

**Technical Evaluation of Diesel Generator Start Time  
and Containment Spray Actuation in Support of  
Proposed 10 CFR 50.46a Rulemaking**

**Division of Systems Analysis and Regulatory Effectiveness  
Office of Nuclear Regulatory Research**

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## ABSTRACT

The staff is considering a risk-informed option for 10 CFR 50.46 that will redefine the maximum break size that must be analyzed as part of the design basis for a nuclear power plant and is expected to eliminate the requirement to analyze double-ended guillotine breaks of large primary pipes as design basis accidents (DBAs). The current 10 CFR 50.46 requires that emergency core cooling system (ECCS) performance be analyzed for a broad range of postulated accidents at different locations and of different sizes up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (RCS). Industry representatives have suggested that risk informing the 10 CFR 50.46 rule may allow the plant to be optimized such that overall safety is improved. Two examples of potential safety improvements associated with break size redefinition are a) improved emergency diesel generator (EDG) reliability resulting from the elimination of the need for a 10 second start time, and b) a delay or elimination of automatic containment spray actuations for LOCA, which preserves refueling water storage tank (RWST) inventory for core cooling and delays the need to depend on water from the containment sump where debris can have a detrimental effect on coolant flows and downstream equipment.

This report presents a series of calculations using the TRACE, TRACE/CONTAIN, and RELAP5 codes to investigate the effect of relaxing diesel start time requirements on thermal-hydraulic safety margins for a Westinghouse designed pressurized water reactor (PWR) for small and intermediate loss of coolant accidents (LOCA). In addition, calculations were performed to estimate the delay in time when the RWST inventory would be depleted if the containment spray system does not automatically actuate on high containment pressure. These calculations are intended to evaluate the feasibility of relaxed diesel start up requirements and the potential for making risk-informed revisions to containment spray actuation set points.

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## 1.0 INTRODUCTION

The staff is considering a risk-informed option for 10 CFR 50.46 that will redefine the maximum break size that must be analyzed as part of the design basis for a nuclear power plant. The current 10 CFR 50.46 requires that emergency core cooling system (ECCS) performance be analyzed for a broad range of postulated accidents of different sizes and locations in order to provide assurance that the most severe loss of coolant accident (LOCA) has been considered. Loss of coolant accidents are defined as those hypothetical accidents that result in the loss of reactor coolant in excess of the capability of the make up system. This includes pipe breaks up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. For most pressurized water reactors, the LOCAs that pose the greatest challenge to the ECCS and containment spray system (CSS) are the double-ended guillotine breaks of the hot and cold leg pipes.

A risk-informed option for 10 CFR 50.46 is expected to eliminate the requirement to analyze double-ended guillotine breaks of large primary pipes as design basis accidents (DBAs). This in turn, will provide licensees with new flexibility to optimize their plants. Industry representatives have suggested that a risk-informed reduction in the maximum break size required in a safety analysis may allow the plant to be optimized such that overall safety is improved. Two examples of potential safety improvements associated with break size redefinition are [1]:

- a. Improved emergency diesel generator (EDG) reliability resulting from the elimination of the need for a 10 second start time, and
- b. A delay or elimination of automatic containment spray actuations for LOCA.

Emergency diesel generators (EDGs) provide power when there is a loss of offsite power in order for several engineered safety features to be available. The EDGs provide power for safety injection pumps, auxiliary feedwater pumps, and the containment spray and fan coolers. To assure that power is available for these systems in the event of an accident, plant Technical Specifications generally require that an EDG be capable of starting within 10 seconds. Periodic testing is performed to demonstrate that the generator meets this requirement. This periodic testing is harsh and may contribute to a reduction in the reliability of that component.

Emergency diesel reliability is expected to improve as a result of the reduction of the wear and tear that occurs in the testing that is performed to insure a 10 second start up time. There may also be an increase in EDG reliability due to a reduction in the need to perform troubleshooting and maintenance to achieve the 10 second start time. Increases in safety margin resulting from relaxed EDG start up requirements would be quantified using plant specific PRA models.

The containment spray system is actuated either automatically by a high containment pressure signal, or is actuated manually by the operator. Borated water from the Refueling Water Storage Tank (RWST) supplies the containment spray system during the initial injection period. Later in time, when the system has entered recirculation mode, the containment sprays are supplied by the containment sump. A delay or elimination of the need for containment sprays preserves RWST inventory for core cooling and delays the need to

depend on water from the containment sump where debris can have a detrimental effect on coolant flows and downstream equipment.

This report presents a series of calculations investigating the effect of relaxing diesel start time requirements on thermal-hydraulic safety margins for a Westinghouse designed pressurized water reactor (PWR). The TRACE, TRACE/CONTAIN and RELAP5 codes were used to assess the effect of delayed diesel start times on margin to 10 CFR 50.46 fuel acceptance criteria for small and intermediate loss of coolant accidents (LOCA). In addition, calculations were performed to estimate the delay in time when the RWST inventory would be depleted if the containment spray system does not automatically actuate on high containment pressure. These calculations, in conjunction with work being performed by the industry, are intended to evaluate the feasibility of relaxed diesel start up requirements and the potential for making risk-informed revisions to containment spray actuation set points.

The focus of this study is on small and intermediate LOCA break sizes, as the expected revision to 10 CFR 50.46 will eliminate large double-ended guillotine breaks from the design basis of most PWRs. The effect of delaying the EDG start up time on mitigation of large break LOCAs has not been addressed in this report, nor does this report consider other modifications, such as power and peaking factor increases, which are likely to accompany applications that make use of a risk-informed option to 10 CFR 50.46.

## 1.1 Background Information

### 1.1.1 Effect of Delayed EDG Start

There have been very few studies performed to investigate the effect of emergency diesel generator start up delay on thermal-hydraulic safety margin. Westinghouse and EPRI performed large break LOCA calculations using a special version of WCOBRA/TRAC [2] for a typical 4-loop Westinghouse PWR. Results of that study found that diesel start up delays as long as 53 seconds could be tolerated without peak cladding temperatures in the subject plant exceeding 2200 degrees F. Containment calculations were also performed as part of the Westinghouse / EPRI study. Containment pressures remained well below the containment design pressure, but equipment qualification considerations limited the diesel start delay time to 30 seconds.

### 1.1.2 Risk-Informed Break Sizes

The NRC developed pipe break frequencies as a function of break size using an expert opinion elicitation process for degradation-related pipe breaks in typical BWR and PWR reactor coolant systems [3]. A transition break size (TBS) was established using these pipe break frequencies. The TBS was adjusted to account for other significant contributing factors that were not explicitly addressed in the expert elicitation process in the following manner:

(1) Break sizes for each reactor type (PWR and BWR) were selected that corresponded to a break frequency of  $1.0E-05$  per reactor year from the expert elicitation results.

(2) The NRC then considered uncertainty in the elicitation process, other potential mechanisms

that could cause pipe failure that were not explicitly considered in the expert elicitation process, and the higher susceptibility to rupture/failure of specific piping in the reactor coolant system (RCS).

The proposed rule would divide the current spectrum of LOCA break sizes into two regions. The division between the two regions is determined by a “transition” break size (TBS). The first region includes small breaks up to and including the TBS. The second region includes breaks larger than the TBS up to and including the double-ended guillotine break (DEGB) of the largest reactor coolant system pipe.

The staff determined that the appropriate TBS should be the cross-sectional area of the largest attached pipe for the respective RCS pipe. For pressurized water reactors (PWRs), it would be the pressurizer surge line size for the hot leg piping and the SI/accumulator line size for the cold leg piping. The TBS will vary from plant to plant depending on the specific piping system design and can vary from about 8 inches to 14 inches. For boiling water reactors (BWRs), the area of the TBS break is the cross-sectional flow area of the larger of either the feedwater or the residual heat removal piping inside primary containment. The BWR TBS corresponds to a pipe diameter of approximately 20 inches.

## 1.2 Scope of Present Investigation

This study investigated a Westinghouse Standard 412 PWR with a large dry containment. This is a 4-loop unit assumed to be uprated to 3565 Mwt and have safety grade charging pumps and safety grade fan coolers. Section 2.0 provides details on design and initial operating conditions assumed in the analysis for both the reactor coolant system and containment.

Calculations were performed with both TRACE/CONTAIN and RELAP5 for the following cases:

1. A set of small break LOCA simulations ranging in equivalent break diameter from 2-inches to 6-inches in order to identify the limiting small break size. The breaks were assumed to occur at the bottom of the cold leg.
2. A complete severance of the accumulator/safety injection (SI) line, using the actual pipe flow area. ECC flow from the broken accumulator/SI line was assumed to spill to containment.
3. Accumulator/SI line break as above, but increasing and decreasing the flow area 20% relative to the actual pipe flow area.
4. A double-ended guillotine break of the pressurizer surge line (PSL) using the actual flow area of the surge line.

The set of calculations were selected to investigate the range of break sizes that would be potentially limiting if the transition break size (TBS) was defined as being the break to the primary system resulting from severance of the largest connecting branch lines. On conventional Westinghouse plants these are typically a 10-inch Schedule 140 accumulator/SI

line connected to one of the cold legs, and the 14-inch Schedule 160 pressurizer surge line. Increasing the area by plus and minus 20% is equivalent to varying the discharge coefficient and accounting for uncertainty in the break flow rate.

The series of 2-inch to 6-inch cold leg small break LOCA calculations, and the accumulator/SI line breaks were simulated to investigate the effect of delaying the EDG start up time. The base case condition assumed for each was a 10 second EDG start up time, which is the delay time currently used in licensing calculations. A second set of calculations were performed assuming a 60 second delay. This value was based on the assumption that a 60 second start up would be sufficient to improve EDG reliability and that reliability data were available to support plant specific PRA models.

The PSL break was simulated because it represents the largest branch line connected to the primary coolant loop, and hot leg breaks are sometimes limiting with respect to containment pressure. This break is expected to therefore be more challenging to containment pressurization and CSS actuation. The pressurizer surge line break results are not expected to be sensitive to EDG start time. Hot leg breaks provide a low resistance path to the break and do not result in loss of safety injection directly to the break. All four accumulators provide ECC and are sufficient to keep the core covered without reliance on early actuation of pumped safety injection. In addition, peak pressures produced by hot leg breaks occur during blowdown, which is over by the time safety injection actuates even with a 10 second delay time. Thus, the pressurizer surge line breaks served to provide limiting mass and energy releases for the transition breaks and to verify a relatively large margin to core uncover.

Because these calculations involve an evaluation of both primary system and containment performance, a version of TRACE coupled with the CONTAIN code referred to as TRACE/CONTAIN, was employed. This allowed core uncover and cladding heatup as well as the containment pressure response to be determined in the same simulation. This also enabled there to be a feedback between containment pressure and core water level for breaks near the transition break size, where it may have an effect. The coupled TRACE/CONTAIN version was also used for the 4-inch through 6-inch small break LOCA simulations, although containment feedback does not have any impact on the RCS for these break sizes.

The same base case calculations and sensitivities to EDG start up delay were also performed using RELAP5/MOD3 Version 3.3ek. Since TRACE is still undergoing development and assessment, calculations with a more mature code previously used for small break LOCA analysis were considered desirable. Therefore, a RELAP5 model of the same Westinghouse Standard 412 plant was developed and simulated with RELAP5. However, rather than coupling RELAP5 to a containment code, the RELAP5 plant model included components meant to mimic the containment and thus approximate its response. As noted previously, containment feedback was not expected to be significant and this approach was considered sufficient for the purposes of this study. An RWST was also explicitly modeled for the RELAP5 calculations to estimate the effect of containment sprays on water inventory.

The remaining sections of this report describe the TRACE/CONTAIN and RELAP5 calculations. Section 2 discusses the plant and containment simulated while Sections 3 and 4 present the calculations and their results. Conclusions and recommendations for further study are summarized in Section 5.

## 2.0 REFERENCE PLANT DESCRIPTION

The Reference Plant is a Westinghouse Standard (STD) 412 pressurized water reactor (PWR) design, with a licensed core power output of 3565 MWt (100% power) and a large dry containment. The nuclear steam supply system (NSSS) consists of a pressurized water reactor and a four-loop reactor coolant system (RCS). The reference plant is similar to the group of Westinghouse designed 4-loop plants including Vogel, Seabrook, and Callaway. Each has a 12 foot core with 17x17 fuel assembly design with a base power of 3411 MWt. Later, some of these units opted for a stretch power of 3565 MWt.

Details on parameter values for the core, RCS, reactor protection setpoint and delay times, auxiliary feedwater system (AFW), steam generators, and safety injection (SI) system (both pumped SI and accumulators) can be found in Reference 4.

### 2.1 Reactor Coolant System

The RCS of the newer class STD 412 design has 4 loops, with a Model 93A-1 reactor coolant pump (RCP) and a Model F U-tube steam generator (SG) in each loop. The Model F SG uses thermally treated Alloy 600 tubes having a nominal wall thickness of 0.041 inches. Initial water mass is 93,134 lbm. Each SG is assumed to have 10% of its tubes plugged. One of the loops contains the primary system pressurizer. The nominal RCS pressure at full power is ~ 15.5 MPa (2250 psia), and the thermal design flow per loop, consistent with 10% SG tube plugging, is 93,600 gpm [4].

The reactor vessel contains the reactor core, core barrel, core support structures, and control rod and instrument component structures. Water from the SGs is pumped through the cold legs by the RCPs to the reactor vessel (RV) inlet nozzles, downcomer, RV lower plenum, lower core support structure, and to the core. The flow continues upwards through the core, exiting the RV via the outlet nozzles, and continues through the hot legs into the SGs again. The reactor core is made up of 50,952 Zirlo clad fuel rods within 193 fuel assemblies (264 rods/assembly). The core active height is 12 ft. (3.66 m). Table 2.1 summarizes the Reference Plant conditions.

### 2.2 Containment

The Reference plant was assumed to have a large dry containment. Figure 2.1 depicts a typical large dry containment, showing the large and relatively open containment space within the reactor building. The open space above the operation deck, surrounding the steam generators, represents approximately 80% of the free volume within the building. The containment concept relies on the passive attribute of a large volumetric and thermal capacitance within the enclosed free volume, and complemented by the active heat removal systems such as containment fan coolers and sprays. Typically, the containment design pressure is between 45-60 psig. The containment free volume is 2.5 million cubic feet, and the initial temperature for the gas volume and heat structures are set to 120 F, with an initial pressure of 14.7 psia. Table 2.2 summarizes the containment heat structures.

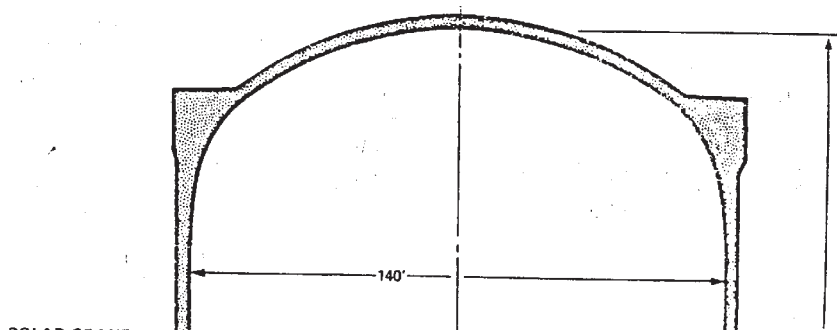


Figure 2.1 Typical large dry containment showing the openness of the large free volume above the lower compartments.

Table 2.1 Reference Plant Parameters

Licensed Core Power (MWt)	3565
Fuel Type	17X17
Total Core Peaking Factor, FQ	2.60
Hot Channel Enthalpy Rise, $F\Delta H$	1.70
Hot Channel Ave Power Factor	1.52
Thermal Design Flow (gpm/loop)	93600
Total Core Bypass (%)	8.4
Pressurizer Pressure (psia)	2250
Reactor Trip Set Point (psia)	1937
Steam Generator Tube Plugging (%)	10
Max. Accumulator Water Temperature (F)	130
Accumulator Water Volume (ft <sup>3</sup> )	900
Min. Accumulator Gas Pressure (psia)	611.3



Table 2.2 Containment Heat Structures

HS #	Description	Surface Area (m <sup>2</sup> )	Total "Slab" Thickness (m)	Material Type *
1	Containment Cylinder	5465.3	0.50	1,2,5,6,7
2	Containment Dome	2883.0	0.50	1,2,5,6,7
3	Unlined Concrete	6118.1	0.50	7
4	Stainless-Stl Lined Conc.	668.9	0.50	4,6,7
5	Galvanized-Stl Lined Conc.	620.7	0.4	3,5,6,7
6	Stainless Steel	1733.1	0.0055	4
7	Galvanized Steel	6361.6	0.00242	3,5
8	Carbon Steel w/o paint	164.4	0.0064	5
9	Painted Carbon Steel	1250.0	0.00276	1,2,5
10	Painted Carbon Steel	7814.9	0.00572	1,2,5
11	Painted Carbon Steel	3761.2	0.00923	1,2,5
12	Painted Carbon Steel	2258.9	0.01863	1,2,5
13	Painted Carbon Steel	1108.9	0.03476	1,2,5
14	Painted Carbon Steel	725.4	0.08568	1,2,5
15	Carbon-Stl Lined Conc.	600.7	0.50	1,2,5,6,7

\* 1 Epoxy; 2 Paint; 3 Zinc; 4 Stainless-Steel; 5 Carbon-Steel; 6 Air; 7 Concrete  
 (Note: left side of the slab is inside containment; right side is assumed adiabatic)

## 3.0 ANALYSIS AND SENSITIVITY STUDIES USING TRACE/CONTAIN

### 3.1 TRACE/CONTAIN Executable Version

Calculations presented in this Section were performed using the TRACE thermal-hydraulics code coupled with the CONTAIN containment code. The TRAC/RELAP Advanced Computational Engine (TRACE - formerly called TRAC-M) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission for analyzing transient and steady-state neutronic-thermal-hydraulic behavior in light water reactors. It is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into a single modernized computational tool. CONTAIN is the code used by the NRC to perform containment thermal-hydraulic calculations for PWRs and BWRs.

The Exterior Communications Interface (ECI) logic in TRACE was used to couple TRACE Version 4.160 with CONTAIN. Coupling the two codes enables feedback between the containment atmosphere and the reactor coolant system. Several modifications were made to TRACE to improve time step control problems observed in check valves in earlier versions of the code and to maintain tighter outer iteration convergence criteria.

The TRACE code is the subject of continuing development and assessment. While a complete assessment of the version used in this study is not currently available, a significant amount of work on versions slightly older than this have been completed. Of particular interest to small and intermediate break LOCA analyses are the simulations of several ROSA-IV small and intermediate break tests [5] and BETHSY International Standard Problem 27 (ISP-27) [6], and two LOFT small break simulations [7].

The assessment against ROSA simulated six different small break tests, and included Test SB-CL-14, which is a 10% cold leg break. The ROSA simulations were performed with TRACE version 4.150 and in general were found to be in reasonable agreement with the experimental data. Simulation of BETHSY ISP-27 and LOFT Tests L3-1 and L3-7 were performed with an older version of TRACE. The simulations of those tests were found to be in good agreement with data. References 5 through 7 should be consulted for specifics on the TRACE calculations and agreement with data.

### 3.2 Plant Modeling and Nodalization

The TRACE model of the reactor coolant system is based on a Westinghouse 4-loop PWR model originally developed for the NRC by Los Alamos National Laboratory (LANL). The original input model, developed for TRAC-P, was converted to the TRACE input format. This TRACE input model was then revised as part of this project in order to make the model a close representation of the 412 Standard Reference Plant described in Section 2.

#### 3.2.1 VESSEL Nodalization

The reactor pressure vessel, reactor internals, and core region were modeled using a TRACE VESSEL component with a detailed nodalization. The reactor vessel model contained 26 axial levels, 4 radial rings and 8 equal theta sectors. Figure 3.1 shows a cross section of one of the axial levels. The 8 theta nodalization places a single hot leg or cold leg in a given sector. The vent valves that are shown in Figure 3.1 can be used to model hot leg nozzle gap leakage. However, this flow path was not activated in the sensitivity studies discussed in this report. Figure 3.2 shows the axial nodalization of the vessel and core in the original USPWR input deck.

Axial levels 1 through 3 represent the lower plenum up to the lower core support plate. The heated core was modeled using the 2 innermost radial rings and was divided into 14 equal length axial levels. Each sector of the inner ring represented 2673 fuel rods (approximately 10 assemblies) while each sector in the second ring represented 3696 fuel rods, or 14 assemblies. The innermost radial ring represents a total of 81 assemblies. Hot rods at the maximum linear heat generation rate (LHGR) were placed in all 16 core sectors. Compared to the original TRAC-P model developed by Los Alamos National Laboratory (LANL), this is an increase in the total number of axial levels used to represent the core. This was done to provide a more accurate tracking of the two-phase level when the level drops below the upper core plate.

The barrel/baffle and downcomer are represented by the third and fourth rings. The core region in the vessel was renodalized from 5 axial levels in the original LANL model to 14 axial levels in the TRACE model used in these calculations. Figure 3.3 shows the axial nodalization of the vessel and core in the revised input deck. This was done in order to better track the two phase fluid level in the core. All other axial levels remain unchanged. Thus, the upper plenum had 6 axial levels, and the upper head 1 level.

