AREVA

BAW-10252(NP)
Revision 0

Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC

July 2004

Framatome ANP, Inc.

Non-Proprietary

ACKNOWLEDGMENT

Framatome ANP gratefully acknowledges the following individuals for their expertise in the subject matter and contributions in the preparation of this document and the supporting calculations.

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Nomenclature

Acronym	Definition
ACC	Accumulator
AFW	Auxiliary Feedwater
BOL	Beginning of Life
BWST	Borated Water Storage Tank
CAC	Containment Air Cooler
CFT	Core Flood Tank
CHF	Critical Heat Flux
CLPD	Cold Leg Pump Discharge
CLPS	Cold Leg Pump Suction
CVTR	Carolina Virginia Tube Reactor
DBA	Design Basis Accident
DEG	Double Ended Guillotine
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EM	Evaluation Model
EOB	End of Blowdown
EOL	End of Life
EQ	Equipment Qualification
FANP	Framatome ANP
FW	Feedwater
GDC	General Design Criteria
HDR	Heissdampfreactor
HL	Hot Leg
HLB	Hot Leg Break
HTC	Heat Transfer Coefficient
LBLOCA	Large Break Loss-of-Coolant Accident
LOCA	Loss-of-Coolant Accident
LOOP	Loss Of Offsite Power
LP	Lower Plenum
M&E	Mass and Energy
MDLM	Mist Diffusion Layer Model
MFLB	Main Feedwater Line Break
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSL	Main Steam Line
MSS	Main Steam System
NCG	Non-Condensable Gas
NPSH	Net Positive Suction Head

NRCV	Non-Return Check Valve(s)
NSSS	Nuclear Steam Supply System
OTSG	Once-Through Steam Generator
PCT	Peak Clad Temperature
PWR	Pressurized Water Reactor
RAS	Recirculation Actuation Signal
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RSG	Recirculating Steam Generator
RV	Reactor Vessel
RVVV	Reactor Vessel Vent Valves
SAR	Safety Analysis Report
SG	Steam Generator
SRP	Standard Review Plan
TSV	Turbine Stop Valve
UFSAR	Updated Final Safety Analysis Report
UPTF	Upper Plenum Test Facility
USNRC	United States Nuclear Regulatory Commission

Abstract

This report summarizes generic methods developed by Framatome ANP (FANP) to deterministically calculate peak pressure and temperature responses from the rupture of high energy lines inside containment using the GOTHIC computer code. The high energy line breaks considered include; (1) reactor coolant system piping, (2) main steam line breaks, and (3) main feedwater line breaks. The methods are applicable for any large dry containment for a Pressurized Water Reactor (PWR) with either Once-Through Steam Generator (OTSG) or Recirculating Steam Generator (RSG) designs.

The models and methods described herein follow the guidance of NUREG-0800 (SRP Section 6) and American National Standard 56.4, where appropriate. The mass and energy release rates, required as input, were conservatively generated with previously NRC-approved deterministic thermal-hydraulic computer codes or other means shown to be bounding in comparison to a detailed analytical solution. The methods consider plant type (OTSG and RSG plant designs), a spectrum of break sizes and locations, failures of mitigating equipment, and the availability of off-site AC power. The analyses utilized existing plant-specific licensing bases for the selection of key inputs and assumptions. Where no guidance is provided, the methods delineated herein take precedence.

The methodology described herein employed by FANP is consistent with industry practices and provides a calculational process that produces conservative results. The results of demonstration calculations for a loss of coolant accident and main steam line break are presented and discussed herein.

1.0 INTRODUCTION

This report describes the methodology used by Framatome ANP (FANP) to predict the maximum containment pressure and temperature response to a spectrum of high energy line breaks using the GOTHIC computer code. These methods are applicable for any large dry containment for a pressurized water reactor (PWR) with either once-through steam generators (OTSGs) or recirculating steam generators (RSGs). Included in the methodology is a description of the inputs to the model including the generation of mass and energy (M&E) release from high energy line breaks. This report provides the general foundation that defines the inputs, assumptions, and acceptance criteria that can be used, along with the licensee's plant-specific licensing and design bases delineated in the applicable licensing documents, to calculate mass and energy release rates and determine containment integrity.

The methods defined in this topical report consider input from various sources including NUREG-0800 and other industry standards. While these sources were used as guidance, their reference in this report should not be construed as complete compliance, especially for application to plants that were licensed prior to the issuance of NUREG-0800. The bases specified in the plant-specific licensing documents govern the spectrum of breaks and the assumed applicable modes of operation that require analysis for the events considered. However, if specific guidance is not provided in the plant-specific licensing basis, then the methods delineated herein will be followed.

The successful application of these methods demonstrate the adherence to the following acceptance criteria:

(1) peak pressure ≤maximum allowable pressure
(2) peak temperature ≤maximum allowable temperature
(3) containment pressure ≤50% of peak within 24 hours after accident
(4) containment pressure < atmospheric pressure within 1 hour for subatmospheric containments

Other possible uses of this methodology include calculating sump pool temperatures to determine the adequacy of NPSH and building temperature response to determine Equipment Qualification (EQ) envelopes.

For containment integrity analyses, there are two types of postulated ruptures that produce break conditions that could challenge the design basis limits of the containment structure: (1) Reactor Coolant System (RCS) pressure boundary ruptures; Loss of Coolant Accident (LOCA), (2) NSSS piping ruptures; Main Steam Line Break and Main Feedwater Line Break (MSLB and MFLB). The nature of these events necessitates different assumptions for each analysis. This topical report discusses the details of the modeling approach for each type of break.

Section 2.0 presents a description of the GOTHIC computer code, the selected tool for performing containment pressure and temperature analysis. Section 3.0 provides reference to the code validation and describes the applicability to the intended applications. Section 4.0 provides a discussion of the GOTHIC modeling options selected for performing containment analysis. Section 5.0 describes the overall analysis approach including the initial and the boundary conditions assumed. Sample cases are provided in Section 6.0. The conclusions of the containment integrity methodology are presented in Section 7.0. Appendix A presents a comparison of the methods discussed herein with the guidance of NUREG-0800 and ANSI-ANS-56.4-1983. Appendix B provides the mass and energy data used in the sample calculations presented in Section 6.0.

2.0 CODE DESCRIPTION

The GOTHIC computer code determines the primary containment response to the mass and energy additions resulting from postulated breaks in the reactor coolant system, main steam and feedwater piping.

GOTHIC is a general purpose thermal-hydraulic code used for the design, licensing and operating analysis of nuclear power plant containments and other confinement buildings (Reference 1). GOTHIC solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer. The interface models allow for the possibility of thermal non-equilibrium between phases and unequal phase velocities. Conservation equations are solved for three fields (steam/gas mixture, continuous liquid, and liquid droplet). The steam/gas mixture is comprised of steam and a number of non-condensable gases. Mass balances are solved for each component; thereby, providing the volume fractions of each type of gas in the mixture.

The GOTHIC models described in this report use lumped parameter (as opposed to subdivided, multi-dimensional) volumes. As such, the mass and energy balances are maintained among the volumes, and the interconnecting junctions pass the flow from one volume to the next.

The code calculates heat transfer between phases, and between surfaces and the fluid. Liquid droplets are transported in the vapor/gas flow.

GOTHIC includes a general model for heat transfer between solid structures and the steam/gas mixture or the liquid. There is no direct heat transfer between solid structures and liquid droplets. Solid structures are modeled as one-dimensional heat slabs (flat plate, cylindrical tube or solid rod) and there is no limit to the number of solid structures that can be assigned to a particular nodal volume. Nodalization of a conductor allows variation of material properties in the direction of heat transfer. Heat generation within thermal conductors may be specified on a node by node basis. A variety of boundary conditions may be specified for determining the heat transfer from the structure.

GOTHIC also includes an extensive set of models for operating equipment (e.g. pumps and fans, valves and doors, heat exchangers, vacuum breakers, spray nozzles, coolers and heaters, etc.).

GOTHIC version 7.1 was used to perform the sample calculations contained in this report. The method, however, is not restricted to a specific GOTHIC computer code version but rather to the code options and modeling approach described in this topical. The use by FANP of future GOTHIC computer code versions will use the modeling approach and code options described herein.

3.0 CODE VALIDATION

The GOTHIC code validation and model benchmarks (Reference 2) establish the appropriateness of the GOTHIC code and models to predict the containment response to LOCA and MSLB/FWLB postulated accidents.

In addition to qualifying GOTHIC with a comprehensive set of standard problems to test code models, the code has also been used to predict test results from several facilities (e.g. the Battelle-Frankfurt Test facility, the Hanford Containment System Test Facility, the Marviken Full Scale Facility, the Carolina Virginia Tube Reactor (CVTR) Facility, and the Heissdampfreactor Facility (HDR).

Of particular interest are the CVTR design basis accident (DBA) tests which are a series of steam blowdown experiments that provide large scale containment response data. Comparisons in Reference 2 of GOTHIC predictions with data from these tests demonstrate the ability of the GOTHIC code to predict the pressure and temperature history of a large-scale containment facility under DBA conditions.

4.0 GOTHIC MODEL DESCRIPTION

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] The specified containment net free volume is conservatively minimized to maximize predicted pressures. Height is input as the volume divided by the transverse cross-sectional area of the containment cylinder to maintain the default liquid/vapor interface area (volume/height) equal to the cross-sectional area. GOTHIC uses the containment volume hydraulic diameter, D_h , to define the surface area of structures within the volume that may be wetted by a liquid film as, $4V/D_h$, where V is the containment net free volume. The input hydraulic diameter is therefore calculated as, 4V/S, where S is the total exposed structural heat sink surface area.

No water is assumed to be in the sump so the initial containment liquid volume fraction is set to zero.

For dual containments, representation of the outer shield building structure is limited to inclusion of the annular air gap and shield wall concrete as part of the conductors modeling the containment vessel wall and dome. Heat transfer across the annular air gap between the containment vessel and shield wall is conservatively calculated assuming conduction through still air.

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For LBLOCA, the long-term containment model includes additional features to enable the calculation of long-term M&Es as well as the long term containment response. The additional features are described in detail in Section 5 of this report.

The GOTHIC computer code allows the user the flexibility of specifying different modeling options. The user specified GOTHIC options are: (1) phase separation assumption and droplet size for the blowdown mass and energy release, (2) spray droplet size, (3) revaporization fraction, (4) wall condensation heat transfer, (5) fog model, (6) maximum mist density, (7) drop diameter from mist, (8) minimum heat transfer coefficient, (9) reference pressure, (10) forced entrainment drop diameter, (11) vapor phase head correction, (12) kinetic energy, (13) vapor phase, (14) liquid phase, (15) drop phase, (16) force equilibrium and (17) drop-liquid conversion. The FANP method establishes the selection of these options on one or more of the following: supporting experimental data, code vendor recommendations (References 2 and 3), regulatory guidelines or industry consensus. A summary of the selected options is provided in Table 4-1. [

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The basis for each of the selected options in the FANP method is discussed in detail below. These modeling options were applied in the sample calculations presented in Section 6 of this report.

4.1 *Droplet Size and Phase Separation during Blowdown*

GOTHIC includes a droplet break-up model. The industry consensus considers the model not to be fully validated to justify its use. The alternative available in GOTHIC is to bypass the droplet breakup model and specify different blowdown droplet sizes. The GOTHIC 7.1 User's manual (Reference 3) recommends a droplet "average" diameter of 100 microns. This droplet size was justified based on the experimental data of Brown & York (Reference 5) and Park & Lee (Reference 6). In the Brown & York experiments, droplet size ranged from 36.6 to 336 microns with mean of 91 microns. In the Park & Lee experiments, droplet size in partially flashing jets ranged from 90 to 200 microns with mean drop diameter in the 90-120 microns range and in fully flashing jets, droplet size ranged from 60 to 90 microns with mean drop diameter of 70 microns. The code vendor's recommendation of 100 micron droplet size falls within the range of mean droplet sizes reported in References 5 and 6. The 100 micron droplet size has also been accepted by USNRC for use (References 13 and 14) in design basis GOTHIC containment analyses. Therefore, the selection for this option is to use 100 microns "average" droplet size for LOCA and MSLB.

A requirement of a phase separation model to be used in pressure and temperature transient analysis in PWR dry containments is given in ANSI/ANS 56.4 guidelines (Reference 4). [

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4.2 *Revaporization Fraction*

This is the fraction of the condensate to be revaporized.

The superheat in the vapor region of the containment is much more significant following a postulated MSLB than LOCA. Therefore, the MSLB is expected to be more sensitive to this input than LOCA. Regulatory guidelines (Reference 12) allow 8% revaporization to be credited. Therefore, for MSLB, the revaporization fraction is limited to 8%.

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4.3 *Spray Droplet Size*

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] The model requires that the fraction of spray liquid that is converted into drops in the containment atmosphere be specified.

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The heat removal effectiveness of the spray is directly calculated by GOTHIC, taking into account the effect of non-condensable gases on the mass and energy transfer at the drop liquid-vapor interface.

4.4 *Wall Condensation Heat Transfer*

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4.5 *Fog and Mist Models*

The fog and mist model options are mutually exclusive. The mist model was installed by the code vendor to remedy a deficiency in the fog model. The fog model generates very small drops when the atmosphere becomes supersaturated. When these small drops are combined with the large drops from a break or spray, the average drop size is not representative of either population, resulting in possible excessive heat and mass transfer at the drop surfaces. If the mist option is used, mist is created when the atmosphere supersaturates. The mist phase is not combined with the drop phase. It is assumed to move with the vapor phase. When the mist concentration exceeds the user specified limit, the excess mist is either converted to drops or is deposited in the liquid phase, depending on the user specification for the drop diameter from excess mist. If the drop diameter from excess mist is set to zero, then the excess mist is deposited in the liquid phase. Otherwise, new drops are created at the specified drop diameter (default is 200 microns). The mist model is recommended for most modeling situations, where a wide range of drop size is anticipated.

