Westinghouse Non-Proprietary Class 3

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Addendum 1
Revision 0

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Westinghouse Containment Analysis Methodology

Westinghouse

WCAP-16608-NP
Addendum 1
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Westinghouse Containment Analysis Methodology Addendum 1 Appendix C PWR LOCA Mass and Energy Release Input Calculation Methodology

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APPENDIX C
PWR LOCA MASS AND ENERGY RELEASE INPUT CALCULATION METHODOLOGY

WCAP-16608 describes the Westinghouse containment analysis methodology. The GOTHIC generic BWR Mark I containment model is documented in Appendix A and the BWR mass and energy release input calculation methodology is documented in Appendix B. Addendum 1 (Appendix C) to WCAP-16608 describes the PWR LOCA mass and energy release input calculation methodology.

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NOMENCLATURE

A	Area, Interfacial area for a single droplet
Cp	Specific heat
D	Diameter
DH	Hydraulic diameter
G	Mass flux
g	Gravitational constant
Hfg	Latent heat of vaporization
h	Heat transfer coefficient
k	Conductivity
Nu	Nusselt number
N	Number of droplets
P	Pressure
Pcrit	Critical pressure
Pr	Prandtl number
Re	Reynolds number
T	Temperature
Vr	Relative velocity
X	Quality
α	Void fraction
σ	Surface tension
μ	Viscosity
π	3.14159
ρ	Density

Subscripts

h	Homogeneous
liq	Liquid
l	Liquid
Sat	Saturation
vap	Vapor
v	Vapor

C.1 INTRODUCTION

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

Traditionally, a LOCA event has been described in four phases: blowdown, refill, reflood, and post-reflood. Sometimes a fifth phase, long-term decay heat removal, is described after the post-reflood phase. The blowdown phase starts when the break occurs and ends when the RCS pressure has equilibrated with the containment pressure. The only source of makeup water to the RCS during this phase is passive injection from the pressurized accumulator water tanks. During the refill phase, which begins just after blowdown, water from these tanks helps to partially refill the vessel prior to actuation of the active safety injection system. The reflood phase begins after the vessel water level reaches the bottom of the active fuel and continues until the core is quenched. The post-reflood phase starts after the core is quenched and continues until the remaining RCS and SG stored energy is released. If the break is located upstream of a steam generator (in a hot leg), the frothy two-phase mixture from the core will exit directly to containment during the early part of the post-reflood phase. Later, after the RCS metal has cooled down and the core decay heat rate decreases, the core will stop boiling and hot water will be released to the containment. If the break is located downstream of a steam generator (in a cold leg or pump suction leg), part of the frothy two-phase mixture from the core will be forced into the broken loop SG tubes during the early part of the post-reflood phase. Energy from the hot SG secondary fluid and metal will be transferred to the froth causing it to become all steam. The steam exiting the steam generator outlet plenum will initially be super-heated but, as the steam generator secondary fluid cools down from the bottom up, a two-phase mixture will begin to exit the outlet plenum. If the break is in the pump suction leg, the safety injection flow to that cold leg will mix with the steam and water coming from the intact loops and spill out the pump side of the break. If the break is in the cold leg, the safety injection line to that cold leg is assumed to be broken and spilling to containment; the steam and water coming from the intact loops will exit the vessel side of the cold leg.

Current Westinghouse-Pittsburgh LOCA M&E Release Methodology

The current Westinghouse LOCA M&E release model methodology is documented in References C-1 and C-2. The model uses a series of three codes to calculate the mass and energy release input for the containment analysis. SATAN-VI (Reference C-3) performs the blowdown phase M&E release calculations. SATAN-VI models the RCS thermal-hydraulic response with a somewhat detailed 1-D nodal network containing one lumped loop (to represent the intact loops) and one broken loop. WREFLOOD (Reference C-4) covers the reflood phase of the LOCA event and performs the M&E release calculations from the end of blowdown to the time the broken loop SG pressure has equilibrated with the containment design pressure. WREFLOOD uses a simple flow resistance model to represent the RCS and calculates the heat transfer from the fuel as the core quenches. The FROTH code (Reference C-2) calculates the heat transfer from the RCS metal and steam generators to the frothy two-phase mixture that exits the core during the post-reflood phase of the event. The WREFLOOD and FROTH codes have been updated and combined to create the REFLOOD10325 code (Reference C-2). A third code, EPITOME, combines the output M&E data files from SATAN-VI and REFLOOD10325 and adjusts the break releases generated during the post-reflood phase of the transient to depressurize all of the steam generators to 14.7 psia at one hour. EPITOME calculates a conservative long-term steaming rate, based on the core decay heat rate, for the rest of the analysis, which is at least 24 hours after event initiation. All of the boil-off is assumed to exit out of the broken loop during the long-term steaming period.

Several simplifying assumptions were made while developing the current LOCA M&E release calculation methodology. These assumptions, which are listed below, were found to yield a conservative calculation of the containment pressure response.

1. The containment is assumed to remain at design pressure during the blowdown phase; the containment backpressure input value can be adjusted during the reflood, post-reflood and long-term steaming phases.

2. The vessel is assumed to be refilled to the bottom of the active fuel at the end of the blowdown phase. This eliminates the calculation of the refill phase. Neglecting the refill phase eliminates the period of reduced break flow that would occur between the end of the blowdown and start of the reflood phase.

3. All of the post-blowdown RCS fluid, metal, and SG energy are assumed to be released to the containment within one hour after event initiation (i.e., the RCS and steam generators are assumed to depressurize to saturated conditions at 14.7 psia within 1 hour). There were several reasons for using this non-mechanistic method to calculate the SG and metal energy release rates. First, at the time the code was written, scalable test data for determining the heat transfer rate from the hot SG to the cooler two-phase RCS mixture was not yet available. Second, since the computer systems memory and processor speeds were not as advanced as they are today, the amount of thermal-hydraulic detail that could be put into the code (e.g., modeling conduction-limited heat conductors) was restricted. Third, since the sub-atmospheric containment design is required to be depressurized to atmospheric pressure within one hour of event initiation (Reference C-5), it was determined that using the one hour time frame to remove all of the remaining RCS metal and SG energy would produce a conservative upper bound containment pressure response for evaluating the design of the sub-atmospheric containment pressure suppression system and the large dry containment design as well.

4. The flow split between the broken and un-broken RCS loops in the post-reflood calculation is assumed to be constant. The selected flow split maximizes the steam release to containment by reducing the amount of condensation via steam/water mixing in the intact loop(s).

This conservative LOCA M&E release calculation methodology has been applied in the design basis accident (DBA) analyses for all Westinghouse containment designs and this method will continue to be used, if requested by our customers.

Current Westinghouse-Windsor (CE/ABB) Methodology

The CE/ABB LOCA M&E release model methodology is documented in References C-6, C-7, and C-8. The model uses a series of three codes to calculate the mass and energy release input for the containment analysis. CEFLASH-4A (Reference C-6) performs the blowdown phase M&E release calculations. CEFLASH-4A models the RCS thermal-hydraulic response with a somewhat detailed 1-D nodal network containing the two hot legs, the two steam generators, and the four cold legs. FLOOD3 (Reference C-7) covers the reflood and post-reflood phases of the LOCA event and performs the M&E release calculations from the end of blowdown to the time the broken loop SG pressure has equilibrated with the containment design pressure. FLOOD3 uses a simple flow resistance model to represent the RCS and calculates the heat transfer from the fuel as the core quenches. A third code, CONTRANS (Reference C-8), is used to calculate the long term boil-off and/or cooldown.

The simplifying assumptions noted above are implemented as follows for the current CE/ABB LOCA M&E release calculation methodology. These assumptions were found to yield a conservative calculation of the containment pressure response.

1. The containment pressure is calculated to increase during the blowdown phase, but is kept constant at slightly below the containment design pressure during the reflood and post-reflood phases.

2. The vessel is assumed to be refilled to the bottom of the active fuel at the end of the blowdown phase.

3. The long term boil-off and/or cooldown is calculated coincident with the containment response.

4. The reflood and post-reflood core exit flow split between the broken and un-broken RCS loops is calculated dynamically using hydraulic resistances in the RCS loops.

This conservative LOCA M&E release calculation methodology has been applied in the DBA analyses for all CE/ABB containment designs and this method will continue to be used, if requested by our customers.

Several developments have occurred since the time the current LOCA M&E release methodology was approved. First, the energy transfer from the hot steam generator secondary fluid to a cooler two-phase mixture flowing through the SG tubes was measured under representative large-LOCA, post-blowdown conditions in the FLECHT-SEASET tests (Reference C-9). The two-phase mixtures, at various flow rates and void fractions, were forced into the SG test assembly to measure the transient heat transfer rates and fluid temperature distribution. The test data demonstrated that the SG quenched from the bottom up and that a complete SG cool down could take considerably more than one hour. Second, the computer processor speeds and memory have increased; this now allows the conduction limited heat transfer from the thick metal in the RCS vessel, piping, and SG inlet/outlet plenums to be modeled. This conduction limited thick metal takes considerably longer than one hour to cool down. Finally, proposed power upratings and limitations in maintenance and operations have increased the need to obtain analysis margin for the containment.

Westinghouse has developed an improved LOCA M&E release calculation methodology, which is described in the sections that follow. This new methodology takes advantage of more realistic modeling capabilities and eliminates the need for some of the simplifying assumptions listed above. Westinghouse intends to offer this new method to its customers after it has been reviewed and approved by the NRC.

C.2 WC/T CODE UPDATES FOR LOCA M&E

C.2.1 Overview of Code Modifications

The approved PWR ECCS evaluation model (Reference C-10) uses the WCOBRA/TRAC (WC/T) code to calculate the RCS thermal-hydraulic response to a pipe rupture. The use of the code and model for these applications has been qualified by comparison with scalable test data covering the expected range of conditions and important phenomena. Therefore, when the input is properly biased, and the options are properly selected, the WC/T ECCS evaluation model can be used to produce the mass and energy release input data for the containment response calculations.

Comparison to experimental data with the WC/T ECCS evaluation model shows that the heat transfer model over-predicts the SG reverse heat transfer. While this is conservative, a more realistic SG heat transfer model will improve the M&E release calculation.

In addition, the WC/T ECCS evaluation model does not represent the wall heat mass of the SG secondary side. The wall energy of the SG secondary side needs to be included for the M&E release calculation.

The containment response for the M&E calculation is done with the GOTHIC code (References C-24 through C-26). In order to calculate the RCS thermal-hydraulics with WC/T and the containment calculations with GOTHIC, WC/T needs to be modified to allow running the code in parallel with GOTHIC.

C.2.2 Steam Generator Interface Heat/Mass Transfer Changes

The simulation of some of the experimental runs of the FLECHT-SEASET Steam Generator Separate Effects Tests (Reference C-9) showed significant differences between the WC/T calculations and the FLECHT-SEASET test data. First, WC/T over-predicted the heat transfer from the secondary side for both high and low quality simulations. Second, WC/T did not calculate a marked temperature stratification seen in the experiments (see Figures 4 – 11 in Reference C-11).

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Model Bases – Saturated Droplet Flow

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]^{a,c} According to several authors (References C-12, C-13, and C-14), a reduction of the interfacial heat transfer coefficient between the droplets and the steam has been observed when the vapor is superheated. This is believed to occur because, for high evaporation rates, the vapor mass flux leaving the surface of the droplet boundary-layer act as a layer decreasing the overall heat transfer rate to the droplet by sort of a "shielding" effect.

Webb and Chen (Reference C-15) proposed a model to account for the vapor generation rate in case of superheated vapor in non-equilibrium conditions. The correlation is based on the two-region hypothesis. This hypothesis is that the vapor generation in the post-critical heat flux region is comprised of two mechanisms:

- A near-field evaporation term to model the active evaporation caused by liquid sputtering of the heated wall in the vicinity of the CHF point.
- A far-fielded evaporation of entrained droplets by heat transfer from the superheated vapor. The near-field term is dominant near the CHF point. The far-field term is important further downstream.

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Model as Coded

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Scaling Considerations

The interfacial heat transfer correlation for the dispersed flow regime is verified through its use in the simulation of the FLECHT-SEASET steam generator tests described later in this section.

Subcooled Dispersed Droplet Flow

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Scaling Considerations

The interfacial heat transfer correlation for the subcooled dispersed droplet flow is verified through its use in the simulation of the full height FLECHT-SEASET steam generator tests described later in this section.

Quench Front Simulation Model Basis

The SG FLECHT-SEASET experimental test results showed the appearance of a quench front inside the primary side tubes (see Test Data curves in Figures C.2.2-2 through C.2.2-8). The dispersed two-phase flow above the quench front provided enough heat transfer and precursory wall cooling so that a quench front advanced up the tubes with time. The abrupt drop in the temperature at certain time was the proof of an active heat transfer process inside the tubes, and the axial stratification of the secondary side liquid temperature was its result.

The WC/T ECCS evaluation model code version was used to simulate these tests. The high quality test simulations (#22701, #23402 and #22503) did not predict the quench front phenomenon during the entire transient. The wall temperature profiles (WC/T Standard curves) in Figures C.2.2-2 through C.2.2-4, suggest that the code was modeling a physically different heat transfer process in the primary side. Only for the low quality runs (#21806, #22314 and #21909) was the code able to simulate such phenomenon, but the timing and the rate of wall temperature cooling was off (Figures C.2.2-5 through C.2.2-7).

According to Collier (Reference C-21, page 135), the heat transfer process in high quality flows is modeled by two main heat transfer regimes separated by the dryout phenomenon. The point of the dryout is also called the quench front. Downstream of the quench front, and before the dry saturated vapor region, there is a region characterized by a thin liquid film wetting the tube walls. The thickness of this film is often such that the effective thermal conductivity is able to prevent the liquid in contact with the wall from being superheated to a temperature which would allow bubble nucleation. The heat transfer process can no longer be called nucleate boiling because nucleation is suppressed. This region is called the two-phase forced convective region. According to Figure 4.14 in Reference C-21, the heat transfer coefficient in this region raises as the film becomes thinner and, when the dryout occurs, there is an abrupt reduction in the value of this parameter.

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Comparison to FLECHT-SEASET Steam Generator Tests

The FLECHT-SEASET Steam Generator Separate Effect tests (Reference C-9) were conducted in 1982. The test facility consisted of a full height U-tube steam generator, boiler, accumulator, and containment tank. The boiler and accumulator supply steam and water to a mixing chamber to generate a two-phase flow regime to supply the steam generator.

These experiments were conducted using high quality two-phase flows. Steam is the continuous phase with liquid dispersed within the steam flow. The two-phase flow in the steam generator hot leg and inlet plenum was generated by spraying liquid into passing steam.

The steam generator tube height and dimensions are typical of Westinghouse series 51 steam generators. A total of 32 of 33 U-tubes were used.

The WC/T simulation model noding structure used to represent the FLECHT-SEASET tests is shown in Figure C.2.2-1. [

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The FLECHT-SEASET test cases listed in Table C.2.2-1 are seven test cases chosen for the comparison. Initial conditions like initial RCS temperature, initial steam generator (SG) pressure and temperature, liquid and vapor mass flow rate, and average inlet quality are listed. Test 22701 was selected as the reference case, test 23402 was a sensitivity to the flow rate (2X increase), test 22503 was a sensitivity to the RCS pressure (2X decrease), and tests 22920, 22314, 21806, and 21909 were sensitivities to the flow quality (1.0 through 0.1). The test data in Reference C-9 shows that increasing the flow rate (23402) or reducing the quality (22314, 21806, and 21909) causes the steam generator secondary side to cooldown faster.

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]^{a,c} Therefore, the initiation and subsequent execution of the WC/T simulation is consistent with the FLECHT test procedure.

The results in Figures C.2.2-2 through C.2.2-8 show a comparison of FLECHT-SEASET test data against results calculated with the WC/T ECCS evaluation model code version (curves identified as WC/T Standard) and the modified M&E version of WC/T (curves identified as WC/T M&E Model).

The results show a marked improvement in the calculation of the steam generator outlet temperature and the calculation of the quench front. All cases underpredict the timing of the quench front, which is conservative for M&E calculations, because this overpredicts the energy removal rate from the SG secondary side. This is supported by the results in Figures C.2.2-9 through C.2.2-15, which show that the SG secondary side temperatures calculated by the modified version of WC/T are always lower than the test data values.