The guide tubes were not explicitly modeled in the current input model. The flow area of the guide tubes was modeled in the vessel level connecting the upper plenum to the upper head.

### 3.2.2 Loop Nodalization

Each of the four loops of the reactor coolant system were modeled explicitly in the TRACE input description. The nodalization for Loop 1, which includes the pressurizer, is shown in Figure 3.4. With the exception of the pressurizer, this nodalization is preserved in the other three loops. (Figures for the other loops are not included.) Table 3.1 summarizes the loop nodalization.

For this investigation, the nodalization in the loop seals was also increased to track the water level more accurately during loop seal clearing. The steam generators are composed of PIPE and TEE components in order to simulate recirculation on the secondary side, with input revised to model the Westinghouse Model F steam generators.

### 3.2.3 Containment Model

The CONTAIN code was used to model the large dry containment of the Reference plant described in section 2.2. Because of the relative openness of this design, containment performance evaluations have usually been performed using a single compartment model for the containment building. The containment free volume is 2.5 million cubic feet, and the initial temperature for the gas volume and heat structures are set to 120 F, with an initial pressure of 14.7 psia. Table 2.2 lists the structural heat sinks. Safety grade containment fan coolers as well

as the containment spray system are also modeled. The fan coolers were assumed to activate on a 44.7 psia signal with a 60 second delay.

### 3.2.4 Steady State Simulation

A steady-state simulation of the Reference Plant was run to help insure the plant was modeled adequately. The steady state calculations were run for 500 seconds and a reasonable steady state was reached, and TRACE conditions were compared to those of the Reference Plant. The Reference Plant conditions were obtained from Reference 4.

Table 3.2 summarizes the steady-state comparison. In general, the code predicted conditions are in fair agreement with the intended conditions of the Reference Plant. The predicted loop temperatures ( $T_{ave}$ ) is 12 F less than the desired Reference Plant value. The maximum difference between loop flows was approximately 1.5 %. The deviations between the TRACE values and the intended values are not expected have a large impact on conclusions based on the sensitivity studies performed in this report.

### 3.3 Transient Simulation Modeling

Three different break locations were considered in this study. For small cold leg breaks (2 to 6 inch equivalent break diameter), the break was assumed to be at the bottom of the Loop 1 cold leg. This was modeled using two TEE components in order to model both the SI connection (TEE 22) and the break connection (TEE 23). The nodalization of the broken cold leg is shown in Figure 3.4. The break node was modeled as a node that was approximately 0.05 meters in length (approximately the thickness of the pipe wall) for all breaks. For the small cold leg breaks, accumulator/SI coolant was injected to the broken cold leg.

The accumulator/SI line break was assumed to be at the actual branch line location, which is on the side of the cold leg. The branch line was also assumed to connect at a 45 degree angle relative to axis of the cold leg, with the safety injection flow directed towards the reactor vessel. In TRACE, this is done by specifying the TEE side pipe orientation so that fluid momentum is appropriately directed. For the accumulator/SI line break, the phase separation model for side connected branch lines was activated. This allows for vapor pull-through to be accounted for from stratified levels in the cold leg. For this break, accumulator/SI coolant was spilled to containment.

The pressurizer surge line break was modeled as shown in Figure 3.5 except that the surge line was changed to 1 cell 0.05 meters long. At the time of the break, a BREAK component was attached to the hot leg (TEE 24). Control logic accounted for the rapid drain of the pressurizer tank and actuation of reactor trip and SI signals on the pressurizer low pressure setpoint. All accumulator/SI coolant was injected for this case, with no spillage to containment.

All calculations remained at constant steady-state power until the reactor trip. Following trip the ANS 71 decay heat standard plus 20% was assumed.

## 3.4 LOCA Calculations

### 3.4.1 Introduction

Three types of breaks were considered in this study; breaks ranging in size from 2 to 6 inches in diameter, breaks the severe the accumulator/SI line, and the pressurizer surge line break. Each of these types of breaks exhibits unique behavior. This sub-section discusses each of these breaks, and presents results of the TRACE/CONTAIN simulations. Results for all cases performed assuming a 10 second diesel start up time are listed in Table 3.3. Results for all cases performed assuming a 60 second diesel start up time are listed in Table 3.4.

### 3.4.2 Small Cold Leg Break Calculations

Breaks ranging from 2 to 6 inches in equivalent diameter are considered small break LOCAs in this report. In general, these breaks depressurize slowly, and the reactor coolant system undergoes a top-down drain. Recovery is a strong function of the pumped safety injection system, although for the larger break sizes in this range the accumulators also play a role. The worst small break in terms of peak cladding temperature is generally the largest break that is not recovered by accumulator injection. Important reactor system parameters that affect small break LOCA peak cladding temperatures are the secondary side relief valve setpoint, the accumulator pressure, and the shape of the high pressure injection delivery curve as a function of reactor pressure. Small break LOCAs can be divided into two phases. In the first phase of the accident the break flow is single phase liquid and the steam generator acts as a heat sink for the primary system. The primary system will depressurize to a pressure slightly above the pressure of the steam generator relief valve setpoint. Heat is generated by the reactor core and remove from the system through the break and heat transfer to the steam generators. The pressure difference between the primary and the secondary depends on the heat transfer to the secondary ( therefore the amount of steam generator tube plugging) and the break flow. Loop seal clearing and the core level depression associated with it occurs during this phase of the accident. The second phase of the accident occurs when the break uncovers and the break flow is dominated by steam flow. This allows the system to depressurize to pressures below the steam generator. The core boils off and uncovers and the fuel rods heat up. The core level starts to recover when the pressure in the system becomes low enough so that the safety injection and accumulator flow exceeds the break flow. The details of each specific small break are described in the paragraphs that follow. Results of the 2-, 3-, 4-, and 6-inch small break LOCA calculations are shown in Figures 3.6, 3.7, and 3.10 through 3.17. The small break calculations describe in this sub-section assumed a 10 and 60 second start up time for the emergency diesel generator.

The 2 inch break had no significant heatup, as seen in Figure 3.8. The high pressure safety injection was sufficient to prevent core uncover and a temperature excursion. Accumulators did not inject for this break size.

Figures 3.12 and 3.13 show the core collapsed liquid level and peak cladding temperature for the 3-inch cold leg break. Note that core liquid volume fraction in Figure 3.12 (and other figures that show this parameter) is equivalent to the core collapsed liquid level. For this break size a cladding heatup was predicted, with a 913 F peak cladding temperature occurring shortly before 2000 seconds. The core temperature excursion was terminated by high pressure injection. The

accumulators did begin to inject for this case, but well after the minimum in core inventory had been reached.

The highest peak cladding temperature for any of the calculations performed with TRACE for this study was obtained for the 4-inch cold leg break. The core collapsed liquid level and peak cladding temperature for the 4-inch break are shown in Figures 3.14 and 3.15. The 4 inch break had a peak cladding temperature of 1335 F at approximately 1400 seconds assuming a 10 second EDG delay (see Table 3.5). Core uncover for the 4 inch break was terminated by accumulator injection. A comparison of Figures 3.13 and 3.15 suggest that the worst break size is between 3- and 4- inches since the 4 inch break is recovered by accumulator injection. The worst break size gives approximately 100 F higher PCT than the 4 inch break.

The 6 inch cold leg break depressurizes rapidly, and accumulator injection begins early in the transient. The core uncover is deeper for this break size in comparison to the 2-, 3-, and 4- inch breaks, but because of the rapid accumulator injection, the cladding heatup is terminated quickly. The 6-inch break was predicted to have a peak cladding temperature of 714 F for the 10 second delay and 736 F for the 60 second delay at approximately 450 seconds. The core collapsed liquid level and peak cladding temperature are shown in Figures 3.16 and 3.17 for the 6-inch break.

#### 3.4.3 Accumulator/SI Line Break Calculations

The second set of calculations performed using TRACE/CONTAIN investigated rupture of the accumulator/SI line. This line connects to the cold leg at the top/side of the main pipe and not at the bottom, so for this break the actual orientation of the break location was accounted for. The accumulator/SI line break assumed a guillotine break of the branch line where it connects to the cold leg. Therefore none of the safety injection water from the pumps or the accumulator in the broken loop was injected into the reactor coolant system, resulting in a reduction in the injected flow.

The Reference Plant accumulator/SI line is a 10-inch Schedule 140 pipe. The inner diameter for this pipe is 8.75 inches. Breaks of this size range cause the system to rapidly depressurize and the core heatup is terminated by accumulator injection. Cases were also run with 80% and 120% of the accumulator/SI line break area. The upper plenum pressure is shown in Figure 3.8 and the integrated break flow is shown in Figure 3.9. The core collapsed liquid level and peak cladding temperature for TRACE/CONTAIN calculations are shown in Figures 3.18 through 3.23. There was no heatup in the SI line break calculation. There was no significant difference between the 10 second and 60 second diesel delay time cases for the SI line break. There was no heatup in these cases and all three behave in a similar fashion.

#### 3.4.4 Pressurizer Surge Line Break Calculation

The pressurizer surge line is a 14-inch Schedule 180 pipe. This pipe has an inner diameter of 11.188 inches, and represents the largest break size that may be evaluated as part of the design basis for a risk-informed option for 10CFR50.46.. There are several fundamental differences in behavior between the pressurizer surge line break and the cold leg breaks. First, the system depressurizes far more rapidly with the pressurizer surge line break compared to the other cold leg breaks investigated because of the larger break area. The depressurization rate is also increased because the large volume of saturated water in the pressurizer is no longer available



to the reactor coolant system. Because of the close proximity of the break to the pressurizer, the pressurizer tank drains and depressurizes very quickly, and the pressurizer low pressure setpoint for reactor trip occurs immediately following the break.

The pressurizer surge line break provides an easy path for steam to escape from the reactor coolant system, thus does not need to clear the loop seals to vent the vapor generated in the core to the break. Breaks of this size range cause the system to rapidly depressurize and the core heatup is terminated by accumulator injection.

The upper plenum pressure is shown in Figure 3.8 and the integrated break flow is shown in Figure 3.9. The core collapsed liquid level and peak cladding temperature for TRACE/CONTAIN calculations are shown in Figures 3.24 through 3.25. The pressurizer surge line break calculation was predicted to have initial heatup due to CHF occurring immediately after the break initiation near the point of peak linear heat generation. Recovery of this temperature excursion was due to the blowdown cooling after the reactor trip. The rapid depressurization of the system allows voids to immediately form in the upper region of the core and the core goes into CHF near the peak power location of the hot rods. A second very brief period of heatup occurred later in the transient, but the increase was only about 150 degrees F and the temperature did not significantly exceed the initial cladding temperature at the start of the transient. This heatup was terminated by accumulator injection. The PCT during the blowdown was approximately 920 F.

### 3.5 Emergency Diesel Generator Delay

A series of calculations were also performed using the TRACE/CONTAIN model assuming a 60 second diesel generator start up time. Results of the 60 second EDG delay time cases are summarized in Tables 3.4 and 3.5, and are discussed below:

There was no significant difference between the 10 second and 60 second diesel delay time cases for the 2 inch break. Two loop seals cleared in both cases.

The 3 inch break had a peak cladding temperature of 913 F for the 10 second delay case and 1063 F for the 60 second delay case (see Figure 3.13). Three loop seals cleared in the 10 second delay case and two loop seals cleared in the 60 second delay case. Cases that clear a smaller number of loop seals have a higher PCT because less water is moved to the vessel causing the core to uncover at a higher decay heat level. There is also a higher flow resistance to the break in this case which causes a deeper core depression.

The 4 inch break had a peak cladding temperature of 1335 F for the 10 second delay case and 1167 F for the 60 second delay case. The 4 inch break had the largest difference between the 10 second delay time and the 60 second delay time calculation. The 10 second delay case actually has a higher peak temperature than the 60 second delay case. This result can be explained by the difference in loop seal clearing behavior between the two calculations. Three loop seals in the 10 second delay case. Four loop seals cleared in the 60 second delay case. The additional loop seal that cleared in the 60 second delay case caused additional water to go to the vessel and delayed the time of core uncover compared to the 10 second delay case.

The counter-intuitive sensitivity found in the 4-inch break with different EDG delay times is partly due to the uncertainties in predicting phenomena that occur during the loop seal clearing period.

The TRACE/CONTAIN model uses an explicit nodalization of each loop. When the loop seals first begin to clear, the liquid levels in the four loops are nearly identical and as a result all loops may begin to starting venting simultaneously. Venting of steam through the loop seals allows steam produced in the core to exit through the break. As steam passes through the loop seals, the inner vessel pressure decreases rapidly, and there is an increase in the inner vessel mixture level. Loop seal venting process is a relatively violent period compared to other periods occurring during a small break, and large loop to loop oscillations can occur as the liquid previously in the bottom of the loop seals is redistributed between the loops, the inner vessel, and the break. This rapid redistribution of inventory, along with uncertainty in the number of loop seals that remain clear, represents a major source of uncertainty in prediction of a small break transient. Reference 8 discusses these uncertainties in more detail.

Thus, the sensitivity observed in the TRACE/CONTAIN 4-inch calculations must be interpreted as demonstration of the relatively small effect of EDG delay time rather than an indication that a longer delay may lead to lower PCTs. The change in PCT is within the uncertainty that is expected due to the non-deterministic nature of the loop seal clearing process.

There was no significant difference between the 10 second and 60 second diesel delay time cases for the 6 inch break, nor were there significant difference between the 10 second and 60 second diesel delay time cases for the SI line breaks. The pressurizer surge line break had a small boiloff heatup for the 60 second delay time compared to no heatup for the 10 second delay time. Four loop seals cleared in all cases for the 6 inch SI line and PSL breaks.

The results of these calculations showed the Reference Plant to be insensitive to diesel generator startup delay over this range. The insensitivity attributed to the small integrated ECCS flow that occurs over the delay period compared to the integrated break flow over the same period. The 4 inch break is the only case with the largest difference between the 10 second delay and the 60 delay. This is due to the difference in loop seal clearing behavior in the two calculations. The differences between the 10 second delay and the 60 second delay cases would be within the uncertainties of a given calculation for all cases if an uncertainty evaluation varying all important variables were performed. For example the safety injection flow for the 4 inch break would be less than 10 percent of the break flow during the 50 additional seconds of no SI flow in the 60 second delay calculation compared to the 10 second delay calculation. This difference is less than the uncertainty in the break flow modeling.

### 3.6 Containment Pressure Response

The coupled TRACE/CONTAIN calculations also provided the containment pressure response.. Coupled TRACE/CONTAIN calculations were performed for the pressurizer surge break, the accumulator/SI line break, and the 6 inch cold leg break. The containment pressure exceeded the containment spray system setpoint of 44.7 psia in all of these calculations, but containment sprays are not needed to remain below the 75 psia design pressure of the containment. The containment pressure responses without sprays activated are shown in Figure 3.26. Peak pressures for the accumulator/SI line break and the small cold leg breaks occur following the initial depressurization at a time when the safety grade fan coolers sufficiently remove decay heat. The



fan coolers are the only available containment heat removal system when sprays are not activated. Figure 3.27 shows the CONTAIN and WOG GOTHIC fan cooler heat removal performance as a function of vapor temperature. The CONTAIN model is a mechanistic model that was set up to match the GOTHIC fan cooler performance curve at one point and can not be made identical to the GOTHIC curves over the whole range.

The time to reach the spray setpoint is listed in Table 3.3. Note that for the 6 inch break it requires over 2000 seconds to reach the spray setpoint and plant specific accident management procedures could change the containment response compared to what was calculated in this case. The significance of the containment pressure reaching the spray setpoint is that the Reactor Water Storage Tank (RWST) will drain at a faster rate if the spray setpoint is reached causing a need to switch over to sump recirculation earlier in time. Significantly delaying or eliminating the switchover to sump recirculation is desirable because it avoid the possibility of sump screen clogging by debris and unqualified coating in containment. The reactor core can uncover and begin to heat up again during the switchover to sump recirculation.

All candidates for transition break size breaks (SI line, PSL ) reach the containment spray setpoint in less than one hour after the break. RWST drain rates and switchover times are analyzed and discussed in more detail in Section 4.

### 3.7 Summary of TRACE/CONTAIN Results

The PCT remained relatively low in all calculations compared to 10 CFR 50.46 acceptance criteria. A summary of the PCT results is shown in Table 3.5. The PCT is designated by N/A in cases where no heatup occurred. A summary of the timing of significant events that occur following the break are also shown in Tables 3.3 and 3.4. The peak containment pressure remained low compared to the design pressure without the benefit of containment sprays.

The sensitivity observed in the TRACE/CONTAIN small calculations must be interpreted as demonstration of the relatively small effect of EDG delay time rather than an indication that a longer delay may lead to lower PCTs. The change in PCT is within the uncertainty that is expected due to the non-deterministic nature of the loop seal clearing process.

The results of the calculations in this section showed the Reference Plant to be insensitive to diesel generator startup delay over the range of break sizes and delay times considered.

Table 3.1 TRACE Loop Nodalization

RCS Region	TRACE Component	Hydraulic Cells (Main Flow path)
Hot Leg	PIPE (Loops 1,3,4) TEE (Loop 2)	6
SG Tube Bundle, Plena	PIPE	18
Pump Suction Piping	PIPE	10
Reactor Coolant Pump	PUMP	2
Cold Leg	TEE	7
SG Secondary	TEE	12
SG Downcomer	TEE	10
Pressurizer	PRIZER	7
Pressurizer Surge Line	TEE Side Leg	2

Table 3.2 Steady State Simulations Results

<u>Parameter</u>	<u>TRACE Value</u>	<u>Reference Plant</u>
Reactor Power (MWt)	3636.3	3636.3
Pressurizer Pressure (psia) (MPa)	2234 15.40	2250 15.51
Thot (EF) (K)	625.	--
Tcold (EF)	562.	--
Tave (EF) (K)	593	588 582
Primary Loop Flow (gpm) (m <sup>3</sup> /sec)	95400 6.02	93600 5.91
Core Bypass (%)	1.02	
Upper Head Bypass (%)	0.20	
Total RCS Inlet Flow Bypass (%)	1.22	8.4
SG Secondary Pressure (psia) (MPa)	994.	
Feedwater Flow per SG (lbm/sec) (kg/sec)	1110 503.5	1126.9 511.2
Steam Flow per SG (lbm/sec) (kg/sec)	1111 503.9	1126.9 511.2
Feedwater Temp. (EF) (K)	436.	448 504.3
SG Tube Plugging (%)	10	10
SI Water Temp. (EF) (K)	120 322	120 322
Accumulator Temp. (EF) (K)	130 327.6	130 327.6
Accumulator Gas Pressure (psia) (MPa)	609.1 4.20	611.3 4.21

Table 3.3 Event Timing 10 second delay time (Time in seconds after the break)

<b><u>Event Break °</u></b>	<b><u>2"</u></b>	<b><u>3"</u></b>	<b><u>4"</u></b>	<b><u>6"</u></b>	<b><u>80% SI</u></b>	<b><u>100 %SI</u></b>	<b><u>120% SI</u></b>	<b><u>Surge Line</u></b>
Start	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor trip signal	26.	12.	7.8	5.0	3.2	3.1	3.0	0.05
SI Signal	39.	21.	14.	9.9	6.0	5.8	5.7	1.8
SI Begins	129.	61.	54.	50.	51.	50.	49.	42.
Spray Signal	N/A	N/A	N/A	2075.	N/A	340.	N/A	106.
Loop seal venting	1230.	593.	327.	180.	110.	90.	75.	47.
Core top uncover	N/A	1470.	855.	410.	N/A	N/A	N/A	N/A
Begin accumulators	N/A	N/A	1335.	427.	218.	172.	141.	120.
PCT occurs	0.0	2070.	1335.	455.	N/A	N/A	N/A	2.0
Core top recovery	N/A	2790.	1995.	480.	N/A	N/A	N/A	N/A

Table 3.4 Event Timing 60 second delay time (Time in seconds after the break)

<b><u>Event Break °</u></b>	<b><u>2"</u></b>	<b><u>3"</u></b>	<b><u>4"</u></b>	<b><u>6"</u></b>	<b><u>80% SI</u></b>	<b><u>100 %SI</u></b>	<b><u>120% SI</u></b>	<b><u>Surge Line</u></b>
Start	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor trip signal	26.	12.	7.8	5.0	3.2	3.1	3.0	0.05
SI Signal	39.	21.	14.	9.9	6.0	5.8	5.7	1.8
SI Begins	179.	111.	104.	100.	51.	50.	49.	92.
Spray Signal	N/A	N/A	N/A	2075.	N/A	340.	N/A	106.
Loop seal venting	1225.	593.	327.	180.	110.	90.	75.	47.
Core top uncover	N/A	1420.	916.	405.	N/A	N/A	N/A	124.
Begin accumulators	N/A	2470.	1555.	431.	218.	178.	141.	118.
PCT occurs	0.0	2177.	1470.	455.	N/A	N/A	N/A	2.0
Core top recovery	N/A	2785.	200.	480.	N/A	N/A	N/A	134.