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4.6 *Minimum Heat Transfer Coefficient*

The minimum heat transfer coefficient (HTC) specifies the lower limit on convection heat transfer that applies to liquid-vapor interfacial heat transfer at a pool surface. The code vendor recommends a value of 0.0 (default). Therefore, the selection for this option is to set the Minimum Heat Transfer Coefficient to 0.0.

4.7 *Reference Pressure*

The code vendor recommends setting this option to IGNORE (default). This means local pressure is used to calculate density in the static head terms of the momentum equation. Therefore, the selection for this option is to set the Reference Pressure to IGNORE.

4.8 *Forced Entrainment Drop Diameter*

This parameter is not applicable since this option is used for subdivided volumes. This option is set to DEFAULT.

4.9 *Vapor Phase Head Correction*

When this parameter is set to INCLUDE, the static head of the pool liquid that is above the volume 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. For the intended application this parameter has no effect (liquid level is significantly lower than the Volume center line). The selection for this option is set to INCLUDE to ensure vapor phase pressure is calculated correctly for deep and shallow pools.

4.10 *Kinetic Energy*

The kinetic energy transport terms are ignored. This is reasonable for large enclosures such as containments. The blowdown mass and energy releases are modeled as boundary condition forcing functions. These forcing functions – when developed by the methods herein – include the effect of kinetic energy. Therefore, the selection for this option is to set Kinetic Energy to IGNORE.

4.11 *Vapor Phase, Liquid and Drop Phases*

The vapor phase, i.e., steam and non-condensables, liquid phase and droplet phase conservation equations are included in the model. Therefore, the selection for each of the phases is set to INCLUDE.

4.12 *Force Equilibrium*

If set to IGNORE, non equilibrium calculations are performed for interfacial heat transfer and momentum. Therefore, the selection for this option is to set Force Equilibrium to IGNORE.

4.13 *Drop-Liquid Conversion*

If set to INCLUDE, GOTHIC accounts for drop phase entrainment, agglomeration and deposition phenomena. Therefore, the selection for this option is to set Drop-Liquid

Conversion to INCLUDE. This option allows the drops in the vapor space to agglomerate and deposit in the liquid region.

5.0 ANALYSIS APPROACH

The PWR containment analysis method described in this topical is used to analyze containment response to design basis LBLOCA and MSLB/FWLB. The LBLOCA analysis includes large breaks in the cold leg pump suction and discharge piping as well as large breaks in the hot leg piping. The MSLB/FWLB includes breaks up to and including the double-ended severance of a main steam line or feedwater system pipe.

The blowdown or short-term mass and energy release used by the GOTHIC computer code can be from either LOCA and MSLB mass and energy release calculations or a plant specific Safety Analysis Report (SAR).

The GOTHIC long-term containment model for MSLB containment response is identical to the short-term model except the mass and energy release data is extended further in time in the long term containment model.

For LBLOCA, the short-term containment model is modified, primarily by adding a vessel node, to enable it to perform long-term mass and energy release as well as long-term containment response calculations.

Further detail of each aspect of the analysis approach is provided in the following sections.

5.1 *Containment Response to LOCA*

Large Break Loss-of-Coolant Accidents (LBLOCA) are performed to generate mass and energy release boundary conditions for determining the containment response to a spectrum of breaks in the reactor coolant system (RCS) piping. For a PWR, LBLOCA mass and energy release analyses and containment response can generally be divided into five phases: blowdown, refill, reflood, post-reflood, and long-term decay heat removal. Each of these phases is discussed in detail in the following paragraphs.

The blowdown phase is characterized by the rapid depressurization of the RCS to a condition nearly in pressure equilibrium with its immediate surroundings. The resultant core flow is variable and dependent on the nature, size, and location of the break. Departure from nucleate boiling (DNB) is calculated to occur quickly, and core cooling is by a film boiling process. Since film boiling amounts to only a small fraction of the steady-state cooling, the cladding temperature increases. Once the RCS pressure decreases below the accumulator cover gas pressure, flow from the emergency core cooling system (ECCS) begins to enter the system. The resulting condensation increases the core velocities and begins to decrease the clad temperature. Blowdown ends when the primary system pressure is approximately equal to the containment pressure. During this phase virtually the entire primary system inventory is discharged to the containment; however, some saturated liquid remains in the reactor vessel (RV).

Following blowdown, a refill period occurs where the ECCS provides sufficient liquid to fill the lower head/plenum region of the reactor vessel. When the ECCS water reaches the bottom of the core, the refill period is over, and the reflood phase begins.

During the reflood phase, a substantial quantity of liquid is carried out of the core with the steam generated by the core-to-coolant heat transfer. For hot leg breaks, this saturated mixture exits the break to the containment atmosphere. For cold leg breaks, this two-phase mixture travels through the steam generators absorbing energy from the secondary side fluid, possibly becoming superheated and then exiting through the break. For B&W plants, some of this two-phase mixture is vented through the reactor vessel vent valves (RVVV) to the reactor vessel downcomer where it has the potential to be condensed by the injecting ECCS fluid prior to exiting the break in the cold leg. This phase ends once the mixture level in the core is sufficient to reduce the cladding to near the saturation temperature of the fluid.

Once the core is quenched, the LBLOCA proceeds into the post-reflood phase. This period is characterized by the core decay heat generating a significant two-phase mixture in the core region. This mixture is carried to the break location as described above. In addition to the core decay heat removal, the reactor coolant system and secondary side fluid and metal sensible heat is assumed released to the containment during this phase.

The final phase of the LBLOCA is the decay heat period, which is characterized by the release of the core decay heat. This period extends from the end of the post-reflood phase.

The following sections describe the modeling approach for containment integrity analyses given a rupture of the RCS pressure boundary.

5.1.1 *Initial Conditions*

GOTHIC requires initial conditions within the containment building to be specified at the start of the simulation. These include initial pressure, temperature and relative humidity.

The maximum allowable containment pressure, consistent with the plant Technical Specification, is used.

For a given pressure, minimizing the initial containment temperature provides a greater mass of non-condensable air in the containment vessel, but also a lower initial heat sink conductor temperature. Conversely, maximizing the initial containment temperature maximizes the temperature of the heat conductor surfaces, making them less effective heat sinks, but also reduces the initial mass of non-condensable air.

A larger mass of non-condensable air tends to increase containment pressure response due to an increase in the partial pressure from non-condensables and a reduction in the Uchida condensing heat transfer coefficient assumed for conductor surfaces exposed to steam due to the decrease in the steam/air mass ratio. These effects may be offset, however, by the potential increase in conductor heat removal capacity due to the lower initial temperature.

Typically, for containments with a limiting blowdown peak, a lower initial temperature may be conservative for calculating the maximum containment pressure. For containments with a limiting reflood peak, a maximum containment initial temperature may result in a higher peak due to the lower heat sink heat absorption. Therefore, if the plant specific licensing basis does not identify the limiting initial temperature, a conservative initial containment temperature will be determined from the results of a sensitivity analysis.

A lower humidity is conservative for the containment maximum pressure analyses. A value consistent with the licensing basis is selected. If this value is not identified in the licensing basis, a lower bound dewpoint will be used.

In addition to the initial containment building conditions, initial conditions for the NSSS must be selected when generating mass and energy releases. The initial conditions used to generate the LOCA mass and energy release to containment are chosen consistent with licensing basis of the analyzed plant. Where no guidance is given in the plant licensing basis documents, the initial conditions will be consistent with the approved LOCA methods (References 18 and 19).

5.1.2 *Boundary Conditions*

The boundary conditions for the GOTHIC containment model can be grouped into three categories: (1) containment heat conducting structures, (2) containment heat removal systems, and (3) break mass and energy releases. The modeling approach used to account for these boundary conditions are described in following sections.

5.1.2.1 *Containment Heat Structures*

GOTHIC conductors are used to model the major structural heat sinks inside the containment vessel, using appropriate regional subdivisions to represent changes in materials and material properties. The subregion nodalizations may be specified by user input or the GOTHIC automatic nodalization feature may be used. The automatic nodalization feature provides appropriate regional subdivision of the conductor based on user selected characterizations of the magnitude of the expected heat transfer coefficient on the conductor inner and outer surfaces, selecting from the following options:

Option	Order of Magnitude of surface HTC (Btu/hr-ft^2-^{o}F)
ZERO	0
V. LOW	1
LOW	10
MEDIUM	100
HIGH	1000
V. HIGH	10000

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Since natural convection heat transfer is generally insensitive to characteristic length for turbulent conditions, the characteristic length used in the convection heat transfer correlations is allowed to default to the input containment volume hydraulic diameter.

When credited, heat transfer to structures from the liquid pool is based on liquid natural convection.

5.1.2.2 *Containment Heat Removal Systems*

Containment Air Cooler(s) (CAC) are modeled (when credited) using the GOTHIC cooler or fan cooler heat exchanger models to remove heat from the containment volume vapor space.

Conservative values of the CAC heat removal capacity as a function of inlet air temperature are specified for the cooler heat removal rate along with the cooler inlet air stream volumetric flow rate. Specification of the inlet volumetric air flow rate limits the heat removal to the contents of the inlet flow stream.

As an alternative, the GOTHIC Fan Cooler heat exchanger model may be employed to model the performance of the CAC. This model uses a detailed description of the CAC heat exchanger design (configuration, number of passes, shell and tube side flow path, heat exchanger tube and fin geometry) to calculate the CAC performance based on standard heat exchanger principles. The fan cooler model allows heat transfer from the primary side steam-gas mixture to a specified secondary side coolant flow stream.

In both CAC modeling options steam in the inlet air flow stream is condensed to the extent allowed by the specified/calculated heat transfer and amount of steam in the flow stream.

Containment sprays are modeled using GOTHIC flow boundary conditions and spray nozzle options to specify conservative values of spray flow rate, temperature, and mean spray drop size.

For long-term containment response applications, recirculation from the containment sump is modeled in GOTHIC as a flow boundary condition.

The performance of the decay/residual heat removal heat exchanger in the recirculation flow path is modeled using the GOTHIC liquid-to-liquid heat exchanger or cooler models with appropriate conservative input. Secondary coolant heat exchanger loops for component cooling and/or service water may also be explicitly included as necessary using similar modeling to capture plant performance characteristics.

Inputs for the containment heat removal systems are chosen in a conservative manner consistent with the specific plant licensing basis.

5.1.2.3 *Mass and Energy Release for LOCA*

The initial and boundary conditions for the mass and energy release calculations are chosen to maximize the stored energy in the primary and secondary coolant systems and to maximize the removal of this energy to containment, respectively. The sources of stored and generated energy in the analyses are the same as those considered in the approved LOCA models for Appendix K analyses. They include: reactor power; decay heat; stored energy in the core; stored energy in the RCS metal, including the reactor vessel (RV) and RV internals; metal-water reaction energy; and stored energy in the secondary system, including SG tubing and secondary water. Maximizing the heat removal will add the most mass and energy to containment in the shortest amount of time, ensuring a conservative containment analysis. Since this requirement is generally contrary to the intent of the PCT calculations, certain changes to the approved LOCA methods are required to conservatively determine the mass and energy release for containment integrity. These modifications as well as certain clarifications are described in the following sections. For items not discussed, the approach used in the approved LOCA methods will be used.

The mass and energy (M&E) release data could be obtained from various approved sources. The different sources could be; (1) approved M&E data in a licensee's Updated Final Safety Analysis Report (UFSAR) and/or other licensing basis documents that remain applicable and appropriate to the configuration of the plant, (2) new M&E data generated with approved methods from another company other than FANP, or (3) new M&E data generated by FANP using the methods described herein.

Where guidance is provided, the plant-specific licensing basis will govern the spectrum of breaks and the assumed applicable modes of operation that require analysis for the events considered. However, if specific guidance is not provided in the plant-specific licensing basis, then the adjustments to the Appendix K LOCA models outlined in the following sections are followed. The RELAP5/MOD2-B&W code package was approved in Reference 10 for use in LOCA and non-LOCA applications, and the approved Appendix K LOCA EM methodologies utilizing RELAP5/MOD2-B&W are documented in References 18 and 19.

5.1.2.3.1 Short-Term Mass and Energy Release

The mass and energy release rates following postulated ruptures of the RCS piping can be determined through a number of methods including (but not limited to) hand calculations, the plant updated final safety analysis report (UFSAR), other licensing basis analysis of record, other vendor calculations, or FANP calculations (system code or GOTHIC). In all cases, the model inputs, initial and boundary conditions, single active failures chosen, and the plant system performance considered result in a conservative estimation of the mass and energy release rates.

For mass and energy release rates determined by FANP, the calculations consider the same licensing basis assumptions specified in plant-specific UFSAR for the spectrum of breaks and the assumed applicable modes of operation that require analysis. If specific guidance is not provided in the plant-specific licensing basis, then the guidance outlined in this section will be followed. When appropriate, the calculations will utilize a large, system thermal-hydraulic code. FANP has approval to utilize the RELAP5/MOD2-B&W thermal-hydraulic analysis code (Reference 10) for the deterministic prediction of the system response to a LOCA for OTSG and RSG plants for compliance with the 10 CFR 50.46 acceptance criteria. The RELAP5/MOD2-B&W code has been reviewed and approved by the NRC to ensure that the requirements of 10 CFR Appendix K have been followed. However, the objective of the Appendix K methods is to limit the energy transfer from the core fuel elements to the RCS fluid such that the cladding temperature is maximized. While this approach is appropriate for analyses pursuant to 10 CFR 50.46, it is not appropriate for the generation of mass and energy release rates for containment integrity analyses. In order to maximize the containment temperature and pressure response following a LOCA, the heat removal from the fuel elements should be maximized.

The Appendix K models (References 18 and 19) are used as a starting point and adjustments made to conservatively determine mass and energy release rates for containment analyses. [

]

The adjustment to the Appendix K methods considered input from various sources including NUREG-0800 and ANSI/ANS-56.4. While these sources were used as guidance, their use does not imply that all analyses will strictly comply with them, especially for application to plants that were licensed prior to the issuance of NUREG-0800.