Table C.2.2-1 FLECHT-SEASET Steam Generator Tests Initial Conditions							
	R22701	**R23402**	**R22503**	**R22314**	**R21806**	**R21909**	**R22920**
Initial Pressure (kPa – abs)	290.9	331.9	166.9	290.9	297.9	304.9	294.2
Initial RCS Temperature (K)	406	410	388	406	406	407	406
Initial RCS Void Fraction	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Initial SG Pressure (MPa – abs)	5.86	5.86	5.86	5.86	5.86	5.86	5.86
Steam Flow Rate (kg/s)	0.179	0.358	0.178	0.112	0.045	0.045	0.225
Steam Temperature (K)	428	436	427	428	421	427	430
Steam Pressure (kPa – abs)	290.9	331.9	166.9	290.9	297.9	304.9	294.2
Water Flow Rate (kg/s)	0.045	0.090	0.045	0.114	0.181	0.384	0.0
Water Temperature (K)	395	399	375	400	401	402	N/A
Water Pressure (kPa – abs)	290.9	331.9	166.9	290.9	297.9	304.9	N/A
Outlet Pressure (kPa – abs)	269.9	269.9	131.9	269.9	269.9	269.9	272.1
Avg. Inlet Quality	0.8	0.8	0.8	0.5	0.2	0.1	1.0

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Figure C.2.2-1 WC/T Simulation Model Noding Structure for SG FLECHT-SEASET Tests

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Figure C.2.2-2 FLECHT-SEASET Test R22701 WC/T Plot Comparisons

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Figure C.2.2-3 FLECHT-SEASET Test R23402 WC/T Plot Comparisons

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Figure C.2.2-4 FLECHT-SEASET Test R22503 WC/T Plot Comparisons

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Figure C.2.2-5 FLECHT-SEASET Test R22314 WC/T Plot Comparisons

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Figure C.2.2-6 FLECHT-SEASET Test R21806 WC/T Plot Comparisons

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Figure C.2.2-7 FLECHT-SEASET Test R21909 WC/T Plot Comparisons

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Figure C.2.2-8 FLECHT-SEASET Test R22920 WC/T Plot Comparisons

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Figure C.2.2-9 R22701 Steam Generator Secondary Fluid Temperatures

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Figure C.2.2-10 R23402 Steam Generator Secondary Fluid Temperatures

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Figure C.2.2-11 R22503 Steam Generator Secondary Fluid Temperatures

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Figure C.2.2-12 R22314 Steam Generator Secondary Fluid Temperatures

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Figure C.2.2-13 R21806 Steam Generator Secondary Fluid Temperatures

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Figure C.2.2-14 R21909 Steam Generator Secondary Fluid Temperatures

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Figure C.2.2-15 R22920 Steam Generator Secondary Fluid Temperatures

C.2.3 Steam Generator Wall Heat Transfer Changes

The STGEN component of the WC/T ECCS Evaluation Model code version does not represent the metal wall of the SG inlet and outlet plenum, or the metal wall of the secondary side shell. For mass and energy release calculations it is important to represent the metal mass of the steam generator inlet and outlet plenum and secondary side shell.

Model Basis

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C.2.4 WCOBRA/TRAC Running in Parallel with GOTHIC

The containment pressure, temperature, and sump temperature response during a LOCA are dependent on the mass and energy releases. The LOCA mass and energy releases on the other hand are dependent on the containment pressure and on the sump temperature when the RHR heat exchanger is in operation. Inter-process communication is available in GOTHIC by specifying read/write run-time from and to specified data files. WC/T was modified to incorporate in the code the read/write run-time files capability consistent with GOTHIC which allows WC/T to run in parallel with GOTHIC.

Code Implementation

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Code Validation

The correctness of the transfer of interfaces between WC/T and GOTHIC is validated by plotting the interface variables from the WC/T and GOTHIC sides. The results coincide identically.

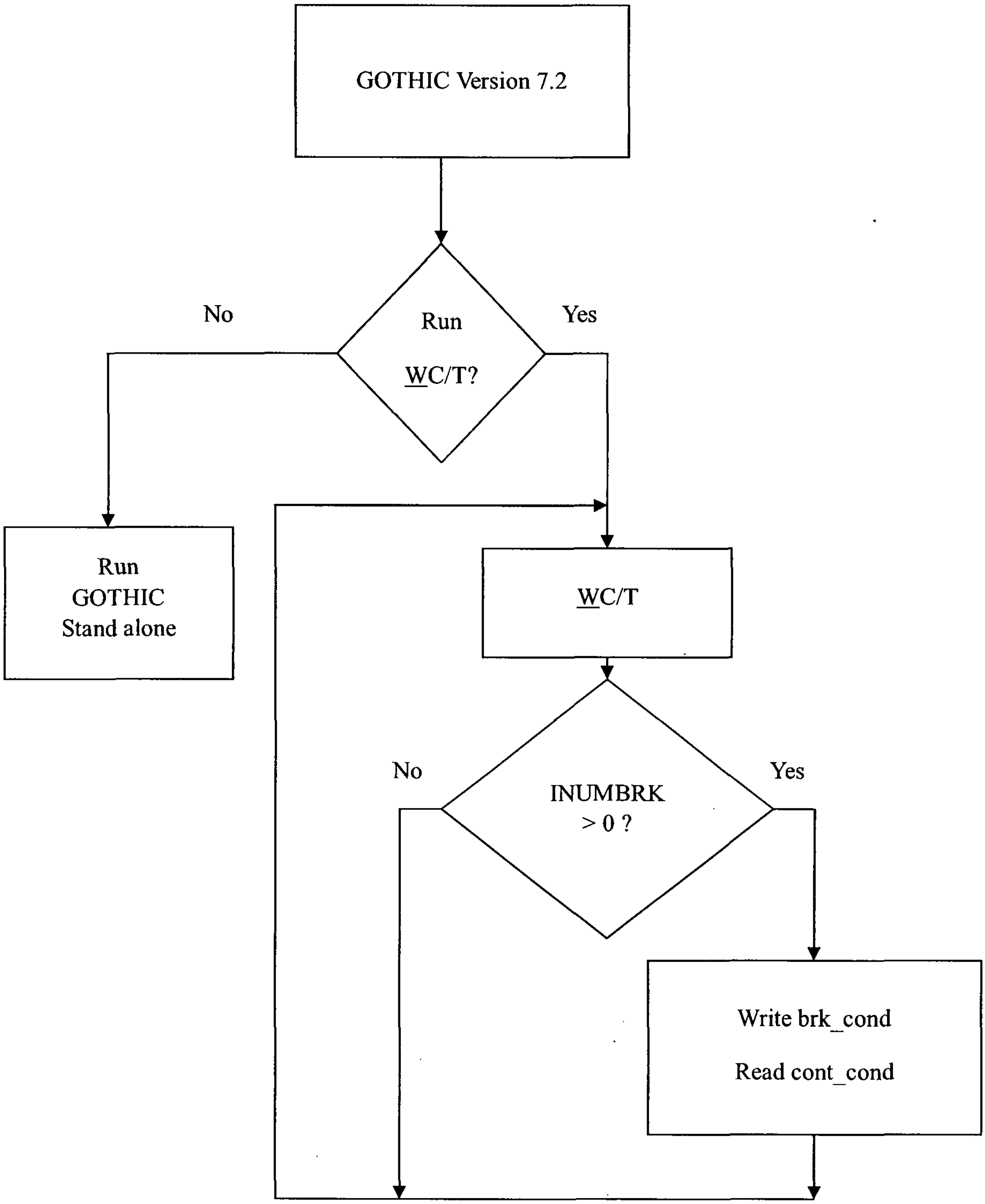

Figure C.2.4-1 Schematic of the GOTHIC – WC/T Execution Control

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Figure C.2.4-2 Schematic of the GOTHIC – WC/T Parallel Execution

C.3 INPUT BIASING FOR THE CONTAINMENT DBA ANALYSES

As described in Section C.1, several simplifying assumptions were made during the development of the currently approved LOCA M&E release methodology. We intend to remove some of these assumptions in the proposed new LOCA M&E release methodology as described below.

1. It is not necessary to assume the containment backpressure remains at a constant value during blowdown or try to define conservative containment backpressure input values during the reflood and post-reflood phases. The new WC/T LOCA M&E release model will be coupled with a GOTHIC containment model to calculate the containment response into the post-reflood phase of the event. The SG fluid, metal, and RCS metal energy remaining at the end of the coupled WC/T+GOTHIC calculation will be released along with the decay heat in the long-term GOTHIC calculation.

2. The assumption that the vessel is refilled to the bottom of the fuel at the end of the blowdown phase (just prior to reflood) is un-realistic. The new WC/T LOCA M&E release model will calculate the refill transient response.

3. The assumption that all the remaining post-blowdown energy in the metal and steam generators can be released to the containment within one hour is overly conservative. Now, with the advent of faster computers with more memory, the current non-mechanistic LOCA M&E model can be replaced with a more advanced model that includes an improved calculation of heat transfer from the RCS metal and steam generators into the post-reflood phase of the event.

4. It is not necessary to assume or force a fixed flow split between the broken and intact loops during the post-reflood phase. The new WC/T LOCA M&E release model will calculate the flow split based on the loop hydraulic resistances.

The proposed LOCA M&E release methodology was developed in a series of steps. In the first step of the process, a phenomena identification and ranking table (PIRT) was developed to identify the important phenomena that need to be considered in the calculation (Reference C-23). For example, the PIRT identified SG heat transfer as one of the highly ranked phenomena that is modeled non-mechanistically in the current LOCA M&E release methodology. Next, an appropriate code, WCOBRA/TRAC (WC/T), was selected for the LOCA M&E release model. The Westinghouse best-estimate LOCA ECCS evaluation model uses the WC/T code (Reference C-10) to calculate the RCS thermal-hydraulic response to a large pipe rupture. The LOCA ECCS evaluation model PIRT is very similar to the LOCA M&E release model PIRT, so WC/T already contains models for most of the important M&E phenomena identified in the PIRT. The code and model have been qualified for large pipe rupture analyses by comparison with scalable test data covering the expected range of conditions and important phenomena.

The LOCA ECCS evaluation model was modified to address the remaining LOCA M&E PIRT items (modeling reverse SG heat transfer and coupling with a containment model for the reflood and post-reflood phases). The modified WC/T code was validated by comparison with SG test data from FLECHT. Finally, the calculated transient response from the proposed LOCA M&E release methodology using the modified WC/T code and model was compared with the calculated transient response from the current LOCA M&E release methodology.

The PWR mass and energy release model input for the containment design basis accident analyses is biased to maximize the initial mass and energy stored in the RCS and to calculate a conservatively rapid release rate. NUREG-0800, Section 6.2.1.3 documents an acceptable practice for the calculation of the LOCA mass and energy release input data. The SRP specifies that the sources of energy available for release are to be based on 10 CFR Part 50, Appendix K, paragraph I.A. A comparison of the proposed Westinghouse methodology to the requirements given in NUREG-0800, Section 6.2.1.3 is shown in Table C.3-1. ANS 56.4-1983 also provides guidance for developing conservative input for the mass and energy release calculation in accordance with the acceptable practice documented in NUREG-0800, Section 6.2.1.3. A comparison of the proposed Westinghouse methodology to the recommendations in ANS 56.4-1983 is shown in Table C.3-2.

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Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements			
	Sources of Energy, 10 CFR 50, Appendix K, I.A	**Current Westinghouse Methodology**	**New Westinghouse Methodology**
1	*Reactor Power* – The reactor should be assumed to have been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.		
2	*Core Stored Energy* – The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.)		
3	*Fission Heat* – Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.		

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
4	*Decay of Actinides* – The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.		
5	*Fission Product Decay* – The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.		
6	*Metal-Water Reaction Rate* – The rate of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just equation. The reaction shall be assumed not to be steam limited.		

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Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

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	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
7	*Reactor Internals Heat Transfer* – Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.		
8	*Fuel Rod Swelling and Rupture* – The calculation of fuel rod swelling and rupture should not be considered for M&E calculations		
9	*Break Size and Location* – Containment design basis calculations should be performed for a spectrum of possible pipe breaks, sizes, and locations to assure that the worst case has been identified.		

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

a,c

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
10	*Calculations, Sub-compartment Analysis* – The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.		
11	*Calculations, Initial Blowdown Phase* – The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 102% times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.		
12	*Calculations, Initial Blowdown Phase* – Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.		

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
13	*Calculations, Initial Blowdown Phase* – Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.		
14	*Calculations, Initial Blowdown Phase* – Calculations of heat transfer from the secondary coolant to the steam generator tubes should be based on natural convection for tubes immersed in water and condensing heat transfer for tubes exposed to steam.		
15	*Calculations, Core Reflood Phase* (cold leg breaks only) – The water remaining in the vessel should be assumed to be saturated. Justification should be provided for the refill period, which is the time from the end of blowdown to the time when the emergency core cooling system (ECCS) refills the vessel lower plenum. An acceptable approach is to assume a water level at the bottom of the active core at the end of blowdown so there is no refill time.		

a,c

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
16	*Calculations, Core Reflood Phase* (cold leg breaks only) – The flooding rate should be based on the ECCS operating condition from the beginning of flooding the core until the time that the core is completely quenched. The carryout fraction should be based on the FLECHT emergency core heat transfer experiments and liquid entrainment should occur until the water level is 2 feet from the top of the core. The carryout rate fraction that is acceptable is 0.05 to the 18 inch level and linearly increasing to 0.80 at the 24 inch level and held constant at 0.8 until the quench front is 2 feet from the top of the core. Above this level, 0.05 may be used.		
17	*Calculations, Core Reflood Phase* (cold leg breaks only) – The assumption of steam quenching should be justified by comparison to applicable experimental data. Liquid entrainment should consider the effect of the carryout rate fraction of the increased core inlet temperature caused by the steam quenching assumed to occur from mixing with the ECCS water.		

a,c

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
18	*Calculations, Core Reflood Phase* (cold leg breaks only) – The steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.		
19	*Calculations, PWR Post-Reflood Phase* – All remaining energy in the primary and the secondary systems should be removed.		

a,c

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

a,c

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
20	*Calculations, PWR Post-Reflood Phase* – Steam quenching should be justified by comparison with applicable experimental data. The results of post-reflood analytical models should be compared to applicable experimental data.		
21	*Calculations, PWR Decay Heat Phase* – The dissipation of 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.		

Table C.3-1 NUREG-0800, Section 6.2.1.3 M&E for LOCA Requirements (cont.)

a,c

	Sources of Energy, 10 CFR 50, Appendix K, I.A	Current Westinghouse Methodology	New Westinghouse Methodology
22	*Calculations, PWR Decay Heat Phase* – Steam from the decay heat boiling in the core should be assumed to flow to the containment by a path which produces the minimum amount of mixing with the ECCS injection water.		

Table C.3-2 ANS 56.4-1983 Recommendations

Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology
1	*3.2.1.1 Reactor Coolant System Water and Metal* – The increase in the reactor coolant system volume resulting from the pressure and temperature expansion to conditions at the initial power level defined in 3.2.2.2 shall be included. Stored energy in all reactor coolant system pressure boundary and internals metal thermally in contact with the reactor coolant system water shall be included.		
2	*3.2.1.2 Steam Generator Secondary Water and Meta*l – Maximizing the steam generator secondary water inventory and metal energy is conservative. The secondary volume resulting from the pressure and temperature conditions at the initial power level defining 3.2.2.2 shall be included.		
3	*3.2.1.3 Core Stored Energy* – The core stored energy and the steady-state core-temperature distribution, adjusted for uncertainties, shall be consistent with the initial conditions and consistent with the time of fuel cycle life required in 3.2.2.1.		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

	Recommendation	Current Westinghouse Methodology	New Westinghouse Methodology
4	*3.2.1.4 Fission Heat* – Fission heat shall be conservatively calculated. Shutdown reactivities resulting from temperature and voids shall assume minimum plausible values including allowances for uncertainties; all data shall be based on their minimum values consistent with the fuel parameters which yield the maximum core stored energy. Rod trip and insertion may be assumed at the time appropriate for the transient being analyzed.		
5	*3.2.1.5 Decay of Actinides* – The heat from the radioactive decay of actinides, including neptunium and plutonium as well as isotopes of uranium generated during operation, shall be calculated in accordance with fuel cycle calculations and shall be appropriate for the time in the fuel cycle that yields the highest calculated core stored energy. The decay heat shall be the values given in American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979 for end-of-life operation time.		
6	*3.2.1.6 Fission Product Decay* – The heat generation rates from radioactive decay of fission products shall be assumed to be equal to at least the values given in ANSI/ANS-5.1-1979 for end-of-life operation time.		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

	Recommendation	Current Westinghouse Methodology	New Westinghouse Methodology
7	*3.2.1.7 Metal-Water Reaction Rate* – The amount of metal-water reaction shall be calculated according to 10 CFR 50.44 and assumed to occur uniformly over a period less than 2 minutes following the end of reactor vessel blowdown.		
8	*3.2.1.8 Main Steam Lines* – Steam flow to the turbine until the main steam isolation valves or turbine stop valves are calculated to close may be included. Flow to the turbine shall be minimized. Delays and valve closure times shall be conservatively short. In lieu of this calculation, flow to the turbine may be conservatively terminated at break initiation.		
9	*3.2.1.9 Main Feedwater Line* – Main feedwater flow shall be included and shall be maximized. Delays and valve closure times used to determine the termination of flow shall be conservatively long.		
10	*3.2.1.10 Auxiliary Feedwater System* – Auxiliary feedwater flow to the steam generators may be included in the analysis if it can be determined that the system is both available and actuated. Flow rates shall be minimized. Delays in actuating the auxiliary feedwater system shall be conservatively long. Alternatively, auxiliary feedwater (AFW) flow may be conservatively assumed to be zero.		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology
11	*3.2.1.11 ECCS Flow* – Flow from the ECCS shall be included. Flows and delay times shall be chosen in accordance with the single active failure consideration which results in the highest peak primary containment pressure.		
12	*3.2.1.12 Safety Injection Tank Nitrogen Expansion* – Nitrogen release to the primary containment from the safety injection tanks after the tanks have emptied shall be included in the calculation. Core heat transfer shall be included if appropriate.		
13	*3.2.2.1 Time of Life* – The time of life of the core shall be that producing the maximum energy from the combination of core stored energy and decay heat assuming power level as required in 3.2.2.2.		
14	*3.2.2.2 Power Level* – The initial power level shall be at least as high as the licensed power level plus uncertainties such as instrumentation error (typically 102 percent of the licensed power level).		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

	Recommendation	**Current Westinghouse Methodology**	**New Westinghouse Methodology**
15	*3.2.2.3 Core Inlet Temperature* – The initial core inlet temperature shall be the normal operating temperature consistent with the initial power level adjusted upward for uncertainties such as instrumentation error. The uncertainties shall be biased to result in maximizing energy releases through the break for the entire transient.		
16	*3.2.2.4 Reactor Coolant System Pressure* – The initial reactor coolant system pressure shall be at least as high as the normal operating pressure consistent with the initial power level plus uncertainties such as instrumentation error.		
17	*3.2.2.5 Steam Generator Pressure* – The initial steam generator pressure shall be at least as high as the normal operating pressure consistent with the initial power level plus uncertainties such as instrumentation error.		
18	*3.2.2.6 Reactor Coolant System Pressurizer Level* – The initial reactor coolant system pressurizer level shall be at least as high as the maximum normal operating level plus uncertainties such as instrumentation error.		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

a,c

Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology
19	*3.2.2.7 Steam Generator Water Level* – The initial steam generator water level shall be at least as high as the normal operating level consistent with the initial power level plus uncertainties such as instrumentation error.		
20	*3.2.2.8 Core Parameters* – Initial core parameters (including physics parameters, fuel properties, and gas conductivity) shall be chosen to maximize core stored energy.		
21	*3.2.2.9 Safety Injection Tanks* – The initial safety injection tank water level and temperature and nitrogen pressure shall be based on normal operating values. Uncertainties shall be biased in the direction which leads to the maximum primary containment pressure.		