Table 3.5 Peak Cladding Temperature Summary

<b><u>Break</u></b>	<b><u>2"</u></b>	<b><u>3"</u></b>	<b><u>4"</u></b>	<b><u>6"</u></b>	<b><u>80% SI</u></b>	<b><u>100% SI</u></b>	<b><u>120% SI</u></b>	<b><u>Surge Line</u></b>
10 second delay PCT	N/A	913 F	1335 F	714 F	N/A	N/A	N/A	922 F
DNB PCT	N/A	N/A	N/A	N/A	N/A	N/A	N/A	922 F
Loop Seal PCT	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Boiloff PCT	N/A	913 F	1335 F	714 F	N/A	N/A	N/A	N/A
60 second delay PCT	N/A	1063 F	1167 F	736 F	N/A	N/A	N/A	922 F
DNB PCT	N/A	N/A	N/A	N/A	N/A	N/A	N/A	922 F
Loop Seal PCT	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Boiloff PCT	N/A	1063 F	1167 F	736 F	N/A	N/A	N/A	640 F

Table 3.6 Number of Loop Seals Cleared

<b><u>Break</u></b>	<b><u>2"</u></b>	<b><u>3"</u></b>	<b><u>4"</u></b>	<b><u>6"</u></b>	<b><u>80% SI</u></b>	<b><u>100% SI</u></b>	<b><u>120% SI</u></b>	<b><u>Surge Line</u></b>
10 second delay	2	3	3	4	4	4	4	4
60 second delay	2	2	4	4	4	4	4	4

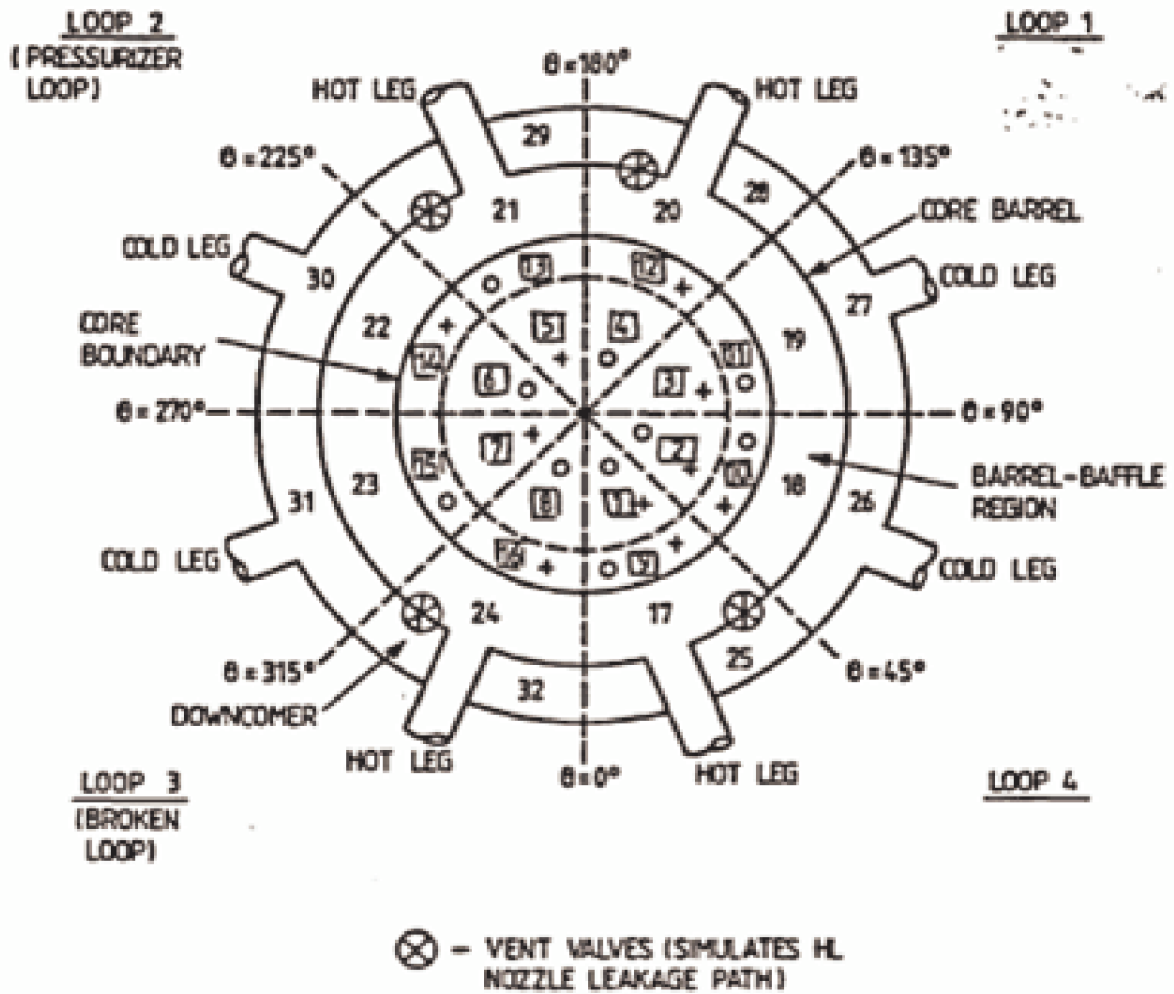


Figure 3.1 Cross section view of USPWR VESSEL component.

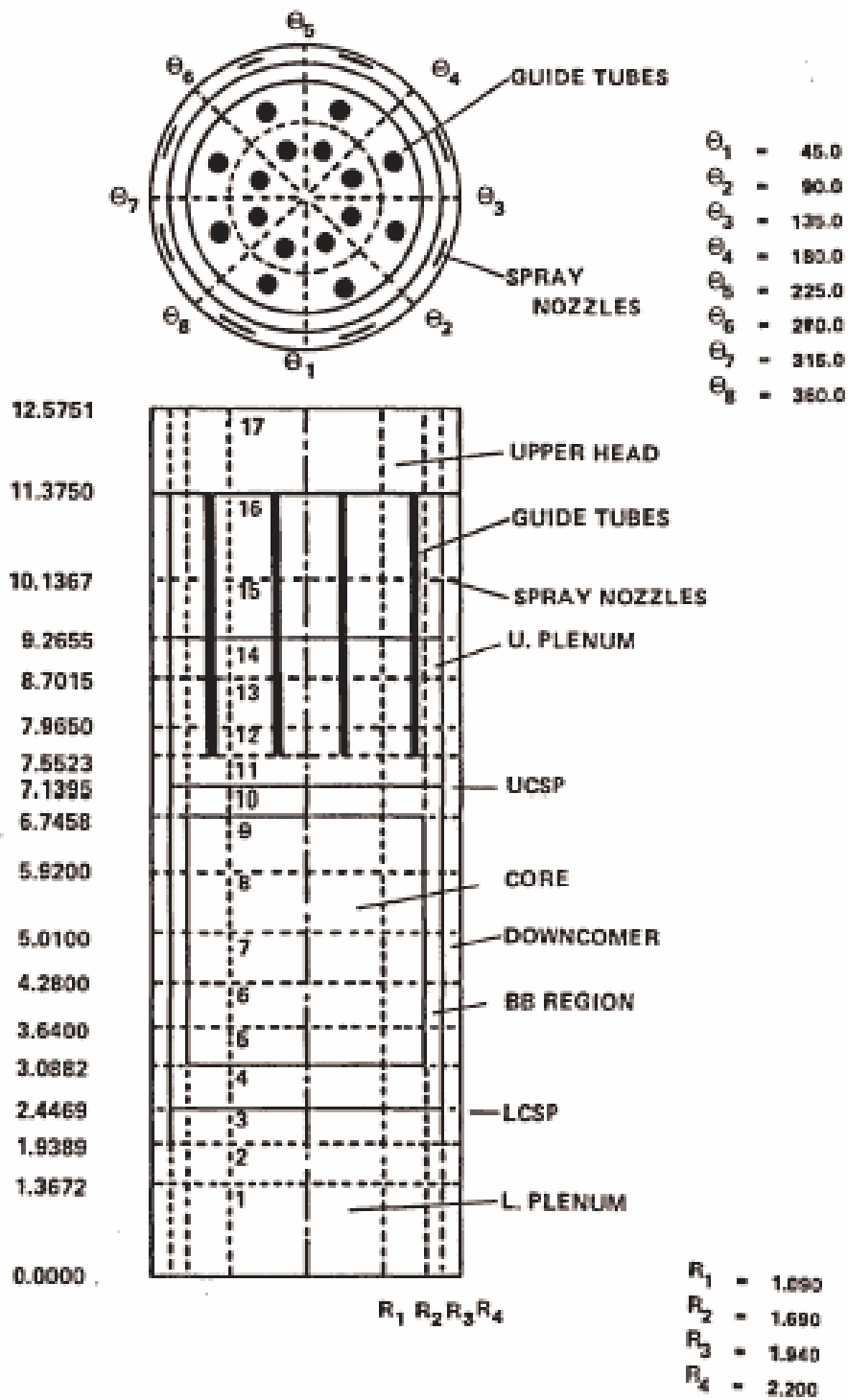
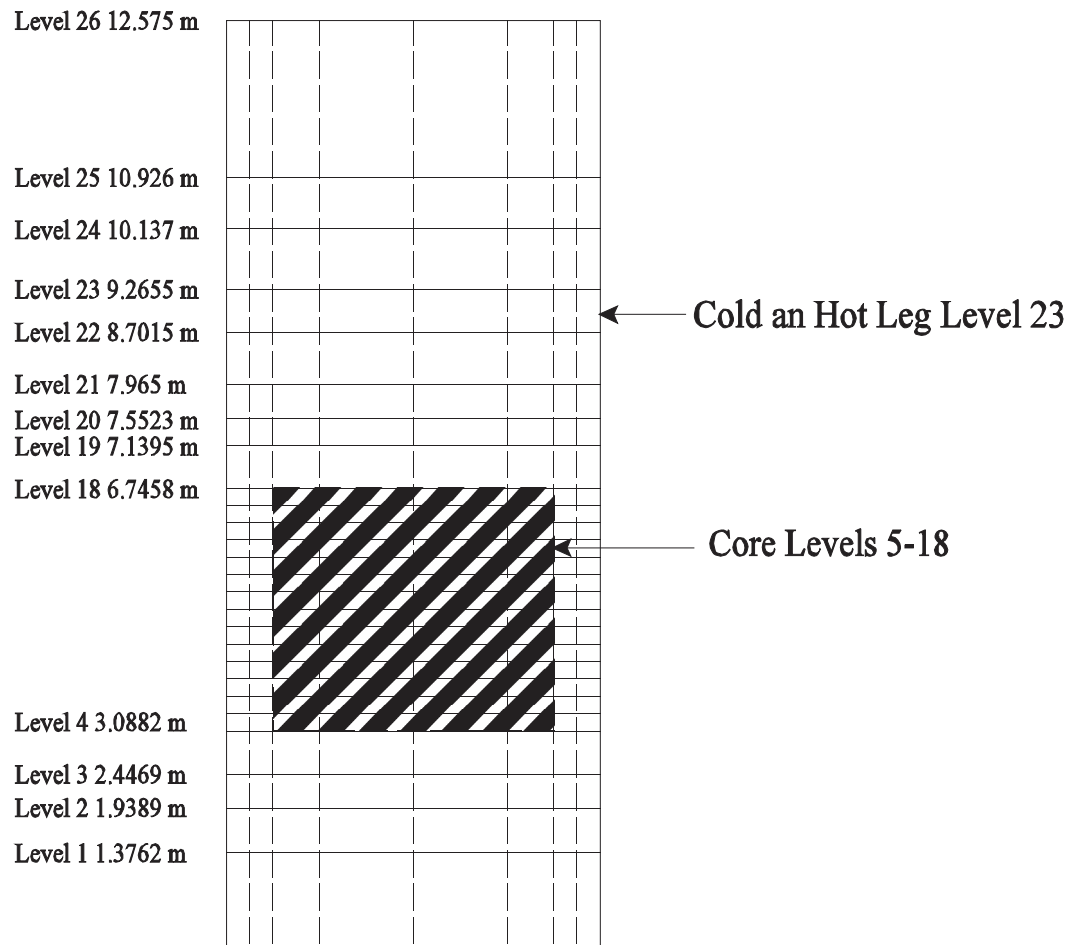
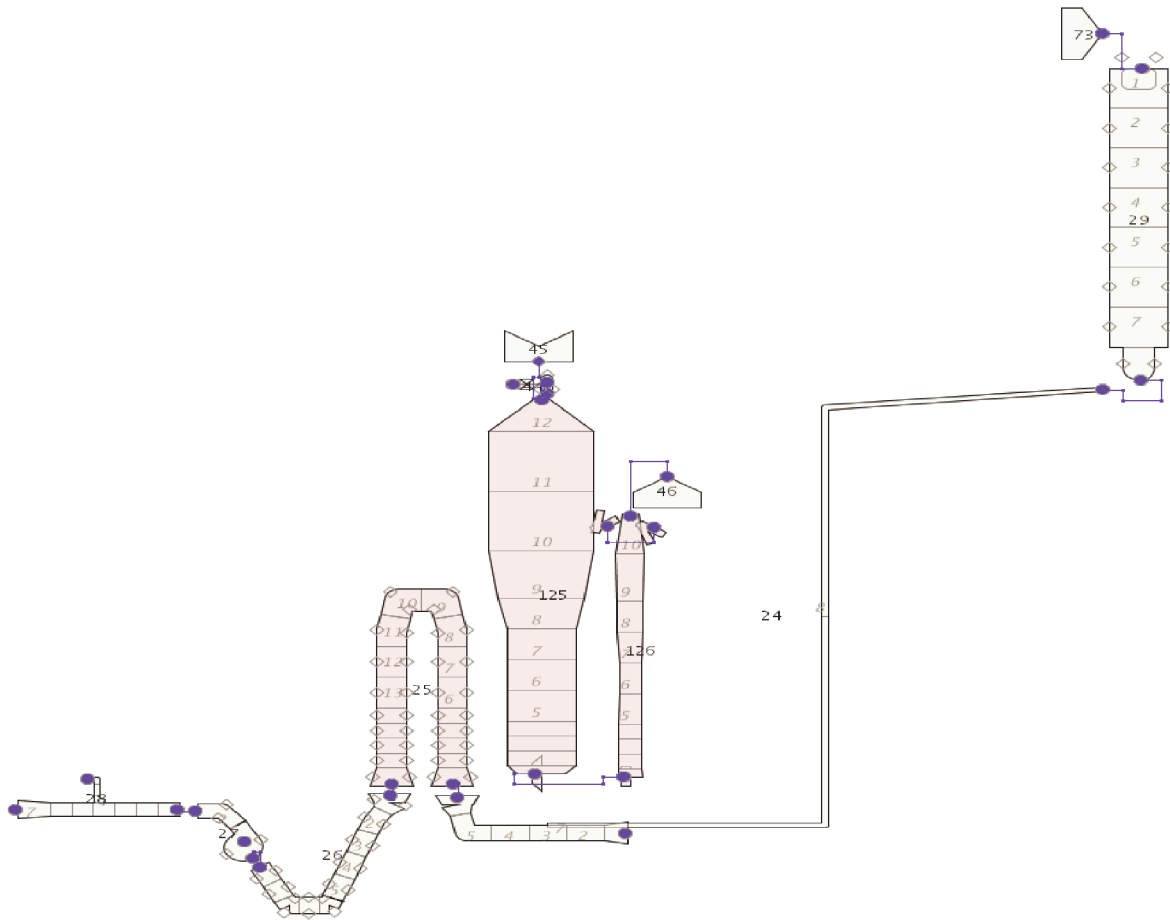


Figure 3.2 Vessel nodalization of original USPWR input deck.