The following sections detail the adjustments to an approved Appendix K model to generate mass and energy release rates for containment analyses. For items not specifically discussed, the methods described in the approved Appendix K LOCA topical reports (Reference 18 and 19) apply.

5.1.2.3.1.1 Sources of Energy

The sources of stored and generated energy in the analyses include: reactor power; decay heat; stored energy in the core; stored energy in the RCS metal, including the RV and RV internals; metal-water reaction energy; and stored energy in the secondary system, including SG tubing and secondary water.

The initial reactor power level is consistent with the rated power level plus an appropriate calorimetric uncertainty. Reactivity components are chosen to provide a conservative insertion of negative reactivity. [

] An appropriate initial stored energy in the core is ensured by utilizing a conservatively high initial fuel temperature. The RCS metal is modeled appropriately in regard to size, location and composition. The secondary side metal mass that is in contact with RCS fluid will be explicitly modeled. The system code includes appropriate computation of the heat transfer across the SG tubes. The energy addition due to the metal-water reaction is calculated based on the same correlation (Baker-Just) specified in the approved Appendix K EM (Reference 18 and 19).

5.1.2.3.1.2 Break Flow Calculations

Break flow calculations are performed with a model that has been shown to be conservative in comparison to experimental data. The LOCA methods established for Appendix K calculations typically demonstrate that this criterion is met. [

]

During certain conditions following a LOCA, the condensing of steam due to the injection of the relatively cold ECC fluid causes a reduction of the RCS pressure below the containment pressure. Once the RCS pressure drops below the containment pressure, the potential exists for inflow from the containment. To maximize the integrated mass and energy release rates, the inflow from the containment during these periods will not be credited.

[

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5.1.2.3.1.3 Heat Transfer

NUREG-0800, Section 6.2.1.3, specifies the heat transfer modes to be considered for mass and energy release calculations. The heat transfer modes in the LOCA models include nucleate boiling in the presence of liquid and forced convection in the presence of steam on the primary side. The models also address natural convection and condensing heat transfer on the secondary side. The mass and energy release calculations compute heat transfer from surfaces exposed to primary coolant using models and heat transfer correlations internal to the RELAP5/MOD2-B&W code package including reflood heat transfer models as benchmarked in the BEACH Topical Report (Reference 20). The BEACH heat transfer models are inclusive in the RELAP5/MOD2-B&W code as approved in Reference 10.

5.1.2.3.1.4 CHF Calculation and Return to Nucleate Boiling

The LOCA methods established for Appendix K calculations typically predict early critical heat flux (CHF) with no return to nucleate boiling. These assumptions maximize the energy retained in the fuel pins thereby maximizing the peak cladding temperature.

[

]

5.1.2.3.1.5 Length of Refill Period

After the blowdown phase, the reactor vessel lower plenum is refilled by ECCS. In Appendix K LOCA calculations, a conservatively long refill period is assumed wherein the heat transfer from the fuel pins is negligible (i.e., the fuel experiences a nearly adiabatic heatup). Modeling the refill period in this manner ensures that a conservative peak cladding temperature is predicted by minimizing the heat removal from the fuel pins.

For LOCA mass and energy release rate calculations, a refill period where the fuel experiences a nearly adiabatic heatup is not appropriate, because the intent is to transport the core energy to the containment environment as quickly as possible. [

]

5.1.2.3.1.6 *Liquid Entrainment and Steam Quenching*

NUREG-0800 Section 6.2.1.3 indicates that the calculation of liquid entrainment should be based on the PWR FLECHT experiments and continue until the water level is two feet from the top of the core. Further, the steam quenching should be justified by comparison with applicable experimental data. [

]

5.1.2.3.1.7 *Superheat of Liquid Exiting SG for CL breaks*

NUREG-0800 Section 6.2.1.3 indicates that for cold leg breaks the steam leaving the SGs should be assumed to be superheated to the temperature of the secondary coolant. For non-mechanistic calculations, this assumption is appropriate to ensure conservative results. However, the current LOCA models mechanistically consider the heat transfer through the SG tubes from the secondary side to the primary fluid using approved correlations that have been benchmarked to test data. These models predict superheated steam at the exit of the steam generator. [

]

5.1.2.3.1.8 Fuel Pin Swelling and Rupture

The EM core models typically include a fuel clad swelling and rupture model that includes provisions for metal-water reaction energy addition. While the swelling and rupture of the cladding increase the gap resistance between the fuel pellet and cladding, it also increases the surface area of the cladding for consideration in the metal-water reaction. [

] Further, the cladding heatup for the mass and energy release calculations is typically less than approximately 1600 F, below which the clad swelling is minimal and rupture is unlikely.

5.1.2.3.1.9 Break Sizes and Locations

In general, three double-ended guillotine pipe break locations are examined for each application: (1) hot leg at the reactor vessel outlet, (2) cold leg pump discharge, and (3) cold leg pump suction. Breaks of lesser flow areas (splits and reduced area guillotine) are considered to ensure that the limiting containment pressure is achieved with the larger breaks. If, for a particular application, the plant design basis incorporates other break sizes or locations, then those would also be modeled to maintain the plant design basis (e.g., postulated SBLOCA scenarios for EQ temperature profile analyses). For plants with once-through steam generators (OTSGs), an additional hot leg break location is examined in the U-bend near the SG inlet. This additional break is postulated because of the elevation change of the hot leg piping from the RV hot leg nozzle to the top of the OTSG.

The following sections discuss each break location in detail.

Hot Leg at Exit of Reactor Vessel

This break is located in the horizontal section of the hot leg (HL) piping near the reactor vessel (RV) outlet nozzle. Generally, the flow area of hot leg piping is the largest in the entire RCS piping. Thus, a double-ended guillotine break at this location represents the largest possible break. The larger break area produces the highest blowdown mass and energy release rate. Consequently, hot leg breaks generally produce the greatest containment pressure near the end of blowdown.

Hot Leg Near Inlet of Steam Generator (OTSG only)

This break is located in the upper U-bend (candy cane) of the hot leg piping near the SG inlet nozzle. As stated earlier, this location is only applicable to the OTSG plant design. While the break is still in the hot leg, the elevation change will cause the event to progress differently than a break at the RV outlet nozzle. The blowdown M&E release rate is expected to be lower than the RV outlet break. However, due to the break location there exists the potential for a more rapid removal of the stored energy from the secondary side of the steam generator.

Cold Leg Pump Discharge

This break is located in the horizontal section of the cold leg piping between the RV and reactor coolant pump (RCP), near the RV inlet nozzle. The blowdown M&E release rates are expected to be smaller in comparison to the HL break due to the resistance of the RCP which causes a reduction in the break flow rate from the SG side of the break. While it is anticipated that this break location will produce lower blowdown mass and energy release rates in comparison to the HL break, the CLPD break is more important for the modeling of the release of non-condensable gas into the containment. This break location provides a direct release path from the RV to the containment for the accumulator cover gas after it empties. Additionally, the potential exists during the core reflood phase of the LOCA for the steam generated in the core to flow through the hot leg and become superheated in the SG before exiting the break in the cold leg. For a

B&W plant, some of the steam will flow through the RV vent valves and be quenched in the upper RV downcomer or discharged into the containment.

Cold Leg Pump Suction

This break is located in the cold leg piping between the SG and the RCP. The blowdown M&E release rates are expected to be smaller in comparison to the HL break due to the resistance of the RCP, which causes a reduction in the break flow rate from the RV side of the break. During the core reflood phase, the CLPD piping is filled by the injected ECCS water. Eventually, the water level in the CLPD will reach the RCP spillover elevation and possibly form a liquid seal. The break effluent from the pump side of the break is then essentially ECCS liquid. For plants without reactor vessel vent valves (RVVVs), this effluent is subcooled. For plants with RVVVs, the core steam passing through the valves is condensed by the ECCS liquid; thus, increasing the ECCS water temperature. Given the possibility of the CLPD piping becoming plugged by the water level, the steam generated in the core may eventually flow through the HL and possibly becoming superheated in the SG before exiting the SG side of the break.

5.1.2.3.1.10 Most Limiting Single Failure

A limiting, single active failure is assumed. For LOCA containment integrity analyses, the failure of a safety-grade electrical distribution is often limiting. This failure results in the loss of a train of ECCS and one train of containment heat removal equipment. However, in some cases, it may be more limiting to maximize the ECCS so that the energy removal rate in the RCS is increased. In this case, the limiting single failure may be the loss of a containment spray pump. Sensitivity studies are used to determine the limiting, single active failure the first time this methodology is applied to a plant.

[

]

5.1.2.3.1.11 Offsite Power

In response to a LOCA, the initiation of a turbine trip may cause grid disturbance that may result in a loss of offsite power (LOOP). Subsequently, the RCPs coastdown and the emergency diesel generators start to provide safety-grade electrical distribution to the ECCS and containment cooling equipment. If offsite power is not lost, the RCPs continue to run and add heat to the RCS fluid directly and by forcing additional flow through the steam generators where additional heat is added from the secondary side. In this case, the pumps are not tripped until some time after the loss of subcooling margin. In each application, a sensitivity study is performed to determine the limiting condition with respect to the availability of offsite power.

5.1.2.3.1.12 Emergency Core Cooling System Injection

The ECCS injection rate and delay times correspond to the choice of single active failure and the availability of offsite power. Typically, LOOP and the loss of a diesel results in the minimum ECCS injection and the longest response times. Conversely, the availability of offsite power and a single failure in the containment heat removal system results in the maximum ECCS injection and the shortest response times. A conservative representation of the delay time consistent with the offsite power assumption is used.

5.1.2.3.1.13 ECCS and Containment Cooling Systems Source

The Technical Specification maximum liquid temperature is assumed for the ECCS liquid. The storage tank that supplies the ECCS and containment cooling systems is not explicitly modeled in the LOCA mass and energy analyses because the codes used to analyze the event are stopped well before the depletion of the storage tank is predicted. The initial level and depletion rate are considered in the determination of the assumed time of Recirculation Actuation Signal (RAS) in the building response portion of the analysis.

5.1.2.3.1.14 Reactor Coolant Pump Operation

The availability of offsite power determines the RCP operation. If offsite power is lost, the pumps coast down coincident with LOOP. If offsite power is available, the pumps continue to run until they are tripped by the operators some time after loss of subcooling margin as defined by the plant procedures. In cases that allow the RC pumps to continue running, the pump degradation can affect the RCS flow rate and the subsequent heat removal from the core. [

]

5.1.2.3.1.15 Main Feedwater

The feedwater pumps continue to run until generation of an appropriate isolation signal or loss-of-offsite power (LOOP) initiates system isolation. The additional mass increases the water level and stored energy on the secondary side; thus, the isolation of the feedwater (FW) system is modeled conservatively slow.

5.1.2.3.1.16 Auxiliary Feedwater System

The actuation of the auxiliary feedwater (AFW) system is either conservatively delayed or not credited. This modeling approach maximizes the secondary side fluid energy available for transfer to the primary system. Because AFW is significantly colder than the fluid in the RCS and SG secondary, the introduction of AFW decreases the temperature difference across the tubes; thus, decreasing the secondary-to-primary heat transfer across the SG tubes.

5.1.2.3.1.17 Accumulator Nitrogen Injection

[

] The presence of NCG affects the partial pressure of the containment atmosphere, effectively raising the total pressure. [

]

5.1.2.3.1.18 Containment Backpressure

The containment backpressure is determined to ensure the calculation of a conservative mass and energy release rate. A time-dependent backpressure may be used that is determined through iteration with the containment code. In this case, the containment pressure will be consistent with the mass and energy release rate for each break size and location. Alternately, a constant value may be chosen. In this case, the value (either maximum or minimum) will be supported by sensitivity studies.

5.1.2.3.2 Long-Term Mass and Energy Release

Once the core is quenched, the LBLOCA proceeds into the long-term cooling phase of the analysis (post-reflood and decay heat phases). During the long term phase of LBLOCA, the core has quenched, the vessel level has recovered to the RCS loop nozzle elevations and the ECCS injection maintains the core covered so that core decay heat removal and sensible heat removal is assured at all times.

In the long-term phase, the vessel thermal hydraulics behaves as a quasi steady state condition. The phenomena that need to be modeled: 1) nearly constant coolant level in the vessel, 2) steam production in the core, and 3) the transfer of heat from the remaining heat sources in the primary and secondary systems (sensible heat). [

] This GOTHIC method calculates mass and energy releases from around the time of core quench (end time of the short term mass and energy releases), extending beyond RAS (recirculation actuation signal), until termination of the simulation (at least 24 hours after the postulated accident). The end time of the short-term M&E is referred to as the transition time to indicate the transition from the boundary condition short-term M&E tables to internal GOTHIC calculations of long-term M&E release rates.

With regard to the containment building, GOTHIC is capable of modeling both the pre-RAS and post-RAS operation of equipment including sump recirculation, the emergency core cooling function of the ECCS, and the containment heat removal functions of the sprays, heat exchangers and fan coolers.

The constituent energy sources (sensible heat) remaining at the transition time are:

(a) primary system fluid stored energy,
(b) primary system passive metal (including core metal stored energy),
(c) secondary system stored energy (fluid + metal), and
(d) core decay heat.

The treatment of each of these sources in the long-term GOTHIC model are described in the following sections. It should be noted that the metal water reaction energy and nitrogen gas source from accumulators are included in the short-term M&E releases because they are released prior to the transition time.

5.1.2.3.2.1 Primary Fluid stored energy

[

] These initial conditions are taken from the system model at the transition time and represent the liquid volume fraction in the RCS and the averaged quantities of RCS liquid and vapor temperatures and RCS pressure.