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

a,c

	Recommendation	Current Westinghouse Methodology	New Westinghouse Methodology
22	*3.2.3 Single Active Failures* – In determining the mass and energy releases following a reactor coolant system break, the most restrictive single active failure shall be considered. The possibility that the highest peak primary containment pressure may occur for the situation where no active failure has occurred shall not be overlooked. No more than one single active failure in the safety systems, (including primary containment heat removal system; see 4.2.5) required to mitigate the consequences of the event, need to be considered.		
23	*3.2.3.2 Single Passive Failures* – Passive failures normally need not be considered.		
24	*3.2.3.3 Non-emergency Power* – The loss of non-emergency power shall be postulated if it results in circumstances (for example, delayed primary containment cooling or safety injection) which lead to higher primary containment pressures.		
25	*3.2.4.1 Nodalization* – Geometric nodalization for the various periods of the reactor coolant system break analysis need not be the same. Since low quality at the break node is conservative during blowdown because it leads to high flow rates, the reactor coolant system shall be modeled with sufficient detail so that the quality at the break location shall not be over predicted.		

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology
26	*3.2.4.2 Thermodynamic Conditions* – The thermodynamic state conditions for steam and water shall be described using real gas equations or industry accepted steam table in such a manner that the resultant steam and water temperature and partial steam pressure are within one percent of that which would result from use of the 1967 ASME Steam Tables with appropriate interpolation.		
27	*3.2.4.3 Flow Modeling* – The following effects may be taken into account in the flow modeling: 1) temporal change in momentum, 2) momentum convection, 3) forces due to wall friction, 4) forces due to fluid pressure, 5) forces due to gravity, 6) forces due to geometric head loss effects. If an uncertainty in a pressure loss exists, the pressure loss shall be conservatively minimized.		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)				a,c
Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology	
28	*3.2.4.4 Pump Characteristics* – The characteristics of the reactor coolant system pumps shall be derived from a dynamic model that includes momentum transfer between the fluid and the impeller with variable pump speed as a function of time. The pump model for the subcooled and two-phase region shall be verified by applicable subcooled and two-phase performance data. In lieu of a full dynamic pump model, any model which can be shown to be conservative by comparison with the test data or by comparison with a full dynamic pump model may be used.			
29	*3.2.4.5.1 Break Sizes* – For reactor coolant system analysis, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the reactor coolant system. The break shall be defined by its location, type, and area.			

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

a,c

	Recommendation	Current Westinghouse Methodology	New Westinghouse Methodology
30	*3.2.4.5.2 Break Flow Model* – Empirical critical break flow models developed from test data may be utilized during the periods of applicability, for example, subcooled, saturated, or two-phase critical flow. Acceptable critical break flow models, when the fluid conditions are subcooled immediately upstream of the break, include the Zaloudek and Henry-Fauske models. During the period when fluid conditions immediately upstream of the break are saturated or two-phase, an acceptable model is the Moody critical flow model. The critical break flow correlations may be modified to allow for a smooth transition between subcooled and saturated flow regions. Other critical flow models may be used if justified by analysis or experimental data. The discharge coefficient applied to the critical flow correlation shall be selected to adequately bound experimental data.		
31	*3.2.4.5.3 ECCS Spillage* – In generating mass and energy release source terms from spillage for primary containment peak pressure determination, the quality shall be selected based on the partial pressure of steam in containment to maximize primary containment pressurization. For the determination of the maximum primary containment sump temperature for calculation of available NPSH, assumptions on generating mass and energy release and spillage source terms shall be biased toward maximizing the sump temperature.		

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

	Recommendation	Current Westinghouse Methodology	New Westinghouse Methodology
32	*3.2.4.6.1 PWR Backpressure* – For blowdown period analysis, the primary containment backpressure is unimportant because the break flow is critical virtually throughout the blowdown period. During the reflood and post-reflood periods, the primary containment backpressure affects the resistance to the flow (steam binding) in the reactor coolant loop and, therefore, affects the rate of mass and energy release. The mass and energy releases calculation shall be coupled to the primary containment pressure calculation or a conservatively high backpressure (constant or time dependent function) shall be used.		
33	*3.2.4.7 Heat Transfer Correlations* – Heat transfer correlations shall be based on experimental data or chosen to predict conservatively high primary containment pressure.		

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Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

a,c

Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology
34	*3.2.4.8 Core Modeling* – Fission heat may be calculated using a core averaged point kinetics model which considers delayed neutrons and reactivity feedback. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowances for uncertainties for the range of power distribution shapes and peaking factors which result in the maximum core stored energy. Rod trip and insertion may be assumed if they are calculated to occur. Reactivity effects shall be consistent with the time of life which leads to the maximum core stored energy. For core thermal hydraulic calculations, the core shall be modeled with sufficient detail so as not to under-predict core-to-reactor coolant heat transfer. Initial core stored energy shall be maximized.		
35	*3.2.4.9 Modeling of Metal Walls* – Heat transfer from metal walls to coolant shall be calculated so as not to under-predict the rate of heat transfer relative to experimental data or the solution of the one-dimensional, time dependent heat conduction equation.		

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

	Recommendation	Current Westinghouse Methodology	New Westinghouse Methodology
36	*3.2.4.10 Modeling of Auxiliary Flows* – Flows from the safety injection tanks and safety injection pumps shall be calculated assuming backpressures less than or equal to the actual pressure at the injection point. The flows shall be based on expected pump performance values. Uncertainties shall be biased in such a way as to maximize primary containment pressure. A single active failure shall be included if conservative as discussed in 3.2.3. Flows from the auxiliary feedwater system may be assumed if they are calculated to occur or they may be conservatively omitted. If flows are assumed, they shall be based on expected pump performance values. Uncertainties shall be biased to minimize flow since this is conservative. A single active failure shall be included if conservative as discussed in 3.2.3.		

a,c

Table C.3-2 ANS 56.4-1983 Recommendations (cont.)

Recommendation		Current Westinghouse Methodology	New Westinghouse Methodology
37	*3.2.4.11 Post-blowdown Modeling* – The reflood of the core following blowdown shall be calculated using a gravity-feed model which considers the pressure distribution around the primary loop. Entrainment of reflood water in the core shall be based on carry-out rate fractions based on the FLECHT or other test data. Parameters which determine the carryout rate fractions, such as core inlet temperature, linear heat rate, core pressure, core height, and core inlet velocity shall be modeled in such a way as to maximize the carryout rate fraction. The height of water in the core at which the core is reflooded shall be based on experimental data or the reflood height may be assumed to be two feet below the top of the active core. If credit for condensing of steam by ECCS water is taken, it shall be justified with experimental data.		

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C.3.1 Biasing for Peak Pressure/Temperature

The following changes must be made to a WC/T ECCS evaluation model input deck to bias the LOCA M&E releases for the peak containment pressure and temperature calculation:

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a,c

Figure C.3.1-1 WC/T Steady State Noding Diagram (4-Loop Plant)

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The following is a list of items that are not included in the LOCA M&E release calculation:

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[

]^{a,c}

C.3.2 Biasing for Long-term EQ Application

A suitably biased WC/T ECCS evaluation model is used to calculate the blowdown, refill, reflood, and post-reflood phase M&E release input for the containment response analysis. The WC/T LOCA M&E model must be run until both sides of the break reach saturation, i.e., there is no superheated steam release. The DEPS and DECL cases are typically run out to at least one hour to cover the transfer to sump recirculation. The energy remaining in the RCS metal, the SG fluid, and the SG metal at the end of the WC/T calculation is inventoried and released during the long-term decay heat boil-off calculation.

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C.3.3 Biasing for Minimum NPSHa Application

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C.4 BENCHMARK COMPARISONS

This section compares the DEPS and DEHL LOCA M&E releases calculated with a modified WC/T ECCS evaluation model to benchmark results calculated with the currently approved LOCA M&E release calculation methodology. The containment response comparison is also included.

C.4.1 LOCA M&E Model Description

An existing WC/T 4-loop plant ECCS evaluation model was modified and used for the DEPS and DEHL LOCA benchmark comparison cases. [

][a,c] The steady state loop noding diagram for the modified WC/T ECCS evaluation model is shown in Figure C.3.1-1. The modified WC/T steam generator noding diagram is shown in Figure C.4.1-1.

The accumulator pressure, temperature, and water volume, along with the SI flow rate and temperature were modified to match the SATAN-VI benchmark model. [
][a,c] The initial RCS pressure, pressurizer level, and fluid and metal temperatures were adjusted to match the SATAN-VI benchmark model. [
][a,c] A 60 second steady state case was used to adjust the SG secondary side pressure and steam/feed flow rates to maintain the desired RCS operating conditions.

The initial stored mass and energy from the modified WC/T ECCS evaluation model are compared with the SATAN-VI benchmark model in Table C.4.1-1. The WC/T model has a slightly higher initial RCS fluid mass and energy, but a substantially higher initial SG fluid mass and energy than SATAN-VI. [

][a,c] The WC/T model SG and RCS metal energies are also substantially higher than SATAN-VI. The difference in the RCS metal energy is primarily due to the difference in vessel metal energy between the two models. All of the initial RCS fluid energy and a small part of the RCS metal, SG metal, and SG fluid energy is released during the LOCA blowdown phase. The rest of the RCS metal, SG metal, and SG fluid energy is released later during the post-reflood and long-term decay heat removal phases of the event.

Table C.4.1-1 Initial Steady State Mass and Energy Comparison

	WC/T Model	SATAN-VI Model
RCS Fluid Mass	571,750 lbm	567,400 lbm
RCS Fluid Energy	349 MBtu	341 MBtu
RCS Metal Energy	203 MBtu	169 MBtu
SG Secondary Fluid Mass	608,600 lbm	546,500 lbm
SG Secondary Fluid Energy	347 MBtu	309 MBtu
SG Secondary Metal Energy	142 MBtu	119 MBtu

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Figure C.4.1-1 WC/T Steam Generator Noding Structure for LOCA M&E

C.4.2 Containment Model Description

The containment model input is based on the COCO containment model from the benchmark analysis case. This model represents a PWR large dry containment with a net free volume of 2.76 x 10^6 ft^3. Twenty passive heat sinks are modeled. The active containment heat removal system includes 2 spray pumps, 5 service water cooled fan coolers, and 2 RHR cooling loops; however, only one electrical train of active containment heat removal is assumed to be in operation. This leaves only 1 spray pump, 2 fan coolers, and 1 RHR pump in service. The low-head RHR pump switches from the injection mode to the sump recirculation mode after the RWST reaches the low-2 level setpoint. The spray pump continues to draw from the RWST until the level reaches the low-3 setpoint. After this, the spray pump suction is transferred from the RWST to the sump to provide recirculation spray.

The GOTHIC containment model was developed following a methodology which is based on previously approved topical reports. The containment model noding diagram is shown in Figure C.4.2-1 and the key containment model input is given in Tables C.4.2-1 and C.4.2-2. [

]^{a,c} The fan cooler heat removal rate is input as a function of the containment saturation temperature as shown in Figure C.4.2-2. [

]^{a,c}

The GOTHIC containment model runs concurrently with the WC/T LOCA M&E release model to calculate the containment response during the blowdown, refill, reflood, and post-reflood phase of the LOCA event. The GOTHIC containment model is used to calculate both the M&E releases and containment response for the long-term decay heat removal phase.

Table C.4.2-1 Key Containment Model Input Values	
Description	**GOTHIC Model**
Containment Data	
Noding Structure	Single lumped
Volume	2,758,000 ft^3
Height	100 ft
Pool Area	27,580 ft^2
Heat Sink Geometry – See Table C.4.2-2	
Heat Transfer Coefficients – LOCA	Tagami+Uchida
Initial Conditions	
Initial Pressure	15.7 psia
Initial Temperature	120°F
Initial Humidity	20 %RH
Boundary Conditions	
Break Flow Phase Separation	Liquid released as drops during blowdown phase
Accumulator Nitrogen Release	Modeled for LOCA
Fan Cooler Initiation	29.7 psia with a 60 second delay
Fan Cooler Heat Removal Rate (Btu/s)	See Figure C.4.2-2
Spray Flow Initiation	44.7 psia with a 30 second delay
Spray Flow Rate	359 lbm/s per pump
Spray Flow Termination	Low-3 RWST Level (5,000 ft^3 remaining)
LOCA Sump Recirculation Modeling	
Transfer to ECCS Recirculation	Low-2 RWST Level (24,530 ft^3 remaining)
RHR Flow Rate	1,000 gpm
RHR Heat Exchanger UA (Btu/hr-F)	Code calculated for the HX type using flow area, D_h and HTA
CCW Flow Rate	5,000 gpm
CCW Heat Exchanger UA (Btu/hr-F)	Code calculated for the HX type using flow area, D_h and HTA
Other CCW Heat Loads	6.8 MBtu/hr
Service Water Flow Rate	690.6 lbm/s
Service Water Temperature	1,000°F

Table C.4.2-2 Heat Sink Geometry								
	Area (ft^2)	Sides	Paint (in)	SS Steel (in)	CS Steel (in)	Air (in)	Concrete (in)	Total (in)
Containment Cylinder	72,740	1	0.01		0.2496	0.017	9	9.2766
Containment Dome	17,550	1	0.01		0.2496	0.017	9	9.2766
Unlined Concrete	16,000	0					9	9
SS Lined Concrete	848	1		0.498		0.017	9	9.515
Unlined Concrete	4,803	1					12	12
CS Lined Concrete	7,702	1	0.01		0.9192	0.017	9	9.9462
Painted Steel Lining	422.3	1	0.01		0.75			0.76
Unlined Concrete	69,540	1					9	9
CS Lined Concrete	3,852	1	0.01		0.048	0.017	9	9.075
CS Lined Concrete	1,571	1	0.01		0.852	0.017	9	9.879
SS Lined Concrete	2,129	1		0.828		0.017	9	9.845
Misc. Steel Plate	19,790	1	0.01		0.5			0.51
Misc. Steel Plate	94,670	1	0.01		0.25			0.26
Polar Crane	14,090	1	0.01		0.912			0.922
Misc. Steel Plate	21,880	1	0.01		0.48			0.49
Misc. Steel Plate	22,530	1	0.01		0.18			0.19
Cable/Conduit Trays	27,095	1	0.01		0.125			0.135
Supports	6,385	1	0.01		0.098			0.108
Misc. Steel Plate	69,860	1	0.01		0.188			0.198
Lined Concrete	9,291	1		0.198		0.017	9	9.215

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Figure C.4.2-1 GOTHIC Containment Model Noding Diagram