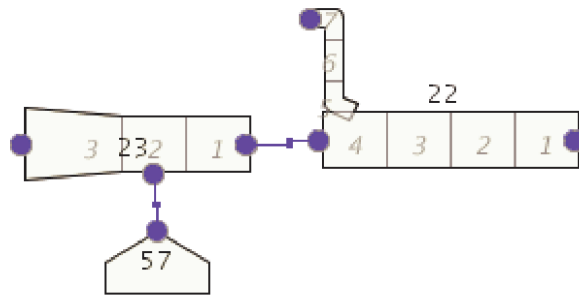


**Figure 3.3** Revised vessel nodalization for LOCA calculations





**Figure 3.4** Loop nodalization for pressurizer loop.



**Figure 3.5** Nodalization of broken cold leg.

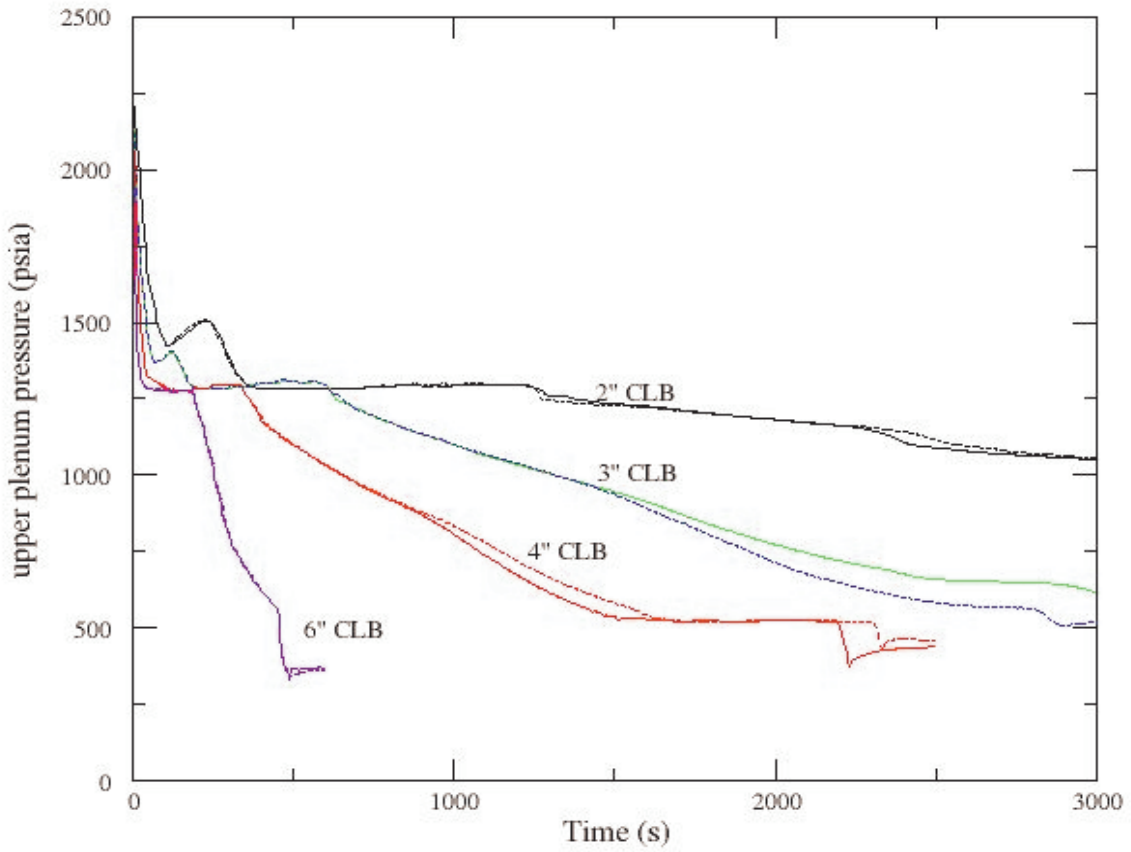
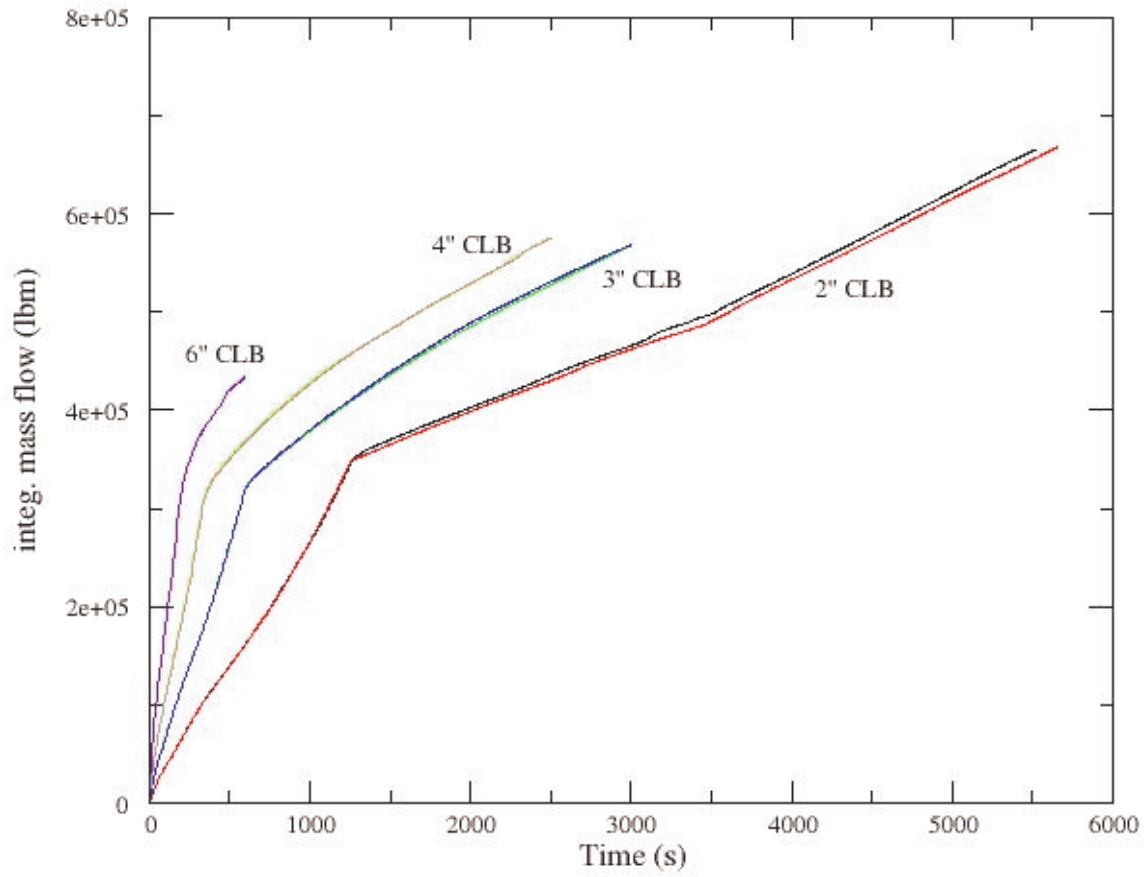
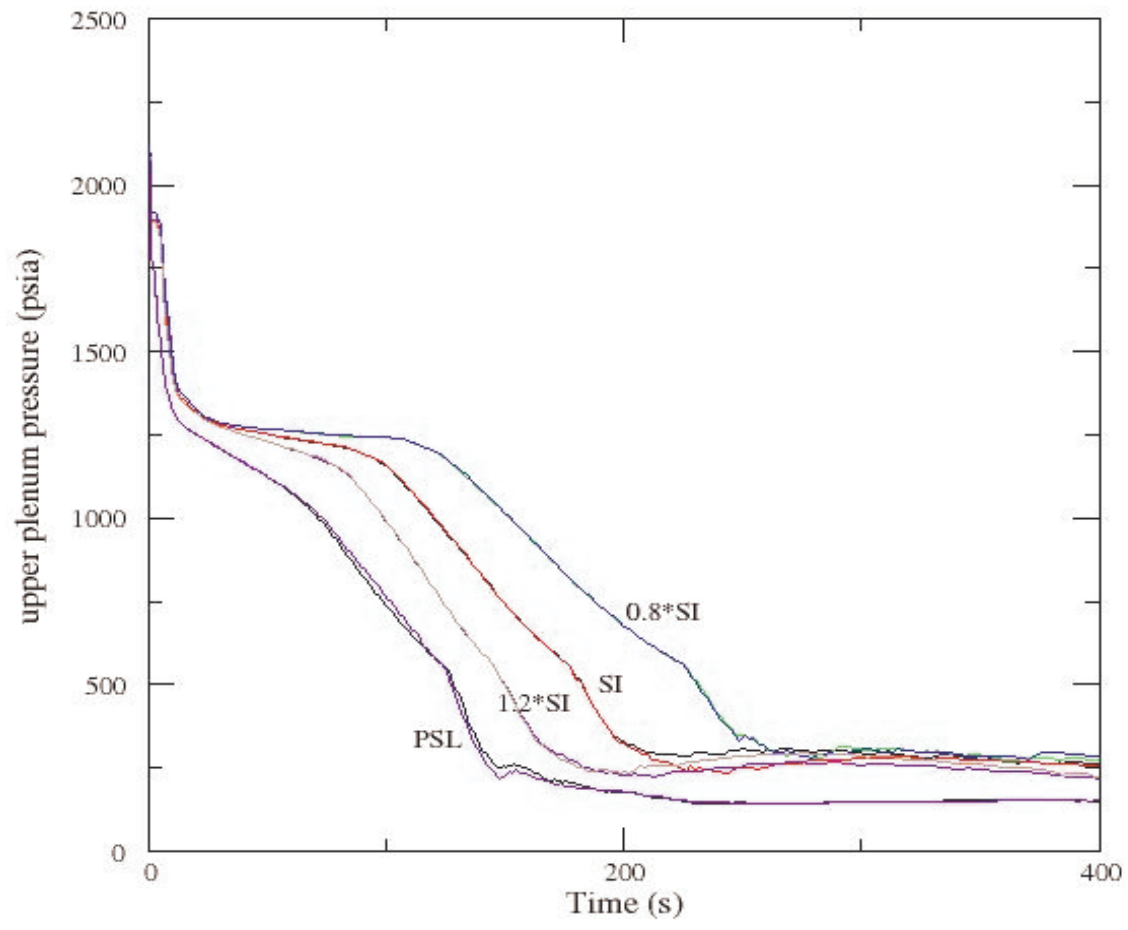


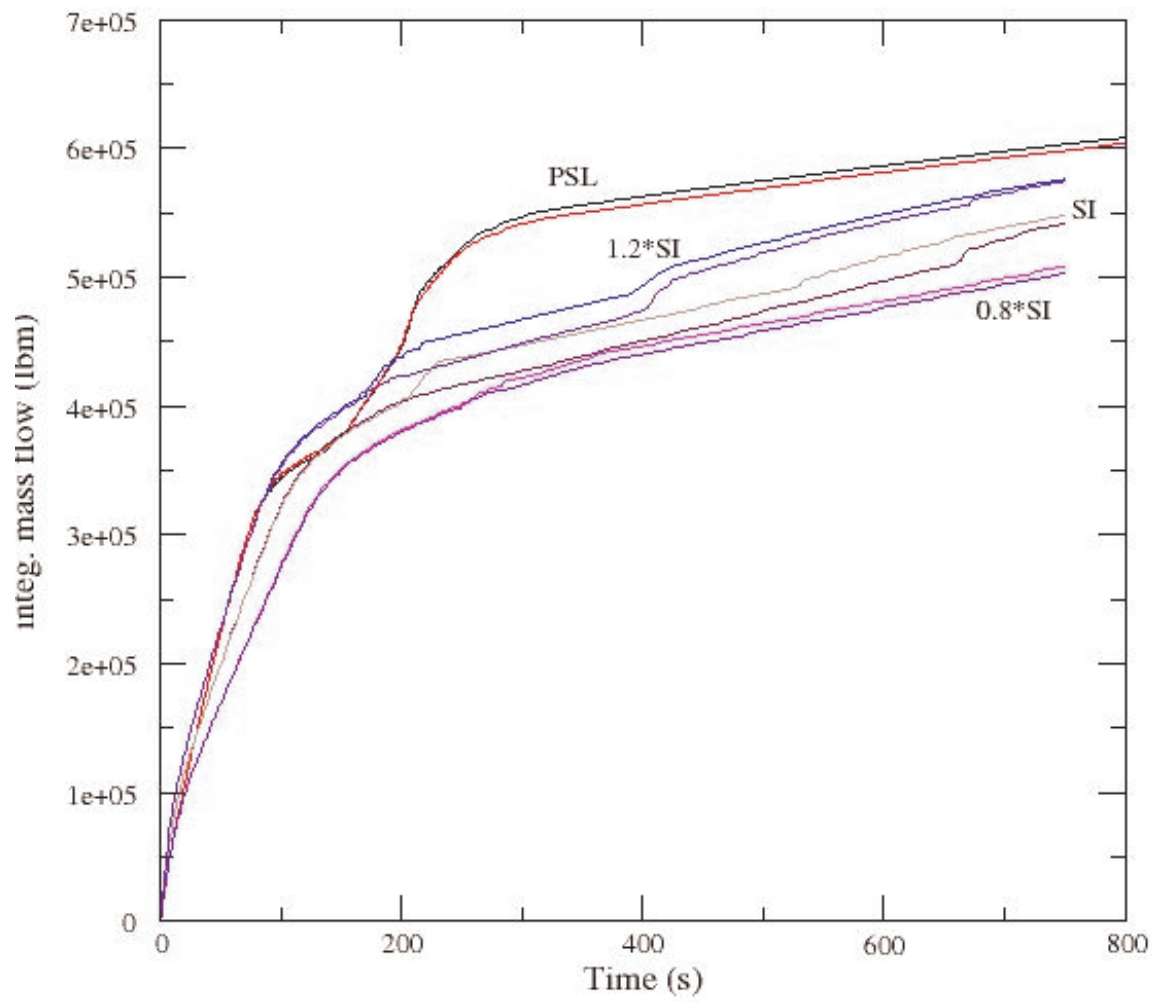
Figure 3.6 Upper plenum pressure for small break calculations.



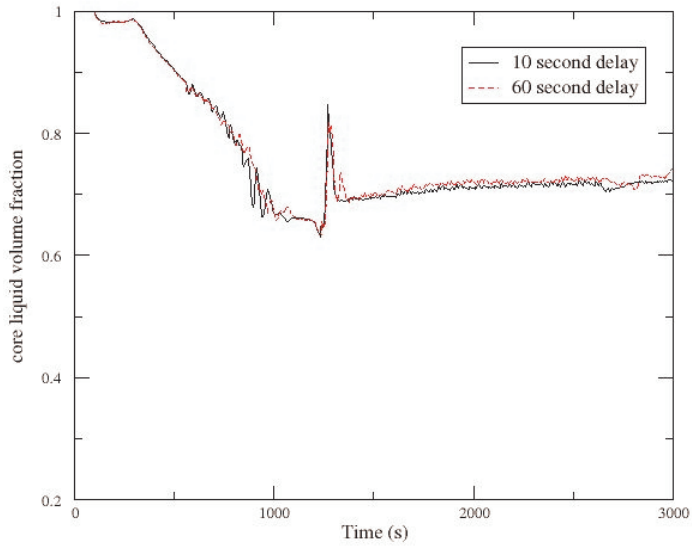
**Figure 3.7** Integrated break flow for small breaks.



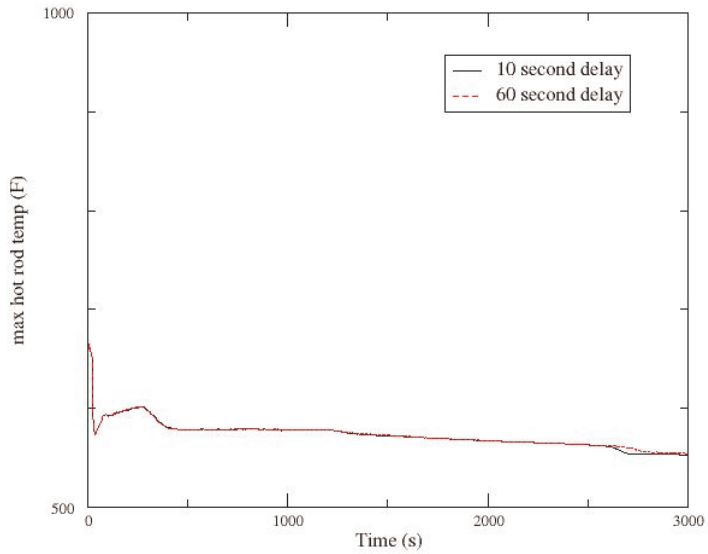
**Figure 3.8** Upper plenum pressure for transition breaks.



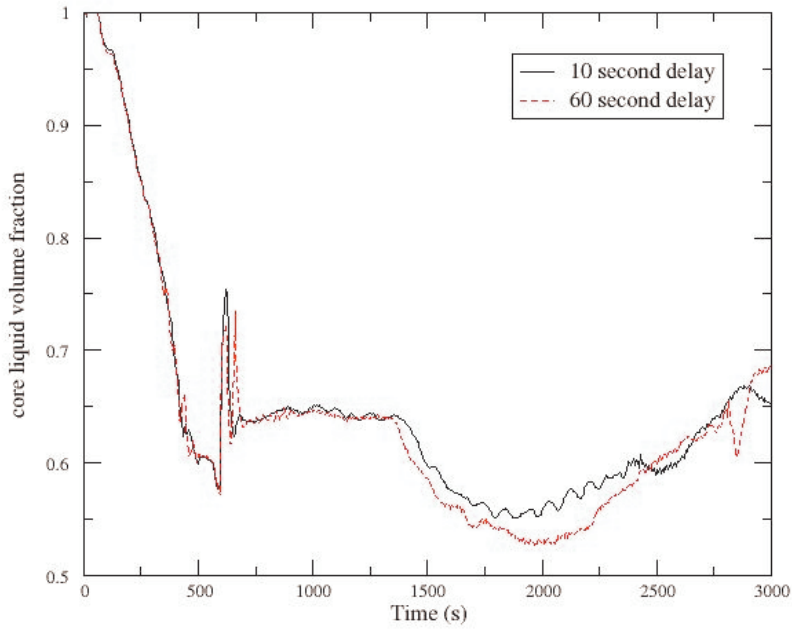
**Figure 3.9** Integrated break flow for transition breaks.



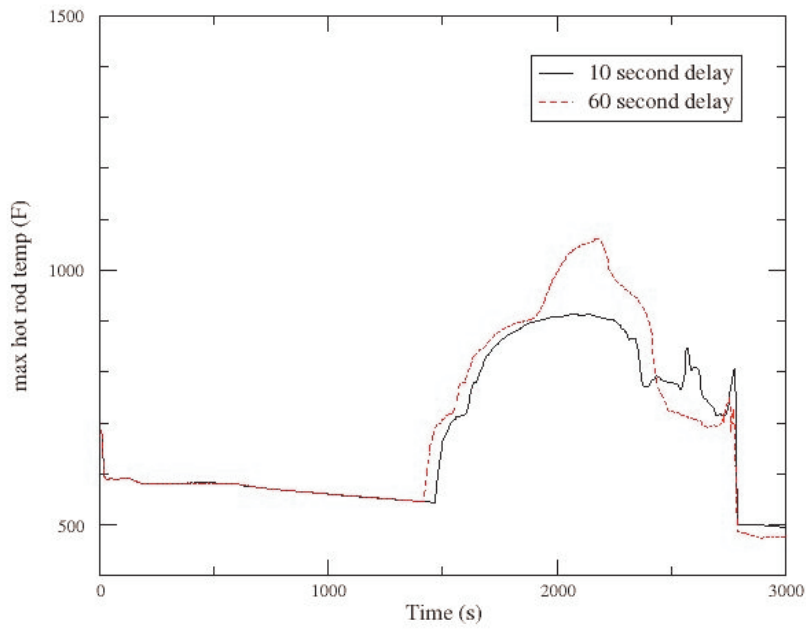
**Figure 3.10** Core collapsed level for 2 inch break.



**Figure 3.11** PCT for 2 inch break.

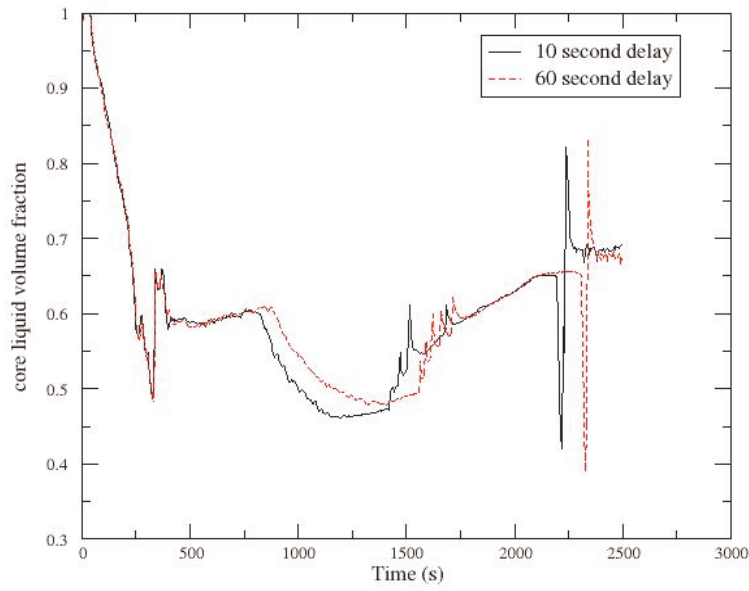


**Figure 3.12** Core collapsed level for 3 inch break.

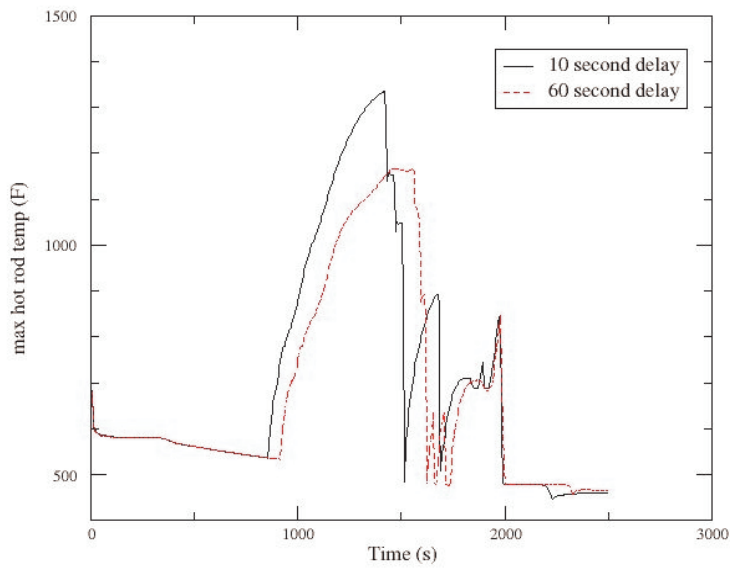


**Figure 3.13** PCT for 3 inch break.

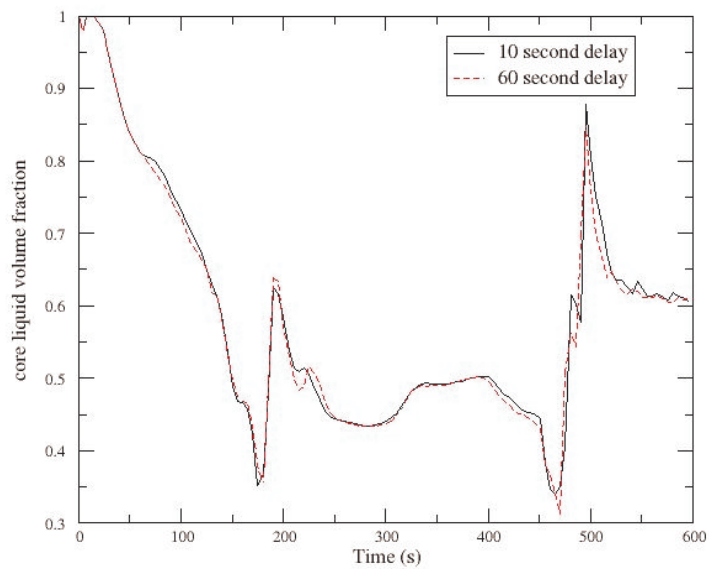




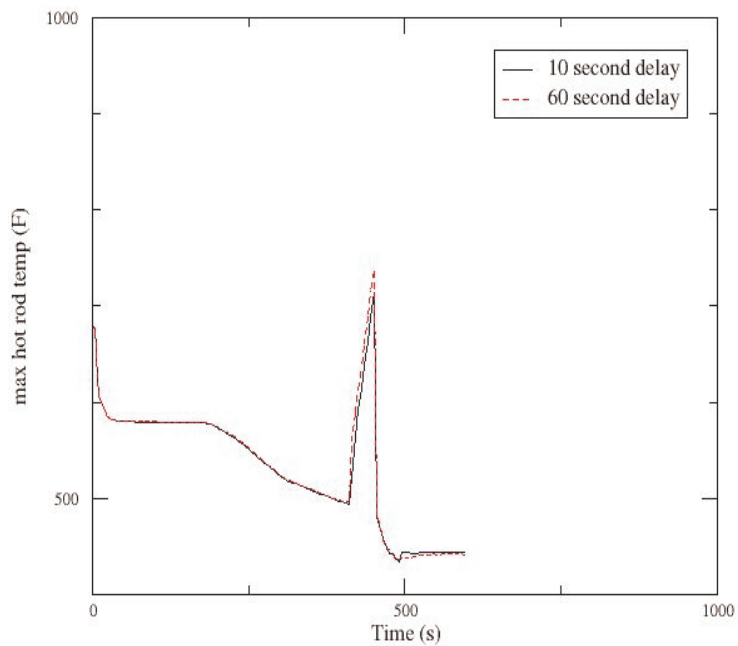
**Figure 3.14** Core collapsed level for 4 inch break.



