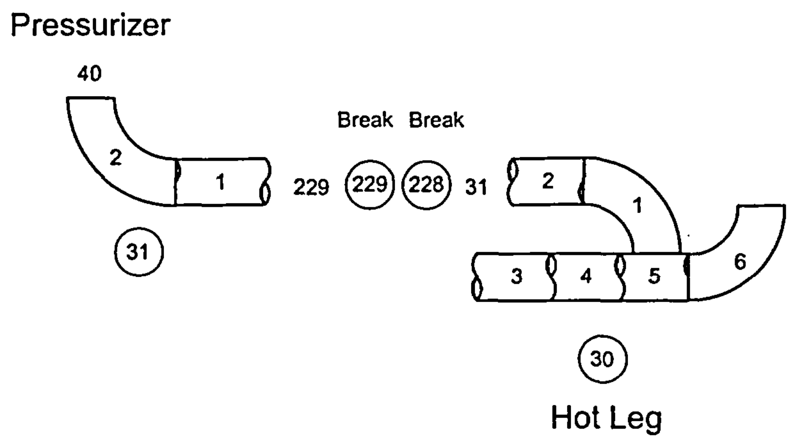


Figure 2-11. Pressurizer Surge Line Break Configuration

2.3.3 Safety Injection Line Break Modification

The safety injection line break was modeled in Coolant Loop 2 by placing BREAK components at the exit of the side tube of TEE 27 and the exit of the main tube of TEE 64 (see **Figure 2-8**). The model configuration for the safety injection line break is shown in **Figure 2-12**.

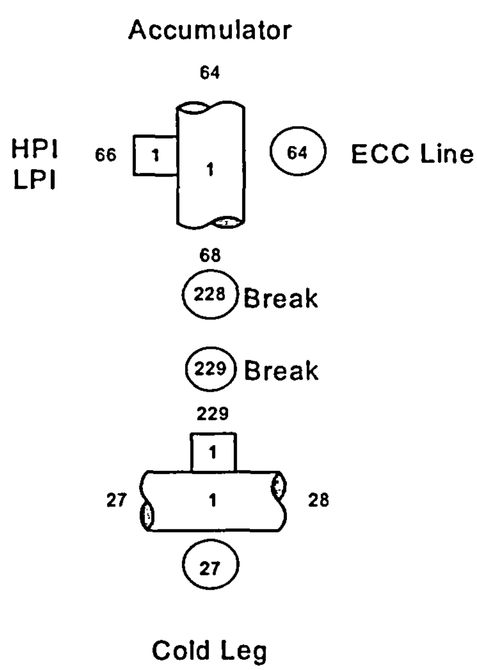


Figure 2-12. Safety Injection Line Break Configuration

3.0 SBLOCAS USING FSAR PLANT CONDITIONS AND ASSUMPTIONS

Seven cold leg SBLOCAs, one pressurizer surge line double-ended guillotine break (DEGB) LOCA, and three safety injection line DEGB LOCAs were simulated with a TRACE model of a typical Westinghouse three-loop plant using conditions and assumptions consistent with the current licensing bases. These eleven transient simulations were repeated with an additional 50-s delay included for the startup of the diesel-electric generator. **Table 3-1** lists the locations and sizes for the eleven simulated SBLOCAs.

The TRACE plant input model used for these calculations is described in Section 2.0. A summary of the licensing-basis analysis conditions and assumptions is given in Section 3.1. The results for a TRACE simulation of plant full-power steady state operation are compared against the desired licensing-basis plant conditions in Section 3.2. The TRACE transient SBLOCA calculations were initiated from this steady state condition and the calculation results are presented in Section 3.3.

Table 3-1. Break Size and Location for the Small Break LOCA Simulations

Break Size and Location	**Break Area (ft^2)**
2 inch – bottom of Loop 2 cold leg	0.0218
2.25 inch – bottom of Loop 2 cold leg	0.0276
2.5 inch – bottom of Loop 2 cold leg	0.0341
2.75 inch – bottom of Loop 2 cold leg	0.0412
3 inch – bottom of Loop 2 cold leg	0.0491
4 inch – bottom of Loop 2 cold leg	0.0873
6 inch – bottom of Loop 2 cold leg	0.1964
Pressurizer Surge Line – Loop 3 (10.5 inch line)	0.6013
Safety Injection Line – Loop 2 – nominal (8.5 inch line)	0.3941
Safety Injection Line – Loop 2 – 80% of nominal	0.3153
Safety Injection Line – Loop 2 – 120% of nominal	0.4729

3.1 Summary of Current Licensing Basis Conditions and Assumptions

The assumptions applied to the TRACE calculations described in this section include the conservatisms typically inherent in Westinghouse three-loop plant licensing calculations. The key assumptions for the SBLOCA analysis are as follows. The plant is assumed to be operating at 102% of its rated thermal power. The core power distribution is such that the peak local power (at the hottest spot on the hottest fuel rod) is at the plant licensing limit ($F_Q = 2.50$). Six percent of the SG tubes are assumed to be plugged. Off-site power is assumed to be lost at the time of the reactor scram. This result trips the three reactor coolant pumps initiating pump coastdown, and causes a delay in the delivery of the pumped ECC system flows. The assumption is made that only one diesel-electric generator starts and loads providing availability of one HPI pump and one LPI pump. The time delays (resulting from the diesel-electric generator start time and electrical load sequencer effects) are 28.5 s for the delivery of HPI flow

and 44.7 s for delivery of LPI flow. **Table 1-1** in Section 1 provides more details of the assumptions used for the calculations presented in this report.

3.2 Steady State Used as Initial Conditions for the SBLOCA Calculations

A TRACE calculation was performed for a period of 1,000 s in order to establish steady conditions consistent with the licensing-basis plant operation. This steady-state calculation, which provides the initial conditions from which the SBLOCA transient calculations were started, was performed using an input deck identified as "hbrss7A.inp." **Table 3-2** compares the TRACE-calculated values for key parameters at the end of the steady state calculation with the desired plant values for those parameters. The table indicates excellent agreement between the calculated and desired plant parameter values. This calculated steady state therefore represents an adequate set of starting conditions for the transient TRACE SLOCA calculations.

Table 3-2. Comparison of TRACE-Calculated and Desired Typical Westinghouse Three-Loop Plant Steady-State Conditions

Parameter	TRACE-Calculated Parameter Value	Desired Value for Typical Westinghouse Three-Loop Plant
Core Power (MWt)	2,346	2,346[a]
Total Peaking Factor, F_Q	2.5	2.5
RCS Average Temperature (°F)	575.3	575.4[b]
Hot Leg Temperature (°F)	605.5	---
Cold Leg Temperature (°F)	545.0	548.4[c]
Pressurizer Pressure (psia)	2,248	2,250
Total Coolant Loop Flow Rate (Mlbm/hr)	100.56	100.3
SG Secondary Pressure (psia)	797	800
SG Flow Rate per Generator (Mlbm/hr)	3.43	3.3
Main Feedwater Temperature (°F)	420	441.5
Number of tubes plugged per SG	193	193
Accumulator Pressure (psia)	615	615
Accumulator Fluid Temperature (°F)	120	120
SI Fluid Temperature (°F)	70	70
SIS Activation Setpoint Pressure (psia)	1,615	1,615

a 102% of nominal power
b Nominal condition value
c Temperature used in the FSAR SBLOCA analysis

3.3 TRACE Simulated Small Break LOCA Results

TRACE simulations of SBLOCAs for a typical Westinghouse three-loop plant were performed over a 6,000 s period or until the fuel rods were quenched and no further heatup was expected. The input deck identifiers for these calculations are listed in **Table 3-3**. Results from the TRACE SBLOCA simulations are presented here. A detailed discussion of the TRACE results for the 2.5-inch cold leg break using typical Westinghouse three-loop plant conditions and assumptions consistent with current licensing basis is presented first. This is the break size for which TRACE calculated the highest PCT. A discussion of the variations observed among all of the SBLOCA simulations is then presented.

3.3.1 2.5 Inch Small Cold Leg Break Simulation

Table 3-4 lists the event timing of key events during the 2.5 inch cold leg break TRACE simulation.

The RCS pressure rapidly declined when the break opened, as shown in **Figure 3-1**. The rapid depressurization slowed when the flashing of RCS coolant began. At 19.5 s the RCS pressure dropped below the low pressurizer pressure setpoint and the reactor tripped. A loss of offsite power was assumed, coincidental with the reactor trip, and the reactor coolant pumps were also tripped. The turbine stop valves in the steam line closed and the SG pressures began to increase, also as shown in **Figure 3-1**. Low pressurizer pressure also led to a safety injection actuation signal at 27.9 s, starting the sequencing to get the diesel driven electric generator up on line and the HPI and LPI pumps started. It was assumed power to the HPI pump was available to deliver flow 28.5 s after the SI actuation signal (for the LPI pump, it was 44.7 s).

Table 3-3. TRACE Input File Identifier for the Small Break LOCA Simulations

Break Simulation	Input File Identifier	Assumption
2 inch Cold Leg	2inSBfromss7A.inp	Base
2.25 inch Cold Leg	2p25inSBfromss7A.inp	Base
2.5 inch Cold Leg	2pt5inSBfromss7A.inp	Base
2.75 inch Cold Leg	2p75inSBfromss7A.inp	Base
3 inch Cold Leg	3inSBfromss7A.inp	Base
4 inch Cold Leg	4inSBfromss7A.inp	Base
6 inch Cold Leg	6inSBfromss7A.inp	Base
Pressurizer Surge Line	PSurgefromss7A.inp	Base
Safety Injection Line 100%	SI-Linefromss7A.inp	Base
Safety Injection Line 80%	SI-Line80fromss7A.inp	Base
Safety Injection Line 120%	SI-Line120fromss7A.inp	Base
2 inch Cold Leg	2inSBfromss7AD.inp	Diesel generator delay[a]
2.25 inch Cold Leg	2p25inSBfromss7AD.inp	Diesel generator delay
2.5 inch Cold Leg	2pt5inSBfromss7AD.inp	Diesel generator delay
2.75 inch Cold Leg	2p75inSBfromss7AD.inp	Diesel generator delay
3 inch Cold Leg	3inSBfromss7AD.inp	Diesel generator delay
4 inch Cold Leg	4inSBfromss7AD.inp	Diesel generator delay

6 inch Cold Leg	6inSBfromss7AD.inp	Diesel generator delay
Pressurizer Surge Line	PSurgefromss7AD.inp	Diesel generator delay
Safety Injection Line 100%	SI-Linefromss7AD.inp	Diesel generator delay
Safety Injection Line 80%	SI-Line80fromss7AD.inp	Diesel generator delay
Safety Injection Line 120%	SI-Line120fromss7AD.inp	Diesel generator delay

a An additional 50 s delay was assumed for the startup of the diesel-electric generator. This delay affects initiation of the SI system flows.

At about 100 s, the mismatch between the declining core power and flow caused the RCS pressure to momentarily increase. Afterward, the RCS pressure began declining again as a result of more RCS energy being removed at the break and to the SGs than produced in the core. The SG secondary pressure had reached the safety valve setpoint and energy was being removed through the valves providing cooling to the primary side. At 145 s the RCS pressure declined below the shutoff head of the HPI system and HPI flow was initiated.