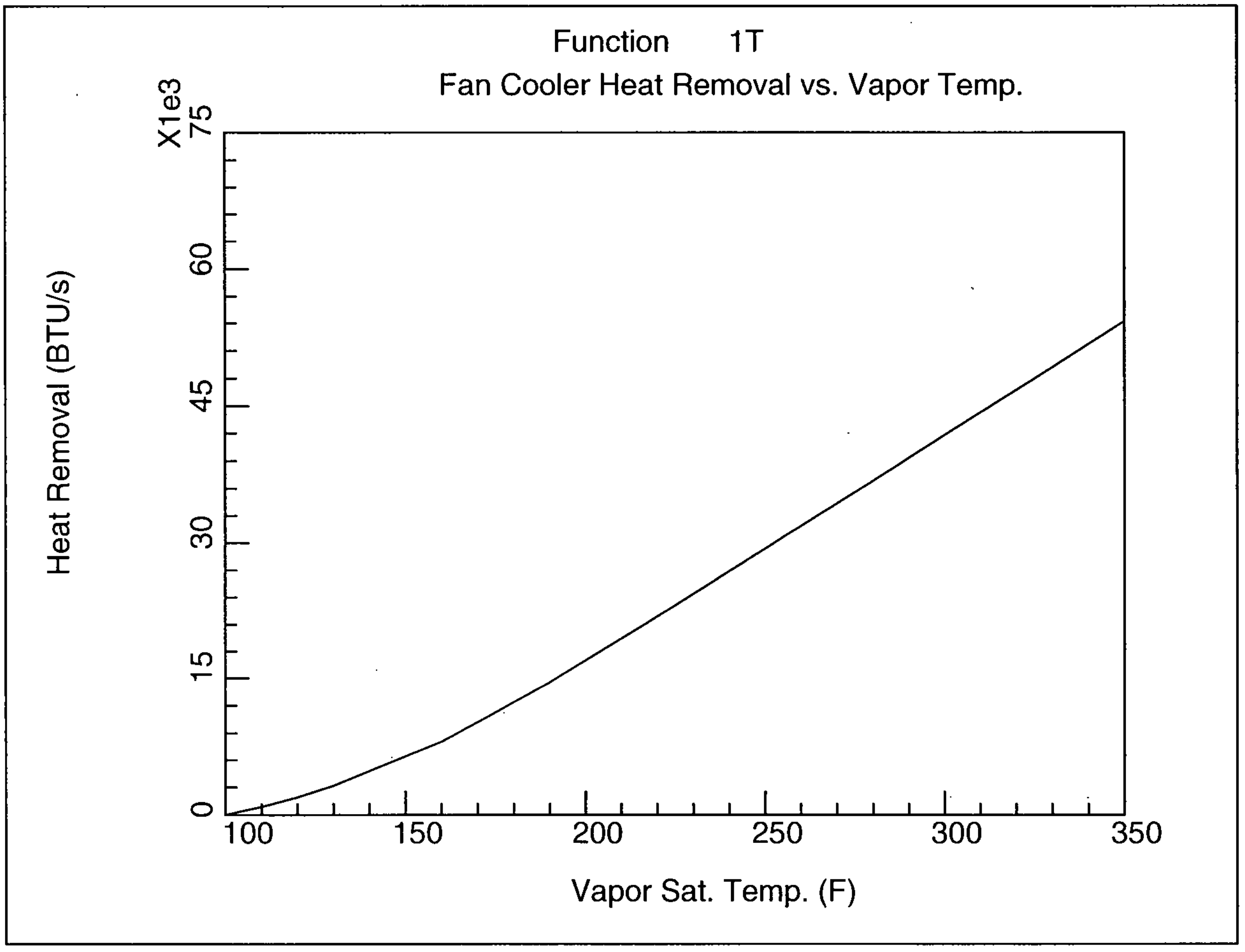

Figure C.4.2-2 Fan Cooler Heat Removal Curve

C.4.3 DEPS LOCA Benchmark Case Results Comparison

The DEPS break is located in the pressurizer loop in both the WC/T and SATAN-VI models. [

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The WC/T DEPS LOCA case was run for at least 2,500 seconds to allow the M&E release and containment response results to be compared with the WCAP-10325 (Reference C-2) benchmark case through sump recirculation. The integrated blowdown break mass and energy release comparison is shown in Figures C.4.3-1 and C.4.3-2. The WC/T model calculates a similar blowdown break mass and energy release. The integrated long-term mass and energy release comparison is shown in Figure C.4.3-3 and C.4.3-4. The integrated long-term mass release comparison shows a difference starting at about 1,100 seconds because the benchmark model simulates a transfer to recirculation at that time; recirculation did not start until later (about 1,400 seconds) in the WC/T model. The WC/T model calculates a lower long-term break energy release than the benchmark model. The lower long-term break energy release rate is due to the improved modeling of the SG quench and RCS metal heat removal in the WC/T model. The impact of the lower metal and SG energy release rates on the GOTHIC calculated containment pressure and temperature is shown in Figures C.4.3-5 through C.4.3-10. The blowdown peak pressure and temperature are about the same since the energy release rate is the same, but because the WC/T long-term energy release rate is much lower, the long-term peak containment pressure and temperature are more than 10 psi and 30°F lower than those predicted using the current WCAP-10325 LOCA M&E release model.

