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Westinghouse Containment Analysis Methodology

Westinghouse

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LIST OF ACRONYMS

ADS	Automatic Depressurization System
ANS	American Nuclear Society
ATWS	Anticipated Transient Without Scram
BC	Boundary Condition
BWR	Boiling Water Reactor
DBA	Design Basis Accident
DLM	Diffusion Layer Model
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HPCI	High Pressure Coolant Injection
IBA	Intermediate Break Accident
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
NAI	Numerical Applications Incorporated
NPSHa	Net Positive Suction Head – available
NRC	Nuclear Regulatory Commission
PCT	Peak Clad Temperature
PRFO	Pressure Regulator Failure – Open to Maximum Demand
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RSLB	Recirculation Suction Leg Break
RV	Relief Valve
SBA	Small Break Accident
SBO	Station Blackout
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SP	Suppression Pool
SRP	Standard Review Plan
SRV	Safety-Relief Valve
SV	Safety Valve
UFSAR	Updated Final Safety Analysis Report
WPS	Water Pressure Suppression

ABSTRACT

This report describes the Westinghouse methodology for the calculation of the containment pressure and temperature response to various postulated breaks in the reactor coolant system. Once approved, Westinghouse intends to use this methodology for the following applications:

1. PWR and BWR Containment Design Analysis for Peak Pressure
2. PWR and BWR Containment Design Analysis for Peak Temperature
3. PWR and BWR Containment Design Analysis for Peak Liner Temperature
4. PWR and BWR Minimum ECCS Containment Backpressure Analysis
5. PWR Containment Analysis for Peak Sump Temperature
6. BWR Containment Analysis for Peak Suppression Pool Temperature
7. PWR and BWR Containment Analysis for Thermal Hydraulic Input to ECCS Pump NPSHa Analysis
8. BWR Containment Analysis for Thermal Hydraulic Input to Hydrodynamic Load Analysis
9. PWR and BWR Containment Analysis for Thermal Hydraulic Input to Equipment Qualification Analysis

The break mass and energy release input data is the primary driver for the calculation of the containment pressure and temperature response. The break mass and energy release input data would come from any of the following sources:

1. Mass and energy release data that is calculated by Westinghouse using previously approved models and methods.

2. Mass and energy release data that is calculated by Westinghouse using an approved ECCS evaluation model appropriately and conservatively biased for the specific containment analysis, as described in the Appendices of this report.

3. Mass and energy release data that is provided by the customer that is either calculated by the customer or another vendor using their approved models and methods.

The GOTHIC computer code is used to construct the containment models that are used to calculate the containment response to the release of mass and energy from a postulated break in the RCS. A generic model for each of the various containment designs is described in the Appendices; results of the model qualification (comparison with test data and/or benchmark results from existing analyses) and sample cases for various applications are also included in the Appendices.

The models and methods described in this report cover the PWR and BWR containment designs and follow the guidance of NUREG-0800 (SRP Section 6.2.1) and American National Standard 56.4, where appropriate.

1 INTRODUCTION

Westinghouse has performed loss of coolant accident (LOCA) and main steam line break (MSLB) containment design analysis calculations for many years. The analyses have been performed for PWR large-dry, sub-atmospheric, ice condenser, and passive containment designs. The containment response calculations use plant specific containment models developed using COCO (Reference 1) or CONTRANS (Reference 2) containment analysis codes for the large-dry or sub-atmospheric containment designs, LOTIC (References 3, 4, 5) for the ice condenser containment design, or WGOTHIC (Reference 6) for the passive containment design. The COCO code (Reference 1) was never formally submitted for generic approval and the NRC reviewed and/or performed independent confirmatory analyses for each plant specific containment model.

The plant specific mass and energy release calculations, which provide the primary driver for the containment pressure and temperature response, were performed according to the SATAN/REFLOOD/EPITOME (References 7, 8) or CEFLASH/FLOOD3 (Reference 9) for the LOCA event, and LOFTRAN or RETRAN (References 10, 11), or SGN-III (Reference 12) for the MSLB event.

Several developments have taken place since the previous methods were approved:

- Westinghouse has been developing a new PWR LOCA mass and energy release methodology (Reference 13); the improved method would eliminate some of the non-mechanistic assumptions that are employed in the current approved method.
- Westinghouse has developed and qualified several plant-specific containment analysis models with the GOTHIC code (References 14, 15, 16). The models documented in References 14 and 15 have been reviewed and approved by the NRC for containment design analyses.
- The NRC has reviewed and approved a generic containment analysis methodology for containment design basis analysis applications (Reference 17).
- Westinghouse has been asked by customers to extend its containment design analysis capability to cover the BWR containment designs.

As a result of these developments, Westinghouse is submitting this topical report for review by the Nuclear Regulatory Commission. The topical report describes the Westinghouse methodology for performing design basis containment analyses with GOTHIC and a proposed new methodology for generating LOCA mass and energy input data for the containment response. Once approved, Westinghouse intends to use this methodology for the following applications:

1. PWR and BWR Containment Design Analysis for Peak Pressure
2. PWR and BWR Containment Design Analysis for Peak Temperature
3. PWR and BWR Containment Design Analysis for Peak Liner Temperature
4. PWR and BWR Minimum ECCS Containment Backpressure Analysis
5. PWR Containment Analysis for Peak Sump Temperature
6. BWR Containment Analysis for Peak Suppression Pool Temperature

7. PWR and BWR Containment Analysis for Thermal Hydraulic Input to ECCS Pump NPSHa Analysis
8. BWR Containment Analysis for Thermal Hydraulic Input to Hydrodynamic Load Analysis
9. PWR and BWR Containment Analysis for Thermal Hydraulic Input to Equipment Qualification Analysis

The base report generically describes the methodologies, the regulatory bases and provides a general overview of the GOTHIC input and modeling. The specific models, benchmarks and verification and sample transients for each type of BWR and PWR containment design are provided in the Appendices.

2 REGULATORY REQUIREMENTS FOR CONTAINMENT ANALYSES

The containment system is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environment. The containment structure and associated heat removal systems are designed to accommodate the pressure and temperature conditions resulting from a postulated break in the reactor coolant system or the steam and feedwater system piping without exceeding the design leakage rate. Containment analyses are performed to confirm that the containment structure meets its functional requirements under the conditions of the specific plant's operating limits.

Containment analyses are performed to demonstrate compliance with the General Design Criteria (GDC) listed in Appendix A of 10 CFR, Part 50, in particular, primarily GDC 16, 38, and 50. NUREG-0800 (Reference 18), Section 6.2.1 and its sub-sections provide guidance to the NRC staff for performing their review and the acceptance criteria for the various analyses.

This topical report outlines the methodology for developing plant specific containment models and performing the following types of containment analyses with the GOTHIC computer code:

- Analyses to determine the peak containment pressure and temperature response following a postulated RCS or main steam line break. These analyses are performed to demonstrate compliance with GDC 16 and 50 (with regard to margin to the maximum containment design pressure) and GDC 38 (with regard to the effectiveness of containment heat removal systems).

- Analyses to determine the long-term containment pressure and temperature response. These analyses are performed to demonstrate compliance with GDC 13 and 64 (for equipment qualification), and GDC 38 (with regard to the effectiveness of containment heat removal systems).

The modeling characteristics for performing containment integrity analyses that meet these requirements are generally outlined in the American National Standard ANSI/ANS-56.4-1983 (Reference 19), "Pressure and Temperature Transient Analysis for Light Water Reactor Containments." Westinghouse and the Nuclear Regulatory Commission jointly participated in the development of this industry standard along with other organizations. The ANSI/ANS standard outlines modeling requirements for calculating mass and energy releases and containment performance and is consistent with the NRC's standard review plan in Reference 18.

The modeling characteristics from Reference 19 required for containment integrity calculations are summarized in Table 2-1. Generally, the discussion of these modeling characteristics and assumptions is biased to maximize containment pressure and temperature. However, some analyses are performed to minimize containment pressure and temperature or maximize containment water temperature. Care must be taken to bias the inputs of the analysis to produce the appropriate results for the purpose of the analysis.

Westinghouse intends to use the GOTHIC code to model PWR and BWR containment response to loss of coolant accidents and main steam line break accidents. The Westinghouse containment methodology is summarized with respect to each requirement outlined in the Table 2-1.

Table 2-1 ANS 56.4-1983 Requirements for Containment Integrity Analysis		
Requirements		Westinghouse Methodology
Analytical Considerations		a,c
1	The analysis shall include a spectrum of accident sequences, break sizes, power levels in the reactor coolant system and secondary system to ensure the accident yielding the maximum or minimum pressure and temperature is identified.	
2	Each analysis will be carried out to a sufficient duration to ensure that the maximum pressure and temperature has been ascertained.	
3	For dry and water pressure suppression (WPS) containments, the bounding LOCA peak pressure case will be carried out to ascertain that the pressure is reduced below 50% of the peak calculated pressure (psig) within 24 hours and maintained below this pressure for the duration of the accident.	
4	For sub-atmospheric containments, the analysis will be carried out to show that the containment pressure is reduced below one atmosphere within one hour and maintained below one atmosphere for the remainder of the accident.	
5	The analysis will incorporate the effects of the most severe single failure.	
6	The single failure chosen will be consistent with the single failure chosen for the generation of the mass and energy release data.	
7	The loss of all non-emergency electric power to the plant shall be postulated concurrent with the LOCA pipe break. Continued availability of offsite power is typically limiting for MSLB analysis.	
8	Initial conditions will be chosen to yield a conservative result.	
Thermodynamic Considerations		
9	The dry containment may be modeled with one control volume, which must be capable of distinctly modeling the containment vapor region and the containment liquid water region	
10	The WPS containment shall be developed to track the thermodynamic response in the drywell and wetwell regions.	

Table 2-1 (cont.) ANS 56.4-1983 Requirements for Containment Analysis

a,c

Requirements		Westinghouse Methodology
11	The volume of the containment will be estimated conservatively for the analysis.	
12	The control volume modeling will be based on a conservation of mass and energy.	
13	The containment atmosphere steam and non-condensable components in the vapor region shall be well-mixed and in thermal equilibrium with each other. Drops may be modeled if the treatment of their thermodynamic and mechanical behavior is justified. Non-condensable components are treated as ideal gases. The steam component is treated with steam tables or equations that result in the temperatures and partial pressures within one percent of the steam tables.	
14	The containment sump or wetwell is in pressure equilibrium with the atmosphere.	
15	The containment sump or wetwell temperature is determined from steam tables or suitable equations to determine the thermodynamic state conditions of the water.	
16	Mass and energy transfer across the sump/atmosphere interface need not be modeled unless the sump becomes saturated at the containment pressure and begins to boil.	
17	For WPS containments, the drywell to wetwell vents shall be included in the analysis	
18	The vent model phases are a vent clearing phase and a vent flow phase. During the vent clearing phase, the boundary conditions are the pressures at the liquid interface in the vents and the wetwell. Pressure losses in the vents due to the flow in the vent system shall be accounted for. There shall be no mass and energy exchange assumed between the wetwell pool and the atmosphere in the vents. The vent flow phase follows the vent clearing phase immediately. Vent flow consists of two-component, two-phase mixture of air, steam and water	

Table 2-1 ANS 56.4-1983 Requirements for Containment Analysis (cont.)

Requirements		Westinghouse Methodology
19	Mass and energy transfer across the wetwell atmosphere-pool interface by condensation and evaporation shall be considered infinite unless justified by experiment	
Mass and Energy Release Considerations		
20	The mass and energy from the break flow will be assumed to go directly to the containment atmosphere region	
21	The mass and energy from the spillage flow will be assumed to go directly to the containment liquid region	
22	For the portion of the break flow going to the atmosphere, a phase separation model will be used that will produce a steam addition at a rate at least as large as the rate computed using the assumption of flashing to the saturation temperature at the transient atmosphere steam partial pressure	
23	Other energy sources not previously accounted for will be considered for their influence on the atmosphere region pressure and temperature. This includes sensible heat, potential chemical reaction terms, accumulator nitrogen and other non-condensable gas sources	
Structural Heat Transfer Considerations		
24	The structural heat sinks are to be modeled conservatively with a lower bound estimate of the number, volume and heat transfer area	
25	The heat sinks are assigned to the vapor region or the liquid region by the arrangement of the containment. The assignment may be transient or split based on the geometric configuration of the sump water pool	
26	The rate of heat transfer within a heat sink is determined by the physical arrangement of the conducting layers, their thermal properties, surface properties and boundary conditions. The temperature profile through the heat sink is determined by an appropriate solution of the transient conduction equation	
27	An appropriate value of contact resistance will be modeled where distinct material layers interface within the heat sink	

a,c

Table 2-1 (cont.) ANS 56.4-1983 Requirements for Containment Analysis

a,c

Requirements		Westinghouse Methodology
28	The thermal properties of the materials will be chosen to provide a conservatively low estimate of thermal capacitance and transmission capabilities	
29	Free convection, condensation and (where appropriate) radiation heat transfer will be addressed at the heat sink surfaces. Forced convection is difficult to defend in a 1-D model	
Containment Spray Considerations		
30	Transient heat removal from the containment vapor region due to sprays will be modeled	
31	If the atmosphere is saturated, the sprays will not evaporate	
32	If the atmosphere is superheated, the evaporation rate of the sprays may be assumed to be unlimited	
33	Unevaporated spray water is assumed to go to the sump region at a temperature in equilibrium with the containment atmosphere	
34	The residence time of the sprays in the containment atmosphere may be assumed to be zero	
Containment Heat Removal Considerations		
35	Mass and energy removal rates of containment heat removal components shall be chosen to be representative of the components performance while retaining conservatism in the analysis	
36	Containment heat removal components, such as fan-coolers and recirculation system heat exchangers must model the mechanism and efficiency of energy removal for each component	
37	Sources of coolant for the containment heat removal components are assumed to be at their highest credible temperature throughout the accident. A transient temperature may be used if it can be justified	
38	Sources that can be depleted during the accident are modeled at their lower bound volume	

Table 2-1 (cont.) ANS 56.4-1983 Requirements for Containment Analysis

a,c

	Requirements	Westinghouse Methodology
39	Modeling of heat removal components shall include effects of fouling, condensate build up or any conditions that may be expected to degrade the performance of the component	
40	Steam condensed from the containment atmosphere by cooler units is added directly to the containment sump at the saturation enthalpy of liquid water.	

3 DESCRIPTION OF ACCIDENT SEQUENCES

The following sections provide a brief description of the spectrum of LOCA and MSLB transients that are considered in a containment analysis and a discussion of factors which affect the containment pressure and temperature results.

3.1 PRESSURIZED WATER REACTOR TRANSIENTS

3.1.1 Long Term LOCA Analysis

Following a postulated rupture of the reactor coolant system (RCS) the containment volume receives large mass and energy releases that produce elevated temperature and pressures that may challenge the containment integrity. Bounding mass and energy release rates for a spectrum of breaks are considered to confirm that the design limits of the containment structure are not exceeded. Also, a spectrum of system failures is considered to establish the worst single failure.

The LOCA mass and energy release transient is typically divided into four phases:

- Blowdown – the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS reaches initial equilibrium with containment.
- Refill – the period of time when the lower plenum is being filled by accumulator and safety injection water.
- Reflood – begins when the water from the lower plenum enters the core and ends when the core is completely quenched.
- Post-Reflood (Froth) – describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the steam generators cool, the break flow becomes two phase.

Mass and energy releases are calculated for each phase of the accident and imposed on the containment to produce the pressure and temperature transient results.

<u>Break Size and Location</u>

Generic Studies have been performed with respect to the effect on the LOCA mass and energy release relative to postulated break size. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant loop can be postulated for pipe rupture:

- Hot Leg (between vessel and steam generator)
- Cold Leg (between pump and vessel)

- Pump suction (between steam generator and pump)

Historically, there were six different cases that were considered. These cases were:

- Double Ended Pump Suction – Maximum Safeguards
- Double Ended Pump Suction – Minimum Safeguards
- Double Ended Pump Suction with a Discharge Coefficient of 0.6
- 3 ft^2 Pump Suction Split
- Double Ended Cold Leg
- Double Ended Hot Leg

Based upon established sensitivities and current LOCA mass and energy release methodology, generally only the Double Ended Pump Suction (maximum & minimum safeguards) and Hot Leg cases are analyzed. These three cases have been shown to be more limiting than the other cases.

Application of Single Failure Criteria

An inherent assumption in the generation of the LOCA mass and energy releases is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety systems. Mass and energy releases are generated for minimum and maximum safeguards cases. In the case of minimum safeguards, the single failure postulated to occur is usually the loss of an emergency diesel generator. This results in the loss of one safety injection train and the containment safeguards components on that diesel, thereby minimizing both the safety injection flow and the containment heat removal. For the maximum safeguards case, all safety injection flow is available and the failure of one containment spray pump or one fan-cooler unit, if more limiting, is usually assumed. The number of fan cooler units and spray pumps assumed to be operating in containment for each accident sequence is chosen consistently with the assumptions of the long term mass and energy release analysis.

3.1.2 Main Steam Line Break Mass and Energy Release Analysis

Steam line ruptures occurring inside the containment structure may result in a significant release of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The impact of the releases following a secondary line rupture is dependant upon factors such as break size, plant operating condition, containment design, and plant steam system configuration. The many variations of these parameters make it difficult to assess the single worst case for containment temperature and pressure following a steam line break.

There are four major factors that influence the release of mass and energy following a steam line break: steam generator fluid inventory, primary-to-secondary heat transfer, protection system operation, and the state of the secondary fluid blowdown. The following is a partial list of those plant variables that determine the influence of each of these factors:

- Plant Power Level.
- Main Feedwater System Design.
- Auxiliary Feedwater System Design.
- Postulated break type, size, and location.

- Availability of Offsite Power.
- Safety System Failures.
- Steam Generator Reverse heat transfer and RCS metal heat capacity.

Each of the above variables is generally incorporated in determining mass and energy release data which is supplied for determination of containment temperature and pressure. However, as discussed earlier, the many variations of plant parameters make it difficult to predetermine the limiting configuration. As a result, the mass and energy releases for a variety of postulated pipe breaks and plant configurations are determined and used for analysis. The various breaks and plant configurations are discussed next.

Plant Power Level

Steam line breaks can be postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power levels may result in greater mass released to the containment. However, because of increased energy storage in the primary plant, increased heat transfer in the steam generators, and the additional energy generation in the nuclear fuel, the energy release to the containment from breaks postulated to occur during power operation may be greater than breaks occurring in a hot shutdown condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with increasing power and have significant influence on both the rate of blowdown and the amount of moisture that could be entrained in the fluid leaving the break following a steam break event.

Because of the opposing effects of changing power level on steam line break releases, no single power level can be identified as a worst case initial condition for a steam line break event. Therefore, several different power levels spanning the operating range as well as the hot shutdown condition are typically analyzed.

Postulated Break Type, Size, and Location

A. Postulated Break Type

Two types of postulated pipe ruptures are considered in evaluating steam line breaks: split breaks and double-ended guillotine breaks.

The first type is the split rupture in which a hole opens at some point on the side of the steam pipe or steam header but does not result in a complete severance of the pipe. A single, distinct break area is fed by all steam generators until steam line isolation occurs. The blowdown flow rates from the individual steam generators are interdependent, since fluid coupling exists between all steam lines.

The second break type is the double-ended guillotine rupture in which the steam pipe is completely (or partially) severed and the ends of the break displace from each other. Guillotine ruptures are characterized by two distinct break locations, each of equal area, but being fed in the forward and reverse flow directions by different steam generators.

The type of break influences the mass and energy releases to containment by altering both the nature of the steam blowdown from the piping in the steam plant and the effective break area fed by each steam generator prior to steam line isolation.

B. Postulated Break Size

Break area is also important when evaluating steam line breaks. It controls the rate of releases to containment, as well as exerts significant influence on the steam pressure decay and the amount of entrained water that could exist in the blowdown flow. Steam line breaks are categorized as either a double-ended (large or small) rupture or a split break as noted in the previous section. The size of the double-ended rupture is dependent on the pipe cross-sectional flow area (large) or the amount of entrained water or the containment revaporization assumed for the (small) break. The split break is defined as a break where the steam line isolation occurs on a containment setpoint signal instead of a secondary side signal. Typically the first SI signal is also due to high containment pressure. For these small breaks, an iterative procedure between the containment response and the mass and energy releases is performed to determine the Hi-1 and Hi-2 containment setpoints so that the steamline break mass and energy releases can accurately take credit for the protection signals.

C. Postulated Break Location

Westinghouse does not take credit for line losses. All breaks are postulated immediately downstream of the flow restrictor.

<u>Availability of Offsite Power</u>

For the containment model, loss of offsite power is assumed, as it delays the actuation of the containment heat removal systems due to time to start the emergency diesel generators. This includes diesel initial sequencer loading time. However, for the mass and energy releases calculation, offsite power is assumed to be available, as it maximizes the mass and energy released into containment due to:

- The continued operation of the reactor coolant pumps, which maximizes the energy transferred from the RCS to the steam generator.
- Continued operation of the feedwater pumps and actuation of the auxiliary feedwater system, which maximizes the steam generator inventories available for release.

Analyses may gain margin by providing consistency between mass and energy release analyses and containment integrity analyses offsite power assumptions. Historically, mass and energy release analyses have assumed no loss of offsite power for the reasons previously described. In containment integrity analysis, however, loss of offsite power is generally assumed to conservatively delay actuation of containment heat removal systems. Inconsistent treatment of offsite power between the mass and energy and containment integrity analyses is overly conservative.

Safety System Failures

The following single active failures are typically considered:

- Loss of one emergency diesel (loss of offsite power)
- Vital bus failure (continued offsite power)
- Failure of one main steam isolation valve
- Failure of one main feedwater isolation valve
- Failure of the auxiliary feedwater runout protection system

3.2 BOILING WATER REACTOR TRANSIENTS

3.2.1 Design Basis Accident Analysis

Following a postulated pipe rupture of the reactor system, the primary containment volume receives large mass and energy releases that produce elevated temperature and pressures that may challenge the containment integrity. Bounding mass and energy release rates for the design basis accident (DBA), a double-ended rupture of one of the recirculation lines or steam lines, is considered to demonstrate that the design limits of the containment structure are not exceeded. A spectrum of system failures is considered to establish the worst single failure for the DBA case. Usually, the double-ended rupture of the recirculation suction line results in the maximum drywell and wetwell pressure since this scenario results in the maximum mass and energy release in a short period of time. The maximum atmospheric temperature in the drywell is usually determined for small steam line breaks, which result in the release of steam in the drywell, creating a superheated condition. The maximum atmospheric temperature so calculated, is usually used for environmental qualification of electrical equipment.

The LOCA mass and energy release transient for the DBA in one of the recirculation lines is typically divided into three phases:

- Blowdown – the period from accident initiation (when the reactor is at steady state operation) to the time that the reactor vessel reaches initial pressure equilibrium with containment.

- Refill / Reflood – the period of time when the core spray pumps are delivering spray water above the core and the low pressure coolant injection pumps are delivering water to the lower plenum. As the two-phase level in the core recovers, continued core cooling is provided by the core spray and low pressure coolant injection pumps.

- Long-term Cooling – the period following the refill / reflood transient. The mass and energy releases are generated by decay heat in the core and any sensible energy left in the reactor vessel metal.

Mass and energy releases are calculated for each phase of the accident and imposed on the containment to produce the pressure and temperature transient results.

Break Size and Location

The double-ended rupture in the recirculation suction line is usually the DBA for the BWR containment maximum pressure consideration. Since the DBA does not represent the limiting case for all structural elements, a spectrum of other break locations and break sizes are also considered. Typically, these breaks include an intermediate break accident (IBA) located in a liquid line (e.g., recirculation suction piping) and a small break accident (SBA) located in the steam line.

The DBA in the recirculation line will result in the propagation a pressure wave into the drywell vent system. The increase in pressure and temperature in the drywell accelerates the water, initially standing in the downcomers, into the pool until the downcomers clear of water. During this water clearing process, a water jet forms in the suppression pool. A bubble of air forms at the exit of the downcomers immediately following the downcomer clearing. When the flow of the air/steam mixture from the drywell becomes established in the vent system, the initial bubble expands, and subsequently collapses as a result of over-expansion. During the early stages of this process, the suppression pool will swell in bulk and as the pool continues to swell, the bubble pressure will fall below the torus airspace pressure. However, the momentum of the water slug causes it to continue to rise, compressing the air volume above the pool. The thickness of the water slug decreases as it rises and it eventually evolves into a two-phase mixture of air and water.

The pool swell phase typically lasts on the order of 3 to 5 seconds and is dominated by the flow of drywell atmosphere through the vent system. Steam flow will follow, beginning near the end of the pool swell transient. Throughout these periods, there is a significant pressure differential between the drywell and the torus. When steam flows through the vent system, pressure oscillations will occur in the vent system and the suppression pool. The oscillations are caused by condensation of steam at the steam / liquid interface near the exit of the downcomers. Initially, the steam flow rate through the vent system is high enough to prevent water in the pool from re-entering the downcomer. However, as the steam flow decreases toward the end of blowdown, the pool water will re-enter the downcomer intermittently setting up irregular pressure pulses (chugging).

After the end of blowdown during the refill / reflood phase of the DBA, ECCS water is injected into the reactor vessel, a portion of which spills into the drywell. This causes the steam in the reactor vessel and drywell to condense, which causes the drywell pressure to decrease. As the drywell pressure falls below the suppression chamber pressure, the vacuum relief system allows non-condensables from the suppression chamber to flow into the drywell.

In the long term, suppression pool water is continuously recirculated through the core by the LPCI and core spray pumps. The energy associated with the core decay heat will result in a slow suppression pool heatup. Operators will initiate suppression pool cooling to control suppression pool temperature.

As the break size decreases, the rate at which the reactor depressurizes slows down and consequently the phenomena associated with clearing of the downcomer become less severe. For those break sizes, which are large enough that the HPCI system cannot maintain the reactor vessel water level, the automatic depressurization system (ADS) actuates to depressurize the reactor.

The SBA is small enough so that HPCI operation can maintain the reactor vessel water level. However, this break will not result in rapid reactor depressurization. Therefore, this event will result in a long duration combination of chugging and multiple SRV discharge loads. This event is assumed to be terminated by operator action to manually actuate the automatic depressurization system (ADS).

Application of Single Failure Criteria

An inherent assumption in the generation of the RSLB LOCA mass and energy releases is that offsite power is lost simultaneously with the pipe break. This results in the actuation of the emergency diesel generators, required to power the safety systems. The mass and energy releases for a particular application are determined in a manner that is conservative, which includes consideration of the break size, break location and single active failure.

Active containment heat removal systems are initiated by the operator. The containment heat removal for each accident sequence is chosen consistently with the ECCS assumptions of the long term mass and energy release analysis.

4 GOTHIC CODE DESCRIPTION

The GOTHIC code is becoming the industry standard for performing containment analyses, as well as analyses for auxiliary buildings outside containment. The code has been developed by Numerical Applications Incorporated (NAI) with funding by the Electric Power Research Institute (EPRI). The GOTHIC code consists of a pre-processor for input generation; a solver, which performs the calculations; and a post-processor, which produces output data tables and plots. The GOTHIC Technical Manual (Reference 21) provides a description of the governing equations, constitutive models, and solution methods in the solver. The GOTHIC Qualification Report (Reference 22) provides a comparison of the solver results with both analytical solutions and experimental data. The GOTHIC User Manual (Reference 23) provides information to help the user develop models for various applications.

GOTHIC solves the integral form of the conservation equations for mass, momentum, and energy for multi-component, two-phase flow. The conservation equations are solved for three fields; continuous liquid, liquid drops, and the steam/gas phase. The three fields may be in thermal non-equilibrium within the same computational cell. This treatment allows the modeling of subcooled drops (e.g., containment spray) falling through an atmosphere of saturated steam. The gas component of the steam/gas field may be comprised of up to eight different non-condensable gases with mass balances performed for each component. Relative velocities are calculated for each field as well as the effects of two-phase slip on pressure drop. Heat and mass transfer between the phases, surfaces, and the fluid is also allowed.

The GOTHIC code is capable of performing calculations in three modes. The code can be used in the lumped parameter nodal network mode, the two-dimensional finite difference mode, and the three-dimensional finite difference mode. Each of these modes may be used within the same model. The capability of multi-dimensional analyses greatly enhances the ability to study non-condensable gases and stratification as well as allowing the calculation of flow field details within any given volume. Most of the GOTHIC containment models described in this report use lumped parameter volumes.

The GOTHIC code also contains the options to model a large number of structures and components. These include, but are not limited to, heated and unheated conductors, pumps, fans, valves, heat exchangers, and ice condensers. These components can be coupled to represent the various systems found in any typical containment.

4.1 GOTHIC VALIDATION

The GOTHIC code has undergone extensive review and validation against an array of tests (Reference 22). The code has been validated against a number of Battelle-Frankfurt tests performed to study steam blowdowns and hydrogen releases. A number of Hanford Engineering Development Laboratory (HEDL) tests were modeled to simulate steam-hydrogen jets. The LACE tests were modeled to validate rapid depressurization events with aerosols. Several of the Heissdampfreaktor (HDR) full scale containment tests were modeled to study steam and water blowdowns and hydrogen releases in a full scale multi-compartment containment geometry.

The flexible noding and conservation equation solutions in GOTHIC allow its application to a wide variety of problems. GOTHIC models have been used to study hydrogen distributions, calculate containment pressure and temperature transients, and perform flow field calculations for particle transport purposes, and surge-line flooding studies for loss of RHR cooling events during shutdown operations.

GOTHIC transient results have been compared with results from other containment design analysis codes (COCO, CONTEMPT, CONRANS, CONTAIN and COPATTA). The Westinghouse benchmark comparisons with COCO, CONTEMPT and CONTRANS results are presented in References 14 through 16. Differences between the GOTHIC results and the results from other codes are attributed to the ability of GOTHIC to better model droplet phase interface heat and mass transfer.

GOTHIC has been reviewed and approved by the NRC for containment design basis analysis applications for several U.S. utilities (References 24 through 30). This list is not intended to be complete, but to demonstrate that GOTHIC modeling has been accepted numerous times for containment analyses.

The containment analysis models and methods described in this report are not intended to be restricted to a specific GOTHIC code version. The code is being continuously maintained and updated by EPRI and NAI to include new features and/or correct problems. Therefore, although the containment models and methods described in this report were developed using GOTHIC version 7.2a, Westinghouse intends to use future versions of GOTHIC for plant specific containment analyses as they become available.

4.2 GOTHIC CONTAINMENT MODEL INPUT

A GOTHIC containment model is built using control volumes, flow paths, thermal conductors, and boundary conditions. Components, such as coolers, heat exchangers, valves, nozzles, and pumps may also be included in the model. This Section describes the general containment model input for GOTHIC. More detailed input descriptions can be found in the GOTHIC User Manual (Reference 23).

4.2.1 GOTHIC Modeling Parameters

4.2.1.1 Revaporization Fraction

The revaporization fraction is the fraction of condensate that re-vaporizes from the surfaces of the thermal conductors. For the recommended condensation heat transfer model (diffusion layer model with no mist), the revaporization fraction is set to DEFAULT. GOTHIC will calculate the condensation that is re-vaporized from the surface of the heat sink. The default revaporization option was used for all of the validation cases in the GOTHIC Qualification Report (Reference 22).

4.2.1.2 Fog and Mist Modeling

The fog model is turned OFF to enable the mist model in GOTHIC. The mist model creates mist in the containment vapor space when the atmosphere becomes supersaturated. The maximum mist density is set to DEFAULT (1 g/m^3). Excess mist will create drops when the mist density exceeds the maximum. Drop diameter from mist is also set to DEFAULT, which creates 200 micron diameter drops.

4.2.1.3 Minimum Heat Transfer Coefficient

The minimum heat transfer coefficient specifies a lower limit on the convection heat transfer coefficient that applies to liquid-vapor interfacial heat transfer at a pool surface. It is set to the default value of 0.0 Btu/hr- ft^2-°F.

4.2.1.4 Reference Pressure

The reference pressure is set to IGNORE or DEFAULT to use the local pressure and temperature to calculate density in the static head terms of the steam/gas momentum equation.

4.2.1.5 Forced Entrainment Drop Diameter

Diameter of entrained drops for specified entrainment modeling in subdivided volumes. This parameter has no impact because forced entrainment is not specified in the modeling. This value is set to DEFAULT.

4.2.1.6 Vapor Phase Head Correction

The value of this parameter is set to INCLUDE. The static head of the pool liquid that is above the cell center line is subtracted from the vapor phase pressure, and the calculated results are physically more realistic than results when this parameter is set to IGNORE. This parameter has no effect in volumes that do not contain deep pools.

4.2.1.7 Kinetic Energy

The kinetic energy transport and storage in the fluid energy equations is set to IGNORE. The kinetic energy is included in the calculated mass and energy releases.

4.2.1.8 Vapor, Liquid and Drop Phases

Vapor, liquid and drop phases are all modeled in the GOTHIC containment analysis. The values of each parameter are set to INCLUDE.

4.2.1.9 Force Equilibrium

If set to INCLUDE, GOTHIC will force thermal equilibrium between the phases in the control volumes and homogeneous flow (equal phase velocities) in all junctions. The value is set to IGNORE to allow the code to perform non-equilibrium calculations of interfacial heat transfer between the vapor and liquid phases. Thermal equilibrium conditions can be modeled in an individual volume by setting the liquid-vapor interface area input value to a large number.

4.2.1.10 Drop-Liquid Conversion

The value of Drop-Liquid Conversion is set to INCLUDE to allow drop phase entrainment, agglomeration and deposition in the GOTHIC model.

4.2.2 Control Volume Input

Control volumes are used to represent the regions of the containment that are occupied by a fluid or ice. The fluid may be steam, non-condensing gases (air, hydrogen, etc.), water or any combination of these fluids. GOTHIC requires an input value for the volume, elevation, height, hydraulic diameter, and liquid-vapor interface area for each component in the model. The calculation of these input values is described below.

The component volume input value represents the free volume occupied by the atmosphere and water within that component. The free volume can be difficult to calculate because of the complex shapes of the structures inside the containment. Typically, the gross volume inside each control volume is calculated using the inner dimensions of the containment. The volume of the equipment located within the containment is subtracted from the gross volume to get an approximate free volume for the containment analysis; an uncertainty is also defined. For containment peak pressure and temperature analyses, uncertainty is subtracted from the free volume to minimize the calculated free volume; this will allow the pressure to be calculated conservatively high. For containment minimum ECCS backpressure or minimum NPSHa analyses, uncertainty is added to the free volume; this will allow the pressure to be calculated conservatively low.