Table 3-4. Timing of Key Events for the 2.5 Inch Cold Leg Break Simulation

Event	**Time (s)**
Break Opens	0.0
Reactor Trip Signal	19.5
SI Actuation Signal	27.9
HPI Flow Initiated	145
Loop 1 Loop Seal Cleared	565
Loop 2 Loop Seal Cleared (BL)	N/A
Loop 3 Loop Seal Cleared	N/A
Two Phase Level Drops Below Top of Core	820
Accumulator Injection Begins	1,925
PCT Reached	1,999
LPI Flow Initiated	N/A
TRACE Calculation Terminated	4,200

By 200 s the primary coolant pumps rotors had stopped turning and the coolant loop flow transitioned from forced to natural circulation flow as shown in **Figure 3-2**. The RCS depressurization continued to slow as a result of flashing in the RCS as shown in **Figure 3-1**.

At about 565 s the loop seal in the Loop 1 cold leg cleared of coolant as shown in **Figure 3-3**. Clearing the loop seal relieved pressure in the upper plenum, resulting in a readjustment of the vessel downcomer and core liquid levels as shown in **Figure 3-4**. During this readjustment the liquid in the cold legs drained into the downcomer and the void fraction at the break increased to 1.0 as shown in **Figure 3-5**. With single-phase steam exiting the break, the break mass flow rate rapidly declined as shown in **Figure 3-6**.

After the loop seal cleared the RCS depressurization rate increased (**Figure 3-1**), and eventually the RCS pressure declined below the SG secondary-side pressures. Because of the declining RCS pressure, the break flow rate continued to decline and the HPI flow rate increased (**Figure 3-6**). However, the break flow rate continued to exceed the HPI flow rate and the RCS

continued to lose coolant mass, as indicated by the decreasing downcomer and core collapsed liquid levels shown in **Figure 3-4**.

At around 820 s the level in the core had declined sufficiently that the fuel rods in the upper region of the core began to heat up as shown in **Figure 3-7**. As the core level continued declining, more and more of the axial length of the fuel rods was exposed to steam and the fuel rods transitioned from nucleate boiling to film boiling, continuing the fuel rod heatup.

The RCS pressure declined to the accumulator initial pressure setpoint at 1,925 s and the accumulators began discharging into the cold legs as shown in **Figure 3-6**. The volume of liquid injected into the RCS by the accumulators was large, overwhelming the break flow, and the core region of the vessel began to refill as shown in **Figure 3-4**. With the core refilling, the fuel rod temperature excursion was mitigated as shown in **Figure 3-7**. The PCT, 1,703 °F, was reached in the central region of the core (Ring 1 of the VESSEL model), which has the highest radial peaking factor.

The steam produced as the core refilled caused the RCS to momentarily repressurize and terminate the accumulator injection. Subsequently, when the RCS depressurization continued, accumulator flow was reestablished. In this manner, the accumulators continued to inject coolant into the system in spurts (**Figure 3-6**) and the core continued to refill (**Figure 3-4**). By about 3,000 s the core was sufficiently refilled that all of the fuel rods had transitioned from film boiling to nucleate boiling and the fuel rods were completely quenched.

The TRACE calculation was terminated by a run failure at 4,200 s (it took 6,2367 CPU seconds to calculate the 4,200 s on a 2.79 GHz personal computer with 1 GB of RAM) . At that time the RCS pressure was still above the LPI system shutoff head (138 psia). However, at the end of the calculation most of the initial accumulator water inventory remained (to that point, only 16% of the initial accumulator water inventory had been expelled). Had the TRACE calculation been continued, the remaining accumulator water inventory is expected to keep the core water-filled and the fuel rods cool until the RCS pressure declines sufficiently to allow for LPI flow.

3.3.2 Comparisons Among the Small Break Simulations

The previous section described in detail the TRACE calculation results for the 2.5-inch cold leg SBLOCA in a Westinghouse three-loop plant using licensing-basis analysis assumptions. This section compares and contrasts the TRACE calculation results for SBLOCAs of various sizes and locations, including the effects of assuming an additional 50-s delay for emergency diesel-electric generator startup.

Table 3-5 compares the timing of key events among the SBLOCA calculations. As expected, the sequence event timing becomes more rapid as the break size increases. For break sizes smaller than 4 inches, no significant differences in event timing are noted as a result of assuming the additional 50-s diesel generator start time. For 4-inch breaks and larger, the system depressurization becomes rapid enough that the RCS pressure declines below the HPI pump

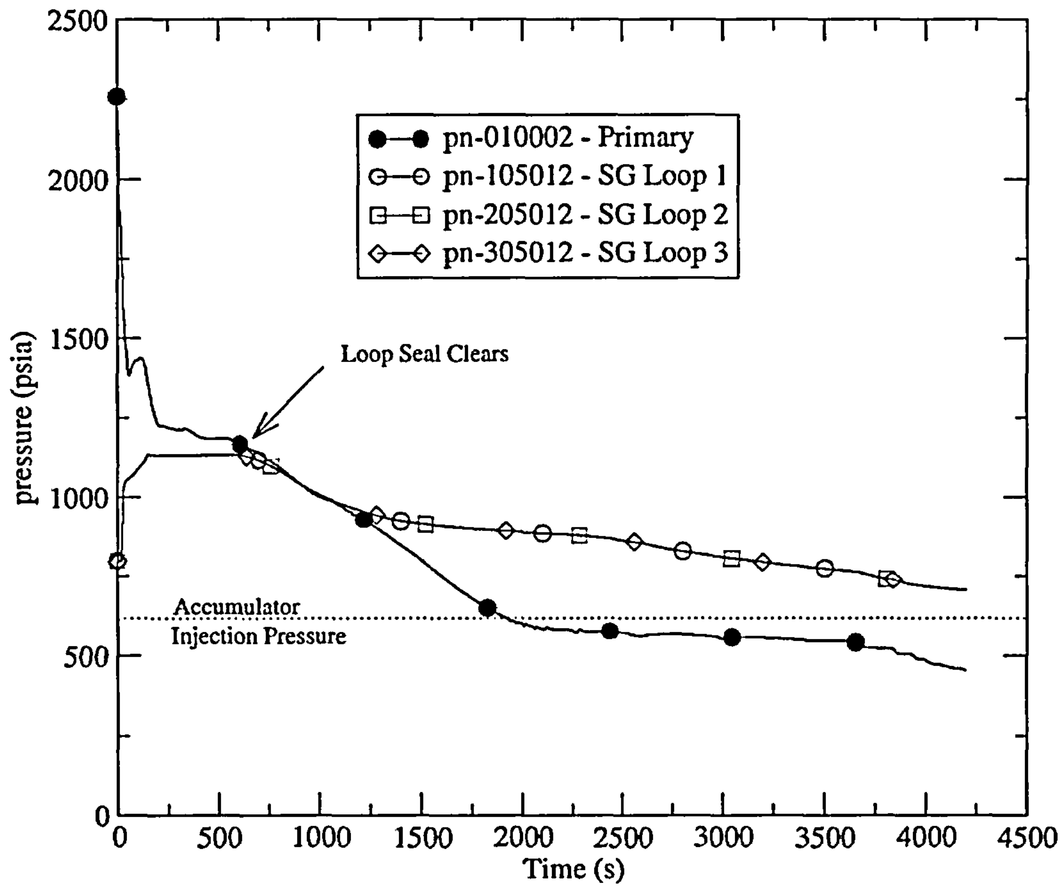


Figure 3-1. Primary and Secondary Pressure Response – 2.5 Inch Cold Leg Break

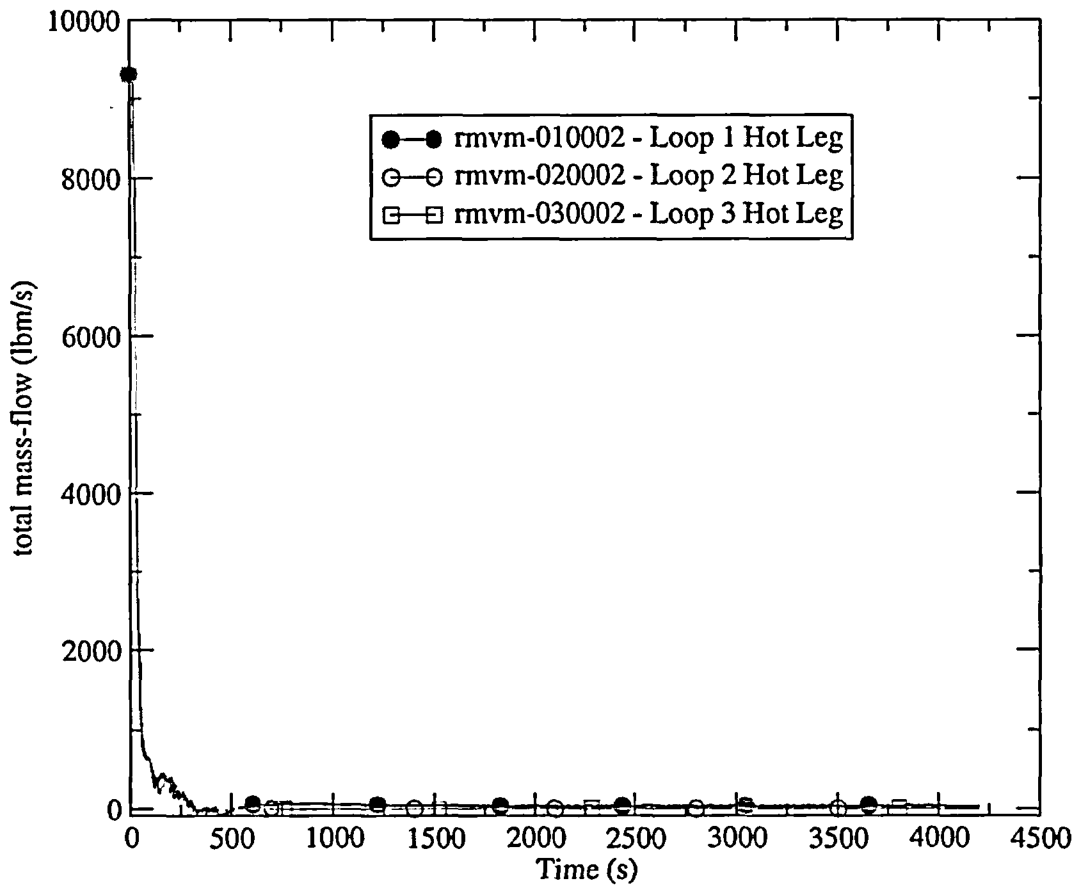


Figure 3-2. Hot Leg Mass Flow Rates – 2.5 Inch Cold Leg Break

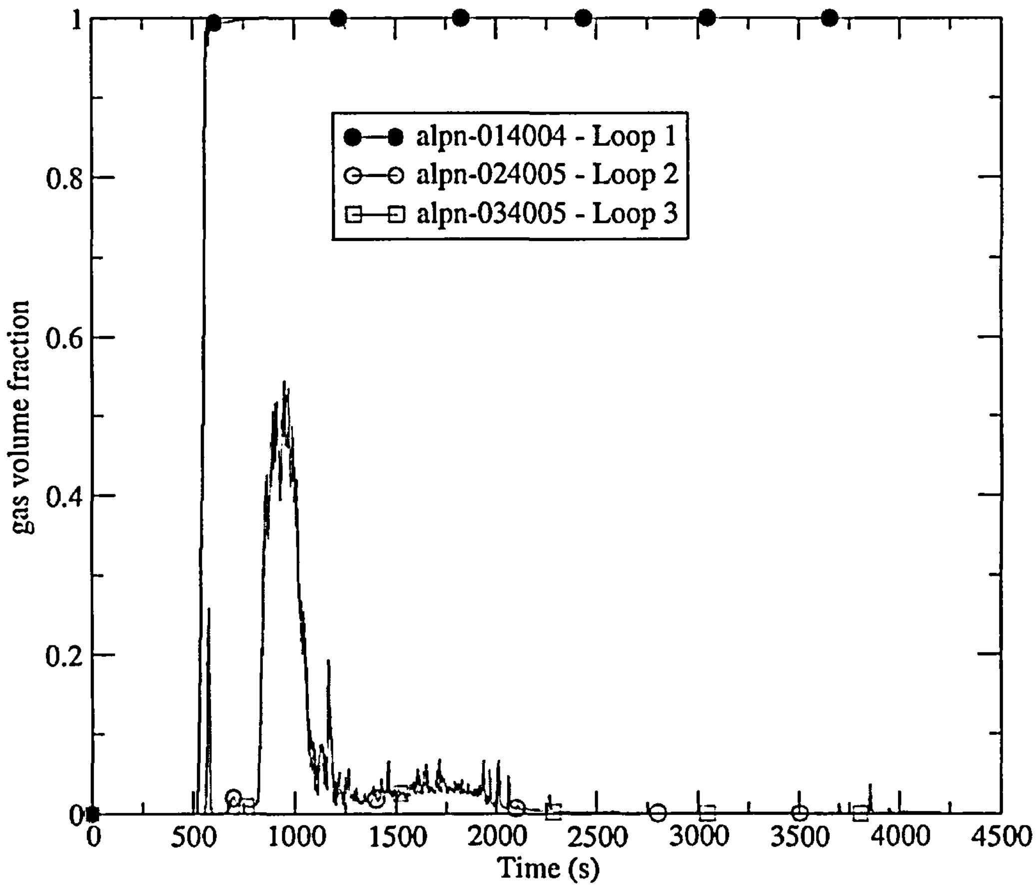


Figure 3-3. Void Fractions in the Bottom of the Loop Seals – 2.5 Inch Cold Leg Break