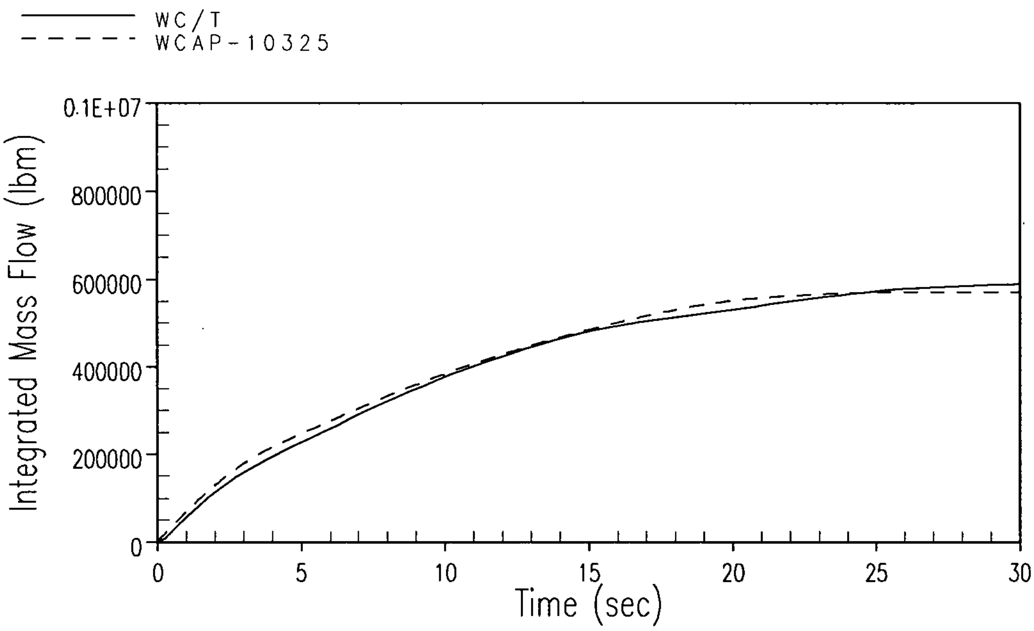

Figure C.4.3-1 Integrated Blowdown Break Mass Release Comparison

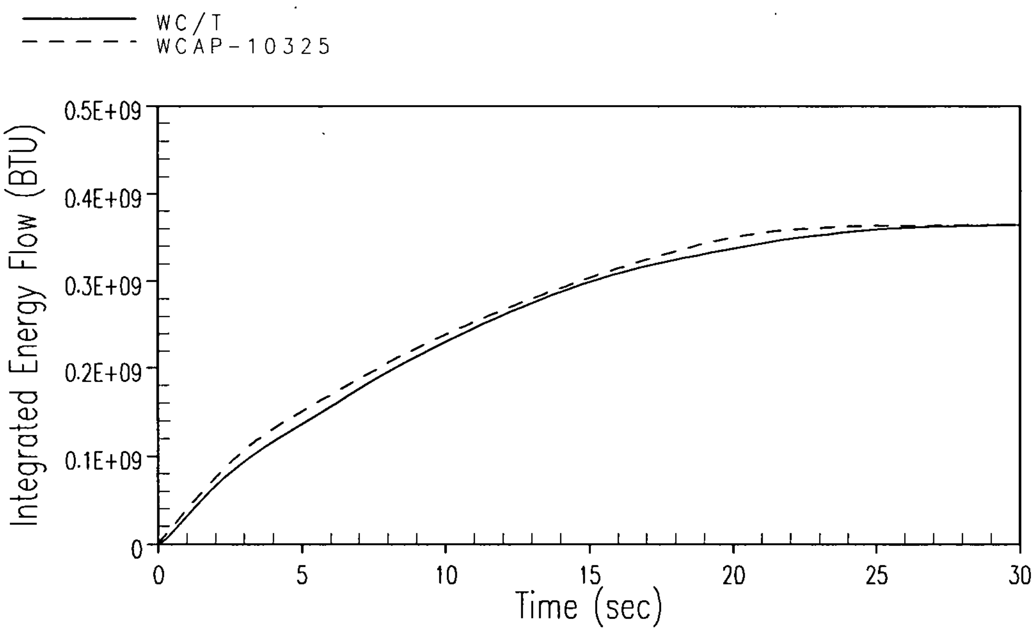

Figure C.4.3-2 Integrated Blowdown Break Energy Release Comparison

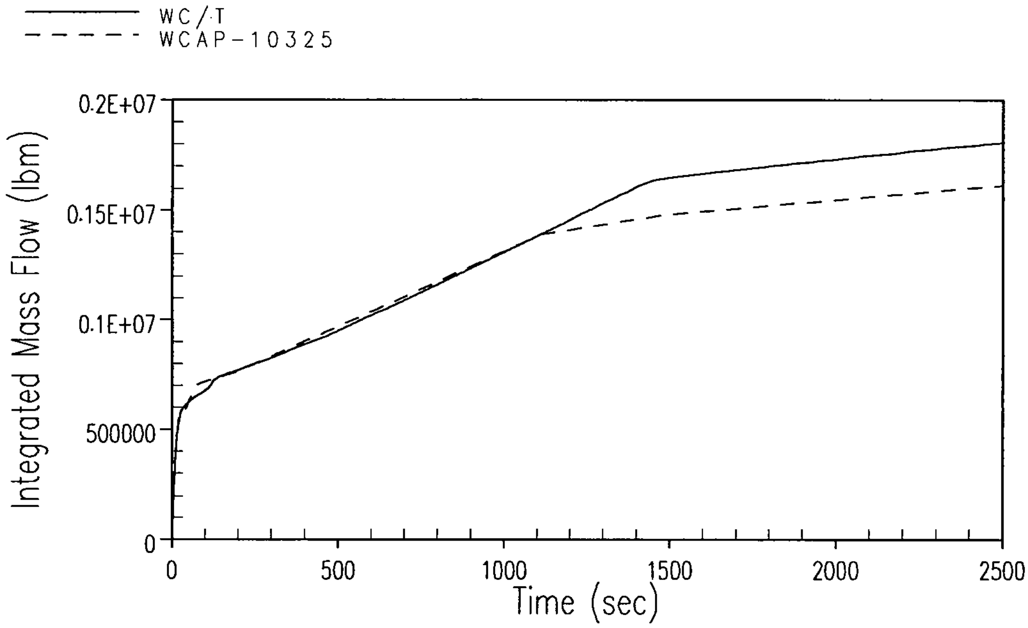

Figure C.4.3-3 Integrated Long-term Break Mass Release Comparison

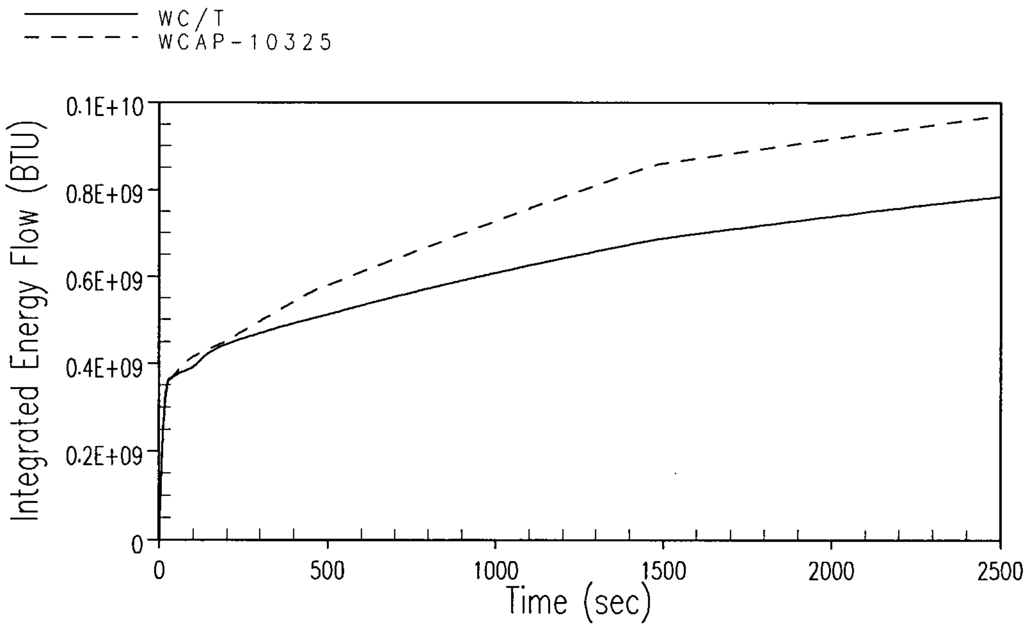

Figure C.4.3-4 Integrated Long-term Break Energy Release Comparison

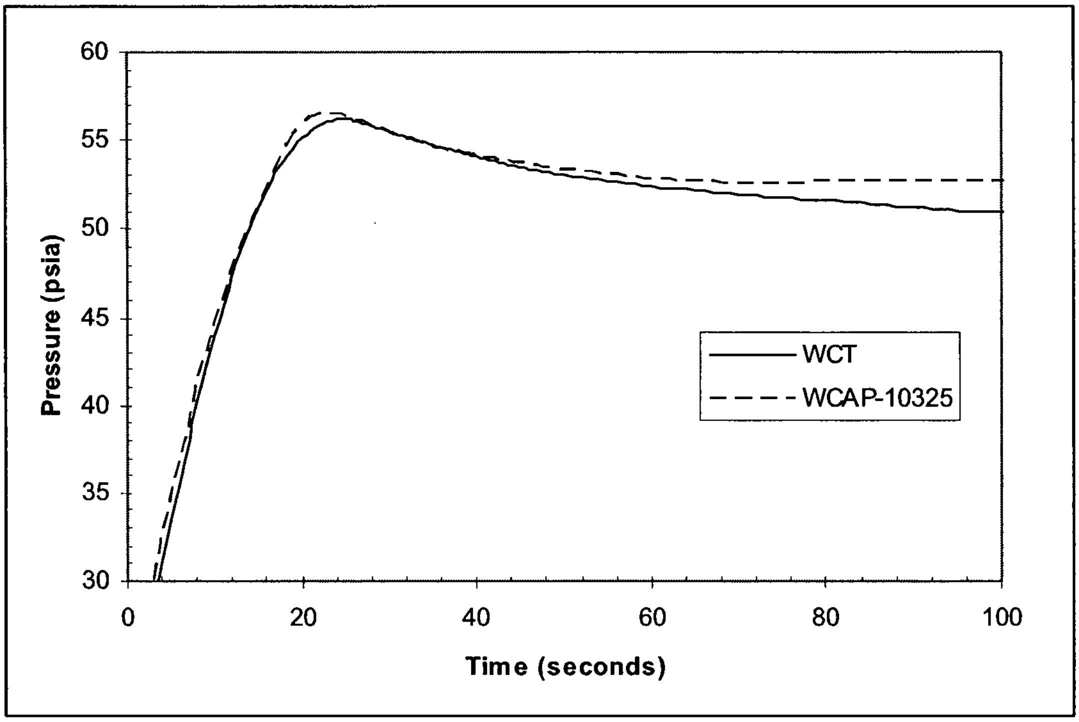

Figure C.4.3-5 Blowdown Containment Pressure Comparison

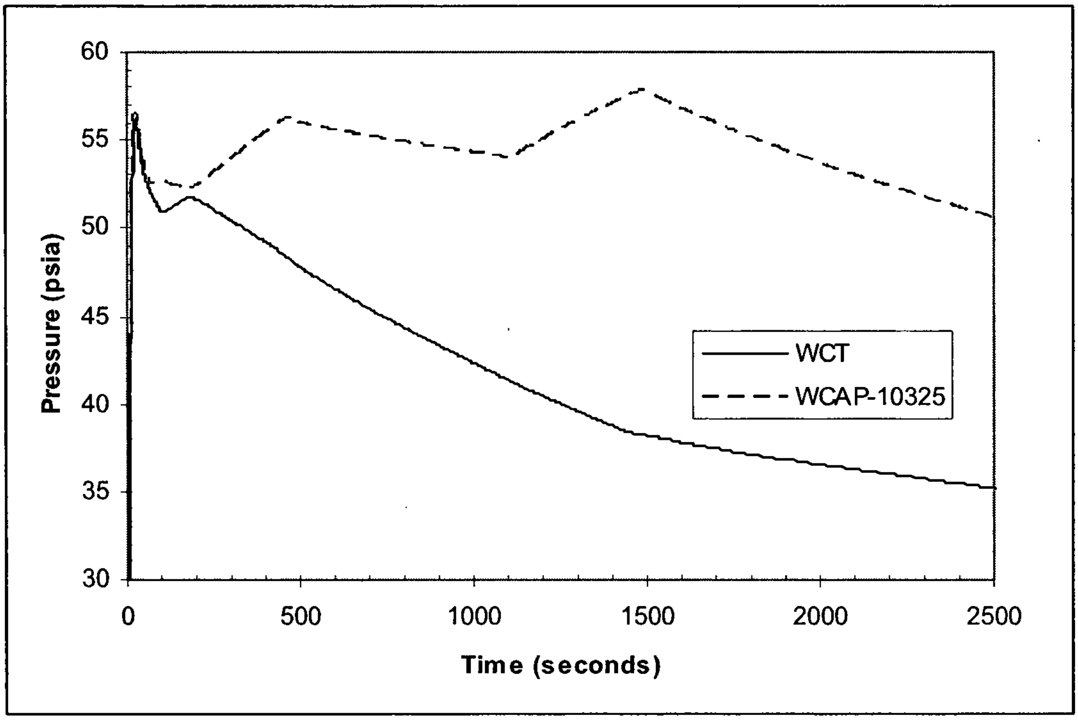

Figure C.4.3-6 Long-term Containment Pressure Comparison

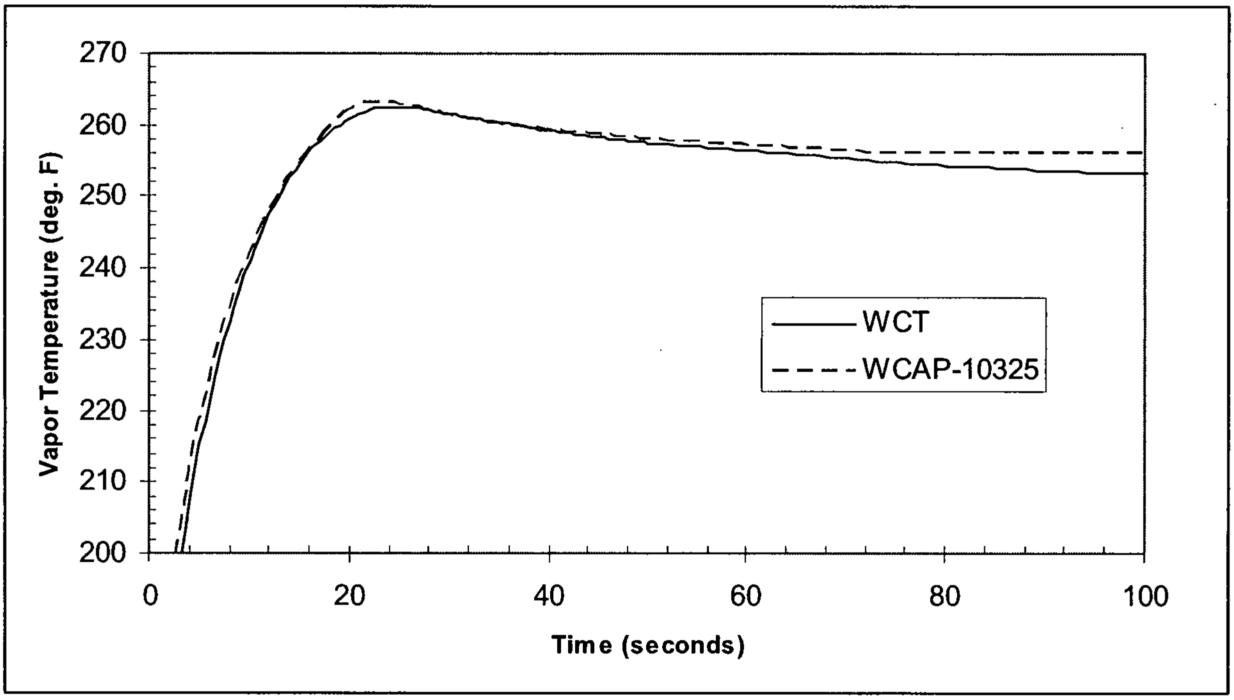

Figure C.4.3-7 Blowdown Containment Vapor Temperature Comparison

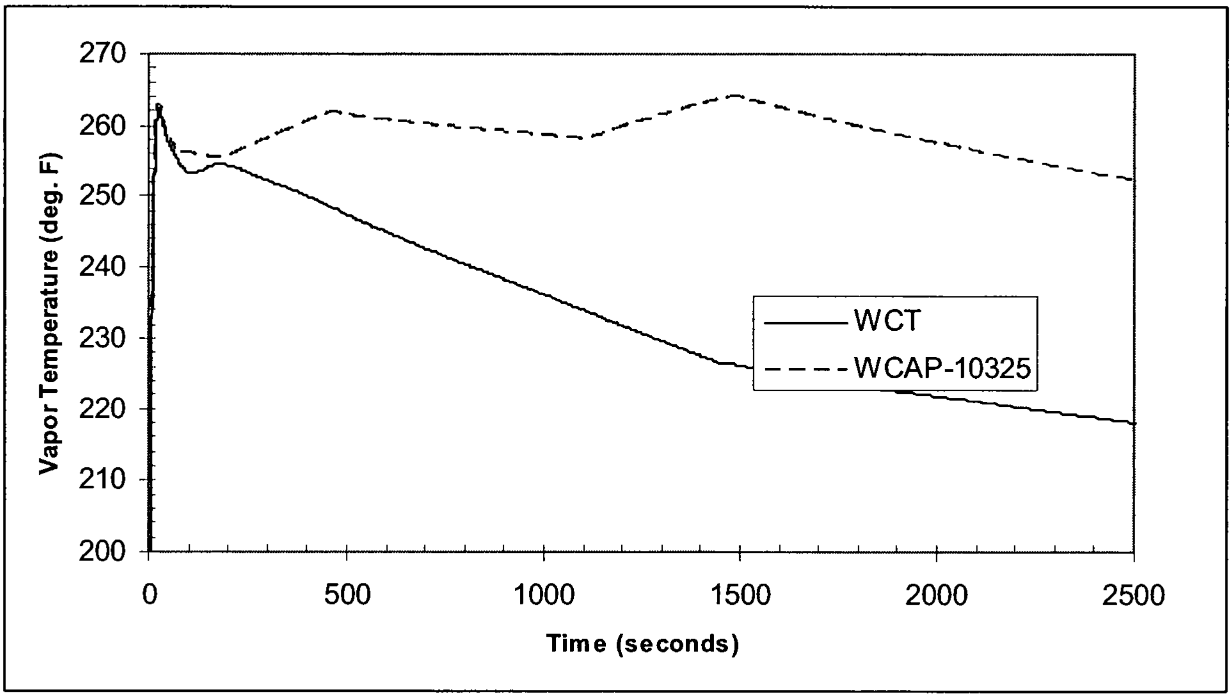

Figure C.4.3-8 Long-term Containment Vapor Temperature Comparison

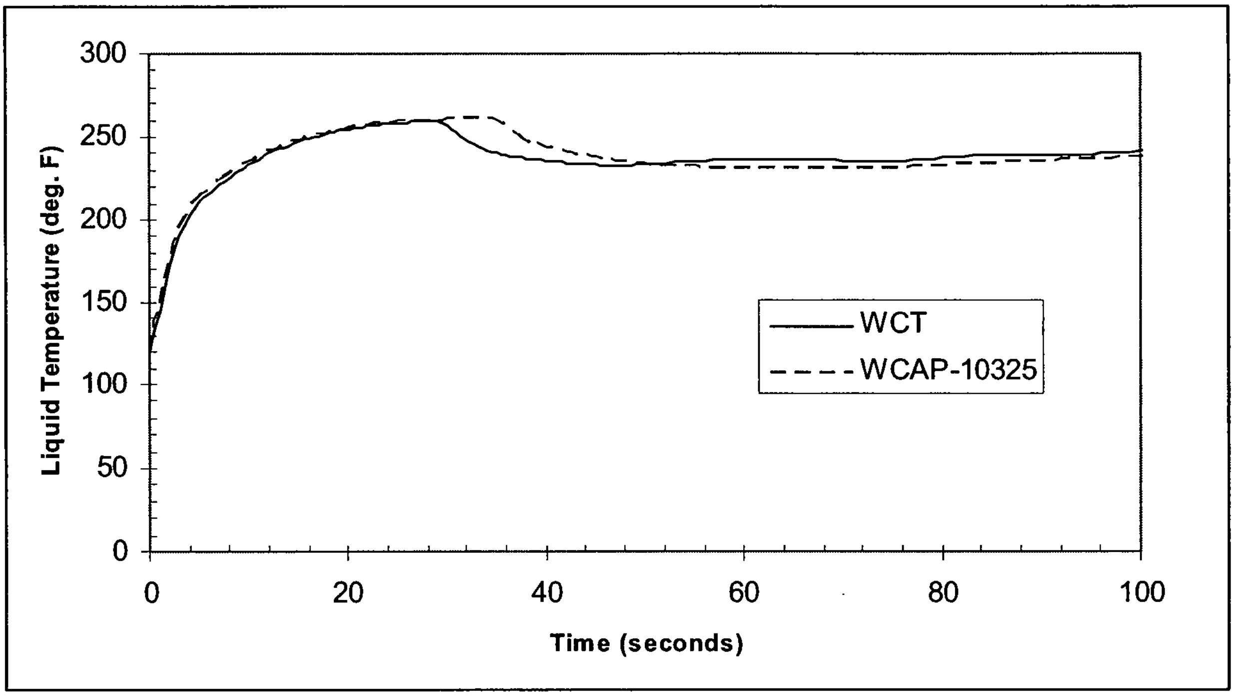

Figure C.4.3-9 Blowdown Containment Sump Temperature Comparison

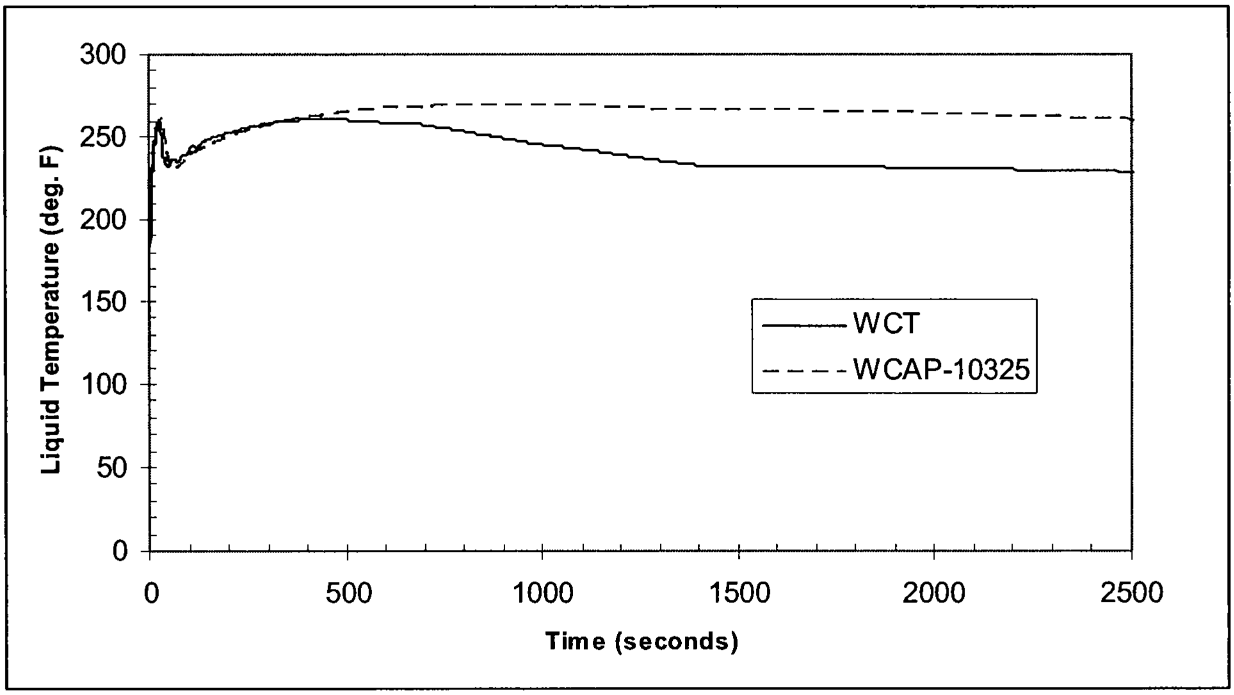

Figure C.4.3-10 Long-term Containment Sump Temperature Comparison

C.4.4 DEHL LOCA Benchmark Case Results Comparison

The DEHL break is located in the pressurizer loop in both the WC/T and SATAN-VI models. [

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The WC/T DEHL LOCA case was run for at least 25 seconds to allow the M&E release and containment response results to be compared with the WCAP-10325 benchmark case. The integrated break mass and energy release comparison is shown in Figures C.4.4-1 and C.4.4-2. The WC/T model calculates a similar blowdown break mass and energy release. The containment response comparison is shown in Figures C.4.4-3 through C.4.4-5. The blowdown peak pressure and temperature are about the same since the energy release rate is nearly the same.