The control volume elevation input value represents the elevation at the bottom of the containment. The elevation can be taken directly from a drawing or be based on an arbitrarily assigned reference value.

The control volume height input value represents the difference in elevation between the bottom and top of the component. The component height input value is used for several purposes: to determine a nominal floor area, to determine the film thickness for the interfacial heat and mass transfer calculation and as the fall height of liquid drops in lumped parameter volumes. The nominal floor area is calculated in GOTHIC by dividing the component volume by its height and is used to calculate the liquid level within the component. The film thickness is determined from the liquid volume and the wetted surface. The wetted surface is the maximum of the pool area and the wettable surface calculated from the hydraulic diameter of the control volume.

The component hydraulic diameter input value is used in GOTHIC to determine the maximum wetted surface area of thermal conductors within the component. It is also used to define the maximum bubble and drop diameters within the component. The component hydraulic diameter input value is calculated as follows:

$$D_H = \frac{4V}{A_S} \qquad \text{(Equation 1)}$$

where:

V	=	the component volume
A_s	=	the total surface area of all of the thermal conductors within the component.

The component interface area input value represents the area for heat and mass transfer between the liquid and vapor phase within the component. The default value is the maximum of the nominal floor area (component volume/height) and the wetted surface area. A very large input value can be used to force the component liquid and vapor phases to be in thermal equilibrium. Likewise, a very small input value can be used to force the component liquid and vapor phases to be thermally isolated from one another. The interface area is generally set to default, or in the case of the BWR Mark I torus, to a large value to force equilibrium.

Initial pressure, temperature, relative humidity and water volume conditions are specified for the control volumes to conservatively bias the results for the particular analysis that is being performed.

4.2.3 Flow Path Input

GOTHIC flow paths are used to connect control volumes to one another or to connect boundary conditions to control volumes. Flow paths transport liquid, vapor, and droplet mass and energy between the control volumes or between boundary conditions and control volumes.

GOTHIC requires input values to specify the flow path elevations, end height, hydraulic diameter, area, friction length, inertia length, and loss coefficient. In junctions where the flow is specified by a flow boundary condition, such as break flow or spray flow, the flow paths act as junctions that direct the flow to the appropriate lumped parameter control volume. In this case, the momentum equation is not being solved by GOTHIC and most of the required junction parameter values can be assigned arbitrary values. In junctions where the flow rate must be calculated, such as pump suction piping in a minimum NSPHa calculation or vent flow rates to a suppression pool, the area, friction length, inertial length and loss coefficients must be appropriately modeled to calculate the correct flow rates and pressure drops. The calculation of junction parameters is described below.

For each end of a flow path, elevation and end height must be specified such that the junction end falls within the elevation range of the control volume to which it is connected. For a flow path taking suction from the containment sump, the elevation and end height are specified so that the junction is submerged in the liquid region of the containment control volume. For a flow path injecting fluid to the containment vapor space, the flow path is connected at an elevation in the vapor region of the control volume. In a lumped parameter control volume, the model is not sensitive to the elevation since the vapor space is assumed to be instantaneously well-mixed. Flow paths and control volumes modeling vents into a suppression pool or wetwell must be modeled carefully with respect to control volume elevation and water level for the purpose of capturing vent clearing characteristics.

The flow area of a junction is the minimum cross sectional area connecting the volumes. The fluid velocities through the junction are based on this area. For junctions that connect a flow boundary condition to a lumped parameter volume, the flow area is arbitrarily assumed to be 1.0 ft^2.

The hydraulic diameter of the junction is used to calculate wall friction head. The hydraulic diameter of the flow path is defined by the equation:

$$D_H = \frac{4A}{P_W} \qquad \text{(Equation 2)}$$

where:

A = the flow area, and
P_W = wetted perimeter of the junction.

For cylindrical junctions, D_H is the physical diameter of the junction. For junctions that connect a flow boundary condition to a lumped parameter volume, the hydraulic diameter is not used since there is no momentum balance solved. It is arbitrarily assumed to be 1.0 ft.

The inertial length of the junction determines the inertia of the junction. Combined with the flow area, the inertial length defines an effective volume of the junction. The inertial length is set to the distance between control volume centers. For junctions that connect a flow boundary condition to a lumped parameter volume, the inertial length is not used. It is arbitrarily assumed to be 1.0 ft.

The frictional length of the junction is used to calculate the wall friction. The frictional length can be included in the fL/D term in the loss coefficients, and the default value is zero. For junctions that connect a flow boundary condition to a lumped parameter volume, the frictional length is not used. It is arbitrarily assumed to be 1.0 ft.

The loss coefficient determines the velocity head loss coefficient, K, in the junction defined by the term:

$$\Delta P = K\frac{\rho V^2}{2} \qquad \text{(Equation 3)}$$

where:

ΔP = the pressure drop
ρ = the density of the fluid, and
V = the velocity of the fluid.

The loss coefficients include the form losses, entrance and exit losses of the junction. If the friction length is set to zero, the friction loss is also included. For junctions that connect a flow boundary condition to a lumped parameter volume, the loss coefficient is not used since there is no momentum balance solved. It is arbitrarily assumed to be 0.0.

4.2.4 Structural Heat Sinks

Structural heat sinks in the containment model are represented by GOTHIC thermal conductors. Generally, one-dimensional wall thermal conductors are used to model the structures in the lumped parameter containment volume. Heat and mass transfer between the control volume vapor or liquid regions and the structures are specified by the heat transfer coefficient types assigned to each of the thermal conductors. This section describes the thermal conductor models and the heat transfer coefficient input to the GOTHIC code.

Thermal conductors are comprised of any number of material layers. A material in GOTHIC is defined by its density, thermal conductivity and specific heat, each of which may be input as a function of temperature. Material properties are selected and biased based on the specific application of the model. For example, for peak pressure analyses, the material properties would be biased low to predict the lower bound conduction rate through the structural heat sinks. For minimum containment backpressure calculations, the material properties are biased high to predict the upper bound conduction rate.

When assembling the layers of a thermal conductor, gap conductance between dissimilar, non-adhering layers in a thermal conductor such as steel-lined concrete is modeled as an air gap. The thickness of the air gap in the model is defined by the effective gap conductance assumed between the layers and the value used for the thermal conductivity of the air comprising the gap. The value of the gap conductance between layers is biased for the specific application.

The thermal conductor layers must be subdivided appropriately to calculate the temperature profile through the layers of the conductor. The GOTHIC preprocessor provides an "auto-divide" utility to automatically subdivide the thermal conductor layers appropriately based on the expected magnitude of the heat flux at the thermal conductor surface and the thermal conductivity of the thermal conductor.

The surface areas assigned to the thermal conductors are based on an assessment of the containment walls and internal structural. The heat sink surface areas are biased conservatively for the specific application. Depending on how the heat sink geometry data was defined, the GOTHIC thermal conductor can be modeled as a full thickness slab with both sides exposed to the containment or a double area, half thickness slab with one side exposed to the containment and the second side assumed to be adiabatic.

Heat transfer coefficients are assigned to the thermal conductor surfaces exposed to the containment. Natural convection, condensation and radiation are credited in the heat transfer coefficient. Forced convection is not modeled. Typical heat transfer coefficient types defined for LOCA and MSLB containment analysis are:

1. The direct heat transfer option with diffusion layer modeling (DLM) condensation (Reference 20), vertical surface natural convection and vapor phase option is used for all thermal conductors exposed to the containment vapor layer. A multiplier is used to conservatively increase the heat transfer rate for the minimum ECCS backpressure and minimum NPSHa analyses.

2. The direct heat transfer option with DLM, vertical surface natural convection and split phase modeling for thermal conductors for sump or suppression pool walls. The liquid fraction for the transition from vapor to liquid heat transfer is determined by the volume of the sump and the volume of the containment. A multiplier is used to conservatively increase the heat transfer rate for the minimum ECCS backpressure and minimum NPSHa analyses.

3. An adiabatic heat transfer coefficient for the back side of one-sided slab heat sinks or insulated surfaces.

4. A constant 2.0 Btu/hr-ft^2-°F heat transfer coefficient to model natural convection heat transfer from the outside of the containment shell to the outside air.

In the GOTHIC Qualification Report (Reference 22), the DLM for condensation heat transfer has been shown to under-predict the condensation rate. The DLM has been previously accepted by the NRC for LOCA and MSLB peak pressure/temperature containment analyses (Reference 25).

4.2.5 Containment Sprays

GOTHIC models containment spray water injected from the refueling water storage tank into the vapor region of the containment. A flow boundary condition linked to the containment control volume with a flow path is used to specify the spray flow rate, temperature of the water source and mean spray drop diameter. GOTHIC calculates the mechanistic heat and mass transfer at the droplet-vapor interface (Reference 21).

In lumped parameter modeling, the spray drops instantaneously fills the entire volume of the containment vapor space when it enters the sprayed control volume. The spray fall height parameter may affect the spray effectiveness by determining the droplet residence time in the vapor space, which impacts the drop concentration and the effective heat capacity of the atmosphere. However, this parameter is not particularly sensitive in the containment application because the fall height is typically large enough that the spray water reaches equilibrium with the containment atmosphere. The spray fall height is specified as the height of the containment volume in the control volumes table.

The spray drop diameter determines the spray fall velocity and heat transfer coefficient. Spray nozzles typically deliver a spectrum of spray drop sizes. The smaller drops fall more slowly and reach equilibrium more quickly than larger drops because of the higher surface to mass ratio. GOTHIC models sprays with a single drop diameter. The mean drop diameter is specified as the plant specific Sauter mean diameter as specified in the GOTHIC user's guidance (Reference 23) or as a conservative value for the analysis being performed.

The initial spray flow is set to zero. The sprays are initiated as an operator action or automatically from a high containment pressure signal that is specified as a GOTHIC "Trip" and assigned to the spray boundary condition. A delay time after the pressure setpoint is reach is specified within the trip. Generally, a loss of offsite power is assumed to occur coincident with the LOCA initiating event. The delay time is associated with the start up time of the emergency diesel generators and spray pumps. The duration of the injection sprays is calculated considering the number of spray pumps operating (consistently with respect to the mass and energy release single failure assumptions) and a conservative estimate of the water source volume.

The spray parameters, operator actuation timing, spray setpoint and the delay time and spray duration are chosen to conservatively bias the spray heat removal for the particular analysis being performed.

4.2.6 Containment Fan Coolers

GOTHIC is able to model containment fan coolers in plants that have emergency fan cooler units. Fan coolers may be modeled with a GOTHIC "cooler" model or as a coupled model of the containment fan and cooling coil unit using a heat exchanger model.

The GOTHIC cooler model allows the user to specify the heat removal and may be used if appropriately conservative heat removal data is provided from an outside source. Heat removal data is typically provided as a function of the containment saturation temperature for a specific bounding cooling water temperature and flow rate. The heat removal function and the containment vapor flow rate through the cooler are the parameters provided to the cooler model. The heat removal data is entered into GOTHIC as a function table or control variable and the code interpolates the heat removal rate based on the containment saturation temperature.

The fan cooler model is placed in the containment control volume vapor space and removes heat from the vapor flowing through the cooler. If the heat is being transferred directly from the fan coolers to the ultimate heat sink, the heat is simply removed from the model. If the heat is being transferred to a cooling water system that is explicitly modeled, a GOTHIC heater model is used to transfer the heat to the cooling water system. The heater model is coupled to the cooler via a control variable to conserve energy in the model.

If a more detailed fan cooler model is required, the fan coolers may be modeled physically by specifying a volumetric flow of the containment vapor through a vapor-to-water heat exchanger model of the fan cooler coils. Cooling water is modeled on the secondary side of the coil heat exchanger. The detailed geometric design of the coils (number of tubes, materials, thickness, number of passes, flow patterns and fin design) is used to calculate the heat removal using standard heat exchanger principles. The physical fan cooler model is benchmarked with available heat removal data to confirm that the performance is appropriate and conservative for the analysis being performed.

The fan-coolers are initiated from a high containment pressure signal that is specified as a GOTHIC "Trip" and assigned to the cooler model. A delay time after the pressure setpoint is reach is specified within the trip. Generally, a loss of offsite power is assumed to occur coincident with the initiating event. The delay time is associated with the start up time of the emergency diesel generators and fan cooler motor. The number of operational fan coolers is considered consistently with respect to the single failure assumptions of the mass and energy releases.

4.2.7 Residual Heat Removal and Recirculation Sprays

Flow boundary conditions are used to model the recirculation of core cooling water and containment sprays from the containment sump. Liquid-to-liquid heat exchanger models calculate the heat removal from the containment recirculation and recirculation spray paths. The residual heat removal and spray heat exchangers are typically cooled with component cooling water, which in turn is cooled through a liquid-to-liquid heat exchanger model cooled by service water that provides the ultimate heat sink. A conservative bounding service water temperature is assumed for the specific analysis. Each of these coolant loops may be modeled explicitly in GOTHIC to analyze the system performance during the transients.

The heat exchangers may be modeled using detailed physical parameters of the tube heat transfer area, tube material properties, primary and secondary flow areas, hydraulic diameters, and fouling resistances. The heat transfer coefficients on the primary and secondary sides of the heat exchanger are calculated using the Dittus-Bolter correlation. Otherwise, the heat transfer area and constant heat transfer coefficient of the heat exchanger can be specified to model an appropriate and conservative UA value.

The heat exchanger models used in the residual heat removal, recirculation spray, component cooling water and service water system models are benchmarked against available design data, operational data, and/or previous analysis to confirm that the systems are modeled appropriately and conservatively for the specific analysis being performed.

5 MASS AND ENERGY RELEASES TO THE CONTAINMENT

The mass and energy release input data is the primary driver for the calculation of the containment pressure and temperature. This section describes how the mass and energy release input for the containment model is generally calculated. Specific modeling for the different plant types and events is outlined in the Appendices.

5.1 REQUIREMENTS

NUREG-0800, Section 6.2.1.3 (Reference 18) documents an acceptable practice for calculating the LOCA mass and energy release input data. The SRP specifies that the sources of energy available for release are to be based on 10 CFR Part 50, Appendix K, paragraph I.A. A list of the requirements given in NUREG-0800, Section 6.2.1.3 is shown in Table 5.1-1.

ANSI/ANS 56.4-1983 (Reference 19) also provides guidance for developing conservative input for the mass and energy release calculation in accordance with the acceptable practice documented in NUREG-0800, Section 6.2.1.3. A list of the recommendations in ANS 56.4 is shown in Table 5.1-2.

A discussion of how the mass and energy release analysis methodologies comply with these requirements is presented in the Appendix B of this WCAP for BWRs and Appendix E of this WCAP for PWRs.

5.2 BLOWDOWN, REFILL AND REFLOOD PHASES OF THE RELEASE

Mass and energy releases from the reactor coolant system during the blowdown, refill, reflood phases of the accident will be provided to the containment model using an NRC approved evaluation model. These methods include:

1. Mass and energy release data that is calculated by Westinghouse using previously approved models and methods.

2. Mass and energy release data that is calculated by Westinghouse using an approved ECCS evaluation model with input parameters appropriately and conservatively biased for the specific containment analysis, as described in this report.

3. Mass and energy release data that is provided by the customer that is either calculated by the customer or another vendor using their approved models and methods.

The mass and energy release from the detailed evaluation model will continue until such time as a conservative release rate is established to transition the calculation from the detailed modeling to a GOTHIC calculated mass and energy release.

Table 5.1-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements	
	Requirements
1	Reactor Power – The reactor should be assumed to have been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.
2	Core Stored Energy – The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.)
3	Fission Heat – Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.
4	Decay of Actinides – The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.
5	Fission Product Decay – The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.
6	Metal-Water Reaction Rate – The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation. The reaction shall be assumed not to be steam limited.
7	Reactor Internals Heat Transfer – Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.
8	Fuel Rod Swelling and Rupture – The calculation of fuel rod swelling and rupture should not be considered for M&E calculations
9	Break Size and Location – Containment design basis calculations should be performed for a spectrum of possible pipe breaks, sizes, and locations to assure that the worst case has been identified.
10	Calculations, Sub-compartment Analysis – The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.
11	Calculations, Initial Blowdown Phase – The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 102% times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

Table 5.1-1 (cont.)	NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements
	Requirements
12	Calculations, Initial Blowdown Phase – Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.
13	Calculations, Initial Blowdown Phase – Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Table 5.1-2 ANS 56.4-1983 M&E for LOCA Recommendations	
	Recommendation
1	Reactor Coolant System Water and Metal – The increase in the reactor coolant system volume resulting from the pressure and temperature expansion to conditions at the initial power level defined in 3.2.2.2 shall be included. Stored energy in all reactor coolant system pressure boundary and internals metal thermally in contact with the reactor coolant system water shall be included.
2	Core Stored Energy – The core stored energy and the steady-state core-temperature distribution, adjusted for uncertainties, shall be consistent with the initial conditions and consistent with the time of fuel cycle life required in 3.2.2.1.
3	Fission Heat – Fission heat shall be conservatively calculated. Shutdown reactivities resulting from temperature and voids shall assume minimum plausible values including allowances for uncertainties; all data shall be based on their minimum values consistent with the fuel parameters which yield the maximum core stored energy. Rod trip and insertion may be assumed at the time appropriate for the transient being analyzed.
4	Decay of Actinides – The heat from the radioactive decay of actinides, including neptunium and plutonium as well as isotopes of uranium generated during operation, shall be calculated in accordance with fuel cycle calculations and shall be appropriate for the time in the fuel cycle that yields the highest calculated core stored energy. The decay heat shall be the values given in American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979 for end-of-life operation time.
5	Fission Product Decay – The heat generation rates from radioactive decay of fission products shall be assumed to be equal to at least the values given in ANSI/ANS-5.1-1979 for end-of-life operation time.
6	Metal-Water Reaction Rate – The amount of metal-water reaction shall be calculated according to 10 CFR 50.44 and assumed to occur uniformly over a period less than 2 minutes following the end of reactor vessel blowdown.
7	Main Steam Lines – Steam flow to the turbine until the main steam isolation valves or turbine stop valves are calculated to close may be included. Flow to the turbine shall be minimized. Delays and valve closure times shall be conservatively short. In lieu of this calculation, flow to the turbine may be conservatively terminated at break initiation.
8	Main Feedwater Line – Main feedwater flow shall be included and shall be maximized. Delays and valve closure times used to determine the termination of flow shall be conservatively long.
9	ECCS Flow – Flow from the ECCS shall be included. Flows and delay times shall be chosen in accordance with the single active failure consideration which results in the highest peak primary containment pressure.
10	Time of Life – The time of life of the core shall be that producing the maximum energy from the combination of core stored energy and decay heat assuming power level as required in 3.2.2.2.
11	Power Level – The initial power level shall be at least as high as the licensed power level plus uncertainties such as instrumentation error (typically 102 percent of the licensed power level).
12	Core Inlet Temperature – The initial core inlet temperature shall be the normal operating temperature consistent with the initial power level adjusted upward for uncertainties such as instrumentation error. The uncertainties shall be biased to result in maximizing energy releases through the break for the entire transient.
13	Reactor Coolant System Pressure – The initial reactor coolant system pressure shall be at least as high as the normal operating pressure consistent with the initial power level plus uncertainties such as instrumentation error.

Table 5.1-2 (cont.)	ANS 56.4-1983 Recommendations
	Recommendation
14	Core Parameters – Initial core parameters (including physics parameters, fuel properties, and gas conductivity) shall be chosen to maximize core stored energy.
16	Single Passive Failures – Passive failures normally need not be considered.
17.	Non-emergency Power – The loss of non-emergency power shall be postulated if it results in circumstances (for example, delayed primary containment cooling or safety injection) which lead to higher primary containment pressures.
18	Nodalization – Geometric nodalization for the various periods of the reactor coolant system break analysis need not be the same. Since low quality at the break node is conservative during blowdown because it leads to high flow rates, the reactor coolant system shall be modeled with sufficient detail so that the quality at the break location shall not be over predicted.
19	Thermodynamic Conditions – The thermodynamic state conditions for steam and water shall be described using real gas equations or industry accepted steam table in such a manner that the resultant steam and water temperature and partial steam pressure are within one percent of that which would result from use of the 1967 ASME Steam Tables with appropriate interpolation.
20	Pump Characteristics – The characteristics of the reactor coolant system pumps shall be derived from a dynamic model that includes momentum transfer between the fluid and the impeller with variable pump speed as a function of time. The pump model for the subcooled and two-phase region shall be verified by applicable subcooled and two-phase performance data. In lieu of a full dynamic pump model, any model which can be shown to be conservative by comparison with the test data or by comparison with a full dynamic pump model may be used.
21	Break Sizes – For reactor coolant system analysis, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the reactor coolant system. The beak shall be defined by its location, type, and area.
22	Break Flow Model – Empirical critical break flow models developed from test data may be utilized during the periods of applicability, for example, subcooled, saturated, or two-phase critical flow. Acceptable critical break flow models, when the fluid conditions are subcooled immediately upstream of the break, include the Zaloudek and Henry-Fauske models. During the period when fluid conditions immediately upstream of the break are saturated or two-phase, an acceptable model is the Moody critical flow model. The critical break flow correlations may be modified to allow for a smooth transition between subcooled and saturated flow regions. Other critical flow models may be used if justified by analysis or experimental data. The discharge coefficient applied to the critical flow correlation shall be selected to adequately bound experimental data.
23	ECCS Spillage – In generating mass and energy release source terms from spillage for primary containment peak pressure determination, the quality shall be selected based on the partial pressure of steam in containment to maximize primary containment pressurization. For the determination of the maximum primary containment sump temperature for calculation of available NPSH, assumptions on generating mass and energy release and spillage source terms shall be biased toward maximizing the sump temperature.
24	Drywell Backpressure – Throughout the post-LOCA transient, the mass and energy release calculation shall be coupled to the drywell pressure calculation, or a conservatively low back-pressure function shall be used.
25	Heat Transfer Correlations – Heat transfer correlations shall be based on experimental data or chosen to predict conservatively high primary containment pressure.

Table 5.1-2 ANS 56.4-1983 Recommendations (cont.)	
	Recommendation
26	Core Modeling – Fission heat may be calculated using a core averaged point kinetics model which considers delayed neutrons and reactivity feedback. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowances for uncertainties for the range of power distribution shapes and peaking factors which result in the maximum core stored energy. Rod trip and insertion may be assumed if they are calculated to occur. Reactivity effects shall be consistent with the time of life which leads to the maximum core stored energy. For core thermal hydraulic calculations, the core shall be modeled with sufficient detail so as not to under-predict core-to-reactor coolant heat transfer. Initial core stored energy shall be maximized.
27	Modeling of Metal Walls – Heat transfer from metal walls to coolant shall be calculated so as not to under-predict the rate of heat transfer relative to experimental data or the solution of the one-dimensional, time dependent heat conduction equation

5.3 LONG TERM DECAY HEAT STEAMING TO CONTAINMENT

GOTHIC will be used to calculate the long term decay heat steaming rate to the containment. The decay heat is calculated using ANS 1979 + 2σ uncertainty for full reactor power plus uncertainty.

5.3.1 Non-Condensable Gas Injection

If the plant design includes nitrogen charged accumulators, the mass of nitrogen in the accumulators is injected into the containment vapor space after all of the accumulator water has been injected.

6 CONCLUSIONS

This report outlines a generic methodology for performing LOCA and MSLB containment safety analyses for PWR and BWR containments. The methods are consistent with the regulations and guidance provided in NUREG-0800 (Reference 18) and ANSI/ANS 56.4 (Reference 19). Application of the methodology to specific plant designs is outlined in the Appendices of this report.

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APPENDIX A
GOTHIC GENERIC BWR MARK I MODEL

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A.1 INTRODUCTION

GOTHIC BWR Mark I containment models were developed and benchmarked against existing FSAR analyses and other sources of public information (Reference A-1). These models were merged to form a GOTHIC generic BWR Mark I containment model, which was used to produce sample transient cases. This Appendix contains a description of the GOTHIC generic BWR Mark I containment model, a summary of the results from the benchmark comparisons for the various GOTHIC BWR Mark I containment models, and results for various sample transients using the GOTHIC generic BWR Mark I containment model. GOTHIC version 7.2a was used for all the analysis cases presented in this Appendix.

A.2 MODEL DESCRIPTION

This section describes the GOTHIC generic BWR Mark I containment model. This model is used (with case specific input changes) to perform the sample cases documented in Section A.4.

A.2.1 Noding Structure

The GOTHIC generic BWR Mark I containment model noding structure is shown in Figures A.2.1-1, A.2.1-2, and A.2.1-3. The noding structure is based on the GOTHIC BWR Mark I containment models that were developed for the various benchmark cases described in Reference A-1. The noding structure for a plant specific model may be slightly different than the sample model; more or less flow paths and boundary conditions may be required for a plant specific model.

The GOTHIC generic BWR Mark I containment model contains [ ]^{a,c} control volumes as follows:
[

]^{a,c}

Flow boundary conditions are used to either supply mass and energy to the model or remove mass and energy from the model. Coupled boundary conditions are used to transport the mass and energy taken from one boundary condition to other locations in the model. Pressure boundary conditions are used to impose a pressure constraint on some element in the model.

a,c

Figure A.2.1-1 GOTHIC Generic BWR Mark I Model Noding Diagram

a,c

Figure A.2.1-2 GOTHIC Generic BWR Mark I Model Noding Diagram
(Expanded Drywell Region)

a,c

Figure A.2.1-3 GOTHIC Generic BWR Mark I Model Noding Diagram (Expanded Wetwell Region)

The GOTHIC generic BWR Mark I model has [ ]^{a,c} flow boundary conditions, [ ]^{a,c} coupled boundary conditions, and [ ]^{a,c} pressure boundary condition as follows:

[

]^{a,c}

The control volumes are connected to one another and to the boundary conditions with [ ]^{a,c} flow paths as follows:

[

]^{a,c}

[

]a,c

A.2.2 Control Volume Input

The control volume input data can either be calculated from drawings, or taken from the most current approved containment model. Most of the control volume input for the GOTHIC generic BWR Mark I containment model was taken from the models developed for the various benchmark cases described in Reference A-1. The control volume input for the vessel and recirculation lines was taken from a GOBLIN ECCS evaluation model described in Appendix B.

GOTHIC requires the following control volume input data: free volume, elevation, height, and hydraulic diameter. The free volume is the volume that can be filled with air or water and is calculated by subtracting the volume of equipment, piping, etc. from the geometric volume determined by the room dimensions. [ ]a,c

[

]^{a,c} The elevation is the location of the bottom of the control volume. The difference between the bottom and top elevations of the control volume is the height. The control volume hydraulic diameter can be set to the actual diameter of the pipe, or calculated as four times the free volume divided by the total wetted surface area within the control volume.

The user can also specify a liquid/vapor interface area input value for the control volume or use the code calculated default value. Using a large interface area will maximize heat and mass transfer between the liquid and vapor phases and tend to keep them in thermodynamic equilibrium. Likewise, using a very small interface area will minimize heat and mass transfer between the liquid and vapor phases.

Drywell

This control volume represents the drywell that contains the reactor vessel. [

]^{a,c} The bottom of the drywell is located at an elevation of 502 feet – 4 inches, and the height of the drywell is approximately 100 feet. The drywell hydraulic diameter, which is based on the volume and wetted surface area, is 23.3 feet.

The default liquid/vapor interface area is used for most of the applications; [

]^{a,c}

[

]^{a,c}

Wetwell

This control volume represents the toroid-shaped wetwell containing the suppression pool. [

]^{a,c} The bottom of the wetwell is 479 feet, and the height of the wetwell is 30 feet. The wetwell hydraulic diameter, which is calculated based on the volume and wetted surface area, is 34.9 feet.

[

]^{a,c}

Reactor Vessel

This control volume represents the reactor vessel. The reactor vessel is used to calculate the long-term steaming mass and energy release input data for the RSLB and MSLB events.

The reactor vessel volume was calculated by [

]^{a,c}. The volume of the reactor vessel (including recirculation lines and pumps) is 20,524 ft^3. The bottom elevation of the vessel is 512 feet – 4 inches and the height is 68.63 feet.

[

]^{a,c}

a,c

Figure A.2.2-1 GOTHIC Vessel Noding Diagram

A.2.3 Flow Path Input

Flow paths connect control volumes and boundary conditions to one another. GOTHIC requires input values for the flow path elevation, height, hydraulic diameter, flow area, inertia length, friction length and loss coefficient. This data can either be calculated from drawings or taken from the most current approved containment model. The flow path input for the GOTHIC generic BWR Mark I containment model was taken from the models developed for the various benchmark cases described in Reference A-1. The flow path input is described below.

Flow Path 1: Break Flow

This flow path is connected to [ ]^{a,c} the reactor vessel for the RSLB cases. It is located at an elevation of 538 ft [ ]^{a,c} and has a flow area of 4.261 ft^2; this area represents the cross sectional flow area of [ ]^{a,c}.

The break flow path is connected to [ ]^{a,c} the reactor vessel for the MSLB and SBA cases. It is located at an elevation of 564 ft [ ]^{a,c} and has an area of 2.808 ft^2 for the MSLB cases and an area of 0.05 ft^2 for the SBA case.

An arbitrary height input value of 4 ft was chosen for the RSLB and MSLB cases [ ]^{a,c}. An arbitrary height input value of 0.1 ft was chosen for the SBA case.

[

]^{a,c}

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

A.2.4 Thermal Conductor Input

GOTHIC requires the following thermal conductor input data: overall thickness, thickness of each layer, material properties (density, specific heat, and thermal conductivity), heat transfer surface area, heat transfer coefficient type, and initial temperature. The thermal conductor input data can either be calculated from drawings or taken from the most current approved containment model. The volume for a wall-type thermal conductor is the product of the thickness and surface area of one side of the wall. The conductor volume should be conserved when calculating the input from drawings or converting input from one code to another. The thermal conductor input for the GOTHIC generic BWR Mark I containment model was taken from the models developed for the various benchmark cases described in Reference A-1; the thermal conductor input for the vessel and recirculation lines was taken from a GOBLIN ECCS evaluation model, described in Appendix B.

Containment Thermal Conductors

There are [ ]a,c thermal conductors in the GOTHIC generic BWR Mark I containment model; [ ]a,c thermal conductors are used to represent the drywell and wetwell heat sinks. The [ ]a,c heat sink represents a 0.5604-inch thick carbon steel wall with an area of 32,250 ft^2. Each of the [ ]a,c heat sinks represents one half of the [ ]a,c; the upper half of the shell is represented by a 0.5820-inch carbon steel wall with an area of 16,120.5 ft^2, and the lower half of the shell is represented by a 0.6528-inch carbon steel wall with an area of 16,120.5 ft^2. The material properties for carbon steel were taken from Reference A-6.

[

]a,c

[

]^{a,c}

Fuel Rods

The fuel rods are modeled as thermal conductor [ ]^{a,c} in the GOTHIC generic BWR Mark I containment model. The fuel rods are placed in [ ]^{a,c} the GOTHIC vessel volume.

The fuel rods contain 3 material types: zirconium cladding, helium gas, and UO_2 fuel. The fuel material properties were taken from the GOTHIC BWR Mark I benchmark model (Reference A-1).

The fuel internal heat generation rate is calculated by dividing the core power by the UO_2 volume. The core power is determined by interpolating the core decay heat fraction function table over time and multiplying this value by the nominal core power plus uncertainty.

Vessel Thermal Conductors

The GOBLIN vessel thermal conductor input is listed under the PLATE section of the input data file. The GOBLIN PLATE data was converted from metric to British units for input to GOTHIC.

The GOBLIN ECCS model contains a large number of plates. The plates represent either stainless steel clad carbon steel or stainless steel that is located in the reactor vessel. The material properties for stainless steel were taken from Reference A-6. These plates were modeled as thermal conductors in the GOTHIC generic BWR Mark I vessel model; plates that were very thin or had no thickness were not modeled.

Heat Transfer Coefficients

Five heat transfer coefficient types are defined in the GOTHIC generic BWR Mark I containment model. The first type uses the DIRECT heat transfer coefficient with SPLIT option to model heat transfer between the vapor and the dry portion of the torus wall. The second type models a constant specified heat flux (0.0 Btu/hr- ft^2-F) to represent either an insulated surface or the centerline of a wall modeled with half thickness and heat transfer on both faces. The third type uses the DIRECT heat transfer coefficient with SPLIT option to model heat transfer between the liquid and the wet portion of the torus wall. The fourth type uses the [ ]^{a,c} heat transfer coefficient to model heat transfer within the reactor vessel. The fifth type uses the DIRECT heat transfer coefficient with SPLIT option to model heat transfer to the drywell shell; it allows the heat transfer to be split between the liquid and vapor phase when the drywell water level is between [ ]^{a,c} of the drywell height.

The DIRECT heat transfer coefficient types use the DLM option for condensation, the vertical surface free convection correlation, the pipe flow forced convection correlation, radiation heat transfer, and have a multiplier function associated with them. The multiplier value should be 1.0 for most transients; however, a value of [ ]^{a,c} should be used for the blowdown portion, and a value of [ ]^{a,c} should be used for the post-blowdown portion of the minimum NPSHa analysis and the minimum ECCS backpressure calculations to conservatively maximize the condensation rate on the containment heat sinks and reduce the calculated pressure.

A.2.5 Boundary Condition Input

The GOTHIC generic BWR Mark I containment model uses boundary conditions to provide [ ][a,c] to the containment, and [ ][a,c] to the reactor vessel. The boundary condition input should be based on plant specific data. Each of these boundary condition functions is described below.

Containment Spray

Containment spray is modeled with [ ][a,c]. In the generic model, [ ][a,c] removes water from the suppression pool through [ ][a,c] at a fixed flow rate of 5000 gpm (11.14 cfs). The spray flow rate is split between the drywell and wetwell (95% to the drywell and 5% to the wetwell) [ ][a,c]. The heat removed from the containment spray flow as it passes through an RHR heat exchanger is calculated with [ ][a,c]. The spray drop diameter is specified in the nozzle components that are connected to the [ ][a,c] spray flow paths. Since flow is being removed from the control volume, the flow boundary condition pressure and temperature input values have no effect on the calculation, so nominal values are specified.

The ON/OFF trips, which operate on [ ][a,c], are used to initiate/terminate the containment spray flow. Plant specific input values should be used to control the containment spray flow; a minimum operator action time of 10 minutes is assumed in the generic model. If the containment conditions exceed the setpoint for spray initiation at 10 minutes, the operator is assumed to trip one of the LPCI pumps and align the flow from the other LPCI pump through the RHR heat exchanger to initiate containment spray to the drywell and wetwell at 600 seconds.