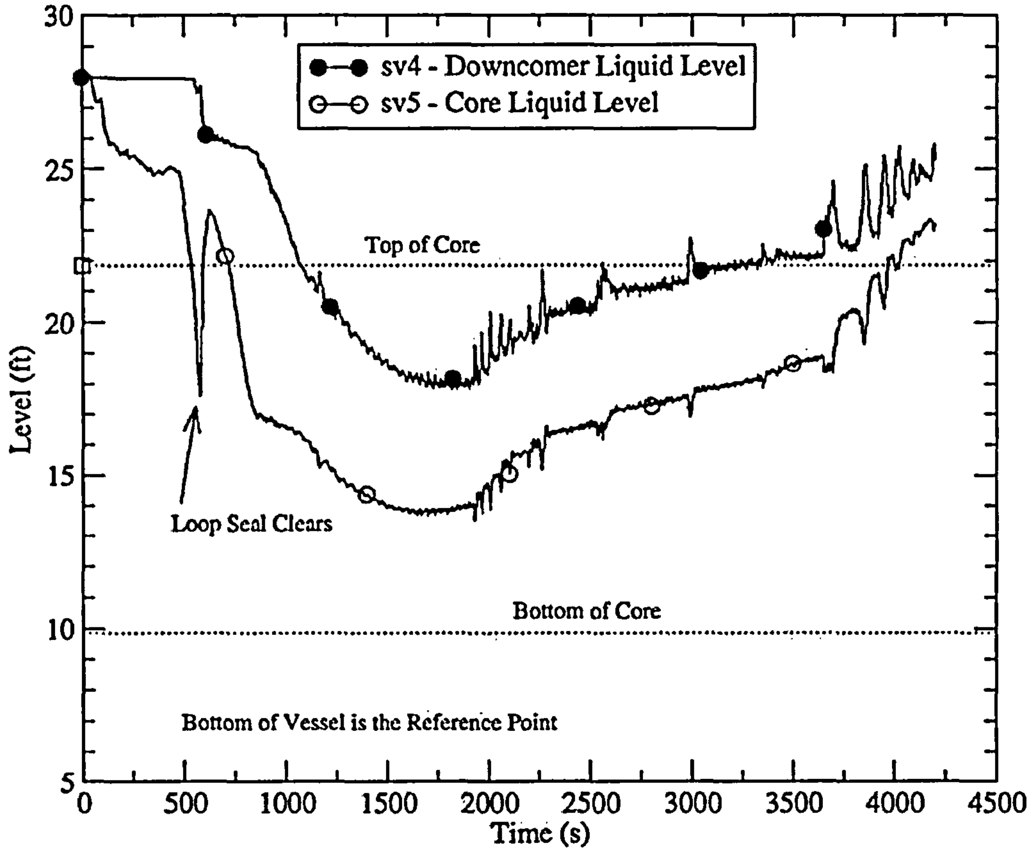


Figure 3-4. Downcomer and Core Collapsed Liquid Levels – 2.5 Inch Cold Leg Break

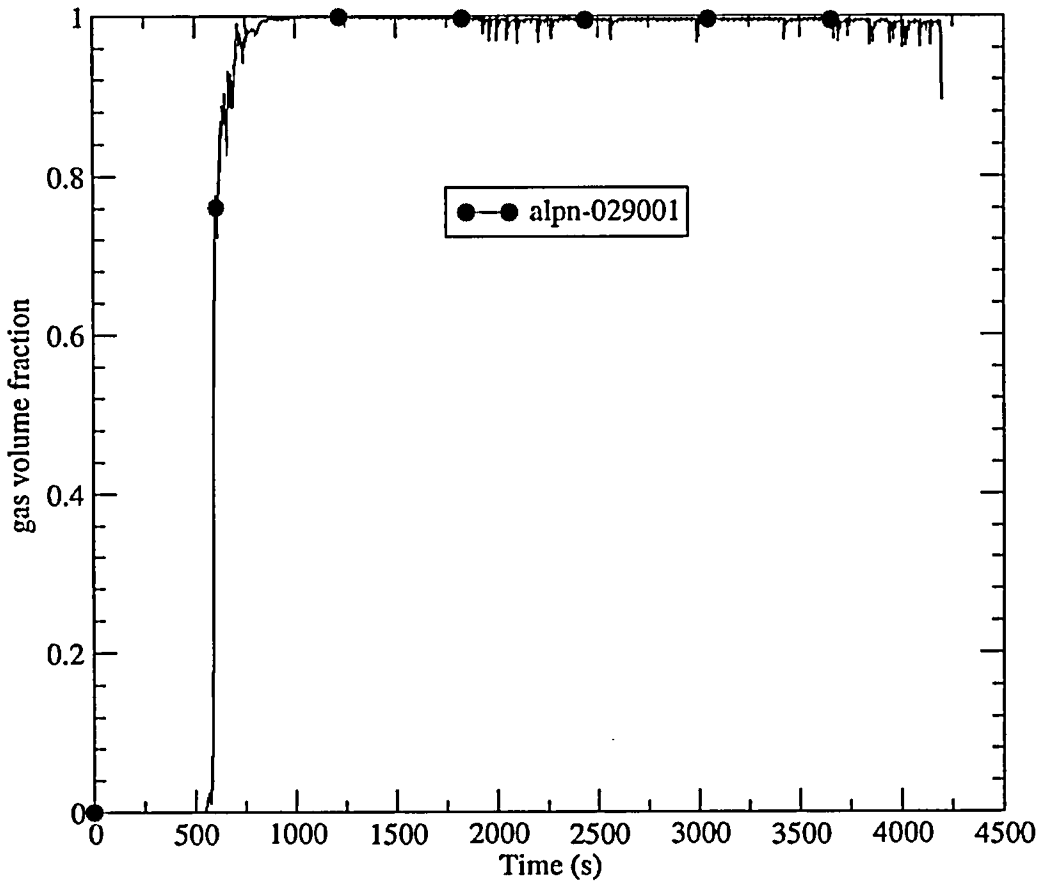


Figure 3-5. Void Fraction at the Break – 2.5 Inch Cold Leg Break

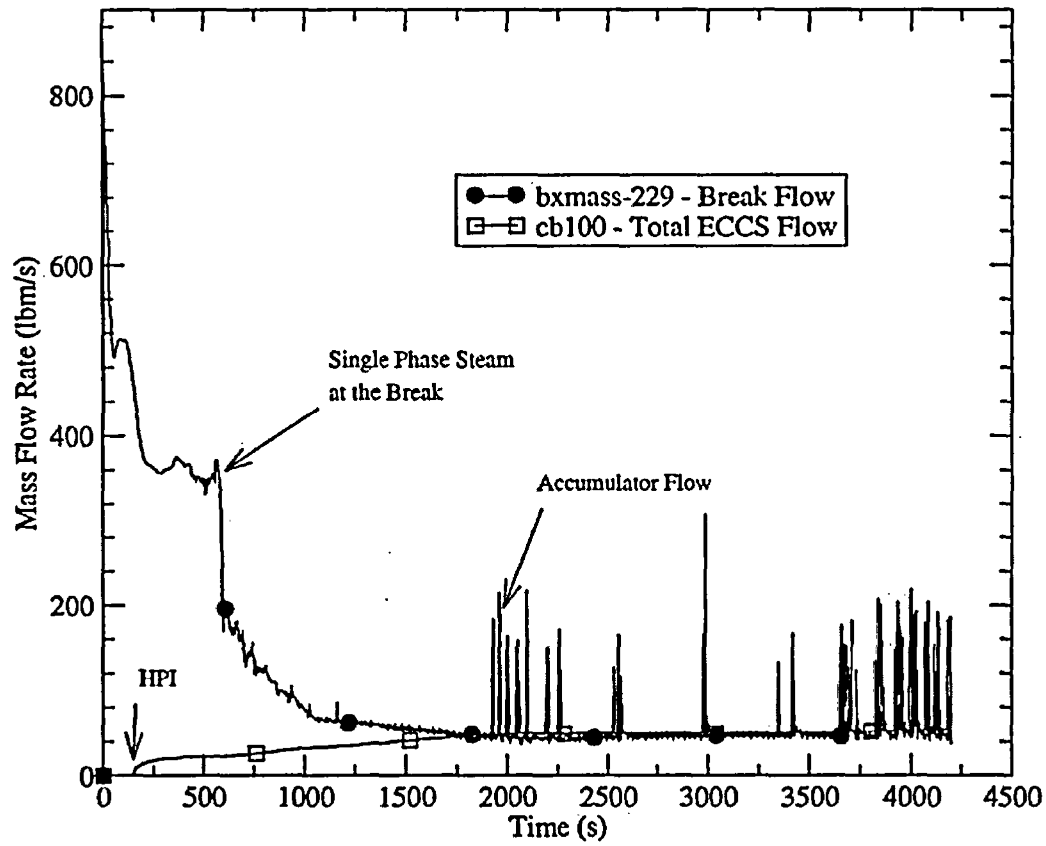


Figure 3-6. Break Mass Flow Rate Versus Total ECCS Mass Flow Rate – 2.5 Inch Cold Leg Break