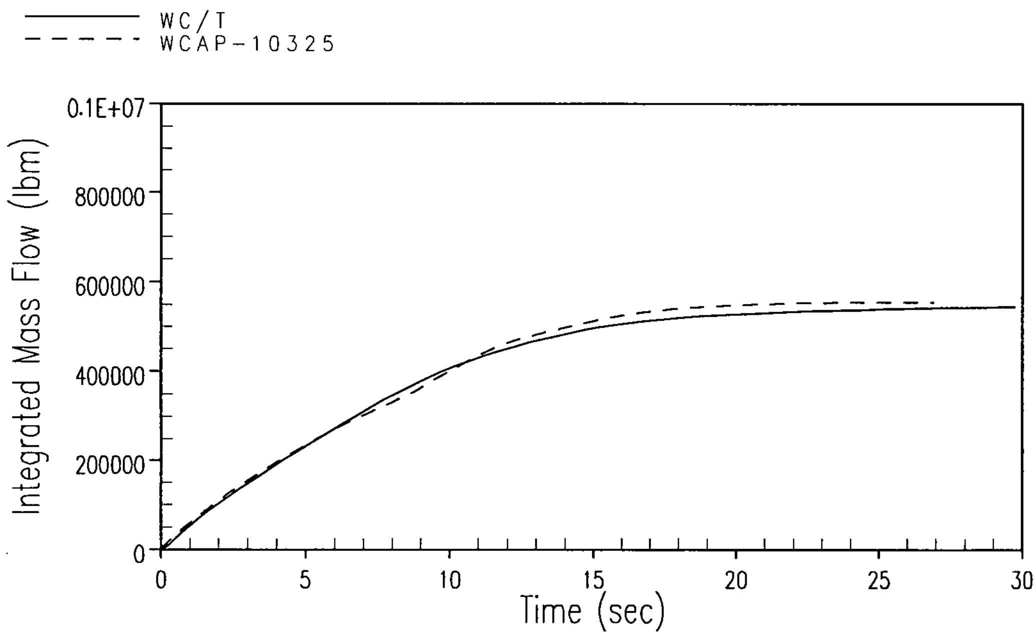

Figure C.4.4-1 Integrated Break Flow Rate Comparison

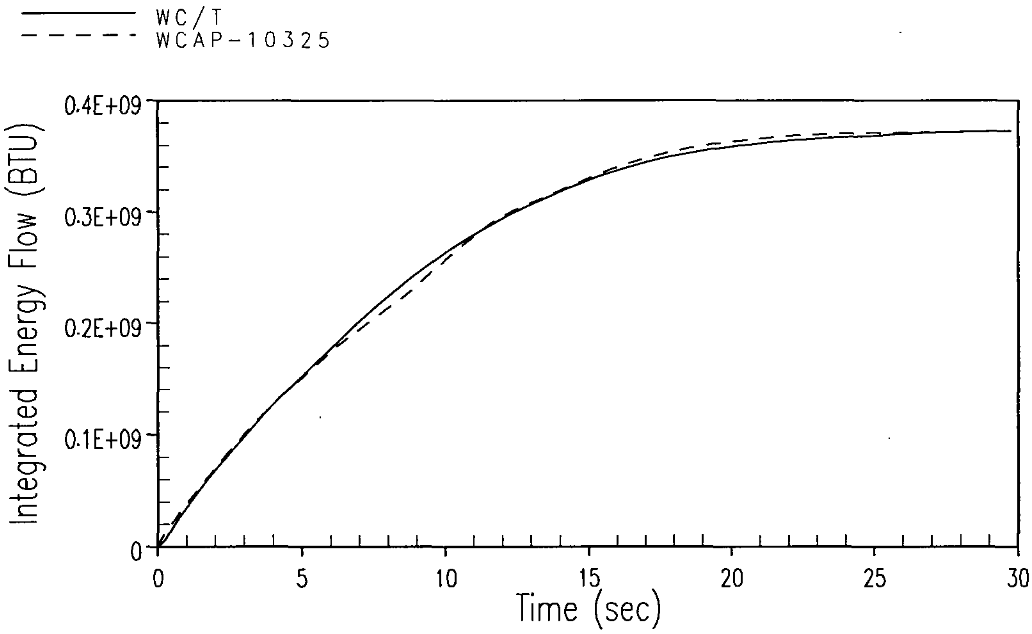

Figure C.4.4-2 Integrated Break Energy Release Rate Comparison

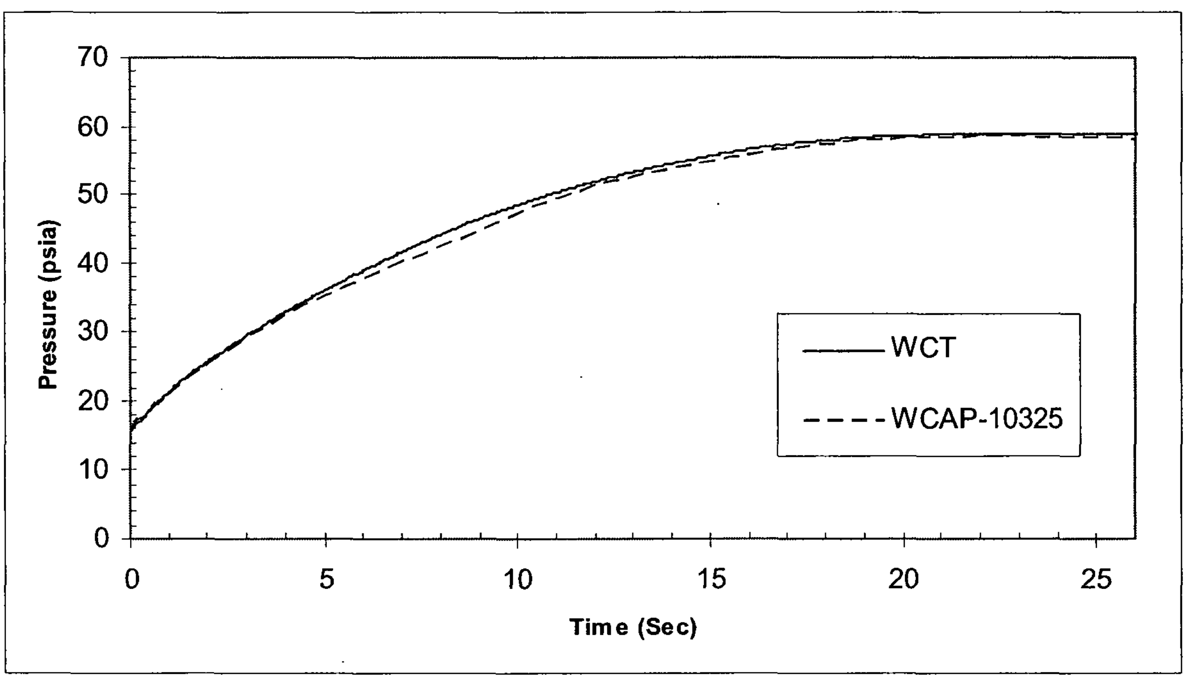

Figure C.4.4-3 Containment Pressure Comparison

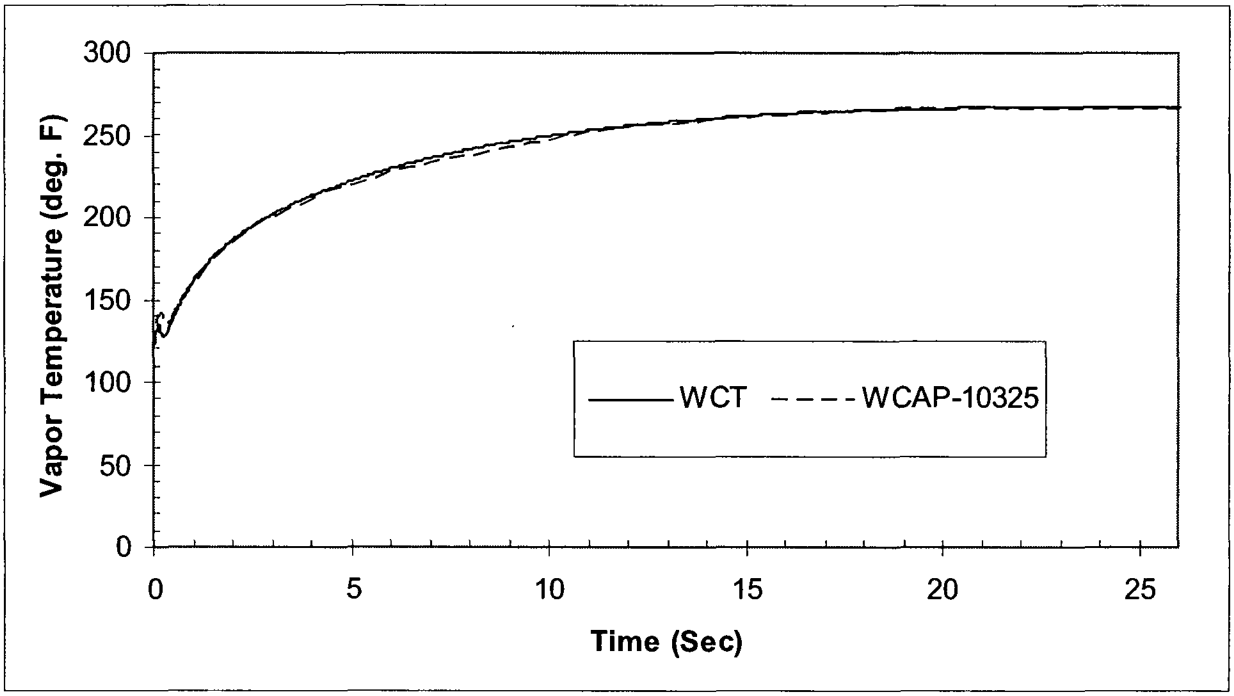

Figure C.4.4-4 Containment Temperature Comparison

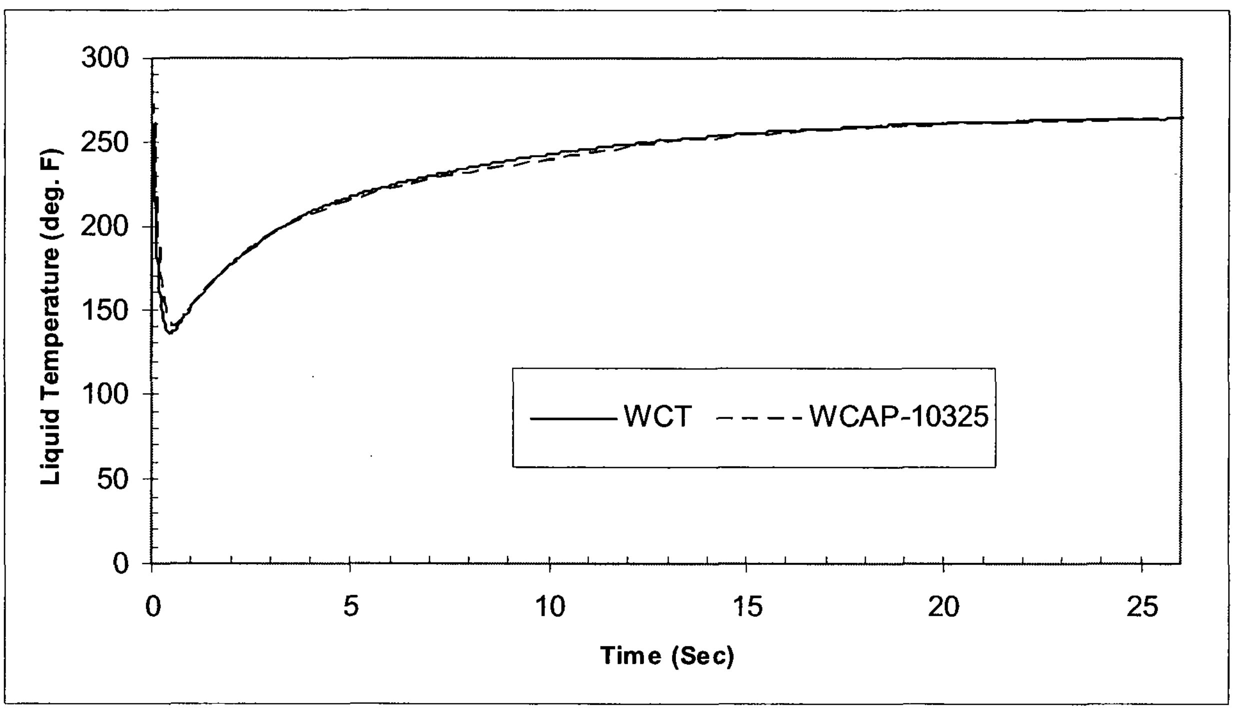

Figure C.4.4-5 Containment Sump Temperature Comparison

C.5 SAMPLE CASES

The WC/T LOCA M&E release and containment models described in Section C.4 were used to produce sample transient cases for the containment peak pressure/temperature application, the long-term equipment qualification (EQ) application and the minimum net positive suction head available (NPSHa) application. This section provides the results from these sample cases.

C.5.1 Peak Containment Pressure/Temperature

LOCA M&E releases for the peak containment pressure/temperature application were generated for the DEPS, DEHL, and DECL LOCA events. [

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The containment pressure, temperature, and sump temperature for the three cases are compared in Figures C.5.1-1 through C.5.1-6. The peak pressure and temperature occur during blowdown for all three cases; the DEHL case peak pressure is highest, but the DECL case pressure peaks first and is slightly higher than the DEPS case. The containment pressure for the DEPS case increases between 100 and 200 seconds as steam produced during the core reflood process, along with energy from the broken loop steam generator, is added to the containment. In the long-term, the containment pressure and temperature remain higher for the DECL and DEPS cases due to the addition of the SG secondary energy to the break flow from the SG side of the break.

The blowdown break mass flow and energy release rates are compared in Figures C.5.1-7 and C.5.1-8. The DECL break flow and energy release rates are much higher than the others during the first 2 seconds. This explains why the containment pressure peaks first for the DECL case. Figure C.5.1-9 compares the average blowdown break enthalpy. The average break enthalpy for the DEHL case is higher than the others since the release is mostly steam; this causes the initial containment pressure for this case to be higher than the others.

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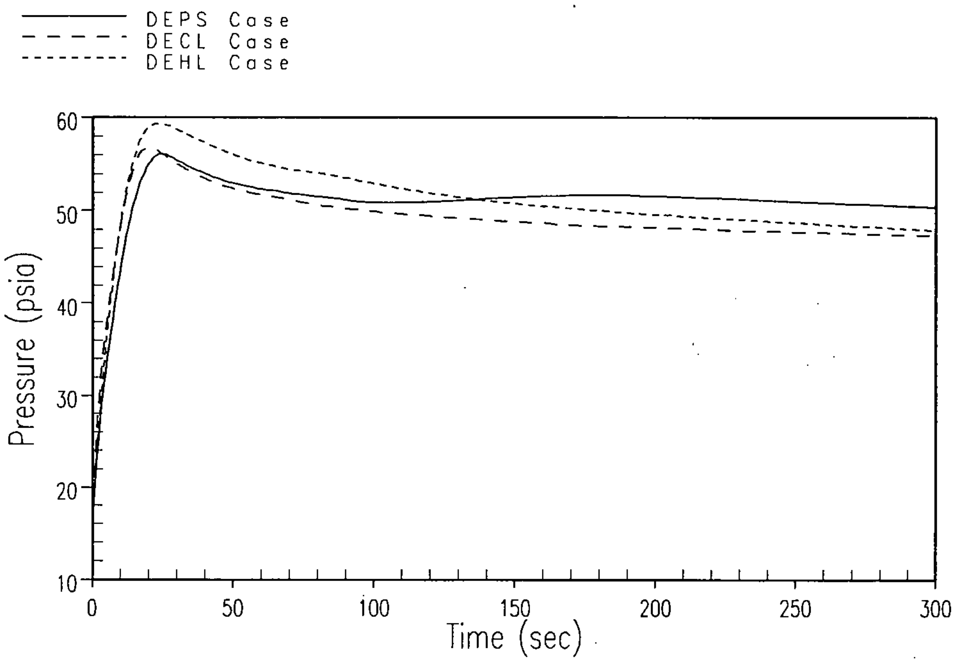