5.1.2.3.2.2 Primary System Passive Metal Stored Energy

[

] The RCS metal energy is lumped based on exposure to either water or steam environments. [

]

5.1.2.3.2.3 Secondary Stored Energy

For cold leg pump suction breaks, after the initial rapid blowdown (within the first 20-30 seconds), the steam generators transfer heat to the steam vented through the break. This heat transfer process continues into the core reflood and post-reflood phases of the accident until the steam generators depressurize to the primary system pressure.

The SG stored energy remaining at the transition time is modeled to obtain a conservative sensible heat released to containment, based on the following methods:

The system calculation is extended for a short time interval beyond the transition time to develop the heat removal rates from the steam generator secondary. [

] The final temperature of the secondary fluid and metal is ensured to be equal to or less than the saturation temperature at the 24-hour containment pressure.

[

]

5.1.2.3.2.4 Core Decay Heat

The LOCA-EM decay heat curve is used in the GOTHIC model. The LOCA-EM method uses 1.2 times the 1971 decay heat standard plus heavy actinide (References 9 and 10). For cooling times greater than 10^3 but less than 10^7 seconds, a multiplier of 10% is applied in conformance with Branch Technical Position ASB 9-2 (Reference 11). [

]

5.2 *Containment Response to MSLB/FWLB*

Main Steam Line and Feedwater Line Breaks (MSLB & MFLB) are performed to generate mass and energy release boundary conditions to determine the containment response to postulated breaks in the secondary system piping. The breaks range in size and vary by location; however, the general plant system response is similar for all secondary system pipe ruptures. In general, all of the steam generators contribute to the break flow through a common header or crossover pipe until the intact steam generator(s) become isolated from the break by the closure of either simple non-return check valves, main steam isolation valves (MSIV) or, in some cases, turbine stop valves. The feedwater system continues to feed the faulted steam generator until closure of the FW isolation valves. Depending upon the auxiliary feedwater system design, actuation of AFW may provide additional mass to the affected steam generator until either automatic or manual isolation.

The transient progression of a MSLB and a MFLB are quite similar; however, the event transition experiences a slightly different effluent discharge sequence. The break effluent in a MSLB event progresses from single phase steam to two-phase discharge, and then back to a single-phase vapor release. Whereas, the MFLB event initially experiences liquid discharge, then progresses to a two-phase release, and then to a single-phase steam discharge. Regardless of the break effluent progression, both scenarios cause a complete blowdown of a single steam generator, as well as the addition of main and possibly auxiliary feedwater until termination. [

]

The following sections describe the modeling approach for containment integrity analyses due to the rupture of secondary system piping.

5.2.1 *Initial Conditions*

The initial conditions used in the GOTHIC model for secondary breaks are the same as discussed in Section 5.1.1 for LOCA.

The initial conditions used to generate the MSLB/MFLB mass and energy release to containment are chosen consistent with licensing basis of the analyzed plant. Where no guidance is given in the plant licensing basis documents, the initial conditions are consistent with the approved non-LOCA methods (References 9 and 17).

5.2.2 *Boundary Conditions*

The boundary conditions for the GOTHIC containment model can be grouped into three categories: (1) containment heat conducting structures, (2) containment heat removal systems, and (3) break mass and energy releases. The modeling approach used to account for these boundary conditions are described in following sections.

5.2.2.1 *Containment Heat Structures*

The containment heat structure discussion provided in Section 5.1.2.1 for LOCA also applies to secondary breaks.

5.2.2.2 *Containment Heat Removal Systems*

The containment heat removal systems discussion provided in Section 5.1.2.2 for LOCA also applies to secondary breaks.

5.2.2.3 *Mass and Energy Release for MSLB/MFLB*

The mass and energy release rates calculated for MSLB/MFLB events are determined on a case specific basis. Sources include (but are not limited to) hand calculations, the plant updated final safety analysis report (UFSAR), other licensing basis documents, other vendor calculations, or FANP calculations. In all cases, the model inputs, initial and boundary conditions, single active failures chosen, and the plant system performance considered result in a conservative estimation of the mass and energy release rates.

Mass and Energy calculations performed by FANP consider the plant-specific licensing basis for determination of the spectrum of breaks, single failures and assumed applicable modes of operation. If specific guidance is not provided in the plant-specific licensing basis as described in the UFSAR or by the licensee, then the guidance outlined in this section is followed. Calculations are performed using RELAP5/MOD2-B&W (Reference10), which has been approved for predicting the plant response to non-LOCA accidents for B&W-designed plants with OTSGs and for plants with recirculating steam generators (References 9 and 17). The MSLB analysis utilized for the prediction of the core response acceptance criteria is the basis for model development of mass and energy release rate calculations for MSLB and MFLB events. However, the MSLB analysis for core response might utilize different assumptions to minimize the stored energy in the primary system to exacerbate the cooldown and maximize the return to power for consideration in the sub-channel CHF analyses. Therefore, adjustments will be made to provide conservative calculations of mass and energy release rates for containment analyses.

The MSLB/MFLB methods consider input from various sources including NUREG-0800 and ANSI/ANS-56.4. While these sources were used as guidance, their use does not imply that all analyses will strictly comply with them, especially for application to plants that were licensed prior to the issuance of NUREG-0800. The licensing bases specified in plant-specific UFSARs will govern the spectrum of breaks, single failures and assumed modes of operation that require analysis for the events considered. However, if specific guidance is not provided in the plant-specific licensing basis, then the methods outlined in the following sections will be followed.

5.2.2.3.1 Sources of Energy

The sources of stored and generated energy in the analyses include: reactor power; decay heat; stored energy in the RCS; reactor coolant pump heat; stored energy in the affected SG metal, including vessel tubing, feedwater line, and steam line; stored energy in the affected SG water, including feedwater injected prior to FW isolation and steam from the unaffected steam generator(s) prior to MSL isolation; and energy from primary coolant to the affected SG.

Conservative reactivity components are chosen to maximize the insertion of positive reactivity during the cooldown. Fuel temperature and moderator density are the dominant reactivity feedback effects and are modeled in accordance with the approved MSLB methods. [

] The RCS metal is modeled appropriately in regard to size location, and composition. The secondary side metal mass that is in contact with the secondary system fluid is explicitly modeled. The NRC-approved system code includes appropriate computation of the heat transfer across the SG tubes.

Depending on the plant licensing basis, a spectrum of initial reactor power levels – including an appropriate calorimetric uncertainty – is analyzed. [

]

5.2.2.3.2 Break Flow Calculations

The same critical flow model and discharge coefficients used in the MSLB core response model are used for the mass and energy release calculations. The total energy release is based on the stagnation properties for the prediction of the peak containment pressure. [

]

5.2.2.3.3 Reactor Vessel Thermal Mixing

During secondary system pipe ruptures, the potential exists for asymmetric coolant loop temperature responses to be generated due to the overcooling from the affected SG blowdown. The mixing of reactor coolant from various loops and the application of reactivity weighting factors are in accordance with the approved MSLB methods (References 9 and 17).

The RV thermal mixing assumption for OTSG plants is consistent with the approved methods in Reference 17. Appendix A, Item A.1.1 of Reference 17 provides justification of the mixing assumption for core power prediction. This justification applies to the core power prediction in response to any secondary side break whether the application is CHF analyses or mass and energy release rate calculations.

The RV thermal mixing assumption for RSG plants is consistent with the approved methods in Reference 9. Additionally, a weighting factor is applied to the reactivity feedback to ensure a conservative calculation of core power as discussed in Reference 9.

5.2.2.3.4 Heat Transfer

NUREG-0800, Section 6.2.1.4 specifies the heat transfer modes that should be considered for mass and energy release calculations. The heat transfer to the water in the affected steam generator should be based upon nucleate boiling heat transfer. As the steam generator discharges its inventory, the steam generator tubes become uncovered (more so in the OTSG design), and the steam is likely to superheat as it transverses the uncovered tubes. Consequently, film boiling and single-phase heat transfer must be included. Each of these heat transfer modes is included in the NRC-approved RELAP5/MOD2-B&W code (Reference 10).

5.2.2.3.5 Liquid Entrainment

A known phenomena related to large ruptures of main steam system piping is the rapid voiding and liquid swell of the SG water and the consequent entrainment from either high velocity jet streams or the flooding of the break location from the liquid swell. While this phenomenon is expected of the rapid depressurization due to the secondary system pipe rupture, the calculation of the liquid entrainment must be benchmarked to experimental data to ensure an appropriate accounting of entrainment by the break flow model.

The containment pressure and temperature response to a secondary side rupture is highly dependent upon the amount of the break effluent that enters the containment atmosphere as steam (saturated and/or superheated). During the depressurization, there are two phenomena that could reduce the steam contribution by forcing liquid effluent into the containment. One mechanism is the entrainment of liquid drops which are swept out the break due to the high steam velocities, and the other is due to the rapid voiding which causes significant level swell to the break location. [

]

NUREG-0800 Section 6.2.1.4 states that the prediction of liquid entrainment should be supported by experimental data. If the system model is qualified for entrainment calculations, the effect on the entrained liquid of steam separators located upstream from the break is taken in account. Furthermore, a spectrum of cases is analyzed beginning with the double-ended guillotine (DEG) break and decreasing in area until no entrainment is predicted to occur; thus, ensuring the limiting containment response is determined. Likewise, if no entrainment is predicted, a spectrum of cases is analyzed beginning with the double-ended guillotine (DEG) break and decreasing in area until it has been demonstrated that the limiting containment response has been considered.

In cases where entrainment is part of the existing plant licensing basis and/or prototypical test data is available, then modeling the liquid entrainment in the steam effluent is appropriate. The validation of the system code entrainment calculations to the available data will be included in the plant-specific analysis.

In cases where entrainment is not part of the existing plant licensing basis, the inclusion of entrainment is not appropriate. During periods of entrainment prediction, the break energy is non-mechanistically set to the energy of saturated steam.

Level swell to the steam exit nozzle, and subsequent liquid discharge, is caused by void formation (i.e., flashing) in the liquid regions of the steam generator. [

]

5.2.2.3.6 Superheat of Break Effluent

Once through steam generators normally operate with between 35 F and 60 F superheated steam. Superheated steam temperature predictions are inherent to any accident analysis of OTSGs. The computer code and methodology, including inter-phase drag models and CHF and post-CHF heat transfer correlations, described in References 10 and 17 are approved for use in predicting the OTSG behavior following postulated pipe ruptures. Therefore, the discussion herein relates to predicting tube bundle uncovering and subsequent steam superheating in recirculating steam generators.

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5.2.2.3.7 Break Sizes and Locations

The break location is assumed to be upstream of the main steam line isolation valve. A break located downstream of the MSIV would not be expected to be limiting because all the steam generators would be isolated relatively quickly by the closure of the MSIVs; thus, limiting the magnitude of the release to the containment. The only single active failure for that break location would be the failure of the MSIV; however, the piping run would introduce additional pressure drop which would decrease the break flow rate.

A break size spectrum is analyzed beginning with the double-ended guillotine (DEG) break and decreasing in area until it has been demonstrated that the limiting containment response has been considered.

If, for a particular application, the plant design basis incorporates different break locations than discussed above, then those would also be modeled to maintain the plant design basis.

5.2.2.3.8 Most Limiting Single Active Failure

For SLB/FLB containment integrity analyses, a limiting single active failure is considered; however, it is dependent upon the assumed initial conditions and the plant system configuration. The failure of a FWIV results in the continuation of feedwater flow to the affected steam generator until either an automatic or manual isolation occurs. Failure of main feedwater pumps to trip on low-steam pressure safety injection or FW isolation signals also allows maximum feedwater flow to the affected steam generator prior to final isolation. The failure of an AFW control valve to close results in the continuation of AFW flow to the affected stream generator until either an automatic or manual isolation occurs. The MSIV failure allows an unaffected steam generator to contribute to the break flow through the streamline crossover piping. The closure of a non-return check valve, if present, would provide the unaffected SG steam flow termination. A failure of an emergency diesel generator (EDG) to start will cause the loss of a single train of safeguards equipment. The reduced ECCS delivery will reduce the rate of delivery of boric acid which provides shutdown reactivity. Furthermore, the EDG failure will cause the loss of a single train of containment heat removal equipment.

Considering the number of potentially limiting single active failures that could be considered for each application (e.g., feedwater control or isolation valve, auxiliary feedwater control valve, main steam isolation valve, and emergency diesel generator failure), a spectrum of single active failures – predicated by the plant design features and plant-specific licensing basis – are assumed. In each application, a sensitivity study is performed to determine the limiting, single active failure. [

]

5.2.2.3.9 Offsite Power

In response to a MSLB/MFLB, the initiation of a turbine trip may cause grid disturbance that might result in a loss of offsite power (LOOP). Subsequently, the RC and main feedwater pumps would coastdown, and the emergency diesel generators would start to provide safety-grade electrical distribution to the ECCS and containment cooling equipment. If offsite power is available, the RCPs continue to run and add heat to the RCS fluid directly and by forcing additional flow through the steam generators where additional heat is added to the secondary side. The continuation of the forced RCS flow increases the energy transferred to the affected steam generator; thus, increasing the energy discharge to the containment. In general, the mass and energy releases with offsite power available are significantly more severe than with LOOP, causing peak containment pressure and temperature to exceed the same case with LOOP, even when significant losses in containment heat removal (e.g., loss of EDG) are considered.

In each application, a sensitivity study is performed to determine the limiting condition with respect to the availability of offsite power.

5.2.2.3.10 Emergency Core Cooling System Injection

The ECCS injection rate and delay times correspond to the choice in the single active failure and the availability of offsite power. Typically, LOOP and the loss of a diesel result in the minimum ECCS injection and the longest response times. Conversely, the availability of offsite power and a single failure in the containment heat removal system results in the maximum ECCS injection and the shortest response times. A conservative representation of the delay time consistent with the offsite power assumption is used.