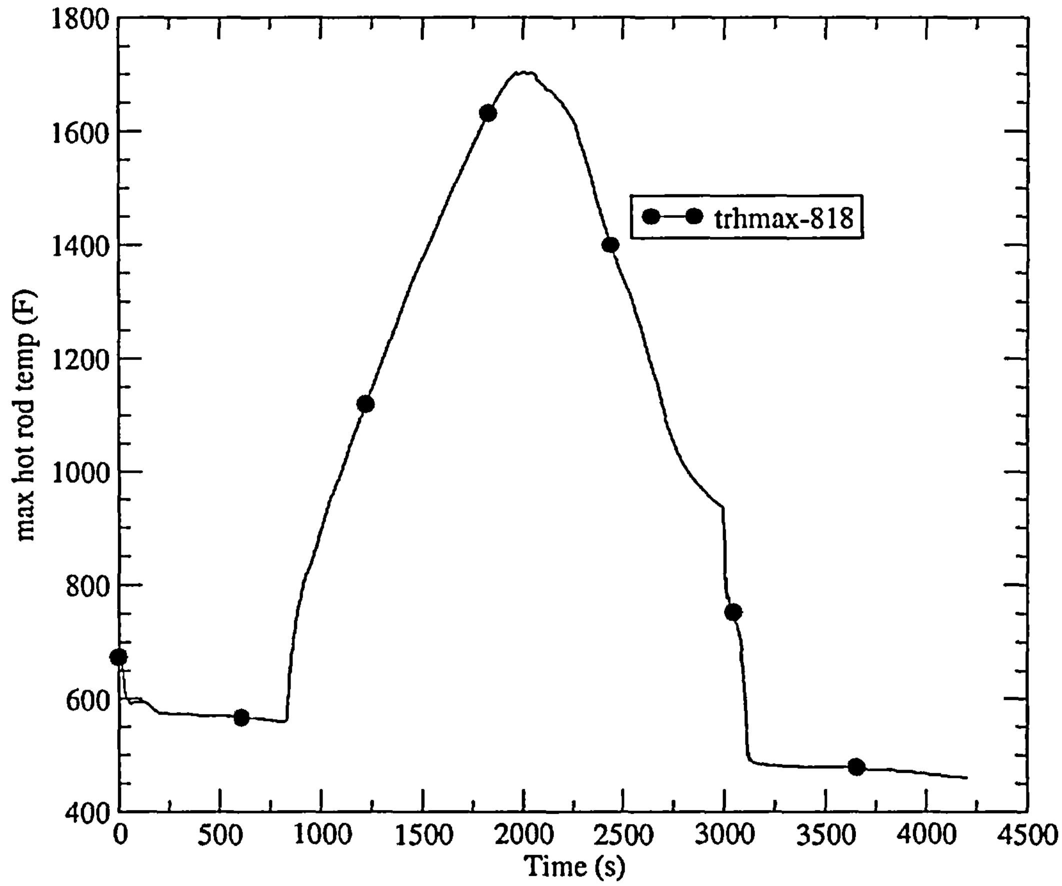


Figure 3-7. Maximum Hot Rod Clad Temperature – 2.5 Inch Cold Leg Break

shutoff head before the HPI pumps are energized. For these larger breaks, the delay in the initiation of the HPI flow fully reflects the additional diesel generator delay.

Figures 3-8, 3-9 and **3-10** compare the calculated RCS pressure response for the SBLOCAs with and without the additional 50-s diesel-electric generator delay. Data are grouped into cold leg breaks smaller than 3 inches, cold leg breaks of 3 inches and larger, and surge and SI line breaks. The calculated pressure response for the different size breaks is as expected (i.e., a larger break leads to a faster depressurization). The RCS pressure for the 2 inch cold leg break did not decline below the accumulator pressure, thus HPI was the only ECCS active. For break sizes between 2.25 and 4 inches, the RCS pressure did not decrease below the LPI pump shutoff head during the calculation period. For all break sizes, there was no significant difference observed in the RCS pressure behavior between the runs with and without the additional 50-s diesel generator delay.

The integrated break mass flow rates among the SBLOCA calculations are compared in **Figures 3-11, 3-12** and **3-13**. The rise in the curves over the early portion of the event sequence reflects discharge of single-phase liquid out the break. As the break flow transitions to two-phase and single-phase steam the break mass flow rate decreases; this is reflected by the change in the slope of the curves. When ECCS is injected at a high rate, the slope of the integrated break mass flow rate again changes, reflecting excess ECCS liquid leaving the RCS through the break. As expected, the larger the break the higher the mass discharge out the break. For all break sizes, there was no significant difference observed in the break flow behavior between the runs with and without the additional 50-s diesel generator delay.

Figures 3-14, 3-15 and **3-16** compare the core collapsed liquid levels among the SBLOCA calculations. When the RCS mass is initially depleted by the flow out the break, the core liquid level declines. Loop seal clearing leads to a recovery in the core level. When the loop seals clear, a more open path to the break is created and this relieves the pressure in the core. This pressure relief results in a flow of liquid from the downcomer into the core. With break flow still exceeding the ECCS flow, the RCS mass inventory continues to deplete and the core level declines again. The core level increases when accumulator and LPI flow are initiated. The exception is for the 2 inch break calculations, for which the HPI injection alone was sufficient to recover the core level. For all break sizes, there was no significant difference observed in the core level behavior between the runs with and without the additional 50-s diesel generator delay.

Referring to **Figure 3-15**, it is noted that in the 6-inch break calculations the core level recovered quickly when the accumulators began to inject. However, the accumulators subsequently emptied before the RCS pressure dropped below the LPI shutoff head. This situation resulted in a second core uncovery; the core level quickly recovered when the LPI flow eventually was initiated.

The hot rod PCTs for the SBLOCA calculations are listed in **Table 3-6** and the responses are compared in **Figures 3-17, 3-18** and **3-19**. The SBLOCA that resulted in the highest PCT was the 2.5 inch cold leg break. The PCT for this calculation was 1,704 °F for the assumption of a 28.5-s delay in the diesel generator startup and 1,730 °F for the assumption of an additional 50 s delay in the diesel generator startup. Differences between the PCTs for the current licensing

Table 3-5. Event Timing Summary for all of the Small Break LOCA Simulations

Event	Break Size (in) or Location										
	2	2.25	2.5	2.75	3	4	6	Surge Line	SI Line 80%	SI Line 100%	SI Line 120%
Current Licensing Basis Assumption	Time (s)										
Break Opens	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	30.2	23.7	19.5	16.5	14.1	8.9	6.3	0.1	6.5	6.5	6.3
SI Actuation Signal	42.7	33.9	27.9	23.7	20.4	13.8	9.8	4.3	10.2	9.7	9.3
HPI Flow Begins	239	186	145	116	88	45	39	33.3	40.5	39.6	39.4
Loop 1 Loop Seal Cleared	868	695	565	468	391	226	117	150	104	86	76
Loop 2 Loop Seal Cleared (BL)	N/A	N/A	N/A	N/A	585	296	123	150	105	87	74
Loop 3 Loop Seal Cleared	N/A	N/A	N/A	725	1,171	301	155	154	111	95	82
Two Phase Level Drops below Top of Core	1,858	1,270	820	880	713	406	150[a]/ 775	N/A	80	70	64
Accumulator Injection Begins	N/A	2,961	1,925	1,440	1,123	575	226	135	171	139	116
PCT Reached	2,949	2,533	1,999	1,521	1,159	607	294	N/A	222	187	386
LPI Flow Begins	N/A	N/A	N/A	N/A	N/A	N/A	913	314	433	317	251
Additional 50 s Delay on Diesel-electric Generator Startup											
Break Opens	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor Trip Signal	30.3	23.7	19.5	16.5	14.1	8.9	6.3	0.1	6.5	6.5	6.3
SI Actuation Signal	42.7	33.9	27.9	23.7	20.4	13.8	9.8	4.3	10.2	9.7	9.3
HPI Flow Begins	236	181	145	116	100	95	90	84.6	91.8	91	88
Loop 1 Loop Seal Cleared	868	695	561	468	394	231	122	151	104	86	76
Loop 2 Loop Seal Cleared (BL)	N/A	N/A	N/A	N/A	N/A	298	128	151	104	87	74
Loop 3 Loop Seal Cleared	N/A	N/A	N/A	716	584	299	162	154	110	96	81
Two Phase Level Drops below Top of Core	1,868	1,251	820	902	736	401	156[a]/ 760	N/A	80	70	64
Accumulator Injection Begins	N/A	3,007	1,899	1,437	1,110	573	233	135	171	139	116
PCT Reached	2,949	2519	1,964	1,511	1,196	597	324	N/A	222	193	175
LPI Flow Begins	N/A	N/A	N/A	N/A	N/A	N/A	859	317	464	311	221

a Two phase level drops below the top of the core prior to loop seal clearing and after the accumulators empty

basis assumption calculations and those with the additional diesel delay time were no more than 108 °F. Because 1,730 °F was the highest calculated cladding temperature, fuel rod heating due to cladding oxidation (metal-water reaction) was not a significant effect.

Table 3-6. Predicted Maximum Hot Rod PCTs for the SBLOCAs

Break Location and Size	Peak Cladding Temperature (°F)	
	Current Licensing Basis Assumption	**Additional 50 s diesel-electric generator delay**
Cold Leg 2.0 in Diameter	1,191	1,195
Cold Leg 2.25 in Diameter	1,514	1,511
Cold Leg 2.5 in Diameter	1,704	1,730
Cold Leg 2.75 in Diameter	1,619	1,605
Cold Leg 3.0 in Diameter	1,568	1,517
Cold Leg 4.0 in Diameter	1,252	1,266
Cold Leg 6.0 in Diameter	1,172	1,209
Pressurizer Surge Line (10.5 in diameter) Double-Ended	No Heatup	No Heatup
Safety Injection Line (Nominal Area, 8.5 in diameter) Double-Ended	1,240	1,232
Safety Injection Line (80% Nominal Area,) Double-Ended	1,311	1,333
Safety Injection Line (120% Nominal Area) Double-Ended	1,363	1,471

For the 2, 2.25, and 2.5 inch cold leg break calculations, only a single fuel rod heatup period was noted (**Figure 3-17**). For all the other SBLOCAs two heatup periods were noted, one prior to loop seal clearing and another just before the core refills. The rod heatups just prior to loop seal clearing were minor for the 2.75, 3, and 4 inch cold leg breaks. However, for the 6 inch cold leg break the heatup just prior to loop seal clearing produced the hottest cladding temperature (**Figure 3-18**).

The fuel rods in the three SI line break calculations took a relatively long time to quench because the flows from one-third of all the ECC systems (HPI, accumulators and LPI) were assumed to be spilled directly to the containment.

The only significant difference in the PCT response between the base calculations and those with the additional diesel generator delay was observed in the SI line break simulations. Because of the large break size and degraded ECC system availability, the additional delay resulted in much less ECC coolant injected into the system. This caused the core level recovery following loop seal clearing to be slightly delayed (**Figure 3-16**) and the second rod temperature excursion to occur earlier (**Figure 3-19**).

The HPI and LPI flow in some Westinghouse three-loop plants passes through a common header before it is split out into the three loops. Details of this configuration (line length, pipe size, friction loss, etc.) were unknown. Therefore the total flow from the HPI and LPI was split

equally amongst the three loops, with the Loop 2 ECCS spilling on the containment floor. However, it is expected that given this ECC line configuration, the ECC flow to the unaffected loops would be somewhat less than is modeled, leading to higher PCTs than are currently calculated.

Figure 3-20 graphically displays the PCTs from **Table 3-6** with and without the additional 50-s delay in the diesel generator startup time. The figure and table show that for some breaks the additional delay resulted in an increase in the PCT while for other breaks it resulted in a decrease in the PCT. These PCT differences (which are relatively small) indicate a randomness in the effect on the PCT which is both expected and typical of SBLOCA analyses in general. This randomness is introduced because any model change alters the course of the transient calculation, such that conditions (pressures, temperatures, flows, etc.) present at certain critical times (for example, at the time when a loop seal clears) are slightly different, thus altering the calculated behavior later in time. Randomness is to some extent physical, as evidenced by the different responses obtained when attempting to repeat a SBLOCA experiment.