Figure C.5.1-1 Peak Containment Pressure Comparison

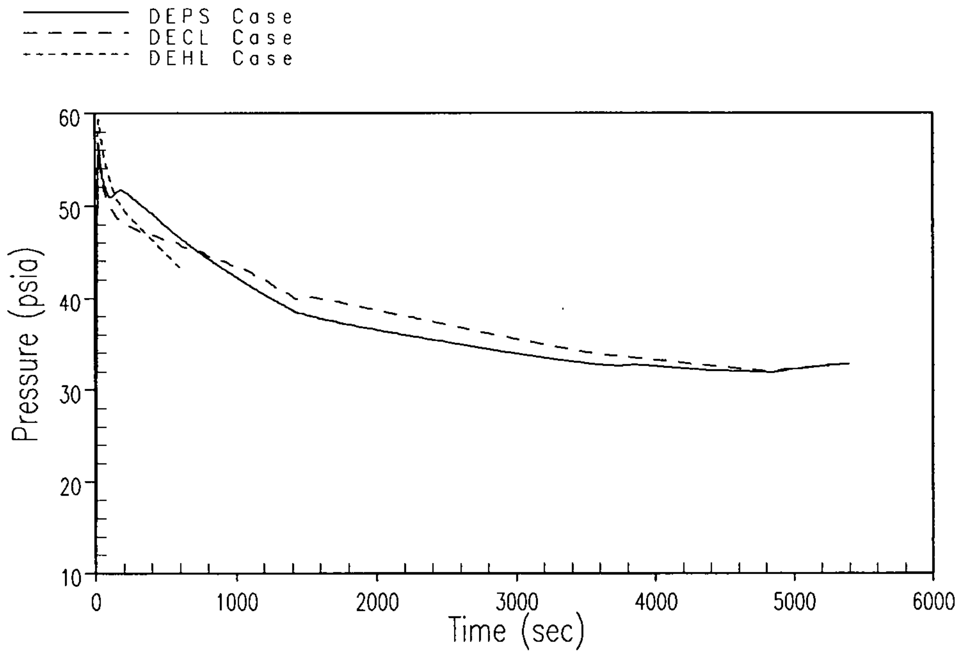

Figure C.5.1-2 Long-term Containment Pressure Comparison

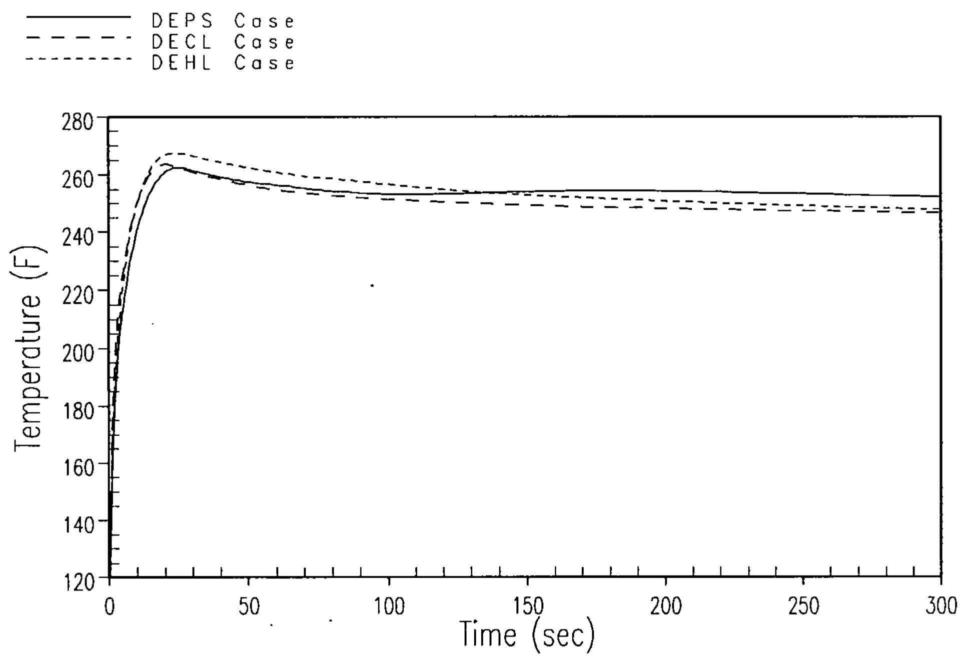

Figure C.5.1-3 Peak Containment Temperature Comparison

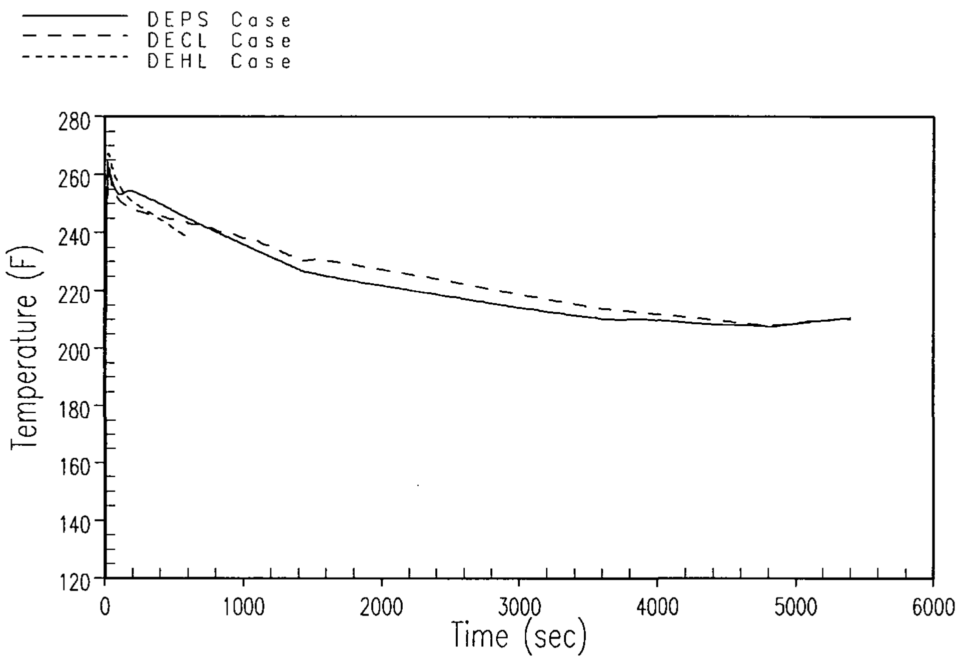

Figure C.5.1-4 Long-term Containment Temperature Comparison

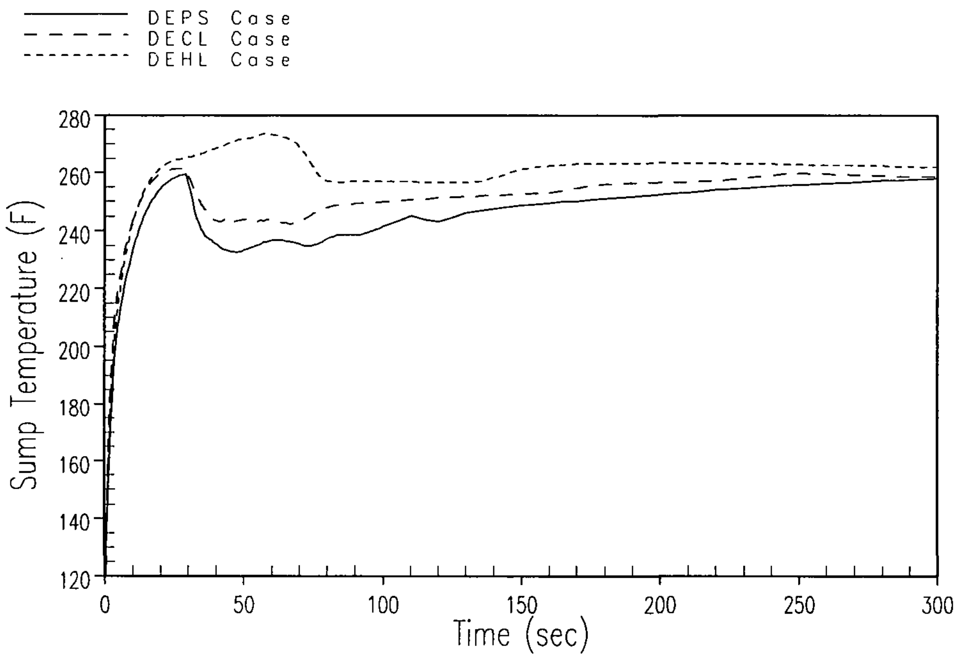

Figure C.5.1-5 Peak Sump Temperature Comparison

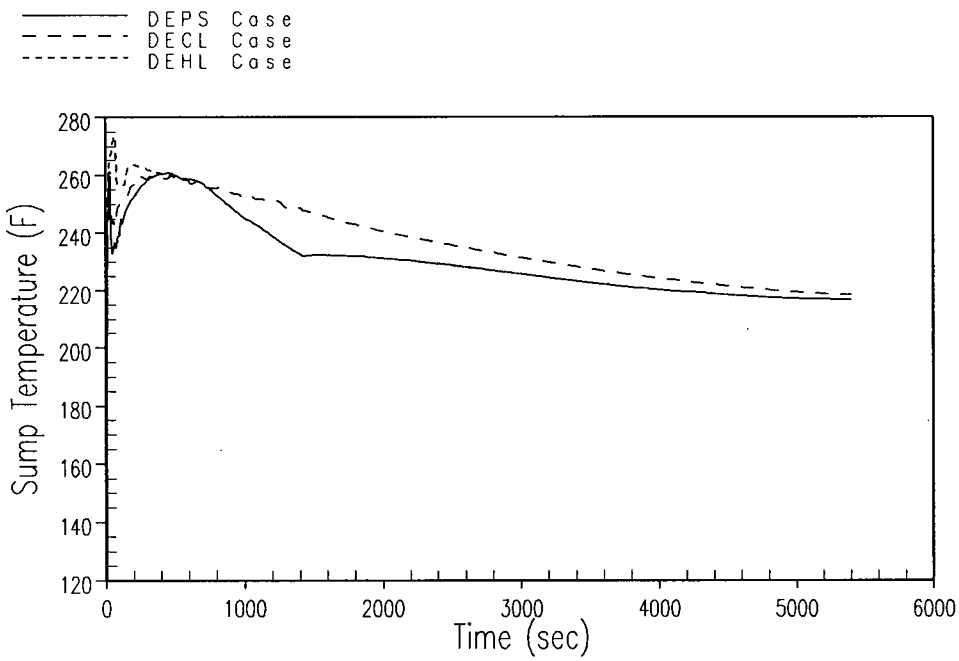

Figure C.5.1-6 Long-term Sump Temperature Comparison

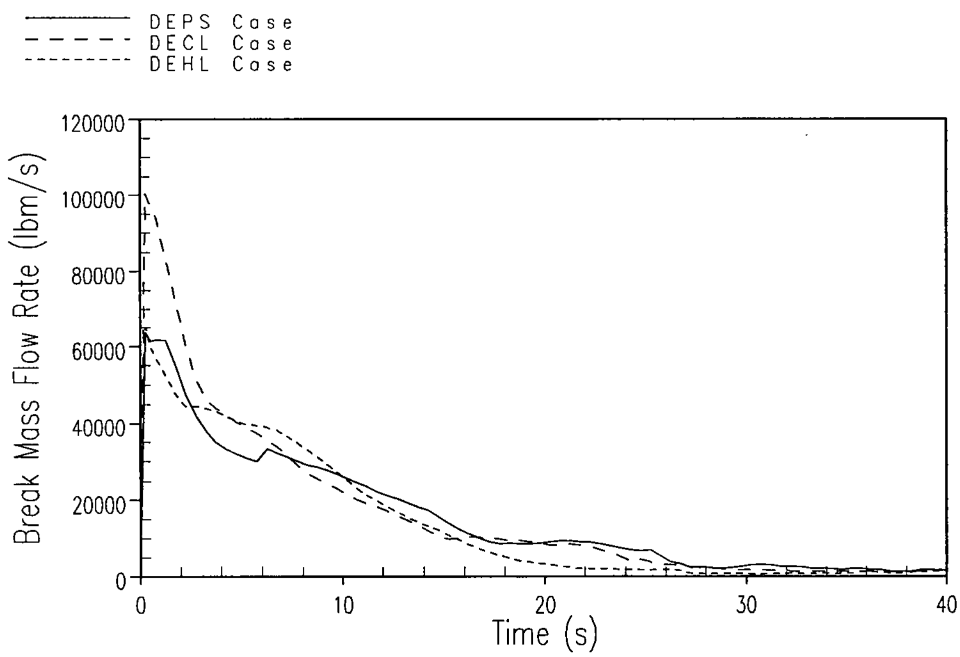

Figure C.5.1-7 Break Mass Flow Rate Comparison

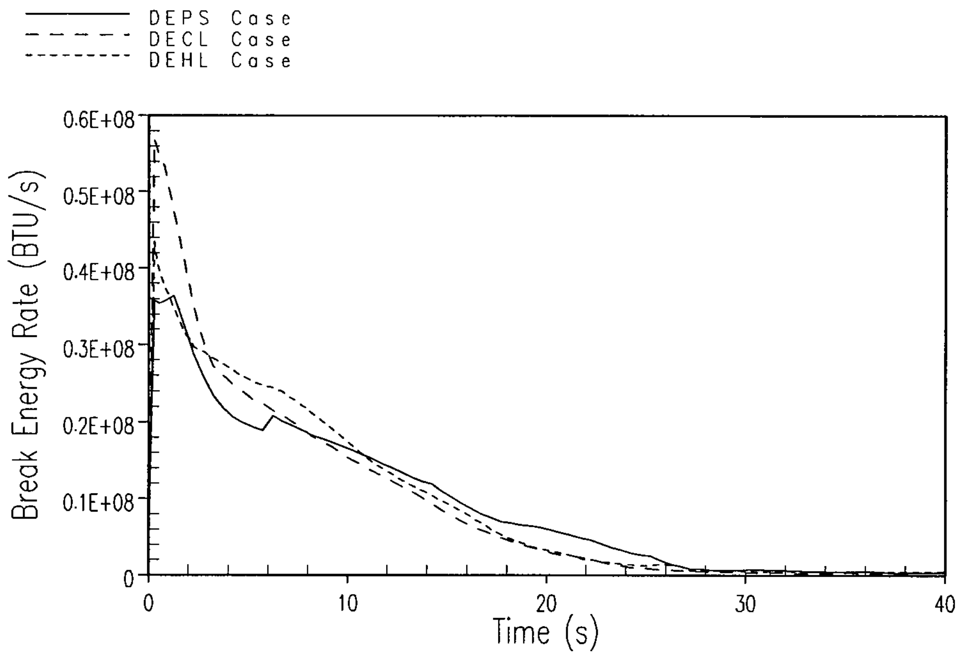

Figure C.5.1-8 Break Energy Flow Rate Comparison

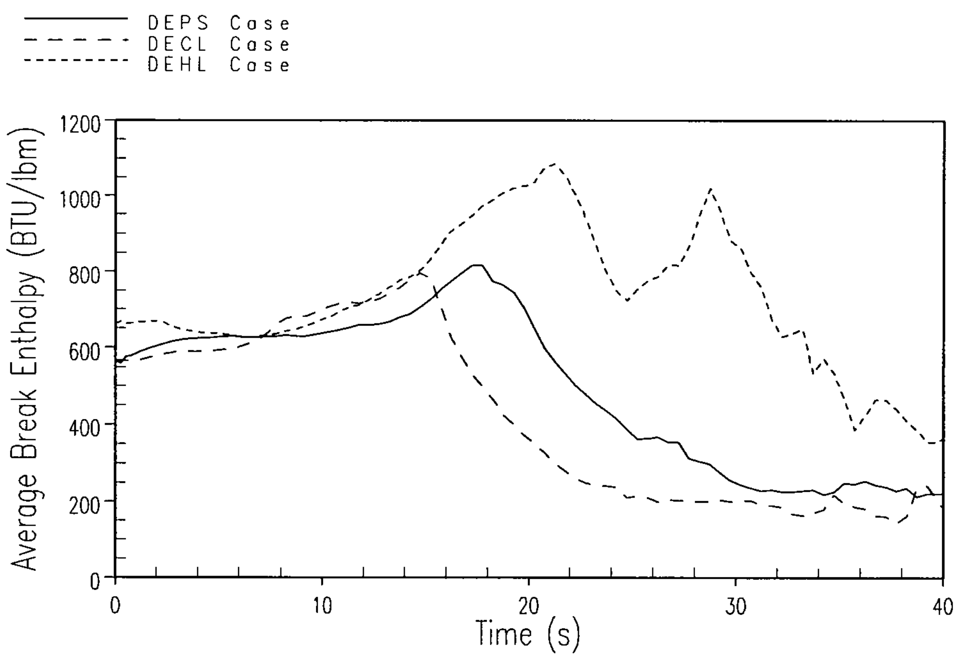

Figure C.5.1-9 Break Enthalpy Comparison

C.5.2 Long-term EQ

As described in Section C.3.2, the long-term LOCA steam release rate is maximized for the GOTHIC long-term EQ analysis. This increases the calculated containment pressure and temperature.

The long-term EQ mass and energy releases for the DEPS LOCA are shown in Figures C.5.2-1 and C.5.2-2. The recirculation flow rate was held constant at approximately 1,000 gpm. The steam mass and energy release rate decreased as the core decay, SG fluid, SG metal, and RCS metal energy release rates decreased. The containment pressure, temperature, and sump temperature response are shown in Figures C.5.2-3 through C.5.2-5. The containment pressure and temperatures decreased as the steam mass and energy release rate decreased.