LPCS Flow

[ ][a,c] are used to model the ECCS flow from the LPCS pumps. [ ][a,c] removes water from the suppression pool through [ ][a,c] at a flow rate defined by [ ][a,c], which is input to the boundary condition via [ ][a,c]. The LPCS flow is supplied to the vessel (just above the core region) using [ ][a,c]. Since flow is being removed from the control volume, the flow boundary condition pressure and temperature input values have no effect on the calculation, so nominal values are specified.

The ON/OFF trips, which operate on [ ][a,c], are used to initiate/terminate LPCS flow. The ON/OFF trips also affect [ ][a,c]. The ON/OFF trip input data should be based on plant specific data. For the generic model, the ON trip is set to start LPCS flow on high drywell pressure (2.5 psig) and the OFF trip is set to stop LPCS flow on high vessel level (46.92 ft). The ON trip delay time input value was based on the data given for a loss of offsite power. The delay time for the pump to reach rated speed is 33.5 seconds and the delay time for the injection valve to fully open is 47.3 seconds. Since flow will start when the injection valve begins to open, an ON trip delay of 35 seconds was chosen.

LPCI Flow

[ ]^{a,c} are used to model the ECCS flow from the LPCI pumps. [ ]^{a,c} removes water from the suppression pool through [ ]^{a,c} at a flow rate defined by [ ]^{a,c}, which is input to the boundary condition via [ ]^{a,c}. The LPCI flow is delivered to the recirculation line discharge legs via [ ]^{a,c}. Since flow is being removed from the control volume, the flow boundary condition pressure and temperature input values have no effect on the calculation, so nominal values are specified.

The ON/OFF trips, which operate on [ ]^{a,c}, are used to initiate/terminate LPCI flow. The ON/OFF trips also affect [ ]^{a,c}. The ON/OFF trip input data should be based on plant specific data. For the generic model, the ON trip is set to start LPCI flow on high drywell pressure (2.5 psig). The ON trip delay time input value was based on the data for a loss of offsite power. The delay time for the pump to reach rated speed is 28 seconds and the delay time for the injection valve to fully open is 56 seconds. Since flow will start when the injection valve begins to open, an ON trip delay of 35 seconds was chosen for the generic model. The OFF trip is set to stop LPCI flow when the operator takes action to either initiate containment spray or pool cooling. A minimum operator action time of 10 minutes was assumed in the generic model.

RHR Cooling

[ ]^{a,c} are used to model the RHR cooling of the suppression pool. In the generic model, [ ]^{a,c} removes water from the suppression pool through [ ]^{a,c} at a constant flow rate of 5000 gpm (11.14 cfs). After removing the RHR heat exchanger heat rate specified by [ ]^{a,c}, the RHR flow is returned to the suppression pool via [ ]^{a,c}. Since flow is being removed from the control volume, the flow boundary condition pressure and temperature input values have no effect on the calculation, so nominal values are specified.

The ON/OFF trips, which operate on [ ]^{a,c}, are used to initiate/terminate RHR cooling. The ON/OFF trip input data should be based on plant specific data; a minimum operator action time of 10 minutes is assumed in the generic model. In the generic model, the ON trip is set to a large time value since the RHR bypass valve is initially open during normal operation.

External Break Mass and Energy Releases

The initial mass and energy release input data for the RSLB and MSLB transient events is calculated externally with GOBLIN as described in Appendix B. The GOBLIN ECCS evaluation model input is biased to calculate conservative mass and energy releases according to References A.2 and A.3. The mass and energy release data is input via [ ]^{a,c}. Nominal pressure, enthalpy, and flow values are multiplied by [

]^{a,c}. The liquid portion of the external break flow is released in the form droplets with a diameter of 100 microns during the blowdown (choked flow) phase of the event (Reference A.12).

The long-term mass and energy release for the RSLB and MSLB events, and the mass and energy release for the SBA event is calculated using the reactor vessel model that is incorporated within the GOTHIC generic BWR Mark I containment model.

Feedwater

The feedwater piping fluid volume, metal mass, and flow paths have been added to the GOBLIN ECCS evaluation model for the RSLB and MSLB M&E release calculations. Therefore, feedwater flow is included in the external mass and energy release data for the RSLB and MSLB events and does not need to be modeled separately in GOTHIC.

Turbine steam flow and feedwater flow may need to be modeled for some portion of time in some of the containment analysis events (IBA, SBA, SBO, and ATWS). [ ]^{a,c} is used to model feedwater flow to the reactor vessel through [ ]^{a,c}. The user may adjust the [ ]^{a,c} pressure, temperature, and flow rate input values as needed for the particular event being modeled.

HPCI Flow

[ ]^{a,c} is used to model ECCS flow from the HPCI pump to the reactor vessel downcomer [ ]^{a,c}. HPCI flow can be drawn from either the CST or the suppression pool. The HPCI flow in the generic model is assumed to be drawn from [ ]^{a,c}. The HPCI flow rate is defined by [ ]^{a,c}, which is input to the boundary condition via [ ]^{a,c}.

The ON and OFF trips, which operate on [ ]^{a,c}, are used to initiate and terminate HPCI flow. In the generic model, the ON trip is set to start HPCI flow on high drywell pressure (2.5 psig) and the OFF trip is set to stop HPCI flow on high vessel level (46.92 ft). The delay time for the turbine to reach rated speed is 25 seconds and the delay time for the injection valve to fully open is 53.5 seconds. Since flow will start when the injection valve begins to open, an ON trip delay of 35 seconds was chosen.

HPCI flow is not modeled in the RSLB or MSLB events; the HPCI pump is assumed to be shutoff when the reactor vessel pressure is less than 150 psig.

External LPCS and LPCI Suction

[ ]^{a,c} are used to model the external ECCS flow that is drawn from the suppression pool. [ ]^{a,c} remove water from the suppression pool through [ ]^{a,c}, corresponding to the external LPCS and LPCI flow rates respectively, as the flow forcing function multipliers. Since flow is being removed from the control volume, the flow boundary condition pressure and temperature input values have no effect on the calculation, so nominal values are specified.

Vessel Pressure

Turbine steam flow and feedwater flow may need to be modeled for some portion of time in some of the containment analysis events (IBA, SBA, SBO, and ATWS). [ ]^{a,c} can be used to control the reactor vessel pressure. The user may adjust the boundary condition pressure or valve position as needed for the particular event being modeled.

[ ]^{a,c}

A.2.6 Components

GOTHIC components are used to model heaters/coolers, valves, and nozzles; a heat exchanger component could also be used to model the RHR heat exchanger. The component input should be based on plant specific data.

Heaters/Coolers

Heater components are used to model the addition of pump heat to the suppression pool. The pump heat rate input value (per pump running) is multiplied by a forcing function representing the number of pumps running; the number of pumps running is specified for each transient event. The number of running pumps is typically a function of time, since one or more of the running LPCI pumps can be turned off.

The pump heat rate input value is calculated [ ]^{a,c}. In the generic model, the LPCI pump horsepower is 700 hp and the LPCS pump horsepower is 800 hp. Therefore, the pump heat rate input to the suppression pool is estimated to be:

[ ]^{a,c}

The HPCI pump is turbine driven; however, a small heat rate input value is specified when the pump is running. This non-zero heat rate is used to trigger a control variable calculation that interpolates the HPCI flow rate from a forcing function input table.

A heater component can also be used to model core heat addition directly to the suppression pool. This component is included in the generic model, but not activated (no heat input value is specified). The core decay heat power fraction is specified in [ ]^{a,c}; this forcing function is interpolated over time using [ ]^{a,c}. The resulting heat rate can be modeled as internal heat generation within the fuel rod thermal conductor for transfer to the reactor coolant and out to containment through the break flow path, or added directly to the suppression pool via the heater component.

Cooler components can be used to model the RHR heat exchanger heat removal from the suppression pool. A cooler component is also included in the generic model, but not activated (no heat input value is specified). The RHR heat removal rate is currently calculated with [ ]^{a,c} in the generic model; the heat is removed within the coupled boundary conditions for the RHR and/or spray flow.

Valves

[ ]^{a,c} valve components are used to model the safety and relief valves, the ADS, and to turn on the break flow path from the vessel (if applicable). A [ ]^{a,c} valve component is used to close the vessel flow path connected to the [ ]^{a,c}.

[

]^{a,c}

The break flow path valve flow area input value is plant and event specific. In the generic model sample cases the break flow path valve area varied from 0.05 ft^2 for a small break case to 4.261 ft^2 for the full double ended RSLB case.

The valve on the flow path that connects to the [ ]^{a,c} is modeled with a small area (0.05 ft^2). The valve is closed with a trip, which is normally set to the time when the external break flow ends and GOTHIC begins its break flow calculation.

The vacuum breaker between the drywell and wetwell is modeled with a pressure dependent valve; plant specific data should be used to model the vacuum breaker. The valve loss coefficient is defined as a function of the flow path differential pressure in [ ]^{a,c}. The valve is modeled to open (has a small loss coefficient) when the differential pressure between the wetwell and vent pipe volumes is greater than 0.5 psi.

Nozzles

Nozzle components are used to model the release of liquid droplets. [ ]^{a,c}

The fraction of the liquid break flow that is converted to drops is determined by the superheat in the incoming flow relative to the saturation temperature at the total containment pressure. The water will break up into small drops (100 microns) with only a few degrees of superheat (~5F). If the water is saturated or subcooled, there will be minimal breakup unless the velocity is very high and there is breakup due to hydrodynamic forces.

In the GOTHIC generic BWR Mark I containment model, the liquid break flow is assumed to be released [ ]^{a,c}

A typical droplet diameter of 500 microns (0.0196 inches) is modeled for containment spray and a droplet diameter of 0.25 inches is modeled for LPCS in the generic model. Plant specific values for these inputs should be used in the design analyses.

A.2.7 Forcing Functions

Tabulated forcing functions are used to provide input data for various boundary conditions or components. There are [ ]^{a,c} forcing function tables in the GOTHIC generic BWR Mark I containment model. Each is described briefly below.

[ ]^{a,c} acts as a switch; the value at 0.0 seconds is 0.0 and the value at 1.0 second (and longer) is 1.0. This forcing function is used with a trip signal to turn a component, flow rate, or heat rate on or off.

[ ]^{a,c} provide the stem travel and valve loss coefficient curves for the vacuum breakers. This data was taken from Reference A-1. Plant specific values for these inputs should be used in the design analyses.

[ ]^{a,c} contains the core power fraction as a function of time. This forcing function is multiplied by the full power core heat generation rate (plus uncertainty) to yield the decay heat power as a function of time.

The ANS 5.1-1979 + 2 sigma decay heat standard was used to produce the data shown in Table A.2.7-1. This data was used in [ ]^{a,c} the generic BWR Mark I containment model. The core power fraction transient data from a BWR ECCS evaluation model (GOBLIN) or BWR transient evaluation model (BISON) calculation could also be used as input for this forcing function.

[ ]^{a,c} contains the HPCI flow rate as a function of the vessel pressure. This forcing function data table is interpolated using [ ]^{a,c} to determine the HPCI flow rate in GOTHIC. Plant specific values for these inputs should be used in the design analyses.

[ ]^{a,c} contain the LPCI flow rate as a function of the differential pressure between the vessel and the wetwell. One of these forcing function data tables is interpolated using a [ ]^{a,c} to determine the LPCI flow rate in GOTHIC. [ ]^{a,c} is used when 4 LPCI pumps are operating, [ ]^{a,c} is used when 2 LPCI pumps are operating, and [ ]^{a,c} is used when 1 LPCI pump is operating. Plant specific values for these inputs should be used in the design analyses.

Table A.2.7-1 ANS 5.1-1979 + 2 Sigma Core Decay Heat Fraction	
Time (sec)	**Decay Heat Generation Rate (Btu/Btu)**
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1,000	0.022156
1,500	0.019921
2,000	0.018315
4,000	0.014781
6,000	0.013040
8,000	0.012000
10,000	0.011262
15,000	0.010097
20,000	0.009350
40,000	0.007778
60,000	0.006958
80,000	0.006424
100,000	0.006021
150,000	0.005323
200,000	0.004847
400,000	0.003770
600,000	0.003201
800,000	0.002834
1,000,000	0.002580
2,000,000	0.001909
4,000,000	0.001355
6,000,000	0.001091
8,000,000	0.000927
10,000,000	0.000808

[ ]^{a,c} contain the LPCS flow rate as a function of the differential pressure between the vessel and the wetwell. One of these forcing function data tables is interpolated using a [ ]^{a,c} to determine the LPCS flow rate in GOTHIC. [ ]^{a,c} is used when 2 LPCS pumps are operating, [ ]^{a,c} is used when 1 LPCS pump is operating prior to switching an LPCI pump to RHR or spray cooling, and [ ]^{a,c} is used when 1 LPCS pump is operating after switching an LPCI pump to RHR or spray cooling (typically 600 seconds). Plant specific values for these inputs should be used in the design analyses.

[ ]^{a,c} contains the LPCI flow rate as a function of the differential pressure between the vessel and wetwell after the pump has been switched to the RHR or spray cooling mode. This forcing function data table is interpolated using a [ ]^{a,c} to determine the LPCI flow rate in GOTHIC. Plant specific values for these inputs should be used in the design analyses.

[ ]^{a,c} are used to transfer the HPCI, LPCI, LPCS, and RHR or containment spray flow rates calculated by [ ]^{a,c} to the [ ]^{a,c}. These [ ]^{a,c} have the corresponding ECCS pump control variable output value (flow rate) as their independent variables.

[ ]^{a,c} is used to specify the feedwater flow rate as a function of time.

[ ]^{a,c} specify the number of LPCI and LPCS pumps running as a function of time. The function values from [ ]^{a,c} and [ ]^{a,c} are multiplied by the LPCI and LPCS pump heat rates to determine the pump heat input to the suppression pool. Although one of the LPCI pumps is switched to the RHR or spray cooling mode after 600 seconds, it is still counted as an LPCI pump for heat input.

[ ]^{a,c} is used to specify a containment heat transfer coefficient multiplier as a function of time. The value should be 1.0 for most transients; however, a value of [ ]^{a,c} should be used for the blowdown portion, and a value of [ ]^{a,c} should be used for the post-blowdown portion of the minimum NPSHa analysis and the minimum ECCS backpressure calculations to conservatively maximize the condensation rate on the containment heat sinks and reduce the calculated pressure.

A.2.8 Control Variable Input

The GOTHIC generic BWR Mark I containment model has [ ]^{a,c} control variables. The control variables are used to calculate the heat exchanger heat removal rate, read external data files, determine the core heat rate, calculate the ECCS pump flow rates, determine the liquid break flow rate drop diameter, and calculate various values for plotting. Each of the control variable functions is described briefly below.

Heat Exchanger Heat Removal Rate Calculation

[ ]^{a,c} is used to calculate the heat removal rate for the RHR heat exchanger. The RHR heat removal rate is defined as:

$$Q_{RHR} = K^*(T_{pool} - T_{SW})$$

The RHR heat exchanger K value and T_{SW} value (service water temperature) are plant specific inputs.

The RHR heat removal rate is calculated using the sum-difference control variable, [ ]^{a,c}; sum-difference control variables have the functional form:

$$Y = G*(a0+a1*x1+a2*x2+...).$$

The control variable constant multiplier (coefficient G) input value is set to the K value (281.7 Btu/s-F), the control variable constant adder (coefficient a0) input value is set to the T_{SW} value (-98°F), and component 1 (x1) is defined as the variable pool temperature.

[ ]^{a,c} use an IF test to check for a minimum RHR heat exchanger flow rate; the value is specified in coefficient a0. If the flow rate to the heat exchanger is less than the specified minimum value (400 lbm/s), the RHR heat removal rate is set to 0.0 Btu/s.

[

]^{a,c}

[

]a,c

Core Heat Rate

The ANS 5.1-1979 decay heat standard + 2 sigma uncertainty (as shown in Table A.2.7-1) defines the core decay heat power fraction values after shutdown. This data is entered into the decay heat power fraction table, [ ]a,c.

[ ]a,c is used to interpolate the decay heat power fraction data table stored in forcing function 4; coefficient G is the full core power heat output (increased by measurement uncertainty) and multiplies the decay heat power fraction output value to yield the core heat rate in Btu/s. [ ]a,c compares the GOBLIN core power generation rate [ ]a,c with the GOTHIC decay heat rate and outputs the maximum value. [ ]a,c uses an IF test to determine when to start the GOTHIC core heat rate calculation; the time is specified in coefficient a0.

ECCS Pump Flow Rates

GOTHIC will calculate the ECCS pump flow rate if the pumps are running and there is no external ECCS flow being drawn from the suppression pool. The ECCS pump flow rate depends on the number of pumps running and the differential pressure between the vessel and suppression pool. A [ ]a,c is used to specify the ECCS pump flow rate as a function of the pressure difference between the vessel and suppression pool for each of the different pump types (HPCI, LPCI, and LPCS). The data for these [ ]a,c is plant specific.

[

]a,c

Break Flow – Liquid Droplet Fraction

The fraction of the liquid break flow that is converted to drops is determined by the superheat in the incoming flow relative to the saturation temperature at the total containment pressure. The water will break up into small drops with only a few degrees of superheat (~5F). If the water is saturated or subcooled, there will be minimal breakup unless the velocity is very high and there is breakup due to hydrodynamic forces.

In the generic BWR Mark I containment model, the liquid break flow is assumed to be released as [

]^{a,c}

[ ]^{a,c} determines the vessel pressure as the maximum of the GOBLIN vessel pressure [ ]^{a,c}, the GOTHIC vessel pressure, or the GOTHIC drywell pressure. If external break flow is being used, the GOBLIN vessel pressure will eventually become less than the GOTHIC calculated drywell pressure since the GOBLIN model assumes a low containment backpressure. If external break flow is not being used, [ ]^{a,c} will be set to a value of 14.7 psia and the GOTHIC vessel pressure should always be greater than the drywell pressure.

A sum-difference control variable, [ ]^{a,c}, is used to calculate the pressure difference between the vessel and drywell; it has 2 components, the vessel pressure from [ ]^{a,c} and the drywell pressure. [ ]^{a,c} checks if the pressure difference is greater than 50 psi; if so, it returns a value of 1.0 and if not, it returns a value of 0.0. [

]^{a,c}

Plot Values or Other Variables

[ ]^{a,c} are used to calculate the drywell and wetwell component phase (steam, air, or water) mass as functions of time for plotting. These multiplier type control variables multiply the cell volume, cell component phase volume fraction, and cell component phase density to get the component phase mass.

[ ]^{a,c} is used to calculate the suppression pool water volume as a function of time for plotting. This multiplier type control variable has two components, the [ ]^{a,c} volume and the [ ]^{a,c} liquid phase component volume fraction which, when multiplied together, yield the pool water volume.

[ ]^{a,c} is used to calculate the total safety and relief valve steam flow rate for plotting. This sum-difference control variable has six components: the vapor flow rate from the [ ]^{a,c} flow paths, the vapor flow rate from the [ ]^{a,c} flow path, and the vapor flow rate from the [ ]^{a,c} flow paths.

[ ]^{a,c} is used to calculate the total break flow rate for plotting. This sum-difference control variable has five components: [

]^{a,c}.

[ ]^{a,c} is used to calculate the total suppression pool heat rate input for plotting. This sum-difference control variable has four components, corresponding to each of the heater components that are located in the suppression pool: the LPCI pump heat, the LPCS pump heat, the HPCI pump heat, and the core heat (if added directly to the pool).

[ ]^{a,c} are used to calculate the total vessel water mass [ ]^{a,c}. The water mass in each of the vessel cells is calculated using a multiplier control variable with the following components: cell volume, cell water volume fraction, cell liquid density, and cell porosity. The control variables representing the various vessel cell masses are summed in [ ]^{a,c}.

A.2.9 Initial Condition Input

As shown in Table A.2.9-1, the containment initial conditions are biased differently for each of the various applications.

For the peak pressure analysis, the initial containment pressure, wetwell temperatures, and suppression pool water volume are [ ]^{a,c}, while the free volume, heat sink area, drywell temperature, and drywell humidity are [ ]^{a,c}. The effect of these biases is to [ ]^{a,c}. This will maximize the calculated short-term containment peak pressure.

The input biasing for the peak suppression pool temperature and peak drywell temperature calculations is not much different than for the peak pressure analysis; the drywell temperature is [ ]^{a,c} and the suppression pool water volume is [ ]^{a,c}

For the minimum ECCS containment backpressure and minimum NPSHa calculations, the initial containment pressure and wetwell water volume is [ ]^{a,c}, while the free volume, heat sink area, temperatures and drywell humidity are [ ]^{a,c}. In addition, the heat transfer coefficient is multiplied by a factor of [ ]^{a,c} during blowdown and [ ]^{a,c} after. The effect of these biases is to [ ]^{a,c}. This will minimize the calculated containment pressure, and [ ]^{a,c}.

Table A.2.9-1 Containment Model Input and Initial Conditions Biasing					
	Peak DW Pressure	ECCS Min Backpressure	Peak SP Temperature	Minimum NPSHa	Peak DW Temperature
Free Volume					
Heat Sink Area					
HTX Multiplier					
DW Pressure					
DW Temperature					
DW Humidity					
WW Pressure					
WW Temperature					
SP Temperature					
WW Water Volume					

a,c

The containment initial conditions used in the GOTHIC generic BWR Mark I containment model sample cases are tabulated below:

Generic BWR Mark I Containment Model Initial Conditions Input					
	Peak DW Pressure	ECCS Min Backpressure	Peak SP Temperature	Minimum NPSHa	Peak DW Temperature
DW Pressure					
DW Temperature					
DW Humidity					
WW Pressure					
WW Temperature					
SP Temperature					
WW Water Vol. Fraction					
DC Water Vol. Fraction					

a,c

The wetwell and downcomer water volume fraction input values were calculated using the following equations:

[

]^{a,c}

where:

[

•

]^{a,c}

A.2.10 Transition from GOBLIN to GOTHIC for the Long-Term M&E Release Calculations

[

]^{a,c}

[

]a,c

A.3 BENCHMARK CASES

Benchmark comparisons with various GOTHIC BWR Mark I containment models are documented in Reference A-1. The benchmark cases were run using GOTHIC version 7.2.

Since then, GOTHIC has been upgraded to version 7.2a to correct some minor errors in version 7.2 and add some additional capabilities. The code changes are described in Appendix A of the GOTHIC Users Manual (Reference A-7).

The benchmark cases were re-run using the more recent GOTHIC version 7.2a; no differences in the results were found. The results from the BWR Mark I containment model benchmark comparisons with GOTHIC version 7.2a are summarized in this section.

Short-Term Recirculation Suction Leg Break Benchmark Comparison

The RSLB event typically results in the peak containment pressure and is the DBA for many BWR containments. The RSLB event assumes a double ended guillotine break in the pump suction leg of one of the two recirculation lines. The reactor vessel blowdown occurs through the pump suction leg and through the jet pump nozzles of the pump discharge leg.

The short-term RSLB benchmark comparison was modeled two ways. The benchmark for Model 1 was an NRC CONTEMPT analysis (Reference A-8); however, much of the Model 1 input data had to be inferred from other sources since the benchmark CONTEMPT input deck was not available. The mass and energy release data for Model 1 was calculated externally and was input using forcing functions in the boundary conditions. The benchmark for Model 2 was a GE M3CPT analysis (Reference A-9, Figures 6.2-31 and 6.2-33); however, the benchmark mass and energy release input data was not available. A simplified GOTHIC vessel model was added to the containment model and this was used to calculate the blowdown mass and energy release along with the containment response in Model 2.

The noding diagrams for the two models are shown in Figures A.3-1 and A.3-2 and a comparison of the key containment input values is shown in Table A.3-1. [

]a,c

The short-term RSLB benchmark comparison results for Model 1 are shown in Figures A.3-3 through A.3-7 and the short-term RSLB benchmark comparison results for Model 2 are shown in Figures A.3-8 through A.3-11. The GOTHIC response is shown as a solid line and the benchmark data points are shown with Δ in the figures. With Model 1, GOTHIC matched the drywell pressure and the drywell and wetwell temperatures very well, but under-predicted the initial wetwell pressure response. The difference in the wetwell pressure response is attributed to using assumed values for the containment model inputs in the GOTHIC model. With Model 2, GOTHIC matched the drywell pressure and temperature and the wetwell pressure very well, but under-predicted the wetwell vapor temperature response. The difference in the wetwell temperature response is attributed to using assumed values for the torus volume and suppression pool water volume in the GOTHIC model.

Table A.3-1 Comparison of Key Containment Model Input Values			
Value	**Model 1**	**Model 2**	**Generic Model**
Drywell Volume			
Torus Volume			
Vent Line/Header/Downcomer Volume			
Reactor Vessel Volume			
Downcomer Flow Area			
Downcomer Loss Coefficient			
Downcomer Inlet Inertia Length			
Downcomer Exit Inertia Length			
Initial Vessel Water Volume			
Initial Torus Water Volume			
Initial Drywell Pressure			
Initial Wetwell Pressure			
Initial Downcomer Submergence			
Initial Vessel Pressure			
Initial Drywell Temperature			
Initial Wetwell Temperature			

a,c

Note:
1. This value was tuned to allow the GOTHIC model to match the RSLB benchmark data.