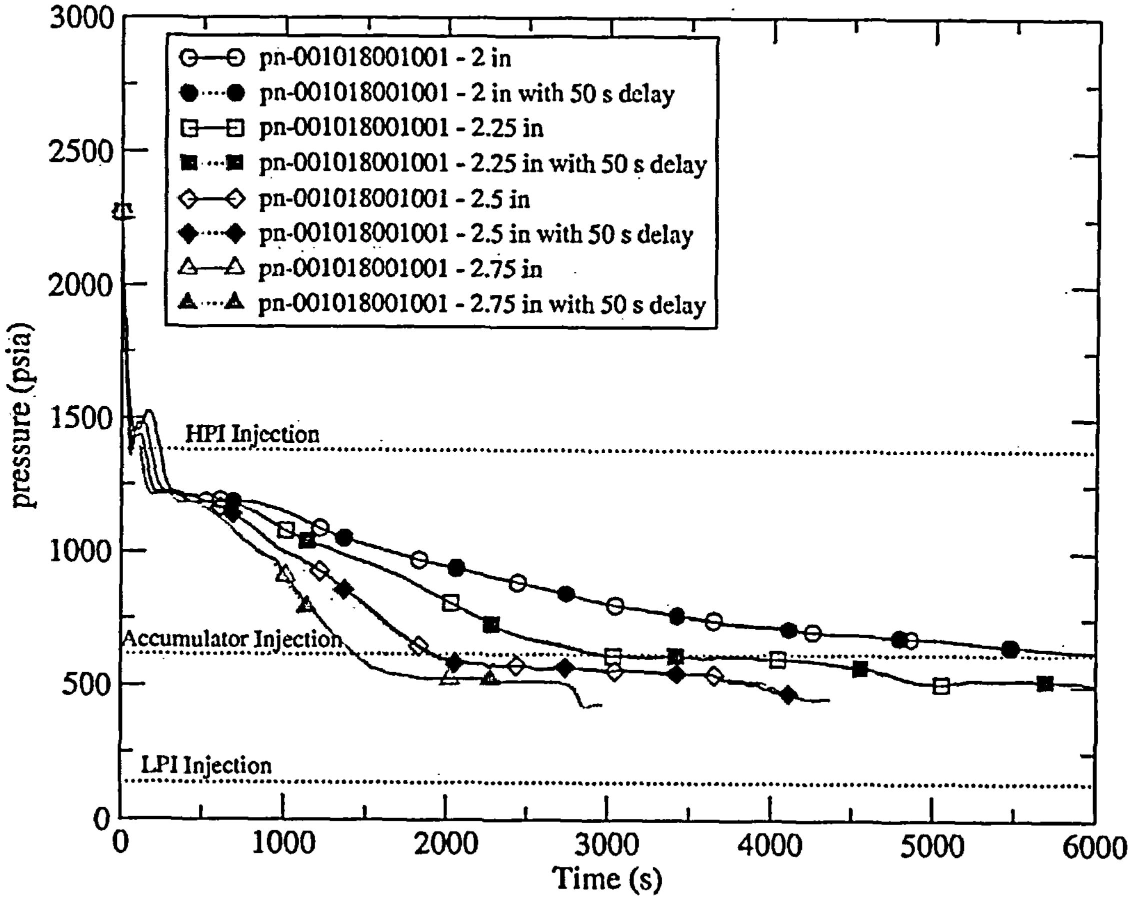


Figure 3-8. Comparison of RCS Pressures for Cold Leg Breaks Smaller than 3 Inches

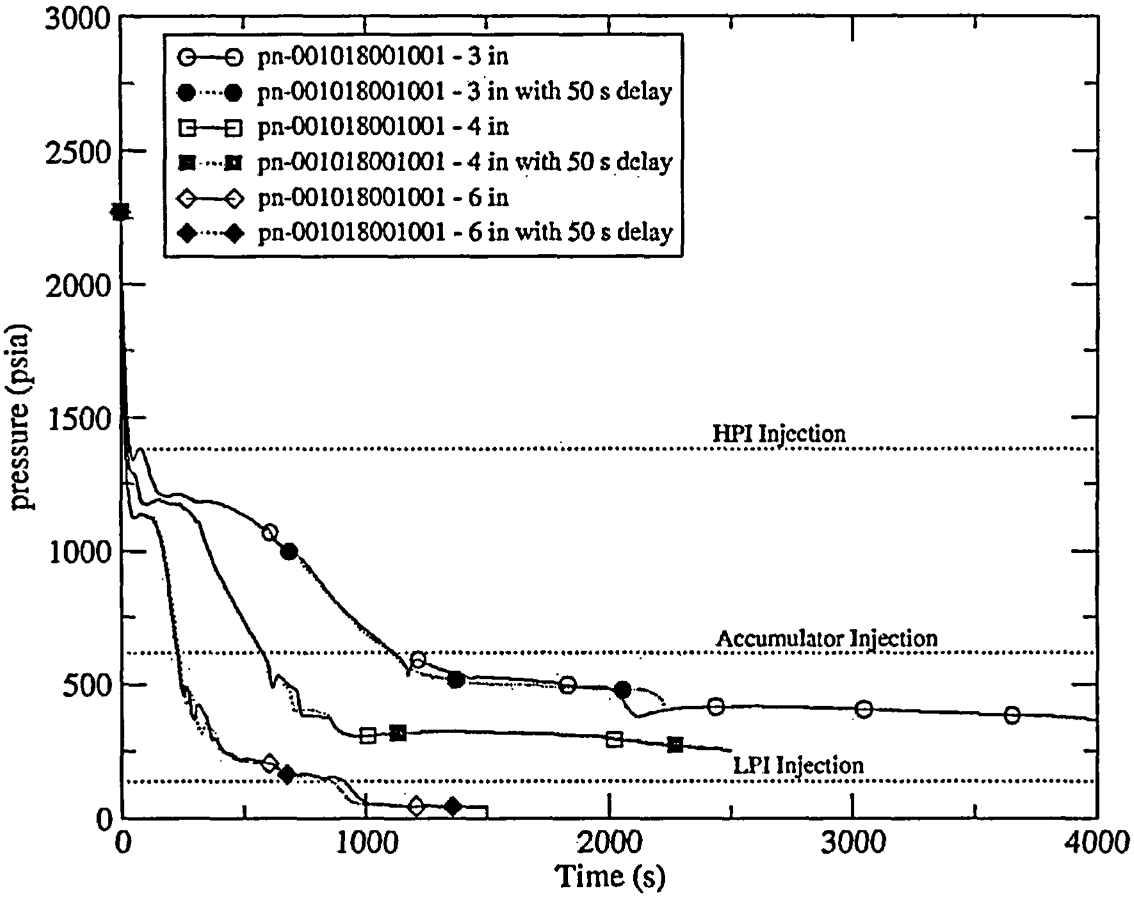


Figure 3-9. Comparison of RCS Pressures for Cold Leg Breaks of 3 Inches and Larger

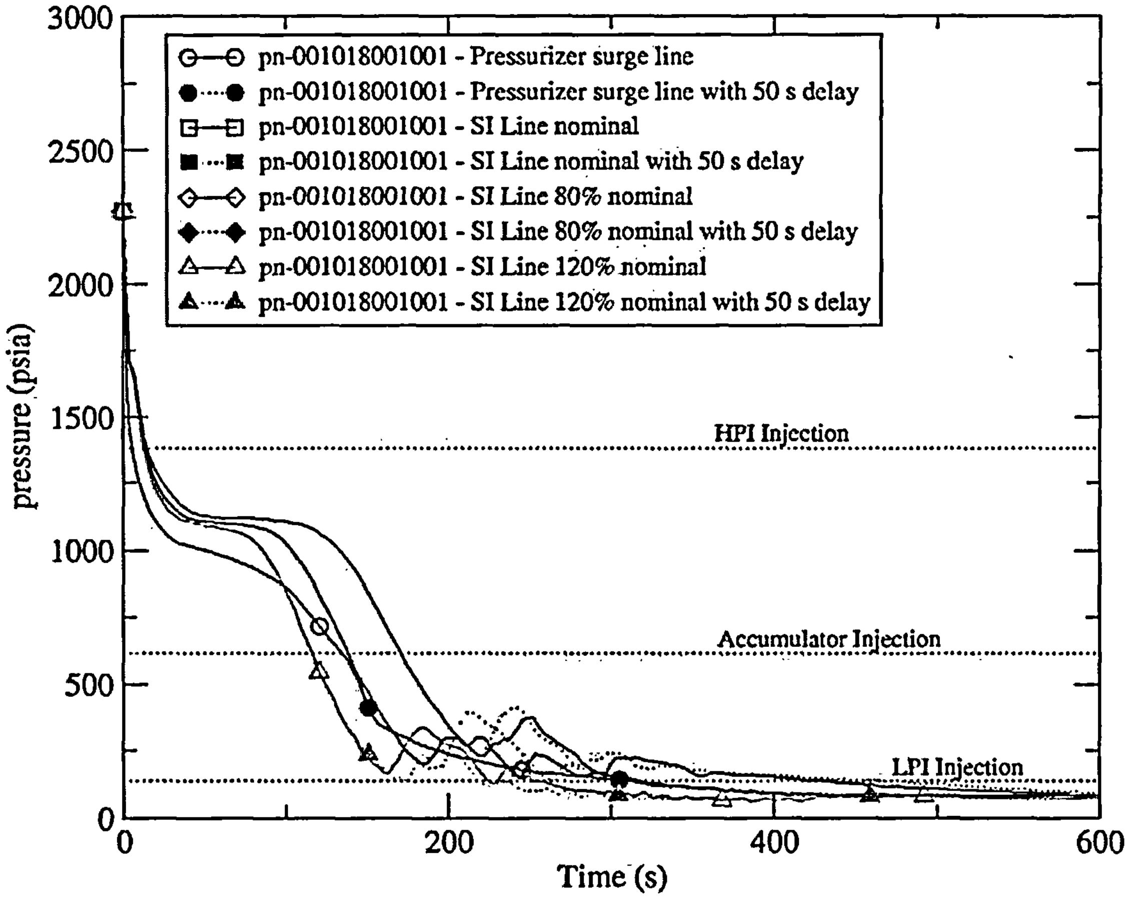


Figure 3-10. Comparison of RCS Pressures for Pressurizer Surge Line and Safety Injection Line Breaks