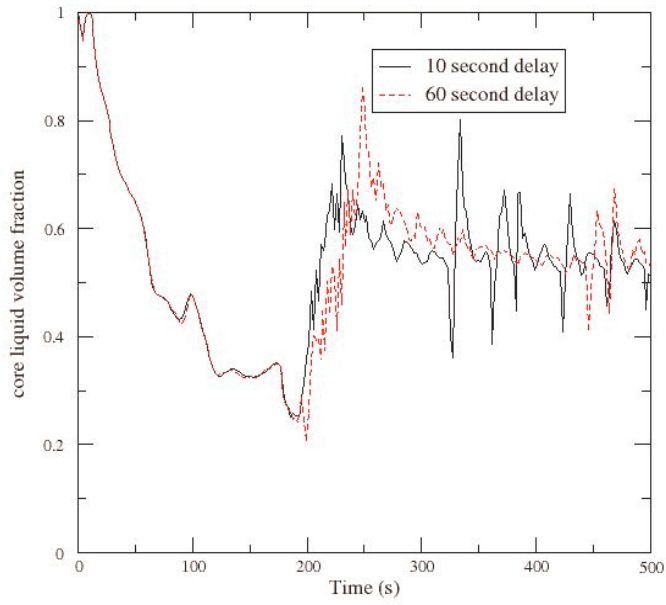
**Figure 3.15** PCT for 4 inch break.



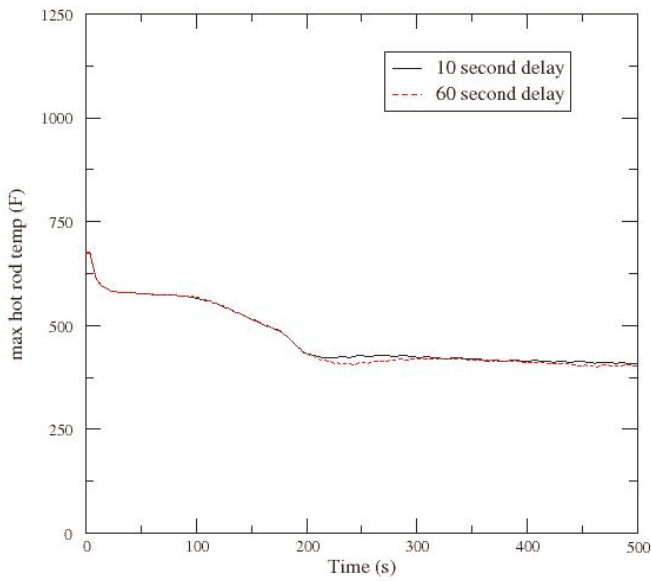
**Figure 3.16** Core collapsed level for 6 inch break.



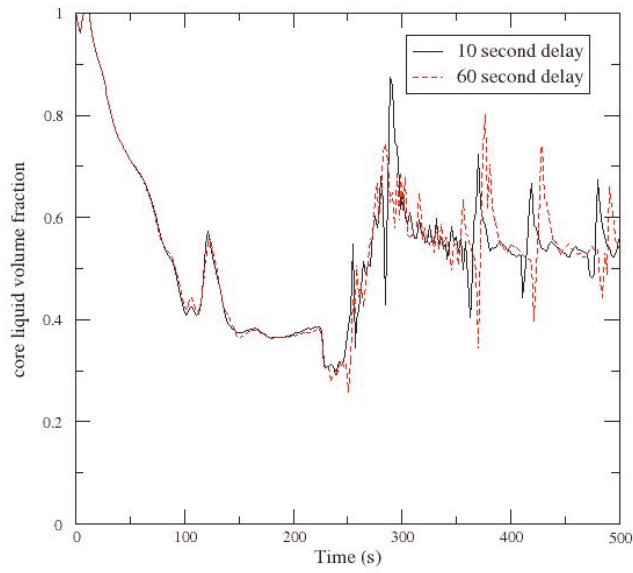
**Figure 3.17** PCT for 6 inch break.



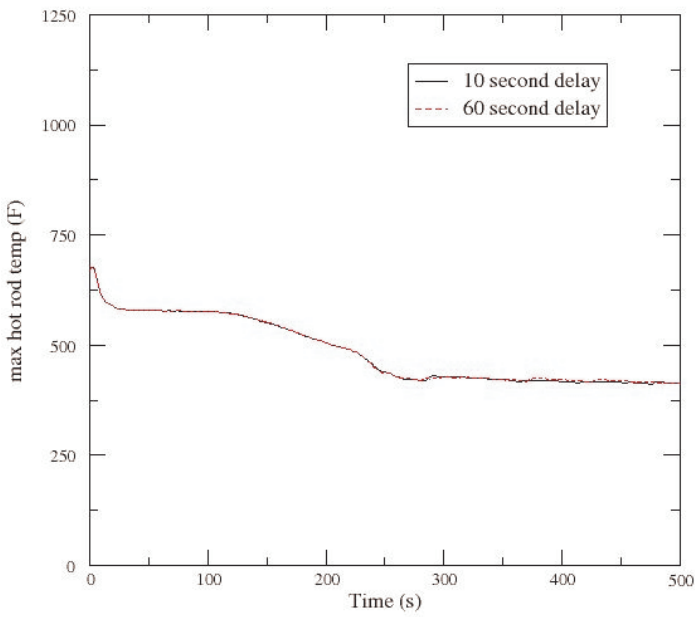
**Figure 3.18** Core collapsed level for SI line break.



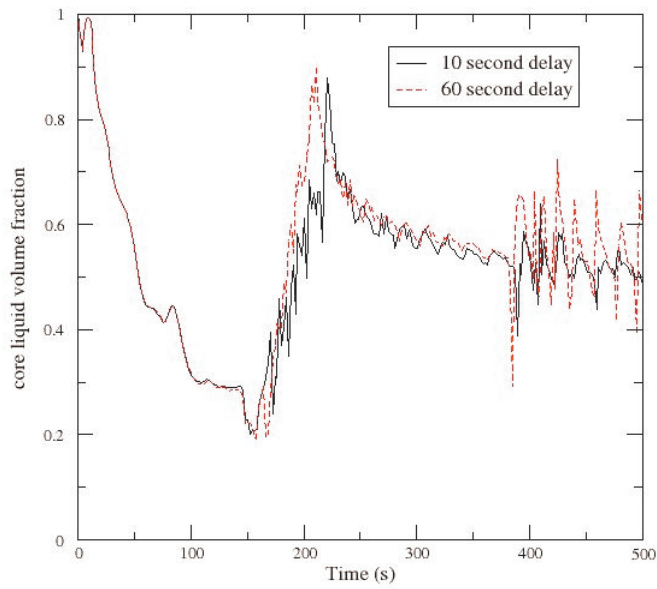
**Figure 3.19** PCT for SI line break



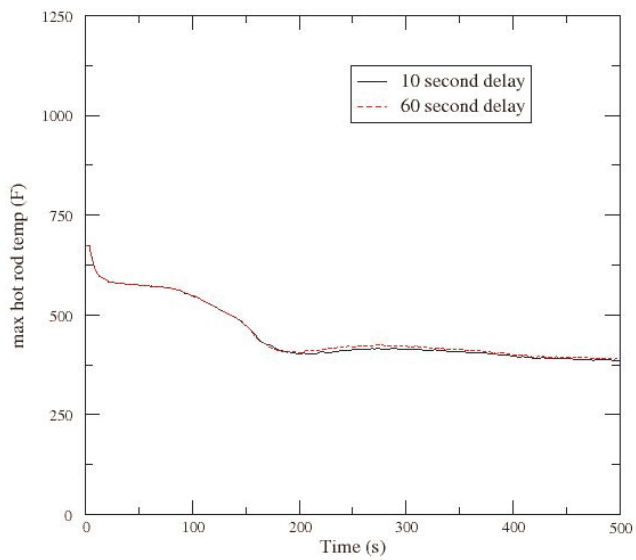
**Figure 3.20** Core collapsed level for the 0.8 SI line break.



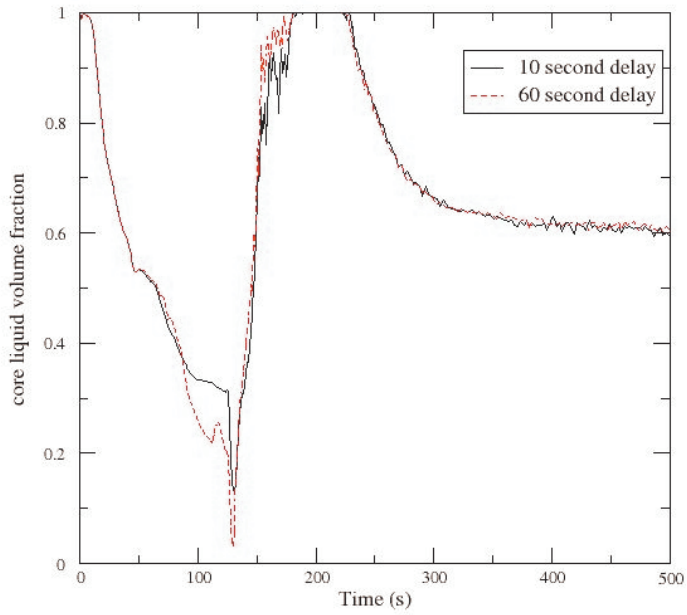
**Figure 3.21** PCT for the 0.8SI line break.



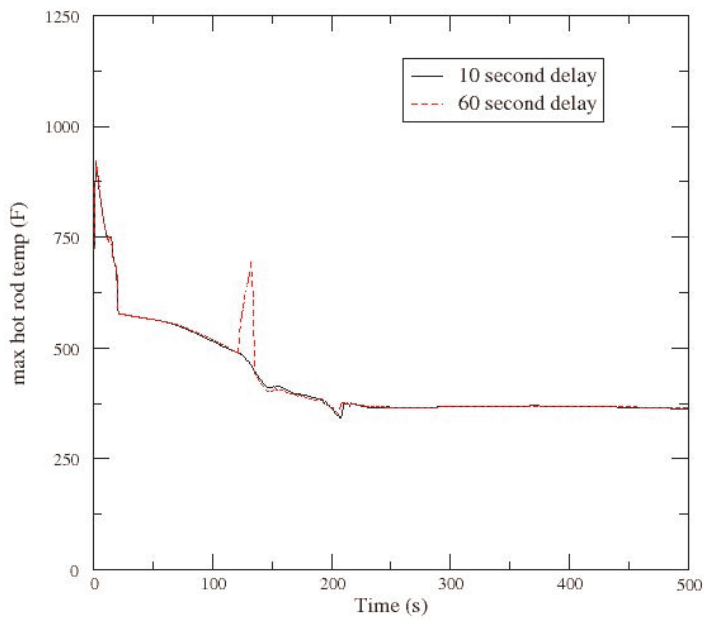
**Figure 3.22** Core collapsed level for the 1.2 SI line break.



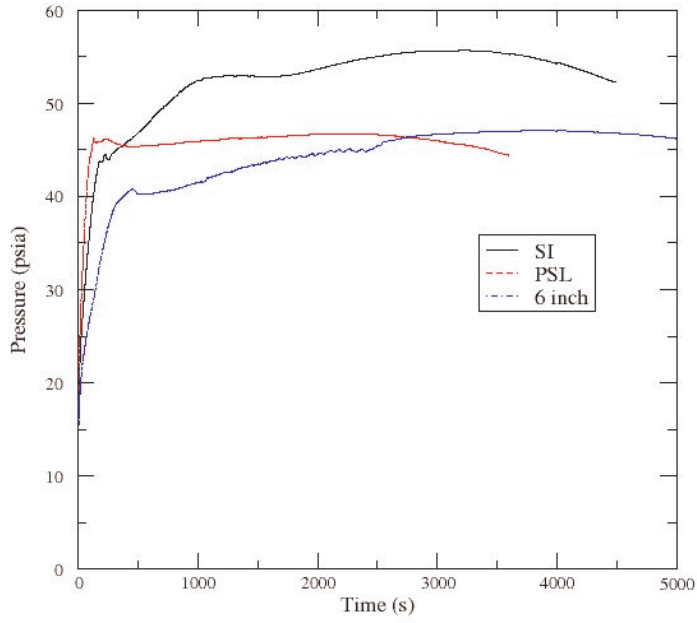
**Figure 3.23** PCT for the 1.2 SI line break.



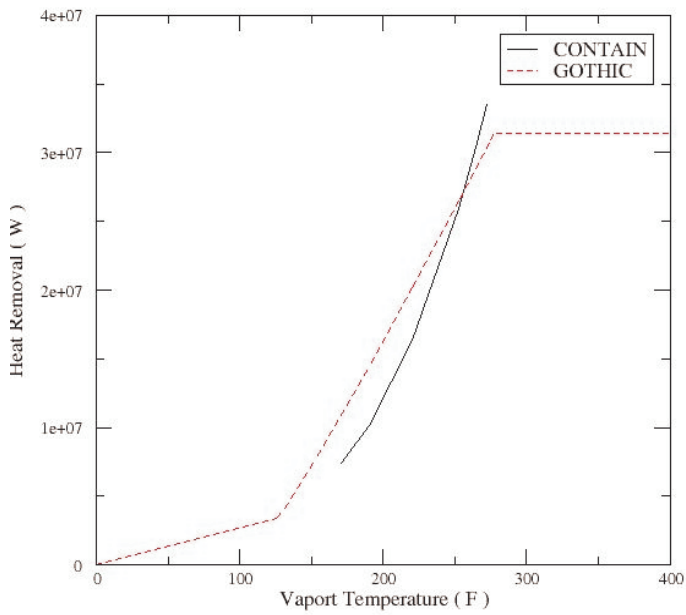
**Figure 3.24** Core collapsed level for the PSL break



**Figure 3.25** PCT for the PSL break



**Figure 3.26** Containment pressure for the SI line and PSL break without sprays.



**Figure 3.27** Fan cooler heat removal.

## 4.0 ANALYSIS AND SENSITIVITY STUDIES USING RELAP5

This section describes the series of calculations to investigate the effect of EDG delay and delayed containment spray actuation on breaks at or below the transition break size using RELAP5. These calculations supplement the TRACE/CONTAIN results in Section 3. The break sizes and locations simulated with RELAP5 are:

Table 4.1 Break Sizes and Locations

	<b>Break</b>	<b>Area (ft<sup>2</sup>)</b>
1	2" break at the bottom of the cold leg pump discharge pipe	0.02181
2	3" break at the bottom of the cold leg pump discharge pipe	0.04908
3	4" break at the bottom of the cold leg pump discharge pipe	0.08727
4	6" break at the bottom of the cold leg pump discharge pipe	0.19632
5	<i>80% SI/accumulator line break</i>	<i>0.3341</i>
6	<b>100% SI/accumulator line break</b>	0.4176
7	<b>120% SI/accumulator line break</b>	0.5011
8	<b>Pressurizer surge line break</b>	0.6827

This set of calculations was performed with a diesel delay time of 10 seconds. Cases 1 through 4 are small break LOCAs. Cases 5, 6 and 7 are transition break sizes. Case 8 was chosen because it is the largest pipe attached to the RCS main coolant piping. A second set of calculations was performed with a diesel delay time of 60 seconds. Enough plant detail was included to be able to estimate RWST water usage and to investigate containment spray delay. These modeling features are described in Section 4.2.

### 4.1 RELAP5 Executable Version

The code version used was RELAP5/MOD3.3 version ek. After discussions with the RELAP5 code developer, it was decided that using a version beyond version ek would not be necessary. All calculations were performed on the RES LINUX TRACE cluster.



## 4.2 Nodalization and Plant Modeling

The plant input model was derived from a Seabrook deck that was originally developed by INEL/EG&G between 1982 and 1992. An updated version was developed for NRC in 1994 [9]. Seabrook is one of several NSSS designated as a Standard 412 plant. Each of the four primary system loops is modeled separately. The containment model in this input deck was relatively simple (see subsection 4.2.2).

### 4.2.1 Reactor Coolant System Modeling

The Seabrook deck was used previously to analyze large breaks and investigate the effects of different metal water and decay heat models. It was also used to explore the effect of downcomer boiling at low containment pressure. Some of the input deck modifications made for this study to reflect the detail needed for this type of study include:

Table 4.2 Input Deck Modifications

1	Increasing the number of axial core nodes from 6 to 21.
2	Developing a single assembly hot channel with cross flow to the average channel

Additional modifications were made to ensure the model was an accurate representation of a Westinghouse Standard 412 plant. A summary of the input conditions appropriate for a Standard 412 plant were provided by Westinghouse [4]. The principal additional modifications are listed in Table 4.3.

Table 4.3 Input Modifications Made for This Study

1	Reactor Power = 3636.3 MWT = 1.02 x 3565 MWT (Standard 412 stretch power) [4].
2	An average channel with 192 assemblies.
3	A hot channel with one assembly.
4	A hot pin in the hot channel with FQ = 2.52 to match Westinghouse 14.974 kw/ft peak LHGR [4]
5	A hot pin in the average channel with FQ = 2.52
6	Top peaked axial power profile with FΔZ (max) = 1.31 [4]
7	Shutdown power table using 1.2 x ANS73 standard plus actinides [4]
8	No auxiliary feedwater to avoid SG overfill without proper controls
9	Downcomer with single azimuthal region to avoid unrealistic circulation seen with multi-dimensional downcomer.
10	Steam generators modified to Model F design with 10% tube plugging assumed [4].
11	Increased secondary safety valve setpoint to 1235 psia [4]
12	Increased secondary pressure to 1000 psia [4]
13	Increased primary pressure from 2230 psia to 2250 psia [4]
14	Closed hot leg nozzle gap
15	Implemented PG (Czech Republic) DNB correlation instead of default Groeneveld model (Table lookup). This eliminated excessive early high power DNB.
16	Implemented Ransom/Trapp critical flow model instead of default Henry/Fauske model, which is closer to the model in TRACE. Improved stability and comparability.
17	Modified core junction hydraulic diameters to values more appropriate for use with mod 3.3 than the original values used in mod 3.1.
18	Used Westinghouse supplied ECCS delivery tables. [4]
19	Modified low pressurizer pressure reactor trip setpoint to 1937 psia [4]
20	Modified main feedwater inlet temperature to 448 F [4]
21	Modified low pressure SI injection signal to 1715 psia [4]
22	Modified accumulator water and gas volumes to 900 and 500 ft <sup>3</sup> respectively [4]

It was decided to run the steady state and transient for these cases from a continuous input stream. That is, a null transient was run, and at an appropriate time the break was opened without stopping and restarting the calculation. This maximized the consistency between the steady state and the transient. Each input deck is the same except for the break section. Each steady state was run for 1000 seconds after which a time trip started the break. Each break was run for 7000 seconds after the steady state principally to run long enough to assure that RWST water inventory can be reasonably calculated or extrapolated.

The nodalization of each break section is shown in Figure 4.1. Figure 4.1a is used for the 2", 3", 4" and 6" small breaks. This configuration is also used for the three transition SI line breaks. A time dependent psuedo-volume is used to receive water spilled from the SI line side

of the break. In this way the low energy spilled water is not emptied into the containment which could lower the containment pressure. However, water from the RWST is still accounted for in estimating switchover from RWST to sump.

For the surge line break (Figure 4.1b), the normal flow path from the pressurizer to the hot leg is treated as a normally open inter-break trip valve for the steady state part of the calculation, and is closed at the time of break initiation.

Figure 4.1c shows the proposed large cold leg break nodalization. Large breaks are not discussed in this report.

The nodalization for the balance of the RCS is as shown in Reference 9 except as explained for the number of axial nodes in the core, core bypass and annulus regions.

#### 4.2.2. Containment Modeling

It has been suggested that one of the potential safety benefits of the revised ECCS rule is to delay or eliminate the need for containment sprays thereby preserving RWST water and delaying switchover to containment sump injection. Sufficient containment input modeling features were developed to be able to reasonably estimate containment spray actuation time based on containment pressure. In addition, an RWST component was added to be a water source for pumped ECCS and containment spray. The RWST water volume was taken to be the usable volume provided in a personal communication from Westinghouse [10]. The RELAP5 Standard 412 plant containment and RWST model are shown in Figure 4.2.

The information used for containment modifications was taken from a WGOTHIC input description provided by Westinghouse [11]. This included:

Table 4.4 Containment Input Information [11]

1	Containment volume = $2.5 \cdot 10^6$ ft <sup>3</sup>
2	15 heat structures as shown in Table 2.2
3	Heat structure conduction properties
4	Fan cooler heat removal as a function of containment vapor temperature
5	Constant flow rate containment spray
6	Spray and fan cooler containment pressure trips

Heat structure properties and dimensions were obtained from a WGOTHIC model of a large dry containment. The conduction nodalization was not exactly the same but very comparable.

The most direct way to model fan coolers in RELAP5 was to utilize heat structures only and not model a piping system. However, the WGOthic fan cooler information required modification to fit the input requirements of RELAP5, which can model tabular heat transfer coefficients as a function of surface temperature. A simple spreadsheet was developed for a heat conduction model that assumed no stored heat in the heat structure, a fixed temperature inner boundary and a heat transfer area. Heat transfer coefficients as a function surface temperature were then calculated from the WGOthic table of heat flux as a function of vapor temperature. The new table was used successfully as a “fan cooler” heat structure boundary condition in RELAP5.

A containment spray system was developed by simply including a time dependent water flow from the RWST to the top of the containment using the default RELAP5 heat transfer model. No attempt was made to simulate a verified spray heat transfer model.