a,c

Figure A.3-1 Short-Term RSLB Benchmark Model 1 Noding Diagram

a,c

Figure A.3-2 Short-Term RSLB Benchmark Model 2 Noding Diagram

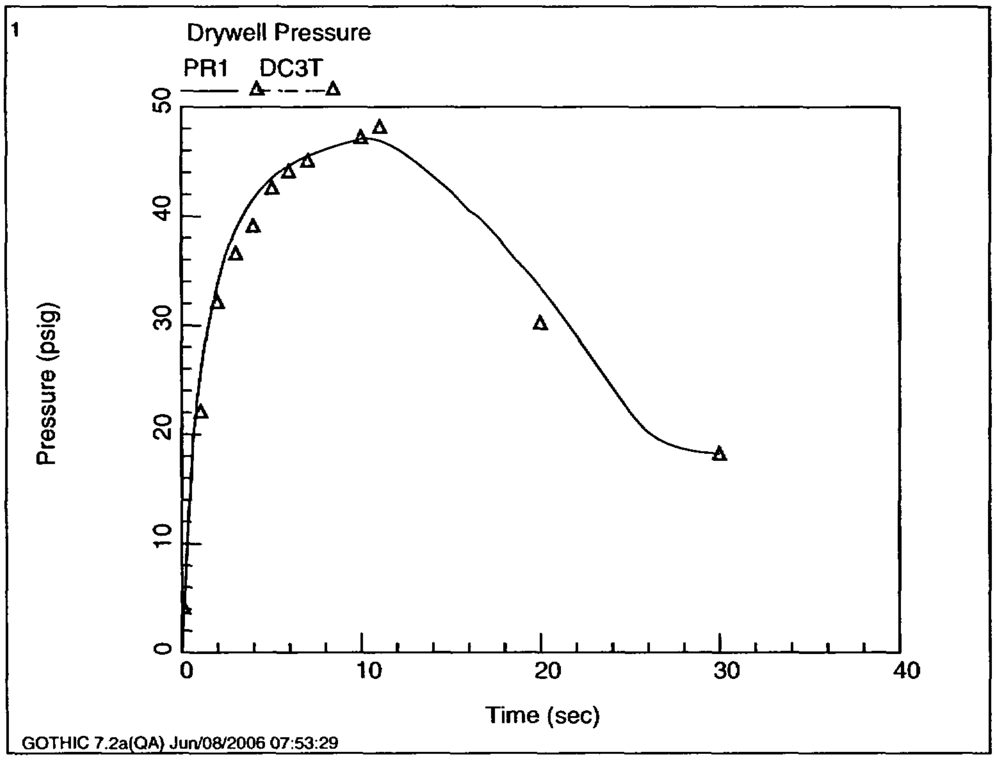

Figure A.3-3 Model 1 Short-Term RSLB Drywell Pressure Comparison

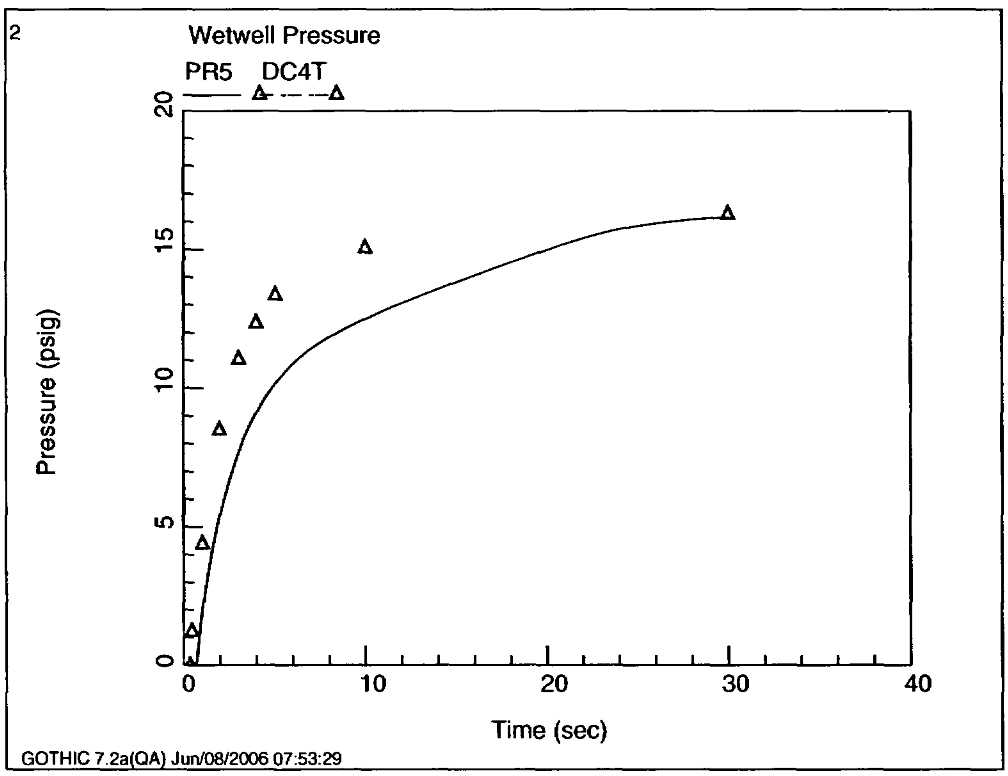

Figure A.3-4 Model 1 Short-Term RSLB Wetwell Pressure Comparison

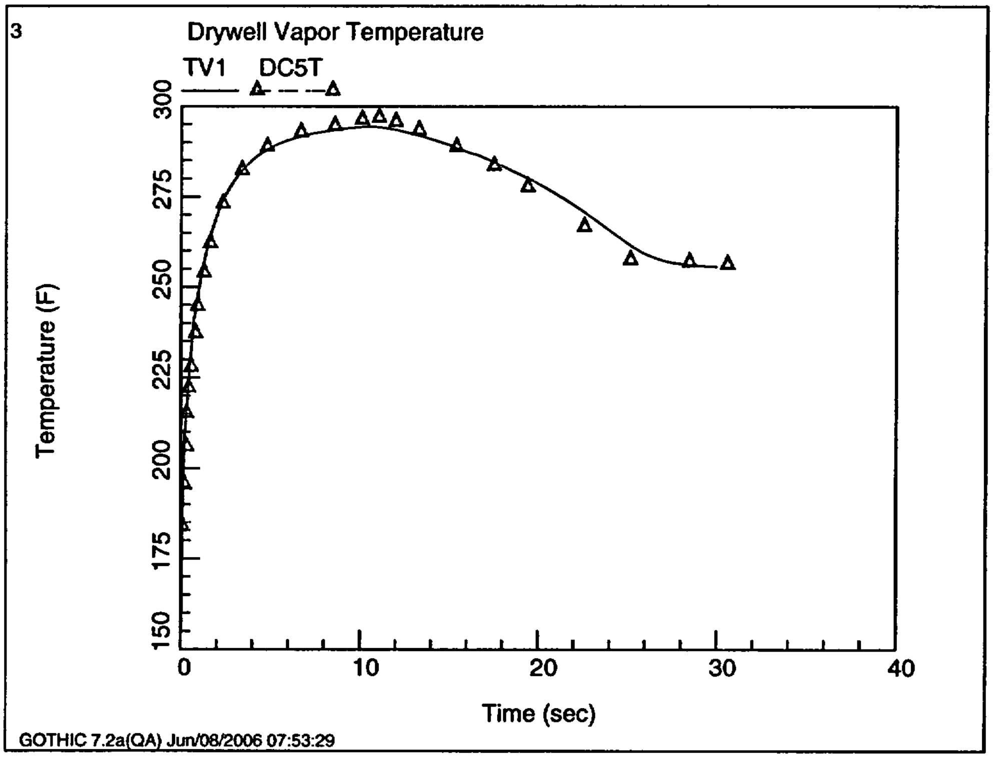

Figure A.3-5 Model 1 Short-Term RSLB Drywell Vapor Temperature Comparison

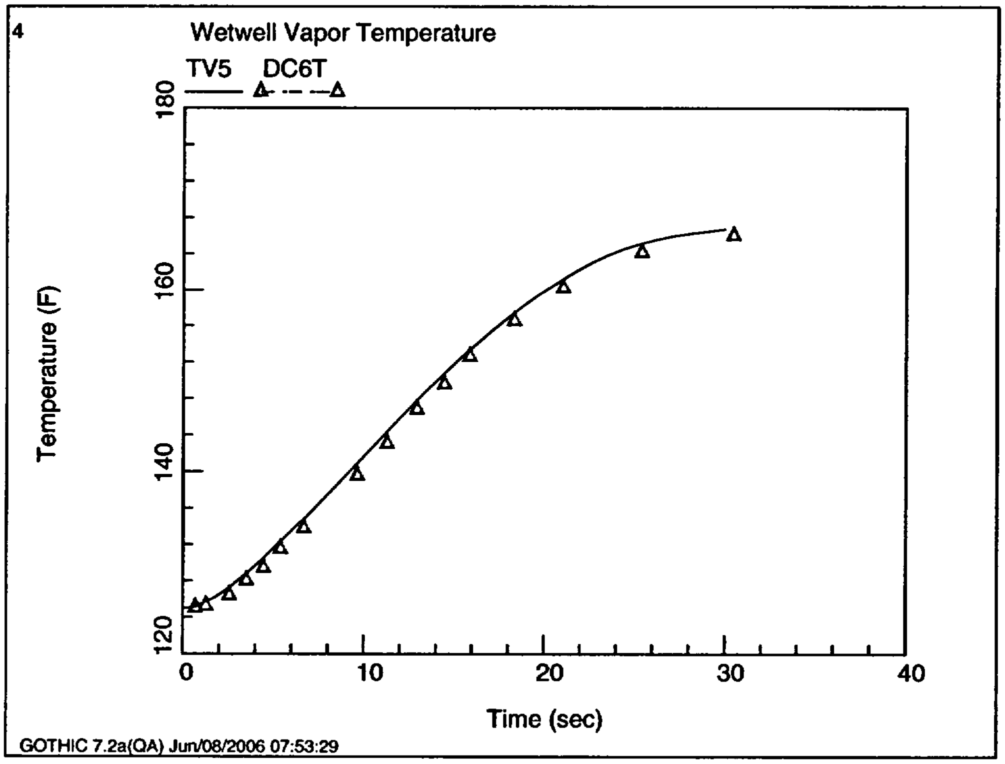

Figure A.3-6 Model 1 Short-Term RSLB Wetwell Vapor Temperature Comparison

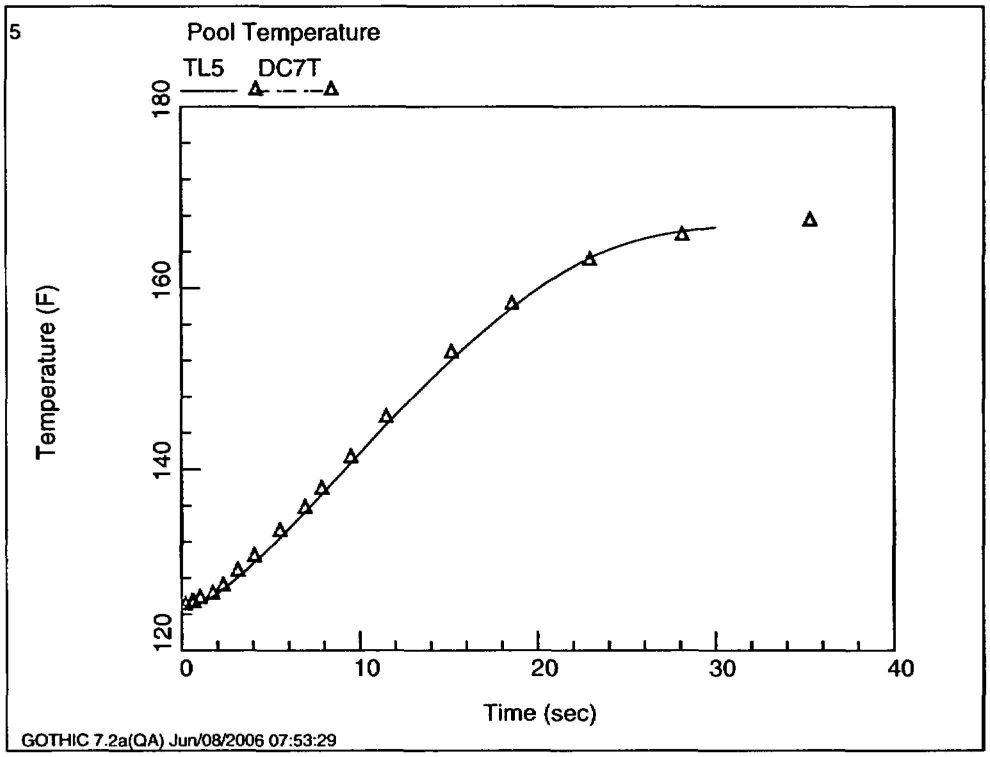

Figure A.3-7 Model 1 Short-Term RSLB Suppression Pool Temperature Comparison

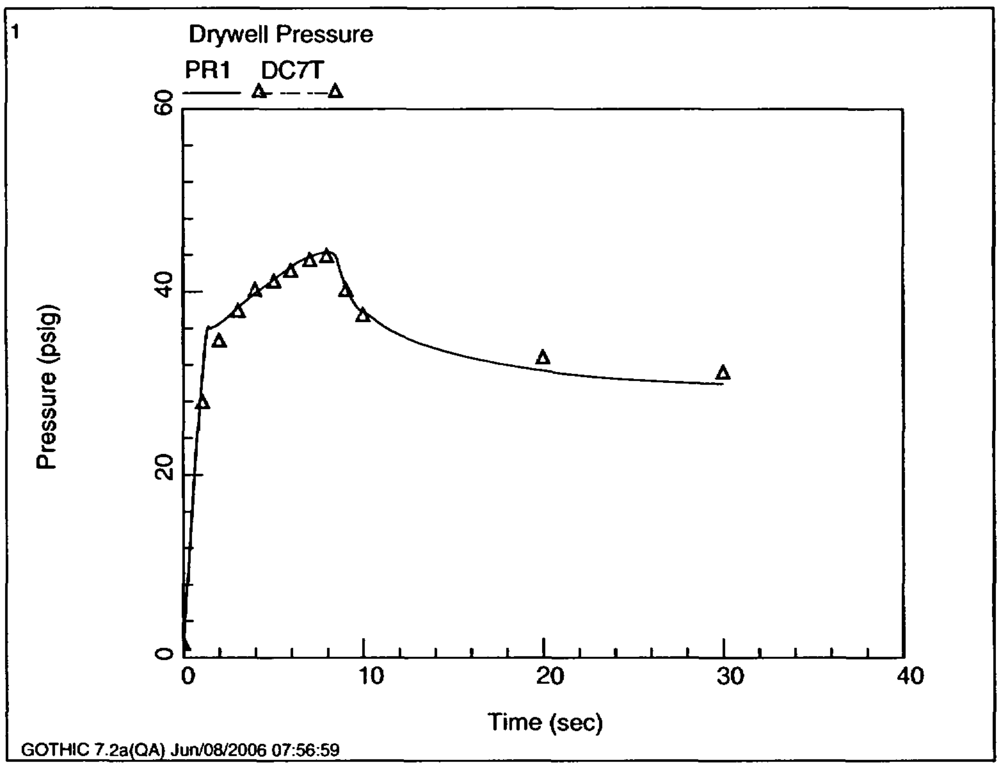

Figure A.3-8 Model 2 Short-Term RSLB Drywell Pressure Comparison

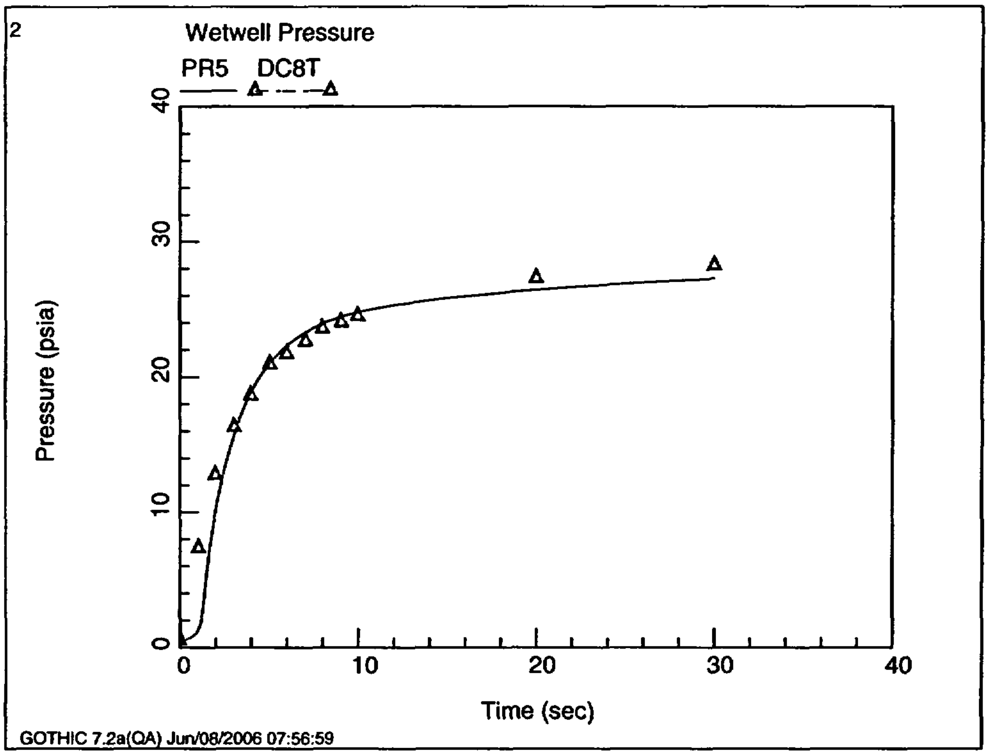

Figure A.3-9 Model 2 Short-Term RSLB Wetwell Pressure Comparison

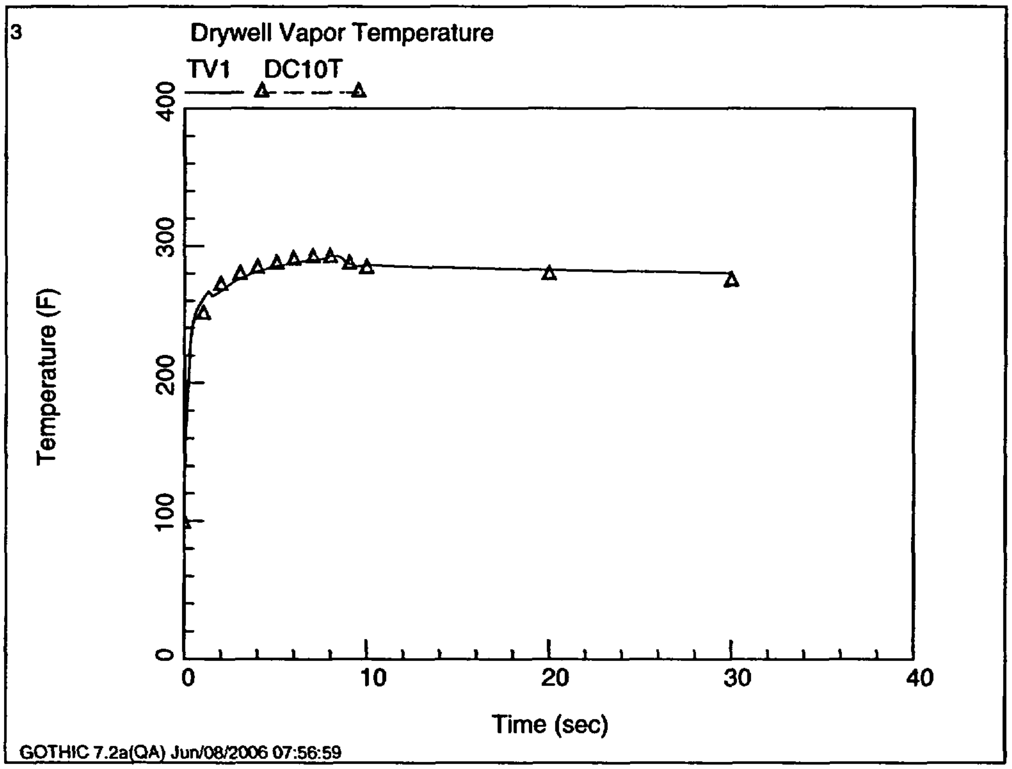

Figure A.3-10 Model 2 Short-Term RSLB Drywell Vapor Temperature Comparison

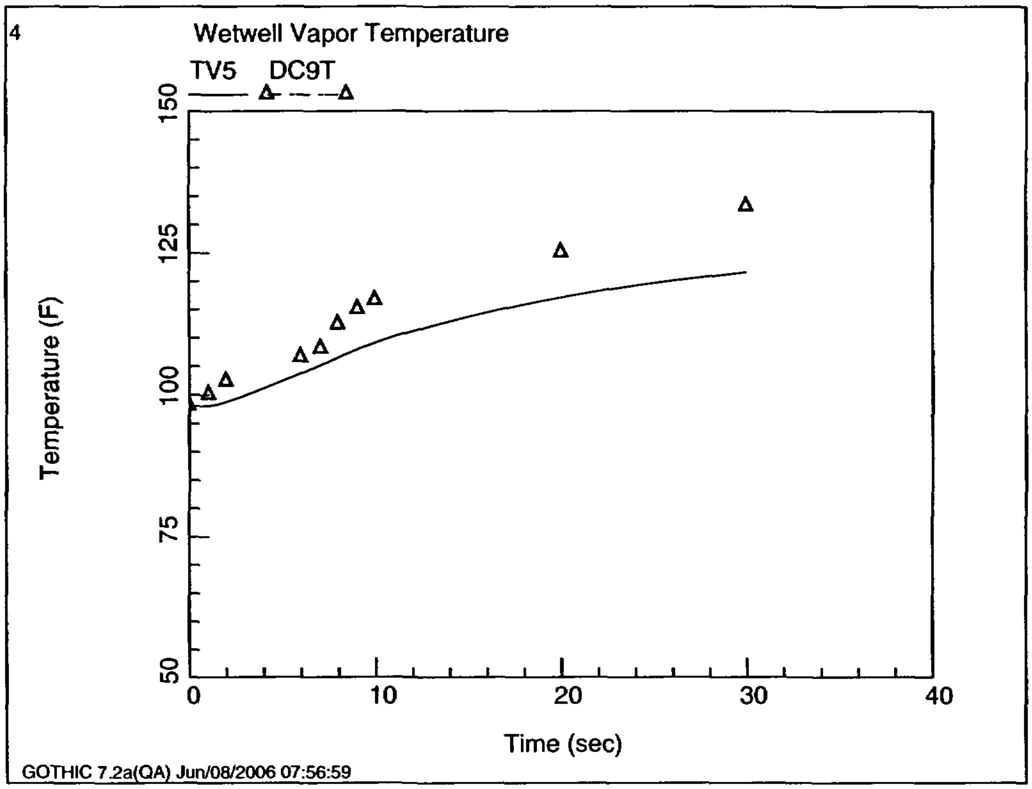

Figure A.3-11 Model 2 Short-Term RSLB Wetwell Vapor Temperature Comparison

Intermediate Break Accident Benchmark Comparison

The intermediate break accident (IBA) represents a liquid line break that is below the threshold where the loop selection logic can detect the broken recirculation loop (0.15 ft^2) and produces a break flow rate that is higher than the injection capability from a single HPCI pump. Typically, the break size is 0.1 ft^2 and it is assumed to be located in the recirculation suction line. The containment thermal-hydraulic response output data from this case is used in the hydrodynamic load analysis. The benchmark for this case was a GE SHEX analysis (Reference A-9, Figures 6.2-35 and 6.2-36).

The GOTHIC IBA benchmark comparison model is similar to Model 2, which was used for the short-term RSLB benchmark comparison. A noding diagram for the GOTHIC IBA benchmark comparison model is shown in Figure A.3-12. [

]a,c

In addition to the changes in the noding structure, changes were made to the torus volume, suppression pool water volume, downcomer loss coefficient, and downcomer exit inertia length input values for the IBA benchmark comparison. These input values were changed to match the values that were assumed to have been used in the SHEX benchmark analysis. A comparison of the key containment model input parameters from the GOTHIC IBA Model with Model 2, and the GOTHIC generic BWR Mark I containment model, is shown in Table A.3-2.

The IBA benchmark comparison results are shown in Figures A.3-13 through A.3-16. The GOTHIC IBA model tends to under-predict the benchmark data. The GOTHIC vessel model was used to calculate the mass and energy release input for the IBA benchmark case since the mass and energy input data for the containment benchmark case was not available. This may be why the containment pressure and temperature were under-predicted.

Table A.3-2 Comparison of Key Containment Model Input Values			
Value	**IBA Model**	**Model 2**	**Generic Model**
Drywell Volume			
Torus Volume			
Vent Line/Header/Downcomer Volume			
Reactor Vessel Volume			
Downcomer Flow Area			
Downcomer Loss Coefficient			
Downcomer Inlet Inertia Length			
Downcomer Exit Inertia Length			
Initial Vessel Water Volume			
Initial Torus Water Volume			
Initial Drywell Pressure			
Initial Wetwell Pressure			
Initial Downcomer Submergence			
Initial Vessel Pressure			
Initial Drywell Temperature			
Initial Wetwell Temperature			

a,c

Note:
1. This value was tuned to allow the GOTHIC model to match the RSLB benchmark data.

a,c

Figure A.3-12 IBA Benchmark Model Noding Diagram

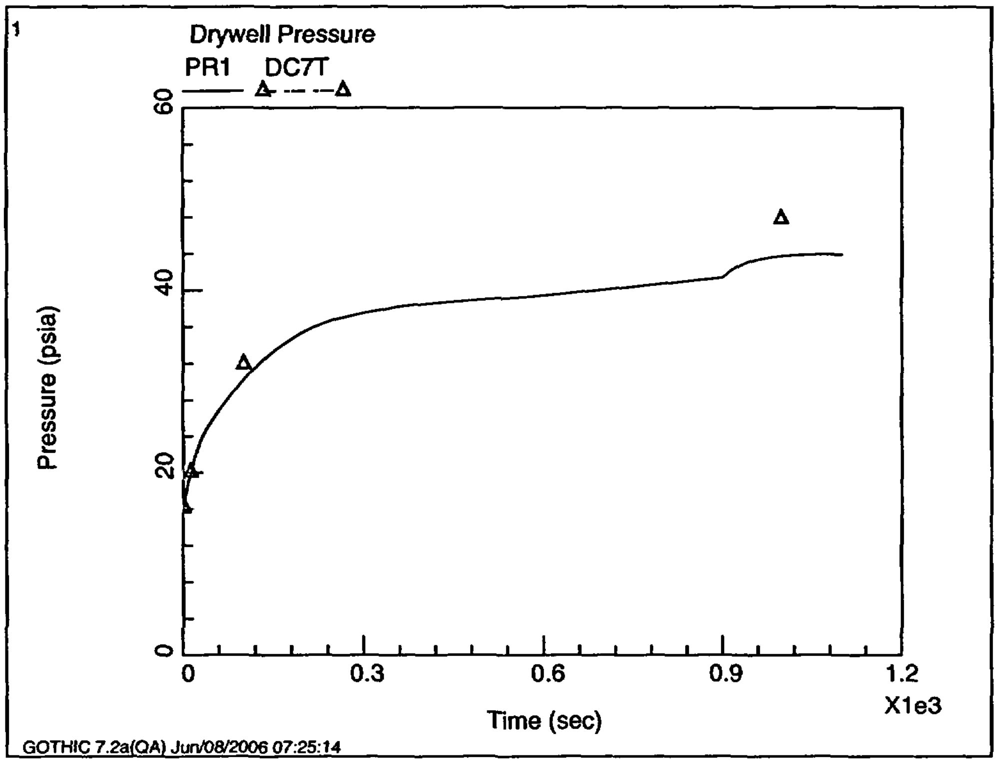

Figure A.3-13 IBA Drywell Pressure Comparison

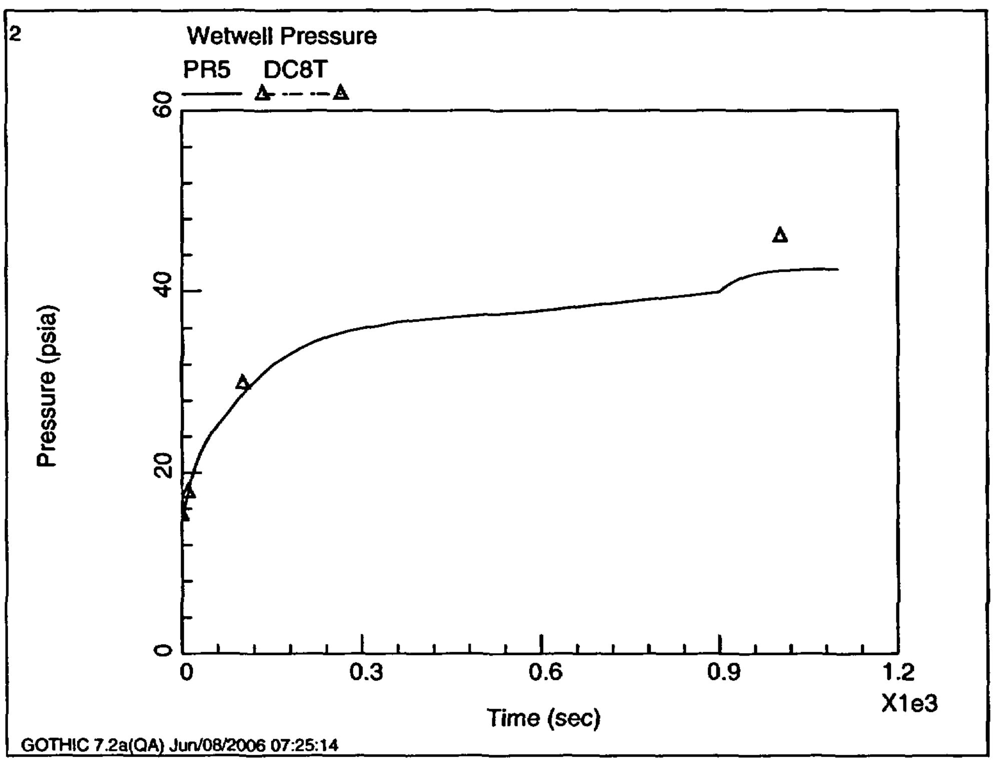

Figure A.3-14 IBA Wetwell Pressure Comparison

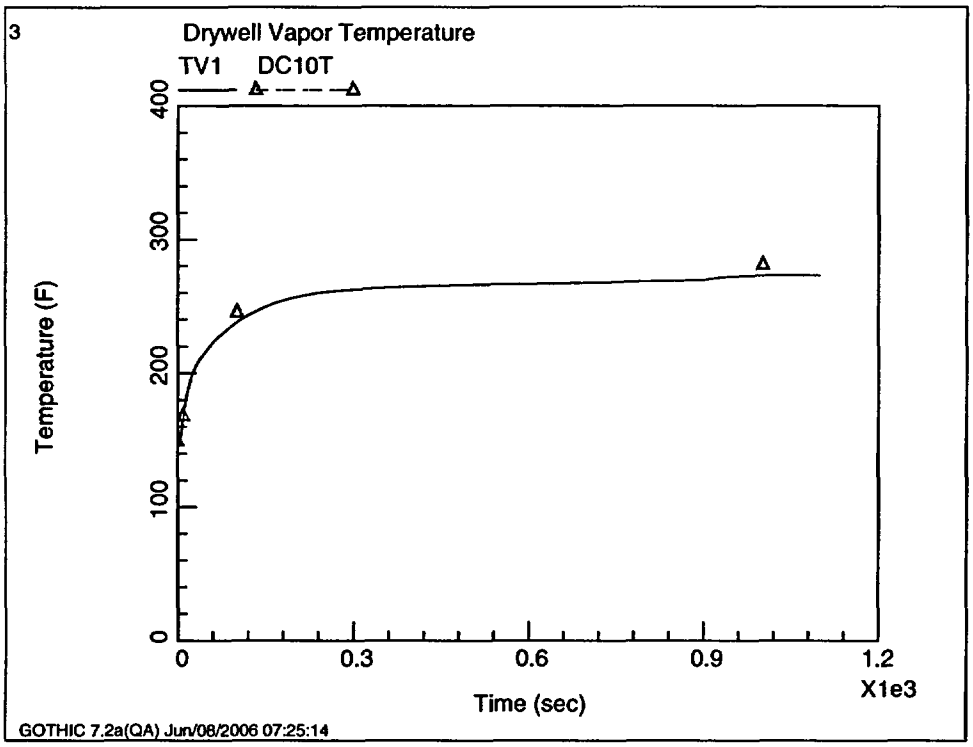

Figure A.3-15 IBA Drywell Vapor Temperature Comparison

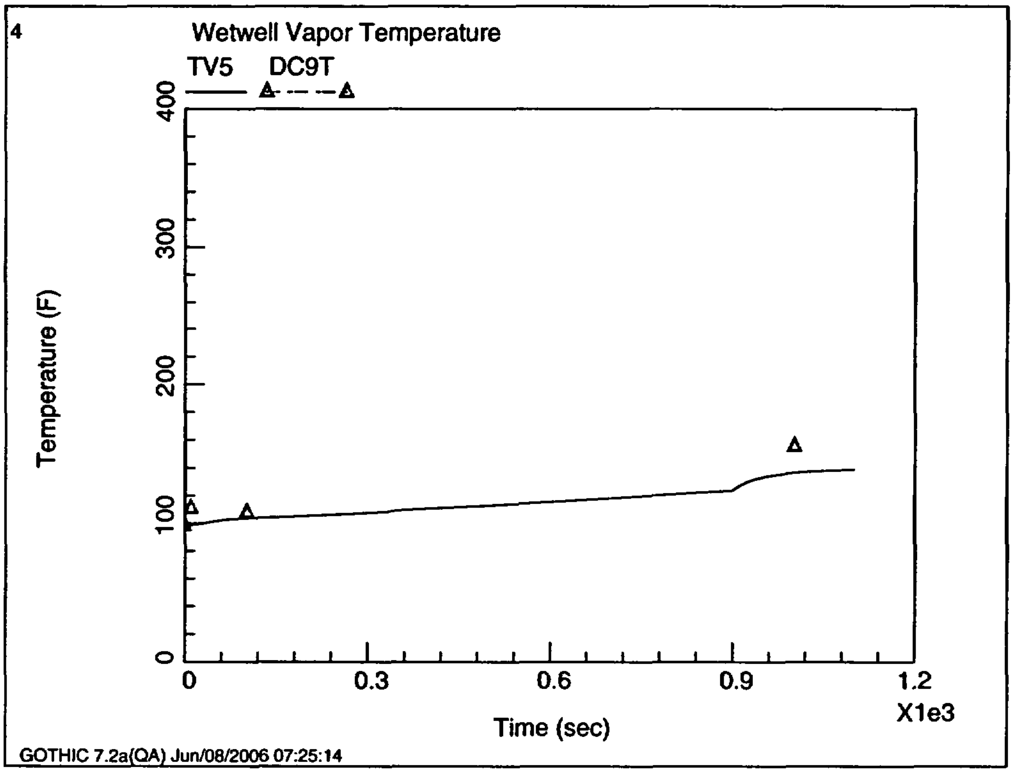

Figure A.3-16 IBA Wetwell Vapor Temperature Comparison

Small Break Accident Benchmark Comparison

This small break accident is assumed to be a small (0.05 ft^2) steam line break. The break is small enough that the reactor control systems would simply treat it as an increase in turbine demand. The SBA event typically results in the peak containment drywell temperature. The benchmark for this case was a Commonwealth Edison Company analysis performed with CONTEMPT (Reference A-10).

The GOTHIC SBA benchmark comparison model is similar to Model 1, which was used for the short-term RSLB benchmark comparison. A noding diagram for the GOTHIC SBA benchmark comparison model is shown in Figure A.3-17. [

]^{a,c}

In addition to modeling the thermal conductors, changes were made to the suppression pool water volume, downcomer loss coefficient, and initial condition input values for the SBA benchmark comparison. A comparison of the key containment model input parameters from the GOTHIC SBA Model with Model 1, and the GOTHIC generic BWR Mark I containment model, is shown in Table A.3-3.

The SBA benchmark comparison results are shown in Figures A.3-18 through A.3-22. The steam release to the drywell causes the pressure and temperature to increase. The water level in the downcomer decreases as the drywell pressure increases; this eventually allows steam and air to enter the suppression pool. The pool temperature begins to increase due to steam condensation, but hot air bubbles rising through the water and compression of the wetwell vapor volume causes the wetwell vapor temperature to be higher than the pool temperature. GOTHIC predicts slightly higher drywell and wetwell pressures, and a lower initial wetwell vapor temperature than the benchmark. These differences were attributed to differences in the modeling of condensation heat and mass transfer between GOTHIC and CONTEMPT.

Table A.3-3 Comparison of Key Containment Model Input Values

a,c

Value	SBA Model	Model 1	Generic Model
Drywell Volume			
Torus Volume			
Vent Line/Header/Downcomer Volume			
Reactor Vessel Volume			
Downcomer Flow Area			
Downcomer Loss Coefficient			
Downcomer Inlet Inertia Length			
Downcomer Exit Inertia Length			
Initial Vessel Water Volume			
Initial Torus Water Volume			
Initial Drywell Pressure			
Initial Wetwell Pressure			
Initial Downcomer Submergence			
Initial Vessel Pressure			
Initial Drywell Temperature			
Initial Wetwell Temperature			

Note:
1. This value was tuned to allow the GOTHIC model to match the RSLB benchmark data.