5.2.2.3.11 ECCS and Containment Cooling Systems Source

The Technical Specification maximum liquid temperature is assumed for the ECCS liquid. The Technical Specification for the minimum boric acid concentration in the Borated Water Storage Tank (BWST) is assumed for the ECCS injection. [

] The storage tank that supplies the ECCS and containment cooling systems is not explicitly modeled in the MSLB/FWLB mass and energy analyses, because the codes used to analyze the event are stopped well before the depletion of the storage tank is predicted. The initial level and depletion rate are considered in the GOTHIC portion of the event.

5.2.2.3.12 Reactor Coolant Pump Operation

The availability of offsite power determines the operability of the RCPs. If offsite power is lost, the pumps coast down coincident with LOOP. If offsite power is available, the RCPs continue to run throughout the transient; however, if the plant-specific licensing basis allows for the tripping of the RCPs by operator action than this may be modeled as appropriate.

5.2.2.3.13 Main Feedwater

The feedwater pumps continue to run until generation of an appropriate isolation signal or loss-of-offsite power (LOOP) initiates system coastdown and isolation. The additional FW mass increases the water level and stored energy on the secondary side.

[ ]

The initial FW system configuration is dependent upon the initial core power level being modeled. The FW system could be; (1) isolated, (2) operating with a reduced number of pumps, or (3) operating with a full system flow for the given power level. The initial feedwater system configuration and associated FW control and isolation systems govern the magnitude of the feedwater system mass added from the break initiation until isolation. Conservative, simple boundary conditions can be utilized to model the FW flow addition or detailed feedwater system models can be utilized to simulate the time-dependent effects of isolation systems and components on the FW system response to the secondary system pipe rupture. [

]

5.2.2.3.14 Main Steam

For large DEG breaks in the main steam system (MSS), the isolation of the turbine and unaffected steam generators may be relatively quick; whereas, the isolation may be delayed for smaller DEG or split breaks. Therefore, the isolation of the main steam system is modeled subsequent to the generation of the appropriate isolation signal. The choice of a conservative time delay for isolation is predicated by the MSS valve configuration. [

]

5.2.2.3.15 Auxiliary Feedwater System

The actuation of the AFW system is conservatively modeled by assuming a short delay time. The purging of the hotter MFW downstream of the AFW injection location is addressed for in accordance with applicable plant designs (i.e., RSGs with feedrings that inject AFW via the main feedwater header). This modeling approach maximizes the secondary side fluid available for energy transfer from the primary system, and discharge out the break. [

]

AFW flow continues until either automatic or manual isolation occurs. Manual isolation times are provided by the licensee in accordance with the existing licensing basis of the plant.

5.2.2.3.16 Containment Backpressure

The containment backpressure is determined to ensure the calculation of a conservative mass and energy release rate. The containment backpressure is either set to a conservatively low value, or a time-dependent backpressure derived through an iterative approach with the containment code. Either approach ensures a conservative prediction of the mass and energy releases.

Table 4-1 Summary of FANP Selected GOTHIC User Options

GOTHIC User Option	FANP Selection
Blowdown Droplet Size	
Phase Separation	
Revaporization Fraction	
Spray Droplet Size	
Wall Condensation Heat Transfer	
FOG/MIST Model	
Maximum Mist Density	
Mist Droplet Diameter	
Reference Pressure	
Forced Entrainment Droplet Diameter	
Vapor Phase Head Correction	
Kinetic Energy	
Vapor Liquid and Drop Phases	
Force Equilibrium	
Drop-Liquid Conversion	]

6.0 SAMPLE CALCULATIONS

Sample calculations of the containment response to LOCA and MSLB events were performed using the GOTHIC containment analysis method described herein. The sample calculations are based on a raised loop NSSS designed by B&W which is housed in a dual containment consisting of an inner steel containment vessel within a reinforced concrete shield building.

The events analyzed were:

1. A 14.14 ft^2 Hot Leg Split Break Near Top of Steam Generator (HLB)
2. Double-Ended Guillotine Cold Leg Pump Suction Break (CLPS)
3. Double-Ended Guillotine Main Steam Line Break (MSLB)
4. DEG Cold Leg Pump Suction (CLPS) Long Term Response

The first three cases comprise a set of short-term benchmark cases comparing GOTHIC predictions against the CONTEMPT computer code. The results of the comparisons are provided in Section 6.2.

The fourth case presents the results of the CLPS long-term response. This case represents a full application of the GOTHIC methodology including the long-term model described in Section 5.1.2.3.2. Results for this case are presented in Section 6.3

Initial conditions and assumed equipment performance for these analyses are provided in Table 6-1.

The analysis of break number 1, the split HLB, also represents a benchmark of the GOTHIC containment model against the analysis method used in the UFSAR analysis of record for this event. The analysis uses the M&E release data for this event from the UFSAR.

The analyses of breaks 2 through 4 use M&E releases for the CLPS and MSLB events developed by FANP.

6.1 *GOTHIC Containment Model*

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Representation of the outer shield wall and air annulus was limited to inclusion of a still air gap and the concrete outer wall material as subregions in the vessel cylindrical wall and dome heat conductors. This model was used for the short-term response for all four sample calculations. In addition, a long-term model (shown in Figure 6-2) was used for the cold leg pump suction case.

6.1.1 *Input Parameters*

The input parameters for the containment model are shown in Tables 6-1, 6-2 and 6-3.

The analyses of the HLB and CLPS LOCA events assumed a combination of Tagami/Uchida condensation HTCs on the heat conductor surfaces exposed to the containment vapor space (all heat sinks except the refueling pool liner wall), with a rapid exponential decay (2.0 exponent) from the peak Tagami HTC at the end of the blowdown period to the Uchida HTC. Heat transfer between the surface of the refueling pool steel liner and the containment liquid region was based on the "FACE-UP" natural convection correlation available in GOTHIC. The exterior surface of the refueling pool concrete wall was assumed to be adiabatic. A 2 Btu/hr-ft^2-^{o}F HTC was assumed on the exterior surface of the concrete shield structure exposed to ambient air at 90^{o}F. However, due to the large still air gap (annulus), heat transfer from the shield structure to the ambient air has negligible impact on the short-term containment response.

The integrated energy releases and lengths of the blowdown period for the HLB and CLPS events were:

Event	Energy Release (Btu)	End-of-Blowdown (sec)
HLB	3.04×10^8	14.31
CLPS	3.19×10^8	24.00

Mass and energy releases for each break are provided in Appendix B.

Containment Spray was not included in the short-term response sample cases since it was not included in the CONTEMPT analyses to which the results are compared due to the short duration of the transient analyzed and the fact that the peak pressure and temperature occur well before spray flow initiation.

The sample cases used a blowdown M&E boundary condition pressure forcing function option which results in phase separation of the blowdown flow based on flashing of the mixture to the saturation temperature at transient containment atmosphere steam partial pressure to maximize the steam addition rate to the atmosphere, as recommended in ANSI/ANS-56-4 section 4.2.3.2.1.

Initial containment vessel atmospheric conditions were selected to match the values assumed in the CONTEMPT analyses to which the results are compared.

6.1.2 *User Options*

The sample calculations used the set of GOTHIC options selected by FANP for LOCA and MSLB analyses using GOTHIC:

[

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[

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Figure 6-1 Short-term GOTHIC Model

[

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Figure 6-2 Long-term GOTHIC Model

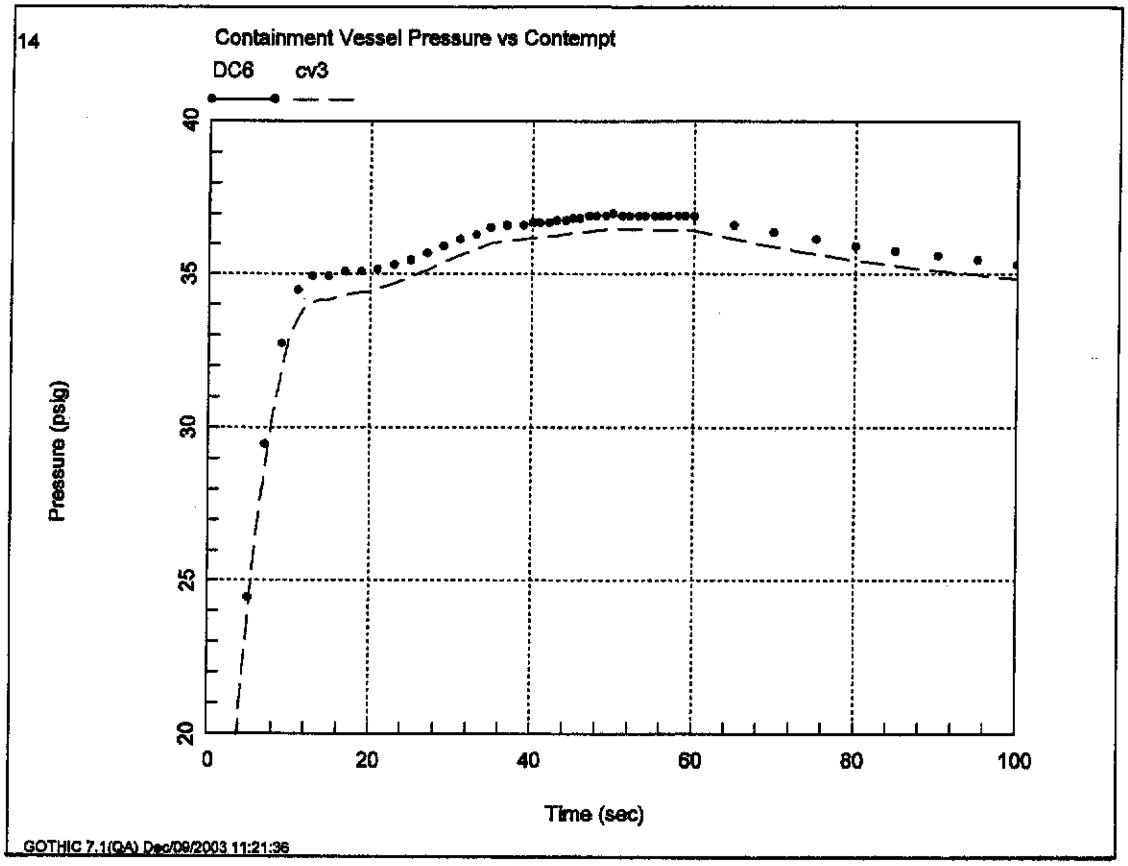
6.2 *Short-Term Benchmark Cases*

6.2.1 *Hot Leg Break Results*

The GOTHIC predictions for containment vessel pressure and vapor space temperature for the 14.14 ft^2 split hot leg break near the steam generator show excellent agreement with the UFSAR and previous CONTEMPT computer code results.

Source	Peak Pressure (psig)	Time of Peak (seconds)	Peak Vapor Temperature (°F)	Time of Peak (seconds)
UFSAR	36.95	50	-	-
CONTEMPT	36.97	50	259.1	49
GOTHIC 7.1a	36.49	50	258.3	50

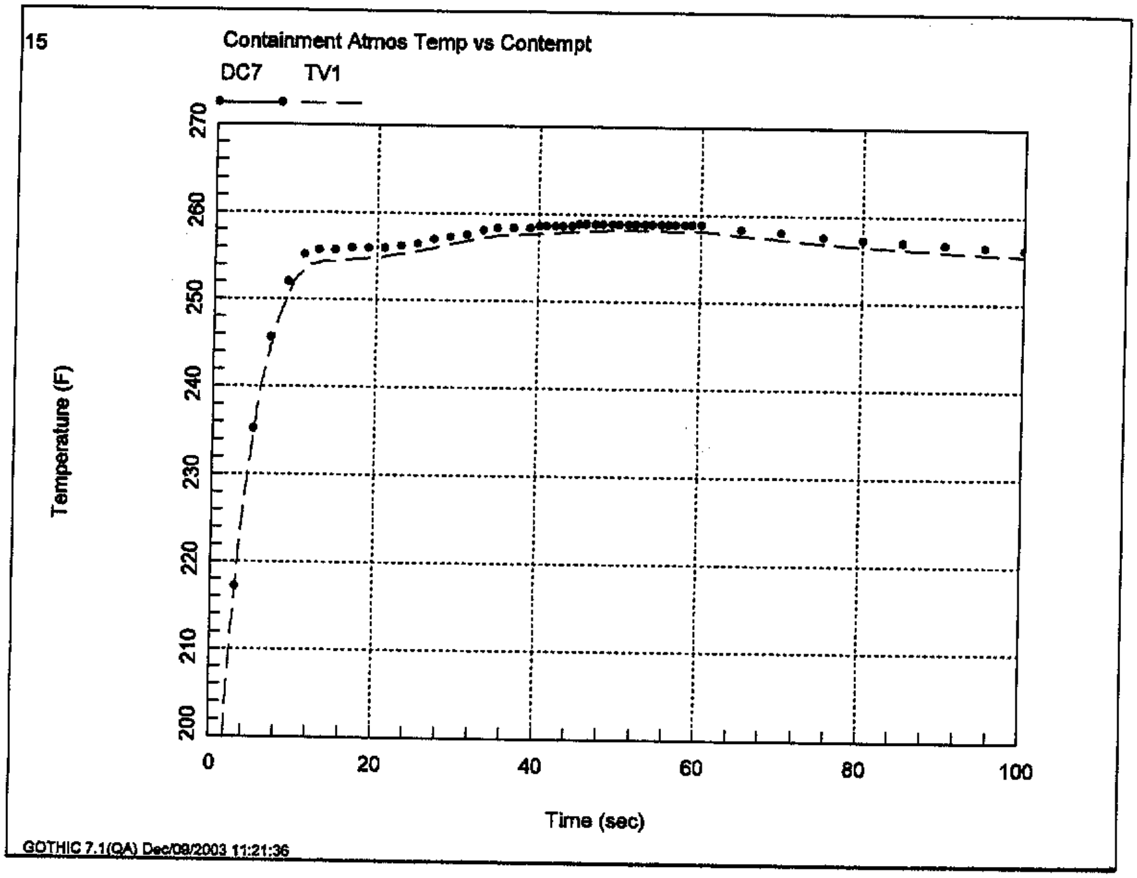
The containment vessel pressure trend predicted by GOTHIC matches the trend observed for the analysis of the same event using the CONTEMPT computer code (Figure 6-3). The peak pressure predicted by GOTHIC is slightly lower as expected, due to the effect of the higher atmospheric heat capacity resulting from the mass of the liquid droplets suspended in the vapor space.