#### 4.3 Steady-State Simulation

Table 4.5 shows selected variables for the steady portion of the transient. The target values used for RELAP5 came from either Reference 4 or personal communication from the vendor Westinghouse [5].

Table 4.5 Standard 412 Plant Steady State Conditions

<u>Parameter</u>	<u>RELAP5 Value</u>	<u>Reference Plant</u>
Reactor Power (MWT)	3636.3	3636.3 [4]
Pressurizer Pressure (psia)	2275.	2250 [4] 2300 [10]
Feedwater Flow (lb./s./gen.)	1127.	1126.9 [10]
Steam Flow (lb./s./gen.)	1127.	1126.9 [10]
Feedwater Temp. (EF)	447.7	448. [4]
T(hot) (EF)	622.	--
T(cold) (EF)	553.	--
T(ave.) (EF)	587.	588. [4]
Primary Loop Flow (GPM)	87500.	93600. [4]
RCS Inlet Flow By-pass (%)	2.4	8.4 [4]
SG Tube Plugging	10	10 [4]
SI Water Temp. (EF)	120.	120. [4]
Accumulator Temp. (EF)	140.	130. [4]

In every calculation the steady state was reached well within the 1000 second duration indicated in Section 4.2.1.

#### 4.4 LOCA Calculations

Results of the eight breaks shown in Table 4.1 are compared in this section. For all the figures and tables the time is reset to zero at the beginning of the break.

##### 4.4.1 Transient Event Timing Analysis

Table 4.6 compares event timing for all eight calculations assuming a standard 10 second diesel delay and Table 4.7 for the case with a 60 second diesel delay. Where the event times are different by more than 10 seconds from one table to the other the values are in bold type. Obviously if an event occurs before SI actuation, the event timings are the same. As expected, SI actuation always varies by 50 seconds from Table 4.3 to 4.4 for each break. Except for the 3", 4" and 6" breaks, the 50 second additional diesel delay has little effect on the event timing. The effect on containment spray signals and spray actuation does not effect other event times and is further discussed in section 4.3. For the 3" break, the only significant differences are the core uncover and recovery times. However this is somewhat deceiving, since the depth of core uncover in the 60 second delay case is very shallow and has a negligible effect on PCT during the core uncover time period. The surprising result is the 4" break, which shows less core uncover with the longer diesel delay and a reduced cladding temperature during the uncover period. This is discussed in the next section.

Table 4.6 Base Case Event Timing (10 sec. Diesel delay)  
(Time in seconds after the break)

<u>Event</u>	<u>Break</u>	<u>2"</u>	<u>3"</u>	<u>4"</u>	<u>6"</u>	<u>80% SI</u>	<u>100% SI</u>	<u>120% SI</u>	<u>Surge Line</u>
Start		0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor trip signal		43.	19.	11.	5.	5.	4.	3.	1.
SI Signal		51.	25	15.	9.	7.	7.	7.	5.
SI Begins		<b>93.</b>	<b>67.</b>	<b>57.</b>	<b>51.</b>	<b>51.</b>	<b>50.</b>	<b>49.</b>	<b>47.</b>
Spray Signal		N/A	N/A	N/A	<b>3419.</b>	1149.	839.	637.	N/A
Spray begins		N/A	N/A	N/A	<b>3455.</b>	1184.	873.	673.	N/A
Loop seal venting		1405	646	348	164.	9.	9.	8.	6.
Core top uncover		N/A	<b>N/A</b>	<b>732.</b>	440.	N/A	N/A	N/A	N/A
Begin accumulators		N/A	N/A	<b>1032.</b>	438.	232.	186.	156.	108.
PCT occurs		0.0	660	<b>1100.</b>	182.	4.	3.	2.	0.0
Core top recovery		N/A	<b>N/A</b>	<b>1540.</b>	448.	N/A	N/A	N/A	N/A

Table 4.7 Event Timing (60 sec. Diesel delay)  
(Time in seconds after the break)

<u>Event</u>	<u>Break °</u>	<u>2"</u>	<u>3"</u>	<u>4"</u>	<u>6"</u>	<u>80% SI</u>	<u>100% SI</u>	<u>120% SI</u>	<u>Surge Line</u>
Start		0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Reactor trip signal		43.	19.	11.	5.	5.	4.	3.	1.
SI signal		51.	25.	15.	9.	7.	7.	7.	5.
SI begins		<b>143.</b>	<b>117.</b>	<b>107.</b>	<b>101</b>	<b>101.</b>	<b>100.</b>	<b>99.</b>	<b>97.</b>
Spray Signal		N/A	N/A	N/A	<b>N/A</b>	1159.	831.	645.	N/A
Spray begins		N/A	N/A	N/A	<b>N/A</b>	1194.	867.	681.	N/A
Loop seal venting		1402.	638.	348.	154	9.	8.	8.	6.
Core top uncover		N/A	<b>1687.</b>	<b>864.</b>	438	N/A	N/A	N/A	N/A
Begin accumulators		N/A	N/A	<b>1082.</b>	433	232.	186.	156.	108.
PCT occurs		0.0	652.	<b>376.</b>	178	5.	4.	3.	0.0
Core top recovery		N/A	<b>2098.</b>	<b>1146.</b>	442	N/A	N/A	N/A	N/A

#### 4.4.2 RCS T/H Behavior and Cladding Temperature Analysis

Primary system pressure is shown for the smaller cold leg breaks in Figure 4.3. As expected the depressurization rate is a direct function of break size and nearly independent of diesel delay time differences in this range. The 2" breaks depressurize only to about 1200 psia, which means that inventory replenishment is principally from the charging pump, and HPSI pumps and not from the accumulators and LPSI pump. For the 3" breaks, the depressurization level is about 600 psia at 3500 sec. At this level the contribution of charging and HPSI is about the same, the accumulators are barely flowing and there is no LPSI flow. For the 4" breaks, the system depressurizes to about 400 psia. This allows accumulator water flow to terminate the cladding temperature transient. For the 6" breaks the pressure drops to below 300 psia rapidly, and accumulator injection prevents cladding heatup.

Figure 4.4 shows primary depressurization for the pressurizer surge line and the safety injection (SI) line. For the SI line breaks, the pressure level drops to between 200 and 250 psia. As was the case for the 6" break, there is sufficient accumulator injection to prevent cladding heatup. Depressurization of the surge line break is sufficient to actuate the LPSI.

Figures 4.5 and 4.6 show the integrated break flow rate for the 16 calculations. The early portion of each curve, before the change in slope reflects the system depressurization. The later portion of each curve reflects the spillage out the break of excess ECCS water. As expected break flow increases with increasing break size and is not influenced by diesel delay time.

Core collapsed water level is shown in Figures 4.7 and 4.8. For the 3", 4" and 6" breaks the water level drops rapidly as inner vessel pressure builds up just prior to loop seal clearing. After the loop seal clears, the water level in the core goes up quickly as its pressure decreases. Figure 4.8 shows the collapsed water level for the transition breaks. Water level recovers quickly for all four of these larger breaks as the pressure drops below about 300 psia allowing accumulator water to refill the vessel long after complete loop seal clearing.

Analysis of the 16 calculations indicated that the PCT could occur during any one of three distinct time periods. The first was a DNB peak that could occur during early depressurization when the power was still relatively high. The second could occur just prior to the initiation of loop seal clearing when the inner vessel loses water then quickly refills after the loop seal clears. The third would be during the potentially relatively long boil-off and refill period. The third period is generally the one of most concern because of its potential duration. Tables 4.8 and 4.9 provide cladding temperature information for the three time periods for the 16 calculations. If there was a very small or no heatup the table indicates "none". It should also be noted that no steady state DNB was observed for any calculation.

Table 4.8 Peak Cladding Temperature Summary (10 sec. Diesel delay)

<u>Event</u>	<u>Break °</u>	<u>2"</u>	<u>3"</u>	<u>4"</u>	<u>6"</u>	<u>80% SI</u>	<u>100% SI</u>	<u>120% SI</u>	<u>Surge Line</u>
DNB PCT (EF)		none	none	none	none	751.	762.	776.	none
t (PCT) (sec.)		-	-	-	-	5.	4.	2.	-
elevation (ft)		-	-	-	-	10.5	9.9	9.3	-
-----									
loop vent PCT (EF)		none	743.	839.	712.	none	none	none	none
t (PCT) (sec.)		-	659.	370.	182.	-	-	-	-
elevation (ft)		-	11.1	10.5	9.9	-	-	-	-
-----									
boil-off PCT (EF)		none	none	1085.	none	none	none	none	none
t (PCT) (sec.)		-	-	1102.	-	-	-	-	-
elevation (ft)		-	-	11.1	-	-	-	-	-

Table 4.9 Peak Cladding Temperature Summary (60 sec. Diesel delay)

<u>Event</u>	<u>BreakY</u>	<u>2"</u>	<u>3"</u>	<u>4"</u>	<u>6"</u>	<u>80% SI</u>	<u>100% SI</u>	<u>120% SI</u>	<u>Surge Line</u>
DNB PCT (EF)		none	none	none	none	751.	762.	776.	none
t (PCT) (sec.)		-	-	-	-	5.	4.	2.	-
elevation (ft)						10.5	9.9	9.3	none
-----									
loop vent PCT (EF)		none	753.	871.	721.	none	none	none	none
t (PCT) (sec.)		-	652.	376.	178.	-	-	-	-
elevation (ft)		-	10.5	10.5	10.5	-	-	-	-
-----									
boil-off PCT (EF)		none	none	695.	none	none	none	none	none
t (PCT) (sec.)		-	-	1088.	-	-	-	-	-
elevation (ft)		-	-	11.1	-	-	-	-	-

As discussed in section 4.4.1, only the 4" break with 10 second diesel delay indicated any significant boil-off PCT. Figures 4.9 and 4.10 provide cladding temperature histories for the hottest elevations in the hot pin within the hot assembly. The cladding temperature heatup and peaks associated with early DNB, the heatup and peaks prior to loop seal clearing, and the boiloff heatup and peaks are identified in the figures.

Because the boil-off PCT for the 4" cole leg break (CLB) with a 60 second diesel delay was significantly lower than the 10 second delay case, it was decided to investigate this anomaly.



It was determined that only the loop seal in one intact loop cleared (Figure 4.10a) for the 10 second delay case, and only the loop seal in the broken loop cleared for the 60 second delay case (Figure 4.10b). No “biasing” model was used in any of the calculations shown in Tables 4.8 and 4.9. That is, no code input was used to influence the venting or non-venting of particular loop seals. This is typical of the phenomenological uncertainty associated with loop seal clearing, which has been known for some time [8]. The problem is addressed with biasing models in PWR small break evaluation models used to demonstrate compliance with 50.46.

For the two 4" CLB cases a biasing model was then employed to force no venting in three of the loops, and allow venting in only one intact loop. This is similar to what occurred in the 10 second case without biasing. The PCTs for the two cases with the vent biasing are shown in Figure 4.10c. The PCT is no longer lower for the 60 second delay case, indicating that uncertainties associated with loop seal clearing is the reason.

#### 4.4.3 Containment and Water inventory Analysis

To investigate the effect of delayed containment spray actuation on breaks at or below the transition break size, the sixteen calculations described above were first performed without containment sprays but with two fan coolers. Figure 4.11 shows that for the eight base case calculations with 10 second diesel delay, the highest containment pressure was 53 psia for the 1.2 X SI line break. Figure 4.12 shows nearly identical pressures for the 60 second diesel delay. Spray actuation pressure is noted on the figures. That pressure plus a 35 second delay time was used to calculate a beginning spray time although sprays were not actuated. As can be seen in Tables 4.3 and 4.4, the spray signal and spray start times are very similar except for the 6" break. For the 6" break 10 second delay case, the containment pressure just reaches the spray signal before containment pressure turnover. For the 60 second delay case, the peak containment pressure is 0.1 psi below the signal pressure.

The effect of containment sprays on water inventory can be evaluated by comparison of Figures 4.13 and 4.14. Figure 4.13 shows the usable RWST water level for the base case with no sprays and 10 second diesel delay. The usable level is the level which originally contains 326,860 gallons of water. Thus Figure 4.13 illustrates the consumption of RWST water without sprays. A third series of eight calculations (Figure 4.14) was run using a 10 second diesel delay, but with the assumed containment spray actuation pressure of 44.7 psia and assumed delay time of 35 seconds. The one exception to this the surge line break which quickly reached a peak pressure less than 2 psi below the actuation pressure (see Figure 4.11). Since the uncertainty in the RELAP5 containment pressure is unknown, and the TRACE/CONTAIN calculation calculated a containment pressure well above the actuation pressure, the RELAP5 surge line break calculation with sprays was modified. For that case, the actuation pressure was dropped by 2 psi to assure actuation. For the three smallest breaks (2", 3", 4" CLB) the spray actuation was not reached, so the water inventory use is the same in both figures. For the larger five breaks, not actuating containment sprays delays reaching the minimum RWST inventory by 3/4 to 2 hours.

#### 4.5. Summary of RELAP5 Results

In the 10 to 60 second diesel delay time range considered for this study, the effect on event timing, PCT and other thermal hydraulic parameters was small.

The importance of loop seal clearing is indicated. Loop seal clearing terminates a cladding heatup at that time and the number of loop seals that clear and the timing of the individual clearings is not easily predicted. However the behavior as a function of break size is predictable.

The RELAP5 calculations showed possible cladding heatup during three distinct time periods: (1) an early short heatup during the first few seconds of depressurization, (2) a heatup just prior to loop seal clearing, and (3) a heatup for smaller breaks (3-4") later during boiloff and recovery. None of these calculations showed any of the heatups to be significant.

The highest containment pressure calculated for the breaks in this series of calculations was about 53 psia. Diesel delay time did not effect containment pressure. For the three smallest breaks, the containment spray actuation was not reached. For the 6" break it was barely reached. For the 6" break and the four transition breaks, if the containment sprays are not automatically actuated, the delay in reaching the minimum RWST inventory is 3/4 to 2 hours.