a,c

Figure A.3-17 SBA Benchmark Model Noding Diagram

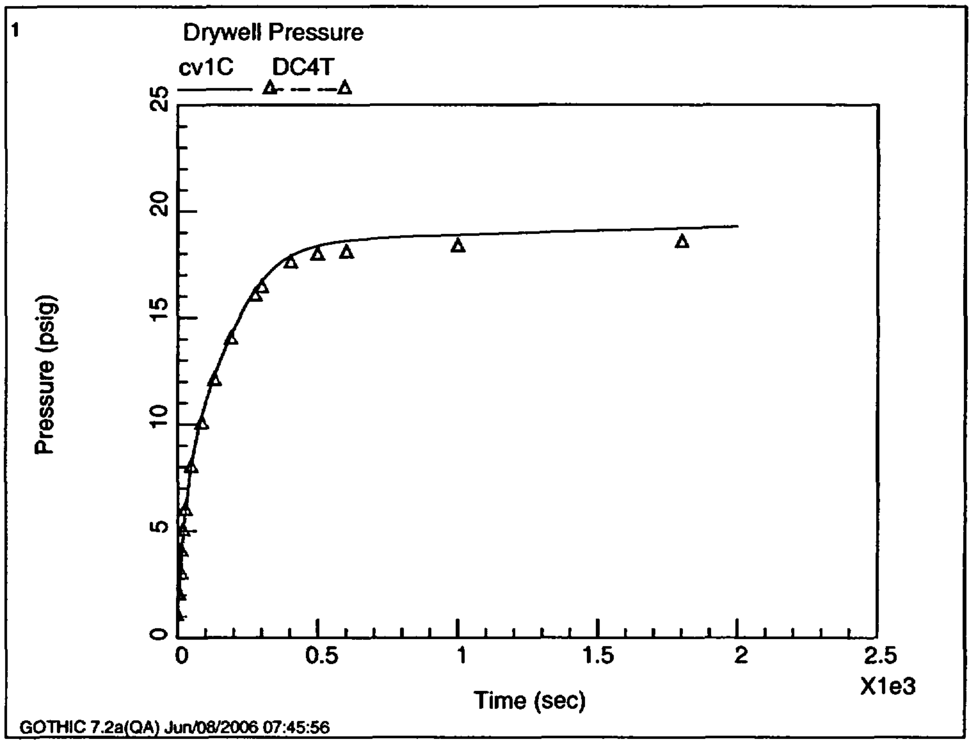

Figure A.3-18 SBA Drywell Pressure Comparison

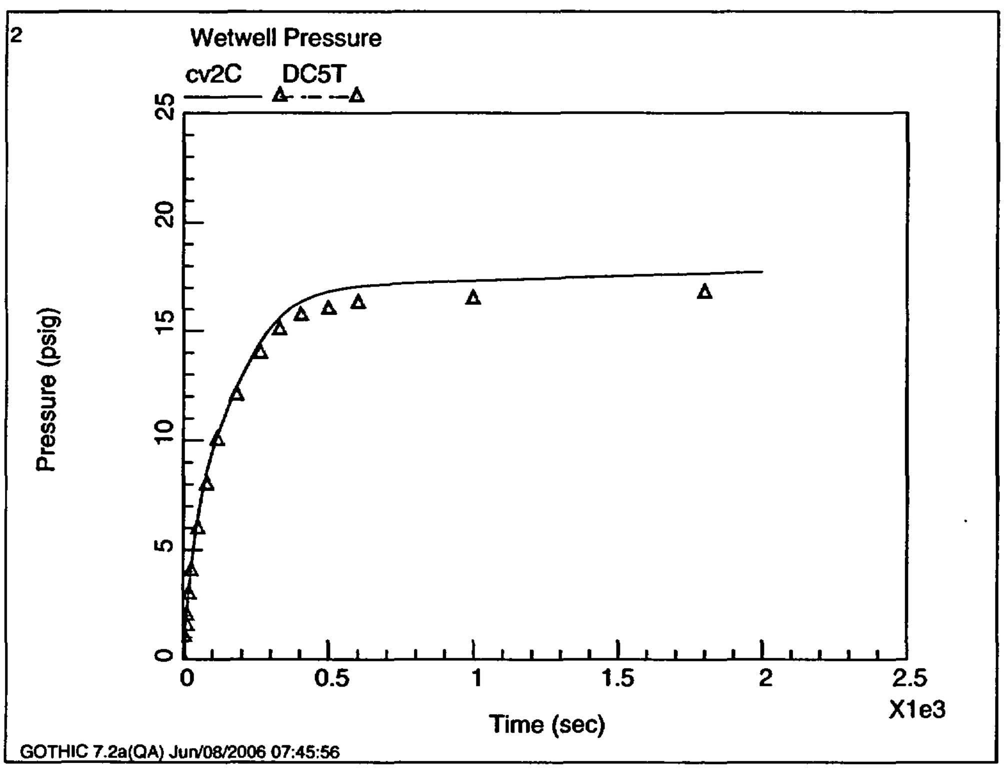

Figure A.3-19 SBA Wetwell Pressure Comparison

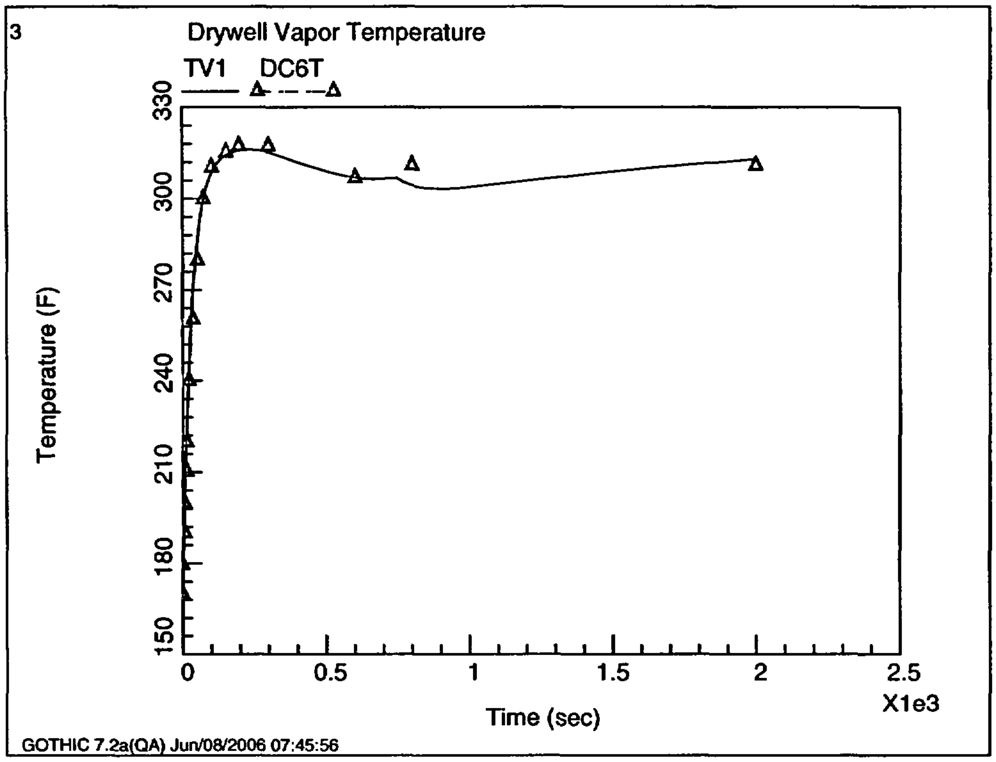

Figure A.3-20 SBA Drywell Vapor Temperature Comparison

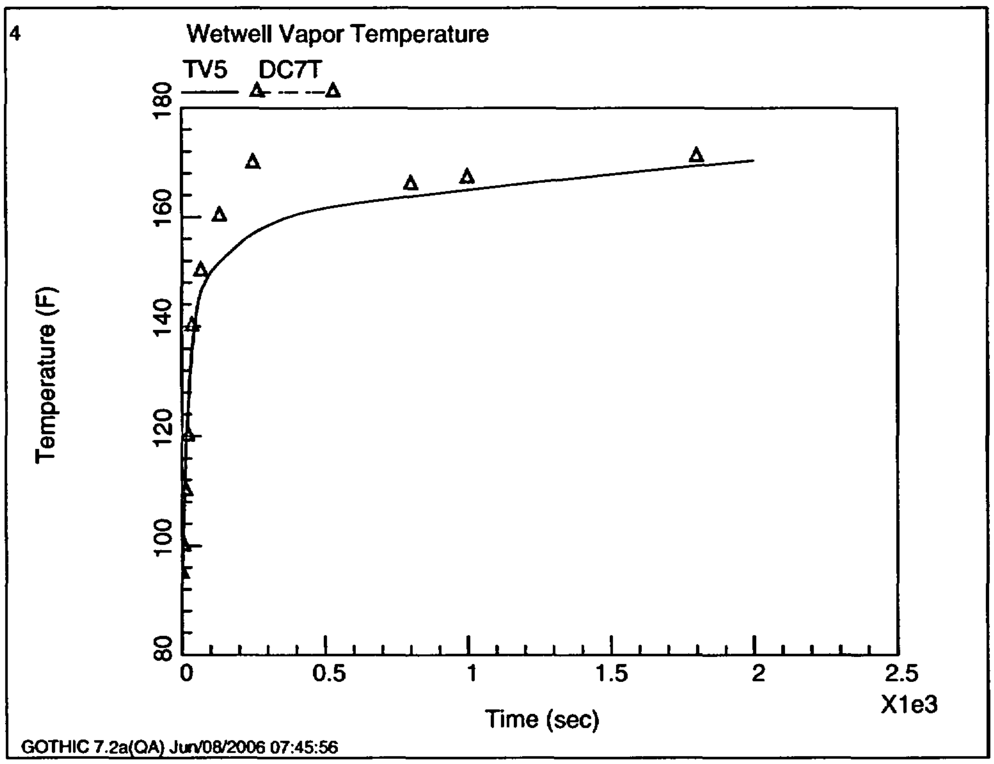

Figure A.3-21 SBA Wetwell Vapor Temperature Comparison

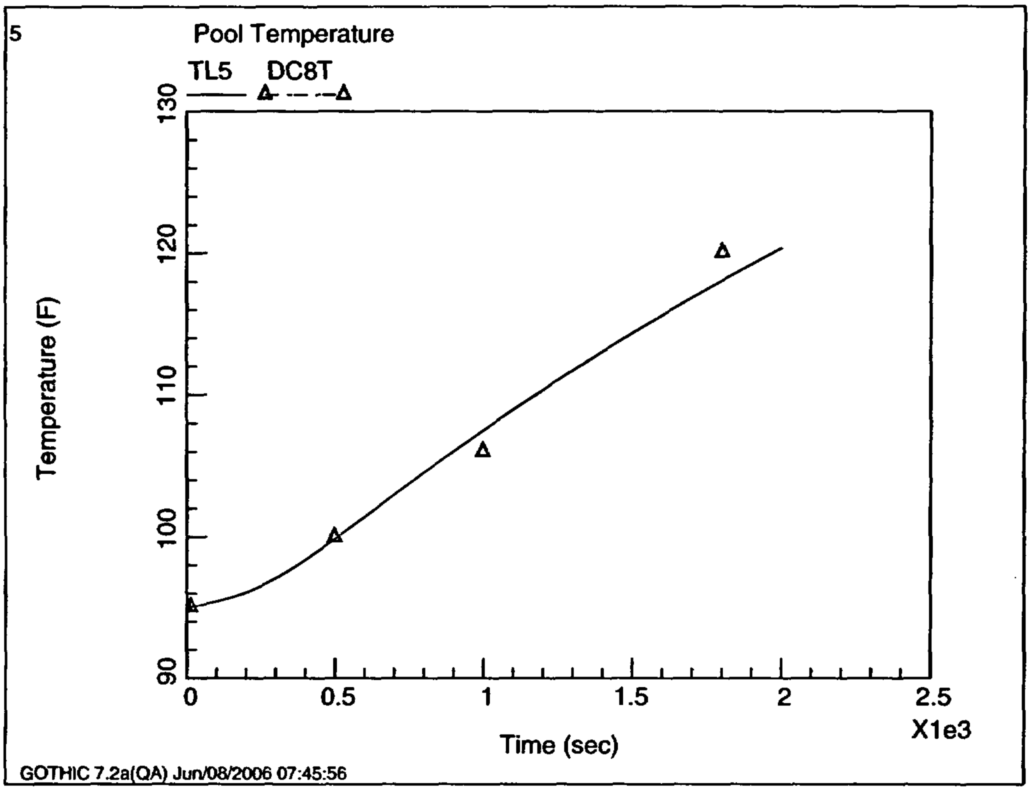

Figure A.3-22 SBA Suppression Pool Temperature Comparison

A.4 SAMPLE TRANSIENT RESULTS

This section presents results for representative containment analysis applications using the GOTHIC generic BWR Mark I containment model described in Section A.2. The purpose of these sample cases is to demonstrate that the GOTHIC model has analysis margin to the containment design limits.

A complete circumferential break of one of the recirculation suction legs is typically the design basis loss of coolant accident (LOCA) for the boiling water reactor (BWR) design. Four separate containment thermal-hydraulic analyses are performed for this event: one to determine the peak containment pressure, one to determine the minimum ECCS containment backpressure, one to provide input for the calculation of the minimum NPSHa for the ECCS pumps, and one to determine the peak suppression pool temperature and long-term containment pressure and temperature response for equipment qualification. This section contains a sample case for each of these four applications.

An MSLB event also releases a large amount of mass and energy to the containment. The MSLB event can be analyzed with various break sizes and initial reactor power levels to identify the limiting case; the flow area of the steam line flow restrictor places an upper limit on the break size. This section contains two MSLB sample cases; one biased to calculate the peak containment pressure, and the other biased to calculate the long-term containment pressure and temperature response for equipment qualification.

The SBA event typically produces the peak containment drywell temperature. Operator action is required to mitigate the consequences of this event. This section contains a sample case for an SBA event; the input is biased to maximize the drywell temperature.

The ATWS and SBO events are also analyzed to determine the containment response. Although these events are typically not included in the containment design basis, one or more of the containment design limits could be challenged without operator recovery action. This section contains a sample case for the ATWS event; the input is biased to maximize the suppression pool temperature.

A.4.1 RSLB Peak Pressure

This case calculates the peak containment pressure and only considers the containment response during the blowdown portion of the RSLB event (prior to ECCS injection). A modified GOBLIN ECCS evaluation model, as described in Appendix B, was used to produce the RSLB mass and energy release input for the GOTHIC containment response calculation. The GOTHIC vessel model is not used to generate mass and energy release data for this case

The GOTHIC generic BWR Mark I containment model was initialized at a steady state condition; the input changes that were made to run the RSLB peak pressure transient are listed below.

[

]a,c

[

]a,c

The results for this case are shown in Figures A.4.1-1 through A.4.1-3. The drywell pressure and temperature increase rapidly due to the large release of steam and hot water from the RSLB (Figures A.4.1-1 and 2). The increase in drywell pressure causes the water level in the downcomer pipes to decrease; a vent path to the suppression pool is created after the downcomer pipes empty. The initial drywell pressure peak (46.5 psia) is determined by the vent path inertia and the subsequent pressure peak (47.1 psia) is determined by the vent path resistance. The wetwell pressure increases as the air from the drywell is transferred to the wetwell and the suppression pool water level increases, due to the break mass flow rate. The suppression pool temperature increases as the steam is condensed (Figure A.4.1-3).

Several sensitivity cases were made using this containment model to determine the impact of changes to various input values on the calculated peak pressure response. The following sensitivity cases were performed:

1. The containment heat sinks were shut off by setting the heat transfer coefficients to 0.0.

2. The ANS 5.1-1979 decay heat standard (+ 2 sigma uncertainty) was used in the GOBLIN mass and energy release calculation.

3. The vent path inertia input value was doubled.

4. The vent path resistance input value was doubled.

The comparison of the drywell pressure response for these cases with the base case is shown in Figure A.4.1-4. Removing the containment heat sinks or using the ANS 5.1-1979 + 2 sigma decay heat standard (instead of the ANS 5.1-1971 + 20% decay heat standard) did not have a significant impact on the drywell peak pressure calculation. The drywell peak pressure calculation was found to be most sensitive to changing the vent path input values. Increasing the vent path inertia input value caused the first peak to increase substantially and increasing the vent path resistance input value caused the second peak to increase substantially.

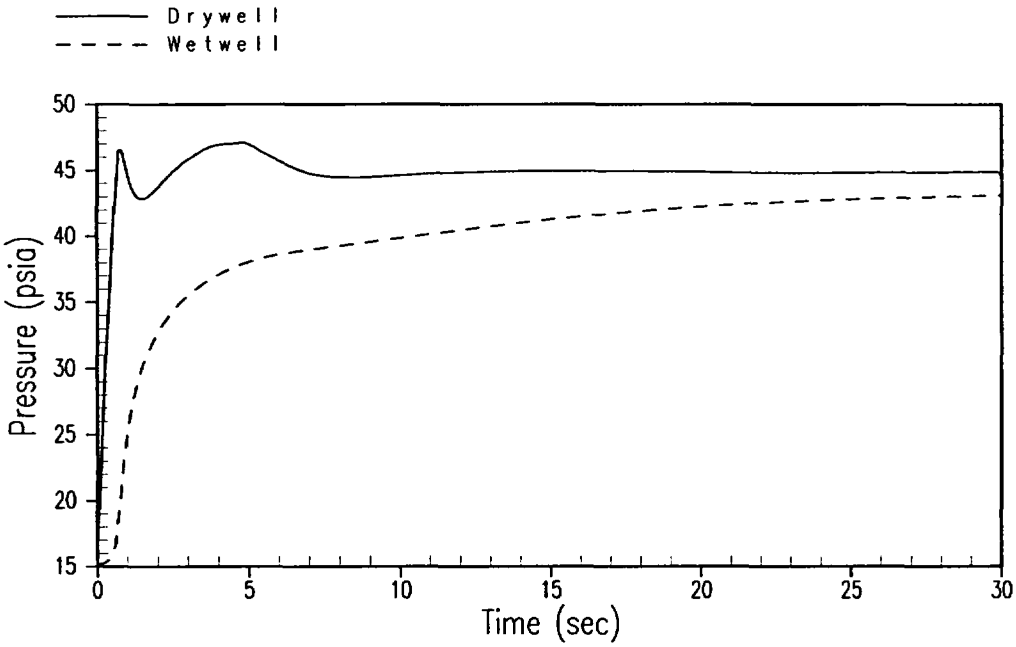

Figure A.4.1-1 RSLB Case 1 Pressure

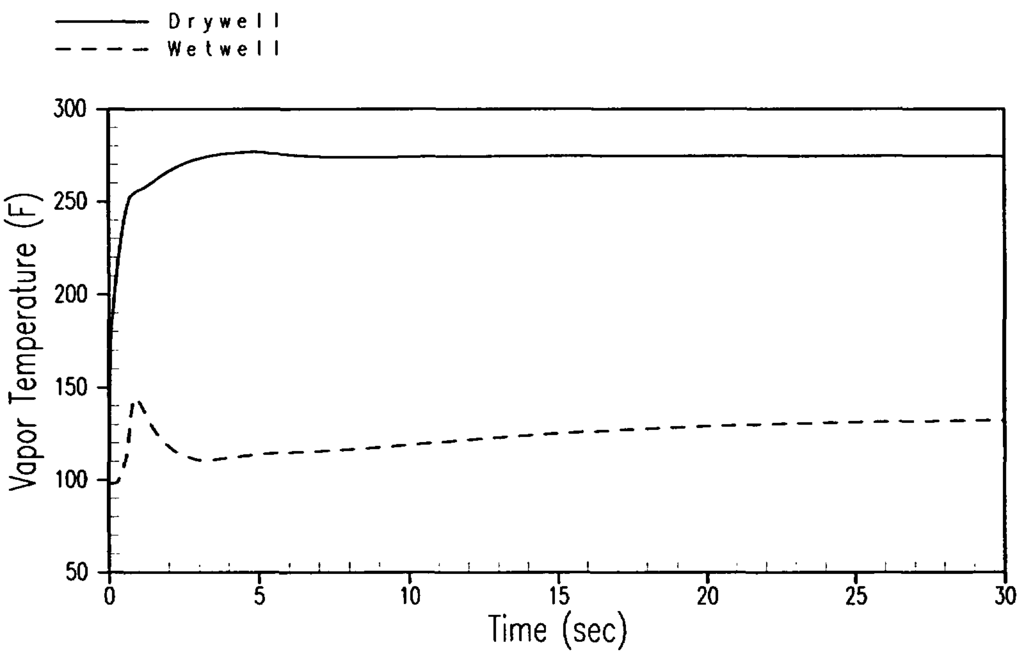

Figure A.4.1-2 RSLB Case 1 Vapor Temperature

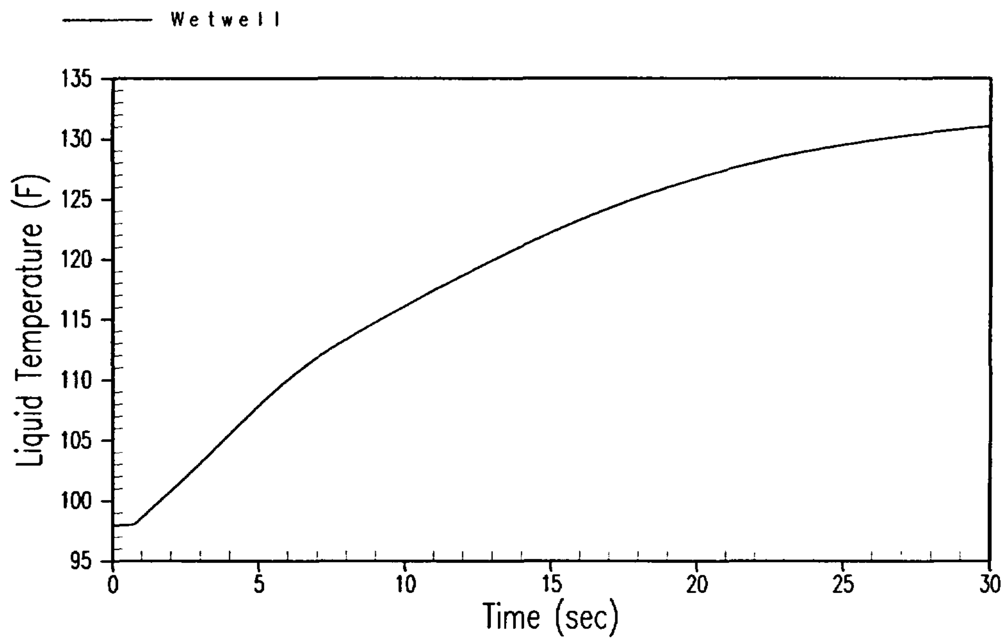

Figure A.4.1-3 RSLB Case 1 Suppression Pool Temperature

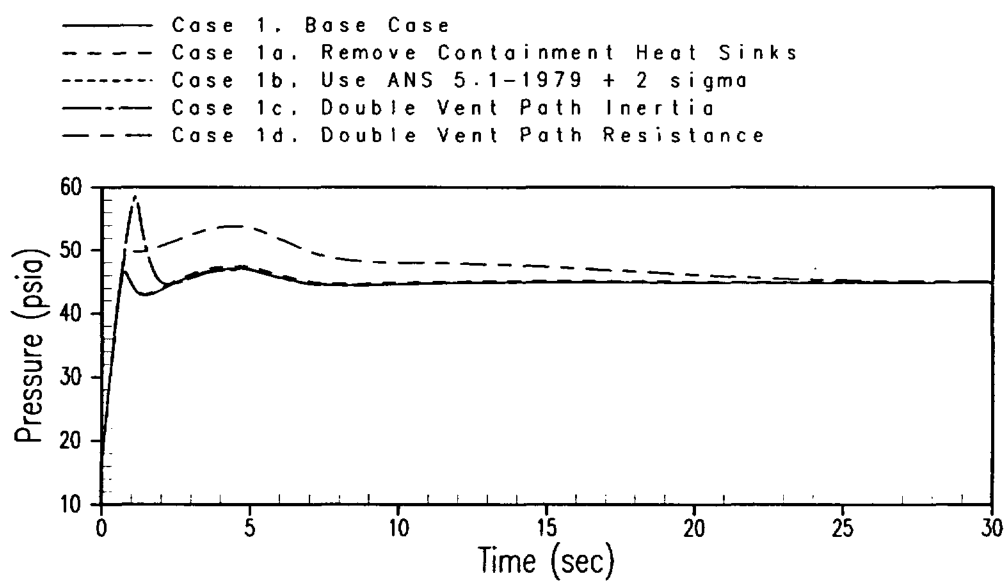

Figure A.4.1-4 RSLB Drywell Pressure Sensitivity Case Comparison

A.4.2 RSLB Minimum ECCS Backpressure

This case calculates a minimum containment backpressure for the ECCS analysis; it covers both the blowdown and reflood phases of the RSLB event. A modified GOBLIN ECCS evaluation model, as described in Appendix B, was used to produce the RSLB mass and energy release input for the GOTHIC containment response calculation. The RSLB mass and energy release data from a limiting PCT case was also used in a sensitivity comparison case. The GOTHIC vessel model was not used to generate the mass and energy release for these cases.

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]^{a,c}

The GOTHIC generic BWR Mark I containment model was initialized at a steady state condition; the input changes that were made to run the RSLB minimum ECCS backpressure transient are listed below.
[

]^{a,c}

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][a,c]

The results for this case are shown in Figures A.4.2-1 through A.4.2-3. The drywell pressure increases, the downcomer vent path to the suppression pool opens, and the large blowdown mass and energy release is over within approximately 30 seconds after the break occurs. After blowdown, the drywell pressure decreases and remains at approximately 35 psia for about 2 minutes while the ECCS pumps refill the reactor vessel (Figure A.4.2-1). After the reactor vessel has refilled, the cooler ECCS water begins to spill from the broken recirculation pump suction line causing the steam in the drywell to begin condensing. The condensation rate has been conservatively enhanced for this case by [ ][a,c]. The vacuum breakers open as the drywell pressure decreases below the wetwell pressure. This allows air from the wetwell to return to the drywell. The containment pressure decreases and remains at approximately 25 psia for the remainder of the 10 minute transient.

A minimum ECCS backpressure sensitivity case was made using the mass and energy release data from a limiting PCT analysis. The limiting PCT analysis assumed the ECCS flow from all 4 of the LPCI pumps was diverted to the intact recirculation line; however, a single failure of the LPCI injection valve to that line prevented this flow from reaching the vessel. The following input changes were made to the original minimum ECCS backpressure model for this case:

1. The GOBLIN limiting PCT mass and energy release nsaplot data file name was entered into the component input for all of the control variables that use the external function.

2. The GOTHIC LPCI flow boundary condition multiplier was set to 0.0 to prevent LPCI flow from being injected to the reactor vessel.

3. The transient end time was set to the end of GOBLIN time for the limiting PCT case (235 seconds).

The drywell pressure and suppression pool temperature results for the sensitivity case are compared with the initial minimum ECCS backpressure case and are shown in Figures A.4.2-4 and A.4.2-5. The drywell pressure response is nearly identical for the first 120 seconds. After this, the drywell steam condensation rate increases and the pressure decreases in the base mass and energy release case because liquid break flow is being released to the drywell atmosphere. It takes longer to refill the reactor vessel and begin spilling liquid break flow to the drywell in the limiting PCT case because none of the LPCI flow is being injected to the vessel. The suppression pool temperature remains slightly lower in the limiting PCT case (Figure A.4.2-5) because the PCT model does not include the [ ][a,c].

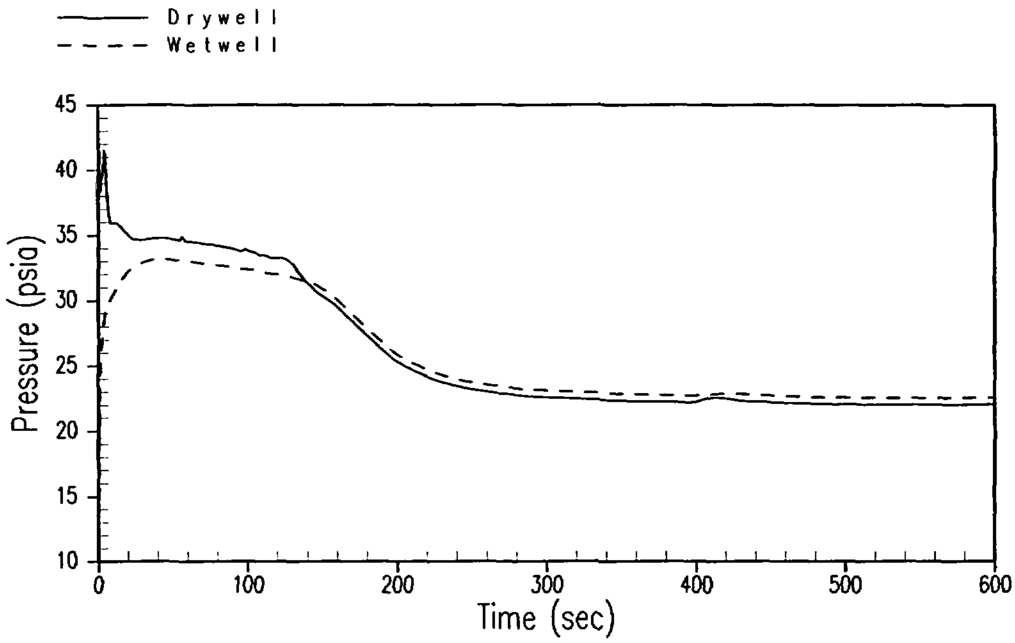

Figure A.4.2-1 RSLB Case 2 Pressure

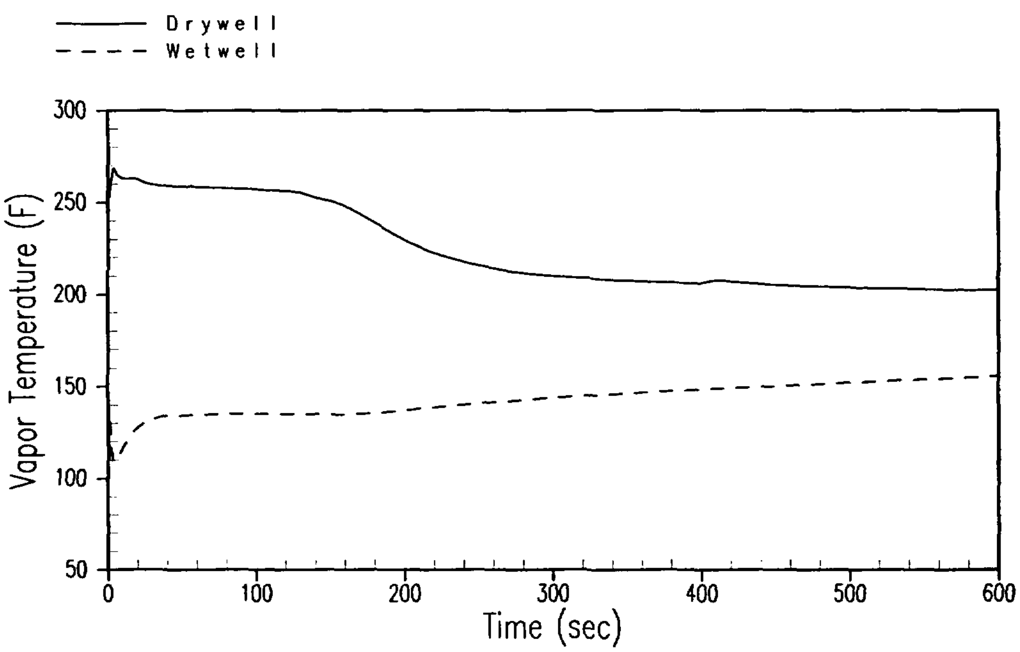

Figure A.4.2-2 RSLB Case 2 Vapor Temperature

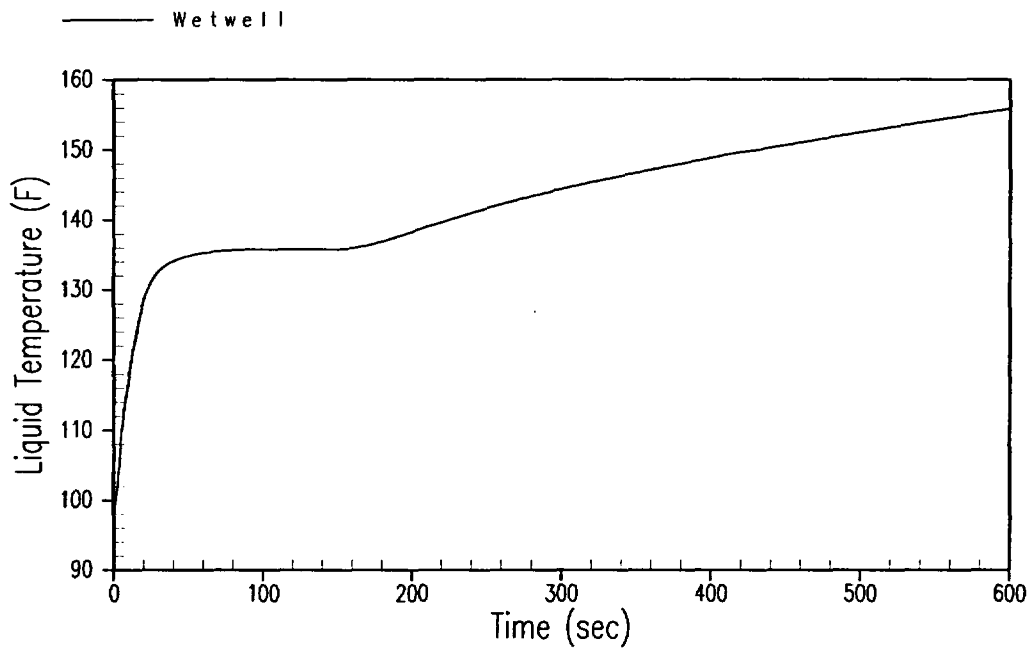

Figure A.4.2-3 RSLB Case 2 Suppression Pool Temperature

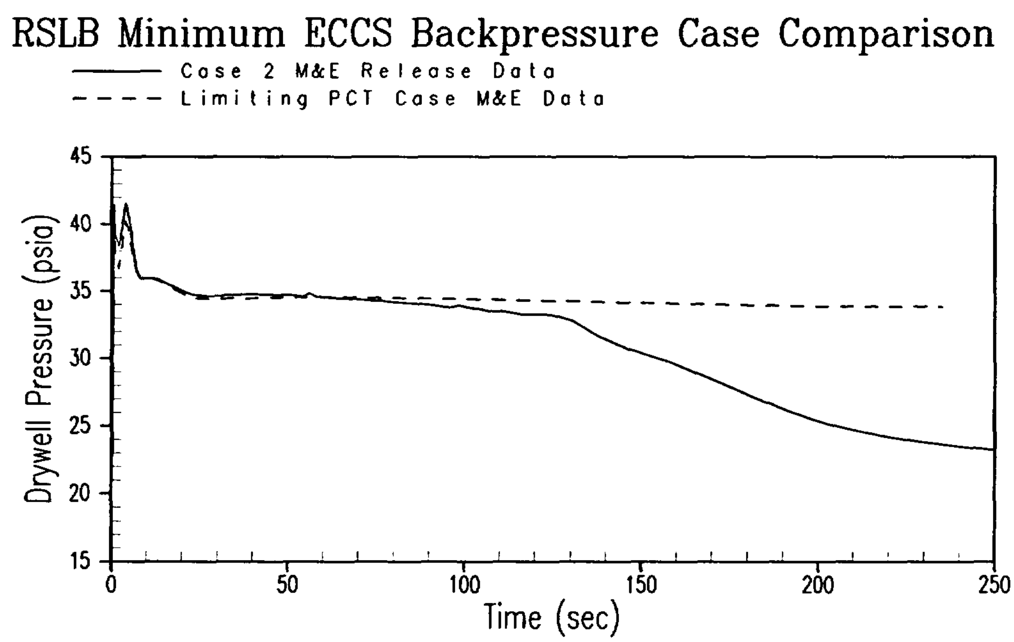

Figure A.4.2-4 RSLB Minimum ECCS Backpressure Case Drywell Pressure Comparison

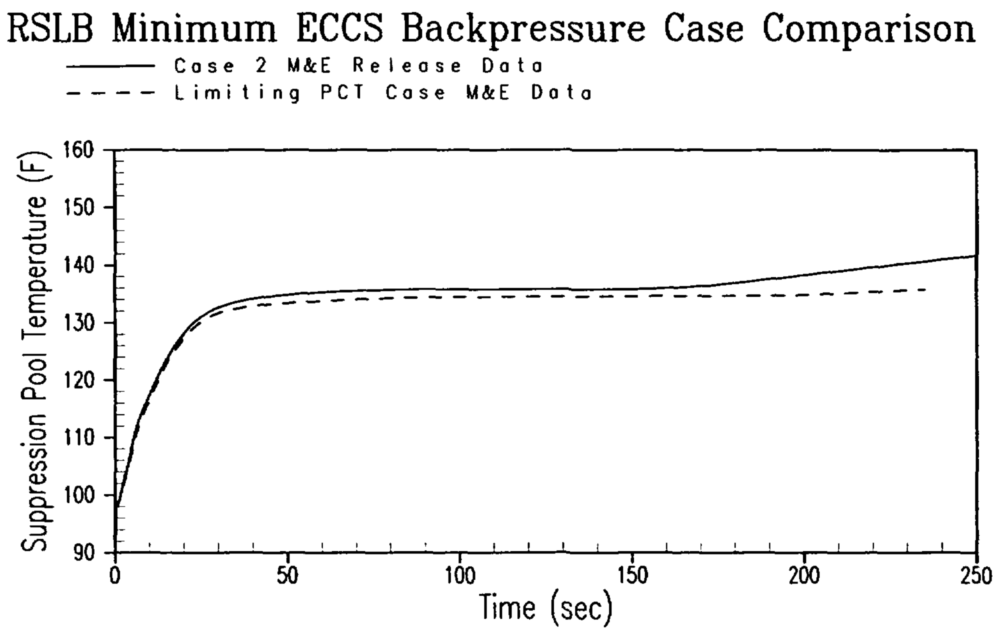

Figure A.4.2-5 RSLB Minimum ECCS Backpressure Case Suppression Pool Temperature Comparison

A.4.3 RSLB Minimum NPSHa

The output from this case is used as input to the minimum net positive suction head available (NPSHa) calculation for the ECCS pumps; it covers both the blowdown and long-term containment response for the RSLB event. A loss of offsite power with a subsequent failure of 1 EDG (loss of 2 LPCI and 1 LPCS pumps) was assumed for this sample case. A modified GOBLIN ECCS evaluation model, as described in Appendix B, was used to produce the RSLB blowdown and reflood phase mass and energy release input for the GOTHIC containment response calculation. The GOTHIC vessel model was used to generate the long-term mass and energy release.

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]^{a,c}

The GOTHIC generic BWR Mark I containment model was initialized at steady state conditions; the input changes that were made to run the RSLB minimum NPSHa transient are listed below.

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]a,c

The results for this case are shown in Figures A.4.3-1 through A.4.3-3; the transient response is similar to the RSLB minimum ECCS backpressure case, but the vessel refills slower because only 2 LPCI pumps and 1 LPCS pump are assumed to be operating. The drywell pressure increases, the downcomer vent path to the suppression pool opens, and the large blowdown mass and energy release is over within approximately 30 seconds after the break occurs. After blowdown, the drywell pressure decreases and remains at approximately 35 psia for about 3 minutes while the ECCS pumps refill the reactor vessel (Figure A.4.3-1). After the reactor vessel has refilled, the liquid break flow begins to spill from the broken recirculation pump suction line causing the steam in the drywell to begin condensing. The condensation rate has been conservatively enhanced for this case by [ ]a,c. The vacuum breakers open as the drywell pressure decreases below the wetwell pressure. This allows air from the wetwell to return to the drywell. The containment pressure decreases and remains between 20 and 25 psia for the remainder of the transient. A minimum operator action time of 10 minutes is assumed in the generic model. Since the containment conditions exceed the setpoint for spray initiation at 10 minutes, the operator is assumed to trip one of the LPCI pumps and align the flow from the other LPCI pump through the RHR heat exchanger to initiate containment spray to the drywell and wetwell at 600 seconds. The suppression pool temperature continues to increase until about 30,000 seconds (Figure A.4.3-3), when the containment spray heat removal rate exceeds the sum of the core decay and pump heat rates.