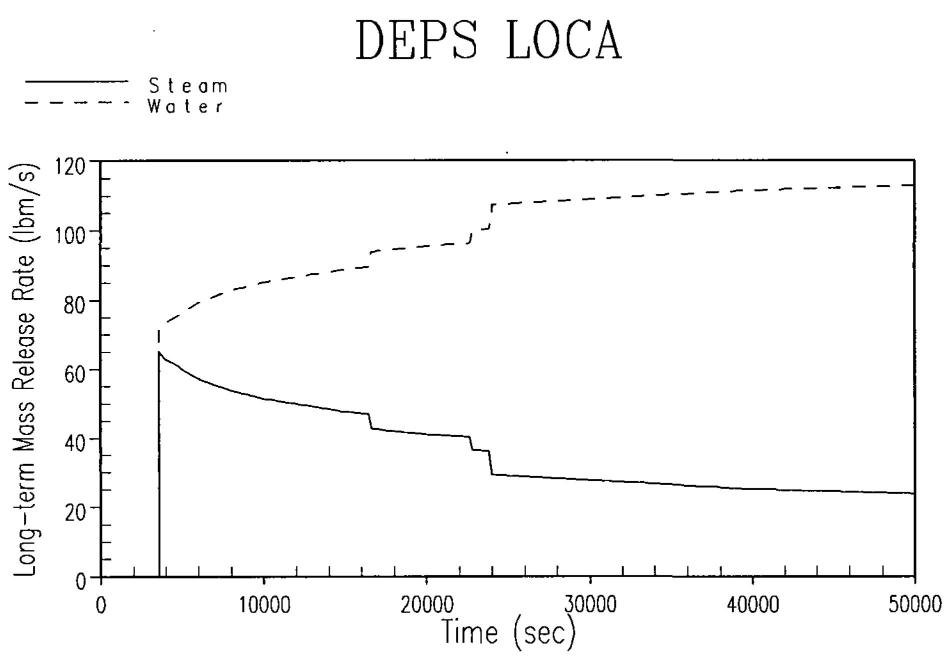

Figure C.5.2-1 Long-term EQ Break Flow Rate

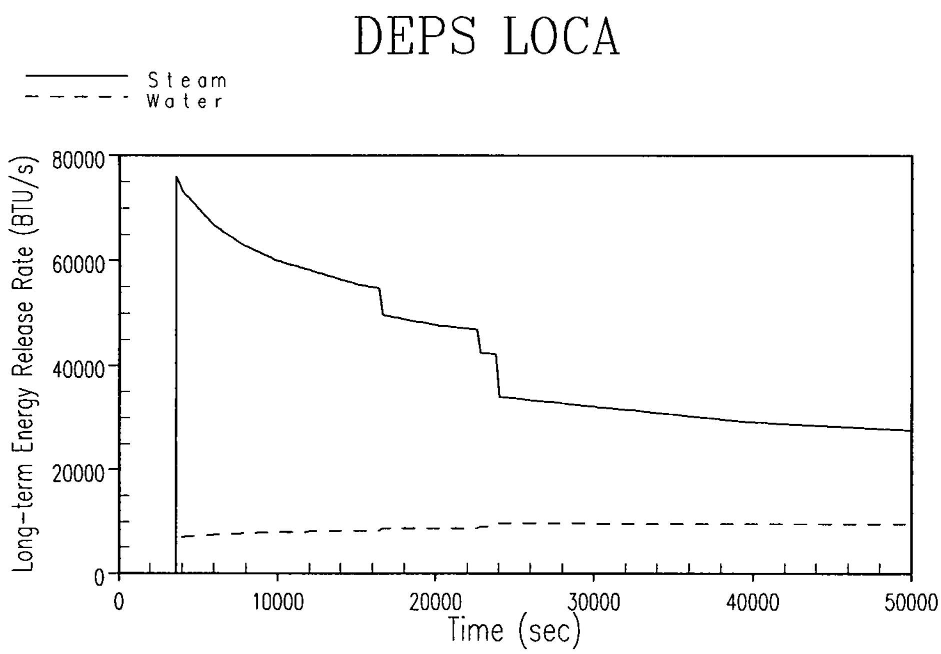

Figure C.5.2-2 Long-term EQ Break Energy Flow Rate

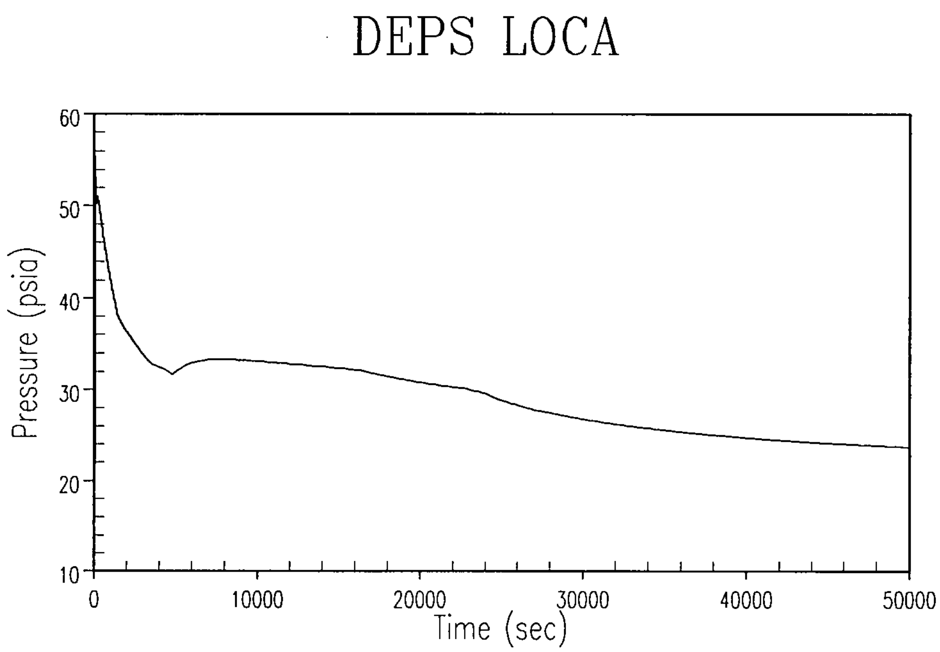

Figure C.5.2-3 Long-term EQ Containment Pressure

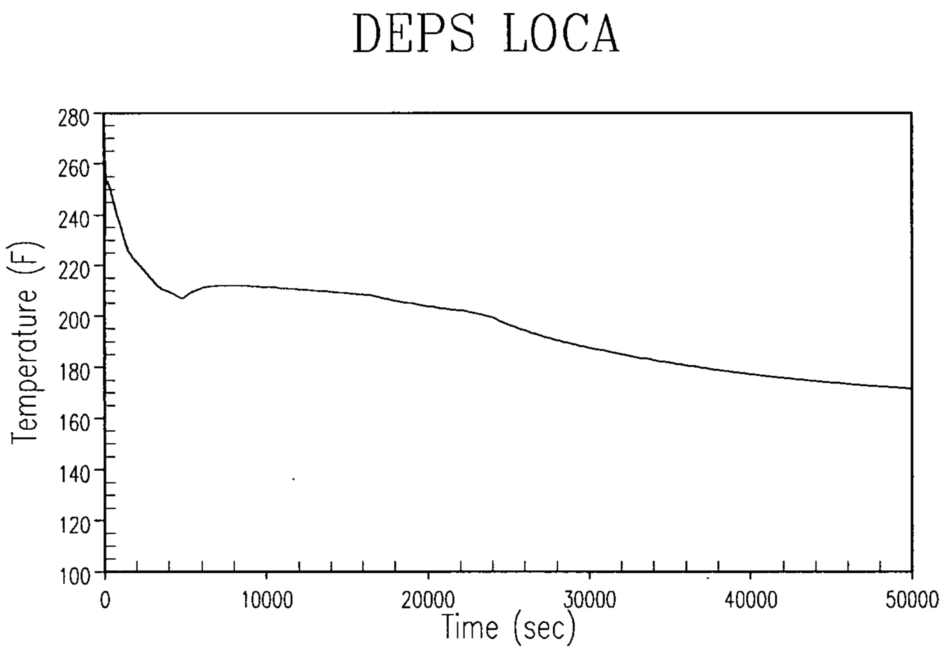

Figure C.5.2-4 Long-term EQ Containment Temperature

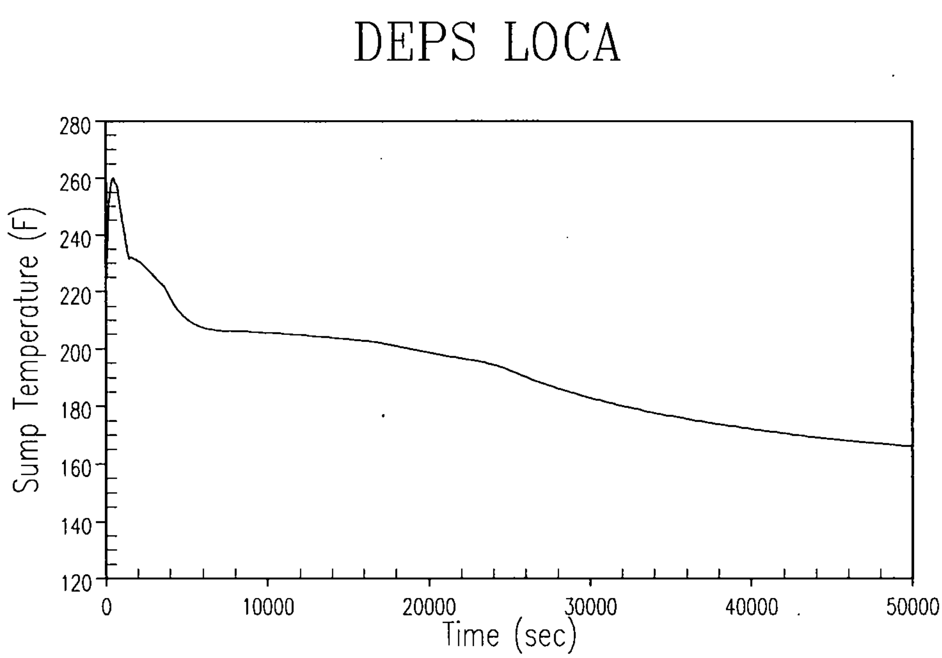

Figure C.5.2-5 Long-term EQ Containment Sump Temperature

C.5.3 Minimum NPSHa

As described in Section C.3.3, the LOCA steam release rate is minimized for the GOTHIC minimum NPSHa analysis. This reduces the containment backpressure and increases the containment sump temperature.

The minimum NPSHa mass and energy releases for the DEPS LOCA are shown in Figures C.5.3-1 and C.5.3-2. The recirculation flow rate was held constant at approximately 1,000 gpm. The steam mass flow rate was lower and the liquid mass flow rate was higher when compared with the long-term EQ sample case results. The energy release rate decreased as the core decay, SG fluid, SG metal, and RCS metal energy release rates decreased. The containment pressure, temperature, and sump temperature response are shown in Figures C.5.3-3 through C.5.3-5. The containment pressure and temperature were slightly lower and the sump temperature was higher when compared with the long-term EQ sample case results.

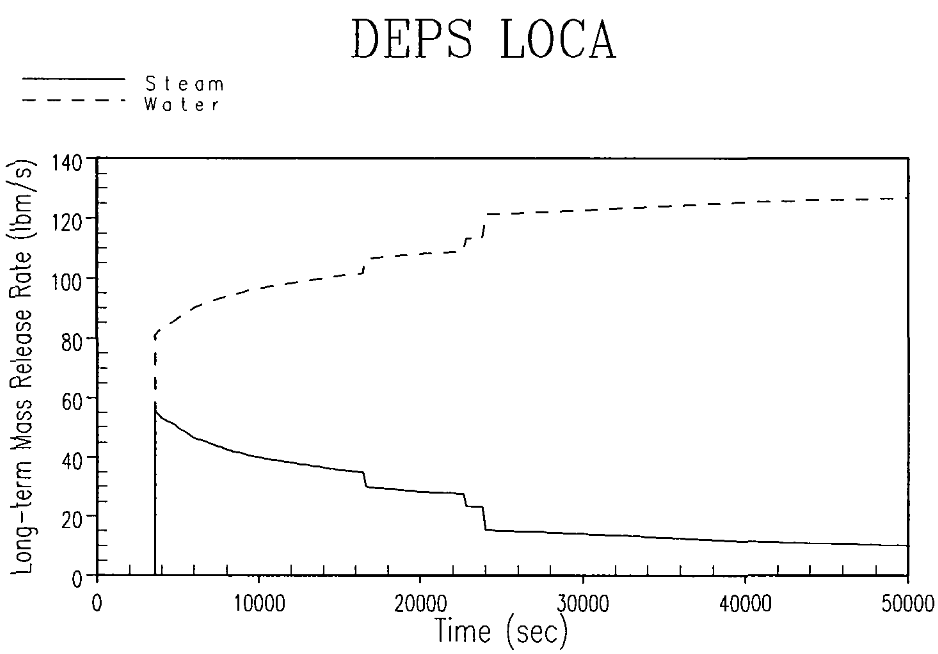

Figure C.5.3-1 Minimum NPSHa Break Flow Rate

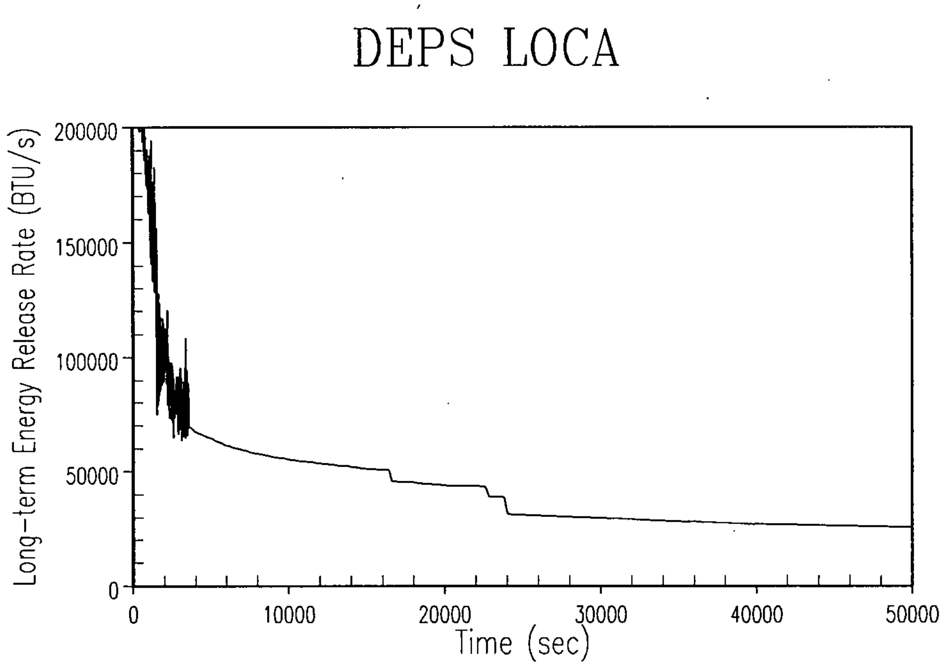

Figure C.5.3-2 Minimum NPSHa Break Energy Flow Rate

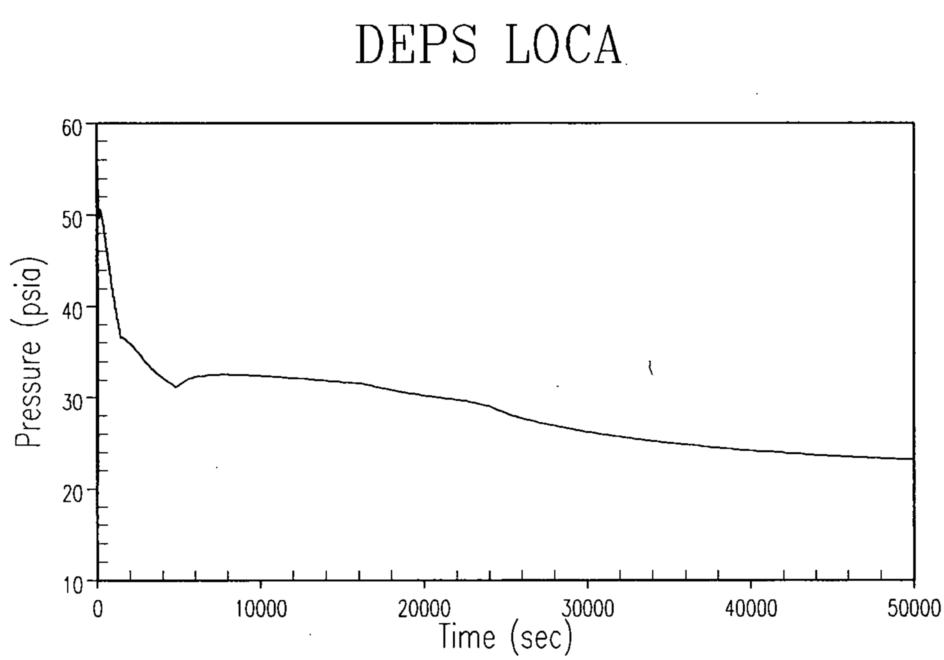

Figure C.5.3-3 Minimum NPSHa Containment Pressure

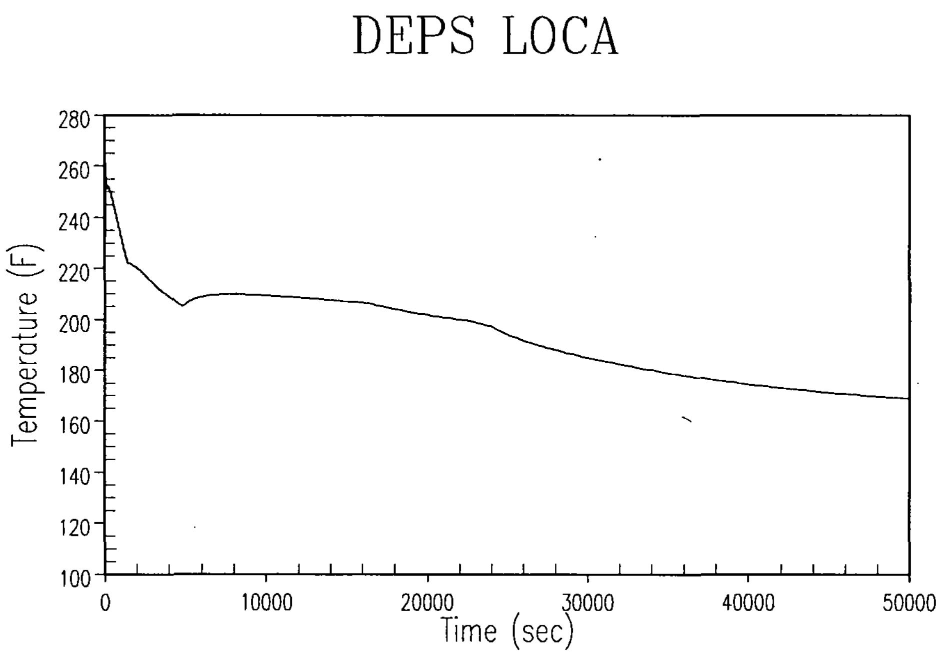

Figure C.5.3-4 Minimum NPSHa Containment Temperature

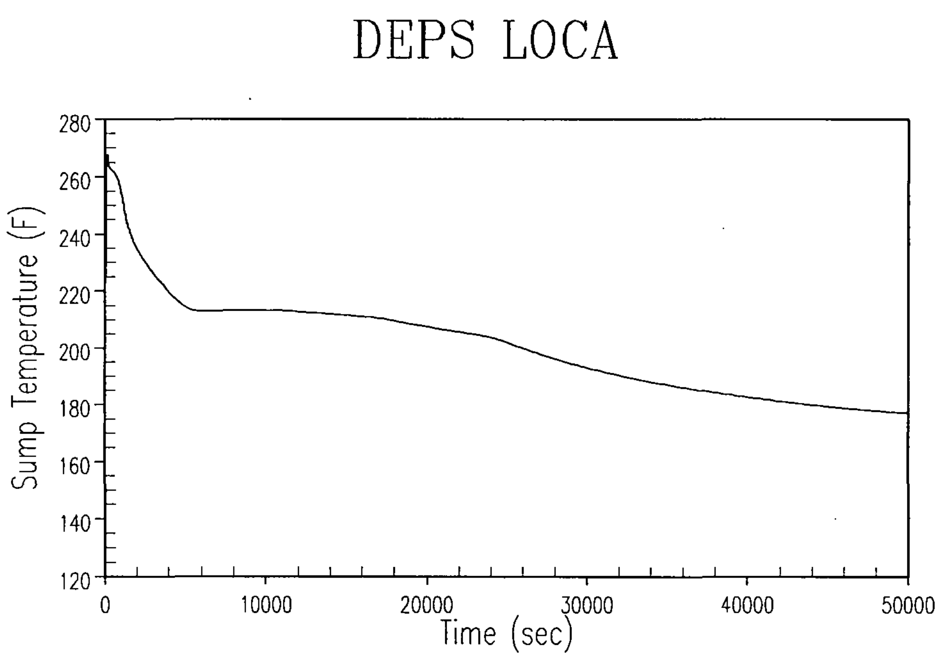

Figure C.5.3-5 Minimum NPSHa Containment Sump Temperature

C.6 CONCLUSIONS

The WC/T ECCS analysis model was modified, as described in this Appendix, to allow it to produce M&E releases for the PWR LOCA containment response calculations. Modifications were made to both the code and the input bias.

The WC/T code changes that were made for the LOCA M&E calculations are transparent to, and do not affect the ECCS analysis. The WC/T code was modified to better model the SG interface heat/mass transfer and SG metal heat transfer, and to allow it to run in parallel with GOTHIC. The SG modeling changes were validated by comparison with data from the FLECHT SEASET test facility. The modified WC/T code conservatively calculates the transfer rate of SG secondary side energy to the primary and runs in parallel with GOTHIC.

The WC/T ECCS model input was biased to produce conservative LOCA M&E releases in accordance with the acceptance criteria documented in the regulations. The WC/T calculated LOCA M&E release data was compared with results from the currently approved model (Reference C-2). The LOCA blowdown M&E releases were essentially the same; however, the WC/T post-reflood energy releases were lower than the currently approved model because the WC/T LOCA M&E release model uses mechanistic steam generator and metal heat release models.

Finally, sample transient results for the containment peak pressure, long-term EQ, and minimum NPSHa applications were produced. The results demonstrate that the WC/T LOCA M&E release model, coupled with a GOTHIC containment response and long-term steaming release model, is capable of performing these types of calculations with analysis margins to the containment design limits.

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