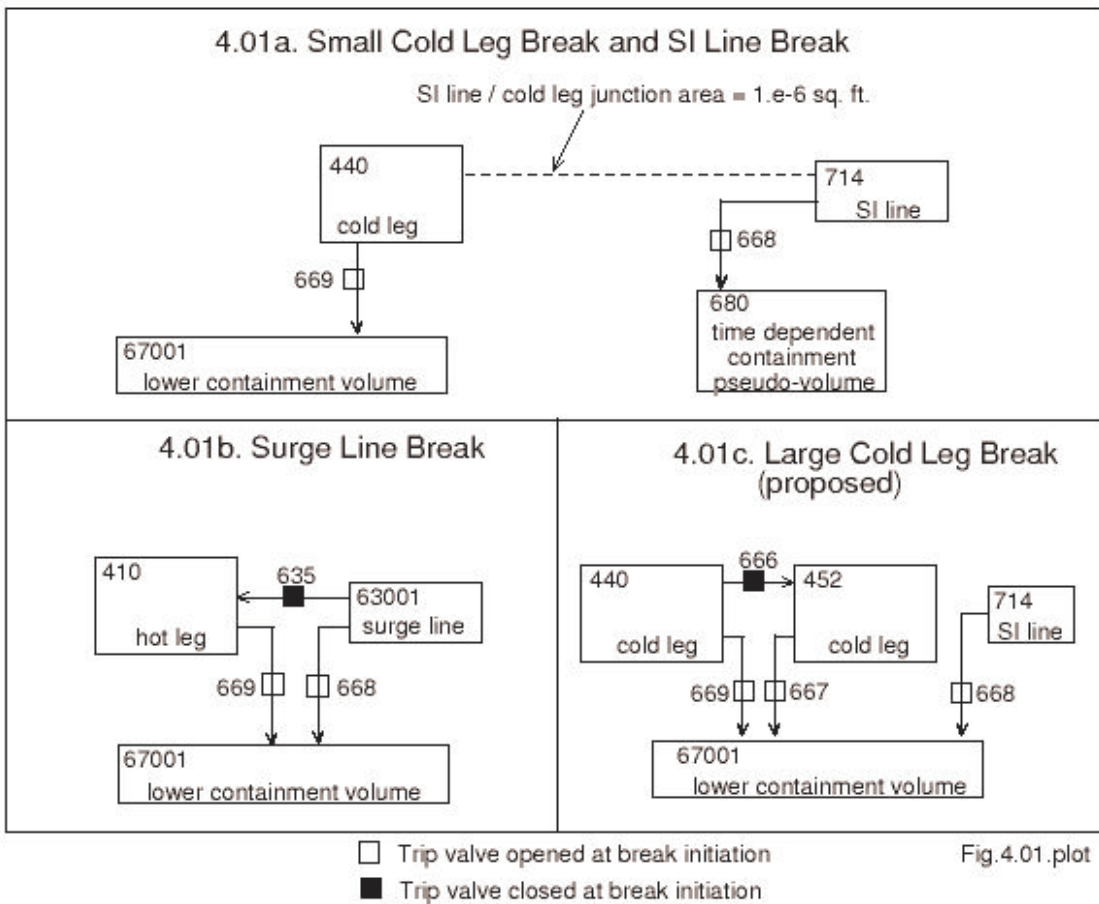


Figure 4.1 Break Nodalization

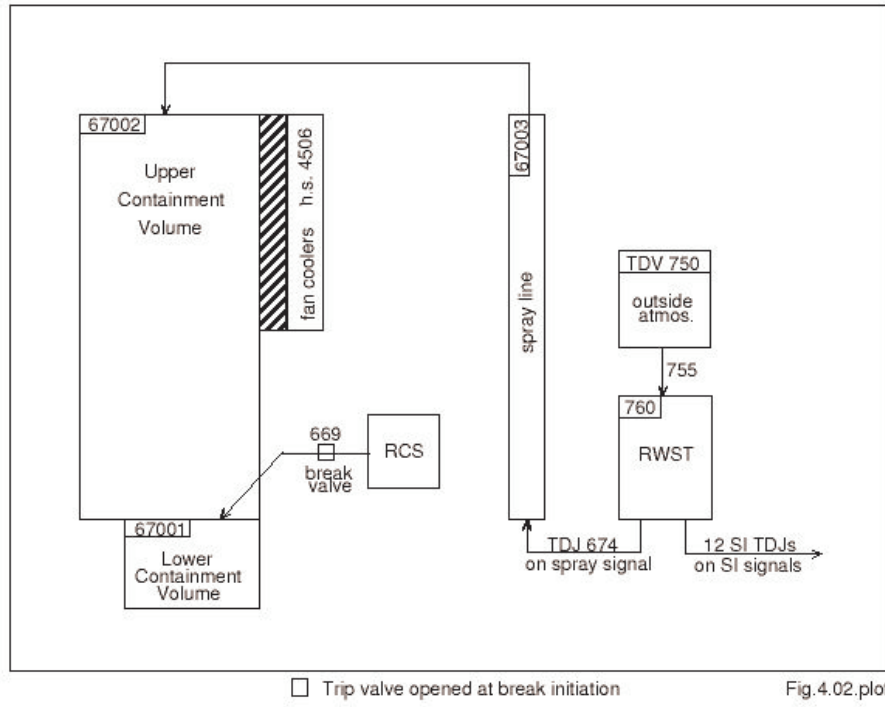


Figure 4.2 Containment Nodalization

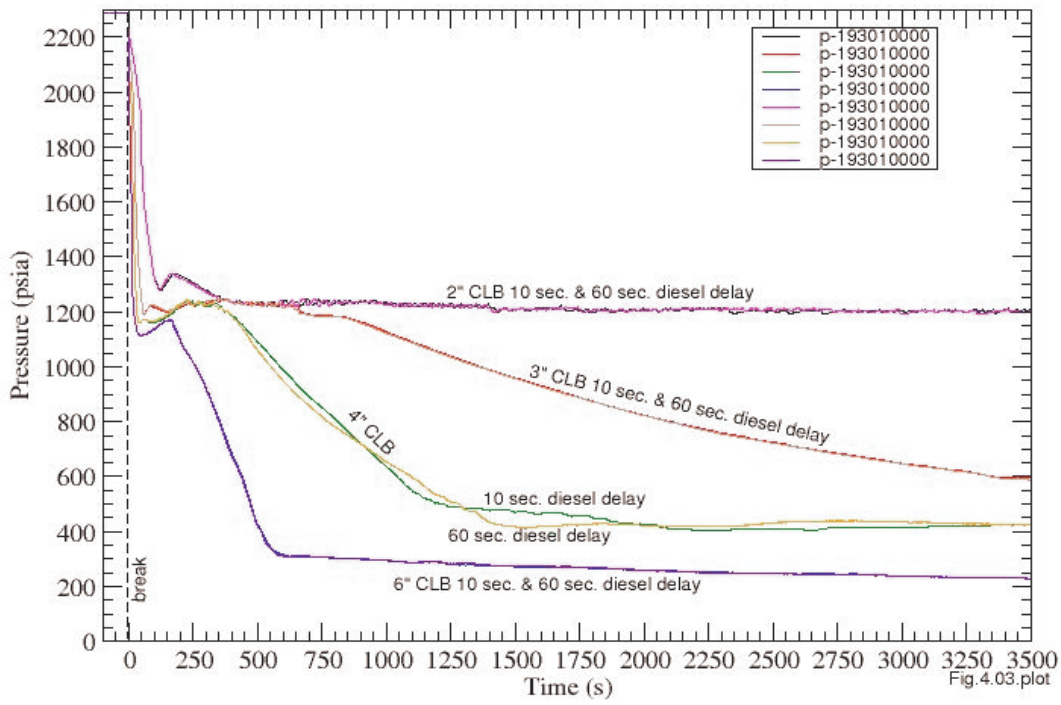


Figure 4.3 Upper Plenum Pressure  
 Std 412 - Cold Leg Breaks, no sprays

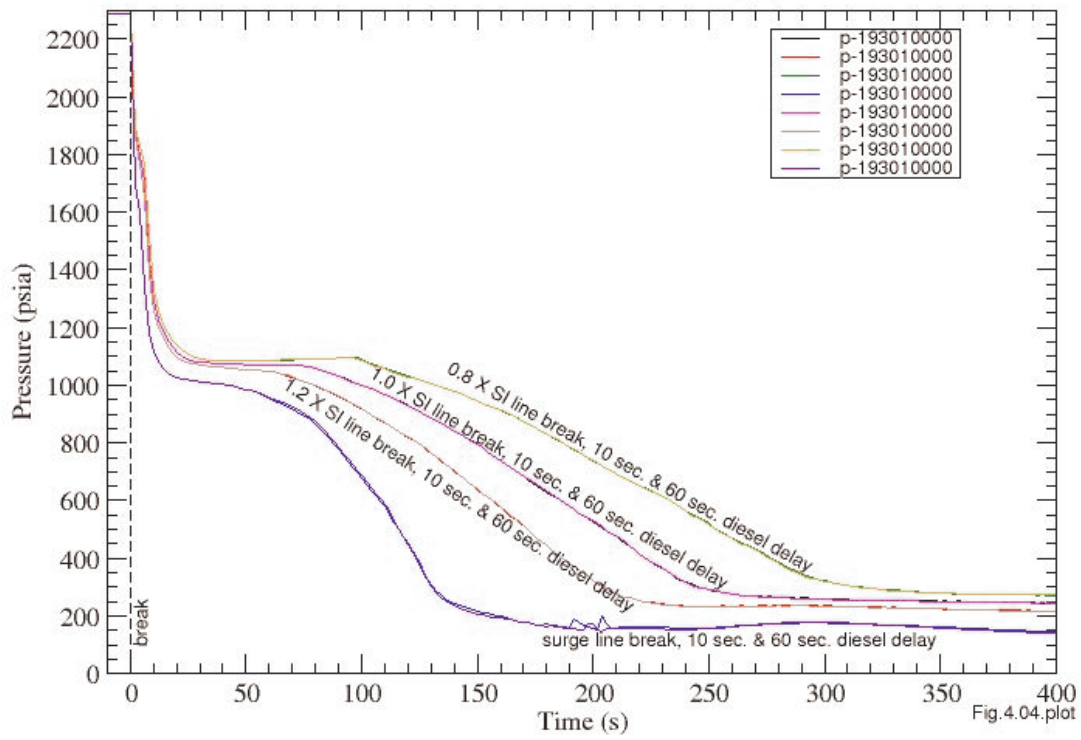


Figure 4.4 Upper Plenum Pressure Accumulator/SI Line Breaks, no sprays

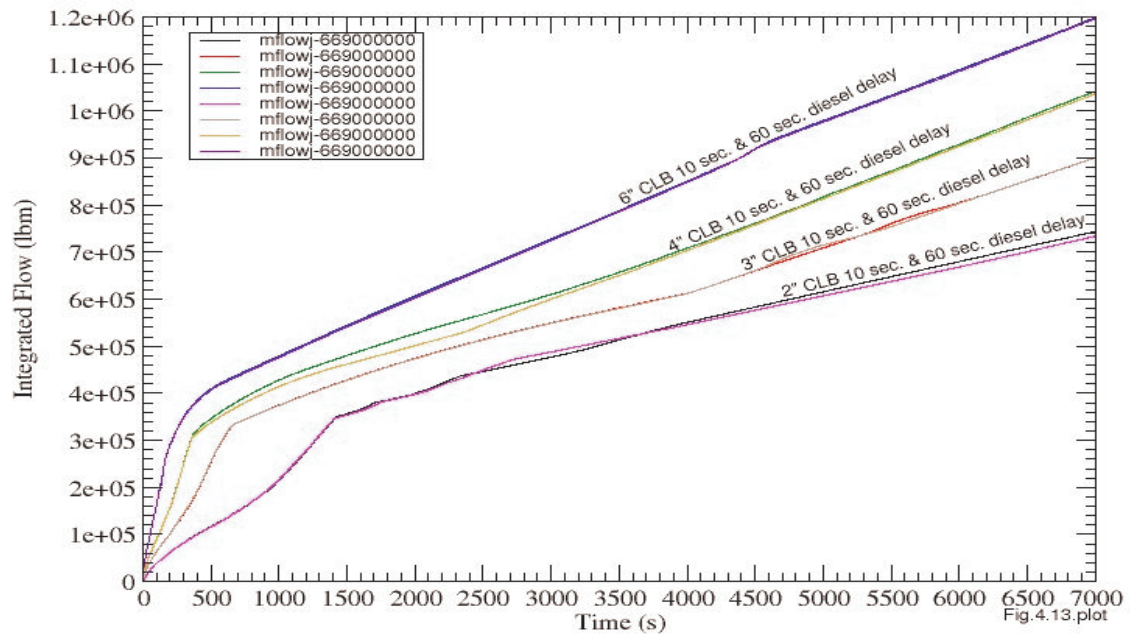


Figure 4.5 Integrated Break Flow, Small Cold Leg Breaks, no sprays

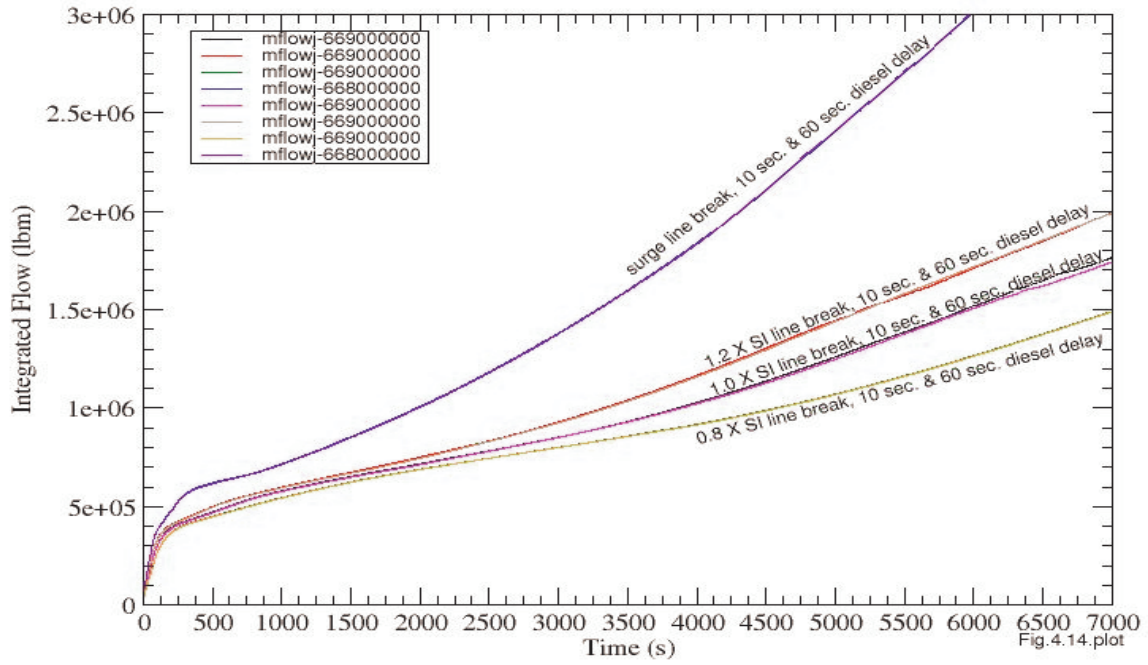


Figure 4.6 Integrated Break Flow  
Accumulator/SI Line Breaks, no sprays



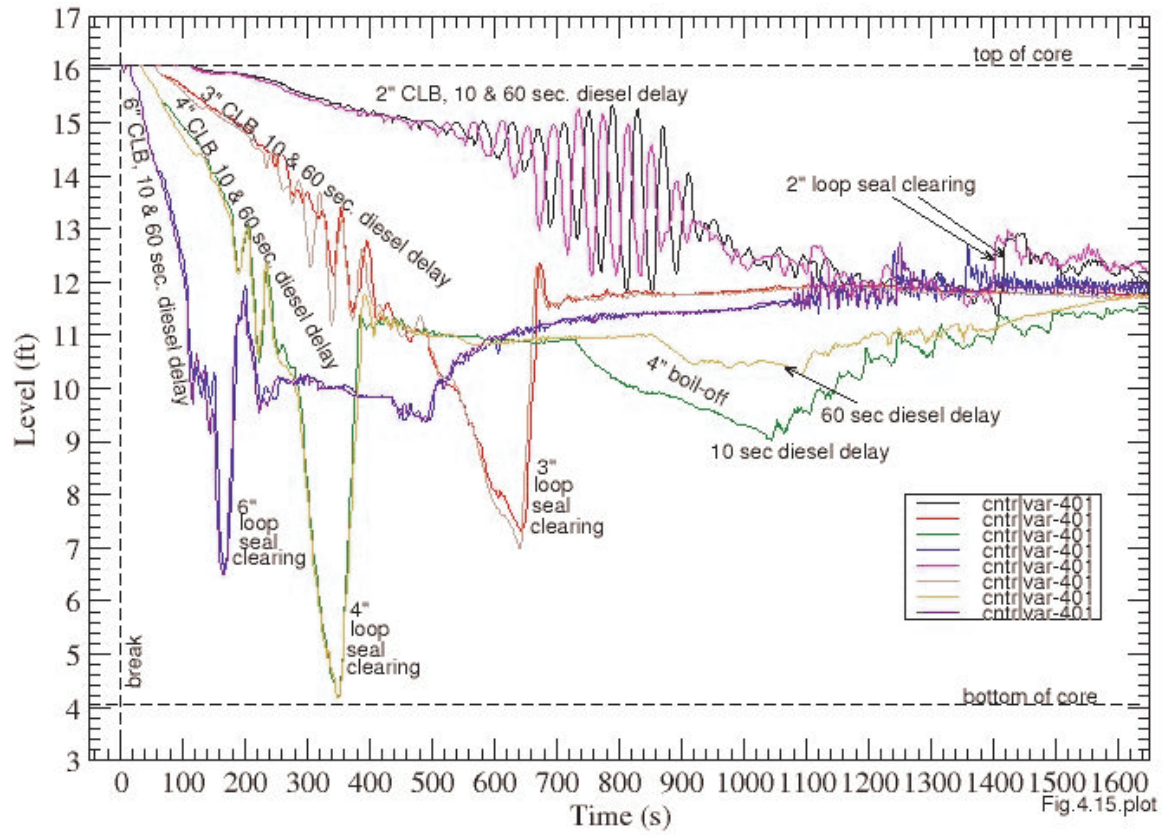


Figure 4.7 Core Collapsed Liquid Level  
Small Cold Leg Breaks, no sprays

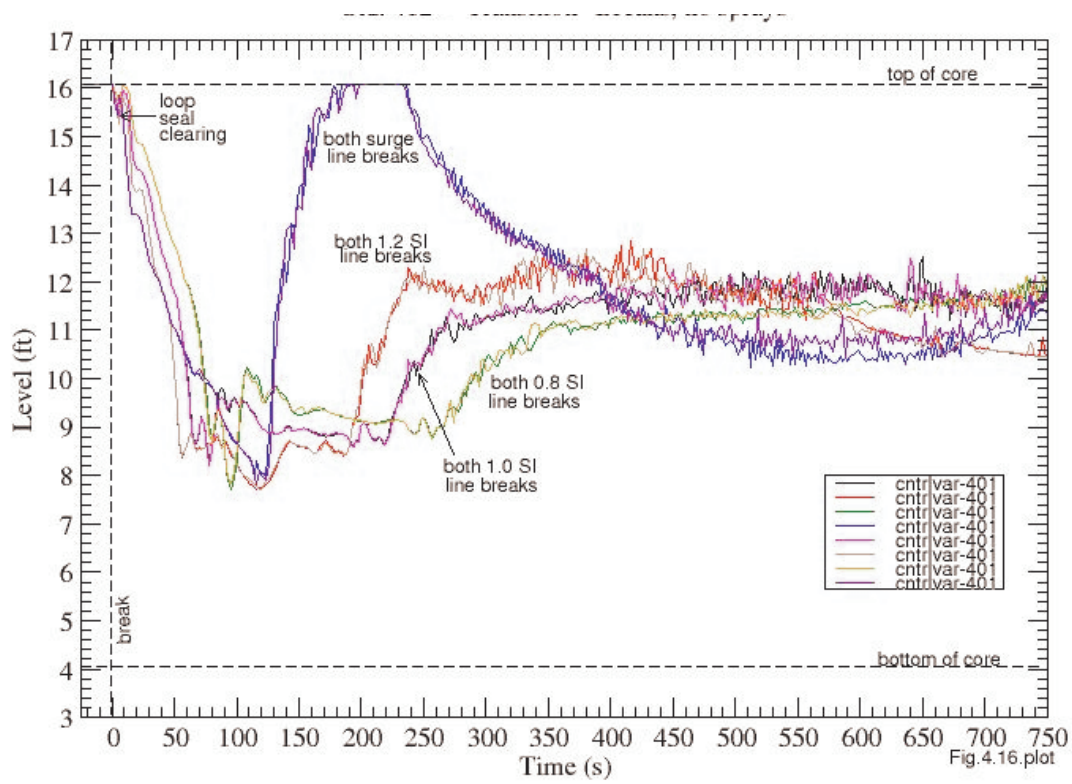


Figure 4.8 Core Collapsed Liquid Level Accumulator/SI Line Breaks, no sprays

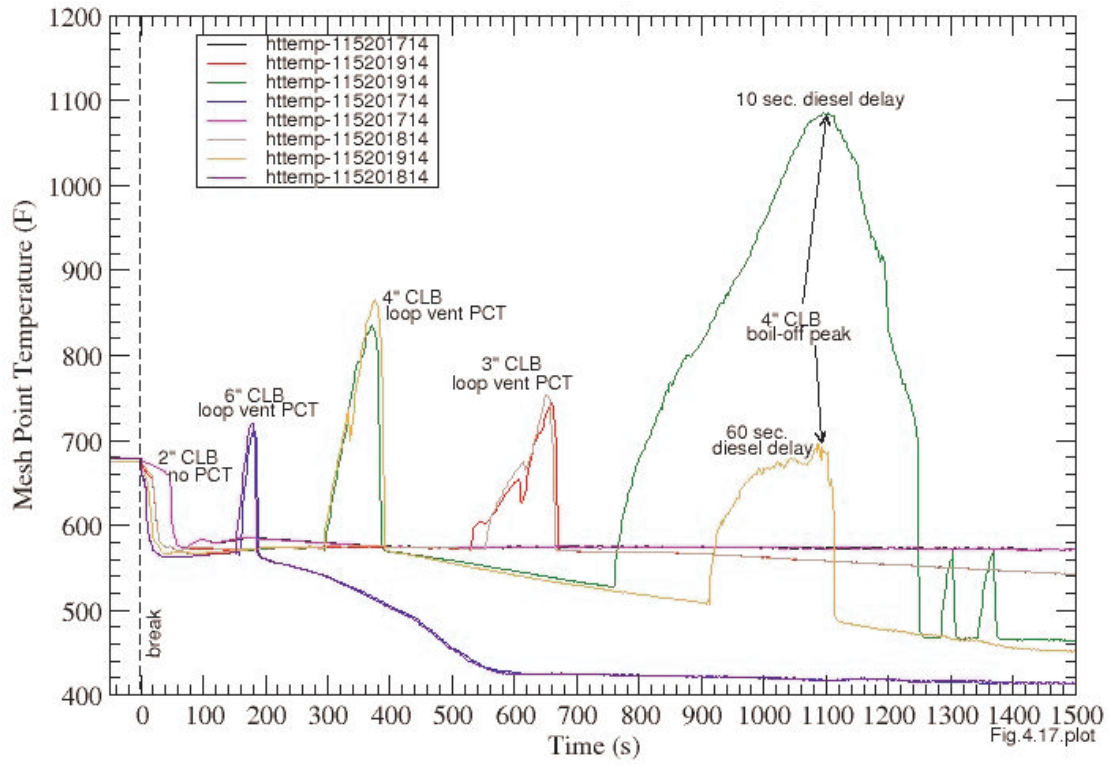


Figure 4.9 Hot Rod Cladding Temperatures  
Small Cold Leg Breaks, no sprays

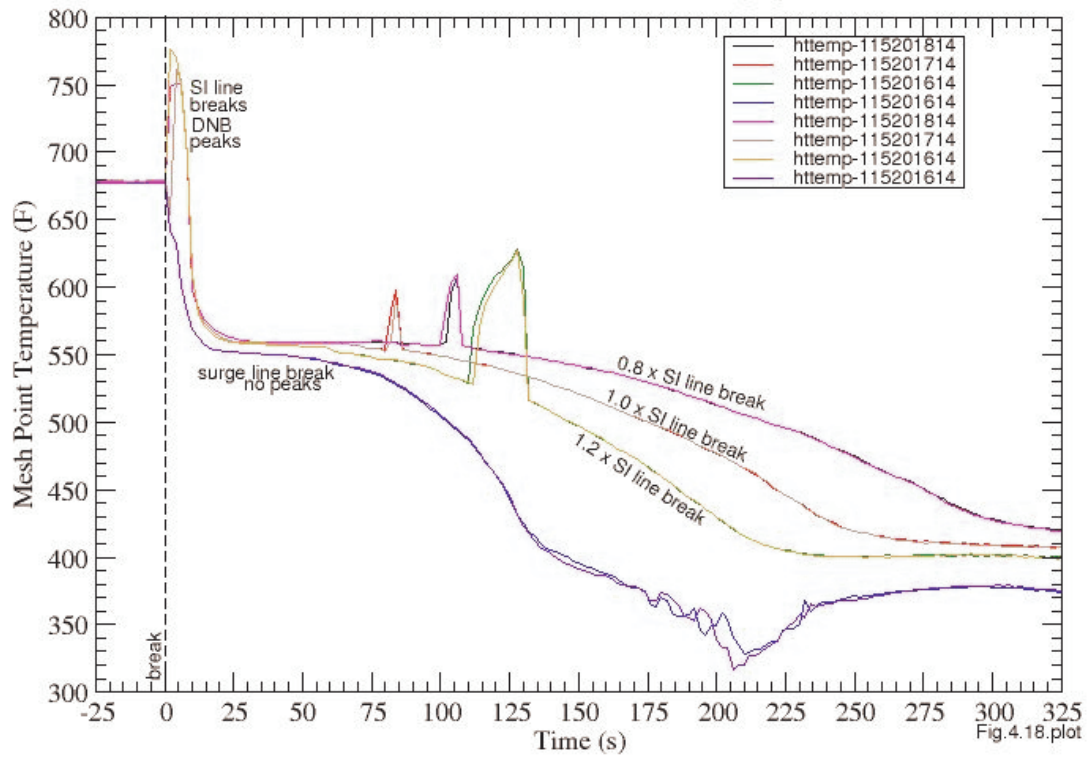
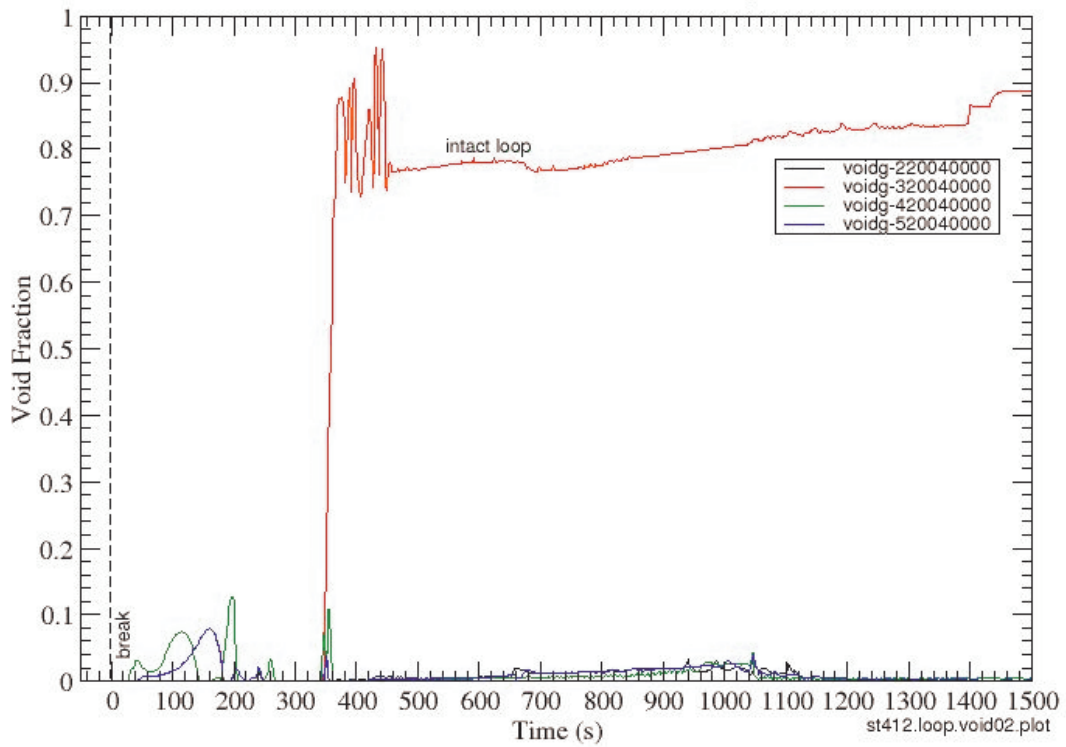