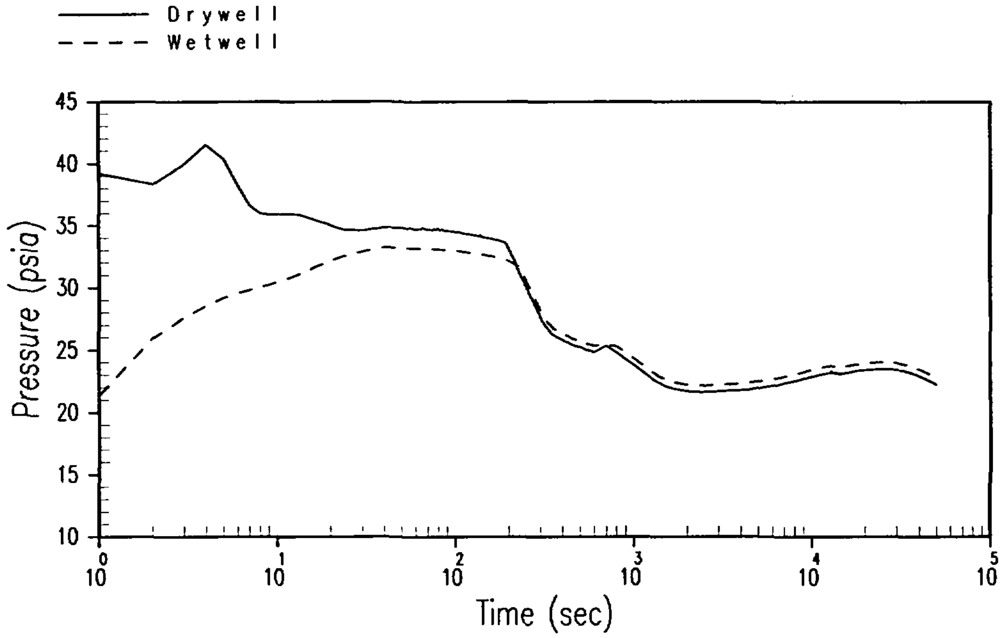

Figure A.4.3-1 RSLB Case 3 Pressure

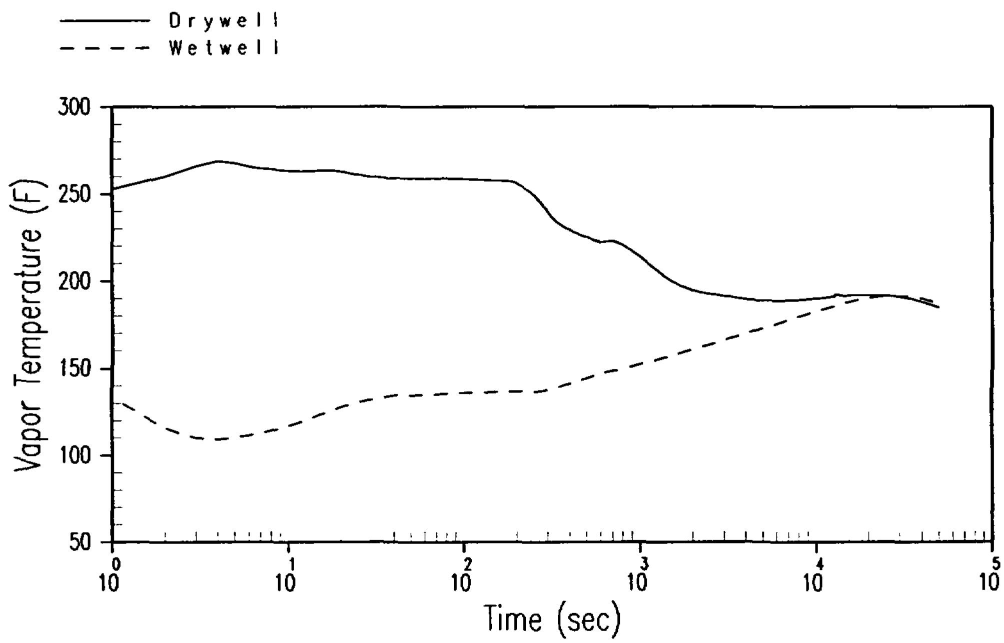

Figure A.4.3-2 RSLB Case 3 Vapor Temperature

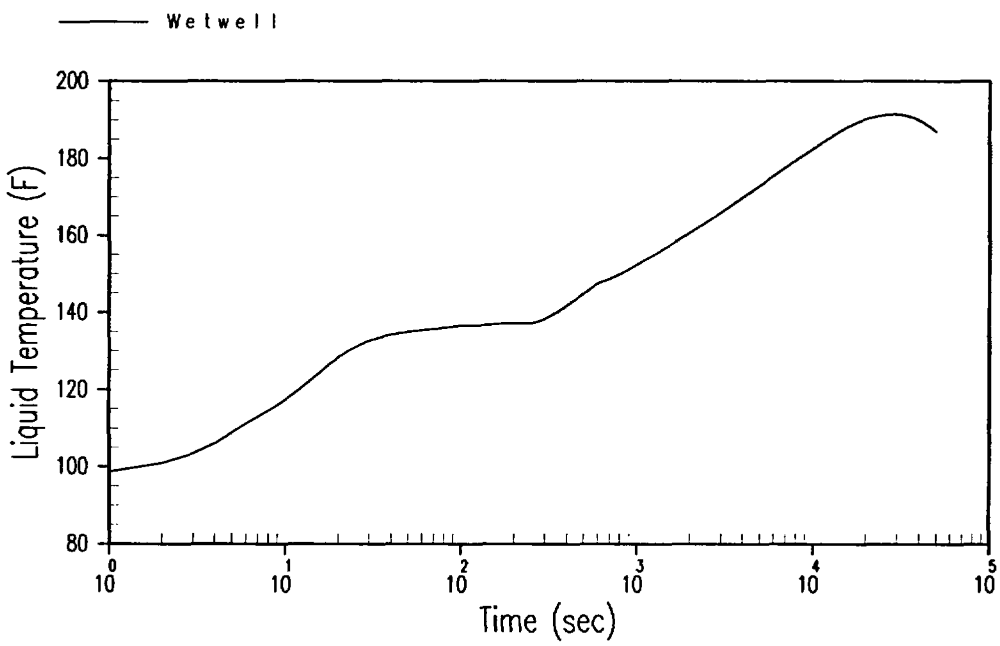

Figure A.4.3-3 RSLB Case 3 Suppression Pool Temperature

A.4.4 RSLB Long-Term Pressure and Temperature

This case calculates the peak suppression pool temperature and long-term containment pressure and temperature response input for equipment qualification; it covers both the blowdown and long-term containment response to the RSLB event. A loss of offsite power with a subsequent failure of 1 EDG (loss of 2 LPCI and 1 LPCS pumps) was assumed for this sample case. To maximize the long-term containment pressure and temperature, an operator action to initiate suppression pool cooling, instead of containment spray cooling, was assumed at 10 minutes. A modified GOBLIN ECCS evaluation model, as described in Appendix B, was used to produce the initial RSLB mass and energy release input for the GOTHIC containment response calculation. The GOTHIC vessel model was used to generate the long-term mass and energy release.

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]a,c

The GOTHIC generic BWR Mark I containment model was initialized at steady state conditions; the input changes that were made to run the RSLB long-term pressure and temperature transient are listed below.

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]^{a,c}

The results for this case are shown in Figures A.4.4-1 through A.4.4-3; the initial transient response is similar to the RSLB peak pressure case. The drywell pressure increases, the downcomer vent path to the suppression pool opens, and the large blowdown mass and energy release is over within approximately 30 seconds after the break occurs. After blowdown, the drywell pressure decreases and remains at approximately 40 psia while the ECCS pumps refill the reactor vessel (Figure A.4.4-1). After the reactor vessel has refilled, the cooler ECCS water begins to spill from the broken recirculation pump suction line causing the steam in the drywell to begin condensing. Unlike the previous cases, [ ]^{a,c} the drywell condensation rate and corresponding depressurization rate are lower in this case. The vacuum breakers open as the drywell pressure decreases below the wetwell pressure. This allows air from the wetwell to return to the drywell. The containment pressure decreases and remains between 30 and 35 psia for the remainder of the transient.

A minimum operator action time of 10 minutes is assumed in the generic model. The operator is assumed to trip one of the LPCI pumps and align the flow from the other LPCI pump through the RHR heat exchanger to initiate suppression pool cooling at 600 seconds. The suppression pool temperature continues to increase until about 30,000 seconds (Figure A.4.4-3), when the RHR heat removal rate exceeds the sum of the core decay and pump heat rates.

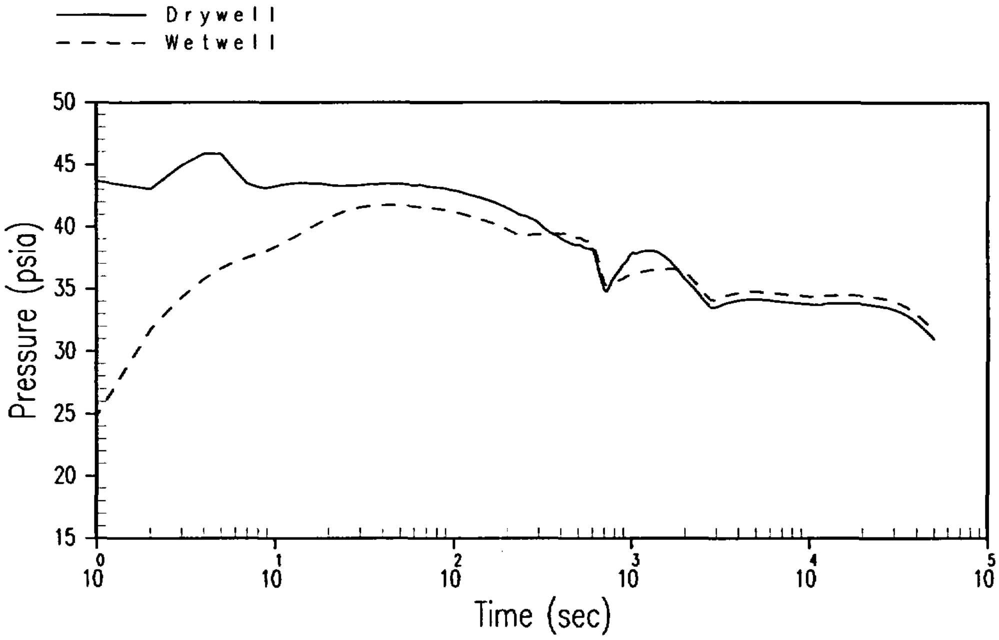

Figure A.4.4-1 RSLB Case 4 Pressure

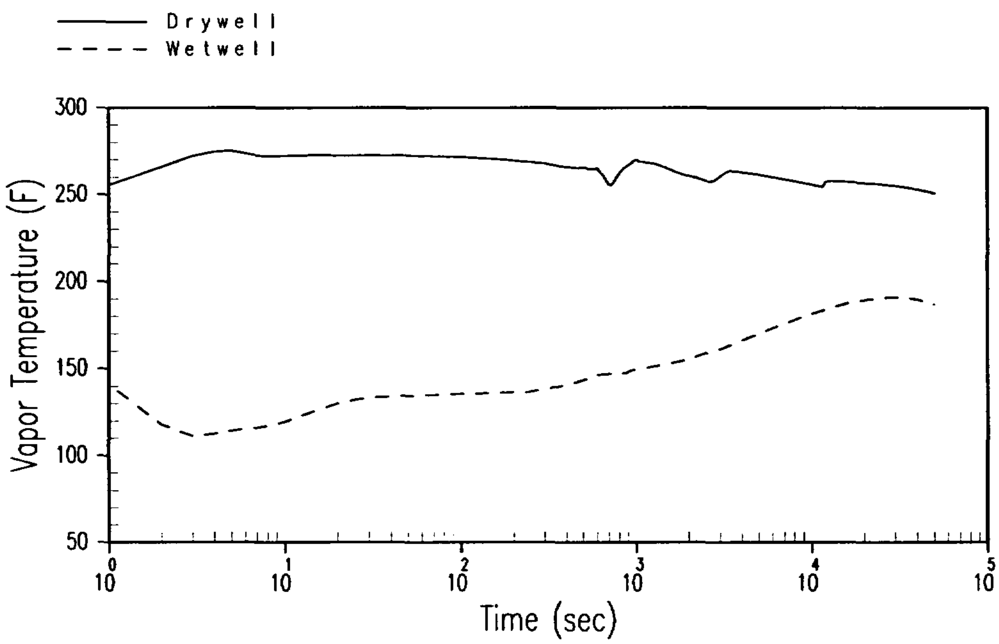

Figure A.4.4-2 RSLB Case 4 Vapor Temperature

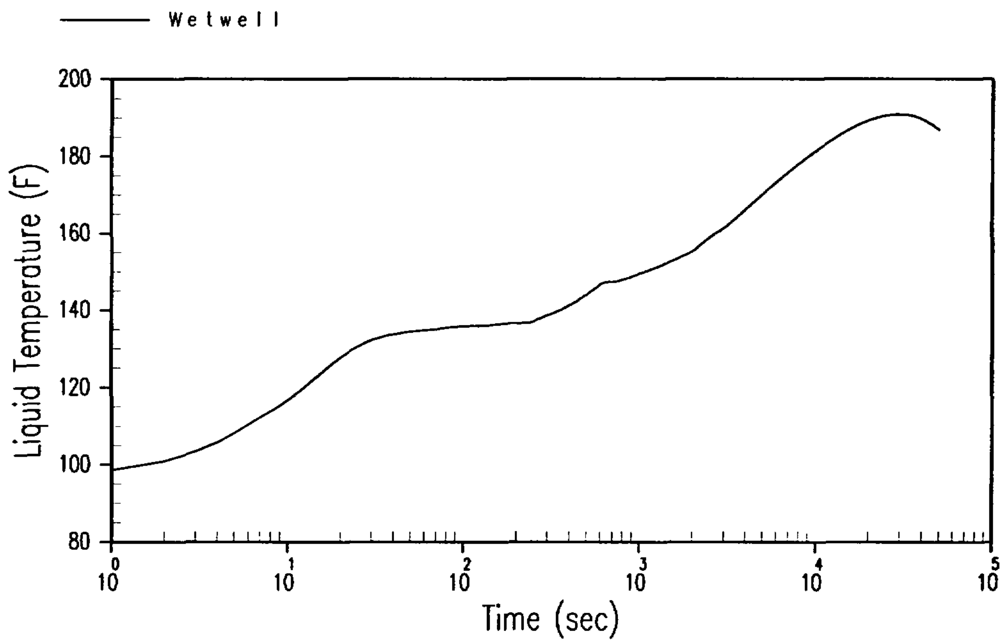

Figure A.4.4-3 RSLB Case 4 Suppression Pool Temperature

A.4.5 MSLB Peak Pressure

This case calculates the containment response for a MSLB event; it covers both the blowdown and reflood phases for the first 10 minutes of the event, before any operations would be assumed to occur. The break is assumed to be located between the flow restrictor and the MSIV. A loss of offsite power with a subsequent failure of 1 EDG (loss of 2 LPCI and 1 LPCS pumps) was assumed for this sample case. A modified GOBLIN ECCS evaluation model, as described in Appendix B, was used to produce the MSLB mass and energy release input for the GOTHIC containment response calculation. The GOTHIC vessel model was not used to generate mass and energy release data for this case

The GOTHIC generic BWR Mark I containment model was initialized at steady state conditions; the input changes that were made to run the MSLB peak pressure transient are listed below.
[

]^{a,c}

The results for this case are shown in Figures A.4.5-1 through A.4.5-3. The drywell pressure and temperature increase rapidly due to the large release of steam from the MSLB (Figures A.4.5-1 and 2). The increase in drywell pressure causes the water level in the downcomer pipes to decrease; a vent path to the suppression pool is created after the downcomer pipes empty. The mixture of steam, air, and hot water passes through the vent path to the suppression pool, causing the suppression pool temperature to increase (Figure A.4.5-3). The initial drywell pressure peak (44.0 psia) is determined by the vent path inertia and the subsequent pressure peak (42.9 psia) is determined by the break flow rate and vent path resistance. The wetwell pressure increases as the air from the drywell is transferred to the wetwell and the suppression pool water level increases, due to the break mass flow rate. The drywell pressure remains approximately 2 psi greater than the wetwell pressure and the drywell temperature remains around 270°F (Figure A.4.5-2) for the remainder of the 600 second transient.

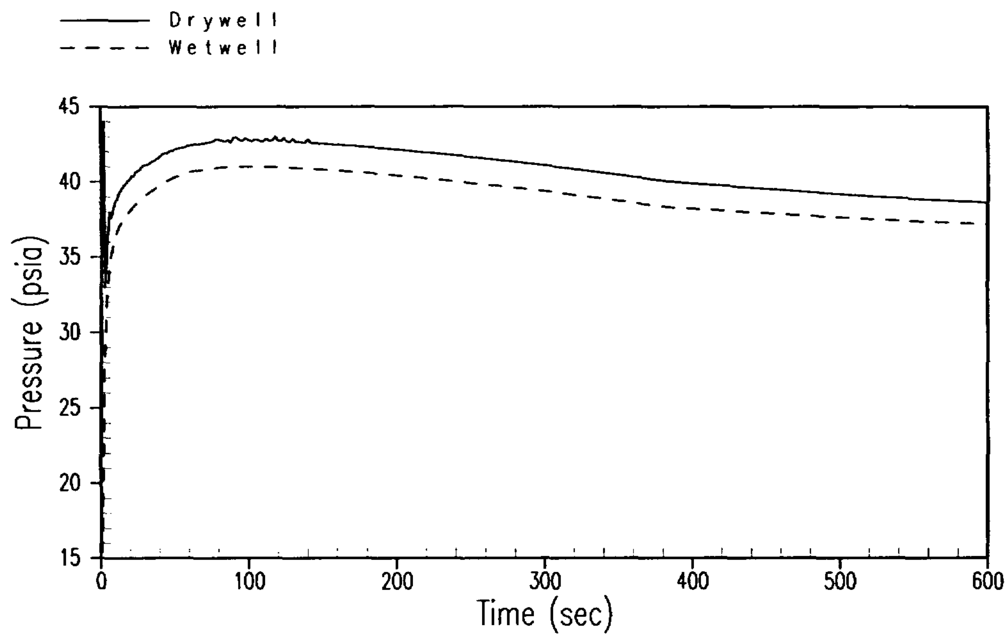

Figure A.4.5-1 MSLB Case 1 Pressure

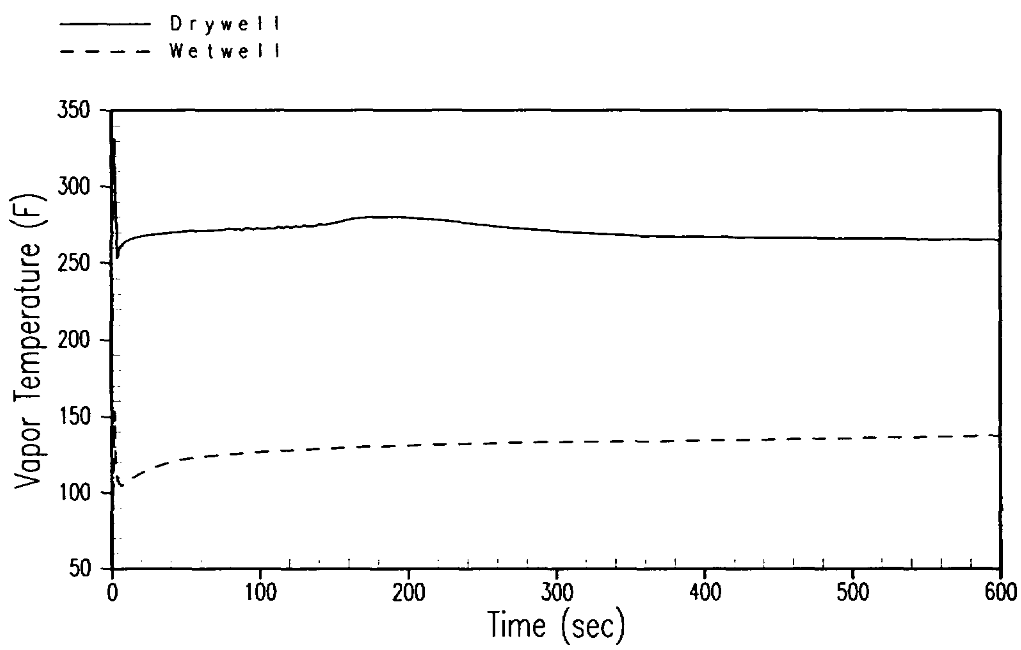

Figure A.4.5-2 MSLB Case 1 Vapor Temperature

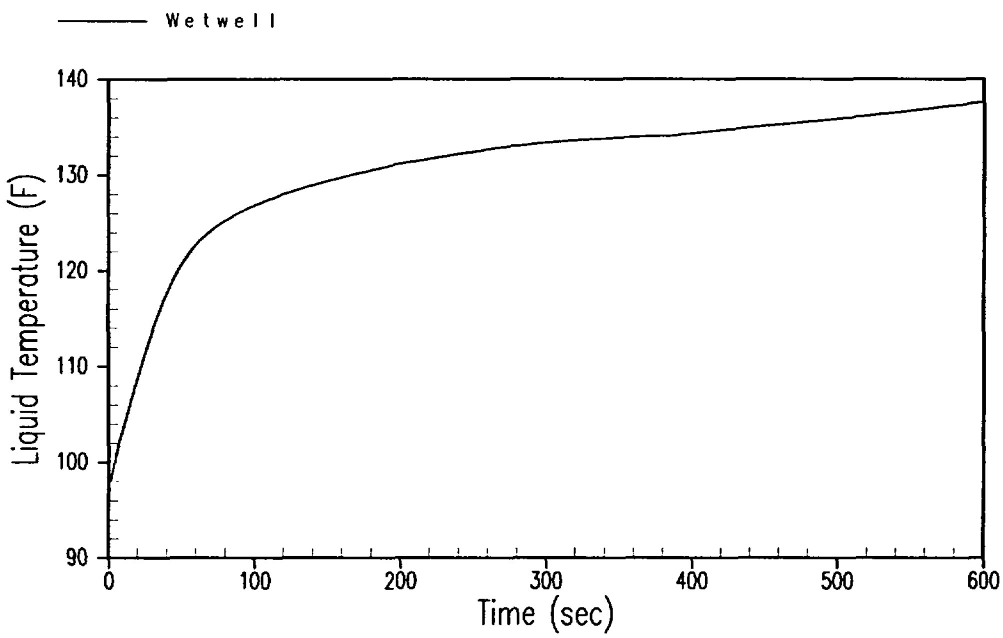

Figure A.4.5-3 MSLB Case 1 Suppression Pool Temperature

A.4.6 MSLB Long-Term Pressure and Temperature

This case calculates the peak suppression pool temperature and long-term containment pressure and temperature response input for equipment qualification; it covers both the blowdown and long-term containment response to the MSLB event. The break is assumed to be located between the flow restrictor and the MSIV. A loss of offsite power with a subsequent failure of 1 EDG (loss of 2 LPCI and 1 LPCS pumps) was assumed for this sample case. To maximize the long-term containment pressure and temperature, an operator action to initiate suppression pool cooling, instead of containment spray cooling, was assumed at 10 minutes. A modified GOBLIN ECCS evaluation model, as described in Appendix B, was used to produce the initial MSLB mass and energy release input for the GOTHIC containment response calculation. The GOTHIC vessel model was used to generate the long-term mass and energy release.

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The GOTHIC generic BWR Mark I containment model was initialized at steady state conditions; the input changes that were made to run the MSLB long-term pressure and temperature transient are listed below.

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]a,c

The results for this case are shown in Figures A.4.6-1 through A.4.6-3; the initial transient response is similar to the MSLB peak pressure case. The drywell pressure increases, the downcomer vent path to the suppression pool opens, and the mass and energy release is vented to the suppression pool. After the initial inertia peak, the drywell pressure remains between 35 and 40 psia while the ECCS pumps refill the reactor vessel.

GOTHIC calculates both the vessel mass and energy release and containment response after 10 minutes. The suppression pool temperature continues to increase as hot water pours from the vessel into the drywell and through the vent pipes back to the suppression pool. The suppression pool temperature continues to increase until about 30,000 seconds (Figure A.4.6-3), when the RHR heat removal rate exceeds the sum of the core decay and pump heat rates; the peak pool temperature was 190.1°F at 30,800 seconds.

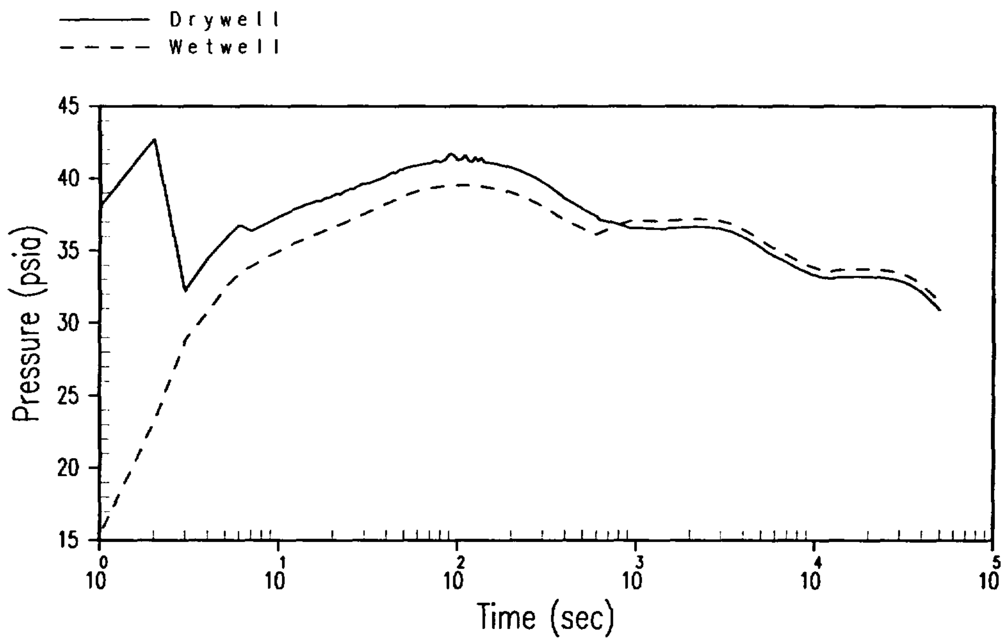

Figure A.4.6-1 MSLB Case 2 Pressure

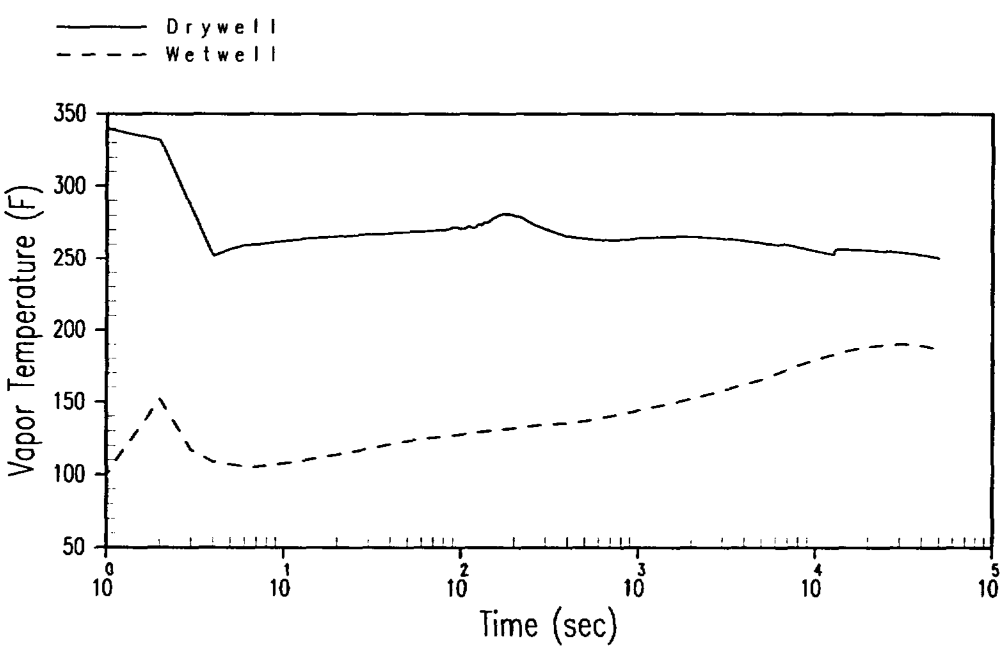

Figure A.4.6-2 MSLB Case 2 Vapor Temperature

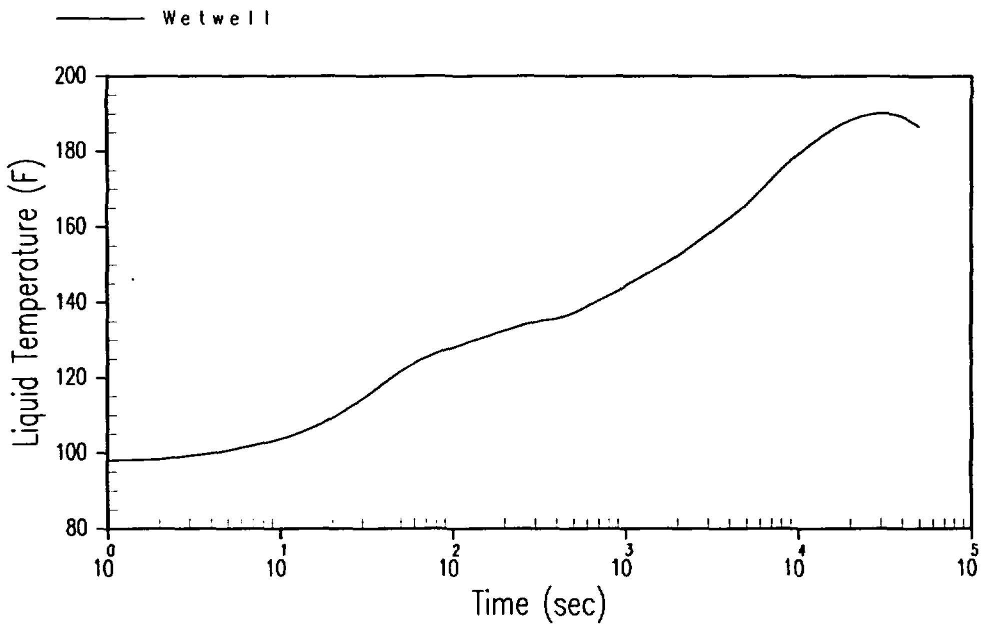

Figure A.4.6-3 MSLB Case 2 Suppression Pool Temperature

A.4.7 SBA

The small break accident (SBA) case represents a small steam line break in the drywell. The break is small enough that the feedwater control system is able to makeup for the loss of inventory while the reactor continues to operate at full power conditions; typically, the break size is 0.05 ft^2. This case usually produces the peak containment drywell temperature.

The reactor was assumed to remain at full power operation conditions after the break occurs. The HEM choked flow correlation was used to calculate the steam flow rate into the drywell through a flow path that is connected to a constant pressure boundary condition representing the full power RCS steam pressure and temperature. The operator was assumed to trip the reactor and initiate a controlled (100°F/hr) cooldown after the suppression pool temperature reached 120°F. The vessel model in the GOTHIC generic BWR Mark I containment model was used to calculate the mass and energy release for this event after the controlled cooldown was started.

The GOTHIC generic BWR Mark I containment model initialized at steady state conditions; the input changes that were made to run the SBA transient are listed below:
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]a,c

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]^{a,c}

The transient plots are shown in Figures A.4.7-1 through A.4.7-6. The containment pressure (Figure A.4.7-1), temperature (Figure A.4.7-2) and suppression pool temperature (Figure A.4.7-3) increase due to energy addition from the small steam line break. The suppression pool temperature reaches 120°F at about 1600 seconds. At this time, the operator is assumed to trip the reactor and initiate a 100°F/hr cooldown. The reactor vessel pressure (Figure A.4.7-4) decreases while the vessel level (Figure A.4.7-5) is controlled with makeup flow. The break flow rate (Figure A.4.7-6) continues to decrease as the reactor cooldown and depressurization progresses. The operator would initiate containment spray and/or suppression pool cooling to reduce containment pressure and the suppression pool temperature; however, these recovery actions were not modeled in this sample case.