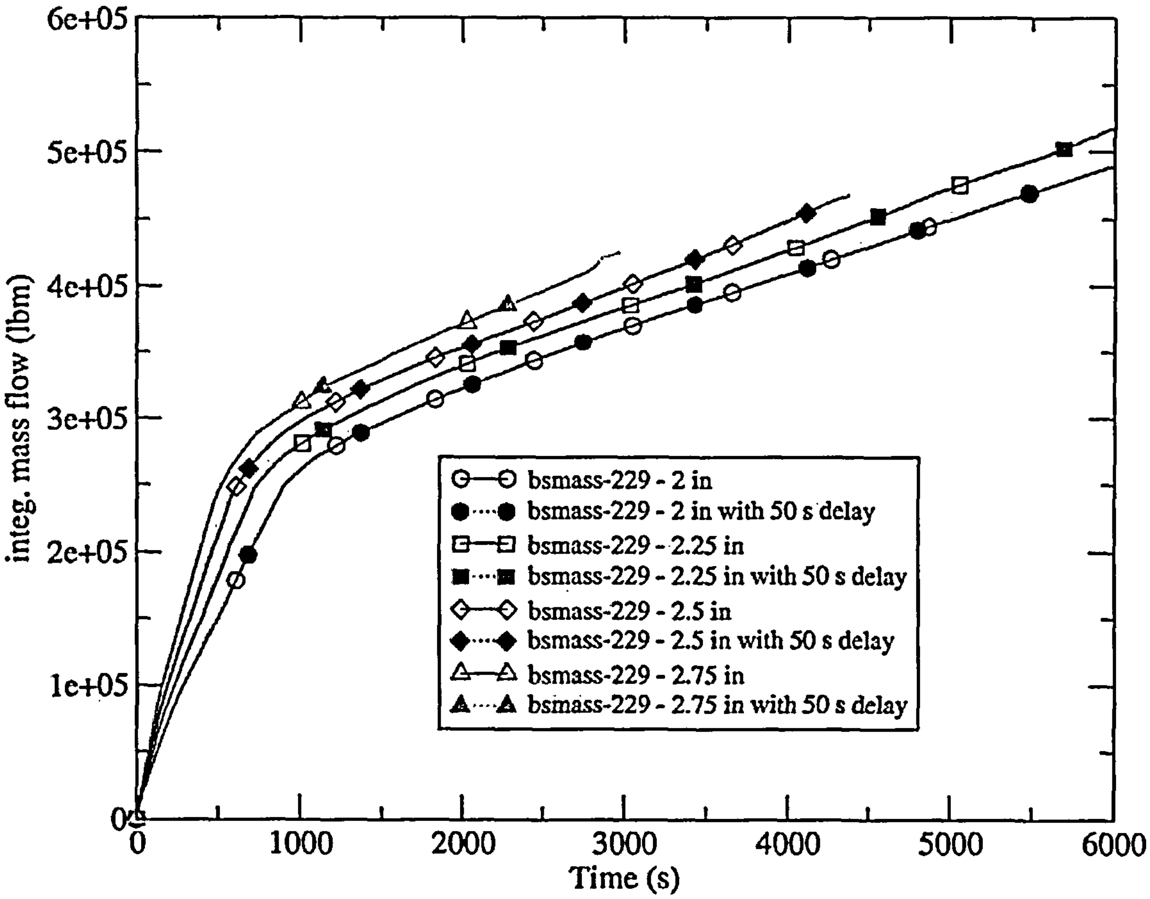


Figure 3-11. Comparison of Integrated Break Mass Flows for Cold Leg Breaks Smaller than 3 Inches

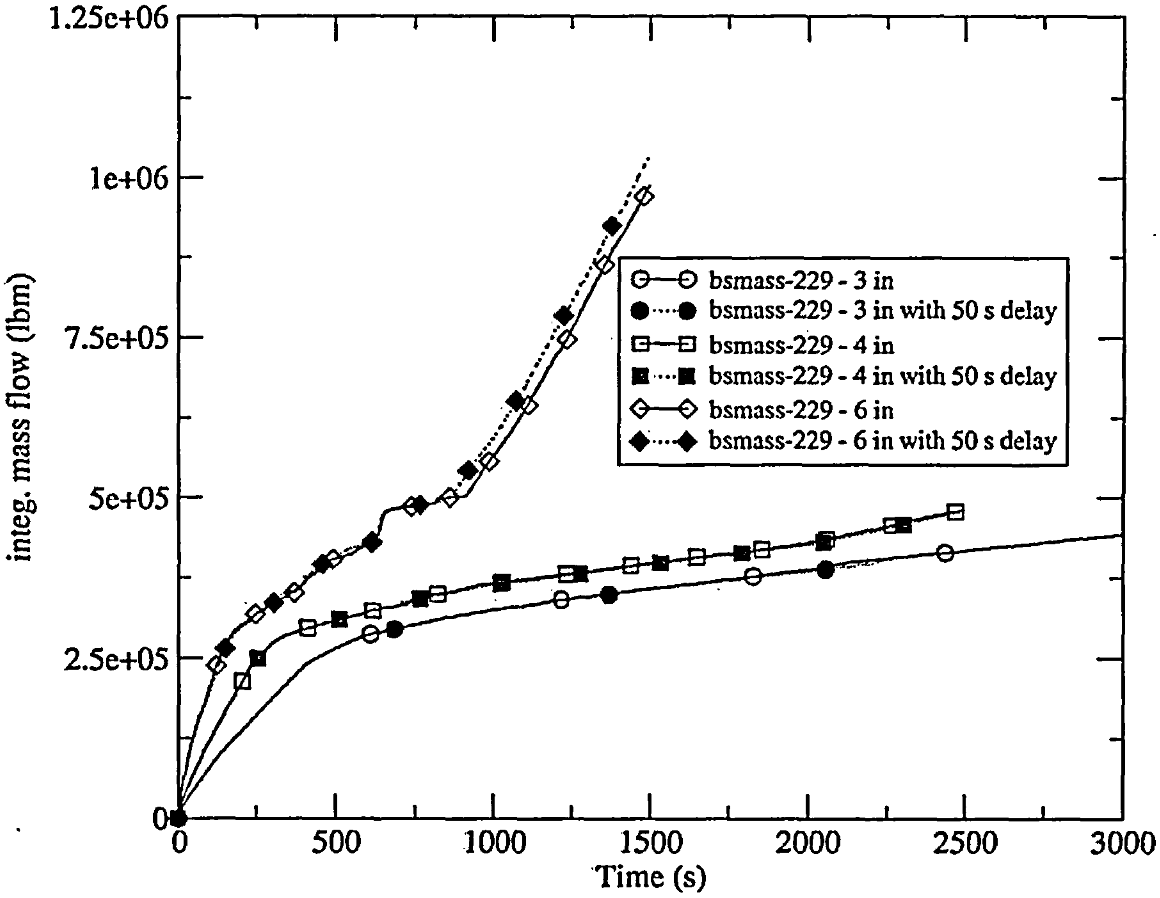


Figure 3-12. Comparison of Integrated Break Mass Flow Rates for Cold Leg Breaks of 3 Inches and Larger

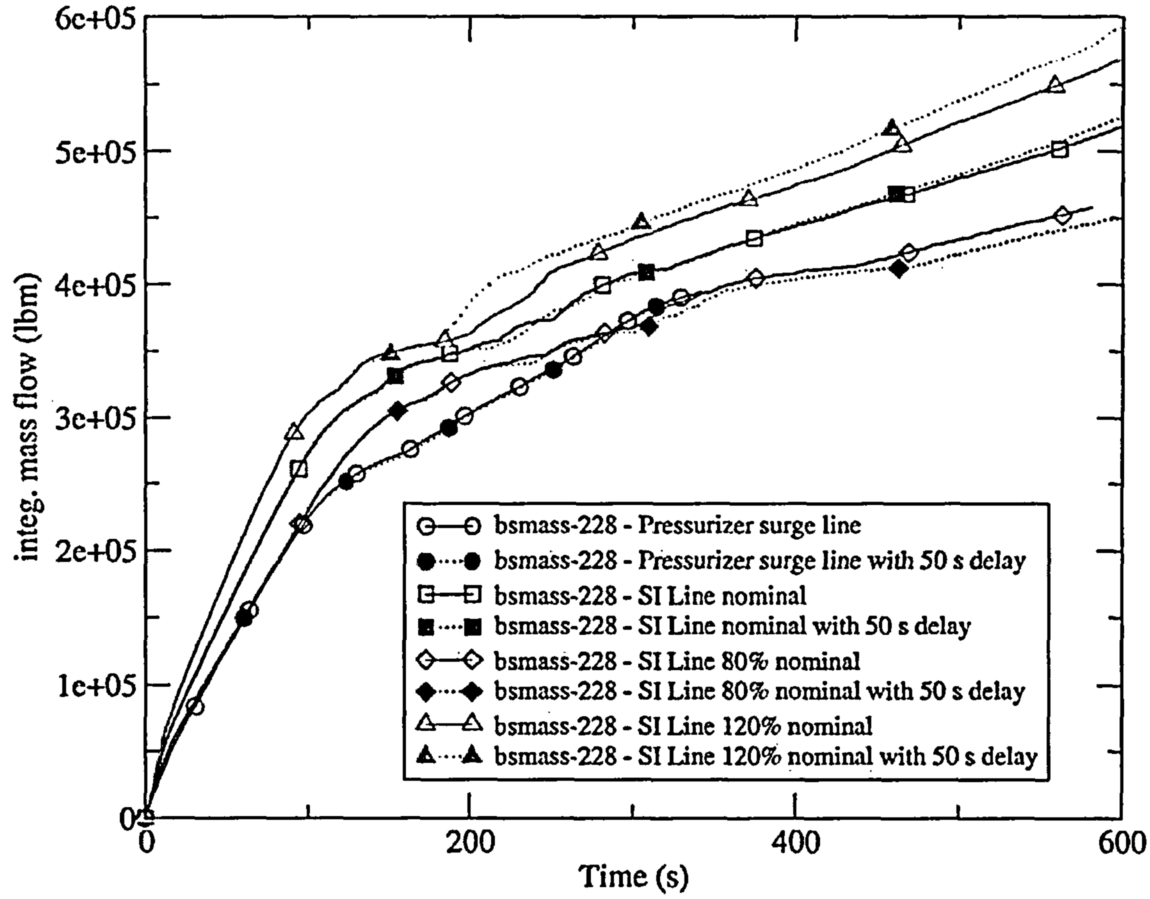


Figure 3-13. Comparison of Integrated Break Mass Flow Rates for Pressurizer Surge Line and Safety Injection Line Breaks

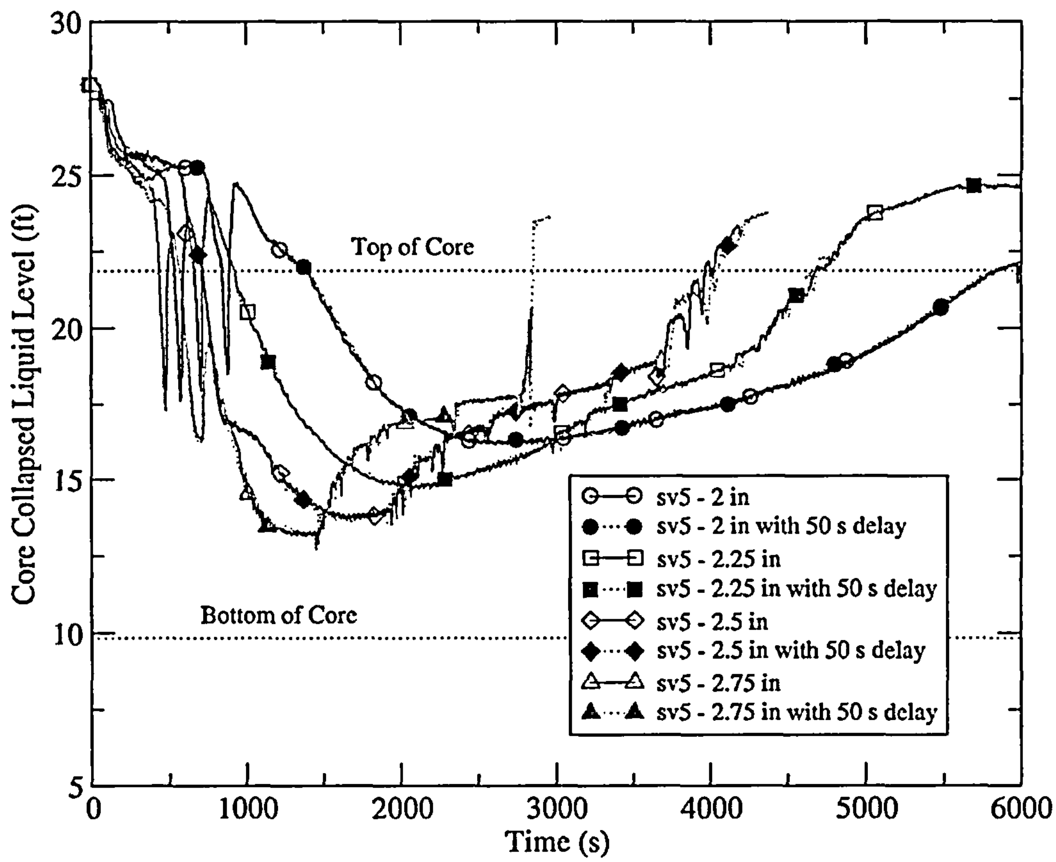


Figure 3-14. Comparison of Core Collapsed Levels for Cold Leg Breaks Smaller than 3 Inches