Figure 4.10 Hot Rod Cladding Temperatures  
Accumulator/SI Line Breaks, no sprays

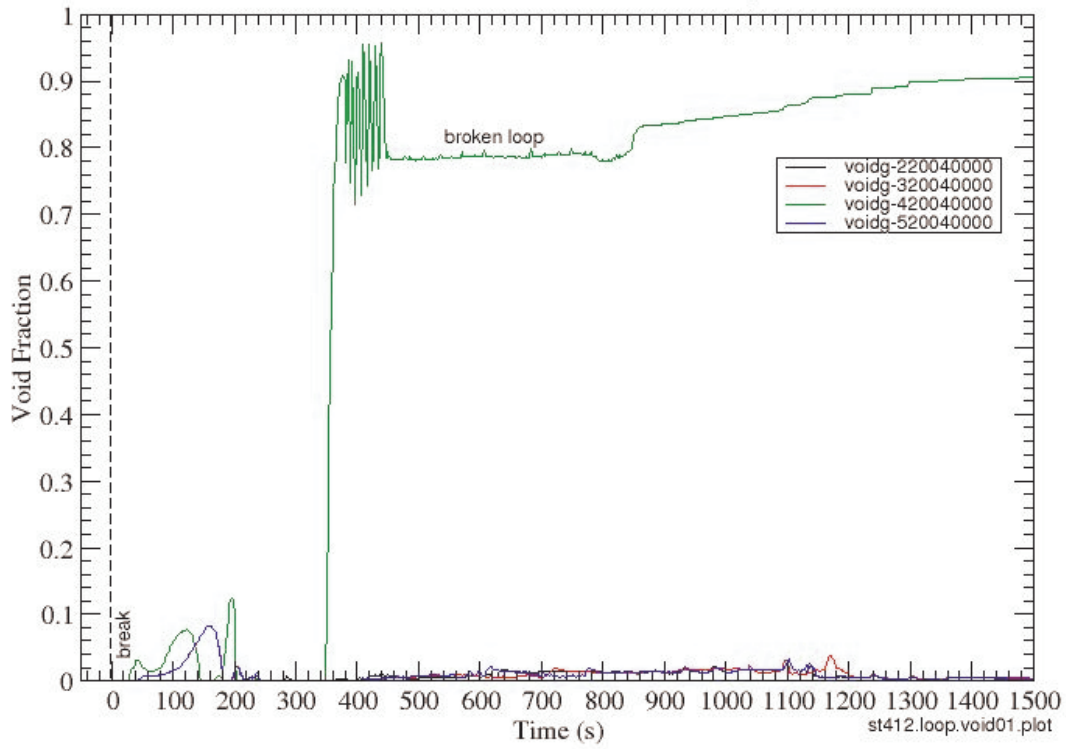
Figure 4.10a. Loop Seal Void Fractions

Std. 412 - 4" CLB (10 sec. diesel delay)



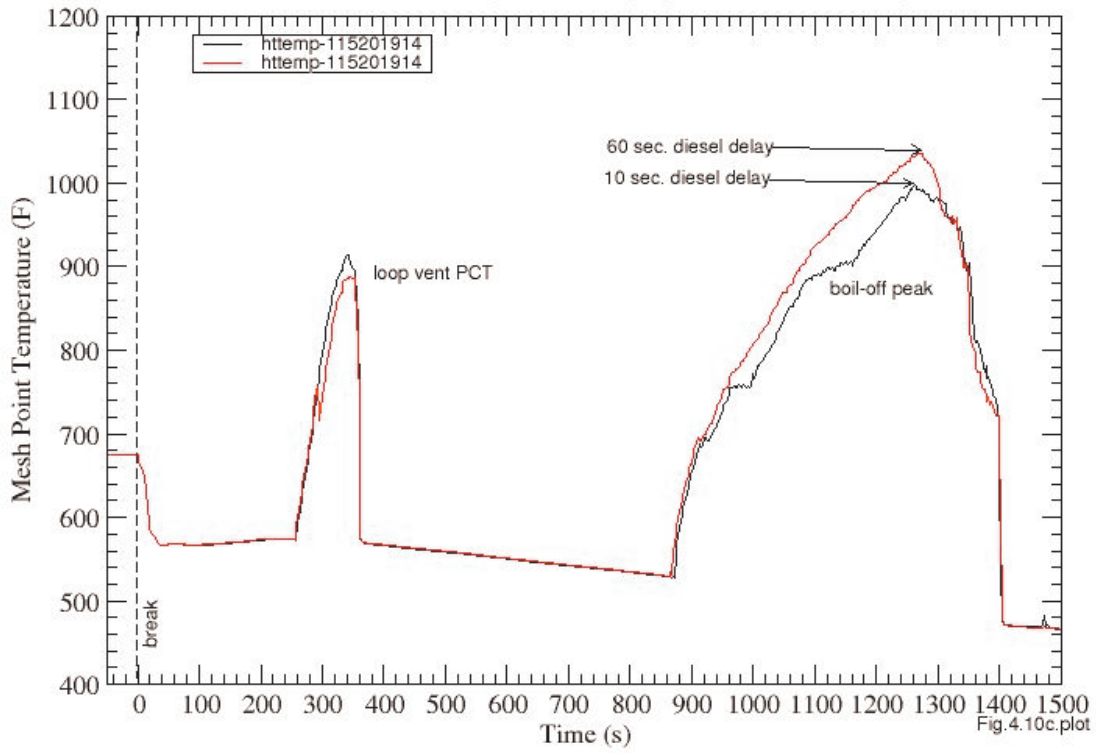
### Figure 4.10b. Loop Seal Void Fractions

Std. 412 - 4" CLB (60 sec. diesel delay)



# Figure 4.10c. Hot Rod Cladding Temperatures

Std. 412 - 4" Cold Leg Break, no sprays, bias one intact loop





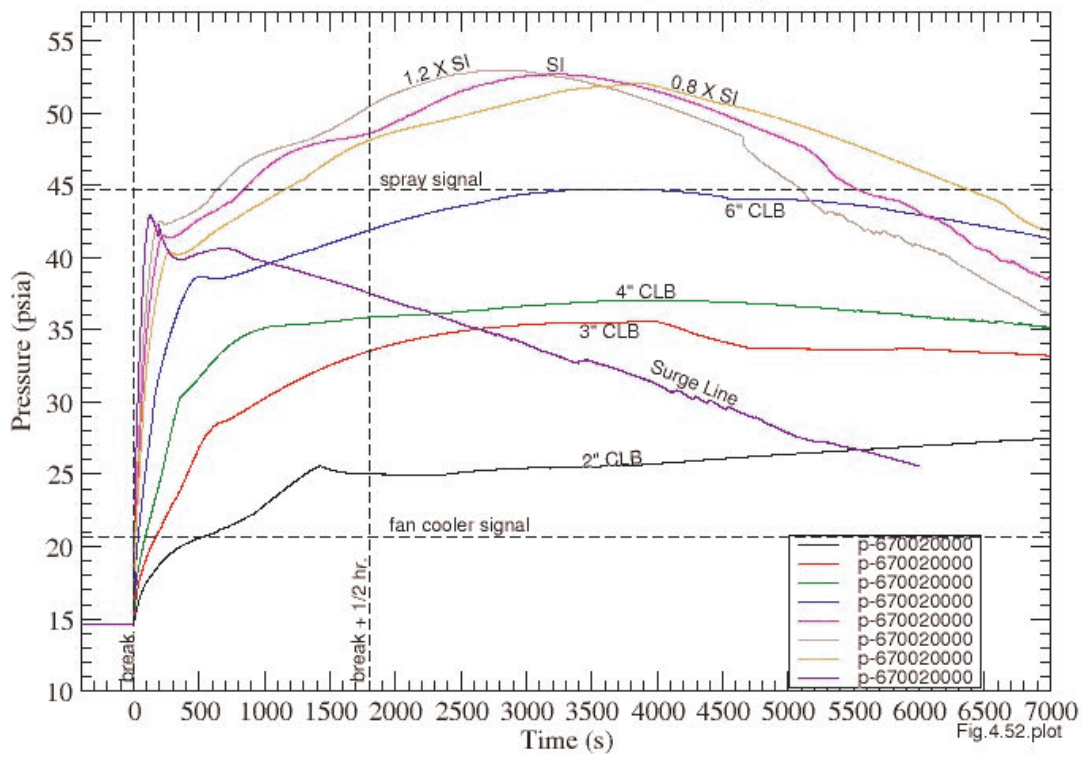


Figure 4.11 Containment Pressure  
Std. 412 (10 sec. diesel delay, no sprays)



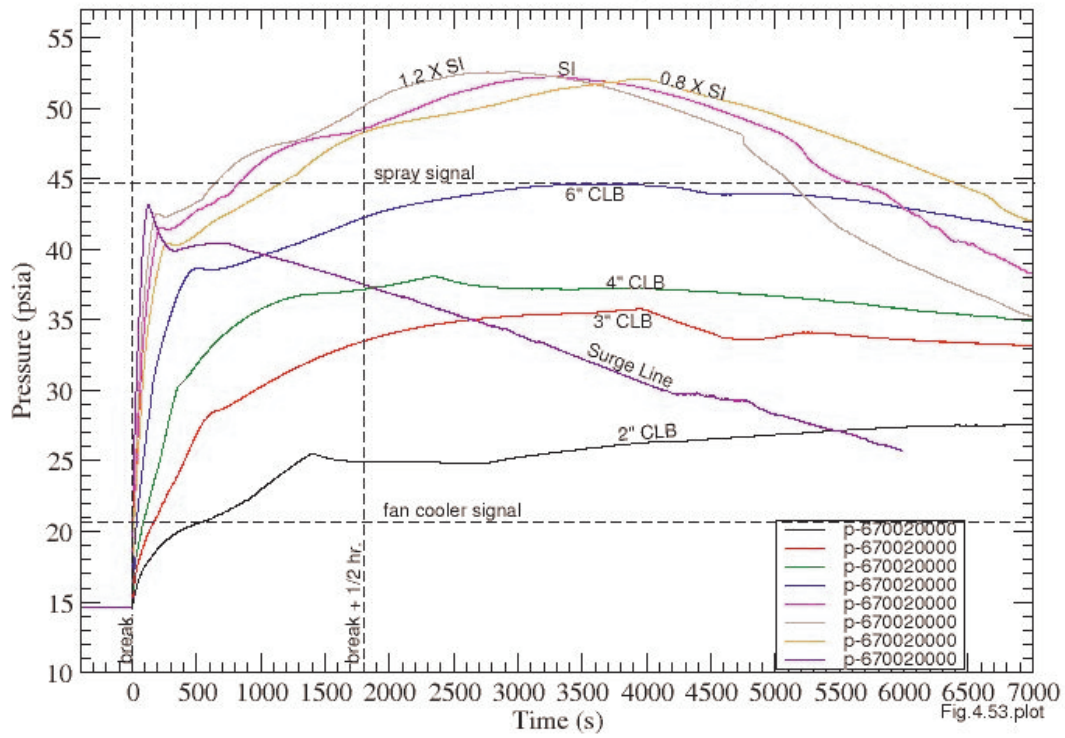


Figure 4.12 Containment Pressure  
Std. 412 (60 sec. diesel delay, no sprays)

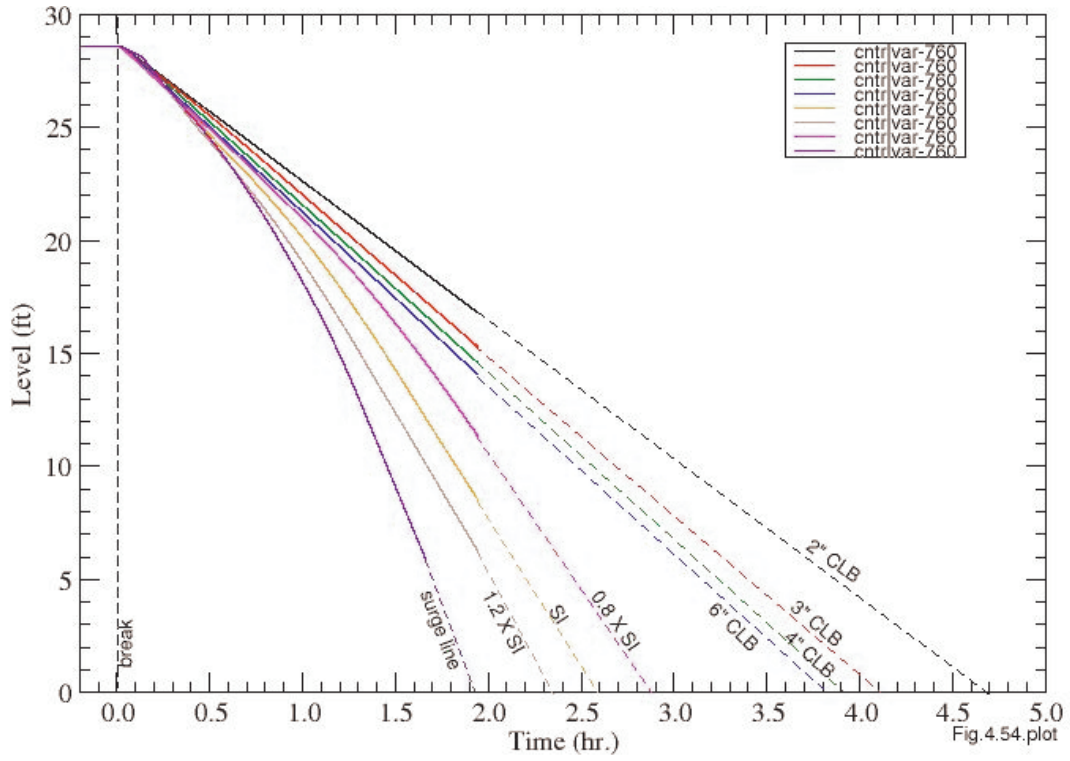


Figure 4.13 Usable RWST Water Level  
Std. 412 (10 sec. diesel delay, no sprays)

Figure 4.14. Usable RWST Water Level

Std. 412 (10 sec. diesel delay, sprays)

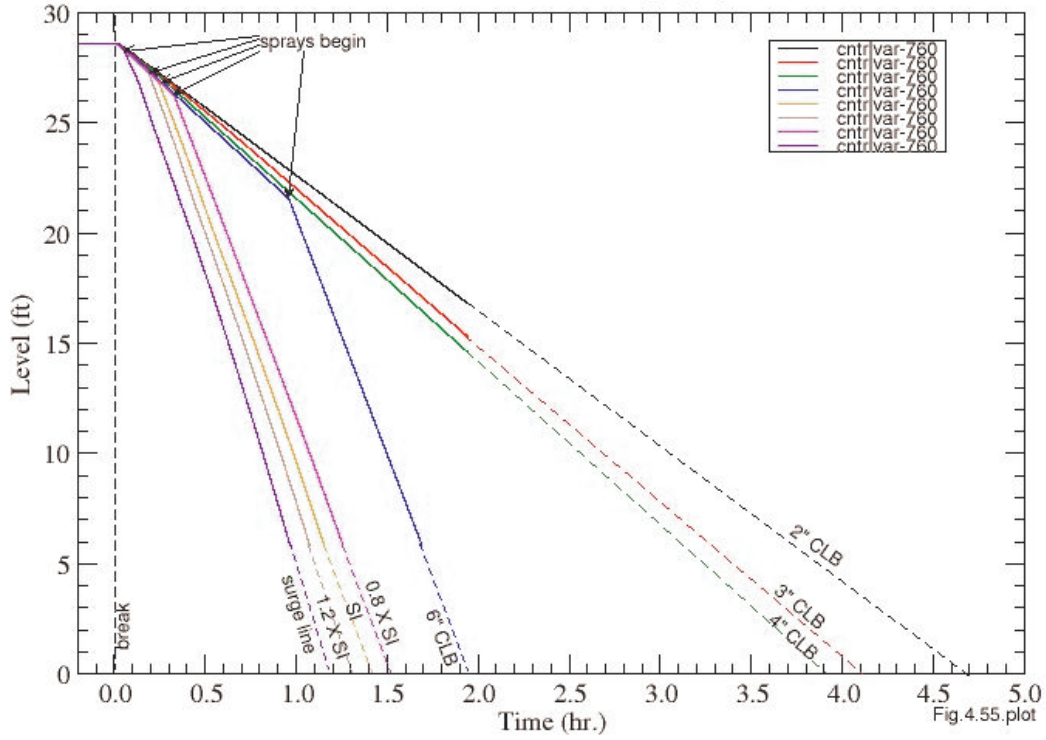


Figure 4.14 Usable RWST Water Level  
Std. 412 (10 sec. diesel delay, sprays)

## 5.0 SUMMARY AND CONCLUSIONS

The RELAP5 and TRACE/CONTAIN calculations discussed in this report support the following conclusions:

1. Increasing the emergency diesel generator (EDG) start up time from 10 seconds to 60 seconds is not expected to be detrimental to core cooling for small break LOCAs and break sizes up to and including major branch lines connected to the primary system for this type of PWR.
2. Delay or elimination of containment spray actuation is feasible for break sizes up to and including major branch lines connected to the primary system.

The sensitivity studies performed with both RELAP5 and TRACE/CONTAIN found that delaying the EDG start up time had only a minor effect on cladding temperatures and system behavior for the range of breaks examined. Small break LOCAs, defined in this report as those breaks with up to a 6-inch equivalent diameter, depressurize relatively slowly and reach a quasi-equilibrium where heat removal is accomplished by the steam generators and the break flow before the loop seals clear. The difference in net contribution to in-vessel inventory by the high head and safety grade charging pumps between the 10 and 60 second delay time cases, while important, is small. For the 2-inch through 4-inch break sizes, which produce the highest peak cladding temperatures, the diesel generator start up is complete before the loop seals begin to clear and any core uncover occurs. The reduction in safety injection in the 60 second delay case results in a small difference in peak cladding temperature. For breaks larger than 6-inches, recovery is primarily due to accumulator injection and the system behavior is unaffected by the selection of choice in EDG start up time.

Thus, with respect to core cooling and sufficiency of the ECCS to meet the 10CFR50.46 fuel acceptance criteria, delaying the diesel start up time from 10 to 60 seconds appears to be feasible for this type of PWR. Plant specific analyses are needed to verify this, but based on the sensitivities exhibited in the RELAP5 and TRACE/CONTAIN calculations, a 60 second EDG delay for small and intermediate breaks should be acceptable.

Calculations to investigate the acceptability of a 60 second diesel start up delay time for large break LOCA were not performed. Calculations to justify a 60 second EDG delay for breaks larger than the connecting branch lines should be performed following the new analysis guidelines for a risk-informed 50.46 option intended to show mitigative capability for these break sizes.

The calculations in previous sections of this report also support a risk-informed revision to containment spray actuation setpoints if breaks are limited to sizes no larger than that of the largest connecting branch lines to the primary system. For small and intermediate sized breaks, which include accumulator/SI line and pressurizer surge line (PSL) breaks, peak containment pressures remained below the containment design pressure without activation of the containment spray system (CSS). Temperatures within containment remained acceptable for the reference design considered in this study, which included safety-grade fan coolers. Plant specific calculations are needed to verify the acceptability of revising the containment spray actuation setpoint for other units.

The effect of delayed CSS actuation on the time to switchover to sump recirculation for breaks at or below the transition break size (TBS) was investigated by running RELAP5 calculations with and without CSS actuation. The calculations showed that for the three smallest breaks (2", 3", 4" CLB), the CSS actuation was not reached, so the RWST water inventory use was the same in both cases. For the 6" break (CLB), not actuating the CSS delayed reaching the minimum RWST inventory by about 2 hours. For the SI line break, the delay was 1.2 hours, and for the PSL break in the hot leg, the delay was 0.7 hours.

In conclusion, increasing the emergency diesel start up time from 10 seconds to 60 seconds, and revising containment spray system setpoints to avoid actuation during a small or intermediate LOCA appears to be acceptable based on the calculations performed in this study for the Westinghouse Standard 412 PWR. Because of the wide variations in ECCS, CSS, and containment types, however, suitable plant specific calculations are necessary to demonstrate this feasibility for any individual unit.

## 6.0 REFERENCES

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