Legend:cv3 - GOTHIC result, DC6 – CONTEMPT result

Figure 6-3 Benchmark of Containment Pressure for Hot Leg Break

The containment vessel vapor space temperature trend shows similar results.



Legend:TV1 - GOTHIC result, DC7 – CONTEMPT result

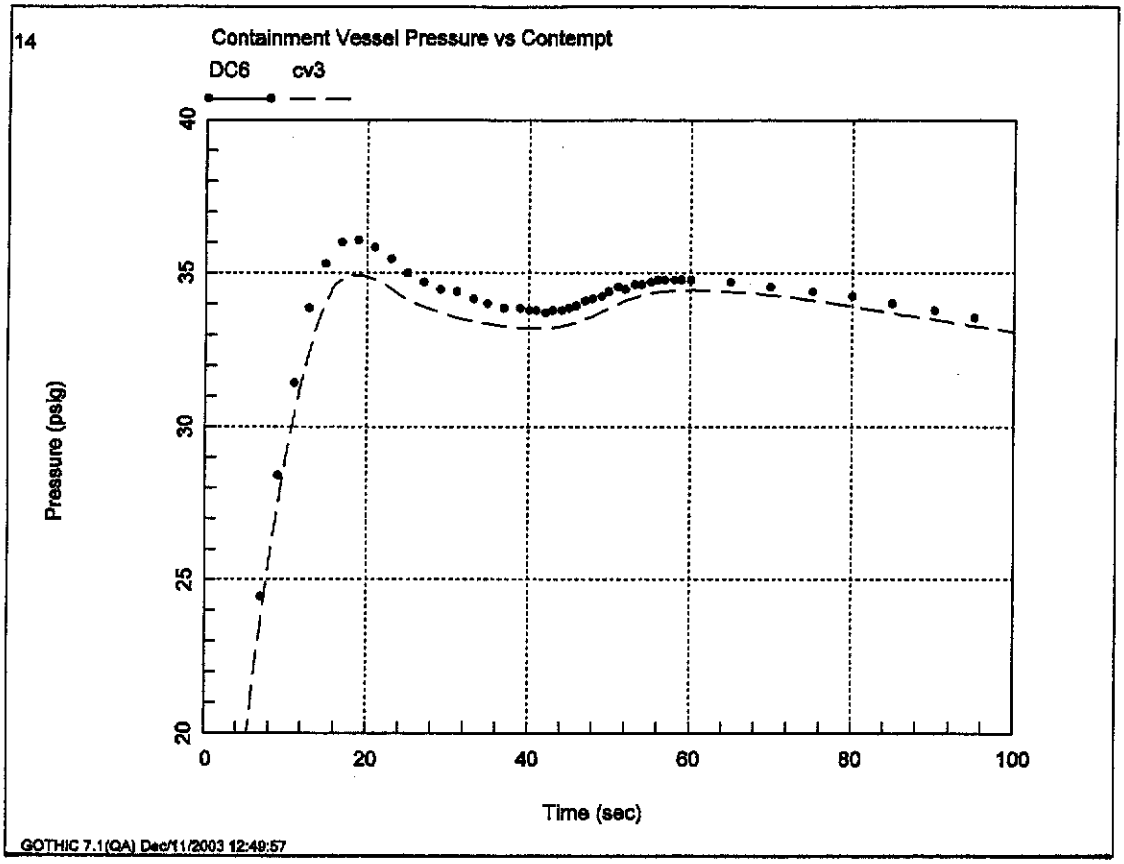
Figure 6-4 Benchmark of Containment Vapor Temperature for Hot Leg Break

6.2.2 *RCP Suction Break Results*

Results for the short-term containment response to the DEG CLPS LOCA event are compared to the predictions for the same M&E release using the CONTEMPT computer code.

Source	Peak Pressure (psig)	Time of Peak (seconds)	Peak Vapor Temperature (°F)	Time of Peak (seconds)
CONTEMPT	36.16	18	257.8	18
GOTHIC 7.1a	34.94	18	255.7	18

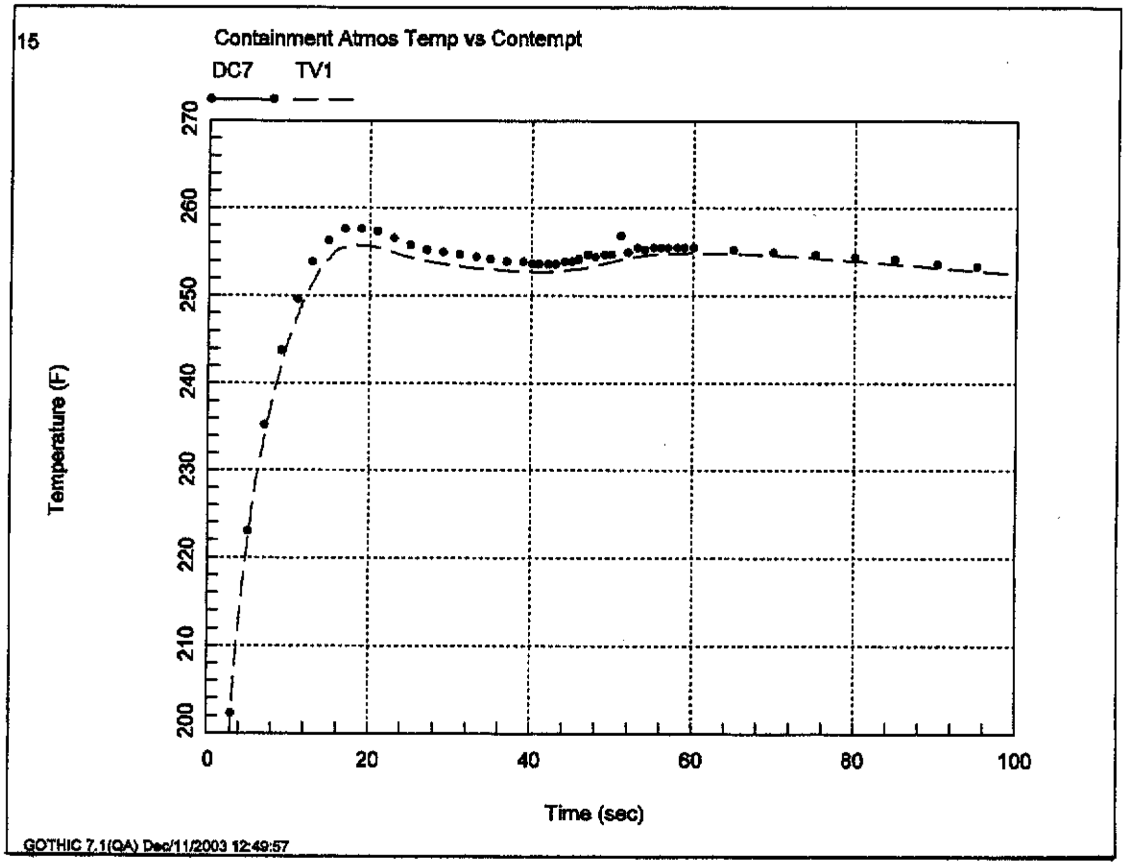
The containment vessel pressure trend predicted by GOTHIC matches the trend observed for the analysis of the same event using the CONTEMPT computer code (Figure 6-5). The peak pressure predicted by GOTHIC is slightly lower as expected, due to the effect of the higher atmospheric heat capacity resulting from the mass of the liquid droplets suspended in the vapor space.



Legend: cv3 – GOTHIC result, DC6 – CONTEMPT result

Figure 6-5 Benchmark of Containment Pressure for RCP Suction Break

Similar results were observed for the containment vessel vapor space temperature response.



Legend: TV1 – GOTHIC result, DC7 – CONTEMPT result

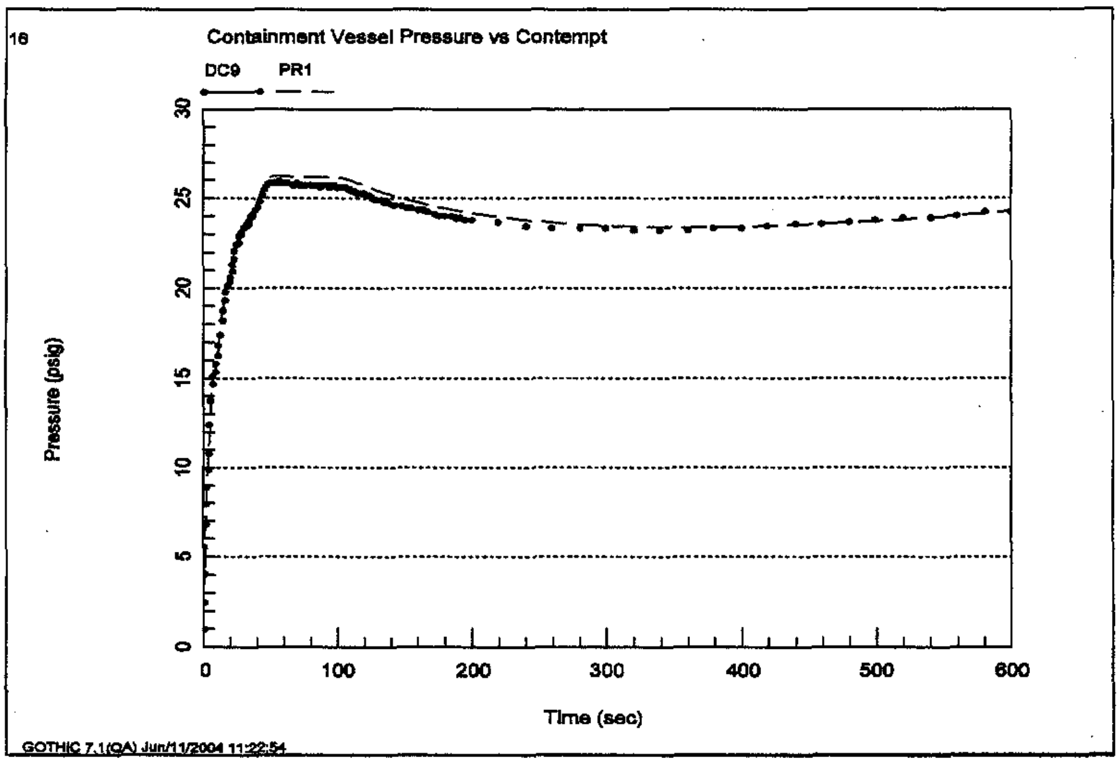
Figure 6-6 Benchmark of Containment Vapor Temperature for RCP Suction Break

6.2.3 *MSLB Results*

Results for the short-term containment response to the DEG MSLB event are compared to the predictions for the same M&E release using the CONTEMPT computer code.

Source	Peak Pressure (psig)	Time of Peak (seconds)	Peak Vapor Temperature (°F)	Time of Peak (seconds)
CONTEMPT	25.90	56.0	353.1	48.0
GOTHIC 7.1a	26.24	55.1	353.8	48.1

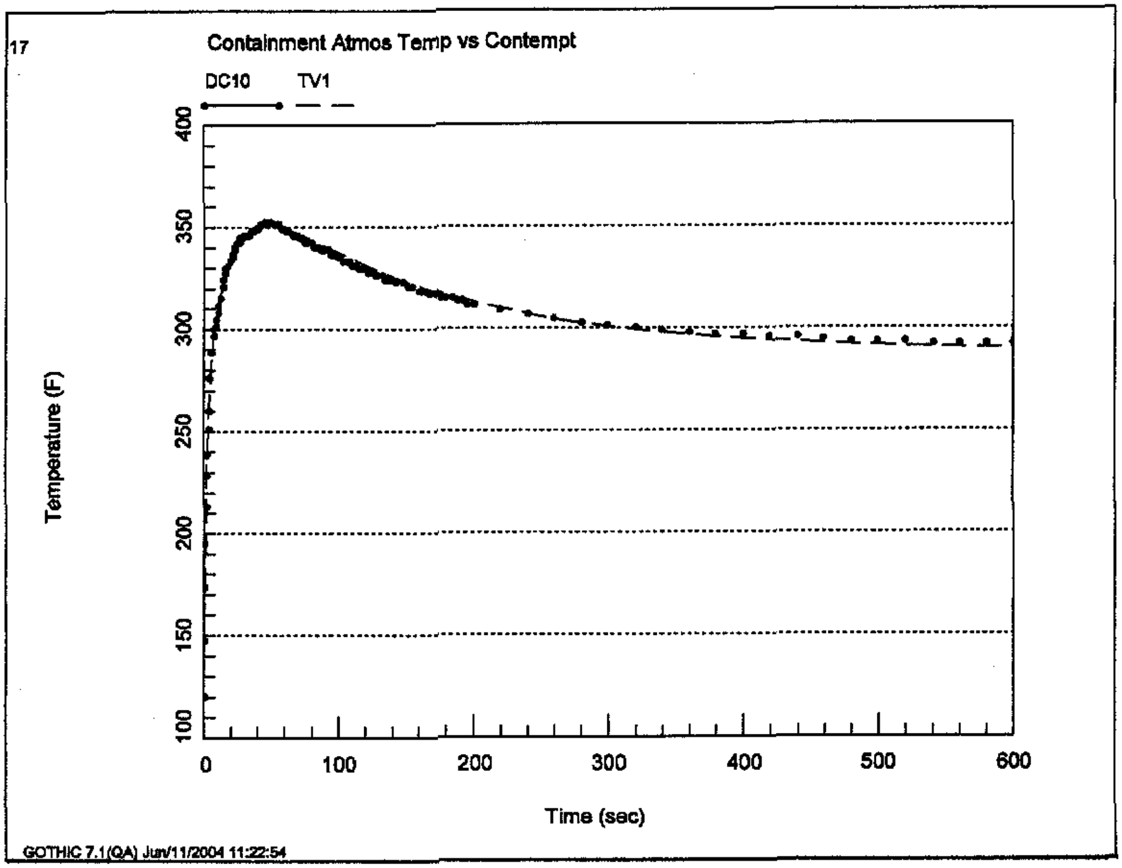
The containment vessel pressure trend predicted by GOTHIC matches the trend observed for the analysis of the same event using the CONTEMPT computer code (Figure 6-7). The peak pressure predicted by GOTHIC closely matches the CONTEMPT result.