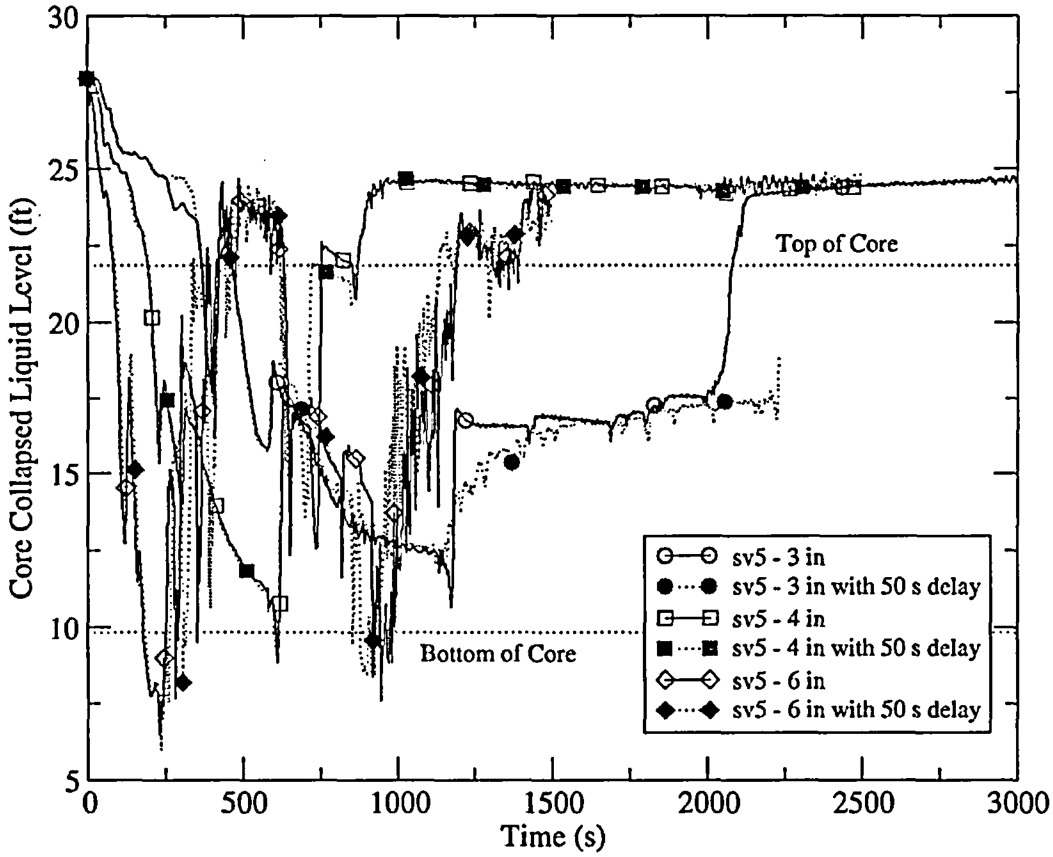


Figure 3-15. Comparison of Core Collapsed Levels for Cold Leg Breaks of 3 Inches and Larger

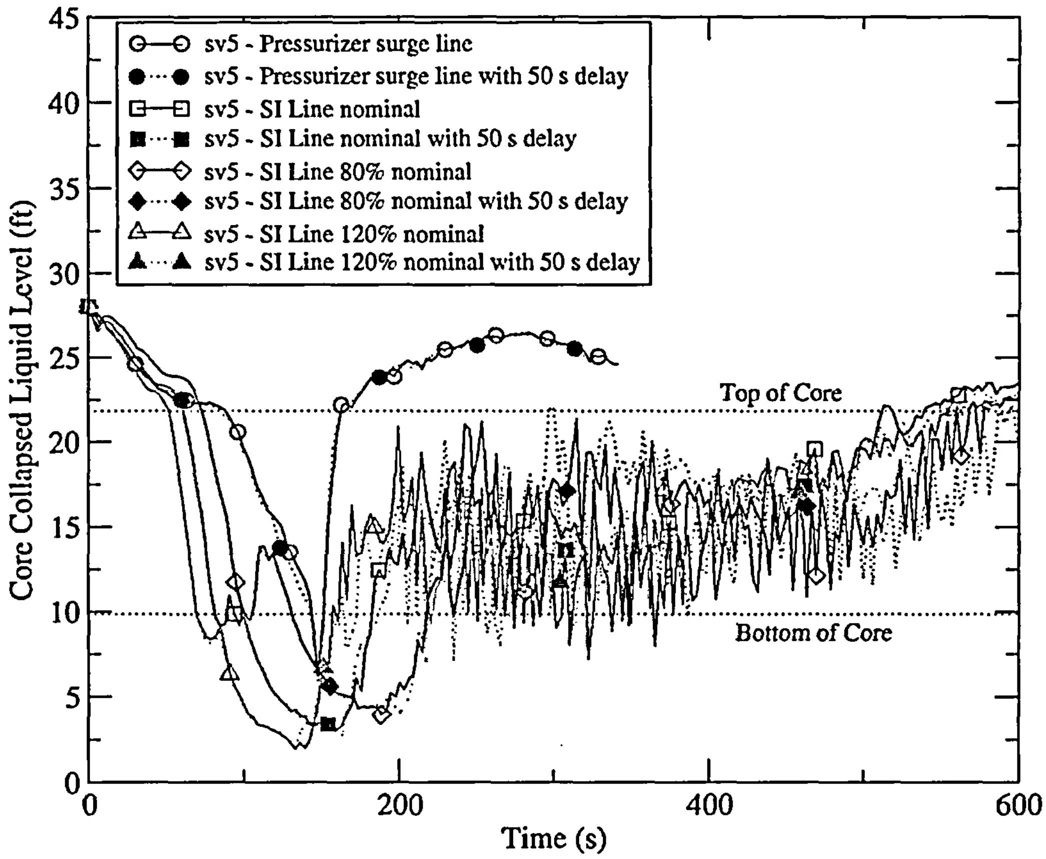


Figure 3-16. Comparison of Core Collapsed Levels for Pressurizer Surge Line and Safety Injection Line Breaks

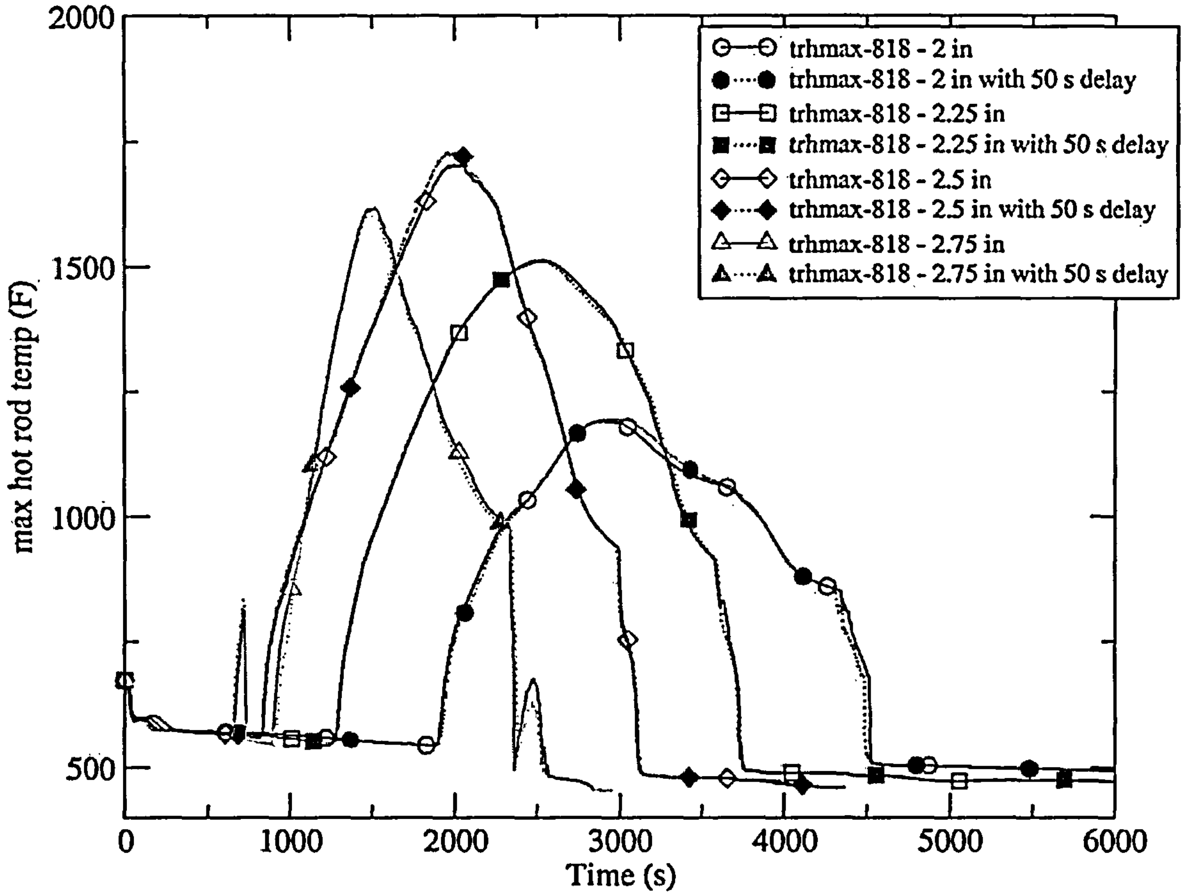


Figure 3-17. Comparison of Maximum Hot Rod PCTs for Cold Leg Breaks Smaller than 3 Inches