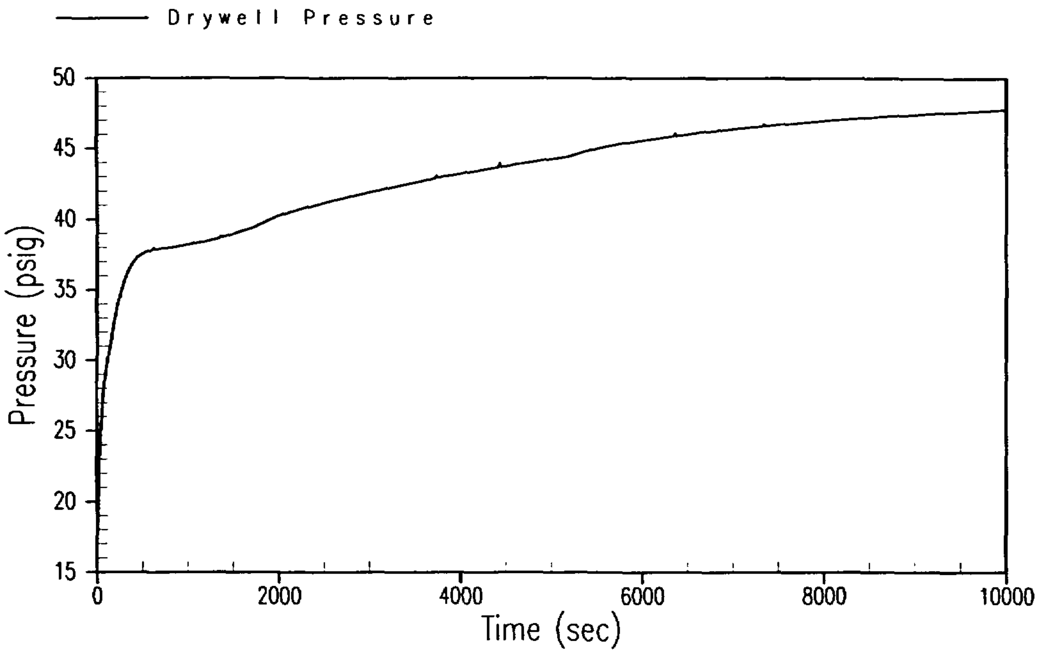

Figure A.4.7-1 SBA Case Drywell Pressure

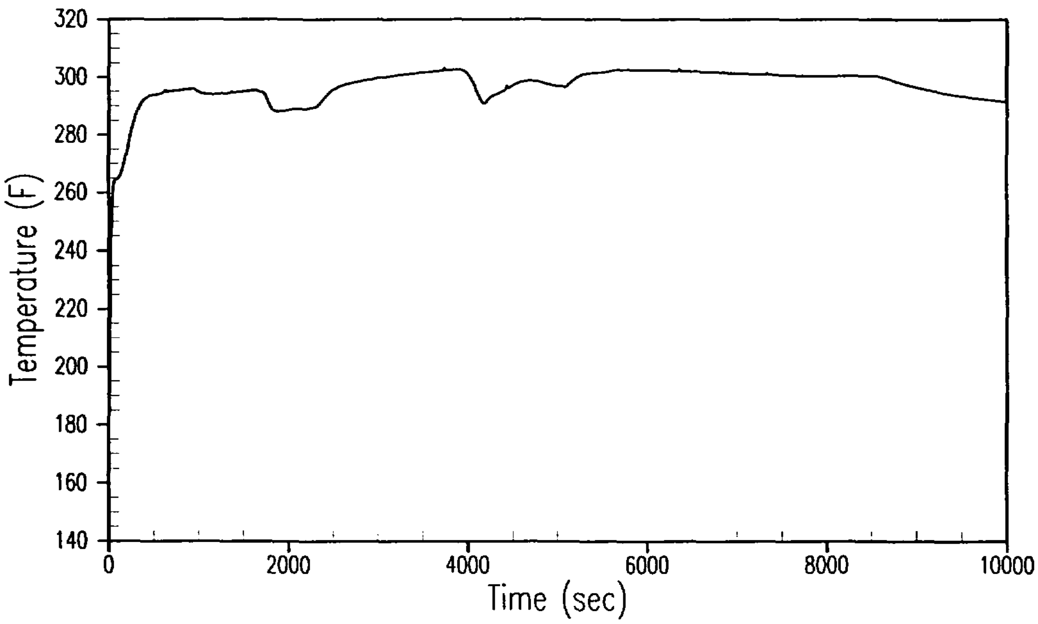

Figure A.4.7-2 SBA Case Drywell Temperature

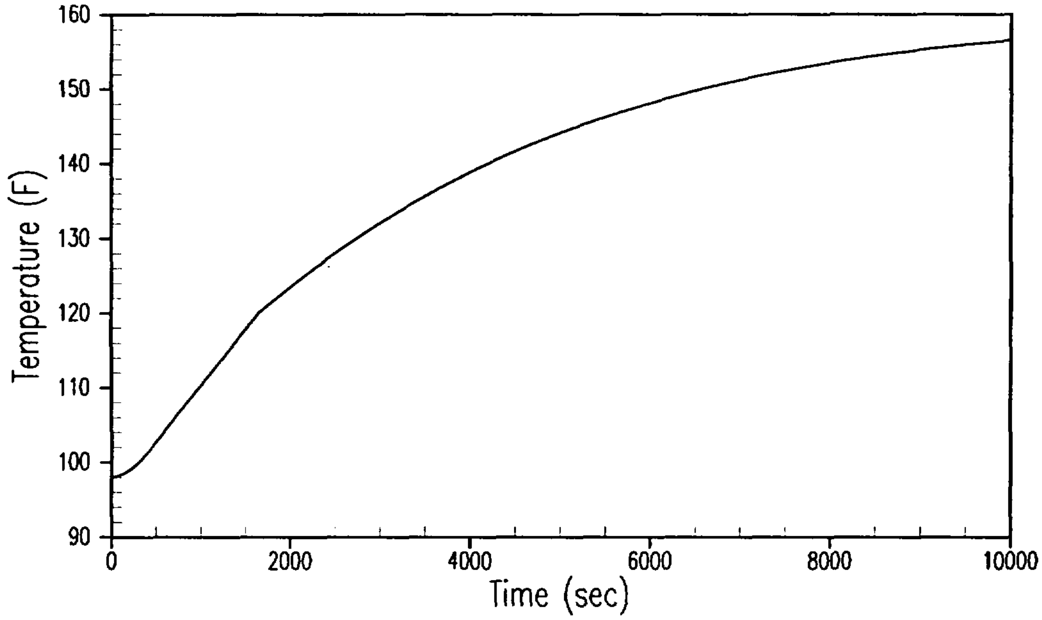

Figure A.4.7-3 SBA Case Suppression Pool Temperature

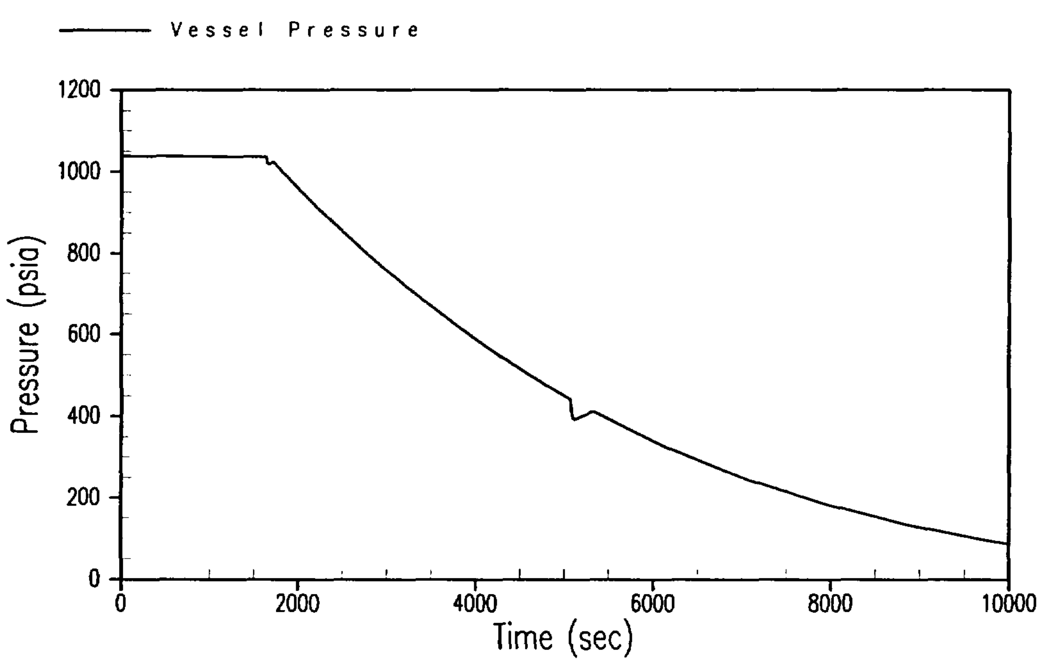

Figure A.4.7-4 SBA Case 7 Vessel Pressure

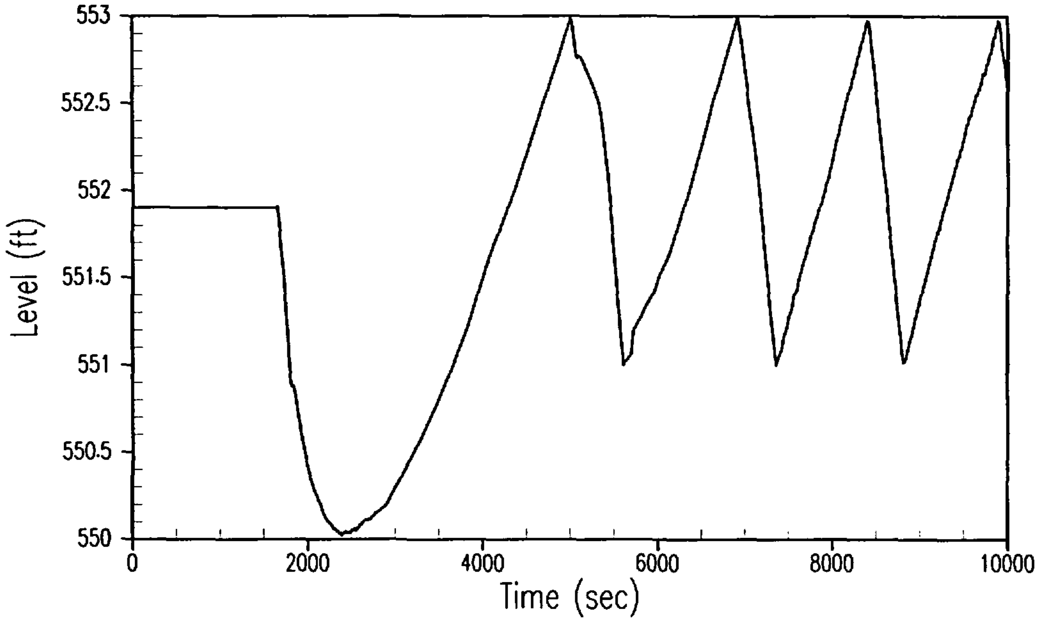

Figure A.4.7-5 SBA Case Vessel Level

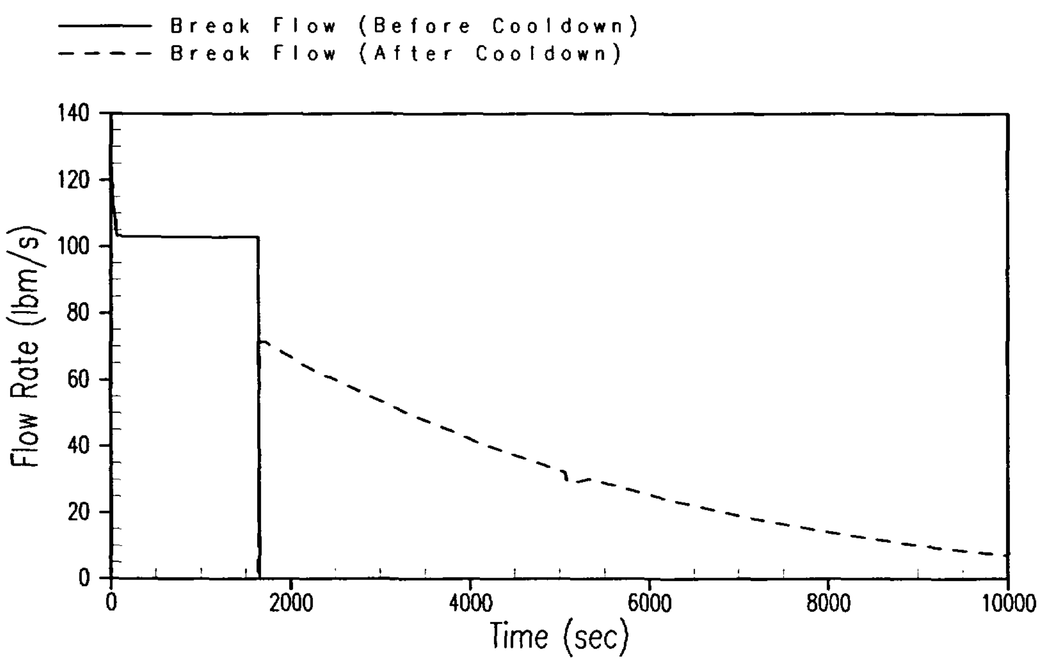

Figure A.4.7-6 SBA Case Break Flow Rate

A.4.8 ATWS

An ATWS event can result in a large, sustained steam release through the reactor safety and relief valves to the suppression pool. The ATWS event can be the limiting event for the suppression pool temperature. To recover from this event, the operator must take action to: shutdown the core, maintain reactor cooling by providing makeup water, and start suppression pool cooling and containment spray. A minimum operator action time of 10 minutes is assumed in the generic model. Since the containment conditions exceed the setpoint for spray initiation at 10 minutes, the operator is assumed to trip two of the LPCI pumps and align the flow from the other two LPCI pumps through the RHR heat exchanger to initiate containment spray to the drywell and wetwell at 600 seconds.

The mass and energy release input for this event was calculated with the BISON code (Reference A-12); the reactor transient response to a pressure regulator failure open to maximum demand (PRFO) was modeled. Operator action to initiate boron injection via the SLCS was modeled after the suppression pool temperature reached 110°F. The vessel water level was maintained using HPCI. The vessel water level was restored to normal once the hot shutdown boron weight was injected into the system. The GOTHIC vessel model could also have been used to calculate the mass and energy release through the safety and relief valves with an appropriate transient core heat rate input from BISON or another approved BWR transient analysis code.

The GOTHIC generic BWR Mark I containment model was initialized at steady state conditions; the input changes that were made to run the ATWS transient are listed below.
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The containment transient plots are shown in Figures A.4.8-1 through A.4.8-3. The drywell and wetwell pressure, drywell temperature, and suppression pool temperature increases rapidly due to the initial large release of steam from the safety and relief valves and subsequent condensation of steam within the suppression pool. The containment pressure and drywell temperature remain high until the containment spray is started at 600 seconds. After spray initiation, the containment pressure decreases to less than 25 psia, the drywell temperature decreases to about 150°F, and the wetwell temperature comes into thermal equilibrium with the suppression pool temperature. The suppression pool temperature continues to increase until the RHR heat removal rate exceeds the sum of the ECCS pump heat and core decay heat rates. The peak suppression pool temperature for this case is 189.1°F at 3347 seconds.

HPCI flow to the reactor vessel from the suppression pool was not modeled in the containment response analysis for this event. A sensitivity case was run to estimate the impact of taking water from the suppression pool and using it for HPCI. [

]^a,c This takes the BISON calculated break mass flow rate from the suppression pool water volume and thus conserves mass in the containment.

The suppression pool temperature comparison is shown in Figure A.4.8-4. Maintaining a constant containment mass by removing the BISON calculated mass flow rate from the suppression pool causes the suppression pool temperature to increase slightly. The peak suppression pool temperature for this case is 192.6°F at 3342 seconds.

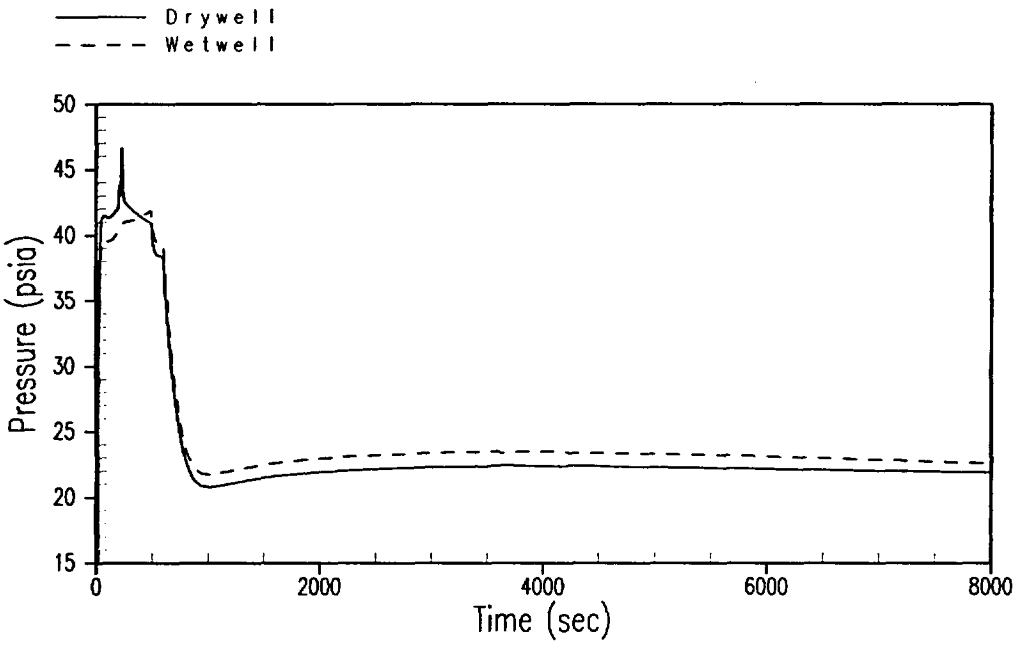

Figure A.4.8-1 ATWS Case Pressure

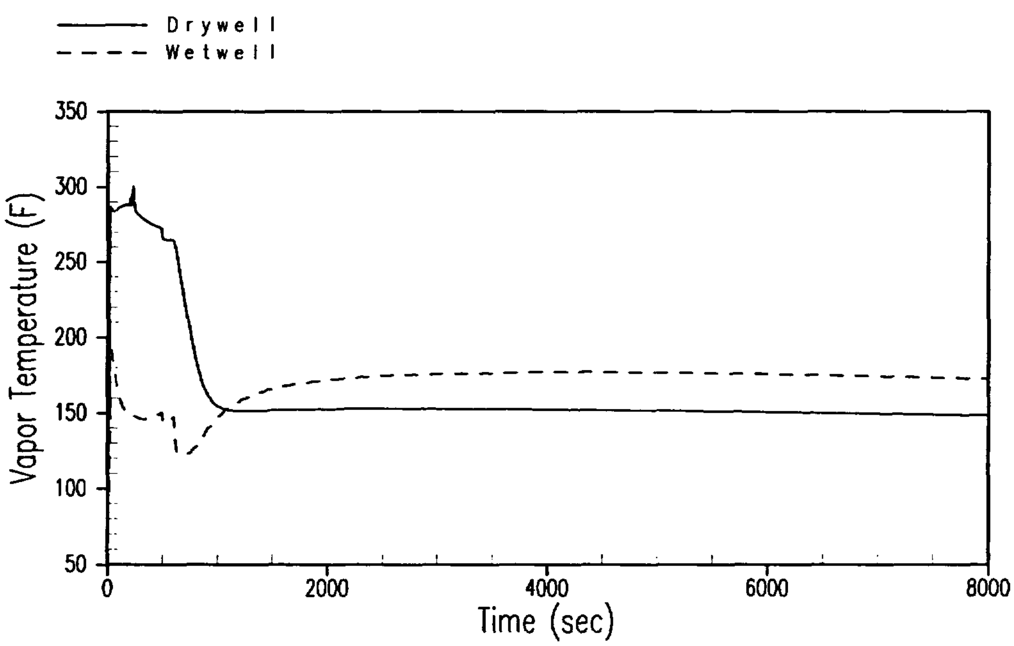

Figure A.4.8-2 ATWS Case Vapor Temperature

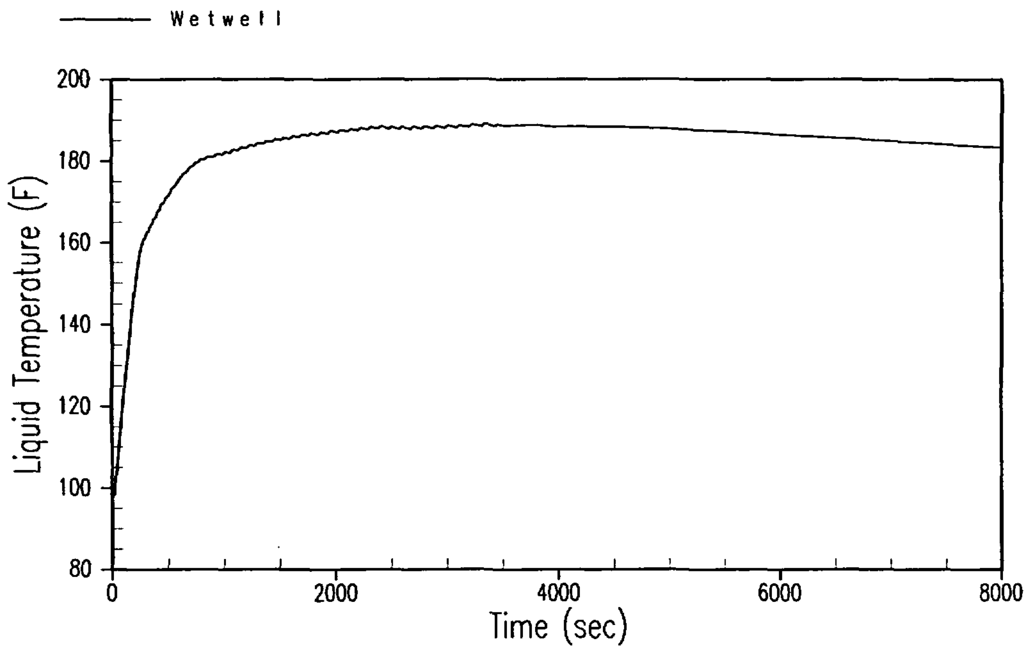

Figure A.4.8-3 ATWS Case 8 Suppression Pool Temperature

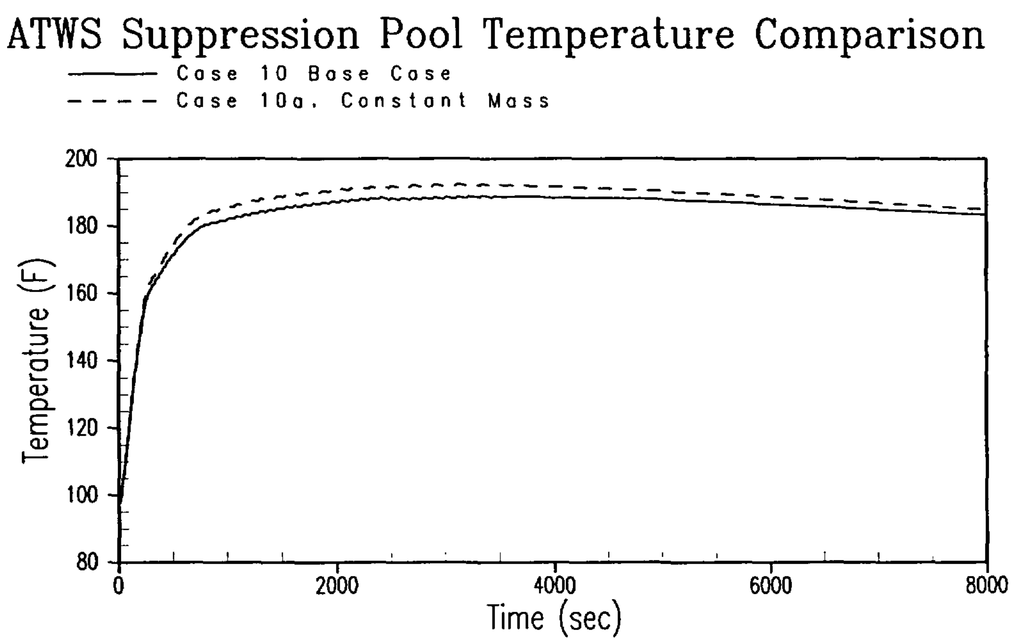

Figure A.4.8-4 ATWS Suppression Pool Temperature Sensitivity

A.5 CONCLUSIONS

The generic GOTHIC BWR Mark I containment model is described in Section A.2. The generic model was developed using information from the Mark I containment models that were built for the benchmark comparisons. The results of the benchmark comparisons are presented in Section A.3. The benchmark comparisons demonstrate that the GOTHIC BWR Mark I containment model maintains a similar degree of conservatism with respect to the present licensing basis models. Several sample transient cases were produced with the generic GOTHIC BWR Mark I containment model. The results of the sample cases are presented in Section A.4. The sample cases demonstrate that the generic model is capable of performing analyses for the various BWR containment applications with analysis margin to the containment design limits.

A.6 REFERENCES

A-1. NAI-1194-001, Revision 2, "GOTHIC BWR Mark I Containment Model," March 2006.

A-2. ANS-56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments," December 1983.

A-3. NUREG-0800, Section 6.2.1.1.C, Revision 6, "Pressure Suppression Type BWR Containments," August 1984.

A-4. NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model," April 1971.

A-5. NEDO-10320, Supplement 1, "The General Electric Pressure Suppression Containment Analytical Model," May 1971.

A-6. Kreith, "Principles of Heat Transfer," 3rd Edition.

A-7. NAI 8907-02, Revision 17, "GOTHIC Containment Analysis Package Users Manual," Version 7.2a, January 2006.

A-8. NRC Docket No. 50-237, LS05-82-08-038, "Subject: Systematic Evaluation Program (SEP) for Dresden Nuclear Power Station, Unit 2 – Evaluation Report on Topics VI-2.D and VI-3 (Docket No. 50-237)," August 19, 1982.

A-9. Dresden UFSAR.

A-10. Commonwealth Edison Company Calculation No. 3C2-0978-001, "Containment Temperature response to 0.05 Sq. Ft. Steam Line Break," December 1978.

A-11. Brown and York, "Sprays Formed by Flashing Liquid Jets," AICHE Journal Volume 8, #2, May 1962.

A-12. CENPD-292-P-A, Rev. 0 "BISON - A One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification," July 1996.

APPENDIX B
BWR MASS AND ENERGY RELEASE INPUT CALCULATION METHODOLOGY

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LIST OF TABLES

LIST OF FIGURES

B.1 INTRODUCTION

The mass and energy release input data is the primary driver for the calculation of the containment pressure and temperature. The mass and energy release input data for the containment response calculation can either be calculated by Westinghouse or provided by the customer. This section describes how Westinghouse calculates the mass and energy release input for the BWR containment model for the various events that are analyzed.

The approved BWR ECCS evaluation model (References B-1 through B-7) uses the GOBLIN code to calculate the RCS thermal-hydraulic response to a pipe rupture. The use of the code and model for these applications has been qualified by comparison with scalable test data covering the expected range of conditions and important phenomena. Therefore, when the input is properly biased, and the options are properly selected, the GOBLIN BWR ECCS evaluation model can be used to produce the mass and energy release input data for the containment response calculations.

B.2 INPUT BIASING FOR CONTAINMENT DBA ANALYSES

The BWR mass and energy release model input for the containment design basis accident analyses is biased to maximize the initial mass and energy stored in the RCS and to calculate a conservative release rate. NUREG-0800, Section 6.2.1.3 documents an acceptable practice for the calculation of the LOCA mass and energy release input data. The SRP specifies that the sources of energy available for release are to be based on 10 CFR Part 50, Appendix K, paragraph I.A. A comparison of the proposed Westinghouse methodology to the requirements given in NUREG-0800, Section 6.2.1.3 is shown in Table B-1. ANS 56.4-1983 also provides guidance for developing conservative input for the mass and energy release calculation in accordance with the acceptable practice documented in NUREG-0800, Section 6.2.1.3. A comparison of the proposed Westinghouse methodology to the recommendations in ANS 56.4-1983 is shown in Table B-2.

The following modifications were made to an existing GOBLIN BWR ECCS evaluation model to demonstrate the application of this methodology; conservative plant specific input values should be used in the actual design analyses. The modified GOBLIN BWR ECCS evaluation model was used to calculate conservative break mass and energy release input data for the containment DBA analysis sample cases presented in Appendix A.

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B.3 INPUT BIASING FOR MINIMUM ECCS BACKPRESSURE AND MINIMUM NPSHA ANALYSES

The following modifications should be made to the GOBLIN BWR ECCS evaluation model to calculate conservative break mass and energy release input data for the GOTHIC containment analyses for the minimum ECCS backpressure and minimum NPSHa analyses:
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B.4 GOBLIN MASS AND ENERGY RELEASE CASE DESCRIPTIONS

Six GOBLIN mass and energy release calculations were performed to generate the input for the GOTHIC containment response sample cases; four RSLB cases and two MSLB cases. In addition, the RSLB mass and energy release from the limiting PCT case was used for a minimum ECCS backpressure sensitivity study. The GOBLIN mass and energy cases are summarized in Table B-3 and the calculated mass and energy releases are shown in Figures B-2 through B-15.

The RSLB cases represent a double-ended rupture of one of the two recirculation suction legs. The break flow area input value (4.261 ft^2) represents the cross sectional flow area of [ ]^{a,c}. Cases 1 and 1b model only the blowdown portion of the event and were terminated prior to ECCS injection. Cases 2 and 3 model the blowdown, reflood, and post-reflood portions of the event out to 600 seconds. Full ECCS injection capability is assumed in Case 2 and a loss of offsite power with a subsequent failure of one EDG (loss of 2 LPCI and 1 LPCS pumps) is assumed in Case 3.

Case 2a is the RSLB mass and energy release from the limiting PCT case. This case assumes full ECCS injection capability; however a failure of the LPCI injection valve prevents LPCI flow from reaching the vessel.

The MSLB cases represent a double-ended rupture of a main steam line pipe between the integral flow restrictor and MSIV (inside the drywell). The steam flow rate (and corresponding break flow area input value of 2.808 ft^2) is limited by the integral flow restrictor. Both Case 4 and Case 5 were run out to 600 seconds; Case 4 assumes full ECCS injection capability, and Case 5 assumes a loss of offsite power with a subsequent failure of one EDG (loss of 2 LPCI and 1 LPCS pumps).

B.5 CONCLUSIONS

The GOBLIN BWR ECCS analysis model input is modified, as described in this Appendix, to produce the mass and energy releases for the RSLB and MSLB containment response calculations. The GOBLIN ECCS model input is biased and produces conservative mass and energy releases in accordance with the acceptance criteria documented in the regulations. The mass and energy release data for several sample RSLB and MSLB cases are provided. These data were used as input for several of the containment response cases that are documented in Appendix A.

Table B-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements

	Sources of Energy, 10CFR50, Appendix K, I.A	Westinghouse Methodology
1	Reactor Power – The reactor should be assumed to have been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.	
2	Core Stored Energy – The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.)	
3	Fission Heat – Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.	
4	Decay of Actinides – The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.	
5	Fission Product Decay – The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.	

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Table B-1 (cont.) NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements		
Sources of Energy, 10CFR50, Appendix K, I.A		Westinghouse Methodology
6	Metal-Water Reaction Rate – The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation. The reaction shall be assumed not to be steam limited.	
7	Reactor Internals Heat Transfer – Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.	
8	Fuel Rod Swelling and Rupture – The calculation of fuel rod swelling and rupture should not be considered for M&E calculations	
9	Break Size and Location – Containment design basis calculations should be performed for a spectrum of possible pipe breaks, sizes, and locations to assure that the worst case has been identified.	
10	Calculations, Sub-compartment Analysis – The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.	
11	Calculations, Initial Blowdown Phase – The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 102% times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.	

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Table B-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

	Sources of Energy, 10CFR50, Appendix K, I.A	Westinghouse Methodology
12	Calculations, Initial Blowdown Phase – Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.	
13	Calculations, Initial Blowdown Phase – Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.	

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Table B-2 ANS 56.4-1983 Recommendations

	Recommendation	Westinghouse Methodology
1	Reactor Coolant System Water and Metal – The increase in the reactor coolant system volume resulting from the pressure and temperature expansion to conditions at the initial power level defined in 3.2.2.2 shall be included. Stored energy in all reactor coolant system pressure boundary and internals metal thermally in contact with the reactor coolant system water shall be included.	
2	Core Stored Energy – The core stored energy and the steady-state core-temperature distribution, adjusted for uncertainties, shall be consistent with the initial conditions and consistent with the time of fuel cycle life required in 3.2.2.1.	
3	Fission Heat – Fission heat shall be conservatively calculated. Shutdown reactivities resulting from temperature and voids shall assume minimum plausible values including allowances for uncertainties; all data shall be based on their minimum values consistent with the fuel parameters which yield the maximum core stored energy. Rod trip and insertion may be assumed at the time appropriate for the transient being analyzed.	
4	Decay of Actinides – The heat from the radioactive decay of actinides, including neptunium and plutonium as well as isotopes of uranium generated during operation, shall be calculated in accordance with fuel cycle calculations and shall be appropriate for the time in the fuel cycle that yields the highest calculated core stored energy. The decay heat shall be the values given in American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979 for end-of-life operation time.	
5	Fission Product Decay – The heat generation rates from radioactive decay of fission products shall be assumed to be equal to at least the values given in ANSI/ANS-5.1-1979 for end-of-life operation time.	
6	Metal-Water Reaction Rate – The amount of metal-water reaction shall be calculated according to 10 CFR 50.44 and assumed to occur uniformly over a period less than 2 minutes following the end of reactor vessel blowdown.	
7	Main Steam Lines – Steam flow to the turbine until the main steam isolation valves or turbine stop valves are calculated to close may be included. Flow to the turbine shall be minimized. Delays and valve closure times shall be conservatively short. In lieu of this calculation, flow to the turbine may be conservatively terminated at break initiation.	

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Table B-2 (cont.) ANS 56.4-1983 Recommendations

	Recommendation	Westinghouse Methodology
8	Main Feedwater Line – Main feedwater flow shall be included and shall be maximized. Delays and valve closure times used to determine the termination of flow shall be conservatively long.	
9	ECCS Flow – Flow from the ECCS shall be included. Flows and delay times shall be chosen in accordance with the single active failure consideration which results in the highest peak primary containment pressure.	
10	Time of Life – The time of life of the core shall be that producing the maximum energy from the combination of core stored energy and decay heat assuming power level as required in 3.2.2.2.	
11	Power Level – The initial power level shall be at least as high as the licensed power level plus uncertainties such as instrumentation error (typically 102 percent of the licensed power level).	
12	Core Inlet Temperature – The initial core inlet temperature shall be the normal operating temperature consistent with the initial power level adjusted upward for uncertainties such as instrumentation error. The uncertainties shall be biased to result in maximizing energy releases through the break for the entire transient.	
13	Reactor Coolant System Pressure – The initial reactor coolant system pressure shall be at least as high as the normal operating pressure consistent with the initial power level plus uncertainties such as instrumentation error.	
14	Core Parameters – Initial core parameters (including physics parameters, fuel properties, and gas conductivity) shall be chosen to maximize core stored energy.	

a,c

Table B-2 (cont.) ANS 56.4-1983 Recommendations		
Recommendation		Westinghouse Methodology a,c
15	Single Active Failures – In determining the mass and energy releases following a reactor coolant system break, the most restrictive single active failure shall be considered. The possibility that the highest peak primary containment pressure may occur for the situation where no active failure has occurred shall not be overlooked. No more than one single active failure in the safety systems, (including primary containment heat removal system; see 4.2.5) required to mitigate the consequences of the event, need to be considered.	
16	Single Passive Failures – Passive failures normally need not be considered.	
17.	Non-emergency Power – The loss of non-emergency power shall be postulated if it results in circumstances (for example, delayed primary containment cooling or safety injection) which lead to higher primary containment pressures.	
18	Nodalization – Geometric nodalization for the various periods of the reactor coolant system break analysis need not be the same. Since low quality at the break node is conservative during blowdown because it leads to high flow rates, the reactor coolant system shall be modeled with sufficient detail so that the quality at the break location shall not be over predicted.	
19	Thermodynamic Conditions – The thermodynamic state conditions for steam and water shall be described using real gas equations or industry accepted steam table in such a manner that the resultant steam and water temperature and partial steam pressure are within one percent of that which would result from use of the 1967 ASME Steam Tables with appropriate interpolation.	
20	Pump Characteristics – The characteristics of the reactor coolant system pumps shall be derived from a dynamic model that includes momentum transfer between the fluid and the impeller with variable pump speed as a function of time. The pump model for the subcooled and two-phase region shall be verified by applicable subcooled and two-phase performance data. In lieu of a full dynamic pump model, any model which can be shown to be conservative by comparison with the test data or by comparison with a full dynamic pump model may be used.	

Table B-2 ANS 56.4-1983 Recommendations (cont.)