Legend: PR1 – GOTHIC result, DC9 – CONTEMPT result

Figure 6-7 Benchmark of Containment Pressure for MSLB

The GOTHIC results for the MSLB case containment atmosphere temperature show excellent agreement relative to CONTEMPT.



Legend: TV1 – GOTHIC result, DC10 – CONTEMPT result

Figure 6-8 Benchmark of Containment Vapor Temperature for MSLB

6.3 *Long-Term RCP Suction Break Results*

Long-term extends from around the time of core quench up to 24 hours after the postulated accident. The major inputs to the long-term model are as follows (refer also to the input parameter Tables 6-1, 6-2, 6-3, B-3, B-4, B-6, and B-7):

(1) The GOTHIC user options selected by FANP are described in Section 4. The Uchida wall condensation correlation was used during post blowdown.

(2) Maximum initial containment pressure of 15.7 psia and minimum initial relative humidity of 10%.

(3) Residual stored energies remaining at 1000 seconds, representation of the transition time from RELAP5/MOD2-B&W mass and energy release boundary condition tables to internal GOTHIC calculation of mass and energy release

(4) Core decay heat curve from transition time to beyond RAS; namely, 24 hours after the postulated accident

(5) The liquid and vapor mass and energy release boundary condition tables for the short term (0-1000 seconds).

(6) The mass and energy release tables of nitrogen injection into the containment

(7) ECCS and containment spray injection during pre-RAS and post-RAS, RAS for this case is modeled at 4500 seconds based on single train operation.

(8) CAC fan cooler performance

(9) Decay heat cooler performance during sump recirculation

(10) Blowdown boundary condition pressure forcing function set to perform phase separation based on the containment atmosphere steam partial pressure per ANSI/ANS 56.4

This model allows GOTHIC to calculate M&E beyond the transition time and extends the calculation of containment pressure and temperature response to beyond RAS.

Figures 6-9 and 6-10 provide the calculated long term containment pressure and temperature response, respectively. The blowdown and reflood peak pressures are nearly equal with the latter peak negligibly lower than the former peak. The long term peak (decay heat peak) pressure is lower than the short term blowdown and reflood peaks. Following the long term peak, the heat removal rates of the decay heat cooler and CAC exceed the core decay heat and passive metal stored energy causing the steady drop in containment pressure to less than half the peak pressure at 24 hours after the postulated accident as required by GDC 38.

The containment vapor temperature response follows the trend of the containment pressure response, i.e. remaining at the saturation temperature corresponding to the containment pressure except for a short period of vapor superheat calculated during post reflood (pre-RAS) due to the discharge into the containment of the residual stored energy; secondary system stored energy and primary system passive metal. As the containment pressure drops during post-reflood, the sump liquid becomes superheated. The decay heat cooler will eventually remove the superheat as the decay heat levels continue to drop beyond 24 hours.

The containment peak pressures as well as the 24-hour pressure and corresponding temperatures are summarized below.

	Time (sec)	Peak Pressure (psig)	Vapor Temperature at Peak Pressure (°F)
Blowdown Peak	19	36.44	254.1
Reflood Peak	62	36.22	253.6
Long-Term Peak	5,540	27.61	238.6
24-hour Pressure	86,400	12.30	189.6

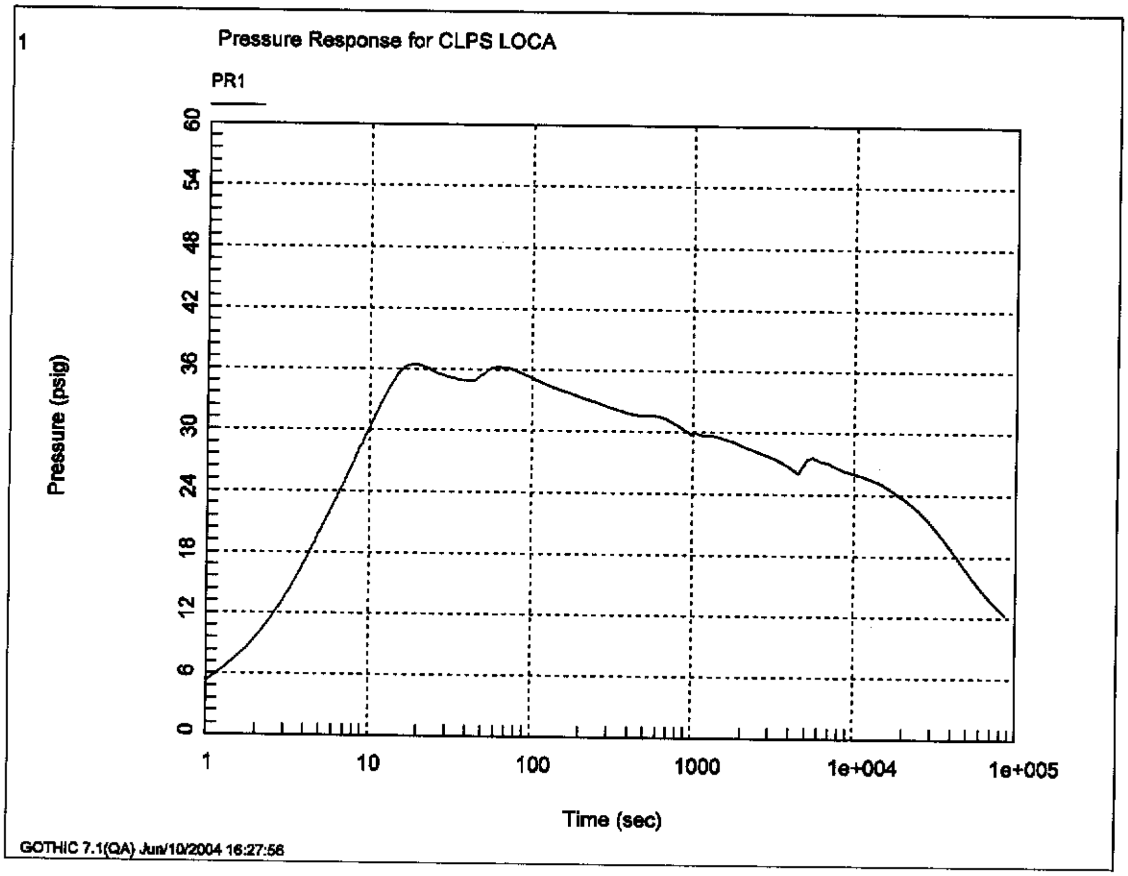


Figure 6-9 Long Term Containment Pressure Response for RCP Suction Break

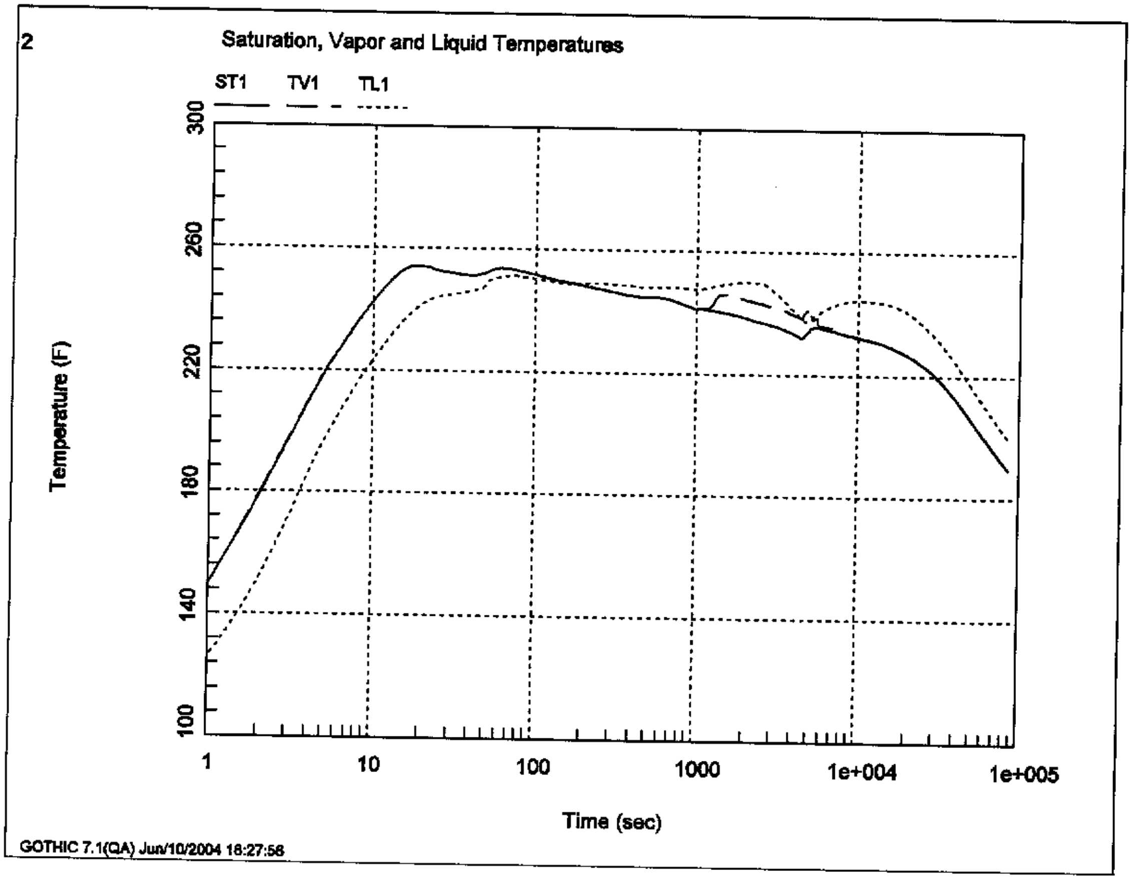


Figure 6-10 Long Term Containment Temperature Response for RCP Suction Break

Table 6-1 Sample Calculation GOTHIC Model Input Parameters

Parameter	Value	Units
Containment Vessel Volume	2,834,000	Ft^3
Containment Vessel Height	213.51	Ft
Containment Vessel D_h	30.48	Ft
Initial Containment Vessel Pressure	14.7 (HLB) 14.7 (short-term CLPS) 15.7 (long-term CLPS) 15.7 (MSLB)	psia
Initial Containment Air Temperature	120	^oF
Initial Containment Air Humidity	70 (HLB) 70 (short-termCLPS) 10 (long-term CLPS) 10 (MSLB)	%RH
CAC Heat Removal Capacity	15,000	Btu/Hr @ 264^oF T_{sat}
CAC Initiation Delay Time	45	sec
CAC Air Flow Rate (single cooler)	45,000	CFM
Containment Spray Actuation	20.83	psia
Containment Spray Flow Rate (single train)	1300	GPM
Containment Spray Delay Time	160	sec
Median Spray Droplet Diameter	780	microns
BWST Temperature	90 (120, long term case)	^oF
LPI Recirculation Flow Rate	3000	GPM
Decay Heat Cooler SW Flow Rate	6000	GPM
Decay Heat Cooler Service Water Inlet Temperature	90	^oF
Decay Heat Cooler Heat Transfer Surface Area	3092	Ft^2
Decay Heat Cooler overall HTC	289.4	Btu/hr-ft^2-^oF

Table 6-2 Contempt Model Heat Structure Data

Item	Composition	Surface Area (ft^2)
Shield Building and Containment Vessel Cylinder Wall	9.5 mil paint, 0.125 ft steel, 4.5 ft air, 2.5 ft concrete	75925
Shield Building and Containment Vessel Dome	5 mil paint, 0.8125 in steel, 4.5 ft air, 2.0 ft concrete	26533
Unlined Concrete	8 mil paint, 1.5 ft concrete[1]	87849[2]
Concrete RV Support	8 mils paint, 0.25 ft concrete[1]	1464[2]
Refueling Pool Liner	0.2 in stainless steel, 1.0 ft concrete	12205
Galvanized Steel	15 mil paint, 0.0107 ft steel[1]	61489[2]
Painted Steel < 0.12 Inches Thick	5 mil paint, 0.0065 ft steel[1]	4772[2]
Painted Steel >= 0.12 and < 0.16 Inches Thick	5 mil paint, 0.0108 ft steel[1]	8779[2]
Painted Steel >= 0.16 and < 0.24 Inches Thick	5mil paint, 0.01725 ft steel[1]	20202[2]
Painted Steel >= 0.24 and < 0.30 Inches Thick	5 mil paint, 0.02142 ft steel[1]	14476[2]
Painted Steel >= 0.30 and < 0.40 Inches Thick	5 mil paint, 0.0285 ft steel[1]	12278[2]
Painted Steel ≥0.40 and < 0.50 Inches Thick	5 mil paint, 0.0376 ft steel[1]	5774[2]
Painted Steel ≥0.50 and < 0.625 Inch Thick	5 mil paint, 0.04217 ft steel[1]	24007[2]
Painted Steel ≥0.625 and < 0.75 Inch Thick	5 mil paint, 0.05342 ft steel[1]	6941[2]
Painted Steel ≥0.75 and < 1.0 Inch Thick	5 mil paint, 0.06358 ft steel[1]	7877[2]
Painted Steel ≥1.0 and < 1.5 Inch Thick	5 mil paint, 0.08867 ft steel[1]	3344[2]
Painted Steel ≥1.5 Inch Thick	5 mil paint, 0.2342 ft steel[1]	10156[2]

[1] Symmetric half-thickness. [2] Total, both sides.