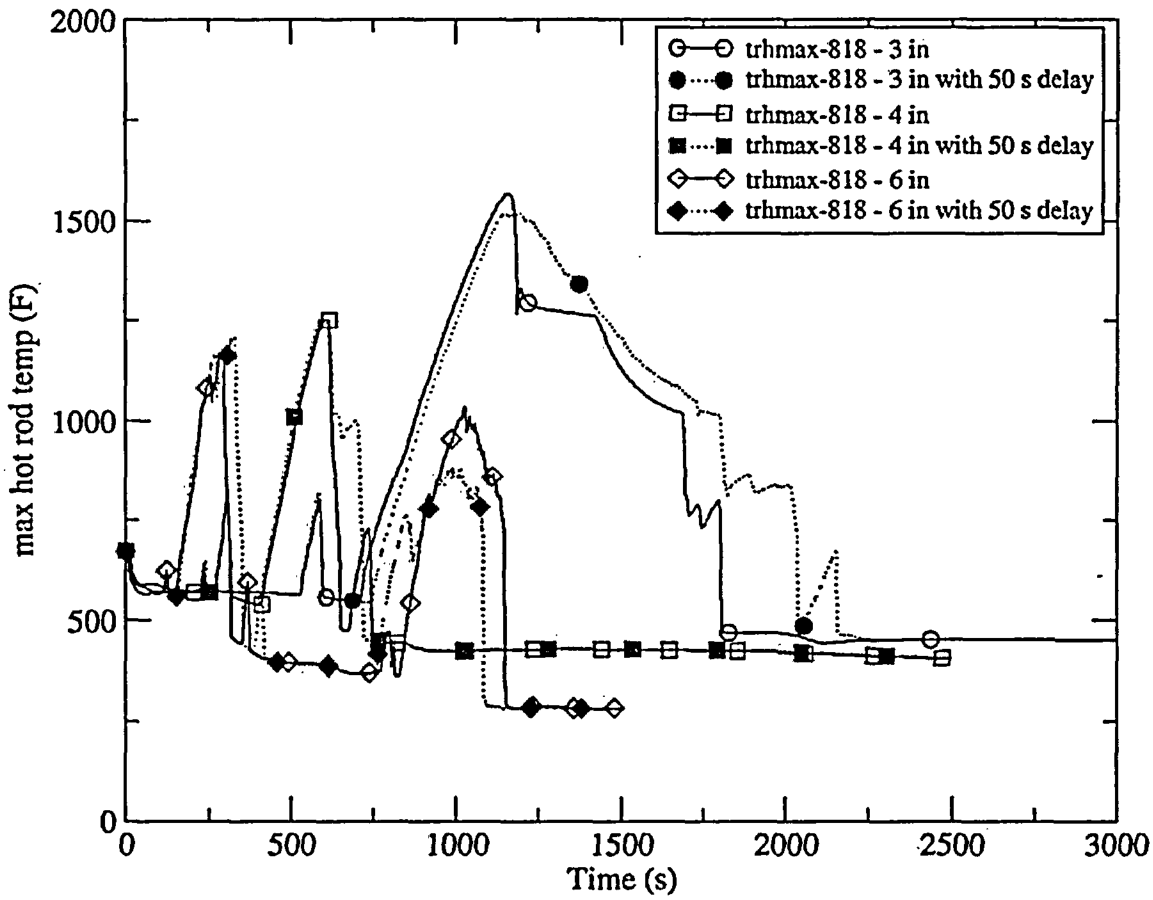


Figure 3-18. Comparison of Maximum Hot Rod PCTs for Cold Leg Breaks of 3 Inches and Larger

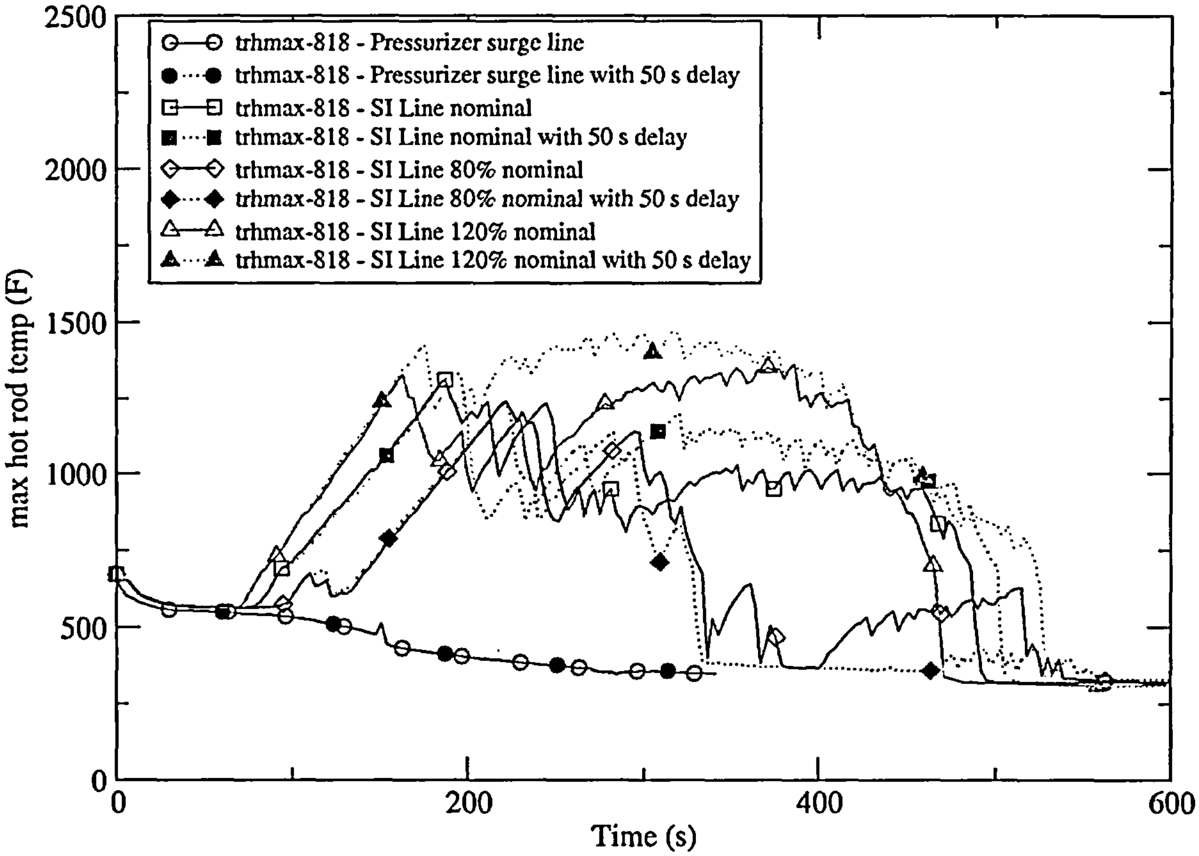


Figure 3-19. Comparison of Maximum Hot Rod PCTs for Pressurizer Surge Line and Safety Injection Line Breaks

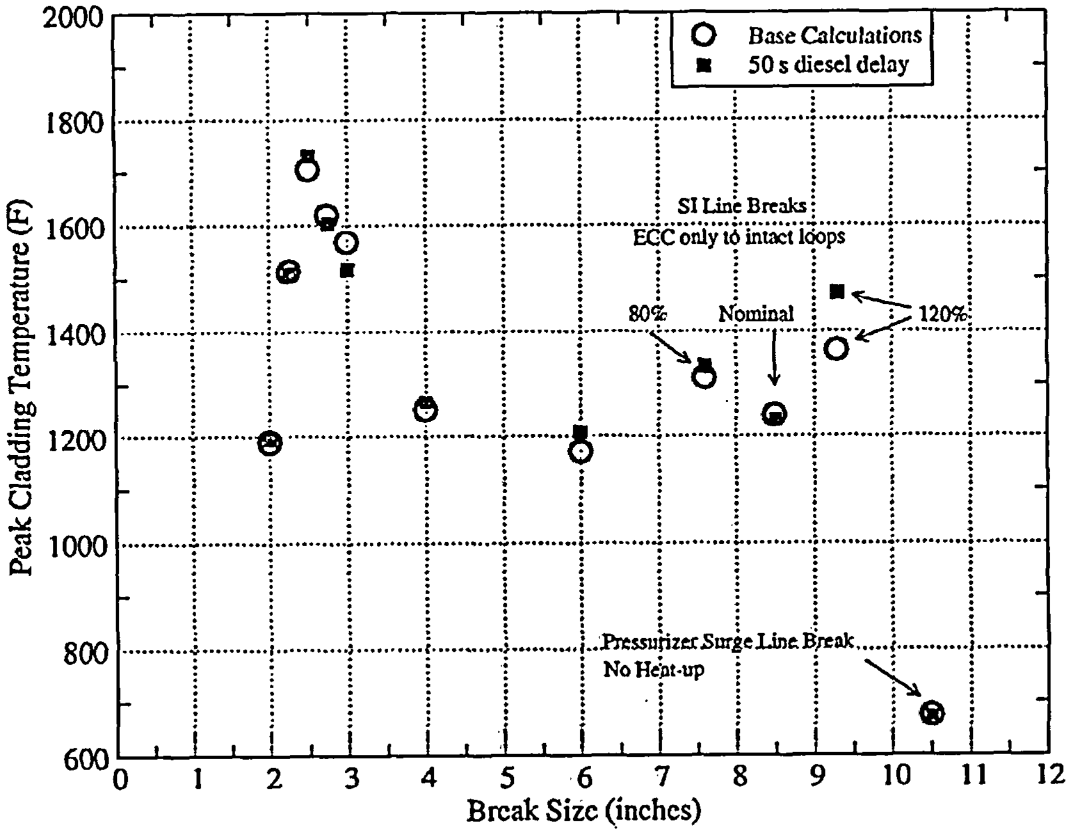


Figure 3-20. PCT Comparison for Simulated SBLOCAs

4.0 CONCLUSIONS

TRACE calculations were performed for a variety of small break loss-of-coolant accidents using conditions and assumptions consistent with typical FSAR licensing calculations for a Westinghouse three-loop plant. These calculations were repeated with an additional 50-s delay for the startup of the diesel electric generator

In the TRACE simulations, difficulties were encountered in obtaining fuel rod heatups as large as those typically seen in the plant licensing-basis SBLOCA calculations. As described in Section 2.0, the input model was adjusted by increasing the reactor coolant pump stopped-rotor flow resistance and by significantly biasing downward the elevation of the loop seals in two of the coolant loops. Without these adjustments PCTs of only about 1,300 °F could be obtained with TRACE (typical SBLOCA licensing-basis calculations for Westinghouse three-loop plants are on the order of 600 °F higher than this). These adjustments are considered beyond the normal range of modeling variations and were justified for the purpose of evaluating the effects of various plant parameters and modeling assumptions on the PCT. That these adjustments were necessary suggests that they are compensating for some shortcoming in the plant input model or TRACE code, or for significant differences between the TRACE code and the codes employed for the licensing-basis calculations.

The small breaks analyzed were 2, 2.25, 2.5, 2.75, 3, 4, and 6 inch diameter in the cold legs, a double-ended guillotine break in the pressurizer surge line (10.5 inches), and three double-ended guillotine breaks in a safety injection line: nominal area (8.5 inches), 80% of nominal, and 120% of nominal. The analysis showed the 2.5 inch cold leg break produced the highest PCT. The PCT for the 2.5 inch cold leg break was 1,704 °F using current licensing basis assumptions and 1,730 °F when an additional 50-s delay for the startup of the diesel generator is assumed. The typical licensing-basis limiting SBLOCA is a cold leg break of about 2-in diameter.

Fuel rod clad temperature excursions were mitigated by flow from the HPI, accumulator and LPI systems. The rod temperature excursion in the 2 inch cold leg break calculation was mitigated by the HPI system alone. For break sizes greater than 2.5 inches, two rod heatup periods were observed; one just prior to loop seal clearing and the other just before the initiation of accumulator injection (for the 6 inch cold leg break calculation, the second rod heatup period occurred after the accumulators had emptied but before LPI flow had started).

The safety injection line breaks showed a relatively long rod heatup period with a later rod quench because one-third of the total ECCS coolant (HPI, accumulators and LPI) was assumed to be spilled directly to the containment.

The additional 50-s delay in the startup of the diesel driven electric generator did not have a significant effect on the results for any of the transients simulated. Because 1,730 °F was the highest calculated cladding temperature, fuel rod heating due to cladding oxiation (metal-water reaction) was not a significant effect.

5.0 REFERENCES

1. *TRACE V4.160 User's Manual, Volume 1: Input Specification*, to be published.
2. J. F. Lime, *Deck Conversion of the H. B. Robinson-2 Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2*, Los Alamos National Laboratory, LA-CP-96-0109, May 1996.
3. C. D. Fletcher, et al., *TRACE Large Break Loss of Coolant Accident Analysis Using Various H. B. Robinson Plant Modeling Assumptions*, Information Systems Laboratories, Inc., ISL-NSAD-TR-04-09 (DRAFT), July 2004.
4. *TRACE V4.160 User's Manual, Volume 2: Input Specification*, to be published.