	Recommendation	Westinghouse Methodology
21	Break Sizes – For reactor coolant system analysis, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the reactor coolant system. The beak shall be defined by its location, type, and area.	
22	Break Flow Model – Empirical critical break flow models developed from test data may be utilized during the periods of applicability, for example, subcooled, saturated, or two-phase critical flow. Acceptable critical break flow models, when the fluid conditions are subcooled immediately upstream of the break, include the Zaloudek and Henry-Fauske models. During the period when fluid conditions immediately upstream of the break are saturated or two-phase, an acceptable model is the Moody critical flow model. The critical break flow correlations may be modified to allow for a smooth transition between subcooled and saturated flow regions. Other critical flow models may be used if justified by analysis or experimental data. The discharge coefficient applied to the critical flow correlation shall be selected to adequately bound experimental data.	
23	ECCS Spillage – In generating mass and energy release source terms from spillage for primary containment peak pressure determination, the quality shall be selected based on the partial pressure of steam in containment to maximize primary containment pressurization. For the determination of the maximum primary containment sump temperature for calculation of available NPSH, assumptions on generating mass and energy release and spillage source terms shall be biased toward maximizing the sump temperature.	
24	Drywell Backpressure – Throughout the post-LOCA transient, the mass and energy release calculation shall be coupled to the drywell pressure calculation, or a conservatively low back-pressure function shall be used.	
25	Heat Transfer Correlations – Heat transfer correlations shall be based on experimental data or chosen to predict conservatively high primary containment pressure.	

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Table B-2 (cont.)	ANS 56.4-1983 Recommendations	
	Recommendation	**Westinghouse Methodology** a,c
26	Core Modeling – Fission heat may be calculated using a core averaged point kinetics model which considers delayed neutrons and reactivity feedback. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowances for uncertainties for the range of power distribution shapes and peaking factors which result in the maximum core stored energy. Rod trip and insertion may be assumed if they are calculated to occur. Reactivity effects shall be consistent with the time of life which leads to the maximum core stored energy. For core thermal hydraulic calculations, the core shall be modeled with sufficient detail so as not to under-predict core-to-reactor coolant heat transfer. Initial core stored energy shall be maximized.	
27	Modeling of Metal Walls – Heat transfer from metal walls to coolant shall be calculated so as not to under-predict the rate of heat transfer relative to experimental data or the solution of the one-dimensional, time dependent heat conduction equation.	

Table B-3 RSLB and MSLB Case Descriptions

Case	Break	Transient end Time(s)	Decay Heat Standard	ECCS Capacity
1	RSLB	30	ANS71 + 20%	N/A
1b	RSLB	30	ANS79 + 2σ	N/A
2	RSLB	600	ANS71 + 20% (0-30 s) ANS79 + 2σ (> 30 s)	2 LPCS + 4 LPCI
2a	RSLB	235	ANS71 + 20%	2 LPCS + 4 LPCI (The flow from the 4 LPCI pumps does not get injected)
3	RSLB	600	ANS71 + 20% (0-30 s) ANS79 + 2σ (> 30 s)	1 LPCS + 2 LPCI
4	MSLB	600	ANS79 + 2σ	2 LPCS + 4 LPCI
5	MSLB	600	ANS79 + 2σ	1 LPCS + 2 LPCI

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Figure B-1 GOBLIN BWR Mass and Energy Release Model Noding Diagram

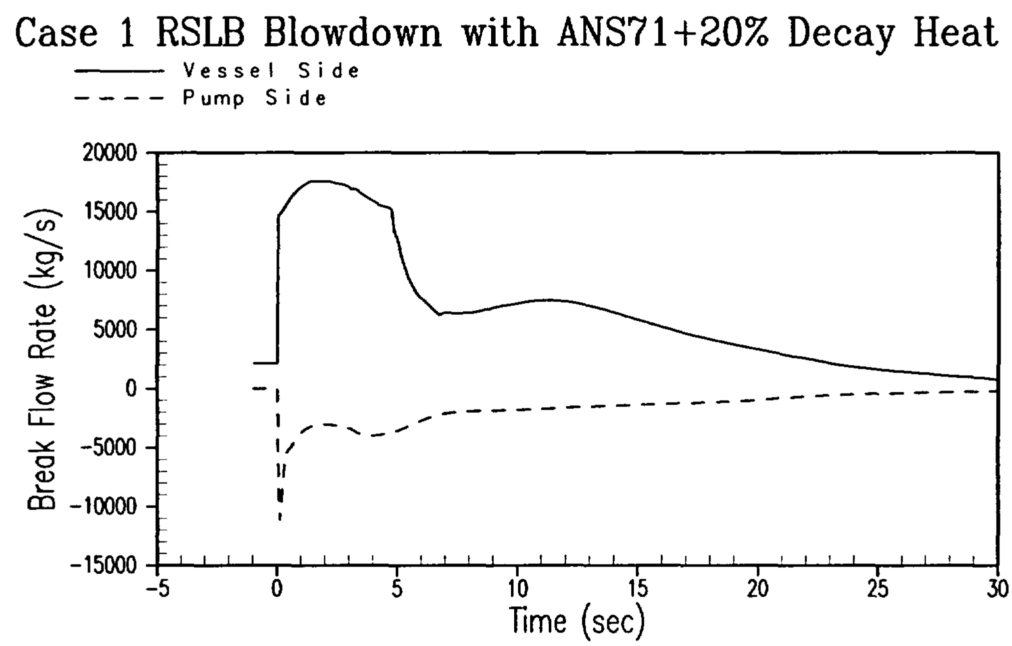

Figure B-2 Case 1 Break Flow Rate

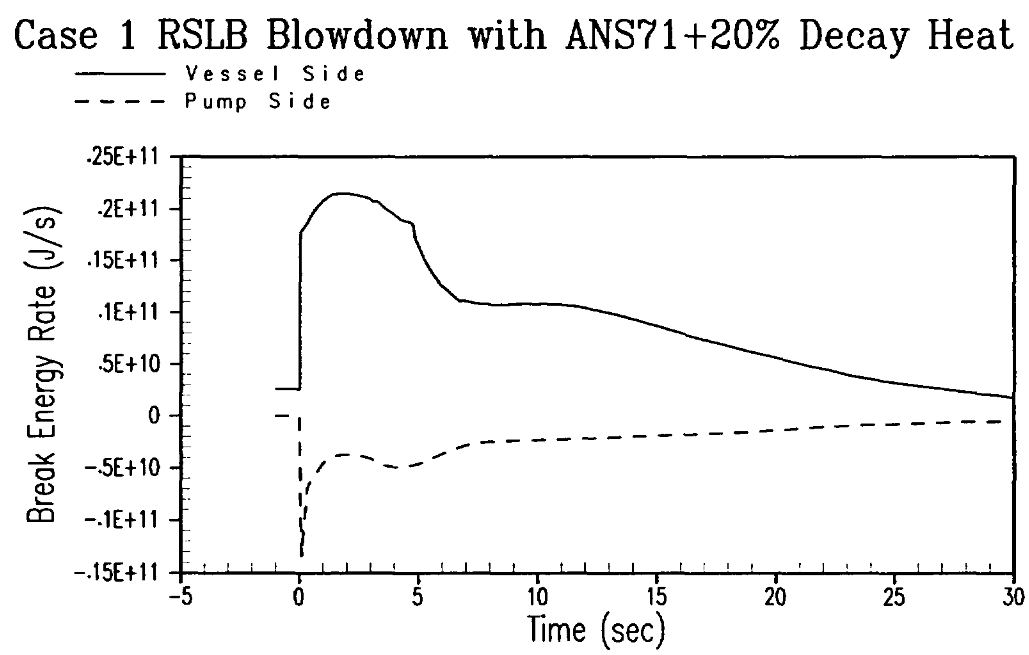

Figure B-3 Case 1 Break Energy Release Rate

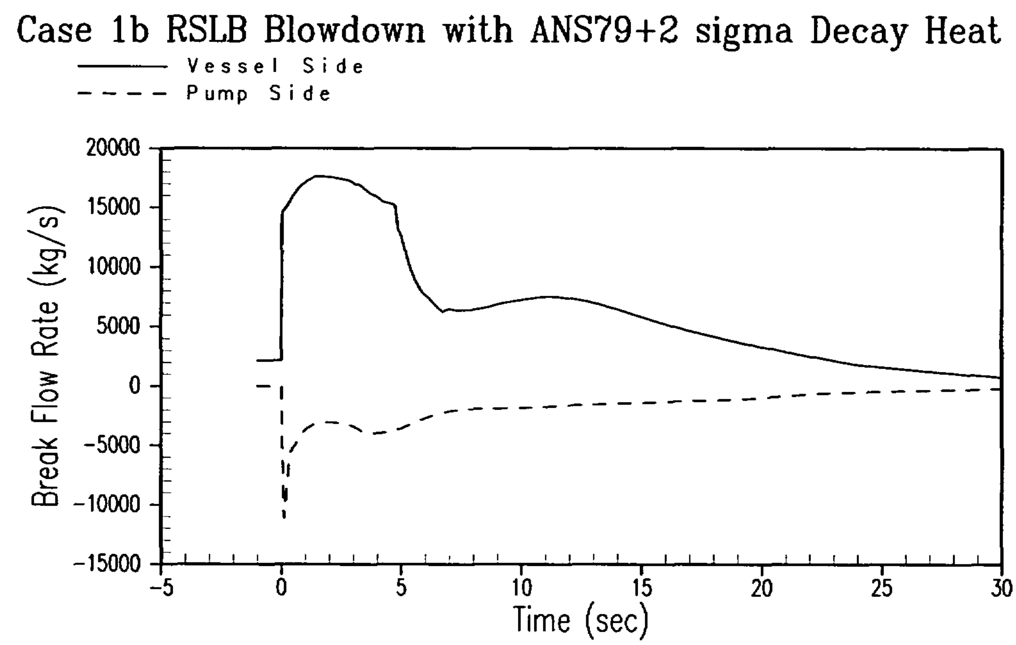

Figure B-4 Case 1b Break Flow Rate

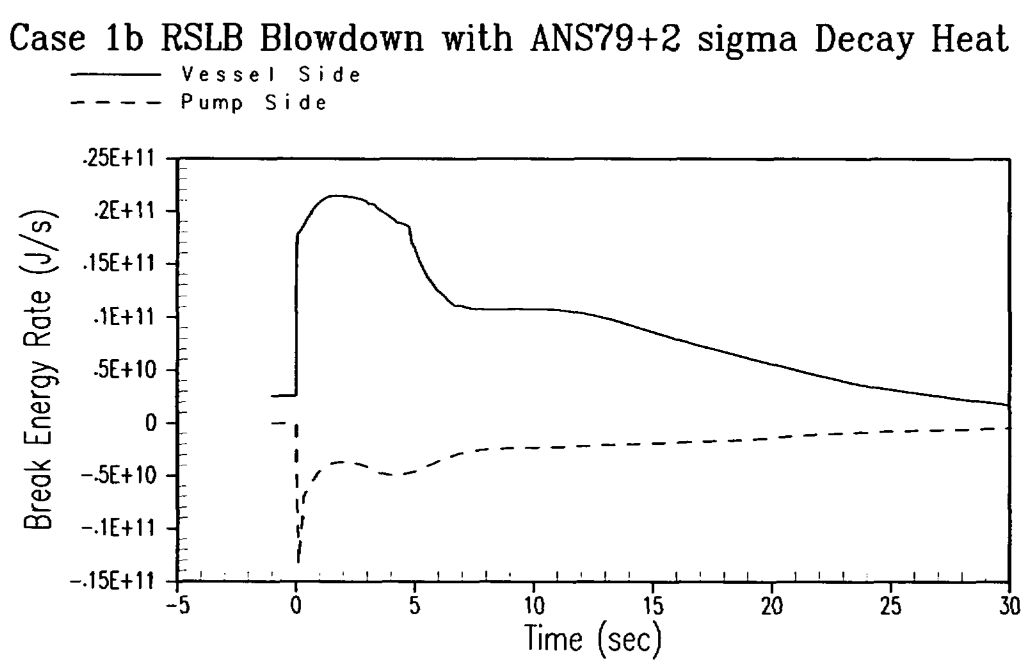

Figure B-5 Case 1b Break Energy Release Rate

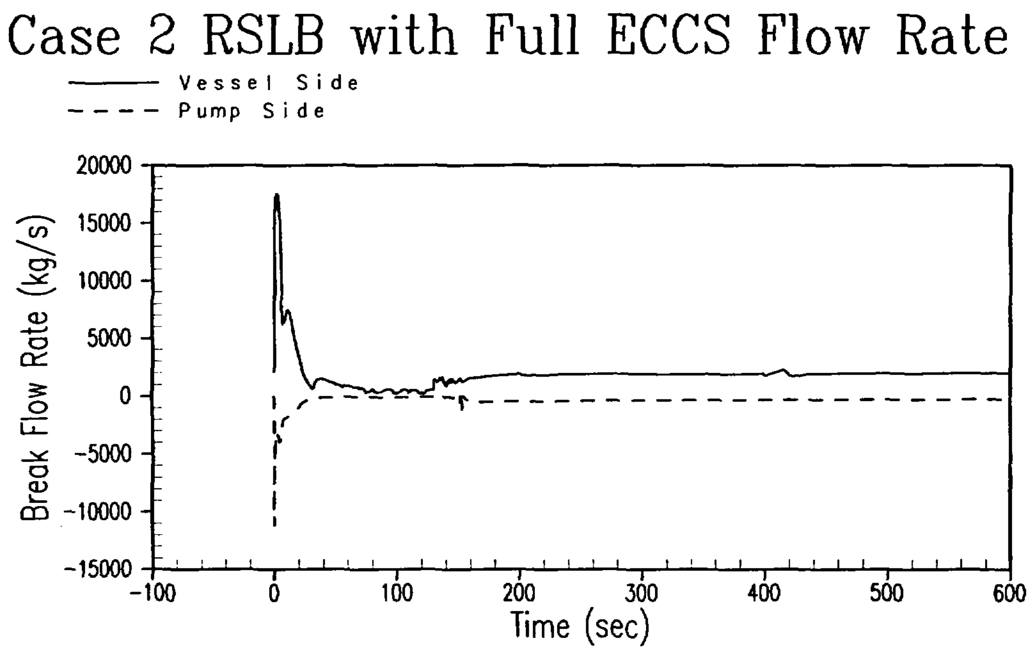

Figure B-6 Case 2 Break Flow Rate

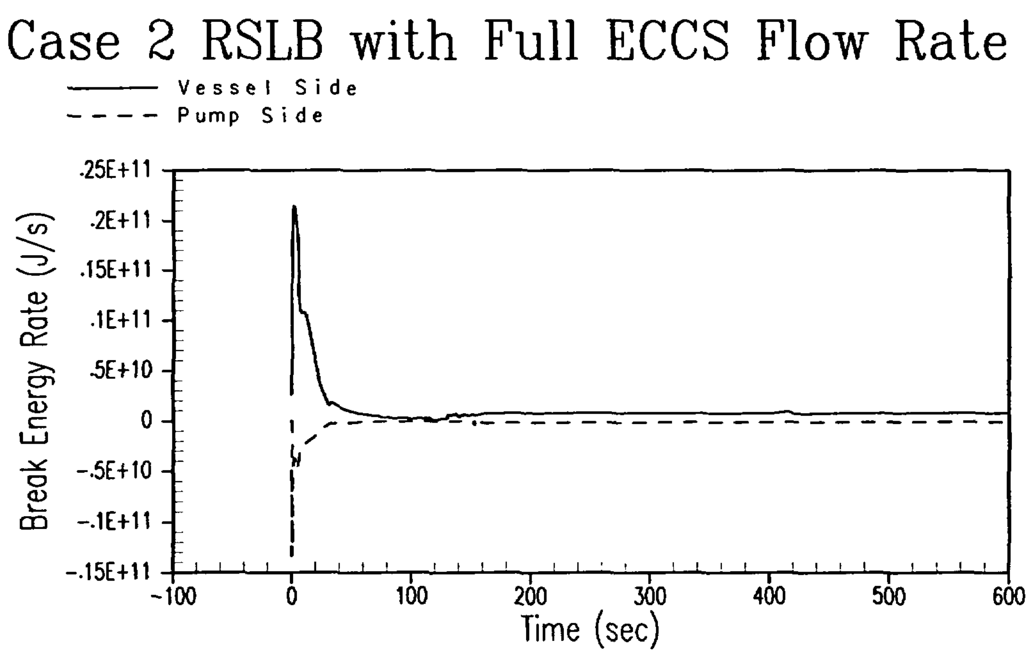

Figure B-7 Case 2 Break Energy Release Rate

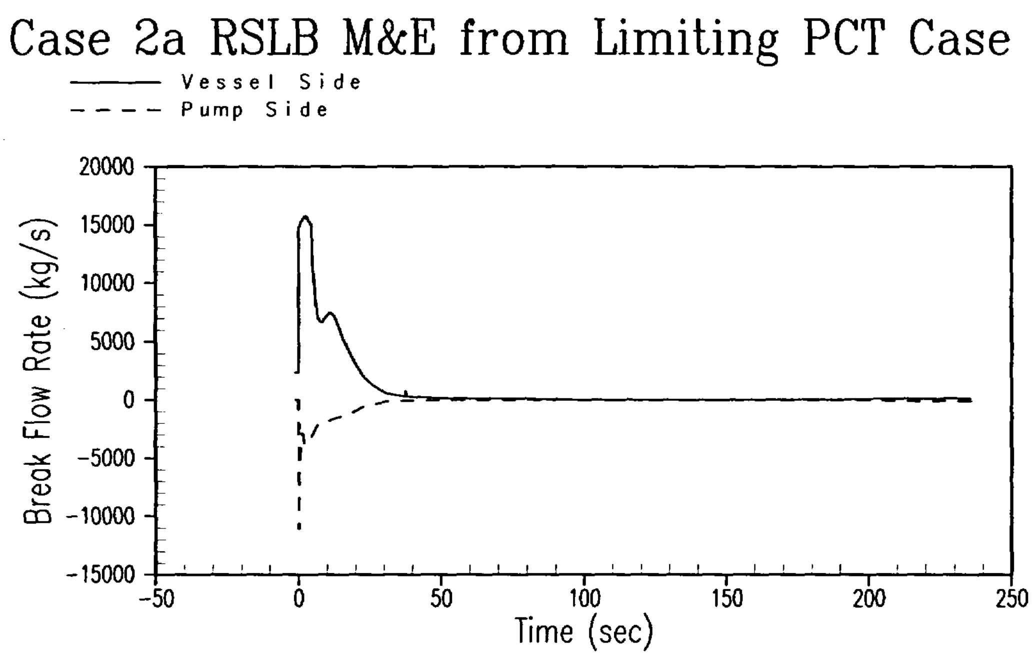

Figure B-8 Case 2a Break Flow Rate

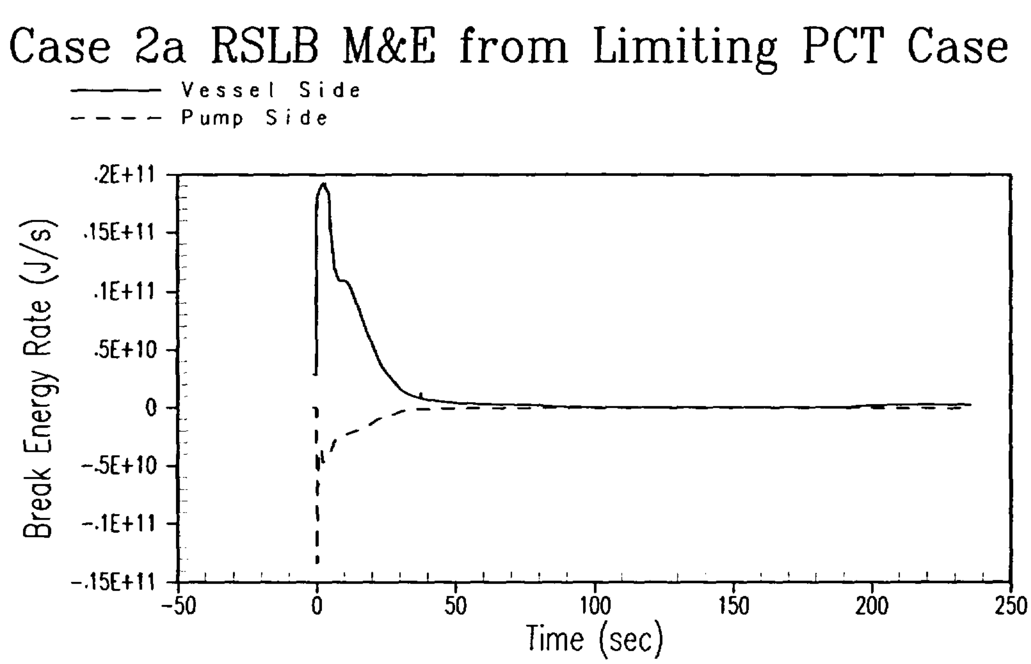

Figure B-9 Case 2a Break Energy Release Rate

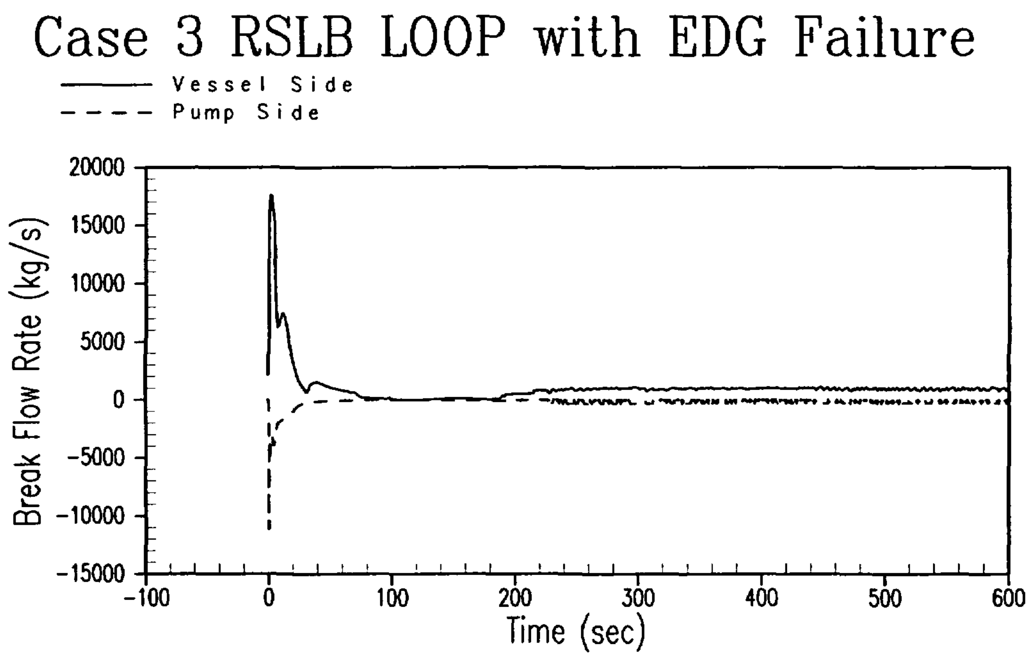

Figure B-10 Case 3 Break Flow Rate

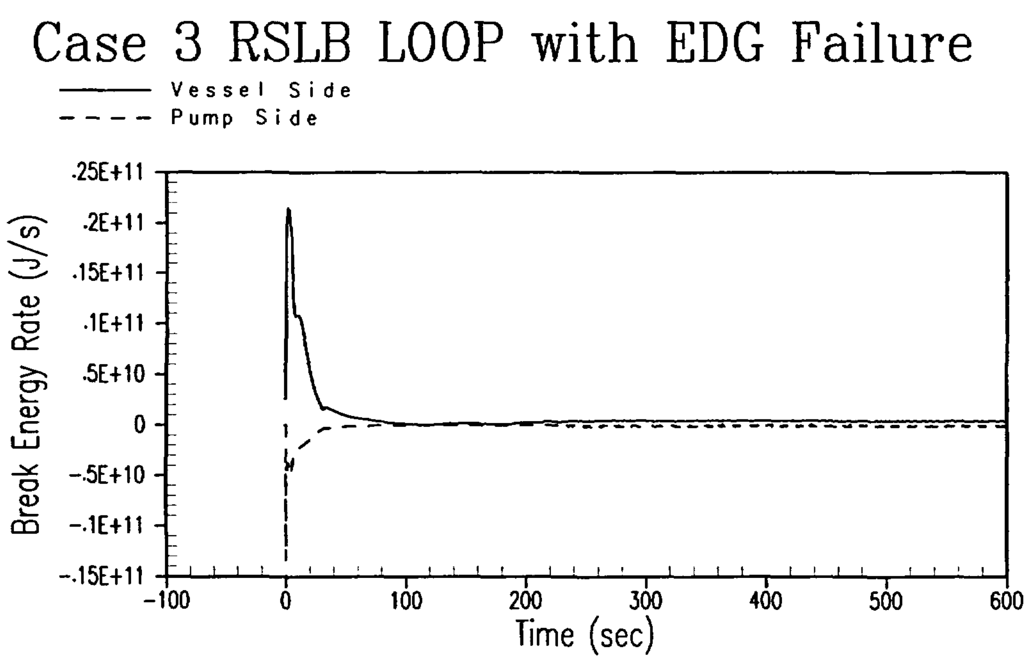

Figure B-11 Case 3 Break Energy Release Rate

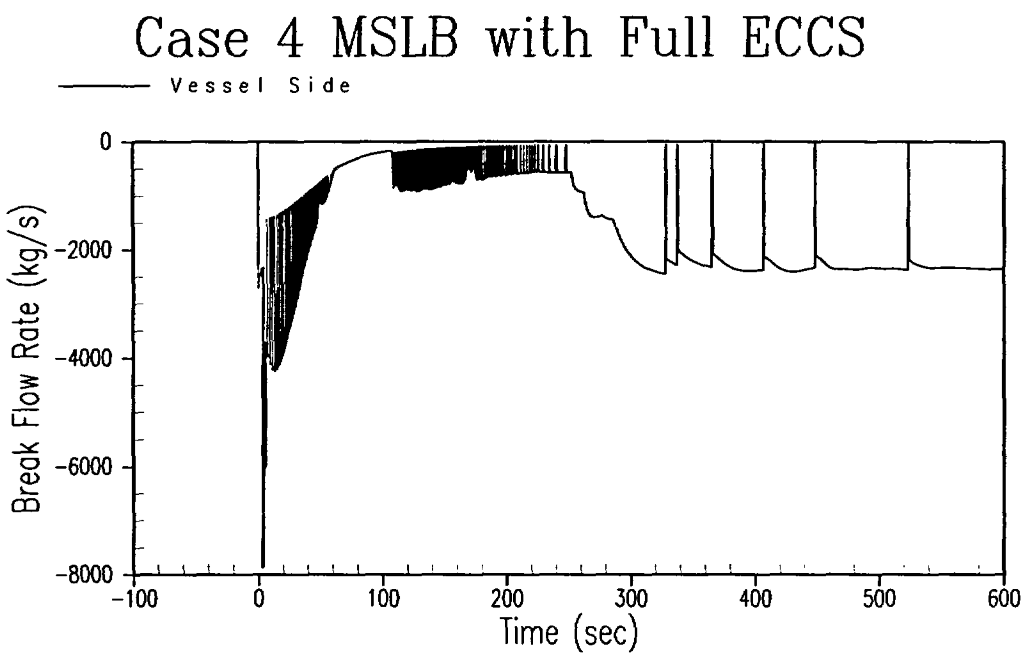

Figure B-12 Case 4 Break Flow Rate

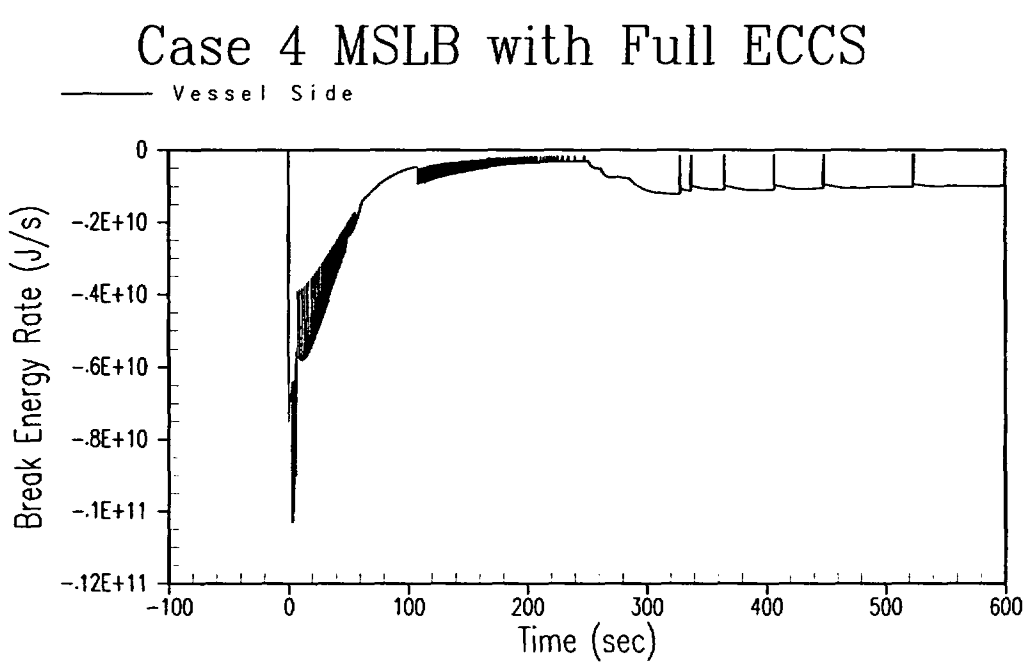

Figure B-13 Case 4 Break Energy Release Rate

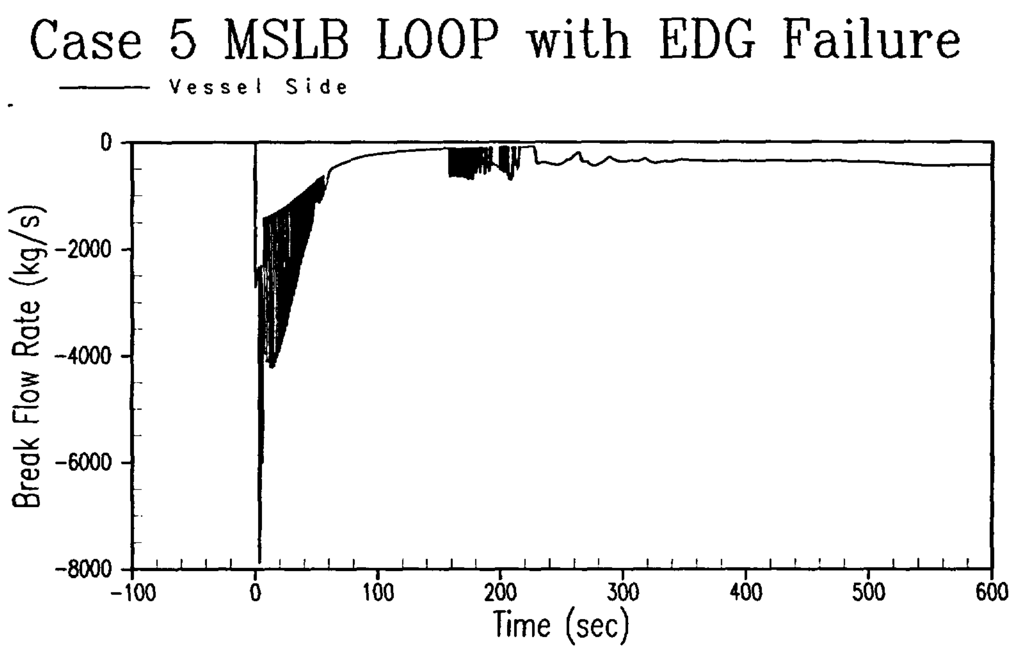

Figure B-14 Case 5 Break Flow Rate

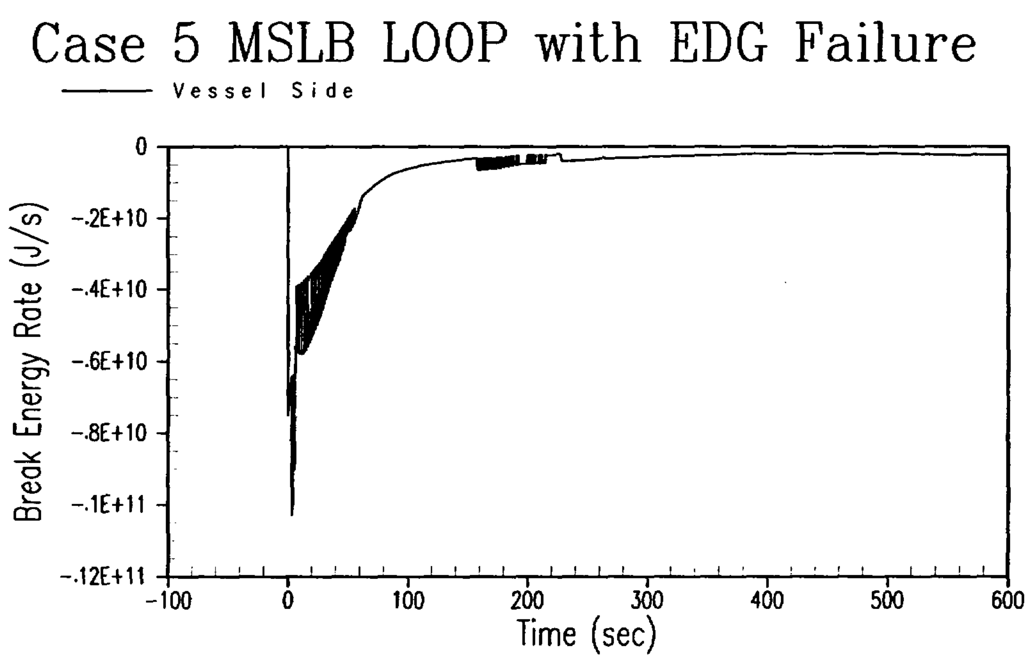

Figure B-15 Case 5 Break Energy Release Rate

B.6 REFERENCES

B-1. "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A, July 1996.

B-2. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification," RPB 90-93-P-A, October 1991.

B-3. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity," RPB 90-94-P-A, October 1991.

B-4. "BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification," CENPD-293-P-A, July 1996.

B-5. "Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel," CENPD-283-P-A, July 1996.

B-6. "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description and Qualification," WCAP-15682-P-A, April 2003.

B-7. "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description and Qualification," WCAP-16078-P-A, November 2004.