Table 6-3 Heat Structure Material Properties

Material	Thermal Conductivity (Btu/hr-ft-°F)	Heat Capacity (Btu/ft^3-°F)
Concrete	0.57	22.3
Carbon steel	29.6	53.6
Stainless steel	8.6	60.1
Paint	1.4	32.0
Air	0.0156	0.01685

7.0 CONCLUSIONS

This report describes a methodology using the GOTHIC computer code to calculate the pressure and temperature response of PWR Dry Containments to postulated primary (LOCA) and secondary (MSLB/MFLB) system breaks. Also included in this methodology description is a process to develop the short-term mass and energy releases used as a boundary condition to GOTHIC. The long-term mass and energy releases are generated within GOTHIC by conservatively accounting for the various mass and energy sources and heat removal mechanisms.

Benchmarks to representative LOCA and MSLB events were performed and it was shown that GOTHIC produced reasonable and consistent results compared to a currently acceptable containment analysis code (CONTEMPT).

The methodology employed by FANP is consistent with industry practices and provides a calculational process that produces conservative results.

8.0 REFERENCE LIST

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APPENDIX A

Comparison to Applicable Standards

This section compares the proposed analysis methodology to NUREG-0800 and ANSI/ANS-56.4 guidelines.

SRP 6.2.1.1.A PWR Dry Containments, Acceptance Criteria		**FANP Methodology**
6.1.a	GDC 16 and 50- The peak calculated containment pressure following a loss-of coolant accident, or a steam or feedwater line break, should be less than the containment design pressure	[ ]
6.1.b	GDC 38 – The containment pressure should be reduced to less than 50% of peak calculated pressure for the design basis loss of coolant accident with 24 hours after the postulated accident	[ ]
6.1.c	GDC 38 – The containment pressure for subatmospheric containments should be reduced to below atmospheric pressure within one hour after the postulated accident, and the subatmospheric condition maintained for at least 30 days.	[ ]
6.1.d	GDC 38 and 50 – For containment response to the loss-of–coolant accident, the analysis should be based on the assumption of loss of off-site power and the most severe single failure in the emergency power system (e.g., a diesel generator failure) the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.	[ ]

SRP 6.2.1.1.A PWR Dry Containments, Acceptance Criteria (cont.)		FANP Methodology
6.1.e	GDC 38 and 50 – For containment response to secondary system pipe ruptures, the analysis should be based on the most severe single failure in the containment heat removal systems (e.g., a fan, pump, or valve failure. The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis.	[ ]

	SRP 6.2.1.3 M&E for LOCA Sources of Energy (II.B.1) [10 CFR Part 50 Appendix K, I.A]	**FANP Methodology**
1	Reactor Power – The reactor should be assumed to have operated continuously at least 1.02 times the licensed power level; however, a lower core power level – no less than licensed power – could be justified.	[ ]
2	Stored Energy in the Core – The steady-state temperature distribution and stored energy in the fuel shall be calculated for the burn-up that yields the highest calculated stored energy.	[ ]
3	Fission Heat - Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivity resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties.	[ ]
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.	[ ]
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 1971 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	Stored Energy in the Reactor Coolant system metal – Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.	[ ]
8	Stored Energy in the Secondary System – Heat transfer between the primary and secondary systems in the SG shall be taken into account.	[
9	Fuel Clad Swelling and Rupture – The prediction of fuel clad swelling and rupture should not be considered.	[ ]

SRP 6.2.1.3 M&E for LOCA Break Size and Location (II.B.2)		FANP Methodology
1	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.	[ ]

SRP 6.2.1.3 M&E for LOCA Blowdown Calculations (II.B.3.b)		FANP Methodology
1	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 existing at 102% of full power.	[ ]
2	Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.	[ ]
3	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.	[ ]
4	Calculations of heat transfer from the secondary coolant to the SG tubes for PWRs should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.	[ ]

SRP 6.2.1.3 M&E for LOCA PWR Core Reflood Calculations (II.B.3.c)		FANP Methodology
1	Following initial blowdown of the RCS, the water remaining in the RV should be assumed to be saturated.	[ ]
2	Justification should be provided for the refill period. An acceptable approach is to assume a water level at the bottom of the active core at the EOB so there is no refill time.	[ ]
3	Calculations of the core flooding rate should be	[

	SRP 6.2.1.3 M&E for LOCA PWR Core Reflood Calculations (II.B.3.c)	FANP Methodology
	based on the ECCS operating condition that maximizes the containment pressure either during the core reflood phase or the post-reflood phase.	]
4	Calculations of liquid entrainment should be based on PWR FLECHT experiments.	[ ]
5	Liquid entrainment should be assumed to continue until the water level in the core is 2ft from the top of the core.	[ ]
6	The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the carryout rate fraction of the increase core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water.	[ ]
7	For Cold Leg Breaks only. Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.	[ ]

	SRP 6.2.1.3 M&E for LOCA PWR Post-Reflood Calculations (II.B.3.d)	FANP Methodology
1	All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.	[ ]
2	Steam quenching should be justified by comparison with applicable experimental data.	[ ]
3	The results of post-reflood analytical models should	[

	be compared to applicable experimental data.	]

SRP 6.2.1.3 M&E for LOCA PWR Decay Heat Phase Calculations (II.B.3.e)		FANP Methodology
1	The dissipation of the core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in Branch Technical Position ASB 9-2 in SRP §9.2.5.	[ ]
2	Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water.	[ ]

SRP 6.2.1.4, M&E for Secondary System Pipe Ruptures, II.1, Sources of Energy		FANP Methodology
1	The stored energy in the affected SG metal, including the vessel tubing, feedwater line, and steam line.	[ ]
2	The stored energy in the water contained within the affected SG.	[.]
3	The stored energy in the feedwater transferred to the affected SG prior to closure of the isolation valve in the FW line.	[ ]
4	The stored energy in the steam from the unaffected SG(s) prior to closure of the isolation valves in the SG crossover lines.	[ ]
5	The energy transferred from the primary coolant to the water in the affected SG during blowdown.	[ ]
6	The SLB should be analyzed for a spectrum of pipe sizes and various plant conditions from hot standby to 102% of full power. Only the 102% power condition need be analyzed provided the applicant can demonstrate that the feedwater flows and fluid inventory are greatest at full power.	[ ]

SRP 6.2.1.4 M&E for Secondary System Pipe Ruptures, II.2, M&E Release Rate Calculations		FANP Methodology
1	Mass release rates should be calculated using the Moody model for saturated conditions, or a model that is demonstrated to be equally conservative.	[ ]
2	Calculations of heat transfer to the water in the affected SG should be based on nucleate boiling heat transfer.	[ ]
3	Calculations of mass release should consider the water in the affected SG and FW line, the FW transferred to the affected SG prior to the closure of the isolation valves in the FW lines, the steam in the affected SG, and the steam coming from the unaffected SG(s) as the secondary system is being depressurized prior to the closure of the isolation valves in the SG crossover lines.	[ ]
4	If liquid entrainment is assumed in the SLB, experimental data should support the predictions of the liquid entrainment model.	[ ]
5	The effect on the entrained liquid of steam separators located upstream from the break should be taken into account.	[ ]
6	A spectrum of SLB should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur, to allow section of the maximum release case.	[ ]
7	If no liquid entrainment is assumed, a spectrum of the SLBs should be analyzed beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered.	[ ]
8	A single active failure in the steam or feedwater line isolation provisions or feedwater pump, such that the containment peak pressure and temperature are maximized, should be assume to occur in steam and feedwater line break analyses. For the assumed failure of a safety grade steam or feedwater line isolation valve, operation of non-safety grade equipment may be relied upon as a backup to the safety grade equipment.	[ ]
9	Operator action to terminate auxiliary feedwater flow will be reviewed by ASB (i.e., must be justified).	[ ]

Conformance to ANSI-ANS-56.4-1983

ANSI-ANS-564.4-1983	FANP Methodology
4.2.1 Postulated Accidents	[ ]
4.2.2 Duration of Analyses	[ ]
4.2.3 Dry Primary Containment Analysis Model	[ ]
4.2.3.1 Thermodynamic State Conditions	[ ]

ANSI-ANS-564.4-1983	FANP Methodology
4.2.3.1.2 Dry Primary Containment Sump Region	[ ]
4.2.3.2 Mass and Energy Transfer Mechanism	[ ]
4.2.3.2.2 Energy Source Terms	[ ]
4.2.3.2.3 Structural Heat Transfer	[ ]
4.2.3.2.4 Dry Primary Containment Spray	[ ]
4.2.3.2.5 Containment Heat Removal Systems (CHRS)	[ ]
4.2.3.2.6 Atmosphere-Sump Interface	[ ]

ANSI-ANS-564.4-1983	FANP Methodology
4.2.3.3 Modeling Consideration	[ ]
4.2.4.1 Net Free Volume	[ ]
4.2.4.2 Heat Sinks	[ ]
4.2.4.3 Primary Containment Pressure	[ ]
4.2.4.4 Primary Containment Atmosphere Temperature.	[ ]
4.2.2.5 Primary Containment Dewpoint (Humidity)	[ ]
4.2.5 Single Failure Criteria	[ ]

APPENDIX B

Mass & Energy Data Used in Sample Calculations

Table B-1 Mass & Energy Addition to Containment Atmosphere for UFSAR Split Hot Leg Break

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
0	0	0	0
0.0005	8196	5035500	614.4
0.005	81960	50355000	614.4
2	81960	50355000	614.4
2.001	58725	35565000	605.6
4	58725	35565000	605.6
4.001	42960	26660000	620.6
6	42960	26660000	620.6
6.001	29055	18770000	646.0
8	29055	18770000	646.0
8.001	15260	11910000	780.5
10	15260	11910000	780.5
10.001	4690	5575000	1188.7
12	4690	5575000	1188.7
12.001	2250	2745000	1220.0
14	2250	2745000	1220.0
14.001	1315	1600000	1216.7
16	1315	1600000	1216.7
16.001	1170	1300000	1111.1
18	1170	1300000	1111.1
18.001	1665	1225000	735.7
20	1665	1225000	735.7
20.001	3421.37	1958784	572.5
25	3421.37	1958784	`572.5
25.001	5146.85	2487512	483.3
30	5146.85	2487512	483.3
30.001	6246.5	2617047	419.0
35	6246.5	2617047	419.0
35.001	810.75	954256	1177.0
40	810.75	954256	1177.0
40.001	824.72	970697	1177.0
50	824.72	970697	1177.0
50.001	580.78	683574	1177.0
60	580.78	683574	1177.0
60.001	236.87	278744	1176.8
80	236.87	278744	1176.8
80.001	175.65	296743	1689.4
100	175.65	296743	1689.4
100.001	163.2	192883	1181.9
120	163.2	192883	1181.9
120.001	166.16	195564	1177.0

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
140	166.16	195564	1177.0
140.001	160.71	189159	1177.0
160	160.71	189159	1177.0
160.001	153.56	180743	1177.0
180	153.56	180743	1177.0
180.001	150.63	177290	1177.0
200	150.63	177290	1177.0
200.001	144.47	170039	1177.0
220	144.47	170039	1177.0
220.001	144.23	168754	1170.0
240	144.23	168754	1170.0
240.001	142.16	167322	1177.0
260	142.16	167322	1177.0
260.001	138.96	163559	1177.0
280	138.96	163559	1177.0
280.001	136.3	160427	1177.0
300	136.3	160427	1177.0
300.001	132.18	155575	1177.0
350	132.18	155575	1177.0
350.001	124.34	146343	1177.0
400	124.34	146343	1177.0
400.001	102.26	120359	1177.0
450	102.26	120359	1177.0
450.001	91.91	108177	1177.0
500	91.91	108177	1177.0
500.001	85.05	100105	1177.0
600	85.05	100105	1177.0
600.001	78.08	91894	1176.9
700	78.08	91894	1176.9
700.001	72.8	85689	1177.0
800	72.8	85689	1177.0
800.001	69.95	82335	1177.0
900	69.95	82335	1177.0
900.001	64.72	76172	1177.0
1000	64.72	76172	1177.0

Table B-2 Liquid Mass & Energy Addition to Containment Sump for UFSAR Split Hot Leg Break

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Table B-3 Mass & Energy Addition to Containment Atmosphere for Cold Leg (Pump Suction) Break Case

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Table B-4 Liquid Mass & Energy Addition to Containment Sump for Cold Leg (Pump Suction) Break Case

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Table B-5 M&E Addition for FANP MSLB Case

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Time (sec)	Flow (lb/s)	Energy (Btu/s)	Enthalpy (Btu/lb) = Energy / Flow
[			]

Table B-6 Long Term Decay Heat Forcing Function

Time (sec)	1.2ANS71 Plus Actinides	ASB 9-2 Multipliers (1.2 for t $\leq$ 1000 sec, 1.1 for t>1000sec)	Core Decay Heat (Based on 2966 MWt * 1.02) (BTU/sec)
[			]

Table B-7 Nitrogen Mass and Energy Release for Cold Leg (Pump Suction) Break Case (Long Term Sample Calculation)

Time (sec)	Flow (lbm/sec)	Energy (BTU/sec)	Enthalpy (BTU/lbm) = Energy/Flow
[			]

Time (sec)	Flow (lbm/sec)	Energy (BTU/sec)	Enthalpy (BTU/lbm) = Energy/Flow
[			]

Time (sec)	Flow (lbm/sec)	Energy (BTU/sec)	Enthalpy (BTU/lbm) = Energy/Flow
[			]

Time (sec)	Flow (lbm/sec)	Energy (BTU/sec)	Enthalpy (BTU/lbm) = Energy/Flow
[			]